REACTOR SAFETY GAP EVALUATION OF ACCIDENT TOLERANT COMPONENTS AND SEVERE ACCIDENT ANALYSIS

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ABSTRACT

The reactor accidents at Fukushima Daiichi have rekindled interest in Light Water Reactor (LWR) severe accident phenomenology. Post-event analyses have identified several areas that may warrant additional research and development to reduce modeling uncertainties and to assist the industry in the development of mitigating strategies and refinement of Severe Accident Management Guidelines to both prevent significant core damage given a beyond design basis event and to mitigate source term release if core damage does occur. On these bases, a technology gap evaluation on accident tolerant components and severe accident analysis methodologies has been completed with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of LWR severe accident research and augmented by insights gained from recent analyses for the Fukushima Daiichi accident. The ultimate benefit of this activity is that the results can be used as a basis for refining research plans to address key knowledge gaps in severe accident phenomenology that affect reactor safety and that are not being directly addressed by the nuclear industry or by the US Nuclear Regulatory Commission. As a result of this study, thirteen gaps were identified in the areas of severe accident tolerant components and accident modeling. The results clustered in three main areas; namely, i) modeling and analysis of in-vessel melt progression phenomena, ii) Emergency Core Cooling System equipment performance under beyond-design-basis accident conditions, and iii) ex-vessel debris coolability and core-concrete interaction behavior relevant to accident management actions. This paper provides a high level summary of the methodology used for the evaluation, the identified gaps, and finally appropriate Research and Development that may be completed to address the gaps.

KEYWORDS

Fukushima Daiichi, Severe Accident, Accident Tolerant Components

1. INTRODUCTION

1.1 Background

In the aftermath of the March 2011 multi-unit accident at the Fukushima Daiichi nuclear power plant (Fukushima), the nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations and emergency actions, particularly with respect to extreme events that might occur and that are beyond current design bases. Because of our significant domestic investment in nuclear reactor technology, the US has been a major leader internationally in these activities. The US nuclear industry is proactively pursuing a number of initiatives regarding enhancing nuclear safety for beyond design basis events (BDBEs); and the US Nuclear Regulatory Commission (NRC) continues to evaluate and, where deemed appropriate, establish new requirements for ensuring adequate protection of public health and safety in the occurrence of low probability events at a licensed commercial nuclear facility; e.g., large external events such as seismic or flooding initiators.

The US Department of Energy (DOE) has also played a major role in the US response to the Fukushima accident. Initially, DOE worked with the Japanese and the international community to help develop a more complete understanding of the Fukushima accident progression and its consequences, and to respond to various concerns regarding nuclear safety for BDBEs emerging from uncertainties about the nature and effects of the accident. DOE Research and Development (R&D) activities have been focused on providing scientific and technical insights, data, and analyses methods that ultimately support industry efforts to enhance safety. These activities are expected to further enhance the safety of currently operating nuclear power plants, as well as improving the safety characteristics of future plant designs.

Shortly after the Fukushima accident, the body of R&D work related to understanding the accident progression and mitigation was evaluated, and some level of effort was applied to start the process of enhancing the knowledge in BDBE response. This review was intended to provide a means for identifying any safety-related knowledge gaps that are currently not being addressed by DOE, industry, or NRC, thereby providing a technical basis for refining DOE's R&D activities in this area.

1.2 Objectives

With this background, the overall objective of this study is to conduct a technology gap evaluation on accident tolerant components and severe accident analysis methodologies with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of LWR severe accident research, and additionally augmented by insights obtained from the Fukushima accident. The ultimate benefit of this activity is that the results can be used to refine DOE's Reactor Safety Technology (RST) R&D plan to address key knowledge gaps in severe accident phenomenology that affect reactor safety and that are not being directly addressed by the nuclear industry or by the NRC.

To this end, the methodology used to carry out this technology gap evaluation is described first, followed by a summary discussion of gaps identified as part of the study, including recommendations on appropriate R&D that may be considered to address the gaps. Underlying details for each gap area, including discussions on safety relevance and a review of any existing R&D that has already been conducted, can be found in [1].

2. METHODOLOGY

2.1 Process Overview

The approach taken to conduct this reactor safety gap evaluation on accident tolerant components and severe accident analysis incorporated familiar features of a traditional Phenomena Identification and Ranking Table (PIRT) process that is designed to identify safety relevant phenomena, evaluate the knowledge base, and rank potential gaps [2]. A PIRT is a systematic method for gathering information from experts on a specific subject and ranking the importance of that information in order to meet an objective, which in this case is research prioritization**.** PIRTs are generally structured to address the scope and level of detail appropriate to a particular system or scenario under consideration. Evaluation of well-developed designs or specific scenarios can be more narrowly focused, while assessment of more generic designs or scenarios can be used to evaluate overall safety characteristics. Because the intent of this work was to conduct a high level reactor safety gap evaluation based on insights obtained from the recent Fukushima accident, the latter approach was adopted.

The process was initiated by forming a panel of US experts in LWR operations and safety with representatives from several DOE laboratories and industry organizations, including the Electric Power Research Institute (EPRI), the PWR Owners Group (PWROG), and the BWR Owners Group (BWROG). Representatives from the NRC and TEPCO also participated as observers to inform the process. A high-level PIRT on accident tolerant components and severe accident analysis was then developed and distributed to participants as preparatory meeting material. General severe accident areas covered in the PIRT included:

- In-vessel behavior
- Ex-vessel behavior
- Containment (and reactor building) response
- Emergency response equipment performance
- Instrumentation performance
- Operator actions to remove decay heat.

The panel interacted through a series of two workshops. This work included development of recommendations on appropriate R&D for addressing the identified gaps.

2.2 Evaluation Criteria

The most important evaluation criterion or figure of merit (FOM) for the phenomena or operation considered as part of this study is the potential impact on the release of radioactive material to the public. In practical terms, this essentially means the influence of the phenomena or operation on maintaining (or compromising) the three engineered barriers for containment of radioactive material (i.e., cladding, primary system, and the reactor containment itself). This is the common FOM for all PIRT-type analyses. Given the fact that this gap analysis was focused not only on BDBE phenomenology but also on accident tolerant components, the evaluation metrics were expanded to include a functional criterion as well. This second criterion was added on the basis that operational data for emergency response equipment is available for design basis accident (DBA) conditions, but the analogous information under BDBE conditions may not be. Thus, the two criteria utilized as figures of merit for the phenomena or operations considered as part of this study are as follows:

- Radiological consequence criterion: dose at the site boundary, worker dose, primary or secondary radioactive material inventory releases;
- Functional criterion: potential impact on system or component operability or functionality under BDBE conditions for the scenario of interest (see Section 2.3)

Importance and knowledge base rankings of a particular phenomenon or operation were evaluated according to the set of criteria noted above. The importance ranking categories were qualitative levels of High (H), Medium (M), and Low (L). Additional details regarding the ranking methodology are provided in [1], which also provides the rationale or justification for the panel rankings.

2.3 Accident Scenario Definition

The specific accident scenario used as a basis for carrying out the gap evaluation was an unmitigated Station Blackout (SBO) involving extended loss of AC power (ELAP). This scenario was selected so that the full array of severe accident conditions ranging from onset of core degradation out through failure of the reactor pressure vessel and discharge of core debris into containment would be addressed as part of the panel evaluation process. Potential operator actions to mitigate severe accident consequences were then considered as a separate, distinct category. Accident progression for both Boiling Water Reactor

(BWR) and Pressurized Water Reactor (PWR) plant designs under these conditions were evaluated as part of the analysis.

3. RESULTS AND DISCUSSION

Thirteen safety-relevant knowledge gaps were identified by the panel in the areas of accident tolerant components and severe accident analysis that are not currently being addressed by industry, NRC, or DOE. The results are listed in Table I. Recommendations on appropriate R&D to address these gaps were also developed by the panel, and are also provided in the table. It is noteworthy that two important areas related to BDBAs were identified in which gaps are known to exist, but it was concluded that efforts currently underway by industry, NRC, or DOE were adequate to address the gaps. Specifically, these areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation.

In broad terms, the gap results could be classified into five categories: i.e., i) in-vessel core melt behavior, ii) ex-vessel core debris behavior, iii) containment – reactor building response to degraded core conditions, iv) emergency response equipment performance during core degradation, and v) additional degraded core phenomenology. The gaps and associated R&D recommendations are summarized under these topical areas below.

3.1 In-Vessel Core Melt Behavior

The first, second, and fourth-ranked gaps all fell under the category on in-vessel core melt behavior. In particular, ***the highest ranked gap is related to fuel assembly/core-level degradation***. A critical aspect of accident progression is the timing of core heatup, degradation, relocation, and radionuclide release and transport. Reflooding and quenching of degraded fuel materials have also been shown to significantly impact accident progression. There are significant differences in PWR and BWR core structures that can impact late-phase in-core degradation. The panel noted that there are gaps in the existing data base for modeling BWR late-phase in-core fuel and structure degradation and relocation, especially with respect to phenomena that affect multiple assemblies. Gaps also exist for PWR late-phase in-core fuel and structure degradation and relocation. These gaps have led to differences in current modeling approaches adopted by accident progression codes that strongly impact the predicted behavior. In particular, based on the recent cross walk activity [3] between severe accident systems analysis codes developed by industry (i.e., MAAP[4]) and by the NRC (i.e., MELCOR[5]), the principal phenomenological uncertainty regarding in-core behavior is the extent that core debris is permeable to gas flow during degradation. Namely, impermeable debris (assumed in MAAP) leads to gradual accumulation of a large high temperature in-core melt accumulation akin to that formed during TMI-2, while permeable debris (assumed in MELCOR) steadily relocates to the lower head where the material collects as a debris bed.

From a reactor safety viewpoint, uncertainty related to in-core melt progression is important as it leads to large variations in the prediction of in-vessel hydrogen production [3]. In addition, these uncertainties have a strong impact on the boundary conditions for the balance of the accident sequence including core debris relocation to the lower head, melt interactions with the lower head, the mechanism(s) and timing of lower head failure, and subsequently ex-vessel debris pour conditions that impact melt spreading, the potential for failing containment structures during spreading such as the Mark I liner, and finally, debris coolability.

Reducing uncertainties related to in-core melt progression would serve to enhance severe accident management guidance related to locations and rates of water addition to the plant, as well as actions such as containment venting. In addition, an increased understanding of in-core phenomenology will improve the ability to train operators on accident management procedures and inform emergency response personnel on the best way to allocate resources (e.g., locations for water addition, power supplies, etc.).

Table I. Summary of identified gaps with associated importance rankings and recommended R&D to address the gaps

Category	Identified Gap	Importance Ranking	Recommended R&D to Address the Gap:
In-Vessel Behavior	Assembly/core-level degradation	1 [a]	• Re-examine existing tests for any additional insights that could reduce modeling uncertainties • Planning to determine if scaled tests are possible • MAAP/MELCOR evaluations to gain a common understanding of regimes where predictions are consistent and regimes where predictions differ qualitatively and quantitatively • Develop real-time tools to support SAMG enhancements and for staff training
	Lower head	2 [a,b]	• Scaled tests addressing melt relocation and vessel wall impingement heat transfer
	Vessel failure	4 [a,b]	• Scaled tests addressing vessel lower head failure mechanisms; focus on penetration-type failures
Ex-Vessel Behavior	Wet cavity melt relocation and CCI	5 [a,b]	• Modify existing models based on ongoing prototypic experiments and investigate the effect of water throttling rate on melt spreading and coolability in BWR containments
Containment-Reactor Building Response	H_2 stratification and combustion	7 [a]	• Analysis and possible testing of combustion in vent lines under prototypic conditions (i.e., condensation, air ingress, hot spots, and potential DDT)
	H_2 /CO monitoring	10	• Leverage ongoing international efforts as a basis for developing a H_2-CO containment monitoring system
	Organic seal degradation	12 [a]	• Similar to process completed by the BWR industry, develop PWR containment seal failure criteria under BDBE conditions based on available information sources
	PAR performance	13	• Evaluate optimal position in containment with existing codes that predict gas distributions • Examine performance with H_2/CO gas mixtures under BDBE environmental conditions
Emergency response equipment performance	RCIC/AFW equipment	3 [a]	• Plan for a facility to determine true BDBE operating envelope for RCIC/AFW pumps • Based on stakeholder input, construct facility and conduct testing
	BWR SRVs	6 [a]	• Testing to determine BDBE operating envelope (in RCIC-AFW test facility)
	Primary PORVs	11 [a]	• Testing to determine BDBE operating envelope (in RCIC-AFW test facility)
Additional Phenomenology	Raw water	8 [a]	• Monitor studies underway in Japan to obtain basic insights into phenomenology. • Develop tools to analyze raw water effects; apply to postulated accident scenarios. • Based on outcome of these activities, formulate additional R&D if uncertainties persist.
	Fission product transport and pool scrubbing	9 [a]	• Leverage existing international facilities to characterize: i) thermodynamics of fission product vapor species at high temperatures with high partial pressures of H_2O and H_2, ii) the effect of radiation ionizing gas within the RCS, and iii) vapor interactions with aerosols and surfaces. • Leverage existing international facilities to address the effect of H_2/H_2O and H_2/CO gas mixtures on pool scrubbing at elevated pressures and saturated conditions.

[a] Panel consensus was that Fukushima forensics offer best opportunity for insights in these areas.

[b] Panel consensus was that uncertainties in these areas are dominated by uncertainties related to assembly/core-level degradation; thus, addressing assembly/core-level uncertainties should be higher priority.

Regarding potential R&D activities to reduce the knowledge gap related to in-core melt progression, the panel grouped the recommendations into two categories; i.e., planning versus actionable items. Regarding planning, the committee concurred that data from the Fukushima reactors offer the best opportunity to fill BWR knowledge gaps with possible application to PWR knowledge gaps. Note that the full-scale data from TMI-2 is limited to one particular accident scenario and one type of reactor design. Because of the potential overall benefits to reactor safety, the panel recommends that an integrated Fukushima inspection plan be developed that identifies the types and density of data that are needed, which in this case relates to the morphology of in-core debris formations. This general observation, which applies to 11 of the 13 gaps in Table I, is being addressed as part of a separate DOE activity [6].

Although Fukushima data generally offer the best option for closing knowledge gaps, this information will take many years (possibly decades) to obtain. On this basis, the panel acknowledged that experiments may be needed to reduce knowledge gaps related to in-core melt progression on a shorter timeframe, but the *scaling rationale* for any proposed testing would need to be well established. In addition, the experiments would need to be specifically targeted at addressing data needs identified in the cross-walk activity [3]. It may be prudent to initiate planning activities to determine if appropriately scaled tests are possible to examine this complicated behavior. The panel drafted test design goals that are summarized in [1].

In terms of actionable R&D items, the panel identified three items for consideration:

- Based on modeling uncertainties identified as part of the cross-walk activity [3], reexamine previously conducted tests related to in-core melt progression (documented in [1]) to determine if additional insights can be obtained to help reduce the gaps.
- Conduct more detailed discussions, analysis and model development activities between development teams for severe accident systems analysis codes (in the US, the MAAP and MELCOR codes) to try to create common understanding between the codes given the current knowledge base in this area.
- Consider developing computational tools to inform evaluators and decision makers in the development of accident management guidelines and supporting accident response. Specifically, real-time data processing systems are needed that can take input from key plant or portable instruments and, on that basis, assess the likely plant state as well as supporting prioritization of actions for best dealing with the situation.

The 2nd ranked gap relates to core melt behavior in the lower head. Phenomena associated with core debris relocation into the head, as well as the resultant heat transfer from this material to the reactor vessel, represent critical aspects of severe accident progression that affect subsequent accident behavior. The limited prototypic data currently available are focused on PWR designs, and the full-scale data from TMI-2 is limited to one particular accident scenario. Uncertainties associated with this limited data have led to differences in current modeling approaches adopted by accident progression analysis codes. From a reactor safety viewpoint, lower head behavior is important as it has a strong impact on the boundary conditions for the balance of the accident sequence; i.e., mechanism(s) and timing of lower head failure, and the resultant ex-vessel debris pour conditions that impact melt spreading, the potential for failing key containment structures during spreading such as the Mark I liner, and finally debris coolability.

Reducing uncertainties related to lower head behavior would enhance severe accident management guidance related to locations and rates of water addition to the plant. In addition, reduction in these uncertainties would improve the ability to train operators on accident management procedures and inform response personnel on the best way to allocate water resources. However, it is noted that *uncertainties in* *lower head behavior are overshadowed by those associated with in-core melt progression*; i.e., the associated impact that those uncertainties have on melt relocation behavior to the lower head.

Regarding potential R&D activities, the committee concurred that data from the Fukushima reactors offer the best opportunity to fill knowledge gaps related to lower head behavior in BWRs with possible application to PWR knowledge gaps. Specific to this particular area, evidence on the mechanism of core debris relocation to the lower head, as well as the morphology of the material in the lower head, would be very helpful in reducing uncertainties related to lower head behavior. Also, information on the extent of thermal attack on the vessel wall and penetrations would be very beneficial.

Additional experiments addressing melt relocation behavior to the lower head and the resultant heat transfer on the vessel wall would also be beneficial, but the scaling basis needs to be well established for any proposed new testing.

In close relationship to lower head behavior, the ***4th ranked gap relates to lower head failure.*** Prototypic data are limited for characterizing the mode and size of vessel lower head failure, either through a breach of the vessel wall or failure of a penetration in the lower head. Hence, there is significant uncertainty in model predictions for the mode and timing of BWR and PWR lower head failure. Such uncertainties significantly impact predictions of subsequent accident progression phenomena, such as the temperature, morphology, and composition of debris exiting the vessel; the potential for ex-vessel debris to form a coolable geometry; and finally, the associated fission product release into the containment. Additionally, for PWRs, some plants are able to submerge the bottom of the reactor vessel which may delay or prevent reactor vessel failure.

Improved understanding of vessel failure mechanisms can lead to enhanced severe accident management guidance for existing plants related to time windows available for water addition to the plant at various locations (e.g., primary containment versus reactor vessel) and the potential for preventing or delaying vessel failure. Additional data to resolve uncertainties in this area would inform accident management strategies and operator training by providing a technical basis for the location and timing of water injection during a severe accident.

Regarding potential R&D activities, the committee again concurred that data from the Fukushima reactors offer the best opportunity to fill knowledge gaps related to lower head failure for BWRs with possible application to PWRs. Specific to this particular area, evidence on the location and nature of lower head failure (e.g., penetration versus vessel creep failure modes) would be very beneficial. Also, general information on the extent of vessel wall/penetration thermal attack in the vessel failure area(s) would be very useful.

Additional experiments addressing lower head failure mechanisms would also be beneficial, with a particular focus on penetration-type failures. However, the scaling basis needs to be well established for any proposed new testing. Again, it is noted that *uncertainties in lower head failure are overshadowed by uncertainties related to in-core behavior*, as these uncertainties impact melt arrival conditions in the lower head which, in turn, impact vessel failure characteristics.

3.2 Ex-Vessel Behavior

The ***5th ranked gap is related to ex-vessel behavior***; specifically, melt relocation from the pressure vessel and subsequent core-concrete interaction behavior under wet cavity conditions. One of the principal knowledge gaps in this area relates to an investigation by the BWROG into an alternate flooding strategy; i.e., gaps exist in the understanding of the impact on throttling water addition rates to preserve the availability of the wetwell vent path. This is the preferred option as it provides scrubbing of radionuclides prior to release and can avoid the need for an additional drywell vent path. Knowledge gaps related to this strategy and to similar possible PWR strategies include the effect of pre-existing water on the drywell/pedestal/cavity floors on melt stream breakup and spreading, as well as the timing of vessel failure and the influence of water throttling rate on spreading behavior and long-term coolability. Other questions include the effect of BWR-specific high metal content melts on core-concrete interaction and debris coolability.

Regarding potential R&D activities, the committee again concurred that data from the Fukushima reactors offer the best opportunity to fill knowledge gaps related to ex-vessel core debris spreading and debris coolability, particularly for Unit 1. Specific to this area, the extent (i.e., floor area coverage) of the debris spreading in the reactor pedestal and drywell needs to be determined, as well as the debris elevation variations. It is also important to characterize the debris morphology (i.e., monolithic crust versus particle bed) as this has a pronounced effect on coolability. If concrete erosion occurred, then it would be beneficial to characterize the cavity ablation profile. Finally, any evidence of contact and thermal attack of the containment liner would be very useful.

Related to the investigation by the BWROG into alternate flooding strategies, an actionable R&D item in this area is to analytically investigate the effect of water addition throttling rate (to preserve wetwell vent path) on core debris spreading and long term debris coolability, after appropriate modeling upgrades are made to the MELTSPREAD and CORQUENCH codes.

3.3 Emergency Response Equipment Performance under BDBA Conditions

The ***3^rd^ ranked gap relates to emergency response equipment performance under BDBE conditions***. Specifically, the reactor core isolation cooling (RCIC) for BWRs and the turbine-drive auxiliary feed water (TDAFW) for PWRs are the key safety systems that are used to remove decay heat from the reactor under a wide-range of conditions ranging from operational pressures down to lower pressures approaching cold shutdown conditions. Based on events at Fukushima [7], it is known that RCIC operation was critical in preventing or delaying core damage for days (almost three days for Fukushima Unit 2). This observation indicates that there is significant margin in RCIC performance that has been neither quantified nor qualified. Technically, this is a highly important lesson-learned from Fukushima that needs to be explored and quantified for the benefit of the operating fleet both domestically and internationally. Furthermore, quantifying emergency response equipment performance under these conditions could form the technical basis for emergency mitigation strategies that would greatly increase options for the successful implementation of FLEX [8] (or equivalent measures for design extension conditions in other countries) and SAMG measures under ELAP conditions for both BWR and PWR designs.

This is recognized as an important area for further research by US industry as well as international organizations. The principal R&D need is to determine the *actual operating envelope* for emergency response equipment performance under BDBE conditions for both BWRs and PWRs; specifically, RCIC and TDAFW systems. A facility to conduct this type of testing may be needed. If this is determined to be the case, then actionable R&D items in this area would be to: i) perform the necessary planning for a facility of this type, ii) construct the facility, and iii) perform testing necessary to determine the actual operating envelopes for RCIC and TDAFW systems under BDBE conditions.

Two other gaps were identified by the panel in the category on emergency response equipment performance. In particular, the ***6^th^ ranked gap relates to BWR safety relief valve (SRV) performance under BDBE conditions,*** while the ***11^th^ ranked gap relates to PWR primary system pilot-operated relief valve (PORV) performance***. In general, data on SRV and PORV performance under design basis accident (DBA) conditions are well known. However, the panel identified a knowledge gap on the performance of these devices involving extended cycling under high temperature in the process gases flowing through the valve as well as the high temperature and pressure conditions expected inside containment during protracted BDBE scenarios such as those experienced at Fukushima. For example, in the case of PWRs, radiation heat transfer from the process gases may cause failure of the solenoid that is used to maintain the PORV in an open position.

Regarding potential R&D activities, the panel noted that appropriate testing to reduce knowledge gaps related to SRV performance under BDBE conditions may be possible in a facility similar to the type that would be used to test RCIC and TDAFW performance, as noted above. Testing of PWR PORV performance may require a facility with significantly higher temperatures and pressures due to the higher operating conditions in a PWR.

3.4 Containment and Reactor Building Response

Four additional gaps were identified under the category of containment and reactor building response. To begin, ***the 7th ranked gap is related to H_2 stratification and combustion***. The panel noted that there are uncertainties in characterizing random ignition sources in plant-level analyses. Other identified information needs include: i) flame front propagation in the containment vent line, ii) stratification in large physical structures exemplified by containments and reactor buildings, iii) methods for modeling combustible gas concentration variations in lumped parameter codes, and finally iv) auto-ignition at high temperatures.

Regarding safety relevance, if uncontrolled deflagrations occur, they can result in direct challenges to containment. In addition, deflagrations occurring outside containment (e.g., the reactor building) can potentially damage safety-significant structures, emergency response equipment and pose a significant safety hazard to plant personnel. They can also inhibit the ability of plant personnel to implement accident management procedures that are required to reestablish or maintain adequate core cooling.

Regarding R&D to address the gaps, the panel noted that US industry and the international community already have substantial work underway in this area. Domestically, the NRC has issued the severe accident vent Order, EA-13-109 [9], and the industry has responded by providing guidance for complying with this order [10]. The MAAP5 enhancement project is examining lumped parameter approaches for evaluating hydrogen transport issues in containment. Internationally, the PANDA and THAI facilities are actively conducting research on gas mixing and stratification in large structures. Despite these efforts, additional R&D may be warranted to consider specific issues of interest, such as combustion in vent lines and factoring in practical considerations such as condensation, air ingress, hot spots, and the potential for deflagration-detonation transition (DDT).

Closely related to this topic, ***the 10th ranked gap relates to H_2/CO monitoring in containment under BDBE conditions.*** Measurements of this type are traditionally made using either a hydrogen analyzer that measures electrical conductivity of containment gases or gas mass spectroscopy. The challenge here is predominately equipment related; i.e., development of a system that can monitor potential flammability from H_2 and CO under ELAP conditions while accounting for practical considerations such as non-homogeneous gas mixtures in containment and steam condensation in the gas sample lines.

Management of combustible gases during a severe accident is a key LWR technical challenge. Events at Fukushima [7] illustrated the point that decision making related to accident management actions (such as venting or actuating containment sprays) could be better informed if the operators had knowledge of the time-dependent gas composition in containment. Thus, instrumentation that can provide this information under BDBE conditions would be very beneficial in supporting this decision making.

Regarding potential R&D to address this gap, the panel noted that a joint CEA-EdF-Canberra-AREVA project is already underway to develop a system that can measure the composition of gases released through the containment vent line [11]. Thus, the panel recommended that industry leverage these efforts as a basis for developing a H_2-CO containment monitoring system. In addition, it should be noted that one US supplier announced they are offering a real-time monitoring system that can reportedly measure hydrogen concentration, pressure, humidity, temperature, and selected fission product gas concentrations in the containment under harsh accident conditions [12]. Thus, any decisions regarding additional R&D in this area should be re-evaluated as more information on this product becomes available. With this development, the importance ranking for this particular gap may be lower than that originally evaluated by the panel (see Table I).

The ***12th ranked gap relates to organic seal degradation under BDBE conditions***. Because elastomeric seals form integral elements of the containment boundary, their ability to remain leak-tight under accident conditions (including BDBE conditions) is key for meeting the principal containment functional requirement to mitigate fission product release to the environment. Knowledge of sealant vulnerabilities can be key to accident management decisions for ventilation of structures adjoining the primary containment. Elastomeric seals also form integral elements of the integrity of the reactor vessel (BWRs) and reactor coolant system boundary (PWRs) whose failure can accelerate the loss rate of cooling water from the reactor vessel. For instance, seal leakage from recirculation pumps in BWRs has been well characterized and is factored into SAM planning.

Seal degradation has been the subject of research in the nuclear industry for some time due to the relevance to containment integrity [13]. In general, seal performance has been reasonably characterized under DBA conditions; however, there is much less information on the ability of seals to remain leak-tight under BDBE conditions that include elevated temperature, pressure, and radiation effects in the presence of high steam concentrations, particularly for seals that have undergone significant aging.

Regarding R&D in this area, in response to the severe accident capable vent Order EA-13-109 [9], the BWR industry has evaluated [10] available test and engineering evaluation information sources [14-19] to develop containment failure criteria that envelopes the range of expected conditions encountered inside containment under extended BDBA conditions involving ELAP. This analysis includes the effects of penetration degradation on containment leakage. However, the analogous investigation has not been completed for PWR containments. This investigation is currently being considered by EPRI and the PWROG. The committee recommends that this investigation be undertaken.

The ***13th ranked gap relates to Passive Autocatalytic Recombiner (PAR) performance under BDBE conditions***. Performance data for these devices with H_2/air gas mixtures are readily available, but the panel noted limited open literature information regarding the effectiveness of PARs to reduce combustible gas levels when high concentrations of aerosol fission products or CO are present. Aside from the performance of the PAR units themselves, an equally important question relates to where these units should be positioned in the containment to optimize their performance. This latter question relates, in turn, to our ability to predict combustible gas distributions in containment during a severe accident (see previous gap discussion on H_2 stratification and combustion).

This gap area was ranked the lowest of all those identified due to the fact that PARs are not deployed in any operating US plants as a severe accident mitigation measure.[1] However, PARs are used in the Westinghouse AP1000 plants being built in the US and are commonly used in other countries, including

[1] There are a limited number of plants in the US that have PARs installed as DBA hydrogen control measures but these PARs are not designed for severe accident flammable gas generation rates.

US-designed plants that are operating or under construction. Thus, this gap is relevant for SAMG planning and implementation for those units.

In terms of potential R&D in this area, it was previously noted that there are only limited data [20] available regarding the effectiveness of these devices to reduce combustible gas levels when a H_2/CO gas mixture is present. Also, degradation of PAR performance due to severe accident conditions is not widely reported in the open literature. There are also questions related to positioning of these devices in containment that could be addressed through analyses with codes that are able to predict combustible gas distributions in containment under severe accident conditions.

3.5 Additional Phenomenology

The panel identified two additional gaps that were classified under the category of additional phenomenology. ***The 8th ranked gap relates to the influence of raw water on the ability to maintain long term core cooling.*** During the Fukushima accidents, large volumes of seawater were injected into Units 1-3 in an effort to cool the reactor cores and stabilize the accident [7,21]. Current US industry guidance [22] calls for the use of seawater or other sources of raw water (e.g., river water with high levels of sediment) to provide core cooling should fresh water sources be exhausted. The main issue with raw (including sea) water injection is that as a result of boiling in the core, large amounts of solute could precipitate on the surface of fuel pins, thereby restricting coolant flow passages and degrading heat transfer. For BDBE conditions involving highly degraded core conditions, there is a similar concern that precipitates could block porosity in the debris, thereby degrading the coolability. There are currently a limited number of laboratory studies being conducted internationally (i.e., in Japan) to address these questions. For PWRs, there is also a concern related to fouling of heat transfer surfaces when raw water is used as a feed source for steam generator heat removal.

Potential impacts associated with the use of raw water to reestablish/maintain core cooling were brought into focus by events at Fukushima. In terms of R&D needs in this area, scoping studies and potentially bench top experiments would provide basic insights into key phenomenology. As noted earlier, benchtop experiments are already underway in Japan, and these efforts should be monitored for potential application to US accident management planning activities.

Questions related to the potential impact of accident management strategies (in this case, raw water injection) are usually addressed with system level codes such as MAAP and MELCOR. However, these codes currently do not have models that account for the effects of water impurities on accident progression. As more information from the scoping studies becomes available, these codes should be upgraded to incorporate any findings. The codes should then be applied to postulated accident sequences to scope out potential consequences related to core debris cooling and fission product release. Depending upon findings from the scoping and plant level studies, additional R&D may be warranted to reduce phenomenological uncertainties and/or develop new models that better reflect physical reality.

The 9th ranked gap relates fission product transport and pool scrubbing. Regarding fission product transport, the panel noted that there has been significant R&D conducted in this area because it is a key factor influencing reactor safety. However, based on events at Fukushima, a few information needs were identified that may warrant additional consideration. In particular, data are needed to characterize the thermodynamics of fission product vapor species in high temperature conditions with high partial pressures of steam and hydrogen; the effects of radiation ionizing gas within the reactor coolant system (RCS); and vapor interactions with aerosols and surfaces. In addition, there are no data for evaluating the effects of raw water addition on fission product transport. Regarding late phase ex-vessel behavior, data are needed to assess the effect of H_2/H_2O and H_2/CO gas mixtures on pool scrubbing at elevated pressures and saturated conditions.

In terms of R&D in this area, the US NRC is currently investigating the effects of raw water on fission product transport in containment [23]. In Japan, the Nuclear Regulatory Authority (NRA) is funding research that may provide insights into the effect of H_2/H_2O and H_2/CO gas mixtures on pool scrubbing [24]. In addition, there is the potential to obtain fission product scrubbing data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France) [25]. One of the principle knowledge gaps to address in this area relates to the influence of elevated pressures and saturated conditions on pool scrubbing (applicable to the suppression chambers of BWRs); these overseas facilities may be capable of providing this information.

4. SUMMARY AND CONCLUSIONS

A technology gap analysis on accident tolerant components and severe accident analysis methodologies has been completed with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of LWR severe accident research and augmented by insights gained from recent analyses for the Fukushima Daiichi accident. The ultimate benefit of this activity is that the results can be used as a basis for refining research plans to address key knowledge gaps in severe accident phenomenology that affect reactor safety and that are not being directly addressed by the nuclear industry or by the US NRC. As a result of this study, thirteen gaps were identified in the areas of severe accident tolerant components and accident modeling. The results clustered in three main areas; namely, i) modeling and analysis of in-vessel melt progression phenomena, ii) ECCS equipment performance under beyond-design-basis accident conditions, and iii) ex-vessel debris coolability and core-concrete interaction behavior relevant to accident management actions. The top five identified gaps were: i) assembly/core-level degradation during a severe accident, ii) core melt behavior in the lower head, iii) emergency equipment performance (i.e. RCIC and TDAFW systems) during a severe accident, iv) lower head failure mode, and v) ex-vessel core melt relocation and core-concrete interaction under flooded cavity conditions. The consensus opinion of the expert panel was that uncertainties related to assembly/core-level degradation overshadow those associated with lower head and ex-vessel behavior, since in-core behavior sets the boundary conditions for the balance of the sequence.

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