## Closure of Tier 3 Recommendations 5.2 and 6

## Reliable Hardened Vents for Other Containment Designs and Hydrogen Control and Mitigation inside Containment and Other Buildings

In SECY-15-0137, "Proposed Plans for Resolving Open Fukushima Tier 2 and 3 Recommendations," the U.S. Nuclear Regulatory Commission (NRC) staff provided an initial assessment and basis for closing the recommendations from the Near-Term Task Force (NTTF) related to: (1) possible regulatory requirements for reliable hardened vents for plants with other than Mark I and II containments, and (2) hydrogen control and mitigation. The staff stated in SECY-15-0137 that additional interactions with the Advisory Committee on Reactor Safeguards (ACRS) and external stakeholders were planned and more detailed documentation, incorporating insights from these interactions, would be provided to the Commission.

This final evaluation addresses the observations provided by the ACRS in their letter dated November 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15320A074), on the staff's initial assessments in SECY-15-0137. The staff met publicly on January 7, 2016, with external stakeholders to gather information and insights regarding the issues and initial conclusions discussed in Enclosure 4 to SECY-15-0137, which deals with Recommendations 5.2 and 6. The staff addresses insights from those interactions in the following justification for the closure of those recommendations. The NRC staff also met with the ACRS Fukushima Subcommittee on February 18, 2016, and the Full Committee on March 3, 2016. The staff considered insights from these interactions with the ACRS during the preparation of this evaluation. For the sake of completeness, this discussion includes the initial assessment from Enclosure 4 to SECY-15-0137, and the marked text provides additional discussions to address questions and insights from interactions with various stakeholders.

### Background

As described in SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," dated December 23, 2011 (ADAMS Accession No. ML11186A950), the NRC's Near-Term Task Force identified Recommendation 5.2, which recommended that the NRC assess the need to require the installation of reliable, hardened venting systems for containments with designs other than Mark I and II (which are addressed as part of Recommendation 5.1). The NTTF also recommended that the staff assess the need to further strengthen requirements associated with hydrogen control and mitigation inside and outside reactor containment buildings as part of NTTF Recommendation 6. In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011 (ADAMS Accession No. ML11272A111), the staff prioritized these as Tier 3 activities because they required further staff study and the insights from implementation of Recommendation 5.1 and related international activities to support a regulatory decision.

In SECY-11-0137, the NRC staff described its proposals for immediate regulatory actions and longer-term evaluations to address the NTTF recommendations. Among the highest-priority Tier 1 actions that the NRC staff proposed was the issuance of orders to address Recommendation 5.1, requiring reliable hardened containment vents for those licensees of boiling-water reactors (BWRs) with Mark I and II containment designs. Venting Mark I and II containments can help prevent the loss of, and facilitate recovery of, important safety functions, such as reactor core cooling, reactor coolant inventory control, containment cooling, and containment pressure control. The NRC issued Order EA-12-050, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents," on March 12, 2012 (ADAMS Accession No. ML12054A694), requiring reliable, hardened vents for these plants. The NRC subsequently revised these requirements by Order EA-13-109, "Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013 (ADAMS Accession No. ML13130A067), to make the venting systems for Mark I and II containments capable of operation during severe accident conditions.

The NRC staff has been actively participating in various international studies, including a working group studying hydrogen generation, transport, and risk management organized by the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA). The NRC staff has also gathered insights from other Fukushima-related activities, as well as probabilistic risk studies, previous evaluations of generic issues, operating experience, and other available information. These insights are being used to help assess whether the results of additional studies of containment performance and the control of hydrogen following potential severe reactor accidents would justify regulatory actions beyond those already taken for plants with Mark I and II containments.

Containment performance and the control of hydrogen have been the focus of a number of previous NRC studies and evaluations. In addition to the recent evaluations related specifically to Mark I and II containments, the NRC completed detailed assessments as part of the Containment Performance Improvement Program (CPIP) in the 1980s, resolved generic safety issues, and established requirements such as Section 50.44, "Combustible Gas Control for Nuclear Power Plants," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." Containment performance and hydrogen-related issues have also been addressed in major studies, such as those documented in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued December 1990, and NUREG-1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," issued November 2012.

The NRC staff described the CPIP effort in SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988. This effort evaluated generic severe accident challenges for each light water reactor (LWR) containment type to determine whether additional regulatory guidance or requirements concerning containment features were warranted. Therefore, the CPIP is especially relevant to this evaluation. The CPIP was initiated to address uncertainties in the ability of LWR containments to successfully survive some severe accident challenges, consistent with the results documented in NUREG-1150. All LWR containment types were assessed in the CPIP, but as in more recent evaluations, many of the activities were focused on Mark I and II containment designs. The CPIP identified potential improvements for Mark I and II designs that were provided to licensees for consideration in performing individual plant examinations (IPEs) and resulted in plant changes for Mark I plants as described in Generic Letter 89-16, "Installation of Hardened Wetwell Vent." Some of these features were further enhanced through the Tier 1 activities associated with Orders EA-12-049, "Mitigating Strategies for Beyond Design Basis External Events," dated March 12, 2012 (ADAMS Accession No. ML12054A735), and Order EA-13-109. As described in NUREG-0933, "Resolution of Generic Safety Issues," published December 2011, the NRC staff did not identify generic improvements that would apply to Mark III, ice condenser, or large dry containments. Rather, the staff requested that licensees with plants with these containment designs consider insights from the CPIP within the IPEs.

The NRC has also addressed containment performance issues and the role of the containment in limiting the consequences of severe accidents in research programs, resolving generic safety issues, and evaluating regulatory actions that were ultimately not pursued because the possible action was found to provide only minimal safety benefits. Many of the NRC-sponsored research projects related to containment performance are described in NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories," published in July 2006. NUREG-0933 describes the NRC's assessment and closure of various containment-related issues, including the activities within the CPIP. The NRC and licensees have also addressed containment performance issues through the development and revision of regulatory requirements within plant technical specifications and in the development of regulations such as 10 CFR 50.44. The NRC staff evaluated various issues and potential improvements to containment performance as part of internal initiatives (e.g., SOARCA) and in response to petitions for enforcement action or rulemaking. These activities have collectively added to the body of knowledge related to containment performance and the control of hydrogen following severe reactor accidents. The activities undertaken in response to the Fukushima accident provide additional insights and have resulted in regulatory actions, such as issuance of Orders EA-12-049 and EA-13-109, which further enhance safety.

The NRC has taken actions, such as issuing Order EA-13-109, to address lessons learned from the Fukushima accident, and evaluated other potential changes to regulatory requirements and agency policies. The staff also benefited from interactions with the Commission, ACRS, and other stakeholders regarding the initial assessment of issues presented in SECY-15-0137. The staff met publicly on January 7, 2016, with external stakeholders to gather information and insights regarding the issues and initial conclusions discussed in Enclosure 4 to SECY-15-0137 dealing with Recommendations 5.2 and 6. The staff addressed insights from those interactions in the following justification for the closure of those recommendations. The staff met with the ACRS Fukushima Subcommittee on February 18, 2016, and the Full Committee on March 3, 2016 (see ACRS letter dated March 15, 2016 (ADAMS Accession No. ML16075A330)). The following discussion includes and expands upon the initial assessment from Enclosure 4 to SECY-15-0137.

**Current Status**

In SECY-15-0137, the staff documented a preliminary analysis to support decisions on whether additional regulatory actions associated with Recommendations 5.2 and 6 might be warranted. The initial conclusion from that analysis was that regulatory actions beyond those completed for Mark I and II containments are not warranted. It is worth noting that industry initiatives to develop and implement improvements in severe accident management guidelines (SAMGs) are proceeding even though regulatory requirements are not being proposed as part of the rulemaking to address mitigating beyond-design-basis events. The staff considered the improvements to the SAMGs and the Commission decisions related to the regulatory treatment of SAMGs within this assessment of possible improvements to containment performance and control of hydrogen during severe accidents. In SECY-15-0137, the staff described plans to obtain stakeholder input, finalize its analysis, and complete its evaluation of these recommendations. These additional activities, discussed in more detail in the following paragraphs, support and provide justification for the closure of Recommendations 5.2 and 6. The staff will continue to evaluate information from ongoing research activities, including international programs, and will, as appropriate, inform the Commission if these activities identify the need for additional safety enhancements.

BWR Mark I and II Containments

In the March 19, 2013, staff requirements memorandum (SRM) (ADAMS Accession No. ML13078A017) to SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling-Water Reactors with Mark I and Mark II Containments," dated November 26, 2012, the Commission directed the NRC staff to: (1) issue a modification to Order EA-12-050 requiring BWR licensees with Mark I and II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions, and (2) develop a technical basis and rulemaking for filtering strategies with drywell filtration and severe accident management for BWR Mark I and II containments. The staff subsequently issued Order EA-13-109[1], which rescinded the requirements imposed by Order EA-12-050 and replaced them with the following requirements for licensees of BWRs with Mark I and II containments:

- Phase 1: Upgrade the venting capabilities from the containment wetwell to provide reliable, severe accident capable hardened vents to assist in preventing core damage and, if necessary, to provide venting capability under severe accident conditions.

- Phase 2: Install a reliable, severe-accident-capable drywell vent, or develop a reliable containment venting strategy that makes it unlikely the site would need to vent from the containment drywell during a severe accident.

The NRC's interim staff guidance (ISG) for Phase 1 of the order was issued in November 2013, which endorsed the guidance developed by the Nuclear Energy Institute (NEI) and an industry working group, NEI 13-02, Revision 0 (ADAMS Accession No. ML13316A853). The NRC issued the ISG for Phase 2 requirements in April 2015. This ISG endorsed the updated industry guidance document, NEI 13-02, Revision 1 (ADAMS Accession No. ML15113B318). As required by Order EA-13-109, licensees with Mark I and II containments submitted their overall integrated plans (OIPs) for Phase 1 by June 30, 2014. The staff has completed its review of Phase 1 plans and has issued interim staff evaluations. Licensees submitted OIPs for Phase 2 of EA-13-109 by the required date of December 31, 2015.

Containment Protection and Release Reduction (CPRR) Rulemaking

As directed by the SRM for SECY-12-0157, the staff assessed possible additional requirements for containment pressure control and venting, to include measures to enhance the capability to maintain containment integrity and to cool core debris. These evaluations formed the draft regulatory basis prepared for the CPRR rulemaking. The main objective of the CPRR

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[1] Order EA-13-109 states that the requirement to provide a reliable hardened containment vent system (HCVS) to prevent or limit core damage upon loss of heat removal capability is necessary to ensure reasonable assurance of adequate protection of public health and safety, while the requirement that the reliable HCVS remain functional during severe accident conditions is a cost-justified substantial safety improvement under 10 CFR 50.109(a)(3).

regulatory basis was to determine what, if any, additional requirements are warranted related to filtering strategies and severe accident management of BWRs with Mark I and II containments assuming the installation of severe-accident-capable hardened vents per Order EA-13-109. The staff interacted with external stakeholders and identified four major alternatives for possible courses of action related to filtering strategies and severe accident management for BWRs with Mark I and II containment designs. The CPRR alternatives were the following:

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

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

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

- Alternative 4: Pursue rulemaking to address both containment protection against multiple failure modes and release reduction measures for controlling releases through the containment venting systems. This alternative would include making Order EA-13-109 generically applicable, requiring external water addition into the reactor pressure vessel or drywell, and requiring that licensees implement a strategy for managing the wetwell and drywell vents to limit releases of fission products and/or the addition of an engineered filter.

The draft regulatory basis document was provided to the Commission in SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities," dated June 25, 2015 (ADAMS Accession No. ML15022A218). In the SRM to SECY-15-0085, dated August 19, 2015 (ADAMS Accession No. ML15231A471), the Commission directed the staff to take no further action beyond those associated with implementation of Order EA-13-109 (Alternative 1). The addition of engineered filters would not provide a substantial additional safety benefit and the safety benefits of severe accident water addition are being provided by licensees for compliance with the order. In addition, the SRM directed the staff to leverage the draft regulatory basis to the extent applicable to support resolution of the post-Fukushima Tier 3 item related to containments of other designs (i.e., Recommendation 5.2).

International Activities

The NRC staff has participated in various international meetings and working groups related to reactor containment performance and has used insights from these activities to identify and evaluate technical and regulatory issues. For example, in "Staff Requirements – Briefing of the Status of Lessons Learned from the Fukushima Dai-ichi Accident," dated August 24, 2012 (ADAMS Accession No. ML122400033), the Commission directed the staff to compare practices for hydrogen control for plants in other countries with those of U.S. plants. The staff from the NRC Office of Nuclear Regulatory Research participated as members of an OECD/NEA working group conducting a study of hydrogen generation, transport, and risk management. The working group issued a report entitled, "Status Report on Hydrogen Management and Related Computer Codes," in June 2014. The report describes various containment designs, national requirements, and actions addressing lessons learned from the Fukushima accident. Measures to control hydrogen during severe accidents, including the use of passive autocatalytic recombiners, have been taken or are being pursued for many foreign plants. Currently, some countries are assessing the need for hydrogen mitigation measures outside containment, but no specific requirements have been imposed in most countries for such measures. The OECD/NEA report provides a comparison of various designs and practices for plants in the U.S. and other countries.

**Discussion**

The staff has used the insights from the technical evaluations discussed above in developing its initial assessment of Recommendations 5.2 and 6. The staff has also considered previous Commission decisions on post-Fukushima matters, regulatory analysis, and longstanding policies related to safety goals and treatment of severe accidents for operating reactors. These decisions are provided in SRMs related to a number of papers, such as the following:

- SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated August 21, 2012 (ADAMS Accession No. ML12173A478)

- SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012

- COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 25, 2013 (ADAMS Accession No. ML13329A918)

- SECY-15-0065, "Mitigation of Beyond-Design-Basis Events," dated May 15, 2015 (ADAMS Accession No. ML15049A201)

- SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities," dated June 18, 2015 (ADAMS Accession No. ML15022A218).

These decisions provide continuity between current assessments and previous evaluations of containment-related safety issues and maintain the relevance of previous regulatory and backfit analyses and the associated decisions.

Table 1 provides a summary of key functional areas related to these two recommendations and how they have been addressed for each containment type, with a more detailed assessment for each containment type provided below.

| Table 1. Recommendation 5.2 and 6 – Other Containment Designs and Hydrogen Control Requirements and Practices | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Core Cooling Functions | Venting/Heat Removal for Containment Pressure Control | | Other Containment Failure Modes/Core Debris Cooling | Release Reduction (Filtering) | Hydrogen Control | |
| | | Pre-Core Damage | Severe Accident | | | Containment | Other |
| Mark I | EA-12-049<br>EA-13-109 | EA-13-109<br>EA-12-049<br>EOPs, FSGs | EA-13-109<br>SAMGs | EA-13-109<br>(CPRR) | N/A<br>(CPRR) | 10 CFR 50.44<br>EA-13-109<br>SAMGs | EA-13-109<br>SAMGs |
| Mark II | EA-12-049<br>EA-13-109 | EA-13-109<br>EA-12-049<br>EOPs, FSGs | EA-13-109<br>SAMGs | EA-13-109<br>(CPRR) | N/A<br>(CPRR) | 10 CFR 50.44<br>EA-13-109<br>SAMGs | EA-13-109<br>SAMGs |
| Mark III | EA-12-049 | EA-12-049<br>EOPs, FSGs | SAMGs | SAMGs | N/A<br>(current assessment) | 10 CFR 50.44<br>GSI-189<br>EA-12-049<br>SAMGs, FSGs | GSI-189<br>EA-12-049<br>SAMGs, FSGs |
| Ice Condenser | n/a | EOPs | SAMGs | SAMGs | N/A<br>(current assessment) | 10 CFR 50.44<br>GSI-189<br>EA-12-049<br>SAMGs, FSGs | GSI-189<br>EA-12-049<br>SAMGs, FSGs |
| Large Dry | n/a | EOPs | SAMGs | SAMGs | N/A<br>(current assessment) | 10 CFR 50.44<br>SAMGs | N/A<br>(current assessment) |

EA-12-049: Mitigation Strategies Order
EA-13-109: BWR Mark I/II Severe accident capable vent order
EOPs: Emergency Operating Procedures
FSGs: FLEX (Mitigating Strategies) Support Guidelines
SAMGs: Severe accident management guidelines
GSI-189: Generic Safety Issue re: Hydrogen Issues

Shaded area indicates scope of Order EA-13-109 and draft regulatory basis document for CPRR rulemaking
10 CFR 50.44 defines generic and containment-specific features and analyses related to combustible gas control

## BWR Mark I and II

### Containment Performance

As part of implementing Order EA-13-109 requirements, licensees are planning to install a wetwell venting system that remains functional under severe accident conditions, and an approach involving severe accident water addition (SAWA). Licensees are expected to implement a severe accident water management (SAWM) strategy to control the water levels in the suppression pool, such that it is unlikely a licensee would need to vent from the containment drywell during severe accident conditions. The NRC staff and industry evaluations have shown that the SAWA/SAWM strategies not only support the planned approach for compliance with Order EA-13-109, but could also prevent containment failure from mechanisms other than over-pressurization. As part of developing the CPRR regulatory basis, the staff analyzed numerous alternatives in relation to the NRC's quantitative health objectives (QHOs) described in the

Safety Goal Policy Statement. As shown in Figure 1, taken from the CPRR draft regulatory basis document, significant margins are calculated between existing plant risks from an extended loss of electrical power and the NRC's safety goals; therefore, changes to Mark I and II containments beyond those required by Order EA-13-109 would not constitute substantial safety improvements.

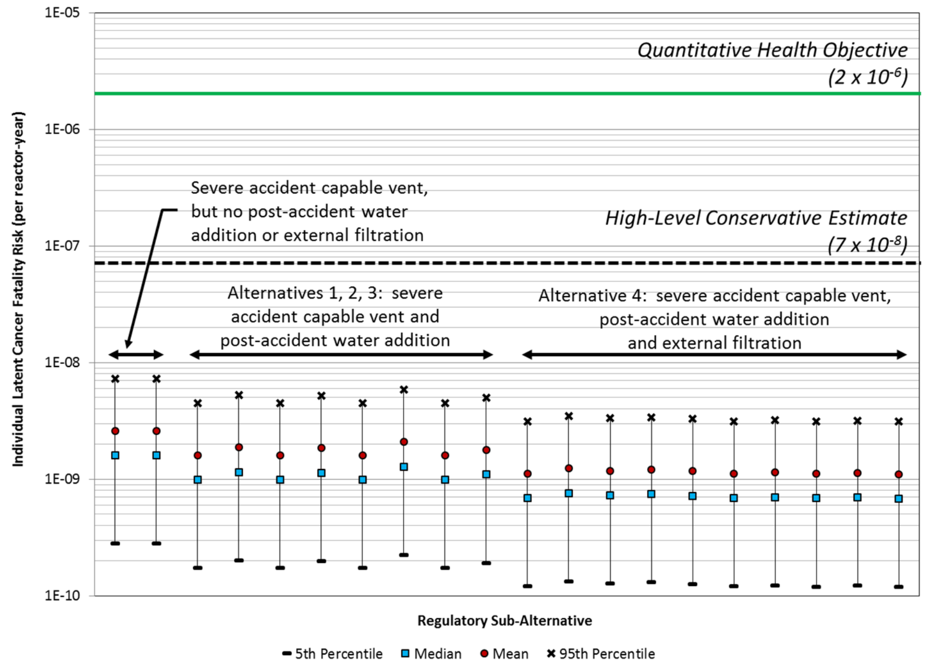


Figure 1

The Commission's SRM related to SECY-15-0085 directed the staff to discontinue further CPRR-related rulemaking activities. The actions taken under Order EA-13-109 and decisions made on SECY-15-0085 have resolved the shaded areas in Table 1 for Mark I and Mark II containments.

<u>Hydrogen Control</u>

The issue of hydrogen control in Mark I and II primary containments was considered in the technical analyses supporting the severe accident functions of Order EA-13-109 and the consideration of additional containment performance issues as part of the CPRR regulatory basis document. Mark I and II containments are inerted during normal operations to address the requirements of 10 CFR 50.44. The Fukushima accident also highlighted the possible migration of hydrogen to buildings outside the primary containment and the need to evaluate possible features or procedures to prevent explosions in the reactor building or other structures.

The NRC staff performed detailed evaluations of possible severe accident conditions, including the generation of hydrogen and other combustible gases within Mark I and II containments, as part of the work supporting Order EA-13-109 and the CPRR draft regulatory basis document. Similar studies performed by the industry are documented in the Electric Power Research Institute (EPRI) report, "Technical Basis for Severe Accident Mitigating Strategies, Volume 1," issued April 2015. In the CPRR draft regulatory basis document, the staff described the benefits of improved venting operations for the control of hydrogen in the primary containment and other buildings as follows:

> The behavior of hydrogen in the containment is shown in Figure 4-19 [Figure 2 below], "Mark I Hydrogen Generation and Transport for Case 9 (SAWA)." The blue line represents the total hydrogen generation which should be almost identical with the amount remaining inside the containment and the amount that is vented (represented by the green line). The amount of hydrogen that remains inside the containment (both the drywell and the wetwell air space as shown by the red line) quickly decreases as a result of venting. With the wetwell vent open during the transient, the total amount of hydrogen is kept very low in the long term (below 30 kg). Therefore, containment venting is very efficient in purging the hydrogen from the containment. The presence of water seems to avoid containment failure and any uncontrolled release of hydrogen to the reactor building which remains intact for the duration of the accident.

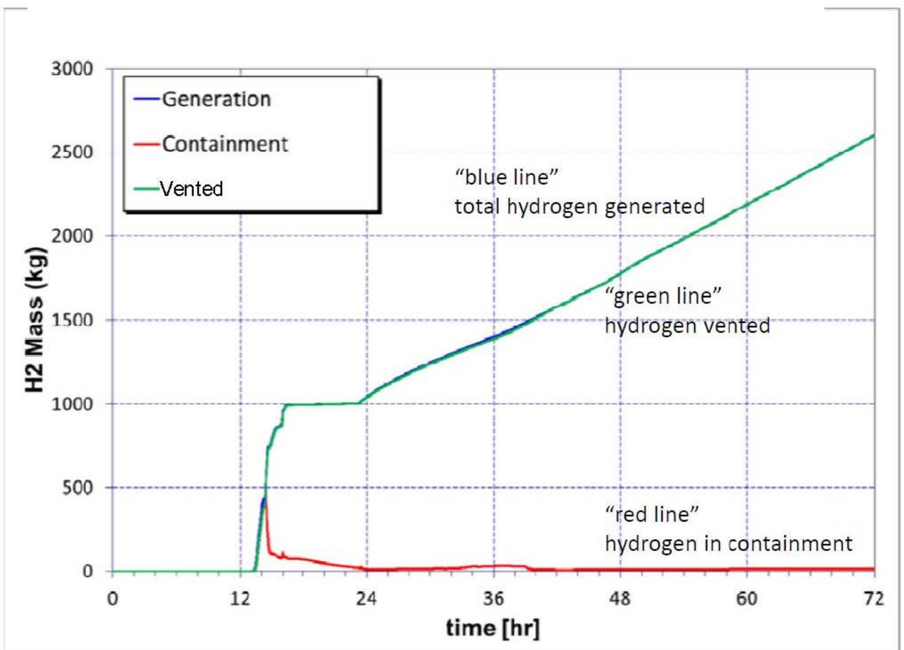


Figure 2

The technical analyses for Order EA-13-109 and the CPRR draft regulatory basis show that the threat of explosions from combustible gases is significantly reduced by effective venting strategies and the SAWA/SAWM approaches being taken as part of implementing the Order. SAMGs, which are maintained by licensees and are being updated after the Fukushima accident, also include specific measures to monitor and vent Mark I and II containments to address hydrogen issues. The enhancements provide some further risk reductions by improving the control of hydrogen in Mark I and II containments, even though more specific regulatory actions would likely not be justified given the large margins between plant risks and the NRC's safety goals (as shown in Figure 1). The improvements to capabilities and guidelines for venting the containment ensures that the hydrogen is discharged to the environment and prevents the migration of hydrogen to the reactor building, as occurred at Fukushima Dai-ichi. Further evaluations to identify other possible improvements for hydrogen control in primary containments or other buildings are unlikely to justify the imposition of additional regulatory requirements. As such, the staff's initial assessment was that Recommendation 6 can be closed for Mark I and II containments.

In its letter to the Commission on SECY-15-0137 dated November 16, 2015, the ACRS stated:

> The staff has concluded that issues of hydrogen mitigation in the reactor buildings of Mark I and Mark II containment BWRs do not merit further consideration because reliable vents will prevent over-pressurization of the containments and massive leakage of hydrogen into the reactor buildings. The conclusion neglects the potential for other pathways of hydrogen release to the reactor building under severe reactor accident conditions. It was, for example, speculated in the immediate aftermath of combustion events during the Fukushima accidents that hydrogen could be leaked to the reactor buildings through failed bellows on the containments, or through thermally or radiolytically degraded seals. Either pathway might be sufficient to release enough hydrogen to pose a combustion hazard while keeping containment pressures below levels mandating vent activation. It may, then, be more prudent for the staff to perform a comprehensive examination of potential hydrogen release pathways before they forego consideration of hydrogen mitigation in the reactor buildings.
>
> We look forward to interacting with the staff as they provide additional evaluation and supporting documentation for their conclusions in early 2016. In that work, the staff should further document the findings derived from their review of international activities and how they have affected their conclusions. We also expect that the staff will maintain their research programs and monitor international research and regulatory programs on Fukushima, and will continue to assess implications for NRC regulation and oversight.

The staff interacted with the industry during the development of guidance for licensees to meet the requirements of Order EA-13-109. An important part of the guidance was the selection of design temperatures for the containment venting systems and the ability of water addition strategies to keep structure temperatures from significantly exceeding those design values. The acceptance of 545 degrees Fahrenheit (F) as the design temperature for the containment venting systems included detailed discussions of predicted temperatures for various scenarios and the ability of water addition systems to significantly lower the temperatures in the drywell. The design values for the venting system and related analyses also demonstrated that overall containment integrity could be maintained because the pressures and temperatures would not be expected to result in the lifting of the drywell head or failure of containment penetrations. The following figure from Revision 1 to NEI 13-02 shows the relationship between the venting system design values and containment capabilities:

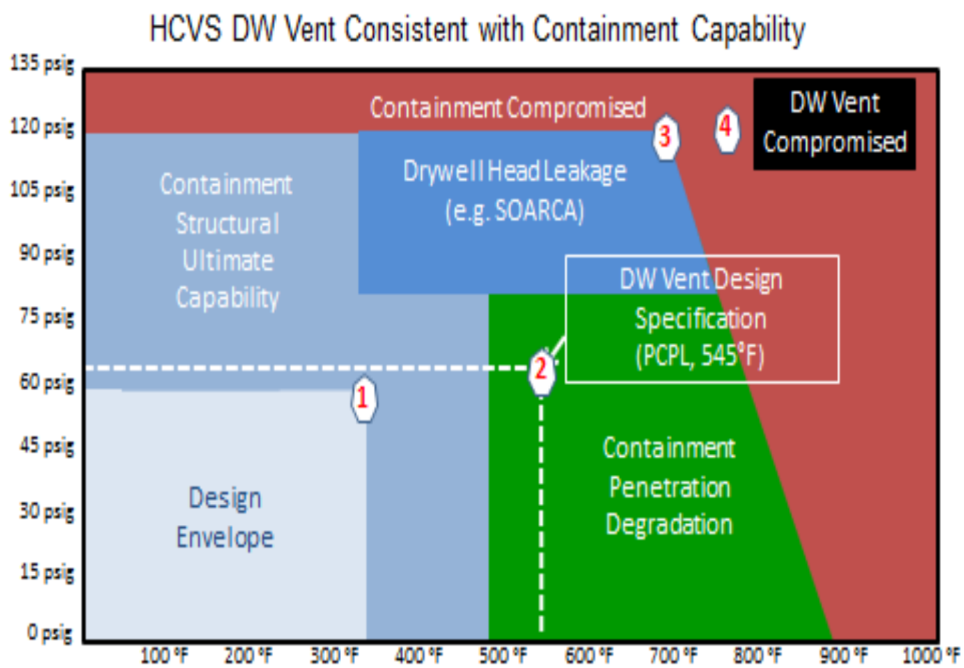


Figure 3

The green area (labeled "Containment Penetration Degradation") in Figure 3 relates to the combinations of pressures and temperatures expected to cause degradation and failure of containment penetrations. The performance of containment structures and penetrations during severe accident conditions was the subject of a series of tests performed at Sandia National Laboratory and other testing facilities in the 1980s. The test results are documented in reports such as NUREG/CR-4944, "Containment Penetration Elastomer Seal Leak Rate Tests," and NUREG/CR-5096, "Evaluation of Seals for Mechanical Penetrations of Containment Buildings," which provide the basis for the containment penetration degradation area in the above plot.[2] The programs included a series of tests of scaled containments (with and without penetrations), sections of containments, and specific tests for a variety of penetration types (e.g., electrical, mechanical, and airlocks) conducted to assess containment integrity under severe accident conditions.

The combinations of accident scenarios and containment conditions, including pressure and temperature, were evaluated not only in terms of overpressure protection, but also in regard to maintaining overall containment integrity. The assessments by the industry and the staff show that overall containment integrity is expected to be maintained for the calculated containment conditions assuming successful water addition. The industry results are documented in EPRI report "Investigation of Strategies for Mitigating Radiological Releases in Severe Accidents, BWR Mark I and Mark II Studies," while the staff's analytical results are provided in the draft regulatory basis document enclosed in SECY-15-0085. The figure below from the draft

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2 Chapter 5, "References," in NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories," provides a useful bibliography of reports and other references for containment-related testing programs, including the performance of penetrations under severe accident conditions.

regulatory basis document, revised to include the EPRI data, shows that the reduction in temperatures is a significant benefit from water addition to containment during severe accident conditions. The associated protection of the containment from not only overpressure and liner melt through mechanisms, but also prevention of over-temperature failures, is one of the primary benefits of water addition discussed in the CPRR draft regulatory basis document. The above discussions related to the hydrogen being discharged from the containment vent are valid assuming water addition capabilities are successful and overall containment integrity is maintained.

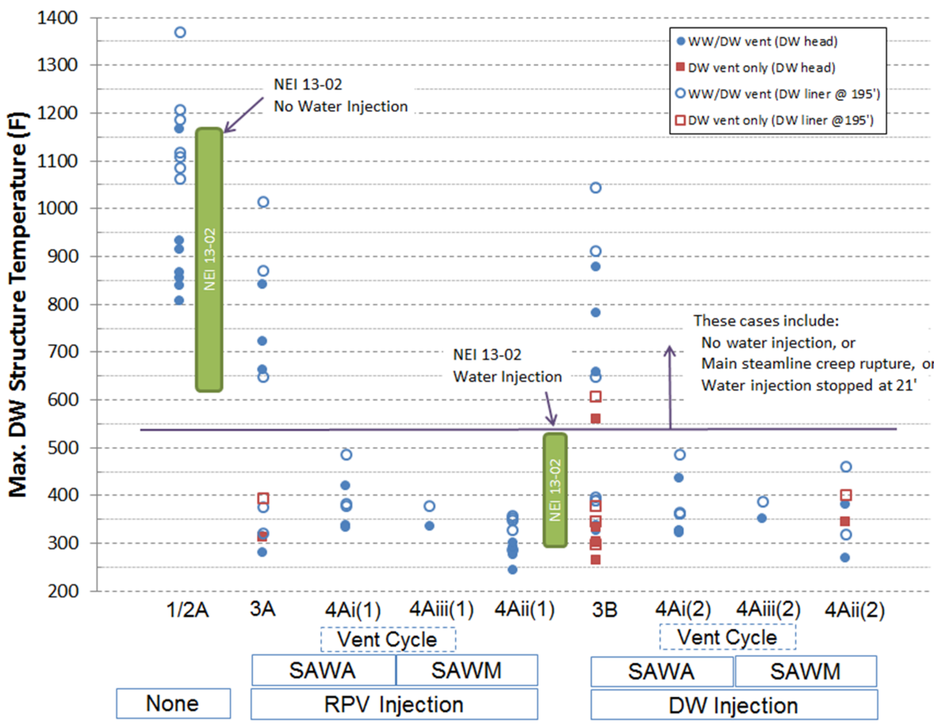


Figure 4

The staff considered the above information as part of the basis for concluding that issues related to hydrogen migration from the primary containment to the reactor building do not warrant further regulatory actions. The improved containment venting systems and additional capabilities for water addition during severe accident conditions decrease the chances of losing containment integrity from the dominant failure mechanisms (over-pressure, over-temperature, liner melt through). The analyses performed to support the guidance for EA-13-109 and the draft regulatory basis for the CPRR rulemaking confirmed that the successful deployment of equipment to cool core debris maintains overall containment integrity, including limiting the degradation of and leakage through containment penetrations. The Commission decided not to pursue the CPRR rulemaking for Mark I and Mark II containments.

The staff is aware that plant modifications are being pursued in some countries to provide additional capabilities for hydrogen control and mitigation within containments and adjacent buildings. Examples include the installation of passive autocatalytic recombiners and venting capabilities to release hydrogen from BWR reactor buildings. Such measures could be helpful in accident scenarios that involve failure of the additional capabilities required by Order EA-13-109. The possible measures for addressing hydrogen do not themselves directly support the cooling of core debris, but could help for some selected scenarios to maintain barriers to the release of radioactive material and prevent explosions that could hamper severe accident management activities. The potential benefits of these measures would be comparable to or less than the alternatives shown in Figure 1, which were determined to be below the threshold for warranting further regulatory actions.[3] Therefore, the staff is closing Recommendation 6 for Mark I and Mark II containment designs.

## BWR Mark III

### Containment Performance

There are four operating BWRs with Mark III containments located at four sites in the U.S. The Mark III containment is approximately five times the volume of the Mark I containment and 65 to 85 percent of the volume of a large dry pressurized water reactor (PWR) containment. The containment design pressure of a Mark III containment is 15 pounds per square inch gauge (psig) (25 percent of a Mark I and 30 percent of a large dry containment). Unlike Mark I and II containments, the Mark III containment is not inerted, but instead has igniters for hydrogen control. The NRC evaluated a Mark III containment (Grand Gulf) as part of the activities associated with NUREG-1150. Supporting evaluations of containment issues for Mark III containments are described in NUREG/CR-5529, "An Assessment of BWR Mark III Containment Challenges, Failure Modes, and Potential Improvements in Performance," published in January 1991. The modern BWR design incorporating the Mark III containment includes a diversity of ways to provide water to the core; therefore, reactors with this type of containment have a relatively low estimated core damage frequency related to plant transients and malfunctions (on the order of $10^{-6}$/year). The pre-Fukushima evaluations of core damage and containment performance for licensed Mark III plants did not identify generic improvements that warranted regulatory actions (see NUREG-0933).

Mark III containments are pressure suppression containments and have system interactions between the core cooling and containment functions, similar to plants with Mark I and II containments. These interactions are especially important during an extended loss of electrical power when cooling systems used for design-basis accidents are not available. Order EA-12-049 requires all operating plants to develop mitigating strategies for events involving extended losses of electrical power and loss of normal access to the plant's ultimate heat sink. The mitigating strategies include three phases: (1) an initial phase which must be survived with installed equipment such as steam-driven pumps; (2) a transition phase which uses portable, onsite equipment; and (3) a final phase which may credit offsite resources.

Suppression pool cooling is an important safety function within the mitigating strategies for the plants with Mark III containments. Venting is not a primary method for suppression pool cooling for three of the Mark III plants, while one does include venting from the suppression pool as part of its mitigating strategies. Instead, licensees for plants with Mark III containments have

[3] The discussions in this enclosure and the referenced NEA activities close WITS Item 201200144, "Staff Requirements – Briefing of the Status of Lessons Learned from the Fukushima Dai-ichi Accident, 9:00 A.M., Tuesday, August 7, 2012, Commissioners Hearing Room, One White Flint North, Rockville, Maryland (Open to Public Attendance) [ADAMS Accession No. ML122400033]."

included in their mitigating strategies additional capabilities to power suppression pool cooling equipment (e.g., through the use of portable power supplies). The NRC staff has reviewed these approaches and issued interim staff evaluations documenting that the licensees for Mark III plants have developed an acceptable approach for addressing core cooling functions and containment pressure control, including the need to remove heat from the suppression pool. These requirements address the functions deemed necessary to provide reasonable assurance of adequate protection of public health and safety in Order EA-13-109 issued to plants with Mark I and II containments.[4] The estimated low frequency for extended losses of electrical power makes it unlikely that further evaluations of means to cool or vent suppression pools in Mark III containments beyond those required under Order EA-12-049 would identify a cost-justified substantial safety improvement.

The activities supporting implementation of Order EA-13-109 for Mark I and II containments highlighted the need to take a holistic approach to considering improvements to containment performance during potential severe accidents. An important insight from the CPRR activities is that potential safety benefits from improvements to address some failure mechanisms or reduce releases by adding engineered filters can be limited by other potential failure mechanisms and accident sequences. For example, the benefits from engineered filters are limited by factors such as the possible failure of equipment leading to releases that would not be scrubbed effectively by the filters. Similar relationships and limitations would likely apply to Mark III reactor designs and thereby limit the potential effectiveness of specific items related to improving containment performance during severe accidents. NUREG/CR-5529 described evaluations of potential severe accident improvements for Mark III containments and informed the NRC's decision that no regulatory actions were warranted, except additional consideration for improving the control of hydrogen (see next section). The closure of the CPIP for Mark III containments and subsequent assessments reflect that the low frequency of severe accidents and expected protective actions ensure significant margins between the risks to public health associated with plant operations and the NRC's safety goals. Insights from the Fukushima accident do not undermine the findings from these previous evaluations and actions taken following the event (e.g., implementation of mitigating strategies) provide additional margins between estimated plant risks and the QHOs.

In summary, the NRC staff concludes that additional detailed study of possible improvements to the performance of Mark III containments during the mitigation of events or during severe accident conditions would be unlikely to identify regulatory actions that would provide a substantial safety improvement. Therefore, the staff's initial assessment was that Recommendation 5.2 can be closed for Mark III containments with no additional requirements beyond those imposed by Order EA-12-049.

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[4] Order EA-13-109 maintained from Orders EA-12-049 and EA-12-050 requirements for addressing core cooling and other safety functions to prevent core damage. The Commission determined that the requirements in Orders EA-12-049 and EA-12-050 were needed to provide reasonable assurance of adequate protection of the health and safety of the public. Severe-accident capabilities required by Order EA-13-109 were deemed cost-justified safety enhancements. U.S. licensees with Mark III containments elected to maintain key safety functions for preventing core damage by improving capabilities for suppression pool cooling versus improving containment venting and makeup capabilities. The NRC staff evaluated potential regulatory actions beyond Order EA-12-049 or related to severe accident conditions using the guidance for analyzing cost-justified safety enhancements.

The NRC staff performed some limited additional analyses of the expected performance of Mark III containments during long-term station blackout (SBO) conditions. The results from MELCOR computer simulations are shown below for cases where reactor core isolation cooling (RCIC) is assumed available until suppression pool temperature reaches 230 degrees F at about 6.5 hours. This case models plant behavior assuming a licensee was unsuccessful at establishing the suppression pool cooling function included in mitigating strategies. The loss of core cooling in this scenario is followed by reactor vessel lower head failure at 18 hours. Regarding containment performance in this simulation, the containment fails by over-pressure soon after lower head rupture if hydrogen igniters are not credited. With credit for igniters (as discussed in the following section on hydrogen control), containment failure by over-pressure is significantly delayed. The actions taken by licensees with Mark III containments as part of compliance with Order EA-12-049 would extend RCIC operation by cooling water in the suppression pool and thereby prevent or further delay breaches of fission product barriers, and would also provide for backup power supplies to hydrogen igniters. An assessment of estimated event frequencies, plant response, the timing of barrier failures, radioactive releases, and other factors show margins to the QHOs would not be substantially changed by additional containment improvements such as improved containment vents (similar to Figure 1 for Mark I and Mark II containments). These findings are consistent with previous generic evaluations, such as NUREG/CR-5529 and plant-specific assessments performed under the IPE program. Additional capabilities for containment venting or other measures to address severe accidents for Mark III containments would not provide a substantial safety enhancement and therefore additional regulatory actions are not warranted.

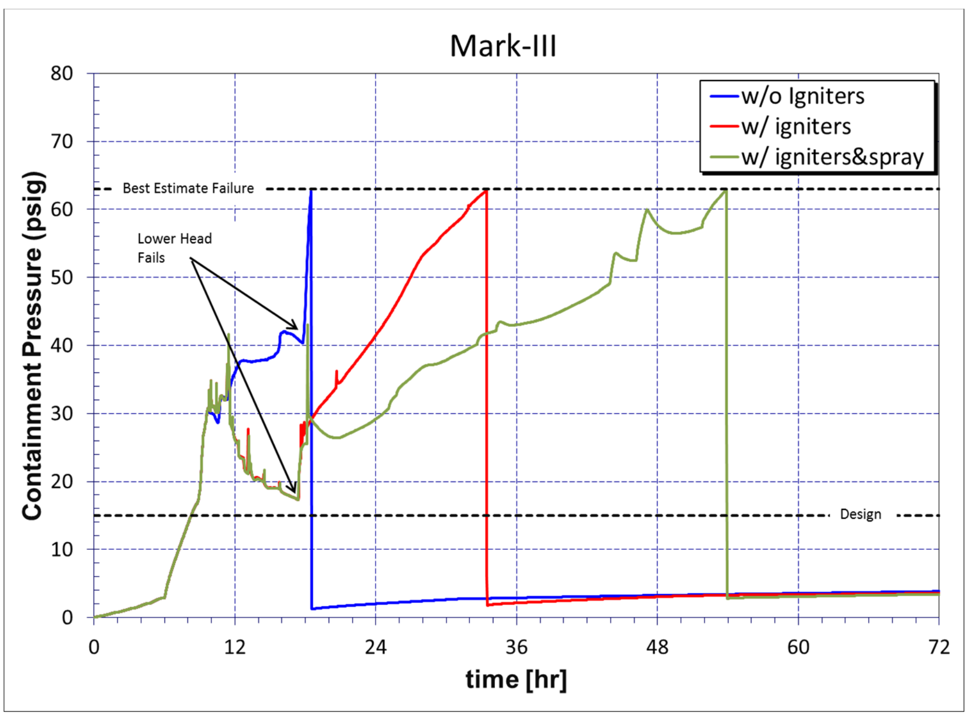


Figure 5

## Hydrogen Control

NUREG-1150 and other studies identified hydrogen issues as a potential concern for Mark III containments. The evaluations documented in NUREG/CR-6427, "Assessment of the DCH [direct containment heating] Issue for Plants with Ice Condenser Containments," issued April 2000, led to GSI-189, "Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion during a Severe Accident," updated February 4, 2009, and assessments of potential safety enhancements related to the reliability of igniter systems. To deal with large quantities of hydrogen, ice condenser and Mark III containments are equipped with alternating current (ac) powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited "burns" of hydrogen. In essence, the igniters prevent the hydrogen (or any other combustible gas) from accumulating in large quantities and then suddenly burning (or detonating) all at once, which could pose a threat to containment integrity. For most accident sequences, the hydrogen igniters can address the potential threat from combustible gas buildup. The situation of interest for GSI-189 related primarily to accident sequences associated with SBOs, where the igniter systems are not available because they are ac-powered. Thus, the concern does not affect the frequency of severe accidents, but does affect the likelihood of a significant release of radioactive material to the environment should such an accident occur.

Because this issue was not incorporated into the original scope of security-related modifications implemented following the September 11, 2001, terrorist attacks, the staff held meetings with licensees to further explore the proper consideration of security insights in providing backup power to the igniter systems. The staff reviewed industry proposals and concluded that the proposed modifications would resolve GSI-189 and provide benefit for some security scenarios. On April 23, 2007, the NRC's Executive Director for Operations issued a memorandum informing the Commission of the staff's intent to accept the commitments associated with providing backup power to hydrogen igniters and perform verification inspections at the affected sites. On June 15, 2007, the NRC staff issued letters to affected licensees accepting these commitments. The regulatory commitments related to backup power to the igniter systems received additional attention during the development of guidance for Order EA-12-049. The guidance documents for compliance with Order EA-12-049 identify the backup power supplies to the igniter systems for ice condenser and Mark III containments to be part of the containment protection features within the scope of the Order. By improving the reliability of igniter systems during SBO scenarios, the actions taken provide confidence that combustible gases will not cause a loss of primary containment integrity and reduce the chances that they will migrate to other structures, as occurred during the Fukushima accident.

Based on the assessments discussed above, the staff's initial evaluation concludes that the additional study of possible improvements to hydrogen control for Mark III containments or other buildings is unlikely to identify regulatory actions that would provide a substantial safety improvement.

The additional analyses performed by the NRC confirm that hydrogen accumulation and potential combustion could challenge the integrity of Mark III containments and shows the benefit of igniters to address this concern. Licensees took action to ensure power is available to the igniter systems during SBO conditions to help resolve GSI-189 and subsequently as part of compliance with Order EA-12-049. As shown in Figure 5, the igniters can extend the integrity of the containment, allowing licensees additional time to use the capabilities required by Order

EA-12-049 and included in their severe accident management strategies. The NRC is incorporating the requirements of Order EA-12-049 into its regulations through the rulemaking for mitigating beyond-design-basis events and revising the reactor oversight process to address licensees' implementation and maintenance of SAMGs.

The staff's analyses included containment conditions potentially affecting overall containment integrity and the potential migration of hydrogen to other structures through penetrations or other pathways. While the local gas temperatures in the upper containment can be high because of hydrogen combustion, the maximum structure temperature does not exceed 300 degrees F. The calculated temperatures are shown in Figure 6. Containment structures and penetrations are expected to maintain their integrity at these temperatures and pressures. Maintaining containment integrity will limit the migration of hydrogen from the containment to other buildings and thereby makes a requirement for monitoring or mitigating hydrogen outside of the containment unnecessary.

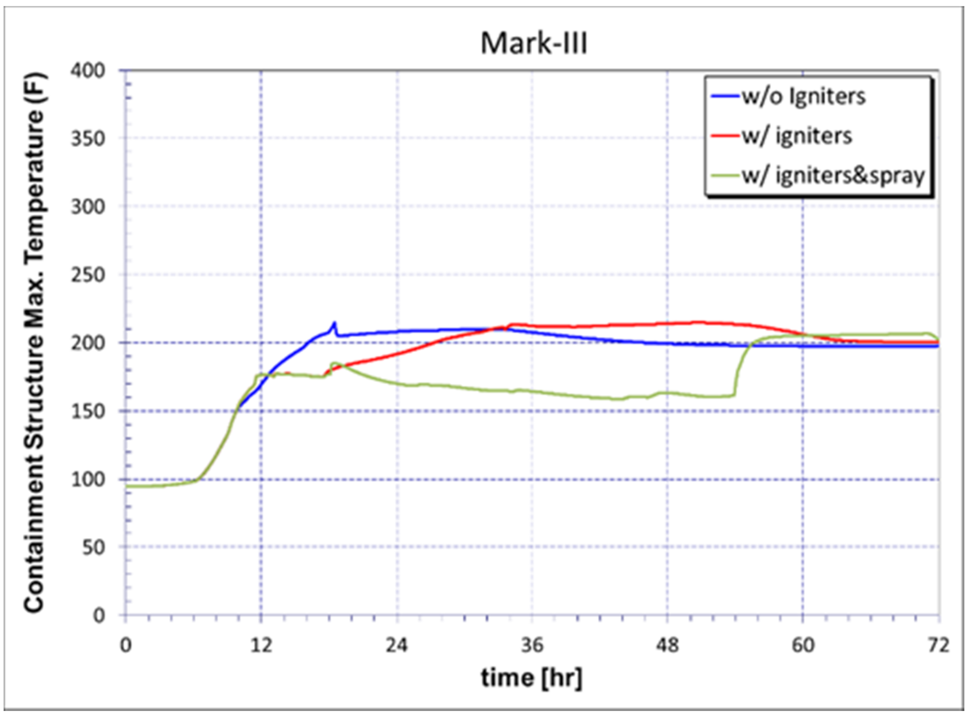


Figure 6

An assessment of estimated event frequencies, plant response, the timing of barrier failures, conditional release fractions, and other factors show margins to the QHOs would not be substantially changed by additional improvements for controlling hydrogen within containments or other structures (similar to Figure 1 for Mark I and Mark II containments). Regulatory actions to require additional capabilities for containment venting, hydrogen control, or other measures beyond the hydrogen igniters and the associated backup power supplies to address severe accidents in plants with Mark III containments are not justified.

**PWR Ice Condenser**

Containment Performance

There are nine operating PWRs with ice condenser containments located at five sites in the U.S. and an additional unit (Watts Bar, Unit 2) is expected to enter commercial operation in the future. The volumes and design pressures for ice condenser containments are similar to the Mark III BWR containments. Ice condenser containments also have igniters for hydrogen control. The NRC evaluated an ice condenser containment (Sequoyah) as part of the activities associated with NUREG-1150. Supporting evaluations of containment issues for ice condenser containments are described in various studies, including NUREG/CR-6427. The pre-Fukushima evaluations of core damage and containment performance for licensed ice condenser plants did not identify generic improvements that warranted regulatory actions (see NUREG-0933).

Ice condenser containments are pressure suppression containments, but like other PWR designs, they do not have direct system interactions between core cooling functions and containment functions, as discussed above for BWRs[5]. As discussed above, Order EA-12-049 requires all operating plants to develop a three phase approach for mitigating events involving extended losses of electrical power and loss of normal access to the ultimate heat sink. Venting is not a primary method for protecting ice condenser containments as part of compliance with Order EA-12-049. Rather, these plants use the presence of the ice and containment sprays to maintain containment pressure and temperature within limits during an extended loss of electrical power. The NRC staff has reviewed these approaches and issued interim staff evaluations documenting that the licensees with ice condenser plants have developed an acceptable approach for addressing core cooling and containment functions. These requirements address the functions deemed necessary to ensure reasonable assurance of adequate protection of public health and safety. The estimated low frequency for extended losses of electrical power and expected plant response to loss of power scenarios, including the implementation of mitigating strategies, makes it unlikely that further evaluations of the means to protect ice condenser containments, beyond those developed for compliance with Order EA-12-049, would identify a cost-justified substantial safety improvement. Therefore, the staff's initial assessment is that Recommendation 5.2 can be closed for ice condenser containments.

Studies documented in reports such as NUREG-1150 and NUREG/CR-6427 evaluated potential severe accident scenarios for ice condenser containments and contributed to the NRC's decision that no regulatory actions were warranted, except additional consideration of improving the control of hydrogen (see next section). The NUREG-1150 accident progression analysis models were used by the staff and its contractors in the evaluation of possible containment improvements for the PWR ice condenser and BWR Mark III designs. The result of the staff reviews of these designs (and all others except the Mark I containments) was that potential improvements would best be pursued as part of the IPE process. The most significant finding discussed in NUREG/CR-6427 was that the early containment failure probability is dominated by hydrogen combustion events not associated with direct containment heating (DCH), which only occur during SBOs, rather than by DCH events. This is because the SBO probability is

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5 Some licensees for plants with various containment designs credit containment accident pressure to ensure sufficient net positive suction head for emergency core cooling pumps under some design-basis accident scenarios. These plants, therefore, have some less direct dependencies between containment and core-cooling functions than was described for the BWRs.

small, the high-pressure melt ejection probability is small, and because containment loads are non-threatening for any reasonable plant damage state associated with a non-SBO event. Insights from the Fukushima accident do not undermine the findings from these previous evaluations. The NRC staff is evaluating an ice condenser plant (Sequoyah) as part of a continuation of the SOARCA project as directed by the Commission in the SRM to SECY-12-0092, “State-of-the-Art Reactor Consequence Analyses - Recommendation for Limited Additional Analysis” (ADAMS Accession No. ML12341A349).

The NRC staff performed additional analyses related to ice condenser containments as part of continuing efforts in the SOARCA project. The staff expects to issue the detailed report for the SOARCA assessment of an ice condenser containment in late 2016. Preliminary results from the study are presented here to confirm the insights from previous studies and the related staff conclusion that additional regulatory actions are not warranted. The results shown in Figure 7 are for a case where turbine-driven auxiliary feedwater is assumed available until station batteries are depleted at 8 hours into the SBO event. Core damage occurs in this scenario shortly before 24 hours, followed by a rupture of the reactor coolant system hot leg piping and failure of the reactor vessel lower head after 24 hours. Without the igniters discussed in the following section, the containment could fail in this scenario soon after hot leg rupture because of an explosion of the hydrogen released into containment. Successful operation of the igniters can control the combustible gases and limit the containment pressure so that a containment failure is not likely within 72 hours.

The actions taken by licensees with ice condenser containments to comply with Order EA-12-049 include the use of installed equipment, such as auxiliary feedwater systems for core cooling and spray systems or cooling units for containment temperature and pressure control. Successful deployment of these capabilities would extend cooling functions and prevent or delay breaches of fission product barriers. An assessment of estimated event frequencies, plant response, the timing of barrier failures, conditional release fractions, and other factors show margins to the QHOs would not be substantially changed by additional containment improvements (similar to Figure 1 for Mark I and Mark II containments). These findings are consistent with previous generic evaluations, such as NUREG/CR-6427 and plant-specific assessments performed under the IPE program. Regulatory actions to require other capabilities for containment venting or other measures to address severe accidents in plants with ice condenser containments are not justified.

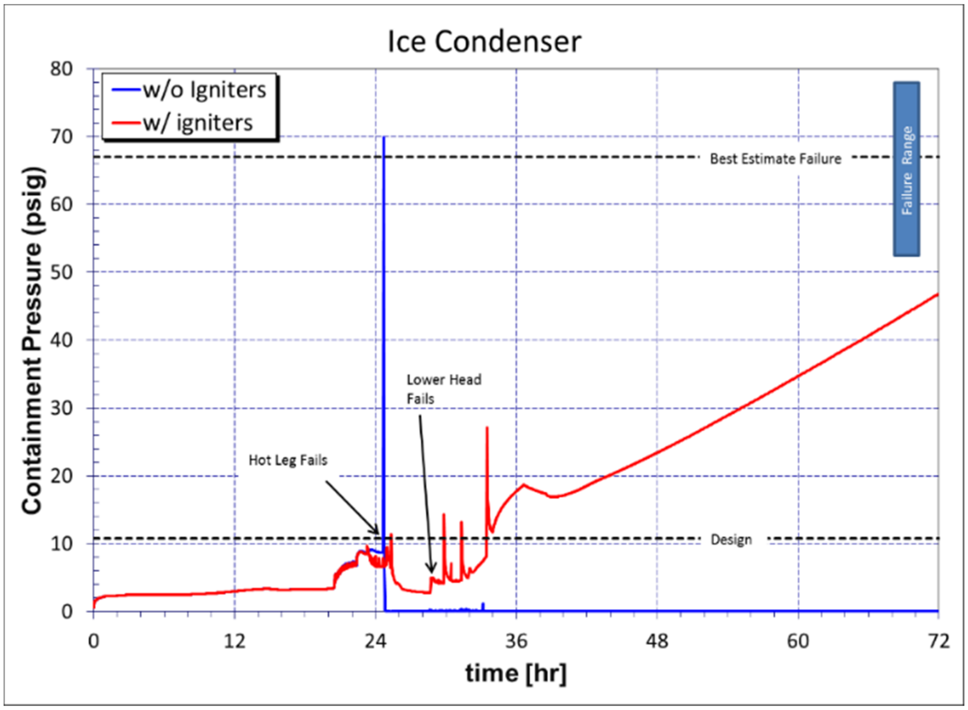


Figure 7

Hydrogen Control

The evaluation of hydrogen control for ice condenser containments follows the above discussion for Mark III containments. By improving the reliability of igniter systems during SBO scenarios, the actions taken provide confidence that combustible gases will not cause a loss of primary containment integrity and will reduce the chances that they will migrate to other structures, as occurred during the Fukushima accident. Based on the assessments discussed above, the staff believes that additional study of possible improvements to hydrogen control for ice condenser containments or other buildings is unlikely to identify regulatory actions beyond those already taken that would provide a substantial safety improvement.

The NRC's ongoing SOARCA study confirms that hydrogen combustion can challenge the integrity of ice condenser containments and shows the benefit of igniters to address this concern. Licensees took action to ensure power is available to the igniter systems during SBO conditions to help resolve GSI-189 and subsequently as part of compliance with Order EA-12-049. As shown in Figure 7, the igniters can extend the integrity of the containment, allowing licensees additional time to use the capabilities required by Order EA-12-049 and included in their severe accident management strategies. The NRC is incorporating the requirements of Order EA-12-049 into its regulations through the rulemaking for mitigating beyond-design-basis events and revising the reactor oversight process to address licensees' implementation and maintenance of SAMGs.

The staff's analyses included containment conditions potentially affecting overall containment integrity and the potential migration of hydrogen to other structures through penetrations or other pathways. Also, while the local gas temperatures in the containment can be high because of hydrogen combustion, the maximum structure temperature does not exceed 300 degrees F. The calculated temperatures are shown in Figure 8. The integrity of containment penetrations and prevention of hydrogen migration outside of containment should be maintained at these temperatures. Limiting the migration of hydrogen to other structures makes a requirement for monitoring or mitigating hydrogen outside of the primary containment unnecessary. The containment pressure remains at or below the design pressure for many hours after breaching the reactor vessel lower head and a containment failure is not likely within 3 days.

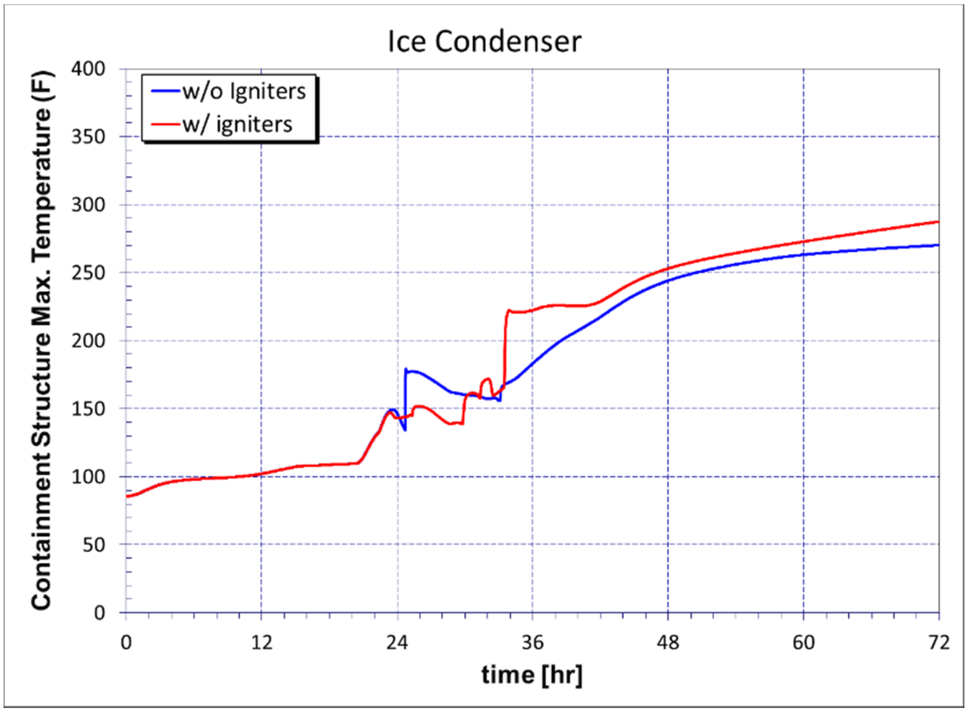


Figure 8

An assessment of estimated event frequencies, plant response, the timing of barrier failures, conditional release fractions, and other factors show margins to the QHOs would not be substantially changed by measures beyond required igniters and associated backup power supplies to control hydrogen within containment structures or other buildings (similar to Figure 1 for Mark I and Mark II containments). Regulatory actions to require additional capabilities for hydrogen control or other measures to address severe accidents are not justified.

**PWR Large Dry**

Containment Performance

There are 56 operating PWRs with large dry containments located at 33 sites in the U.S. Four PWRs with AP1000 designs are under construction and are discussed in a following section for reactors licensed under the provisions of 10 CFR Part 52, "Licenses, Certification, and Approvals for Nuclear Power Plants." For the sake of this discussion, large dry containments also include those maintained at sub-atmospheric conditions during normal operations. A large dry containment is designed to contain the blowdown mass and energy from a large break loss of coolant accident, assuming any single active failure in the containment heat removal systems. These systems may include containment sprays and/or fan coolers, depending on the particular design. Large dry containments can be of either concrete or steel construction. All U.S. concrete containments have steel liners to assure leak tightness. Large dry (and all other) containments have a large, thick basemat that provides seismic capability, supports the structures, and may serve to contain molten material during a severe accident. PWR designs with large dry containments do not have direct system interactions between core cooling functions and containment functions as discussed for BWRs (see footnote 5).

- As discussed above, Order EA-12-049 requires all operating plants to develop a three phase approach for mitigating events involving an extended loss of electrical power and loss of normal access to the ultimate heat sink. Venting is not a primary method for protecting large dry containments as part of compliance with this order. Instead, plants use containment sprays or restore containment cooling functions to maintain containment pressure and temperature within limits during an extended loss of electrical power. The NRC staff has reviewed licensees' approaches for compliance with this Order and issued interim staff evaluations documenting that the licensees for plants with large dry containments have developed (or will develop) acceptable approaches for addressing core cooling and containment functions. These requirements address the functions deemed necessary to ensure reasonable assurance of adequate protection of public health and safety. The estimated low frequency for extended losses of electrical power and the expected plant response to the loss of power scenarios, including the implementation of mitigating strategies, make it unlikely that further evaluations of the means to protect large dry containments beyond the overall integrated plans developed for Order EA-12-049 would identify a cost-justified substantial safety improvement. Therefore, the staff's initial assessment was that Recommendation 5.2 can be closed for large dry containments.

Large dry containments have been evaluated in terms of severe accident behavior in several major NRC studies, including NUREG-1150 and most recently in NUREG-1935 (SOARCA). Results from the SOARCA study for the PWR large dry containment pilot plant, Surry, are provided on the right side of Figure 9 in terms of individual latent cancer fatality risk. Similar to the evaluations presented in the CPRR regulatory basis document, the figure shows a significant (orders of magnitude) margin between the risks associated with an extended loss of ac power event and the QHOs defined by the NRC's Safety Goal Policy Statement. A preliminary assessment performed by the NRC staff determined that site-to-site variations related to reevaluated external hazards would not challenge the conclusion that a generic requirement is not warranted for severe accident measures beyond those already in place for large dry containments. Insights from the Fukushima accident do not undermine the findings from these previous evaluations.  As directed by the Commission in the SRM to SECY-12-0092, an uncertainty analysis of the SOARCA Surry unmitigated short-term SBO (STSBO) scenario is underway which will provide additional insights.

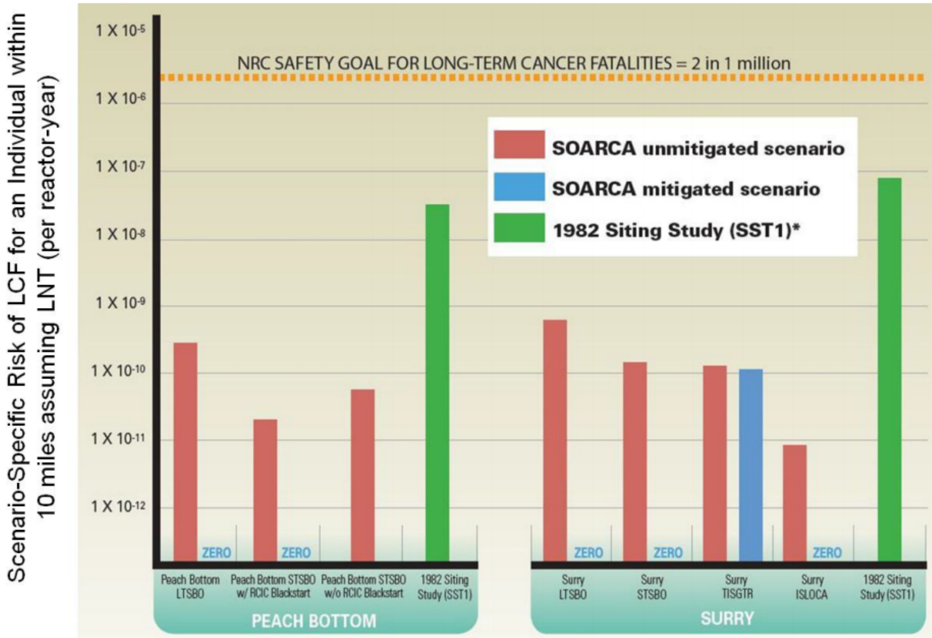


Figure 9 – Comparison of Individual Latent Cancer Fatality (LCF) Risk Results for SOARCA Scenarios to the NRC Safety Goal and to Extrapolations of the 1982 Siting Study SST1 (taken from NUREG-1935 Figure ES-3)[6]

Although Figure 9 displays similar LCF risks for the two plants modeled in the SOARCA studies, accident conditions involving a loss of heat removal functions can challenge the integrity of Mark I containments more quickly than large dry containments.  The containment pressure response for an unmitigated LTSBO scenario for Surry is shown in Figure 10.  The longer times to over-pressurize large dry containments provides additional opportunities for emergency responders to restore key safety functions prior to the containment being breached.  The low LCF risks estimated in the SOARCA study, which reflect the ability of large dry containments to limit the release of radioactive materials for many hours into the unmitigated LTSBO scenario, confirms the NRC staff's initial assessment of the adequacy of containment performance and finding that additional regulatory actions such as requiring improved containment vents are not warranted for large dry containments.

[6] LTSBO: long-term station blackout; LNT: linear no-threshold; RCIC: reactor core isolation cooling; ISLOCA: interfacing systems loss-of-coolant accident; SST: siting source term

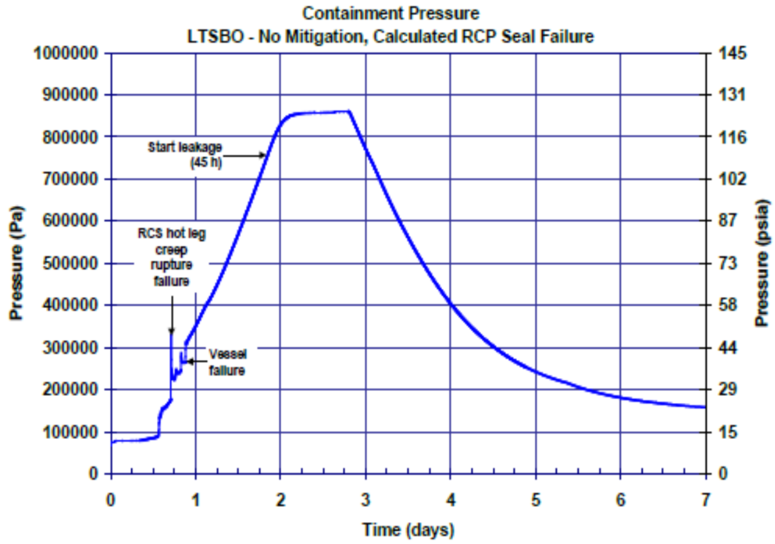


**Figure 5-5 Unmitigated long-term station blackout containment pressure history**

Figure 10 (from NUREG/CR-7110, Volume 2)

<u>Hydrogen Control</u>

The capabilities for large dry containments to withstand possible hydrogen combustion events have been addressed in several NRC risk studies and documentation related to the development of 10 CFR 50.44 (SECY-03-0127), including the associated regulatory analysis. A detailed assessment is documented in NUREG/CR-5662, "Hydrogen Combustion, Control, and Value-Impact Analysis for PWR Dry Containments," issued in June 1991. NUREG/CR-5662 discusses that additional requirements for hydrogen control, such as a requirement for hydrogen igniters, were not justified for reactors with large dry containment designs due to the large size and robust design of these containments. Based on the assessments discussed above, the staff concludes that additional study of possible improvements to hydrogen control for large dry containments or other buildings would be unlikely to identify regulatory actions meeting the threshold for a substantial safety improvement.

Additional insights related to large dry containments are provided in the Surry SOARCA analyses (NUREG/CR-7110, Volume 2). The unmitigated LTSBO scenario results in high-steam concentrations sufficient to inert the containment and suppress hydrogen combustion (see Figure 11). Although the in-vessel hydrogen production is very significant, combustible conditions did not exist in the containment through 72 hours of the simulation.

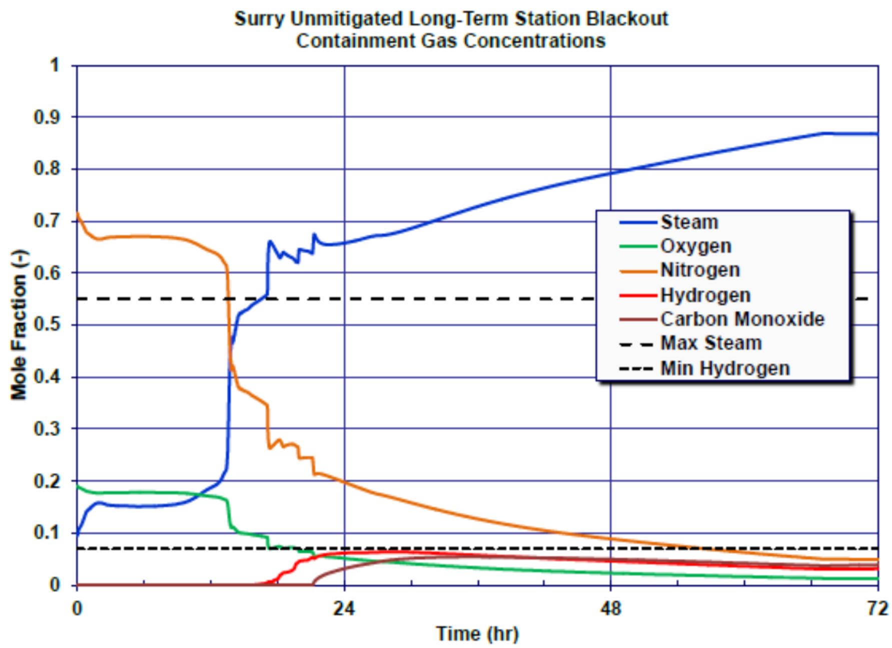


**Figure 5-6 Unmitigated long-term station blackout containment gas concentrations**

Figure 11 (from NUREG/CR-7110, Volume 2)

It is noted that the Surry SOARCA analyses do identify that hydrogen migration to adjacent buildings may occur for ISLOCA scenarios. However, the SOARCA risk estimates generally confirm previous assessments (e.g., GSI-105, "Interfacing Systems LOCA at LWRs"), which found that the estimated frequency of ISLOCAs leading to core damage is very low (see summary table provided as Figure 16). The staff's evaluation of recent studies confirms that possible improvements to hydrogen control for large dry containments or other buildings do not meet the thresholds for a substantial safety improvement.

**Additional Regulatory Evaluation**

In its SRM for SECY-15-0085, the Commission directed the NRC staff to leverage the draft regulatory basis for the CPRR rulemaking for BWRs with Mark I and Mark II containments when evaluating other containment designs to address NTTF Recommendation 5.2. The NRC staff did not perform detailed assessments of the potential benefits and costs of possible regulatory actions to improve the performance of containments other than the Mark I and Mark II or to improve the control of hydrogen during severe accident conditions. Instead, the staff reviewed previous assessments for each containment, performed limited analyses for selected scenarios, and compared the available information to the results and decision criteria discussed in the draft regulatory basis for the CPRR activity.

The first step taken for the regulatory evaluation within the CPRR draft regulatory basis document was to follow the safety goal screening process described in NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines for the U.S. Nuclear Regulatory Commission" (ADAMS Accession No. ML042820192). This approach is similar to the screening process documented in the analysis described in COMSECY-13-0030, "Staff Evaluation and

Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 25, 2013 (ADAMS Accession No. ML13329A918). The Commission decisions associated with both evaluations affirmed the staff's approach, including the use of the safety goal screening process. The NRC staff's regulatory evaluation for the CPRR draft regulatory basis included developing a high-level conservative estimate of individual latent cancer fatality risks introduced by an extended loss of alternating current power (ELAP) at a BWR with Mark I or Mark II containment and the potential benefits of various containment modifications in reducing those risks. The staff summarized the development of the high-level conservative estimate in the following figure from the CPRR draft regulatory basis:

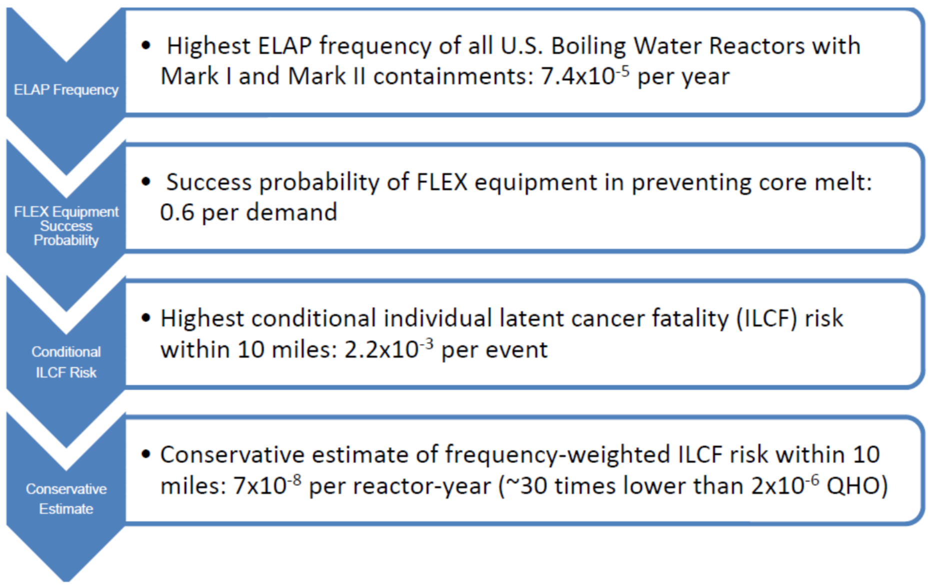


Figure 12 (from Enclosure to SECY-15-0085)

The conservative estimate of frequency-weighted individual LCF risk for ELAP events at BWRs with Mark I and Mark II containments was estimated to be a small fraction of the related safety goal QHO. The staff considered existing Commission policy as defined by the Severe Accident Policy Statement and more recent Commission decisions such as the SRM to SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated March 20, 2013 (ADAMS Accession No. ML13079A055), and the SRM to SECY-14-0087, "Qualitative Consideration of Factors in the Development of Regulatory Analyses and Backfit Analyses," dated March 4, 2015 (ADAMS Accession No. ML15063A568). Based on these considerations and the expected significant costs associated with achieving these marginal risk reductions, the staff determined that a more detailed cost-benefit assessment was not warranted. The margins to the QHO and the relative benefits of various alternatives considered within the CPRR draft regulatory basis is shown in Figure 1.

The NRC staff has not performed plant simulations or risk evaluations for each containment type to the degree performed for the regulatory evaluations prepared for Mark I and Mark II containments. As previously described, the staff reviewed various studies and performed confirmatory analyses to determine if additional studies or regulatory actions might be warranted to address Recommendations 5.2 and 6. Studies and safety issues related to each containment type were evaluated and re-assessed in light of the Fukushima accident and other updated information.  The NRC staff reviewed available probabilistic risk assessments and supporting information to support the regulatory evaluation.  An example is provided in Figure 13, which provides an estimate of the seismic core damage frequency (SCDF) for the nuclear power plants in the U.S. based on updated seismic hazard curves.  The figure was prepared by EPRI using information from the NRC and licensees in support of the NRC's decisions related to the reevaluation of seismic hazards under Recommendation 2.1 (see letter dated March 12, 2014, ADAMS Accession No. ML14083A596).  The NRC staff used the EPRI results as part of the basis for the letter dated May 9, 2014, "Seismic Screening and Prioritization" (ADAMS Accession No. ML14111A147).  The curve is especially relevant to this assessment because seismic events are a significant contributor to the scenarios evaluated for Mark I and Mark II containments and events leading to extended losses of electrical power.

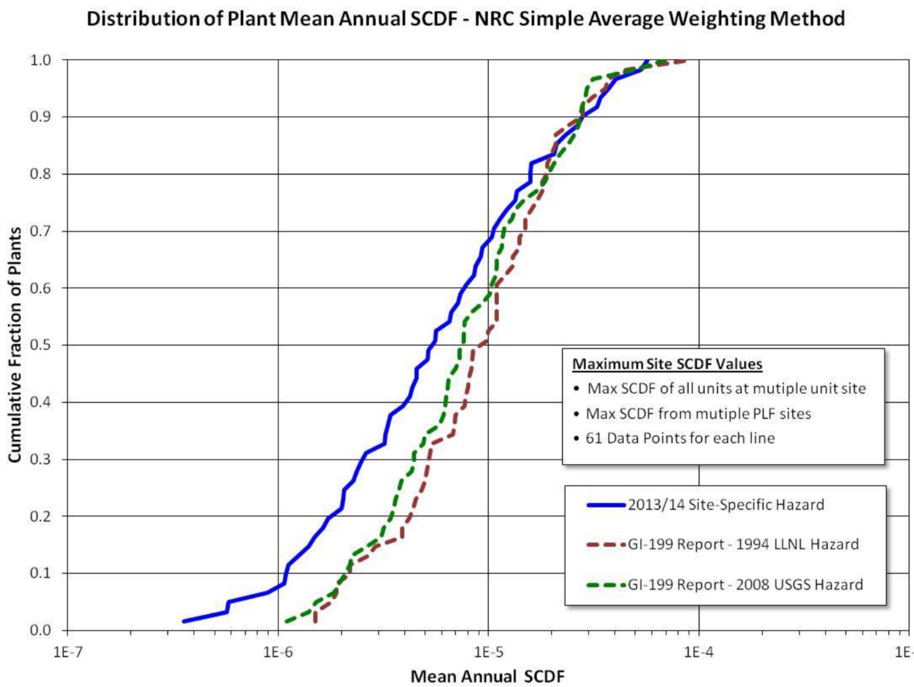


Figure 13

The fleet-wide estimates in Figure 13 for SCDF are comparable to or below the high-level conservative estimate of extended losses of electrical power used for the evaluation of Mark I and Mark II containments.  In addition, plants of all containment designs are required by Order EA-12-049 to develop and implement mitigating strategies for beyond-design-basis events.  The estimated success rate for mitigating strategies used in the high-level conservative estimates for Mark I and Mark II containments is also reasonable for other reactor designs given the similarities in system design characteristics for installed plant equipment.  Lastly, a review of the timings of containment failures, dispersion of radioactive materials from the various plant types, populations, and other factors supports that the individual latent cancer fatality risk from the regulatory evaluation of the plants with Mark I and Mark II containments is reasonable to use in an assessment of other containment designs.  Therefore, the NRC staff concludes that significant margins exist between estimated plant risks that might be influenced by improvements in containment performance or hydrogen control and the NRC established safety goals.[7] Possible plant changes related to Recommendations 5.2 and 6 would not provide a substantial safety enhancement and therefore additional regulatory actions are not warranted.

In the previously discussed evaluations of the various containment types, the staff generally presented comparisons for scenarios involving LTSBO. The LTSBO scenarios for both Peach Bottom (BWR) and Surry (PWR) were considered to be the most likely severe accident scenario for each plant considered in SOARCA. The LTSBO was analyzed assuming no mitigation (except for the initial operation of installed turbine-driven systems), and resulted in core damage beginning in 9 hours for Peach Bottom and 6 hours for Surry, and reactor vessel failure at about 20 hours.[8] Offsite radiological release because of containment failure begins at about 20 hours for Peach Bottom and at 45 hours for Surry. STSBO scenarios also include the assumed immediate loss of turbine-driven systems through loss of direct current control power or other failure and therefore the plant proceeds to core damage more rapidly. For the most rapid events (i.e., the unmitigated STSBO in which core damage may begin in 1 hour for Peach Bottom to 3 hours for Surry), reactor vessel failure begins at roughly 8 hours. If core cooling is not restored in these cases, containment failure and radiological release begins at about 8 hours for Peach Bottom and at 25 hours for Surry. A summary of the offsite consequence results for the Peach Bottom and Surry SOARCA studies (NUREG-1935) are provided below:

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[7] The safety goals and related QHOs were developed to assess aggregate risks and to be used for making decisions on rulemakings or other major agency actions. It is necessary to keep this in mind when using the QHOs to evaluate specific issues or plant specific concerns. In this case, additional regulatory requirements for containment venting capabilities at plants with other than Mark I and Mark II containments and requirements for hydrogen control and mitigation beyond those already imposed would not significantly change the margins between the overall risks from nuclear power plants and the established safety goals. The evaluated scenarios involve important contributors to overall risk and yet potential regulatory actions result in benefits (i.e., risk reductions) that are a small fraction of the safety goal values.

[8] In the SOARCA studies, the cases that credit successful implementation of equipment and procedures developed to address the loss of large areas of the plant due to explosions or fire (10 CFR 50.54(hh)) in addition to actions directed by the EOPs and SAMGs are referred to as the mitigated cases. The analyses without credit for 10 CFR 50.54(hh) equipment and procedures are referred to as the unmitigated cases (SAMGs were considered but not implemented in the unmitigated case). Mitigating strategies developed by licensees after the Fukushima accident for beyond-design-basis external events provide similar or more extensive capabilities for key safety functions as those developed for 10 CFR 50.54(hh), and might further reduce the quantified risks.

**Table ES-1 Offsite Consequence Results for Peach Bottom Scenarios Assuming Linear No-Threshold (LNT) Dose-Response Model**

| Scenario | Core damage frequency (CDF) (per reactor-year)* | Mitigated | | Unmitigated | |
|---|---|---|---|---|---|
| | | Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles | Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year) | Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles | Scenario-specific risk (CDF x Conditional) of latent cancer fatality for an individual located within 10 miles (per reactor-year) |
| Long-term SBO | $3\times10^{-6}$ | No Core Damage | | $9\times10^{-5}$ | $\sim 3\times10^{-10}$ **** |
| Short-term SBO with RCIC Blackstart** | $3\times10^{-7}$ | No Core Damage *** | | $7\times10^{-5}$ | $\sim 2\times10^{-11}$ **** |
| Short-term SBO without RCIC Blackstart | | Not Applicable *** | | $2\times10^{-4}$ | $\sim 6\times10^{-11}$ **** |

* The CDF assumes that 10 CFR 50.54(hh) equipment and procedures were not used.

** Blackstart of the reactor core isolation cooling (RCIC) system refers to starting RCIC without any ac or dc control power. Blackrun of RCIC refers to the long-term operation of RCIC without electricity, once it has been started. This typically involves using a portable generator to supply power to indications such as reactor pressure vessel (RPV) level to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. STSBO RCIC blackstart and limited blackrun is credited as an unmitigated case for SOARCA purposes because the licensee has included its use in procedures. Past NRC severe accident analyses of STSBO scenarios did not credit blackstart of RCIC. A sensitivity calculation without blackstart was therefore performed to provide a basis for comparison to past analyses.

*** A scenario with 10 CFR 50.54(hh) mitigation, but without RCIC blackstart was not analyzed.

**** Estimated risks below $1 \times 10^{-7}$ per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

Figure 14 (Table ES-1 from NUREG-1935)

**Table ES-2 Offsite Consequence Results for Surry Scenarios Assuming LNT Dose-Response Model**

| Scenario | Core damage frequency [CDF] (per reactor-year)* | Mitigated | | Unmitigated | |
|---|---|---|---|---|---|
| | | Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles | Scenario-specific risk [CDF x Conditional] of latent cancer fatality for an individual located within 10 miles (per reactor-year) | Conditional scenario-specific probability of latent cancer fatality for an individual located within 10 miles | Scenario-specific risk [CDF x Conditional] of latent cancer fatality for an individual located within 10 miles (per reactor-year) |
| Long-term SBO | $2\times10^{-5}$ | No Core Damage | | $5\times10^{-5}$ | $\sim 7\times10^{-10}$ **** |
| Short-term SBO | $2\times10^{-6}$ | No Containment Failure ** | | $9\times10^{-5}$ | $\sim 1\times10^{-10}$ **** |
| Short-term SBO with TISGTR | $4\times10^{-7}$ | $3\times10^{-4}$ *** | $\sim 1\times10^{-10}$ **** | $3\times10^{-4}$ | $\sim 1\times10^{-10}$ **** |
| Interfacing systems LOCA | $3\times10^{-8}$ | No Core Damage | | $3\times10^{-4}$ | $\sim 9\times10^{-12}$ **** |

* The CDF assumes that 10 CFR 50.54(hh) equipment and procedures were not used.

** Accident progression calculations showed that source terms in the mitigated case are smaller than in the unmitigated case. Offsite consequence calculations were not run, since the containment fails at about 66 hours. A review of available resources and emergency plans shows that adequate mitigation measures could be brought onsite within 24 hours and connected and functioning within 48 hours. Therefore 66 hours would allow ample time for mitigation through measures transported from offsite.

*** Containment failure is delayed by about 46 hours in the mitigated case relative to the unmitigated case. Rounding to one significant figure shows conditional LCF probabilities of $3\times10^{-4}$ for both mitigated and unmitigated cases, however the original values were $2.8\times10^{-4}$ for the mitigated case and $3.2\times10^{-4}$ for the unmitigated case.

**** Estimated risks below $1 \times 10^{-7}$ per reactor year should be viewed with caution because of the potential impact of events not studied in the analyses and the inherent uncertainty in very small calculated numbers.

Figure 15 (Table ES-2 from NUREG-1935)

Uncertainty analyses were subsequently completed for the Peach Bottom unmitigated LTSBO (NUREG/CR-7155) and Surry unmitigated STSBO scenarios. The uncertainty analyses showed that the consideration of uncertainties in model inputs results in variations in release start times, containment failure times, release magnitudes, and latent cancer fatality risks. However, the uncertainty analyses corroborate the conclusions from the original SOARCA studies, including essentially zero early fatality risk, and confirms the NRC staff's initial assessment that additional regulatory actions are not warranted.

The actions taken for all plant types through the issuance of Order EA-12-049 requiring enhanced mitigating strategies for extended losses of electrical power reflect the importance of maintaining or restoring core cooling and thereby preventing the release of radioactive materials. The NRC issued Order EA-13-109 for Mark I and Mark II containments to address the shorter times available before containment failure for some scenarios. The guidance provided in NEI 13-02 for complying with Order EA-13-109 includes assessments to ensure containment venting and water addition capabilities can be established under severe accident conditions associated with a STSBO. As possible accident sequences progress to include core damage, operators for all plant types would transition to SAMGs and attempt to restore key safety functions associated with core cooling and containment integrity. The staff concludes that appropriate actions have been taken to address STSBO for plants with Mark I and Mark II containments and that consideration of STSBO versus LTSBO does not alter the conclusions that additional actions are not warranted for plants with other containment designs.

The additional regulatory evaluation supports the staff's initial finding that regulatory actions in response to Recommendations 5.2 and 6 are not warranted for operating nuclear power plants. The staff bases this finding on conservative estimates of frequency-weighted risks to public health and safety in comparison to the NRC's established safety goals, insights from evaluations and agency decisions for Mark I and Mark II containments, past studies on the performance of other containment designs in terms of plant response and the timing of possible failures during severe accidents, considering both LTSBO and STSBO scenarios.

**Reactors Licensed Under 10 CFR Part 52**

For nuclear power plants licensed under 10 CFR Part 52, including the AP1000 plants currently under construction, the NRC imposes additional requirements for containments beyond those for currently operating plants. This practice is consistent with the NRC's Severe Accident Policy Statement that new nuclear power plants should incorporate improvements during design and construction that were not practical or cost-effective to require as modifications to existing plants. New reactors licensed under 10 CFR Part 52 must address similar design basis accidents as operating plants, but must also have severe accident design features to increase the ability of containments to maintain their integrity during severe accident conditions. In addition, more conservative hydrogen generation rates and related controls are imposed in 10 CFR 50.44 for plants licensed after 2003. As a result, new plants have design features such as hydrogen igniters for AP1000 design reactors and inerted containments and passive autocatalytic recombiners for Economic Simplified Boiling Water Reactors (ESBWR). The NRC staff assessed potential further enhancements and found that such measures would not likely be justified under the finality provisions established under 10 CFR Part 52 (similar to backfit requirements defined in 10 CFR 50.109, "Backfitting").

In response to questions from stakeholders related to SECY-15-0137, the staff offers the following additional discussion of the design features for severe accident conditions and hydrogen control for the certified reactor designs. More details related to these reactor designs, including severe accident design features, are available in the design certification documents associated with each reactor design (see Appendices A, D, and E to 10 CFR Part 52).

Advanced Boiling-Water Reactor (ABWR)

The ABWR containment has a specific design feature to increase the ability of its containment to maintain its integrity during severe accident conditions and maintain containment integrity during an ELAP concurrent with a loss of normal access to the ultimate heat sink. The design feature is called the containment over-pressure protection system (COPS). The COPS is a hardened vent system consisting of two overpressure relief rupture disks that relieve pressure from the top of the suppression pool air space to the atmosphere via the plant stack. Once the COPS rupture disk relieves, containment pressure and temperature decrease. Additionally, the ABWR has several other design features to mitigate severe accidents, which the staff concluded meet the applicable requirements. These design features include the following:

- ac-independent water addition system (provides RPV injection and upper drywell spray)
- lower drywell flooder (provides alternate cavity flooding with thermally activated flooder valves)
- vessel depressurization
- lower drywell design (a sacrificial, low gas content concrete)
- inerted containment (minimizes the impact from combustible gases)
- drywell-wetwell vacuum breakers (reduce loads during pool swell)

AP1000

Unlike current operating plants, the AP1000 containment has a specific design feature to increase the ability of its containment to maintain its integrity during severe accident conditions and maintain containment integrity during an ELAP and loss of access to the normal heat sink. The design feature is called the passive containment cooling system and consists of a tank of water above a steel containment shell. When activated, water from the tank pours over the steel containment shell cooling the containment. Once the system activates, containment pressure and temperature decrease. Additionally, AP1000 has several other design features to mitigate severe accidents, which the staff concluded meet the applicable requirements. These design features include the following:

- cavity flooding system
- a reactor vessel bottom head that has no penetrations
- an RPV thermal insulation system that provides an engineered pathway to supply water for cooling the vessel, and to vent steam from the reactor cavity, during severe accidents (cools the external surface of reactor vessel to provide in-vessel retention of core debris)
- reactor coolant system depressurization
- hydrogen igniters (minimizes the effect from combustible gases)
- containment spray (provides fission product scrubbing)

Economic Simplified Boiling-Water Reactor (ESBWR)

The ESBWR design also incorporates a passive containment cooling system, which uses gravity-driven natural circulation to cool the containment atmosphere. Additionally, ESBWR has

several other design features to mitigate severe accidents, which the staff concluded meet the applicable requirements. These design features include the following:

- isolation condenser supporting passive decay heat removal
- containment spray system
- gravity-driven cooling system and the basemat internal melt arrest and coolability device (prevents significant ablation of the concrete in the lower drywell)
- reactor coolant system depressurization
- inerted containment (minimizes the impact from combustible gases)

**Stakeholder Interactions**

The NRC staff held numerous public meetings with nuclear industry representatives related to the activities for Mark I and II containments. The staff also made presentations to subcommittees and the full committee of the ACRS. The NRC staff interacted with other interested stakeholders during discussions on petitions for enforcement actions, at public meetings, and in correspondence related to Mark I and II containments and various proposals for improvements, including the installation of engineered filters.

The NRC staff had additional interactions focused on Recommendations 5.2 and 6 after the issuance of SECY-15-0137. Specifically, a public meeting was held on January 7, 2016, when the staff heard from representatives of the nuclear industry, nongovernment organizations, and members of the public. The comments received during the January 7, 2016, public meeting can be found in the meeting summary dated January 20, 2016 (ADAMS Accession No. ML16013A277). NEI noted during the meeting that the industry agreed with the NRC staff's finding that further study is unlikely to identify a need for regulatory actions related to containment vents or hydrogen control. Dr. Edwin Lyman of the Union of Concerned Scientists discussed possible changes to the NRC's approach to regulatory analyses and for addressing societal measures, noting that such measures might justify additional containment-related requirements. Dr. Lyman also noted that hydrogen explosions could complicate efforts to deal with a severe accident. The staff considered insights from this meeting and the ACRS letter dated November 16, 2015, on SECY-15-0137 and prepared a "white paper" to support further interactions with the public and ACRS (ADAMS Accession No. ML16020A245). The staff's final evaluations benefited from interactions with the ACRS in February and March 2016, as discussed in the letter from ACRS dated March 15, 2016 (ADAMS Accession No. ML16075A330).

**Conclusion**

The staff has determined that there is adequate information to conclude that regulatory actions to impose further improvements to containment venting and hydrogen control are not warranted. The staff bases this finding on a conservative estimation of frequency-weighted risks to the public health and safety in comparison to the NRC's established safety goals. The staff used insights from the evaluations and agency decisions for Mark I and Mark II containments and considered the performance of other containment designs in terms of plant response and the timing of possible failures during severe accidents. The containment responses were obtained from previous studies and more recent evaluations such as the SOARCA reports and specific simulations performed for this assessment. The evaluations considered the benefits from previous regulatory actions for controlling hydrogen in Mark III and ice condenser containments and requiring mitigating strategies for beyond-design-basis external events. The NRC staff confirmed that significant margins exist between the NRC established safety goals and estimated plant risks that might be reduced by improvements in containment performance or hydrogen control. The staff's conclusion that regulatory actions are not needed to resolve Recommendations 5.2 and 6 are supported by the evaluations of these event frequencies, plant responses, the timing of barrier failures, conditional release fractions, and potential for plant changes to influence margins to the QHOs. The NRC staff is therefore closing Recommendations 5.2 and 6.

The NRC staff will continue to monitor information emerging from ongoing international research activities on containment integrity and hydrogen risk. The staff will finalize the ongoing SOARCA evaluation of ice condensers, SOARCA uncertainty study for a large dry PWR, and the Level 3 probabilistic risk assessment study. The staff will perform these longer-term activities under established programs and processes. The staff will, to the extent practical, address the conclusions and recommendations of the ACRS letter dated March 15, 2016, by using ongoing projects and future activities defined by the NRC's research needs to further examine possible mechanisms and pathways for the release of hydrogen from primary containments into surrounding structures and the consequences of hydrogen combustion.

Related to the staff's consideration of Recommendation 6, the Natural Resources Defense Council submitted a petition for rulemaking (PRM) on October 14, 2011, requesting the NRC revise 10 CFR 50.44 regarding the measurement and control of combustible gas generation and dispersal within a power reactor system (PRM-50-103). The petition addresses several issues beyond those identified in the NTTF report and will be addressed in a separate paper to the Commission.