Draft Regulatory Basis for Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors
(10 CFR Part 50)

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Office of Nuclear Regulatory Research

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Executive Summary

In the United States, nuclear power plants have a containment barrier to confine the fission products within the plant should an accident lead to a compromise of the barriers provided by the fuel design and the reactor coolant pressure boundary. For currently operating plants, this barrier is provided by containments that include either (1) a large air volume to address the energy released from a design-basis loss of coolant accident (LOCA) while not exceeding the design pressure for the containment, or (2) smaller volumes and systems that include water or ice to absorb the energy released from a LOCA and thereby suppress the increase in pressure to values below the design limits for the containment. Boiling water reactors (BWRs) employ such pressure suppression containment designs. Mark I and Mark II BWR containments use water suppression pools to remove energy from the reactor following a LOCA or other plant transients or accidents. In addition to the suppression pools, plant systems consisting of pumps and heat exchangers are needed to maintain Mark I and Mark II containments within their established design limits. Venting of Mark I and Mark II containments may be necessary for potential accident scenarios involving an extended loss of electrical power or other failures of the active heat removal systems.

On March 11, 2011, a 9.0-magnitude earthquake struck Japan and was followed by a 45-foot tsunami, resulting in extensive damage to the nuclear power reactors at the Fukushima Dai-ichi facility. The damaged reactors had BWR Mark I containments. The Fukushima accident disabled the plants' ability to cool their reactor cores, causing heat and pressure to build within the concrete containment buildings that surround the reactors. This buildup eventually allowed hydrogen to migrate outside containment and explode, which damaged the buildings and made it possible for radioactive material to reach the environment.

As a result of the events at the Fukushima Dai-ichi Nuclear Power Plant in 2011, the U.S. Nuclear Regulatory Commission (NRC) Near Term Task Force (NTTF) created a series of recommendations intended to outline a path to enhance the ability of nuclear power plants to respond to beyond-design-basis events. The NRC staff in SECY-12-0157, “Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments,” dated November 26, 2012 (ADAMS Accession No. ML12325A704), analyzed whether additional requirements might be warranted to address venting from BWRs with Mark I and II containments after core damage and whether filtering of radionuclides that may be released from the vents would be necessary. The NRC staff evaluated four alternatives, including severe accident capable vents, external filters, and a performance based confinement strategy.[1] The staff’s analysis in that SECY paper is discussed in more detail in Section 2, “Background.”

Commission staff requirements memorandum (SRM) for SECY-12-0157 (ADAMS Accession No. ML13078A017) directed the staff to order severe accident capable vents and to pursue a rulemaking to evaluate the design and installation of an engineered filtered containment venting system and a performance based confinement strategy for BWRs with Mark I and Mark II

[1] The performance based strategy alternative was evaluated at a high level, but was not expressly evaluated.

containments.[2] The Commission also directed the staff to assume that severe accident capable vents have been ordered and to explore requirements associated with measures to enhance the capability to maintain confinement integrity and to cool core debris. The Commission also directed the staff to evaluate a number of performance criteria including decontamination factor and equipment and procedure availability similar to those required to implement Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(hh).

In response to SRM-SECY-12-0157, the staff issued Order EA-13-109 (ADAMS Accession No. ML13143A321) and began developing the regulatory basis for the containment protection and release reduction (CPRR) rulemaking for BWRs with Mark I and Mark II containments. Order EA-13-109 has two primary requirements: first, Mark I and Mark II containments must have a wetwell venting system that remains functional during severe accident conditions; second, licensees with Mark I and Mark II containments must either install a severe accident capable drywell venting system or develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions. These two requirements are being implemented sequentially as Phase 1 and Phase 2, respectively. The implementation of Order EA-13-109 has proceeded in parallel with the development of this regulatory basis.

In the course of preparing for Phase 2, the nuclear power industry proposed an approach in which licensees would use one of two strategies: severe accident water addition (SAWA) in combination with either (a) a severe accident capable drywell vent designed to lower temperature limits, or (b) severe accident water management[3] (SAWM) to control the water levels in the suppression pool such that it is unlikely a licensee would need to vent from the containment drywell during severe accident conditions. The NRC staff included the industry's proposal in the analysis presented in this regulatory basis as Section 4, "Technical Evaluation." The NRC's analysis shows that the SAWA/SAWM strategies not only serve the purposes of Order EA-13-109 intended to address, but could also prevent containment failure modes other than over-pressurization. The NRC has endorsed the use of SAWA/SAWM strategies for compliance with Phase 2 of EA-13-109 (ADAMS Accession No. ML14343A818).

Although Phase 2 of EA-13-109 is expected to include SAWA/SAWM and thereby help address containment failure modes other than over-pressurization, licensees continue to have the option to develop approaches that would not include SAWA/SAWM. Therefore, the staff has included a rulemaking to require SAWA/SAWM among the alternatives considered in this regulatory basis.

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2 Evaluations of other reactor and containment designs are within NTTF Recommendation 5.2 and have been deemed Tier 3 activities.

3 SAWM refers to throttling the flex pump flow rate in order to avoid submerging the wetwell vent and thereby retaining the beneficial fission product scrubbing capability of the suppression pool.

In this regulatory basis, the staff considers four alternatives:

- Alternative 1 (the status quo): Take no action (Order EA-13-109 implemented without rulemaking).[4]

- Alternative 2: Pursue rulemaking to make Order EA-13-109 generically applicable for protection of BWR Mark I and II containments against over-pressurization.

- Alternative 3: Pursue rulemaking to address overall BWR Mark I and Mark II containment protection against multiple failure modes by making Order EA-13-109 generically applicable and requiring external water addition points that would allow for water addition into the reactor pressure vessel (RPV) or drywell (DW) to prevent containment failure from both over-pressurization and liner melt-through.

- Alternative 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 would include making Order EA-13-109 generically applicable and require external water addition into the RPV or DW. In addition, licensees would be required to reduce the fission products released from the containment by either implementing strategies to maximize the availability and efficiency of the wetwell in scrubbing or filtering fission products before venting from containment and/or installing an engineered filter in the containment vent paths.

These four alternatives are discussed in Section 3, "Regulatory Evaluation." To support the regulatory analysis of these four alternatives, the staff performed detailed thermal-hydraulic and severe accident progression computer simulations with MELCOR, performed offsite release and consequence analysis of radioactive materials with the MELCOR Accident Consequence Code System (MACCS), and developed a probabilistic risk assessment (PRA) of the accident scenarios. Because SAWA/SAWM is not required by EA-13-109, it was not credited in the staff's evaluation of alternative 1 or 2. The staff considered the results of those analyses in part using the Commission's 1986 Safety Goal Policy Statement.

The Safety Goal Policy Statement includes two quantitative health objectives (QHOs) regarding first, prompt fatality risk, and second, latent cancer fatality risk. For the first quantitative safety goal, the staff determined that for the severe reactor accidents modeled there is zero conditional early fatality risk. This occurs in part because the modeled accident progression results in releases that begin late when compared to the time needed for evacuation. For the second, the staff determined that the latent cancer fatality risk is already well below the QHO, and therefore none of the alternatives meet the threshold for a substantial safety enhancement.

The staff evaluated the benefits of each alternative against alternative 1, in which the staff would take no additional action on CPRR. Alternative 2 would not include any new requirements, but by making the requirements generically applicable would have benefits related to improved reporting, change control, and other aspects of controlling licensing basis information. Alternatives 3 and 4 would provide these same benefits but could also include new

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4 For the purposes of the quantitative safety benefit analysis, the staff assumed that the status quo does not include the adoption of SAWA/SAWM for compliance with Order EA-13-109.

requirements that must satisfy the backfitting requirements of 10 CFR 50.109. Alternative 3 provides a small safety benefit between alternative 1 and the QHO, but alternative 3 also ensures that licensees would take steps to provide protection against multiple containment failure modes, enhancing defense-in-depth. Alternative 4 can provide additional safety benefits over alternative 3 by reducing the amount of radioactive materials released if containment venting was needed during a severe accident.

The staff estimates SAWA installation costs to the licensee will range from approximately $2.6 million to $3.2 million. However, licensees are already planning to install SAWA/SAWM as a part of Phase 2 compliance with Order EA-13-109. Therefore Alternative 3 would not impose any additional costs on licensees. The expected installation costs for engineered filtered vents range from $11 million to $64 million. These would be new costs imposed by alternative 4.

Based on the information presented in this regulatory basis, the NRC staff expects to adopt alternative 3: the development of a proposed rule that would address the overall BWR Mark I and Mark II containment protection issue against multiple failure modes by installing severe accident capable venting systems and external water addition points that would require water addition into the reactor pressure vessel or drywell.

1. Introduction

As a result of the accident at Fukushima Dai-ichi in Japan, the U.S. Nuclear Regulatory Commission (NRC) initiated a rulemaking effort to determine the need for additional regulations in the area of filtration of containment vents for boiling water reactors (BWRs) with Mark I and Mark II containments. This effort is intended to study the current regulatory framework associated with various aspects of beyond-design-basis severe accidents and the effects on BWR Mark I and II containments.

2. Background

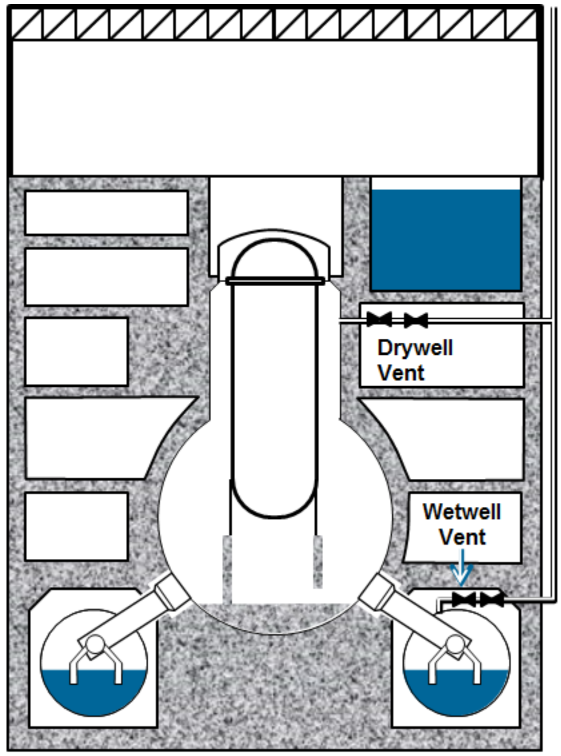
SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012, provides background information on the regulatory history associated with BWR Mark I and Mark II containments. The related staff requirements memorandum (SRM) dated March 19, 2013, documents the Commission's decisions on SECY-12-0157 and provides direction to the staff concerning immediate and longer-term actions to address containment-related lessons learned from the March 2011 Fukushima Dai-ichi accident. This section will summarize the background information in SECY-12-0157 and describe the activities leading up to the findings and recommendations described in this regulatory basis document.

2.1 BWR Mark I and Mark II Containments — General Background

A key element of the design of nuclear power plants is the inclusion of multiple barriers to the potential release of radioactive materials created within the fuel by the fission process. In the United States, a containment barrier is included to confine the fission products within the plant should an accident lead to a compromise of the barriers provided by the fuel design and the reactor coolant pressure boundary. For currently operating plants, this barrier is provided by containments that include either (1) a large air volume to address the energy released from a design-basis loss of coolant accident (LOCA) while not exceeding the design pressure for the containment, or (2) systems that include water or ice to absorb the energy released from a LOCA and thereby suppress the increase in pressure to values below the design limits for the containment. BWRs employ such pressure suppression containment designs.

Mark I and Mark II are specific containment configurations for BWRs that use water suppression pools to remove energy from the reactor following a LOCA or other plant transients or accidents. Most other containment designs provide a larger volume to accommodate the increased pressure during a LOCA and are less reliant on successful venting to perform their function. The primary focus of containment designs is to limit the potential exposure of the public from radioactive materials following a design-basis reactor accident, as represented by conditions defined in NRC regulations and related guidance. Figure 2-1, "Typical BWR Mark 1 Containment Vent Locations," shows the typical layout for the wetwell and drywell vents for a Mark I BWR containment. The role of the containments in helping deal with beyond-design-basis events, including severe accidents with significant damage to a reactor core, was evaluated for operating nuclear power plants as part of risk studies and in response to operating experience beginning in the 1980s.

Figure 2-1: Typical BWR Mark 1 Containment Vent Locations



In SECY-88-147, “Integration Plan for Closure of Severe Accident Issues,” dated May 25, 1988 (ADAMS Accession No. ML12250A921), the staff presented to the Commission its plan to evaluate potential generic severe accident containment vulnerabilities in a research effort called 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. These were to 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 light water reactor (LWR) containments to successfully survive some severe accident challenges, as indicated by the results documented in NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” (ADAMS Accession No. ML040140729). All LWR containment types were assessed in the CPIP, beginning with the BWRs with Mark I containments. Potential improvements for BWRs with Mark I containments were documented in NUREG/CR-5225 (including Addendum 1), “An Overview of BWR Mark-I Containment Venting Risk Implications,” (ADAMS Accession No. ML101870670) and SECY-89-017, “Mark I Containment Performance Improvement Program,” dated January 23, 1989 (ADAMS Accession No. ML12251A419). 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 in Performance” (ADAMS Accession No. ML101890887).

The Commission documented in the related SRMs that the majority of the staff’s recommended safety improvements would be evaluated by licensees as part of the Individual Plant Examination Program. The only exception was the hardened vent capability recommendation

for BWR Mark I containments. The Commission directed the staff to approve installation of hardened vents under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59, “Changes, Tests, and Experiments,” for licensees that would voluntarily implement this improvement and perform a backfit analysis for requiring a hard vent installation at those plants that declined voluntary installation. Generic Letter 89-16, “Installation of a Hardened Wetwell Vent,” (ADAMS Accession No. ML060760371) was issued in September 1989 and provided an example of an acceptable design that used the suppression pool to achieve as much reduction in effluent radioactivity. In response to the issuance of the generic letter, all Mark I licensees installed a hardened vent under 10 CFR 50.59. Concern about a large release from a severe accident was the key consideration in the decision to recommend a hardened vent for the Mark I containment, and not to recommend a hardened vent for the Mark II containment following completion of the CPIP. For the Mark I containments, where the wetwell might provide some scrubbing of a release, a wetwell vent was recommended. For the Mark II containments, where severe accident risk was influenced by a potential bypass of the wetwell, the NRC found that possible approaches were not substantial cost-effective safety enhancements and these plants were not addressed by the generic letter.

The accident at the Fukushima Dai-ichi nuclear power station, which included core damage and releases from three BWRs with Mark I containments, reinforced the importance of reliable operation of containment vents for plants with Mark I and II containments. The accident was initiated by a 9.0-magnitude earthquake off the coast of Japan and a subsequent tsunami that caused extensive damage to the nuclear power reactors at the Fukushima Dai-ichi facility. The tsunami flooded plant buildings and disabled various normal and emergency systems used to remove heat from the reactor cores and related systems, such as the suppression pools inside the containments. Systems able to function without alternating current (ac) electrical power sources were able to successfully remove heat from the reactors for a period of time but ultimately failed. The loss of the heat removal systems resulted in rising temperatures and pressures, and the generation of combustible gases which eventually damaged the buildings and made it possible for radioactive material to reach the environment.

As part of its response to the lessons learned from the accident, the NRC issued Order EA-12-050 requiring licensees to upgrade or install a reliable hardened containment venting system (HCVS) for Mark I and II BWR containments. The requirements in EA-12-050 for Mark I and II containments expanded upon the activities associated with Generic Letter 89-16 and required the HCVS to remove decay heat and maintain control of containment pressure within acceptable limits following the loss of the containment heat removal capability or a prolonged station blackout (SBO). The hardened vent system was to be accessible and operable under a range of plant conditions, including a prolonged SBO and inadequate containment cooling. While developing the requirements for Order EA-12-050, the NRC acknowledged that questions remained about maintaining containment integrity and limiting the release of radioactive materials if the venting systems were used during severe accident conditions. In the SRM for SECY-11-0137, “Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned,” dated December 15, 2011 (ADAMS Accession No. ML113490055), the Commission directed the staff to more quickly address the issue of filtration of containment vents because the analysis and interaction with stakeholders needed to inform a decision on whether filtered vents should be required could be performed concurrently with the development of the technical basis, acceptance criteria, and design expectations for reliable hardened vents.

2.2 SECY-12-0157 and Follow-up Activities

The staff presented options to address the issue of filtration of containment vents for Commission consideration in SECY-12-0157. Specifically, the options presented included: (1) maintaining the status quo including completing the requirements established in Order EA-12-050 for reliable hardened vents; (2) issuing orders requiring containment venting systems capable of operating under severe accident conditions; (3) issuing orders requiring containment venting systems capable of operating under severe accident conditions that have filters within the controlled release pathways; and (4) developing a performance-based severe accident confinement strategy for BWRs with Mark I and Mark II containments.

The staff performed various assessments and analyses of possible requirements for licensees to have containment venting systems capable of operating under severe accident conditions and possible requirements for the installation of containment vent filtration systems. The staff recommended the installation of engineered filtered venting systems for Mark I and II containments (Option 3). The staff stated that this option would provide the most regulatory certainty and the fastest implementation, while acknowledging that a comparison of only the quantifiable costs and benefits of the proposed modifications, if considered safety enhancements, would not, by themselves, demonstrate that the benefits exceed the associated costs. However, the NRC staff stated that when qualitative factors were considered, a recommendation to require the installation of engineered filtered vent systems was justified. The qualitative factors in the evaluation of the various options documented in SECY-12-0157 included:

- providing defense-in-depth (including importance of containment function).
- addressing significant uncertainties (frequencies and consequences).
- supporting severe accident management and response.
- improving hydrogen control.
- addressing external events.
- addressing multi-unit events.
- considering independence of barriers.
- improving emergency planning.
- considering consistency between reactor technologies.
- considering severe accident policy statement.[5]
- addressing international experience and practices (including availability of technology).

The staff's regulatory analysis associated with SECY-12-0157 focused on Option 2 (severe accident capable vent order) and Option 3 (filtered vent order) as those options involved potential near term regulatory action. Option 4 (i.e., rulemaking) was seen as involving longer-term efforts, and the staff stated that a cost-benefit assessment would be developed once the approach and possible regulatory changes were better defined. While all of the options in SECY-12-0157 focused on containment venting, the staff acknowledged that various risk assessments performed for BWRs with Mark I or II containments have concluded that

[5] 50 FR 32138, dated August 8, 1985, "Policy Statement on Severe Accidents Regarding Future Designs and Existing plants".

adding water to the drywell significantly benefits controlling the release of radioactive materials for those severe accident scenarios involving fuel melting through the reactor vessel. The water added to the drywell cools the molten fuel, which can arrest its progression and prevent a loss of the drywell containment function (e.g., liner melt-through, containment over-pressurization failure, containment over-temperature failure). This capability would be important to the success of any of the options in SECY-12-0157 for those potential scenarios with a core melting through the reactor pressure vessel, which could then lead to containment failure. The staff stated that the longer-term rulemaking associated with the proposed Options 2, 3, or 4 could consider adding more explicit requirements for the capability of cooling of the core debris during severe accident scenarios.

To facilitate its decision, the Commission was provided with the staff's evaluation as well as views from the Advisory Committee on Reactor Safeguards (ACRS), the nuclear industry, and other external stakeholders. One industry comment about the options presented in SECY-12-0157 was that the NRC focused on containment venting and engineered filters while not proposing measures to address containment failure modes not related to over-pressure conditions and not addressing broader issues related to severe accident management. The staff's focus on containment venting and engineered filters resulted from specific actions to address lessons learned from Fukushima and subsequent questions and taskings related to the issue of filtration of containment vents. The need to address overall containment performance before and during severe accident conditions was deferred until a possible future rulemaking. This approach to containment performance issues for BWR Mark I and II containments was highlighted during a Commission meeting held on January 9, 2013 (ADAMS Accession No. ML13029A516), and subsequent information provided in the Nuclear Energy Institute's letter to the NRC regarding filtering strategies and filtered vents dated January 25, 2013 (ADAMS Accession No. ML13030A145).

In the SRM for SECY-12-0157, the Commission directed the staff 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 to define requirements and schedules for licensees with Mark I and II containments to install severe accident capable containment vents. The staff also initiated a series of meetings with the nuclear industry and other stakeholders to discuss the development of a possible rulemaking for filtering strategies and severe accident management for BWR Mark I and II containments.

2.3 Order EA-13-109

On June 6, 2013, the staff issued the modified 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), to ensure that vents on BWR Mark I and II containments will remain functional in the conditions following reactor core damage. The revised order contains two distinct phases of implementation.

Phase 1, which all licensees are required to implement by June 2018, requires licensees to upgrade the venting capabilities from the containment wetwell to provide reliable hardened vents to assist in preventing core damage and also to remain functional during severe accident

conditions. Phase 2, which all licensees are required to implement by June 2019, requires licensees to: (a) provide additional protections for severe accident conditions through installation of a reliable severe-accident-capable drywell vent system, or (b) develop a reliable containment venting strategy that makes it unlikely to need to vent from the containment drywell during severe accident conditions. The Phase 2 portion of Order EA-13-109 builds on the Phase 1 activities, and also takes advantage of the analyses supporting the CPRR rulemaking (described in Section 3, "Regulatory Evaluation").

Order EA-13-109, Phase 2, requires that licensees either install a severe accident capable drywell vent or develop a strategy that makes it unlikely that venting from the drywell would be needed to control containment pressure during a severe accident. The possible need for venting from the containment drywell primarily results from plant operators adding water into the reactor vessel or containment to cool core debris.[6] In such circumstances, the water level in the suppression pool could rise above the location of the containment vent in the wetwell. As a result, the added water could block the release of steam and gases rather than allowing it to filter it through the suppression pool. To prevent containment over-pressure conditions using the wetwell vent plant operators would need to vent containment using the drywell.

Studies of the effects of water addition to the containment during a severe accident show that the water results in reduced containment temperatures. However, evaluations performed by licensees and the NRC concluded that it could be difficult to add water into containments because access to piping connections and other equipment could be hampered by environmental conditions created by the accident (e.g., high radiation fields). Improved capabilities to provide cooling water to the containment during potential accidents with significant core damage is referred to as severe accident water addition (SAWA). For facilities that elect a strategy that includes SAWA, drywell vents would not experience the same high temperatures in a severe accident as they would without SAWA. Figure 4-17, "Mark I Maximum Drywell Structure Temperature," shows the calculated drywell containment temperatures for a large number of computer simulations. The inclusion of SAWA can therefore be used to lower the range of temperatures that would need to be considered when designing and installing equipment for drywell venting during severe accident conditions.

On September 10, 2014, in an NEI letter titled, "Compliance with Phase 2 of NRC Order to Modify the Licenses of Boiling Water Reactors (BWRs) with Mark I and II Containments with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (EA-13-109)" (ADAMS Accession No. ML14259A186), the industry proposed an approach to implement Phase 2 of EA-13-109 in which licensees would use one of two strategies: SAWA in combination with a severe accident capable drywell vent designed to lower temperature limits, or SAWA with severe accident water management (SAWM) to control the water levels in the suppression pool such that licensees would maintain capabilities to address over-pressure conditions without a severe accident capable drywell vent. SAWM refers to controlling the SAWA flow rate in order to avoid submerging the wetwell vent and thereby

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[6] The addition of water is called for in severe accident management guidelines (SAMGs) but requirements for such capabilities are not currently addressed by NRC regulations. The NRC is proposing to require SAMGs for all nuclear power licensees in the Mitigation of Beyond-Design-Basis Events rulemaking (see NRC docket ID "NRC-2014-0240" at www.regulations.gov for documents related to this rulemaking).

retaining the beneficial fission product scrubbing capability of the suppression pool. The NRC endorsed the use of SAWA/SAWM strategies for compliance with Phase 2 of EA-13-109 on December 10, 2014 (ADAMS Accession No. ML14343A818).

2.4 Current Activities (CPRR Rulemaking)

As previously discussed, the initial focus of the NRC orders and SECY-12-0157 was shaped by the Fukushima accident and related to containment venting to support decay heat removal and preventing containment failure from over-pressure conditions. Other significant potential challenges to the integrity of BWR Mark I and II containments during severe accident conditions include high temperatures and direct interactions with molten core debris following a failure of the reactor vessel (often referred to as liner melt-through). Both of these containment failure modes can be addressed by adding water to the containment drywells, either directly or through the reactor vessel. SECY-12-0157 acknowledged the potential need to address these other challenges to Mark I and II containments under severe accident conditions and noted that “the longer-term rulemaking associated with the proposed Options 2, 3, or 4 could consider adding more explicit requirements for the capability of core debris cooling during severe accident scenarios.” Some comments received by the NRC staff in correspondence and during public meetings following the issuance of SECY-12-0157 called for a more holistic approach to assessing and addressing the risks of uncontrolled and controlled releases of radioactive material during severe accident conditions. The Commission’s SRM directed the staff to assess not only possible filtration strategies for controlled releases during the venting of BWR Mark I and II containments but also broader severe accident management issues related to uncontrolled releases resulting from various containment failure modes.

A logical organization of rulemaking alternatives was developed using insights from risk assessments performed by the staff and the nuclear industry. The alternatives are separated by those associated with preventing uncontrolled releases of radioactive materials through reducing the probability of breaching containment and those associated with reducing the amount of radioactive materials released during controlled venting operations. The alternatives considered as part of this assessment of potential changes to NRC regulations for BWR Mark I and II containments are as follows:

1. Take no action (maintain status quo, including finalizing the implementation of Order EA-13-109).

2. Pursue rulemaking to make generically applicable the requirements of Order EA-13-109 for severe accident capable containment vents to support decay heat removal and containment over-pressure conditions.

3. Pursue rulemaking to make generically applicable the requirements of Order EA-13-109 and to require additional capabilities for water addition (i.e., SAWA/SAWM) to containment drywells during severe accident conditions to address uncontrolled releases resulting from containment over-temperature conditions or breach of containment liner by molten core debris.

4. Pursue rulemaking to make generically applicable the requirements of Order EA-13-109, require additional capabilities for water addition (i.e., SAWA/SAWM) to containment drywells during severe accident conditions to address uncontrolled releases resulting

from containment over-temperature conditions or breach of containment liner by molten core debris, and require additional capabilities to reduce or filter the release of radioactive materials during controlled venting of containments during severe accident conditions.[7]

Additional discussions and evaluations of each of these of alternatives are provided in Section 3, “Regulatory Evaluation.” Additional discussion of the alternatives and the various analyses performed to evaluate them is provided in Section 4, “Technical Evaluation.”

7 The alternatives can be considered sequential with each alternative incorporating the one before it. The Commission’s decision to include severe accident capable containment venting in Order EA-13-109 was, in part, intended to minimize the delays in licensees addressing a major lesson from the Fukushima Dai-ichi accident (importance of venting for decay heat removal and over-pressure protection) while also avoiding duplicative efforts in the design and implementation of plant changes related to containment venting before and after core damage.

3. Regulatory Evaluation

In addition to directing the staff to issue Order EA-13-109 requiring the installation of severe accident capable containment venting systems for BWRs with Mark I and II containments, the Commission's SRM related to SECY-12-0157 instructed the staff to do the following:

- The CPRR rulemaking should consider Option 3 of SECY-12-0157 (design and installation of an engineered filtered containment venting system 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[8]) and Option 4 of SECY-12-0157 (development of requirements and technical acceptance criteria for confinement strategies and requirements for licensees to justify operator actions and systems or combination of systems, such as suppression pools, containment sprays, and separate filters to accomplish the function and meet the requirements).

- The rulemaking should assume the installation of severe accident capable hardened venting systems ordered under Option 2 of SECY-12-0157. The rulemaking should also evaluate a variety of performance criteria.

- The rulemaking should fully explore the requirements associated with measures to enhance the capability to maintain containment integrity and to cool core debris.

- The staff should engage a diversity of external stakeholders throughout the development of the rulemaking and should present to ACRS at the appropriate points in the process.

- The staff should keep the Commission periodically informed of their progress. If policy issues emerge, the staff should provide a notation vote paper.

- The staff should seek detailed Commission guidance regarding the use of qualitative factors in a future notation vote paper.

The staff initiated a series of 13 public meetings and two ACRS meetings following the issuance of the SRM. The meetings involved discussions of the possible alternatives to address the directions in the SRM as well as coordination of technical analyses by the staff and nuclear industry. A summary of the technical analyses performed by the NRC staff is provided in Section 4, "Technical Evaluation." A summary of the technical analyses performed by the nuclear industry is provided in the Electric Power Research Institute (EPRI) report, "Technical Basis for Severe Accident Mitigating Strategies, Volume 1," dated April 2015. The staff considered possible performance criteria for the proposed regulatory requirements as well as broader performance criteria for NRC decisionmaking. These broader criteria for decisionmaking are discussed in Section 5, "Performance Criteria Information," and are

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8 Evaluations of other reactor and containment designs are within NTTF Recommendation 5.2 and have been deemed Tier 3 activities.

provided for information only. The evaluation of alternatives discussed below is performed in accordance with established guidance such as the NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook” (ADAMS Accession No. ML050190193), NUREG-1409, “Backfitting Guidelines” (ADAMS Accession No. ML032230247), the Safety Goal Policy Statement,[9] and the Severe Accident Policy Statement.[10]

In this section the staff considers all the information and policies relevant to making the regulatory determination of whether to proceed with a rulemaking. That includes the specific direction from the Commission in the SRM to SECY-12-0157 on factors to consider and assumptions to make in the preparation of this CPRR regulatory basis. Section 3.1 documents the staff’s the safety goal screening evaluation, as described in NUREG/BR-0058, “Regulatory Analysis Guidelines for the U.S. Nuclear Regulatory Commission” (ADAMS Accession No. ML042820192). To conduct the screening evaluation, the staff used information from the computer simulations and PRA documented below in Section 4, “Technical Evaluation,” to consider the safety benefits of four alternatives and their variants in light of the NRC’s stated quantitative health objectives. Section 3.2 documents the staff’s evaluation of alternatives, which includes not only the quantitative benefit conclusions from the analysis in Section 4, but also other considerations such as qualitative factors and how the alternatives fit with the NRC’s other post-Fukushima activities. Section 3.3 documents the cost-benefit assessment for SAWA/SAWM as well as cost information on engineered filtered vents.

3.1 Safety Goal Screening Evaluation

As a first step in the regulatory analysis, the staff followed the safety goal screening process described in NUREG/BR-0058, “Regulatory Analysis Guidelines for the U.S. Nuclear Regulatory Commission” (ADAMS Accession No. ML042820192). This is similar to the screening process that was performed in the analysis described in COMSECY-13-0030, “Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel,” dated November 25, 2013 (ADAMS Accession No. ML13329A918). To determine whether or not a regulatory action may be further warranted, the staff conducted a safety goal screening evaluation using the Commission’s 1986 Safety Goal Policy Statement. The Safety Goal Policy Statement establishes two qualitative safety goals and two quantitative objectives. Both the goals and objectives apply only to the risks to the public from the accidental or routine release of radioactive materials from nuclear power plants.

The two qualitative safety goals are as follows:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

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9 51 FR 30028, dated August 21, 1986, "10 CFR Part 50 - Safety Goals for the Operations of Nuclear Power Plants" (ADAMS Accession No. ML011210381).

10 50 FR 32138, dated August 8,1985, “Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants.”

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives (QHOs) are to be used in determining achievement of the above safety goals:

(1) The risk to an average individual within 1 mile of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 1/10 of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

(2) The risk to the population within 10 miles of a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The safety goal screening evaluation is designed to answer whether a regulatory requirement should not be imposed generically on nuclear power plants because the risk to the population is already acceptably low. This evaluation is intended to eliminate some proposed requirements from further consideration independently of whether they could be cost-beneficial.

The Safety Goal Policy Statement defines the early fatality area calculation as that within 1.6 kilometers (1 mile) from the site boundary. The prompt fatality QHO represents a $5x10^{-7}$ per year objective for an average individual within 1 mile.[11]

The second quantitative objective of the policy relates to ensuring that the cancer fatality risks from nuclear power plant operations remain a small fraction of the overall cancer risks from all causes. The latent cancer fatality QHO represents a $2x10^{-6}$ per year objective for an average individual within 16 kilometers (10 miles) from the site boundary. The staff reassessed the criteria in COMSECY-13-0030 based on recent data and found that the total fatality rate from cancer in the United States is 580,350[12] per 315,747,500 persons[13] or a risk of $1.84x10^{-3}$ per year. One tenth of 1 percent of this value results in a safety goal of $2x10^{-6}$ per year.[14]

The staff analyzed numerous alternatives in relation to the QHOs described in the Safety Goal Policy Statement. Each of the alternatives was compared to alternative 1 (regulatory baseline - no action) to determine the relative benefits and costs of the alternative. The alternatives are described in Section 2.4, "Current Activities (CPRR Rulemaking)". Alternatives 3 and 4 have numerous sub-alternatives, which are described in Table 4-3, "Summary of Regulatory Analysis Alternatives."

11 NUREG-0880, Revision 1, "Safety Goals for Nuclear Power Plant Operation," (ADAMS Accession No. ML071770230).

12 http://www.cancer.org/research/cancerfactsfigures/index.

13 http://www.census.gov/popclock/.

14 The calculation yields $1.84x10^{-6}$, however, the NRC has rounded this value to $2x10^{-6}$.

For the first quantitative safety goal, the staff determined that for the severe reactor accidents modeled there is zero conditional early fatality risk. This occurs in part because the modeled accident progression results in releases that begin late when compared to the time needed for evacuation.

The staff then performed a screening analysis for the second QHO. The staff evaluated all U.S. BWRs with Mark I containments (a total of 22 units at 15 sites) and Mark II containments (a total of eight units at five sites) (see Table 4-4, "ELAP Frequencies"). Using the highest extended loss of offsite alternating current power (ELAP) frequency[15] value among all Mark I and II BWRs (7.4×10^{-5} per year), a success probability in the flexible coping strategies (FLEX) equipment of 0.6 per demand[16] (i.e., that FLEX will work 6 out of 10 times in arresting an accident involving an ELAP), and the highest conditional individual latent cancer fatality (ILCF) risk among all Mark I and II BWRs (2.2×10^{-3} per event), this yields a conservative high estimate of frequency-weighted individual latent cancer fatality risk of approximately 7×10^{-8} per reactor year.[17] The latent cancer fatality QHO is 2×10^{-6} per year. Therefore, the conservative high estimate is more than an order of magnitude lower than the QHO. The factors leading to this low likelihood, as discussed above, are summarized in Figure 3-1, "Calculation of High Level Conservative Estimate."

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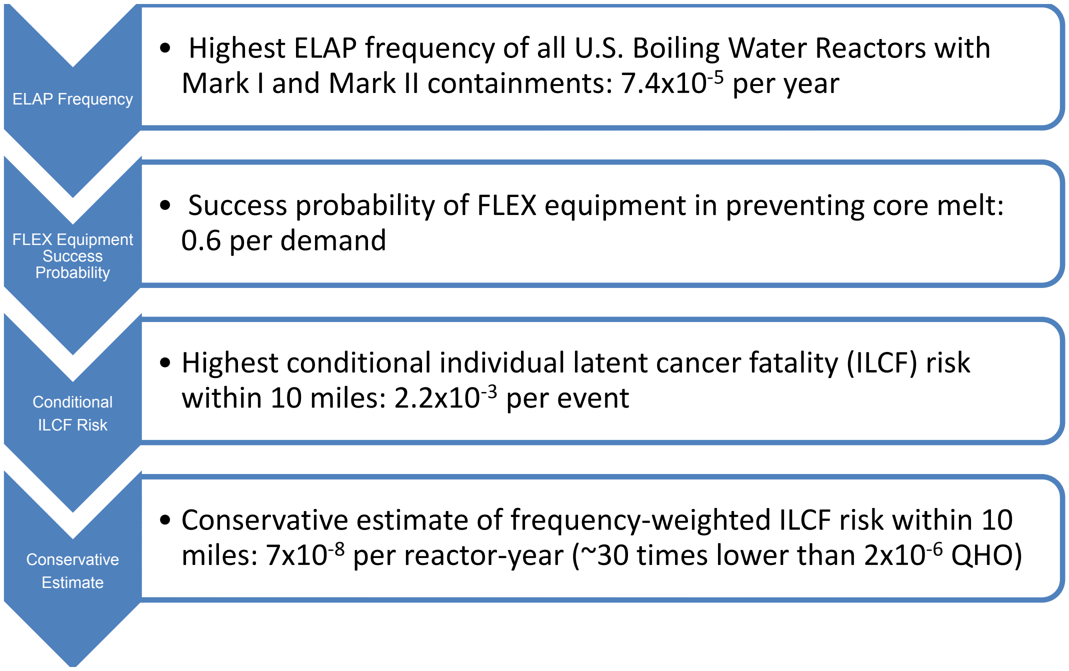
15 The ELAP frequency was calculated using the seismic hazard evaluations provided in industry responses to the letter "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident" (ADAMS Accession No. ML12053A340). The frequencies and durations of the loss-of-offsite-power (LOOP) caused by internal events was evaluated in NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," (ADAMS Accession No. ML060200477). Random and common-cause failures of the onsite emergency power as evaluated in the 2011 update (issued February 2013) of NUREG/CR-5500, Vol. 5, "Emergency Diesel Generator Power System Reliability, 1987-1993," (http://nrcoe.inl.gov/resultsdb/SysStudy/EPS.aspx) and seismic equipment failures as estimated in licensee responses to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (http://www.nrc.gov/reading-rm/doc-collections/gen-comm/gen-letters/1988/gl88020s4.html).

16 The FLEX success probability was estimated based on the results of the risk evaluation (PRA). It includes random failures of the FLEX equipment, seismically induced failures of in-plant equipment that interface with the FLEX equipment, and operator performance (based on a scoping human reliability analysis that was informed by NRC review of licensee mitigating strategies, including plant walkdowns).

17 The calculation yields 6.6×10^{-8}, however, the NRC has rounded this value to 7×10^{-8}.

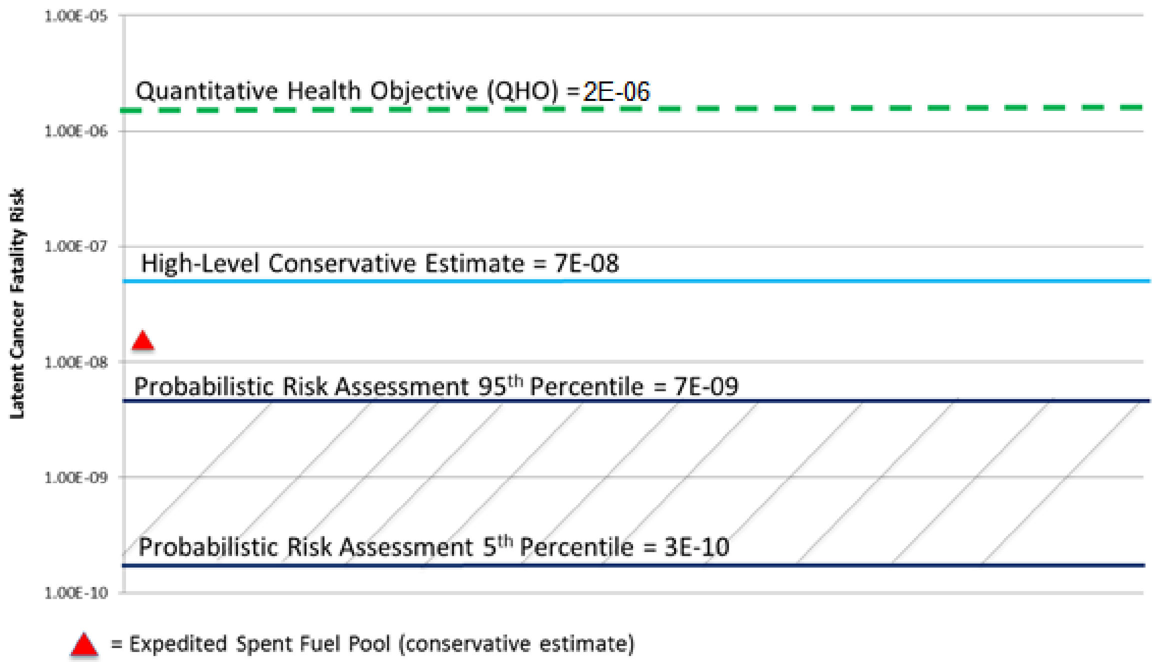
Figure 3-1: Calculation of High-Level Conservative Estimate



This conservative estimate uses ELAP and ILCF risk values from more than one plant that are the highest among BWRs with Mark I and II containments. Using ELAP and ILCF risk values at individual plants would have resulted in values lower than the conservative high-level estimate calculated during this screening process and would be even lower than the QHO and would be in the same order of magnitude as the Expedited Spent Fuel Pool ILCF risk discussed in COMSECY-13-0030, “Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel,” dated November 25, 2013 (ADAMS Accession No. ML13329A918).[18] This is depicted in Figure 3-2, “Relative Latent Cancer Fatality Risk.”

18 Using the bounding frequency of damage to the spent fuel pool (3.5×10^{-5} per year) which considers all initiators that could challenge spent fuel pool cooling or integrity, and the estimated conditional ILCF risk of 4.4×10^{-4} from NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor” (ADAMS Accession No. ML14255A365), yields a conservative high estimate of ILCF risk of 1.5×10^{-8} per reactor year.

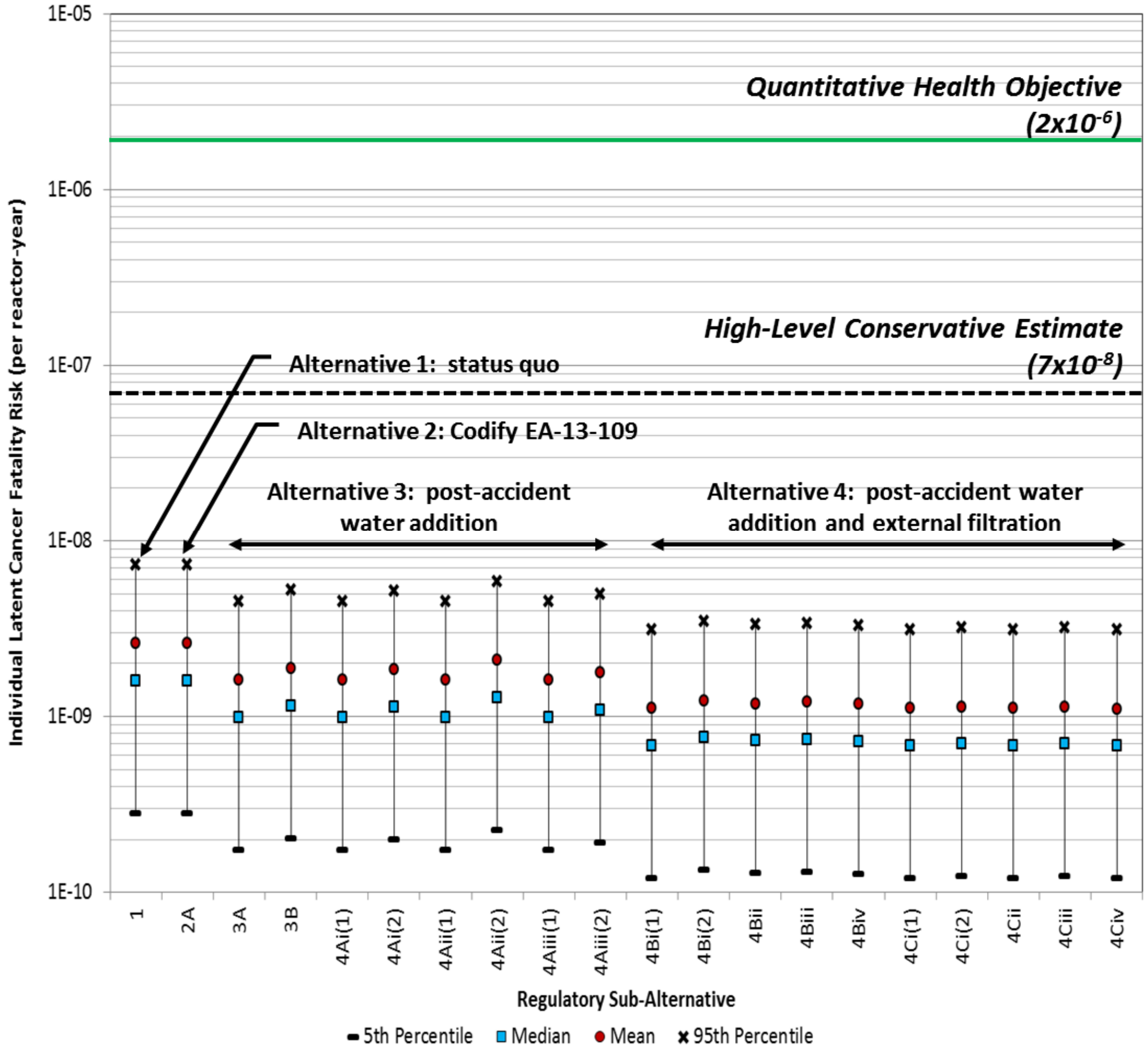
Figure 3-2: Relative Latent Cancer Fatality Risk



Comparing the high-level conservative estimate to the safety goal and QHO involves important considerations. First, the safety goal and QHO are intended to encompass all accident scenarios at a nuclear power plant, including the reactor and spent fuel pool. This regulatory basis does not examine all scenarios that would need to be considered in a full scope PRA, although ELAP is considered an important contribution to plant risk. While this comparison is incomplete it does show that the high-level conservative estimate has a margin that is greater than one order of magnitude to the QHO and therefore the ILCF risk is unlikely to challenge the NRC's safety goal policy.

The staff concludes that none of the alternatives would provide a substantial safety benefit based on this comparison to the QHOs. The conservative high-level estimate above takes no credit for any of the accident strategies and capabilities described in the 20 CPRR alternatives and sub-alternatives (see Table 4-3, “Summary of Regulatory Analysis Alternatives”). The delta benefit for each alternative and sub-alternative, compared to the status quo and Order EA-13-109, are shown in Figure 3-3, “Uncertainty Bounds for Individual Latent Cancer Fatality Risk.” The status quo band and the band for EA-13-109 are also shown in this Figure. Note that if licensees elect to implement SAWA/SAWM as part of compliance with EA-13-109, the risk bands labeled as alternative 3 would apply. However, because SAWA/SAWM is not specifically required by EA-13-109, it is not credited in Figure 3-3 for alternatives 1 and 2.

Figure 3-3: Uncertainty Bounds for Individual Latent Cancer Fatality Risk



If an ELAP occurs and results in core damage, an engineered filtered containment venting system would provide a notable reduction in offsite consequences (see Figures 4-23 and 4-24). However, because the frequency-weighted ILCF risk for alternative 1 (i.e., status quo) is already well below the associated QHO, the staff has concluded that the design and installation of an engineered filtered containment venting system or a performance based confinement strategy for BWRs with Mark I and Mark II containments does not meet the threshold for a substantial safety enhancement (see Figure 3-3). Therefore, the staff is recommending that rulemaking not be pursued for options 3 or 4 from SECY-12-0157 and is not recommending performing a more detailed analysis based on the margin to the QHOs. In the staff's judgment, a detailed regulatory analysis of the various alternatives is not warranted and would provide little additional insight into the regulatory decision. The staff instead provides the following discussion of the alternatives in terms of relationships to the ongoing regulatory activities such as implementation of Order EA-13-109 and qualitative factors discussed in SECY-12-0157. In addition, the staff

considered Commission decisions such as the SRM related to SECY-12-0110,[19] "Staff Requirements–SECY-12-0110–Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework" (ADAMS Accession No. ML13079A055).

3.2 Evaluation of Alternatives

Alternative 1: No Action (maintain status quo)

A plausible alternative is for the NRC to maintain the current regulatory requirements defined in regulations and licenses, including Order EA-13-109, for BWR Mark I and II containments. This approach was taken by the NRC following previous assessments, such as the CPIP, and is consistent with the established NRC policy and guidance for assessing potential requirements to address severe reactor accidents. The requirements included in EA-13-109 would remain in effect unless amended by a future generic or plant-specific licensing action. The significant safety benefit of improved containment venting capabilities for decay heat removal and containment over-pressure protection would be maintained.

Although Order EA-13-109 was imposed as a means to improve capabilities for decay heat removal and containment over-pressure protection, the approach proposed by industry and endorsed by the NRC to implement the order also includes the addition of water to the drywell during severe accident conditions and thereby provides benefits related to reducing temperatures and cooling molten core debris.

The implementation options proposed by industry include SAWA/SAWM to address Phase 2 of Order EA-13-109, which requires either (1) installation of a severe accident capable vent from the drywell, or (2) develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell before alternate reliable containment heat removal and pressure control is reestablished.

Although Phase 2 of EA-13-109 is expected to include SAWA/SAWM and thereby help address containment failure modes other than over-pressurization, licensees continue to have the option to develop other approaches to compliance with the order that would not include SAWA/SAWM or otherwise address dominant failure modes for Mark I and II containments.

This alternative is the status quo case for evaluating the other three alternatives in terms of potential risk reductions, regulatory effectiveness, and the additional costs to develop and implement plant changes and accident management strategies.

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19 At the time that SECY-12-0157 was prepared, the Commission was considering the possible adoption of revised decisionmaking criteria as discussed in SECY-12-0110. Commission decisions related to SECY-12-0110 and other policy matters related to possible lessons learned from the Fukushima accident have clarified the positions of the agency and reinforced the applicability of well-established guidance and practices related to evaluating potential regulatory actions.

Alternative 2: Rulemaking To Make Generically Applicable Order EA-13-109

The NRC usually incorporates requirements imposed by orders into its regulations or plant licenses. This standard practice ensures that the requirements included in the order are better integrated with other technical and administrative requirements. The process of incorporating order-imposed requirements into regulations also provides an opportunity for public involvement. In this case, however, the preparation of SECY-12-0157, the Commission deliberations, and the subsequent development and issuance of Order EA-13-109 included public interactions comparable to the rulemaking process.

As shown in Figure 3-3, there is no risk reduction in this alternative compared with the status quo. The potential benefits are from making generically applicable the requirements in Order EA-13-109 related to improved reporting, change control, and other aspects of controlling licensing basis information. Similar to the first alternative, the expected method of compliance for Order EA-13-109 and the subsequent rulemaking includes SAWA/SAWM and thereby helps address containment failure modes other than over-pressurization. However, the basis for the rulemaking and related technical and change control requirements would remain focused on over-pressure protection (consistent with Order EA-13-109). Licensees could develop other approaches to the order and rule developed under this alternative that would not include SAWA/SAWM or otherwise address dominant failure modes for Mark I and II BWR containments. The potential costs of this alternative relative to the rulemaking process and other administrative costs incurred by the NRC and nuclear industry is expected to be minimal and documented during the proposed rule phase.

Alternative 3: Rulemaking To Make Generically Applicable Order EA-13-109 and Additional Requirements for SAWA To Address Uncontrolled Releases from Major Containment Failure Modes

This alternative includes the requirements imposed by Order EA-13-109 to provide additional capabilities to reduce the likelihood of BWR Mark I and II containments failing from over-pressure conditions and introduces requirements for capabilities to reduce the likelihood of failure from over-temperature conditions or liner melt-through. The general approach to address over-temperature conditions and prevent liner melt-through has long been included in severe accident management guidelines (SAMGs) and includes the same measures discussed above as SAWA. The proposed inclusion of SAWA as an element of implementing Phase 2 of Order EA-13-109 and its use to address other containment failure modes results in a possible synergistic means to improve overall severe accident management for BWR Mark I and II containments. This alternative therefore focuses on this approach and discussions were limited to only variations of SAWA involving flow paths directly to the drywell or indirectly to the drywell through the reactor vessel and the breach in the vessel from the molten core debris. In addition, this alternative provides the same administrative benefits as alternative 2 related to improved reporting, change control, and other aspects of controlling licensing basis information.

In its SRM related to SECY-12-0157, the Commission offered the following direction to the NRC staff:

> … In the rulemaking technical bases, the staff should evaluate 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),

> or other measures that may be developed during the stakeholder engagement. The rulemaking technical bases should include a discussion of validation and testing that would be required to support various options. The staff should ensure that the performance and risks of the various filtering strategies and equipment considered are fully evaluated. The rulemaking should fully explore the requirements associated with measures to enhance the capability to maintain containment integrity and to cool core debris.

The results from the safety goal screening evaluation concluded that the benefits from additional containment protection measures are well below the threshold for proceeding with a detailed regulatory analysis or the substantial safety benefit criteria typically used for the NRC's backfit requirements (see Figure 3-2). The results of the QHO screening evaluation for this assessment are generally consistent with the findings discussed in SECY-12-0157. In that case, the staff proceeded with a more detailed cost-benefit assessment and ultimately provided recommendations to the Commission based largely on the consideration of qualitative factors. The NRC guidance for considering regulatory actions recognizes that quantitative assessments are supplemented by other factors and the technical judgment of the staff and Commission. For example, NUREG/BR-0058 offers the following guidance for situations related to containment-related issues:

> The NRC recognizes that in certain instances, the screening criteria [related primarily to core damage frequency] may not adequately address certain accident scenarios of unique safety or risk interest. An example is one in which certain challenges could lead to containment failure after the time period adopted in the safety goal screening criteria, yet early enough that the contribution of these challenges to total risk would be non-negligible, particularly if the failure occurs before effective implementation of accident management measures. In these circumstances, the analyst should make the case that the screening criteria do not apply and the decision to pursue the issue should be subject to further management decision.

The staff reviewed the results from various analyses in deciding if the subject alternatives to maintain containment integrity and cool core debris warranted further consideration. Although analytical results were available to assess the proposals directly to the NRC quantitative health objectives (QHOs), the staff also describes several other parameters in Section 5, "Performance Criteria Information," that can be used within the regulatory analyses to characterize the possible safety benefits. Comparisons of the various alternatives in terms of risk reductions and other parameters to the status quo (i.e., no action) are provided in Figure 3-3, "Uncertainty Bounds for Individual Latent Cancer Fatality Risk," and in the discussions in Section 4, "Technical Evaluation." Additional information on comparisons of alternatives was provided during many public meetings and in the EPRI report "Technical Basis for Severe Accident Mitigating Strategies, Volume 1," dated April 2015. The staff also considered that SAWA results in a similar risk reduction —in fact, without measures to address other failure mechanisms (e.g., liner melt-through), the benefits from severe accident containment vents are significantly reduced. Based on these considerations, the staff pursued a preliminary cost-benefit assessment[20] (see Section 3.3, "Cost-Benefit Assessment for Severe Accident Water Addition").

20 The safety goal screening evaluation, as outlined in the regulatory analysis guidelines

However, SAWA is being pursued as an integral part of containment over-pressure protection addressed by Order EA-13-109. In this case, the safety benefit of SAWA in terms of broader containment protection, from over-temperature conditions and liner melt-through, are provided and the costs of the modifications are incurred as part of implementing the order. The inclusion of SAWA helps to maintain consistency with established severe accident management guidance and provides a more logical approach to containment protection than does the current approach of requiring measures only for containment over-pressure conditions. Table 3-1, "Containment Failure Modes by Alternatives," summarizes the containment failure modes addressed by each alternative.

Table 3-1: Containment Failure Modes by Alternatives

Reactor Conditions	Containment Failure Mode and Release Reduction	Alternative 1 (No Action)	Alternative 2 (Rulemaking EA-13-109)	Alternative 3 (Rulemaking EA-13-109 with SAWA Requirement)	Alternative 4 (Release Reduction)
No Core Damage	Over-Pressure	Addressed	Addressed	Addressed	Addressed
Core Damage (Severe Accident)	Over-Pressure	Addressed	Addressed	Addressed	Addressed
	Over-Temperature	No Requirement	No Requirement	Addressed	Addressed
	Liner Melt-Through	No Requirement	No Requirement	Addressed	Addressed
	Release Reduction during Controlled Venting	Not Addressed	Not Addressed	Not Addressed	Addressed

The discussion of qualitative factors within SECY-12-0157 provides additional support for making generically applicable the requirements of Order EA-13-109 with additional measures to include SAWA. The combination of the order, SAWA, and other post-Fukushima activities related to mitigating strategies, severe accident management, and emergency preparedness provide benefits in terms of each of the qualitative factors listed in SECY-12-0157. Table 3-2, "Qualitative Factors for Alternative 3 (Containment Protection)," provides a summary of each factor for improved containment protection for BWR Mark I and II containments.

(NUREG/BR-0058), is designed to answer when a regulatory requirement should not be imposed generically on nuclear power plants because the risk is already acceptably low. This evaluation is intended to eliminate some proposed requirements from further consideration independently of whether they could be justified by a regulatory analysis on their net value basis. The safety goal evaluation can also be used for determining whether the substantial added protection standard of 10 CFR 50.109(a)(3) is met. However, the guidance is not intended to remove all flexibility and judgment from the backfit process and therefore points out that the safety goal screening evaluation is not intended to block worthwhile safety or security improvements that would otherwise be found to be cost-beneficial. Use of this guidance therefore requires a judgment by the NRC as to whether the safety goal screening evaluation provides an unreasonable finding on whether a proposed action provides a marginal or substantial safety improvement.

Table 3-2: Qualitative Factors for Alternative 3 (Containment Protection)

Factor	Discussion
Providing defense-in-depth (including importance of containment function).	Alternative 3 provides containment protection measures against the dominant failure mechanisms (over-pressure, over-temperature, and liner melt-through) and thereby supports the general desire for defense-in-depth. The technical analysis supports that addressing gross containment failures via venting and SAWA provides a safety enhancement with minimal costs and addresses much of the defense-in-depth concerns identified in SECY-12-0157 (i.e., balancing measures to (1) prevent accidents from occurring or progressing, (2) contain radioactive materials if released from the fuel, and (3) mitigate the possible release through protective actions, such as evacuation). Although not needed to meet the established safety goal, alternative 4 would provide additional benefits in terms of reducing releases (an added barrier) during venting operations.
Addressing significant uncertainties (frequencies and consequences).	Alternative 3 provides an additional capability to deal with uncertainties in severe accident frequencies and consequences. While the consequences from all scenarios modeled would be further reduced by installation of engineered filters (i.e., alternative 4), the additional requirements associated with alternative 3 provide comparable results when using the NRC's established safety goals. The alternative addresses many but not all of the issues related to analytical uncertainties raised in SECY-12-0157 and addressed by alternative 4. Section 4, "Technical Evaluation," reviews the issues related to analytical uncertainties.
Supporting severe accident management and response.	The development of strategies for implementing Order EA-13-109 and the related introduction of SAWA has been proposed by the industry and is being coordinated with their efforts to implement other Fukushima-related improvements to severe accident management. With the inclusion of SAWA, this alternative better addresses the issues related to coordination of additional requirements for containment

Factor	Discussion
	protection and severe accident management than did the options discussed in SECY-12-0157. Whereas alternative 4 could also be incorporated in SAMG, the potential addition of a requirement for engineered filters via rulemaking could delay the ongoing EA-13-109 activities by causing licensees to reevaluate planned modifications and approaches for SAWA and containment venting systems, including the incorporation of these approaches into SAMGs.
Improving hydrogen control.	The installation of severe accident capable vents (Order EA-13-109) provides the majority of the improvements in hydrogen control that were raised in SECY-12-0157. The SAWA and venting components in both alternatives 3 and 4 result in comparable benefits for hydrogen control.
Addressing external events.	The installation of severe accident capable vents (Order EA-13-109) and the coordination of this activity with mitigating strategies (Order EA-12-049) addresses external events with no significant differences between the evaluated alternatives (see SECY-12-0157 and its Enclosure 3 for seismic-related scenarios).
Addressing multi-unit events.	Each alternative, including implementation of Order EA-13-109, is being developed to provide safety enhancements assuming multi-unit events. As discussed in SECY-12-0157, the primary benefit is from having installed equipment and strategies, and alternative 3 addresses this factor nearly as well as alternative 4.
Considering independence of barriers.	The development of approaches for mitigating strategies (Order EA-12-049) and containment venting (Order EA-13-109) reinforced that the best overall approach to safety for BWR Mark I and II plants includes venting in order to maintain or restore core cooling. Complete independence of barriers is therefore not necessary or practical for these plants but alternative 3 does provide capabilities (venting and SAWA) to help maintain the containment function during severe accident conditions. While the consequences from all scenarios modeled would be further reduced by installation of engineered filters (i.e. alternative

Factor	Discussion
	4), containment venting is still required to maintain core cooling. The additional requirements associated with alternative 3 provide comparable results when using the NRC's established safety goals.
Improving emergency planning.	Alternative 3 provides capabilities to address major containment failure modes and to delay and limit radiological releases, albeit not to the degree of engineered filters for some specific scenarios. This alternative does address much of the concern identified in SECY-12-0157 in terms of addressing uncontrolled releases due to containment failures from over-pressurization and the inclusion of SAWA addresses other failure modes not addressed by venting operations.
Considering consistency between reactor technologies.	As discussed in SECY-12-0157, any action to address severe accident conditions for BWRs does introduce the potential for some inconsistencies between the regulatory requirements for the various reactor technologies. NRC decisions on possible changes in requirements for other containments (Recommendation 5.2) will result from longer term studies. However, alternative 3 does improve the consistency between the protection provided by reactor containment technologies by providing capabilities to improve the performance of Mark I and II containments, which have been previously shown to have a higher conditional probability of failing during severe accident conditions. Insights from this evaluation and related Commission decisions will support future assessments and consistent treatment of other reactor technologies.
Considering severe accident policy statement.	As discussed in SECY-12-0157, the rulemaking process and the regulatory analyses are more consistent with the NRC's severe accident policy statement than were the orders recommended in that paper. The importance of qualitative factors in the justifications provided in SECY-12-0157 suggested a need to revisit portions of the current regulatory framework (including the Severe Accident Policy Statement). The adoption of alternative 3 would require some

Factor	Discussion
	qualitative considerations to support a finding of cost-justified substantial safety enhancement but is generally consistent with the established backfit process (i.e., consistent with the policy statement).
Addressing international experience and practices (including availability of technology).	Sections 3 and 4 describe several different criteria for regulatory decisionmaking and include some measures different from the NRC's but used by regulators in other countries. Most regulators are implementing requirements like alternative 4, but do not have analogous cost/benefit considerations, safety goals, and severe accident policies in their decisionmaking processes. For the majority of their nuclear plants most countries are requiring external filters based on a reduction of radiological impacts. In these countries it appears sufficient to conclude that filtered containment venting systems are beneficial to address severe accidents. However, in the United States the calculated benefits are offset by the low probability associated with accidents where the filtered containment venting option would be useful.

Although the calculated safety benefits from SAWA in providing additional containment protection capabilities would not be justified using the NRC's established screening guidance for backfit evaluations, the staff finds that the relationship to Order EA-13-109 and consideration of other factors provide a logical and sufficient basis for the NRC to incorporate this alternative into the CPRR rulemaking. The planned implementation of SAWA as an element of Order EA-13-109 allows for its use to address other containment failure modes without significant additional costs to licensees or the NRC. The relationship between Order EA-13-109 and the rulemaking activities has always been recognized and was discussed within the Commission's SRM for SECY-12-0157, Order EA-13-109, and the numerous public meetings related to implementation of the order and evaluation of the rulemaking. The potential costs of this alternative relative to the rulemaking process and other administrative costs incurred by the NRC and nuclear industry is expected to be minimal and will be documented during the proposed rule phase.

Alternative 4: Reduce Releases during Controlled Venting (Filtering Strategies, Engineered Filters)

In addition to measures to prevent an uncontrolled release of radioactive materials resulting from containment failures, the Commission directed that the staff assess possible requirements to reduce or filter planned releases during containment venting operations following core damage. As described above for the evaluation of potential requirements for SAWA/SAWM, the process involves first determining if imposing such requirements provide a substantial safety

enhancement, and if so, whether such a requirement would be justified in light of the associated costs. This alternative would include all the costs and benefits of alternative 3, but would add a requirement for licenses to be able to reduce or filter potential releases. The options considered under the alternative associated with reducing potential releases from planned venting of containments during severe accident conditions include:

- filtering strategies using containment wetwells and existing plant features

- additional fission product removal in the venting flow path using:

    - an external filter with a large capacity to handle aerosols and other contaminants (e.g., not expected to block or clog filter during event)

    - an external filter with a limited capacity to handle aerosols and other contaminants (e.g., potential need to bypass filter for longer-term venting operations)

In addition, the staff considered existing Commission policy as defined by the Severe Accident Policy Statement and more recent Commission decisions such as in SRM-SECY-12-0110,[21] "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated March 20, 2013 (ADAMS Accession No. ML13079A055) and SRM-SECY-14-0087, "Qualitative Consideration of Factors in the Development of Regulatory Analyses and Backfit Analyses," dated March 4, 2015 (ADAMS Accession No. ML15063A568). Based on these considerations and the expected significant costs associated with achieving these marginal risk reductions, the staff determined that a more detailed cost-benefit assessment was not warranted.[22] Unlike the earlier case of including SAWA/SAWM in the proposed rulemaking, the possible addition of requirements for further release reductions from controlled venting operations (filtering strategies or engineered filters) is not directly related to or addressed by ongoing activities. The assessment of the qualitative factors provided in SECY-12-0157 and the previous discussion under alternative 3 (see Table 3-2) concluded that much of the unquantified benefits from possible regulatory actions are achieved through the containment protection measures (alternative 3). A discussion of the factors relating only to reducing the planned releases from containment venting operations is provided below in Table 3-3, "Qualitative Factors for Alternative 4 (Release Reduction)."

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21 At the time that SECY-12-0157 was prepared, the Commission was considering the possible adoption of revised decisionmaking criteria as discussed in SECY-12-0110. Recent Commission decisions related to SECY-12-0110, SECY-14-0087 and other policy matters related to possible lessons learned from the Fukushima accident have clarified the positions of the agency and reinforced the applicability of well-established guidance and practices related to evaluating potential regulatory actions.

22 Notwithstanding the decision to not pursue a detailed cost-benefit assessment, cost information was solicited and obtained from the nuclear industry. The cost information for the engineered filters is provided in a letter from NEI dated May 31, 2014 (ADAMS Accession No. ML14188C457) and reviewed in Section 3.3, "Cost-Benefit Assessment for Severe Accident Water Addition".

Table 3-3: Qualitative Factors for Alternative 4 (Release Reduction)

Factor	Discussion
Providing defense-in-depth (including importance of containment function).	Alternative 4 reduces the release related to all scenarios modeled and thereby provides the greatest defense-in-depth measures beyond the containment protection capabilities compared to alternative 3. However, unlike pursuing only alternative 3, there is a significant cost to achieve the additional safety benefits as evaluated against the NRC's established safety goal policy statement and guidance related to imposing backfits.
Addressing significant uncertainties (frequencies and consequences).	Alternative 4 provides some advantages in terms of reducing the consequences from venting operations during a severe accident. This alternative addresses many of the issues related to analytical uncertainties raised in SECY-12-0157 but the analytical results provided in Section 4 and Figure 3-2 above show limited advantages for alternative 4 over alternative 3 when evaluated against NRC safety goals and related guidance. However, when evaluated against the additional offsite consequence measures analyzed in Section 4.5 below, alternative 4 does show advantages.
Supporting severe accident management and response.	Alternative 4 is not currently being integrated into the SAMGs being developed by the nuclear industry. While filtering strategies or engineered filters could be incorporated into the guidelines, a proposal to further investigate filtering strategies or the installation of engineered filtering systems could delay the current efforts to integrate containment venting, SAWA, and other severe accident management activities.
Improving hydrogen control.	The installation of severe accident capable vents (Order EA-13-109) provides the majority of the improvements in hydrogen control that were raised in SECY-12-0157. The SAWA and venting components in both alternatives 3 and 4 result in comparable benefits for hydrogen control.
Addressing external events.	The installation of severe accident capable vents (Order EA-13-109) and the coordination of this activity with mitigating strategies (Order EA-12-049) address external events with no significant differences between the evaluated

Factor	Discussion
	alternatives (see SECY-12-0157 and Section 4 for seismic-related scenarios).
Addressing multi-unit events.	Each alternative, including implementation of Order EA-13-109, is being developed to provide safety enhancements assuming multi-unit events. As discussed in SECY-12-0157, the primary benefit is from having installed equipment and strategies so alternative 4 offers no significant advantage for this factor over alternative 3 except for the previously mentioned reduction in releases associated with containment venting during severe accident conditions.
Considering independence of barriers.	While the consequences from all scenarios modeled would be further reduced by installation of engineered filters, the additional requirements associated with alternative 3 provide comparable results when using the NRC's established safety goals. Alternative 4 addresses most of the independence-related issues identified in SECY-12-0157 but at a significant cost given the limited benefits over the venting and SAWA measures that would be required under alternative 3.
Improving emergency planning.	This alternative does address most of the concerns identified in SECY-12-0157 in terms of controlled releases during venting operations following significant core damage. Section 4 and Figure 3-3 show the relative benefits from the various alternatives and shows only minimal benefits for additional release reduction measures (e.g., engineered filters) when using the NRC's established safety goals and related guidance.
Considering consistency between reactor technologies.	(See Alternative 3 discussion) Insights from this evaluation and related Commission decisions will support future assessments and consistent treatment of other reactor technologies.
Considering severe accident policy statement.	Unlike the discussions provided for alternative 3, proposals to require additional release reduction measures such as engineered filtering systems would, similar to SECY-12-0157, require a heavy reliance on qualitative factors to overcome the negative results from detailed quantitative analyses. Alternative 4 not only has limited safety

Factor	Discussion
	benefits but also relatively high costs compared to alternative 3, for which the costs are largely a result of the implementation of Order EA-13-109.
Addressing international experience and practices (including availability of technology).	Sections 3 and 4 describe several different criteria for regulatory decisionmaking and includes some measures different from the NRC's but used by regulators in other countries. Most regulators are implementing requirements like alternative 4, but do not have analogous cost/benefit considerations, safety goals, and severe accident policies in their decisionmaking processes. For the majority of their nuclear plants most countries are requiring external filters based on a reduction of radiological impacts. In these countries it appears sufficient to conclude that filtered containment venting systems are beneficial to address severe accidents. However, in the United States the calculated benefits are offset by the low probability associated with accidents where the filtered containment venting option would be useful.

As previously discussed, the possible pursuit of additional measures to reduce releases associated with containment venting during severe accident conditions provides only minimal safety benefits while having a significant cost of development and implementation. The limited benefits and high costs result in alternative 4 not being a cost-justified substantial safety enhancement or otherwise being worth pursuing under the NRC's established regulation and guidance for regulatory analyses and plant backfits.

3.3 Cost-Benefit Assessment for Severe Accident Water Addition

The staff considered two sub-alternatives for SAWA/SAWM involving different flow paths for the addition of water:

- water addition to the reactor vessel
- water addition to the containment drywell

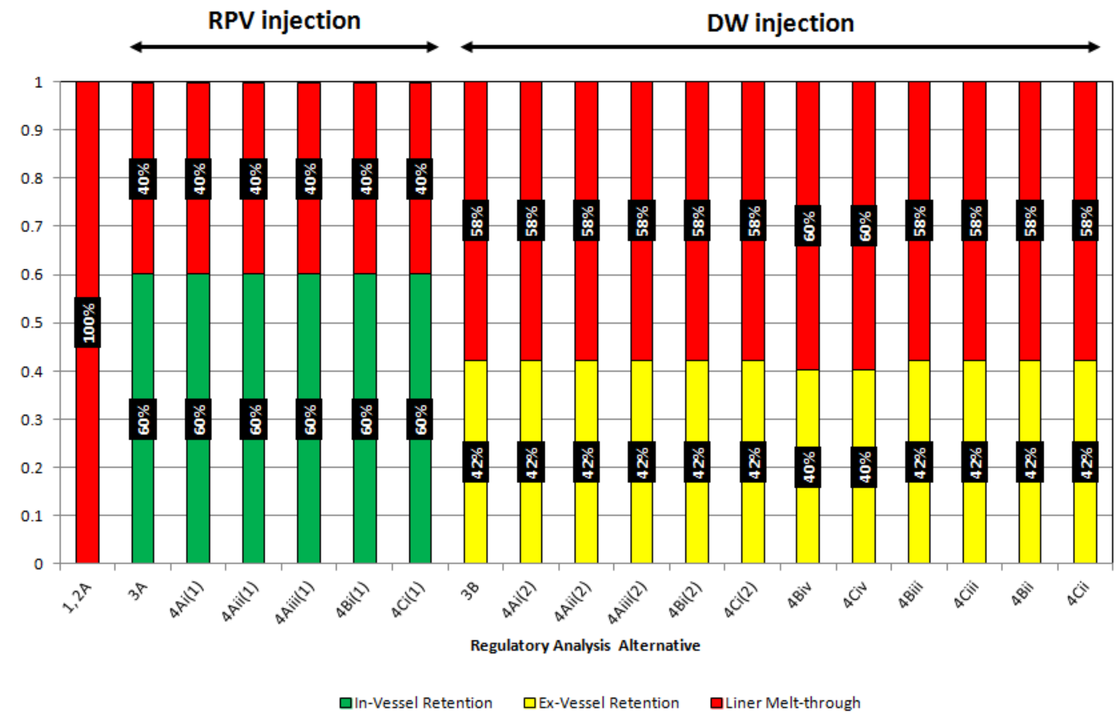
Before conducting the detailed cost-benefit assessment of both sub-alternatives, the staff considered whether there was a significant difference between the safety benefits associated with the two sub-alternatives. This approach reflects that while the NRC is allowed under its backfit regulation to impose plant modifications to address safety concerns, if there are two or more ways to reach a level of protection which is adequate, then the NRC could allow the applicant or licensee to choose the way which best suits its purposes.

SAWA/SAWM installation costs the licensee from approximately \$2.6 million to \$3.2 million. The expected installation costs for engineered filtered vents range from \$11 million to \$64

million. A more detailed analysis of alternative 3 will occur during the proposed rulemaking phase. These estimates may increase when more engineering details are available.

Figure 3-4, “Comparison of Ability To Retain Core Debris,” shows the calculated benefit in terms of the risk of containment failures and large releases for adding water (i.e., SAWA) to the reactor pressure vessel (RPV) or drywell (DW). This small difference in calculated risk reduction between the RPV and DW relates to the opportunity in some selected accident sequences to retain the molten core within the reactor vessel.

Figure 3-4: Comparison of Ability to Retain Core Debris



The differences between the two sub-alternatives are not significant in terms of conditional containment failure probabilities or other parameters considered and are small in comparison to the overall uncertainties in the computer simulations. The staff is, therefore, not proposing to require one sub-alternative over another and conducted the assessment on the assumption that licensees could pursue either sub-alternative—perhaps with a preference to water addition to the reactor vessel where plant modifications were practical for either flow path.

3.4 Backfitting Considerations

The proposed CPRR rulemaking would make generically applicable in § Part 50 the requirements in Order EA-13-109 with an additional requirement for the use of SAWA/SAWM (i.e., alternative 3) such that they would become requirements for BWR nuclear power plants with Mark I and II containments. To the extent that the proposed CPRR rule would make the requirements of Order EA-13-109 generically applicable, the proposed rule, as applied to existing licensees to whom Order EA-13-109 was directed, would not constitute backfitting under § 50.109.

The addition of a requirement for a SAWA/SAWM capability could constitute a backfit. However, the NRC staff has concluded that SAWA/SAWM provide worthwhile additional protection for public health and safety by protecting the integrity of the containment and reducing the release of radioactive materials in some severe accident events. Therefore, should the NRC staff determine, after receiving comments on this draft Regulatory Basis, that the rulemaking described as alternative 3 remains the preferred approach, the NRC staff will determine if a backfit analysis is required and ensure that any proposed rule is in compliance with the requirements of 10 CFR 50.109.

3.5 Regulatory Evaluation Conclusion

Based on the considerations discussed above, the staff is planning to pursue a rulemaking to make the requirements of Order EA-13-109 generically applicable, with an additional requirement for the use of SAWA/SAWM (i.e., alternative 3). This approach would provide the administrative benefits described for alternative 2, while also including the potential synergistic severe accident mitigation opportunities associated with SAWA/SAWM. Unlike alternative 4, a rulemaking to make Order EA-13-109 generically applicable with additional requirements for SAWA/SAWM seems unlikely to have significant additional costs for licensees because, as far as the staff is aware, licensees are currently planning to adopt SAWA/SAWM strategies as part of the implementation of Order EA-13-109. Alternative 4 would provide some additional safety enhancements, but as discussed above provides only minimal safety benefits with regard to the QHOs while having a significant cost of development and implementation.

The Commission in SRM-SECY-12-0157 directed the NRC staff to include a discussion of validation and testing that would be required to support the various options. The testing and validation discussed in SECY-12-0157 related to the abilities of filtering strategies and engineered filters to limit the release of radioactive materials. The technical evaluation for this regulatory basis, included below as Section 4, assumed various filter efficiencies (i.e., decontamination factors) but none were found to warrant further regulatory action. The staff is not considering any additional validation and testing requirements in the proposed CPRR rulemaking outside of the expected demonstrations of compliance with Order EA-13-109.

4. Technical Evaluation

This section summarizes the staff's technical evaluation of venting and severe accident water addition (SAWA) and management (SAWM) strategies that are likely to be adopted during the accident progression following an extended loss of ac power (ELAP).

The SRM to SECY-12-0157 directed the staff to develop the regulatory basis and rulemaking for filtering strategies with drywell filtration and severe accident management of BWR Mark I and II containments. Specifically, the SRM to SECY-12-0157 directed that the technical basis and rulemaking should consider Option 3 from SECY-12-0157—design and installation of an engineered filtered containment venting system 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; and Option 4 from SECY-12-0157—development of requirements and technical acceptance criteria for confinement strategies and requirements for licensees to justify operator actions and systems or combination of systems, such as suppression pools, containment sprays, and separate filters to accomplish the function and meet the requirements.

In addition, the SRM to SECY-12-0157 directed that the regulatory basis 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. The SRM directed the staff to include the following in the development of the regulatory basis:

- evaluate 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), or other measures that may be developed during the stakeholder engagement

- include a discussion of validation and testing that would be required to support various options

- ensure that the performance and risks of the various filtering strategies and equipment considered are evaluated

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

To respond to this direction, the staff has developed a technical approach to provide the following for Mark I and Mark II BWRs:

- a risk evaluation that includes quantitative risk estimates and qualitative risk insights for various CPRR strategies

- 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 the MELCOR code to provide inputs to MACCS (MELCOR Accident Consequence Code System) and to provide insights on the effect of the different strategies

- offsite consequence calculations using the MACCS code to obtain conditional health and societal risks for selected MELCOR source terms

- integration of the results of the steps above to arrive at the frequency-weighted offsite consequences corresponding to each of the different CPRR accident strategy alternatives.

In this evaluation, a number of accident progression scenarios are evaluated based on postulated failures to complete actions necessary to prevent core damage following ELAP initiation. These actions have been defined by the plant operators in the context of event time lines submitted to the NRC. Some of the actions considered are shown in Table 4-1, "Some Operator Actions Considered for ELAP Mitigation Analyses." If some of the actions to be taken before core damage are unsuccessful, then core damage would likely result. Other actions listed in Table 4-1 could be taken after core damage to arrest further core damage, prevent containment failure, and/or mitigate releases to the environment. This evaluation looks at the effects of taking the various actions.

Table 4-1: Some Operator Actions Considered for ELAP Mitigation Analyses

Events and Actions (~Elapsed Time)	Core Damaged?	Vessel Failed?	Remarks
DC power lost (0-6 hr); or recharging unsuccessful.	No	No	Batteries could fail initially or be depleted and not re-charged.
Early depressurization of the RPV (1-3 hr).	No	No	Cooldown at 80 ºF /hr.
Control RPV pressure between 200 and 400 psi (until RCIC fails).	No	No	Maximizes time for RCIC operation.
Early venting (4-10 hr).	No	No	Allows a heat removal path.
Further depressurize RPV after RCIC failure.	No	No	Allows for injection into RPV after RCIC lost.
Close vent at core damage.	Yes	No	Minimize fission product releases.
After RCIC fails, inject water into RPV when pressure low enough, <u>or</u> flood DW after vessel failure.	Yes	No	Arrest core damage, or prevent containment failure and reduce off-site consequences.
Re-open Wetwell (WW) vent at Pressure Suppression Pressure or Primary Containment Pressure Limit (PCPL).	Yes	Maybe	Maintain containment integrity; possible vent cycling.
Throttle flow and/or close WW vent if WW level too high.	Yes	Yes	Most likely at top of level instrument range. Prevents flooding the WW vent.
Open DW vent at PCPL.	Yes	Yes	Only if WW vent closed. Maintain containment integrity; possible vent cycling.
Turn off flow at high DW level.	Yes	Yes	Four feet above DW floor in plant studied.

In developing the analytical approach to evaluate the various strategies, a number of accident prevention and mitigation measures were considered. These were informed by:

- the lessons learned from the events at the Fukushima Dai-ichi Nuclear Power Plant.

- the accident management alternatives being contemplated by the industry.

- the current state of knowledge of severe accident progression and mitigation alternatives in a BWR.

- the experience gained from the previous effort documented in SECY-12-0157 to address the NTTF Recommendation 5.1.

The accident scenarios considered are those associated with an ELAP event caused by an external hazard (e.g., beyond-design-basis earthquake), thereby resulting in one of three possible outcomes: containment over-pressure 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 water level would decrease and the core would uncover, leading to heatup, degradation, and relocation of core debris into the lower plenum. The thermal loading of the RPV lower head would consequently cause the lower head to fail and relocation of core debris into the reactor cavity. The containment would pressurize during this event until it failed by over-pressure, over-temperature, liner melt-through, or other related mechanisms.

By far the most important mitigation strategy considered in the development of the regulatory basis relates to SAWA and SAWM, both in the reactor vessel and in the containment. For most BWRs, the reactor core isolation cooling (RCIC) system is designed to prevent loss of core cooling and subsequent accident progression until other DC-powered (battery or diesel generator) and portable mitigation systems become available. The operation of RCIC (when and how it fails) is a major consideration in developing the MELCOR calculation matrix. Once RCIC is lost, water would need to be supplied from an external source and pumped into either the RPV or the containment. The industry has committed to using this approach to mitigate the consequences of an ELAP by preventing core damage in compliance with Order EA-12-049 by using diverse and flexible coping strategies (FLEX). Other actions are required to make FLEX successful. The most important actions are shown as the first five events and actions in Table 4-1. If core damage cannot be averted, then the remaining actions in Table 4-1 apply.

Post-core damage mitigation strategies provide for water addition into either the RPV or the containment using severe accident-capable equipment. The SAWM strategy provides for controlled water addition (i.e., throttling water flow) to achieve a specified purpose or goal, for example, ensuring the wetwell is not flooded to avoid losing the fission product scrubbing benefit it provides relative to the drywell. The objective is to prevent containment failure and consequent uncontrolled release of radioactivity to the environment.

Along with water addition and management, containment venting is necessary to prevent containment failure until ac power has been restored and a normal decay heat removal path re-established. SECY-12-0157 considered venting when the containment pressure exceeded the primary containment pressure limit (PCPL). Since then, the BWR Owners Group introduced the concept of anticipatory early venting (pre-core damage venting at a pressure significantly below the PCPL) in the process of updating its severe accident management guidance (SAMG). It also is considering venting after core damage when the containment pressure reaches the pressure suppression pressure (PSP).

The goal of early venting is to establish a temporary heat removal path so that core damage and containment failure can be prevented. This action, by reducing the pre-core damage containment load, affords greater opportunity to implement mitigation measures to address

post-core damage containment performance. The early containment venting action is included in the MELCOR matrix, as well as vent cycling and transitioning from wetwell (WW) venting to drywell (DW) venting. Sensitivity to post core damage containment venting at PSP is included to simulate the venting required to maintain the pressure suppression function.

4.1 Risk Evaluation

The following subsections delineate the four CPRR alternatives (see Section 2.4, "Current Activities (CPRR Rulemaking)") considered in the risk evaluation, summarizes the technical approach and information used to develop the risk evaluation, and summarizes the results obtained.

4.1.1 Identification of Regulatory Basis Alternatives

To facilitate development of the risk evaluation, the four alternatives that broadly address Options 2, 3 and 4 in SECY-12-0157 were defined, as shown in Table 4-2, "Overview of CPRR Alternatives Considered in the Risk Evaluation."

Consideration of the information in Table 4-2 indicates that each of the four alternatives can be implemented using sub-alternative approaches, which are shown in Table 4-3, "Summary of Regulatory Basis Sub-Alternatives."

Table 4-2: Overview of CPRR Alternatives Considered in the Risk Evaluation

Alternative	Objective	Implementation Alternatives
1. Alternatives 1 and 2 - Require capability for post-core-damage containment venting. Note that the evaluation does not credit water injection since it is not specifically required by Order EA-13-109 but the expected implementation of the order includes SAWA/SAWM.	Prevent containment failure due to over-pressurization	• Location: - Wetwell - Drywell • Actuation: - Manual (operator action) - Passive (rupture disk)

Alternative	Objective	Implementation Alternatives
2. Alternative 3 - Requires capability to cool core debris, i.e., water injection pathway and containment venting. Note that the planned implementation of Order EA-13-109 includes SAWA/SAWM and this alternative only proposes to capture that capability as a regulatory requirement.	Prevent containment failure due to liner melt-through	• Injection Pathway: - Reactor pressure vessel - Drywell • Injection Control: - SAWA - SAWM – injected water flow is throttled as necessary to prevent submerging the wetwell vent, which helps to ensure that releases are scrubbed. • Venting – Same as Alternatives 1 and 2
3. Alternative 3 - Operate post-core-damage containment vents and water injection in a manner than enhances the containment's ability to retain and scrub radioactive releases.	Reduce the quantity of radioactivity released to the environment following a core-damage accident	• Vent operation: - Open-and-leave-open - Vent cycling – vent is opened as needed to prevent over-pressurization, then reclosed • Injection control: - SAWA - SAWM
4. Alternative 4 - Install engineered filters on the containment vents.	Reduce the quantity of radioactivity released to the environment following a core-damage accident	• Filter Capacity: - Large - Small

In order to fully evaluate the risks of the four alternatives, twenty sub-alternatives were defined, as shown in Table 4-3, "Summary of Regulatory Basis Sub-Alternatives." Risk estimates and results were developed for each of the twenty regulatory analysis alternatives.

Table 4-3: Summary of Regulatory Basis Sub-Alternatives

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

Venting Priority
DWF - drywell-first venting strategy
WWF - wetwell-first venting strategy
Venting Actuation
M - manual venting
P - passive venting (rupture disk)
Venting Operation Mode
AV - anticipatory venting; open at 15 psig and leave open
OLO - open at PCPL and leave open
VC - vent cycling at PCPL with 10 psig band

Post-Accident Injection Location
DW - drywell via external connection
RPV - reactor pressure vessel via external connection
Post-Accident Injection Operating Mode
SAWA - severe accident water addition
SAWM - severe accident water management
Filter Size
L - large filter
S - small filter

4.2 Technical Approach

The risk evaluation is a simplified probabilistic risk assessment (PRA) that models the response of a generic BWR plant with a Mark I containment design and a RCIC system to an ELAP event. The risk evaluation begins with the occurrence of an ELAP event, models the accident progression to core damage (Level 1 PRA), models the subsequent release of radioactive materials to the environment (Level 2 PRA), and combines estimates of the release frequencies

and their associated consequences to produce risk estimates (Level 3 PRA). The following sections summarize the technical approach used to complete each portion of the risk evaluation.

4.2.1 ELAP Modeling

In accordance with NTTF Recommendation 5.1, the risk evaluation of the CPRR alternatives was focused on accidents that are initiated by “a prolonged station blackout (SBO) event". An ELAP event is an SBO (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 frequency of ELAP is plant-specific because SBO coping duration depends on:

- the redundancy of the onsite emergency ac power sources,
- the reliability of the onsite emergency ac power sources,
- the expected frequency of loss of offsite power, and
- the probable time needed to restore offsite power.

The risk evaluation includes 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, "Reevaluation of Station Blackout Risk at Nuclear Power Plants" (ADAMS Accession No. ML060200477) and by seismic events. Focusing on this set of ELAPs made the risk evaluation tractable within the allotted schedule and budget. However, the staff recognizes that the four alternatives considered may also be beneficial in 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, accidents that are initiated by internal floods, internal fires, external floods and other types of external events, and accidents caused by deliberate malevolent acts such as sabotage or terrorism).

A complete assessment of CPRR alternatives that includes these types of accidents would require the development of plant-specific, internal and external event Level 3 PRAs for each BWR Mark I and II plant and is beyond the scope of this rulemaking. Nevertheless, the staff’s analysis of the accident scenarios considered in this regulatory basis captures a sufficient proportion of the risk to allow the staff to reach conclusions regarding the regulatory actions within the scope of this rulemaking.

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. LOOP frequencies and offsite power recovery curves were obtained from NUREG/CR-6890. The probability of EPS failure was obtained from the 2011 update to NUREG/CR-5500, Volume 5, “Reliability Study: Emergency Diesel Generator Power System, 1987-1993,” 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.

The frequency of seismic ELAPs was estimated by probabilistic convolution of the site-specific seismic hazard curve with a model 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. Seismic hazard curves were obtained from licensee responses to the 10 CFR 50.54(f) request associated with NTTF Recommendation 2.1. Seismic fragilities were obtained by developing a population variability distribution from information contained in Individual Plant Examination of External Events. No credit was taken for recovering offsite power after a seismic event.

Table 4-4 lists the estimated ELAP frequencies.

Table 4-4: ELAP Frequencies

Site	Containment Type	Notes	EPS Class	SBO Coping Time (hours)	ELAP Frequencies (/ry)		
					Internal Events	Seismic	Total
Browns Ferry	Mark I		4	4	4.1×10^{-7}	2.2×10^{-5}	2.3×10^{-5}
Brunswick	Mark I		2	4	6.9×10^{-6}	1.1×10^{-5}	1.8×10^{-5}
Columbia	Mark II		2	4	6.9×10^{-6}	5.0×10^{-5}	5.7×10^{-5}
Cooper	Mark I		2	4	6.9×10^{-6}	4.9×10^{-6}	1.2×10^{-5}
Dresden	Mark I	Isolation Condenser	4	4	4.1×10^{-7}	1.7×10^{-5}	1.7×10^{-5}
Duane Arnold	Mark I		2	4	6.9×10^{-6}	2.4×10^{-6}	9.2×10^{-6}
Fermi	Mark I		4	4	4.1×10^{-7}	8.8×10^{-6}	9.2×10^{-6}
FitzPatrick	Mark I		4	4	4.1×10^{-7}	3.5×10^{-6}	3.9×10^{-6}
Hatch	Mark I		3	4	3.4×10^{-6}	9.6×10^{-6}	1.3×10^{-5}
Hope Creek	Mark I		3	4	3.4×10^{-6}	7.6×10^{-6}	1.1×10^{-5}
La Salle	Mark II		3	4	3.4×10^{-6}	3.4×10^{-5}	3.8×10^{-5}
Limerick	Mark II		4	4	4.1×10^{-7}	1.0×10^{-5}	1.0×10^{-5}
Monticello	Mark I		2	4	6.9×10^{-6}	6.7×10^{-6}	1.4×10^{-5}
Nine Mile Point	Mark I – Unit 1 Mark II – Unit 2	Unit 1 – Isolation Condenser	2	4	6.9×10^{-6}	3.8×10^{-6}	1.1×10^{-5}
Oyster Creek	Mark I	Isolation Condenser	2	4	6.9×10^{-6}	8.3×10^{-6}	1.5×10^{-5}
Peach Bottom	Mark I		3	8	2.0×10^{-6}	3.8×10^{-5}	4.0×10^{-5}
Pilgrim	Mark I		2	8	4.1×10^{-6}	7.0×10^{-5}	7.4×10^{-5}
Quad Cities	Mark I		4	4	4.1×10^{-7}	7.0×10^{-6}	7.4×10^{-6}
Susquehanna	Mark II		3	4	3.4×10^{-6}	4.6×10^{-6}	8.0×10^{-6}

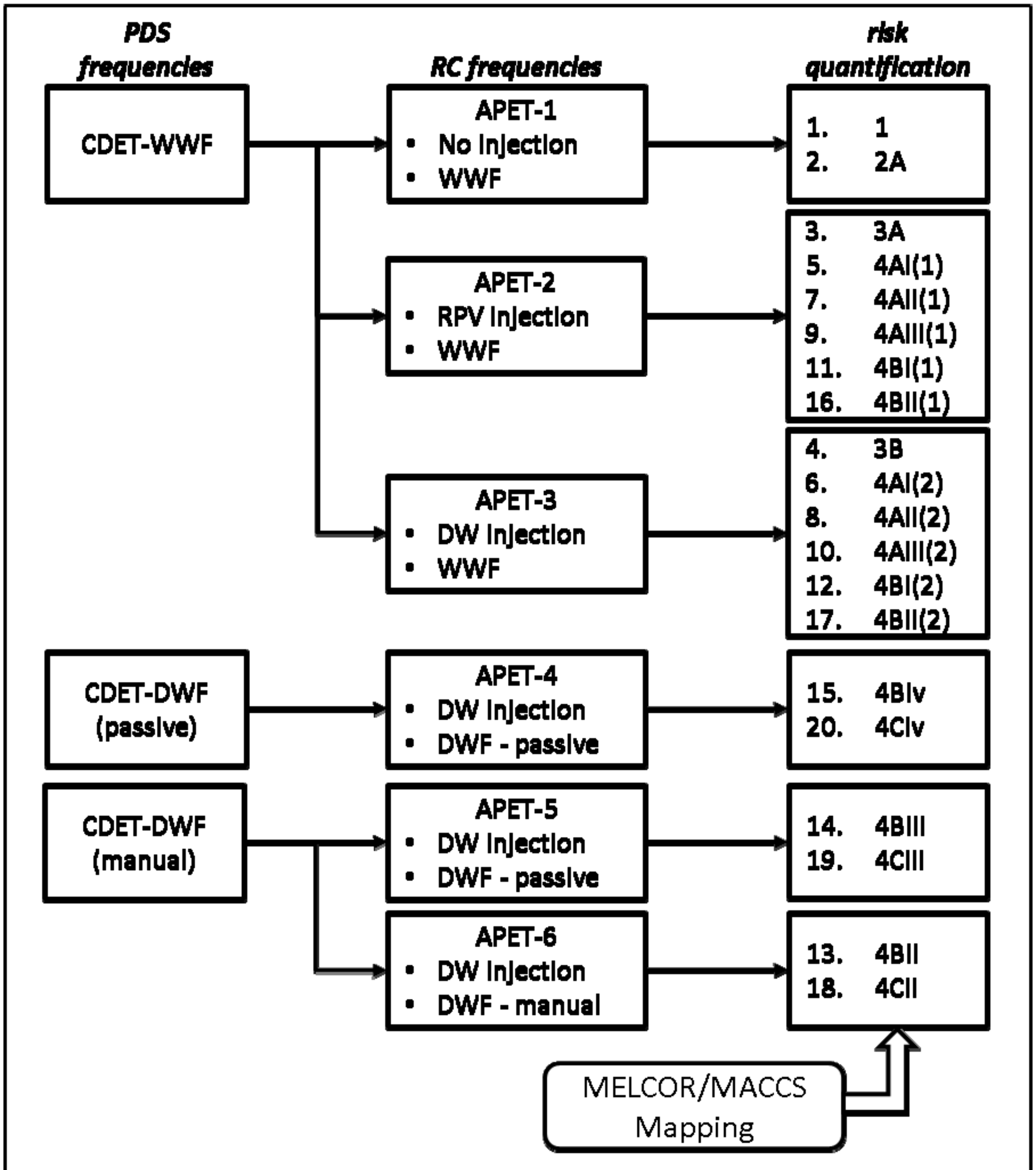
4.2.2 Accident Sequence Delineation

To fully evaluate the risk of the CPRR alternatives and consistent with current state-of-practice in PRA, accidents sequence were delineated by developing a set of core-damage event trees (CDETs) and a set of accident progression event trees (APETs). The CDET logic explicitly models the SBO mitigation strategies reflected in the industry's FLEX approach, and is a substantial expansion of the modeling approach taken in SECY-12-0157.

Core-damage sequences were binned into plant damage states (PDSs) that were subsequently fed into the APETs, which explicitly model the various CPRR alternatives. The end states of the APETs were binned into release categories (RCs) in order to match them to the consequence

assessments (MACCS calculations). Figure 4-1, "Risk Quantification Process," illustrates the overall risk evaluation quantification process.

Figure 4-1: Risk Quantification Process



A review of the licensee approaches to implementing the SBO mitigation (i.e., FLEX) strategies shows a wide variation of implementation details. To make the risk evaluation tractable, the CDETs and APETs were developed for a generic BWR plant having a RCIC system and a Mark I containment using the following assumptions and ground rules:

- 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 stage (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.

- 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:

    - No is credit taken for supplying the RCIC pump from the condensate storage tank (assumed to be non-seismically qualified).

    - The CDETs credit local manual operation of the RCIC pump (termed "blackstart and blackrun") if dc power fails.

    - No credit is 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.

    - There is no need to vent the containment during the first phase.

    - 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.

    - The CDETs credit local manual operation of SRVs if dc power fails.

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

    - Core cooling is maintained by operation of the RCIC pump. Makeup to the suppression pool is provided by the portable FLEX pump.

    - Except for regulatory basis sub-alternatives 4Biv and 4Civ, the operators will initiate anticipatory containment venting at 15 psig in order to minimize heatup of the suppression pool and prolong RCIC operation. (Regulatory basis sub-alternatives 4Biv and 4Civ reflect a passive drywell-first venting strategy.)

    - 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.

    - The CDETs credit local manual operation of the containment vent valves if dc power fails.

    - There is no need to provide RCIC pump room ventilation.

    - A portable generator must be aligned to provide dc power within 4 hours (the assumed battery depletion time).

- 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.

- 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.

- In the APETs, containment over-pressurization failure is prevented by opening the containment vent valves:

  - 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).

  - The CDETs credit local manual operation of the containment vent valves if dc power fails.

  - MELCOR calculations indicate that the containment must be vented before vessel breach because of the build-up of non-condensable gases generated during fuel-clad oxidation.

  - 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, "Containment design," and consistent with the Commission's direction in the SRM to SECY-12-0157, Option 2, to assume the installation of severe accident capable hardened venting system.

- In the APETs, post-core-damage water injection into the RPV will prevent vessel breach if it is initiated prior to core relocation and the RPV is depressurized below the shutoff head of the portable FLEX pump using the SRVs. In contrast, the regulatory basis alternatives involving post-core-damage DW injection cannot prevent vessel breach.

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). As a result, the CDETs and APETs are too voluminous to present in graphical format. Each CDET has 280 core-damage sequences that were binned into 139 PDSs. Each APET contains 72 to 84 release sequences (depending on the regulatory basis alternative being modeled) that were mapped into 24 release categories. Tables 4-5, "Plant Damage State Naming Scheme," and 4-6, "Release Category Naming Scheme," present the PDS and RC naming schemes, respectively.

Table 4-5: Plant Damage State Naming Scheme

Attribute	Possible Values
RCIC Failure Time	E: Early (0-4hr) M: Mid-term (4-16hr) L: Long-term (at 16hr)
RPV Pressure	HP: High Pressure (SRV cycling) MP: Medium Pressure (200-400 psig) LP: Low Pressure (below portable FLEX pump shutoff head)
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
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)
FLEX Equipment Status	OK: Portable FLEX pump is working F: FLEX equipment hardware is failed H: Operator fails to align FLEX equipment prior to core damage XX: Indeterminate (FLEX equipment status not asked in the core-damage sequence)

Table 4-6: Release Category Naming Scheme

Attribute	Possible Values
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
Containment Vent Status	WW: Wetwell vent is open DW: Drywell vent is open OP: Containment over-pressurization failure
Core Debris Location	IVR: In-vessel retention EVR: Ex-vessel retention LMT: Liner melt-through

4.2.3 Human Reliability Analysis

The SBO mitigation strategies represented in the CDETs and the CPRR alternatives 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 has proven to be challenging. The CPRR alternatives are conceptual designs; accordingly, they are not currently incorporated into the EPG/SAGs or into licensee training programs. The staff has gained some insight into how the alternatives would be implemented through interactions with external stakeholders during the development of the risk evaluation. However, these interactions do not provide an adequate technical basis for completing a detailed human reliability analysis (HRA) of the CPRR alternatives. Currently, the staff does not

have an HRA method that is capable of providing detailed HEP estimates of post-core-damage operator actions at the conceptual design stage. Specifically;

- The 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.

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

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

- In general, there is less frequent training on EPG/SAGs as compared to the EOPs.

- Information on some parameters may not be available to the operators or may be ambiguous for decisionmaking.

- The nature of the teamwork among the licensee staff responding to beyond-design-basis external events and severe accidents would be challenging as compared to responding to an off-normal situation.

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

- Site-wide events that involve multiple radiological sources would be challenging for plant staff.

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

As directed by SRM-M061020, "Meeting with Advisory Committee on Reactor Safeguards" (ADAMS Accession No. ML063120582), SRM-M090204B, "Staff Requirements - Briefing on Risk-Informed, Performance-Based Regulation" (ADAMS Accession No. ML090490812), and the SRM to SECY-11-0172, "Response to Staff Requirements Memorandum COMGEA-11-0001, 'Utilization of Expert Judgment in Regulatory Decision Making'" (ADAMS Accession No. ML120380251) the staff has been developing the Integrated Decision-tree Human Event Analysis System (IDHEAS) which, when completed, will provide a methodology for estimating post-core-damage HEPs which should address the challenges identified above. However, because the IDHEAS method had not been finalized at the time of this evaluation, the staff was not able to use it.

The staff used a set of scoping HEP estimates in the CPRR risk evaluation, supplemented with various sensitivity analyses that are focused on understanding the importance of operator actions to the CPRR alternatives. These scoping HEPs are not intentionally conservative; rather, they represent high-level estimates developed from the information and methods available at the time when this risk evaluation was completed. Table 4-7, "Pre-Core-Damage Human Failure Events," lists the pre-core-damage human failure events (HFEs) that are incorporated in the CDETs, along with their scoping HEPs. Table 4-8, "Post-Core Damage

Human Failure Events," lists the post-core-damage HFEs that are incorporated into the APETs, along with their scoping HEPs.

Table 4-7: Pre-Core-Damage Human Failure Events

Identifier	Description	Location	HEP
HFE-CVENT	Operator fails to reclose the containment vents upon core damage.	Control Room	0.1
HFE-CVENT-NODC	Operator fails to reclose the containment vents upon core damage (no DC power).	Local	0.3
HFE-DEPRESS	Operator fails to depressurize to 200-400 psia.	Control Room	0.1
HFE-DEPRESS-NODC	Operator fails to depressurize to 200-400 psia (no DC power).	Local	0.3
HFE-DW	Operator fails to open the drywell vent.	Control Room	0.1 - manual 0.01 - passive
HFE-DW-NODC	Operator fails to open the drywell vent (no DC power).	Local	0.3 – manual 0.01 - passive
HFE-GEN	Operator fails to align portable generator.	Local	0.3
HFE-INVDP	Operator inadvertently depressurizes RPV below the pressure needed to operate RCIC.	Control Room	0.1
HFE-INVDP-NODC	Operator inadvertently depressurizes RPV below the pressure needed to operate RCIC (no DC power).	Local	0.3
HFE-MSCL	Operator fails to depressurize at minimum steam cooling level.	Control Room	0.1
HFE-MSCL-NODC	Operator fails to depressurize at minimum steam cooling level (no DC power).	Local	0.3
HFE-PINJ	Operator fails to depressurize for FLEX RPV injection.	Control Room	0.1
HFE-PINJ-NODC	Operator fails to depressurize for FLEX RPV injection (no DC power).	Local	0.3
HFE-RCIC-BLACKRUN	Operator fails to blackrun RCIC during Phase 2.	Local	0.3
HFE-RCIC-BLACKSTART	Operator fails to blackstart and blackrun RCIC during Phase 1.	Local	0.3
HFE-RPV	Operator fails to align FLEX pump for RPV injection.	Local	0.3
HFE-SP	Operator fails to align FLEX pump for suppression pool makeup.	Local	0.3
HFE-WW	Operator fails to open the wetwell vent.	Control Room	0.1 - manual 0.01 - passive
HFE-WW-NODC	Operator fails to open the wetwell vent (no DC power).	Local	0.3 - manual 0.01 - passive

Table 4-8: Post-Core Damage Human Failure Events

Identifier	Description	APET	PDS Attribute and HEP*
DCRR	Failure to align portable generator after core damage.	All	DC status: OK 0 ST 1 LT 0.3 XX 0.3
RPVINJCRH	Failure to align portable FLEX pump for post-core-damage RPV injection prior to core relocation.	1	FLEX Pump Status: OK 0 F 0 H 0 XX 0
		2	FLEX Pump Status: OK 0 F 0 H 0.3 XX 0.3
PRPVINJCR	Failure to depressurize RPV to allow post-core-damage RPV injection prior to core relocation.	1,2	RPV Pressure: HP 0.1 MP 0.1 LP 0
PRPVINJCRnodc	Failure to depressurize RPV to allow post-core-damage RPV injection prior to core relocation (no DC power).	1,2	RPV Pressure: HP 0.3 ML 0.3 LP 0
RPVINJLMTH	Failure to align portable FLEX pump for post-core-damage RPV injection prior to liner melt-through.	1	FLEX Pump Status: OK 0 F 0 H 0 XX 0
		2	FLEX Pump Status: OK 0 F 0 H 0.3 XX 0.3
DWINJH	Failure to align portable FLEX pump for post-core-damage DW injection prior to liner melt-through.	3,4,5,6	FLEX Pump Status: OK 0 F 0 H 0.3 XX 0.3
WWPCD	Failure to open the WW vent after core damage.	1,2,3,6	Containment Vent Status: WW 0 DW 1 IS 0.1
		4,5	Containment Vent Status: WW 0 DW n/a IS 0.1

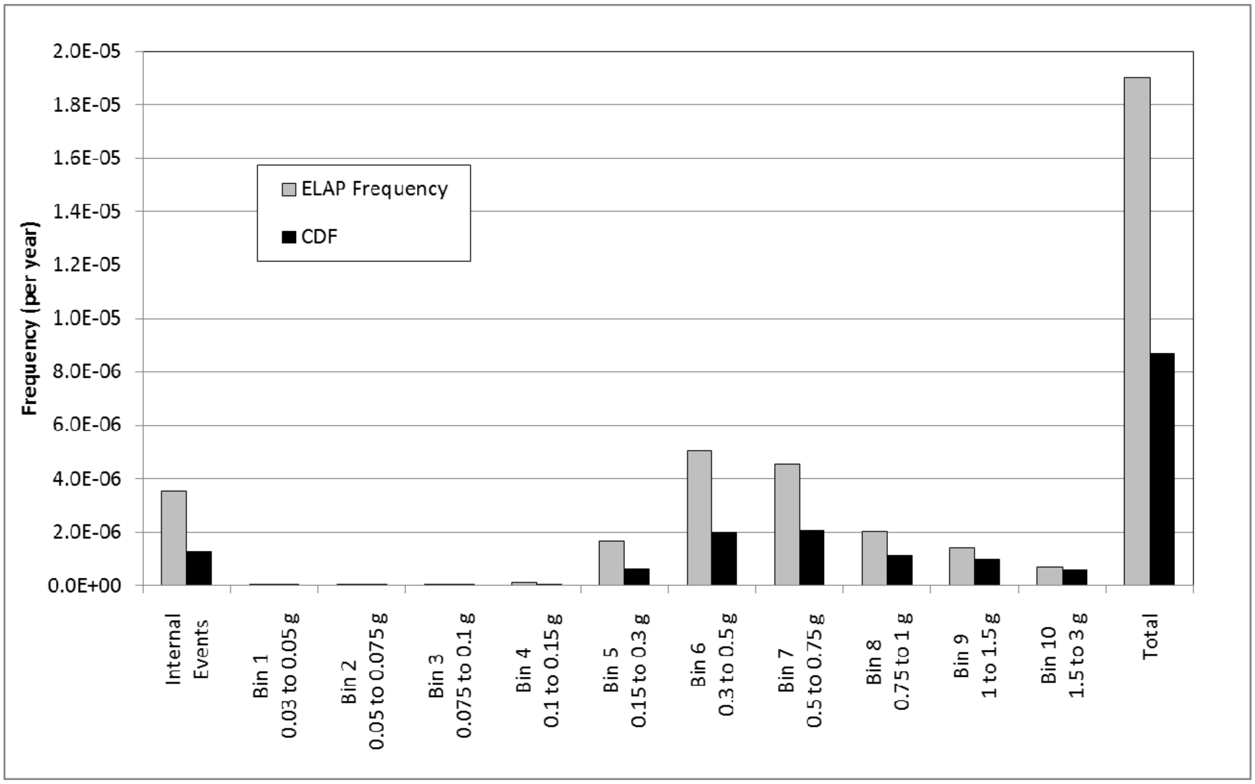
Identifier	Description	APET	PDS Attribute and HEP*
WWPCDnodc	Failure to open the WW vent after core damage (no DC power).	1,2,3,6	Containment Vent Status: WW 0 DW 1 IS 0.3
		4,5	Containment Vent Status WW 0 DW n/a IS 0.3
DWPCD	Failure to open the DW vent after core damage.	1,2,3,6	Containment Vent Status: WW n/a DW 1 IS 0.1
		4,5	Containment Vent Status: WW 1 DW 0 IS 0.01
DWPCDnodc	Failure to open the DW vent after core damage (no DC power).	1,2,3,6	Containment Vent Status: WW n/a DW 1 IS 0.3
		4,5	Containment Vent Status: WW 1 DW 0 IS 0.01
PDWINJCPCD	Failure to depressurize RPV after core damage (during DW injection).	3,4,5,6	RPV Pressure: HP 0.1 MP 0.1 LP 0
PDWINJCPCDnodc	Failure to depressurize RPV after core damage (during DW injection; no DC power).	3,4,5,6	RPV Pressure: HP 0.3 ML 0.3 LP 0

***** Note: See Table 4-5 for attribute naming scheme.

4.2.4 Summary of Technical Approach

The core-damage frequency (CDF) due to ELAPs is calculated to be 8.9×10^{-6}/ry, which is 2 times lower than the value of 1.6×10^{-5}/ry that was estimated for SECY-12-0157. The CDF was calculated by averaging together the CDF for each BWR plant that has a Mark I containment and a RCIC system. As shown in Figure 4-2, “Contributions to ELAP Frequency and Core-Damage Frequency,” the internal event ELAPs and seismic ELAPs caused by earthquakes with peak ground accelerations ranges from 0.3 to 0.7g are notable contributors to the CDF. Figure 4-2 also indicates that the conditional core-damage probability (CCDP) given the occurrence of an ELAP is about 47 percent (i.e., the SBO mitigation strategies reduce the CDF by about 53 percent).

Figure 4-2: Contributions to ELAP Frequency and Core-Damage Frequency



* Note: The contributions to ELAP frequency and CDF were developed from the probabilistic convolution of the seismic hazard curve with the ELAP and core-damage PRA models. The results are similar to results from previous PRA studies (e.g., NUREG-1150 and the IPEEE submittals).

Table 4-9, "Significant Contributors to Core-Damage Frequency," and Figure 4-3, "Identification of Significant Contributors to Core-Damage Frequency," identify the significant contributors to CDF, as determined by importance analysis. Consistent with Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities" (ADAMS Accession No. ML090410014), an event is a significant contributor if it has a risk achievement worth (RAW[23]) greater then 2.0, or a Fussell-Vesely (FV[24]) importance measure greater than 0.005. 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 equipment (failure to start and failure to run), and random failures of the RCIC pump (failure to start).

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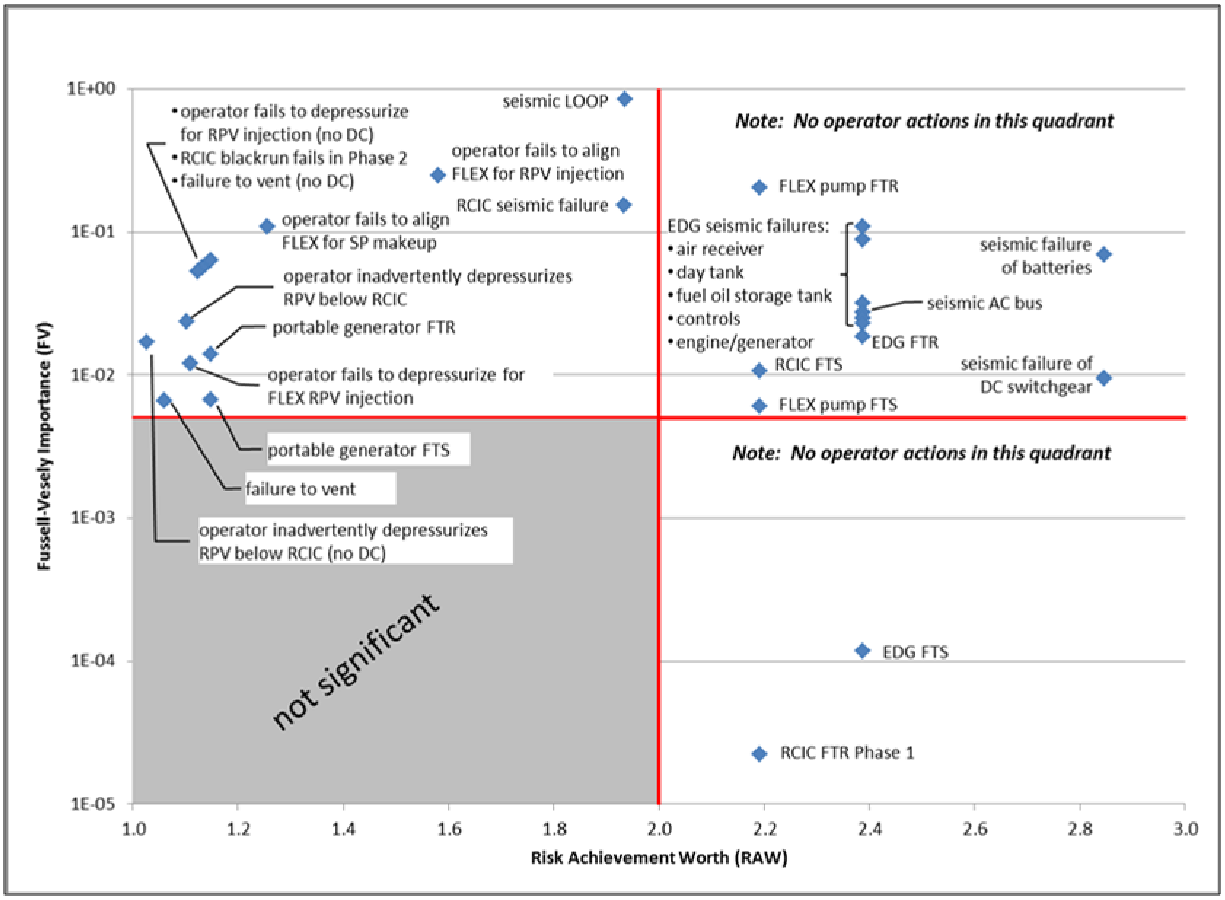
23 Risk achievement worth (RAW) importance measure: For a specified basic event, risk achievement worth reflects the increase in a selected figure of merit when a structure, system, or component (SSC) is assumed to be unable to perform its function due to testing, maintenance, or failure. It is the ratio of the figure of merit, evaluated with the SSC's basic event probability set to one, to the base case figure of merit.

24 Fussell-Vesely (FV) importance measure: For a specified basic event, the fractional contribution to the total of a selected figure of merit from all accident sequences containing that basic event.

Table 4-9: Significant Contributors to Core-Damage Frequency

Basic Event	Description	RAW	FV
LOOP-S	Seismic LOOP	1.9	0.854
HFE-RPV	Operator fails to align FLEX pump for RPV injection	1.6	0.249
FLEXP-FTR	FLEX equipment fails to run	2.2	0.204
RCIC-S	RCIC seismic failure	1.9	0.155
EDG-S-SAR	Seismic failure of EDG starting air receiver	2.4	0.110
HFE-SP	Operator fails to align FLEX pump for SP makeup	1.3	0.109
EDG-S-DT	Seismic failure of EDG day tank	2.4	0.088
BATTERY-S	Seismic failure of batteries	2.8	0.069
HFE-GEN	Operator fails to align portable generator	1.1	0.063
HFE-PINJ-NODC	Operator fails to depressurize for FLEX RPV injection (no DC power)	1.1	0.056
HFE-RCIC-BLACKRUN	Operator fails to blackrun RCIC during Phase 2	1.1	0.056
HFE-WW-NODC	Operator fails to open the wetwell vent (no DC power)	1.1	0.053
HFE-DW-NODC	Operator fails to open the drywell vent (no DC power)	1.1	0.053
EDG-S-FOST	Seismic failure of EDG fuel oil storage tank	2.4	0.032
ACSWGR-S	Seismic failure of AC switchgear	2.4	0.028
EDG-S-CTRL	Seismic failure of EDG controls	2.4	0.025
HFE-INVDP	Operator inadvertently depressurizes RPV below RCIC	1.1	0.024
EDG-S-EG	Seismic failure of EDG engine or generator	2.4	0.023
EDG-FTR	EDGs fail to run	2.4	0.019
HFE-INVDP-NODC	Operator inadvertently depressurizes RPV below RCIC (no DC power)	1.0	0.017
GEN-FTR	Portable generator fails to run	1.1	0.014
HFE-PINJ	Operator fails to depressurize for FLEX RPV injection	1.1	0.012
RCIC-FTS	RCIC fails to start	2.2	0.011
DCSWGR-S	Seismic failure of DC switchgear	2.8	0.010
GEN-FTS	Portable generator fails to start	1.1	0.007
HFE-WW	Operator fails to open the wetwell vent	1.1	0.007
HFE-DW	Operator fails to open the drywell vent	1.1	0.007
FLEXP-FTS	FLEX pump fails to start	2.2	0.006
SORV2-DP	Stuck-open SRV in Phase 2 given Phase 1 depressurization and cooldown	1.0	0.006
SORV1-NODP	Stuck-open SRV given no Phase 1 depressurization and cooldown	1.0	0.005
EDG-FTS	EDGs fail to start	2.4	0.000
RCIC-1	RCIC fails to run in Phase 1	2.2	0.000

Figure 4-3: Identification of Significant Contributors to Core-Damage Frequency



To gain further perspective on the importance of operator actions and the adequacy of the scoping HEPs used in the risk evaluation, an *en masse* sensitivity analysis was performed. The results of this analysis, which are depicted as a "heat map" in Figure 4-4, "En Masse Sensitivity to Human Error Probability," indicate that the CDF ranges from a low of 5.1×10^{-6}/ry (assuming no operator errors) to a high of 1.9×10^{-5}/ry (giving no credit to the plant operators). Two conclusions may be drawn from these results:

- About 57 percent (5.1×10^{-6}/ry / 8.9×10^{-6}/ry) of the total CDF is due to accident sequences that only involve equipment failure. The remaining 43 percent is due to accident sequences that involve combinations of equipment failures and operator errors.

- Operator actions to implement mitigation strategies are important contributors to CDF. The scoping human error probabilities could be refined to provide additional risk insights. However, additional effort to refine the scoping human error probabilities may not be warranted because the change in CDF resulting from the use of refined human error probabilities may be masked by the parametric uncertainties of the data used in the analysis.

Figure 4-4: En Masse Sensitivity to Human Error Probability*

	Ex-Control-Room HEP													
In-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
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

* Note: Cells that are shaded in green have lower core-damage frequencies, which reflects increased credit for operator actions (i.e., lower human error probabilities). Cells that are shaded in red have higher core-damage frequencies, which reflects reduced credit for operator actions (i.e., higher human error probabilities). As less credit is given for operator actions (i.e., human error probabilities approach 1.0), the core-damage frequency approaches the ELAP frequency since FLEX implementation requires operator actions.

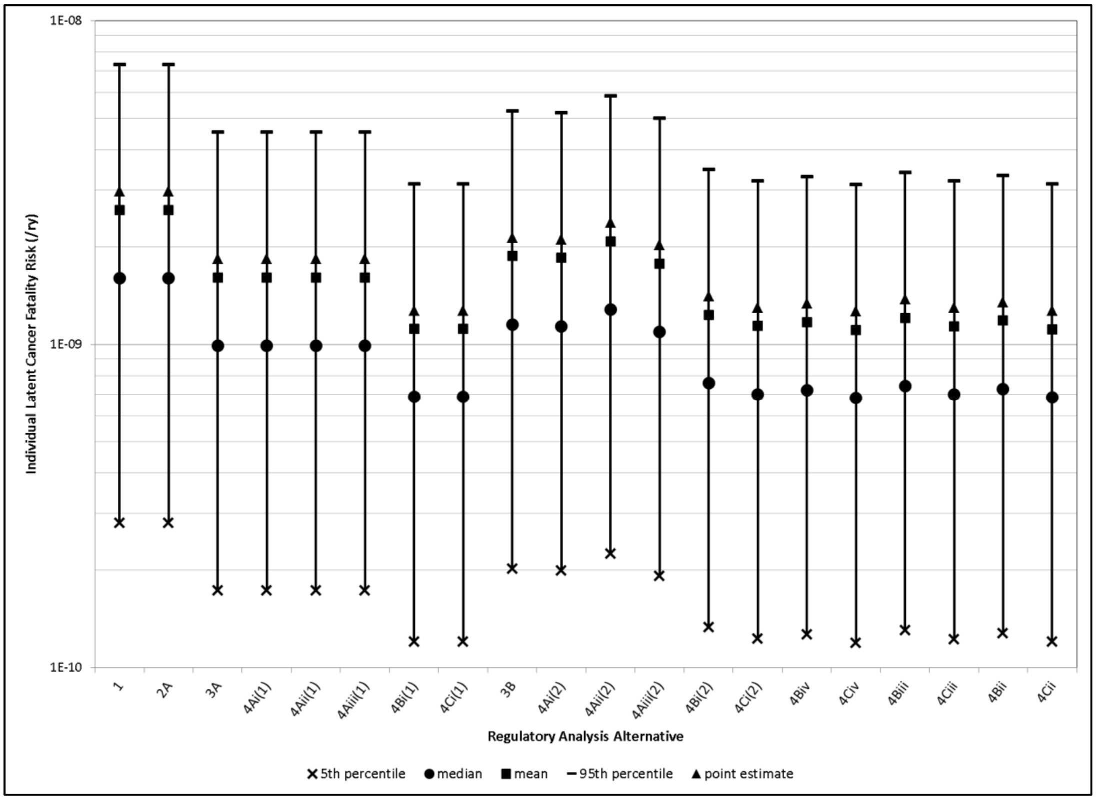
A parametric Monte Carlo uncertainty analysis was performed to gain additional perspective into the point-estimate risk evaluation results. Parametric uncertainties were considered for all inputs, specifically:

- seismic hazard curves
- seismic fragility curves
- random equipment failures

- operator actions
- consequences

The results of the parametric uncertainty analysis, displayed in Figure 4-5, “Parametric Uncertainty Analysis Results,” show greater than an order of magnitude uncertainty in the risk. By far, the major contribution to parametric uncertainty is seismic hazard but the results show that the ILCF risk is almost two orders of magnitude below the QHO of $2x10^{-6}$/ry.

Figure 4-5: Parametric Uncertainty Analysis Results



4.3 MELCOR Analysis

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 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 bases 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 are used in MACCS to calculate offsite consequences.

The selection of accident sequences covered by the MELCOR calculation matrix was informed by the staff’s comprehensive PRA. The PRA treatment, as discussed above, developed a CDET and an APET, and binned a rather large number of possible end states to a manageable

number of categories with similar outcomes. These categories were ranked in descending order of frequency. The MELCOR calculation matrix covers almost all of the possible end states in the APET.

Another important consideration factored into the development of the MELCOR calculation matrix is tied to the alternatives considered in the regulatory basis. Specifically, these alternatives fall into the three following categories:

Category 1—Over-pressure 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/SAWM) is considered in this category. As such, liner melt-through in the Mark I containment is not prevented nor is the consequent uncontrolled release of radioactivity to the environment.

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 4-10, “Alternative Numbering and Actions” (“alternative” designation refers to regulatory basis alternatives being considered as shown in Table 4-3).

Table 4-10: Alternative Numbering and Actions

Alternative Number	Alternative Action
Alternative 1	Order EA-13-109, EPG/SAG Rev 3,[25] 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 calculation matrix for a representative BWR with a Mark I containment is shown in Table 4-11, “MELCOR Calculation Matrix for a Representative BWR with Mark I Containment.” The calculation matrix for a representative BWR with a Mark II containment is shown in Table 4-12, “MELCOR Calculation Matrix for a Representative BWR with Mark II Containment.” The gray boxes in the tables signify the major deviations from the assumed initial and boundary conditions as part of sensitivity calculations.

[25] 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.

Table 4-11: MELCOR 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 4-12: MELCOR 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											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	60
	1p1	72	16	SP	230	N	15	-	-	Y	WW	45
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	60
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	60
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	30
3A	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
3A	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
3B	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
3A	42	4	4	SP	230	N	15	RPV	500	N	WW/DW	60
3B	44	4	4	SP	230	N	15	DW	500	N	WW/DW	60
3B	45	-	16	SP	230	-	-	DW	500	Y	DW	60
3B	49	-	0	-	-	-	-	DW	500	Y	WW/DW	60
3B	51	-	16	SP	230	-	15	DW	500	Y	DW	15
3B	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										

4.3.1 MELCOR Input Models and Assumptions

The MELCOR input models for a representative BWR with a Mark I and II containment follow the modeling best practices used in the State-of-the-Art Reactor Consequence Analysis (SOARCA) study (NUREG-1935, NUREG/CR-7110, NUREG/CR-7008) and reflect current understanding in severe accident modeling with the capability for modeling full-power steady-state operating conditions. The models were informed by the events at the Fukushima Dai-ichi Nuclear Power Plant. The SOARCA input deck for the Peach Bottom Atomic Power Station in Delta, Pennsylvania, was used as the baseline model and some modifications were made for the present study. There are differences in design details between various BWR Mark I plants so the results may need to be appropriately qualified for specific designs.

The MELCOR input model for a representative BWR with a Mark II containment was based on a previously developed deck resembling the LaSalle County Generating Station in Illinois. Variants of this base case were used to model two other Mark II cavity configurations.

The MELCOR analysis assumptions are as follows:

- All MELCOR transients start with an ELAP.
- All transients are run for a duration of 72 hours.[26]
- Industry EPG/SAG Revision 3 is in place.
- FLEX equipment is in place for both pre- and post-core damage:
    - 500 gpm injection into RPV or drywell from external source at vessel breach
    - provision for SAWA or SAWM
- Initial buildup of water in the drywell from nominal leakage.
- Possible end states of accident progression include:
    - liner melt-through (LMT) – Mark I only
    - main steam line creep rupture (MSLCR)
    - drywell head flange leakage by over-pressure and over-temperature
- RCIC operation includes:
    - suction from SP (option for suction from CST/SP)
    - flow rate 600 gpm
    - RPV level control via throttling of RCIC
- RPV pressure control includes:
    - initial pressure control in 800-1000 psig band after 10 minutes
    - controlled depressurization after 1 hour
    - subsequent pressure control in 200-400 psig band for continued RCIC operation
- Containment venting includes:
    - anticipatory venting prior to core damage (15 psig)
        - not performed if RCIC already failed
    - upon entry into SAG, vent closes; reopens at PCPL (60 psig), option to reopen at PSP considered
    - transition from WW to DW venting at SP high water level (21’ above bottom of torus for Mark I and 50’ for Mark II)

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26 The assumptions regarding offsite support in this study are similar to those used in the SOARCA study and 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 (see Table 4-14, “Summary of Mark I Analysis Results”).

- vent cycling in (PCPL)/(PCPL-10/20) band; option with PSP considered
- vent sizing consistent with industry assumptions

4.3.2 MELCOR Results

MELCOR provides source term (i.e., quantity and timing of fission product releases) in addition to various thermal-hydraulic outputs. The output can be tracked by control volume regions (representing physical and geometric attributes of a plant such as core, RPV, lower plenum, drywell, etc.), and both spatial and temporal evolutions of various parameters can be generated. The output parameters of interest for this study included:

- drywell pressure—determines likelihood of containment failure by over-pressure.
- drywell structure temperature—determines likelihood of failure of various components.
- wetwell water level—determines if wetwell is flooded.
- wetwell (suppression pool) water temperature—determines the effectiveness of pool scrubbing.
- in-vessel and ex-vessel hydrogen generation and distribution—important contributor to containment over-pressurization and hydrogen combustion risk; also, hydrogen and other non-condensable gas concentration in control volumes determines the combustion potential in various physical volumes (e.g., vent path, reactor building, spent fuel pool).
- axial and radial erosion of cavity—determines likelihood of containment breach and consequent environmental release.
- Cesium release—important contributor to source term for long-term phase of offsite consequences.
- Iodine release—important contributor to source term for early phase of offsite consequences.

The above parameters are tied to certain performance measures such as component integrity, effectiveness of mitigation measures, and confinement of radioactivity.

The radionuclide release model in MELCOR is of particular importance because the output is used for offsite consequence calculations. MELCOR categorizes radionuclides and other pertinent materials into elemental classes that exhibit similar chemistry. The modeling and treatment of radionuclides include: (1) release of radionuclides from intact fuel and from core debris; (2) transport and deposition of radionuclide vapors and aerosols through the reactor coolant system; (3) behavior of radionuclides and radioactive aerosols in the reactor containment; and (4) effects of engineered safety systems (excluding an external filter) on the amount of radioactive material that can be released from the reactor containment.

Sample MELCOR results are provided in Table 4-13, "Timing of Key Events in Hours for Selected Mark I Cases," and in Figures 4-6 through 4-14 for three examples of BWR Mark I scenarios representing different accident mitigation strategies. MELCOR case 1 is provided as an example of an over-pressure protection scenario in which the severe accident capable vent is in place, however, there is no water addition (i.e., alternatives 1 and 2). MELCOR case 9 and

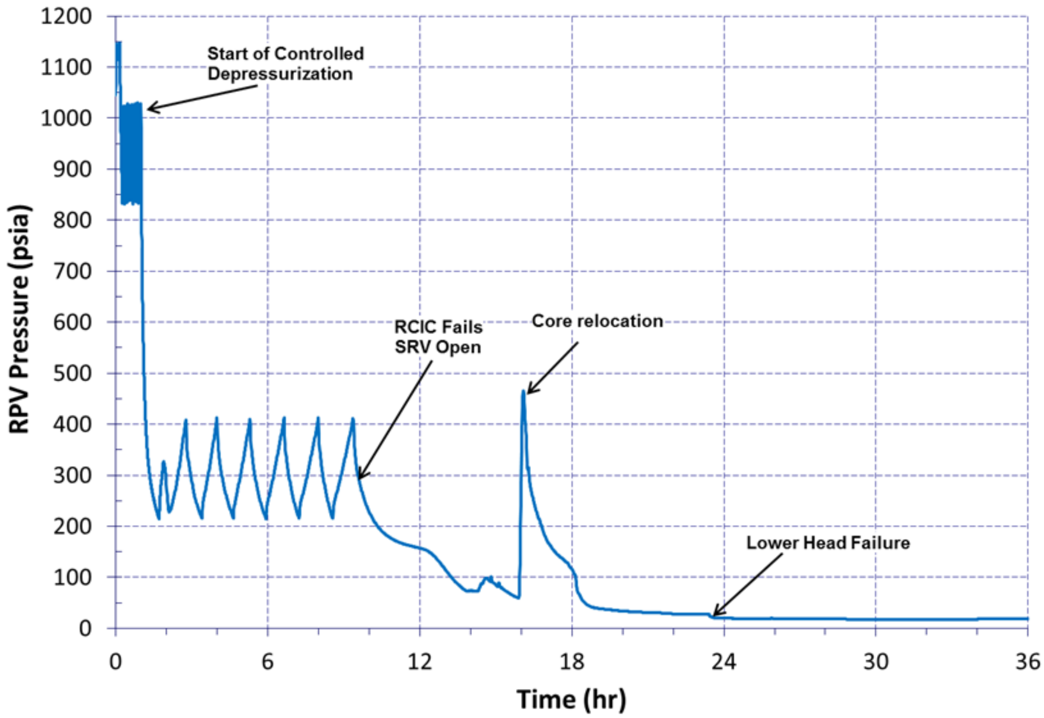
10 are provided as examples of SAWM and SAWA (i.e., alternative 3), respectively. These three cases are provided as examples to illustrate differences and similarities in accident progression and source terms but they are not intended to represent the “best estimate” of each accident mitigation strategy; many other variations were run in MELCOR which are also used to characterize the results.

Table 4-13: Timing of Key Events in Hours for Selected Mark I Cases

Key Event	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
Water level reaches TAF	12.4	11.9	11.9
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	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 (kg)	292,000	280,000	287,000
In-vessel hydrogen generated (kg)	1,195	1,032	1,232
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

Figure 4-6, “Mark I RPV Pressure History for Case 9 (SAWM),” shows the RPV pressure history for case 9 that characterizes the boundary conditions for pressure control. During the first 10 minutes, a single SRV keeps the pressure below the lowest SRV setpoint. At 10 minutes, the operators control the SRV by maintaining the pressure between 800 and 1000 psig. At one hour, the operators begin controlled depressurization of the RPV and maintain the pressure between 200 and 400 psig to allow RCIC to operate. RCIC fails at about 9.6 hours due to over-temperature in the suppression pool (230 degrees Fahrenheit). For case 9, the operators depressurize the RPV by opening the SRV. Water injection is not successful, however, and core damage results. The core damage progresses until core debris falls to the lower head. The water in the lower head quenches the debris until it evaporates. Then, the debris re-heats until the lower head fails at 23 hours.

Figure 4-6: Mark I RPV Pressure History for Case 9 (SAWM)



Figures 4-7 through 4-9 show the containment pressure and the integral mass flow through the WW and DW containment vents for cases 1, 10, and 9. For case 1, the absence of water injection leads to DW head leakage at 27 hours followed by liner melt-through at 31 hours. In case 10, the WW vent is isolated at high WW level (21 feet above the bottom of the torus) at 42 hours, and the DW vent is opened at 54 hours when the containment pressure reaches the PCPL set point. In case 10 the water addition rate continues to be 500 gpm. In case 9 the WW vent remains open, because the flow is reduced to maintain the suppression pool level below 21 feet.

Figure 4-7: Mark I Containment Pressure and Integral Vent Mass Flow Case 1 (no water)

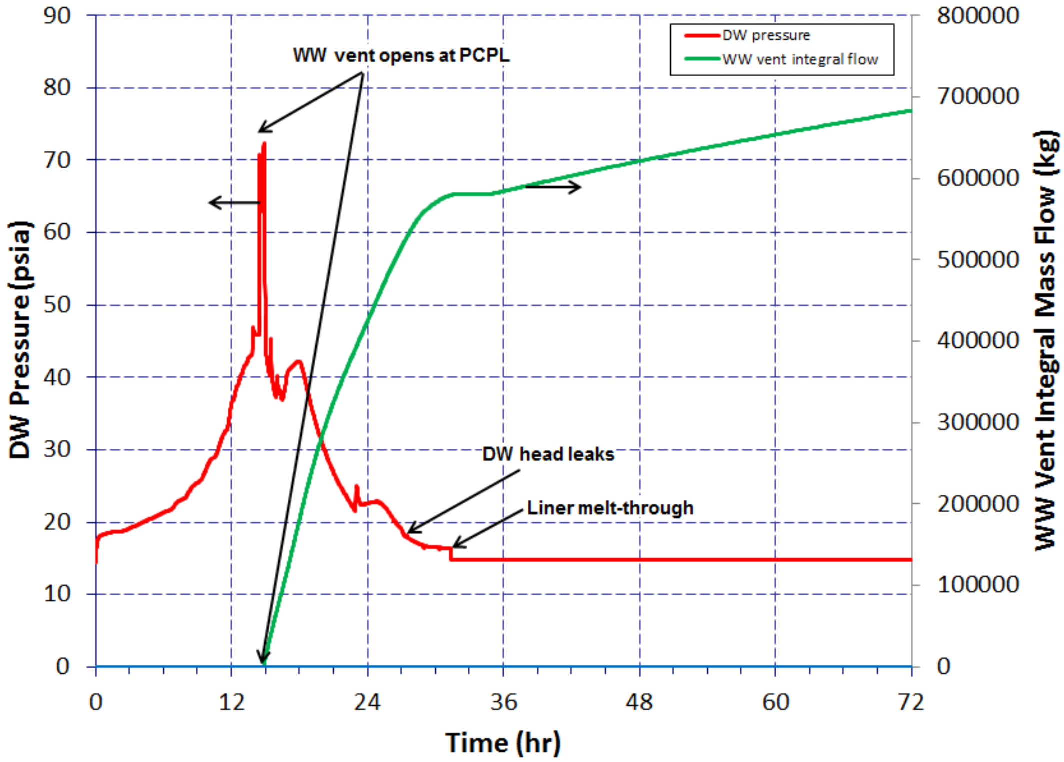


Figure 4-8: Mark I Containment Pressure and Integral Vent Mass Flow for Case 10 (SAWA)

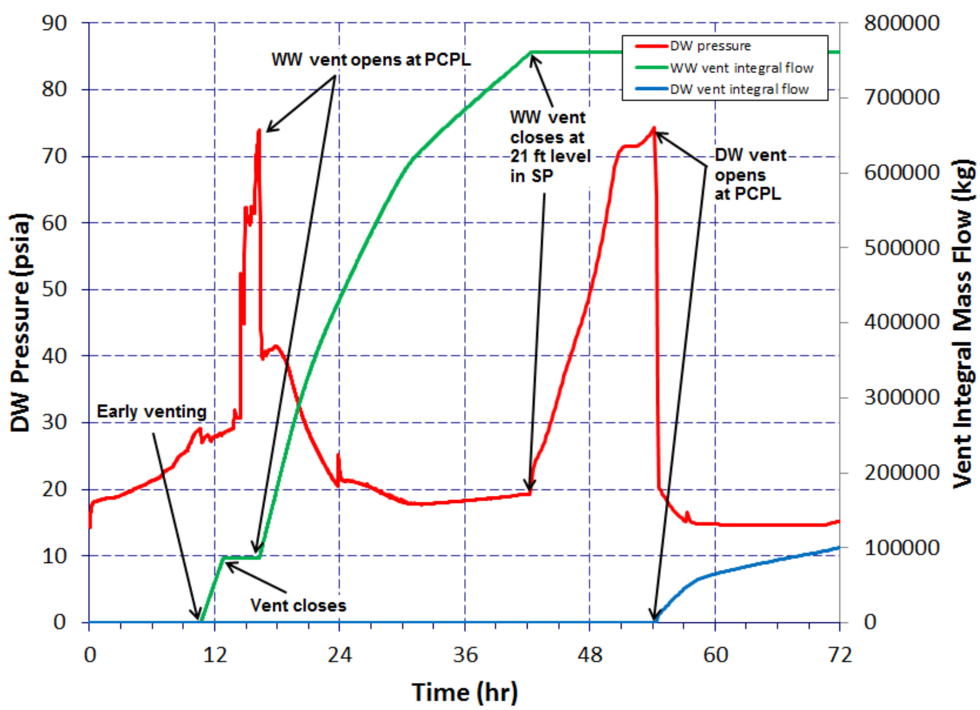
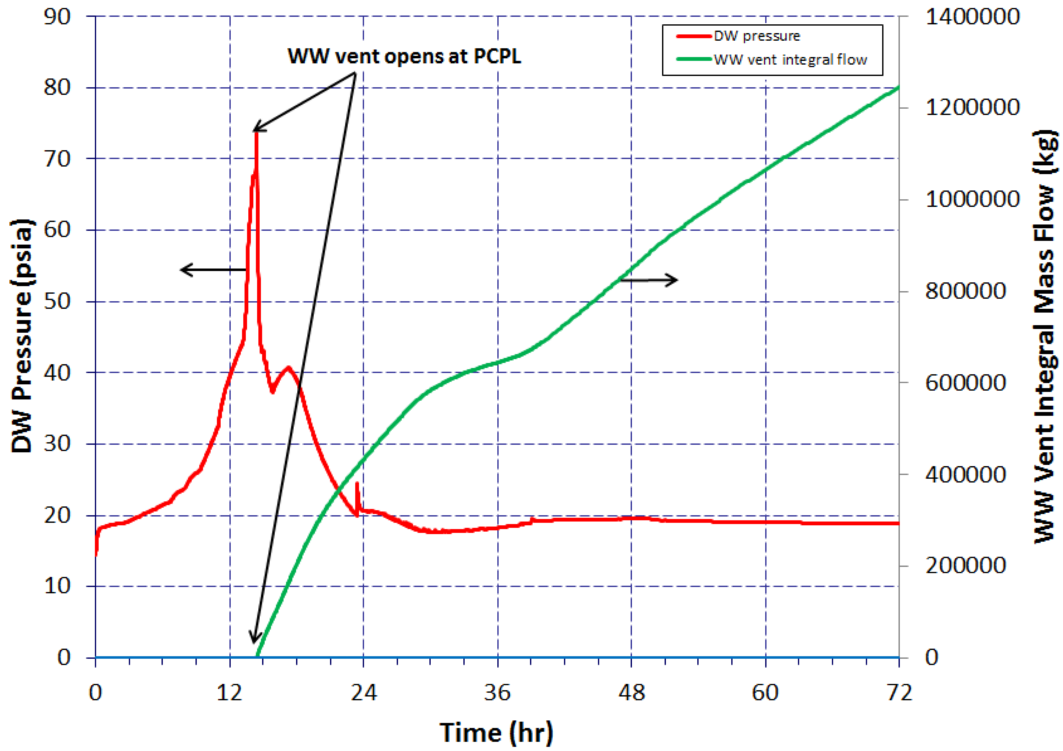


Figure 4-9: Mark I Containment Pressure and Integral Mass Flow for Case 9 (SAWM)



The containment gas temperatures are shown in Figures 4-10 and 4-11 for cases 1 and 9, and the containment water levels are shown in Figures 4-12 through 4-14. Without SAWA, the containment temperature keeps increasing after the core debris dries out the remaining water from the recirculation line leakage (total of 36 gpm). The injection of water at lower head failure for cases 10 and 9 keeps the containment temperature relatively low and prevents both DW head leakage and liner melt-through. The behavior is similar for cases 10 and 9. The sudden increase in the DW water level in case 10 is due to rapid depressurization of the containment and the backflow of water from the WW to the DW through the main vents.

Figure 4-10: Mark I Containment Gas Temperature for Case 1 (no water)

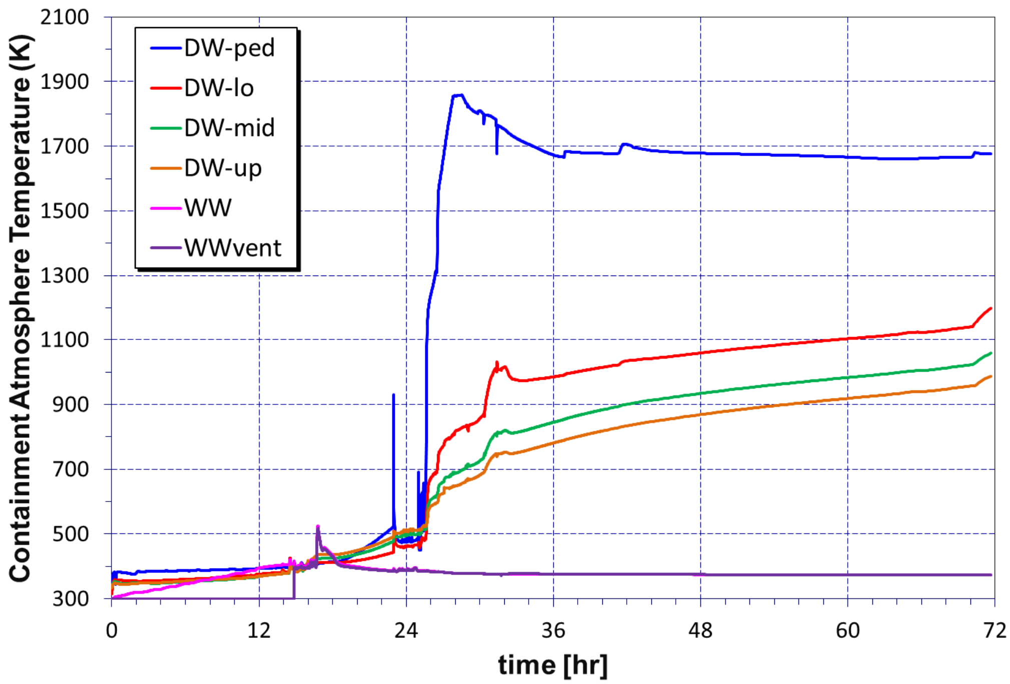


Figure 4-11: Mark I Containment Gas Temperature for Case 9 (SAWM)

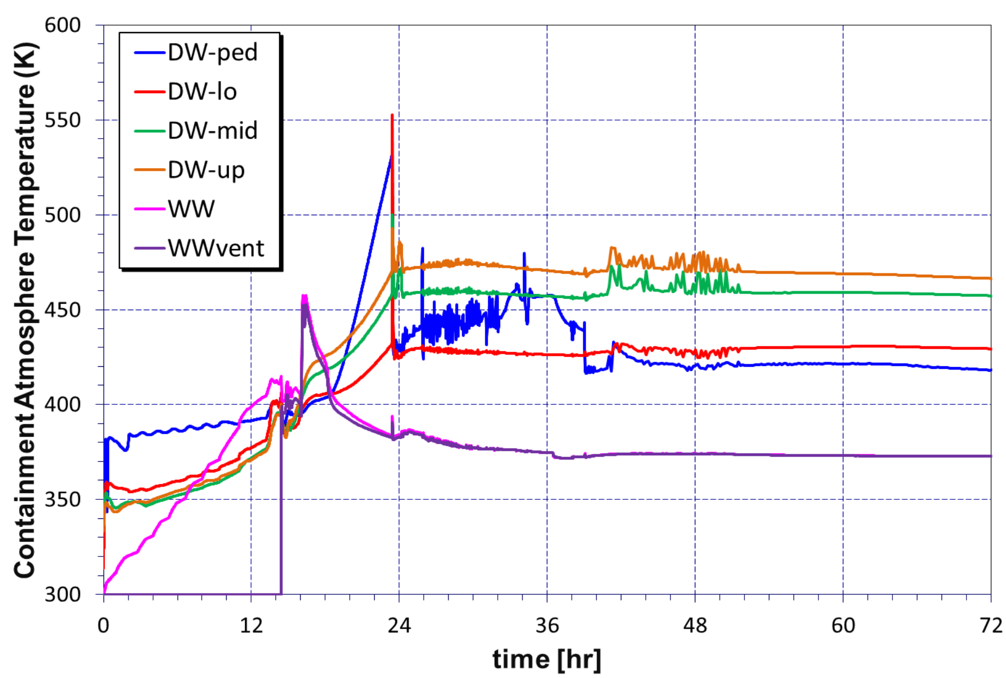


Figure 4-12: Mark I Containment Water Level for Case 1 (no water)

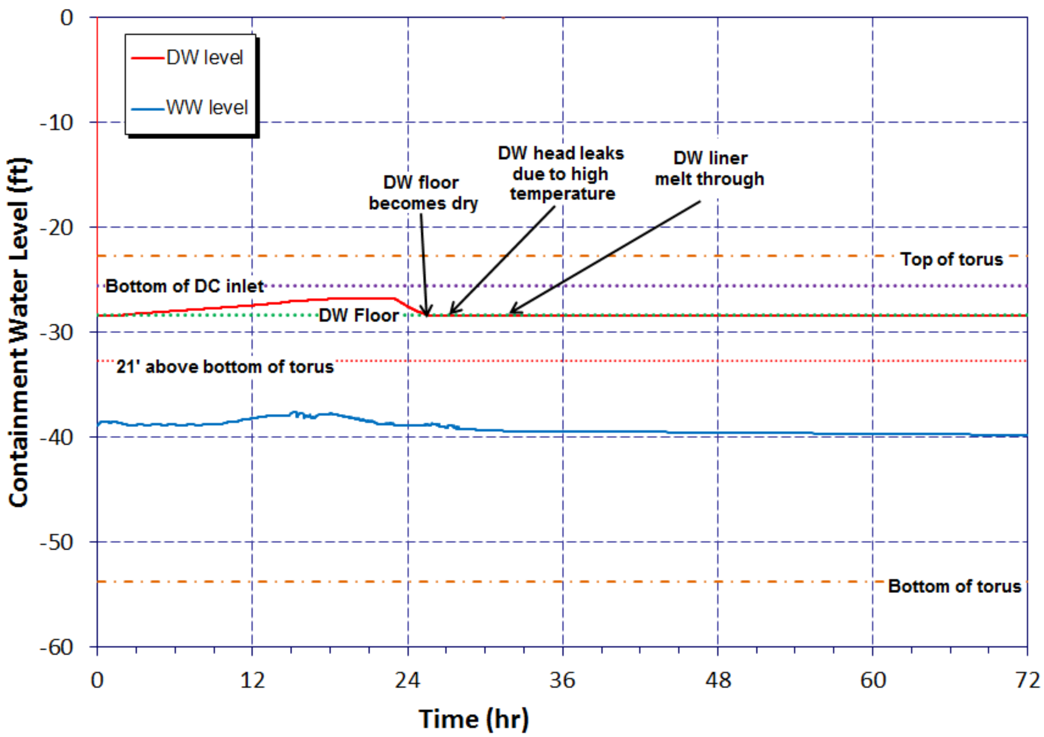


Figure 4-13: Mark I Containment Water Level for Case 10 (SAWA)

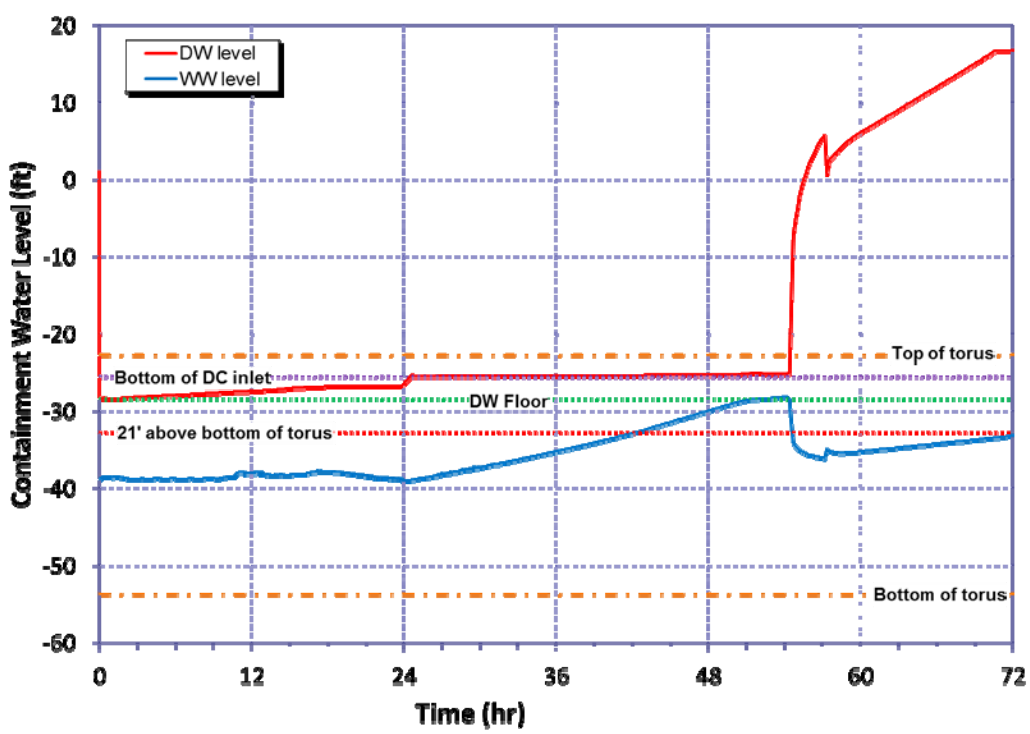
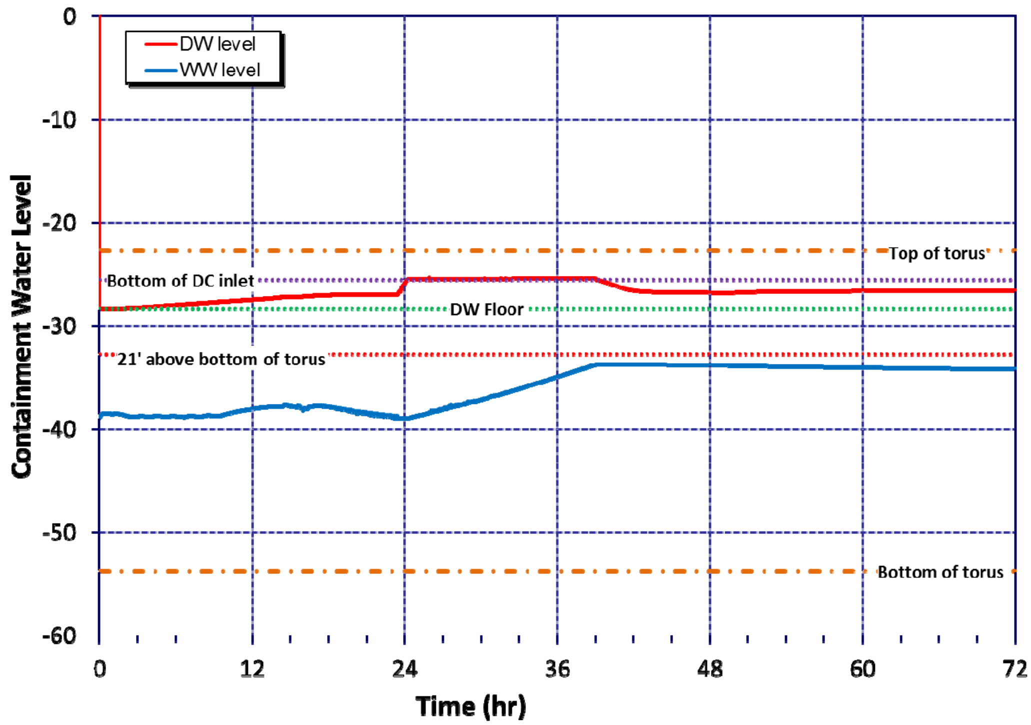


Figure 4-14: Mark I Containment Water Level for Case 9 (SAWM)



The cesium environmental release fractions for the various scenarios are given in Figure 4-15, "Mark I Cesium Environmental Release Fractions." A summary of accident progression results is provided in Table 4-14, "Summary of Mark I Analysis Results." In most cases with water injection, the cesium release fraction remains below 1 percent (i.e., a system decontamination factor of greater than 100) and the mode of containment venting and water injection does not greatly affect the cesium release. Changes in boundary conditions such as opening SRVs by the operators before core damage, the fractional open area of thermally seized SRV, early (pre-core damage) containment venting, and RCIC injection source (SP vs. CST) can affect the source term magnitude. These boundary conditions affect core degradation, thermodynamic conditions inside the RPV, and ultimately the distribution of fission products in the RPV and containment.

In general, cases without water injection or when water injection stops at high torus level show higher release fractions. However, the source term is not significantly affected by the containment failure mechanism (liner melt-through vs. DW head leakage). In these cases, no water has been injected until vessel breach. In all cases the early release is characterized by a single concentrated "puff" at the time of containment venting (see Table 4-13, "Timing of Key Events in Hours for Selected Mark I Cases") but the injection of water after the lower head failure stabilizes the environmental release (see, for example Figure 4-16, "Effect of Water Addition on Cesium Release for Selected Mark I Cases").

Figure 4-15: Mark I Cesium Environmental Release Fractions

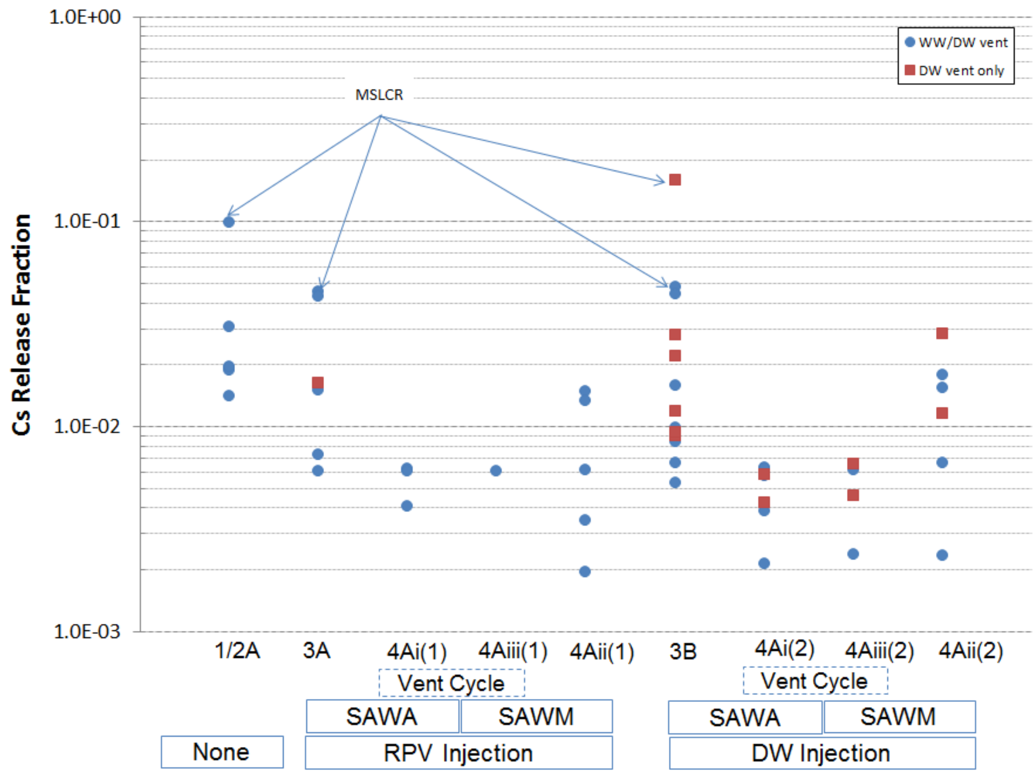


Table 4-14: Summary of Mark I Analysis Results

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

Figures 4-16, "Effect of Water Addition on Cesium Release for Selected Mark I Cases," and 4-17, "Mark I CsI Particle Size Distribution for Case 9 (SAWA)," show 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). It is important to note that the release of radionuclides that immediately follows containment venting is characterized by a single concentrated "puff" release well before vessel breach and the start of water injection. 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. Moreover, because the fission products released to the environment through the wetwell vent have passed

through the suppression pool first, only the smaller particles remain (see Figure 4-17 for case 9). Any scrubbing of aerosols in this size range is expected to be minimal.

Figure 4-16: Effect of Water Addition on Cesium Release for Selected Mark I Cases

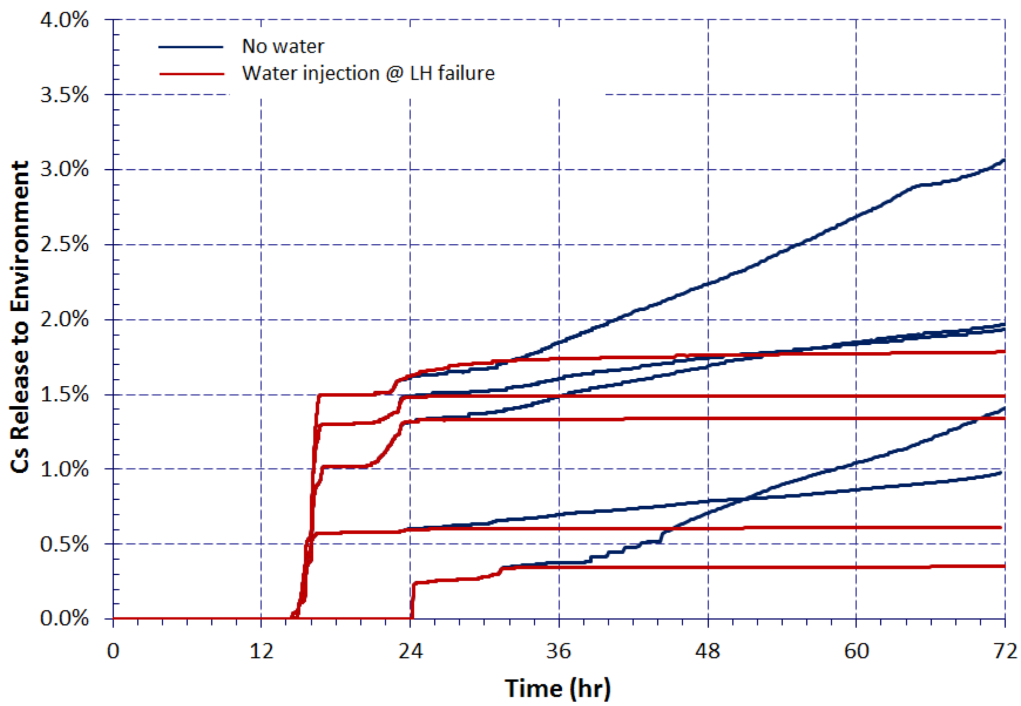


Figure 4-17: Mark I CsI Particle Size Distribution for Case 9 (SAWA)

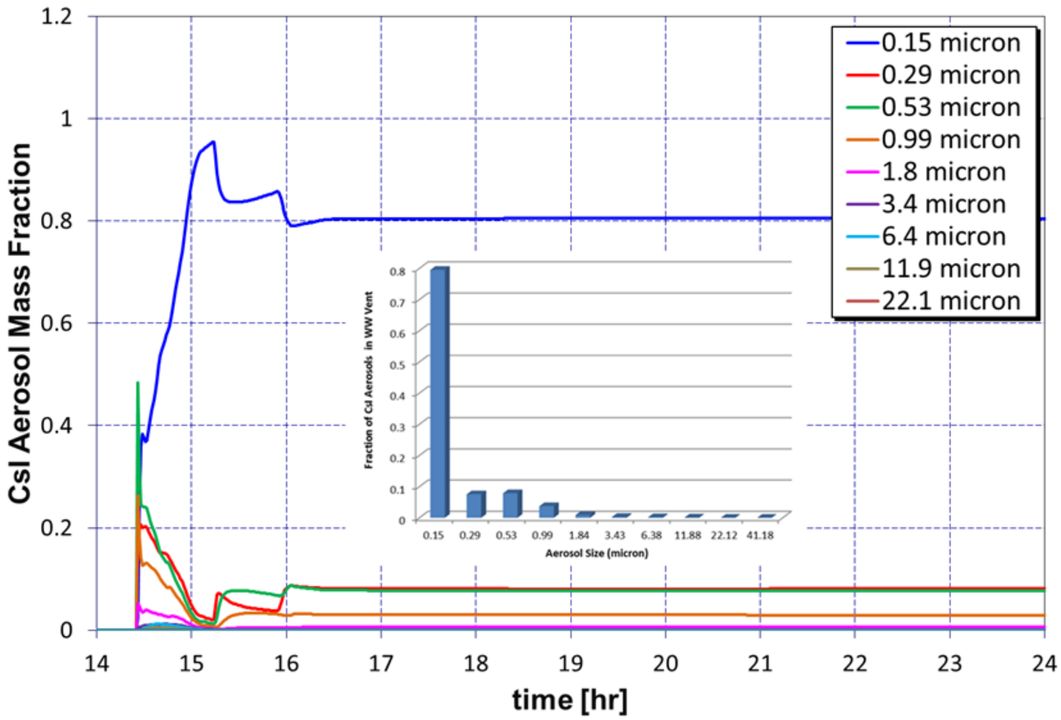
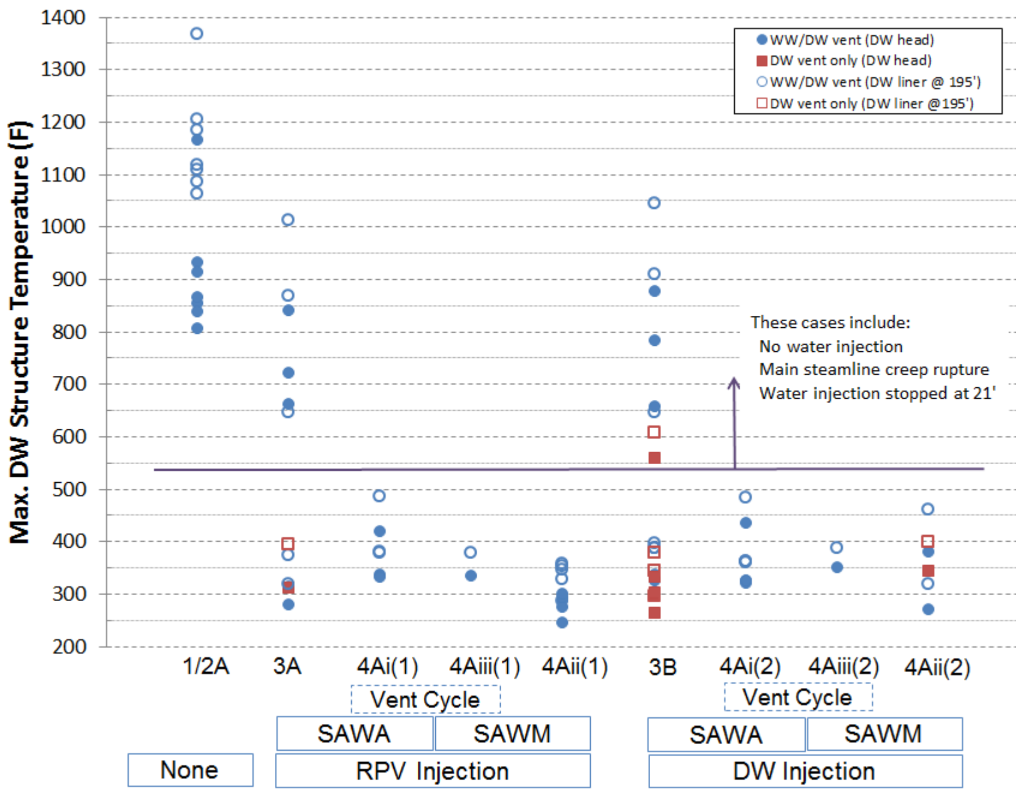


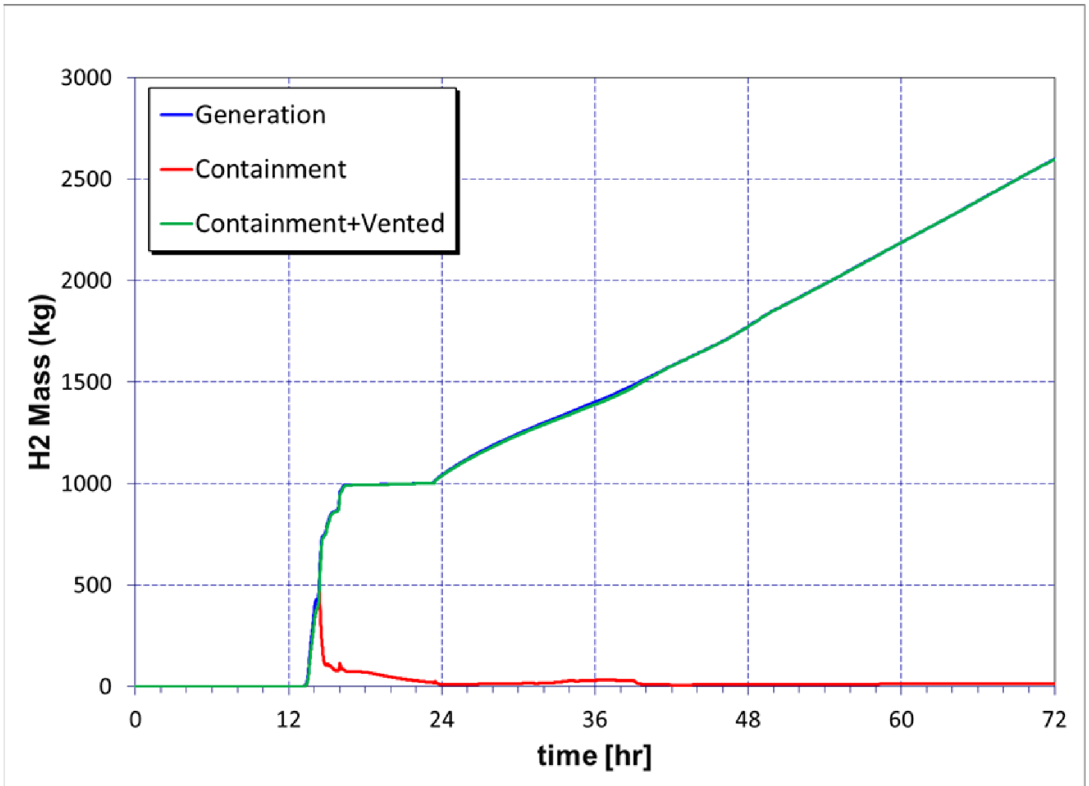
Figure 4-18, "Mark I Maximum Drywell Structure Temperature," shows that for all the cases with sustained water injection into the drywell or the RPV (that eventually accumulates on the drywell floor), the maximum structure temperatures at the drywell upper head or the drywell liner near the elevation of the drywell vent remains below 500 degrees Fahrenheit (260 degrees Celsius). The cases without water injection in general experience the highest temperatures.

Figure 4-18: Mark I Maximum Drywell Structure Temperature



The behavior of hydrogen in the containment is shown in Figure 4-19, "Mark I Hydrogen Generation and Transport for Case 9 (SAWA)." 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 accident.

Figure 4-19: Mark I Hydrogen Generation and Transport for Case 9 (SAWA)



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 heatup. This variation is labeled case 9-IVR because the earlier water addition leads to in-vessel retention (IVR) of the core. The water level (see Figure 4-20, “Mark I RPV Water Level for Case 9-IVR”) 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 4-21, “Mark I Cesium Release Fraction From Fuel for Case 9-IVR,” indicates that about 30 percent 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 4-22, “Mark I Cesium Release Fraction to Environment for Case 9-IVR,” 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 two hours as shown in Figure 4-21). In all the base MELCOR calculations, it was assumed that water injection begins at the time of lower head failure and the initial venting puff release was predicted to occur much sooner. Therefore, the water injection timing is important in determining if there is any reduction in release.

Figure 4-20: Mark I RPV Water Level for Case 9-IVR

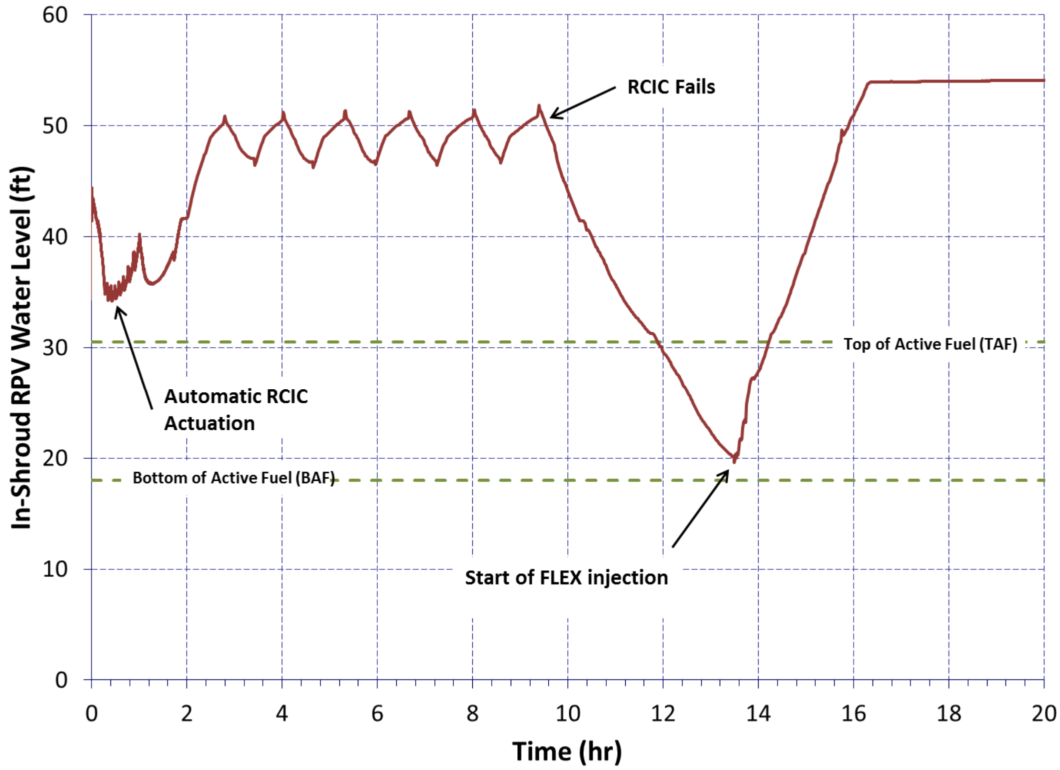


Figure 4-21: Mark I Cesium Release Fraction From Fuel for Case 9-IVR

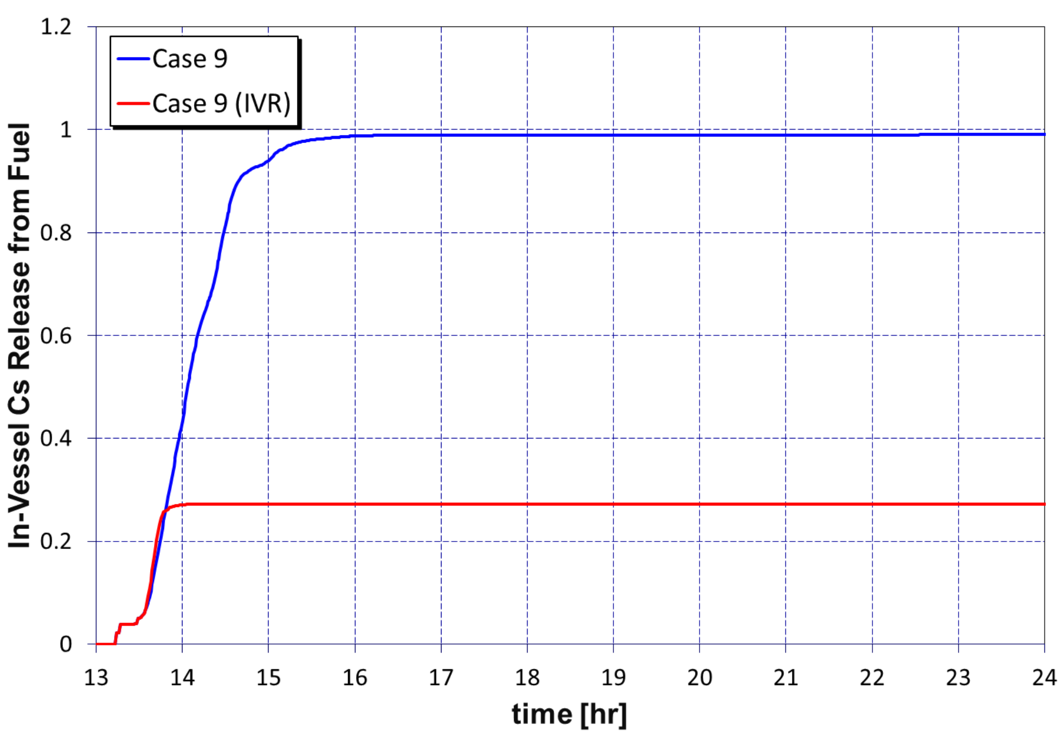
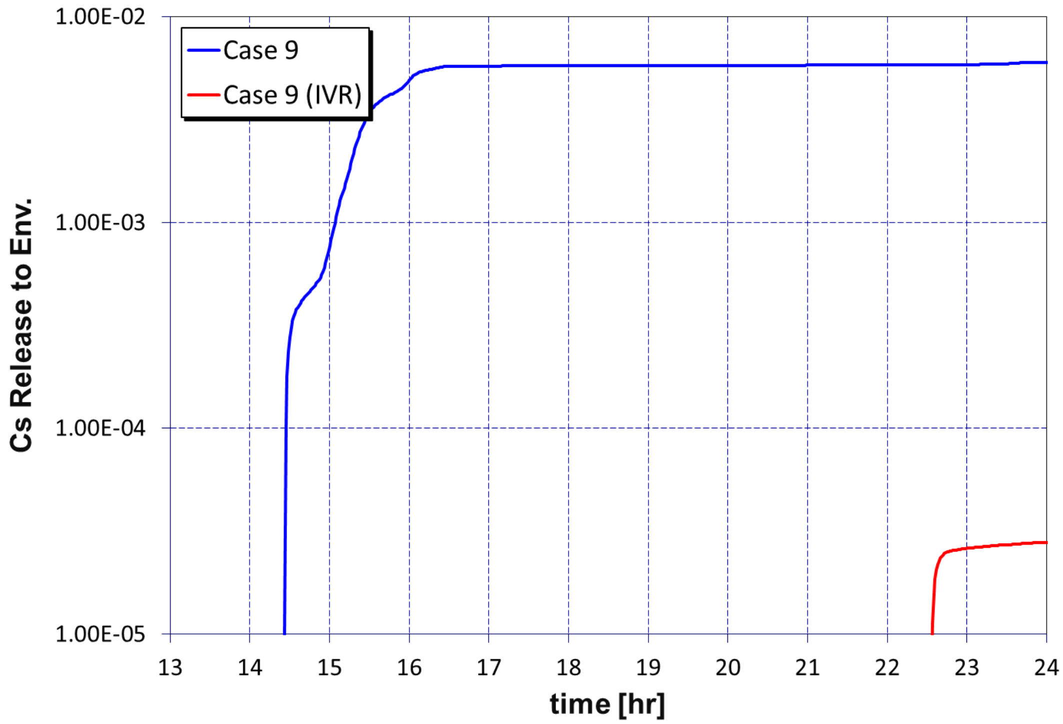
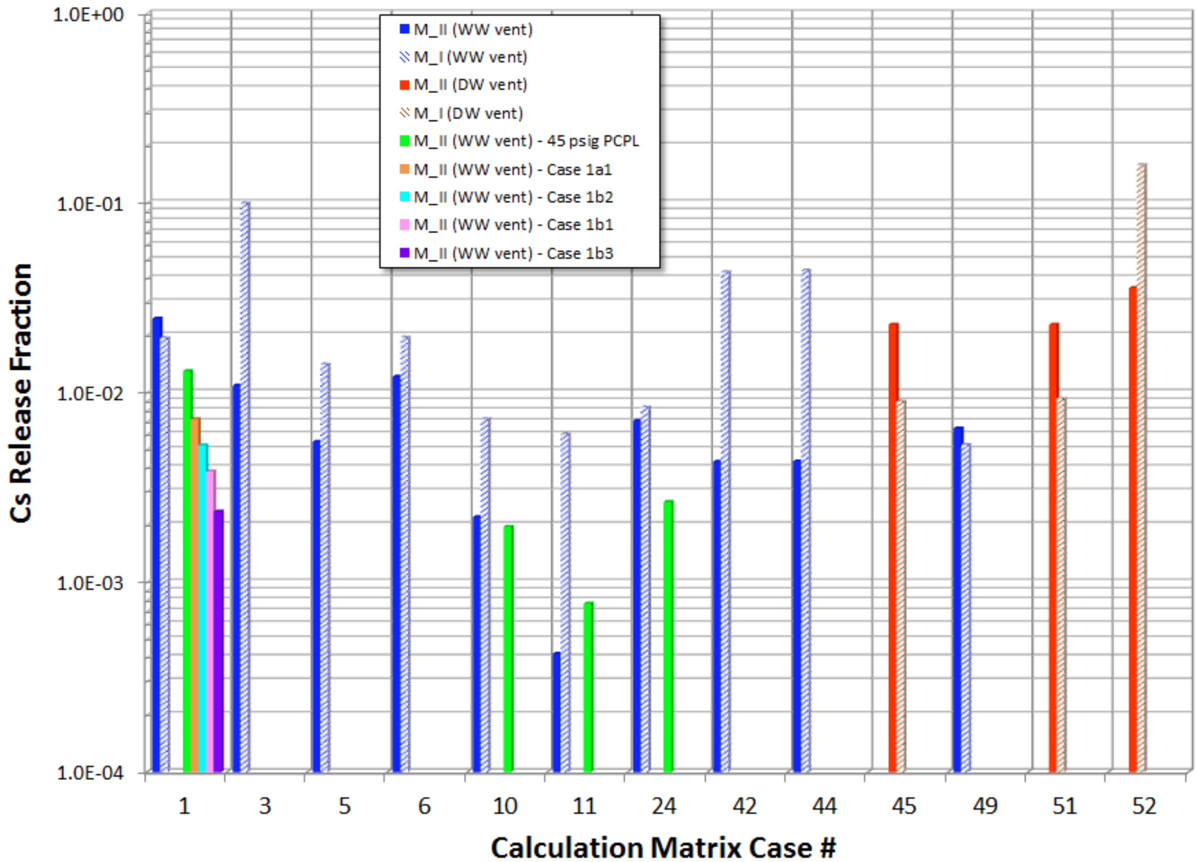


Figure 4-22: Mark I Cesium Release Fraction to Environment for Case 9-IVR



For the Mark II containment, a scoping analysis was performed to investigate different lower cavity configurations by modifying the base model (see Table 4-12). The environmental releases are within the range of source terms predicted based on the variations in the scenario boundary conditions. Figure 4-23, "Environmental Cesium Release Fraction for Mark I and Mark II Cases," presents the cesium release observed for the Mark II analyses (M_I refers to Mark I and M_II refers to Mark II). Creep rupture of the main steam line is unlikely if the pressure is maintained low (either through intentional depressurization or as a result of stuck open SRV during core degradation). Nevertheless, because of the larger containment volume in Mark II containments compared to Mark I containments, MSLCR scenarios do not lead to head flange failure in the Mark II model and the bypass of the suppression pool, thus leading to lower releases compared to Mark I containments. The environmental releases for the Mark II BWR containments are, in general, comparable to or lower than those in the Mark I containments.

Figure 4-23: Environmental Cesium Release Fraction for Mark I and Mark II Cases



Even though MELCOR is considered a state-of-the-art code for severe accident modeling and analysis, and it has a reached a reasonably high level of maturity over the years, it is important to recognize the phenomenological uncertainties in modeling and their effects on the results. Moreover, it is important to understand the compounding effect of various uncertainties on the ultimate parameters of interest (i.e., such as the cesium and iodine release fractions).

The present analyses did not consider a detailed uncertainty analysis, but limited sensitivity analysis was carried out to assess the range of MELCOR results and to further confirm the bounding range of source terms. 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. This did not change the overall conclusions of the analysis.

MELCOR calculations and results for the BWR Mark I and Mark II representative plants will be discussed in more detail in a draft NUREG report which will be released with the proposed CPRR rule.

4.4 MACCS Analyses

The MELCOR Accident Consequence Code System (MACCS) was used to calculate offsite consequences for the source terms generated by MELCOR corresponding to different CPRR accident management strategies following an ELAP event. MACCS was selected because it is one of the standard code systems used for probabilistic consequence analysis. MACCS has a record of continuous development with rigorous quality control and quality assurance processes, an extensive history of applications to a wide variety of assessments, and provides a

capability to model a wide variety of features, events, and processes (i.e., atmospheric transport and deposition, exposure pathways, protective actions, acute and stochastic health effects, and economic impacts) in a fully coupled fashion. No other available code system has the combination of characteristics needed to support this analysis. The code was used as a tool to assess the risk and consequences associated with accidental releases of radioactive material into the atmosphere that are postulated to happen at some unknown point in time in the future.

MACCS uses a polar grid to model the atmospheric transport and dispersion of radionuclides, protective actions, exposure pathways, health effects, and economic costs. The conceptual model in MACCS uses three distinct time phases, as defined in Table 4-15, "Phases of the MACCS Conceptual Model," to model the different types of protective actions and exposure pathways expected for an accident. These are based upon the Environmental Protection Agency's (EPA's) guidance for the offsite response to a radiological accident. First, the early (emergency) phase is used for the period of up to one week following the start of the initiating event which 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 up to 50 or more years.

Table 4-15: 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	~ 1 week, starting at the time of the accident's initiating event.	Weeks to years, starting at the end of the early phase.	Months to decades, starting at the end of the intermediate phase.
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

The results of the consequence analyses are presented in terms of individual health risks to the public (including early and latent cancer fatality risk) and collective consequences including population dose, land contamination, population subject to long-term protective actions, and economic costs. MACCS consequence results are calculated on a conditional basis

(i.e., assuming that the accident occurs). The consequences used here are mean values over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular or annular region.

4.4.1 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 (see Table 4-4). 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.

4.4.1.1 Source Term Modeling

Several proposed CPRR accident management strategies include the use of external filters attached to either the WW or DW vent to reduce environmental releases. While MELCOR allows for detailed modeling of fission product scrubbing provided by the suppression pool, MELCOR does not model fission product scrubbing that would result from an external filter. Therefore the filtration provided by an external filter was modeled by applying a DF to the MELCOR source term in a post-processing step. A range of DFs was selected (10, 100, and 1000) to capture the uncertainty with regard to external filter scrubbing, and different filter designs and sizes. The DF was applied uniformly across all chemical groups and particle sizes. In addition, the DF was applied only to the radionuclides escaping through the WW or DW vent. For MELCOR cases in which other pathways are created, the benefit of an external filter is much smaller because the radionuclides follow the path of least resistance which is often not the WW or DW vent pathway.

For the reference BWR Mark I MELCOR model (Peach Bottom), 41 unique MELCOR simulations were initially run. For the reference BWR Mark II MELCOR model (LaSalle County Generating Station), 12 unique MELCOR simulations were run. With four variations of each (no filter (DF=1), DF=10, DF=100, and DF=1000), a source term binning strategy was employed for optimal efficiency. 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 start time of release is sufficiently delayed for all source terms so this was not used in the binning strategy.

The source term bins were developed based on logarithmic spacing with finer resolution at higher magnitudes. The representative source term from each bin was selected by choosing the source term which was most similar to the average cesium and iodine release fractions for all the source terms in that bin. Tables 4-16, "Binning Strategy for Mark I Source Terms," and 4-17, "Binning Strategy for Mark II Source Terms," show the source term binning strategy for the 18 Mark I source term bins and 9 Mark II source term bins, respectively.

Table 4-16: 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 4-17: 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 (hrs)
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

Tables 4-18, “Identification of Source Term Bin for each Mark I Source Term Case,” and 4-19, “Identification of Source Term Bin for each Mark II Source Term Case,” show the source term bins that correspond to all of the MELCOR case variations. 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 source term bins.

Table 4-18: 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 4-19: 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

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 4-20, "External Filter Effectiveness for Three Example Cases," shows some of the variation in external filter effectiveness through examples of three MELCOR cases.

MELCOR case 1 is shown as an example of a "status quo" accident management strategy (alternative 1) in which no water (i.e., no SAWA/SAWM) is injected and the accident results in DW LMT. Among the many MELCOR cases which result in DW LMT, on average about half the cesium is released through a vent pathway and half escapes through other non-vented pathways. However, in MELCOR case 1, much of the release (78.2% of the cesium) is through a vent pathway[27] so the external filter can 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 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 (17) so the consequences remain effectively unchanged.

For the SAWA/SAWM example, 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.

[27] Section 4.3.2, "MELCOR Results," provides a discussion of the reasons for why the filterable release percent varies among the different MELCOR cases resulting in DW LMT.

Table 4-20: External Filter Effectiveness for Three Example Cases

CPRR Alternatives	MELCOR 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		
Alternative 1 Status Quo (No SAWA/SAWM) 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	
Alternative 1 Status Quo (No SAWA/SAWM) 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	
Alternative 4 SAWA/SAWM	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	

4.4.1.2 MACCS Site Models

Site Data: The SecPop preprocessor code was used to generate site data that is needed for the consequence calculations. SecPop accesses population, land use, and economic value databases for the United States to obtain the data and then uses various algorithms to map the data to each of the 1,664 individual polar grid elements in the MACCS modeling domain for each site. Population data for each site was obtained from the 2010 U.S. Census and was scaled to 2013 using state-level population growth estimates. Land use and economic data was obtained from various U.S. Department of Agriculture[28] (USDA) and U.S. Bureau of Economic Analysis[29] (BEA) databases; economic values were also scaled to 2013 using the consumer price index for all urban consumers inflator.

Meteorological Data: Hourly raw weather data was obtained for Peach Bottom and Limerick from each plant's on-site meteorological tower. The data includes wind direction, wind speed, precipitation, and vertical temperature differential. Raw data was reviewed and converted to

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28 U.S. Department of Agriculture, 2007 Census of Agriculture, http://www.agcensus.usda.gov/Publications/2007/.

29 U.S. Department of Commerce, U.S. Bureau of Economic Analysis, http://www.bea.gov/.

MACCS-specific input. 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. Weather data for these years was compared to that of other years to confirm that these chosen years were representative of the local climate and were not outliers.

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 growth of plume dimensions in the vertical and cross-wind directions was modeled using an Eimutis and Konicek formulation converted into a MACCS lookup table by Bixler et al.[30] Surface roughness, which characterizes the interaction of a plume with ground-based topological features, used the USDA CropScape database[31] for land use within a 30-mile radius of the Peach Bottom and Limerick plants. Dry and wet deposition modeling generally followed expert elicitation guidance in NUREG/CR-7161.[32] Plume meander followed the NRC Regulatory Guide 1.145 model.[33]

Early Phase: MACCS input parameters related to evacuation modeling were developed primarily from the site-specific evacuation time estimate (ETE) report for the Peach Bottom[34] and Limerick[35] plant that were submitted to NRC in early 2014. Evacuation model parameters describe the time delays and travel speeds for different segments (cohorts) of the population such as schools, transit-dependent residents, etc. following a postulated accident. The evacuation model considers that some people may not evacuate (nonevacuees) and others will evacuate before officially ordered to do so (shadow evacuees). Modeling of KI ingestion was informed by discussions with officials at the Pennsylvania Emergency Management Agency. Early phase relocation model parameters were informed by the Environmental Protection Agency (EPA) Protective Action Guides (PAG) Manual,[36] the ETEs, and source term timing data. The daily per person compensation cost associated with evacuation and relocation was selected considering the cost of lodging, food, and lost income.

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30 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.

31 U.S. Department of Agriculture, National Agricultural Statistics Service, CropScape 2013, http://nassgeodata.gmu.edu/CropScape.

32 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.

33 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.

34 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.

35 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.

36 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.

Intermediate Phase: 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. The planning activities include defining areas of interest, characterizing the contamination, identifying decontamination equipment and personnel needed, developing a waste management plan, acquiring decontamination equipment and bringing it onsite, and training personnel and bringing them onsite. All baseline calculations use a period of 3 months for this phase with a dose-dependent relocation criterion (125 mrem) based on linear interpolation of the long-term phase habitability criterion (500 mrem per year in Pennsylvania). Sensitivity calculations were run in which these parameters were varied. The daily per person compensation cost for the population relocated for 3 months is identical to the analogous early phase compensation cost.

Long-Term Phase: This phase starts at the end of the intermediate phase and lasts for 50 years to approximate the time needed to complete all long-term recovery and cleanup activities. Protective actions are implemented to minimize the dose to an individual by external (e.g., groundshine) and internal (e.g., food/water ingestion and resuspension inhalation) pathways. Land is considered uninhabitable and farmland is considered unfit for agricultural production if the dose exceeds the habitability criterion, set at 500 mrem per year based on Pennsylvania-specific guidance. Most states would likely follow the guidance in the EPA PAGs of 2 rem in the first year and 500 mrem per year, each year thereafter. The EPA PAG variation is used in sensitivity calculations. Decontamination is modeled if cost-effective for farmland and non-farmland. Decontamination cost, time, and effectiveness input parameters are consistent with those of past studies such as SOARCA[37, 38] and SECY-12-0157. If the cost to decontaminate exceeds the value of the land, the land is condemned.

Dosimetry: MACCS computes dosimetric quantities including whole body dose and organ-specific doses for use in health effects modeling and protective action decisionmaking. The doses are then summed across all radionuclides and the relevant exposure pathways. In general, the 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 dose conversion factors are based on EPA's Federal Guidance Report Number 13[39] and are consistent with SOARCA and SECY-12-0157. Shielding factors and usage factors are generally consistent with past analyses.

Health Effects: MACCS considers deterministic health effects arising from acute exposures during the early phase and stochastic health effects arising from acute exposures during the early phase and chronic exposures during the intermediate and long-term phases. The

37 U.S. Nuclear Regulatory Commission, NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," Washington, DC, November 2012.

38 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.

39 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.

deterministic health effects model for early fatality risk calculates the likelihood of dying of a fatal cancer based on exceeding an acute dose threshold for the specific organ (red bone marrow, lung, stomach, respectively). The stochastic health effects model for latent cancer fatality risk uses the linear-no-threshold (LNT) model. The model uses cancer risk factors to convert organ doses to 8 different fatal cancer types including leukemia, bone, breast, lung, thyroid, liver, colon, etc. All stochastic and deterministic health effect input parameters are consistent with past studies including SOARCA and SECY-12-0157.

4.4.2 MACCS Off-Site Consequences Results

The summary results for the 18 Mark I source term bins and 9 Mark II source term bins are provided in Tables 4-21, "MACCS Results for 18 Mark I Source Term Bins," and 4-22, "MACCS Results for 9 Mark II Source Term Bins," for the following offsite consequence metrics.

Individual Early Fatality Risk: This is the risk of an average individual in the region of interest contracting a fatal type of acute radiation sickness in the early phase, conditional on the accident occurring. This risk metric 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 risk QHO applies.

Individual Latent Cancer Fatality Risk: This is the risk of an average individual in the region of interest contracting a fatal type of latent cancer from acute exposures in the early phase and chronic exposures in all accident phases, conditional on the accident occurring. This risk metric is presented for the areas within 10, 50, and 100 miles of the site. Results for the 10-mile area is presented because it corresponds to the QHO for cancer latent fatality risk, for the 50-mile area because that region is used in NRC's regulatory analyses, and for the 100-mile area for sensitivity calculations in the regulatory analysis.

Population Dose (person-rem): This is a measure of collective offsite consequences because it sums the doses from all grid elements within the region of interest for all time phases and all exposure pathways, multiplied by the number of people receiving the dose. Results are provided for the 50- and 100-mile areas for use in the regulatory analysis and sensitivities, respectively.

Offsite Cost: This measure is quantified in 2013 dollars because population and economic values were scaled to the most recent year for which all scaling data was available. This metric sums various cost components over the region of interest for all accident phases. Results are provided for the 50- and 100-mile areas for use in the regulatory analysis and sensitivities, respectively. The cost components include:

- evacuation and relocation costs,
- moving expenses for people displaced in the long-term phase,
- decontamination costs,
- costs due to loss of use of property,
- cost of condemned lands, and
- cost of lost agricultural sales.

Land Contamination: This item describes the extent of land in a specified region that exceeds the long-term phase habitability criterion of 500 mrem per year in square miles. Note that this does not indicate the length of time that the habitability criterion is exceeded. Results are provided for the 50- and 100-mile areas for use in the regulatory analysis and sensitivities, respectively.

Population Subject to Long-term Protective Actions: This metric describes how many people could be displaced in the long-term phase for a minimum of one year, based on their land exceeding the 500 mrem per year habitability criterion. Note that this does not indicate the length of time that people are displaced. Results are provided for the 50- and 100-mile areas for use in the regulatory analysis and sensitivities, respectively.

Table 4-21: 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

Table 4-22: 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 Tables 4-21 and 4-22 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 percent of that source term's total cumulative cesium release to the environment. Cesium, rather than iodine, which has a relatively short half-life, was selected here because all of the resulting offsite consequences are driven by long-term phase exposures.

The results from Tables 4-21 and 4-22 are described below:

<u>Individual Early Fatality Risk:</u> Individual early fatality risk is zero for all Mark I and II source term bins within 1.3 miles of the site and beyond. 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.

<u>Individual Latent Cancer Fatality Risk:</u> The ILCF risk generally increases with source term magnitude for the 18 Mark I and 9 Mark II source term bins. The calculated ILCF risk ranges from $5x10^{-7}$ to $7x10^{-4}$ among the Mark I source term bins and from $1x10^{-7}$ to $5x10^{-4}$ among the 9 Mark II source term bins, per event. Because this is a population-weighted consequence metric, the results are generally similar between Peach Bottom and Limerick for a given source term size, despite the fact that Limerick has a substantially higher population in the surrounding 10-mile area.

The risk is dominated by long-term phase exposures to lightly contaminated areas (under the 500 mrem per year habitability criterion threshold). Because the habitability criterion acts as a threshold to limit exposures and health risk, the ILCF risk often has a nonlinear dependence on source term magnitude. A large source term always necessitates more extensive protective actions than a smaller source term (assuming all other conditions are identical), however the increase in ILCF risk may be negligible if the smaller source term was already sufficient to exceed the habitability criterion. Therefore a larger release may displace more people for more

time, and incur a much greater societal cost, while the health risk to the public, measured in ILCF risk, is effectively capped by the habitability criterion.

There are certain cases in which a smaller cesium release causes a larger ILCF risk than a slightly larger cesium release. In these cases, there are other factors, such as the time signature of the release that help explain the consequences. For example, a gradual release, compared to a concentrated puff release of the same magnitude, can be shifted in more compass directions and cause more lightly contaminated land because of the increased likelihood that the wind shifts direction. Depending on how the resulting doses compare to the habitability criterion, having more lightly contaminated area and less highly contaminated area, can increase the ILCF risk estimate.

<u>Population Dose:</u> 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 ILCF 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 non-frequency weighted population dose ranges from 1,600 to 3.8 million person-rem among the 18 Mark I source term bins and from 300 to 6.1 million person-rem among the 9 Mark II source term bins within 50 miles, per event. Unlike the ILCF risk, this metric is directly dependent on population and therefore consequences are generally higher for Limerick than Peach Bottom for a comparable source term size.

The groundshine pathway during the intermediate and long-term phases is generally the largest contributor among the exposure pathways for population dose. Like ILCF risk, the habitability criterion acts as a threshold to limit exposures, and therefore population dose generally shows a nonlinear dependence on source term magnitude. Also like ILCF risk, there are certain cases in which a smaller cesium release causes a larger population dose than a slightly larger cesium release. In these cases, there are other factors, such as the time signature of the release that help explain the consequences. For example, a gradual release, compared to a concentrated puff release of the same magnitude, can be shifted in more compass directions and cause more lightly contaminated land because of the increased likelihood that the wind shifts direction. Depending on how the resulting doses compare to the habitability criterion, having more lightly contaminated area and less highly contaminated area, can increase the population dose.

<u>Offsite Economic Cost:</u> 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. The total non-frequency weighted offsite cost, measured in 2013 dollars, ranges from $80 million to $50 billion among the 18 Mark I source term bins and from $400 million to $90 billion among the 9 Mark II source term bins within 50 miles, per event. Like population dose, this metric is directly dependent on population and therefore consequences are generally higher for Limerick than Peach Bottom for a comparable source term size.

For the smallest source terms (Mark I bins 1-3 and Mark II bins 1-3), the total non-frequency weighted offsite cost is dominated by the cost of evacuating the EPZ population for the duration of the early phase. For larger source terms, the two largest components of total offsite cost are the cost of intermediate phase relocation and the cost of interdicting populated land in the long-term phase. The contribution of population-dependent decontamination costs increases with source term magnitude but doesn’t account for more than 10 percent of the total 50-mile offsite cost, even for the largest source terms. 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 available for this regulatory basis, 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.

There are significant uncertainties in the offsite economic cost model as with other models in the code. In order to better understand the uncertainty related to offsite economic cost, the Fukushima accident recovery provides a useful comparison. In December 2013, the central Japanese government raised the loan ceiling for the Nuclear Damage Liability Facilitation Fund to 9 trillion yen ($76 billion)[40] under the fiscal year 2014 budget.[41] The fund allows Tokyo Electric Power Company to provide compensation to those affected by the nuclear accident and for decontamination and waste storage efforts at least until the ceiling is reached, at which point the Fund may be increased at the government's discretion. These funds pay for decontamination and waste storage efforts (2.5 trillion and 1.1 trillion yen, respectively) for major cleanup efforts in the "targeted decontamination areas"[42], as well as compensation for nuclear damages (5 trillion yen). This compensation is mainly given to 86,000 individuals that have been displaced from areas of the government mandated evacuation and affected businesses.[43]

In comparison to a cost of about $76 billion for an accident affecting 86,000 individuals, Mark I source term bin 14, represented by case 22dw, subjects 93,700 individuals to long-term protective actions for the area within 50 miles. This case has a total offsite economic cost of about $12.9 billion associated with the 50-mile area. Comparing the two costs for accidents that displace a roughly similar population size shows that the Government of Japan projects the cost to be roughly 6 times higher than the MACCS calculation in this analysis. The reason for this difference is not well understood yet, as the estimates are based on a number of factors such as the length of time before people return and the level of necessary cleanup efforts. As more information becomes available the NRC will evaluate the assumptions in the MACCS calculations and modify the analysis, as necessary, to reduce the uncertainties.

<u>Land Contamination:</u> Beyond total population dose and total offsite economic cost, the extent of land contamination represents an additional measure of the societal cost of a nuclear accident. Land contamination is measured as the area of land that exceeds the long-term phase habitability criterion, which is modeled 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

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40 This is based on a currency conversion rate of 100 Japanese yen is equal to $0.85 as of January 27, 2015.

41 Asahi Shimbun. "Cabinet approves new approach to rebuilding Fukushima." http://ajw.asahi.com/article/0311disaster/recovery/AJ201312210036, December 2013.

42 Ministry of the Environment, Government of Japan. "Measures for Decontamination of Radioactive Materials Discharged by TEPCO's Fukushima Daiichi NPS Accident." http://josen.env.go.jp/en/.

43 Ministry of Economy, Trade, and Industry, Government of Japan. "Designating and Rearranging the Areas of Evacuation." http://www.meti.go.jp/english/earthquake/nuclear/roadmap/pdf/20120723_01.pdf, July 2012.

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 or suitable for agricultural production.

Land contamination ranges from 0 to approximately 1,400 square miles among the 18 Mark I source term bins and from 0 to approximately 900 square miles among the 9 Mark II source term bins within 50 miles, per event. (The maximum value of 1,400 square miles within a 50 mile radius represents about 18 percent of the area.) This metric is particularly useful because it is independent of population size and distribution.

Population Subject to Long-Term Protective Actions: The extent of population subject to long-term protective actions is yet another metric for quantifying the societal cost of a nuclear accident. This metric does not consider the duration of time for which people are displaced from their land; some of the population might be able to return within a few years while others might be displaced for a decade or more, or permanently.

The population subject to long-term protective actions is the population residing in a spatial grid element where the land is not habitable because the 500 mrem per year habitability criterion is exceeded. The value ranges from 0 to approximately 500,000 people among the 18 Mark I source term bins and from 0 to approximately 700,000 people among the 9 Mark II source term bins within 50 miles, per event. (Considering that about 8.1 million people live within 50 miles of Limerick, the maximum value of 700,000 represents about 9 percent of the population.) This metric is directly dependent on population and therefore consequences are generally higher for Limerick than Peach Bottom for a comparable source term size.

4.4.3 MACCS Sensitivity Calculation Results

There is considerable variation in many different characteristics at the 18 Mark I and 9 Mark II BWR sites. Sites differ with respect to population, economic values, land use, weather, emergency response characteristics, long-term protective actions, and many others. The reference Mark I MACCS model (Peach Bottom) and Mark II MACCS model (Limerick) were adjusted to assess the sensitivity of the offsite consequence results as a result of varying these inputs:

- population (low, medium, high)
- evacuation delay (1 hour)
- nonevacuating cohort size (5 percent of EPZ population)
- intermediate phase duration (0, 3 months, and 1 year)
- long-term habitability criterion (500 mrem/year and 2 rem/year)

The results of these more than 100 calculations will be documented in a separate draft NUREG report that will be issued with the CPRR proposed rule. However, the following are some general conclusions:

- Early fatality risk remains zero for all sensitivity calculations.

- Individual latent cancer fatality risk remains sufficiently low that when multiplied by the accident frequency, there is still substantial margin to the NRC Safety Goal for ILCF risk.

- An evacuation delay of 1 hour applied uniformly to all evacuation cohorts has zero effect on any of the results because the releases begin late enough that the surrounding population is expected to still have time to evacuate.

- Sensitivities using a larger nonevacuating cohort (5 percent of EPZ population compared to 0.5 percent or 2 percent) show a small (≤ 20 percent) increase to the ILCF risk for the 10-mile area. Other consequence metrics are less sensitive to the size of the nonevacuating cohort.

- Sensitivities using the higher habitability criterion of 2 rem per year generally result in higher ILCF risk and population dose, and lower offsite cost, land contamination, and population subject to long-term protective actions.

- Sensitivities using a shorter (zero) or longer (1 year) intermediate phase show both an increase and a decrease in each consequence metric depending on the size of the source term. The largest increase in offsite cost results from using the longer intermediate phase with the largest source term (up to 60 percent higher within 50 miles).

- Population sensitivities generally show that population dose, offsite cost, and population subject to long-term protective actions depend on population while ILCF risk and land contamination have no correlation. The results, however, depend on the source term magnitude.

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 percent of the cesium and 1.7 percent 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.

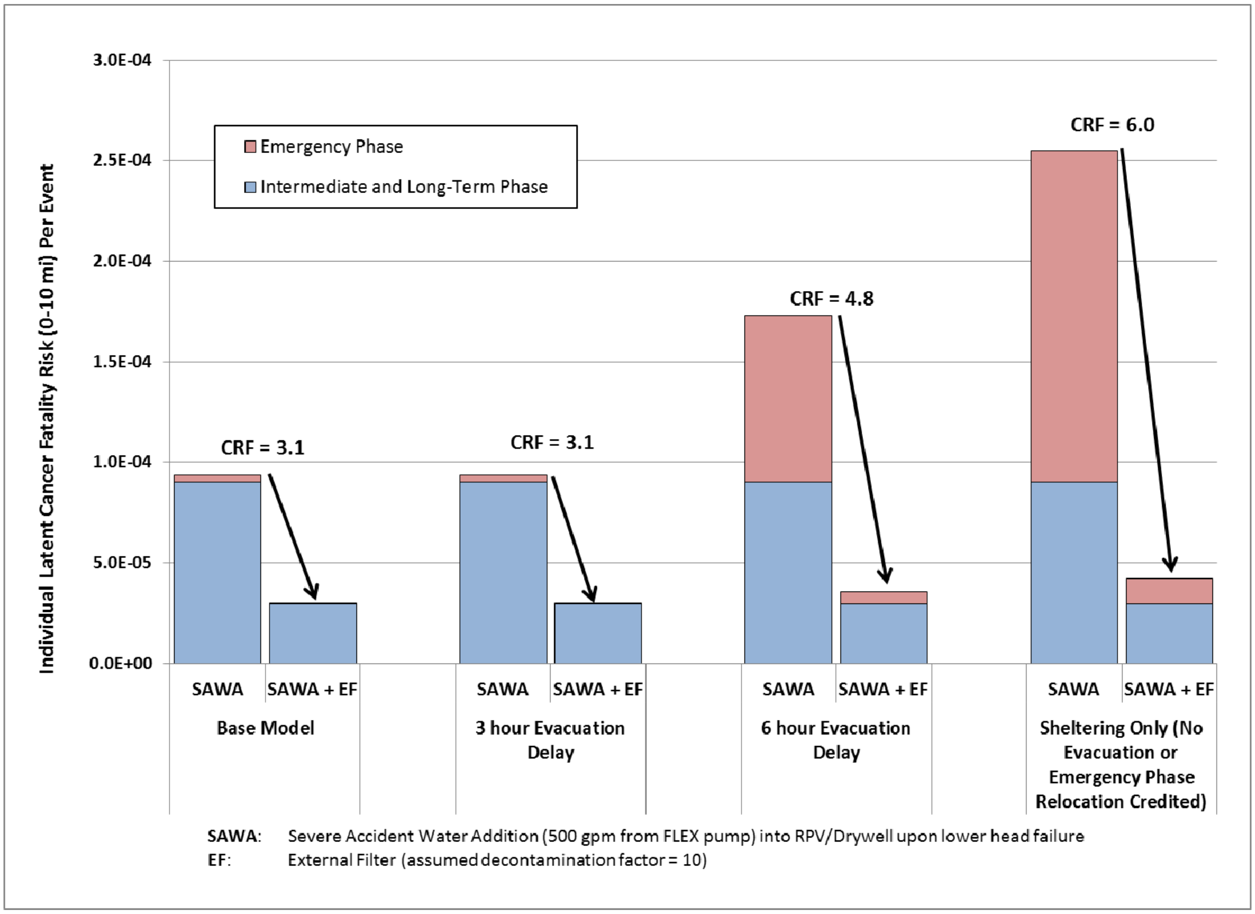
These sensitivity calculations are designed to simulate intentionally unrealistic emergency response situations. Emergency response programs are developed, tested, and evaluated by the NRC as an element of defense-in-depth. Detailed plans for onsite and offsite responses are approved by NRC and FEMA 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 4-24, "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 4-24, 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 4-24, 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 4-24, 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 percent. 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 4-24.

Figure 4-24: 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 4-24 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 zero individual early fatality risk.

4.5 Conditional Consequences of Different CPRR Alternatives

The NRC has evaluated the conditional offsite consequences associated with the three major accident response strategies; (1) alternative 1 (status quo), (2) alternative 3 (SAWA/SAWM), and (3) alternative 4 (SAWA/SAWM with an external filter). For the Mark I analyses, water injection 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 (e.g., status quo (alternative 1) for Mark I is represented by MELCOR cases 1, 2, 4, 5, and 6). 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 Tables 4-23, "Average Mark I Conditional Offsite Consequences for the Different MELCOR Cases Associated with the CPRR Alternatives," and 4-24, "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 Tables 4-25, "Conditional Mark I Offsite Consequences and Consequence Reduction Factor (CRF) for each CPRR Alternative," and 4-26, "Conditional Mark II Offsite Consequences and Consequence Reduction Factors (CRF) for each CPRR Alternative." Tables 4-25 and 4-26 also show the consequence reduction factor (CRF) associated with each alternative relative to the status quo.

Table 4-23: 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)
Status Quo (No Water)	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***
SAWA/ SAWM	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***
SAWA/ SAWM + External Filter	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 4-24: 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)
Status Quo (No Water)	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***
SAWA	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***
SAWA + External Filter	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 4-25: Conditional Mark I Offsite Consequences and Consequence Reduction Factor (CRF) for each CPRR Alternative

Offsite Consequences (per event) / CPRR Alternative	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
Status Quo (No Water)	3.3E-04	1.0	1,600,000	1.0	12,000,000,000	1.0	530	1.0	69,000	1.0
SAWA/SAWM	1.3E-04	2.5	720,000	2.2	4,000,000,000	3.0	230	2.3	23,000	3.0
SAWA/SAWM + External Filter	3.8E-05	8.7	120,000	13.3	490,000,000	24.5	62	8.5	1,100	62.7

Table 4-26: Conditional Mark II Offsite Consequences and Consequence Reduction Factors (CRF) for each CPRR Alternative

Offsite Consequences (per event) / CPRR Alternative	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
Status Quo (No Water)	3.4E-04	1.0	4,100,000	1.0	63,000,000,000	1.0	620	1.0	470,000	1.0
SAWA	1.5E-04	2.3	1,000,000	4.1	9,700,000,000	6.5	160	3.9	26,000	18.1
SAWA + External Filter	5.5E-05	6.2	140,000	29.3	690,000,000	91.3	30	20.7	690	681.2

Figures 4-25, “Conditional Mark I Offsite Consequences for each CPRR Alternative as a Percentage of the Status Quo,” and 4-26, “Conditional Mark II Offsite Consequences for each CPRR Alternative as a Percentage of the Status Quo,” show graphical depictions of the offsite consequences of the four CPRR alternatives as a percentage of the status quo.

Figure 4-25: Conditional Mark I Offsite Consequences for each CPRR Alternative as a Percentage of the Status Quo

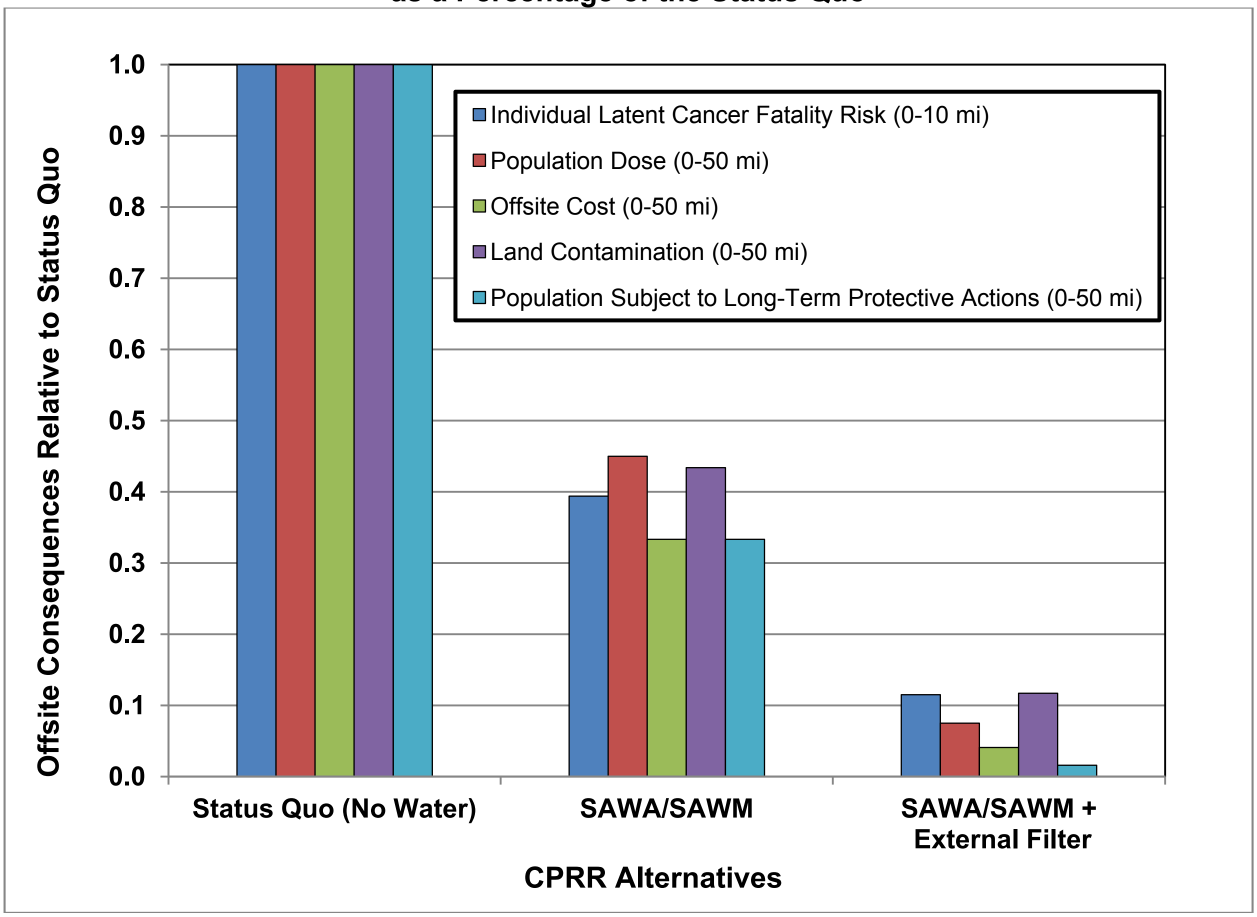
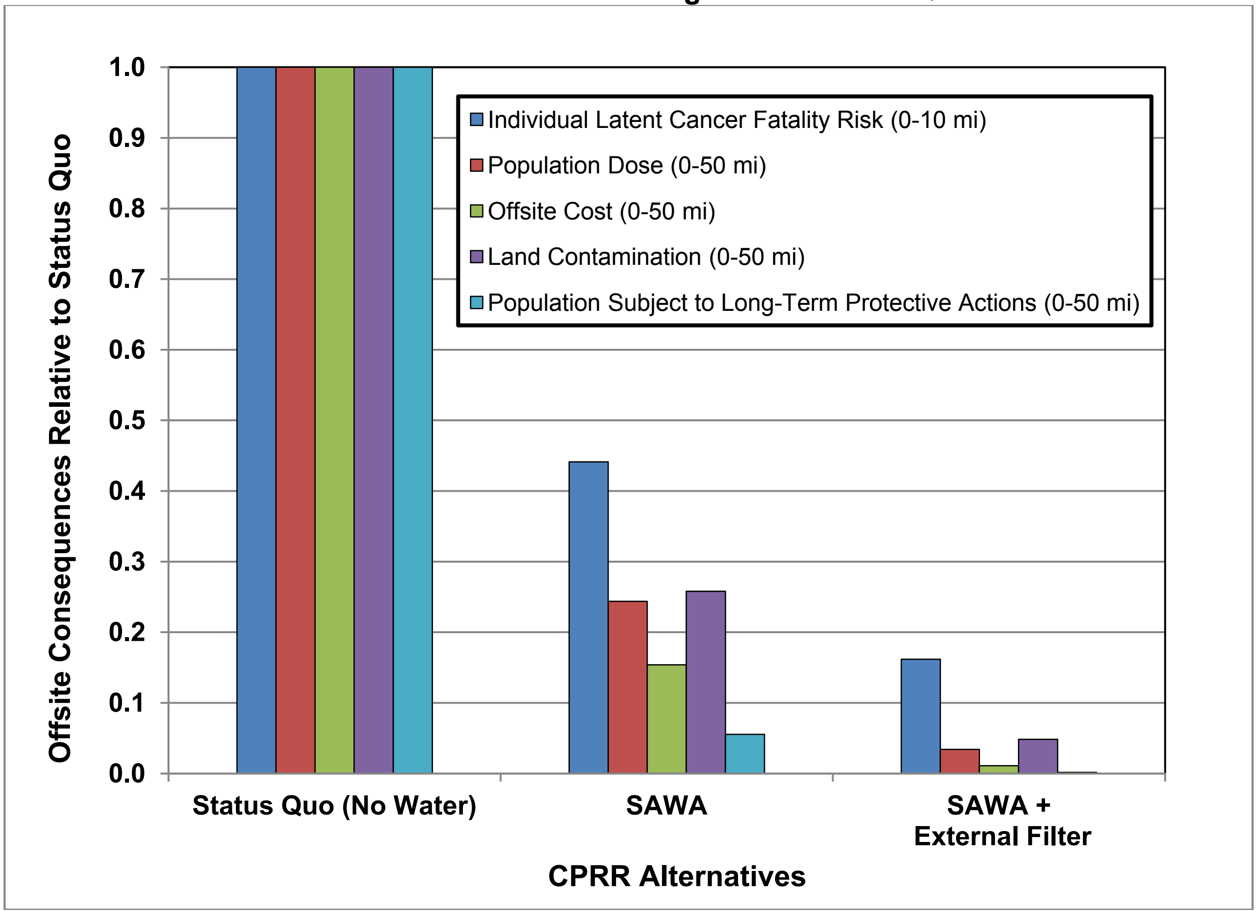


Figure 4-26: Conditional Mark II Offsite Consequences for each CPRR Alternative as a Percentage of the Status Quo



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 the status quo, water injection (SAWA/SAWM) results in a notable reduction in offsite consequences for all five metrics (ILCF risk, population dose, offsite cost, land contamination, and population subject to long-term phase relocation).

- For the Mark I analysis, relative to the status quo, the combination of 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 the status quo, 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 the status quo 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 status quo alternative, $3x10^{-4}$ 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 Technical Evaluation Summary

4.6.1 Comparisons with SECY-12-0157

SECY-12-0157 did not include a detailed PRA compared to what was done for this regulatory basis, but instead developed a fairly simple release event tree, which traced the accident progression starting from the onset of core damage. The initial event tree headings parsed the total CDF according to the type of initiating event and core-damage sequence. Subsequent event tree headings considered operation of the severe accident-capable (SA) hardened vent and the availability of a water supply to the drywell. Each sequence was assigned to a unique containment status:

- Vented: The SA vent is opened, preventing containment over-pressurization failure. A source of water to the drywell exists, preventing liner melt-through.

- LMT: The SA vent is opened, preventing containment over-pressurization failure. No source of water to the drywell exists, and liner melt-through occurs.

- OP: The SA vent is closed, resulting in containment over-pressurization failure. A source of water to the drywell exists, preventing liner melt-through.

- OP + LMT: The SA vent is closed, resulting in containment over-pressurization failure. No source of water to the drywell exists, and liner melt-through occurs.

The release event tree consisted of six event tree headings (top events), which are described as follows:

<u>Event “CD”:</u> Represents the occurrence of core damage, which is the starting point of the risk evaluation. It should be noted that the risks resulting from radiological releases are directly proportional to the CDF.

<u>Event “Hazard”:</u> Partitions core-damage sequences according to their initiating event hazard type; either internal hazards (such as a loss-of-coolant accident (LOCA) or external hazards (such as a seismic event). This partitioning is included in the event tree structure to determine if offsite power is recoverable.

<u>Event “Sequence Type”:</u> Partitions core-damage sequences according to their timing or influence on containment integrity such as:

- Sequence “other” denotes the internal hazard sequences that are not “SBO,” “bypass,” or “fast.”

- Sequence type “SBO” denotes core-damage sequences that involves station blackout. In these sequences, it may be possible to recover offsite power, which allows the use of in-plant systems (such as condensate) to provide a source of water to the containment drywell.

- Sequence type “bypass” denotes core-damage sequences that involve containment bypass (such as interfacing systems LOCAs). In these sequences, venting the containment is not helpful because the containment has already functionally failed.

- Sequence type “fast” denotes sequences that evolve quickly (such as medium LOCAs, large LOCAs, and anticipated transients without scram) and, therefore, reduce the available time for the operator to manually open the SA vent.

- Sequence “other” denotes the external hazard sequences that are not “bypass.”

- Sequence type “bypass” denotes core-damage sequences that involve containment bypass (such as large seismic events that directly damage the containment). In these sequences, venting the containment is not helpful because the containment has already functionally failed.

<u>Event “Vent”:</u> Identifies if the SA vent is opened.

<u>Event “OSP Recovery”:</u> Identifies if offsite power is recovered.

<u>Event “Portable Pump”:</u> Identifies if a portable pump is used to provide water to the drywell floor via the core spray system or drywell spray system following core damage.

This logic led to a tree with only 15 sequences to analyze.

In contrast, the risk evaluation described in Section 4.1, “Risk Evaluation,” was based on a rather elaborate core damage event tree and an equally elaborate accident progression event tree that are used to evaluate a very large number of sequences representing various RPV depressurization, wetwell and drywell venting, and water addition strategies. More than 50 MELCOR accident progression sequences were run to capture the required level of detail to evaluate the various alternatives discussed in this regulatory basis.

Also, in contrast to the SECY-12-0157 study, seismic initiators and equipment failures were analyzed in detail, and human error probabilities were estimated in a fairly simple fashion. None of these tasks were carried out in SECY-12-0157.

There was no fundamental shift in the technical approach or results with regard to the MELCOR and MACCS analyses presented in SECY-12-0157 in comparison with the draft regulatory basis document supporting the current CPRR rulemaking. The scope of the analysis was expanded to also include MELCOR and MACCS modeling of a Mark II BWR containment and site. In both descriptions, the scope of the MELCOR analysis covers thermal-hydraulic performance of the containment and source term estimates for risk significant accident sequences arising from an ELAP event. The technical approach for the MELCOR analysis in both cases 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. Notably, the current analysis incorporates

opening and closing the wetwell vent early in an ELAP rather than keeping the vent closed until core damage is imminent. This is the strategy selected by the industry to comply with Order EA-12-049, "Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735). The results were very similar and verified that both water addition and venting are required to maintain containment structural integrity and that water addition is a beneficial strategy to influence containment conditions, cool core debris, and reduce radiological releases.

The MACCS analysis provided in the draft regulatory basis document, likewise, is similar to that in SECY-12-0157 with the exception that the MACCS analysis now also includes Mark II BWR containments. 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, ILCF risk, population dose, offsite cost, contaminated land area, and population subject to long-term protective actions.

The quantitative results and the key insights from the current analysis and the analysis included in SECY-12-0157 are quite similar. Notably, there is no early fatality risk for all cases analyzed and the ILCF risk is more than a factor of 10 lower than the QHO level. Conditional ILCF risk (per event) is dominated by long-term phase exposures to contaminated areas and because of the habitability criterion, this metric is relatively insensitive to the source term magnitude whereas the societal consequence metrics are often more sensitive. After the staff issued SECY-12-157, the Commission issued SRM-SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated March 20, 2013 (ADAMS Accession No. ML13079A055), which stated "that the NRC's current approach to the issue of land contamination from reactor accidents is sound" and "that economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety." The Commission direction in the SRM was not available to the staff when SECY-12-0157 was sent to the Commission but has been considered by the staff in the CPRR regulatory basis to reduce the reliance on the consideration of qualitative factors, and the consideration and weighting of land contamination and other factors.

The scope and level of detail of the PRA model developed for SECY-12-0157 has been substantially expanded. The PRA model used in SECY-12-0157 did not delineate core-damage sequences; rather, it relied on a generic estimate of core-damage frequency developed from previous NRC staff and licensee PRAs. In order to provide a quantitative basis for regulatory decisionmaking, the draft CPRR regulatory basis PRA includes the following features:

- Models to estimate the frequency of ELAP events resulting from internal events and earthquakes, based on seismic hazard estimates developed by the industry.

- Core-damage event trees (CDETs) that delineate accident sequences from the occurrence of an ELAP event to the onset of core damage. The CDETs reflect station blackout mitigation strategies using installed plant and portable equipment.

- Accident progression event trees (APETs) that delineate accident sequences from the onset of core damage to the release of radioactive materials. The APETs reflect CPRR strategies such as post-core-damage containment venting and water addition.

- Random and seismically induced equipment failures.

- In-control room and local manual operator actions specified by the emergency procedure and severe accident guidelines.

- Identification of important contributors to core-damage frequency.

- Sensitivity evaluations to gain insight into how human error probability affects the quantitative results.

The results of the revised PRA show a low value for core damage frequency from an ELAP event, and provide insights into which initiating events (e.g. the strength of an earthquake), equipment failures, and human errors contribute the most to overall core damage frequency for the plants with Mark I and Mark II containments.

4.6.2 Key Insights from the Risk Evaluation

The point-estimate risk for each CPRR alternative is presented in Table 4-27, "Summary of Point-Estimate Risk Evaluation Results." Note that the order of the alternatives used in Table 4-27 is different than in Table 4-3; in Table 4-27, regulatory basis alternatives that use similar CPRR strategies have been grouped together. Figures 4-27 through 4-31 present the same information graphically.

Table 4-27: Summary of Point-Estimate Risk Evaluation Results.

Number	Regulatory Basis Alternative	Core-Damage Frequency (/ry)	Conditional Containment Failure Probability	Individual Early Fatality Risk (0-1.3 miles and beyond (/ry))	Individual Latent Cancer Fatality Risk (0-10 miles (/ry))	Population Dose Risk (0-50 miles (person-rem/ry))	Offsite Cost Risk (0-50 miles ($ thousands/ry))	Land Exceeding Long-Term Habitability Criterion (0-50 miles (square miles/ry))	Population Subject to Long-Term Protective Actions (0-50 miles (persons/ry))
Only WW severe accident venting									
1	1	$8.9x10^{-6}$	100%	0	$3.0x10^{-9}$	13	99	0.0044	0.51
2	2A	$8.9x10^{-6}$	100%	0	$3.0x10^{-9}$	13	99	0.0044	0.51
WW severe accident venting and RPV injection with number 11 & 16 filtered venting									
3	3A	$8.9x10^{-6}$	42%	0	$1.8x10^{-9}$	9	65	0.0029	0.33
5	4Ai(1)	$8.9x10^{-6}$	42%	0	$1.8x10^{-9}$	9	65	0.0029	0.33
7	4Aii(1)	$8.9x10^{-6}$	42%	0	$1.8x10^{-9}$	9	65	0.0029	0.33
9	4Aiii(1)	$8.9x10^{-6}$	42%	0	$1.8x10^{-9}$	9	65	0.0029	0.33
11	4Bi(1)	$8.9x10^{-6}$	42%	0	$1.3x10^{-9}$	5	29	0.0016	0.15
16	4Ci(1)	$8.9x10^{-6}$	42%	0	$1.3x10^{-9}$	5	29	0.0016	0.15
WW severe accident venting and DW injection with number 12 & 17 filtered venting									
4	3B	$8.9x10^{-6}$	58%	0	$2.1x10^{-9}$	11	74	0.0034	0.41
6	4Ai(2)	$8.9x10^{-6}$	58%	0	$2.1x10^{-9}$	10	68	0.0032	0.36
8	4Aii(2)	$8.9x10^{-6}$	58%	0	$2.4x10^{-9}$	12	89	0.0039	0.48
10	4Aiii(2)	$8.9x10^{-6}$	58%	0	$2.0x10^{-9}$	9	62	0.0030	0.31
12	4Bi(2)	$8.9x10^{-6}$	58%	0	$1.4x10^{-9}$	5	31	0.0018	0.16
17	4Ci(2)	$8.9x10^{-6}$	58%	0	$1.3x10^{-9}$	4	30	0.0016	0.15
DW severe accident filtered venting and DW injection									
13	4Bii	$8.9x10^{-6}$	58%	0	$1.4x10^{-9}$	5	30	0.0017	0.15
14	4Biii	$8.9x10^{-6}$	58%	0	$1.4x10^{-9}$	5	31	0.0017	0.15
15	4Biv	$8.9x10^{-6}$	60%	0	$1.3x10^{-9}$	5	30	0.0017	0.15
18	4Cii	$8.9x10^{-6}$	58%	0	$1.3x10^{-9}$	4	29	0.0015	0.15
19	4Ciii	$8.9x10^{-6}$	58%	0	$1.3x10^{-9}$	4	30	0.0016	0.15
20	4Civ	$8.9x10^{-6}$	60%	0	$1.3x10^{-9}$	4	29	0.0015	0.15

* Alternatives which use external filters are highlighted in the above table.

Figure 4-27: Comparison of Alternatives Using Individual Latent Cancer Fatality Risk

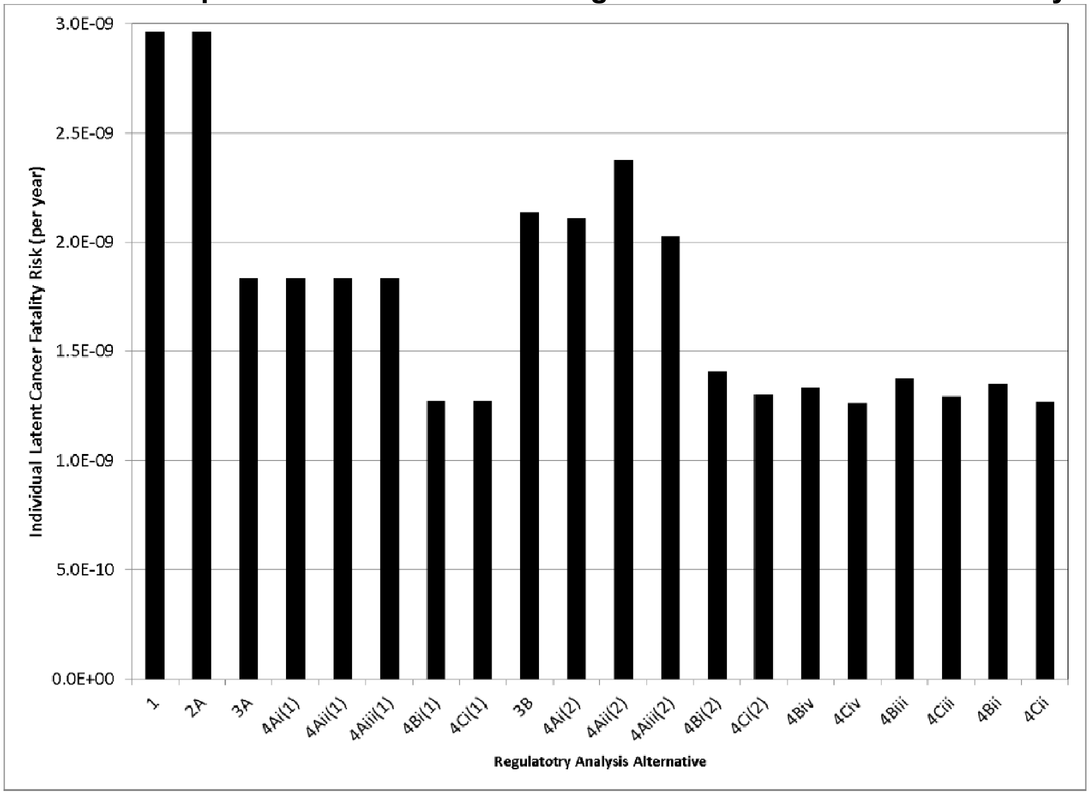


Figure 4-28: Comparison of Alternatives Using Population Dose Risk

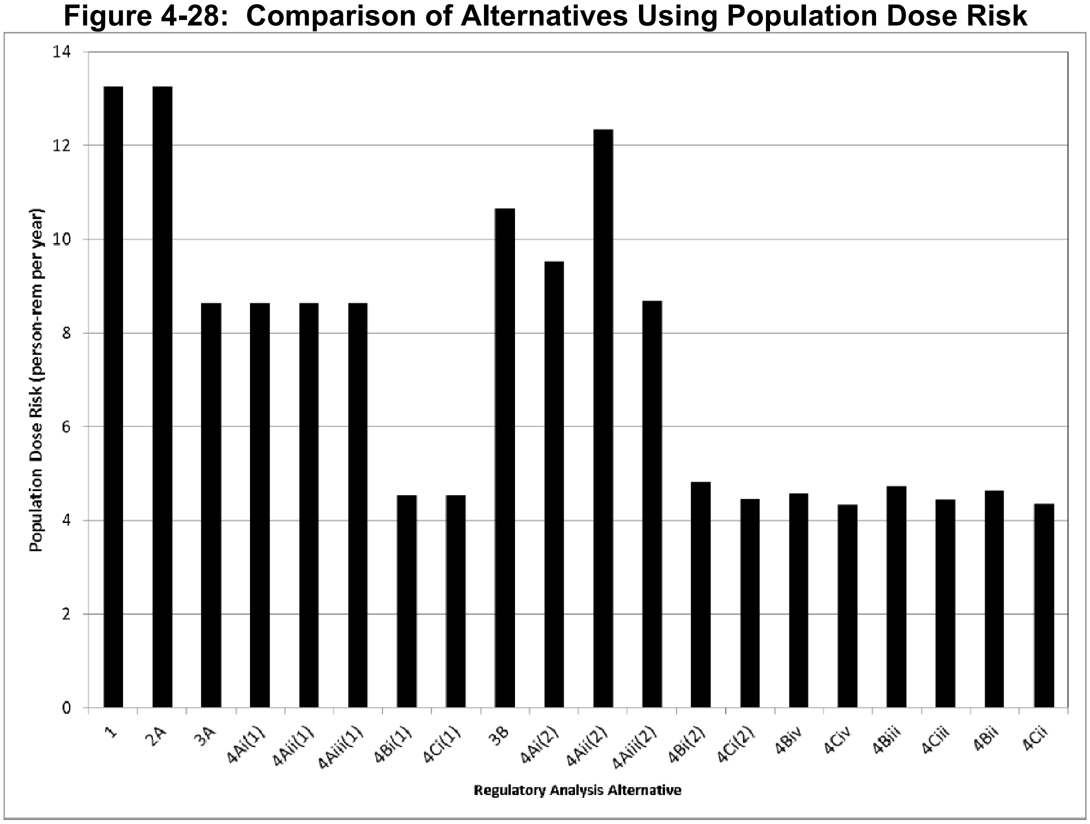


Figure 4-29: Comparison of Alternatives Using Offsite Cost Risk

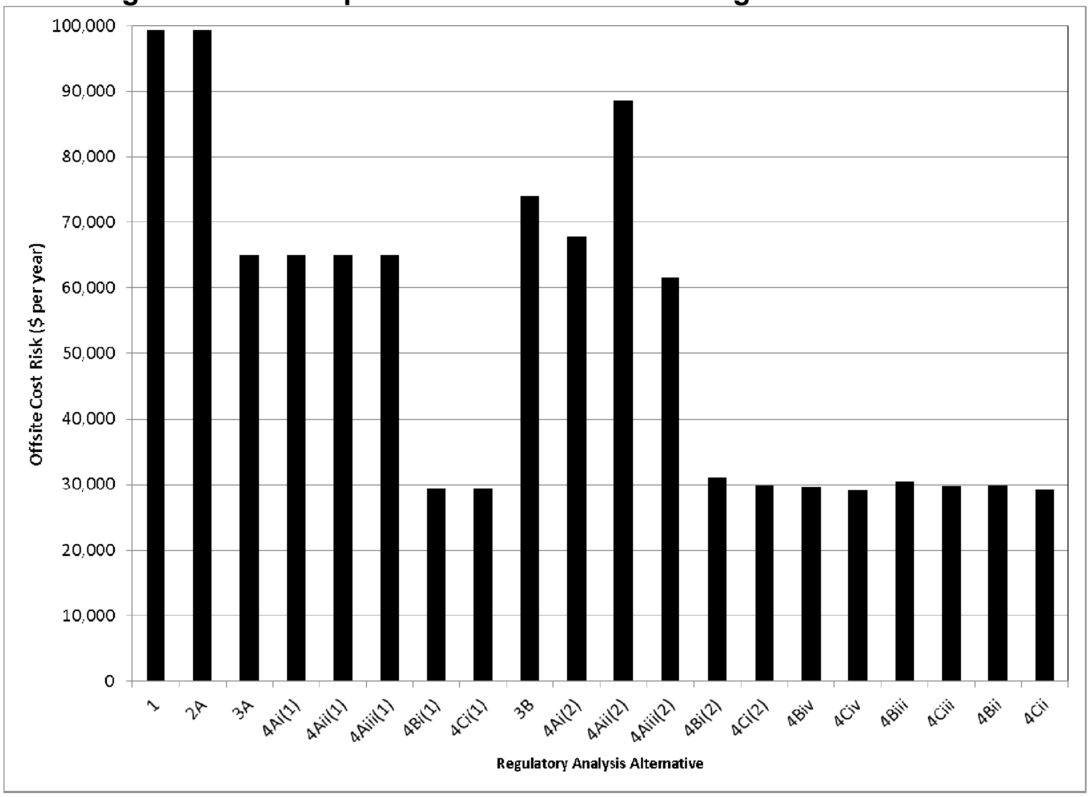


Figure 4-30: Comparison of Alternatives Using Area of Land Exceeding Long-Term Habitability Criterion

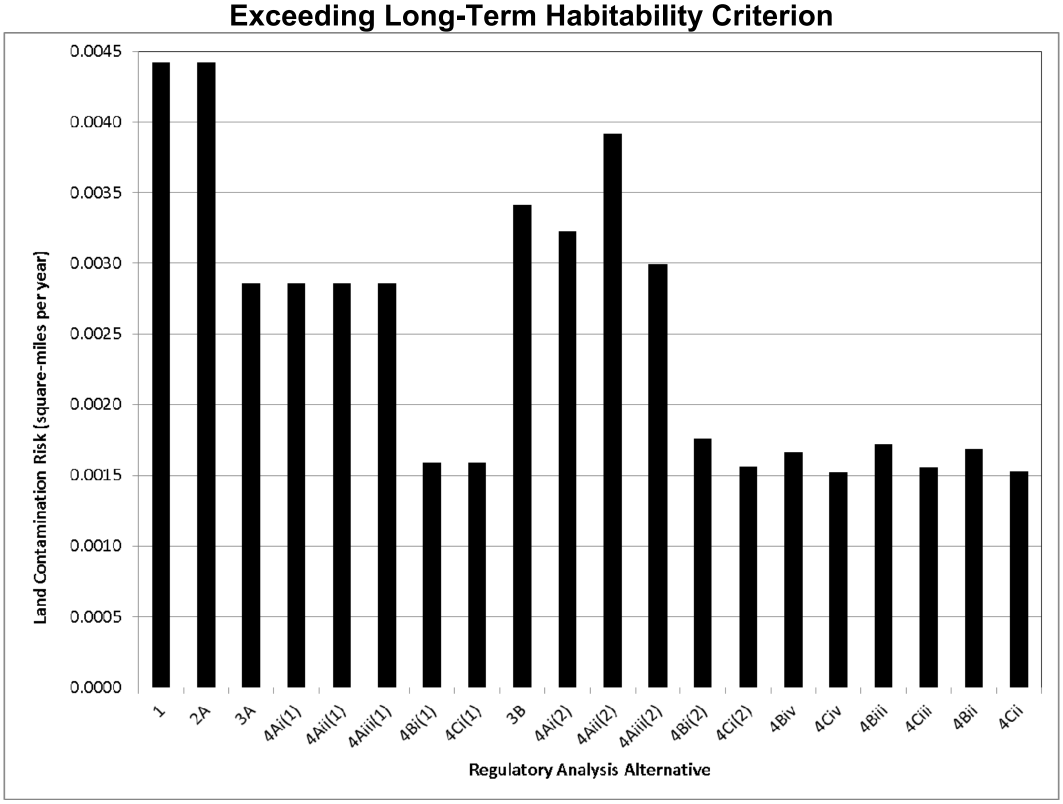
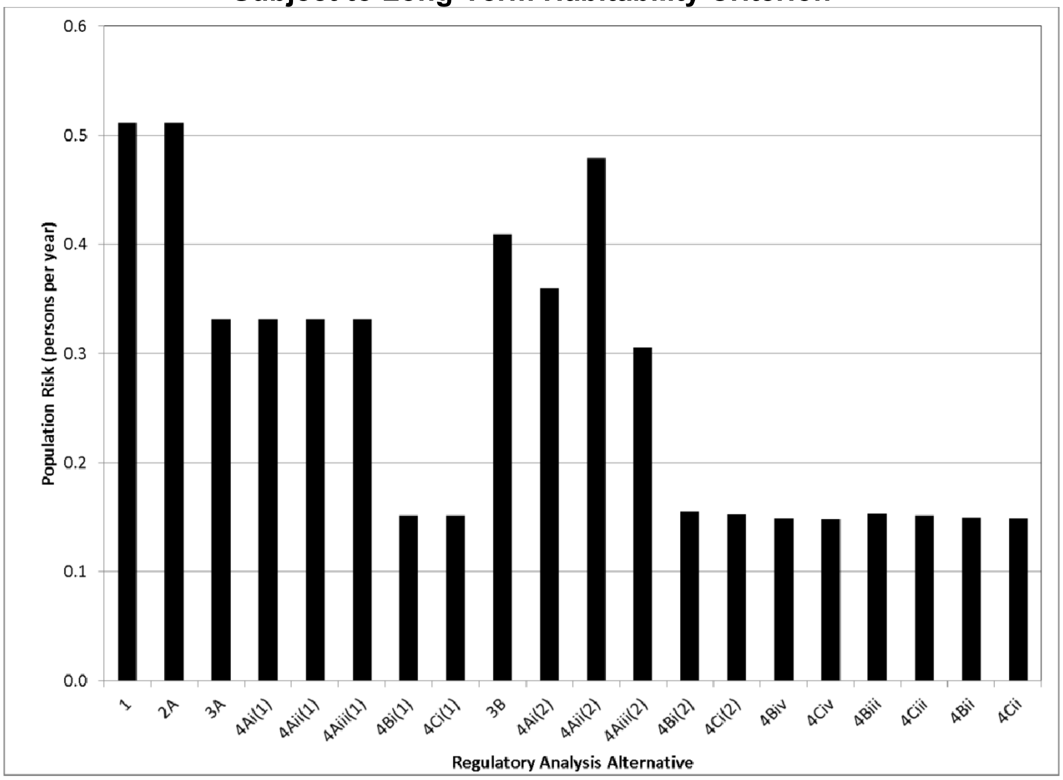


Figure 4-31: Comparison of Alternatives Using Population Subject to Long-Term Habitability Criterion



Two key risk insights may be developed from the point-estimate risk evaluation results:

- As shown in the results for alternatives 1 and 2A, BWR plants with Mark I containments and RCIC systems currently meet the QHOs defined in the Commission's Safety Goal Policy. Specifically, individual early fatality risk is zero and the individual latent cancer fatality risk is almost two orders of magnitude below its QHO of 2×10^{-6}/ry. These results are consistent with those obtained in the SOARCA study.

- Alternative 4 use of engineered filters has lower release rates than alternative 3 which relies on the operation of the SA vents and SAWA.

As illustrated in Figures 4-32, "Comparison of Alternatives Using Conditional Containment Failure Probability," and 4-33, "Comparison of Alternatives Using Ability to Retain Core Debris," post-core-damage water injection to provide core debris cooling is an important CPRR strategy for preventing containment liner melt-through. Water injection into the RPV is preferable to water injection directly to the drywell because it provides the capability to arrest core damage prior to vessel breach.

Figure 4-32: Comparison of Alternatives Using Conditional Containment Failure Probability

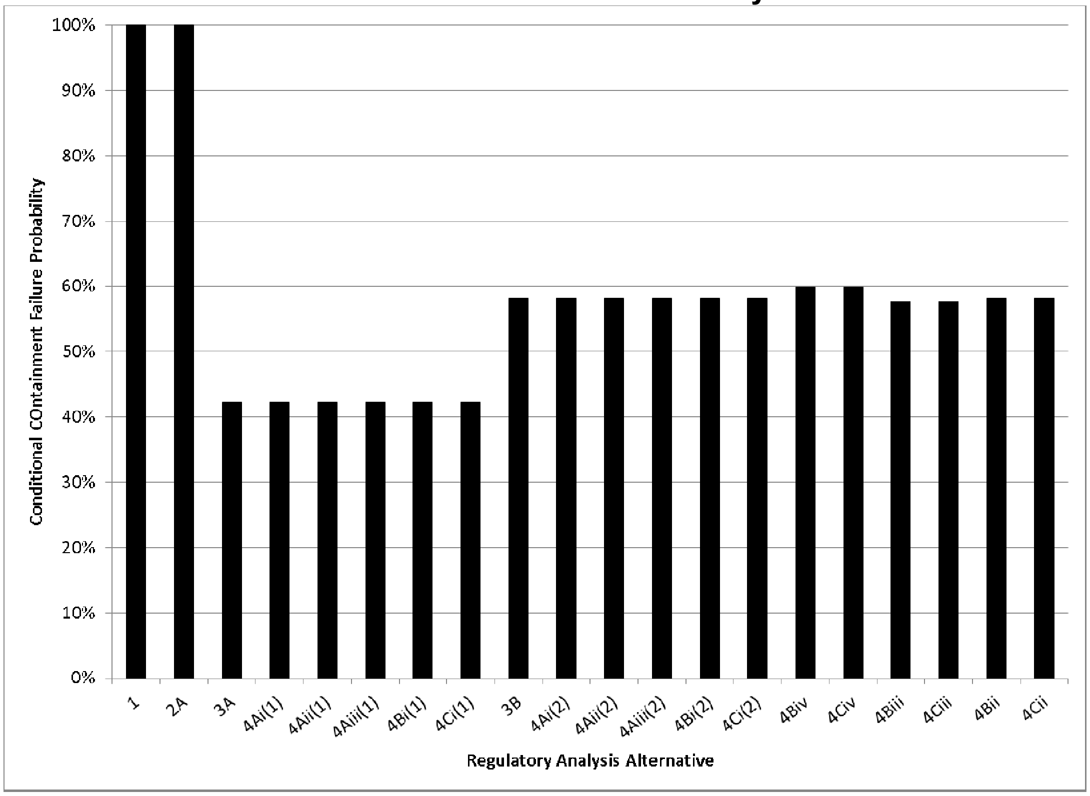
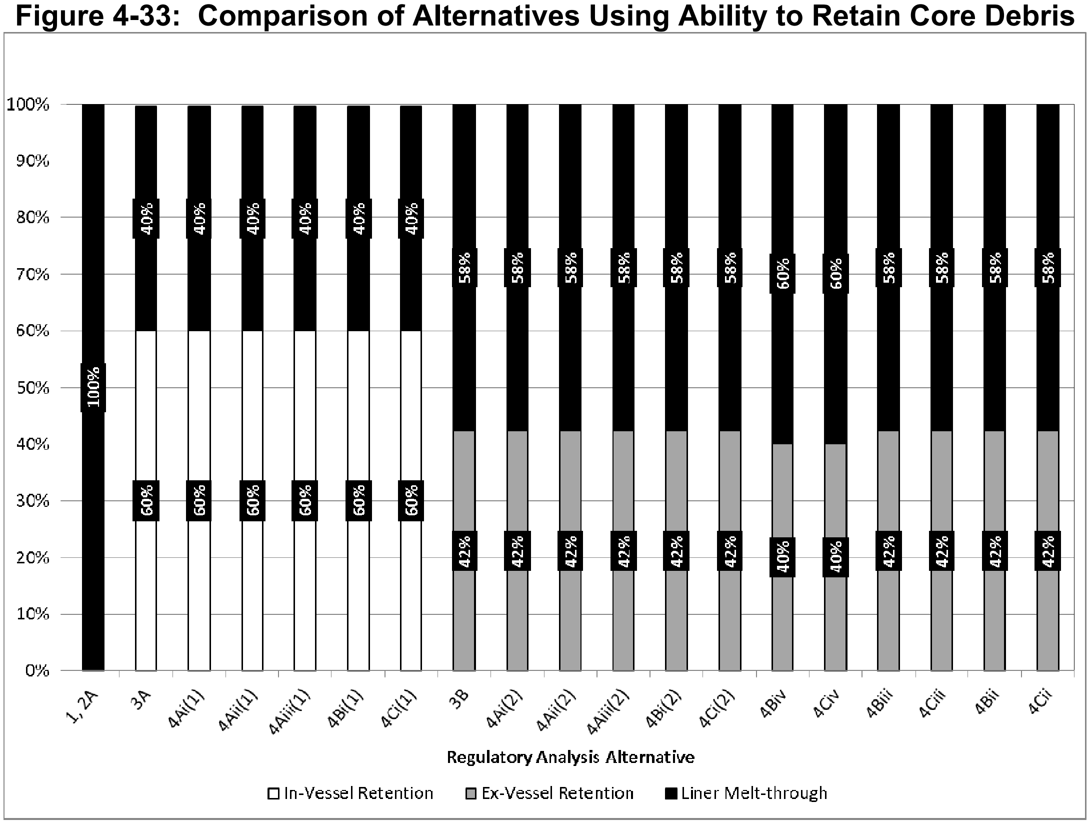


Figure 4-33: Comparison of Alternatives Using Ability to Retain Core Debris



Other important insights from the risk evaluation:

- 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.

- 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 is also significant.

- Over a reasonable range of values, CDF is not particularly sensitive to the human error probabilities for in-control-room and ex-control-room operator actions.

- The 5 percent/95 percent parametric uncertainty interval of the estimated risks is more than one order of magnitude, and is largely driven by uncertainty in the seismic hazard curves.

- The evaluation assumed that 60 percent of the time the pre-core-damage water addition (FLEX) will be successful in preventing core damage. This assumption is informed by the results of the risk evaluation, which used scoping estimates of human error probabilities, and the NRC review of licensees' mitigating strategies, including plant walkdowns. Half of the failure probability is due to failure to implement needed operator actions and the remaining half is due to equipment failures.

- If core damage occurs, there would be a release due to containment venting and/or containment failure caused by over-pressurization or liner melt-through for all CPRR alternatives. However, the estimated mean individual latent cancer fatality risk (0-10 miles) is more than two orders of magnitude below the relevant NRC Safety Goal Quantitative Health Objective. The risk is low because the core-damage frequency is low and the conditional latent cancer fatality risk is low.

4.6.3 Key Insights from MELCOR Analyses

Both water addition and venting are required to maintain containment integrity and reduce source terms. As discussed above, challenges related to operator actions and possible equipment failures result in the Mark I containment being estimated to fail almost half the time in the event of an ELAP. Additional capabilities related to protecting containment and cooling core debris were evaluated. The post-core damage water addition is a SAMG action. As previously stated in Section 2.3, "Order EA-13-109," the staff is proposing in the Mitigation of Beyond-Design-Basis Events rulemaking to require SAMGs for all licensees. In addition, the industry is proposing water addition as their response to Phase 2 of Order EA-13-109.

The MELCOR analysis led to several important findings, many of which relate to operator actions under severe accident conditions. Vent cycling is found to have very little beneficial effect on release reduction but does support containment protection. There is hardly any difference between water addition (SAWA) and water management (SAWM) with regard to their respective effects on release reduction but both are helpful in reducing the offsite release due to a scrubbing effect. Vent cycling and SAWM do require operator actions under severe accident conditions.

The MELCOR analysis also found that RCIC suction initially taken from the condensate storage tank (CST) provides a better alternative to suction from suppression pool (SP) as this action will

likely extend the RCIC duration. However, this requires the CST to be at least "seismically robust" (not necessarily seismic Category I) to assure its availability and functionality during a beyond design basis seismic event.

Operator actions to prevent or mitigate severe accidents are contingent on the availability and functionality of equipment and diagnostic instruments under severe accident conditions. The MELCOR analysis provides insights on the timeline for such actions. The SRM to SECY-12-0157 mentions consideration of equipment availability as one of several performance measures. The impacts of equipment availability can be quantitatively measured by the conditional uncontrolled release index (CURI) which is reviewed in Section 5, "Performance Criteria Information". The operator relies on instruments to know when to add water and/or to take other accident management actions. Therefore, instrument availability and reliability play an important role in this respect. In the PRA done as part of this evaluation, FLEX was assumed to be 60 percent successful. In the accident progression analysis using MELCOR, instruments measuring the RPV and containment water levels and pressures were assumed to be available. Note that 10 CFR 50.34(f)(2)(xix) requires licensees to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

Other important insights from the MELCOR analyses are as follows:

- Containment venting is generally assumed to occur at the Primary Containment Pressure Limit (PCPL), which is after the onset of core damage but before vessel failure. The venting leads to a puff release to the environment.

- For most of the cases considered, water addition is assumed to occur just after vessel failure. If water addition is assumed to occur before venting, then releases to the environment would be reduced and vessel failure would be averted.

- Anticipatory venting (before core damage) is beneficial to reduce the containment pressure and delay the radionuclide release to the environment.

- Containment venting is efficient in purging hydrogen and non-condensable gases. Water injection is also helpful in maintaining an inert atmosphere where steam can preclude energetic hydrogen combustion.

- Depressurizing the RPV below 200 psia after RCIC failure allows for the possibility of water injection into the RPV before vessel failure and reduces the release to the environment.

- The highest calculated release to the environment results from a main steam line creep rupture scenario, which is one of the least likely scenarios.

- The release to the environment calculated in the Mark II analysis are generally comparable to or lower than those in the Mark I analysis.

- For the Mark II analysis, scoping analyses were performed to investigate different lower cavity configurations. The environmental releases are within the range of source terms predicted based on the variations in the scenario boundary conditions.

4.6.4 Key Insights from MACCS Analyses

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, there is zero early fatality risk 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. 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.

All of the source terms begin their release to the environment long enough after the accident's initiation to allow time for the 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). Because the habitability criterion acts as a threshold to limit exposures and health risk, the ILCF risk often has a nonlinear dependence on source term magnitude. A large source term always necessitates more extensive protective actions than a smaller source term (assuming all other conditions are identical), however the increase in ILCF risk may be negligible if the smaller source term was already sufficient to exceed the habitability criterion. Thus 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.

When comparing conditional consequences for the Mark I and Mark II containment analyses (see Figures 4-25 and 4-26), relative to the status quo, water addition (SAWA/SAWM) results in a notable reduction in offsite consequences in terms of ILCF risk, population dose, offsite cost, land contamination, and population subject to long-term phase protective actions. For the Mark I analysis, relative to the status quo, the combination of water addition with an external filter results in a greater reduction in the same offsite consequences. The CPRR alternatives generally show greater reduction in offsite consequences relative to the status quo for the Mark II analysis compared to the Mark I analysis. This is likely due to the higher population around the Mark II reference site.

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 resulting in an uncontrolled release (e.g., drywell liner melt-through or a main steam line creep rupture), there is less benefit from an external filter since some of the release may bypass the venting system.

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, evacuation timing, a larger non-evacuating population cohort, the intermediate phase duration, and the long-term habitability criterion. Sensitivity calculations do not change any of the consequence analysis insights related to the QHO comparisons. Early fatality risk remains zero for all sensitivity calculations. ILCF risk remains well below the QHO for all sensitivity calculations, even those assuming a larger habitability criterion (e.g., 2 rem per year instead of 500 mrem per year).

5. Performance Criteria Information

In SRM-SECY-12-0157 the Commission directed the staff to evaluate a variety of performance criteria as part of the CPRR rulemaking:

> In the rulemaking technical bases, the staff should evaluate 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), or other measures that may be developed during the stakeholder engagement.

The NRC considered possible performance criteria for the proposed regulatory requirements as well as broader performance criteria that could be used within the deliberative process. It should be noted that the criteria discussed in this Section may differ from existing agency policies and guidance and are provided for information only.

Since the staff prepared SECY-12-0157, the Commission has given the staff additional direction on factors and approaches to consider in the regulatory analysis for this rulemaking. In SRM-SECY-12-0110, the Commission stated "that the NRC's current approach to the issue of land contamination from reactor accidents is sound" and "that economic consequences should not be treated as equivalent in regulatory character to matters of adequate protection of public health and safety." In addition, in the SRM to SECY-13-0132, "U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report," dated May 19, 2014 (ADAMS Accession No. ML14139A104), the Commission disapproved the proposed recommendation to establish a design-basis extension category of events and associated regulatory requirements, and to establish Commission expectations for defense in depth. Instead, the Commission directed that the staff should reevaluate the issue, as appropriate, in the context of the Risk Management Regulatory Framework activities.

Performance Criterion 1: Conditional Containment Failure Probability

The consideration of conditional containment failure probability (CCFP) was discussed in SECY-12-0157 and remains an important factor. CCFP is not a candidate for defining a specific regulatory requirement (i.e., not a practical parameter for use in monitoring facility performance) but can be used to assess the possible safety benefits to containment protection of the CPRR alternatives. The importance of CCFP is closely related to the broader discussions of defense in depth. These factors are especially relevant to the CPRR activity and its evaluation against the established safety goals because the QHO-related analyses results are highly influenced by the estimated core damage frequency. The CCFP for Mark I and Mark II BWR containments has been recognized as being higher than other containment designs due to their smaller available volumes and use of active heat removal systems to support the normal pressure suppression functions. The CCFP was included within the qualitative factors discussed in SECY-12-0157 and was a significant part of the rationale for recommending the addition of severe accident hardened capable vents and filtering strategies.

The analyses supporting the CPRR conclusion (Section 3.2, "Evaluation of Alternatives") were used to assess various alternatives being considered in terms of improving the protection of containment integrity. The combination of containment venting and SAWA/SAWM provide operators with the capability to prevent containment failure. However, the differences between various alternatives for providing these functions are relatively small compared to the overall uncertainties in plant response and analytical results. It is therefore not practical to use calculated CCFP as a definitive performance measure but it is useful to see relative benefits

between the alternatives. The combination of quantitative and qualitative assessments support the NRC's recommended approach to pursue containment protection requirements (i.e., severe accident hardened vents and SAWA/SAWM).

Conditional Uncontrolled Release Index (CURI)

The NRC also developed a supplementary containment protection performance measure to CCFP called the conditional uncontrolled release index (CURI) to assess the potential benefits of the proposed alternatives on containment performance. The CURI is conditional upon the initiation of a beyond-design-basis event, and is a measure of the ability to protect the containment during a severe accident. In the context of the CPRR rulemaking, it is a containment protection parameter. A high CURI value means that there is a high likelihood of a significant release of radionuclides during a severe accident. It is defined as the ratio of the frequency of an uncontrolled release of radioactivity to the environment during a severe accident to the initiating frequency of a beyond-design-basis event. With respect to the CPRR evaluation, the initiating event is an extended loss of ac power (ELAP).

The CURI is similar to the CCFP, which is one of the two containment performance measures used by the NRC in its review of new reactor designs. In NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, it states that the staff should address the containment performance goals identified in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993 (ADAMS Accession No. ML003708021) and SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990 (ADAMS Accession No. ML003707849) as approved by the associated SRMs. Specifically, the staff is directed to address a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges, and a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA. After 24 hours, fission product releases must be controlled. In the reviews of the new BWR certified designs, the NRC considered containment failure to be "an uncontrollable leakage substantially greater than the design basis resulting from loss of containment integrity following the onset of severe core damage," as stated in SECY-90-016. Therefore, containment venting is not considered containment failure for purposes of using the concept of CCFP for new reactors.

For Mark I and Mark II BWRs, there is not sufficient capacity to maintain containment integrity for 24 hours after the onset of an ELAP even if there is no core damage, unless the containment is vented. The potential failure modes include drywell head leakage, failures at penetrations, failures at points where corrosion has compromised structural integrity, and downcomer vent bellows failures (Mark I containments). After core damage and vessel failure, drywell liner melt-through is possible in a Mark I BWR unless external water is added to cool the debris. Because venting is necessary to prevent core damage and maintain containment structural integrity, it is not considered a failure mode. In the context of the CPRR evaluations, the NRC considered a possible measure involving an uncontrolled release with respect to the onset of the ELAP, not from when core damage begins (i.e., CCFP).

Consider the example of venting through the wetwell and injecting water into the RPV of a Mark I BWR after RCIC failure. Assume that a PRA estimates the ELAP frequency to be

$2x10^{-5}$ per year, the core damage frequency following the ELAP is $8.7x10^{-6}$ per year, and frequency of uncontrolled release to the environment (mostly from drywell line melt-through) is $3.7x10^{-6}$ per year. Also assume that, if all pre-core damage operator actions to be carried out during the ELAP are successful, the core damage frequency drops to $5x10^{-6}$ per year and the uncontrolled release frequency drops to $2.9x10^{-6}$ per year. Results of calculating the CCFP are shown in Table 5-1, "Conditional Containment Failure Probability Results".

Table 5-1: Conditional Containment Failure Probability Results

Results	Base Case	All HEP Prior to Core Damage = 0
ELAP frequency (per year)	$2x10^{-5}$	$2x10^{-5}$
Core damage frequency (per year)	$8.7x10^{-6}$	$5x10^{-6}$
Conditional core damage probability (CCDP)	0.434	0.248
Uncontrolled release frequency (per year)	$3.7x10^{-6}$	$2.9x10^{-6}$
CURI (relative to start of ELAP)	0.183	0.142
CCFP (relative to start of core damage)	0.42	0.58

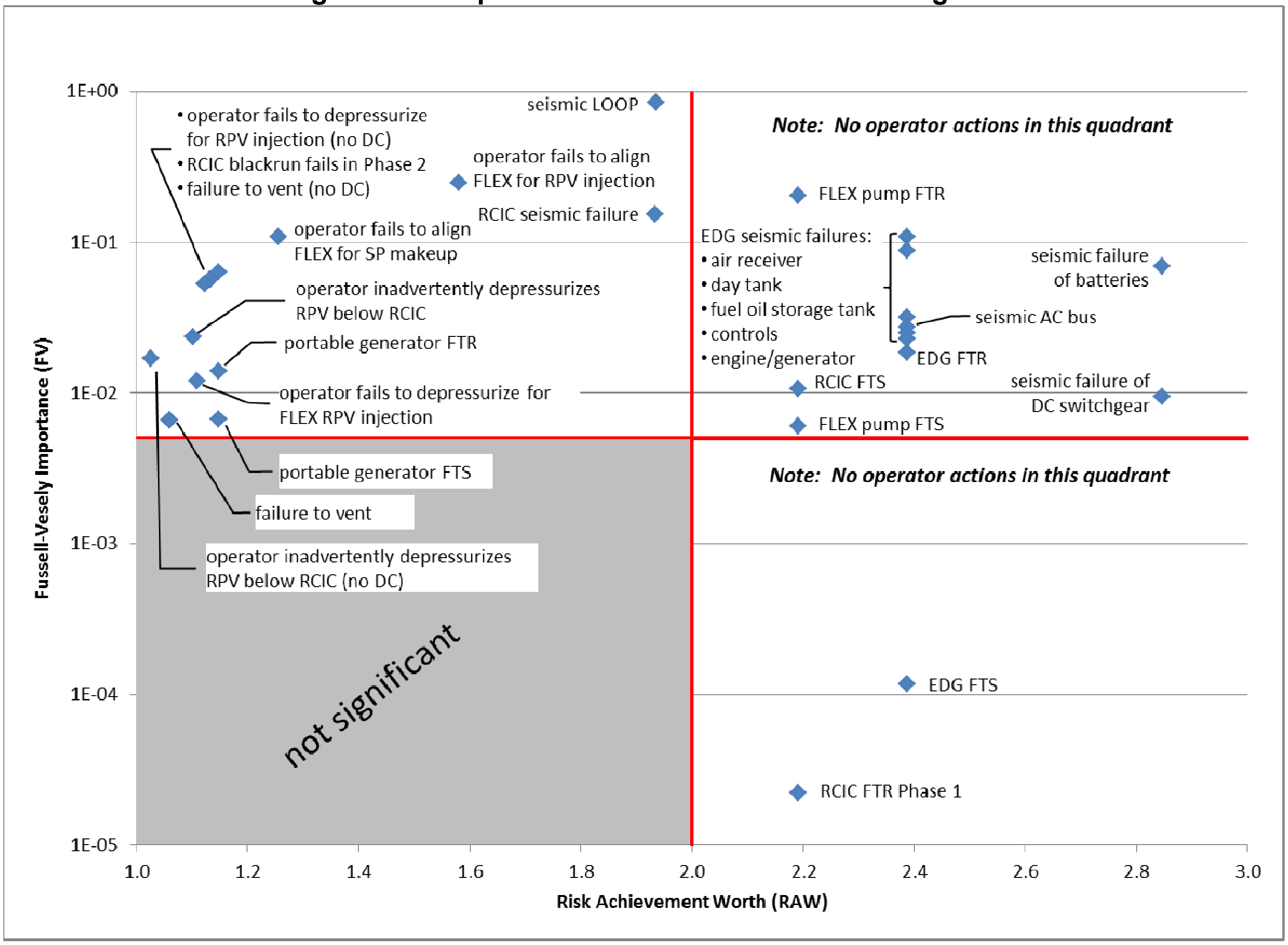
Using the CCFP, which is measured relative to the onset of core damage, can give a different impression of the success of some potential measures to address beyond-design-basis events. For example, one might conclude from Table 5-1 that taking the correct actions before core damage actually makes matters worse (if focusing on CCFP), even with the CCDP being reduced by nearly a factor of two. In fact, for this circumstance, there are only two general ways for core damage to result: 1) the FLEX equipment fails and liner melt-through would result; or 2) RCIC fails early, such that there is not sufficient time to align the FLEX equipment before core damage. In this second way, the operators could succeed in preventing a vessel failure by aligning the FLEX equipment prior to core relocation or, failing this, aligning it while the core debris is still on the lower head prior to vessel failure.

The CURI, however, indicates that there is a very good chance that an uncontrolled release would not result if the FLEX equipment is aligned to inject water into the RPV, either before or after core damage. Moreover, successful operator actions pre-core damage can improve these chances even more. Furthermore, improving the reliabilities of RCIC and the FLEX equipment to withstand the beyond-design-basis event can further improve the situation. Such improvements could decrease the probabilities of RCIC failure to start; failure of the FLEX equipment to run; seismic failures of ac buses, batteries, dc switchgear; etc. In short, the CURI can be a measure of the effectiveness of these improvements in equipment reliability.

For the PRA that produced the values in Table 5-1, Figure 5-1, "Importance Measures for Core Damage", shows the important measures for core damage (and, by extrapolation, for uncontrolled release). Fussell-Vesely Importance (FV) of a modeled plant feature (usually a component, train, or system) is defined as the fractional decrease in total risk level (usually CDF) when the plant feature is assumed perfectly reliable (failure rate = 0.0). The risk achievement worth (RAW) of a plant feature (usually a component, train, or system) is the increase in risk if that feature is assumed to be failed at all times. It is expressed in terms of the ratio of the risk with the event failed to the baseline risk level. The features in the upper right quadrant of Figure 5-1 are the most significant and offer the greatest potential for reducing the CURI. Note that these are all failures of systems or components, and not human errors. The modeled operator actions have relatively low RAW and won't increase the risk much if they have relatively low likelihood of success, although some of them, such as successfully aligning of the FLEX equipment for RPV injection or successfully venting when there is no dc power, can significantly reduce the risk if they can be carried out with high reliability. Therefore, evaluating

equipment availability and reliability, and human actions, in the context of CURI, can help quantify increased containment protection.

Figure 5-1: Importance Measures for Core Damage



Performance Criterion 2: **Decontamination Factor**

SECY-12-0157 discusses the possible use of decontamination factors as performance criteria for a regulatory requirement in combination with several possible decision-making criteria (e.g., extent of land contamination, population dose, economic consequences, and available technologies). The use of a decontamination factor as a performance measure would most logically be associated with the release reduction aspects of the CPRR evaluations. In SECY-12-0157, one potential approach was based on a required decontamination factor for the available combination of plant systems, such as RPV or DW sprays, the suppression pool, the reactor building, and, if necessary, an engineered filter. Another approach stated in SECY-12-0157 is to limit the release of radioactive materials as low as reasonably achievable using currently available filtering technologies. The NRC performed the technical analyses described in Section 4, "Technical Evaluation," and performed sensitivity studies on the potential value of strategies or engineered filters with a variety of assumed decontamination factors. The NRC concluded that filtering strategies and engineered filters would not meet the thresholds for a substantial safety enhancement no matter what efficiency was assumed. The NRC did not further define possible criteria related to decontamination factors because filtering strategies or

engineered filters (alternative 4) is not expected to be included in the proposed CPRR rulemaking.

The rulemaking is expected to adopt alternative 3, which involves additional measures, like SAWA/SAWM, to protect containment integrity and reduce uncontrolled releases. This alternative provides capabilities to protect the containment from over-pressure and over-temperature conditions, and to cool core debris. These capabilities are not effectively addressed by a performance measure related to decontamination factors.

Performance Criterion 3: **Equipment and Procedure Availability**

A performance-based approach discussed in SECY-12-0157 involves developing strategies or contingencies and not defining in regulations specific requirements for individual structures, systems, or components (e.g., the aircraft impact assessment rule in 10 CFR 50.150 and the loss of large area requirements defined in 10 CFR 50.54(hh)). This general approach has been used to define the requirements for Order EA-12-049 for mitigating strategies for beyond-design-basis external events and Order EA-13-109. This approach logically applies to alternative 3 since it is directly related to those orders and the NRC expects to use the analyses and guidance documents developed for Order EA-13-109 to support the CPRR rulemaking. The NRC is likely to adopt a performance measure and language similar to Order EA-13-109 and related guidance documents when developing the rule language for the proposed CPRR regulation.

Performance Criterion 4: **Total Population Dose**

The radiation dose to the public following an assumed severe accident is considered within the NRC's assessments of proposed regulatory actions. However, the dose calculations are considered along with other factors within the analysis of benefits and costs of the actions, which are presented in terms of dollars. Population dose provides a possible alternative way to characterize the societal consequences of an accident. This metric sums the doses from all exposure pathways and multiplies them by the size of the population that would be expected to receive them for the calculated time duration. Unlike the ILCF risk, this metric is directly dependent on population and therefore consequences would generally be higher for plants near cities and lower for plants located in more rural areas. The total population dose is also very dependent on assumptions used for allowing evacuated populations to return to their homes (i.e., habitability criteria).

Performance Criterion 5: **Margin to the Quantitative Health Objectives**

The safety goals and related QHOs were developed to assess aggregate risks and are used to help make decisions on rulemakings or other major agency actions. It is necessary to keep this in mind when using the QHOs to evaluate specific issues or plant specific concerns. Results from Section 4, "Technical Evaluation," associated with assessing the CPRR alternatives were often put into the context of the "margin to the QHOs," which is appropriate when evaluating specific initiating events or issues.

The NRC presents the analytical results related to a conservative representation or estimate for the potential risks and comparison to the QHOs (see Section 3.1, "Safety Goal Screening Evaluation" and Figure 3-2, "Relative Latent Cancer Fatality Risk"). The risks are comparable to those associated with releases from spent fuel pool events (COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel") for which the Commission determined that no additional regulatory actions were warranted. Some discussions during the CPRR-related assessments included the

possibility of defining specific performance criteria regarding the margin (e.g., orders of magnitude) between the calculated risks and the QHOs.

Performance Criterion 6: Long Term Relocation

Another parameter related to the amount of radioactive material released from a severe accident and the resultant contamination of areas near nuclear power plants involves measures of the number of people displaced or relocated. The technical analysis in Section 4, "Technical Evaluation," presents the results in terms of the number of people that would need long term (greater than one year) relocation following a severe accident at a nuclear power plant. Limiting the relocation of populations is discussed in some international venues as a goal (but not necessarily a defined performance measure) of post-Fukushima activities. As with total population dose, this measure is heavily dependent on local population densities and the habitability criteria.

5.1 Performance Criteria Conclusion

As directed by the SRM to SECY-12-0157, the NRC has identified and evaluated several possible performance measures as discussed above but did not identify a potential regulatory requirement that would provide a cost-justified substantial safety improvement. The use of the above performance criteria could involve significant changes to existing NRC policies and practices, and may not be consistent with the broader guidance issued for federal agencies via executive orders and guidance from the Office of Management and Budget. However, various societal measures and performance criteria such as those discussed within this Section are sometimes discussed in international meetings and academic papers. These discussions can help put such discussions of different societal measures into context and help explain some differences in decisions on engineered filters taken by other regulatory bodies.

6. Impact Analysis

6.1 Impact on Licensees

The impact of this proposed rulemaking on licensees of BWRs with Mark I and Mark II containments will be determined by the regulatory requirements actually incorporated into the NRC's regulations. Based on stakeholder feedback, the NRC expects that the impacts of the rulemaking to pursue alternative 3 would be limited because licensees will be making the associated plant and procedure changes as part of their implementation of Order EA-13-109, Phases 1 (severe accident hardened vents) and 2 (severe accident water addition/management).

Guidance for licensees developing and implementing overall integrated plans related to Phase 1 of Order EA-13-109 is provided in Interim Staff Guidance (ISG) JLD-ISG-2013-02, "Compliance with Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated November 14, 2013 (ADAMS Accession No. ML13304B836), which endorses, with exceptions and clarifications, the methodologies described in the industry guidance document Nuclear Energy Institute (NEI) 13-02, "Industry Guidance for Compliance with Order EA-13-109," Revision 0 (ADAMS Accession No. ML13316A853).

The approach to address Phase 2 of Order EA-13-109 was developed in coordination with the technical analyses and discussions conducted to support the CPRR rulemaking. An approach developed by the industry for Phase 2 of Order EA-13-109 is described in ongoing revisions to NEI 13-02 and in the letter from NEI dated September 10, 2014 (ADAMS Accession No. ML14259A186). The industry's current approach for implementation of Phase 2 of Order EA-13-109 would have licensees incorporate a SAWA capability either as part of a design that allows for severe accident drywell venting or as part of a severe accident water management (SAWM) strategy. In the case of the SAWM strategy, licensees control the water levels in the suppression pool during severe accident water addition and thereby make it unlikely that drywell venting would be needed during severe accident conditions. The inclusion of SAWA/SAWM as an element of implementing Phase 2 of Order EA-13-109 provides a comprehensive means to improve overall severe accident management for Mark I and II BWR containments and it also provides the capability to address other containment failure modes (e.g., over-temperature and liner melt-through) without significant additional costs to licensees.

The NRC has concluded that alternative 3 could be supported by the analyses developed for this regulatory basis and for the implementation of Order EA-13-109. The costs incurred by licensees' participation in the alternative 3 rulemaking and other administrative costs are expected to be minimal because these actions will be part of the implementation of Phase 2 of Order EA-13-109.

6.2 Applicability

A CPRR rule would be intended for all holders of or applicants for an operating license under 10 CFR Part 50 with BWR Mark I and Mark II containments, except those: 1) Part 50 licensees who have permanently ceased operations and 2) who certified that fuel has been permanently removed from the reactor vessel.

6.3 Implementation Plan and Development of Supporting Guidance

To provide guidance for the industry on an approved approach to meeting the requirements of a new rule, the NRC plans to work with the industry and public to create supporting draft

guidance, most likely in the form of a draft Regulatory Guide, concurrent with the development of the proposed rule per the goals of the cumulative effects of regulation. The NRC anticipates that new technical guidelines will be developed with input from the industry owners' groups, and NEI will submit an industry-recommended approach to the rulemaking concurrently with this rulemaking effort. The NRC intends to evaluate the approach developed by the industry when developing the Regulatory Guide. The draft Regulatory Guide will be published for public comment at the same time as the proposed rule. The final Regulatory Guide will be published with the final CPRR rule.

The NRC will also develop a NUREG that will document the technical analysis performed to support the CPRR activities summarized in this document. The draft NUREG will be published for public comment at the same time as the proposed rule. The final NUREG will be published with the final CPRR rule.

In addition to the input provided by the industry, the NRC will formally respond to public comments received during the proposed rule phase of the rulemaking to ensure that all stakeholders have input into the rulemaking process and the development of the corresponding draft Regulatory Guide. The NRC's goal is to develop a final rule that efficiently and effectively addresses all regulatory concerns, with adequate supporting guidance, and considers all stakeholder input.

7. Conclusion

The NRC finds that a continued rulemaking effort is justified for containment protection and release reduction. The regulatory and technical analysis in this area identified opportunities for regulatory improvement that could further reduce the risk to public health and safety in the event of an accident similar in scale to that experienced at Fukushima Dai-ichi. The NRC proposes to continue the rulemaking effort with alternative 3 from Section 2.4. Alternative 3 would establish new requirements within the existing framework for severe accident conditions for BWR Mark I and Mark II containments by making generically applicable the requirements of Order EA-13-109 and establish requirements for severe accident water addition and management.

Appendix A: Stakeholder Involvement

The NRC involved stakeholders in developing this regulatory basis. These efforts to meet with external stakeholders are consistent with the intent of the formal Cumulative Effects of Regulation (CER) requirements promulgated by SRM-SECY-0032, "Consideration of the Cumulative Effects of Regulation in the Rulemaking Process," dated October 11, 2011 (ADAMS Accession No. ML112840466).

The NRC held thirteen public meetings and two ACRS subcommittee meetings between June 2013 and December 2014. The meetings attracted members of the public, representatives from NEI, EPRI, and industry owners groups, and non-government organizations (NGOs). Information about each meeting and the meeting summaries are describe in the table below.

Meeting Date	ADAMS No.
June 13, 2013	ML13199A216
June 26, 2013	ML13203A074
July 11, 2013	ML13211A395
August 14, 2013	ML13238A328
September 19, 2013	ML13277A332
November 6, 2013	ML13324A953
December 12, 2013	ML13357A794
March 26-27, 2014	ML14093A098
April 30, 2014	ML14136A292
June 18-19, 2014	ML14176B132
August 21, 2014	ML14279A343
October 14, 2014	ML14317A060
December 11, 2014	ML15008A002

As this rulemaking effort continues, the next opportunity for public comment will be during the proposed rule phase. During the proposed rule phase, the NRC will solicit input from stakeholders on the draft rule language and supporting guidance.

Appendix B: Other Regulatory Considerations

Environmental Analysis

During the proposed rule phase, the proposed rule language will be analyzed for its potential effects on the environment. The NRC does not anticipate that this rule will have any negative impact on the environment.

Impact on State, Local, or Tribal Governments

The CPRR rulemaking should not affect state and local government resources during a severe accident exercise because it will address onsite activities only. In addition, it is unlikely that any drill or exercise that is extended to meet any requirement for severe accident exercises would require extended participation by state and local governments.

Impact on the NRC

A new CPRR rule may require inspection resources from the regional NRC offices to support follow-on inspections of licensee programs associated with accident mitigating procedures, training, and exercises. The expansion of requirements for exercises would require additional hours from resident inspectors, regional inspectors and potentially emergency response organizations at the regional offices and will be documented in the CPRR regulatory analysis during the proposed rule phase.

Appendix C: List of Acronyms

ACRS	Advisory Committee for Reactor Safeguards
APET	Accident Progression Event Tree
AV	Anticipatory Venting
BEA	U.S. Bureau of Economic Analysis
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CCFP	Conditional Containment Failure Probability
CDET	Core Damage Event Tree
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CPIP	Containment Performance Improvement Program
CPRR	Containment Protection and Release Reduction
CRF	Consequence Reduction Factor
DF	Decontamination Factor
DW	Drywell Vent
ELAP	Extended Loss of AC Power
EOP	Emergency Operating Procedures
EPA	U.S. Environmental Protection Agency
EPG	Emergency Procedure Guideline
EPRI	Electric Power Research Institute
EPS	Emergency Power System
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
FLEX	Diverse and Flexible Coping Strategies
FR	Federal Register
FV	Fussell-Vesely
HCVS	Hardened Containment Venting System
HEP	Human Error Probability
HFE	Human Failure Event
HRA	Human Reliability Analysis
IDHEAS	Integrated Decision-tree Human Event Analysis System
ILCF	Individual Latent Cancer Fatality
ISG	Interim Staff Guidance
IVR	In-Vessel Retention
LOCA	Loss of Coolant Accident
LOP	Loss of Offsite Power
LNT	Linear-No-Threshold
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
MELCOR	A computer code whose primary purpose is to model the progression of accidents in nuclear reactors
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NUREG	Technical Report Prepared by the NRC
PCPL	Primary Containment Pressure Limit
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment

Appendix C: List of Acronyms

PSP	Pressure Suppression Pressure
QHO	Quantitative Health Objectives from the NRC's Safety Goal Policy Statement
r-y	Reactor Year
RAW	Risk Achievement Worth
RC	Release Category
RCIC	Reactor Core Isolation Cooling
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SA	Severe Accident-Capable
SAG	Severe Accident Guideline
SAMG	Severe Accident Management Guideline
SAWA	Severe Accident Water Addition
SAWM	Severe Accident Water Management
SBO	Station Blackout
SECY	Office of the Secretary
SOARCA	State-of-the-Art Reactor Consequence Analyses
SP	Suppression Pool
SPAR	Standardized Plant Analysis of Risk
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
USDA	U.S. Department of Agriculture
WW	Wetwell Vent

Appendix D: References

Date	Document	ADAMS Accession Number
May 25, 1988	SECY-88-147, “Integration Plan for Closure of Severe Accident Issues”	ML12250A921
January 23, 1989	SECY-89-017, “Mark I Containment Performance Improvement Program”	ML12251A419
January 12, 1990	SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements”	ML003707849
April 2, 1993	SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”	ML003708021
November 8, 2006	SRM-M061020, "Meeting with Advisory Committee on Reactor Safeguards"	ML063120582
February 18, 2009	SRM-M090204B, "Staff Requirements - Briefing on Risk-Informed, Performance-Based Regulation"	ML090490812
October 3, 2011	SECY-11-0137, “Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned”	ML11269A204
October 11, 2011	SRM-SECY-0032, “Consideration of the Cumulative Effects of Regulation in the Rulemaking Process”	ML112840466
December 15, 2011	SRM-SECY-11-0137, “Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned”	ML113490055
February 7, 2012	SRM-SECY-11-0172, "Response to Staff Requirements Memorandum COMGEA-11-0001, ‘Utilization of Expert Judgment in Regulatory Decision Making’"	ML120380251
February 17, 2012	SECY-12-0025, “Proposed Orders and Requests for Information in Response to Lessons Learned from Japan’s March 11, 2011, Great Tohoku Earthquake and Tsunami”	ML12039A111
March 12, 2012	Order EA-12-049, “Issuance of Order to Modify Licenses With Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events”	ML12054A735

Date	Document	ADAMS Accession Number
March 12, 2012	Order EA-12-050, “Issuance of Order to Modify Licenses With Regard to Reliable Hardened Containment Vents”	ML12054A694
November 26, 2012	SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments”	ML12325A704
January 25, 2013	G20130080/LTR-13-0075 - Anthony R. Pietrangelo Ltr. re: “Filtering Strategies and Filtered Vents”	ML13030A145
January 29, 2013	SRM-M130109B, "Staff Requirements-January 9, 2013, Briefing on Venting Systems for Mark I and Mark II Containments"	ML13029A516
March 19, 2013	SRM-SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments”	ML13078A017
March 20, 2013	SRM-SECY-12-0110, “Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission’s Regulatory Framework”	ML13079A055
June 6, 2013	Order EA-13-109, “Issuance of Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions”	ML13143A321
November 12, 2013	Nuclear Energy Institute 13-02, Revision 0, “Industry Guidance for Compliance with Order EA-13-109”	ML13316A853
November 14, 2013	Interim Staff Guidance JLD-ISG-2013-02, “Compliance with Order EA-13-109, Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions"	ML13304B836
November 25, 2013	COMSECY-13-0030, “Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel”	ML13329A918
December 6, 2013	SECY-13-0132, “U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report”	ML13277A413

Date	Document	ADAMS Accession Number
May 19, 2014	SRM-SECY-13-0132, "U.S. Nuclear Regulatory Commission Staff Recommendation for the Disposition of Recommendation 1 of the Near-Term Task Force Report"	ML14139A104
May 23, 2014	SRM-COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel"	ML14143A360
September 10, 2014	Letter to Jack R. Davis (NRC) from Steven P. Kraft (NEI), "Compliance with Phase 2 of NRC Order to Modify the Licenses of Boiling Water Reactors (BWRs) with Mark I and II Containments with Regard to Reliable Hardened Containment Vents Capable of Operation Under Sever Accident Conditions (EA-13-109)"	ML14259A186
December 10, 2014	Letter to Anthony R. Pietrangelo (NEI) from William M. Dean (NRC), NRC response to a request to endorse a proposal dated September 10, 2014, submitted by the Nuclear Energy Institute (NEI) on behalf of the nuclear industry.	ML14343A818
March 4, 2015	SRM-SECY-14-0087, "Qualitative Consideration of Factors in the Development of Regulatory Analyses and Backfit Analyses"	ML15063A568