U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-2206

Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments

Office of Nuclear Regulatory Research

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at the NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. **The Superintendent of Documents**
   U.S. Government Publishing Office
   Mail Stop SSOP
   Washington, DC 20402-0001
   Internet: http://bookstore.gpo.gov
   Telephone: 1-866-512-1800
   Fax: (202) 512-2104

2. **The National Technical Information Service**
   5301 Shawnee Road
   Alexandria, VA 22161-0002
   http://www.ntis.gov
   1-800-553-6847 or, locally, (703) 605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

U.S. Nuclear Regulatory Commission
Office of Administration
Multimedia, Graphics, Storage, and Distribution Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at the NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
http://www.ansi.org
(212) 642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG-2206

United States Nuclear Regulatory Commission

Protecting People and the Environment

Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments

Manuscript Completed: December 2016
Date Published: March 2018

Prepared by:
Jonathan Barr, Sudhamay Basu,
Hossein Esmaili, and Martin Stutzke

Office of Nuclear Regulatory Research

ABSTRACT

The work documented in this report focused on developing the technical basis for a potential rulemaking action on containment protection and release reduction (CPRR) strategies for boiling water reactors with Mark I and Mark II containments. The work covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by extended loss of ac power (ELAP) due to internal events and seismic events deemed to be the most significant risk contributors; (2) accident progression analysis of these sequences and assessment of radiological source terms; and (3) analysis of offsite consequences including individual early fatality risk and latent cancer fatality risk, land contamination, and economic consequences. The calculated offsite consequences were weighted by accident frequency to assess relative public health risk reduction associated with various CPRR strategies. Important findings and key insights from the work are delineated.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

This report documents research conducted by the staff of the Office of Nuclear Regulatory Research (RES) to support the agency initiative to evaluate containment venting and filtration for boiling water reactor (BWR) Mark I and Mark II containments. Containment venting and filtration was identified by the Near Term Task Force (NTTF), put together in the aftermath of the Fukushima Dai-ichi nuclear power plant accident in Japan, for all containment types but prioritized as a short-term action item for the two containment types mentioned above. The initiative was led by the Office of Nuclear Reactor Regulation (NRR) Japan Lessons Learned Directorate (JLD).

The containment venting and filtration issue has a long history behind it. Subsequent to the Three Mile Island Unit 2 nuclear plant core melt event in 1979, controlled (and potentially filtered) release was identified in NUREG-0585 as a favorable alternative to catastrophic failure of the containment. In SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988, the U.S. Nuclear Regulatory Commission (NRC) staff presented to the Commission its plan to evaluate potential generic severe accident containment vulnerabilities in a research effort entitled the "Containment Performance Improvement Program" (CPIP). All light water reactor (LWR) containment types were considered in the program. Potential improvements for Mark I containments, documented in NUREG/CR-5225, included: (1) hydrogen control; (2) alternate water supply for reactor vessel injection and containment drywell sprays; (3) containment pressure relief capability (venting); (4) enhanced reactor pressure vessel (RPV) depressurization system reliability; (5) core debris control; and (6) emergency procedures and training.

Potential improvements for Mark II containments, identified in NUREG/CR-5528, were largely the same as those for Mark I containments. However, less definitive conclusions were reached regarding venting of Mark II containments for a number of reasons. Potential improvements for other containment types were not studied in as much detail as for Mark I and Mark II containments.

The events at the Fukushima Dai-ichi nuclear power plant brought the containment venting and filtration issue to the forefront once again. In SECY-11-0137, the NRC staff described its proposals for regulatory actions to address containment venting related NTTF recommendations. The insights gained from Fukushima Dai-ichi led the agency to impose additional requirements for reliable hardened venting systems for plants with Mark I and Mark II containments through the Order EA-12-050, on the basis of ensuring adequate protection of public health and safety. This order provided requirements to ensure reliable operation of the hardened venting system; however, it did not include requirements for reliable operation under severe accident conditions. One recommendation, identified in SECY-11-0137 and further developed in SECY-12-0025, was consideration of additional performance requirements, including filters, for hardened containment vent systems for Mark I and Mark II containment designs. This would improve reliability during severe accident conditions and limit the release of radioactive materials if the venting systems were used after significant core damage had occurred.

To support formulation of possible additional regulatory actions related to the performance of Mark I and Mark II containments during severe accidents, RES performed a systematic analysis of accident source terms and consequences for a representative BWR plant with Mark I containment and for representative accident scenarios, predicated on an extended loss of alternating current power (ELAP) as in Fukushima. The technical analysis, completed by RES in 2012 and documented in SECY-12-0157, concluded that the plants with Mark I and Mark II containments would benefit from a containment venting and water addition strategy for a vast majority of severe accident sequences (whether the vent includes an engineered filter or not). Of particular

importance to note in SECY-12-0157 was an accident scenario without venting which resulted in containment failure with significant release of radioactive materials to the environment. This lent further support to having a venting provision, capable of operating under severe accident conditions, for Mark I and Mark II containments. This also affirmed the Commission's earlier action, in the immediate aftermath of Fukushima, to impose Order EA-12-050 on the basis of ensuring adequate protection of public health and safety. The Order EA-12-050 was subsequently rescinded and replaced by the Order EA-13-109 emphasizing, in particular, the functionality of the reliable hardened vents under severe accident conditions.

Subsequently, the RES staff initiated additional technical work to develop technical basis for potential rulemaking involving filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments. This additional technical work was an extension of the previous work by RES documented in SECY-12-0157, and covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by extended loss of ac power (ELAP) due to internal events and seismic events deemed to be the most significant risk contributors; (2) accident progression analysis of these sequences and assessment of radiological source terms; and (3) analysis of offsite consequences using NRC's probabilistic offsite consequence computer code, MACCS (MELCOR Accident Consequence Code System). The calculated offsite consequences were weighted by accident frequency to assess relative public health risk reduction associated with various containment protection and release reduction measures.

The report documents the technical work to support the regulatory basis[1] for potential CPRR rulemaking activity (SECY-15-0085). Chapter 1 provides a historical perspective of the containment venting issue and a brief narrative on the resurgence of the issue in the aftermath of Fukushima. Chapter 2 of the report documents the staff's accident sequence analysis that was used to select risk-significant accident sequences and quantify their frequency. Chapter 3 describes in detail the MELCOR analysis, the scope of which falls broadly into two categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions, and (2) assessment of source terms (i.e., timing and magnitude of fission product releases to the environment). Chapter 4 describes in detail the offsite consequence analysis using MACCS, which evaluates health risks as well as land contamination and economic consequences. Chapter 5 describes the results of accident sequence analysis integrated with the results of offsite consequence analysis, and provides a discussion of integrated risk. Key insights from the technical work are summarized in Chapter 6 in three major areas: risk evaluation, source term assessment using MELCOR, and offsite consequence assessment using MACCS. In summary, the report provides technical inputs to the regulatory analysis, which is documented in an enclosure to SECY-15-0085.

The technical work in each of the three major disciplines was based on a number of assumptions or considerations that are delineated in detail in the respective chapters describing the work. Some top-level considerations are:

1 The complete draft regulatory basis attached to SECY-15-0085 includes relevant background, a regulatory evaluation, a technical evaluation that is based on the technical work documented in this report, performance criteria information, impact analysis, and the staff's conclusion.

- SRM-SECY-12-0157 direction that the regulatory basis should assume the benefits of severe accident capable hardened venting systems (EA-13-109) that would accrue equally to engineered filters and to filtration strategies;

- Consideration of a variety of performance criteria such as a decontamination factor, equipment and procedure availability similar to those required to implement 10 CFR 50.54(hh)(2), or other measures (e.g., quantitative health objective or QHO) that may be developed during the stakeholder engagement;

- Consideration of requirements associated with measures to enhance the capability to maintain containment integrity and to cool core debris; and

- SRM-SECY-12-0110 direction that economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety.

The first of these considerations essentially set the boundary conditions for all technical analyses documented in this report. For example, the CPRR alternatives defined in the risk analysis were predicated on the established capability of post-core damage containment venting. In two of four alternatives selected in the risk analysis (see Chapter 2 for details), additional capability to inject water (in the core and/or the containment) was not considered, whereas, in the other two alternatives, water injection was probabilistically considered. Moreover, in one of the two latter alternatives, an external engineered filter was considered as a further accident management feature, and its potential benefit was investigated. The accident sequence analysis was also informed by: (1) the lessons learned from the events at the Fukushima Dai-ichi Nuclear Power Plant; (2) the accident management alternatives being contemplated by the industry; and (3) the current state of knowledge of severe accident progression and mitigation alternatives in a BWR.

The accident sequence analysis involved development of a core damage event tree (CDET) and an accident progression event tree (APET), and binning of a rather large number of possible end states to a manageable number of categories with similar outcomes. In accordance with NTTF Recommendation 5.1, the evaluation of CPRR alternatives was focused on accidents that are initiated by a prolonged station blackout (SBO) event, i.e., an extended loss of alternating current power (ELAP) event with loss of all offsite and onsite ac power sources that lasts longer than the SBO coping duration specified in 10 CFR 50.63, “Loss of all alternating current power.” The human reliability aspect was considered in the accident sequence formulation and despite an initial attempt to develop a comprehensive human reliability assessment (HRA), only a bounding approach to incorporating HRA into accident sequence analysis was implemented at the end.

The accident sequence analysis results show a low value for core damage frequency (CDF) from an ELAP event, and provide insights into which initiating events (e.g. an earthquake), mitigation system performance (e.g., RCIC failure), or operator actions (equivalently, human error probability associated with such actions) contribute the most to overall CDF for the BWR plants with Mark I and Mark II containments. For example, the major contribution to seismically induced ELAP is from earthquakes that cause site peak ground accelerations in the range of 0.3 to 0.75g. Also, significant contributors to CDF include seismic failures of the batteries, DC switchgear, and the emergency diesel generators (EDGs) and their supporting equipment. Failure of the portable FLEX pump and failure to start of the reactor core isolation cooling (RCIC) pump is also significant. Over a reasonable range of values, CDF is not particularly sensitive to human error probabilities for in-control-room and ex-control-room operator actions.

The scope of MELCOR analysis covered broadly two categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions, and (2) assessment of source terms i.e., timing and magnitude of fission product releases to the environment. The development of the MELCOR calculation matrix was based on the CPRR alternatives defined by the accident sequence analysis.

The MELCOR analysis investigated detailed accident progression, source terms, and the containment response for representative Mark I and Mark II containment designs following an ELAP. The selection of accident scenarios considered for MELCOR analyses is informed by the recent state-of-the-art reactor consequence analysis or SOARCA, the Fukushima accident reconstruction study, and also by the work documented in SECY-12-0157. The representative Mark I containment selected was Peach Bottom Unit 2 like configuration, and the representative Mark II was Lasalle like configuration. The calculation matrix for Mark I included sensitivities to: (1) mode of venting; (2) status of RPV depressurization; (3) mode of FLEX water injection (to RPV or drywell); and (4) water management (i.e., water injection control by throttling flow). The matrix for the Mark II analysis included a subset of the Mark I matrix based on the insights from the Mark I MELCOR calculations. Additionally, for the Mark II analysis, sensitivities were performed to examine the impact of the pedestal and lower cavity designs among the fleet by modifying the base model.

The outcome of MELCOR analysis for the first category includes containment temperature and pressure signatures and hydrogen distribution in the containment, reactor building, vent line, etc. - all indicative of the state of containment vulnerability under severe accident conditions. These quantities provide needed information to assess containment integrity and also provide technical insights for developing staff guidance for the severe accident capable hardened vent Order EA-13-109. The outcome of the MELCOR analysis for the second category includes estimates of fission products release to the environment.

There was no fundamental shift in the scope and technical approach with regard to MELCOR analysis performed in support of the CPRR rulemaking (SECY-15-0085) when compared to what was done in SECY-12-0157. The technical approach in both cases considered best estimate modeling of accident progression, and incorporated both preventative and mitigative accident management measures including venting, water addition and/or water management, as well as the option of using engineered filters. However, it is important to recognize that in SECY-12-0157, water addition was considered in a general way as the industry's post-Fukushima severe accident management strategies were still evolving and the concept of severe accident water addition (SAWA) and severe accident water management (SAWM) had not yet emerged. Moreover, industry's approach to adapt its flexible coping strategy (FLEX), initially developed to meet 10 CFR 50.54(hh)(2) requirements, was being formulated for severe accident mitigation applications at the time. In contrast, during the effort leading to SECY-15-0085, these various concepts and severe accident management measures became more mature. The technical analysis documented here was informed by this new development, and the analysis resulted in findings that are no longer supportive of the recommendation in SECY-12-0157 related to an external engineered filter.

The offsite consequences were calculated using MACCS with site-specific population, economic, land use, weather, and evacuation data for a reference Mark I site and a reference Mark II site. The Peach Bottom Atomic Power Station and the Limerick Generating Station were selected as the site-specific reference models for offsite consequence analysis to enable greater modeling fidelity for the high population sites (Peach Bottom has the second highest population within a 50 mile radius among the 15 Mark I sites and Limerick has the highest population within a 50 mile

radius among the five Mark II sites). Offsite consequence calculations were run for the source terms generated by MELCOR corresponding to different CPRR accident management strategies following an ELAP event. The results of offsite consequence analysis were used to assess relative public health risk reduction associated with various containment protection and release reduction measures in terms of a variety of consequence measures including individual early and latent fatality risk, population dose, land contamination, economic cost, and displaced population. Like the MELCOR analysis, there was no fundamental shift in the scope and technical approach with regard to MACCS analysis performed in support of the CPRR rulemaking (SECY-15-0085) when compared to what was done in SECY-12-0157. The analysis in both cases used source term estimates from MELCOR to calculate with MACCS atmospheric transport and dispersion, protective actions, exposures, and resulting offsite consequences.

The quantitative results from the current MACCS analysis and the analysis included in SECY-12-0157 are quite similar. However, it is important to recognize that the second top-level consideration concerning performance criteria has important implications on the final outcome of the technical analysis in SECY-15-0085. Though not explicitly stated in SECY-12-0157, the staff's previous analysis effort to address containment venting and filtration implicitly assumed decontamination factor (DF) as a performance criterion. Specifically, a DF value of 1000 (equivalent to one-tenth of one percent cesium release to the environment – a measure related to latent cancer fatality risk and land contamination) was targeted in the previous work, consistent with the international nuclear safety practices and guidelines. In contrast, the top-level performance criterion used in SECY-15-0085 is QHO – a measure related to early and latent fatality risk. The MACCS results for the CPRR alternatives showed that not only is there essentially zero individual early fatality risk for all cases analyzed in SECY-15-0085 but also the individual latent cancer fatality risk is orders of magnitude lower than the QHO level.

Important findings and key insights from the technical work are delineated below:

- Venting of Mark I and Mark II containments effectively prevents containment overpressure failure. Pre-core damage anticipatory venting reduces the containment base pressure at the time of core damage and results in a delay when post core damage venting is required. Post-core damage containment venting is efficient in purging hydrogen and other non-condensables from the containment.

- Venting alone, however, is not adequate, as it does not prevent other modes of containment failure such as liner melt-through and over-temperature failure of the upper drywell head, bypass of the suppression pool and direct release of radioactivity to the environment. A combination of venting and water injection is required to prevent such failures, and the current work provides a sound technical basis to that effect thus supporting the adequate protection argument.

- Addition of water either into the RPV or the drywell has the following benefits: (1) cooling of the core debris and containment atmosphere; (2) preventing over-temperature failure of the upper drywell head; (3) preventing and/or delaying liner melt through in Mark I containments; (4) maintaining a steam inerted atmosphere which can preclude an energetic hydrogen combustion; and (5) mitigating radiological releases as it effectively provides means for fission product scrubbing.

- The environmental releases from Mark II containments are in general comparable to or lower than those calculated from Mark I containments. Sensitivity analysis performed to investigate variations in lower cavity configurations of Mark II containments indicate the

environmental releases for all configurations are within the range of releases predicted for Mark I containments.

- The major contribution to seismically induced ELAP is from earthquakes with ground motion exceeding the plant design basis (the safe shutdown earthquake). Specifically, earthquakes with peak ground accelerations in the range of 0.3 to 0.75g are the major contributors.

- Significant contributors to CDF include seismic failures of the batteries, DC switchgear, and the EDGs and their supporting equipment. Failure of the portable FLEX pump and failure to start of the RCIC pump are also significant contributors. CDF is not particularly sensitive to human error probabilities for in-control-room and ex-control-room operator actions.

- The estimated mean individual latent cancer fatality risk (0-10 miles) is more than two orders of magnitude below the NRC Safety Goal QHO. The risk is low because the core-damage frequency is low and the conditional latent cancer fatality risk is low. The range of parametric uncertainty in the risk estimates is more than one order of magnitude, and is largely driven by uncertainty in the seismic hazard curves.

- The estimated individual early fatality risk is essentially zero in all cases and for all alternatives considered, consistent with the findings previously in the State-of-the-Art Reactor Consequence Analyses (SOARCA) and spent fuel pool (SFP) consequence studies. This risk remains unchanged for a wide range of sensitivity analysis.

- The release to the environment is delayed long enough after the accident initiation to allow time for the emergency planning zone (EPZ) population to evacuate. Therefore the ILCF risk is dominated by long-term phase exposures to slightly contaminated areas (under the 500 mrem per year habitability criterion threshold). A larger release may displace more people for more time, and therefore incur a larger societal cost, but the health risk to the public, measured in ILCF risk, is effectively capped by the habitability criterion.

In general, larger releases of radioactive materials to the environment displace more people for more time, and therefore incur larger societal costs. However, for a larger release, the cancer fatality risk to the public shows a nonlinear response because protective actions are in place primarily to reduce exposures (habitability criterion) at the tradeoff of other societal costs such as land contamination, displaced population, and economic losses. Releases that span a longer duration were often seen to result in higher societal costs because longer durations allow more time for the wind to shift direction and thus spread plumes in more directions.

- The potential effectiveness of an external filter on reducing the environmental release is heavily influenced by release pathways. For accident cases in which the entire release flows through a vent pathway, the external filter can reduce the environmental release substantially. For accident cases resulting in a bypass, there is less benefit from an external filter because some of the release may bypass the venting system. Generally, while an engineered filter might accrue additional incremental benefits in terms of further reducing the long-term public health risk, it is not warranted for adequate protection as significant margins exist between estimated plant risks and the NRC established safety goals.

- Sensitivity calculations do not change any of the consequence analysis insights related to the QHO metrics. Individual early fatality risk remains essentially zero for all sensitivity calculations performed, and ILCF risk remains well below the QHO, even in calculations assuming a larger habitability criterion (e.g., 2 rem per year instead of 500 mrem per year).

In summary, the work documented in this NUREG report provides the technical basis to address containment venting and filtration issue (NTTF 5.1) for BWRs with Mark I and Mark II containments. The results support the overall conclusion in SECY-15-0085 that no additional regulatory action beyond the implementation of the severe accident capable vent order EA-13-109 is required based on adequate protection. The analysis confirms that significant margins exist between estimated plant risks that might be influenced by improvements in containment performance and the NRC established safety goals. However, these margins may be eroded somewhat when considering other accident scenarios and/or precursor events in any reasonable combination, and will likely retain the public health risks to an acceptable value from the adequate protection standpoint. That said, the NRC will continue to assess information emerging from ongoing international research activities on containment performance, and will continue to engage in long-term activities to enhance safety under established research programs.

Coincident with the analyses by the NRC staff and industry related to the CPRR rulemaking and related Orders, licensees were developing revisions to the severe accident managmenet guidelines (SAMGs) to address lessons learned from the Fukushima accident. The analyses performed to address issues related to containment venting and severe accident water addition provided valuable insights and supported actual revisions to the SAMGs for plants with Mark I and Mark II containments. This incorporation of insights from the modelling of beyond-design-basis events and severe accidents into plant guidance documents provides a useful example of the potential benefits of the efforts by the NRC and industry to develop improved analytical capabilities. Another example is the use of the results and technical insights in formulating the regulatory basis for the mitigation of beyond design basis events rulemaking. This rulemaking, though not imposing any regulatory footprint on SAMG, nevertheless directs the NRC staff to provide periodic oversight to industry’s SAMG implementation through NRC’s updated Reactor Oversight Program (ROP). A third and related example is that of a recent initiative in many Organization for Economic Cooperation and Development (OECD) countries and a collective effort in the OECD program of work to develop technical insights on how the type of beyond design basis analysis work documented in this report can inform SAMG in a positive way and in so doing, enhance the safety of nuclear power plants.

ACKNOWLEDGMENTS

The work documented in the report was done by the staff of the Office of Nuclear Regulatory Research (RES) in response to a Commission direction to address containment venting and filtration for boiling water reactor (BWR) Mark I and Mark II containments. The initiative to address the issue was led by the Office of Nuclear Reactor Regulation (NRR) Japan Lessons Learned Directorate (JLD). RES interacted closely with the JLD staff, as well as staff from technical divisions of NRR and also staff from other NRC program offices in this effort. The authors wish to acknowledge valuable suggestions and advice from these offices. In particular, the authors wish to acknowledge the contribution of Rajender Auluck, Robert Beall, and William Reckley of JLD and also Aaron Szabo (former NRR staff) in this respect.

The technical work was a collaborative effort between the Division of Risk Analysis (DRA) and the Division of Systems Analysis (DSA) within RES. Besides the authors, other staff from these two divisions contributed to this collaborative effort in various ways, most notably by providing expert advice and guidance to the authors and by conducting a peer review of the product. The authors wish to acknowledge the contributions of Edward Fuller, Tina Ghosh, John Lane, Allen Notafrancesco, and Nathan Siu in this regard. The authors also acknowledge William Reckley from NRR who reviewed a draft version of the report. The authors further wish to acknowledge James Chang from DRA who provided insights into probabilistic risk assessment by performing scoping human reliability analysis, and Jesse Phillips from Sandia National Laboratories (SNL) who performed accident analysis of BWR Mark II containments. The authors also wish to acknowledge Nathan Bixler of SNL and Joseph Jones, formerly of SNL, for their support and peer review of the offsite consequence analysis and for their affirmation that it is consistent with SOARCA best practices modeling approaches. Finally, the authors acknowledge earlier contribution of the SNL staff in developing MELCOR input models for BWR Mark I and Mark II containments.

During the conduct of the technical work and preparation of the report, the authors benefitted greatly from interactions with the ACRS members and other external stakeholders. Their collective contribution in shaping the scope and nature of the report is gratefully acknowledged.

ABBREVIATIONS AND ACRONYMS

AC	alternating current
ACRS	Advisory Committee on Reactor Safeguards
ANS	American Nuclear Society
APET	accident progression event tree
ARTIST	Aerosol Trapping in a Steam Generator
ATD	atmospheric transport and dispersion
atm	atmosphere
AV	anticipatory venting
BEIR	Biological Effects of Ionizing Radiation
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners' Group
CCDP	conditional core damage probability
CDET	core-damage event tree
CDF	core damage frequency
CPIP	Containment Performance Improvement Program
CPRR	containment protection and release reduction
CRAC	Calculation of Reactor Accident Consequences
CRF	consequence reduction factor
CST	condensate storage tank
DC	direct current
DCF	dose conversion factor
DF	decontamination factor
DW	drywell
DWF	drywell first strategy
DW LMT	drywell liner melt through
EAS	emergency alert system
ECCS	emergency core cooling systems
EDG	emergency diesel generator
EDMG	extended damage mitigation guideline
ELAP	extended loss of alternating current power
EOP	emergency operating procedures
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPS	emergency power system
EPZ	emergency planning zone

ETE	evacuation time estimate
FCVS	filtered containment venting system
FGR	Federal Guidance Report
FLEX	flexible coping strategy
FP	flow path
FPT	fission product tests
FV	Fussell-Vesely
gpm	gallons per minute
Gy	gray
HEPs	human error probabilities
HFE	human failure events
HP	high pressure
HPCI	high pressure cooling injection system
HPS	Health Physics Society
HRA	human reliability assessment
HTC	heat transfer coefficient
ILCF	individual latent cancer fatality
IPE	individual plant examination
IPEEE	individual plant examination of external events
JLD	Japanese lessons learned directorate
LCF	latent cancer fatality
LMT	liner melt through
LNT	linear no threshold
LOOP	loss of offsite power events
LP	low pressure
LTSBO	long-term station blackout
LWR	light water reactor
m	meter
MAAP	Modular Accident Analysis Program
MACCS	MELCOR Accident Consequence Code System
MCCI	molten core concrete interactions
MDBDE	mitigation of beyond-design-basis event
MP	medium pressure
mph	miles per hour

MSIV	main steam isolation valve
MSL	main steam line
MSLCR	main steam line creep rupture
mSv	millisievert
NCG	non-condensable gas
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSRC	National SAFER Response Centers
NTTF	Near Term Task Force
OECD	Organization for Economic Cooperation and Development
OLO	open at PCPL and leave open
ORO	offsite response organization
PAG	protective action guide
PAR	protective action recommendation
PCPL	primary containment pressure limit
PDS	plant damage state
PEMA	Pennsylvania Emergency Management Agency
PRA	probabilistic risk assessment
PRT	pressure relief tank
PWR	pressurized water reactor
QHO	qualitative health objective
RAW	risk achievement worth
RCIC	reactor core isolation cooling
RES	Office of Nuclear Regulatory Research
ROP	reactor oversight program
RPV	reactor pressure vessel
SAE	Site Area Emergency
SAGs	severe accident guidelines
SAMG	severe accident management guidelines
SAWA	severe accident water addition
SAWM	severe accident water management
SBO	station blackout
scfh	standard cubic feet per hour
SFP	spent fuel pool
SG	steam generator

SGTR	steam generator tube rupture
SNL	Sandia National Laboratories
SOARCA	State of the Art Reactor Consequence Analyses
SPAR	standardized plant analysis risk
SRM	staff requirements memorandum
SRV	safety relief valve
SV	safety valve
TAF	top of active fuel
UA	uncertainty analysis
URC MINALT	upper reactor cavity ablation depth reaching minimum altitude of the concrete
VC	vent cycling
WWF	wetwell first strategy

1 INTRODUCTION

This report documents the results of research conducted by the staff of the Office of Nuclear Regulatory Research (RES) to support the agency initiative to address the containment venting issue for the boiling water reactor (BWR) Mark I and Mark II containments. The venting issue was identified by the Near Term Task Force (NTTF), put together in the aftermath of the Fukushima Dai-ichi nuclear power plant accident in Japan, for all containment types, but prioritized as a short-term action item for the two containment types mentioned above. The initiative to address the venting issue was led by the Office of Nuclear Reactor Regulation (NRR) Japan Lessons Learned Directorate (JLD).

The RES work focused on developing the technical basis for a potential rulemaking action on containment protection and release reduction (CPRR), previously also known as filtered containment venting system (FCVS). The RES work covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by extended loss of ac power (ELAP) due to internal events and seismic events deemed to be the most significant risk contributors; (2) accident analysis of these sequences and assessment of radiological source terms; and (3) analysis of consequences with particular emphasis on health effects, both short term and long term. The results of consequence analysis were used to assess relative public health risks (more appropriately, health risk reduction) associated with various containment protection and release reduction measures. The relative risk measures were used by NRR to perform regulatory analysis for various mitigation options considered.

1.1 Containment Venting Issue – A Historical Perspective

The containment venting issue is not new and has a long history behind it. The U.S. Nuclear Regulatory Commission (NRC) and nuclear industry have recognized the potential need to vent Mark I and Mark II containment designs to cope with severe accident conditions since at least the early 1980s. These containment designs as well as other pressure suppression containments have been shown to be capable of addressing the requirements related to the design-basis accidents. However, various studies (e.g., NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants” [1]) have shown that the Mark I and Mark II containments do not have the same margins of safety that other containments (e.g., large dry ones) have during accidents that exceed the conditions established by design basis events. In 1983, the NRC approved Revision 2 to the Boiling Water Reactor Owners’ Group (BWROG) Emergency Procedure Guidelines (EPG), which included guidance for operators to vent Mark I and Mark II containments in response to containment overpressure conditions. Though emergency procedures have existed since the 1980s for Mark I and Mark II containment venting systems for beyond-design-basis accidents, the NRC’s actions to date for operating reactors have not required containment venting systems for Mark I and Mark II containments be designed for severe accident conditions.

The NRC evaluated the possible imposition of additional containment functional requirements for operating reactors in other previous studies as well. Subsequent to the Three Mile Island Unit 2 nuclear plant core melt event in 1979, NUREG-0585, “TMI-2 Lessons Learned Task Force Final Report,” October 1979, [2] stated:

> *Available studies indicate that controlled venting of the containment to prevent failure due to overpressure could be an effective means of delaying ultimate containment failure by melting through. If appropriately filtered to partially decontaminate the gases that would be*

released in order to avoid overpressurization, such venting may significantly reduce the consequences and risk from core-melt accidents.

A controlled (and potentially filtered) release was identified as a favorable alternative to catastrophic failure of the containment.

In SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988, [3] the U.S. Nuclear Regulatory Commission (NRC) staff presented to the Commission its plan to evaluate potential generic severe accident containment vulnerabilities in a research effort entitled the "Containment Performance Improvement Program" (CPIP). This effort was predicated on the presumption that there are generic severe accident challenges to each light water reactor (LWR) containment type that should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features were warranted, and to confirm the adequacy of the existing Commission policy. These assessments were needed because of the uncertainty in the ability of LWR containments to successfully survive some severe accident challenges, as indicated by the results documented in NUREG-1150.

All LWR containment types were considered in the CPI program, beginning with the BWR Mark I containments. The potential improvements for Mark I containments were documented in NUREG/CR-5225 (including Addendum 1), "An Overview of BWR Mark-I Containment Venting Risk Implications," [4] and SECY-89-017, "Mark I Containment Performance Improvement Program," dated January 23, 1989 [5]. In the latter document, the staff described its findings associated with six areas of potential improvement for Mark I containments. These were: (1) hydrogen control, (2) alternate water supply for reactor vessel injection and containment drywell sprays, (3) containment pressure relief capability (venting), (4) enhanced reactor pressure vessel (RPV) depressurization system reliability, (5) core debris control, and (6) emergency procedures and training. Each area was evaluated to determine the potential benefits in terms of reducing the core melt frequency, containment failure probability, and offsite consequences. The staff provided cost-justification for, and recommended implementation of, all the aforementioned improvements with two exceptions: hydrogen control (beyond then the existing rule) and core debris control (i.e., feasibility of confining core debris through design of curbs in the drywell and curbs or weir walls in the torus room below the wetwell).

In the subsequent staff requirements memorandum (SRM), however, the Commission concluded that the majority of the staff's recommended safety improvements would be evaluated by licensees as part of the Individual Plant Examination (IPE) Program [6]. The only exception was the hardened vent capability recommendation. The Commission directed the staff to approve installation of hardened vents under the provisions of 10 CFR 50.59, "Changes, Tests, and Experiments," [7] for licensees that would voluntarily implement this improvement and perform a back-fit analysis for requiring a hard vent installation at those plants who declined voluntary installation. Thus, NRC issued Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," [8] to all licensees of BWRs with Mark I containments to encourage licensees to voluntarily install a hardened wetwell vent in September 1989 providing an example of an acceptable design that used the suppression pool to achieve as much reduction in effluent radioactivity as possible.

In response to the issuance of the generic letter, all Mark I licensees installed a version of a hardened vent under 10 CFR 50.59. Some licensees also installed a hardened vent branch line from the drywell. The Boiling Water Reactor Owners' Group (BWROG) developed a general design criteria document that was subsequently approved by the staff (with clarifications). The hardened vent was specifically to provide an exhaust line from the wetwell air space to a suitable release point (e.g., stack, reactor building or turbine building roof). The basic design objective of

the hardened vent was to mitigate the loss of decay heat removal accident sequence, and not for operation during a severe accident. Because the modifications to the plant were performed in accordance with the Code of Federal Regulations, 10 CFR 50.59, “Changes, tests and experiments,” detailed information regarding individual plant configurations was not submitted to the NRC staff for review.

In concluding the CPIP effort, the NRC determined that the low probability of severe accidents resulted in the costs of plant modifications beyond the installation of a hardened vent exceeded the calculated benefits and, as such, were not cost-justified for Mark I and Mark II containment designs. Legislators and regulators in other countries did impose additional requirements in the aftermath of the accidents at Three Mile Island and Chernobyl. In effect, those other regulatory authorities assessed filtered vents and other severe accident management strategies with an emphasis on the defense-in-depth argument and with less or no consideration of cost/benefit analyses.

The potential improvements for Mark II containments were published in NUREG/CR-5528, “An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements.” [9] Mark II containment vulnerabilities and potential improvements identified in this document were largely the same as those for Mark I containments. However, less definitive conclusions were reached regarding venting of Mark II containments for a number of reasons. The findings and recommendations for Mark II containments as well as other containment types were documented in SECY-90-120, “Recommendations of Containment Performance Improvement Program for Plants with Mark II, Mark III, Ice Condenser, and Dry Containment." [10] However, unlike for Mark I containments, no generic letter requiring containment improvement was issued for these other containment types.

1.2 Post-Fukushima Development

The events at the Fukushima Dai-ichi nuclear power plant brought the containment venting issue once again to the forefront. The accidents involved an extended loss of electrical power and heat-removal systems, resulting in containment pressures that exceeded the containment design pressure. Plant conditions at Fukushima Dai-ichi (e.g., loss of all electrical power or station blackout) hampered the efforts of operators to address the containment overpressure conditions using the installed venting systems, which ultimately contributed to the compromise of all fission product barriers and significant releases of radioactive material. The events highlighted the need for safety improvements for nuclear power plants related to beyond-design-basis natural hazards, and the resulting effects on plant systems and barriers from an extended loss of electrical power and access to heat removal systems.

In SECY-11-0137, “Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned,” dated October 3, 2011 [11], the NRC staff described its proposals for the regulatory actions to address containment venting related NTTF recommendations [12]. Venting containment can help prevent or delay the loss of, or facilitate recovery of, important safety functions such as reactor core cooling, reactor coolant inventory control, containment cooling, and containment pressure control. The insights gained from Fukushima Dai-ichi led the agency to impose additional requirements for reliable hardened venting systems for plants with Mark I and Mark II containments. As such, on March 11, 2012, the NRC issued an order (EA-12-050) [13] to all licensees of BWR facilities with Mark I and Mark II containment designs to require a reliable hardened vent.

The EA-12-050 order provided requirements to ensure reliable operation of the hardened venting system in support of strategies relating to the prevention of core damage. However, EA-12-050 did not include requirements for reliable operation under severe accident conditions; rather, it focused on requirements prior to the onset of core damage. As such, EA-12-050 did not prescribe the venting location (drywell or wetwell) as essentially all vent flow prior to RPV breach would pass through the suppression pool. Nevertheless, the existing emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and extended damage mitigation guidelines (EDMGs) for BWRs with Mark I and Mark II containments contain provisions for venting containment following core damage.

One of the six additional recommendations identified in SECY-11-0137, and further developed in SECY-12-0025 [14], was consideration of additional performance requirements, including filters for hardened containment vent systems for Mark I and Mark II containment designs. This would improve reliability during severe accident conditions and limit the release of radioactive materials if the venting systems were used after significant core damage had occurred. The NRC staff identified an additional issue in SECY-11-0137 related to possible modifications of the containment vents, including the addition of engineered filters. In the SRM for SECY-11-0137, dated December 15, 2011 [15], the Commission directed the NRC staff as follows:

> *The staff should quickly shift the issue of "Filtration of Containment Vents" from the "additional issues" category and merge it with the Tier 1 issue of hardened vents for Mark I and Mark II containments such that the analysis and interaction with stakeholders needed to inform a decision on whether filtered vents should be required can be performed concurrently with the development of the technical bases, acceptance criteria, and design expectations for reliable hardened vents.*

In accordance with the direction in SRM-SECY-11-0137, the additional issue of filtration of containment vents was merged with the Tier 1 issue of hardened vents for Mark I and Mark II containments to facilitate further analysis and interaction with stakeholders so as to inform the need and benefit of filtered vents. In SECY-12-0025, the staff explained that it needed to resolve technical and policy issues before regulatory action could be proposed that would require licensees to install filters, or change any other performance requirement, for hardened containment vent systems.

To support additional regulatory actions related to the performance of Mark I and Mark II containments during severe accidents, RES performed a systematic analysis of accident source terms and consequences in late 2011 and mid 2012. This analysis was conducted for a representative BWR plant with Mark I containment and for representative accident scenarios, predicated on an extended loss of alternating current power (ELAP) as in Fukushima. The analysis used NRC severe accident code MELCOR and the consequence code MACCS (MELCOR Accident Consequence Code System) for source term and consequence assessments, respectively, and was informed by lessons-learned and best practices from the State-of-the-Art Reactor Consequence Analysis (SOARCA) project [16]. The analysis provided an assessment of the sensitivity of the plant risks to selected accident management strategies, keeping in mind that such strategies were still evolving at the time. The MELCOR and MACCS simulations were used along with insights from previous studies (e.g., individual plant examinations, NUREG-1150, CPIP, severe accident mitigation alternatives) to evaluate the potential benefits of features in Mark I and Mark II containment designs. These designs included containment venting systems with and without engineered filters, and provision for water addition into the pressure vessel and/or the containment.

The technical analysis, documented in SECY-12-0157 [17], concluded that the plants with Mark I containments (and by extrapolation Mark II containments as well) would benefit from a containment venting and water addition strategy for a vast majority of severe accident sequences (whether the vent includes an engineered filter or not). In addition to its own assessments and analyses, the staff relied on information gained through interactions with various stakeholders including the nuclear industry, which provided insights to the NRC staff during several public meetings, and also through a report the Electric Power Research Institute published.

From a regulatory perspective, the staff presented four options in SECY-12-0157 for Commission consideration to address the containment venting issue. The options presented included: (1) maintaining the requirements established in Order EA-12-50 for reliable hardened vents and do nothing else; (2) issuing a new order requiring containment venting systems to be capable of operating under severe accident conditions; (3) issuing an order requiring containment venting systems capable of operating under severe accident conditions with additional external filtering feature to reduce release of radioactivity to environment through controlled release pathways; and (4) developing a performance-based severe accident management strategy for BWRs with Mark I and Mark II containments. The staff's regulatory analysis focused on Option 2 (severe accident capable vent order) and Option 3 (filtered vent order) as those options involved potential near term regulatory action, and recommended that the Commission approve Option 3. This recommendation was not based solely on quantitative analysis; rather, a combination of quantitative analysis and qualitative arguments invoking the long established policy statements by the Commission on severe accident, defense-in-depth, and other related topics.

The Commission directed the staff in SRM-SECY-12-0157 [18] to: (1) issue a modification to Order EA-12-050 requiring licensees with Mark I and Mark II containments to "upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions," and (2) develop technical basis and rulemaking for filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments. The NRC staff subsequently issued Order EA-13-109, "Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions" (ADAMS Accession No. ML13143A321) [19], to ensure that vents on BWR Mark I and II containments will remain functional in the conditions following a reactor core melt accident.

In parallel, the staff initiated additional technical work to develop technical basis for potential rulemaking involving filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments. As mentioned previously, this additional technical work by RES focused on developing the technical basis for a potential rulemaking action on containment protection and release reduction (CPRR) supporting SECY-15-0085, "Evaluation of the Containment Protection & Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (ADAMS Accession No. ML15005A1079) [20]. The technical work covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by extended loss of ac power (ELAP) due to internal events and seismic events deemed to be the most significant risk contributors; (2) accident analysis of these sequences and assessment of radiological source terms; and (3) analysis of consequences with particular emphasis on health effects, both short term and long term.

1.3 Content and Structure of Report

The report documents the outcome of the technical work to support regulatory basis of the potential CPRR rulemaking activity. The regulatory analysis for rulemaking activity is documented in an enclosure to SECY-15-0085.

Chapter 2 of the report documents the staff's accident sequence analysis that was used to select risk-significant accident sequences. This work involved development of a core damage event tree (CDET) and an accident progression event tree (APET), and binning of a rather large number of possible end states to a manageable number of categories with similar outcomes. Another important aspect of the accident sequence analysis is the development of CPRR alternatives considered in the regulatory basis. In accordance with NTTF Recommendation 5.1, the evaluation of CPRR alternatives was focused on accidents that are initiated by a prolonged station blackout (SBO) event, i.e., an extended loss of alternating current power (ELAP) event with loss of all offsite and onsite ac power sources that lasts longer than the SBO coping duration specified in 10 CFR 50.63, "Loss of all alternating current power." [21]

Chapter 3 describes in detail the MELCOR analysis, the scope of which falls broadly into two categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions, and (2) assessment of source terms (i.e., timing and magnitude of fission product releases to the environment). The development of the MELCOR calculation matrix was tied to the regulatory basis alternatives mentioned in the previous paragraph.

The outcome of MELCOR analysis for the first category includes containment temperature and pressure signatures and hydrogen distribution in the containment, reactor building, vent line, etc., all indicative of the state of containment vulnerability under severe accident conditions. These quantities provided needed information to assess containment integrity and also provided the technical basis for developing staff guidance, for example, the severe accident capable hardened vent Order EA-13-109. The outcome of the MELCOR analysis for the second category is environmental source term release estimates, which were used to calculate offsite consequences.

Chapter 4 describes in detail the consequence analysis using MELCOR Accident Consequence Code System (MACCS). The code calculates offsite consequences for the source terms generated by MELCOR corresponding to different CPRR accident management strategies following an ELAP event. The results of consequence analysis were used to assess relative public health risks (more appropriately, health risk reduction) associated with various containment protection and release reduction measures. The staff evaluated the conditional offsite consequences associated with the CPRR alternatives discussed in the accident sequence analysis and the results are described in this chapter.

Chapter 5 provides the main results of the risk integration, i.e., risk estimates corresponding to regulatory analysis alternatives and sub-alternatives considered in the study and described in Chapter 2. The integration process involved combining the outcome of MELCOR analysis (Chapter 3) and MACCS analysis (Chapter 4).

Key insights from the technical work are summarized in Chapter 6 in three major areas: accident sequence analysis, source term assessment using MELCOR, and consequence assessment using MACCS. It is important to note that the accident sequence analysis results show a low core damage frequency from an ELAP event for BWR plants with Mark I and Mark II containments, and provide insights into the relative contributions of various factors (e.g., external hazards, equipment failures, human errors, etc.) to overall core damage frequency. The MELCOR results show both

water addition and venting are required to maintain containment integrity and reduce source terms, and finally, MACCS results show essentially zero early fatality risk and sufficiently low individual latent cancer fatality (ILCF) risk with two order of magnitude or more safety margin relative to quantitative health objective (QHO).

As with any complex analysis work involving multiple technical disciplines, integration of the work requires a logical structure. To that end, the CPRR technical basis work adopted a technical approach that consists of the following steps:

- Accident sequence analysis that includes quantitative risk estimates and qualitative risk insights for various CPRR strategies

- Development of an accident progression calculation matrix, consistent with the risk evaluation, for analyzing the range of different accident management strategies

- Accident progression and source term calculations using MELCOR

- Offsite consequence calculations using MACCS for selected MELCOR source terms

- Integration of the results of the steps above to generate frequency-weighted offsite consequences corresponding to each of the different CPRR strategies

The technical work in each of the three major disciplines was based on a number of assumptions or considerations, which are delineated in detail in the respective chapters describing the work. It is, however, important to highlight some top-level considerations. These are:

- SRM-SECY-12-0157 direction that the regulatory basis should assume the benefits of severe accident capable hardened venting systems (EA-13-109) that would accrue equally to engineered filters and to filtration strategies

- Consideration of a variety of performance criteria such as a decontamination factor, equipment and procedure availability similar to those required to implement 10 CFR 50.54(hh) [22], or other measures that may be developed during the stakeholder engagement

- Consideration of requirements associated with measures to enhance the capability to maintain containment integrity and to cool core debris

The first of these considerations essentially set the boundary conditions for all technical analyses. For example, the CPRR alternatives defined in the risk analysis were predicated on the established capability of post-core damage containment venting. In two of four alternatives selected in the risk analysis (see Chapter 2 for details), additional capability to inject water (in the core and/or the containment) was not considered whereas, in the other two alternatives, water injection was probabilistically considered. Moreover, in one of the two latter alternatives, an external engineered filter was considered as a further accident management feature, and its potential benefit was investigated. To that end, the MELCOR matrix for source term calculations was mapped to these four CPRR alternatives, and subsequent MACCS consequence calculations were mapped to these alternatives as well.

The second consideration concerning performance criteria has important implications on the final outcome of the technical analysis. Though not explicitly stated in SECY-12-0157, the staff's previous analysis effort to address the containment venting issue implicitly assumed decontamination factor (DF) as a performance criterion. Specifically, a DF value of 1000 (equivalent to one-tenth of one percent cesium release to the environment – a measure related to latent cancer fatality risk and land contamination) was targeted in the previous work, consistent with the international nuclear safety practices and guidelines. In contrast, the top-level performance criterion used in SECY-15-0085 is QHO – a measure exclusively related to public health risk. As will be seen later in the document, for all accident scenarios considered in the analysis, the risk was assessed to be well below the QHO limit. This obviated the need for otherwise meeting a more stringent target of a DF value of 1000.

The third consideration associated with measures to enhance the capability to maintain containment integrity and to cool core debris, likewise, has important implications. In SECY-12-0157, water addition was considered, albeit in a general way, as the industry's post-Fukushima severe accident management strategies were still evolving and the concept of severe accident water addition (SAWA) and severe accident water management (SAWM) did not yet emerge. Moreover, industry's approach to adapt its flexible coping strategy (FLEX), initially developed to meet 10 CFR 50.54(hh) requirements, was being formulated for severe accident mitigation application at the time. In contrast, during the effort leading to SECY-15-0085, these various concepts and severe accident management measures became more mature. The technical analysis documented here was informed by this new development, and the analysis resulted in findings that are far less supportive of the recommendation in SECY-12-0157 related to an external engineered filter. It was recognized that while an engineered filter might accrue additional incremental benefits in terms of further reducing the long-term public health risk, its implementation could not be justified as a cost-beneficial safety enhancement measure.

A few other high level assumptions, specific to each of the three areas of analysis (accident sequence analysis, MELCOR, and MACCS) are worth noting here. The accident sequence analysis was informed by: (1) the lessons learned from the events at the Fukushima Dai-ichi Nuclear Power Plant; (2) the accident management alternatives being contemplated by the industry; (3) the current state of knowledge of severe accident progression and mitigation alternatives in a BWR; and (4) the experience gained from the previous effort documented in SECY-12-0157. The human reliability aspect was considered in the event tree formulation and despite an initial attempt to develop a comprehensive human reliability assessment (HRA), only a bounding approach to incorporating HRA into the accident sequences was implemented at the end.

For source term assessment, the version of MELCOR used is consistent with other recent MELCOR applications such as the State-of-the-Art Reactor Consequence Analysis (SOARCA) and the spent fuel pool (SFP) study [23]. MELCOR embodies the current state of knowledge of severe accident progression and mitigation; however, as with any other complex analysis tools, the models in MELCOR are based, in part, on phenomenological studies and, to a degree, on physical abstraction. As such, modeling and parametric uncertainties in MELCOR are recognized and addressed through uncertainty analysis and/or sensitivity studies.

For consequence analysis, likewise, the version of MACCS used is consistent with other applications such as SOARCA and SFP. None of these applications put economic consequence on the same footing as the public health consequence, consistent with a recent commission deliberation on the subject (see SRM-SECY-12-0110) [24]. On a specific technical note, aqueous release paths are currently not modeled in consequence analysis.

1.4 References for Chapter 1

1. U.S. Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, December 1990.

2. U.S. Nuclear Regulatory Commission, “TMI-2 Lessons Learned Task Force Final Report,” NUREG-0585, October 1979.

3. U.S. Nuclear Regulatory Commission, “Integration Plan for Closure of Severe Accident Issues,” SECY-88-147, May 25, 1988.

4. U.S. Nuclear Regulatory Commission, “An Overview of BWR Mark-I Containment Venting Risk Implications,” NUREG/CR-5225.

5. U.S. Nuclear Regulatory Commission, “Mark I Containment Performance Improvement Program,” SECY-89-017, January 23, 1989.

6. U.S. Nuclear Regulatory Commission, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” NUREG-1560, November 1996.

7. Code of Federal Regulations, Title 10, Part 50, “Changes, Tests, and Experiments,” 10 CFR 50.59.

8. U.S. Nuclear Regulatory Commission, “Installation of a Hardened Wetwell Vent,” Generic Letter 89-16, September 1989.

9. Kelly, D.L., et al., “An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements in Performance,” NUREG/CR-5528, Nuclear Regulatory Commission, Washington DC, July 1990.

10. U.S. Nuclear Regulatory Commission, “Recommendations for Containment Performance Improvement Program for Plants with Mark II, Mark III, Ice Condenser, and Dry Containment,” SECY-90-120, 1990.

11. U.S. Nuclear Regulatory Commission, “Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned,” SECY-11-0137, October 3, 2011.

12. U.S. Nuclear Regulatory Commission, “Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” July 11, 2011, ADAMS Accession No. ML111861807.

13. U.S. Nuclear Regulatory Commission, “Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents,” Order EA-12-050, March 12, 2012, ADAMS Accession No. ML12054A694.

14. U.S. Nuclear Regulatory Commission, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami,” SECY-12-0025, February 17, 2012.

15. U.S. Nuclear Regulatory Commission, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," SRM-SECY-11-0137, December 15, 2011.

16. Bixler N., et al. "State-of-the-Art Reactor Consequence Analyses (SOARCA) Project Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110 Vol. 1, Rev. 1, Nuclear Regulatory Commission, Washington DC, May 2013.

17. U.S. Nuclear Regulatory Commission, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments (REDACTED VERSION)," SECY-12-0157, November 26, 2012, ADAMS Accession No. ML12345A030.

18. U.S. Nuclear Regulatory Commission, "Consideration Of Additional Requirements For Containment Venting Systems For Boiling Water Reactors With Mark I and Mark II Containments," SRM-SECY-12-0157, March 2013.

19. U.S. Nuclear Regulatory Commission, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," Order EA-13-109, June 6, 2013, ADAMS Accession No. ML13143A334.

20. U.S. Nuclear Regulatory Commission, "Evaluation of the Containment Protection & Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26)," SECY-15-0085, June 18, 2015, ADAMS Accession No. ML15022A218.

21. Code of Federal Regulations, Title 10, Part 50, "Loss of All Alternating Current Power," 10 CFR 50.63.

22. Code of Federal Regulations, Title 10, Part 50, "Conditions of Licenses," 10 CFR 50.54(hh).

23. U.S. Nuclear Regulatory Commission, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," NUREG-2161, September 2014.

24. U.S. Nuclear Regulatory Commission, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework" SECY-12-0110, August 14, 2012.

2 ACCIDENT SEQUENCE ANALYSIS

This section details the accident sequence analysis that was performed to support the draft regulatory basis attached to SECY-15-0085 [1]. Using a semi-plant-specific approach, the risk evaluation assessed the risk impacts of potential containment protection and release reduction (CPRR) strategies for severe accidents initiated by extended loss of ac power (ELAP) events due to internal events and seismic events occurring at operating boiling water reactors (BWRs) with Mark I containment designs and reactor core isolation cooling (RCIC) systems. The focus on ELAP events is consistent with the wording of NTTF Recommendation 5.1, which alludes to "prolonged SBOs." In addition to internal events and seismic events, ELAPs may also be caused by other types of external events (e.g., fires, floods, high winds); however, these were not included due to the amount of time and effort need to the collect the site-specific information needed to develop an appropriate logic model. It is also recognized that CPRR strategies may be beneficial in mitigating severe accidents that do not involve ELAP. As a result, the accident sequence analysis underestimates the potential benefits of CPRR strategies. NTTF Recommendation 5.1 applies to all BWRs with Mark I and Mark II containment designs. However, the accident sequence analysis excluded BWRs with Mark I containment designs and isolation condensers and BWRs with Mark II containments because the staff determined that it could develop sufficient information to inform the regulatory analysis by evaluating only the BWRs with Mark I containment designs and RCIC systems. This reduced the time and effort needed to complete the draft regulatory evaluation attached to SECY-15-0085. Table 2-1 lists all BWRs with Mark I and Mark II containment designs, and indicates their disposition in the accident sequence analysis.

Table 2-1 Consideration of BWR Mark I and Mark II plants in the accident sequence analysis

Included in the Accident Sequence Analysis	Excluded from the Accident Sequence Analysis
<u>BWR Mark I and RCIC:</u> Browns Ferry, Units 1, 2 and 3 Brunswick, Units 1 and 2 Cooper Duane Arnold Fermi, Unit 2 FitzPatrick[2] Hatch, Units 1 and 2 Hope Creek, Unit 1 Monticello Peach Bottom, Units 2 and 3 Pilgrim, Unit 1[2] Quad Cities, Units 1 and 2	<u>BWR Mark I and RCIC:</u> Vermont Yankee[1] <u>BWR Mark I and Isolation Condensers:</u> Dresden, Units 2 and 3 Nine Mile Point, Unit 1 Oyster Creek <u>BWR Mark II:</u> Columbia La Salle, Units 1 and 2 Limerick, Units 1 and 2 Nine Mile Point, Unit 2 Susquehanna, Units 1 and 2

[1] Shortly after the accident sequence analysis was commenced, the owner of the Vermont Yankee plant announced that it would cease operations in the fourth quarter of 2014; as a result, Vermont Yankee was excluded from the scope of the accident sequence analysis.

[2] Shortly after SECY-15-0085 was issued, the owners of the FitzPatrick and Pilgrim plants announced that these plants would cease operations in 2016 and 2019, respectively.

During the risk integration effort, the following risk metrics for each of the 20 regulatory analysis sub-alternatives defined in SECY-15-0085:

- Individual early fatality risk (0-1.3 miles and beyond)
- Individual latent cancer fatality risk (0-10 miles, 0-50 miles, and 0-100 miles)
- Population dose risk (0-50 miles and 0-100 miles)
- Offsite costs (0-50 miles and 0-100 miles)
- Land exceeding long-term habitability criterion (0-50 miles and 0-100 miles)
- Population subject to long-term protective actions (0-50 miles and 0-100 miles)

The following sections describe the rationale used to identify potential CPRR strategies (including their relationship to the options provided in SECY-12-0157 [2] and the alternatives provided in SECY-15-0085), explain the technical approach used to estimate the risk metrics, present the results obtained, discuss the sensitivity and parametric uncertainty analyses that were conducted, and provide the conclusions of the accident sequence analysis.

2.1 <u>Identification of CPRR Strategies</u>

The CPRR accident sequence analysis began in March 2013 with the issuance of the staff requirements memorandum on SECY-12-0157 and concluded in June 2015 with the issuance of SECY-15-0085. During this period, the scope of the accident sequence analysis evolved as the staff identified various CPRR strategies, conducted analysis, solicited stakeholder input during public meetings, and incorporated the impacts of Fukushima-related regulatory actions. The following sections describe the factors that influenced the selection of CPRR strategies addressed by the accident sequence analysis, and identify the specific combinations of CPRR strategies that were addressed.

2.1.1 Influencing Factors

A CPRR strategy is an action taken prior to or during the course of a severe accident to protect the containment's structural integrity or to reduce the amount of radioactive material released to the environment. Examples include containment venting following core damage (a containment protection strategy) and the installation of engineered filters on the containment vent lines (a release reduction strategy). High-level strategies (e.g., containment venting) may be divided into more specific categories according to how they are implemented (e.g., wetwell venting or drywell venting). In order to conduct the accident sequence analysis, it is essential to define a set of possible CPRR strategies and to specify their implementation details so that a probabilistic logic model can be developed and quantified. The major factors that influenced the set of CPRR strategies addressed in the accident sequence analysis are summarized below.

2.1.1.1 Commission Direction

SECY-12-0157 identified four options to address the issue of containment venting for BWRs with Mark I and Mark II containments, which are reproduced below verbatim:

1. Reliable hardened vents (Status Quo): Continue with the implementation of Order EA-12-050 [3] for reliable hardened vents to reduce the likelihood of core damage and failure of BWR Mark I and Mark II containments and take no additional regulatory action to improve their ability to operate under severe accident conditions or to require the installation of an engineered filtered vent system.

2. Severe accident capable vents order: Upgrade or replace the reliable hardened vents required by EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions.

3. Filtered vents order: Design and install an engineered filtered containment venting system that is intended to prevent the release of significant amounts of radioactive material following the dominant severe accident sequences at BWRs with Mark I and Mark II containments.

4. Severe accident confinement strategy: Pursue development of requirements and technical acceptance criteria for confinement strategies and require licensees to justify operator actions and systems or combinations of systems, such as suppression pools, containment sprays, and separate filters to accomplish the function and meet the requirements.

In response to SECY-12-0157, the Commission:

1. Approved Option 2 to issue a modification to Order EA-12-050 to require licensees of Boiling Water Reactors (BWRs) with Mark I and Mark II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions,

2. Approved the development of technical bases and rulemaking for filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments to further consider Option 3 and Option 4, and

3. Directed that the technical bases should assume the installation of severe accident capable hardened venting systems ordered under Option 2 and, as a consequence of that action, should assume that the benefits of these vents accrue equally to engineered filters and to filtration strategies.

In order to respond to the Commission's direction, three high-level CPRR strategies were identified for subsequent analysis:

1. Severe accident containment venting (containment protection),

2. Installation of engineered filters on the containment vent lines (release reduction), and

3. Severe accident mitigation strategies, including termination of core melt progression and core debris cooling (containment protection and release reduction).

2.1.1.2 Post-Accident Water Injection

The technical analysis performed to support SECY-12-0157 identified the need for post-accident water injection, which provides the following benefits:

1. Water injection to the reactor pressure vessel (RPV) can arrest a severe accident before vessel breach.

2. Water injection to the containment, either directly into the drywell (DW) or indirectly into the RPV (which subsequently flows through the vessel breach to the drywell), can prevent loss of containment integrity due to liner melt-through.

3. Water injection reduces the temperature of the drywell atmosphere, which simplifies the design of a severe accident capable drywell venting system.

The staff identified post-accident water injection as a potential CPRR strategy early in the accident sequence analysis.

2.1.1.3 Actions Taken Prior to Core Damage

During a public meeting held in the spring of 2013, industry described an early (i.e., prior to core damage) venting strategy termed “anticipatory venting,” which was developed as part of the mitigating strategies required by Order EA-12-049 [4]. Prior to this meeting, it was assumed that containment venting would be initiated when the containment pressure approached the primary containment pressure limit (PCPL), approximately 60 psig. Anticipatory venting initiated prior to core damage at a pressure substantially lower than the PCPL would help to cool the wetwell inventory and, thus, prolong the operation of the RCIC pump. As a result of this meeting, the staff identified anticipatory venting as a potential CPRR strategy.

2.1.1.4 Order EA-13-109

As directed by the Commission in the staff requirements memorandum on SECY-12-0157, the staff rescinded Order EA-12-050 and issued Order EA-13-109 [5] to implement requirements for reliable hardened containment vents capable of operation under severe accident conditions. A phased approach to implementation was used to minimize delays in implementing the requirements originally imposed by EA-12-050. Phase 1 involves upgrading the venting capabilities from the containment wetwell to provide reliable, severe accident capable hardened vents to assist in preventing core damage and, if necessary, to provide venting capability during severe accident conditions. Phase 2 involves providing additional protections for severe accident conditions through installation of a reliable, severe accident capable drywell vent system or the development of a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions.

If the post-accident water injection flow rate exceeds what is needed to replenish the wetwell inventory lost due to venting, then wetwell venting capability will eventually be lost because the wetwell vent connection will become submerged. Thus, the operating mode of post-accident water injection was identified as a potential CPRR strategy having two alternatives:

1. Severe accident water addition (SAWA): no effort is taken to prevent submergence of the wetwell vent

2. Severe accident water management (SAWM): the post-accident water injection flow is throttled as needed to prevent submergence of the wetwell vent.

2.1.2 Regulatory Analysis Sub-Alternatives

As the analysis progressed, specific combinations of CPRR strategies were termed “options” or “alternatives,” and were identified by a scheme proposed by industry. This approach was not entirely satisfactory because it caused confusion between the CPRR strategy options/alternatives

being considered in the accident sequence analysis and the options/alternatives presented for the Commission's consideration in SECY papers. Moreover, in August 2014, the staff determined that the original rulemaking name (filtering strategies) no longer matched the purpose of the activity. The staff felt it was more logical to have the rulemaking reflect the two issues being analyzed – enhanced containment protection and release reduction. Accordingly, the various combinations of CPRR strategies were organized into a set of regulatory analysis alternatives and sub-alternatives in preparation for developing a SECY paper to respond to the staff requirements memorandum on SECY-12-0157, as indicated in the following outline:

1. No Action/Base Case (leave Order EA-13-109 in place)
2. Overpressurization Measures
   A. Make Order EA-13-109 generically applicable
3. Containment Failure Prevention Measures
   A. Water addition via RPV
   B. Water addition via DW
4. Release Reduction and Containment Failure Prevention Measures
   A. Filtration Strategies
      i. Vent Cycling
         (1) Water addition via RPV
         (2) Water addition via DW
      ii. Water Management
         (1) Water addition via RPV
         (2) Water addition via DW
      iii. Vent Cycling and Water Management
         (1) Water addition via RPV
         (2) Water addition via DW
   B. Small Engineered Filter
      i. Manual WWF before core damage, and manual WWF after core damage
         (1) Water addition via RPV
         (2) Water addition via DW
      ii. Manual DWF before core damage and manual DWF after core damage
      iii. Manual DWF before core damage and passive DWF after core damage
      iv. Passive DWF before core damage and passive DWF after core damage
   C. Large Engineered Filter
      i. Manual WWF before core damage, and manual WWF after core damage
         (1) Water addition via RPV
         (2) Water addition via DW
      ii. Manual DWF before core damage and manual DWF after core damage
      iii. Manual DWF before core damage and passive DWF after core damage
      iv. Passive DWF before core damage and passive DWF after core damage

Several features of this outline are noteworthy:

- The second regulatory analysis alternative was defined in order to consider the costs and benefits associated with making Order EA-13-109 generically applicable, i.e., the second regulatory analysis alternative explores a change in the "regulatory footprint" concerning severe accident containment venting. There is no technical difference between the first and second regulatory analysis alternatives.

- A single sub-alternative (2A) was identified for the second regulatory analysis alternative; no other sub-alternatives for the second regulatory analysis alternative were defined (i.e., the "A" is superfluous).

- The outline contained ambiguities about some alternatives. Specifically, no post-accident water injection location was specified for sub-alternatives that utilize DWF venting and engineered filters.

Table 2-2 lists the complete and fully specified regulatory analysis sub-alternatives that were considered in the accident sequence. It also traces each regulatory analysis sub-alternative back to the options provided in SECY-12-0157.

Additionally, Table 2-2 maps each regulatory analysis sub-alternative onto the alternatives provided in SECY-15-0085:

1. Take no action (Order EA-13-109 implemented without additional regulatory actions)

2. Pursue rulemaking to make Order 13-109 generically applicable

3. Pursue rulemaking to address containment protection against multiple failure modes by making Order EA-13-109 generically applicable and requiring external water addition points (external to the reactor building) that would allow for post-accident water injection into the RPV or DW.

4. Pursue rulemaking to address both containment protection against multiple failure modes and release reduction measures for controlling releases through the containment venting systems. This alternative included making Order EA-13-109 generically applicable, requiring post-accident water injection, and reducing the fission products released from the containment by either implementing filtering strategies or installing engineered filters.

As the regulatory alternatives for CPRR were being developed, it became apparent that all licensees intended to comply with Phase 2 of Order EA-13-109 by implementing a post-accident water injection strategy. Thus, there is no technical difference among the first three alternatives presented in SECY-15-0085; rather, the first three alternatives in SECY-15-0085 explore differences in the “regulatory footprint” for various CPRR strategies.

Table 2-2 Regulatory analysis sub-alternatives

Index	Regulatory Analysis Sub-Alternative	SECY-12-0157 Option	SECY-15-0085 Alternative(s)	Before Core Damage				After Core Damage					
				Vent Priority	Venting Actuation	Venting Operation Mode	Reclose Vent if Core Damage is Imminent	Post-Accident Water Injection Location	Post-Accident Water Injection Operating Mode	Vent Priority	Venting Actuation	Venting Operation Mode	Filter Size
1	1	2	n/a	WWF	M	AV	yes	no	n/a	WWF	M	OLO	no
2	2A	2	n/a	WWF	M	AV	yes	no	n/a	WWF	M	OLO	no
3	3A	2	1,2,3	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	no
4	3B	2	1,2,3	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	no
5	4Ai(1)	4	4	WWF	M	AV	yes	RPV	SAWA	WWF	M	VC	no
6	4Ai(2)	4	4	WWF	M	AV	yes	DW	SAWA	WWF	M	VC	no
7	4Aii(1)	4	4	WWF	M	AV	yes	RPV	SAWM	WWF	M	OLO	no
8	4Aii(2)	4	4	WWF	M	AV	yes	DW	SAWM	WWF	M	OLO	no
9	4Aiii(1)	4	4	WWF	M	AV	yes	RPV	SAWM	WWF	M	VC	no
10	4Aiii(2)	4	4	WWF	M	AV	yes	DW	SAWM	WWF	M	VC	no
11	4Bi(1)	3	4	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	S
12	4Bi(2)	3	4	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	S
13	4Bii	3	4	DWF	M	AV	yes	DW	SAWA	DWF	M	OLO	S
14	4Biii	3	4	DWF	M	AV	yes	DW	SAWA	DWF	P	OLO	S
15	4Biv	3	4	DWF	P	OLO	no	DW	SAWA	DWF	P	OLO	S
16	4Ci(1)	3	4	WWF	M	AV	yes	RPV	SAWA	WWF	M	OLO	L
17	4Ci(2)	3	4	WWF	M	AV	yes	DW	SAWA	WWF	M	OLO	L
18	4Cii	3	4	DWF	M	AV	yes	DW	SAWA	DWF	M	OLO	L
19	4Ciii	3	4	DWF	M	AV	yes	DW	SAWA	DWF	P	OLO	L
20	4Civ	3	4	DWF	P	OLO	no	DW	SAWA	DWF	P	OLO	L

Venting Priority
DWF drywell first strategy
WWF wetwell first strategy

Venting Actuation
M manual
P passive (rupture disk)

Venting Operation Mode
AV anticipatory venting; open at 15 psig and leave open
OLO open at PCPL and leave open
VC venting cycling at PCPL with 10 psi band

Post-accident Water Injection Location
DW drywell via external connection
RPV reactor pressure vessel via external connection

Post-accident Water Injection Operating Mode
SAWA severe accident water addition
SAWM severe accident water management

Filter Type
L large (DF=1000)
S small (DF=10)

The staff considered a variety of implementation details associated with the CPRR strategies identified in the previous sections:

1. Venting priority
   a. Wetwell-first venting (WWF): the wetwell vent is preferential opened, with the drywell providing redundancy
   b. Drywell-first venting (DWF): the drywell vent is preferentially opened, with the wetwell vent providing redundancy

2. Venting actuation
   a. Manual: the plant operators open the vents
   b. Passive: the vents are provided with rupture discs, which improves their reliability since operator action is not required

3. Venting operation mode
   a. Before core damage
      i) Anticipatory venting (AV) at 15 psig or less
      ii) Open-and-leave-open (OLO) venting at the PCPL or lower
   b. After core damage
      i) Vent cycling (VC) at the PCPL within a 10 psi band
      ii) Open-and-leave-open (OLO) venting at the PCPL

4. Vent reclosure if core damage is imminent: yes or no

5. Post-accident water injection location: RPV or DW

2.2 Technical Approach

The technical approach used to develop the accident sequence analysis of potential CPRR strategies was based on simplified probabilistic risk assessment (PRA) methods. An overview of NRC policy regarding the level of detail to be provided in regulatory analyses is provided in Chapter 4 of the NUREG/BR-0058 [6]. As discussed in NUREG/BR-0184 [7], the emphasis in implementation of the NRC regulatory analysis guidelines should be on simplicity, flexibility, and commonsense, both in terms of the type of information supplied and in the level of detail provided. The level of treatment given to a particular issue in a regulatory analysis should reflect how crucial that issue is to the bottom line recommendation of the regulatory analysis.

The following sections discuss the rationale used to develop the technical approach, summarize the technical approach used, detail the logic model (event tree) development, describe supporting data analyses, and explain how the logic models were quantified and combined with results from the consequence analysis to produce risk metrics for each regulatory analysis sub-alternative.

2.2.1 Rationale Used to Develop the Technical Approach

The following factors were considered during the development of the technical approach for the CPRR accident sequence analysis:

1. The risk integration should provide risk metrics for each of the 20 CPRR regulatory analysis sub-alternatives according to the schedule established by the Commission and the resources allotted by NRC management.

2. As discussed in NUREG/BR-0058 and NUREG/BR-0184, the risk integration should provide fleet-average risk estimates. As a result, the technical approach should consider the impacts of plant-to-plant variability.

3. Consistent with Recommendation 5.1 in the Fukushima Near-Term Task Force report [8], the accident sequence analysis should focus on accidents initiated by ELAP events.

4. The generic estimates of release sequence frequencies and conditional consequences provided in NUREG/BR-0184 were developed from previous probabilistic risk assessments that did not consider CPRR strategies and, therefore, cannot be used to provide an adequate technical basis for the CPRR risk integration.

5. CDETs should be developed in order to:

    a. Model the impact of equipment failures and operator actions occurring prior to core damage that affect severe accident progression and the probability that CPRR strategies are successfully implemented,

    b. Match the initial and boundary conditions used in the thermal-hydraulic simulation of severe accidents (MELCOR calculations),

    c. Probabilistically consider mitigating strategies for beyond design basis external events required by Order EA-12-049 (e.g., the FLEX strategies, including anticipatory venting).

6. The CPRR strategies addressed in the set of 20 regulatory analysis sub-alternatives are specified at a conceptual level. As a result, it is acceptable to develop high-level generic APETs to model the CPRR strategies because no information is available about their specific design details.

2.2.2 Summary of the Technical Approach

Consideration of the factors listed above resulted in development of the following technical approach to conducting the accident sequence analysis:

1. Accident sequences are initiated by ELAP events due to internal events and seismic events. An ELAP is defined as a station blackout (SBO) that lasts longer than the SBO coping duration specified in 10 CFR 50.63. ELAP frequencies are semi-plant-specific since they are based on the plant’s emergency power system (EPS) class, SBO coping time, and site-specific seismic hazard.

2. The core-damage event trees (CDETs) and accident progression event trees (APETs) model a stylized (representative) BWR plant having a Mark I containment design and RCIC system.

3. The CDETs and APETs were developed using a modular approach that allows them to be combined and configured as needed to model each regulatory analysis sub-alternative.

4. The CDETs and APETs are quantified with industry-average reliability parameters (failure rates and failure-on-demand probabilities) and seismic fragilities developed from individual plant examination of external events (IPEEE) information.

5. The CDETs and APETs are quantified with scoping estimates of human error probabilities. Sensitivity studies confirmed that altering the human error probabilities (HEPs) values did not impact the results in any significant manner.

6. Similar core-damage sequences are grouped together using plant damage states (PDSs), which provide the input to the APETs.

7. The CDETs are solved for each plant within the scope of the analysis, then used to determine fleet-average PDS frequencies.

8. Similar APET sequences are grouped together using release categories (RCs).

9. Mean risk estimates are developed by multiplying the frequencies of significant RCs by the RC conditional consequences, then summing over all RCs.

The risk integration is not considered to be a PRA because it does not include all of the technical elements specified in Regulatory Position 1.2 of RG 1.200 [9].

2.2.3 Logic Model Development

As shown in Table 2-2, each regulatory analysis subalternative is defined by a specific combination of CPRR strategies that are intended to prevent the occurance of a severe accident or to mitigate its consequences should it occur. A probabilistic perspective recognizes that one or more of the CPRR strategies may not be successfully implemented. An accident sequence consists of an initiating event (the ELAP event), followed by a unique combination of CRPP strategy successes and failures that results in core damage and the subsequent release of radioactive materials to the environment. Logic model development uses a systematic process (event tree analysis) to identify the set of possible accident sequences associated with a specific regulatory analysis subalternative that might occur, and to estimate their frequency of occurrence. The logic model provides the fundamental probabilistic framework for assessing the risk associated with a specific regulatory analysis subalternative.

Logic model development relies on the results of the accident progression analysis (MELCOR analysis). An accident progression analysis is a simulation of a specific accident sequence that is conducted in order to (a) understand how the specific combination of CPRR strategy successes and failures affects the plant, and (b) estimate the fission product release (the source term) resulting from the accident sequence. The nomenclature used to identify MELCOR calculations somewhat overlaps with the nomenclature used to identify the regulatory analysis subalternatives. The reader is cautioned to remember that, in the context of the accident sequence analysis, a

specific regulatory analysis subalternative refers to a set of CPRR strategies and accident sequences, and that each accident sequence is linked to a specific MELCOR calculation.

The following sections provide the assumptions and ground rules used to develop the CDETs and APETs, describe the modular approach used for their development, and provide additional supporting information.

2.2.3.1 Assumptions and Ground Rules

During development of the accident sequence analysis, the set of strategies used to comply with the requirements of Order EA-12-049 were referred to as “SBO mitigating strategies,” “FLEX strategies,” and “mitigation of beyond-design-basis events (MDBDE) strategies.” The term “FLEX strategies” is a reference to the set of diverse and flexible coping strategies (FLEX) developed by the Nuclear Energy Institute, as defined in NEI-12-06 [10], to provide guidance to its members about complying with Order EA-12-049. The accident sequence analysis nomenclature uses the modifier “FLEX” in CDET and APET headings to refer to portable equipment and associated operator actions related to the operation of portable equipment.

A review of licensee approaches to implementing the requirements of Order EA-12-049 showed a wide variation of implementation details. To make the accident sequence analysis and risk integration tractable within the allotted schedule and budget, the CDETS and APETs were developed for a generic BWR plant having a reactor core isolation cooling (RCIC) system and a Mark I containment using the following assumptions and ground rules:

1. The CDETs model the first phase (use of installed plant equipment) and the second phase (use of onsite portable equipment) of the SBO mitigation strategy. The third phase (use of offsite portable equipment) was not considered due to lack of a suitable probabilistic approach for modeling the interactions between the plant and the National Response Centers established by industry’s Strategic Alliance for FLEX Emergency Response (SAFER).

2. In the CDETs, the first phase of the SBO mitigation strategy was assumed to last four hours. During this first phase, core cooling is provided by the RCIC system drawing from the suppression pool:

    a. No probabilistic consideration was taken for supplying the RCIC pump from the condensate storage tank (assumed to be non-seismically qualified).
    b. Based on the MELCOR analysis, the RCIC pump only needs to operate for two hours during the first phase. If the RCIC pump fails after two hours, then core damage will occur at about four hours after the ELAP occurs.
    c. No probabilistic consideration was taken for using the high-pressure coolent injection (HPCI) system. HPCI is a relatively high flowrate system (approximately 10 times higher than the portable FLEX pump), and would need to be manually cycled on and off to prevent overfilling the RPV. There is no information that can be used to estimate the reliability of HPCI while it is operating in a cyclical mode.
    d. The CDETs probabilistically consider local manual operation of the RCIC pump (termed “blackstart and blackrun”) if dc power fails.

e. No probabilistic consideration was taken for using the portable FLEX pump to provide core cooling if the RCIC pump fails because there is not sufficient time to get it aligned during the first phase.
f. There is no need to vent the containment during the first phase.
g. The plant operators will reduce RPV pressure using the safety relief valves (SRVs) to a range of 200-400 psig in order to minimize SRV cycling and to minimize heatup of the suppression pool.
h. The CDETs probabilistically consider local manual operation of SRVs if dc power fails.

3. In the CDETs, the second phase of the SBO mitigation strategy was assumed to last for 68 hours (i.e., the accident sequence analysis used a total mission time of 72 hours):

    a. The 72-hour total mission time used in the accident sequence analysis consists of the time needed for onsite diagnosis leading to a decision to request assistance from the National SAFER Response Centers (NSRC) (nominally 1 hour from the occurance of station blackout), the time needed to load and transport offssite portable equipment from the NSRC to the site (nominally 24 hours after the request is received), and the time needed to align the offsite portable equipment once it arrives at the site. Each of these times is highly uncertain; the 72-hour mission time is believed to be conservative. It should be noted that the only basic event in the accident sequence analysis whose probability depends on the assumed 72-hour mission time is the event FLEXP-FTR, "FLEX pump fails to run."
    b. Core cooling is maintained by operation of the RCIC pump. Makeup to the suppression pool is provided by the portable FLEX pump.
    c. Except for regulatory analysis alternatives 4Biv and 4Civ, the operators will initiate anticipatory containment venting in order to minimize heatup of the suppression pool and prolong RCIC operation. (Regulatory analysis alternatives 4Biv and 4Civ reflect a passive drywell-first venting strategy.)
    d. The WW and DW vents are redundant, i.e., the DW vent can be used to provide anticipatory venting if the WW vent fails closed, and vice versa.
    e. The CDETs probabilistically consider local manual operation of the containment vent valves if dc power fails.
    f. There is no need to provide RCIC pump room ventilation.
    g. A portable generator must be aligned to provide dc power within four hours. Many plants have battery lifetimes longer than four hours, which may be further extended by shedding non-essential dc loads. The scoping approach used to conduct the human reliability analysis (HRA) does not depend on the assumed battery lifetime.
    h. If the RCIC pump fails, core cooling can be provided by aligning the portable FLEX pump for RPV injection and depressurizing the RPV below the portable FLEX pump's shutoff head.

4. In the CDETs, it is assumed that the operators will attempt to reclose the containment vent valves, in accordance with the BWR Owners' Group Emergency Procedure and Severe Accident Guidelines (EPG/SAGs), if they recognize that core damage is occurring.

5. In the APETs, containment overpressurization failure is prevented by opening the containment vent valves:

    a. The WW and DW vents are redundant, i.e., the DW vent can be used to provide anticipatory venting if the WW vent fails closed, and vice versa.
    b. The CDETs probabilistically consider local manual operation of the containment vent valves if dc power fails.
    c. MELCOR calculations indicate that the containment must be vented prior to vessel breach due to the buildup of non-condensable gases generated during fuel-clad oxidation.
    d. Successful post-core-damage containment venting is a controlled release of radioactive materials to the environment, which is allowable under 10 CFR Part 50, Appendix A, General Design Criterion 16 and consistent with the Commission’s direction in the SRM to SECY-12-0157 to assume the installation of severe accident capable hardened venting systems ordered under Option 2.

6. In the APETs, post-core-damage water injection to the RPV can prevent vessel breach if it is initiated prior to lower plenum dryout and the RPV is depressurized below the shutoff head of the portable FLEX pump using the SRVs. In contrast, regulatory analysis alternatives involving post-core-damage DW injection cannot prevent vessel breach.

2.2.3.2 Modular Approach to Logic Model Development

A modular approach was used to develop the CDETs and APETs in order to streamline the development of risk estimates. As shown in Figure 2-1, three CDETs and six APETS were developed and subsequently combined as appropriate to develop the set of RC frequencies for each regulatory analysis sub-alternative. This modeling technique proved to be a responsive and efficient approach during the CPRR accident sequence analysis since the set of regulatory analysis sub-alternatives slowly evolved as the analysis progressed.

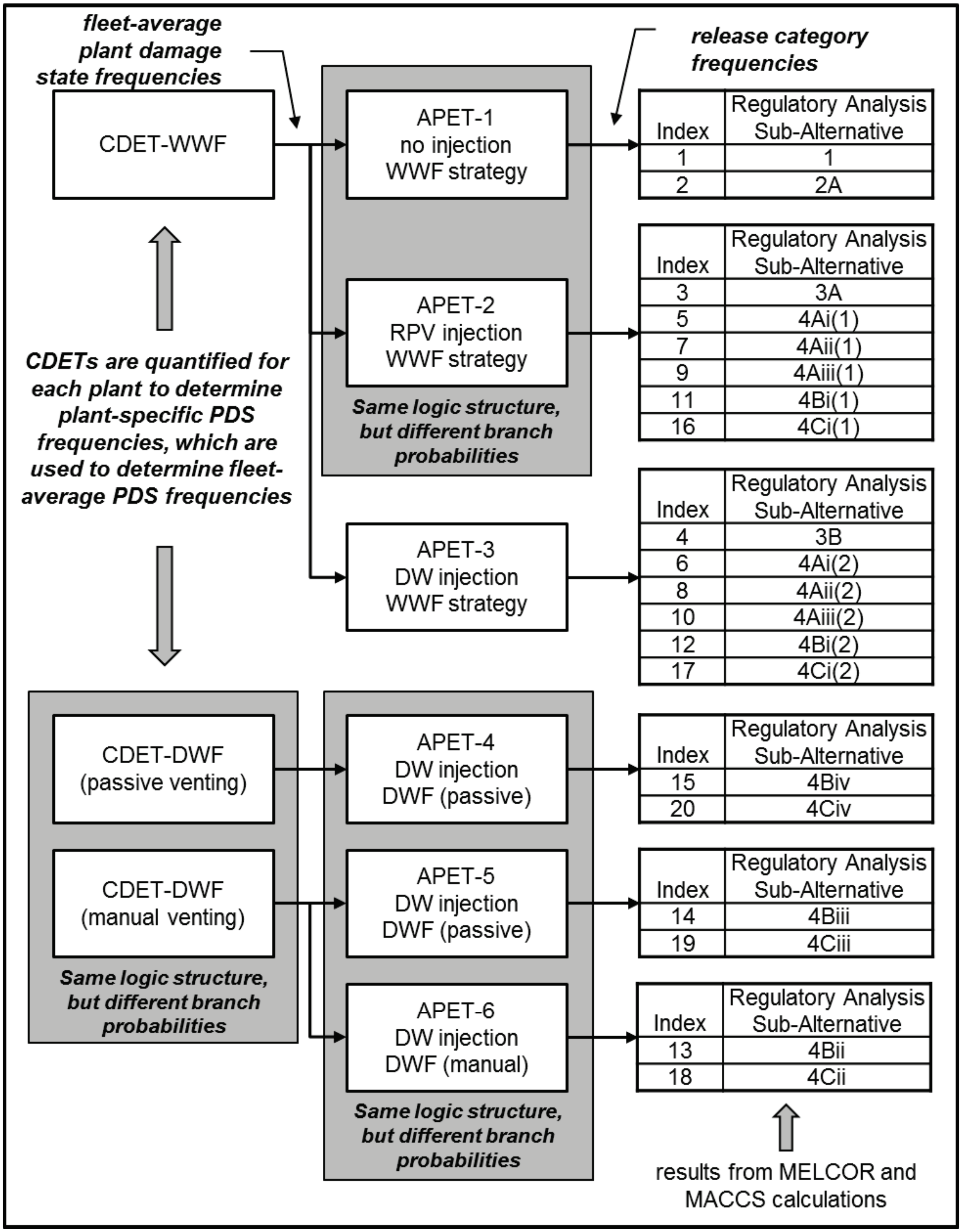


Figure 2-1 Modular approach to event tree development

2.2.3.3 Event Tree Development

The CDETs and APETs do not utilize supporting fault tree logic (i.e., they are not linked event tree/fault tree models that require the use of Boolean solution methods to generate minimal cut sets for each accident sequence). The CDET diagrams are shown in Appendix A, and supporting information is provided in Appendix B. The APET diagrams are shown in Appendix C, and supporting information is provided in Appendix D.

Each CDET has 280 core-damage sequences that were binned into 139 PDSs. Each APET contains 72-84 release sequences (depending on the regulatory analysis alternative being modeled) that were mapped into 24 release categories. Table 2-3 and Table 2-4 present the PDS and RC naming schemes.

Table 2-3 Plant damage state naming scheme

A plant damage state name consists of five attributes, arranged as follows: **T-PP-VV-DD-FF**			
PDS Attribute		**Possible Values**	
Code	**Description**	**Code**	**Description**
T	RCIC failure time	E	Early (0-4 hours)
		M	Mid-term (4-16 hours)
		L	Long-term (at 16 hours)
PP	RPV pressure at time of core damage	HP	High pressure (SRV cycling)
		MP	Medium pressure (200-400 psig)
		LP	Low pressure (below portable FLEX pump shutoff head)
VV	Containment vent status	DW	Drywell vent is open at time of core damage
		IS	Both vents are closed at the time of core damage
		WW	Wetwell vent is open at the time of core damage
DD	dc power status	LT	dc power fails long-term (unrecovered battery depletion)
		OK	dc power is available throughout the accident
		ST	dc power fails short-term (before battery depletion)
		XX	Indeterminate (dc power status not important to subsequent logic)
FF	FLEX pump status	OK	Portable FLEX pump is working
		F	FLEX pump hardware has failed
		H	Operator fails to align FLEX prior to core damage
		XX	Indeterminate (FLEX status not asked in the core-damage sequence)

Table 2-4 Release category naming scheme

A release category name consists of five attributes, arranged as follows: RRRRR-VV-DDD			
Attribute		**Possible Values**	
Code	**Description**	**Code**	**Description**
RRRRR	RPV depressurization	SRV	RPV is depressurized using the SRVs
		HP	RPV is at high pressure (high-pressure melt scenario)
		MSLCR	RPV depressurized due to main steamline creep rupture
VV	Containment vent status	WW	Wetwell vent is open
		DW	Drywell vent is open
		OP	Containment overpressurization failure
DDD	Core debris location	IVR	In-vessel retention
		EVR	Ex-vessel retention
		LMT	Liner melt-through

2.2.4 Supporting Data Analysis

The following sections describe how the frequency of ELAP events was estimated, identify the data source used to estimate hardware-related failure events, and discuss the reasons why a scoping human reliability analysis was used.

2.2.4.1 ELAP Frequency

The accident sequence analysis included ELAPs that are initiated by the four categories of loss-of-offsite-power events (LOOPs) that are included in the staff's Standardized Plant Analysis of Risk (SPAR) internal event models (plant-centered, switchyard-centered, grid-related, and weather-related, as defined in NUREG/CR-6890 [11]) and by seismic events. It was recognized that CPRR strategies may also be beneficial during other types of accidents (e.g., short duration SBOs, LOOPs that do not degenerate into SBOs, accidents that do not involve LOOP such as loss-of-coolant accidents, and accidents that are initiated by internal floods, internal fires, external floods and other types of external events). A complete assessment of CPRR strategies that includes these types of accidents would have required the development of plant-specific, internal and external event Level 3 PRAs for each BWR Mark I plant, which exceeds the required level-of-detail established in NUREG/BR-0058 and NUREG/BR-0184 for regulatory analyses.

The frequencies of ELAP events used in the accident sequence analysis are semi-plant-specific since they are based on the plant's emergency power system (EPS) class (the amount of redundancy provided by the onsite emergency ac power sources), the SBO coping time, and the site-specific seismic hazard.

The frequency of internal event ELAPs was estimated by probabilistic convolution of the associated LOOP frequency, the probability of emergency power system (EPS) failure, and the probability that offsite power was not recovered within the SBO coping duration. Specifically:

$$f_{ELAP} = f_{LOOP} \int_0^{\infty} \left[1 - \Phi\left(\frac{\ln(t + t_{SBO}) - \mu}{\sigma} \right) \right] \lambda e^{-\lambda t} \, dt$$

where:

f_{ELAP}	= ELAP frequency
f_{LOOP}	= LOOP frequency
Φ	= standard normal cumulative distribution function
t_{SBO}	= SBO coping time
μ,σ	= log-normal parameters of the offsite power recovery curve
λ	= EPS failure rate

LOOP frequencies and offsite power recovery curves (which are assumed to have a log-normal distribution) were obtained from NUREG/CR-6890. The probability of EPS failure was obtained from the 2011 update to NUREG/CR-5500, Vol. 5 [12], which is based on the staff's internal event SPAR models, reflects the number of onsite emergency power sources (the EPS class), and includes contributions from random equipment failures, common-cause failures, test/maintenance unavailability, and pre-initiator human failure events. Specific parameter values and the results of the probabilistic convolution are provided in Appendix E.

The frequency of seismic ELAPs was estimated by probabilistic convolution of the site-specific seismic hazard curve for peak ground acceleration with a model (top logic) of the EPS that included non-seismic failure modes and seismic failures of the emergency diesel generator equipment including the engine, generator, controls, day tank and fuel oil storage tank), ac switchgear, and dc switchgear. The convolution was performed using ten seismic bins that span the range of 0.03g to 3g peak ground acceleration. Seismic failures were assumed to be completely correlated (i.e., the seismically induced failure of a specific component implies the failure of all other similar and redundant components). No probabilistic consideration was given for recovering offsite power after a seismic event.

Seismic hazard curves were obtained from licensee responses (Browns Ferry [13], Brunswick [14], Cooper [15], Duane Arnold [16], Fermi [17], FitzPatrick [18], Hatch [19], Hope Creek [20], Monticello [21], Peach Bottom [22], Pilgrim [23], and Quad Cities [24]) to the 10 CFR 50.54(f) information request associated with NTTF Recommendation 2.1 [25].

Industry-average seismic fragilities were developed from information contained in the Individual Plant Examinations of External Events and the Screening, Prioritization and Implementation Details (SPID) document developed by the Electric Power Research Institute (EPRI) in response to NTTF Recommendation 2.1 [25]. Seismic fragility is characterized by three parameters:

C_{50}	= median seismic capacity
β_R	= logarithmic standard deviation due to randomness
β_U	= logarithmic standard deviation due to uncertainty

For each component, a linear opinion pool (discrete mixture distribution) was formed to develop an industry-average value of the median seismic capacity:

$$C_{50,avg} = \sum_i w_i C_{50,1}$$

where:

$C_{50,avg}$	= industry average median seismic capacity
w_i	= subjectively assigned weight of the i'th IPEEE fragility data source
$C_{50,I}$	= median seismic capacity of the i'th IPEEE fragility data source

IPEEE fragility data sources that did not contain complete information (C_{50} and β_U) were weighted less than complete sources.

A similar linear opinion pool was developed to estimate percentiles of the seismic capacity distribution function:

$$\alpha = \sum_i w_i \Phi\left[\frac{\ln C_\alpha - \ln C_{50,i}}{\beta_{U,i}}\right]$$

where C_α denotes the α'th percentile of seismic capacity distribution function, which found using numerical methods for α = 0.05 and α = 0.95. The industry-average logarithmic standard deviation for uncertainty was then determined from:

$$\beta_{U,avg} = \frac{1}{z_{0.95}}\sqrt{\frac{C_{0.95}}{C_{0.05}}}$$

where $z_{0.95}$ denotes the 95th percentile of the standard normal distribution (approximately 1.645).

Appendix F provides the seismic top logic, a summary of the seismic fragility parameter s, worksheets showing the details of the linear opinion pools used to develop industry-average fragility parameters, and plant-specific results of the seismic convolutions.

2.2.4.2 Hardware-Related Failure Data

Reliability parameters (failure rates and failure-on-demand probabilities) were based on estimates used in the staff's Standardized Plant Analysis of Risk (SPAR) models [28]. Appendix B lists the hardware-related failure event probabilities that are incorporated in the CDETs. Appendix D provides the hardware-related failure event probabilities that are incorporated into the APETs.

2.2.4.3 Human Reliability Analysis

The FLEX mitigation strategies represented in the CDETs and the CPRR strategies represented in the APETs rely on operator actions for their implementation. Some of these operator actions are to be performed in the main control room, and some are to be performed at various locations through the plant. All operator actions may need to be performed following a seismic event that is large enough to cause the occurrence of an ELAP event.

The assessment of the human error probabilities (HEPs) needed to quantify the CDETs and APETs proved to be challenging. The CPRR strategies are conceptual designs; accordingly, they are not incorporated into the EPG/SAGs or into licensee training programs. The staff gained some insight into how the CPRR strategies would be implemented through interactions with external stakeholders

during the development of the event trees. However, these interactions did not provide an adequate technical basis for completing the steps of an HRA of the CPRR strategies. No HRA method is capable of providing detailed HEP estimates at the conceptual design stage. Moreover:

1. Current HRA methods are inadequate for post-core-damage analysis because they are geared to supporting at-power, Level 1, internal events PRA, and therefore fail to recognize and appropriately capture the increased complexity of post-core-damage scenarios.

2. There is little actual experience with severe accidents to guide our understanding of operator responses in post-core-damage conditions.

3. EPG/SAGs differ from Emergency Operating Procedures (EOPs) in a number of ways including format, level of detail, prescriptiveness, and requirements for decision-making.

4. In general, there is less frequent training on EPG/SAGs as compared to the EOPs. In addition, most training simulators are not equipped to model plant behavior after the onset of core damage.

5. Cues to the operators may not be available or may be ambiguous. As a result, there is less accurate information on plant conditions that are important inputs to decision making.

6. The nature of the teamwork among the licensee staff responding to a severe accident is different as compared to responding to an off-normal situation. Prior to core damage, a small cohesive team in the main control room is tasked with responding. In contrast, following core damage, a larger number of people are involved who are situated at multiple distributed locations.

7. Following core damage, assessment responsibilities shift from the control room operators to technical support center.

8. Staffing may be inadequate for responding to site-wide events that involve multiple radiological sources.

9. Access to vital plant locations may be impaired due to the damage caused by seismic events or radiological hazards created by core damage.

Therefore, the accident sequence analysis used a set of scoping HEP estimates, supplemented with various sensitivity analyses that are focused on understanding the importance of operator actions to the CPRR strategies. Actions that take place in the main control room were assigned a scoping failure probability of 0.1. Actions that take place outside of the main control were assigned a scoping failure probability of 0.3. Appendix B provides the pre-core-damage human failure events (HFEs) that are incorporated in the CDETs, along with their scoping HEPs. Appendix D provides the post-core-damage HFEs that are incorporated into the APETs, along with their scoping HEPs.

2.2.5 Risk Quantification

The initial step in the risk quantification process was the determination of fleet-average PDS frequencies for each CDET, which was quantified twice for each plant included in the scope of the

accident sequence analysis. As shown in Figure 2-2, the first quantification was done using the semi-plant-specific internal events ELAP frequencies, and the second quantification was made using a set of four seismic ELAP frequencies that account for the success or failure of the dc power system and the RCIC pump. This approach was convenient to use since the only headings in the CDET that include seismic failures are ELAP, DC1 (failure of dc power in Phase 1), and RCIC1 (failure of RCIC in Phase 1).

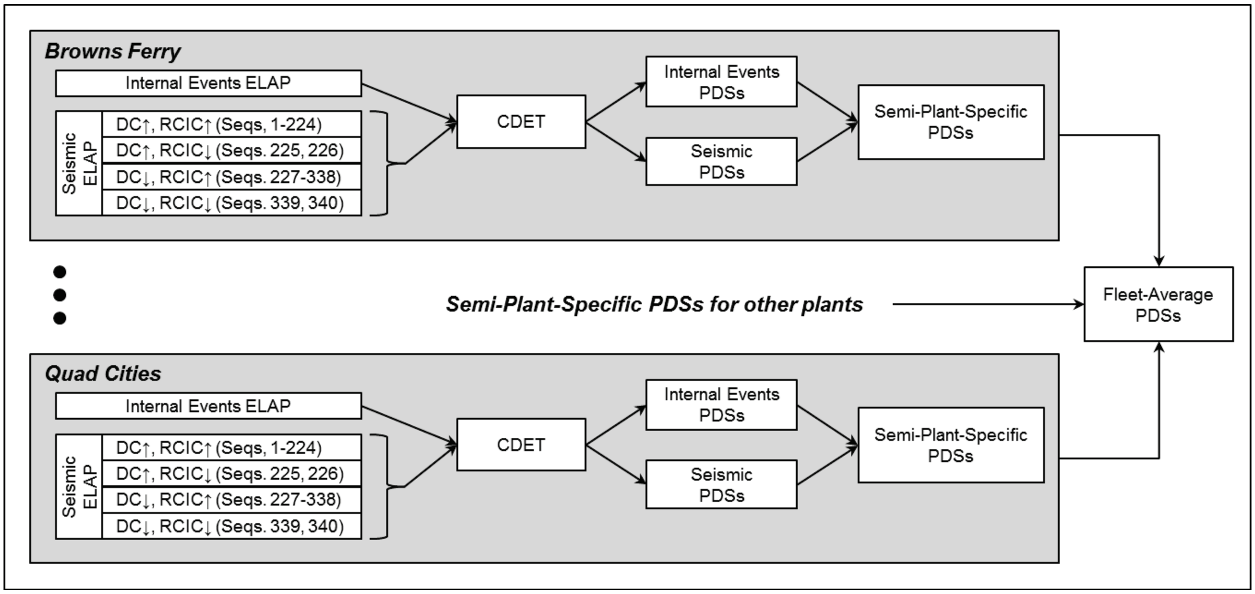


Figure 2-2 Core-damage event tree quantification

The second step in the risk quantification process was the determination of RC frequencies for each APET. The APET down-branch probabilities depend on the specific PDS that is input to the APET; as result, each APET was quantified 139 times (the number of unique PDSs for each CDET). The total set of RC frequencies was determined by summing the RC frequencies generated by the APET quantification process.

The third step in the risk quantification process was the identification of significant RCs. A significant RC is one of the set of RCs that, when ranked, compose 95% of the CDF or that individually contributes more than 1% to the CDF.

The fourth step is the risk quantification process is the determination of mean risk, R_{avg}, which is defined as:

$$R_{avg} = \sum_i f_i C_i$$

where f_i denotes the frequency of the i’th significant RC, and C_i denotes the conditional consequence associated with the i’th RC as determined by the consequence analysis. Conditional consequences were assigned by mapping each significant RC for a given regulatory analysis sub-alternative to an associated MELCOR case. Specifically, the mapping was achieved by matching the plant status specified by the RC attributes (i.e., the combination of successes and failures that led to release, including their time of occurrence) to the initial and boundary conditions of the MELCOR case. In turn, each MELCOR case was mapped to an associated MACCS bin based on

consideration of the source term, as further discussed in the consequence analysis chapter. Some RCs could not be directly mapped to one of the MELCOR cases; in this situation, an appropriate "composite" source term was developed by reviewing the set of MELCOR cases for similar plant conditions, including the timing and quantity of releases. Table 2-5 presents the mapping of significant RCs to MELCOR cases for each regulatory analysis sub-alternative. In this table, composite MELCOR cases are indicated by the letter "C," followed by a unique three-digit identifier. 1.1.1.1.1 APPENDIX A describes the build-up of composite MELCOR cases and MACCS bins.

Table 2-5 Mapping release categories to MELCOR cases

APET	**APET-1**	**APET-2**				
Regulatory Analysis Sub-Alternative **Release Category**	**1 – 1** **2 – 2A**	**3 - 3A** **5 - 4Ai(1)** **7 - 4Aii(1)** **9 - 4Aiii(1)**	**11 - 4Bi(1)**	**16 - 4Ci(1)**		
HP-DW-EVR						
HP-WW-EVR						
SRV-DW-EVR		11	11DF10	11DF1000		
SRV-WW-EVR		11	11DF10	11DF1000		
SRV-DW-IVR		C108	C108DF10	C108DF1000		
SRV-OP-IVR		C108	C108	C108		
SRV-WW-IVR		C108	C108DF10	C108DF1000		
HP-DW-LMT	2	2	2DF10	2DF1000		
HP-OP-LMT	2	2	2DF10	2DF1000		
HP-WW-LMT	2	2	2DF10	2DF1000		
SRV-DW-LMT	C104	C104	C104DF10	C104DF1000		
SRV-OP-LMT	C104	C104	C104DF10	C104DF1000		
SRV-WW-LMT	C104	C104	C104DF10	C104DF1000		
APET	**APET-3**					
Regulatory Analysis Sub-Alternative **Release Category**	**4 - 3B**	**6 - 4Ai(2)**	**8 - 4Aii(2)**	**10 - 4Aiii(2)**	**12 - 4Bi(2)**	**17 - 4Ci(2)**
HP-DW-EVR						
HP-WW-EVR	C101	30	C105	28	C106DF10	C106DF1000
SRV-DW-EVR	26dw	31dw	25dw	29dw	26dwDF10	26dwDF1000
SRV-WW-EVR	26	31	25	29	26DF10	26DF1000
SRV-DW-IVR						
SRV-OP-IVR						
SRV-WW-IVR						
HP-DW-LMT	2	2	2	2	2DF10	2DF1000
HP-OP-LMT	2	2	2	2	2DF10	2DF1000
HP-WW-LMT	2	2	2	2	2DF10	2DF1000
SRV-DW-LMT	C104	C104	C104	C104	C104DF10	C104DF1000
SRV-OP-LMT	C104	C104	C104	C104	C104DF10	C104DF1000
SRV-WW-LMT	C104	C104	C104	C104	C104DF10	C104DF1000

APET	APET-4		APET-5		APET-6	
Regulatory Analysis Sub-Alternative Release Category	15 - 4Biv	20 - 4Civ	14 - 4Biii	19 - 4Ciii	13 - 4Bii	18 - 4Cii
HP-DW-EVR	53DF10	53DF1000	53DF10	53DF1000	53DF10	53DF1000
HP-WW-EVR						
SRV-DW-EVR	51DF10	51DF1000	51DF10	51DF1000	51DF10	51DF1000
SRV-WW-EVR			C107DF10	C107DF1000	C107DF10	C107DF1000
SRV-DW-IVR						
SRV-OP-IVR						
SRV-WW-IVR						
HP-DW-LMT	2DF10	2DF1000	2DF10	2DF1000	2DF10	2DF1000
HP-OP-LMT	2DF10	2DF1000	2DF10	2DF1000	2DF10	2DF1000
HP-WW-LMT	2DF10	2DF1000	2DF10	2DF1000	2DF10	2DF1000
SRV-DW-LMT	C104DF10	C104DF1000	C104DF10	C104DF1000	C104DF10	C104DF1000
SRV-OP-LMT	C104DF10	C104DF1000	C104DF10	C104DF1000	C104DF10	C104DF1000
SRV-WW-LMT	C104DF10	C104DF1000	C104DF10	C104DF1000	C104DF10	C104DF1000

2.3 Results

In addition to the risk estimates developed to support the draft regulatory basis attached to SECY-15-0085, a variety of intermediate results were developed to provide risk insights about the CPRR strategies. These results are presented in the following sections.

2.3.1 Results from the CDET Quantification

Table 2-6 provides the point-estimate ELAP frequency, core-damage frequency (CDF), and conditional core-damage probability (CCDP) by site for the wetwell-first containment venting strategy. Table 2-7 and Table 2-8 provide the same information for the drywell-first (passive actuation) and the drywell-first (manual actuation) containment venting strategies. For ELAPs initiated by internal events, plant-to-plant variations of ELAP frequencies and core-damage frequencies (CDFs) are due to differences in the EPS class and SBO coping time. For ELAPs initiated by seismic events, plant-to-plant variations are due to differences in the seismic hazard curves.

The CCDP values given in Tables 2-6, 2-7 and 2-8 provide insight into the efficacy of the FLEX mitigating strategies. Overall, the FLEX strategies reduce the probability of core damage due to ELAP events by more than 50%. For a given pre-core-damage venting strategy:

- Each plant has the same CCDP for ELAPs initiated by internal events, which is due to the use of a generic BWR plant model in the accident sequence.

- The CCDP values exhibit minor variations for ELAPs initiated by seismic events, which is due to differences in the seismic hazard curves.

The CDF and CCDP values are somewhat lower for passive venting than for manual venting since the passive rupture disk is more reliable than operator action.

Table 2-6 Core-damage frequencies by site for Wetwell-First strategy

Site	EPS Class	SBO Coping Time (h)	Internal Events			Seismic			Total		
			ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP
Browns Ferry	4	4	4.1E-07	1.5E-07	36%	2.2E-05	1.0E-05	46%	2.3E-05	1.0E-05	45%
Brunswick	2	4	6.9E-06	2.5E-06	36%	1.1E-05	4.7E-06	44%	1.8E-05	7.2E-06	41%
Cooper	2	4	6.9E-06	2.5E-06	36%	4.9E-06	2.2E-06	45%	1.2E-05	4.7E-06	40%
Duane Arnold	2	4	6.9E-06	2.5E-06	36%	2.4E-06	1.1E-06	47%	9.2E-06	3.6E-06	39%
Fermi	4	4	4.1E-07	1.5E-07	36%	8.8E-06	4.1E-06	47%	9.2E-06	4.3E-06	46%
FitzPatrick	4	4	4.1E-07	1.5E-07	36%	3.5E-06	1.6E-06	46%	3.9E-06	1.7E-06	45%
Hatch	3	4	3.4E-06	1.2E-06	36%	9.6E-06	4.4E-06	46%	1.3E-05	5.6E-06	43%
Hope Creek	3	4	3.4E-06	1.2E-06	36%	7.6E-06	3.4E-06	44%	1.1E-05	4.6E-06	42%
Monticello	2	4	6.9E-06	2.5E-06	36%	6.7E-06	3.2E-06	47%	1.4E-05	5.6E-06	42%
Peach Bottom	3	8	2.0E-06	7.4E-07	36%	3.8E-05	1.9E-05	50%	4.0E-05	2.0E-05	50%
Pilgrim	2	8	4.1E-06	1.5E-06	36%	7.0E-05	3.4E-05	49%	7.4E-05	3.6E-05	48%
Quad Cities	4	4	4.1E-07	1.5E-07	36%	7.0E-06	3.3E-06	47%	7.4E-06	3.4E-06	47%
Fleet Average			**3.5E-06**	**1.3E-06**	**36%**	**1.6E-05**	**7.6E-06**	**48%**	**1.9E-05**	**8.9E-06**	**46%**

Table 2-7 Core-damage frequencies by site for Drywell-First (passive) strategy.

Site	EPS Class	SBO Coping Time (h)	Internal Events			Seismic			Total		
			ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP
Browns Ferry	4	4	4.1E-07	1.4E-07	34%	2.2E-05	9.6E-06	43%	2.3E-05	9.8E-06	43%
Brunswick	2	4	6.9E-06	2.3E-06	34%	1.1E-05	4.5E-06	41%	1.8E-05	6.8E-06	38%
Cooper	2	4	6.9E-06	2.3E-06	34%	4.9E-06	2.0E-06	42%	1.2E-05	4.4E-06	37%
Duane Arnold	2	4	6.9E-06	2.3E-06	34%	2.4E-06	1.0E-06	44%	9.2E-06	3.4E-06	36%
Fermi	4	4	4.1E-07	1.4E-07	34%	8.8E-06	3.9E-06	44%	9.2E-06	4.0E-06	44%
FitzPatrick	4	4	4.1E-07	1.4E-07	34%	3.5E-06	1.5E-06	43%	3.9E-06	1.6E-06	42%
Hatch	3	4	3.4E-06	1.2E-06	34%	9.6E-06	4.1E-06	43%	1.3E-05	5.3E-06	40%
Hope Creek	3	4	3.4E-06	1.2E-06	34%	7.6E-06	3.2E-06	42%	1.1E-05	4.3E-06	39%
Monticello	2	4	6.9E-06	2.3E-06	34%	6.7E-06	3.0E-06	44%	1.4E-05	5.3E-06	39%
Peach Bottom	3	8	2.0E-06	6.9E-07	34%	3.8E-05	1.8E-05	48%	4.0E-05	1.9E-05	47%
Pilgrim	2	8	4.1E-06	1.4E-06	34%	7.0E-05	3.2E-05	46%	7.4E-05	3.4E-05	46%
Quad Cities	4	4	4.1E-07	1.4E-07	34%	7.0E-06	3.1E-06	45%	7.4E-06	3.3E-06	44%
Fleet Average			**3.5E-06**	**1.2E-06**	**34%**	**1.6E-05**	**7.2E-06**	**45%**	**1.9E-05**	**8.4E-06**	**43%**

Table 2-8 Core-damage frequencies by site for Drywell-First (manual) strategy

Site	EPS Class	SBO Coping Time (h)	Internal Events			Seismic			Total		
			ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP	ELAP Frequency (/y)	CDF (/y)	CCDP
Browns Ferry	4	4	4.1E-07	1.5E-07	36%	2.2E-05	1.0E-05	46%	2.3E-05	1.0E-05	45%
Brunswick	2	4	6.9E-06	2.5E-06	36%	1.1E-05	4.7E-06	44%	1.8E-05	7.2E-06	41%
Cooper	2	4	6.9E-06	2.5E-06	36%	4.9E-06	2.2E-06	45%	1.2E-05	4.7E-06	40%
Duane Arnold	2	4	6.9E-06	2.5E-06	36%	2.4E-06	1.1E-06	47%	9.2E-06	3.6E-06	39%
Fermi	4	4	4.1E-07	1.5E-07	36%	8.8E-06	4.1E-06	47%	9.2E-06	4.3E-06	46%
FitzPatrick	4	4	4.1E-07	1.5E-07	36%	3.5E-06	1.6E-06	46%	3.9E-06	1.7E-06	45%
Hatch	3	4	3.4E-06	1.2E-06	36%	9.6E-06	4.4E-06	46%	1.3E-05	5.6E-06	43%
Hope Creek	3	4	3.4E-06	1.2E-06	36%	7.6E-06	3.4E-06	44%	1.1E-05	4.6E-06	42%
Monticello	2	4	6.9E-06	2.5E-06	36%	6.7E-06	3.2E-06	47%	1.4E-05	5.6E-06	42%
Peach Bottom	3	8	2.0E-06	7.4E-07	36%	3.8E-05	1.9E-05	50%	4.0E-05	2.0E-05	50%
Pilgrim	2	8	4.1E-06	1.5E-06	36%	7.0E-05	3.4E-05	49%	7.4E-05	3.6E-05	48%
Quad Cities	4	4	4.1E-07	1.5E-07	36%	7.0E-06	3.3E-06	47%	7.4E-06	3.4E-06	47%
Fleet Average			**3.5E-06**	**1.3E-06**	**36%**	**1.6E-05**	**7.6E-06**	**48%**	**1.9E-05**	**8.9E-06**	**46%**

As shown in Table 2-6, Table 2-7 and Table 2-8, the contribution of ELAPs initiated by seismic events is generally greater than the contribution of ELAPs initiated by internal events. However, this observation is reversed for plants located in areas with low seismicity with only two onsite emergency power sources (EPS Class) and 4-hour SBO coping times. Figure 2-3 shows the contribution to fleet-average ELAP frequency and CDF by seismic bin. In general, the contribution from ELAPs initiated by seismic events is due to earthquakes whose ground motion exceeds the plant design basis (the safe shutdown earthquake).

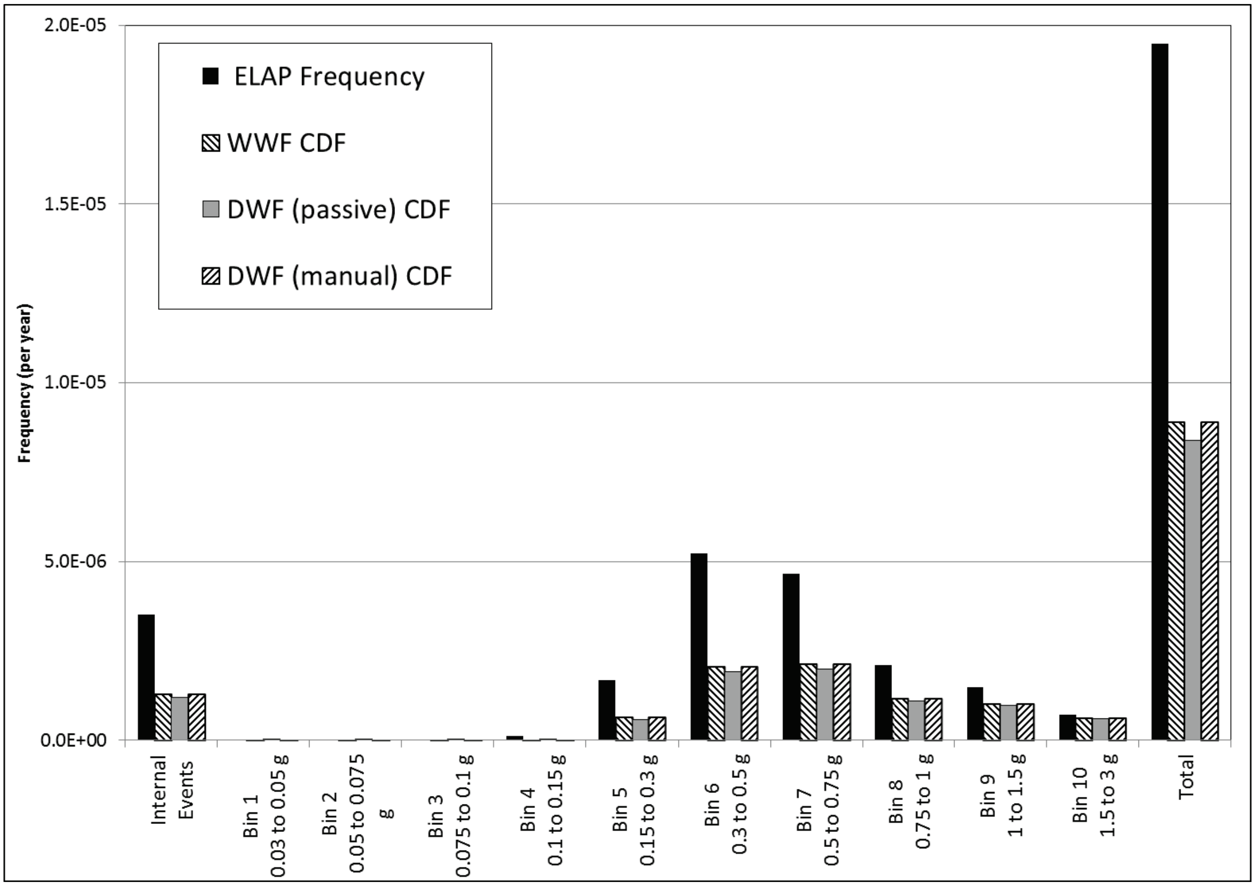


Figure 2-3 Contributions to ELAP frequency and core-damage frequency

Table 2-9 identifies the significant contributors to CDF, as determined by importance analysis. Consistent with Regulatory Guide 1.200, an event is a significant contributor if it has a risk achievement worth (RAW) greater then 2.0, or a Fussell-Vesely (FV) importance measure greater than 0.005. Figure 2-4 presents the information in Table 2-9 in graphical form. The most important contributors to CDF are seismic failures of the station batteries or dc switchgear, seismic failures of the emergency diesel generators (EDGs) and their supporting systems, random failures of the portable FLEX pump (failure to start and failure to run), and random failures of the RCIC pump (failure to start).

Failures of dc equipment (batteries, switchgear) are important because it is used to actuate the EDGs, RCIC, the SRVs and containment vents. The CDETS assume that failures of RCIC, the SRVs and the containment vents can be recovered by local, manual operator actions. These recovery actions, which need to occur in addition to aligning the FLEX pump, take place at diverse plant locations. The accident sequence analysis assumed that the plant is adequately staffed to achieve all of these operators, and did not explore resource constraints. Failure of plant personnel to successfully perform these recovery actions would contribute toward an increase on core-damage frequency.

Table 2-9 Importance measures with respect to core-damage frequency

Identifier	Description	Wetwell-First Venting Strategy		Drywell-First Venting Strategy (Passive)		Drywell-First Venting Strategy (Manual)	
		RAW[a]	FV[b]	RAW	FV	RAW	FV
Equipment Failures							
EDG-FTR	EDGs fail to run	2.4	1.8E-02	2.4	1.8E-02	2.4	1.8E-02
EDG-FTS	EDGs fail to start	2.4	1.2E-04	2.4	1.2E-04	2.4	1.2E-04
FLEXP-FTR	FLEX pump fails to run	2.2	2.0E-01	2.3	2.3E-01	2.2	2.0E-01
FLEXP-FTS	FLEX pump fails to start	2.2	6.1E-03	2.3	6.8E-03	2.2	6.1E-03
GEN-FTR	portable generator fails to run	1.1	1.4E-02	1.1	1.1E-02	1.1	1.4E-02
GEN-FTS	portable generator fails to start	1.1	6.6E-03	1.1	5.0E-03	1.1	6.6E-03
RCIC-1	RCIC fails to run in Phase 1	2.2	2.2E-05	2.3	2.5E-05	2.2	2.2E-05
RCIC-FTS	RCIC fails to start	2.2	1.1E-02	2.3	1.2E-02	2.2	1.1E-02
Human Failure Events							
HFE-DW	operator fails to open the drywell vent	1.1	6.6E-03	< 2	< 0.005	1.1	6.6E-03
HFE-DW-NODC	operator fails to open the drywell vent (no DC power)	1.1	5.3E-02	< 2	< 0.005	1.1	5.3E-02
HFE-GEN	operator fails to align portable generator	1.1	6.3E-02	1.1	4.8E-02	1.1	6.3E-02
HFE-INVDP	operator inadvertently depressurize RPV below RCIC	1.1	2.3E-02	1.1	2.5E-02	1.1	2.3E-02
HFE-INVDP-NODC	operator inadvertently depressurize RPV below RCIC (no DC power)	1.0	1.7E-02	1.0	2.0E-02	1.0	1.7E-02
HFE-PINJ	operator fails to depressurize for FLEX RPV injection	1.1	1.2E-02	1.1	1.3E-02	1.1	1.2E-02
HFE-PINJ-NODC	operator fails to depressurize for FLEX RPV injection (no DC power)	1.1	5.6E-02	1.2	6.6E-02	1.1	5.6E-02
HFE-RCIC-BLACKRUN	operator fails to blackrun RCIC during Phase 2	1.1	5.6E-02	1.2	6.5E-02	1.1	5.6E-02
HFE-RPV	operator fails to align FLEX pump for RPV injection	1.6	2.5E-01	1.7	2.8E-01	1.6	2.5E-01
HFE-SP	operator fails to align FLEX pump for SP makeup	1.3	1.1E-01	1.3	1.2E-01	1.3	1.1E-01
HFE-WW	operator fails to open the wetwell vent	1.1	6.6E-03	< 2	< 0.005	1.1	6.6E-03
HFE-WW-NODC	operator fails to open the wetwell vent (no DC power)	1.1	5.3E-02	< 2	< 0.005	1.1	5.3E-02
Seismic Failures							
ACSWGR-S	seismic failure of AC switchgear	2.4	2.8E-02	2.4	2.8E-02	2.4	2.8E-02
BATTERY-S	seismic failure of batteries	2.8	6.9E-02	2.7	6.2E-02	2.8	6.9E-02
DCSWGR-S	seismic failure of DC switchgear	2.8	9.6E-03	2.7	8.2E-03	2.8	9.6E-03
EDG-S-CTRL	seismic failure of EDG controls	2.4	2.5E-02	2.4	2.5E-02	2.4	2.5E-02
EDG-S-DT	seismic failure of EDG day tank	2.4	8.8E-02	2.4	8.8E-02	2.4	8.8E-02
EDG-S-EG	seismic failure of EDG engine or generator	2.4	2.3E-02	2.4	2.3E-02	2.4	2.3E-02
EDG-S-FOST	seismic failure of EDG fuel oil storage tank	2.4	3.2E-02	2.4	3.2E-02	2.4	3.2E-02
EDG-S-SAR	seismic failure of EDG starting air receiver	2.4	1.1E-01	2.4	1.1E-01	2.4	1.1E-01
LOOP-S	seismic LOOP	1.9	8.6E-01	1.9	8.6E-01	1.9	8.6E-01
RCIC-S	RCIC seismic failure	1.9	1.6E-01	2.0	1.8E-01	1.9	1.6E-01

[a] Risk achievement worth
[b] Fussell-Veseley importance measure

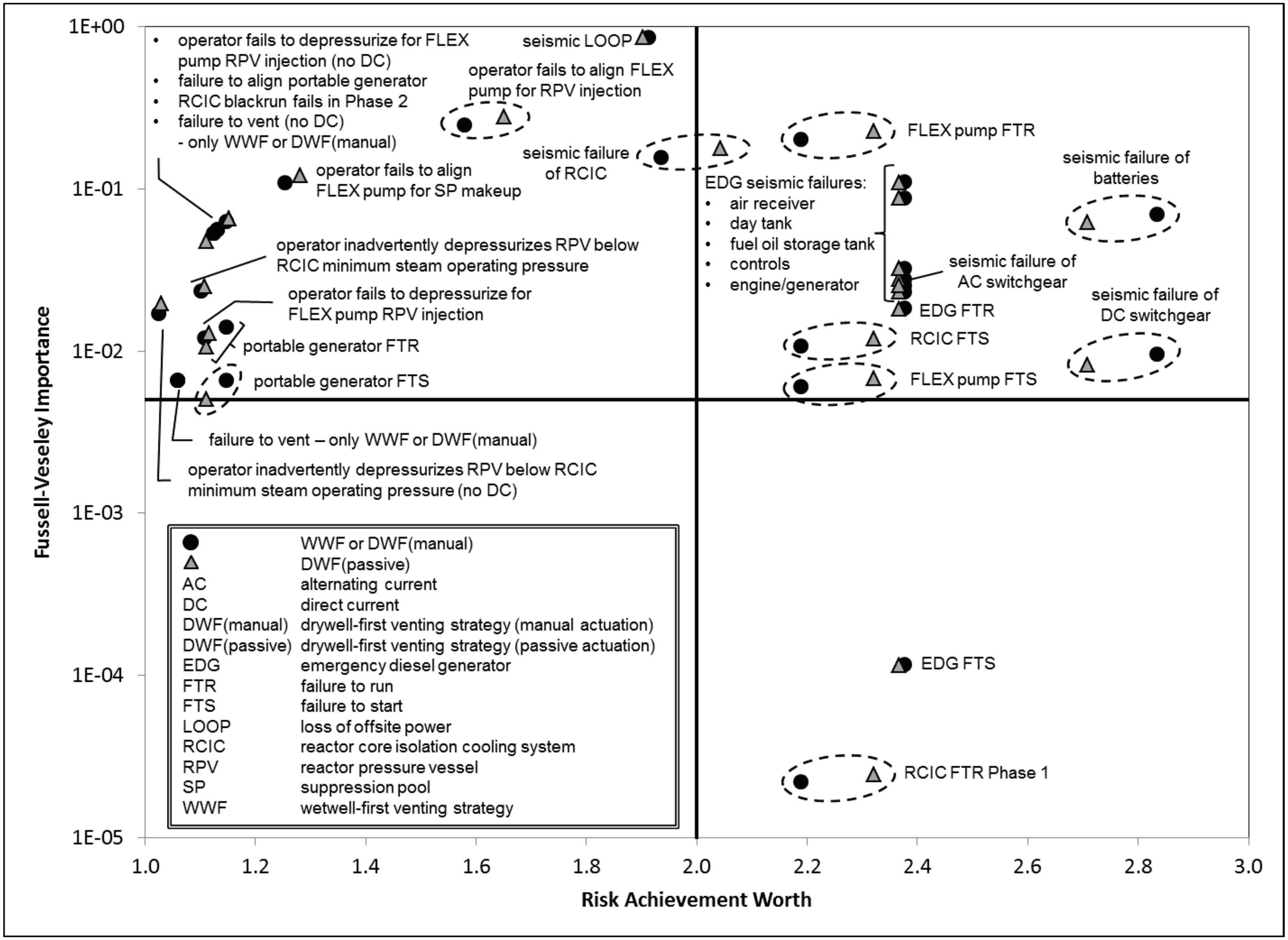


Figure 2-4 **Identification of significant events with respect to core damage frequency**

APPENDIX G: provides the complete list of PDS frequencies, in alphabetic order, that were input to the APETs. Table 2-10, Table 2-11 and Table 2-12 list the significant PDS for each CDET in descending order. A significant PDS is one of the set of PDSs that, when ranked, compose 95% of the CDF or that individually contributes more than 1% to the CDF. Review of these tables shows that failures that occur during FLEX Phase 1 (before the portable FLEX pump can be aligned) contribute about 30% of the total CDF.

Table 2-10 Significant plant damage states – Wetwell First strategy

Rank	Wetwell First Strategy			
	Plant Damage State	Frequency (/y)	Contribution (%)	Cumulative (%)
1	E-LP-IS-ST-XX	1.01E-06	11.4	11.4
2	E-LP-IS-XX-XX	9.81E-07	11.0	22.4
3	L-MP-IS-OK-F	6.74E-07	7.6	30.0
4	M-LP-IS-OK-H	5.70E-07	6.4	36.4
5	L-MP-IS-OK-H	4.44E-07	5.0	41.3
6	E-HP-IS-ST-XX	4.34E-07	4.9	46.2
7	M-MP-IS-LT-XX	3.67E-07	4.1	50.3
8	M-LP-IS-LT-H	3.29E-07	3.7	54.0
9	M-LP-IS-ST-H	2.94E-07	3.3	57.3
10	M-MP-IS-LT-H	2.34E-07	2.6	60.0
11	M-LP-IS-OK-F	2.00E-07	2.2	62.2
12	L-MP-IS-LT-F	1.67E-07	1.9	64.1
13	M-MP-IS-ST-XX	1.51E-07	1.7	65.8
14	M-MP-IS-LT-OK	1.39E-07	1.6	67.3
15	M-LP-IS-LT-F	1.16E-07	1.3	68.6
16	L-MP-IS-LT-H	1.10E-07	1.2	69.9
17	E-HP-IS-XX-XX	1.09E-07	1.2	71.1
18	M-LP-WW-LT-H	1.09E-07	1.2	72.3
19	M-LP-IS-ST-F	1.03E-07	1.2	73.5
20	M-LP-IS-ST-XX	9.97E-08	1.1	74.6
21	M-LP-WW-ST-H	9.68E-08	1.1	75.7
22	M-MP-IS-ST-H	9.61E-08	1.1	76.8
23	L-MP-IS-OK-OK	8.79E-08	1.0	77.7
24	M-MP-IS-LT-F	8.22E-08	0.9	78.7
25	M-MP-WW-LT-H	7.72E-08	0.9	79.5
26	M-LP-IS-LT-XX	6.86E-08	0.8	80.3
27	L-MP-IS-ST-F	6.84E-08	0.8	81.1
28	L-MP-WW-OK-F	6.81E-08	0.8	81.8
29	M-HP-IS-ST-XX	6.77E-08	0.8	82.6
30	L-MP-IS-LT-OK	6.52E-08	0.7	83.3
31	L-HP-IS-OK-F	6.51E-08	0.7	84.1
32	M-MP-IS-OK-XX	6.42E-08	0.7	84.8
33	M-LP-WW-OK-H	5.75E-08	0.6	85.4
34	M-MP-IS-ST-OK	5.71E-08	0.6	86.1
35	L-MP-WW-LT-F	5.50E-08	0.6	86.7
36	M-MP-WW-LT-OK	4.59E-08	0.5	87.2
37	L-MP-IS-ST-H	4.50E-08	0.5	87.7
38	L-MP-WW-OK-H	4.48E-08	0.5	88.2
39	M-HP-IS-ST-H	4.32E-08	0.5	88.7
40	L-HP-IS-OK-H	4.28E-08	0.5	89.2
41	L-HP-IS-ST-F	4.21E-08	0.5	89.7
42	M-LP-WW-LT-F	3.81E-08	0.4	90.1
43	L-MP-WW-LT-H	3.62E-08	0.4	90.5

Rank	Wetwell First Strategy			
	Plant Damage State	Frequency (/y)	Contribution (%)	Cumulative (%)
44	M-LP-WW-ST-F	3.40E-08	0.4	90.9
45	M-MP-IS-ST-F	3.37E-08	0.4	91.2
46	M-HP-IS-LT-XX	3.33E-08	0.4	91.6
47	M-LP-DW-LT-H	3.26E-08	0.4	92.0
48	M-MP-WW-ST-H	3.17E-08	0.4	92.3
49	M-LP-DW-ST-H	2.90E-08	0.3	92.7
50	L-HP-IS-ST-H	2.77E-08	0.3	93.0
51	M-MP-WW-LT-F	2.71E-08	0.3	93.3
52	L-MP-IS-ST-OK	2.68E-08	0.3	93.6
53	M-HP-IS-ST-OK	2.57E-08	0.3	93.9
54	M-MP-DW-LT-H	2.32E-08	0.3	94.1
55	L-MP-WW-ST-F	2.26E-08	0.3	94.4
56	L-MP-WW-LT-OK	2.15E-08	0.2	94.6
57	M-HP-IS-LT-H	2.12E-08	0.2	94.9
58	L-HP-IS-LT-F	2.07E-08	0.2	95.1

Table 2-11 Significant plant damage states – Drywell-First (passive) strategy.

Rank	Wetwell First Strategy			
	Plant Damage State	Frequency (/y)	Contribution (%)	Cumulative (%)
1	E-LP-IS-ST-XX	1.01E-06	12.1	12.1
2	E-LP-IS-XX-XX	9.81E-07	11.7	23.7
3	L-MP-DW-OK-F	7.49E-07	8.9	32.7
4	M-LP-DW-OK-H	6.33E-07	7.5	40.2
5	M-LP-DW-LT-H	5.12E-07	6.1	46.3
6	L-MP-DW-OK-H	4.93E-07	5.9	52.2
7	M-LP-DW-ST-H	4.56E-07	5.4	57.6
8	E-HP-IS-ST-XX	4.34E-07	5.2	62.8
9	M-MP-DW-LT-H	3.64E-07	4.3	67.1
10	L-MP-DW-LT-F	2.59E-07	3.1	70.2
11	M-LP-DW-OK-F	2.22E-07	2.6	72.9
12	M-MP-DW-LT-OK	2.16E-07	2.6	75.4
13	M-LP-DW-LT-F	1.80E-07	2.1	77.6
14	L-MP-DW-LT-H	1.70E-07	2.0	79.6
15	M-LP-DW-ST-F	1.60E-07	1.9	81.5
16	M-MP-DW-ST-H	1.49E-07	1.8	83.3
17	M-MP-DW-LT-F	1.28E-07	1.5	84.8
18	E-HP-IS-XX-XX	1.09E-07	1.3	86.1
19	L-MP-DW-ST-F	1.06E-07	1.3	87.4
20	L-MP-DW-LT-OK	1.01E-07	1.2	88.6
21	L-MP-DW-OK-OK	9.77E-08	1.2	89.8
22	M-MP-DW-ST-OK	8.88E-08	1.1	90.8
23	L-HP-DW-OK-F	7.24E-08	0.9	91.7
24	L-MP-DW-ST-H	6.99E-08	0.8	92.5
25	M-HP-DW-ST-H	6.72E-08	0.8	93.3
26	L-HP-DW-ST-F	6.55E-08	0.8	94.1
27	M-MP-DW-ST-F	5.24E-08	0.6	94.7
28	L-HP-DW-OK-H	4.76E-08	0.6	95.3

Table 2-12 Significant plant damage states – Drywell-First (manual) strategy.

Rank	Wetwell First Strategy			
	Plant Damage State	Frequency (/y)	Contribution (%)	Cumulative (%)
1	E-LP-IS-ST-XX	1.01E-06	11.4	11.4
2	E-LP-IS-XX-XX	9.81E-07	11.0	22.4
3	L-MP-IS-OK-F	6.74E-07	7.6	30.0
4	M-LP-IS-OK-H	5.70E-07	6.4	36.4
5	L-MP-IS-OK-H	4.44E-07	5.0	41.3
6	E-HP-IS-ST-XX	4.34E-07	4.9	46.2
7	M-MP-IS-LT-XX	3.67E-07	4.1	50.3
8	M-LP-IS-LT-H	3.29E-07	3.7	54.0
9	M-LP-IS-ST-H	2.94E-07	3.3	57.3
10	M-MP-IS-LT-H	2.34E-07	2.6	60.0
11	M-LP-IS-OK-F	2.00E-07	2.2	62.2
12	L-MP-IS-LT-F	1.67E-07	1.9	64.1
13	M-MP-IS-ST-XX	1.51E-07	1.7	65.8
14	M-MP-IS-LT-OK	1.39E-07	1.6	67.3
15	M-LP-IS-LT-F	1.16E-07	1.3	68.6
16	L-MP-IS-LT-H	1.10E-07	1.2	69.9
17	E-HP-IS-XX-XX	1.09E-07	1.2	71.1
18	M-LP-DW-LT-H	1.09E-07	1.2	72.3
19	M-LP-IS-ST-F	1.03E-07	1.2	73.5
20	M-LP-IS-ST-XX	9.97E-08	1.1	74.6
21	M-LP-DW-ST-H	9.68E-08	1.1	75.7
22	M-MP-IS-ST-H	9.61E-08	1.1	76.8
23	L-MP-IS-OK-OK	8.79E-08	1.0	77.7
24	M-MP-IS-LT-F	8.22E-08	0.9	78.7
25	M-MP-DW-LT-H	7.72E-08	0.9	79.5
26	M-LP-IS-LT-XX	6.86E-08	0.8	80.3
27	L-MP-IS-ST-F	6.84E-08	0.8	81.1
28	L-MP-DW-OK-F	6.81E-08	0.8	81.8
29	M-HP-IS-ST-XX	6.77E-08	0.8	82.6
30	L-MP-IS-LT-OK	6.52E-08	0.7	83.3
31	L-HP-IS-OK-F	6.51E-08	0.7	84.1
32	M-MP-IS-OK-XX	6.42E-08	0.7	84.8
33	M-LP-DW-OK-H	5.75E-08	0.6	85.4
34	M-MP-IS-ST-OK	5.71E-08	0.6	86.1
35	L-MP-DW-LT-F	5.50E-08	0.6	86.7
36	M-MP-DW-LT-OK	4.59E-08	0.5	87.2
37	L-MP-IS-ST-H	4.50E-08	0.5	87.7
38	L-MP-DW-OK-H	4.48E-08	0.5	88.2
39	M-HP-IS-ST-H	4.32E-08	0.5	88.7
40	L-HP-IS-OK-H	4.28E-08	0.5	89.2
41	L-HP-IS-ST-F	4.21E-08	0.5	89.7
42	M-LP-DW-LT-F	3.81E-08	0.4	90.1
43	L-MP-DW-LT-H	3.62E-08	0.4	90.5
44	M-LP-DW-ST-F	3.40E-08	0.4	90.9
45	M-MP-IS-ST-F	3.37E-08	0.4	91.2
46	M-HP-IS-LT-XX	3.33E-08	0.4	91.6
47	M-LP-WW-LT-H	3.26E-08	0.4	92.0
48	M-MP-DW-ST-H	3.17E-08	0.4	92.3
49	M-LP-WW-ST-H	2.90E-08	0.3	92.7
50	L-HP-IS-ST-H	2.77E-08	0.3	93.0
51	M-MP-DW-LT-F	2.71E-08	0.3	93.3
52	L-MP-IS-ST-OK	2.68E-08	0.3	93.6
53	M-HP-IS-ST-OK	2.57E-08	0.3	93.9

Rank	Wetwell First Strategy			
	Plant Damage State	Frequency (/y)	Contribution (%)	Cumulative (%)
54	M-MP-WW-LT-H	2.32E-08	0.3	94.1
55	L-MP-DW-ST-F	2.26E-08	0.3	94.4
56	L-MP-DW-LT-OK	2.15E-08	0.2	94.6
57	M-HP-IS-LT-H	2.12E-08	0.2	94.9
58	L-HP-IS-LT-F	2.07E-08	0.2	95.1

2.3.2 Results from the APET Quantification

Table 2-13 summarizes the results of the APET quantification in terms of containment failure mode probabilities, which provides insight into the efficacy of the CPRR strategies. The capability to vent the containment during a severe accident, as required by Order EA-13-109, limits the probability of containment structural failure due to overpressurization to a few percent. However, it is noted that operation of the containment venting system during a severe accident also results in a release (albeit, controlled) of fission products to the environment.

The provision of post-accident water injection capability noticeably reduces the probability of containment failure due to liner melt-through. The largest reductions are associated with APET-2, which applies to regulatory analysis sub-alternatives that utilize RPV injection (as opposed to direct DW injection). Sub-alternatives involving post-accident injection to the RPV provide the capability to arrest a severe accident before vessel breach occurs.

Table 2-13 Containment failure mode probabilities

Description	Accident Progression Event Trees and Associated Regulatory Analysis Sub-Alternatives					
	APET-1 1 - 1 2 - 2A	APET-2 3 - 3A 5 - 4Ai(1) 7 - 4Aii(1) 9 - 4Aiii(1) 11 - 4Bi(1) 16 - 4Ci(1)	APET-3 4 - 3B 6 - 4Ai(2) 8 - 4Aii(2) 10 - 4Aiii(2) 12 - 4Bi(2) 17 - 4Ci(2)	APET-4 15 - 4Biv 20 - 4Civ	APET-5 14 - 4Biii 19 - 4Ciii	APET-6 13 - 4Bii 18 - 4Cii
Containment Pressure Control						
Drywell Venting	15%	15%	21%	99%	96%	85%
Wetwell Venting	81%	81%	77%	1%	3%	13%
Overpressurization Failure	4%	4%	3%	0.05%	0.1%	3%
Core Debris Location						
In-Vessel Retention	0%	42%	0%	0%	0%	0%
Ex-Vessel Retention	0%	13%	42%	40%	42%	42%
Liner Melt-Through	100%	45%	58%	60%	58%	58%
Conditional Containment Failure Probability	100%	47%	58%	60%	58%	58%
Note: Containment overpressurization failure and liner melt-through, taken together, contain cross-product terms which are eliminated in arriving at the total conditional containment failure probabilitiy.						

Table 2-14 lists the significant RCs for each APET, which were input to the risk calculations made for each regulatory analysis sub-alternative.

Table 2-14 Frequencies of significant release categories

Index	Release Cateogry	Accident Progression Event Trees and Associated Regulatory Analysis Sub-Alternatives					
		APET-1 **1 - 1** **2 - 2A**	**APET-2** **3 - 3A** **5 - 4Ai(1)** **7 - 4Aii(1)** **9 - 4Aiii(1)** **11 - 4Bi(1)** **16 - 4Ci(1)**	**APET-3** **4 - 3B** **6 - 4Ai(2)** **8 - 4Aii(2)** **10 - 4Aiii(2)** **12 - 4Bi(2)** **17 - 4Ci(2)**	**APET-4** **15 - 4Biv** **20 - 4Civ**	**APET-5** **14 - 4Biii** **19 - 4Ciii**	**APET-6** **13 - 4Bii** **18 - 4Cii**
1	HP-DW-EVR				1.7E-07	2.1E-07	2.0E-07
2	HP-DW-LMT			1.2E-07	4.0E-07	4.0E-07	3.3E-07
3	HP-WW-EVR			1.8E-07			
4	HP-WW-LMT	2.8E-07	1.6E-07	2.9E-07			
5	SRV-DW-EVR		1.7E-07	5.6E-07	3.1E-06	3.4E-06	3.1E-06
6	SRV-DW-IVR		5.7E-07				
7	SRV-DW-LMT	1.3E-06	5.5E-07	1.1E-06	4.5E-06	4.5E-06	3.8E-06
8	SRV-OP-IVR		1.6E-07				
9	SRV-OP-LMT	3.5E-07	1.2E-07	1.6E-07			1.6E-07
10	SRV-WW-EVR		9.1E-07	2.9E-06		9.9E-08	3.6E-07
11	SRV-WW-IVR		3.0E-06				
12	SRV-WW-LMT	6.9E-06	3.1E-06	3.4E-06		1.7E-07	6.8E-07
Total of Significant RCs		**8.8E-06**	**8.8E-06**	**8.7E-06**	**8.2E-06**	**8.7E-06**	**8.6E-06**
Core-Damage Frequency		**8.9E-06**	**8.9E-06**	**8.9E-06**	**8.4E-06**	**8.9E-06**	**8.9E-06**
Fraction of CDF Captured in the APET		**99%**	**99%**	**98%**	**98%**	**98%**	**97%**

2.4 References for Chapter 2

1. U.S. Nuclear Regulatory Commission, SECY-15-0085, "Evaluation of the Containment Protection & Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26)," June 18, 2015, ADAMS Accession No. ML15022A218.

2. U.S. Nuclear Regulatory Commission, SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments (REDACTED VERSION)," November 26, 2012, ADAMS Accession No. ML12345A030.

3. U.S. Nuclear Regulatory Commission, Order EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," March 12, 2012, ADAMS Accession No. ML12054A694.

4. U.S. Nuclear Regulatory Commission, Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events", March 12, 2012, ADAMS Accession No. ML 12054A736.

5. U.S. Nuclear Regulatory Commission, Order EA-13-109, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," June 6, 2013, ADAMS Accession No. ML13143A334.

6. U.S. Nuclear Regulatory Commission, NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," September 2004, ADAMS Accession No. ML042820192.

7. U.S. Nuclear Regulatory Commission, NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," October 2010, ADAMS Accession No. ML050190193.

8. U.S. Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 11, 2011, ADAMS Accession No. ML111861807.

9. U.S. Nuclear Regulatory Commission, RG 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009, available through the NRC's public Web site under "Regulatory Guides" at http://www.nrc.gov/reading-rm/doc-collections/.

10. Nuclear Energy Institute, NEI 12-06, Rev. 2, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," December 2015, ADAMS Accession No. ML16005A625.

11. U.S. Nuclear Regulatory Commission, NUREG/CR-6890, Vol. 1, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," December 2005, ADAMS Accession No. ML060200477.

12. Idaho National Laboratory, INL/EXT-15-34441, "System Study: Emergency Power System 1998–2013," January 2015, http://nrcoe.inel.gov/resultsdb/publicdocs/SystemStudies/eps-SPAR-unreliability-external-2013.pdf

13. Tennessee Valley Authority, "Tennessee Valley Authority's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014, ADAMS Accession No. ML14098A478.

14. Duke Energy Progress, Inc., "Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014, ADAMS Accession No. ML14106A461.

15. Nebraska Public Power District, "Nebraska Public Power District's Seismic Hazard and Screening Report (CEUS Sites) - Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," March 31, 2014, ADAMS Accession Nos. ML14094A040 and ML14094A042.

16. NextEra Energy Duane Arnold, LLC, “NextEra Energy Duane Arnold, LLC Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 28, 2014, ADAMS Accession No. ML14092A331.

17. DTE Electric Company, “DTE Electric Company's Seismic Hazard and Screening Report, Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 31, 2014, ADAMS Accession No. ML14090A326.

18. Entergy Nuclear Northeast, “Entergy Seismic Hazard and Screening Report (CEUS Sites), Response NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 31, 2014, ADAMS Accession No. ML14090A243.

19. Southern Nuclear Operating Company, Inc., Edwin I. Hatch Nuclear Plant - Units 1 and 2, Seismic Hazard and Screening Report for CEUS Sites,” March 31, 2014, ADAMS Accession No. ML14092A017.

20. PSEG Nuclear LLC, “PSEG Nuclear LLC's Seismic Hazard and Screening Report (CEUS Sites) Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident - Hope Creek Generating Station,” March 28, 2014, ADAMS Accession No. ML14087A436.

21. Exel Energy, “MNGP Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” May 14, 2014, ADAMS Accession No. ML14136A288.

22. Exelon Generation Company LLC, “Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 31, 2014, ADAMS Accession No. ML14090A247.

23. Entergy Nuclear Operations, Inc., “Entergy's Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Seismic Aspects of Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 31, 2014, ADAMS Accession No. ML14092A023.

24. Exelon Generation Company LLC, “Exelon Generation Company, LLC, Seismic Hazard and Screening Report (Central and Eastern United States (CEUS) Sites), Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 31, 2014, ADAMS Accession No. ML14090A526.

25. U.S. Nuclear Regulatory Commission, “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident,” March 12, 2012, ADAMS Accession No. ML12056A046.

26. Electric Power Research Institute, EPRI Report 1025287, “Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” November 2012, ADAMS Accession No. ML12333A170.

27. Idaho National Laboratory, “Summary of SPAR Component Unreliability Data and Results: 2010 Parameter Estimation Update,” http://nrcoe.inel.gov/resultsdb/publicdocs/AvgPerf/ComponentUR2010.pdf

28. U.S. Nuclear Regulatory Commission, NUREG/CR-7155 (draft), “State-of-the-Art Reactor Consequence Analyses Project, Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” August 2013, ADAMS Accession No. ML13189A145.

3 ACCIDENT PROGRESSION ANALYSIS

This report documents analysis of selected accident scenarios in representative boiling-water reactor (BWR) plants with Mark I and Mark II containments. This supports the staff's ongoing effort to address the Near-Term Task Force (NTTF) recommendation related to the containment venting issue [1]. Specifically, the work reported herein was performed in support of the ongoing Containment Protection and Release reduction (CPRR) rulemaking activities in response to the staff requirement memorandum for SECY-12-0157 [2], using the U.S. Nuclear Regulatory Commission (NRC) severe accident analysis code MELCOR [3]. This former reference documents staff's previous analytical work of a similar nature, which formed the technical basis of the SECY.

The scope of MELCOR analysis falls broadly into two categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions; and (2) assessment of source terms, i.e., timing and magnitude of fission products releases to the environment. The outcome of MELCOR analysis in the first category includes, for example, containment temperature and pressure signatures and hydrogen distribution in the containment, reactor building, vent line, etc. – all indicative of the state of containment vulnerability under severe accident conditions. These quantities provide needed information to assess containment integrity and also provide technical basis for developing staff guidance; for example, for the severe accident capable hardened vent order EA-13-109 [4]. The outcome of MELCOR analysis in the second category is environmental release estimates that are used to calculate health consequence and offsite property damage assessment using the MELCOR Accident Consequence Code System, MACCS [5], discussed in Chapter 4. The MELCOR/MACCS results, along with the results of the accident sequence analysis, are used to estimate the relative public health risk reduction associated with the various accident prevention and mitigation strategies.

The selection of accident scenarios considered for MELCOR analyses is informed by the recent state-of-the-art reactor consequence analysis or SOARCA [6], the Fukushima study [7], and also by the previous analytical work documented in SECY-12-0157. Specifically, the accident scenarios selected for MELCOR/MACCS analyses relate to extended loss of alternating-current power (ELAP). The ELAP results in a loss of offsite power (LOOP), failure of onsite power, and failure of the grid. All systems dependent on alternating current (AC) power are unavailable. The turbine-driven reactor core injection cooling (RCIC) system (equivalently, the isolation condenser in some plants) is the only system assumed available. It is assumed that the high pressure coolant injection or high pressure core spray system is not available.

For various accident progression events (defined in in terms of accident progression event tree or APET), MELCOR cases were run simulating different possible outcomes or plant damage states (e.g., containment failure by overpressurization, drywell liner melt-through, main steam line rupture). Consideration was given to various preventative and mitigative measures and how these influence the failure modes.

The MELCOR calculations, described in considerable detail in the rest of this Chapter, are deterministic in nature. The calculations produce point estimates of the quantities of interest (e.g., radionuclide release fractions). There are phenomenological uncertainties in the code models, and, as a result, the predicted point estimates also have some uncertainties. With regard to the containment venting issue, pertinent uncertainties are discussed in a latter section of this chapter.

MELCOR BWR models are described in considerable detail in Section 3.1 . The representative Mark I plant model is based on the SOARCA study referenced above, and the insights from the

accident progression results (e.g., mode of containment failure or venting) are used in the risk assessment of other Mark I containments as well. The representative Mark II plant model is based on a LaSalle-type configuration. Scoping analysis was done to examine the response of the plant for other Mark II configurations. Section 3.2 discusses the basis of the accident scenario development and formulation of the MELCOR run matrix. Results of MELCOR analysis are discussed in detail in Section 3.3 for both Mark I and Mark II representative plants. Section 3.4 provides a discussion on uncertainties and the implications on the results. A summary of important results and the conclusions are documented in Section 3.5 .

3.1 MELCOR BWR Models

The following sections detail the representative Mark I and Mark II models used in the present analyses. The goal of this section is to provide sufficient information for readers to understand the system and models. The representative nuclear power plants for the BWR Mark I and Mark II are the Peach Bottom Unit 2 and LaSalle Unit 2 Power Stations.

3.1.1 Description of Mark I MELCOR Model

The BWR/4 Mark I input model described here follows the “best practice” used in the SOARCA study and reflects the current understanding in severe accident modeling. Modification of the model was necessary to perform the present analyses given the inclusion of the FLEX equipment and sequence events. The model was updated to permit the use of the most recent MELCOR 2.1 computer code. Details of the model are provided in SECY-12-0157. Some of the major model characteristics are given below.

3.1.1.1 Reactor Pressure Vessel and Reactor Coolant System

Figure 3-1 shows a representation of the MELCOR control volumes and flow paths for the reactor coolant system, and Figure 3-2 provides a detailed reactor vessel nodalization comparing MELCOR modeling features to actual vessel design. Control volumes are indicated by “CV” followed by the three-digit control volume number, and flow paths are indicated by “FL” followed by the three-digit flow path number.

The reactor pressure vessel is modeled with seven control volumes outside of the core region:

- Lower plenum (CV320)
- Downcomer (CV310)
- Shroud dome or upper plenum (CV345)
- Steam separators (CV350)
- Steam dryers (CV355)
- Steam dome (CV360)
- Jet pumps (CV300)

Reactor vessel upper internals are modeled in detail. The steam dryer region includes all volume inside of the dryer skirt and the dryers from the top of the steam separators to the top of the steam dryers. Water stripped from steam in the separators and dryers is returned to the downcomer volume.

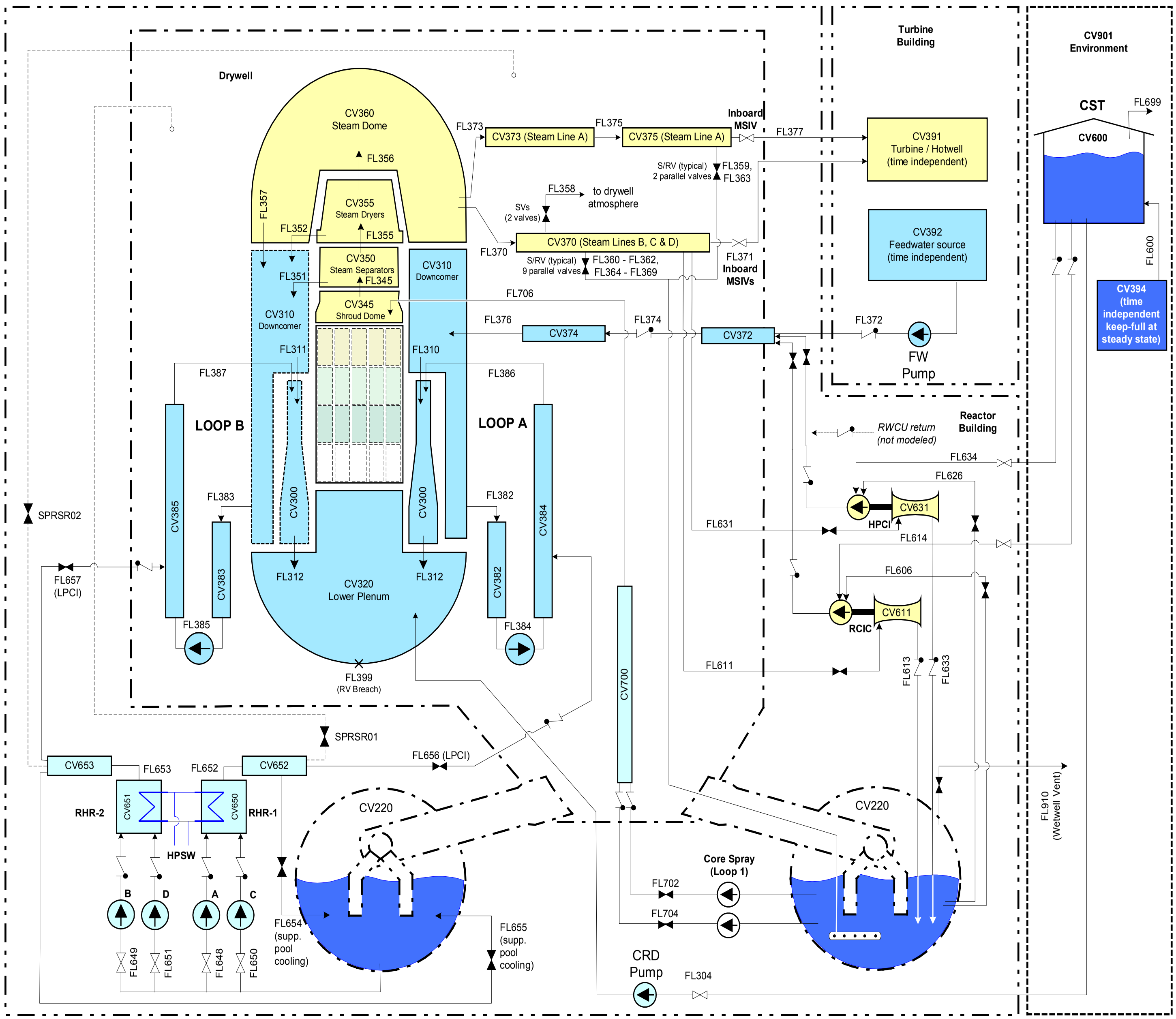


Figure 3-1 MELCOR reactor coolant system nodalization for Mark I analysis

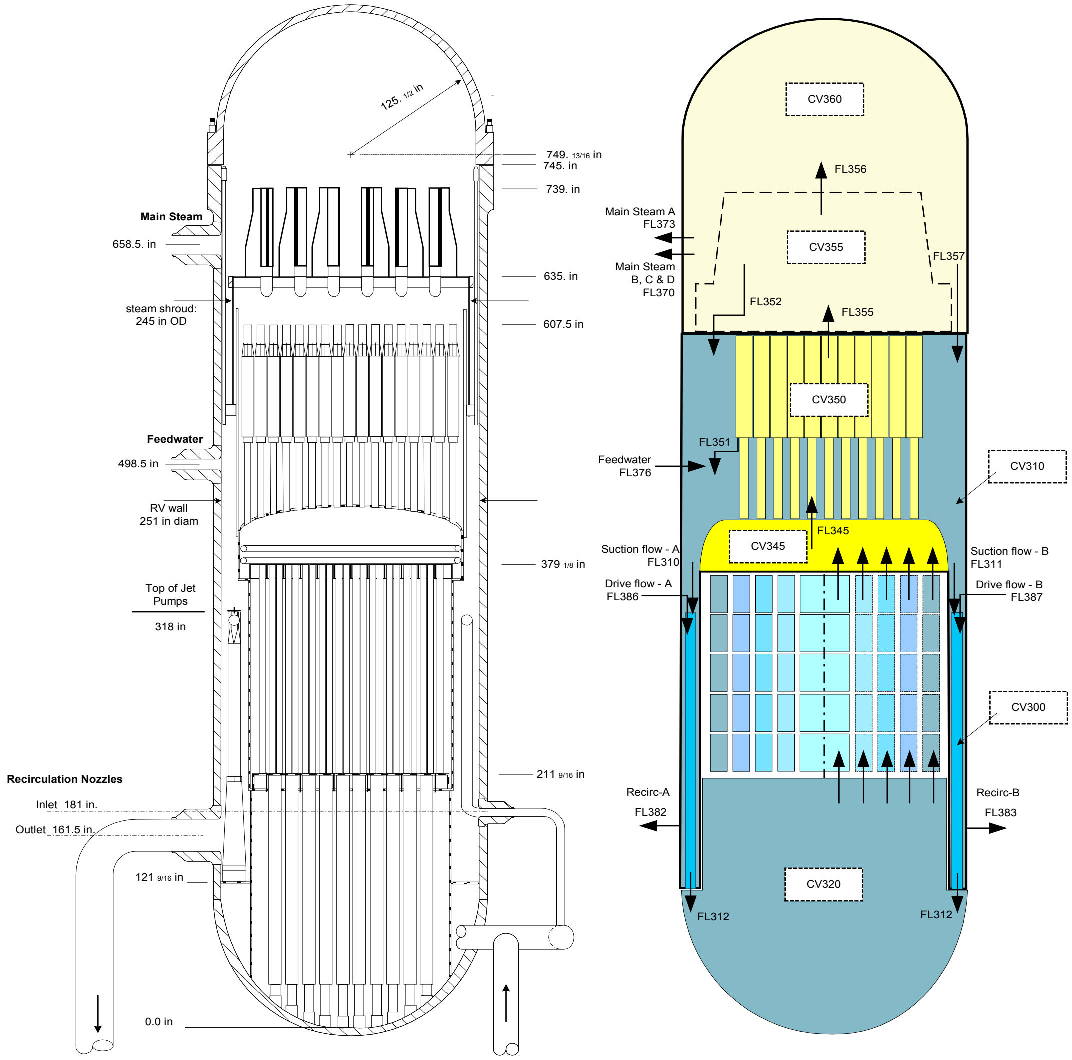


Figure 3-2 MELCOR reactor vessel nodalization for Mark I analysis

Flow paths are designed to represent all potential fluid pathways between the control volumes defined above. The 9 flow paths modeled connecting reactor pressure vessel control volumes include flow between:

- Jet pumps and lower plenum (FL312)
- Shroud dome/upper plenum and steam separator standpipes (FL345)
- Steam separators and steam dryers (FL355)
- Steam dryers and steam dome (FL356)
- Steam dome and downcomer (FL357)
- Loop A suction flow from the downcomer to the jet pumps (FL310)
- Loop B suction flow from the downcomer to the jet pumps (FL311)
- Steam separators and downcomer (FL351)
- Steam dryers and downcomer (FL352)

The heat capacity and radionuclide deposition surface of a number of structures associated with the reactor pressure vessel are modeled via heat structures including:

- Cylindrical portion in the lower downcomer region
- Cylindrical portion in the upper downcomer region
- Cylindrical portion adjacent to the steam dryers
- Hemispherical upper head
- Shroud baffle
- Standpipes and steam separators
- Steam dryers
- Core shroud

Depressurization of the RCS is performed with 11 flow paths (FL359-FL369) that represent the S/RVs, and flow path FL358, which represents the two safety valves (SV). Each valve within a group is represented with a differentiating pressure of 1 psi between each SRV actuation set point starting from the prescribed group actuation set pressure. This small differential pressure is sufficient to impose nearly all automatic SRV operations to a single SRV as modeled by MELCOR.

3.1.1.2 Reactor Core

In MELCOR, the region tracked directly by the COR package model includes a cylindrical space extending axially from the inner surface of the vessel bottom head to the core top guide and radially from the vessel centerline to the inside surface of the core shroud. The region tracked by the COR package also includes the region of the lower plenum outside of the core shroud and below the downcomer. The core and lower plenum regions are divided into concentric radial rings and axial levels. A particular radial ring and a particular axial level define a core cell (node) which has a cell number defined with a three digit integer IJJ, where the first digit represents the radial ring number and the last two digits represent the axial level number. For example, core cell number 314 specifies a cell located in radial ring three and axial level 14. The numbering of axial segments begins at the bottom of the vessel. Each core cell may contain one or more core components, including fuel pellets, cladding, canister walls, supporting structures (e.g., the lower core plate and control rod guide tubes), non-supporting structures (e.g., control blades, the upper tie plate and core top guide) and particulate debris.

The MELCOR core nodalization for the current containment filtered venting study is the same as SOARCA analysis as shown in Figure 3-3. The entire core and lower plenum regions are divided into six radial rings and 17 axial segments. Axial levels 1 through 6 represent the entire lower plenum and the unfueled region of the core immediately above the lower core plate. Initially this region has no fuel and no internal heat source. However, during the core degradation phase, the fuel, cladding and other core components may enter the lower plenum in the form of particulate or molten debris by relocation from the upper core nodes. Axial node 6 represents the steel associated with assembly lower tie plates, fuel nose pieces and the lower core plate and its associated supports. Particulate debris formed by fuel, canister, and control blade failures above the lower core plate will be supported at this level until the lower core plate yields. Axial segments 7 through 16 represent the active fuel region. All fuel is initially in this region and generates the fission and decay power. Axial level 17 represents the nonfuel region above the core, including the top of the canisters, the upper tie plate and the core top guide. Radial ring 6 represents the region in the lower plenum outside of the core shroud inner radius and below the downcomer region.

Core cell geometry and masses for nonfuel-related core components (e.g., control rod guide tubes, lower core plate, and core top guide) are obtained from a variety of references. Axial level 1 through 5 in rings 1 through 5 contains control rod stub tubes, control rod drives, and instrument guide tubes. Axial level 1 includes the region from the lower head to the top of the control rod stub tubes. Control rod stubs are modeled as tubes with a specified inner diameter and an outer diameter. Control rod drives are modeled as a solid shaft with a specified diameter representative of a BWR Mark I design. Fifty-five instrument tubes are modeled with each one including a guide tube with a specified inner diameter and an outer diameter, and a central shaft with a specified diameter. Control rod stub/drive and instrument tubes are distributed between the rings. The combined mass of the control rod stub tubes, control rod drives, and instrument tubes within axial level 1 are modeled as a stainless steel supporting structure. The surface area for this component is modeled as the outer surface area of the control rod stub tubes. Axial level 1 in ring 6, which is outside of the lower head curvature, does not contain any core components and is simply deactivated within the model.

Axial level 6 in rings 1 through 5 includes the fuel support pieces, lower core plate, lower core plate support structures and fuel assembly lower tie plates. The total mass for the fuel support pieces and lower core plate is distributed between the core rings based on the fraction of the area inside of the core shroud represented by the ring. Assembly lower tie plate mass depends on the type of fuel assemblies modeled, and is distributed based on the number of assemblies per ring. The combined mass of these structures is modeled as a steel support structure representing the lower core plate.

All control blades are assumed to be inserted in the core region at the time of the accident. Axial levels 7 through 16 in rings 1 through 5 contain the control blades distributed as described in axial level 1. Axial level 17 in rings 1 through 5 contains the core top guide and the fuel assembly upper tie plates.

Core cells within the five concentric rings modeling the active fuel region and the core top guide from axial levels 7 through 17 are coupled with a total of 40 hydrodynamic control volumes. Within each radial ring, five axially-stacked control volumes represent coolant flow through the core channels and five parallel (axially-stacked) control volumes represent the neighboring bypass regions of the core. This reflects a coupling between core cells and hydrodynamic control volumes within the core region.

Four distinct groups of flow paths are modeled to represent all potential flow within the core region. Axial core flow within the fuel assemblies is modeled with the channel flow area for each ring excluding flow area internal to the water rods. Axial flow paths from the lower plenum into the fuel assembly channel include pressure losses associated with flow through the fuel support piece orifices and the lower tie plate. Axial flow paths between volumes within the core region include friction losses for flow through fuel rods over a volume-center to volume-center length and form losses based on grid spacers. Axial flow from the upper fuel region control volume and the upper plenum includes form losses for flow through the upper tie plate. The MELCOR axial flow blockage model is activated for each of these flow paths. Axial bypass core flow between canisters and through the peripheral bypass is modeled with the bypass flow area in the core region, including flow area internal to the water rods.

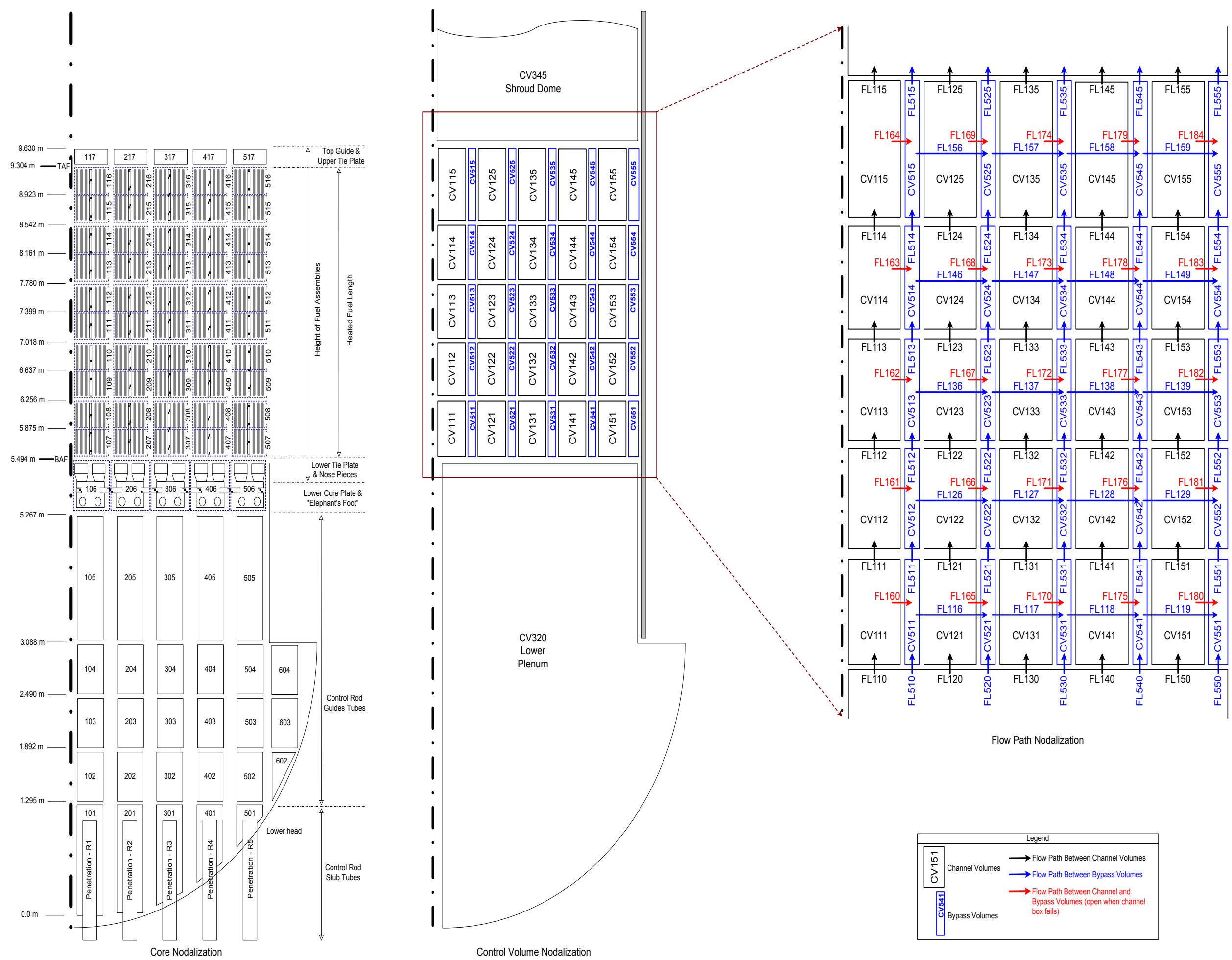


Figure 3-3 MELCOR core nodalization for Mark I analysis

At each axial level of the core, horizontal flow paths model the possibility of coolant cross-flow between channel and bypass areas. The open fraction for these flow paths is connected to control logic that monitors channel box integrity (i.e., the flow paths are closed when the channel box is intact and open if the channel box fails in a particular ring). In addition, coolant cross-flow between bypass regions is modeled by horizontal flow paths between each ring at each axial level.

The lower head is modeled as a hemisphere with an inner radius and thickness representative of a BWR Mark I plant. The lower head region extends to the downcomer baffle plate where it connects with the reactor pressure vessel. The hemispherical region of the lower head is represented by eight segments, and the cylindrical region of the lower head below the baffle plate is represented by a single segment. A one-dimensional model of the stress and strain distribution in the lower head is applied. Heat transfer coefficients from particulate debris to the lower head and penetrations are modeled with a temperature-dependent control function which reflects conduction-based heat transfer through a frozen crust at temperatures of 2650K and below, a conduction enhanced heat transfer coefficient as the debris reaches the eutectic melting temperature of UO_2 and ZrO_2, and a convective heat transfer coefficient as the debris exceeds the eutectic melting temperature and forms a circulating molten pool. By default, the penetration failure model is deactivated and the lower head failures occur due to creep rupture (same assumption as in SOARCA).

3.1.1.3 Containment

The primary containment is subdivided into seven distinct control volumes. The drywell is represented by the following four control volumes:

- Region internal to the reactor pedestal including the drywell sumps (CV205)
- Region external to the drywell pedestal from the floor to an elevation of 165' (CV200)
- Region from 165 feet to the drywell head flange (CV201)
- Region above the drywell head flange (CV202)

The wetwell is represented by the following two control volumes:

- 1/16th segment of the wetwell (CV221)
- Remaining 15/16th of the wetwell (CV222)

One control volume represents the vent pipes and downcomers connecting the drywell to the wetwell (CV210). The MELCOR nodalization of the primary containment is shown in Figure 3-4.

A total of 18 flow paths represent the containment flow pathways. Of these, two flow paths (FL200 and FL202) connect the three drywell regions external to the reactor pedestal. Each of these flow paths is modeled with 50% of the interfacing flow area between the control volumes. This assumes a 50% obstruction by equipment and structures of the interface between the drywell regions. In addition, two flow paths (FL201 and FL203) allow for natural circulation inside the drywell.

Three flow paths (FL014, FL015, and FL016) connect the reactor pedestal to the lower drywell. The open fraction of the personnel doorway is reduced based on the core debris elevation in the reactor pedestal after vessel failure (debris elevation determined from CAV package). Two additional flow paths (FL012 for flow from the drywell to the vent pipes and FL017 for nominal drywell leakage from the lower drywell to the reactor building) represent flow from the drywell. The nominal drywell leakage flow area, friction, and form losses are defined to match the nominal drywell leak rate. The elevation of nominal containment leakage through the drywell is modeled at the dominant location of drywell penetrations.

The downcomer vent exits to the two wetwell control volumes are represented by FL801 and FL802. Each flow path has the SPARC fission product pool scrubbing model activated within MELCOR for aerosols and vapors across all fission product classes.

Four flow paths (FL821, FL822, FL022, and FL023) model vacuum breakers intended to limit under-pressure failures of the drywell and wetwell. The wetwell-drywell vacuum breakers (FL821 and FL822, representing connections with each wetwell control volume) open whenever the wetwell pressure exceeds the vent pipe pressure by 0.5 psid. The reactor building-wetwell vacuum breakers connect the wetwell airspace of the larger wetwell control volume with the northeast and southeast torus corner rooms, and open whenever the pressure in the wetwell drops 2 psi below the pressure in the reactor building.

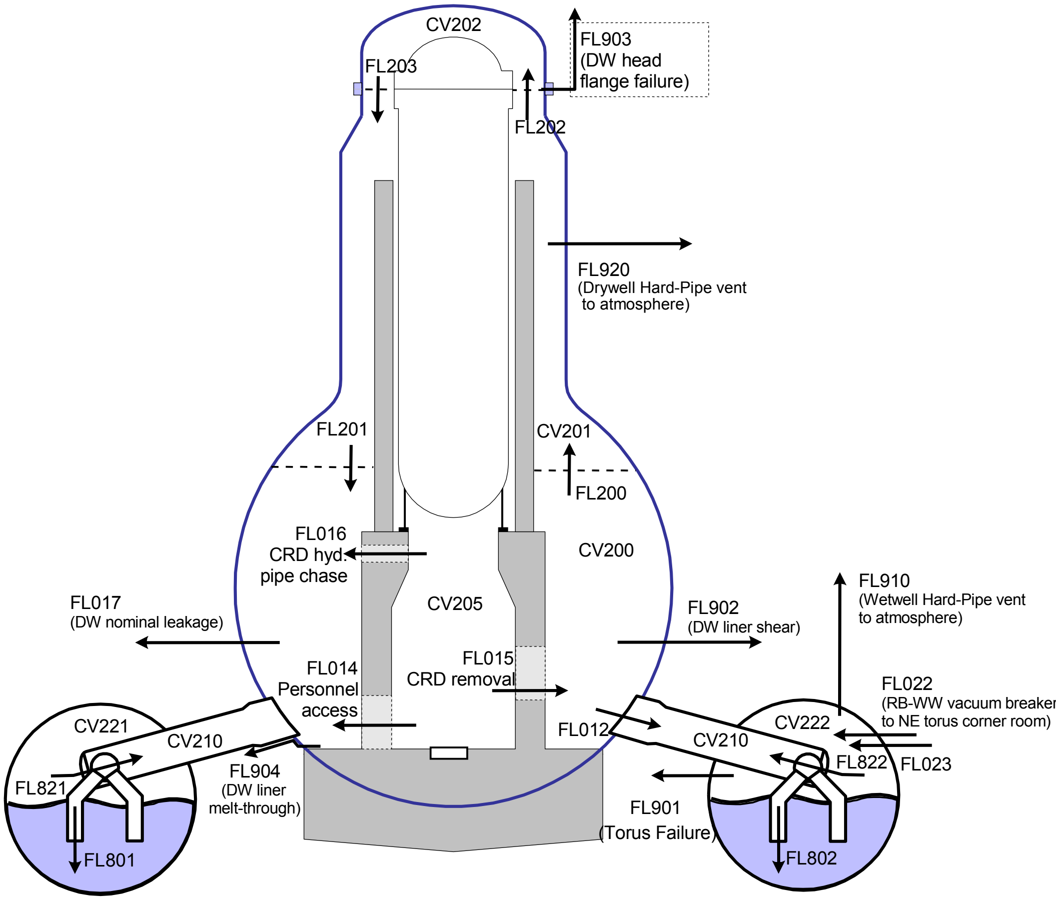


Figure 3-4 MELCOR nodalization of the primary containment for Mark I analysis

Four flow paths (FL901, FL902, FL903, and FL904) represent the flow through various potential breach locations. FL901 represents the torus failure location, FL902 the drywell liner shear, FL903 the head flange leakage, and FL904 the drywell liner melt-through.

The wetwell is modeled using two control volumes to capture the RCIC and single SRV exhaust and improve pressurization trends observed during the SBO accidents observed at Fukushima. A single well-mixed control volume under-predicts containment pressurization rates observed during an SBO when only the lowest set point SRV is operating. Enhanced evaporation of pool water and uncondensed vapor escaping the pool due to increased local wetwell temperatures cannot be captured with a single well-mixed control volume. A 1/16th segment of the wetwell torus (CV221) represents the near-field region of the lowest set point SRV t-quencher to promote higher local temperature predictions. The remaining wetwell torus control volume (CV222) interacts with the remaining SRVs, the RCIC system, vacuum breakers, wetwell vent lines, and torus failure. The two control volumes are connected through flow path FL851 to permit the transport of water and atmospheric material between the wetwell control volumes. The SRV exhaust while using a single SRV in the MELCOR model was set to the larger wetwell control volume (CV222) to enforce uniform heatup in agreement with operators alternating among SRVs during relief mode operations.

Heat structures in the containment model are represented include the following:

- Drywell floor outside of the reactor pedestal
- Drywell floor inside of the reactor pedestal
- Drywell liner-air gap-concrete wall (between primary and secondary containment)
- Drywell liner representing the cylindrical and dome portions
- Biological shield wall in the lower drywell
- Biological shield wall in the mid-drywell
- Reactor pedestal
- Misc. drywell steel in the lower drywell
- Misc. drywell steel in the mid-drywell
- Misc. horizontal deposition surfaces in the lower drywell
- Wetwell torus liner
- Wetwell miscellaneous steel (equipment and structures)

The heat structure film-tracking model is activated to connect water film flows between the appropriate drywell liner heat structures.

3.1.1.4 Reactor Cavity

The drywell floor is subdivided into three regions for the purposes of modeling molten-core/concrete interactions. The first region (which receives core debris exiting the reactor vessel) corresponds to the reactor pedestal and sump floor areas (CAV 0). Debris that accumulates in the pedestal can flow out into the second region (through a doorway in the pedestal wall), corresponding to a 90 degrees sector of the annular portion of the drywell floor (CAV 1). If sufficient debris accumulates in this region, it can spread further into the third region, which represents the remaining portion of the drywell floor (CAV 2). This discrete representation of debris spreading is illustrated in Figure 3-5.

Two features of debris relocation within the three cavities are modeled. The first models debris overflow from one cavity to another. The second manages debris spreading radius within the drywell floor region cavities (CAV 1 and 2). Control functions monitor debris elevation and temperature within each region, both of which must satisfy user-defined threshold values for debris to move from one region to its neighbor. More specifically, when debris in a cavity is at or above the liquidus temperature of concrete, all material that exceeds a predefined elevation above the floor/debris surface in the adjoining cavity is relocated (6 inches for CAV 0 to CAV 1, and 4 inches for CAV 1 to CAV 2). When debris in a cavity is at or below the solidus temperature of concrete, no flow is permitted. Between these two debris temperatures, restricted debris flow is permitted by increasing the required elevation difference in debris between the two cavities (more debris head required to flow).

Debris entering CAV 1 and CAV 2 are not immediately permitted to cover the entire surface area of the cavity floor. The maximum allowable debris spreading radius is defined as a function of time. When the cavity debris temperature is at or above the liquidus, the shortest transit time (and therefore maximum transit velocity) of the debris front to the cavity wall is determined (10 minutes for CAV 1 as defined in MELCOR control function CF960, and 30 minutes for CAV 2 as defined in control function CF961). When the debris temperature is at or below the solidus, the debris front is assumed to be frozen. A linear interpolation is performed to determine the debris front velocity at temperatures between these two values. The CAVITY package model implemented enforces full mixing of all debris into a single mixed layer.

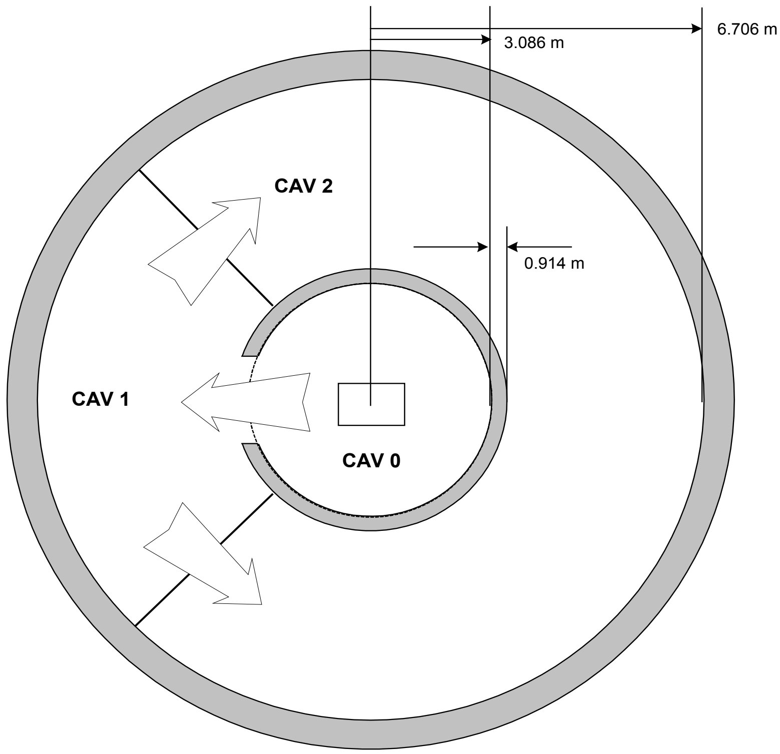


Figure 3-5 Discrete representation of debris spreading in the cavity

The solidus and liquidus temperatures in the parametric model that govern the rate of debris spreading on the drywell floor were modified in the present study. Original values of solidus and liquidus temperatures in the SOARCA model were 1,420K and 1,670K, respectively. These temperatures are representative of concrete solidus and liquidus. For containment venting calculations, the solidus and liquidus temperatures were changed to 1,700K and 2,800K, respectively which are more representative of the debris properties rather than concrete to better approximate the debris composition shortly after lower head failure which impacts debris spreading and the potential for liner melt-through. The revised liquidus temperature is representative of the liquidus temperature of a eutectic UO_2/ZrO_2 mixture. The revised solidus temperature was set at 1,700K to represent the lower bound of average melt temperature at vessel breach, and happens to coincide approximately with the melting point of steel. In the model, spreading is disallowed at debris temperatures less than the solidus temperature and occurs at a maximum rate (0.259 m/min) when debris temperature is above the liquidus temperature. Spreading rate varies linearly at temperatures between the solidus and liquidus temperatures.

3.1.1.5 Balance of Plant

A total of 41 control volumes, 71 flow paths, and 85 heat structures are modeled to represent all pertinent structures external to primary containment. These model elements represent the reactor building, turbine building, radwaste building, and the environment. Given its importance as a fission product release pathway if the containment fails, the reactor building is modeled in significant detail (30 control volumes and 80 heat structures).

A sectional view of the reactor building is shown in Figure 3-6. It is modeled on a level-by-level basis, beginning in the basement (i.e., torus room) and sequentially rising up through the main floors to the refueling bay.

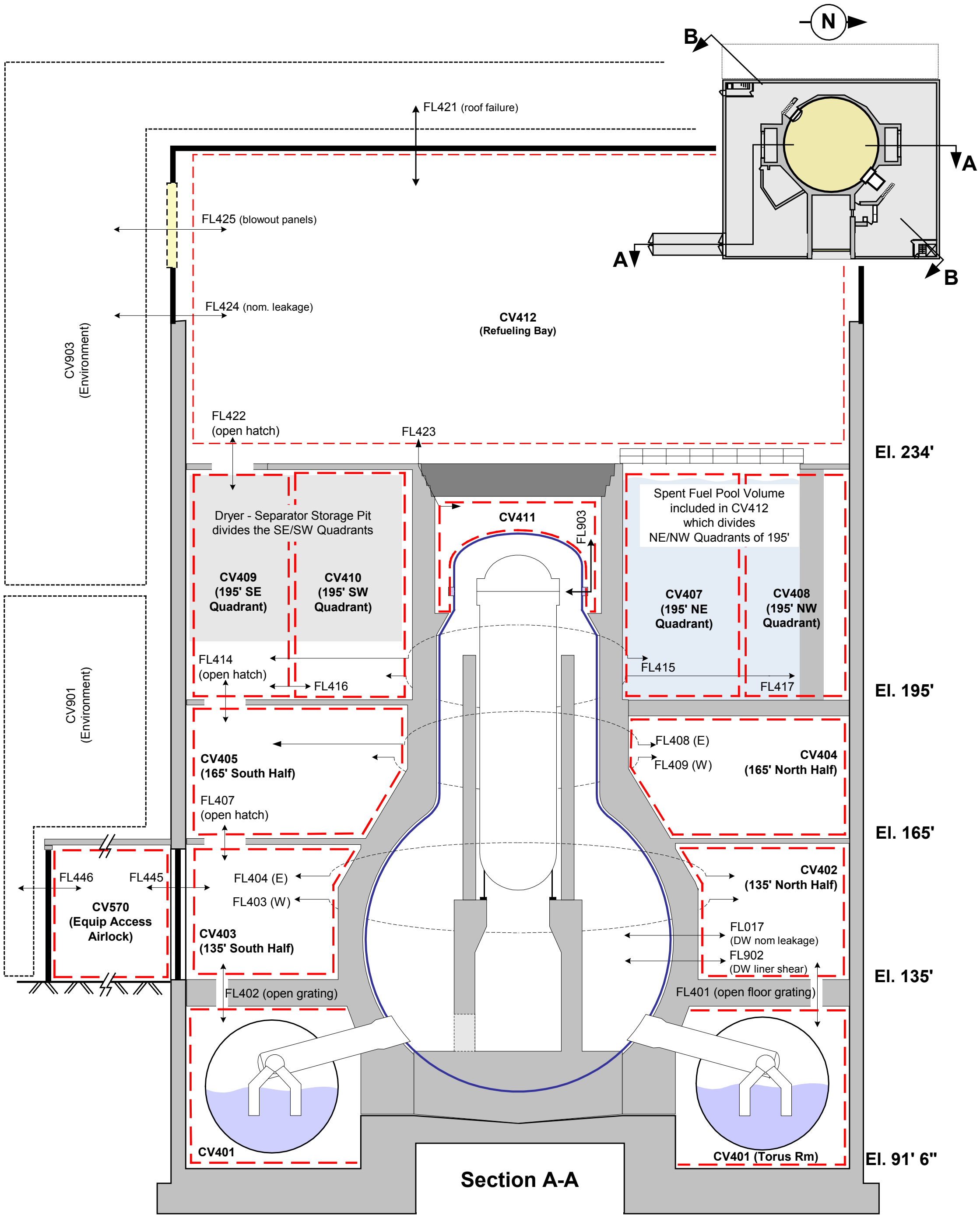


Figure 3-6 MELCOR nodalization of the reactor building for Mark I analysis

The torus room level of the reactor building is represented by eight control volumes. These include volumes representing the main torus room, the northeast corner room, the stairwell in the northeast corner of the building, the southeast corner room, and the RHR A, B, C, and D heat exchanger and pump rooms. The next higher level of the reactor building is modeled by five control volumes. These include volumes representing the southern half of the building, the northern half of the building, the southwest stairwell enclosure, the northeast stairwell enclosure, and the steam tunnel. The next higher level of the reactor building is represented by five control volumes. The next higher level of the reactor building is represented by eight control volumes. These volumes represent the northeast, southeast, northwest, and southwest quarters of the floor; the reactor building ventilation room; the drywell enclosure; the southwest stairwell enclosure; and the northeast stairwell enclosure. The refueling bay level (highest level) of the reactor building is represented by four control volumes. These volumes represent the open refueling bay (including the spent fuel pool but neglecting the separator/dryer storage pit), the southwest stairwell enclosure, the northeast corner room and the northeast stairwell enclosure. The flow paths modeled within the reactor building can be classified into the following categories: same level flows between distinct control volumes, open hatches, doors, blowout panels, flow pathways through walls, leakage pathways, stairwells and concrete hatches.

3.1.2 Description of Mark II MELCOR Model

To support the filtered/venting analyses, the original model from Reference [8] was converted to the input format standards of MELCOR 2.1. Unlike the extensive SOARCA input development performed for the Mark I analysis, the original BWR/5 Mark II input deck was modified primarily with the intention of performing the present analyses.

Any significant variation between the input used here and that of the original model are presented below in addition to the general details necessary to illustrate the model.

A detailed description of the secondary containment system is not provided. Given the containment failures assumed for the Mark II model, discussed in Section 3.1.3.4 , liner melt through is not considered. Furthermore, neither head flange leakage nor failure were observed during the accident analyses. These findings reduce the interactions between the primary and secondary containment systems to heat transfer through the boundary structures for the analyses provided. The reactor building is not a significant contribution to the findings of the analyses for Mark II.

3.1.2.1 Reactor Pressure Vessel and Reactor Coolant System

The RPV model is composed of 7 control volumes, 14 flow paths, and 18 heat structures. Figure 3-7 depicts the facility nodalization and corresponding flow path connections of the RPV, with the exception of the channel and bypass control volumes.

The RPV modeled includes the following control volumes:

- Lower plenum (CV100)
- Channel (CV111)
- Bypass (CV121)
- Upper plenum/separators (CV103)
- Dryers/steam dome (CV104)
- Downcomer (CV105)
- Jet pumps (CV300)

The downcomer control volume (CV105) represents the volume between the core barrel and the reactor vessel wall (excluding jet pump volume) from the baffle plate to the top of the steam separators. The downcomer control volume includes all volume external to the steam separators in the region above the core shroud dome. The lower plenum control volume (CV100) includes all reactor vessel volume below the top of the core support plate excluding the downcomer region and jet pumps. All volume internal to the jet pumps is represented by CV300.

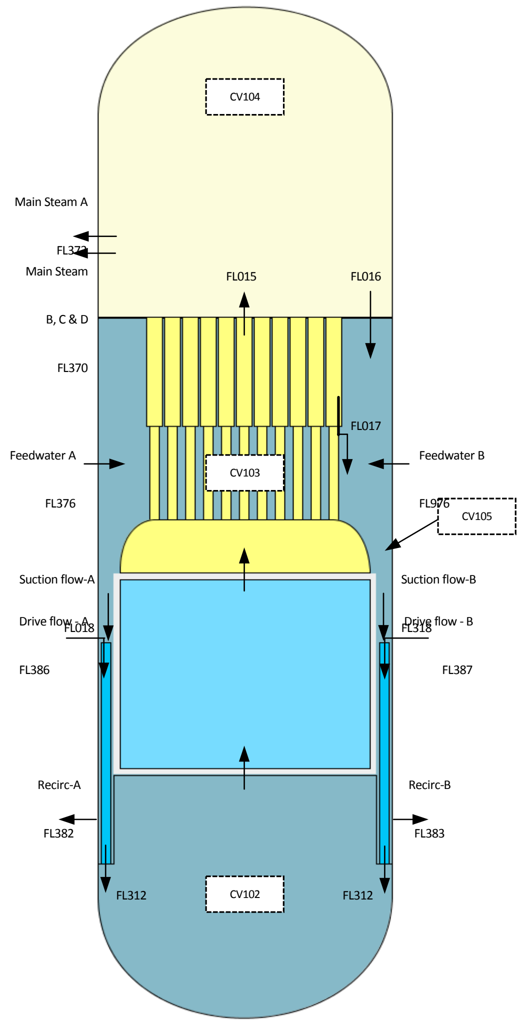


Figure 3-7 MELCOR reactor vessel nodalization for Mark II analysis

The total jet pump volume was extracted from the original model's lower plenum control volume, which included the jet pumps. The inlet and outlet height of the recirculation lines as well as the steady state flow control logic were adapted from the Mark I model. This practice was also performed to include the feedwater lines, feedwater pump control logic, and main steam lines. The original volumes of the recirculation loops and jet pumps were preserved from the original model.

A single lumped control volume represents the upper plenum mixing region in the shroud dome as well as the stand pipes and steam separators (CV103). In total, CV103 models the region

between the core top guides to the top of the steam separators. CV104, which is composed of the steam dryers and steam dome, receives steam from CV103. Water separated from the steam is returned to the downcomer control volume (CV105).

The flow paths present in the RPV are the following:

- Downcomer to jet pump A entrance (FL018)
- Downcomer to jet pump B entrance (FL318)
- Jet pump exit to lower plenum (FL312)
- Loop A recirculation water drawn from downcomer (FL382)
- Loop A recirculation water injected into jet pump loop (FL386)
- Loop B recirculation water drawn from downcomer (FL383)
- Loop B recirculation water injected into jet pump (FL387)
- Lower plenum to bypass (FL061)
- Lower plenum to channel (FL051)
- Bypass to upper plenum/steam separators (FL081)
- Channel to upper plenum/steam separators (FL071)
- Upper plenum/steam separators to downcomer (FL017)
- Upper plenum/steam separators to dryers/steam dome (FL015)
- Dryers/steam dome to downcomer (FL016)

Structures located within the RPV, not internally modeled as a core component, are represented as heat structures. The following heat structures are present in the RPV model:

External heat structures

- Cylindrical portion in the downcomer region (HS10501)
- Upper cylindrical portion of the vessel wall above the downcomer region (HS10403)
- Upper reactor pressure vessel hemispherical head (HS10402)

Internal heat structures

- Dryers (HS10401)
- Separators (HS10303)
- Core shroud in axial level JJ, (HS120JJ (HS12106 – HS12113))
- Lower plenum shroud (HS10004, HS10005, and HS10014)
- Upper plenum shroud (HS10301)
- Shroud dome (HS10302)

Given the single control volume representation of the drywell, the convective boundary for the outer surface of the heat structures representing the vessel cylinder and head (HS10402, HS10403, and HS10501) are bounded by the drywell volume (CV205). The cylindrical HS10403 and HS10501 inner convective boundary is set to the downcomer (CV105) and dryers/steam dome (CV104), respectively. The hemispherical HS10402 convective boundary is assigned to the dryers/steam dome (CV104). The separators (HS10303) are bounded by the downcomer (CV105) and upper plenum/separators (CV103). However, the dryers are bound on both convective surfaces to the dryers/steam dome (CV104) region. HS11101, the support plate, has convective boundaries assigned to the channel (CV111) and the lower plenum (CV100).

The modeling requirements for heat structures representing the shroud are divided between two regions, the shroud located above the core, which includes the shroud dome (HS10302) and upper plenum shroud (HS10301), and the shroud heat structures that are associated with core assigned control volumes. The representative heat structures modeling the shroud above the core are similarly modeled as the heat structures already discussed. The inner and outer convective boundaries of HS10301 and HS10302 are the upper plenum/separators and the downcomer (CV105). The shroud heat structures located in the core region (HS12106 – HS12113) each identify the bypass (CV121) as the inner convective boundary and the downcomer (CV105) as the outer convective boundary, the shroud located in the lower plenum (HS10004, HS10005, HS10014) identifies the lower plenum (CV100) and the downcomer (CV105) as the inner and outer convective boundaries, respectively.

The lowest set point SRV, represented by FL021, was separated from the lumped representation of all the SRVs modeled in the original input as FL022. Representing the lowest set point SRV as a single flow path allowed direct integration of the failure mode models from the Mark I model discussed in Section 3.1.3.3 . More significant pressure events may still actuate additional SRVs.

3.1.2.2 Reactor Core

The general discussion provided in Section 3.1.1.2 for the Mark I model is largely applicable to Mark II model; therefore, the provided discussion is limited to the model description and necessary modification.

The core and lower plenum region, which represents the physical space where core degradation is evaluated, is divided into 5 radial rings and 13 axial segments. Axial levels 1 through 5 represent the lower plenum and unfueled region immediately above the core plate. Axial level 5 represents the steel associated with the assembly lower tie plate, fuel support pieces, and core plate. Axial segments 6 through 12 represent the fueled region and axial segment 13 represents the region above the active fuel, e.g., tie plate, cladding, and core top guide. Ring 5 is only present in the lower plenum and characterizes the region outside the inner radius of the core shroud located beneath the downcomer baffle plate; no intact structural material exists within this region.

The core cell geometry and masses are retained from the original model. However, some modifications were necessary to accommodate the current lower head modeling approach in MELCOR. The current lower head model considers the curvature of the lower head unlike the flat cylindrical representation in the original model. The region impacted by the curvature of the lower head (axial levels 1 and 2) was therefore adjusted. These two axial levels were combined to form a single axial level, axial level 1, and the total component masses were redistributed per core cell volume fraction. In addition, a new axial level was added, dividing the original axial level 3, to indicate the base of the downcomer baffle plate. The vessel inner radius was used as an approximation for the radius for the hemispherical lower head. The resulting modifications maintained the same total number of axial segments within the model.

Coupling between the core cells and the hydrodynamic modeling performed in the control volumes is provided in Figure 3-8. Given the reduction in control volume fidelity, in comparison to the SOARCA model, the channel region for each core cell is associated with the control volume representing the channel (CV111) and bypass region of each core cell is associated with the control volume representing the bypass (CV121). Penetration failure modeling was deactivated in accordance with the Mark I modeling methodology discussed in Section 3.1.1.2 .

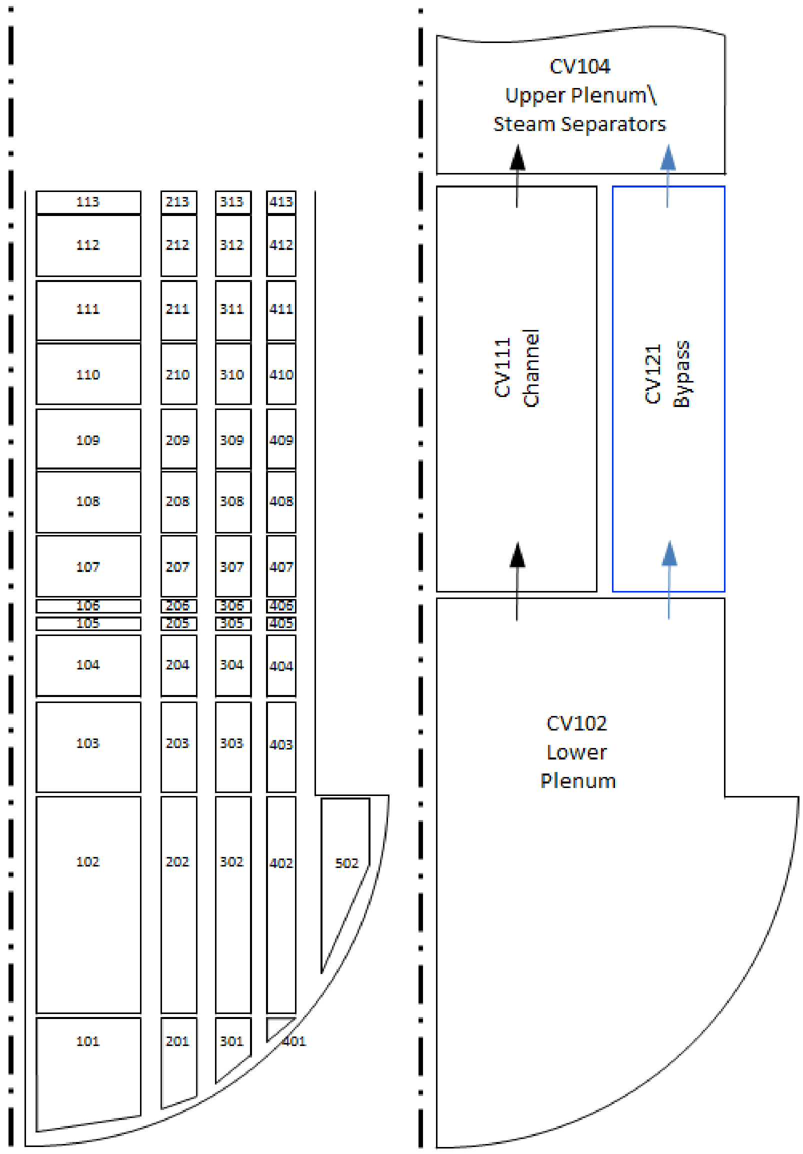


Figure 3-8 MELCOR core nodalization for Mark II analysis

3.1.2.3 Containment

The Mark II containment is represented by the following 5 control volumes:

- Wetwell (CV200)
- Downcomer vents (CV201)
- Lower reactor cavity (CV203)
- Upper reactor cavity (CV204)
- Drywell (CV205)

A total of 9 flow paths provide the interconnection among the control volumes representing the containment. The MELCOR nodalization of the primary containment is shown in Figure 3-9.

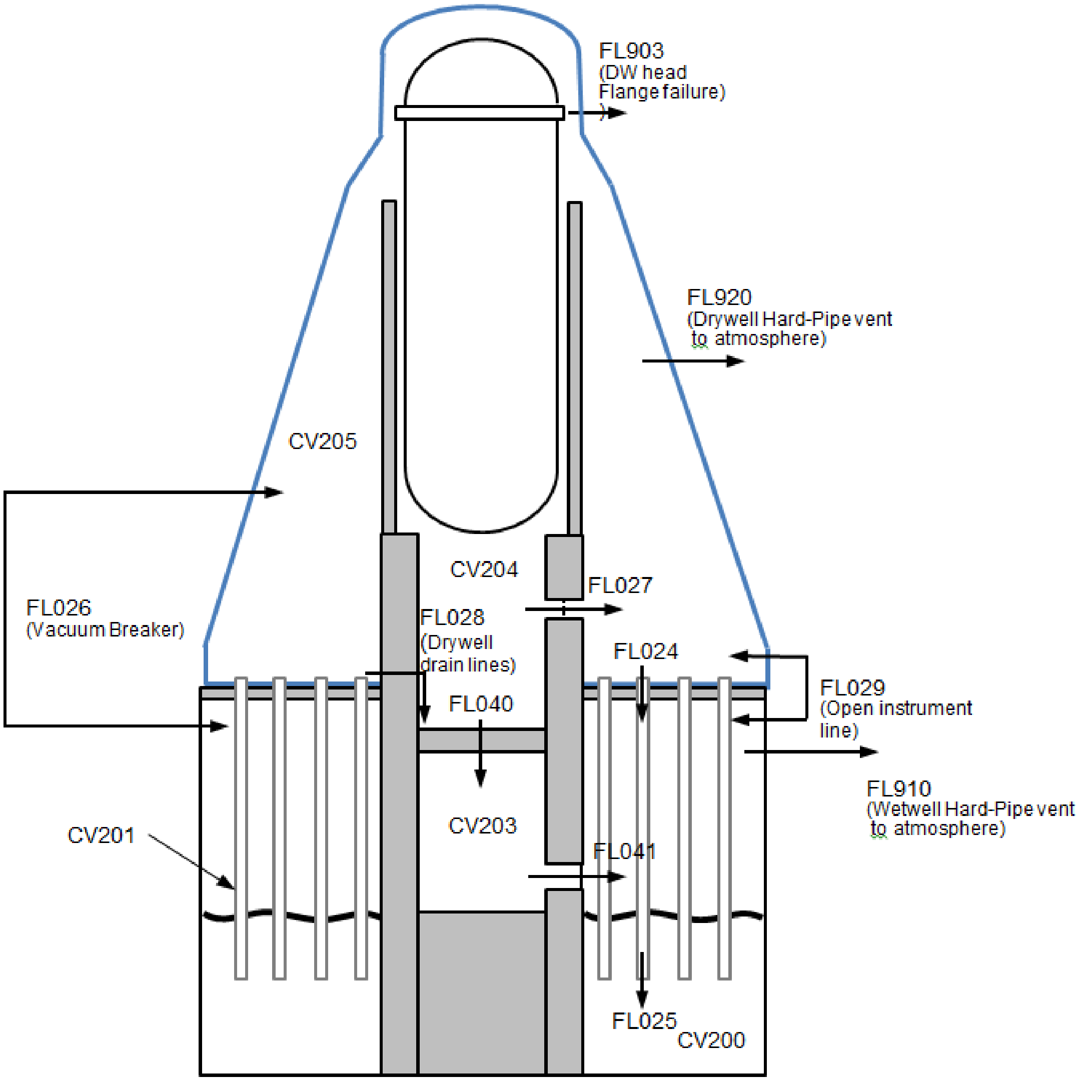


Figure 3-9 MELCOR nodalization of the primary containment for Mark II analysis

Interconnectivity between the drywell and wetwell is modeled by 4 flow paths. The inlet and outlet of the downcomer vents (CV201) are represented by FL024 and FL025, respectively to CV205 and CV200. The containment vacuum breakers are modeled by FL026 with a linear open fraction model with a differential pressure range of 0.0 and 3.0 psi. FL029 models an open ½-inch inerting system line connecting the drywell and wetwell.

Interconnectivity between the upper reactor cavity and drywell is modeled with 2 flow paths. Various accesses and ports are lumped into FL027 between the two volumes. Two 8-inch drain lines from the drywell floor to the upper reactor cavity are modeled with FL028. The access ports connecting the lower reactor cavity and wetwell are lumped together and modeled as FL041 (assumed open throughout the transient). FL040 represents two 4-inch sump drain lines through the upper reactor cavity floor to model prescribed failure between the upper and lower reactor cavity after lower head failure and debris relocation to the upper reactor cavity. Once the drain line fails, this flow path becomes fully open.

Containment heat structures present in the model are listed below.

- Wetwell wall (HS20001)
- Wetwell base slab (HS20002)
- Drywell floor support columns (HS20003)
- Wetwell pedestal section 2 (HS20004)
- Wetwell pedestal section 3 (HS20005)
- Wetwell pedestal section 4 (HS20006)
- Wetwell misc. steel (above water) (HS20007)
- Wetwell misc. steel (below water) (HS20008)
- Downcomers (HS20101)
- Drywell pedestal floor (HS20301)
- Drywell floor (HS20501)
- Reactor shield (HS20502)
- Drywell wall (HS20503)
- Drywell pedestal Section 1 (HS20504)
- Drywell head (HS20505)
- Drywell steel (vertical) (HS20506)
- Drywell steel (horizontal) (HS20507)
- Lower reactor cavity floor (HS10002) [added in the current model]

3.1.2.4 Reactor Cavity

Two cavities were modeled to represent the upper and lower reactor cavities in the representative Mark II containment. After lower head failure, debris relocates into the upper reactor cavity and initiates molten core concrete interaction. The ex-vessel debris is modeled as a single well-mixed metal/oxide layer assumed instantly covering the cavity floor.

Drain lines leading to the in-pedestal sump are a potential failure mode for the upper reactor cavity [9]. The challenge is that after core debris contacts the drainlines and fails them, it could provide a bypass from the drywell directly to the wetwell without going through the suppression pool. Within the model, failure of the upper reactor cavity may be identified through user input by selecting one of two methods. The first option initiates a timer to relocate the debris from the upper cavity to the lower cavity using a default delay of 20 minutes based on information provided in [9]. As a second option, the debris relocates to the lower cavity once the ablation depth exceeds the drywell pedestal floor thickness. However, this second option is used to investigate different cavity configuration among the fleet. The flow path representing the drain lines, FL040, opens upon upper reactor cavity failure, increasing the total bypass area available between the wetwell and drywell vapor space.

The addition of the debris from the upper reactor cavity instantly mixes with the existing steel mass in the lower cavity and core concrete interaction can resume once debris temperature exceed the ablation temperature.

3.1.3 Description of Mitigation Systems and Failure Modes

Performing the ELAP sequences with the Mark I and II models required modeling sequence events detailing operator actions as well as system availability and failure. The initial and boundary conditions for these systems are based on the interaction with the industry through public meetings (see Reference [10]).

The descriptions provided in this section are applicable to both the Mark I and II models, unless stated otherwise below. The following sections discuss the reactor core isolation cooling system, FLEX injection, SRV operations and failure modes, containment failure modes, and vent operations.

3.1.3.1 Reactor Core Isolation Cooling

Reactor core isolation cooling (RCIC) is the only considered in-situ injection system for the sequences analyzed. Nodalization for the RCIC system includes:

- RCIC turbine (CV611)
- Flow from main steam line C to the RCIC turbine (FL611)
- Flow from the RCIC turbine to the suppression pool (FL613)
- Flow from the CST to feedwater piping including the RCIC pump (FL614)
- Flow from the suppression pool to feedwater piping including the RCIC pump (FL606)

The model includes a constant-flow pump, delivering 600 gpm via velocity-specified flow paths, with suction initially aligned to the condensate storage tank (CST). Switchover of pump suction to the suppression pool occurs upon receipt of a low CST water level signal. The model permits initialization with the CST failed at the time of scram if desired (in this case suction is aligned to the suppression pool).

Steam flow through the RCIC turbine is modeled to account for the transfer of energy from the steam line to the suppression pool during RCIC operation. The flow of steam from main steam line to the RCIC turbine is modeled as a function of the pressure difference between the main steam line and the suppression pool. RCIC is modeled with automatic initiation and termination criterion. RCIC is initiated on receipt of a reactor vessel low level-2 signal[2]. RCIC is terminated on receipt of a reactor vessel high level-8 signal when DC power is available. For this analysis, manual pump operation is modeled where operators throttle the RCIC turbine/pump to maintain RPV water level after automatic initiation.

In general, RCIC failure is assumed if the temperature of the suction source for the RCIC pump exceeds 383 K (230 F) since a portion of water is used as coolant for the pump. This can happen only when the suction is aligned to the suppression pool. RCIC can also fail upon loss of dc power (in SOARCA, this led to overfilling of the steamline and flooding the turbine). In addition, consistent with assumption in SECY-12-0157, RCIC is assumed to fail at 16 hours if the other conditions are not met. These assumptions can lead to failure of RCIC anytime between zero to 16 hours.

3.1.3.2 FLEX Injection

FLEX injection provides a constant 500 gpm with a water temperature of 300K assuming availability of on-site water supplies. Injection can be specified either to the vessel or the drywell via the core sprays or drywell sprays, respectively. FLEX injection occurs at the time of lower head failure for the all the scenarios considered here if water addition is to be probabilistically considered in the accident sequence analysis.

[2] For the Mark II analysis, RCIC initiates at 31 seconds, which is prior to L2 water level signal. There is no significant impact on the overall response of the plant.

Injection can be specified as either continuous injection, injection termination upon reaching the top of wetwell level instrumentation range, or controlled injection prior to exceeding the wetwell level instrumentation range. Injection is full rated flow during for the continuous injection and injection with termination option. Controlled option injects at full flow until the wetwell level reaches close to the upper instrumentation range, and the flow is decreased to maintain water in the pedestal region of the drywell (but sufficiently low to inhibit drainage from the drywell through the downcomers into the wetwell).

3.1.3.3 SRV Operations and Failure Modes

The SRV operations performed for the Mark I analyses include two pressure management stages (see Reference [10]). The first, occurring 10 minutes into the transient, models the operator performing managed SRV operations, operating available SRVs to manage wetwell heat-up, while maintaining RPV pressure between 800 and 1000 psig. After 1 hour, the second pressure management stage initiates as an operator depressurizes the reactor to 200 psig and maintains pressure between 200 psig and 400 psig[3]. The 200 psig limit maintains sufficient turbine pressure to continue RCIC operation during the transient. For the Mark II analysis, the pressure band for 10 minutes to 1 hour was set between 900 and 1000 psig. The SRV operation between 10 minutes and 1 hour does not significantly impact the Mark I and Mark II analyses.

The SRV failure models are unchanged from the values provided in NUREG/CR-7110 [11]. The present model assumes either a stochastic failure or high temperature failure. Valve failure is modeled as fully open by default for either criterion. User options were included to permit further investigations into the SRV failure states by allowing the user to specify the open fraction of the failed SRV as well as disabling either the stochastic and/or high temperature failure criteria.

3.1.3.4 Containment Failure Modes

Two containment pressure boundary failure modes modeled in the Mark I containment venting analysis include the drywell liner failure resulting from debris interaction and drywell head flange leakage due to containment over pressure or drywell head flange failure due to over temperature. The Mark II model only incorporates the Mark I drywell head flange failure. Liner melt through is only modeled for the Mark I model since it was not identified as a failure mode in the Mark II design.

The Mark I debris spreading model, discussed in Section 3.1.1.4 , calculates the spreading of material from the pedestal region into the drywell as well as the time of debris-liner contact. Given limitations in the cavity model, temperature response of the liner due to heat transfer between the liner and debris is not explicitly modeled. Without direct simulation of liner failure, an approximation of 15 minutes was assumed for the duration of the liner remaining intact after debris contact (SOARCA assumption). Once failed, a flow path (FL904) is opened between the lower drywell control volume (CV200) and adjacent reactor building compartment (CV401), but debris is not permitted to relocate into the reactor building.

Drywell head flange overpressurization leakage and high temperature failure are modeled with flow path (FL903). Overpressurization simulates head flange bolt strain in response to containment pressure. The head flange bolts are pretensioned, mating the flange surfaces while compressing a gasket seal. As pressure increases within containment, the pretensioning is overcome, the gasket seal decompresses over a distance of 0.03in, and continual strain opens

---

[3] The operators try to control the depressurization rate to 80ºF/hr. No controlled depressurization is performed in the present analysis for simplicity and is not believed to significantly impact the results.

the drywell head. The high temperature failure mode corresponds to the temperature (644K) at which ductility of the gasket material is lost. When drywell atmosphere temperature exceeds 644K, an assumed open area of 0.1 ft^2 is initiated [10].

3.1.3.5 Vent Operations

Hardened vent lines connecting the wetwell and drywell atmosphere spaces with the environment have been included in the models to investigate containment venting strategies. The wetwell vent line is modeled as a 16 in. line between the reactor building stack and wetwell using a discharge coefficient of 0.3 (see Reference [10]). The physical volume of the wetwell vent line is represented with CV223 and is connected to the wetwell volume with FL910 and to the environment with FL911. A simpler representation was applied for the drywell vent line, which is simply represented with a single flow path (FL920) using a 12 in. line (see Reference [10]).

Anticipatory venting (pre-core damage) is performed to cool the suppression pool and avoid RCIC failure. This action is to be performed prior to RCIC failure, but an option to permit anticipatory venting after RCIC failure has occurred is available. However, if the RPV water level reaches the minimum steam cooling (2/3 core height), the vent is closed (see Reference [10]).

Post core damage venting operation stage is performed to protect containment integrity and/or control releases. User options provide selection of either wetwell or drywell vent line operation when containment pressure exceeds primary containment pressure limit (prescribed as 60 psig) or at pressure suppression pressure (prescribed as 30 psig) as the actuation criterion (see Reference [10]). Additionally, vent cycling can be imposed with a pressure band of 10 or 20 psi [10] below the actuation pressure, i.e., 50-60 psig or 40-60 psig. The cycling option is only exercised for the PCPL venting cases. The Mark I PCPL of 60 psig was retained for the Mark II analysis, but additional sensitivities were performed with a corrected value of 45 psig (see Table 3-3).

If severe accident water addition (SAWA) is in effect, and wetwell venting is isolated upon high suppression water level (i.e., closing of the vent), then a switchover to drywell venting is required. However, the switchover to the drywell vent is not immediate. The pressure is allowed to return to PCPL or PSP before drywell venting actuates. Cycling can be enforced following switchover but only for PCPL venting cases.

If severe accident water management (SAWM) is in effect, and wetwell venting is the preferred path, then no switchover to drywell venting is required.

3.2 Scenario Development and Run Matrix

In developing the MELCOR calculation matrix for containment filtered venting system analysis, a set of accident prevention and mitigation measures were considered, informed by the lessons learned from the Fukushima event, accident management alternatives contemplated by the industry, the current state of knowledge of severe accident progression in a BWR and mitigation alternatives, and by the experience gained from the previous effort (SECY-12-0157) to address the Near Term Task Force (NTTF) Recommendation 5.1. The accident scenarios considered are those associated with an extended loss of alternating power (ELAP) event related to BWRs with Mark I and Mark II containments caused by an external hazard (e.g., beyond design basis earthquake), thereby resulting in one of three possible outcomes: containment overpressure or over-temperature failure, liner melt-through failure, or maintaining the containment largely intact (i.e., without any significant loss of its radioactivity confinement function) as a result of venting or other mitigation measures.

In an ELAP with the loss of all cooling function and absent any mitigation measures, the core is going to uncover leading to heatup, degradation, relocation of degraded core into the lower plenum, thermal loading of the reactor pressure vessel (RPV) lower head and consequent lower head failure, relocation of core debris into the reactor cavity, and ultimate containment failure by overpressure or other mechanisms. For this type of situation, the reactor core isolation cooling (RCIC) system is designed to provide core cooling, thus delaying core uncovery and subsequent accident progression until such time other DC-powered (battery or diesel generator) and portable mitigation systems become available. Hence, the operation of RCIC is considered as an important element in developing the MELCOR calculation matrix.

Containment venting is considered as another strategy to prevent catastrophic containment failure (by overpressure or otherwise) and consequent large release of radioactivity. SECY-12-0157 considered venting when the containment pressure exceeded the primary containment pressure limit (PCPL). Since then, the BWR industry introduced the concept of anticipatory early venting (pre-core damage venting at a pressure significantly below PCPL) in the process of updating its severe accident management guidelines (SAMGs). The idea behind the anticipatory early venting is to reduce the containment load pre-core damage, thereby affording more opportunity to put in place mitigation measures to address post-core damage containment performance. The early containment venting feature is included in the MELCOR matrix as are vent cycling and transition from wetwell venting to drywell venting.

By far the most important mitigation strategy, considered in the development of MELCOR matrix, relates to water addition, both in the reactor vessel and in the containment. Supplementing pre-core damage water addition, this mitigation strategy calls for post-core damage water addition to both RPV and containment (using FLEX [10]). Moreover, this strategy calls for water management (i.e., controlled water addition to achieve a specified purpose or goal, for example, ensuring the wetwell is not flooded).

In summary, the MELCOR run matrix presented in this section and the analysis presented in the next section are based on a number of assumptions which fall into two broad categories: (1) general assumptions and (2) specific assumptions. The general assumptions are as follows (see Reference [10]):

- All MELCOR transients start with an ELAP
- All transients are 72 hour in duration[4]
- Industry (BWROG) EPG/SAG Rev. 3 is in place
- FLEX is in place both pre- and post-core damage
- Possible end states of accident progression: liner melt-through (LMT), main steam line creep rupture (MSLCR), drywell head flange leakage by overpressure and overtemperature – all consequential to environmental release
- Recirculation pump leakage of 18 gpm per pump starts at the time of the initiating event

Specific MELCOR assumptions are focused on mitigation systems functions and include RCIC, RPV pressure control, containment venting, and water injection into either RPV or drywell. The assumptions are as follows:

[4] The assumptions regarding offsite support in this study are similar to those used in the SOARCA study (NUREG-1935) and the spent fuel pool study (NUREG-2161). The 72-hour time was chosen to perform a detailed assessment of the accident progression and to capture the release characteristics. In all cases where external water addition is successful, the radioactive releases have stabilized well before 72 hours.

- RCIC suction is generally from SP (though option for suction initially from CST, then transitioning to SP, is also considered as a sensitivity)
- Initial RCIC flow rate is 600 gpm and throttling of RCIC considered as an option for RPV level control
- RPV pressure control in 800 – 1000 psig band at 10 minutes into ELAP followed by controlled depressurization in one hour; subsequent pressure control in 200 – 400 psig band for continued RCIC operation
- Anticipatory early containment venting prior to core damage (generally at 15 psig containment pressure but a sensitivity case was run with 5 psig)
- Upon entry into SAG, vent closes; reopens at PCPL (60 psig nominally) with option to reopen at PSP also considered
- Transition from wetwell to drywell venting is at high SP water level (nominally 21 feet above the bottom of torus)
- The option of vent cycling is considered in (PCPL/PCPL-20) band and sensitivity cases were run with (PCPL/PCPL-10) band
- Water injection into drywell at vessel breach from an external source under severe accident conditions with 500 gpm flow rate; sensitivity of flow rate control (water management) considered for drywell injection to prevent wetwell flooding to avoid the need for switchover to drywell venting
- Initial buildup of water in the drywell from nominal leakage

The above assumptions were vetted with the industry stakeholders in several public meetings to assure that they are consistent with either the current or the planned SAMG practices. The industry has used largely the same or similar assumptions in their analysis using the MAAP code. It should be noted, however, that the industry is in the midst of updating their SAMG.

Selection of accident sequences covered by MELCOR calculation matrix was informed by staff's accident sequence analysis. The accident sequence analysis, discussed elsewhere in detail, consisted of core damage event tree (CDET) and accident progression event tree (APET) development and binning of a rather large number of possible end states to a manageable fewer categories of similar outcome, and ranking these categories in descending order of frequencies. The MELCOR calculation matrix covered all event categories comprising 98% of possible end states.

Another important consideration factored into the development of the MELCOR calculation matrix is tied to alternatives considered for regulatory analysis. Specifically, these alternatives fall into three categories discussed below:

Category 1—Overpressure Protection (Alternatives 1 and 2)*:* This category considers the severe accident capable vent, as called for in Order EA-13-109, is in place to provide over-pressure protection to the containment. Venting action, both pre- and post-core damage, is assumed, and available vent paths (wetwell and/or drywell) are considered. However, no further action (i.e., SAWA) is considered in this category. As such, liner melt-through in the Mark I containment and consequent uncontrolled release of radioactivity to the environment are not prevented.

Category 2—Liner Melt-through Protection (Alternative 3)*:* This category involves Category 1 over-pressure protection in addition to water injection (i.e., SAWA/SAWM) to either the RPV or the containment (drywell) or both for Mark I BWRs. This action has a high likelihood of preventing liner melt-through and slowing down the containment basemat erosion. This results in a smaller environmental release due, in large part, to the removal of fission products by the wetwell.

***Category 3—Enhanced Measures to Reduce Radioactivity Release (Alternative 4)**:* This category considers various enhanced measures to further reduce radioactivity release to the environment. This measure includes Category 2 water management (i.e., controlled water addition to ensure that the wetwell vent remains available (SAWM, vent cycling, etc.) and the use of engineered filters (small or large)).

Within Categories 2 and 3, several subcategories are considered delineating specific actions, individually and collectively. These actions include vent cycling, wetwell to drywell vent transition, etc. Accordingly, the alternatives (including sub-alternatives) considered in the construction of the MELCOR calculation matrix are listed in Table 3-1.

The MELCOR calculation matrix for a representative BWR with Mark I containment is shown in Table 3-2. The calculation matrix for a representative BWR with Mark II containment is shown in Table 3-3. The gray boxes in the tables signify the major deviations from the assumed initial and boundary conditions as part of sensitivity calculations.

The Mark I analyses in Table 3-2 include several sensitivities to investigate uncertainties in equipment availability and operator actions. The outcome of these sensitivities as well as the scenarios investigated were used to inform the analyses selected for the Mark II calculation matrix. In addition, the water management option was removed from consideration for the Mark II calculation matrix. Given the increased suppression chamber volume, wetwell flooding up to the vent line was perceived as less significant for the Mark II analyses. These considerations resulted in a significant reduction to the total number of analyses performed for the Mark II calculation matrix.

Table 3-1 Listing of the alternative options and actions

Alternative Number	Alternative Action
Alternative 1	Order EA-13-109, EPG/SAG Rev 3[5], anticipatory venting, RPV pressure control.
Alternative 2A	Make generically applicable Alternative 1 in rulemaking.
Alternative 3A	Alternative 2 plus wetwell venting (no vent cycling) and severe accident water addition (SAWA) into RPV.
Alternative 3B	Alternative 2 plus wetwell venting (no vent cycling) and SAWA into drywell.
Alternative 4Ai(1)	Alternative 3A with vent cycling.
Alternative 4Ai(2)	Alternative 3B with vent cycling.
Alternative 4Aii(1)	Alternative 2 plus wetwell venting (no vent cycling) and severe accident water management (SAWM) into RPV.
Alternative 4Aii(2)	Alternative 2 plus wetwell venting (no vent cycling) and SAWM into drywell.
Alternative 4Aiii(1)	Alternative 4Aii(1) with vent cycling.
Alternative 4Aiii(2)	Alternative 4Aii(2) with vent cycling.
Alternative 4Bi(1)	Alternative 4Ai(1) with an external filter (Decontamination factor [DF]=10).
Alternative 4Bi(2)	Alternative 4Ai(2) with an external filter (DF=10).
Alternative 4Bii	Alternative 4Bi(2) with both pre- and post-core damage manual venting through drywell and an external filter (DF=10).
Alternative 4Biii	Alternative 4Bi(2) with pre-core damage manual venting and post-core damage passive venting through drywell and an external filter (DF=10).
Alternative 4Biv	Alternative 4Bi(2) with both pre- and post-core damage passive venting through drywell and an external filter (DF=10).
Alternative 4Ci(1)	Alternative 4Ai(1) with an external filter (DF=1000).
Alternative 4Ci(2)	Alternative 4Ai(2) with an external filter (DF=1000).
Alternative 4Cii	Alternative 4Bi(2) with both pre- and post-core damage manual venting through drywell and an external filter (DF=1000).
Alternative 4Ciii	Alternative 4Bi(2) with pre-core damage manual venting and post-core damage passive venting through drywell and an external filter (DF=1000).
Alternative 4Civ	Alternative 4Bi(2) with both pre- and post-core damage passive venting through drywell and an external filter (DF=1000).

The MELCOR results, presented in the next section, can be classified into two broad categories: (1) thermal-hydraulic output; and (2) source term output. The thermal-hydraulic output includes the following:

- RPV pressure, temperature, and water level – determine the likelihood of main steam line creep rupture failure

[5] Elements of BWR Owners' Group (BWROG) Emergency Procedure guidelines (EPG) and Severe Accident Guidelines (SAG), Revision 3, have been communicated to NRC by Nuclear Energy Institute (NEI) and BWROG.

- Containment (drywell) pressure and temperature – determine the likelihood of failure of containment and various components (e.g., drywell head flange) by overpressure and/or overtemperature
- Hydrogen and other non-condensable gas generation and migration – contribute to containment overpressurization and hence, the timing of vent operation; also, determine the potential for combustion in reactor building, vent line, etc.

- Wetwell (suppression pool) temperature and water level – determine the effectiveness of pool scrubbing and the likelihood of wetwell vent becoming unavailable

The source term output provides input to consequence analysis using MACCS and includes, among others, the following quantities:

- Cesium release – an important contributor to latent cancer fatality risk and land contamination
- Iodine release – an important contributor to early fatality risk

Table 3-2 Calculation matrix for a representative BWR with Mark I containment

		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	WW Level Control Injection @ 21' (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	1S1	72	16	SP	230	N	5	-	-	Y	WW	PCPL
1/2A	2	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	PCPL
1/2A	4	72	16	SP	240	N	15	-	-	Y	WW	PCPL
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	PCPL
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	PSP
3A	7	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
	7dw	72	16	SP	230	N	15	RPV	0	Y	DW	PCPL
3A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	PCPL
4Aii(1)	8	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	9	72	16	SP	230	Y	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	12	72	16	SP	230	N	15	RPV	throttle	Y	WW	PSP
4Aii(1)	13	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	14	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
4Aiii(1)	15	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	18	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
4Ai(1)	16	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3B	21	72	16	SP	230	N	15	DW	0	Y	WW	PCPL
3B	24	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
	24dw	72	16	SP	230	N	15	DW	500	Y	DW	PCPL
4Aii(2)	22	72	16	SP	230	N	15	DW	throttle	Y (50%)	WW	PCPL
	22dw	72	16	SP	230	N	15	DW	throttle	Y	DW	PCPL
4Aii(2)	23	72	16	SP	230	N	15	DW	throttle	Y	WW	PCPL
4Aii(2)	25	72	16	SP	230	Y	15	DW	throttle	Y	WW	PCPL
3B	26	72	16	SP	230	Y	15	DW	500	Y	WW	PCPL
4Ai(2)	27	72	16	SP	230	N	15	DW	0	Y	WW	PCPL
4Aiii(2)	28	72	16	SP	230	N	15	DW	throttle	Y	WW	PCPL
	28dw	72	16	SP	230	N	15	DW	throttle	Y	DW	PCPL
4Ai(2)	32	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
4Ai(2)	30	72	16	SP	230	N	15	DW	500	Y	WW/DW	PCPL
	30dw	72	16	SP	230	N	15	DW	500	Y	DW	PCPL
4Aiii(2)	29	72	16	SP	230	Y	15	DW	throttle	Y	WW	PCPL
	29dw	72	16	SP	230	Y	15	DW	throttle	Y	DW	PCPL
4Ai(2)	31	72	16	SP	230	Y	15	DW	500	Y	WW/DW	PCPL
	31dw	72	16	SP	230	Y	15	DW	500	Y	DW	PCPL
3A	41	4	4	SP	230	N	15	RPV	0	N	WW	PCPL
3B	43	4	4	SP	230	N	15	DW	0	N	WW	PCPL
3A	42	4	4	SP	230	N	15	RPV	500	N	WW/DW	PCPL
3B	44	4	4	SP	230	N	15	DW	500	N	WW/DW	PCPL
4Aii(1)	47	4	4	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(2)	48	4	4	SP	230	N	15	DW	throttle	Y	WW	PCPL
3B	45	-	16	SP	230	-	-	DW	500	Y	DW	PCPL
3B	46	-	16	SP	230	-	-	DW	500	Y	WW/DW	PCPL
3B	49	-	0	-	-	-	-	DW	500	Y	WW/DW	PCPL
4Ai(2)	50	-	0	-	-	-	-	DW	500	Y	WW/DW	PCPL
3B	51	-	16	SP	230	-	15	DW	500	Y	DW	15
3B	52	-	16	SP	230	-	15	DW	500	N	DW	15
3B	53	-	16	SP	230	-	15	DW	500	Y	DW	15

Table 3-3 Calculation matrix for a representative BWR with Mark II containment

		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	SAWA Injection rate (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1	1	72	16	SP	230	N	15	-	-	Y	WW	60
	1p1	72	16	SP	230	N	15	-	-	Y	WW	45
1	3	4	4	SP	230	N	15	-	-	N	WW	60
1	5	72	16	CST	230	N	15	-	-	Y	WW	60
1	6	72	16	SP	230	N	15	-	-	Y	WW	30
2A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	60
	10p1	72	16	SP	230	N	15	RPV	500	Y	WW/DW	45
2A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	60
	11p1	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	45
3A	24	72	16	SP	230	N	15	DW	500	Y	WW/DW	60
	24p1	72	16	SP	230	N	15	DW	500	Y	WW/DW	45
2A	42	4	4	SP	230	N	15	RPV	500	N	WW/DW	60
3A	44	4	4	SP	230	N	15	DW	500	N	WW/DW	60
3A	45	-	16	SP	230	-	-	DW	500	Y	DW	60
3A	49	-	0	-	-	-	-	DW	500	Y	WW/DW	60
3A	51	-	16	SP	230	-	15	DW	500	Y	DW	15
3A	52	-	16	SP	230	-	15	DW	500	N	DW	15
		Sensitivity to DW-WW bypass and lower reactor cavity (LRC) pool										
1	1a1	Same as case 1 but assuming suppression pool bypass is delayed until upper reactor cavity (URC) floor is completely ablated										
1	1b1	Same as case 1 but assuming lower reactor cavity is filled with water										
1	1b2	Same as case 1b1 but assuming immediate suppression pool bypass (0 min delay)										
1	1b3	Same as case 1b1 but assuming suppression pool bypass is delayed until URC floor is completely ablated										

3.3 Accident Progression Results

This section provides a detailed analysis of the MELCOR calculations for the Mark I and Mark II representative models. The initial and boundary conditions for the analysis and the accident scenarios are provided in Sections 3.1.3 and 3.2 .

3.3.1 BWR Mark I Results

It is important to understand the basic response of the plant. Three cases were selected to examine accident progression under severe accident conditions for cases with and without water addition or management. These cases are described in detail to represent the phenonmena pertinent to the run matrix. Cases 9 and 10 from the run matrix (see Table 3-2) are selected as representative runs for the water management and water addition (Section 3.3.1.1). Case 1 is without water addition and is discussed in Section 3.3.1.2 . A timing of key events is shown in Table 3-4.

Table 3-4 Timing of key events for selected Mark I cases

Event Timing (hr)	Case 1 (no water)	Case 9 (SAWM)	Case 10 (SAWA)
Start of ELAP	0.0	0.0	0.0
Operators first open SRV to control pressure	0.17	0.17	0.17
Low-level 2 and RCIC actuation signal	0.18	0.18	0.18
Operators open SRV to control pressure (200-400 psig)	1.0	1.0	1.0
RCIC flow terminates	9.6	9.6	9.6
SRV sticks open or operators open SRV after RCIC fails	16.0	9.6	16.0
Downcomer water level reaches TAF	12.0	11.6	12.0
First hydrogen production	13.7	13.2	13.7
First fuel-cladding gap release	13.7	13.2	13.7
Start of containment venting at 60 psig	14.9	14.4	16.3
Relocation of core debris to lower plenum	15.6	15.5	15.5
RPV lower head dries out	18.1	18.2	18.9
RPV lower head fails grossly	23.0	23.4	23.1
Drywell head flange leakage	27.1	-	-
Hydrogen burn in reactor building refueling bay	28.8	-	-
Drywell liner melt-through	31.4	-	-
Calculation terminated	72	72	72
Selected MELCOR Results	**Case 1**	**Case 9**	**Case 10**
Debris mass ejected (1000 kg)	292	280	287
In-vessel hydrogen generated (kg)	1195	1032	1232
Iodine release fraction at 72 hr	2.28E-01	7.86E-02	8.10E-02
Cesium release fraction at 72 hr	1.94E-02	6.12E-03	7.26E-03

3.3.1.1 Accident Progression with Water Management/Addition

Figure 3-10 shows the RPV pressure for Case 9. During the first 10 minutes, there is no pressure control and a single SRV keeps the pressure at its lowest pressure setpoint. The hydraulic transient immediately following reactor scram and isolation results in a gradual decrease in water level because of coolant evaporation and discharge through the cycling SRV to the suppression pool as shown in Figure 3-11. At 10 minutes, the operators maintain the pressure between 800-1000 psig by opening one or more SRVs. At about the same time, the RCIC starts automatically when the water level in the RPV reaches the low level L2.

After RCIC initiation, the RPV water level is gradually restored by injecting water from the suppression pool at 600 gpm. At one hour, the operators begin controlled depressurization of the RPV and maintain the pressure between 200-400 psig (the lower pressure of 200 psig would allow enough steam for the continued operation of the RCIC turbine). The operators take manual control of the RCIC to maintain the level by throttling the water injection. The cycling of the RPV pressure after this time causes changes in the effective level of water because of variations in the average coolant density.

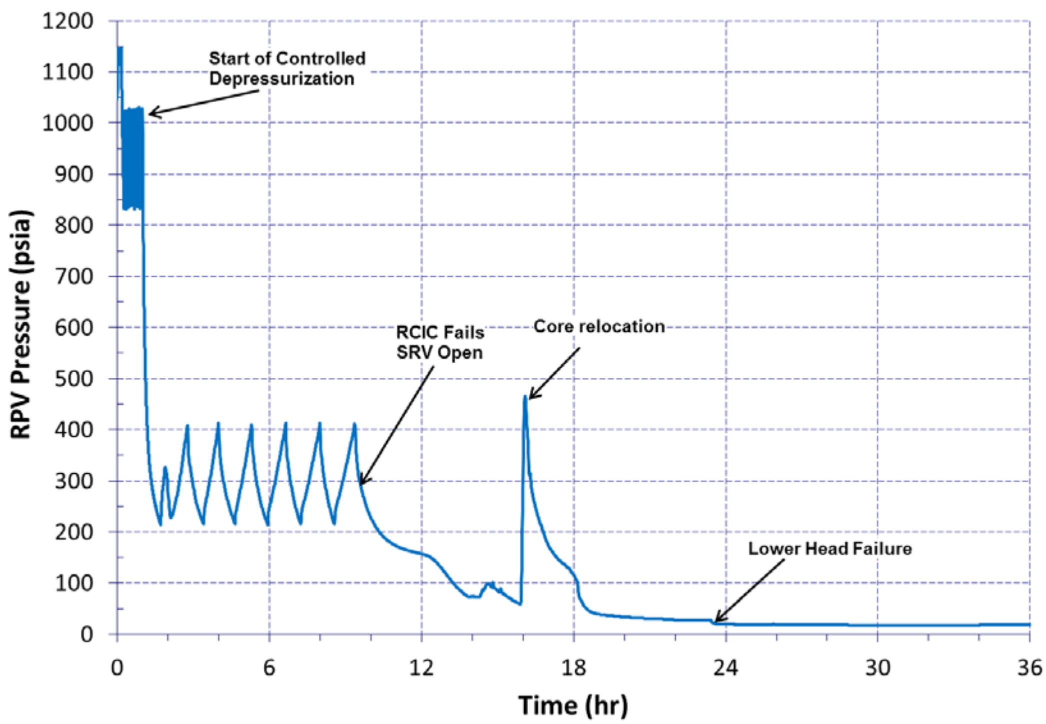


Figure 3-10 Mark I RPV pressure for Case 9 (SAWM)

RCIC fails at 9.6 hours due to over-temperature in the suppression pool (230°F). For case 9, the operators depressurize the RPV by opening the SRV. Between the start of initial depressurization at 1 hour and RCIC failure, there are six openings of the SRVs. The operators cycle different SRVs during this time to provide a more uniform heatup of the suppression pool. Without water injection from RCIC, the RPV water level decreases gradually and reaches the top of active fuel by 11.6 hours. It takes more than 2 hours for the water to reach the bottom of the active fuel and an additional 2 hours for the core to relocate to the lower plenum.

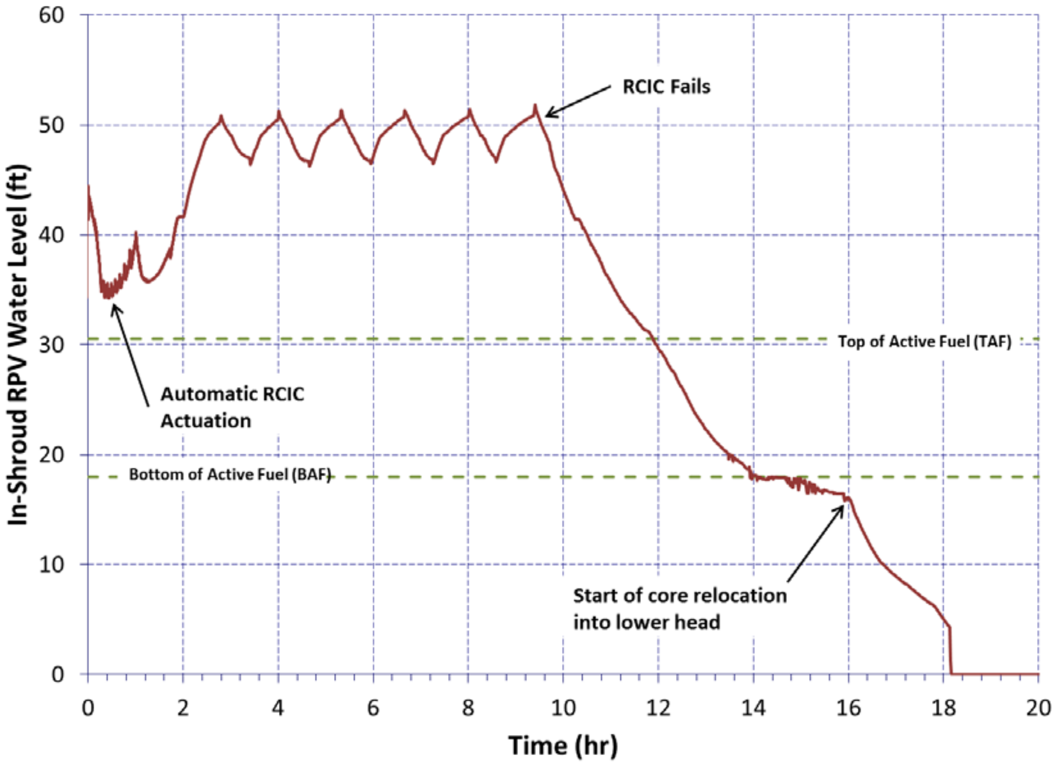


Figure 3-11 Mark I RPV water level for Case 9 (SAWM)

The thermal response of fuel in the core is illustrated in Figure 3-12 and Figure 3-13, which show the calculated temperature of fuel cladding in the inner core ring and across the core mid-plane. Temperatures of fuel cladding at the top of the core begin to rise when the mixture level decreases below the top of active fuel. As the mixture level decreases toward the bottom of the core, fuel temperatures increase rapidly due to runaway oxidation of Zircaloy cladding. The close relationship between the rate at which fuel cladding temperature increase and Zircaloy oxidation is shown in Figure 3-14, which compares clad temperatures (left-hand scale) to total in-vessel hydrogen generation (right-hand scale).

Mechanical failure of fuel at the top of the core occurs when Zircaloy clad material either melts and drains to lower regions of the core, or oxides to form a thin, fragile ZrO_2 shell around over-heated fuel. This mechanically weak material fragments into particulate debris, which relocates toward the lower core plate as rubble. Particulate and molten debris continue to move downward in the core until 16 hours, when the lower core plate yields, releasing molten core debris into the reactor vessel lower head. The interaction between hot debris and residual water in the lower head increases the rate of coolant evaporation, as indicated in Figure 3-11. It also causes the molten debris to freeze on surfaces of the control rod guide tubes, which are submerged in the large body of water that remains in the lower plenum. The changes in core geometry during this time frame, which are caused by the formation and downward relocation of molten and particulate debris, are illustrated in Figure 3-14.

The core debris cools as it enters the water-filled lower plenum. When residual water in the lower plenum is completely evaporated at 18 hours, debris temperatures begin to increase. Heat transfer from debris to the inner surface of the lower head causes the lower head temperature to

increase as well. Because reactor vessel pressure is relatively low during the heat up of debris in the lower plenum, the failure of the lower head is more strongly influenced by thermal rather than mechanical stresses.

Figure 3-14 also illustrates changes in the configuration of core debris and lower plenum structures between the time of RPV dryout (18 hrs) and lower head failure (23.4 hours). At the time of lower head dryout, most of the core fuel assemblies (i.e., the central four of five radial rings in the MELCOR model) have collapsed into the lower plenum. Highly oxidized, but vertically intact assemblies remain standing in the outer ring of the core. Debris in the lower plenum surrounds a forest of intact control rod guide tubes. As the temperature of lower plenum debris increases, it causes the structural failure of the control rod guide tubes, which collapse and are mixed into the growing debris bed. Immediately prior to lower head failure at 23.4 hours, the debris bed represents the mass of most of the core plus structural materials below the core. Failure of the lower head results in the rapid ejection of 280 metric tons of core debris onto the floor of the reactor pedestal in the drywell. By the time of the lower head failure, the outer ring is still intact and the injection of water into the RPV causes the temperature to decrease.

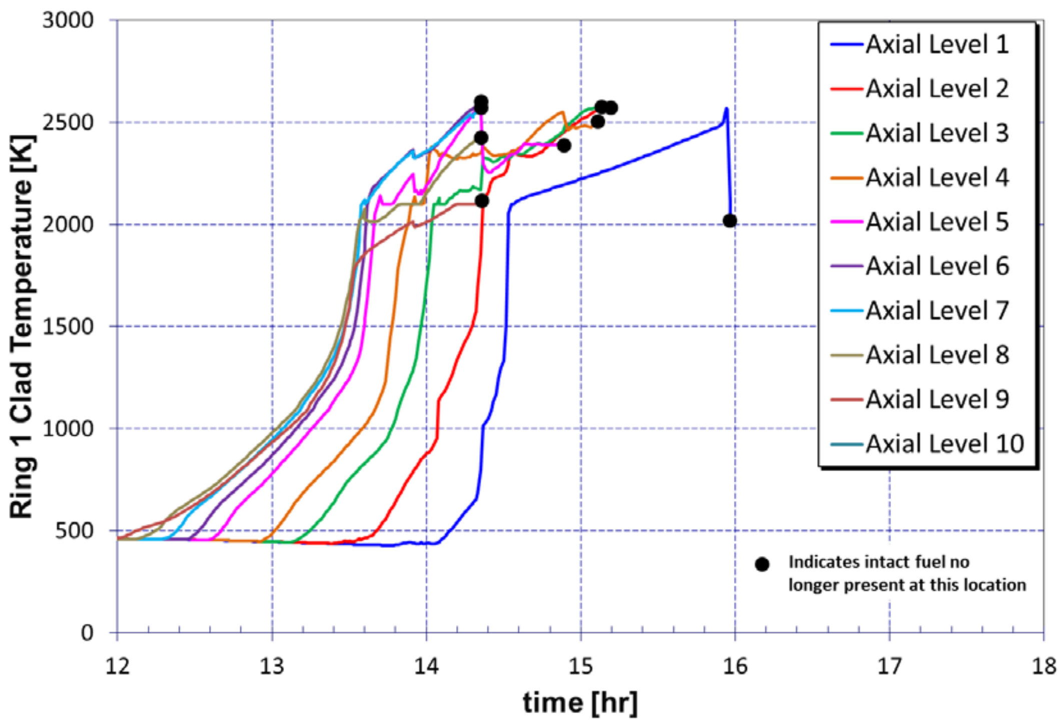


Figure 3-12 Mark I fuel cladding temperature in Ring 1 for Case 9 (SAWM)

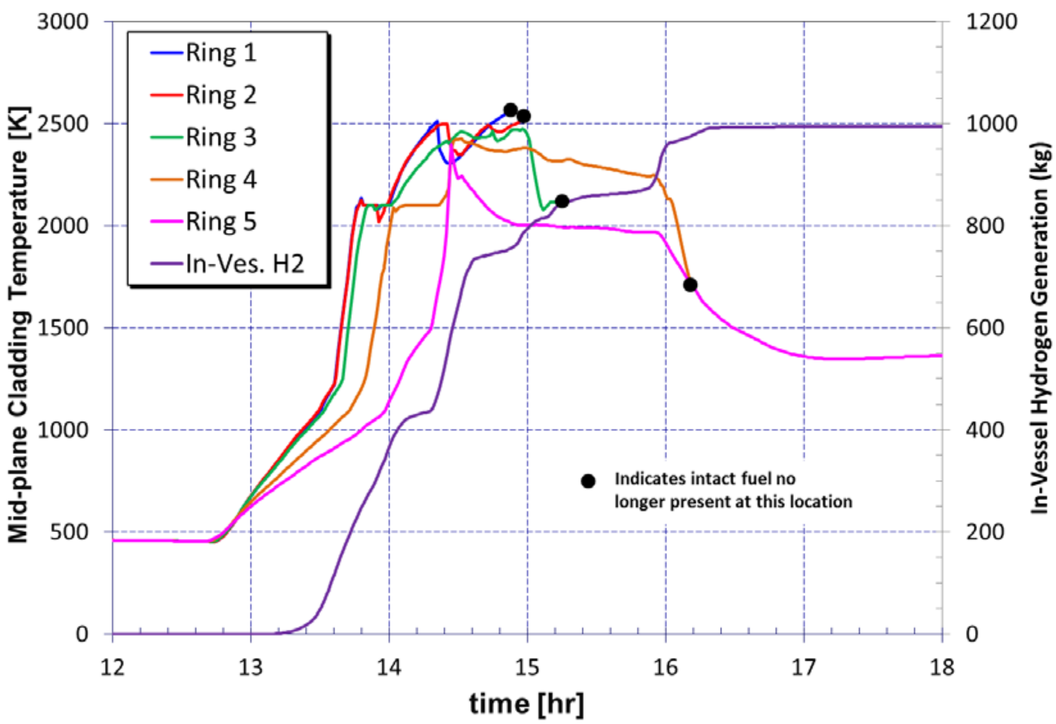


Figure 3-13 Mark I fuel cladding temperatures at core mid-plane for Case 9 (SAWM)

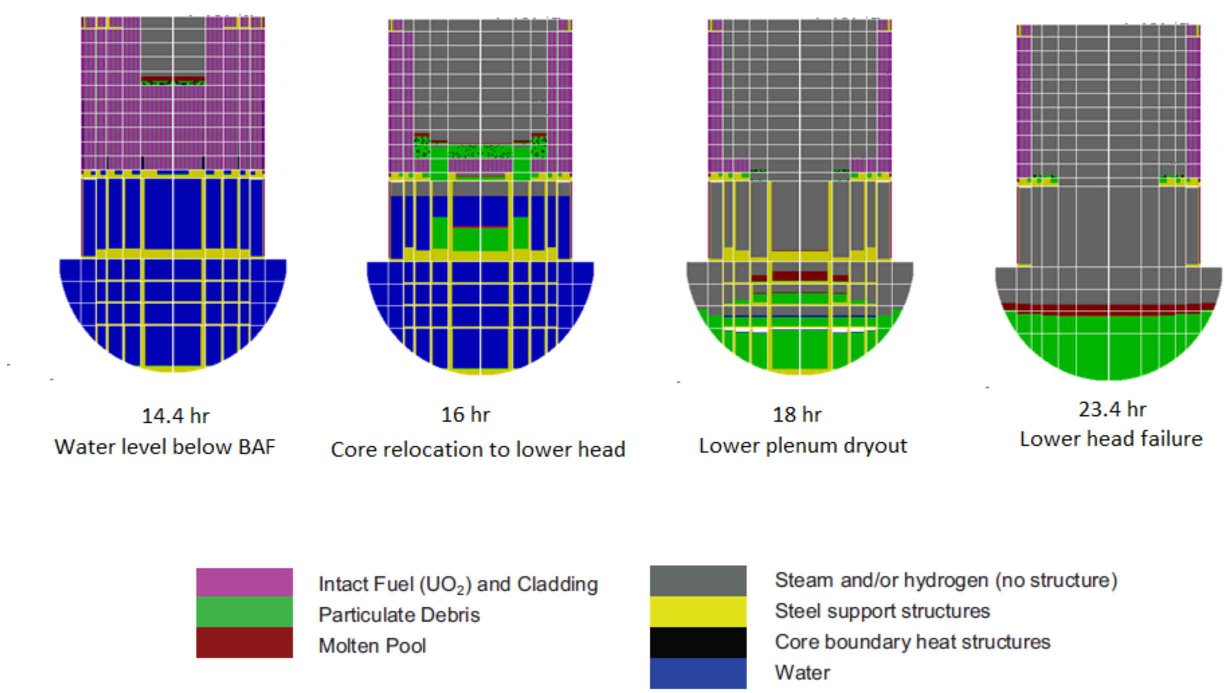


Figure 3-14 Mark I core degradation and relocation for Case 9 (SAWM)

During the pre-core damage phase of the accident, the thermal response of the containment is governed by the operation of RCIC and SRVs. The RCIC injection of suppression pool water into the RPV and the removal of decay heat through the SRVs back to the suppression pool caused the gradual heat up of the water as shown in Figure 3-15. The step changes in the pool temperature coincide with the opening of the SRVs (see Figure 3-10). The pool temperature reaches 230°F at 9.6 hours causing the failure of RCIC due to high water temperature. The saturated pool continues to heat up for about an additional 5 hours as a result of opening of the SRVs until the water level in the RPV reaches the bottom of the active fuel. There is a gradual decrease in the suppression pool temperature following the opening of the wetwell vent at 14.3 hours. The suppression pool remains saturated for the remainder of the accident, as the wetwell vent is kept open to stabilize the pressure in the containment.

The changes in the containment pressure and the total integral vent flow are shown in Figure 3-16. By the time RCIC fails, the containment pressure has reached 12.5 psig, which is below the threshold of 15 psig for anticipatory venting. The containment pressure starts to rapidly increase due to release of hydrogen through the SRV to the torus (i.e., over 500 kg before venting). At 14.3 hours after the initiating event, the containment pressure increases above the venting pressure of 60 psig, and the wetwell vent is opened. The rapid rise in containment pressure post core damage requires a timely response to vent the containment.

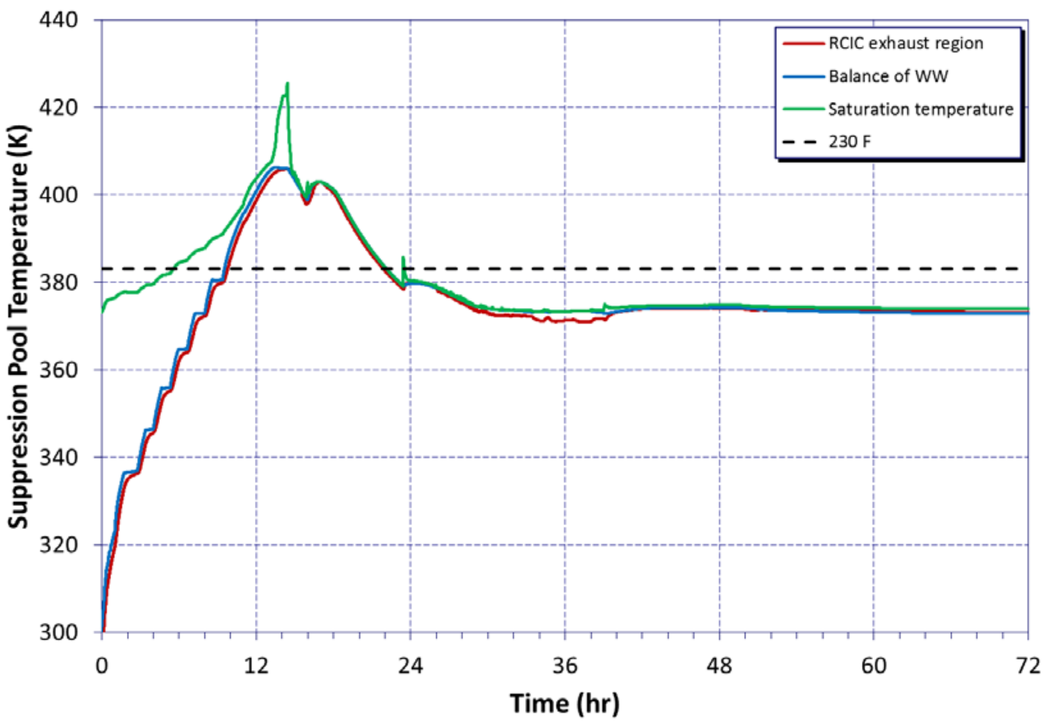


Figure 3-15 Mark I suppression pool temperature for Case 9 (SAWM)

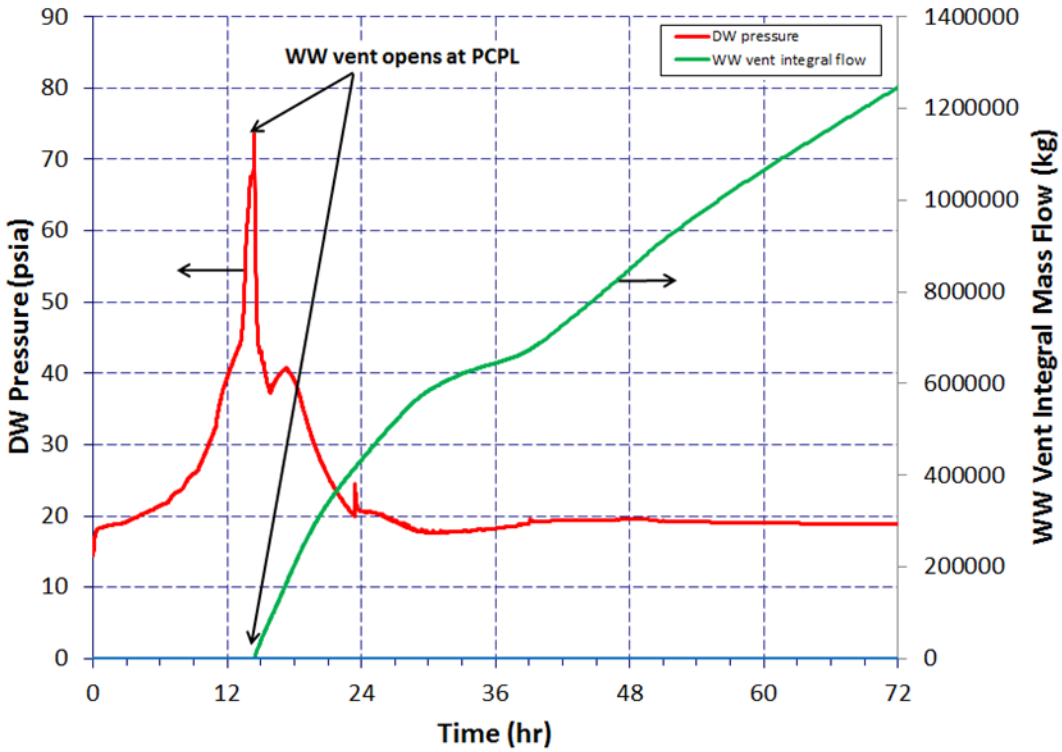


Figure 3-16 Mark I containment pressure and vent flow for Case 9 (SAWM)

Figure 3-17 shows the containment water level. By the time of the lower head failure, there is an accumulation of 1.4 ft of water above the drywell floor as a result of leakage from the recirculation lines. After venting the containment, the water level in the suppression pool continues to decrease as coolant is discharged from the containment. Following lower head failure, water injection into the RPV begins as shown in Figure 3-18. The initial flow rate is 500 gpm, which continues for the next 15.6 hours when the water level in the torus approaches 21 ft. At this time, there is a gap of about 2 hours[6] with no water injection as the operators try to throttle the water and maintain the level below 21 ft.

Containment atmosphere temperatures (see Figure 3-19) remain relatively low even after core damage because the steam/hydrogen mixture cools as it bubbles through the suppression pool. Immediately following vessel breach, containment atmosphere temperatures increase due to the accumulation of core debris on the reactor pedestal and drywell floors. However, water injection maintains the drywell temperature below 500 K in the long term.

Soon after debris is released onto the reactor pedestal floor, it flows laterally out of the cavity through the personnel access doorway and spreads out across the main drywell floor. Lateral movement and spreading of debris across the drywell floor shown in Figure 3-20 indicates that the debris does not reach the steel shell at the outer perimeter of the drywell. A combination of water injection and containment venting prevents containment failure.

[6] This is a modeling assumption to control the water flow rate.

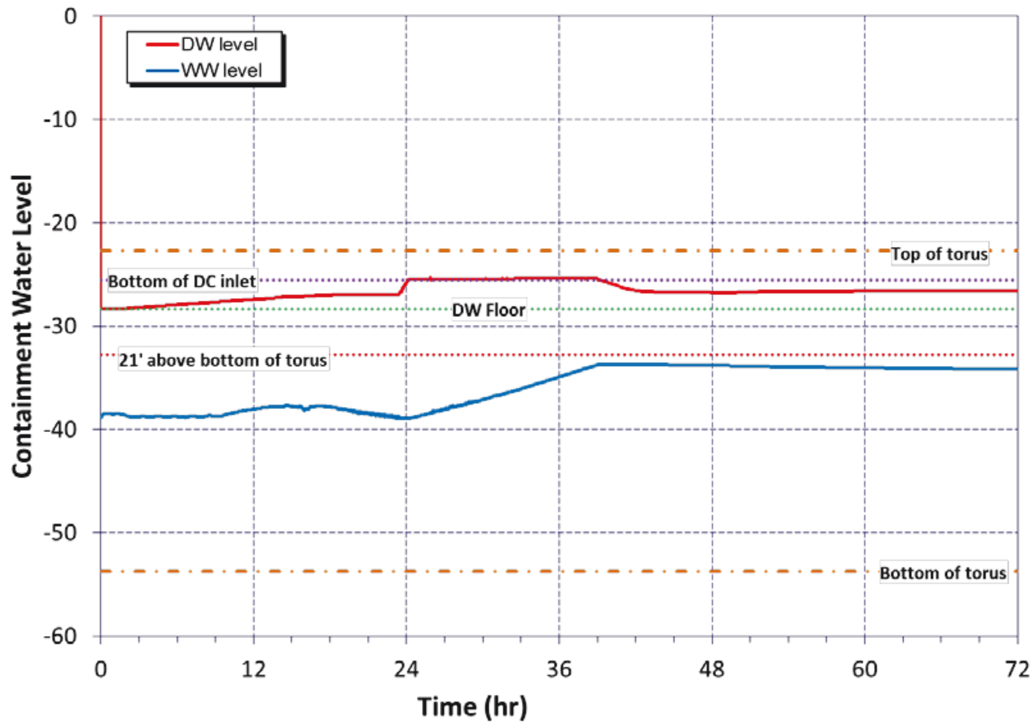


Figure 3-17 Mark I containment water level for Case 9 (SAWM)

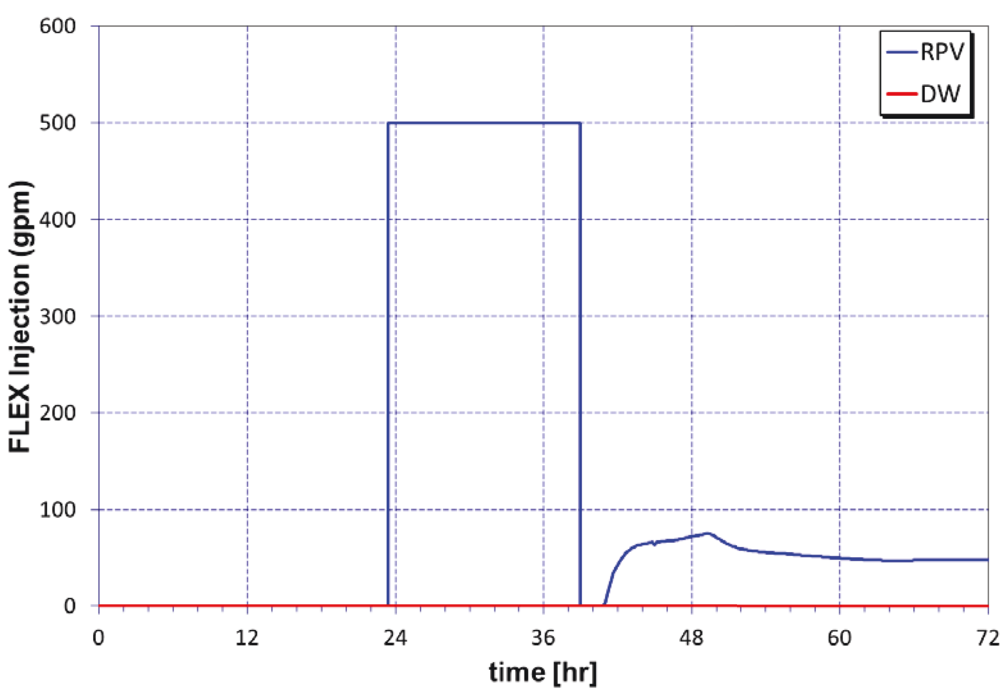


Figure 3-18 Mark I RPV water injection rate for Case 9 (SAWM)

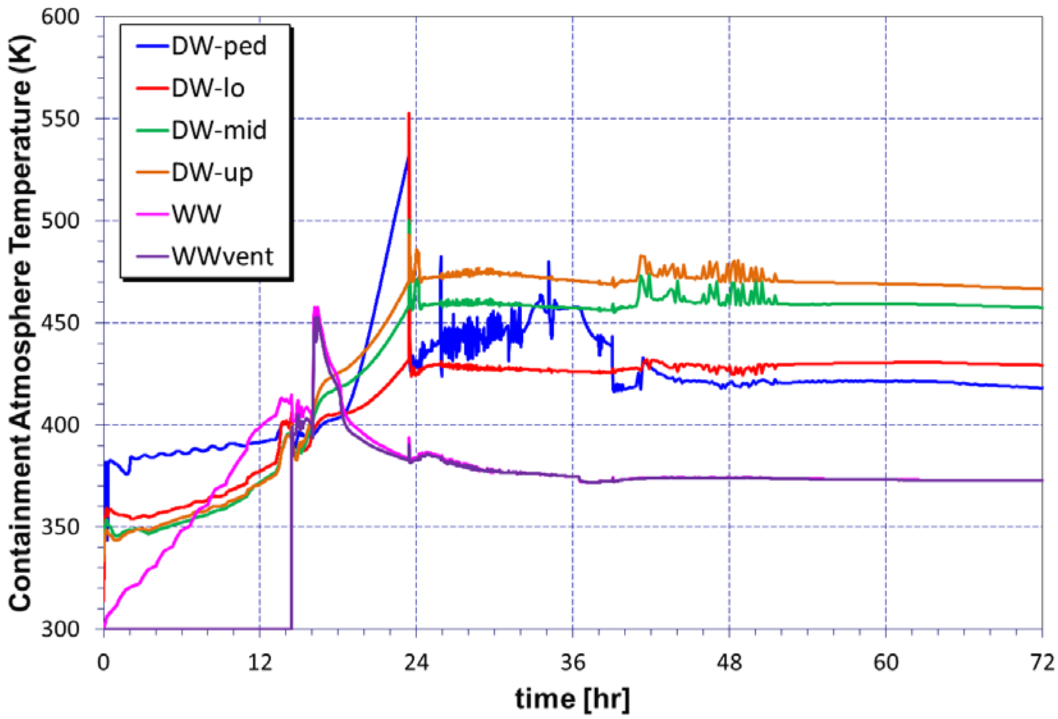


Figure 3-19 Mark I containment atmosphere temperature for Case 9 (SAWM)

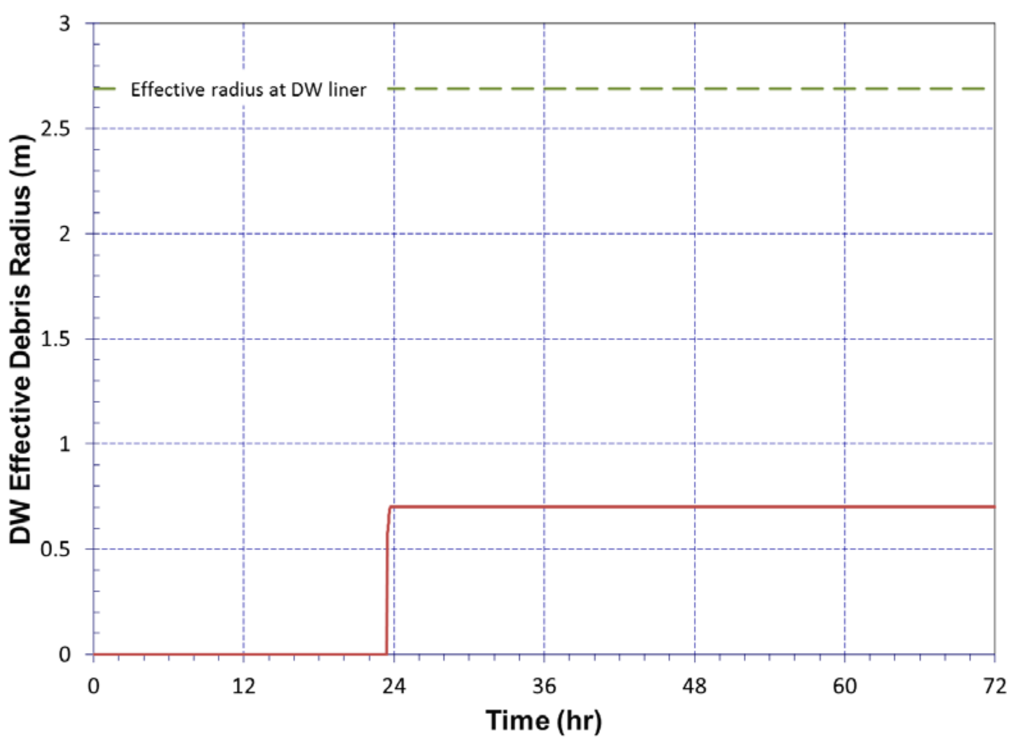


Figure 3-20 Mark I debris spreading in DW for Case 9 (SAWM)

Volatile fission products release from fuel begins at 13.2 hours, when a portion of the fuel gap inventory is released due to early fuel cladding failures. As fuel temperatures rise (see Figure 3-12, diffusion-driven release of fission products out of the fuel matrix rapidly increases the amount of volatile species released into the reactor coolant system. The cumulative release of volatile species such as cesium and iodine from the fuel and the distribution in the RPV and the containment are shown in Figure 3-21 and Figure 3-22.

The spatial distribution of cesium does not significantly change with time in the same way the distribution changed for iodine. This is most evident by the behavior inside the RPV. Even though the fraction of the initial inventory released from fuel that is deposited initially (e.g., on the steam separators and dryers) is consistent, a larger proportion of deposited cesium is retained inside the vessel. These differences in iodine and cesium behavior can be attributed to differences in the physical properties of their dominant chemical forms. Iodine is transported as CsI. The cesium contribution to CsI represents only 6% of the total cesium inventory. The vast majority of the cesium inventory is transported in the form of cesium molybdate (Cs_2MoO_4). Cesium molybdate is less volatile than the iodide and remains deposited on in-vessel structures. However, iodine is preferentially transported to the torus, but cesium remains deposited on in-vessel structures. This is clear from Figure 3-22 that shows between the time of containment venting at 14.3 hours and vessel breach, most of the iodine inventory in the RPV is purged to the suppression pool through the open SRV (see Figure 3-10).

It is important to note that the release of radionuclides that follows containment venting at 14.3 hours is characterized by a single sudden release about 9 hours before vessel breach and start of water injection. The injection of water after lower head failure stabilizes the environmental release.

The particle size distribution of CsI released through the wetwell vent (see Figure 3-23) clearly shows that the majorty of aerosols are in the submicron size range (~99%) with almost 80% in the lower bin size of 0.15 micron. The insert in the figure is the mass averaged particle size. The transport and scrubbing of the fission product aerosols and vapors through the suppression pool in the short time between the time of core damage at 13.2 hours and containment venting at 14.4 hours (see Table 3-4) results in small particle sizes that make it through the wetwell vent in the initial sudden release. Any scrubbing of aerosols in this size range is expected to be minimal.

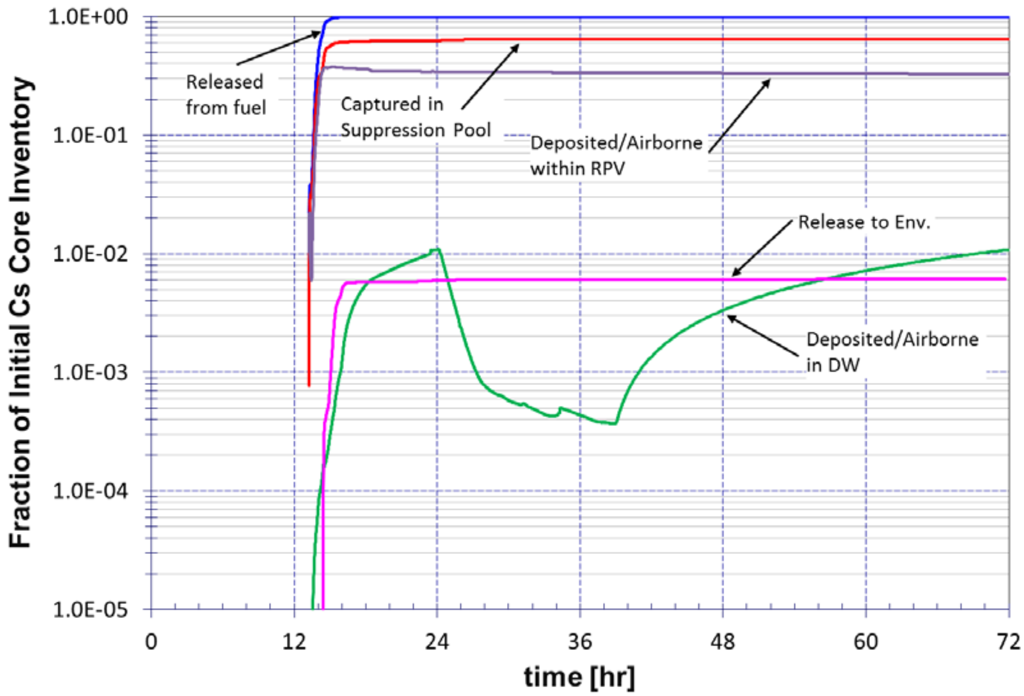


Figure 3-21 Mark I cesium fission product distribution for Case 9 (SAWM)

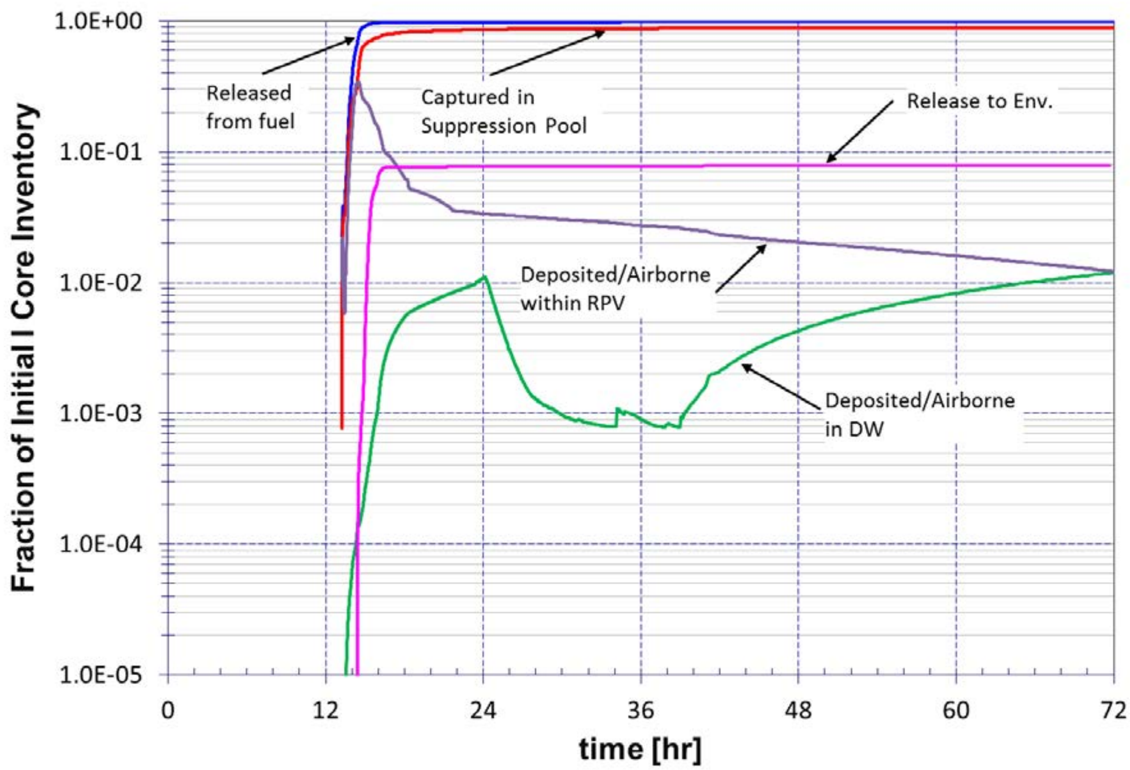


Figure 3-22 Mark I iodine fission product distribution for Case 9 (SAWM)

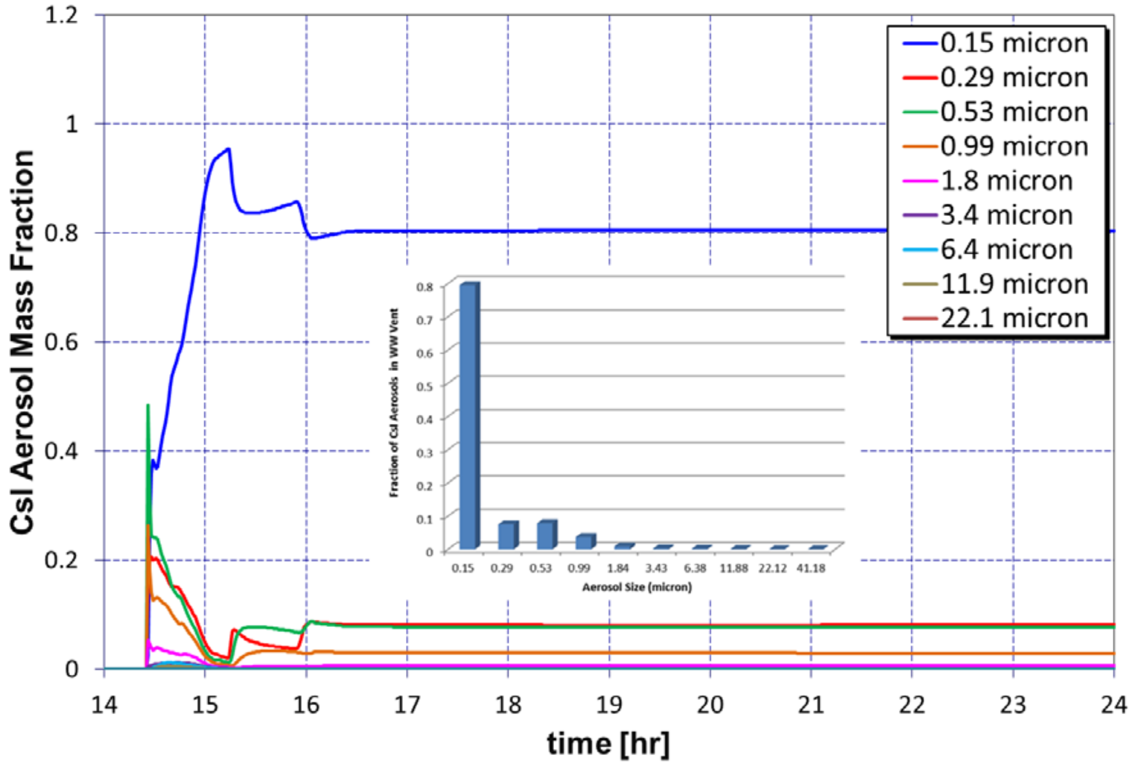


Figure 3-23 Mark I CsI particle size distribution for Case 9 (SAWM)

Since the release to the environment is dominated by containment venting before vessel breach and vessel injection, the mode of water control (i.e., water addition vs. water management) is not expected to significantly change the source term. Case 10 is a representative run to see the effects of continuous water injection that requires a switchover to drywell venting. In this case, it is assumed that once the water level in the torus reaches 21 ft, the wetwell vent is isolated and a switchover to drywell venting is initiated when the containment pressure once again reaches 60 psig. Two other changes in this scenario compared to Case 9 include the assumption that there is no depressurization of the RPV following failure of the RCIC and anticipatory venting is allowed even after RCIC failure.[7]

Figure 3-24 shows the RPV pressure for Case 10. As mentioned before, the RPV is not depressurized after RCIC failure but continues to cycle between 200-400 psig for an additional 6 hours until the SRV is stuck in an open position. The higher frequency of SRV cycles is a direct result of core degradation during this time. Even though the SRV fails to open due to thermal failure, there is a period of vessel pressurization following core debris relocation to the lower head.

The containment pressure response is shown in Figure 3-25. There is only an hour difference between the failure of RCIC at 9.6 hours and the time that the pressure inside the containment reaches the threshold of 15 psig for early venting. Here it is assumed that anticipatory venting

[7] These assumptions were made to see the response of the RPV to elevated pressure that could lead to possible main steam line creep rupture following core degradation and the timing of the stuck open SRV due to thermal seizure.

occurs and the wetwell vent is opened. Once the water level inside the RPV reaches the minimum steam cooling at 12.8 hours, the wetwell vent is isolated. The containment starts to pressurize as a result of core degradation as shown before (see Figure 3-16 for Case 9). The early venting in this case results in a time delay for post core damage containment venting at 60 psig as compared to Case 9 (at 16.3 hours or about 2 hours later). With water injection at lower head failure (23.8 hours), it takes 18.4 hours to increase the torus water level to 21 ft. The containment is once again isolated and the pressure continues to increase due to decay heat addition and non-condensable gas generation as a result of molten core concrete interaction. The containment pressure reaches 60 psig at 54.3 hours or more than two days after the initiating event.

The sudden containment depressurization from the drywell causes a back flow of water from downcomer vents onto the drywell floor as shown in Figure 3-26. The vacuum breaker model takes into account the hydrostatic head of water inside the vent lines as water injection leads to an increase in the downcomer level. This back flow of water from the suppression pool and water injection into the RPV leads to a buildup of water of more than 15 ft above the bottom of containment vessel by the end of the calculation at 72 hours.

The fractional distribution of cesium and iodine are shown in Figure 3-27 and Figure 3-28. The total release of cesium (7.3E-3 for Case 10 and 6.1E-3 for Case 9) and iodine (8.1E-2 for Case 10 and 7.9E-2 for Case 9) are comparable between the water addition case (case 10) and water management case (case 9) even though the fractional distributions inside the RPV and containment are initially somewhat different due to differences in boundary conditions and core degradation process.

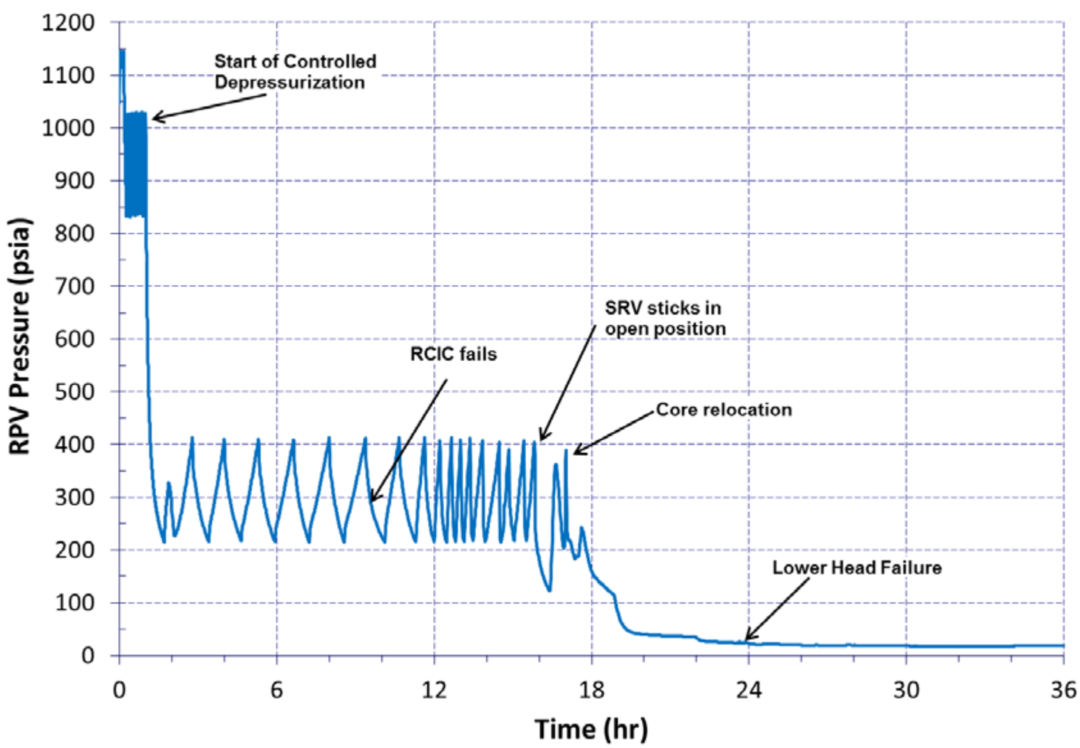


Figure 3-24 Mark I RPV pressure for Case 10 (SAWA)

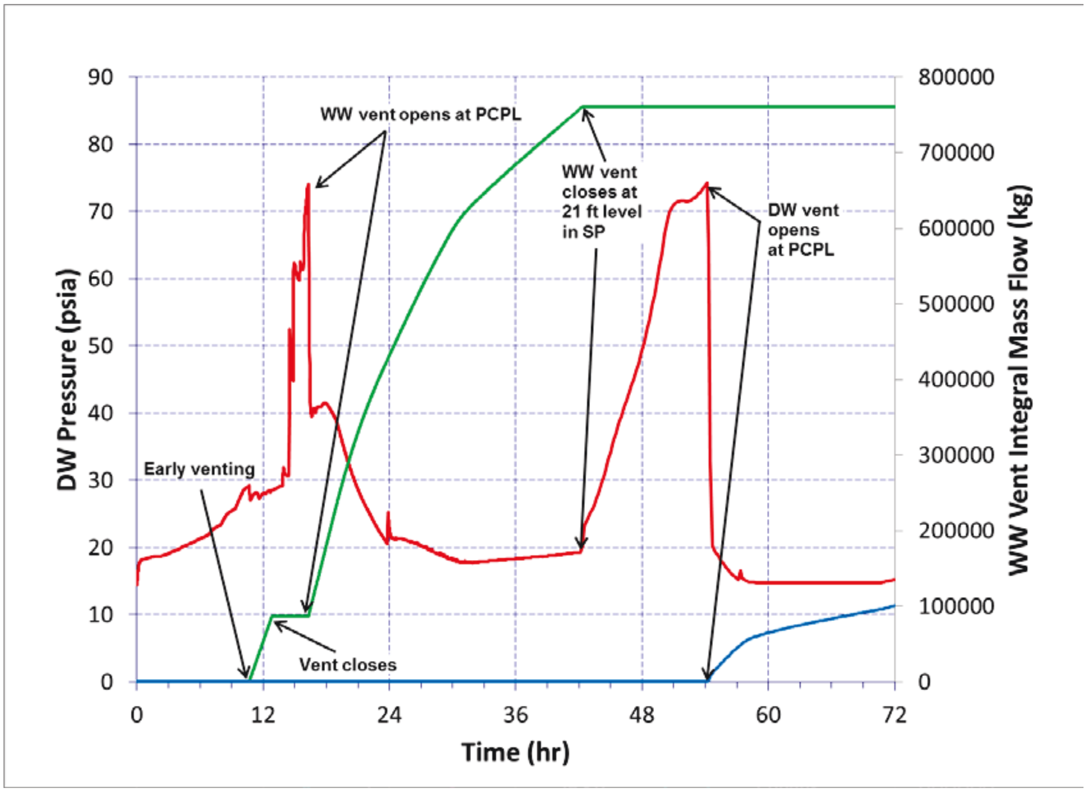


Figure 3-25 Mark I containment pressure and vent flow for Case 10 (SAWA)

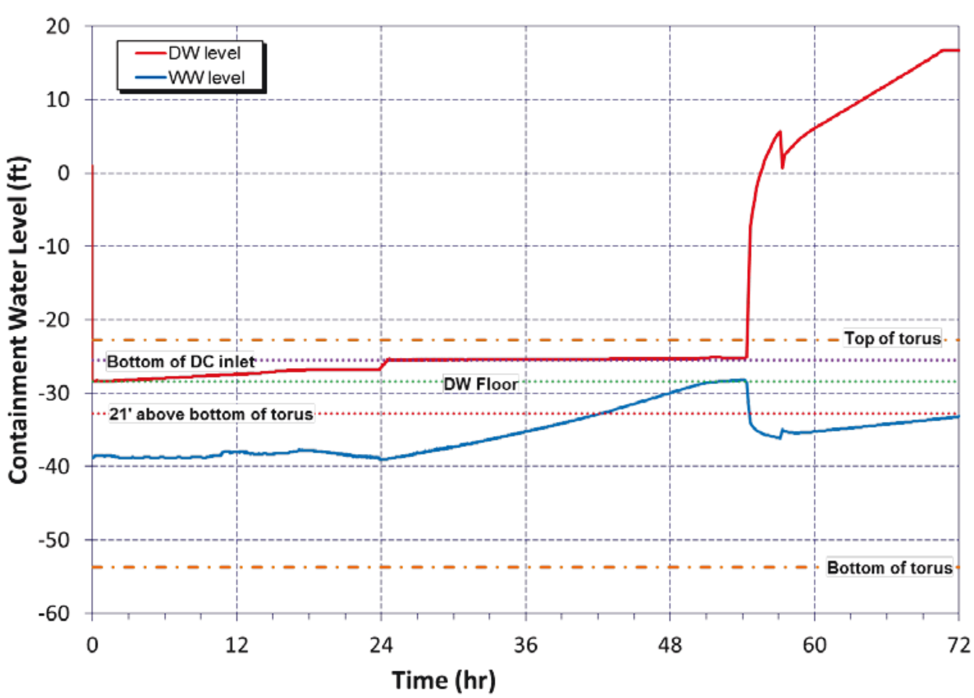


Figure 3-26 Mark I containment water level for Case 10 (SAWA)

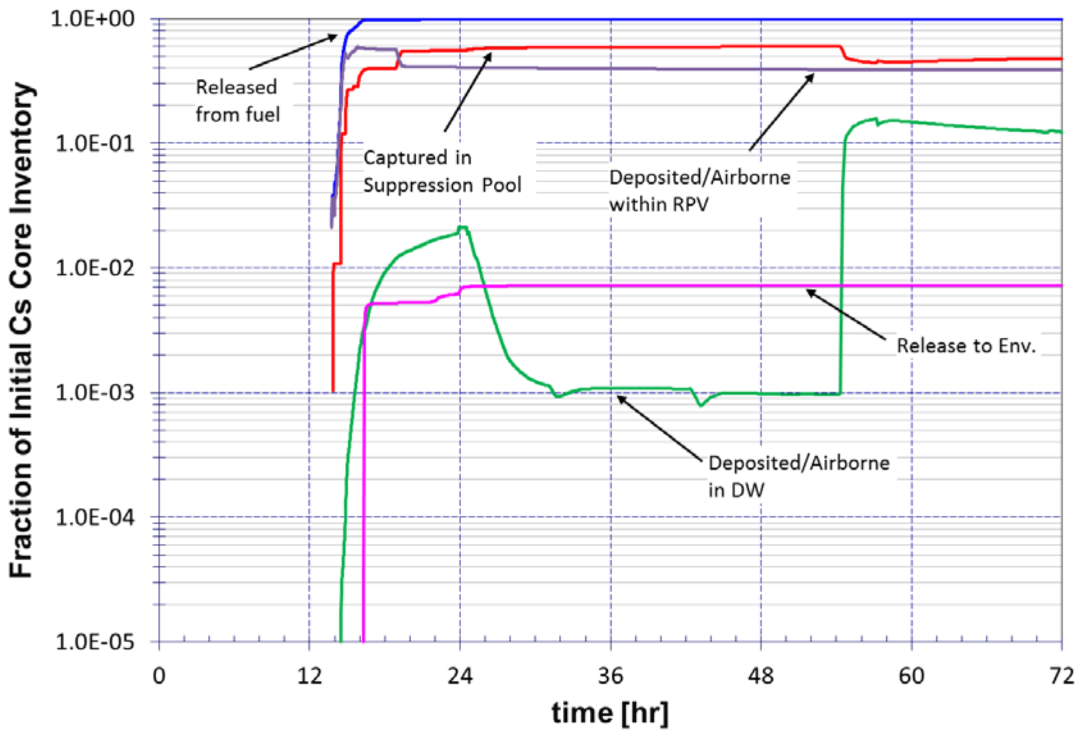


Figure 3-27 Mark I cesium fission product distribution for Case 10 (SAWA)

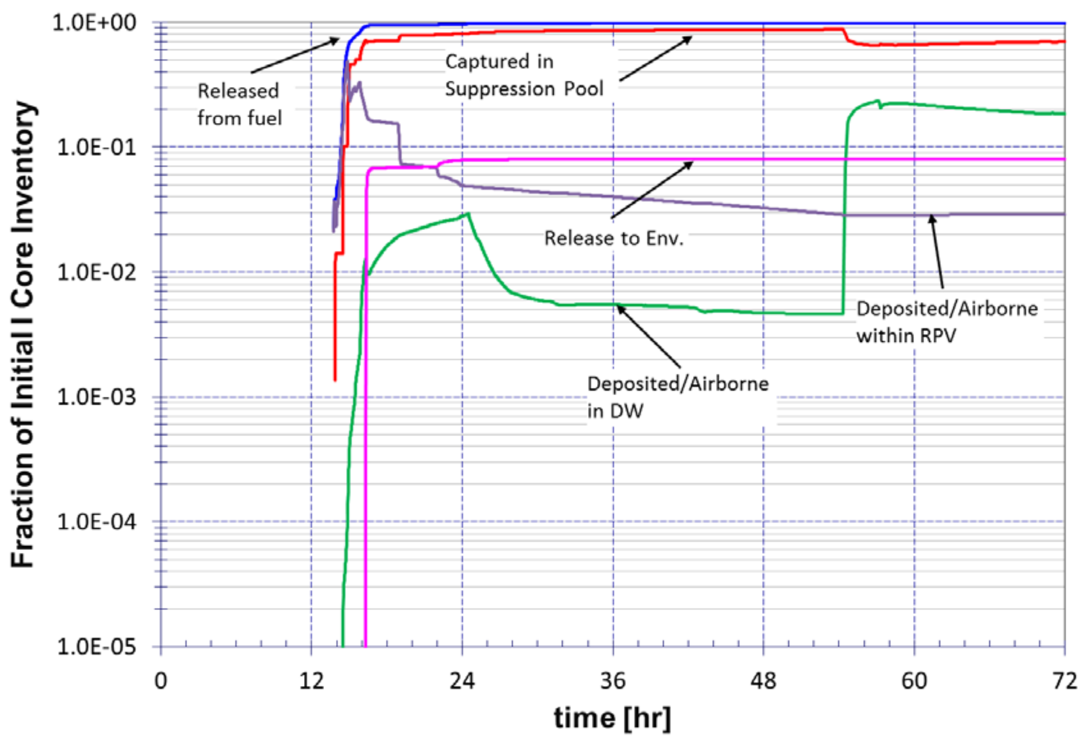


Figure 3-28 Mark I iodine fission product distribution for Case 10 (SAWA)

3.3.1.2 Accident Progression without Water Management/Addition

This section describes the thermal response of the plant without water injection after core damage. The representative scenario is Case 1 (see Table 3-2 for the list of boundary conditions). The focus here is only on parameters of interest such as the water level and thermodynamic conditions inside the containment, the mode of containment failure, and the source term. The timing of key events is provided in Table 3-4 that shows comparable timings with Cases 9 and 10.

As mentioned before, the leakage of water (total of 36 gpm) from the recirculation lines leads to a buildup of water inside the pedestal and the lower drywell as shown in Figure 3-29. By the time of lower head failure at 23 hours there is an accumulation of 1.6 ft of water on the drywell floor. Once the debris is ejected from the RPV (see Figure 3-30), it starts vaporizing the water and at the same time the heat transfer to water cools the debris. It takes 2.7 hours to completely deplete the existing water on the drywell floor. Without interaction with the water, the debris starts to heat up once again.

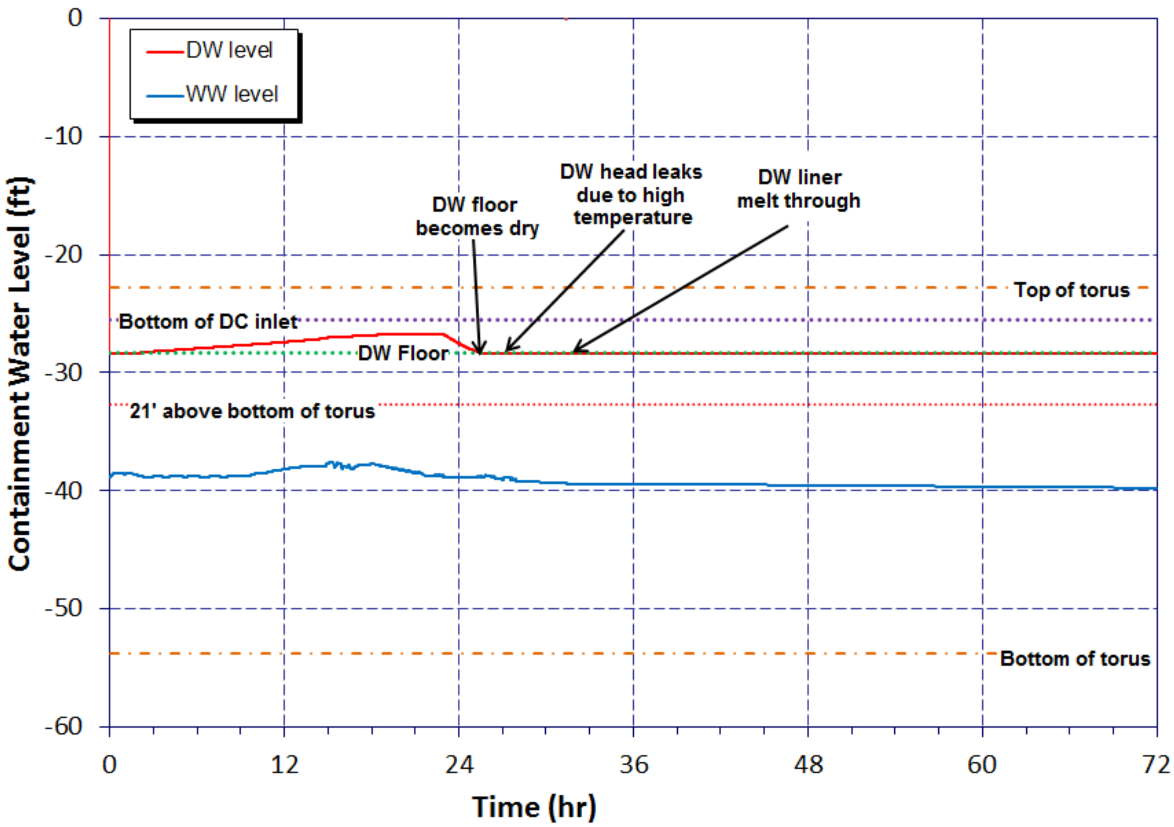


Figure 3-29 Mark I containment water level for Case 1

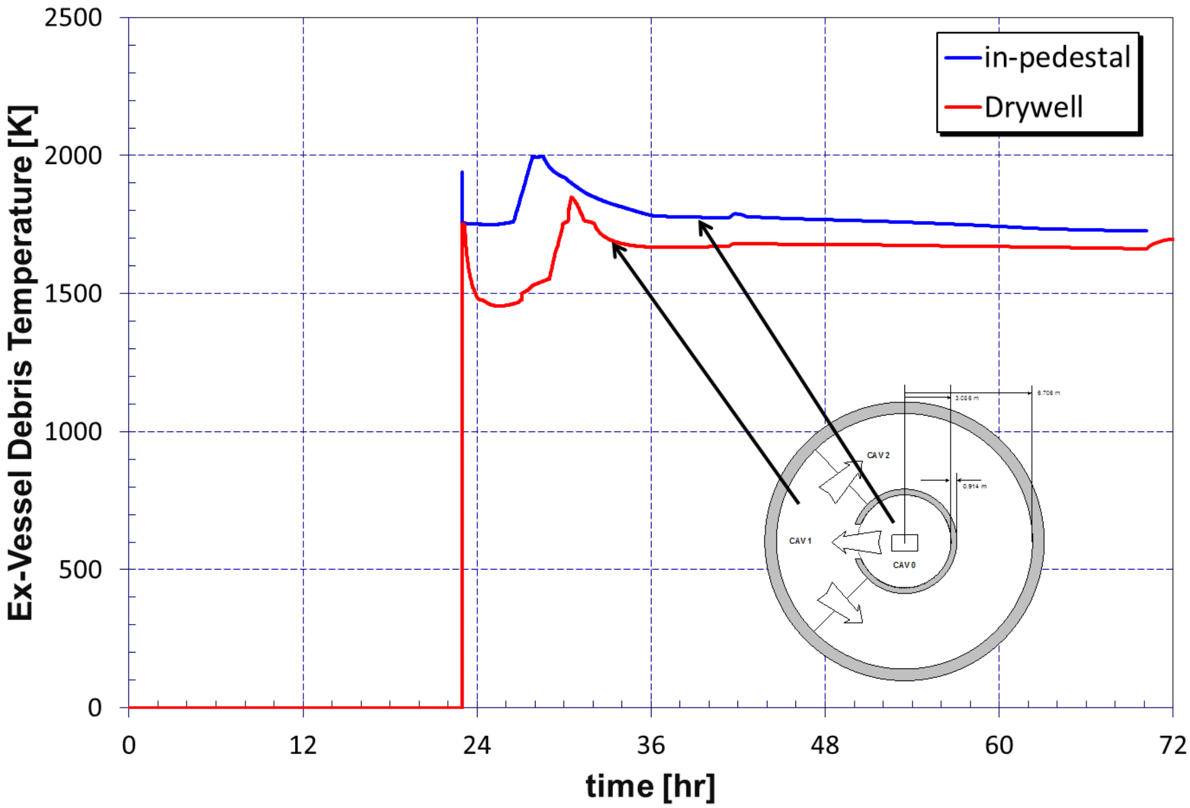


Figure 3-30 Mark I ex-vessel debris temperature for Case 1

The movement of debris out of the pedestal and on to the drywell floor is shown in Figure 3-31. The debris flows laterally out of the cavity through the open personnel access doorway and spreads out across the main drywell floor. However, the cooling of debris temporarily stops the flow towards the drywell liner. After the debris heats up again, the debris reaches the steel shell at the outer perimeter of the drywell and the thermal attack of the debris against the steel shell results in shell penetration. Because of wetwell venting and containment depressurization earlier, the shell failure and opening of a release pathway for fission products into the torus room of the reactor building does not result in any significant release to the environment.

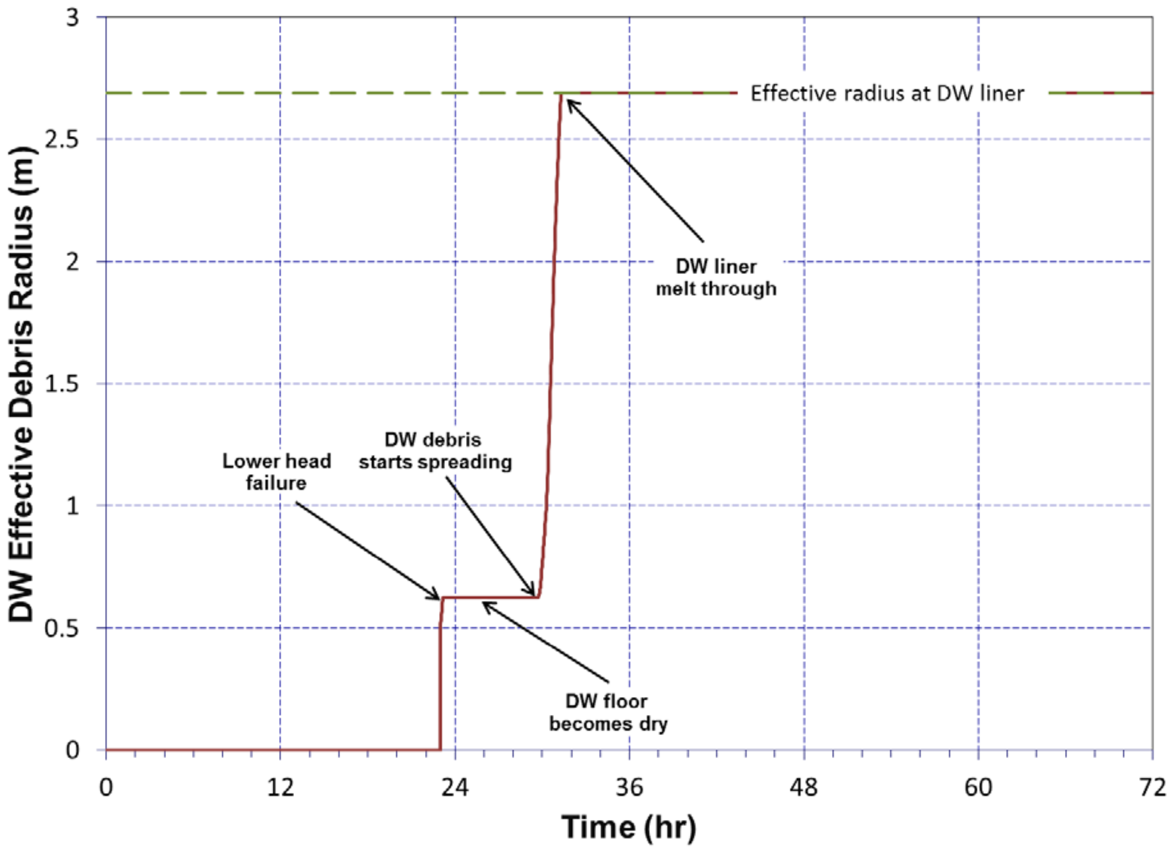


Figure 3-31 Mark I debris spreading in DW for Case 1

The containment atmosphere temperatures remain relatively low before lower head failure as seen before (see Figure 3-33). The presence of the water in the drywell also limits the temperature rise until the dryout in the drywell at 25.7 hours. At this time, the atmosphere temperature in the pedestal increases to nearly 1900 K and continues to increase in the long term. High gas temperature near the top the drywell leads to failure of the drywell head seals and the model assumes leakage through the head flange with a constant leak area of 0.1 ft^2. The leak rate is relatively small since the venting has already depressurized the containment.

Before drywell shell melt-through occurs, hydrogen leaks through the drywell head flange and accumulates in the reactor building refueling bay. A flammable mixture quickly develops in the refueling bay, causing a hydrogen combustion (see Figure 3-32). Small increases in internal pressure (0.25 psig) cause the blowout panels in the refueling bay to open; the roof also fails at an overpressure of 0.5 psig (see Reference [11] and Figure 3-6)[8]. Therefore, the failure of the refueling bay offers another release pathway to the environment immediately after a hydrogen burn occurs within the building. After melt-through of the drywell liner, additional hydrogen is released from the drywell into the torus room and is transported upward through open floor gratings into the ground level of the reactor building. The pressure rise within the building causes several doorways to open, including the large equipment access doorway at grade level. Several

[8] In the MELCOR model, combustible gases can still accumulate in the control volume leading to the second hydrogen combustion even though the roof has failed. This does affect the overall results since the containment fails by liner melt-through shortly after.

other doorways also open within the building, including personnel access doorways into the building stairwells. The large opening at grade level, coupled with the open flow areas in the refueling bay at the top of the building, creates an efficient transport pathway to the environment for material released from containment.

The fractional distribution of cesium and iodine are shown in Figure 3-34 and Figure 3-35. The environmental release is dominated by the initial sudden venting of containment followed by a gradual release after vessel breach due to revaporization of CsI as discussed before.

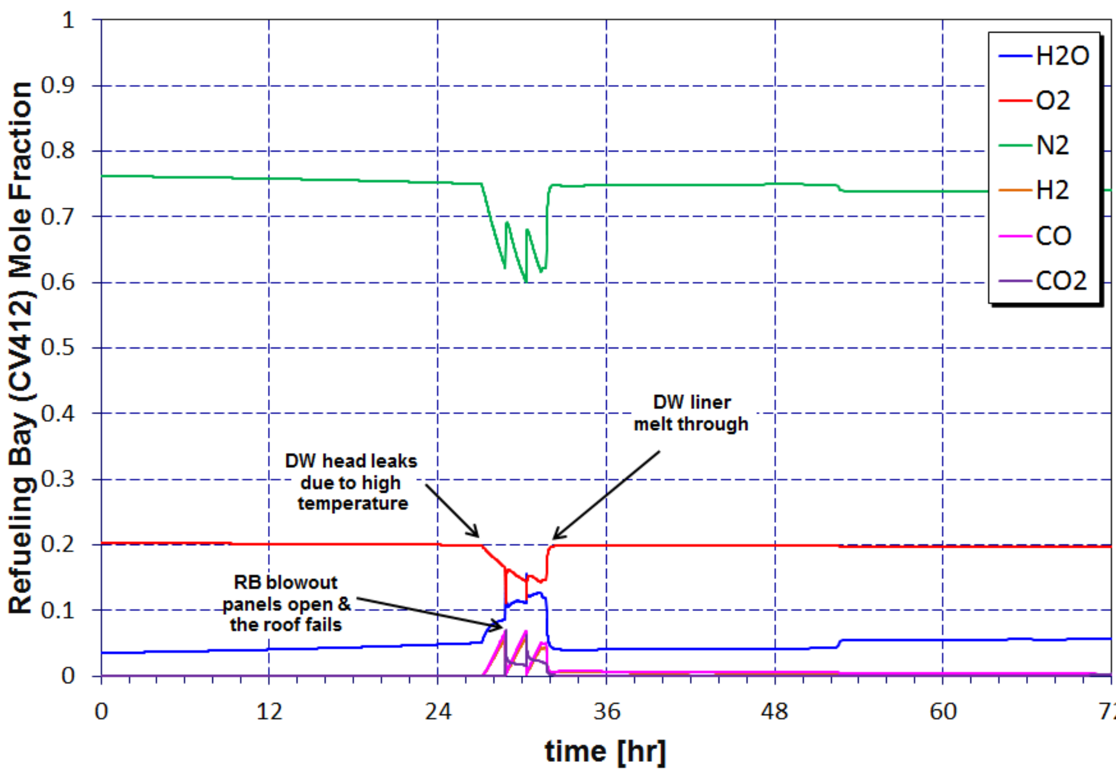


Figure 3-32 Mark I refueling bay gas concentration for Case 1

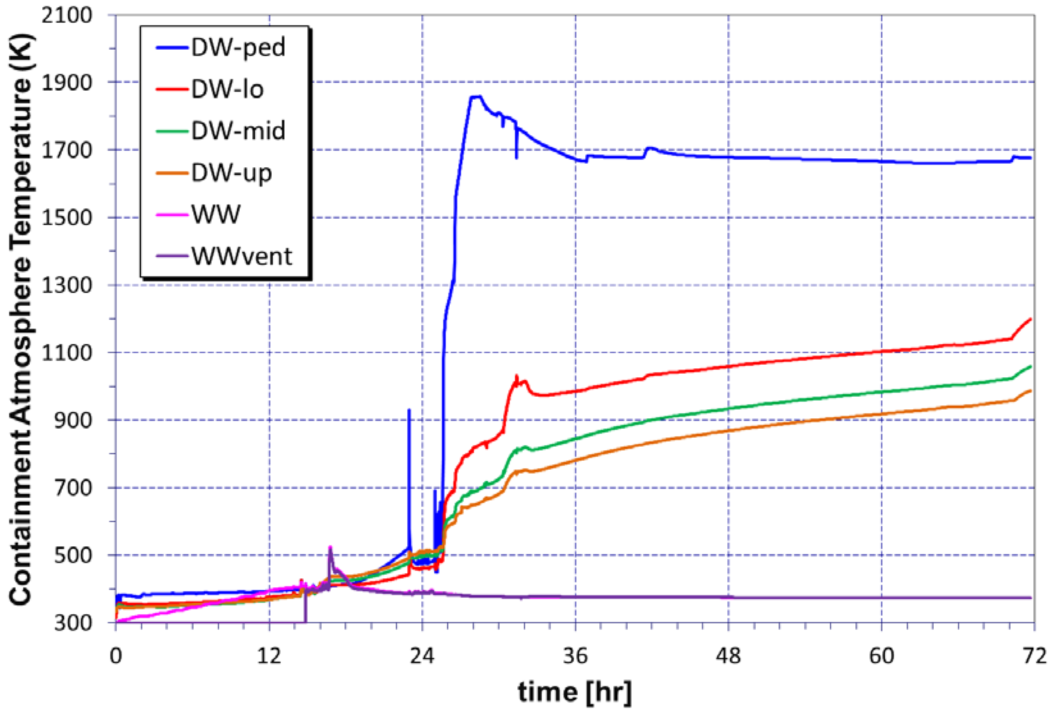


Figure 3-33 Mark I containment atmosphere temperature for Case 1

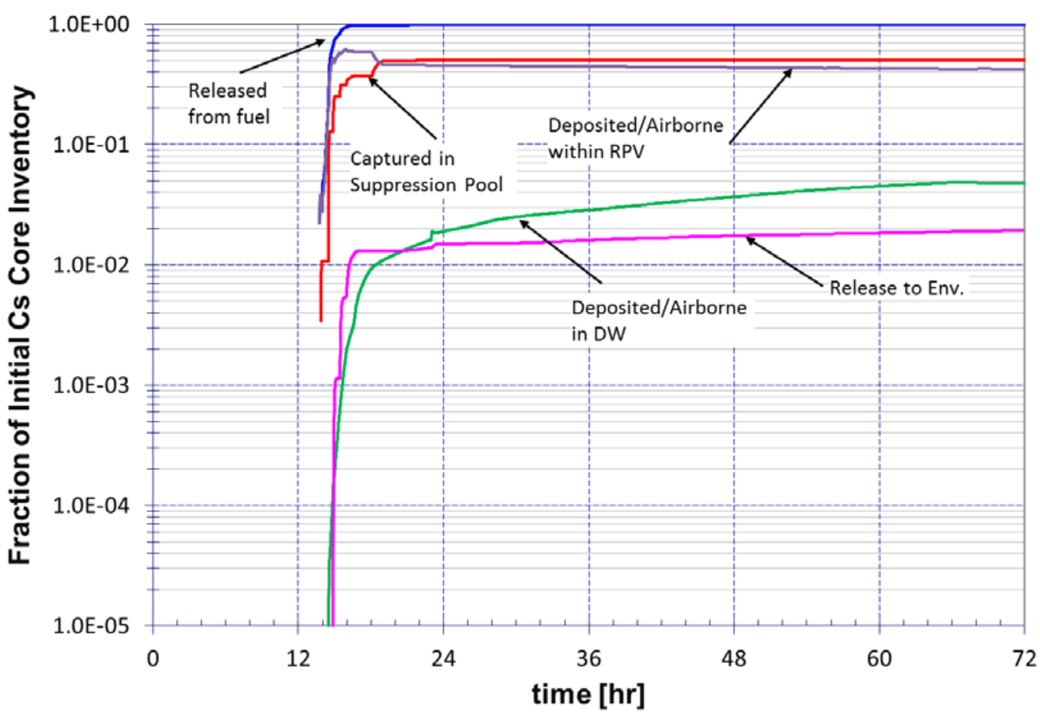


Figure 3-34 Mark I cesium fission product distribution for Case 1

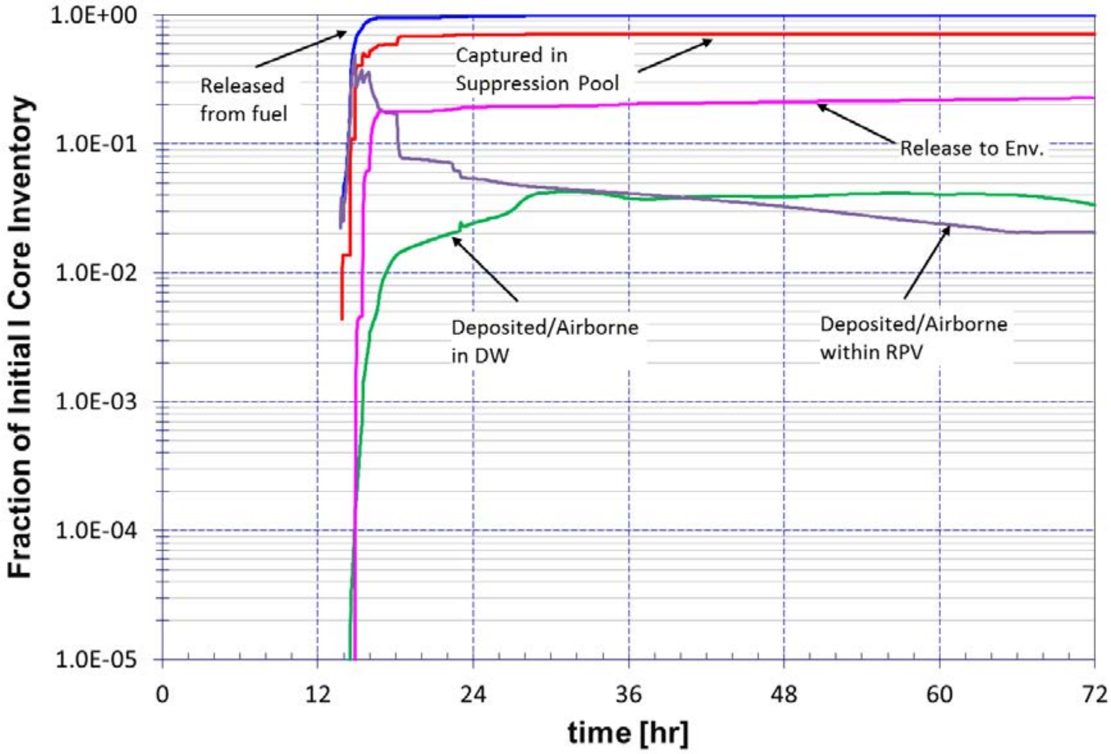


Figure 3-35 Mark I iodine fission product distribution for Case 1

3.3.1.3 Source Term and Containment Response

A summary of cesium release fractions for the different scenarios in the run matrix (see Table 3-2) is given in Figure 3-36. The summary of main parameters is provided in Table 3-5. In most cases with water injection, the cesium release fraction remains below 1% (or a DF of greater than 100) and the mode of containment venting and water injection do not greatly affect the cesium release. Changes in the boundary conditions such as opening of the SRVs by the operators before core damage, the fractional open area of thermally seized SRV, early (pre-core damage) containment venting, and injection source of RCIC (SP vs. CST) can affect the release fractions. These boundary conditions affect the core degradation and the thermodynamic conditions inside the RPV and ultimately the distribution of the fission products in the RPV and containment.

In general, the cases without water injection or when water injection stops at high water level (see, for example, case 7 in Table 3-2) show higher release fractions. However, the release fraction of cesium is not significantly affected by the containment failure (liner melt-through and DW head leakage) because water injection occurs at the time of vessel breach whereas controlled containment venting has occurred much sooner during core damage. Therefore, in all cases the early release is characterized by a sudden release at the time of venting.

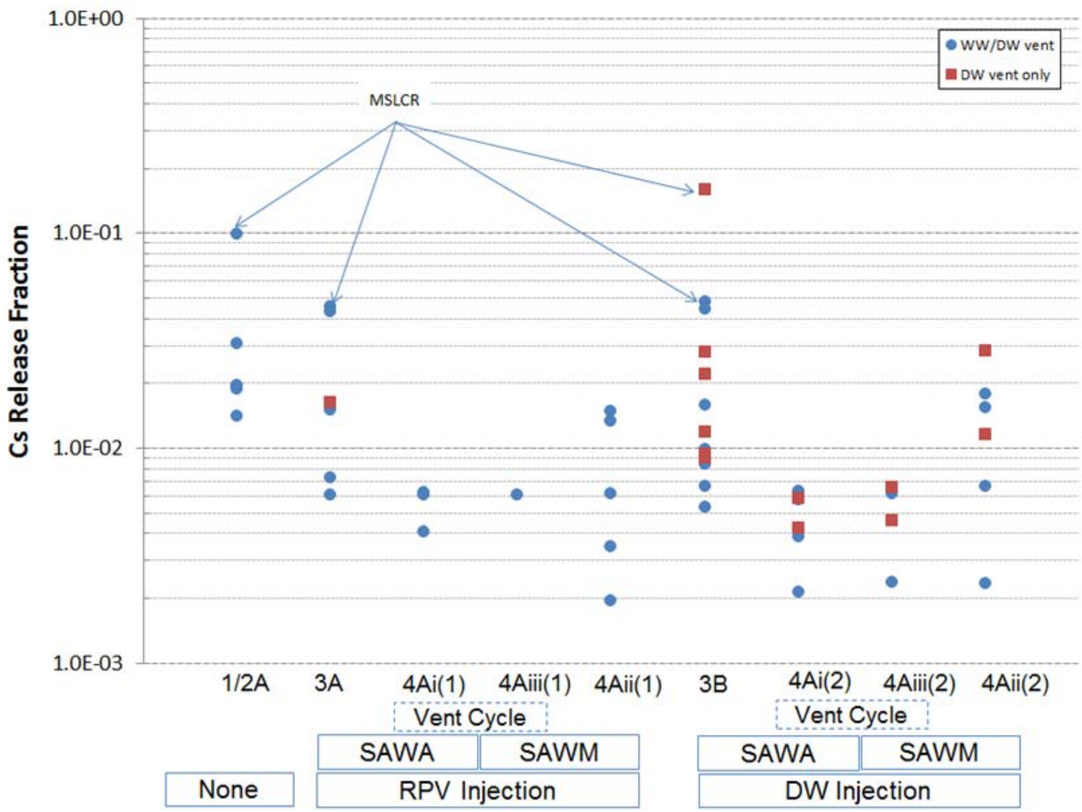


Figure 3-36 Mark I cesium environmental release fraction

Figure 3-37 shows the effects of containment venting, water injection, and suppression pool scrubbing on fission product releases to the environment for several cases (with and without water injection at lower head failure). The injection of water after lower head failure stabilizes the environmental release, but clearly does not affect the magnitude of release at the time of venting.

The highest releases are associated with main steam line creep rupture cases since suppression pool can be bypassed. For these scenarios, the DW head leakage and hydrogen combustion in the refueling bay occurs almost immediately after the release into the drywell. This provides a direct path for release to the environment.

Figure 3-38 shows that for all the cases with sustained water injection into the drywell or the RPV (that eventually accumulates on the drywell floor), the maximum structure temperatures at the drywell upper head or the drywell liner near the elevation of the drywell vent remains below 500°F. The cases without water injection in general experience the highest temperatures at the time of vessel breach, since the exposure of the debris to the drywell atmosphere as it exits the vessel and the circulation of hot gases inside the drywell can heat up both the atmosphere and the surrounding structures. Water injection and submergence of the debris result in direct heat transfer from the debris to the overlying water, and the drywell atmosphere directly transfers heat to the cooler pool surface. In addition, the water cools the concrete decomposition gases before they enter the drywell atmosphere.

The relation between the maximum gas and structure temperatures in the middle drywell is shown in Figure 3-39. As stated before, the cases without water injection or main steam line creep

rupture result in higher temperatures in the drywell. For the sustained water injection cases, the structure temperature on the average remains about 100ºF cooler than the atmosphere.

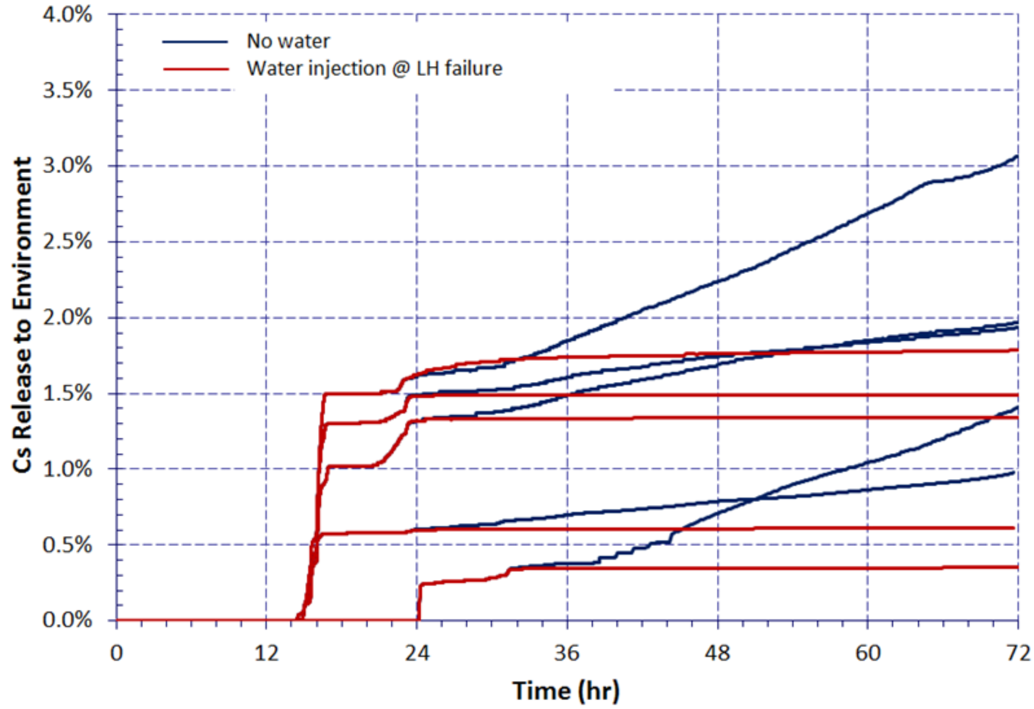


Figure 3-37 Effect of water injection on selected cesium releases for selected cases

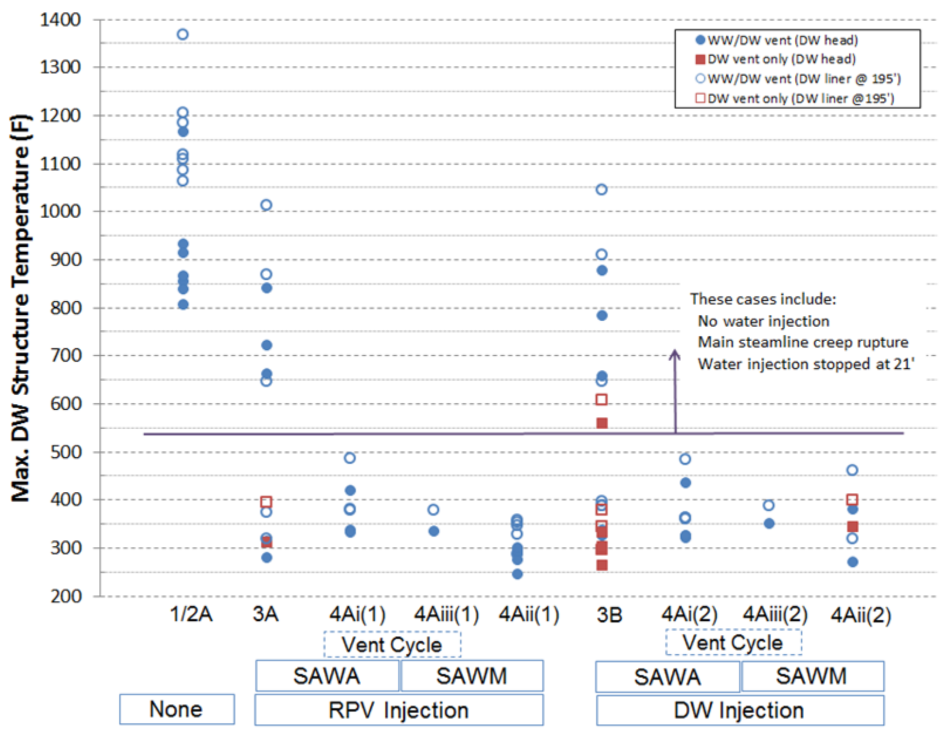


Figure 3-38 Mark I containment gas and structure temperatures

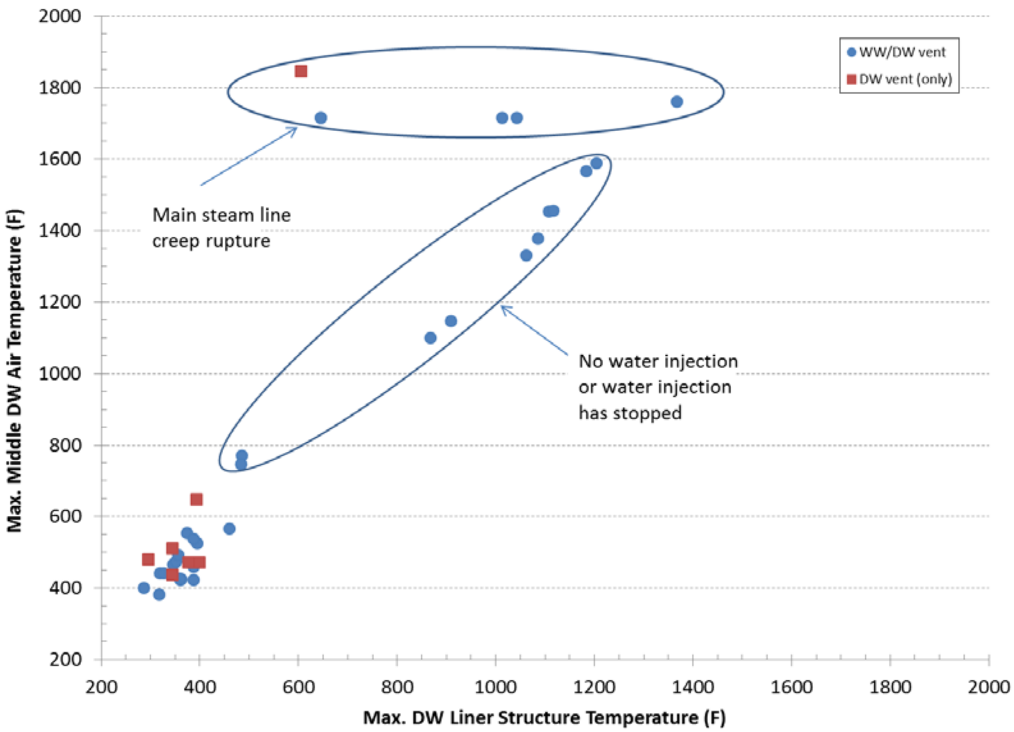


Figure 3-39 Mark I maximum containment gas and DW liner temperatures

The behavior of hydrogen in the containment is shown in Figure 3-40 and Figure 3-41. The total mass of hydrogen produced during the transient for Case 9 is about 2600 kg with about 1000 kg generated in-vessel. The blue line represents the total hydrogen generation, which should be almost identical with the amount remaining inside the containment and the amount that is vented (represented by the green line). The amount of hydrogen that remains inside the containment (both the drywell and the wetwell air space as shown by the red line) quickly decreases as a result of venting. With the wetwell vent open during the transient, the total amount of hydrogen is kept very low in the long term (below 30 kg). Therefore, containment venting is very efficient in purging the hydrogen from the containment. The presence of water seems to avoid containment failure and any uncontrolled release of hydrogen to the reactor building, which remains intact for the duration of the transient.

The conditions inside the containment show that following vessel breach and water injection, evaporation of water leads to a high mole fraction of steam in excess of 80% (see Figure 3-41 and Figure 3-42). The mole fraction of steam in the wetwell at the time of containment venting is also very high, which leads to steam inerting of the containment vent line in a very short time (see Figure 3-43 and Figure 3-44). These conditions are not conducive to an energetic hydrogen combustion.

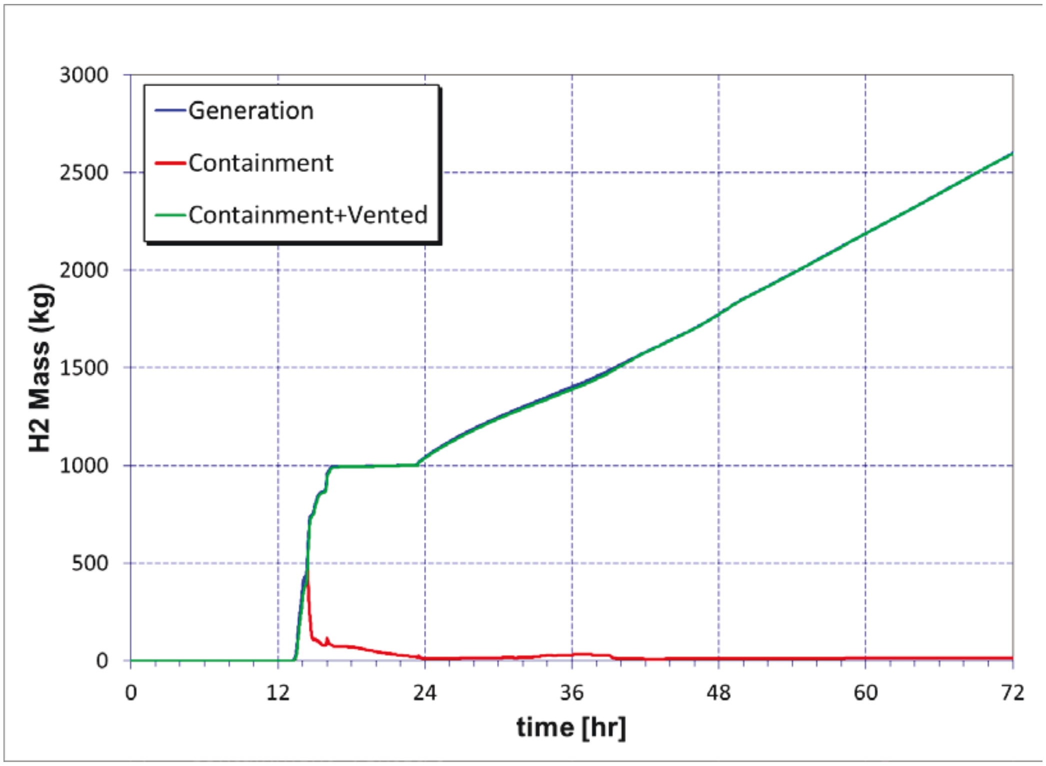


Figure 3-40 Mark I hydrogen generation and transport for Case 9

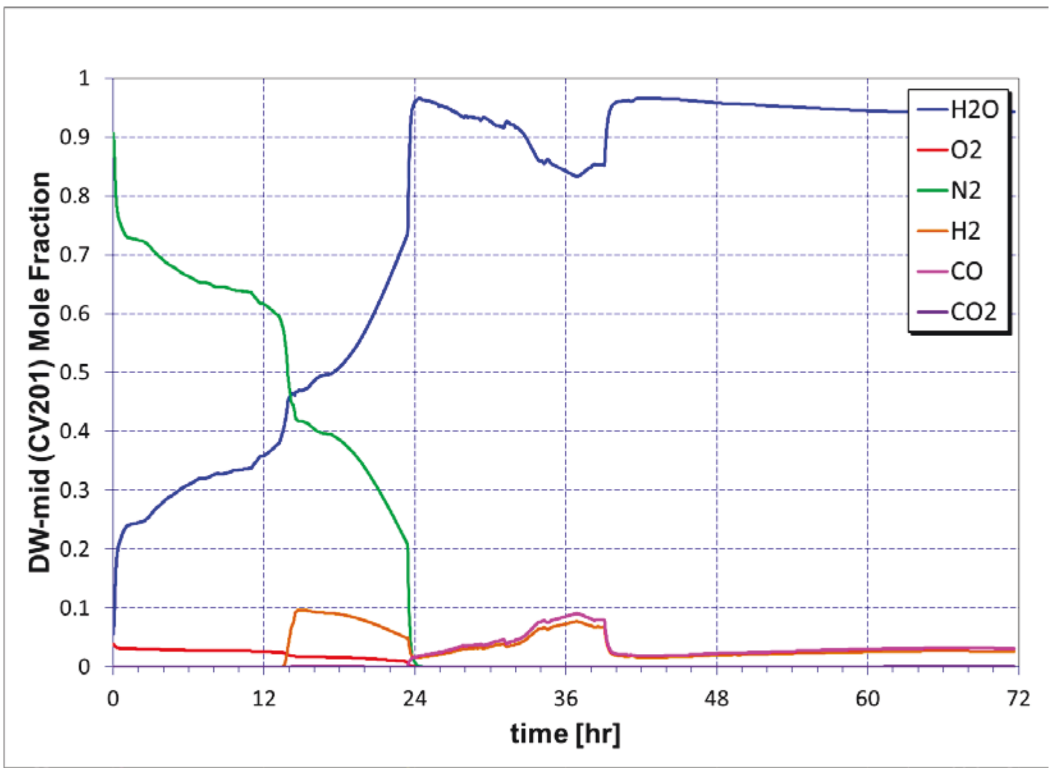


Figure 3-41 Mark I drywell gas distribution for Case 9

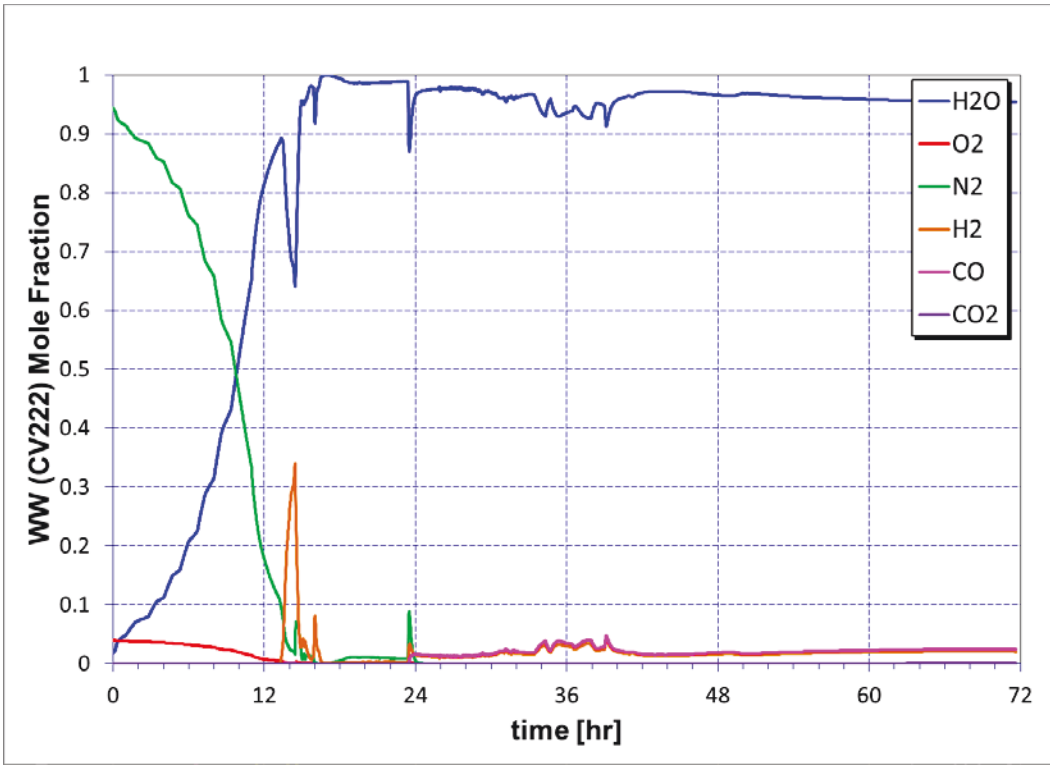


Figure 3-42 Mark I wetwell gas distribution for Case 9

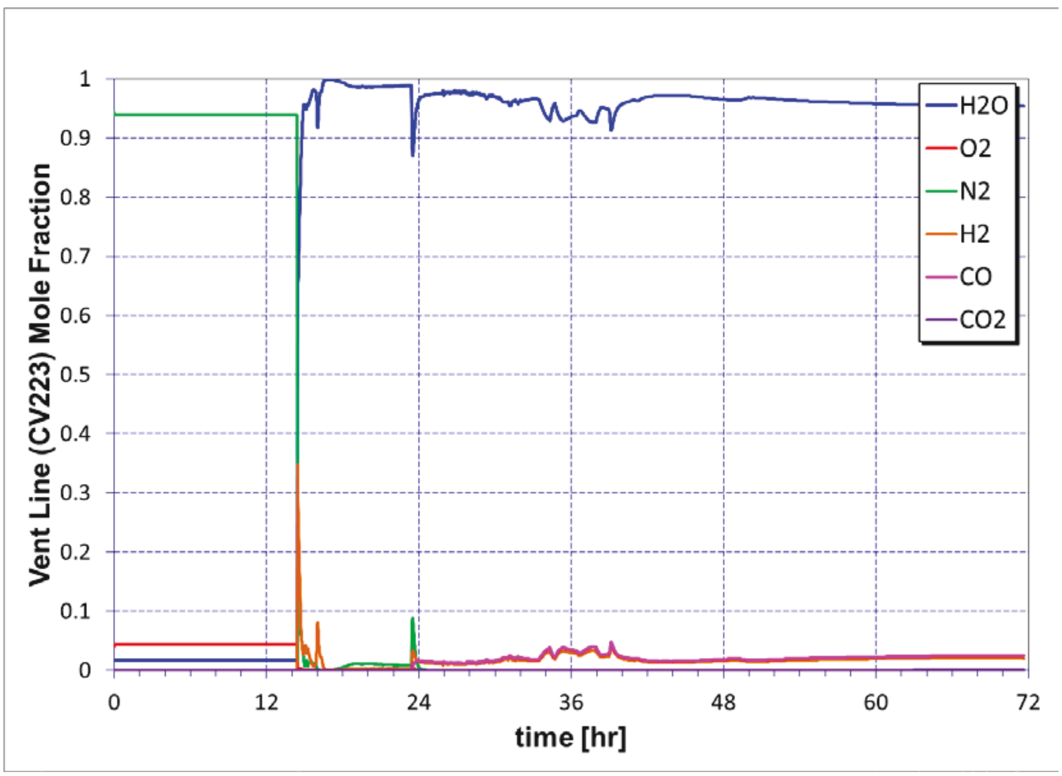


Figure 3-43 Mark I wetwell vent line gas distribution for Case 9

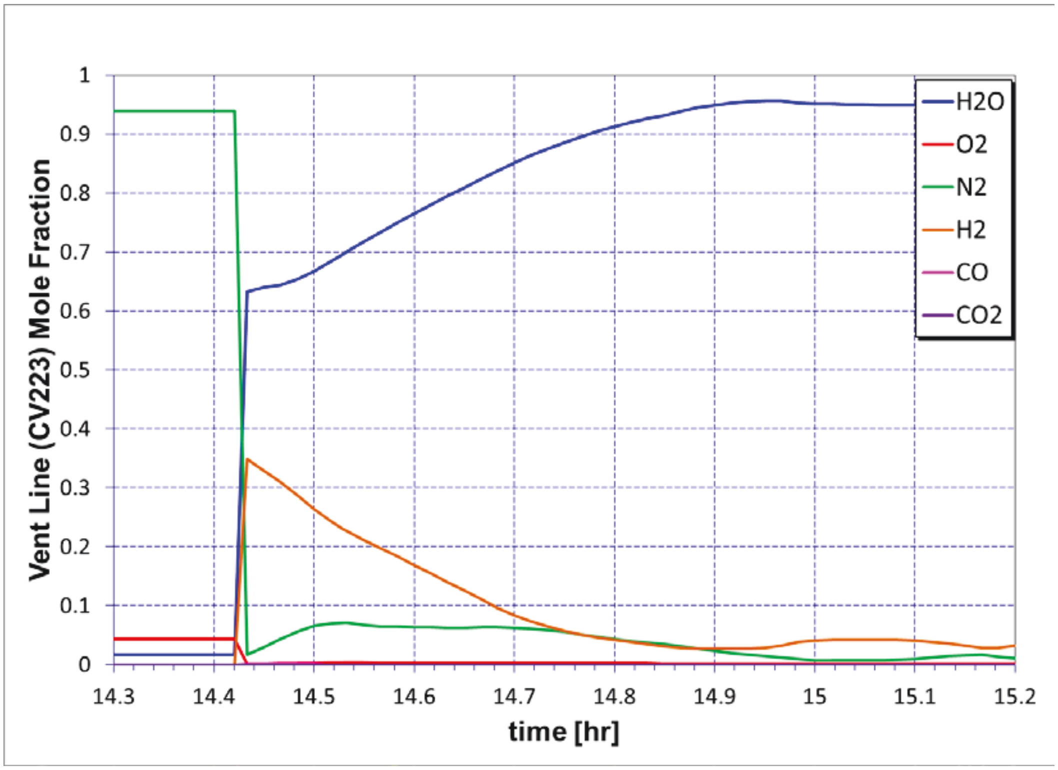


Figure 3-44 Mark I wetwell vent line gas distribution at venting for Case 9

A calculation was performed to investigate the impact of water addition prior to vessel breach on the fission product release to the atmosphere. This is a variation of Case 9 with the assumption that RPV injection begins at 13.5 hours or shortly after core heat up (see Table 3-4). The water level is shown in Figure 3-45. The water level is near the bottom of active fuel by the time the 500 gpm FLEX injection starts. It takes about 0.8 hours before the level is restored above the top of active fuel and some fission product release from the fuel has already occurred. Figure 3-46 indicates that about 30% of the cesium inventory in the core has already been released from the fuel. However, the release to the environment is significantly lower as shown in Figure 3-47, and it occurs at a much later time. The cooling of the core affects the containment pressurization and delays the timing of venting. There is only a small time window before the cesium release from the core is almost complete (about 2 hours as shown in Figure 3-46). In all the base MELCOR calculations, it was assumed that injection begins at the time of lower head failure and the initial venting sudden release was predicted to occur much sooner. Therefore, the injection timing is important in determining if there is any reduction in release.

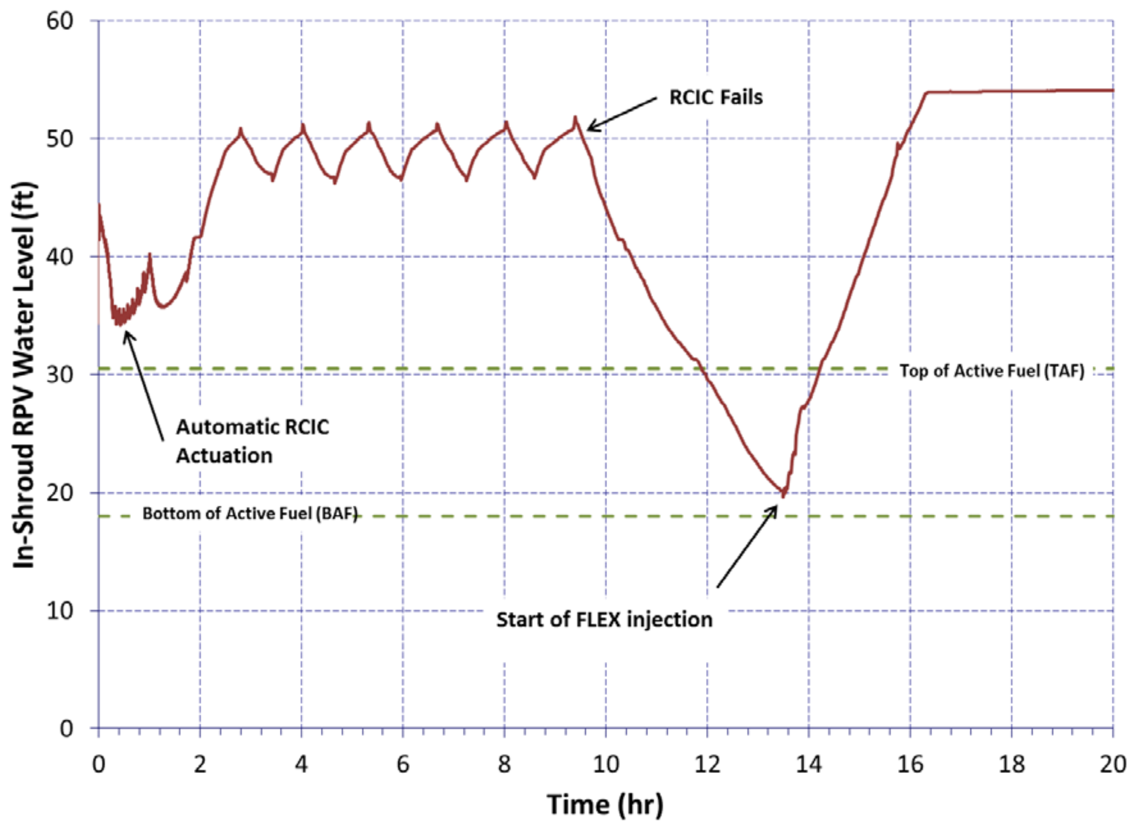


Figure 3-45 Mark I RPV water level for Case 9 (IVR)

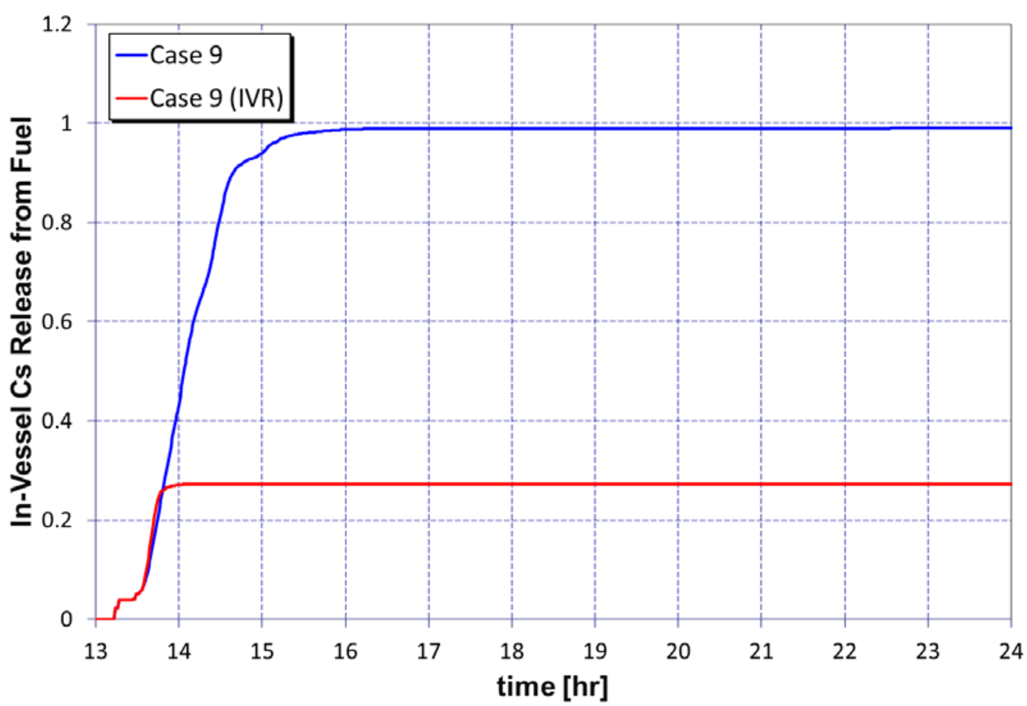


Figure 3-46 Mark I cesium release fraction from fuel for Case 9 (IVR)

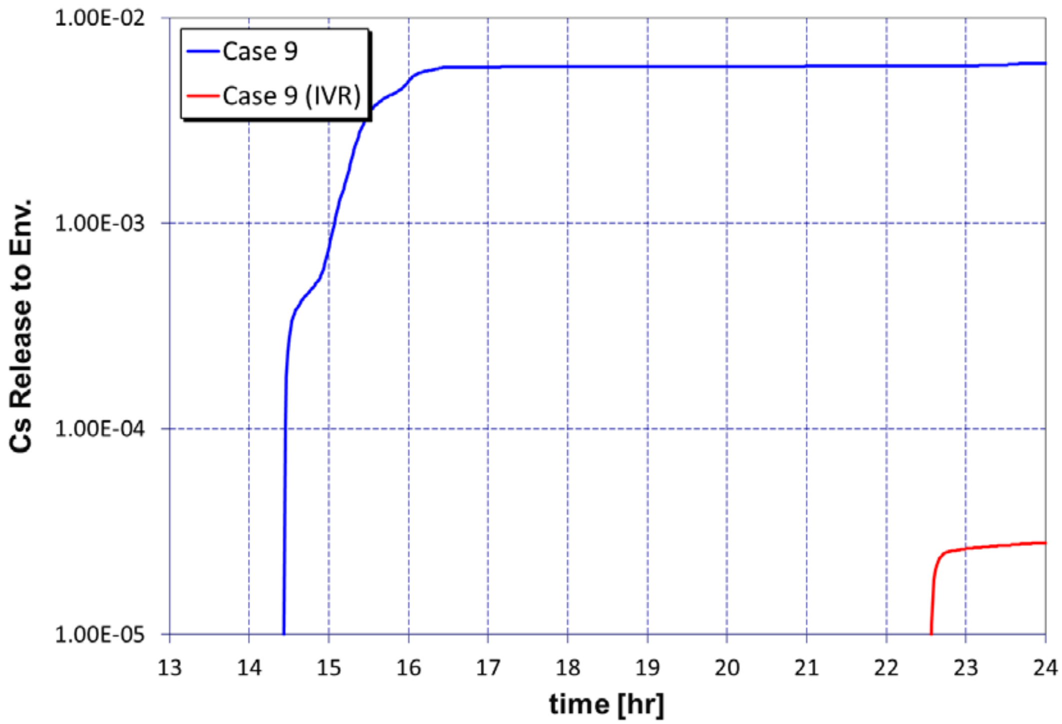


Figure 3-47 Mark I cesium release fraction to environment for Case 9 (IVR)

Table 3-5 Summary of main parameters for Mark I analysis

Option	Case	Cs (RF) @ 72 hours	Cs (DF = 1/RF)	I (RF) @ 72 hours	Early venting (opn/cls) (hr)	RCIC failure (hr)	SRV failure (hr/# cycles)	MSL failure (hr)	WW venting (opn/cls) (hr)	DW venting (opn/cls) (hr)	LH fails (hr)	FLEX stops (hr)	DW head leaks (hr)	Liner failure (hr)	In-core H2 (kg)	Meject (ton)
1/2A	1	1.94E-02	52	2.28E-01	-	9.6	16/49	-	14.9/-	-	23.0	-	27.1	31.4	1195	292
1/2A	1S1	6.56E-03	152	6.36E-02	5.3/17.9	14.2	21/53	-	23.3/-	-	29.4	-	34.1	38.0	1167	297
1/2A	2	3.06E-02	33	2.64E-01	-	9.6	16/49	-	14.9/-	-	22.6	-	26.9	30.8	1198	274
1/2A	3	9.94E-02	10	3.08E-01	-	4.0	-/368	9.82	9.82/-	-	16.6	-	9.8	19.2	1165	345
1/2A	4	1.87E-02	53	9.80E-02	10.8/14.4	11.1	17/50	-	18/-	-	24.7	-	28.2	31.7	1330	277
1/2A	5	1.41E-02	71	6.50E-02	11.4/19.8	16.0	23.2/54	-	24.2/-	-	31.2	-	36.3	43.8	1235	279
1/2A	6	1.96E-02	51	2.03E-01	-	9.6	16/49	-	13.9/-	-	23.1	-	26.9	30.5	1224	305
3A	7	1.51E-02	66	1.95E-01	-	9.6	16/49	-	14.9/-	-	23.0	40.7	52.2	-	1195	291
	7dw	1.62E-02	62	1.77E-01	-	9.6	15.5/49	-	-	14.9/-	23.1	43.7	-	-	1165	298
3A	10	7.26E-03	138	8.10E-02	10.7/12.8	9.6	16/49	-	16.3/42.2	54.3/-	23.8	72	-	-	1232	287
3A	11	6.04E-03	165	7.81E-02	-	9.5	9.5/40	-	14.4/41.5	56.5/-	23.4	72	-	-	1016	280
4Aii(1)	8	1.49E-02	67	1.93E-01	-	9.6	16/49	-	14.9/-	-	23.0	72	-	-	1180	291
4Aii(1)	9	6.12E-03	163	7.86E-02	-	9.5	9.5/40	-	14.4/-	-	23.4	72	-	-	1032	280
4Aii(1)	12	1.34E-02	75	1.64E-01	-	9.6	15.6/49	-	13.9/-	-	23.1	72	-	-	1226	289
4Aii(1)	13	3.50E-03	286	3.91E-02	11.4/19.8	16.0	23.2/49	-	24.2/-	-	31.2	72	-	-	1173	255
4Ai(1)	14	6.18E-03	162	6.10E-02	-	9.6	16/49	-	14.9/44.5	-	25.5	38.3	46.5	-	1101	336
4Aiii(1)	15	6.05E-03	165	5.90E-02	-	9.6	16/49	-	14.9/...	-	25.5	72	-	-	1101	336
4Ai(1)	18	4.07E-03	246	5.00E-02	10.7/12.8	9.6	16/49	-	16.3/31.0	35.9/...	23.9	72	-	-	1201	284
4Ai(1)	16	6.08E-03	164	5.90E-02	-	9.6	16/49	-	14.9/37.8	41.3/-	25.5	72	-	-	1101	336
3B	21	1.59E-02	63	2.00E-01	-	9.6	16/49	-	14.9/-	-	23.0	40.2	48.5	-	1196	300
3B	24	8.37E-03	119	8.30E-02	10.7/12.8	9.6	16/49	-	16.3/41.5	58.0/-	23.8	72	-	-	1240	298
	24dw	2.19E-02	46	8.61E-02	10.8/12.8	9.6	15.5/49	-	-	16.0/-	23.6	72	-	-	1148	264
4Aii(2)	22	1.78E-02	56	2.23E-01	-	9.6	16/49	-	14.9/-	-	22.6	72	-	-	1152	274
	22dw	2.82E-02	35	1.87E-01	-	9.6	15.5/49	-	-	14.9/-	23.0	72	-	-	1121	269
4Aii(2)	23	1.54E-02	65	1.95E-01	-	9.6	15.5/49	-	14.9/-	-	23.0	72	-	-	1194	300
4Aii(2)	25	6.60E-03	152	8.10E-02	-	9.6	9.6/40	-	14.4/-	-	23.4	72	-	-	1063	283
	25dw	1.16E-02	86	1.21E-01	-	9.5	9.5/40	-	-	14.4/-	23.5	72	-	-	1107	296
3B	26	6.65E-03	150	8.15E-02	-	9.5	9.5/40	-	14.4/-	-	23.4	72	-	-	1060	283
	26dw	1.19E-02	84	1.20E-01	-	9.5	9.5/40	-	-	14.4/-	23.5	72	-	-	1108	296
4Ai(2)	27	6.28E-03	159	6.10E-02	-	9.6	16/49	-	14.9/44.2	-	25.5	37.9	46.0	-	1096	337
4Aiii(2)	28	6.11E-03	164	5.80E-02	-	9.6	16/49	-	14.9/...	-	25.5	72	-	-	1096	337
	28dw	6.53E-03	153	4.88E-02	-	9.6	16/49	-	-	14.9/...	23.6	72	-	-	1153	291
4Ai(2)	32	3.86E-03	259	4.87E-02	10.7/12.8	9.6	16/49	-	16.3/25.2	38.5/...	23.9	72	-	-	1212	284
4Ai(2)	30	6.18E-03	162	5.91E-02	-	9.6	16/49	-	14.9/28.6	41.7/...	25.5	72	-	-	1096	337
	30dw	5.82E-03	172	4.31E-02	-	9.6	16/49	-	-	14.9/-	23.6	72	-	-	1154	303
4Aiii(2)	29	2.38E-03	420	2.19E-02	-	9.5	9.5/40	-	14.4/...	-	23.2	72	-	-	1083	254
	29dw	4.62E-03	216	1.93E-02	-	9.5	9.5/40	-	-	14.4/...	23.6	72	-	-	1126	323
4Ai(2)	31	5.75E-03	174	2.39E-02	-	9.5	9.5/40	-	14.4/...	-	23.2	72	-	-	1092	254
	31dw	4.23E-03	236	1.34E-02	-	9.5	9.5/40	-	-	14.4/...	23.6	72	-	-	1126	323
3A	41	4.56E-02	22	1.41E-01	-	4.0	-/368	9.82	9.82/-	-	16.6	32.6	9.8	42.2	1178	343
3B	43	4.78E-02	21	1.61E-01	-	4.0	-/368	9.82	9.82/-	-	16.6	32.6	9.8	41.7	1164	345
3A	42	4.33E-02	23	1.07E-01	-	4.0	-/368	9.82	9.82/32.6	-	16.6	72	9.8	-	1178	343
3B	44	4.42E-02	23	1.08E-01	-	4.0	-/368	9.82	9.82/32.6	-	16.6	72	9.8	-	1164	345
4Aii(1)	47	1.94E-03	515	1.25E-02	-	4.0	7.9/270	-	11.4/-	-	14.9	72	-	-	1105	339
4Aii(2)	48	2.35E-03	426	1.70E-02	-	4.0	7.9/270	-	11.4/-	-	14.9	72	-	-	1104	339
3B	45	8.99E-03	111	9.61E-02	-	9.1	5.2/270	-	-	14.8/-	24.1	72	-	-	1106	286
3B	46	9.80E-03	102	1.10E-01	-	9.1	5.2/270	-	14.8/41.2	57.2/-	23.6	72	-	-	1210	268
3B	49	5.31E-03	188	1.68E-02	-	0.0	1.7/158	-	7.3/20.6	33.7/-	7.3	65	-	-	1258	348
4Ai(2)	50	2.13E-03	469	1.10E-02	-	0.0	1.7/158	-	7.3/15.2	19.4/...	7.3	56	-	-	1255	348
3B	51	9.30E-03	108	1.01E-01	9.2/-	9.1	5.2/270	-	-	9.2/-	23.4	72	-	-	1335	317
3B	52	1.59E-01	6	3.44E-01	7.7/-	14.1	-/824	19.2	-	7.7/-	26.8	72	19.2	-	1378	347
3B	53	2.79E-02	36	2.91E-01	7.7/-	14.1	18.5/785	-	-	7.7/-	26.7	72	-	-	1324	351

3.3.2 BWR Mark II Results

The following analyses are presented to facilitate a general understanding of the Mark II containment response. The representative cases discussed below document cases with and without SAWA. Case 11 has been selected as the representative water injection case and is discussed in Section 3.3.2.1 whereas Case 1 is discussed in Section 3.3.2.2 to document a case without water injection. Event outlines are provided in Table 3-6 for the selected representative cases.

Table 3-6 Timing of key events for selected Mark II cases

Event Timing (hr)	Case 1 (no water)	Case 11 (SAWA)
Start of ELAP	0.0	0.0
Operators first open SRV to control pressure	0.17	0.17
RCIC actuation signal	0.01	0.01
Operators open SRV to control pressure (200-400 psig)	1.0	1.0
RCIC flow terminates	8.4	8.4
SRV sticks open or operators open SRV after RCIC fails	16.8	8.4
Water level reaches TAF[9]	10.7	10.3
First hydrogen production	12.9	12.1
First fuel-cladding gap release	13.0	12.2
Start of containment venting at 60 psig	22.8	20.3
Relocation of core debris to lower plenum	15.3	14.5
RPV lower head dries out	20.3	18.5
RPV lower head fails grossly	22.0	20.0
Calculation terminated	72	72
Selected MELCOR Results	**Case 1**	**Case 11**
Debris mass ejected (1000 kg)	248	220
In-vessel hydrogen generated (kg)	1232	1307
Iodine release fraction at 72 hr	1.98e-1	4.50e-3
Cesium release fraction at 72 hr	2.46e-2	4.22e-4

*RCIC actuates erroneously prior to low level signal. Corrected runs give RCIC actuation at 0.07 hours. No significant deviation is

3.3.2.1 Accident Progression with Water Addition

Case 11 is presented here as the representative water addition case. The options in Table 3-3 dictate that containment venting will commence using the wetwell vent line when containment pressure reaches PCPL and FLEX injection will begin at lower head failure. FLEX injection is modeled at a continuous volumetric rate of 500 gpm injection sourced to the RPV. As the wetwell fills due to continuous FLEX injection, switchover from the wetwell vent to the drywell vent will be available should this occur.

The initiating event results in the loss of AC power, containment isolation, reactor scram, MSIV closure, feedwater coastdown, and recirculation pump trip. The loss of injection and closure of the MSIVs permits RPV pressure to increase due to continual heat generation. RPV pressure

---

[9] This is the time when the water level in the downcomer reaches the top of active fuel. The water level in the core is slightly different.

eventually increases until the lowest set point SRV actuates automatically. Figure 3-48 displays the RPV pressure for Case 11.

RCIC injection initiates shortly after the reactor scram. RCIC pump suction is aligned to the suppression pool. RCIC injects approximately 600 gpm of suppression pool water into the vessel while steam drawn to operate the RCIC turbine is exhausted to the suppression pool. Operators assume control of the RCIC system by manually throttling RCIC injection to prevent a high reactor water level trip shutting down RCIC. Figure 3-49 illustrates the RPV water level for Case 11.

Operators initiate controlled depressurization of the RPV after 1 hour. Pressure is maintained above the operating pressure of the RCIC turbine as operators cycle available SRVs to maintain RPV pressure within the range of 200-400 psig. SRV operations and RCIC turbine exhaust results in an increase of the suppression pool temperature. At 8.4 hours, RCIC fails as a result of suppression pool water temperature exceeding 230ºF. The RPV is depressurized by operator opening an SRV following loss of RCIC for Case 11.

Without RCIC injection, boil-off of the RPV water level commences. The water level falls to the top of active fuel at 11.0 hours and below the bottom of active fuel at 13.3 hours. The thermal response of intact cladding in the active fuel region is illustrated in Figure 3-50 and Figure 3-51, which show the calculated temperature of the fuel cladding in the inner core ring and across the core mid-plane, respectively. As temperatures increase in the uncovered regions of the fuel, exothermic oxidation of the Zircaloy cladding initiates. The integral mass of hydrogen generated in-vessel is displayed in Figure 3-51.

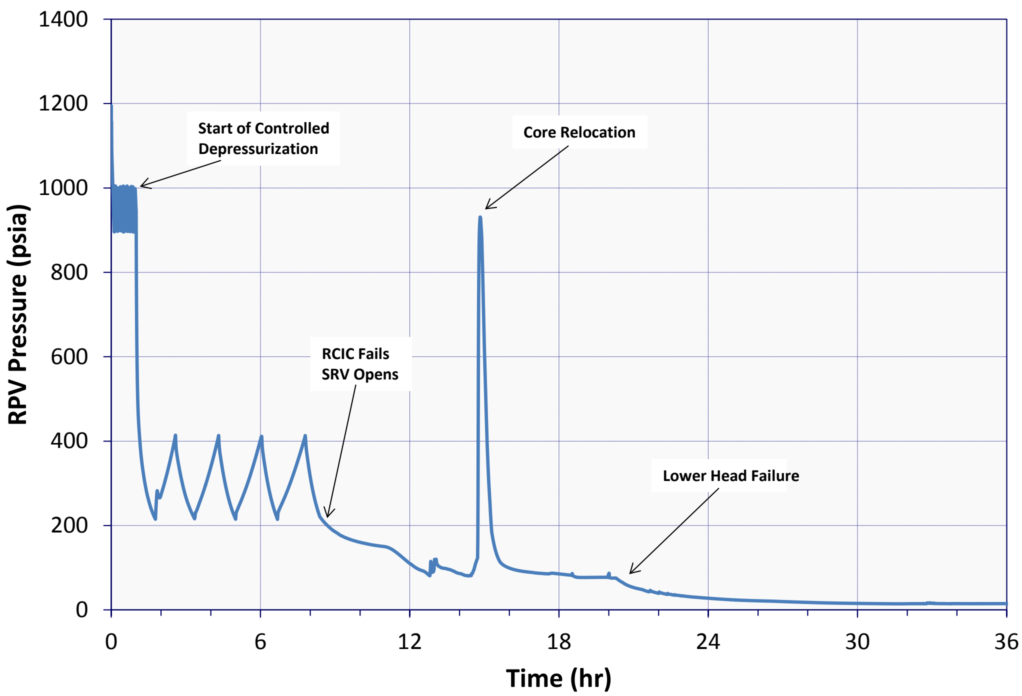


Figure 3-48 Mark II RPV pressure for Case 11

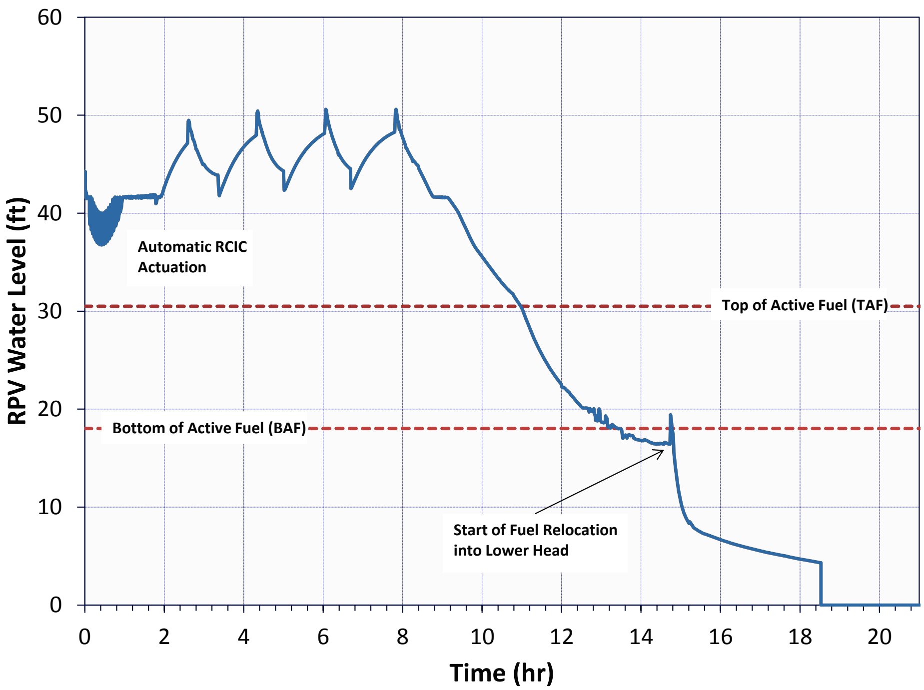


Figure 3-49 Mark II RPV water level for Case 11

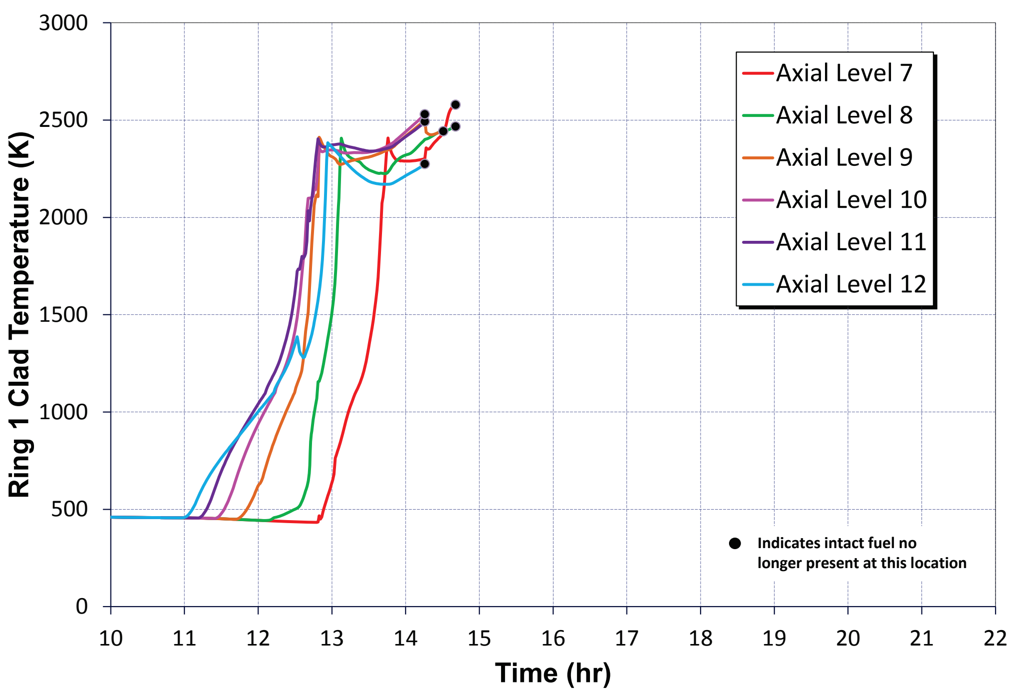


Figure 3-50 Mark II fuel cladding temperature in Ring 1 for Case 11

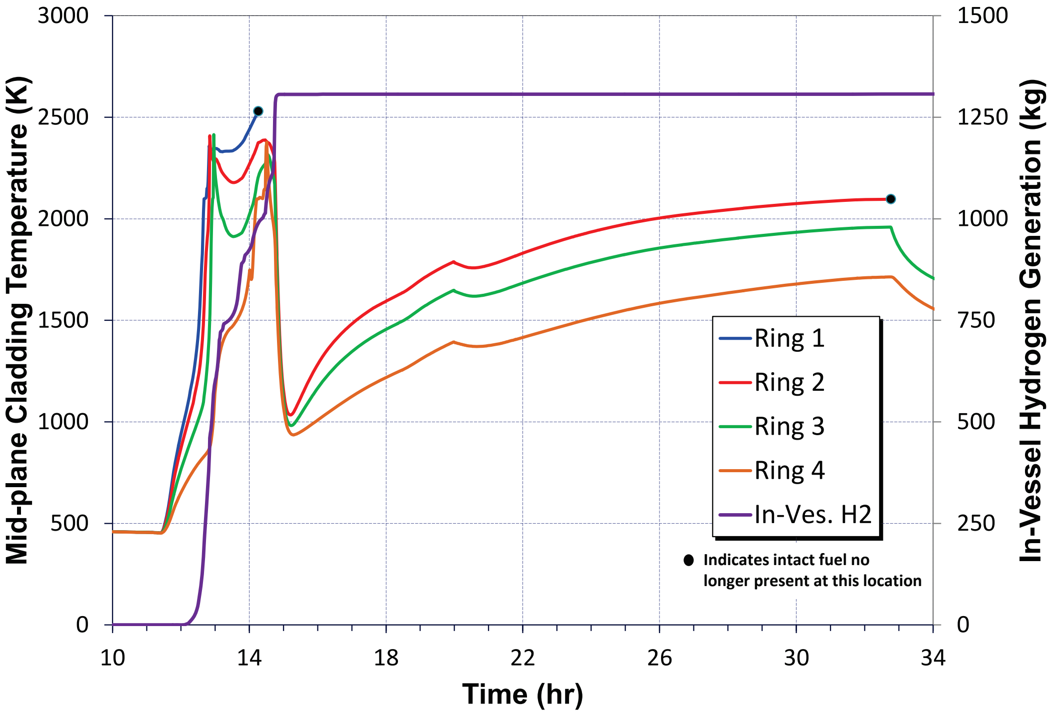


Figure 3-51 Mark II fuel cladding temperatures at core mid-plane for Case 11

As the core degrades, debris moves downward through available free volume. Large relocation initiates in the bypass region as control material fails early on. While intact fuel is modeled to restrict downward propagation of fuel debris due to the limited available volume, canister wall failures permit radial relocation of fuel debris into the bypass region. Radial relocation of fuel debris back into the channel region is permissible as the canister wall fails in lower regions of the core.

As the core degradation continues, debris begins to accumulate onto the core plate. While in contact, conduction between the debris and the core plate commences and the core plate temperature begins to increase. The increase in temperature of the core plate along with the physical loading of the debris eventually cause the plate to yield locally, allowing debris to relocate into the lower plenum. Rapid quenching of the debris is imposed as debris enters the pool of water in the lower head. Interaction between the debris, coolant, and lower head determines the eventual boil-off of the remaining coolant and thermal response of the lower head. Ultimately, failure of the lower head occurs at 20.0 hours from strain. Various stages of core degradation are presented in Figure 3-52 to illustrate the core degradation progression.

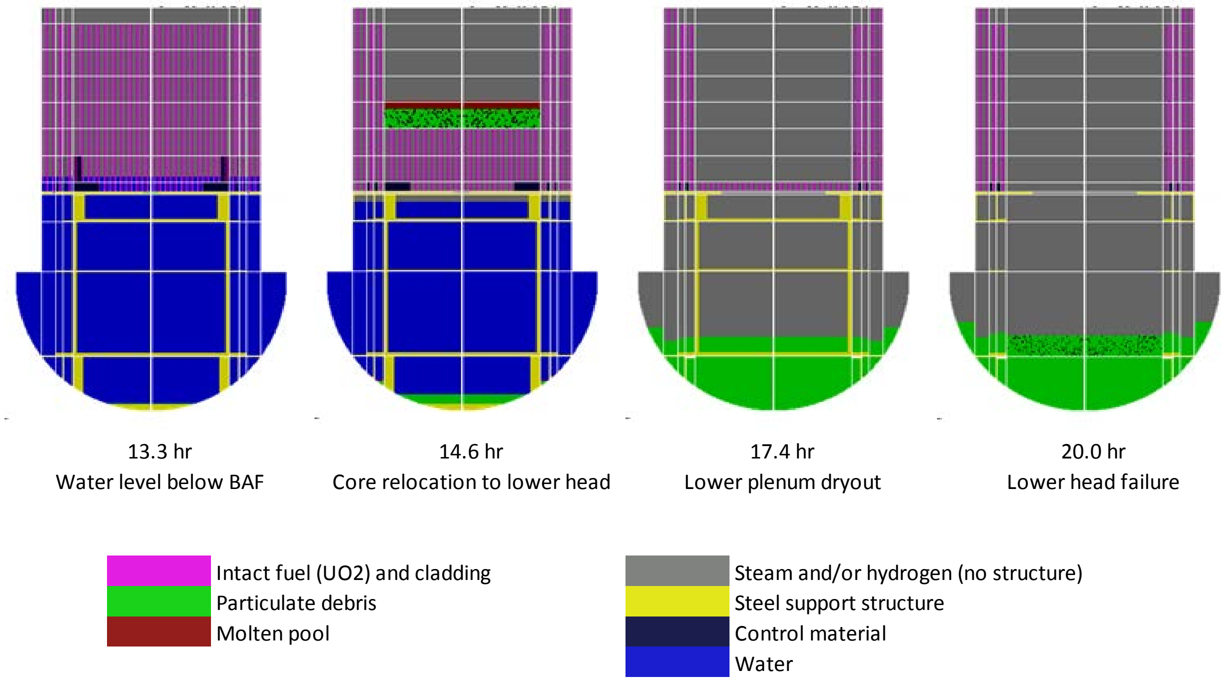


Figure 3-52 Mark II core degradation and relocation for Case 11

Upon lower head failure, approximately 220 metric tons of corium are ejected to the upper reactor cavity within the reactor pedestal. Corium concrete interaction commences as the relocated debris within the upper reactor cavity begins to decompose the concrete. Sump drain lines penetrating the concrete structure between the upper and lower reactor cavity are assumed to fail 20 minutes after debris relocation to the upper reactor cavity. At this time, the sump drain lines are modeled in the analysis as fully open, thus allowing debris to relocate to the lower reactor cavity along with water. Concrete ablation continues in the lower reactor cavity throughout the remainder of the accident sequence, as shown in Figure 3-53.

The containment pressure response throughout the transient can be seen in Figure 3-54. The operation of the RCIC system and actuation of the SRVs result in a slow pressurization of the containment prior to suppression pool saturation (see Figure 3-55). Once the pool becomes saturated, the containment pressure rapidly increases due to steam released from the vessel and the subsequent vaporization of wetwell pool water. Since the containment pressure does not reach 15 psig prior to RCIC failure, venting is not performed. However, the post core damage venting initiates at 20.3 hours as the containment pressure exceeds the prescribed PCPL of 60 psig. The containment pressure excursion during core damage fails to achieve 60 psig.

Vessel leakage through the reactor circulation pumps and to a lesser extent steam condensed and drained from the drywell has over time increases the water level in the upper reactor cavity, as seen in Figure 3-56. At the time of lower head failure, debris comes into contact with the upper reactor cavity water and sufficient containment pressure results in actuation of the wetwell vent. The integral vent flow is provided in Figure 3-54.

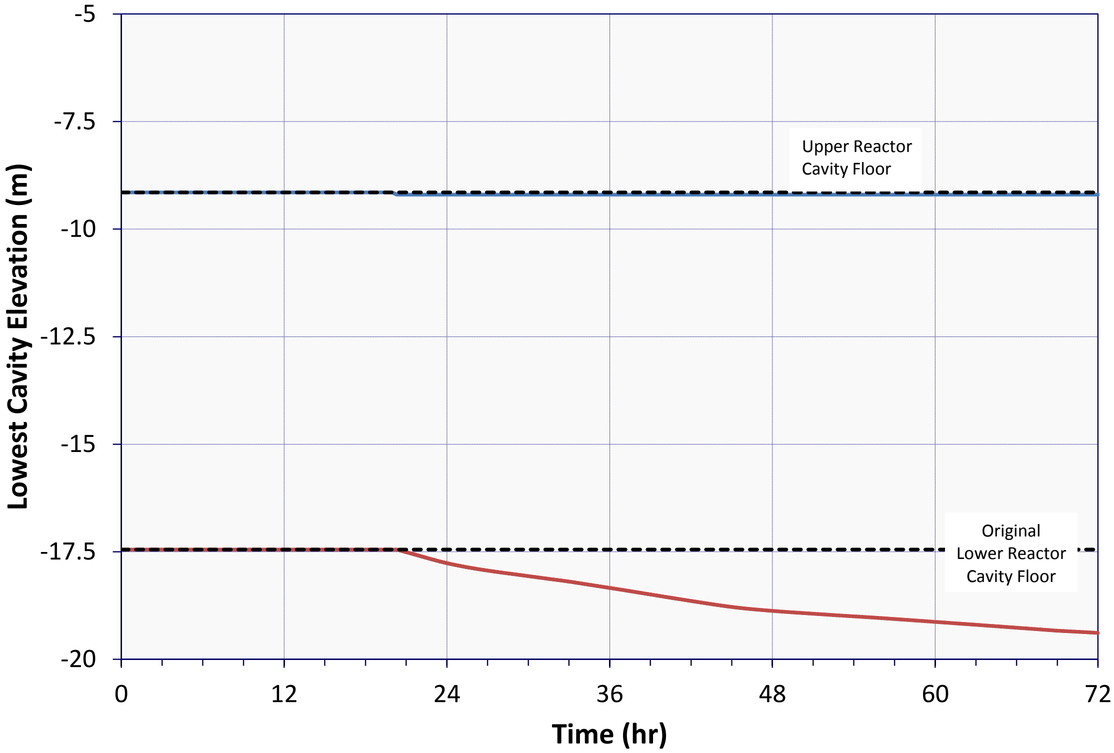


Figure 3-53 Mark II concrete ablation depth due to MCCI for Case 11

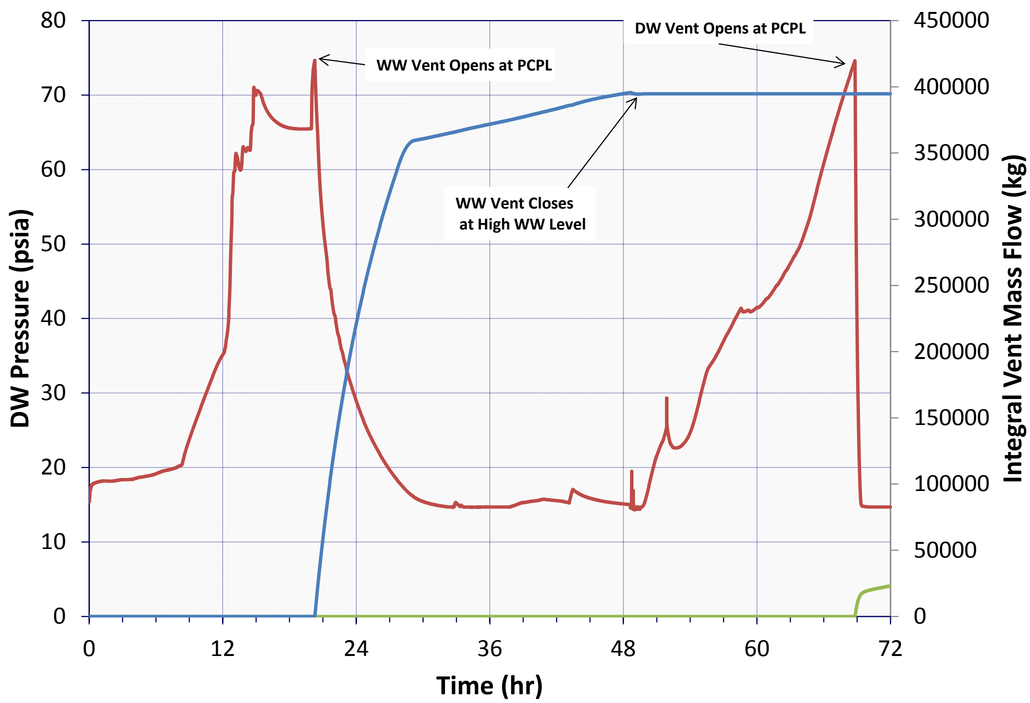


Figure 3-54 Mark II containment pressure and vent flow for Case 11

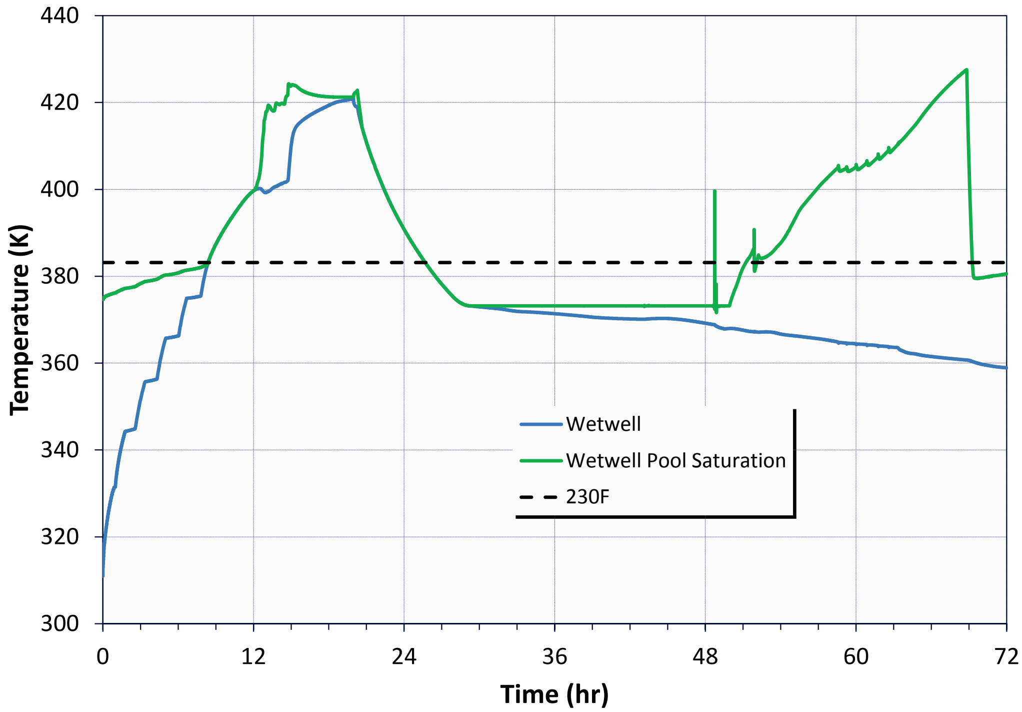


Figure 3-55 Mark II suppression pool temperature for Case 11

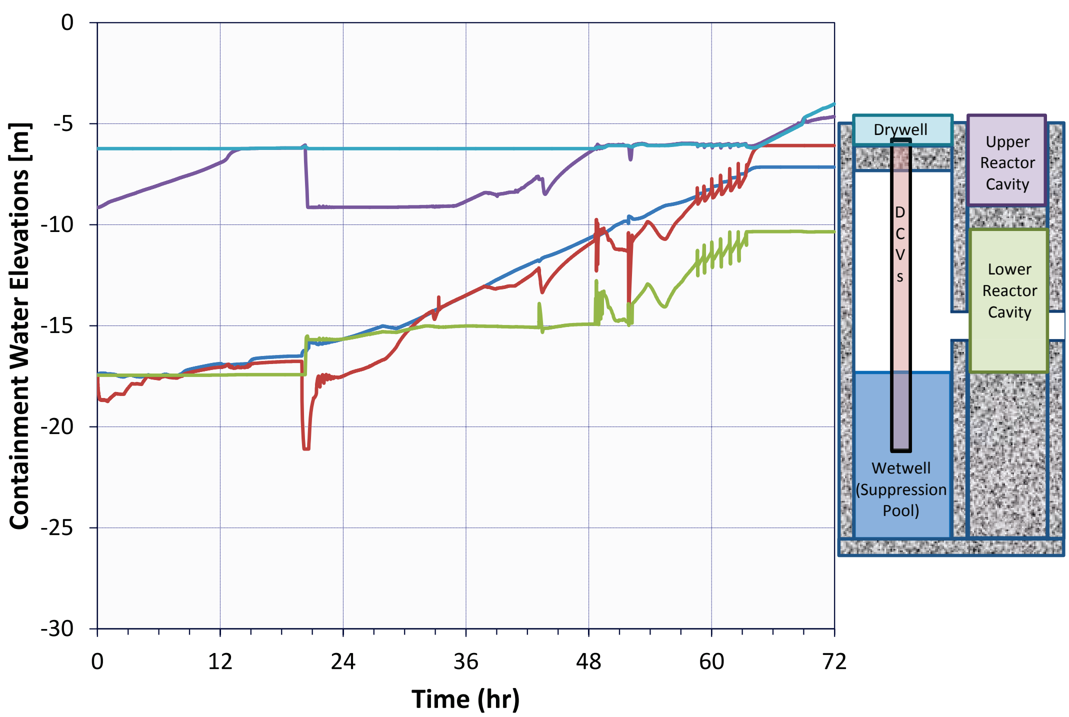


Figure 3-56 Mark II containment water levels for Case 11

RPV FLEX injection initiates at the time of lower head failure and water is permitted to drain from the vessel through the lower head failure into the upper reactor cavity. Once the sump drain lines fail, water is able to freely flow into the lower reactor cavity and enter the wetwell. With continuous FLEX injection of 500 gpm, the suppression pool level eventually exceeds the assumed high water level and the wetwell vent is isolated. Containment pressure is permitted to pressurize until 60 psig is achieved at 68.8 hours, at which time the drywell vent line is opened.

Containment atmosphere temperature and pressure remain sufficiently low to prevent drywell head failure (see Figure 3-57). Ex-vessel gas generation and heat transfer from the debris cause temperatures in the containment to rise while FLEX injection prevents significant temperature increase.

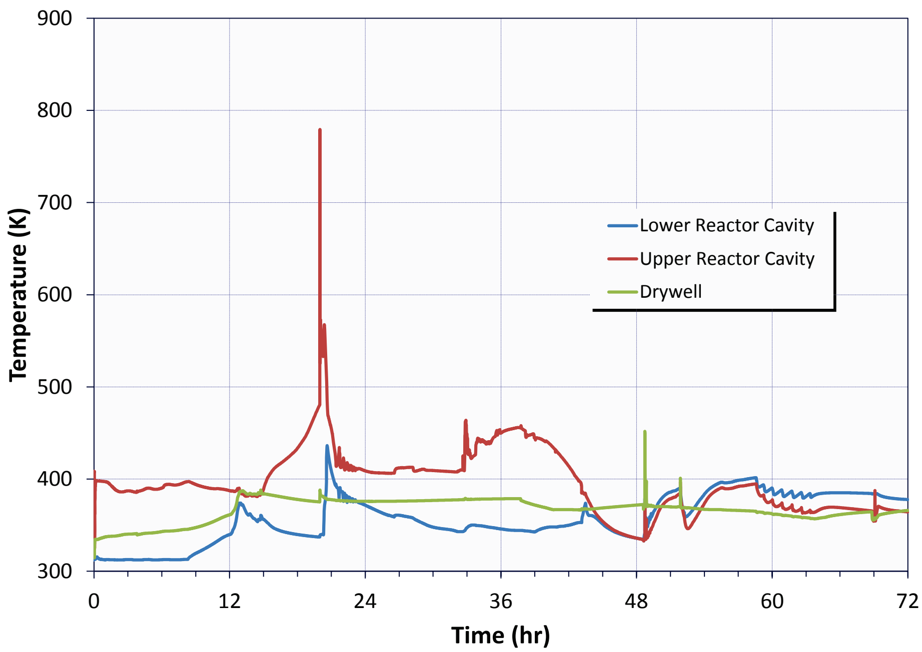


Figure 3-57 Mark II containment atmosphere temperatures for Case 11

Due to the temperature transient and fuel failure, fission products are eventually released from the core and transported throughout the primary containment until vent operations permit release to the environment. The onset of fission product release occurs once fuel clad gap fails at 12.2 hours. This timing coincides well with the onset of vigorous oxidation of the cladding. As fuel temperatures rapidly increase, fission product diffusion enhances and volatile fission products are released from the fuel. Cesium and iodine, which are significant contributors to public risk, are distributed as seen in Figure 3-58 and Figure 3-59.

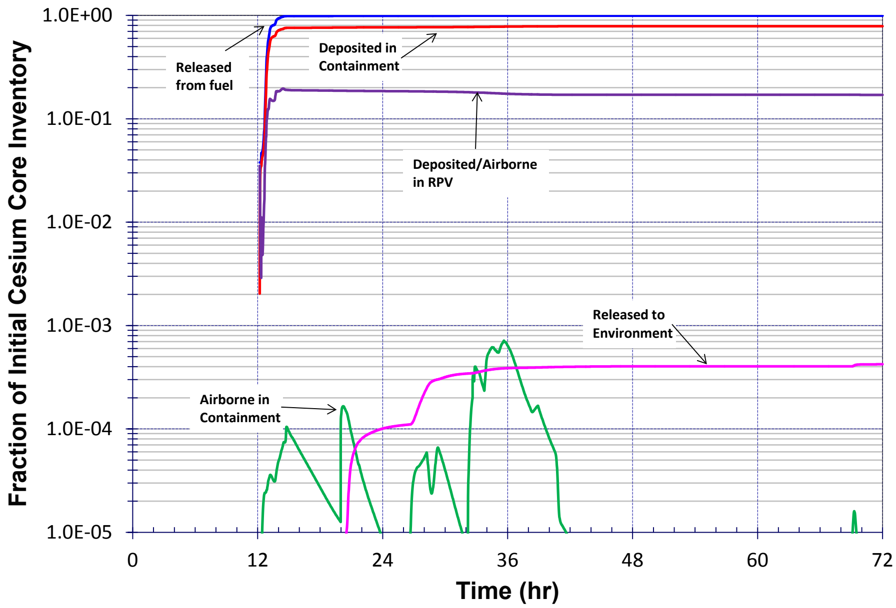


Figure 3-58 Mark II cesium fission product distribution for Case 11

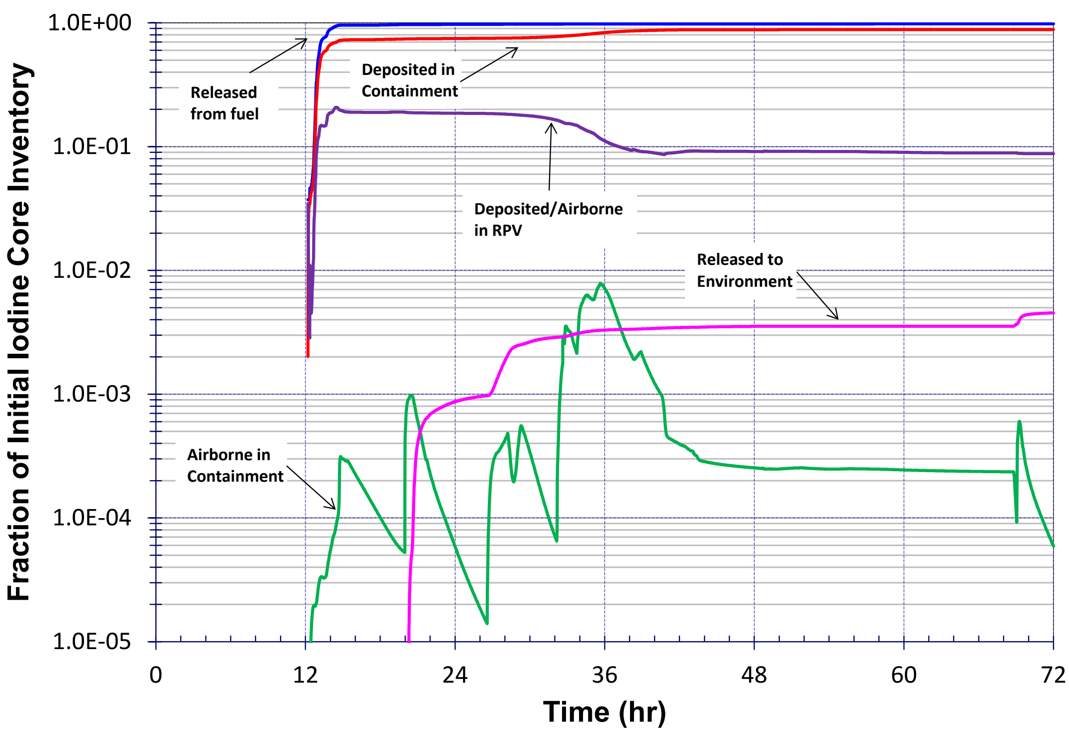


Figure 3-59 Mark II iodine fission product distribution for Case 11

Oxidation results in a rapid temperature increase such that the release of cesium and iodine from the fuel is nearly complete within 2.5 hours of the initial gap release; more than 95% of the original inventory is predicted to be released from the fuel at this time. Released fission products that are not deposited within the RCS enter containment primarily through the open SRV. Fission product

scrubbing occurs as steam is passed through the suppression pool. Uncaptured fission products are transported through the wetwell/drywell vacuum breakers to the drywell vapor space, where suspended aerosols continue to deposit over time. Airborne fission products at the time of vent operation are available for environment release.

Once the lower head has failed, fission products are transported from the reactor vessel to containment through the lower head failure. Pool scrubbing is performed, albeit less efficiently, through the downcomer vents while differential pressures between the drywell and wetwell are sufficient to clear the downcomer vents. The differential pressure is a product of the MCCI gas generation and the open wetwell vent line. However, with the assumed failure of the in-pedestal drain lines after 20 minutes, suppression pool bypass occurs and pool scrubbing through the downcomer vents terminates now that flow through the upper/lower reactor cavity is possible.

After suppression pool bypass has occurred, the late in-vessel release, characterized by the revaporization of RCS deposits, is the dominant contributor to the environmental release. For Case 11, the operator action to open an SRV at the time of RCIC failure provided a continuous release path from the RCS to the containment throughout core degradation, permitting relatively large amounts of fission products to be released to the containmaint. By enhancing fission product release to the containment, the fission product inventory deposited within the RCS is reduced. This limits the available fission product release from the deposited fission product in the RCS by suppressing the total deposited mass available for release, and similarly, the available decay heat to promote revaporization. Therefore, for Case 11, suppression pool bypass is not observed to produce a significant impact to the environmental releases. The initial vent operation and resulting sudden release are followed by revaporization of deposited cesium-iodide from the RCS where vapor pressures are sufficient. Iodine is subjected to greater revaporization from the RCS than is observed for cesium, as is discussed in Section 3.3.1.1 . A slow liberation of total iodine, and to a lesser extent cesium, is observed in RPV inventory seen in Figure 3-59 and corresponding environmental release.

Suppression pool water, due to the continual RPV FLEX injection of 500 gpm, eventually exceeds the assumed upper instrumentation level range and the wetwell vent line is closed at 49.9 hours, and the containment pressure is permitted to repressurize to 60 psig. At 68.8 hours environmental releases continue for the remainder of the analysis with the actuation of the drywell vent line.

3.3.2.2 Accident Progression without Water Addition

In contrast to the Mark II analysis with water addition, provided in Section 3.3.2.1 , Case 1 is presented to demonstrate system response in the absence of FLEX injection. Additionally, RPV depressurization after RCIC failure is not performed. Table 3-3 lists the specifics of the sequence employed for Case 1.

RPV pressure and water level responses are presented for Case 1 in Figure 3-60 and Figure 3-61, respectively. As anticipated, the pressure response remains identical to that observed in Case 11 until 8.4 hours when RCIC fails. In Case 1, pressure control, initiated at 1 hour, continues and the RPV pressure is maintained between 200-400 psig. Without RCIC maintaining reactor water level, oxidation of zirconium increases fuel and vapor temperatures. Transitioning from intact components to particulate debris, fuel and in-core structures relocate downward, settling upon the core plate. As heat is conducted from the debris, the core plate temperature rises and the plate fails at 15 hours. Debris enters the pool in the lower head and the resulting steam generation and pressure event are clearly seen in Figure 3-60 even though an SRV is open. As the core degradation continues and vapor temperatures further increase, the internal components

of the operating SRV achieve temperatures in excess of 900K. Loss of mechanical strength due to excessive temperatures is assumed to fail the operating SRV in the open position, which permits the RPV to depressurize at 16.8 hours.

Core degradation, unlike that observed in Case 11, continues after lower head failure as degraded fuel in core ring two becomes particulate debris and relocates to upper reactor cavity. This process increases the observed cumulative mass ejected from the vessel from the 220 metric tons observed in Case 11 to 248 metric tons observed in Case 1. Figure 3-62 outlines various core degradation states throughout the transient.

Initial gap release and hydrogen generation occur at approximately 13 hours. SRV actuations allow airborne fission products to be released to the suppression pool until the lower head fails, whereby fission products are transported directly from the vessel to the containment. Downcomer vent scrubbing halts once the upper reactor cavity sump drain lines fail. Fission product distributions for cesium and iodine are provided in Figure 3-63 and Figure 3-64, respectively. Late in-vessel release significantly contributes to the environment releases as a larger fraction of fission products were observed to deposit in the RCS in comparison to those in Case 11. Unlike Case 11, suppression pool bypass timing impacts a greater fraction of the total fission product inventory released to the environment. Release fractions of total cesium and iodine inventories to the environment are calculated as 2.46E-2 and 1.98E-1, respectively.

During the transient, the containment pressure is observed to slowly rise while the suppression pool has yet to saturate. SRV operations and hydrogen generation produce rapid pressurization of the containment system once saturation of the suppression pool occurs. As seen in Figure 3-65, the drywell pressure reaches 60 psig at 22.8 hours and the wetwell vent is actuated. Following lower head failure and the expulsion of debris, boiloff of the pool water accumulated in the upper reactor cavity as well as the progression of MCCI increases local temperatures within the upper reactor cavity, and eventually the lower reactor cavity following sump drain line failure. Pool water and debris are then transferred to the lower reactor cavity; where without the addition of FLEX injection, the containment temperatures observed in Figure 3-66 steadily increase. The debris bed becomes uncovered at 26.0 hrs, as shown in Figure 3-67. The wetwell vent line remains open for the remainder of the analysis.

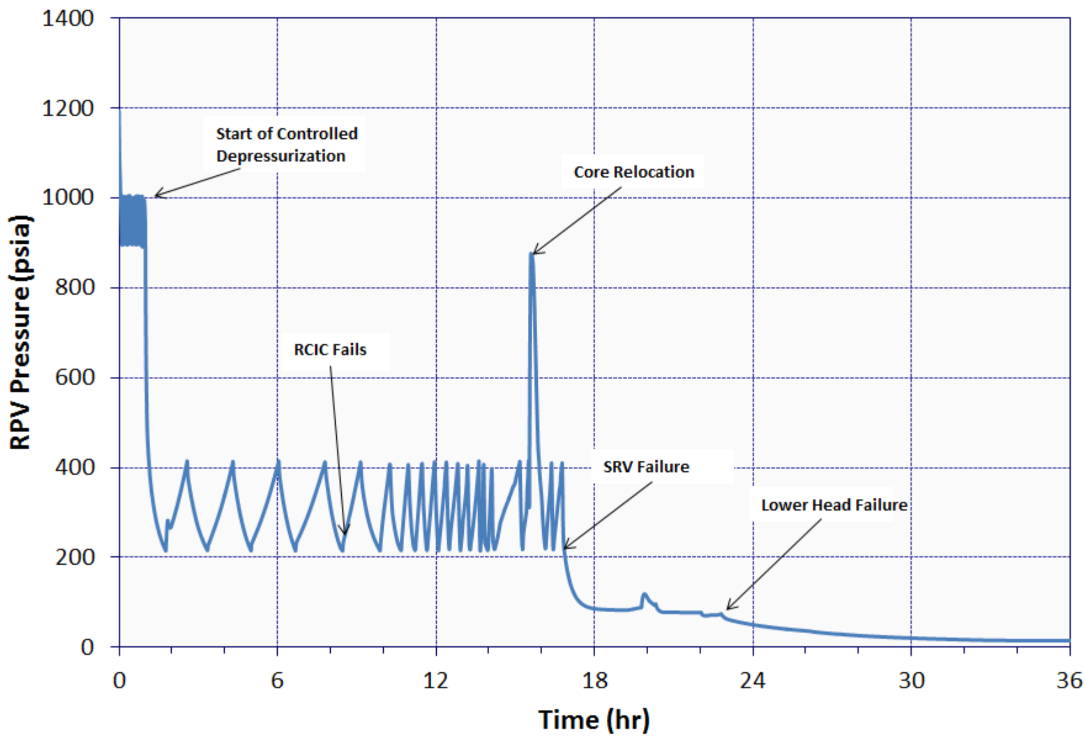


Figure 3-60 Mark II RPV pressure for Case 1

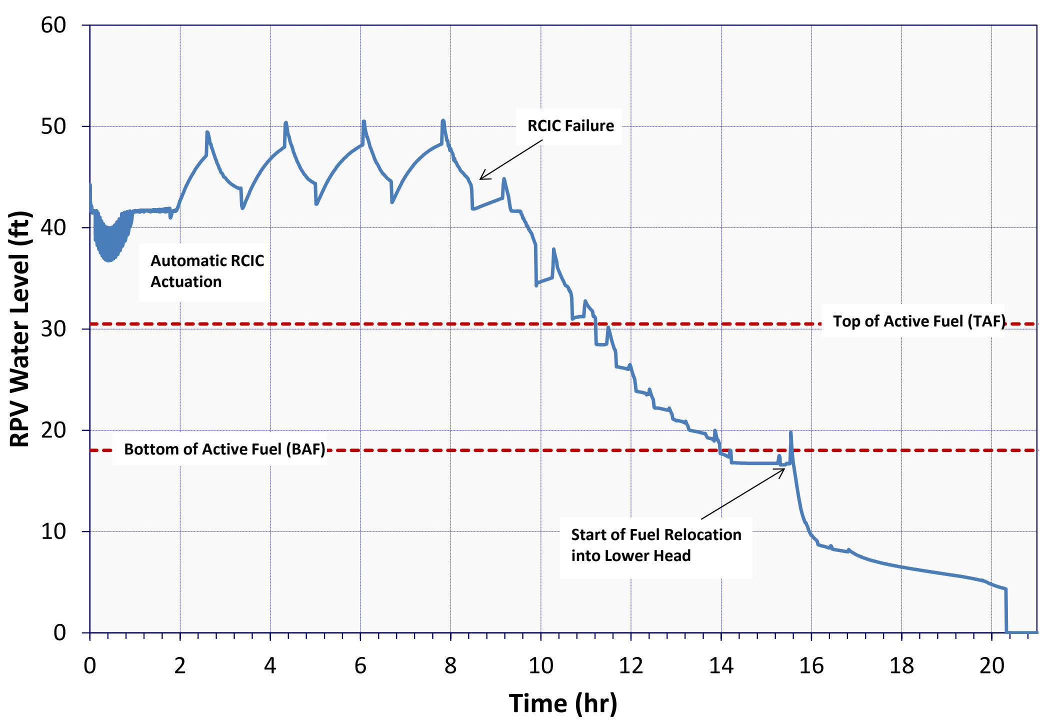


Figure 3-61 Mark II RPV water level for Case 1

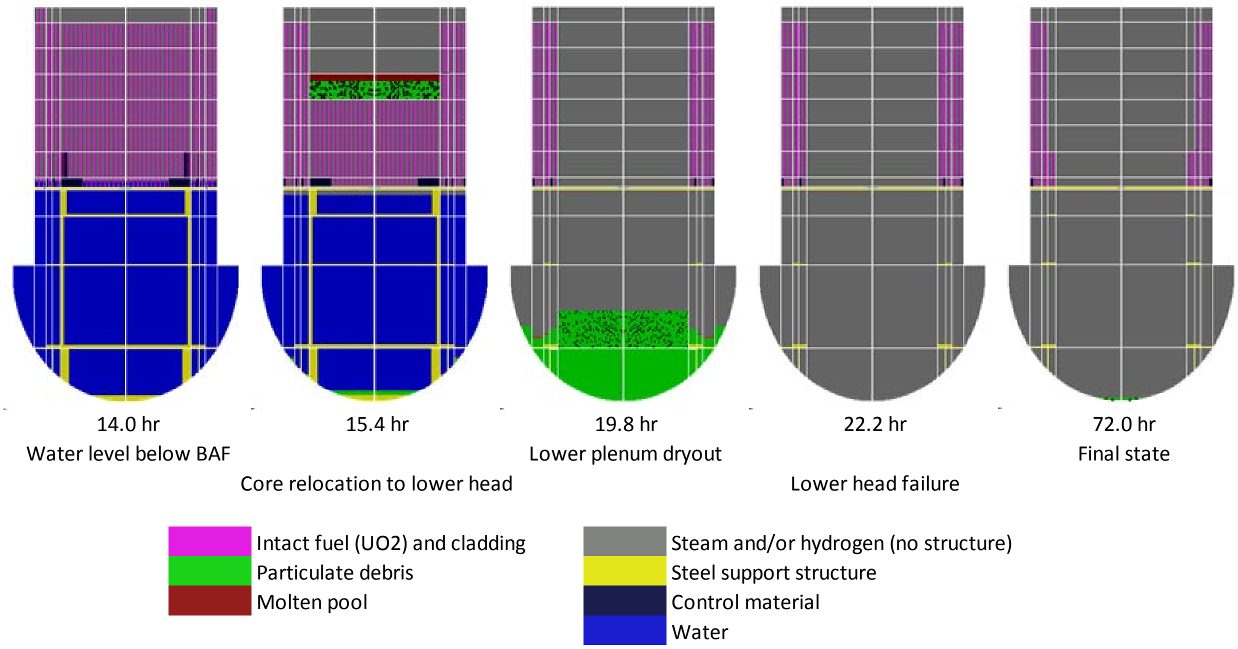


Figure 3-62 Mark II core degradation and relocation for Case 1

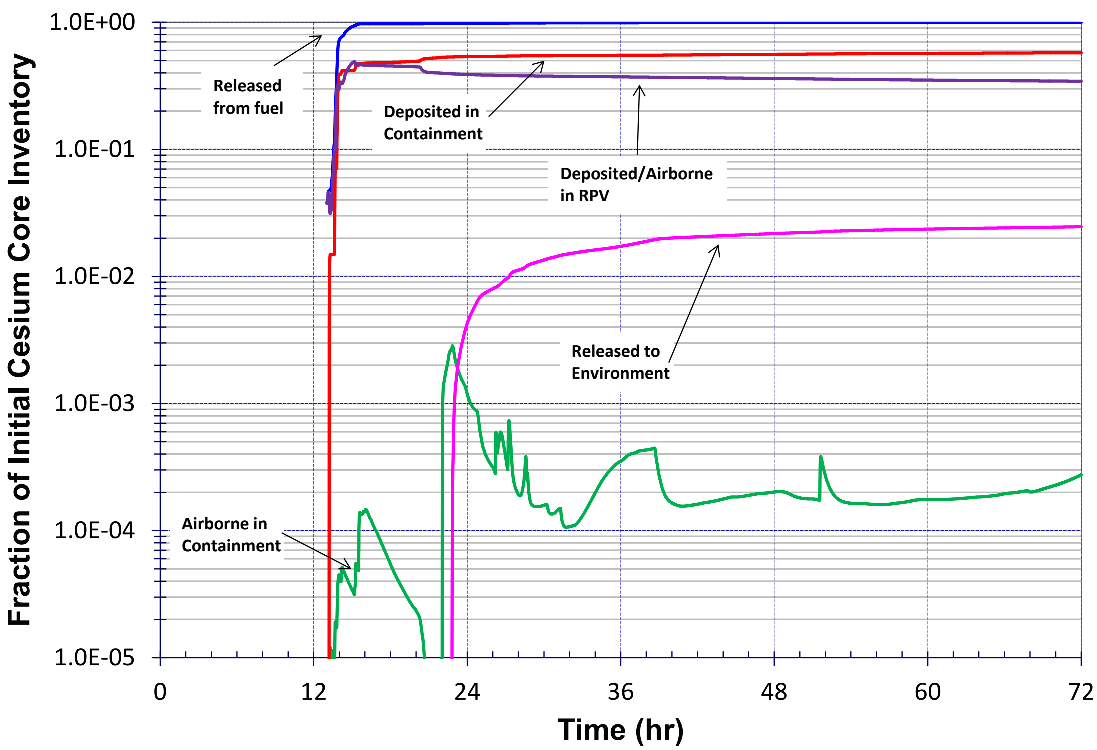


Figure 3-63 Mark II cesium fission product distribution for Case 1

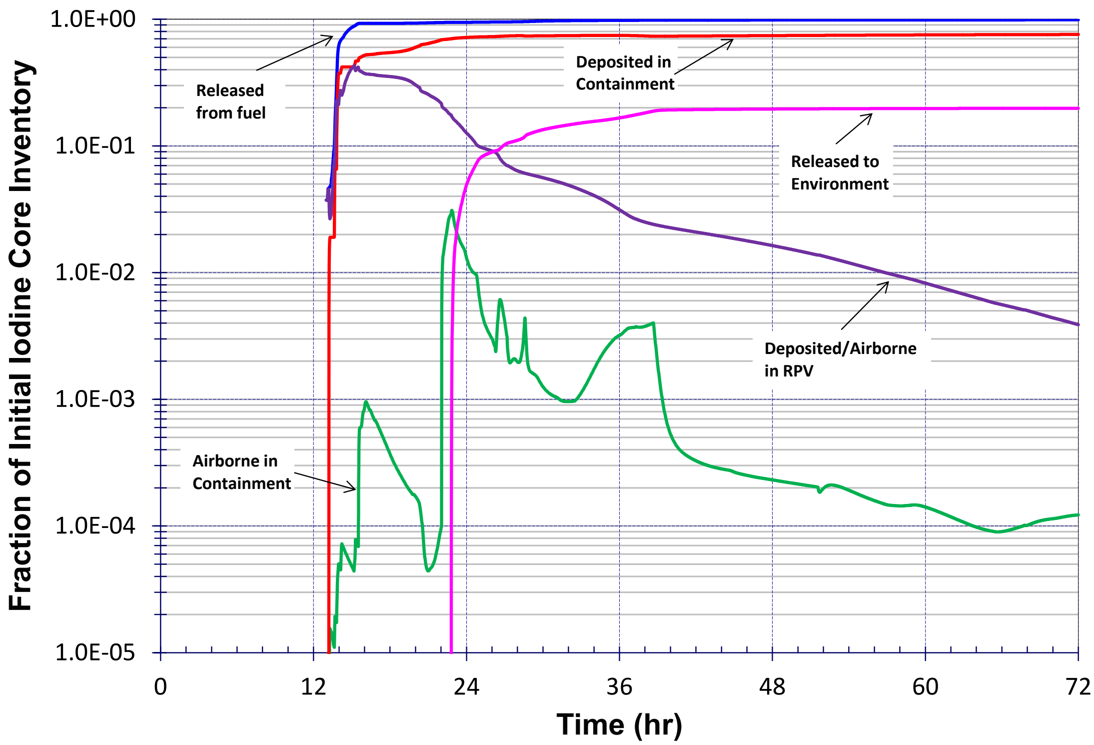


Figure 3-64 Mark II iodine fission product distribution for Case 1

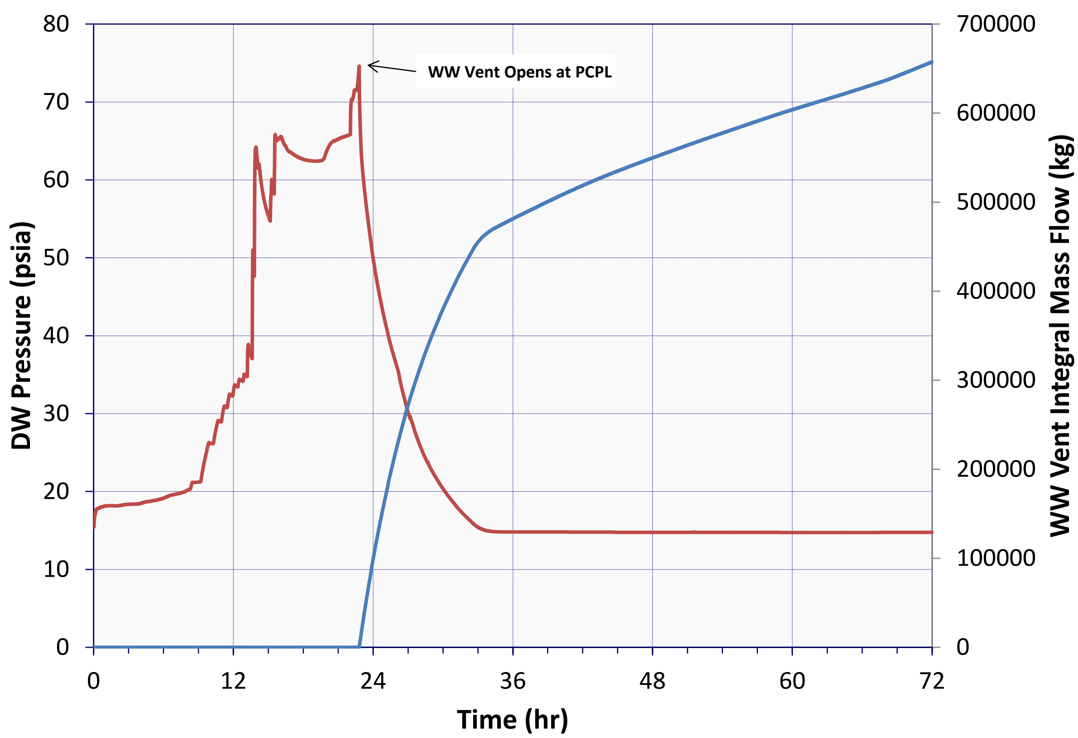


Figure 3-65 Mark II containment pressure and vent flow for Case 1

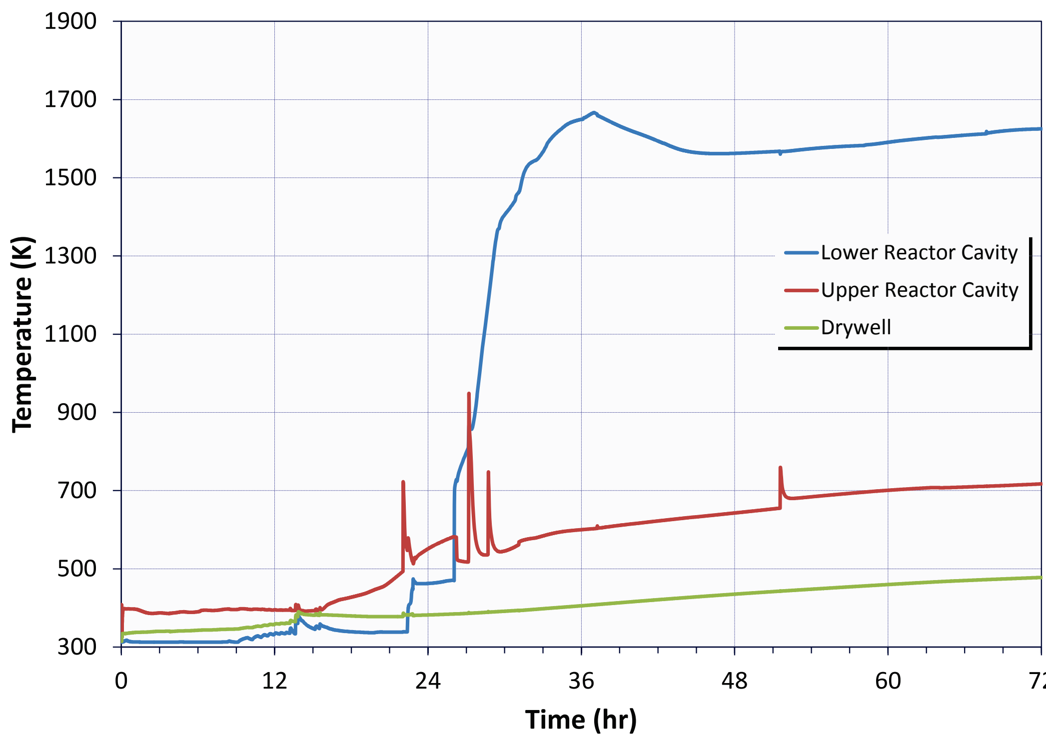


Figure 3-66 Mark II containment atmosphere temperatures for Case 1

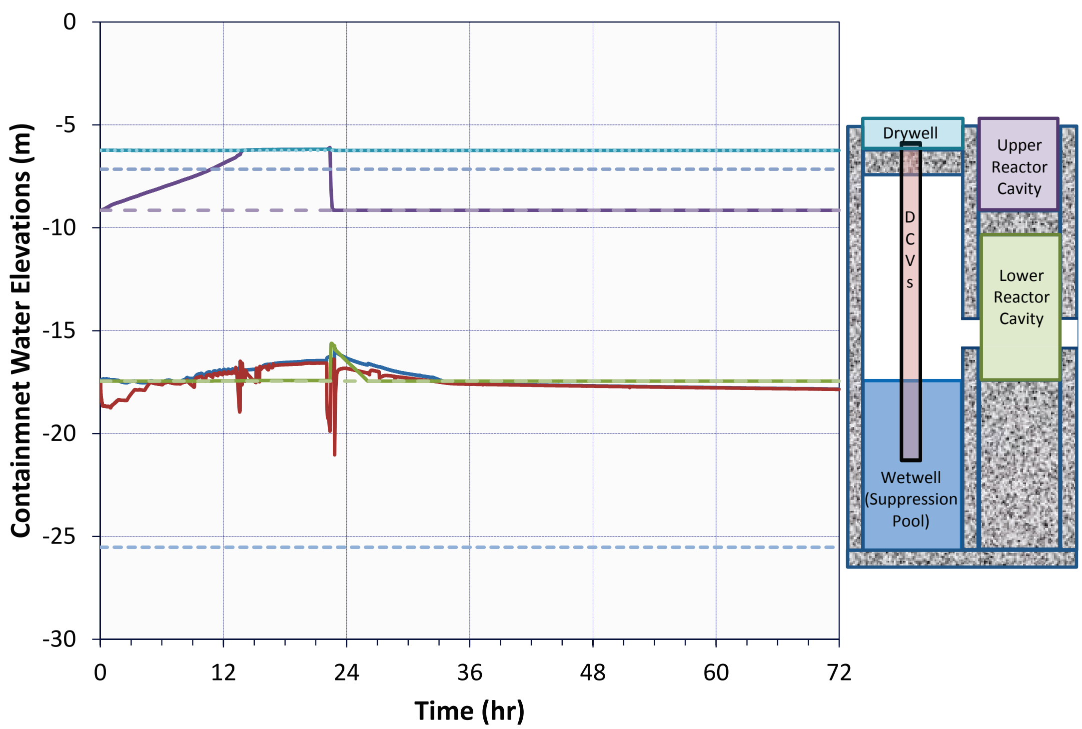


Figure 3-67 Mark II containment water levels for Case 1

3.3.2.3 Source Term and Containment Response

The U.S. Mark II fleet employs several cavity designs. Should core degradation result in lower head failure, containment response and ultimately fission product release could vary among these designs. Regardless of containment design, a reasonable assumption is fission product transport prior to lower head failure would undergo similar decontamination within containment; therefore, differences are anticipated to occur after lower head failure among the various containment configurations, see Figure 3-68.

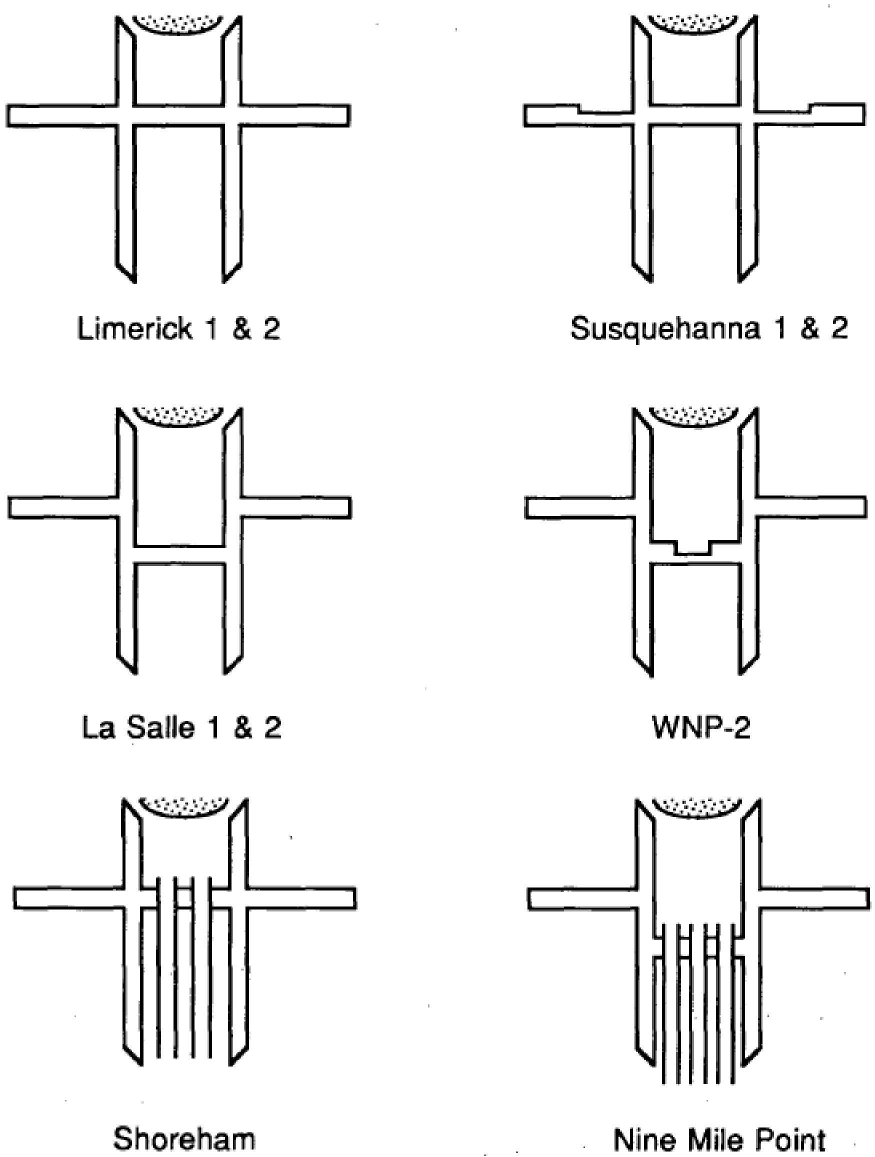


Figure 3-68 Mark II containment configurations

The upper reactor cavity floor, drywell floor, downcomer vents (see for example, Figure 3-67), and drain lines (not shown) are the components which comprise the drywell pressure boundary separating the atmospheres of the drywell and wetwell. Debris exiting the vessel accumulates within the upper reactor cavity and interacts with the upper reactor cavity floor and penetrations

within, which compromises the integrity of the pressure boundary. Should the pressure boundary between the drywell and wetwell fail, atmospheric material can be freely exchanged between the drywell and wetwell atmospheres. The loss of the pressure boundary prevents the downcomer vents from passing drywell atmosphere through the suppression pool; therefore, vapor condensation from drywell atmosphere bubbled through the pool and suppression pool scrubbing are lost. This event is commonly referred to as suppression pool bypass.

Three distinct fission product transport phases are characterized in Figure 3-69 to emphasize the importance of suppression pool bypass. The first phase occurs when the RCS pressure boundary is maintained and the predominant fission product releases from the vessel occur through open SRVs. The second phase initiates once lower head failure has occurred and debris is transported to the upper reactor cavity. Fission products released from RCS enter the drywell atmosphere predominantly through the lower head failure. In additional, MCCI generated aerosols and gases enter the drywell atmosphere as well. The pressure differential between the drywell and wetwell results in the downcomer vents clearing and atmospheric material is transmitted to the suppression pool. The third phase follows suppression pool bypass, where flow between the atmospheres of the wetwell and drywell becomes prevalent. Aerosol and vapors released from submerged debris do undergo scrubbing; however, volatile fission products, such as cesium and iodine, have predominantly been released from the fuel prior to lower head failure.

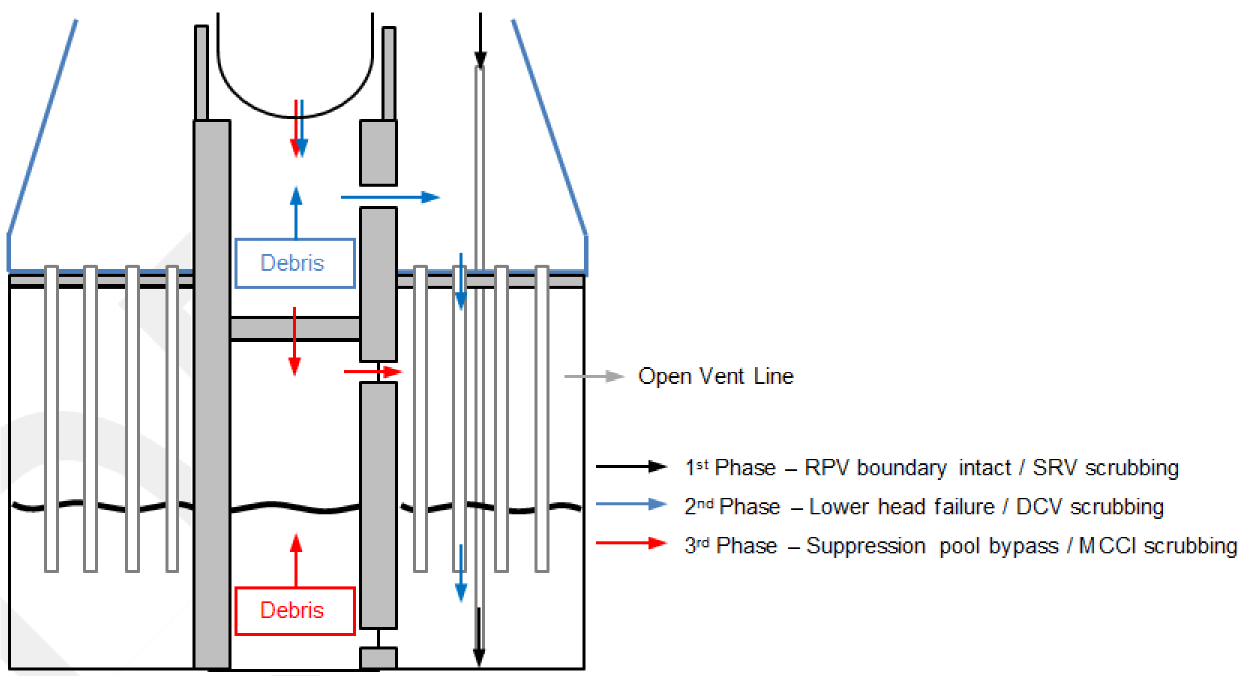


Figure 3-69 Mark II fission product transport paths during accident phases

The analyses presented assume the timing of suppression pool bypass is the most significant contribution to containment decontamination. Rather than attempt to address each containment configuration, suppression pool bypass timing was modified as a surrogate for direct simulation of each containment configuration. This was performed under the limitation of available Mark II models for severe accident analysis. Furthermore, the in-pedestal region beneath the upper reactor cavity is flooded in all other configurations. To address this difference, analyses with and without lower reactor cavity flooding are presented in the sensitivities discussed below. The sensitivities presented use Case 1 due to the large fission product masses deposited in the RCS

as compared to Case 11. Suppression pool bypass and cavity configurations will influence the environmental release during the late in-vessel release phase of the accident.

The first sensitivity presented increases the time to suppression pool bypass for the default Mark II model. In the original model, suppression pool bypass is assumed to occur 20 minutes after debris enters the upper reactor cavity as a result of in-pedestal sump drains failing due to debris contact. This failure criterion was suspended and debris was permitted to remain in the upper reactor cavity until the ablation depth due to MCCI exceeded the total thickness of the cavity floor. This case is presented as Case 1 with URC MINALT (upper reactor cavity ablation depth reaching minimum altitude of the concrete).

A set of sensitivities is presented that employ a modified Mark II model to investigate a flooded lower reactor cavity. The lower reactor cavity of the representative Mark II model is a partially filled, dry volume. This cavity volume was increased by extending the volume downward until the base altitude of the lower reactor cavity agreed with the bottom of the wetwell volume. The lower reactor cavity volume was combined with wetwell control volume to create a contiguous representation of the suppression pool, a single, well-mixed pool region defining the in-pedestal and suppression pool water. The original suppression pool water level in the Mark II representative model was maintained. This configuration is presented in Figure 3-69 (see Figure 3-9 for comparison with the original model). The three sensitivities performed with the modified containment model include the following:

- Case 1 LRC MINALT – Case 1 was performed with the minimum ablation rule enabled.
- Case 1 LRC 20 min delay – Case 1 was performed with the default 20-minute delay after debris enters the upper reactor cavity before suppression pool bypass.
- Case 1 LRC 0 min delay – Case 1 was performed; however, the debris relocating from the reactor was passed directly to the suppression pool/lower reactor cavity, producing an instant suppression pool bypass at the time of lower head failure.

The environmental releases of cesium and iodine are presented in Figure 3-70 and Figure 3-71 for the default Mark II model and the modified Mark II model with the flooded lower reactor cavity, respectively. Environment release initiates at the time of wetwell venting in the analyses performed. In the case of the base analysis (Case 1) and the Case 1 sensitivity (URC MINALT), significant reductions in the release of cesium and iodine result. The increased duration of downcomer vent scrubbing through prolonging the bypass of the suppression pool, a significant reduction in the released cesium and iodine are realized.

The sensitivities performed with the Mark II model with the flooded lower in-pedestal reactor cavity modification produce similar reductions in the environment release of cesium and iodine. These analyses deviate from the base case as the vent line operation, initiating at PCPL, was reached prior to lower head failure[10]. Regardless of chronology of vent line operation and lower head failure, a significant reduction in the overall release of cesium and iodine is released by extending the suppression pool bypass timing.

The timing of suppression pool bypass is considered uncertain and assumed to be the largest significant variation among the different containment configurations. While no qualitative discussion is presented regarding the variation of suppression pool bypass timing for each design,

[10] The modified Mark II model increases the total suppression pool inventory, which prolongs RCIC operations. These differences ultimately permit the first significant pressurization event to exceed the PCPL. The Mark I and the modified Mark II model pressure responses are similar (see the Figure 3-16 for the Mark I containment pressure response).

early suppression pool bypass and a protracted suppression pool bypass was performed for the Mark II and a modified Mark II model. Reductions in total material released were within an order of magnitude for the sensitivities performed. In comparison, the range of results observed for the Case 1 sensitivities is presented along with the remaining Mark II analyses in Figure 3-77. The releases are comparable to those observed among the scenarios investigated and are therefore considered no more significant than the variations performed. If these sensitivities showed more significance, additional investigations would be justified.

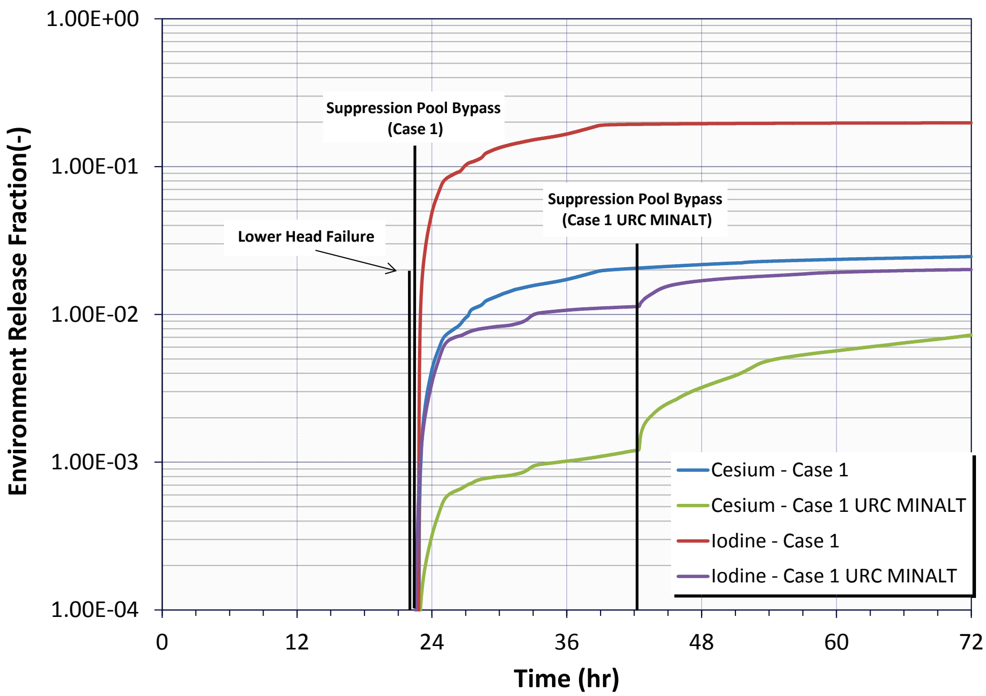


Figure 3-70 Mark II environment release sensitivity to suppression pool bypass timing

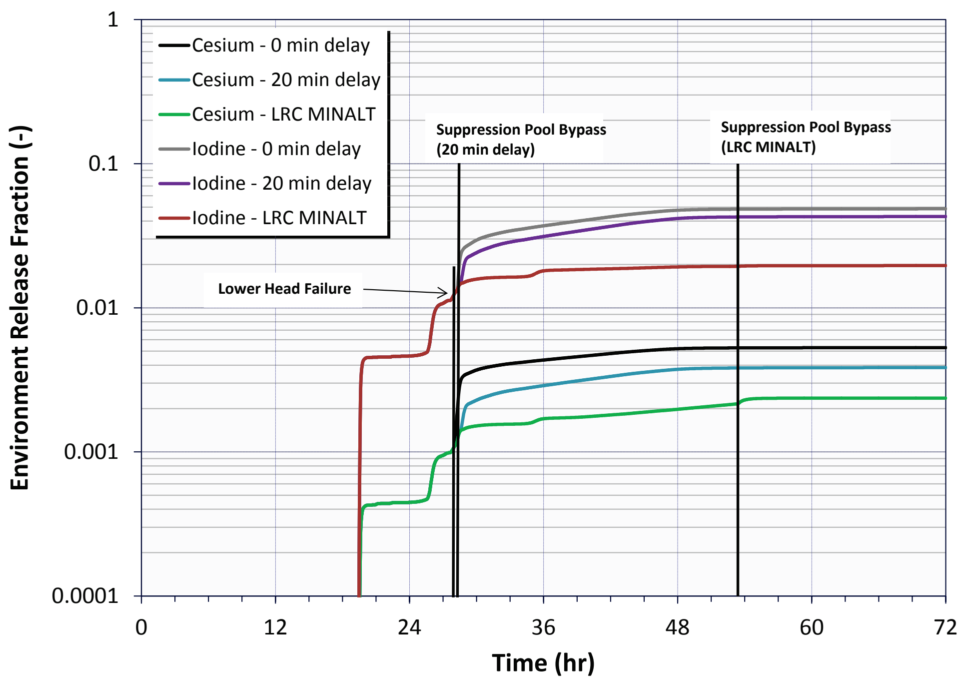


Figure 3-71 Mark II with flooded lower reactor cavity sensitivity to suppression pool bypass timing

The hydrogen discussion presented below incorporates Case 11p1 from Table 3-3 (the PCPL was corrected from 60 psig to 45 psig). The wetwell vent line is initialized with noncondensible mole fractions representative of standard air (79% N_2 and 21% O_2). For the Case 11p1 transient, approximately 1300 kg of hydrogen is produced in-vessel while the total mass of hydrogen generation is calculated as approximately 2700 kg. Figure 3-72 shows the integral hydrogen generated, both in-vessel and ex-vessel, compared to the hydrogen mass remaining in containment. Atmospheric mole fractions are presented in through Figure 3-75 depicting the distribution of relevant gases. Carbon monoxide, generated during MCCI, is an additional combustion concern and is added to Figure 3-73 throughFigure 3-75. Combustible concentrations are precluded within containment prior to vent line operations due to high concentrations of nitrogen and steam even though significant amounts of hydrogen are present. Once the wetwell and eventual drywell vent line actuations are performed, hydrogen and carbon monoxide are rapidly dispersed from containment at 13 and 60 hrs. Without containment failure, hydrogen is not distributed to the reactor building.

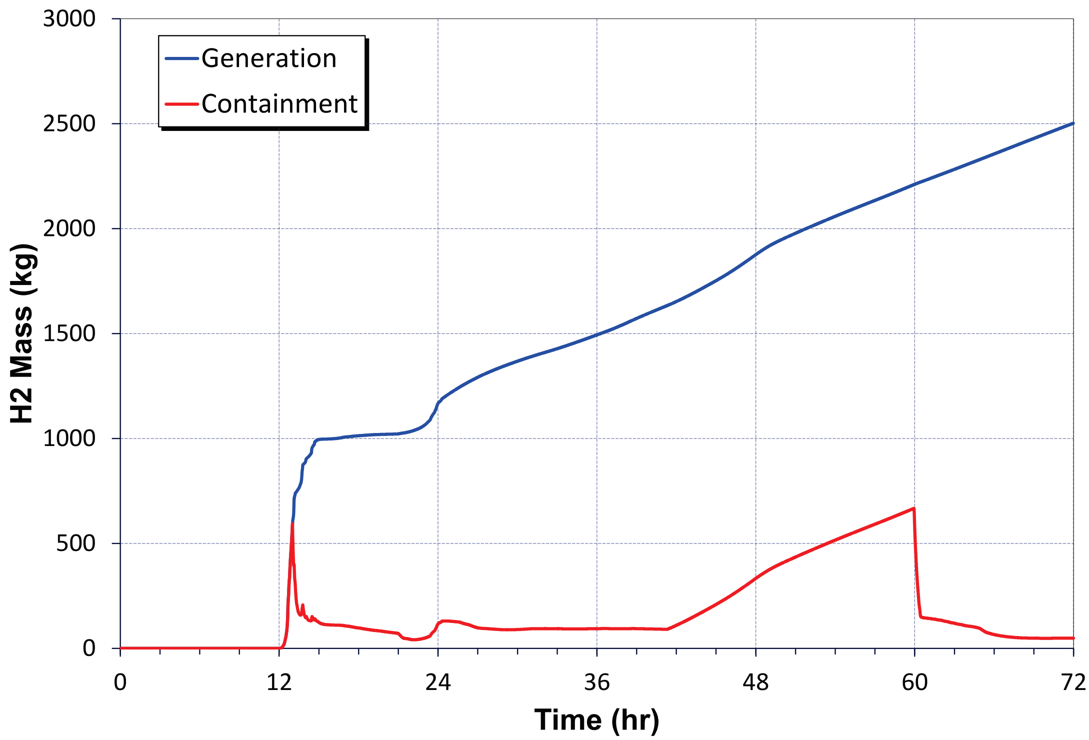


Figure 3-72 Mark II hydrogen generation and transport for Case 11p1

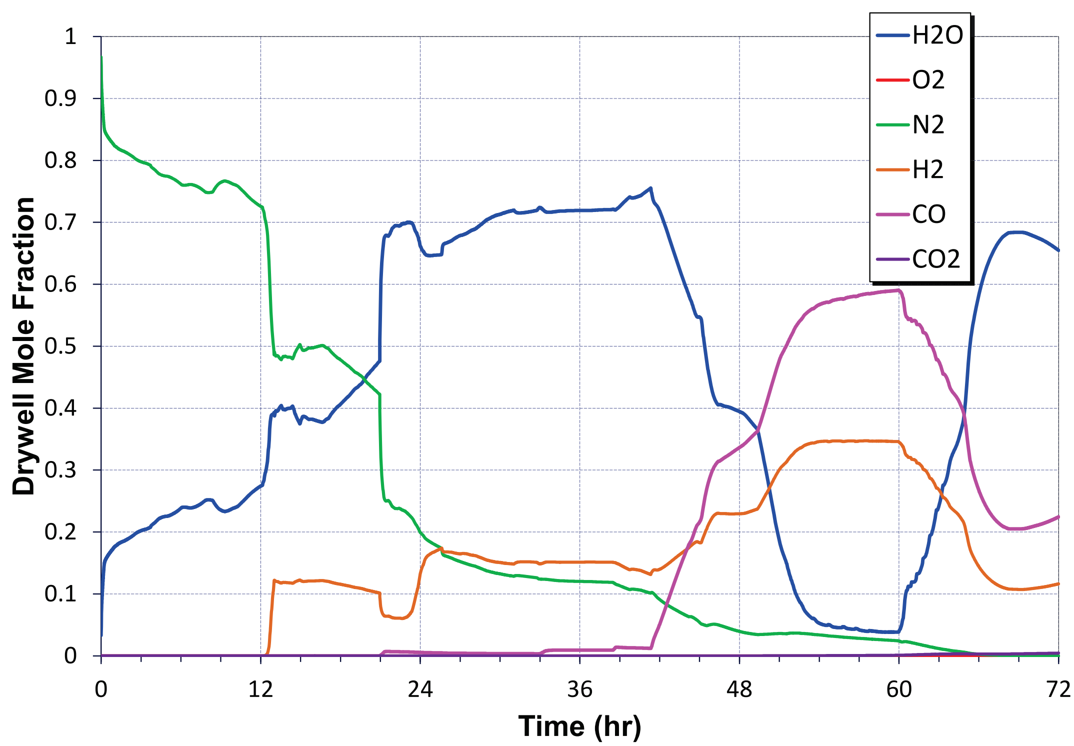


Figure 3-73 Mark II drywell gas distribution for Case 11p1

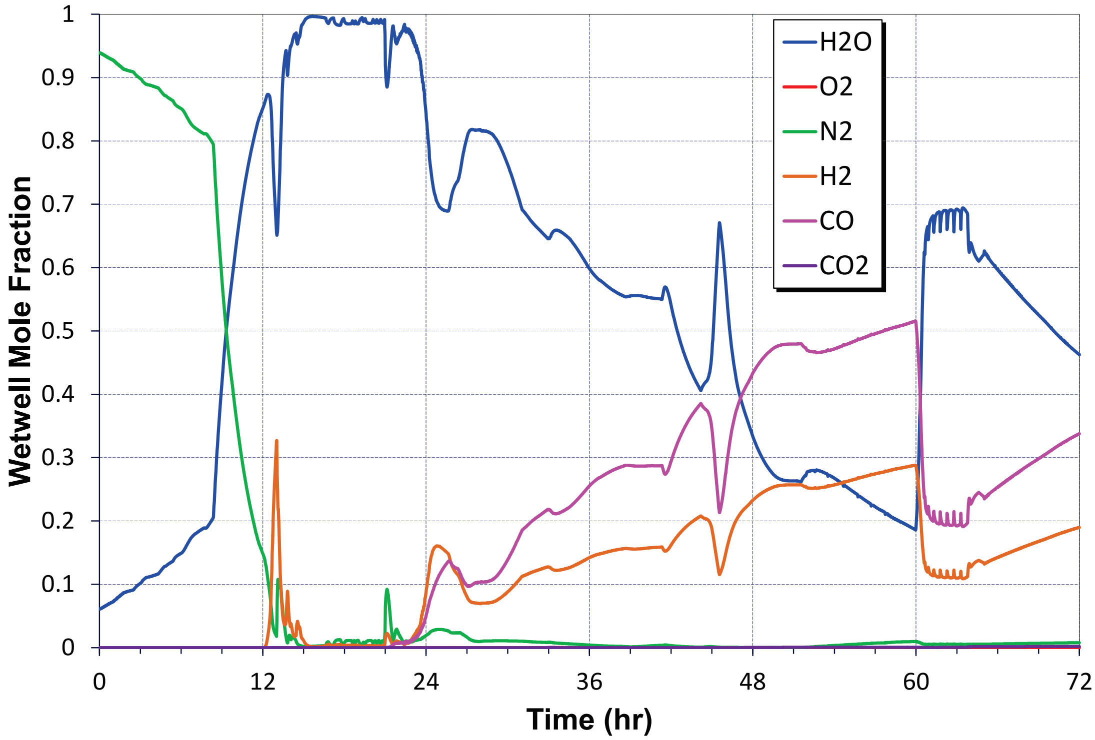


Figure 3-74 Mark II wetwell gas distribution for Case 11p1

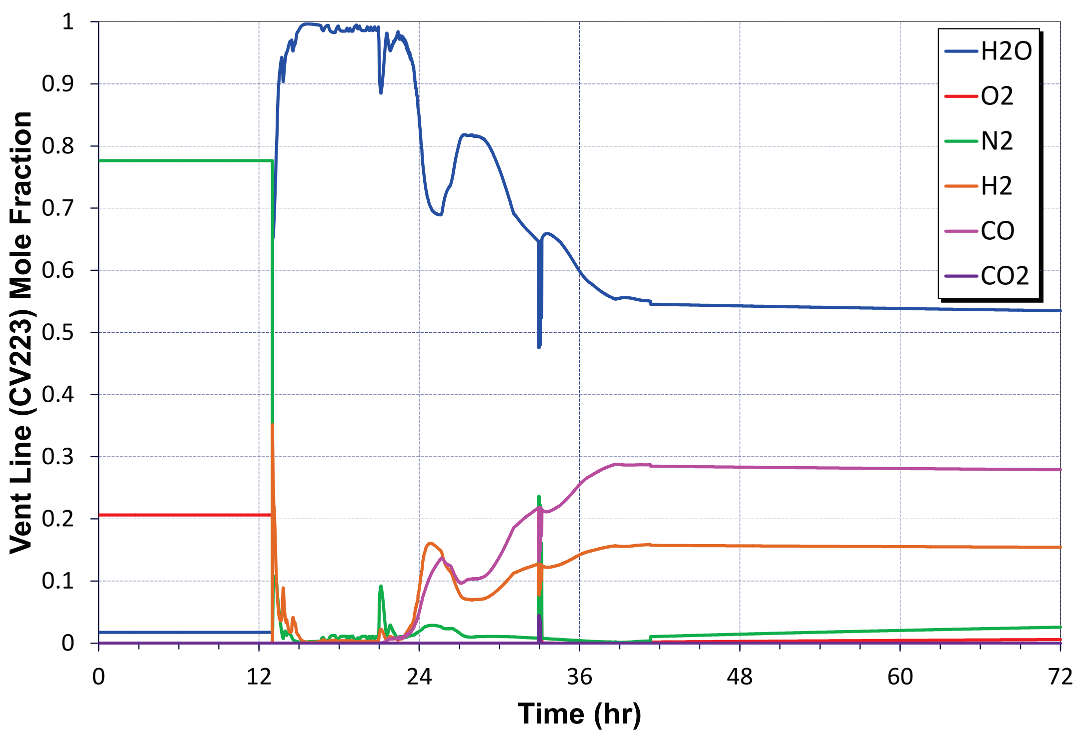


Figure 3-75 Mark II wetwell vent gas distribution for Case 11p1

Releases to the environment observed from the accidents analyzed (see Table 3-3) are presented in Figure 3-76. The summary of the main parameters for the Mark II analyses are provided in Table 3-7 (see Table 3-3 for details of the scenarios). The observations made in Section 3.3.1.3 for the Mark I accident analyses are largely applicable to the Mark II accident analyses. Similarly, FLEX injection appears to reduce overall releases of cesium whereas drywell venting produces larger environmental releases, as would be anticipated.

Noticeably, cases representing MSLCR for the Mark II model are not necessarily dominant release cases as was observed in Mark I results. Unlike the Mark I sequence of events, MSLCR does not produce head flange failure in the Mark II model; therefore the environment release path remains the designated vent line in the analyses performed with the Mark II model[11].

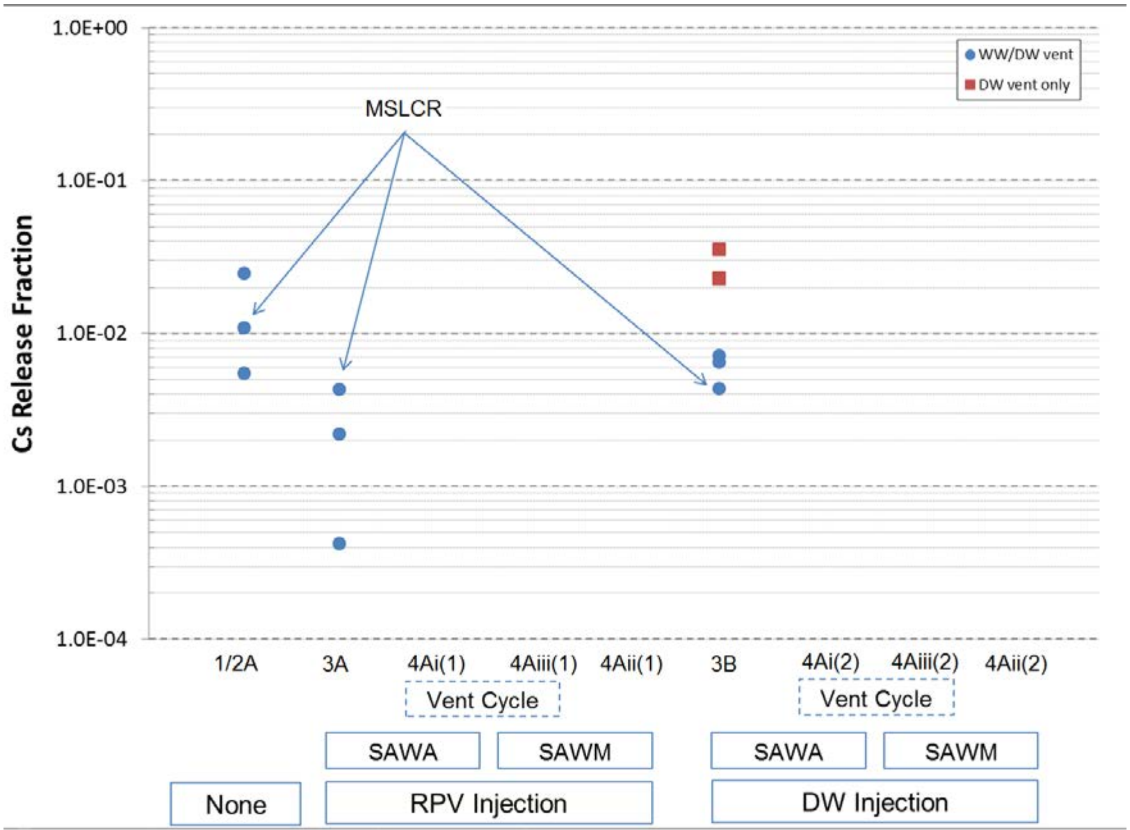


Figure 3-76 Mark II cesium environmental release fraction

Figure 3-77 presents the cesium release observed for the cavity sensitivities performed as well as the calculation matrix for the Mark II analyses (M_I refers to Mark I and M_II refers to Mark II). The designation of PCPL as 60 psig was changed to 45 psig for several of the calculations provided. While the final magnitude of the release remains reasonably comparable, it should be reflected upon that the event timings, in particular the vent actuation prior to lower head failure, does become prominent. Cesium releases remain below 4% of total inventory for the analyses performed.

[11] The two models do differ in regard to drywell nodalization, where the Mark I model segregates the drywell atmosphere space into three separate volumes, the drywell in the Mark II model is a single well mixed receptor. Note in addition to the nodalization variations, the vapor space volume in Mark II is larger than found at Mark I.

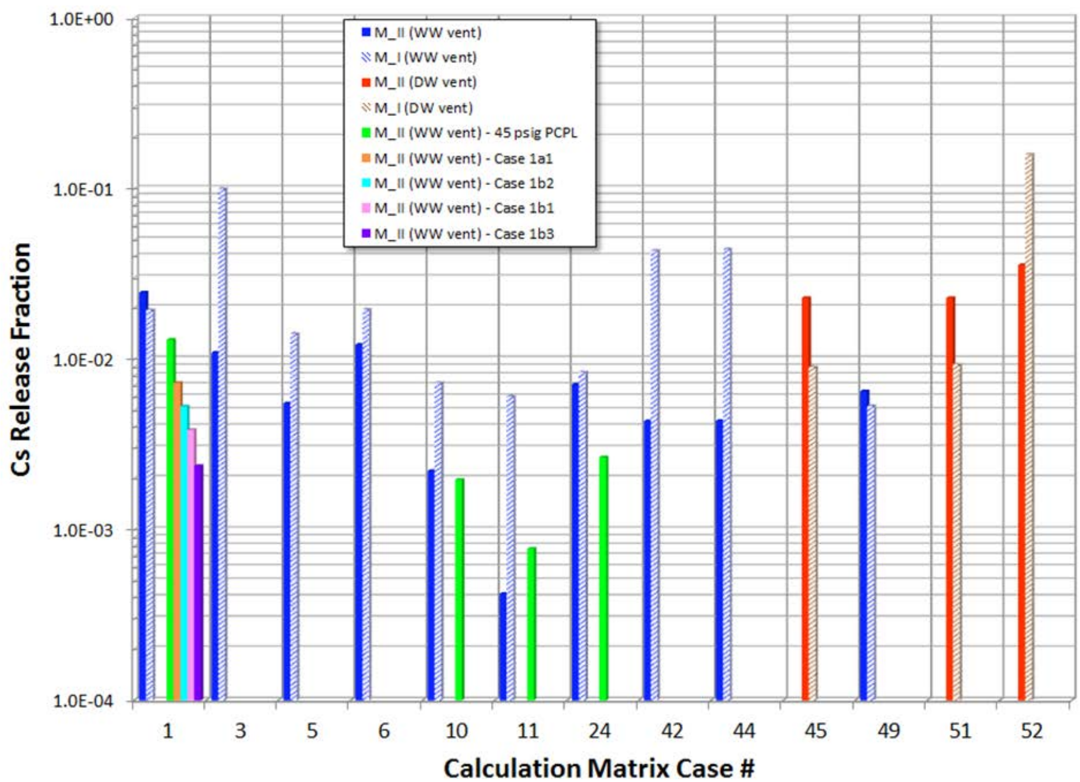


Figure 3-77 Cesium release fractions for Mark II cases and sensitivities as well as corresponding Mark I cases

Table 3-7 Summary of main parameters for Mark II analysis

Option	Case	Cs (RF) @ 72 hours	Cs (DF = 1/RF)	I (RF) @ 72 hours	Early venting (opn/cls) (hr)	RCIC failure (hr)	SRV failure (hr/# cycles)	MSL failure (hr)	WW venting (opn/cls) (hr)	DW venting (opn/cls) (hr)	LH fails (hr)	FLEX stops (hr)	DW head leaks (hr)	In-core H2 (kg)	Meject (ton)
1/2A	1	2.46E-02	41	1.98E-01	-	8.4	16.8/39	-	22.8/-	-	22.0	-	-	1232	248
	1p1	1.30E-02	77	9.45E-02	-	8.4	16.4/38	-	13.8/-	-	21.8	-	-	1300	242
1/2A	3	1.09E-02	92	1.03E-01	-	4.0	-/202	14.4	14.4/-	-	22.8	-	-	1095	323
1/2A	5	5.52E-03	181	4.94E-02	-	13.4	23.4/43	-	32.2/-	-	37.3	-	-	1223	323
1/2A	6	1.22E-02	82	1.24E-01	-	8.4	17.4/41	-	13.6/-	-	22.6	-	-	1482	260
3A	10	2.22E-03	451	2.53E-02	-	8.4	16.8/39	-	22.2/52.0	70.9/-	22.0	72	-	1154	222
	10p1	1.96E-03	509	1.95E-02	-	8.4	16.4/38	-	13.8/42.4	60.5/-	21.8	72	-	1204	213
3A	11	4.22E-04	2372	4.50E-03	-	8.4	7.8/24	-	20.3/49.9	68.8/-	20.0	72	-	1307	220
	11p1	7.76E-04	1289	5.04E-03	-	8.4	7.8/24	-	13.0/41.3	55.3/-	20.6	72		1157	232
3B	24	7.12E-03	140	5.80E-02	-	8.4	16.8/39	-	30.5/51.7	68.4/-	22.0	72	-	1234	222
	24p1	2.66E-03	376	1.88E-02	-	8.4	16.4/38	-	13.8/42.8	57.1/-	21.8	72	-	1271	244
3A	42	4.33E-03	231	3.65E-02	-	4.0	-/202	14.4	14.4/51.2	62.0/-	22.8	72	-	1095	323
3B	44	4.36E-03	230	3.68E-02	-	4.0	-/202	14.4	14.4/51.8	66.7/-	22.8	72	-	1095	323
3B	45	2.29E-02	44	1.93E-01	-	12.2	20.8/237	-	-	18.3/-	23.9	72	-	1202	280
3B	49	6.49E-03	154	7.62E-02	-	0.0	2.3/75	-	11.2/33.0	46.7/-	6.4	72	-	1231	316
3B	51	2.29E-02	44	2.02E-01	-	12.2	19.7/235	-	-	13.8/-	23.9	72	-	1152	278
3B	52	3.57E-02	28	2.87E-01	-	12.2	-/244	-	-	13.8/-	21.6	72	-	1217	272

3.4 Consideration of Uncertainty

MELCOR is considered a state-of-the-art system-level integral code for severe accident modeling and analysis, and it has reached a reasonably high level of maturity over the years as evidenced from its wide acceptability and its broad range of applications. MELCOR embodies the current state of knowledge of severe accident phenomena. However, as for any system-level codes with similar capabilities, it is important to recognize that there are phenomenological uncertainties in severe accident progression that have a direct bearing on modeling uncertainties. Moreover, it is important to understand the compounding effect of various uncertainties on the output parameters of interest (e.g., hydrogen generation, release of fission products to the environment, etc.).

Given the state of severe accident modeling and the residual uncertainties therein, an "adequate for purpose" approach is to do bounding analysis reflecting best estimate outcome supplemented by some measures of uncertainties. The bounding values of the output parameters of interest are then compared to those considered acceptable from a safety margin standpoint to determine if further reduction of residual uncertainties is warranted. In the remainder of this section, a brief discussion is provided regarding some of the more important modeling uncertainties in MELCOR and potential implications of such uncertainties. Also, a brief discussion of uncertainties in reference to mitigation systems modeling in MELCOR is provided.

The in-vessel melt progression modeling in MELCOR starting with the loss of intact core geometry to clad oxidation, in-vessel hydrogen generation, molten core relocation to lower plenum, and subsequent lower head failure are based on small scale experiments which were conducted with the primary objective of gaining an understanding of these phenomena in relation to the observation and experience from plant accidents such as Three Mile Island.

MELCOR, for example, has a parametric model for evaluating fuel mechanical response whereby a temperature-based criterion is used to define the threshold beyond which normal ("intact") fuel rod geometry can no longer be maintained, and the core materials at a particular location collapse into particulate debris. The relocation of molten and particulate debris to the lower plenum is controlled by the relocation time constant parameter in MELCOR. This parameter is used as a surrogate for the broad uncertainty in the debris relocation rate into water in the lower head. The choice of relocation time constant affects the potential for debris coolability in the lower head (faster relocation rates decrease coolability; slower rates improve coolability). These and other related in-vessel melt progression modeling attributes in MELCOR affect the timing of lower head failure as well as the characteristics of melt (temperature, mass, and composition) exiting the vessel which provide the initial and boundary conditions for ex-vessel melt progression. These attributes also affect the amount of hydrogen generation – a parameter of interest from the containment integrity standpoint. The current state of BWR modeling in MELCOR does not consider the effects of structures (such as control rod drive mechanisms) beneath the lower head. Such structures can provide an energy sink to materials that relocate to the lower head, thus potentially delaying the lower head failure. Also, the core materials can transfer heat and potentially freeze onto these structures as they are ejected from the vessel into the cavity.

As in the case of in-vessel melt progression, the ex-vessel phenomenological modeling is based on experiments which were conducted to gain an understanding of melt spreading on the drywell floor, debris quenching in the presence of water, and molten core-concrete interaction, among others. After the core debris is released from the reactor vessel lower head, it flows out of the reactor pedestal onto the main drywell floor. The precise conditions under which core debris would flow out of the pedestal and across the drywell floor are uncertain. These uncertainties are captured in MELCOR in a parametric manner. Phenomenological models are being continuously

updated in MELCOR based on data generated through international research programs. The improved models are expected to reduce the uncertainties to some extent.

Partitioning the initial core inventory of radionuclides (cesium and iodine in particular) among certain allowable chemical forms (for release and transport) is performed within MELCOR input files that define the initial spatial mass distribution of each chemical species and its associated decay heat. Changes to the mass fractions assumed for a particular chemical group directly affect the mass fractions of other chemical groups. Due to the complexity of this modeling approach, five alternative sets of MELCOR input files are used to bound uncertainties by spanning the range of plausible combinations of chemical forms of key radionuclide groups.

Gaseous iodine remains another source term issue with uncertainties, especially with respect to long-term radioactive release mitigation issues after the comparatively much larger airborne aerosol radioactivity has settled from the atmosphere. Mechanistic modeling of gaseous iodine behavior is a technology still under development with important international research programs to determine the dynamic behavior of iodine chemistry with respect to paints, wetted surfaces, buffered and unbuffered water pools undergoing radiolysis, and gas phase chemistry.

Several other sources of uncertainties, not specifically discussed here, can have an impact on MELCOR results. Moreover, there are uncertainties in modeling various mitigation features (e.g., reactor core injection cooling or RCIC performance, drywell water addition mode and effectiveness, suppression pool decontamination factor, and in-containment radionuclide retention factor). Given these various sources of uncertainties, it is not uncommon to find an order of magnitude or more variation in the MELCOR prediction of the source term.

An uncertainty analysis for a long term station blackout was recently performed for Peach Bottom following the completion of SOARCA (a Surry uncertainty analysis is currently in progress). The key MELCOR model parameters for Peach Bottom included the following.

- Zircaloy melt breakout temperature
- Molten clad drainage rate
- Fuel failure criterion (transformation of intact fuel to particulate debris)
- Radial debris relocation time constants
- Debris Pool Interface Heat Transfer
- Debris lateral relocation—cavity spillover criteria and spreading rate
- Chemical forms of iodine and cesium (I2, CH3I, CsI, CsOH, and Cs2MoO4)

The results of the SOARCA uncertainty analysis confirmed that prediction of the source term can have an order of magnitude or more variation.

The analyses presented in this report did not consider a detailed uncertainty analysis. Limited sensitivity analysis was carried out to assess the range of MELCOR results and to further confirm the bounding range of source terms. The run matrix given in Table 3-2 contains mainly variations in the accident boundary conditions such as operation of RCIC, suction source of RCIC injection, etc. In addition, the run matrix also includes sensitivity to the fractional open area of a thermally seized SRV, which was considered an important parameter in the SORACA uncertainty analysis. This parameter was varied to observe its effect on the possibility of main steam line creep rupture (MLSCR). In all the MELCOR calculations, MSLCR only occurred by disabling the SRV failure or intentional depressurization. MSLCR scenarios represent the highest environmental releases due to early bypass of the suppression pool. The results of the calculations showed that there are

variations in the source term by as much as an order of magnitude, especially when the releases are low.

3.5 Accident Progression Summary and Conclusions

The MELCOR analysis investigated detailed accident progression, source terms, and the containment response following an ELAP subject to appropriate initial and boundary conditions for representative Mark I and Mark II containment designs. The run matrix for the Mark I analysis included sensitivities to the following main parameters:

- Mode of venting (e.g., WW or DW first, vent cycling)
- Status of RPV depressurization (e.g., SRV open pre-core damage, SRV stuck open)
- Mode of FLEX injection (either to the RPV or the DW at lower head failure)
- Water injection control (e.g., water management by throttling flow at high level in wetwell)

The run matrix for the Mark II analysis included a subset of the Mark I runs based on the insights from the Mark I MELCOR calculations. The base case Mark II MELCOR model had a dry lower cavity without in-pedestal downcomers that was chosen mainly because of the availability of the MELCOR model. However, sensitivities were performed to examine the impact of the pedestal and lower cavity designs among the fleet by modifying the base model (e.g., replace the concrete lower cavity with water).

The following are the main observations from the calculations:

- A combination of venting and water injection is required to prevent containment failure and is a beneficial strategy for mitigating radiological releases.

- In all MELCOR calculations for the Mark I analysis, containment venting after core damage occurs well before the lower head failure and injection of water. For this reason, the sudden release at the time of venting is only sensitive to core degradation and fission product transport and deposition rather than the late water injection at the time of lower head failure.

- Pre-core damage anticipatory venting reduces the containment base pressure at the time of core damage and results in a delay when post core damage venting is required. This time delay affects fission product behavior inside the RPV and containment and impacts the sudden release at the time of venting.

- Creep rupture of the main steam line seems unlikely if the reactor pressure is maintained low. For the cases that the creep ruptured was forced by disabling the stuck open SRV model (either through high temperature or excessive number of cycles), the results show the highest releases to the environment. The failure of the main steam line results in failure of the containment (opening of the upper drywell head) and bypass of the suppression pool. The failure also results in migration of hydrogen to the refueling bay of the Mark I containment, and consequent hydrogen combustion and release of fission products directly to the environment.

- Addition of water either into the RPV or the drywell has the benefit of cooling the core debris and containment atmosphere and can prevent the over-temperature failure of the upper drywell head. For the Mark I analysis with water addition, the maximum structure

temperatures at the drywell upper head or the drywell liner near the elevation of the drywell vent remains below 500ºF.

- For the Mark I analysis, the presence of water in the pedestal and lower drywell can cool the debris and delay liner melt through. However, without water injection at the time of lower head failure, the debris eventually heats up and contacts the drywell liner, leading to its failure. With water addition, liner melt through is averted.

- Containment venting is efficient in purging the hydrogen and non-condensibles from the containment. Water injection is also helpful in maintaining a steam inerted atmosphere, which can preclude an energetic hydrogen combustion.

- The calculations show that the environmental releases from the Mark II containment are in general comparable to or lower than those from the Mark I containment.

- For the Mark II analysis, additional analysis was performed to investigate different lower cavity configurations by modifying the base model. The environmental releases are within the range of source terms predicted based on the variations in the scenario boundary conditions.

- In the present analysis, hydrogen migration outside the containment through nominal leakage pathways does not lead to accumulation of combustible mixtures in the reactor building. Therefore, a hydrogen burn is not predicted in the reactor building as long as the containment does not fail due to drywell head failure or liner melt-through.

3.6 References for Chapter 3

1. U.S. Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century, The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 2011.

2. U.S. Nuclear Regulatory Commission, "Consideration Of Additional Requirements For Containment Venting Systems For Boiling Water Reactors With Mark I and Mark II Containments," SECY-12-0157, November 2012.

3. "MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.1.6840," SAND 2015-6691 R, Sandia National Laboratories, August 2015 (ADAMS Accession No. ML15300A479).

4. U.S. Nuclear Regulatory Commission, "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Sever Accident Conditions," Order EA-13-109, June 2013.

5. U.S. Nuclear Regulatory Commission, D. Chanin, M. L. Young, J. Randall, "Code Manual for MACCS2: User's Guide," NUREG/CR-6613, Vol. I (1998).

6. U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses (SOARCA) report", NUREG-1935, January 2012.

7. Sandia National Laboratories, "Fukushima Daiichi Accident Study (Status as of April 2012)," SAND2012-6173, July 2012.

8. Shaffer, C.J., et al., “Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant,” NUREG/CR-5305 Volume 3, Nuclear Regulatory Commission: Washington DC, October 1992.

9. Kelly, D.L., et al., “An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements in Performance,” NUREG/CR-5528, Nuclear Regulatory Commission, Washington DC, July 1990.

10. U.S. Nuclear Regulatory Commission, Filtering Strategies Rulemaking Public Meeting, December 12, 2013 (ADAMS Accession No. ML13357A794).

11. Bixler N., et al. “State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis,” NUREG/CR-7110 Vol. 1, Rev. 1, Nuclear Regulatory Commission, Washington DC, May 2013

4 OFFSITE CONSEQUENCE ANALYSIS

This section documents the offsite consequence analyses of the accident progression cases discussed in the MELCOR accident analysis section. The MELCOR Accident Consequence Code System (MACCS) was used to calculate offsite doses and land contamination, and their effect on members of the public with respect to individual early and latent cancer fatality risk, land contamination areas, population dose, and economic costs. MACCS was selected as the consequence analysis tool for this project because it is NRC's code for performing offsite consequence analyses for severe accident risk assessments.

This section begins with a general description of MACCS and is then followed by a discussion of the modeling approach for the Mark I and Mark II source terms. The discussion of the modeling approach begins with a description of the radionuclide release modeling including the source term binning strategy. The discussion continues with a description of the calculational grid; site data; meteorological data; atmospheric transport; early, intermediate, and long-term phase exposure pathways; protective actions and costs; dosimetry; and health effects. The results are then presented and explained. These results are used to estimate the relative public health risk reduction associated with the various CPRR alternatives. This section then continues with a description of the sensitivity analyses and finally concludes with a discussion of the major insights gained from this effort.

4.1 MACCS Conceptual Models

The MACCS code was developed for NRC to evaluate offsite consequences from a hypothetical release of radioactive materials into the atmosphere. The code is used as a tool to assess the risk and consequences associated with accidental releases of radioactive material into the atmosphere in probabilistic risk assessment studies. The code models atmospheric transport and dispersion, emergency response and long-term phase protective actions, exposure pathways, health effects, and economic costs. While MACCS models consequences of airborne releases depositing onto water bodies, MACCS does not model transport and dispersion of aqueous source terms consistent with MELCOR, which does not estimate aqueous releases. The Fukushima accidents demonstrated that large volumes of contaminated water can be generated which can disperse through surface water, sediments, soils, and groundwater; however, this is a gap in existing PRA modeling technology. Past assessments have shown that aqueous releases pose less overall health and environmental risk than airborne releases [1].

MACCS estimates consequences in four steps:

1. atmospheric transport and deposition of radioactive materials onto land and water bodies,

2. the estimated exposures and health effects for up to seven days following the beginning of release (early phase),

3. the estimated exposures and health effects during an intermediate time period of up to one year (intermediate phase), and

4. the estimated long-term (e.g., 50 years) exposures and health effects (late-phase model).

The assessment of offsite property damage in terms of contaminated land and economic costs uses all four parts of the modeling. An overview of the code is provided below.

MACCS[12] version 3.7.5 was used for the consequence analyses [2] [3]. The WinMACCS graphical user interface (version 3.7.5) was used to input data into MACCS [4]. Site file data including population, economic values, and land use data was prepared using the SecPop preprocessor code version 4.3 [5]. MELCOR source terms were converted to MACCS input format using the MelMACCS version 1.7.3 code [6]. These codes have been developed by the NRC and Sandia National Laboratories (Sandia) over multiple decades and have rigorous quality control and quality assurance processes in place. Code capabilities described in this section are specific to the exact version used in the analysis.

MACCS is used by U.S. nuclear power plant license renewal applicants to support the plant specific evaluation of severe accident mitigation alternatives (SAMAs) that may be required as part of the applicant's environmental report for license renewal. MACCS is also routinely used in severe accident mitigation design alternative (SAMDA) or severe accident consequence analyses for environmental impact statements (EISs) supporting design certification, early site permit, and combined construction and operating license reviews for new reactors. The NRC's regulatory analysis guidelines in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," [7] and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," [8] recommend the use of MACCS to estimate the averted "offsite property damage" cost (benefit) and the averted offsite dose cost elements. The information from MACCS code runs supports a cost-benefit assessment for various potential plant improvements as part of SAMAs or SAMDAs.

MACCS has also been used in a variety of NRC research studies. MACCS was used in the State-of-the-Art Reactor Consequence Analyses (SOARCA) project, which aimed to calculate the accident progression and consequences in a very detailed manner for the most important severe accident scenarios at the Peach Bottom Atomic Power Station (Peach Bottom) and the Surry Power Station (Surry). These analyses were documented in NUREG-1935 [9], NUREG/CR-7110 [10] [11], and NUREG/BR-0359 [12]. The MACCS best practices as applied in the SOARCA project were documented in NUREG/CR-7009 [13]. Following the SOARCA project was an uncertainty analysis of one of the SOARCA scenarios, the Peach Bottom unmitigated long-term station blackout (LTSBO), documented in NUREG/CR-7155 [14]. This study propagated uncertainty for a variety of key uncertain MELCOR and MACCS parameters to develop insights into the overall sensitivity of SOARCA results and conclusions and to identify the most influential input parameters for consequences. The results of the Peach Bottom unmitigated LTSBO uncertainty analysis corroborated the conclusions of the SOARCA project.

MACCS was also used in a consequence study of a beyond design basis earthquake affecting the spent fuel pool for a U.S. Mark I BWR and this is documented in NUREG-2161 [15]. MACCS was used in the SECY-12-0157 technical basis related to containment venting systems for BWRs with Mark I and Mark II containments [16]. In addition, MACCS is currently being used for the offsite consequence analyses supporting the NRC's Full-Scope Level 3 Probabilistic Risk Assessment (PRA) for the Vogtle Electric Generating Plant [17].

[12] Recent versions of MACCS have been known as "MACCS2" however NRC has decided to remove the "2" so that the code has just one version number (version 3.7.5 in this case). MACCS version 3.7.5 was developed in 2013 and was the most current version available at the time.

4.1.1 Atmospheric Transport and Dispersion (ATD)

MACCS models dispersion of radioactive materials released into the atmosphere using the straight-line Gaussian plume segment model with provisions for meander and surface roughness effects. The ATD model treats the following: plume rise resulting from the sensible heat content (i.e., buoyancy), initial plume size caused by building wake effects, release of up to 200 plume segments, dispersion under statistically representative meteorological conditions, deposition under dry and wet (precipitation) conditions, and decay and ingrowths of up to 150 radionuclides and a maximum of six generations. The model does not treat in detail irregular terrain, spatial variations in the wind field, and temporal variations in wind direction.

The user has the option to use a single weather sequence or multiple weather sequences. Sampling among multiple weather sequences is used in PRA studies to evaluate the effect of weather conditions at the time of the hypothetical accident.

The results generated by the ATD model include contaminant concentrations in air, on land, and as a function of time and distance from the release source; these results are subsequently used in early, intermediate, and long-term phase exposure modeling.

4.1.2 Phases of the MACCS Conceptual Model

The U.S. Environmental Protection Agency (EPA), in its "Manual of Protective Action Guides and Planning Guidance for Radiological Incidents," has characterized the response to a nuclear accident in three distinct phases of activity. MACCS has the capability to characterize the different exposure pathways, protective actions, and costs associated with the accident for each of the three phases. These are summarized in Table 4-1. First, the early (emergency) phase is used for the period of up to 7 days following the start of the initiating event that causes the accident. The intermediate phase starts at the end of the early phase and in MACCS can last up to one year. The long-term phase starts at the conclusion of the intermediate phase and can last 50+ years.

Table 4-1 Phases of the MACCS conceptual model

	Early Phase (Emergency)	Intermediate Phase	Long-Term Phase
Primary Offsite Accident Response Objective(s)	• Protect public from plume exposures.	• Protect public from exposures to deposited materials. • Plan for long-term cleanup and recovery activities.	• Protect public from exposures to deposited materials. • Conduct long-term cleanup and recovery activities.
Typical Duration and Time Frame	Days to weeks, starting at the time of the accident's initiating event. 7 days was used in all CPRR calculations.	Weeks to years, starting at the end of the early phase. 3 months was used in CPRR base calculations.	Months to decades, starting at the end of the intermediate phase. 50 years was used in all CPRR calculations.
Exposure Pathways	• Cloudshine • Groundshine • Inhalation • Skin deposition	• Groundshine • Inhalation of resuspended materials.	• Groundshine • Inhalation of resuspended materials. • Food and water ingestion.
Protective Actions	• Sheltering • Evacuation • Relocation • KI ingestion	• Relocation	• Interdiction • Decontamination • Condemnation

4.1.2.1 Early (Emergency) Phase Protective Actions and Exposure Pathways

The early phase model in MACCS assesses the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase and it can extend up to seven days after the arrival of the first plume at any downwind spatial interval. Early exposures in this phase account for emergency planning (i.e., sheltering, evacuation, and relocation of the population). MACCS models sheltering and evacuation actions for user-specified population cohorts. Different shielding factors for the different exposure pathways (cloudshine, groundshine, inhalation, and deposition on the skin) are associated with three types of activities: normal activity, sheltering, and evacuation.

For population cohorts that are not explicitly modeled to evacuate in MACCS, dose-dependent relocation actions may take place during the emergency phase. If individuals at any location are projected to exceed either of two user-specified dose thresholds (a larger, "hotspot" threshold, and a smaller, "normal threshold") over the duration of the emergency phase, they are relocated at a user-specified time after plume arrival and are modeled to receive no further early phase exposures.

MACCS also models the beneficial effect of populations consuming potassium iodide (KI) to reduce radioiodine inhalation doses to the thyroid. KI can saturate the thyroid with stable iodine and thereby reduce the amount of radioiodine that can be absorbed. KI is distributed near some nuclear power plants. MACCS allows the user to specify which population cohorts would take KI, the expected fraction of the population within each cohort that would take it, and the efficacy of the KI in reducing thyroid doses.

4.1.2.2 Intermediate Phase Protective Actions and Exposure Pathways

MACCS can model an intermediate phase with a duration of up to one year following the end of the early phase. The only protective action modeled in this phase is relocation. If the projected dose to a population exceeds a user-specified threshold over a user-specified time duration, the population is assumed to be relocated to an uncontaminated area for the entire duration of this phase. A corresponding per-capita per diem economic cost is defined by the user. If the projected dose does not reach the user specified threshold, exposure pathways for groundshine and inhalation of resuspended material are modeled. The food and water ingestion pathway is not modeled in the intermediate phase because of the assumption that uncontaminated food and water would be brought in from outside the affected region during this interim period.

4.1.2.3 Long-Term Phase Protective Actions and Exposure Pathways

In the long-term phase (typically 30-50 years following the end of the intermediate phase), protective actions are defined to minimize the dose to an individual by external (e.g., groundshine) and internal (e.g., food consumption and resuspension inhalation) pathways. Decisions on protective actions are based on two sets of independent actions — i.e., decisions relating to whether land, at a specific location and time, is suitable for human habitation (habitability) or agriculture production (farmability). Habitability and farmability are defined by a set of user-specified maximum doses and a user-specified exposure periods to receive those doses. Habitability and farmability decision-making can result in four possible outcomes:

1. land is immediately habitable or farmable,
2. land is habitable or farmable after decontamination,
3. land is habitable or farmable after decontamination and interdiction[13], or
4. land is deemed not habitable after decontamination plus 30 years of interdiction or land is deemed not farmable after decontamination plus 8 years of interdiction (i.e., it is condemned).

Land is also condemned if the cost of decontamination exceeds the value of the land. The dose criterion for the MACCS modeling of individuals returning back to the affected (i.e., contaminated) area is a user input and is typically based on the U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAGs) [18] or state-specific guidelines. The decision on whether land is suitable for farming is first based on prior evaluation of its suitability for human habitation.

Decisions on decontamination are made using a decision tree. The first decision is whether land is habitable. If it is, then no further actions are needed. The population returns to their homes and receive a small dose from any deposited radionuclides for the entire long-term phase. If land is not habitable, the first option considered is to decontaminate at the lowest level of dose reduction, which is also the cheapest to implement. If this level is sufficient to restore the land to habitability, then it is performed. Following the decontamination, the population return to their homes and receive a small dose based on the residual contamination for the duration of the long-term phase. If the first level of decontamination is insufficient to restore habitability, then successively higher levels are considered. MACCS considers up to three decontamination levels. If the highest level of

[13] In this context, interdiction generally refers to the period of time in which residents are not permitted to return to live on their property because the radiation doses they would receive (from external sources and inhalation) exceed the habitability criterion. Interdiction allows for radioactive decay, decontamination, and weathering to potentially bring these doses to a point where they would no longer exceed the habitability criterion.

decontamination is insufficient, then interdiction for up to 30 years is considered following the highest level of decontamination. During the interdiction period, radioactive decay and weathering work to reduce the dose rates that would be received by the returning population. If the highest level of decontamination followed by interdiction is sufficient to restore habitability, then it is employed and the population is allowed to return. Doses are accrued for the duration of the long-term phase. If habitability cannot be restored by any of these actions, then the land is condemned. The land is also condemned if the cost of the required action to restore habitability is greater than the value of property.

The decision tree for farmability is first based on prior evaluation of its suitability for human habitation—land cannot be used for agriculture unless it is habitable. Furthermore, farmland must be able to grow crops or produce dairy products that meet the user-specified farmability criterion, which is an ingestion dose equivalent threshold, and in this analysis is set to equal the habitability criterion. If farmland is habitable and farmable, a food chain model is used to determine doses that would result from consuming the food grown or produced on this land. The COMIDA2 food chain model is the latest model developed for use in MACCS. This model contains data on expected radionuclide uptake in nine foodstuff types for different seasons of the year and for different contamination levels and food category consumption rates for an average adult.

MACCS values of total long-term population dose and health effects account for exposures received by workers performing decontamination. While engaged in cleanup efforts, workers are assumed to wear respiratory protection devices; therefore, they only accumulate doses from groundshine.

4.1.3 Offsite Consequence Measures

The results of the consequence analyses are presented in terms of health risks to the public, population dose, land contamination, population subject to long-term protective actions, and economic costs. All consequence results are presented in this section as conditional consequences (i.e., assuming that the accident occurs), and show the risks to individuals as a result of the accident (i.e., LCF risk per event or early fatality risk per event). Therefore in this section, there is no consideration of the different probabilities/frequencies of the different accident progression scenarios. The risks, population dose, and economic costs are mean values (i.e., expectation values) over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular or annular region.

4.1.3.1 Health Effects

Populations located on the MACCS computational grid receive doses from the passing plume (cloudshine), by exposure to materials deposited on the ground (groundshine), by inhalation of airborne radioactive materials (from the plume or from mechanical or wind-driven resuspension of materials deposited on the ground), and by ingestion of contaminated food and water.

MACCS uses a dose conversion factor file based on EPA's Federal Guidance Report No. 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides," [19] which converts the integrated air concentration and ground deposition of 825 radionuclides to a whole body effective dose and individual organ doses for 26 tissues and organs and for four exposure pathways. The whole body effective dose and individual organ doses are then used to calculate health effects. In general, the radiological dose to a receptor in a given spatial element is the product of the radionuclide concentration or quantity, the exposure duration, the shielding factor, the dose conversion factor, and the usage factor (e.g., breathing rate). The total dose to an organ or the

whole body used for modeling of health effects or protective action decisionmaking is then summed across the relevant exposure pathways and radionuclides.

MACCS considers deterministic and stochastic health effects and estimates the likelihood that an exposed individual may experience a specific health effect (e.g., lung impairment, breast cancer). Deterministic health effects (early injuries and early fatalities) are calculated using nonlinear dose response models. These models consider the dose equivalent delivered to the target organ, a dose equivalent threshold below which the effect is not expected to occur and thus the risk is zero, a dose equivalent that would induce the effect in half the exposed population, and a shape factor that affects the range over which the likelihood of the health effect goes from zero to one for the overall population. For stochastic health effects (latent cancer fatalities), the NRC uses the linear no-threshold (LNT) model for analyses serving regulatory purposes, and this model is adopted in MACCS here. The LNT dose-response relationship suggests that any increase in dose, no matter how small, results in an incremental risk. The parameters supporting these models are discussed in Section 4.2.10.

4.1.3.2 Land Contamination and Populations Subject to Long-Term Protective Actions

Land contamination can be computed in two different ways in MACCS. The first is to compute the area of land exceeding a user-specified areal concentration of a user-specified radionuclide. For example, the area of land exceeding a cesium-137 ground concentration of 15 $\mu Ci/m^2$ (15 Ci/km^2) is a useful metric for quantifying the long-term land contamination caused by the accident because cesium-137 is the most important long-term contaminant that limits habitability. This metric was used to quantify the land contamination caused by the Chernobyl accident. The second method is to compute the area of land for which human habitation and farm production are temporarily restricted or permanently restricted (condemned). This metric can be reported over any user-specified circular area or ring. This metric is more useful for characterizing the extent of land over which long-term phase protective actions are necessary.

An additional consequence metric is the population subject to long-term protective actions, including land and property interdiction and condemnation. In addition to the area of land that must be restricted from human habitation and farm production, this metric is useful for assessing the societal consequence of an accident.

4.1.3.3 Economic Consequences

Every protective action modeled in MACCS to reduce radiation exposures to the public (except for sheltering and potassium iodide (KI) ingestion[14] during the early phase) has an associated cost. The offsite economic consequence model in MACCS sums the costs for protective actions over the region of interest and includes the six categories as follows. (Costs for medical care, life-shortening, and litigation are not calculated by MACCS.)

1. Evacuation and relocation costs. These are per-diem costs associated with the population that is temporarily relocated. This includes the population that is explicitly modeled to evacuate during the early phase, the population that would relocate during the early phase due to the hotspot or normal relocation criteria, and the population that would need to be relocated for the entire duration of the intermediate phase. These costs are calculated by

[14] There is a cost to maintain a KI acquisition, distribution, and awareness program; however, this cost is incurred regardless of the occurrence of the accident and is therefore not included in the offsite cost calculation.

adding up the number of displaced people times the number of days they are displaced from their homes.

2. Moving expenses for people displaced. This is a one-time moving expense for the population displaced from their homes because of long-term phase relocation for decontamination, interdiction, or condemnation. The modeling can include loss of wages.
3. Decontamination costs. These are the costs associated with decontaminating farm and nonfarm property and include labor, materials, and equipment for performing the decontamination as well as the cost to dispose of the contaminants. They depend on the population and size of the area that needs to be decontaminated as well as the level of decontamination that needs to be performed. The model estimates the costs only if decontamination is cost effective.
4. Costs due to loss of use of property. These costs are associated with the lost return on investment and on depreciation caused by lack of routine maintenance for farm and nonfarm properties during the period of interdiction, the time when the property cannot be used.
5. Disposal of contaminated food grown or produced locally. For farmland that exceeds the farmability criterion and is modeled to be temporarily unable to produce crops, meat, and dairy products that are suitable for human consumption, this food must be disposed for the current growing season. This cost sums the expected food sales per area times the affected area. The site data file includes a parameter to estimate the fraction of annual farm sales for dairy products and thus the disposal cost for dairy products is reported separately from the disposal cost for all other agricultural products.
6. Cost of condemned lands. For farmland and nonfarmland that cannot be restored to usefulness or is not cost-effective to do so, the land is condemned. These are costs of condemning property, i.e., the value of the property condemned.

All of the costs for the six cost categories are summed over the entire region of interest affected by the atmospheric release to get the total offsite economic costs. Many of the values affecting the economic cost model are user inputs and thus can account for a variety of costs and can be adjusted for inflation, new technology, or changes in policy. Other data for the cost model come from the site file, which is discussed in more detail in section 3.3. The site file uses external data from the U.S. Census, the U.S. Bureau of Economic Analysis, and the U.S. Agricultural Census. These values are scaled to the year of interest.

4.2 Modeling Approach

The CPRR technical evaluation is designed to be applicable to all Mark I and II BWR containment sites in the United States. There is considerable variation in many different characteristics at the 15 Mark I and 5 Mark II sites. Sites differ with respect to population (number and distribution); economic values; land use (land vs. water, farmland vs. developed land, etc.); weather (wind, precipitation, etc.); emergency response characteristics (time to evacuate 10-mile emergency planning zone (EPZ), use of potassium iodide (KI), etc.); long-term protective actions (habitability criterion); and many others. To capture the variation among these characteristics in the most resource-efficient manner, one site-specific Mark I reference MACCS model was developed and one site-specific Mark II reference model was developed. These reference models were then adapted in a series of over 100 sensitivity calculations to assess the potential impact of site-specific parameters on offsite consequence results.

The Peach Bottom Atomic Power Station and the Limerick Generating Station were selected as the site-specific reference models to enable greater modeling fidelity for the high population sites (Peach Bottom has the second highest population within a 50 mile radius among the 15 Mark I sites and Limerick has the highest population within a 50 mile radius among the five Mark II sites). The modeling approach used the most current sources of information and consequence modeling best practices to make the Peach Bottom and Limerick MACCS models as realistic as practically possible.

This section discusses the modeling approach for each of the components of the analysis. Greater detail is provided for those modeling parameter selections that differ from those used in SOARCA, SECY-12-0157, and other recent analyses.

4.2.1 Radionuclide Release

The preceding MELCOR accident progression modeling section describes the assumptions, calculations, and results of the different accident progression scenarios. For the reference BWR Mark I site MELCOR model (Peach Bottom), 41 unique MELCOR simulations were run. For the reference BWR Mark II site MELCOR model (LaSalle), 12 unique MELCOR simulations were run. MELCOR provides the following data for each source term:

- Time-dependent release fraction of 9 chemical groups for each MELCOR release pathway: Noble Gases (Xe), Alkali Metals (Cs), Alkali Earths (Ba), Halogens (I), Chalcogens (Te), Platinoids (Ru), Early Transition Elements (Mo), Tetravalents (Ce), and Trivalents (La)
- Time-independent distribution by particle size diameter for 10 aerosol size bins characterized by geometric mean diameters from 0.15 μm to 41.2 μm for each chemical group
- Height of each MELCOR release pathway
- Time-dependent plume rise data including rate of release of sensible heat (W), mass flow (kg/s), and gas density (kg/m^3)

In addition, the following data is needed to characterize each source term. For each item, the modeling approach is described.

- Radionuclides to include in consequence analysis and their assignment to chemical groups
    - Both the BWR Mark I reference site MACCS model and the BWR Mark II reference site MACCS model use the identical modeling approach as the SOARCA project for both the Peach Bottom and Surry analyses. The full details are available in Table A-1 of NUREG/CR-7110, Vol. 1, Rev. 1 [10].

- Pseudostable radionuclides to include in consequence analysis
    - Both the BWR Mark I reference site MACCS model and the BWR Mark II reference site MACCS model use the identical modeling approach as the SOARCA project for both the Peach Bottom and Surry analyses. The full details are available in Table B-1 of NUREG/CR-7110, Vol. 1, Rev. 1 [10].

- Radionuclide inventory (activity) at the time of reactor shutdown

- The BWR Mark I reference site MACCS model uses the identical modeling approach as the SOARCA project for the Peach Bottom analyses. The inventory was computed using the SCALE code to model the specific fuel management strategy used at Peach Bottom. The SCALE model assumed that the accident occurs mid-cycle and that the peak fuel rod burnup is 49 MWd/kg fuel. Both radial and axial variations in burnup were modeled with SCALE. The inventory was calculated by integrating the isotopic inventory over the whole core.
- The BWR Mark II reference site MACCS model uses the same radionuclide inventory as the Mark I site reference MACCS model because the reference plants are very similar and have essentially identical thermal power output; Peach Bottom is licensed at 3,514 MW_t and Limerick is licensed at 3,515 MW_t [20]. In addition, the same mid-cycle accident timing assumption is made for the Mark II reference site. The full details are available in Table A-2 through Table A-10 of NUREG/CR-7110, Vol. 1, Rev. 1 [10].

- Plume segmentation characteristics including duration and thresholds for inclusion
    - Consistent with the SOARCA project, the radionuclide release is divided into hourly plume segments to be consistent with the resolution of the accompanying meteorological data. Also consistent with the SOARCA project, 0.001 is used as the threshold for inclusion of each MELCOR release pathway and each plume segment. Thus, a MELCOR release pathway is used only if more than 0.001 (0.1%) of the total release of any chemical group occurs through it. A plume segment is evaluated only if any of the chemical groups in that segment contribute 0.001 (0.1%) to the total release of that chemical group.

- Identification of the most risk-significant plume segment
    - For each source term, the most risk-significant plume segment needs to be identified to align the release with the weather data for each weather bin. The risk-significant plume segment is considered to be the one that causes the highest risk for early fatalities. The modeling approach was to select the plume segment among the first few that has the highest iodine chemical group release fraction. All representative cases for the Mark I and Mark II source term bins used either the first or second plume segment as most risk-significant.

- Building height and initial vertical and horizontal plume size to characterize the initial dispersion of the plume and address building wake effects
    - Both the BWR Mark I and II reference plants are approximately 50 m in height and width. Consistent with the SOARCA project, the initial horizontal dispersion uses the equation $\sigma_{y0} = 0.23 * W_b$ and the initial vertical dispersion uses the equation $\sigma_{z0} = 0.47 * H_b$ [4]. Therefore both BWR Mark I and II MACCS models use an initial horizontal dispersion of 11.6 m and vertical dispersion of 23.3 m.

- Ground height in the MELCOR reference frame used to adjust MELCOR release heights to release heights relative to grade
    - The BWR Mark I analyses use the Peach Bottom MELCOR model, which has grade level at an elevation of -4.04 m. The BWR Mark II analyses use the LaSalle MELCOR model, which has grade level at an elevation of -14.5 m. These values were obtained by reviewing plant schematic diagrams and MELCOR model nodalization diagrams.

4.2.1.1 Modeling the Decontamination Factor Provided by an External Filter

The CPRR rulemaking technical analysis for BWR Mark I and Mark II plants includes options for small and large external engineered filters attached to wetwell and drywell vents. MELCOR reference plant models allow for very detailed modeling of accident progression and they model filtration by the wetwell/suppression pool; however, they do not currently provide mechanistic modeling of the presence of external filters. Therefore the filtration provided by an external filter is modeled by applying a decontamination factor (DF) to the MELCOR source term. In many accident progression cases, a direct pathway to the environment other than a wetwell or drywell vent path is created. For these cases, the DF is applied only to the wetwell and/or drywell vent release paths. The presence of a release pathway other than the wetwell or drywell vent and the fraction that goes through this pathway, are therefore major drivers of the potential effectiveness of an external filter. This topic is discussed further in Section 3.1.3.

In the absence of a research program demonstrating the effectiveness of different filter designs for removing aerosols across the particle size spectrum, the team decided to consider a range of decontamination factors to apply to all source terms. The range includes a DF of 10, 100, and 1000 (which is consistent with the range reported in the OECD/NEA/CSNI Status Report on Filtered Containment Venting [21]) and each is applied uniformly among all chemical groups despite their different particle size distributions. An attempt to identify chemical group-specific DFs based on their particle size distribution was considered for this analysis but is being deferred to future analyses based on schedule and resource constraints.

4.2.1.2 Source Term Binning Process

The CPRR rulemaking accident progression analysis produced 41 Mark I source terms and 12 Mark II source terms. Considering the range of external filter DFs for each (unfiltered, DF = 10, DF = 100, and DF = 1000), this yields 164 Mark I source terms and 48 Mark II source terms. Many of the source terms have very similar release fraction and release timing characteristics, so rather than running all 212 source terms in MACCS, a binning strategy was developed for each containment type. The binning process was based on cumulative cesium and iodine release fractions, because the cesium group is most important for long-term offsite consequences and the iodine group is most important for early offsite consequences. The time of release of the risk-significant plume segment was also considered. However, in all cases it was after the time at which the 10-mile emergency planning zone (EPZ) is expected to be evacuated, and therefore was not considered as important. For example, for Peach Bottom, the EPZ evacuation is expected to be completed about four hours after the time of notification [22]. (This topic is discussed in more detail in Section 3.6.2). Assuming public notification via general emergency (GE) siren at 1.5 hours after the accident, the EPZ is cleared in 5.5 hours. In comparison, the earliest risk-significant plume segment starts at 7.3 hours from the start of the accident and is more generally in the range of 10 to 24 hours from the start of the accident.

The source term bins were developed based on logarithmic spacing with finer resolution at higher magnitudes. The bin definitions for the Mark I source terms are provided in Table 4-2 and they span a very large (> 4 orders of magnitude) range of cesium release from 0.0006% to about 16%. The representative source term from each bin was selected by choosing the source term that was most similar to the average cesium and iodine release fractions for all the source terms in that bin. Table 4-2 shows the summary source term information for each of the 18 Mark I source term bins.Table 4-3 identifies which source term bin corresponds to each of the 164 unique Mark I source terms.

The binning strategy for the Mark II source terms was very similar to that used for the Mark I source terms but since there were far fewer source term cases, the bin spacing for iodine release fraction was chosen to be somewhat discontinuous for the highest consequence bins. The cesium and iodine release fractions were used to group the source terms into bins. Consistent with the Mark I source terms, the start of release to the environment was sufficiently later than the expected EPZ evacuation completion time so the Mark II source terms were not binned considering release start timing. Table 4-4 shows the summary source term information for each of the 9 Mark II source term bins. Table 4-5 identifies which source term bin corresponds to each of the 48 unique Mark II source terms. Additional details for each individual source term case are provided in Appendix A.

Table 4-2 Binning strategy for Mark I source terms

Bin	Bin Cs Range (%)	Bin I Range (%)	Representative Case	Rep Case Cs (%)	Rep Case I (%)	Start of Release to Environment (hours)
1	0.0002 - 0.001	0.001 - 0.01	28DF1000	0.0006%	0.006%	15.9
2	0.001 - 0.003	0.01 - 0.03	48DF100	0.002%	0.02%	11.4
3	0.003 - 0.01	0.03 - 0.1	10DF100	0.01%	0.08%	16.3
4	0.01 - 0.03	0.1 - 0.3	7DF1000	0.02%	0.26%	14.9
5	0.03 - 0.1	0.3 - 1.0	11DF10	0.06%	0.78%	14.4
6	0.1 - 0.3	1.0 - 3.0	48	0.23%	1.69%	11.4
7	0.3 - 1.0	3.0 - 10.0	15	0.60%	5.85%	15.9
8	0.3 - 1.0	10.0 - 20.0	46	0.98%	11.01%	14.8
9	1.0 - 2.0	2.0 - 4.0	5DF10	1.05%	2.89%	24.2
10	1.0 - 2.0	4.0 - 10.0	5	1.39%	6.46%	24.2
11	1.0 - 2.0	10.0 - 20.0	8	1.49%	19.25%	14.9
12	1.0 - 2.0	20.0 - 40.0	1	1.93%	22.68%	14.9
13	2.0 - 4.0	3.0 - 10.0	41DF1000	3.40%	7.65%	9.8
14	2.0 - 4.0	10.0 - 20.0	22dw	2.82%	18.64%	15.9
15	2.0 - 4.0	20.0 - 40.0	53	2.79%	29.05%	18.4
16	4.0 - 10.0	10.0 - 20.0	41	4.54%	14.10%	9.8
17	4.0 - 10.0	20.0 - 40.0	3DF10	8.85%	24.65%	9.8
18	10.0 - 20.0	20.0 - 40.0	52	15.90%	34.32%	18.4

Table 4-3 Identification of source term bin for each Mark I source term case

Case	Bin	Case	Bin	Case	Bin	Case	Bin	Case	Bin	Case	Bin
1	12	7	11	13	7	22dw	14	41	16	48	6
1DF10	7	7DF10	6	13DF10	5	22dwDF10	6	41DF10	13	48DF10	4
1DF100	7	7DF100	5	13DF100	3	22dwDF100	4	41DF100	13	48DF100	2
1DF1000	7	7DF1000	4	13DF1000	1	22dwDF1000	2	41DF1000	13	48DF1000	1
1S1	7	7dw	11	14	7	24	7	42	16	49	7
1S1DF10	7	7dwDF10	6	14DF10	5	24DF10	5	42DF10	13	49DF10	5
1S1DF100	6	7dwDF100	4	14DF100	4	24DF100	4	42DF100	13	49DF100	3
1S1DF1000	6	7dwDF1000	2	14DF1000	4	24DF1000	2	42DF1000	13	49DF1000	2
2	15	8	11	15	7	24dw	13	43	16	50	6
2DF10	10	8DF10	6	15DF10	5	24dwDF10	6	43DF10	13	50DF10	4
2DF100	10	8DF100	4	15DF100	3	24dwDF100	4	43DF100	13	50DF100	2
2DF1000	10	8DF1000	2	15DF1000	1	24dwDF1000	2	43DF1000	13	50DF1000	1
3	17	9	7	16	7	27	7	44	16	51	8
3DF10	17	9DF10	5	16DF10	5	27DF10	5	44DF10	13	51DF10	5
3DF100	17	9DF100	3	16DF100	3	27DF100	4	44DF100	13	51DF100	3
3DF1000	17	9DF1000	1	16DF1000	2	27DF1000	4	44DF1000	13	51DF1000	1
4	10	10	7	18	7	28	7	45	7	52	18
4DF10	10	10DF10	5	18DF10	5	28DF10	5	45DF10	5	52DF10	13
4DF100	10	10DF100	3	18DF100	3	28DF100	3	45DF100	3	52DF100	13
4DF1000	10	10DF1000	1	18DF1000	3	28DF1000	1	45DF1000	1	52DF1000	13
5	10	11	7	21	11	30	7	46	8	53	15
5DF10	9	11DF10	5	21DF10	6	30DF10	5	46DF10	5	53DF10	6
5DF100	9	11DF100	3	21DF100	5	30DF100	3	46DF100	3	53DF100	4
5DF1000	9	11DF1000	2	21DF1000	5	30DF1000	2	46DF1000	2	53DF1000	2
6	12	12	11	22	12	32	7	47	6		
6DF10	7	12DF10	6	22DF10	6	32DF10	5	47DF10	4		
6DF100	7	12DF100	4	22DF100	4	32DF100	3	47DF100	2		
6DF1000	7	12DF1000	2	22DF1000	2	32DF1000	1	47DF1000	1		

Table 4-4 Binning strategy for Mark II source terms

Bin	Bin Cs Range (%)	Bin I Range (%)	Representative Case	Rep Case Cs (%)	Rep Case I (%)	Start of Release to Environment (hours)
1	0.00001 - 0.0001	0.0001 - 0.001	11DF1000	0.00004%	0.0005%	20.3
2	0.0001 - 0.001	0.001 - 0.01	5DF1000	0.0006%	0.005%	32.2
3	0.001 - 0.01	0.01 - 0.1	42DF100	0.0043%	0.037%	14.3
4	0.01 - 0.1	0.1 - 1.0	11	0.042%	0.45%	20.3
5	0.1 - 0.4	1.0 - 3.0	51DF10	0.23%	2.01%	16.6
6	0.4 - 1.0	3.0 - 10.0	5	0.55%	4.94%	32.2
7	1.0 - 2.0	~ 10.0	3	1.09%	10.26%	14.3
8	2.0 - 3.0	~ 20.0	1	2.46%	19.81%	22.8
9	3.0 - 4.0	~ 30.0	52	3.57%	28.67%	16.6

Table 4-5 Identification of source term bin for each Mark II source term case

Case	Bin	Case	Bin	Case	Bin	Case	Bin	Case	Bin	Case	Bin
1	8	5	6	11	4	42	6	45	8	51	8
1DF10	5	5DF10	4	11DF10	3	42DF10	4	45DF10	5	51DF10	5
1DF100	4	5DF100	3	11DF100	2	42DF100	3	45DF100	4	51DF100	4
1DF1000	3	5DF1000	2	11DF1000	1	42DF1000	2	45DF1000	3	51DF1000	3
3	7	10	5	24	6	44	6	49	6	52	9
3DF10	5	10DF10	4	24DF10	4	44DF10	4	49DF10	4	52DF10	5
3DF100	4	10DF100	3	24DF100	3	44DF100	3	49DF100	3	52DF100	4
3DF1000	3	10DF1000	2	24DF1000	2	44DF1000	2	49DF1000	2	52DF1000	3

4.2.1.3 External Filter Effectiveness

For accident progression cases in which all releases to the environment are through a vent path, an external filter can reduce the source term and offsite consequences. However, for accident progression cases that lead to containment failure, for example, via drywell liner

melt-through (DW LMT) or main steam line creep rupture (MSLCR), an external filter is less effective. Table 4-6 shows some of the variation in external filter effectiveness through examples of three MELCOR Mark I cases.

MELCOR case 1 is shown as an example in which there is no post-core-damage external water addition (either because the plant lacks the capability or because external water additional is unsuccessful) and the accident results in an uncontrolled release via DW LMT. In MELCOR case 1, much of the release (78.2% of the cesium) is through a vent pathway[15] so the external filter can

[15] Chapter 3 provides a discussion of the reasons for why the filterable release percent varies among the different MELCOR cases resulting in DW LMT.

substantially reduce the environmental release. However, the incremental benefit of increasing the external filter DF becomes very small.

For a similar but less likely case in which the containment fails instead via MSLCR, most of the release is uncontrolled and goes through an unvented pathway so the external filter has a very small effect on the total source term released to the environment. Even though the release to the environment continues to decrease as the external filter DF increases, the source term remains within the existing source term bin (Mark I bin 17) so the offsite consequences remain essentially unchanged.

For MELCOR Mark I Case 10 in which external water addition is successful, all of the released cesium flows through a vented pathway and therefore the external filter can potentially reduce the environmental release. Note that the MACCS source term bin number decreases with each incremental DF applied.

Table 4-6 External filter effectiveness for three example BWR Mark I cases

CPRR Alternatives	MELCOR Mark I Case and External Filter DF	Percent of Source Term Released Through Vented Pathway		Total Source Term Released to Environment		MACCS Source Term Bin	Description of External Filter Effectiveness
		Cesium	Iodine	Cesium	Iodine		
No External Water Addition Resulting in DW LMT	1	78.2%	85.5%	1.93%	22.70%	12	External filter has a notable effect on reducing environmental release for DF=10 but smaller incremental benefit for higher DF
	1DF10			0.57%	5.24%	7	
	1DF100			0.44%	3.49%	7	
	1DF1000			0.42%	3.32%	7	
No External Water Addition resulting in MSLCR	3	11.5%	21.6%	9.88%	30.20%	17	External filter has an insignificant effect on reducing environmental release
	3DF10			8.85%	24.32%	17	
	3DF100			8.75%	23.74%	17	
	3DF1000			8.74%	23.68%	17	
External Water Addition Successful	10	100.0%	100.0%	0.72%	8.04%	7	External filter reduces environmental release
	10DF10			0.07%	0.80%	5	
	10DF100			0.007%	0.08%	3	
	10DF1000			0.0007%	0.008%	1	

4.2.2 Calculational Grid

MACCS uses a polar grid to model radionuclide releases to the environment, exposures to people, land contamination, and protective actions of people and land. MACCS allows the user to divide the grid into 16, 32, 48, or 64 angular sectors. Consistent with modeling best practices and recent consequence analyses studies, 64 angular sectors were used in this project because this provides the greatest resolution.

MACCS allows the user to divide the polar grid into a maximum of 35 radial rings. For this project, 26 radial sectors were selected. The boundaries of each sector are described in Table 4-7. The

boundaries are closer together nearer to the site to enable higher resolution at areas of greater interest. Boundaries are selected to be consistent with certain areas of interest for the analysis. For example, the radial boundary at 1.33 miles from the site is used to approximate the area within a mile of the site boundary, 10 miles is used to approximate the plume exposure EPZ, 15 miles is used to approximate the outer distance for shadow evacuation (see Section 3.6.2 for discussion), and 50 miles is used to approximate the ingestion exposure EPZ.

The use of 64 angular sectors and 26 radial rings enables results to be reported over 1,664 grid elements. A discussion on which grid elements are used for the output metrics is provided later in Section 4.

Table 4-7 Radial boundaries used in the polar calculational grid

Radial Boundary Number	Radial Boundary Distance (mi)	Radial Boundary Distance (km)	Radial Boundary Number	Radial Boundary Distance (mi)	Radial Boundary Distance (km)
1	0.1	0.16	14	15	24.1
2	0.33	0.53	15	20	32.2
3	0.75	1.21	16	25	40.2
4	1	1.61	17	30	48.3
5	1.33	2.1	18	40	64.4
6	2	3.2	19	50	80.5
7	2.5	4.0	20	70	113
8	3	4.8	21	100	161
9	3.5	5.6	22	150	241
10	5	8.0	23	200	322
11	7.5	12.1	24	350	563
12	10	16.1	25	500	805
13	12.5	20.1	26	1000	1610

4.2.3 Site Data

The SecPop preprocessor code (version 4.3) was used to generate site data that is needed for the consequence calculations. SecPop accesses external population, land use, and economic databases to obtain the data and then uses various algorithms to map the data to each of the 1,664 individual grid elements. The data types and sources are summarized in Table 4-8. More detail on the data sources and the algorithms used are provided in the SecPop reference manual, NUREG/CR-6525, Rev. 1 [5].

Table 4-8 Site data types and sources

Data Type	Source
Population	• U.S. Census Bureau, 2010 Census
Land fraction	• U.S. Census Bureau, 2010 Census
Fraction of land used for farming	• U.S. Department of Agriculture, 2007 Census of Agriculture
Farm value per hectare	• U.S. Department of Agriculture, 2007 Census of Agriculture
Annual farm sales per hectare	• U.S. Department of Agriculture, 2007 Census of Agriculture
Fraction of annual farm sales from dairy products	• U.S. Department of Agriculture, 2007 Census of Agriculture
Non-farm wealth per capita	• U.S. Census Bureau, 2010 Census • U.S. Census Bureau, 2007 American Housing Survey • U.S. Department of Agriculture, Economic Research Service • U.S. Department of Agriculture, Natural Resources Conservation Service, 2007 National Resources Inventory Report • U.S. Department of Commerce, Bureau of Economic Analysis

Population data was scaled forward from 2010 to the year of interest based on state level population growth data available from the U.S. Census Bureau. The year of interest was selected as 2013 because it was the most recent year for which population projection and consumer price index data were available. For sites such as Peach Bottom and Limerick, where the 50-mile radial area includes multiple states, an approximate land fraction was used to weight the state-specific population growth rates. This is presented in Table 4-9 and shows a weighted average value of 1.016 for Peach Bottom and 1.009 for Limerick. SecPop reads population data at the census block level and uses various algorithms to map the data to the user-specified MACCS polar grid.

Table 4-9 Population growth multipliers for Peach Bottom and Limerick

State	Census 2010	Census 2013 Est.	2010 to 2013 Multiplier	Peach Bottom Approximate 50-mile Area Fraction	Limerick Approximate 50-mile Area Fraction
PA	12,702,379	12,773,801	1.006	0.5	0.7
MD	5,773,552	5,928,814	1.027	0.35	0.05
DE	897,934	925,749	1.031	0.1	0.05
NJ	8,791,894	8,899,339	1.012	0.05	0.2
Population Multiplier for 2010 to 2013:				**1.016**	**1.009**

Economic values are based on 2007 data and are scaled to 2013 using the consumer price index for all urban consumers (CPI-U) [23]. Based on a national CPI-U of 207.342 in 2007 and a national CPI-U of 232.957 in 2013, a multiplier of 1.124 is used for all economic values.

4.2.4 Meteorological Data

The atmospheric transport and dispersion model in MACCS relies on the following types of meteorological data for the following purposes. Data for wind direction, wind speed, precipitation, and stability class can be provided hourly or every 15 or 30 minutes and should span an entire year to capture daily and seasonal variations.

- Hourly wind direction data is used to identify the direction each plume segment will travel
- Hourly wind speed data is used to characterize the speed of plume travel away from the site and the plume meander factor. Plume speed is adjusted each hour for each plume segment.
- Hourly precipitation data is used to determine the timing and magnitude of wet deposition and the timing of reduced evacuation travel speeds
- Hourly Pasquill-Gifford stability class data is used to characterize the dispersion of the plume in the vertical and cross-wind directions and the plume meander factor. Stability class is adjusted each hour for each plume segment.
- Diurnal (morning and afternoon) seasonal mixing layer height data is used to determine the upper boundary of the region in which each plume may expand.

4.2.4.1 Hourly Meteorological Data

The hourly data was developed through an analysis of the raw weather data from site meteorological towers provided by Exelon, the licensee of Peach Bottom and Limerick. Exelon provided raw weather data for Peach Bottom for 2005 and 2006 and for Limerick for 2012 and 2013. MACCS requires meteorological data to be provided for each time point of the entire year (8,760 hourly data points for a non-leap year). Therefore missing data was reviewed and was filled in by NRC meteorologists and in accordance with EPA's "Meteorological Monitoring Guidance for Regulatory Applications" [24]. For Peach Bottom and Limerick, the 2006 and 2013 years, respectively, were chosen because they had the higher data recovery rates of the two years available. A summary of meteorological statistics for the data sets is provided in Table 4-10. The wind rose for each data set is provided in Figure 4-1 on a 16 sector grid. These show that for both sets, the most common wind direction is toward the southeast.

Table 4-10 Summary of meteorological data sets used for Peach Bottom and Limerick

		Peach Bottom Year 2006	Limerick Year 2013
Average Wind Speed (m/s)		2.12	2.36
Precipitation	Total (in)	44.42	44.92
	Hours	602	650
	Frequency (%)	6.87%	7.42%
Stability Class Frequency (%)	Unstable	17.75%	7.33%
	Neutral	24.57%	47.91%
	Stable	57.68%	44.76%
Joint Data Recovery (%)		99.25%	95.19%

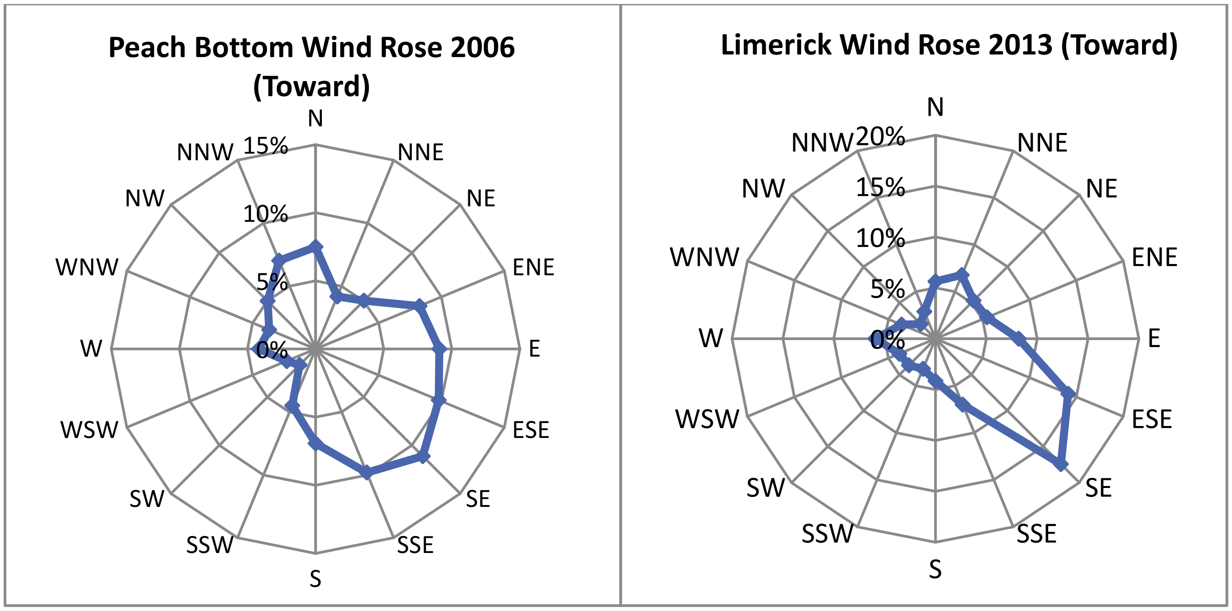


Figure 4-1 Wind Rose for Peach Bottom 2006 and Limerick 2013

The methodology described in NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data” [25] was used to perform quality assurance evaluations of all meteorological data. In accordance with Regulatory Guide 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” [26], a data recovery rate greater than 90% was achieved for the wind speed, wind direction, and atmospheric stability parameters. In addition, atmospheric stability data was reviewed to determine if the time of occurrence and duration of reported stability conditions were generally consistent with expected meteorological conditions (e.g., neutral and slightly stable conditions predominating during the year with stable and neutral conditions occurring at night and unstable and neutral conditions occurring during the day).

The weather data was also reviewed against data from other years to assess how representative the chosen year seemed for the site’s climate. Data from other sources, such as annual effluent reports and dose assessment reports, for ~10 other years were analyzed and compared to the selected year and showed that they were indeed representative of the climate conditions.

4.2.4.2 Mixing Height Data

In addition to hourly observation data, the MACCS meteorological file requires morning and afternoon mixing height data for four meteorological seasons, for a total of eight entries. The morning mixing height is the minimum mixing height used in the code, and the afternoon mixing height is the maximum mixing height. MACCS uses the site longitude and latitude coordinates to determine the time of sunrise and sunset and it estimates the mixing height by linear interpolation between the minimum and the maximum, based on the time of day. Mixing height data is based on upper air measurements that are only available at selected locations across the United States. The mixing height data used for this analysis came from the U.S. Environmental Protection Agency’s (EPA’s) SCRAM database [27] for the nearest weather stations to Peach Bottom and Limerick at Pittsburgh, PA, and Sterling, VA, for the three most recent years available, 1989-1991. Data from January through March was used for winter, April-June for spring, July-September for summer, and October-December for autumn. In addition, the EPA report, “Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution throughout the Contiguous United States,” [28],

which provides mixing height contour maps across the U.S., was used to provide an estimate of the mixing heights. Data from these sources were averaged to yield the mixing heights used in the analysis. The values are rounded to the nearest tens of meters to comply with the MACCS formatting requirements and these are provided in Table 4-11.

Table 4-11 Seasonal diurnal mixing heights (m) used for Peach Bottom and Limerick

	Winter	Spring	Summer	Fall
Morning	760	650	500	570
Afternoon	1000	1700	1680	1130

4.2.4.3 Weather Bin Sampling

In MACCS, a weather trial is defined by the starting hour of weather data from the meteorological data file and each trial uses as many of the following hours as are required to transport plume segments through and out of the computational grid. Rather than using all 8,760 available weather trials (corresponding to each starting hour of the year) in an offsite consequence analysis, weather bins are defined to categorize similar sets of weather conditions, and a sampling approach is used to randomly select the weather trials within each bin. This analysis used the nonuniform weather bin sampling approach and yielded approximately 1,000 weather samples based on 36 weather bins. The 36 weather bins are based on wind speed, stability class, and the occurrence of precipitation. Sixteen bins were defined based on combinations of stability class and wind speed and the remaining 20 bins were defined based on rain occurrence and intensity at various downwind locations within 20 miles of the site. The parameters used to define the bins are the same as those used in previous studies such as SOARCA [9] and are documented in NUREG/CR-7009 [13].

The nonuniform sampling approach allows the user to specify a different number of random samples for each bin. Consistent with the SOARCA project, the number of trials selected from each bin is the maximum of 12 trials and 10% of the number of trials in the bin. For bins containing fewer than 12 trials, all of the trials within the bin are used for sampling. This strategy resulted in 984 weather trials for Peach Bottom and 963 for Limerick.

4.2.4.4 Boundary Weather

Boundary weather refers to the use of artificial weather conditions for the outermost ring of the calculational grid. Continuous rain is specified in the ring spanning 500-1000 miles as a way to ensure that all of the radionuclides released into the atmosphere will be deposited within the computational domain. This is clearly unrealistic and therefore consequences for the 500-1000-mile ring are not presented in this report.

4.2.5 Atmospheric Transport and Dispersion

MACCS models dispersion of radioactive materials released into the atmosphere using the straight-line Gaussian plume segment model with provisions for meander and surface roughness effects. The ATD model treats the following: plume rise resulting from the sensible heat content (i.e., buoyancy), initial plume size caused by building wake effects, release of up to 200 plume segments, dispersion under statistically representative meteorological conditions, deposition under dry and wet (precipitation) conditions, and decay and ingrowths of up to 150 radionuclides

and a maximum of six generations. The model does not treat in detail irregular terrain and spatial variations in the wind field. The model considers temporal variations in wind direction at the time each plume segment is released. Once a plume segment moves in the direction of the initial wind, the plume segment is modeled to continue in that same direction even if the wind field changes. However the next plume segment released at the next time step (typically 1 hour) would move in a different direction if the wind field changed.

4.2.5.1 Dispersion Parameters

The Gaussian plume segment model uses two spatially dependent dispersion parameters to estimate the atmospheric dispersion, σ_y for the horizontal, cross-wind dispersion and σ_z for the vertical dispersion. These parameters are specified for each stability condition as a function of downwind distance and can be modeled in two ways, either using power-law functions or a lookup table.

The growth of plume dimensions during downwind transport to short distances (1 km) has been experimentally determined [29] over flat terrain covered by prairie grass for short release durations (10 min) during stable, neutral, and unstable atmospheric conditions. Pasquill used this data to develop curves that depict the increase of plume dimensions (σ_y and σ_z values) with downwind distance for each of the six Pasquill-Gifford stability classes used in MACCS, A-F. Although measurements had only been made to 1 km, Pasquill extrapolated the curves to 100 km and they have subsequently been extrapolated farther. Tadmor and Gur [30] developed a power-law correlation to fit to the original experimental data. Eimutis and Konicek developed a separate correlation to better represent the different power-law fit coefficients for different distance ranges [31]. The Eimutis and Konicek formulation was recently converted into a MACCS lookup table by Bixler, Napier, and Rishel [32].

For horizontal, cross-wind dispersion, the coefficient has a relatively constant slope on a log-log plot of distance and both the power-law and lookup table approaches approximate the prairie grass experimental data similarly well. However, the dispersion coefficient has a more nonlinear slope as a function of distance and stability class on a log-log plot and therefore the lookup table approach can approximate vertical dispersion better than power law functions. Despite this, power law functions have the advantage of requiring far fewer values to specify dispersion, and therefore uncertainty analysis sampling is made much more efficient. Power-law functions based on Tadmor and Gur have been commonly used in past studies such as SOARCA for this reason. Uncertainty in these parameters, along with a variety of others, was propagated in the SOARCA Peach Bottom Uncertainty Analysis [14]. Because this project is not aimed at re-characterizing dispersion parameter uncertainty, the lookup table approach based on the Bixler et al. conversion of the Eimutis and Konicek formulation was used because it best approximates vertical dispersion in addition to horizontal dispersion.

4.2.5.2 Surface Roughness

The surface roughness of a land area characterizes the amount of interaction a plume would have with the ground based on topographical features such as row crops, trees, and houses. MACCS requires a single surface roughness value for each site studied. To select a single value, the USDA CropScape database [33] was accessed, which provides land use data for the EPA land use categories [34] for any user-specified area in the continental U.S. The land use area fractions were used to weight various typical land use surface roughness values to yield one representative value for each site. This process is summarized in Table 4-12, which led to a value of 26 cm for Peach Bottom and 33 cm for Limerick. Land use data for a circular area of approximately 30-mile radius from each site was chosen because the 30-mile distance is considered a reasonable

estimate of the distance plumes travel before becoming well-mixed between the ground and the mixing height layer. The typical surface roughness values for the 10 land types are from DOE/RL/87-09, "The Remedial Action Priority System (RAPS): Mathematical Formulations, 1987 [35] and NUREG/CR-7110 [10].

Table 4-12 Surface roughness calculation for Peach Bottom and Limerick

Land Use Type	Typical Surface Roughness (cm)	Peach Bottom Land Fraction	Limerick Land Fraction
Open Water	0.03	2.12%	0.77%
Barren	1	0.15%	0.13%
Grass/Pasture	3	23.46%	17.38%
Developed/Open Space and Developed/Low Intensity	5	17.39%	27.50%
Shrubland	5	1.45%	2.65%
Wetlands	5	2.41%	0.40%
Farmland	14	20.73%	12.76%
Forest	60	29.40%	31.12%
Developed/Medium Intensity	70	2.06%	4.96%
Developed/High Intensity	300	0.84%	2.32%
Weighted Average Site Surface Roughness:		**26 cm**	**33 cm**

In MACCS, surface roughness affects both vertical dispersion and dry deposition velocities. The effect on vertical dispersion has traditionally been modeled by means of a multiplicative factor. The empirical expression for this factor is the ratio of surface roughness at the site in question to a standard value of surface roughness to the 1/5th power. Most of the data upon which empirical dispersion models have been based were taken at a site characterized by prairie grass [29], which was estimated to have a surface roughness of 3 cm. Thus, the empirical equation used to scale vertical dispersion uses the actual surface roughness divided by 3 cm to the 1/5th power. The standard multiplicative factor for Peach Bottom corresponding to a 26 cm surface roughness is $(26 / 3)^{0.2} = 1.54$ and for Limerick corresponding to a 33 cm surface roughness is 1.62. The effect of surface roughness on dry deposition velocities is described in the following section.

4.2.5.3 Dry Deposition

The dry deposition velocity of an aerosol particle is a function of particle size and of the degree of turbulence in the atmosphere, which is affected by wind speed and surface roughness. The effect of surface roughness on deposition velocity has been characterized by Bixler et al. in NUREG/CR-7161 [36] based on expert elicitation data in NUREG/CR-6545 [37]. Bixler et al. provides a set of correlations for estimating deposition velocity as a function of aerosol diameter, wind speed, surface roughness, and percentile representing degree of belief by the experts. This correlation is valid for aerosol diameters up to about 20 μm and surface roughness up to about 60 cm. For the largest particle size bin of 41.2 μm, the correlation is no longer valid so the effect of gravitation settling alone is considered. The mean wind speed used in the correlation was taken from the year of weather data selected, 2006 for Peach Bottom (2.12 m/s) and 2013 for Limerick (2.36

m/s). Table 4-13 provides the set of dry deposition velocities used in the analysis for Peach Bottom and Limerick for the 10 aerosol particle size bins calculated by MELCOR.

Table 4-13 Dry deposition velocities for Peach Bottom and Limerick

Particle Size Bin	Particle Diameter (μm)	Dry Deposition Velocities (cm/s)	
		Peach Bottom	Limerick
1	0.15	0.095	0.115
2	0.29	0.087	0.105
3	0.53	0.114	0.138
4	0.99	0.192	0.233
5	1.84	0.375	0.456
6	3.43	0.767	0.932
7	6.38	1.480	1.798
8	11.88	2.427	2.948
9	22.12	3.005	3.650
10	41.18	5.151	5.151

4.2.5.4 Wet Deposition

Wet deposition is an important phenomenon that strongly affects atmospheric transport. Under heavy rains, wet deposition rapidly depletes the plume. Even under light rains, the plume is depleted much faster than by dry deposition alone. The wet deposition process can produce concentrated deposits on the ground and create what is often referred to as a hot spot (i.e., an area of higher radioactivity than the surrounding areas). While rain occurs less than 10% of the time for most of the U.S., it can significantly affect consequence calculations when it does occur.

The wet deposition model predicts how much radioactive material is deposited on the ground by rainfall. Wet deposition model parameters were derived based on expert elicitation data [36]. The linear washout coefficient was selected to be 1.89E-5 per second and the exponential washout coefficient was selected to be 0.664.

4.2.5.5 Plume Meander

Plume meander refers to the broadening of the plume in the crosswind (σ_y) direction as a result of wind direction fluctuations. MACCS provides three options for plume meander modeling. The first option is the original MACCS plume meander model, which considers plume segment duration and derives a meander scaling factor by comparing to the Pasquill-Gifford experimental data. The second option is based on NRC Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” [38] and described in more detail in NUREG/CR-2260 [39]. This option is based on plume segments of one hour duration, which is the same as the plume segment durations used in this study. The third option is to not model plume meander. The Regulatory Guide 1.145 model considers stability class and wind speed for the meander factor and was selected for this analysis because it represents the NRC’s most recent evaluation of plume meander for consequence analysis. This model requires user-specified values of wind speed, distance, and meander factor; all selected values are taken

directly from Regulatory Guide 1.145. The effect of plume meander using this model is greatest around 800 m from the site and diminishes farther from the site.

4.2.6 Early Phase Exposure Pathways, Protective Actions, and Costs

The early phase model in MACCS assesses the time period immediately following a radioactive release. The EPA PAG Manual describes this phase as “the beginning of a radiological incident when immediate decisions for effective use of protective actions are required and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions”. This period is commonly referred to as the emergency phase and it can last for days to weeks following the start of the accident. Early exposures account for emergency planning (i.e., sheltering, evacuation, relocation of the population, and ingestion of potassium iodide). MACCS models sheltering, evacuation, and KI ingestion for user-specified population cohorts. Different shielding factors for the different exposure pathways (cloudshine, groundshine, inhalation, and deposition on the skin) are associated with three types of activities: normal activity, sheltering, and evacuation.

4.2.6.1 Emergency Response Timeline

Emergency response programs for nuclear power plants are designed to protect public health and safety in the unlikely event of a radiological accident. These emergency response programs are developed, tested, and evaluated and are in place as an element of the NRC’s defense-in-depth policy. Detailed plans for onsite and offsite response are approved by the NRC and the Federal Emergency Management Agency respectively. Offsite response organization (ORO) emergency plans are required to include detailed evacuation plans for the 10-mile plume exposure emergency planning zone (EPZ) [40]. Site-specific information was obtained from ORO emergency response plans to support development of timelines for protective action implementation. Site-specific planning elements were modeled, for example whether evacuation of schools follows declaration of a site area emergency (SAE) or a general emergency (GE).

One emergency response timeline was developed for each reference site for all accident scenarios using information from the MELCOR analyses, expected timing of emergency classification declarations, and information from the evacuation time estimate (ETE) reports. The timeline identifies points at which population cohorts would receive instruction from OROs to implement protective actions. In practice, initial evacuation orders are based on the severity of the accident and in Pennsylvania would likely include an evacuation of the entire EPZ. In contrast, it is expected that nuclear power plants in most if not all other states would implement a keyhole-shaped evacuation consisting of an inner circular region and a region extending outward based on the expected direction of the prevailing winds.

One of the objectives of the SOARCA project was to model emergency response in a more detailed and realistic manner than past studies, using site-specific emergency planning information. Therefore, for each scenario at each SOARCA pilot plant, a unique emergency response timeline was developed and was reviewed with the licensee for accuracy. This project seeks to model a significantly larger number of accident progression scenarios compared to SOARCA. Therefore, to develop the timeline for implementation of protective actions, the accident initiator and expected accident progression conditions and timing were reviewed for similarity to the SOARCA scenarios. The initiating event in this analysis is an extended loss of alternating current (AC) power (ELAP) postulated to be caused by an unspecified internal event or seismic event. In the majority of the MELCOR simulations, the reactor core isolation cooling (RCIC) system is assumed to be available for at least 4 hours and therefore the series of events closely

aligns with the accident timeline of the SOARCA Peach Bottom unmitigated long-term station blackout (LTSBO). Thus the SOARCA Peach Bottom unmitigated LTSBO emergency response timeline was adopted for this project and is used for Peach Bottom and Limerick because they would both follow the same Pennsylvania-specific guidance. The timeline is described in detail in Table 4-14.

Table 4-14 Emergency response timeline

Time	Event
0:00	ELAP
0:15	SAE is declared via MS1 of the SAE Emergency Action Level (EAL) based on loss of all AC power.
0:45	GE is declared based on EAL MG1, 45 minutes into the event (coincidentally 15 minutes before the issue of the first Emergency Alert System (EAS) message related to the SAE) when it is assumed that operators have determined that offsite power will not be restored within 2 hours. An EAS message for the GE is then broadcast which would include instructions for implementing protective actions.
1:00	Sirens for SAE sound about 45 minutes after SAE declaration. An EAS message is broadcast at this time providing notification to residents and transients within the EPZ that there is an incident and instructing them to monitor the situation for further information.
1:30	Sirens for GE sound again 45 minutes after the GE declaration, which is 30 minutes after the siren and initial EAS message for the SAE.

4.2.6.2 Evacuation Modeling

MACCS now enables two different types of evacuation regions. The first type is an evacuation of the full 360° EPZ region surrounding the plant. The second type is a recent advancement to MACCS and involves a keyhole-shaped evacuation region consisting of an inner circular region and a region extending outward in the direction of the prevailing winds. Most states are expected to implement a keyhole-shaped evacuation region; however, Pennsylvania would likely implement a full 360° EPZ region according to officials with the Pennsylvania Emergency Management Agency (PEMA). Because both reference plants for this project, Peach Bottom and Limerick, are in Pennsylvania, the full EPZ evacuation region approach is used.

MACCS input parameters related to evacuation modeling were developed primarily from the site-specific ETE reports, which are required to be developed and updated by the licensee under 10 CFR 50.47 (b)(10), “Emergency Plans.” ETEs provide the time required to evacuate various sectors and distances within the EPZ for transient and permanent residents, and these times are used to develop response timing and travel speeds for evacuating cohorts in MACCS. The ETEs provided the mobilization times and travel times of different segments of the population as well as evacuation routing information. The guidance in NUREG/CR-7002 [41] describes the detail included in an ETE study. Important information in an ETE report includes demographic and response data for the following four population segments and may be readily converted into cohorts, if appropriate: (1) permanent residents and transient population, (2) transit-dependent permanent residents—people who do not have access to a vehicle or are dependent upon help from outside the home to evacuate, (3) special facility residents—people in nursing homes,

assisted living centers, hospitals, jails, prisons, etc., and (4) schools, including all public and private educational facilities within the EPZ.

The ETE typically includes about 10 scenarios that vary by season, day of the week, time of day, and weather conditions, as well as other EPZ-specific situations such as special events. The ETEs do not consider a large seismic event and its impact on road infrastructure, which could be substantial. This study considers the impact of a seismic event slowing evacuation in a set of sensitivity calculations described in Section 4.4.4. Consistent with past analyses such as SOARCA for Peach Bottom and Surry, the ETE scenario for a winter weekday, mid-day, good/fair weather accident was used for the development of MACCS evacuation model parameters. This scenario includes residents at work and children at school at the time of declaration of the emergency.

ETEs compute the vehicle demand for each scenario based on population information from the licensee, census reports, and telephone surveys of local residents. These telephone surveys are also used to estimate the mobilization times for each population group, which is the time to learn of the event and prepare to evacuate. ETE studies use detailed link-node representations of the road network to estimate all the routing pathways and the travel speeds of the public through the road network. These studies consider that the public mobilizes and evacuates over a period of time as a distribution of data. In contrast, MACCS models the evacuation process as a series of discrete events for each population cohort. The use of multiple cohorts allows a more realistic modeling of the evacuation process and helps to better represent the evacuating public as a distribution.

The initial response parameter in MACCS is OALARM, which can be used to uniformly adjust protective action timing for all cohorts. Consistent with the SOARCA analysis of Peach Bottom, the OALARM time is set to zero, the time of accident initiation and reactor scram. For each population cohort, MACCS requires the following duration and travel speed parameters:

- Delay to shelter (DLTSHL) represents the duration of time from the accident initiation until the population learns of the event and begins sheltering. During this period shielding parameters are applied assuming normal activity.

- Delay to evacuate (DLTEVA) represents the duration of time from the start of sheltering to the start of evacuation. During this period, shielding parameters are applied assuming sheltering.

- Duration of beginning phase (DURBEG) of evacuation is the first of three time periods and is used to represent the time of travel from one’s starting point until they are in the evacuation queue. This period begins when the sheltering period ends. During this phase a travel speed (ESPEED1) is applied for each cohort. For each of the three evacuation phases, shielding parameters are applied assuming evacuation.

- Duration of middle phase (DURMID) of evacuation is used to represent the time of travel after DURBEG to exit the EPZ. This period begins when DURBEG ends. During this phase a second travel speed (ESPEED2) is applied for each cohort.

- The third and final phase of evacuation is defined as the period of time from the end of DURMID to the end of the early phase (7 days from the time of the first radiation release). During this phase a third travel speed (ESPEED3) is used to represent the travel speed outside the EPZ on large roads such as interstate highways. The population

travels at ESPEED3 until they have reached a distance of 50 miles from the nuclear power plant, at which point they are modeled to receive no further early phase radiation exposure.

For both Peach Bottom and Limerick, the following cohorts are used:

1. Schools within 0-10 miles. This cohort includes elementary, middle, and high school student populations within the EPZ. Based on Pennsylvania protocols, schools would receive early and direct warning from OROs and have response plans in place to support busing of students out of the EPZ. The ETE provides considerable detail regarding schools, including the number and location of schools, student and staff population, number of buses required to evacuate the students, and the ETE, which considers whether return trips are required. Students evacuated by bus would be brought to pre-designated reception centers where parents would pick them up. Preschools and daycare facilities are described in the ETE but are not included in this cohort because, unlike elementary school students, younger children would be expected to be picked up by parents or caregivers and evacuated with the family and not evacuated on buses. College students are also not included in this cohort because they would be expected to evacuate largely via private cars rather than by buses.

2. Special Facilities within 0-10 miles. The special facilities population includes residents of hospitals, nursing homes, prisons, etc. This cohort was modeled differently for Peach Bottom and Limerick and is therefore discussed in the following sections of site-specific evacuation characteristics.

3. Transit-Dependent Residents within 0-10 miles. This population refers to evacuees who do not have access to transportation and confined persons who require special transportation assistance. This population is specifically described and modeled in the ETE reports. Population estimates are based upon U.S. Census data, results of a telephone survey, and for Peach Bottom on data provided by the Lancaster County Geographic Information System (GIS) Department. The ETEs assume a fraction of this population will evacuate with a neighbor or friend and the rest will be evacuated by bus to a reception center outside the EPZ.

4. Early Evacuees within 0-10 miles. Pennsylvania guidance is for sirens to be sounded at SAE. This is a local decision implemented by the OROs. This evacuation is considered early because it is before the OROs officially instruct the public to evacuate. A population fraction of 20% was selected based on data from a national telephone survey of residents of EPZs [42].

5. General Public within 0-10 miles. This cohort represents the population that is not included in any of the other 0-10-mile cohorts. The population fraction is calculated by subtracting the population fractions for all of the other 0-10-mile cohorts.

6. Tail within 0-10 miles. The 0-10-mile tail is defined as the last 10% of the public to evacuate [43] from the 10-mile EPZ. The evacuation tail takes longer to evacuate for valid reasons, such as shutting down farming or manufacturing operations, performing other time consuming actions prior to evacuating, or they may have missed the initial notification.

7. <u>Nonevacuating Public within 0-10 miles</u>. This cohort represents a portion of the public residing in the EPZ who would either not receive any notification of the event or would choose not to evacuate. SOARCA and other studies have assumed this group to be 0.5% of the population. This percentage is consistent with research on large-scale evacuations that has shown a small percentage of the public refuses to evacuate [44]. This value is used in the Limerick evacuation model; however, a higher value is used for Peach Bottom reflecting the expected response of the Pennsylvania Dutch (Amish) community. This is described further in the following section.

8. <u>Shadow Evacuees within 10-15 miles</u>. A shadow evacuation occurs when members of the public evacuate from areas that are not under official evacuation orders and typically begins when a large-scale evacuation is ordered [44]. This study assumed that 20% of the residents in the area between 10-15 miles of the nuclear power plant would evacuate without being ordered to do so. The 20% value was obtained from results of a telephone survey of residents of EPZs conducted by the NRC in 2008 [42]. The location of this cohort, 10-15 miles from the plant, was selected based on guidance in NUREG/CR-7002 [41].

A normal weather winter weekday ETE scenario was selected to develop the evacuation delay and travel speed parameters. However, real weather data is used in the calculation, so to represent the impact of adverse weather, the travel speed of each cohort is reduced when adverse weather occurs. Adverse weather is typically defined as rain, ice, or snow that affects the response of the public during an emergency. The meteorological data file includes hourly precipitation but does not distinguish between rain, ice, snow, etc. MACCS uses the evacuation speed multiplier (ESPMUL) parameter for each cohort to reduce the defined travel speed when precipitation is occurring, as indicated from the meteorological data file. The ESPMUL factor was set at 0.7, which effectively slows down the evacuating public to 70% of the established travel speed during the time precipitation occurs. This value was based on existing ETE guidance [41] and is consistent with SOARCA.

In addition to specifying evacuation travel speeds for each cohort, MACCS includes a network evacuation model that allows the user to select the direction of travel in each spatial grid element in the polar calculational grid. This feature was used in this analysis for Peach Bottom and Limerick. Maps of each site were displayed against the spatial grid, allowing selection of travel directions to represent the likely flow of traffic based on the road infrastructure.

4.2.6.2.1 Peach Bottom Site-Specific Evacuation Model Considerations

A summary of evacuation data for Peach Bottom is provided in Table 4-15. Eight evacuation cohorts were modeled based on the ETE report submitted by Exelon to NRC in June 2014 [22]. The eight cohorts and their general characteristics were described in the previous section. All eight cohorts are modeled as if the population exists uniformly across the entire EPZ. As described above, the nonevacuating cohort is used to represent a very small fraction of the public that would either not receive notification of the event or would choose not to evacuate. A value of 0.5% was used in the SOARCA Peach Bottom analyses; however, this study uses a higher value based on information in the recently submitted ETE, which considered the expected response of the Pennsylvania Dutch (Amish) community. Based on discussions with Lancaster County emergency management officials, the Amish men (considered age 15 or older) would remain on their land while the women and children would evacuate in the event of an incident at Peach Bottom. The number of Amish men is approximately 1.5% of the EPZ population, so therefore

2.0% was selected as the nonevacuating cohort fraction. The Amish women and children are modeled with the transit-dependent population. This is clearly a Peach Bottom site-specific consideration and the use of a larger than typical nonevacuating cohort fraction is discussed later in the results section.

Figure 4-2 shows the emergency response timeline for Peach Bottom in a chart to better illustrate the components of the timeline for each cohort and the differences among the cohorts. The time duration parameters (DLTSHL, DLTEVA, DURBEG, and DURMID) for the schools, special facilities, and transit-dependent cohorts were selected by reviewing the ETE, which models the mobilization and travel time for each individual school, medical facility, etc. in the EPZ. The time duration parameters for the tail cohort were selected to align the response with the ETE mobilization time and evacuation time for 100% of the EPZ to be cleared. For the selected ETE scenario (winter, midweek, mid-day, good/fair weather), the ETE 100% time was 3:55 from the time of notification. Assuming this cohort is notified via the GE siren and corresponding emergency alert system (EAS) messaging at 1:30, the total evacuation time is approximately 5:30 as shown in the figure below. The time duration parameters for the rest of the cohorts were selected by approximating the mobilization and evacuation distribution curves in the ETE report.

The evacuation travel speeds are shown in the figure below for the first two evacuation phases in units of miles per hour. They were selected considering the DURBEG and DURMID evacuation travel durations to make the travel distance approximately 10 miles. In reality, the centroid of population within the EPZ would not be very close to the plant, and thus the average distance to the EPZ boundary might only be about 5 miles (or less). However, the travel distance of 10 miles is approximated because the path of travel must follow the existing road network and is assumed to not be directly outward. For example, the fastest travel path may involve driving on a small local road toward the plant to then enter a limited-access highway that goes generally away from the plant at a higher speed. The evacuation travel speeds were also selected so that travel for all of the cohorts would be about the same at the same time. The first few cohorts to enter the road network travel at about 15 mph. Shortly after, a larger fraction of the population enters the roads and all speeds drop to about 5 mph on average. These travel speeds include the delays and queue times that occur at intersections. The tail cohort is the last group in the EPZ so they travel at a faster average speed (20 mph) because the roads are much less crowded than before. The 10-15-mile shadow evacuation cohort travels at speeds different from the others because it is on different road sections that are less congested.

Table 4-15 Peach Bottom evacuation cohorts and parameters

Population					Response Durations (hrs)		Evacuation Phase Durations (hrs)		Evacuation Travel Speeds (mph)		
	Cohort Title	Cohort Location (mi)	Population	Location Population Fraction	Delay to Shelter	Delay to Evacuate	Early	Middle	Early	Middle	Late
1	Schools	0 – 10	7,846	0.173	0.25	1.75	0.50	0.50	15	5	20
2	Special Facilities	0 – 10	363	0.008	0.25	2.00	0.25	0.75	15	5	20
3	Transit-Dependent	0 – 10	1,859	0.041	0.25	2.25	0.50	2.00	5	5	20
4	Early	0 – 10	9,070	0.2	1.00	1.00	0.50	0.50	15	5	20
5	General	0 – 10	20,771	0.458	1.50	1.00	0.50	1.50	5	5	20
6	Tail	0 – 10	4,535	0.1	2.00	1.50	1.75	0.25	5	20	20
7	Nonevacuees	0 – 10	907	0.02	-	-	-	-	-	-	-
8	Shadow	10 – 15	20,803	0.2	1.00	2.00	0.50	1.50	20	20	20

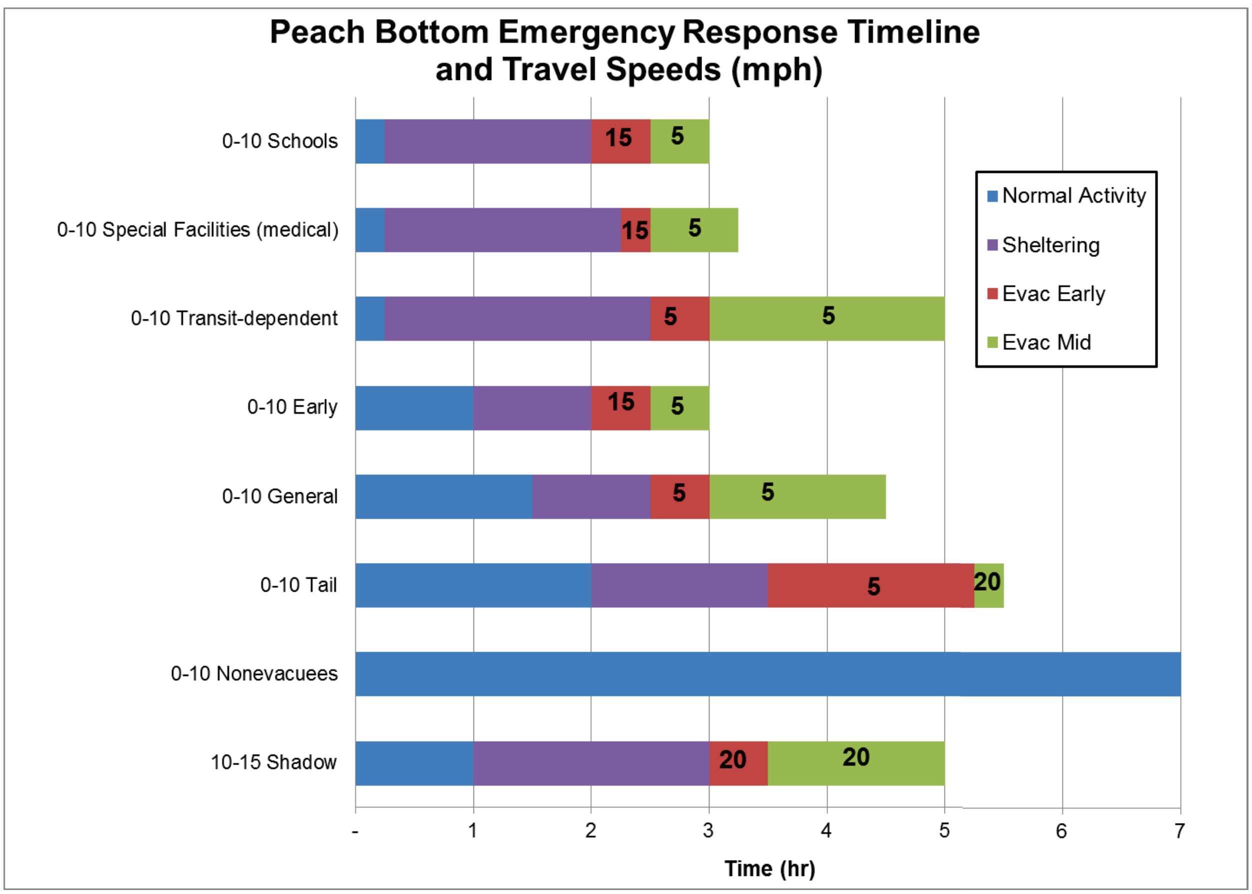


Figure 4-2 Peach Bottom emergency response timeline and travel speeds (mph)

4.2.6.2.2 Limerick Site-Specific Evacuation Model Considerations

A summary of evacuation data for Limerick is provided in Table 4-16. Nine evacuation cohorts were modeled based on the ETE report submitted by Exelon to NRC in January 2014 [45]. Compared to the Peach Bottom evacuation model, an additional cohort was used for Limerick to distinguish between two types of special facility populations, those at medical and assisted living facilities vs. those at correctional facilities. (Peach Bottom has no correctional facilities in the EPZ.) According to the Limerick ETE, the 6,037 inmates (3,957 at the State Correctional Institute at Graterford in Graterford, PA and 2,080 at the Montgomery County Correctional Facility in Eagleville, PA) would shelter in place rather than evacuate, unlike the medical facility residents who would evacuate in the transit-dependent cohort. Because this significant fraction of the population would not evacuate and it exists in two specific locations, effort was taken to model them in the specific calculational grid elements where they exist, rather than applying the population uniformly across the EPZ as is done with the rest of the Limerick cohorts. The Graterford, PA prison inmates are all modeled in the 7.5-10-mile ring in compass sector 16 and the Eagleville, PA prison inmates are all modeled in the 7.5-10-mile ring in compass sector 22 (compass sector 1 is centered on due north of the plant, compass sector 17 is centered on due east of the plant, etc.). The ability to model cohorts in specific grid elements is a recent advancement to MACCS and improves realism. Unlike in the Peach Bottom EPZ, the Amish community represents a negligible fraction of the population and is therefore not considered as part of the nonevacuating cohort in the Limerick evacuation model.

Table 4-16 Limerick evacuation cohorts and parameters

Population					Response Durations (hrs)		Evacuation Phase Durations (hrs)		Evacuation Travel Speeds (mph)		
Cohort Title		Cohort Location (mi)	Population	Location Population Fraction	Delay to Shelter	Delay to Evacuate	Early	Middle	Early	Middle	Late
1	Schools	0 – 10	49,321	0.19	0.25	1.75	0.50	3.0	10	2	5
2	Special Facilities (Medical)	0 – 10	2,741	0.011	0.25	3.00	0.50	2.0	2	2	5
3	Special Facilities (Correctional)	Individual Grid Elements	6,037	0.023 (of EPZ)	-	-	-	-	-	-	-
4	Transit-Dependent	0 – 10	3,764	0.015	0.25	2.75	0.50	3.75	2	2	5
5	Early	0 – 10	51,949	0.2	1.00	1.00	0.50	3.0	10	2	5
6	General	0 – 10	118,602	0.457	1.50	1.00	0.75	3.25	2	2	5
7	Tail	0 – 10	25,974	0.1	3.00	2.50	1.50	1.50	2	5	5
8	Nonevacuees	0 – 10	1,299	0.005	-	-	-	-	-	-	-
9	Shadow	10 – 15	64,018	0.2	1.00	3.00	0.50	2.00	5	5	5

Figure 4-3 shows the emergency response timeline for Limerick in a chart to better illustrate the components of the timeline for each cohort and the differences among the cohorts. The time duration parameters and evacuation travel speeds were developed using the same approach as was described for Peach Bottom. Limerick has an ETE population more than five times higher than Peach Bottom; however, the ETE for 100% of the EPZ to be cleared for the same (winter, midweek, mid-day, good/fair weather) scenario is 6:50 from the notification time, just ~3 hrs longer

than Peach Bottom. The ETE for 90% EPZ clearance is also considered a useful metric for emergency response officials and is 5:05 for Limerick and 2:30 for Peach Bottom (both from the time of notification). One of the primary reasons for the highly sublinear increase in evacuation time as a function of population is that the Limerick road network is larger and can accommodate a higher vehicle flow rate on average compared to Peach Bottom.

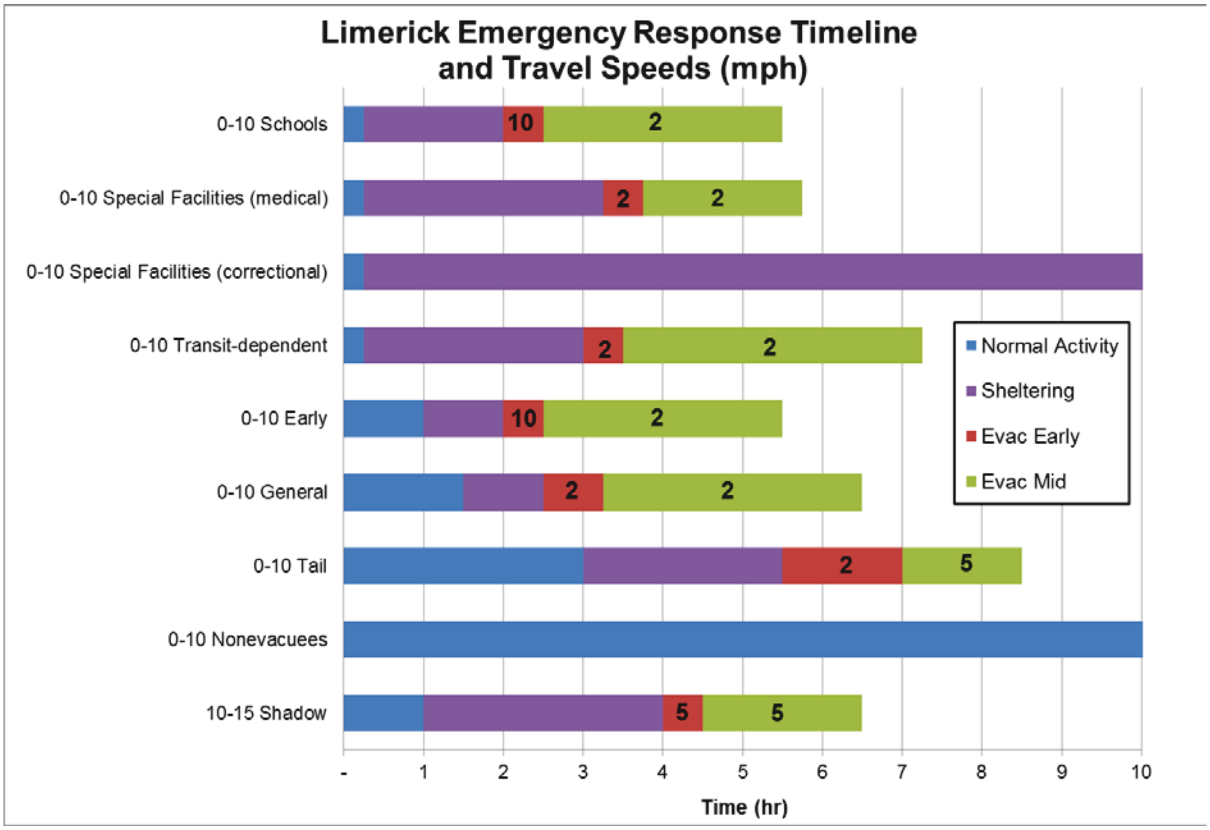


Figure 4-3 Limerick emergency response timeline and travel speeds (mph)

4.2.6.3 Shielding

In MACCS, shielding and protection factors are specified for each dose pathway and directly affect the doses received by individuals at each location. The shielding and protection factors are used as multipliers on the dose that a person would receive if there were no shielding. Thus, a shielding factor of one represents the limiting case of a person receiving the full dose (i.e., standing outdoors and completely unprotected from the exposure); a shielding factor of zero represents the limiting case of complete shielding from the exposure.

Three types of activity: normal, sheltering, and evacuation, are evaluated for each dose pathway for each population cohort which evacuates, while just one type of activity is evaluated for the nonevacuating public. In this context, normal activity refers to a combination of activities that are averaged over a typical week and over the population, including being indoors at home, commuting, being indoors at work, and being outdoors. These values are applied to the time periods in Table 4-15 and Table 4-16 and continue for evacuees until they have traveled to a point 50 miles from the nuclear plant; for nonevacuees they are applied until the early phase concludes (7 days from the start of the radionuclide release to the environment) or until they relocate. Table 4-17 and Table 4-18 show the cloudshine shielding factor, groundshine shielding factor, inhalation

protection factor, and skin protection factor values used in this analysis. The values in Table 4-17 are consistent with those used for Peach Bottom in SOARCA [10] and were developed for NUREG-1150 [46] considering region-specific housing stock. The development of shielding parameters for NUREG-1150 is described in greater detail in NUREG/CR-4551, Vol. 2, Rev. 1, Part 7, “Evaluation of Severe Accident Risks: Quantification of Major Input Parameters: MACCS Input” [47]. Special facilities are typically larger and more robust structures than residential housing stock and therefore shielding factors are lower than for other population cohorts.

Table 4-17 Shielding and protection factors for evacuating cohorts

Population Facility Type	Cloudshine Shielding			Inhalation and Skin Protection			Groundshine Shielding		
	Normal Activity	Sheltering	Evacuating	Normal Activity	Sheltering	Evacuating	Normal Activity	Sheltering	Evacuating
Regular Facilities	0.60	0.50	1.00	0.46	0.33	0.98	0.18	0.10	0.50
Special Facilities (Medical)	0.31	0.31	1.00	0.33	0.33	0.98	0.05	0.05	0.50

Table 4-18 Shielding and protection factors for nonevacuating population

Population and Facility Type	Cloudshine Shielding	Inhalation and Skin Protection	Groundshine Shielding
Limerick Special Facilities (Correctional)	0.31	0.33	0.05
Nonevacuees Within EPZ	0.60	0.46	0.18
Remaining Population Beyond EPZ	0.50	0.33	0.10

For the nonevacuating population groups, only one shielding value for each exposure pathway is defined. For the Limerick special facilities cohort corresponding to the two correctional facilities in Graterford, PA and Eagleville, PA, shielding factors were selected to be consistent with the normal activity/sheltering shielding for the medical facilities. For the nonevacuees within the EPZ, shielding factors were selected to be consistent with the normal activity values for regular facility populations. Finally, for the remaining population outside the EPZ (80% of the population between 10-15 miles and everyone beyond 15 miles), shielding factors were selected to be consistent with the sheltering values for regular facilities. This is based on the expected protective action recommendation of the OROs for the remaining affected population to shelter in place and monitor the situation as it evolves based on television, phone, radio, or other means.

4.2.6.4 Potassium Iodide

Some States have distributed potassium iodide (KI) tablets to people who live near commercial NPPs. KI has been distributed within the Peach Bottom and Limerick EPZ according to officials with PEMA. The purpose of the KI is to saturate the thyroid gland with stable iodine so that further uptake of radioiodine by the thyroid is diminished. If taken at the right time, KI can nearly eliminate doses to the thyroid gland from inhaled radioiodine. Ingestion of KI is modeled for half of the EPZ population for each site, applied uniformly across all population cohorts. A further assumption is that most residents do not take KI at the optimal time (from shortly before to immediately after plume arrival), so the efficacy is only 70% (i.e., the thyroid dose from inhaled radioiodine is reduced by 70%). KI ingestion is not modeled for any of the population beyond 10 miles from each site.

4.2.6.5 Early Phase Relocation

In addition to evacuation, sheltering, and KI ingestion, OROs would implement the additional protective action of relocation. In the event of an accident at the NPP, OROs would use State, utility, and Federal agency computer models and field measurements to project dose levels in the area considering available source term information and current and projected meteorology. Based on these dose projections, OROs would inform the affected population directly and through available communication channels and instruct them to relocate. The U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAGs) [48] provide guidance to OROs for relocation based on a range of 1 to 5 rem dose projected over a 4-day period. In reality, relocation would be instructed on a gradual scale, starting with those most at risk and then moving to populations at progressively lower risk levels. In MACCS, emergency phase relocation is modeled with two user-specified dose criteria to trigger the action, and a relocation time for the population affected by each dose. The higher dose level, used first, is set to the maximum point of the range in the EPA PAGs (5 rem projected over a 4-day period) and is referred to as the hotspot relocation dose criterion. The lower dose level, used next, is set to the lower end of the range in the EPA PAGs (1 rem projected over a 4-day period) and is referred to as the normal relocation dose criterion. The projected dose in MACCS would include the cloudshine, groundshine, direct inhalation, and resuspension inhalation exposure pathways and also considers shielding. In MACCS, individuals who reside in a grid element where either relocation dose criterion is exceeded, and who are modeled to have not already evacuated, are exposed for a user-specified time duration and then are removed from the computational grid so that they receive no further early phase dose. This time duration depends on the source term and on the size of the population affected. Larger source terms would trigger relocation farther from the plant than smaller source terms and areas with larger population would take longer to notify and to evacuate than those with a smaller population.

Table 4-19 provides the relocation dose criteria and exposure times for Peach Bottom and Limerick used in this project. The dose criteria were selected based on the upper and lower end of the range of dose levels specified in the EPA PAGs. The exposure times were selected considering the range of areas and population that might reach either dose criterion. The hotspot relocation exposure time duration was selected to account for relocation that would apply to the closest group of people that would not have already evacuated, those in the 10-20-mile rings. The normal relocation exposure time duration was selected to account for relocation that would apply to the population farther from the site, approximately 20-30 miles and potentially farther.

Table 4-19 Early phase relocation parameters

NPP Site	Relocation Type	Dose Criterion	Exposure Time Duration After Plume Arrival
Peach Bottom	Hotspot	5 rem	3 hrs
	Normal	1 rem	24 hrs
Limerick	Hotspot	5 rem	21 hrs
	Normal	1 rem	48 hrs

The hotspot relocation exposure time duration considers the time when the 10-20-mile population would be advised to relocate, the time this population would take to mobilize and travel past 20 miles, the time of the start of the release of radionuclides to the environment, and the travel time for the radionuclide plume to reach the 10-20-mile rings. There is a large degree of uncertainty

with all of these times and they depend on the accident, the ORO decisionmaking process, the population affected, and the meteorological conditions. For Peach Bottom, and for the quickest environmental release timing (~ 7 hours corresponding to Mark I MELCOR case 49, which assumes a short-term station blackout with no batteries available and no RCIC), notification is estimated to align with the GE siren (1.5 hours), plume travel is estimated based on average annual wind speed (10 mi / ~5 mph = ~2 hours), and the mobilization and travel time is estimated at about 10 hours for 450,000 people. This leads to an exposure duration of about 3 hours. For Limerick, the same source term timing is assumed (7 hours) and the same plume travel time is assumed (2 hours). Because Limerick has a much larger EPZ population, ORO officials would likely wait to advise relocation for the 10-20-mile population until the EPZ population would have time to leave first, since they would be at higher risk. The EPZ evacuation is modeled to conclude around 8.5 hours (1.5 hours for GE siren and 7 hours for the 100% ETE); therefore, the relocation notification is assumed at 9 hours. The mobilization and travel time for the 10-20-mile population is about 21 hours for 930,000 people, a linear extrapolation from the ETE of 7 hours for 300,000 people. This leads to an exposure duration of about 21 hours for Limerick.

The normal relocation exposure durations are designed to simulate the potential for evacuation of the 20-30-mile ring and potentially farther. For Limerick, the population from 20 miles to 30 miles is approximately 2.8 million. To approximate the time to evacuate such a large population, the times for large-scale evacuations for hurricanes Rita and Ivan were reviewed. These hurricanes included populations in this range and as described in NUREG/CR-6981, "Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations," [49] the largest numbers of evacuees took on the order of 36 to 48 hours to complete their evacuation. Therefore, 48 hours was selected as an approximate time to evacuate 2.8 million people from the 20-30-mile area. The notification timing for this population is even more uncertain than for the 10-20-mile area and thus 48 hours is estimated as the normal relocation exposure duration. The population for this ring around Peach Bottom is about 1 million, and therefore 24 hours is estimated for the Peach Bottom normal relocation based on extrapolation of 48 hours for Limerick for about three times higher population.

In addition to advising relocation for populations projected to exceed the EPA PAGs, OROs would advise the public to shelter in place. This protective action is captured in MACCS using a shielding factor for this group based on sheltering rather than normal activity.

4.2.6.6 Early Phase Costs

MACCS calculates the cost of the early (emergency) phase of the accident by multiplying a user-specified daily per person compensation cost by the number of people affected by the number of days each person is affected. The population affected includes those who evacuate in any of the defined evacuation population cohorts as well as those who would be relocated by exceeding either the hotspot or normal relocation dose criterion. The number of people and number of days affected is computed at the grid element level.

The daily compensation cost, defined in MACCS as EVACST, was selected considering the cost of lodging, food, and lost income. Studies show that the majority of people who evacuate their homes do not use shelter facilities for overnight stay [50]. People most often stay with relatives and friends or in hotels. For those who must leave their homes during the early phase, the costs of hotels and meals are additional costs that would not have been incurred under normal conditions. A typical federal per diem lodging rate is about $150 per night and considering the median household size of 2.58 persons per house from the U.S. Census, the lodging cost would be $58/person. This cost varies regionally and seasonally. For displaced people who stay with

family or friends, there are other tangible and intangible costs associated with these stays. Food costs can vary considerably depending on the restaurant category. For this estimate, typical meal prices for diner type restaurants were reviewed. An estimate of $35 per person per day includes the meals, beverage, tax, and tip. Restaurant costs also vary between rural and metropolitan areas, but the difference is not as significant as lodging. The lost income was selected based on the median annual household income and dividing by the number of working days in a year and by the median number of people per household. The US Census identifies the 2012 median annual household income as approximately $51,000. There are 365 days in a year, minus 104 weekend days and 11 holidays, leaving 250 working days. Income divided by working days equals $204/per day per household and dividing this by 2.58 persons per household equals $79 per person per day. Therefore the total evacuation compensation cost per person per day is $172 ($58 + $35 + $79).

4.2.7 Intermediate Phase Exposure Pathways, Protective Actions, and Costs

The intermediate phase begins after the source and releases have been brought under control and accounts for the time needed to plan the long-term restoration and cleanup activities before they can begin. These activities include the following:

1. defining the areas of interest,
2. characterizing the contamination using dose data and field surveys,
3. identifying the types of materials to be decontaminated and the equipment and personnel needed,
4. developing a waste management plan including estimating waste volumes, storage requirements, storage locations, and acquiring storage materials,
5. acquiring decontamination equipment and bringing it onsite, and
6. training personnel and bringing them onsite.

The length of time to conduct these processes would depend on a variety of factors including the extent and location of the contamination, cleanup criteria, material types, and the state and local decision processes. As the distance from the plant increases, the level of contamination may decrease; however, other influences may increase because a wider variety of materials may be involved and additional towns, counties, and stakeholders would become involved.

Based on the number and types of planning processes that are needed prior to starting decontamination, 3 months was selected as an average intermediate phase duration and was used for all source terms in this analysis. Some decontamination would be expected to start prior to 3 months; however, much would likely take longer to begin. The first documented decontamination efforts after the March 11, 2011, Fukushima accident began with removal of contaminated soils from schools in May 2011, about 10 weeks after the accident [51]. In contrast, there is still much area for which decontamination has not begun years after the accident. According to the Japanese Ministry of the Environment in an April 2014 presentation, “Progress on Off-site Cleanup Efforts in Japan,” seven of the eleven affected municipalities had not started decontamination work as of April 2012, over a year after the accident. In addition, one of the municipalities, Futaba, had not started decontamination work as of April 2014, over three years after the accident [52].

The only protective action modeled in the intermediate phase is relocation. If the projected dose to a population exceeds a user-specified threshold projected over the duration of the intermediate

phase, the population is assumed to be relocated to an uncontaminated area for the entire duration of this phase. A corresponding per-capita per diem compensation cost is defined by the user. If the projected dose does not reach the user specified threshold, exposure pathways for groundshine and inhalation of resuspended material are modeled.

The intermediate phase relocation dose criterion was selected based on linear extrapolation of Pennsylvania's annual habitability criterion of 500 mrem to the duration of 3 months, equaling 125 mrem. MACCS calculates the cost of the intermediate phase of the accident very similar to the process for the early phase. A user-specified daily per person compensation cost is multiplied by the number of people affected by the number of days each person is affected (91 days). The intermediate phase daily per person compensation cost, defined in MACCS as RELCST, was selected to be identical to the early phase compensation cost, $172/person-day, because the costs incurred are similar.

4.2.8 Long-Term Phase Exposure Pathways, Protective Actions, and Costs

In MACCS, the long-term phase starts at the conclusion of the intermediate phase and lasts for a user-specified duration, which was selected in this project as 50 years, consistent with SOARCA. This period accounts for the time needed complete all recovery and restoration actions. Protective actions are implemented to minimize the dose to an individual by external (e.g., groundshine) and internal (e.g., food and water ingestion and resuspension inhalation) exposure pathways. Protective actions are based on decisions relating to whether farmland, at a specific location (grid element) and time, is suitable for agricultural production (farmability) and whether nonfarmland, at a specific location and time, is suitable for human habitation (habitability). Habitability is defined by a user-specified maximum dose and a user-specified exposure period to receive that dose. Habitability decision making for nonfarmland can result in four possible outcomes:

1. land is immediately habitable,
2. land is habitable after decontamination,
3. land is habitable after decontamination and interdiction, or
4. land is not deemed habitable after decontamination plus 30 years of interdiction and is therefore condemned.

Land is also condemned if the cost of decontamination exceeds the value of the land. The dose criterion for the MACCS modeling of individuals returning back to the affected (i.e., contaminated) area is based on a Pennsylvania state-specific dose criterion of 500 mrem-per-year for each year; whereas most other states would likely follow the guidance in the EPA PAGs [18] of 2 rem in the first year and 500 mrem-per-year each year thereafter.

The decision on whether land is suitable for farming is first based on prior evaluation of its suitability for human habitation—land cannot be used for agriculture unless it is habitable. Furthermore, farmland must be able to grow crops or produce meat or dairy products that meet the user-specified farmability criterion, which is 500 mrem-per-year effective dose from ingestion. This dose criterion was selected to be consistent with the Pennsylvania state-specific habitability criterion. If farmland is habitable and farmable, a food chain model is used to determine doses that would result from consuming the food grown or produced on this land. The COMIDA2 food chain model is the latest model developed for use in MACCS. This model contains data on expected radionuclide uptake in nine foodstuff types for different seasons of the year for different contamination levels and food category consumption rates for an average adult.

Decisions on decontamination are made using a decision tree. The first decision is whether land is habitable. If it is, then no further actions are needed. The population returns to their homes and receive a dose not to exceed 500 mrem-per-year from any deposited radionuclides for the entire long-term phase. If land is not habitable, the first option considered is to decontaminate at the lower level of dose reduction, which is also cheaper to implement. If this level is sufficient to restore the land to habitability, then it is performed. Following the decontamination, the population returns to their homes and receives a dose not to exceed 500 mrem-per-year based on the residual contamination for the duration of the long-term phase. If the first level of decontamination is insufficient to restore habitability, then a higher and more expensive level is considered. If the higher level of decontamination is insufficient, then interdiction for up to 30 years is considered following the decontamination. During the interdiction period, radioactive decay and weathering reduce the dose rates that would be received by the returning population. If the higher level of decontamination followed by interdiction is sufficient to restore habitability, then it is implemented and the population is allowed to return. Doses are accrued for the duration of the long-term phase. If habitability cannot be restored by any of these actions, then the land is condemned. The land is also condemned if the cost of the required action to restore habitability is greater than the value of property.

To support the decision process of whether decontamination is cost-effective for a land use type for a grid element, MACCS requires one parameter to characterize the value of farm wealth (VALWF in $ per hectare) and one parameter to characterize the value of nonfarm wealth (VALWNF in $ per person). If the cost to decontaminate farmland in a grid element exceeds VALWF, then the farmland in the grid element is condemned. If the per person cost to decontaminate nonfarmland in a grid element exceeds the cost of condemning the property (VALWNF plus the cost to permanently relocate), then the nonfarmland in the grid element is condemned. If condemned, the actual cost of condemnation is taken from the site data file, discussed previously in Section 3.3, which includes a unique value of these parameters for each grid element. To identify one value for VALWF and one value for VALWNF that would represent the potential range of affected land, the individual values from the site file for each grid element out to 50 miles were weighted by the number of people and the area of farmland they contain. This resulted in a weighted average farm wealth value of $24,400 per hectare and a weighted average nonfarm wealth value of $518,000 per person for Peach Bottom and $28,600 per hectare and $528,000 per person for Limerick.

MACCS values of total long-term population dose and health effects account for exposures received by workers performing decontamination. While engaged in cleanup efforts, workers are assumed to wear respiratory protection devices; therefore, they only accumulate doses from groundshine. Table 20 provides the decontamination plan data used in this analysis for both Peach Bottom and Limerick. The farm costs ($ per hectare) and nonfarm costs ($ per person) are based on the NUREG-1150 values and are scaled from the NUREG-1150 target year, 1986, to this project's target year, 2013, using the CPI-U inflator of 2.126. The MACCS decontamination model also requires a decontamination worker cost which is used to convert the total decontamination cost to a number of worker-years, which is then used to compute the total population dose to the decontamination workers. The decontamination worker cost was estimated at $76,000 per worker per year. This is based on the median annual wage for hazardous materials removal workers of $37,590 according to the U.S. BLS Occupational Outlook Handbook [53] and adding a factor of 100% to cover overhead and all other non-direct labor costs.

Table 4-20 Decontamination plan data for Peach Bottom and Limerick

Dose Reduction Factor (Reduction Percent)	Farm Decontamination Cost (per hectare)	Nonfarm Decontamination Cost (per person)	Decontamination Time (days)
3 (67%)	$1,200	$6,400	60
15 (93%)	$2,700	$17,000	120

For population displaced from their home in the long-term phase, a one-time relocation cost is applied (POPCST). This cost is assessed if decontamination alone, decontamination followed by interdiction, or condemnation is required. The value is intended to account for personal and corporate income losses for a transitional period, as well as moving expenses. The value used in this analysis is $10,600 per person which is based on the same CPI-U inflator of 2.126 from the NUREG-1150 study, based on 1986, to this project's target year, 2013. There is no analogous cost for farmland.

The cost of interdiction for nonfarmland includes a component to capture the loss of use of the property. For farmland, this is the only cost for interdiction. This equation considers the per capita or per hectare value (from the site file), a typical inflation-adjusted annual rate of return on investments, a typical depreciation rate, and a fraction of the per capita and per hectare value that is due to improvements (which are subject to depreciation). The depreciation rate (DPRATE) used is 20% per year, consistent with SOARCA and previous studies. Past studies used a 12% return on investments (DSRATE), which is not representative of the current investment climate. Therefore a current DSRATE value was computed by a simple average of inflation-adjusted annual returns on two investment indexes over a period of greater than 20 years: the Case-Shiller U.S. National Home Price Index [54] for residential property and the National Council of Real Estate Investment Fiduciaries (NCREIF) Property Index [55] for commercial property. The average of the average annual inflation-adjusted return on each index resulted in a DSRATE of about 3%. The fraction of farm wealth in the region due to improvements (FRFIM, considered everything other than the value of the land itself) used was 0.25, consistent with SOARCA and past studies. The fraction of nonfarm wealth in the region due to improvements (FRNFIM) was selected to be 0.62 based on a study by Davis and Heathcote [56], which concluded that from 1970 to 2003, the nominal value of land accounts for 38% of the nominal market value of homes.

Finally, for farmland that exceeds the farmability criterion and is modeled to be temporarily unable to produce crops, meat, and dairy products that are suitable for human consumption, this food for the current growing season must be disposed. MACCS calculates these costs by multiplying the farm area of each affected grid element by the annual agricultural sales per hectare and summing this over all grid elements. The site data file includes a parameter to estimate the fraction of annual farm sales for dairy products and thus the disposal cost for dairy products is reported separately from the disposal cost for all other agricultural products.

4.2.9 Dosimetry

The approach used for dosimetry modeling in this analysis is identical to that used in the SECY-12-0157 offsite consequence analysis [16]. The dosimetric quantities computed by MACCS for use in modeling protective action decisionmaking or health effects are based on a dose conversion factor approach. In general, the radiological dose to a receptor in a given spatial element is the product of (1) the integrated air concentration or total ground deposition of a radionuclide, (2) the exposure duration for an exposure pathway, (3) the shielding factor for an

exposure pathway, (4) the dose conversion factor for a radionuclide and pathway, and (5) the usage factor for an exposure pathway. The total dose to an organ or the whole body used for modeling of health effects or protective action decisionmaking is then summed across all radionuclides and the relevant exposure pathways.

The detailed model formulation for each exposure pathway is discussed in Chapter 3 of NUREG/CR-4691 [2]. The exposure pathways considered during the emergency phase include cloudshine, groundshine, cloud inhalation, and inhalation of resuspended radionuclides. The exposure pathways for the intermediate phase include groundshine and inhalation of resuspended radionuclides. The long-term phase includes groundshine, inhalation of resuspended radionuclides, food ingestion, and water ingestion. For the early phase, two kinds of doses are calculated: (1) acute doses used for calculating early fatalities and injuries and (2) lifetime dose commitment used for calculating cancers resulting from the early exposure. For the long-term phase, only lifetime dose commitments are calculated.

The quantities used in the dose equations depend on the exposure pathway and are either user inputs or are computed internally by MACCS. The radionuclide concentrations are calculated by the ATD code module at ground level along the plume centerline. In order to calculate the doses at different locations within a grid element, a correction factor (discussed in Section 3.1.1 and 3.2.1 of NUREG/CR-4691, Volume 2 [2]) is derived to adjust the concentration to account for the reduction in concentration away from the plume centerline. The duration of exposure depends on the exposure pathway and the protective actions in a grid element, and is either calculated by MACCS or specified by the user. The shielding factor is a dimensionless quantity used to reduce the radiation dose as a result of shielding protection provided by a given protective action for a given exposure pathway. Shielding factors for the early phase were discussed in Section 4.2.6.3. Shielding factors for the long-term phase use the early phase normal activity value for the entire population. The dose conversion factors for all exposure pathways are provided via the MACCS dose conversion factor file and have the following units for each exposure pathway:

- Cloudshine: Sv/s per Bq/m^3
- Groundshine: Sv/s per Bq/m^2
- Acute Inhalation: Sv/Bq
- Chronic Inhalation: Sv/Bq
- Ingestion: Sv/Bq

The dose conversion factors used in this analysis are identical to those used in SOARCA. They are based on a methodology that considers the updated dosimetry and health effects models from FGR-13 [19], as well as the instantaneous dose rate values provided in the supplemental files provided with FGR-13. This allows both a consideration of the acute effects due to short-term exposure, as well as the ability to consider annual doses as well as committed doses. The dose conversion factor file set used in this analysis, "FGR13GyEquivDCF.INP," together with its annual dose files, contains dose conversion factors based on FGR-13 for 825 radionuclides, 26 tissues, organs, the whole body effective dose, and four exposure pathways. MACCS contains a more limited set of organ dose quantities than are available in the DCF file based on FGR-13. MACCS considers nine organs (including whole body) for stochastic effects from chronic exposures and six organs for deterministic effects from acute exposures. Due to a current limitation of eight cancer sites (organs), MACCS calculates the dose to seven specific cancer sites and one residual cancer site. To estimate residual cancers, the dose coefficients for the pancreas are used as a

surrogate for dose to soft tissue, following recommendations of the Oak Ridge National Laboratory letter report [58].

The dosimetry calculation also considers a usage factor for the inhalation and ingestion exposure pathways. For cloudshine and groundshine, the value in the calculation is one. For inhalation, the usage factor is a volumetric breathing rate and is specified by the user. Consistent with past studies, one value of this parameter is used for all populations and time periods: 2.66E-4 m^3/s. This value was derived in NUREG/CR-4551, Vol. 2, Rev. 1, Part 7 [47] for an adult man who sleeps 8 hours per day and engages in light activity when awake. This value is slightly higher than one that considered men, women, and children. The usage factor for the ingestion pathway has two components, both of which are specified in the COMIDA2 food chain file that is an input to MACCS. The first describes the uptake of different radionuclides into different foodstuff types for different seasons of the year. The second component describes the consumption rate of each foodstuff type for an average adult.

The effective dose (ICRP60ED) is used internally by MACCS for simulating protective action decisions based on dosimetric quantities computed under a system of radiation protection. This requires a specification of the tissue weighting factors to compute an effective dose. The tissue weighting factors used in the computation of the dose conversion factor for the ICRP60ED effective dose are taken from ICRP-60 [60] and are identical to those used in SOARCA.

4.2.10 Health Effects

The approach used for health effects modeling in this analysis is identical to that used in the SECY-12-0157 offsite consequence analysis [16].

MACCS considers two types of health effects: deterministic health effects arising from acute exposures during the early phase of an accident, and stochastic health effects arising from acute exposures during the early phase of an accident and chronic exposures during the intermediate and long-term phases. The health effects models in MACCS are based on the models described in the NUREG/CR-4214 series of reports [57], as reflected in NUREG/CR-4691 [2] and NUREG/CR-6613 [3]. The models presented in these reports provide estimates of the likelihood that an exposed individual may experience a specific health effect (e.g., lung impairment, breast cancer). Depending upon the exposure pathway, MACCS considers three types of populations: (1) individuals residing in the area surrounding the accident site who are directly exposed to contaminated media, (2) individuals who reside in unspecified locations that consume food grown in, or drink water originating in, the spatial elements surrounding the accident site, who are therefore indirectly exposed to contaminated media, and (3) decontamination workers who reside in unspecified locations. Health effects for all three populations are attributed to the grid element in which the contamination exists that created the exposure, even if the exposed population resides elsewhere.

The distinction between the population types is used in the reporting of individual vs. collective health effects as shown in Table 4-21. After average individual risks have been estimated using the individual risk models, total cases of health effects are calculated in MACCS by multiplying the average individual risk of experiencing an effect by the number of people who receive the same dose that leads to the risk. However, health effect cases also account for the ingestion pathway, whereas individual risk does not. Furthermore, health effect cases also count doses to decontamination workers, whereas, individual risk does not. Measures of individual risk of health effects are based only on the direct pathways. Doses from these direct pathways are attributed to individual grid elements, and because the exposed population is confined to that grid element, estimates of individual risk can be made. Quantitative output for collective measures such as

population dose includes all three types of populations. Although doses from indirect pathways (such as food or water ingestion or decontamination worker doses) are attributed to individual grid elements, they are not included in individual risk measures because the exposure population for these pathways may be very different from the population residing in that grid element.

Table 4-21 Exposure pathways and population types used to compute individual and collective health effects

Exposure Pathway and Population Type	**Individual Health Effects** (Early Fatality Risk and Latent Cancer Fatality Risk)	**Collective Health Effects** (Population Dose)
Cloud Inhalation	Early	
Cloudshine	Early	
Groundshine to Population Residing on Spatial Grid	Early, Intermediate, and Long-Term	
Groundshine to Decontamination Workers	-	Long-Term
Inhalation of Resuspended Materials to Population Residing on Spatial Grid	Early, Intermediate, and Long-Term	
Skin Dose	Early	
Food and Water Ingestion	-	Long-Term

The early health effect risk model implemented in MACCS is described in NUREG/CR-4691 [2] and in NUREG/CR-6613 [3]. This model is based on NUREG/CR-4214 [57] and has a sigmoidal dependence of individual risk on dose to the target organ in an exposed individual.

The model considers a shape parameter that determines the steepness of the sigmoidal dose response curve, a dose equivalent delivered to the target organ, a dose equivalent threshold below which the effect is not expected to occur and thus the risk is zero, and a dose equivalent that would induce the effect in half the exposed population (D_{50}).

Parameters representing acute health effects are derived from expert elicitation data documented in NUREG/CR-6545 [37] and in NUREG/CR-7161 [36]. The most sensitive organ is typically the red bone marrow, which has a dose threshold of 2.32 Sv and a D_{50} of 5.6 Sv for causing fatal hematopoietic syndrome. Other fatal acute health effects include pulmonary syndrome from sufficiently large acute lung dose and gastrointestinal syndrome from sufficiently large acute stomach dose.

For stochastic health effects, NRC uses the linear-no-threshold (LNT) model for analyses serving regulatory purposes and this model is adopted in MACCS here. The LNT dose-response relationship suggests that any increase in dose, no matter how small, results in an incremental increase in risk. To support the SOARCA project, an Oak Ridge National Laboratory (ORNL) letter report “Radiation Dose and Health Risk Estimation: Technical Basis for the State-of-the-Art Reactor Consequence Analysis Project,” [58] was written to provide a basis for the risk factors which are taken from FGR-13 [19] and have their origin in the BEIR V report [59]. The cancer risk factors are used to convert doses to 8 different organs to the risk of an individual contracting 8

different fatal cancers: leukemia, bone, breast, lung, thyroid, liver, colon, and residual. Residual cancers represent all types of cancer that are not explicitly treated.

The BEIR V report also specified the dose and dose rate effectiveness factor (DDREF), which is used to reduce the health impact of low doses and dose rates. The DDREF is applied to all doses in the intermediate and long-term phases and to those doses in the early phase that are less than a threshold value of 20 rem (0.2 Sv) to the target organ. This guidance for the application of the DDREF is identical to the recommendations provided in ICRP 60 [60]. The DDREF is given a value of 2 for central estimates of most cancer types with the exception of breast and thyroid cancers, for which the DDREF is assigned a value of 1.

4.3 Offsite Consequence Results

The summary results for the 18 Mark I source term bins and 9 Mark II source term bins are provided in Table 4-22 and Table 4-23, respectively so that they can be mapped to the release categories developed in the project's accident sequence analysis. Individual early fatality risk is presented for the area within 1.3 miles of the site because this ring most closely approximates the area within 1 mile from the site boundary, the area for which NRC's early fatality quantitative health objective (QHO) applies [61]. Individual latent cancer fatality risk is presented for the areas within 10, 50 and 100 mi from the site. The 10-mile area is presented because it corresponds to the QHO for cancer fatality risk [61] and to the plume exposure EPZ. Results are displayed for the 50-mile area because that region corresponds to the ingestion exposure EPZ and is used in NRC's regulatory analyses. Results are also displayed for the 100-mile area for sensitivity calculations in the estimates of relative public health risk reduction. Population dose, offsite cost, land contamination, and population subject to long-term protective actions are provided for the 50-mile and 100-mile areas; the 50-mile area is used for the base case regulatory analysis calculations and the 100-mile area is used for sensitivity calculations in the regulatory analysis. Results for the 100-mile area are also provided to serve as an indication of the extent/fraction of offsite consequences captured within the 50-mile area.

Table 4-22 MACCS results for 18 Mark I source term bins

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individiual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
2	48DF100	0.002%	0.02%	11.4	8	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
3	10DF100	0.01%	0.08%	16.3	6	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
4	7DF1000	0.02%	0.26%	14.9	20	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
5	11DF10	0.06%	0.78%	14.4	4	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
6	48	0.23%	1.69%	11.4	8	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
7	15	0.60%	5.85%	14.9	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
8	46	0.98%	11.01%	14.8	17	0	1.53E-04	4.59E-05	2.34E-05	790,000	1,410,000
9	5DF10	1.05%	2.89%	24.2	34	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
10	5	1.39%	6.46%	24.2	41	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
11	8	1.49%	19.25%	14.9	5	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
12	1	1.93%	22.68%	14.9	22	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
13	41DF1000	3.40%	7.65%	9.8	17	0	5.22E-04	1.49E-04	7.89E-05	1,900,000	3,610,000
14	22dw	2.82%	18.64%	14.9	27	0	4.27E-04	1.28E-04	6.57E-05	1,830,000	3,320,000
15	53	2.79%	29.05%	17.4	13	0	2.59E-04	1.19E-04	6.96E-05	1,740,000	3,520,000
16	41	4.54%	14.10%	9.8	16	0	5.57E-04	1.75E-04	9.82E-05	2,300,000	4,520,000
17	3DF10	8.85%	24.65%	9.8	63	0	7.10E-04	2.95E-04	1.68E-04	3,830,000	7,720,000
18	52	15.90%	34.32%	17.4	11	0	5.39E-04	2.23E-04	1.50E-04	3,080,000	6,870,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	78,900,000	78,900,000	0	0	-	-
2	48DF100	0.002%	0.02%	11.4	8	79,700,000	79,700,000	1	1	0	0
3	10DF100	0.01%	0.08%	16.3	6	98,100,000	98,700,000	10	11	1	1
4	7DF1000	0.02%	0.26%	14.9	20	141,000,000	141,000,000	23	23	7	7
5	11DF10	0.06%	0.78%	14.4	4	220,000,000	240,000,000	41	65	118	118
6	48	0.23%	1.69%	11.4	8	1,150,000,000	1,390,000,000	116	175	3,440	3,440
7	15	0.60%	5.85%	14.9	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
8	46	0.98%	11.01%	14.8	17	3,760,000,000	5,220,000,000	242	506	20,700	27,400
9	5DF10	1.05%	2.89%	24.2	34	7,290,000,000	8,600,000,000	351	429	35,200	35,200
10	5	1.39%	6.46%	24.2	41	9,900,000,000	12,000,000,000	479	715	51,400	51,500
11	8	1.49%	19.25%	14.9	5	5,960,000,000	9,720,000,000	286	673	40,500	55,800
12	1	1.93%	22.68%	14.9	22	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
13	41DF1000	3.40%	7.65%	9.8	17	19,400,000,000	24,700,000,000	783	1,170	168,000	190,000
14	22dw	2.82%	18.64%	14.9	27	12,900,000,000	18,300,000,000	544	1,010	93,700	114,000
15	53	2.79%	29.05%	17.4	13	15,700,000,000	26,500,000,000	573	1,290	111,000	142,000
16	41	4.54%	14.10%	9.8	16	25,500,000,000	35,400,000,000	904	1,500	235,000	281,000
17	3DF10	8.85%	24.65%	9.8	63	47,000,000,000	68,100,000,000	1,360	2,470	417,000	504,000
18	52	15.90%	34.32%	17.4	11	46,500,000,000	87,700,000,000	987	2,170	467,000	873,000

∗ Note: For the purpose of quantifying the time signature of a source term release, an hourly plume segment is considered “significant” if it contributes at least 0.5% of that source term’s total cumulative cesium release to the environment. Cesium, rather than iodine, was selected here because all of the resulting offsite consequences are driven by long-term phase exposures.

Table 4-23 MACCS results for 9 Mark II source term bins

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	0	9.72E-08	1.03E-08	3.45E-09	282	345
2	5DF1000	0.0006%	0.005%	32.2	20	0	1.15E-06	1.81E-07	6.35E-08	4,340	5,440
3	42DF100	0.0043%	0.037%	14.3	13	0	6.58E-06	8.67E-07	3.02E-07	20,700	26,700
4	11	0.042%	0.45%	20.3	20	0	7.90E-05	9.68E-06	3.27E-06	202,000	261,000
5	51DF10	0.23%	2.01%	16.6	9	0	1.35E-04	3.39E-05	1.21E-05	689,000	888,000
6	5	0.55%	4.94%	32.2	20	0	2.29E-04	1.05E-04	4.01E-05	2,160,000	2,900,000
7	3	1.09%	10.26%	14.3	20	0	3.08E-04	1.88E-04	7.43E-05	4,140,000	5,580,000
8	1	2.46%	19.81%	22.8	25	0	4.70E-04	3.17E-04	1.25E-04	6,110,000	8,260,000
9	52	3.57%	28.67%	16.6	10	0	4.03E-04	2.46E-04	1.01E-04	5,430,000	7,440,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	381,000,000	381,000,000	-	-	-	-
2	5DF1000	0.0006%	0.005%	32.2	20	381,000,000	381,000,000	0	0	-	-
3	42DF100	0.0043%	0.037%	14.3	13	393,000,000	393,000,000	2	2	0	0
4	11	0.042%	0.45%	20.3	20	844,000,000	846,000,000	44	47	1,030	1,030
5	51DF10	0.23%	2.01%	16.6	9	4,250,000,000	4,380,000,000	130	221	15,400	15,400
6	5	0.55%	4.94%	32.2	20	24,000,000,000	28,000,000,000	303	551	62,400	62,400
7	3	1.09%	10.26%	14.3	20	80,800,000,000	105,400,000,000	698	1,200	619,000	649,000
8	1	2.46%	19.81%	22.8	25	85,500,000,000	109,300,000,000	854	1,680	721,000	741,000
9	52	3.57%	28.67%	16.6	10	53,600,000,000	63,800,000,000	618	1,400	414,000	449,000

∗ Note: For the purpose of quantifying the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5% of that source term's total cumulative cesium release to the environment. Cesium, rather than iodine, was selected here because all of the resulting offsite consequences are driven by long-term phase exposures.

4.3.1 Individual Early Fatality Risk

Individual early fatality risk was calculated to be zero for all Mark I and Mark II source term bin cases within 1.3 miles of the site and beyond as shown in Table 4-22 and Table 4-23. This is because the source terms are not large enough to exceed the threshold for the acute dose to the red bone marrow, which is typically the most sensitive tissue for early fatalities. As described previously in Section 3.10, the dose threshold for the red bone marrow is 2.32 Sv (232 rem). Among the 18 Mark I and 9 Mark II calculations, the largest peak dose to the red bone marrow, averaged over all weather trials, is about 35 rem, well below the dose threshold, for the closest populated ring between 0.33 and 0.75 miles of the site. The computed results depend on meteorological conditions and population distribution, which are clearly site-specific. Thus, had other sites been selected for offsite consequence modeling than Peach Bottom and Limerick, calculated individual early fatality risk might have been nonzero, however recent NRC consequence studies such as SOARCA [9] and SOARCA uncertainty analyes [14] suggest they could still be characterized as "essentially zero."

4.3.2 Individual Latent Cancer Fatality Risk

Individual latent cancer fatality (LCF) risk is defined as the risk of an average individual in a given spatial grid element contracting a fatal cancer from the radiation exposure pathways that may be applied to the residents of the area. These include early phase cloudshine, groundshine, inhalation, and skin deposition; intermediate phase groundshine and inhalation of resuspended materials; and long-term phase groundshine and inhalation of resuspended materials. Individual LCF risk is a population-weighted metric and is calculated by dividing the expected number of

fatal latent cancers in a spatial grid element by the population residing in that grid element. The expected number of fatal latent cancers considers the doses to each population in a grid element for the relevant pathways and multiplies them by the cancer risk factors specified for the 8 fatal cancer types considered. The risk to each organ is then added together and the risk from each phase is added together to arrive at an expected number of latent cancer deaths.

In general, individual LCF risk increases with source term magnitude for the 18 Mark I and 9 Mark II source term bins. This is shown in Figure 4-4 for the Mark I source terms and Figure 4-5 for the Mark II source terms. Larger source terms generally result in higher radionuclide concentrations across the spatial grid resulting in higher expected doses and therefore higher numbers of health effects. However, there are certain cases when a slightly larger source term can result in a lower individual LCF risk. For example, Mark I bin 10 has a slightly smaller cesium release (1.39%) than Mark I bin 11 (1.49%), yet bin 10 leads to a higher individual LCF risk for all three areas considered. (Mark I bin 10 causes a conditional (per event) individual LCF risk of 4E-04 within 10 miles whereas Mark I bin 11 causes a conditional individual LCF risk of ~1E-04 for the same area.) This is likely explained by the different time signature of each source term. Mark I bin 10 has a more gradual release; whereas, Mark I bin 11 is more of a pulse-type release. One way to quantify the time signature of a release is to identify the number of hours in which a significant portion of the source term is released. Because the different source term bins span many orders of magnitude, the determination of significant is based on a percent of the individual source term's total release, rather than an activity level. A threshold of 0.5% would consider any hourly plume segment to be significant if it releases half a percent of that individual source term's total cesium. For example, if the total cumulative environmental release of a source term was 2% of the core inventory of cesium, an hourly plume segment would be considered in this quantification if it released at least 0.01% of the total core inventory. Using this threshold, one can quantify the difference in release profiles. Mark I bin 10 contains 41 hours in which at least 0.5% of the source term's total Cs is released, while Mark I bin 11 contains just 5 hours in which at least 0.5% of the source term's total Cs is released. Mark I bin 10 causes higher individual LCF risks because the longer release duration allows more time for the wind to change direction, resulting in radionuclides being transported across a larger portion of the grid. When radionuclides are spread in more directions across the grid, there is a greater chance that the plume will intersect with population centers leading to a higher individual LCF risk. The larger consequences of a longer, more gradual release compared with a shorter, more punctuated release are more pronounced at shorter distances and less apparent at longer distances. This trend results from the fact that the exposures from narrower, more concentrated plume patterns provide significant exposures at longer distances; whereas, less concentrated plumes that are spread over multiple compass sectors become depleted at shorter distances.

Individual LCF risk generally decreases when larger areas are considered because contamination levels generally decrease with distance. Thus the 10-mile area risk is the largest for each source term followed by the 50-mile area and then the 100-mile area has the smallest average individual risk for each source term. Smaller source terms generally result in a larger difference in individual LCF risk between small (10-mi) and larger (50-, 100-mi) areas because smaller source terms generally lead to smaller exposures and therefore the expected number of fatal latent cancers drops faster as a function of distance from the site. For example, Mark II source term bin 1 has a 10-mile individual LCF risk about 28 times higher than the 100-mile value; whereas, Mark II source term bin 9 has a 10-mile individual LCF risk just 4 times higher than the 100-mile value.

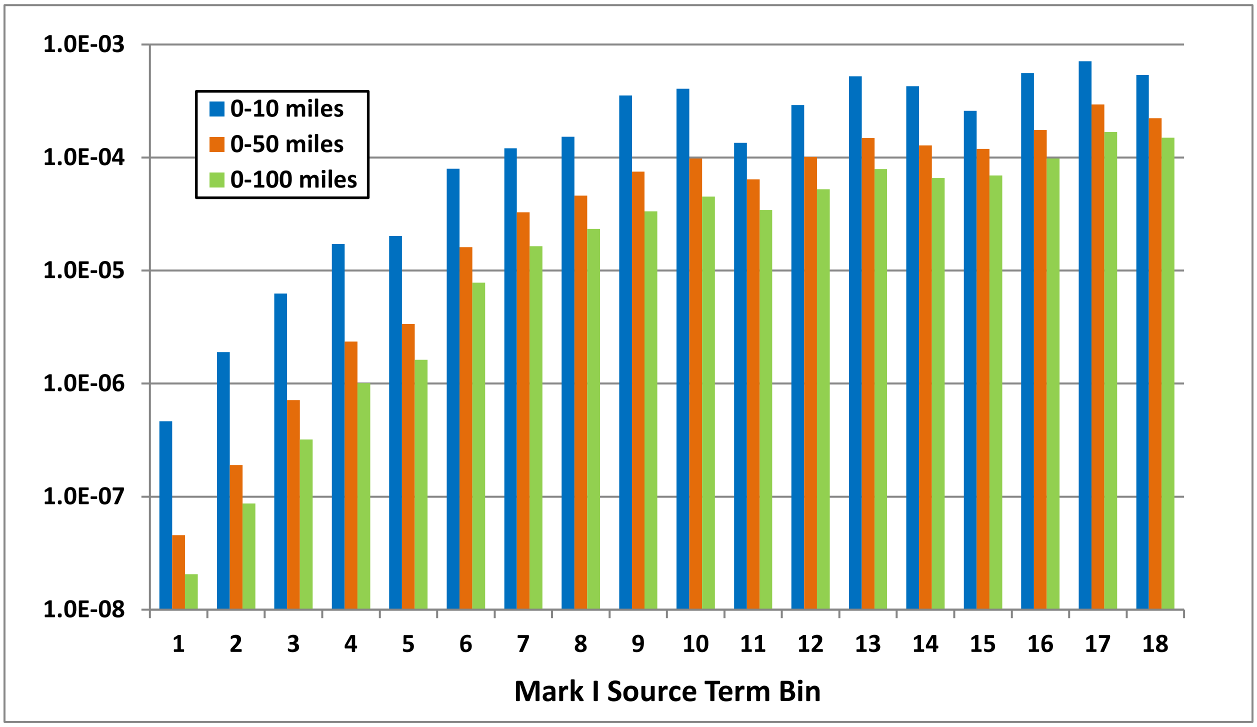


Figure 4-4 Mark I source terms – individual latent cancer fatality risk per event

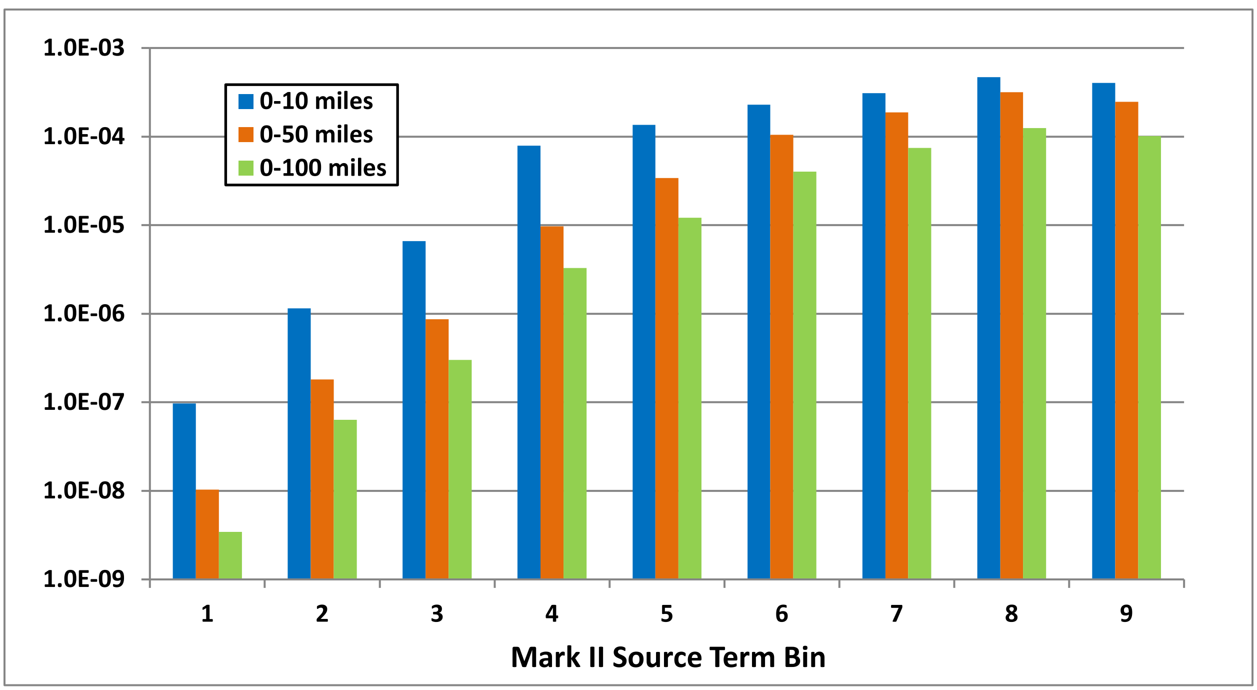


Figure 4-5 Mark II source terms – individual latent cancer fatality risk per event

Figure 4-6 and Figure 4-7 show the fraction of the average individual LCF risk that results from the emergency phase and from the combination of the intermediate and long-term phases for the Mark I and Mark II source term bins, respectively, for the 10- and 50-mile areas. These figures show that the intermediate and long-term phases dominate compared to the emergency phase. Also, they show that the emergency phase has a significantly smaller contribution to individual LCF risk for the 10-mile area compared to the 50-mile area for all Mark I and Mark II source terms. The emergency phase contributes 2% on average (simple average, not frequency-weighted) for the 18 Mark I source terms and 3% on average for the 9 Mark II source terms for the 10-mile area; whereas, the emergency phase contributes 25% and 28% on average for the Mark I and Mark II source term bins, respectively, for the 50-mile area. This difference is a result of the enhanced protective actions for the 10-mile area relative to areas beyond 10 miles that are expected in a real accident and modeled in MACCS. The EPZ population (excluding a very small nonevacuating cohort) is modeled to receive early notification, shelter, and evacuate; and half of the EPZ population is modeled to consume KI.

Observation of Figure 4-6 and Figure 4-7 also shows that the relative contributions of the different accident phases to individual LCF risk vary with the source term. This is largely a function of the relative cesium and iodine source term sizes for each bin. Iodine has greater potential for early health effects whereas cesium has greater potential for long-term health effects. Source terms with a relatively high iodine release lead to relatively larger contributions from the emergency phase. For example, Mark I bin 11 and bin 12 have an iodine release about 12 times larger (in terms of percent of core inventory) than the cesium release and these lead to the highest contribution from the emergency phase for the 50-mile area (44% for bin 11, 37% for bin 12). In contrast, Mark I bin 9 has an iodine release fraction just about three times higher than its cesium release fraction and this leads to the smallest contribution from the emergency phase for the 50-mile area (11% compared to 89% from the intermediate and long-term phases).

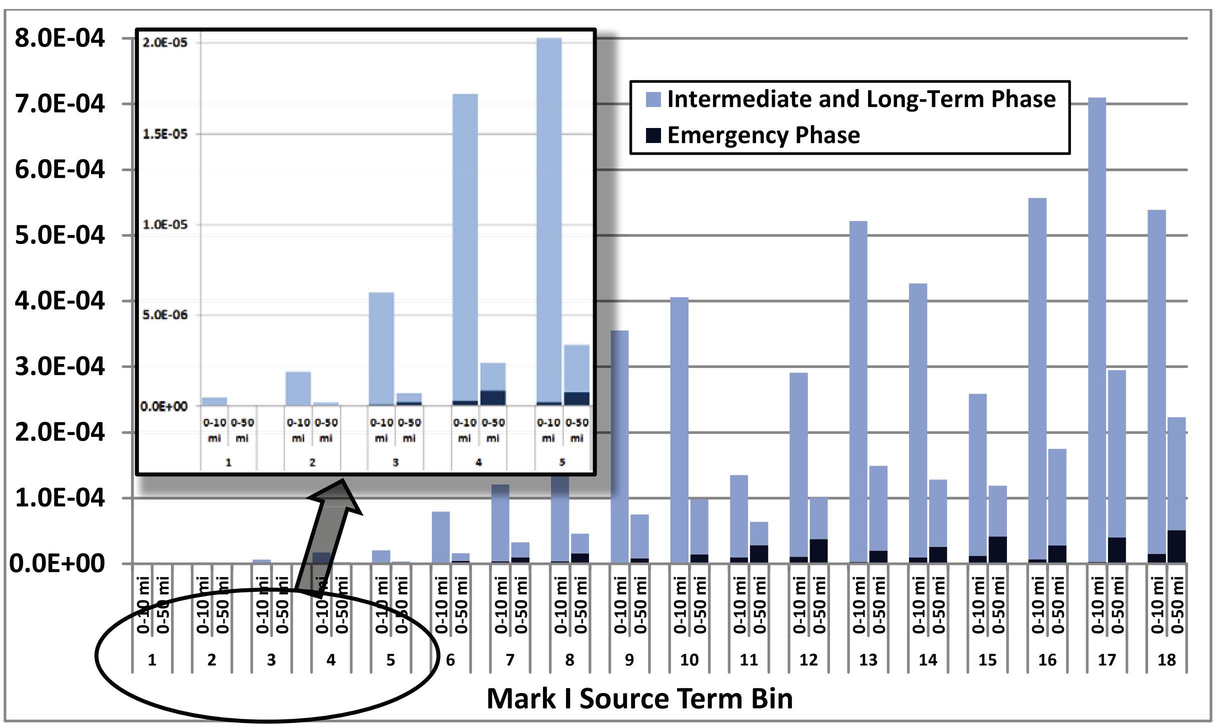


Figure 4-6 Mark I source terms – individual latent cancer fatality risk per event with breakdown by accident phase

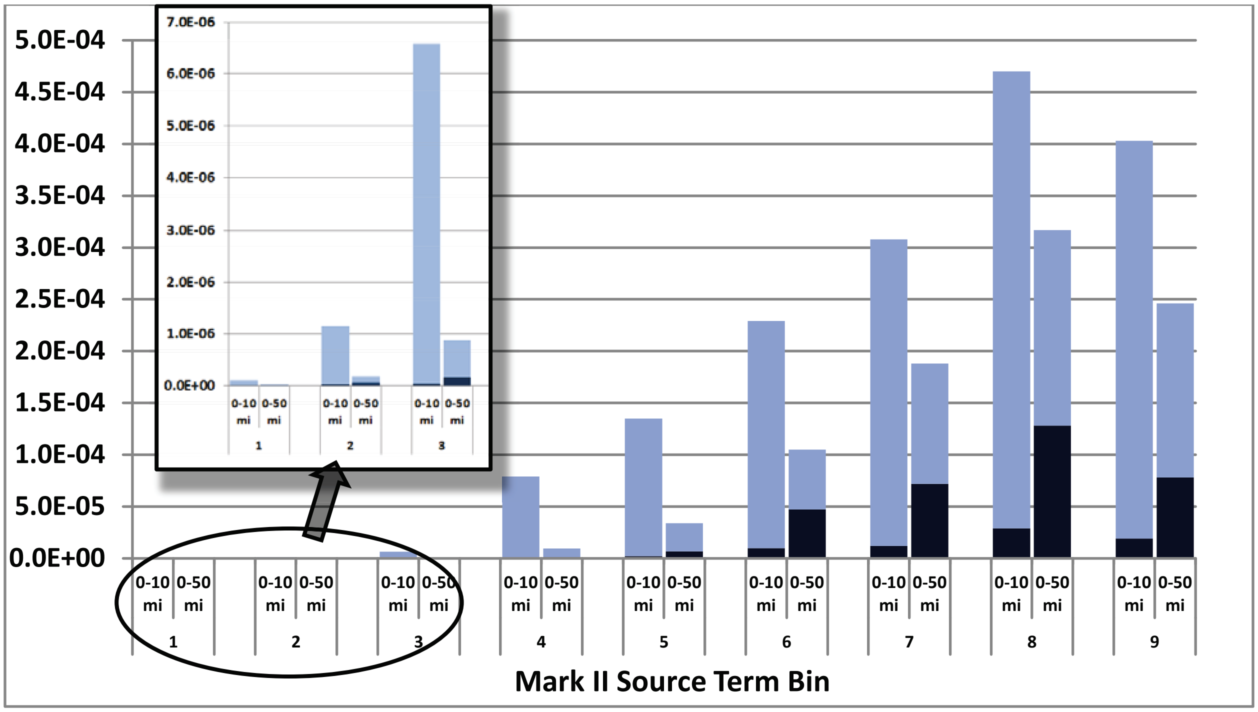


Figure 4-7 Mark II source terms – individual latent cancer fatality risk per event with breakdown by accident phase

4.3.3 Population Dose

Population dose provides a way to characterize the societal consequences of an accident. This metric sums the doses from all exposure pathways (those used in the latent cancer fatality risk calculation as well as food and water ingestion and groundshine to decontamination workers) and multiplies them by the size of the population that would be expected to receive them for the calculated time duration. The population dose, measured in person-rem, is shown in Figure 4-8 and Figure 4-9 for the Mark I and Mark II source terms, respectively. Each figure shows both the 50-mile and 100-mile area. As with individual LCF risk, the population dose generally increases with source term magnitude. As a secondary consideration, the population dose also generally increases for source terms that span a longer duration. For example, Mark I bin 18 has a cesium release almost double Mark I bin 17, however Mark I bin 18 results in a lower weather-averaged population dose (~3,100,000 person-rem) than Mark I bin 17 (~3,800,000 person-rem). Mark I bin 17 likely has a higher population dose because it has a much longer release duration with more hours of significant cesium release. Mark I bin 17 has 63 1-hour periods in which at least 0.5% of the total source term is being released; whereas, Mark I bin 18 has just 11 1-hour periods in which at least 0.5% of the total source term is being released. A longer release duration increases the probability that the plumes intersect with population centers, which drives up the population dose.

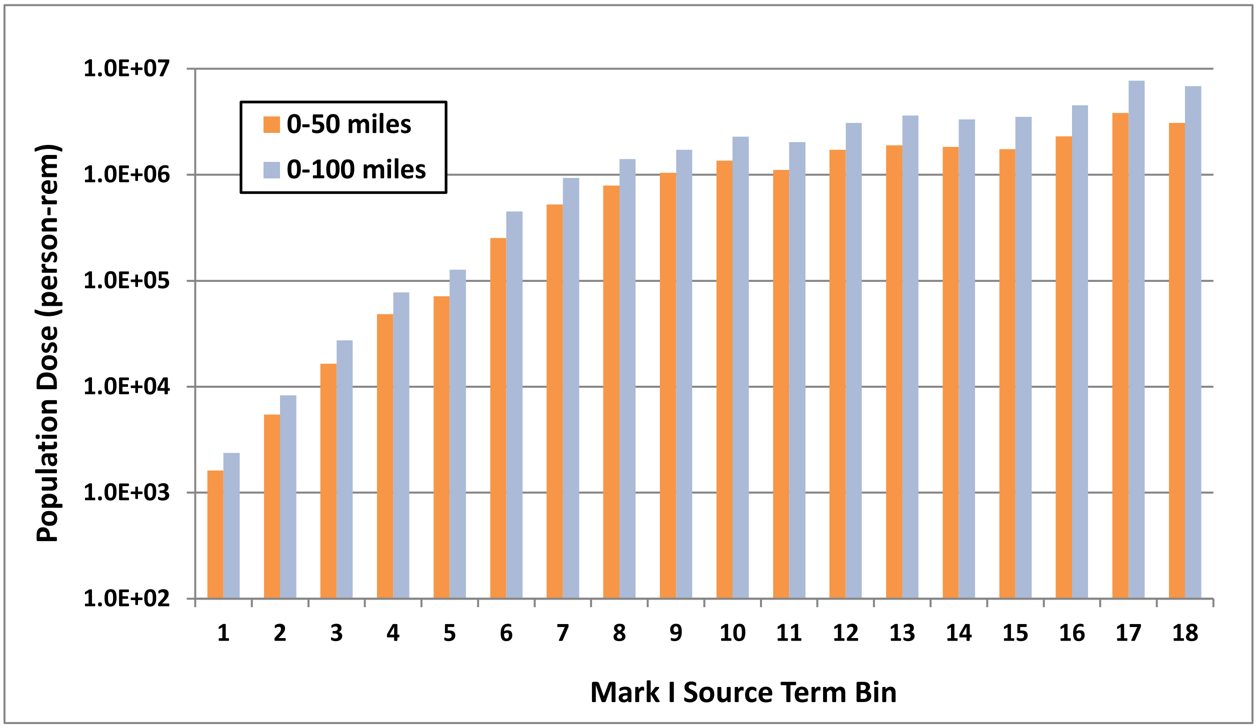


Figure 4-8 Mark I source terms – population dose per event (person-rem)

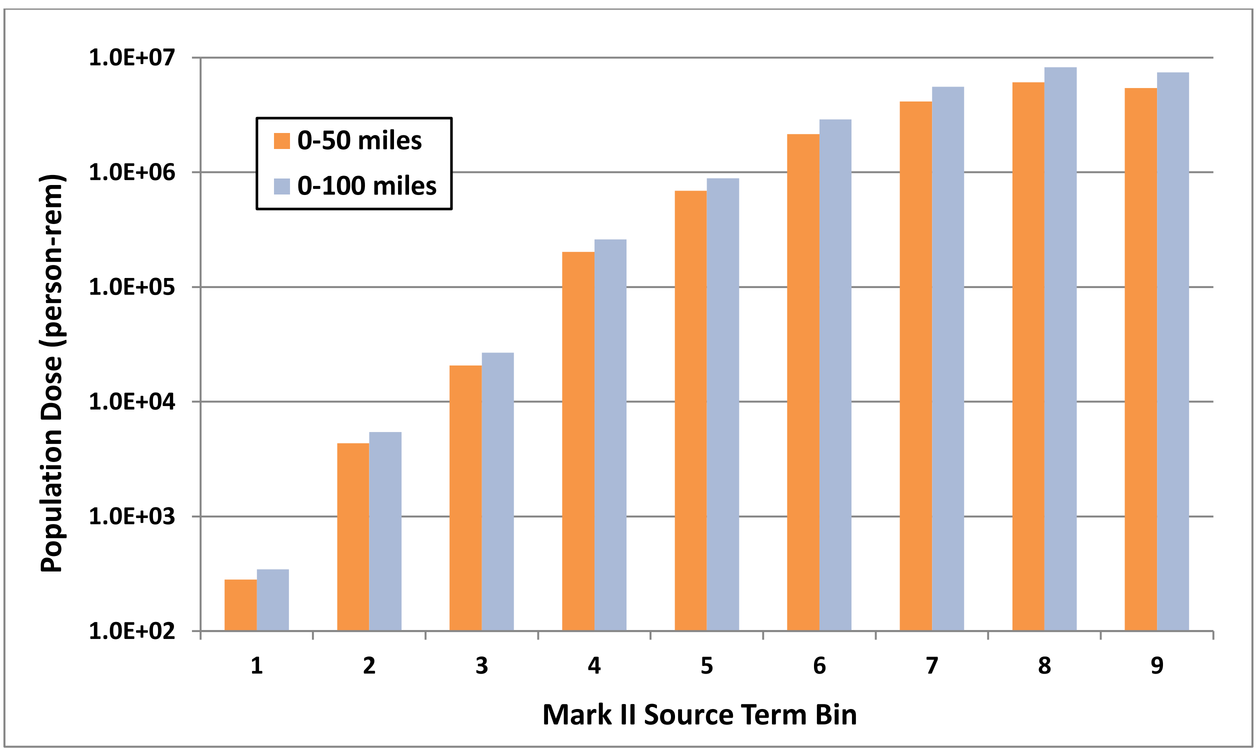


Figure 4-9 Mark II source terms – population dose per event (person-rem)

Figure 4-10 and Figure 4-11 show the population dose fraction that results from the different accident phases and exposure pathways for the area within 50 miles of each site. Each figure includes an inset chart to rescale the population dose for the smaller source term bins so they are legible. The fraction from the emergency phase ranges from about 15% to 60% of the total population dose. This fraction depends on the size of the source term's iodine release relative to its cesium release. Source terms with relatively high iodine releases (Mark I bin 11 and Mark I bin 12) have a larger fraction of the population dose from the emergency phase. For the Mark II source terms, the emergency phase fraction generally increases with source term magnitude. The ratio of iodine release to cesium release is fairly consistent, around an order of magnitude, so the trend is more consistent than for the Mark I source term bins.

The groundshine exposure pathway, considering both the intermediate and long-term phases, is also a major contributor to population dose, ranging from about 20% to 70%. This pathway's contribution generally increases with the magnitude of the cesium release because this creates more offsite contamination for a longer period of time. The groundshine exposure pathway doses are accumulated for the population that live on land that has been contaminated but at doses that are under the intermediate phase relocation criterion and the long-term phase habitability criterion.

The long-term phase food and water ingestion exposure pathway is a significant contributor to population dose for the smaller source terms, but this pathway's relative contribution (fraction of dose) drops as the source term increases. This is because larger source terms create progressively more groundshine exposure dose so the relative contribution from ingestion decreases as a result.

Intermediate and long-term phase resuspension dose is a very small contributor to population dose for all source terms because this pathway is less effective in causing doses to humans than others, like groundshine.

The long-term phase decontamination exposure doses, both to farmland and populated land, are also very small contributors to population dose. These increase for larger cesium source terms because more cesium release contaminates more land and thus requires more decontamination. Research efforts are underway to evaluate newly emerging information from the Fukushima accident recovery experience, and in particular develop MACCS decontamination plan input parameters based on Fukushima. The decontamination plan input parameters include the costs to decontaminate, the dose reductions achieved, and the times required to perform decontamination. These research efforts were not completed in time for this analysis, but preliminary information, had it been used in this project, may have shown a higher contribution to total population dose from the decontamination exposure pathway. Additional information from the Fukushima accident related to total offsite accident cost is provided in the following section.

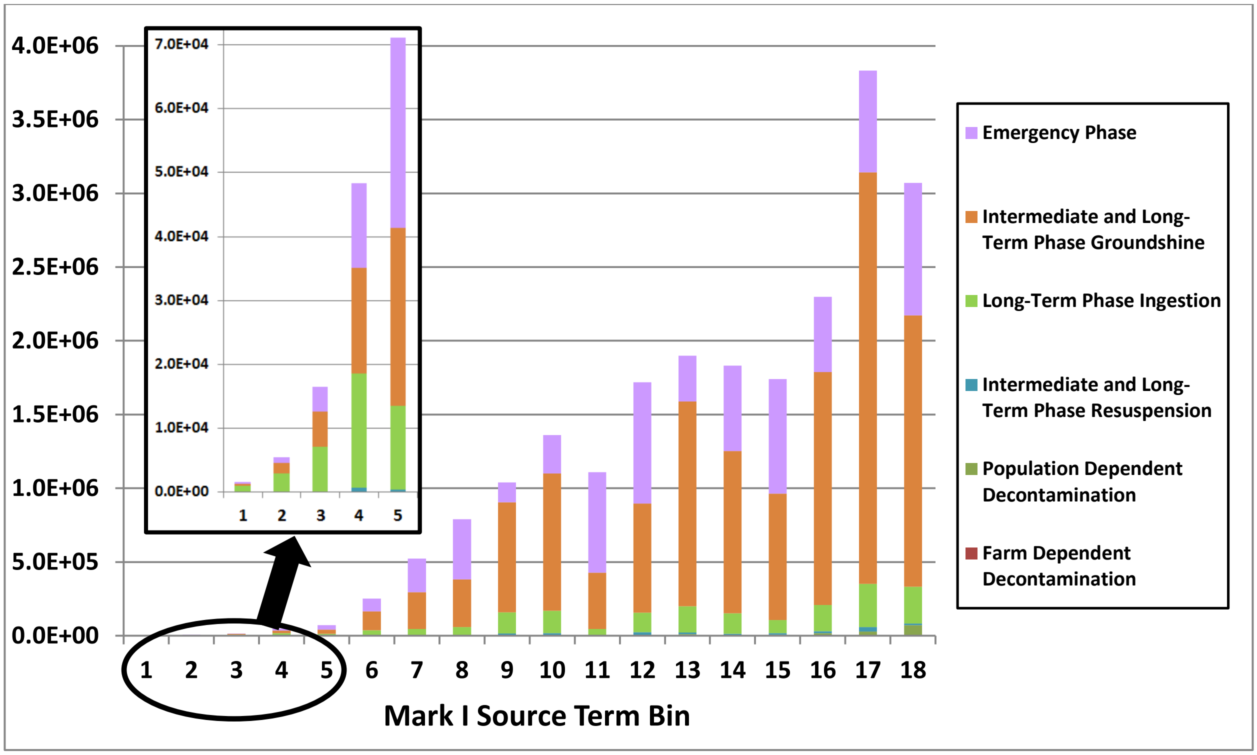


Figure 4-10 Mark I source terms – population dose per event (person-rem) with breakdown by accident phase and exposure pathway (0-50 miles)

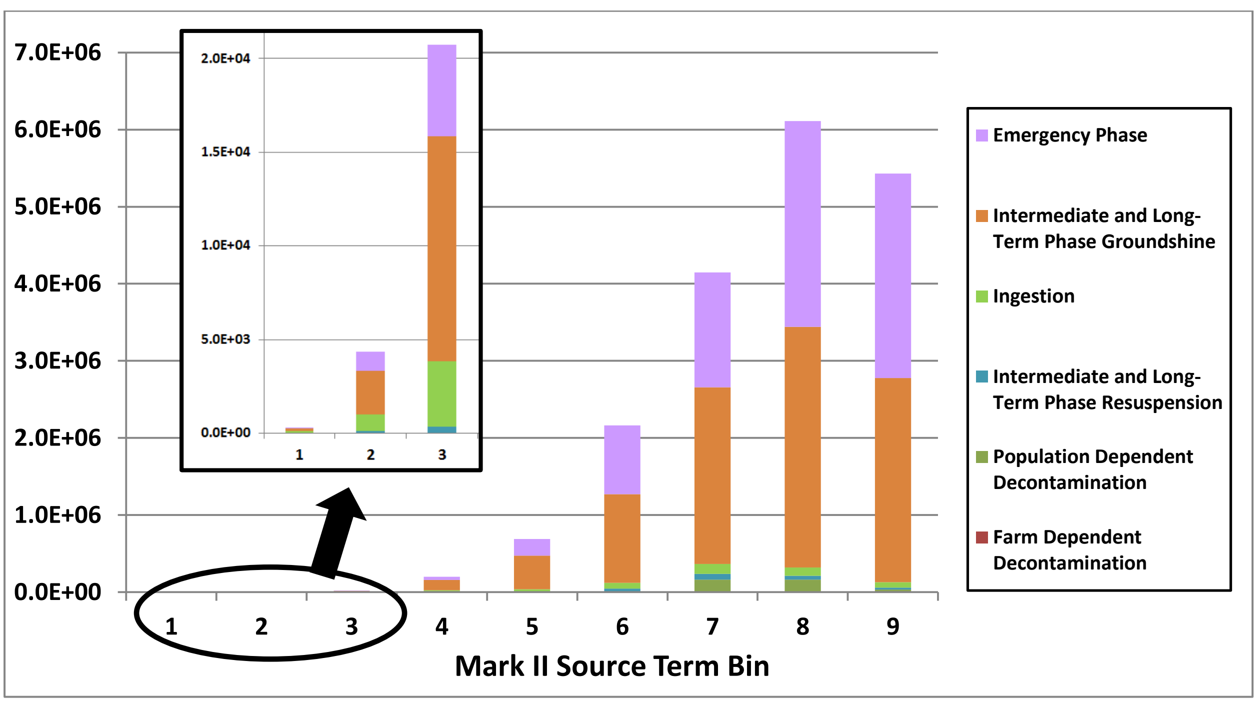


Figure 4-11 Mark II source terms – population dose per event (person-rem) with breakdown by accident phase and exposure pathway (0-50 miles)

4.3.4 Offsite Economic Cost

As with total population dose, offsite economic cost is a way to characterize the societal consequences of an accident. This metric sums the costs of the protective actions that need to be taken to reduce offsite exposures to avoid and minimize health effects and to restore land to usability and habitability. These costs include evacuation and relocation costs, moving expenses for people displaced, decontamination costs, costs due to loss of use of property, disposal of contaminated food, and the costs of condemning land. These cost categories are described in more detail in Section 2.3.3.

The total offsite cost, measured in 2013 dollars, is shown in Figure 4-12 and Figure 4-13 for the Mark I and Mark II source terms, respectively. Each figure shows both the 50-mile and 100-mile area. Like population dose, the total offsite cost generally increases with source term magnitude. As a secondary consideration, the offsite cost also generally increases for source terms that span a longer duration. For example, Mark II bin 9 has a larger cesium release than Mark II bin 8 but results in a lower offsite cost ($ ~54B) than Mark II bin 8 ($ ~86B). Mark II bin 8 has a higher offsite cost because it has a significantly longer release duration. Mark II bin 8 has 25 hours in which a "significant" portion of the source term is released (for the purpose of quantifying the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5**%** of that source term's total cumulative cesium release to the environment); whereas, Mark II bin 9 has just 10 hours. A longer release duration increases the probability that the plumes intersect with population centers and drives up the economic cost. The larger consequences of a longer, more gradual release compared with a shorter, more punctuated release are more pronounced at shorter distances and less apparent at longer distances. This trend results from the fact that the exposures from narrower, more concentrated plume patterns provide significant exposures at longer distances; whereas, less concentrated plumes that are spread over multiple compass sectors become depleted at shorter distances.

For the smallest Mark I and Mark II source terms, the radionuclide release is too small to necessitate intermediate phase and long-term phase protective actions. Therefore the total offsite cost is essentially just the fixed cost of evacuating the EPZ population for the duration of the emergency phase, 1 week. For example, Mark I bin 1 has an offsite cost of ~ $79M. This cost represents evacuation of the EPZ population and is equal to the population times the per diem cost times the number of days (7). The total offsite cost for the smallest Mark II source term, bin 1, is $381M. This value is larger than for Mark I bin 1 solely as a result of the Mark II reference MACCS model having a higher population density.

Figure 4-12 and Figure 4-13 also show that for the smaller source terms, the total offsite cost is attributed essentially entirely to the area within 50 miles of the site. Using the total offsite cost values from Table 4-22 and Table 4-23, rounding to one significant figure shows that a cesium release of at least about 0.5% is needed to see any difference in the 50-mile and 100-mile area values.

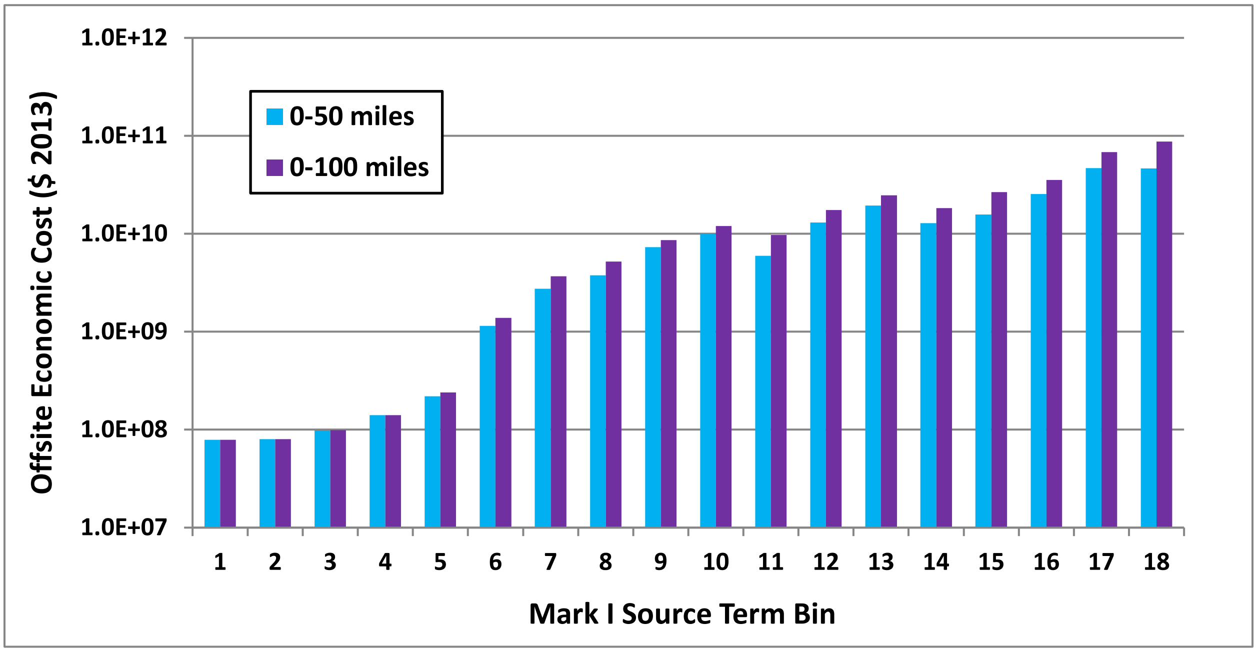


Figure 4-12 Mark I source terms – offsite economic cost ($ 2013)

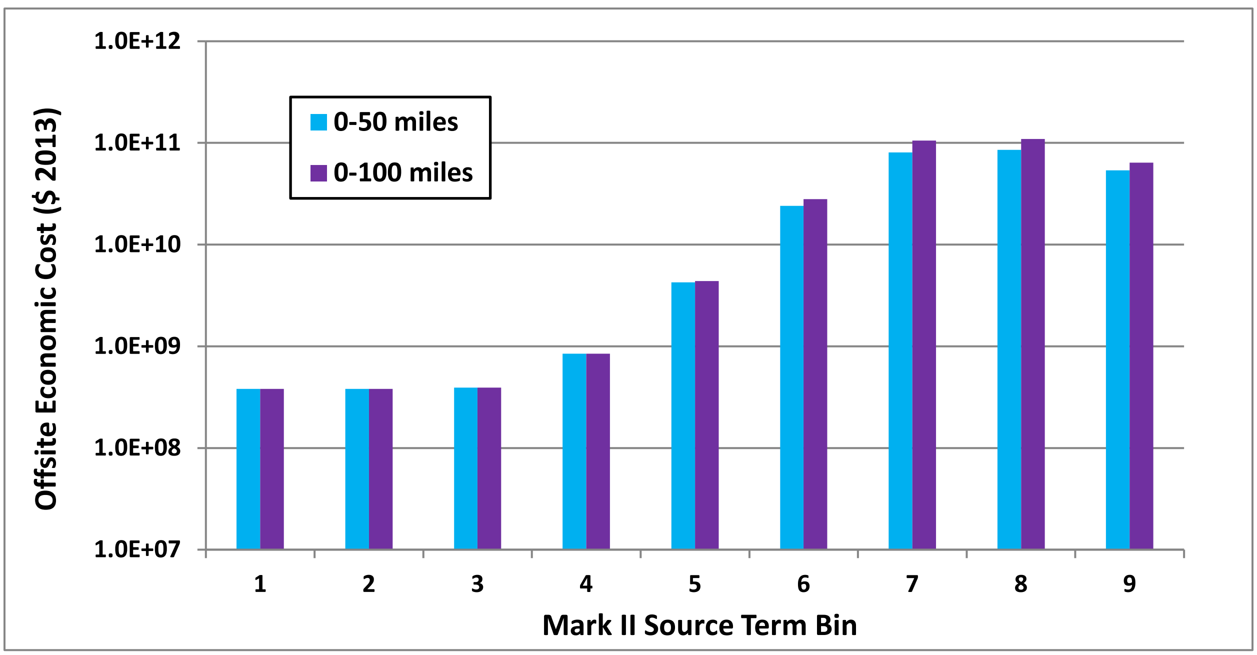


Figure 4-13 Mark II source terms – offsite economic cost ($ 2013)

Figure 4-14 and Figure 4-15 show the total offsite cost fraction that results from the different cost components for the area within 50 miles of each site. For the smaller source terms, the emergency phase cost dominates the total offsite cost because offsite contamination is minimal and therefore long-term phase protective actions are not warranted. As the source terms increase in magnitude, the intermediate phase and population-dependent interdiction costs represent an increasingly dominant share of the total offsite cost. As the source term magnitudes increase further, the farm-based costs start contributing to the total offsite costs (Mark I bins 3-5 and Mark II

bin 4). Farmland costs include the value of milk and crops that cannot be consumed as well as farmland interdiction. However, the farm-based costs quickly decrease in share of the total cost because the population-based costs grow much faster for larger source terms.

The contribution of population-dependent decontamination costs increases with source term magnitude but doesn't account for more than 10% of the total 50-mile offsite cost, even for the largest source terms. As described in the previous section, research efforts are underway to evaluate information from the Fukushima accident recovery experience, and in particular develop MACCS decontamination plan input parameters based on Fukushima. The decontamination plan input parameters include the costs to decontaminate, the dose reductions achieved, and the times required to perform decontamination. These research efforts were not completed in time for this analysis, but preliminary information, had it been used in this project, may have shown higher decontamination costs and a higher contribution to total offsite cost from the decontamination cost component.

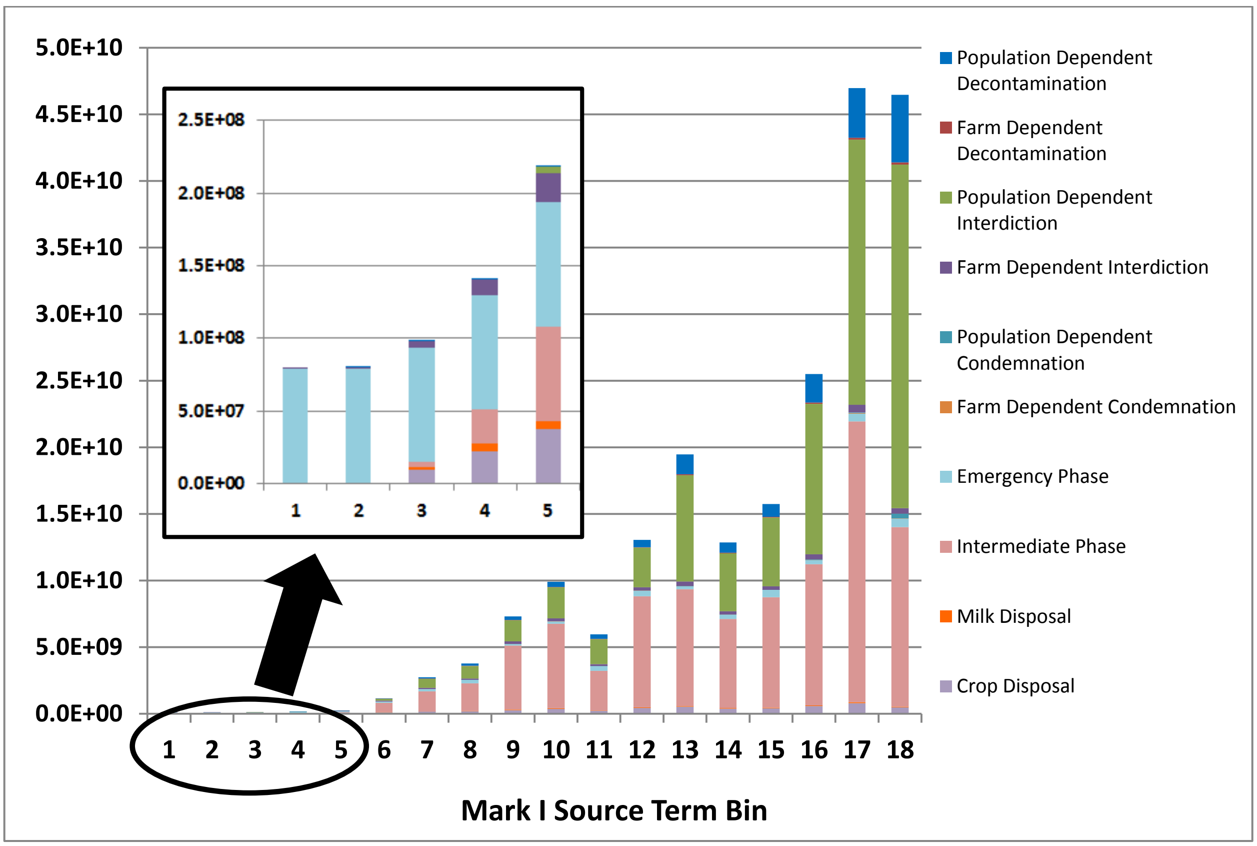


Figure 4-14 Mark I source terms – offsite economic cost ($ 2013) with breakdown by accident phase and protective action type (0-50 miles)

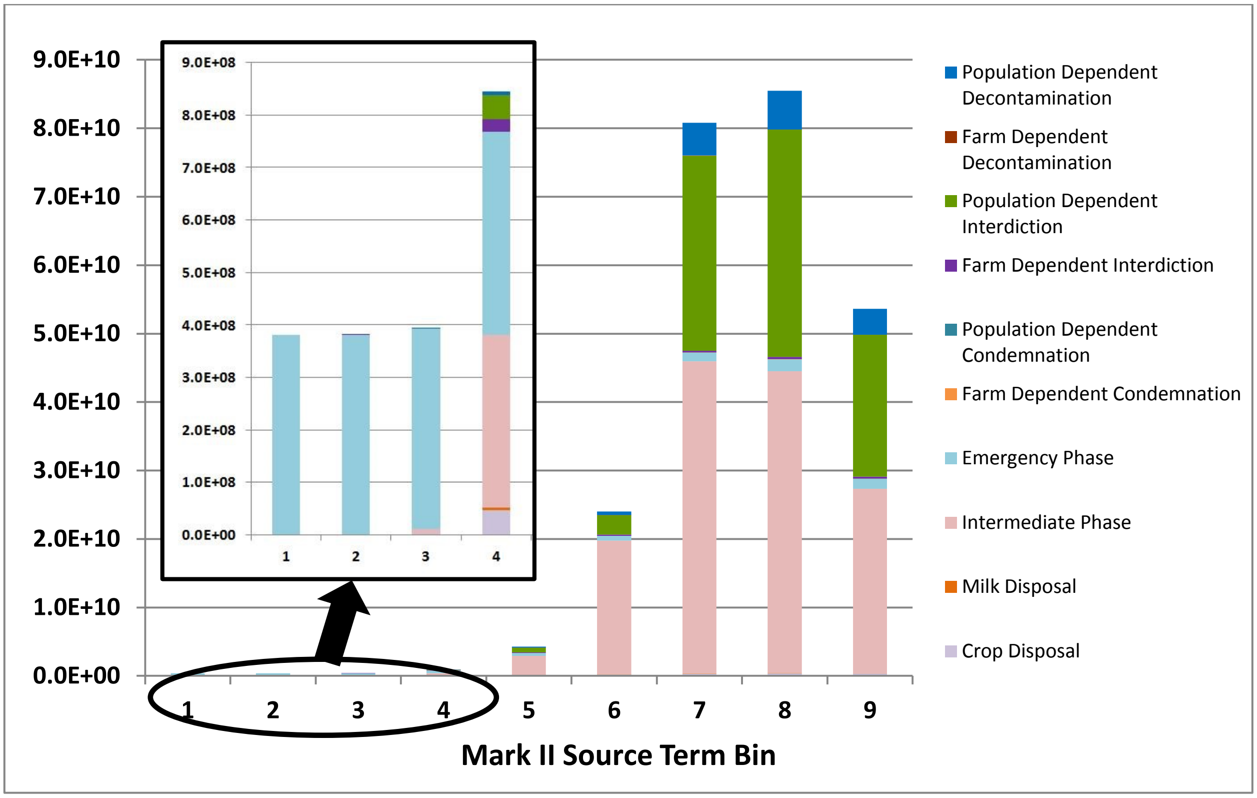


Figure 4-15 Mark II source terms – offsite economic cost ($ 2013) with breakdown by accident phase and protective action type (0-50 miles)

4.3.5 Land Contamination and Population Subject to Long-Term Protective Actions

Beyond total population dose and total offsite economic cost, extent of land contamination and population subject to long-term protective actions represent two additional measures of the societal cost of a nuclear accident. These have the advantage that they are measured in terms (area, number of people) that may be more meaningful to some stakeholders (compared to person-rem or dollars).

Land contamination is measured as the area of land that exceeds the long-term phase habitability criterion, which is modeled here as 500 mrem-per-year, starting in the first year of the long-term phase. This is the land that is either temporarily interdicted with or without decontamination or is condemned, and considers both farmland and populated land. The long-term phase begins after the end of the three-month intermediate phase, which begins after the end of the week-long emergency phase. This metric does not consider the duration of time for which the land exceeds the habitability criterion and is considered too contaminated to be habitable.

Figure 4-16 and Figure 4-17 show the area (square miles) of land exceeding the long-term phase habitability criterion for the 50-mile and 100-mile areas around each site for the Mark I and Mark II source terms. The area of land contaminated increases with source term magnitude and with release duration because a longer release allows more time for the wind to change direction and thus increases the probability of spreading contamination over a larger area. The weather-averaged area of land contamination ranges from essentially zero for the smallest Mark II source term bins to about

1,400 square miles for Mark I bin 17 within 50 miles of the site. This maximum represents about 18% of the land within 50 miles.

Examination of the blue bars representing the 50-mile area and the orange bars representing the 100-mile area shows that the contaminated land is concentrated in the inner 50-mile circle for the small source terms. However for the largest source terms, more than half of the contaminated land is beyond 50 miles from the site. For example, 2,170 square miles are contaminated within 100 miles for Mark I bin 18; 987 square miles (~45%) of which are within 50 miles, and 1,183 square miles (~55%) of which are between 50 and 100 miles.

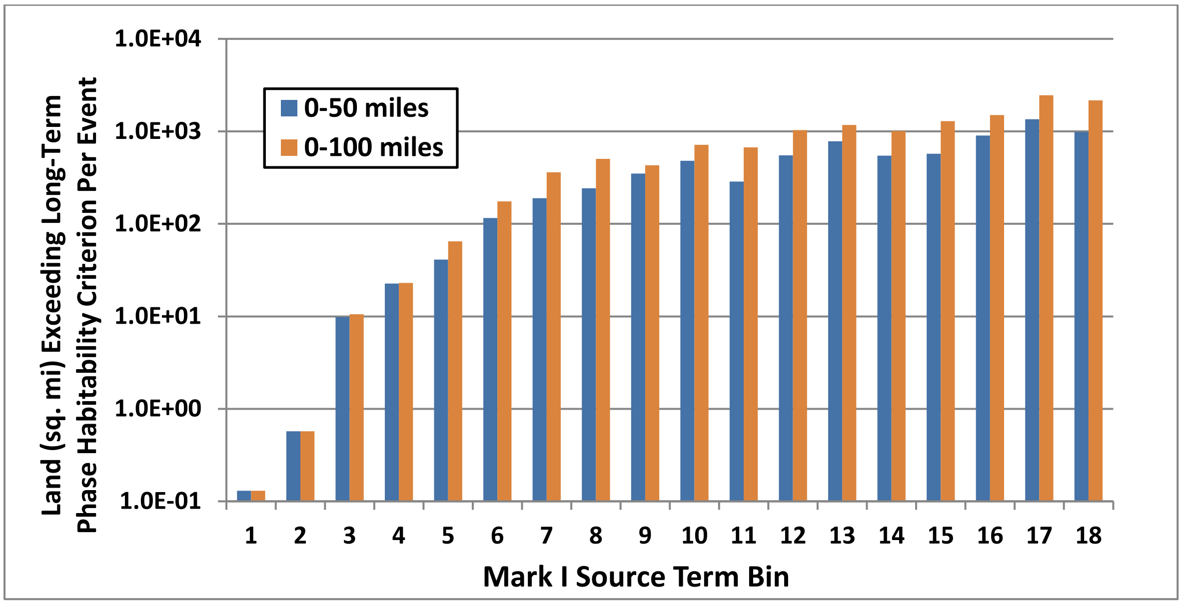


Figure 4-16 Mark I source terms – land exceeding long-term phase habitability criterion (square miles) per event

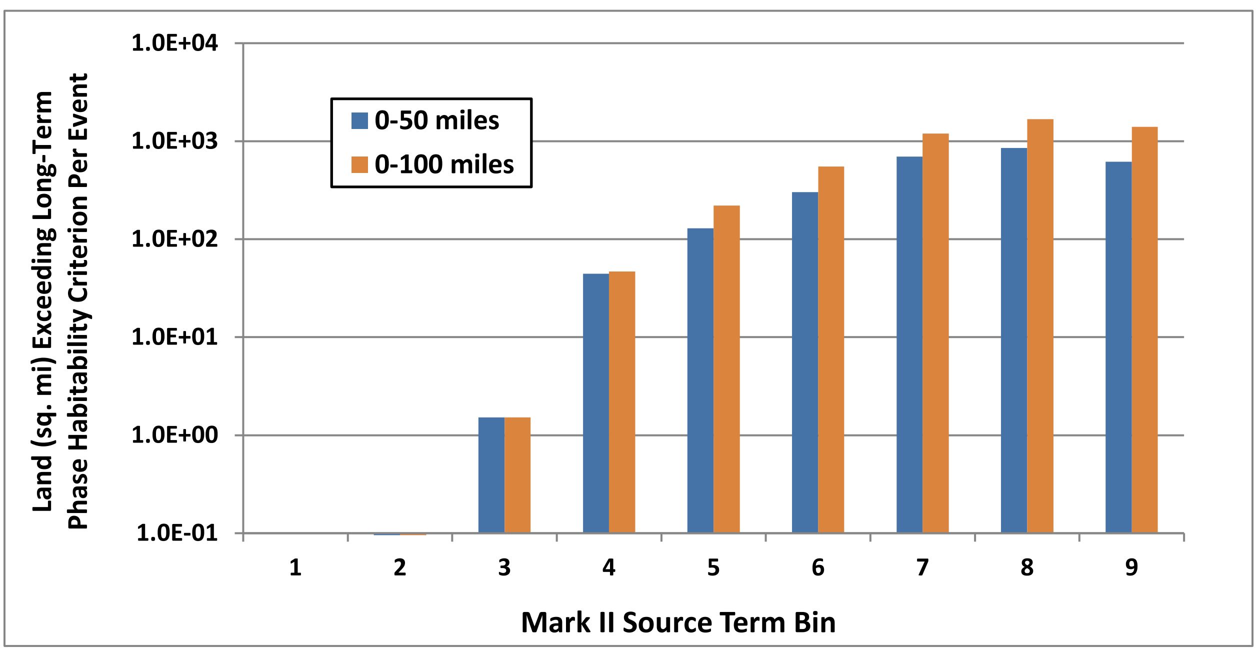


Figure 4-17 Mark II source terms – land exceeding long-term phase habitability criterion (square miles) per event

The weather-averaged number of individuals displaced from their land for long-term phase interdiction or condemnation is shown in Figure 4-18 and Figure 4-19 for the 50-mile and 100-mile areas around each site. This metric does not consider the duration of time for which the land exceeds the habitability criterion and is considered contaminated. Some of the population might be able to return within one year following decontamination while others might need to stay away for a decade or more, or permanently.

The number of people displaced from interdicted and condemned land increases with source term magnitude and with release duration because a longer release allows more time for the wind to change direction and thus increases the probability of spreading contamination over population centers. The weather-averaged population displaced ranges from essentially zero for the smallest source terms to about 720,000 people within 50 miles for Mark II bin 8. This maximum represents about 9% of the total population within 50 miles of the Mark II reference site, Limerick. As with land contamination area, the people displaced from interdicted and condemned land is concentrated in the 50-mile area around each site. Even for the largest source terms, the 50-mile area dominates compared to the 50-100-mile area. For example, Mark II bin 8, which causes the highest population displaced, affects 721,000 people within 50 miles (~97% of the 100-mile population total) compared to just 20,000 people between 50 and 100 miles (~3% of the 100-mile population total).

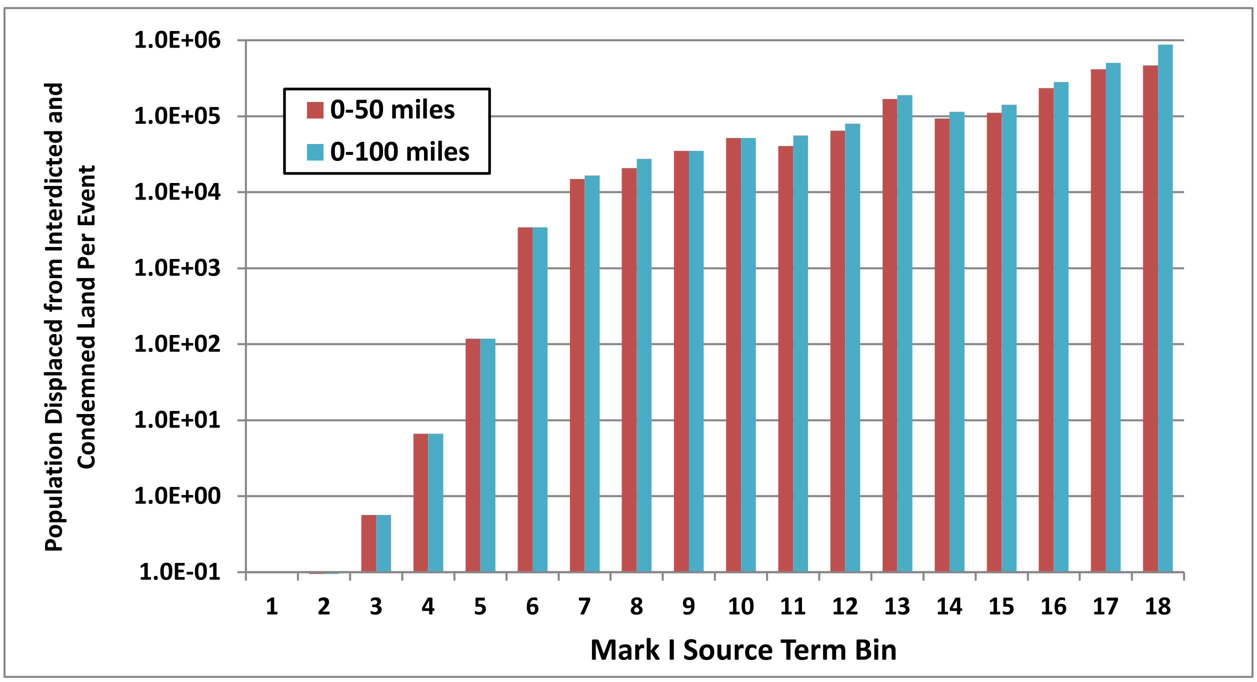


Figure 4-18 Mark I source terms – population displaced from interdicted and condemned land per event

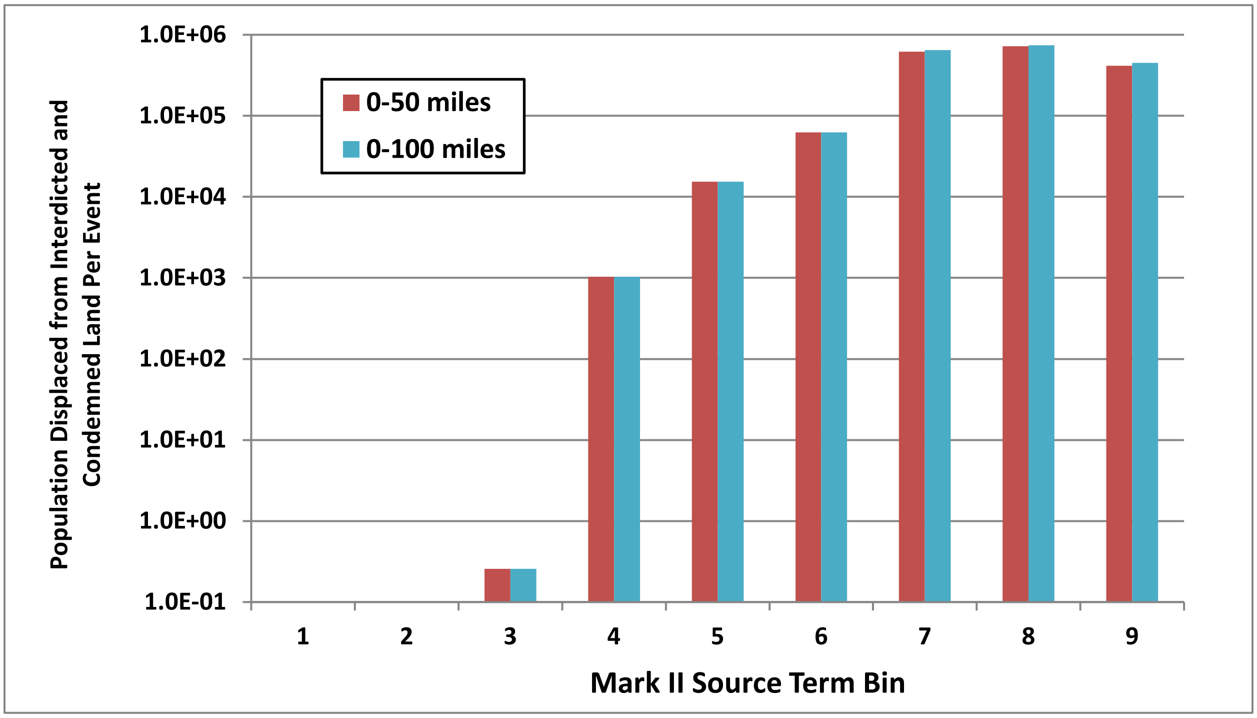


Figure 4-19 Mark II source terms – population displaced from interdicted and condemned land per event

4.4 Sensitivity Analyses

4.4.1 Approach

The overall modeling approach for the offsite consequence analysis component of the CPRR rulemaking effort was to develop and use site-specific MACCS models of a reference BWR Mark I site and of a reference BWR Mark II site. Site-specific features of the model include meteorology, population distribution, land use, economic values, evacuation characteristics and timing, use of KI, and intermediate and long-term phase habitability criteria. However, this rulemaking is designed to apply to the entire fleet of U.S. BWR Mark I and Mark II plants and their various sites have many different features. There are 15 different Mark I sites and 5 different Mark II sites. In order to better capture some of the effects of different site features on the project's results, a number of parameters and features were varied in sensitivity studies. These sensitivities help gauge the impact of different population sizes, different evacuation timelines, different nonevacuating cohort sizes, different intermediate phase durations, and different long-term phase habitability criteria on the calculated offsite consequence metrics. The following sections discuss the approach to each of these.

4.4.1.1 Population

A SecPop site file was created for each Mark I and Mark II site and the 50-mile population value was ranked for each. High, medium, and low population sites were selected to approximate the 90th, 50th, and 10th percentiles, respectively. The population is based on the 2010 U.S. Census and is scaled to 2013 using state level population growth projections from the Census Bureau using the same process described earlier in Section 3.3. Peach Bottom and Limerick were already selected as the reference Mark I and Mark II sites for the baseline calculations and these are used

as the high population sites for the sensitivity calculations. Vermont Yankee and Hatch were selected as the medium and low population Mark I sites. Susquehanna and Columbia were selected as the medium and low population Mark II sites. Population and site data is summarized in Table 4-24. Population data for each Mark I site is presented in Table 4-25 and for each Mark II site in Table 4-26. Population growth multipliers are provided in Table 4-27 for Hatch, Vermont Yankee, Columbia, and Susquehanna.

The SecPop site file contains one value of farmland ($ per hectare) and one value of populated land ($ per person) for each individual spatial grid element across the entire modeling domain. These values for the 50-mile area around each plant were weighted by the quantity of farmland in each and by the population in each to develop a weighted average value for the 50-mile area. These values are also presented in Table 26 and are used in the population sensitivities.

The sensitivity calculations that use Vermont Yankee, Hatch, Susquehanna, or Columbia site data do not use any other site-specific modeling features that would pertain to these four sites. For example, the Mark II population sensitivity calculations using the low population site (Columbia) still use all other parameters and features for the Mark II reference site, Limerick. Thus the Mark II calculations with the Columbia site data still use Limerick meteorology, Limerick evacuation characteristics, and the Pennsylvania state-specific habitability criterion. These sensitivities are included simply to assess how the consequence metrics might change for a less populated area with lower weighted average economic values.

Table 4-24 Sites selected for population sensitivity calculations

		Site	Population within 50 miles	Weighted Average Value of Farmland ($ 2013 per hectare)	Weighted Average Value of Populated Land ($ 2013 per person)
Mark I	High	Peach Bottom	5,645,811	$24,400	$518,000
	Medium	Vermont Yankee	1,536,793	$20,000	$475,000
	Low	Hatch	453,404	$7,800	$285,000
Mark II	High	Limerick	8,108,436	$28,600	$528,000
	Medium	Susquehanna	1,790,924	$14,800	$399,000
	Low	Columbia	464,310	$5,800	$359,000

Table 4-25 Mark I site population information

Mark I	Site	Population within 50 mi	Population Density (per sq mi)	Rank	Percentile
	Dresden	7,374,320	939	1	1.00
HIGH	**Peach Bottom**	**5,645,811**	**719**	**2**	**0.93**
	Hope Creek	5,633,411	717	3	0.86
	Pilgrim	4,851,642	618	4	0.79
	Fermi	4,808,370	612	5	0.71
	Oyster Creek	4,567,689	582	6	0.64
	Monticello	3,052,698	389	7	0.57
MEDIUM	**Vermont Yankee**	**1,536,793**	**196**	**8**	**0.50**
	Browns Ferry	997,194	127	9	0.44
	Fitzpatrick & Nine Mile Point	923,614	118	10	0.36
	Duane Arnold	673,752	86	11	0.29
	Quad Cities	653,780	83	12	0.21
	Brunswick	479,743	61	13	0.14
LOW	**Hatch**	**453,404**	**58**	**14**	**0.07**
	Cooper	159,946	20	15	0.00

Table 4-26 Mark II site population information

Mark II	Site	Population within 50 mi	Population Density (per sq mi)	Rank	Percentile
HIGH	**Limerick**	**8,108,436**	**1,032**	**1**	**1.00**
	LaSalle	1,909,500	243	2	0.75
MEDIUM	**Susquehanna**	**1,790,924**	**228**	**3**	**0.50**
	Nine Mile Point	923,614	118	4	0.25
LOW	**Columbia**	**464,310**	**59**	**5**	**0.00**

Table 4-27 Population growth multipliers for selected sensitivity sites

	Census 2010	Census 2013 Est.	2010 to 2013 Multiplier	Hatch	Vermont Yankee	Columbia	Susquehanna
				Approximate 50-mile Area Fraction			
GA	9,687,653	9,992,167	1.031	1	-	-	-
MA	6,547,629	6,692,824	1.022	-	0.4	-	-
NH	1,316,470	1,323,459	1.005	-	0.3	-	-
NY	19,378,102	19,651,127	1.014	-	0.05	-	-
OR	3,831,074	3,930,065	1.026	-	-	0.1	-
PA	12,702,379	12,773,801	1.006	-	-	-	1
VT	625,741	626,630	1.001	-	0.25	-	-
WA	6,724,540	6,971,406	1.037	-	-	0.9	-
Population Multiplier for 2010 to 2013:				**1.031**	**1.012**	**1.036**	**1.006**

4.4.1.2 Selection of Source Terms for Use in Sensitivity Analyses

The source term bins used in the Mark I and Mark II baseline analyses span over 4 orders of magnitude for cesium release, and the results clearly depend on the size of the release. Rather than run sensitivity calculations for all 18 Mark I source term bins and all 9 Mark II source term bins, a low, medium, and high source term was selected for each containment type. Similar to the population values, the high source term was selected to approximate the 90th percentile, the medium source term was selected to represent the median value, and the low source term was selected to approximate the 10th percentile. Note that these percentiles are computed from considering the spread of bin results; these percentiles do not represent the distribution of frequency-weighted releases.

As shown in Table 4-28, bins 3, 10, and 17 were selected as the low, medium, and high source terms for the Mark I sensitivity calculations. Similarly, Table 4-29 shows that bins 2, 5, and 8 were selected as the low, medium, and high source terms for the Mark II sensitivity calculations.

Table 4-28 Selection of source terms for Mark I sensitivity calculations

	Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Percentile
	1	28DF1000	0.0006%	0.006%	14.9	7	0.000
	2	48DF100	0.002%	0.02%	11.4	8	0.058
LOW	**3**	**10DF100**	**0.0073%**	**0.08%**	**16.3**	**6**	**0.117**
	4	7DF1000	0.02%	0.26%	14.9	20	0.176
	5	11DF10	0.06%	0.78%	14.4	4	0.235
	6	48	0.23%	1.69%	11.4	8	0.294
	7	15	0.60%	5.85%	14.9	7	0.352

	Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Percentile
	8	46	0.98%	11.01%	14.8	17	0.411
	9	5DF10	1.05%	2.89%	24.2	34	0.470
MEDIUM	**10**	**5**	**1.39%**	**6.46%**	**24.2**	**41**	**0.529**
	11	8	1.49%	19.25%	14.9	5	0.588
	12	1	1.93%	22.68%	14.9	22	0.647
	13	41DF1000	3.40%	7.65%	9.8	17	0.823
	14	22dw	2.82%	18.64%	14.9	27	0.764
	15	53	2.79%	29.05%	17.4	13	0.705
	16	41	4.54%	14.10%	9.8	16	0.882
HIGH	**17**	**3DF10**	**8.85%**	**24.65%**	**9.8**	**63**	**0.941**
	18	52	15.90%	34.32%	17.4	11	1.000

Note: For the purpose of quantifying the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5% of that source term's total cumulative cesium release to the environment

Table 4-29 Selection of source terms for Mark II sensitivity calculations

	Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Percentile
	1	11DF1000	0.00004%	0.0005%	20.3	20	0
LOW	**2**	**5DF1000**	**0.00055%**	**0.005%**	**32.2**	**20**	**0.125**
	3	42DF100	0.0043%	0.037%	14.3	13	0.25
	4	11	0.042%	0.45%	20.3	20	0.375
MEDIUM	**5**	**51DF10**	**0.23%**	**2.01%**	**16.6**	**9**	**0.5**
	6	5	0.55%	4.94%	32.2	20	0.625
	7	3	1.09%	10.26%	14.3	20	0.75
HIGH	**8**	**1**	**2.46%**	**19.81%**	**22.8**	**25**	**0.875**
	9	52	3.57%	28.67%	16.6	10	1

Note: For the purpose of quantifying the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5% of that source term's total cumulative cesium release to the environment

4.4.1.3 Evacuation Delay

Each accident progression and source term scenario modeled in this rulemaking technical basis begins with an ELAP. The ELAP is postulated to come from both internal and external events. For certain external events that could cause the ELAP, e.g., earthquake or flood, they could also slow the EPZ evacuation effort. For example, an earthquake could damage roads or bridges and cut power to traffic signals. A flood could obstruct access to certain road sections and also cut power to traffic signals. These would result in a slower evacuation because people would need to travel farther to avoid obstructed areas and traffic signals might take longer to pass through. Therefore a sensitivity calculation was included that delays the start of evacuation for all population cohorts by 1 hour to assess the impact of this delay on the offsite consequence metrics. The uniform delay of 1 hour was modeled by changing the MACCS OALARM parameter from 0 to 1 hour.

4.4.1.4 Nonevacuating Cohort

The EPZ population around the reference Mark I site, Peach Bottom, and the reference Mark II site, Limerick, were studied and segmented into population cohorts with distinct evacuation delay and travel characteristics. For each evacuation model, there is a cohort that is modeled to not evacuate, for any number of reasons including refusing to respond to EAS messaging and sirens or not receiving emergency alert communication. SOARCA and other studies have assumed this group to be 0.5% of the EPZ population. This percentage is consistent with research on large-scale evacuations that has shown a small percentage of the public refuses to evacuate [44]. The MACCS model for the reference Mark II site uses 0.5% for the nonevacuating cohort. The MACCS model for the reference Mark I site, however, uses 2% for the nonevacuating cohort to represent the expected emergency response of the Amish community within the Peach Bottom EPZ. According to the recently submitted ETE for Peach Bottom, Amish men and boys aged 15 and older would not evacuate and would stay on their land [22].

The nonevacuating cohort size is commonly perceived to influence consequences related to early phase exposures and is therefore included as a sensitivity case. For each set of Mark I and Mark II sensitivity calculations, a sensitivity run is included that increases the nonevacuating cohort to 5% of the EPZ population. The 5% value was selected based on discussion with senior NRC emergency preparedness staff. To accommodate the increase in population of the nonevacuating cohort, the “general” cohort for each site was reduced by the same percentage. Sensitivity calculations that use a 5% nonevacuating cohort do not also use the 1-hour evacuation delay described in the previous section because the causes of these hypothetical situations are considered independent.

4.4.1.5 Intermediate Phase Duration

The intermediate phase begins after the reactor and releases have been brought under control and accounts for the time needed to plan the long-term restoration and cleanup activities before they can begin. These activities include the following:

1. defining the areas of interest,
2. characterizing the contamination using dose data and field surveys,
3. identifying the types of materials to be decontaminated and the equipment and personnel needed,
4. developing a waste management plan including estimating waste volumes, storage requirements, storage locations, and acquiring storage materials,

5. acquiring decontamination equipment and bringing it onsite, and
6. training personnel and bringing them onsite.

The length of time to conduct these processes would depend on a variety of factors, including the extent and location of the contamination, cleanup criteria, material types, and the state and local decision processes. As the distance from the plant increases, the level of contamination is expected to decrease; however, other influences may increase because a wider variety of materials may be involved and additional towns, counties, and stakeholders would become involved.

Based on the number and types of planning processes that are needed prior to starting decontamination, 3 months was selected as an average intermediate phase duration and was used for all baseline calculations. Some decontamination would be expected to start prior to 3 months; however, much could take longer to begin. To assess the impact of the intermediate phase duration, sensitivity calculations are included using a low value (zero) and a high value (1 year). The dose criterion applied to the intermediate phase uses the long-term phase criterion scaled to the applicable time duration.

4.4.1.6 Long-Term Phase Habitability Criterion

The long-term phase habitability criterion refers to a projected dose accumulated over a specific time and is used to determine whether people can reside on land and whether people can use farmland for agricultural production (assuming the crops meet the farmability criterion). The U.S. EPA provides guidance in its PAG Manual [18] for a habitability criterion of 2 rem in the first year followed by 500 mrem each year thereafter. Pennsylvania, however, uses a stricter habitability criterion of 500 mrem-per-year, starting in the first year. Because both Mark I and Mark II reference sites are located in Pennsylvania, the 500 mrem-per-year long-term phase habitability criterion was used in all baseline calculations. Most BWR Mark I/II sites are not in Pennsylvania and would likely implement the EPA's recommended habitability criterion. To assess the impact of this parameter on the consequence metrics, the EPA's recommended value was used in sensitivity calculations. MACCS requires one dose and one time period, so to accommodate this, 2 rem-per-year was used for each year. The specific intermediate phase and long-term phase habitability parameters are provided in Table 4-30 below. The second and third columns show the sensitivity parameter choices. The next four columns show the implementation in MACCS, which in some cases is slightly different. For variations 4 and 6 in which 2 rem is applied in year 1 of the long-term phase, some or all of this 2 rem is applied to the intermediate phase. In variation 6, the intermediate phase is a full year, so all of the 2 rem is applied to the intermediate phase and then the long-term phase uses 500 mrem each year. In variation 4, the intermediate phase is 3 months so 1/4th of the 2 rem is applied to the intermediate phase (500 mrem) and the remaining 1.5 rem is applied to the long-term phase. Because this 1.5 rem would apply to the first 9 months of the long-term phase year 1, an additional 0.125 rem is included (1/4th of the second year's 500 mrem) totaling 1.625 rem.

Table 4-30 Intermediate and long-term phase sensitivity calculation parameters

			Implementation in MACCS			
			Intermediate Phase		Long-Term Phase	
Variation	Intermediate Phase Duration (yrs)	Long-Term Phase Habitability Criterion	Duration and Dose Projection Period (yrs)	Dose Criterion	Dose Projection Period (yrs)	Dose Criterion
1	0.25	500 mrem	0.25	125 mrem	1	500 mrem
2	0	500 mrem	0	0	1	500 mrem
3	1	500 mrem	1	500 mrem	1	500 mrem
4	0.25	2 rem	0.25	500 mrem	1	1.625 rem
5	0	2 rem	0	0	1	2 rem
6	1	2 rem	1	2 rem	1	500 mrem

4.4.2 Sensitivity Calculations Supporting Regulatory Analysis: Results and Discussion

The results of the 144 sensitivity calculations are provided in Table I-1 through Table I-6 in Appendix I. Each table contains the results of 24 calculations corresponding to a base MACCS model (either the reference site for Mark I, Peach Bottom, or the reference site for Mark II, Limerick) and a low, medium, or high population site file. Each table contains 8 calculations for each of the three source terms used—low, medium, and high. The 8 calculations for each base model, site file, and source term are as follows:

Run 1) Baseline parameters

Run 2) Baseline parameters but with 1 hour evacuation delay

Run 3) Baseline parameters but with no intermediate phase

Run 4) Baseline parameters but with 1 year intermediate phase

Run 5) Baseline parameters but with EPA PAG recommended long-term phase habitability criterion

Run 6) Baseline parameters but with no intermediate phase and EPA PAG recommended long-term phase habitability criterion

Run 7) Baseline parameters but with 1 year intermediate phase and EPA PAG recommended long-term phase habitability criterion

Run 8) Baseline parameters but with 5% nonevacuating cohort fraction

4.4.2.1 Impact of Evacuation Delay

Comparing the Run 2 sensitivity to the Run 1 baseline sensitivity for each of the 18 groupings (2 reference models × 3 site files × 3 source terms) shows zero change to all of the offsite consequences as a result of the 1-hour evacuation delay. All of the source terms used in the sensitivity calculations have an environmental release that is delayed sufficiently to allow time for evacuation of the EPZ. Table 4-31 shows the ratio of each consequence for the evacuation delay case to the baseline case for each of the 18 groupings.

Table 4-31 Ratio of consequences for evacuation delay sensitivity cases to baseline cases

Base Model	Site File	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	High - Peach Bottom	Mark I - Low (Bin 3)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Medium - Susquehanna	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	High - Limerick	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

* Indicates that both the numerator and denominator in the ratio are zero

4.4.2.2 Impact of Larger Nonevacuating Cohort

Comparing the Run 8 sensitivity to the Run 1 baseline sensitivity for each of the 18 groupings (2 reference models × 3 site files × 3 source terms) shows either small or zero change to the offsite consequences as a result of the larger nonevacuating cohort. Table 4-32 shows the ratio of each consequence for the nonevacuating cohort sensitivity case to the baseline case for each of the 18 groupings. This table uses colored formatting to help visualize trends. Higher ratios appear in green while lower ratios appear in red.

Individual early fatality risk is essentially zero for all cases analyzed because even the largest source terms combined with meteorological conditions are not able to cause sufficiently large acute doses. The individual LCF risk within 10 miles shows the largest difference among the consequence metrics. For the Mark II sensitivities, in which the nonevacuating cohort size increased from 0.5%to 5%, the 10-mile area LCF risk increased by 3-4% for the small and medium source terms and by 10-20% for the high source term. For the Mark I sensitivities, in which the nonevacuating cohort is increased from 2-5%, the increase in individual LCF risk is smaller, about 1-2%for all three source terms. The increase is smaller for the Mark I sensitivities because the increase in fraction of the nonevacuating cohort is smaller. Overall, the impact of a larger nonevacuating cohort is fairly small because early phase relocation is still projected and modeled, which would ultimately help this population avoid or minimize exposures.

The 50-mile LCF risk and the 50-mile total population dose show an increase of between 0 and 1% and the increase drops when larger areas (0-100 miles) are considered. The land area exceeding the long-term habitability criterion and the population subject to long-term protective actions show no change because these are not affected by the early phase. The total offsite cost decreases slightly because when fewer people evacuate, fewer people would hypothetically be paid the per diem cost. This would decrease the evacuation cost of the accident. The evacuation cost is a large contributor to the total offsite cost for the low source term, so the low source terms

show a decrease in total offsite cost by about 1-3%. For the medium and large source terms, the evacuation cost is a very small component of the total offsite cost, so these show no impact.

Table 4-32 Ratio of consequences for larger nonevacuating cohort sensitivity cases to baseline cases

Base Model	Site File	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.02	1.00	1.00	1.00	1.00	0.98	0.98	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		1.02	1.00	1.00	1.00	1.00	0.98	0.98	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	High - Peach Bottom	Mark I - Low (Bin 3)		1.02	1.00	1.00	1.00	1.00	0.98	0.98	1.00	1.00	1.00	1.00
		Mark I - Med (Bin 10)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark I - High (Bin 17)		1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		1.04	1.00	1.00	1.00	1.00	0.99	0.99	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.03	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.13	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Medium - Susquehanna	Mark II - Low (Bin 2)		1.04	1.01	1.00	1.00	1.00	0.97	0.97	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.03	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.17	1.01	1.00	1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	High - Limerick	Mark II - Low (Bin 2)		1.04	1.01	1.01	1.01	1.00	0.96	0.96	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.04	1.00	1.00	1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00
		Mark II - High (Bin 8)		1.20	1.01	1.01	1.01	1.00	1.00	1.00	1.00	1.00	1.00	1.00

* Indicates that both the numerator and denominator in the ratio are zero

4.4.2.3 Impact of Intermediate Phase Duration

The impacts of changing the intermediate phase duration to zero and to one year can be seen by comparing the Run 3 and 4 sensitivities to the Run 1 baseline sensitivity for each of the 18 groupings. Table 4-33 and Table 4-34 show the ratio of each consequence for the one-year and zero intermediate phase duration sensitivities, respectively, compared to the baseline cases using a 0.25 year intermediate phase duration. These tables use colored formatting; larger ratios appear in green while smaller ratios appear in red.

For individual LCF risk, using zero intermediate phase increases the results for the small source term (~7-14%) and decreases the results for the medium and large source terms (~4-17%). Using a 1 year intermediate phase generally shows the opposite trend: individual LCF risk decreases for the small source term (~10%) and generally increases for the large source terms (2-18%). For the medium source term the results are within +/- 10%.

Using no intermediate phase generally increases the total population dose, whereas using a 1 year intermediate phase generally decreases the total population dose. The total offsite cost varies between about +60% for the high source term for a 1 year intermediate phase to about -40% for the medium and large source term for the cases with no intermediate phase.

The use of a longer intermediate phase (1 year) allows more time for radionuclides to decay and be removed by natural weathering and therefore less land (~4-40% reduction) is contaminated when the long-term phase begins. Therefore fewer people are subject to long-term phase protective actions (~60-90% reduction) assuming the source term is sufficiently large (medium and high source terms). The low source term is not high enough to cause offsite contamination so land contamination is unaffected. The opposite is true for the use of a shorter (zero) intermediate

phase. The long-term phase starts sooner and because there is less time for radionuclide decay and weathering, more land is contaminated and more people are displaced. The population subject to long-term protective actions increases by up to 550% but is more commonly in the range of 200-300%. The area of land exceeding the long-term phase habitability criterion increases in the range of 2-80%.

The total offsite cost generally increases with the longer intermediate phase for the large source term and decreases with zero intermediate phase. Even though a longer intermediate phase allows more time for decay and weathering, the cost of intermediate phase relocation also increases (50-60% within 50 miles) because the per diem cost is applied for each relocatee for 4 times longer. The small source term is not large enough to necessitate nearly as much, if any, intermediate phase relocation, so there is no effect on offsite cost. The medium source term sensitivity case with the longer intermediate phase results in an increased cost of about 0-40%. The total offsite cost generally decreases by about 20-% for the sensitivities with no intermediate phase for the medium and large source terms and is less affected for the small source term (0-6%).

Table 4-33 Ratio of consequences for 1-year intermediate phase duration sensitivity cases to baseline cases

Base Model	Site	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	0.88	0.89	0.88	0.98	0.99	0.98	0.98	1.00	1.00	0.00	0.00
		Mark I - Med (Bin 10)		1.07	0.93	0.91	0.97	0.97	1.38	1.18	0.86	0.92	0.48	0.48
		Mark I - High (Bin 17)		1.04	0.98	0.93	0.98	0.96	1.61	1.39	0.80	0.87	0.60	0.53
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		0.88	0.88	0.88	0.94	0.95	0.96	0.96	1.00	1.00	0.14	0.14
		Mark I - Med (Bin 10)		1.06	0.92	0.89	0.93	0.92	1.39	1.04	0.73	0.86	0.57	0.57
		Mark I - High (Bin 17)		1.02	0.97	0.91	0.97	0.92	1.58	1.33	0.71	0.82	0.59	0.46
	High - Peach Bottom	Mark I - Low (Bin 3)		0.88	0.89	0.88	0.95	0.95	0.97	0.97	1.00	1.00	0.16	0.16
		Mark I - Med (Bin 10)		1.07	0.92	0.90	0.93	0.92	1.31	1.16	0.91	0.94	0.39	0.39
		Mark I - High (Bin 17)		1.04	0.97	0.93	0.97	0.94	1.60	1.46	0.86	0.89	0.55	0.51
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.99	0.99	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		0.96	0.92	0.92	0.98	0.98	1.00	1.00	0.99	1.00	0.29	0.29
		Mark II - High (Bin 8)		1.18	0.98	0.98	0.98	0.98	1.50	1.49	0.86	0.90	0.20	0.19
	Medium - Susquehanna	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.96	0.96	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		0.98	0.93	0.90	0.95	0.93	1.18	1.11	0.94	0.97	0.44	0.44
		Mark II - High (Bin 8)		1.18	0.98	0.98	0.97	0.97	1.63	1.49	0.62	0.81	0.26	0.21
	High - Limerick	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.93	0.94	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.00	0.92	0.91	0.94	0.93	1.08	1.06	0.96	0.97	0.45	0.45
		Mark II - High (Bin 8)		1.17	0.97	0.98	0.95	0.96	1.57	1.48	0.68	0.81	0.21	0.20

* Indicates that both the numerator and denominator in the ratio are zero

Table 4-34 Ratio of consequences for zero intermediate phase sensitivity cases to baseline cases

Base Model	Site	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.14	1.09	1.09	1.01	1.01	0.97	0.97	1.00	1.00	11.27	11.27
		Mark I - Med (Bin 10)		0.87	0.99	1.05	1.00	1.02	0.63	0.60	1.44	1.27	2.45	2.56
		Mark I - High (Bin 17)		0.94	0.92	0.99	0.98	1.00	0.66	0.62	1.37	1.29	1.72	2.13
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		1.12	1.09	1.09	1.05	1.04	0.94	0.95	1.03	1.02	16.29	16.29
		Mark I - Med (Bin 10)		0.88	1.01	1.08	1.02	1.06	0.61	0.48	1.77	1.42	2.58	2.92
		Mark I - High (Bin 17)		0.96	0.93	1.00	0.97	1.01	0.65	0.58	1.52	1.40	1.87	2.63
	High - Peach Bottom	Mark I - Low (Bin 3)		1.14	1.09	1.09	1.04	1.04	0.97	0.97	1.00	1.00	20.08	20.08
		Mark I - Med (Bin 10)		0.86	1.00	1.05	1.01	1.04	0.62	0.56	1.29	1.21	3.12	3.34
		Mark I - High (Bin 17)		0.94	0.92	0.98	0.96	0.99	0.65	0.62	1.27	1.27	1.85	2.17
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		1.07	1.05	1.05	1.01	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		0.87	1.03	1.04	1.01	1.01	0.70	0.73	1.02	1.02	5.51	5.51
		Mark II - High (Bin 8)		0.83	0.91	0.94	1.04	1.03	0.83	0.83	1.29	1.45	2.22	2.69
	Medium - Susquehanna	Mark II - Low (Bin 2)		1.07	1.05	1.05	1.03	1.03	1.00	1.00	1.02	1.02	*	*
		Mark II - Med (Bin 5)		0.84	1.02	1.06	1.02	1.04	0.58	0.58	1.24	1.12	3.32	3.32
		Mark II - High (Bin 8)		0.85	0.93	1.05	1.09	1.11	0.77	0.70	1.78	1.73	2.62	6.04
	High - Limerick	Mark II - Low (Bin 2)		1.07	1.05	1.05	1.05	1.05	1.00	1.00	1.00	1.00	**	**
		Mark II - Med (Bin 5)		0.84	1.04	1.05	1.04	1.04	0.55	0.55	1.18	1.10	3.33	3.33
		Mark II - High (Bin 8)		0.87	0.90	0.96	1.10	1.11	0.83	0.78	1.65	1.67	2.57	3.09

* Indicates that both the numerator and denominator in the ratio are zero

** Indicates that the denominator in the ratio is zero but the numerator is nonzero (~0.16 persons subject to long-term protective actions for Mark II – High Site File – Low Source Term

4.4.2.4 Impact of Long-Term Phase Habitability Criterion

Comparing the Run 5 sensitivity to the Run 1 baseline sensitivity for each of the 18 groupings shows a much larger impact on the results than the early phase model sensitivities. Table 4-35 shows the ratio of each consequence metric for the 2 rem-per-year habitability criterion sensitivity case to the baseline case using 500 mrem-per-year. This table uses colored formatting to facilitate observation of trends. Larger ratios appear in green, while smaller ratios appear in red.

The 2 rem-per-year habitability criterion allows up to 4 times greater annual dose in the first year than the Pennsylvania criterion of 500 mrem-per-year and therefore the population is exposed to more radiation. Because a higher exposure level is tolerated, less land is considered contaminated, fewer people need to be displaced from land, and there is a lower cost reflecting less decontamination needed, less property loss-of-use costs, and less compensation costs.

The two health effect consequence metrics, individual LCF risk and total population dose, increase when using the 2 rem-per-year habitability criterion. Individual LCF risk within 10 miles increases by 6-7% for the small source term but increases by about 50-250% for the medium and large source terms. The largest change is for the Mark I reference model with the Peach Bottom site file and the large source term: individual LCF risk increases from 7.1E-04 to 1.74E-03, an increase of about 246%. The increases become smaller when larger areas are considered. The total population dose for the 50-mile area shows a negligible increase for the small source term but increases by about 4-48% for the medium and large source terms. The increase to the 50-mile population dose is clearly larger than the increase to the 10-mile individual LCF risk but is very similar to the increase in individual LCF risk for the 50-mile area.

Offsite cost and population and land exceeding the long-term phase habitability criterion decrease when using the 2-rem-per-year habitability criterion. There is either no change or a very small

change for the cases using the small source term. For the 50-mile area and for the medium and large source terms, the offsite cost decreases by about 30-50%, the land contaminated decreases by about 10-50%, and the population subject to long-term phase protective actions decreases by about 70-95%.

Table 4-35 Ratio of consequences for 2 rem-per-year long-term phase habitability criterion sensitivity cases to baseline cases

Base Model	Site	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.06	1.01	1.01	1.00	1.00	0.97	0.97	1.00	1.00	0.00	0.00
		Mark I - Med (Bin 10)		2.29	1.32	1.19	1.15	1.06	0.38	0.40	0.79	0.88	0.21	0.21
		Mark I - High (Bin 17)		2.41	1.63	1.35	1.33	1.16	0.42	0.37	0.67	0.80	0.32	0.27
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		1.07	1.02	1.01	1.01	1.00	0.93	0.93	1.00	1.00	0.01	0.01
		Mark I - Med (Bin 10)		2.38	1.29	1.12	1.24	1.09	0.31	0.25	0.59	0.78	0.32	0.32
		Mark I - High (Bin 17)		2.42	1.56	1.25	1.46	1.21	0.34	0.24	0.51	0.71	0.31	0.21
	High - Peach Bottom	Mark I - Low (Bin 3)		1.06	1.00	1.00	1.00	1.00	0.97	0.97	1.00	1.00	0.03	0.03
		Mark I - Med (Bin 10)		2.29	1.24	1.16	1.19	1.12	0.27	0.25	0.87	0.91	0.14	0.14
		Mark I - High (Bin 17)		2.46	1.57	1.36	1.48	1.30	0.33	0.28	0.77	0.83	0.26	0.23
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.49	1.14	1.12	1.04	1.02	0.52	0.57	0.99	0.99	0.05	0.05
		Mark II - High (Bin 8)		2.24	1.35	1.32	1.18	1.13	0.53	0.53	0.89	0.92	0.32	0.30
	Medium - Susquehanna	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.68	1.20	1.09	1.14	1.06	0.34	0.37	0.91	0.96	0.11	0.11
		Mark II - High (Bin 8)		2.35	1.35	1.24	1.28	1.19	0.46	0.37	0.66	0.82	0.32	0.26
	High - Limerick	Mark II - Low (Bin 2)		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.74	1.17	1.14	1.14	1.11	0.29	0.30	0.93	0.96	0.16	0.16
		Mark II - High (Bin 8)		2.32	1.34	1.30	1.29	1.25	0.46	0.41	0.72	0.83	0.30	0.29

* Indicates that both the numerator and denominator in the ratio are zero

4.4.2.5 Impact of Population Density and Site File Data

For each source term and containment base model, the high, medium, and low site file cases can be examined to assess the impact of population and economic values on the consequence results. Table 4-36 provides ratios of each consequence for the medium and high site files compared to the low site file for each of the 18 groupings.

Individual LCF risk is relatively insensitive to site file data. Comparing the medium and high population cases to the low population case, individual LCF risk within 10 miles increases by up to 60% but also decreases by up to about 10%. When considering larger areas, individual LCF risk generally decreases as population increases.

Population dose is directly related to population size, so the sensitivity cases show a strong increase in population dose for larger population site files. For example, for the Mark II high source term, the high site file case has a population dose about 11 times higher than the low site file case. There are some cases in which a higher population site file does not lead to a higher population dose. For the Mark II low source term cases, the medium site file (Susquehanna) has a lower 50-mile population dose than the low site file (Columbia). This is likely due to the population distribution across the spatial grid in relation to the wind rose. Susquehanna has a 50-mile population almost 4 times larger than Columbia, yet Columbia may happen to have more people distributed in spatial grid elements that intersect with the prevailing wind directions characterized in the weather file.

For a given source term, the total offsite cost always increases with higher population site files. Higher population site files also have more valuable farmland and populated land, so both population and economic values drive the costs up. Comparing the medium to low site file baseline sensitivity calculations, the medium site file calculations have about 2-7 times higher costs. Comparing the high to low site file baseline sensitivity calculations, the high site file calculations have about 4-20 times higher costs.

The population subject to long-term protective actions increases with higher population site files as expected. As expected, the area of land exceeding the long-term phase habitability criterion does not show any clear trends with the site file population and economic values. Land areas depend primarily on land fraction (land versus water), which is another type of information contained in the site file.

Table 4-36　Results for baseline cases with different site files

Base Model	Source Term	Site File	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Mark I - Low (Bin 3)	Med (VT Yankee) / Low (Hatch)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.52	0.98	0.90	0.92	1.19	2.79	2.75	0.39	0.43	6.20	6.20
		High (Peach Bottom) / Low (Hatch)		0.94	0.74	0.96	2.82	2.07	4.65	4.57	1.53	1.45	2.07	2.07
	Mark I - Med (Bin 10)	Med (VT Yankee) / Low (Hatch)		1.25	0.98	0.97	1.88	2.37	3.08	3.60	0.67	0.72	2.91	2.92
		High (Peach Bottom) / Low (Hatch)		1.02	0.83	1.02	5.83	4.00	8.84	8.22	1.28	1.08	7.15	7.15
	Mark I - High (Bin 17)	Med (VT Yankee) / Low (Hatch)		1.23	1.05	1.08	2.26	3.33	3.58	4.95	0.82	0.82	3.11	4.16
		High (Peach Bottom) / Low (Hatch)		1.00	0.89	1.00	6.78	5.04	11.11	9.33	1.11	0.98	9.96	9.59
Mark II - Limerick	Mark II - Low (Bin 2)	Med (Susquehanna) / Low (Columbia)		1.20	0.93	0.49	0.70	1.00	4.90	4.90	3.93	3.93	*	*
		High (Limerick) / Low (Columbia)		1.63	1.10	0.69	2.33	2.25	20.48	20.48	12.79	12.79	*	*
	Mark II - Med (Bin 5)	Med (Susquehanna) / Low (Columbia)		0.94	0.86	0.49	1.38	1.96	2.32	2.33	0.40	0.56	6.35	6.35
		High (Limerick) / Low (Columbia)		1.17	1.03	0.65	6.53	4.82	11.71	10.63	0.52	0.61	28.96	28.96
	Mark II - High (Bin 8)	Med (Susquehanna) / Low (Columbia)		0.89	0.85	0.59	2.06	3.71	3.07	6.60	0.61	0.76	3.00	3.42
		High (Limerick) / Low (Columbia)		1.07	1.04	0.68	10.82	9.32	18.49	17.97	0.69	0.75	17.87	17.09

* Indicates that both the numerator and denominator in the ratio are zero

4.4.3 Environmental Release Timing Sensitivity

A separate sensitivity calculation was performed to examine the effect of the start time of environmental release on the consequence metrics. The representative source term case for each source term bin was selected by choosing the one with a cesium and iodine release closest to the average among the source terms in that bin. The selection of the representative source term case did not consider the time at which release to the environment begins because all of them provide sufficient time for evacuation of the EPZ population prior to the environmental release. To confirm this, a sensitivity calculation was run to compare the offsite consequence results for the source term case with the fastest environmental release of all the cases to the representative source term case for that source term's bin. The fastest environmental release among all considered was 7.3 hours for case 50 and is compared to case 48, the representative case for bin 6, which has a release to the environment beginning at 11.4 hours. The cesium release is very similar for each of these but they differ in many other ways beyond just the environmental release start time. The iodine release is larger in case 48 (1.69% vs. 1.09%). The release profile is also somewhat different: case 50 releases 75% of the source term's cesium in the first hour whereas case 48's hour-long plume segment with the largest cesium release is only about 43% of the source term's total cesium.

Table 4-37 shows the offsite consequence metrics for Mark I case 48 and case 50. Consistent with all other calculations, individual early fatality risk is essentially zero for both cases. For individual LCF risk, population dose, offsite cost, and land exceeding the long-term phase habitability criterion, the consequences are about 10-30% higher for case 48. However, the

population subject to long-term protective actions is 33% larger for case 50. The consequences are generally but not always higher for case 48 because of its slightly larger source term magnitude and slightly longer release duration.

Table 4-37 Comparison of source term environmental release timing for Mark I Case 48 and Mark I Case 50

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
6	48	0.23%	1.69%	11.4	8	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
Test	50	0.21%	1.09%	7.3	6	0	7.11E-05	1.42E-05	6.75E-06	204,000	354,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
6	48	0.23%	1.69%	11.4	8	1,150,000,000	1,390,000,000	116	175	3,440	3,440
Test	50	0.21%	1.09%	7.3	6	1,000,000,000	1,190,000,000	91	135	4,590	4,590

4.4.4 Evacuation Sensitivity Calculations for Fastest Release (Mark I Case 49)

As discussed above, an evacuation delay of 1 hour applied uniformly to all evacuation cohorts has zero effect on the results because the releases begin late enough that the surrounding population is expected to still have time to evacuate. Additional calculations were performed to assess the sensitivity of extended delays in evacuation assumptions on the individual early fatality and ILCF risk measures. The calculations used the accident progression and source term data from Mark I MELCOR case 49, which had the fastest environmental release of all scenarios analyzed. This scenario assumes a short-term station blackout with no batteries available and no RCIC at a BWR with a Mark I containment. In this case, the release to the environment starts at 7.3 hours and releases 0.5% of the cesium and 1.7% of the iodine. The MACCS calculations were run for four variations of the scenario, each with and without an external filter. The cases with an external filter assumed a decontamination factor of 10 for all particle sizes. The four variations are as follows:

- *Base Case:* This uses the expected evacuation timing for Peach Bottom based on the ETE report submitted to NRC in 2014. Based on the ETE data and emergency declaration and notification assumptions, the EPZ would be cleared of evacuees in about 5.5 hours.
- *3 Hour Delay:* The evacuation timeline is delayed by 3 hours and is applied uniformly to all evacuation cohorts.
- *6 Hour Delay:* The evacuation timeline is delayed by 6 hours and is applied uniformly to all evacuation cohorts.
- *No Evacuation*: A hypothetical situation in which the EPZ population does not evacuate and instead shelters in place. The MACCS evacuation model was turned off and shielding parameters were adjusted to simulate sheltering.

These sensitivity calculations are designed to simulate intentionally unrealistic emergency response situations. Emergency response programs are developed, tested, and evaluated by the NRC as an element of defense-in-depth. Detailed plans for onsite and offsite responses are approved by NRC and FEMA, and it is expected that the plans will be implemented as written.

In addition, for all of the MACCS calculations, the protective actions for the intermediate phase and long-term phase were kept in full effect, so the model changes only affect the emergency phase which was modeled to last for one week.

The summary results for the conditional ILCF risk for the 0-10 mile area are shown in

Figure 4-20, "CRF for Conditional ILCF Risk (0-10 mi) for Evacuation Sensitivity Calculations for BWR Mark I MELCOR Case 49." For the base case in Figure 4-20, a scenario with an external filter results in a consequence reduction factor (CRF) of about 3 for the ILCF risk, conditional on the accident occurring. For the 3 hour evacuation delay in Figure 4-20, there is no change to the ILCF risk total and no change to the CRF for the external filter. In this case the release would start before the EPZ is evacuated; however, by the time the plume travels out to the more populated areas, the population would effectively have left the area.

For the 6 hour evacuation delay in Figure 4-20, the total ILCF risk approximately doubles relative to the base case without an external filter. The contribution to the ILCF risk from the emergency phase increases significantly, by about a factor of 25; however, the emergency phase contribution was originally just 3%. Given that the contribution from the intermediate and long-term phase remains constant, the overall increase is fairly minor. For the case with an external filter, the increase in ILCF risk from the emergency phase is much smaller. The overall CRF increases to 4.8 relative to the base case as shown in Figure 4-20.

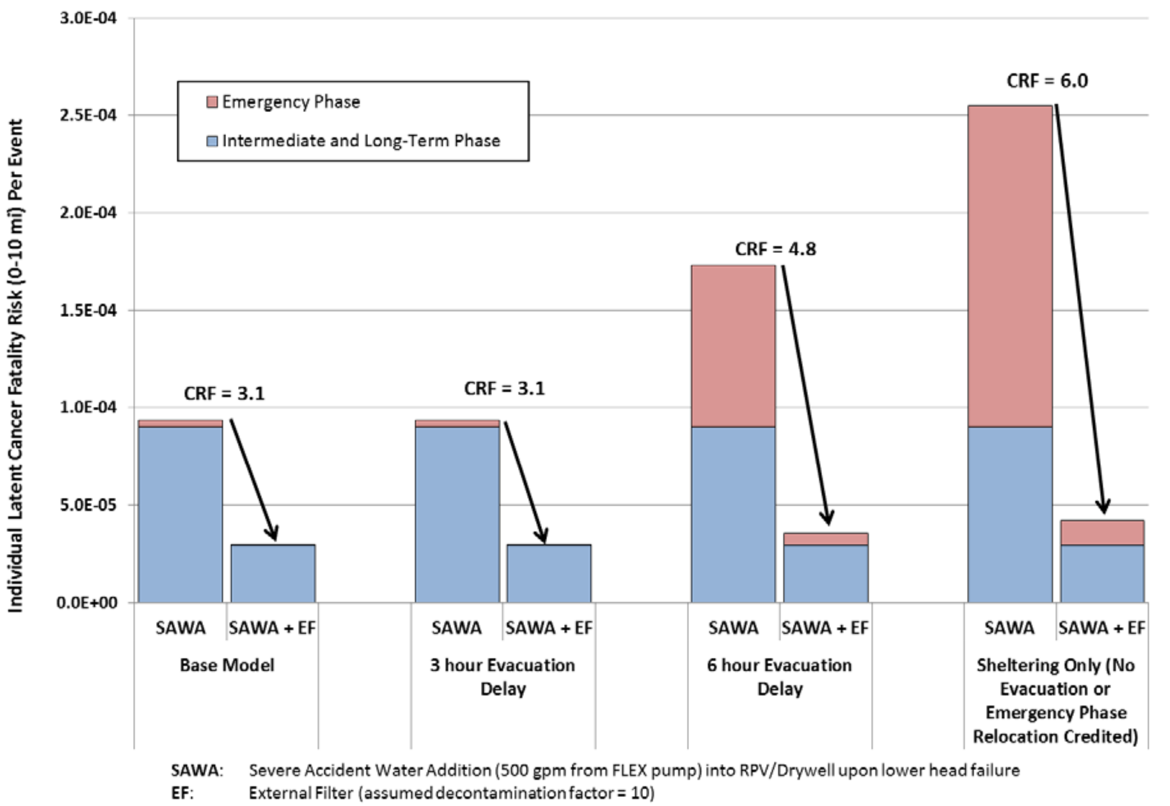


Figure 4-20 CRF for conditional ILCF risk (0-10 mi) for evacuation sensitivity calculations for BWR Mark I MELCOR Case 49

For the extremely unlikely and hypothetical situation in which the EPZ population does not evacuate and instead shelters in place, Figure 4-20 shows the total ILCF risk increases for each case (with and without the external filter) because of the increase to the emergency phase. Without an external filter, the emergency phase contribution to the ILCF risk is about 50 times larger than for the base case; however, the total ILCF risk is only about 2.5 times larger than the base case. Overall, the CRF increases to 6.0.

The following are conclusions from the additional evacuation sensitivity calculations for BWR Mark I MELCOR case 49:

- Changes to the evacuation model show an increase in the potential benefit of an external filter; however, the effect is relatively small (up to about a factor of two for a CRF=6.0 compared to CRF=3.1).
- Assuming protective actions are taken in the intermediate and long-term phases, the ILCF risk is maintained at a level well below the QHO, when multiplied by the accident frequency.
- For all evacuation sensitivities for BWR Mark I MELCOR case 49, there is essentially zero individual early fatality risk.

4.5 Conditional Offsite Consequences of Different CPRR Alternatives

The NRC has evaluated the conditional offsite consequences associated with the three major accident response strategies: (1) no post-core damage external water addition, (2) successful post-core damage external water addition, and (3) external water addition with an external filter. For the Mark I analyses, water addition cases (SAWA and SAWM) are grouped together for consequence analysis purposes because there is little difference between them. For the Mark II analysis, the water management strategy does not apply because of the containment geometry. All cases representing an external filter assume a uniform decontamination factor of 10.

For each alternative, a subset of MELCOR cases was selected which equally represent the alternative. The consequences of the MACCS source term bin corresponding to each of the applicable MELCOR cases are averaged to yield an overall set of offsite consequence results. This process is shown in Table 4-38, “Average Mark I Conditional Offsite Consequences for the Different MELCOR Cases Associated with the CPRR Alternatives,” and Table 4-39, “Average Mark II Conditional Offsite Consequences for the Different MELCOR Cases Associated with the CPRR Alternatives,” for the Mark I and Mark II analysis, respectively. The average offsite consequences for each of the CPRR alternatives are then compared in Table 4-40, “Conditional Mark I and Mark II Offsite Consequences and Consequence Reduction Factor (CRF) for each CPRR Alternative.” Table 4-40 also shows the consequence reduction factor (CRF) associated with each alternative relative to the case with no external water addition.

Table 4-38 Average Mark I conditional offsite consequences for the different MELCOR cases associated with the CPRR alternatives

	MELCOR Case	MACCS Bin	Individual Latent Cancer Fatality Risk (0-10 mi)	Population Dose (rem) (0-50 mi)	Offsite Cost ($ 2013) (0-50 mi)	Land Contamination (sq. miles) (0-50 mi)	Population Subject to Long-Term Protective Actions (0-50 mi)
No External Water Addition	1	12	2.91E-04	1,720,000	13,000,000,000	549	64,500
	2	15	2.59E-04	1,740,000	15,700,000,000	573	111,000
	4	10	4.06E-04	1,360,000	9,900,000,000	479	51,400
	5	10	4.06E-04	1,360,000	9,900,000,000	479	51,400
	6	12	2.91E-04	1,720,000	13,000,000,000	549	64,500
	Average:		***3.30E-04***	***1,600,000***	***12,000,000,000***	***530***	***69,000***
Successful External Water Addition	8	11	1.35E-04	1,110,000	5,960,000,000	286	40,500
	9	7	1.21E-04	524,000	2,740,000,000	190	15,000
	10	7	1.21E-04	524,000	2,740,000,000	190	15,000
	11	7	1.21E-04	524,000	2,740,000,000	190	15,000
	12	11	1.35E-04	1,110,000	5,960,000,000	286	40,500
	13	7	1.21E-04	524,000	2,740,000,000	190	15,000
	14	7	1.21E-04	524,000	2,740,000,000	190	15,000
	15	7	1.21E-04	524,000	2,740,000,000	190	15,000
	16	7	1.21E-04	524,000	2,740,000,000	190	15,000
	21	11	1.35E-04	1,110,000	5,960,000,000	286	40,500
	22	12	2.91E-04	1,720,000	13,000,000,000	549	64,500
	23	11	1.35E-04	1,110,000	5,960,000,000	286	40,500
	25	7	1.21E-04	524,000	2,740,000,000	190	15,000
	26	7	1.21E-04	524,000	2,740,000,000	190	15,000
	28	7	1.21E-04	524,000	2,740,000,000	190	15,000
	29	6	7.95E-05	253,000	1,150,000,000	116	3,440
	30	7	1.21E-04	524,000	2,740,000,000	190	15,000
	Average:		***1.30E-04***	***720,000***	***4,000,000,000***	***230***	***23,000***
Successful External Water Addition & External Filter (DF=10)	8DF10	6	7.95E-05	253,000	1,150,000,000	116	3,440
	9DF10	5	2.03E-05	71,200	220,000,000	41	118
	10DF10	5	2.03E-05	71,200	220,000,000	41	118
	11DF10	5	2.03E-05	71,200	220,000,000	41	118
	12DF10	6	7.95E-05	253,000	1,150,000,000	116	3,440
	13DF10	5	2.03E-05	71,200	220,000,000	41	118
	14DF10	5	2.03E-05	71,200	220,000,000	41	118
	15DF10	5	2.03E-05	71,200	220,000,000	41	118
	16DF10	5	2.03E-05	71,200	220,000,000	41	118
	21DF10	6	7.95E-05	253,000	1,150,000,000	116	3,440
	22DF10	6	7.95E-05	253,000	1,150,000,000	116	3,440
	23DF10	6	7.95E-05	253,000	1,150,000,000	116	3,440
	25DF10	5	2.03E-05	71,200	220,000,000	41	118
	26DF10	5	2.03E-05	71,200	220,000,000	41	118
	28DF10	5	2.03E-05	71,200	220,000,000	41	118
	29DF10	4	1.72E-05	48,400	141,000,000	23	7
	30DF10	5	2.03E-05	71,200	220,000,000	41	118
	Average:		***3.80E-05***	***120,000***	***490,000,000***	***62***	***1,100***

Table 4-39 Average Mark II conditional offsite consequences for the different MELCOR cases associated with the CPRR alternatives

	MELCOR Case	MACCS Bin	Individual Latent Cancer Fatality Risk (0-10 mi)	Population Dose (rem) (0-50 mi)	Offsite Cost ($ 2013) (0-50 mi)	Land Contamination (sq. miles) (0-50 mi)	Population Subject to Long-Term Protective Actions (0-50 mi)
No External Water Addition	1	8	4.70E-04	6,110,000	85,500,000,000	854	721,000
	5	6	2.29E-04	2,160,000	24,000,000,000	303	62,400
	6	7	3.08E-04	4,140,000	80,800,000,000	698	619,000
	Average:		***3.40E-04***	***4,100,000***	***63,000,000,000***	***620***	***470,000***
Successful External Water Addition	10	5	1.35E-04	689,000	4,250,000,000	130	15,400
	11	4	7.90E-05	202,000	844,000,000	44	1,030
	24	6	2.29E-04	2,160,000	24,000,000,000	303	62,400
	Average:		***1.50E-04***	***1,000,000***	***9,700,000,000***	***160***	***26,000***
Successful External Water Addition & External Filter (DF=10)	10DF10	4	7.90E-05	202,000	844,000,000	44	1,030
	11DF10	3	6.58E-06	20,700	393,000,000	2	0
	24DF10	4	7.90E-05	202,000	844,000,000	44	1,030
	Average:		***5.50E-05***	***140,000***	***690,000,000***	***30***	***690***

Table 4-40 Conditional Mark I and Mark II offsite consequences and consequence reduction factor (CRF) for each CPRR alternative

	Offsite Consequences (per event) / CPRR Cases	Individual Latent Cancer Fatality Risk (0-10 mi)		Population Dose (0-50 mi)		Offsite Cost (0-50 mi)		Land Contamination (0-50 mi)		Population Subject to Long-Term Protective Actions (0-50 mi)	
		-	CRF	Person-rem	CRF	$ 2013	CRF	Square Miles	CRF	People	CRF
Mark I	No External Water Addition & No External Filter	3.3E-04	1.0	1,600,000	1.0	12,000,000,000	1.0	530	1.0	69,000	1.0
	Successful External Water Addition	1.3E-04	2.5	720,000	2.2	4,000,000,000	3.0	230	2.3	23,000	3.0
	Successful External Water Addition & External Filter (DF=10)	3.8E-05	8.7	120,000	13.3	490,000,000	24.5	62	8.5	1,100	62.7
Mark II	No External Water Addition & No External Filter	3.4E-04	1.0	4,100,000	1.0	63,000,000,000	1.0	620	1.0	470,000	1.0
	Successful External Water Addition	1.5E-04	2.3	1,000,000	4.1	9,700,000,000	6.5	160	3.9	26,000	18.1
	Successful External Water Addition & External Filter (DF=10)	5.5E-05	6.2	140,000	29.3	690,000,000	91.3	30	20.7	690	681.2

Figure 4-21 and Figure 4-22 show graphical depictions of the offsite consequences of the CPRR alternatives as a percentage of the status quo.

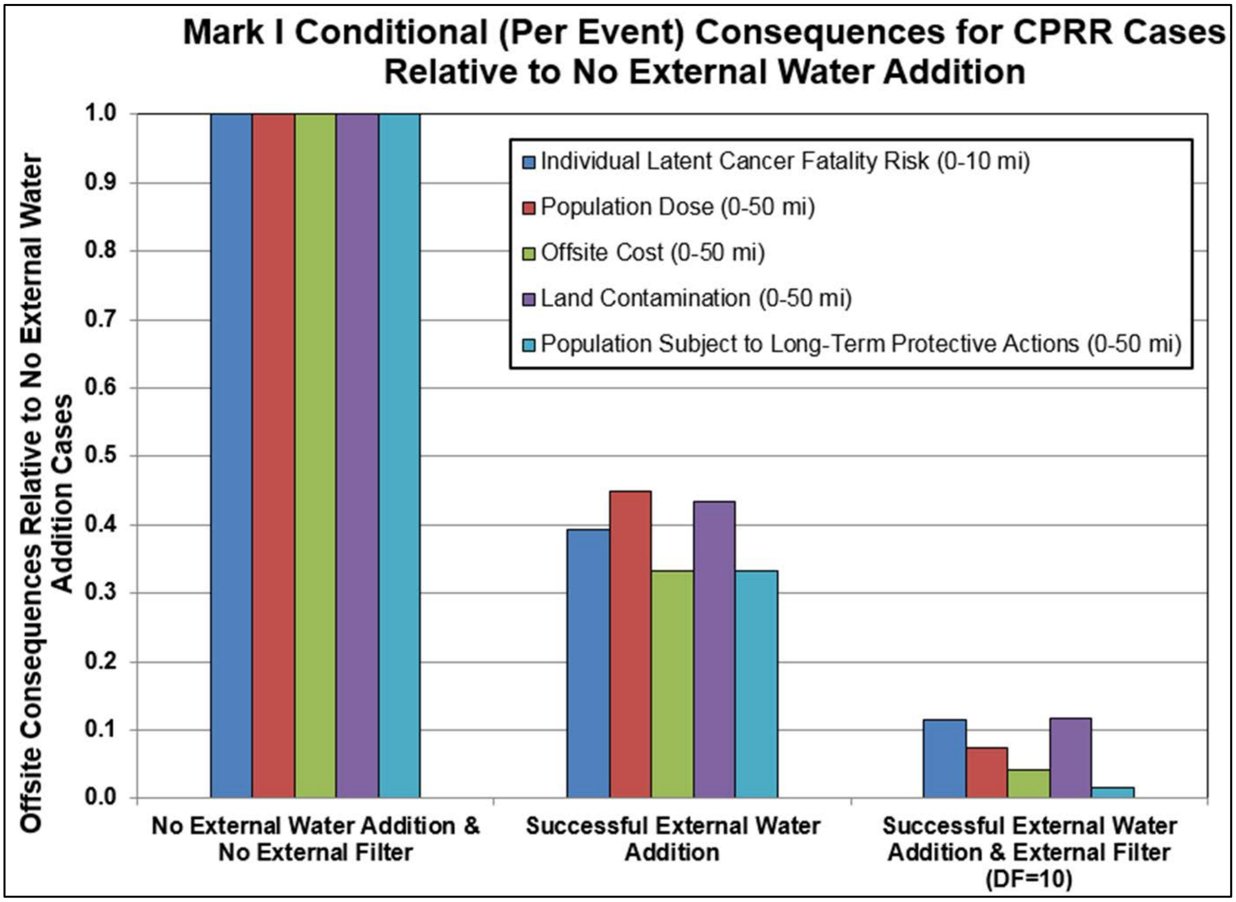


Figure 4-21 Conditional Mark I offsite consequences for each CPRR alternative as a percentage of the case with no external water addition

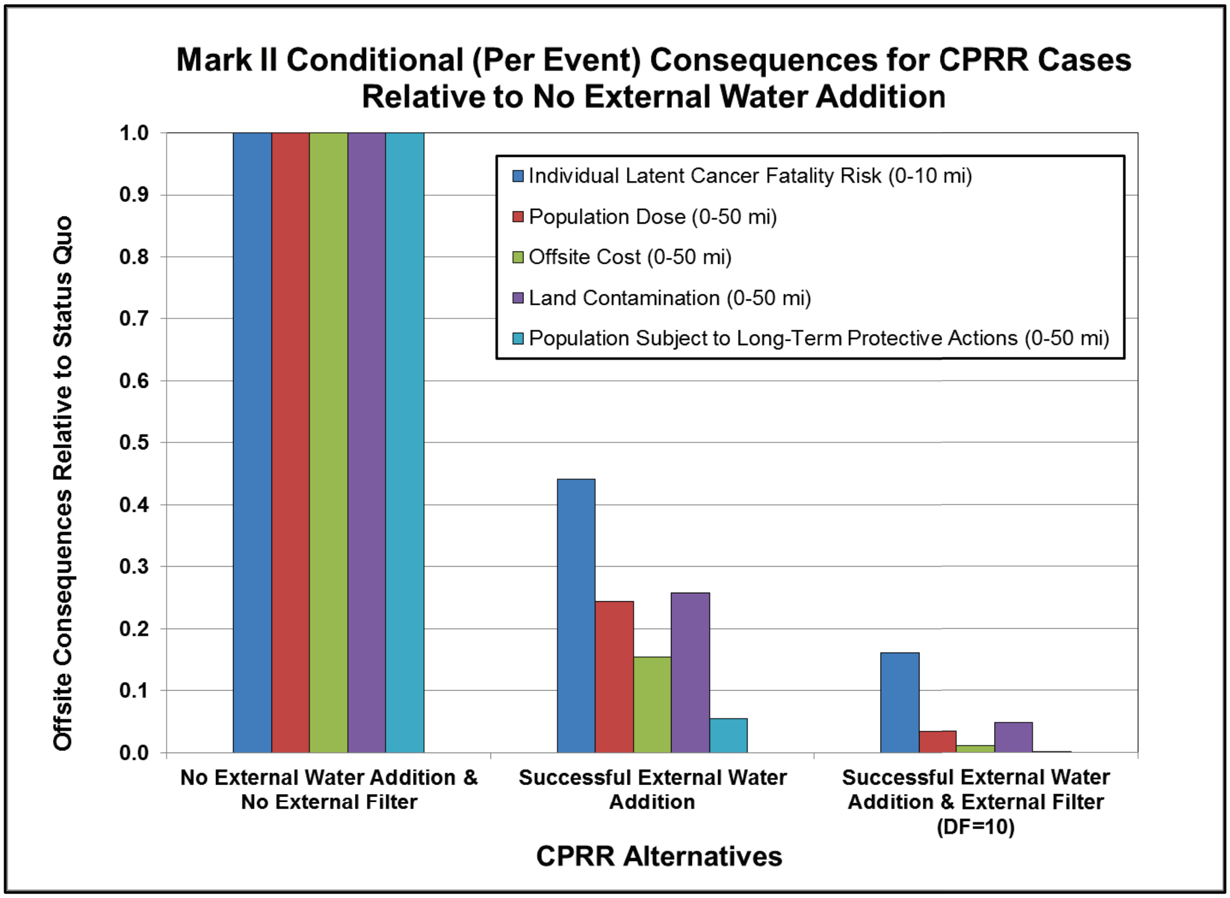


Figure 4-22 Conditional Mark II offsite consequences for each CPRR alternative as a percentage of the case with no external water addition

The following general conclusions can be drawn from the tables and figures above regarding the conditional offsite consequences (per event) for the CPRR alternatives:

- For the Mark I and Mark II analysis, relative to cases without external water addition, successful water injection (SAWA/SAWM) results in a notable reduction in offsite consequences for all five measures (ILCF risk, population dose, offsite cost, land contamination, and population subject to long-term phase protective actions).
- For the Mark I analysis, relative to cases without external water addition, the combination of successful water injection with an external filter results in a greater reduction of offsite consequences.
- For both the Mark I and Mark II analysis, the offsite consequence metric that shows the largest reduction factor is the population subject to long-term protective actions. Compared to cases without external water addition, successful water addition combined with an external filter provides a consequence reduction factor of approximately 63 for Mark I and approximately 680 for Mark II.
- The CPRR alternatives generally show greater reduction in offsite consequences relative to cases without external water addition for the Mark II analysis compared to the Mark I analysis. This is likely due to the higher population around the Mark II site (Limerick).

- The conditional ILCF risk for the cases without external water addition, 3×10^{-4} per event for both Mark I and Mark II analyses, is sufficiently low that when multiplied by the accident frequency, there is substantial margin to the NRC Safety Goal (QHOs) for the ILCF risk.

4.6 Offsite Consequence Summary and Conclusions

In support of the CPRR rulemaking, offsite consequence calculations were performed using the MACCS code. The modeling approach was to develop site-specific MACCS models for a reference Mark I site and a reference Mark II site based on the most current information available and the latest modeling best practices. Site-specific features of the model include weather, population, land use, economic values, emergency response characteristics and timing, and the long-term phase habitability criterion. Source terms, generated by MELCOR, were each considered in four variations to evaluate the potential range of external filter performance: with no external filter (DF=1) and with an external filter (DF=10, DF=100, and DF=1000). After applying the DFs, the source terms were binned according to cesium and iodine release magnitudes. Individual source term cases were then selected to represent the 18 Mark I source term bins and 9 Mark II source term bins.

The potential effectiveness of an external filter on reducing the environmental release is heavily influenced by the fraction of the source term that flows through the wetwell vent or drywell vent, where the external filter is attached. For accident cases in which the entire release flows through a vent pathway, the external filter can reduce the environmental release, and there is significant incremental benefit to increasing the filter DF. In contrast, for accident cases in which most of the release does not flow through the vent, perhaps due to drywell liner melt-through or a main steam line creep rupture, there is little benefit that can be achieved from an external filter. There are also many cases in between these, in which much, but not all of the release flows through a vent pathway. In these cases, an external filter with DF=10 can reduce the environmental release, However, there is little or no incremental benefit of having an external filter that can achieve a higher DF (100 or 1000). Note that these analyses do not consider the reliability of the external filter; each case assumes that it always performs at its specified DF for the release going through the filter.

Results calculated and presented in this section include individual early fatality risk, individual LCF risk, total population dose, total offsite cost, land exceeding the long-term phase habitability criterion, and population subject to long-term phase protective actions. All results are mean values over approximately 1,000 weather trials and are presented as conditional (per event) on the accident scenario occurring (no frequency consideration).

For all Mark I and Mark II source terms, there is essentially zero individual early fatality risk, because even the largest source terms do not cause acute doses that exceed the dose threshold for acute radiation syndrome fatalities. All of the source terms begin their release to the environment long enough after the accident's initiation to reasonably allow time for the EPZ population to evacuate. Thus the contribution to individual LCF risk and total population dose from the early phase is very low, especially for the 10-mile area. All other calculated consequences beside early fatality risk generally increase with source term magnitude and with source term duration. Larger source terms create more offsite contamination, leading to higher consequence results. Source terms that are released over a longer period of time allow more time for the wind to shift direction, thereby increasing the probability that contamination is spread over a larger area. The larger consequences of a longer, more gradual release compared with a shorter, more punctuated release are more pronounced at shorter distances and less apparent at longer

distances. This trend results from the fact that the exposures from narrower, more concentrated plume patterns provide significant exposures at longer distances, whereas less concentrated plumes that are spread over multiple compass sectors become depleted at shorter distances.

Sensitivity studies were conducted to assess the impact of many of the site-specific modeling features on the calculated results. These modeling features include site population and economic values, an evacuation delay, a larger nonevacuating population cohort, the intermediate phase duration, and the long-term habitability criterion. The sensitivity calculations lead to the following general insights:

- An evacuation delay of 1 hour applied uniformly to all cohorts has no effect on any of the results.
- Baseline calculations use 0.5% or 2% for the nonevacuating cohort fraction. Sensitivities using a larger nonevacuating cohort (5%) show a small (≤ 20%) increase to the individual LCF risk for the 10-mile area.
- Baseline calculations use the Pennsylvania long-term phase habitability criterion of 500 mrem-per-year. Sensitivities using the less strict 2 rem-per-year habitability criterion based on the EPA PAG Manual [18] generally show:
    - higher individual LCF risk (~50-250% higher within 10 miles for the medium and large source terms)
    - higher population dose (up to ~50% higher within 50 miles for the medium and large source terms)
    - lower offsite cost (up to ~75% lower within 50 miles for the medium and large source terms)
    - lower land contamination (up to ~50% lower within 50 miles for the medium and large source terms)
    - lower population subject to long-term protective actions (up to ~95% lower within 50 miles for the medium and large source terms)
- Baseline calculations used 3 months as the duration of the intermediate phase. Compared to baseline calculations, sensitivities using no intermediate phase generally show:
    - higher individual LCF risk for the small source term (up to ~15% higher within 10 miles) and lower individual LCF risk for the medium and large source terms (decrease of ~4-17% within 10 miles)
    - higher population dose (up to ~10% higher within 50 miles)
    - lower offsite cost (up to ~45% lower within 50 miles for the medium and large source terms)
    - higher land contamination (up to ~80% higher within 50 miles for the medium and large source terms)
    - higher population subject to long-term protective actions (up to ~550% higher within 50 miles for the medium and large source terms)
- Compared to baseline calculations, sensitivities using a 1-year intermediate phase generally show:
    - lower individual LCF risk for the small source term (~10% lower within 10 miles) and higher individual LCF risk for the medium and large source terms (increase of up to ~18% within 10 miles)

- lower population dose (up to ~7% lower within 50 miles)
  - higher offsite cost (up to ~60% higher within 50 miles for the large source term)
  - less land contamination (up to ~40% lower within 50 miles for the medium and large source terms)
  - lower population subject to long-term protective actions (up to ~80% lower within 50 miles for the medium and large source terms)

- Population sensitivity calculations were performed by using site file data for low, medium, and high population Mark I and Mark II sites. Site files contain population, land use, and economic values. Population dose, offsite cost, and population subject to long-term protective actions generally increase with population. Since the Mark I and Mark II reference sites were the high population sites (Peach Bottom and Limerick), these three result types were generally lower for the low and medium sites compared to those reported in Section 4 of this report. Individual LCF risk is a population-weighted metric so it does not scale as clearly with population; rather, it depends on population distributions compared with wind rose probabilities. The area of land exceeding the long-term phase habitability criterion also does not show any clear relationship with population and economic data; rather, it depends on land fraction (land versus water) for the area around the site.

- MACCS sensitivity calculations, which generally showed the greatest increase in individual latent cancer fatality risk were those with medium and large releases in which the EPA PAG habitability criterion (2 rem in the first year), were used instead of the Pennsylvania habitability criterion (500 mrem in the first year), thus allowing a first year dose up to four times higher. The increase in individual latent cancer fatality risk was commonly a factor of 2-3 higher than for the base cases.

Overall, sensitivity calculations do not change any of the consequence analysis insights related to the QHO metrics. Individual early fatality risk remains essentially zero for all sensitivity calculations conducted, and ILCF risk remains well below the QHO, even those assuming a larger habitability criterion (e.g., 2 rem per year instead of 500 mrem per year).

4.7 References for Chapter 4

1. Electric Power Research Institute, EPRI Report 1025287, “Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” November 2012, ADAMS Accession No. ML12333A170.
2. U.S. Nuclear Regulatory Commission, NUREG/CR-4691, Vol. 2, “MELCOR Accident Consequence Code System (MACCS): Model Description,” Washington, DC, February 1990.
3. U.S. Nuclear Regulatory Commission, NUREG/CR-6613, “Code Manual for MACCS2: Volume 1, User’s Guide,” Washington, DC, May 1998.
4. U.S. Nuclear Regulatory Commission, Draft NUREG/CR, “WinMACCS, a MACCS2 Interface for Calculating Health and Economic Consequences from Accidental Release of Radioactive Materials into the Atmosphere: User’s Guide and Reference Manual for WinMACCS Version 3,” Washington, DC, December 2009.

5. U.S. Nuclear Regulatory Commission, NUREG/CR-6525, Rev. 1, SAND2003-1648P, "SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program," Washington, DC, August 2003.
6. McFadden, Katherine et al, "MELMACCS Models Document," Sigma Software L.L.C., Draft, January 2011.
7. U.S. Nuclear Regulatory Commission, NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Washington, DC, September 2004.
8. U.S. Nuclear Regulatory Commission, NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Washington, DC, January 1997.
9. U.S. Nuclear Regulatory Commission, NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," Washington, DC, November 2012.
10. U.S. Nuclear Regulatory Commission, NUREG/CR-7110, Vol. 1, Rev. 1, "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis," Washington, DC, May 2013.
11. U.S. Nuclear Regulatory Commission, NUREG/CR-7110, Vol. 2, Rev. 1, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis," Washington, DC, August 2013.
12. U.S. Nuclear Regulatory Commission, NUREG/BR-0359, Rev. 1, "Modeling Potential Reactor Accident Consequences," Washington, DC, December 2012.
13. U.S. Nuclear Regulatory Commission, NUREG/CR-7009, "MACCS Best Practices as Applied in the SOARCA Project," Washington, DC, September 2014.
14. U.S. Nuclear Regulatory Commission, Draft NUREG/CR-7155, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," Washington, DC, June 2013.
15. U.S. Nuclear Regulatory Commission, NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor," Washington, DC, September 2014.
16. U.S. Nuclear Regulatory Commission, SECY-12-0157, Enclosure 5B, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments: MACCS Consequence Analysis," Washington, DC, December 2012.
17. U.S. Nuclear Regulatory Commission, SECY-12-0123, "Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC's Regulatory Framework," September 2012.
18. U.S. Environmental Protection Agency, EPA 402-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," Washington, DC, May 1992.
19. U.S. Environmental Protection Agency, EPA 402-R-99-001, "Federal Guidance Report No. 13, Cancer Risk Coefficients for Environmental Exposure to Radionuclides," Washington, DC, September 1999.
20. U.S. Nuclear Regulatory Commission, NUREG-1350, Vol. 25, "Information Digest, 2013-2014," Washington, DC, August 2013.
21. OECD/NEA/CSNI Status Report on Filtered Containment Venting, NEA/CSNI/R(2014)7, July 2014, https://www.oecd-nea.org/nsd/docs/2014/csni-r2014-7.pdf

22. KLD TR-636, Rev. 0, “Development of Evacuation Time Estimates: Peach Bottom, Units, 1, 2, and 3,” April 2014, http://pbadupws.nrc.gov/docs/ML1414/ML14141A046.html.
23. U.S. Department of Labor, Bureau of Labor Statistics, http://www.bls.gov/cpi
24. U.S. Environmental Protection Agency, EPA-454/R-99-005, “Meteorological Monitoring Guidance for Regulatory Modeling Applications,” Research Triangle Park, NC, February 2000.
25. U.S. Nuclear Regulatory Commission, NUREG-0917, “Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data,” Washington, DC, July 1982.
26. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.23, Rev. 1, “Meteorological Monitoring Programs for Nuclear Power Plants,” Washington, DC, March 2007.
27. U.S. Environmental Protection Agency, Support Center for Regulatory Atmospheric Modeling, “SCRAM Surface Meteorological Archived Data: 1984-1992,” http://www.epa.gov/ttn/scram/surfacemetdata.htm
28. U.S. Environmental Protection Agency, Office of Air Programs, AP-101, “Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States,” Research Triangle Park, NC, January 1972.
29. Pasquill, F., “The Estimation of the Dispersion of Windborn Material,” *The Meteorological Magazine*, 90:33-49, 1961.
30. Tadmor, J., and Y. Gur., “Analytical Expressions for the Vertical and Lateral Dispersion Coefficients in Atmospheric Diffusion,” *Atmospheric Environment*, Vol. 3, pp. 688–689, 1969.
31. Eimutis E.C. and M.G. Konicek, “Derivations of Continuous Functions for the Lateral and Vertical Atmospheric Dispersion Coefficients,” *Atmospheric Environment*, 6:859-63, 1972.
32. Napier, B.A., J.P. Rishel, and N.E. Bixler, PNNL-20990, “Final Review of Safety Assessment Issues at Savannah River Site, August 2011,” Pacific Northwest National Laboratory, December 2011.
33. U.S. Department of Agriculture, National Agricultural Statistics Service, CropScape 2013, http://nassgeodata.gmu.edu/CropScape
34. U.S. Environmental Protection Agency, Multi-Resolution Land Characteristics Consortium, National Land Cover Class Definitions, http://www.epa.gov/mrlc/definitions.html
35. Whelan, G., et al., DOE/RL/87-09, “The Remedial Action Priority System (RAPS): Mathematical Formulations, U.S. Department of Energy, Washington, DC, August 1987.
36. U.S. Nuclear Regulatory Commission, NUREG/CR-7161, “Synthesis of Distributions Representing Important Non-Site-Specific Parameters in Offsite Consequence Analyses,” Washington, DC, April 2013.
37. U.S. Nuclear Regulatory Commission and Commission of European Communities, NUREG/CR-6545, “Probabilistic Accident Consequence Uncertainty Analysis: Early Health Effects Uncertainty Analysis,” Washington, DC, December 1997.
38. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Washington, DC, November 1982.
39. U.S. Nuclear Regulatory Commission, NUREG/CR-2260, “Technical Basis for Regulatory Guide 1.145, ‘Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,’” Washington, DC, October 1981.

40. U.S. Nuclear Regulatory Commission, NUREG-0654/FEMA-REP-1, Rev. 1, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Washington, DC, November 1980.
41. U.S. Nuclear Regulatory Commission, NUREG/CR-7002, SAND2010-0016P, "Criteria for Development of Evacuation Time Estimate Studies," Washington, DC, November 2011.
42. U.S. Nuclear Regulatory Commission, NUREG/CR-6953, Vol. 2, “Review of NUREG-0654, Supplement 3, ‘Criteria for Protective Action Recommendations for Severe Accidents’: Focus Groups and Telephone Survey”, Washington, DC, October 2008.
43. Wolshon, Brian, J. Jones, and F. Walton, “The Evacuation Tail and Its Effect on Evacuation Decision Making," *Journal of Emergency Management*. Vol. 8, No. 1, January/February 2010.
44. U.S. Nuclear Regulatory Commission, NUREG/CR-6864, "Identification and Analysis of Factors Affecting Emergency Evacuations," Washington, DC, January 2005.
45. KLD TR-617, Rev. 0, “Development of Evacuation Time Estimates: Limerick Units 1 and 2,” January 2014, http://pbadupws.nrc.gov/docs/ML1404/ML14042A181.pdf.
46. U.S. Nuclear Regulatory Commission, NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” Washington, DC, December 1990.
47. U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 2, Rev. 1, Part 7, “Evaluation of Severe Accident Risks: Quantification of Major Input Parameters: MACCS Input,” Washington, DC, December 1990.
48. U.S. Environmental Protection Agency, “PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents: Draft for Interim Use and Public Comment,” Washington, DC, March 2013.
49. U.S. Nuclear Regulatory Commission, NUREG/CR-6981, “Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations,” Washington, DC, October 2008.
50. Mileti, Dennis S., John H. Sorensen and Paul W. O’Brien, “Toward an Explanation of Mass Care Shelter Use in Evacuations,” *International Journal of Mass Emergencies and Disasters*. 10 (1): 25-42, 1992.
51. NHK News, “Earthquake Report – JAIF,” No. 95: 18:00, May 28, 2011. http://www.jaif.or.jp/nglish/news_images/pdf/ENGNEWS01_1306574680P.pdf
52. Japan Ministry of the Environment, “Progress on Off-site Cleanup Efforts in Japan,” April 2014.
53. U.S. Department of Labor, “U.S. Bureau of Labor Statistics Occupational Outlook Handbook,” January 2014. http://www.bls.gov/ooh
54. S&P Case-Shiller National Home Price Index, http://www.us.spindices.com/index-family/real-estate/sp-case-shiller-us-national-home-price-index
55. National Council of Real Estate Investment Fiduciaries, U.S. National Property Index, https://www.ncreif.org/property-index-returns.aspx
56. Davis, M.A. and J. Heathcote, “The Price and Quantity of Residential Land in the United States,” *Journal of Monetary Economics*, 54 (2595-2620), June 2007.
57. U.S. Nuclear Regulatory Commission, NUREG/CR-4214, “Health Effects Model for Nuclear Power Plant Accident Consequence Analysis,” Washington, DC, July 1985.

58. Eckerman, K.F., Letter Report, “Radiation Dose and Health Risk Estimation: Technincal Basis for the State-of-the-Art Reactor Consequence Analysis Project,” ML12159A259, Oak Ridge National Laboratory, Oak Ridge, TN, January 2012.

59. National Academy of Sciences, “Health Effects of Exposure to Low Levels of Ionizing Radiation: BEIR V,” National Research Council, National Academy Press, Washington, DC, 1990.

60. International Commission on Radiological Protection, ICRP Publication 60, “1990 Recommendations of the International Commission on Radiological Protection,” 1991.

61. 51 FR 30028, “Safety Goals for the Operation of Nuclear Power Plants,” August 1986.

5 RISK INTEGRATION

5.1 Results

Table 5-1 provides the main results of the risk integration, the risks estimates of each regulatory analysis sub-alternative. These results are graphically depicted in Figure 5-1 through Figure 5-5. The similarity among these figures implies that risk insights developed by comparing regulatory analysis sub-alternatives do not strongly depend on which risk metric is used to make the comparison.

The estimates of individual early fatality risk for all sub-alternatives are essentially zero since the conditional individual early fatality risks developed by the offsite consequence analysis are calculated as zero for all cases analyzed.

The risk estimates (prompt or early fatality risk and latent cancer fatality risk) show a reduction in risk when post-accident water injection is used. The risks of sub-alternatives 1 and 2A are larger than any of the other sub-alternatives since sub-alternatives 1 and 2A cannot prevent containment structural failure caused by liner melt-through. As discussed in Section 2, it became apparent that, as the project was nearing completion and SECY-15-0085 was being developed, all licensees intended to comply with Phase 2 of Order EA-13-109 by implementing the post-accident water injection strategy.

The risk estimates show relatively minor decreases when post-accident water injection is made to the RPV (sub-alternatives 3A, 4Ai(1), 4Aii(1) and 4Aiii(1)) as compared to injection made to the DW (sub-alternatives 3B, 4Ai(2), 4Aii(2) and 4Aiii(2)). Sub alternatives involving post-accident injection to the RPV provide the capability to arrest a severe accident before vessel breach occurs.

Table 5-1 Risk estimates by regulatory analysis sub-alternative

Index	Regulatory Analysis Sub-Alternative	Fraction of Core-Damage Frequency		Individual Early Fatality Risk (/y)	Individual Latent Cancer Fatality Risk (/y)			Population Dose (person-rem/y)		Offsite Cost ($ 2013/y)		Land Exceeding Long-Term Habitability Criterion (square miles/y)		Population Subject to Long-Term Protective Actions (persons/y)	
		Vented	Uncontrolled Release	0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	1	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
2	2A	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
3	3A	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
4	3B	42%	58%	0.0E+00	2.1E-09	6.7E-10	3.4E-10	1.1E+01	1.9E+01	7.4E+04	1.0E+05	3.4E-03	6.4E-03	4.1E-01	4.9E-01
5	4Ai(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
6	4Ai(2)	42%	58%	0.0E+00	2.1E-09	6.1E-10	3.1E-10	9.5E+00	1.7E+01	6.8E+04	9.0E+04	3.2E-03	5.8E-03	3.6E-01	4.1E-01
7	4Aii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
8	4Aii(2)	42%	58%	0.0E+00	2.4E-09	7.7E-10	3.9E-10	1.2E+01	2.2E+01	8.9E+04	1.2E+05	3.9E-03	7.3E-03	4.8E-01	5.8E-01
9	4Aiii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
10	4Aiii(2)	42%	58%	0.0E+00	2.0E-09	5.6E-10	2.7E-10	8.7E+00	1.5E+01	6.2E+04	7.9E+04	3.0E-03	5.1E-03	3.1E-01	3.4E-01
11	4Bi(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
12	4Bi(2)	42%	58%	0.0E+00	1.4E-09	3.3E-10	1.5E-10	4.8E+00	8.2E+00	3.1E+04	3.8E+04	1.8E-03	2.7E-03	1.6E-01	1.6E-01
13	4Bii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.6E+00	7.9E+00	3.0E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
14	4Biii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.7E+00	8.1E+00	3.1E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.6E-01
15	4Biv	40%	60%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.6E+00	7.8E+00	3.0E+04	3.6E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
16	4Ci(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
17	4Ci(2)	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.5E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
18	4Cii	42%	58%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.4E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01
19	4Ciii	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.4E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
20	4Civ	40%	60%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.3E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01

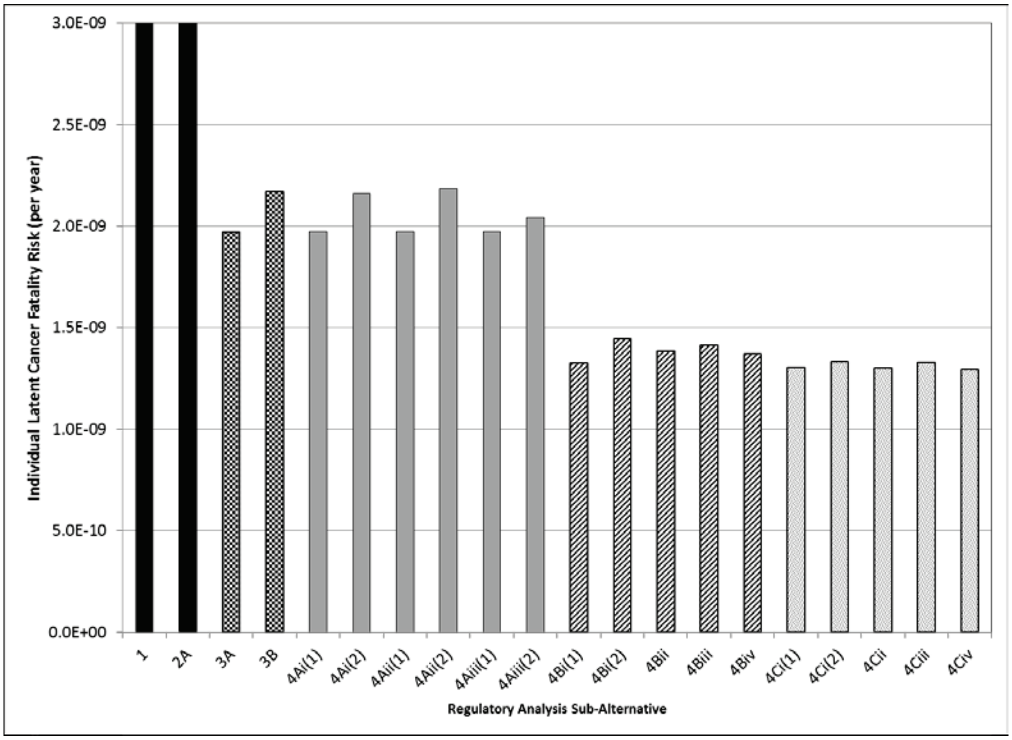


Figure 5-1 Comparison of regulatory analysis sub-alternatives using individual latent cancer fatality risk (0-10 miles)

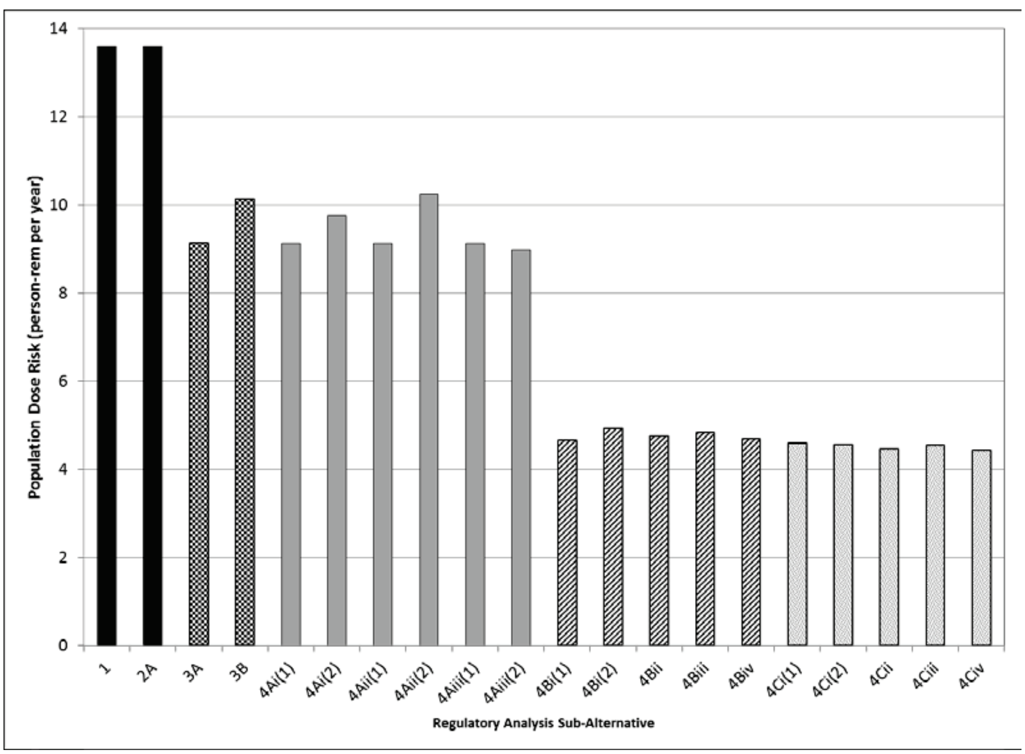


Figure 5-2 Comparison of regulatory analysis sub-alternatives using population dose risk (0-50 miles)

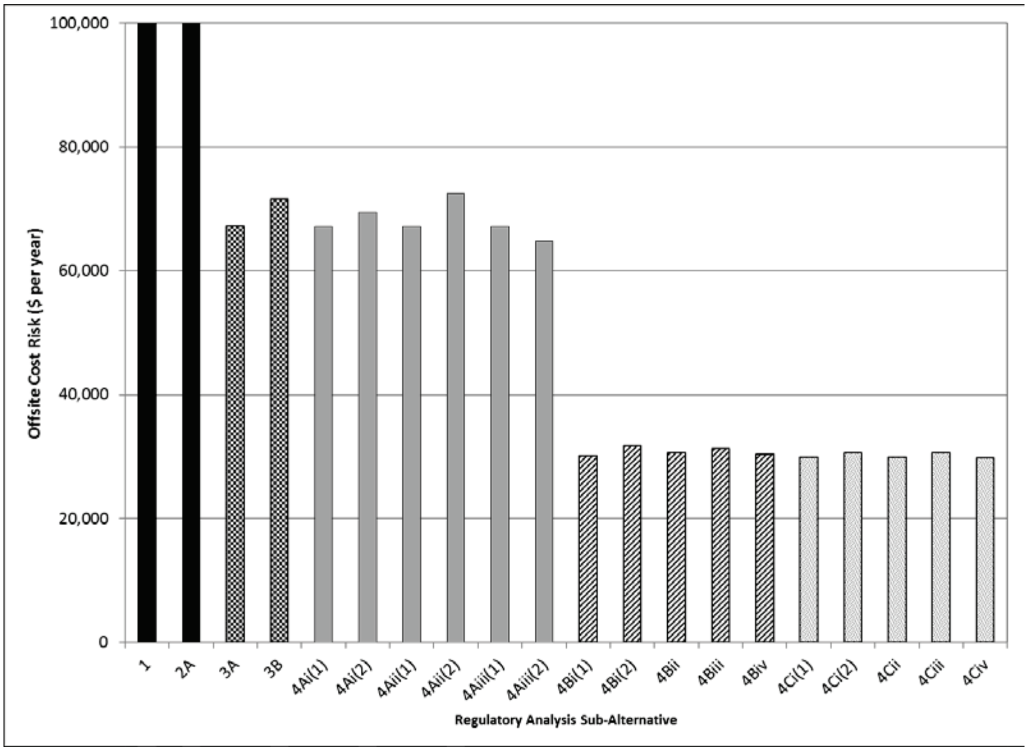


Figure 5-3 **Comparison of regulatory analysis sub-alternatives using offsite cost risk (0–50 miles)**

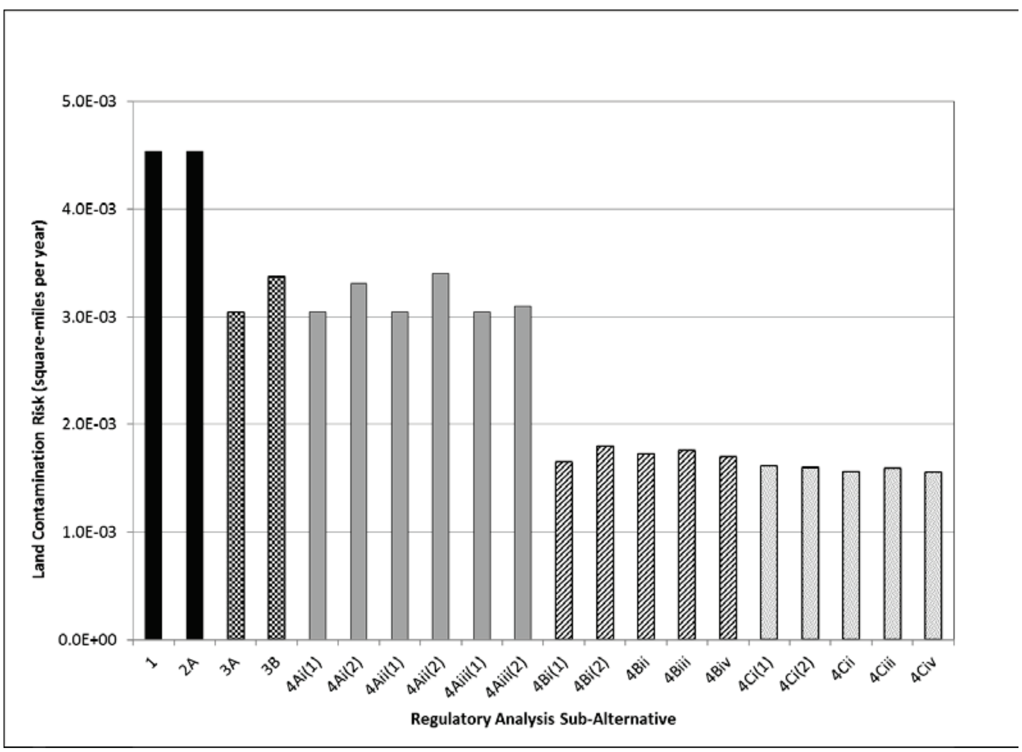


Figure 5-4 **Comparison of regulatory analysis sub-alternatives using area of land exceeding long-term habitability criterion (0-50 miles)**

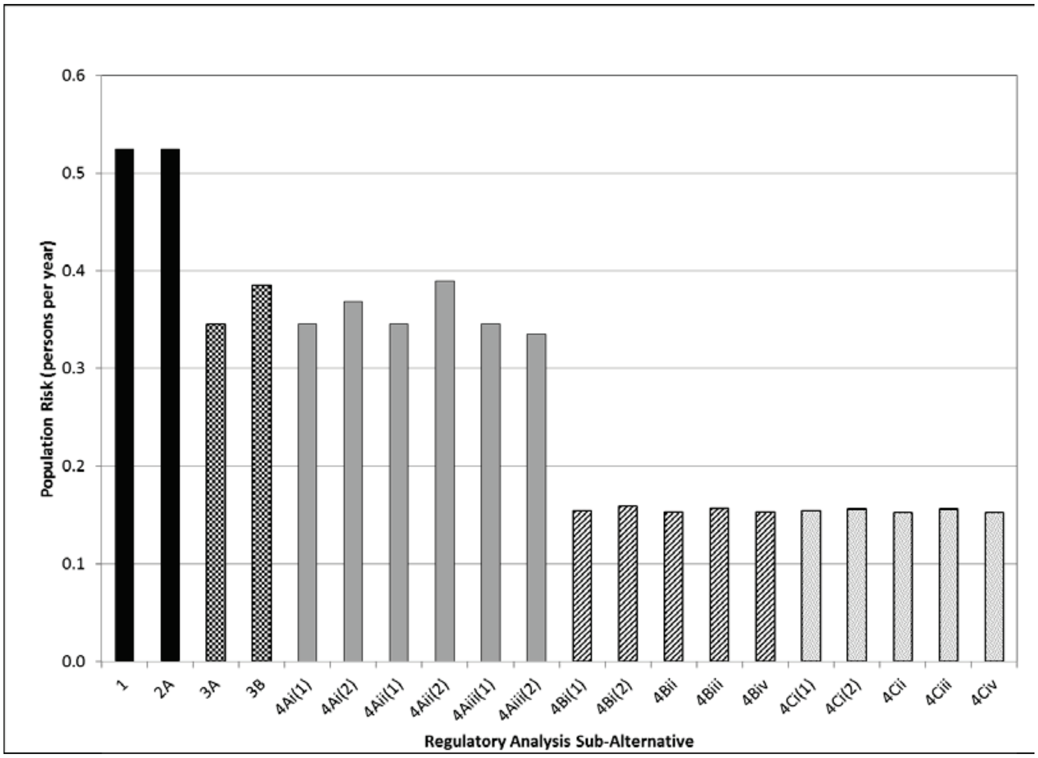


Figure 5-5 Comparison of regulatory analysis sub-alternatives using population subject to long-term habitability criterion (0-50 miles)

The risk estimates show no discernable reduction in risk when filtration strategies (containment venting cycling and severe accident water management) are used. This observation is based on comparison of sub-alternatives 4Ai(1), 4Aii(1) and 4Aiii(1) with sub-alternative 3A, and on comparison of sub-alternatives 4Ai(2), 4Aii(2) and 4Aii(2) with sub-alternative 3B.

The risk estimates show a noticeable reduction in risk when engineered filters are used. Specifically, the risk estimates for sub-alternatives 4Bi(1), 4Bi(2), 4Bii, 4Biii, 4Biv, 4Ci(1), 4Cii, 4Ciii and 4Civ are noticeably smaller as compared to the risk estimates of the other sub-alternatives. However, there is no apparent benefit to using large engineered filters (which have an assumed decontamination factor of 1000) as opposed to using small engineered filters (which have an assumed decontamination factor of 10). Specifically, the risk estimates show no discernable reduction in risk for sub-alternatives 4Ci(1), 4Ci(2), 4Cii, 4Ciii and 4Civ as compared to sub-alternatives 4Bi(1), 4Bi(2), 4Bii, 4Biii and 4Biv.

5.2 Sensitivity and Uncertainty Analysis

In order to develop additional risk insights about the CPRR strategies, two supplemental analyses were performed. First, the sensitivity of core-damage frequency to the scoping human error probabilities was explored. Second, an approximate parametric uncertainty analysis of the individual latent cancer fatality risk estimates was conducted. Each of these supplemental analyses is discussed in the following sections.

5.2.1 Sensitivity of Core Damage Frequency to Human Error Probabilities

To gain further perspective on the importance of operator actions and the adequacy of the scoping HEPs used in the accident sequence analysis, an *en masse* sensitivity analysis was performed. The *en masse* sensitivity analysis was conducted by simultaneously varying the HEPs of all operators contained in the CDETs. The results of this analysis, depicted as "heat maps," are shown in Figure 5-6, Figure 5-7 and Figure 5-8. Note that the lowest core-damage frequency appears in the upper left corner of the heat map. The core-damage frequency increases as the in-control room HEPs increase (towards the bottom of the heat map) and as the ex-control HEPs increase (towards the righthand side of the heat map). The highest core-damage frequency appears in the lower right corner of the heat map.

		Ex-Control_Room HEP													
		0	0.001	0.002	0.003	0.006	0.01	0.02	0.03	0.06	0.1	0.2	0.3	0.6	1
In-Control-Room HEP	0	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.001	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.002	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.003	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.006	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
	0.01	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
	0.02	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.6E-06	1.5E-05	1.9E-05
	0.03	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.9E-06	8.6E-06	1.5E-05	1.9E-05
	0.06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.2E-06	5.4E-06	5.7E-06	7.0E-06	8.7E-06	1.5E-05	1.9E-05
	0.1	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.3E-06	5.3E-06	5.5E-06	5.9E-06	7.2E-06	8.9E-06	1.5E-05	1.9E-05
	0.2	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.6E-06	5.7E-06	5.9E-06	6.3E-06	7.6E-06	9.3E-06	1.5E-05	1.9E-05
	0.3	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.1E-06	6.2E-06	6.4E-06	6.8E-06	8.1E-06	9.8E-06	1.6E-05	1.9E-05
	0.6	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.8E-06	8.9E-06	9.2E-06	1.0E-05	1.1E-05	1.6E-05	1.9E-05
	1	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.4E-05	1.4E-05	1.5E-05	1.7E-05	1.9E-05

Figure 5-6 En masse sensitivity of core-damage frequency to human error probability for Wetwell-First venting strategy

		Ex-Control_Room HEP													
		0	0.001	0.002	0.003	0.006	0.01	0.02	0.03	0.06	0.1	0.2	0.3	0.6	1
In-Control-Room HEP	0	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.3E-06	5.5E-06	6.5E-06	8.0E-06	1.4E-05	1.9E-05
	0.001	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.3E-06	5.5E-06	6.5E-06	8.0E-06	1.4E-05	1.9E-05
	0.002	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.3E-06	5.5E-06	6.5E-06	8.0E-06	1.4E-05	1.9E-05
	0.003	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.5E-06	6.5E-06	8.0E-06	1.4E-05	1.9E-05
	0.006	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.5E-06	6.6E-06	8.0E-06	1.4E-05	1.9E-05
	0.01	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.5E-06	6.6E-06	8.1E-06	1.4E-05	1.9E-05
	0.02	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.6E-06	8.1E-06	1.4E-05	1.9E-05
	0.03	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.6E-06	8.1E-06	1.4E-05	1.9E-05
	0.06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.4E-06	5.7E-06	6.8E-06	8.3E-06	1.4E-05	1.9E-05
	0.1	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.4E-06	5.7E-06	6.9E-06	8.4E-06	1.4E-05	1.9E-05
	0.2	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.3E-06	5.5E-06	5.9E-06	7.1E-06	8.7E-06	1.4E-05	1.9E-05
	0.3	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.3E-06	5.3E-06	5.6E-06	6.0E-06	7.3E-06	8.8E-06	1.4E-05	1.9E-05
	0.6	5.1E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.3E-06	5.4E-06	5.7E-06	6.2E-06	7.6E-06	9.1E-06	1.4E-05	1.9E-05
	1	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.3E-06	5.3E-06	5.4E-06	5.5E-06	5.9E-06	6.4E-06	7.8E-06	9.3E-06	1.5E-05	1.9E-05

Figure 5-7 En Masse sensitivity of core-damage frequency to human error probability for Drywell-First (passive actuation) venting strategy

		Ex-Control_Room HEP													
		0	0.001	0.002	0.003	0.006	0.01	0.02	0.03	0.06	0.1	0.2	0.3	0.6	1
In-Control-Room HEP	0	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.001	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.002	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.003	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.7E-06	8.5E-06	1.5E-05	1.9E-05
	0.006	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
	0.01	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.5E-06	1.5E-05	1.9E-05
	0.02	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.8E-06	8.6E-06	1.5E-05	1.9E-05
	0.03	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.3E-06	5.6E-06	6.9E-06	8.6E-06	1.5E-05	1.9E-05
	0.06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.1E-06	5.2E-06	5.2E-06	5.2E-06	5.4E-06	5.7E-06	7.0E-06	8.7E-06	1.5E-05	1.9E-05
	0.1	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.2E-06	5.3E-06	5.3E-06	5.5E-06	5.9E-06	7.2E-06	8.9E-06	1.5E-05	1.9E-05
	0.2	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.5E-06	5.6E-06	5.7E-06	5.9E-06	6.3E-06	7.6E-06	9.3E-06	1.5E-05	1.9E-05
	0.3	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.0E-06	6.1E-06	6.2E-06	6.4E-06	6.8E-06	8.1E-06	9.8E-06	1.6E-05	1.9E-05
	0.6	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.7E-06	8.8E-06	8.9E-06	9.2E-06	1.0E-05	1.1E-05	1.6E-05	1.9E-05
	1	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.5E-05	1.4E-05	1.4E-05	1.5E-05	1.7E-05	1.9E-05

Figure 5-8 En Masse sensitivity of core-damage frequency to human error probability for Drywell-First (manual actuation) venting strategy

The results of the en masse sensitivity analysis show that CDF ranges from a low of $5.1\text{x}10^{-6}$/ry (assuming no operator errors) to a high of $1.9\text{x}10^{-5}$/ry (giving no credit to the plant operators). Two conclusions may be drawn from these results:

- About 60% of the total CDF is due to accident sequences that only involve equipment failure.
- Assuming the HEP scoping values are appropriate, the CDF approximately doubles when operators completely fail to take beneficial actions (1.9E-5 / 8.9E-6 = 2.1).

5.2.2 Parametric Uncertainty Analysis

Monte Carlo methods were used to conduct an approximate parametric uncertainty analysis of the individual latent cancer fatality risk estimates for each regulatory analysis sub-alternative. Parametric uncertainties were considered for all inputs, as explained in Table 5-2:

Table 5-2 Uncertainty analysis inputs

Events	Distribution	Remarks
Frequency of ELAPs due to internal events	Log-normal Mean = point estimate Error factor =15	An error factor of 15 maximizes the ratio of the 95th percentile to the mean value. This approach does not explicitly consider the uncertainty in the offsite power recovery curves or the uncertainty in the EPS reliability parameters (failure rate and failure-on-demand probability)
Seismic hazard curves	Log-normal	Normal parameters were developed for each point on the seismic hazard curve using the fractile information provided by licensees in their responses to the 10 CFR 50.54(f) information request concerning NNTF Recommendation 2.1
Seismic fragilities	Double log-normal, using the developed values of C_{50}, β_R, and β_U	Traditional approach to modeling uncertainty in seismic fragility.
Hardware-related failures	Log-normal Mean = point estimate Error factor = 15	An error factor of 15 maximizes the ratio of the 95^{th} percentile to the mean value.

Events	Distribution	Remarks
Human failure events	Constrained non-informative prior	A constrained non-informative prior distribution is a beta distribution with mean = point estimate and α = 0.5.
Conditional consequences	Log-normal Mean = point estimate Error factor = 10	Informed by preliminary results of the SOARCA uncertainty analysis project [1].

In essence, the parametric uncertainty analysis consists of recalculating the logic model using a random sample of parameter values. Each CDET was requantified 5,000 times for each plant within the scope of the accident sequence analysis, and the resulting semi-plant-specific random samples of PDS frequencies were averaged to develop 5,000 random samples of the fleet-average PDS frequencies. Each APET was recalculated 500 times for each input PDS to determine 500 random samples of the RC frequencies. The random samples of RC frequencies were combined with 500 random samples of the conditional consequences to develop 500 random samples of the individual latent cancer fatality risk for each regulatory analysis sub-alternative. The results of the parametric uncertainty analysis are illustrated in Figure 5-9. It should be noted that, in contrast to the uncertainty analysis results presented in the draft regulatory basis attached to SECY-15-0085, the regulatory analysis sub-alternatives in Figure 13 match the ordering used in Table 2-1.

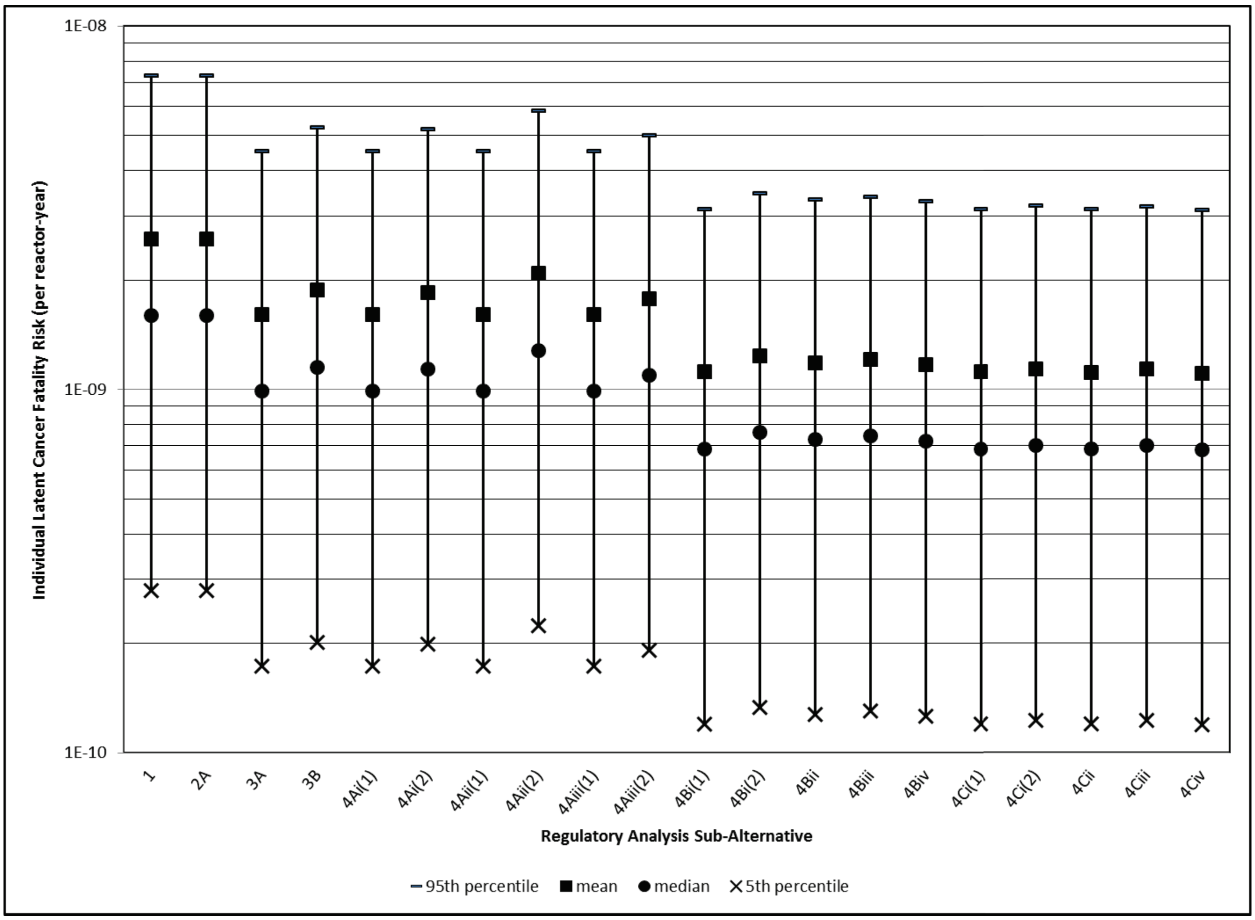


Figure 5-9 Results of the parametric uncertainty analysis

Uncertainty in the individual latent cancer fatality risk estimates ranges about two orders of magnitude. The major source of uncertainty is due to uncertainty in the seismic hazard curves, a result that is consistent with previous seismic PRAs.

As shown in Figure 5-9, the changes in risk among the various CPRR sub-alternatives are much smaller than the estimated parametric uncertainty ranges. This result does not necessarily imply that differences among the CPRR sub-alternatives are insignificant. The parametric uncertainty risk estimates of the various DPRR sub-alternatives are stochastically dependent (correlated) because they share a common database; as the correlation increases, the uncertainty in the changes in risk decreases. It should be noted that the impact evaluation contained in the draft regulatory basis attached to SECY-15-0085 uses the mean difference between sub-alternatives.

As shown in Figure 5-9, the mean risk estimate for each CPRR sub-alternative is approximately 1,000 times lower than the QHO, and the 95th percentiles risk estimate for each CPRR sub-alternative is more than 100 times lower than the QHO. As a result, it may concluded that the risk due to ELAP events is very small, regardless of which specific CPRR regulatory analysis sub-alternative is implemented.

5.3 <u>Conclusions</u>

The risk integration, which was performed to support the draft regulatory basis attached to SECY-15-0085, assessed the risk impacts of twenty potential CPRR strategies for severe accidents initiated by ELAP events due to internal events and seismic events occurring at operating BWRs with Mark I containment designs and reactor core isolation cooling (RCIC) systems. Based on the results of the risk integration, the following conclusions were developed:

1. The risk due to ELAP events is very small, regardless of which specific CPRR regulatory analysis sub alternative is implemented.

2. FLEX strategies reduce the probability of core damage due to ELAP events by more than 50%.

3. The contribution of ELAPs initiated by seismic events is generally greater than the contribution of ELAPs initiated by internal events.

4. The most important contributors to CDF from ELAPs are seismic failures of the station batteries or dc switchgear, seismic failures of the emergency diesel generators (EDGs) and their supporting systems, random failures of the portable FLEX pump (failure to start and failure to run), and random failures of the RCIC pump (failure to start).

5. The capability to vent the containment during a severe accident limits the probability of containment structural failure due to overpressurization to a few percent.

6. There is a reduction in risk when containment liner melt-through is prevented through the use of post-accident water injection. Injection to the RPV somewhat lowers the risk as compared to injection directly to the DW.

7. There is no discernable reduction in risk relative to sub-alternatives 3A and 3B (the baseline for the impact evaluation provided in the draft regulatory evauation attached to SECY-15-0085) when different filtration strategies (containment vent cycling and severe accident water management) are considered.

8. There is a reduction in risk (for many of the accident sequences evaluated) when engineered filters are used. However, there is no apparent benefit to using large type (DF = 1000) engineered filters as opposed to using small type (DF = 10) engineered filters.

9. About 60% of the total CDF is due to accident sequences that only involve equipment failures. About 40% of the total CDF is due to accident sequences that involve combinations of equipment failures and operator actions.

5.4 References for Chapter 5

1. U.S. Nuclear Regulatory Commission, NUREG/CR-7155 (draft), “State-of-the-Art Reactor Consequence Analyses Project, Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” August 2013, ADAMS Accession No. ML13189A145.

6 TECHNICAL EVALUATION SUMMARY AND KEY INSIGHTS

This report documents the results of research conducted to support the agency initiative to address the containment venting issue for the boiling water reactor (BWR) Mark I and Mark II containments. The research focused on developing the technical basis for a potential rulemaking action on containment protection and release reduction (CPRR), and covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences deemed to be the most significant risk contributors; (2) accident analysis of these sequences and assessment of radiological source terms; and (3) analysis of consequences with particular emphasis on health effects, both short term and long term.

The accident sequence analysis evaluated a large number of sequences representing various operator actions such as RPV depressurization, wetwell and drywell venting, and water addition strategies. Also, seismically induced equipment failures were analyzed in detail. Models to estimate the frequency of ELAP events resulting from internal events and earthquakes were based on seismic hazard estimates developed by the industry. The human reliability aspect was considered in the accident sequence formulation, and despite an initial attempt to develop a comprehensive human reliability assessment (HRA), only a bounding approach to incorporating HRA into the accident sequences was implemented at the end. Sensitivity evaluations were done to gain insight into how human error probability affects the accident sequence frequencies.

In the accident sequence analysis for the CPRR rulemaking (SECY-15-0085), the core-damage frequency (CDF) due to ELAPs is calculated to be 8.9×10^{-6} per reactor year, which is two times lower than the value of 1.6×10^{-5} that was estimated for SECY-12-0157. The CDF was calculated by averaging together the CDF for each BWR plant that was included in the scope of the accident sequence analysis (see Table 2-1). Also, the conditional core-damage probability (CCDP) given the occurrence of an ELAP was calculated to be about 47% (i.e., the mitigation strategies required by Order EA-12-049 reduce the CDF by about 53%).

There was no fundamental shift in the scope and technical approach with regard to MELCOR analysis performed in support of the CPRR rulemaking when compared to what was done in SECY-12-0157. The scope of MELCOR analysis falls broadly into two categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions, and (2) assessment of source terms (i.e., timing and magnitude of fission product releases to the environment). The technical approach in both cases (SECY-12-0157 and SECY-15-0085) takes into account best estimate modeling of accident progression, and incorporates both preventative and mitigative accident management measures including venting, water addition and/or water management, and the use of engineered filters. The selection of accident sequences covered by the MELCOR calculation matrix was informed by the set of accident sequences delineated in the accident sequence analysis. That said, the current analysis incorporates operator actions (e.g., opening and closing the wetwell vent early in an ELAP, anticipatory early venting rather than keeping the vent closed until core damage is imminent, RPV depressurization to 200-400 psig in order to minimize SRV cycling and heatup of the suppression pool, severe accident water management, etc.) that were not considered in SECY-12-0157. Some of these are strategies selected by the industry to comply with Order EA-12-049, “Mitigation Strategies for Beyond-Design-Basis External Events.”

The outcome of MELCOR analysis in the thermal-hydraulics category includes RPV and containment temperature and pressure signatures, and hydrogen distribution in the containment, reactor building, and vent line, all indicative of the state of containment vulnerability under severe accident conditions. These quantities provide needed information to assess containment integrity and also provide technical insights for developing staff guidance for Order EA-13-109 (severe accident capable hardened vent). The MELCOR analysis also provides source terms (estimates of fission products release to the environment), which are used by MACCS to calculate offsite consequences.

The MACCS analysis documented in this report, likewise, is similar to that in SECY-12-0157. With regard to the technical approach, the analysis in both cases uses source term estimates from MELCOR to calculate atmospheric transport and dispersion, protective actions, exposures, and resulting offsite consequences. The offsite consequence results are presented in terms of individual early fatality risk, individual latent cancer fatality (ILCF) risk, population dose, offsite cost, contaminated land area, and population subject to long-term protective actions.

The quantitative results from the current analysis indicate no early fatality risk for all cases analyzed and the ILCF risk a factor of 10 lower than the recommended Safety Goal QHO acceptance level. The conditional ILCF risk (per event) is dominated by long-term phase exposures to contaminated areas. Because of the habitability criterion, this metric is relatively insensitive to the source term magnitude, whereas the societal consequence metrics are often more sensitive. The Commission direction in the SRM-SECY-12-0110, “Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission’s Regulatory Framework,” was considered by the staff in weighing land contamination and other factors.

6.1 <u>Key Insights from Accident Sequence Analysis and Risk Integration</u>

The accident sequence analysis results show a low value for core damage frequency from an ELAP event, and provide insights into which initiating events, mitigation systems, and operator actions contribute the most to overall core damage frequency for the BWR plants with Mark I and Mark II containments. Key insights from the accident sequence analysis include:

- The major contribution to seismically induced ELAP is from earthquakes whose ground motion exceeds the plant design basis (the safe shutdown earthquake). Specifically, earthquakes with peak ground accelerations in the range of 0.3 to 0.75g are the major contributors.

- Significant contributors to CDF include seismic failures of the batteries, DC switchgear, and the EDGs and their supporting equipment. Failure of the portable FLEX pump and failure to start of the RCIC pump are also significant contributors.

- CDF is not particularly sensitive to the human error probabilities for in-control-room and ex-control-room operator actions.

- The estimated mean individual latent cancer fatality risk (0-10 miles) is more than two orders of magnitude below the NRC Safety Goal QHO. The risk is low because the core-damage frequency is low and the conditional latent cancer fatality risk is low. The range of parametric uncertainty in the risk estimates is more than one order of magnitude, and is largely driven by uncertainty in the seismic hazard curves.

- The estimated individual early fatality risk is essentially zero in all cases analyzed and for all alternatives considered, consistent with the findings previously in SOARCA and SFP studies. This risk remains unchanged for a wide range of sensitivity analysis.

6.2 Key Insights from Accident Progression Analysis

The MELCOR analysis investigated detailed accident progression, source terms, and the containment response following an ELAP subject to appropriate initial and boundary conditions for the representative Mark I and Mark II containment designs. The analysis included sensitivities to venting, RPV depressurization, water injection (to RPV or drywell), and water management strategy. Additional sensitivities were performed for Mark II containments to examine the impact of the different cavity designs among the fleet. Following are some key insights from the MELCOR analysis:

Venting of Mark I and Mark II containments effectively prevents containment overpressure failure. Pre-core damage anticipatory venting reduces the containment base pressure at the time of core damage and results in a delay when post core damage venting is required. Post-core damage containment venting is efficient in purging hydrogen and other non-condensables from the containment, and reduces the likelihood of combustible quantities of non-condensable accumulation in the reactor building.

- Creep rupture of the main steam line seems unlikely if the reactor pressure is maintained low. The failure of the main steam line results in failure of the containment (opening of the upper drywell head) and bypass of the suppression pool. The failure also results in migration of hydrogen to the refueling bay of the Mark I containment, and consequent hydrogen combustion and release of fission products directly to the environment.

- Venting alone, however, is not adequate, as it does not prevent other modes of containment failure such as liner melt-through and over-temperature failure of the upper drywell head, bypass of the suppression pool and direct release of radioactivity to the environment. A combination of venting and water injection is required to prevent such failures, and the current work provides a sound technical basis to that effect thus supporting adequate protection argument.

- Addition of water either into the RPV or the drywell has the following benefits: (1) cooling of the core debris and containment atmosphere; (2) preventing over-temperature failure of the upper drywell head; (3) preventing and/or delaying liner melt through in Mark I containments; (4) maintaining a steam inerted atmosphere which can preclude an energetic hydrogen combustion; and (5) mitigating radiological releases as it effectively provides means for fission products scrubbing.

- Environmental releases from Mark II containments are in general comparable to or lower than those from Mark I containments. Sensitivity analysis performed to investigate variations in lower cavity configurations of Mark II containments indicate the environmental releases for all configurations are within the range of releases predicted for Mark I containments.

- Assuming the condensate storage tank survives a beyong design basis accident, RCIC suction initially taken from the CST provides a better alternative to suction from suppression pool (SP) as this action will likely extend the duration of RCIC duration.

6.3 Key Insights from Offsite Consequence Analysis

The MACCS evaluation results include individual early fatality risk, ILCF risk, total population dose, total offsite cost, land contamination, and population subject to long-term phase protective actions for all Mark I and II source terms calculated by MELCOR. The analysis included several sensitivities such as evacuation delay, non-evacuating population, and habitability criteria. Following are key insights from the MACCS analysis:

- There is essentially zero individual early fatality risk for all Mark I and Mark II source term bin cases analyzed because the releases are not large enough to exceed the threshold for the acute dose to the red bone marrow, which is typically the most sensitive tissue for early fatalities.

- For all Mark I and II source terms analyzed, the conditional ILCF risk is sufficiently low that when multiplied by the accident frequency, there is at least a two order of magnitude margin to the QHO.

- The calculated releases to the environment are delayed long enough after the accident initiation to allow time for the emergency planning zone (EPZ) population to evacuate. Therefore the ILCF risk is dominated by long-term phase exposures to slightly contaminated areas (under the 500 mrem per year habitability criterion threshold).

- In general, a larger release of radioactive materials to the environment displaces more people for more time, and therefore incurs a larger societal cost. However, for a larger release, the cancer fatality risk to the public shows a nonlinear effect because protective actions are in place primarily to reduce exposures (habitability criterion) at the tradeoff of other societal costs such as land contamination and economic losses.

- The potential effectiveness of an external filter on reducing the environmental release is heavily influenced by release pathways. For accident cases in which the entire release is through the venting system, there can be significant reductions to the environment. However, there is considerably less benefit from an external filter when any of the release bypasses the venting system.

- MACCS sensitivity calculations which generally showed the greatest increase in individual latent cancer fatality risk were those with medium and large releases in which the EPA PAG habitability criterion (2 rem in the first year) was used instead of the Pennsylvania habitability criterion (500 mrem in the first year), thus allowing a first year dose up to four times higher. The individual latent cancer fatality risk was commonly 2-3 times higher than for the base cases.

- All sensitivity calculations conducted, including those with changes to population, evacuation characteristics, intermediate phase duration, and long-term habitability criterion, do not change any of the consequence analysis insights related to the QHO metrics. Individual early fatality risk remains essentially zero for all sensitivity calculations conducted, and ILCF risk remains well below the QHO.

6.4 Concluding Remarks

Coincident with the analyses by the NRC staff and industry related to the CPRR rulemaking and related Orders, licensees were developing revisions to the severe accident managmenet guidelines (SAMGs) to address lessons learned from the Fukushima accident. The analyses performed to address issues related to containment venting and severe accident water addition provided valuable insights and supported actual revisions to the SAMGs for plants with Mark I and Mark II containments. This incorporation of insights from the modelling of beyond-design-basis events and severe accidents into plant guidance documents provides a useful example of the potential benefits of the efforts by the NRC and industry to develop improved analytical capabilities. Another example is the use of the results and technical insights in formulating the regulatory basis for the mitigation of beyond design basis events rulemaking. This rulemaking, though not imposing any regulatory footprint on SAMG, nevertheless directs the NRC staff to provide periodic oversight to industry's SAMG implementation through NRC's updated Reactor Oversight Program (ROP). A third and related example is that of a recent initiative in many Organization for Economic Cooperation and Development (OECD) countries and a collective effort in the OECD program of work to develop technical insights on how the type of beyond design basis analysis work documented in this report can inform SAMG in a positive way and in so doing, enhance the safety of nuclear power plants.

The NRC staff are exploring possible ways to ensure that future insights from severe accident research and analyses are shared with and considered by licensees even when some of those insights are unlikely to initiate regulatory actions. The NRC staff, through its continued engagement in severe accident research, plans to remain cognizant about both the industry and the regulatory activities in other countries.

APPENDIX A

CORE DAMAGE EVENT TREES

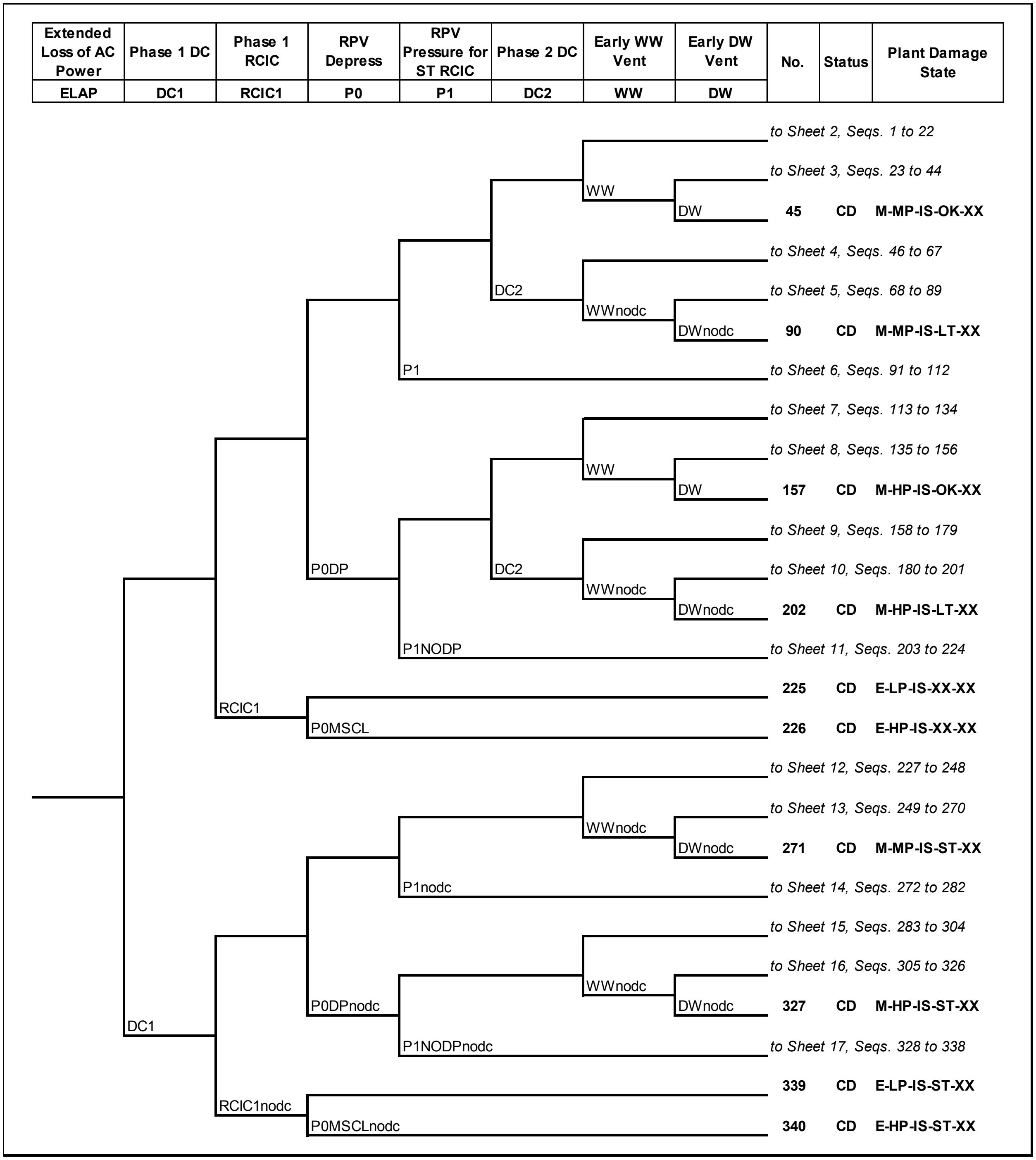


Figure A-1 CDET for WWF venting strategy, Sheet 1

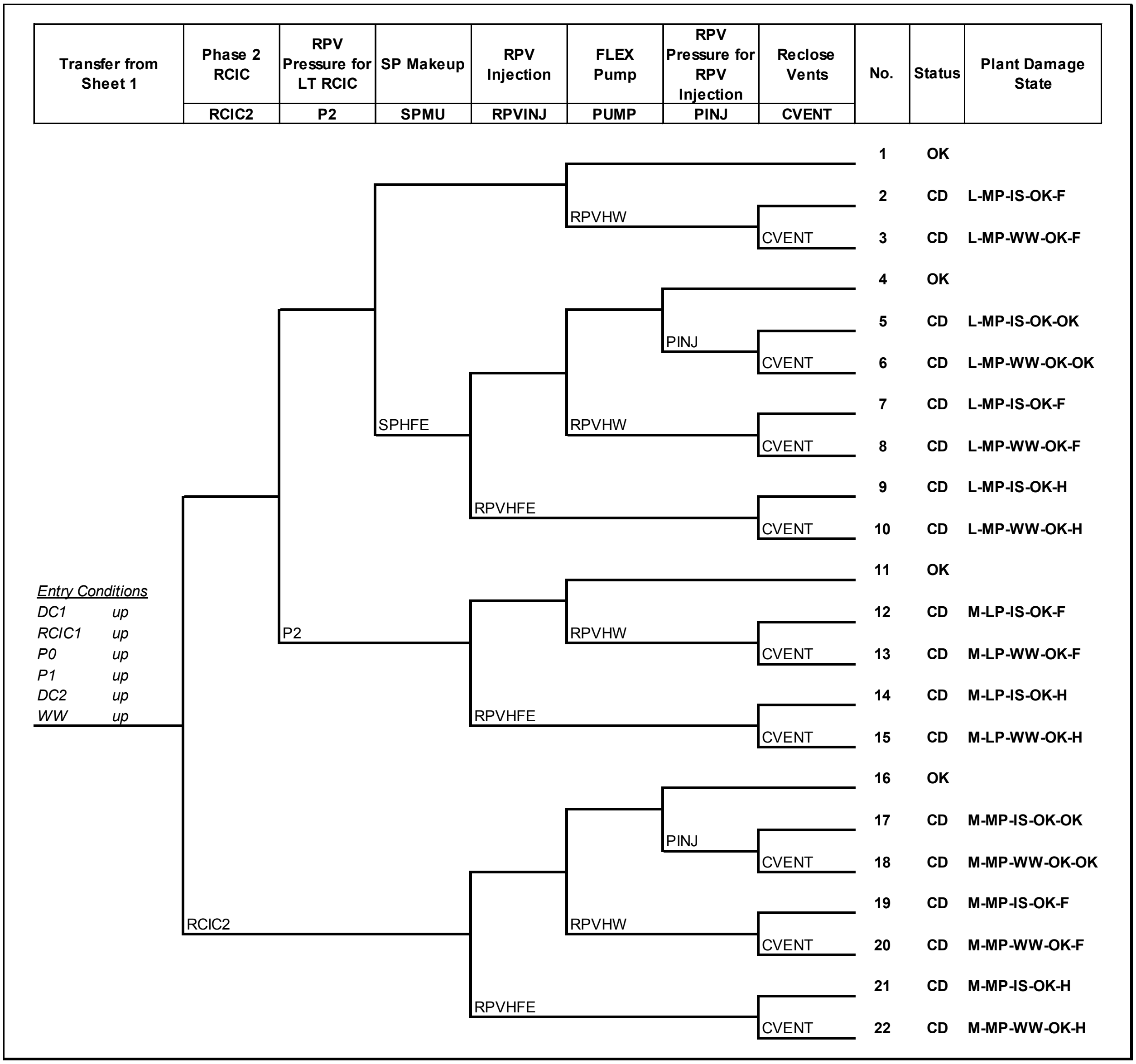


Figure A-2 CDET for WWF venting strategy, Sheet 2, Sequences 1 to 22

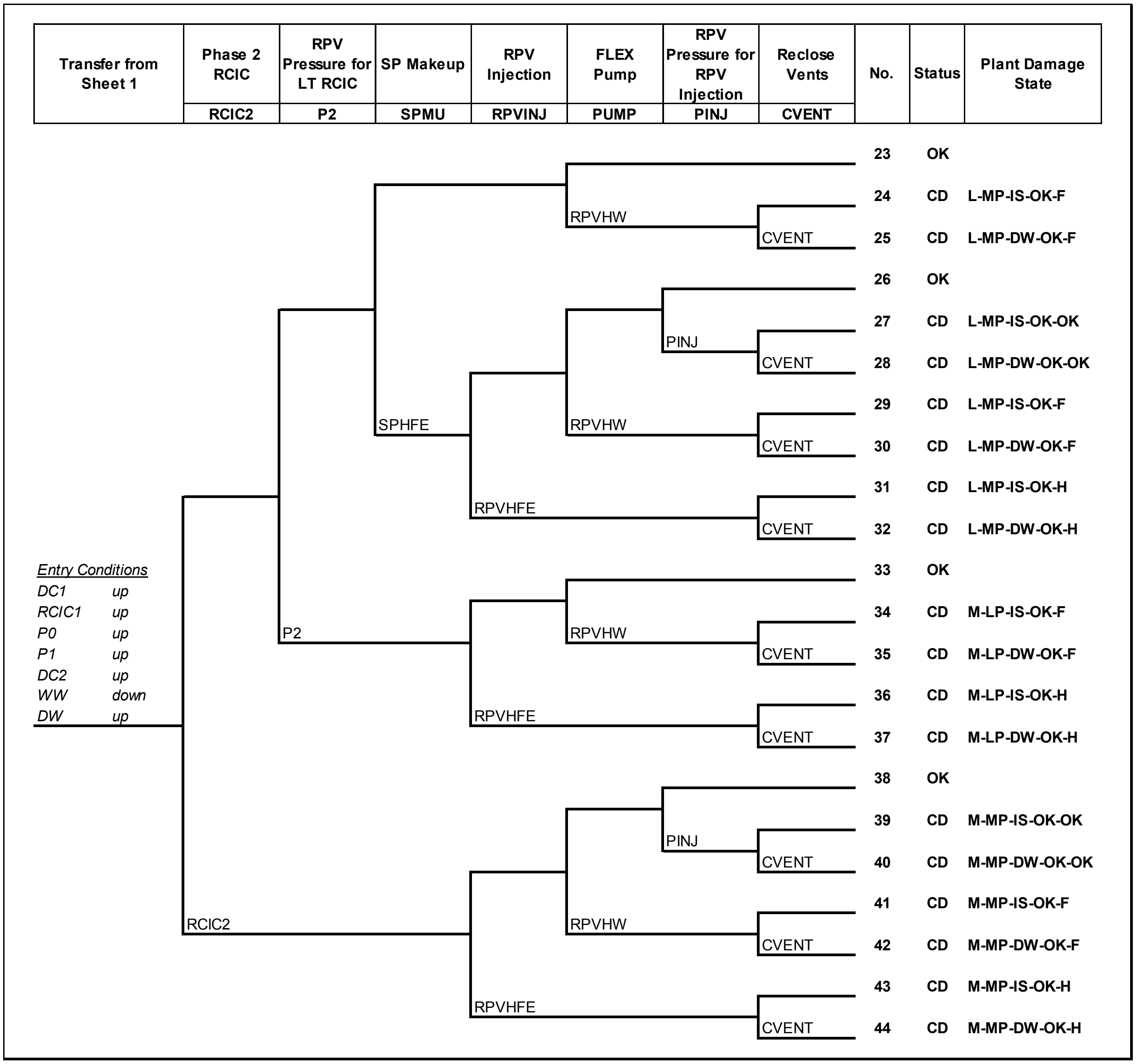


Figure A-3 CDET for WWF venting strategy, Sheet 3, Sequences 23 to 44

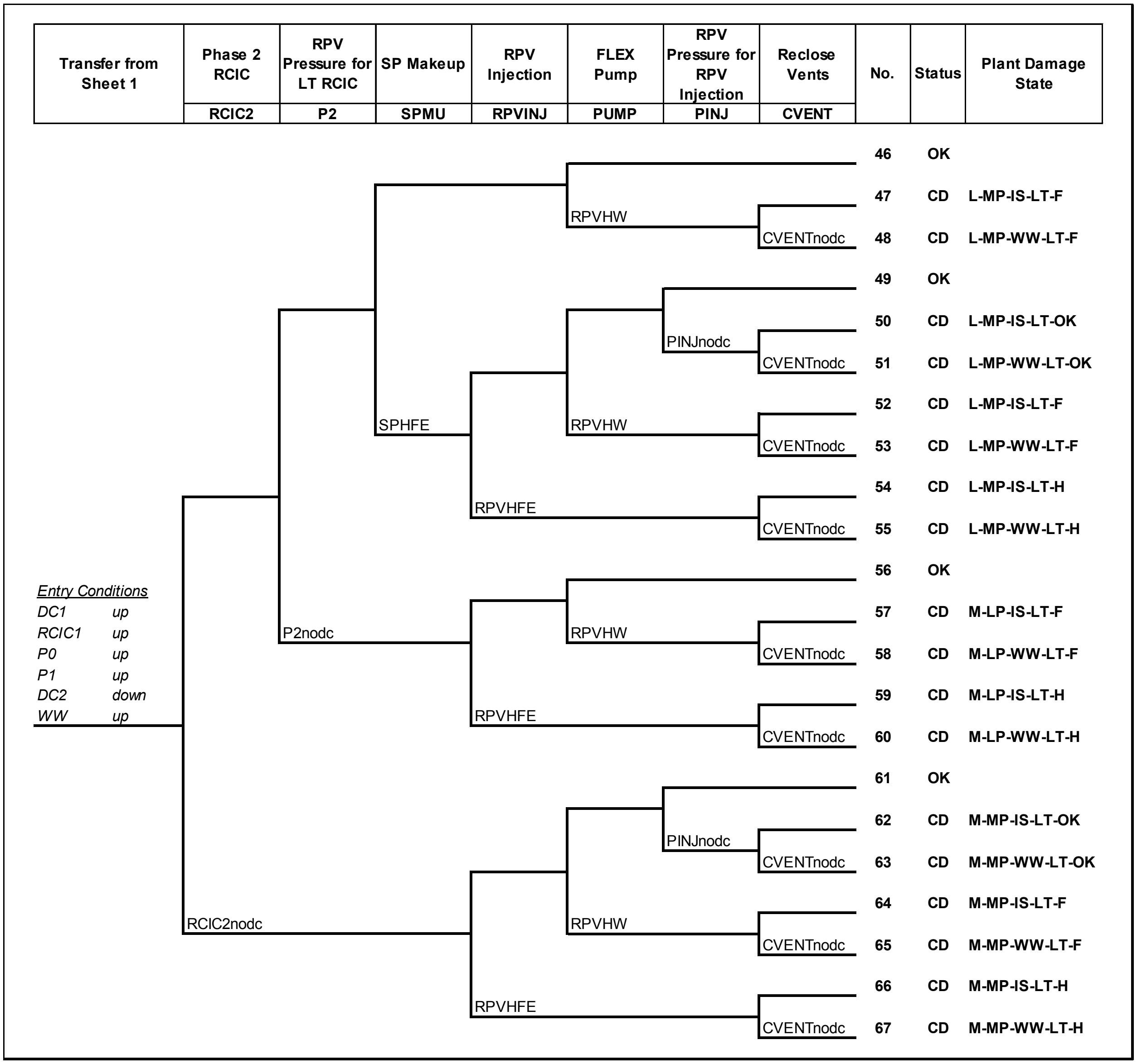


Figure A-4 CDET for WWF venting strategy, Sheet 4, Sequences 46 to 67

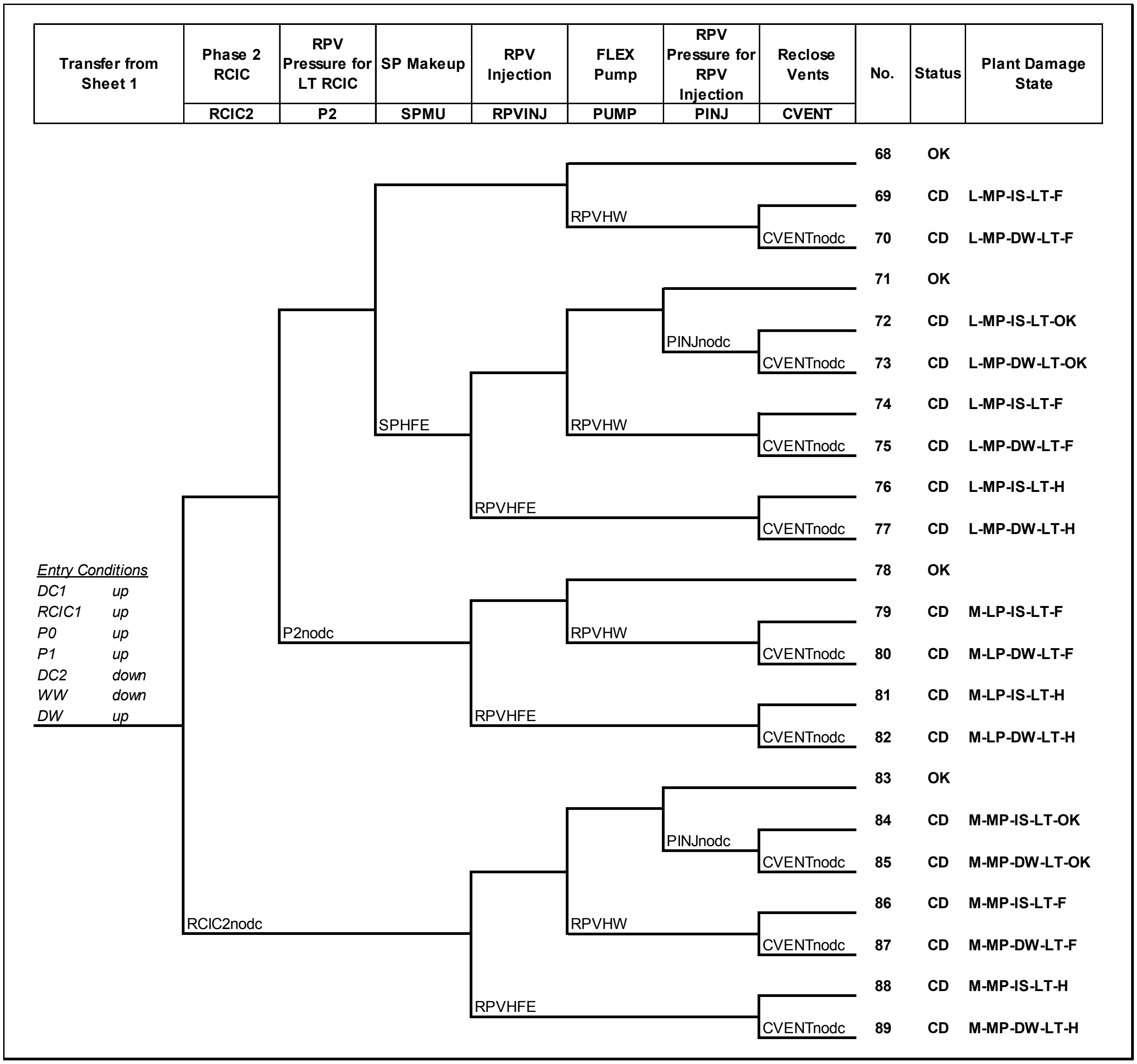


Figure A-5 CDET for WWF venting strategy, Sheet 5, Sequences 68 to 89

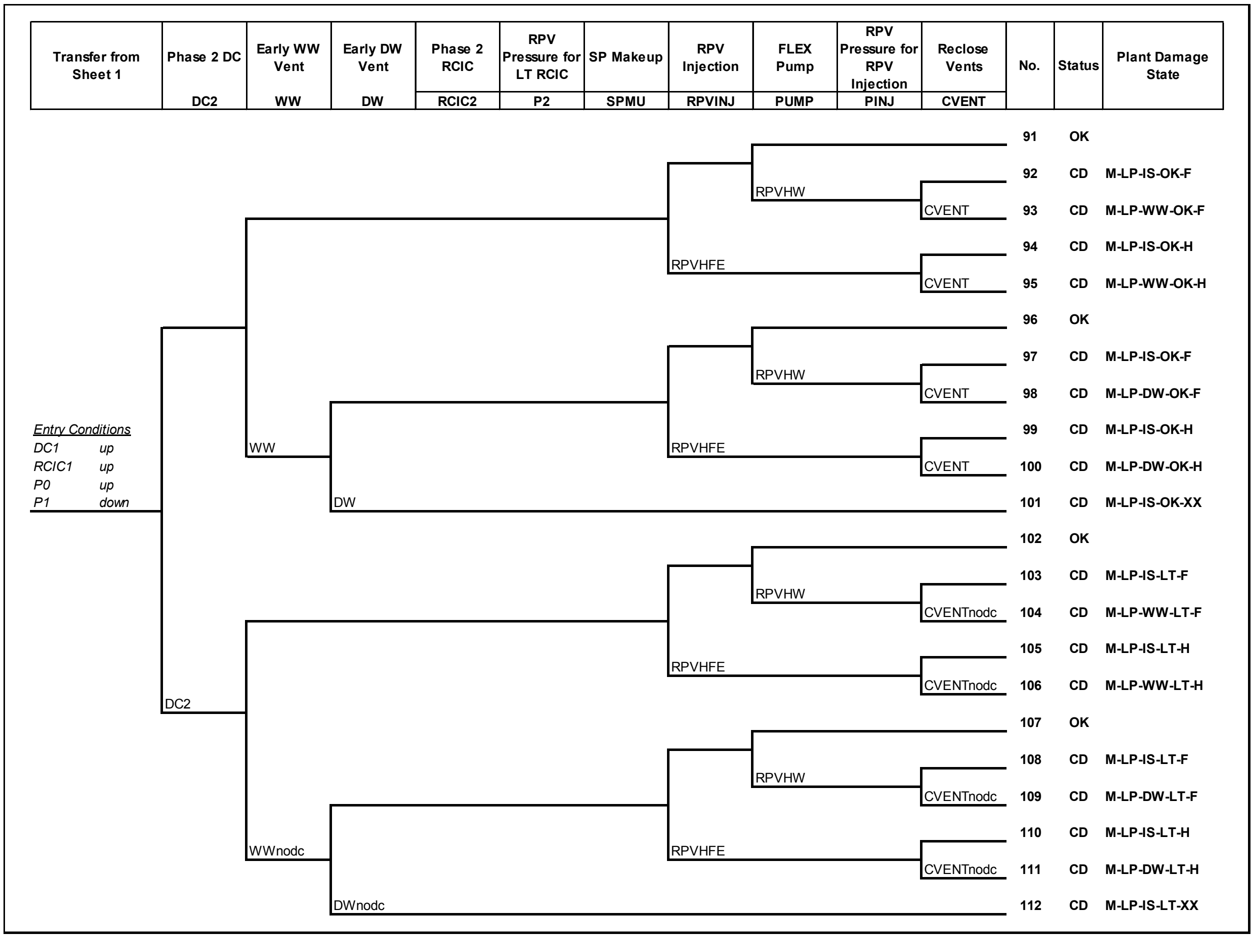


Figure A-6 CDET for WWF venting strategy, Sheet 6, Sequences 91 to 112

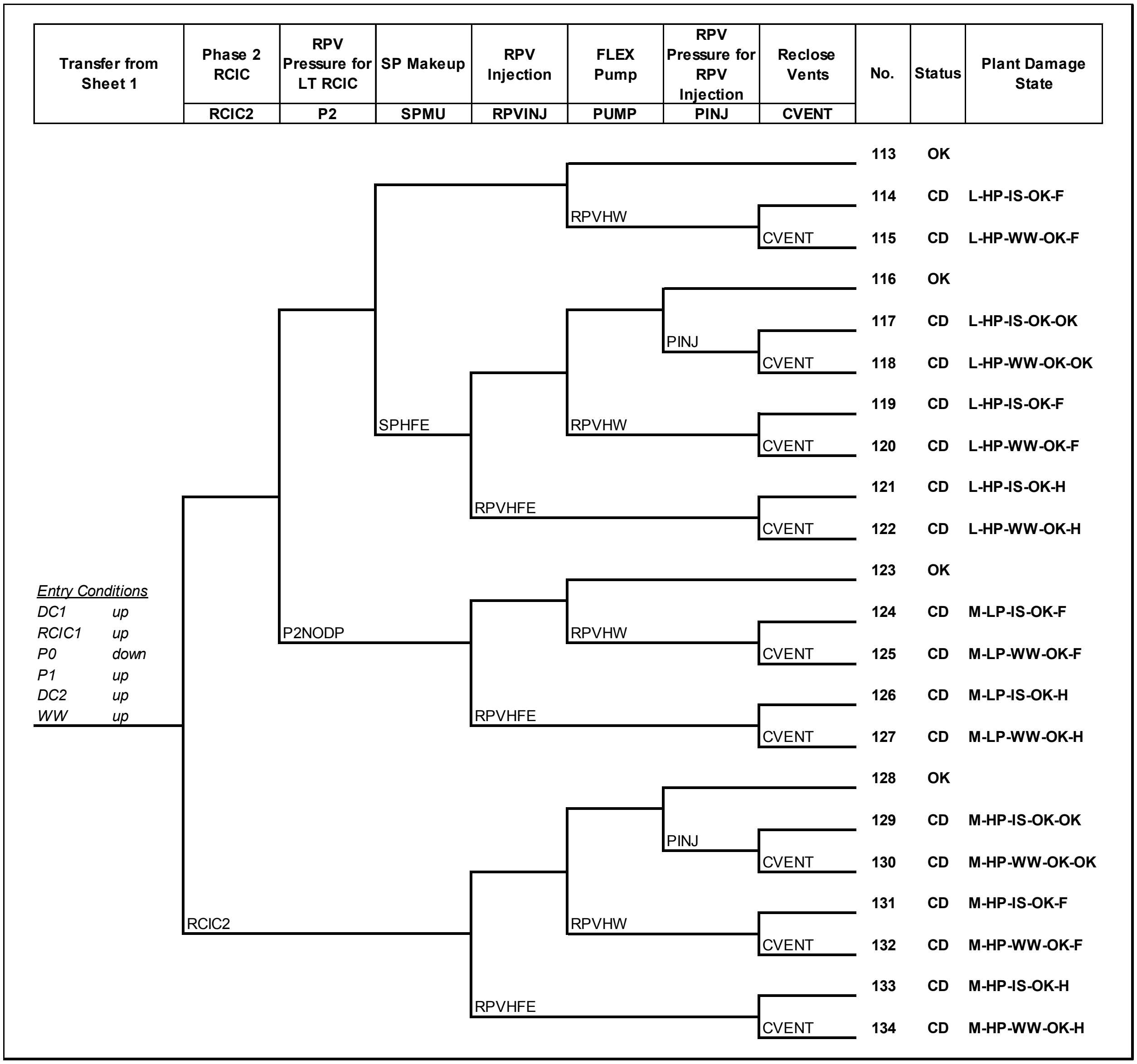


Figure A-7 CDET for WWF venting strategy, Sheet 7, Sequences 113 to 134

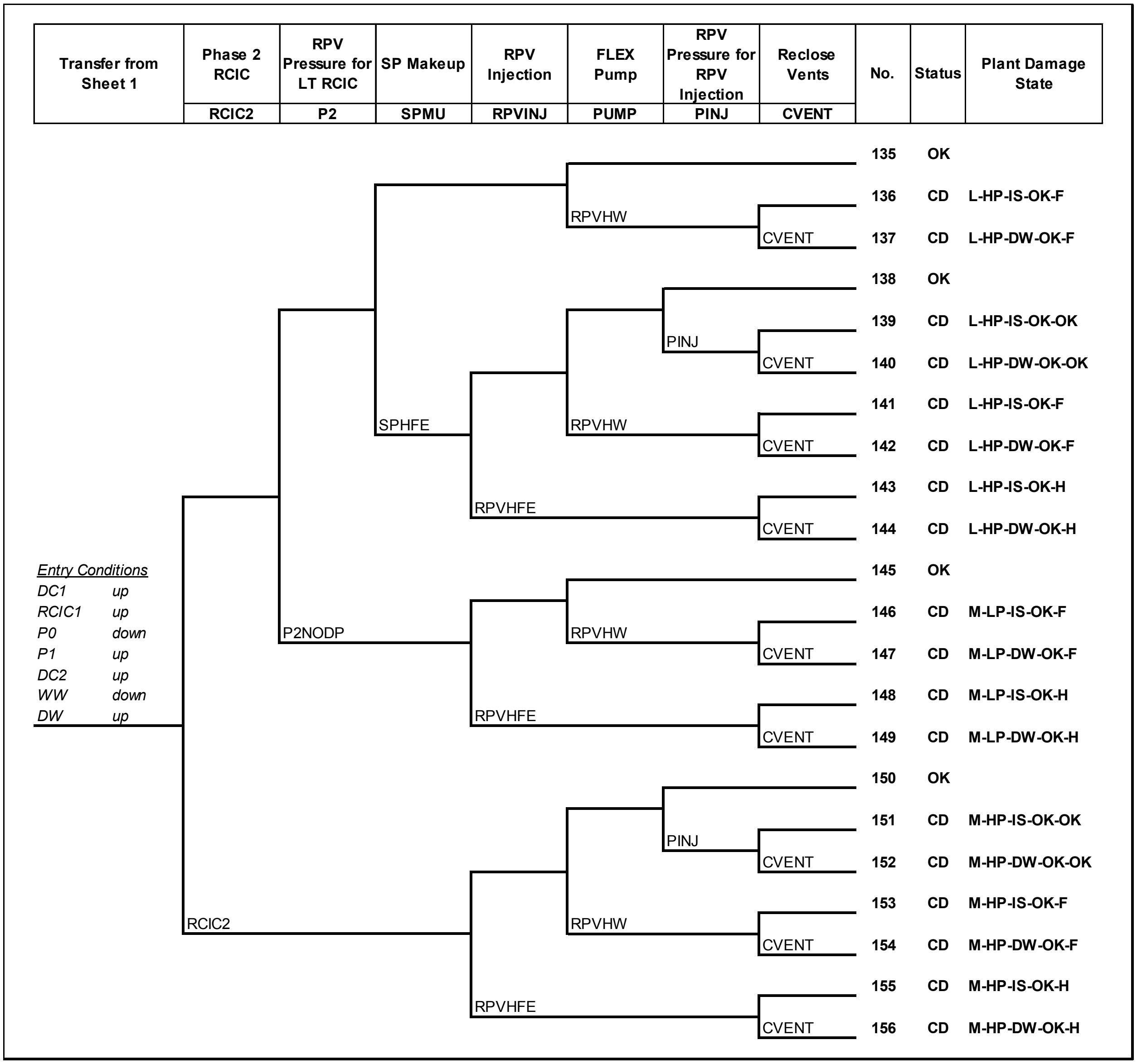


Figure A-8 CDET for WWF venting strategy, Sheet 8, Sequences 135 to 156

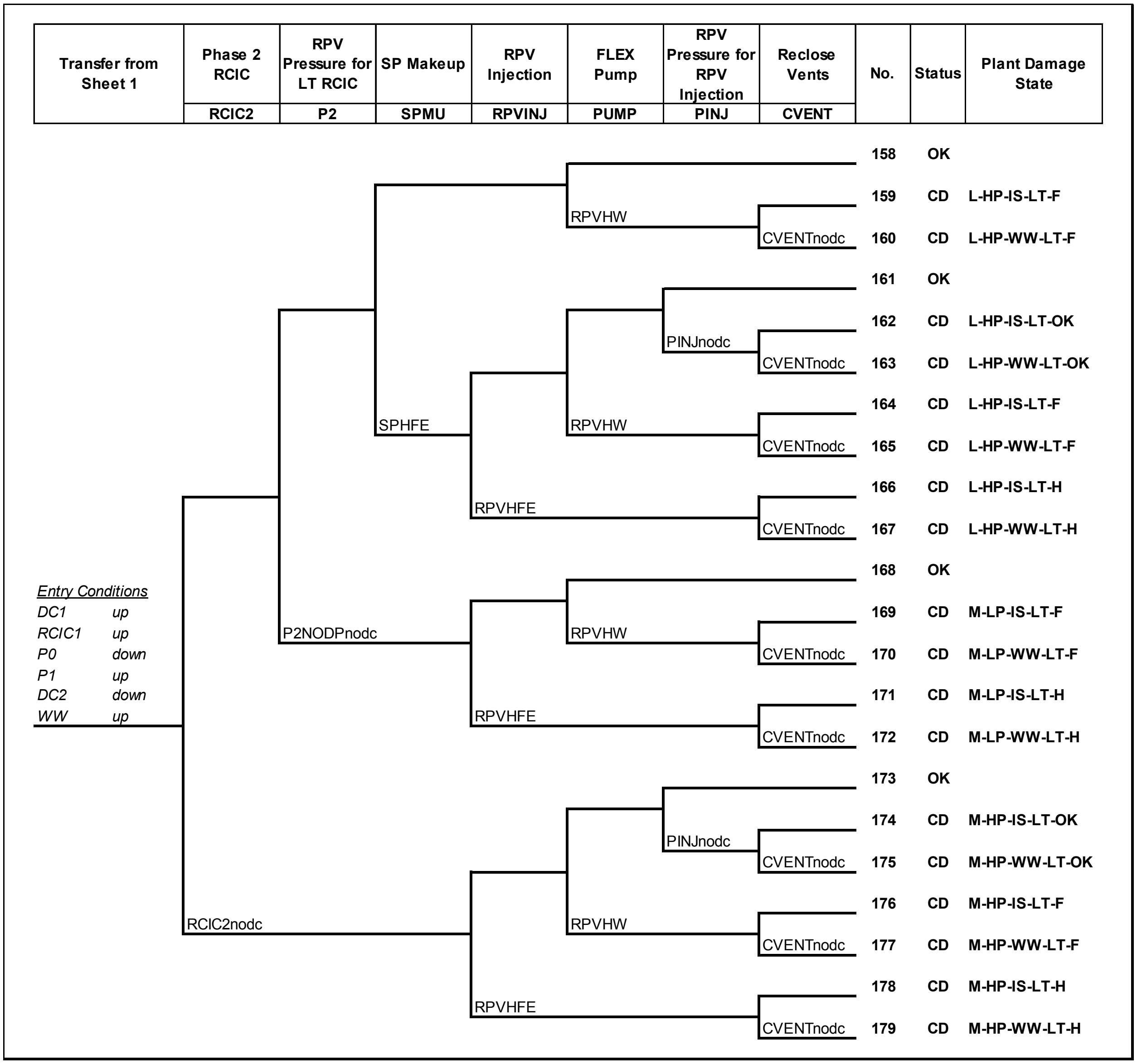


Figure A-9 CDET for WWF venting strategy, Sheet 9, Sequences 158 to 179

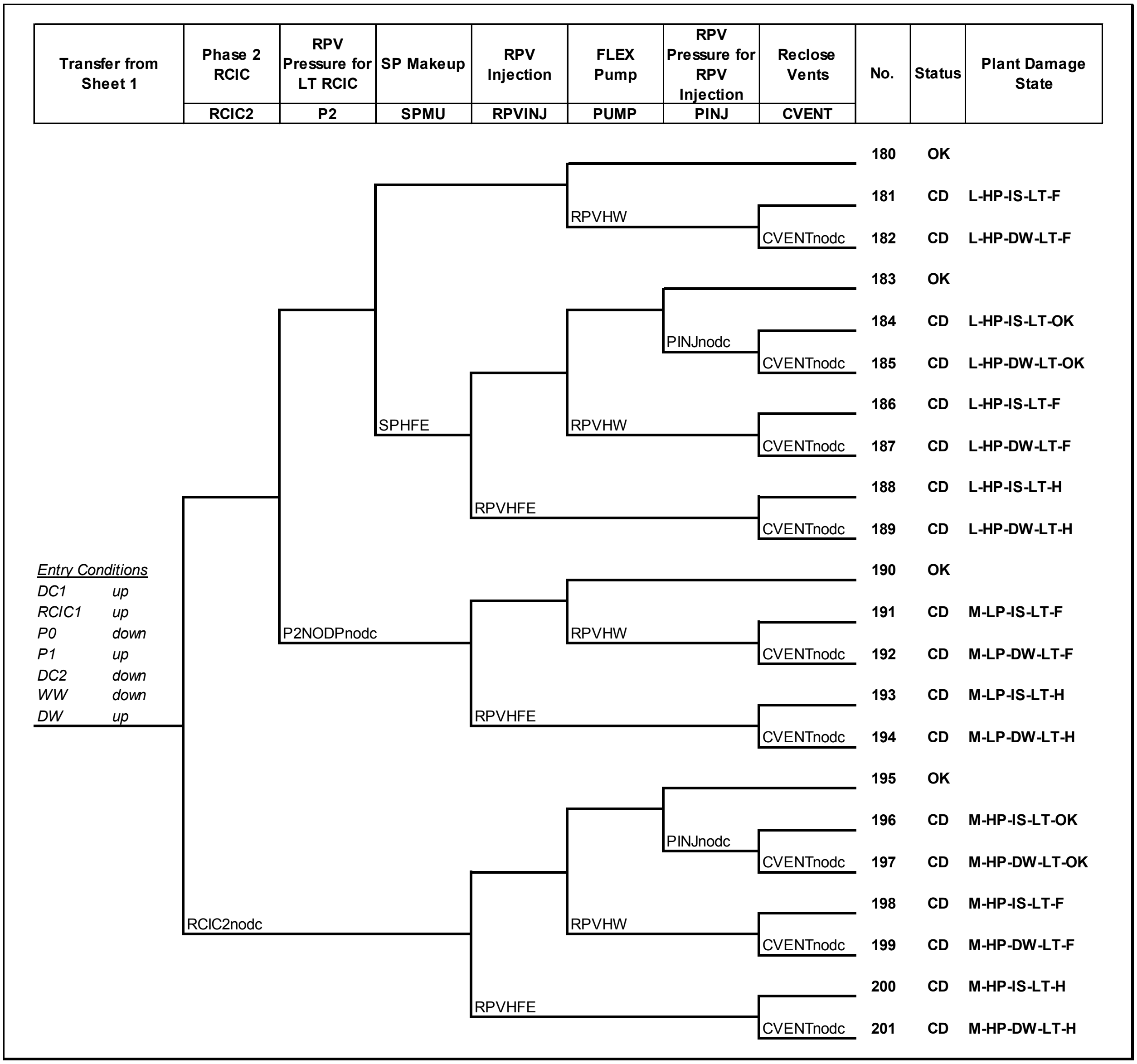


Figure A-10 CDET for WWF venting strategy, Sheet 10, Sequences 180 to 201

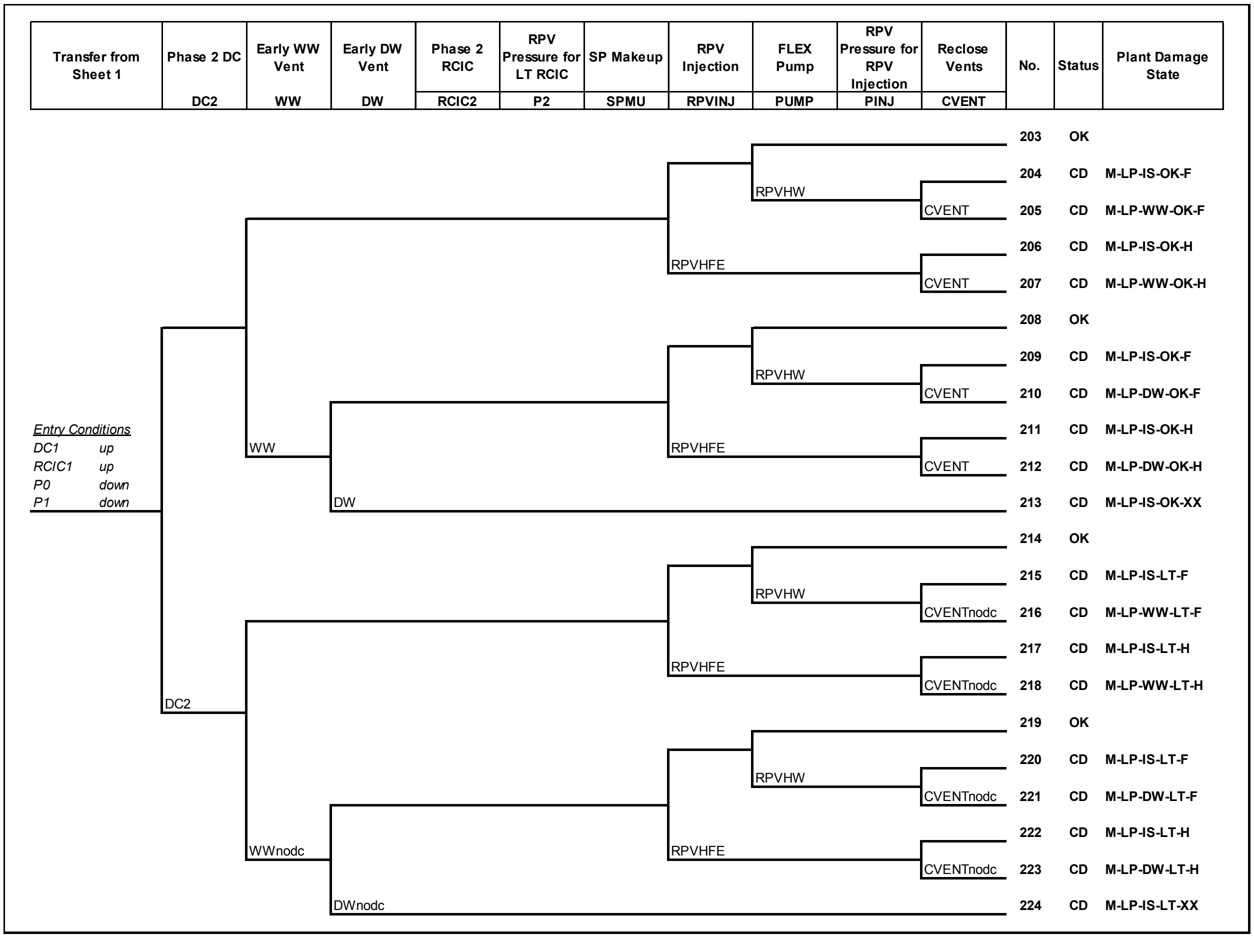


Figure A-11 CDET for WWF venting strategy, Sheet 11, Sequences 203 to 224

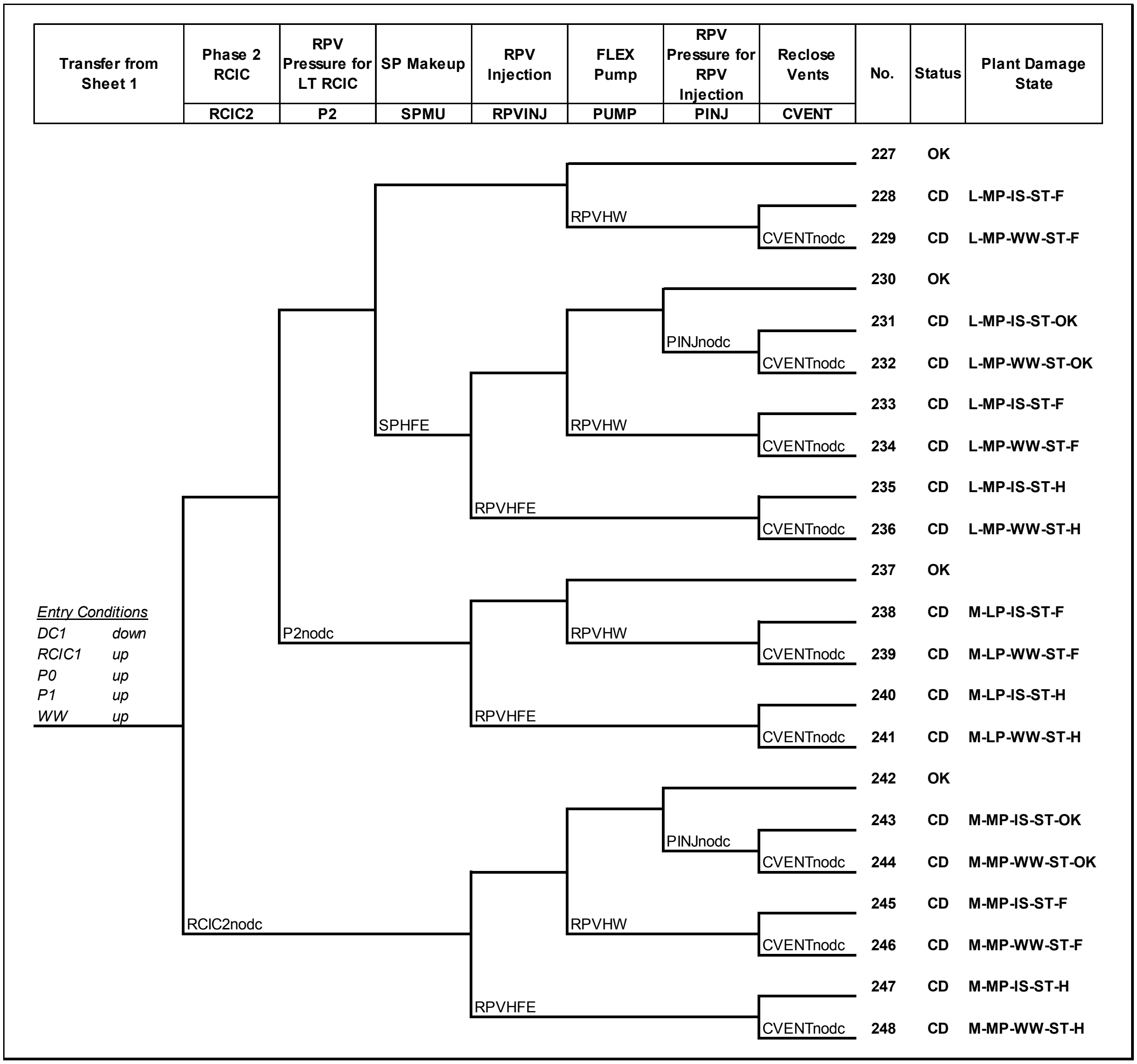


Figure A-12 CDET for WWF venting strategy, Sheet 12, Sequences 227 to 248

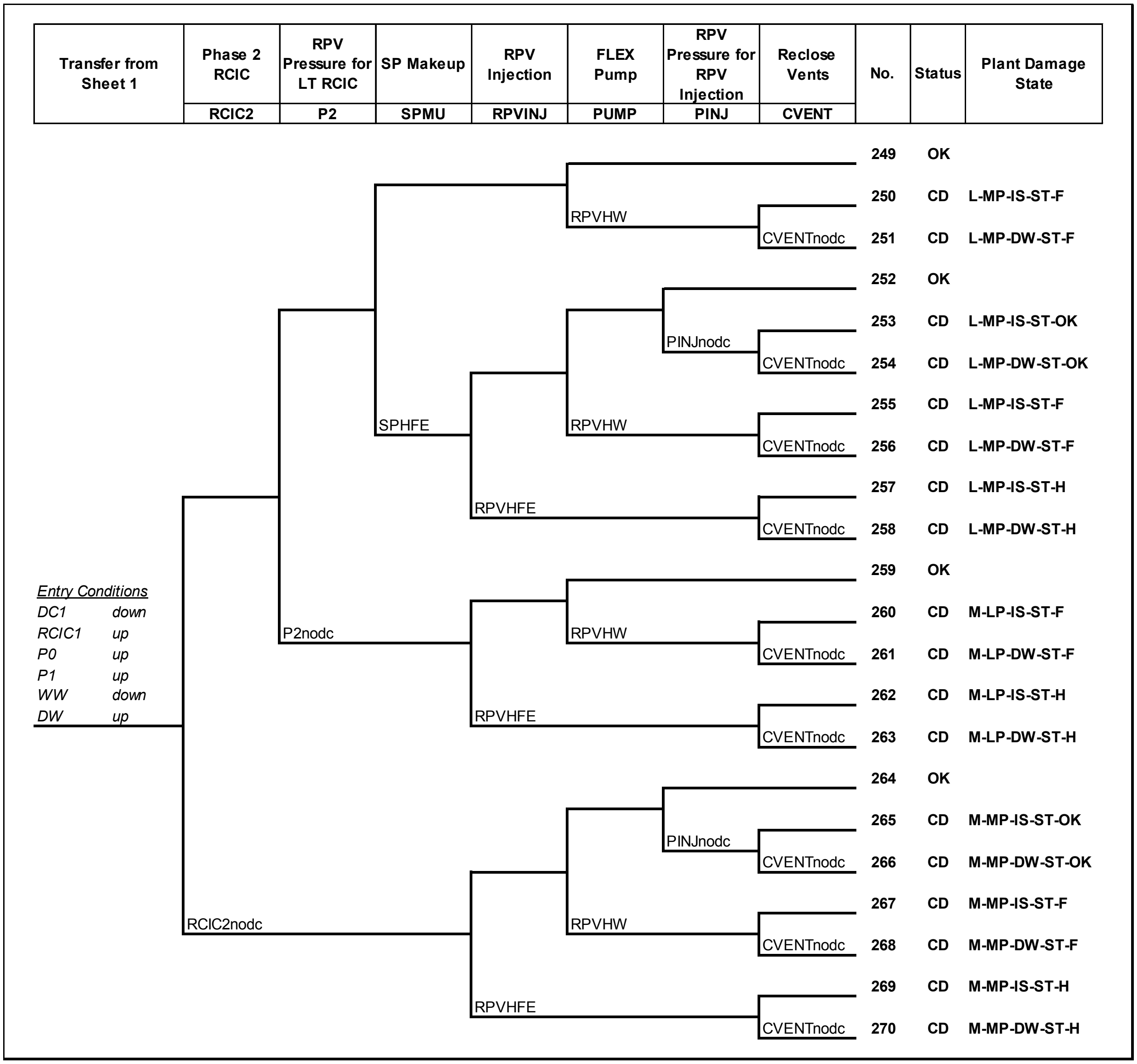


Figure A-13 CDET for WWF venting strategy, Sheet 13, Sequences 249 to 270

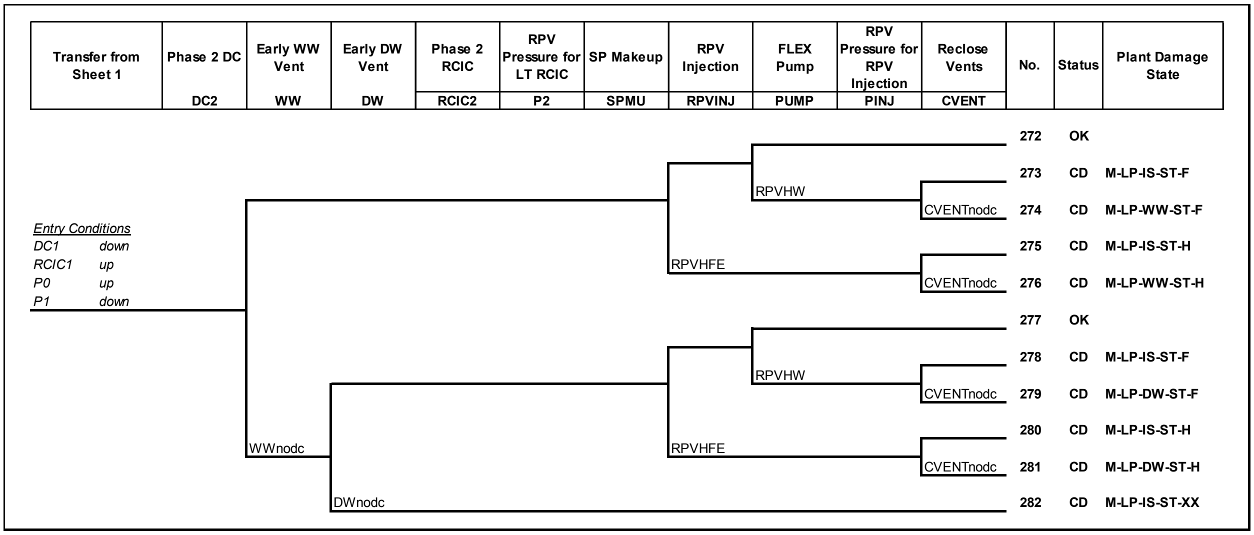


Figure A-14 CDET for WWF venting strategy, Sheet 14, Sequences 272 to 282

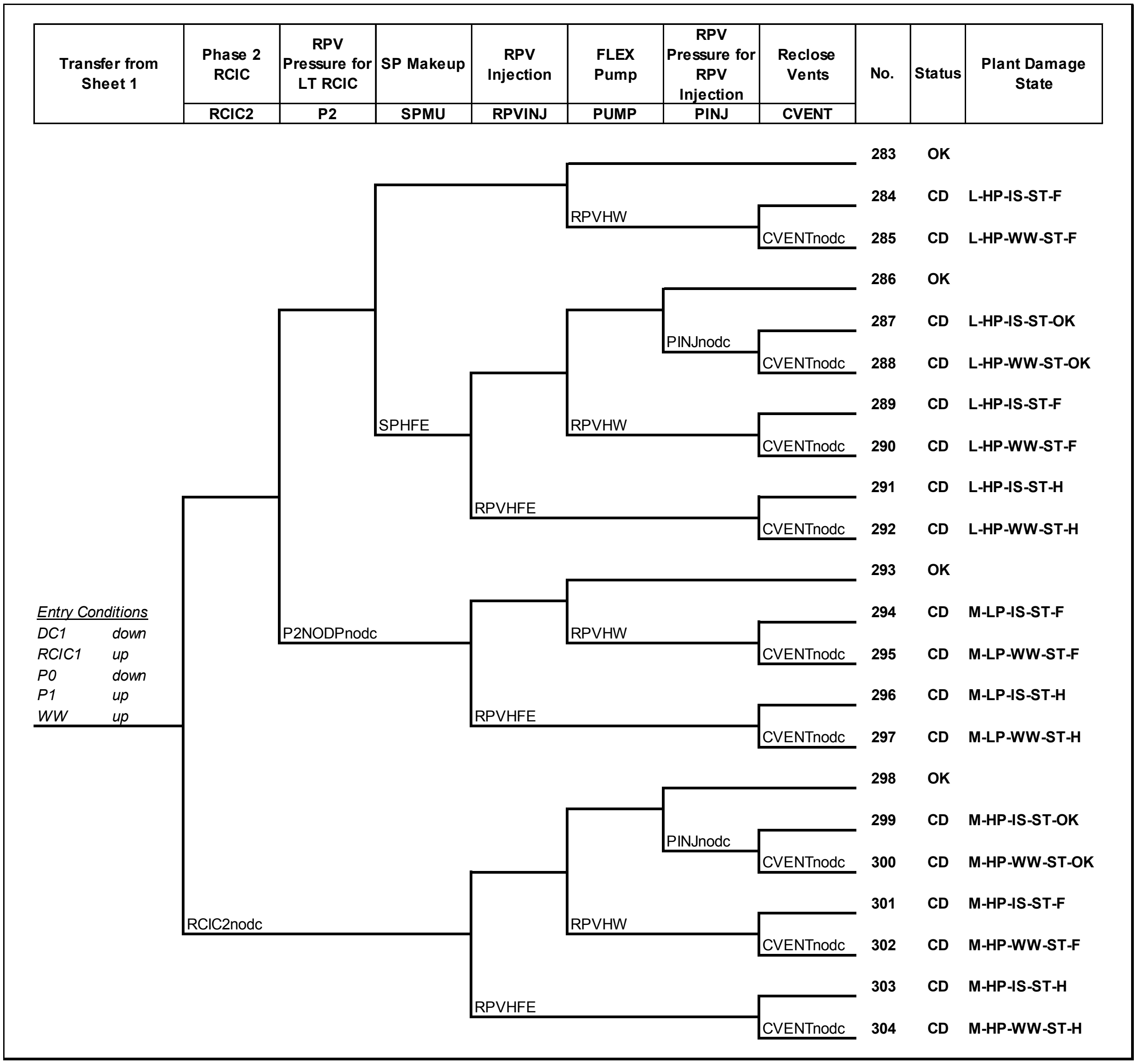


Figure A-15 CDET for WWF venting strategy, Sheet 15, Sequences 283 to 304

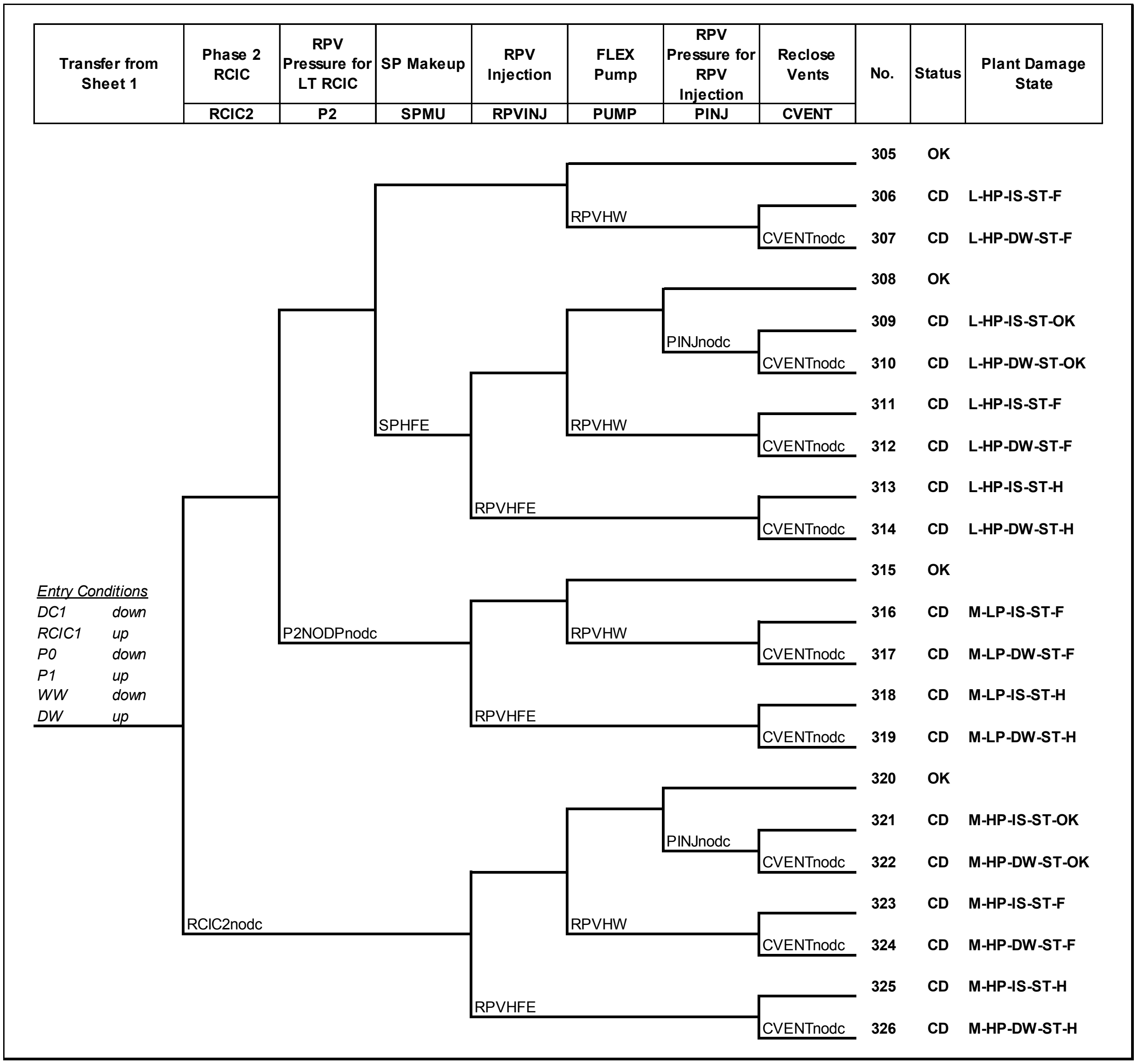


Figure A-16 CDET for WWF venting strategy, Sheet 16, Sequences 305 to 326

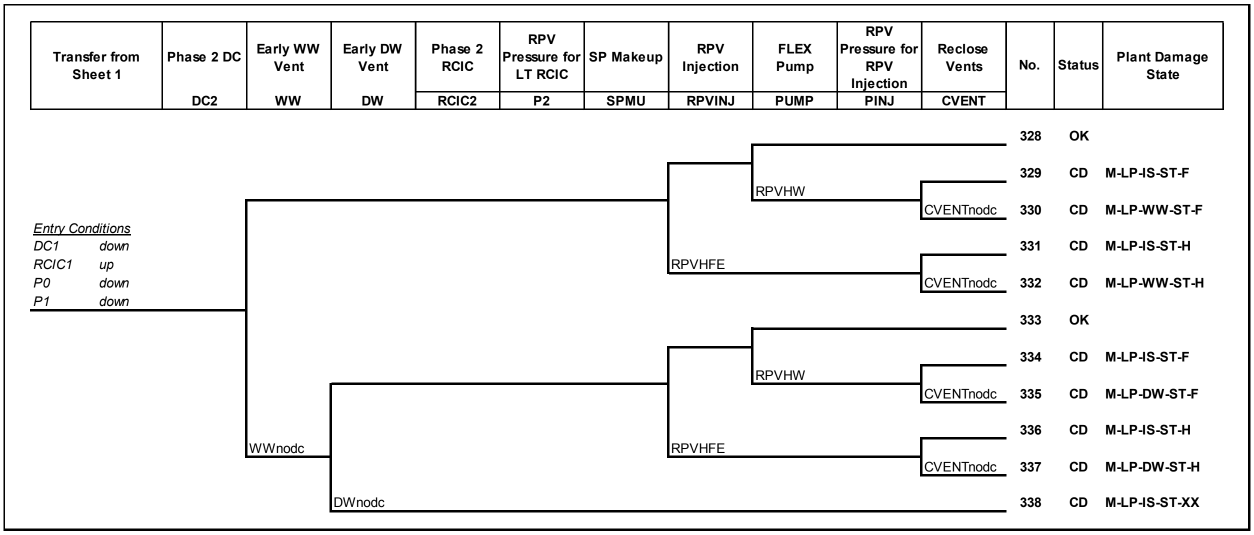


Figure A-17 CDET for WWF venting strategy, Sheet 17, Sequences 328 to 338

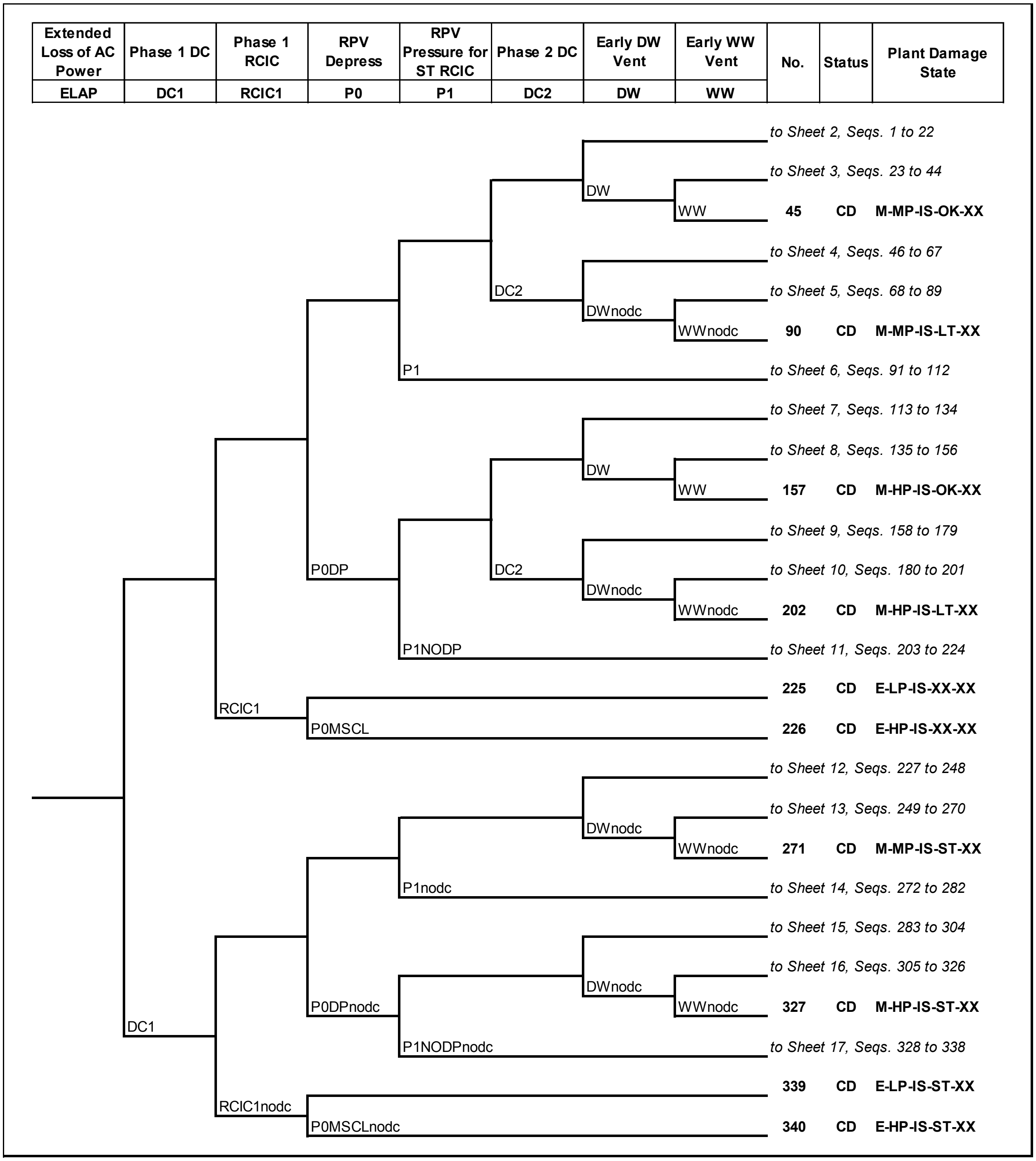


Figure A-18 CDET for DWF venting strategies, Sheet 1

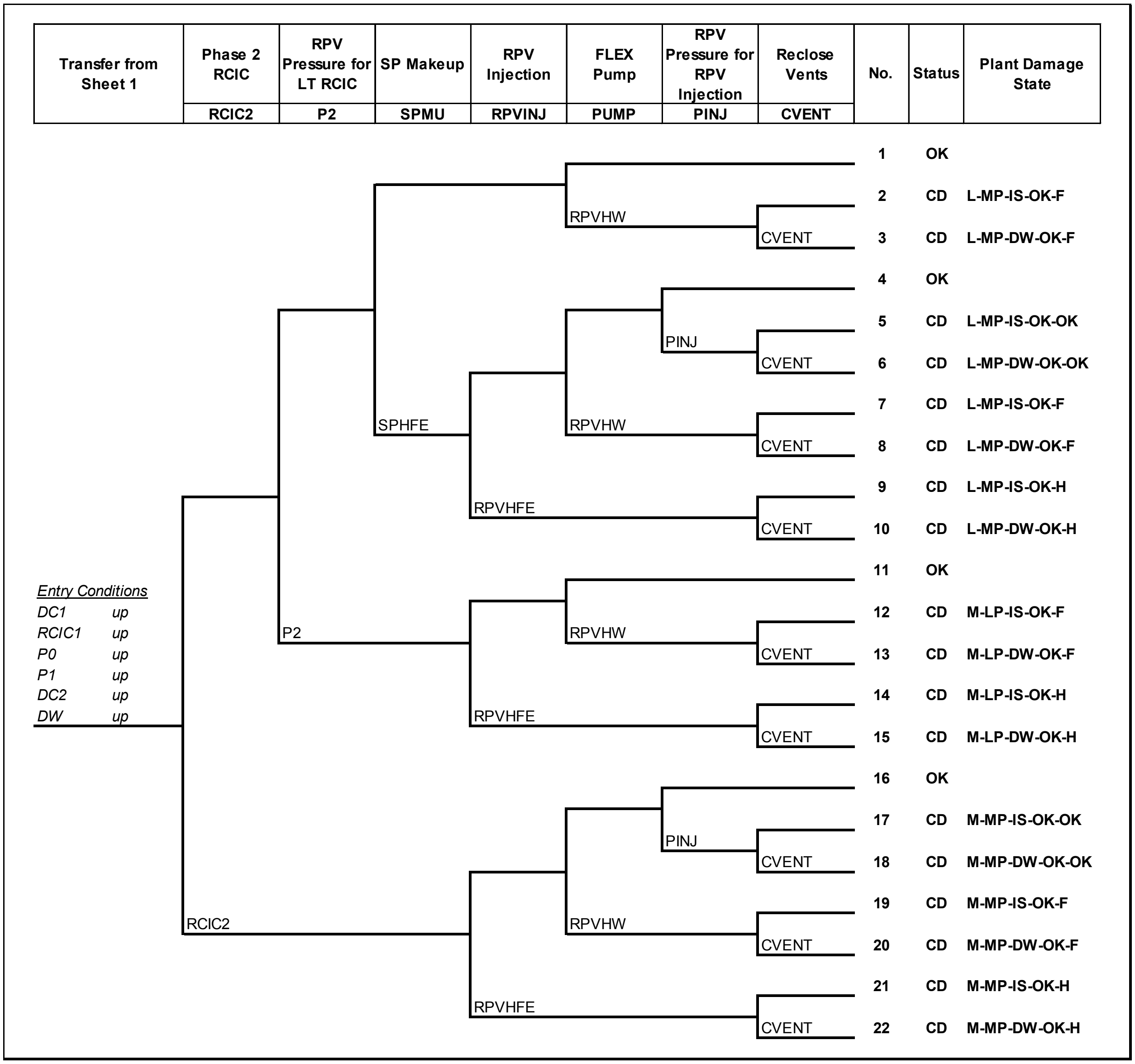


Figure A-19 CDET for DWF venting strategies, Sheet 2, Sequences 1 to 22

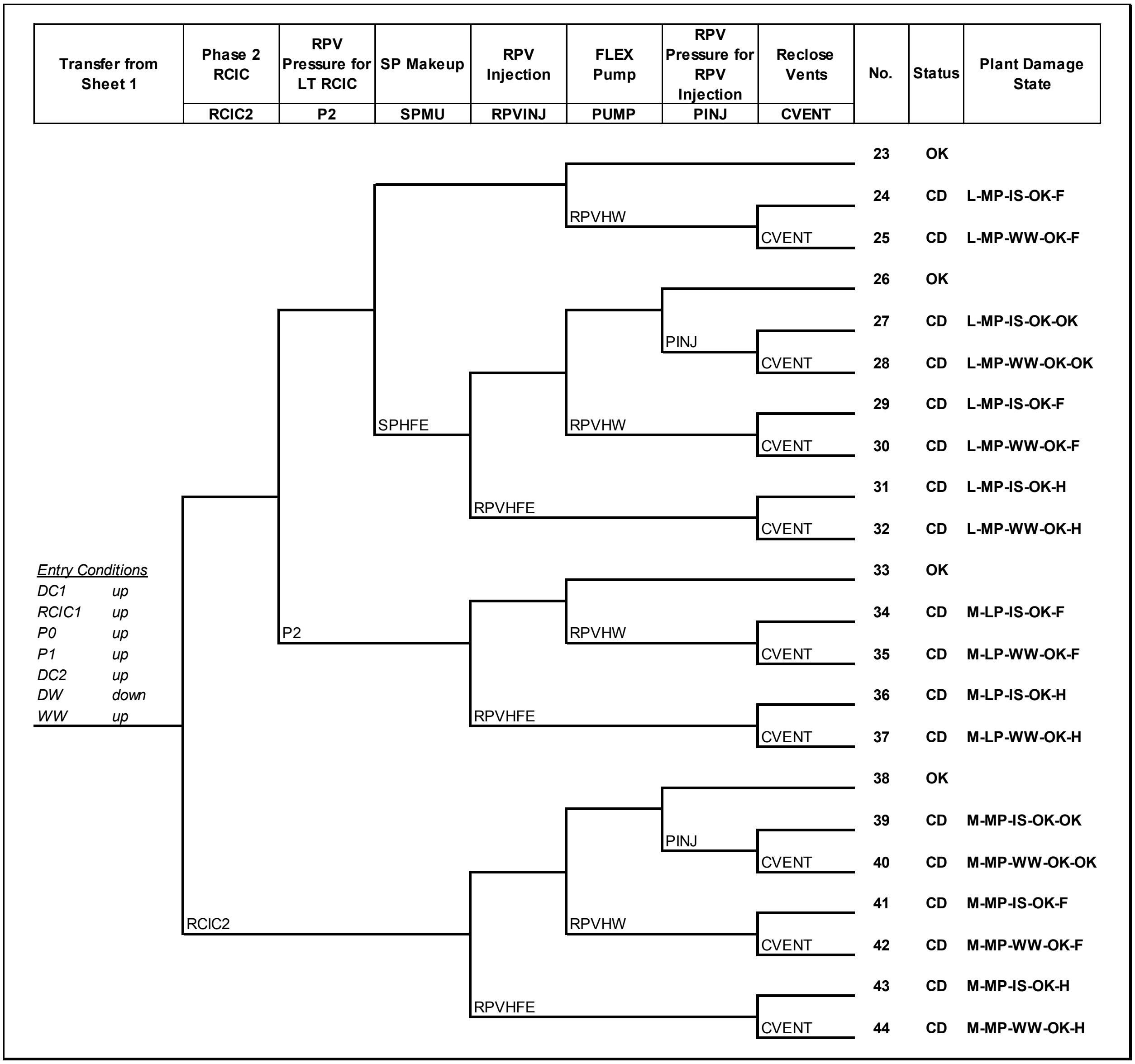


Figure A-20 CDET for DWF venting strategies, Sheet 3, Sequences 23 to 44

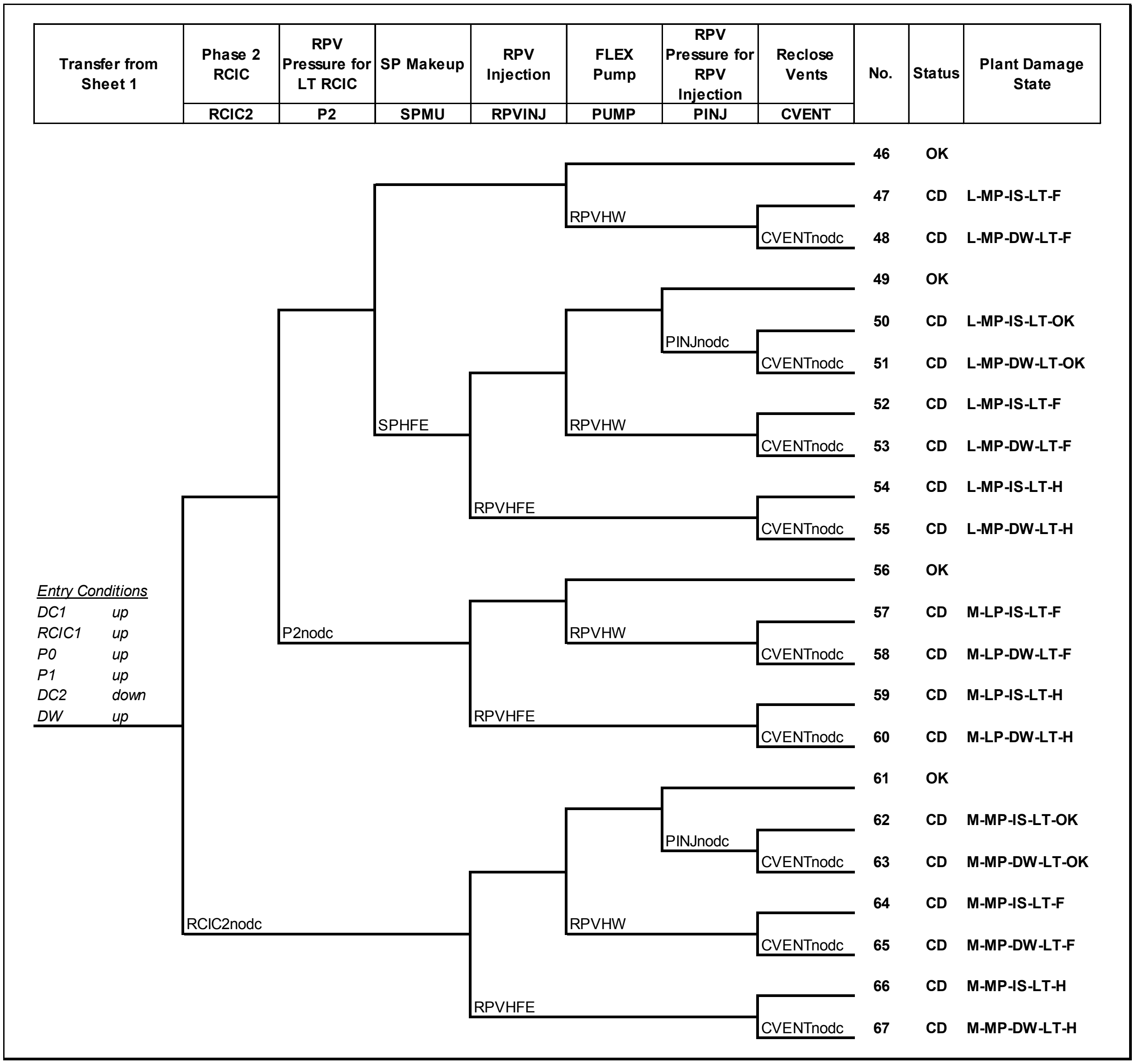


Figure A-21 CDET for DWF venting strategies, Sheet 4, Sequences 46 to 67

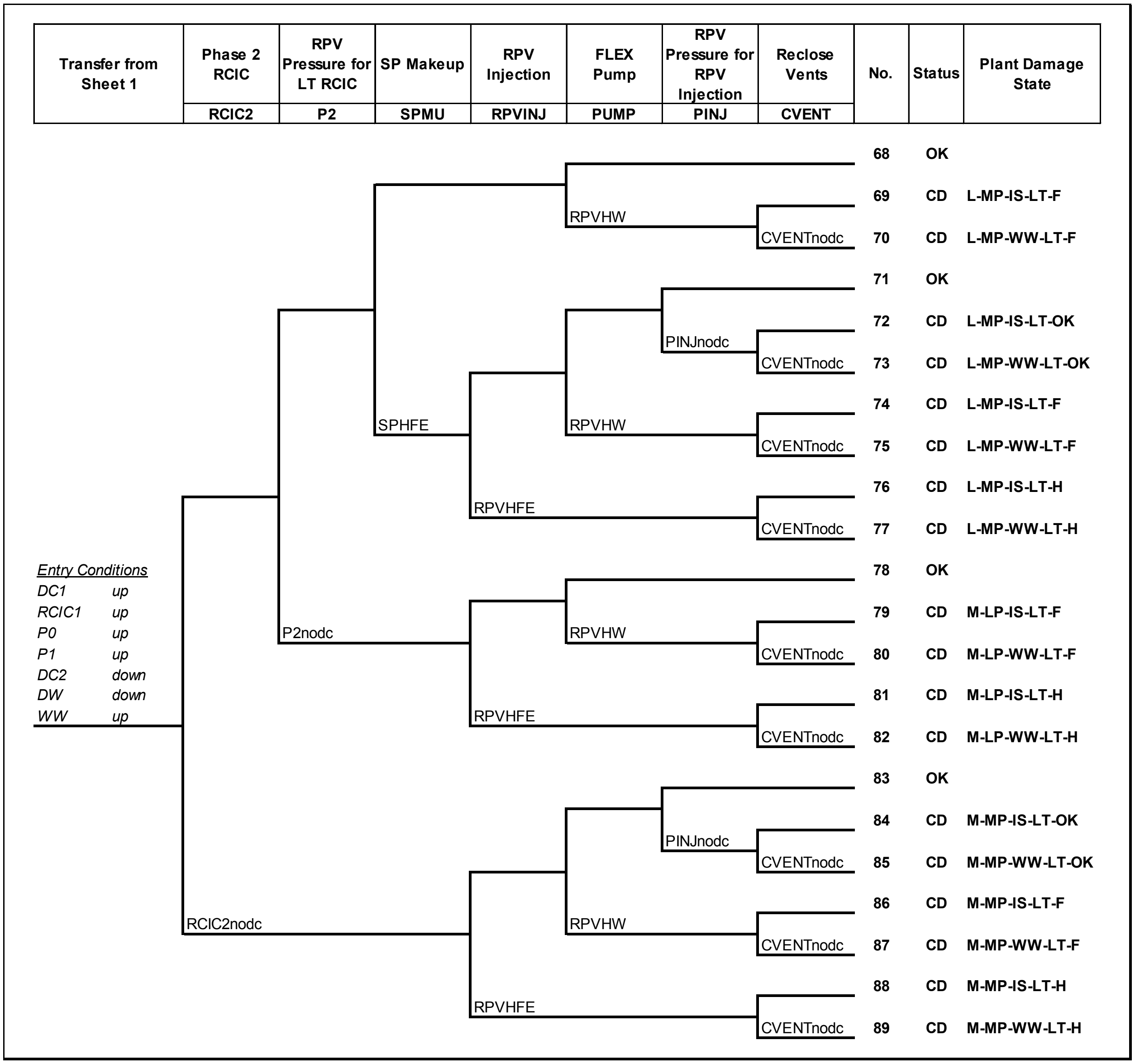


Figure A-22 CDET for DWF venting strategies, Sheet 5, Sequences 68 to 89

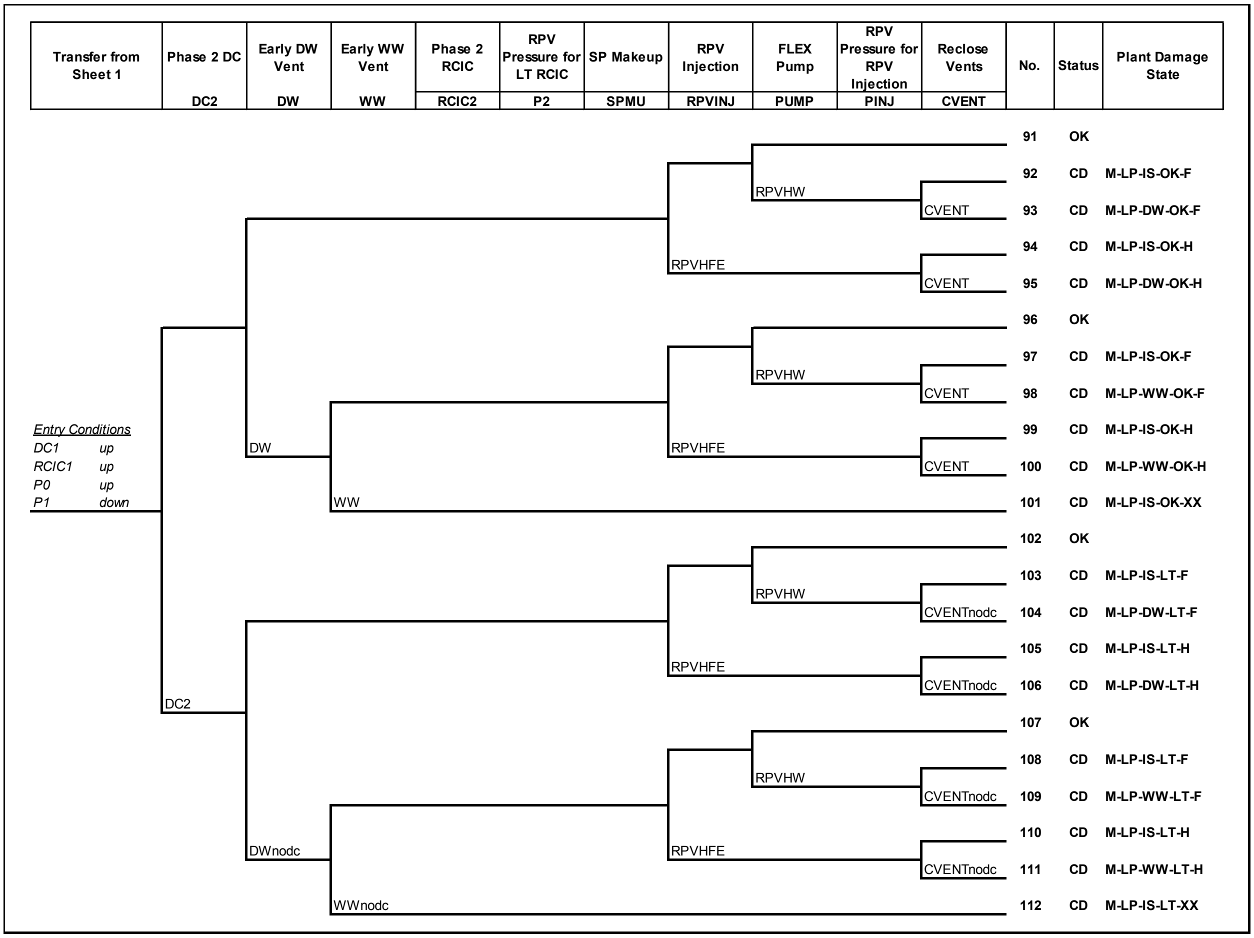


Figure A-23 CDET for DWF venting strategies, Sheet 6, Sequences 91 to 112

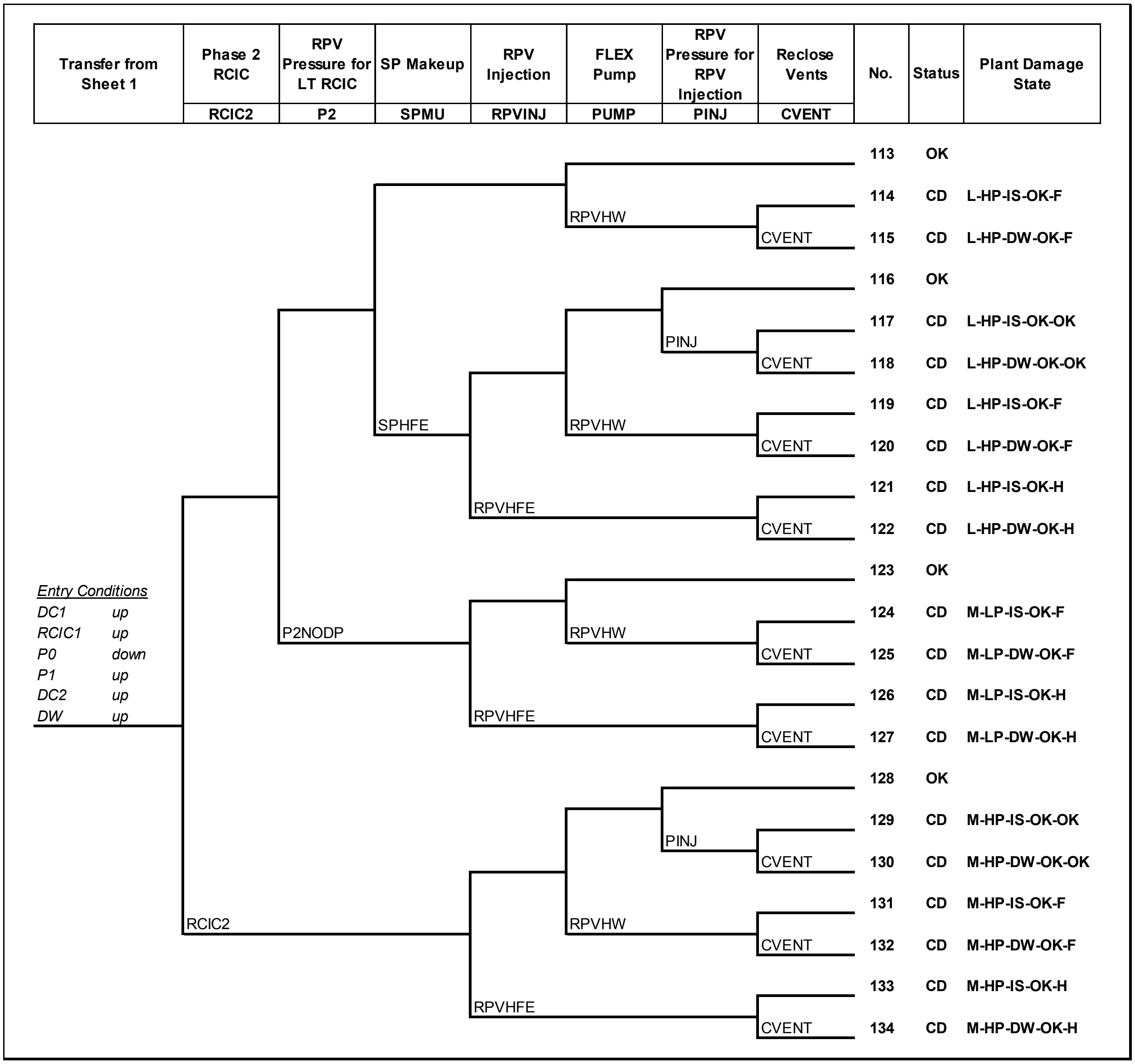


Figure A-24 CDET for DWF venting strategies, Sheet 7, Sequences 113 to 134

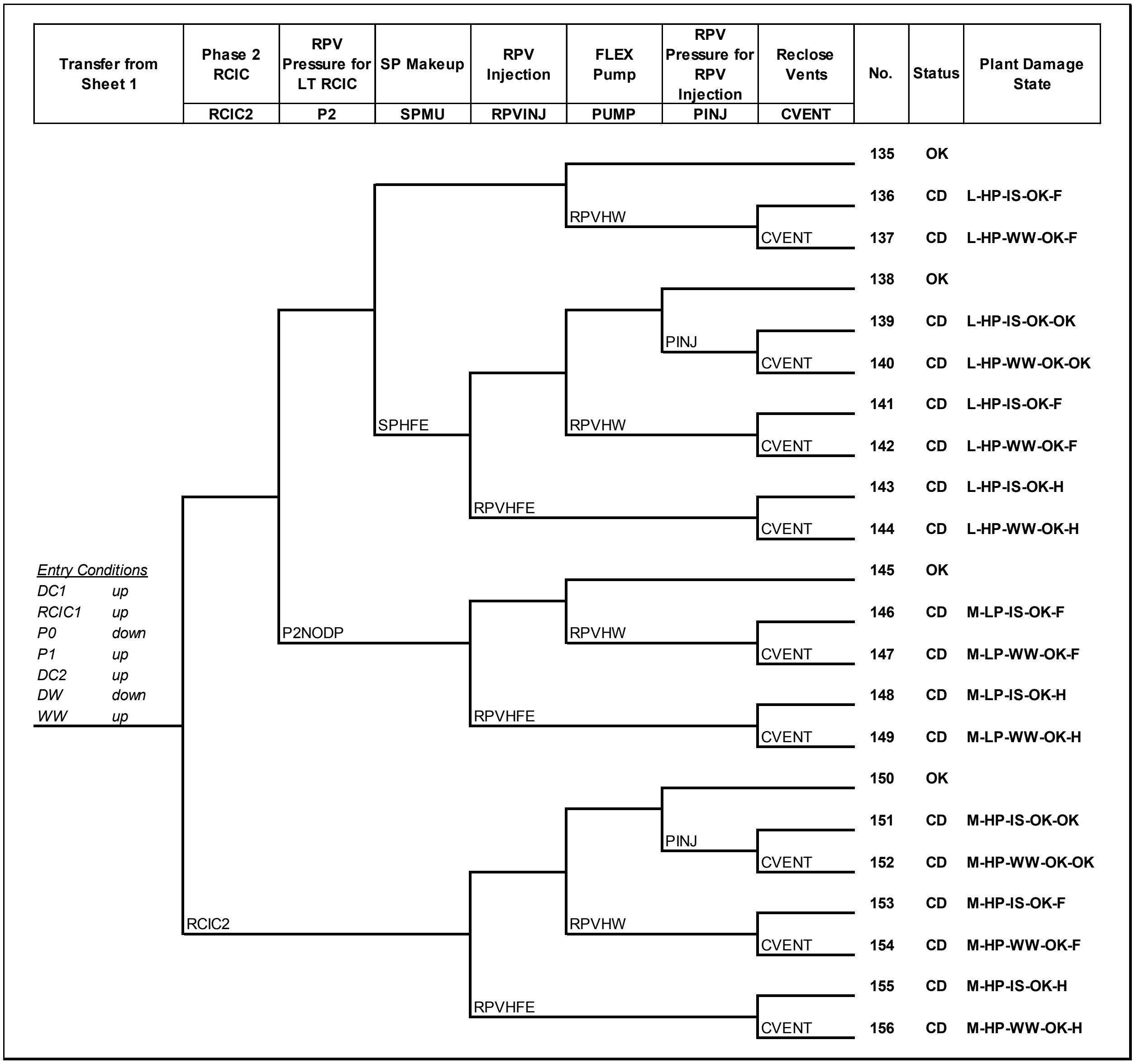


Figure A-25 CDET for DWF venting strategies, Sheet 8, Sequences 135 to 156

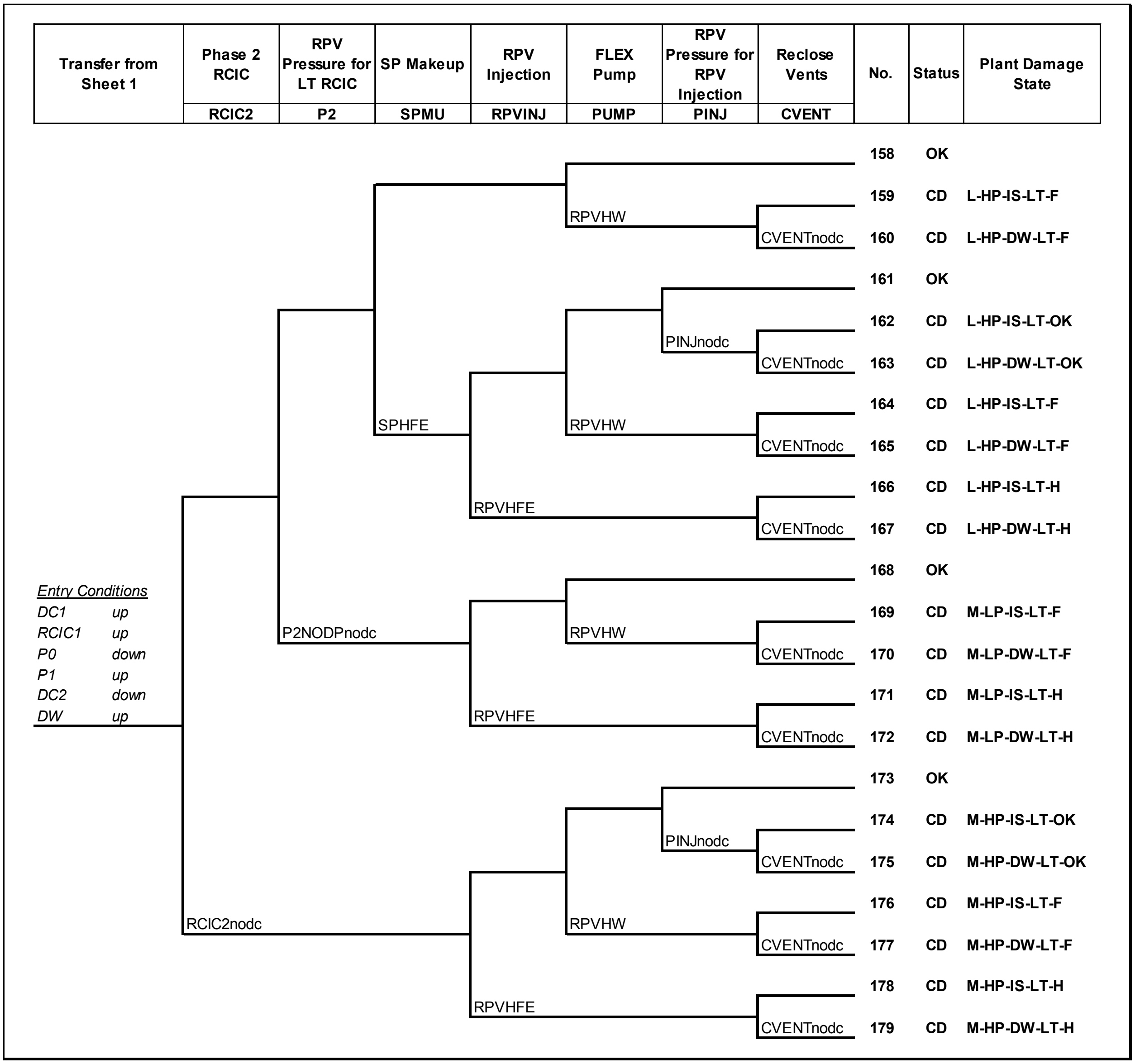


Figure A-26 CDET for DWF venting strategies, Sheet 9, Sequences 158 to 179

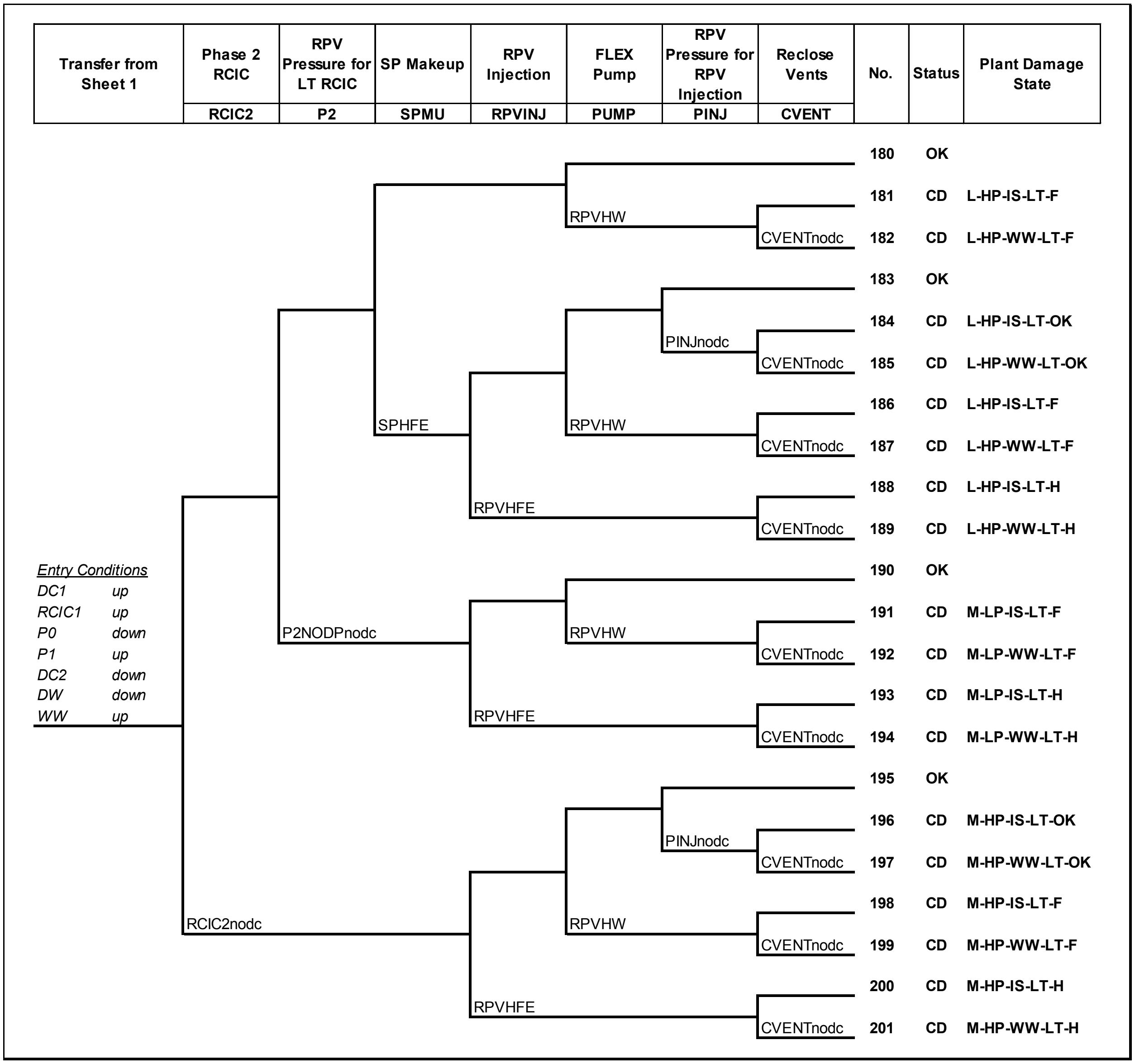


Figure A-27 CDET for DWF venting strategies, Sheet 10, Sequences 180 to 201

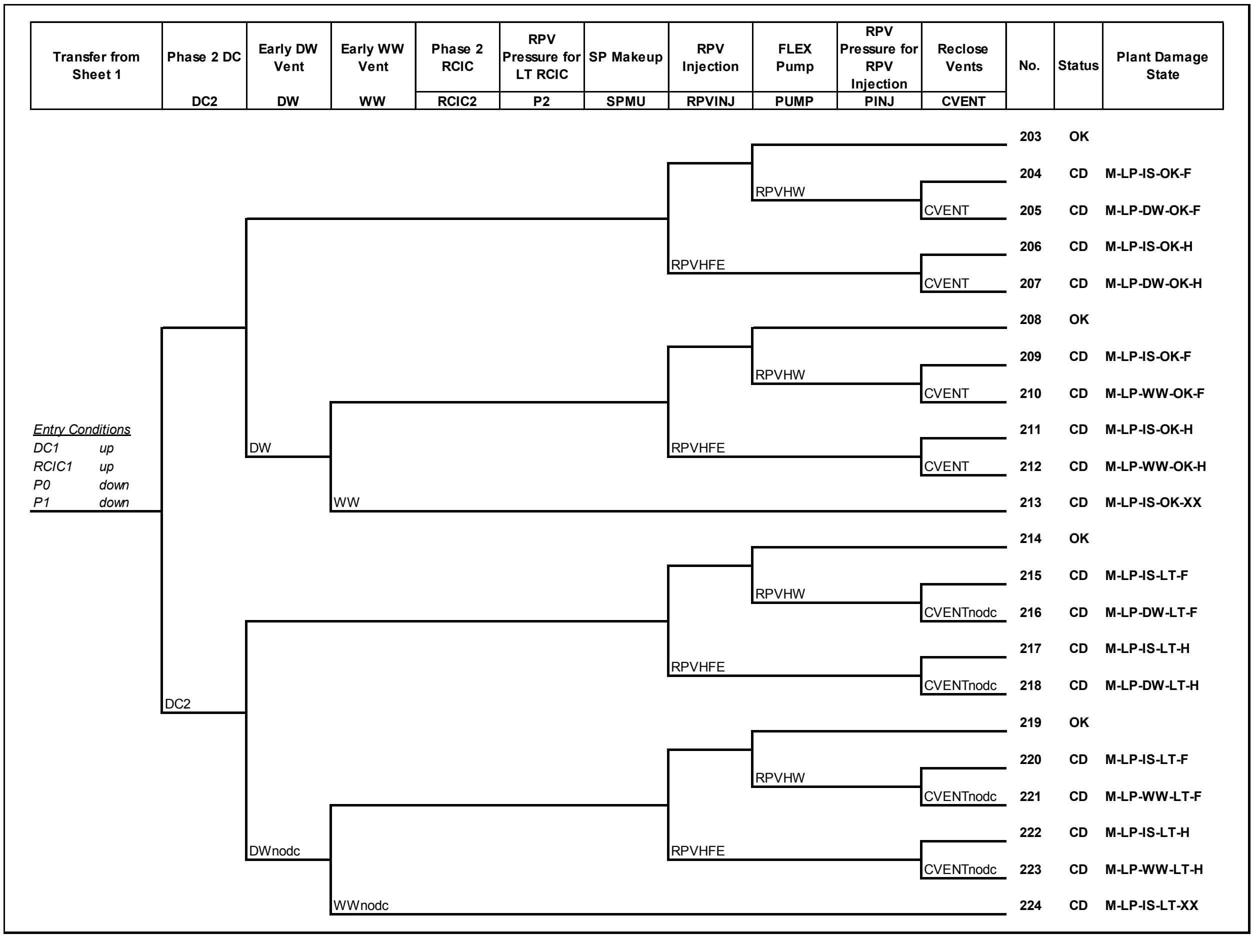


Figure A-28 CDET for DWF venting strategies, Sheet 11, Sequences 203 to 224

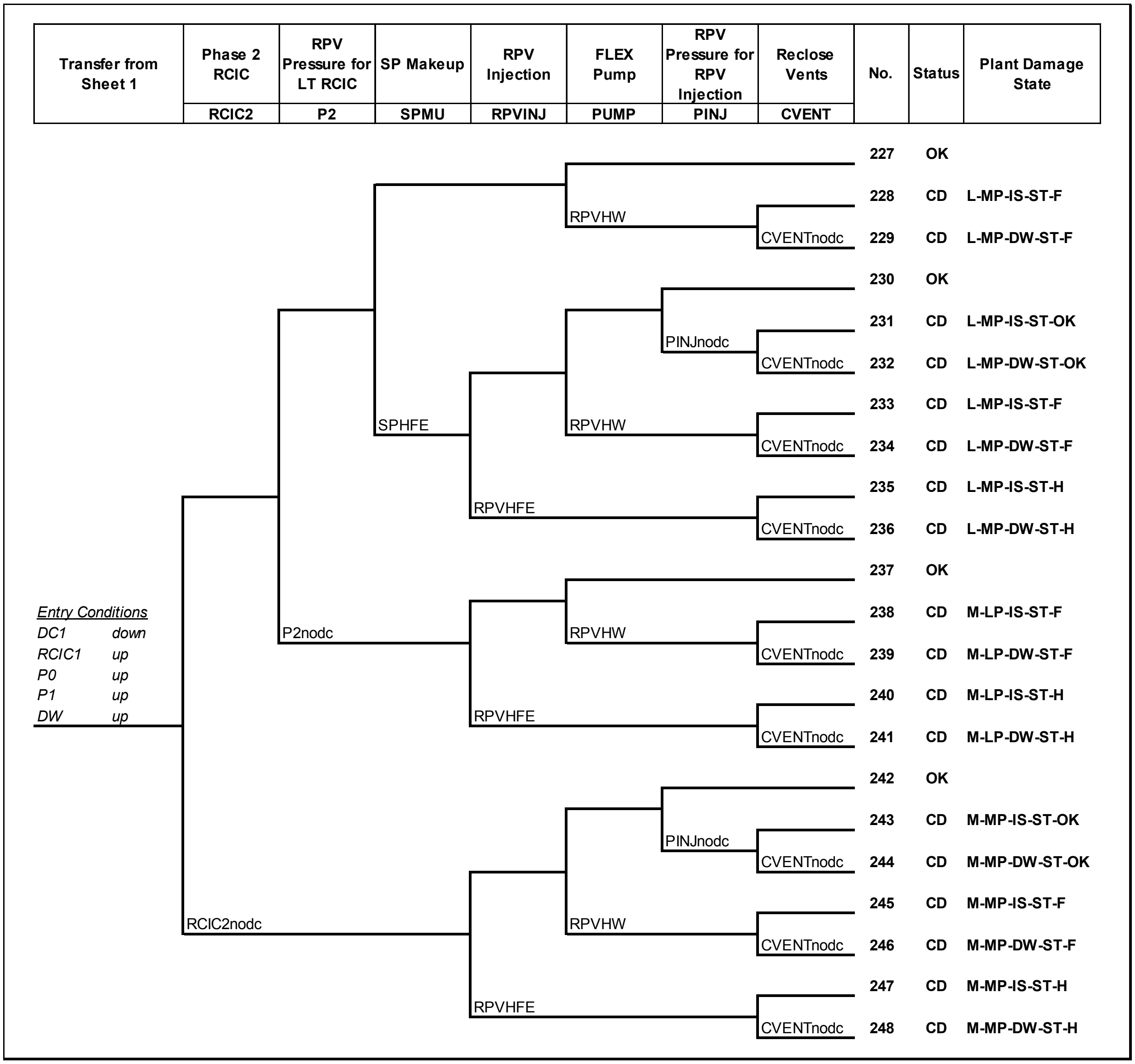


Figure A-29 CDET for DWF venting strategies, Sheet 12, Sequences 227 to 248

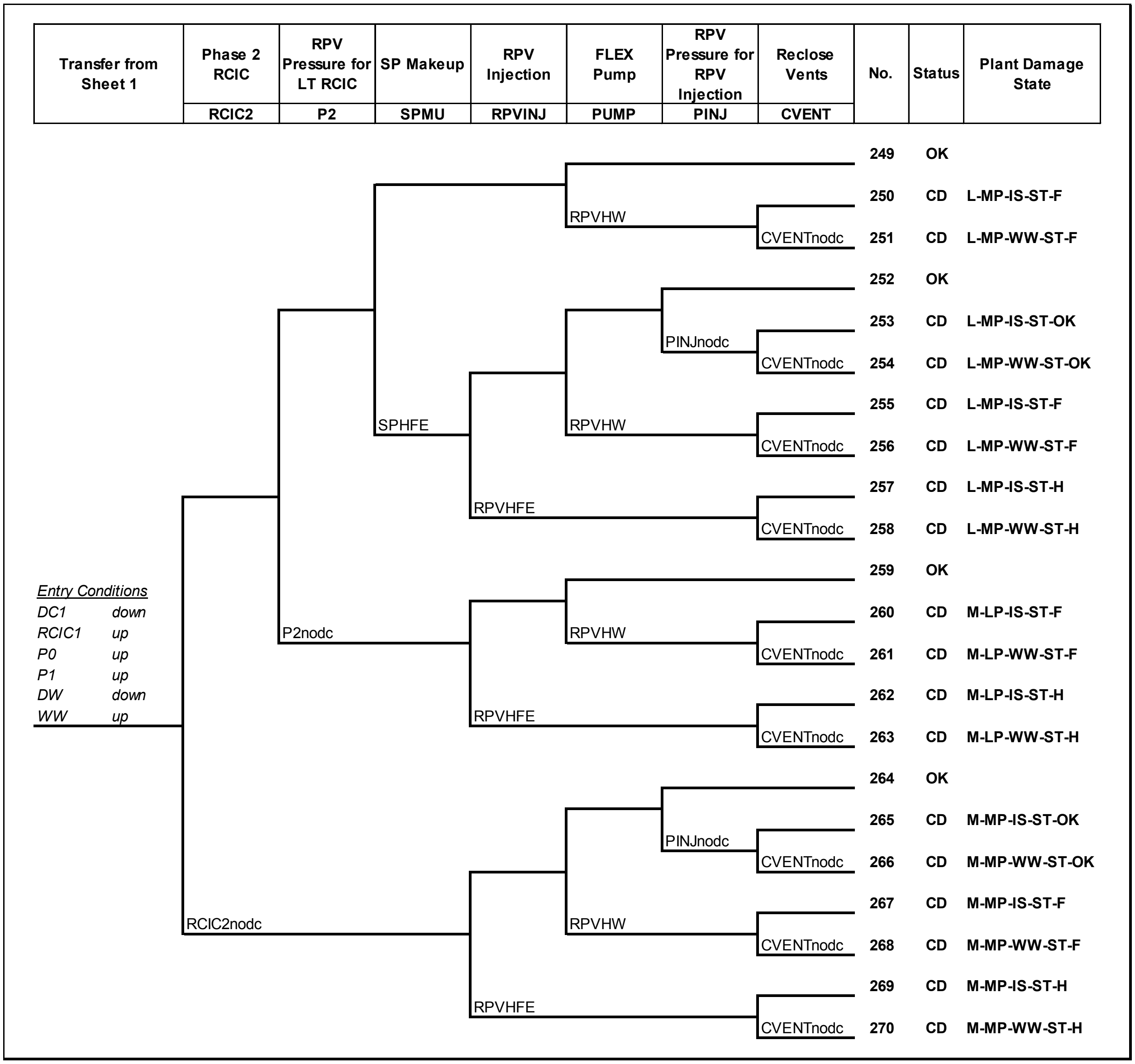


Figure A-30 CDET for DWF venting strategies, Sheet 13, Sequences 249 to 270

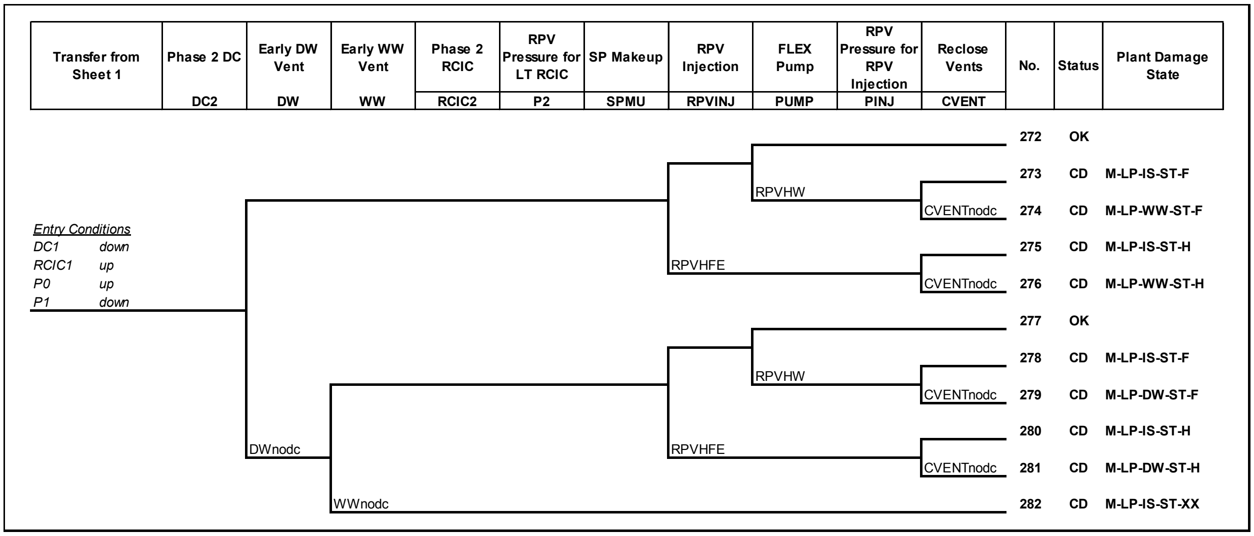


Figure A-31 CDET for DWF venting strategies, Sheet 14, Sequences 272 to 282

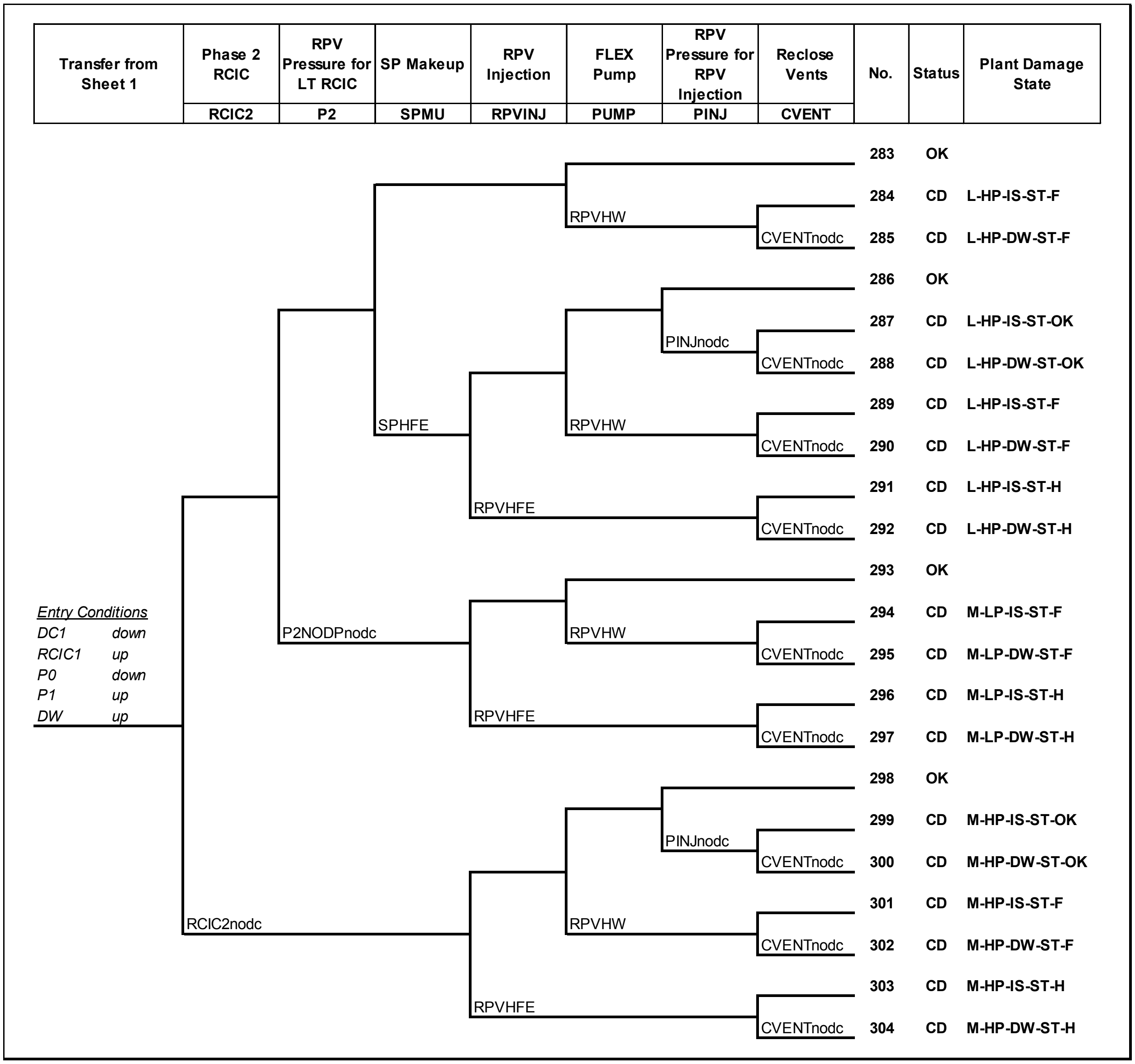


Figure A-32 CDET for DWF venting strategies, Sheet 15, Sequences 283 to 304

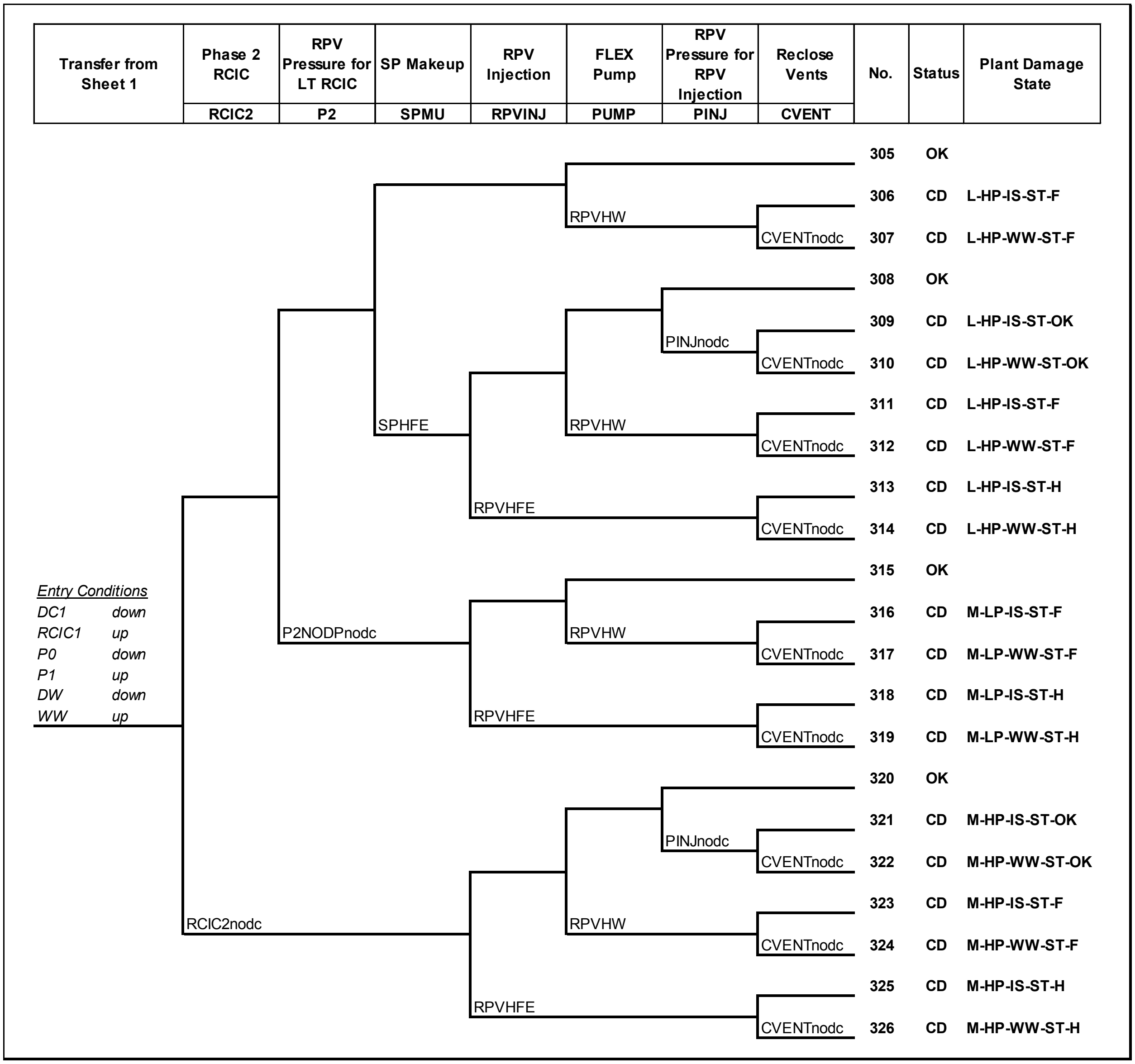


Figure A-33 CDET for DWF venting strategies, Sheet 16, Sequences 305 to 326

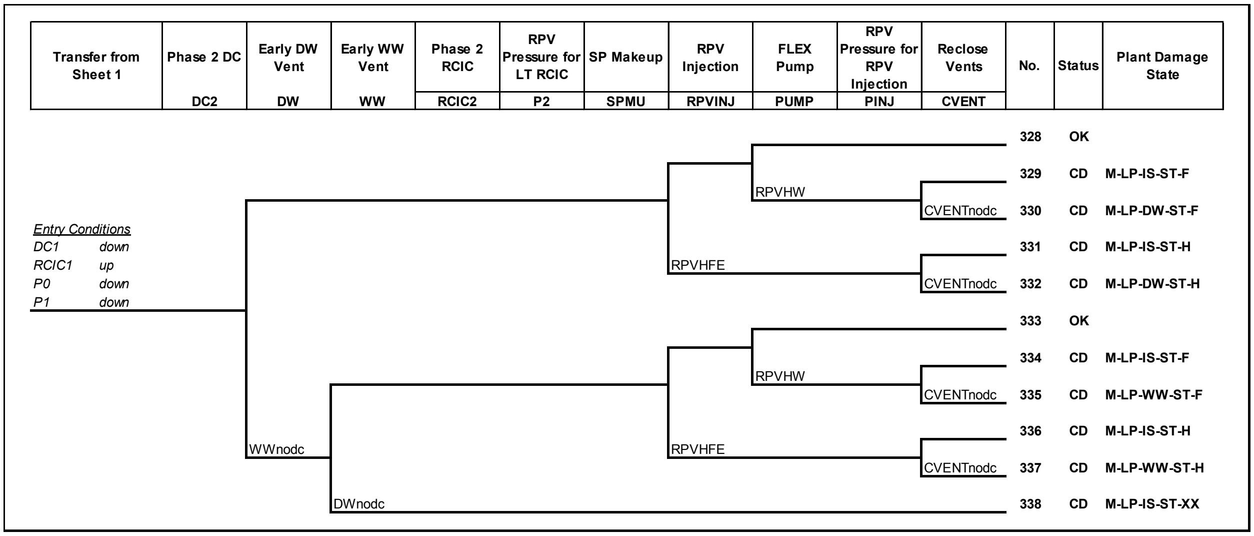


Figure A-34 CDET for DWF venting strategies, Sheet 17, Sequences 328 to 338

APPENDIX B

CORE DAMAGE EVENT TREE SUPPORTING INFORMATION

Table B-1 Core-damage event tree top logic

Top Event	Description	Boolean Expression	Probability		
			WWF	DWF passive	DWF manual
CVENT	failure to reclose containment vents	= HFE-CVENT	0.1	1	0.1
CVENTnodc	failure to reclose containment vents (no DC power)	= HFE-CVENT-NODC	0.3	1	0.3
DC1	DC power fails short-term	= BAT-ST	1.2E-06		
DC2	DC power fails long-term	= HFE-GEN + GEN-FTS + GEN-FTR	3.9E-01		
DW	failure to open drywell vent	= HFE-DW	0.1	0.01	0.1
DWnodc	failure to open drywell vent (no DC power)	= HFE-DW-NODC	0.3	0.01	0.3
P0DP	failure to depressurize to 200-400 psia	= HFE-DEPRESS	0.1		
P0DPnodc	failure to depressurize to 200-400 psia (no DC power)	= HFE-DEPRESS-NODC	0.3		
P0MSCL	failure to depressurize at minimum steam cooling level	= HFE-MSCL	0.1		
P0MSCLnodc	failure to depressurize at minimum steam cooling level (no DC power)	= HFE-MSCL-NODC	0.3		
P1	RPV pressure inadequate for RCIC short-term	= SORV1-DP + HFE-INVDP	1.3E-01		
P1nodc	RPV pressure inadequate for RCIC short-term (no DC power)	= SORV1-DP + HFE-INVDP-NODC	3.2E-01		
P1NODP	RPV pressure inadequate to RCIC short-term (no depress)	= SORV1-NODP	2.9E-01		
P1NODPnodc	RPV pressure inadequate to RCIC short-term (no depress or DC power)	= SORV1-NODP	2.9E-01		
P2	RPV pressure inadequate for RCIC long-term	= SORV2-DP + HFE-INVDP	1.4E-01		
P2nodc	RPV pressure inadequate for RCIC long-term (no DC power)	= SORV2-DP + HFE-INVDP-NODC	3.3E-01		
P2NODP	RPV pressure inadequate to RCIC long-term (no depress)	= SORV2-NODP	8.2E-02		
P2NODPnodc	RPV pressure inadequate to RCIC long-term (no depress or DC power)	= SORV2-NODP	8.2E-02		
PINJ	failure to depressurize RPV for FLEX injection	= HFE-PINJ	0.1		
PINJnodc	failure to depressurize RPV for FLEX injection (no DC power)	= HFE-PINJ-NODC	0.3		

Top Event	Description	Boolean Expression	Probability		
			WWF	DWF passive	DWF manual
RCIC1	RCIC fails short-term	= RCIC-FTS + RCIC-1	8.9E-03		
RCIC1nodc	RCIC fails short-term (no DC power)	= RCIC-FTS + RCIC-1 + HFE-RCIC-BLACKSTART	3.1E-01		
RCIC2	RCIC fails long-term	= RCIC-2	6.5E-04		
RCIC2nodc	RCIC fails long-term (no DC power)	= RCIC-2 +HFE- RCIC-BLACKRUN	3.0E-01		
RPVHFE	operator fails to align RPV injection using FLEX	= HFE-RPV	0.3		
RPVHW	FLEX pump fails during RPV injection	= FLEX-FTS + FLEXP-FTR	1.50E-01		
SPHFE	operator fails to align SP makeup	= HFE-SP	0.3		
SPHW	FLEX pump fails while supplying SP makeup	= FLEXP-FTS + FLEXP-FTR	1.5E-01		
WW	failure to open wetwell vent	= HFE-WW	0.1		
WWnodc	failure to open wetwell vent (no DC power)	= HFE-WW-NODC	0.3		

Table B-2 Core-damage event tree human failure events

Human Failure Event	Description	Probability		
		WWF	DWF passive	DWF manual
HFE-CVENT	operator fails to reclose the containment vents upon CD	0.1	1	0.1
HFE-CVENT-NODC	operator fails to reclose the containment vents upon CD (no DC power)	0.3	1	0.3
HFE-DEPRESS	operator fails to depressurize to 200-400 psia	0.1		
HFE-DEPRESS-NODC	operator fails to depressurize to 200-400 psia (no DC power)	0.3		
HFE-DW	operator fails to open the drywell vent	0.1	0.01	0.1
HFE-DW-NODC	operator fails to open the drywell vent (no DC power)	0.3	0.01	0.3
HFE-GEN	operator fails to align portable generator	0.3		
HFE-INVDP	operator inadvertantly depressurizes RPV below RCIC	0.1		
HFE-INVDP-NODC	operator inadvertantly depressurizes RPV below RCIC (no DC power)	0.3		
HFE-MSCL	operator fails to depressurize at minimum steam cooling level	0.1		
HFE-MSCL-NODC	operator fails to depressurize at minimum steam cooling level (no DC power)	0.3		
HFE-PINJ	operator fails to depressurize for FLEX RPV injection	0.1		
HFE-PINJ-NODC	operator fails to depressurize for FLEX RPV injection (no DC power)	0.3		
HFE-RCIC-BLACKRUN	operator fails to blackrun RCIC during Phase 2	0.3		
HFE-RCIC-BLACKSTART	operator fails to blackstart and blackrun RCIC during Phase 1	0.3		

Human Failure Event	Description	Probability		
		WWF	DWF passive	DWF manual
HFE-RPV	operator fails to align FLEX pump for RPV injection	0.3		
HFE-SP	operator fails to align FLEX pump for SP makeup	0.3		
HFE-WW	operator fails to open the wetwell vent	0.1		
HFE-WW-NODC	operator fails to open the wetwell vent (no DC power)	0.3		

Table B-3 Core-damage event tree hardware-related basic events

Name	Description	Probability	Calculation Details
BAT-ST	station batteries fail short-term	2.3E-06	P=1-exp(-λT) λ = 5.9E-07 per hour Basis: SPAR 2010 Parameter Estimation Update [X] T = 2 hours Basis: Phase 1 mission time
FLEXP-FTR	FLEX pump fails to run	1.4E-01	P=1-exp(-λ_1 – λ_2T) λ_1 = 1.3E-03 /h = failure rate < 1 h Basis: SPAR 2010 Parameter Estimation Update λ_2 = 2.3E-03 /h = failure rate > 1 h Basis: SPAR 2010 Parameter Estimation Update T = 67 h Basis: Phase 2 mission time – 1 h
FLEXP-FTS	FLEX pump fails to start	5.1E-03	Failure probability per demand Basis: SPAR 2010 Parameter Estimation Update
GEN-FTR	portable generator fails to run	8.5E-02	P=1-exp(-λT) λ = 1.3E-03 per hour Basis: SPAR 2010 Parameter Estimation Update T = 68 hours Basis: Phase 2 mission time
GEN-FTS	portable generator fails to start	4.3E-02	Failure probability per demand Basis: SPAR 2010 Parameter Estimation Update
RCIC-1	RCIC pump fails to run in Phase 1	3.7E-05	P=1-exp(-λT) λ = 9.3E-06 per hour Basis: SPAR 2010 Parameter Estimation Update T = 2 hours Basis: Phase 1 mission time
RCIC-2	RCIC pump fails to run in Phase 2	6.4E-04	P=1-exp(-λT) λ = 9.3E-06 per hour Basis: SPAR 2010 Parameter Estimation Update T = 68 hours Basis: Phase 2 mission time
RCIC-FTS	RCIC fails to start	8.9E-03	Failure probability per demand Basis: SPAR 2010 Parameter Estimation Update
SORV1-DP	stuck-open SRV given Phase 1 depressurization and cooldown	3.4E-02	P=1-(1-p_D)^n p_D = 8.6E-04 per demand Basis: SPAR 2010 Parameter Estimation Update n = 40 demands Basis: thermal-hydraulic calculations
SORV1-NODP	stuck-open SRV given no Phase 1 depressurization and cooldown	2.9E-01	P=1-(1-p_D)^n p_D = 8.6E-04 per demand Basis: SPAR 2010 Parameter Estimation Update n = 400 demands Basis: thermal-hydraulic calculations

Name	Description	Probability	Calculation Details
SORV2-DP	stuck-open SRV in Phase 2 given Phase 1 depressurization and cooldown	4.2E-02	$P=1-(1-p_D)^n$ p_D = 8.6E-04 per demand Basis: SPAR 2010 Parameter Estimation Update n = 50 demands Basis: thermal-hydraulic calculations
SORV2-NODP	stuck-open SRV in Phase 2 given no Phase 1 depressurization and cooldown	8.2E-02	$P=1-(1-p_D)^n$ p_D = 8.6E-04 per demand Basis: SPAR 2010 Parameter Estimation Update n = 100 demands Basis: thermal-hydraulic calculations

APPENDIX C

ACCIDENT PROGRESSION EVENT TREES

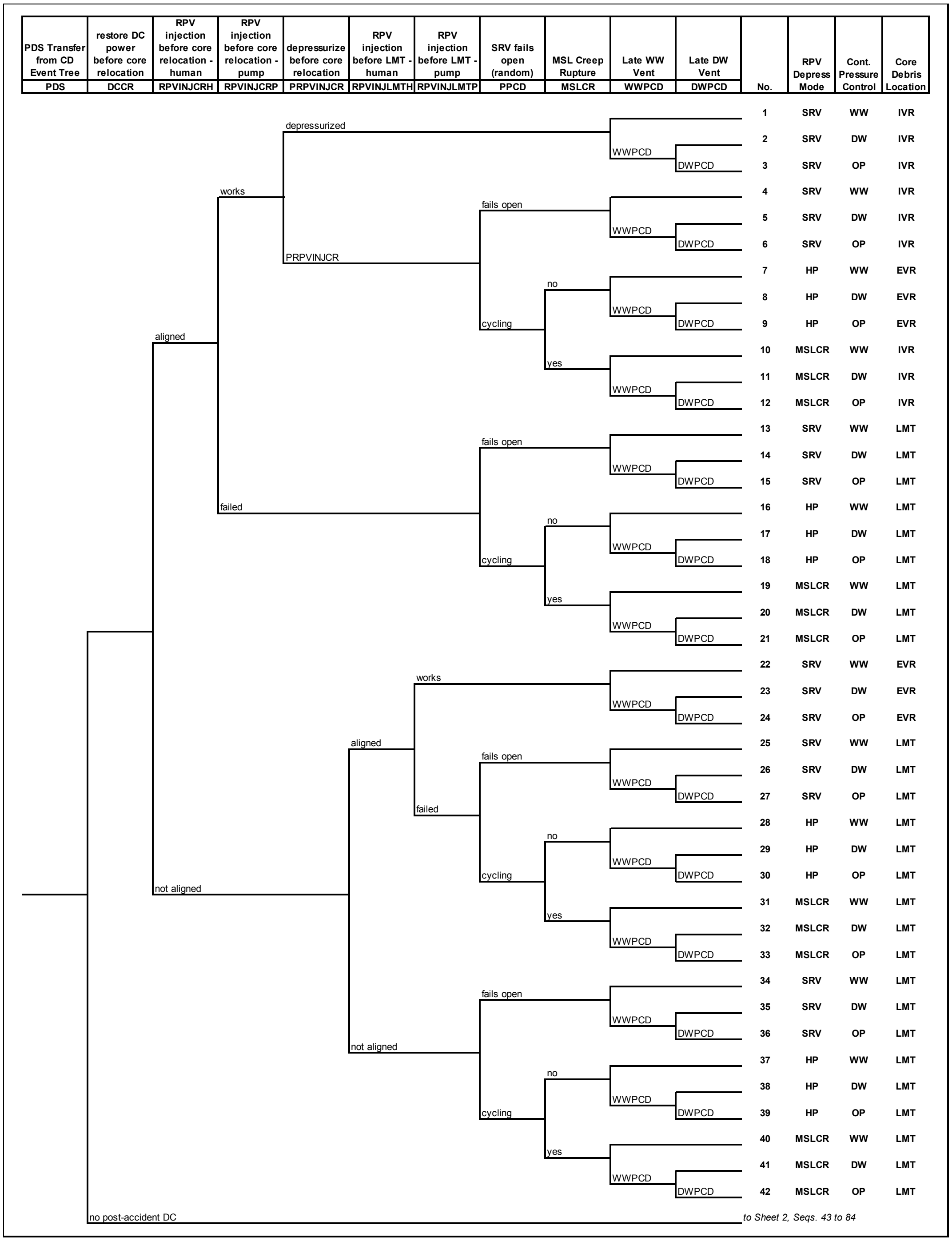


Figure C-1 APET for RPV injection and WWF venting strategies (APETs 1 and 2), Sheet 1, Sequences 1 to 42

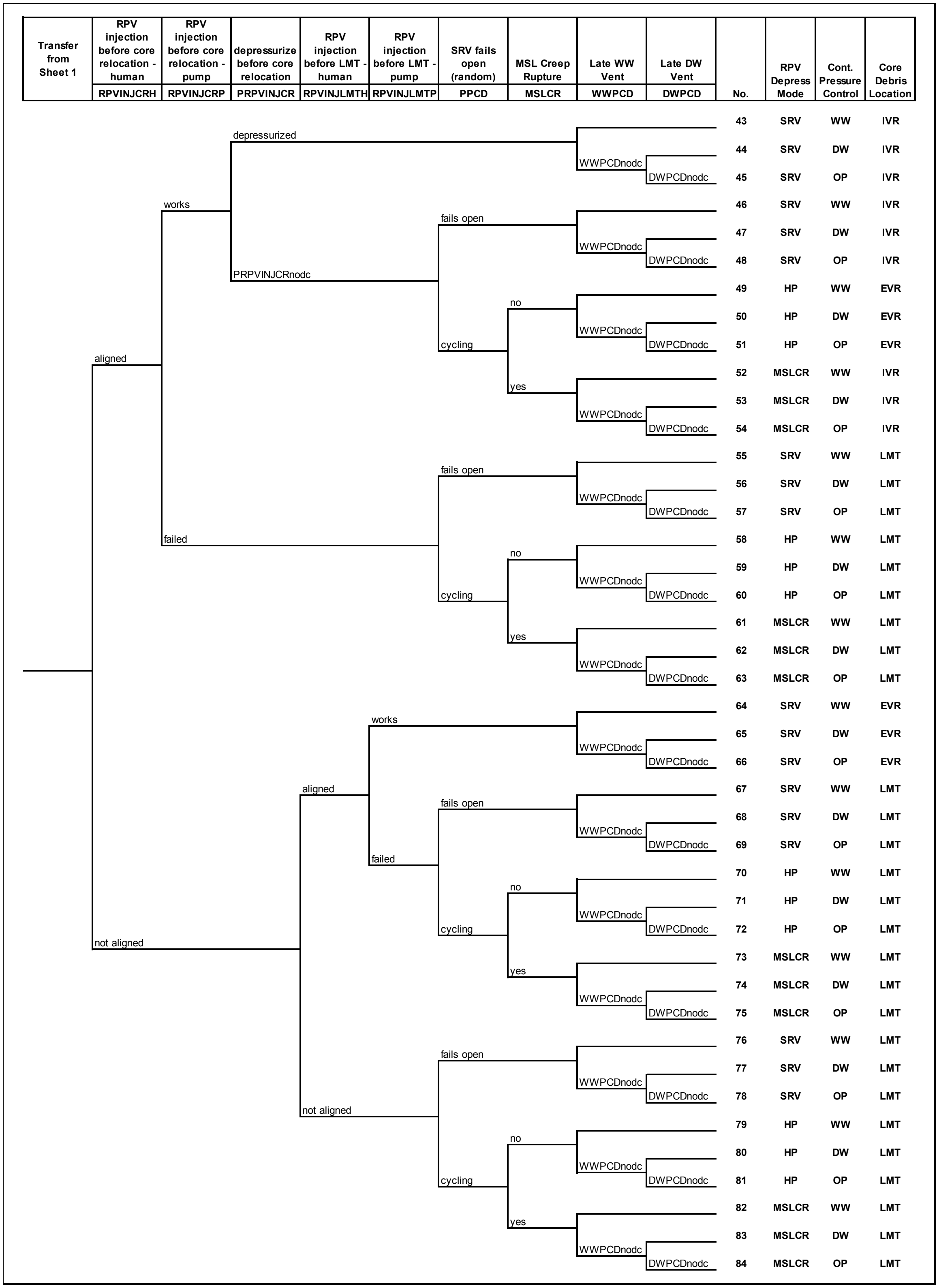


Figure C-2 APET for RPV injection and WWF venting strategies (APETs 1 and 2), Sheet 2, Sequences 43 to 84

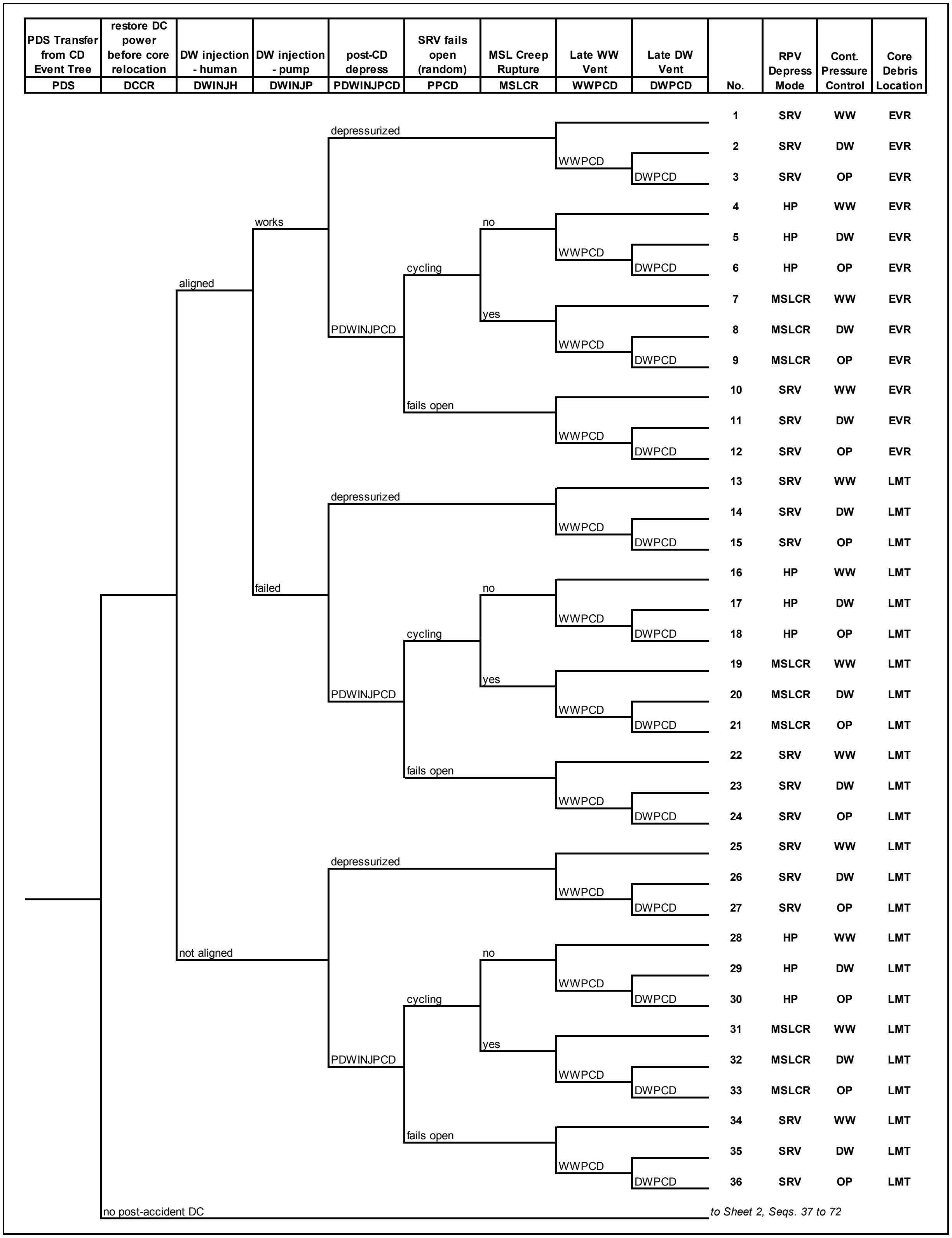
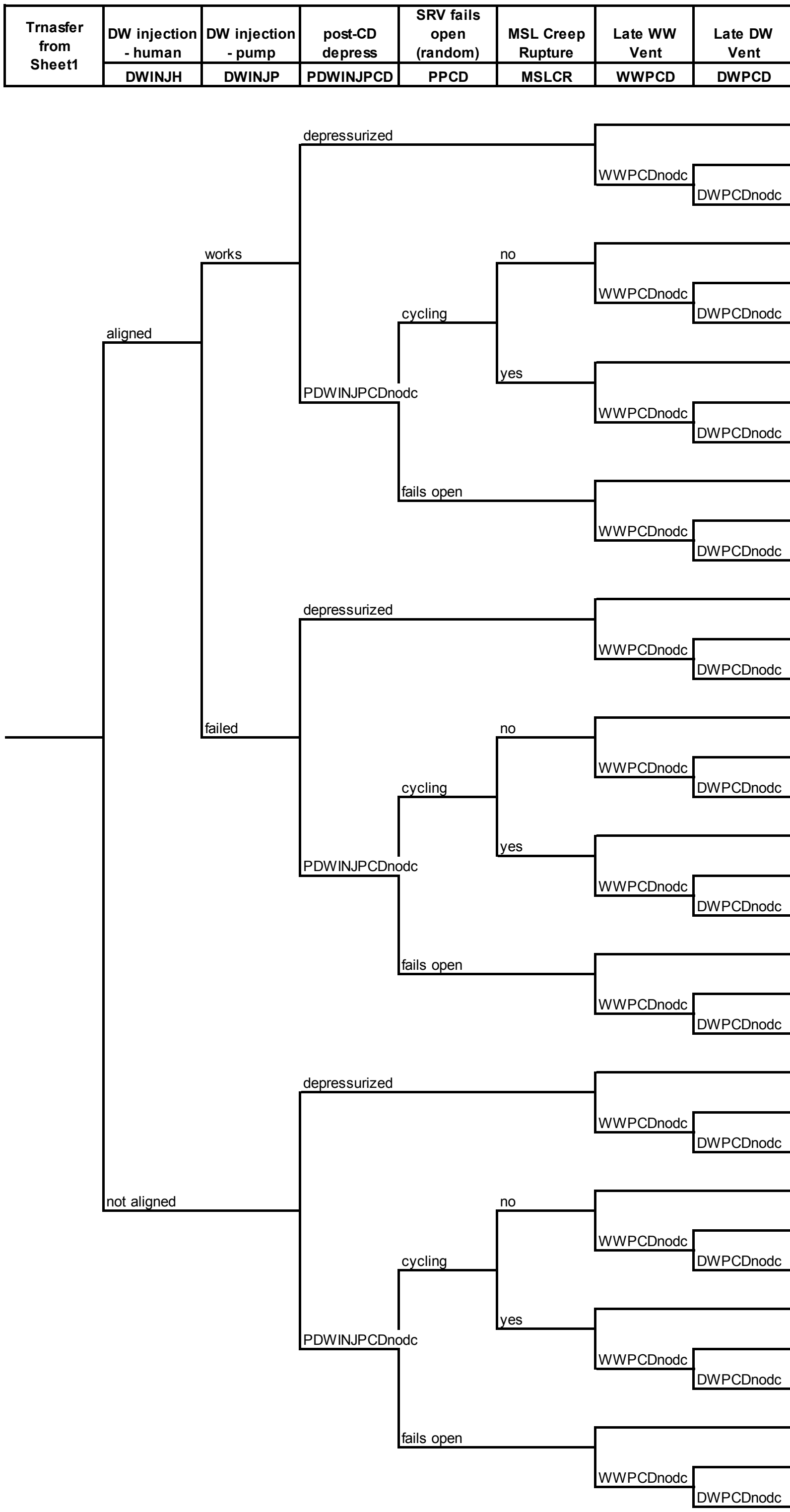
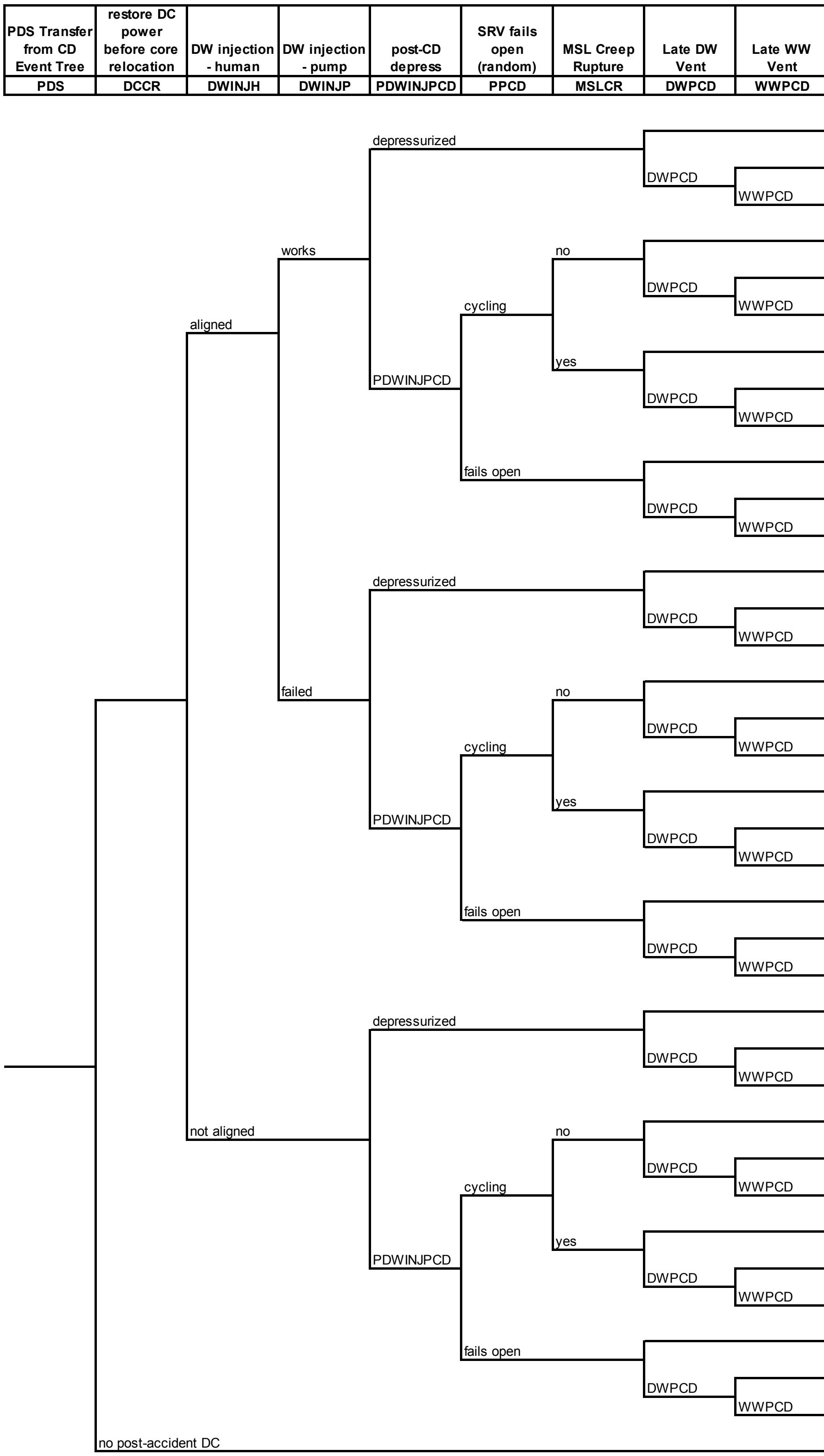


Figure C-3 APET for DW Injection and WWF Venting Strategies (APET 3), Sheet 1, Sequences 1 to 36



No.	RPV Depress Mode	Cont. Pressure Control	Core Debris Location
37	SRV	WW	EVR
38	SRV	DW	EVR
39	SRV	OP	EVR
40	HP	WW	EVR
41	HP	DW	EVR
42	HP	OP	EVR
43	MSLCR	WW	EVR
44	MSLCR	DW	EVR
45	MSLCR	OP	EVR
46	SRV	WW	EVR
47	SRV	DW	EVR
48	SRV	OP	EVR
49	SRV	WW	LMT
50	SRV	DW	LMT
51	SRV	OP	LMT
52	HP	WW	LMT
53	HP	DW	LMT
54	HP	OP	LMT
55	MSLCR	WW	LMT
56	MSLCR	DW	LMT
57	MSLCR	OP	LMT
58	SRV	WW	LMT
59	SRV	DW	LMT
60	SRV	OP	LMT
61	SRV	WW	LMT
62	SRV	DW	LMT
63	SRV	OP	LMT
64	HP	WW	LMT
65	HP	DW	LMT
66	HP	OP	LMT
67	MSLCR	WW	LMT
68	MSLCR	DW	LMT
69	MSLCR	OP	LMT
70	SRV	WW	LMT
71	SRV	DW	LMT
72	SRV	OP	LMT

Figure C-4 APET for DW Injection and WWF Venting Strategies (APET 3), Sheet 2, Sequences 37 to 72



No.	RPV Depress Mode	Cont. Pressure Control	Core Debris Location
1	SRV	DW	EVR
2	SRV	WW	EVR
3	SRV	OP	EVR
4	HP	DW	EVR
5	HP	WW	EVR
6	HP	OP	EVR
7	MSLCR	DW	EVR
8	MSLCR	WW	EVR
9	MSLCR	OP	EVR
10	SRV	DW	EVR
11	SRV	WW	EVR
12	SRV	OP	EVR
13	SRV	DW	LMT
14	SRV	WW	LMT
15	SRV	OP	LMT
16	HP	DW	LMT
17	HP	WW	LMT
18	HP	OP	LMT
19	MSLCR	DW	LMT
20	MSLCR	WW	LMT
21	MSLCR	OP	LMT
22	SRV	DW	LMT
23	SRV	WW	LMT
24	SRV	OP	LMT
25	SRV	DW	LMT
26	SRV	WW	LMT
27	SRV	OP	LMT
28	HP	DW	LMT
29	HP	WW	LMT
30	HP	OP	LMT
31	MSLCR	DW	LMT
32	MSLCR	WW	LMT
33	MSLCR	OP	LMT
34	SRV	DW	LMT
35	SRV	WW	LMT
36	SRV	OP	LMT

Figure C-5 APET for DW Injection and DWF Venting Strategies (APETs 4 to 6), Sheet 1, Sequences 1 to 36

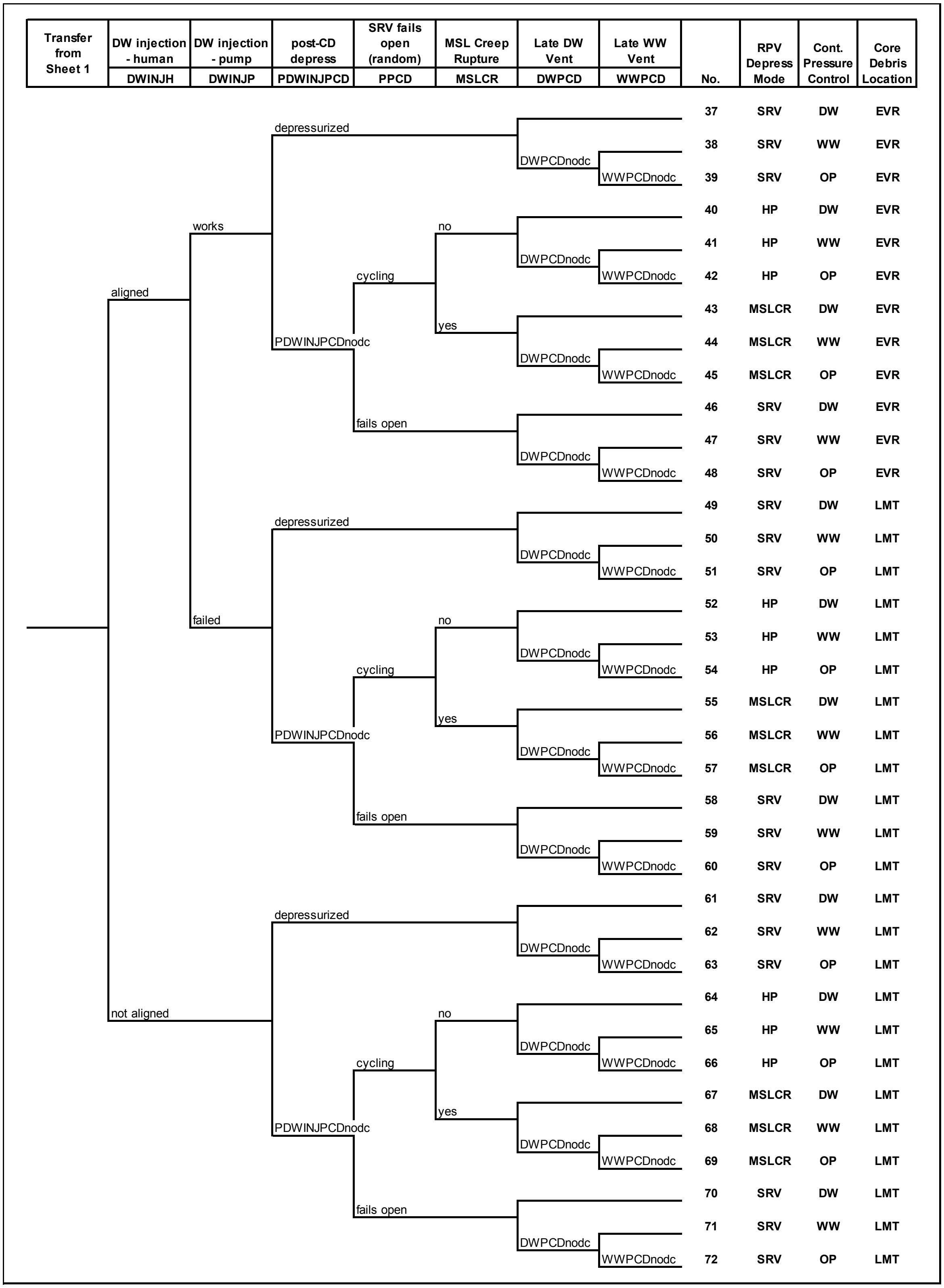


Figure C-6 APET for DW Injection and DWF Venting Strategies (APETs 4 to 6), Sheet 2, Sequences 37 to 72 accident progression event tree supporting information

APPENDIX D

ACCIDENT PROGRESSION EVENT TREE SUPPORTING INFORMATION

Table D-1 provides the down-branch probabilities used to quantify each accident progression event tree (APET). The six APETs (APET-1 through APET-6) were developed using a common set of APET headers (listed in the first column of Table C-1), although a specific APET may not use some of the APET common headers. For example, APET header RPVINJCRP (align the portable pump to provide post-accident water injection via the RPV prior to core relocation) only appears in APET-1 and APET-2. Similarly, APET header DWINJP (align the portable pump to provide post-accident water injection via the drywell) only appears in APET-3, APET-4, APET-5, and APET-6.

The APET down-branch probabilities depend on which plant damage state (PDS) is input to the APET. The second column of Table C-1 indicates which PDS attribute controls the down-branch probability, and the fourth column indicates the down-branch probability used for specific values of the controlling PDS attribute. For example, if the PDS E-LP-IS-ST-XX is input to APET-2, then the following down-branch probabilities are used to quantify APET-2:

Pr{DCRR}	= 1
Pr{DWPCD}	= 0.1
Pr{DWPCDnodc}	= 0.3
Pr{MSLCR}	= 0
Pr{PPCD}	= 0
Pr{PRPVINJCR}	= 0
Pr{PRPVINJCRnodc}	= 0
Pr{RPVINJCRH}	= 0.3
Pr{RPVINJCRP}	= 0.147
Pr{RPVINJLMTH}	= 0.3
Pr{RPVINJLMTP}	= 0.147
Pr{WWPCD}	= 0.1
Pr{WWPCDnodc}	= 0.3

Table D-1 Accident progression event tree down-branch probabilities.

APET Heading	PDS Attribute	APET Down-Branch Identifier	PDS Attribute Value	APET-1	APET-2	APET-3	APET-4	APET-5	APET-6
DCCR	DC status	DCRR	OK	0	0	0	0	0	0
			ST	1	1	1	1	1	1
			LT	0.3	0.3	0.3	0.3	0.3	0.3
			XX	0.3	0.3	0.3	0.3	0.3	0.3
DWINJH	pump status	DWINJH	OK	n/a	n/a	0	0	0	0
			F			0	0	0	0
			H			0.3	0.3	0.3	0.3
			XX			0.3	0.3	0.3	0.3
DWINJP	pump status	RDWINJP	OK	n/a	n/a	1	1	1	1
			F			1	1	1	1
			H			0.147	0.147	0.147	0.147
			XX			0.147	0.147	0.147	0.147
DWPCD	vent status	DWPCD	WW	n/a	n/a	n/a	0	0	0
			DW	0	0	0	1	1	1
			IS	0.1	0.1	0.1	0.01	0.01	0.1
		DWPCDnodc	WW	n/a	n/a	n/a	0	0	0
			DW	0	0	0	1	1	1
			IS	0.3	0.3	0.3	0.01	0.01	0.3
MSLCR	RPV pressure	MSLCR	HP	0.5	0.5	0.5	0.5	0.5	0.5
			MP	0	0	0	0	0	0
			LP	0	0	0	0	0	0
PDWINJPCD	RPV pressure	PDWINJPCD	HP	n/a	n/a	0.1	0.1	0.1	0.1
			MP			0.1	0.1	0.1	0.1
			LP			0	0	0	0
		PDWINJPCDnodc	HP			0.3	0.3	0.3	0.3
			MP			0.3	0.3	0.3	0.3
			LP			0	0	0	0
PPCD	RPV pressure	PPCD	HP	8.21E-02	8.21E-02	8.21E-02	8.21E-02	8.21E-02	8.21E-02
			MP	8.21E-02	8.21E-02	8.21E-02	8.21E-02	8.21E-02	8.21E-02
			LP	0	0	0	0	0	0
PRPVINJCR	RPV pressure	PRPVINJCR	HP	0.1	0.1	n/a	n/a	n/a	n/a
			MP	0.1	0.1				
			LP	0	0				
		PRPVINJCRnodc	HP	0.3	0.3				
			MP	0.3	0.3				
			LP	0	0				
RPVINJCRH	pump status	RPVINJCRH	OK	0	0	n/a	n/a	n/a	n/a
			F	0	0				
			H	0	0.3				
			XX	0	0.3				
RPVINJCRP	pump status	RPVINJCRP	OK	1	1	n/a	n/a	n/a	n/a
			F	1	1				
			H	1	0.147				
			XX	1	0.147				
RPVINJLMTH	pump status	RPVINJLMTH	OK	0	0	n/a	n/a	n/a	n/a
			F	0	0				
			H	0	0.3				
			XX	0	0.3				
RPVINJLMTP	pump status	RPVINJLMTP	OK	0	0	n/a	n/a	n/a	n/a
			F	1	1				
			H	1	0.147				
			XX	1	0.147				

APET Heading	PDS Attribute	APET Down-Branch Identifier	PDS Attribute Value	APET-1	APET-2	APET-3	APET-4	APET-5	APET-6
WWPCD	vent status	WWPCD	WW	0	0	0	n/a	n/a	n/a
			DW	1	1	1	0	0	0
			IS	0.1	0.1	0.1	0.1	0.1	0.1
		WWPCDnodc	WW	0	0	0	n/a	n/a	n/a
			DW	1	1	1	0	0	0
			IS	0.3	0.3	0.3	0.3	0.3	0.3

Note: Down-branch probabilities based on the scoping HRA are highlighted.

APPENDIX E

INTERNAL EVENT ELAP FREQUENCY SUPPORTING INFORMATION

Table E-1 Loss of offsite power frequencies and recovery curve parameters

LOOP Category	LOOP Frequency (per year)	Offsite Power Recovery Curve Lognormal Parameters	
		μ	σ
Plant Centered	2.07E-03	-0.76	1.287
Switchyard Centered	1.04E-02	-0.391	1.256
Grid Related	1.86E-02	0.3	1.064
Weather Related	4.83E-03	0.793	1.982
Source: U.S. Nuclear Regulatory Commission, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," NUREG/CR-6890, Vol. 1, ADAMS Accession No. ML060200477, November 2005.			

Table E-2 Emergency power system reliability parameters

EPS Class	EPS Probability of Failure to Start (per demand)	EPS Rate of Failure to Run (per hour)
2	3.1E-04	1.5E-04
3	1.6E-04	7.2E-05
4	2.5E-05	7.5E-06
Source: Derived from information in the 2011 Update to NUREG/CR-5500, Vol. 5		

Table E-3 Internal event elap frequenciesby eps class and sbo coping time

EPS Failure Mode	Description	Plant Centered	Switchyard Centered	Grid Related	Weather Related	Total
EPS Class 2; SBO Coping Time = 4 hours						
EPS Failure Mode	**Description**	**Plant Centered**	**Switchyard Centered**	**Grid Related**	**Weather Related**	**Total**
EPS failure to start	frequency (per year)	3.0E-08	2.5E-07	8.8E-07	5.7E-07	1.7E-06
	contribution	0.4%	3.6%	12.7%	8.2%	25.1%
EPS failure to run	frequency (per year)	5.3E-08	4.7E-07	1.6E-06	3.0E-06	5.2E-06
	contribution	0.8%	6.9%	23.1%	44.2%	74.9%
Total	frequency (per year)	8.3E-08	7.2E-07	2.5E-06	3.6E-06	6.9E-06
	contribution	1.2%	10.5%	35.8%	52.5%	100.0%
EPS Class 3; SBO Coping Time = 4 hours						
EPS Failure Mode	**Description**	**Plant Centered**	**Switchyard Centered**	**Grid Related**	**Weather Related**	**Total**
EPS failure to start	frequency (per year)	1.5E-08	1.3E-07	4.5E-07	2.9E-07	8.8E-07
	contribution	0.4%	3.7%	12.9%	8.4%	25.4%
EPS failure to run	frequency (per year)	2.6E-08	2.4E-07	7.9E-07	1.5E-06	2.6E-06
	contribution	0.8%	6.8%	23.0%	44.0%	74.6%
Total	frequency (per year)	4.2E-08	3.6E-07	1.2E-06	1.8E-06	3.4E-06
	contribution	1.2%	10.5%	35.9%	52.4%	100.0%
EPS Class 4; SBO Coping Time = 4 hours						
EPS Failure Mode	**Description**	**Plant Centered**	**Switchyard Centered**	**Grid Related**	**Weather Related**	**Total**
EPS failure to start	frequency (per year)	2.4E-09	2.0E-08	7.0E-08	4.5E-08	1.4E-07
	contribution	0.6%	4.9%	17.3%	11.2%	34.0%
EPS failure to run	frequency (per year)	2.7E-09	2.5E-08	8.3E-08	1.6E-07	2.7E-07
	contribution	0.7%	6.1%	20.3%	39.0%	66.0%
Total	frequency (per year)	5.2E-09	4.5E-08	1.5E-07	2.0E-07	4.1E-07
	contribution	1.3%	11.0%	37.6%	50.1%	100.0%
EPS Class 2; SBO Coping Time = 8 hours						
EPS Failure Mode	**Description**	**Plant Centered**	**Switchyard Centered**	**Grid Related**	**Weather Related**	**Total**
EPS failure to start	frequency (per year)	8.7E-09	7.9E-08	2.7E-07	3.8E-07	7.4E-07
	contribution	0.2%	1.9%	6.6%	9.4%	18.1%
EPS failure to run	frequency (per year)	2.2E-08	2.1E-07	6.7E-07	2.4E-06	3.3E-06
	contribution	0.5%	5.2%	16.4%	59.7%	81.9%
Total	frequency (per year)	3.1E-08	2.9E-07	9.4E-07	2.8E-06	4.1E-06
	contribution	0.8%	7.1%	23.0%	69.1%	100.0%
EPS Class 3; SBO Coping Time = 8 hours						
EPS Failure Mode	**Description**	**Plant Centered**	**Switchyard Centered**	**Grid Related**	**Weather Related**	**Total**
EPS failure to start	frequency (per year)	4.4E-09	4.0E-08	1.4E-07	1.9E-07	3.8E-07
	contribution	0.2%	2.0%	6.7%	9.5%	18.4%
EPS failure	frequency (per year)	1.1E-08	1.1E-07	3.3E-07	1.2E-06	1.7E-06

to run	contribution	0.5%	5.2%	16.3%	59.5%	81.6%
Total	frequency (per year)	1.5E-08	1.5E-07	4.7E-07	1.4E-06	2.0E-06
	contribution	0.8%	7.1%	23.0%	69.1%	100.0%

EPS Class 4; SBO Coping Time = 8 hours

EPS Failure Mode	Description	Plant Centered	Switchyard Centered	Grid Related	Weather Related	Total
EPS failure to start	frequency (per year)	7.0E-10	6.3E-09	2.2E-08	3.1E-08	5.9E-08
	contribution	0.3%	2.7%	9.3%	13.2%	25.4%
EPS failure to run	frequency (per year)	1.2E-09	1.1E-08	3.5E-08	1.3E-07	1.7E-07
	contribution	0.5%	4.7%	14.9%	54.4%	74.6%
Total	frequency (per year)	1.9E-09	1.7E-08	5.6E-08	1.6E-07	2.3E-07
	contribution	0.8%	7.4%	24.2%	67.6%	100.0%

APPENDIX F

SEISMIC ELAP FREQUENCY SUPPORTING INFORMATION

Table F-1 Seismic ELAP top logic

Top Event	Description	Boolean Expression	Basic Event Description
SELAP	ELAP due to seismic event	= SLOOP * (SP + DC)	seismic LOOP onsite power fails after seismic event DC power fails after seismic event
SELAP1 (applies to CDET Seqs. 1 to 224)	ELAP due to seismic event <u>and</u> DC succeeds <u>and</u> RCIC-ST succeeds	= SLOOP * SP * /D * /RCI -ST	seismic LOOP onsite power fails after seismic event complement of top event DC complement of top event RCIC-ST
SELAP2 (Applies to CDET Seqs. 225, 226)	ELAP due to seismic event <u>and</u> DC succeeds <u>and</u> RCIC-ST fails	= SLOOP * SP * /D * R IC-ST	seismic LOOP onsite power fails after seismic event complement of top event DC RCIC fails after seismic event
SELAP3 (Applies to CDET Seqs. 227 to 338)	ELAP due to seismic event <u>and</u> DC fails <u>and</u> RCIC-ST succeeds	= SLOOP * D * /RCI -ST	seismic LOOP DC power fails after seismic event complement of top event RCIC-ST
SELAP4 (Applies to CDET Seqs. 339, 340)	ELAP due to seismic event <u>and</u> DC fails <u>and</u> RCIC-ST fails	= SLOOP * D * R IC-ST	seismic LOOP DC power fails after seismic event RCIC fails after seismic event
OSP	Onsite power fails after seismic event	= ACSWGR-S + EDG-S-CTRL + EDG-S-DT + EDG-S-EG + EDG-S-FOST + EDG-S-SAR + EPS-FTR + EPS-FTS	seismic failure of AC switchgear seismic failure of EDG controls seismic failure of EDG day tank seismic failure of EDG engine or generator seismic failure of EDG fuel oil storage tank seismic failure of EDG starting air receiver EPS fails to run (random) EPS fails to run (random)
DC	DC power fails after seismic event	= BATTERY-S + DCSWGR-S	seismic failure of batteries seismic failure of DC switchgear
RCIC-ST	RCIC fails after seismic event	= RCIC-FTR1 + RCIC-FTS + RCIC-S	RCIC fails to run in Phase 1 RCIC fails to start RCIC seismic failure

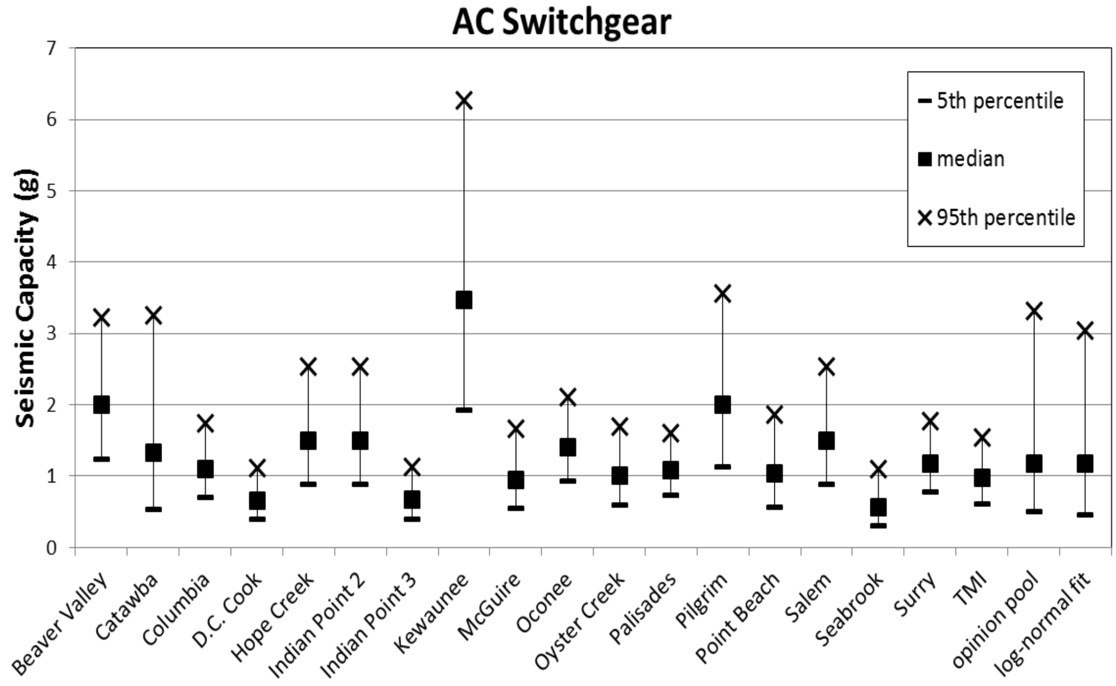
Table F-2 Summary of seismic fragility information

Identifier	Description	Median Seismic Capacity[a], C_{50} (g)	Logarithmic Standard Deviations	
			Randomness[b] β_R	Uncertainty[a] β_U
ACSWGR-S	seismic failure of AC switchgear	1.17	0.24	0.58
BATTERY-S	seismic failure of batteries	1.11	0.24	0.63
DCSWGR-S	seismic failure of DC switchgear	1.39	0.24	0.40
EDG-S-CTRL	seismic failure of EDG controls	1.00	0.24	0.45
EDG-S-DT	seismic failure of EDG day tank	1.33	0.24	0.98
EDG-S-EG	seismic failure of EDG engine or generator	1.16	0.24	0.54
EDG-S-FOST	seismic failure of EDG fuel oil storage tank	1.03	0.24	0.52
EDG-S-SAR	seismic failure of EDG starting air receiver	0.78	0.24	0.61
LOOP-S	seismic LOOP	0.29	0.24	0.45
RCIC-S	seismic failure of RCIC	1.33	0.24	0.59

[a]Inputs and results of the opinion pool used to estimate the median seismic capacity (C_{50}) and logarithmic standard deviation due to uncertainty (β_U) are provided in the seismic fragility worksheets.
[b]The logarithmic standard deviation due to randomness (β_R) is generic value recommended in the SPID

Table F-3 AC switchgear seismic fragility worksheet

Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	2	0.29	1.24	3.22	2	6.9%
Catawba	1.32	0.55	0.53	3.26	2	6.9%
Columbia	1.1	0.28	0.69	1.74	2	6.9%
D.C. Cook	0.66	0.32	0.39	1.12	1	3.4%
Hope Creek	1.5	0.32	0.89	2.54	1	3.4%
Indian Point 2	1.5	0.32	0.89	2.54	1	3.4%
Indian Point 3	0.67	0.321	0.40	1.14	2	6.9%
Kewaunee	3.47	0.36	1.92	6.27	2	6.9%
McGuire	0.95	0.34	0.54	1.66	2	6.9%
Oconee	1.4	0.25	0.93	2.11	1	3.4%
Oyster Creek	1	0.32	0.59	1.69	1	3.4%
Palisades	1.08	0.24	0.73	1.60	1	3.4%
Pilgrim	2	0.35	1.12	3.56	2	6.9%
Point Beach	1.03	0.36	0.57	1.86	2	6.9%
Salem	1.5	0.32	0.89	2.54	1	3.4%
Seabrook	0.57	0.4	0.30	1.10	2	6.9%
Surry	1.17	0.25	0.78	1.77	2	6.9%
Three Mile Island 1	0.97	0.28	0.61	1.54	2	6.9%
opinion pool	1.17	n/a	0.49	3.31	n/a	n/a
log-normal fit	1.17	0.58	0.45	3.04	n/a	n/a



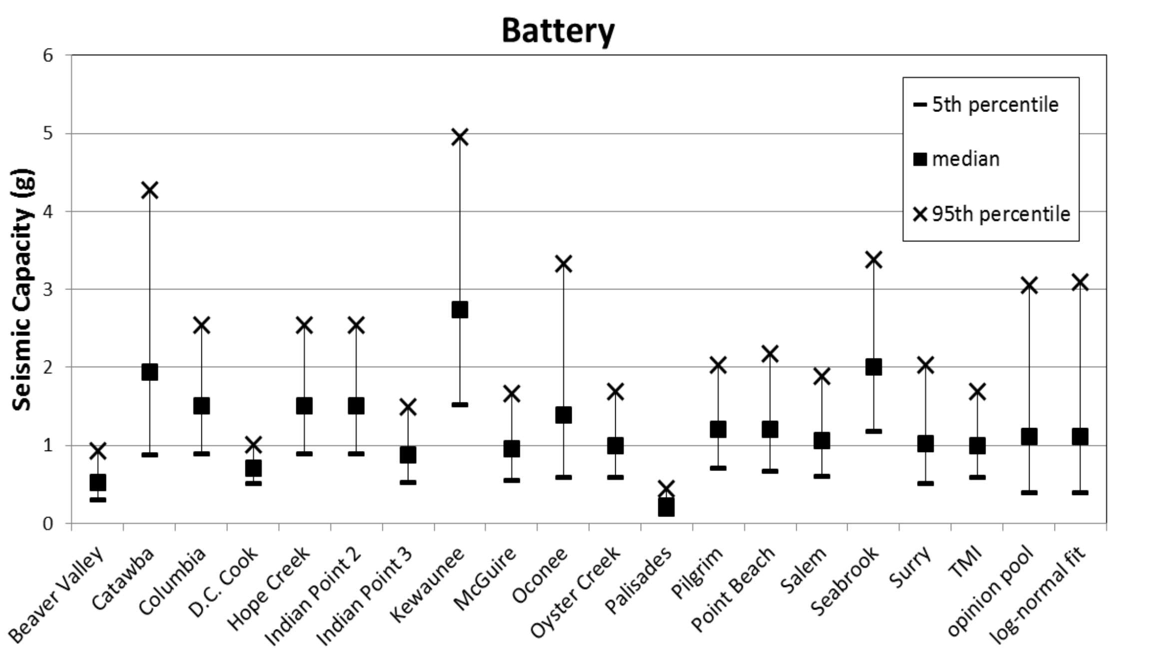
Shaded cells:

C_{50} screening value as reported in the IPEEE

β_U generic value recommended in the SPID

Table F-4 Battery seismic fragility worksheet

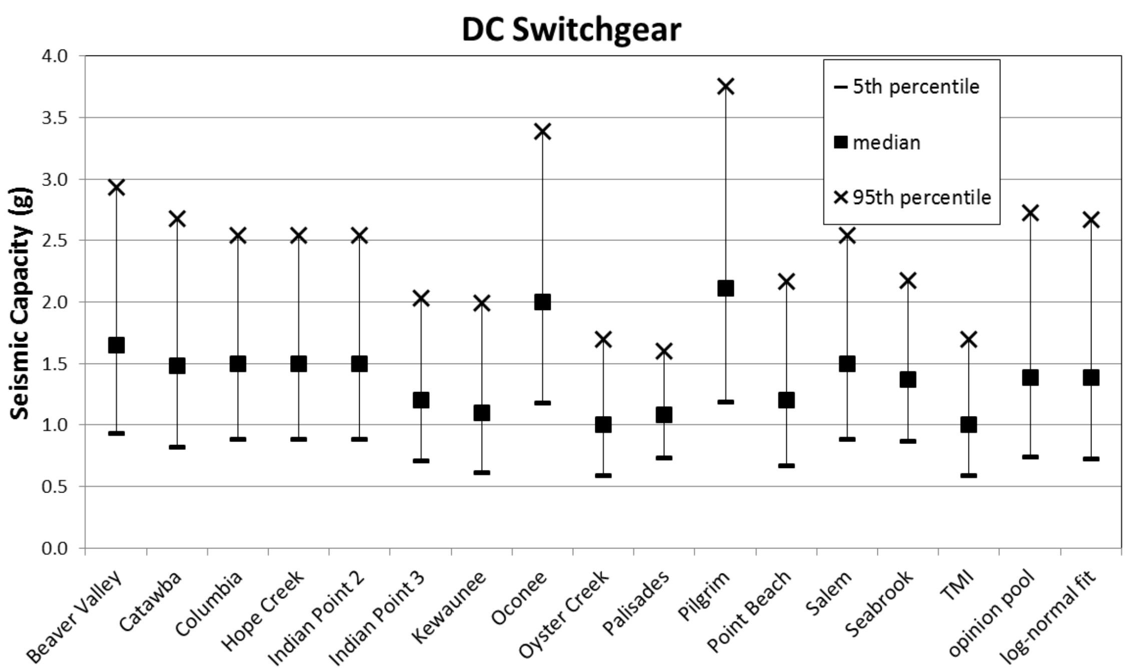
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	0.52	0.35	0.29	0.92	2	7.1%
Catawba	1.94	0.48	0.88	4.27	2	7.1%
Columbia	1.5	0.32	0.89	2.54	1	3.6%
D.C. Cook	0.71	0.21	0.50	1.00	2	7.1%
Hope Creek	1.5	0.32	0.89	2.54	1	3.6%
Indian Point 2	1.5	0.32	0.89	2.54	1	3.6%
Indian Point 3	0.88	0.321	0.52	1.49	2	7.1%
Kewaunee	2.74	0.36	1.52	4.95	2	7.1%
McGuire	0.95	0.34	0.54	1.66	2	7.1%
Oconee	1.39	0.53	0.58	3.32	2	7.1%
Oyster Creek	1	0.32	0.59	1.69	1	3.6%
Palisades	0.22	0.42	0.11	0.44	1	3.6%
Pilgrim	1.2	0.32	0.71	2.03	1	3.6%
Point Beach	1.2	0.36	0.66	2.17	2	7.1%
Salem	1.06	0.35	0.60	1.89	2	7.1%
Seabrook	2	0.32	1.18	3.39	1	3.6%
Surry	1.02	0.42	0.51	2.04	2	7.1%
Three Mile Island 1	1	0.32	0.59	1.69	1	3.6%
opinion pool	1.11	n/a	0.39	3.05	n/a	n/a
log-normal fit	1.11	0.63	0.40	3.10	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-5 DC switchgear seismic fragility worksheet

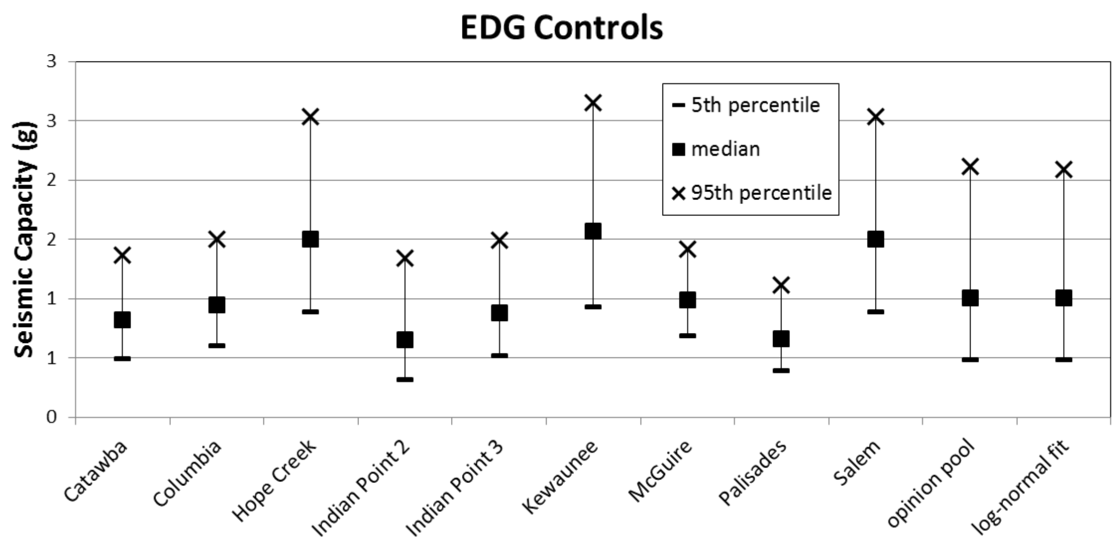
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	1.65	0.35	0.93	2.93	2	9.5%
Catawba	1.48	0.36	0.82	2.68	2	9.5%
Columbia	1.5	0.32	0.89	2.54	1	4.8%
Hope Creek	1.5	0.32	0.89	2.54	1	4.8%
Indian Point 2	1.5	0.32	0.89	2.54	1	4.8%
Indian Point 3	1.2	0.32	0.71	2.03	1	4.8%
Kewaunee	1.1	0.36	0.61	1.99	2	9.5%
Oconee	2	0.32	1.18	3.39	1	4.8%
Oyster Creek	1	0.32	0.59	1.69	1	4.8%
Palisades	1.08	0.24	0.73	1.60	1	4.8%
Pilgrim	2.11	0.35	1.19	3.75	2	9.5%
Point Beach	1.2	0.36	0.66	2.17	2	9.5%
Salem	1.5	0.32	0.89	2.54	1	4.8%
Seabrook	1.37	0.28	0.86	2.17	2	9.5%
Three Mile Island 1	1	0.32	0.59	1.69	1	4.8%
opinion pool	1.39	n/a	0.74	2.73	n/a	n/a
log-normal fit	1.39	0.40	0.72	2.67	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-6 EDG controls seismic fragility worksheet

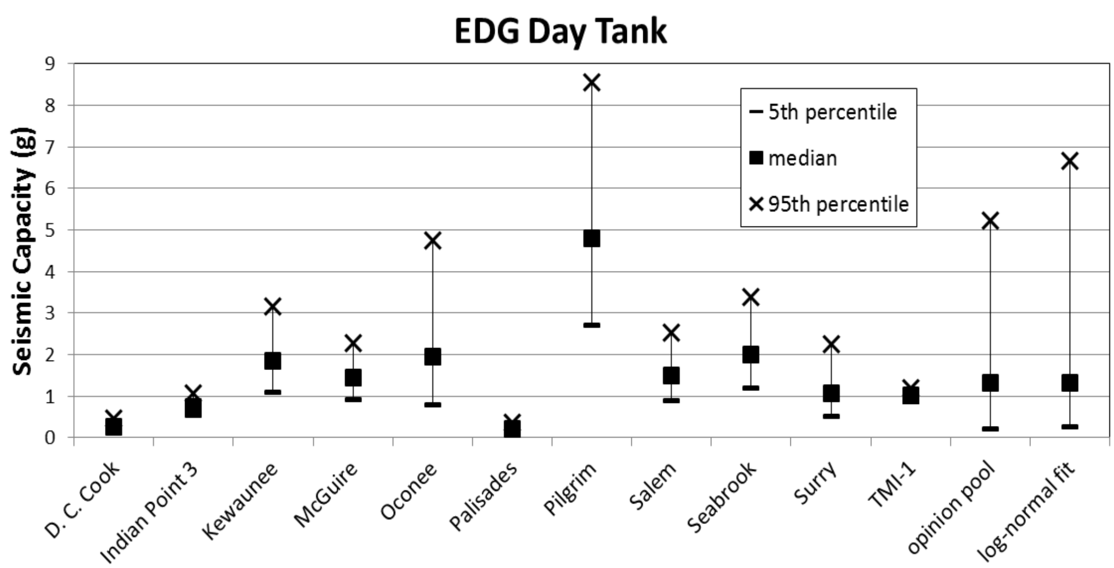
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Catawba	0.82	0.31	0.49	1.37	2	13.3%
Columbia	0.95	0.28	0.60	1.51	2	13.3%
Hope Creek	1.5	0.32	0.89	2.54	2	13.3%
Indian Point 2	0.65	0.44	0.32	1.34	1	6.7%
Indian Point 3	0.88	0.321	0.52	1.49	1	6.7%
Kewaunee	1.57	0.32	0.93	2.66	1	6.7%
McGuire	0.99	0.22	0.69	1.42	2	13.3%
Palisades	0.66	0.32	0.39	1.12	2	13.3%
Salem	1.5	0.32	0.89	2.54	2	13.3%
opinion pool	1.00	n/a	0.49	2.12	n/a	n/a
log-normal fit	1.00	0.45	0.48	2.09	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-7 EDG day tank seismic fragility worksheet

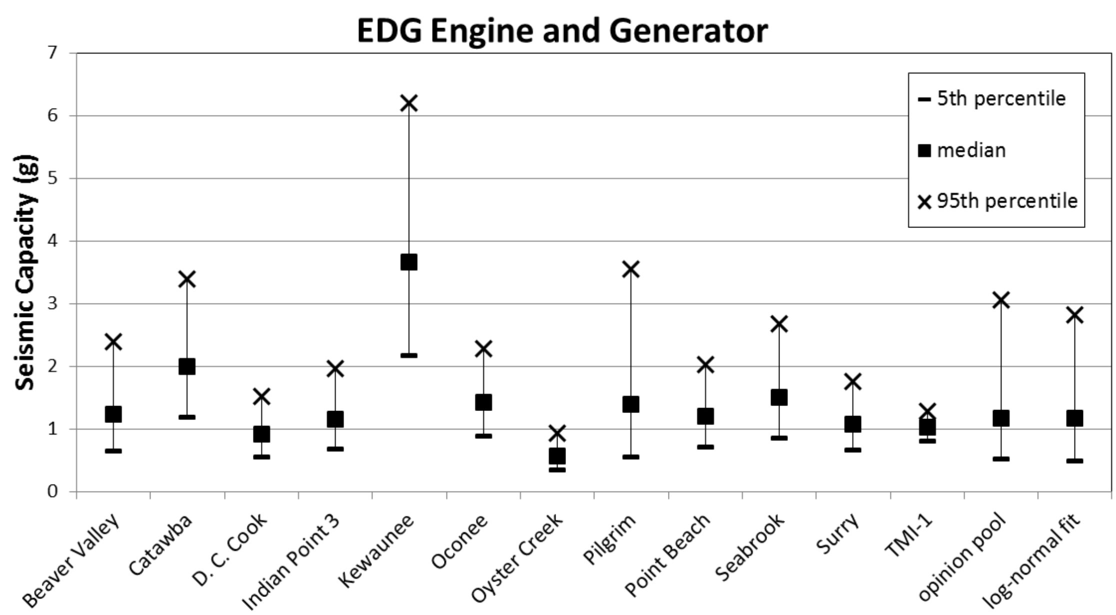
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
D. C. Cook	0.27	0.32	0.16	0.46	2	11.8%
Indian Point 3	0.75	0.213	0.53	1.06	2	11.8%
Kewaunee	1.86	0.32	1.10	3.15	2	11.8%
McGuire	1.44	0.28	0.91	2.28	1	5.9%
Oconee	1.95	0.54	0.80	4.74	1	5.9%
Palisades	0.22	0.32	0.13	0.37	1	5.9%
Pilgrim	4.81	0.35	2.70	8.55	2	11.8%
Salem	1.5	0.32	0.89	2.54	2	11.8%
Seabrook	2	0.32	1.18	3.39	2	11.8%
Surry	1.08	0.45	0.52	2.26	1	5.9%
TMI-1	1.01	0.1	0.86	1.19	1	5.9%
opinion pool	1.33	n/a	0.21	5.23	n/a	n/a
log-normal fit	1.33	0.98	0.27	6.66	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-8 EDG engine and generator seismic fragility worksheet

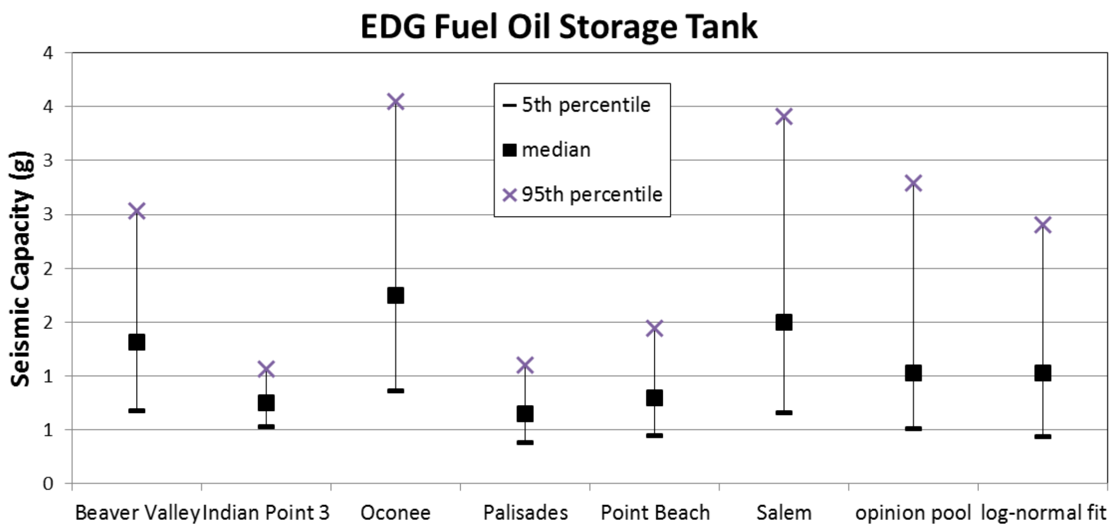
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	1.24	0.4	0.64	2.39	2	9.5%
Catawba	2	0.32	1.18	3.39	1	4.8%
D. C Cook	0.91	0.31	0.55	1.52	2	9.5%
Indian Point 3	1.16	0.321	0.68	1.97	2	9.5%
Kewaunee	3.67	0.32	2.17	6.21	1	4.8%
Oconee	1.42	0.29	0.88	2.29	2	9.5%
Oyster Creek	0.56	0.31	0.34	0.93	2	9.5%
Pilgrim	1.39	0.57	0.54	3.55	2	9.5%
Point Beach	1.2	0.32	0.71	2.03	1	4.8%
Seabrook	1.51	0.35	0.85	2.69	2	9.5%
Surry	1.07	0.3	0.65	1.75	2	9.5%
TMI-1	1.02	0.14	0.81	1.28	2	9.5%
opinion pool	1.16	n/a	0.52	3.06	n/a	n/a
log-normal fit	1.16	0.54	0.48	2.82	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-9 EDG fuel oil storage tank seismic fragility worksheet

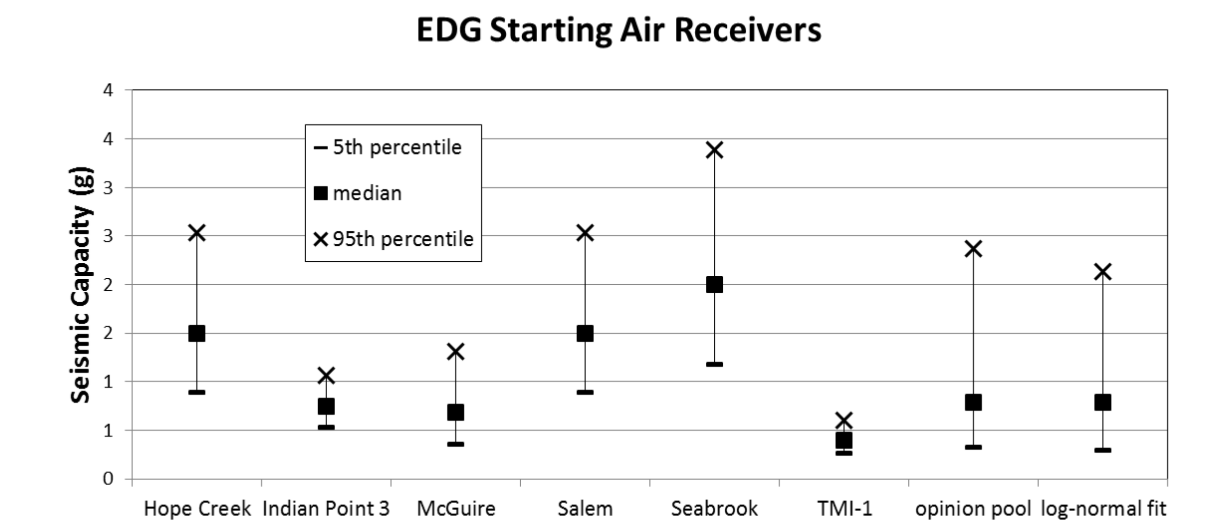
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	1.31	0.4	0.68	2.53	2	18.2%
Indian Point 3	0.75	0.213	0.53	1.06	2	18.2%
Oconee	1.75	0.43	0.86	3.55	2	18.2%
Palisades	0.65	0.32	0.38	1.10	1	9.1%
Point Beach	0.8	0.36	0.44	1.45	2	18.2%
Salem	1.5	0.5	0.66	3.41	2	18.2%
opinion pool	1.03	n/a	0.51	2.79	n/a	n/a
log-normal fit	1.03	0.52	0.44	2.41	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-10 EDG starting air receiver seismic fragility worksheet

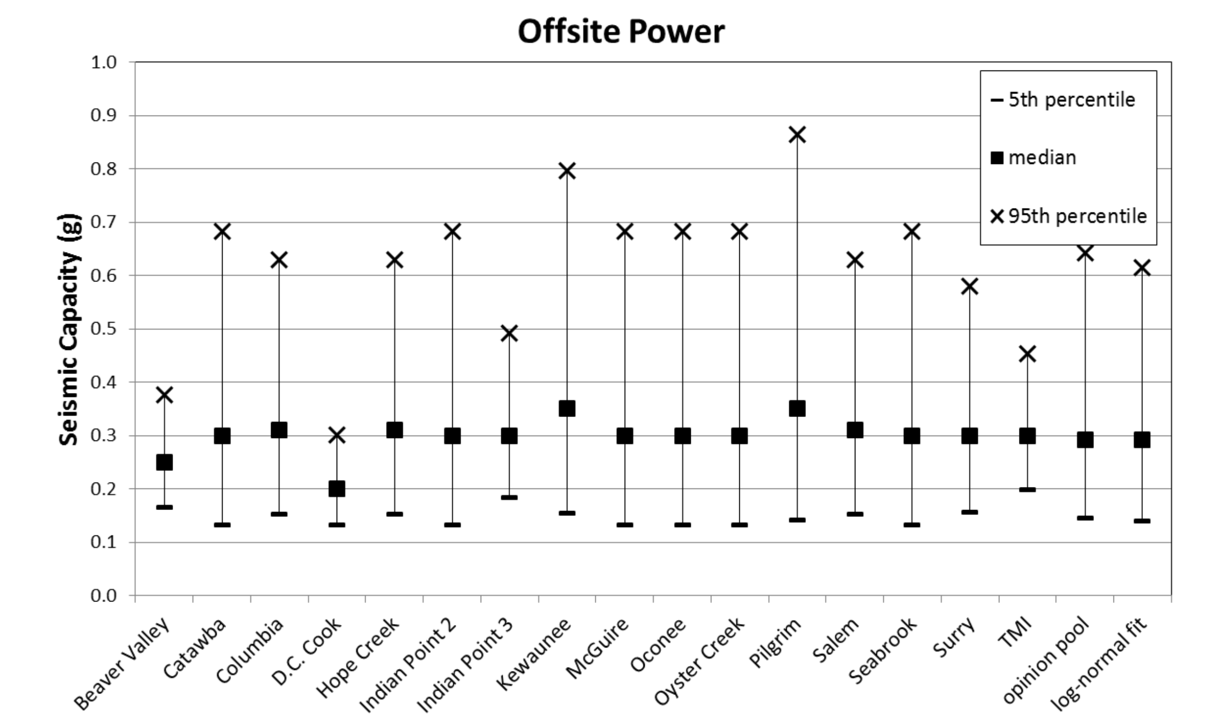
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Hope Creek	1.5	0.32	0.89	2.54	1	11.1%
Indian Point 3	0.75	0.213	0.53	1.06	2	22.2%
McGuire	0.68	0.4	0.35	1.31	2	22.2%
Salem	1.5	0.32	0.89	2.54	1	11.1%
Seabrook	2	0.32	1.18	3.39	1	11.1%
TMI-1	0.4	0.25	0.27	0.60	2	22.2%
opinion pool	0.79	n/a	0.32	2.37	n/a	n/a
log-normal fit	0.79	0.61	0.29	2.13	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-11 Loss of offsite power seismic fragility worksheet

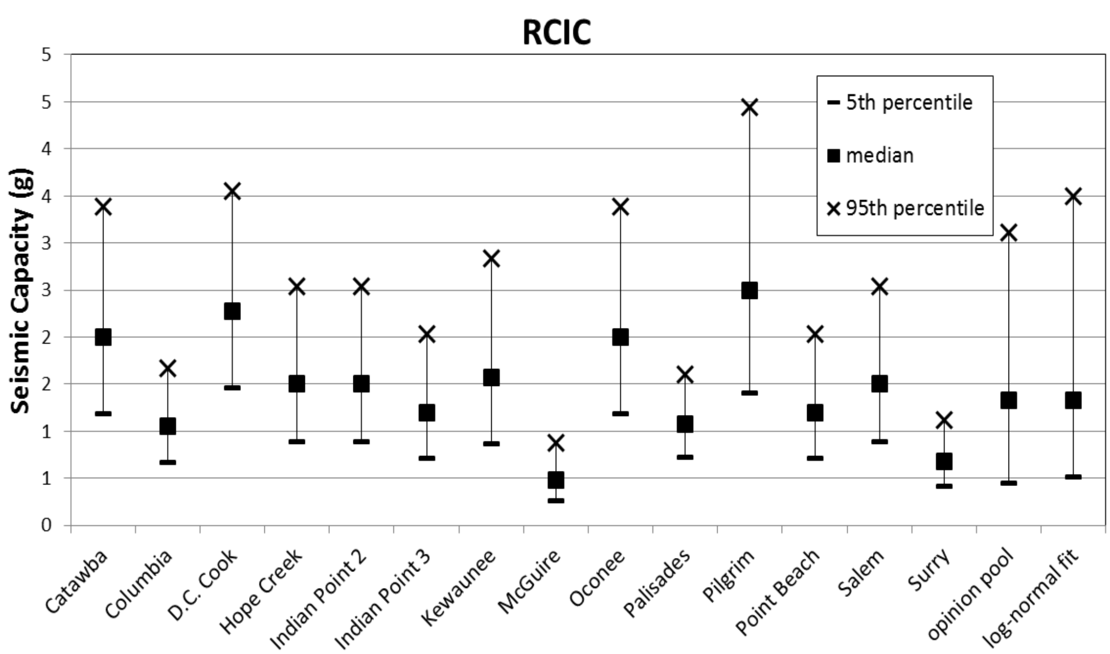
Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Beaver Valley	0.25	0.25	0.17	0.38	2	6.3%
Catawba	0.3	0.5	0.13	0.68	2	6.3%
Columbia	0.31	0.43	0.15	0.63	2	6.3%
D.C. Cook	0.2	0.25	0.13	0.30	2	6.3%
Hope Creek	0.31	0.43	0.15	0.63	2	6.3%
Indian Point 2	0.3	0.5	0.13	0.68	2	6.3%
Indian Point 3	0.3	0.3	0.18	0.49	2	6.3%
Kewaunee	0.35	0.5	0.15	0.80	2	6.3%
McGuire	0.3	0.5	0.13	0.68	2	6.3%
Oconee	0.3	0.5	0.13	0.68	2	6.3%
Oyster Creek	0.3	0.5	0.13	0.68	2	6.3%
Pilgrim	0.35	0.55	0.14	0.86	2	6.3%
Salem	0.31	0.43	0.15	0.63	2	6.3%
Seabrook	0.3	0.5	0.13	0.68	2	6.3%
Surry	0.3	0.4	0.16	0.58	2	6.3%
Three Mile Island 1	0.3	0.25	0.20	0.45	2	6.3%
opinion pool	0.29	n/a	0.14	0.64	n/a	n/a
log-normal fit	0.29	0.45	0.14	0.62	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-12 RCIC seismic fragility worksheet

Plant	IPEEE Information		Median Capacity Percentiles		Opinion Pool	
	C_{50}	β_U	5th	95th	importance	weight
Catawba	2.00	0.32	1.18	3.39	1	4.8%
Columbia	1.05	0.28	0.66	1.66	2	9.5%
D.C. Cook	2.28	0.27	1.46	3.55	2	9.5%
Hope Creek	1.50	0.32	0.89	2.54	1	4.8%
Indian Point 2	1.50	0.32	0.89	2.54	1	4.8%
Indian Point 3	1.20	0.32	0.71	2.03	1	4.8%
Kewaunee	1.57	0.36	0.87	2.84	2	9.5%
McGuire	0.48	0.37	0.26	0.88	2	9.5%
Oconee	2.00	0.32	1.18	3.39	1	4.8%
Palisades	1.08	0.24	0.73	1.60	2	9.5%
Pilgrim	2.50	0.35	1.41	4.45	2	9.5%
Point Beach	1.20	0.32	0.71	2.03	1	4.8%
Salem	1.50	0.32	0.89	2.54	1	4.8%
Surry	0.68	0.3	0.42	1.11	2	9.5%
opinion pool	1.33	n/a	0.45	3.11	n/a	n/a
log-normal fit	1.33	0.59	0.51	3.50	n/a	n/a



Shaded cells:
C_{50} screening value as reported in the IPEEE
β_U generic value recommended in the SPID

Table F-13 Browns Ferry seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	1.5E-03	6.9E-04	3.1E-04	2.5E-04	1.7E-04	3.1E-05	8.0E-06	2.4E-06	1.5E-06	6.5E-07	n/a	n/a
Total ELAP	1.2E-10	2.4E-09	1.5E-08	1.6E-07	3.3E-06	8.6E-06	6.0E-06	2.3E-06	1.5E-06	6.5E-07	2.2E-05	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	1.2E-10	2.3E-09	1.5E-08	1.5E-07	3.0E-06	6.9E-06	3.8E-06	9.1E-07	2.2E-07	4.6E-09	1.5E-05	66.8%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	1.1E-12	2.1E-11	1.4E-10	1.4E-09	3.3E-08	2.5E-07	5.2E-07	3.2E-07	1.8E-07	1.5E-08	1.3E-06	5.9%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	1.9E-14	6.8E-12	2.1E-10	6.1E-09	3.1E-07	1.3E-06	1.4E-06	8.1E-07	5.9E-07	1.4E-07	4.6E-06	20.6%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	1.7E-16	6.1E-14	1.9E-12	5.6E-11	3.4E-09	4.8E-08	2.0E-07	2.8E-07	4.9E-07	4.8E-07	1.5E-06	6.7%

Table F-14 Brunswick seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	7.6E-04	3.8E-04	1.7E-04	1.4E-04	8.5E-05	1.5E-05	3.5E-06	9.4E-07	5.2E-07	2.0E-07	n/a	n/a
Total ELAP	6.7E-10	1.0E-08	4.0E-08	2.2E-07	2.1E-06	4.2E-06	2.6E-06	9.0E-07	5.1E-07	2.0E-07	1.1E-05	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	6.6E-10	1.0E-08	3.9E-08	2.1E-07	2.0E-06	3.4E-06	1.7E-06	3.5E-07	7.5E-08	1.4E-09	7.7E-06	71.5%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	6.0E-12	9.1E-11	3.5E-10	1.9E-09	2.2E-08	1.2E-07	2.3E-07	1.2E-07	6.2E-08	4.6E-09	5.6E-07	5.2%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	9.6E-15	3.8E-12	1.2E-10	3.4E-09	1.6E-07	6.3E-07	6.2E-07	3.1E-07	2.1E-07	4.4E-08	2.0E-06	18.4%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	8.7E-17	3.4E-14	1.1E-12	3.1E-11	1.8E-09	2.3E-08	8.5E-08	1.1E-07	1.7E-07	1.5E-07	5.3E-07	4.9%

Table F-15 Cooper seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	4.8E-04	1.6E-04	5.7E-05	4.3E-05	2.9E-05	6.5E-06	1.8E-06	5.1E-07	2.7E-07	9.6E-08	n/a	n/a
Total ELAP	4.2E-10	4.1E-09	1.3E-08	6.7E-08	7.3E-07	1.8E-06	1.3E-06	4.8E-07	2.7E-07	9.6E-08	4.9E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	4.2E-10	4.1E-09	1.3E-08	6.5E-08	6.7E-07	1.5E-06	8.6E-07	1.9E-07	4.0E-08	6.8E-10	3.3E-06	68.9%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	3.8E-12	3.7E-11	1.2E-10	5.9E-10	7.4E-09	5.5E-08	1.2E-07	6.6E-08	3.3E-08	2.3E-09	2.8E-07	5.8%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	6.0E-15	1.5E-12	3.9E-11	1.0E-09	5.5E-08	2.8E-07	3.2E-07	1.7E-07	1.1E-07	2.1E-08	9.6E-07	19.7%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	5.5E-17	1.4E-14	3.5E-13	9.6E-12	6.1E-10	1.0E-08	4.4E-08	5.8E-08	9.1E-08	7.2E-08	2.8E-07	5.7%

Table F-16 Duane Arnold seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	1.9E-04	6.0E-05	2.1E-05	1.6E-05	1.1E-05	2.8E-06	9.1E-07	3.1E-07	2.0E-07	8.6E-08	n/a	n/a
Total ELAP	1.7E-10	1.6E-09	4.8E-09	2.5E-08	2.8E-07	7.9E-07	6.9E-07	2.9E-07	1.9E-07	8.6E-08	2.4E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	1.7E-10	1.6E-09	4.8E-09	2.4E-08	2.5E-07	6.4E-07	4.4E-07	1.2E-07	2.8E-08	6.1E-10	1.5E-06	64.1%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	1.5E-12	1.4E-11	4.3E-11	2.2E-10	2.8E-09	2.3E-08	6.0E-08	4.0E-08	2.4E-08	2.0E-09	1.5E-07	6.4%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	2.4E-15	5.9E-13	1.4E-11	3.9E-10	2.1E-08	1.2E-07	1.6E-07	1.0E-07	7.8E-08	1.9E-08	5.0E-07	21.4%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	2.2E-17	5.3E-15	1.3E-13	3.5E-12	2.3E-10	4.3E-09	2.2E-08	3.5E-08	6.5E-08	6.4E-08	1.9E-07	8.1%

Table F-17 Fermi seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	5.1E-04	1.9E-04	7.6E-05	6.2E-05	4.5E-05	1.1E-05	3.6E-06	1.2E-06	7.0E-07	2.9E-07	n/a	n/a
Total ELAP	4.1E-11	6.5E-10	3.8E-09	4.0E-08	9.0E-07	3.1E-06	2.7E-06	1.1E-06	7.0E-07	2.9E-07	8.8E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	4.0E-11	6.4E-10	3.7E-09	3.8E-08	8.0E-07	2.5E-06	1.7E-06	4.3E-07	1.0E-07	2.1E-09	5.6E-06	63.6%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	3.6E-13	5.8E-12	3.4E-11	3.4E-10	8.8E-09	9.1E-08	2.3E-07	1.5E-07	8.5E-08	7.0E-09	5.7E-07	6.5%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	6.5E-15	1.9E-12	5.2E-11	1.5E-09	8.5E-08	4.8E-07	6.4E-07	3.8E-07	2.8E-07	6.6E-08	1.9E-06	22.0%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	5.8E-17	1.7E-14	4.7E-13	1.4E-11	9.3E-10	1.8E-08	8.7E-08	1.3E-07	2.3E-07	2.2E-07	6.9E-07	7.8%

Table F-18 FitzPatrick seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	4.9E-04	1.6E-04	5.8E-05	4.1E-05	2.3E-05	4.6E-06	1.3E-06	4.1E-07	2.5E-07	1.1E-07	n/a	n/a
Total ELAP	3.9E-11	5.6E-10	2.9E-09	2.6E-08	4.7E-07	1.3E-06	9.7E-07	3.9E-07	2.5E-07	1.1E-07	3.5E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	3.9E-11	5.5E-10	2.8E-09	2.5E-08	4.2E-07	1.0E-06	6.2E-07	1.5E-07	3.6E-08	7.5E-10	2.3E-06	65.6%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	3.5E-13	5.0E-12	2.6E-11	2.2E-10	4.6E-09	3.7E-08	8.4E-08	5.3E-08	3.0E-08	2.5E-09	2.1E-07	6.1%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	6.2E-15	1.6E-12	3.9E-11	9.8E-10	4.4E-08	2.0E-07	2.3E-07	1.4E-07	1.0E-07	2.4E-08	7.3E-07	21.2%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	5.6E-17	1.4E-14	3.5E-13	9.0E-12	4.8E-10	7.1E-09	3.2E-08	4.7E-08	8.3E-08	8.0E-08	2.5E-07	7.2%

Table F-19 Hatch seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	7.0E-04	3.2E-04	1.4E-04	1.1E-04	6.8E-05	1.3E-05	3.3E-06	1.0E-06	6.4E-07	3.0E-07	n/a	n/a
Total ELAP	3.2E-10	4.5E-09	1.8E-08	1.2E-07	1.5E-06	3.6E-06	2.5E-06	9.6E-07	6.4E-07	3.0E-07	9.6E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	3.1E-10	4.5E-09	1.8E-08	1.1E-07	1.4E-06	2.9E-06	1.6E-06	3.8E-07	9.3E-08	2.1E-09	6.5E-06	67.3%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	2.8E-12	4.1E-11	1.6E-10	1.0E-09	1.5E-08	1.0E-07	2.1E-07	1.3E-07	7.7E-08	7.1E-09	5.5E-07	5.8%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	8.8E-15	3.1E-12	9.3E-11	2.6E-09	1.3E-07	5.4E-07	5.9E-07	3.4E-07	2.5E-07	6.7E-08	1.9E-06	20.1%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	8.0E-17	2.8E-14	8.4E-13	2.4E-11	1.4E-09	2.0E-08	8.1E-08	1.2E-07	2.1E-07	2.2E-07	6.5E-07	6.8%

Table F-20 Hope Creek seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	5.5E-04	2.2E-04	9.0E-05	7.3E-05	5.2E-05	1.1E-05	2.8E-06	7.4E-07	3.7E-07	1.1E-07	n/a	n/a
Total ELAP	2.5E-10	3.1E-09	1.2E-08	7.8E-08	1.2E-06	3.1E-06	2.1E-06	7.1E-07	3.7E-07	1.1E-07	7.6E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	2.5E-10	3.1E-09	1.2E-08	7.5E-08	1.0E-06	2.5E-06	1.3E-06	2.8E-07	5.3E-08	8.0E-10	5.3E-06	69.8%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	2.2E-12	2.8E-11	1.1E-10	6.9E-10	1.2E-08	9.1E-08	1.8E-07	9.7E-08	4.4E-08	2.7E-09	4.3E-07	5.6%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	6.9E-15	2.1E-12	6.1E-11	1.8E-09	9.7E-08	4.7E-07	5.1E-07	2.5E-07	1.5E-07	2.5E-08	1.5E-06	19.6%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	6.2E-17	1.9E-14	5.5E-13	1.6E-11	1.1E-09	1.7E-08	6.9E-08	8.6E-08	1.2E-07	8.4E-08	3.8E-07	5.0%

Table F-21 Monticello seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	2.8E-04	1.1E-04	4.7E-05	4.0E-05	3.0E-05	8.0E-06	2.6E-06	8.8E-07	5.6E-07	2.3E-07	n/a	n/a
Total ELAP	2.5E-10	3.0E-09	1.1E-08	6.2E-08	7.6E-07	2.3E-06	2.0E-06	8.4E-07	5.6E-07	2.3E-07	6.7E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	2.5E-10	3.0E-09	1.1E-08	6.1E-08	7.0E-07	1.8E-06	1.3E-06	3.3E-07	8.1E-08	1.6E-09	4.3E-06	64.0%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	2.2E-12	2.7E-11	9.7E-11	5.5E-10	7.7E-09	6.7E-08	1.7E-07	1.2E-07	6.8E-08	5.5E-09	4.4E-07	6.5%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	3.6E-15	1.1E-12	3.2E-11	9.7E-10	5.7E-08	3.4E-07	4.7E-07	2.9E-07	2.2E-07	5.2E-08	1.4E-06	21.5%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	3.2E-17	1.0E-14	2.9E-13	8.9E-12	6.3E-10	1.2E-08	6.5E-08	1.0E-07	1.9E-07	1.7E-07	5.4E-07	8.0%

Table F-22 Peach Bottom seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	4.8E-04	2.4E-04	1.2E-04	1.2E-04	1.1E-04	3.9E-05	1.6E-05	6.2E-06	4.6E-06	2.6E-06	n/a	n/a
Total ELAP	2.2E-10	3.5E-09	1.7E-08	1.3E-07	2.6E-06	1.1E-05	1.2E-05	5.9E-06	4.6E-06	2.6E-06	3.8E-05	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	2.2E-10	3.4E-09	1.6E-08	1.2E-07	2.3E-06	8.8E-06	7.4E-06	2.3E-06	6.7E-07	1.8E-08	2.2E-05	56.8%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	1.9E-12	3.1E-11	1.5E-10	1.1E-09	2.6E-08	3.2E-07	1.0E-06	8.0E-07	5.5E-07	6.1E-08	2.8E-06	7.3%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	6.1E-15	2.4E-12	8.4E-11	2.9E-09	2.2E-07	1.7E-06	2.8E-06	2.1E-06	1.8E-06	5.7E-07	9.1E-06	23.9%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	5.5E-17	2.2E-14	7.6E-13	2.7E-11	2.4E-09	6.1E-08	3.8E-07	7.1E-07	1.5E-06	1.9E-06	4.6E-06	12.0%

Table F-23 Pilgrim seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	1.1E-03	5.3E-04	2.6E-04	2.5E-04	2.3E-04	7.3E-05	2.9E-05	1.1E-05	7.5E-06	3.5E-06	n/a	n/a
Total ELAP	9.5E-10	1.4E-08	5.9E-08	3.9E-07	5.7E-06	2.1E-05	2.1E-05	1.0E-05	7.5E-06	3.5E-06	7.0E-05	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	9.4E-10	1.4E-08	5.9E-08	3.8E-07	5.2E-06	1.7E-05	1.4E-05	4.1E-06	1.1E-06	2.5E-08	4.1E-05	59.5%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	8.5E-12	1.3E-10	5.3E-10	3.4E-09	5.7E-08	6.2E-07	1.9E-06	1.4E-06	9.1E-07	8.3E-08	4.9E-06	7.1%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	1.4E-14	5.2E-12	1.8E-10	6.0E-09	4.3E-07	3.1E-06	5.1E-06	3.6E-06	3.0E-06	7.8E-07	1.6E-05	23.1%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	1.2E-16	4.7E-14	1.6E-12	5.5E-11	4.7E-09	1.1E-07	7.0E-07	1.2E-06	2.5E-06	2.6E-06	7.2E-06	10.3%

Table F-24 Quad Cities seismic ELAP frequencies

Frequency (per year)	Bin 1 0.03 to 0.05 g	Bin 2 0.05 to 0.075 g	Bin 3 0.075 to 0.1 g	Bin 4 0.1 to 0.15 g	Bin 5 0.15 to 0.3 g	Bin 6 0.3 to 0.5 g	Bin 7 0.5 to 0.75 g	Bin 8 0.75 to 1 g	Bin 9 1 to 1.5 g	Bin 10 1.5 to 3 g	Total	%
Seismic Bin	4.7E-04	1.6E-04	6.2E-05	4.8E-05	3.4E-05	8.7E-06	2.8E-06	9.4E-07	6.0E-07	2.6E-07	n/a	n/a
Total ELAP	3.7E-11	5.6E-10	3.1E-09	3.1E-08	6.8E-07	2.4E-06	2.1E-06	9.0E-07	6.0E-07	2.6E-07	7.0E-06	100.0%
ELAP and DC success and RCIC-ST success (Seqs. 1 to 224)	3.7E-11	5.5E-10	3.1E-09	2.9E-08	6.1E-07	1.9E-06	1.3E-06	3.6E-07	8.7E-08	1.9E-09	4.4E-06	62.6%
ELAP and DC success and RCIC-ST fails (Seqs. 225, 226)	3.3E-13	5.0E-12	2.8E-11	2.7E-10	6.7E-09	7.0E-08	1.8E-07	1.2E-07	7.3E-08	6.2E-09	4.6E-07	6.6%
ELAP and DC fails and RCIC-ST success (Seqs. 227 to 338)	5.9E-15	1.6E-12	4.2E-11	1.2E-09	6.4E-08	3.7E-07	5.1E-07	3.1E-07	2.4E-07	5.9E-08	1.6E-06	22.4%
ELAP and DC fails and RCIC-ST fails (Seqs. 339, 340)	5.4E-17	1.5E-14	3.8E-13	1.1E-11	7.0E-10	1.3E-08	6.9E-08	1.1E-07	2.0E-07	2.0E-07	5.9E-07	8.4%

APPENDIX G

PLANT DAMAGE STATE FREQUENCIES

Table G-1 Plant damage state frequencies (per year)

Index	Plant Damage State	WWF Venting Strategy	DWF Venting Strategy (Passive)	DWF Venting Strategy (Manual)
1	E-HP-IS-ST-XX	4.3E-07	4.3E-07	4.3E-07
2	E-HP-IS-XX-XX	1.1E-07	1.1E-07	1.1E-07
3	E-LP-IS-ST-XX	1.0E-06	1.0E-06	1.0E-06
4	E-LP-IS-XX-XX	9.8E-07	9.8E-07	9.8E-07
5	L-HP-DW-LT-F	2.0E-09	3.2E-08	6.8E-09
6	L-HP-DW-LT-H	1.3E-09	2.1E-08	4.5E-09
7	L-HP-DW-LT-OK	8.0E-10	1.3E-08	2.7E-09
8	L-HP-DW-OK-F	6.6E-10	7.2E-08	6.6E-09
9	L-HP-DW-OK-H	4.3E-10	4.8E-08	4.3E-09
10	L-HP-DW-OK-OK	8.6E-11	9.4E-09	8.6E-10
11	L-HP-DW-ST-F	4.2E-09	6.5E-08	1.4E-08
12	L-HP-DW-ST-H	2.7E-09	4.3E-08	9.1E-09
13	L-HP-DW-ST-OK	1.6E-09	2.6E-08	5.4E-09
14	L-HP-IS-LT-F	2.1E-08	0.0E+00	2.1E-08
15	L-HP-IS-LT-H	1.4E-08	0.0E+00	1.4E-08
16	L-HP-IS-LT-OK	8.1E-09	0.0E+00	8.1E-09
17	L-HP-IS-OK-F	6.5E-08	0.0E+00	6.5E-08
18	L-HP-IS-OK-H	4.3E-08	0.0E+00	4.3E-08
19	L-HP-IS-OK-OK	8.5E-09	0.0E+00	8.5E-09
20	L-HP-IS-ST-F	4.2E-08	0.0E+00	4.2E-08
21	L-HP-IS-ST-H	2.8E-08	0.0E+00	2.8E-08
22	L-HP-IS-ST-OK	1.6E-08	0.0E+00	1.6E-08
23	L-HP-WW-LT-F	6.8E-09	2.3E-10	2.0E-09
24	L-HP-WW-LT-H	4.5E-09	1.5E-10	1.3E-09
25	L-HP-WW-LT-OK	2.7E-09	8.9E-11	8.0E-10
26	L-HP-WW-OK-F	6.6E-09	6.6E-10	6.6E-10
27	L-HP-WW-OK-H	4.3E-09	4.3E-10	4.3E-10
28	L-HP-WW-OK-OK	8.6E-10	8.6E-11	8.6E-11
29	L-HP-WW-ST-F	1.4E-08	4.6E-10	4.2E-09
30	L-HP-WW-ST-H	9.1E-09	3.0E-10	2.7E-09
31	L-HP-WW-ST-OK	5.4E-09	1.8E-10	1.6E-09
32	L-MP-DW-LT-F	1.6E-08	2.6E-07	5.5E-08
33	L-MP-DW-LT-H	1.1E-08	1.7E-07	3.6E-08
34	L-MP-DW-LT-OK	6.5E-09	1.0E-07	2.2E-08
35	L-MP-DW-OK-F	6.8E-09	7.5E-07	6.8E-08
36	L-MP-DW-OK-H	4.5E-09	4.9E-07	4.5E-08
37	L-MP-DW-OK-OK	8.9E-10	9.8E-08	8.9E-09
38	L-MP-DW-ST-F	6.8E-09	1.1E-07	2.3E-08

Index	Plant Damage State	WWF Venting Strategy	DWF Venting Strategy (Passive)	DWF Venting Strategy (Manual)
39	L-MP-DW-ST-H	4.5E-09	7.0E-08	1.5E-08
40	L-MP-DW-ST-OK	2.6E-09	4.2E-08	8.8E-09
41	L-MP-IS-LT-F	1.7E-07	0.0E+00	1.7E-07
42	L-MP-IS-LT-H	1.1E-07	0.0E+00	1.1E-07
43	L-MP-IS-LT-OK	6.5E-08	0.0E+00	6.5E-08
44	L-MP-IS-OK-F	6.7E-07	0.0E+00	6.7E-07
45	L-MP-IS-OK-H	4.4E-07	0.0E+00	4.4E-07
46	L-MP-IS-OK-OK	8.8E-08	0.0E+00	8.8E-08
47	L-MP-IS-ST-F	6.8E-08	0.0E+00	6.8E-08
48	L-MP-IS-ST-H	4.5E-08	0.0E+00	4.5E-08
49	L-MP-IS-ST-OK	2.7E-08	0.0E+00	2.7E-08
50	L-MP-WW-LT-F	5.5E-08	1.8E-09	1.6E-08
51	L-MP-WW-LT-H	3.6E-08	1.2E-09	1.1E-08
52	L-MP-WW-LT-OK	2.2E-08	7.2E-10	6.5E-09
53	L-MP-WW-OK-F	6.8E-08	6.8E-09	6.8E-09
54	L-MP-WW-OK-H	4.5E-08	4.5E-09	4.5E-09
55	L-MP-WW-OK-OK	8.9E-09	8.9E-10	8.9E-10
56	L-MP-WW-ST-F	2.3E-08	7.5E-10	6.8E-09
57	L-MP-WW-ST-H	1.5E-08	4.9E-10	4.5E-09
58	L-MP-WW-ST-OK	8.8E-09	2.9E-10	2.6E-09
59	M-HP-DW-LT-F	7.4E-10	1.2E-08	2.5E-09
60	M-HP-DW-LT-H	2.1E-09	3.3E-08	7.0E-09
61	M-HP-DW-LT-OK	1.2E-09	2.0E-08	4.2E-09
62	M-HP-DW-OK-F	3.6E-13	4.0E-11	3.6E-12
63	M-HP-DW-OK-H	1.0E-12	1.1E-10	1.0E-11
64	M-HP-DW-OK-OK	2.0E-13	2.2E-11	2.0E-12
65	M-HP-DW-ST-F	1.5E-09	2.4E-08	5.0E-09
66	M-HP-DW-ST-H	4.3E-09	6.7E-08	1.4E-08
67	M-HP-DW-ST-OK	2.5E-09	4.0E-08	8.5E-09
68	M-HP-IS-LT-F	7.5E-09	0.0E+00	7.5E-09
69	M-HP-IS-LT-H	2.1E-08	0.0E+00	2.1E-08
70	M-HP-IS-LT-OK	1.3E-08	0.0E+00	1.3E-08
71	M-HP-IS-LT-XX	3.3E-08	1.1E-09	3.3E-08
72	M-HP-IS-OK-F	3.6E-11	0.0E+00	3.6E-11
73	M-HP-IS-OK-H	1.0E-10	0.0E+00	1.0E-10
74	M-HP-IS-OK-OK	2.0E-11	0.0E+00	2.0E-11
75	M-HP-IS-OK-XX	5.8E-09	5.8E-10	5.8E-09
76	M-HP-IS-ST-F	1.5E-08	0.0E+00	1.5E-08
77	M-HP-IS-ST-H	4.3E-08	0.0E+00	4.3E-08
78	M-HP-IS-ST-OK	2.6E-08	0.0E+00	2.6E-08
79	M-HP-IS-ST-XX	6.8E-08	2.3E-09	6.8E-08
80	M-HP-WW-LT-F	2.5E-09	8.2E-11	7.4E-10
81	M-HP-WW-LT-H	7.0E-09	2.3E-10	2.1E-09
82	M-HP-WW-LT-OK	4.2E-09	1.4E-10	1.2E-09
83	M-HP-WW-OK-F	3.6E-12	3.6E-13	3.6E-13

Index	Plant Damage State	WWF Venting Strategy	DWF Venting Strategy (Passive)	DWF Venting Strategy (Manual)
84	M-HP-WW-OK-H	1.0E-11	1.0E-12	1.0E-12
85	M-HP-WW-OK-OK	2.0E-12	2.0E-13	2.0E-13
86	M-HP-WW-ST-F	5.0E-09	1.7E-10	1.5E-09
87	M-HP-WW-ST-H	1.4E-08	4.7E-10	4.3E-09
88	M-HP-WW-ST-OK	8.5E-09	2.8E-10	2.5E-09
89	M-LP-DW-LT-F	1.1E-08	1.8E-07	3.8E-08
90	M-LP-DW-LT-H	3.3E-08	5.1E-07	1.1E-07
91	M-LP-DW-OK-F	2.0E-09	2.2E-07	2.0E-08
92	M-LP-DW-OK-H	5.8E-09	6.3E-07	5.8E-08
93	M-LP-DW-ST-F	1.0E-08	1.6E-07	3.4E-08
94	M-LP-DW-ST-H	2.9E-08	4.6E-07	9.7E-08
95	M-LP-IS-LT-F	1.2E-07	0.0E+00	1.2E-07
96	M-LP-IS-LT-H	3.3E-07	0.0E+00	3.3E-07
97	M-LP-IS-LT-XX	6.9E-08	2.3E-09	6.9E-08
98	M-LP-IS-OK-F	2.0E-07	0.0E+00	2.0E-07
99	M-LP-IS-OK-H	5.7E-07	0.0E+00	5.7E-07
100	M-LP-IS-OK-XX	1.2E-08	1.2E-09	1.2E-08
101	M-LP-IS-ST-F	1.0E-07	0.0E+00	1.0E-07
102	M-LP-IS-ST-H	2.9E-07	0.0E+00	2.9E-07
103	M-LP-IS-ST-XX	1.0E-07	3.3E-09	1.0E-07
104	M-LP-WW-LT-F	3.8E-08	1.3E-09	1.1E-08
105	M-LP-WW-LT-H	1.1E-07	3.6E-09	3.3E-08
106	M-LP-WW-OK-F	2.0E-08	2.0E-09	2.0E-09
107	M-LP-WW-OK-H	5.8E-08	5.8E-09	5.8E-09
108	M-LP-WW-ST-F	3.4E-08	1.1E-09	1.0E-08
109	M-LP-WW-ST-H	9.7E-08	3.2E-09	2.9E-08
110	M-MP-DW-LT-F	8.1E-09	1.3E-07	2.7E-08
111	M-MP-DW-LT-H	2.3E-08	3.6E-07	7.7E-08
112	M-MP-DW-LT-OK	1.4E-08	2.2E-07	4.6E-08
113	M-MP-DW-OK-F	4.0E-12	4.4E-10	4.0E-11
114	M-MP-DW-OK-H	1.1E-11	1.2E-09	1.1E-10
115	M-MP-DW-OK-OK	2.2E-12	2.5E-10	2.2E-11
116	M-MP-DW-ST-F	3.3E-09	5.2E-08	1.1E-08
117	M-MP-DW-ST-H	9.5E-09	1.5E-07	3.2E-08
118	M-MP-DW-ST-OK	5.7E-09	8.9E-08	1.9E-08
119	M-MP-IS-LT-F	8.2E-08	0.0E+00	8.2E-08
120	M-MP-IS-LT-H	2.3E-07	0.0E+00	2.3E-07
121	M-MP-IS-LT-OK	1.4E-07	0.0E+00	1.4E-07
122	M-MP-IS-LT-XX	3.7E-07	1.2E-08	3.7E-07
123	M-MP-IS-OK-F	3.9E-10	0.0E+00	3.9E-10
124	M-MP-IS-OK-H	1.1E-09	0.0E+00	1.1E-09
125	M-MP-IS-OK-OK	2.2E-10	0.0E+00	2.2E-10
126	M-MP-IS-OK-XX	6.4E-08	6.4E-09	6.4E-08
127	M-MP-IS-ST-F	3.4E-08	0.0E+00	3.4E-08
128	M-MP-IS-ST-H	9.6E-08	0.0E+00	9.6E-08

Index	Plant Damage State	WWF Venting Strategy	DWF Venting Strategy (Passive)	DWF Venting Strategy (Manual)
129	M-MP-IS-ST-OK	5.7E-08	0.0E+00	5.7E-08
130	M-MP-IS-ST-XX	1.5E-07	5.0E-09	1.5E-07
131	M-MP-WW-LT-F	2.7E-08	9.0E-10	8.1E-09
132	M-MP-WW-LT-H	7.7E-08	2.6E-09	2.3E-08
133	M-MP-WW-LT-OK	4.6E-08	1.5E-09	1.4E-08
134	M-MP-WW-OK-F	4.0E-11	4.0E-12	4.0E-12
135	M-MP-WW-OK-H	1.1E-10	1.1E-11	1.1E-11
136	M-MP-WW-OK-OK	2.2E-11	2.2E-12	2.2E-12
137	M-MP-WW-ST-F	1.1E-08	3.7E-10	3.3E-09
138	M-MP-WW-ST-H	3.2E-08	1.1E-09	9.5E-09
139	M-MP-WW-ST-OK	1.9E-08	6.3E-10	5.7E-09
		8.9E-06	8.4E-06	8.9E-06

APPENDIX H

SOURCE TERM INFORMATION TO MAP EACH MELCOR CASE VARIATION TO A MACCS SOURCE TERM BIN

Table H-1 Mark I source term detailed information

Case	Cesium Release (%)	Iodine Release (%)	Fraction of Cesium Release Through Vented Pathway	Fraction of Iodine Release Through Vented Pathway	Start of Release to Environment (hours)	Source Term Bin
1	1.93%	22.70%	78.2%	85.5%	14.9	12
1DF10	0.57%	5.24%				7
1DF100	0.44%	3.49%				7
1DF1000	0.42%	3.32%				7
1S1	0.65%	6.33%	55.4%	54.0%	23.3	7
1S1DF10	0.33%	3.25%				7
1S1DF100	0.29%	2.94%				6
1S1DF1000	0.29%	2.91%				6
2	3.06%	26.40%	53.9%	83.0%	14.9	15
2DF10	1.58%	6.69%				10
2DF100	1.43%	4.72%				10
2DF1000	1.41%	4.52%				10
3	9.88%	30.20%	11.5%	21.6%	9.8	17
3DF10	8.85%	24.32%				17
3DF100	8.75%	23.74%				17
3DF1000	8.74%	23.68%				17
4	1.86%	9.85%	27.3%	59.2%	18.1	10
4DF10	1.40%	4.60%				10
4DF100	1.36%	4.08%				10
4DF1000	1.35%	4.03%				10
5	1.39%	6.43%	27.6%	61.7%	24.2	10
5DF10	1.04%	2.86%				9
5DF100	1.01%	2.50%				9
5DF1000	1.01%	2.46%				9
6	1.97%	20.30%	68.5%	80.8%	13.9	12
6DF10	0.76%	5.54%				7
6DF100	0.63%	4.06%				7
6DF1000	0.62%	3.92%				7
7	1.51%	19.40%	98.7%	99.0%	14.9	11
7DF10	0.17%	2.12%				6
7DF100	0.03%	0.39%				5
7DF1000	0.02%	0.22%				4
7dw	1.62%	17.70%	100.0%	100.0%	14.9	11
7dwDF10	0.16%	1.77%				6
7dwDF100	0.02%	0.18%				4
7dwDF1000	0.002%	0.02%				2

Case	Cesium Release (%)	Iodine Release (%)	Fraction of Cesium Release Through Vented Pathway	Fraction of Iodine Release Through Vented Pathway	Start of Release to Environment (hours)	Source Term Bin
8	1.49%	19.20%	100.0%	100.0%	14.9	11
8DF10	0.15%	1.92%				6
8DF100	0.01%	0.19%				4
8DF1000	0.001%	0.02%				2
9	0.61%	7.84%	100.0%	100.0%	14.4	7
9DF10	0.06%	0.78%				5
9DF100	0.01%	0.08%				3
9DF1000	0.001%	0.01%				1
10	0.72%	8.04%	100.0%	100.0%	16.3	7
10DF10	0.07%	0.80%				5
10DF100	0.01%	0.08%				3
10DF1000	0.001%	0.01%				1
11	0.61%	7.79%	99.7%	100.0%	14.4	7
11DF10	0.06%	0.78%				5
11DF100	0.01%	0.08%				3
11DF1000	0.003%	0.01%				2
12	1.34%	16.40%	100.0%	100.0%	13.9	11
12DF10	0.13%	1.64%				6
12DF100	0.01%	0.16%				4
12DF1000	0.001%	0.02%				2
13	0.35%	3.90%	100.0%	100.0%	24.2	7
13DF10	0.04%	0.39%				5
13DF100	0.004%	0.04%				3
13DF1000	0.0004%	0.004%				1
14	0.62%	6.03%	97.1%	95.9%	14.9	7
14DF10	0.08%	0.83%				5
14DF100	0.02%	0.31%				4
14DF1000	0.019%	0.26%				4
15	0.61%	5.83%	100.0%	100.0%	14.9	7
15DF10	0.06%	0.58%				5
15DF100	0.01%	0.06%				3
15DF1000	0.001%	0.01%				1
16	0.61%	5.88%	99.7%	99.9%	14.9	7
16DF10	0.06%	0.60%				5
16DF100	0.01%	0.07%				3
16DF1000	0.003%	0.01%				2

Case	Cesium Release (%)	Iodine Release (%)	Fraction of Cesium Release Through Vented Pathway	Fraction of Iodine Release Through Vented Pathway	Start of Release to Environme nt (hours)	Source Term Bin
18	0.41%	4.97%	98.8%	100.0%	16.3	7
18DF10	0.05%	0.50%				5
18DF100	0.01%	0.05%				3
18DF1000	0.005%	0.005%				3
21	1.58%	19.90%	97.5%	97.5%	14.9	11
21DF10	0.19%	2.44%				6
21DF100	0.06%	0.69%				5
21DF1000	0.04%	0.52%				5
22	1.78%	22.30%	100.0%	100.0%	14.9	12
22DF10	0.18%	2.23%				6
22DF100	0.02%	0.22%				4
22DF1000	0.002%	0.02%				2
22dw	2.82%	18.60%	100.0%	100.0%	14.9	14
22dwDF10	0.28%	1.86%				6
22dwDF100	0.03%	0.19%				4
22dwDF1000	0.003%	0.02%				2
24	0.84%	8.24%	99.8%	99.8%	16.3	7
24DF10	0.09%	0.84%				5
24DF100	0.01%	0.10%				4
24DF1000	0.003%	0.03%				2
24dw	2.20%	8.59%	100.0%	100.0%	16	13
24dwDF10	0.22%	0.86%				6
24dwDF100	0.02%	0.09%				4
24dwDF1000	0.002%	0.01%				2
27	0.63%	6.03%	96.7%	95.9%	14.9	7
27DF10	0.08%	0.83%				5
27DF100	0.03%	0.31%				4
27DF1000	0.02%	0.26%				4
28	0.61%	5.78%	100.0%	100.0%	14.9	7
28DF10	0.06%	0.58%				5
28DF100	0.01%	0.06%				3
28DF1000	0.001%	0.01%				1
30	0.62%	5.88%	99.7%	100.0%	14.9	7
30DF10	0.06%	0.59%				5
30DF100	0.01%	0.06%				3
30DF1000	0.002%	0.01%				2

Case	Cesium Release (%)	Iodine Release (%)	Fraction of Cesium Release Through Vented Pathway	Fraction of Iodine Release Through Vented Pathway	Start of Release to Environment (hours)	Source Term Bin
32	0.39%	4.85%	100.0%	100.0%	16.3	7
32DF10	0.04%	0.49%				5
32DF100	0.004%	0.05%				3
32DF1000	0.0004%	0.01%				1
41	4.58%	14.10%	25.1%	46.0%	9.8	16
41DF10	3.55%	8.27%				13
41DF100	3.44%	7.68%				13
41DF1000	3.43%	7.63%				13
42	4.33%	10.70%	26.6%	60.6%	9.8	16
42DF10	3.30%	4.87%				13
42DF100	3.19%	4.28%				13
42DF1000	3.18%	4.23%				13
43	4.77%	16.00%	24.9%	40.5%	9.8	16
43DF10	3.70%	10.17%				13
43DF100	3.59%	9.58%				13
43DF1000	3.58%	9.53%				13
44	4.40%	10.80%	27.0%	60.0%	9.8	16
44DF10	3.33%	4.97%				13
44DF100	3.22%	4.38%				13
44DF1000	3.21%	4.33%				13
45	0.90%	9.60%	100.0%	100.0%	14.8	7
45DF10	0.09%	0.96%				5
45DF100	0.01%	0.10%				3
45DF1000	0.001%	0.01%				1
46	0.98%	11.00%	100.0%	100.0%	14.8	8
46DF10	0.10%	1.10%				5
46DF100	0.01%	0.11%				3
46DF1000	0.001%	0.01%				2
47	0.19%	1.24%	100.0%	100.0%	11.4	6
47DF10	0.02%	0.12%				4
47DF100	0.002%	0.01%				2
47DF1000	0.0002%	0.001%				1
48	0.24%	1.69%	100.0%	100.0%	11.4	6
48DF10	0.02%	0.17%				4
48DF100	0.002%	0.02%				2
48DF1000	0.0002%	0.002%				1

Case	Cesium Release (%)	Iodine Release (%)	Fraction of Cesium Release Through Vented Pathway	Fraction of Iodine Release Through Vented Pathway	Start of Release to Environment (hours)	Source Term Bin
49	0.53%	1.67%	99.6%	100.0%	7.3	7
49DF10	0.06%	0.17%				5
49DF100	0.01%	0.02%				3
49DF1000	0.002%	0.002%				2
50	0.21%	1.09%	99.8%	100.0%	7.3	6
50DF10	0.02%	0.11%				4
50DF100	0.003%	0.01%				2
50DF1000	0.001%	0.001%				1
51	0.93%	10.20%	100.0%	100.0%	14.9	8
51DF10	0.09%	1.02%				5
51DF100	0.009%	0.10%				3
51DF1000	0.0009%	0.01%				1
52	15.90%	34.30%	83.6%	77.0%	18.6	18
52DF10	3.93%	10.54%				13
52DF100	2.73%	8.16%				13
52DF1000	2.61%	7.93%				13
53	2.79%	29.10%	100.0%	100.0%	18.6	15
53DF10	0.28%	2.91%				6
53DF100	0.03%	0.29%				4
53DF1000	0.003%	0.029%				2

Table H-2 Mark II source term detailed information

Case	Cesium Release to Environment (%)	Iodine Release to Environment (%)	Start of Release to Environment (hours)	Fraction of Cesium and Iodine Release Through Vented Pathway	Source Term Bin
1	2.46%	19.81%	22.8	100%	8
1DF10	0.246%	1.981%			5
1DF100	0.025%	0.198%			4
1DF1000	0.002%	0.020%			3
3	1.09%	10.26%	14.3	100%	7
3DF10	0.109%	1.026%			5
3DF100	0.011%	0.103%			4
3DF1000	0.001%	0.010%			3
5	0.55%	4.94%	32.2	100%	6
5DF10	0.055%	0.494%			4
5DF100	0.006%	0.049%			3
5DF1000	0.001%	0.005%			2
10	0.23%	2.67%	22.2	100%	5
10DF10	0.023%	0.267%			4
10DF100	0.002%	0.027%			3
10DF1000	0.0002%	0.003%			2
11	0.04%	0.45%	20.3	100%	4
11DF10	0.004%	0.045%			3
11DF100	0.0004%	0.005%			2
11DF1000	0.00004%	0.0005%			1
24	0.71%	5.79%	30.5	100%	6
24DF10	0.071%	0.579%			4
24DF100	0.007%	0.058%			3
24DF1000	0.001%	0.006%			2
42	0.43%	3.65%	14.3	100%	6
42DF10	0.043%	0.365%			4
42DF100	0.004%	0.037%			3
42DF1000	0.0004%	0.004%			2
44	0.44%	3.68%	14.3	100%	6
44DF10	0.044%	0.368%			4
44DF100	0.004%	0.037%			3
44DF1000	0.0004%	0.004%			2

Case	Cesium Release to Environment (%)	Iodine Release to Environment (%)	Start of Release to Environment (hours)	Fraction of Cesium and Iodine Release Through Vented Pathway	Source Term Bin
45	2.29%	19.26%	18.3	100%	8
45DF10	0.229%	1.926%			5
45DF100	0.023%	0.193%			4
45DF1000	0.002%	0.019%			3
49	0.65%	7.62%	11.2	100%	6
49DF10	0.065%	0.762%			4
49DF100	0.006%	0.076%			3
49DF1000	0.001%	0.008%			2
51	2.29%	20.14%	16.6	100%	8
51DF10	0.229%	2.014%			5
51DF100	0.023%	0.201%			4
51DF1000	0.002%	0.020%			3
52	3.57%	28.67%	16.6	100%	9
52DF10	0.357%	2.867%			5
52DF100	0.036%	0.287%			4
52DF1000	0.004%	0.029%			3

APPENDIX I

MACCS SENSITIVITY CALCULATION RESULTS

Table I-1 Sensitivity calculation results for Mark I source terms with low population site file (Hatch)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	Mark I - Peach Bottom	Low - Hatch	Mk I - Low (Bin 3)	Base Case	0.00E+00	6.63E-06	9.74E-07	3.33E-07	5.83E+03	1.32E+04	2.11E+07	2.16E+07	6.38E+00	7.22E+00	2.73E-01	2.73E-01
2				1-hour Evacuation Delay		6.63E-06	9.74E-07	3.33E-07	5.83E+03	1.32E+04	2.11E+07	2.16E+07	6.38E+00	7.22E+00	2.73E-01	2.73E-01
3				No Intermediate Phase		7.53E-06	1.06E-06	3.63E-07	5.91E+03	1.33E+04	2.05E+07	2.10E+07	6.42E+00	7.26E+00	3.08E+00	3.08E+00
4				1-year Intermediate Phase		5.82E-06	8.63E-07	2.95E-07	5.73E+03	1.30E+04	2.06E+07	2.11E+07	6.38E+00	7.22E+00	1.11E-03	1.11E-03
5				EPA Habitability Criterion (2 rem per year)		7.03E-06	9.84E-07	3.36E-07	5.84E+03	1.32E+04	2.04E+07	2.09E+07	6.38E+00	7.22E+00	0.00E+00	0.00E+00
6				No Intermediate Phase and EPA Hab.		7.84E-06	1.07E-06	3.64E-07	5.92E+03	1.33E+04	2.04E+07	2.09E+07	6.38E+00	7.22E+00	1.26E-01	1.26E-01
7				1-year Intermediate Phase and EPA Hab.		5.93E-06	8.65E-07	2.95E-07	5.74E+03	1.30E+04	2.04E+07	2.09E+07	6.38E+00	7.22E+00	1.11E-03	1.11E-03
8				Larger Nonevacuating Cohort		6.75E-06	9.76E-07	3.34E-07	5.83E+03	1.32E+04	2.07E+07	2.12E+07	6.38E+00	7.22E+00	2.73E-01	2.73E-01
9			Mk I - Med (Bin 10)	Base Case	0.00E+00	3.98E-04	1.17E-04	4.44E-05	2.33E+05	5.74E+05	1.12E+09	1.46E+09	3.73E+02	6.63E+02	7.19E+03	7.21E+03
10				1-hour Evacuation Delay		3.98E-04	1.17E-04	4.44E-05	2.33E+05	5.74E+05	1.12E+09	1.46E+09	3.73E+02	6.63E+02	7.19E+03	7.21E+03
11				No Intermediate Phase		3.46E-04	1.16E-04	4.65E-05	2.33E+05	5.85E+05	7.11E+08	8.81E+08	5.39E+02	8.39E+02	1.76E+04	1.85E+04
12				1-year Intermediate Phase		4.25E-04	1.09E-04	4.04E-05	2.25E+05	5.56E+05	1.54E+09	1.73E+09	3.21E+02	6.11E+02	3.43E+03	3.43E+03
13				EPA Habitability Criterion (2 rem per year)		9.11E-04	1.55E-04	5.30E-05	2.67E+05	6.11E+05	4.27E+08	5.79E+08	2.96E+02	5.85E+02	1.51E+03	1.51E+03
14				No Intermediate Phase and EPA Hab.		8.91E-04	1.67E-04	5.76E-05	2.78E+05	6.31E+05	2.98E+08	4.45E+08	3.26E+02	6.16E+02	3.70E+03	3.70E+03
15				1-year Intermediate Phase and EPA Hab.		5.19E-04	1.24E-04	4.36E-05	2.38E+05	5.70E+05	6.62E+08	8.09E+08	3.21E+02	6.11E+02	3.43E+03	3.43E+03
16				Larger Nonevacuating Cohort		4.02E-04	1.17E-04	4.44E-05	2.33E+05	5.74E+05	1.12E+09	1.46E+09	3.73E+02	6.63E+02	7.19E+03	7.21E+03
17			Mk I - High (Bin 17)	Base Case	0.00E+00	7.12E-04	3.32E-04	1.69E-04	5.65E+05	1.53E+06	4.22E+09	7.28E+09	1.22E+03	2.52E+03	4.19E+04	5.25E+04
18				1-hour Evacuation Delay		7.12E-04	3.32E-04	1.69E-04	5.65E+05	1.53E+06	4.22E+09	7.28E+09	1.22E+03	2.52E+03	4.19E+04	5.25E+04
19				No Intermediate Phase		6.66E-04	3.07E-04	1.67E-04	5.55E+05	1.54E+06	2.80E+09	4.54E+09	1.67E+03	3.26E+03	7.22E+04	1.12E+05
20				1-year Intermediate Phase		7.40E-04	3.26E-04	1.57E-04	5.52E+05	1.47E+06	6.79E+09	1.01E+10	9.72E+02	2.20E+03	2.51E+04	2.79E+04
21				EPA Habitability Criterion (2 rem per year)		1.72E-03	5.41E-04	2.27E-04	7.50E+05	1.78E+06	1.76E+09	2.72E+09	8.12E+02	2.02E+03	1.33E+04	1.40E+04
22				No Intermediate Phase and EPA Hab.		1.76E-03	5.59E-04	2.43E-04	7.75E+05	1.85E+06	1.14E+09	1.87E+09	9.40E+02	2.17E+03	2.31E+04	2.65E+04
23				1-year Intermediate Phase and EPA Hab.		8.26E-04	3.81E-04	1.76E-04	6.01E+05	1.55E+06	3.71E+09	4.75E+09	9.72E+02	2.20E+03	2.51E+04	2.79E+04
24				Larger Nonevacuating Cohort		7.17E-04	3.32E-04	1.69E-04	5.65E+05	1.53E+06	4.22E+09	7.28E+09	1.22E+03	2.52E+03	4.19E+04	5.25E+04

Table I-2 Sensitivity calculation results for Mark I source terms with medium population site file (Vermont Yankee)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
25	Mark I - Peach Bottom	Medium - Vermont Yankee	Mk I - Low (Bin 3)	Base Case	0.00E+00	1.01E-05	9.53E-07	2.99E-07	5.35E+03	1.56E+04	5.89E+07	5.94E+07	2.50E+00	3.14E+00	1.69E+00	1.69E+00
26				1-hour Evacuation Delay		1.01E-05	9.53E-07	2.99E-07	5.35E+03	1.56E+04	5.89E+07	5.94E+07	2.50E+00	3.14E+00	1.69E+00	1.69E+00
27				No Intermediate Phase		1.13E-05	1.04E-06	3.25E-07	5.61E+03	1.62E+04	5.56E+07	5.62E+07	2.56E+00	3.20E+00	2.76E+01	2.76E+01
28				1-year Intermediate Phase		8.86E-06	8.42E-07	2.64E-07	5.02E+03	1.48E+04	5.65E+07	5.71E+07	2.49E+00	3.14E+00	2.35E-01	2.35E-01
29				EPA Habitability Criterion (2 rem per year)		1.08E-05	9.70E-07	3.01E-07	5.40E+03	1.57E+04	5.50E+07	5.55E+07	2.49E+00	3.14E+00	1.26E-02	1.26E-02
30				No Intermediate Phase and EPA Hab.		1.21E-05	1.06E-06	3.27E-07	5.67E+03	1.63E+04	5.48E+07	5.53E+07	2.50E+00	3.14E+00	2.70E-01	2.70E-01
31				1-year Intermediate Phase and EPA Hab.		9.16E-06	8.49E-07	2.65E-07	5.04E+03	1.48E+04	5.48E+07	5.54E+07	2.49E+00	3.14E+00	2.35E-01	2.35E-01
32				Larger Nonevacuating Cohort		1.03E-05	9.57E-07	2.99E-07	5.37E+03	1.56E+04	5.76E+07	5.81E+07	2.50E+00	3.14E+00	1.69E+00	1.69E+00
33			Mk I - Med (Bin 10)	Base Case	0.00E+00	5.00E-04	1.15E-04	4.29E-05	4.38E+05	1.36E+06	3.45E+09	5.25E+09	2.49E+02	4.80E+02	2.09E+04	2.11E+04
34				1-hour Evacuation Delay		5.00E-04	1.15E-04	4.29E-05	4.38E+05	1.36E+06	3.45E+09	5.25E+09	2.49E+02	4.80E+02	2.09E+04	2.11E+04
35				No Intermediate Phase		4.41E-04	1.15E-04	4.63E-05	4.47E+05	1.45E+06	2.09E+09	2.50E+09	4.41E+02	6.83E+02	5.39E+04	6.15E+04
36				1-year Intermediate Phase		5.30E-04	1.06E-04	3.83E-05	4.09E+05	1.25E+06	4.80E+09	5.45E+09	1.83E+02	4.13E+02	1.20E+04	1.20E+04
37				EPA Habitability Criterion (2 rem per year)		1.19E-03	1.48E-04	4.81E-05	5.42E+05	1.48E+06	1.06E+09	1.29E+09	1.47E+02	3.77E+02	6.71E+03	6.71E+03
38				No Intermediate Phase and EPA Hab.		1.16E-03	1.61E-04	5.27E-05	5.84E+05	1.60E+06	6.03E+08	7.82E+08	1.87E+02	4.17E+02	1.24E+04	1.24E+04
39				1-year Intermediate Phase and EPA Hab.		6.31E-04	1.18E-04	4.02E-05	4.47E+05	1.29E+06	2.19E+09	2.37E+09	1.83E+02	4.13E+02	1.20E+04	1.20E+04
40				Larger Nonevacuating Cohort		5.05E-04	1.15E-04	4.29E-05	4.39E+05	1.36E+06	3.45E+09	5.25E+09	2.49E+02	4.80E+02	2.09E+04	2.11E+04
41			Mk I - High (Bin 17)	Base Case	0.00E+00	8.79E-04	3.49E-04	1.82E-04	1.28E+06	5.10E+06	1.51E+10	3.60E+10	9.96E+02	2.08E+03	1.30E+05	2.19E+05
42				1-hour Evacuation Delay		8.79E-04	3.49E-04	1.82E-04	1.28E+06	5.10E+06	1.51E+10	3.60E+10	9.96E+02	2.08E+03	1.30E+05	2.19E+05
43				No Intermediate Phase		8.47E-04	3.23E-04	1.83E-04	1.24E+06	5.16E+06	9.89E+09	2.10E+10	1.51E+03	2.92E+03	2.44E+05	5.75E+05
44				1-year Intermediate Phase		8.98E-04	3.37E-04	1.66E-04	1.23E+06	4.71E+06	2.39E+10	4.79E+10	7.08E+02	1.70E+03	7.66E+04	1.01E+05
45				EPA Habitability Criterion (2 rem per year)		2.12E-03	5.44E-04	2.28E-04	1.87E+06	6.18E+06	5.16E+09	8.74E+09	5.07E+02	1.48E+03	3.99E+04	4.50E+04
46				No Intermediate Phase and EPA Hab.		2.23E-03	5.67E-04	2.48E-04	1.97E+06	6.68E+06	3.03E+09	4.72E+09	6.68E+02	1.67E+03	7.14E+04	9.89E+04
47				1-year Intermediate Phase and EPA Hab.		9.72E-04	3.92E-04	1.84E-04	1.40E+06	5.14E+06	1.31E+10	1.80E+10	7.08E+02	1.70E+03	7.66E+04	1.01E+05
48				Larger Nonevacuating Cohort		8.88E-04	3.49E-04	1.82E-04	1.28E+06	5.10E+06	1.51E+10	3.60E+10	9.96E+02	2.08E+03	1.30E+05	2.19E+05

Table I-3 Sensitivity calculation results for Mark I source terms with high population site file (Peach Bottom)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
49	Mark I - Peach Bottom	High - Peach Bottom	Mk I - Low (Bin 3)	Base Case	0.00E+00	6.25E-06	7.16E-07	3.21E-07	1.65E+04	2.73E+04	9.81E+07	9.87E+07	9.80E+00	1.05E+01	5.65E-01	5.65E-01
50				1-hour Evacuation Delay		6.25E-06	7.16E-07	3.21E-07	1.65E+04	2.73E+04	9.81E+07	9.87E+07	9.80E+00	1.05E+01	5.65E-01	5.65E-01
51				No Intermediate Phase		7.12E-06	7.78E-07	3.49E-07	1.72E+04	2.84E+04	9.52E+07	9.58E+07	9.83E+00	1.05E+01	1.14E+01	1.14E+01
52				1-year Intermediate Phase		5.47E-06	6.35E-07	2.84E-07	1.56E+04	2.59E+04	9.55E+07	9.61E+07	9.80E+00	1.05E+01	9.12E-02	9.12E-02
53				EPA Habitability Criterion (2 rem per year)		6.60E-06	7.20E-07	3.22E-07	1.65E+04	2.73E+04	9.48E+07	9.54E+07	9.80E+00	1.05E+01	1.46E-02	1.46E-02
54				No Intermediate Phase and EPA Hab.		7.37E-06	7.80E-07	3.49E-07	1.72E+04	2.84E+04	9.48E+07	9.54E+07	9.80E+00	1.05E+01	1.31E-01	1.31E-01
55				1-year Intermediate Phase and EPA Hab.		5.56E-06	6.36E-07	2.85E-07	1.56E+04	2.59E+04	9.48E+07	9.54E+07	9.80E+00	1.05E+01	9.12E-02	9.12E-02
56				Larger Nonevacuating Cohort		6.36E-06	7.17E-07	3.22E-07	1.65E+04	2.73E+04	9.64E+07	9.70E+07	9.80E+00	1.05E+01	5.65E-01	5.65E-01
57			Mk I - Med (Bin 10)	Base Case	0.00E+00	4.06E-04	9.78E-05	4.51E-05	1.36E+06	2.30E+06	9.90E+09	1.20E+10	4.80E+02	7.14E+02	5.14E+04	5.15E+04
58				1-hour Evacuation Delay		4.06E-04	9.78E-05	4.51E-05	1.36E+06	2.30E+06	9.90E+09	1.20E+10	4.80E+02	7.14E+02	5.14E+04	5.15E+04
59				No Intermediate Phase		3.49E-04	9.80E-05	4.71E-05	1.37E+06	2.38E+06	6.16E+09	6.72E+09	6.20E+02	8.65E+02	1.60E+05	1.72E+05
60				1-year Intermediate Phase		4.34E-04	8.97E-05	4.07E-05	1.26E+06	2.12E+06	1.30E+10	1.39E+10	4.36E+02	6.71E+02	2.02E+04	2.02E+04
61				EPA Habitability Criterion (2 rem per year)		9.32E-04	1.21E-04	5.24E-05	1.62E+06	2.58E+06	2.64E+09	2.95E+09	4.16E+02	6.50E+02	7.01E+03	7.01E+03
62				No Intermediate Phase and EPA Hab.		9.06E-04	1.32E-04	5.74E-05	1.75E+06	2.78E+06	1.60E+09	1.80E+09	4.41E+02	6.76E+02	2.39E+04	2.39E+04
63				1-year Intermediate Phase and EPA Hab.		5.35E-04	1.01E-04	4.42E-05	1.39E+06	2.25E+06	4.41E+09	4.62E+09	4.36E+02	6.71E+02	2.02E+04	2.02E+04
64				Larger Nonevacuating Cohort		4.11E-04	9.79E-05	4.51E-05	1.36E+06	2.30E+06	9.90E+09	1.20E+10	4.80E+02	7.14E+02	5.14E+04	5.15E+04
65			Mk I - High (Bin 17)	Base Case	0.00E+00	7.10E-04	2.95E-04	1.68E-04	3.83E+06	7.72E+06	4.69E+10	6.79E+10	1.36E+03	2.47E+03	4.17E+05	5.04E+05
66				1-hour Evacuation Delay		7.10E-04	2.95E-04	1.68E-04	3.83E+06	7.72E+06	4.69E+10	6.79E+10	1.36E+03	2.47E+03	4.17E+05	5.04E+05
67				No Intermediate Phase		6.66E-04	2.72E-04	1.64E-04	3.66E+06	7.66E+06	3.04E+10	4.19E+10	1.73E+03	3.13E+03	7.71E+05	1.10E+06
68				1-year Intermediate Phase		7.40E-04	2.87E-04	1.57E-04	3.72E+06	7.26E+06	7.49E+10	9.88E+10	1.16E+03	2.19E+03	2.29E+05	2.55E+05
69				EPA Habitability Criterion (2 rem per year)		1.74E-03	4.62E-04	2.29E-04	5.68E+06	1.01E+07	1.57E+10	1.93E+10	1.04E+03	2.04E+03	1.10E+05	1.17E+05
70				No Intermediate Phase and EPA Hab.		1.77E-03	4.84E-04	2.45E-04	5.97E+06	1.07E+07	9.12E+09	1.10E+10	1.14E+03	2.17E+03	2.07E+05	2.37E+05
71				1-year Intermediate Phase and EPA Hab.		8.27E-04	3.38E-04	1.78E-04	4.27E+06	8.06E+06	3.95E+10	4.49E+10	1.16E+03	2.19E+03	2.29E+05	2.55E+05
72				Larger Nonevacuating Cohort		7.15E-04	2.95E-04	1.68E-04	3.83E+06	7.72E+06	4.69E+10	6.79E+10	1.36E+03	2.47E+03	4.17E+05	5.04E+05

Table I-4 Sensitivity calculation results for Mark II source terms with low population site file (Columbia)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
73	Mark II - Limerick	Low - Columbia	Mk II - Low (Bin 2)	Base Case	0.00E+00	7.05E-07	1.64E-07	9.15E-08	1.86E+03	2.42E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
74				1-hour Evacuation Delay		7.05E-07	1.64E-07	9.15E-08	1.86E+03	2.42E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
75				No Intermediate Phase		7.55E-07	1.72E-07	9.58E-08	1.87E+03	2.43E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
76				1-year Intermediate Phase		6.34E-07	1.53E-07	8.54E-08	1.85E+03	2.40E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
77				EPA Habitability Criterion (2 rem per year)		7.05E-07	1.64E-07	9.15E-08	1.86E+03	2.42E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
78				No Intermediate Phase and EPA Hab.		7.55E-07	1.72E-07	9.58E-08	1.87E+03	2.43E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
79				1-year Intermediate Phase and EPA Hab.		6.34E-07	1.53E-07	8.54E-08	1.85E+03	2.40E+03	1.86E+07	1.86E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
80				Larger Nonevacuating Cohort		7.32E-07	1.64E-07	9.16E-08	1.86E+03	2.42E+03	1.84E+07	1.84E+07	4.10E-04	4.10E-04	0.00E+00	0.00E+00
81			Mk II - Med (Bin 5)	Base Case	0.00E+00	1.15E-04	3.29E-05	1.86E-05	1.06E+05	1.84E+05	3.63E+08	4.12E+08	2.52E+02	3.64E+02	5.31E+02	5.31E+02
82				1-hour Evacuation Delay		1.15E-04	3.29E-05	1.86E-05	1.06E+05	1.84E+05	3.63E+08	4.12E+08	2.52E+02	3.64E+02	5.31E+02	5.31E+02
83				No Intermediate Phase		9.98E-05	3.38E-05	1.93E-05	1.06E+05	1.86E+05	2.54E+08	3.02E+08	2.58E+02	3.70E+02	2.92E+03	2.92E+03
84				1-year Intermediate Phase		1.10E-04	3.04E-05	1.70E-05	1.03E+05	1.81E+05	3.63E+08	4.11E+08	2.51E+02	3.62E+02	1.54E+02	1.54E+02
85				EPA Habitability Criterion (2 rem per year)		1.72E-04	3.76E-05	2.08E-05	1.10E+05	1.88E+05	1.88E+08	2.36E+08	2.50E+02	3.62E+02	2.62E+01	2.62E+01
86				No Intermediate Phase and EPA Hab.		1.89E-04	4.16E-05	2.30E-05	1.13E+05	1.93E+05	1.71E+08	2.19E+08	2.51E+02	3.62E+02	9.77E+01	9.77E+01
87				1-year Intermediate Phase and EPA Hab.		1.39E-04	3.30E-05	1.82E-05	1.05E+05	1.83E+05	1.86E+08	2.34E+08	2.51E+02	3.62E+02	1.54E+02	1.54E+02
88				Larger Nonevacuating Cohort		1.18E-04	3.29E-05	1.86E-05	1.06E+05	1.84E+05	3.63E+08	4.12E+08	2.52E+02	3.64E+02	5.31E+02	5.31E+02
89			Mk II - High (Bin 8)	Base Case	0.00E+00	4.40E-04	3.06E-04	1.84E-04	5.65E+05	8.87E+05	4.63E+09	6.12E+09	1.23E+03	2.25E+03	4.03E+04	4.34E+04
90				1-hour Evacuation Delay		4.40E-04	3.06E-04	1.84E-04	5.65E+05	8.87E+05	4.63E+09	6.12E+09	1.23E+03	2.25E+03	4.03E+04	4.34E+04
91				No Intermediate Phase		3.65E-04	2.77E-04	1.72E-04	5.87E+05	9.16E+05	3.82E+09	5.09E+09	1.59E+03	3.26E+03	8.94E+04	1.17E+05
92				1-year Intermediate Phase		5.20E-04	3.01E-04	1.80E-04	5.54E+05	8.72E+05	6.96E+09	9.09E+09	1.07E+03	2.03E+03	8.20E+03	8.22E+03
93				EPA Habitability Criterion (2 rem per year)		9.87E-04	4.13E-04	2.42E-04	6.65E+05	1.00E+06	2.47E+09	3.25E+09	1.10E+03	2.07E+03	1.30E+04	1.31E+04
94				No Intermediate Phase and EPA Hab.		8.80E-04	4.18E-04	2.54E-04	7.11E+05	1.07E+06	2.05E+09	2.65E+09	1.29E+03	2.38E+03	4.22E+04	4.86E+04
95				1-year Intermediate Phase and EPA Hab.		6.17E-04	3.55E-04	2.16E-04	6.03E+05	9.40E+05	4.00E+09	4.81E+09	1.07E+03	2.03E+03	8.20E+03	8.22E+03
96				Larger Nonevacuating Cohort		4.96E-04	3.06E-04	1.84E-04	5.65E+05	8.87E+05	4.63E+09	6.12E+09	1.23E+03	2.25E+03	4.03E+04	4.34E+04

Table I-5 Sensitivity calculation results for Mark II source terms with medium population site file (Susquehanna)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
97	Mark II - Limerick	Medium - Susquehanna	Mk II - Low (Bin 2)	Base Case	0.00E+00	8.47E-07	1.52E-07	4.46E-08	1.29E+03	2.43E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
98				1-hour Evacuation Delay		8.47E-07	1.52E-07	4.46E-08	1.29E+03	2.43E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
99				No Intermediate Phase		9.10E-07	1.60E-07	4.68E-08	1.33E+03	2.50E+03	9.12E+07	9.12E+07	1.64E-03	1.64E-03	0.00E+00	0.00E+00
100				1-year Intermediate Phase		7.63E-07	1.41E-07	4.15E-08	1.24E+03	2.32E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
101				EPA Habitability Criterion (2 rem per year)		8.49E-07	1.52E-07	4.46E-08	1.29E+03	2.43E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
102				No Intermediate Phase and EPA Hab.		9.10E-07	1.60E-07	4.68E-08	1.33E+03	2.50E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
103				1-year Intermediate Phase and EPA Hab.		7.63E-07	1.41E-07	4.15E-08	1.24E+03	2.32E+03	9.12E+07	9.12E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
104				Larger Nonevacuating Cohort		8.79E-07	1.53E-07	4.47E-08	1.30E+03	2.43E+03	8.83E+07	8.83E+07	1.61E-03	1.61E-03	0.00E+00	0.00E+00
105			Mk II - Med (Bin 5)	Base Case	0.00E+00	1.08E-04	2.82E-05	9.18E-06	1.46E+05	3.60E+05	8.43E+08	9.58E+08	1.00E+02	2.03E+02	3.37E+03	3.37E+03
106				1-hour Evacuation Delay		1.08E-04	2.82E-05	9.18E-06	1.46E+05	3.60E+05	8.43E+08	9.58E+08	1.00E+02	2.03E+02	3.37E+03	3.37E+03
107				No Intermediate Phase		9.03E-05	2.86E-05	9.75E-06	1.48E+05	3.75E+05	4.93E+08	5.58E+08	1.24E+02	2.26E+02	1.12E+04	1.12E+04
108				1-year Intermediate Phase		1.06E-04	2.61E-05	8.23E-06	1.38E+05	3.37E+05	9.91E+08	1.06E+09	9.47E+01	1.97E+02	1.48E+03	1.48E+03
109				EPA Habitability Criterion (2 rem per year)		1.81E-04	3.38E-05	1.00E-05	1.66E+05	3.81E+05	2.86E+08	3.51E+08	9.16E+01	1.94E+02	3.71E+02	3.71E+02
110				No Intermediate Phase and EPA Hab.		1.86E-04	3.66E-05	1.09E-05	1.76E+05	4.02E+05	2.06E+08	2.71E+08	9.48E+01	1.97E+02	1.42E+03	1.42E+03
111				1-year Intermediate Phase and EPA Hab.		1.35E-04	2.86E-05	8.60E-06	1.47E+05	3.45E+05	3.77E+08	4.42E+08	9.47E+01	1.97E+02	1.48E+03	1.48E+03
112				Larger Nonevacuating Cohort		1.11E-04	2.83E-05	9.20E-06	1.46E+05	3.61E+05	8.40E+08	9.55E+08	1.00E+02	2.03E+02	3.37E+03	3.37E+03
113			Mk II - High (Bin 8)	Base Case	0.00E+00	3.93E-04	2.61E-04	1.08E-04	1.16E+06	3.29E+06	1.42E+10	4.04E+10	7.48E+02	1.70E+03	1.21E+05	1.48E+05
114				1-hour Evacuation Delay		3.93E-04	2.61E-04	1.08E-04	1.16E+06	3.29E+06	1.42E+10	4.04E+10	7.48E+02	1.70E+03	1.21E+05	1.48E+05
115				No Intermediate Phase		3.34E-04	2.42E-04	1.13E-04	1.27E+06	3.64E+06	1.10E+10	2.81E+10	1.33E+03	2.93E+03	3.17E+05	8.95E+05
116				1-year Intermediate Phase		4.63E-04	2.56E-04	1.05E-04	1.13E+06	3.20E+06	2.31E+10	6.01E+10	4.67E+02	1.37E+03	3.16E+04	3.18E+04
117				EPA Habitability Criterion (2 rem per year)		9.24E-04	3.53E-04	1.34E-04	1.50E+06	3.92E+06	6.60E+09	1.49E+10	4.95E+02	1.40E+03	3.88E+04	3.92E+04
118				No Intermediate Phase and EPA Hab.		8.30E-04	3.68E-04	1.54E-04	1.71E+06	4.59E+06	4.75E+09	8.41E+09	7.93E+02	1.83E+03	1.35E+05	2.42E+05
119				1-year Intermediate Phase and EPA Hab.		5.43E-04	3.10E-04	1.33E-04	1.31E+06	3.83E+06	1.18E+10	1.91E+10	4.67E+02	1.37E+03	3.16E+04	3.18E+04
120				Larger Nonevacuating Cohort		4.59E-04	2.63E-04	1.08E-04	1.17E+06	3.30E+06	1.42E+10	4.04E+10	7.48E+02	1.70E+03	1.21E+05	1.48E+05

Table I-6 Sensitivity calculation results for Mark II source terms with high population site file (Limerick)

Run	Base Model	Site File	Source Term	Type	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
					0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
121	Mark II - Limerick	High - Limerick	Mk II - Low (Bin 2)	Base Case	0.00E+00	1.15E-06	1.81E-07	6.35E-08	4.34E+03	5.45E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
122				1-hour Evacuation Delay		1.15E-06	1.81E-07	6.35E-08	4.34E+03	5.45E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
123				No Intermediate Phase		1.24E-06	1.91E-07	6.68E-08	4.55E+03	5.69E+03	3.81E+08	3.81E+08	5.27E-03	5.27E-03	1.60E-01	1.60E-01
124				1-year Intermediate Phase		1.04E-06	1.68E-07	5.90E-08	4.04E+03	5.10E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
125				EPA Habitability Criterion (2 rem per year)		1.15E-06	1.81E-07	6.36E-08	4.34E+03	5.45E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
126				No Intermediate Phase and EPA Hab.		1.24E-06	1.91E-07	6.68E-08	4.55E+03	5.69E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
127				1-year Intermediate Phase and EPA Hab.		1.04E-06	1.68E-07	5.90E-08	4.05E+03	5.10E+03	3.81E+08	3.81E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
128				Larger Nonevacuating Cohort		1.20E-06	1.83E-07	6.40E-08	4.36E+03	5.47E+03	3.67E+08	3.67E+08	5.25E-03	5.25E-03	0.00E+00	0.00E+00
129			Mk II - Med (Bin 5)	Base Case	0.00E+00	1.35E-04	3.39E-05	1.21E-05	6.89E+05	8.88E+05	4.25E+09	4.38E+09	1.30E+02	2.21E+02	1.54E+04	1.54E+04
130				1-hour Evacuation Delay		1.35E-04	3.39E-05	1.21E-05	6.89E+05	8.88E+05	4.25E+09	4.38E+09	1.30E+02	2.21E+02	1.54E+04	1.54E+04
131				No Intermediate Phase		1.12E-04	3.51E-05	1.27E-05	7.13E+05	9.24E+05	2.32E+09	2.41E+09	1.53E+02	2.44E+02	5.12E+04	5.12E+04
132				1-year Intermediate Phase		1.34E-04	3.13E-05	1.10E-05	6.46E+05	8.30E+05	4.57E+09	4.65E+09	1.24E+02	2.15E+02	6.97E+03	6.97E+03
133				EPA Habitability Criterion (2 rem per year)		2.34E-04	3.98E-05	1.38E-05	7.83E+05	9.82E+05	1.22E+09	1.31E+09	1.21E+02	2.12E+02	2.42E+03	2.42E+03
134				No Intermediate Phase and EPA Hab.		2.40E-04	4.30E-05	1.50E-05	8.38E+05	1.05E+06	8.64E+08	9.46E+08	1.24E+02	2.15E+02	7.50E+03	7.50E+03
135				1-year Intermediate Phase and EPA Hab.		1.72E-04	3.37E-05	1.18E-05	6.86E+05	8.69E+05	1.82E+09	1.91E+09	1.24E+02	2.15E+02	6.97E+03	6.97E+03
136				Larger Nonevacuating Cohort		1.39E-04	3.40E-05	1.21E-05	6.93E+05	8.92E+05	4.24E+09	4.37E+09	1.30E+02	2.21E+02	1.54E+04	1.54E+04
137			Mk II - High (Bin 8)	Base Case	0.00E+00	4.70E-04	3.17E-04	1.25E-04	6.11E+06	8.26E+06	8.56E+10	1.10E+11	8.53E+02	1.68E+03	7.21E+05	7.41E+05
138				1-hour Evacuation Delay		4.70E-04	3.17E-04	1.25E-04	6.11E+06	8.26E+06	8.56E+10	1.10E+11	8.53E+02	1.68E+03	7.21E+05	7.41E+05
139				No Intermediate Phase		4.09E-04	2.86E-04	1.20E-04	6.72E+06	9.14E+06	7.07E+10	8.56E+10	1.40E+03	2.81E+03	1.85E+06	2.29E+06
140				1-year Intermediate Phase		5.51E-04	3.09E-04	1.22E-04	5.84E+06	7.95E+06	1.34E+11	1.63E+11	5.81E+02	1.37E+03	1.49E+05	1.49E+05
141				EPA Habitability Criterion (2 rem per year)		1.09E-03	4.26E-04	1.62E-04	7.89E+06	1.03E+07	3.92E+10	4.55E+10	6.11E+02	1.40E+03	2.16E+05	2.16E+05
142				No Intermediate Phase and EPA Hab.		9.89E-04	4.49E-04	1.76E-04	9.25E+06	1.21E+07	3.02E+10	3.29E+10	8.97E+02	1.80E+03	8.11E+05	8.74E+05
143				1-year Intermediate Phase and EPA Hab.		6.35E-04	3.77E-04	1.49E-04	6.91E+06	9.36E+06	6.92E+10	7.39E+10	5.81E+02	1.37E+03	1.49E+05	1.49E+05
144				Larger Nonevacuating Cohort		5.64E-04	3.66E-04	3.20E-04	6.16E+06	8.30E+06	8.56E+10	1.10E+11	8.53E+02	1.68E+03	7.21E+05	7.41E+05

APPENDIX J

OFFSITE CONSEQUENCE RESULTS FOR COMPOSITE MELCOR CASES

Table J-1 Build-up of composite MELCOR cases and MACCS bins: Conditional consequences for individual early fatality, individual latent cancer fatality, and population dose

MELCOR Case	MACCS Bin	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
		0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C101	**101**	**0**	**1.28E-04**	**4.85E-05**	**2.54E-05**	**817,000**	**1,481,000**
21	11	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
24	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C101DF10	**102**	**0**	**4.99E-05**	**9.73E-06**	**4.71E-06**	**162,100**	**288,500**
21DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
24DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
C101DF100	**103**	**0**	**1.88E-05**	**2.86E-06**	**1.32E-06**	**59,800**	**102,300**
21DF100	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
24DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
C101DF1000	**104**	**0**	**1.11E-05**	**1.78E-06**	**8.53E-07**	**38,340**	**67,630**
21DF1000	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
24DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
C102DF10	**105**	**0**	**3.72E-05**	**7.00E-06**	**3.38E-06**	**123,143**	**219,286**
21DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
22DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
24DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
27DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
28DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
30DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
32DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
C102DF100	**106**	**0**	**1.30E-05**	**1.79E-06**	**8.02E-07**	**37,986**	**63,100**
21DF100	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
22DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
24DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
27DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
28DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
30DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
32DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300

MELCOR Case	MACCS Bin	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
		0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C102DF1000	**107**	**0**	**6.30E-06**	**9.10E-07**	**4.19E-07**	**19,897**	**33,449**
21DF1000	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
22DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
24DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
27DF1000	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
28DF1000	1	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
30DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
32DF1000	1	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
C103	**108**	**0**	**1.21E-04**	**3.28E-05**	**1.64E-05**	**524,000**	**932,000**
27	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
30	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
32	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C104	**109**	**0**	**3.49E-04**	**9.94E-05**	**4.87E-05**	**1,540,000**	**2,690,000**
1	12	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
4	10	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
5	10	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
6	12	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
C104DF10	**110**	**0**	**2.51E-04**	**5.96E-05**	**2.79E-05**	**862,000**	**1,468,500**
1DF10	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
4DF10	10	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
5DF10	9	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
6DF10	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C104DF100	**111**	**0**	**2.51E-04**	**5.96E-05**	**2.79E-05**	**862,000**	**1,468,500**
1DF100	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
4DF100	10	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
5DF100	9	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
6DF100	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C104DF1000	**112**	**0**	**2.51E-04**	**5.96E-05**	**2.79E-05**	**862,000**	**1,468,500**
1DF1000	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
4DF1000	10	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
5DF1000	9	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
6DF1000	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C105	**113**	**0**	**2.13E-04**	**8.26E-05**	**4.33E-05**	**1,415,000**	**2,560,000**
22	12	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
23	11	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000

MELCOR Case	MACCS Bin	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
		0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C106	**114**	**0**	**1.71E-04**	**6.55E-05**	**3.43E-05**	**1,116,000**	**2,020,500**
21	11	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
22	12	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
23	11	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
24	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C106DF10	**115**	**0**	**6.47E-05**	**1.29E-05**	**6.25E-06**	**207,550**	**369,250**
21DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
22DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
23DF10	6	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
24DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
C106DF100	**116**	**0**	**1.80E-05**	**2.60E-06**	**1.16E-06**	**54,100**	**89,950**
21DF100	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
22DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
23DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
24DF100	4	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
C106DF1000	**117**	**0**	**6.50E-06**	**9.83E-07**	**4.70E-07**	**21,910**	**37,945**
21DF1000	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
22DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
23DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
24DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
C107	**118**	**0**	**1.21E-04**	**3.28E-05**	**1.64E-05**	**524,000**	**932,000**
25	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
26	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
28	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
30	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
C107DF10	**119**	**0**	**2.03E-05**	**3.36E-06**	**1.62E-06**	**71,200**	**127,000**
25DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
26DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
28DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
30DF10	5	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
C107DF100	**120**	**0**	**6.25E-06**	**7.16E-07**	**3.21E-07**	**16,500**	**27,300**
25DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
26DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
28DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
30DF100	3	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300

MELCOR Case	MACCS Bin	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
		0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C107DF1000	**121**	**0**	**8.24E-07**	**8.18E-08**	**3.72E-08**	**2,585**	**3,850**
25DF1000	1	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
26DF1000	1	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
28DF1000	1	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
30DF1000	2	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
C108	**122**	**0**	**1.28E-04**	**4.01E-05**	**2.02E-05**	**652,000**	**1,156,000**
1 prior to CR	60% of 12	0	1.75E-04	6.06E-05	3.14E-05	1,032,000	1,854,000
4 prior to CR	20% of 10	0	8.12E-05	1.96E-05	9.02E-06	272,000	458,000
5 prior to CR	20% of 10	0	8.12E-05	1.96E-05	9.02E-06	272,000	458,000
6 prior to CR	60% of 12	0	1.75E-04	6.06E-05	3.14E-05	1,032,000	1,854,000
C108DF10	**123**	**0**	**7.44E-05**	**1.85E-05**	**8.85E-06**	**277,200**	**480,100**
1DF10 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200
4DF10 prior to CR	20% of 10	0	8.12E-05	1.96E-05	9.02E-06	272,000	458,000
5DF10 prior to CR	20% of 9	0	7.10E-05	1.50E-05	6.70E-06	208,000	344,000
6DF10 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200
C108DF100	**124**	**0**	**7.44E-05**	**1.85E-05**	**8.85E-06**	**277,200**	**480,100**
1DF100 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200
4DF100 prior to CR	20% of 10	0	8.12E-05	1.96E-05	9.02E-06	272,000	458,000
5DF100 prior to CR	20% of 9	0	7.10E-05	1.50E-05	6.70E-06	208,000	344,000
6DF100 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200
C108DF1000	**125**	**0**	**7.44E-05**	**1.85E-05**	**8.85E-06**	**277,200**	**480,100**
1DF1000 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200
4DF1000 prior to CR	20% of 10	0	8.12E-05	1.96E-05	9.02E-06	272,000	458,000
5DF1000 prior to CR	20% of 9	0	7.10E-05	1.50E-05	6.70E-06	208,000	344,000
6DF1000 prior to CR	60% of 7	0	7.26E-05	1.97E-05	9.84E-06	314,400	559,200

Table J-2 Build-up of composite MELCOR cases and MACCS Bins: Conditional consequences for offsite cost, land exceeding long-term habitability criterion, and population subject to long-term protective action

MELCOR Case	MACCS Bin	Conditional Offsite Consequences					
		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
		0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C101	**101**	**4,350,000,000**	**6,705,000,000**	**238**	**517**	**27750**	**36200**
21	11	5,960,000,000	9,720,000,000	286	673	40,500	55,800
24	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C101DF10	**102**	**685,000,000**	**815,000,000**	**78.5**	**120**	**1779**	**1779**
21DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
24DF10	5	220,000,000	240,000,000	41	65	118	118
C101DF100	**103**	**180,500,000**	**190,500,000**	**32**	**44**	**62.5**	**62.5**
21DF100	5	220,000,000	240,000,000	41	65	118	118
24DF100	4	141,000,000	141,000,000	23	23	7	7
C101DF1000	**104**	**149,850,000**	**159,850,000**	**21**	**33**	**59**	**59**
21DF1000	5	220,000,000	240,000,000	41	65	118	118
24DF1000	2	79,700,000	79,700,000	1	1	0	0
C102DF10	**105**	**485,714,286**	**568,571,429**	**62.4**	**96.4**	**1,067.1**	**1,067.1**
21DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
22DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
24DF10	5	220,000,000	240,000,000	41	65	118	118
27DF10	5	220,000,000	240,000,000	41	65	118	118
28DF10	5	220,000,000	240,000,000	41	65	118	118
30DF10	5	220,000,000	240,000,000	41	65	118	118
32DF10	5	220,000,000	240,000,000	41	65	118	118
C102DF100	**106**	**133,900,000**	**137,014,286**	**20.0**	**23.9**	**20.3**	**20.3**
21DF100	5	220,000,000	240,000,000	41	65	118	118
22DF100	4	141,000,000	141,000,000	23	23	7	7
24DF100	4	141,000,000	141,000,000	23	23	7	7
27DF100	4	141,000,000	141,000,000	23	23	7	7
28DF100	3	98,100,000	98,700,000	10	11	1	1
30DF100	3	98,100,000	98,700,000	10	11	1	1
32DF100	3	98,100,000	98,700,000	10	11	1	1

MELCOR Case	MACCS Bin	Conditional Offsite Consequences					
		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
		0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C102DF1000	**107**	**108,271,429**	**111,128,571**	**9.6**	**13.0**	**25.0**	**25.0**
21DF1000	5	220,000,000	240,000,000	41	65	118	118
22DF1000	2	79,700,000	79,700,000	1	1	0	0
24DF1000	2	79,700,000	79,700,000	1	1	0	0
27DF1000	4	141,000,000	141,000,000	23	23	7	7
28DF1000	1	78,900,000	78,900,000	0	0	-	-
30DF1000	2	79,700,000	79,700,000	1	1	0	0
32DF1000	1	78,900,000	78,900,000	0	0	-	-
C103	**108**	**2,740,000,000**	**3,690,000,000**	**190.0**	**361.0**	**15,000.0**	**16,600.0**
27	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
30	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
32	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C104	**109**	**11,450,000,000**	**14,700,000,000**	**514**	**878**	**57,950**	**65,600**
1	12	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
4	10	9,900,000,000	12,000,000,000	479	715	51,400	51,500
5	10	9,900,000,000	12,000,000,000	479	715	51,400	51,500
6	12	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
C104DF10	**110**	**5,667,500,000**	**6,995,000,000**	**303**	**467**	**29,150**	**29,975**
1DF10	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
4DF10	10	9,900,000,000	12,000,000,000	479	715	51,400	51,500
5DF10	9	7,290,000,000	8,600,000,000	351	429	35,200	35,200
6DF10	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C104DF100	**111**	**5,667,500,000**	**6,995,000,000**	**303**	**467**	**29,150**	**29,975**
1DF100	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
4DF100	10	9,900,000,000	12,000,000,000	479	715	51,400	51,500
5DF100	9	7,290,000,000	8,600,000,000	351	429	35,200	35,200
6DF100	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C104DF1000	**112**	**5,667,500,000**	**6,995,000,000**	**303**	**467**	**29,150**	**29,975**
1DF1000	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
4DF1000	10	9,900,000,000	12,000,000,000	479	715	51,400	51,500
5DF1000	9	7,290,000,000	8,600,000,000	351	429	35,200	35,200
6DF1000	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C105	**113**	**9,480,000,000**	**13,560,000,000**	**418**	**857**	**52,500**	**67,750**
22	12	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
23	11	5,960,000,000	9,720,000,000	286	673	40,500	55,800

MELCOR Case	MACCS Bin	Conditional Offsite Consequences					
		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
		0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C106	**114**	**6,915,000,000**	**10,132,500,000**	**328**	**687**	**40,125**	**51,975**
21	11	5,960,000,000	9,720,000,000	286	673	40,500	55,800
22	12	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
23	11	5,960,000,000	9,720,000,000	286	673	40,500	55,800
24	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C106DF10	**115**	**917,500,000**	**1,102,500,000**	**97**	**148**	**2,610**	**2,610**
21DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
22DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
23DF10	6	1,150,000,000	1,390,000,000	116	175	3,440	3,440
24DF10	5	220,000,000	240,000,000	41	65	118	118
C106DF100	**116**	**160,750,000**	**165,750,000**	**28**	**34**	**35**	**35**
21DF100	5	220,000,000	240,000,000	41	65	118	118
22DF100	4	141,000,000	141,000,000	23	23	7	7
23DF100	4	141,000,000	141,000,000	23	23	7	7
24DF100	4	141,000,000	141,000,000	23	23	7	7
C106DF1000	**117**	**114,775,000**	**119,775,000**	**11**	**17**	**30**	**30**
21DF1000	5	220,000,000	240,000,000	41	65	118	118
22DF1000	2	79,700,000	79,700,000	1	1	0	0
23DF1000	2	79,700,000	79,700,000	1	1	0	0
24DF1000	2	79,700,000	79,700,000	1	1	0	0
C107	**118**	**2,740,000,000**	**3,690,000,000**	**190**	**361**	**15,000**	**16,600**
25	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
26	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
28	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
30	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
C107DF10	**119**	**220,000,000**	**240,000,000**	**41**	**65**	**118**	**118**
25DF10	5	220,000,000	240,000,000	41	65	118	118
26DF10	5	220,000,000	240,000,000	41	65	118	118
28DF10	5	220,000,000	240,000,000	41	65	118	118
30DF10	5	220,000,000	240,000,000	41	65	118	118

MELCOR Case	MACCS Bin	Conditional Offsite Consequences					
		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
		0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
C107DF100	**120**	**98,100,000**	**98,700,000**	**10**	**11**	**1**	**1**
25DF100	3	98,100,000	98,700,000	10	11	1	1
26DF100	3	98,100,000	98,700,000	10	11	1	1
28DF100	3	98,100,000	98,700,000	10	11	1	1
30DF100	3	98,100,000	98,700,000	10	11	1	1
C107DF1000	**121**	**79,100,000**	**79,100,000**	**0**	**0**	**0**	**0**
25DF1000	1	78,900,000	78,900,000	0	0	-	-
26DF1000	1	78,900,000	78,900,000	0	0	-	-
28DF1000	1	78,900,000	78,900,000	0	0	-	-
30DF1000	2	79,700,000	79,700,000	1	1	0	0
C108	**122**	**4,890,000,000**	**6,420,000,000**	**213**	**384**	**24,490**	**29,060**
1 prior to CR	60% of 12	7,800,000,000	10,440,000,000	329	624	38,700	47,820
4 prior to CR	20% of 10	1,980,000,000	2,400,000,000	96	143	10,280	10,300
5 prior to CR	20% of 10	1,980,000,000	2,400,000,000	96	143	10,280	10,300
6 prior to CR	60% of 12	7,800,000,000	10,440,000,000	329	624	38,700	47,820
C108DF10	**123**	**1,681,500,000**	**2,137,000,000**	**99**	**166**	**8,830**	**9,315**
1DF10 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960
4DF10 prior to CR	20% of 10	1,980,000,000	2,400,000,000	96	143	10,280	10,300
5DF10 prior to CR	20% of 9	1,458,000,000	1,720,000,000	70	86	7,040	7,040
6DF10 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960
C108DF100	**124**	**1,681,500,000**	**2,137,000,000**	**99**	**166**	**8,830**	**9,315**
1DF100 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960
4DF100 prior to CR	20% of 10	1,980,000,000	2,400,000,000	96	143	10,280	10,300
5DF100 prior to CR	20% of 9	1,458,000,000	1,720,000,000	70	86	7,040	7,040
6DF100 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960
C108DF1000	**125**	**1,681,500,000**	**2,137,000,000**	**99**	**166**	**8,830**	**9,315**
1DF1000 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960
4DF1000 prior to CR	20% of 10	1,980,000,000	2,400,000,000	96	143	10,280	10,300
5DF1000 prior to CR	20% of 9	1,458,000,000	1,720,000,000	70	86	7,040	7,040
6DF1000 prior to CR	60% of 7	1,644,000,000	2,214,000,000	114	217	9,000	9,960

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-2206
2. TITLE AND SUBTITLE Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments	3. DATE REPORT PUBLISHED MONTH: March YEAR: 2018 4. FIN OR GRANT NUMBER
5. AUTHOR(S) Jonathan Barr, Sudhamay Basu, Hossein Esmaili, and Martin Stutzke	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Systems Analysis and Division of Risk Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Same as above

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The work documented in this report focused on developing the technical basis for a potential rulemaking action on containment protection and release reduction (CPRR) strategies for boiling water reactors with Mark I and Mark II containments. The work covered three areas of analyses: (1) accident sequence analysis (event tree development) to identify accident sequences initiated by extended loss of ac power (ELAP) due to internal events and seismic events deemed to be the most significant risk contributors; (2) accident progression analysis of these sequences and assessment of radiological source terms; and (3) analysis of offsite consequences including individual early fatality risk and latent cancer fatality risk, land contamination, and economic consequences. The calculated offsite consequences were weighted by accident frequency to assess relative public health risk reduction associated with various CPRR strategies. Important findings and key insights from the work are delineated.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
CPRR containment protection release reduction ELAP MELCOR MACCS severe accident water addition external filter	13. AVAILABILITY STATEMENT: unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

NUREG-2206

Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments

March 2018