POLICY ISSUE
NOTATION VOTE

November 26, 2012 SECY-12-0157

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: CONSIDERATION OF ADDITIONAL REQUIREMENTS FOR CONTAINMENT VENTING SYSTEMS FOR BOILING WATER REACTORS WITH MARK I AND MARK II CONTAINMENTS

PURPOSE:

The purpose of this paper is to provide the U.S. Nuclear Regulatory Commission (NRC) with information, options, and a recommendation from the NRC staff to impose new requirements for containment venting systems for boiling-water reactors (BWRs) with Mark I and Mark II containments. This paper is provided in response to the Commission's staff requirements memorandum (SRM) for SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated December 15, 2011.

SUMMARY:

As directed by the Commission, the NRC staff evaluated the addition of filtered containment venting systems to BWRs with Mark I and Mark II containments to address lessons learned from the events at the Fukushima Dai-ichi nuclear accident in Japan. Specifically, the options presented include: (1) the status quo including completing requirements established for reliable hardened vents; (2) issuance of Orders requiring containment venting systems capable of operating under severe accident conditions; (3) issuance of Orders requiring containment venting systems capable of operating under severe accident conditions that have filters within

CONTACT: David L. Skeen, NRR/JLD
301-415-3091

the controlled release pathways; and (4) developing a severe accident confinement strategy for BWRS with Mark I and Mark II containments. The NRC 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. Several public meetings and other interactions with stakeholders, including the NRC's Advisory Committee on Reactor Safeguards (ACRS), informed the NRC staff's assessments. The evaluation of options used existing NRC processes and addressed possible updates to associated regulatory guidance and insights from the Fukushima Dai-ichi accident.

Based on its regulatory analyses, the staff concludes that installation of engineered filtered venting systems for Mark I and Mark II containments is the option that would provide the most regulatory certainty and the timeliest implementation. The vast majority of Mark I and Mark II severe accident sequences would benefit from a containment vent, (whether the vent includes an engineered filter or not) and the addition of an engineered filter reduces the release of radioactive materials should a severe accident occur. 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, when qualitative factors such as the importance of containment systems within the NRC's defense-in-depth philosophy are considered, as is consistent with Commission direction, a decision to require the installation of engineered filtered vent systems is justified.

BACKGROUND:

The accident at the Fukushima Dai-ichi nuclear facility in Japan highlighted the need for safety improvements for nuclear power plants related to beyond-design-basis natural hazards and the resulting effects on plant systems and barriers from an extended loss of electrical power and access to heat removal systems. In SECY-11-0137, "Prioritization of Recommended Actions To Be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the NRC staff described its proposals for the regulatory actions to address the recommendations of the Fukushima Near-Term Task Force (NTTF). Among the immediate (Tier 1) actions that the NRC staff proposed was the issuance of orders requiring reliable hardened containment vents for those licensees with BWR facilities with Mark I and Mark II containment designs. Ensuring the availability of reliable, hardened containment vents addresses some of the problems encountered during the Fukushima Dai-ichi accident by providing plant operators with improved methods to vent containments during beyond-design-basis accidents (but before core melt). Venting containment can help prevent or delay the loss of, or facilitate recovery of, important safety functions such as reactor core cooling, reactor coolant inventory control, containment cooling, and containment pressure control. The NRC subsequently issued orders requiring reliable hardened vents for these plants on March 12, 2012. The NRC staff identified an additional issue in SECY-11-0137 related to possible upgrading of the containment vents, including the addition of engineered filters, to improve reliability during severe accident conditions and limit the release of radioactive materials if the venting systems were used after significant core damage had occurred.

In the staff requirements memorandum (SRM) for SECY-11-0137, dated December 15, 2011, the Commission directed the NRC staff as follows:

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

In response to the SRM, the staff included plans to address the filtered venting issue for Mark I and Mark II containments in 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," dated February 17, 2012. The staff explained the proposed evaluations and need for timely consideration as follows:

> The staff has determined that some of the additional issues should be included in existing Tier 1 activities. In accordance with the direction in SRM-SECY-11-0137, the additional issue of filtration of containment vents was merged with the Tier 1 issue of hardened vents for Mark I and Mark II containments such that further analysis and interaction with stakeholders will inform whether filtered vents should be required. The staff has determined that consideration of severe accident conditions in the design and operation of the vent, the addition of filters to hardened reliable vents, and consideration of vents in areas other than primary containment, will be the topic of a policy paper to the Commission in July 2012.[1]
>
> The staff believes that the requirements for hardened reliable vents in the proposed order are important to ensure core and containment cooling, and that these requirements should be imposed before the staff completes its evaluation of the technical and policy issues associated with imposing additional requirements, as described above. In public meetings, the staff has encouraged licensees to consider the potential for the later addition of filters. However, the industry has stated that the addition of filters to hardened containment vents may require modifications to the vent design. In light of this, a consideration in the staff's proposal to issue the proposed order now is that the proposed order requires submission of integrated plans for implementing the requirements of the order by February 28, 2013, eight months after the staff plans to send the July 2012 policy paper to the Commission for consideration. As a result, licensees should have time to revise draft plans in response to any new Commission direction before the integrated implementation plans are due.

In SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012, the staff described the current course of action as follows:

> One of the six additional recommendations identified in SECY-11-0137, and further developed in SECY-12-0025, was consideration of additional performance

[1] The schedule for this paper was subsequently extended to November 30, 2012, in a memorandum dated August 6, 2012, "Staff Requirements - COMSECY-12-0014 - Revised Schedule and Plans for Japan Lessons- Learned."

> requirements, including filters, for hardened containment vent systems for boiling-water reactor Mark I and Mark II containment designs. In SECY-12-0025, the staff explained that it needed to resolve technical and policy issues before regulatory action could be proposed that would require licensees to install filters, or change any other performance requirement, for hardened containment vent systems. The staff's recommendation on additional performance requirements for containment vents will be provided in a separate paper.

On August 7, 2012, the staff briefed the Commission on the status of actions taken in response to lessons learned from the Fukushima Dai-ichi accident. In the resulting SRM, "Briefing on the Status of Lessons Learned from the Fukushima Dai-ichi Accident (M120807B)," dated August 24, 2012, the Commission provided the following direction to the staff:

> In the forthcoming notation vote paper on filtered vents, the staff should include a discussion of accident sequences where the filters are and are not beneficial.

This paper provides the staff's assessment and recommendation on the installation of filtered vents and provides a discussion of those accident sequences in which the filters are both beneficial and nonbeneficial.

DISCUSSION:

A key element of the design of nuclear power plants is the inclusion of multiple barriers to prevent or contain potential release of radioactive materials created within the fuel by the fission process. In the United States, multiple structural barriers always have been required to confine the fission products to the plant should an accident lead to a compromise of one or more of the barriers provided by the fuel design, the reactor coolant pressure boundary, and the containment.

For currently operating plants, the design of the containment barrier provides either (1) a large enough air volume to accommodate 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 by condensing steam and thereby suppressing the increase in pressure to values below the design pressure for the containment. BWRs employ such pressure suppression containment designs. Mark I and Mark II containments are specific containment configurations for BWRs that use water suppression pools to condense the steam released from the reactor following a LOCA or other plant transients or accidents. As a result of the heat capacity of a suppression pool (i.e., the ability to condense steam), Mark I and Mark II containments have relatively small free volumes compared to other types of containments (e.g., large dry containments). For additional background information on Mark I and Mark II containments, see Enclosure 2.

Mark I and Mark II containments (as well as other pressure suppression containments) have been shown to be capable of addressing the requirements related to the design-basis accidents that the NRC and its predecessor (Atomic Energy Commission) established for the licensing of currently operating plants. However, various studies (e.g., NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants") and events have shown that the Mark I and Mark II containments do not have the same margins of safety that other containments (e.g., large dry ones) have during accidents that exceed the conditions established by design basis events. These include events that result in an extended addition of energy (i.e., decay heat from the reactor core) to the containment and suppression pool without having available heat removal systems that include pumps and heat exchangers to direct that energy to the ultimate heat sink (e.g., the atmosphere, a nearby river, reservoir), and events that result in the production of significant quantities of noncondensable gases (e.g., hydrogen, carbon monoxide) that are released into the containment. The events at the Fukushima Dai-ichi nuclear power plant involved an extended loss of electrical power and heat-removal systems, resulting in containment pressures that exceeded the containment design pressure. Plant conditions at Fukushima Dai-ichi (e.g., loss of all electrical power or station blackout) hampered the efforts of operators to address the containment overpressure conditions using the installed venting systems, which ultimately contributed to the compromise of all fission product barriers and significant releases of radioactive material. The insights that the NRC gained from Fukushima Dai-ichi on the difficulties in venting the containments led the agency to impose additional requirements for reliable hardened venting systems for plants with Mark I and Mark II containments. It also led the NRC to initiate proposed new regulations for all plants to improve operator readiness to respond to severe accident conditions.

The NRC and nuclear industry have recognized the potential need to vent Mark I and Mark II containment designs to cope with severe accident conditions since at least the early 1980s. In 1983, the NRC approved Revision 2 to the Boiling Water Reactor Owners' Group Emergency Procedure Guidelines, which included guidance for operators to vent Mark I and Mark II containments in response to containment overpressure conditions. The emergency procedure guidelines are used to develop plant specific emergency operating procedures. Though emergency procedures have existed since the 1980s for Mark I and Mark II containment venting systems for beyond-design-basis accidents and severe accidents, the NRC's actions to date, for operating reactors, have not required containment venting systems for Mark I and Mark II containments be designed for severe accident conditions. In keeping with its Severe Accident Policy Statement, the NRC defined in Section 52.79 of Title 10 of the *Code of Federal Regulations* (10 CFR 52.79) requirements for new reactor designs to include in applications " … a description and analysis of design features for the prevention and mitigation of severe accidents."

The NRC has evaluated the possible imposition of such design requirements for operating reactors in previous studies (e.g., the containment performance improvement program (CPIP) in the late 1980s) and has determined that the low probability of such events resulted in the costs of design improvements exceeding the calculated benefits. While the cost/benefit assessment performed by the NRC at that time determined that additional requirements were not cost-justified for Mark I and Mark II containment designs, legislators and regulators in other countries did impose requirements in the aftermath of the accidents at Three Mile Island and Chernobyl. In effect, those other regulatory authorities assessed filtered vents and other severe accident protections with an emphasis on the defense-in-depth argument and with less or no consideration of cost/benefit analyses. A discussion of the requirements in various countries can be found in Enclosure 3. Through interactions with nuclear safety regulators and licensees in other countries and in conducting independent assessments, the staff did not identify any adverse systems interactions or potential negative consequences associated with the installation of filtered containment venting systems.

The performance of existing plant features is an important consideration in evaluating plant behavior under severe accident conditions and possibly adding regulatory requirements to address such conditions. Although not specifically designed to address severe accident conditions, existing plant systems for core cooling, coupled with containment cooling and the suppression pool, can serve to limit the releases of radioactive materials from the plant. Additional plant capabilities and guidelines for accident management have come from previous plant studies and response to events such as the Three Mile Island accident and the terrorist attacks on September 11, 2001. A discussion of the potential capabilities and limitations of existing systems to limit the release of radioactive materials following significant core damage at plants with Mark I or Mark II containments is available in Enclosure 4.

To support deliberations of possible actions related to the performance of Mark I and Mark II containments during severe accidents, the NRC staff, with assistance from Sandia National Laboratories, performed various simulations using the MELCOR and MACCS2 computer codes to evaluate plant response and possible releases from a representative plant assuming various capabilities and configurations. As discussed in Enclosure 5, these simulations provide an assessment of the sensitivity of the plant risks to particular features or parameters. The NRC used lessons-learned and best practices from the recently completed State-of-the-Art Reactor Consequence Analysis (SOARCA) project in conducting the MELCOR and MACCS2 simulations. The simulations in Enclosure 5 were used along with insights from previous studies (e.g., individual plant examinations, NUREG-1150, CPIP, severe accident mitigation alternatives) to help evaluate the potential benefits of features such as revising Mark I and Mark II containment designs to ensure that containment venting systems are capable of working under severe accident conditions and adding engineered filters to the containment venting systems. The technical analysis, discussed in Enclosure 5, includes an assessment of various scenarios to determine those that might benefit from proposed severe accident features, such as engineered filters, and those that would not benefit because the release would bypass such features. In general, the vast majority of Mark I and Mark II severe accident sequences would benefit from a containment vent (whether the vent includes an engineered filter or not). Examples of those sequences for which such vents (with or without engineered filters) would not be beneficial include containment bypass events and intersystem LOCAs, which represent a small fraction of the failure modes for Mark I and Mark II containments. The staff notes that while an engineered filtered vent or a severe accident capable vent (without an engineered filter) is beneficial to many accident sequences, additional measures are needed to provide cooling to core debris released to the containment and to prevent other types of containment failure (e.g., Mark I liner melt through, Mark II suppression pool bypass). The staff's evaluation includes consideration of the need for additional measures to provide core debris cooling.

In addition to its own assessments and analyses, the staff also relied on information gained through interacting with various stakeholders. The nuclear industry provided insights to the NRC staff during several public meetings and through a report that the Electric Power Research Institute prepared. Several nongovernmental organizations and individuals in correspondence and during public meetings provided information to the NRC staff. Enclosure 6 provides additional information on the NRC staff's interactions with external stakeholders.

The NRC staff used the assessments, analyses, and interactions that are discussed above to inform its evaluation of options in Enclosure 1 and the regulatory analysis available in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML12312A456. In evaluating the possible approaches to address the issue of containment venting for BWRs with Mark I and Mark II containments, the staff identified the following options:

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

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

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

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

In evaluating these options, the staff assumed, to the extent practical, the completion of the post-Fukushima Tier 1 items (e.g., implementation of mitigating strategies, reliable hardened containment vents, and integration of accident-related procedures). In its evaluation of the above options, the NRC staff chose not to apply any of the exceptions to the backfit regulations prior to conducting the cost-benefit analysis. The staff proceeded with analyses of the proposed venting modifications as possible cost-justified substantial safety improvements. As stated in Enclosure 1, the staff performed its cost and benefit evaluation using established agency practices for evaluating potential safety enhancements as described in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission." In that evaluation, the NRC staff did not assume any changes in the traditional approaches to imposing safety enhancement requirements or the consideration of economic consequences (e.g., land contamination) from possible severe accidents. The evaluation of options is therefore presented in terms of a comparison of the possible benefits from new requirements and the associated costs of those requirements. The staff's regulatory analysis focuses on Option 2 (severe accident capable vent order) and Option 3 (filtered vent order) as those options involve potential near term regulatory action. Option 4 involves a longer-term effort, and the associated regulatory analysis, which includes a cost/benefit assessment, would be developed once the approach and possible regulatory changes are better defined.

The staff conducted sensitivity studies to evaluate the implications of possible changes to assumptions used in the cost-benefit analysis and major uncertainties in factors, such as event frequencies and consequences. The best-estimate quantitative evaluations, excluding any qualitative factors and sensitivity analysis, indicate that the costs of the proposed actions outweigh the benefits. However, when values from the higher end of the uncertainty bands are assumed for event frequencies or event consequences, the calculated benefits from the proposed options can exceed the estimated costs. The following table provides a summary of the quantitative evaluation from Enclosure 1, showing the sensitivity to event frequency:

Quantitative Cost/Benefit Analysis Per Plant				
	Severe Accident Capable Venting System		Engineered Filtered Venting System	
Total Costs ($k)	(2,027)[1]		(16,127)	
Core Damage Frequency	$2x10^{-5}$/yr	$2x10^{-4}$/yr	$2x10^{-5}$/yr	$2x10^{-4}$/yr
Total Benefits ($k)	938	9,380	1,648	16,480
Net Value (Benefits – Costs)	(1,089)	+7,353	(14,479)	+353

- Total costs include industry and NRC development, implementation, and operating costs.
- Total benefits include averted dose (offsite and occupational assuming $4K/person-rem) and averted property damage (offsite and onsite)

(1) As discussed in Enclosures 1 and 4, the costs for severe accident capable vents for Mark II containment designs will likely be higher than for Mark I plants. The higher cost reflects the likely need to modify the containments to prevent molten core debris in the lower drywell sump drain lines or downcomers from causing a bypass of the suppression pool. Avoidance of suppression pool bypass is needed to make the severe accident capable vents a viable option for the Mark II containment design.

In addition to the analyses discussed above, the NRC staff identified other factors that are not readily represented in quantitative terms but nevertheless warrant consideration in making a decision on possible changes to the performance requirements under severe accident conditions for BWRs with Mark I and Mark II containments. Inclusion of these factors in decisionmaking is consistent with the Commission's guidance on the Backfit Rule in the June 30, 1993, SRM on SECY-93-086, "Backfit Considerations." This guidance is reflected in NUREG/BR-0058. The qualitative factors included in this evaluation include:

- 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; and
- addressing international experience and practices (including availability of technology).

The majority of these qualitative factors, which are discussed in Enclosure 1, provide additional support to pursuing an improved containment venting system for BWRs with Mark I or Mark II containments to address specific design concerns (e.g., high conditional containment failure probability given a core melt); to support severe accident management functions by preventing releases of radioactive materials, hydrogen, and steam into the reactor building or other locations on the site; to minimize the contamination of the site environs; and to reduce the reliance on long term emergency planning for protection of public safety. The summary section of Enclosure 1 provides a discussion of the positive and negative attributes (i.e., pros and cons) of each option with respect to these qualitative factors.

The staff concludes that considering both the quantitative and qualitative factors shows the direct and indirect costs associated with Options 2 and 3 are cost-justified in light of the substantial increase in the overall protection of the public health and safety that is provided by addressing severe accident conditions for BWRs with Mark I and Mark II containments. Option 4 also appears to be justified; however, the staff would need to complete the regulatory analysis once the potential technical requirements were better defined, which, if successful, would likely take several years. The uncertainties, schedules, and resource requirements associated with Option 4 are described in Enclosure 1 and are identified as significant challenges to implementing this approach. The timeliness of developing and implementing Option 4 is a potential issue because the Commission identified the evaluation of filtered vents as a Tier 1 issue, which is reserved for those actions to be initiated without unnecessary delay. Based on the assessments completed this past year, the staff concludes that approaches, such as filtering technologies, currently exist and could be implemented in the near term to resolve issues related to Mark I and Mark II severe accident containment venting. These technologies are technically feasible and have been demonstrated through significant testing and application at nuclear power plants worldwide. Furthermore, the staff concludes that the best solution to address the combination of quantitative and qualitative factors (e.g., providing improved defense in depth) is the installation of passive, engineered filtered venting systems at BWRs with Mark I and Mark II containments.

On June 20, September 5, October 3, October 31, and November 1, 2012, the staff briefed the ACRS on the results of its assessments and evaluations, and the resulting conclusions and recommendations. In a letter dated November 8, 2012, the ACRS provided its own recommendations and views on the staff's recommendations. The ACRS concluded that additional defense in depth measures should be considered for plants with BWR Mark I and Mark II containments and they recommended Option 4. The ACRS noted that severe accident capable vents (Option 2) are an essential part of any controlled venting strategy and that installation of additional engineered filters (Option 3) may be an outcome of the assessments associated with Option 4.[2]

Regarding the need for reliable hardened vents, severe accident capable vents, or engineered filtered containment vents for containment designs other than Mark I and Mark II (e.g., Mark III, ice condenser, and large dry containments), the staff stated in SECY-12-0095 that it would

[2] The ACRS reviewed a draft of this Commission paper in which Option 4 was entitled "Performance-based approach." The NRC staff's internal review and concurrence process led to revisions to the paper, including clarifying the title and descriptions of Option 4 as developing a severe accident confinement strategy for Mark I and Mark II containments. The general proposal and most of the discussions related to Option 4 remain the same as that reviewed and commented on by the ACRS.

revise and develop a program plan with an appropriate schedule and milestones following the Commission's decision on the need for severe accident venting or filtered venting for BWRs with Mark I and Mark II containments. Accordingly, following the Commission's decision and direction on this paper, the staff will revise the program plan and proceed with the evaluation of the technical and safety merits of venting for each particular class of containment designs. The staff noted in SECY-12-0095 that expecting different decisions for each class of containment designs is reasonable. The staff continues to believe this is the case, and will address the specifics for each containment design.

RECOMMENDATION

The staff recommends that the Commission approve Option 3 to require the installation of an engineered filtered containment venting system for BWRs with Mark I and Mark II containments. If the Commission approves Option 3, the staff will engage stakeholders on possible implementation issues (e.g., schedules) related to the draft proposed order provided in Enclosure 7b.[3] Within 60 days of the staff requirements memorandum, the staff will provide the Commission a summary of the stakeholder interactions via a Commission Note and the final order via a Regulatory Notification. The staff would likewise engage stakeholders on the draft proposed order in Enclosure 7a if the Commission were to chose Option 2 or Option 4 with a more immediate requirement to make the containment vents capable of operation during severe accident conditions while relying on existing containment systems to limit possible releases.

RESOURCES

	Option 1		Option 2		Option 3		Option 4		Option 2/4	
	FTE	$K	FTE	$K	FTE	$K	FTE	$K	FTE	$K
FY 2013	2	$500	1	$100	1.5	$100	1.5	$175	1.75	$175
FY 2014	Official Use Only – Sensitive Internal Information									
FY 2014 Unbudgeted										

OUO

NRR has budgeted approximately 2 full-time equivalent (FTE) and $500K in the fiscal year (FY) 2013 current estimate (CE) budget and FTE and $ K in the FY 2014 Performance Budget. If the Commission approves Option 2, 3, or 4, the NRC staff will reallocate additional resources associated with the Fukushima near-term task force tier 3 recommendations during the FY 2015 Planning, Budgeting, and Performance Management process. This reallocation would be less than the 4 FTE or $500,000 that requires Commission approval. Resources beyond 2014 will be addressed during the Planning, Budgeting, and Performance Management process.

[3] It is likely that the draft proposed orders for Options 2 and 3, provided in Enclosure 7a or 7b respectively, will require revision based on interactions with stakeholders and continuing internal discussions on technical or legal issues. If the Commission approves Option 2 or Option 3, the staff will provide the Commisison with a final order via a Regulatory Notification.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. The Office of the Chief Financial Officer has reviewed this paper for resource implications and has concurred.

/RA/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Evaluation of Options
2. Design and Regulatory History
3. Foreign Experience
4. BWR Mark I and Mark II Containment Performance during Severe Accidents
5. Technical Analyses
6. Stakeholder Interactions
7. Draft Proposed Orders

Enclosure 1
Evaluation of Options

CONTENTS

1. INTRODUCTION

The purpose of this enclosure is to describe the various technical and policy evaluations that the U.S. Nuclear Regulatory Commission (NRC) staff conducted to support an integrated decision on the need for additional requirements for severe accident containment venting of boiling-water reactors (BWRs) with Mark I and Mark II containments. Fundamental to this evaluation is the regulatory analysis, which is available in the NRC's Agencywide Documents Access and Management System (ADAMS) at Accession No. ML1212A456. This enclosure provides the results of the NRC staff's development and consideration of various factors, and it summarizes the basis for the staff's recommendations.

The NRC performs regulatory analyses as part of its process in evaluating the merits of imposing new requirements on its licensees. Both NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," describe the methodology and standard assumptions. The methodology includes the consideration of various costs and benefits associated with a possible change in regulatory requirements as well as the considerations of qualitative factors and arguments that are difficult to present in quantitative measures, such as financial costs or averted radiation exposures.

Within the regulatory analysis, several key assumptions and factors are important in evaluating the costs and benefits and representing them in a common term (dollars). The development of NUREG/BR-0184, published in 1997, determined many of these factors. The NRC staff considered updating the regulatory analysis guidance before the Fukushima accident. The accident provided other insights into some of the assumptions in the NRC's approach to performing regulatory analyses. An example of a factor that is subject to change in updating the guidance includes the conversion factor of $2,000 per person-rem for averted radiation exposures. The staff has performed a regulatory analysis of the proposed options (severe accident capable vents order and engineered filtered vents order) using existing guidance. Section 2 of this enclosure summarizes this analysis. To evaluate the possible sensitivity of the regulatory analysis to changes in the standard factors described in existing guidance, the staff provides a summary of a regulatory analysis using revised values for selected assumptions and factors in Section 3 of this enclosure.

1.1 <u>Identification of Options</u>

As discussed in 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," dated February 17, 2012, the staff was asked to evaluate several possible options for revising the severe accident capabilities of BWRs with Mark I and Mark II containments. The possible options evaluated are listed below.

<u>Option 1: Reliable Hardened Vents (Status Quo or Base Case)</u>

Description: Continue with the implementation of Order EA-12-050, "Reliable Hardened Containment Vents," for reliable hardened vents to reduce the likelihood of core damage and failure of BWR Mark I and Mark II containments and take no additional action to improve their ability to operate under severe accident conditions or to require the installation of an engineered filtered vent system.

The base case used in the regulatory analysis is the current fleet of affected boiling-water reactor plants (31 units located at 20 sites with an average remaining license term of 25 years) assuming, to the extent practical, the completion of the post-Fukushima Tier 1 items (e.g., implementation of mitigating strategies, reliable hardened containment vents, and integration of accident-related procedures). There are, however, significant uncertainties associated with the analyses and consequence evaluations related to the base case and the assessment of options. Some examples include the following:

- The frequency and consequences of severe accident conditions (i.e., core damage, hydrogen generation, and containment challenge from high pressures); the experience at Fukushima; current U.S. plant designs and procedures; and planned enhancements to designs and procedures.

- The efficiency of the suppression pool and plant systems (e.g., containment sprays or systems to flood the drywell cavity) in capturing and removing fission products (i.e., providing a decontamination function), which limits the release of radioactive materials to the site environs.

Option 2: Severe Accident Capable Venting System Order (without Filter)

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

This alternative involves upgrading or replacing the reliable hardened vents required by NRC Order EA-12-050 with a venting system designed and installed to remain functional during severe accident conditions (i.e., release of fission products, hydrogen, and high containment pressures and temperatures)[1]. This modification would be pursued to increase confidence in maintaining the containment function following core damage events. Although venting the containment during severe accident conditions could result in a significant release of radioactive materials, the act of venting could prevent gross containment failures that would hamper accident management (e.g., continuing efforts to cool core debris) and result in larger releases of radioactive material.

In addition to ensuring the containment venting system, its supporting equipment, and instrumentation are capable of functioning in severe accident conditions, reviews of plant shielding and other protections for personnel would be required for operation of the vents under harsh conditions. Similar requirements were included in NUREG-0737, “Clarification of TMI [Three Mile Island] Action Plan Requirements,” as Action Item II.B.2, “Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which May be

[1] Varying terms have been used to describe BWR containment venting capability. In accordance with the requirements defined in Order EA-12-050, BWRs with Mark I and Mark II containments shall have a “reliable hardened containment venting system” or HCVS. The HCVS provides improved reliability over the “hardened wetwell vent” or “reliable hardened vents” installed in BWR Mark I containments following the issuance of Generic Letter 89-16, “Installation of a Hardened Wetwell Vent,” but do not specifically address operations during severe accident conditions. Option 2 provides additional requirements for the HCVS ordered by EA-12-050 to ensure reliable operation under severe accident conditions (i.e., following core damage). Under Option 3, the severe accident capable HCVS with an engineered filtration capability is designated as the “filtered containment venting system” or FCVS. The FCVS not only provides the venting function to address overpressure and other conditions within the containment, but also uses an engineered filter to limit the release of radioactive materials.

Used in Post-Accident Operations," and subsequently incorporated into the NRC's standard review plan for nuclear reactors (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"). The TMI action item was developed before the development of severe accident management guidelines (SAMGs) and may not have been performed for some later activities related to responding to severe accidents.

Section 2 provides an analysis of this option, using existing regulatory analysis guidance to determine if the benefits justify the approximate $2 million cost of plant modifications. Section 3 provides a revised analysis to address concerns about the possible need to update or change the regulatory analysis guidance. The NRC staff notes that Option 2 could be pursued as part of an overall severe accident management strategy, in a manner similar to that proposed by the nuclear industry (see letter from the Nuclear Energy Institute (NEI) on "Containment Filtration Strategies for Mitigating Radiological Releases in Severe Accidents for BWR Mark I and Mark II Plants to Reduce the Risk of Land Contamination," dated October 5, 2012). The combination of a severe accident capable vent and an accident management strategy that uses various mechanisms to reduce the release of fission products differs from Option 4 described in this paper in that a specific performance measure (e.g., a combined decontamination factor), would not be treated as a firm regulatory requirement.

A complicating factor in developing Option 2 for Mark II containments is the possibility that molten core material on the drywell floor of the Mark II containment may fail the downcomers or the drywell sump drain lines and result in suppression pool bypass. Enclosure 4 describes this issue in more detail. A bypass of the suppression pool would negate the possible benefits of a severe accident capable venting system in terms of avoiding containment overpressure conditions and a scrubbed release through the suppression pool. The staff concludes that Option 2 for Mark II containments may need to include plant design changes to minimize the possibility of such a bypass event. For example, design features were incorporated into the advanced boiling-water reactor (ABWR) to prevent core debris from entering the lower drywell sump and ablating concrete and breaching the embedded drywell liner. These design changes would likely result in higher costs for Mark II containments, but the average plant costs (including Mark I and Mark II) is expected to remain close to the staff's estimate of $2 million.

If Option 2 is selected, the staff recommends that it be imposed by issuing a new order or amending existing Order EA-12-050. Enclosure 7 provides a draft proposed order.[2] The upgrading of the venting system to ensure its functionality during severe accident conditions also would be required for Option 3 (filtered vents order) and Option 4 (severe accident confinement strategies) and would need to be addressed within the development and implementation of those options should they be selected. The staff would use the draft proposed order to support interactions with stakeholders and provide the final order to the Commission via a Regulatory Notification. The staff plans to complete stakeholder interactions and issue the final order within 60 days of the staff requirements memorandum related to this paper. To the extent practical, the order would include performance-based attributes, rather than prescriptive requirements, to allow licensees flexibility in determining how to meet the requirements of the order.

[2] It is likely that the draft proposed order for Option 2, provided in Enclosure 7a, will require revision based on interactions with stakeholders and continuing internal discussions on technical or legal issues. If the Commission approves Option 2, the staff will provide the Commission with the final order via a Regulatory Notification.

Option 3: Filtered Severe Accident Venting System Order

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

This option involves the installation of an engineered filtered containment vent system to prevent the release of significant amounts of radioactive material following most severe accident scenarios at BWRs with Mark I and Mark II containments. The filtering system and connections to the containment wetwell and drywell would need to operate during conditions associated with significant core damage, including breaching of the reactor vessel. Similar to Option 2 (severe accident capable venting system), the approach significantly increases the chances of preventing gross containment failure and substantially supports accident management efforts to arrest further plant degradation and the release of radioactive materials. The inclusion of an engineered filter reduces the amount of radioactive material released to the environment during the venting of containments during severe accidents.

The assumed approach involves the installation of filtering technologies that currently exist to significantly reduce the release of radioactive material in the event of a severe accident. Examples of this filtering technology have been installed at some foreign plants following the accidents at Three Mile Island and Chernobyl (see Enclosures 3 and 4). Section 2 provides an analysis of this option using existing regulatory analysis guidance to determine if the benefits outweigh the approximate $15 million cost of plant modifications. Section 3 provides a revised analysis that addresses concerns about the possible need to update or change the regulatory analysis guidance. If Option 3 is selected, the staff recommends imposing the related requirements by issuing a new order or amending Order EA-12-050. Enclosure 7 provides a draft proposed order.[3] The staff would use the draft proposed order to support interactions with stakeholders and provide the final order to the Commission via a Regulatory Notification. The staff plans to complete stakeholder interactions and issue the final order within 60 days of the staff requirements memorandum related to this paper. Similar to Option 2 and to the extent practical, the order would include performance-based attributes, rather than prescriptive requirements, to allow licensees flexibility in determining how to meet the requirements of the order.

Option 4: Severe Accident Confinement Strategies[4]

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

---

[3] It is likely that the draft proposed order for Option 3, provided in Enclosure 7b, will require revision based on interactions with stakeholders and continuing internal discussions on technical or legal issues. If the Commission approves Option 2, the staff will provide the Commission with the final order via a Regulatory Notification

[4] The Advisory Committee for Reactor Safeguards (ACRS) reviewed a draft of this Commission paper in which Option 4 was entitled “Performance-based approach.” The NRC staff’s internal review and concurrence process led to revisions to the paper, including clarifying the title and descriptions of Option 4 as developing a severe accident confinement strategy for Mark I and Mark II containments. The general proposal and most of the discussions related to Option 4 remain the same as that reviewed and commented on by the ACRS.

A possible approach to containment venting for BWRs with Mark I and Mark II containments involves establishing technical acceptance criteria (e.g., defined decontamination factor or site-specific cost/benefit analysis) and allowing licensees to select and justify systems or combinations of systems, such as suppression pools, containment sprays, or engineered filters, to accomplish the function and meet the criteria. For this option, the staff did not analyze a specific filtering system; instead, it drew on insights from various sensitivity studies to define a possible approach. Section 4 of this enclosure discusses this option in more detail.

Whereas Options 2 and 3 would have performance-based attributes, Option 4 could potentially result in performance-based regulatory requirements. The development of performance-based approaches tends to involve extensive interactions with stakeholders and the preparation of detailed industry and regulatory guidance documents. Since this process and the related extended time periods are envisioned for Option 4, it may be appropriate to proceed with Option 2 and the related order to ensure the venting systems currently being designed and implemented under EA-12-050 are made severe accident capable. The draft proposed order provided in Enclosure 7a includes highlighted language that would be included if the Commission selects Option 4 with a more immediate requirement to make the containment vents capable of operation during severe accidents. This approach would support the longer term development of the severe accident confinement strategies while possibly reducing the net costs for the changes to containment venting systems. Whichever regulatory process is chosen, it would include performing a regulatory analysis for the proposed requirements, which would depend on the chosen performance measure. For the purpose of this paper, the regulatory analysis for Option 4 addresses a more subjective discussion dealing with possible benefits, costs, and uncertainties. If Option 4 is selected, the staff would engage stakeholders to develop appropriate performance measures, identify the appropriate regulatory process for establishing requirements (e.g., order or rulemaking), and develop the necessary project plans and schedules.

1.2 Other Items

As mentioned in the discussion of the proposed options, the uncertainties associated with the assessment of these approaches are important in attempting to reach a regulatory decision. In addition to the quantitative evaluations in Sections 2 and 3, several other qualitative factors and policy issues directly affect the issue of requiring severe accident capable or filtered vents. Section 5 discusses these qualitative factors and policy issues, which include the following:

- 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; and
- addressing international experience and practices (including availability of technology).

Beyond these options, there are other issues relating to containment venting that are worth considering.

1.2.1 Vents in Areas other than Primary Containment

This issue involves the possible installation of vents in areas other than primary containment. For example, vents could be installed in other areas to prevent deflagration or detonation of hydrogen within the reactor building, as occurred at Fukushima. Given that this topic is associated with the control of hydrogen, it will ultimately be resolved through the Tier 3 item associated with the Near-Term Task Force (NTTF) Recommendation 6, "Hydrogen Control and Mitigation Inside Containment or in Other Buildings." However, there is a significant relationship between the control of hydrogen within the primary containment and other plant areas and the decisions associated with severe accident capable or filtered containment venting. The staff will consider the outcomes from this paper in its assessment and proposals for possible paths to resolving Recommendation 6. If Option 2 or 3 are pursued, the resulting containment venting system could play a substantial role in resolving Recommendation 6 for Mark I and Mark II containments. The most likely remaining issues would be an assessment of hydrogen release pathways from containment bypass events and the performance of containment seals, drywell head, and penetrations, if post-severe accident high-pressure and high-temperature conditions were maintained in the containment. Resolving this issue would depend significantly on ensuring a reliable engineered pathway for releasing the hydrogen from the containment and ensuring that there was minimal differential pressure across containment seals and penetrations following venting operations. The staff notes that venting strategies involving maintaining containment pressure at elevated levels, or strategies involving containment vent cycling at elevated pressures, would continue to present the potential for hydrogen leakage from the primary containment to other buildings and may not be as beneficial in resolving NTTF Recommendation 6. Industry proposed approaches, as described in the letter from NEI dated October 5, 2012, might employ such elevated-pressure strategies.

1.2.2 Drywell Flooding Capabilities

Various risk assessments that the NRC and industry have performed for BWRs with Mark I or Mark II containments have concluded that 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 overpressurization failure, containment overtemperature failure). The importance of providing cooling water to protect the containment was a factor in establishing the mitigating strategies and capabilities associated with the possible loss of large areas of the plant due to explosions or fire in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh). Current capabilities are addressed in the NRC-endorsed guidance document NEI-06-12, "B.5.b Phase 2 & 3 Submittal Guideline," Revision 2, issued December 2006, which calls for adding approximately 300 gallons per minute through a portable pump and flow paths into the drywell or reactor vessel. For the purpose of this assessment, the staff has incorporated this capability into its characterization of the status quo and has not proposed additional requirements within the proposed options for severe accident capable or filtered containment vents. This capability is important to the success of Options 2, 3, or 4 for scenarios in which the core melts through the reactor pressure vessel, which could then lead to containment failure. The importance of this capability to any severe accident venting requirements may warrant a more specific requirement than is currently in place under 10 CFR 50.54(hh) and the related guidance documents. Because there are existing requirements and guidance related to this capability, the NRC staff has not included a similar requirement in the draft proposed orders provided in Enclosure 7 for Options 2 and 3. However, 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. An additional consideration is the degree to which core or drywell sprays are credited for providing a scrubbing or decontamination function for the radioactive materials within the drywell during a severe accident. The staff will, if necessary, address this issue as part of its implementation of the decisions reached on possible requirements for severe accident capable or filtered containment venting systems.

1.3 Justification for Imposing Requirements

In developing new or revised regulatory requirements, the NRC uses regulatory analyses such as those discussed in Sections 2 and 3 to help in the decisionmaking process. However, the agency is not constrained by quantitative cost/benefit calculations. There are two primary cases in which the agency's deliberations might lead to an action even though the costs of that action might appear to outweigh the benefits. These cases involve one of the following:

(1) finding that one or more of the options discussed is needed to provide reasonable assurance of adequate protection of the public health and safety, or[5]

(2) finding that one or more of the options justify the associated costs as a result of the combination of the standard regulatory analysis and other qualitative factors.

Adequate Protection

The first case involves specific exceptions in 10 CFR 50.109, "Backfitting," on the need to perform cost/benefit analyses for some NRC actions that impose new requirements on licensees. The exceptions listed in 10 CFR 50.109(a)(4) are listed below:

> (i) That a modification is necessary to bring a facility into compliance with a license or the rules or orders of the Commission, or into conformance with written commitments by the licensee; or
>
> (ii) That regulatory action is necessary to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security; or
>
> (iii) That the regulatory action involves defining or redefining what level of protection to the public health and safety or common defense and security should be regarded as adequate.

In the case of the potential options under consideration (Options 2, 3, or 4), exceptions (ii) or (iii) could be invoked if the Commission were to determine that such changes were needed to address the current or a revised standard for adequate protection. A discussion of the history

[5] In the case of a finding that an action is needed for adequate protection of public health and safety, the NRC is actually not allowed to consider costs in its decisions. Therefore, a finding should be made about adequate protection independent of costs instead of invoking the adequate protection provisions because the costs have been found to exceed the calculated benefits.

and traditional use of the NRC invoking the standard of reasonable assurance of adequate protection is provided in SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated August 14, 2012.

The NRC staff assessed the possible benefits associated with the options described in this paper for improving containment venting at BWRs with Mark I and II containments. The assessment and lessons learned from the Fukushima accident indicate that functions to delay core damage and containment failure in combination with protective actions taken to evacuate or shelter the public are able to minimize risks to the public health and safety. The NRC has traditionally reserved the use of the adequate protection standard for the protection of public health and safety and has invoked it for design-basis accidents, selected functions to prevent core damage (e.g., EA-12-050), and programs to ensure licensees have strategies or contingencies for severe accidents (e.g., emergency planning, EA-12-049 ("Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, dated March 12, 2012) and 10 CFR 50.54(hh)). The NRC has previously considered incorporating into its approach to defense in depth a balancing of prevention and mitigation measures. However, such an approach and the related criteria for achieving a balance between elements of defense in depth were not formally adopted nor included in the NRC's guidance documents. For the purpose of this analysis, the staff did not apply any of the exceptions to the Backfit Rule. The staff has proceeded with analyses of proposed venting modifications as possible cost-justified substantial safety improvements. The staff's decision to proceed with a cost-benefit analysis does not represent a staff recommendation regarding whether Option 2, 3, or 4 could be pursued under one of the exceptions to the Backfit Rule.

The NRC staff does not currently consider the potential economic consequences of an accident within its deliberations on adequate protection. A Commission decision to revise the agency's accounting of offsite land contamination (Option 3 in SECY-12-0110) could affect arguments related to finding whether the addition of a filtered vent system for BWRs with Mark I or II containments might be needed for a revised adequate protection standard or a separate equivalent standard for economic consequences. Even in the absence of Commission direction to revise the current focus on public health and safety in deliberations on adequate protection (or equivalent standard for economic consequences), the current assessment process for a regulatory analysis or a backfit analysis includes consideration of offsite costs—a topic discussed within the additional qualitative factors in Section 5.

Cost-Justified Safety Enhancements

For the purpose of this paper, a two-part backfit analysis is applied, as described in 10 CFR 50.109(a)(3). Before proceeding to a comparison of costs and benefits, the first part of the test under (a)(3) is to determine whether there is a "substantial increase in the overall protection of the public health and safety or the common defense and security derived from the backfit." NUREG/BR-0058 includes the following explanation from staff requirements memorandum (SRM), "SRM-SECY-93-086—Backfit Considerations," dated June 30, 1993, on the need for plant backfits to provide a substantial increase in safety:

> The Commission has stated that "substantial" means important or significant in a large amount, extent, or degree. Applying such a standard, the Commission would not ordinarily expect that safety-applying improvements would be required as backfits that result in an insignificant or small benefit to the public health and safety, regardless of costs. On the other hand, the standard is not intended to be interpreted in a manner that would result in disapprovals of worthwhile safety or

> security improvements having costs that are justified in view of the increased protection that would be provided. This approach is flexible enough to allow for qualitative arguments that a given proposed rule would substantially increase safety. The approach is also flexible enough to allow for arguments that consistency with national and international standards, or the incorporation of widespread industry practices, contributes either directly or indirectly to a substantial increase in safety. Such arguments concerning consistency with other standards, or incorporation of industry practices, would have to rest on the particulars of a given proposed rule. The Commission also believes that this approach of “substantial increase” is consistent with the Agency’s policy of encouraging voluntary initiatives.

NUREG/BR-0058 describes the use of the NRC safety goals as a way to evaluate if a proposed backfit provides substantial safety improvements. However, it also recognizes the limitations of this approach for modifications that do not change core damage estimates but provide improvements to containment performance. Specifically, the guidance states:

> The NRC recognizes that in certain instances, the screening criteria 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 nonnegligible, 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.
>
> Furthermore, note that the safety goal screening criteria described in these Guidelines do not address issues that deal only with containment performance. Consequently, issues that have no impact on core damage frequency (ΔCDF of zero) cannot be addressed with the safety goal screening criteria. However, because mitigative initiatives have been relatively few and infrequent compared with accident preventive initiatives, mitigative initiatives will be assessed on a case-by-case basis with regard to the safety goals. Given the very few proposed regulatory initiatives that involve mitigation, this should have little overall impact from a practical perspective on the usefulness of the safety goal screening criteria.

Senior NRC managers on the Japan Lessons-Learned Steering Committee (SECY-11-0117, “Proposed Charter for the Longer-Term Review of Lessons Learned from the March 11, 2011, Japanese Earthquake and Tsunami,” dated August 26, 2011) assessed the issue of whether the possible imposition of requirements for severe accident capable or filtered venting systems satisfy the “substantial safety improvement” standard. The managers decided that the possible modifications should proceed to the estimation and evaluation of values and impacts within the regulatory analysis process. The following sections of this enclosure provide these estimates and evaluations.

1.4 Performance-based Approaches

The Commission provided the staff with direction on the need to consider performance-based approaches in its SRM for SECY-11-0124, “Recommended Actions to be Taken Without Delay from the Near-Term Task Force Report,” dated October 18, 2011. In that SRM, the Commission stated:

> As the staff evaluates Fukushima lessons-learned and proposes modifications to NRC’s regulatory framework, the Commission encourages the staff to craft recommendations that continue to realize the strengths of a performance-based system as a guiding principle. In order to be effective, approaches should be flexible and able to accommodate a diverse range of circumstances and conditions. In consideration of events beyond the design basis, a regulatory approach founded on performance-based requirements will foster development of the most effective and efficient, site-specific mitigation strategies, similar to how the agency approached the approval of licensee response strategies for the “loss of large area” event under its B.5.b program.

A performance-based regulatory approach is one that establishes performance and results as the primary basis for regulatory decisionmaking, and incorporates the following attributes:

(1) measurable (or calculable) parameters (i.e., direct measurement of the physical parameter of interest or of related parameters that can be used to calculate the parameter of interest) exist to monitor system, including facility and licensee, performance,

(2) objective criteria to assess performance are established based on risk insights, deterministic analyses, and/or performance history,

(3) licensees have flexibility to determine how to meet the established performance criteria in ways that will encourage and reward improved outcomes, and

(4) a framework exists in which the failure to meet a performance criterion, while undesirable, will not in and of itself constitute or result in an immediate safety concern.

In the development and assessment of Options 2, 3, and 4, the staff considered the potential incorporation of performance-based approaches. Options 2 and 3 include performance-based attributes, and Option 4 involves development of a performance-based regulatory requirement.

2.0 EVALUATION OF OPTIONS USING EXISTING REGULATORY ANALYSIS GUIDANCE

The staff, with assistance from Sandia National Laboratories, performed analyses using MELCOR and MACCS2 computer simulations to characterize the expected plant response and offsite consequences for an extended loss of electrical power at a representative BWR with a Mark I containment design. The following key assumptions were used in the simplified regulatory analyses provided in this enclosure:

- Base event frequency for events in which the severe accident capable or filtered venting system would add significant value is assumed to be $2x10^{-5}$ per reactor-year. This value is taken from the results of individual plant examinations, NRC standardized plant analysis risk (SPAR) models, and engineering judgment. This value is considered representative of the core damage frequency for the operating plants with Mark I and II containment designs.

- To address the uncertainties associated with event frequencies, the assessment is also performed assuming a core damage frequency of $2x10^{-4}$ per reactor-year, which is a factor of 10 above the base event frequency.[6]

- Assuming a lower CDF value would reduce the calculated benefits in a similar fashion but in the opposite direction, thereby making the proposals less cost-effective. Since the reduction is proportional to the CDF assumption (i.e., reducing CDF by a factor of 10 reduces the calculated benefit by a factor of 10), the staff has not specifically included the sensitivity to lower CDFs within the discussions or tables.

- The technical analyses sections included in Enclosure 5 discuss specific assumptions about transients, equipment performance, and recovery actions.

The base case and sensitivity analyses are summarized below in terms of the various factors used in the regulatory analysis guidelines. A more complete assessment of uncertainties and sensitivities can be found in Enclosure 5 and in the regulatory analysis available in ADAMS at Accession No. ML12312A456.

2.1 Public Health (Accident)

For the purpose of establishing the base case, scenarios involving the potential for a significant release of radioactive material through a containment vent path are identified and evaluated in terms of consequences and estimated accident frequencies. In the case of BWRs with Mark I and II containment designs, this subset of severe accidents makes up the majority of the sequences involving large releases (with the remainder involving failures of containment and releases through pathways other than a controlled and possibly filtered pathway). Containment failures, for example, could occur as a result of severe accident conditions that involve high pressures in the containment (e.g., venting failures) or scenarios that involve a molten core breaching both the reactor vessel and drywell liner (e.g., lack of drywell spray).

[6] The range was selected to provide decisionmakers with information about sensitivities to certain assumptions and to address uncertainties, plant-to-plant variations, and the limited number of PRAs including external events. The NRC staff is not placing any particular importance on the upper value used except as a part of sensitivity studies provided for CDF and other parameters. The factor of 10 in the simplified analysis provided in this enclosure generally corresponds to the 95% confidence levels used in Enclosure 5C and the regulatory analysis.

The results from the simulations of an extended loss of electrical power transient are consistent with previous evaluations and the experience from the Fukushima Dai-ichi accident in the viability of avoiding large exposures to the general public by the evacuation of populations near a nuclear power plant. The analysis assumes, however, that populations are instructed to return to their homes following an accident if projected dose rates fall below the defined criteria (e.g., 500 mrem/year). This longer-term exposure of populations from the residual contamination of the countryside is controllable but is assumed to estimate a plausible balancing of public health and economic impacts. In this case, reducing public exposures by limiting the return of populations to affected areas would result in an increase in the economic consequences by preventing the use of homes and businesses.

For the status quo, Case 6, a related scenario described in Enclosure 5, includes failure of containment on overpressure and a long-term population dose of 310,000 rem to the public within 80 kilometers (50 miles) of the site. As discussed in Enclosure 5, consideration of various possible sequences of events, with assumed probabilities, leads to an estimated 80-kilometer (50-mile) population dose risk of 10.2 rem/reactor year (rem/ry).

2.1.1 Option 2—Severe Accident Capable Vents

To estimate the potential benefits of requiring a severe accident capable venting system, the staff used the simulations and risk estimates from Enclosure 5. The estimated population dose risk for a severe accident capable vent is 5.9 rem/ry or a net benefit of 4.3 rem/ry when compared to the base case. Using the existing guidance for NRC regulatory analyses, the staff converted the estimated dose savings into dollars using the following equation:

[(Estimated Accident Frequency) x (Change in Population Dose)] x ($2,000/person-rem) x
[1- exp (-(discount rate) x (remaining reactor life)]/(discount rate)

Where: 4.3 rem/ry reflects the frequency and change in estimated dose
conversion factor of $2,000 per rem
discount rates are assumed to be 3 percent[7]
remaining reactor life assumed to be 25 years

Using the assumptions above, the benefits of the severe accident capable vent in terms of avoiding doses to the population are estimated to be $150,000 per reactor unit.

The benefits, estimated above, are proportional to the estimated accident frequency and the related uncertainties. If, for example, the estimated frequency related to a severe accident were raised to $2x10^{-4}$/reactor year, the associated benefits would increase to $1.50 million per unit.

2.1.2 Option 3—Filtered Vents

The installation of a filtering system with expected performance requirements would significantly reduce the release and subsequent exposure of the population. For the sake of this evaluation, the values associated with Modification 6 from Enclosure 5 are used. These estimates include a risk evaluation estimate for population dose of 2.0 rem/ry or a projected reduction of

[7] A complete regulatory analysis is available in ADAMS at Accession No. ML12312A456 and includes an alternate assessment using a 7 percent discount rate. The 3-percent discount rate provides a higher calculated benefit and is used for the remainder of this enclosure.

8.2 rem/ry when compared to the base case. Using the equation above, the reduction in projected dose risks translates into a net benefit of $290,000 per reactor unit.

The benefits estimated above would increase to $ 2.90 million per unit if the estimated accident frequency were raised to $2x10^{-4}$/reactor year.

The uncertainties associated with expected decontamination factors for suppression pools and sprays were assessed by performing additional simulations with the MELCOR and MACCS2 computer codes. Sensitivity studies related to various scenarios and decontamination factors are provided in Enclosure 5. A very conservative estimate with limited credit for scrubbing by the suppression pool or sprays and venting from the drywell resulted in a reduction in dose for a filtered vent path of nearly 4 million rem for a population within 80 kilometers (50 miles). That value would, in turn, translate into a calculated benefit of $2.8 million per unit in current dollars using the above equation and core damage frequency of $2x10^{-5}$ per reactor-year.

2.2 Occupational Health (Accident)

Accidents involving significant core damage will result in an increase in occupational exposures at the plant. A range of estimated occupational exposures were taken from NUREG/BR-0184 to simulate the possible effects of severe accident capable and filtered venting systems.
A containment failure due to overpressure or liner melt-through was assumed to result in the highest estimate of immediate occupational dose from the regulatory analysis handbook, which is 14,000 person-rem. The conditions associated with severe accident capable vents were assumed to reduce the associated occupational exposure to 3,300 person-rem. Finally, the filtered release was assumed to result in the lowest immediate occupational exposure of 1,000 person-rem, which is approximately the occupational dose received from the Three Mile Island accident. The risk assessment provided in Enclosure 5 considered the possible end states and their likelihood for the various possible modifications and provided dose risk for the immediate accident period. The following total occupational dose risks are derived from combining the immediate occupational doses and the longer term (cleanup) doses from NUREG/BR-0184.

- status quo 0.88 person-rem/ry
- severe accident capable (Mod 2) 0.56 person-rem/ry
- filtered vent (Mod 6) 0.33 person-rem/ry

Using the same equations and assumptions ($2,000 per person-rem and CDF of $2x10^{-5}$ per reactor-year) as used above for consideration of public doses results in an estimated benefit of $11,000 per unit for severe accident capable vents and $19,000 per unit for filtered vents. Increasing the estimated frequency of core damage to $2x10^{-4}$ per reactor-year would result in an increase of estimated benefit for the severe accident capable vents to $110,000 per unit and to $190,000 per unit for filtered vents.

Another potential impact in terms of evaluating filtered vents would be the number of workers added to participate in offsite cleanup activities following a major release. However, decisions related to cleanup activities for the nearby countryside could consider and assess the expected dose to workers versus the economic impact of not recovering the affected areas. The potential dose-related costs for the cleanup of contaminated offsite areas are accounted for in the assessment of potential effects on offsite property.

2.3 Offsite Property

The United States has an existing structure for nuclear power plants that involves measures to prevent, contain, and mitigate releases of radioactive materials and, if necessary, to compensate individuals for the potential damages to health, property, or income. For the purpose of this discussion, prevention and containment relate to attempts to arrest a nuclear accident and maintain the radioactive material within the plant (including confining materials within containment or within a filter). Mitigation relates to limiting the impact on public health through protective actions such as sheltering or evacuation. The Price-Anderson Act and related NRC regulations address provisions for compensation. Regulatory analyses do not usually address compensation for nuclear accidents since it involves the source and flow of funds but does not influence the actual amount of damages that a potential nuclear accident causes. The funding from current insurance pools available to address a major nuclear accident in the United States is approximately $12 billion.

The results from the computer simulations include estimates for the amount of land area that could be contaminated following the modeled scenarios as well as an estimate of total economic costs (assuming loss of use of property, businesses, etc.). The results from the analyses for one of the cases (Case 6 with containment failure on overpressure) used in the risk assessments described in Enclosure 5 is a land contamination area of 72 km^2, and an offiste property damage estimate of $850 million. Consideration of various possible sequences of events, with assumed probabilities, leads to an estimated offsite cost risk of $630,000 per reactor-year.

2.3.1 Option 2—Severe Accident Capable Vents

Applying the same assumptions and cases discussed for population doses, the estimated difference in the offsite cost risk for Modification 2 (assumed passive vent from wetwell) is $19,767 per reactor-year. Using the existing guidance and assumptions for NRC regulatory analyses, the estimated difference in economic consequences in current dollars (i.e., the benefit of the severe accident capable vent) is $348,000 per reactor unit. Assuming an event frequency of $2x10^{-4}$ per reactor-year would increase the calculated benefit to $3.48 million per unit.

2.3.2 Option 3—Filtered Vents

The installation of a filtering system with expected performance requirements would significantly reduce the estimated affected land area and related economic consequences. The filtered venting system in this assessment uses the offsite cost risk reductions from Enclosure 5 for Modification 6 (assumed passive vent from wetwell with filter), which were estimated to be $34,166 per reactor-year. Using the established assumptions and conversions, the avoided economic consequences translates in current dollars to a benefit of $600,000 per reactor unit. As with the other factors, this result is directly correlated to estimated accident frequencies and increases to $6.0 million per unit if a frequency of $2x10^{-4}$ per reactor-year is assumed. Section 3 provides additional discussion on the uncertainties and other issues associated with estimating economic consequences.

2.4 Onsite Property

A severe accident at a nuclear power plant is assumed to result in the loss of the affected unit in terms of the future electrical output and early decommissioning (complicated by the post-accident conditions). The installation of a filter within the containment vent path would not likely change the total loss of the unit experiencing significant fuel damage. However, a filter could limit contamination of nearby units and the associated increase in onsite property damage, including loss of generation from the co-located units. The factor related to occupational health was used to address radiation exposure for site cleanup. Other cleanup costs are addressed using guidance from NUREG/BR-0184 and the estimates of risk factors provided in Enclosure 5.

The onsite property costs address the possible loss of electrical generation resulting from an accident. For the purposes of this evaluation, the radioactive releases from either the base case or Option 2 are assumed to result in the permanent closure of not only the unit with the damaged core but also units located on the same site. In accordance with existing practices, the impact of these shutdowns is modeled as the replacement costs for a 10-year period (after which alternate energy supplies would become available). The filtered venting case is assumed to result in the loss of the co-located units for 1 year. Of the 31 BWR units with Mark I or II containments, 8 are single unit sites, 16 could affect one other operating unit, and 7 could affect 2 other operating units. Based on these site combinations, consideration of the loss of co-located facilities on a generic basis for Mark I and II units is addressed by multiplying the loss of electrical generation by a factor of 1.75.

2.4.1 Option 2—Severe Accident Capable Vents

The estimated difference in the onsite cost risk for Modification 2 (assumed passive vent from wetwell) is $15,185 per reactor-year. Using the existing guidance and assumptions for NRC regulatory analyses, the estimated difference in onsite costs in current dollars (i.e., the benefit of the severe accident capable vent) is $268,000 per reactor unit. Assuming an event frequency of $2x10^{-4}$ per reactor-year would increase the calculated benefit to $2.68 million per unit.

The cost from the loss of electrical generation from co-located facilities was estimated assuming an average value of $9.9 million per reactor-year. Using the generic factor of 1.75 and a period of 10 years for needed power replacement results in an undiscounted consequence estimate of $173.25 million. Considering the likelihood of such events results in a value of $3,500 for an event frequency of $2x10^{-5}$ per reactor-year and of $35,000 for the value of $2x10^{-4}$ per reactor-year. However, since this loss is the same for the base case, it is not used directly, except to estimate savings for the following filtered vent option.

2.4.2 Option 3—Filtered Vents

The estimated difference in the onsite cost risk for Modification 6 (assumed passive vent from wetwell with filter) is $24,485 per reactor-year, which translates into an estimated difference in onsite costs in current dollars (i.e., the benefit of the filtered vent) of $430,000 per reactor unit. Assuming an event frequency of $2x10^{-4}$ per reactor-year would increase the calculated benefit to $4.3 million per unit.

The cost from the loss of electrical generation from co-located facilities was estimated assuming an average value of $9.9 million per reactor-year. Using the generic factor of 1.75 and a period of 1 year for needed power replacement for the undamaged unit results in an undiscounted consequence estimate of $106.425 million. Considering the likelihood of such events results in a value of $2,100 for an event frequency of $2x10^{-5}$ per reactor-year and of $21,000 for the value of 2×10^{-4} per reactor-year. This can be represented as a savings of $1,400 for the $2x10^{-5}$ per reactor-year frequency and $14,000 for an assumed event frequency of $2x10^{-4}$ per reactor-year.

2.5 Industry Implementation

The base case involves implementing current requirements (e.g., EA-12-049, "Mitigation Strategies for Beyond-Design-Basis External Events," and EA-12-050); therefore, it does not involve additional costs. The implementation costs for providing a severe accident capable reliable hardened vent could vary significantly between plants based on equipment configurations and plans regarding the implementation of EA-12-050. An assumed cost for this evaluation is $2 million per unit, which is based primarily on judgment and gross estimates of time and materials for many of the plants that would need to perform modifications. As discussed in Enclosure 4, the costs for severe accident capable vents for Mark II containment designs will likely be higher than for Mark I units. The higher cost reflects the likely need to modify containments to prevent a molten core from causing a bypass of the suppression pool because of failure of downcomers and drain lines below the reactor vessel. Given that avoiding bypass of the wetwell is necessary to make the severe accident capable vents a viable option for the Mark II design, protection of the downcomers and drain lines are included in the cost of this option for Mark II containments. The implementation costs for the filtered venting system are estimated based on discussions with foreign plants, vendors, and other stakeholders. The estimated costs used in this assessment are $15 million per unit.[8]

2.6 Industry Operation

The base case involves implementing current requirements (e.g., EA-12-049 and EA-12-050); therefore, it does not involve additional costs. The upgrading of venting systems to be compatible with severe accident conditions is not expected to add significantly to the operating costs of a nuclear power plant and is therefore not estimated for this evaluation. The operating costs for maintaining the filtered venting system, including training, are estimated based on discussions with foreign plants, vendors, and other stakeholders. The estimated costs used in this assessment are $60,000 per unit per reactor-year in current dollars for a present value of $1.1 million (3 percent discount rate and a 25-year license term).

2.7 NRC Implementation

The base case involves implementing current requirements (e.g., EA-12-049 and EA-12-050); therefore, it does not involve additional costs. The implementation costs for developing regulations for a severe accident capable or filtered vent and subsequent reviews and inspections are estimated to involve a total NRC cost of $830,000 or approximately $27,000 per unit.

Longer-term NRC operating costs are not expected to change as a result of the possible addition of these requirements and are not included in this evaluation.

[8] Some stakeholders have noted that an estimate of $15 million seems low and that the price could be factors of 2 or 3 higher. The costs could be significantly above $15 million if the system is designed and installed as safety-related equipment or needed to be protected from beyond-design-basis external events.

2.8 Summary

The results of the evaluation of the costs and benefits of a severe accident capable and filtered vent system using the existing regulatory analysis guidelines are summarized below.

Table 1: Summary of Quantified Cost/Benefit Assessment for Options 2 and 3

Costs ( ) and Benefits of Severe Accident Capable and Filtered Vent System $ K Per Unit				
	Severe Accident Capable Venting Systems		**Engineered Filtered Venting Systems**	
Factor	**Best Estimate Frequency of $2x10^{-5}$ / ry**	**Accident Frequency of $2x10^{-4}$ / ry**	**Best Estimate Frequency of $2x10^{-5}$ / ry**	**Accident Frequency of $2x10^{-4}$ / ry**
Public Health	150	1,500	290	2,900
Occupational Health	11	110	19	190
Offsite Property	348	3,480	600	6,000
Onsite Property	268	2,680	430	4,300
Industry Implementation	(2,000)	(2,000)	(15,000)	(15,000)
Industry Operation	n/a	n/a	(1,100)	(1,100)
NRC Implementation	(27)	(27)	(27)	(27)
TOTAL	(1,250)	+5,743	(14,778)	(2,737)

3. EVALUATION OF OPTIONS INCLUDING POSSIBLE CHANGES TO REGULATORY ANALYSIS GUIDANCE

The previous section discussed the base case and related sensitivities for the evaluation. Much of that information is used for the revised analysis in this section, which focuses on possible updates or changes to the regulatory analysis guidance or assumptions related to the costs and benefits of a severe accident capable or filtered venting system for BWRs with Mark I or II containment designs. There are several possible changes that would affect the evaluation of the severe accident capable or filtered vent options. In general, the consequence analyses from Section 2 are carried forward to this assessment and revised factors are used to represent those consequences in terms of the cost/benefit calculations.

3.1 Public Health (Accident)

Section 2 described the evaluation of the base case and options in terms of possible exposures to the populations within 80 kilometers (50 miles) of a plant undergoing a severe accident for which the installation of severe accident capable or filtered vents could reduce the offsite consequences. A discussion of sensitivities to accident frequency and retention of fission products by suppression pools and sprays is provided in Enclosure 5. The other major factor in the assessment of possible public health benefits is the value used to convert population dose (roentgen equivalent man (rem)) into dollars based on various health studies and the valuation of impacts on life and health. The NRC staff is currently assessing a possible revision of the $2,000 per person-rem conversion factor, including a revision of the factor to $4,000 per person–rem.

The sensitivity of this assessment of the costs and benefits of installing a severe accident capable or filtered venting system for BWRs with Mark I or II containments is directly proportional to the assumed conversion factor. A doubling of the factor, to $4,000 per person-rem, would double the previously calculated benefits of the severe accident capable vent to $300,000 per unit while the benefit of a filtered system would be increased to $580,000 per unit. An increase in assumed accident frequency to $2x10^{-4}$ per reactor-year would then increase the benefits to $3.0 million and $5.8 million per unit, respectively, for the severe accident capable and filtered venting systems. The estimated benefit of an engineered filter for the case in which possible retention of fission products within the suppression pool is largely neglected by venting from the drywell would increase to $5.6 million dollars per unit (assuming accident frequency of $2x10^{-5}$ per reactor-year).

3.2 Occupational Health (Accident)

As above, an increase in the dollars per person-rem conversion factor to $4,000 would double the estimates provided in Section 2. The estimated benefits would be $22,000 per unit for a severe accident capable vent and event frequency of $2x10^{-5}$ per reactor-year and $220,000 for an estimated event frequency of $2x10^{-4}$ per reactor-year. Likewise, the estimated benefits of a filtered vent system would increase to $38,000 and $380,000 per unit, respectively, for the frequencies of $2x10^{-5}$ and $2x10^{-4}$ per reactor-year.

3.3 Offsite Property

Estimates of the long-term economic consequences of the Fukushima Dai-ichi accident continue to evolve and ultimately may be used to update NRC guidance for performing regulatory analyses. As discussed in SECY-12-0110, the NRC staff is evaluating possible updates to the computer codes and models used to assess offsite property damages.

There continues to be a fairly wide range of estimates for the actual economic impact of previous events, such as Hurricane Katrina, which struck the southern United States in 2005. This highlights the difficulty in predicting potential impacts for future disasters, including potential nuclear reactor accidents. Several journals provide estimates of around $125 billion, including the loss of oil production and refining, for the economic impacts of Hurricane Katrina. Other major disasters, such as Hurricane Andrew in 1992 and Hurricane Irene in 2011, have been estimated to have caused around $45 billion in economic losses. A conservative simulation using MACCS2, discussed in Enclosure 5, addresses uncertainties in the performance of the suppression pool and sprays in limiting the release of radioactive materials. The simulation calculated total economic costs at $33 billion for that conservative representation of a large release from the modeled BWR.[9] In terms of a typical regulatory analysis, an estimated offsite cost of $33 billion translates (assuming an event frequency of 2×10^{-5} per reactor-year) into a net benefit (averted cost) of $11.6 million per unit. Given the ongoing efforts to assess and update capabilities to estimate economic consequences, the staff is not providing additional sensitivities here about the estimation of offsite property damage. This issue will be discussed again in qualitative terms in Section 5.

3.4 Onsite Property

As mentioned in Section 2, a severe accident at a nuclear power plant is assumed to result in the loss of the affected unit in terms of the future electrical output and early decommissioning (complicated by the post-accident conditions) for both the base case and the proposed options. The installation of a filter within the containment vent path could, however, limit contamination of nearby units and the associated increase in onsite property damage, including loss of generation from the co-located unit. The potential impacts could range from a temporary loss of the unaffected unit to its permanent closure because of economic, technical, or societal factors. The regulatory analysis includes sensitivities to a range of electrical energy costs, but these were not found to affect the assessment dramatically. The results from Section 2 are as follows:

Table 2: Onsite Property Damage Estimates for Options 2 and 3

Modification	Unit Cost	
	2×10^{-5}/yr Event Frequency	2×10^{-4}/yr Event Frequency
Severe Accident Capable Venting Systems	$268,000	$2.68 million
Engineered Filtered Venting Systems	$430,000	$4.3 million

9 Note that under the provisions of the Price-Anderson Act, damages that exceed the available insurance pools (currently at approximately $12 billion) would require actions on the part of the U.S. government to increase nuclear utility liability or contribute to the compensation funds.

Although the replacement energy costs for the affected and co-located units do not appear to significantly affect the results of the regulatory analysis, the Fukushima accident also led to the shutdown of other nuclear units located away from the direct effects of the accident. Such shutdowns might result from new regulatory reviews or requirements, caution on the part of plant operators, or other societal factors. The possibility of such shutdowns and the resulting increase in replacement power is addressed as a sensitivity case in the regulatory analysis and could increase the calculated benefits from the installation of a filtering system. Early shutdown of a large number of units would also entail the costs from decommissioning and disturbance of broader energy markets.

3.5 Industry Implementation

As discussed in Section 2, the costs of industry implementation are estimated to be $2 million for severe accident capable vents and $15 million for a filtered venting system. While there is considerable uncertainty with these estimates, the handling of industry implementation costs is not likely to be a significant issue within the updating of the regulatory analysis guidance and no additional discussion of sensitivities is provided here.

3.6 Industry Operation

The industry operating costs for maintaining the filtered venting system were estimated in Section 2 to be $60,000 per unit per reactor-year in current dollars for a present value of $1.1 million (3 percent discount rate and a 25-year license term). As with the industry implementation costs, there are uncertainties associated with NRC estimates of industry operating costs, but they are not likely to be identified as a significant issue when updating the regulatory analysis guidance. Therefore, no additional discussion of sensitivities is provided here.

3.7 NRC Implementation

As discussed for the previous two factors, NRC implementation costs for the development of regulations have uncertainties, but this element of the regulatory analysis is not likely to be a major issue for updating the regulatory analysis guidance. The NRC implementation costs are estimated to be approximately $27,000 per unit.

3.8 Summary

The results of the evaluation of the costs and benefits of a severe accident capable and filtered vent system using possible revision of the regulatory analysis guidelines are summarized below:

Table 3: Sensitivity Study for Quantified Cost/Benefit Assessment for Options 2 and 3

Costs ( ) and Benefits of Modified Vent System ($ K) Per Unit						
Factor	Best Estimate (from Section 2)		Revised to Address Sensitivity to Changes to Regulatory Analysis Assumptions			
	Severe Accident Capable	Filtered	Severe Accident Capable[(1)] (at $2x10^{-5}$/ry)	Severe Accident Capable[(1)] (at $2x10^{-4}$/ry)	Filtered (at $2x10^{-5}$/ry)	Filtered (at $2x10^{-4}$/ry)
Public Health	150	290	300	3,000	580	5,800
Occupational Health	11	19	22	220	38	380
Offsite Property	348	600	348	3,480	600*	6,000
Onsite Property	268	430	268	2,680	430	4,300
Industry Implementation	(2,000)	(15,000)	(2,000)		(15,000)**	
Industry Operation	n/a	(1,100)	n/a		(1,100)	
NRC Implementation	(27)	(27)	(27)		(27)	
TOTAL	(1,250)	(14,778)	(1,089)*	+7,353	(14,479)	+353

(1) As discussed in Enclosures 4, the costs for severe accident capable vents for Mark II containment designs will likely be higher than for Mark I units. The higher cost reflects the likely need to modify containments to prevent a molten core from causing a bypass of the suppression pool due to failure of drain lines and downcomers below the reactor vessel. Avoidance of suppression pool bypass is needed to make the severe accident capable vents a viable option for the Mark II design.

* Uncertainties in estimating consequences is addressed further as a qualitative factor in Section 5. As previously mentioned, a largely unmitigated release leads to offsite property damage on the order of $33 billion, which in turn translates into a benefit for filtered vents of approximately $11.6 million per unit.

** Note that some stakeholders have stated that the price of a filtered vent system could range from $30 – 45 million

4. SEVERE ACCIDENT CONFINEMENT STRATEGIES

As previously noted in Sections 2 and 3, there are significant uncertainties associated with some of the key parameters used in the regulatory analyses. These include the frequency of the scenarios that would benefit from severe accident capable or filtered vents, the efficiency of various systems in limiting the release of radioactive materials, and the economic consequences of a severe accident that results in the contamination of environs near a reactor facility. An issue related to uncertainties is the plant-to-plant variations that limit the effectiveness of generic assessments and generic solutions. The various BWRs with Mark I and II containments have similarities, but also differences in design features, system capabilities and vulnerabilities, risk contributors, number of co-located units, and geographic locations. Such differences between plants have given rise to the possible benefits of developing a performance-based approach, which would require each licensee to evaluate the needed performance of the containment venting function and to implement appropriate design and procedure changes to satisfy the performance requirement. While there are performance-based attributes in the orders that would be issued under Options 2 and 3, Option 4 would include broader consideration of a performance-based regulatory approach. A discussion of the Commission's direction on performance-based approaches is in Section 1.4 of this enclosure.

The NRC traditionally has approached the development of performance-based regulations using the rulemaking process to accommodate the necessary interactions with stakeholders and the appropriate development of performance standards. Simpler measures might be effectively imposed through the issuance of orders, but measures for which additional research is needed or involve other policy issues (e.g., broader societal measures such as land contamination) would more likely be pursued through the rulemaking process. The staff would include in any proposed rulemaking for this option an assessment of costs and benefits related to the performance-based approach.

In a letter dated October 5, 2012, NEI proposed that licensees for each plant with Mark I or Mark II containments develop a "filtering strategy" that could include the use of existing systems and, if deemed appropriate, additional equipment such as engineered filters. A performance-based option and the NEI proposal would seem to require, at a minimum, a venting system capable of operating under severe accident conditions (Option 2). The establishment of a performance measure could, for some plants, result in the installation of an engineered filtering system (Option 3) if it is determined that such a system is necessary to meet the performance measure with the required level of confidence.

<u>Performance Measures</u>

One potential approach to defining a performance measure would be to define a parameter such as a required decontamination factor (DF) for the available combination of plant systems, such as core or drywell sprays, the suppression pool, the reactor building, and, if necessary, an installed filtering system. The basis for the selected decontamination factor should first be defined, and could potentially range from decontamination factors intended to address issues such as containment performance (conditional containment failure probability), land contamination (extent of contaminated property), offsite consequences (health consequences and population return criteria), economic consequences (monetary value of damages), or an equivalence to available filtering technologies (i.e., Option 3) in terms of reliability of systems and confinement of radioactive releases. A traditional NRC approach would be to define a source term (defined radionuclides and chemical forms) and require licensees to analyze the effectiveness of the various systems and ensure plant capabilities satisfied the acceptance criteria (including adding an engineered filtering system if necessary). The NRC could prescribe methods for analyses or review the analyses performed for the various plants and their specific configurations. Requirements placed on the analyses could include validation against tests, experiments, and operating histories. This type of approach probably would not specifically account for plant-specific risk profiles but instead establish specific accident conditions to analyze. Defining a specific collective DF also would be consistent with the traditional NRC practice for design-basis accidents of defining regulatory limits in terms of radiation dose to a representative individual (or contamination per unit area) at a specified distance from the release. However, for severe accident conditions, the NRC has more recently required the development of strategies or contingencies and not established 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)). Development of a severe accident confinement strategy without defining a specific performance measure was discussed as a possible approach under Option 2.

Another approach could be to define performance measures based on what can be reasonably achieved using currently available filtering technologies. This concept is used as the basis for the draft proposed order in Enclosure 7b for an engineered filtering system. Such an approach would ensure that the technical requirements associated with Option 4 could provide a comparable decontamination of the containment atmosphere as that provided by an engineered filtering system. Engineered filtering systems could provide the actual means of satisfying the performance measure if the evaluation of other confinement strategies failed to provide the needed confidence in decontamination factors or system reliability. The availability of engineered filtered systems could also provide the basis for establishing schedules for this approach that are consistent with the proposed timeframe proposed for Option 3 (e.g., 2017).

The consideration of risk contributors and importance measures could be included in establishing the performance measure to address significant plant differences. The performance goal could be established for event frequencies above an established criterion, or the event frequency and DF could be considered together in a more complicated consideration of limiting the exposure to a representative individual (or contamination per unit area). This type of an approach recognizes and tries to address the differences in plant designs and related differences in the importance of various accident sequences to core damage and containment failure. The following figure from NUREG-1560, “Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance,” presents the range in accident sequence contributions for various accident scenarios for BWR 3/4 plants.

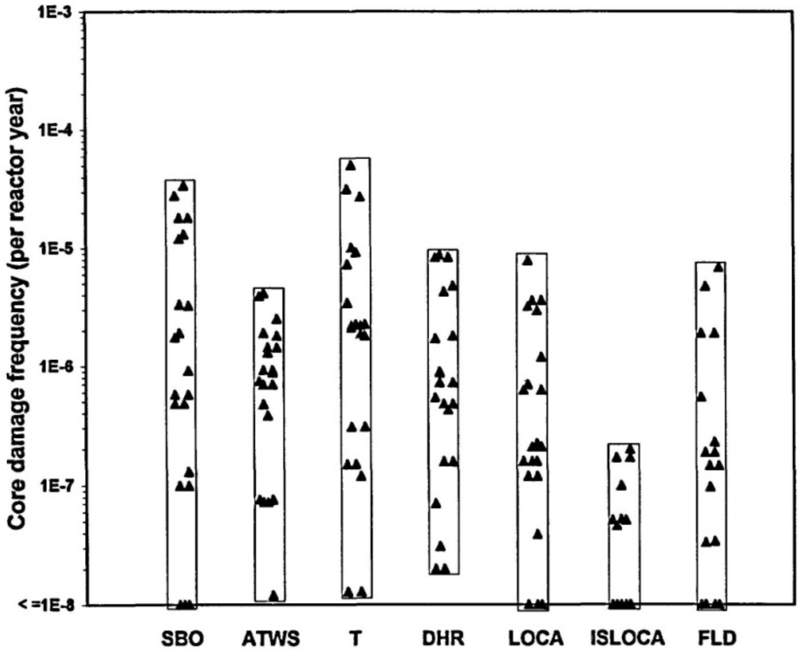


Figure 3.6 Reported IPE accident sequence CDFs for BWR 3/4 plants with RCIC.

SBO – Station Blackout
ATWS – Anticipated Transients without Scram
DHR – Transients with Loss of Decay Heat Removal
FLD – Internal Flood Initiators
T – Other Transients
LOCA – Loss of Coolant Accidents
ISLOCA – Interfacing System LOCA

Figure 1: BWR Accident Sequence Core Damage Frequencies
Source: NUREG-1560

A third alternative for a performance-based approach consists of including additional measures in the determination of the required performance of the collective systems to limit the release of radioactive materials. An example would be to define as low as reasonably achievable requirements similar to those described in NUREG-2150, “A Proposed Risk Management Regulatory Framework,” issued April 2012, and current severe accident mitigation alternatives (SAMA) assessments. This approach would not only account for design differences but also factors such as the differences in potential economic consequences because of plant location. Such an approach would differ from the traditional calculation of doses to a representative individual, which tends to make requirements largely independent of location. Hypothetically, under this approach a plant located in an economically developed area might need to install additional measures to contain radioactive materials as compared to a very similar plant located in a less economically developed location. It should be noted that the second and third alternatives would likely require the licensees to have and maintain plant-specific PRAs. Therefore, these approaches may have a relationship to activities such as the resolution of NTTF Recommendation 1 on possible changes to the NRC’s regulatory framework (which might require the licensees to have and maintain a plant-specific PRA).

The NRC staff envisions that the development of a severe accident confinement strategy would involve many interactions with stakeholders. These interactions would help inform the regulatory analysis that would be performed to support developing regulatory requirements associated with Option 4. Given that the process would involve developing specific performance measures and subsequent analysis of the resultant costs and benefits, the NRC staff has not specifically addressed Option 4 within the regulatory analyses, described in Sections 2 and 3. However, any approach to using the containment venting systems during severe accident conditions would require modifications to existing systems (or planned systems to satisfy EA-12-050) to ensure that they were capable of operating following core damage and related conditions. Option 2 would therefore appear to set the minimum costs and related benefits for the performance-based approach. Additional costs for Option 4 would likely include additional studies and possibly scaled testing or experiments to demonstrate the ability of sprays, pools, and engineered filters to contain radioactive materials through the implementation of a predictable and repeatable strategy as suggested by the recent study by the Electric Power Research Institute (EPRI) report entitled "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents – BWR Mark I and Mark II Studies (EPRI Product No. 1026539)." The staff expects that the costs and related benefits of Option 4 lies between Options 2 and 3, both of which might be cost-justified safety enhancements upon consideration of uncertainties and qualitative factors.

While the costs of Option 4 could be compared to Option 2 or 3, the completion schedule for the activity would likely be at least several years longer. All of the uncertainties mentioned throughout this paper would complicate any efforts to define, review, and implement a system that meets the selected performance measure with the desired level of confidence. In its letter dated October 5, 2012, NEI noted that considerable time would be required to determine if the EPRI approach was feasible and without unintended consequences:

> Applying the findings of the EPRI study to individual plants will take significant effort and time. At a minimum, each plant (or class of plants) will have to perform a specific evaluation based on the EPRI methodology to determine the appropriate strategy to implement. This would require, prior to initiation of the study, alignment with NRC on the filtering strategy performance-basis, development of a regulatory vehicle, implementation guidance, design basis assumptions, severe hazard considerations, accident scenario requirements, etc. Experience suggests that this will involve numerous meetings among NRC staff, industry and other stakeholders over at least 24 months.
>
> Following development of the performance-basis, etc., a significant amount of time is required to perform the required analysis, engineering, design, development, procurement, plant walk-downs, installation, testing, training, and so on.

The significance of the longer implementation period for Option 4 depends on the characterization of the safety issue being addressed. For people who consider containment venting improvements an important enhancement or possibly even necessary for reasonable assurance of adequate protection of public safety, a delay of several years would be a significant negative for this option. However, for those who view possible improvements to severe accident features as worthwhile, but not necessarily urgent safety enhancements, the longer schedule can be viewed as providing an opportunity to coordinate the venting issue with other improvement efforts (e.g., NTTF Recommendation 1, SECY-12-0110) and development of policies applicable to all reactor technologies.

5. OTHER FACTORS AND POLICY ISSUES

The regulatory analyses in Sections 2 and 3 assessed the possible imposition of requirements for venting systems for Mark I and Mark II containments and whether such requirements met the standard to be cost-effective, substantial safety improvements. The assessments were performed using the process described in established guidance and considered, where possible, uncertainties in the assumptions and possible changes to the guidance under consideration at the time of this assessment. The analyses considered severe accident capable vents and filtered venting systems. Section 4 discusses another option for a performance based approach, which probably falls between the other options in terms of expected costs and benefits.

A regulatory analysis using only quantitative factors, including standard assumptions, would not appear to justify the imposition of additional requirements on the venting systems for BWR Mark I and Mark II containments. However, sensitivity studies and analyses using values of event frequency and accident consequence in the upper range of the uncertainty bands result in the calculated benefits potentially justifying the likely costs of improved venting systems. The existing guidance in NUREG/BR-0058 discusses the consideration of qualitative factors instead of or as a supplement to the quantitative analyses such as those in Sections 2 and 3 of this enclosure, and in more detail in the complete regulatory analysis. In this case, the NRC staff is considering several qualitative factors to supplement the previous discussions. A tool that is sometimes useful to decisionmakers in making cost/benefit decisions is a break-even assessment such as shown in the following figure for the engineered filtered vent modification. The figure shows values of when the modification would be justified in terms of limiting consequences (expressed in dollars) for core-damage events of certain frequencies. In this case, plant modification costs were assumed for the filtered vent ($15 to $45 million) with other data associated with the BWRs with Mark I and II containments.

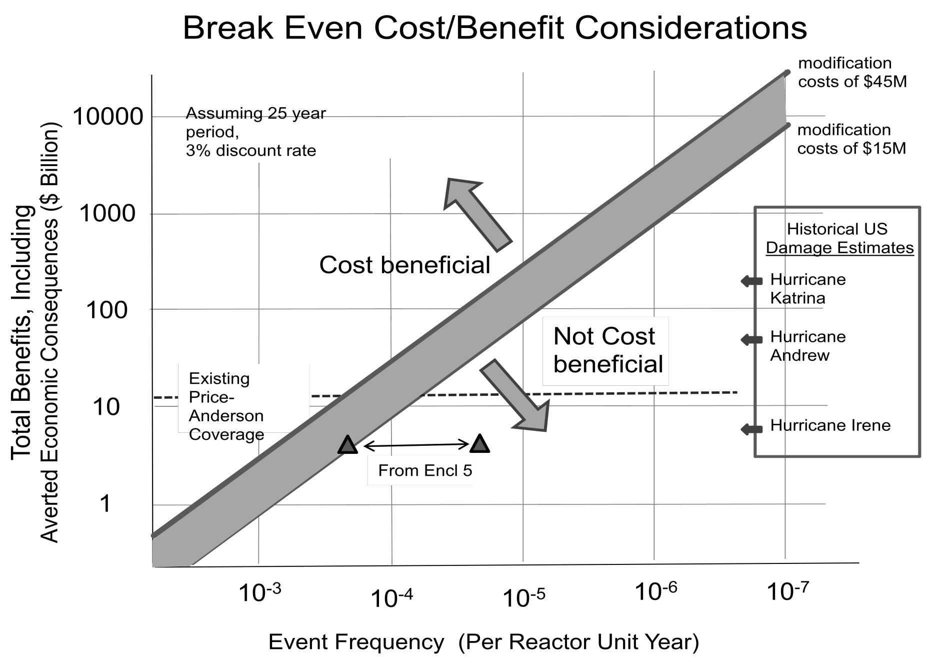


Figure 2: Break-Even Values for Option 3 (Filtered Vent)

As shown above, the "best estimate" valuation (event frequency of 2 x10^{-5}/yr) is outside the break-even region while assuming an event frequency of 2x10^{-4}/yr would appear to strengthen the argument for making the filtered vents on the basis of it being a cost justified safety enhancement. Although the staff was not able to assign numerical values to the various qualitative factors discussed in the following sections, they can likewise be viewed as either affecting the frequency of challenges to containment integrity or affecting the release of a large amount of radioactive material from the plant (which results in economic consequences) and thereby moving toward or away from the break-even region shown in the figure.

A discussion of several significant qualitative factors is provided below.

5.1 Defense in Depth

A key principle of NRC's regulation and oversight of nuclear power plants has historically been and continues to be "defense in depth." An aspect of defense in depth traditionally has been to have multiple barriers to the release of radioactive materials and to have equipment and personnel 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. The containment systems at nuclear power plants play a key role in helping confine fission products within the plant if an accident progresses to a point where significant core damage has occurred. Containment designs also help to control accidents by absorbing the energy released from the reactor coolant system, holding water for long-term core cooling, and protecting systems from external hazards. Given the key role of containment performance as an essential element of defense in depth, concerns about the performance of Mark I and II containments during severe accident conditions have been discussed for many years.

The logic underlying this set of basic goals is that each level of defense represents a threshold where failure to accomplish the prior goal introduces a significantly greater potential for consequences and a greater uncertainty in the phenomenology, accident progression, and, therefore, the ability to control the outcome of an event.

<u>Prevention</u>

The first defense-in-depth goal, prevention of severe accidents, recognizes that there is little threat to public health and safety in the absence of core damage, while there is a significant increase in the potential for major consequences once fission products are released from the fuel and cladding. In addition, much larger phenomenological uncertainties are introduced under severe accident conditions than when the core is undamaged and is in a fixed geometric position. Finally, considerable uncertainty in the availability and functionality of core cooling equipment is also indicated, since major failures must have already occurred to arrive at a severe accident condition.

<u>Containment</u>

The second defense-in-depth goal is containment of fission products on site in the event of a severe accident. This is a critical threshold because containment of fission products on site results in minimal impact to public health and the environment, while failure to contain radioactive material leads to the potential for widespread health, environmental, and socio-economic consequences. Furthermore, once a large release has occurred, the ability to influence outcomes is limited by uncontrollable factors, such as weather and public response.

Thus, the containment goal provides a reliable backstop against uncertainties in the prevention of severe accidents and protects against the uncertainties associated with uncontrollable releases and the potentially large and varied consequences.

The event at TMI showed the importance of a reliable containment design—the second element of the defense-in-depth strategy. Despite extensive core damage, the containment was successful, limiting fission product release to insignificant levels. The passive attributes of the containment building (i.e., the large volume and inherent strength) were critical to prevent the release of radioactive materials despite the hydrogen burn that ensued. At TMI, the containment barrier provided sufficient time for event diagnoses and recovery from operator errors that occurred earlier in the event. However, the accident at TMI was not complicated by an extended loss of electrical power and heat removal systems, as was the case at Fukushima.

<u>Mitigation of Release (Emergency Preparedness)</u>

Emergency planning and response is the third and final element of defense in depth. This element provides protection against uncertainties in containment performance under severe accident conditions. Evacuation and sheltering protect against acute doses of radioactive materials. Relocation protects against long-term health effects in the event of containment failure. This element does not, however, protect against environmental or socio-economic consequences.

The containment failures at Fukushima showed the importance of emergency planning for protection against acute doses of radiation. Evacuation, shelter, and relocation were very effective in limiting doses to the public. The Fukushima event also confirmed that when containment fails in a severe accident, the consequences (economic, social, and long-term health) are large, difficult to estimate, and depend upon critical but uncontrollable factors, such as weather and public reaction.

In considering additional requirements for venting systems for BWRs with Mark I or II containments, the deliberations ultimately will need to determine whether those additional protections are reasonable in light of the costs and the benefits, including the desire for effective defense in depth for dominant severe accident sequences. A process to consider in deliberating the containment improvement options is to follow the progression of accidents and determine at what point does the combination of event probability and consequence, with consideration of related uncertainties, warrant regulatory controls. For BWRs, estimates of low core melt frequencies have, in part, justified the NRC's previous acceptance of the estimated high conditional failure probability of the Mark I and II containments. The containments did fail, however, during the accident at the Fukushima Dai-ichi facility, as predicted for those plant conditions. Further, the failure of containments during the Fukushima accident resulted in a large release of radioactive material and greatly complicated the attempts of plant operators to stop conditions from worsening. For example, the loss of the reactor buildings (secondary containments) resulted from hydrogen explosions, which occurred because of difficulties in venting to maintain pressures and hydrogen levels within the containment structures.

Summary—Defense in Depth

The relatively high likelihood of a failure of Mark I and II containments following a core melt accident questions the level of defense in depth that this intended barrier to the release of radioactive material provides. Improving the chances that the containment venting function is available under severe accident conditions reduces the chances of failure and uncontrolled releases. Providing filters in the venting system significantly reduces release of radioactive materials for the dominant core melt scenarios.

Option 1

Option 2 ↑

Option 3 ↑↑↑

Option 4 ↑↑

5.2 Uncertainties

As discussed above, there are significant uncertainties in estimating the frequency of events for which a severe accident capable or filtered venting system would be a useful severe accident design feature. The results of the regulatory analyses are sensitive to the event frequency, and as shown above, a frequency assumption of $2x10^{-4}$ per reactor-year is sufficient to make the filtered vent marginally cost effective. There are also significant uncertainties in the calculation of event consequences in terms of the dispersion of radioactive material into the site environs. This is due in part to significant uncertainties about the degree to which radioactive materials would be retained within the plant as a result of systems such as sprays and suppression pools. Estimating economic consequences given a large release of radioactive material also includes large uncertainties related to modeling the many different aspects of local economies and their impact on the larger economy. An example of this is the supply chain disruptions that followed the tsunami in Japan or the flooding in Thailand. Just as an increase in event frequency by approximately an order of magnitude was sufficient to change the results of the cost/benefit analyses, so would an increase in consequences by an order of magnitude appear to change the balance between costs and benefits.

Summary—Uncertainties

Significant uncertainties exist in the estimation of event frequencies and consequences. This factor provides support for taking additional action. The benefits from the proposed changes, in terms of reducing the consequences from severe accidents, would be greatest for Option 3 (filter) while the least would be from Option 2 (unfiltered venting).

Option 1

Option 2 ↑

Option 3 ↑↑↑

Option 4 ↑↑

5.3 Severe Accident Management

The Fukushima experience demonstrated that responding to and arresting the accident was complicated by the problems associated with venting containment and the failure of containment. The failure of containments as a result of overpressure conditions creates harsh environments in the reactor building and other plant locations. In turn, the elevated temperatures and radiation levels can impede operators in their attempts to restore installed equipment or put into service temporary equipment such as what EA-12-049 requires. Severe accident capable vents would not only include equipment that could remain functional and support venting operations during severe accident conditions, but they would also address shielding and equipment operation to ensure personnel could execute needed tasks during a severe accident. Some severe accident capable venting designs include the use of passive features, such as rupture disks, to provide additional confidence that the system would operate and prevent failure of containment structures because of overpressure conditions.

The filtered vent designs would provide the same improvements to the plant to prevent containment failures and help control conditions within the reactor building and other site areas. The filtered system could provide an additional advantage in that decisionmakers could be more confident (or at least less stressed) about ordering the venting operation knowing that the filter would contain the vast majority of radioactive materials. From an accident management perspective, this increased confidence in the venting operation would enable measures to restore installed equipment, connect temporary equipment, or otherwise take measures to arrest the accident.

Summary—Severe Accident Management

Improving the containment venting systems to support operation under severe accident conditions would enhance the possible management of the accident by allowing operators to focus on other recovery actions. Each proposed option provides some benefit, but filtered systems are the simplest while a performance-based approach could be integrated into other severe accident management activities and procedures.

Option 1	
Option 2	↑
Option 3	↑↑
Option 4	↑

5.4 <u>Hydrogen Control</u>

In addition to providing pressure control, severe accident capable or filtered venting systems also can remove hydrogen from the containment spaces and lessen the likelihood of hydrogen deflagration and detonations in the containment structures or the reactor building. The primary consideration for improving the control of hydrogen during a severe accident is associated with the Tier 3 item related to NTTF Recommendation 6, “Hydrogen Control and Mitigation Inside Containment or in Other Buildings.” However, the successful venting of containments during severe accidents can help address the potential problems of the buildup of hydrogen in primary and secondary containment systems. Selection of any of the severe accident capable venting options proposed in this paper will therefore influence and potentially help resolve hydrogen control issues for Mark I and II containments.

The benefits of venting hydrogen for BWRs with Mark I or II containments were evident during the Fukushima accident. Hydrogen generated by various mechanisms associated with severe accidents made its way to the reactor buildings and exploded. Those explosions, in turn, increased the amount of radioactive materials escaping from the facility, complicated operators efforts to respond to the event, and increased concerns about the integrity of spent fuel pools.

The location of the spent fuel pools within the BWR reactor buildings is another feature that makes the venting function and control of hydrogen especially important to these reactor designs. Proper venting of hydrogen would alleviate concerns associated with hydrogen burns within the reactor building, possibly affecting the integrity of the spent fuel pool.

Summary – Hydrogen Control

The experience at Fukushima Dai-ichi demonstrated the importance of effective control of hydrogen generated during severe accidents. The possible containment venting systems discussed in this paper (Options 2, 3, or 4) could provide a way to improve the control of hydrogen.

Option 1	
Option 2	↑↑
Option 3	↑↑
Option 4	↑

5.5 External Events

The technology comparison above may not fully address the influence of external events and the fact that such hazards could be major contributors to the risk profiles for operating nuclear power plants. The estimated core damage frequencies for BWRs from internal events are lower than those for PWRs, in part because of the multiple systems available to add water to the reactor core. However, events such as an extended loss of electrical power renders some of these systems unavailable and potentially reduces the BWR advantage for such events, which are likely to be caused by a major external event (e.g., a beyond-design-basis seismic or flooding event). Provided that the enhanced venting systems, either severe accident capable or filtered, are able to survive the external event and remain available for use if the accident progresses to involve significant core damage, then the system could be a major part of the accident response. As mentioned under the severe accident management factor, the availability of a reliable venting system during severe accident conditions could help prevent conditions from degrading further and enable responders to continue efforts to cool the molten core. The venting system thereby compliments the ability of the portable equipment to help arrest an event even if previous efforts had failed to prevent core damage.

Summary – External Events

Beyond design basis external events such as the 2011 earthquake and tsunami in Japan will challenge normal and emergency power and cooling systems at a nuclear power plant. There is a significant advantage to having installed equipment and strategies in place to address such events and conditions and thereby avoid the nuclear power plant compounding the consequences from such events.

Option 1	
Option 2	↑
Option 3	↑↑
Option 4	↑↑

5.6 Multi-Unit Events

The quantitative evaluations performed in Sections 2 and 3 did not consider potential scenarios involving accidents at more than one unit at a multiple unit site. The tsunami that flooded the Fukushima site initiated a series of events that resulted in core damage accidents at three of the six units sharing the site. The most likely cause of multi-unit accidents is a major beyond-design-basis external event, such as what occurred at Fukushima and discussed above. Although the frequency of such events might be estimated for particular sites, the uncertainties are relatively large given the limited recorded histories and limited knowledge of hazards, such as large (beyond-design-basis) seismic or flooding events. In addition, the possibility of core damage events at multiple units has the potential for larger releases and increased economic damage. By improving severe accident management functions and, especially in the case of the filtered vent, reducing the releases from each unit, the enhanced venting systems could help address concerns about concurrent core damage events at multiple units.

Summary—Multi-unit Events

Conditions or events (e.g., external hazards) that challenge multiple units at a nuclear facility is a concern that the Fukushima accident highlighted. There is a significant advantage to having installed equipment and strategies in place to address such multi-unit events.

Option 1	
Option 2	↑
Option 3	↑↑
Option 4	↑↑

5.7 Independence of Barriers

The events at Fukushima highlighted the interdependence between the performance of core cooling functions and the pressure suppression containment designs used for BWRs with Mark I or Mark II containment designs. This dependent relationship between what is generally thought of as individual barriers to the release of radioactive materials has been noted in several severe accident studies and during the operating history of BWRs with Mark I or Mark II containments (see Enclosure 2). Although the primary fission product barriers usually are discussed as being largely independent of each other, the NRC has previously recognized and accepted some dependencies, such as the crediting of containment accident pressure for supplying net positive suction head for pumps in the emergency core cooling system. In its SRM for SECY-11-0014, “Staff Requirements–SECY-11-0014–Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents,” dated March 15, 2011, the Commission directed the NRC staff to continue to use existing guidance in the standard review plan, which states:

> Defense in depth is preserved (for example, system redundancy, diversity, and independence are maintained commensurate with the expected frequency and consequence of challenges to the system; defenses against potential common cause failures are maintained and the introduction of new common cause failure mechanisms is assessed; and defenses against human errors are maintained).

Although the discussion above relates to design-basis functions, previous (pre-Fukushima) evaluations that the NRC performed also found that the expected frequency and consequences of severe accidents involving potential releases through established vent pathways for BWRs did not warrant additional severe accident design features (see SECY-89-017, “Mark I Containment Performance Improvement Program,” dated January 23, 1989, and related SRM). However, the Commission could find that the Fukushima accident has changed our understanding of severe accident frequencies and consequences such that measures are needed to address this issue and compensate for the lack of independence between the core cooling and containment functions. The installation of a filtered vent would be a plausible approach to improving the defense-in-depth attributes for BWRs with Mark I or Mark II containments. In its efforts to address lessons learned from Fukushima, the industry, to date, has emphasized additional measures for preventing core damage (e.g., making available portable pumps for injection into the core or drywell) versus the installation of an additional barrier (filters) on a dedicated vent pathway from containment.

A focus on preventing or arresting the progression of core damage is also consistent with EA-12-050, which requires modifications to ensure BWRs with Mark I and II containments have a reliable hardened vent to control containment pressure. The NRC issued EA-12-050 with a finding that the action was needed for adequate protection and included the following explanation:

> The events at Fukushima Dai-ichi highlight the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. The inability to reduce containment pressure inhibited efforts to cool the reactor core. If additional backup or alternate sources of power had been available to operate the containment venting system remotely, or if certain valves had been more accessible for manual operation, the operators at Fukushima may have been able to depressurize the containment earlier. This, in turn, could have allowed operators to implement strategies using low-pressure water sources that may have limited or prevented damage to the reactor core. Thus, the events at Fukushima demonstrate that reliable hardened vents at BWR facilities with Mark I and Mark II containment designs are important to maintain core and containment cooling.

Summary—Independence of Barriers

While it may not be necessary or practical to ensure the complete independence of each barrier to the release of radiation, it is desirable to minimize dependencies and address the high conditional failure probability of Mark I and Mark II containments following a compromise of the preceding barriers (fuel and coolant system). The filtered system would provide the most independence while the unfiltered vent could result in large releases in the attempts to reduce containment overpressure conditions.

Option 1	
Option 2	↑
Option 3	↑↑↑
Option 4	↑↑

5.8 Emergency Planning

The installation of severe accident capable or filtered venting systems can add to existing emergency planning margins (e.g., effective evacuation periods) by controlling the release of radioactive materials as compared to containment failure by overpressurization. The filtered vent system provides additional advantages by dramatically reducing the amount of radioactive material released through containment venting during severe accident conditions. This could allow different protective action recommendations that would reduce the number of evacuees, thereby reducing the stress and risks associated with such emergency measures. In addition to the effects on immediate protective measures to protect public health and safety, the filtered vent option reduces or eliminates concerns about the return of populations following a possible release of radioactive materials and long-term exposures associated with contamination of the countryside through the failure of containment or the release from an unfiltered venting operation. The issue of long-lasting effects from a release also relates to other qualitative factors, such as societal considerations and uncertainties in estimating economic consequences.

Summary—Emergency Planning

Improving containment venting functions during severe accidents would reduce uncertainties and releases, thereby enabling improvements in emergency planning or reducing the need to evacuate large numbers of people. The most benefit in reducing the demands on emergency planning would be associated with Option 3 (filter) while the proposed change with the least benefit would be from Option 2 (unfiltered venting).

Option 1	
Option 2	↑
Option 3	↑↑↑
Option 4	↑↑

5.9 Consistency between Reactor Technologies

NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” provides a comparison between a Mark I containment and a PWR containment of the conditional containment failure probability given various core damage events. The following figure from NUREG-1150 shows that the conditional failure probability for Mark I containments is relatively high (approximately 0.75 for the plant evaluated in that study).

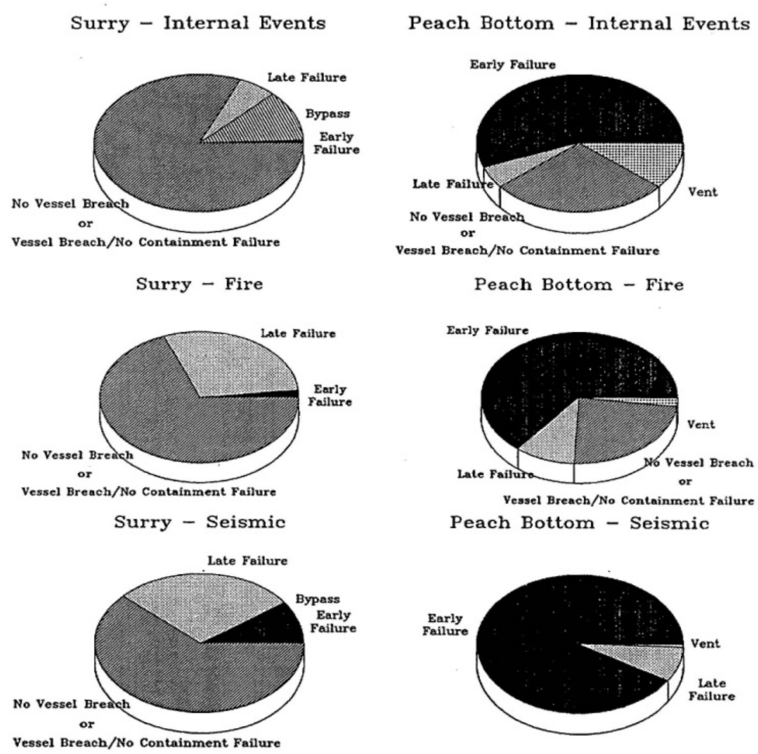


Figure 9.5 Relative probability of containment failure modes (internal and external events, Surry and Peach Bottom).

Figure 4: Comparison of Containment Failure Modes
Source: NUREG-1150

However, as pointed out in NUREG-1150 and NUREG-1560, and shown in the following figures, when combined with estimated frequencies of core damage events, the risk of large releases from BWRs with Mark I and Mark II containments is comparable to other plant designs. A lower core damage frequency is estimated because of a more diverse set of plant equipment that is able to add water to the reactor core under most plant conditions. The weighting of the defense-in-depth approaches to emphasize minimizing core damage can result in similar overall risk profiles for large releases. However, many of these core-cooling systems would be rendered unavailable for events such as an extended station blackout that occurred at Fukushima Dai-ichi. Thus, given a core damage event, the higher conditional failure probability of containment failure means that a release is more likely than not.

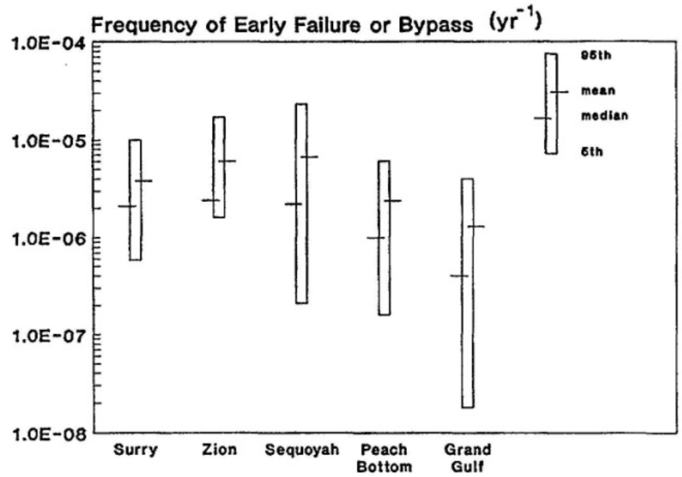


Figure 9.3 Frequency of early containment failure or bypass (all plants).

Figure 5: Frequency of Containment Failure or Bypass
Source: NUREG-1150

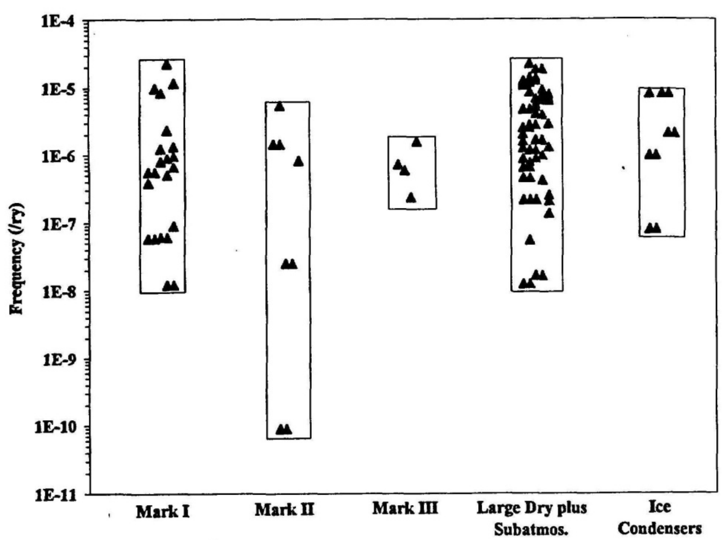


Figure E.3 Frequencies of significant early release (by containment type) as reported in the IPEs.

Figure 6: Frequency of Significant Early Release
Source: NUREG-1560

Summary—Consistency with Other Technologies

While the proposed improvements to venting systems for BWRs with Mark I and II containments address a known weakness in the severe accident performance for those plants, the pursuit of these improvements without resolving broader issues (e.g., NTTF Recommendation 1 and the Severe Accident Policy Statement) introduces the possibility for inconsistent treatment of severe accident capabilities for the various reactor technologies.

Option 1	⇧⇧⇧
Option 2	
Option 3	
Option 4	⇧

5.10 <u>Severe Accident Policy Statement</u>

Following the 1979 accident at TMI, the United States and the international nuclear safety community recognized that severe accidents needed further attention. The NRC evaluated, generically, the capability of existing plants to tolerate a severe accident and found that the design-basis approach contained significant safety margins for the analyzed events. These margins permitted operating plants to accommodate a large spectrum of severe accidents. Based on this information, the Commission, in its “Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants” (50 FR 32138; August 8, 1985), concluded that existing plants posed no undue risk to public health and safety. The Commission also concluded that no basis existed for immediate action on generic rulemaking or other regulatory changes affecting these plants because of the risks that a severe accident posed. To address this issue for operating plants in the long term, the NRC issued SECY-88-147, “Integration Plan for Closure of Severe Accident Issues,” in May 1988. This document identified the following necessary elements for closure of severe accidents:

- performance of an individual plant examination
- assessment of generic containment performance improvements
- improved plant operations
- a severe accident research program
- an external events program
- an accident management program

Each of these programs and the conclusions reached are discussed elsewhere in this paper and its enclosures. The portion of the policy statement that deals with operating plants states:

In light of the above principles and conclusions, the Commission's policy for operating reactors includes the following guidance:

- Operating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new safety information arises to question whether there is adequate assurance of no undue risk to public health and safety.

- In the latter event, a careful assessment shall be made of the severe accident vulnerability posed by the issue and whether this vulnerability is plant or site specific or of generic importance.

- The most cost-effective options for reducing this vulnerability shall be identified and a decision shall be reached consistent with the cost-effectiveness criteria of the Commission's backfit policy as to which option or set of options (if any) are justifiable and required to be implemented.

- In those instances where the technical issue goes beyond current regulatory requirements, generic rulemaking will be the preferred solution. In other cases, the issue should be disposed of through the conventional practice of issuing Bulletins and Orders or Generic Letters where modifications are justified through backfit policy, or through plant-specific decision making along the lines of the Integrated Safety Assessment Program (ISAP) conception.

- Recognizing that plant-specific PRAs have yielded valuable insight to unique plant vulnerabilities to severe accidents leading to low-cost modifications, licensees of each operating reactor will be expected to perform a limited-scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents. These plant-specific studies will serve to verify that conclusions developed from intensive severe accident safety analyses of reference or surrogate plants can be applied to each of the individual operating plants. During the next two years, the Commission will formulate a systematic approach, including the development of guidelines and procedural criteria, with an expectation that such an approach will be implemented by licensees of the remaining operating reactors not yet systematically analyzed in an equivalent or superior manner.

For advanced nuclear power plants, including both the evolutionary and passive designs, the NRC concluded that vendors should address severe accidents during the design stage. Designers can take full advantage of the insights gained from such input as probabilistic safety assessments, operating experience, severe accident research, and accident analysis by designing features to reduce the likelihood that severe accidents will occur and, in the unlikely occurrence of a severe accident, to mitigate the consequences of such an accident. Incorporating insights and design features during the design phase is much more cost effective than modifying existing plants.

Summary—Severe Accident Policy Statement
Although the Severe Accident Policy Statement specifies that severe accident design features could be imposed on operating reactors using the established backfit process, the importance of the qualitative factors suggests a need to revisit portions of the current regulatory framework (including the Severe Accident Policy Statement). The status quo option best fits the current policy statement and its traditional application.

Option 1	⬆⬆
Option 2	
Option 3	
Option 4	⬆

5.11 <u>International Practices</u>

A description of the staff's collection and assessment of information from various countries related to decisions on filtered venting systems is provided in Enclosure 3. As discussed in that enclosure, the majority of countries with BWRs using Mark I and Mark II containment designs have modified or plan to modify the designs to include filtered containment venting systems. In addition, some countries are requiring filtered venting systems on other reactor containment designs. As previously mentioned, in the discussions on determining whether a proposed change meets the standard of a substantial increase in safety, the Commission stated:

> …The approach is also flexible enough to allow for arguments that consistency with national and international standards, or the incorporation of widespread industry practices, contributes either directly or indirectly to a substantial increase in safety. Such arguments concerning consistency with other standards, or incorporation of industry practices, would have to rest on the particulars of a given proposed rule…

Although no particular international standard exists that calls specifically for filtered vents for Mark I and Mark II containments, Option 3 is consistent with the general standards and guides that call for improving the ability of containments to contain radioactive materials during severe accident conditions. Pursuing Option 3 would also place the United States among the majority of countries that have required filtered venting systems, and maintain its stature as a leader in nuclear safety. Another significant benefit from the international experience is the development and installation of various filtering systems. This lessens concerns that requiring filtered vents would necessitate research and development programs to design and test a new technology.

Many countries that have pursued filtered venting systems have done so coincident with the development of the defense-in-depth system described in guidance from the International Atomic Energy Agency (IAEA) and Western European Nuclear Regulators' Association (WENRA). This defense-in-depth logic includes a specific level for dealing with severe accidents and minimizing the need to displace populations near nuclear power plants. The logic is shown below, along with the corresponding regulatory structure in the United States.

	International System of Defense in Depth	Corresponding US Considerations	
1	Normal Operations	Normal Operations	Risk Informed Licensing Reactor Oversight Process
2	Anticipated Operational Occurrences	Anticipated Operational Occurrences	
3	Design Basis Accidents	Design Basis Accidents	
	Design Extension Events	Beyond Design Basis Events (Design Extension considered under Recommendation 1)	
4	Severe Accident	Safety Goal Policy Statement **Severe Accident Policy Statement** - **Operating Plants** - New Reactors	
5	Emergency Planning	Emergency Planning	

Figure 3: Comparison of Defense-in-Depth Approaches

As shown above, the regulatory systems are similar in most areas, but they differ in the treatment of beyond-design-basis and severe accidents. The Severe Accident Policy Statement is discussed as a separate qualitative factor.

Summary—International Practices

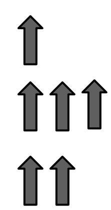
As discussed in Enclosure 3, most countries that have reactors with Mark I or Mark II containments require or plan to require filtered vent systems. Although not specifically included in an international standard, a desire to maintain consistency with international practices would support taking action (in order of Option 3, Option 4, and then Option 2)

Option 1

Option 2

Option 3

Option 4



6. SUMMARY

Based on the quantitative and qualitative considerations discussed above, some of the more significant positive and negative attributes (i.e., pros and cons) for each of the options are as follows:

Option 1: Status Quo: Continue with the implementation of EA-12-050 for reliable Hardened vents to reduce the likelihood of core damage and failure of BWR Mark I and Mark II containments and take no additional action to improve their ability to operate under severe accident conditions or to require the installation of a filtered vent system

Pros:

- Consistent with Severe Accident Policy Statement that no additional measures are needed for operating reactors
- No additional costs to industry and the NRC
- Consistent with quantitative cost-benefit analysis findings using current framework and assumptions
- Consistent with findings from SAMA analyses

Cons:

- Maintains defense-in-depth “imbalance” between prevention of core damage and mitigation (i.e., while measures have been taken to reduce chances of core melt, high conditional failure probability remains for containment if core melt does occur)
- Of the four options, results in highest doses and highest economic consequences in the unlikely event of a severe accident
- Inconsistent with international practices that emphasize reliable containment as a critical function

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

Pros:

- Supports severe accident management by improving hydrogen control, pressure control (supports low-pressure injection), and minimizing radiation releases to reactor building
- Reduces doses to emergency workers (relative to an uncontrolled containment failure)
- Consistent with industry approach in EPRI study (without performance measure)
- Involves limited changes to existing EA-12-050, related guidance, and implementation schedules

Cons:

- Would involve significant release of radioactive materials when venting operations are performed during severe accident conditions

- Uncertainty of decontamination factor is large and highly dependent on the specifics and timing of the accident scenario

- Does not resolve issues about the use of drywell path for venting

- Not supported by quantitative cost benefit analysis using current framework and assumptions

- Could be viewed as inconsistent with both the NRC's Severe Accident Policy Statement and international practices

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

Pros:

- Supports severe accident management by improving hydrogen control, pressure control (supports low-pressure injection), and minimizing radiation releases to reactor building

- Reduces doses to emergency workers (relative to an uncontrolled containment failure) without increasing offsite releases

- Ensures high decontamination factors that are independent of specifics of the accident sequence (excluding containment bypass sequences)

- Confidence in decontamination factor supports use of system from both wetwell and drywell

- Improves defense-in-depth balance between prevention and mitigation (i.e., addition of filter directly addresses containment performance issues)

- More consistent with international approach to containment reliability

Cons:

- Not supported by quantitative cost benefit analysis using current framework and assumptions (highest cost of proposed options)

- Could be viewed as inconsistent with NRC's Severe Accident Policy Statement

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

Pros:

- Consistent with Commission policy to encourage use of performance-based requirements

- Possible to integrate with the NRC's resolution of other regulatory policy issues and development of revised guidance on defense in depth and industry's evaluation of strategies and technologies

- Improves defense-in-depth balance between prevention and mitigation

Cons:

- Requires development of performance standards and acceptable methods for demonstration of compliance (difficult task given high uncertainties, limited testing, and nature of severe accident conditions)

- Would likely extend the resolution of this issue by several years

- Large uncertainties in both the NRC and industry costs and schedules

6.1 Conclusion

Based on its regulatory analyses, the staff concludes that installation of engineered filtered venting systems for Mark I and Mark II containments is the option that would provide the most regulatory certainty and the timeliest implementation. The NRC performed a cost-benefit assessment considering both quantifiable and qualitative factors after NRC senior managers determined that the possible imposition of requirements for severe accident capable or filtered venting systems satisfy the "substantial safety improvement" standard of 10 CFR 50.109. 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, revising assumptions related to event frequencies or event consequences to address the significant uncertainties in modeling severe accident scenarios could lead to a conclusion that the proposed options are at least marginally cost-effective. In addition, the majority of the qualitative factors discussed in Section 5 (1) support pursuing an improved venting system for BWRs with Mark I or Mark II containments to address specific design concerns (e.g., high conditional failure probability for containment failure given core melt); (2) support severe accident management functions by preventing releases of radioactive materials, hydrogen, and steam into the reactor building or other locations on the site; (3) minimize the contamination of the site environs; and (4) reduce the reliance on emergency planning for protection of public safety. The staff concludes that considering both the quantitative and qualitative factors shows the direct and indirect costs associated with Options 2 and 3, and most likely Option 4, are cost-justified in light of the substantial increase in the overall protection of the public health and safety that is provided by addressing severe accident conditions for BWRs with Mark I and Mark II containments. In addition, the NRC staff finds that the combination of quantitative and qualitative factors (e.g., providing improved defense in depth) best supports the installation of engineered filtered venting systems at BWRs with Mark I and II containments.

ENCLOSURE 2
BWR MARK I AND MARK II CONTAINMENT REGULATORY HISTORY

CONTENTS

1. INTRODUCTION

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 has always been 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. This philosophy was described in a report prepared in 1965 for the U.S. Atomic Energy Commission (AEC) by Oak Ridge National Laboratory (ORNL) that compiled the early practices and approaches for containment designs. The report provided the following summary:

> The need for a containment system in the large power reactor installation is well established by convention and precedent in the United States, and the specific design requirements are determined by the reactor safety analysis. Philosophically, containment is provided so that the risk that cannot be disassociated from the operation of a particular reactor can be reduced to acceptable proportions with respect to the corresponding gain that is expected to result from its operation. However, such a balance of gain versus risk is impossible to attain on a quantitative basis, and only the risk enters into the evaluation that is made in connection with every reactor safety analysis. The specific function of the containment system is to reduce the consequences of the maximum credible accident so that a particular facility may fulfill siting requirements as defined in the Code of Federal Regulations. On this basis, containment systems may be called upon to effect a reduction in the activity released in an accident by a factor of 10^2 to 10^5.
>
> The accident that could occur and would have associated with it the most severe set of consequences as far as the radiation exposure of offsite personnel is termed the "maximum credible accident" (mca). Although this accident is a characteristic of a given plant, there are only two types of accidents that comprise the mca. The first is the loss of coolant accident, with subsequent core melting or possible nuclear excursion and release of fission products. The second is the fuel handling accident in which a fuel element, or assembly, is dropped or allowed to fail in such a way that its fission products are released. After these initiating events occur, the released fission products disperse through the system and leak to the environment at some rate determined by the containment vessel in question.

For currently operating plants, this barrier is provided by containments that include either (1) a large enough 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. Boiling-water reactors (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. The pressure suppression designs were summarized as follows in the early ORNL report:

> In an effort to reduce the cost of containment, the concept of pressure suppression has been employed with water-cooled reactors. In principle, this technique is especially suited to water-cooled reactors, since the major portion of

> the energy released upon occurrence of an mca is in the form of saturated steam, which may be removed by condensation and thereby greatly reduce the final pressure to be withstood by the containment building. This scheme uses the “dry well” and vent piping to direct the steam that is released into the water of the suppression pool, where the steam is condensed and fission products may be partially removed.

As mentioned above, the primary focus of containment designs was, and largely remains, the demonstration that it addresses the “maximum credible accident” and limits the potential exposure of the public from radioactive materials. The maximum credible accident and its role in siting decisions and containment functions was described as follows in another early and key guidance document, TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” (AEC-1962):

> In evaluating proposed reactor sites, the basic safety questions involve the possibility of accidents which might cause radioactivity release to areas beyond the site, the possible magnitudes of such releases and the consequences these might have. Practically, there are two difficult aspects to the estimation of potential accidents in a proposed reactor which affect the problem of site evaluation.
>
> (1) The necessity for site appraisal arises early in the life of a project when many of the detailed features of design which might affect the accident potential of a reactor are not settled.
>
> (2) The inherent difficulty of postulating an accident representing a reasonable upper limit of potential hazard.
>
> In practice, after systematic identification and evaluation of foreseeable types of accidents in a given facility, a nuclear accident is then postulated which would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility. Such an accident has come to be known as the "maximum credible accident".
>
> For pressurized and boiling water reactors, for example, the "maximum credible accident" has frequently been postulated as the complete loss of coolant upon complete rupture of a major pipe, with consequent expansion of the coolant as flashing steam, meltdown of the fuel and partial release of the fission product inventory to the atmosphere of the reactor building. There may be other combinations of events which could also release significant amounts of fission products to the environment, but in every case, for the events described above to remain the maximum credible accident the probability of their occurrence should be exceedingly small, and their consequences should be less than those of the maximum credible accident. In the analysis of any particular site-reactor combination, a realistic appraisal of the consequences of all significant and credible fission release possibilities is usually made to provide an estimate in each case of what actually constitutes the "maximum credible" accident. This estimated or postulated accident can then be evaluated to determine whether or not the criteria set out in 10 CFR 100 are met. As a further important benefit,

> such systematic analyses of potential accidents often lead to discovery of ways in which safeguards against particular accidents can be provided.
>
> Since a number of analyses have indicated that the pipe rupture-meltdown sequence in certain types of water cooled reactors would result in the release of fission products not likely to be exceeded by any other "credible" accident, this accident was designated the "maximum credible accident" (MCA) for these reactors. The remainder of this discussion will refer chiefly to this type of reactor and this type of accident. Corresponding maximum credible accidents can by similar analyses be postulated for gas-cooled, liquid metal cooled, and other types of reactors.

The above discussion remains largely relevant today as the limits in Title10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," are unchanged, and some plants continue to be evaluated using the estimates in TID-14844 to assess the adequacy of containment designs.[1] Other aspects of the containment design and evaluation are also derived from the establishment of a large pipe break as the maximum credible accident. Such design requirements include the ability of structures, systems, and components to withstand the pressures, temperatures, and hydrodynamic forces associated with pipe breaks within the containment, as well as withstanding external hazards such as seismic events.

There have been several significant issues related to the performance of BWR containments during design-basis accidents. These problems and their resolution are discussed in Section 4, "Other Design Issues," but are not related to the primary issue of this paper, which deals with beyond-design-basis accidents and the importance of containment venting during such scenarios.

In SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988, the U.S. Nuclear Regulatory Commission (NRC) staff presented to the Commission its plan to evaluate potential generic severe accident containment vulnerabilities in a research effort entitled the containment performance improvement (CPI) program. This effort was predicated on the presumption that there are generic severe accident challenges to each light water reactor (LWR) containment type that should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features were warranted, and to confirm the adequacy of the existing Commission policy. These assessments were needed because of the uncertainty in the ability of LWR containments to successfully survive some severe accident challenges, as indicated by the results documented in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,". All LWR containment types were assessed in the CPI program, beginning with the boiling-water reactors (BWRs) with Mark I containments. The 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," and SECY-89-017, "Mark I Containment Performance Improvement Program," dated January 23, 1989. The potential improvements for Mark II containments were published in NUREG/CR-5528, "An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements."

---

[1] Licensees are allowed but not required by NRC regulations defined in 10 CFR 50.67. "Accident Source Term," to use revised accident source terms to take advantage of research and knowledge gained since the issuance of TID-14844.

2. BWR MARK I AND MARK II CONTAINMENTS

The key design attributes of Mark I and Mark II containments relevant to the need for containment venting during severe accidents such as Fukushima are: (1) the containment free gas volumes are relatively small compared to other light-water reactors, so gas and steam buildup in containment will cause the pressure to rise more dramatically, (2) BWR reactor cores have about three times the zirconium inventory compared to pressurized-water reactors (PWRs) with comparable power levels, so there is a greater potential to generate significant amounts of hydrogen gas which also will increase containment pressures. These design attributes, in comparison with other containment types, are illustrated in Figures 1 and 2.

2.1 Mark I Containment Designs

As shown in Figure 3, the Mark I containment design is a drywell in the shape of an inverted common incandescent light bulb containing the reactor vessel and primary piping attached with several large vent pipes to a torus shaped suppression chamber located below the drywell. The steam escaping from the break in the reactor coolant piping would vent, along with the drywell atmosphere, down into the suppression chamber. It would be distributed through a header to many downcomer pipes whose open ends were submerged in the suppression pool, which fills about half the suppression chamber.

Presently, worldwide a total of 37 commercial nuclear power units (reactors) use a Mark I-type (drywell /toroidal suppression pool) pressure suppression containment. Twenty-three—or roughly 60 percent—are licensed by the NRC to operate in the United States. All but one (Fermi 2) have been granted a license extension, with the earliest expiring in 2029 (Dresden 2) and the latest expiring in 2038 (Hatch 2). Twenty have been granted a power uprate between 1.5 percent (Pilgrim) and 20 percent (Brunswick 1, 2). Additional information is provided in Table 3.

Table 1. BWR Mark I Containments by Country

Country	Number	Name
US	23	See Table 3
Japan	8	Fukushima I 1-5 Hamaoka 1 Shimane Tsuruga
India	2	Tarapur 1,2
Taiwan	2	Chinsan 1,2
Spain	1	Santa Maria de Garona
Switzerland	1	Muehleberg

The General Electric (GE) BWR Mark I containment was an early design and evolutionary step in the development of the containment technology seen in the industry today. As knowledge and experience were acquired, shortcomings in the understood safety margins were identified and assessed. Over time, extensive improvement modifications have been made to restore those safety margins (See Section.4 in this Enclosure).

The Mark I pressure-suppression concept containment design was based on experimental information obtained from testing performed for the Humboldt Bay and Bodega Bay Power Plants. (The Humboldt Bay Nuclear Power Plant was rated for 63 megawatts electric (MWe) operated from August 1963 to July 1976 just south of Eureka, California. The Bodega Bay Power Plant was to be rated for 313 MWe, but construction at the site 50 miles north of San Francisco was cancelled about 1964.)

The purpose of these initial tests, performed from 1958 through 1962, was to demonstrate the viability of the pressure-suppression concept for reactor containment design. The tests were designed to simulate loss-of-coolant-accidents (LOCAs), with breaks in piping sized up to approximately twice the cross-sectional break area of the design-basis LOCA. The tests were instrumented to obtain quantitative information for establishing containment design pressures. The data from these tests were the primary experimental bases for the design and the initial staff approval of the Mark I containment system. Dresden Generating Station (also known as Dresden Nuclear Power Plant or Dresden Nuclear Power Station) was the first privately financed nuclear power plant built in the United States. Dresden Unit 1, which had a Mark I type containment, received a construction permit in 1959, and was decommissioned in 1978.

Given that the primary function of this containment is to contain radioactive material following an accident, designers and regulators are faced with a challenge when it comes to maintaining the integrity of the containment when it is challenged by high pressures. Historically, primary containment pressure control to prevent structural failure, and thus unrecoverable loss of the primary function, was to be achieved by multiple, diverse active and passive systems (spray, fan-coolers, vents to suppression pools) and not by a simple relief valve or rupture disk discharging containment atmosphere directly to the environment as would be the practice for most other pressure vessels Thus, the American Society of Mechanical Engineers (ASME) created an exception to the general practice of requiring a passive relief device in the ASME Boiler and Pressure Code Section III Article NE-7000, which states:

> A containment vessel shall be protected from the consequence arising from the application of conditions of pressure and coincident temperature that would cause the Service or Test Limits specified in the Design Specification to be exceeded. Pressure relief devices are not required where the Service or Test Limits specified are not exceeded. It is recognized that the fundamental purpose of a containment vessel may be nullified by the incorporation of pressure relief valves discharging directly into the environment.

However, a controlled (and potentially filtered) release was identified as a favorable alternative to catastrophic failure of the containment. Subsequent to the Three Mile Island Unit 2 nuclear plant core melt event in 1979, NUREG-0585, “TMI-2 Lessons Learned Task Force Final Report,” October 1979, stated:

> Available studies indicate that controlled venting of the containment to prevent failure due to overpressure could be-an effective means of delaying ultimate containment failure by melting through. If appropriately filtered to partially decontaminate the gases that would be released in order to avoid over-pressurization, such venting may significantly reduce the consequences and risk from core-melt accidents… It appears to us that sufficient studies have been completed to support a preliminary conclusion that controlled filtered venting of containments is an effective and feasible means of mitigating the consequences of core-melting.

As probabilistic risk assessment (PRA) methods continued to mature, the Reactor Safety Study, "An Assessment of Risks in U.S. Commercial Nuclear Power Plants [NUREG-75/014 (WASH 1400)]," found that, for the Peach Bottom BWR Mark I nuclear plant, even though the core melt probability was relatively low, the containment could be severely challenged if a large core melt occurred. Based on this conclusion, and reinforced by the anticipation of similar findings (subsequently confirmed) in the draft Reactor Risk Reference Document (NUREG-1150, February 1987) a five element program was proposed in June 1986 to enhance the performance of the BWR Mark I containment. After the initial proposal, the staff held two separate meetings in early 1987 with researchers representing NRC contractors and industry. There was a wide range of views expressed regarding accident phenomenology as well as the efficacy of the various improvements. In view of the lack of technical consensus on the effectiveness of the proposed improvements, the staff decided to undertake additional efforts. In July 1987, the staff briefed the Commission on an integrated approach to resolve all severe accident issues, including matters relating to BWR Mark I containments. The integrated approach was to be comprised of four main programs: (1) the Individual Plant Evaluation Program (IPE), (2) the Containment Performance Program, (3) a program to improve plant performance, and (4) a program to implement guidance on Severe Accident Management Strategies.

The staff proposed a broad-based plan in December 1987 to address the performance issues of Mark I containments (SECY-87-297). The proposal listed several, relatively low-cost improvements whose purpose was to substantially mitigate potential offsite releases. This list of possible improvements included: hydrogen control, alternate water supplies for the containment spray system, venting, core debris control, enhancing reactor building fission product attenuation, basemat isolation, improving the automatic depressurization system, and improving existing emergency procedures and training to include coping with severe accidents.

SECY-87-297 also laid out a two-stage strategy to attempt resolving such a large-scale set of technical issues. The first stage would consist of characterizing an issue and performing parametric studies and experimental assessments to assist in focusing on the most relevant technical aspects. After initial issue characterization, a meeting would be held with representatives from the staff, contractors, the industry, and other experts and interested members of the public on each issue. During the second stage, the staff would evaluate and sort each issue into one of three categories: (1) resolved or unimportant, (2) potentially resolvable by future research, or (3) candidates for regulatory initiatives.

The staff returned to the Commission in January 1989 to present recommendations on Mark I containment performance improvements and other safety enhancements (SECY-89-17, "Mark 1 Containment Performance Improvement Program"). In that paper, the staff described their findings associated with examining six areas of potential improvement for Mark I containments. These were: (1) hydrogen control, (2) alternate water supply for reactor vessel injection and containment drywell sprays, (3) containment pressure relief capability (venting), (4) enhanced reactor pressure vessel (RPV) depressurization system reliability, (5) core debris controls, and (6) procedures and training. Each area was evaluated to determine the potential benefits in terms of reducing the core melt frequency, containment failure probability, and offsite consequences.

The staff concluded there was no significant risk reduction associated with additional hydrogen control (beyond the existing rule, see Hydrogen Control section below for details). The primarily reason was because, during a severe accident, reactor pressure is anticipated to increase, releasing steam and noncondensable gases into the containment. This would increase containment pressure, preventing ingress of air. Therefore, the containment atmosphere would not become de-inerted for an extended period of time. Since offsite supplies of nitrogen could readily be obtained during this period, an onsite backup supply of nitrogen would not significantly reduce risk.

Additionally, the staff determined that more research was necessary to ensure the technical feasibility of core debris controls (e.g. curbs in the drywell or curbs or weir walls in the torus room under the wetwell). The design and installation costs, as well as the occupational exposure during installation, were also significant deterrents from pursuing further actions in this improvement area.

Aside from these two exceptions, the staff provided cost-justification for, and recommended implementation of, all the aforementioned improvements including: (1) improved hardened venting capability, (2) improved RPV depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) emergency procedures and training.

In the subsequent SRM, however, the Commission concluded that the majority of the staff's recommended safety improvements would be evaluated by licensees as part of the IPE Program. The only exception was the hardened vent capability recommendation. The Commission directed the staff to approve installation of hardened vents under the provisions of 10 CFR 50.59, "Changes, Tests, and Experiments," for licensees that would voluntarily implement this improvement and perform a back-fit analysis for requiring a hard vent installation at those plants who declined voluntary installation. Thus, Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," was issued in September 1989 providing an example of an acceptable design that used the suppression pool to achieve as much reduction in effluent radioactivity as possible without the cost of an external filter making the change more cost-beneficial.

In response to the issuance of the generic letter, all Mark I licensees installed a version of a hardened vent under 10 CFR 50.59. The Boiling Water Reactor Owners' Group (BWROG) developed a general design criteria document that was subsequently approved by the staff (with clarifications).

The hardened vent was specifically to provide an exhaust line from the wetwell vapor space to a suitable release point (e.g. stack, reactor building or turbine building roof). The basic design objective of the hardened vent was to mitigate the loss of decay heat removal accident sequence. As such, the piping was designed (sized) to accommodate a steam flow equivalent of 1 percent decay heat power assuming a pressure equal to the primary containment pressure limit (PCPL), and not designed for operation during a severe accident.

The staff requested that the capability for the initiation (although not termination) of utilizing the hardened vent be in the control room, and that radiation monitoring devices be required to alert control room operators of radioactive releases during venting. It was proposed in the staff recommendation in SECY 89-17 that the hardened vent isolation valves be capable of being opened from the control room under station blackout conditions beyond the then-established coping time; however, the generic letter only requested that the licensee include costs for electrical modifications in a plant-specific basis for why the vent was not cost beneficial if a vent was not voluntarily installed. The installed vents in most cases were dependent on alternating current power.

The newly installed hardened vents were subject to pre-existing technical specifications for containment isolation valves and containment integrity, but the system itself had no imposed limiting conditions of operation (LCO) or surveillance requirements. The valves were, however, subject to the local leak rate testing and inservice testing requirements (10 CFR 50.54(o) and 10 CFR 50.55a(f), respectively) of all containment penetrations and isolation devices.

2.2 Mark II Containment Designs

The Mark II containment concept (Figure 4) evolved the drywell and suppression pool to a simpler truncated cone over the cylindrical suppression chamber. Currently, there are a total of 17 commercial nuclear power units using a Mark II-type pressure suppression containment worldwide. The NRC has granted operating licenses to eight of these BWRs with Mark II containments on five different sites. Columbia, Nine Mile Point 2, and Susquehanna 1&2, have also been granted license extensions, and the application for license extension at Limerick was received by the NRC in June 2011.

The details of the design of the Mark II containment dry well floor directly below the reactor vessel, the in-pedestal region, greatly affects the accident progression, and thus the uncertainty in predicting consequences of a severe accident. The design of this in-pedestal region varies from plant to plant. The designs of the Shoreham and Nine Mile Point 2 containments include downcomers inside the pedestal region. At La Salle, Columbia and Nine Mile Point 2, the in-pedestal region is at a lower elevation than the ex-pedestal drywell floor. Columbia has two sumps cast into the in-pedestal floor. All Mark IIs, with the possible exception of the two Susquehanna units, have drain lines through the dry well floor in this area. Failure of a drywell floor penetration, or the floor itself (by core-concrete attack or from excessive differential pressure across the floor) would allow fission products in the dry well to bypass the wet well, thus resulting in no decontamination before release by a hardened vent from the wet well air space.

Table 2. BWR Mark II Containments by Country

Country	Number	Name
U.S.	8	Columbia LaSalle 1,2 Limerick 1,2 Nine Mile Point 2 Susquehanna 1,2
Japan	7	Fukushima I 6 Fukushima II 1-4 Hamaoka 2 Tokai 2
Mexico	2	Laguna Verde 1,2

In July 1990, the NRC published NUREG/CR-5528, “An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements.” The conclusions of this containment performance improvement program study, with respect to containment venting, are excerpted in the following paragraphs.

> Severe accident sequences at the Mark II plants can be grouped into two general categories: one where containment integrity is challenged before core

> degradation, the other where core damage precedes any threat to containment integrity. In the first category, which includes loss of long-term containment heat removal with reactor scram (TW) and anticipated transient without scram (ATWS) sequences, the challenge to containment is from overpressurization due to inadequate containment heat removal. In the second category, which includes station blackout (SBO) and other transients where reactor scram occurs, the challenge can be from either overpressurization at or near the time of reactor vessel failure or overpressure or overtemperature failure several hours after vessel failure. Potential improvements addressing the first category of containment challenges include containment pressure control. Examples could include venting from the wetwell through a hardened vent pipe, and containment pressure control and fission product scrubbing, such as the use of containment sprays with a backup water supply. A hardened vent line would allow excess energy in the containment to be rejected to the environment, while avoiding concerns associated with venting through existing "soft" heating, ventilating, and air conditioning (HVAC) ductwork. However, with the high estimated probability of suppression pool bypass in the base case via failure of in-pedestal drain lines shortly after vessel breach, the vent systems would need an external filter, such as the Swedish multiventuri scrubbing system, to prevent a severe offsite release of fission products. Containment sprays could be used to condense steam in the containment, thus delaying overpressurization failure.
>
> For the second category of containment challenges (core melt before containment failure), potential improvements include: (a) containment pressure control, such as a hardened vent from the wetwell, (b) improved means of depressurizing the reactor, such as enhancements to the automatic depressurization system (ADS) and the safety/relief valves (SRVs), (c) containment temperature control and fission product scrubbing, such as containment sprays with a backup water supply and external cooling of the drywell head, and (d) mitigation of the fission product releases, such as the use of reactor building fire protection sprays to enhance fission product retention in the secondary containment. The hardened vent line (with or without an external filter) could be used to mitigate late overpressurization challenges.

The rationale for making an external filter optional for late containment failures at the time of the report was that the release would be less threatening than an early release and would likely not result in prompt fatalities if evacuation was not successful; the release to the environment would still be substantial. In summary, an external filter was indicated for the dominant failure modes of the Mark II containment.

The report summarized the benefits of a filtered containment venting system as:

(1) prevents overpressure failures for transients with scram
(2) delays overpressure failures for ATWS
(3) reduces base pressure through preemptive (early) venting before core damage
(4) mitigates hydrogen burns in secondary containment
(5) ensures scrubbing of aerosol releases
(6) is unaffected by suppression pool bypass

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, where the wet well might provide some scrubbing of a release, a wet well vent was recommended, despite the potential for a low degree of decontamination in the wet well. For the Mark II, where risk was dominated by bypass of the wet well and thus no wet well decontamination at all, a vent was not recommended without an external filter. However, a filter was judged to not be cost effective based on published cost estimates at the time, e.g., multiventuri scrubber system (MVSS) approximately $5 million plus the cost of the vent.

3. HYDROGEN CONTROL INSIDE CONTAINMENT—MARK I AND MARK II

One of the key considerations associated with the protection of containment integrity is the control of the hydrogen which is produced by the coolant-zirconium reaction during a severe accident. Hydrogen gas can also be produced by radiolysis of the coolant and by core-concrete interaction; however, the main contributor to the production of hydrogen is the aforementioned coolant-zirconium reaction.

In October 1978, the NRC adopted a new rule, 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” specifying the standards for primary containment combustible gas control systems. The rule required the applicant or licensee to show that during the time period following a postulated LOCA, but prior to effective operation of the combustible gas control system, either: (1) an uncontrolled hydrogen-oxygen recombination would not take place in the containment, or (2) the plant could withstand the consequences of an uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions could be shown, the rule required that the containment be provided with an inerted atmosphere to provide protection against hydrogen burning and explosion. The rule assumed a release of hydrogen corresponding to 5 percent oxidation of the fuel cladding in determining compliance.

Subsequently, the NRC reassessed the vulnerability of various containment designs to hydrogen burning and adopted amendments to 10 CFR 50.44, including one in 1981 that added a requirement for an inerted atmosphere for BWR Mark I and Mark II containments. Results of research at the time were incorporated into various studies (e.g., NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light-water reactors. The result of these studies was an improved understanding of combustible gas behavior during severe accidents and confirmation that the hydrogen release postulated from a design-basis LOCA was not risk significant because it was not large enough to lead to early containment failure, and that the risk associated with hydrogen combustion was from beyond-design-basis (e.g., severe) accidents. Combustible gas generated from design-basis accidents was not risk-significant for any containment type, given intrinsic design capabilities or installed mitigative features. The studies also concluded that combustible gas generated from severe accidents was not risk significant for Mark I and II primary containments, provided that the required inerted atmosphere was maintained.

A September 2003 amendment to 10 CFR 50.44 retained the requirement to inert Mark I and II type containments while removing the requirement for hydrogen recombiners or backup hydrogen purge systems. Given the large zirconium inventory in these reactors and their relatively small primary containment volumes, these containments, without inerting, would have a high likelihood of failure from hydrogen combustion due to the potentially large concentration of hydrogen that a severe accident could cause. The regulatory analysis found the cost of maintaining the recombiners exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late time frame, well beyond 24 hours, by the long-term generation of oxygen through radiolysis. The regulatory analysis for this rulemaking found the cost of maintaining the recombiners, and thus likely also the hydrogen purge systems, exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late timeframe. The rule retained existing requirements for ensuring a mixed atmosphere; inerting Mark I and II containments, and hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The technical bases for the regulations were established from experience at

Three Mile Island along with bounding estimates for the amount of hydrogen likely to be generated by a severe core damage accident.

This rule also specified requirements for combustible gas control in future water-cooled reactors which are similar to the requirements specified for existing plants. However, a key difference is the need to accommodate an equivalent amount of hydrogen as would be generated from a 100 percent (active fuel) clad-coolant reaction. Particularly, if a containment does not have an inerted atmosphere, it must limit hydrogen concentrations in containment during and following an accident that releases hydrogen (equivalent to 100 percent fuel-coolant reaction) when uniformly distributed to less than 10 percent (by volume); and maintain containment structural integrity and appropriate accident mitigating features.

As stated in the rule, all BWRs with Mark I or Mark II type containments must have an inerted atmosphere. This concept reduces oxygen enough to suppress combustion; thereby, a hydrogen generation limit is not specified. The result of a hydrogen combustion event is characterized as a relatively sharp pressure pulse, and thus the intent of rule precludes this occurrence inside containment; but does not recognize the slow buildup of containment pressure as a result of the hydrogen gas generated by postulated severe core damage accidents. Therefore, containment pressure control is addressed in the severe accident management guidelines (SAMG). Essentially, pressure control for severe accidents in Mark I and Mark IIs are also related to hydrogen control for the containment.

4. OTHER DESIGN ISSUES—MARK I AND MARK II

The following issues deal primarily with design-basis issues, which are not directly relevant to the topic of containment venting. They do involve considerations of defense in depth and some early recognition that pressure suppression containments involved additional complexities compared to large dry containments and introduced concerns, such as bypassing the pressure suppression features. Such a bypass would lead to rapid over-pressurization given the smaller volumes of these containment designs (e.g., the relationships shown in Figure 1).

Hydrodynamic Forces

Between 1972 and 1974, the Mark III containment system design was undergoing large-scale testing of the new suppression pool hydrodynamic loads which were identified for the postulated LOCAs. GE was testing the Mark III containment concept at that time because of configurational differences between the previous containment concepts and the Mark III design.

The Mark I containment design is a drywell in the shape of an inverted common incandescent light bulb containing the reactor vessel and primary piping attached with several large vent pipes to a torus-shaped suppression chamber located below the drywell. The steam escaping from the break in the reactor coolant piping would vent, along with the drywell atmosphere, down into the suppression chamber where it would be distributed through a header to many downcomer pipes whose open ends were submerged in the suppression pool, which filled about half the suppression chamber. The Mark II containment concept evolved the drywell and suppression pool to a simpler truncated cone over the cylindrical suppression chamber. The Mark III containment concept involved more fundamental changes in the containment layout with the drywell being completely within the suppression chamber which formed the entire containment boundary.

More sophisticated instrumentation and data analysis was available for the Mark III tests and led to a better understanding of short-term dynamic effects of drywell air being forced into the suppression pool in the initial stage of the postulated LOCA. This air injection into the suppression pool water results in a pool swell event of short duration but with substantial forces associated with the water impacting the suppression chamber walls and internal structures. Additional LOCA-related dynamic load information was obtained from foreign testing programs for similar pressure-suppression containments, including the occurrence of oscillatory condensation loads during the later stages of a postulated LOCA blowdown. Actual experience at operating plants indicated that reactor vessel safety/relief valves (SRVs) discharging via tailpipes to the suppression pool would cause oscillatory hydrodynamic loads on the suppression chamber.

Consequently, in February and April 1975, the NRC transmitted letters to all utilities owning BWR facilities with the Mark I containment system design, requesting that the owners quantify the hydrodynamic loads and assess the effect of these loads on the containment structure. As a result of these letters from the NRC, and recognizing that the additional evaluation effort would be very similar for all Mark I BWR plants, the affected utilities formed an “ad hoc” Mark I Owners Group with the objective to determine the magnitude and significance of these dynamic loads and identify courses of action needed to resolve outstanding safety concerns. This task was divided into a short-term program (STP) to be completed in early 1977 and a long-term program (LTP). The STP objective was to verify that each Mark I containment system would maintain its integrity and functional capability when subjected to the most probable loads induced by a postulated design-basis LOCA, and to verify that licensed Mark I BWR facilities could continue to operate safely, without endangering the health and safety of the public, while working on the comprehensive LTP. The STP acceptance criteria were based on providing adequate margins of safety, i.e., a safety-to-failure factor of 2, to justify continued interim operation of the plants.

The staff's conclusions relative to the STP are described in NUREG-0408, “Mark I Containment Short-Term Program Safety Evaluation Report,” issued December 1977. The objective of the LTP was to establish design-basis (conservative) loads appropriate for the anticipated life of each Mark I BWR facility and restore the originally intended design-safety margins. The requirements resulting from the LTP (described in NUREG-0661 “Mark I Containment Long-Term Program,” issued July 1980) were used by each BWR/Mark I licensee to perform a plant-specific analyses and identify plant modifications needed to restore margins of safety in the-containment design. Modifications included:

- Torus–Vent System—Considerable additional steel in the way of reinforcement for ring girders, miter joints, vent header and downcomers, internal catwalk and conduit. Torus temperature monitoring instrumentation system. Torus tie-downs and dynamic motion restraints (snubbers).

- Torus attached piping—Considerable additional steel in the way of reinforcement of torus attached piping at the penetration area and supports within the torus.

- SRVs—Added T-quencher spargers at the discharge point within the suppression pool, vacuum breakers for the discharge lines and control scheme circuitry to prevent immediate reopening of an SRV before the vacuum breakers function. Added SRV position monitoring instrumentation.

The Mark II containment suppression chamber dynamic load re-evaluation followed a similar course. NUREG-0487, “Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria,” was issued in October 1978. NUREG-0487, Supplement 2, issued February 1981, completed the lead plant program after addressing the condensation–oscillation or chugging loads. NUREG-0808, “Mark II Containment Program Load Evaluation and Acceptance Criteria,” issued August 1981, provides a discussion of LOCA-related suppression-pool hydrodynamic loads in the Mark II containment design and staff acceptance criteria for pool-swell loads from the lead-plant program and new criteria for steam loads developed in the LTP.

Emergency Core Cooling System Suction Strainers

In May 1996, NRC Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” was issued requesting BWR operators to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of emergency core cooling system (ECCS) suppression pool suction strainers by debris generated during a LOCA. The bulletin cited an event at a Swedish BWR, Barsebäck 2, which involved plugging of two containment spray system suction strainers with mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened.

Subsequent to this event, the NRC issued Information Notice (IN) 92-71, “Partial Plugging of Suppression Pool Strainers at a Foreign BWR,” in September 1992, to alert addressees of the potential for loss of ECCS that was identified as a result of the Barsebäck 2 event. It was expected that recipients would review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems.

Two earlier events involving the clogging of ECCS strainers had occurred at the Perry Nuclear Power Plant, a domestic BWR in 1993. Based on these earlier happenings, the NRC issued Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," in May 1993. In it, the staff requested licensees to identify fibrous air filters or other temporary sources of fibrous material, not designed to withstand a LOCA, which were installed or stored in primary containment. The licensees were to take any immediate compensatory measures to assure the functional capability of the ECCS and promptly remove any such material.

Because of the apparent trend identified in these events, the staff conducted a detailed study of a reference BWR 4 plant with a Mark I containment and issued NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," in October 1995. A suction strainer debris plugging event at Limerick Unit 1 in September 1995, led to further evaluation. Eventually, all BWRs implemented programs to reduce potential strainer blockage debris in containment and improve suppression pool cleanliness and installed large capacity passive strainer designs by the mid-1990's to ensure ECCS pump net positive suction head available for emergency core cooling system during a LOCA.

GSI-191 Implications for BWRs

Because of the information the NRC learned during the assessment of BWR suction strainers and oversight of BWR plant-specific evaluations and modifications, the NRC sponsored a new research effort to study the accumulation of debris on PWR containment sump screens. Based on the most recent research study, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," the NRC concluded that its guidance needed revision for PWRs. In November 2003, the NRC issued Revision 3 of Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident." Currently, the NRC is implementing its plan to have all PWR licensees perform a plant-specific evaluation for the potential for excessive head loss across the containment sump screen because of the accumulation of debris on the containment sump screen. The NRC also expects licensees to evaluate the effects of debris that might pass through the sump screens.

Based on the information available to date, continued operation of PWRs is justified until plant-specific evaluations are completed. To provide additional assurance regarding the continued operation of PWRs, the NRC asked the licensees of PWRs to implement compensatory measures. This was done through the issuance of Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," in June 2003. If the results of ongoing NRC inspections and reviews or ongoing and planned studies indicate that unsafe conditions exist at any operating PWR, the NRC will take actions to ensure the continued health and safety of the public. Also, if a licensee discovers that it is not in compliance with the NRC regulations during the implementation of the requested actions in Bulletin 2003-01, it is required to take prompt corrective actions.

In 2007, the NRC did a preliminary area-by-area comparison of regulatory and technical treatment of BWRs vs. PWRs. The NRC's initial conclusion was that there were disparities in treatment, but there was not enough information to validate the issues or their significance. The NRC concluded additional evaluations were needed to determine the safety significance of these issues. The NRC's Office of Nuclear Regulatory Research and the BWROG have initiated additional work on BWR strainer performance.

The NRC and the BWROG have met on a number of occasions to discuss a path forward. The NRC staff has provided perspective to the Owners Group on some of the subject areas related to strainer performance based on lessons learned from evaluations of PWR sump performance. The BWROG continues to apply the lessons learned from GSI-191 and re-evaluate the modifications and analyses for BWRs completed in the 1990s. Final guidance for BWRs is scheduled for May 2016.

Generic Issue-193, "BWR ECCS Suction Concerns"

In May 2002, the staff opened Generic Issue (GI)-193, "BWR ECCS Suction Concerns," which evaluates possible failure of the ECCS pumps (or degraded performance) caused by unanticipated large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I, II, and III containments during LOCA conditions that could cause gas binding, vapor locking, or cavitation. As a result of the initial screening, a task action plan (TAP) for the technical assessment of this issue was approved in May 2004. The staff completed a literature search for information on ECCS pump performance during intake conditions at high voiding in March 2005, and the staff also found experimental evidence that gas may reach the ECCS pumps during a LOCA. Although it appears the pumps can recover given a limited amount of void fraction, the impact of voiding on the operation of the pumps is a concern.

The TAP to resolve this GI involves an evaluation of suppression pool designs, the dynamics of air entrainment in the suppression pool, and the impact on ECCS pump performance. A review of wetwell and suppression pool designs was made to establish bounding parameters. Relevant experiments on pool dynamics were reviewed to identify pre-existing sources of data.

Completed portions of the TAP resulted in a basic understanding of the overall phenomena and a preliminary assessment that continued work on the GI is warranted. The next phase will involve a multi-step estimation of the maximum potential void fraction (MPVF) occurring at different stages of a large and medium LOCA and will attempt to quantify an upper bound for voids present at the ECCS pump suction strainer in the wetwell. The MPVF appears to be influenced by a number of phenomena, many of which overlap in time, such as the gas/liquid jet coming from the downcomer and noncondensable gas injection from the drywell. An estimate of the MPVF (based on a simplified, worst-case scenario for a generic containment) will be made. Ultimately, it is expected that this may provide licensees with insight on how to calculate the MPVF based on their plant-specific geometrical and operational characteristics. Initial emphasis will be placed on the calculations for the Mark I containment.

Based on a staff request, BWROG agreed to provide voluntary input that would provide insights into the characteristics of LOCA phenomena at the earliest stages of the postulated accidents, plus general information about wetwell geometries in relation to ECCS suction strainers. This proprietary input was received on October 29, 2009.

An experimental testing program was proposed in 2009 to help assess the complex phenomenology involved with bubble creation, injection, and transport into the containment wetwell. Modifications to the experimental facility at Purdue University began in fall 2009 in order to simulate the creation and behavior of voids following their injection into a BWR Mark I suppression pool. The testing program, underway during 2010, was completed at Purdue University to promote understanding of complex void-transport phenomena. The final report was received in March 2011. The results of the experimental program have shed light on the behavior of voids in the BWR Mark I wetwell design in regard to the potential transport of bubbles resulting from the LOCA blowdown. This information will be valuable in assessing the capability of bubbles to be transported to the suction strainer of ECCS pumps. The issue remains open.

Table 3. BWR Mark I Containments in the United States

BWR Mark I's Data
Sorted Alphabetically by Plant Name (Column A)

A	B	C	D	E						
Unit	License Renewal		Power Uprate		Power	Power	Containment Free Volume	Containment Free Volume	SP Water Volume	SP Water Volume
	Yes/No	Date of Expiration	Yes/No	%	MWth	MWe	Cubic Feet	Cubic Meters	Cubic Feet	Cubic Meters
Browns Ferry 1	Yes	12/20/2033	Yes	5	3458	1065	278,000	7872	85,000	2,407
Browns Ferry 2	Yes	06/28/2034	Yes	5	3458	1104	278,000	7872	85,000	2,407
Browns Ferry 3	Yes	07/02/2036	Yes	5	3458	1115	278,000	7872	85,000	2,407
Brunswick 1	Yes	09/08/2036	Yes	20	2923	938	288,100	8158	87,600	2,481
Brunswick 2	Yes	12/27/2034	Yes	20	2923	937	288,100	8158	87,600	2,481
Cooper	Yes	01/18/2034	No		2419	830	255,240	7228	87,660	2,482
Dresden 2	Yes	12/22/2029	Yes	17	2957	867	275,481	7801	112,203	3,177
Dresden 3	Yes	01/12/2031	Yes	17	2957	867	275,236	7801	112,203	3,177
Duane Arnold	Yes	02/21/2034	Yes	15.3	1912	640	225,560	6387	61,500	1,741
Fermi 2	No	01/23/2028	Yes	5	3430	1122	291,490	8254	121,080	3,429
Fitzpatrick	Yes	10/17/2034	No		2536	852	264,000	7476	105,000	2,973
Hatch 2	Yes	06/13/2038	Yes	15	2804	883	257,190	7283	87,660	2,482
Hatch 1	Yes	08/06/2034	Yes	15	2804	876	257,190	7283	87,660	2,482
Hope Creek 1	Yes	04/11/2046	Yes	16.6	3840	1061	302,500	8566	122,000	3,455
Monticello	Yes	08/08/2030	Yes	7	1775	572	242,450	6865	77,970	2,208
Nine Mile Point 1	Yes	08/22/2029	No		1850	621	300,000	8495	89,000	2,520
Oyster Creek	Yes	04/09/2029	No		1930	619	308,000	8722	77,970	2,208
Peach Bottom 2	Yes	08/08/2033	Yes	6.62	3514	1112	303,500	8594	123,000	3,483
Peach Bottom 3	Yes	07/02/2034	Yes	6.62	3514	1112	303,500	8594	123,000	3,483
Pilgrim	Yes	06/08/2032	Yes	1.5	2028	685	257,000	7277	84,000	2,379
Quad Cities 1	Yes	12/14/2032	Yes	17	2957	882	275,236	7794	115,600	3,273
Quad Cities 2	Yes	12/14/2032	Yes	17	2957	882	275,236	7794	115,600	3,273
Vermont Yankee	Yes	03/21/2032	Yes	20	1912	620	242,450	6865	77,970	2,208

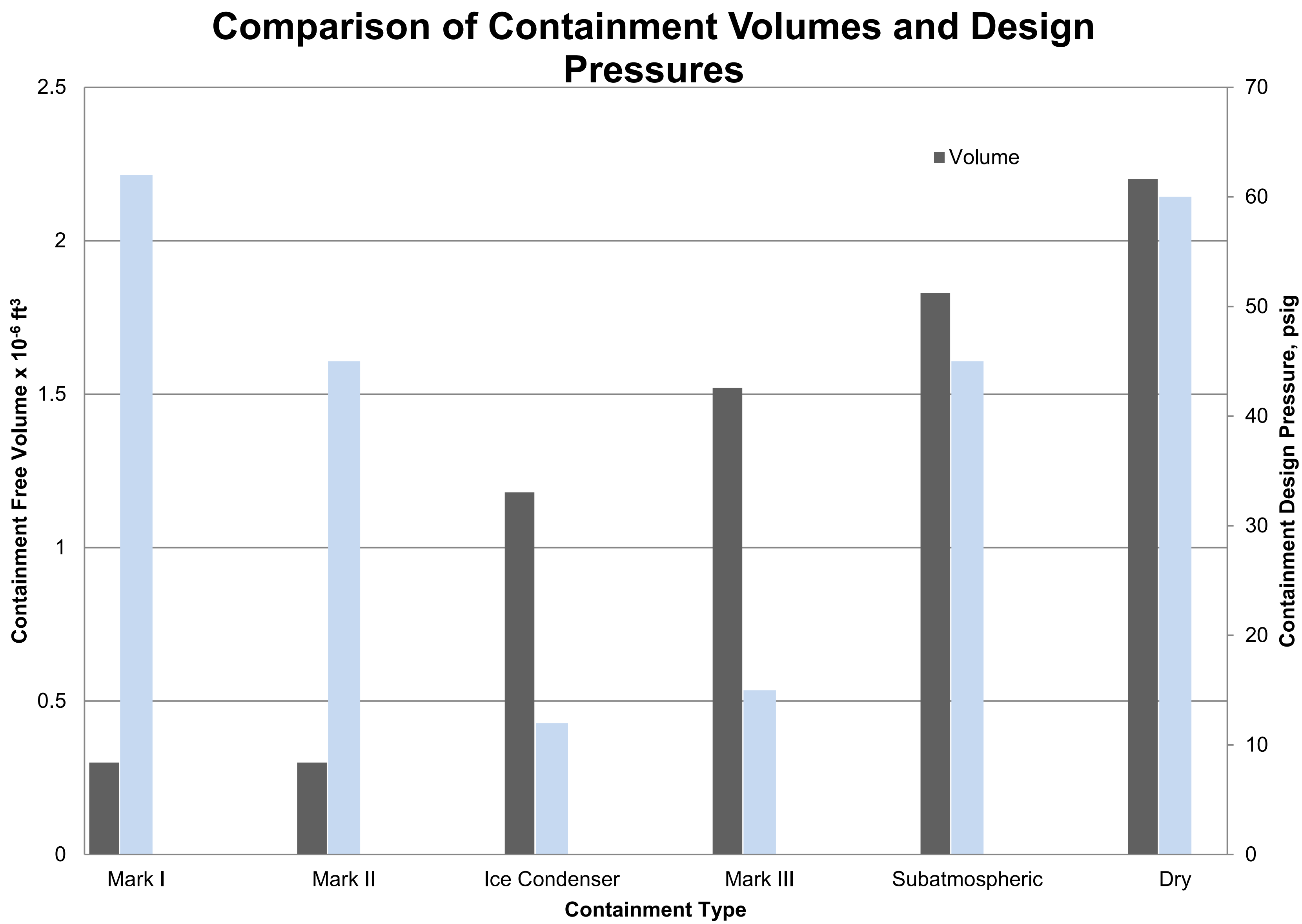


Figure 1. Comparison of containment volumes and design pressures

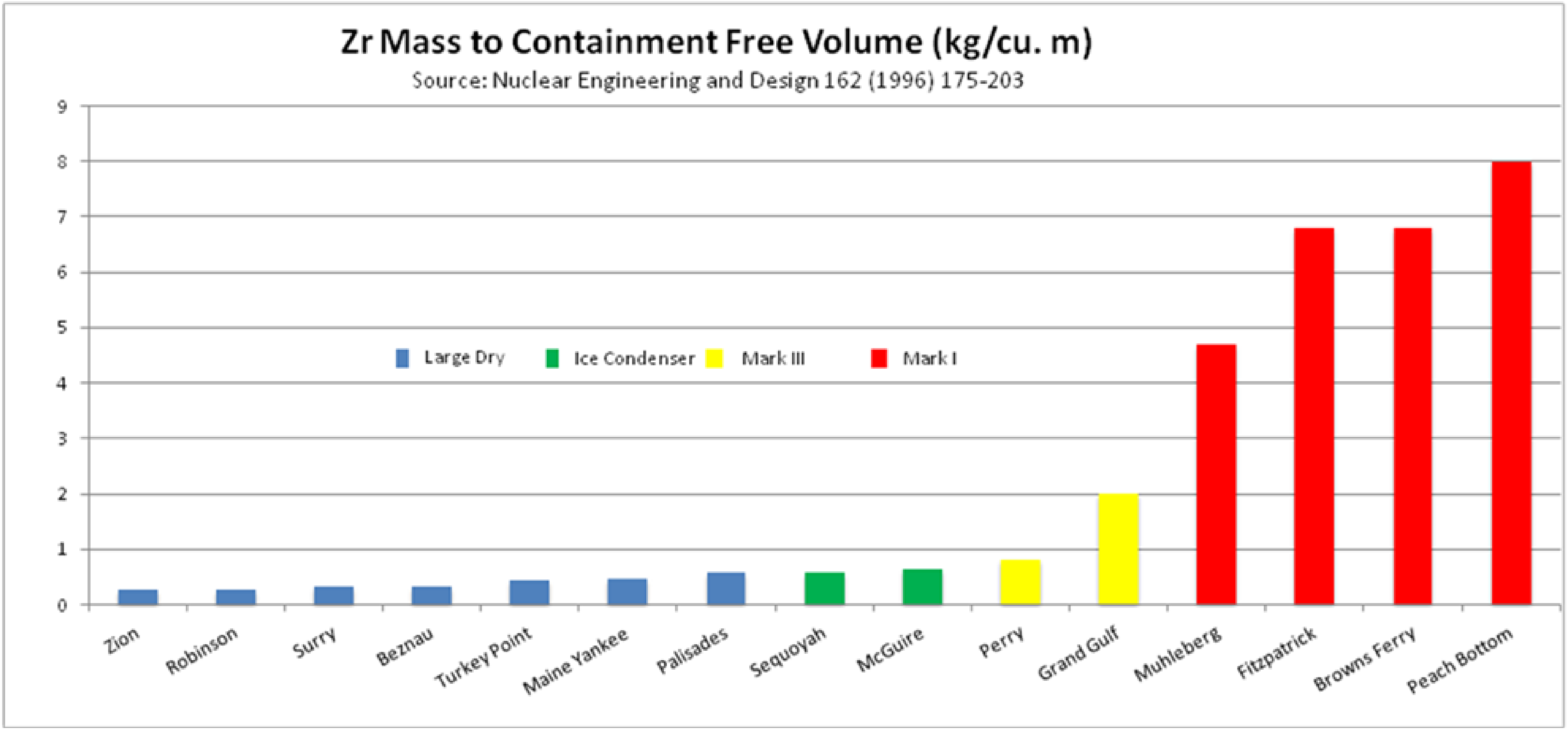


Figure 2. Zirconium mass to containment free volume

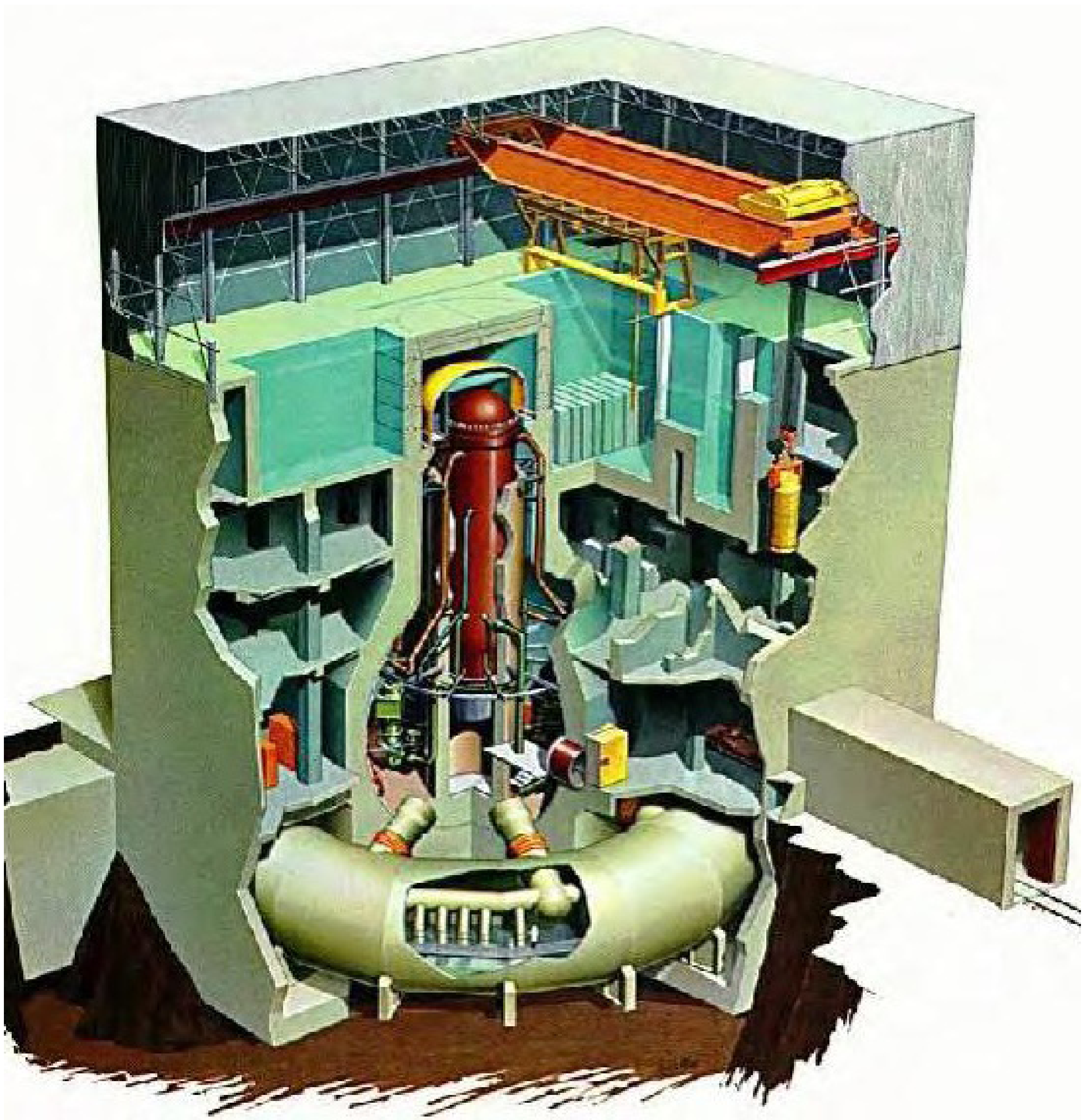


Figure 3. BWR Mark I containment cross section

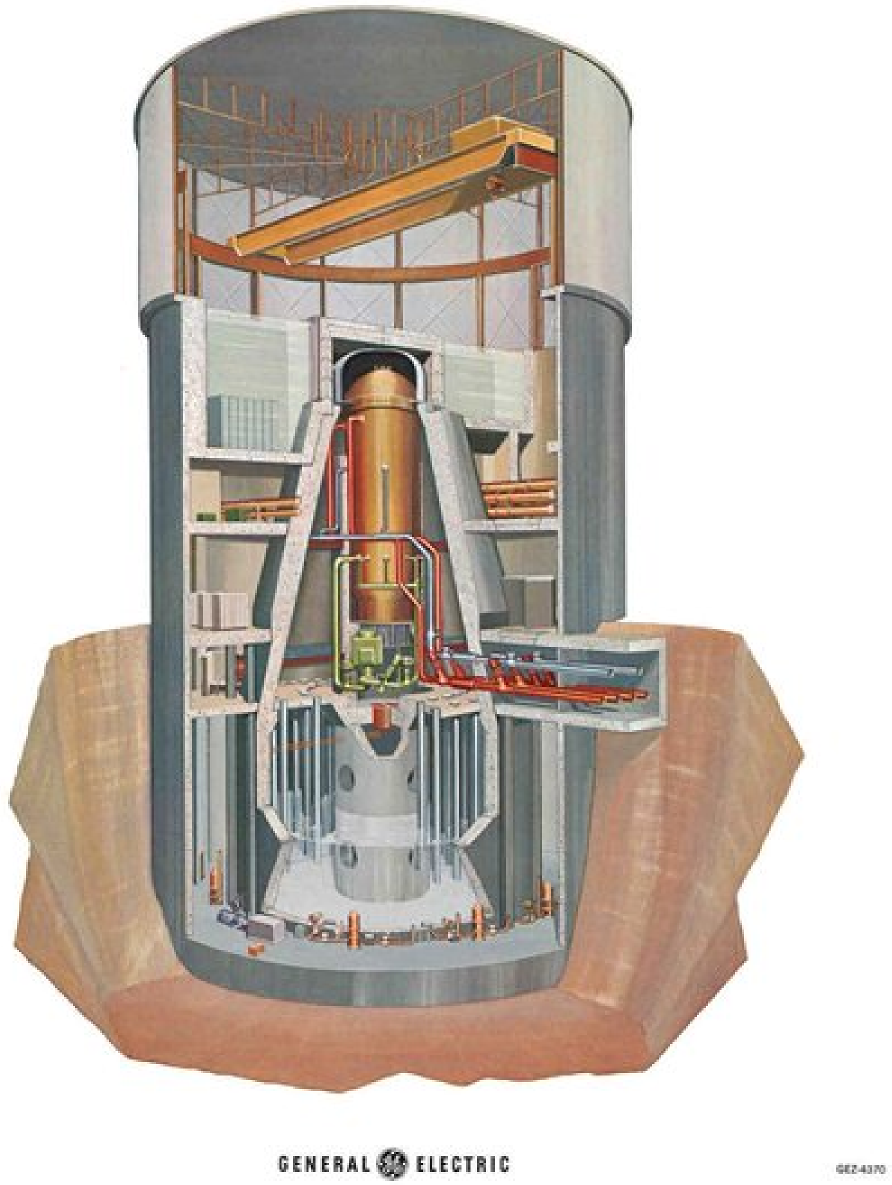


Figure 4. BWR Mark II containment cross section

ENCLOSURE 3

FOREIGN EXPERIENCE

Contents

Foreign Experience

1.0 Introduction

Many nuclear regulatory authorities in Europe and other parts of the world require filtered containment venting systems (FCVS) or are currently considering the safety benefit of installing FCVS. Following the accidents at Three Mile Island (TMI) in 1979, and Chernobyl in 1986, FCVS were installed on a significant number of reactors worldwide. After the Fukushima accidents, decisions were made to install FCVS on many more reactors. In Sweden, Finland, Germany, France and Switzerland, regulators have evaluated the technical issues for achieving severe accident mitigation, and averting potential radiation doses to members of the public and land contamination in the event of a severe accident. The conclusions reached in those countries was that severe core damage operating experience warranted increased defense in depth, especially in the ability of the primary containment to passively retain accident fission product releases and manage the hydrogen produced in a severe accident. FCVS were deemed essential safety enhancements and are required by many regulatory authorities.

The U.S. Nuclear Regulatory Commission (NRC) identified filtered containment venting as a possible safety enhancement following the Three Mile Island accident. In 1979, industrial-size fission product filtering was performed only in conjunction with military defense program production and fabrication facilities, using solid filter media such as sand and gravel. Sand and gravel filters were briefly reviewed, in concept, by the Electric Power Research Institute (EPRI) and the NRC, but not pursued further. Filtered containment venting was discussed in NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident,” (Reference 1) under task items II.B.5, “Research phenomena associated with core degradation and fuel melting” and II.B.8, “Rulemaking proceeding on degraded core accidents.” However, the items were not included in the TMI-related items approved for implementation by the Commission (Reference 2). After Chernobyl, the issue evolved into Task Item CH3.2, “Filtered Venting,” in NUREG-0933, “Resolution of Generic Safety Issues” (Reference 3). The GSI program stated that work related to filter vents is a non-distinguishable part of the development of accident management strategies and containment performance assessments. In pursuing this issue, the staff was expected to increase its knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety; therefore, the issue was considered to be a licensing issue. The program concluded that staff has been doing that by its involvement in various on-going activities related to severe accidents. Because licensing issues are not considered safety issues by the generic safety issues (GSI) program guidance, filtered venting was not pursued any further under NUREG-0933. Filtering was not seriously considered for the Mark I hardened vents requested in Generic Letter (GL) 89-16 “Installation of a Hardened Wetwell Vent” (Reference 4), since venting the primary containment to atmosphere was intended to prevent and not mitigate core damage.

In the 10 years immediately following TMI, Sweden and other countries pursued FCVS technology based on dry filtering, and on wet venturi scrubbing principles first applied in the 1950s to coal-fired power plant emission scrubbing. As a result, the technical and operational expertise and experience for FCVS mostly resides outside of the United States. Early filtering methods and the improved technologies that are currently available in the market are discussed in Enclosure 4 to this Commission Paper. In April and May 2012, the NRC staff took part in bilateral meetings with representatives from regulatory authorities and licensees in Sweden, Switzerland, and Canada to learn more about the experiences of FCVS in those countries. All three countries had required FCVS prior to the accident at Fukushima. The staff exchanged detailed information with plant operators and participated in walkdowns of FCVS installations at the Forsmark (Sweden), Ringhals (Sweden), Leibstadt (Switzerland), Mühleberg (Switzerland), and Point Lepreau (Canada) nuclear power plants (NPPs).

The staff learned that each of these countries share much in common with respect to FCVS: (1) the regulator requires what are considered reasonable-cost safety upgrades consistent with industry progress in safety technology; (2) the regulator and the licensees agree on strengthening containment integrity as a goal, regardless of the calculated value of core damage frequency, for defense in depth recognizing the uncertainties in probabilistic safety assessments (PSAs); (3) a Level 2 PSA is performed to identify the benefit of severe accident management features to strengthen containment, using a large release and/or land contamination criterion; and (4) the regulator and the licensee agree on features that will (a) prevent core-concrete interaction and liner melt through, by flooding containment and covering core debris with water, and (b) remove heat using the FCVS, which also manages overpressure scenarios (e.g., prolonged station blackout (SBO)), including those resulting from arrested core melt outside the reactor vessel.

The information in this enclosure is based on various sources, including the staff's bilateral meetings and discussions with representatives of foreign countries and walkdown of FCVS installations; CSNI Report 148, "OECD Specialist Meeting on Filtered Containment Vent System" conducted in Paris, France on May 17–18, 1988 (Reference 5); responses from staff questions to nuclear authorities in foreign countries; and information from European Stress Test Reports (Reference 6).

The following describes, in part, the rationale for filtered venting in countries that pioneered the early development of FCVS technology. Sweden is representative of countries that decided early-on to install FCVS, and is discussed at length. Consistent with the purpose of this paper, greater emphasis is placed on the boiling water reactor (BWR) Mark I and Mark II type containments.

2.0 Foreign Experience with FCVS

2.1 Sweden

There were 12 light-water reactors in the Swedish nuclear power program at the time the Swedish government ordered a comprehensive review of the lessons learned from the TMI accident. In December 1979, the "Report by the Swedish Government Committee on Nuclear Reactor Safety" recommended that all existing Swedish NPPs be capable of withstanding a core melt accident without any casualties or "ground contamination of importance to the population." Many of the experts who authored the report considered that in cases where the containment was threatened by overpressure, controlled release of limited amounts of activity was preferable to a possible catastrophic failure of the containment barrier with a large and uncontrolled release that would likely result.

The Swedish Parliament later established new general guidelines for the country's reactor safety program as part of its 1980/81 Energy Bill, which was reconfirmed in the 1984/85 Energy Bill. According to the 1981 guidelines, the main priority for Swedish NPPs was for operators to remain focused on the prevention of core damage. However, Sweden's reactor safety program also recognized that despite efforts to prevent core damage, accidents involving severe core damage may still occur, and that no matter how small the probability may be for such accidents, measures should be taken to ensure that releases from severe accidents are kept low.

The 1980/81 Energy Bill also required the owners and operators of the Barsebäck NPP, located approximately 20 km from Copenhagen, Denmark, to expedite the installation of a FCVS at the facility. Barsebäck NPP consists of two BWR units with many similarities to U.S. BWR Mark II pressure suppression containments. Sweden later followed up this action by issuing a "regulatory decree," or order, in October 1981, further requiring that the FCVS for Barsebäck be operational no later than 1985. Completion of the project became a condition of its operating license. The bill gave priority to prevention of ground contamination due to the social consequences that can be anticipated in connection with large-scale evacuation. The bill did not provide any avenues, such as cost-effectiveness arguments, for nonimplementation of the FCVS. In a letter dated October 15, 1981, to the owner of Barsebäck, the Swedish Government provided performance requirements for the new filtering system by requiring that 99.9 percent of radioactive isotopes, excluding noble gases, be retained in either the containment or the new filter, when venting during a severe accident. The decree further stated that filtered venting of reactor containments at Sweden's remaining operating nuclear power plants located at Ringhals, Oskarsham, and Forsmark is a future possibility after taking into consideration the experience from Barsebäck's implementation of the FCVS and the research and technical developments underway within the field of severe accidents. The chosen mitigation concept at Barsebäck was based on the FILTRA research project carried out in 1980–1982. The filtered vent at Barsebäck consists of a 10,000 m^3 gravel bed connected to the containment of each of the two reactor units. Venting is made feasible through a pipe connected to the upper drywell and via a vent pipe and rupture disc connected to the gas volume of the wetwell. Additional discussion of the filtered venting at Barsebäck is not provided herein as the gravel bed filters are not the current state of art. In addition, the two Barsebäck units were permanently shut down, prior to Fukushima events, over concerns of a severe accident impacting the population centers of Copenhagen in Denmark, and Malmo in Sweden.

Upon request from the Swedish Nuclear Power Inspectorate (SKI), the utilities operating the remaining NPPs provided reports applicable to their plants in 1985, on ongoing severe accident development projects. After reviewing the results of the utility studies and the research conducted under the FILTRA and RAMA programs, SKI and the National Institute of Radiation Protection (SSI) submitted their recommendations to the Swedish government, following which the remaining operating nuclear plants located at Forsmark, Ringhals, and Oskarsham, were given the formal requirements in 1986 for new mitigating systems during severe accidents and were asked to install the mitigating features by January 1989. Forsmark and Oskarsham consist of three BWRs at each site for a total six BWRs of ASEA-Atom design with similarities in concept to BWR Mark I and Mark II, and the Ringhals site consists of one BWR of ASEA-Atom design and three pressurized-water reactors (PWRs) of Westinghouse design. Swedish authorities determined that cost-benefit considerations would not be the deciding factor in whether or not to ultimately require a FCVS at Forsmark, Ringhals and Oskarsham. This conclusion was consistent with a 1981 decision by the Swedish government that "such measures should be taken even if they involve a not insignificant cost for the owners, as seen in relation to the reduction of the release risk."

The basic guidelines and criteria for severe accident management and release mitigations measures at the Swedish NPPs are:

- There shall be no early fatalities resulting from radiation injuries.

- Ground contamination that would make it impossible to use large areas for long periods of time shall be prevented.
- Events with extremely low probabilities such as major reactor vessel ruptures need not be considered.
- The same basic requirements with regard to the maximum radioactive release are to apply to all reactors, regardless of location or power output.

Figure 1 – Implementation of severe accident consequence mitigating measures in Sweden

The Swedish BWRs consist of two types: BWR-A, in which the lower section of the containment forms the condensation or suppression pool covering the whole bottom area, somewhat similar to a GE BWR Mark II; and BWR-B in which the condensation pool is annular and the space below the reactor vessel (lower drywell) is dry, somewhat similar to GE BWR Mark I. The analysis of core melt progression and possible radioactive releases were performed through the research project RAMA and utility plant specific studies. The frequency of core melt and probability of containment failure, early or late, were analyzed using a PSA. Two cases were considered, Case 1 is a loss of all core cooling due to loss of all AC power, combined with a loss of steam driven feedwater. Case 2 is a large loss-of-coolant accident (LOCA) with degraded pressure suppression function. Case 1 is the main beyond-design-basis event where the core is damaged and measures to mitigate external release from the containment is required. Case 2 is a design-basis event with respect to early containment overpressurization in a BWR. However, emergency core cooling systems and/or electrical power systems are not affected in this case. A separate unfiltered vent will provide the necessary protection against early rapid overpressurization for Case 2.

Sweden investigated several severe accident mitigating strategies, including alternatives to FCVS. This effort led to the identification of a number of measures that could be taken to protect containment integrity during severe accidents. These measures focused on vulnerabilities uncovered by risk analyses related to containment overpressure failure, containment liner/concrete failure from core debris scenarios, and electrical and mechanical penetration failures in the lower drywell that could occur in Case 1 scenarios in BWR-B containments. These measures included:

- Containment over-pressure suppression
- Lower drywell flooding to protect the basement
- Independent containment spray and water fill systems
- Filtered containment venting systems

The new 1986 guidelines led to extensive safety improvements to Sweden's NPPs, including:

- Filtered containment venting for BWRs and PWRs
- Containment overpressure protection for BWRs, an unfiltered vent to relieve pressure from an early and rapid increase in pressure, as discussed above for Case 2. The unfiltered vent is for a design-basis accident case with loss of pressure suppression function.
- Lower drywell flooding from the wetwell by opening dump valves
- Independent containment spray and containment water filling from an external source
- Containment instrumentation for severe accidents (radioactivity, temperature, pressure, water level, hydrogen concentration)
- Containment penetration shielding in lower drywell for BWRs

To address severe core damage, licensees were also required to prepare plant-specific strategies in order to protect the reactor containment function and to allow the reactor to reach a stable condition where the core is cooled and covered by water. The containment function was required to remain intact during the first 10 to 15 hours after core damage.

The 1986 guidelines also limited radioactive releases to the environment to a maximum 0.1 percent of the reactor core content of Cesium-134 and Cesium-137 in a reactor core of 1,800 MW thermal power, provided that "other nuclides of significance, from the use of land viewpoint, are limited to the same extent as Cesium." This release limit is well below 200 terabecquerel (TBq) and results in a very limited area (considered much less than 50 km^2 off site) for potential first year dose resulting from ground contamination of greater than 50 mSv. Accordingly, filtered containment venting through an inerted multi-venturi scrubber system (MVSS) needed to have a decontamination factor (DF) of at least 100 for BWRs and 500 for PWRs to meet the 1986 guidelines.

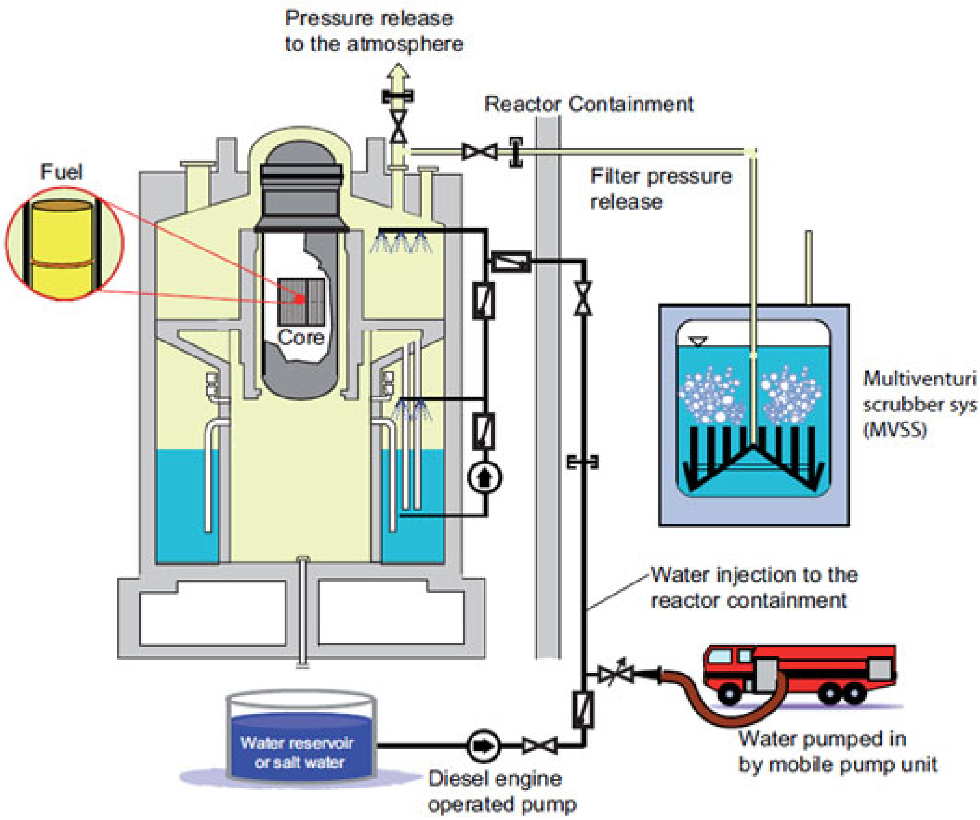


Figure 2 – Severe accident mitigating measures at BWRs in Sweden

According to the Swedish analysis, several potential threats to containment integrity occur during the core melt process. These threats are categorized as: (1) pressure loads due to gas and steam generation, (2) temperature loads due to the high temperature of the molten core, (3) impulse loads due to the interaction between the molten core and water, and (4) concrete ablation due to interaction between the high temperature corium and concrete.

In Sweden, the chosen severe accident design scenario is an SBO, further defined as a loss of all AC, and loss of steam-driven pumps, with no manual actions credited during the first 8 hours. Without manual actions or mitigating systems, this scenario will typically result in serious core degradation within 1 to 2 hours; reactor pressure vessel failure within 2 to 4 hours, followed by containment overpressurization and gross failure of containment with large releases of fission products, unless mitigating measures are taken.

In the design scenario for BWRs, the pressure in the containment will not reach the design pressure within 8 hours and the actuation of an independent containment spray (ICS) at this time will significantly delay the overpressurization of the containment until more than 24 hours. Somewhat later, the FCVS is assumed to be actuated manually, but in case of no manual actions, the FCVS is automatically actuated through the bursting of a rupture disc.

However, if no other means of cooling of the containment becomes available, the ICS injection cannot continue indefinitely because it will fill up the containment and, thus, it is terminated after approximately 30 hours. The pressure will then rise and there will be repeated activations of the FCVS with an energy balance established with “feed-and-boil” with the ICS intermittently injecting water and the FCVS dissipating energy through a filtered release of steam. The ability to passively (no operator decision or action required) prevent a containment overpressurization is an important feature for protecting containment integrity in Sweden (Reference 6a).

Operational Findings from Forsmark and Ringhals

In April 2012, members of the NRC staff participated in bilateral meetings with representatives from the Swedish Radiation Safety Authority (SSM) and Vattenfall AB, the licensee for Forsmark and Ringhals NPPs. During this time, the staff had the opportunity to ask questions, exchange information, and perform walkdowns of the FCVS at Forsmark, Unit 1, and Ringhals, Unit 2. The staff gained considerable knowledge and valuable insights into filtered containment venting at the facilities. These insights included:

- Sweden considered the following alternatives before choosing FCVS
  - Pressure relief without filter
  - Filtered pressure relief using a small sand filter
  - Pressure relief of residual heat removal system to avoid a containment bypass flow path
  - Sprays and containment (under pedestal) flooding systems
  - Other improvements to avoid containment leakage
  - Administrative measures

- High filter efficiencies were required to meet the 1986 guidelines to limit radioactive releases to a maximum 0.1 percent of the reactor core content of Cesium-134 and Cesium-137 in a reactor core of 1,800 MW (thermal power).

- Representatives from SSM and Vattenfall stated that Sweden considered various alternatives to filtered venting, and that they would still choose FCVS due to the uncertainties of these alternatives to provide a reliable means to decontaminate aerosols. They generally believe that FCVS is a cost-effective solution to the prevention of land contamination.

- Installation costs (in 1988 dollars) were estimated be approximately $12.5 million per unit at Forsmark and approximately $9 million per unit at Ringhals NPPs.

- Vattenfall representatives estimated that annual maintenance, testing, and inspection costs for the FCVS is approximately $10,000 to $30,000 per unit.

- FCVS is included in plant technical specifications. Allowed outage times (AOTs) are 30 days.

- Licensee representatives did not consider design obstacles to be significant, and the system was designed to be largely independent of existing plant systems in order to minimize construction costs and any loss of electrical production as a result of possible increases in refueling outage-related delays.

- The FCVS was located immediately adjacent to the reactor building and is physically separated from other plant mitigation equipment. In addition, the filter is constructed of concrete and provides significant radiation shielding to plant workers following an accident. Thus, the licensee and SSM officials did not see any concerns regarding an adverse impact from the FCVS to plant workers due to high radiation doses.

- The investigations of severe accidents at the Swedish plants were summarized in a report called "MITRA." MITRA summarizes PSA studies of potential improvements for a number of initiating events. The initiating events studied included transients, LOCAs, external events, and internal events.

- PSA-studies also included consideration of
  - Filtered containment vents similar to Barsebäck with a 99.9 percent removal efficiency (1,000 decontamination factor)
  - Diversified containment spray system

Current Regulatory Requirements

The regulatory approach adopted in Sweden by SSM has been described as being more process-oriented and is less prescriptive. Regulations provide general requirements that focus on the required licensee processes and the outcome of these processes. Regulatory guidance documents provide only limited details on how licensees are to perform the various process requirements.

The most current regulations regarding nuclear power plants, SSMFS 2008:17, "The Swedish Radiation Safety Authority's Regulations Concerning the Design and Construction of Nuclear Power Reactors," are intended to provide more explicit requirements instead of more generalized guidelines for the industry. The new regulations pertaining to severe accident mitigation and FCVS now include:

- Additional requirements needed for modernization of nuclear facilities.
- A definition of "highly improbable events." These are events which are not expected to occur. However, if the event should nevertheless occur, it can result in major core damage. These events serve as the basis of the nuclear power reactor's mitigating systems for severe accidents.
- Chapter 5 of SSMFS 2008:17 states that the "reactor containment shall be designed taking into account phenomena and loads that can occur in connection with events in the event class highly improbable events, to the extent needed in order to limit the release of radioactive substances to the environment." To meet the requirement in Chapter 5, a safety evaluation is to be performed for events and phenomena that may be of importance for containment integrity following "highly improbable events."

In other words, current regulations require containments to be able to perform their intended safety function following a severe accident, and licensees are required to have mitigating systems to limit radioactive releases to the environment in the event of core damage. The present regulatory framework for FCVS was guided by a study regarding the impacts of cleanup efforts following the Chernobyl accident.

2.2 Finland

Finland operates four NPPs, two PWRs at Loviise site of VVER-40 design with ice-condenser containments, and two BWRs of AB ASEA-Atom design at Olkiluoto site with containments somewhat similar to a GE BWR Mark II. A new AREVA PWR unit is currently under construction at Olkiluoto. The following information is based on the staff's understanding of

Finland’s filtered venting requirements and its National Report on European Stress Tests for Nuclear Power Plants (Reference 6b).

Filtered containment venting systems are not installed on the Loviise units due to concerns that venting could lead to sub-atmospheric pressures and possible collapse of the containment. Filtered containment venting systems were installed at the two operating BWRs, Olkiluoto Unit 1 and Unit 2. The plant was modified due to STUK requirement in 1986 (after the Chernobyl accident) that the containment of operating Finnish plants must be equipped with systems ensuring containment integrity in severe accidents. Severe accidents were not part of the plant’s original design basis. FCVS were installed in 1990 at both units as a plant modification.

- The design purpose of the filtered vent is to decrease the containment pressure in severe accident sequences when energy and fission products are released into the containment, if the pressure exceeds a specified limit.

- System components include:
  - A vent line from the wetwell equipped with two manually operated valves in series and connected to the filter unit, with valves normally closed.
  - A vent line from the drywell, which is divided into two parallel lines, one equipped with two manually operated valves in series and connected to the filter unit, with valves normally closed, and the other equipped with a break disc and two manually operated valves in series and connected to the filter unit, with valves normally open.

- The reasoning behind the manual valves is that requiring electric power to the accident management system is inconsistent with the postulation that a total loss of power is the most probable cause of a severe accident, and more so because the filters do not depend on power.

- The valves are not optimally located because of reliance on existing valves. The locations require workers to climb several stairs to access the valves. Mechanical remote operation handles and protection to operators from doses were considered in the design.

- Venting is only required if the power outage lasts for about seven hours or more after the core has melted, and decay heat removal from the wetwell cannot be started.

- Lower drywell can be flooded by gravity feed from the wetwell.

- The exhaust gas line from the filter unit is filled with nitrogen.

- The filter unit is a wet scrubber system with chemical control and fine steel fibers to prevent the water droplets from escaping the filter tank.

- Containment filtered venting in severe accidents is actuated by opening of the break disc at 0.55 MPa containment pressure. The system can also be actuated by opening the manual valves from wetwell or drywell. Use of the system is included in the severe accident procedures.

The capacity (thermal hydraulic design) of the filtered venting system is 12 kg/s of saturated steam at a containment pressure of 0.6 MPa and 6 kg/s at 0.3 MPa. The corresponding containment heat removal rates are 25 MW and 12.5 MW. These decay heat levels will be reached in approximately 5 and 60 hours, respectively, after reactor shutdown. The design criteria of the filter were that the release would start at 8 hours, continue for 16 hours, and terminate at 24 hours. It is assumed in the Olkiluoto 1 and 2 severe accident management that AC power is restored at 24 hours and the decay heat would be removed by the containment cooling system after that. The filter unit and the piping up to the filter is designed and insulated so that makeup of scrubber water is not required for a period of 24 hours.

The filter unit must be capable of arresting at least 99.9 percent of the particle activity of the gas and 99 percent of the gaseous iodine. Capability of the unit to reach these values has been verified by testing at the manufacturer (KWU Siemens). The system is not as effective for gaseous organic iodine compounds, with only 60–80 percent of these arrested.

Installation and maintenance costs were not available to the NRC staff. Since the system is passive, STUK believes the maintenance costs are limited.

Olkiluoto 3 under construction will have a similar system. The requirements of the system have changed since 1990. According to present Finnish regulation, it is not allowed to design the filtered venting system as the principal means for decay heat removal from the containment. The system should be used in a very late stage of the severe accidents (after about 1 week) to remove noncondensable gases from the containment and by this means to decrease the containment pressure and subsequently fission product leakage. A separate system for containment heat removal in severe accidents must be provided

2.3 Germany

After the Fukushima events, Germany terminated the licenses for power operation of seven older plants (BWRs and PWRs) commissioned before 1980, by the amended Atomic Energy Act on August 6, 2011. Currently, there are eight PWRs and two BWRs operational in Germany. There are no regulatory standards that require FCVS at German NPPs. However, in the aftermath of the Chernobyl disaster, German utilities decided in December 1986 to voluntarily install FCVS in all PWRs and BWRs. Germany shares a similar philosophy with Sweden in that, while the prevention of core damage is the priority, the mitigation of severe core damage must be considered by licensees.

The decision for filtered venting systems was based, in part, on plant-specific accident analyses that considered containment venting systems to be relatively important accident management systems. The analyses showed that:

- The most frequent severe accidents are likely to lead to a medium or long-term containment failure.
- If severe core damage can be stopped or if an early containment failure can successfully be prevented, there still remains the potential for late overpressure failure.

Other advantages of FCVS at German NPPs cited by Germany's participants at the Organization for Economic Co-operation and Development (OECD) Specialist Meeting on Filtered Containment Venting Systems (Reference 5), included:

- Beyond-design-basis plant conditions are difficult to predict. With increasing plant degradation during a severe accident, the uncertainties regarding relevant phenomena, further development of the accident, and possible containment failure modes increase considerably.
- Risk assessments predict lower releases of fission products to the environment, because the release path is through filters, and the containment release pathway can be closed.
- Flexibility for plant personnel is increased substantially. Although venting may be designed to cope only with one specific goal (e.g., the avoidance of an overpressure failure) FCVSs can be used for several purposes, (e.g., decay heat can be removed before a basemat melt-through, containment pressure can be reduced to minimize the flow into the ground, and containment atmosphere can be purged, if desirable.)

The German Reactor Safety Commission (RSK) specified the requirements for filtered containment venting in December 1986 for PWRs, and June 1987 for BWRs. Venting flow (gas and steam) at saturated steam conditions shall correspond to 1 percent of the thermal reactor power. The basis for the 1 percent heat removal is that it corresponds to the decay heat rate after the entire heat capacity of the pressure suppression pool is utilized. The RSK specified that FCVSs must be able to remove 99.9 percent of all aerosols and 90 percent of all elemental Iodine. To meet these requirements, many German PWRs were equipped with dry filter systems. The dry filter systems comprised primarily of a series or packs of metal fiber fleeces with decreasing fiber diameters that were installed upstream of conventional HEPA filters. In addition, Germany developed its own “wet filter method” to clean fission products resulting from severe core damage. The German wet filters include a venturi scrubber system with a metal fiber droplet/mist/particle filter unit similar in design to the FILTRA/MVSS systems developed in Sweden (Reference 7).

The vent line in BWRs is connected directly via a line coming from the suppression pool. The procedure for filtered containment venting is described in the emergency procedures manual. The aim of containment venting is to reduce pressure in the containment from approximately the design pressure to half that value. It is estimated that this process in a BWR takes approximately 10 to 20 minutes.

The critical containment pressure for venting is 1 to 1.2 times containment design pressure. The FCVS is to be used during beyond-design-basis events with loss of containment cooling. The venting procedure consists of several steps, including consultation with regulatory and civilian authorities, confirmation that containment pressure is ascending, and emission monitoring instrumentation is functional.

Installation and maintenance costs were not available to the NRC staff.

2.4 France

France has a significant presence in nuclear power with 58 pressurized water reactors. Following the TMI accident, France took a similar approach as other West European countries. France’s Institute of Radiological Protection and Nuclear Safety (IRSN) established additional procedures to manage accidental situations leading to the loss of redundant safety systems and to limit the accident consequences whatever its cause. These procedures included the management of containment leakage, the suppression of leakage paths in the basemat, and the management of slow containment overpressurization by a filtered containment venting. These procedures led to the installation of sand filter systems in France's PWRs by the early 1990s (Reference 8).

French NPPs are PWRs with large dry containments. They are all equipped with filtered containment venting systems. The purpose of the venting system is to avoid any containment failure in the long-term phase of a severe accident that could for instance be due to overpressure resulting from gases from molten core concrete interactions. The opening of the vent system, which is an ultimate reactor containment protection measure, would not take place until after 24 hours. To prevent or mitigate the risks from short-term containment failure due to dynamic events, other prevention systems are used, such as pressurizer safety valves to limit the reactor coolant system pressure, direct containment heating, and induced steam generator tube rupture, and passive autocatalytic recombiners to limit the loads due to hydrogen combustion.

The FCVS includes:

- A metallic filter inside the containment that can retain a large fraction of the aerosols
- A sand-bed filter outside the containment that retains most of the remaining aerosols

The venting line is heated to avoid steam condensation and limit the risk of hydrogen combustion within the venting line.

The decontamination factors that are credited in safety analyses are derived from small-scale and full-scale (FUCHIA program) experiments. The decontamination factors include 1,000 for aerosol particles, 10 for inorganic gaseous iodine, and 1 for organic gaseous iodine.

IRSN recognizes that significant uncertainties still remain in the evaluation of severe accident consequences and launched, among other actions, the OECD/STEM Project (and its foreseen considered follow-up STEM2) to address the following issues:

- The in-containment source term for gaseous iodine in the mid and long term (STEM Phase 1)
- The stability under radiation of iodine-bearing aerosol particles deposited in containment (STEM Phase 1)
- The transport of ruthenium in case of an air ingress accident with special emphasis on the gaseous ruthenium tetroxide issue (STEM Phase 1)
- The efficiency of presently used filtering media and possible new ones for retention of gaseous species, especially iodine and ruthenium gaseous compounds (STEM Phase 2)

The issue of land contamination and evacuation is addressed in PSA Level-2 studies conducted by IRSN. As for long-term effects such as land contamination, reference is made to the Chernobyl accident. Three thresholds for land contamination based on surface activity of Cesium-137 are used in the analyses, with the highest thresholds (15 and 40 Ci/km^2) requiring mandatory permanent relocation. The threshold values are not a regulation but only used for an analysis of consequences.

For 900 MWe French NPPs, the latest version of PSA level 2 studies indicates that no permanent relocation is needed and that the threshold for relocation on a voluntary basis is reached at 4 km from the power plant.

Cost information was not available to the NRC staff. IRSN is performing analyses related to the costs induced by a NPP severe accident for several degrees of severity that include the consequences of the so-called S3 source term which corresponds to what would happen in case of filtered containment venting for a core meltdown accident. Direct on-site (decontamination and material replacement) and off-site (prohibition of foods, long-term sanitary effects) as well as indirect costs will be included in the analyses.

The French Report on European Stress Tests conducted after Fukushima (Reference 6c) indicated potential deficiencies of the filter system, such as the seismic capability (it is not seismically designed after the outboard containment isolation valve) and the adequacy of the system for venting at multi-unit sites. In November 2011, IRSN announced a new approach to ensuring the safety of nuclear installations and a number of measures designed to prevent severe accidents from becoming catastrophic ones as a result of the Fukushima accident. The new measures that were outlined may result in the replacement of some accident management systems, such as the filtered containment vents (sand filters) at French reactors. In announcing the new approach, IRSN stated that the new measures are an implicit recognition that a severe accident can happen, even at plants that have implemented post-TMI and post-Chernobyl accident management measures. The Nuclear Safety Authority (ASN) requested licensees to submit a detailed study of possible improvements to the U5 venting-filtration systems, with respect to their resistance to hazards (e.g., seismic), improved filtration of fission products, consequences of opening the vent on accessibility to the site and on the control room and emergency premises, and risks of hydrogen combustion.

2.5 Switzerland

Switzerland has five operating nuclear power plant units, with three PWRs and two BWRs of GE design (a Mark III and a Mark I containment). In April 2012, members of the NRC staff participated in bilateral meetings with representatives from the Swiss Federal Nuclear Safety Inspectorate (ENSI), Kernkraftwerk Leibstadt (KKL) the licensee for NPP Leibstadt (GE Mark III), and Kernkraftwerk Mühleberg (KKM/BKW) the licensee for NPP Mühleberg (GE Mark I). As in Sweden, the staff had the opportunity to ask questions, exchange information and perform walkdowns of the FCVS at Leibstadt and Mühleberg. The following information is based on the insights gained in these meetings, and the Swiss National Report in response to European Stress Tests (Reference 6d).

Although many similarities to the actions taken by Sweden exist, the Swiss took a slightly different approach following the accident at TMI. The Swiss Nuclear Energy Act states that licensees shall backfit to the extent necessary, in keeping with operational experience and currently available technology, to further reduce risk to people and the environment. Following the TMI accident, the Swiss Safety Authority (HSK) did not require FCVS, but it required all nuclear power plants in Switzerland to install other severe accident mitigation systems. For example, at the Mühleberg NPP, the licensee (KKM) installed a bunkered emergency building housing an additional independent emergency core cooling systems. The facility, known as "SUSAN," provided fully redundant and diverse sources of cooling water at Mühleberg beyond what was originally designed for the plant.

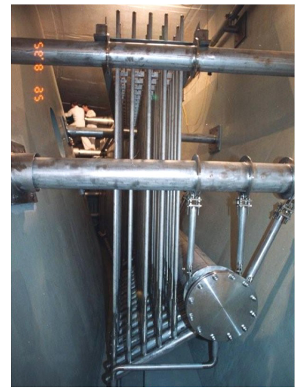
In response to Chernobyl (1986), HSK directed licensees to evaluate the FCVS. By that time, Sweden had made considerable progress on the FILTRA/MVSS effort, and the "wet" containment filters using venturi-type scrubbers were considered the best available technology. HSK required the backfit based on a defense-in-depth argument with the goal of preventing an uncontrolled radioactive release due to loss of integrity of the containment during a severe accident. The FCVS increases the fourth safety layer of defense in depth (onsite accident management). The Swiss Nuclear Energy Act (Article 22, paragraph 2, item g) states that "the licensee holder shall backfit the installation to the necessary extent that it is in keeping with operational experience and the current state of the backfitting technology, and beyond insofar as further upgrading is appropriate and results in further reduction of risk to humans and the environment." FCVS was considered in the Level 2 PSA, but the decision to install FCVS was not based on probabilistic arguments. Uncertainties (e.g., initiating event, human error, severe accident phenomena, system success criteria, emergency response) also did not specifically play any role in the decision to require FCVS. The decision was based on defense in depth and deterministic considerations.

Once the FCVS became a requirement, HSK regulatory guidance was drafted in 1988, with final guidance in 1993 contained in HSK R-40. Plant installations were completed during the 1989–1993 timeframe. The final guidance directed that the designs possess:

- Heat removal capacity of 1 percent thermal power. For PWRs, 0.5 percent thermal power was acceptable.
- Passively actuate via rupture disc so as not to require intervention for 24 hours.
- Allow operation from the control room and separate remote panel. Flow rate though the venting device shall be adjustable.
- Contain their own dedicated power for instrumentation and valve operation.
- Earthquake resistance based on Seismic Class 1 requirements.
- High decontamination factors, 1,000 for aerosols and 100 for elementary iodine (based on available technology)
- Decontamination factors shall be demonstrated on the basis of experiments within a flow rate of 10 percent to 100 percent of the nominal flows. With respect to the expected maximum load on the filters consisting of radioactive and non-radioactive materials, an amount of 150 kg of aerosols is postulated to go into the vent system. A large part of this aerosol mass consists of inactive material.
- Sections of the venting system beyond the second containment isolation valve and ahead of an eventual throttling device shall be deigned to Safety Class 4 (Swiss design rule R-06) and safe-shutdown earthquake. Design pressure shall be a factor of 1.5 times the nominal relief pressure specified above. Consideration of dynamic loads (e.g., condensation shocks) shall be included.
- Conservative consideration of the temperatures to be expected during operation including the possibility of local accumulation of hydrogen gas.
- For BWR Mark I containments, the system should still work after filling the wetwell and flooding the drywell.
- Instrumentation for the venting system shall work in stand-alone mode for 100 hours.

Operational Findings in Switzerland

- Installation costs were estimated to be approximately $11 million (1993 dollars) at Leibstadt, and $6 million (1990 dollars) at Mühleberg. The primary reason for the significantly lower cost for the FCVS at Mühleberg was due to the plant's unique design. Mühleberg has a second "outer torus" that provides pressure suppression capability for the secondary containment similar to that provided by the suppression pool in BWR primary containments. As a result, KKM/BKW was able to take advantage of the outer torus to install venturi-type scrubbers, as shown in the picture to the right.



- Estimated cost for maintenance, testing, and inspection costs for the FCVS is approximately $50,000 to $100,000.
- FCVS is included in plant technical specifications. Allowed out-of-service time is 10 days.
- Enhanced water chemistry is planned to improve iodine retention.
- As in Sweden, licensee representatives did not consider design obstacles to be significant, and the system was also designed to be largely independent of existing plant systems in order to minimize construction costs and any loss of electrical production as a result of possible increases in refueling outage-related delays.

2.6 Canada

In May 2012, members of the NRC staff participated in meetings with representatives of the Canada Nuclear Safety Commission (CNSC), NB Power (Point Lepreau Owner/Operator), and Ontario Power Generation. During this time, the staff had the opportunity to ask questions, exchange information and perform a walkdown of the Point Lepreau FCVS. As in Sweden and Switzerland, the staff also presented questions to Canadian authorities and licensee representatives.

Point Lepreau is a CANDU 6 that has undergone substantial refurbishment beginning in 2000. In 2007, the regulator and the utility discussed the installation of a FCVS similar to those on Swiss plants for severe accident management. The value of a FCVS was assessed by the licensee in a complete Level 2 PSA, including external events, in accordance with CSNC Regulatory document S-294. The analysis uses severe core damage frequency (SCDF), and large release frequency (>1 percent Cs-137 inventory) as decision metrics that align with IAEA SSG-3 and SSG-4. The FCVS, costing approximately $14 million Canadian, was found to be cost-beneficial when using the large release frequency metric. The stated purpose of the FCVS is "to prevent failure of containment integrity due to the increase of containment pressure beyond the failure pressure" of approximately 220-230 kPa(g) or 31.9 to 33.4 psig.

In Canada, an Integrated Safety Review (ISR) is conducted as part of the regulatory requirements for life extension/refurbishment. The ISR requires that licensees address modern codes and standards and state-of-the art knowledge to enhance safety to a level approaching that of a modern plant. The FCVS at Pt. Lepreau was the option that NB power implemented to allow the SCDF/LRF to be met. NB power considered other options and determined that a FCVS was the optimal solution to reduce the large release frequency below the limit.

The implementation rationale for the FCVS on the CANDU 6 parallels the basis found at other foreign sites visited in Sweden and Switzerland. That is: (1) the regulator requires reasonable cost safety upgrades consistent with industry progress in safety technology, (2) the regulator and the licensees agree on strengthening containment integrity as the goal, regardless of the calculated value of core damage frequency, for defense in depth recognizing the uncertainties in PSA, (3) a Level 2 PSA is performed to identify the benefit of severe accident management features to strengthen containment, using a large release and/or land contamination criterion, (4) the regulator and the licensee agree on features that will (a) flood the containment to prevent core-concrete interaction, liner melt through, cover core debris, and (b) remove heat using the FCVS, which also manages overpressure scenarios (e.g., prolonged SBO), including those resulting from arrested core melt outside the reactor vessel.

Point Lepreau installed an AREVA-designed passive filter next to the containment. This design is significantly smaller than the FILTRA design, with the scrubber tank measuring 6.5 meters high and 4 meters in diameter. The scrubber tank contains venturi nozzles in a sparger array, and a metal fiber filter for micro-aerosols. It is operated by hard-linkage isolation valves from a shielded location. It does have a rupture disk, but an isolation valve in the same path is normally closed. It is designed to retain greater than 99.9 percent of aerosols, greater than 99.5 percent elemental iodine, and greater than 99 percent organic iodine. The filter house is seismically qualified.

The CNSC Fukushima Task Force recommended that similar venting provisions as implemented at Pt. Lepreau be considered for all other Canadian NPPs (Reference 9). During the meetings, the staff learned that Canada is also planning to install a shared filter at the four unit Darlington site. The filter will be designed to handle simultaneous accidents at all four units. Filters under consideration are the AREVA design and Westinghouse dry filter design using metal fiber filter and a zeolite “molecular sieve.”

2.7 Japan

There were a total of 54 reactors licensed to operate at the time of the Fukushima accidents. The Fukushima accident caused core meltdowns at three reactors with BWR Mark I containments at the Fukushima Dai-ichi site. Nuclear power in Japan was shut down after the Fukushima accident. Japan’s Nuclear Regulatory Authority (NRA), comprised of five commissioners, unanimously approved a directive on October 10, 2012, to Tokyo Electric Power Company (TEPCO), the owner and operator of the NPPs at the Fukushima Dai-ichi site, to decommission Units 1 to 4, and maintain cold shutdown of Units 5 and 6. As of September 2012, there are only two reactors operating in Japan.

In an announcement on February 7, 2012, the chairman of a committee on nuclear development measures under Japan’s Federation of Electrical Power Companies (FEPC) announced that new vent facilities containing filters would be installed at all nuclear reactors in Japan. The NRA also unanimously approved on October 10, 2012, to formulate regulations to implement the law passed by the Japanese Parliament stipulating that NPP operators must prevent the release of radioactive materials at abnormal levels following severe accidents. The NRA intends to formulate two sets of regulations with regard to detailed design of plant systems and severe accident management procedures aimed at preventing or mitigating those severe accidents.

2.8 Taiwan

Taiwan has four BWRs (two GE Mark I and two Mark III) and two PWRs. In an information exchange between the Atomic Energy Council (AEC) staff and the NRC staff at the NRC Headquarters on October 10, 2012, the AEC stated that filtered vents were ordered for their plants in August 2012.

2.9 Spain

There are two BWRs (one GE Mark I and one GE Mark III) and six PWRs in operation. The installation of FCVS is currently under consideration in Spain.

2.10 Mexico

There are two BWR Mark II units at the coastal site of Laguna Verde. During a visit by the NRC staff members to Mexico in August 2012, the staff was informed that CNSNS is considering what actions they should require.

2.11 Belgium

There are seven PWRs in operation. Belgium has included FCVS in the long-term operation project for the older plants (Doel 1 and 2, and Tihange 1), and is studying the requirement for FCVS in the newer plants.

2.12 China

There are 14 operating reactors in China, all PWRs. China has been pursuing the construction of 26 new generation nuclear power plants, all PWRs, in different stages of construction. The staff understands that China may be planning to install FCVS on two operating reactors and possibly other new reactors. However, recent press reports indicate that as a result of Fukushima, China may be lowering its target by not building new plants in inland locations, and to only build them in coastal areas.

2.13 Netherlands

There is one PWR of Kraftwork Union (KWU) in operation. It is equipped with a wet scrubbing FCVS (Reference 6f).

2.14 Romania

The two CANDU 6 units at Cernavoda are operated by Romanian state nuclear power corporation Societatea Nationala Nuclearelectrica (SNN). They began operation in 1996 and 2007. It was announced in January 2012 that AREVA had been awarded a contract to provide its filtered containment venting systems for the two Romanian plants with a completion scheduled for 2013. The AREVA system uses wet scrubbing technology followed by dry metal fiber filters.

2.15 South Korea

There are a significant number of operating reactors (various PWR types) and new constructions in South Korea. A South Korean representative attending the International

Society of Nuclear Air Treatment Technologies (ISNATT) meeting in the U.S. in 2012, stated that South Korea has decided to install FCVS on all its containment types.

2.16 Other Countries

Hungary, Slovenia, and Slovakia have reactors of western PWR or VVER-440/213 (PWR) design. Hungary stated that FCVS will be one of the many concepts that will be considered for containment overpressure protection (Reference 6g). Slovakia did not mention FCVS as being under consideration (Reference 6i). Slovenia stated that it was considering alternatives to FCVS in its response to the European Stress Tests (Reference 6h); however, Slovenia recently committed to installing a dry filter method FCVS at its only reactor (Krsko), a PWR.

Ukraine has 15 reactors of VVERs type (PWRs) in operation. None of them have FCVS. The National Report of Ukraine on the European Stress Tests (Reference 6k) states that FCVS is one of the measures under consideration for containment overpressure protection.

United Kingdom has many reactors in operation, with only one PWR (Sizewell B) and FCVS for the PWR is under consideration (Reference 6j). The remaining reactors are gas cooled.

Table 1 summarizes the implementation of FCVS in many nations, both BWRs and PWRs, including decisions taken post-Fukushima for future implementation. The table is generally limited to reactors larger than 400 MWe and does not include all countries (e.g., Argentina and Brazil). However, it includes all reactors with BWR Mark I and Mark II type containments. In assessing the status of worldwide implementation of FCVS, the staff relied on news releases from both U.S. and foreign organizations and the European Stress Test Reports.

Country	Boiling Water Reactors (BWR) by Containment Types																													
	GE Mark I					GE Mark II					ABB Mark II					GE Mark III					Other					ABWR				
	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS
Belgium																														
Bulgaria																														
Canada																														
China																														
Czech Republic																														
Finland											2	X																		
France																														
Germany																					2	X								
Hungary																														
India	2				X																									
Japan	4*		X			7		X								3		X			4		X			3		X		
South Korea (ROK)																														
Mexico						2				X																				
Netherlands																														
Romania																														
Russia																														
Slovakia																														
Slovenia																														
South Africa																														
Spain	1			X												1			X											
Sweden											4	X									3	X								
Switzerland	1	X														1	X													
Taiwan	2		X													2		X												
Ukraine																														
United Kingdom																														

* Does not include the 4 reactors damaged by the earthquake and tsunami at Fukushima Dai-ichi.

Country	Other Reactor Designs (non-BWR)																				Notes
	PWR					PHWR/ Candu					VVER					Other					
	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	No. of Rx	FCVS Operational	Committed	Considering	No FCVS	
Belgium	7			X																	
Bulgaria											2	X									
Canada						18	X	X													
China	10					2					2										Information is unavailable.
Czech Republic	6			X																	
Finland											2	X									
France	58	X																			
Germany	11	X																			
Hungary	4																				Information is unavailable.
India																					
Japan	24		X																		
South Korea (ROK)	17			X		4			X												
Mexico																					
Netherlands	1	X																			
Romania						1		X													
Russia											17										Information is unavailable.
Slovakia											4				X						
Slovenia	1		X																		
South Africa	2																				Information is unavailable.
Spain	6			X																	
Sweden	3	X																			
Switzerland	3	X																			
Taiwan	2		X																		
Ukraine											15			X							
United Kingdom	1		X													18					Other reactor types are gas-cooled.

3.0 Unintended Consequences

In the CSNI Report (Reference 5), Sweden included a section in its presentation titled, "Independence of the filter venting system, eventual implications for the existing plant." Several design considerations for the FCVS were included in this section, including:

- FCVS shall have no detrimental influence on the normal operation
- FCVS shall have no detrimental influence on the other safety functions and systems, especially the isolation of the containment should not be impaired (in particular with respect to design-basis accidents)
- The instrumentation of the venting system has to work in stand-alone mode for 100 hours
- The availability of the system must be assured even in case of failure of electrical equipment not pertaining to the vent system

In addition, Sweden also stated in its presentation:

> The filtered vent is essentially a safety valve function which would come into operation only if the containment pressure significantly exceeds the design pressure. However, the installation of such systems requires attention to its possible negative impact on safety, during accidents within traditional design basis as well as during severe accidents. Situations such as release through inadvertent opening of the vent system and containment sub-pressure as a result of venting non-condensable gases and use of containment spray must be analyzed. By appropriate system design and also operator procedures and training, possible negative impact can be kept a minimum and be clearly outbalanced by the overall benefits.

In the bilateral meeting with the staff, representatives from SSM and Vattenfall did not identify any drawbacks or unintended consequences to FCVS. The system was designed to be independent of other plant systems, including systems installed to mitigate the consequences of severe accidents. As a result, there was no interference with other safety systems. Also, the Vattenfall engineering staff did not discover any concerns regarding on-site radiation release, seismic hazards, or maintenance problems (e.g., system corrosion).

The staff understands that Germany may have identified a possible deficiency of the FCVS. It has indicated that, in some plants, the venting mass flow is not guided via a separate line to the end of stack; instead, it is mixed with the off-gas at the stack entrance. This would possibly allow that explosive H2-gas mixtures are released to buildings before they reach the vent outlet port. Germany has also indicated that it may provide further details in its ENSREG Stress Test in addition to the already-performed national stress test on the venting system design.

4.0 Summary

The staff notes that most countries did not rely on cost-benefit analysis to require FCVS on NPPs. The regulatory approach to severe accidents in many countries requires multiple improvements, such as accident management procedures, making equipment available to mitigate the accident (e.g., flooding, H_2 control), and training procedures, with filtered containment venting as one component. No single improvement would provide adequate management of severe accidents. To a large extent, they are interdependent for a successful management of a severe accident. The FCVS is an important aspect of the improvements, because it provides flexibility for the operators to vent the containment for unforeseen sequences without being overly concerned about releases. The preferred option is to keep containment intact without venting. However, filtered containment venting is considered a final option preferable to uncontrolled leakage or failure of containment if pressure within containment rises above the normal design pressure. Additional insights gained by the staff include:

- The regulator and licensee agreed on strengthening containing integrity as a goal, regardless of the calculated value of core damage frequency, and to improve defense in depth recognizing the uncertainties in PSAs.

- A level 2 PSA is performed to identify the benefit of severe accident management features to strengthen containment, using a large release and/or land contamination criterion.

- The regulator and the licensee agreed on features that will:

  (1) flood containment to prevent core concrete interaction, liner melt through by covering the core debris with water, and

  (2) remove heat by the FCVS, thus managing overpressure scenarios

- The regulator requires what are considered reasonable-cost safety upgrades consistent with progress in safety technology

The issues discussed under Section 4.0, “Unintended Consequences,” are resolvable through consideration in the design and operation of the FCVS. The foreign countries did not consider them to be serious enough to question the need for FCVS. Within the U.S., existing plant procedures for venting containment results in the venting of non-condensable gases. As such, the addition of FCVS does not introduce a new vulnerability. Rather, the vulnerability of containment implosion due to the removal of non-condensable gases already exists and is actively managed and minimized through plant procedures and controls.

A number of countries in Western Europe have provided FCVS on both PWRs and BWRs. Canada started implementing FCVS on their operating reactors, starting with Point Lepreau in 2007. After Fukushima events, a significant number of countries, including Japan, have started implementing FCVS or declared their intention to proceed with FCVS. A summary of the current status of FCVS worldwide, to the extent information is available, is provided in Table 1. The European Commission press release on October 4, 2012, states that out of approximately 145 reactors in the EU member states, only 32 reactors are not equipped with FCVS. Further, all the EU neighboring countries that responded to the stress tests have already installed FCVS or are in the process of doing so. For pressure suppression containments such as GE BWR Mark I and GE BWR Mark II, the countries that have not made a decision regarding the implementation of FCVS are the U.S., Spain, Mexico, and India.

5.0 References

1. NUREG-0660, “NRC Action Plan Developed as A Result of the TMI-2 Accident,” May 1980.

2. NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.

3. NUREG-0933, “Resolution of Generic Safety Issues,” (Formerly entitled “A Prioritization of Generic Safety Issues), Item CH3.2 “Filtered Venting.”

4. Generic Letter 89-16, “Installation of a Hardened Wetwell Vent,” September 1 1989.

5. Organization for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), CSNI Report 148, “OECD Specialist Meeting on Filtered Containment Venting Systems; Paris, France, May 17 – 18, 1988,” pp 32 – 39 (P. M. Herttrich and E. F. Hicken).

6. European Stress Tests for Nuclear Power Plants
   a. Swedish National Report, December 29, 2011
   b. National Report for Finland, December 30, 2011
   c. Complementary Safety Assessments of the French Nuclear Power Plants (European Stress Tests), Report by the French Nuclear Safety Authority, December 2011
   d. Swiss National Report, December 29, 2011
   e. Slovenia National Report, December 2011
   f. Netherlands Final Report, December 21, 2011
   g. Hungary National Report, December 29, 2011
   h. Slovenia National Report, December 31, 2011
   i. Slovakia National Report, December 30, 2011
   j. United Kingdom National Report, December 30, 2011
   k. National Report for Ukraine, 2011

7. 24th DOE/NRC Nuclear Air Cleaning and Treatment Conference, Session 6, Provisions for Containment Venting in Germany, by J.G. Wilhelm, Consultant, pp 371 – 373.

8. E. Rainmond, et al., “Continued Efforts to Improve the Robustness of the French Gen II PWRs with Respect to the Risks of Severe Accidents: Safety Assessment and Research Activities’,” Eurosafe, 2011, Pages 6 and 7.

9. Canadian Fukushima Task Force Report, INFO-0824, October 2011.

ENCLOSURE 4

BWR MARK I AND MARK II CONTAINMENT PERFORMANCE DURING SEVERE ACCIDENTS

Contents

BWR Mark I and Mark II Containment Performance during Severe Accidents

1.0 Introduction

This enclosure provides an overview of various plant design features that help protect boiling water reactors (BWRs) with Mark I and Mark II containments from certain severe accident challenges, a brief assessment of these design features for reducing radiological releases resulting from severe accidents, as well as an assessment of external containment filters commercially available today. In addition, Enclosure 4 provides the NRC staff's initial assessment of a report that was prepared by the Electric Power Research Institute (EPRI) on the topic of limiting radiological releases and made available to the public through its web site.

2.0 Containment Systems and Severe Accident Management

Emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and extreme damage mitigation guidelines (EDMGs) for BWRs with Mark I and Mark II containments provide strategies for protecting the containment under accident conditions or the loss of large areas of the plant. These strategies include use of drywell and wetwell spray systems and venting to remove heat, steam, and non-condensable gases from the containment, and protect the containment from structural failure as a result of overpressure challenges. In addition, if molten core debris were to melt through the reactor pressure vessel (RPV) and relocate to the drywell floor, the procedures instruct plant operators to flood the containment to assist in cooling the core debris, minimize core-concrete interactions and protect the containment wall (Mark I liner melt-through containment breach) and drywell floor penetrations (Mark II suppression pool bypass).

2.1 Containment Spray Systems

Containment heat removal may be accomplished during and after design basis accidents by the containment cooling modes of the residual heat removal (RHR) system. Containment cooling includes suppression pool cooling and containment spray (drywell and wetwell) modes. The containment spray mode is accomplished in most Mark I and II containments by diverting water flow from the RHR system to the drywell or suppression chamber spray headers. The purpose of these two RHR modes is to prevent containment temperatures and pressures from exceeding design values in order to maintain containment integrity following an accident. Under postulated accident conditions, water is drawn from the suppression pool, pumped through one or both RHR heat exchanger loops, and delivered to the drywell spray header or to the suppression chamber spray header. For design basis accidents, the RHR system is only realigned for the containment spray mode by the plant operator after verifying flow is not needed for RPV injection or suppression pool cooling. If the operator chooses to use containment spray, the associated low-pressure coolant injection (LPCI) valve to the core is closed (low pressure water sources are no longer sent to the RPV to cool the core) and the spray valves are opened. Under postulated accident conditions, the typical containment drywell spray system design flow rates range between 3,000 and 10,000 gallons per minute. If RHR pumps are not available, such as during an extended station blackout (SBO), the portable temporary pumps currently required by (10 CFR 50.54(hh)(2)) provide flow rates in the range of 100 to 300 gallons per minute.

2.2 Containment Flooding

Another severe accident management strategy included in EOPs, SAMGs and EDMGs is containment (drywell) flooding. The drywell flooding strategy is intended to provide water on the lower drywell floor should core melt appear imminent, or by the time a melted core breaches the RPV. Water on or around the core debris on the drywell floor serves to quench, immobilize, and inhibit the molten core debris from flowing across the drywell floor and melting through the drywell wall (i.e., Mark I liner melt-through containment breach) or penetrations that would result in bypassing of the suppression pool (i.e., Mark II suppression pool bypass). Water on the drywell floor would also reduce core-concrete interactions and the resulting flammable and non-condensable gases that contribute to containment pressurization. An additional strategy involves flood up of the containment into the drywell to a level as high as the top of the fuel zone elevation in the reactor vessel. This strategy is designed to provide RPV exterior cooling for the damaged core debris remaining in the vessel and water depth over exposed core debris.

BWRs with Mark I and Mark II containments are required to be capable of injecting water into the drywell by an AC-power-independent means as a result of Section B.5.b of Order EA-02-026, “Order for Interim Safeguards and Security Compensatory Measures,” the corresponding license conditions, and 10 CFR 50.54(hh)(2). Nuclear Energy Institute (NEI) 06-12, “B.5.b Phase 2 & 3 Submittal Guideline,” Revision 2, Section 3.4.9, identifies the objectives of injecting the water as providing cooling of core debris and scrubbing of fission products, in the event core damage and vessel failure cannot be prevented. The injection flow could use a portable pump or other existing sources. Detailed procedural guidance for implementing this injection capability is also required. The injection flow, using a portable pump or other existing sources, could be routed through the drywell spray system, emergency core cooling system, or any other system providing a suitable pathway to the drywell. Following core melt-through of the RPV, injection of water into the reactor vessel would reach the drywell floor through the opening in the RPV caused by the core melt-through. Although some scrubbing of fission products will likely occur, injection flows in the range of the required capability are primarily for decay heat removal and would not be expected to result in appreciable fission product decontamination of the containment atmosphere. The required AC-power-independent injection capability is 300 gallons per minute or less, and the low pressure portable pumps are not expected to provide much more flow than that through the entire range of flow resistances and back pressures that could be experienced.

The drywell flooding strategy may completely flood the wetwell within 12 to 24 hours, as the water drains from the drywell floor into the suppression chamber through the drywell to suppression chamber vent system. The amount of time to fill the suppression pool with water depends upon the portable pump’s flow rates, and how long these flow rates exceed the amount necessary to remove decay heat. Prior to the suppression pool becoming fully flooded with water and sealing off the wetwell vent penetrations, emergency procedures direct operators to vent the containment through the drywell without regard to the potential radiological consequences.

2.3 Containment Venting

The EOPs, SAMGs, and EDMGs for BWRs with Mark I and Mark II containments include provisions for venting containment prior to the pressure exceeding the primary containment pressure limit (PCPL). Due to the small size of the Mark I and Mark II containments and their response to severe accidents, the need for containment venting has been recognized for a long time. In 1983, the NRC approved Revision 2 to the Boiling Water Owners’ Group Emergency Procedure Guidelines which included guidance for operators to vent Mark I and Mark II containments in response to containment overpressure conditions. The Emergency Procedure Guidelines are used to develop plant specific Emergency Operating Procedures. In 1988, the NRC approved Revision 4 to the BWR Emergency Procedure guidelines, which provided improved guidance for venting, in particular guidance on establishing the containment vent initiation pressure. In approving venting for the BWRs with Mark I and Mark II containments, the staff noted its basic concern that:

> [V]enting even if it results in some radiological consequences should only be undertaken as an extreme means to prevent core melt or as a last resort measure to prevent the irreversible and unpredictable rupture of the containment which could otherwise lead to a large release.

Though procedures have existed for some time for Mark I and Mark II containment venting systems for beyond design basis accidents and severe accidents, the NRC's actions to date have not specifically required that plants with Mark I and Mark II containments be designed with systems, structures, and components to limit the releases from potential beyond design basis scenarios, such as an extended station blackout involving significant core damage and an inability to remove energy from the suppression pool (primary containment) by means other than containment venting. In the staff's evaluation of Revision 4 to the emergency procedure guidelines, the staff noted the following concerns with venting wherein the venting systems where not designed for the expected loadings:

> However, there are downsides to a strategy which intentionally releases containment atmosphere to the reactor building or the environs. If the vent path is not capable of bearing the associated pressure and consequently ruptures upon initiation of venting, then the reactor building could become highly contaminated and operator access will be impractical. Thus, recovery of failed equipment may be prevented. Further, rupture of a vent line in the reactor building will unnecessarily threaten the functioning of safety equipment or instrumentation which was operating by exposing that equipment to a high temperature, steam, and radiation environment.

In 1989, the NRC issued Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," to all licensees of BWRs with Mark I containments to encourage licensees to voluntarily install a hardened wetwell vent. In response, licensees installed a hardened vent pipe from the wetwell to some point outside the secondary containment envelope (usually outside the reactor building). Some licensees also installed a hardened vent branch line from the drywell. Because the modifications to the plant were performed in accordance with 10 CFR 50.59, "Changes, tests and experiments," detailed information regarding individual plant configurations was not submitted to the NRC staff for review.

On March 11, 2012, the NRC issued an order (EA-12-050) to all licensees of BWR facilities with Mark I and Mark II containment designs to require a reliable hardened vent (RHV). The order provided requirements to ensure reliable operation of the hardened venting system in support of strategies relating to the prevention of core damage. EA-12-050 did not include requirements for reliable operation under severe accident conditions. Because the order focused on requirements prior to the onset of core damage, EA-12-050 did not prescribe the venting location (drywell or wetwell) as essentially all vent flow prior to RPV breach would pass through the suppression pool regardless of vent origination from wetwell or drywell. Nevertheless, the existing EOPs, SAMGs, and EDMGs for BWRs with Mark I and Mark II containments contain provisions for venting containment following core damage.

2.3.1 Wetwell Venting

Venting from the wetwell is preferred because a wetwell vent ensures the maximum available decontamination scrubbing action from the suppression pool. However, there are circumstances where suppression pool scrubbing may be bypassed or, otherwise, unavailable. For example, wetwell venting would not be available in the event of failure of the venting valves, loss of motive power to venting valves, lack of operator access to actuate the venting valves, or high level in the suppression pool.

A reactor vessel breach would result in a flow of the drywell atmosphere to the wetwell via the downcomer pipes with much-reduced scrubbing effect when compared to releases through the safety relief valve lines. In addition, the suppression pool may be bypassed if efforts employed by operators to flood the lower drywell floor are unsuccessful and result in a Mark I drywell liner melt-through or a Mark II vessel drain line or downcomer melt-through. Also, as previously noted, wetwell venting may become unavailable within 12 to 24 hours following efforts to flood the drywell floor under the RPV in order to prevent the complete bypass of containment.

2.3.2 Drywell Venting

EOPs and SAMGs direct operators to vent the containment to avoid exceeding the primary containment pressure limit (PCPL) or avoid combustible gas concentrations in the primary containment. Venting from the wetwell is the preferred venting path; however, if the wetwell vent is not available or effective at reducing pressure or hydrogen concentration, then the operators are directed to vent from the drywell regardless of the radiological release consequences. This is in accordance with existing procedures.

A drywell vent would provide the same suppression pool scrubbing for the steam, radionuclides, and hydrogen gas that is discharged into the suppression pool via the safety-relief valve discharge line and T-quenchers. In this case, the wetwell atmosphere (i.e., nitrogen/air, steam, and other non-condensable gases) exhausts to the drywell atmosphere via vacuum breakers, and the resulting drywell atmosphere is vented. However, for accident sequences involving breaks in piping within the drywell or for accident sequences where the molten core exits the RPV, any discharge from drywell venting would be unscrubbed by the suppression pool.

A drywell vent, especially if it exits high in the drywell, will discharge more drywell heat and hydrogen, and reduce the potential for drywell penetration gross leakage and the amount of hydrogen available for leakage into the secondary containment (reactor building).

3.0 Containment Design Features to Limit Radiological Releases

3.1 Decontamination by Drywell Spray

In international severe accident strategy, the drywell spray headers are used as the pathway for getting water into the primary containment to cover core debris and to provide makeup for feed-and-bleed heat removal using the filtered containment venting system. This provides a means to stabilize the core melt to protect penetrations and avoid containment breach and bypass. The spray is not relied upon for fission product removal because the decontamination provided by the limited capacity severe accident spray has not been demonstrated to provide sufficient coverage and performance.

Reactors with Mark I and Mark II containments have drywell spray systems or subsystems for design basis accidents. Their function is to provide a means of containment pressure control and, using emergency service water cooled heat exchangers, to remove heat from the containment. They were not designed or intended for aerosol particle decontamination. Drywell spray pumps and valves are dependent on alternating current (AC) electrical power, and are not functional in a prolonged station blackout as was experienced at Fukushima. The drywell spray equipment useable under prolonged SBO is the passive drywell spray ring headers. Their use also presumes a flow path to the header unobstructed by several inoperable valves. Because of the potential for opening containment vacuum breaker valves and letting air/oxygen in, or of collapsing the containment by inadvertently operating drywell spray, the decision to initiate containment sprays requires due consideration, even at the low flow rates considered for severe accident purposes. The use of containment sprays might also present a concern due to the potential for condensing steam. Steam assists in maintaining an inert environment in the containment to avoid any burning of hydrogen gas produced during a severe accident, in the event air is introduced into the containment.

In contrast with BWRs with Mark I and Mark II containments, PWRs with large dry containments have containment spray systems that were originally designed to provide a decontamination function and many included a means to add pH elevating chemicals to the spray flow for improved iodine retention in the emergency sump water**.** The testing of spray for PWR atmosphere decontamination has been performed in geometries that attempt to model the large free volumes of large dry PWR containments. The vast majority of this testing has been performed by France, which uses large dry containments for their PWRs (they have no BWRs) and the results are not in the public domain. The spray testing cited in the literature and known to the staff consists of 20 data points for a single set of steady state conditions in a large volume from an experiment by the Department of Energy (DOE), Office of Scientific and Technical Information (OSTI), and documented in BNWL-1592, “Removal of Iodine and Particles from Containment Atmospheres by Sprays: Containment Systems Experiment Interim Report,” July 1971.

The many variables and uncertainties which must be understood to assess the value of drywell spray for fission product decontamination using computer models include: the rate and pressure of the flow through the drywell header nozzles, which affect droplet size, spray trajectory, and velocity; the volume of the drywell that will be swept by the spray due to drywell geometry, structures and equipment installed in the drywell between the spray header and the drywell floor; the height through which the spray droplets will fall; the thermodynamic conditions in the containment that will affect spray distribution, e.g., convection currents; and the uncertainties inherent in modeling complex aerosol physics, in particular the removal efficiencies. The uncertainties in modeling aerosol physics have been exhaustively analyzed in NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” June 1993. Estimates for drywell spray decontamination factors, including estimates of uncertainty, were calculated in NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," for Mark I containments and the results are shown in Figure 1.

The 2009 Organization for Economic Co-operation and Development (OECD) report, the Nuclear Energy Agency Committee on the Safety of Nuclear Installations, “State-of-the-Art Report on Nuclear Aerosols,” December 2009 (ADAMS Accession Number ML11355A245), gave the following summary of the state of knowledge for fission products in the containment atmosphere:

Mixed aerosols in condensing atmospheric conditions: Although there has been considerable progress in modeling aerosol deposition as a function of relative humidity, a comparison of the adequacy of code results from ISP 37 and ISP 44 indicate that there is still some work to be done to ensure satisfactory coupling between thermal hydraulic and aerosol models so that these capture correctly aerosol behavior in most environments. An additional uncertainty in modeling aerosol behavior in the containment in humid conditions arises from determining the hygroscopicity associated with a mixture of aerosols of different compositions. Finally, there is some uncertainty regarding the density of multi-component aerosols, and whether this parameter is important for accident conditions with a wide variety of aerosol components.

Removal by sprays: This issue has been extensively investigated by the French organizations CEA and IRSN using specific apparatuses and the CARAIDAS, MISTRA and TOSQAN test facilities. The data should be made accessible to the nuclear community, at least the OECD partners. Validated modeling based on these experimental investigations has been implemented in the codes ASTEC and TONUS. The ASTEC model can be found in the open literature. Further work on containment sprays is low priority for countries that have access to this data but in other countries and for certain advanced designs it remains important to establish effective removal by spray systems and both experimental and analytical efforts continue.

With respect to the Mark I containment spray system, the staff reached the following conclusion through the Containment Performance Improvement Program (CPIP):

> A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for the drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared to having the water run out of the spray nozzles. Fission product removal in the small crowded volume in which the sprays would be effective was judged to be small compared to the benefit of having a water pool on top of the core debris. Therefore, modifications to the spray nozzles are not considered warranted. (SECY 89-17)

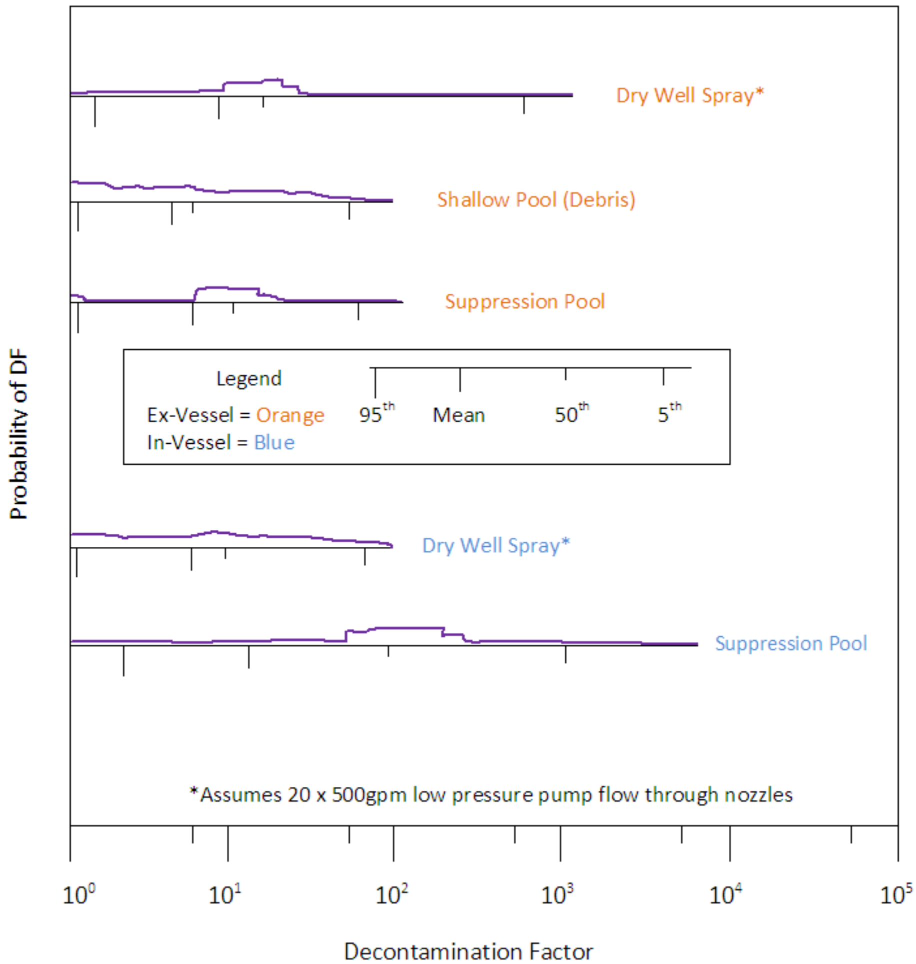


Figure 1 – Uncertainty distributions for Cesium decontamination factors (DFs)
Mark I Containment – Peach Bottom

Source: "Assessment of In-Containment Aerosol Removal Mechanisms."
BNL Technical Report L-1535, 1992

3.2 Decontamination by the Wetwell (Suppression Pool)

BWR Mark I and Mark II pressure suppression primary containments include a large pressure suppression water pool within a pressure suppression chamber (wetwell). As the name "suppression pool" implies, the wetwell was designed to condense steam from a design basis accident and limit the peak design basis accident pressure in the relatively small total volume of the drywell/wetwell combination. The suppression pool was not designed with a fission product decontamination function in mind. However, because of its size (depth and capacity) and the possible routing of fission products through the pool prior to release from containment, it has been analyzed as a passive "ad hoc" filter for severe accident mitigation. This was the basis for preferring a wetwell hardened vent in Generic Letter 89-16, "Installation of a Hardened Wetwell Vent."

3.2.1 Mark I Containments

As a potential fission product filter, the wetwell has its greatest value when (1) the core damage is arrested in the reactor vessel, (2) the reactor vessel and attached piping remain intact relieving through the safety relief valves (SRVs), (3) the SRV tailpipes to the T-quenchers (spargers, pipes with many holes approximately 1 centimeter in diameter to spread the discharge and assist with pool mixing to avoid local boiling and containment pressurization above the pool) at the bottom of the wetwell remain intact, and (4) the wetwell water remains substantially subcooled. At Fukushima Units 2 and 3, extended reactor core isolation cooling (RCIC) and high pressure coolant injection (HPCI) operation resulted in SRV discharge pathway transfer of enough decay heat from the RPV to the suppression pools to bring them to saturation conditions.

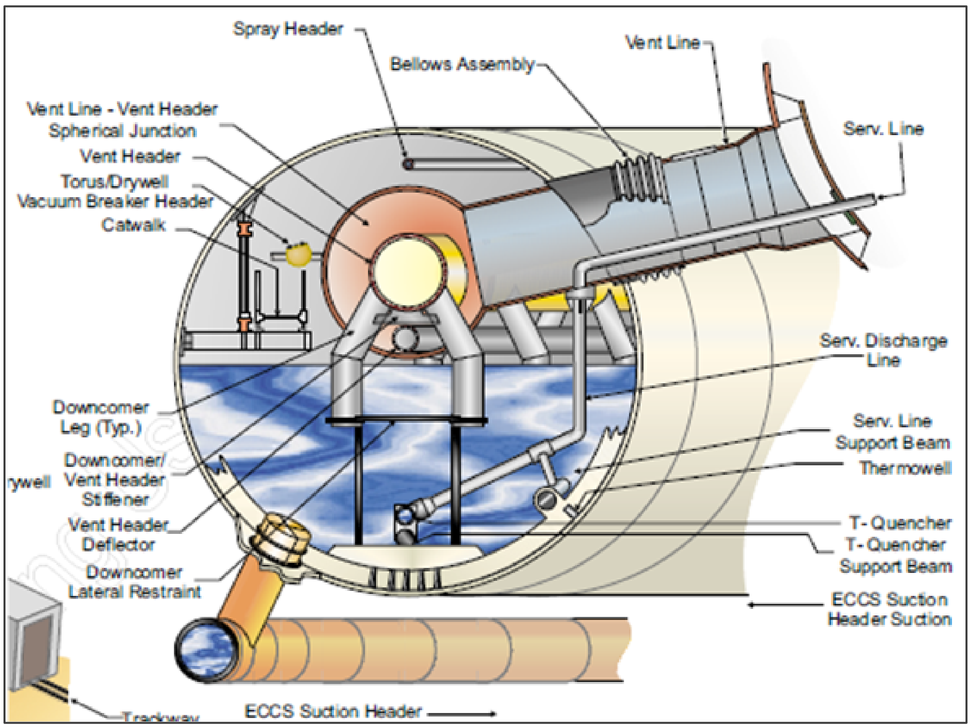


Figure 2 - Suppression chamber cross-section

The suppression pool scrubbing effect diminishes as the pool temperature approaches saturation, especially when non-condensable gasses are a significant portion of the flow entering the pool. A decrease in pool water pH (i.e., acidification) also results in further reducing the scrub effect by lessening the capture and retention of iodines. SRVs discharge near the bottom of the suppression pool through diffusers (T-quenchers). When the reactor vessel boundary is breached, the reactor vessel communicates directly with the drywell and the flow path into the suppression pool is via the downcomers, which are many large pipes with open ends with much less submergence in the suppression pool. The decay heat generated steam and non-condensable gas flow through these downcomer pipes hours after reactor shutdown would be relatively non-energetic (low velocity) and removal of entrained aerosol radionuclides via this pathway is thus much less effective than via the SRVs. With the transition from SRV discharge to downcomer discharge, the bottom third of the suppression pool becomes thermally uncoupled from the upper portion requiring less decay heat passing through the downcomers to keep that upper portion of the pool involved with scrubbing at or very close to saturation temperature. A cross-section of the Mark I suppression chamber is provided in Figure 2.

Some important variables and uncertainties in calculating the DF for the wetwell are: pool temperature, submergence of injection point, size of the bubbles, injection flow velocity and gas composition (percent noncondensables) and temperature. Other variables related to the physics of aerosol removal are also important and uncertain, but probably less so than the variables mentioned, with one exception, which is important in considering the efficiency of an engineered filter on the vent from the air space of the wetwell. That variable is the distribution of aerosol particle sizes leaving the wetwell and going through the engineered filter, if installed. The physical processes involved in wetwell or pool scrubbing are described and analyzed in NUREG/CR-6153, “A Simplified Model of Decontamination by BWR Steam Suppression Pools.” The overall DF of the suppression pool and external filter or drywell spray and an external filter is not a direct multiple of their individual DFs given that the filtration efficiency is different for different particle sizes. This is not an overriding concern since currently available external filters have very high removal efficiencies for even the most difficult particle sizes. Wetwell decontamination factors were calculated in NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," for Mark I containments. The calculated estimates and uncertainties are shown in Figure 1 and Table 1**.**

Table 1 – Uncertainty Distributions for Cesium Decontamination Factors (DFs)
Mark I Containment Suppression Pool – Peach Bottom

	Decontamination Factor (DF)			
Conditions	5th Percentile	Median	Mean	95th Percentile
During In-vessel Release Phase (through T-Quenchers)				
Peach Bottom	2.3	81	14.5	1,200
LaSalle & Grand Gulf	1.8	56	10.5	2,500
During Ex-vessel Release Phase (through Vent Pipes)				
Peach Bottom	1.2	9.5	5.1	50
LaSalle & Grand Gulf	1.2	6.8	4	72

Source: “Assessment of In-Containment Aerosol Removal Mechanisms.”
BNL Technical Report L-1535, 1992

The 2009 OECD state of the art report (SOAR) gave the following summary of the state of knowledge for wetwell (pool) scrubbing:

> ***Pool scrubbing***: Some BWR and PWR severe accident scenarios involve transport of radioactive aerosols through pools of water where particles can be retained. This phenomenon, known as pool scrubbing, has the potential to

reduce the source term. Results provided by both stand-alone and integral code models indicate satisfactory agreement with simple experiments for integral retention. However, a systematic experimental database is required for validation purposes. Particular attention should be given to removal of aerosols during formation and subsequent disintegration and coalescence of bubbles, and the effects of submerged structures and contaminants (surfactants).

3.2.2 Mark II Containments

In the Mark II containment design, a severe accident proceeds in a similar manner to that in a Mark I containment. Before vessel breach, the SRVs discharge to the bottom of the suppression pool and aerosol fission products not retained in the suppression pool pass into the drywell with accumulated gasses via the suppression chamber-to-drywell vacuum breakers. Barring significant leakage from the RPV and attached piping boundary in the drywell, any containment atmosphere leakage or vent discharge from either the wetwell or drywell benefits greatly from suppression pool scrubbing. Once the core debris breaches the bottom of the RPV, SRV flow to the suppression pool ceases and any steam and other noncondensables generated will enter the suppression pool via the downcomers , unless exiting containment via a drywell vent. However, molten core debris on the drywell floor may enter and melt through and breach the drain lines or downcomer pipes that pass through the drywell floor. When this happens, there is a direct pathway from the drywell to suppression chamber atmosphere and nearly all the scrubbing subsequently performed by the suppression pool is of that portion of the core debris that falls into and is submerged in the pool. Analyses of severe accident progression have concluded that this bypass of the suppression pool in Mark II containments may occur soon after molten core debris reaches the floor under the reactor vessel.

The details of the design of the Mark II containment drywell floor directly below the reactor vessel, the in-pedestal region, greatly affects the accident progression, and thus the uncertainty in predicting consequences of a severe accident. The design of this in-pedestal region varies from plant to plant (Figures 3 and 4). The Nine Mile Point 2 containments have downcomers inside the pedestal region. The La Salle, WNP-2 and Nine Mile Point 2 primary containments have an in-pedestal region at a lower elevation than the surrounding ex-pedestal drywell floor. Nearly all Mark II containments have drain lines through the in-pedestal drywell floor. Failure of a drywell floor penetration (drain line or downcomer), or the floor itself (by core-concrete attack and stress from the core debris weight) would allow fission products in the drywell atmosphere to bypass the suppression pool, thus resulting in much higher release of radioactivity via a hardened vent, even if from the wetwell air space.

NUREG/CR-5528 stated for the Mark II containment:

> [G]iven a severe core damage accident, there is a 55% chance of recovering the sequence in-vessel, with no significant release from containment. Should the sequence progress to vessel failure, there still is a 24.9% chance of establishing a coolable debris bed inside containment, again with no significant release to the environment. However, there is an 11.8% chance that a severe core damage sequence will lead to early overpressure containment failure. Of these early failures, ~90% will involve suppression pool bypass, because of either in-pedestal drain line failure or a failure location in the drywell.

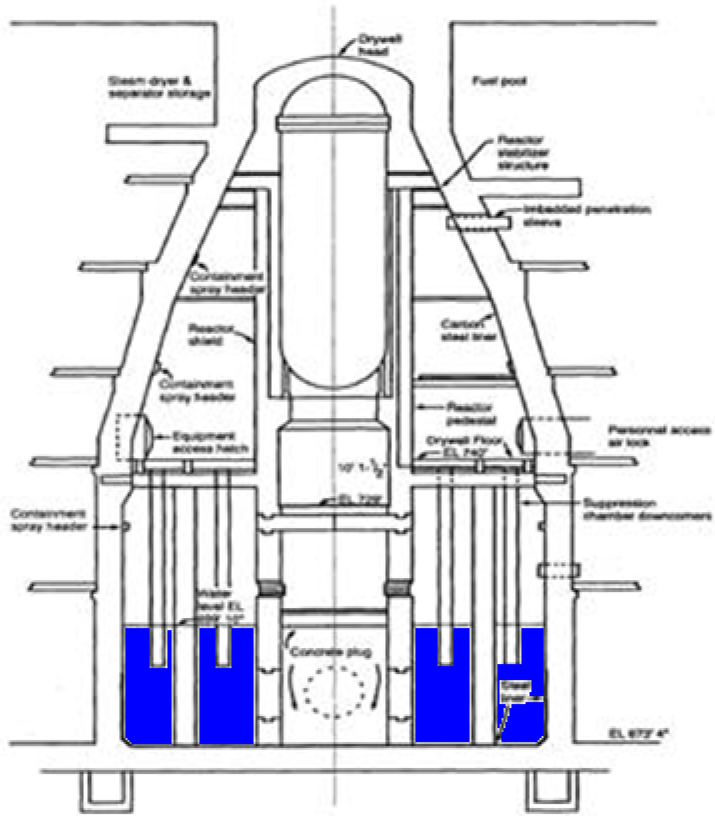


Figure 3 – BWR Mark II containment with lowered floor below RPV (pool not below floor under RPV)

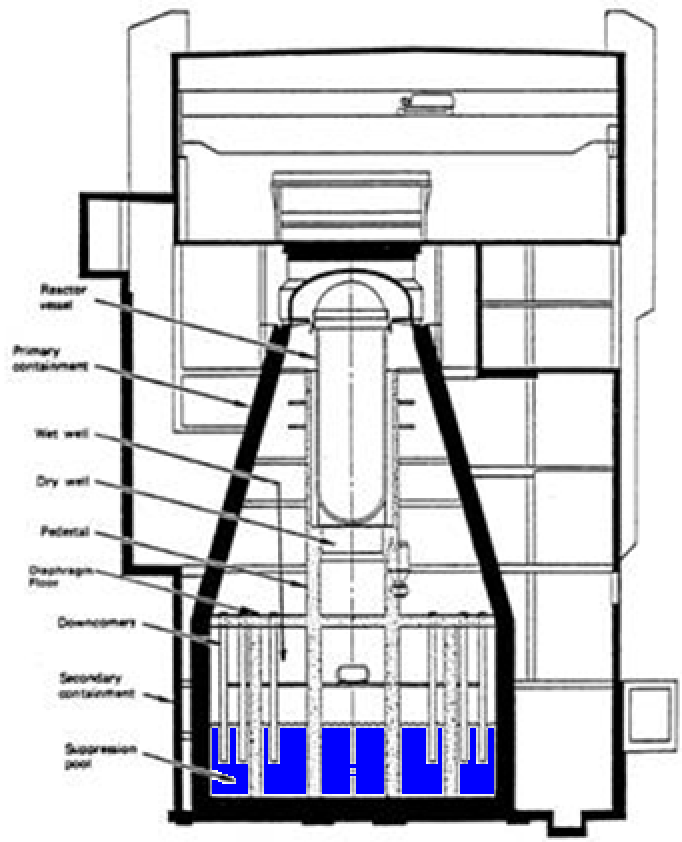


Figure 4 – BWR Mark II containment with level floor (pool below floor and under RPV)

3.3 Decontamination by External Engineered Filter Systems

Engineered containment external filter systems deployed throughout the world have evolved considerably since the first gravel bed filter was installed at the Barsebäck Nuclear Power Plant in Sweden in the mid-1980s. Since that time, engineers have been able to significantly reduce the physical size of the filter and improve the decontamination efficiency for iodine and aerosols. In particular, designers have developed and tested the technology to better retain organic iodine, and to trap more of the most penetrating aerosol particle sizes (less than one micron), those in the mid-range referred to as “the filter gap.”

The benefits of current filter designs, shown in Table 1, rest primarily on extensive full-scale vendor testing. Many of the individuals involved in this testing participate as experts in international efforts such as the preparation of the OECD/SOAR on aerosols referenced earlier. The validity of the testing has been accepted by regulators and plant owners and operators outside the U. S. In preparing this paper, the staff had extensive interaction with foreign regulatory authorities and owner/operators of plants equipped with primary containment external filters (see Enclosure 3, “Foreign Experience.”) The staff was also briefed by representatives of AREVA, IMI Nuclear, Paul Scherrer Institute, and Westinghouse. During the public meetings, AREVA, IMI Nuclear, PSI and Westinghouse provided extensive information regarding filter designs, capabilities and validation testing.

3.3.1 Removal of Radioactive Aerosols

The staff's assessment did not have the benefit of independent testing of the current filtered vent technologies. However, the staff notes that two vendors are getting similar results using multi-venturi nozzle sparger arrays. In 1992, the Electric Power Research Institute (EPRI) published the results of extensive third-party testing of eight filter designs of late 80s vintage as part of the Advanced Containment Experiments (ACE) Project. The testing of the containment venting filtration devices was done by Westinghouse Hanford Company as a subcontractor to Battelle Pacific Northwest Laboratories. Both DOE and NRC were members of the consortium led by EPRI.

Decontamination Factor (DF) values claimed and/or warranted by the current containment filter vendors are shown in Table 2. These values are consistent with DF values measured in the ACE Program. The staff notes that the sand and gravel filters are considered obsolete as the size/volume of the filters necessary to achieve the DFs makes them impractically large for installation at most nuclear plant sites.

External wet filters are specifically designed for achieving high DFs when operating at saturation temperatures. Vent flow enters the filter pool through either high speed venturi nozzles or high speed convergent jet nozzles and impingement/baffle plates. The resultant process maximizes the interface area of the filter liquid and the high relative velocity of entering gas for maximum particulate capture across the particle size distribution. Subsequent bubble rise is either

Table 2 – Containment Severe Accident External Filter Designs

Type	Aerosol Particulate DF	Elemental Iodine DF	Organic Iodine DF	Current Vendor
Dry – Sand Bed	100	10		Installed on French PWRs, design not currently marketed
Dry – Large Gravel Bed	10,000	100		Swedish FILTRA project early design installed at Barseback, not currently marketed
Wet – Multi-venturi + water pH elevation + metal fiber filter	10,000	10,000	5	Westinghouse FILTRA-MVSS
Wet – Multi-nozzle + impingement plates + mixing elements + elevated pH and enhanced iodine capture and retention chemistry	10,000	1000	1000	IMI (Paul Scherrer Institute, PSI-CCI AG)
Wet – Venturi + Metal Fiber	10,000	200		AREVA FCVS
Dry – Metal Fiber + Silver Zeolite	10,000	100	10	Westinghouse Dry Filter Method (DFM)

Note: Decontamination Factors (DFs) are the filter vendor literature stated minimums for a defined range of operating variables with the dominant variable being vent flow rate.

through a deep water pool or through a mixing section that ensures a long dwell time with small bubbles for maximum diffusion capture of aerosol small particles. Alternatively, many filter designs have a second stage filter of small diameter metal fiber beds that remove water droplets and small aerosol particles. See Figures 5 through 7 showing the design features utilized by various filter manufacturers that are currently available on the market.

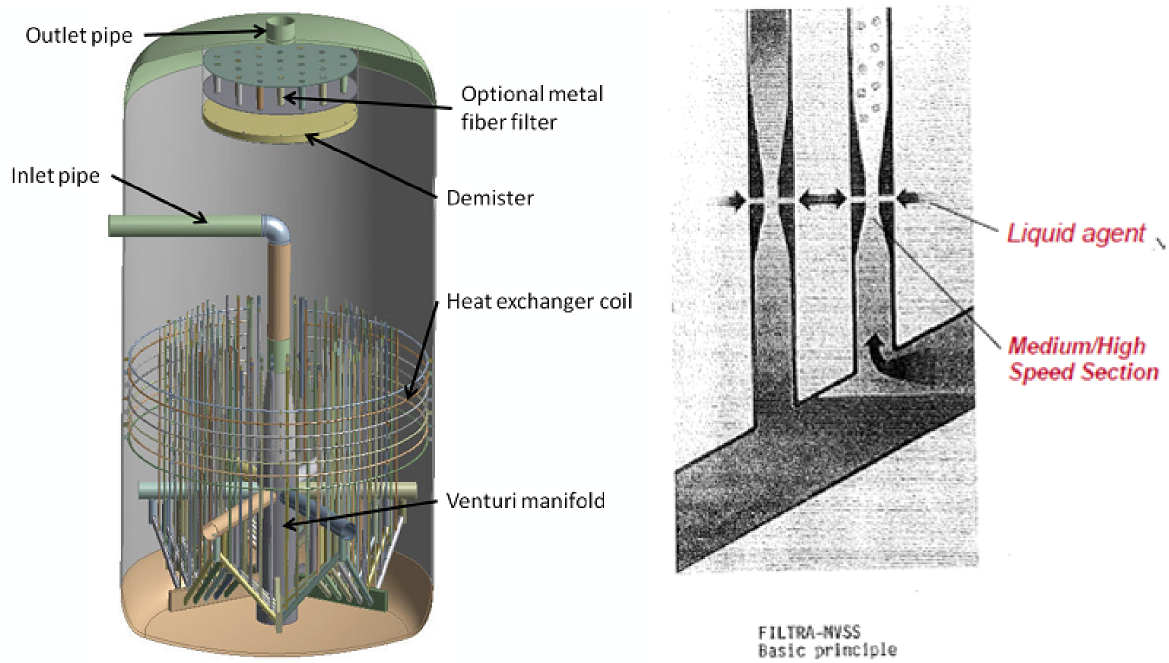


Figure 5 – Westinghouse FILTRA/MVSS multi-venturi scrubber technology

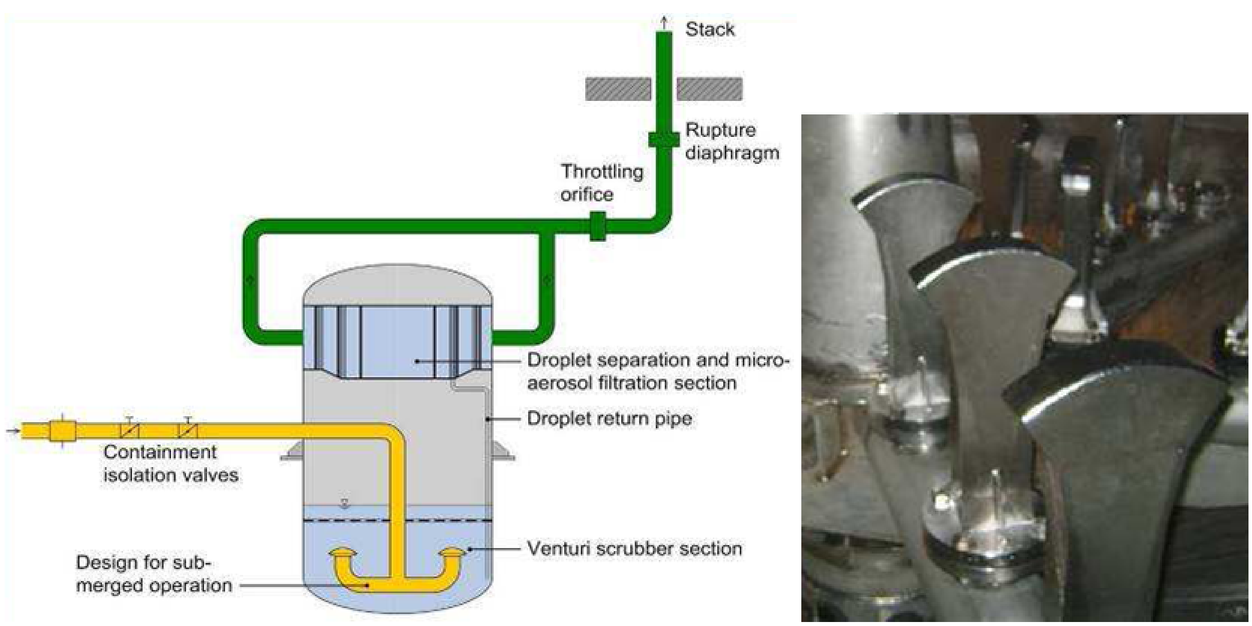


Figure 6 – AREVA venturi nozzle filter technology

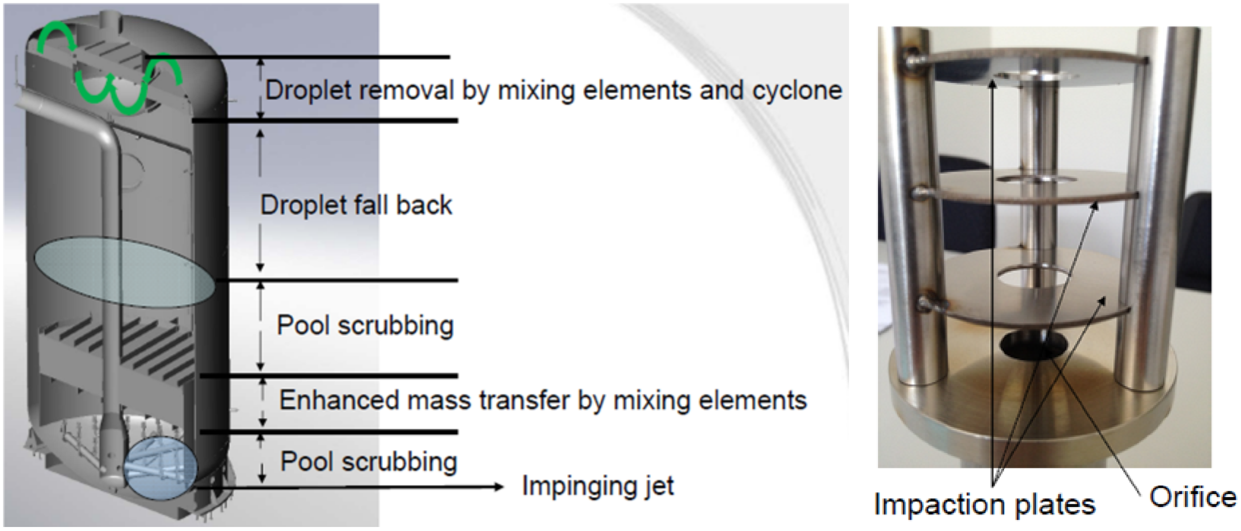


Figure 7 – IMI Nuclear (CCI) filter technology with impaction plate nozzles

3.3.2 Removal of Iodine

Chemicals present in the filter pool water elevate the pH and catalyze iodine's for high levels of short term retention in the filter pool. Some use additional chemicals for essentially 100 percent iodine capture and long term retention. Some foreign PWRs have used dry filters of metal fibers and silver zeolite beds where venting strategy involves delay of more than a day after reactor shutdown resulting in much lower decay heat loadings consistent with the heat dissipation capability of the metal fiber filters and the longer time for particulate settling in containment, reducing the likelihood of fiber filter clogging and blockage.

3.3.3 Wet vs. Dry Filter Technology

The wet filters appear more suitable for venting BWR containments as the filter can be placed in service early in the event given their inherent higher tolerance for particulate loading and decay heat dissipation capacity. The wet filters with venturi nozzles achieve a high DF over a large containment pressure range by having nozzles inject into the filter pool at different elevations such that the nozzles are operating at high efficiency through a large filter flow range. This does require a deeper pool and thus a larger filter vessel. Alternatively, the filter outlet line can be throttled to choke and control flow such that all filter venturi's can be at the same injection submergence and near constant nozzle flow velocity but the filter operates at a higher pressure. This achieves a somewhat different form of "sliding pressure control" and allows for a smaller filter vessel size, but may limit the rate of containment depressurization. Wet filters can be designed/sized with a water volume capable of from 24 hours to several days' operation without operator action. With wet filters, water over the injection nozzles forms a loop seal, thus the containment would not depressurize all the way down to atmospheric through the filter. Most existing wet filter installations include a nitrogen blanket within the filter and inlet/outlet piping to maintain inert conditions for combustible gas control and minimize chemical degradation.

4.0 Electric Power Research Institute (EPRI) Evaluation of Severe Accident Venting Strategies for Mitigation of Radiological Releases

4.1 Background

On September 25, 2012, the Electric Power Research Institute (EPRI) published a study relating to BWR Mark I and Mark II containment venting. The report titled, "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents - *BWR Mark I and Mark II Studies*," (EPRI Final Report 1026539), was made available to the NRC staff through EPRI's public Web site (http://my.epri.com/portal/server.pt?Product_id=000000000001026539). The report was not provided directly to the NRC, and it is not expected to be formally submitted to the staff for review.

The purpose of the report was to document research on investigations into potential strategies for reducing the environmental and public health effect consequences of severe reactor accidents. The essence of the report was also the subject of two public meetings. On August 8, 2012, the staff held a public meeting where representatives from EPRI provided an overview and preliminary results of the research efforts documented in the September 25 report. In addition, EPRI briefed the Advisory Committee on Reactor Safeguards (ACRS) Fukushima Subcommittee on September 5, 2012, providing information relating to computer modeling and preliminary evaluation of strategies for mitigating radiological releases during severe accidents at BWRs with Mark I and II containments.

By letter dated October 5, 2012, the Nuclear Energy Institute (NEI) presented the industry's position with respect to possible implementation of the results of EPRI's research. In the letter, NEI recommended that the NRC staff pursue a more performance-based approach to ensure that radionuclide aerosols are filtered and retained in containment during severe events. NEI stated that:

> [EPRI's] findings demonstrate that substantial decontamination factors for radioactive releases can be achieved by a comprehensive strategy that includes installed equipment, operator actions and capabilities that are largely consistent with the diverse and flexible coping strategy (FLEX).

In addition, the October 5th letter stated that:

> A combination of these actions would result in 99.9 percent removal of radionuclides that have the potential to contaminate the environment. (They provide for a containment system decontamination factor (DF) of greater than 1000, which is a common international requirement.)

The following represents the NRC staff's preliminary assessment of EPRI's September 25, 2012, study. Because of the report's timing, and the fact that it was not submitted to the NRC for review, the staff is only able to provide its initial impressions of the report.

4.2 Overview

The EPRI report evaluates certain strategies that are intended to maintain or enhance the containment function in scenarios involving long-term loss of electric power. The strategies evaluated include water injection (by flooding or spraying), alternative containment heat removal, venting, controlled venting, filtered venting, and combinations of these plant features. Based on the results of its research, EPRI noted seven “key insights” from the analysis, including:

- No single strategy is effective
- Active core debris cooling is required
- Existing severe accident management guidelines (SAMGs) strategies provide substantial benefit
- Spraying the containment atmosphere is beneficial
- Venting prevents uncontrolled release and manages hydrogen
- Control of the vent provides benefit
- Low-efficiency filters can further reduce radionuclide releases

The staff is in general agreement with many of the report’s insights; however, many concerns remain about strategies that use existing containment features and their ability to achieve a dependable and adequate decontamination of radionuclides following a severe accident. The staff’s preliminary assessment of EPRI’s key insights is presented below.

4.3 Staff Assessment

4.3.1 No Single Strategy is Effective

The EPRI report concluded that “no single strategy is optimal in retaining radioactive fission products in the containment system.” The NRC staff agrees with this conclusion. Uncertainties surrounding severe accidents resulting from accident progression, status of plant systems and components, and operator response make it highly unlikely that accidents can be modeled and procedures developed to account for all potential scenarios.

4.3.2 Active Core Debris Cooling is Required

The insights presented included confirmation that sufficient water injection into the drywell was needed, whatever the pathway, to cool core debris on the drywell floor to immobilize it and prevent molten core debris flow out to and melt through of the drywell wall in Mark I containments or of the downcomer or drain pipes in the drywell floor below the reactor vessel in Mark II containments.

The staff agrees that an active debris cooling strategy is essential to protecting the containment wall at drywell floor level in Mark I containments, and it supports the following conclusion:

> Core debris cooling is an important element of a robust strategy for mitigating releases. If debris cooling is not provided through water injection or spray into the drywell, containment failure or bypass is likely. Without core debris cooling, the containment can be challenged in several ways. Molten debris can come into direct contact with the containment wall, melting the liner and providing a release path to the environment. Elevated drywell temperatures in the containment atmosphere can cause seals and other containment penetrations to fail, leading to containment bypass. Finally, core–concrete interactions can generate large quantities of noncondensable gases that increase containment pressure and also can accelerate concrete erosion that could challenge containment integrity over time.

The analysis also confirmed that Mark I drywell wall breach would largely negate any additional benefit of a hardened vent and external filter, if installed, in reducing releases or in preserving secondary containment (reactor building) accessibility and subsequent usefulness of equipment installed there for stabilizing plant conditions and avoiding or minimizing additional releases.

Mark II containment downcomer or drain line breach would result in suppression pool bypass and a potentially marked increase in radioactivity released if an external filter was not in the vent pathway.

4.3.3 Existing SAMG Strategies Provide Substantial Benefit

The EPRI study also addressed strategies defined in existing Severe Accident Management Guidelines (SAMGs). The guidelines assist operators with symptom-based strategies and include provisions for active debris cooling and containment flooding by using temporary portable equipment. However, the ability of portable pumps to provide sufficient flow rates and provide even limited decontamination of radionuclides raises serious doubts. Drywell spray systems are designed for flow rates that range from 3,000 to 10,000 gallons per minute (GPM). Portable pumps normally provide a maximum flow rate of 300 GPM; however, some pumps may provide up to 500 GPM but require larger and heavier hoses that are more difficult to position for use. As discussed further in section 4.3.4, the staff is concerned that reduced capacity drywell sprays will not provide a reliable means to scrub radioactive aerosols to sufficiently limit releases during venting operations.

4.3.4 Spraying the Containment Atmosphere is Beneficial

The staff recognizes that spraying the drywell atmosphere provides a benefit; however, because of inherent uncertainties in spray systems' capability to provide adequate decontamination factors (DFs), questions always remain as to how much, and whether or not they are reliable. The Mark I and Mark II containment drywells are highly congested areas that contain numerous piping systems (e.g., reactor recirculation, emergency core cooling). In addition to the piping itself, there are numerous piping supports, snubbers, sway struts, catwalks, and other interferences that limit the spray systems' ability to provide adequate spray coverage even under ideal conditions. Therefore, the ability of computer models to accurately calculate decontamination factors presents a significant challenge.

The report presented an optimum outcome and involved a water injection flow rate of 500 GPM. This would be well in excess of what is needed for decay heat removal, and it will maintain considerable suppression pool subcooling while providing some drywell spray scrubbing of the containment atmosphere. The staff considers this spray scrubbing to be very limited given the spray headers are typically designed for several thousand gallons per minute flow rate (up to 10,000 GPM) and flow rates of 500 GPM or less would yield a spray of pattern, droplet size and velocity with minimum decontamination potential, especially with obstructions in the drywell removing most of the spray flow from the atmosphere long before reaching the floor. The benefit of this low spray flow beyond pool subcooling may be more from the cooling of core debris on the floor and cooling of drywell surfaces for better aerosol settling and plate-out with less revolatilization.

4.3.5 Venting Prevents Uncontrolled Release and Manages Hydrogen

The severe accident scenarios evaluated in this report assume that core debris is discharged into the containment. As previously noted, water is needed to cool the debris. The quenching of the debris is beneficial; however, it produces a large amount of steam which contributes to containment pressurization. Unless active heat removal systems are available to remove the steam, pressurization will continue beyond containment design pressure to the point of containment failure. Therefore, even if water is available to cool the core debris, containment venting is required to avoid containment failure. Venting also helps manage the buildup of hydrogen and other noncondensable gases generated during the core melting and relocation process. Up to 20 percent of the pressure inside containment can be the result of hydrogen and other noncondensable gases. Venting could maintain the containment pressure below the design pressure and removes hydrogen and other gases from containment.

4.3.6 Control of the Vent Provides Benefit

The innovative feature developed in the EPRI study involve the active management and control of containment venting by plant operators during severe accident conditions in order to achieve sufficient decontamination of radioactive aerosols to limit releases to the public. The report concludes:

> The key to controlling the amount of radioactive material released to the environment is minimizing the amount of contaminants that are airborne in containment during venting. Opening and closing the vent at the most appropriate times is essential. Such controlled venting strategies could be beneficial, but additional analysis is needed to more fully understand this option and ensure coordination with the plant's emergency procedures.

As previously noted, there are many unknowns and variables that affect the conditions in the containment during in a severe accident. These unknowns include:

- pump start and stop times
- ability to sustain an injection flow rate close to 500 GPM
- severe accident phenomenological uncertainties
- rate of hydrogen generation
- success in setting up emergency pumps
- timing and availability of AC power
- battery life
- human reliability
- collateral damage from external events

The strategy presented would require a significant number of operator actions in order to obtain the decontamination factors achieved by the model. Operators must actively manage containment DF by simultaneously controlling containment pressure, water level and temperature (and hydrogen) under conditions that may not include reliable instrumentation and involve the burden of continuous operator monitoring and repeated actions.

In its letter dated October 5, 2012, NEI appears to acknowledge that significant challenges remain to be solved before such a single scenario-specific strategy could even be implemented in the field:

> Applying the findings of the EPRI study to individual plants will take significant effort and time. At a minimum, each plant (or class of plants) will have to perform a specific evaluation based on the EPRI methodology to determine the appropriate strategy to implement. This would require, prior to initiation of the study, alignment with NRC on the filtering strategy performance-basis, development of a regulatory vehicle, implementation guidance, design basis assumptions, severe hazard considerations, accident scenario requirements, etc. Experience suggests that this will involve numerous meetings among NRC staff, industry and other stakeholders over at least 24 months.

Additionally, the October 5 letter recognizes that operator actions and containment venting control remain concerns by the NRC staff:

> We understand the need to provide appropriate reliability to this operation whether it will be a self-actuating relief valve, an instrumented valve capable of operating during station blackout conditions, a manual valve or a combination. The actual duty cycle for this valve will be determined by plant specific analysis. While not downplaying the importance of the reliability of this operation and potential service conditions, the valve would not have to actuate repeatedly throughout the life of the plant.

This scheme also allows for more settling and plate-out of airborne radioactivity in containment and subsequently a more energetic discharge into the suppression pool or more dwell time for the spray header flow to scrub drywell atmosphere aerosols than would occur with continuous venting.

The modeling results indicate an effective overall containment decontamination factor of a 1,000 or more can be achieved by sequential opening and closing of the wetwell vent in order to maintain containment pressure between 60 pounds per square inch gauge (psig) and 40 psig. When the wetwell water level rises to where it prevents further wetwell vent use (approximately 18–20 hours from event start), any benefits of wetwell scrubbing is lost, a drywell vent path is needed and is subsequently cycled opened and closed for containment pressure control. Because suppression pool scrubbing is lost, radioactive releases are expected to be much greater.

In their presentation to the NRC staff, EPRI suggested that to accomplish the automatic vent cycling suggested in their report as being means to achieve the high DFs, the vent valves could be outfitted with a programmable controller to reduce the uncertainty of operator ability to maintain the venting strategy given other demands of the event on their time and attention. This scheme would also place continuous reliance on containment water level and pressure instrumentation as well as that of the vent status and valve actuators and power supplies to achieve the maximum possible reduction in airborne radioactivity released. The staff notes that the containment barrier has traditionally been recognized as a passive barrier with the exception of the need for an initial isolation of any open valves. The EPRI concept appears to potentially change the passive barrier concept, and result in the containment being an actively managed system.

EPRI stated that the drywell was modeled as a single node and no evaluation was made for thermal stratification and temperatures that could be experienced by penetration/seals located in the Mark I containment upper cylindrical section, the higher drywell spray ring header normally being just below the transition to the spherical portion of the drywell. The water injection would appear to provide little cooling effect above the spray ring header elevation and maintaining containment at or near 60 psig may not be prudent with potentially large quantities of light combustible gases being generated within containment and susceptible drywell penetrations potentially compromised by excessive temperatures. Gross leakage into the reactor building may be much larger with pressure being maintained near 60 psig if susceptible penetrations have been overheated rather than reducing pressure to lower values by continuous venting. The EPRI analyses were conducted for 72 hours. At the end of this time period, the containment pressures and bulk average temperatures are still significantly elevated at 60 psig and 300 degrees F. While probably an artifact of the analysis, the staff notes that success in mitigating severe accidents should not be dependent upon elevated containment pressures and temperatures for extended periods. A safe steady state end point should be identified that is not challenging barriers to the release of radioactive material.

In summary, the study's models focused on identifying actions that could be taken given a few plausible but specific severe accident event scenarios with existing equipment, or with modifications short of installing external vent filters, that could reduce airborne releases to levels approaching those reliably obtainable with the external filters. However, the conceptual strategy requires a high degree of confidence that current plant systems (i.e., suppression pools and sprays) can achieve a reliable DF under accident conditions. There is limited availability of testing data (if any) supporting the efficacy of sprays using FLEX flow rates within crowded BWR Mark I containments. Decontamination effectiveness highly depends upon containment conditions, and DFs of 1,000 are possible only if containment conditions are controllable and controlled. The industry acknowledges that further and significant developments, including plant-specific analyses, will be required over the next two or more years before it can be confirmed that the concept strategy is even feasible.

4.3.7 Low-efficiency Filters Can Further Reduce Radionuclide Releases

The EPRI report also mentions the possibility of installing a new design, low efficiency filter in order to further reduce radiological releases:

> The analyses conducted for this research indicate that several of the combined strategies could reduce radiological releases significantly, with DFs greater than 1000. These combined strategies could potentially be enhanced by adding a low efficiency filter to the vent path to provide additional fission product capture. However, the aerosol remaining after using the strategies would be composed of much smaller particles, and the efficiency of the removal of these very small particles has not been demonstrated with current filter designs. Additional research is needed to assess the efficacy of current filter designs when used in combination with the combined strategies to evaluate whether new filter designs significantly change radiological releases.

The report states that the removal of "very small particles has <u>not been demonstrated</u> with current filter designs" (emphasis added). The staff believes this effectively ignores the significant developments and advancements made by filter design engineers and manufacturers over the past 25 years to specifically capture these hard-to-remove particle sizes.

During the course of its investigations, the NRC staff has had the opportunity to discuss filter designs and decontamination effectiveness with filter manufacturers (AREVA, IMI Nuclear, and Westinghouse) as well as with representatives from foreign regulatory authorities in Sweden, Switzerland, and Canada. All parties recognize that submicron particles (penetrating particles) are hard to stop with sprays and simple water pools (e.g., suppression pool). As a result, filter design engineers and scientists have come up with innovate ways to specifically address and improve submicron particle capture. These innovations include improved venturi scrubbers, nozzle designs with impaction plates, methods to recirculate water within filters, and dry filter technology to enhance submicron particle removal. Manufacturers have cited thousands of tests performed by reputable testing agencies and laboratories (e.g., Paul Scherrer Institute, Battelle, and the U.S. Department of Energy National Laboratories (ACE testing)). Although the NRC staff has not performed a detailed review of test reports provided by the laboratories, foreign nuclear safety regulatory authorities have reviewed test results and have accepted decontamination factors of at least 1,000 (aerosols) for designs currently on the market. The ability of these external filters to capture and retain radioactive iodine is similarly recognized and impressive. Therefore, based on its review, the staff has reason to believe that the various engineered filter designs readily available today will provide a more effective, and at a minimum, a more reliable and predictable means of capturing all particle sizes, including submicron particles, than a wetwell with an unknown temperature and length of decontamination (bubble rise) path.

4.4 Other Concerns

In a letter dated July 24, 2012, the BWR Owners' Group (BWROG) submitted a request to the NRC to review and approve changes to the BWROG emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs) for venting operations during station blackout scenarios. These changes are referred to as the "early venting concept." Under the early venting concept, containment pressure would be kept below 25 psig. The NRC staff understands that early venting may be necessary in order to maintain RCIC injection flow cooling to the reactor as well as be necessary to support certain strategies under NEI's FLEX response strategy. In contrast, the EPRI strategy/concept requires that the containment pressure be kept between 40 and 60 psig in order to achieve proper hold up and decontamination factors. This strategy may be inconsistent and at odds with BWROG early venting concept (<25 psig). As such, the EPRI optimum decontamination strategy may not allow the implementation of venting strategies that are necessary to support certain FLEX strategies designed to maintain RCIC and provide alternate water supplies to cool the core and/or limit core damage. The emergency procedures are developed to guide the operator's response to the "symptoms" that the plant is showing in response to an accident, rather than requiring the operator to determine the actual accident underway. The notion of early venting for certain accidents would have to be evaluated in terms of consistency with the development and purpose of symptom-based procedures.

5.0 Passive Containment Vent Actuation Capability

Many of the Mark I containment plants in the U.S. have a rupture disk in the hardened vent line in series with normally closed valve(s). The burst pressures range from about one-half of containment design pressure up to the containment design pressure. Some have the capability of pressurizing between the valve(s) and rupture disk and enabling early venting to better support injection via low pressure, low capacity pumps. Opening the valves requires operator action and active function of the valves. Given the unpredictability of an event and its impact on licensee's performance, a passive activation feature may be appropriate to reduce uncertainty in successful venting when containment conditions are beyond design values. Even close physical proximity to vent valves for local opening and subsequent closing efforts may be extremely difficult or dangerous due to radiological, thermal, lighting, and sound conditions, or other access impairments due to the initiating event or to available capable personnel.

Mark I and II containments typically have maximum calculated design basis accident pressures several pounds per square inch below the containment design pressure. A rupture disk providing for design basis integrity with a burst pressure at or moderately above containment design pressure could support passive initiation of vent function. In addition, early venting may be appropriate to extend RCIC pump operation or ensure low pressure pump injection capability to maintain RPV water level above the fuel to avoid or arrest core damage in the RPV. Valve(s) in series with the rupture disk would normally be open, but capable of closure during or after the event. Early venting with this configuration would require closing a vent line valve, injecting nitrogen/air pressurizing the volume between the valve and rupture disk to the burst pressure. The valve would subsequently have to be opened to vent. This requires two strokes of the valve and availability and introduction of the gas to burst the rupture disk and the additional uncertainty of successful completion and personnel resources required. A simpler arrangement for both active and passive deployment involves having two branches, one passive with an exposed rupture disk and valve(s) for subsequent closure, the other with normally closed valves that could be opened for early venting. This arrangement also provides the feature of redundancy for the vent function in the case a closed valve cannot be opened. Having two valves in series provides for redundancy of containment function in case one of the valves cannot be closed. See Figure 8 for a simplified filtered containment vent system applied to a Mark I containment.

Venting from the drywell after reactor vessel breach would result in a much higher release of airborne radioactivity. This potential release could be greatly reduced by addition of an external vent filter. An external filter would also support justification of exposed rupture disk for fully passive vent actuation as the impact of inadvertent initiation would likely result in a minimal release. It could also support justification for a single containment isolation valve in series with the rupture disk

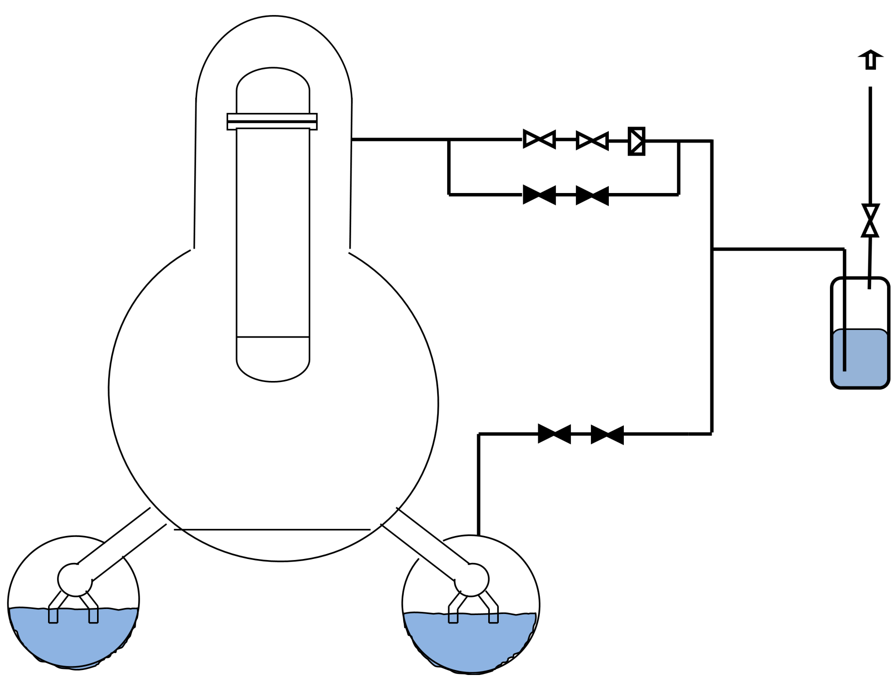


Figure 8 – Potential containment venting arrangement for BWR Mark I containments

6.0 Early Venting

As previously noted, in a letter dated July 24, 2012, the BWROG requested NRC staff review of their Emergency Procedure and Severe Accident Guidelines (EPGs/SAGs) changes recently approved by their Emergency Procedures Committee. The letter states that the primary objective of the changes to the guidance is the maintenance of adequate core cooling and prevention of core damage during extended station blackout conditions. Procedures would be changed to indicate that containment should be vented early, at pressures below the PCPL value, to reduce pressure as necessary to restore and maintain core cooling or reduce the potential total offsite radiation dose. This would be before significant core damage had occurred, in anticipation that containment pressure may well rise above the design or limiting pressure values and the ability to provide adequate low pressure injection for core cooling could become impaired. This guidance would allow for venting and releasing airborne radioactivity in excess of normal release limits in anticipation that the event may progress to a severe accident status with significant core damage and possibly much larger later releases if containment pressure reduction is not accomplished without further delay.

Early venting, similar to full passive activation with an exposed rupture disk, is more easily justified with an external filter that would likely limit early venting releases to the range of normal release limits.

Enclosure 5
Technical Analysis

Technical Analysis

To support the staff's assessment of the quantitative costs and benefits of severe accident capable vents (Option 2) and filtered containment venting (Option 3), the Office of Nuclear Regulatory Research (RES) performed an analysis of selected accident scenarios for a boiling-water reactor (BWR) plant with a Mark I containment. The analysis was conducted using the NRC's severe accident analysis code MELCOR, and its companion code, the MELCOR Accident Consequence Code System, Version 2 (MACCS2). RES staff was assisted by Sandia National Laboratories. The staff used the MELCOR code to calculate fission product release estimates for each of the selected accident scenarios, and this information was used to calculate health consequence and offsite property damage assessments using MACCS2. The staff used the results to inform the cost-benefit analyses of various accident prevention and mitigation options. The NRC's regulatory analysis guidelines in NUREG/BR-0058 and NUREG/BR-0184 recommend the use of MACCS2 to estimate the averted "offsite property damage" cost (benefit) and the offsite averted dose cost elements.

The selected scenarios are illustrative of potential accident sequences and serve as a means to provide comparisons of the quantifiable benefits for each of the proposed options. Selected cases are not meant to provide any insights into what the staff may believe is "the next accident" or even what it considers as bounding. Additionally, the staff notes that uncertainty always accompanies specific plant responses and timing during potential accident scenarios. Therefore, the most useful information stemming from this analysis are not individual results or consequences; rather, the "deltas" or comparisons between the selected cases.

The staff also performed a risk evaluation to estimate the reduction in risk resulting from the installation of a severe accident (SA) capable venting system in a BWR with either a Mark I or Mark II containment design. This information provides a major input to the regulatory and backfit analyses of the SA and filtered containment venting systems.

Finally, on September 25, 2012, the Electric Power Research Institute (EPRI) published a study relating to BWR Mark I and Mark II containment venting. The report titled, "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents - *BWR Mark I and Mark II Studies*," (EPRI Final Report 1026539), was made available to the NRC staff through EPRI's public web site (http://my.epri.com/portal/server.pt?Product_id=000000000001026539).

The purpose of the report was to document research on investigations into potential strategies for reducing the environmental and public health effect consequences of severe reactor accidents. The results of the report were also the subject of two public meetings. On August 8, 2012, the staff held a public meeting where representatives from EPRI provided an overview and preliminary results of the research efforts documented in the September 25 report. In addition, EPRI briefed the Advisory Committee on Reactor Safeguards (ACRS) Fukushima Subcommittee on September 5, 2012, and provided information relating to its preliminary evaluation of strategies for mitigating radiological releases during severe accidents at BWRs with Mark I and II containments.

The MELCOR analysis and results are generally consistent with the insights provided in the EPRI report with one notable exception. MELCOR calculations do not show vent cycling to be any more effective than once-open venting. The release estimates in both cases are on the same order of magnitude. The EPRI calculations concluded vent cycling to be more effective. Even if vent cycling is demonstrated to be effective, the feasibility of its operation needs to be carefully examined. Note the insights in the EPRI report recognize that an external filter can further reduce the fission product release to the environment – consistent with the conclusion from MELCOR/MACCS analysis.

In summary, the staff's technical analysis provided relevant insights into the merits of severe accident capable venting and filtered containment venting. The results were used to help quantify the benefits of various options under the NRC's current regulatory framework. As such, the reader is cautioned in drawing additional insights and conclusions from the results of the staff's analysis. The following sections describe the NRC staff's technical analysis:

- Enclosure 5a - MELCOR Accident Analysis
- Enclosure 5b - MACCS Analysis
- Enclosure 5c - Probabilistic Risk Evaluation

Enclosure 5a
MELCOR Accident Analysis

CONTENTS

1. INTRODUCTION

This enclosure documents MELCOR analysis of selected accident scenarios in a boiling-water reactor (BWR) plant with a Mark I containment in support of the staff's ongoing effort to address the Near-Term Task Force (NTTF) recommendation related to the containment venting [1]. Specifically, the work reported herein relates to the calculations of fission product release estimates using the U.S. Nuclear Regulatory Commission (NRC) severe accident analysis code MELCOR [2]. The release estimates are used to calculate health consequence and offsite property damage assessment using the MELCOR Accident Consequence Code System, Version 2, or MACCS2 [3], discussed in Enclosure 5b. The MELCOR/MACCS2 results, along with consideration of probabilistic risk assessment (PRA) as discussed in Enclosure 5c, are used in regulatory analyses of various accident prevention and mitigation strategies.

MELCOR has a long history of systematic development whereby each release version provides an update of code capabilities with regard to phenomenological modeling, code assessment, and other code improvements. The code has an extensive assessment database and is routinely benchmarked against other codes as well as experimental data. The code is also routinely subjected to rigorous quality assurance processes.

The selection of accident scenarios considered for MELCOR and MACCS analyses is informed by the recent state-of-the-art reactor consequence analysis or SOARCA [4] and also by the recent Fukushima study [5]. Specifically, two accident scenarios were selected for MELCOR/MACCS analyses as in the SOARCA Peach Bottom plant consequence analysis. These are: long-term station blackout (LTSBO) and short-term station blackout (STSBO) as defined in the SOARCA study, both initiated by a seismic event. The LTSBO results in a loss of offsite power (LOOP), failure of onsite power, and failure of the grid. All systems dependent on AC power are unavailable. The turbine-driven reactor core injection cooling (RCIC) system is available until battery depletion and, for the current study, it is assumed that the high-pressure coolant injection (HPCI) system is not available. For STSBO, it is further assumed that the RCIC is initially not available.

The primary focus is on the LTSBO scenario and a large number of MELCOR cases were run simulating different possible outcomes (e.g., containment failure by overpressurization, drywell liner melt-through, main steam line rupture). Consideration was given to various preventative and mitigative measures and how these influence the failure modes. Accident scenarios other than station blackout (SBO) were left out following the same considerations (i.e., core damage frequency cutoff, generic containment performance improvements to reduce the accident frequency or the severity of consequences, etc.) as in the SOARCA study. It is noted that the Electric Power Research Institute (EPRI), on behalf of the industry, has performed similar analysis in support of strategies for mitigating radiological releases from severe accidents at BWR Mark I and Mark II containments.

The MELCOR code calculations, described in considerable detail in the rest of this document, are deterministic in nature. The calculations produce point estimates of the quantities of interest (e.g., radionuclide release fractions). There are phenomenological uncertainties in the code and, as a result, the predicted point estimates also have some uncertainties. For the containment venting issue, the most pertinent uncertainties are related to core melt progression in a BWR in the presence of one or more mitigation measures, ex-vessel core debris behavior (e.g., molten core-concrete interaction, melt spreading), and fission product decontamination.

There are also modeling uncertainties in MACCS; in particular, those related to atmospheric transport of fission product aerosols. Given these uncertainties, the MELCOR deterministic safety analysis and MACCS consequence analysis are often supplemented by uncertainty analyses and sensitivity studies to provide a bounding estimate of the parameters of interest for regulatory analysis and decisionmaking.

Another source of uncertainty not discussed in the present report relates to that associated with the implementation of prevention or mitigation features used in the MELCOR analysis. It is assumed that in an SBO situation, such features or measures will be available. The report makes no statement, implied or otherwise, regarding the effectiveness and human reliability of operator actions in a severe accident situation; nor does it make any statement regarding equipment availability, operability, and system monitoring in a severe accident situation. These elements play a significant role in determining the feasibility and efficacy of any prevention and mitigation measures.

The report provides a discussion of the deterministic analysis of accident progression and its consequence given a core melt accident, and makes no assumption of the core damage frequency or the probability of a particular mode of failure (e.g., liner melt-through). The latter information is important for an estimation of risk and for regulatory analysis. It is provided in a separate enclosure.

Section 2 of this report provides a general description of the MELCOR code and focuses on the features of the code that are relevant for the containment venting analysis. Section 3 provides some general discussion of uncertainties in relation to MELCOR analysis of accident scenarios. Section 4 discusses the BWR MELCOR model used in the current study. As will be elaborated in this section later, the Peach Bottom SOARCA BWR model is used with a few modifications. Section 5 delineates the MELCOR calculation matrix comprising a large number of cases covering variations of LTSBO as well as various prevention and mitigation measures. This section also discusses the results of baseline MELCOR calculations and selected sensitivity cases highlighting the relative effects of various prevention and mitigation measures. Conclusions from MELCOR analysis are drawn in Section 6 of this report. Corresponding MACCS calculations and a discussion of results are provided in a separate enclosure.

2. GENERAL DESCRIPTION OF MELCOR

MELCOR is an integrated system-level computer code for modeling progression of severe accidents (i.e., accidents resulting in severe core damage, possibly melting of the core, leading to release of radioactivity) in nuclear power reactors. The scope of accident progression modeling includes:

- core uncovery (due to loss of coolant), fuel heatup, candling, clad ballooning, clad oxidation, fuel degradation (loss of geometry), and core material melting and relocation
- heatup of reactor vessel lower head from relocated core materials, subsequent failure of the lower head from thermal and mechanical loading, and release of molten core debris to the reactor cavity
- molten core-concrete interaction in the reactor cavity and ensuing aerosol generation
- in-vessel and ex-vessel hydrogen production, transport, and combustion
- fission product (aerosol and vapor) release from the core, and transport and deposition in the containment
- containment loading from high-pressure melt ejection, overpressurization from noncondensable gas generation including hydrogen, or other mechanisms (e.g., hydrogen burning, thermal attack of liner), and subsequent failure of the containment
- fission product release into the environment

MELCOR development was started in the 1980s by the NRC to provide an estimate of risk associated with a core melt accident in nuclear power plants. The initial thrust of code development was to have an analytical tool for adequate quantification of severe accident risks, yet a reasonably fast-running code that embodied, in a parametric manner, the then state of phenomenological knowledge on severe accidents.

In the years following the initial development of MELCOR, significant advances were made to the phenomenological understanding of severe accidents as a result of extensive research both in the experimental and in the analytical fronts. This together with the advent of faster and more powerful computing capabilities facilitated further development of MELCOR in primarily two areas—development of more mechanistic modeling of severe accident phenomena and numerical improvement for a faster running code. As a result of modeling improvements, MELCOR has become the repository of an improved understanding of severe accident phenomena, and a code of choice for confirmatory safety analysis of nuclear power plants. The code has a substantial worldwide community of users, and its use has been expanded to include both power and nonpower reactors, other nuclear systems (e.g., spent fuel pool, dry cask storage), and advanced reactor concepts, including non-light-water reactor designs. The code is routinely used as a confirmatory analysis tool to provide technical basis in support of a variety of regulatory applications, including power uprate, design-basis containment performance, risk-informing loss-of-coolant accident criteria, and review of new and advanced reactor designs.

Many MELCOR models are mechanistic; however, some are parametric, particularly those related to phenomena with large uncertainties where consensus is lacking concerning an acceptable mechanistic approach. Current use of MELCOR for deterministic safety analysis is often supplemented by uncertainty analyses and sensitivity studies. To facilitate this, many of the mechanistic models have been coded with optional adjustable parameters. These parameters can be varied one at a time as well as multivariate effects can be examined in a systematic manner. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient.

MELCOR has a modular architecture consisting of a number of “packages” that address different aspects of reactor accident analyses. The packages come in three categories: (1) basic physical phenomena (i.e., hydrodynamics, heat and mass transfer to structures, core degradation and relocation, core-structure and fuel-coolant interactions, gas combustion, and aerosol and vapor physics), (2) reactor design specific information (i.e., decay heat generation, sprays, and engineering safety systems, etc.), and (3) support functions (thermodynamics, equations of state, material properties, data-handling utilities, and equation solvers). The important phenomenological packages (first category) are listed in Table 1.

Table 1. Important Phenomenological Packages in MELCOR

Acronym	Package Name	Functional Description
BUR	Burn package	Models the combustion of gases in control volumes. The models consider the effects of burning on a global basis and are based on the deflagration models in the HECTR 1.5 code.
CAV	Cavity package	Models core-concrete interaction (an ex-vessel phenomenon) and melt spreading. The effects of heat transfer, concrete ablation, cavity shape change, and gas generation are included, using models taken from the CORCON-Mod3 code.
COR	Core package	Models thermal response of the core and lower plenum internal structures, including the portion of the lower head directly below the core. The package also models the relocation of core and lower plenum structural materials during melting, slumping, formation of molten pool and debris, failure of the reactor vessel, and ejection of debris into the reactor cavity.
CVH/FL	Control volume hydrodynamic and flow path	Models of the thermal-hydraulic behavior of water, vapor and gases in control volumes connected by flow paths, including evaporation and condensation phenomena.
FDI	Fuel dispersal package	Models both low-pressure molten fuel ejection and high-pressure molten fuel ejection from the reactor vessel, and the behavior of dispersed debris in containment (direct containment heating phenomenon).
HS	Heat structure package	Models heat conduction within an intact, solid structure and energy transfer across its boundary surfaces. The modeling capabilities of heat structures are general and can include pressure vessel internals and walls, containment structures and walls, fuel rods, steam generator tubes, piping walls, etc.

RN	Radionuclide package	Models the behavior of fission product aerosols and vapors released from fuel and debris, aerosol dynamics with vapor condensation and revaporization, deposition on structure surfaces, transport through flow paths, and removal by engineered safety features.

2.1 Radionuclide Package in MELCOR

The radionuclide (RN) package is of particular importance since the output of this package is used for dose calculations by MACCS. Within the RN package, the MELCOR code categorizes radionuclides and other pertinent materials into elemental classes that exhibit similar chemistry.

These elemental classes and their representative elements are shown in Table 2. The modeling and treatment of radionuclides in the RN package include:

- release of radionuclides from intact fuel and from core debris
- transport and deposition of radionuclide vapors and aerosols through the reactor coolant system
- behavior of radionuclides and radioactive aerosols in the reactor containment
- effects of engineered safety systems on the amount of radioactive material that can be released from the reactor containment

Table 2. Elemental Classes and Representative Radionuclides in the RN Package

Class #	Class Name	Representative	Member Elements
1	Noble Gases	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
2	Alkali Metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
3	Alkaline Earths	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
4	Halogens	I	F, Cl, Br, I, At
5	Chalcogens	Te	O, S, Se, Te, Po
6	Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
7	Early Transition Elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
8	Tetravalent	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
9	Trivalents	La	Al, Sc, Y, La, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf
10	Uranium	U	U
11	More Volatile Main Group	Cd	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi
12	Less Volatile Main Group	Sn	Ga, Ge, In, Sn, Ag
13	Boron	B	B, Si, P
14	Water	H2O	H2O
15	Concrete	- -	- -
16	Cesium Iodide	CsI	CsI

17	Cesium Molybdate	Cs_2MoO_4	Cs_2MoO_4
18	Non-Radioactive Tin	Sn	Sn

MELCOR considers radionuclide release from fuel both within the reactor vessel and when reactor fuel has been expelled from the reactor coolant system into the containment. Radionuclide release from fuel within the reactor vessel can be calculated using one of three closely related models: CORSOR, CORSOR-M, and CORSOR-Booth [6]. All three of these models have an empirical relationship derived from data on tests of fission product release from fuel heated usually out of pile. Diffusion coefficients in these models have been adjusted to match more recent tests such as those being done as part of the PHÉBUS-FP project [7].

Ex-vessel release of radionuclides is done with the VANESA model [8] developed based on experimental data explicitly for this purpose. The model considers fission product release by vaporization into bubbles of gas sparging through core debris attacking structural concrete. It also considers the formation of aerosols due to the bursting of bubbles at the surface of molten core debris. Radionuclide release can be retarded substantially by the presence of a water pool over the surface of the core debris. Modeling of this attenuation of the ex-vessel release is akin to that used in MELCOR to model decontamination of aerosol-laden gas flows through steam suppression pools. The suppression pool decontamination, including uncertainties, is discussed below in more detail.

Modeling of agglomeration and deposition of aerosol particles is done in MELCOR using the MAEROS model [9]. Deposition mechanisms considered in MAEROS are: gravitational settling, diffusion, thermophoresis, diffusiophoresis, and inertial impaction. The code also models vapor deposition by condensation as well as vapor chemisorption onto surfaces. Further, a model for hygroscopicity effects is also available in MELCOR. As with any phenomenological modeling, the models in MAEROS were validated with the then available but limited data. There are underlying uncertainties in these models that need to be assessed systematically with more recent data to determine their impact on the overall release estimates. This is, however, beyond the scope of the present study.

The MELCOR code considers the decontamination effects of the containment design and engineered safety features on fission products scrubbing. Specific features that are modeled include decontamination by: (1) suppression pool, (2) spray systems, and (3) filters. Details of the modeling are discussed in the following paragraphs.

2.1.1 Suppression Pool

Pool scrubbing is a relevant issue in nuclear safety since it provides a means to reduce source term to the environment during hypothetical severe accidents. Several severe accident scenarios involve the transport paths of fission product aerosols which include passages through stagnant pools of water where pool scrubbing can occur. Although the pressure suppression pool in BWRs is primarily designed to avoid overpressurization of the wetwell space, scrubbing in such pools has been given credit for mitigating the source term and hence the associated risk posed by accidents.

Several fundamental processes take place during aerosol pool scrubbing: diffusiophoresis, thermophoresis, inertial impaction, gravitational settling, centrifugal deposition, diffusion during bubbles rise, Brownian diffusion, etc. Aerosol characteristics (i.e., size, hygroscopicity, etc.) are the key factors for the effectiveness of these removal processes. Gas hydrodynamics plays an essential role determining key variables for pool scrubbing such as bubbles size and surface/volume ratio. In addition, other parameters like pool depth (injection point submergence), water subcooling, carrier gas composition and temperature and velocity, injection mode, water composition, etc., heavily influence individual pool scrubbing processes. In addition to the main aerosol removal processes, change in the particle size directly affects the pool scrubbing.

Decontamination by a steam suppression pool is done with the SPARC-90 model [10]. This model calculates removal of both aerosol and iodine gas from gases sparging through the suppression pool. Pool scrubbing or wet scrubbing is the removal of aerosol particles in gas bubbles rising in a water pool. The pool thus acts as a filter. Traditionally, the scrubbing efficiency has been expressed in terms of a decontamination factor (DF), which is defined by the ratio of the aerosol mass flow rate entering (m_{in}) and leaving (m_{out}) the pool. The path of aerosols along the pool height is usually split into three regions: injection (bubble formation) region, bubble rise region, and pool surface (bubble collapse) region. The overall DF is a multiplication of individual DFs of the three regions of the pool.

Past investigations have shown that decontamination by bubble formation and equilibration in a water pool can be significant, both in BWR's and PWR's risk relevant sequences. For shallow pools, the relative significance of the bubble formation and equilibration processes in determining the decontamination can be even larger than that by the decontamination process during the bubble rise through the pool height. Past investigations have also shown that the DF displays an inverted Gaussian type of trend as a function of particle diameter with a minimum at about 0.1 μm. Uncertainties in the particle size distribution at the inlet can largely influence DF estimates. Also, the DF increases smoothly and exponentially with submergence. Increased gas residence time through the pool efficiently raises the DF. WASH-1400, "The Reactor Safety Study," assumed a DF of 100 for subcooled pools and 1 for saturated pools.

2.1.2 Spray Systems

The drywells of most BWRs are equipped with water spray systems. These spray systems were installed to condense steam and reduce the pressure of the containment or drywell atmosphere in the event of a design-basis break in the reactor coolant system. Sprays are also very effective at removing aerosol particles from the containment or drywell atmospheres during severe reactor accidents. The spray systems consist of a large number of spray nozzles oriented differently near the top of the containment or drywell, and the header and spray nozzle configurations were designed for optimum spray pattern and droplet size with flows of several thousand gallons per minute to the drywell and several hundred gallons per minute to the suppression chamber. These nozzles discharge large numbers of water droplets that fall along ballistic trajectories through the atmosphere and sweep out aerosol particles.

Spray droplets remove aerosol particles from the containment or drywell atmospheres by several mechanisms:

- diffusiophoresis: steam condensing on the droplets and sweeping aerosol particles
- impaction: aerosol particles colliding with the droplet
- interception: aerosol particles adhering to the droplets
- diffusion: Brownian motion carrying aerosol particles in contact with falling droplets

The diffusiophoresis mechanism is only important early in an accident when the atmosphere is steam rich and aerosol concentrations are quite low. Consequently, this mechanism is not usually considered in the analysis of the steady state effectiveness of aerosol removal by sprays. The efficiency of aerosol removal by impaction, interception, and diffusion is expressed as the ratio of the number of particles actually removed from the atmosphere by a particular mechanism to the number of fixed particles that would be removed by a droplet along the same trajectory.

The removal efficiency is highly dependent on both the particle size and the effective droplet diameter. Diffusion is effective at the removal of very small aerosol particles (<0.1 μm). Impaction affects mostly aerosol particles larger than about 5 μm. Interception affects particles in the size range of 0.5 to 2 μm. Consequently, there is a minimum in the total aerosol removal efficiency when plotted against aerosol particle size. This minimum depends on the droplet diameter.

Reductions in the aerosol concentration by a factor of 10 can initially be achieved within 1 hour with full design spray flow. Further reduction in the aerosol concentration can be slower because the action of the spray alters the size distribution of the aerosol so that particles are less efficiently removed.

2.1.3 Filters

The requirements for the design of a filter system in removing the fission products depend on the thermal-hydraulic conditions (temperature, pressure, humidity, flow rate through the filter system) and concentration of the fission products in gaseous and aerosol form. Containment or auxiliary (reactor) building filtration systems are designed to avoid any substantial release of activity transported by aerosol particles and gaseous iodine. Of course, the main assumption here is that the containment is isolated and there are no uncontrolled leak paths.

As a result of the emerging new regulatory requirements for severe accidents, new filtration concepts were developed starting in the 1980s to backfit the current operating reactors in some countries. The main emphasis in the new regulatory requirements is to minimize potential land contamination by keeping the pressure in the containment under the design limits in order to avoid catastrophic containment failures and gross penetration leakage. This is done by venting through a containment filtered vent which should, at the same time, remove the aerosol particles and molecular gaseous iodine with certain efficiencies.

For most filtration devices, the efficiency of collection depends strongly on the particle size. For the purpose of MELCOR/MACCS analysis, the efficiency of filters is characterized by a specified DF. Further discussion of filter efficiency is provided in Section 5 of this report. For wetwell venting where the fission product aerosols are already scrubbed by the suppression pool, thus altering their size distribution, the DF range for filter is assumed to be relatively low. In the MACCS analysis reported in Enclosure 5b, the assumed range of DF is between 2 and 10. Some calculations were performed with a DF of 100. For drywell venting, if the feature is present in the design, a much higher DF (on the order of 1,000) may be attributed to the filter since the aerosols are not pre-scrubbed.

3. CONSIDERATION OF PHENOMENOLOGICAL UNCERTAINTIES IN MELCOR

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 as evidenced from its wide acceptability and its broad range of applications. Nevertheless, it is important to recognize the phenomenological uncertainties in MELCOR and their significance to MELCOR results. Moreover, it is important to understand the compounding effect of various uncertainties on the ultimate parameter of interest i.e., source term for all practical purposes. Some of the more important uncertainties are briefly discussed in this section.

The in-vessel melt progression modeling in MELCOR starting with the loss of intact core geometry to clad oxidation, in-vessel hydrogen generation, molten core relocation to lower plenum, and subsequent lower head failure are based on experiments which were conducted with the primary objective of gaining an understanding of these phenomena in relation to the observation and experience from plant accidents such as Three Mile Island. There are uncertainties associated with these phenomena. For example, the clad oxidation model in MELCOR is predicated on certain minimum thickness of pre-oxidized clad layer and certain minimum clad temperature. Any change in the values of these parameters may have an impact on the quantity of in-vessel hydrogen generation, melt temperature, and lower head failure timing.

MELCOR lacks a mechanistic model for evaluating fuel mechanical response to the effects of clad oxidation, material interactions (i.e., eutectic formation), zircaloy melting, fuel swelling, and other processes that occur at very high temperatures. The code uses a simple temperature-based criterion to define the threshold beyond which normal ("intact") fuel rod geometry can no longer be maintained, and the core materials at a particular location collapse into particulate debris. The temperature-based criterion attempts to bound uncertainties in phenomenological processes that affect fuel rod integrity.

The rate of movement of radial molten and solid debris to the center of the core and the time it takes the debris to move to the lower plenum are controlled by the relocation time constant parameter in MELCOR. This parameter is used as a surrogate for the broad uncertainty in the debris relocation rate into water in the lower head. This, in turn, affects the potential for debris coolability in the lower head (faster relocation rates decrease coolability; slower rates improve coolability). Debris relocation in MELCOR occurs when the lower core plate in a ring yields. Molten material and particulate debris from the ring immediately moves toward the center of the core and falls into the lower head. Thus, adjustments in this relocation time constant parameter affect the overall rate at which debris enters the lower head after support plate failure. For MELCOR calculations reported in this document, the relocation time constant value in the SOARCA study was used.

As in the case of in-vessel melt progression, the ex-vessel phenomenological modeling is based on experiments which were conducted to gain an understanding of melt spreading on the drywell floor, debris quenching in the presence of water, and molten core-concrete interaction, among others. The dominant mechanism of containment failure in accident sequences such as the LTSBO, is thermal failure (melting) of the drywell liner following contact with molten core debris (i.e., drywell liner melt-through). Containment failure by this mechanism occurs after debris is released from the reactor vessel lower head and flows out of the reactor pedestal onto the main drywell floor. The precise conditions under which core debris would flow out of the pedestal and across the drywell floor are uncertain. These uncertainties are currently captured in MELCOR in a parametric manner.

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

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

Several other sources of phenomenological uncertainties, not specifically discussed here, may be present in MELCOR. Moreover, there are uncertainties in modeling various mitigation features described previously (e.g., drywell spray effectiveness, suppression pool scrubbing decontamination factor, and external filter efficiency). Given these various sources of uncertainties, the MELCOR prediction of the source term can have a wide range and it is not uncommon to find an order of magnitude or more variation. A comprehensive MELCOR uncertainty analysis is being done for SOARCA Peach Bottom LTSBO.

4. MELCOR BWR INPUT MODELS

The BWR input models described here follow the “best practice” used in the SOARCA study 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 recent Fukushima study. The Peach Bottom SOARCA input deck was used as the baseline and a few modifications were made to the deck for the present containment venting study. These modifications are described later in appropriate subsections. The present study focuses on BWR Mark I containments. It is recognized that there are differences in design details between Mark I and Mark II containments so the results from this study may need to be appropriately qualified for Mark II containment types.

4.1 Reactor Pressure Vessel and Reactor Coolant System Models

Excluding the core region, the reactor pressure vessel is represented by 7 control volumes, 9 flow paths, and 24 heat structures. Nodalization for the core region between the core top guide and the bottom of active fuel are described later in the text. Figure 1 shows a representation of the MELCOR control volumes and flow paths for the reactor coolant system. Figure 2 provides a reactor vessel nodalization detail comparing MELCOR modeling features to actual vessel design. Control volumes are indicated by “CV” followed by the three-digit control volume number, and flow paths are indicated by “FL” followed by the three-digit flow path number.

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

- lower plenum (CV320)
- downcomer (CV310)
- shroud dome or upper plenum (CV345)
- steam separators (CV350)
- steam dryers (CV355)
- steam dome (CV360)
- jet pumps (CV300)

The downcomer control volume (CV310) represents the volume between the core barrel and reactor vessel wall (excluding jet pump volume) from the baffle plate to the top of the steam separators. The downcomer control volume includes all volume external to the steam separators in the region above the core shroud dome. The lower plenum control volume (CV320) includes all reactor vessel volume below the bottom of active fuel excluding the downcomer region and jet pumps. All volume internal to the 20 jet pumps is represented by CV300.

Reactor vessel upper internals are modeled in detail. Four control volumes, linked in series, are used to represent changes in the quality and temperature of core exit gases as they travel from the top of the core to the main steam line nozzles. The shroud dome control volume (CV345) represents the upper mixing plenum within the core shroud dome (from the top of the core top guide to the top of the shroud dome). The steam separators control volume (CV350) comprises the steam separator standpipes and the steam separators. The steam dryer region is represented by CV355 and includes all volume inside of the dryer skirt and the dryers from the top of the steam separators to the top of the steam dryers. Water stripped from steam in the separators and dryers is returned to the downcomer volume. The reactor vessel steam dome control volume (CV360) includes the dome region of the reactor vessel above the downcomer and steam dryer volumes.

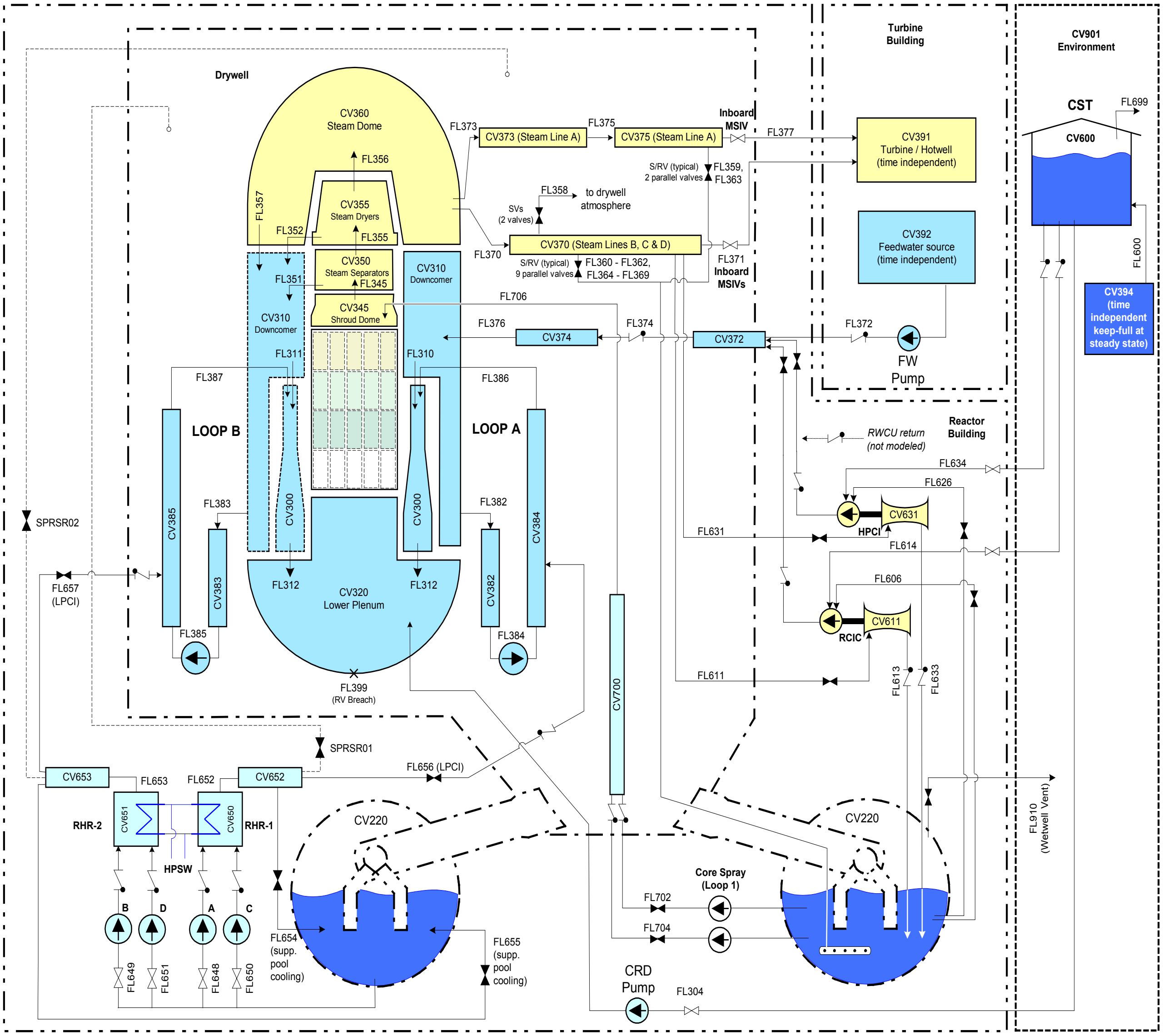


Figure 1. MELCOR control volumes and flow paths for the reactor coolant system

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

- the jet pumps and lower plenum (FL312)
- the shroud dome/upper plenum and steam separator standpipes (FL345)
- the steam separators and steam dryers (FL355)
- the steam dryers and steam dome (FL356)
- the steam dome and downcomer (FL357)
- Loop A suction flow from the downcomer to the jet pumps (FL310)

- Loop B suction flow from the downcomer to the jet pumps (FL311)
- the steam separators and downcomer (FL351)
- the steam dryers and downcomer (FL352)

The heat capacity and radionuclide deposition surface of a number of structures associated with the reactor pressure vessel are modeled via heat structures. The reactor pressure vessel itself is represented by four heat structures that include:

- the cylindrical portion in the lower downcomer region (HS31001)
- the cylindrical portion in the upper downcomer region (HS31011)
- the cylindrical portion adjacent to the steam dryers (HS36003)
- the hemispherical upper head (HS36002)

Cylindrical HS31001 is bounded by the downcomer (CV310) on the inside surface and the lower drywell (CV200) on the outside surface, and models the reactor vessel from the base of the downcomer to the elevation of the reactor building floor. Cylindrical HS31011 is bounded by the downcomer (CV310) on the inside surface and the mid-drywell (CV201) on the outside surface and models the reactor vessel from the elevation of the reactor building floor to the top of the steam separators. Cylindrical HS36003 is bounded by the steam dome (CV360) on the inside surface and the mid-drywell (CV201) on the outside surface and models the remaining cylindrical region of the reactor vessel from the top of the steam separators to the start of the hemispherical upper head. Hemispherical HS36002 is bounded by the steam dome (CV360) on the inside surface and the mid-drywell (CV201) on the outside surface, and models the hemispherical region of the upper head. The reactor vessel lower head is modeled within the core package and not included as a heat structure.

The core shroud is represented by 17 heat structures. Core shroud heat structures below the downcomer region represent the lower core shroud (HS32004) and the core shroud support (HS32003). Each of these structures is bounded by the lower plenum (CV320) on both surfaces. The upper shroud and dome are modeled by three heat structures. The first two structures represent the cylindrical region of the dome from the top of active fuel to the top of the core top guide (HS33017) and from the top of the core top guide to the hemispherical head (HS33018). The shroud dome head (HS34501) is represented by a horizontal rectangular heat structure. Both of these structures are bounded by the upper plenum (CV345) on the inner surface and the downcomer (CV310) on the outside surface.

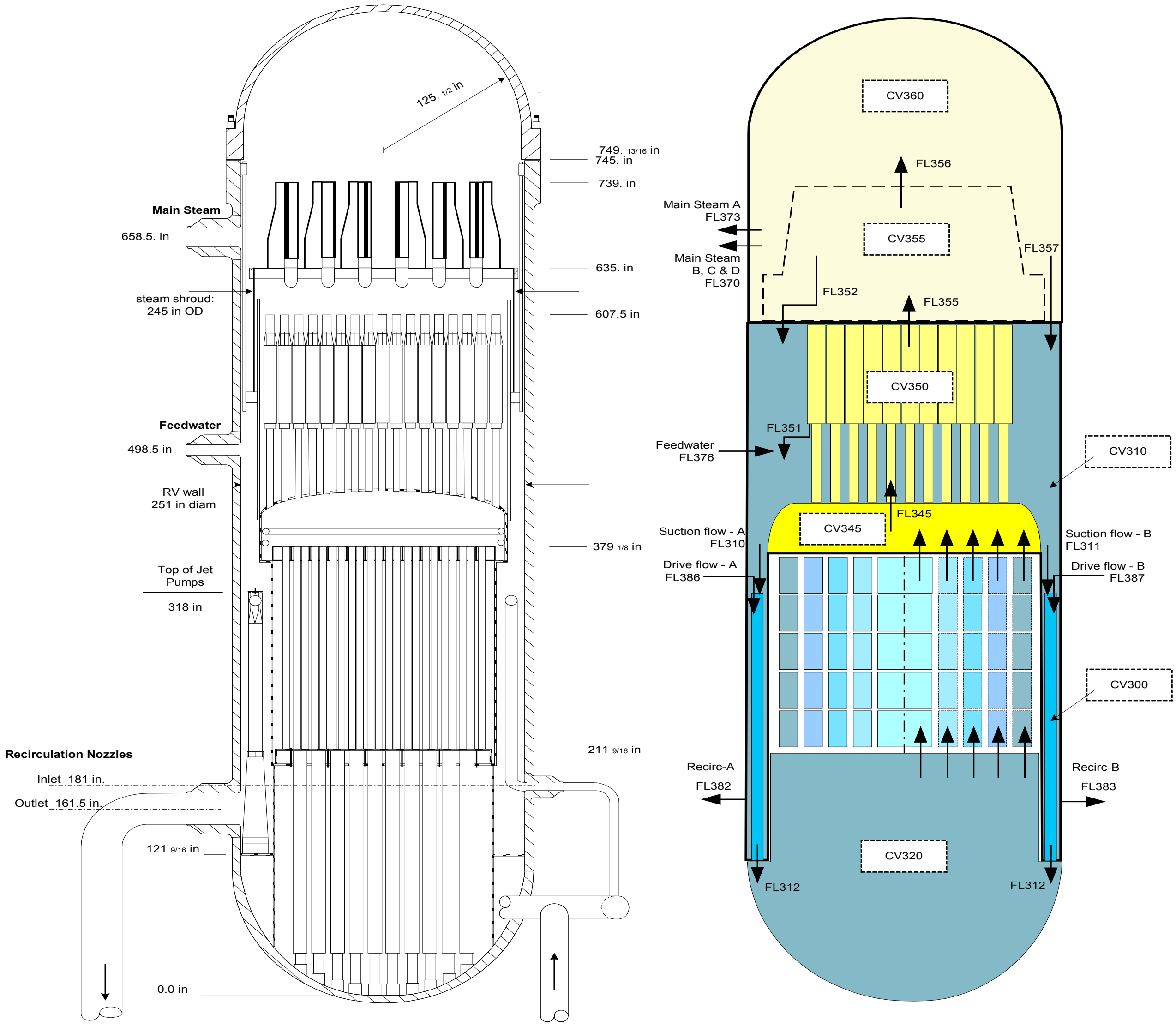


Figure 2. Reactor vessel nodalization detail

Three additional heat structures model other miscellaneous structures within the reactor vessel:

- the shroud baffle (HS31002)
- the standpipes and steam separators (HS35003)
- the steam dryers (HS36001)

The shroud baffle (HS31002) represents the boundary between the base of the downcomer (CV310) and the lower plenum (CV320). It is modeled by a rectangular heat structure (slab) with a surface area representing the base of the downcomer between the core shroud and reactor vessel (excluding the jet pumps). Vertical cylindrical heat structure 35003 represents the standpipes and steam separators and is bounded by the steam separators volume (CV350) on the inner surface and the downcomer (CV310) on the outer surface. Vertical rectangular heat structure 36001 represents the steam dryers and is bounded by the steam dryer volume (CV355) on one surface and the steam dome (CV360) on the other surface.

4.2 Core Model

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

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

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

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

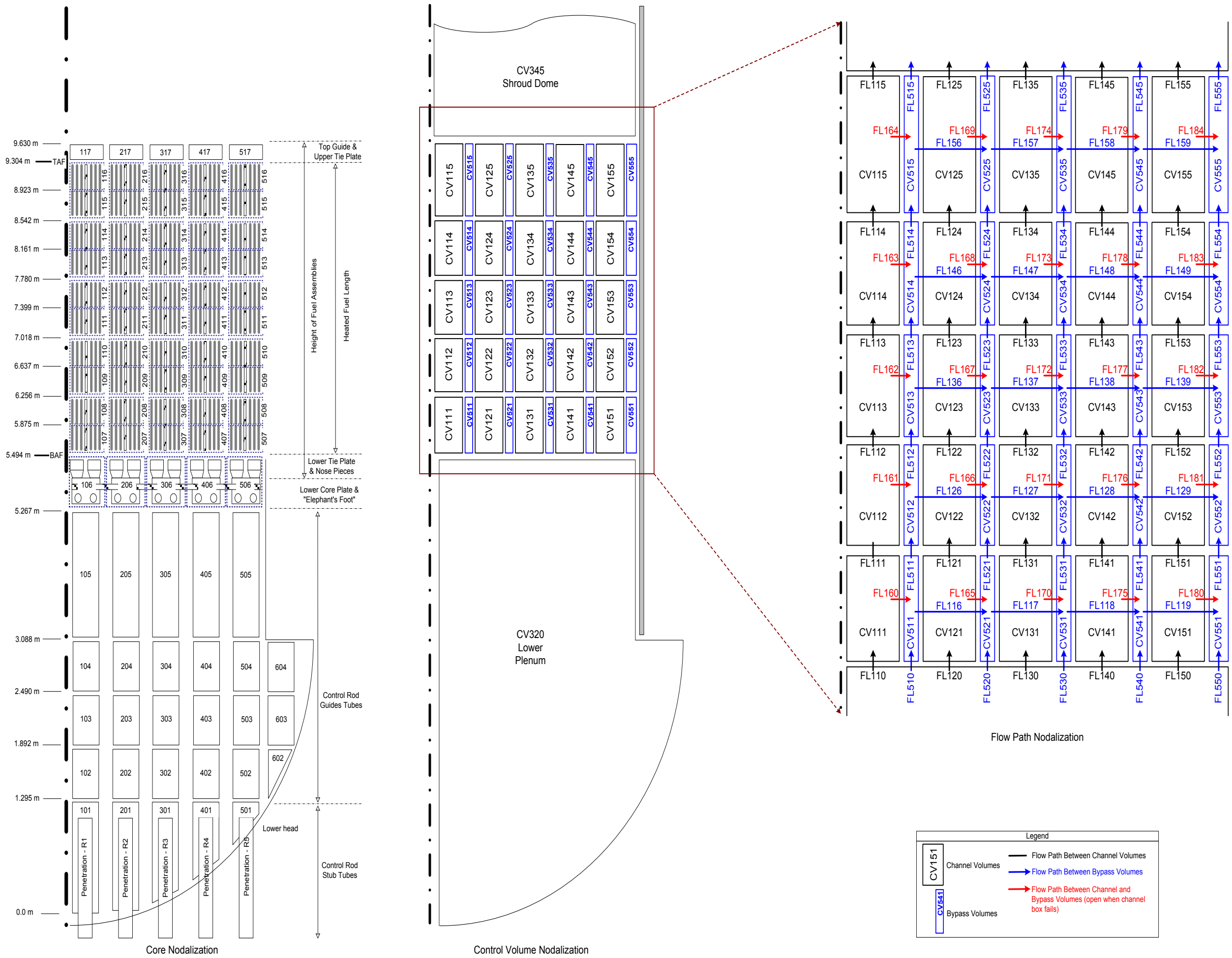


Figure 3. MELCOR core nodalization for the containment filtered venting study

All control blades are assumed to be inserted in the core region, regardless of the transient time (before or after SCRAM) or the type of transient (normal, ATWS). Axial levels 7 through 16 in rings 1 through 5 contain the control blades distributed as described in axial level 1. The combined stainless steel and B_4C mass is modeled as a nonsupporting structure in MELCOR with the surface area estimated from control blade dimensions.

Axial level 17 in rings 1 through 5 contains the core top guide and the fuel assembly upper tie plates. The total mass for the core top guide is distributed between the core rings based on the fraction of the area inside of the core shroud represented by the ring. Assembly upper tie plate mass depends on the type of fuel assemblies modeled, and is distributed based on the number of assemblies per ring. The combined mass of these structures is modeled as a nonsupporting steel structure.

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

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

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

The lower head is modeled as a hemisphere with an inner radius and thickness representative of a BWR Mark I plant. The lower head region extends to the downcomer baffle plate where it connects with the reactor pressure vessel. The hemispherical region of the lower head is represented by eight segments, and the cylindrical region of the lower head below the baffle plate by a single segment. A one-dimensional model of the stress and strain distribution in the lower head is applied. The temperature at which the yield stress in the lower head vanishes is set to 1,700°K to ensure creep-rupture of the lower head when it reaches the steel melting point. Heat transfer coefficients from particulate debris to the lower head and penetrations are modeled with a temperature-dependent control function which reflects conduction-based heat transfer through a frozen crust at temperatures of 2650 K and below, a conduction enhanced heat transfer coefficient as the debris reaches the eutectic melting temperature of UO_2 and ZrO_2, and a convective heat transfer coefficient as the debris exceeds the eutectic melting temperature and forms a circulating molten pool.

A single lower head penetration is modeled within each of the five inner most radial rings. This penetration models the heat capacity, surface area, and axial conduction area of a single control rod stub tube (excluding the drive shaft). By default, the penetration failure model is deactivated and the lower head failures occur due to creep rupture.

4.3 Residual Heat Removal System Models

Major modes of the residual heat removal (RHR) system are included in the MELCOR model. These include low-pressure coolant injection (LPCI), drywell sprays, and suppression pool cooling. Each train of RHR is modeled separately to allow for the possibility that under certain circumstances, one train might be aligned for operation in a different mode. The model for each train includes options for operating one or two trains of pumps and heat exchangers. RHR heat exchangers operate in all modes of operation whenever high-pressure service water (HPSW) is available. RHR pumps trip under the following conditions:

- loss of ac power
- suppression pool temperatures exceed pump NPSH limits
- suppression pool level below pump suction vortex limits
- pump failure flags in the sequence trip file

RHR operation in LPCI mode draws water from the suppression pool and delivers it to the reactor vessel via the recirculation loop discharge lines upstream of discharge valves on each RHR side. The LPCI model allows for automatic or manual initiation of the system. Automatic initiation of LPCI occurs upon receipt of a reactor vessel Low-level 1 signal. LPCI is terminated when the RCIC shutdown criterion is reached (operators are assumed to shut down LPCI when this criterion is reached). Suppression pool cooling mode draws water from the suppression pool, delivers it through the RHR pumps and heat exchangers for cooling, and returns it to the suppression pool.

Drywell sprays are modeled separately from the LPCI and shutdown cooling modes of operation using the MELCOR Containment Sprays package. The suppression pool is modeled as the source control volume with the mid-drywell (CV201) modeled as the location of the spray header for RHR train I and the lower-drywell (CV200) modeled as the location of the spray header for RHR train II. Drywell spray temperatures are calculated based on suppression pool temperatures and RHR heat exchanger operation. The drywell sprays mode of RHR allows for automatic or manual initiation of the system. Manual operation of drywell sprays may be specified through single initiation and termination times. Automatic initiation of drywell sprays (assuming operator actions to follow emergency procedures) is determined based on generic spray actuation limits provided in the BWROG Emergency Procedure Guidelines:

- drywell atmosphere temperatures exceed 350^{o}F
- drywell pressures below 3.0 psig @ 0^{o}F, 3.0 psig @ 100^{o}F and 7.2 psig @ 350^{o}F

4.4 Emergency Core Cooling Systems Models

Three emergency core cooling systems (ECCS) models are included in MELCOR. They include the reactor core isolation cooling (RCIC) system, the high-pressure coolant injection (HPCI) system and the low-pressure core spray (LPCS) system.

Operation of the turbine-driven RCIC system is modeled in detail. Nodalization for the RCIC system includes:

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

The model includes a constant-flow pump, delivering 600 gpm via velocity-specified flow paths, with suction initially aligned to the CST. Switchover of pump suction to the suppression pool occurs upon receipt of a low CST water level signal. Within the MELCOR model, CRDHS suction is modeled at an elevation common to the RCIC/HPCI suction header and also accesses this dedicated volume. The RCIC system nodalization does not include heat structures.

Steam flow through the RCIC turbine is modeled to account for the transfer of energy from the steam line to the suppression pool during RCIC operation. The flow of steam from main steam line to the RCIC turbine is modeled as a function of the pressure difference between the main steam line and the suppression pool. RCIC is modeled with automatic initiation and termination criterion. RCIC is initiated on receipt of a reactor vessel low level-2 signal. RCIC is terminated on receipt of a reactor vessel high level-8 signal. User input (CF937) may also be selected to model manual pump operation where operators throttle the RCIC turbine/pump to maintain water levels after automatic initiation.

Upon receipt of a RCIC actuation signal, the RCIC pumps reach full flow after a 30 second delay and 1 second ramp up in flow. The duration of dc power (station batteries) is specified by CF901 in the sequence trip input file. When the pump is manually operated, user input may also be selected so that RCIC turbine/pump operation continues at its current speed when station batteries are depleted (CF933).

The LPCS system in the plant consists of two loops, each with two pumps, which draw suction from the suppression pool and deliver flow to the reactor vessel via a spray header just above the core. One loop (i.e., 2 pumps) of the low-pressure core spray system is modeled, with pump suction aligned to the suppression pool. Nodalization for the LPCS system includes:

LPCS discharge piping from the LPCS pumps to spray header (CV700)
flow from the suppression pool to LPCS piping including LPCS Pump A (FL702)
flow from the suppression pool to LPCS piping including LPCS Pump C (FL704)
flow from LPCS piping to the spray header in the shroud dome (FL706)

LPCS operation is modeled with two modes of operation. LPCS delivery to the reactor vessel requires a low reactor vessel pressure permissive of 400 psig. LPCS pumps start immediately upon receipt of an actuation signal. In mode one, LPCS is terminated when the RCIC shutdown criterion is reached (operators are assumed to shut down LPCS when this criterion is reached). In mode two, the operators are assumed to throttle pump speed to maintain level just above the top of the core.

4.5 Containment Model

The primary containment is subdivided into six distinct control volumes. The drywell is represented by four control volumes:

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

One control volume represents the vent pipes and downcomers connecting the drywell to the wetwell (CV210), and one control volume represents all remaining volume within the wetwell (CV220). The MELCOR nodalization of the primary containment is shown in Figure 4.

A total of 17 flow paths represent intact containment flow pathways. Of these, two flow paths (FL200 and FL202) connect the three drywell regions external to the reactor pedestal. Each of these flow paths is modeled with 50 percent of the interfacing flow area between the control volumes. This assumes a 50 percent obstruction by equipment and structures of the interface between the drywell regions.

Three flow paths (FL014, FL015, and FL016) connect the reactor pedestal to the lower drywell. The open fraction of the personnel doorway is reduced based on the core debris elevation in the reactor pedestal after vessel failure (debris elevation determined from CAV package). Two additional flow paths (FL012 for flow from the drywell to the vent pipes and FL017 for nominal drywell leakage from the lower drywell to the reactor building) represent flow from the drywell. Flow path 012 includes flow from the drywell into all eight vent pipes in the drywell. The nominal drywell leakage flow area, friction, and form losses are defined to match the nominal drywell leak rate. The elevation of nominal drywell leakage is modeled at the dominant location of drywell penetrations.

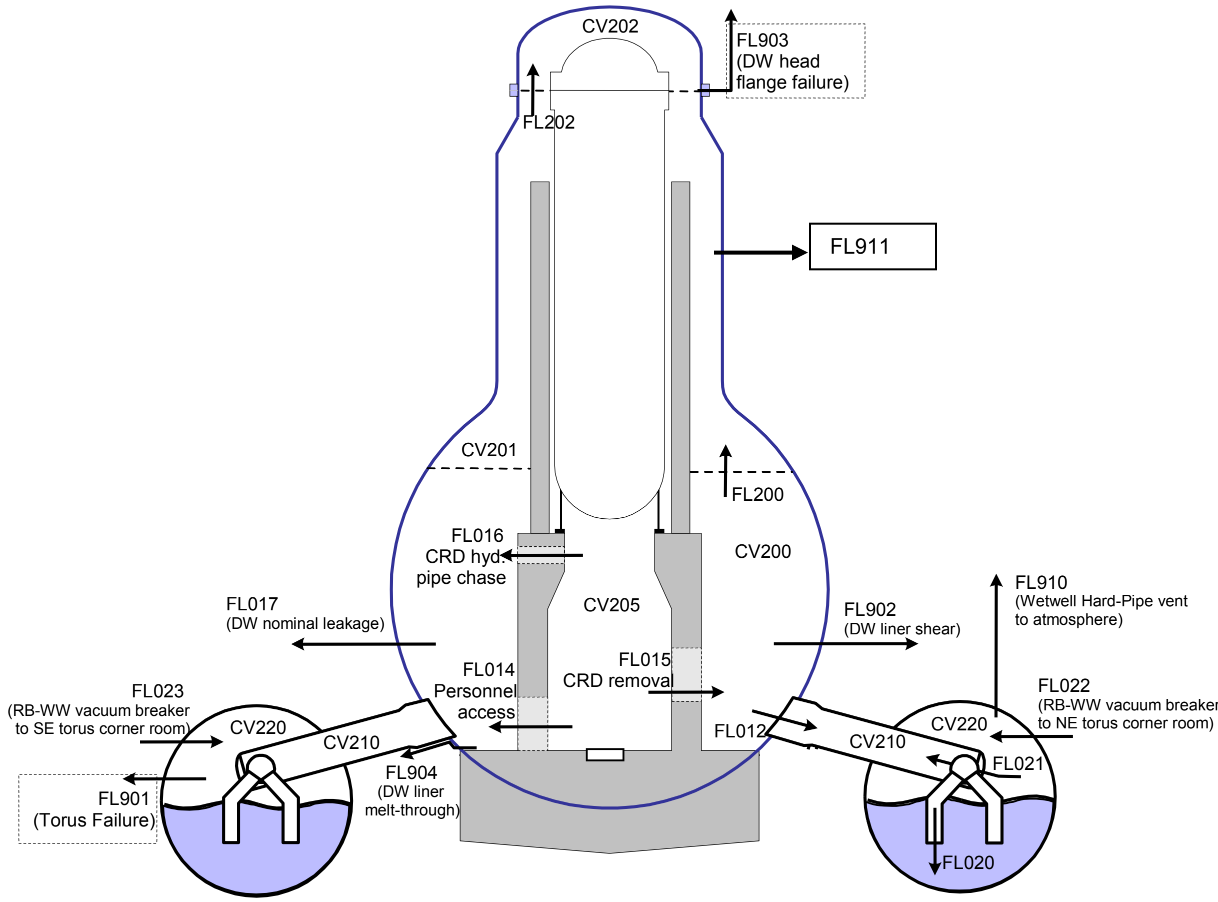


Figure 4. MELCOR nodalization of the primary containment

A single flow path represents flow from the downcomer pipes to the wetwell (FL020). The exit of this flow path has the SPARC pool fission product scrubbing model activated within MELCOR for aerosols and vapors across all fission product classes.

Three flow paths (FL021, FL022, and FL023) model vacuum breakers intended to limit under-pressure failures of the drywell and wetwell. The wetwell-drywell vacuum breakers open whenever the wetwell pressure exceeds the vent pipe pressure by 0.5 psid. The reactor building-wetwell vacuum breakers connect the wetwell airspace with the northeast and southeast torus corner rooms, and open whenever the pressure in the wetwell drops 2 psi below the pressure in the reactor building.

One additional flow path is modeled to represent manual wetwell venting (FL910). Based on user input, a hard-pipe vent line in the wetwell atmosphere may be actuated when containment pressure exceeds 60 psig. This line vents to the environment at an elevation equal to the top of the reactor building. Drywell venting was not modeled in the SOARCA study; however, in the current study two cases of drywell venting were considered. For this, an additional flow path (FL911) was added in the control volume CV201.

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

The SOARCA wetwell model used a single control volume to represent the hydrodynamic volume in the torus, one downcomer flow path from the lumped vent volume to the wetwell, and one vacuum breaker flow path from the torus airspace back up to the lumped vent volume. While this may have been sufficient to capture the containment pressurization rate for the accident scenarios defined in SOARCA, this same nodalization was found to underpredict the containment pressure for an accident scenario with extended safety relief valve (SRV) cycling and RCIC/HPCI operation as in Unit 3 reactor at the Fukushima Dai-ichi plant where RCIC and SRV discharged steam to the suppression pool for over 20 straight hours.

A refined wetwell model was used for the first set of 15 MELCOR runs for the containment venting issue. The refined model discretizes the torus into 16 equally sized control volumes the sum of which is equal to the original hydrodynamic volume in the PB SOARCA model so the total pool volume is preserved. There is only one volume in the axial direction, and there is still the single lumped vent volume from the SOARCA model. There are 16 interior flow paths connecting each circumferential volume to allow thermal-hydraulic communication between the wetwell volumes. Segmenting the wetwell into smaller circumferential volumes does not treat thermal stratification; nor does it treat wetwell mixing, but it is intended to provide a first-order prediction of asymmetric wetwell heating due to SRV and turbine exhaust.

While the refined model resulted in some improvement in containment pressure prediction, it also added significantly larger computation time for some scenarios and in a few cases, numerical convergence became an issue. For subsequent MELCOR runs, a 2-volume wetwell representation was used but only after checking that the 2-volume representation provided results which are reasonably close to those obtained with a 16-volume representation. Containment structures in the containment model are represented by 23 MELCOR heat structures. Eleven of these heat structures represent the drywell liner-air gap-concrete wall that makes up the boundary between primary and secondary containment (HS10010-HS10020). One drywell liner heat structure is modeled for each reactor building control volume in the reactor building. These rectangular heat structures are made up of carbon steel to represent the drywell liner, an air gap, and a concrete wall. The drywell liner interacts with the drywell control volume, and the concrete wall surface communicates with the appropriate reactor building control volume. The height of these heat structures matches the reactor building control volume in which it resides. Heat structure surface area is calculated so that drywell liner mass is appropriately modeled. Two additional drywell liner heat structures represent the cylindrical (HS10021) and dome (HS10022) portion of the drywell liner within the drywell enclosure. The heat structure film-tracking model is activated to connect film flows between the appropriate drywell liner heat structures.

Eight heat structures are modeled to represent the remaining drywell structures:

- the drywell floor outside of the reactor pedestal (HS10001)
- the drywell floor inside of the reactor pedestal (HS10002)
- the biological shield wall in the lower drywell (HS10003)

- the biological shield wall in the mid-drywell (HS10007)
- the reactor pedestal (HS10004)
- miscellaneous drywell steel in the lower drywell (HS10005)
- miscellaneous drywell steel in the mid-drywell (HS10008)
- miscellaneous horizontal deposition surfaces in the lower drywell (HS10006)

The heat structures representing the drywell floor are modeled as horizontal rectangular heat structures with an insulated boundary condition on one side and the drywell or drywell pedestal region as the other boundary condition. The biological shield wall is split between the lower and mid-drywell volumes as two vertical cylinders that communicate with the drywell at both boundaries. The bottom of the biological shield wall meets the top of the reactor pedestal. The reactor pedestal is represented by a thick vertical cylinder. Drywell miscellaneous steel structures represent equipment within these regions and are modeled by vertical rectangular heat structures.

Miscellaneous horizontal deposition surfaces within the drywell are modeled as upward facing rectangular heat structures with negligible heat capacity (relative to drywell atmosphere) and high thermal conductivity to track drywell temperatures. These heat structures are intended to represent all upward facing fission product deposition surfaces within the drywell (e.g., equipment, cable trays, piping) except for the floor.

Two additional heat structures model the wetwell liner (HS20001) and miscellaneous steel (HS20002). The wetwell liner is modeled as a thick horizontal cylindrical heat structure with a length representing the major torus diameter, and the wetwell (CV220) and main torus room (CV401) as surface boundary conditions. Wetwell miscellaneous steel represents equipment and structures within the wetwell and is modeled by a vertical rectangular heat structure.

Critical pool fractions for heat transfer to the pool and atmosphere are both set at 0.5 for all heat structure surfaces inside primary containment. This allows heat transfer between the heat structure and either the pool or atmosphere, but not both simultaneously. The transition from heat transfer with the pool to heat transfer with the atmosphere occurs when the fraction of the heat structure that is submerged in the pool drops below 0.5. Radiation heat transfer is not modeled for structures within primary containment.

4.6 Reactor Cavity Model

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

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

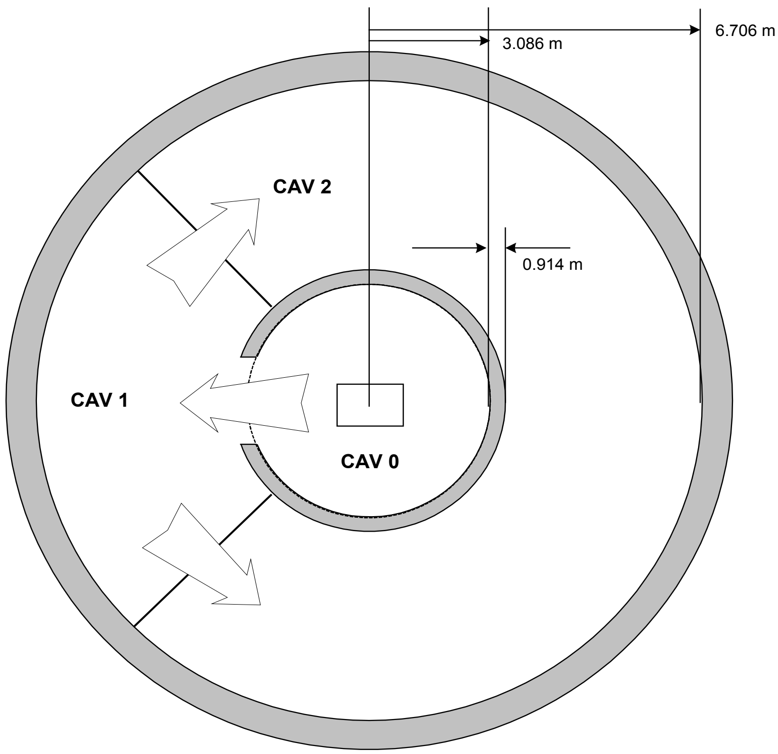


Figure 5. Discrete representation of debris spreading in the cavity

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

The solidus and liquidus temperatures in the parametric model that governs the rate of debris spreading on the drywell floor were modified in the present study. Original values of solidus and liquidus temperatures in the PB SOARCA model were 1,420^{o}K and 1,670^{o}K, respectively. These temperatures are representative of concrete solidus and liquidus. For containment venting calculations, the solidus and liquidus temperatures were changed to 1,700^{o}K and 2,800^{o}K, respectively. The revised liquidus temperature is representative of the liquidus temperature of a eutectic UO_2/ZrO_2 mixture. The revised solidus temperature was set at 1,700^{o}K to represent the lower bound of average melt temperature at vessel breach, and happens to coincide approximately with the melting point of steel. In the model, spreading is disallowed at debris temperatures less than the solidus temperature and occurs at a maximum rate (0.259 m/min) when debris temperature is above the liquidus temperature. Spreading rate varies linearly at temperatures intermediate between the solidus and liquidus temperatures.

4.7 Balance of Plant Models

A total of 41 control volumes, 71 flow paths, and 85 heat structures are modeled to represent all pertinent structures external to primary containment. These model elements represent the reactor building, turbine building, radwaste building, and the environment. Given its importance as a fission product release pathway, the reactor building is modeled in significant detail (30 control volumes and 80 heat structures). The turbine and radwaste buildings are considered to have a second order impact on fission product releases to the environment due to the large scale of these buildings and the limited pathways for fission products to enter them. Based on these considerations, the turbine and radwaste buildings are each modeled as single control volumes with one heat structure representing the floor (with the building cross-sectional area) and one nominal leakage flow path. In addition, a single heat structure with surface area equivalent to floor area models horizontal deposition surfaces within the turbine building. The other control volumes external to primary containment represent the reactor building ventilation system (a time-independent control volume fixing reactor building pressure during steady-state conditions), the condensate storage tank (CST), the equipment access lock connected to the reactor building, and the environment.

The reactor building is represented by 30 control volumes. A sectional view of the reactor building is shown in Figure 6. It is modeled on a level-by-level basis, beginning in the basement (i.e., torus room) and sequentially rising up through the main floors to the refueling bay. Control volumes are defined for each region of the reactor building where a volume is deemed to be large in comparison to its flow connectivity areas to other regions of the building. In addition, a finer control volume nodalization is implemented when flow resistances from one building level to another might impact fission product transport (such as in stairwell volumes). When determining the free volume available within control volumes where data on equipment and interior wall displacement are unavailable, it is assumed that 25 percent of the volume calculated based on room dimensions is displaced by these items. For stairwell volumes, it is assumed that 10 percent of the calculated volume is displaced by equipment or walls. For all other reactor building volumes, equipment and miscellaneous displaced volume is either calculated from data (Main Torus Room, Steam Tunnel) or neglected.

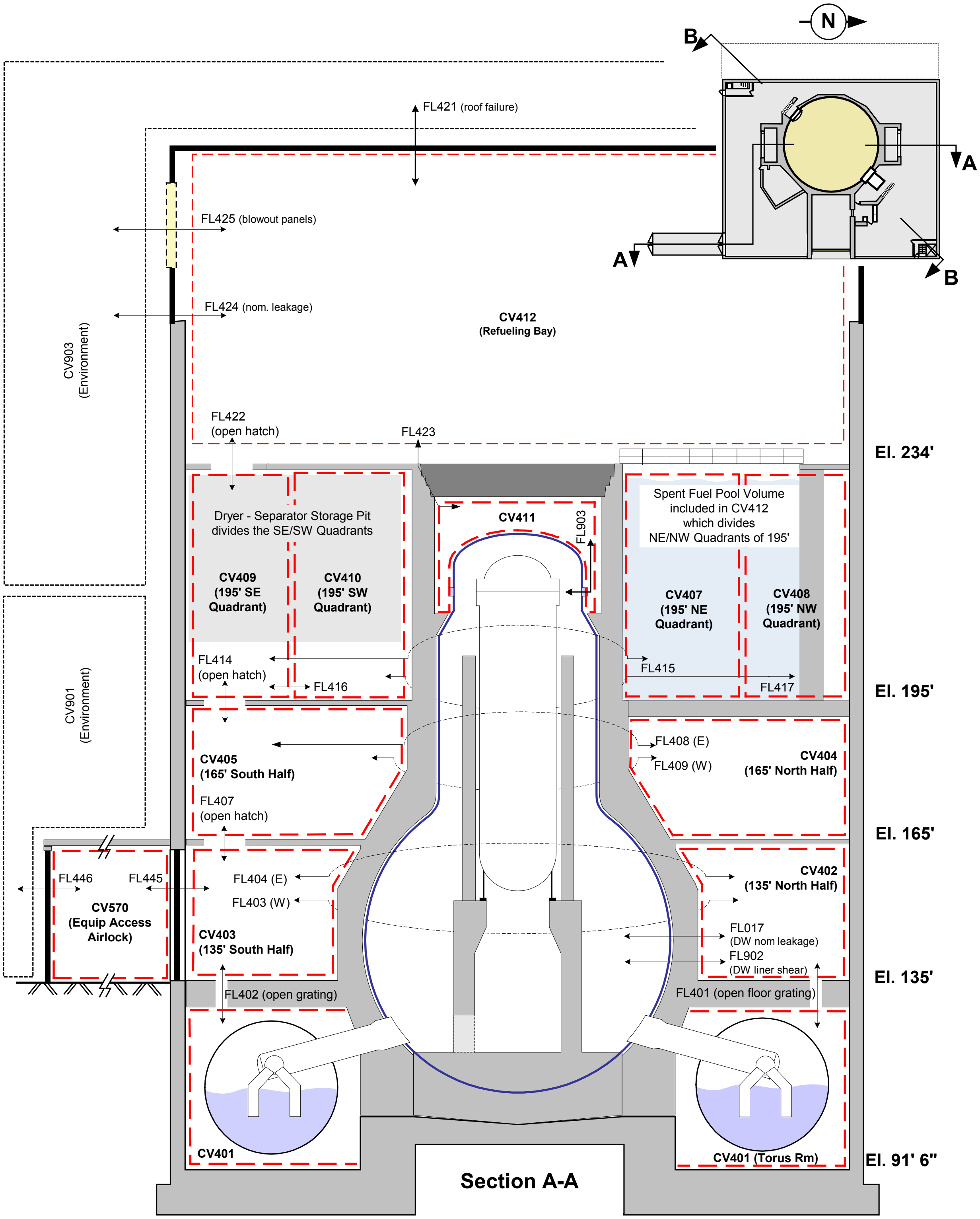


Figure 6. MELCOR nodalization of a sectional view of the reactor building

The torus room level of the reactor building is represented by eight control volumes. These include volumes representing the main torus room, the northeast corner room, the stairwell in the northeast corner of the building, the southeast corner room, and the RHR A, B, C, and D heat exchanger and pump rooms. The next higher level of the reactor building is modeled by five control volumes. These include volumes representing the southern half of the building, the northern half of the building, the southwest stairwell enclosure, the northeast stairwell enclosure, and the steam tunnel. The next higher level of the reactor building is represented by five control volumes. The next higher level of the reactor building is represented by eight control volumes. These volumes represent the northeast, southeast, northwest, and southwest quarters of the floor; the reactor building ventilation room; the drywell enclosure; the southwest stairwell enclosure; and the northeast stairwell enclosure. The refueling bay level (highest level) of the reactor building is represented by four control volumes. These volumes represent the open refueling bay (including the spent fuel pool but neglecting the separator/dryer storage pit), the southwest stairwell enclosure, the northeast corner room and the northeast stairwell enclosure. The 68 flow paths modeled within the reactor building can be classified into the following categories: same level flows between distinct control volumes, open hatches, doors, blowout panels, flow pathways through walls, leakage pathways, stairwells and concrete hatches. Same level flow paths are modeled to connect the distinct control volumes on each floor level (FL403, FL404, FL408, FL409, FL410, FL415, FL416, and FL417). These flow paths are modeled as horizontal. Open hatches connect each of the reactor building levels. Grated hatches exist between the main torus room and both the north and south control volumes (FL401 and FL402). An open hatch pathway exists in the southeast corner of the building to the refueling bay (FL407, FL414, and FL422). Flow paths representing each of these open hatches are modeled as vertical flows with the area of the open hatch.

A total of 25 flow paths are modeled representing doors within the reactor building. Both double and single door characteristics are modeled. Each flow path representing a single-style door is modeled as a horizontal flow. Each door has a combination of a valve and control functions to model door failure based on building overpressures. Each double door is assumed to be leaktight under nominal conditions, but has a combination of a valve and control functions to model door failure based on building overpressures.

Three flow paths are modeled representing blowout panels within the reactor building. Each of these is represented as a horizontal flow path with a valve and control function logic managing open fraction. Ten flow paths are modeled to represent open pipe chases and fire dampers through walls or floors (FL430-FL435, FL437, FL443-FL444, and FL451). Two flow paths are modeled to represent leak-type pathways. FL424 is a horizontal pathway representing nominal leakage through the refueling bay walls and ceiling. FL423 is a vertical pathway representing the leakage from the drywell enclosure through the concrete plug gap to the refueling bay. Seven flow paths represent vertical flows through the southwest (FL482, FL484, and FL486) and northeast stairwells (FL462, FL465, FL470, and FL474) in the reactor building. Six vertical flow paths represent concrete hatches that may be displaced by building overpressures. One additional reactor building flow path (FL450) connects the reactor building ventilation system (CV450) with the northern half of the reactor building. Nominal reactor building leakage occurs through the refueling bay walls/ceiling and closed doorways connecting the reactor building to the environment, turbine building, and radwaste building.

Structures within the reactor building are represented by 83 MELCOR heat structures. These heat structures can be classified in one of the following categories: floor/ceiling, exterior walls, interior walls, horizontal fission product deposition surfaces, or miscellaneous steel. Each reactor building control volume representing part of the primary room at each building level (CV401-410, 412) is modeled with heat structures representing the room's floor, ceiling, exterior walls, and miscellaneous steel. Each of these control volumes, excluding the refueling bay, also contains a heat structure representing internal walls. Each room floor is modeled as a horizontal slab with a surface area equal to the projected area of the room. For floors between two building levels, a two-sided heat structure represents the floor for the upper volume and the ceiling for the lower volume.

Exterior concrete walls are modeled as vertical slabs with an adiabatic boundary condition on the outside surface (due to interfaces with multiple external volumes and assumption that concrete wall thickness allows adiabatic assumption). Miscellaneous steel and internal walls are both modeled as rectangular-vertical heat structures. Internal walls within these volumes represent spent fuel pool walls, the separator/dryer storage pit walls, or miscellaneous structures. Miscellaneous steel represents equipment located within each volume. Miscellaneous internal wall structures and steel are modeled with model legacy values.

Horizontal fission product deposition surfaces within the reactor building are modeled as upward facing rectangular heat structures with negligible heat capacity (relative to drywell atmosphere) and high thermal conductivity to track drywell temperatures. This heat structure is intended to represent all upward facing fission product deposition surfaces (e.g., equipment, cable trays, piping) located in a particular region of the building (with the exception of the floor). Since fission product releases may occur at higher elevations in the reactor building (via drywell liner penetration shear, interfacing-systems loss-of-coolant-accident breaks), horizontal fission product deposition surfaces are modeled at these reactor building levels. Additional horizontal deposition surface area within these regions was estimated as projected floor area.

Rooms modeled within the reactor building that are accessible only via doorways are considered of secondary importance to fission product distribution. For these control volumes (CV452-458, 460), modeling of heat structures is limited to a slab representing projected floor area and steel representing grated floors in the RHR heat exchanger and pump rooms. Stairwell control volumes are only accessible via initially closed doorways, and heat structure modeling for these spaces is limited to a slab representing projected floor area and steel representing the stairwell structures.

5. MELCOR CALCULATIONS FOR CONTAINMENT FILTERED VENTING SYSTEM ANALYSIS

In developing the MELCOR calculation matrix for containment filtered venting system analysis, a set of accident prevention and mitigation measures were considered, informed by the lessons learned from the Fukushima event, accident management alternatives contemplated by the industry, the current state of knowledge of severe accident progression in a BWR and mitigation alternatives, and by the experience gained from the SOARCA study. The accident scenarios considered are both long-term and short-term station blackout (SBO) leading to one of three possible outcomes: containment overpressure failure, liner melt-through failure, or maintaining the containment intact as a result of venting or other mitigation measures.

In a SBO with the loss of all cooling function and absent any mitigation measures, the core is going to uncover leading to heatup, degradation, relocation of degraded core into lower plenum, thermal loading of the reactor pressure vessel (RPV) lower head and consequent lower head failure, relocation of core debris into the reactor cavity, and ultimate containment failure by overpressure or other mechanisms. It is assumed that low-pressure core injection (LPCI), high-pressure core injection (HPCI), drywell spray, and other engineered safety features (ESF), normally designed to run by AC power, become unavailable for an extended period of time.

For this type of situation, the reactor core isolation cooling (RCIC) system is designed to provide core cooling, thus delaying core uncovery and subsequent accident progression until such time other DC-powered (battery or diesel generator) and portable mitigation systems become available. The RCIC operation is controlled by battery, which acts as a power source for control valves that run the RCIC pump on and off. Before battery depletion, the RCIC is throttled to maintain a nominal RPV water level. In the SOARCA model, the RCIC continues operating after battery depletion, albeit in a “locked” state (i.e., without throttling). Cooling of the core by RCIC continues during this period.

The operation of RCIC was considered as the first preventative/mitigative feature in developing the MELCOR calculation matrix. For most MELCOR cases documented in this report, core cooling by RCIC continues for 2 hours or so after battery depletion until the main steam lines are flooded. In a few cases, the RCIC operation was specified so as not to have an additional period of core cooling from steam line flooding.

The SOARCA study assumed RCIC operation for 4 hours. Many, if not most, BWR Mark I plants are equipped with batteries that will allow RCIC to run for an extended period of as much as 8 hours. Moreover, in the post-911 development of accident management strategies, conceivably even a longer battery life for RCIC operation may have been considered. In Fukushima Dai-ichi Unit 2, RCIC operation in excess of 70 hours has been reported although the reason for such an extended operation is yet unknown. Likewise, in Fukushima Dai-ichi Unit 3, RCIC operation on the order of 20 hours has been reported, followed by another 16 hours of HPCI operation that kept the core cooled. With these considerations in mind, RCIC operation of 16 hours has been assumed in most of the MELCOR calculations reported here. For sensitivity analysis, one calculation with RCIC operation time of 4 hours (so the results can be compared with the SOARCA results) and a limited number of calculations with RCIC operation time of 8 hours were also performed. Also, a calculation was performed with RCIC failing to start, simulating a short-term SBO scenario.

Upon termination of RCIC operation, the next mitigation feature considered in the current study is actuation of core spray. As it is not clear at this time that the HPCI system can be actuated with portable devices, a diesel generator driven fire water system was considered to feed the low-pressure core spray system but only after RPV depressurization. A 300 gpm flow rate for the core spray was used in the analysis.

Another mitigation feature considered in the current study is drywell spray with a nominal flow rate of 300 gpm. As in the case of core spray, the drywell spray is assumed to be operated by a diesel-powered portable device. The drywell spray is actuated at 24 hours which, in most cases, correspond to the timing of RPV lower head failure. Variation in drywell spray actuation time was considered as part of the sensitivity study.

In addition to the mitigation features above, containment venting was considered in the current study in a number of ways. The primary function of venting is to prevent containment failure by overpressure from steam and other noncondensable gases. The BWR Mark I plants were originally designed with wetwell vents that had a low pressure capacity. As a result of post-TMI improvements, the wetwell vents in many of these plants have been upgraded and “hardened” for a high pressure capacity. Nevertheless, the vents were not designed or upgraded for operation under severe accident conditions. Core degradation and consequent hydrogen generation from steam oxidation of the degraded core and other core structures will add to containment loading resulting in containment overpressurization. In this situation, venting will prevent containment failure by overpressure, and greatly reduce the hydrogen, steam, and radioactive airborne contamination leaking into the secondary containment which could have resulted in a high dose environment, thus impeding accident mitigation and recovery actions by the operators. However, venting will also create a leakage path for fission products to escape to the environment, thus increasing health and land contamination risk. For these reasons, venting alone is not considered an adequate accident management measure; rather, venting in combination with other mitigation features is considered for further investigation in the current study. In all cases where venting is considered, it is initiated at a pressure of 60 psig.

Venting through the wetwell has the advantage of attenuating fission products through suppression pool scrubbing. Generally speaking, the fission products or aerosol particle size distribution is altered through the suppression pool scrubbing process as shown in Figure 7 below for a 300 cm deep pool and representative accident conditions. This figure illustrates the aerosol removal efficiency being highly dependent on particle size. The probability distributions in the figure are mass weighted and normalized to the total mass input to the pool. The dashed portion of the input size distribution curve (green) in the figure denotes large particles (>10 μm) that are typically deposited in transport and do not actually reach the suppression pool. Particles larger than 1 μm are efficiently removed by gravitational settling or inertial impaction. Very small particles (<0.1 μm) are removed by the diffusion process. Particles of intermediate size are removed by interception with the bubble in the suppression pool. When all removal mechanisms are considered, the efficiency of removal passes through a minimum when plotted against particle size. The particle size corresponding to the minimum efficiency (correspondingly known as the “maximum penetration size”) is around 0.2 μm. The difference in removal efficiency between the larger particles and the maximum penetration size particles can be two orders of magnitude or more.

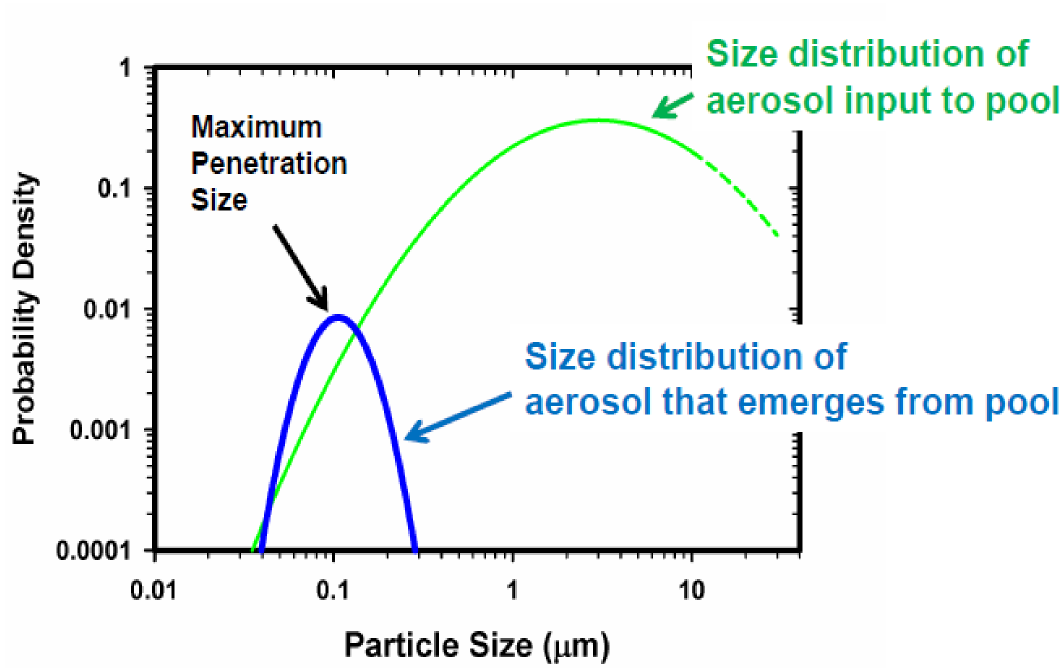


Figure 7. Aerosol removal efficiency as a function of particle size

Decontamination from pool scrubbing both attenuates the amount of mass and narrows the size distribution so the altered size distribution (blue) is centered around the maximum penetration size. Additional decontamination may result from adding a filter on the vent line. The effectiveness of the filter at the wetwell end varies depending on the filter design and construction. The altered particle size distribution emerging from the suppression pool is not nearly as amenable to further decontamination by a filter of the traditional variety. More recent filtration technology appears to provide a DF far in excess of the somewhat low range of DF achievable by traditional filters. Note the suppression pool itself has a DF calculated internally within the SPARC90 module of the MELCOR code, which was benchmarked against the Electric Power Research Institute (EPRI)-sponsored Advanced Containment Experiment (ACE) data [11] and Battelle Columbus Laboratories experiments [12]. The combined DF (i.e., suppression pool and wetwell filter) can be quite significant.

In the current MELCOR analysis, the calculated suppression pool DF varies from nominally 100 to 300 depending on the pool depth, pool temperature, and other factors. This range of calculated DF is bounded by the estimated pool DF (see Figure 8), which has a much larger variation and correspondingly, large uncertainties [13]. Also, in the current MELCOR analysis, a DF in the range of 2 to 10 is assumed for the wetwell filter instead of a DF of 1,000 or more that the currently available filtration technology can provide. This assumption is predicated upon the fact that the aerosol size distribution is altered after going through the pool scrubbing process, and the altered size distribution is not nearly as amenable to high decontamination as the original size distribution would be.

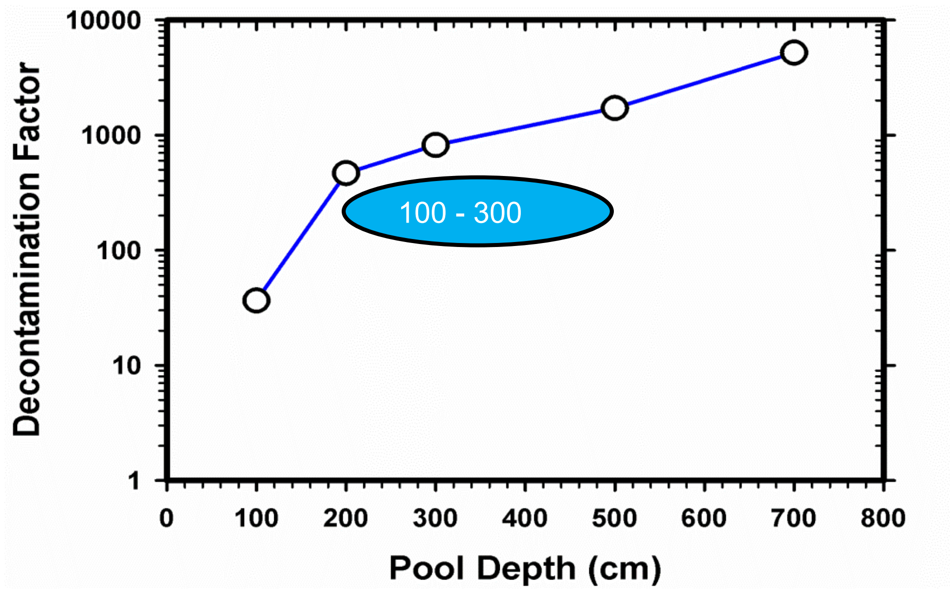


Figure 8. Estimated decontamination factor as a function of pool depth

The BWR Emergency Procedure Guidelines, which form the basis for plant specific emergency operating procedures, contain provisions for containment venting through the wetwell and drywell. Some BWR Mark I plants have drywell vents in addition to wetwell vents. In those plant configurations where the wetwell vent path is blocked (e.g., high suppression pool level), drywell venting essentially serves the same purpose as the wetwell venting in most designs. The drywell venting does not have the benefit of suppression pool scrubbing upstream of the vent. However, since the fission product aerosols are not scrubbed by the pool after reactor vessel breach, they retain their original size distribution by and large and are, therefore, amenable to significant attenuation by a filter at the drywell end. As a variation to wetwell venting, the current analysis considered two cases of drywell venting for comparison.

A large number of MELCOR cases were run for the containment venting study as described below. Most of the cases represent long-term SBO as in the Fukushima event and the SOARCA studies. Also, these cases consider RCIC operation and a combination of one or more mitigation features such as core spray, containment spray, and venting. The cases with venting include the option of wetwell (majority of the cases) and drywell venting (two cases). Collectively, the MELCOR cases provide all representative combinations of prevention and mitigations measures which are considered in the description of options used in the regulatory analysis (Enclosure 1). The MELCOR cases are summarized below in Table 3.

MELCOR does not model the effect of an external filter on fission product releases. This effect is considered in the MACCS analysis through the use of a prescribed DF value. In other words, in those cases where venting is present, release fractions calculated by MELCOR are used to perform two sets of MACCS calculations—one using the prescribed filter DF and the other as the unfiltered case. The comparison between the filtered case and the unfiltered case provides an indication of the effectiveness of filter.

Table 3. Matrix of MELCOR Cases for Containment Venting Study

MELCOR Case Description	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10
RCIC with 4-hour battery life	X									
RCIC with 8-hour battery life								X	X	X
RCIC with 16-hour battery life		X	X	X	X	X	X			
16-hour extended RCIC operation with 8-hour battery life										
Wetwell venting at 60 psig, vent open			X				X		X	
Wetwell vent cycled, open at 60 psig and close at 45 psig				X						
Drywell venting at 24 hours										
Core spray after RPV lower head failure						X	X			X
Drywell spray at 24 hours										
Drywell spray at 16 hours										
Drywell spray at 8 hours										
SRV stuck-open mechanism disabled—MSL creep rupture										
Traveling in-core probe leak to containment										
SRV seal leakage										
Short term SBO with RCIC failure to start										

Notes:

- Case 5 is a variation of Case 2 where the CST inventory is reduced to half its volume to determine the sensitivity. Makeup water for RCIC operation is provided from the CST until it is empty. After that, suction is taken from the suppression pool.
- Core spray flow rate is 300 gallons per minute (gpm) for Cases 6, 7, and 10.

Table 3. Matrix of MELCOR Cases for Containment Venting Study (continued)

MELCOR Case Description	Case 11	Case 12	Case 13	Case 14	Case 15	Case 16	Case 17	Case 18	Case 19	Case 20
RCIC with 4-hour battery life										
RCIC with 8-hour battery life	X									
RCIC with 16-hour battery life		X	X	X	X					
16-hour extended RCIC operation with 8-hour battery life						X	X	X	X	X
Wetwell venting at 60 psig, vent open	X									
Wetwell vent cycled, open at 60 psig and close at 45 psig					X					
Drywell venting at 24 hours		X	X							
Core spray after RPV lower head failure	X									
Drywell spray at 24 hours			X	X	X					X
Drywell spray at 16 hours									X	
Drywell spray at 8 hours								X		
SRV stuck-open mechanism disabled —MSL creep rupture		X	X							
Traveling in-core probe leak to containment										
SRV seal leakage										
Short term SBO with RCIC failure to start										

Notes:

- Drywell spray flow rate is 300 gpm for cases 13, 14, 15, 18, 19, and 20. Variations of flow rate considered in sensitivity analysis (Cases 28 through 30).
- Cases 16 through 25 were run with 2-volume wetwell nodalization in contrast to Cases 1 through 15, which were run with 16-volume nodalization. Two-volume nodalization provided improved computational efficiency.

Table 3. Matrix of MELCOR Cases for Containment Venting Study (continued)

MELCOR Case Description	Case 21	Case 22	Case 23	Case 24	Case 25	Case 26	Case 27	Case 28	Case 29	Case 30
RCIC with 4-hour battery life										
RCIC with 8-hour battery life										
RCIC with 16-hour battery life						X		X	X	X
16-hour extended RCIC operation with 8-hour battery life	X	X	X	X	X		X			
Wetwell venting at 60 psig, vent open	X	X	X	X	X			X	X	X
Wetwell vent cycled, open at 60 psig and close at 45 psig						X	X			
Drywell venting at 24 hours										
Core spray after RPV lower head failure										
Drywell spray at 24 hours								X	X	X
Drywell spray at 16 hours										
Drywell spray at 8 hours	X	X	X	X	X	X	X			
SRV stuck-open mechanism disabled —MSL creep rupture		X								
Traveling in-core probe leak to containment			X							
SRV seal leakage				X						
Short term SBO with RCIC failure to start					X					

Notes:

- Case 28 is variation of Case 14 with 100 gpm drywell flow rate, Case 29 is variation with 500 gpm flow rate, and Case 30 is variation with 1,000 gpm flow rate.

Cases 2, 3, 6, 7, 12, 13, 14, and 15 are selected as MELCOR base cases, the results of which are used for MACCS consequence calculations and for regulatory analysis. The rest of the cases were run as variations of the base cases for sensitivity analyses. The base cases represent no venting or spray (Case 2), wetwell venting but no spray (Case 3), core spray only (Case 6), core spray with wetwell venting (Case 7), drywell venting (Case 12), drywell venting and drywell spray (Case 13), drywell spray only (Case 14) and drywell spray with wetwell venting (Case 15). Collectively, the base cases provide all representative combinations of prevention and mitigations measures which are considered in the description of options used in the regulatory analysis (Enclosure 1). For example, Case 2 with no venting or spray maps to Option 1 (status quo) in the regulatory analysis. Likewise, all venting cases (Cases 3, 7, 12, 13, and 15) map to Option 2 (severe accident capable vent) and, when considered in combination with an external filter, to Option 3 (filtered vent). Case 6 and Case 14 (both nonventing but with sprays) may be considered variations of Option 1. Note the base cases are similar to the cases used in EPRI's analysis mentioned before with one exception. EPRI considered cycled venting as a mitigation strategy in its analysis. The MELCOR base cases do not include cycled venting; however, this mitigation feature was considered as part of additional MELCOR sensitivity analysis. As discussed later, MELCOR analysis did not find any significant differences between cycled venting cases and once-open venting cases with regard to fission product release estimates.

All the base cases assumed RCIC operation with 16-hour battery life. Each calculation was terminated after 48 hours of transients, consistent with the SOARCA study and based on observations therein that fission product releases occur mostly in the first 48 hours. MELCOR calculations and the results are discussed in detail below. Table 4 shows the timing of key events and MELCOR results for selected base cases. A discussion of the sensitivity cases and their results is provided following the discussion of the base cases.

5.1 Case 2 (No Venting or Spray)

Case 2 represents a long-term SBO situation resulting in the loss of all cooling functions. RCIC is operational by battery power with a mission time of 16 hours. The RCIC flow terminates at about 18 hours after SBO (additional 2 hours after depletion of battery). The core is subsequently uncovered at about 23 hours after SBO. Core oxidation starts shortly thereafter, resulting in hydrogen production. In the meantime, core degradation proceeds, resulting in core relocation to the lower head and subsequent lower head dryout at about 30 hours. The thermal loading of the lower head at this time and forward ultimately leads to its gross failure at about 37 hours.

Since this case does not allow any venting, the pressure from steam and noncondensable builds up in the containment leading to drywell head flange leakage at a pressure exceeding its design limit of 80 psig. (This assumed leakage scenario is based on the information available and analysis performed on Fukushima Dai-ichi Unit 1.) The leakage starts long before the RPV lower head failure. With the leakage path created, any pressure buildup in excess of 80 psig due to continued noncondensable production is relieved, and the drywell pressure remains at the design limit until the failure of the lower head by thermal loading.

Table 4. Matrix of MELCOR Calculations Showing Timing of Key Events

Event Timing (hr.)	Case 2 (no venting)	Case 3 (wetwell venting)	Case 6 (core spray)	Case 7 (core spray + wetwell venting)
Station blackout	0.0	0.0	0.0	0.0
RCIC flow terminates	17.9	17.9	17.9	18.0
Active fuel uncovery	22.9	22.9	22.9	22.9
First hydrogen production	23.6	23.6	23.6	23.2
Relocation of core debris to lower plenum	25.9	25.9	25.9	25.8
RPV lower head dries out	30.3	28.6	29.6	28.1
RPV lower head fails grossly	37.3	34.3	36.7	33.8
Drywell pressure > 60 psig—vent opens if applicable	22.8	22.8	23.3	23.2
SRV sticks open	22.7	22.7	22.7	22.7
Drywell head flange leakage (>80 psig)—overpressure failure	25.5	---	25.4	---
Drywell liner melt-through	40.3	36.6	---	---
Calculation terminated	48	48	48	48
Selected MELCOR Results	**Case 2**	**Case 3**	**Case 6**	**Case 7**
Debris mass ejected (1,000 kg)	286	270	255	302
In-vessel hydrogen generated (kg-mole)	525	600	500	600
Ex-vessel hydrogen generated (kg-mole)	461	708	276	333
Other noncondensable generated (kg-mole)	541	845	323	390
Iodine release fraction at 48 hrs	2.00E-02	2.81E-02	1.70E-02	2.37E-02
Cesium release fraction at 48 hrs	1.32E-02	4.59E-03	3.76E-03	3.40E-03

At lower head failure, the molten core debris relocates to the drywell cavity and spreads to the cavity perimeter and to the drywell liner. The thermal loading imparted on the liner by core debris challenges the liner integrity and the liner ultimately breaches by melt-through at about 40 hours or about 3 hours after the RPV failure. Core-concrete interactions, initiated due to molten core relocation to the drywell floor, generate noncondensable gases and fission product aerosols. These fission products along with those generated in-core are released to the environment at liner melt-through and the release is not scrubbed or filtered as the release path bypasses the wetwell.

Table 4. Matrix of MELCOR Calculations Showing Timing of Key Events (continued)

Event Timing (hr.)	Case 12 (drywell venting)	Case 13 (drywell venting + drywell spray)	Case 14 (drywell spray)	Case 15 (drywell spray + wetwell venting)
Station blackout	0.0	0.0	0.0	0.0
RCIC flow terminates	17.9	17.9	17.9	18.0
Active fuel uncovery	24.0	24.0	22.9	22.9
First hydrogen production	24.3	25.0	23.2	23.2
Relocation of core debris to lower plenum	28.3	28.7	25.7	25.6
RPV lower head dries out	28.9	29.1	29.4	29.3
RPV lower head fails grossly	34.2	34.7	36.6	35.3
Drywell pressure > 60 psig—vent opens if applicable	27.7	27.7	23.2	23.3
SRV sticks open	27.2	27.3	22.7	22.4
Drywell head flange leakage (>80 psig)—overpressure failure	27.6	---	35.2	---
Drywell liner melt-through	34.8	35.1	---	---
Calculation terminated	48	48	48	48
Selected MELCOR Results	**Case 12**	**Case 13**	**Case 14**	**Case 15**
Debris mass ejected (1,000 kg)	345	351	267	257
In-vessel hydrogen generated (kg-mole)	670	750	614	650
Ex-vessel hydrogen generated (kg-mole)	774	410	327	276
Other noncondensable generated (kg-mole)	922	485	383	270
Iodine release fraction at 48 hrs	4.90E-01	4.84E-01	5.41E-03	1.86E-02
Cesium release fraction at 48 hrs	1.93E-01	1.86E-01	1.12E-03	3.01E-03

5.2 Case 3 (Wetwell Venting)

Case 3 is basically identical to Case 2, but this time venting is in effect. The wetwell vent opens at about 23 hours after SBO when the drywell pressure exceeds 60 psig and remains open. This prevents containment failure by overpressure. However, as in Case 2, at lower head failure, the relocated core debris on the drywell floor spreads to the liner, and attacks the same leading to melt-through and containment breach. Also, as in Case 2, fission products are released to the environment at liner melt-through and this release is not scrubbed or filtered as the release path bypasses the wetwell. However, any release through wetwell vent prior to liner melt-through (a duration of about 14 hours between vent opening and liner melt-through timing) is scrubbed efficiently by the suppression pool.

5.3 Case 6 (Core Sprays)

Case 6 examines the mitigation effect of core spray. In the present study, it is assumed that the core spray can only be actuated at a sufficiently low pressure when using a hookup from a portable fire-water system. Specifically, it is assumed that the core spray system is actuated at reactor vessel failure, which depressurizes the vessel. Moreover, it is assumed that a nominal flow rate of 300 gpm is achievable using a portable system. With the core spray actuation at vessel breach, water finds its way to the drywell floor. The net effect is slowing down of core debris spreading on the floor to the point of effectively freezing the debris, thus arresting further progression. As a result, the liner melt-through is prevented. However, since this case does not involve any venting, the drywell pressure builds up, leading eventually to containment overpressure failure (through head flange leakage).

5.4 Case 7 (Core Sprays and Wetwell Venting)

Case 7 builds on Case 6 by adding wetwell venting. As in Case 6, actuation of core spray at vessel breach using a fire water system at 300 gpm flow rate provides water to the drywell floor. The net effect is slowing down of core debris spreading and prevention of liner melt-through. Any fission product release, in this case, goes through wetwell vent and in the process, gets scrubbed by the suppression pool.

Case 7 may be contrasted with Case 6 as well as the two previous cases (Case 2 and Case 3) with no spray action. Case 6 examines the effect of core spray only with no venting in effect. While the absence of venting in Case 6 leads to containment overpressure failure by drywell head flange leakage as in Case 2, actuation of core spray has a scrubbing effect on in-core fission products. As a result, the total amount of fission products released in Case 6 is smaller than those for both Case 2 and Case 3. Case 7 examines the combined effect of core sprays and venting. The combined effect shows smaller release amounts in Case 7 than those in Case 2 and Case 3, and similar releases compared to Case 6.

5.5 Case 12 (Drywell Venting)

Case 12 explores the efficacy of drywell venting with an external filter downstream of the vent. In Case 12 (also in Case 13 discussed later), the SRV sticking mechanism was disabled and the wetwell vent was assumed closed. This is to simulate a transport path of steam, noncondensable gases, and fission products through the drywell vent.

Some BWR Mark I plants implemented a severe accident management strategy whereby the reactor cavity is flooded above the wetwell vents making wetwell venting inoperable. Some Mark I plants in the United States are equipped with both drywell and wetwell vents and depending on the accident sequences and failure modes, either or both may be operable. Any fission products released through a drywell vent will not be scrubbed at all unless there is a filter. By the same token, fission products passing through a filtered drywell vent will be greatly attenuated since the size distribution of fission product aerosols is amenable to a high degree of decontamination.

Case 12 is simulated in MELCOR in a manner that is similar to a main steam line rupture scenario considered for the Fukushima Dai-ichi Unit 1. The MELCOR analysis of Unit 1 shows that core exit gas temperatures began to significantly increase, with superheated steam and hydrogen gas flowing into the main steam line associated with the cycling SRV (achieved in MELCOR by disabling SRV sticking open mechanism). This hot gas heated the steam line significantly, and because the reactor pressure was high (at the lowest SRV set point), thermal creep in the hottest steam line eventually led to failure of the main steam line and resulting depressurization of the RPV. The loss of pressure boundary in this mode is believed to be consistent with the data from Fukushima Dai-ichi Unit 1.

5.6 Case 13 (Drywell Venting and Drywell Sprays)

Case 13 builds on Case 12 by adding a drywell spray operation initiated at 24 hours. As in Case 12, the SRV sticking mechanism is disabled and the wetwell vent is closed to simulate the transport path of fission products through the drywell vent. Also, as in Case 12, this case is simulated in MELCOR in a manner that is similar to a main steam line rupture scenario.

A nominal 300 gpm drywell spray flow rate was assumed in this scenario. This rate is an order of magnitude less than the design-basis flow rate of the installed drywell spray header. However, this nominal flow rate is considered reasonable when a portable or diesel operated device is used in an SBO. It is anticipated that the sprays would provide the benefits of washing some fraction of the fission product aerosols from the drywell atmosphere, thus making them unavailable for release to the environment.

5.7 Case 14 (Drywell Sprays)

Case 14 explores the effect of drywell sprays which differ from core sprays (Case 6) in terms of their influence on containment pressure and aerosol washoff. Sprays with a flow rate of 300 gpm are actuated at 24 hours (about 2 hours before the drywell head flange leakage and more than 12 hours before the lower head failure). Since there is no vent opening, the case results in containment failure by overpressure. Fission products, leaked through the head flange leakage path, are scrubbed to some degree by the drywell spray action, which reduces the release to the environment. Likewise, fission products released after vessel failure are scrubbed by the spray.

5.8 Case 15 (Drywell Sprays and Wetwell Venting)

Case 15 is a variation of Case 14 in which wetwell venting is added. This action prevents containment overpressure failure. The RPV lower head fails in this case at about 35 hours (as opposed to 36 hours for Case 14), whereas the containment spray starts at 24 hours (same as in Case 14). The spray action creates a pool of water in the cavity and in the pedestal region. At vessel failure when the fission products are released, the pool of water provides some scrubbing effect. Further, as the fission products are released through the wetwell vent to the environment, additional scrubbing by the suppression pool takes place. The combined effect of the two scrubbing processes is significant, resulting in a smaller release to the environment.

Figure 9 shows a comparison of drywell pressure for six cases, three of which are not vented (Cases 2, 6, and 14) and the other three (Cases 3, 7, and 15) vented through the wetwell. As seen in this figure, both Case 2 and Case 3 lead to liner melt-through since there is no provision of water in these two cases to cool the core debris and prevent melt spreading to the liner. Case 7 and Case 14, on the other hand, have provision for water as do Case 6 and Case 15. Of these latter four cases, Case 6 and Case 14 do not have venting. As a result, these two cases lead to containment failure by overpressure, indicated in the figure by drywell head flange leakage. Case 2 also has no venting and hence, leads to overpressure failure. It is interesting to note that the drywell spray action in Case 14 relieves the containment pressure for a while and delays the overpressure failure by over 10 hours (~25 hours in Case 6 versus ~37 hours in Case 14). It is also interesting to note that in Case 14, gross failure of the RPV lower head precedes the head flange leakage, though not by much, whereas in Case 2 and Case 6, the lower head fails much later. Cases 3, 7, and 15 all have venting which prevents containment overpressure failure.

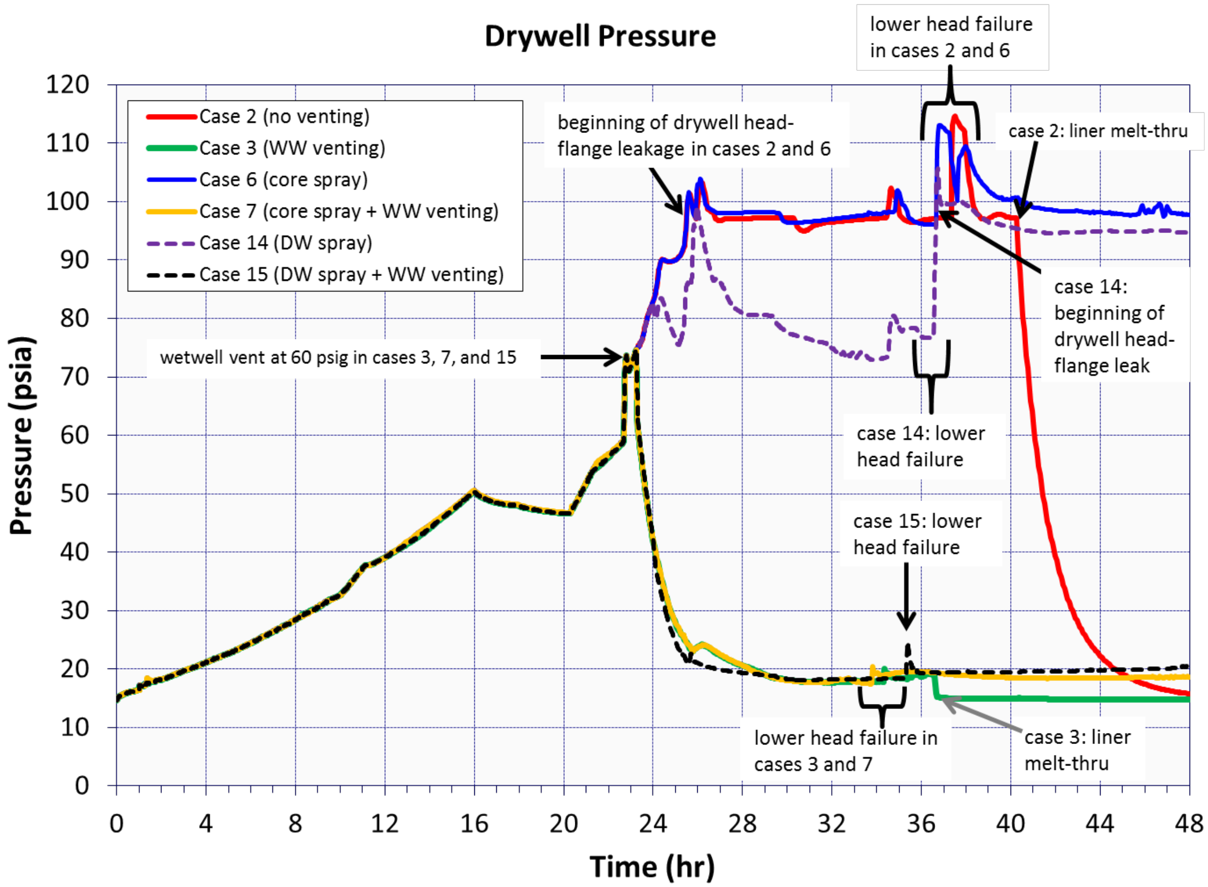


Figure 9. Comparison of drywell pressures for selected cases

Figure 9 also indicates that in all venting cases, the lower head failure occurs earlier than in nonventing cases (by about 2 to 3 hours). It is important to note the degree of core oxidation strongly affects the timing of lower head failure and that oxidation is steam limited at times. Steam evolves largely from the boiling of water in the reactor lower plenum but not entirely. It also evolves from flashing liquid as the reactor depressurizes. The rate of depressurization governs the rate that steam via flashing. As long as choked flow persists in the SRV (the lowest set-point SRV) reactor pressure is not responsive to containment pressure. However, choked flow will abate at some point as the reactor depressurizes through a failed (stuck-open) SRV. Once choked flow through the SRV abates, containment pressure influences reactor pressure and hence steam evolution from flashing. Lower containment pressures relate to lower reactor pressure and more flashing. The LTSBO cases where the containment is vented with the vent left open lead to very low containment pressures and hence, more steam production from flashing, lower reactor pressure, more oxidation, and hydrogen generation seen in Figure 10 below. Finally, increased oxidation leads to hotter debris relocated to the lower plenum and earlier lower head failure.

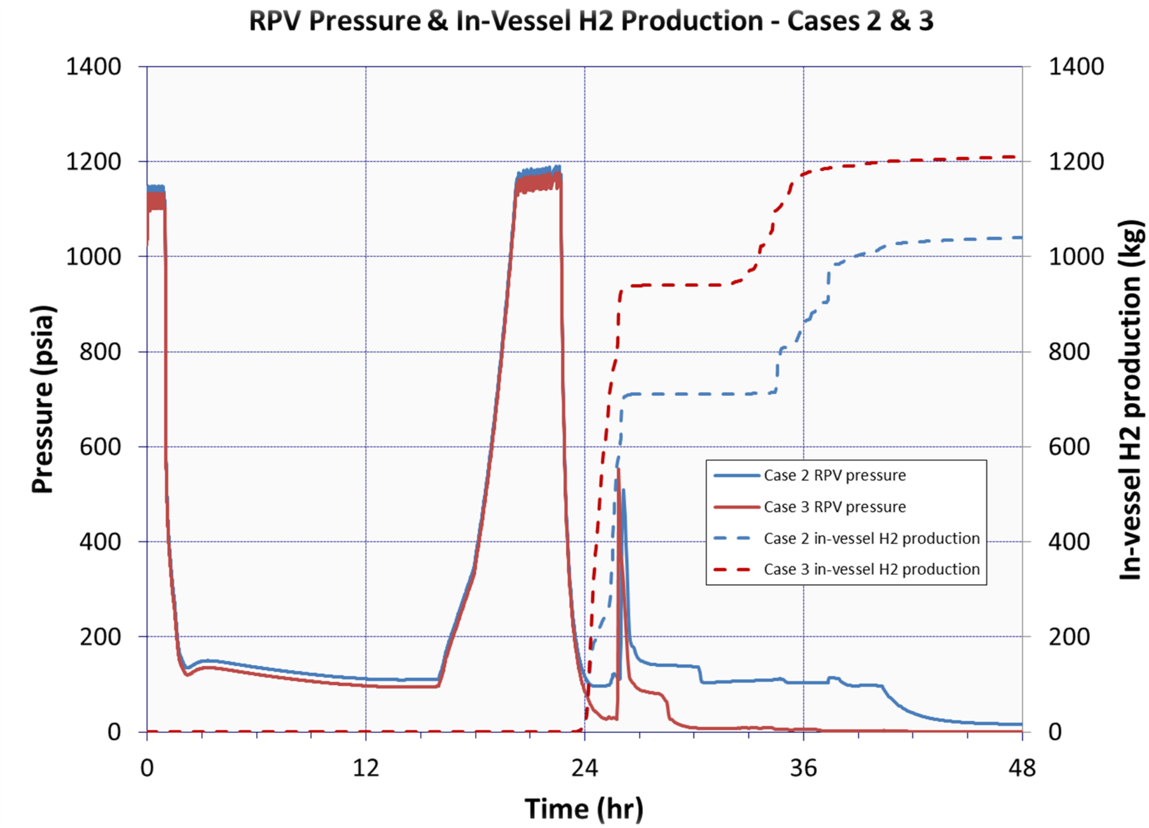


Figure 10. Comparison of RPV pressure and in-vessel hydrogen production in Cases 2 and 3

Figure 11 shows another comparison of drywell pressure for six cases—this time two of the venting cases involve drywell venting (Cases 12 and 13) and the other four are Cases 2, 3, 6, and 7 as before.

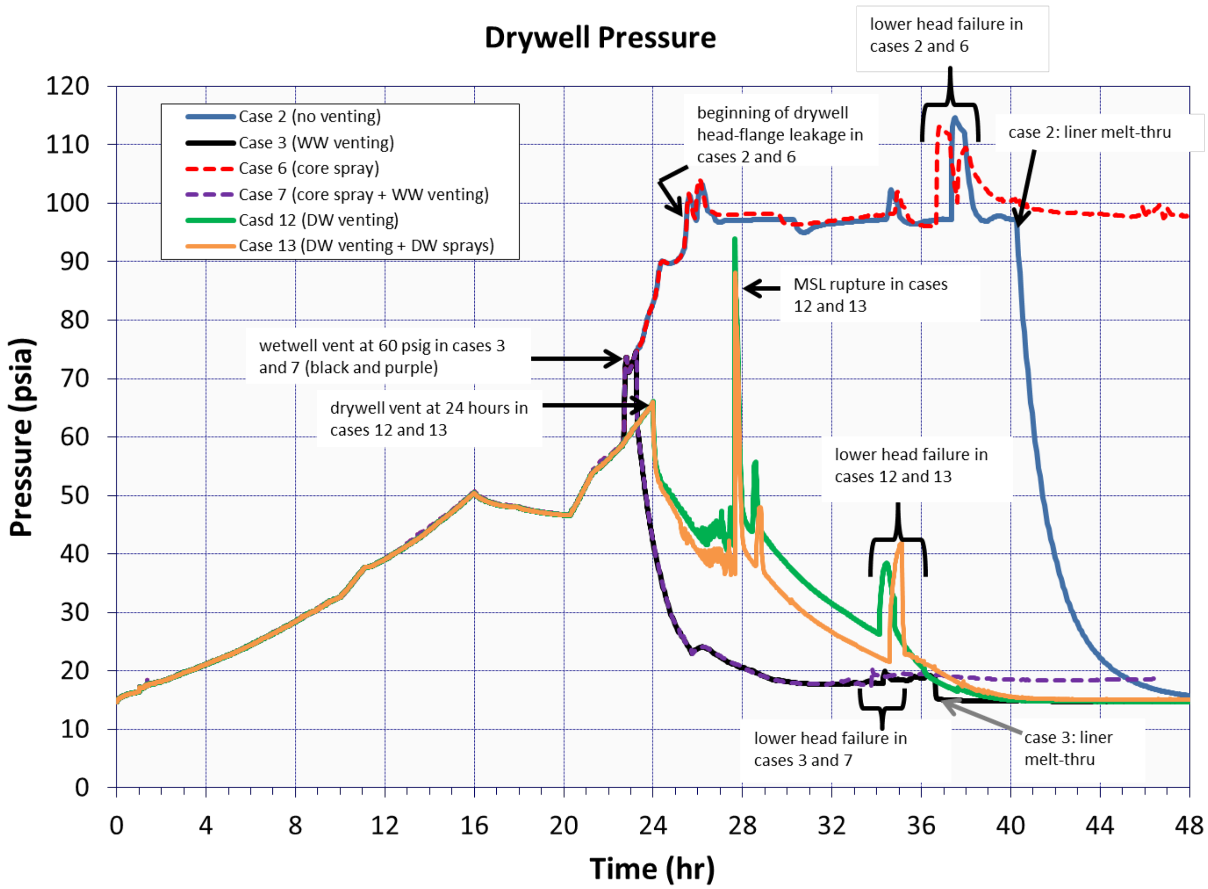


Figure 11. Comparison of drywell pressures for selected cases highlighting drywell venting

Disabling the SRV failure mechanisms in Cases 12 and 13 led to main steam line creep rupture as seen in Figure 11. Generally, in the LTSBO calculations, developing a pool on the containment floor prior to reactor lower head failure allowed core debris to quench to the point that it could not migrate to the drywell liner and hence could not melt through the liner. In Case 13, however, disabling the SRV failure mechanisms led to more core degradation occurring at pressure (i.e., at SRV safety set-point pressure). More core damage occurring at pressure led to more oxidation and hotter debris temperatures. The hotter temperature of debris exiting the vessel kept the pool on the drywell floor from quenching the debris enough that it could not migrate to the drywell liner. The debris cooled substantially but still managed to move to and melt through the liner.

The cesium release fractions for the cases shown in Figure 9 are compared in Figure 12 below. Cases 2 and 3 both of which lack any mitigation measure involving water (i.e., core sprays or drywell sprays) show the highest release fractions, as expected from a liner melt-through type failure. The fission product releases in these two cases bypass the wetwell after the core exits

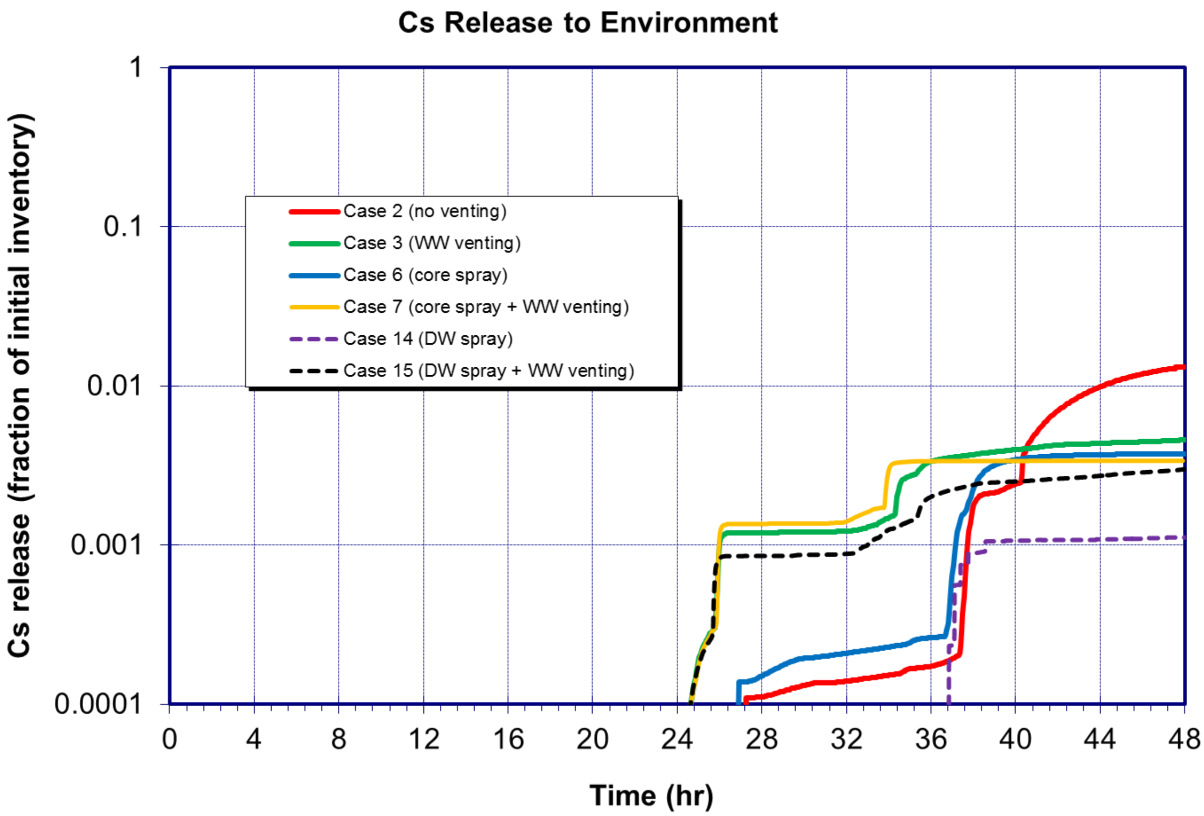


Figure 12. Comparison of cesium release fractions for selected cases

the reactor vessel and, as a result, are not scrubbed by the suppression pool. Cases 6 and 7 involving core spray action show moderate effect on fission product attenuation and resulting lower amount of cesium release fractions. The drywell spray action (Cases 14 and 15) show the largest reduction in cesium release fractions. In Case 14, drywell sprays provide significant scrubbing whereas in Case 7, core sprays provides limited scrubbing.

In contrasting Case 15 with Case 14, the difference in the release of cesium to the environment appears to be counterintuitive. Release in Case 15 is higher than in Case 14, even though Case 15 has the supposed benefit of wetwell venting complete with pool scrubbing. There are a couple of reasons for this difference. First, drywell sprays are efficient in Case 14 at keeping containment pressure low enough that there is very little gas leakage past the drywell head flange relative to the amount of gas relieved through the wetwell vent in Case 15. Second, the lower containment pressure in Case 15 (resulting from the wetwell venting) fosters substantially more revaporization of cesium (and other fission products such as iodine) off reactor vessel internals. The vapors escape the reactor and condense to aerosols that are carried towards the wetwell vent. Some of the aerosols are scrubbed in the wetwell pool but not all of them. The aerosols not scrubbed in the pool release to the environment through the wetwell vent. In considering the scrubbing taking place in the wetwell pool during wetwell venting in Case 15, note the flow to the wetwell is through the downcomer vents rather than through a T-quencher. The DF of 10 associated with a downcomer vent is markedly less than the DF of 1,000 associated with a T-quencher. Evidently the increased revaporization of cesium off reactor internals combined with the larger vent flows and less effective wetwell scrubbing in Case 15 lead to the larger releases of fission products to the environment in Case 15 relative to Case 14.

In considering the releases in Case 14, it is worth noting that the drywell head flange leakage model implemented in MELCOR assumes elastic deformation of head bolts and flange seal and does not address inelastic deformation or temperature dependent effects. In reality, the head flange is likely to experience permanent deformation, in part due to aging and other degradation processes over time, and thus the flange gap is likely to widen over time, leading to higher leakage of fission products as well as noncondensable gases.

Cesium release fractions in Figure 12 may be contrasted with those in Figure 13 below, particularly, release fractions pertaining to drywell venting cases (Case 12 and Case 13). Evidently, both Cases 12 and 13 show nearly two orders of magnitude higher release relative to the wetwell venting cases in Figure 12. This is not unusual considering that the drywell venting cases do not have the benefit of pool scrubbing or other forms of decontamination. For the wetwell venting cases considered in the present study, the decontamination factors range between 100 and 300 as shown in Figure 9 above.

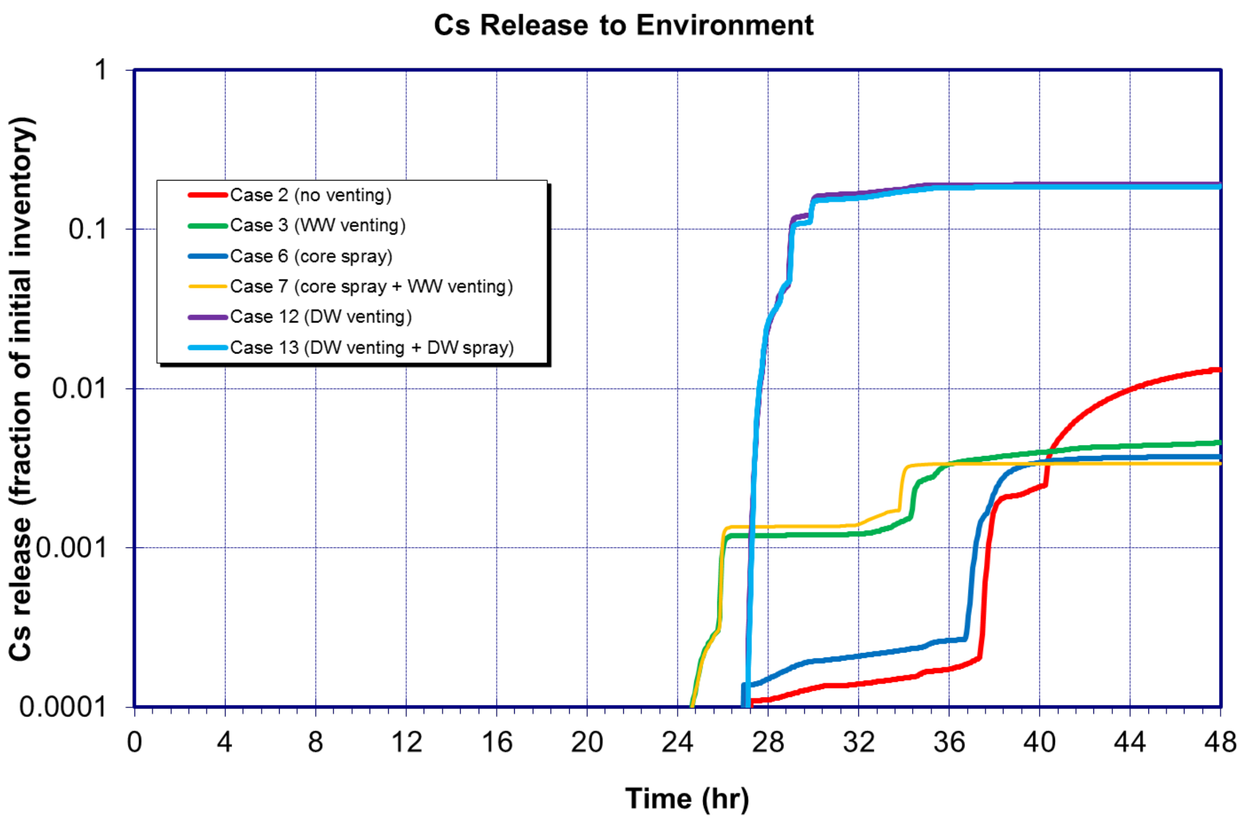


Figure 13. Comparison of cesium release fractions highlighting drywell venting

It is important to understand that two fundamental differences were introduced to the accident progression in Cases 12 and 13—both conducive to larger releases to the environment. First, when the MSL rupture took place, fission products escaped the RPV to the drywell rather than to the wetwell for a period of time preceding RPV lower head failure. The fission products introduced to the drywell were available for release through the drywell vent. Second, more core damage occurred at pressure (i.e., at SRV safety set-point pressure). Most all of the cesium released to the environment in the LTSBO calculations can be traced to the revaporization of material deposited on reactor internals during core degradation. The degree of revaporization increases with increasing time spent at pressure. Consequently, more cesium revaporized from reactor internals when the SRV failure mechanisms were disabled in Case 12. Some of the cesium vapors escaped the reactor vessel and condensed to aerosol, which was available for release through the drywell vent. Cesium and iodine releases to the environment were lower in Case 13 than in Case 12.

Another point is worth noting with regard to drywell/wetwell venting operation. The venting cases presented here do not consider any scenario where the venting is initiated at the wetwell and is transitioned later to drywell. Some plants have reportedly the capability to vent through both wetwell and drywell, and the severe accident management guidance may specify a combination of venting operation. In such a case, the initial release through the wetwell vent will be scrubbed by the suppression pool in the usual course, and the later release through the drywell vent will not be scrubbed. The total release in this case will be lower than that corresponding to a drywell venting only case, and somewhat higher corresponding to a wetwell venting only case. In that sense, release estimates in Figure 13 may be considered bounding.

Note that the fission products are released through several pathways, some of which provide decontamination by natural means (deposition, settling, etc.) or by other means (e.g., suppression pool scrubbing). Other pathways do not provide any decontamination. Depending on the failure mode and location, the various decontamination processes can provide significant attenuation. As two examples, cesium release fractions by different release pathways are shown for two cases in Figure 14 (Case 7) and Figure 15 (Case 14), respectively. In Case 7, nearly 80 percent of cesium release fractions are associated with the wetwell vent path, whereas in Case 14, only about 50 percent is associated with the same path. In creating the input for MACCS calculations, release fractions from different paths are summed up taking into account the appropriate decontamination factors. MACCS calculations are then performed in two sets—one using an external filter with a defined decontamination factor and the other with no filter. The results are contrasted to determine the effect of external filter on consequences.

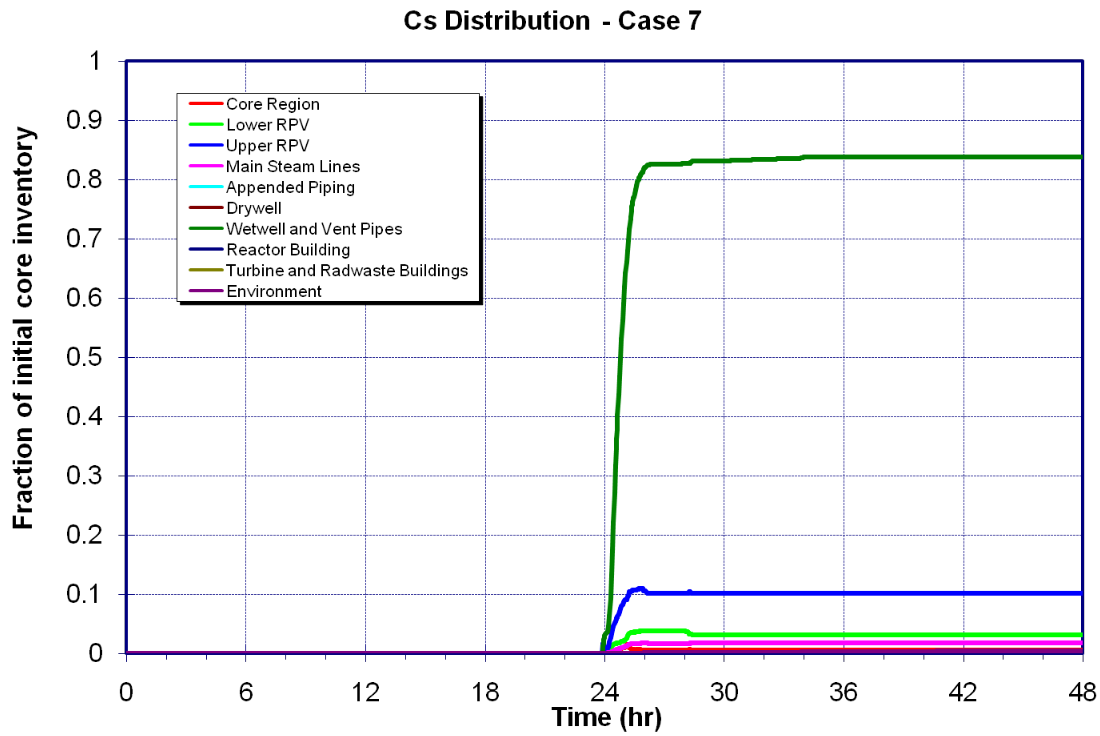


Figure 14. Cesium distribution by various pathways for Case 7

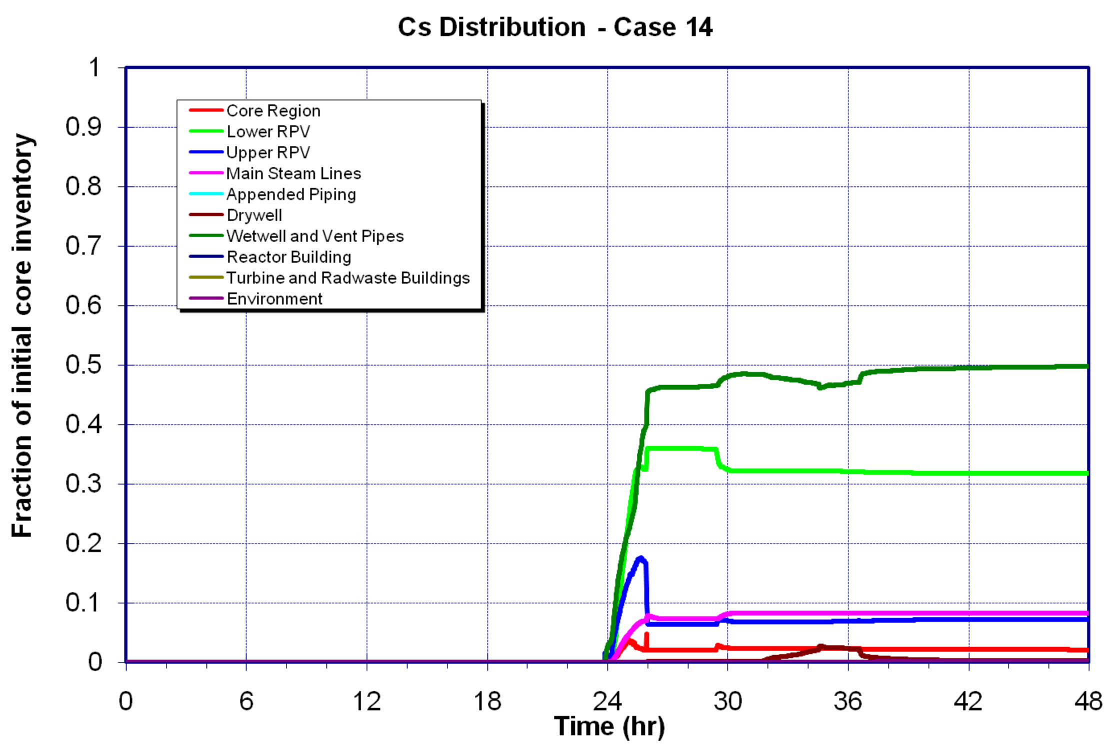


Figure 15. Cesium distribution by various pathways for Case 14

While the presence of water in one form or another has a beneficial effect on fission product scrubbing, this action can also influence the generation of hydrogen as may be evident from the comparison shown in Figure 16 for in-vessel hydrogen production. Consistent with the explanation provided earlier while contrasting venting versus nonventing cases, it is noted that the venting cases considered in the study generally produced 100 to 150 kg-mole (alternatively, 200 to 300 kg) of additional hydrogen in-vessel.

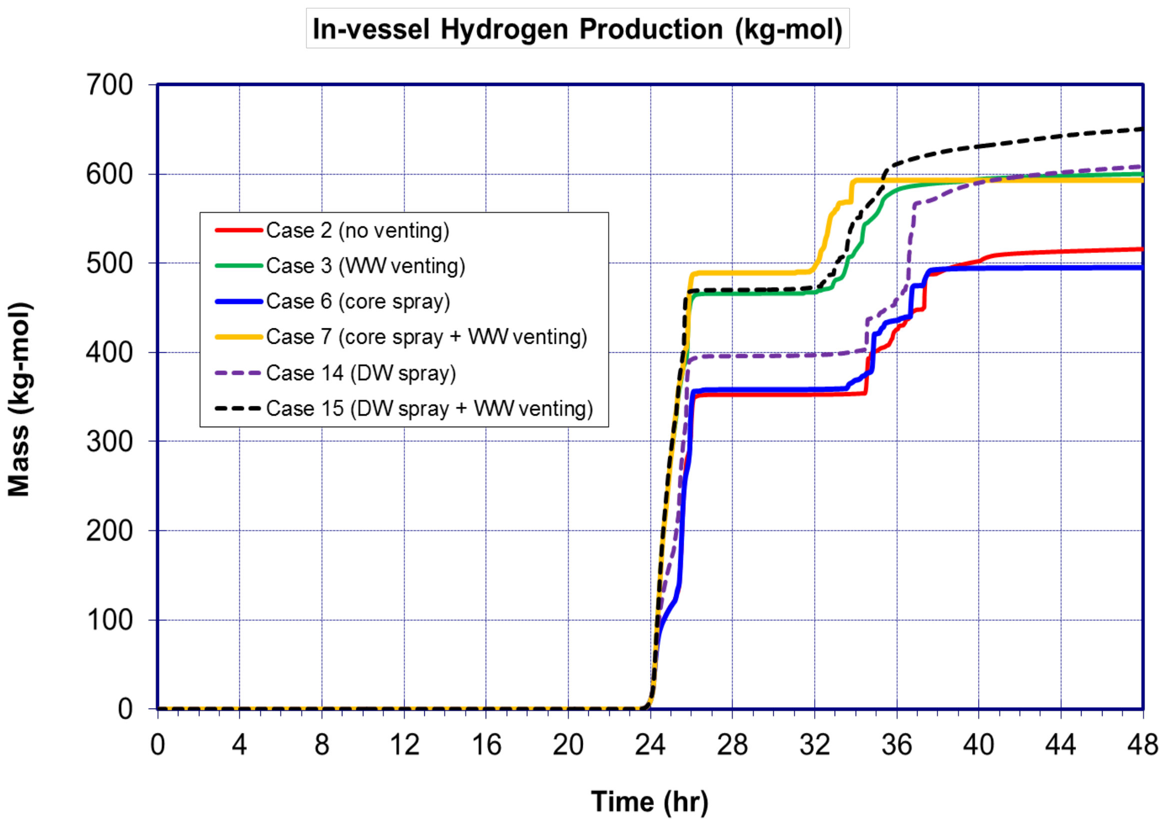


Figure 16. Comparison of in-vessel hydrogen production for selected cases

In-vessel generation in Figure 16 shows temporary cessation of hydrogen production as indicated by plateaus (horizontal segments). This is an artifact of MELCOR modeling of clad oxidation. The code considers clad oxidation to be in effect when certain criteria (e.g., minimum pre-oxide layer thickness and minimum temperature) are met. There is also the effect of steam starvation during which clad oxidation cannot take place.

Additional amount of hydrogen and other noncondensable gases (mostly carbon monoxide) are generated from core-concrete interactions (CCI) once the core debris relocates on the drywell floor as can be seen in Figures 17 (for hydrogen) and 18 (for carbon monoxide). The presence of water on the drywell floor has a slowing down effect on CCI and consequent noncondensable gas generation. As a result, less hydrogen and carbon monoxide is produced in all but two cases (Case 2 and Case 3). In these two cases, the amount of hydrogen generated is quite comparable to in-vessel generation amount.

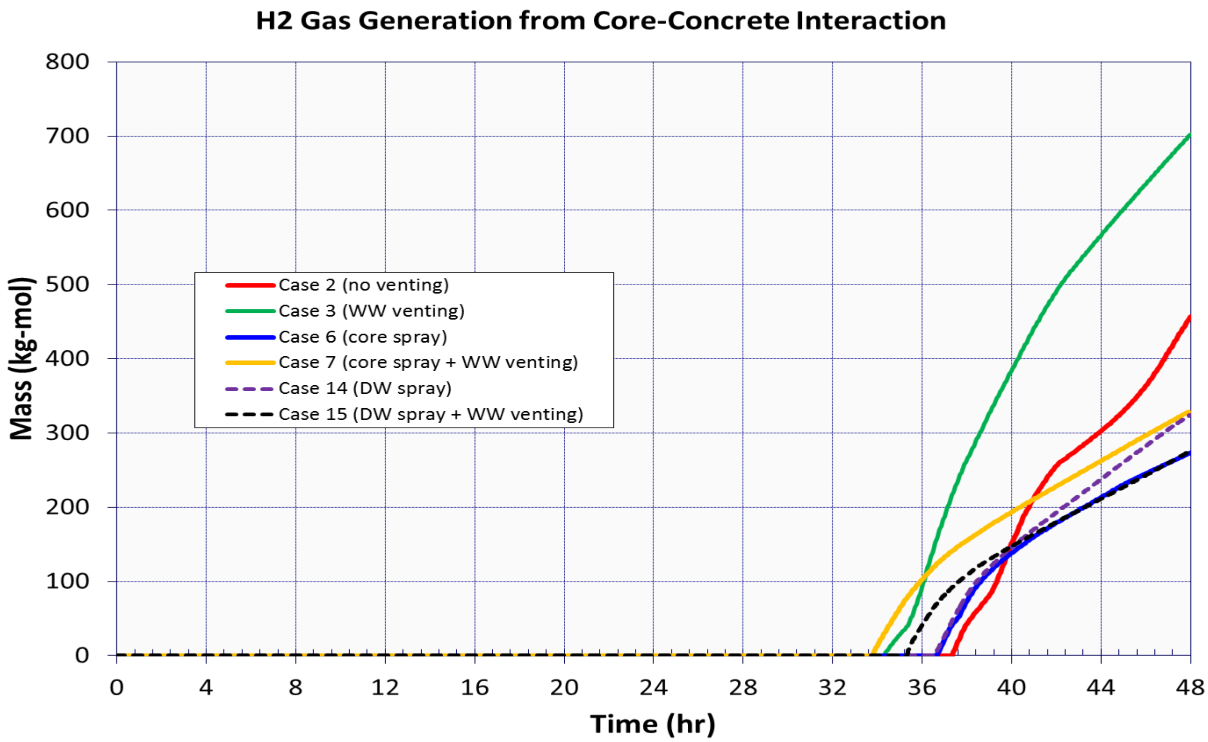


Figure 17. Comparison of ex-vessel hydrogen production

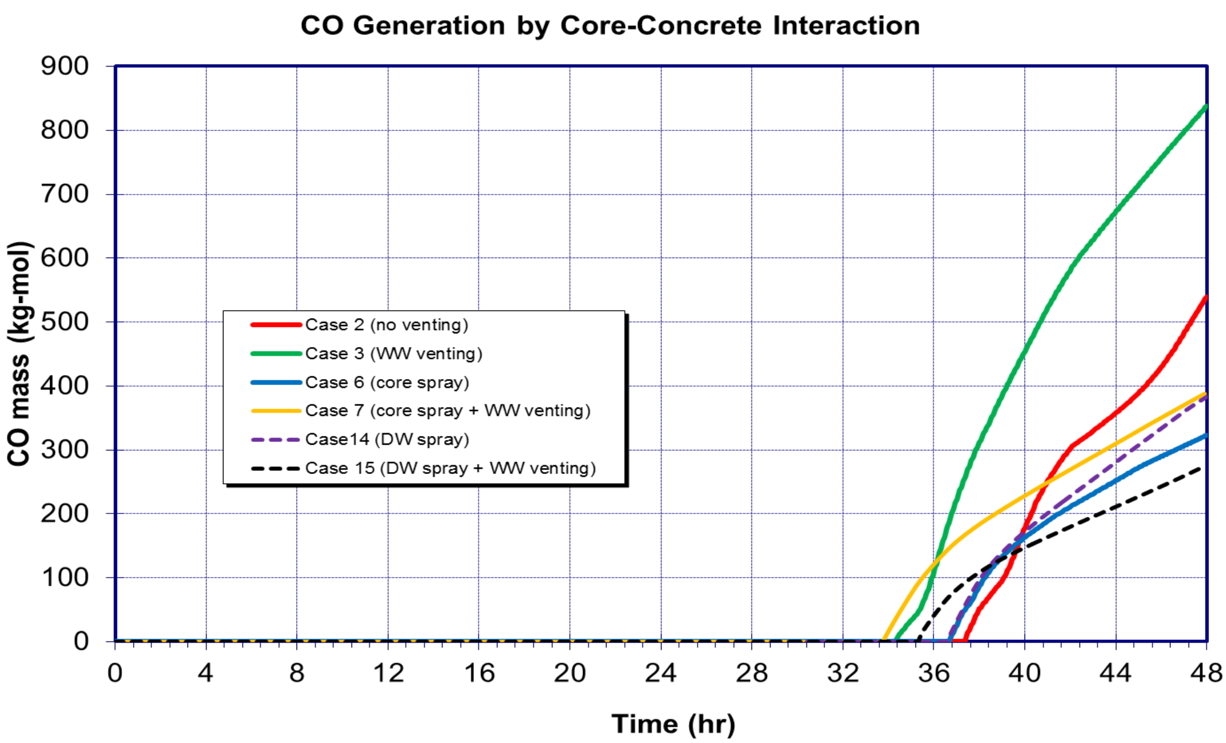


Figure 18. Comparison of ex-vessel carbon monoxide production

5.9 Additional MELCOR Cases for Sensitivity Analysis

Case 1 is identical to Case 2 with the exception of RCIC operation for a 4-hour battery time instead of 16-hour battery time. Likewise, Case 8 is identical to Case 2, except for a RCIC operation with 8-hour battery time. These three cases together are considered as sensitivity cases with variation in RCIC operation time, and are discussed later in more detail.

Case 4 is a venting case where the vent is allowed to cycle between 60 psig (opening) and 45 psig (closing). This case may be contrasted with Case 3 where venting, once opened, remained so through the duration of calculation. Comparison of MELCOR results of these two cases provides an insight of the relative merit of vent cycling.

Case 5 is identical to Case 2 with the added feature that only half of the inventory of the CST is provided for RCIC operation. Case 5 was a sensitivity calculation designed to investigate the dependence of containment pressurization on the source of RCIC suction. RCIC suction is initially from the CST. RCIC suction can optionally be from the wetwell, and by design, an automatic switchover of RCIC suction from the CST to the wetwell would occur as the CST neared depletion. In Case 5, the CST was initialized only half-full, forcing a switchover of RCIC suction to the wetwell.

Cases 8 through 11 represent RCIC operation for an 8-hour battery time. Case 8 is a variation of Case 2, which is already described above. Case 9 is a variation of Case 8 with wetwell venting operation and may be contrasted with Case 3, which has 16-hour RCIC. Case 10 is another variation of Case 8 with core sprays and may be contrasted with Case 6. Finally, Case 11 is a variation of Case 8 with venting and drywell sprays and may be contrasted with Case 7. Generally, the pronounced effect of the duration of RCIC operation is the delay of the onset of core melt progression and subsequent RPV failure.

Case 16 is a variation of Case 2 whereby a 2-CV (control volume) representation of the wetwell was adapted as opposed to 16-CV representation in all previous cases. This is to determine if a coarser representation is adequate for the purpose of MELCOR calculations while still capturing the effect of local temperature variation in the wetwell. Figure 19 shows a comparison of wetwell temperature between a 16-CV representation (Case 2) and a 2-CV representation (Case 16). There is clearly some difference in wetwell temperature between the two cases up to the time of core uncover (about 23 hours), beyond which both representations yield similar wetwell temperature.

Case 17 is a scenario where RCIC operation beyond the battery mission time is allowed by disabling the RCIC failure logic. This scenario is similar to Fukushima Dai-ichi Unit 2 where an extended RCIC operation was observed. Cases 18 through 20 represent sensitivity cases where the drywell spray actuation time was varied (8 hours after SBO in Case 18, 16 hours in Case 19, and 24 hours in Case 20) to examine the spray actuation timing effect on fission product attenuation. These cases were not vented, thus leading to containment overpressure failure. A variation of Case 18 was run with venting in effect (Case 21), which prevented overpressure failure. Note there was no liner melt-through in the last four cases.

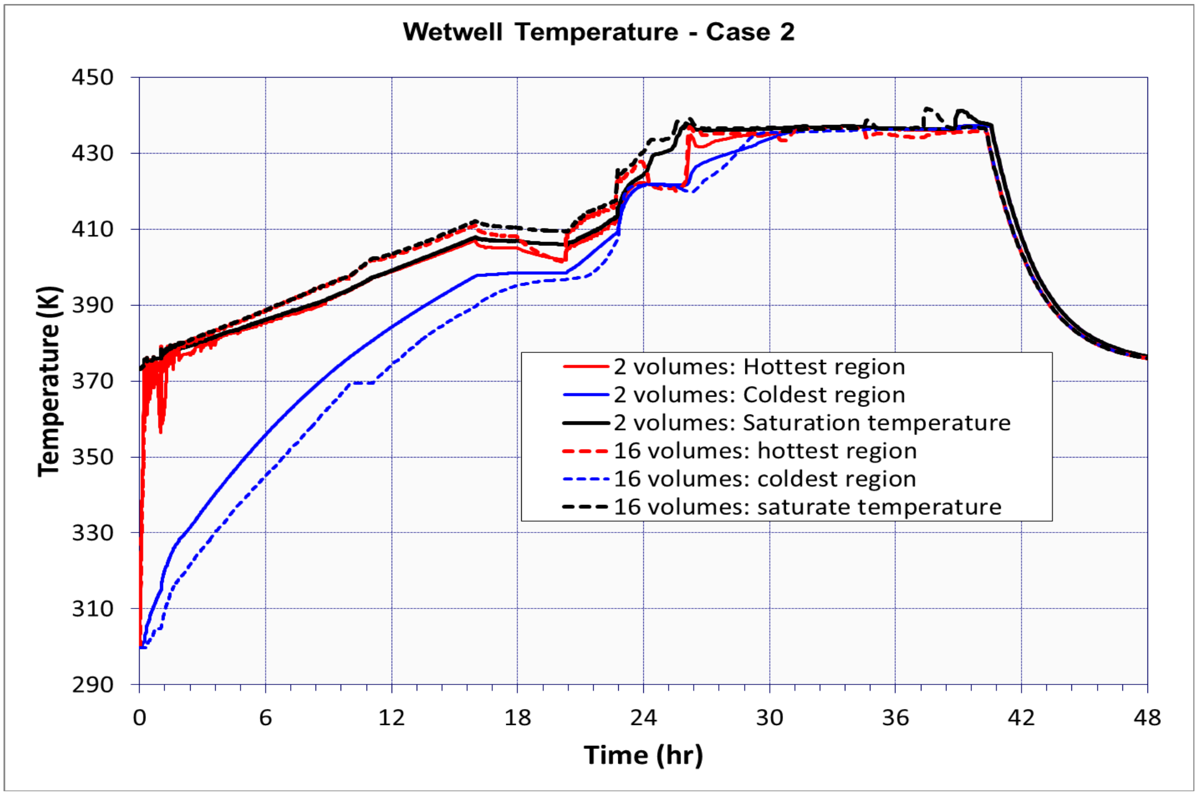


Figure 19. Comparison of wetwell temperatures between 16-CV and 2-CV representations

The next three cases involve different failure modes other than the gross lower head failure. Case 22 simulates main steam line creep rupture. Case 23 simulates a traveling in-core probe leak to containment. Case 24 simulates an SRV seal leakage. These various failure mechanisms have been postulated and are being examined to explore the events in Fukushima Dai-ichi plants. Case 25 represents a short-term SBO situation with RCIC failure to start. Note Cases 17 through 25 were all run with RCIC operation of 16 hours, well beyond the battery mission time of 8 hours assumed in these cases. In that sense, these cases may be considered as informed by what was observed in Fukushima Dai-ichi, Unit 2 and Unit 3.

Five additional cases were run to examine additional sensitivities. Case 26 and Case 27 examine the combined effect of vent cycling and drywell spray. Cases 28 through 30 examine the drywell spray flow rate sensitivity.

The MELCOR results pertaining to various sensitivity studies are discussed below.

5.10 RCIC Operation Sensitivity

RCIC is designed to provide core cooling, thus delaying core uncovery and subsequent accident progression until such time other DC-powered and portable mitigation systems become available. Cooling of the core by RCIC continues, however, for a period defined by the battery mission time. Following failure of RCIC to operate, the core uncovery will begin. As mentioned previously, baseline calculations were performed with 16-hour RCIC operation time (multiple cases), one sensitivity calculation (Case 1) was performed with RCIC operation time of 4 hours (so the results can be compared with the SOARCA results), and a limited number of calculations (Cases 8 through 11) were performed with RCIC operation time of 8 hours. Some of these cases involved consideration of additional mitigation measures such as spray.

Three cases (Case 1, Case 2, and Case 8)—all with no additional prevention or mitigation features—are compared here to provide an understanding of RCIC operation sensitivity. Figure 20 shows, as an example, the comparison of hydrogen production at reactor vessel failure for different duration of RCIC operation. The primary benefit of extended RCIC operation time is to delay the reactor vessel failure, thereby gaining additional time to implement other prevention and mitigation measures, as they become available.

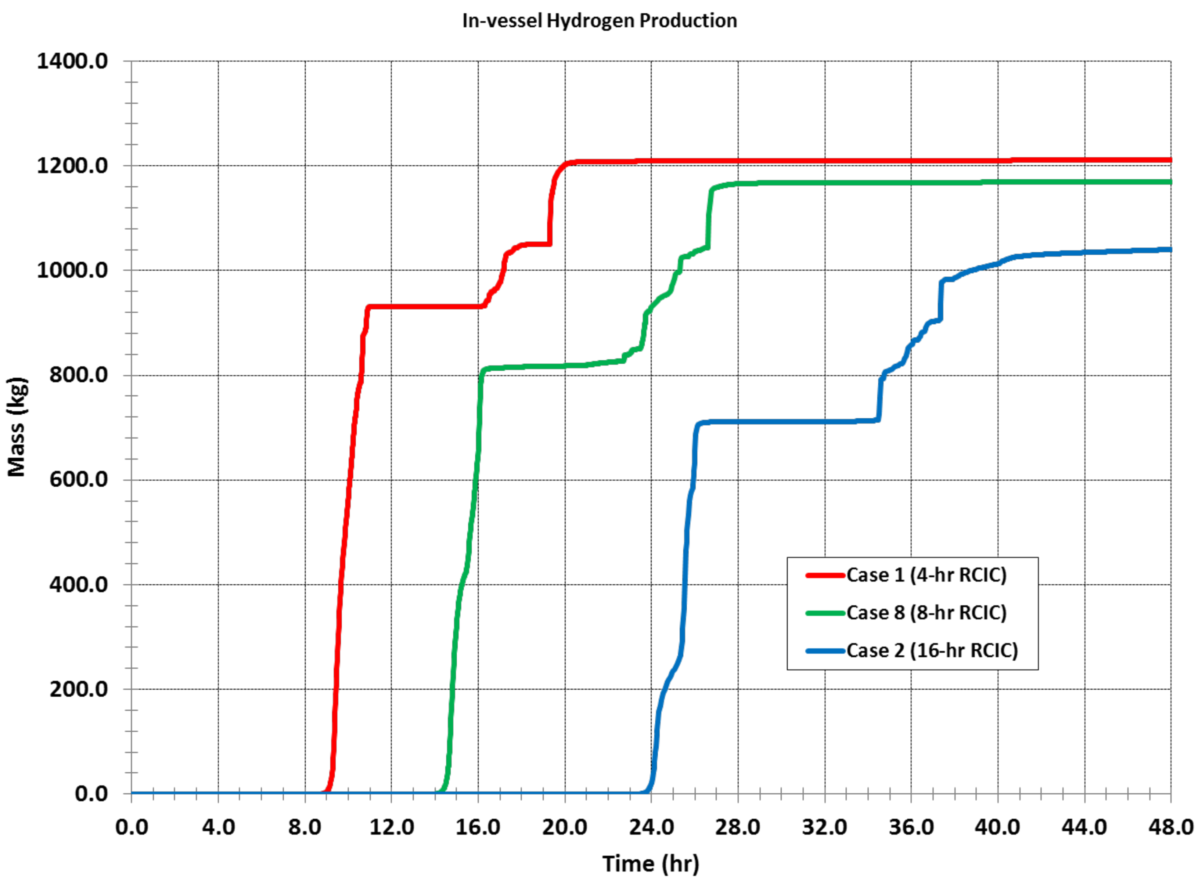


Figure 20. Comparison of in-vessel hydrogen production for various RCIC durations

The difference in hydrogen production between these cases is evident. With shorter duration RCIC (e.g., 4 hours), the core is at a higher decay power than that with a longer duration RCIC. This difference in decay power can alter the accident progression. There is a direct correlation between cladding temperature and in-vessel hydrogen production, and it is not uncommon to see a change in cladding temperature on the order of 200 to 300 degrees in these calculations. The corresponding change in hydrogen production could be on the order of 200 kg or so. Note that since none of these cases consider any mitigation measure involving water addition, they all lead to containment failure by liner melt-through—a bypass type of failure mode in which an external filter, whether at the wetwell end or at the drywell end, provides no benefit of fission product decontamination.

5.11 Effect of Spray

To illustrate the effect of spray, cesium release fractions for four cases (Case 2, Case 3, Case 6, and Case 14) were contrasted, all with 16-hour RCIC operation. Of these, Case 6 involved core spray actuation with a 300 gpm flow rate at vessel failure. Case 14 involved drywell spray actuation, also with a 300 gpm flow rate, but at 24 hours (shortly after vessel failure). Case 2 involved no venting or spray, and Case 3 involved wetwell venting but no spray. Core spray actuation at vessel failure was selected based on the consideration that the hookup of a transportable fire water system may not be feasible when the reactor vessel is at high pressure. Cesium release fractions for these cases are plotted in Figure 21 below. Needless to say, with spray action, the liner melt-through mode of failure is prevented.

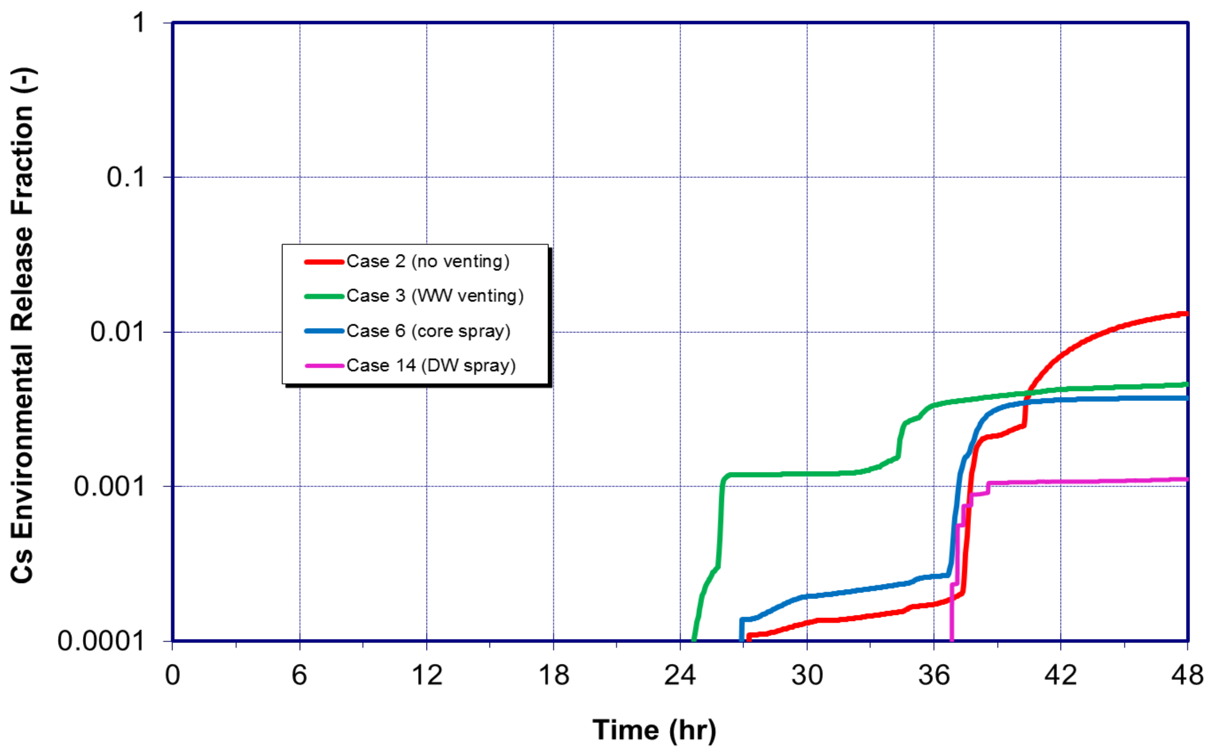


Figure 21. Comparison of cesium release fraction showing spray effect

A comparison of cesium release fractions in Figure 21 shows gradual reduction of releases from no venting case to venting to core spray and finally, to drywell spray. Venting alone or core spray alone provides about a two-thirds reduction in releases relative to the no venting case, whereas the drywell spray action provides a considerably higher reduction.

5.12 Combined Effect of Venting and Spray

Several cases were run where the venting was in effect. These cases include venting in either a passive mode (i.e., vent once opened remains open) or an active mode (i.e., vent opening and closing are cycled through operator action), in combination with spray action (core or drywell). Venting in passive mode is initiated at a drywell pressure exceeding 60 psig or about 75 psia (design pressure). Vent opening in active mode is initiated likewise at a pressure of 60 psig and closing is initiated when the drywell pressure drops to 45 psig.

Figure 22 provides a comparison of cesium release estimates for combined cases of venting and spray (Case 7 with venting and core spray and Case 15 with venting and drywell spray) contrasted to Case 2 (no venting or spray). Unless otherwise stated, the venting is always considered through wetwell. As long as the drywell is at a higher pressure, that will be the preferential vent path. If the suppression pool level is increased significantly, the wetwell vent path cannot be used, in which case fission products will be transported through drywell vent. Note in such a case (Case 12), fission products will not be scrubbed as the releases bypass the suppression pool and, as such, the releases will be much higher than those of the wetwell vent cases.

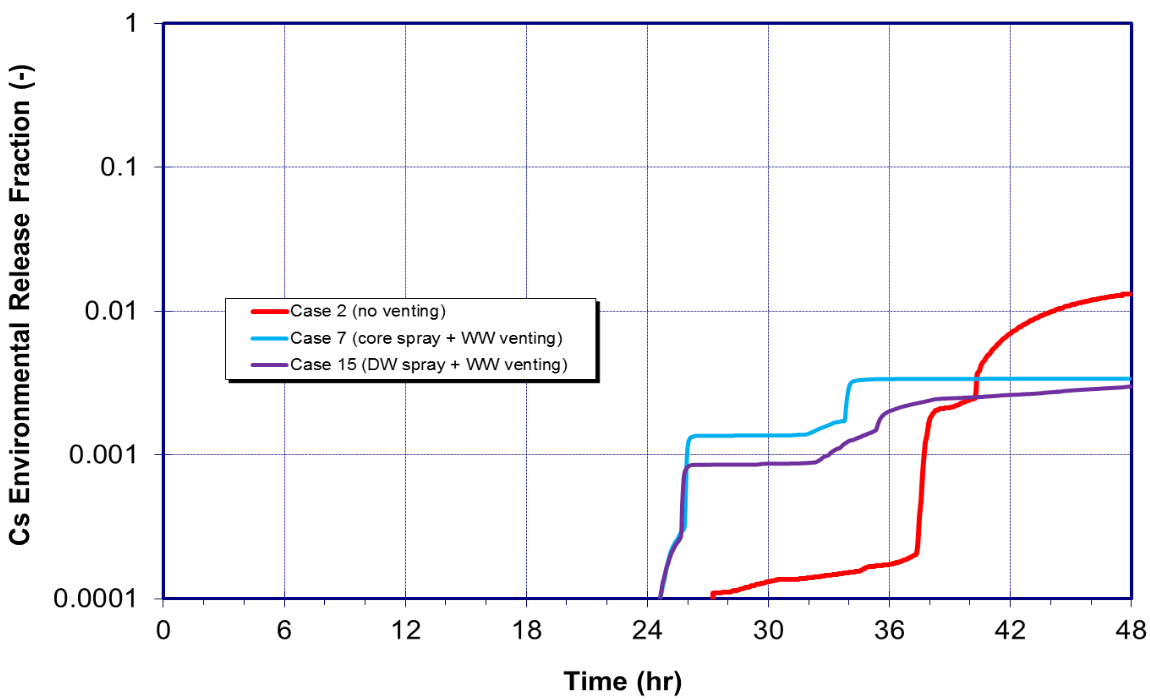


Figure 22. Cesium release fraction showing combined venting and spray effects

When compared with Figure 21 above, there appears to be modest additional reduction of releases in the cases with combined venting and spray actions. Core spray, upon actuation at or shortly after vessel failure, provides some degree of cooling of the remainder of the degraded core that may still be held up structurally inside the vessel. Moreover, the water from core spray finds its way to the drywell floor, thus effectively flooding the cavity. Because the core spray flow rate is not high, the pool created in the drywell is not expected to be deep. Also, the water will likely be saturated by the time it ends up on the drywell floor. Nevertheless, fission products will be modestly scrubbed by the flooded cavity before they are transported to the wetwell and go through suppression pool scrubbing. This appears to be the reason for getting a nominally incremental attenuation by a combination of venting and core spray actions.

The drywell spray action, likewise, provides scrubbing of the airborne fission product aerosols. When combined with venting, any additional attenuation of fission products depends on the resulting aerosol size distribution and the corresponding suppression pool scrubbing efficiency as the fission products are transported through the wetwell vent. In the particular example shown in Figure 22 (reference Case 15), there was no incremental benefit with the combined venting and spray action. Note the drywell spray was initiated after venting in Case 15 so the initial aerosol inventory did not benefit from the spray action. Generally, a significant amount of the initial fission product inventory will likely go through the wetwell prior to drywell spray actuation. This may be the reason for a slightly lower decontamination in Case 15 when compared to Case 14. In cases where venting is initiated after spray actuation, there appears to be a nominally incremental attenuation by a combination of venting and drywell spray action in contrast to spray action alone, much like the combined effect of core spray and venting.

Another grouping of selected MELCOR case runs was examined to determine the relative effect of vent cycling versus venting once and keeping it open.

Figure 23 shows the comparison between once-opened vent and vent cycling cases with and without spray action. Case 18 in this figure is no venting, Case 21 is that of once-opened wetwell vent, and Case 27 that of vent cycling—all with spray action. The MELCOR results indicate slightly smaller releases in the case of vent cycling when compared to once-opened vent cases; however, both are within the same order of magnitude. Note that EPRI's preliminary findings indicate vent cycling to be more effective in reducing fission product release relative to once-open venting. It appears that EPRI's analysis may have accounted for only the wetwell releases whereas in their model, both drywell and wetwell release paths were considered.

Note this report makes no a priori assumption regarding the implementation of vent cycling operation (i.e., feasibility of such operation, effectiveness and timeliness of operator actions in an accident situation). Even if vent cycling is demonstrated to be effective, the feasibility of its operation needs to be carefully examined.

5.13 Drywell Spray Sensitivity

The drywell spray sensitivity was explored in two different ways. First, the effect of drywell spray actuation timing was investigated in a series of three runs, all without venting. Case 18 represents drywell spray actuation time of 8 hours, Case 19 represents an actuation time of 16 hours, and Case 20 an actuation time of 24 hours. Second, the spray flow rate sensitivity was explored by varying the flow rate from 100 gpm to 1,000 gpm. For this, the 24-hour core spray actuation with 300 gpm flow rate (Case 14) was selected as the base case. Case 28 was a variation of Case 18 with 100 gpm flow rate; Case 29 a variation with 500 gpm flow rate; and finally, Case 30 a variation with 1,000 gpm flow rate.

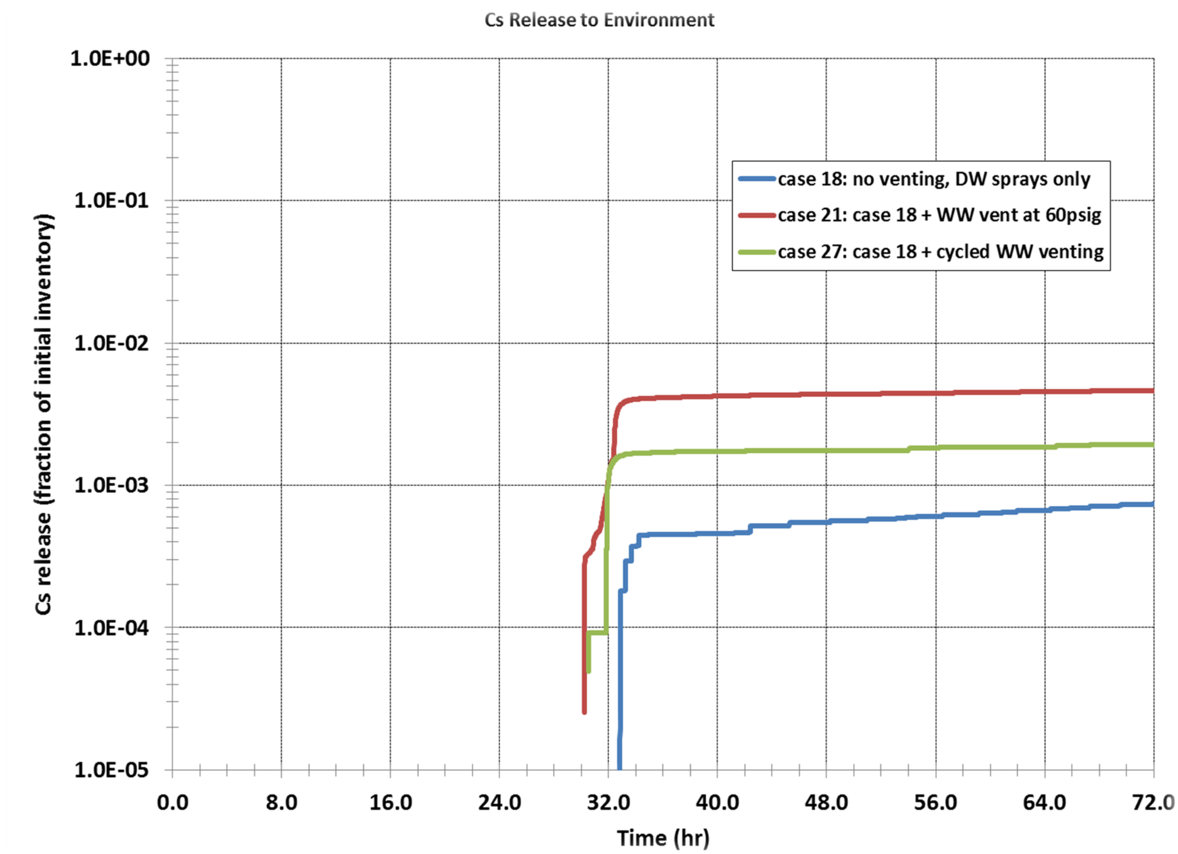


Figure 23. Comparison of cesium release factions for vent once and vent cycling cases

For the spray actuation time sensitivity, cesium release fractions are plotted in Figure 24 below. Generally, late spray actuation provides less opportunity for scrubbing as may be evident by comparing Case 19 and Case 20 results individually with Case 18 results. The release estimates, however, are within the bounds of uncertainties so it cannot be readily concluded that when the drywell spray is used as the only mitigation measure, its actuation time impact significantly the release estimates. Note the early actuation of drywell spray may have a concomitant effect of flooding the cavity and the pedestal region to the point that the wetwell venting becomes ineffective.

As mentioned earlier, the three cases considered for the spray actuation timing sensitivity do not consider venting, meaning these cases eventually lead to overpressure failure of the containment through head flange leakage. The head flange leakage model in MELCOR considers only elastic deformation of bolts, based on pressure differential. As a result, the flange opens and closes during the transient as an artifact of the model, thus limiting somewhat the releases. The timing of opening and closing of the flange is not the same in every case of drywell spray actuation, and that explains the different trend in releases between Case 19 and Case 20.

For the flow rate sensitivity analysis, cesium release fractions are plotted in Figure 25 below. Within the flow rates considered in the present analysis, there does not appear to be a large sensitivity with regard to release estimates. The 100 gpm flow rate (Case 28)—more like sprinkling flow—provides the least amount of fission products scrubbing and hence the most release, albeit still less than 1 percent at 48 hours. The higher flow rates (Case 14 with 300 gpm as the best case, Case 29 with 500 gpm, and Case 30 with 1,000 gpm) result in

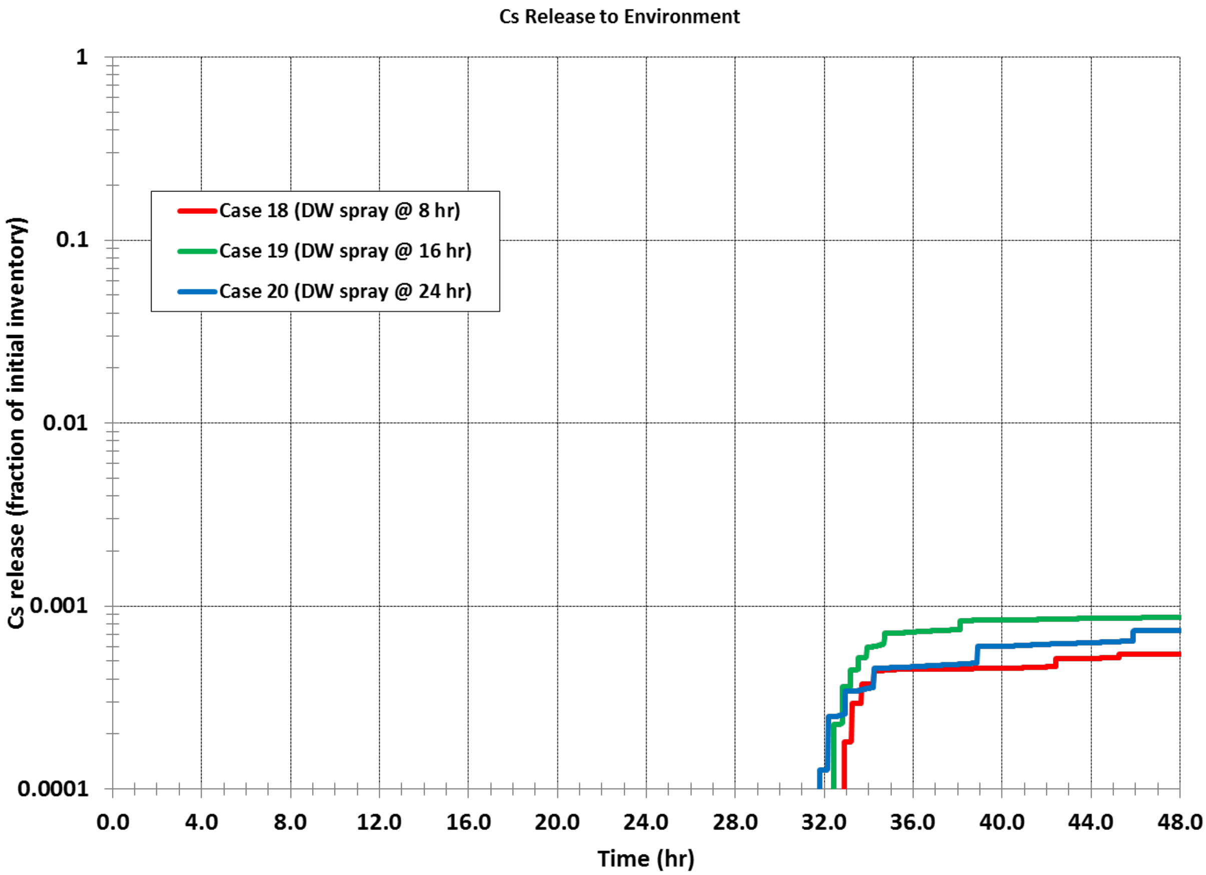


Figure 24. Effect of drywell spray actuation time on cesium release fraction

lesser release amounts (between 0.05 to 0.1 percent of initial inventory). Given the trend, a much higher flow rate is expected to provide further reduction in release amounts.

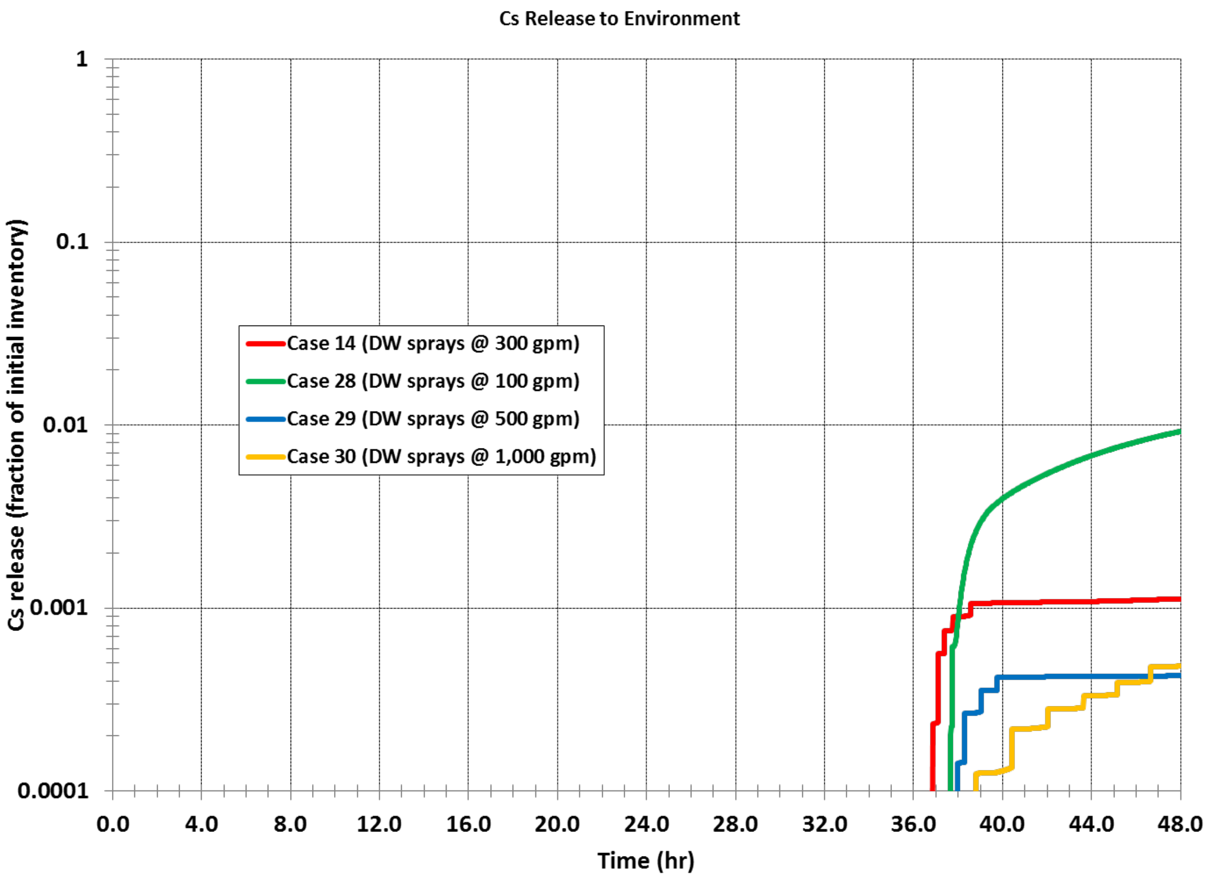


Figure 25. Effect of drywell spray flow rate on cesium release fraction

6. CONCLUSIONS FROM MELCOR ANALYSIS

The MELCOR analysis presented above, when considered in combination with the MACCS analysis in Enclosure 5b, makes a compelling technical argument for a strategy to mitigate radiological consequences of severe accidents in BWR Mark I containments that includes a combination of venting and spray action, supplemented further by the installation of an external filter. In other words, the MELCOR/MACCS analyses provide a technical basis to support Option 3 in the regulatory analysis. The external filter aspect of the technical basis is discussed further in the context of the MACCS analysis (Enclosure 5b) of health effects and property damage consequence (land contamination). The MELCOR analysis presented here leads to the following specific conclusions on containment venting and other mitigation actions.

- A combination of venting and spraying (or any mitigation action including water on the drywell floor) is required for an effective strategy for mitigating radiological releases. Venting alone or spraying alone does not provide sufficient reduction in radiological releases. The combined action results in significant reduction of fission product release. In some accident sequences, venting and provision of water, when supplemented with natural decontamination processes (e.g., fission product deposition on structural surfaces), can provide an overall decontamination factor approaching 1,000.

- An external filter is capable of providing additional fission product attenuation of already scrubbed aerosols (by spray or flooding action, or by suppression pool scrubbing). Thus, external filtering has a direct influence on reducing further the amount of fission product release to the environment, and consequent reduction in health effects and land contamination.

- Venting through the wetwell is preferred as it provides an opportunity for fission product scrubbing in the suppression pool. Pool scrubbing efficiency can be appreciable (decontamination factor in the range between 100 and 300 in the MELCOR analysis). Venting through drywell does not have pool scrubbing benefit. As such, if the drywell vent is used for the purpose, external filtration would be necessary to reduce the amount of fission product release to the environment.

- Venting prevents overpressurization failure, and excessive buildup of hydrogen and other noncondensable gases in the reactor building and other areas, thereby providing an effective means of combustible gas control. Though the hydrogen issue is not the focus of the current study, this particular insight lends further credibility to the efficacy of venting.

- MELCOR analysis, results therein, and the conclusions drawn above are consistent with the insights provided in the EPRI report [14] with one notable exception. MELCOR calculations do not show vent cycling to be any more effective than once-open venting. The release estimates in both cases are on the same order of magnitude. Preliminary EPRI calculations concluded vent cycling to be more effective. Even if vent cycling is demonstrated to be effective, the feasibility of its operation needs to be carefully examined. Note that the insights in the EPRI report recognize that an external filter can further reduce the fission product release to the environment—consistent with the conclusion from MELCOR/MACCS analysis.

- Limited sensitivity analysis carried out indicates that the release estimates are not sensitive to spray flow rate in the low flow regimes that are practically achievable in an accident situation. The estimates are also not particularly sensitive to spray actuation timing. Early actuation of drywell sprays may have a concomitant effect of flooding the cavity and the pedestal region to the point that the wetwell venting becomes no longer effective.

7. REFERENCES

[1] U.S. Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century, The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," July 12, 2011.

[2] MELCOR Computer Code Manuals, Version 1.8.6, Vol. 1: Primer and Users' Guide, NUREG/CR-6119, Vol. 1, Rev. 3, SAND 2005-5713, September 2005.

[3] D. Chanin, M. L. Young, J. Randall, "Code Manual for MACCS2: User's Guide," NUREG/CR-6613, Vol. I (1998).

[4] State-of-the-Art Reactor Consequence Analyses (SOARCA) report, NUREG-1935, January 2012.

[5] Fukushima Daiichi Accident Study (Status as of April 2012), SAND2012-6173, July 2012.

[6] Ramamurthi, M., and M. R. Kuhlman, "Final Report on Refinement of CORSOR—An Empirical In-Vessel Fission Product Release Model," Battelle Report, October 31, 1990.

[7] B. Clement, T. Haste "Comparison Report on International Standard Problem ISP-46 (FPT1)" Note Technique SEMAR 03/021, 2003.

[8] D.A. Powers, J.E. Brockmann, A.W. Shiver, VANESA: A Mechanistic Model of Radionuclide Release and Aerosol Generation during Core Debris Interactions with Concrete, NUREG/CR-4308, Sandia National Laboratories, Albuquerque, NM, July 1986.

[9] F. Gelbard, "MAEROS User Manual," NUREG/CR-1391, Sandia National Laboratories, Albuquerque, NM, December 1982.

[10] SPARC-90: A Code for Calculating Fission Product Capture in Suppression Pools, NUREG/CR-5765, October 1991.

[11] Cunane, J. C., et al., "The Scrubbing of Fission Product Aerosols in LWR Water Pools Under Severe Accident Conditions - Experimental Results," EPRI NP-4113-85, Electric Power Research Institute, Palo Alto, California, 1985.

[12] Owczarski, P. C., and W. K. Winegardner, "Validation of SPARC, A Suppression Pool Aerosol Capture Model." Paper IAEA-SM-281/29, presented at IAEA International Symposium on Source Term Evaluation for Accident Conditions, Columbus, Ohio, 1985.

[13] D. A. Powers and J. L. Sprung, "A Simplified Model of Aerosol Scrubbing by a Water Pool Overlying Core Debris Interacting With Concrete," NUREG/CR-5901, SAND92-1422, Sandia National Laboratories, Albuquerque, New Mexico, 1992.

[14] Electric Power Research Institute, "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents: BWR Mark I and Mark II Studies," 2012 Technical report, September 2012.

Enclosure 5b
MACCS Consequence Analysis

CONTENTS

MACCS CONSEQUENCE ANALYSIS

This enclosure documents the MACCS2 analysis of the same selected accident scenarios (cases) discussed in the MELCOR accident analysis enclosure. The MACCS2 consequence model (Version 2.5.0.9) was used to calculate offsite doses and land contamination, and their effect on members of the public with respect to individual prompt and latent cancer fatality risk, land contamination areas, population dose, and economic costs. This enclosure begins with a general description of MACCS, followed by the consequence analyses for the selected cases. The results are used in the regulatory cost-benefit analyses of various accident prevention and/or mitigation strategies.

1. GENERAL DESCRIPTION OF MACCS

The MELCOR code provides input to a companion code, MELCOR Accident Consequence Code System (MACCS) Version 2, or MACCS2 for short, for the analysis of radioactive material dispersion in the environment and the consequences of this dispersion. The code was specifically developed for the U.S. Nuclear Regulatory Commission (NRC) to evaluate offsite consequences from a hypothetical release of radioactive materials into the atmosphere. The code is used as a tool to assess the risk and consequences associated with accidental releases of radioactive material into the atmosphere in probabilistic risk assessment (PRA) studies. The code models atmospheric transport and dispersion, emergency response actions, exposure pathways, health effects, and economic costs.

MACCS2 is used by U.S. nuclear power plant license renewal applicants to support the plant-specific evaluation of severe accident mitigation alternatives (SAMAs) that may be required as part of the applicant's environmental report for license renewal. MACCS2 is also routinely used in severe accident mitigation design alternative (SAMDA), or severe accident consequences analyses for environmental impact statements (EISs) supporting design certification, early site permit, and combined construction and operating license reviews for new reactors. The NRC's regulatory analysis guidelines in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" [1], and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook" [2], recommend the use of MACCS2 to estimate the averted "offsite property damage" cost (benefit) and the averted offsite dose cost elements. The information from MACCS2 code runs supports a cost-benefit assessment for various potential plant improvements as part of SAMAs or SAMDAs.

MACCS2 estimates consequences in four steps:

(1) atmospheric transport and deposition onto land and water bodies

(2) the estimated exposures and health effects for up to 7 days following the beginning of release (early phase)

(3) the estimated exposures and health effects during an intermediate time period of up to 1 year (intermediate phase)

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

The assessment of offsite property damage in terms of contaminated land and economic consequences uses all four parts of the modeling. An overview of the code is provided below to explain the assessment of health effects and offsite property damage in MACCS2.

1.1 Atmospheric Transport and Dispersion (ATD) Model

MACCS2 models dispersion of radioactive materials released into the atmosphere using the straight-line Gaussian plume model with provisions for meander and surface roughness effects. The ATD model treats the following: plume rise resulting from the sensible heat content (i.e., buoyancy), initial plume size caused by building wake effects, release of up to 200 plume segments, dispersion under statistically representative meteorological conditions, deposition under dry and wet (precipitation) conditions, and decay and ingrowths of up to 150 radionuclides and a maximum of six generations. The model does not treat in detail irregular terrain, spatial variations in the wind-field, and temporal variations in wind direction.

The user has the option to select meteorological sampling, such as a single weather sequence or multiple weather sequences. The latter of these weather sampling options is used in PRA studies to evaluate the effect of weather conditions at the time of the hypothetical accident.

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

1.2 Early Phase (Emergency Phase) Model

The early-phase model in MACCS2 assesses the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase and it can extend up to 7 days after the arrival of the first plume at any downwind spatial interval. Early exposures in this phase account for emergency planning (i.e., sheltering, evacuation, and relocation of the population). The early-phase modeling in MACCS2 is limited to 7 days from the beginning of release. MACCS2 models sheltering and evacuation actions within the emergency planning zone (EPZ). Different shielding factors for exposure to cloudshine, groundshine, inhalation, and deposition on the skin are associated with three types of activities: normal activity, sheltering, and evacuation.

Outside the sheltering/evacuation zone, dose dependent relocation actions may take place during the emergency phase. That is, if individuals at a specific location are projected to exceed either of two dose thresholds (i.e., the hotspot relocation (5 rem in 12 hours) and normal relocation (0.5 rem in 24 hours) MACCS2 inputs) over the duration of the emergency phase, they are relocated at a specified time after plume arrival.

For a radioactive release containing radioiodine, some of the iodine may be absorbed by the thyroid. As a consequence, the chance of thyroid cancer to the individual may be increased. Potassium iodide (KI) can saturate the thyroid with iodine and thereby reduce the amount of radioiodine that can be absorbed. KI is distributed near some nuclear power plants. MACCS2/WinMACCS has implemented a KI model to account for the beneficial effect of taking KI. This model accounts for the fraction of the population taking KI and the efficacy, or dose reduction, provided by the KI.

1.3 Intermediate Phase Model

MACCS2 can model an intermediate phase with duration of up to 1 year following the early phase. The only mitigative action modeled in this phase is relocation. That is, if the projected dose leads to doses in excess of a threshold, the population is assumed to be relocated to an uncontaminated area for the entire duration of this phase, with a corresponding per-capita economic cost defined by the user. The intermediate phase duration can be modeled as being zero (i.e., no intermediate phase). If the projected dose does not reach the user-specified threshold, exposure pathways for groundshine and inhalation of resuspended material are treated.

1.4 Long-Term Phase Model

In the long-term phase (e.g., 50 years of potential exposure), protective actions are defined to minimize the dose to an individual by external (e.g., groundshine) and internal (e.g., food consumption and resuspension inhalation) pathways. Decisions on mitigative actions are based on two sets of independent actions (i.e., decisions relating to whether land, at a specific location and time, is suitable for human habitation (habitability) or agriculture production (farmability)). Habitability is defined by a maximum dose and an exposure period to receive that dose. Habitability decisionmaking can result in four possible outcomes:

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

Land is also condemned if the cost of decontamination exceeds the value of the land. The dose criterion for the MACCS2 modeling of individuals returning back to the affected (i.e., contaminated) area is a user input and is typically from the U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAGs)[2]. The decision on whether land is suitable for farming is first based on prior evaluation of its suitability for human habitation.

1.4.1 Decontamination Model

Decisions on decontamination are made using a decision tree. The first decision is whether land is habitable. If it is, then no further actions are needed. The population returns to their homes and receives a small dose from any deposited radionuclides for the entire long-term phase. If land is not habitable, the first option considered is to decontaminate at the lowest level of dose reduction, which is also the cheapest to implement. If this level is sufficient to restore the land to habitability, then it is performed. Following the decontamination, the population returns to their homes and receives a small dose based on the residual contamination for the duration of the long-term phase. If the first level of decontamination is insufficient to restore habitability, then successively higher levels are considered. MACCS2 considers up to three decontamination levels. If the highest level of decontamination is insufficient, then interdiction for up to 30 years is considered following the decontamination. During the interdiction period, radioactive decay and weathering work to reduce the dose rates that would be received by the returning population. If the highest level of decontamination followed by interdiction is sufficient to restore habitability, then it is employed and the population is allowed to return. Doses are accrued for the duration of the long-term phase. If habitability cannot be restored by any of these actions, then the land is condemned. The land is also condemned if the cost of the required action to restore habitability is greater than the value of property.

The decision tree for farmability is similar, but the decision on whether land is suitable for farming is first based on prior evaluation of its suitability for human habitation. That is, land cannot be used for agriculture unless it is habitable. Furthermore, farmland must be able to grow crops or produce dairy products that meet the U.S. Food and Drug Administration (FDA) requirements (i.e., it must be farmable). If farmland is habitable and farmable, a food chain model is used to determine doses that would result from consuming the food grown or produced on this land. The COMIDA2 food chain model is the latest model developed for use in MACCS2. COMIDA2 represents a significant improvement over the older food-chain model

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

[2] EPA developed the PAG Manual to provide guidance to State and local authorities on actions to help protect the public during emergencies. The manual can be found at http://www.epa.gov/rpdweb00/rert/pags.html.

embodied in the original MACCS code and used in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants."

MACCS2 values of total long-term population dose and health effects account for exposures received by workers performing decontamination. While engaged in cleanup efforts, workers are assumed to wear respiratory protection devices; therefore, they only accumulate doses from groundshine.

1.4.2 Land Contamination Areas

Land areas contaminated above a threshold level can be calculated in several ways. The simplest is to report land areas that exceed activity levels per unit area for one or more isotopes. This is the approach used to report contaminated areas following the Chernobyl accident (i.e., land areas exceeding threshold levels of Cs-137 activity were reported). Currently, MACCS2 estimates such areas based on the Gaussian plume segment model for atmospheric transport and deposition.

1.5 MACCS2 Economic Consequence Model

The economic consequence model in MACCS2 includes costs associated with various actions or modeling within six categories as follows:

(1) Evacuation and relocation costs (e.g., a per diem cost associated with displaced individuals). The per-diem costs are associated with the population that is temporarily relocated. These costs are calculated by adding up the number of displaced people times the number of days they are displaced from their homes.

(2) Moving expenses for people displaced (i.e., a one-time expense for moving people out of a contaminated region). There is a one-time moving expense for the population displaced from their homes because of decontamination, interdiction, or condemnation. The modeling can include loss of wages.

(3) Decontamination costs (e.g., labor, materials, equipment, and disposal of contaminants). These are the costs associated with decontaminating property. These costs include labor and materials for performing the decontamination. They depend on the population and size of the area that needs to be decontaminated as well as the level of decontamination that needs to be performed. They can include the cost to dispose of contaminated material. The model estimates the costs only if decontamination is cost effective.

(4) Cost due to loss of land use of property (e.g., costs associated with lost return on investment and for depreciation of property that is not being maintained). These costs are associated with loss of use of property. These costs include an expected rate of return on property and depreciation caused by lack of routine maintenance during the period of interdiction, the time when the property cannot be used.

(5) Disposal of contaminated food grown locally (e.g., crops, vegetables, milk, dairy products, and meat).

(6) Cost of condemned lands (i.e., land that cannot be restored to usefulness or is not cost effect to do so). These are costs of condemning property that cannot be restored to meet the habitability criterion.

All of the costs for the six cost categories are summed over the entire offsite area affected by the assumed atmospheric release to get the total offsite economic costs. Nearly all of the values affecting the economic cost model are user inputs and thus can account for a variety of costs and can be adjusted for inflation, new technology, or changes in policy. Also, the isotopic composition of the source term significantly impacts the costs that would be needed to decontaminate. Some isotopes require no decontamination at all while others might require extensive decontamination. Thus applying a decontamination factor (DF) to the particulate source term release fraction will not result in a linear extrapolation of the results.

1.6 Recent Improvements to the MACCS2 Code

The MACCS2 code has gone through improvements since its original release in 1997. Version 2.5 of the code has been released recently together with the graphical user interface (GUI), WinMACCS Version 3.6 [1]. The three most important modeling features implemented in WinMACCS are:

(1) the ability to easily evaluate the impact of parameter uncertainty

(2) the ability to manipulate input parameters for network evacuation modeling

(3) the ability to model alternative dose-response relationships for latent cancer fatality evaluation (e.g., linear with threshold model)

Uncertainty in the source term and in most of the other MACCS2 input parameters, including parameters related to emergency response, can be treated through WinMACCS.

2. CONSEQUENCE ANALYSES

The MACCS2 consequence model (Version 2.5.0.9) was used to calculate offsite doses and land contamination, and their effect on members of the public with respect to fatality risk, land contamination areas, population dose, and economic costs for the cases considered in this study. Updates to the SOARCA version of the MACCS2 code (Version 2.5.0.0) used for offsite consequence predictions are discussed in NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report: Draft Report for Comment," Section 5 [4]. The following are the MACCS2 code version updates from SOARCA to this study:

- Provide file locations on MACCS2 cyclical files (e.g., MELMACCS source term files) to provide enhanced traceability between inputs and results. This update did not affect the results.

- A lower plume density limit (PLMDEN) consistent with the MACCS2 User Manual [4]. This update did not affect the results. It only allowed calculations to be performed over a wider range of input parameters.

- Change to a FORTRAN compiler compatible with the Windows 7 operating system. This change did create minor differences (i.e., less than 10 percent). The new compiler uses a different representation for real numbers. Slight changes in the real values affect the rounding of these values to create integer values, which in turn affect the random values that are calculated; particularly the set of weather trials that are selected. This difference is considered acceptable and not an error because there is no reason to think that one set of random choices is better than the others.

- Correction of the NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," plume meander model [6]. This correction did not have any impact on the SOARCA results or this study's results because neither of these analyses used this model.

The principal phenomena considered in MACCS2 are atmospheric transport using a straight-line Gaussian plume segment model of short-term and long-term dose accumulation through several pathways including cloudshine, groundshine, inhalation, deposition onto the skin, and food and water ingestion. The ingestion pathway model was used in these analyses. The following dose pathways are included in the reported latent cancer fatality (LCF) risk metrics:

- Cloudshine during plume passage.

- Groundshine during the emergency and long-term phases from deposited aerosols.

- Inhalation during plume passage and following plume passage from resuspension of deposited aerosols. Resuspension is treated during both the emergency and long-term phases.

MACCS2 does not include ingestion of contaminated food or water in the LCF risk calculation. However, the ingestion pathway is included in the population dose calculation.

Another risk metric considered in this study is prompt fatality risk. The NRC quantitative health object (QHO) for prompt fatalities ($5x10^{-7}$ per reactor year (pry)) is generally interpreted as the absolute risk within 1 mile of the exclusion area boundary (EAB). For Peach Bottom, the EAB is 0.5 mile from the reactor building from which release occurs, so the outer boundary of this 1-mile zone is at 1.5 miles. The closest MACCS2 grid boundary to 1.5 miles used in this set of calculations is at 1.3 miles. Evaluating the risk within 1.3 miles should reasonably approximate the risk within 1 mile of the EAB.

Prompt fatality risk is based on doses large enough to exceed the dose thresholds for early fatalities for the 0.5 percent of the population that are modeled as refusing to evacuate. The red bone marrow is usually the most sensitive organ for prompt fatalities. The minimum acute exposure that can cause a prompt fatality is about 2.3 gray (Gy) (1 Gy = 100 rad) to the red bone marrow. Additional acute exposure thresholds are also considered for the lungs (13.6 Gy) and the stomach (6.5 Gy). None of the cases considered for this study exceeded the lung and stomach acute exposure thresholds.

This work uses the Peach Bottom unmitigated long-term station blackout (LTSBO) MACCS2 input deck from the SOARCA project as a starting point (the Peach Bottom SOARCA analysis is documented in NUREG/CR-7110, Volume 1, "State-of-the-Art Reactor Consequence Analyses Project—Volume 1: Peach Bottom Integrated Analysis" [7]). One basic change is that the ingestion pathway was modeled in this study, but was excluded in the SOARCA analyses. The only other changes were to use the modified source terms, as calculated from the MELCOR analyses for this study to account for variation in the LTSBO scenario, and the effect of adding an external filter to the vent paths. None of the source terms considered in this study are the same as the LTSBO source term used in SOARCA. This difference in source term is in part due to the difference in DC station battery duration (i.e., 4 hours for SOARCA and 16 hours for this study). However, additional mitigative actions discussed in Section 2.1, also contribute to the differences in the source term.

As part of SOARCA, a number of code enhancements were made to MACCS2 [7]. In general, these enhancements implemented some of the recommendations obtained during the SOARCA external peer review and needs identified by the broader consequence analysis community [8]. The code enhancements implemented for SOARCA were primarily to improve realism and code performance and to enhance existing functionality.

Many of the user-specified modeling practices used for consequence analysis in SOARCA are different than previous studies. SOARCA applied the most current weather sampling and updated modeling techniques, and multiple alternate dose-response options to create a more detailed, integrated, and realistic analysis than past consequence analyses. In this study, only the linear-no-threshold dose-response model is used, while SOARCA reported additional results for two linear-*with*-threshold dose-response models as well. Some of the MACCS2 enhancements used in SOARCA and this study included increased angular resolution, updated dose conversion factors, and a larger number of evacuation cohorts

Studies prior to the SOARCA analyses used 16 compass directions. For SOARCA, 64 compass directions were used [7], and are maintained for this study.

MACCS2 analyses prior to SOARCA used dose conversion factors based on the International Commission on Radiological Protection (ICRP) publications ICRP 26 [9] and ICRP 30 [10]. The SOARCA project used dose conversion factors from Federal Guidance Report 13 [11], which are also used in this study.

MACCS2 previously allowed up to three emergency-phase cohorts. A cohort is a population group that mobilizes or moves differently from other population groups. Each emergency-phase cohort represents a fraction of the population who behave in a similar manner, although MACCS2 allows response times to be a function of radius, so there can be some limited variation within a single cohort. As an example, a cohort might represent a fraction of the population who rapidly evacuate after officials instruct them to do so. To treat public response more realistically, the number of emergency phase cohorts allowed in MACCS2 was increased to 20. This allows significantly more variations in emergency response (e.g., variations in preparation time before evacuation) to more accurately reflect the movement of the public during an emergency. In a similar way, modeling evacuation routes using the network-evacuation model in MACCS2 adds more realism than had been employed in previous studies.

The population near the Peach Bottom plant was modeled in SOARCA using six cohorts [7], and this approach was maintained in this study. Cohorts were established to represent members of the public who may evacuate early, evacuate late, those who refuse to evacuate, and those who evacuate from areas not under an evacuation order (e.g., the shadow evacuation). The following cohorts were used for these analyses:

Cohort 1: 0 to 10 Public. This cohort includes the public residing within the emergency planning zone (EPZ) which is the radial area within 10 miles of the plant.

Cohort 2: 10 to 20 Shadow. This cohort includes the shadow evacuation from the 10-mile to 20-mile area beyond the EPZ.

Cohort 3: 0 to 10 Schools and 0 to 10 Shadow. This cohort includes elementary, middle, and high school student populations within the EPZ. A shadow evacuation from within the EPZ is included that is assumed to mobilize at the same time as the schools. Both the evacuation of the schools and the shadow evacuation are triggered by the sounding of sirens indicating a site area emergency (SAE).

Cohort 4: 0 to 10 Special Facilities. The special facilities population includes residents of hospitals, nursing homes, assisted-living communities, and prisons. Special facility residents are assumed to reside in robust facilities such as hospitals, nursing homes, or similar structures that provide additional shielding. Shielding factors for this population group consider this fact.

Cohort 5: 0 to 10 Tail. The 0 to 10 tail is defined as the last 10 percent of the public to evacuate from the 10-mile EPZ.

Cohort 6: Non-Evacuating Public. This cohort represents a portion of the public from 0 to 10 miles who are assumed to refuse to evacuate. In this study, this cohort is assumed to be 0.5 percent of the population and they are modeled as though they continuing to perform normal activities.

In the SOARCA analyses, SECPOP2000 [12] was used to estimate the population within 50 miles of each plant. The population for each site was projected from 2000 to 2005 using a national population growth multiplier of 1.0533 obtained from the Census Bureau. SECPOP2000 interpolates U.S. census data at the block level onto a MACCS2 grid. SECPOP2000 also interpolates U.S. land-use and economic data at the county level onto a MACCS2 grid. The economic values used in SOARCA are from the Bureau of Economic Analysis (BEA) for the year 2002. These values were scaled to 2005 dollars by applying a multiplier of 1.0900, which is the ratio of the consumer price index for 2005 to the value for 2002. The MACCS2 model used in this study uses the same Peach Bottom site-specific files for population data and economic data based on the year 2005.

2.1 Consequence Analyses Overview

The results of the consequence analyses are presented in terms of risks to the public, population dose, land contamination, and economic costs for each of the cases. All consequence results are presented as conditional consequences (i.e., assuming that the accident occurs), and show the risks to individuals as a result of the accident (i.e., LCF risk per event or prompt-fatality risk per event).

The risk metrics are LCF risk and prompt fatality risk to residents in circular regions surrounding the plant. The risks, population dose, and economic costs are mean values (i.e., expectation values) over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular region. The land contamination areas are total areas of land exceeding a certain threshold of Cesium areal concentration (and unlike the other consequence metrics, is not limited to the 50-mile circular region). The risk values represent the predicted number of fatalities divided by the population. LCF risks are calculated for a linear no-threshold (LNT) dose-response model. These risk, population dose, and economic cost metrics account for the distribution of the population, land use, and property within the circular region and for the interplay between these distributions and the wind rose probabilities.

Table 1 provides a brief description for each MELCOR scenario used in the regulatory analysis (i.e., Case 2, Case 3, Case 6, Case 7, Case 12, Case 13, Case 14, and Case 15).

Table 1 Matrix of MELCOR Scenarios Used in the Consequence Analyses

Case	DC Battery time (16 hours)	Core spray after RPV failure	Drywell spray at 24 hours	Wetwell venting at 60 psig	Main steam line (MSL) failure	Drywell venting at 24 hours
2	X					
3	X			X		
6	X	X				
7	X	X		X		
12	X				X	X
13	X		X		X	X
14	X		X			
15	X		X	X		

For ease of discussion, four groups were constructed to compare the effect of venting and additional mitigative actions (e.g., core spray and drywell spray). The MELCOR cases were grouped as follows:

- Base case—Case 2 and Case 3
- Core spray—Case 6 and Case 7
- Drywell venting with MSL failure—Case 12 and Case 13
- Drywell spray—Case 14 and Case 15

A discussion of health effect risks (Section 2.2 through Section 2.5), land contamination (Section 2.2 through Section 2.5), population dose (Section 2.6), and economic costs (Section 2.7) is provided for each group of cases.

A 48-hour truncation time was assumed for this work. This is the same truncation time used in SOARCA [7]. The 48-hour truncation time for SOARCA was based on the many resources available at the State, regional, and national level that would be available to mitigate a severe reactor accident. For the SOARCA project, the staff reviewed available resources and emergency plans and determined that adequate mitigation measures (i.e., at minimum, the ability to flood the reactor building) could be brought on site within 24 hours and connected and functioning within 48 hours. The decision to truncate releases at 48 hours was made well before the Fukushima Daiichi accident[3]. Based on the assumptions made for SOARCA, the releases that would occur within 48 hours for the Peach Bottom unmitigated LTSBO scenario cease because of reactor building flooding. Note that past studies, including PRAs such as NUREG-1150, typically truncated releases after 24 hours.

For this work, neither MELCOR nor MACCS2 were used to mechanistically model the decontamination effect of an external filter for the wetwell or drywell vent path. Instead, a prescribed DF value is assigned to represent the external filter. This DF is applied to the portion of the environmental source term released that would flow through the filtered vent and is not a noble gas. The DF is applied uniformly to all of the aerosol sizes and is assumed to be time independent. A more realistic approach would account for the DF for each aerosol size bin and possibly account for the effect of temperature and radionuclide concentration in the external filtration system.

The relationship between the DF value and the reduction in environmental consequence (e.g., land contamination) is nonlinear. A DF of 10 does not usually translate to a 10-fold reduction in consequence. Some of the results presented in this study are inherently nonlinear. Land contamination area is a good example because this includes thresholds for which values are only tabulated when the threshold is exceeded. Depending on the accident sequence under consideration and the consequence metric being evaluated, the effect of a DF can be modest to significant.

For the calculations presented in this study, a minimum DF value of 2 was considered for the wetwell external filter. The external filter DF is considered in addition to any type of DF that occurs from the scrubbing effects within the wetwell. In the filtered cases analyzed for this study (e.g., Case 15), part of the source term is from fission products in the water flowing from the drywell through the containment downcomers and into the wetwell. This path bypasses the

[3] For Fukushima, the operators delayed releases beyond the SOARCA assumption, so substantial releases occurred beyond 48 hours. In addition, the operators at Fukushima were not able to flood the reactor buildings, as assumed for SOARCA. For mitigated cases, the SOARCA analysis assumed the effectiveness of mitigation measures well within 48 hours. This assumption is considered reasonable, given the vast network of resources available in the United States. These resources include an offsite emergency operations facility, which would provide access to fleetwide emergency response personnel and equipment, including the 10 CFR 50.54(hh) mitigation measures and equipment from sister plants. These assets, as well as those from neighboring utilities and State preparedness programs, could be brought to bear on the accident if needed. In addition, SOARCA did not assume a tsunami, and such an event is considered highly unlikely at Peach Bottom. If sites were subject to tsunamis, these events could affect the availability and effectiveness of mitigation measures. In response to the recommendation of the NRC's Near-Term Task Force report, SECY-11-0093, dated July 12, 2011, the NRC is currently evaluating changes to mitigation strategies.

T-quenchers during wetwell venting. When the T-quenchers are bypassed, a lower DF occurs for the wetwell. The wetwell DF is typically considered to be an order of magnitude higher when the T-quenchers are not bypassed. The reduced DF in the wetwell will cause more of the radionuclides to be scrubbed in the external filters and thus increase the DF for the external filters. With this in mind, the environmental consequences reported for a DF value of 2 for the external filters should be viewed with reservation. Additional MACCS2 calculations were carried out for all wetwell venting cases included in this study with DF values of 10 and 100. The results show a reduction of consequences for the filtered cases.

For the calculations presented in this study, a minimum DF value of 1,000 was considered for the drywell external filter. Since there are no scrubbing effects from the wetwell for drywell venting, the external filter is considered to be 99.9 percent efficient. As a sensitivity study, a DF of 5,000 was applied to Case 12 (i.e., external filter is 99.98 percent efficient) to determine the effect of an increased efficiency.

In terms of the type of long-term radiation that would be emitted, the most important isotope is cesium-137 (Cs-137). Cs-137 decays to Ba-137m, which rapidly decays and emits gamma radiation. Most of the resulting doses are from groundshine; resuspension inhalation and ingestion of cesium are relatively unimportant because cesium is rapidly excreted from the body, and so these pathways do not lead to large doses. Groundshine from deposited cesium continues until the land has been decontaminated or the cesium has decayed.

The noble gases, primarily xenon and krypton, are responsible for a significant amount of the released radioactivity that results from a severe accident. However, these gases do not deposit and do not contribute significantly to doses to humans because they are very inert (i.e., they are nonreactive and do not absorb onto surfaces). Since the noble gases do not absorb onto the surfaces of the lungs and are thus quickly exhaled, they insignificantly contribute to the inhalation dose. As a result of these attributes, the noble gases contribute little to health risk.

2.2 Base Cases

Table 2 provides a brief description of source terms for the Peach Bottom accident scenarios analyzed for Case 2 and Case 3. The filtered cases include an applied DF of 2, 10, and 100 for the wetwell vent path. When a DF is applied to the pathway for flow through the filtered vent (i.e., Case 3—wetwell vent left open), the relationship is nonlinear between the inverse of DF and the source term. The reason is that for the filtered cases, the wetwell vent path is not the only release pathway to the environment. At 36.5 hours, the containment fails due to core melt through of the drywell liner. The drywell liner failure provides a lower resistance pathway to the environment than through the wetwell vent. Unlike drywell head flange leakage, the flow path opened by melt-through of the drywell liner can never be reclosed. The drywell line failure is a permanent leak path out of the containment to the environment without any benefit of wetwell pool scrubbing associated with the wetwell vent.

Table 2. **Brief Source Term Description for MELCOR Scenarios Discussed in the Base Cases Consequence Analyses**

Scenario	Integral Release Fractions by Chemical Group									Atmospheric Release Timing	
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Case 2 Base case	0.77	0.013	0.0014	0.019	0.016	0	0.003	0	0	25.7	48
Case 3 Base case with wetwell venting	1.00	0.0046	0.0081	0.028	0.033	0	0.0004	0.0002	0	23.9	48
Case 3 DF=2	1.00	0.0029	0.0047	0.017	0.022	0	0.0003	0.0001	0	23.9	48
Case 3 DF=10	1.00	0.0015	0.0020	0.0077	0.013	0	0.0002	0.00002	0	23.9	48
Case 3 DF=100	1.00	0.0011	0.0014	0.0057	0.011	0	0.0002	0.000002	0	23.9	48

2.2.1 Base Cases—Latent Cancer Fatality and Prompt Fatality Risk

Exposure of the public to a radioactive release and the risk associated with that exposure can be analyzed with MACCS2. One of the risk metrics used in these analyses is LCF risk for residents in circular regions surrounding the plant. The risks are averaged over the entire residential population within the circular region, and represent the calculated number of fatalities for all dose pathways, except ingestion, divided by the population. The LCF risk metric accounts for the distribution of the population within the circular region and for the relationship between the population distribution and the wind rose probabilities, as well as other meteorological characteristics. LCF risk results are presented for the linear no-threshold (LNT) dose-response model.

Table 3 shows the individual, mean LCF risks per event for residents within a circular area at specified radial distances for Case 2 and Case 3. All the vented cases (Case 3) result in smaller risks[4] than the base case, Case 2. The addition of a filter to the vented cases results in an additional reduction in risk. As seen in Table 3, when a DF is applied to the pathway that flows through the filtered vent (i.e., Case 3—wetwell vent left open), the relationship is nonlinear between the inverse of DF and LCF risk, for the reasons described below.

As discussed above for the filtered case, the wetwell vent path is not the only release pathway to the environment. As a result of the additional environmental release pathway (i.e., the drywell liner failure), the relationship between the assumed DF and the LCF risk is sublinear. In addition, the sublinear behavior is more pronounced at shorter distances, primarily due to short-term and long-term mitigative actions. For smaller releases, less offsite protective actions are needed and employed. Thus, doses and LCF risks diminish less than linearly. The offsite protective actions implemented in the MACCS2 model that are responsible for these trends are relocation during the emergency phase and enforcement of the habitability criterion during the long-term phase.

[4] This is despite the fact that the release fractions for some chemical groups are higher in Case 3 compared to Case 2. The LCF risk is dominated by Cesium isotopes (as discussed in Section 2.1), whose release fractions are higher in Case 2 compared to Case 3.

Table 3. Individual, Mean LCF risk per Event for Residents within a Circular Area at Specified Radial Distances for the Base Cases

	Case 2 Base case	**Case 3** Base case with wetwell venting	**Case 3 DF 2**	**Case 3 DF 10**	**Case 3 DF 100**
0-10 miles	1.6×10^{-4}	9.6×10^{-5}	8.0×10^{-5}	5.6×10^{-5}	4.5×10^{-5}
0-20 miles	1.2×10^{-4}	8.7×10^{-5}	6.5×10^{-5}	3.9×10^{-5}	3.1×10^{-5}
0-30 miles	8.4×10^{-5}	6.1×10^{-5}	4.4×10^{-5}	2.6×10^{-5}	2.1×10^{-5}
0-40 miles	5.7×10^{-5}	4.0×10^{-5}	2.8×10^{-5}	1.6×10^{-5}	1.3×10^{-5}
0-50 miles	4.8×10^{-5}	3.3×10^{-5}	2.3×10^{-5}	1.3×10^{-5}	1.0×10^{-5}

Figure 1 shows the individual, mean LCF risk per event using the LNT model for residents within a circular area at specified radial distances for Case 2 and Case 3. Each column is the combined (total) LCF risk from the emergency and long-term phases (i.e., the results shown in Table 3). Table 3 and Figure 1 show that the vented base case (Case 3) has a lower total LCF risk than the base case with no venting (Case 2), and all the filtered cases have a lower total LCF risk than the unfiltered cases (i.e., Case 2 or Case 3 without filter). For Case 3, assuming a DF of 100 for the external filter, the total LCF risk is reduced by 53 percent for the 10-mile radial distance to 70 percent for the 50-mile radial distances, compared to the unfiltered Case 3.

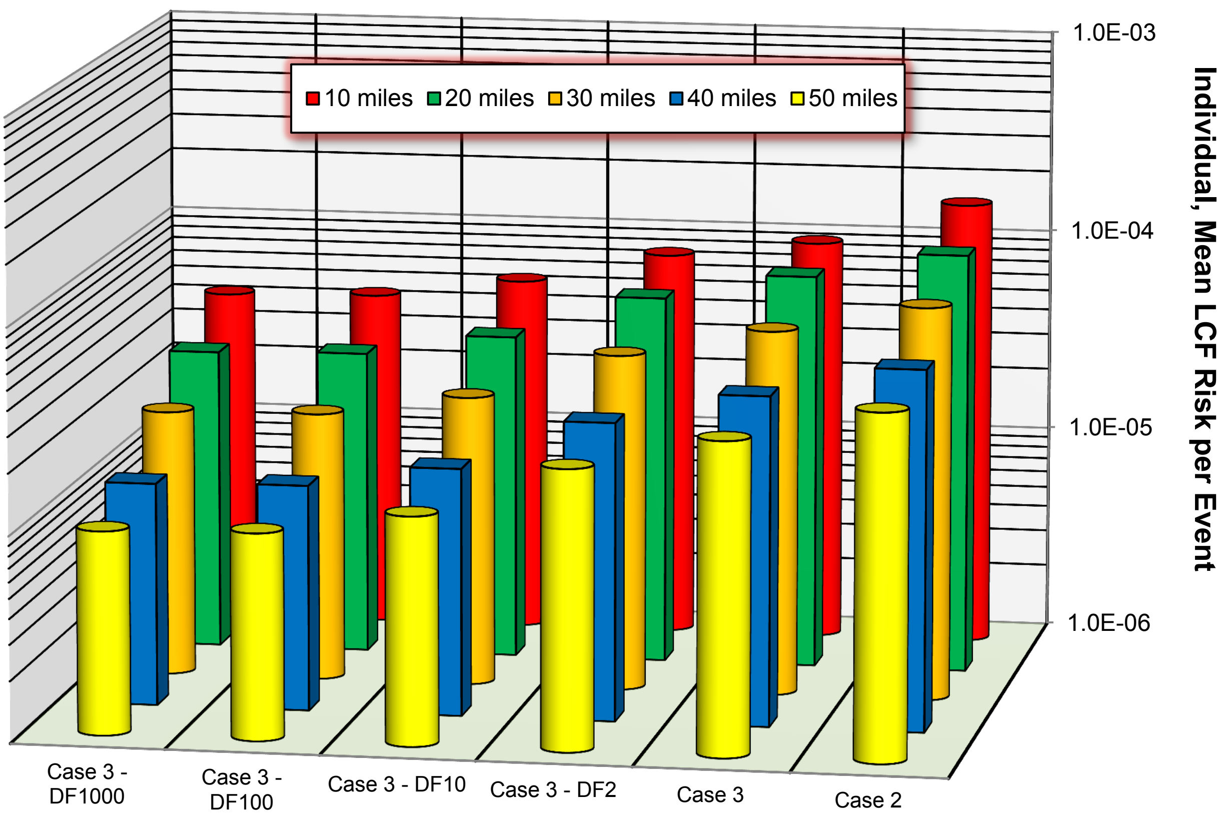


Figure 1**. Individual, mean LCF risk per event for residents within a circular area at specified radial distances for the base cases**

Figure 2 shows the individual, mean LCF risk per event using the LNT dose-response model for residents within a circular area at the specified radial distances for Case 2. The figure shows the emergency and long-term phases. The entire height of each column shows the combined (total) LCF risk for the two phases (i.e., the results shown in Table 3). The emergency response is very effective within the EPZ (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population that are modeled as refusing to evacuate. The emergency phase accounts for ~15 percent of the total LCF risk within the 50-mile radial distance.

The long-term phase risk dominates the total risks for this case with the LNT dose-response model. These long-term risks are controlled by the habitability (return) criterion, which is the dose rate at which residents are allowed to return to their homes following the emergency phase. For Peach Bottom, the State of Pennsylvania's guideline of a dose rate of 500 mrem/yr (i.e., starting in the first year) is used as the habitability criterion[5].

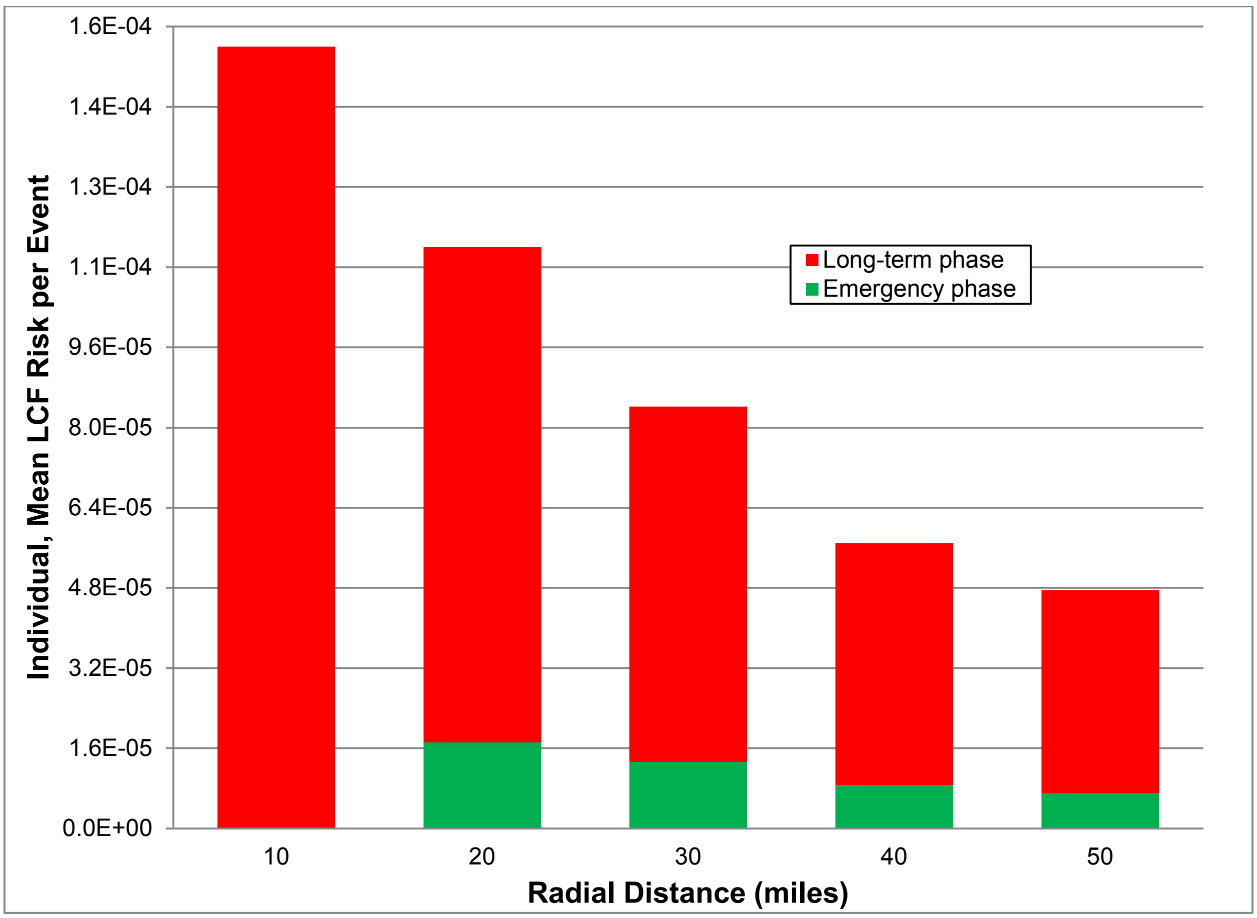


Figure 2. Case 2 individual, mean LCF risk per event for residents within a circular area at specified radial distances

[5] The U.S. Environmental Protection Agency's Protective Action Guideline is 2 rem the first year, followed by 500 mrem/year starting the second year. States can choose a more restrictive guideline.

Figure 3 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances using the LNT dose-response model for Case 3 with each of the DFs applied. Again, the emergency response is very effective within the evacuation zone (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population who are modeled as refusing to evacuate. The explanations provided for Figure 2 also apply to Figure 3. The emergency phase accounts for ~40 percent of the total LCF risk within the 50-mile radial distance for all DF values.

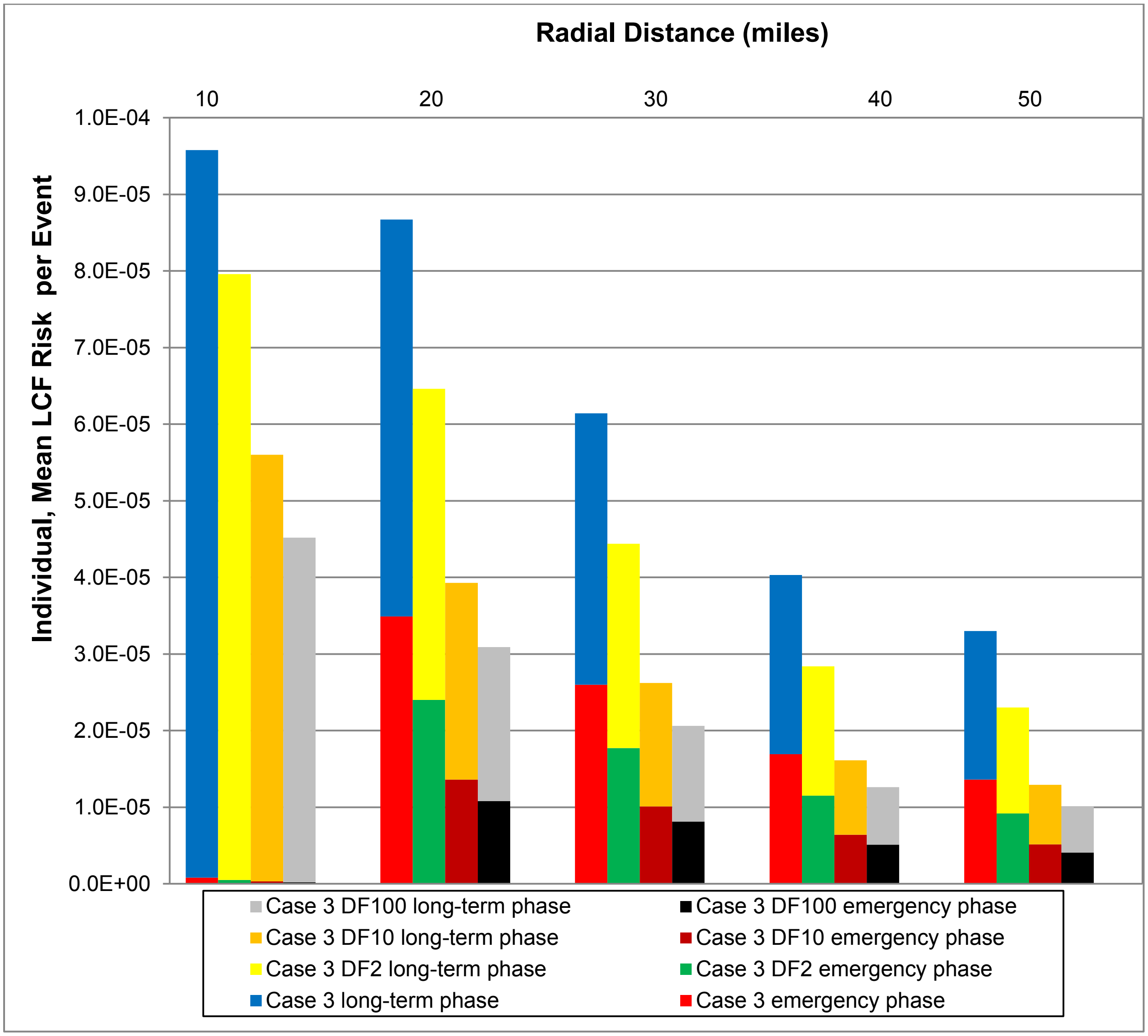


Figure 3. Case 3 individual, mean LCF risk per event for residents within a circular area at specified radial distances with specified decontamination factors

The prompt fatality risks are zero for these cases. This is because the release fractions (i.e., see in Table 2) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5 percent of the population that are modeled as refusing to evacuate. The largest value of the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers from the plant) for these cases is about 0.06 Gy to the red bone marrow. As discussed previously, the red bone marrow is usually the most sensitive organ for prompt fatalities, but the minimum acute dose that can cause an early fatality is about 2.3 Gy to the red bone marrow. As a result, the calculated exposures are all well below this threshold.

2.2.2 Base Cases—Land Contamination

Land areas contaminated above a threshold level can be calculated several ways in MACCS2, the simplest of which is to report land areas that exceed activity levels per unit area for one or more of the isotopes. This is the approach used here, and using the same threshold levels of Cs-137 as were used following the Chernobyl accident [13].

Other than the noble gases, each of the isotopes can deposit onto surfaces and cause contamination, but most of them have short half-lives and only remain in the environment for days or weeks. For example, iodine-131 has an 8-day half-life. Thus, in 80 days (i.e., 10 half-lives) its concentration is diminished to $2^{-10} \approx 0.001$ of its initial activity. As a result, it contributes to short-term doses but does not require decontamination because it disappears on its own. A relatively small number of the isotopes that could potentially be released from a nuclear reactor are radiologically important and require effort to decontaminate. Among these are Cs-134 and Cs-137, which have half-lives of 2 years and 30 years.

Cs-137 land contamination discussed by the International Atomic Energy Agency (IAEA) for the Chernobyl accident were reported at levels of 1, 5, 15, and 40 Ci/km^2, which are the same as 1, 5, 15, and 40 μCi/m^2, respectively. Based on these land contamination levels, the IAEA report was able to estimate annual effective external doses. Table 4 provides the annual effective external dose estimates based on Cs-137 soil-surface contamination[6] [13].

Table 4. Chernobyl Annual Effective External Dose Estimates for 1986 to 1995

Soil Deposition (μCi/m^2 of ^{137}Cs)	Annual Effective External Dose (rem)*									
	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995
15	0.79	0.20	0.19	0.18	0.18	0.18	0.17	0.15	0.14	0.13
5	0.25	0.06	0.06	0.06	0.06	0.06	0.05	0.05	0.04	0.04
1	0.06	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01

* 100 rem = 1 Sievert (Sv)

Table 5 provides the mean, contaminated area prior to decontamination for specified Cs-137 contamination levels for Case 2 and Case 3. There is an inherently nonlinear relationship between the size of the source term and land contamination area. This is primarily because land contamination area is calculated using a threshold (i.e., land areas are only tabulated when they exceed a threshold ground concentration). It turns out that the relationship between the inverse of DF (i.e., the quantity released) and land contamination area is superlinear.

Figure 4 shows the mean, land contamination area per event for Case 2 and Case 3. When the unvented unfiltered case (Case 2) is compared with the filtered case, a DF of 10 or 100 results in a one or two order-of-magnitude reduction in land contamination area. The filtered cases of DF of 10 or 100 result in a factor of ~5–10 reduction compared to the vented unfiltered case (Case 3).

6 Conversion of the 40 μCi/m^2 of Cs-137 soil deposition to Chernobyl annual effective external dose was not provided in the IAEA report.

Table 5. Mean, Contaminated Area per Event Above the Specified Contamination level for the Base Cases

Contamination Level ($\mu Ci/m^2$ of ^{137}Cs)	Contaminated Area (km^2)*				
	Case 2 Base case	**Case 3** Base case with wetwell venting	**Case 3 DF 2**	**Case 3 DF 10**	**Case 3 DF 100**
1	8,900	2,000	1,100	430	340
5	1,000	250	130	49	39
15	280	54	24	8	6
40	74	11	4	1	1

* 2.59 km^2 = 1 $mile^2$

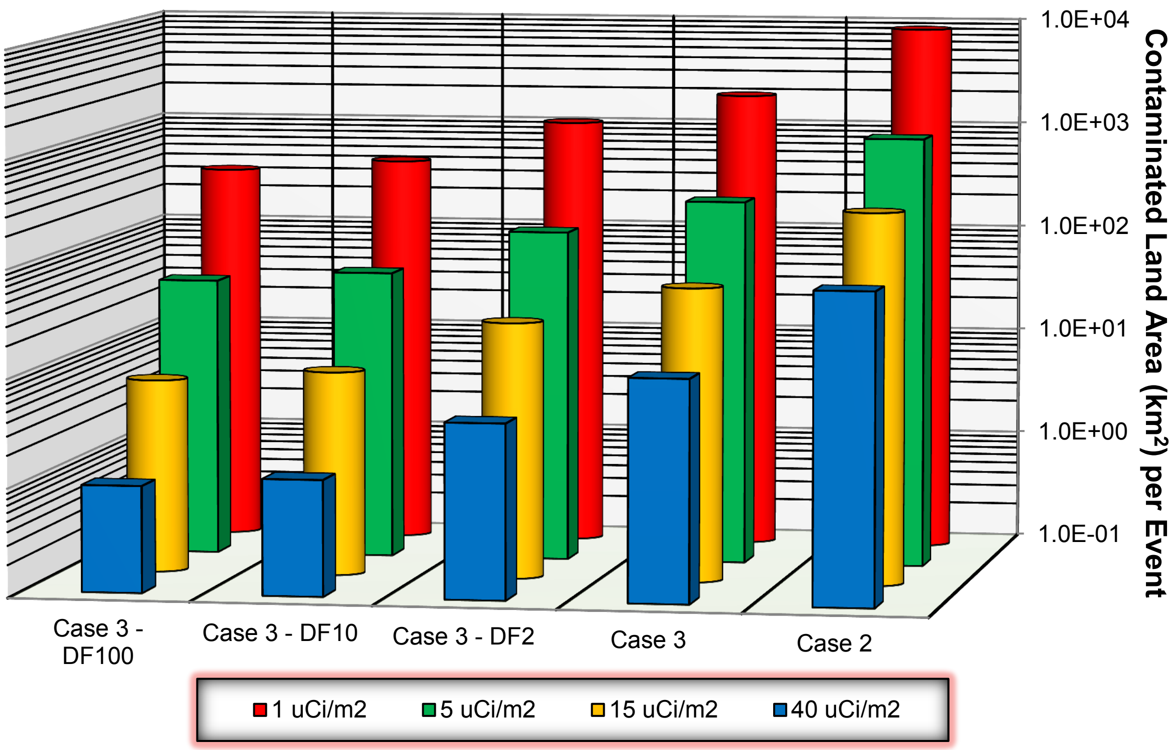


Figure 4. Mean, land contamination area per event for the base cases

2.3 Core Spray Cases

Table 6 provides a brief description of source terms for the Peach Bottom accident scenarios analyzed for Case 6 and Case 7. Each of the filtered cases has an applied DF of 2, 10, and 100 for the wetwell vent path. When a DF is applied to the pathway for flow through the filtered vent (i.e., Case 7—wetwell vent left open), the relationship is linear between the inverse of DF and the source term. The reason is that for the filtered cases, the wetwell vent path is the only release pathway to the environment.

Table 6. Brief Source Term Description for MELCOR Scenarios Discussed in the Core Spray Cases Consequence Analyses

Scenario	Integral Release Fractions by Chemical Group									Atmospheric Release Timing	
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Case 6 Base case with core spray	0.73	0.004	0.001	0.016	0.035	0	0	0	0	25.7	48
Case 7 Base case with wetwell venting and core spray	1.00	0.003	0.001	0.024	0.009	0	0	0	0	23.9	48
Case 7 DF=2	1.00	0.002	0.0005	0.012	0.005	0	0	0	0	23.9	48
Case 7 DF=10	1.00	0.0003	0.0001	0.002	0.001	0	0	0	0	23.9	48
Case 7 DF=100	1.00	0.00003	0.00001	0.0002	0.0001	0	0	0	0	23.9	48

2.3.1 Core Spray Cases—Latent Cancer Fatality and Prompt Fatality Risk

LCF risk results are presented for the LNT dose-response model. Table 7 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances for Case 6 and Case 7. As seen in Table 7, when a DF is applied to the pathway that flows through the filtered vent (i.e., Case 3—wetwell vent left open), the relationship is nonlinear between the inverse of DF and LCF risk.

For the filtered cases, even though the only release pathway to the environment is through the wetwell vent, the relationship between the assumed DF and the LCF risk is sublinear. The sublinear behavior is more pronounced at shorter distances, primarily due to short-term and long-term mitigative actions, as discussed in 2.2.1.

Table 7. Individual, Mean LCF risk per Event for Residents within a Circular Area at Specified Radial Distances for the Core Spray Cases

	Case 6 Base case with core spray	**Case 7** Base case with wetwell venting and core spray	**Case 7 DF 2**	**Case 7 DF 10**	**Case 7 DF 100**
0-10 miles	8.5×10^{-5}	6.4×10^{-5}	4.4×10^{-5}	1.3×10^{-5}	1.5×10^{-6}
0-20 miles	6.6×10^{-5}	4.6×10^{-5}	2.7×10^{-5}	7.2×10^{-6}	1.4×10^{-6}
0-30 miles	4.6×10^{-5}	3.1×10^{-5}	1.8×10^{-5}	4.6×10^{-6}	1.0×10^{-6}
0-40 miles	3.0×10^{-5}	2.0×10^{-5}	1.1×10^{-5}	2.8×10^{-6}	6.4×10^{-7}
0-50 miles	2.5×10^{-5}	1.6×10^{-5}	9.1×10^{-6}	2.2×10^{-6}	5.2×10^{-7}

Figure 5 shows the individual, mean LCF risk per event using the LNT model for residents within a circular area at specified radial distances for Case 6 and Case 7. Each column is the combined (total) LCF risk from the emergency and long-term phases (i.e., the results shown in Table 7). Table 7 and Figure 5 show that all the vented cases, unfiltered or filtered, have a lower total LCF risk than the unvented case (i.e., Case 6). If venting is used, assuming a DF of 100 for the external filter, the total LCF risk is reduced by ~98 percent at the five specified radial distances, compared to the unfiltered vented case (Case 7).

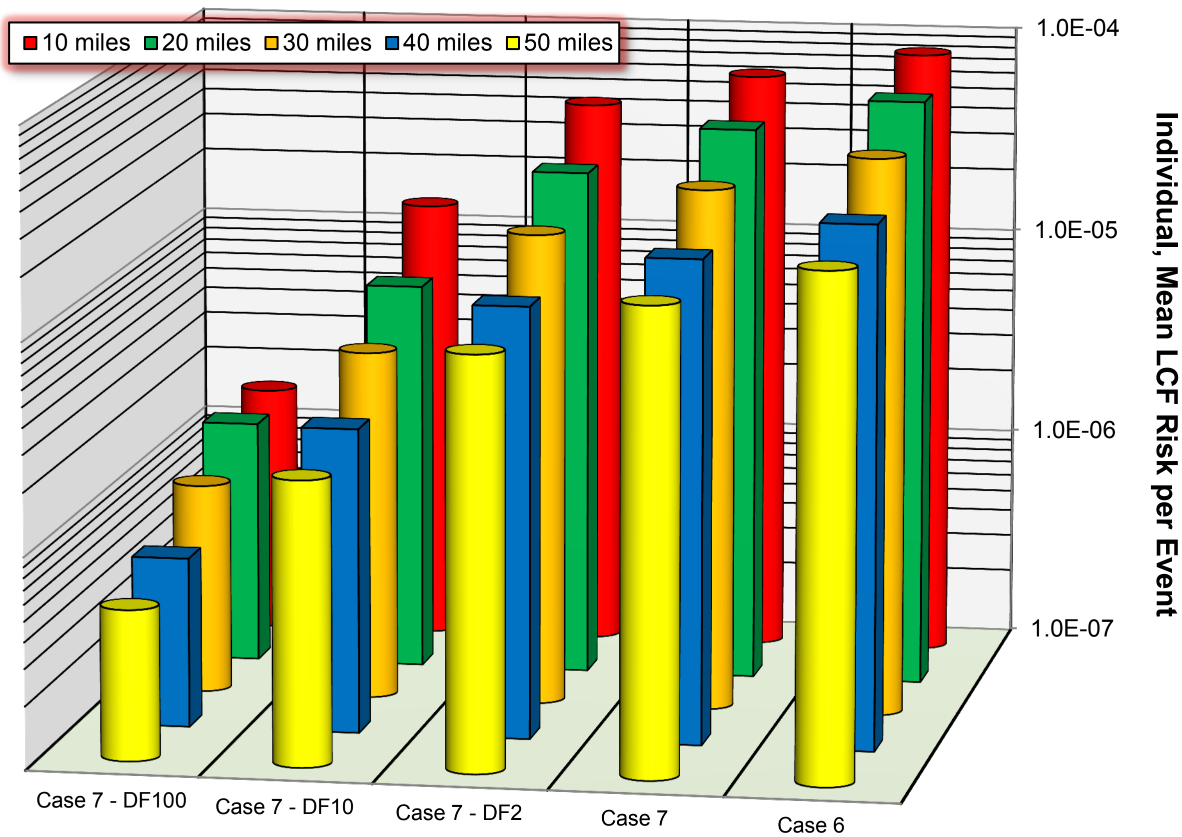


Figure 5. Individual, mean LCF risk per event for residents within a circular area at specified radial distances for the core spray cases

Figure 6 shows the individual, mean LCF risk per event using the LNT dose-response model for residents within a circular area at the specified radial distances for Case 6. The figure shows the emergency and long-term phases. The entire height of each column shows the combined (total) LCF risk for the two phases (i.e., the results shown in Table 7). The emergency response is very effective within the EPZ (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population who are modeled as refusing to evacuate. The emergency phase accounts for ~35 percent of the total LCF risk within the 50-mile radial distance.

The long-term phase risk dominates the total risks for this case when the LNT dose-response model is used. These long-term risks are controlled by the habitability (return) criterion, which is the dose rate at which residents are allowed to return to their homes following the emergency phase. For Peach Bottom, the State of Pennsylvania's guideline of a dose rate of 500 mrem/yr is used for the habitability criterion.

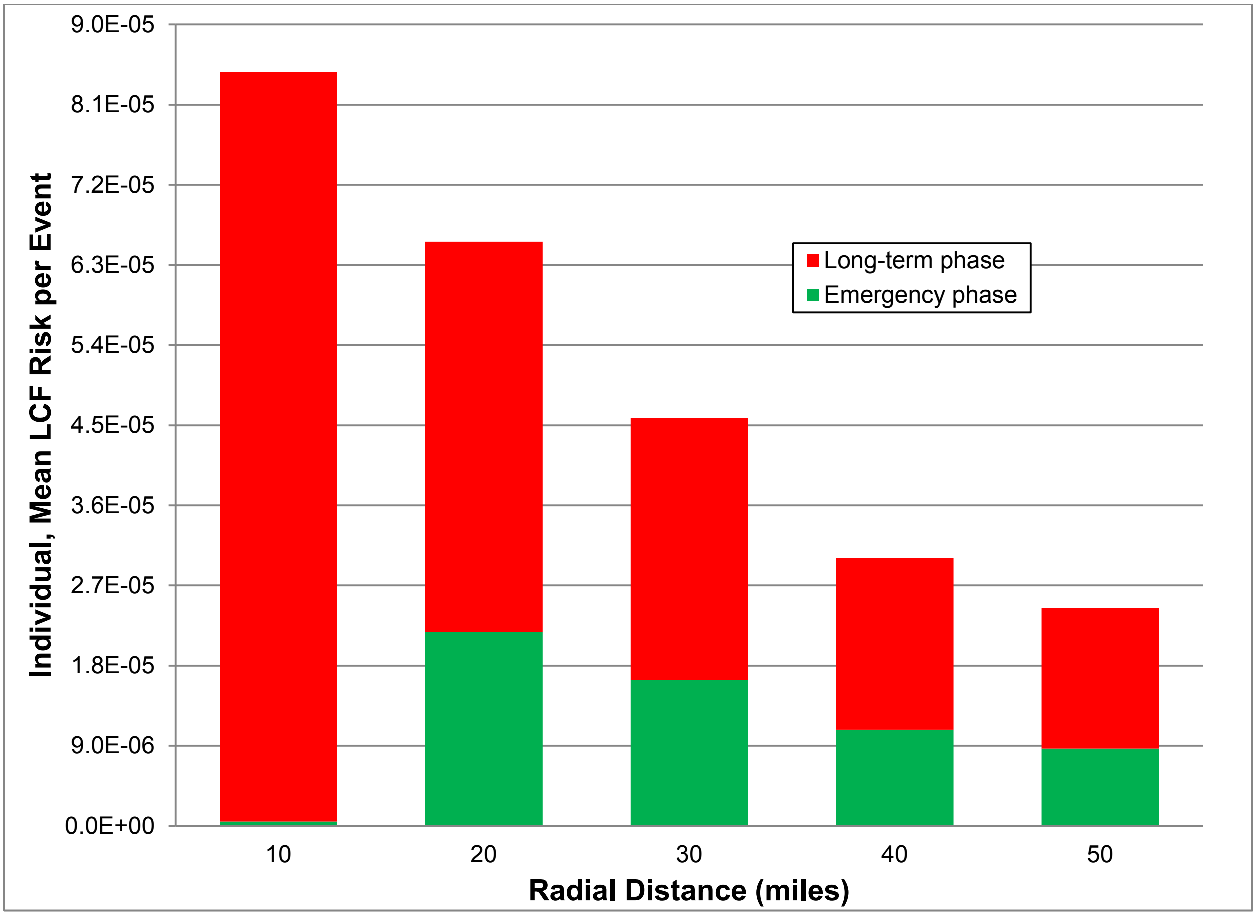


Figure 6. Case 6 individual, mean LCF risk per event for residents within a circular area at specified radial distances

Figure 7 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances using the LNT dose-response model for Case 7 with three values of DF applied. Again, the emergency response is very effective within the evacuation zone (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population who are modeled as refusing to evacuate. The explanations provided for Figure 6 also apply to Figure 7. The emergency phase accounts for 30–70 percent of the total LCF risk within the 50-mile radial distance for all DF values.

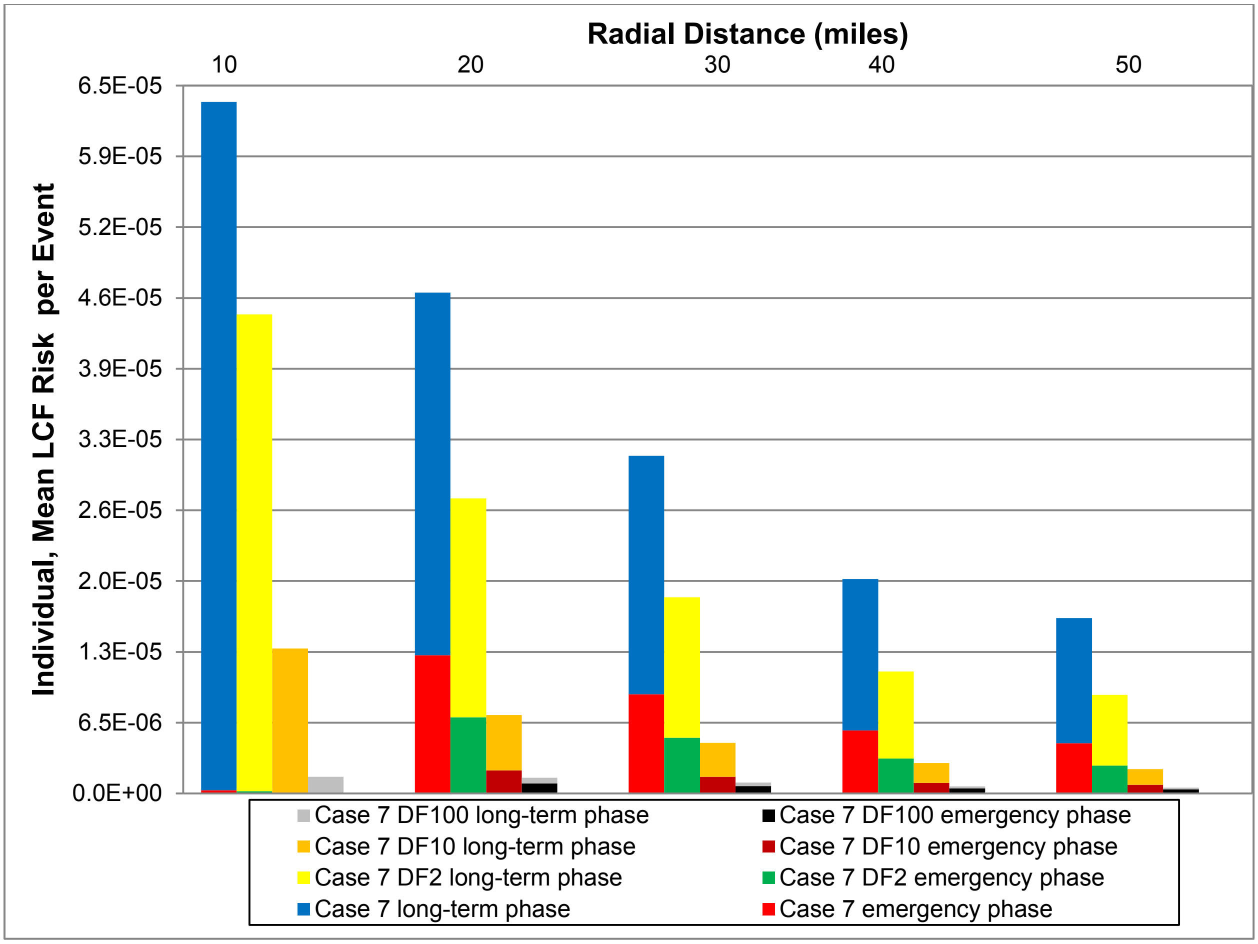


Figure 7. Case 7 individual, mean LCF risk per event for residents within a circular area at specified radial distances with specified decontamination factors

The prompt fatality risks are zero for these cases. This is again because the release fractions (i.e., see in Table 6) are too low to produce doses large enough to exceed the dose thresholds for early fatalities (see discussion under Section 2.2.1 above). The largest value of the mean, acute exposure for the closest resident for these cases is about 0.06 Gy to the red bone marrow.

2.3.2 Core Spray Cases—Land Contamination

Table 8 provides the mean, contaminated area prior to decontamination for specified Cs-137 contamination levels for Case 6 and Case 7. As discussed in Section 2.2.2 above, the relationship between the inverse of DF (i.e., the quantity released) and land contamination area is superlinear.

Figure 8 shows the mean, land contamination area per event for Case 6 and Case 7. When the unfiltered case (i.e., Case 6) is compared with the filtered case, a DF of 10 or 100 results in a several order-of-magnitude reduction in land contamination area.

Table 8. Mean, Contaminated Area per Event above the Specified Contamination Level for the Core Spray Cases

Contamination Level ($\mu Ci/m^2$ of ^{137}Cs)	Contaminated Area (km^2)*				
	Case 6 Base case with core spray	**Case 7** Base case with wetwell venting and core spray	**Case 7 DF 2**	**Case 7 DF 10**	**Case 7 DF 100**
1	1,800	1,400	590	62	1
5	270	180	62	4	0.02
15	72	34	11	0.4	0.002
40	19	7	2	0.04	0.0001

* 2.59 km^2 = 1 $mile^2$

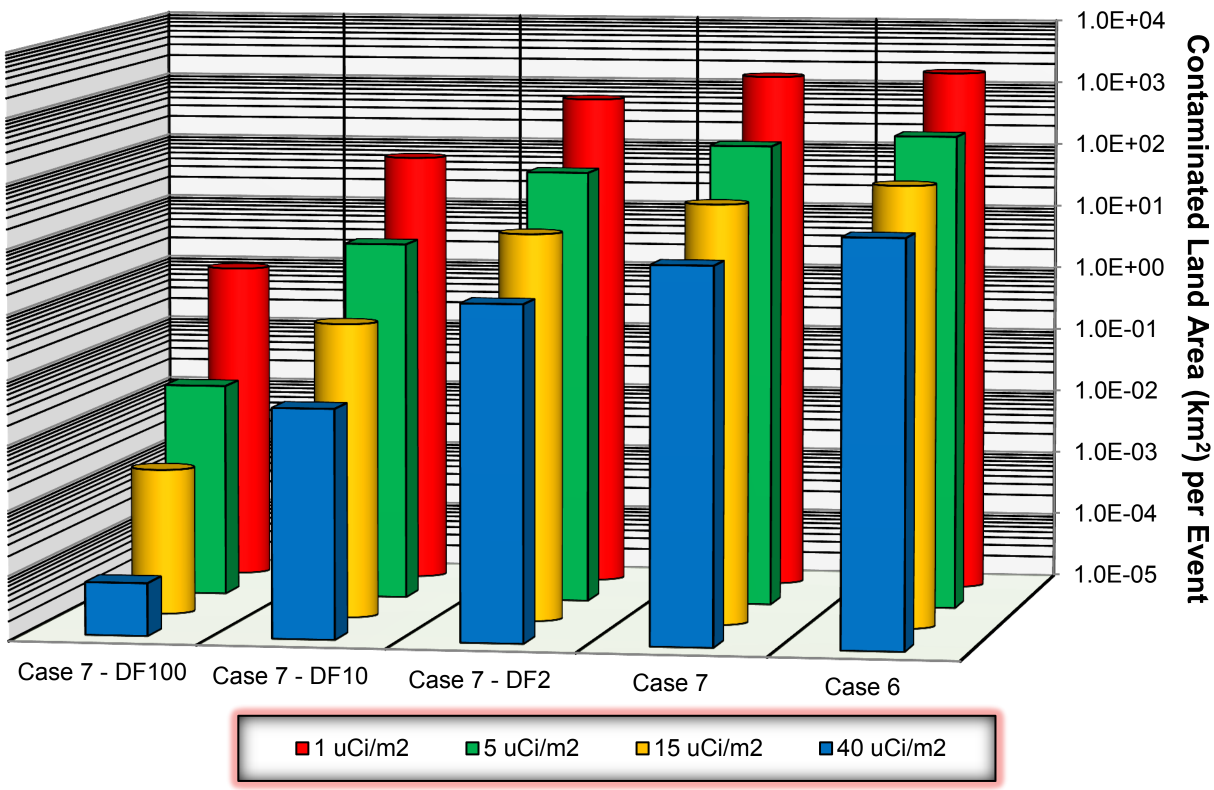


Figure 8. Mean, land contamination area per event for the core spray cases

2.4 Drywell Venting Cases

Case 12 and Case 13 are unique when compared to the other accident scenarios analyzed for this study in that containment is vented via the drywell vent path, and both cases experience a main steam line failure. These two cases were considered as an alternative to wetwell venting. If the cavity is deeply flooded, as in some European plants, the wetwell vent path will be ineffective in which case venting will occur through the drywell vent.

Additionally, the safety relief valve (SRV) stochastic failure probability was disabled (i.e., the SRV stochastic failure probability was set to zero—no failure) in MELCOR, which resulted in failure of the main steam line. With a longer valve cycling period, the main steam line experiences high temperature gases exiting the reactor pressure vessel (RPV) to the wetwell via the SRV. These increased temperatures ultimately result in a failure of the main steam line at 27.7 hours. The main steam line failure allows radionuclides released from the fuel to bypass the wetwell and directly enter the drywell. This results in a larger environmental release when either drywell venting occurs or when containment fails.

For Case 12 and Case 13, drywell venting occurs before the main steam line failure. Since the main steam line failure is such a large pressure transient (i.e., >50 psid in 2 seconds in the drywell), that even when the use of containment sprays (i.e., Case 13) is considered, the unfiltered drywell vent path results in a large environmental release.

Table 9 provides a brief description of source terms for the Peach Bottom accident scenarios analyzed for Case 12 and Case 13. Since there are no scrubbing effects from the wetwell for drywell venting, the external filter is considered to be 99.9 percent efficient (i.e., DF = 1,000). As a sensitivity study to determine the effect of increased filter efficiency, Case 12 assumes the external filter is 99.98 percent efficient (i.e., DF = 5,000).

When a DF is applied to the pathway for flow through the filtered vent (i.e., Case 12—drywell vent left open), the relationship is nonlinear between the inverse of DF and the source term. The reason is that for the filtered cases, the drywell vent path is not the only release pathway to the environment. At ~35 hours, the containment fails due to core melt through of the drywell liner for both cases. The drywell liner failure provides a lower resistance pathway to the environment than through the drywell vent.

Table 9. Brief Source Term Description for MELCOR Scenarios Discussed in the Drywell Venting Cases Consequence Analyses

Scenario	Integral Release Fractions by Chemical Group									Atmospheric Release Timing	
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Case 12 Base case with drywell venting	1.00	0.194	0.037	0.490	0.364	0.001	0.043	0.003	0	25.5	48
Case 12 DF=1000	1.00	0.0012	0.002	0.015	0.010	0	0	0.0001	0	25.5	48
Case 12 DF=5000	1.00	0.0010	0.002	0.014	0.010	0	0	0.0001	0	25.5	48
Case 13 Base case with drywell venting and drywell spray	1.00	0.186	0.048	0.484	0.380	0.001	0.041	0.005	0	25.5	48
Case 13 DF=1000	1.00	0.0002	0.0005	0.001	0.0005	0	0	0	0	25.5	48

2.4.1 Drywell Venting Cases—Latent Cancer Fatality and Prompt Fatality Risk

LCF risk results are presented for the LNT dose-response model. Table 10 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances for Case 12 and Case 13. As seen in Table 10, when a DF is applied to the pathway for flow through the drywell filtered vent (i.e., either case), the relationship is nonlinear between the inverse of DF and LCF risk.

As discussed above for both cases, the drywell vent path is not the only release pathway to the environment. As a result of this additional environmental release pathway (i.e., the drywell liner failure), the relationship between the assumed DF and the LCF risk is sublinear. The sublinear behavior is more pronounced at shorter distances, for reasons discussed in Section 2.2.1.

Table 10. Individual, Mean LCF Risk per Event for Residents within a Circular Area at Specified Radial Distances for the Drywell Venting Cases

	Case 12 Base case with drywell venting	**Case 12 DF 1000**	**Case 12 DF 5000**	**Case 13** Base case with drywell venting and drywell spray	**Case 13 DF 1000**
0-10 miles	4.0×10^{-4}	1.1×10^{-4}	9.3×10^{-5}	4.0×10^{-4}	3.6×10^{-5}
0-20 miles	8.5×10^{-4}	5.7×10^{-5}	5.0×10^{-5}	9.3×10^{-4}	1.5×10^{-5}
0-30 miles	5.8×10^{-4}	3.4×10^{-5}	3.1×10^{-5}	6.3×10^{-4}	8.5×10^{-6}
0-40 miles	3.8×10^{-4}	2.1×10^{-5}	1.8×10^{-5}	4.0×10^{-4}	4.8×10^{-6}
0-50 miles	3.2×10^{-4}	1.6×10^{-5}	1.4×10^{-5}	3.3×10^{-4}	3.7×10^{-6}

Figure 9 shows the individual, mean LCF risk per event using the LNT model for residents within a circular area at specified radial distances for Case 12 and Case 13. Each column is the combined (total) LCF risk from the emergency and long-term phases (i.e., the results shown in Table 10). Table 10 and Figure 9 show that the filtered cases have a lower total LCF risk than the unfiltered cases. Assuming a DF of 1,000 for the external filter, the total LCF risk for Case 12 is reduced by ~70 percent for the 10-mile radial distances and ~95 percent within the 50-mile radial distance. Assuming a DF of 1,000 for the external filter, the total LCF risk for Case 13 is reduced by ~90 percent for the 10-mile radial distances and ~99 percent within the 50-mile radial distance.

An interesting observation is seen when the LCF risk for Case 12 is compared with Case 13. Even though containment spray is on for Case 13, the LCF risks are higher. The majority of the source term for these unfiltered cases occurs when the main steam line fails. When the source terms are compared, Case 13 has a slightly higher barium (Ba), tellurium (Te), and cerium (Ce) release fraction and a slightly lower iodine (I) and cesium (Cs) release fraction (i.e., see Table 9).

Figure 10 shows the individual, mean LCF risk per event using the LNT dose-response model for residents within a circular area at the specified radial distances for the unfiltered cases. The figure shows the emergency and long-term phases. The entire height of each column shows the combined (total) LCF risk for the two phases (i.e., the results shown in Table 7). As shown in Figure 10, the two unfiltered cases show similar long-term LCF risk. However, the short-term LCF risk for Case 13 is higher. This is attributed to slightly higher short-term LCF risk contributors from the Ce (e.g., Pu-238 and Pu-239) and Ba classes for acute inhalation dose. Additionally, the emergency phase accounts for ~50-55 percent of the total LCF risk within the 50-mile radial distance for both unfiltered cases.

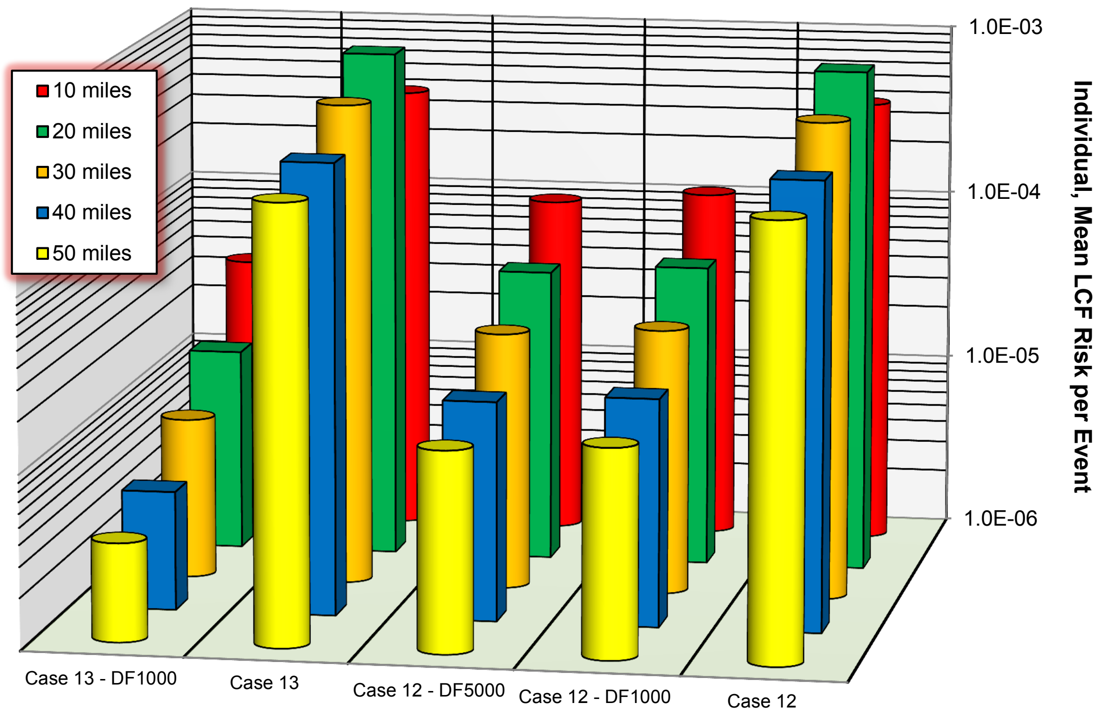


Figure 9. Individual, mean LCF risk per event for residents within a circular area at specified radial distances for the drywell venting cases

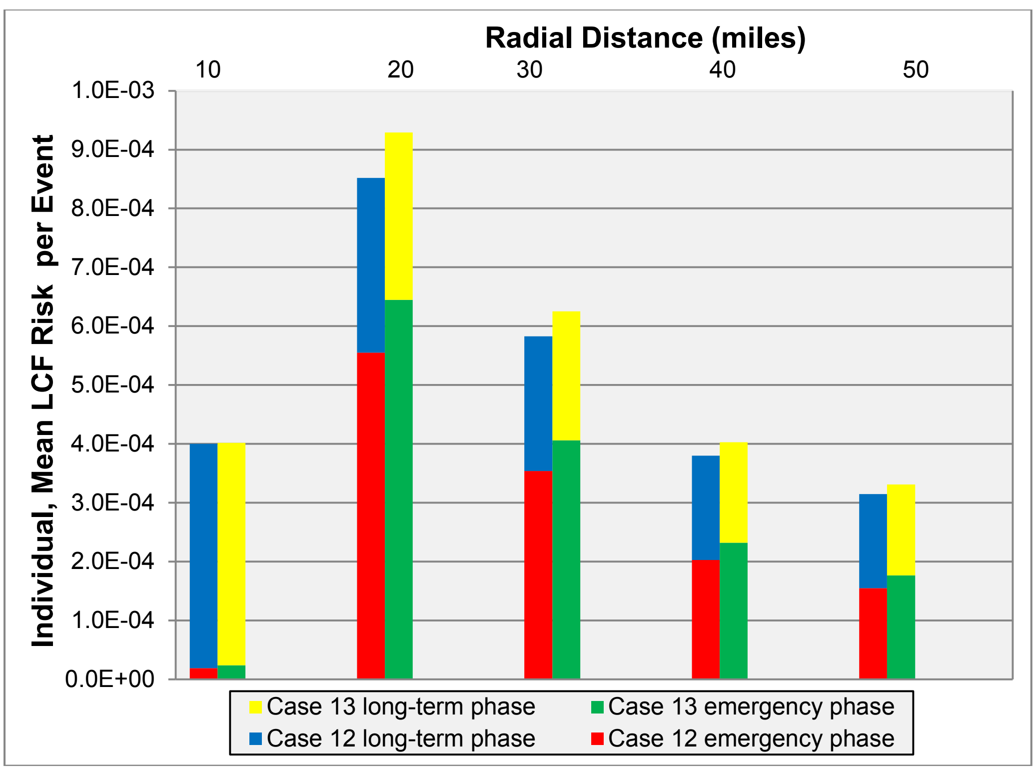


Figure 10. Individual, mean LCF risk per event for residents within a circular area at specified radial distances for unfiltered Case 12 and unfiltered Case 13

Figure 11 shows the individual, mean LCF risk per event using the LNT dose-response model for residents within a circular area at the specified radial distances for Case 12 with respective DFs applied. The figure shows the emergency and long-term phases. The entire height of each column shows the combined (total) LCF risk for the two phases (i.e., the results shown in Table 3). The emergency response is very effective within the EPZ (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population that are modeled as refusing to evacuate. The emergency phase accounts for ~30 percent of the total LCF risk when a DF is applied, and ~50 percent of the total LCF risk for the unfiltered case, within the 50-mile radial distance.

When a DF is applied, the long-term phase risk dominates the total risks for this case. These long-term risks are controlled by the habitability (return) criterion.

For the unfiltered case, the emergency phase risk is equal to or dominates the total risk due to the main steam line failure. The emergency phase risk is controlled by inhalation doses during the emergency phase as a result of the large iodine release fraction.

For the sensitivity study where a DF of 5,000 is applied for Case 12, there is a sublinear relationship with the filtered Case 12 where a DF of 1,000 is applied. This sublinear relationship is attributed to the additional release pathway. As discussed above, the drywell vent path is not the only release pathway to the environment. As a result of this additional environmental release pathway (i.e., the drywell liner failure), when a DF ≥1,000 is applied the fraction of the source term that is released through the drywell liner failure dominates the overall source term (see Table 9).

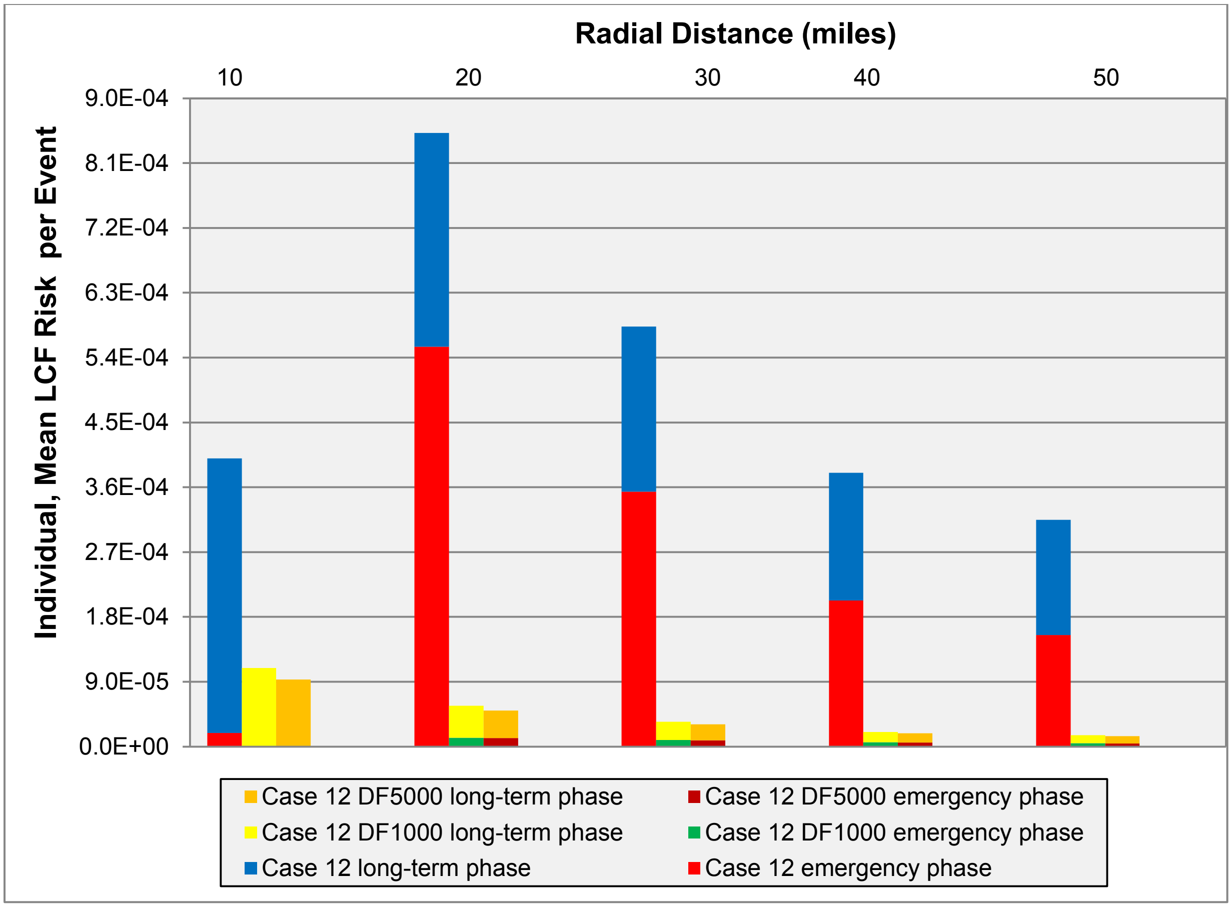


Figure 11. Case 12 individual, mean LCF risk per event for residents within a circular area at specified radial distances with specified decontamination factors

Figure 12 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances using the LNT dose-response model for Case 13 with the respective DF applied. Again, the emergency response is very effective within the evacuation zone (10 miles) during the early phase. The explanations provided for Figure 11 also apply to Figure 12. The emergency phase accounts for ~30 percent of the total LCF risk when a DF is applied, and ~55 percent of the total LCF risk for the unfiltered case, within the 50-mile radial distance.

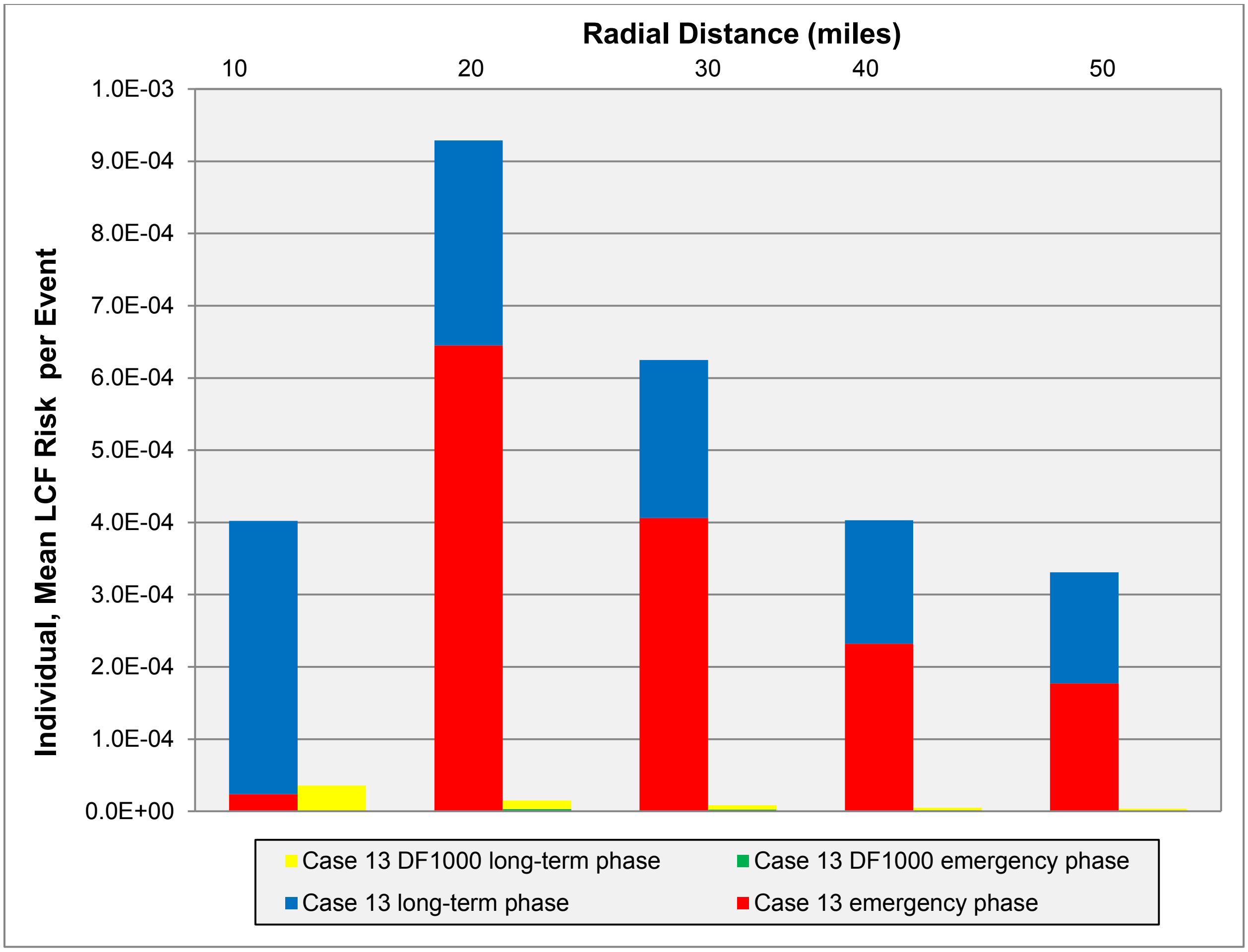


Figure 12. Case 13 individual, mean LCF risk per event for residents within a circular area at specified radial distances with a specified decontamination factor

The prompt fatality risks are zero for all cases, except unfiltered Case 13. For the cases that resulted in a zero prompt fatality risk, this is because the release fractions (i.e., see in Table 9) are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5 percent of the population that are modeled as refusing to evacuate. The largest value of the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers[7] from the plant) for these cases is about 0.8 Gy to the red bone marrow (i.e., unfiltered Case 12). As discussed previously, the red bone marrow is usually the most sensitive organ for prompt fatalities, but the minimum acute dose that can cause an early fatality is about 2.3 Gy to the red bone marrow. The calculated mean, acute exposures are all well below this threshold.

For unfiltered Case 13, Table 11 provides the mean, individual prompt fatality risk per event within the 3-mile radial distance. Beyond 3 miles, prompt fatality risk is zero. For unfiltered Case 13, the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers[7] from the plant) is about 1.0 Gy to the red bone marrow. While this is below the red bone marrow threshold for an early fatality, 0.5 percent of the MACCS2 weather trials produced an acute exposure greater than the threshold. As a result of these few weather trials, a nonzero mean prompt fatality risk was observed. Based on this observation and since the mean, prompt

[7] 1.6 km = 1 mile

fatality risk for the 2-mile and 2.5-mile radial distances are so low, the mean, individual prompt fatality risk per event at these distances are considered essentially zero.

Table 11. Mean, Individual Prompt Fatality Risk per Event for Unfiltered Case 13

Radius of Circular Area (mi)	Unfiltered Case 13 Base case with drywell venting and drywell spray
1.3	0.0
2	1.9×10^{-9}
2.5	1.1×10^{-9}

2.4.2 Drywell Venting Cases—Land Contamination

Table 12 provides the mean, contaminated area prior to decontamination for specified Cs-137 contamination levels for Case 12 and Case 13. The relationship between the inverse of DF (i.e., the quantity released) and land contamination area is again superlinear for reasons discusses in Section 2.1.2.

Figure 13 shows the mean, land contamination area per event for Case 12 and Case 13. When the unfiltered cases are compared with the filtered case, a DF of 1,000 results in a several order-of-magnitude reduction in land contamination area.

For the sensitivity study where a DF of 5,000 is applied for Case 12, there is a sublinear relationship with the filtered Case 12 where a DF of 1,000 is applied. This sublinear relationship is attributed to the additional release pathway. As discussed above, the drywell vent path is not the only release pathway to the environment. As a result of this additional environmental release pathway (i.e., the drywell liner failure), when a DF ≥1,000 is applied the fraction of the source term that is released through the drywell liner failure dominates the overall source term (i.e., see Table 9). Thus, a higher DF has little effect on the overall contaminated land area.

Table 12. Mean, Contaminated Area per Event above the Specified Contamination Level for the Drywell Venting Cases

Contamination Level ($\mu Ci/m^2$ of ^{137}Cs)	Contaminated Area (km^2)				
	Case 12 Base case with drywell venting	**Case 12 DF 1000**	**Case 12 DF 5000**	**Case 13** Base case with drywell venting and drywell spray	**Case 13 DF 1000**
1	83,000	590	510	86,000	110
5	29,000	110	93	29,000	13
15	9,200	28	25	8,800	2
40	3,300	7	6	3,000	0.02

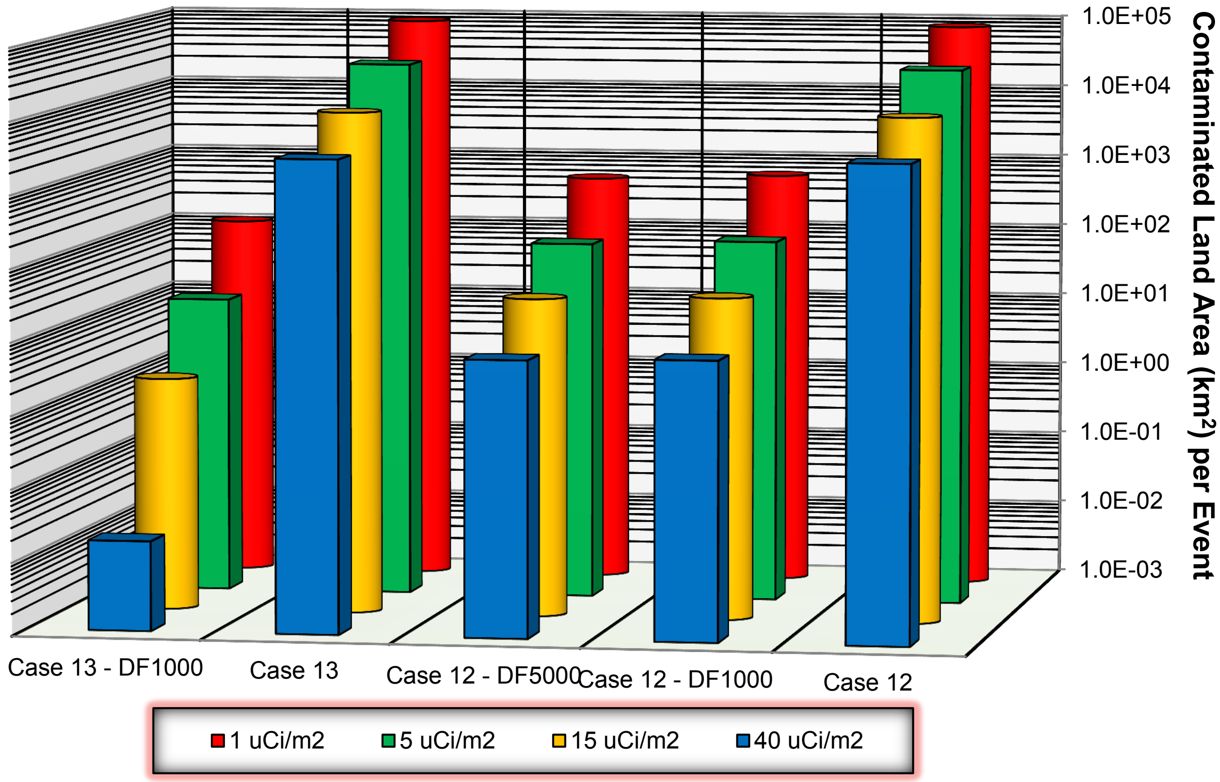


Figure 13. Mean, land contamination area per event for the drywell venting cases

2.5 Drywell Spray Cases

Table 13 provides a brief description of source terms for the Peach Bottom accident scenarios analyzed for Case 14 and Case 15. Each of the filtered cases has an applied DF of 2, 10, and 100 for the wetwell vent path. When a DF is applied to the pathway for flow through the filtered vent (i.e., Case 15—wetwell vent left open), the relationship is linear between the inverse of DF and the source term, with the exception of the noble gases. For Case 15, the wetwell vent path is the only release pathway to the environment.

Table 13. Brief Source Term Description for MELCOR Scenarios Discussed in the Drywell Spray Cases Consequence Analyses

Scenario	Integral Release Fractions by Chemical Group									Atmospheric Release Timing	
	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Case 14 Base case with drywell spray	0.68	0.001	0	0.004	0.005	0	0	0	0	28.2	48
Case 15 Base case with wetwell venting and drywell spray	1.00	0.003	0.002	0.019	0.021	0	0	0	0	23.9	48
Case 15 DF=2	1.00	0.002	0.001	0.010	0.011	0	0	0	0	23.9	48
Case 15 DF=10	1.00	0.0003	0.0002	0.002	0.002	0	0	0	0	23.9	48
Case 15 DF=100	1.00	0.00003	0.00002	0.0002	0.0002	0	0	0	0	23.9	48

The reason the source term with drywell sprays only (i.e., Case 14) is lower than the source term with drywell sprays and wetwell venting (i.e., Case 15) is mostly due to the much greater flow rate through the opened wetwell vent in Case 15 than the flow through the leaking drywell head flange in Case 14. The pressure suppression by the drywell sprays minimizes leakage from the drywell head flange, which is the primary model of containment overpressure failure and is the only pathway for radionuclide release to the environment for Case 14. The head flange leakage in the MELCOR model behaves elastically. Thus, after a high pressure excursion that temporarily lifts the head flange at ~26 hours for 20 minutes, the head flange is assumed to reseat perfectly with no residual leakage as long as the containment sprays reduce drywell pressure below 80 psig. The head flange doesn't lift again until RPV lower vessel head failure at 36.6 hours, and after about 4.5 hours the head flange reseats and intermittently reopens for the rest of the MELCOR simulation.

A secondary reason is that the lower containment pressure in Case 15 resulting from the wetwell venting fosters slow revaporization of cesium and iodine from the RPV internals. The vapors escape the RPV and condense into aerosols that are carried towards the wetwell vent. Some of the aerosols are scrubbed in the wetwell pool but not all of them. The aerosols not scrubbed in the pool release to the environment through the wetwell vent path. In considering the scrubbing taking place in the wetwell pool during wetwell venting for Case 15, the flow to the wetwell is through the downcomer vents rather than through the T-quenchers. A DF of 10 associated with the downcomer vents is markedly less than a DF of 1,000 associated with the T-quenchers as reported by MELCOR for Case 15.

The much higher flow rates through the vent, combined with increased revaporization of cesium and iodine from RPV internals and attendant imperfect wetwell scrubbing of small aerosols produced after revaporization for Case 15, the elastic drywell head flange model in MELCOR, and the effectiveness of the drywell containment sprays lead to the nonintuitive larger environmental release for Case 15 relative to Case 14.

2.5.1 Drywell Spray Cases—Latent Cancer Fatality and Prompt Fatality Risk

LCF risk results are presented for the LNT dose-response model. Table 14 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances for Case 14 and Case 15. As seen in Table 14, when a DF is applied to the pathway for flow through the filtered vent (i.e., Case 15—wetwell vent left open), the relationship is sublinear between the inverse of DF and LCF risk.

Table 14. Individual, Mean LCF Risk per Event for Residents within a Circular Area at Specified Radial Distances for the Drywell Spray Cases

	Case 14 Base case with drywell spray	**Case 15** Base case with wetwell venting and drywell spray	**Case 15 DF 2**	**Case 15 DF 10**	**Case 15 DF 100**
0-10 miles	$3.3x10^{-5}$	$9.3x10^{-5}$	$6.1x10^{-5}$	$1.8x10^{-5}$	$2.1x10^{-6}$
0-20 miles	$2.1x10^{-5}$	$6.2x10^{-5}$	$3.6x10^{-5}$	$9.2x10^{-6}$	$1.7x10^{-6}$
0-30 miles	$1.3x10^{-5}$	$4.1x10^{-5}$	$2.3x10^{-5}$	$5.8x10^{-6}$	$1.1x10^{-6}$
0-40 miles	$8.0x10^{-6}$	$2.6x10^{-5}$	$1.4x10^{-5}$	$3.5x10^{-6}$	$7.1x10^{-7}$
0-50 miles	$6.4x10^{-6}$	$2.1x10^{-5}$	$1.1x10^{-5}$	$2.7x10^{-6}$	$5.7x10^{-7}$

Figure 14 shows the individual, mean LCF risk per event using the LNT model for residents within a circular area at specified radial distances for Case 14 and Case 15. Each column is the combined (total) LCF risk from the emergency and long-term phases (i.e., the results shown in Table 14). Table 14 and Figure 14 show that unlike previous filtered cases, the vented case has a higher total LCF risk than the unfiltered case (i.e., Case 14) for a DF somewhat less than 10. The much higher flow rates through the vent, combined with increased revaporization of cesium and iodine from RPV internals and attendant imperfect wetwell scrubbing of small aerosols produced after revaporization for Case 15, the elastic drywell head flange model in MELCOR, and the effectiveness of the drywell containment sprays lead to the nonintuitive larger environmental release for Case 15 relative to Case 14. Assuming a DF of 100 for the external filter, the total LCF risk is reduced by ~97 percent for the five specified radial distances.

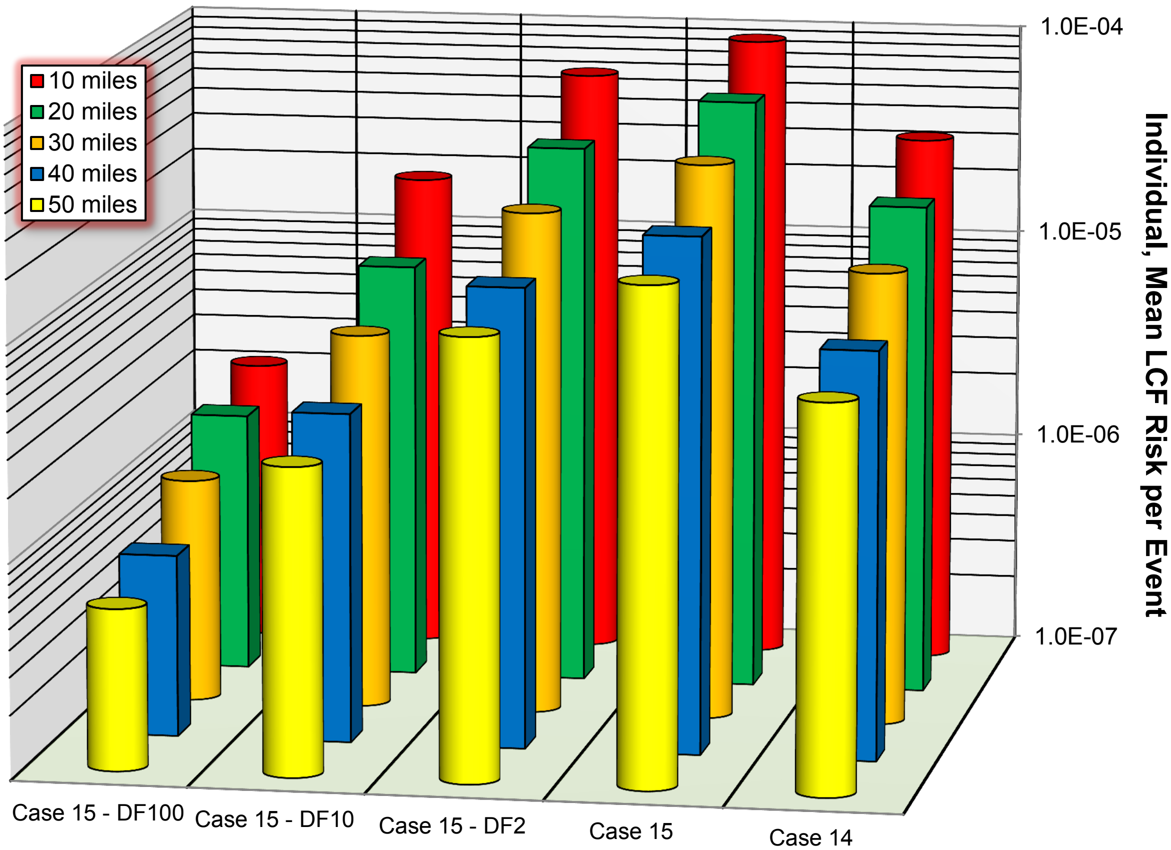


Figure 14. Individual, mean LCF risk per event for residents within a circular area at specified radial distances for the drywell spray cases

Figure 15 shows the individual, mean LCF risk per event using the LNT dose-response model for residents within a circular area at the specified radial distances for Case 14. The figure shows the emergency and long-term phases. The entire height of each column shows the combined (total) LCF risk for the two phases (i.e., the results shown in Table 14). The emergency response is very effective within the EPZ (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population that are modeled as refusing to evacuate. The emergency phase accounts for 30 percent of the total LCF risk within the 50-mile radial distance.

The long-term phase risk dominates the total risks for this case using the LNT dose-response model. These long-term risks are controlled by the habitability (return) criterion.

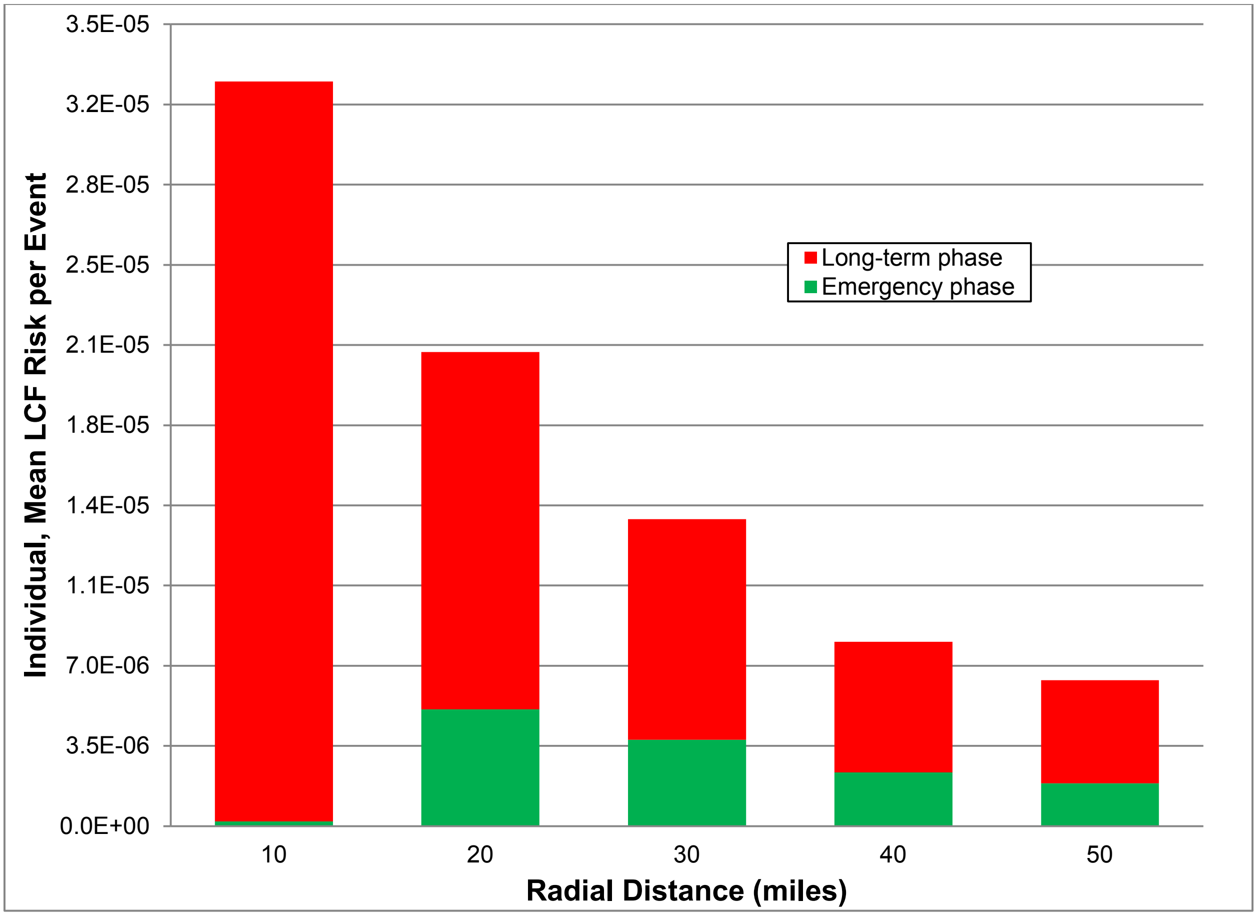


Figure 15. Case 14 individual, mean LCF risk per event for residents within a circular area at specified radial distances

Figure 16 shows the individual, mean LCF risk per event for residents within a circular area at specified radial distances using the LNT dose-response model for Case 15 with respective DFs applied. Again, the emergency response is very effective within the evacuation zone (10 miles) during the early phase. The explanations provided for Figure 15 also apply to Figure 16. The emergency phase accounts for 35–70 percent of the total LCF risk within the 50-mile radial distance for all DF values.

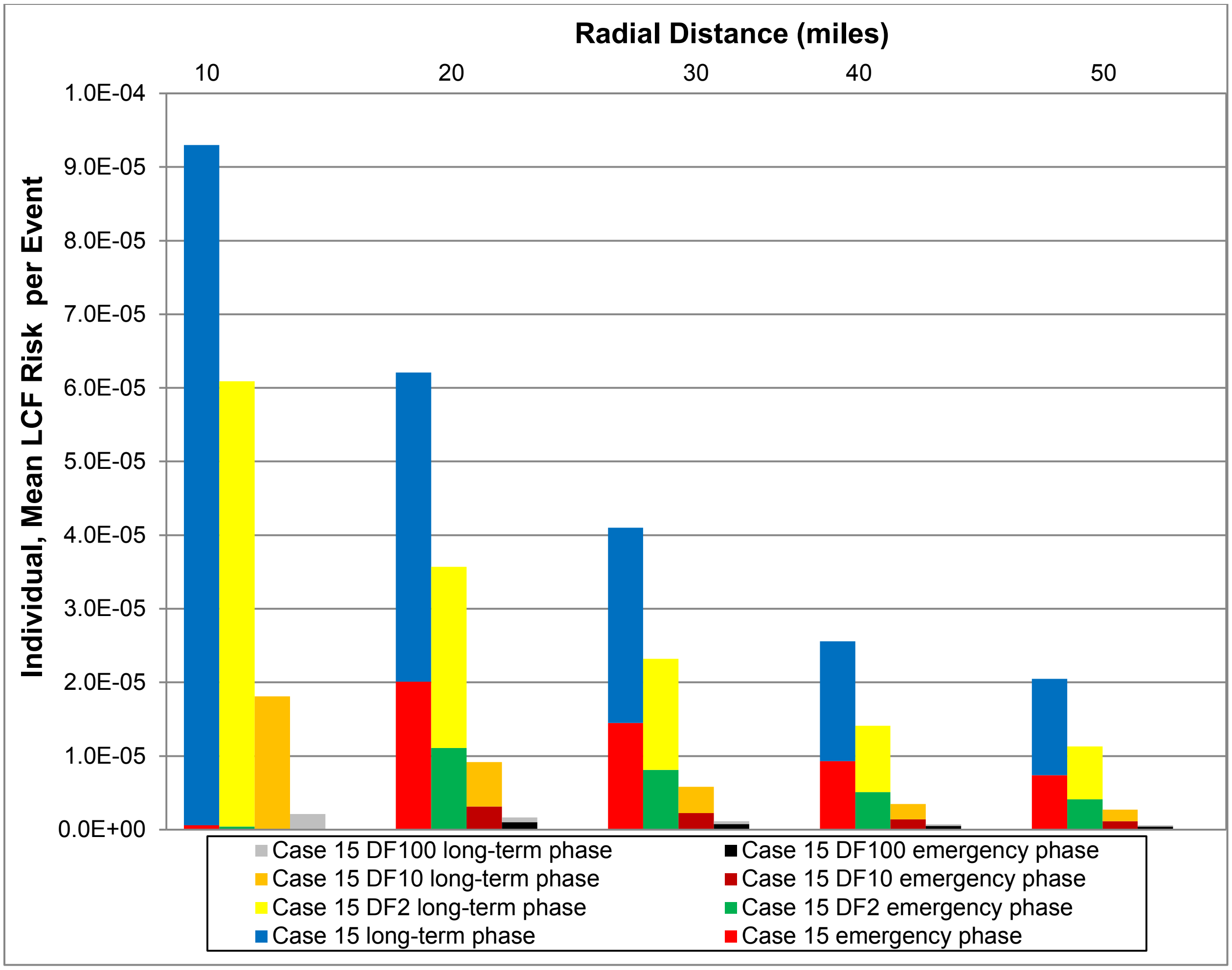


Figure 16. Case 15 individual, mean LCF risk per event for residents within a circular area at specified radial distances with specified decontamination factors

The prompt fatality risks are zero for these cases, because the release fractions (i.e., see Table 13) are too low to produce doses large enough to exceed the dose thresholds for early fatalities. The largest value of the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers from the plant) for these cases is about 0.06 Gy to the red bone marrow.

2.5.2 Drywell Spray Cases—Land Contamination

Table 15 provides the mean, contaminated area prior to decontamination for specified Cs-137 contamination levels for Case 14 and Case 15. As with the other cases, the relationship between the inverse of DF (i.e., the quantity released) and land contamination area is superlinear.

Figure 17 shows the mean, land contamination area per event for Case 14 and Case 15. When the unfiltered case (i.e., Case 15) is compared with the filtered case, a DF of 10 or 100 results in a several order-of-magnitude reduction in land contamination area.

As with the LCF risk, Table 15 and Figure 17 show that unlike previous filtered cases, the vented case has a higher mean land contamination area than the unvented unfiltered case (i.e., Case 14) for a DF somewhat less than 10. The much higher flow rates through the vent, combined with increased revaporization of cesium and iodine from RPV internals and attendant imperfect wetwell scrubbing of small aerosols produced after revaporization for Case 15, the elastic drywell head flange model in MELCOR, and the effectiveness of the drywell containment sprays lead to the nonintuitive larger environmental release for Case 15 relative to Case 14.

Table 15. Mean, Contaminated Area per Event above the Specified Contamination Level for the Drywell Spray Cases

Contamination Level (μCi/m^2 of ^{137}Cs)	Contaminated Area (km^2)*				
	Case 14 Base case with drywell spray	**Case 15** Base case with wetwell venting and drywell spray	**Case 15 DF 2**	**Case 15 DF 10**	**Case 15 DF 100**
1	390	1,200	480	53	1
5	51	140	53	3	0.01
15	10	28	8	0.3	0.001
40	2	5	1	0.02	0

* 2.59 km^2 = 1 mile^2

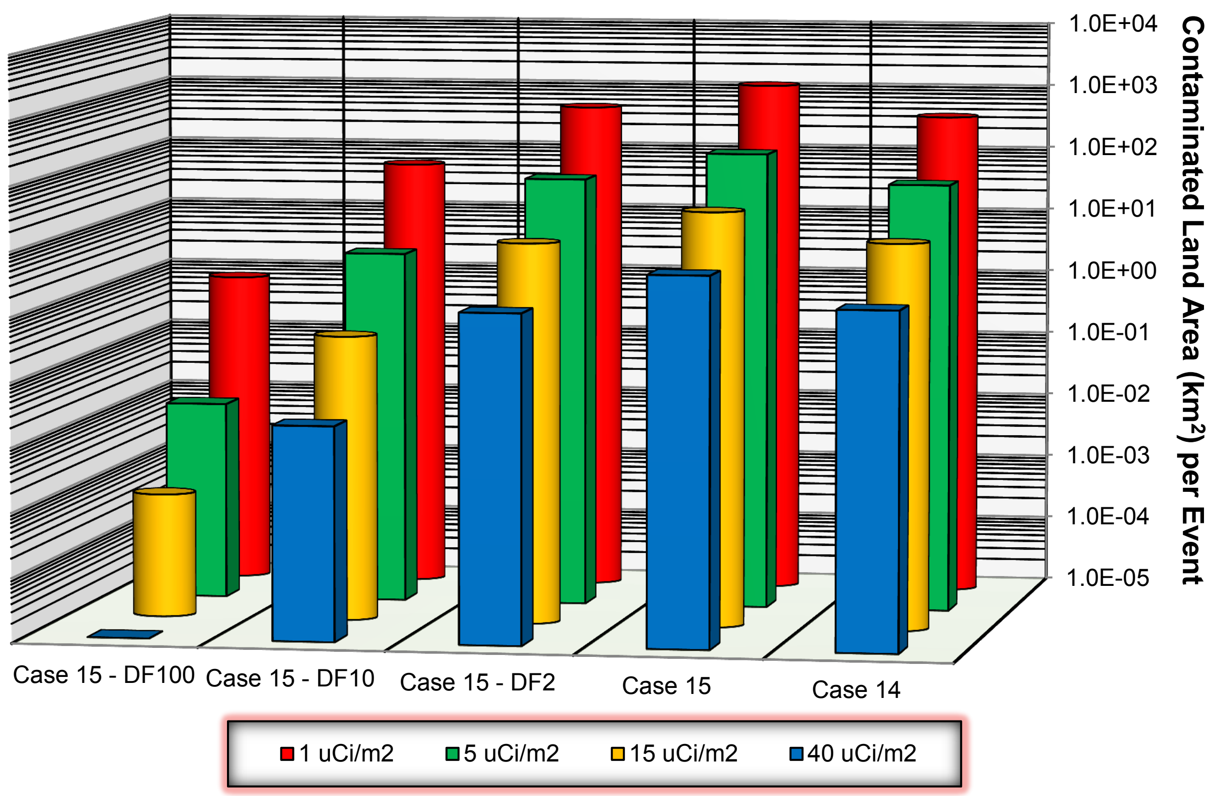


Figure 17. Mean, land contamination area per event for the drywell spray cases

2.6 Population Dose

A sum of all the effective doses to all the individuals within a given radial distance is roughly proportional to the number of radiation-induced health effects, using the LNT model. The proportionality is not perfect because latent health effects are calculated using a dose and dose-rate effectiveness factor that treats doses above 20 rem as being more effective for cancer induction than those below 20 rem. Furthermore, MACCS2 models cancers for individual organs, which is more complicated than basing them on an effective dose representing an average for the whole body.

The total, effective population dose from the plume and deposited contamination, subject to remedial actions to reduce dose levels, within a 50-mile radius of the plant is shown in Table 16 for each of the cases. The population dose is for a lifetime (i.e., 50-year dose commitment period), effective dose calculated for the population residing within a 50-mile radius. The relationship between population dose and inverse DF is sublinear because less remedial action is taken at lower contamination levels.

Table 16. Mean Population Dose (person-rem) per Event for Residents within a Circular Area of 50-mile Radius for Specified Decontamination Factors and for All the Cases Considered

Case 2 Base case	**Case 3** Base case with wetwell venting	**Case 3 DF 2**	**Case 3 DF 10**	**Case 3 DF 100**
580,000	460,000	320,000	180,000	140,000

Case 6 Base case with core spray	**Case 7** Base case with wetwell venting and core spray	**Case 7 DF 2**	**Case 7 DF 10**	**Case 7 DF 100**
310,000	240,000	140,000	37,000	8,200

Case 12 Base case with drywell venting	**Case 12 DF 1,000**	**Case 12 DF 5,000**	**Case 13** Base case with drywell venting and drywell spray	**Case 13 DF 1,000**
3,800,000	230,000	210,000	3,900,000	60,000

Case 14 Base case with drywell spray	**Case 15** Base case with wetwell venting and drywell spray	**Case 15 DF 2**	**Case 15 DF 10**	**Case 15 DF 100**
86,000	280,000	160,000	43,000	8,800

The composition and properties of the source terms affect the population dose through deposition rates, half-lives, and the types of radiation emitted. As described in the LCF risk sections, various phenomena affect dose depending on the phase of the event. During the emergency phase, evacuation within the EPZ significantly reduces population dose within the 10-mile radial distance. The only dose contribution within the EPZ is entirely represented by the 0.5 percent of the population that is modeled as refusing to evacuate. Emergency phase doses generally contribute less than half of the overall population dose for the cases considered. Case 7 with a DF=100 and Case 15 with a DF=100 are the only cases for which over half (i.e., 55 percent for both cases) of the population dose is from the emergency phase. Most of the long-term doses are controlled by the habitability (return) criterion, which is the dose rate at which residents are allowed to return to their homes following the emergency phase. For Peach Bottom, the State of Pennsylvania's guideline for habitability criterion is a dose rate of 500 mrem/yr starting the first year.

Unlike the doses included in LCF risks, population doses also include the ingestion pathway. The population doses include both public doses from the ingestion pathway and doses to decontamination workers working in the offsite contaminated area; LCF risk does not include either of these doses. Ingestion is considered during the long-term phase from contaminated food and water. The ingestion pathway accounts for:

- 10–20 percent of the population dose for the wetwell venting unfiltered cases considered
- 15–30 percent of the population dose for the wetwell venting filtered cases considered
- 5 percent of the population dose for the drywell venting unfiltered cases considered
- 20-30 percent of the population dose for the drywell venting filtered cases considered.

Figure 18 shows the mean population dose per event within a 50-mile radius for all cases considered. Table 16 and Figure 18 show that a DF of 10 or more for all wetwell venting filtered cases and a DF of 1,000 for all drywell venting filtered cases result in lower population doses than their respective unfiltered cases.

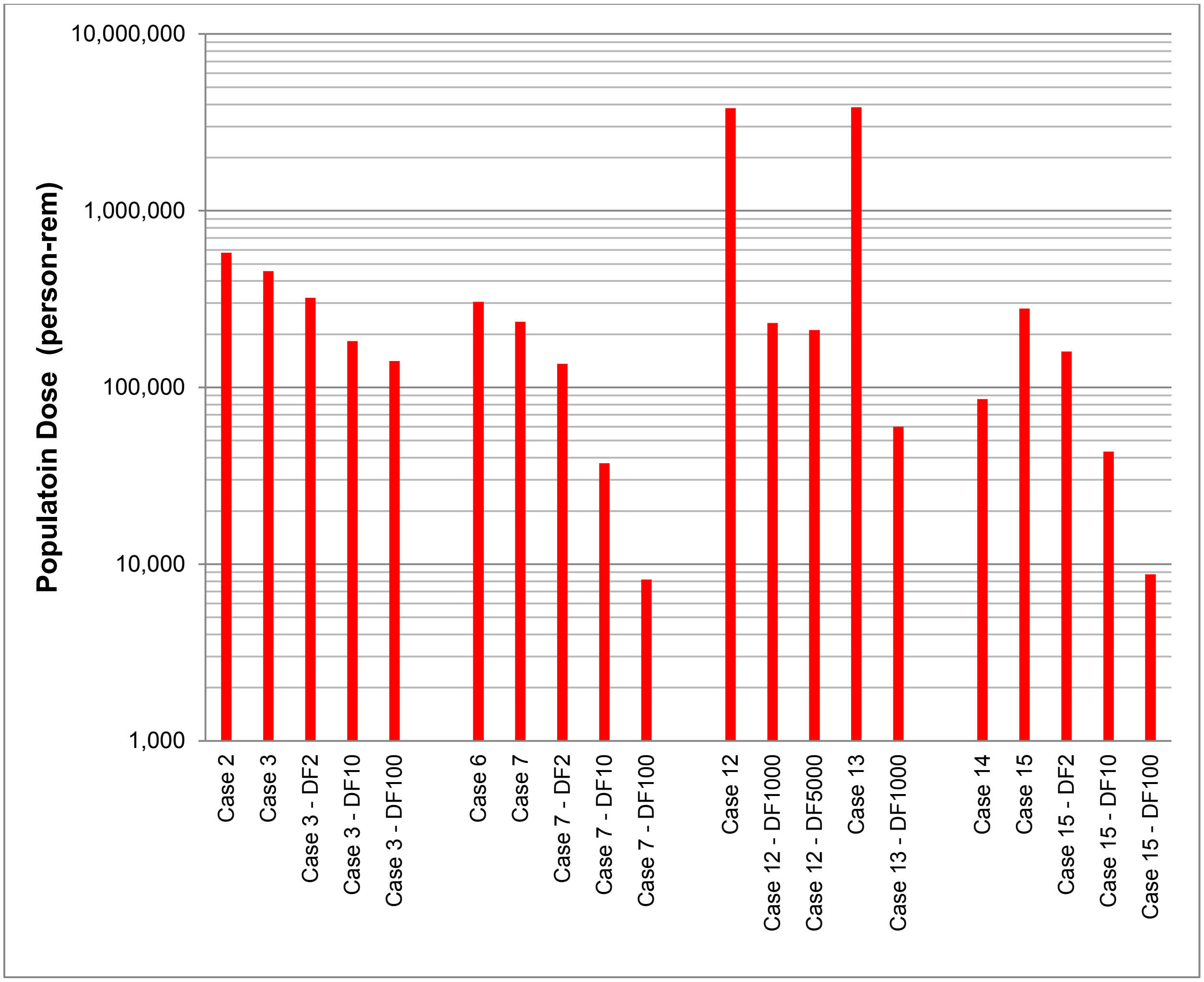


Figure 18. Mean population dose per event for residents within a circular area at the 50-mile radial distance with specified decontamination factors for all the cases considered

2.7 Offsite Economic Costs

The economic model in MACCS2 includes costs that fall within six categories as follows:

- evacuation and relocation costs
- moving expenses for people displaced
- decontamination
- cost due to loss of property use
- loss of contaminated food grown locally
- cost of condemned lands

The isotopic composition of the source term is one element that impacts the costs of decontamination. Some isotopes require no decontamination at all while others can more be difficult to decontaminate.

Other than the noble gases, each of the isotopes can deposit onto surfaces and cause contamination, but most of them have short half-lives and only remain in the environment for days or weeks. For example, iodine-131 has an 8-day half-life. Thus, in 80 days (i.e., 10 half-lives) its concentration is diminished to $2^{-10} \approx 0.001$ of its initial activity. As a result, it contributes to short-term doses but does not require decontamination because it disappears on its own. A relatively small number of the isotopes that could potentially be released from a nuclear reactor are radiologically important and require effort to decontaminate. Among these are Cs-134 and Cs-137, which have half-lives of 2 years and 30 years, respectively, and are important isotopes for a typical nuclear reactor accident in terms of decontamination costs.

In terms of the type of long-term radiation that would be emitted, the most important radionuclide, Cs-137, decays to Ba-137m, which rapidly decays and emits gamma radiation. Most of the resulting doses are from groundshine; inhalation and ingestion are relatively unimportant because cesium is rapidly excreted from the body and so these pathways do not lead to large doses. On the other hand, groundshine from deposited cesium can continue for tens or hundreds of years. Buildings and other structures can provide significant shielding from these gamma doses. The purpose of decontamination is to remove enough of the cesium to reduce the level of radiation from ground and building surfaces to acceptable levels (i.e., below the habitability limit).

Implementation of decontamination, which along with the associated interdiction of land is the dominant contributor to the overall economic costs, depends on whether or not the habitability criterion is exceeded. Remedial actions considered in the long-term phase depend on two criteria: habitability and farmability. Both of these criteria are based on contamination thresholds, which lead to inherently nonlinear relationships between source term magnitude and economic costs. This compounds the nonlinear effect between a DF and source term magnitude due to the DF applying to only the release pathway where the filter is connected. Thus applying a DF to represent an external filter does not result in a linear relationship between release (i.e., reciprocal of DF) and economic costs.

Table 17 provides the mean, total offsite economic costs shown in millions of 2005 dollars for the 10-mile and 50-mile radial distances for the cases considered in this study. A DF of 10 for the wetwell venting cases results in about an order-of-magnitude reduction.

Table 17. Mean, Total Offsite Economic Costs ($M–2005) per Event within a Circular Area at Specified Radial Distances with Specified Decontamination Factors for the Cases Considered

	Case 2 Base case	**Case 3** Base case with wetwell venting	**Case 3 DF 2**	**Case 3 DF 10**	**Case 3 DF 100**
0-10 miles	220	200	150	89	67
0-50 miles	1,900	1,700	890	270	190

	Case 6 Base case with core spray	**Case 7** Base case with wetwell venting and core spray	**Case 7 DF 2**	**Case 7 DF 10**	**Case 7 DF 100**
0-10 miles	130	71	38	8.0	0.58
0-50 miles	850	480	180	18	0.81

	Case 12 Base case with drywell venting	**Case 12 DF 1,000**	**Case 12 DF 5,000**	**Case 13** Base case with drywell venting and drywell spray	**Case 13 DF 1,000**
0-10 miles	1,400	150	140	1,300	30
0-50 miles	33,000	390	370	33,000	38

	Case 14 Base case with drywell spray	**Case 15** Base case with wetwell venting and drywell spray	**Case 15 DF 2**	**Case 15 DF 10**	**Case 15 DF 100**
0-10 miles	34	100	58	11	0.56
0-50 miles	120	590	240	20	0.70

All of the costs for the six cost categories are summed over the entire offsite area (to a maximum radius of 50 miles) affected by the assumed atmospheric release considered to obtain the total offsite economic costs. As an example of the detailed costs estimates, Table 18 provides the mean cost data for the 50-mile radial distance for Case 12. All costs listed in Table 18 are shown in millions of 2005 dollars.

Table 18. Case 12 Detailed Mean, Economic Model Output

Mean, Total Offsite Economic Cost Measures per Event for the 0-50 mile radial distance	($M–2005)
Population Dependent Nonfarm Decontamination Cost	8,840
Population Dependent Nonfarm Interdiction Cost	21,400
Population Dependent Nonfarm Condemnation Cost	1,190
Farm Dependent Decontamination Cost	224
Farm Dependent Interdiction Cost	277
Farm Dependent Condemnation Cost	84.8
Emergency Phase Cost	1,010
Milk Disposal Cost	20.5
Crop Disposal Cost	309
Total Offsite Economic Costs	**33,300**

Figure 19 shows the mean, total offsite economic costs in millions of 2005 dollars per event for the 10-mile and 50-mile radial distances for all the cases considered. Table 17 and Figure 19 show that a DF of 10 or more for all wetwell venting filtered cases and a DF of 1,000 for all drywell venting filtered cases results in a lower economic costs than their respective unfiltered case.

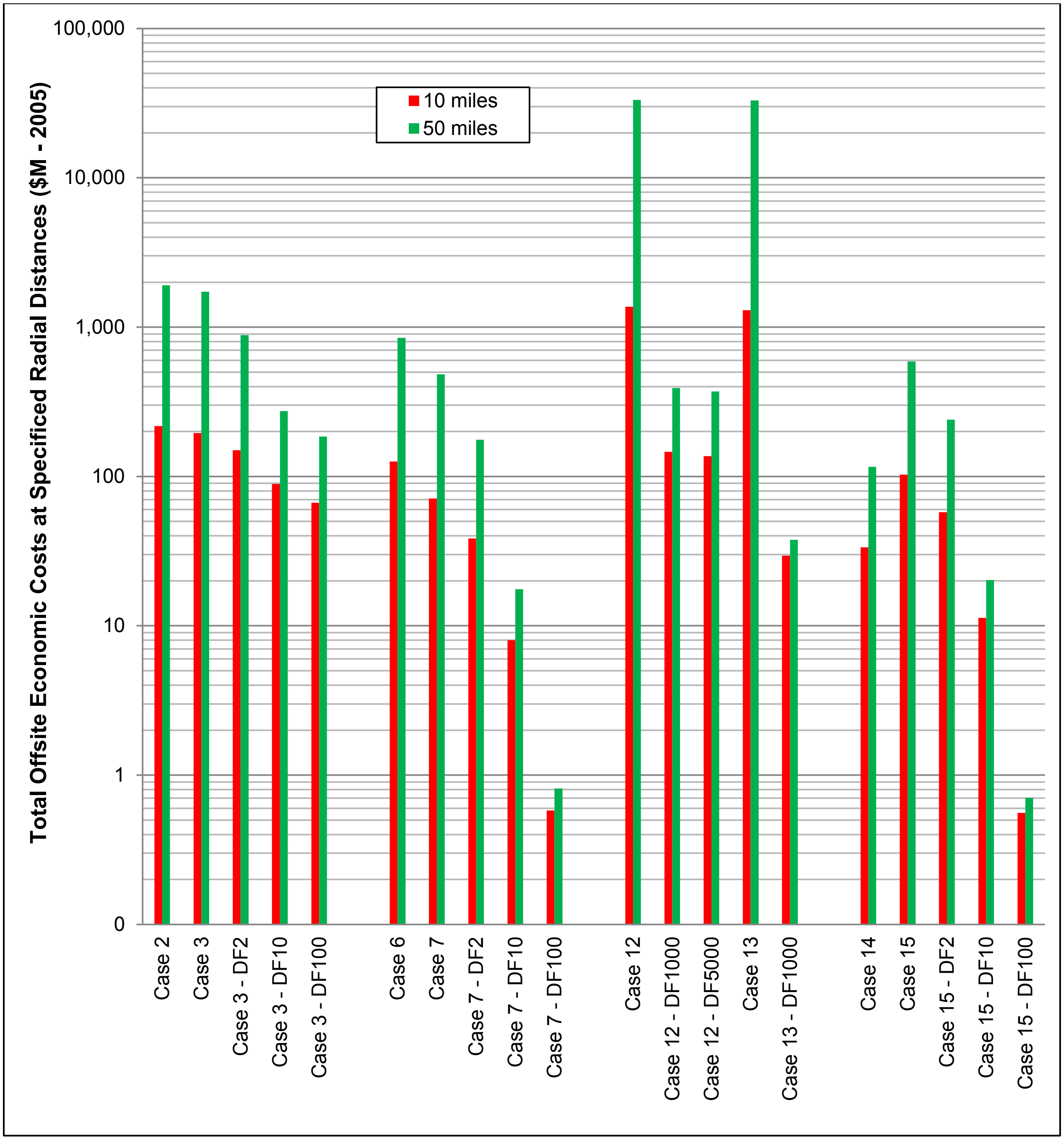


Figure 19. Mean, total offsite economic costs per event within a circular area at specified radial distances with specified decontamination factors for all the cases considered

To better identify which filtered cases have costs that are directly correlated to land contamination, Figure 20 shows the ratio of economic costs to contaminated land area; more specifically, the ratio of the mean, total offsite economic costs in millions of 2005 dollars per event for the 50-mile radial distance to the area of land exceeding the 15 μCi/m^2 of Cs-137 areal concentration, for all the cases considered. The ratio varies from ~3 to ~800. Figure 20 shows that the economic cost computation is more complicated than a constantly proportional relationship to contaminated land area.

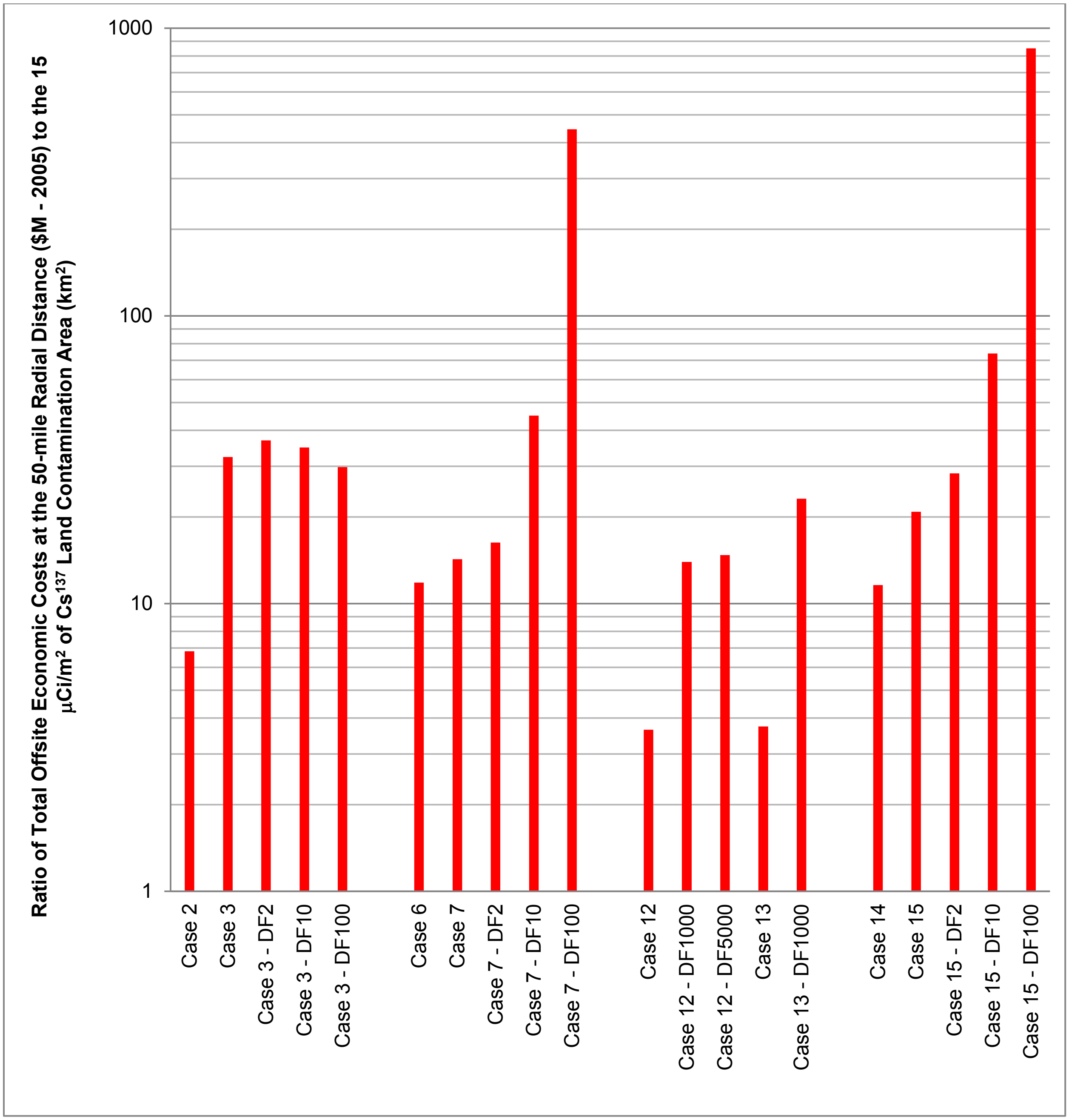


Figure 20. Ratio of mean, total offsite economic costs per event within a circular area of 50-mile radius to the land contamination area exceeding 15 μCi/m^2 of Cs-137 for all the cases considered

3. CONSEQUENCE ANALYSES SUMMARY

The MACCS2 results for this study consider the mitigative measures listed in Table 19, and the benefit of an external filter on the wetwell or drywell vent path. For wetwell venting, Case 3, Case 7, and Case 15 consider a DF associated for the external filter of 2, 10, and 100. For drywell venting, Case 12 and Case 13 consider a DF associated for the external filter of 1,000.

Table 19. Matrix of Scenarios Used in the Consequence Analyses

Case	DC Battery time (16 hours)	Core spray after RPV failure	Drywell spray at 24 hours	Wetwell venting at 60 psig	Main steam line failure	Drywell venting at 24 hours
2	X					
3	X			X		
6	X	X				
7	X	X		X		
12	X				X	X
13	X		X		X	X
14	X		X			
15	X		X	X		

The results of the consequence analyses are presented in terms of public individual fatality risks, land contamination, population dose, and economic costs for each of the cases. All individual fatality risk results are presented as conditional risk (i.e., assuming that the accident occurs), and show the risks to individuals as a result of the accident (i.e., latent cancer fatality [LCF] risk per event or prompt-fatality risk per event). Table 20 shows the results for all four consequence metrics for the eight cases at the 50-mile radial distance and the 15 $\mu Ci/m^2$ contamination threshold with specified DFs for the wetwell and drywell venting cases considered.

The risk metrics are LCF risk and prompt fatality risks to residents in circular regions surrounding the plant. The risk values represent the predicted number of fatalities divided by the population. LCF risks are calculated using the LNT dose-response model. The risks, land contamination, population dose, and economic costs are mean values (i.e., expectation values) over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular region. These risk, population dose, and economic cost metrics account for the distribution of the population within the circular region and for the interplay between the population distribution and the wind rose probabilities.

Table 20. Summary of MACCS2 Results for the 50-mile Radial Distance and the 15 μCi/m^2 Contamination Threshold for the Cases Considered

Event	Case 2 Base case	Case 3 Base case with wetwell venting Unfiltered Filtered DF = 10	Case 6 Base case with core spray	Case 7 Base case with wetwell venting and core spray Unfiltered Filtered DF = 10
Population dose at the 50-mile radius per event (rem)	580,000	460,000 180,000	310,000	240,000 37,000
LCF risk at the 50-mile radius per event	4.8×10^{-5}	3.3×10^{-5} 1.3×10^{-5}	2.5×10^{-5}	1.6×10^{-5} 2.2×10^{-6}
Contaminated area (km^2) for levels exceeding 15 μCi/m^2 per event	280	54 8	72	34 0.4
Total economic cost at the 50-mile radius per event ($M–2005)	1,900	1,700 270	850	480 18

Event	Case 12 Base case with drywell venting Unfiltered Filtered 1 DF = 1,000 Filtered 2 DF = 5,000	Case 13 Base case with drywell venting and drywell spray Unfiltered Filtered DF = 1,000	Case 14 Base case with drywell spray	Case 15 Base case with wetwell venting and drywell spray Unfiltered Filtered DF = 10
Population dose at the 50-mile radius per event (rem)	3,800,000 230,000 210,000	3,900,000 60,000	86,000	280,000 43,000
LCF risk at the 50-mile radius per event	3.2×10^{-4} 1.6×10^{-5} 1.4×10^{-5}	3.3×10^{-4} 3.7×10^{-6}	6.4×10^{-6}	2.1×10^{-5} 2.7×10^{-6}
Contaminated area (km^2) for levels exceeding 15 μCi/m^2 per event	9,200 28 25	8,800 2	10	28 0.3
Total economic cost at the 50-mile radius per event ($M–2005)	33,000 390 370	33,000 38	120	590 20

3.1 Wetwell Venting—Latent Cancer Fatality and Prompt Fatality Risk

For the filtered wetwell venting cases, when a DF is applied to the pathway that flows through the filtered vent (i.e., Case 3—wetwell vent left open), the relationship is sublinear between the inverse of DF and LCF risk. This sublinear behavior is more pronounced at shorter distances. This trend is primarily due to short-term and long-term mitigative actions. For smaller releases, the implementation of offsite protective actions is triggered less often. Thus, doses and LCF risks diminish less than linearly. The offsite protective actions implemented in the MACCS2 model that are responsible for these trends are relocation during the emergency phase and enforcement of the habitability criterion during the long-term phase.

Additionally for Case 3, the wetwell vent path is not the only release pathway to the environment. As a result of the additional environmental release pathway (i.e., the drywell liner failure), the relationship between the assumed DF and the LCF risk contributes to the sublinearity of the LCF risk results.

Case 15 does not produce lower environmental consequences than the unfiltered case (Case 14). However, when a DF of 10 or greater is applied to the wetwell vent pathway to represent the effect of the external filters, the environmental consequences are lowered.

For all cases, the emergency response is very effective within the EPZ (10 miles) during the early phase, so those risks are very small and entirely represent the 0.5 percent of the population that are modeled as refusing to evacuate.

For all wetwell venting cases, except Case 7 and Case 15 each with a DF greater than 10 (where total LCF risks are lowest), the long-term phase LCF risk dominates the total LCF risks for these cases when the LNT dose-response model is used. These long-term risks are controlled by the habitability (return) criterion, which is the dose rate at which residents are allowed to return to their homes following the emergency phase. For Peach Bottom, the State of Pennsylvania's guideline of habitability criterion is a dose rate of 500 mrem/yr.

For filtered wetwell venting Case 7 and Case 15 with a DF greater than 10, the emergency phase LCF risk dominates the total LCF risks. This is due the reduced source term from core spray or drywell spray, respectively. Table 21 shows the percent contribution of the emergency phase LCF risk to the total LCF risk for each of the wetwell venting cases considered for the specified radial distances.

The prompt fatality risks are zero for these cases. This is because the release fractions are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5 percent of the population that are modeled as refusing to evacuate. The largest value of the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers from the plant) is about 0.06 Gy to the red bone marrow. As discussed previously discussed, the red bone marrow is usually the most sensitive organ for prompt fatalities, but the minimum acute dose that can cause an early fatality is about 2.3 Gy. The calculated mean, acute exposures are all well below this threshold.

Table 21. Percent Contribution of the Emergency Phase LCF Risk to the Total LCF Risk for All Wetwell Venting Cases Considered at the Specified Radial Distances

	Case 2 Base case	**Case 3** Base case with wetwell venting	**Case 3 DF 2**	**Case 3 DF 10**	**Case 3 DF 100**
0-10 miles	0%	1%	0.5%	0.5%	0.5%
0-50 miles	15%	40%	40%	40%	40%
	Case 6 Base case with core spray	**Case 7** Base case with wetwell venting and core spray	**Case 7 DF 2**	**Case 7 DF 10**	**Case 7 DF 100**
0-10 miles	0.5%	0.5%	0.5%	0%	1.5%
0-50 miles	35%	30%	30%	35%	70%
	Case 14 Base case with drywell spray	**Case 15** Base case with wetwell venting and drywell spray	**Case 15 DF 2**	**Case 15 DF 10**	**Case 15 DF 100**
0-10 miles	0.5%	0.5%	1%	0.5%	1.5%
0-50 miles	30%	35%	35%	40%	70%

3.2 Drywell Venting—Latent Cancer Fatality and Prompt Fatality Risk

When a DF is applied to the pathway that flow through the drywell filtered vent (i.e., Case 12 and Case 13), the relationship is nonlinear between the inverse of DF and LCF risk.

The drywell vent path is not the only release pathway to the environment. This additional environmental release pathway (i.e., drywell liner failure) influences the relationship between the assumed DF and the LCF risk to be sublinear. The sublinear behavior is more pronounced at shorter distances, primarily due to short-term and long-term mitigative actions (see discussion in Section 3.1).

An interesting observation is that when the LCF risk for the unfiltered Case 12 is compared with that for unfiltered Case 13 (i.e., no DF is applied for an external filter on the drywell vent path), the LCF risks are higher for Case 13 even though containment spray is on. The majority of the source term for these unfiltered cases occurs when the main steam line fails. The two unfiltered cases have similar long-term LCF risk. However, the emergency phase LCF risk for Case 13 is higher. This is attributed to slightly higher short-term LCF risk contributors in the cerium class (e.g., Pu-238 and Pu-239) for acute inhalation dose. The emergency phase accounts for 50–70 percent of the total LCF risk beyond 20 miles for both unfiltered cases.

The emergency response is very effective within the EPZ (10 miles) during the emergency phase, so those risks are very small and entirely represent the 0.5 percent of the population that are modeled as refusing to evacuate.

When an external filter is employed on the vent, the long-term phase risk dominates the total risks for these cases. These long-term risks are controlled by the habitability (return) criterion, which is the dose rate at which residents are allowed to return to their homes following the emergency phase. For Peach Bottom, the State of Pennsylvania’s habitability criterion is a dose rate of 500 mrem/yr.

For the unfiltered cases, the emergency phase risk dominates the total risk due to the main steam line failure. The emergency phase risk is controlled by inhalation doses during the emergency phase as a result of the large iodine release fraction. Table 22 shows the percent contribution of the emergency phase LCF risk to the total LCF risk for each of the drywell venting cases considered for the specified radial distances.

Table 22. Percent Contribution of the Emergency Phase LCF Risk to the Total LCF Risk for All Drywell Venting Cases Considered at the Specified Radial Distances

	Case 12 Base case with drywell venting	Case 12 DF 1,000	Case 12 DF 5,000	Case 13 Base case with drywell venting and drywell spray	Case 13 DF 1,000
0-10 miles	5%	0%	0.5%	5%	0.5%
0-50 miles	50%	30%	30%	55%	30%

For the sensitivity study where a DF of 5,000 is applied for Case 12, there is a sublinear relationship with the filtered Case 12 where a DF of 1,000 is applied. This sublinear relationship is attributed to the additional release pathway. As a result of this additional environmental release pathway (i.e., the drywell liner failure), when a DF ≥1,000 is applied the fraction of the source term that is released through the drywell liner failure dominates the overall source term. Thus, a higher DF has little effect on the LCF risk.

The prompt fatality risks are zero for all cases, except unfiltered Case 13. For those cases that resulted in a zero prompt fatality risk, this is because the release fractions are too low to produce doses large enough to exceed the dose thresholds for early fatalities, even for the 0.5 percent of the population that are modeled as refusing to evacuate. The largest value of the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers from the plant) for these cases is about 0.8 Gy to the red bone marrow (i.e., unfiltered Case 12). As discussed previously, the red bone marrow is usually the most sensitive organ for prompt fatalities, but the minimum acute dose that can cause an early fatality is about 2.3 Gy. The calculated mean, acute exposures are all well below this threshold.

For unfiltered Case 13, there is a nonzero mean, individual prompt fatality risk per event at the 2-mile and 2.5-mile radial distances. Beyond 2.5 miles, all prompt fatality risk is zero. For unfiltered Case 13, the mean, acute exposure for the closest resident (i.e., 0.5 to 1.2 kilometers from the plant) is about 1.0 Gy to the red bone marrow. While this is below the red bone marrow threshold for an early fatality, 0.5 percent of the MACCS2 weather trials produced an acute exposure greater than the threshold. As a result of these few weather trials, a nonzero mean prompt fatality risk was observed. Based on this observation and since the mean, prompt fatality risk for the 2-mile and 2.5-mile radial distances are so low, the mean, individual prompt fatality risk per event at these distances are considered essentially zero.

3.3 Land Contamination

Land areas contaminated above a threshold level can be calculated several ways in MACCS2, the simplest of which is to report land areas that exceed activity levels per unit area for one or more of the isotopes. This is the approach used here, and areas are reported using the same threshold levels of Cs-137 as were reported following the Chernobyl accident [13].

A relatively small number of the isotopes that could potentially be released from a nuclear reactor are radiologically important and require effort to decontaminate. Among these are Cs-134 and Cs-137, which have half-lives of 2 years and 30 years, respectively, and are important isotopes for a typical nuclear reactor accident in terms of decontamination.

There is an inherently nonlinear relationship between the size of the source term and land contamination area. This is primarily because land contamination area is calculated using a threshold (i.e., land areas are only tabulated when they exceed a threshold ground concentration). It turns out that the relationship between the inverse of DF (i.e., the quantity released) and land contamination area is superlinear for all filtered cases.

The mean contaminated area for specified Cs-137 contamination levels for all cases show the same trends when a DF is applied to the filtered cases. When the unfiltered unvented case (e.g., Case 2) is compared with the filtered case (e.g., Case 3), a DF of 10 or 100 for wetwell venting and a DF 1,000 for drywell venting results in a several order-of-magnitude reduction in land contamination area.

3.4 Population Dose

The relationship between population dose and inverse DF is sublinear because less remedial action is taken at lower contamination levels. For the cases considered, a DF of 10 or more for all wetwell venting filtered cases and a DF of 1,000 for all drywell venting filtered cases result in lower population doses than their respective unfiltered cases. The discussion for individual LCF and prompt fatality risks in 3.1 and 3.2 apply for population dose too.

One difference is that the population dose results include public doses from the ingestion pathway and doses to offsite decontamination workers; LCF risks do not include either of these doses. Ingestion is considered during the long-term phase from contaminated food and water. The ingestion pathway accounts for:

- 10–20 percent of the population dose for the wetwell venting unfiltered cases considered
- 15–30 percent of the population doses for the wetwell venting filtered cases considered
- 5 percent of the population doses for the drywell venting unfiltered cases considered
- 20–30 percent of the population doses for the drywell venting filtered cases considered

3.5 Economic Costs

The isotopic composition of the source term is one element that impacts the costs of decontamination. Some isotopes require no decontamination at all while others can be more difficult to decontaminate. The purpose of decontamination is to remove enough of the cesium to reduce the level of radiation from ground and building surfaces to acceptable levels (i.e., habitability limit).

Implementation of decontamination, which along with the associated interdiction of land is the dominant contributor to the overall economic costs, depends on whether or not the habitability criterion is exceeded. Remedial actions considered in the long-term phase depend on two criteria: habitability and farmability. Both of these criteria are based on contamination thresholds, which lead to inherently nonlinear relationships between source term magnitude and economic costs. Thus applying a DF to represent an external filter does not result in a linear relationship between release (i.e., reciprocal of DF) and economic costs.

A DF of 10 for the wetwell venting cases results in an order-of-magnitude reduction in economic cost. For the cases considered, a DF of 10 or more for all wetwell venting filtered cases and a DF of 1,000 for all drywell venting filtered cases results in a lower economic costs than their respective unfiltered cases.

4. CONCLUSIONS

These MACCS consequence analyses show a clear benefit in applying an external filter to either the wetwell or drywell vent path[8]. More specifically:

- The filtered cases with an external filter on either the wetwell or drywell vent path and a DF ≥10 for wetwell venting or a DF ≥1,000 for drywell venting results in a lower conditional latent cancer fatality [LCF] risk (i.e., 40–95 percent reduction) when compared to the unfiltered cases.
- The filtered cases with an external filter on either the wetwell or drywell vent path and a DF ≥10 for wetwell venting or a DF ≥1,000 for drywell venting results in a lower population dose (i.e., 50–95 percent reduction) when compared to the unfiltered cases. Unlike the LCF risk calculations, the population dose includes public doses from the ingestion pathway and doses to offsite decontamination workers.
- All the filtered cases with an external filtered vent path , results in a several order-of-magnitude reduction in Cs-137 land contamination.
- For all cases considered, the conditional prompt fatality risk is either zero or essentially zero.
- For the cases considered, a DF ≥10 for all wetwell venting filtered cases and a DF ≥1,000 for all drywell venting filtered cases results in lower economic costs (i.e., >60 percent to orders of magnitude reduction) than their respective unfiltered cases.

When a DF is applied to a filtered vent path, the LCF risk, population dose, contaminated land area, and economic consequence results are all nonlinearly related to the inverse of the DF (which represents the release magnitude). The relationship is sublinear between the inverse of DF and LCF risk or population dose. This relationship is sublinear because less remedial action is taken at lower contamination levels. In some cases, it is also sublinear because a portion of the release bypasses the filter vent path. The relationship between the inverse of DF and land contamination area is observed to be superlinear, because in this analysis land contamination is defined as exceeding particular thresholds of Cs-137 areal concentration. Lastly, economic costs are dominated by the implementation of decontamination, which depends on whether or not the habitability or farmability criterion is exceeded. Since habitability and farmability criteria are based on contamination thresholds, there is an inherently nonlinear relationship between source term magnitude and economic costs.

[8] With the exception that the external filter was not beneficial for a DF=2 for the Case 15 with drywell spray and wetwell vent path, compared to Case 14 with drywell spray and no venting.

5. REFERENCES

[1] U.S. Nuclear Regulatory Commission (NRC). NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Washington D.C.: NRC, 2004.

[2] U.S. Nuclear Regulatory Commission. NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," Washington D.C.: NRC, 1997

[3] K. McFadden, N. E. Bixler, Lee Eubanks, R. Haaker, "WinMACCS, a MACCS2 Interface for Calculating Health and Economic Consequences from Accidental Release of Radioactive Materials into the Atmosphere User's Guide and Reference Manual for WinMACCS Version 3", DRAFT NUREG/CR.

[4] U.S. Nuclear Regulatory Commission. Draft NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report: Draft Report for Comment," Washington D.C.: NRC, January 2012.

[5] U.S. Nuclear Regulatory Commission. NUREG/CR-6613, "Code Manual for MACCS2: Volume 1, User's Guide," Washington D.C.: NRC, 1997.

[6] U.S. Nuclear Regulatory Commission. Regulatory Guide 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Washington D.C.: NRC, November 1982.

[7] U.S. Nuclear Regulatory Commission. Draft NUREG/CR-7110, Volume 1, "State-of-the-Art Reactor Consequence Analyses Project—Volume 1: Peach Bottom Integrated Analysis," Washington D.C.: NRC, January 2012.

[8] U.S. Nuclear Regulatory Commission. "Meeting with Sandia National Laboratories and an Expert Panel on MELCOR/MACCS Codes in Support of the State of the Art Reactor Consequence Analysis Project," Washington D.C.: NRC, September, 2006. Agencywide Documents Access and Management System (ADAMS) Accession No. ML062500078.

[9] International Commission on Radiological Protection (ICRP). ICRP 26, "Recommendations of the International Commission on Radiological Protection," Volume 1, No. 3, Pergamon Press Elmsford, NY, 1977.

[10] International Commission on Radiological Protection. ICRP 30, "Limits for Intakes of Radionuclides by Workers," Volume 6, No. 2/3, Pergamon Press Elmsford, NY, 1981.

[11] U.S. Environmental Protection Agency (EPA). EPA 402-R-99-001, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides—Federal Guidance Report 13," Washington D.C.: EPA, September 1999.

[12] U.S. Nuclear Regulatory Commission. NUREG/CR-6525, Rev. 1, SAND2003-1648P, "SECPOP2000: Sector Population, Land Fraction, and Economic Estimation Program," Washington D.C.: NRC, 2003.

[13] International Atomic Energy Agency (IAEA). IAEA-TECDOC-1240, "Present and Future Environmental Impact of the Chernobyl Accident," Vienna, Austria: IAEA, August 2001.

Enclosure 5c
Probabilistic Risk Evaluation

CONTENTS

1. INTRODUCTION

A risk evaluation was performed to estimate the reduction in risk resulting from the installation of a severe accident (SA) venting system in a boiling-water reactor (BWR) with either a Mark I or Mark II containment design. This information provides a major input to the regulatory and backfit analyses of the SA venting system. In addition, the risk evaluation discusses accident sequences where the inclusion of filters to the SA venting system is and is not beneficial, as directed by the Commission in a staff requirements memorandum (SRM) (M120807B) issued on August 24, 2012, following a staff briefing held August 7, 2012, on the status of actions taken in response to lesson learned from the Fukushima Dai-ichi accident.

The purpose of an SA venting system is to prevent an uncontrolled large release of radioactive material during a severe accident as a result of containment failure due to overpressurization from the buildup of steam and noncondensable gases generated during core degradation. An SA venting system should significantly reduce the amount of radioactive material released from the containment when compared to an uncontrolled release. An SA venting system is different than the reliable, hardened venting system mandated by Order EA-12-050, as shown in Table 1.

Table 1. Differences Between a Severe Accident Venting System and a Reliable Hardened Venting System

Characteristic	Severe Accident Venting System	Reliable Hardened Venting System
Purpose	Prevent containment overpressurization failure after core damage	Provide a pathway for decay heat removal in order to prevent core damage
Period of Use	After core damage	Prior to core damage
Vented Materials	Radioactive steam and noncondensable gases resulting from core damage	Mildly radioactive steam (limited to activity contained in the reactor coolant system that exists during normal operations)
Release of Radioactive Materials to the Environment	Small if the severe accident venting system operates as designed to prevent containment overpressurization failure, includes a filter or other means to scrub fission products, and other containment failure modes (such as liner melt-through) are prevented Otherwise, potentially large	Very small if the reliable, hardened vent operates as designed to prevent core damage

The following sections discuss risk insights related to SA venting obtained from previous analyses, explain the technical approach used, list the assumptions used, describe the delineation of post-core-damage accident sequences pertaining to SA venting, provide the quantitative information used, and present the results of the risk evaluation.

2. RISK INSIGHTS FROM PREVIOUS ANALYSES

As an initial step in the risk evaluation, the staff reviewed information from the individual plant examinations completed in response to Generic Letter 88-20, license amendment requests for integrated leak rate testing (ILRT) extensions, and severe accident mitigation alternatives (SAMA) analyses submitted with license renewal requests. The purpose of this review was to gain insight into the causes and likelihood of containment failure and to understand how SA venting has been considered in previous probabilistic risk assessments (PRAs) and risk-informed applications. The following sections summarize the information obtained.

2.1 Individual Plant Examinations

The results of individual plant examinations (IPEs) indicated that the likelihood of Mark I and Mark II containment failure due to severe accident phenomena is not insignificant. Figure 1 illustrates the range of conditional containment failure probabilities for BWR Mark I containments as reported in the IPE submittals.

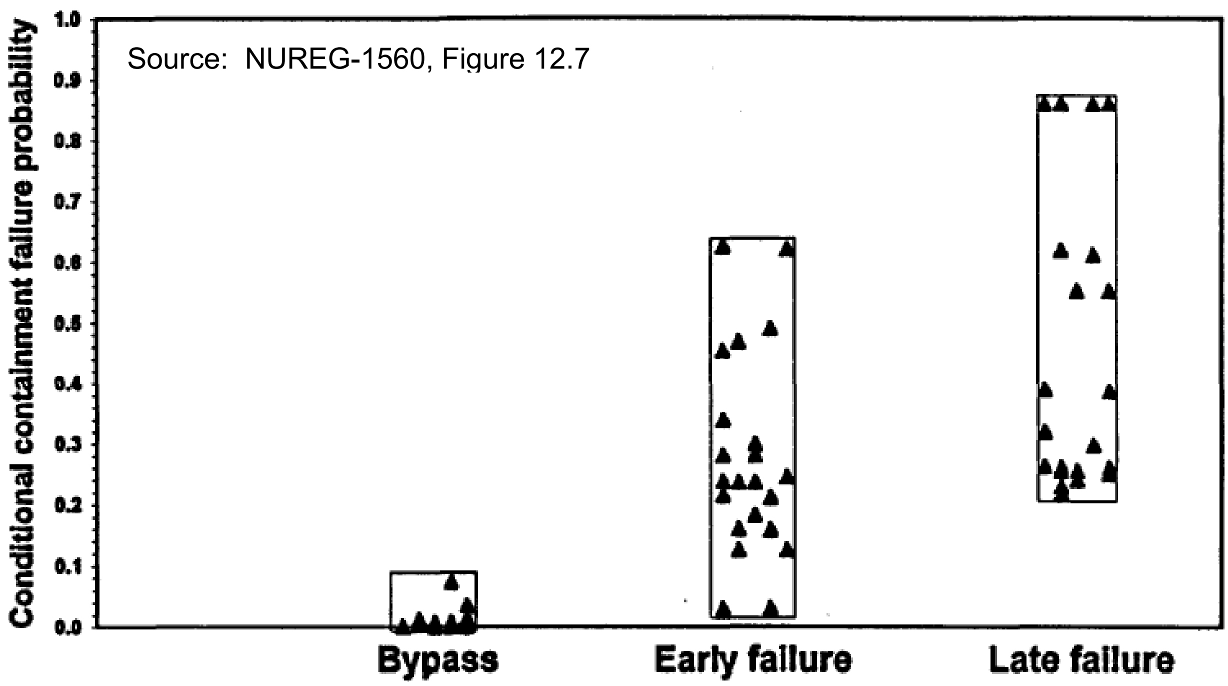


Figure 1. Reported IPE conditional probabilities of failure for BWR Mark I containments

With respect to the likelihood of BWR Mark I containment failure modes, NUREG-1560 indicates that liner melt-through is the most important contributor to early containment failure. Overpressurization failures are generally associated with late containment failure, as discussed in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance":

> Because of a high containment pressure capability and the energy absorbing capacity of the suppression pool, a typical Mark I containment is unlikely to fail because of overpressure early in the accident sequence. However, accidents in which both containment heat removal and containment venting are not available

or inadequate (such as occurs in some sequences in which the reactor vessel fails at high pressure, or in some anticipated transient without scram (ATWS) sequences) can cause early containment failure. For these sequences, containment may fail either before or at vessel breach because of the high containment pressures.

As noted in Table 10.4 of NUREG-1560, the design pressures for BWR Mark I containments range from 56 to 62 psig, and the median failure pressures estimated for the IPEs range from 98 to 190 psig.

Figure 2 illustrates the range of conditional containment failure probabilities for BWR Mark II containments as reported in the IPE submittals.

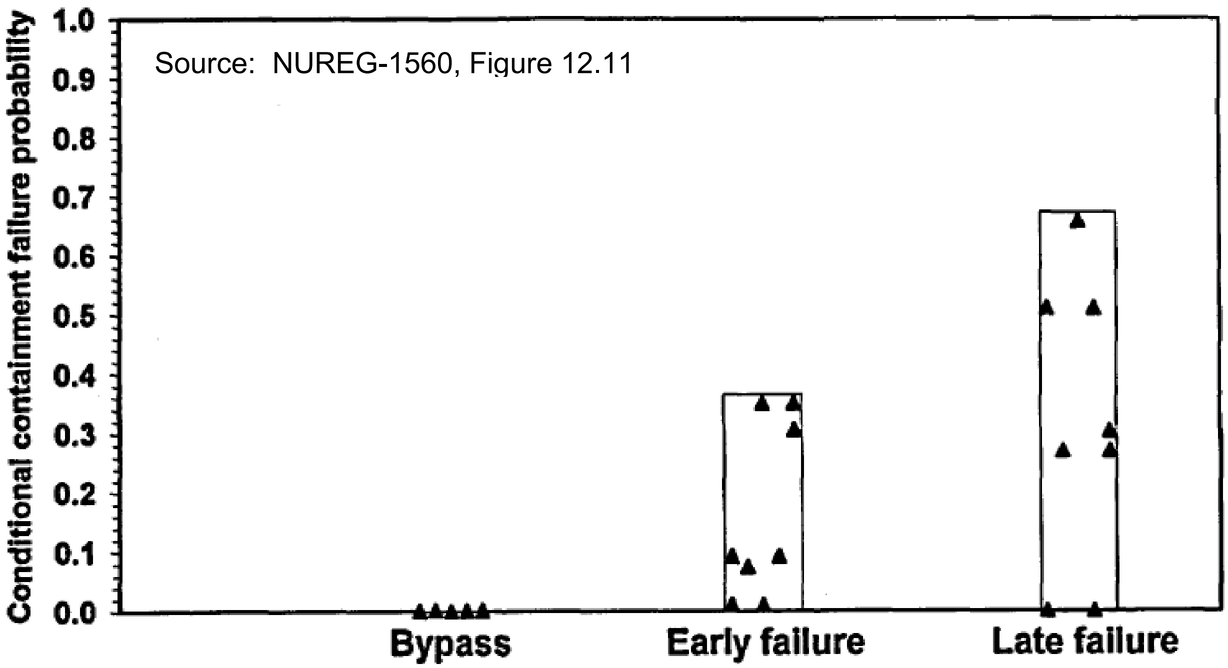


Figure 2. Reported IPE conditional probabilities of failure for BWR Mark II containments

NUREG-1560 states that containment overpressure failure caused by a loss of containment heat removal (primarily during ATWS sequences) is important in most Mark II IPE analyses, and that rapid pressure and temperature increases at the time of reactor vessel failure are significant in only a few Mark II IPE analyses. Specific plant features play an important role in accident progression in Mark II containments. As noted in Table 10.4 of NUREG-1560, the design pressures for BWR Mark II containments range from 45 to 55 psig, and the median failure pressures estimated for the IPEs range from 140 to 191 psig.

2.2 Integrated Leak Rate Test Extensions

In recent years, a number of BWR Mark I and Mark II plants have applied for and been granted extensions of their ILRT intervals. ILRTs are conducted to satisfy the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Relevant to SA venting, license amendments requests for

ILRT extension provide information that can be used to estimate conditional containment failure probabilities. A review of this information was made to determine how licensees' understanding of conditional containment failure probability has evolved since completion of their IPEs. Table 2 summarizes the contributions to conditional containment failure probability for selected ILRT extension requests. Note that the contributions to conditional containment failure probability by specific accident-induced failures (overpressurization, liner melt-through, etc.) are not provided in ILRT extension requests.

Table 2. Conditional Containment Failure Probabilities from ILRT Extensions

Plant	Type	ILRT Interval	Accident Phenomena	Bypass (ISLOCA)	Isolation Failures	Total CCFP
Cooper	Mark I	3 in 10y 1 in 10y 1 in 15y	94.6% 94.6% 94.6%	0.0% 0.0% 0.0%	1.0% 1.0% 1.0%	95.6% 95.6% 95.6%
Nine Mile Point 1	Mark I	3 in 10y 1 in 10y 1 in 15y	62.4% 62.4% 62.4%	2.7% 2.7% 2.7%	9.7% 9.7% 9.8%	74.8% 74.9% 74.9%
Peach Bottom	Mark I	3 in 10y 1 in 10y 1 in 15y	61.1% 61.1% 61.1%	2.4% 2.4% 2.4%	2.7% 3.4% 4.0%	66.2% 67.0% 67.5%
Pilgrim	Mark I	3 in 10y 1 in 10y 1 in 15y	97.7% 97.7% 97.7%	0.6% 0.6% 0.6%	0.0% 0.1% 0.1%	98.3% 98.3% 98.4%
Vermont Yankee	Mark I	1 in 10y 1 in 15y	86.8% 86.8%	1.1% 1.1%	0.1% 0.2%	88.0% 88.1%
LaSalle	Mark II	3 in 10y 1 in 10y 1 in 15y	82.9% 82.9% 82.9%	2.4% 2.4% 2.4%	0.4% 0.6% 0.8%	85.7% 85.9% 86.1%
Limerick	Mark II	3 in 10y 1 in 10y 1 in 15y	62.4% 62.4% 62.4%	1.3% 1.3% 1.3%	0.7% 1.5% 2.0%	64.4% 65.2% 65.7%

2.3 <u>Severe Accident Mitigation Alternatives</u>

Table 3 provides a breakdown by plant type of how filtered containment vent (FCV) systems have been considered in SAMA analyses. SAMA analyses have used two approaches when considering FCV systems. A screening approach compares the cost of a FCV system to the monetized baseline risk of the plant. This approach is conservative since it assumes that installation of a FCV system will completely eliminate all plant risk. A detailed approach attempts to approximate the risk reduction that would be achieved by installing a FCV system by adjusting the source terms that are used in a Level 3 PRA. Three early SAMA analyses stated that they had considered FCV systems, but the discussion does not describe the approach taken to assess the risk reduction or provide the numerical results. To date, no SAMA analysis has determined that FCV systems are cost justified.

Table 3. Consideration of Filtered SA Venting in SAMA Analyses, as of February 2012						
Plant Type	**FCV Not Considered**	**FCV Considered (Screening Analysis)**	**FCV Considered (Detailed Analysis)**	**License Renewal Granted, but Limited SAMA**	**License Renewal Application Not Submitted**	**Total**
BWR Mark I	5	11	5	1	1	23
BWR Mark II	1	3		2	2	8
BWR Mark III			1		3	4
PER large dry containment	22	10	14		9	55
PWR subatmospheric containment			5			5
PWR ice condenser		2	4		3	9
Totals	28	26	29	3	18	104

3. TECHNICAL APPROACH

The addition of an SA venting system does not change a plant's core damage frequency (CDF); rather, it affects the frequency of releases to the environment resulting from core damage and also the consequences of these releases. Release frequencies are estimated using Level 2 PRA methods, and consequences are estimated using Level 3 PRA methods. The staff has developed three proof-of-concept Level 2 Standardized Plant Analysis of Risk (SPAR) models, but does not routinely use them to support regulatory decisionmaking. In addition, the staff does not have any Level 3 PRA models. As a result, a simplified event tree was constructed to estimate the frequencies of the MELCOR scenarios developed to support the assessment of SA venting, as described in Enclosure 5a. Coupled with the MACCS2 consequence results, described in Enclosure 5b, developed for each MELCOR scenario, this simplified event tree provides the information needed to assess the reduction in risk resulting from the installation of an SA venting system.

There are a variety of ways to design an SA venting system, depending on where the vent attached (wetwell or drywell), how the vent is actuated (manually by the operator or passively using a rupture disk), and whether the SA venting system has a filter. The simplified event tree structure used to estimate release sequence frequencies was designed to allow assessment of a wide range of SA vent system designs. Specifically, the same simplified event tree structure was used to assess nine hypothetical plant modifications ("mods"), which are defined in Table 4.

Table 4. Hypothetical Plant Modifications Assessed in the Risk Evaluation

Plant Modification Identifier	SA Vent Filter	SA Vent Location	SA Vent Actuation
Mod 0 (current situation)	n/a	None	n/a
Mod 1	No	Wetwell	Manual
Mod 2	No	Wetwell	Passive
Mod 3	No	Drywell	Manual
Mod 4	No	Drywell	Passive
Mod 5	Yes	Wetwell	Manual
Mod 6	Yes	Wetwell	Passive
Mod 7	Yes	Drywell	Manual
Mod 8	Yes	Drywell	Passive

The first two characteristics that define the plant modification (the presence of a filter and the vent location) only affect the consequences associated with the release sequences defined in the simplified event tree. For example, the addition of a filter or venting through the wetwell would reduce the consequences. The third characteristic (vent actuation method) only affects the frequency of the release sequences. For example, utilization of a passive mechanism (e.g., rupture disk) to actuate the vent path is expected to be more reliable than operator action, and therefore, the frequency of large releases is expected to decrease more when a passive vent is used than when relying on manual operation.

In order to support the regulatory and backfit analyses, the following risk metrics were estimated for each hypothetical plant modification:

- 50-mile population does risk (person-rem/reactor-year)
- 50-mile offsite cost risk ($/reactor-year)
- onsite worker dose risk (person-rem/reactor-year)
- onsite cleanup and decontamination cost ($/reactor-year)

Using the risk metrics identified above, the risk reductions (relative to Mod 0, which is the current situation) due to implementation of each hypothetical plant modification (Mod 1 through Mod 8) were estimated. These risk reductions are used as an input to the regulatory and backfit analyses:

- Reduction in 50-mile population dose risk (Δperson-rem/reactor-year)
- Reduction in 50-mile offsite cost risk (Δ$/reactor-year)
- Reduction in onsite worker dose risk (Δperson-rem/reactor-year)
- Reduction in onsite cleanup and decontamination cost (Δ$/reactor-year)

In addition to the risk metrics listed above, a risk metric pertaining to land contamination was estimated. It should be noted that the impact of accident releases on land contamination that occurs within 50 miles of the site is included in the offsite cost risk. A direct measure of land contamination risk (including contaminated land that is farther than 50 miles from the site) is desirable to gain perspective on the risk reductions that can be achieved through implementation of the hypothetical plant modifications. Mathematically, risk is defined as the sum of the product of the release sequence frequency and the consequence of the release:

$$R = \sum_i f_i c_i$$

where R denotes the risk, f_i denotes the frequency of the ith release sequence, and c_i denotes the consequences associated with the ith release sequence, and the summation is taken over all release sequences. One measure of the consequences of a release with respect to land contamination is the amount of area (in km^2) that is contaminated above 15 μCi/m^2 with cesium-137 (^{137}Cs)[1]. Using this consequence measure, land contamination risk has the units of km^2/ry (square kilometers/reactor year), which is rather difficult to interpret. A potentially more insightful risk metric is conditional contaminated land area (CCLA), as defined by:

$$CCLA = \frac{\sum_i f_i c_i}{\sum_i f_i} = \frac{R}{CDF}$$

That is, the CCLA is the frequency-weighted average area contaminated above 15 μCi/m^2 with ^{137}Cs, conditional on the occurrence of a core-damage accident. Accordingly, a reduction in CCLA due to implementation of one of the hypothetical SA vent modifications measures the effectiveness of that modification with respect to reducing land contamination.

---

1 Annex I to IAEA TECDOC-1240, "Present and future environmental impact of the Chernobyl accident," zoned land surrounding the Chernobyl site according to the level of radionuclide soil deposition. Land that was contaminated above 15 μCi/m^2 with ^{137}Cs was called an "obligatory (subsequent) resettlement zone." Permanent residence and the production of commodities within the obligatory (subsequent) resettlement zone is forbidden.

3.1 Assumptions

The following assumptions were used to conduct the risk evaluation:

(1) The existing regulatory analysis guidance provided in NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” and NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” have been used. Accordingly:

   (a) The risk evaluation was developed on a “per-reactor” basis

   (b) Multi-unit accidents were not addressed

   (c) Accidents involving spent fuel (stored either in the spent fuel pool or in dry casks) were not addressed

(2) Except for bypass sequences (ISLOCAs (interfacing-systems loss-of-coolant accidents) and large external hazards that directly fail the containment), severe accident containment venting is always required to prevent a containment overpressurization failure. This assumption follows from the results of the MELCOR calculations performed to support the regulatory and backfit analyses of SA venting.

(3) No credit was given for recovering offsite power if core damage was caused by an external hazard (seismic event, high winds, etc.).

(4) The consequences of accident sequence that result in radioactive releases are reasonably approximated by determining the consequences of station blackout (SBO) sequences.

(5) The reactor core isolation cooling (RCIC) system operates for 16 hours (16-hour battery depletion).

(6) If the SA venting system includes a filter, then it has a decontamination factor of 10 (wetwell venting) or 1,000 (drywell venting)..

(7) If an accident sequence involves failure to open the SA vent or containment bypass (such as an ISLOCA) then use of a portable pump to provide core spray or drywell spray following core damage is precluded due to a harsh work environment (high dose rates, high temperatures, etc.).

3.2 Delineation of Accident Sequences

The simplified release event tree (Figure 3) traces the accident progression starting from the onset of core damage. The initial event tree headings parse the total CDF according to the type of initiating event and core-damage sequence. Subsequent event tree headings consider operation of the SA vent and the availability of a water supply to the drywell. Each sequence has been assigned to a unique containment status:

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

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

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

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

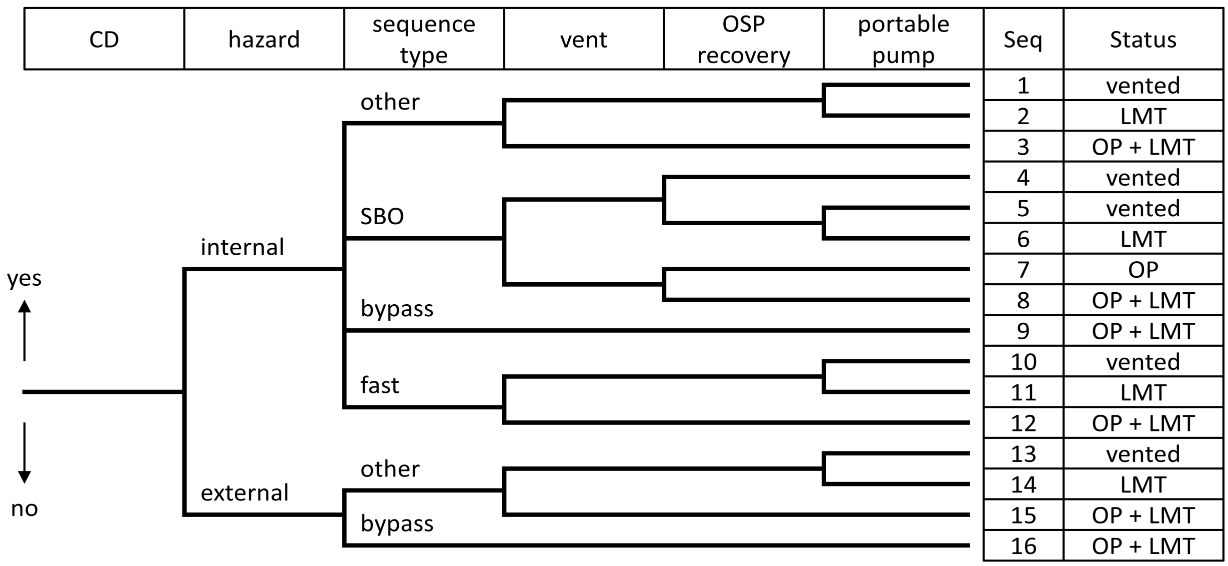


Figure 3. Simplified release event tree

3.2.1 List of Top Events

The release event tree consists of six event tree headings (top events), which are described in the following sections.

- Event “CD”: 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.

- Event “hazard”: 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.

- Event “sequence type”: Partitions core-damage sequences according to their timing or influence on containment integrity. For internal hazards:

  - 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 ISLOCAs). 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 (MLOCAs), large LOCAS (LLOCAs), and anticipated transients without scram (ATWS)) and, thus, reduce the available time for the operator to manually open the SA vent.

  For external hazards:

  - 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.

- Event “vent”: Identifies if the SA vent is opened.

- Event “OSP recovery”: Identifies if offsite power is recovered.

- Event “portable pump”: 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.

3.2.2 List of Sequences

The release event tree delineates 16 post-core-damage accident sequences, which are summarized in the following paragraphs.

- Sequence 1 (status “vented”): Following core damage caused by an internally initiated sequence that does not involve SBO, ISLOCAs, or quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the emergency core cooling system (ECCS)) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). However, a portable pump is successfully installed and operated to provide water to the drywell floor, thereby preventing liner melt-through.

- Sequence 2 (status “LMT”): Following core damage caused by an internally initiated sequence that does not involve SBO, ISLOCAs, or quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the ECCS) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). Moreover, a portable pump to provide water to the drywell floor is either not installed or fails. As a result, liner melt-through occurs.

- Sequence 3 (status “OP + LMT”): Following core damage caused by an internally initiated sequence that does not involve SBO, ISLOCAs, or quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent remains closed and the containment fails due to overpressurization. In-plant equipment (such as the ECCS) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through occurs.

- Sequence 4 (status “vented”): Following core damage caused by an internally initiated SBO sequence, the SA vent is opened, thereby preventing containment overpressurization failure. Offsite power is recovered, which allows the use of in-plant equipment (such as the condensate system) to provide water to the drywell floor and avoid liner melt-through.

- Sequence 5 (status “vented”): Following core damage caused by an internally initiated SBO sequence, the SA vent is opened, thereby preventing containment overpressurization failure. Offsite power is not recovered, which prevents the use of in-plant equipment to provide water to the drywell floor. However, a portable pump is successfully installed and operated to provide water to the drywell floor, thereby preventing liner melt-through.

- Sequence 6 (status “LMT”): Following core damage caused by an internally initiated SBO sequence, the SA vent is opened, thereby preventing containment overpressurization failure. Offsite power is not recovered, which prevents the use of in-plant equipment to provide water to the drywell floor. Moreover, a portable pump to provide water to the drywell floor is either not installed or fails. As a result, liner melt-through occurs.

- Sequence 7 (status “OP”): Following core damage caused by an internally initiated SBO sequence, the SA vent remains closed and the containment fails due to overpressurization. Offsite power is recovered, which allows the use of in-plant equipment (such as the condensate system) to provide water to the drywell floor and avoid liner melt-through.

- Sequence 8 (status “OP + LMT”): Following core damage caused by an internally initiated SBO sequence, the SA vent remains closed and the containment fails due to overpressurization. Offsite power is not recovered, which prevents the use of in-plant equipment to provide water to the drywell floor. Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through also occurs.

- Sequence 9 (status “OP + LMT”): Core damage occurs due to an internally initiated ISLOCA sequence. Venting the containment is not necessary because overpressurization cannot occur (the steam and noncondensable gases caused by core degradation pass through the ISLOCA and, hence, bypass the containment). The risk evaluation assumes that the consequences resulting from containment bypass are the same as the consequences resulting from containment overpressurization, followed by liner melt-through. Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through also occurs.

- Sequence 10 (status “vented”): Following core damage caused by an internally initiated, quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the ECCS) is assumed to be unavailable due to equipment failure or a nonrecoverable loss of offsite power (if it was available, core damage would not have occurred in the first place). However, a portable pump is successfully installed and operated to provide water to the drywell floor, thereby preventing liner melt-through. This sequence is similar to Sequence 1; however, there is less available time to open the SA vent.

- Sequence 11 (status “LMT”): Following core damage caused by an internally initiated, quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the ECCS) is assumed to be unavailable due to equipment failure or a nonrecoverable loss of offsite power (if it was available, core damage would not have occurred in the first place). Moreover, a portable pump to provide water to the drywell floor is either not installed or fails. As a result, liner melt-through occurs. This sequence is similar to Sequence 2; however, there is less available time to open the SA vent.

- Sequence 12 (status “OP + LMT”): Following core damage caused by an internally initiated, quickly developing sequences (e.g., MLOCA, LLOCA, or ATWS), the SA vent remains closed and the containment fails due to overpressurization. In-plant equipment (such as the ECCS) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through also occurs. This sequence is similar to Sequence 3; however, there is less available time to open the SA vent.

- Sequence 13 (status “vented”): Following core damage caused by an externally initiated sequence that does not involve containment bypass, the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the ECCS) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). However, a portable pump is successfully installed and operated to provide water to the drywell floor, thereby preventing liner melt-through. This sequence is similar to Sequence 1; however, it is an external hazard sequence rather than an internal hazard sequence.

- Sequence 14 (status “LMT”): Following core damage caused by an externally initiated sequence that does not involve containment bypass, the SA vent is opened, thereby preventing containment overpressurization failure. In-plant equipment (such as the ECCS) is assumed to be unavailable due to equipment failure or a nonrecoverable loss of offsite power (if it was available, core damage would not have occurred in the first place). Moreover, a portable pump to provide water to the drywell floor is either not installed or fails. As a result, liner melt-through occurs. This sequence is similar to Sequence 2; however, it is an external hazard sequence rather than an internal hazard sequence.

- Sequence 15 (status “OP + LMT”): Following core damage caused by an externally initiated sequence that does not involve containment bypass, the SA vent remains closed and the containment fails due to overpressurization. In-plant equipment (such as the ECCS) is assumed to be unavailable (if it was available, core damage would not have occurred in the first place). Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through also occurs. This sequence is similar to Sequence 3; however, it is an external hazard sequence rather than an internal hazard sequence.

- Sequence 16 (status “OP + LMT”): Core damage occurs due to an externally initiated sequence that involves containment bypass. Venting the containment is not necessary because overpressurization cannot occur (the steam and noncondensable gases caused by core degradation bypass the containment). The risk evaluation assumes that the consequences resulting from containment bypass are the same as the consequences resulting from containment overpressurization, followed by liner melt-through. Moreover, use of a portable pump to provide water to the drywell floor is precluded since the operator cannot access areas of the plant needed to install the pump and associated equipment. As a result, liner melt-through also occurs. This sequence is similar to Sequence 9; however, it is an external hazard sequence rather than an internal hazard sequence.

3.2.3 Mapping Sequences to MELCOR/MACCS2 Calculations

As previously discussed, each sequence in the simplified release event tree has been assigned to a unique containment status. This mapping has been used, along with the definitions of the hypothetical plant modifications, to determine the specific MELCOR/MACCS2 (Enclosures 5a and 5b) calculation that applies to each sequence as shown in Table 5.

<table>
<tr><th colspan="8">Table 5. Mapping to Release Sequences to MELCOR/MACCS2 Calculations</th></tr>
<tr><th colspan="4">Modification Description</th><th colspan="4">Release Sequence Status</th></tr>
<tr><th>Mod</th><th>Filter</th><th>Location</th><th>Actuation</th><th>Vented
• Vent: open
• DW: wet
• Seq: 1, 4, 5, 10, and 13</th><th>LMT
• Vent: open
• DW: dry
• Seq: 2, 6, 11, and 14</th><th>OP
• Vent: closed
• DW: wet
• Seq: 7</th><th>OP + LMT
• Vent: closed
• DW: dry
• Seq: 3, 8, 9, 12, 15, and 16</th></tr>
<tr><td>0</td><td>n/a</td><td>n/a</td><td>None</td><td>n/a</td><td>n/a</td><td>Case 6</td><td>Case 2</td></tr>
<tr><td>1</td><td>No</td><td>Wetwell</td><td>Manual</td><td rowspan="2">Case 7 or 15 (no filter)</td><td rowspan="2">Case 3 (no filter)</td><td rowspan="2">Case 6</td><td rowspan="2">Case 2</td></tr>
<tr><td>2</td><td>No</td><td>Wetwell</td><td>Passive</td></tr>
<tr><td>3</td><td>No</td><td>Drywell</td><td>Manual</td><td rowspan="2">Case 13 (no filter)</td><td rowspan="2">Case 12 (no filter)</td><td rowspan="2">Case 14</td><td rowspan="2">Case 2</td></tr>
<tr><td>4</td><td>No</td><td>Drywell</td><td>Passive</td></tr>
<tr><td>5</td><td>Yes</td><td>Wetwell</td><td>Manual</td><td rowspan="2">Case 7 or 15 (filter)</td><td rowspan="2">Case 3 (filter)</td><td rowspan="2">Case 6</td><td rowspan="2">Case 2</td></tr>
<tr><td>6</td><td>Yes</td><td>Wetwell</td><td>Passive</td></tr>
<tr><td>7</td><td>Yes</td><td>Drywell</td><td>Manual</td><td rowspan="2">Case 13 (filter)</td><td rowspan="2">Case 12 (filter)</td><td rowspan="2">Case 14</td><td rowspan="2">Case 2</td></tr>
<tr><td>8</td><td>Yes</td><td>Drywell</td><td>Passive</td></tr>
</table>

3.2.4 Quantitative Information

Parameters values used to estimate the release sequence frequencies were taken from a variety of sources, as shown in Table 6.

Table 6. Parameter Values Used in the Risk Evaluation

Parameter	Value		Basis
CDF	2E-5/reactor-year		SPAR external hazard models
Fraction of total CDF due to external hazards	0.8		SPAR external hazard models; review of previous PRAs
Breakdown of sequence types for internal hazards	Other (not SBO, bypass or fast)	0.83	SPAR internal hazard models
	SBO	0.12	
	Bypass (ISLOCAs)	0.05	
	Fast (MLOCAs, LLOCAs, ATWS)	0.01	
Breakdown of sequence types for external hazards	Other (not bypass)	0.95	Review of previous PRAs; engineering judgment
	Bypass	0.05	
Probability that SA vent fails to open	Mod 0	1	
	Mods 1, 3, 5, 7—other or SBO	0.3	SPAR-H method (manual vent; longer available time)
	Mods 1, 3, 5, 7—fast	0.5	SPAR-H method (manual vent; shorter available time)
	Mods 2, 4, 6, 8	0.001	Engineering judgment (passive vent mechanical failure)
Conditional probability that offsite power is not recovered by the time of lower head failure given not recovered at the time of core damage (internal hazards)	0.38		Historical data (NUREG-6890)
Probability that portable pump for core spray or drywell spray fails	0.3		SPAR-H; consistent with SPAR B.5.b study done by Idaho National Laboratory

The consequence per release for population dose, offsite cost, and contaminated area were obtained from MELCOR/MACCS2 calculations (Enclosures 5a and 5b). Table 7 lists the results of these calculations which have been used in the risk evaluation.

Table 7. Consequences Determined by MELCOR/MACCS2 Calculations							
Case	Core Spray	Drywell Spray	Venting	Location	Population Dose (person-rem/event)	Offsite Cost ($/event)	Land Contamination (km^2/event)
2	no	no	no	n/a	514,000	$1,910,000,000	354
3F	no	no	yes	wetwell	183,000	$274,000,000	8
3NF	no	no	yes	wetwell	397,000	$1,730,000,000	54
6	yes	no	no	n/a	305,000	$847,000,000	91
7F	yes	no	yes	wetwell	37,300	$17,600,000	0.4
7NF	yes	no	yes	wetwell	235,000	$484,000,000	34
12F	no	no	yes	drywell	232,000	$391,000,000	28
12NF	no	no	yes	drywell	3,810,000	$33,300,000,000	9,150
13F	no	yes	yes	drywell	59,990	$37,700,000	2
13NF	no	yes	yes	drywell	3,860,000	$33,000,000,000	8,830
14	no	yes	no	n/a	86,100	$116,000,000	12
15F	no	yes	yes	wetwell	43,300	$20,200,000	0.3
15NF	no	yes	yes	wetwell	280,000	$588,000,000	28

Table 8 lists the onsite consequences that were used in the risk evaluation, consistent with the existing regulatory analysis guidance in NUREG/BR-0184.

Table 8. Onsite Consequences		
Release End State	Onsite Worker Dose Risk (person-rem/event)	Onsite Cost ($/event)
vented—filtered	1,000	$1,900,000,000
vented—unfiltered	3,300	$2,390,000,000
LMT, OP, or OP + LMT	14,000	$3,190,000,000

4. RESULTS

Table 9 provides the frequencies and percent contributions for each end state defined in the risk evaluation. The frequencies and contributions are identical for those modifications that have the same vent actuation method (either manual or passive). This is expected since the venting actuation method (and its associated failure probability) is the only characteristic among the group of characteristics that define the hypothetical plant modifications which influences the event tree sequence frequencies. Comparison of the information in this table to the CCFP values presented in Table 2 demonstrates that the installation of an SA venting system helps to lower the CCFP.

Table 9. Breakdown of Containment Failure Modes

Mod	Vent Filtered	Vent Location	Vent Actuation	End State			
				vent	LMT	OP	LMT + OP
0	n/a	None	n/a	0 0%	0 0%	3E-7 1.5%	2E-5 98.5%
1	No	Wetwell	Manual	9E-6 46.8%	4E-6 19.6%	9E-8 0.4%	7E-6 33.1%
3	No	Drywell	Manual				
5	Yes	Wetwell	Manual				
7	Yes	Drywell	Manual				
2	No	Wetwell	Passive	1E-5 66.9%	6E-6 28.0%	3E-10 0.0%	1E-6 5.1%
4	No	Drywell	Passive				
6	Yes	Wetwell	Passive				
8	Yes	Drywell	Passive				

Table 10 provides the point estimates of the risks for each of the nine hypothetical plant modifications (Mod 0, which is the current situation and Mods1 through 8). In comparison to Mod 0, in the available SAMA analyses, the baseline 50-mile population dose risks range from 3.3 to144 person-rem/ry, and the offsite cost risks range from $5,614/ry to $976,847/ry.

Table 10. Risk Evaluation Results (Point Estimates)								
Mod	**Vent Filtered**	**Vent Location**	**Vent Actuation**	**50-mile Population Dose Risk (person-rem/ry)**	**50-mile Offsite Cost Risk ($/ry)**	**Onsite Worker Dose Risk (person-rem/ry)**	**Onsite Cost risk ($/ry)**	**CCLA (km^2)**
0	n/a	None	n/a	10.2	$37,884	0.28	$63,800	350.1
1	No	Wetwell	Manual	7.2	$24,041	0.14	$53,166	144.1
2	No	Wetwell	Passive	5.9	$18,117	0.08	$48,615	55.9
3	No	Drywell	Manual	54.5	$452,466	0.14	$53,166	6,048.4
4	No	Drywell	Passive	73.5	$630,000	0.08	$48,615	8,487.8
5	Yes	Wetwell	Manual	4.5	$13,958	0.11	$46,653	119.3
6	Yes	Wetwell	Passive	2.0	$3,717	0.03	$39,315	20.5
7	Yes	Drywell	Manual	4.9	$14,540	0.11	$46,653	123.6
8	Yes	Drywell	Passive	2.6	$4,642	0.03	$39,315	27.2

Table 11 provides the risk reductions (relative to Mod 0, the current situation) associated with implementation of the SA venting system plant modifications (Mods 1 through 8). Figures 4 through 8 graphically depict the information contained in Table 11.

Table 11. Risk Reduction Results (Point Estimates)								
Mod	**Vent Filtered**	**Vent Location**	**Vent Actuation**	**Reduction in 50-mile Population Dose Risk (Δperson-rem/ry)**	**Reduction in 50-mile Offsite Cost Risk (Δ$/ry)**	**Reduction in Onsite Worker Dose Risk (Δperson-rem/ry)**	**Reduction in Onsite Cost risk (Δ$/ry)**	**Reduction in CCLA ($\Delta km^2/ry$)**
1	No	Wetwell	Manual	3.0	$13,842	0.14	$10,634	206.0
2	No	Wetwell	Passive	4.3	$19,767	0.29	$15,185	294.2
3	No	Drywell	Manual	-44.3	-$414,582	0.14	$10,634	-5,698.3
4	No	Drywell	Passive	-63.3	-$592,117	0.20	$15,185	-8,137.7
5	Yes	Wetwell	Manual	5.7	$23,926	0.17	$17,147	230.8
6	Yes	Wetwell	Passive	8.2	$34,166	0.25	$24,485	329.5
7	Yes	Drywell	Manual	5.3	$23,344	0.17	$17,147	226.4
8	Yes	Drywell	Passive	7.6	$33,242	0.25	$24,485	322.9

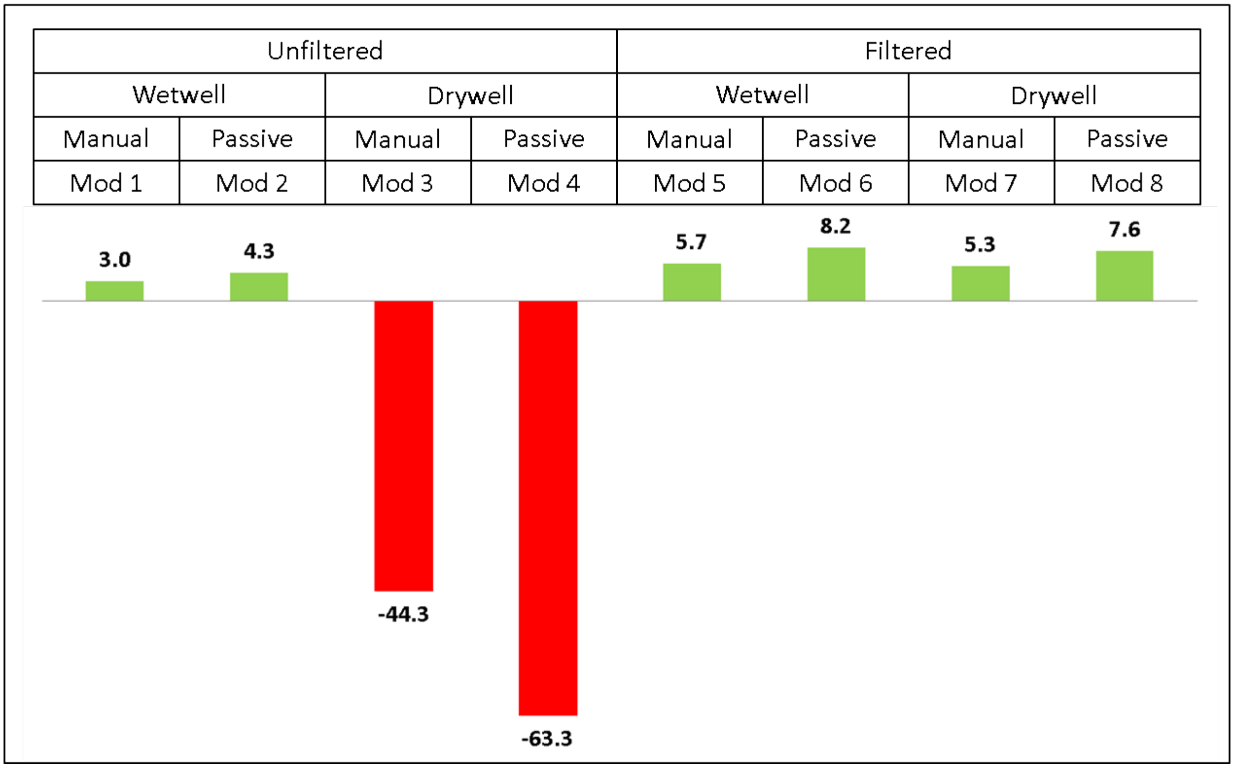


Figure 4. Reduction in population dose risk

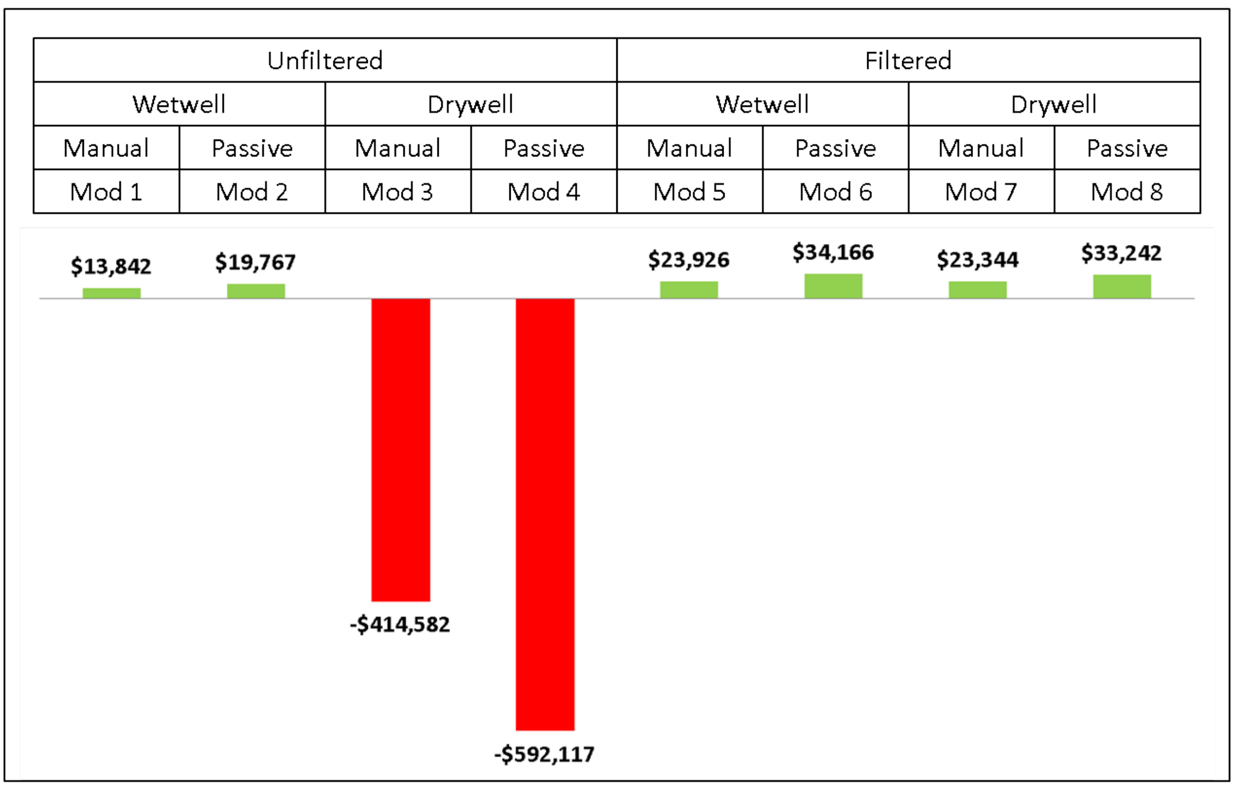


Figure 5. Reduction in offsite cost risk

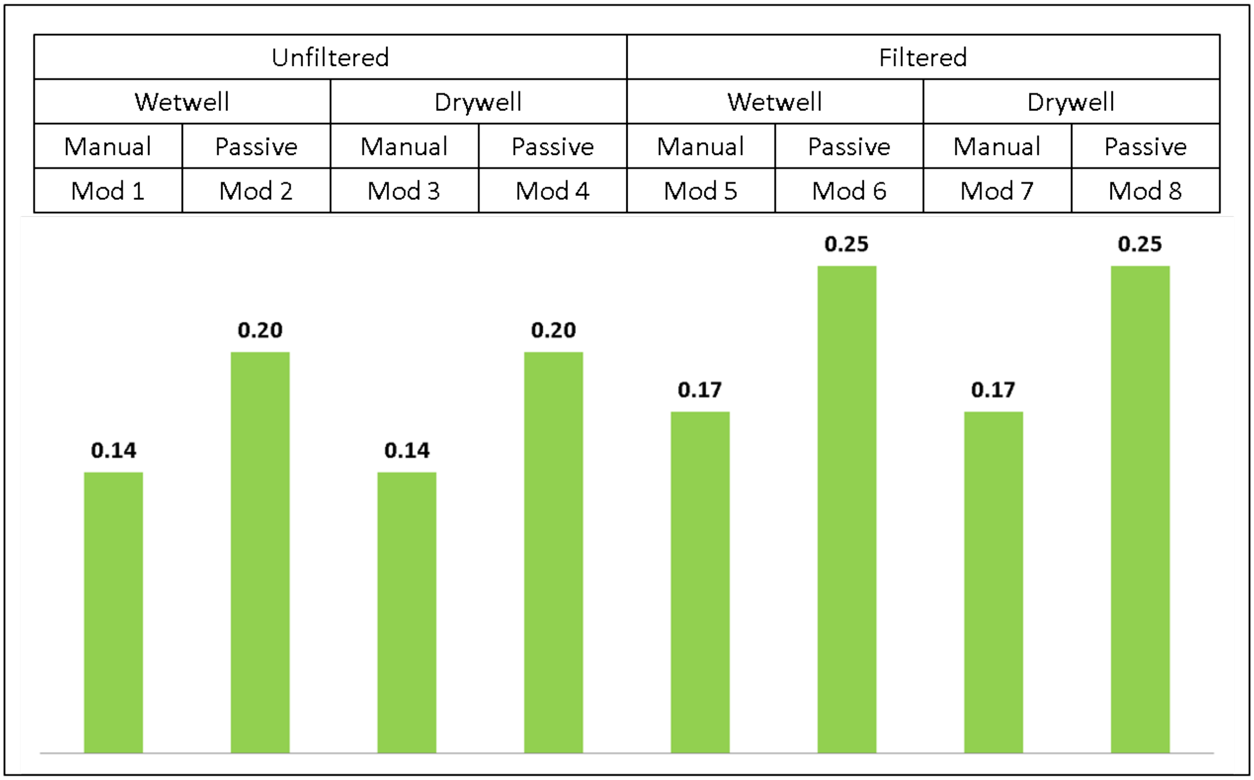


Figure 6. Reduction in onsite worker dose risk

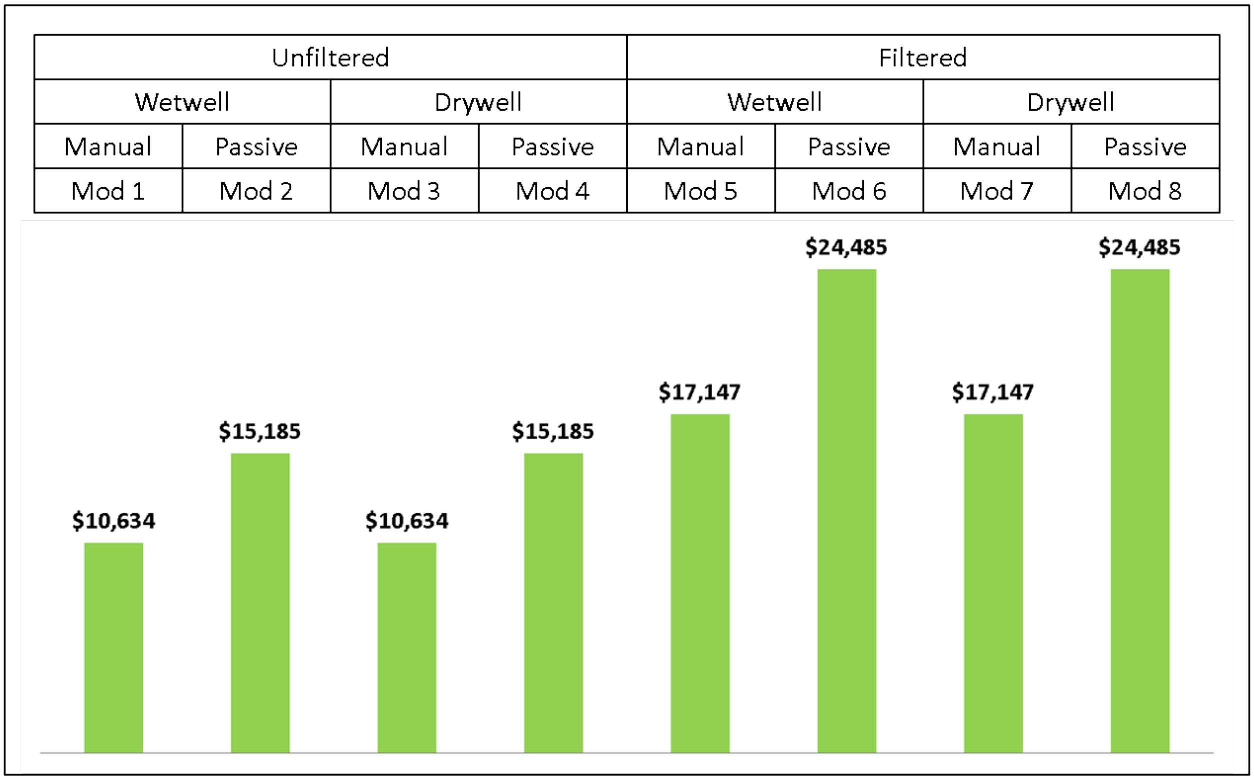


Figure 7. Reduction in onsite cost risk

Unfiltered				Filtered			
Wetwell		Drywell		Wetwell		Drywell	
Manual	Passive	Manual	Passive	Manual	Passive	Manual	Passive
Mod 1	Mod 2	Mod 3	Mod 4	Mod 5	Mod 6	Mod 7	Mod 8

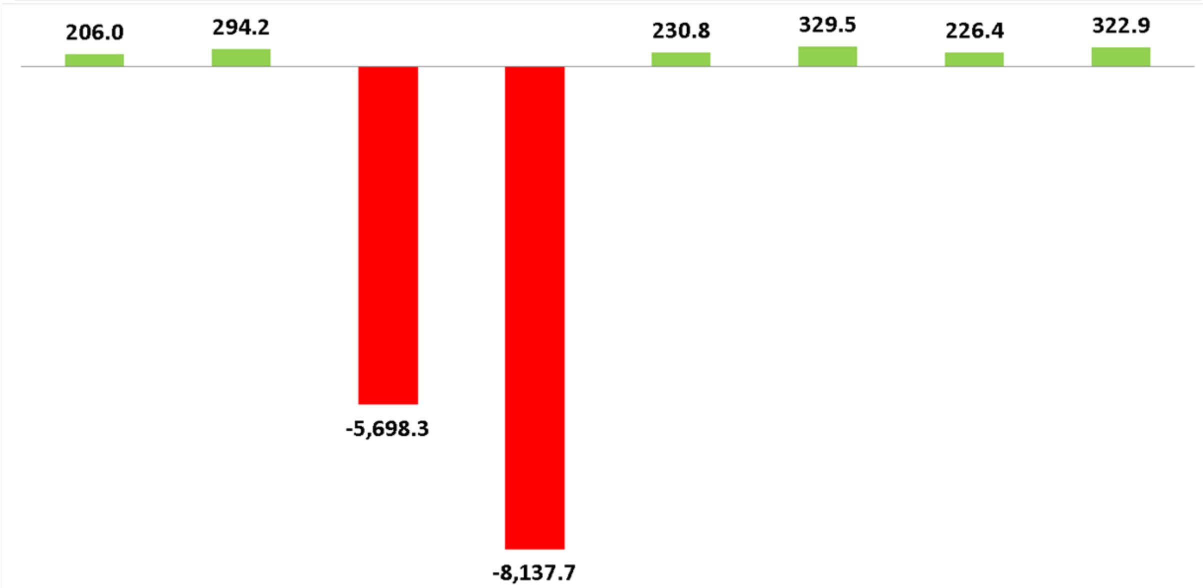


Figure 8. Reduction in conditional contaminated land area

In order to gain further insight into the risk reductions afforded by the hypothetical plant modifications, a simple parametric Monte Carlo uncertainty analysis was performed. Each of the parameters used to quantify the sequence frequencies and each of the consequences was assigned a distribution as described in Table 12.

Table 12. Uncertainty Distributions

Parameter	Mean		Distribution
CDF	2E-5/reactor year		Lognormal; error factor = 10
Fraction of total CDF due to external hazards	0.8		Beta; α = 0.5, β = 0.125
Breakdown of sequence types for internal hazards	Other (not SBO, bypass or fast)	0.83	Dirichlet α_1 (other) = 41 α_2 (SBO) = 6 α_3 (bypass) = 2.5 α_4 (fast) = 0.5
	SBO	0.12	
	Bypass (ISLOCAs)	0.05	
	Fast (MLOCAs, LLOCAs, ATWS)	0.01	
Breakdown of sequence types for external hazards	Other (not bypass)	0.95	Beta; α (bypass) = 0.5, β (bypass) = 9.5
	Bypass	0.05	
Probability that SA vent fails to open	Mod 0	1	Held constant
	Mods 1, 3, 5, 7—other or SBO	0.3	Beta; α = 0.5, β = 1.167
	Mods 1, 3, 5, 7—fast	0.5	Beta; α = 0.5, β = 0.5
	Mods 2, 4, 6, 8	0.001	Beta; α = 0.5, β = 499.5
Conditional probability that offsite power is not recovered by the time of lower head failure given not recovered at the time of core damage (internal hazards)	0.38		Beta; α = 0.5, β = 0.816
Probability that portable pump for core spray or drywell spray fails	0.3		Beta; α = 0.5, β = 1.167
Consequences	Per Tables X-7 and X-8		Lognormal; error factor = 10 Within a given consequence category, consequences were assumed to be totally dependent.

Results of the parametric uncertainty analysis are shown in Figures 9 through 13. The mean values are very close, although somewhat higher, to the point estimates. In general, the ratio of the 95th percentile to the point estimate varies from about 3.5 to 4.0 depending on the consequence category. The major contributors to uncertainty in the risk reduction results are uncertainty in the core-damage frequency and uncertainty in the sequence consequences.

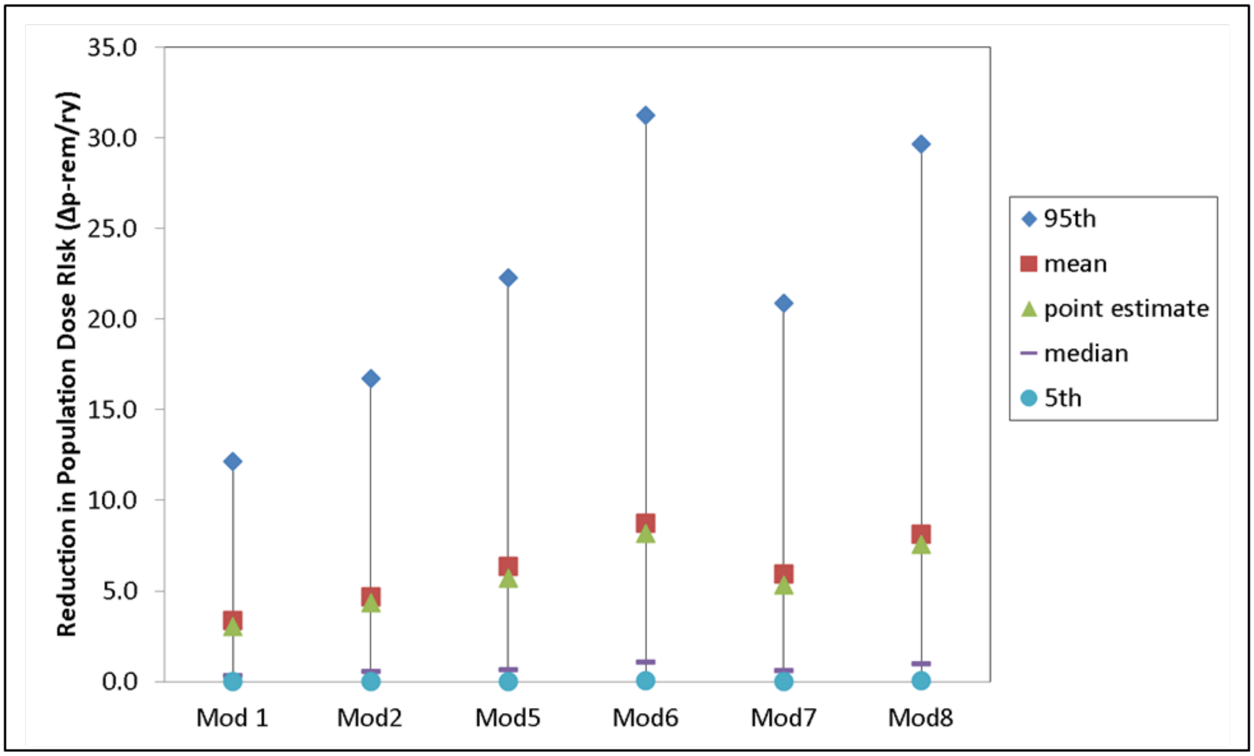


Figure 9. Uncertainty in the reduction in population dose risk

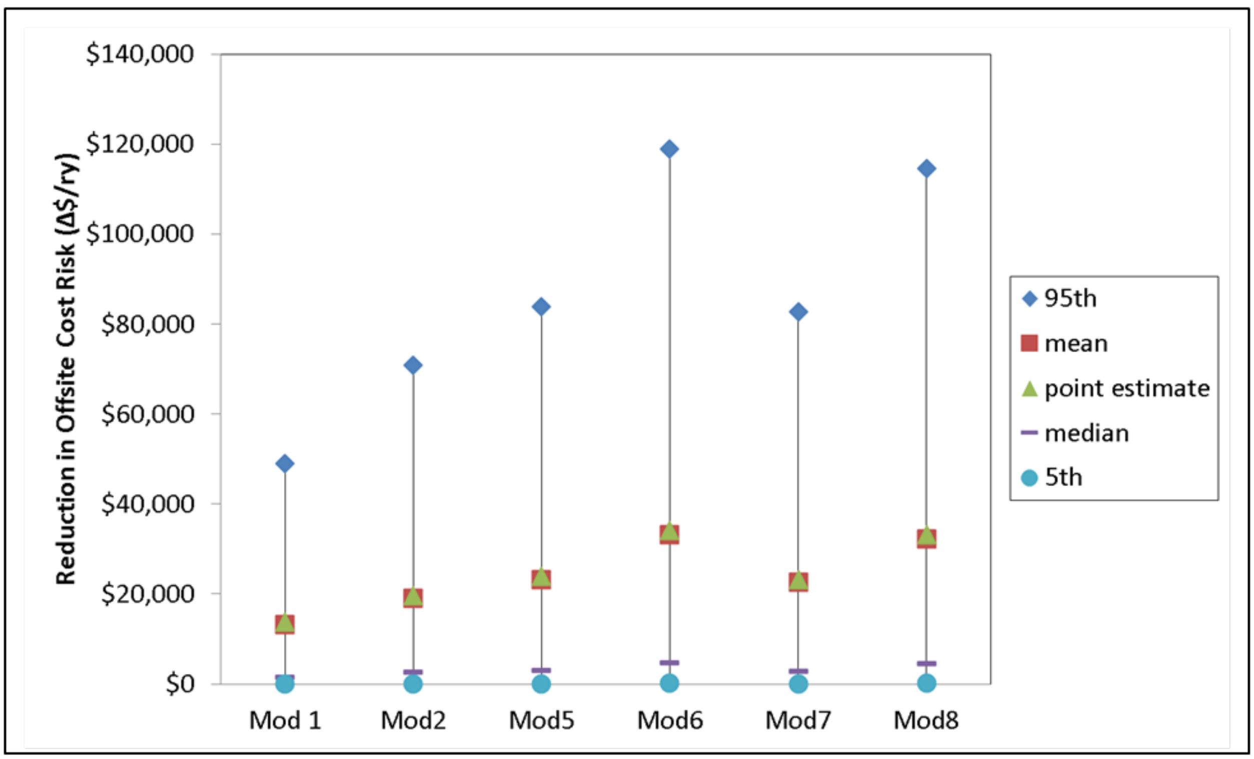


Figure 10. Uncertainty in the reduction in offsite cost risk

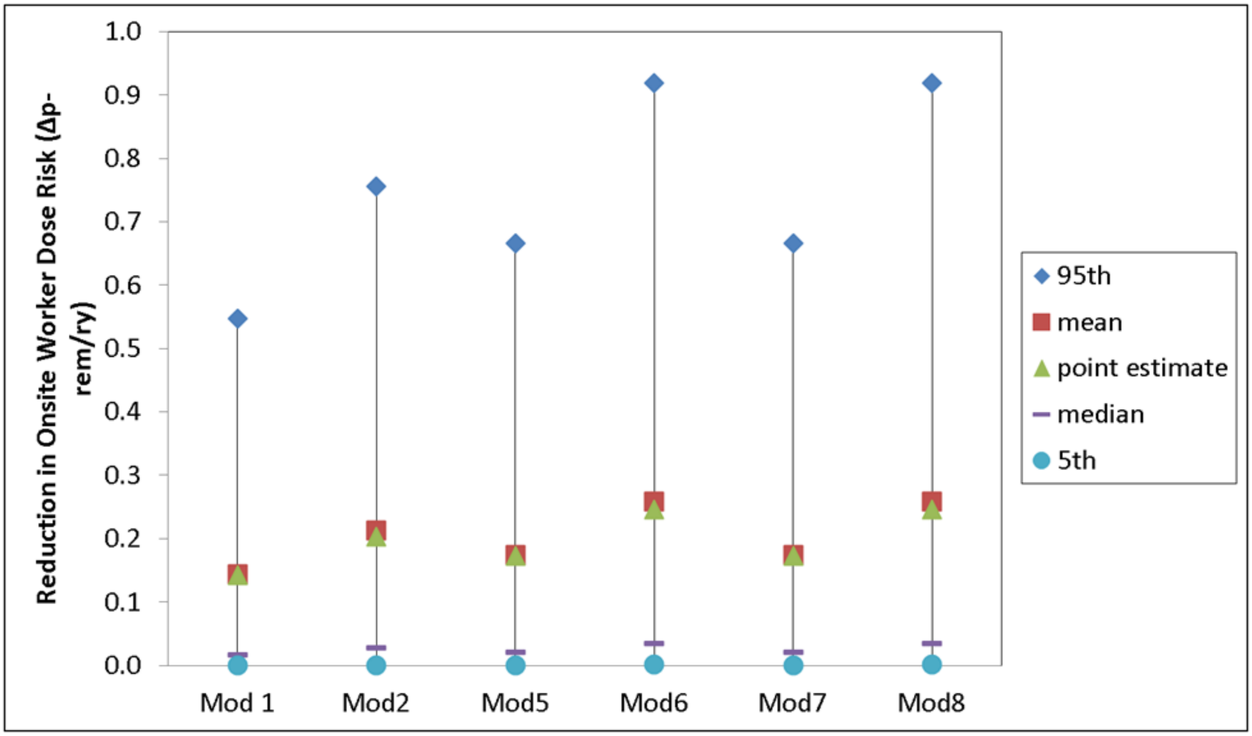


Figure 11. Uncertainty in the reduction in onsite worker dose risk

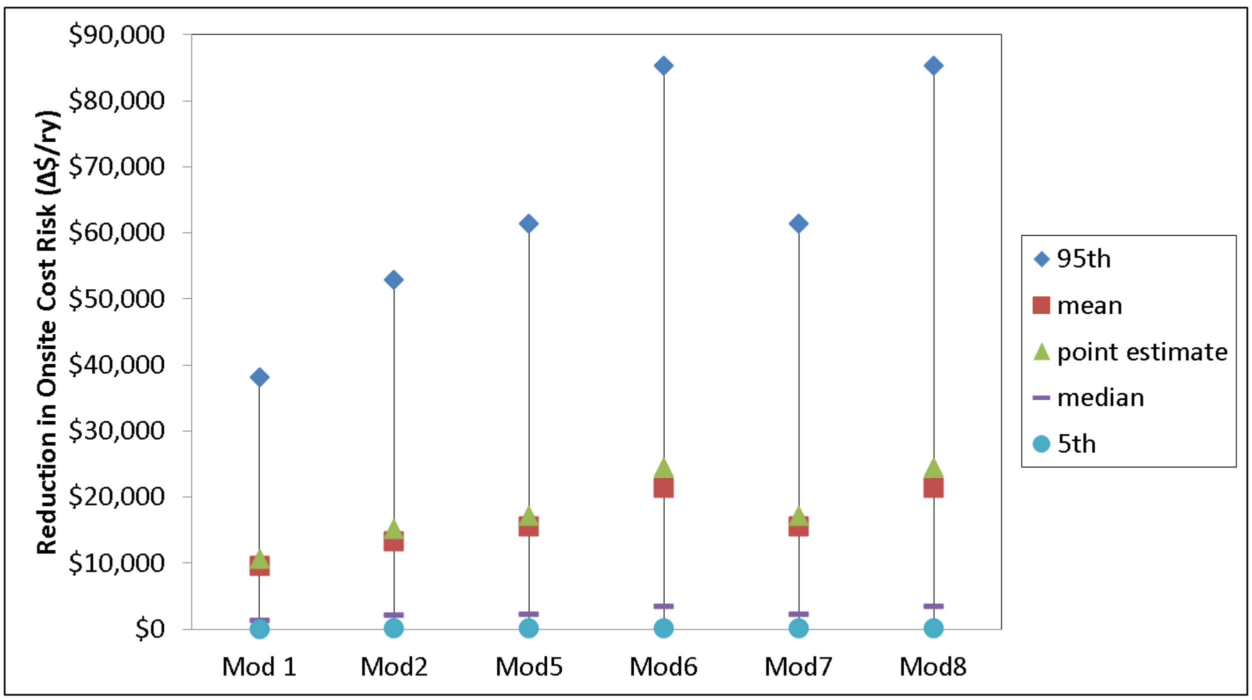


Figure 12. Uncertainty in the reduction in onsite cost risk

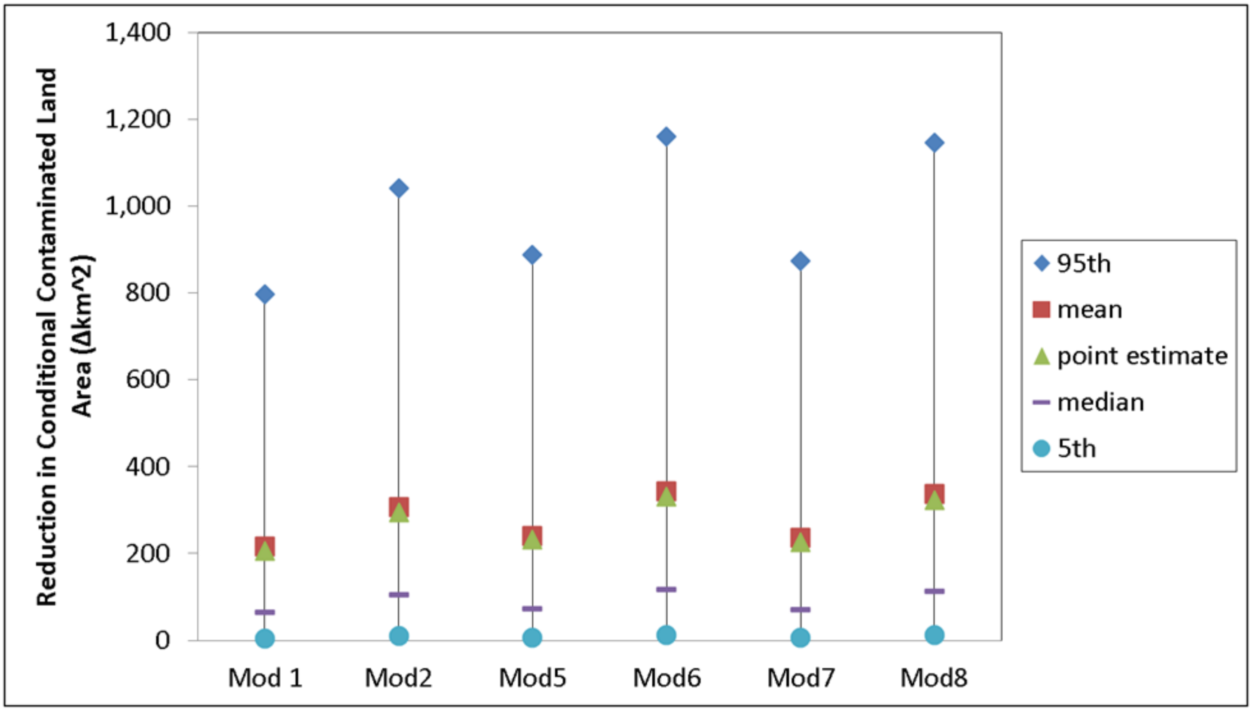


Figure 13. Uncertainty in the reduction in conditional contaminated land area

5. CONCLUSIONS

The risk evaluation presented above, which incorporates information and insights from the MELCOR analysis in Enclosure 5a and the MACCS analysis in Enclosure 5b, makes a compelling technical argument for a strategy to mitigate the radiological consequences of severe accidents in BWR Mark I containments that includes a combination of SA venting and core debris cooling, supplemented further by the installation of an external filter. In other words, the risk evaluation provides a technical basis to support Option 3 in the regulatory analysis. The risk evaluation presented here leads to the following specific conclusions on SA venting:

- The installation of an unfiltered wetwell SA venting system would reduce public health risk, offsite economic cost risk, and land contamination risk. In contrast, the installation of an unfiltered drywell SA venting system would increase public health risk, offsite economic cost risk, and land contamination risk.
- The installation of a filtered SA venting system (attached to either the wetwell or the drywell) would reduce public health risk, offsite economic cost risk, and land contamination risk. That is, the incorporation of an external filter into the SA venting systems is preferable.
- By preventing containment overpressurization failure, the successful operation of an SA venting system promotes access to plant areas where portable pumps would be installed to provide core debris cooling.
- The installation of an SA venting system (unfiltered or filtered, attached to the wetwell or the drywell) would reduce onsite worker health risk and onsite cost risk.
- Passive actuation (via a rupture disk) is preferred to manual actuation because it is more reliable and, hence, results in larger risk reductions.
- The uncertainty in the amount of risk reduction achieved by the installation of an SA venting system is mainly due to uncertainty in the CDF and uncertainty in the offsite and onsite consequences resulting from radiological releases.

ENCLOSURE 6

STAKEHOLDER INTERACTIONS

Contents

Stakeholder Interactions

1.0 Introduction

To better inform its regulatory analysis, the staff conducted ten public meetings with stakeholders to better understand their views and obtain feedback on severe accident and filtered containment venting. Summaries of meetings related to severe accident and filtered containment venting are provided in this enclosure.

2.0 Public Meetings

2.1 December 15, 2011

Purpose: The purpose of this meeting was to begin discussions with stakeholders on implementation strategies the NRC was considering taking to address Recommendation 5.1, Reliable Hardened Vents, of the Near-Term Task Force (NTTF) Recommendations. The meeting focused on a general approach and introduction to the implementation of this recommendation.

Summary: The staff provided an overview of the Fukushima accident, describing the difficulty that plant operators faced when attempting to vent the containments at Units 1, 2 and 3. The staff noted that ensuring that BWR Mark I and Mark II containments have reliable hardened venting capability would have a significant safety benefit. In addition the staff indicated that it was considering the idea that the reliable hardened venting system be equipped with a filter to preserve the containment function as a barrier to fission products. Representatives from the BWR Owners' Group stated that it was looking into alternative approaches to filtering, and the staff recommended that the BWROG provide any insights into the alternatives to an external filter as soon as possible.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML11348A100
Stakeholder Presentation Slides - ML11353A002 (BWROG)

2.2 January 17, 2012

Purpose: The purpose of this meeting was to continue discussions with stakeholders on implementation strategies the NRC was considering taking to address Recommendation 5.1, Reliable Hardened Vents, of the Near-Term Task Force (NTTF) Recommendations. The meeting focused on hardened vent performance requirements and implementation of this recommendation.

Summary: The NRC staff provided an update since the previous meeting, including an accelerated schedule for all Tier 1 NTTF recommendations to be issued by March 9, 2012, as well as the NRC Japan Lessons Learned Steering Committee decision that additional information is needed on potential filters for the reliable hardened vent for applicable licensees. The Steering

Committee has asked the NRC staff to prepare a policy issue paper which will be presented to the Commission for a notation vote by the summer of 2012 relating to the filtered vent issue. The staff also outlined its current views relating to possible new regulatory requirements for reliable hardened vents.

The industry and BWROG representatives presented their proposed response to the December 15, 2011, public meeting related to a hardened filtered vent in the terms of two distinct phases. Phase 1 would employ a reliable hardened vent integrated with the Nuclear Energy Institute's (NEI) "Integrated, Diverse & Flexible Mitigation Capability" (FLEX) initiative. Phase 2 would focus on a post-core damage response strategy to reliably vent containment and manage radiological release for an extended station blackout.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12013A230
Stakeholder Presentation Slides - ML12019A122 (BWROG)
Meeting Summary - ML12025A020

2.3 May 2, 2012

Purpose: The purpose of this meeting was to discuss the implementation of Order EA-12-050, regarding reliable hardened containment vents at BWR facilities with Mark I and Mark II containments. The staff also discussed development of interim staff guidance (ISG) relating to this order that was to be issued by August 31, 2012, and the staff requested input from stakeholders regarding the implementation of order requirements. In addition, the staff sought input relating to the issue of filtered vents as described in 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*," issued February 17, 2012.

Summary: The NRC staff provided an overview of the plan to issue the ISG for Order EA-12-050 no later than August 31, 2012. The staff provided a general outline of the draft ISG contents: (1) definitions, (2) administrative requirements, (3) reporting requirements, and (4) NRC staff positions on each of the order's technical requirements. The NRC staff reviewed each of the order's administrative, reporting, and technical requirements, and presented preliminary staff viewpoints on each of the requirements.

The issue of early containment venting was noted as a particular interest to the BWROG, and the owner's group was interested in learning whether or not the NRC staff has changed its views on containment venting as a "last resort" in light of the lessons learned from Fukushima. The staff noted that any changes to Emergency Procedure Guidelines would likely have to be reviewed by the NRC staff prior to implementation.

The NRC staff sought input and comments from members of the public and non-governmental organization representatives on the issue of filtered containment vents. The staff noted that the Commission directed the staff in SRM-SECY-11-0137, to address the issue of 'Filtration of Containment Vents' in conjunction with the Tier 1 issue on hardened vents for Mark I and Mark II containments. The introduction of this issue prompted numerous comments from members of the public. Examples included: (1) concerns as to why the NRC did not require the vents to be able to handle severe accidents and the presence of hydrogen gas following a severe accident; (2) many considered containment vent filters an obvious solution and stated that filters should be made a requirement to ensure that the containment is able to "do what it is suppose to do;" (3) another person commented that filtered vents should be required because operators never know when core damage really begins; (4) the NRC is not serving the public interest by assuming no core damage is present with the hardened vents that were ordered by the NRC in March.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12124A132
Stakeholder Presentation Slides - ML12124A130 (BWROG)
Meeting Summary - ML12130A369

2.4 May 14, 2012

Purpose: The purposes of this meeting were to brief stakeholders on the staff's preliminary plans for implementation of the Tier 3 recommendations, provide an opportunity for stakeholders and the NRC staff to exchange information on the Tier 3 recommendations, afford stakeholders an opportunity to ask the NRC staff clarifying and amplifying questions on the plans, and provide input for consideration before the plans were finalized. The staff also gave a presentation on information gathered for Recommendation 5.1, "Reliable Hardened Vents for Mark I and II Containments." The public had an opportunity to comment and discuss the recommendation following each individual staff presentation.

Summary: Some audience and teleconference members felt very strongly about requiring filters for Mark I and II containments, and urged the staff to permanently shutdown boiling-water reactors with Mark I and II containments if filters were not installed. Another audience member encouraged the staff to think beyond the hardened filtered vent systems that some European nations had installed in the 1980s. A Nuclear Energy Institute (NEI) representative indicated that hardened, filtered ventilation was a complex topic, and that it was more important to do it right, rather than quickly. Accordingly, NEI would be sending a letter to the Commission requesting that staff perform a more comprehensive analysis that considers other alternatives for precluding and mitigating potential releases from core damage events, and credits safety improvements being installed under FLEX.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12137A008
Stakeholder Presentation Slides - N/A
Meeting Summary - ML12160A097

2.5 July 12, 2012

Purpose: The purpose of this meeting was to discuss testing programs and technology developments on wet and dry filtered containment venting systems (FVCS) with Dr. Bernd Eckardt, AREVA NP Canada LTD. AREVA provided information to the staff on European and world experience relating to FCVS since the late 1980s.

Summary: Representatives from AREVA NP Canada opened the technical discussions by providing an overview of FCVS at Canada's CANDU nuclear plants. CANDU plants have containment structures that are similar in design to large dry containments in the U.S.; however, the use of FCVS technology is also applicable to BWR Mark I and Mark II containment designs. Additional details of the FCVS system installed at Point Lepreau were also provided.

AREVA presented information about FCVS technology and historical developments since the 1980s. AREVA also discussed the issue of the "filter gap." The filter gap issue is primarily concerned about the filter's ability to retain particles less than one micron in size. AREVA stated that, depending on the particle diameter, a filter's retention efficiency has been shown to vary. He added that every type of filter technology appears to have a "filter gap" where lower removal efficiencies are observed for particles of a particular size. As a result, filter engineers have designed ways to overcome concerns relating to the filter gap in order to achieve improved particle retention.

AREVA discussed the development of scrubbers, filters, sorbents, media, standards and new liquid agents from a historical perspective of the development of FCVS since the 1980s. The principles of venturi scrubbing were presented, including the engineered features being employed by filter designers to eliminate the filter gap. AREVA further explained that rigorous testing was performed in order to verify aerosol retention capabilities and that very high decontamination factors (DFs) have been verified by thousands of laboratory tests under prototypical operating conditions. An AREVA representative stated that filters have achieved efficient retention (high DFs) of large and fine aerosol fractions for aerosols (fine Aerosols > 10,000 and large Aerosols > 100,000) during ACE/JAVA testing. AREVA also presented information on dry filter technology including metal fiber and sand bed filters.

Following the formal presentations and discussions on filtered containment venting technology, the NRC staff sought input and comments from members of the public and non-governmental organization representatives.

Related ADAMS Documents:

NRC Staff Presentation Slides - N/A
Stakeholder Presentation Slides - ML12206A263 (AREVA - Dr. Eckardt)
ML12206A266 (AREVA NP Canada)
Meeting Summary - ML12319A530

2.6 August 8, 2012

Purpose: The purpose of this meeting was to discuss with the Electric Power Research Institute (EPRI), industry representatives, and members of the public to the results of industry's analysis and assessment of possible severe accidents in BWRs with Mark I and Mark II containments using various codes and models for radiological releases. In addition, the staff discussed the role of uncertainty in risk-informed decision making.

Summary: EPRI provided an overview and preliminary results of the research efforts that were later documented in its September 25 report. EPRI provided preliminary information relating to computer modeling and preliminary evaluation of strategies for mitigating radiological releases during severe accidents at BWRs with Mark I and II containments.

The EPRI report evaluates certain strategies that are intended to maintain or enhance the containment function in scenarios involving long-term loss of electric power. The strategies evaluated include water injection (by flooding or spraying), alternative containment heat removal, venting, controlled venting, filtered venting, and combinations of these plant features. Based on the results of its research, EPRI noted seven "key insights" from the analysis, including:

- No single strategy is effective
- Active core debris cooling is required
- Existing severe accident management guidelines (SAMGs) strategies provide substantial benefit
- Spraying the containment atmosphere is beneficial
- Venting prevents uncontrolled release and manages hydrogen
- Control of the vent provides benefit
- Low-efficiency filters can further reduce radionuclide releases

The staff was in general agreement with many of EPRI's insights; however, many concerns remained about strategies that use existing containment features and their ability to achieve a dependable and adequate decontamination of radionuclides following a severe accident.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12229A303
Stakeholder Presentation Slides - ML12229A293 (EPRI)
Meeting Summary - ML12233A085

2.7 September 4, 2012

Purpose: The purpose of this meeting was to discuss testing programs and technology developments on filtered containment venting systems (FVCS) with representatives from the Paul Scherrer Institute (PSI), Villigen, Switzerland. PSI is a multi-disciplinary research organization that has considerable experience relating to research and development of FCVS.

Summary: IMI Nuclear (IMI) is a supplier of filters for containment venting applications and has a working relationship with PSI. IMI opened the meeting with discussions on (1) venturi scrubbing, (2) metal fiber filtration, and (3) iodine adsorption by molecular sieve based adsorbents. IMI representatives then provided their perspectives on the suitability of these technologies for filtered venting applications. IMI also contrasted the aerosol removal performance of the CCI FCVS (CCI is affiliated with IMI Nuclear). One of the more notable features of the CCI FCVS is its sparger assembly. The spargers operate by directing a fraction of the airstream from the nozzle to an opening with a restricted flow path. Larger aerosol particles enter the nozzle opening, forming a "virtual surface," to become entrained in a minor flow of aerosols at a reduced velocity. Smaller aerosol particles follow the major flow and are ultimately captured in the liquid. This process is repeated two more times in the nozzle. IMI stated that mockup testing produced extremely high DF of aerosols, and iodine species under all conditions.

Representatives from PSI presented information about its experience in the area of FCVS technology and provided a short history of PSI's knowledge in aerosol and iodine research. During the 1980s PSI participated in the LACE and DEMONA tests and development of on-line aerosol concentration measurement devices and LOFT research. PSI also supported development of Sulzer's FCVS during this time. In the 1990s, PSI performed aerosol research (aerosol generation by plasma, POSEIDON pool scrubbing, GE-SBWR PCCS and SIEMENS SWR100 PCCS behavior testing). More recently, PSI has conducted further research and development in the area of FCVS technology: ARTIST project studying aerosol and droplet retention in steam generators, qualification tests for CCI-FCVS, severe accident safety studies for Swiss plants in support of PSA, hydrogen behavior in containments (PANDA), and research on aerosol behavior to support IMI (CCI-AG) for demonstration of FCVS performance under utility specified conditions.

Related ADAMS Documents:

NRC Staff Presentation Slides - N/A
Stakeholder Presentation Slides - ML12248A019 (Paul Scherrer Institute)
ML12248A021 (IMI Nuclear)
Meeting Summary - ML12319A541

2.8 September 13, 2012

Purpose: The purpose of this public meeting was to discuss initial results from the NRC staff's analysis of various strategies or methods to manage radiological releases following a severe accident in BWR Mark I and Mark II containments. Discussions focused on the staff's use of severe accident analysis codes such as MELCOR in the assessment of severe accident progression. Scenarios with various containment venting, spraying, and flooding strategies were discussed. The staff also allotted a significant portion of the meeting agenda to allow representatives from public interest groups to provide technical insights relating to the issue of filtered containment venting.

Summary: Members of the NRC staff from the Office of Nuclear Regulatory Research (RES) provided an overview of severe accident management and containment venting, and strategies to protect containment and limit radiological releases. RES then discussed the analysis it performed using MELCOR. The purpose of the MELCOR analysis is to support the regulatory analyses on filtered venting. The filtered venting regulatory analysis will draw upon the results of MACCS calculations based on representative MELCOR cases. The MELCOR cases focused on Mark I containments, and were informed by Fukushima and SOARCA. This information was then used to perform MACCS consequence calculations using MELCOR output.

David Lochbaum, Director, Nuclear Safety Project, Union of Concerned Scientists (UCS), made a presentation regarding filtered venting. UCS noted that radioactive releases during routine operations and design basis accidents are filtered through the standby gas treatment system (SGTS); however, radioactive releases during severe accidents are not filtered. UCS's argument is that, when the highest amount of radioactivity is likely present, the lowest protection to plant workers and members of the public is provided. In addition, UCS noted that there is a large uncertainty associated with the analysis of severe accident progression and modeling.

Mary Lampert, Director, Pilgrim Watch (PW), then presented information on its perspectives of filtered venting. PW recommended that hardened vents now required by Order EA-12-050 be equipped with rupture discs and filters to help ensure that operators are not reluctant to follow orders when containment venting is required. PW stated that an unfiltered vent releases up to 200 times more radioactivity than do commercially available filtered systems now being used in Europe. The PW presentation turned to the issue of how offsite consequences are being calculated. PW stated that MACCS2 under predicts or understates the consequences of a severe accident. One of the primary concerns stated is that MACCS2 does not calculate consequences of aqueous releases. PW was also concerned about the analysis assumptions, such as core damage frequency, when the NRC staff performs its cost-benefit calculations. PW requested that the staff review the reports it provided to the NRC (see ADAMS Accession Numbers below).

Mr. Mark Leyse spoke on behalf of the National Resources Defense Council (NRDC), and raised concerns relating to the NRC staff's analysis. NRDC stated that, in a BWR severe accident, "hundreds of kilograms of non-condensable hydrogen gas would also be produced (up to over 3000 kg) - at rates as high as between 5.0 and 10.0 kg per second, if there were a reflooding of an overheated reactor core - which would increase the internal pressure of the primary containment. If enough hydrogen were produced, the containment could fail from becoming over-pressurized." NRDC recommended the installation of high capacity filtered containment venting systems in order to accommodate the potentially high production of hydrogen during an accident.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12256A849

Stakeholder Documents - ML12254A871 – Pilgrim Watch Document #1
ML12254A869 – Pilgrim Watch Document #2
ML12254A865 – Mark Leyse (NRDC) Document #1
ML12254A850 – Mark Leyse (NRDC) Document #2

Stakeholder Presentation Slides - ML12256A913 - UCS Presentation Slides
ML12256A853 - Pilgrim Watch Presentation Slides

Meeting Transcript - ML12320A324
Meeting Summary - ML12319A545

2.9 October 4, 2012

Purpose: The purpose of this meeting was to hold follow-up discussions to the public meeting held on September 13, 2012. The staff discussed the results from its analysis of various strategies or methods to manage radiological releases following a severe accident in BWR Mark I and Mark II containments. Discussions focused on the use of MELCOR and MACCS in the staff's regulatory analysis, probabilistic risk assessment insights, and initial regulatory analysis insights. The staff also provided opportunities for members of the public to provide technical insights relating to the issue of filtered containment venting.

Summary: As part of its follow up to presentation and discussions of the MELCOR analysis during the September 13, 2012, public meeting, the NRC staff presented material on the following topics: (1) design and regulatory history, and foreign experience, (2) FCVS in severe accident management, (3) MELCOR analysis, (4) MACCS2 analysis, (5) risk evaluation, (6) regulatory analysis, and (7) qualitative arguments. The staff noted that technical and policy assessments and evaluations were ongoing, and that the preliminary results being shared at the meeting were subject to change. In addition, the NRC staff stated that it would be continuing to engage the NRC's Steering Committee on path forward, and that staff recommendations will be made when technical evaluations and policy assessments were completed.

Related ADAMS Documents:

NRC Staff Presentation Slides - ML12283A288
Stakeholder Presentation Slides - N/A
Meeting Summary - ML12319A547

2.10 October 11, 2012

Purpose: The purpose of this meeting is to discuss testing programs and technology developments on filtered containment venting systems (FVCS) with representatives from the Westinghouse Electric Company (WEC). Westinghouse has considerable experience relating to research and development of FCVS.

Summary: The NRC staff opened the meeting with a brief update on the status of its effort to evaluate the merits of severe accident and filtered containment venting. Representatives from WEC then presented information on its two proven filtered containment venting technologies: (1) FILTRA - MVSS (venturi based scrubber system) and (2) dry filter method (DFM) system. The Westinghouse FILTRA program has been developed in conjunction with Alstom Thermal Power. Alstom is a provider of equipment and services to various power generation and rail transportation companies. The FILTRA MVSS technology has been installed in all Swedish BWRs and PWRs and at one Swiss BWR. The DFM technology has been installed at seven German PWR facilities. Both designs were included in the ACE testing in the 1980s. WEC most recent developments in FCVS technology includes: (1) high DF >10,000 for aerosols, (2) the scrubbing of aerosols down to 0.5 microns, (3) protection against iodine release (both elemental and organic), (4) passive operation for at least 24 hours, and (5) the ability to handle relatively high decay heat loads in its filter designs.

Related ADAMS Documents:

NRC Staff Presentation Slides - N/A
Stakeholder Presentation Slides - ML12312A110 Westinghouse Technology Overview
ML12312A111 MVSS (Wet Filter) Technology
ML12312A112 DFM (Dry Filter) Technology
Meeting Summary - ML12319A549

Enclosure 7A

Draft Proposed Order

Option 2

Severe Accident Capable Vents

Or

Option 4

Pursue Development of

a Severe Accident Confinement Strategy

for BWRS with Mark I and Mark II Containments

(Additional Material Highlighted if order

included as part of Option 4 implementation)

Note: It is likely that this draft proposed order will require revision based on interactions with stakeholders and continuing internal discussions on technical and legal issues. If the Commission approves Option 2 (or Option 4 with more immediate action to make containment vents capable of operation under severe accident conditions), the staff will provide the Commission with a final order via a Regulatory Notification.

7590-01-P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ALL OPERATING BOILING WATER	)	Docket Nos. (as shown in Attachment 1)
REACTOR LICENSEES WITH	)	License Nos. (as shown in Attachment 1)
MARK I AND MARK II CONTAINMENTS	)	EA-12-XXX
	)	

[NRC-20YY-XXXX]

ORDER MODIFYING LICENSES WITH REGARD TO RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS (EFFECTIVE IMMEDIATELY)

I.

The Licensees identified in Attachment 1 to this Order hold licenses issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation of nuclear power plants in accordance with the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, these Licensees operate boiling-water reactors (BWRs) with Mark I and Mark II containment designs.

II.

On March 12, 2012, the NRC issued Order EA-12-050 requiring the Licensees identified in Attachment 1 to this Order to implement requirements for reliable hardened vents for Mark I and Mark II containments. This Order supersedes Order EA-12-050 by revising requirements imposed in Order EA-12-050 and imposing additional requirements on reliable hardened vent systems and related procedures to ensure that the venting function is maintained during severe accident conditions (i.e., following significant core damage).

Order EA-12-050 required that licensees of BWR facilities with Mark I and Mark II containment designs shall ensure that these facilities have a containment venting system that meets certain requirements for reliable and dependable operation to support strategies to control containment pressure and prevent core damage following events causing a loss of heat removal systems (e.g., an extended loss of electrical power). The NRC determined that the issuance and implementation of the requirements in Order EA-12-050 were necessary to provide reasonable assurance of adequate protection of the public health and safety and assurance of the common defense and security. As described in Order EA-12-050:

> To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.
>
> The events at Fukushima Dai-ichi highlight the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. The inability to reduce containment pressure inhibited efforts to cool the reactor core. If additional backup or alternate sources of power had been available to operate the containment venting system remotely, or if certain valves had been more accessible for manual operation, the operators at Fukushima may have been able to depressurize the containment earlier. This, in turn, could have allowed operators to implement strategies using low-pressure water sources that may have limited or prevented damage to the reactor core. Thus, the events at Fukushima demonstrate that reliable hardened vents at BWR facilities with Mark I and Mark II containment designs are important to maintain core and containment cooling.

> The Commission has determined that ensuring adequate protection of public health and safety requires that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability for events that can lead to core damage. These new requirements provide greater mitigation capability consistent with the overall defense-in-depth philosophy, and therefore greater assurance that the challenges posed by severe external events to power reactors do not pose an undue risk to public health and safety. To provide reasonable assurance of adequate protection of public health and safety, all licenses identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order.
>
> Accordingly, the NRC has concluded that these measures are necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii), and is requiring Licensee actions. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

In developing the requirements included in Order EA-12-050, the NRC acknowledged that questions remained about possible ways to limit the release of radioactive materials if the venting systems were used after significant core damage had occurred. The NRC staff described in SECY-12-XXXX, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," various options for Commission consideration. One of the options in SECY-12-XXXX (Option 2) was to revise the requirements in Order EA-12-050 to ensure that the venting function is maintained during severe accident conditions. Another option included in the Commission Paper (Option 4) called for the NRC to develop a severe accident confinement strategy for BWRs with Mark I and Mark II containments to limit the release of radioactive materials. In its Staff Requirements Memorandum (SRM) for SECY-12-XXXX, the Commission documented its decision to pursue the development of the severe accident confinement strategy but also to more immediately require the affected licensees to make the containment venting systems capable of operation under severe accident conditions and directed the NRC staff to implement that requirement through the issuance of this Order.

The desire to ensure that the venting function is maintained during severe accident conditions is to provide protection from events that might otherwise cause containment failure due to high pressures. It is equally important to prevent core debris that has melted through the reactor vessel from breaching the containment structures. New regulatory requirements were imposed as item B.5.b in Order EA-02-026 and later incorporated into 10 CFR 50.54(hh) that require licensees to have the capability to direct water into the drywells to reduce the chances of containment failures from a molten core. The NRC is also pursuing development of the severe accident confinement strategy to minimize the release of radioactive materials should it ever be necessary to vent a Mark I or Mark II containment during severe accident conditions.

III.

The NRC may impose safety measures on licensees or applicants over and above those required by the adequate-protection standard cited in Order EA-12-050. Such requirements may be pursued to protect health or to minimize danger to life or property. As described in various NRC regulations and guidance documents, the requirements established to reduce risk (beyond measures needed for adequate protection) will not attempt to eliminate all risk but will instead pursue reasonable reductions. An evaluation of the costs and benefits of proposals within this category has been used as part of the determination of what is a reasonable requirement to reduce risks to public health and safety and the common defense and security. The NRC process for evaluating costs and benefits to help determine if additional requirements should be imposed are defined in Section 50.109, "Backfitting," of Title 10 of the *Code of Federal Regulations.* Additional information and guidance related to these assessments are provided in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook."

An evaluation of the costs and benefits associated with providing a venting function for BWRs with Mark I and Mark II containments that remains available during severe accident conditions was summarized in SECY-12-XXXX. As discussed in SECY-12-XXXX, the NRC's determination that a venting system should be available during severe accident conditions considered both quantitative assessments of costs and benefits as well as various qualitative factors. SECY-12-XXXX identified changes to make the venting systems capable of operating during severe accident conditions as a prerequisite to developing a severe accident confinement strategy aimed at improving venting operations for BWR Mark I and Mark II containments. Among the qualitative factors, one of the more important is the desire to improve the defense-in-depth characteristics of Mark I and Mark II containments by addressing the high conditional failure probabilities that those containments have should an event lead to a core melt. As discussed in SECY-12-XXXX, other qualitative factors supporting installation of severe accident capable vents include addressing significant uncertainties in the understanding of severe accident events, supporting severe accident management and response, improving the control of hydrogen generated during severe accidents, improving readiness for external and multi-unit events, reducing uncertainties about radiological releases and thereby improving emergency planning and response, and maintaining consistency with interactional practices.

As previously described, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.

The events at Fukushima Dai-ichi highlight the possibility that events such as extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly hindered attempts by the operators to prevent core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. These problems with venting the containments under the challenging conditions following the tsunami contributed to the progression of the accident from inadequate cooling of the core leading to core damage, to compromising containment functions from overpressure conditions, and to the hydrogen explosions that destroyed the reactor buildings (secondary containments). The loss of the various barriers led to the release of radioactive materials and further hampered operator efforts to arrest the accidents that ultimately led to the contamination of large areas surrounding the plant. The evacuation of local populations minimized the immediate danger to public health and safety from the loss of control of the large amount of radioactive materials within the reactor cores.

The actions imposed by this Order are intended to increase confidence in maintaining the containment function following core damage events. Although venting the containment during severe accident conditions could result in the release of radioactive materials, the act of venting could prevent gross containment failures that would hamper accident management (e.g., continuing efforts to cool core debris) and result in larger releases of radioactive material later in the progression of the accident. Further actions to reduce the release of radioactive materials during such venting operations are being pursued as a longer-term program to develop a severe accident confinement strategy.

The NRC is empowered to require plant improvements beyond those needed to provide reasonable assurance of adequate protection of public health and safety when engineering approaches are available to provide a cost-justified substantial safety improvement. An evaluation of possible improvements for Mark I and Mark II containment vents was provided in SECY-12-XXXX and a more detailed regulatory analysis is available in the NRC's agencywide documents access and management system (ADAMS). These evaluations included both quantitative and qualitative factors and led the Commission to determine that the safety improvements required by this Order are justified. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees to take appropriate action to reduce the risks posed to the public from the operation of nuclear power plants.

The Commission has determined that it is a cost-justified substantial safety improvement to require BWR facilities with Mark I and Mark II containments to make the necessary plant modifications and procedure changes to provide a reliable hardened venting capability that is capable of performing under severe accident conditions. These new requirements protect health and minimize danger to life or property by having licensees provide greater capabilities to respond to severe accidents and contain radioactive materials, which is consistent with the NRC's overall defense-in-depth philosophy. These requirements are also a prerequisite to developing a severe accident confinement strategy and defining an overall set of requirements for venting operations during severe accidents for BWR Mark I and Mark II containments. To provide an enhanced level of safety, all licenses identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order.

Accordingly, the NRC has concluded that these measures are an appropriate cost-justified safety improvement under the provisions of the backfit rule, 10 CFR 50.109(a)(3), and is requiring Licensee actions. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

IV.

Accordingly, pursuant to Sections 161b, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202, "Orders," and 10 CFR Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT ALL LICENSES IDENTIFIED IN ATTACHMENT 1 TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. All Licensees shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the license. The requirements of Attachment 2 to this Order supersede those set forth in Attachment 2 to Order EA-12-050 dated March 12, 2012. These Licensees shall promptly start implementation of the requirements in Attachment 2 to the Order and shall complete full implementation **no later than December 31, 2017**.

B. 1. All Licensees shall, within **twenty (20) days** of the date of this Order, notify the Commission (1) if they are unable to comply with any of the requirements described in Attachment 2, (2) if compliance with any of the requirements is unnecessary in their specific circumstances, or (3) if implementation of any of the requirements would cause the Licensee to be in violation of the provisions of any Commission regulation or the facility license. The notification shall provide the Licensee's justification for seeking relief from or variation of any specific requirement.

2. Any Licensee that considers that implementation of any of the requirements described in Attachment 2 to this Order would adversely affect the safe and secure operation of the facility must notify the Commission, within **twenty (20) days** of this Order, of the adverse safety impact, the basis for its determination that the requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in the Attachment 2 requirement in question, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, the Licensee must supplement its response to Condition B.1 of this Order to identify the condition as a requirement with which it cannot comply, with attendant justifications as required in Condition B.1.

C. 1. All Licensees shall, **within 6 months following the issuance of interim staff guidance (ISG) for this order**, submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements described in Attachment 2 will be achieved. The NRC staff plans to issue the ISG no later than **[insert date 120 days from date of this order]**.

2. All Licensees shall provide status reports at **six (6)-month intervals** following submittal of the overall integrated plan, as required in Condition C.1, which delineates progress made in implementing the requirements of this Order.

3. All Licensees shall report to the Commission when full compliance with the requirements described in Attachment 2 is achieved.

4. All Licensees shall, **by August 31, 2015**, submit to the Commission a progress report on the development of a severe accident confinement strategy for limiting the release of radioactive materials should it ever be necessary to vent containment during severe accident conditions. The report shall describe progress made on selecting specific performance measures and the development of analyses tools, research, and testing related to those performance measures.

Licensee responses to Conditions B.1, B.2, C.1, C.2, C.3, and C.4 above shall be submitted in accordance with 10 CFR 50.4, "Written Communications."

The Director, Office of Nuclear Reactor Regulation may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee of good cause.

V.

In accordance with 10 CFR 2.202, the Licensee must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this Order, within twenty (20) days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to answer or to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order.

If a hearing is requested by a Licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Pursuant to 10 CFR 2.202(c)(2)(i), the licensee or any other person adversely affected by this Order, may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, web-based submission form. In order to serve documents through the Electronic Information Exchange, users will be required to install a web browser plug-in from the NRC web site. Further information on the web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd1.nrc.gov/ehd/*, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

If a person other than the Licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d).

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Dated this th day of

OPERATING BOILING WATER REACTOR LICENSES
WITH MARK I AND MARK II CONTAINMENTS

Browns Ferry Nuclear Plant, Units 1, 2, and 3	BWR-Mark I
Brunswick Steam Electric Plant, Units 1 and 2	BWR-Mark I
Columbia Generating Station	BWR-Mark II
Cooper Nuclear Station	BWR-Mark I
Dresden Nuclear Power Station, Units 2 and 3	BWR-Mark I
Duane Arnold Energy Center	BWR-Mark I
Edwin I. Hatch Nuclear Plant, Units 1 and 2	BWR-Mark I
Fermi	BWR-Mark I
Hope Creek Generating Station	BWR-Mark I
James A. FitzPatrick Nuclear Power Plant	BWR-Mark I
LaSalle County Station, Units 1 and 2	BWR-Mark II
Limerick Generating Station, Units 1 and 2	BWR-Mark II
Monticello Nuclear Generating Plant	BWR-Mark I
Nine Mile Point Nuclear Station, Units 1 and 2	BWR-Mark I & II
Oyster Creek Nuclear Generating Station	BWR-Mark I
Peach Bottom Atomic Power Station, Units 2 and 3	BWR-Mark I
Pilgrim Nuclear Power Station	BWR-Mark I
Quad Cities Nuclear Power Station, Units 1 and 2	BWR-Mark I
Susquehanna Steam Electric Station, Units 1 and 2	BWR-Mark II
Vermont Yankee Nuclear Power Station	BWR-Mark I

REQUIREMENTS FOR RELIABLE HARDENED VENT SYSTEMS
CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS
AT BOILING-WATER REACTOR FACILITIES WITH
MARK I AND MARK II CONTAINMENTS

In accordance with Order EA-12-050 dated March 12, 2012, Boiling-Water Reactors (BWRs) with Mark I and Mark II containments were required to have a reliable hardened containment venting system (HCVS). This Order requires that these facilities ensure that the HCVS originally required by Order EA-12-050 also provide a reliable hardened venting capability from the wetwell and drywell under severe accident conditions. The severe accident capable HCVS is intended to prevent severe accidents from occurring, and to help mitigate the consequences of a severe accident should one occur. The HCVS shall meet the requirements in Sections 1, 2, and 3, below. In addition, the Licensee shall meet the requirements of Section 4.

1. <u>HCVS Functional Requirements</u>

BWRs with Mark I and Mark II containments shall have a reliable HCVS to remove decay heat; vent the containment atmosphere including steam, hydrogen, non-condensable gases, aerosols, and fission products; and control containment pressure within acceptable limits. The HCVS shall be designed for those accident sequences wherein containment venting is expected or relied upon to prevent containment failure; including accident sequences that result in the loss of active containment heat removal capability or prolonged Station Blackout (SBO).

1.1 The design of the HCVS shall consider the following performance objectives:

1.1.1 The HCVS shall be designed to minimize the reliance on operator actions through incorporation of passive features to the extent practical.

1.1.2 The HCVS shall be designed to minimize plant operators' exposure to occupational hazards, such as extreme heat stress, while operating the HCVS system.

1.1.3 The HCVS shall also be designed to minimize radiological consequences that would impede personnel actions needed for event response.

1.1.4 The HCVS shall be accessible and operable under a range of plant conditions, including a severe accident environment, prolonged SBO and inadequate containment cooling.

1.2 The HCVS shall include the following reliable hardened venting design features:

1.2.1 The HCVS shall have the capacity to vent the steam/energy equivalent of 1 percent of licensed/rated thermal power (unless a lower value is justified by analyses), and be able to maintain containment pressure below the primary containment design pressure and the primary containment pressure limit (PCPL).

1.2.2 The HCVS shall be capable of venting from the suppression chamber (wetwell) and the drywell. The wetwell vent path shall include means for passive (i.e. rupture disk) and active operation, and include means to isolate the passive vent pathway.

1.2.3 The HCVS shall discharge the effluent to a release point above main plant structures.

1.2.4 The HCVS shall include design features to preclude cross flow of vented fluids within a unit and between units on the site.

1.2.5 The HCVS shall be designed to be manually operated during sustained operations from a control panel located in the main control room or a remote but readily accessible location. "Sustained operations" means until such time that reliable containment heat removal and pressure control is reestablished independent of the HCVS.

1.2.6 The HCVS shall also be capable of local manual operation (e.g., reach-rod with hand wheel or manual operation of pneumatic supply valves from a shielded location). All local manual HCVS controls shall be accessible to plant operators during sustained operations.

1.2.7 The HCVS shall be capable of operating with dedicated and permanently installed equipment for at least 24 hours following the loss of normal power or loss of normal pneumatic supplies to air operated components during the prolonged SBO.

1.2.8 The HCVS shall include means to prevent inadvertent actuation.

1.2.9 The HCVS shall include means to monitor the status of the vent system (e.g., valve position indication) from the control panel installed in accordance with requirement 1.2.5. The monitoring system shall be designed for sustained operation during a prolonged SBO.

1.2.10 The HCVS shall include a means to monitor the effluent discharge for radioactivity that may be released from operation of the HCVS. The monitoring system shall provide indication from the control panel installed in accordance with requirement 1.2.5, and shall be designed for sustained operation during a prolonged SBO.

1.2.11 The HCVS (excluding the rupture disk) shall be designed for pressures that are consistent with containment design pressures and expected temperatures during a severe accident as well as dynamic loading resulting from system actuation.

1.2.12 The HCVS shall be designed and operated to ensure the flammability limits of gases passing through the system are not reached; otherwise, the system shall be designed to withstand dynamic loading resulting from hydrogen deflagration and detonation.

1.2.13 The HCVS shall incorporate strategies for hydrogen control that minimizes the potential for hydrogen gas migration and ingress into the reactor building or other buildings.

1.2.14 The HCVS shall include features and provision for the operation, testing, inspection and maintenance adequate to ensure that reliable function and capability are maintained.

2. <u>HCVS Quality Standards</u>

The following quality standards are necessary to fulfill the requirements for a reliable HCVS:

2.1 The HCVS vent path up to and including the second containment isolation barrier shall be designed consistent with the design basis of the plant. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators and containment isolation valve position indication components.

2.2 All other HCVS components shall be designed for reliable and rugged performance that is capable of ensuring HCVS functionality following a seismic event. These items include electrical power supply, valve actuator pneumatic supply and instrumentation (local and remote) components.

2.3 All FCVS instrumentation shall be designed and constructed to withstand seismic loadings consistent with the design basis of the plant.

3. <u>HCVS Programmatic Requirements</u>

3.1 The Licensee shall develop, implement, and maintain procedures necessary for the safe operation of the HCVS. Procedures shall be established for system operations when normal and backup power is available, and during SBO conditions.

3.2 The Licensee shall train appropriate personnel in the use of the HCVS. The training curricula shall include system operations when normal and backup power is available, and during SBO conditions.

4. <u>Additional Requirements</u>

4.1 Licensees shall make necessary modifications to address the potential for suppression pool bypass due to molten core debris melting through susceptible drain lines and downcomers. Acceptable approaches could include providing protection for the susceptible drain lines and downcomers, or installation of an engineered filtered containment venting system.

Enclosure 7B

Draft Proposed Order

Option 3

Filtered Containment Vents

Note: It is likely that this draft proposed order will require revision based on interactions with stakeholders and continuing internal discussions on technical and legal issues. If the Commission approves Option 3, the staff will provide the Commission with a final order via a Regulatory Notification.

7590-01-P

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
ALL OPERATING BOILING WATER	)	Docket Nos. (as shown in Attachment 1)
REACTOR LICENSEES WITH	)	License Nos. (as shown in Attachment 1)
MARK I AND MARK II CONTAINMENTS	)	EA-12-XXX
	)	

[NRC-20XX-XXXX]

ORDER MODIFYING LICENSES WITH REGARD TO FILTERED RELIABLE HARDENED CONTAINMENT VENTS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS (EFFECTIVE IMMEDIATELY)

I.

The Licensees identified in Attachment 1 to this Order hold licenses issued by the U.S. Nuclear Regulatory Commission (NRC or Commission) authorizing operation of nuclear power plants in accordance with the Atomic Energy Act of 1954, as amended, and Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Specifically, these Licensees operate boiling-water reactors (BWRs) with Mark I and Mark II containment designs.

II.

On March 12, 2012, the NRC issued Order EA-2012-050 requiring the Licensees identified in Attachment 1 to this Order to implement requirements for reliable hardened vents for Mark I and Mark II containments. This Order supersedes Order EA-12-050 by revising requirements imposed in Order EA-12-050 and imposing additional requirements on reliable hardened vent systems and related procedures to ensure that the venting function is maintained during severe accident conditions (i.e., following significant core damage) and incorporates an engineered filter in the venting discharge paths from the suppression pool and drywell to limit the release of radioactive materials.

Order EA-12-050 requires that licensees of BWR facilities with Mark I and Mark II containment designs shall ensure that these facilities have a containment venting system that meets certain requirements relating to reliable and dependable operation to support strategies relating to the control of containment pressure and prevention of core damage following events causing a loss of heat removal systems. The NRC determined that the issuance and implementation of the requirements in Order EA-12-050 were necessary to provide reasonable assurance of adequate protection of the public health and safety and assurance of the common defense and security. As described in Order EA-12-050:

> To protect public health and safety from the inadvertent release of radioactive materials, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.
>
> The events at Fukushima Dai-ichi highlight the possibility that extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. The inability to reduce containment pressure inhibited efforts to cool the reactor core. If additional backup or alternate sources of power had been available to operate the containment venting system remotely, or if certain valves had been more accessible for manual operation, the operators at Fukushima may have been able to depressurize the containment earlier. This, in turn, could have allowed operators to implement strategies using low-pressure water sources that

> may have limited or prevented damage to the reactor core. Thus, the events at Fukushima demonstrate that reliable hardened vents at BWR facilities with Mark I and Mark II containment designs are important to maintain core and containment cooling.
>
> The Commission has determined that ensuring adequate protection of public health and safety requires that all operating BWR facilities with Mark I and Mark II containments have a reliable hardened venting capability for events that can lead to core damage. These new requirements provide greater mitigation capability consistent with the overall defense-in-depth philosophy, and therefore greater assurance that the challenges posed by severe external events to power reactors do not pose an undue risk to public health and safety. To provide reasonable assurance of adequate protection of public health and safety, all licenses identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order.
>
> Accordingly, the NRC has concluded that these measures are necessary to ensure adequate protection of public health and safety under the provisions of the backfit rule, 10 CFR 50.109(a)(4)(ii), and is requiring Licensee actions. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

In developing the requirements included in Order EA-12-050, the NRC acknowledged that questions remained about possible ways to limit the release of radioactive materials if the venting systems were used after significant core damage had occurred. The NRC staff described in SECY-12-XXXX, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," various options for Commission consideration. One of the options in SECY-12-XXXX (Option 3) was to revise the requirements in Order EA-12-050 to ensure that the venting function is maintained during severe accident conditions and incorporates a filter technology to limit the release of radioactive materials. In its Staff Requirements Memorandum (SRM) for SECY-12-XXXX , the Commission documented its decision to require the affected licensees to provide a containment venting systems capable of operation under severe accident conditions with filters in the discharge paths and directed the NRC staff to implement that requirement through the issuance of this Order.

The desire to ensure that the venting function is maintained during severe accident conditions is to provide protection from events that might otherwise cause containment failure due to high pressures. The addition of filters to the discharge paths from the wetwell and drywell is intended to limit the release of radioactive materials that would occur during a containment venting operation during severe accident conditions. It is equally important to prevent core debris that has melted through the reactor vessel from breaching the containment structures. New regulatory requirements were imposed as item B.5.b in Order 02-026 and later incorporated into 10 CFR 50.54(hh) that require licensees to have the capability to direct water into the drywells to reduce the chances of containment failures from a molten core.

III.

The NRC may impose safety measures on licensees or applicants over and above those required by the adequate-protection standard cited in Order EA-12-050. Such requirements may be pursued to protect health or to minimize danger to life or property. As described in various NRC regulations and guidance documents, the requirements established to reduce risk (beyond measures needed for adequate protection) will not attempt to eliminate all risk but will instead pursue reasonable reductions. An evaluation of the costs and benefits of proposals within this category has been used as part of the determination of what is a reasonable requirement to reduce risks to public health and safety and the common defense and security. The NRC process for evaluating costs and benefits to help determine if additional requirements should be imposed are defined in Section 50.109, "Backfitting," of Title 10 of the *Code of Federal Regulations.* Additional information and guidance related to these assessments are provided in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook."

An evaluation of the costs and benefits associated with providing a venting function for BWRs with Mark I and Mark II containments that include filtering the releases from the wetwell and drywell during severe accident conditions was summarized in SECY-12-XXXX. As discussed in SECY-12-XXXX, the NRC's determination that a filtered venting system should be available during severe accident conditions considered both quantitative assessments of costs and benefits as well as various qualitative factors. Among the qualitative factors, one of the more important is the desire to improve the defense-in-depth characteristics of Mark I and Mark II containments by addressing the high conditional failure probabilities that those containments have should an event lead to a core melt. As discussed in SECY-12-XXXX, other qualitative factors supporting installation of a filtered venting system include addressing significant uncertainties in the understanding of severe accident events, supporting severe accident management and response, improving the control of hydrogen generated during severe accidents, improving readiness for external and multi-unit events, reducing uncertainties about radiological releases and thereby improving emergency planning and response, and maintaining consistency with interactional practices.

As previously discussed, the NRC's defense-in-depth strategy includes multiple layers of protection: (1) prevention of accidents by virtue of the design, construction and operation of the plant, (2) mitigation features to prevent radioactive releases should an accident occur, and (3) emergency preparedness programs that include measures such as sheltering and evacuation. The defense-in-depth strategy also provides for multiple physical barriers to contain the radioactive materials in the event of an accident. The barriers are the fuel cladding, the reactor coolant pressure boundary, and the containment. These defense-in-depth features are embodied in the existing regulatory requirements and thereby provide adequate protection of public health and safety.

The events at Fukushima Dai-ichi highlight the possibility that events such as extreme natural phenomena could challenge the prevention, mitigation and emergency preparedness defense-in-depth layers. At Fukushima, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to prevent core damage and containment failure. In particular, the operators were unable to successfully operate the containment venting system. These problems with venting the containments under the challenging conditions following the tsunami contributed to the progression of the accident from inadequate cooling of the core leading to core damage, to compromising containment functions from overpressure conditions, and to the hydrogen explosions that destroyed the reactor buildings (secondary containments). The loss of the various barriers led to the release of radioactive materials and further hampered operator efforts to arrest the accidents that ultimately led to the contamination of large areas surrounding the plant. The evacuation of local populations minimized the immediate danger to public health and safety from the loss of control of the large amount of radioactive materials within the reactor cores.

The actions imposed by Order are intended to increase confidence in maintaining the containment function following core damage events and filtering releases associated with the venting operations.

The NRC is empowered to require plant improvements beyond those needed to provide reasonable assurance of adequate protection of public health and safety when engineering approaches are available to provide a cost-justified substantial safety improvement. An evaluation of possible improvements for Mark I and Mark II containment vents was provided in SECY-12-XXXX and a more detailed regulatory analysis that is available in the NRC's agencywide documents access and management system (ADAMS). These evaluations included both quantitative and qualitative factors and led the Commission to determine that the safety improvements required by this Order are justified. In such situations, the Commission may act in accordance with its statutory authority under Section 161 of the Atomic Energy Act of 1954, as amended, to require Licensees to take appropriate action to reduce the risks posed to the public from the operation of nuclear power plants.

The Commission has determined that it is a cost-justified substantial safety improvement to require BWR facilties with Mark I and Mark II containments to make the necessary plant modifications and procedure changes to provide a reliable hardened venting capability that is capable of performing under severe accident conditions and incorporates filtering technologies to limit the release of radioactive materials from venting from either the suppression pool or drywell. These new requirements protect health and minimize danger to life or property by having licensees provide greater capabilities to respond to severe accidents and contain radioactive materials, which is consistent with the NRC's overall defense-in-depth philosophy. To provide an enhanced level of safety, all licenses identified in Attachment 1 to this Order shall be modified to include the requirements identified in Attachment 2 to this Order.

Accordingly, the NRC has concluded that these measures are an appropriate cost-justified safety improvement under the provisions of the backfit rule, 10 CFR 50.109(a)(3), and is requiring Licensee actions. In addition, pursuant to 10 CFR 2.202, the NRC finds that the public health, safety and interest require that this Order be made immediately effective.

IV.

Accordingly, pursuant to Sections 161b, 161i, 161o, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR 2.202, “Orders,” and 10 CFR Part 50, IT IS HEREBY ORDERED, EFFECTIVE IMMEDIATELY, THAT ALL LICENSES IDENTIFIED IN ATTACHMENT 1 TO THIS ORDER ARE MODIFIED AS FOLLOWS:

A. All Licensees shall, notwithstanding the provisions of any Commission regulation or license to the contrary, comply with the requirements described in Attachment 2 to this Order except to the extent that a more stringent requirement is set forth in the license. The requirements of Attachment 2 to this Order supersede those set forth in Attachment 2 to Order EA-12-050 dated March 12, 2012. These Licensees shall promptly start implementation of the requirements in Attachment 2 to the Order and shall complete full implementation **no later than December 31, 2017**.

B. 1. All Licensees shall, within **twenty (20) days** of the date of this Order, notify the Commission (1) if they are unable to comply with any of the requirements described in Attachment 2, (2) if compliance with any of the requirements is unnecessary in their specific circumstances, or (3) if implementation of any of the requirements would cause the Licensee to be in violation of the provisions of any Commission regulation or the facility license. The notification shall provide the Licensee’s justification for seeking relief from or variation of any specific requirement.

2. Any Licensee that considers that implementation of any of the requirements described in Attachment 2 to this Order would adversely affect the safe and secure operation of the facility must notify the Commission, within **twenty (20) days** of this Order, of the adverse safety impact, the basis for its determination that the

requirement has an adverse safety impact, and either a proposal for achieving the same objectives specified in the Attachment 2 requirement in question, or a schedule for modifying the facility to address the adverse safety condition. If neither approach is appropriate, the Licensee must supplement its response to Condition B.1 of this Order to identify the condition as a requirement with which it cannot comply, with attendant justifications as required in Condition B.1.

C. 1. All Licensees shall, **within 6 months following the issuance of interim staff guidance (ISG) for this order**, submit to the Commission for review an overall integrated plan including a description of how compliance with the requirements described in Attachment 2 will be achieved. The NRC staff plans to issue the ISG no later than **[insert date 120 days from date of this order]**.

2. All Licensees shall provide status reports at **six (6)-month intervals** following submittal of the overall integrated plan, as required in Condition C.1, which delineates progress made in implementing the requirements of this Order.

3. All Licensees shall report to the Commission when full compliance with the requirements described in Attachment 2 is achieved.

Licensee responses to Conditions B.1, B.2, C.1, C.2, and C.3 above shall be submitted in accordance with 10 CFR 50.4, “Written Communications.”

The Director, Office of Nuclear Reactor Regulation may, in writing, relax or rescind any of the above conditions upon demonstration by the Licensee of good cause.

V.

In accordance with 10 CFR 2.202, the Licensee must, and any other person adversely affected by this Order may, submit an answer to this Order, and may request a hearing on this

Order, within twenty (20) days of the date of this Order. Where good cause is shown, consideration will be given to extending the time to answer or to request a hearing. A request for extension of time in which to submit an answer or request a hearing must be made in writing to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and include a statement of good cause for the extension. The answer may consent to this Order.

If a hearing is requested by a Licensee or a person whose interest is adversely affected, the Commission will issue an Order designating the time and place of any hearings. If a hearing is held, the issue to be considered at such hearing shall be whether this Order should be sustained. Pursuant to 10 CFR 2.202(c)(2)(i), the licensee or any other person adversely affected by this Order, may, in addition to demanding a hearing, at the time the answer is filed or sooner, move the presiding officer to set aside the immediate effectiveness of the Order on the ground that the Order, including the need for immediate effectiveness, is not based on adequate evidence but on mere suspicion, unfounded allegations, or error.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC E-Filing rule (72 FR 49139, August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at (301) 415-1677, to request (1) a digital ID certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, web-based submission form. In order to serve documents through the Electronic Information Exchange, users will be required to install a web browser plug-in from the NRC web site. Further information on the web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with NRC guidance available on the

NRC public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail at MSHD.Resource@nrc.gov, or by a toll-free call at (866) 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format. Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in NRC's electronic hearing docket, which is available to the public at *http://ehd1.nrc.gov/ehd/*, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

If a person other than the Licensee requests a hearing, that person shall set forth with particularity the manner in which his interest is adversely affected by this Order and shall address the criteria set forth in 10 CFR 2.309(d).

In the absence of any request for hearing, or written approval of an extension of time in which to request a hearing, the provisions specified in Section IV above shall be final twenty (20) days from the date of this Order without further order or proceedings. If an extension of time for requesting a hearing has been approved, the provisions specified in Section IV shall be final when the extension expires if a hearing request has not been received. AN ANSWER OR A

REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

FOR THE NUCLEAR REGULATORY COMMISSION

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Dated this th day of

OPERATING BOILING WATER REACTOR LICENSES
WITH MARK I AND MARK II CONTAINMENTS

Browns Ferry Nuclear Plant, Units 1, 2, and 3	BWR-Mark I
Brunswick Steam Electric Plant, Units 1 and 2	BWR-Mark I
Columbia Generating Station	BWR-Mark II
Cooper Nuclear Station	BWR-Mark I
Dresden Nuclear Power Station, Units 2 and 3	BWR-Mark I
Duane Arnold Energy Center	BWR-Mark I
Edwin I. Hatch Nuclear Plant, Units 1 and 2	BWR-Mark I
Fermi	BWR-Mark I
Hope Creek Generating Station	BWR-Mark I
James A. FitzPatrick Nuclear Power Plant	BWR-Mark I
LaSalle County Station, Units 1 and 2	BWR-Mark II
Limerick Generating Station, Units 1 and 2	BWR-Mark II
Monticello Nuclear Generating Plant	BWR-Mark I
Nine Mile Point Nuclear Station, Units 1 and 2	BWR-Mark I & II
Oyster Creek Nuclear Generating Station	BWR-Mark I
Peach Bottom Atomic Power Station, Units 2 and 3	BWR-Mark I
Pilgrim Nuclear Power Station	BWR-Mark I
Quad Cities Nuclear Power Station, Units 1 and 2	BWR-Mark I
Susquehanna Steam Electric Station, Units 1 and 2	BWR-Mark II
Vermont Yankee Nuclear Power Station	BWR-Mark I

REQUIREMENTS FOR FILTERED CONTAINMENT VENT SYSTEMS CAPABLE OF OPERATION UNDER SEVERE ACCIDENT CONDITIONS AT BOILING-WATER REACTOR FACILITIES WITH MARK I AND MARK II CONTAINMENTS

In accordance with Order EA-12-050 dated March 12, 2012, Boiling-Water Reactors (BWRs) with Mark I and Mark II containments were required to have reliable hardened containment venting system (HCVS). This Order requires that these facilities ensure that the HCVS originally required by Order EA-12-050 also provide a reliable hardened venting capability from the wetwell and drywell under severe accident conditions that include an engineered filtering system. The HCVS with an engineered filtration venting capability is designated as the filtered containment venting system (FCVS). The FCVS is intended to prevent severe accidents from occurring, and to mitigate the consequences of a severe accident should one occur. The FCVS shall meet the requirements in Sections 1, 2, and 3 below.

1. FCVS Functional Requirements

BWR with Mark I and Mark II containments shall have a reliable FCVS to remove decay heat; vent the containment atmosphere including steam, hydrogen, non-condensable gases, aerosols, and fission products; capture fission products released during a severe accident; and control containment pressure within acceptable limits. The FCVS shall be designed for those accident sequences wherein containment venting is expected or relied upon to prevent containment failure; including accident sequences that result in the loss of active containment heat removal capability or prolonged Station Blackout (SBO).

1.1 The design of the FCVS shall consider the following performance objectives:

1.1.1 The FCVS shall be designed to minimize the reliance on operator actions through the incorporation of passive features to the extent practical.

1.1.2 The FCVS shall be designed to minimize plant operators' exposure to occupational hazards, such as extreme heat stress, while operating the FCVS system.

1.1.3 The FCVS shall also be designed to minimize radiological consequences that would impede personnel actions needed for event response.

1.1.4 The FCVS shall be accessible and operable under a range of plant conditions, including a severe accident environment, prolonged SBO, and inadequate containment cooling.

1.2 The FCVS shall include the following reliable hardened venting design features:

1.2.1 The FCVS, including filter, shall have the capacity to vent the steam/energy equivalent of 1 percent of licensed/rated thermal power (unless a lower value is justified by analyses), and be able to maintain containment pressure below the primary containment design pressure and the primary containment pressure limit (PCPL).

1.2.2 The FCVS shall be capable of venting from the suppression chamber (wetwell) and the drywell. The drywell vent path shall include means for passive (i.e. rupture disk) and active operation, and include means to isolate the passive vent pathway.

1.2.3 The FCVS shall discharge the effluent to a release point above main plant structures.

1.2.4 The FCVS shall include design features to preclude cross flow of vented fluids within a unit and between units on the site.

1.2.5 The FCVS shall be designed to be manually operated during sustained operations from a control panel located in the main control room or a remote but readily accessible location. "Sustained operations" means until such time that reliable containment heat removal and pressure control is reestablished independent of the FCVS.

1.2.6 The FCVS shall also be capable of local manual operation (e.g., reach-rod with hand wheel or manual operation of pneumatic supply valves from a shielded location). All local manual FCVS controls shall be accessible to plant operators during sustained operations.

1.2.7 The FCVS shall be capable of operating with dedicated and permanently installed equipment for at least 24 hours following the loss of normal power or loss of normal pneumatic supplies to air operated components during the prolonged SBO.

1.2.8 The FCVS shall include means to prevent inadvertent actuation.

1.2.9 The FCVS shall include means to monitor the status of the vent system (e.g., valve position indication, important filter parameters, such as water level) from the control panel installed in accordance with requirement 1.2.5, and shall be designed for sustained operation during a prolonged SBO.

1.2.10 The FCVS shall include a means to monitor the effluent discharge for radioactivity that may be released from operation of the FCVS. The monitoring system shall provide indication from the control panel installed in accordance with requirement 1.2.5, and shall be designed for sustained operation during a prolonged SBO.

1.2.11 The FCVS (excluding the rupture disk) shall be designed for pressures that are consistent with containment design pressures and expected temperatures during a severe accident as well as dynamic loading resulting from system actuation.

1.2.12 The FCVS shall be designed and operated to ensure the flammability limits of gases passing through the system are not reached or the system shall be designed to withstand dynamic loading resulting from hydrogen deflagration and detonation.

1.2.13 The FCVS shall incorporate strategies for hydrogen control that minimizes the potential for hydrogen gas migration and ingress into the reactor building or other buildings.

1.2.14 The FCVS shall include features and provision for the operation, testing, inspection and maintenance adequate to ensure that reliable function and capability are maintained.

1.3 The FCVS shall include a filter that is capable of reducing the release of radioactive materials passing through the venting system by an amount that is reasonably achievable (e.g., decontamination factors on the order of 1000 for aerosols and 100 for iodine) using filtering technologies available as of the date of this Order.

1.3.1 The FCVS filter shall be sized based on the results of severe accident analyses, including consideration of both in-vessel and ex-vessel severe accident phenomena. The analyses shall include consideration of those accident sequences wherein containment venting is expected or relied upon to prevent containment failure. The analyses shall form the basis for FCVS sizing parameters such as the quantity, type, size, and form of radioactive and non-radioactive aerosols; containment atmosphere pressure and temperature; vent flow rates and gas composition including steam, aerosols, and non-condensable gases; and decay heat to be removed.

1.3.2 The FCVS filter shall be capable of passive operation with no operator actions required for 24 hours following the initiation of containment venting.

2. <u>FCVS Quality Standards</u>

The following quality standards are necessary to fulfill the requirements for a reliable FCVS:

2.1 The FCVS vent path up to and including the second containment isolation barrier shall be designed consistent with the design basis of the plant. These items include piping, piping supports, containment isolation valves, containment isolation valve actuators and containment isolation valve position indication components.

2.2 All other FCVS components shall be designed for reliable and rugged performance that is capable of ensuring FCVS functionality following a seismic event. These items include electrical power supply, valve actuator pneumatic supply and instrumentation (local and remote) components.

2.3 All FCVS instrumentation shall be designed and constructed to withstand seismic loadings consistent with the design basis of the plant.

3. <u>FCVS Programmatic Requirements</u>

3.1 The Licensee shall develop, implement, and maintain procedures necessary for the safe operation of the FCVS. Procedures shall be established for system operations when normal and backup power is available, and during SBO conditions.

3.2 The Licensee shall train appropriate personnel in the use of the FCVS. The training curricula shall include system operations when normal and backup power is available, and during SBO conditions.