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Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis

Nuclear Engineering Division

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March 31, 2015

Executive Summary: Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis

The overall objective of this study was 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 light water reactor (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 the Department of Energy's (DOE) Reactor Safety Technology (RST) research and development (R&D) program plan to address key knowledge gaps in severe accident phenomena and analyses that affect reactor safety and that are not currently being addressed by the industry or the Nuclear Regulatory Commission (NRC).

In the aftermath of the March 2011 accident at the Fukushima Daiichi nuclear power plant (Fukushima; see Figure 1), 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.

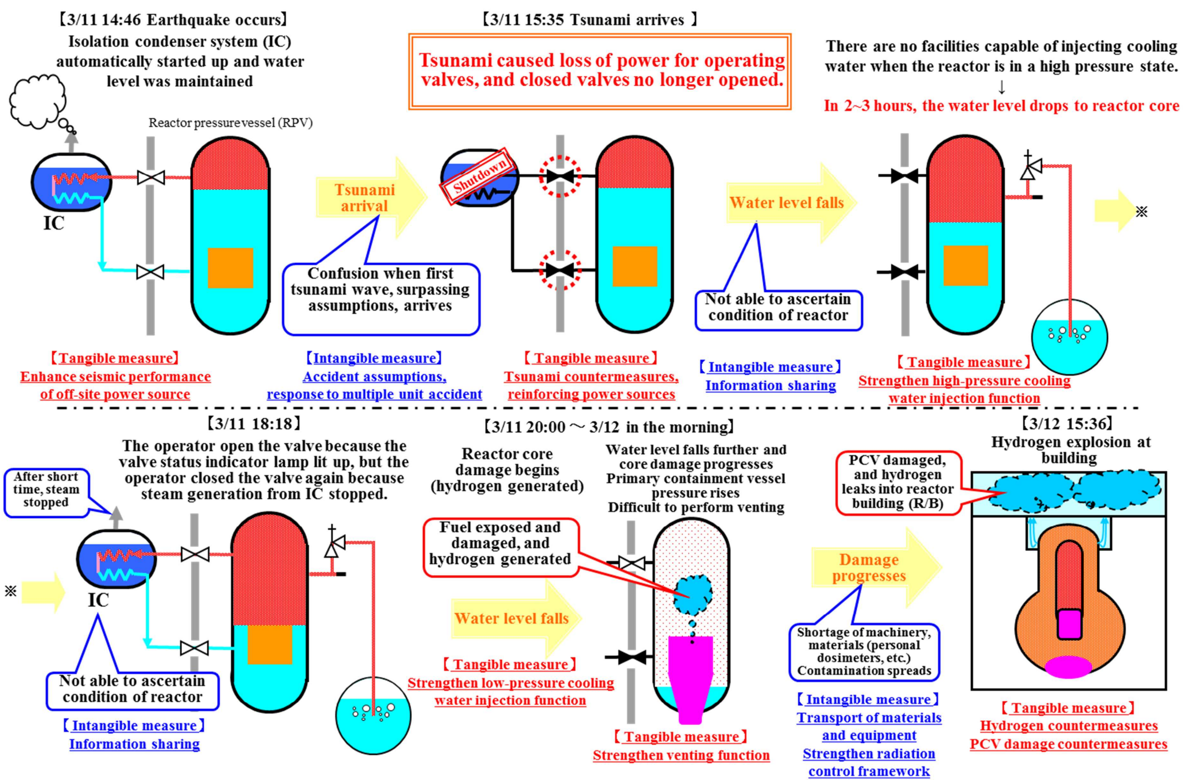

Figure 1. Summary of Accident Progression at Fukushima Daiichi Unit 1 and Necessary Countermeasures (Courtesy of TEPCO).

Because of our significant domestic investment in nuclear reactor technology, the United States (US) has been a major leader internationally in these activities. The US nuclear industry is pursuing a number of safety initiatives on its own, and the NRC continues to evaluate and, where deemed appropriate, establish new requirements for ensuring adequate protection of public health and safety considering risk significant events at a licensed commercial nuclear facility; e.g., extreme external events such as seismic or flooding initiators. Based on these activities by industry, NRC, and DOE, several areas have been identified that may warrant additional R&D to reduce modeling and analysis uncertainties and to assist the industry to develop mitigating strategies to prevent significant core damage given a beyond design basis event (BDBE), and to refine Severe Accident Management Guidelines (SAMGs) that can mitigate challenges to remaining fission product boundaries that could result in a release if core damage does occur.

In addition to this US gap evaluation effort, it is noteworthy that complimentary activities are underway internationally. In particular, a senior expert group on SAfety REsearch opportunities post-Fukushima (SAREF) was established in 2013 by the Committee for the Safety of Nuclear Installations (CSNI) to establish a process for identifying and following up on research opportunities for addressing safety research gaps and advancing safety knowledge related to the Fukushima Daiichi nuclear accident [1]. Organizations from twelve countries (including the NRC and DOE within the US) are participating in this activity. The work scope includes identifying research opportunities that use information from Fukushima Daiichi, either available now or to be obtained during decommissioning, that will provide additional safety knowledge of common interest to the member countries. The group will submit a report with prioritized recommendations for safety research activities to the CSNI in 2016.

The approach taken to conduct this US gap evaluation incorporated familiar features of a traditional Phenomena Identification and Ranking Table (PIRT) process. PIRTs are generally structured to address the scope and level of detail appropriate to a particular system or scenario under consideration; e.g., 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 gap evaluation based on insights from the Fukushima accident, the latter approach was adopted.

The process used a panel of US experts in LWR operations and safety with representatives from the DOE staff, DOE laboratories [i.e., Argonne National Laboratory (ANL), Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), and Sandia National Laboratories (SNL)] and industry [the Electric Power Research Institute (EPRI), boiling water reactor owner's group (BWROG), and pressurized water reactor owner's group (PWROG)] to identify and rank knowledge gaps, and also to identify appropriate R&D actions that may be considered to close these gaps. Representatives from the NRC and the Tokyo Electric Power Company (TEPCO) participated as observers in this process. General severe accident areas covered in this evaluation included:

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

Panel deliberations led to the identification of thirteen (13) knowledge gaps on accident tolerant components and severe accident analysis methods that were deemed to be important to reactor safety and are not being currently addressed by industry, NRC, or DOE. The results are summarized in Table 1. The recommended R&D actions developed by the panel to address the gaps are also summarized in the table. The panel noted that information from the damaged Fukushima reactors provides the potential for key insights that could be used to address virtually all the identified gaps (i.e., 11 out of 13). Because of this potential the panel recommended that an integrated Fukushima examination plan be developed, from the US perspective, that identifies the types and density of data needed from the reactors to address these gaps.

It is noteworthy that the panel identified two important areas related to beyond design basis accidents (BDBAs) in which gaps are known to exist, but it was concluded that efforts currently underway by industry, NRC, DOE, and the international community could address the gaps. Specifically, these areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation. For completeness, background in these two areas regarding the known gaps as well as efforts underway to address these gaps is provided in Appendix B. These efforts should be monitored to ensure that existing gaps are addressed.

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 first, second, and fourth-ranked gaps are all under the category on in-vessel core melt behavior. In particular, the highest ranked knowledge gap is associated with assembly/core-level degradation; the second is core melt behavior in the lower head; and the fourth is lower head failure. Recent complimentary analyses of the Fukushima reactor accidents using the MELCOR [2] and MAAP [3] codes have been completed as part of a joint industry-DOE effort, along with a supplemental DOE-sponsored study on ex-vessel behavior for Unit 1 with the MELTSPREAD and CORQUENCH codes [4] that utilized the ex-vessel debris pour conditions from the MELCOR [2] and MAAP [3] calculations as input. As part of this supplemental study [4], it was noted that the ex-vessel debris pour conditions between the two codes varied dramatically, and this led to profound differences in the predicted ex-vessel behavior that determines thermal loads on containment, as well as the potential for failing key containment structures such as the Mark I liner. These findings prompted a follow-on jointly sponsored (industry-DOE) cross-walk

Table 1. 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 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 a 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 the facility and conduct the 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, the latter should be higher priority.

activity between the MAAP and MELCOR teams that focused on identifying key modeling differences between the two codes that could lead to large differences in predicted ex-vessel melt release conditions [5]. The results indicate that the discrepancy begins in the early phases of in-core melt progression; i.e., the different methods used to model assembly blockages, the resultant debris size, the porosity of these blockages, and how these formations influence the overall progression of in-core melt front propagation. Unfortunately, there are currently insufficient experiment and full-scale prototype [i.e., Three Mile Island Unit 2 (TMI-2), Fukushima] data that can be used to assess these modeling differences. From a reactor safety viewpoint, this is an important issue as the modeling differences lead to large disparities in the accident signature; i.e., in-vessel hydrogen production, lower head behavior, predicted lower head failure mode, and ex-vessel behavior.

Closely related to this topic, the fifth ranked gap is under the category on ex-vessel behavior; specifically, melt relocation from the pressure vessel and subsequent core-concrete interaction (CCI) behavior under wet cavity conditions. From a reactor safety viewpoint, although containment failure by ex-vessel core debris interacting with structural concrete is categorized as a late phase event, the potential radiological consequences could be substantial and warrant effective strategies to prevent or mitigate such a release. As one of several strategies, severe accident management guidelines (SAMGs) for many operating LWRs include flooding the reactor cavity in the event of an ex-vessel core debris release. New reactor designs also incorporate provisions for cavity flooding as a mitigation feature. 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. There is strong international interest in this area, and there is the potential for international collaboration in assessing data from Fukushima and in conducting additional large-scale experiments on this topic.

The third highest ranked gap overall is under the category on emergency response equipment performance under beyond design basis extension (BDBE) conditions. Specifically, the Reactor Core Isolation Cooling (RCIC) for boiling water reactors (BWRs) and the Turbine Driven Auxiliary Feedwater (TDAFW) systems for pressurized water reactors (PWRs) are the key safety systems that are used to remove decay heat from the reactor under a wide-range of conditions. Both systems use steam produced by water boiling from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core (BWRs) or steam generators (PWRs) to maintain the needed water inventory for long-term core cooling for a wide set of operating pressures. In many cases, the same turbine / pump is used in both BWRs and PWRs.

Based on events at Fukushima [6], it is known that RCIC operation was critical in delaying core damage for days (almost three days for Fukushima Unit 2) even though the turbine-pump system ran without direct current (DC) power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently operated in a self-regulating mode supplying water to the core and maintaining core-cooling until it eventually failed at about 72 hours. This observation indicates that there may be 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 US operating fleet. Furthermore, quantifying emergency response equipment performance under these conditions could aid in providing safety margins for current license renewals, subsequent license renewals, as well as assist internationally. Finally, this expanded understanding could form the technical basis for emergency mitigation strategies that could greatly increase options for the successful implementation of SAMG and diverse and flexible coping capability or "FLEX" SAMG measures under extended loss of alternating current (AC) power conditions for both BWR and PWR reactor designs. On these bases, the panel acknowledged a need to determine the true operating envelopes of both RCIC and TDAFW pumps under severe accident conditions. There is also international interest in this area, and there is the potential for conducting large-scale experiments on this topic.

Two other gaps were identified by the panel in the category on emergency response equipment performance under BDBE conditions, and one of these was ranked in the upper 50 % of all identified gaps in terms of safety relevance. In particular, BWR safety relief valve (SRV) performance under BDBE conditions was ranked by the panel as the 6^{th} most important gap. In addition, PWR primary system Pilot-Operated Valve (PORV) performance was also identified as a gap, but the ranking was lower; i.e., 11^{th} out of 13 in total. In general, SRV and PORV performance under DBA conditions is well known. However, the panel identified knowledge gaps on the performance of these devices involving extended cycling under high temperature conditions expected in the process gases flowing through the valve as well as the high temperature and pressure conditions inside containment during protracted BDBE scenarios such as those experienced at Fukushima. For example, for 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. On these bases, the panel acknowledged a need to determine the actual operating envelope of BWR SRVs and PWR primary side PORVs under severe accident conditions.

Four additional gaps were identified under the category of containment and reactor building response; i.e., H_2 stratification and combustion was ranked 7^{th}, H_2/CO monitoring was 10^{th}, organic seal degradation was ranked 12^{th}, and Passive Autocatalytic Recombiner (PAR) performance was ranked 13^{th}.

The events at Fukushima [6] clearly illustrated the effect that combustible gas production can have on the course of a severe accident. In particular, due to over-pressurization, combustible gases were able to leak from the containments, accumulate in the reactor building, and subsequently explode leading to significant damage to the buildings at three of the four affected units. The panel noted that there are uncertainties on characterizing random ignition sources in plant-level analyses. Identified data needs in this area 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.

Closely related to this topic, the ability to monitor combustible gas levels in containment under BDBE conditions was also identified as a gap. 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. Events at Fukushima [6] 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.

Organic seal degradation under BDBE conditions was also identified as a gap. Typical containments include hundreds of penetrations for piping, instrument and power cabling. Often these seals are made using organic materials. Although seal performance under design basis accident (DBA) conditions has been well characterized, there is much less information on the ability of these seals to remain leak-tight under BDBE conditions that include elevated temperature, pressure, steam concentrations and radiation effects, particularly for seals that have undergone significant aging.

The final gap identified by the panel under the category of containment and reactor building response is PAR performance for CO/H_2 gas mixtures that can arise under conditions involving core-concrete interaction. Performance data for these devices with H_2/air gas mixtures are readily available, but the panel noted limited knowledge regarding the effectiveness of PARs on reduction of combustible gas levels when high concentrations of aerosol fission products or CO are present. This gap was ranked the lowest of all those identified due to the fact that PARs are not deployed in any operating US plants as severe accident flammable gas control measures[1]. However, PARs are used in the Westinghouse AP1000 plant design being built in the US and are commonly used in other countries, including US-designed plants that are operating or under construction. Thus, this gap is relevant for SAMG planning and implementation for those units.

The panel deliberations identified two other gaps that were classified under the category of additional phenomenology. In particular, the influence of raw water on accident management procedures was ranked as the 8th most important gap, followed by fission product transport and pool scrubbing that was ranked 9th. 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. There are currently a limited number of studies being conducted in Japan investigating the thermal-

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[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.

hydraulic characteristics of saline solutions in annular tube geometries as well as small-scale simulated debris configurations [7]. The Japan Atomic Energy Agency (JAEA) is also investigating the impact of salt on the chemical and physical form of solidified $(U,Zr)O_2$ debris [8]. Although these studies are providing some preliminary information, the panel judged that there were still knowledge gaps in this area related to the effect of raw water on fission product transport and the coolability of highly degraded core debris; in particular, the potential for precipitates to block coolant passageways and degrade cooling.

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 knowledge gaps have been 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. The US NRC and the Japan Nuclear Regulatory Authority (NRA) are funding research that may provide insights about these latter two issues. In addition, there is the potential to obtain data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France) [9].

ACKNOWLEDGEMENTS

Successful preparation of this report required input and support from several individuals and organizations. The US Department of Energy, Office of Nuclear Energy, Light Water Reactor Sustainability program funded the participation of present and former national laboratory participants as well as the Reactor Safety Technologies pathway leader. In addition, various industry organizations provided substantial in-kind contributions by providing technical experts to participate in this process; these organizations included the Electric Power Research Institute, Exelon Corporation, GE-Hitachi, the Pressurized Water Reactors Owners Group, the Boiling Water Reactor Owners Group, and Southern Nuclear. Finally, two organizations provided technical experts to participate in the panel meetings as observers to the overall process. In particular, Mr. Yasunori Yamanaka and Mr. Kenji Tateiwa from Tokyo Electric Power Company attended, as well as Drs. Sudhamay Basu and Richard Lee from the US Nuclear Regulatory Commission Office of Nuclear Regulatory Research. These individuals facilitated the overall process by providing key clarifications in various areas as the meetings progressed. These efforts are greatly appreciated.

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Acronyms

AC	Alternating current
AFW	Auxiliary feedwater
ANL	Argonne National Laboratory
BDBA	Beyond design basis accident
BDBE	Beyond design basis event
BWR	Boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CCI	Core-concrete interaction
CSNI	Committee on the Safety of Nuclear Installations
CST	Condensate storage tank
DBA	Design basis accident
DC	Direct current
DOE	Department of Energy
ELAP	Extended loss of AC power
EPRI	Electric Power Research Institute
FOM	Figure of merit
HPCI	High pressure coolant injection
IC	Isolation condenser
ICI	In-core instrument
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
LOCA	Loss of coolant accident
LWR	Light water reactor
MCCI	Molten core-concrete interaction
MSIV	Main steam isolation valve
NRA	Nuclear Regulatory Authority
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PAR	Passive autocatalytic recombiner
PIRT	Phenomena identification and ranking table
PORV	Pilot operated relief valve
PWR	Pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and development
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RST	Reactor safety technology
RPV	Reactor pressure vessel

Acronyms (Contd.)

SAM	Severe accident management
SAMG	Severe accident management guidance
SAREF	Safety research opportunities post-Fukushima
SBO	Station blackout
SG	Steam generator
SNL	Sandia National Laboratory
SRV	Safety relief valve
TBR	Technical basis report
TDAFW	Turbine driven auxiliary feed water
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
US	United States

Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis

1.0 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 basis events. 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 BDBEs, and the US NRC continues to evaluate and, where deemed appropriate, establish new requirements for ensuring adequate protection of public health and safety in the occurrence of risk significant events at a licensed commercial nuclear facility; e.g., extreme external events such as seismic or flooding initiators.

The 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 beyond design basis events emerging from uncertainties about the nature and effects of the accident. DOE 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. DOE recognizes that the commercial nuclear industry is ultimately responsible for the safe operation of licensed nuclear facilities. As such, industry is considered the primary “end user” of the results from DOE-sponsored R&D work in this area.

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. The Reactor Safety Technologies R&D program is a pathway that is part of the LWRS program. The objective of this pathway is to improve understanding of beyond design basis events and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. This information will be used to aid in developing mitigating strategies for beyond design basis events and improving severe accident management guidelines for the current light water reactor fleet.

Thus, there exists a need for a more comprehensive review on what the industry has been engaged in regarding BDBEs, as well as what R&D activities NRC is supporting in this area. This review would provide a means for identifying any safety-related knowledge gaps that are currently not being addressed by DOE or industry, thereby providing a technical basis for refining DOE's R&D activities in this area.

It is noteworthy that complimentary activities are underway in this area internationally. In particular, a senior expert group on SAfety REsearch opportunities post-Fukushima (SAREF) was established in 2013 by the CSNI to establish a process for identifying and following up on research opportunities for addressing safety research gaps and advancing safety knowledge related to the Fukushima Daiichi nuclear accident [1]. Organizations from twelve countries (including the NRC and DOE within the US) are participating in this activity. The work scope includes identifying research opportunities that use information from Fukushima Daiichi, either available now or to be obtained during decommissioning, that will provide additional safety knowledge of common interest to the member countries. The group will submit a report with prioritized recommendations for safety research activities to the CSNI by 2016.

1.2 Objectives

Various analyses by DOE and industry in the wake of Fukushima have identified a few areas that may warrant additional R&D to reduce modeling uncertainties and to assist the industry in the development and refinement of Severe Accident Management Guidelines (SAMG) to both prevent significant core damage given a beyond design basis event and to mitigate challenges to remaining fission product boundaries and releases if a core damage event does occur. Both the PWR and BWR Owners Groups (PWROG and BWROG) have updated their generic SAMGs after the Fukushima event based on initial insights gained from the reconstruction of the scenarios at each of the damaged units. The PWROG and BWROG will continue to follow new information and insights and incorporate them into future updates of the guidance as appropriate to address accident management and train their reactor operators on these SAMG's.

Accident management is the diagnosis and selection of appropriate strategies for implementation based on direct or indirect indications of plant status. Typically accident management is symptom based diagnosis, and decision making is based on instrumentation indications or derived values from available parameters. However, severe accident management is not rule based (e.g., Emergency Operating Procedures) but rather knowledge based wherein the user (typically the Technical Support Center staff) needs to have a basic understanding of the potential severe accident progression and phenomena. The decision process is based on assessing the past and projecting the present plant conditions to identify future plant conditions, especially challenges to fission product boundaries (e.g., containment and steam generator tubes for PWRs). The severe accident management training at each site includes severe accident progression and phenomena that are based on the current understanding of severe accidents as embodied in the simulation codes such as MAAP. The training is re-enforced by drills wherein the user has the opportunity to practice severe accident management principles in response to scenarios developed from simulation codes. Therefore, any research that can reduce uncertainties and thus improve the severe accident progression prediction can be useful to enhance accident management guidelines and associated training programs.

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 summarized in Section 2. The results are then provided in Section 3, which begins by providing a high-level overview of the identified gaps, followed by presentation of technical details for each gap including the safety relevance and a brief review of any existing R&D that has already been conducted in the area. Section 4 then provides a summary of the findings, including recommendations on appropriate R&D that may be considered to address the gaps.

2.0 GAP EVALUATION APPROACH

2.1 Process Overview

The approach taken to conduct this reactor safety gap evaluation on accident tolerant components and severe accident analysis incorporates familiar features of a traditional PIRT process that is designed to identify safety relevant phenomena, evaluate the knowledge base, and rank potential gaps [10]. 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. The overall process is illustrated in Figure 2-1.

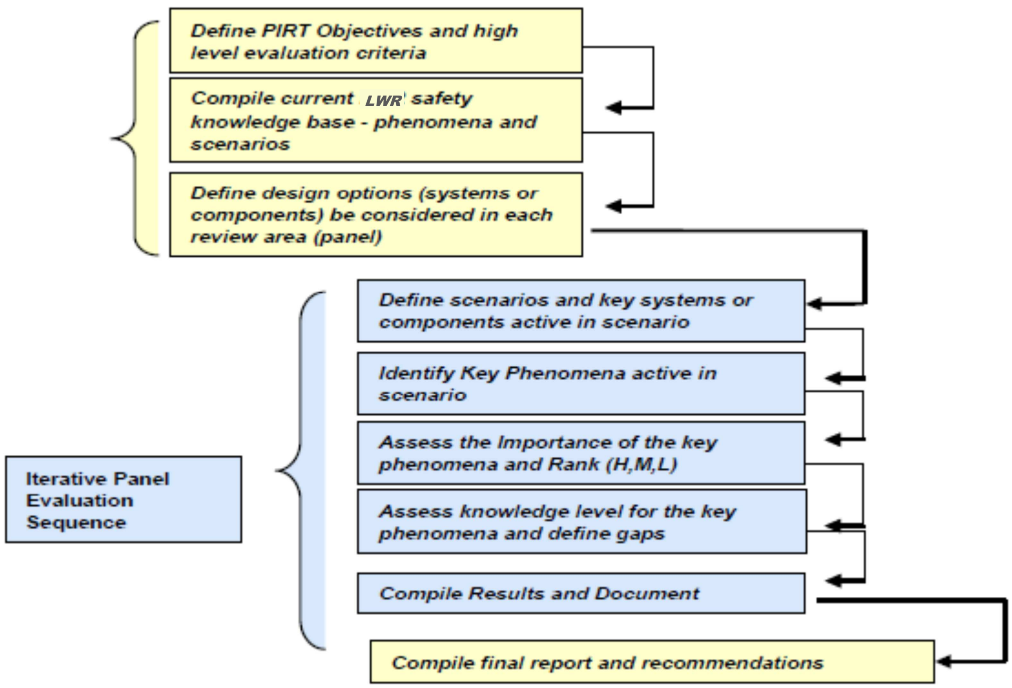


Figure 2-1. Sequence of Gap Analysis Activities and Panel Process.

Note that this process has been used extensively in reactor safety evaluations; e.g., severe accident PIRT analyses have been previously conducted for existing plant designs (i.e., Gen II; e.g., see [11]), as well as advanced plant concepts (i.e., Gen III+ and IV; e.g., see [12]).

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 is to conduct a high level reactor safety gap evaluation based on insights obtained from the recent Fukushima accident, the latter approach is adopted.

This effort required preparatory activities that included development of panel study objectives and criteria, identifying expert participants, preparation of an initial high level draft PIRT table that served as a template to guide panel discussions, and finally organizing and coordinating panel interactions. Consistent with the overall approach taken for this gap evaluation, the panel process included: a) defining a generic accident scenario applicable to both BWR and PWR designs, b) identifying key phenomena and components active in that scenario, c) assessing the importance of these phenomena and components to overall plant safety, d) evaluating the adequacy of currently available information for addressing any identified issues, and finally e) documenting the work.

The process was initiated by forming a panel of US experts in LWR operations and safety with representatives from several DOE laboratories (i.e., ANL, INL, ORNL, and SNL) and industry (EPRI, BWROG, PWROG). All panel members co-authored this report; see the cover pages for a list of participants and their respective organizations. Representatives from the NRC and TEPCO also participated as observers in the process; these individuals are recognized in the acknowledgments. 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 include:

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

Work began by holding an initial panel meeting at ANL on 30-31 October 2014. A follow-up meeting was held at ANL on 7-8 January 2015 in order to finalize the results. During the second meeting, the panel also worked to define appropriate R&D for addressing identified gaps.

Additional details on the gap evaluation criteria, gap importance and knowledge base rankings, and the specific accident scenario considered as part of this study are provided below.

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 2.5)

2.3 Gap Importance Ranking

The importance ranking of a particular phenomenon or operation was evaluated according to the set of criteria (figures of merit) noted above. The importance ranking categories are qualitative levels of High (H), Medium (M), and Low (L). These rankings have been found in previous studies to provide adequate resolution and to be consistent with an expert opinion process. The detailed gap evaluation results are summarized in the form of a table, which includes comment sections for each ranking. Subsequent sections of the report provide details of the rationale or justification for the panel importance level ranking. The general descriptions of these importance ranking levels based on the evaluation criteria are summarized in Table 2-1.

Table 2-1. Phenomena or Operation Importance Ranking Scale.

Rank	Definition
High (H)	Phenomenon/operation has a controlling impact on the FOM
Medium (M)	Phenomenon/operation has a moderate impact on the FOM
Low (L)	Phenomenon/operation has a minimal impact on the FOM

2.4 Gap Knowledge Base Ranking

Evaluating the knowledge state of a phenomenon or operation generally involves the assessment of both the modeling capabilities and the database to validate the model or operation. The specific criteria used to characterize knowledge states as part of this study are summarized in Table 2-2. In general, the knowledge state is ranked as High (H) if a physics-based or correlation-based model is available that adequately represents the phenomenon or operation over the parameter space of interest, which for this study is a BDBE condition. Furthermore, this ranking is appropriate if a database adequate to validate the relevant model or operation exists or the data are available to make an assessment.

The knowledge status is ranked as Medium (M) if a candidate model or correlation is available that addresses most of the phenomenon or operation over at least some portion of the parameter space. In this case, data are available but are not necessarily complete or of high fidelity, allowing only moderately reliable assessments.

The knowledge state is ranked as Low (L) if no model or operational data are available, and/or the applicability of an existing model is uncertain or speculative. In this case, there is no existing database, and assessments cannot be made reliably.

The gap analysis knowledge results are also provided in the summary table (Appendix A), which includes comments on the ranking where appropriate. In addition, subsequent sections of this report provide details of the rationale or justification for the panel knowledge level ranking.

Table 2-2. Phenomena or Operation Knowledge Ranking Scale.

Rank	Definition
High (H)	• Phenomenon or operation is well understood. Uncertainty in experimental-operational data is small. • Analysis or performance models have been or could be applied to plant design.
Medium (M)	• Phenomenon or operation is generally understood, but experimental-operational data are limited or uncertain, and additional study may be necessary. • Technical challenges remain.
Low (L)	• Phenomenon or operation is not really understood. • Limited if any experimental or operational data, such that modeling or operational performance, if any, would depend largely on assumptions. • Further studies are essential if phenomenon or operation is important

2.5 Accident Scenario Definition

The specific accident scenario used as a basis for carrying out the gap evaluation is 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 BWR and PWR plant designs under these conditions were evaluated as part of the analysis.

3.0 GAP ANALYSIS RESULTS AND DISCUSSION

The overall objective of this section is to present and discuss knowledge gaps identified by the expert panel in the areas of accident tolerant components and severe accident analysis that are not currently being addressed by industry, NRC, or DOE. The high level PIRT that was developed as part of this process is provided in Appendix A. This table provides importance and knowledge level rankings for each identified gap as evaluated by the panel based on the metrics described in Sections 2.2 to 2.4.

As described in greater detail below, several gaps were identified by the panel in various BDBE topical areas (summarized in Section 2.1) that are not currently being addressed by industry, DOE, or NRC. However, there are two important areas related to BDBEs in which gaps are known to exist, but the panel concluded that efforts currently underway by industry, NRC, DOE and the international community should address the gaps. Specifically, these areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation. For completeness, background in these two areas regarding the known gaps as well as efforts that are underway to address these gaps is provided in Appendix B. These efforts should be monitored to ensure that existing gaps are addressed.

The balance of this section begins by providing a high-level overview of knowledge gaps identified by the panel in the area of accident tolerant components and severe accident analysis. This overview is followed by additional sections that provide technical details for each gap including a brief review of any existing R&D that has already been conducted in the area, as well as the safety relevance of the gap.

3.1 Summary of High Level Gaps and Associated Rankings

Panel deliberations led to the identification of thirteen (13) knowledge gaps on accident tolerant components and severe accident analysis methods that were deemed to be important to reactor safety and are not being currently addressed by industry, NRC, or DOE. The results are summarized in Table 3-1. The thirteen gaps were ranked in terms of their relative importance to safety using a straightforward voting process involving all panel members. The results were then combined to yield the cumulative importance rankings for the panel as a whole; the outcome is also shown in Table 3-1. In broad terms, the gap results could be classified into five categories: i.e., i) in-vessel behavior, ii) ex-vessel behavior, iii) containment – reactor building response, iv) emergency response equipment performance, and v) additional phenomenology.

The first, second, and fourth-ranked gaps are all under the category on in-vessel core melt behavior. In particular, the highest ranked knowledge gap is associated with assembly/core-level degradation; the second is core melt behavior in the lower head; and the fourth is lower head failure. To provide some background as to why these items were so highly ranked, recent complimentary analyses of the Fukushima reactor accidents using the MELCOR [2] and MAAP [3] codes were completed as part of a joint industry-DOE effort. A follow-on DOE-sponsored study on ex-vessel behavior for Unit 1 with the MELTSPREAD and CORQUENCH codes [4] utilized the ex-vessel debris pour conditions from the MELCOR [2] and MAAP [3] calculations as input. As part of this study [4], it was noted that the ex-vessel debris pour conditions between the two codes varied dramatically, and this led to profound differences in the predicted ex-vessel behavior that determines thermal loads on containment, as well as the potential for failing key containment structures such as the Mark I liner. These findings prompted a follow-on jointly sponsored (industry-DOE) cross-walk activity between the two code development teams that focused on identifying key modeling differences between the two codes that lead to large differences in predicted ex-vessel melt release conditions [5]. The results of the crosswalk indicate that the discrepancy begins in the early phases of in-core melt progression; i.e., the different methods used to model assembly blockages, the resultant porosity of these blockages, and how these formations influence the overall progression of in-core melt front propagation. Unfortunately, there are currently insufficient experiment and full-scale prototype (i.e., TMI-2, Fukushima) data that can be used to assess these modeling differences. From a reactor safety viewpoint, this is an important issue as the modeling differences lead to large disparities related to in-vessel hydrogen production.

Table 3-1. Summary of Identified Knowledge Gaps and Associated Rankings for BDBEs.

Category	Identified Gap	Importance Ranking
In-Vessel Behavior	Assembly/core-level degradation	1
	Lower head	2
	Vessel lower head failure	4
Ex-Vessel Behavior	Wet cavity melt relocation and CCI	5
Containment- Reactor Building Response	H_2 stratification and combustion	7
	H_2 and CO monitoring	10
	Organic seal degradation	12
	PAR Performance	13
Emergency response equipment performance	RCIC/AFW equipment	3
	BWR SRVs	6
	Primary side PORVs	11
Additional Phenomenology	Raw Water	8
	Fission product transport and pool scrubbing	9

Closely related to this topic, the fifth ranked gap is under the category on ex-vessel behavior; specifically, melt relocation from the pressure vessel and subsequent core-concrete interaction behavior under wet cavity conditions. From a reactor safety viewpoint, although containment failure by ex-vessel core debris interacting with structural concrete is categorized as a late phase event, the potential radiological consequences could be substantial and warrant effective strategies to prevent or mitigate such a release. As one of several strategies, SAMGs for many operating LWRs include flooding the reactor cavity in the event of an ex-vessel core debris release. New reactor designs also incorporate provisions for cavity flooding as a mitigation feature. Knowledge gaps related to this topic include the effect of pre-existing water on the drywell/pedestal floors on melt stream breakup and spreading, as well as 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. There is international interest in this area, and there is the potential for international collaboration in assessing data from Fukushima and in conducting additional large-scale experiments on this topic.

The third highest ranked gap from Table 3-1 is under the category on emergency response equipment performance under BDBE conditions. Specifically, the RCIC for BWRs and the TDAFW system for PWRs are the key safety systems that are used to remove decay heat from the reactor under a wide range of conditions. Both systems use steam produced by water boiling from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core (BWRs) or steam generators (PWRs) to maintain the needed water inventory for long-term core cooling under a broad range of operating pressures. In many cases, the same turbine/pump is used in both BWRs and PWRs.

Based on events at Fukushima [6], it is known that RCIC operation was critical in minimizing core damage for days (almost three days for Fukushima Unit 2) even though the turbine-pump system ran without DC power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently (based on information available to date) operated in a self-regulating mode supplying water to the core and maintaining core-cooling until it eventually failed at about 72 hours. This observation indicates that there is significant operating 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 US operating fleet. Furthermore, quantifying emergency response equipment performance under these conditions would aid in providing safety margins for current license renewals, subsequent license renewals, as well as assist internationally. Finally, this expanded understanding would inform enhancements to the technical basis for emergency mitigation strategies that could greatly increase options for the successful implementation of FLEX 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 panel acknowledged a need to determine the true operating envelopes of both the RCIC and the TDAFW systems under severe accident conditions.

Two other gaps were identified by the panel in the category on emergency response equipment performance under BDBE conditions, and one of these was ranked in the upper 50 % of all identified gaps in terms of safety relevance. In particular, BWR SRV performance under BDBE conditions was ranked by the panel as the 6th most important gap. In addition, PWR primary system PORV performance was also identified as a gap, but the ranking was lower; i.e., 11^{th} out of 13 in total. In general, data on SRV and PORV performance under DBA conditions is well known. However, the panel identified knowledge gaps 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, for 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. The panel acknowledged a need to determine the true operating envelopes of both BWR SRVs and PWR primary side PORVs under severe accident conditions.

Four additional gaps were identified under the category of containment and reactor building response; i.e., H_2 stratification and combustion was ranked 7^{th}, H_2/CO monitoring was 10^{th}, organic seal degradation was ranked 12^{th}, and Passive Autocatalytic Recombiner (PAR) performance was ranked 13^{th}.

The events at Fukushima [6] clearly illustrated the effect that combustible gas production can have on the course of a severe accident. In particular, due to over-pressurization, combustible gases were able to leak from the containment, accumulate within the plant, and subsequently explode leading to significant damage to the reactor buildings at three of the four affected units. It is thought the main collection point was on the Refueling Floors due to head seal leakage. The panel noted that there are uncertainties on characterizing random ignition sources in plant-level analyses. Identified information needs in this 7^{th} ranked area 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. One unique aspect of the Fukushima accidents, at least for Unit 1 [6], is the likely production of CO_2/CO gases from core-concrete interaction, in addition to H_2. The production of CO is an additional combustible gas source not normally encountered under DBA conditions since the reactor pressure vessel (RPV) is assumed to remain functional. In the Fukushima cases, H_2/H_2O/CO gas mixtures likely resulted, and the data for combustion of CO_2/CO gases from core-concrete interaction are much more limited in comparison to that for typical air/H_2 mixtures. For example, high-temperature auto-ignition data exist for dry air/H_2 mixtures, but similar information is not available for H_2/H_2O/CO mixtures, which are expected for ex-vessel sequences involving core-concrete interaction.

Closely related to this topic in the 10^{th} ranked area, the panel noted a knowledge gap in the area of combustible gas monitoring in containment under BDBE conditions involving ELAP. 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 hydrogen and carbon monoxide without external AC/DC power for an extended period, and account for practical considerations such as non-homogeneous gas mixtures in containment and steam condensation in the gas sample lines.

The 12th ranked area of organic seal degradation under BDBE conditions was also identified as a gap. Typical containments include hundreds of penetrations for piping, instrument and power cabling. Often these seals are made using organic materials. Although seal performance under DBA conditions has been well characterized, there is much less information on the ability of these seals to remain leak-tight under BDBE conditions that can result in elevated temperature, pressure, steam concentrations and radiation effects, particularly for seals that have undergone significant aging.

The final gap identified by the panel under the category of containment and reactor building response is PAR performance for CO/H_2 gas mixtures that can arise under conditions involving core-concrete interaction. Performance data for these devices with H_2/air gas mixtures are readily available, but the panel noted limited knowledge regarding the effectiveness of PARs on reduction of combustible gas levels when high concentrations of aerosol fission products or CO are present. This gap was ranked the lowest (13th) of all those identified due to the fact that PARs are not deployed in any operating US plants[2]. However, PARs are used in the Westinghouse AP1000 plant design being built in the US and are commonly used in other countries, including US-designed plants both operating or under construction. Thus, this gap is relevant for SAMG planning and implementation for those units.

The panel deliberations identified two additional gaps that were classified under the category of additional phenomenology. In particular, the influence of raw water on accident management procedures was ranked as the 8th most important gap, followed by fission product transport and pool scrubbing that was ranked 9th.

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 [6,13]. Seawater was also injected into the SFP for Unit 4. Current US industry guidance [14] calls for the use of seawater or other sources of raw water to provide core cooling should fresh water sources be exhausted or unavailable during the course of accident management procedures. 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 studies being conducted in Japan investigating the thermal-hydraulic characteristics of saline solutions in annular tube geometries as well as small-scale simulated debris configurations [7]. JAEA is also investigating the impact of salt on the chemical and physical form of solidified $(U,Zr)O_2$ debris [8]. Although these

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

studies are providing some preliminary information, the panel concluded that there were still knowledge gaps in this area (ranked 8th) related to the effect of raw water on fission product transport and the coolability of highly degraded core debris; in particular, the potential for precipitate to block coolant passageways and degrade cooling.

Regarding the 9th ranked area of 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 knowledge gaps have been 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 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. The US NRC and the Japan NRA are funding research that may provide insights about issues related to pool scrubbing [15]. In addition, there is the potential to obtain data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France) [9].

The balance of this section provides additional details for each gap including a brief review of any related R&D that has already been carried out, as well as the safety relevance of the gap.

3.2 In-Vessel Behavior

As discussed in the above summary, the first, second, and fourth-highest ranked gaps all fell under the category on modeling of in-vessel core melt behavior. This is principally due to the fact that there are currently large differences between the two US codes for plant-level analyses (i.e., MELCOR [2] and MAAP [3]) in the prediction of core degradation behavior for a scenario similar to Fukushima Unit 1 [5]. However, as the discussion below will illustrate, there are insufficient data at the present time that can be used to assess these modeling differences, particularly for BWRs. From a reactor safety viewpoint, this is an important issue as the modeling differences lead to large disparities in predictions related to the balance of the accident including in-vessel hydrogen production, lower head behavior, and finally ex-vessel behavior that affect thermal loads on containment and long-term debris coolability [4].

3.2.1 Assembly/Core Level Degradation

Background

The cross-walk study [5] identified a number of areas in which MAAP5 and MELCOR have implemented different models of core degradation phenomena inside the RPV. These modeling differences reflect uncertainty that persists in the understanding of severe accident phenomena, principally due to a lack of experiment data that can be used to resolve such differences.

During the early phases of in-core degradation, the two codes have adopted similar modeling approaches and, for a given scenario, produce similar results regarding initial fuel heatup, oxidation, formation and relocation of molten core debris. The debris accumulates in the originally open flow channels, and the rod-like geometry is lost. The primary modeling differences arise when fuel assembly collapse begins. Both codes utilize time-dependent models to determine when collapse occurs, but the models are quite different and lead to differences in the timing of assembly collapse for a common scenario. Additional modeling deviations arise when considering particle bed formation and core-wide melt zone propagation (see Figure 3-1).

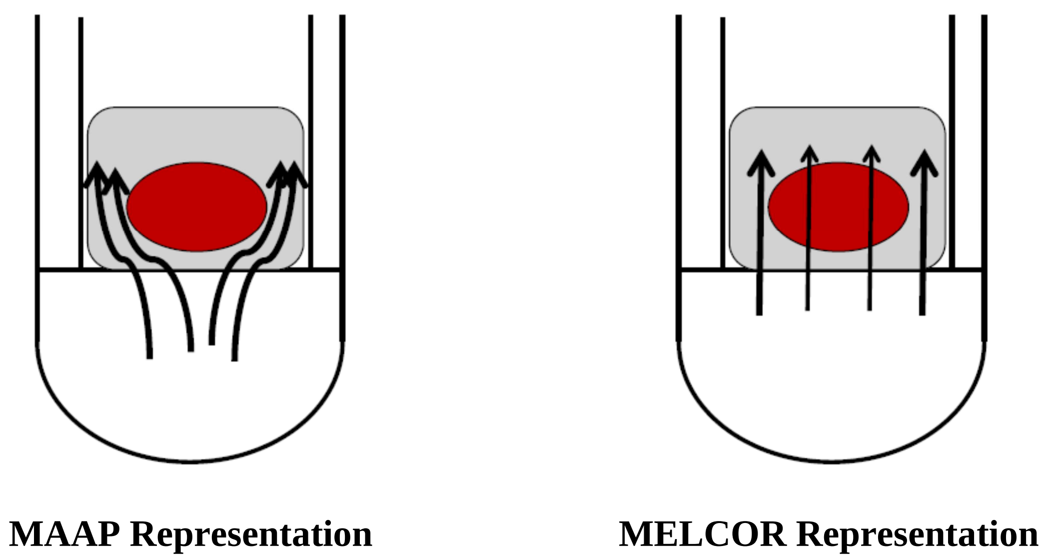


Figure 3-1. Conceptual Differences in MAAP and MELCOR Modeling of Heat Transfer from a Degraded Core [5].

In particular, MAAP models particle beds assuming that they have lower heat transfer surface areas than the rod-like geometry. Moreover, MAAP models predict that porosity of the debris decreases as additional debris is generated, eventually leading to impervious bundle blockages. In this state, the loss of cooling leads to: i) formation of a highly superheated molten zone in the core similar to that formed in TMI-2 [16]; and ii) a reduced amount of in-core hydrogen production as steam flow is vented around molten core material encased within crust, because these formations are treated as impervious to flow. Conversely, MELCOR assumes particulate material that forms in coolant channels remains porous to steam flow and drops (at a fixed velocity specified by a sensitivity coefficient) until it lands on either intact fuel or the lower core plate. Thus, in MELCOR simulations for BWRs, large in-core molten debris zones are not formed; rather, the material steadily drains down through the assembly and then through the core plate. Because steam continues to flow through core debris as it forms, cladding can continue to oxidize, leading to much higher in-core hydrogen generation compared to MAAP simulations of an identical Fukushima-like scenario [5]; see Figure 3-2.

As summarized, in-vessel severe accident analysis results are dominated by models that predict 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. Table 3-2 summarizes experimental data used to develop and validate in-vessel phenomena [17-23]. As indicated, data are primarily from smaller-scale experiments (with the exception of TMI-2 data) which represent localized phenomenon rather than full core response. There are fewer BWR tests (~10) in comparison to PWR tests (~40). Furthermore, all 10 of the BWR experiments were initiated in a dry environment unlike the events at Fukushima. The principal reason why the MELCOR and MAAP predictions of late phase in-core melt progression differ so dramatically for a Fukushima-like scenario [5] is that there are insufficient data at the present time that can be used to assess these modeling differences.

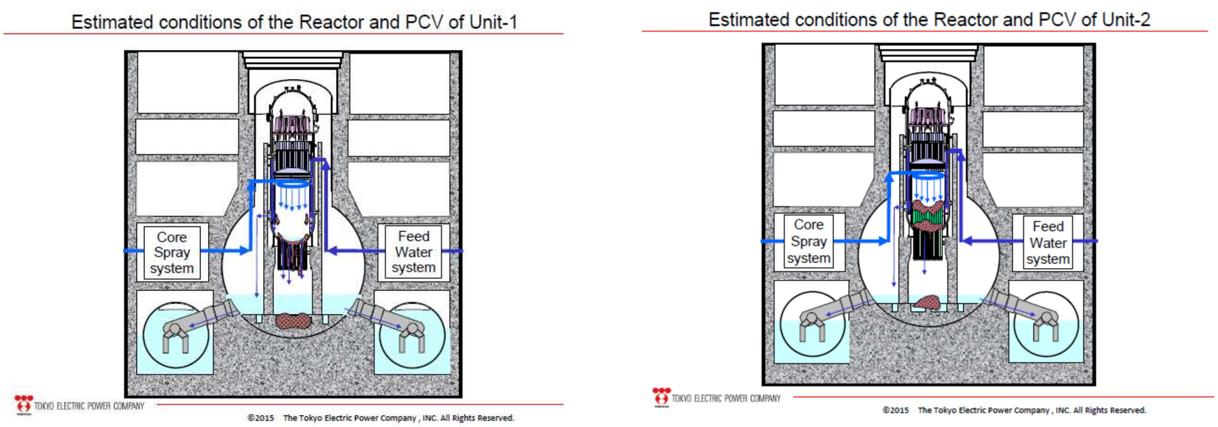


Figure 3-2. Estimated Conditions at Fukushima Units 1-2 (courtesy of TEPCO).

It is noteworthy that post-event analyses of Fukushima Units 1-3 with the MAAP5 code suggest that some of the system depressurization effects can be rationalized by In-Core Instrument (ICI) well failures [5] during the in-core melt progression phase. This subject is being addressed in greater detail as part of the MAAP5 code enhancement project.

Safety Relevance

From a reactor safety viewpoint, uncertainty related to in-core melt progression phenomenology is important as it leads to large variations in the prediction of in-vessel hydrogen production. 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) of lower head failure, and finally 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.

Improved understanding of in-core melt progression would 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, as well as inform response personnel on the best way to allocate resources.

Table 3-2. Assembly /Core Level Degradation Data [17-23].

Test - Accident	Description	Phenomena Tested
PWR		
Loss Of Fluid Test (LOFT)		
FP-2	Large scale fuel bundle severe damage test with reflood	Fuel heatup, cladding oxidation, H_2 generation, quench behavior
Power Burst Facility Severe Fuel Damage (PBF SFD)		
SFD ST, 1-1, 1-4	Small scale and fuel assembly severe damage tests with boil-off and steam flow	Fuel heatup, boil-off cladding oxidation, H_2 generation.
SNL ACRR		
MP & ST series	Small tests with irradiated clad fuel; simulation of the heatup of PWR in-core debris bed	Fission product release from irradiated fuel; debris bed melting
Full-Length, High-Temperature (FLHT)		
FLHT-2, 4, 5	Heatup of full-length PWR fuel assembly; colant boil-off	Boiloff, fuel heatup and damage, H_2 generation, noble gas release
CORA		
CORA-2, 3, 5, 7, 9, 10, 12, 13, 15, 29, 30	Fuel assembly with electrical heater rods, Inconel spacers, Ag-In-Cd absorber	Fuel heatup and damage, cladding oxidation, H_2 generation, reflood and quench
PHEBUS		
B9+, FPT-1 to 4	Fuel assembly and integral severe fuel damage tests: steam generator deposition, containment aerosol/chemistry; melt progression in debris bed geometry with irradiated fuel.	Fuel heatup, liquefaction, collapse, eutectic behavior, H_2 generation, FP release, speciation and volatility, transport and deposition, containment chemistry and deposition, and iodine partitioning; late phase melt progression and low volatility FP release
QUENCH		
QUENCH-1 to 15	Small fuel assembly with electrical heater rods, Ag-In-Cd absorber	Fuel heatup and damage, cladding oxidation, H_2 generation, quenching.
TMI-2 accident	Full scale PWR accident.	System pressure, RCS piping heatup and final state of reactor core. Indirect measurement of H_2 production.
BWR		
Annular Core Research Reactor Damage Fuel Tests (ACRR DF)		
DF-4	Small bundle test that included fuel, channel box and SS control blade with B_4C	Fuel heatup, cladding oxidation, H_2 generation, B_4C-SS eutectic interaction, fuel liquefaction, fuel rod collapse
CORA		
CORA-16, 17, 18, 28, 31, 33	Small electrically-heated fuel assembly with channel walls, with channel walls and B_4C/SS control blade; steam/Ar flow	Fuel heatup, damage, cladding oxidation, H_2 generation, quenching (1 test)
XR		
XR1-1, 2; XR2-1	Fuel assemblies, channel walls and B_4C/SS control blade.	Full scale BWR core cross-section with core-plate structures represented. Response of lower core structures to prototypic relocating liquid materials from upper core.

Knowledge Gaps

The primary knowledge gap during this phase of the accident progression relates to the different methods used to model assembly blockages, the resultant porosity of these blockages, and how these formations influence the overall progression of in-core melt front propagation.

There is currently insufficient experiment data or insights from reactor accidents that can be used to assess these modeling differences.

While the XR2-1 experiment was testing the response of lower core structures in a BWR, the core plate did not fail. Therefore, a knowledge gap exists on how the core plate might fail and how core material (particulate or molten) enters the lower plenum. Furthermore, very limited effort has been spent on the response of the upper internals in a BWR during a severe accident.

Depending upon the extent of core degradation, prototypic full-scale data obtained from Fukushima Daiichi Units 1, 2, and 3 may offer the unique opportunity to reduce modeling uncertainties related to in-core melt progression. This information would also support development of new models for phenomena not currently treated in existing severe accident analysis tools. In addition, this improved modeling will provide valuable insights to Emergency Response personnel tasked with decision making to select the appropriate compensatory actions to limit the impact of the accident progression. The information can be provided in generic owner's group guidance bases.

3.2.2 Lower Head Behavior

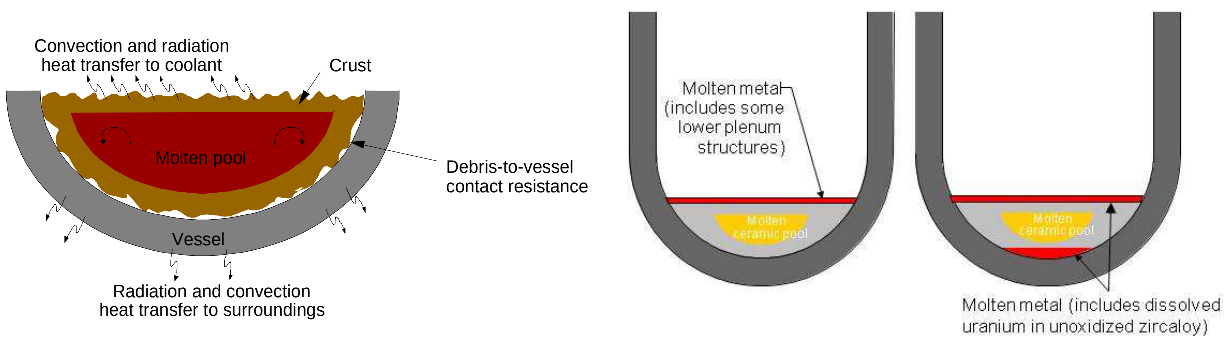
Background

As shown in Table 3-3 [16], limited prototypic material data are available to characterize the behavior of materials relocating to the lower head. TMI-2 post-accident examinations [24] provide the only data that can be used to assess how well different analytical models represent actual core melt progression. Data from prototypic experiments (FARO, RASPLAV, KROTOS, MASCA; [25-33]) that mock up melt relocation and molten pool behavior do not include prototypic structures encountered in actual plants. There are no prototypic data to characterize the effects of raw water addition, which was a significant concern during Fukushima.

Table 3-3. Lower Plenum Melt Relocation and Interaction Data.

Source	Description	Phenomena Tested
TMI-2 Accident and Post-Accident Examinations [24]	Full scale PWR accident.	Melt relocation; melt/water interactions; melt/structure interactions; vessel and penetration heat-up
FARO (Fuel melt And Release Oven) [24,26,27]		
L-5 to L-33	Prototypic materials relocating through water	Melt stream breakup and quench
KROTOS [28-31]		
K-21 to K-58	Prototypic materials poured into a water pool	Melt stream breakup and quench
RASPLAV [32,33]		
AW-200-1 through AW-200-4	Prototypic material test with electrical heating to observe molten corium materials.	Natural convection, stratification in molten pools
MASCA [32,33]		
RCW-1 (RCW); MA-1 through MA-4 (RASPLAV-2)	Prototypic material test with electrical heating to observe stratification, natural convection, and fission product distribution in stratified corium materials	Natural convection, stratification, and fission product distribution in molten pools

While TMI-2 data provide insights needed to extrapolate smaller-scale data to PWR evaluations, TMI-2 data are limited to PWRs and one type of accident. Differences in lower head structures make it difficult to justify extrapolations of TMI-2 PWR data to BWRs. For example, there is more mass associated with BWR internal structures and components and more penetrations within and external to the lower plenum that may act as fins that augment heat transfer from debris in the lower plenum. Likewise, phenomena not observed in TMI-2 post-accident examinations are considered credible in other types of events that can significantly affect severe accident progression. For example, if most relocated core materials contained oxidized zirconium and these materials are allowed to remain in the lower head without quenching, a molten pool may form that is comprised of a light steel/unoxidized zirconium layer above a heavy oxide layer at the bottom (see Figure 3-3). If a significant amount of un-oxidized zirconium is retained in relocated core debris, RASPALV and MASCA test data indicate that this zirconium can reduce some of the uranium dioxide to elemental uranium. This elemental uranium can combine with zirconium and iron to form a separate heavy metal layer on the bottom of the molten pool in the lower head. Although there are no criticality concerns related to this heavy metal layer, it could result in a more concentrated heat source at the bottom central location of the vessel and reduce the thickness of the steel layer above the oxide pool.



Homogeneous Stratified Pools

Figure 3-3. Possible Lower Head Molten Pool Configurations [18].

As for in-core melt progression phenomena, MAAP [34] and MELCOR [35] modeling approaches differ regarding the simulation of core debris relocating into and retained within the lower plenum. These modeling approaches reflect different interpretations of the limited data available for simulating these phenomena. As emphasized in [5], many of these differences result from variations regarding in-core model predictions (e.g., the mass, temperature, physical state, and timing of materials entering the lower plenum), the predicted pathways from the core into the lower plenum (e.g., openings in the core plate, a failure in the core plate, and pathways such as the bypass region between the core shroud or baffle and the core barrel), and estimated thermal hydraulic conditions in the reactor pressure vessel. The manner in which the different models and simplifying assumptions embedded in these codes are invoked can depend on user-modeling choices and the scenario being simulated. As outlined below, there are two primary areas in which differences exist in MAAP and MELCOR representations of lower plenum phenomena.

The first area is how MAAP5 and MELCOR represent molten debris slumping into the lower plenum. MAAP5 assumes that relocating molten debris forms a jet that interacts primarily with lower plenum water. At sufficiently high pour rates, limited interaction occurs. As a result, a substantial amount of the energy in the molten jet is retained. Upon contact with the vessel lower head, pronounced temperature excursions are possible. In MELCOR, core debris slumping to the lower plenum is not modeled as a molten jet or stream; rather, the core debris (molten or particulate) is relocated to the lower plenum when the debris is no longer supported in the active fuel region. Unsupported molten material can flow through the assembly bottom end fittings. The lower core plate will support particulate debris; failure of this plate allows particulate debris to enter the lower plenum. MELCOR does consider hold-up on guide tubes; this feature is not represented in MAAP5.

The second area in which MAAP5 and MELCOR modeling approaches differ relates to how core debris geometry and heat transfer inside the lower plenum is represented. MELCOR models the debris in terms of a set of nodes occupying fixed sub-volumes in the lower plenum (see Figure 3-4). MELCOR considers several types of representations (e.g., conglomerated debris attached to lower plenum structures, particulate debris, an oxide molten pool, an overlying metal molten pool, and lower plenum structures). However, the type of debris in a node is determined based on the characteristics of debris relocating to the lower plenum. For example, a node will only be included in a lower plenum molten pool if molten material relocates into it or if particulate debris in the node melts. In contrast, MAAP5 assumes a lower plenum debris bed (see Figure 3-4) with predefined constituents (e.g., particulate, light metallic, upper oxide crust, molten oxide and lower oxide crust debris). Consistent with observations from the MASCA program, MAAP allows for a fraction of the oxide debris to relocate into a lower heavy metal layer composed primarily of metallic U, Zr, and stainless steel. The volume of each of these debris constituents can vary based on the amount of core material that forms in each region.

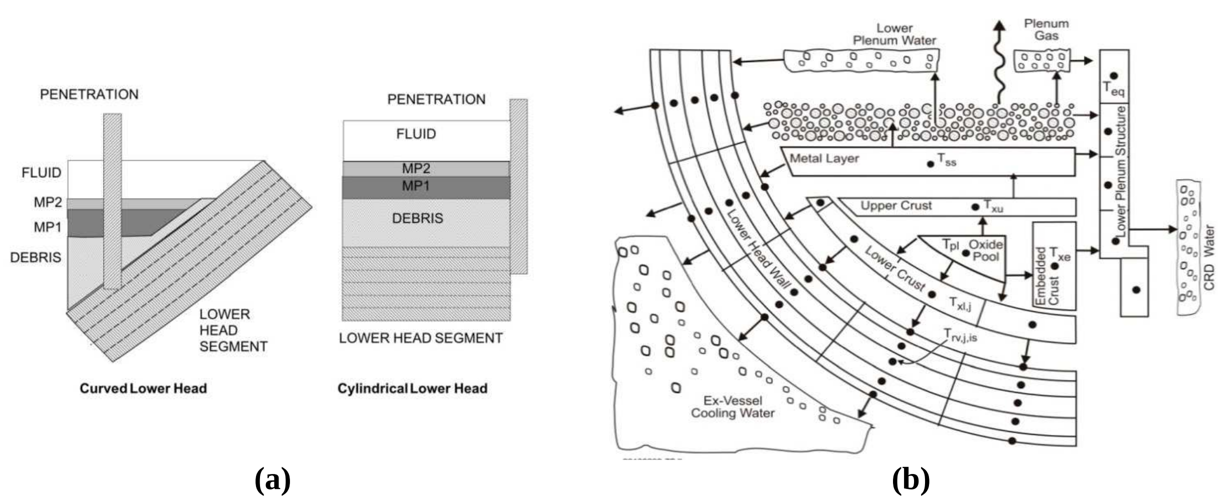


Figure 3-4. Lower Plenum Representations in (a) MELCOR and (b) MAAP [34,35].

If sufficient heat transfer from particulate debris occurs during relocation into the lower plenum, MELCOR may not predict formation of a coherent molten oxide pool. Conversely, MAAP5 assumes that with sufficient accumulation of debris, conduction heat transfer limitations in the oxide layer will result in formation of a molten pool in which decay heat is convectively dissipated to the outer surfaces of the debris. After molten pool formation, MAAP5 predicts that a fraction of the decay heat will be rejected by convection from the pool to the lower head wall; the balance is rejected upward through an oxide crust, the overlying metallic layer, and ultimately radiates to RPV structures [36]. Although MELCOR can simulate convection from molten pools that form in the lower plenum, lower head heatup and molten pool formation are delayed until all water in the lower plenum has boiled away.

Other differences exist between MELCOR and MAAP5 lower plenum heat transfer models. For example, MAAP assumes heat transfer from particulate debris is conduction- and radiation-limited, while heat transfer from the debris crusts is conduction-limited. This limits the fraction of debris that can be maintained as particulate and results in particulate debris and molten pool temperatures much higher than the vessel wall temperature. MELCOR assumes heat transfer from lower plenum debris nodes is not conduction limited, allowing larger amounts of debris decay heat to be transferred to the lower head vessel wall. The treatment of particulate debris also differs in MELCOR and MAAP. MELCOR assumes that particulate debris sinks in molten pools. Conversely, MAAP5 assumes that particulate formed from jet fragmentation by relocation through water in the lower plenum is deposited on top the continuous debris bed.

MAAP and MELCOR lower plenum modeling differences impact predictions for peak vessel temperature distributions, potential failure locations in the vessel lower head, and the conditions (mass, temperature, morphology, flowrate, and composition) of debris exiting the vessel. These differences affect predictions for subsequent accident phenomena such as core concrete interactions and fission product release. In addition, neither code contains models for characterizing the effects of raw water addition.

Safety Relevance

There are limited data for characterizing phenomena associated with materials relocating into and retained within the lower plenum. Hence, there is significant uncertainty associated with predicting both BWR and PWR phenomena on this topic. Such uncertainties significantly impact predictions of the heat load from relocating debris and subsequent accident progression phenomena, such as the time and mode of reactor pressure vessel breach; the mass, temperature, morphology, and composition of debris exiting the vessel; the potential for debris to form a coolable geometry in the containment; and finally, the associated fission product release into the containment.

Reduction of uncertainties related to the behavior of core debris relocating to the lower head can also enhance severe accident management guidance related to locations and rates of water addition to the plant. In addition, an increased understanding of lower head behavior will improve the ability to train operators on accident management procedures, as well as inform response personnel on the best way to allocate water resources.

Knowledge Gaps

As the above discussion indicates, there is a lack of prototypic data for characterizing melt relocation phenomena, such as melt/water interactions, debris coolability, heat transfer from core materials relocating to the lower head, and the effects of raw water addition on these phenomena. This lack of data and the associated increase in uncertainty lead to significant differences in late-phase models in severe accident analysis codes [5]. Such differences significantly impact model predictions of subsequent accident progression phenomena, including ex-vessel behavior. However, at this time, the consensus of the expert panel was that uncertainties in late phase lower plenum phenomena are dominated by uncertainties related to in-core behavior, such as the timing, mass, composition, temperature, morphology, and heat capacitance of relocating core materials.

Prototypic full-scale data obtained from Fukushima Daiichi Units 1, 2, and 3 offers the unique opportunity to resolve many of these modeling uncertainties or supporting development of new models for phenomena not currently treated in existing severe accident analysis tools. Available information suggests that post-accident examinations could provide significant insights into key late phase lower plenum phenomena**.**

3.2.3 Lower Head Failure

Background

As shown in Table 3-4 [37], limited prototypic material data are available to characterize the potential for vessel penetration failure. Information from TMI-2 post-accident examinations [24] provides the only data from a full scale vessel exposed to prototypic core melt. However, the TMI-2 data only represent information from one type of accident and reactor design. Although several tests [38-40] were completed with full scale penetrations, there are questions related to test geometry and the use of simulant melts. The 1/5th -scale tests completed at SNL [41,42] provide the most detailed data for benchmarking vessel creep rupture. However, there are limitations associated with the applicability of this data to BWR vessel geometries, which have significantly different in-vessel and ex-vessel structures that may affect vessel failure. There are no prototypic data to characterize the effects of raw water addition on lower head failure.

Fauske and Associates and the Paul Scherrer Institute (Switzerland) evaluated the potential for vessel penetration failures by completing laboratory tests with full penetrations (a BWR control rod and a BWR drain line) welded to a flat plate at the base of a cylinder. The tests consider attack by iron, alumina, and iron-alumina melt. The Royal Institute of Technology evaluated heat loads and vessel failure phenomena using a calcium metaborate melt. Results indicate that the presence of water significantly impacts melt coolability and the potential for molten material to breach the penetration. However, it is recognized that the use of simulant materials, the absence of decay heat simulation, and the selected test geometry may limit the applicability of these test results.

Table 3-4. Vessel Lower Head and Penetration Failure Data.

Source	Description	Phenomena Tested
TMI-2 Accident and Post-Accident Examinations [24]	Full scale PWR accident.	Melt relocation; melt/water interactions; melt/structure interactions; vessel and penetration heatup
FAI Lower Head Failure Tests [38]		
	Tests for evaluating full-scale instrumentation tube and drain line failure using iron alumina thermite for cases with and with water initially present within penetrations.	Penetration failure
PSI CORVIS Tests [39]		
	Test for evaluating full scale BWR drain line failure using iron alumina thermite in a dry drain line.	Penetration failure
LHF/OLHF (Lower Head Failure/Organization for Economic Development Lower Head Failure) [41,42]		
LHF-1 to 6 OLHF-1	1/5th-scale tests for predicting vessel failure with and without penetrations when subjected to well-defined electric heat load distributions and pressure history.	Vessel and penetration failure
KTH FOREVER and EC-FOREVER [40]		
FOREVER/C1 and C2 EC-FOREVER 1 to 6	1/10th scale tests for evaluating vessel failure with and without penetrations when subjected to a multi-layer molten pool using $CaO + B_2O_3$ with and without water injection	Vessel and penetration failure

The lower head failure (LHF) and OECD LHF [(OLHF)] tests conducted at SNL provide valuable data for benchmarking vessel creep rupture models. This 1/5th-scale test series focused on PWR vessel geometries. Electrical heating was used to create well-characterized distributions of possible heat loads (e.g., uniform, bottom-center peaked, and upper-side peaked). The potential for penetration failure was evaluated by including instrumentation tube penetrations in two tests. Results emphasize the importance of debris heat load, variations in vessel thickness (e.g., variations consistent with manufacturing tolerances), and the influence of accident conditions (e.g., pressure) on the vessel failure area. There are some limitations in the use of data from these tests. For example, test vessel material was fabricated using SA533B1 steel, but subsequent tests revealed that the test vessel material creep properties differed from LWR reactor vessel steel with the same identification [43]. Furthermore, it is difficult to justify extrapolation of data from these one-fifth scale tests (a vessel with an inner diameter 0.91 m and wall thickness of 0.3 cm) to full-scale PWR vessels (typically, ~14 cm thick vessel with 50 to 60, 3-5 cm diameter instrument tube penetrations) or to BWR vessels of significantly different geometries (typically, ~21 cm thick vessel with 55, 5-6 cm diameter instrument tubes, 185 ~12 cm diameter control rod structures, and a 6 cm outer diameter drain line with no internal structure).

Post-accident examinations of TMI-2 instrumentation nozzles and vessel steel provide the only full-scale prototypic data. Video examinations indicate that some nozzles experienced significant damage due to attack from relocated core materials (see Figure 3-5). Examinations of instrumentation nozzles removed from the vessel indicate that melt relocating to the lower head was unable to fail the penetration-to-vessel welds and that melt entering damaged penetrations was unable to cause ex-vessel failure. Examinations of vessel steel samples (see Figure 3-5) indicate that an elliptical region (0.8 x 1.0 m) of the vessel reached a peak temperature of 1100 °C during the accident. Examinations further indicate that steel from this "hot spot" may have remained at peak temperatures for as long as 30 minutes before experiencing fairly rapid cooling (e.g., rates of 10°C to 100 °C /min). Metallurgical examinations of cracks or "tears" in the vessel stainless steel cladding (in samples taken near nozzles located near the hot spot) indicate that the damage extended down to, but not into, the carbon steel RPV. Subsequent evaluations suggest that these cracks were due to differential thermal expansion between the stainless steel and the carbon steel when these materials were subjected to rapid cooling.

Clearly, the TMI-2 data provides insights not possible to obtain from smaller scaled tests. However, it is difficult to justify extrapolation of TMI-2 event- and design-specific information to BWR vessels containing more penetrations with different materials and a drain line without any in-vessel structure. The one possible exception is the hypothesized in-core instrument (ICI) tube failure inferred from the TMI-2 pressure and radiation data [5]. Although data suggest that this in-core instrumentation tube failure may have allowed the primary system to depressurize thereby reducing the potential for subsequent vessel failure, it is not considered a source of vessel lower head failure. Post-event analyses of Fukushima Units 1-3 with the MAAP5 code suggest that some of the system depressurization effects can be rationalized by ICI well failures [5], as well as other potential vessel leakage paths. This subject is being addressed in greater detail as part of the MAAP5 code enhancement project.

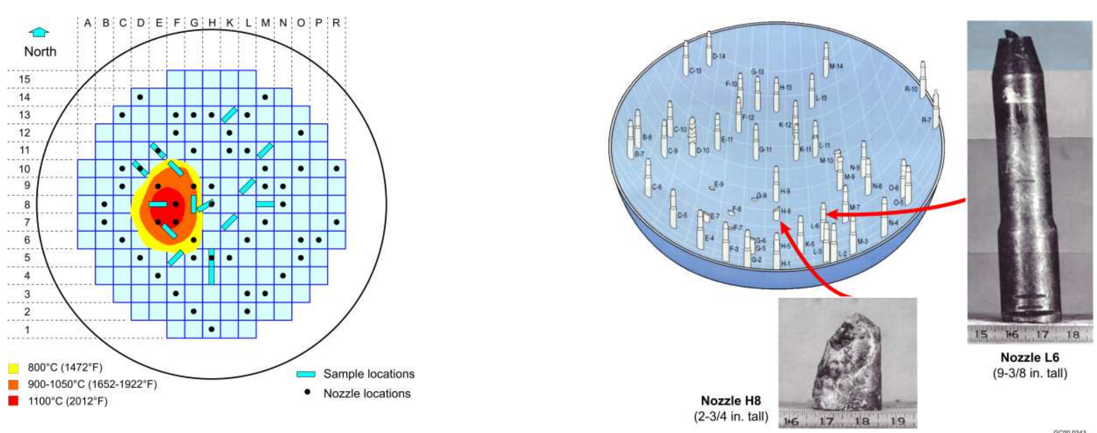


Figure 3-5. Results from OECD TMI-2 Vessel Investigation Program [16].

MAAP [34] and MELCOR [35] modeling approaches differ for predicting vessel failure. As emphasized in [35], modeling differences reflect different interpretations of the limited data available for simulating these phenomena. MAAP5 models the following reactor pressure vessel lower head breach mechanisms:

- Molten material relocation through a penetration leading to ex-vessel penetration wall thermal failure
- Molten debris thermal attack of lower head penetration welds causing penetration ejection
- Lower head wall creep failure
- Thermal ablation of a localized region of the lower head wall by molten debris jet impingement
- Lower head wall thermal erosion due to heat flux focusing from an overlying metallic layer

In contrast, vessel wall creep failure is typically the only vessel breach mechanism used in MELCOR code calculations. In both MAAP and MELCOR, lower head wall creep failure is treated as a localized failure as opposed to extensive creep rupture of the lower head. In a high pressure sequence, MAAP would predict that a localized failure occurs on the side of the vessel expelling only the top debris layer above the failure location and retaining most of the debris within the lower plenum. This localized failure would reduce the pressure inside the vessel and the associated pressure load acting on the vessel lower head. Hence, the debris would continue heating the lower plenum and may eventually lead to an extensive creep failure of the lower head with all the debris relocating into the containment. In MELCOR, assumptions related to debris relocating and heat transfer in the lower plenum often lead to a bottom peaked temperature distribution in the reactor vessel steel. In such cases, a creep rupture is predicted at the bottom center of the vessel, and all molten debris is expelled. The initial size of the hole associated with the predicted creep rupture is less important in MAAP and MELCOR models. Rather, subsequent accident predictions are affected primarily by the resulting vessel depressurization and the amount of mass predicted to be expelled from the vessel.

Safety Relevance

Prototypic data are limited for characterizing the mode and size of vessel failure, either through a breach of the vessel lower head or a 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 vessel 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 debris to form a coolable geometry in the containment; and finally, the associated fission product release into the containment. Improved understanding of the 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).

Additional data to resolve uncertainties in this area could inform accident management strategies and operator training by providing a technical basis for the location and timing of water injection during a severe accident.

Knowledge Gaps

There is a lack of prototypic data for characterizing vessel failure phenomena, especially data that consider BWR-specific in-vessel and ex-vessel structures exposed to prototypic melt attack and the effects of raw water addition. Prototypic full-scale data from Fukushima Daiichi Units 1, 2, and 3 offer the unique opportunity for assessing many modeling issues, or possibly developing new models for phenomena not currently treated in existing severe accident analysis tools. Available information suggests that post-accident examinations could provide insights into the following vessel failure phenomena:

- The mode of vessel failure (e.g., the potential for penetration weld failures, ex-vessel penetration failure, global vessel creep, and localized vessel failure),
- Impact of saltwater addition (deposits on penetrations, corrosion attack on penetrations and welds, etc.),
- The mass, composition, distribution, and morphology of relocated melt (because of its impact on thermal loading to the vessel and penetrations)

The above items are provided as examples of data that could reduce uncertainties about the events at Fukushima and improve our general understanding of severe accident progression.

Nonetheless, the consensus of the expert panel was that current uncertainties in lower head failure phenomena are dominated by uncertainties related to in-core behavior, such as the timing, mass, composition, temperature, morphology, and heat capacitance of relocating core materials and uncertainties related to their behavior in the lower plenum.

3.3 Ex-Vessel Behavior

Background

During a severe accident, if all prevention and mitigation actions fail, the core debris will eventually fail the reactor vessel lower head and relocate into the cavity. Factors that influence the relocation and spreading behavior therein include: i) melt composition, flowrate, and temperature; ii) below-vessel structures that can provide a heat sink to retain material as well as influence the mass source distribution during relocation; iii) cavity geometry characteristics such as the presence of sumps, sump cover plates, doorways, etc.; and finally, iv) the potential for water on the cavity floor that can cause fragmentation and cooling as the material relocates through water, as well as augmenting debris cooling as the material accumulates and spreads. Both of the latter factors are expected to reduce the spreading rate. Spreading behavior is important from two viewpoints: i) the relocating melt can impinge upon and thermally load safety-significant containment structures (e.g., the Mark I BWR containment shell), and ii) the extent of spreading ultimately determines the depth of core material that must be covered with water, quenched, and thermally stabilized in order to effectively terminate the accident sequence. This latter consideration is commonly referred as the 'debris coolability' issue.

Depending upon the debris pour and cavity conditions, molten core-concrete interaction (MCCI) may ensue either during or soon after the spreading phase is completed. If this process continues unabated, then threats to containment may develop that include the potential for basemat penetration by erosion, undermining critical support structures (e.g., reactor pedestal wall in the Mark I), containment over-pressurization from gases generated by concrete decomposition, production of combustible gases by reaction of concrete decomposition gases with metals present in the melt (yielding H_2 and CO), and the possibility of containment bypass for certain designs.

In support of the development of the European Pressurized Reactor (EPR) core catcher concept [44], extensive work has been conducted on melt spreading behavior in Europe. This work includes reactor material spreading tests under dry cavity conditions [45-47]; limited testing was also conducted under wet conditions involving a very shallow water layer [47]. In parallel with the experiments, spreading codes have also been developed [48-51] for application to the EPR as well as other spreading-related issues.

Spreading work has also been conducted in the US. However, these studies have predominately been analytical in nature [52] with an original focus on resolution of the Mark I shell vulnerability issue [53,54]. The MELTSPREAD-1 code for the analysis of transient spreading in containments was developed as part of this work. As part of these efforts, analysis was also conducted to quantify the effects of below-vessel structure on melt relocation for a Mark I BWR [55]. The code was subsequently used to support the safety case for the AP600 design [56], and this work was directly carried over to the AP1000 design. The code was also applied to support EPR licensing with the NRC [57]. Recently, MELTSPREAD was also used to analyze postulated spreading behavior in Fukushima Daiichi Unit 1 [4].

As discussed in [16,58], extensive analytical and experimental investigations have been conducted addressing MCCI issues under both wet and dry cavity conditions. Recently, a series of large scale tests have been conducted at ANL as part of OECD/MCCI programs. One of the principal findings is that cavity erosion shape (i.e., lateral/axial power split) is dependent on concrete type for oxide core materials. Additionally, core debris coolability has been extensively studied as part of the MACE and OECD/MCCI programs involving large reactor material tests ranging in scale up to two metric tons with sustained electrical heating [58]. These tests revealed three physical mechanisms that may provide a pathway for quenching and stabilizing molten core debris that has failed the reactor vessel; i.e., i) water ingression into cracks/fissures that form in the material as it is cooled, ii) melt eruptions due to entrainment by sparging concrete decomposition gases that lead to porous particle beds, and iii) large scale crust breach that allows water to ingress beneath solidified material at the debris/water interface.

Aside from test programs, significant effort on development of code modules for examining MCCI under both wet and dry cavity conditions has occurred. Internationally, the Germans have developed the COSACO [59] and WECHSL [60] codes, while the French have developed TOLBIAC-IBC [61] and MEDICIS [62]. In the US, the primary tools for analysis of MCCI behavior are the CORCON module within MELCOR [35], and the DECOMP module within MAAP [34]. More recently, the CORQUENCH code has been developed [63] with a primary focus on debris coolability modeling; i.e., it incorporates phenomenological models for the previously mentioned water ingression, melt eruption, and crust breach cooling mechanisms identified in the MACE and OECD/MCCI programs. This code was used to scope out a range of conditions under which debris coolability may be achievable for PWR plant conditions [64]. The code was also used to perform debris coolability analyses as part of an enhanced ex-vessel melt progression analysis for Fukushima Unit 1 [4]. Various forms of the coolability models embedded in CORQUENCH are currently being implemented in TOLBIAC-ICB, MEDICIS, MELCOR, and MAAP.

Safety Relevance

By the time the reactor vessel has failed, two of the three barriers for containment of radioactive material (i.e., the fuel cladding and the primary system) have been breached, leaving the reactor containment as the final barrier for prevention/mitigation of fission product release to the environment. Although this is a late phase event, the potential radiological consequences from containment failure by the above described mechanisms (in terms of land and groundwater contamination, as well as latent cancer risk) could be substantial and warrant effective strategies to prevent or mitigate such a release. As one of several strategies, the severe accident management guidance for many LWR plants includes flooding the reactor cavity in the event of an ex-vessel core debris release. New reactor designs also incorporate the provision of flooding the cavity as a mitigation feature.

Specific to the BWR plants, current guidance calls for flooding the drywell to a level of approximately 1.2 m (4 feet) above the drywell floor once vessel breach has been determined. While this action can help to submerge the ex-vessel core debris, it can also result in flooding the wetwell and rendering the wetwell vent path unavailable. An alternate strategy is being developed in the industry guidance [65] for responding to the severe accident capable vent Order, EA-13-109 [66]. The alternate strategy being proposed would attempt to throttle the flooding rate to achieve a stable wetwell water level while preserving the wetwell vent path.

Knowledge Gaps

Regarding melt relocation from the reactor vessel, there have been limited studies to evaluate the amount of material that may be retained on below vessel structures that act as BWR heat sinks [55]. The main conclusion of this work is that the amount of material that could be retained is a relatively small fraction (i.e., <10 %) of the expected overall core debris pour mass under severe accident conditions. This heat sink effect has not been factored into spreading analyses conducted to date. Moreover, the effects of pre-existing water on the containment floor on melt stream breakup and quench during relocation from the reactor vessel have not been quantified. Moreover, there are no existing test data regarding prototypic melt spreading behavior on concrete surfaces with more than a few centimeters of water initially present. This may be an important area as the extent of spreading can impact safety-significant structures, as well as the depth of material to cover with water to provide cooling and fission product scrubbing.

Regarding general MCCI behavior, the results of multiple core oxide tests carried in the OECD/MCCI [64,67] and VULCANO [68] programs indicate that the long-term multi-dimensional concrete ablation behavior is closely linked to concrete type. Although there is a substantial amount of test data, there is currently not a phenomenological explanation for this behavior, and the behavior is repeatable. This is potentially an important area as it can affect the timing of basemat penetration as well as the potential for attack of containment support structures.

The data also indicate that the composition and quantity of noncondensable/combustible gases generated during ablation is a function of concrete type. For instance, the quantity of gas produced during ablation of limestone-common sand (LCS) concrete is considerably more than that produced from siliceous concrete. In addition, the ablation of LCS concrete produces a much higher fraction of CO_2 in comparison to siliceous concrete, which predominately yields H_2O upon decomposition. However, the reasons for these differences are well known and adequate models for predicting this behavior are incorporated into existing severe accident analysis codes [34-35,60-63].

A related issue involves extrapolation of experimental data to reactor scale, particularly with regard to accident progression duration. A recent OECD study showed that the OECD/MCCI data cannot be extrapolated to reactor scale with high confidence for an accident progression duration exceeding 24 hours. Events at Fukushima suggest the need to consider a longer duration accident progression. This, in turn, suggests the possible need for tests of longer duration in comparison to those conducted to date.

Regarding melt composition effects, it is worth mentioning that only one of eight tests conducted in the MCCI test facility involved melt composition representative of BWRs. Moreover, this test did not involve early flooding. Thus, there is a data gap regarding the efficacy of early cavity flooding as a severe accident management strategy for BWRs.

Structural concrete in reactor plants contains a large amount of steel (rebar) that will be incorporated into the melt during MCCI. This metal source will lead to additional non-condensable gas production through oxidation reactions with concrete decomposition gases, and may impact coolability. The MOCKA test program at the Karlsruhe Institute of Technology [69] is being conducted to investigate the interaction of a simulant oxide and metal melt in a stratified configuration interacting with concrete laden with rebar. To allow for a longer-term interaction without the use of an external power supply, additional energy is added to the system by alternating additions of thermite and Zr metal. These tests have shown that the rebar may influence erosion behavior. Thus, it may be worthwhile to investigate the effects of rebar on MCCI under more prototypic conditions.

Related to the investigation by the BWROG into an alternate flooding strategy, there are gaps in our understanding of the impact on throttling water addition rates to preserve the availability of the wetwell venting path. This is preferable as it provides scrubbing of radionuclides prior to release and can avoid the need for an additional drywell vent path. After appropriate modifications are implemented, the MELTSPREAD and CORQUENCH codes could be exercised to study the impact that reduced water addition will have on the overall response of the containment.

In terms of importance to reactor safety, this area was ranked 5th by the expert panel. Research on the effects of cavity flooding on spreading behavior and long term coolability would provide accident management developers and accident management guideline users with supplemental knowledge on factors that can influence accident management.

As discussed, the identified knowledge gaps are as follows:

- The effect of below vessel structure on the arrival conditions of melt onto the containment floor (breakup/hangup).
- The effect of deep water pools on the arrival conditions of melt onto the containment floor (breakup).
- The spreading characteristics of melt in deep water pools.
- Understanding the effect of concrete composition on anisotropic ablation.
- Longer term test data more representative of timescales experienced during the accidents at Fukushima Daiichi.
- The efficacy of early flooding on melt coolability for BWR debris.
- The effects of rebar on the progression of MCCI (ablation, gas generation, coolability).
- The impact of throttle water addition rates with respect to preserving wetwell vent path.

Prototypic full-scale data from any of the Fukushima units for which the reactor vessel has failed (leading to discharge of core material into containment) would provide the unique opportunity to resolve many modeling questions related to ex-vessel behavior. Information that could address questions related to melt spreading behavior includes the total floor area covered by relocating melt; the upper surface elevation profile of the debris; and finally evidence of thermal attack of the containment liner if the melt made contact with this structure. Information that could resolve questions related to core-concrete interaction and debris coolability includes the concrete basemat erosion profile, as well as the surface morphology of the core debris (e.g., porous particle bed versus monolithic crust material). As noted within this section, there is strong international interest in this area, and there is the potential for collaboration in assessing data from Fukushima and in conducting additional large-scale experiments on this topic.

3.4 Emergency Response Equipment Performance Under BDBE Conditions

This section addresses gaps identified by the panel in the area of emergency response equipment performance under BDBE conditions. Specifically, knowledge gaps regarding the performance of BWR RCIC and SRV components under extended BDBE conditions was identified, as well as the performance of TDAFW and primary side PORVs for PWRs. Additional details on the knowledge gaps identified for these safety-relevant components are provided below. As motivation for the gaps that were identified in this section a technical evaluation and discussion regarding emergency response equipment performance during the Fukushima Daiichi accidents is provided in Appendix C.

3.4.1 RCIC/AFW Equipment Performance

Background

The RCIC for BWRs and 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. Both systems use steam produced by water boiling from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core (BWR) and into the steam generators (PWR) to maintain the needed water inventory for long-term core cooling.

For the BWR RCIC system, the steam flow is drawn off directly from the boiling water in the core upstream of the SRVs and the main steam isolation valves (MSIVs), powering the turbine-pump system injecting water from the condensate storage tank (CST) or from the BWR wetwell. For the PWR TDAFW system, steam is drawn from the steam lines upstream of the MSIVs to the turbine-pump with the water source for steam generator injection taken from the CST.

Based on events at Fukushima, [6] it is known that RCIC operation was critical in delaying core damage for days (almost three days for Fukushima Unit 2) even though the turbine-pump system ran without DC power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently operated in a self-regulating mode supplying water to the core and maintaining core-cooling until it eventually failed at about 72 hours.

Based on these observations, DOE [70] is currently supporting efforts, that includes researchers at Texas A&M and at SNL, to develop a thermo-mechanical analytical model of the steam-driven RCIC system operation with mechanistic accounting of liquid water carryover and pump performance degradation. This model is targeted for implementation in system level severe accident codes, such as MELCOR [35] and MAAP [34], to increase the fidelity to which these tools can analyze beyond-design basis events (e.g., BDBEs involving ELAP). Effects of operator actions would also be included. Initially, the Fukushima Unit 2 accident reconstruction will be used as the basis for benchmarking this model. A second key objective of this task is to use insights developed from RCIC model application as a technical basis for developing a RCIC testing program that would obtain data on RCIC operation under ELAP conditions.

Safety Relevance

Except for loss-of-coolant accidents (LOCAs), where the primary system depressurizes down to containment pressure, RCIC and TDAFW are the major long-term heat removal systems employed under a wide range of transients and accidents for the two reactor types. All PRA analyses indicate that the dominant accident sequences that are beyond-design basis events (e.g., BDBEs involving ELAP) would involve RCIC operation for BWRs and TDAFW operation for PWRs. Thus, extended performance of RCIC and TDAFW systems under BDBE conditions is very important to overall plant safety in terms of reducing both the likelihood and the consequences of core damage events involving ELAP.

For PWRs, the TDAFW pump also provides a means to reduce pressure in the RCS thereby reducing any inventory loses and prolonging the time to core damage, particularly for SBO or ELAP events. The importance of the TDAFW pump has increased in recent years with the installation (or planned installation) of low leakage RCP seals in most PWRs. If core damage occurs due to RCS inventory loses, the TDAWF pump also has a high importance in preventing fission product releases from the plant in that it keeps the steam generator (SG) tubes submerged and protects them from high temperature creep rupture failures. For extreme external events, TDAFW can also extend the time at which containment venting might be required.

Implementation of EA-12-049 [65], mitigating strategies for BDBEs, relies on the use of portable systems to provide core cooling (BWR) and secondary side makeup (PWR). A better understanding of the performance of these two systems will directly inform the strategies (i.e., available time) for use of the portable equipment. In particular, any information related to extending the time/conditions under which these systems will continue to operate will provide additional margin to potentially time critical actions related to both core damage prevention and mitigation.

Knowledge Gaps

The preceding discussion indicates that there is significant margin in these emergency core cooling systems that has neither been quantified nor qualified with the US NRC. Technically, this is a highly important lesson-learned from the Fukushima accident that needs to be explored and quantified for the benefit of the US operating fleet. Furthermore, quantifying emergency response equipment performance under BDBE conditions involving ELAP would aid in providing safety margins for current license renewals, subsequent license renewals, as well as assist internationally. Based on data from Daini, this is a longer-term (>15-16 hours) equipment performance issue. Finally, this expanded understanding would form the technical basis for emergency mitigation strategies that could greatly increase options for the successful implementation of FLEX measures under ELAP conditions for both BWR and PWR designs.

This is recognized as a very important area for further research by US industry as well as internationally. The principal objective of R&D in this area would be to reduce knowledge gaps on emergency response equipment performance under BDBE conditions for both BWRs and PWRs; specifically, RCIC and TDAFW systems. In effect, there is a need to determine the actual operating envelope of these components under BDBE conditions in order to expand the time margin before transition to portable systems is needed. In addition, the evaluations should focus on quantifying performance under a range of conditions and defining operational regimes where these pumps will no longer be able to supply core (for RCIC) or steam generator (for TDAFW) cooling. The evaluations should further focus on identifying any potential down sides to extending operation such as development of RCIC leak paths that could drain down the BWR suppression pool.

3.4.2 BWR Safety Relief Valves

Background

Primary system SRVs are the essential components for controlling RPV pressure as a part of accident management procedures for BWRs. In general, SRV performance under DBA conditions is well known. However, the panel identified a knowledge gap on the performance of these components involving extended cycling under high temperature in the process gas flowing through the valve as well as high temperature and pressure conditions expected inside containment during protracted BDBE conditions such as those experienced at Fukushima.

Safety Relevance

SRVs are the essential components for controlling RPV pressure during BWR severe accidents. Thus, data on the reliability of these components under extended BDBE conditions are important for reducing modeling uncertainties related to severe accident progression, as well as supporting accident management planning.

Knowledge Gaps

The principal knowledge gap for these components relates to their reliability (i.e., failure rate as well as failure mode) under high temperature in the process gases flowing through the valve as well as high temperature and pressure conditions expected inside containment during protracted BDBE conditions, such as those experienced at Fukushima. See Appendix C for additional details regarding the expected conditions. The panel acknowledged that if testing infrastructure is identified or developed to evaluate emergency response equipment performance under BDBE conditions (as outlined in Section 3.4.1), then appropriate R&D to address this gap could likely be carried out in the same facility.

3.4.3 PWR Pilot Operated Relief Valves

Background

Primary side PORVs are the essential components for controlling primary system pressure as a part of accident management procedures for PWRs. In general, PORV performance under DBA conditions is well known. However, the panel identified a knowledge gap on the performance of these components involving extended cycling under high temperature in the process gases flowing through the valve.

Safety Relevance

PORVs are the essential components for controlling primary system pressure during a severe accident in PWRs. Thus, data on the reliability of these components under extended BDBE conditions are important for reducing modeling uncertainties related to severe accident progression, as well as supporting accident management planning.

Knowledge Gaps

The primary knowledge gap for these components relates to their reliability (i.e., failure rate as well as failure mode) under high temperature conditions expected for the gases flowing through the valve. In particular, for PWRs radiation heat transfer from the hot gases flowing through the PORV may cause failure of the solenoid (compressed air or DC power) that is used to maintain the PORV in an open position. This solenoid is located just above the tailpipe of the PORV and can be subject to severe heating. The panel acknowledged that if testing infrastructure is identified or developed to evaluate emergency response equipment performance under BDBE conditions (as outlined in Section 3.4.1), then appropriate R&D to address this gap could likely be carried out in the same facility.

3.5 Containment and Reactor Building Response

This section addresses gaps identified by the panel in the area of containment and reactor building response under BDBE conditions. Specifically, knowledge gaps were identified in the areas of H_2 stratification and combustion, containment organic seal degradation, and Passive Autocatalytic Recombiner (PAR) performance for H_2/CO combustible gas mixtures. Additional details on the knowledge gaps identified for these safety-relevant areas and components are provided below.

Other potential issues associated with containment response under BDBA conditions were discussed by the panel. One such issue pertains to the potential for natural convection, mixing and thermal stratification to occur in the large suppression pools of Mark I BWR containments under BDBA conditions. A technical evaluation of the potential for this phenomenon to impact the accident sequence at Fukushima is provided in Appendix C. Stratification could degrade RCIC performance by causing cavitation at the pump inlet. However, EPRI is currently funding research to address this particular question; thus, this knowledge gap is currently being addressed by industry.

3.5.1 H_2 Stratification and Combustion

Background

The explosions at Fukushima [6] clearly illustrated the effect that combustible gas production can have on the course of a severe accident. In particular, due to over-pressurization, combustible gases were able to leak from the containments, accumulate on the refueling floors, and subsequently explode, leading to significant damage to the reactor buildings at three of the four affected units.

Relevant metal oxidation reactions that can occur over the course of the in-vessel and ex-vessel stages of an LWR severe accident are summarized in Table 3-5 [71][3]. For in-vessel conditions, the primary chemical reaction driving combustible gas generation is steam oxidation of Zircaloy (Zr) cladding that yields H_2. As is evident from Table 3-5, this reaction is highly exothermic. Experiments have shown that this reaction begins in earnest above ~1200 °C [73] and can lead to breakaway oxidation [74] in which the fuel heatup rate is substantially increased by chemical heating. The effects of this reaction on accident progression for Fukushima-like conditions as calculated with the MAAP and MELCOR codes are described in [5]. In the event that all cladding is oxidized, then oxidation of metals in structural stainless steel (principally Cr and Fe) can occur, leading to additional H_2 production.

Under ex-vessel conditions, combustible gas production can continue if core-concrete interactions occur due to metals oxidation in the melt by the sparging concrete decomposition gases H_2O and CO_2 [71]. As shown in Table 3-5, the hierarchy of reactions will be the same as for in-vessel conditions, but concurrent metals reduction by CO_2 will yield the additional combustible gas CO. There is also the potential for Si oxidation reactions resulting from SiO_2-Zr reactions that can produce metallic Si in the melt. Finally, depending upon the extent of core-concrete interaction, rebar in the structural concrete can be melted and then oxidized, thus providing an additional source of combustible gas in containment (predominately H_2 from Fe-H_2O reactions).

Table 3-5. In-Vessel (IV) and Ex-Vessel (EV) Metals Oxidation Reactions [71].

Oxidation Sequence	Metal	Oxidation Reactions	Relevance to Accident Phase
1	Zr	$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 6.6\ MJ/kg_{Zr}$ $Zr + 2CO_2 \rightarrow ZrO_2 + 2CO + 6.1\ MJ/kg_{Zr}$	Zr-H_2O: IV and EV Zr-CO_2: EV
2	Si	$Si + 2H_2O \rightarrow SiO_2 + 2H_2 + 15\ MJ/kg_{Si}$ $Si + 2CO_2 \rightarrow SiO_2 + 2CO + 14\ MJ/kg_{Si}$	EV
3	Cr	$2Cr + 3H_2O \rightarrow Cr_2O_3 + 3H_2 + 2.8\ MJ/kg_{Cr}$ $2Cr + 3CO_2 \rightarrow Cr_2O_3 + 3CO + 2.1\ MJ/kg_{Cr}$	Cr-H_2O: IV and EV Cr-CO_2: EV
4	Fe	$Fe + H_2O \rightarrow FeO + H_2 + 0.04\ MJ/kg_{Fe}$ $Fe + CO_2 + 0.4\ MJ/kg_{Fe} \rightarrow FeO + CO$	Fe-H_2O: IV and EV Fe-CO_2: EV

The potential for hydrogen generation during the course of a severe accident is a well-known LWR technical challenge. On this basis, a variety of stand-alone computational tools have been developed to analyze H_2 distribution and combustion in containments, including lumped parameter, computational fluid dynamic (CFD), and hybrid codes. System-level codes, such as MAAP [34] and MELCOR [35], also include distribution and combustion models that utilize a lumped parameter approach that can be used to analyze compartmentalized geometries. In addition, a large number of both small and large scale experimental programs have been

[3]The metals are listed in Table 3-5 in their expected order of oxidation based on the stability of their corresponding oxides as they appear in an Ellingham diagram [72].

conducted to investigate hydrogen distribution, combustion, and detonation thresholds. The reader is referred to recent IAEA technical documents [75-76] for a review of these analytic methods and supporting experiments. Notable facilities that are still operational include PANDA [77] and THAI [78]; these are large scale, multi-compartmental thermal hydraulic test facilities that are providing combustible gas mixing and stratification data for code validation purposes.

Safety Relevance

If deflagrations occur, they can result in direct challenges to containment. Accident management guidance includes strategies to intentionally create deflagrations as well as to steam inert and/or vent containment to prevent deflagrations. Choosing the optimal strategy based on available information is important to maintain long term containment integrity. In addition, for any deflagrations occurring outside containment (i.e., the reactor building), they can potentially damage safety-significant structures and emergency response equipment. Both of these occurrences can inhibit the ability of plant personnel to implement accident management procedures that are required to reestablish or maintain adequate core cooling during the course of an accident.

In the wake of Fukushima, significant efforts have been devoted to the issue. The NRC has issued the severe accident vent Order, EA-13-109 [66], and the industry has responded by providing guidance for complying with this order [65]. Both the PWR and BWR Owners Groups (PWROG and BWROG) have updated their generic SAMGs to reflect lessons learned from the Fukushima accidents. The MAAP5 enhancement project is examining lumped parameter approaches for evaluating hydrogen transport issues in containment.

Knowledge Gaps

Based on events at Fukushima, the panel identified several knowledge gaps in this area. In particular, there are uncertainties on characterizing random ignition sources in plant-level analyses. Specific gaps in this area include:

- flame front propagation in the containment vent line,
- stratification in large physical structures exemplified by containments and reactor buildings,
- methods for modeling combustible gas concentration variations in lumped parameter codes, and
- auto-ignition at high temperatures.

One unique aspect of the Fukushima accidents, at least for Unit 1 [6], is the likely production of CO_2/CO gases from core-concrete interaction, in addition to H_2. As noted, CO production is an additional combustible gas source not normally encountered under DBA conditions. In this case, $H_2/H_2O/CO$ gas mixtures result; the data base under these conditions is available but much more limited in comparison to that for typical air/H_2 mixtures [76]. For example, high-temperature auto-ignition data exist for dry air/H_2 mixtures, but similar information is not available for $H_2/H_2O/CO$ mixtures that are expected for ex-vessel sequences involving core-concrete interaction.

A second issue that has arisen since Fukushima is the potential for flame front propagation in the containment vent line. Here, the issue is that the line will contain an H_2O/H_2 mixture after vent closure; steam condensation will induce suction that will draw air into the line, and this can produce a detonable mixture. This particular scenario has not been examined experimentally.

3.5.2 H_2/CO Monitoring

Background

Closely related to the topic of hydrogen stratification and combustion (See Section 3.5.1), events at Fukushima [6] illustrated the importance of being able to characterize combustible gas concentrations in containment to support decision making related to operator actions for coping with a severe accident. Currently, operating US LWRs do not deploy instrumentation for measuring combustible gas concentrations in containment under BDBE conditions involving ELAP. AC power is required for the analyzer equipment located outside containment as well as for trace heating the lines connecting the analyzer to containment (to prevent steam condensation in the lines) and to open the isolation valves for the connecting lines. In addition, the instrumentation may not detect the presence of carbon monoxide because it operates on the principle of differences in electrical conductivity of monatomic gases (hydrogen) versus diatomic gases (e.g., air and carbon monoxide). The analyzer has a significant lag time for indicating containment conditions that can be crucial in accident management decision making. Finally, the analyzer only samples from a single or limited multiple locations in containment based on an assumption of a homogeneous mixture of gases.

Regarding relevant work in this area, a joint CEA-EdF, Canberra, and AREVA project known as DECA-PF (Diagnosis of a degraded reactor core through Fission Product measurements) [79] is underway to develop and explore the feasibility of such a system that can measure the composition of gases released from the containment. For this project, it is planned for measurements to be made at the outlet of sand beds filters in EdF plants. However, it should be noted that as this report was being finalized, 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 [80]. With this development, the importance ranking for this particular gap may be lower than that originally evaluated by the panel (see Table 3-1).

Safety Relevance

Management of combustible gases during a severe accident is a key LWR technical challenge. Events at Fukushima [6] 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 accident management decision making.

Knowledge Gaps

On-line gas composition measurements are typically made using a device that measures the electrical conductivity of the gases. Composition measurements using gas mass spectroscopy are also available but suffer from significant lag times to provide information in a dynamic environment. The knowledge gap in this area is predominately equipment related; i.e., development of a system that:

- is located inside containment and can survive in-containment environmental conditions for an extended period of time
- provide rapid response of conditions
- can be deployed in multiple locations inside containment
- can function without external AC/DC power for an extended period of time, and account for practical considerations such as condensation in the gas sample line.
- measure overall flammability in the presence of hydrogen and carbon monoxide.

It should be noted that such a device, based on passive autocatalytic recombiner technology, is currently available; but its performance under BDBE conditions has not been reported in the open literature.

3.5.3 Organic Seal Degradation

Background

Organic materials used within nuclear power plants include electrical insulation, elastomeric seals, gaskets, lubricants, coatings and adhesives. Furthermore, typical containments include hundreds of penetrations for piping, instrument and power cabling. Often these seals are made using organic (epoxy) materials. Aging-related degradation of seals (and of elastomeric materials in general) has been the subject of research in the nuclear industry for some time due to the relevance to reactor safety [81]. For instance, seal leakage from recirculation pumps in BWRs has been well characterized and is factored into SAM planning. In general, seal performance has been reasonably characterized under DBA conditions; 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. However, in response to the severe accident capable vent Order EA-13-109 [66], the BWR industry has evaluated [65] available test and engineering evaluation information sources [82-87] to develop containment failure criteria that envelopes the range of expected conditions encountered inside containment under extended BDBA conditions involving ELAP; see Figure 3-6. This analysis includes the effects of penetration degradation on containment leakage.

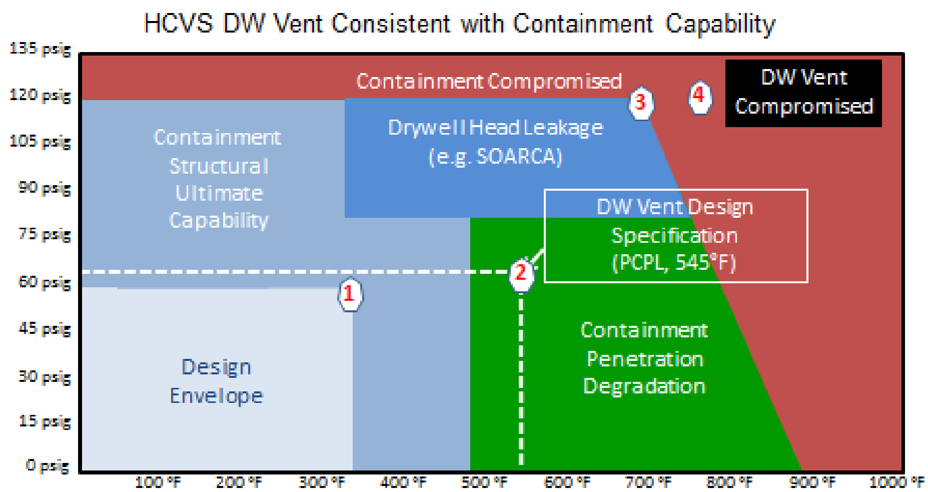


Figure 3-6. Representative Margin of the BWR Containment Based on the Design Envelope [65].

Safety Relevance

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.

Knowledge Gaps

Seal performance under DBA conditions has been characterized as part of the original plant licensing processes. However there is much less information on the ability of these seals to remain leak-tight under BDBE conditions that include simultaneous elevated temperature, pressure, steam concentrations and radiation effects, particularly for seals that have undergone significant aging. Thus, the knowledge gap in this area relates to seal performance under the latter set of conditions.

As noted above, the industry has evaluated [65] available test and engineering evaluation information sources to develop best estimate containment failure criteria under BDBA conditions for BWR systems (see Figure 3-6). However, an analogous assessment has not been performed for PWR containments. This assessment would provide a technical basis for improving SAMGs for PWRs in important areas such as ventilating structures adjoining the primary containment.

3.5.4 Passive Autocatalytic Recombiner Performance

Background

Events at Fukushima [6] reinforced the importance of hydrogen mitigation and control during a severe accident. For reactors in which PARs are deployed, quantifying the performance of these devices over the full range of accident conditions, including BDBE conditions, is key to understanding plant safety performance and supporting accident management planning. The panel identified a gap in this area related to PAR performance under ex-vessel conditions involving ex-vessel core-concrete interaction. In particular, PAR performance with H_2/air gas mixtures has been well-characterized [75], but the effectiveness of these units on reduction of combustible gas levels when CO is present has been much less characterized [88]. This gap was not highly ranked as PARs are not deployed as a severe accident flammable gas control measure in any operating US plants[4]. PARs are included in several of the next generation plants that are under construction in the US. PARs are also commonly used in other countries, including US-designed plants that are operating or under construction. Thus, this gap is relevant for SAMG planning and implementation for those units.

Safety Relevance

Hydrogen mitigation and control is an important element of reactor safety. Thus, for plants in which these devices are deployed, PAR performance under the full range of BDBE conditions, including ex-vessel conditions with core-concrete interaction, is an important element of the overall severe accident mitigation approach.

Knowledge Gaps

The knowledge gap in this area relates to PAR performance under ex-vessel conditions involving ex-vessel core-concrete interaction. In particular, there is only limited PAR performance data regarding the effectiveness of these devices on reduction of 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. Much of the available information is from PAR vendor testing of their units under a range of BDBE conditions.

3.6 Additional Phenomenology

3.6.1 Raw Water Effects

Background

One of the significant accident management actions taken by operators at Fukushima was the injection of large quantities of seawater over an extended period of time in an attempt to cool the damaged reactor cores at Units 1-3 [6,13]. The actual quantity of seawater injected into the cores is uncertain, but the upper limit of the equivalent volume of salt that could have been extracted through boiling is significant; see Table 3-6. For reference, the internal volumes of the RPVs for Units 1-3 are of the order of 300 m^3.

In the US, approximately 20% of the reactor fleet is located near coastal areas. However, most plants are located next to other sources of non-potable "raw" water. Should fresh water sources be exhausted during accident management, seawater or other sources of raw water may be used to provide core cooling [14]. The influence of raw water injection on accident

[4] PARs are installed as DBA hydrogen control measures in a few plants in the US but are not capable of controlling severe accident flammable gas generation rates.

progression has not been extensively investigated. Analyses in the Technical Basis Report (TBR) [14] indicate that the presence of salt could inhibit initial core cooling due to the formation of blockages; however, once the molten debris has relocated into the lower head, debris cooling could in fact improve due to the presence of salt [14]. Other work has been conducted to evaluate the potential for fuel-coolant interactions between molten core material and salt; in particular, elevated pressures are thought to suppress the chances for explosions to occur [89,90].

Table 3-6. Seawater Injection During Fukushima.

Unit	Time of Seawater Injection	Upper Limit on Salt formed by Boiling [6,13]	Likely State at Time of Injection [6,13]	Likely Coolant Behavior
1F1	~28 hr	50-120 m^3	RPV failed; core on the floor	Drained through hole in RPV onto core debris
1F2	~77 hr	160-200 m^3	RPV intact; degraded core	Heat removal in-vessel; leakage out of vessel
1F3	~46 hr	80-200 m^3	RPV intact; degraded core	Heat removal in-vessel; leakage out of vessel

It is currently not clear how seawater affected the stabilization of the Fukushima Daiichi reactors. Current code analyses of the accident sequence with the MAAP [2] and MELCOR [3] codes neglect the influence of raw water impurities on accident progression. A few possible effects are outlined below:

- Water Thermophysical Properties**:** Raw water has different thermophysical properties than that of pure water. Changes in the water thermophysical properties affect heat transfer and water flow rates. Some changes can be beneficial to some phenomena while others are detrimental.
- Fouling Injection Lines and Sump Screens: Raw water may lead to fouling of water injection lines. For example, sprays could plug, or impurities could contribute to fouling of screens or filters on sumps or other equipment.
- Criticality Control: The standby liquid control (SLC) system can inject a boron solution into the RPV as an alternate criticality control measure. Raw water impurities may influence the types of borates formed (with differing solubilities) and therefore affect the efficacy of the SLC system. In contrast, the Cl-35 isotope, a major constituent of seawater and of ~75% natural isotopic abundance, has a moderate thermal neutron capture cross section which may decrease system reactivity [91].
- In-Vessel Core Cooling/Stabilization: Differences in raw water thermophysical properties and the behavior of insoluble impurities and precipitates result in an unclear picture of how raw water would affect core cooling. For example, seawater may initially have a higher critical heat flux, augmenting core coolability. Longer in time, the formation of scale on the fuel rods may inhibit coolant transport through the core, thereby decreasing core coolability.

In a degraded core configuration, if enough precipitate is formed, a unique configuration may occur due to the salts' thermophysical properties, whereby a liquid salt layer may augment in-vessel cooling of the debris.

- <u>Ex-Vessel Core Debris Cooling/Stabilization</u>: As discussed in Section 3.3, crust formation, cracking, and collapse are significant phenomena with respect to core debris coolability. Impurities may impact crack formation and degrade coolability by plugging cracks, inhibiting water ingression. Alternatively, impurities may have a negligible effect if they are swept away by the steam generation. Or, similar to that postulated for in-vessel conditions, a liquid/boiling salt/impurity layer could form that augments debris cooling. The addition of impurities to core debris could also alter the material properties (e.g., adding silicates would impact melt viscosity and spreading characteristics). The breakdown of carbonates could contribute to the generation of CO.
- <u>Fission Product Chemistry and Transport</u>: The primary concern during a severe accident is fission product release. The major objective of severe accident modeling is to predict the timing, type, and amount of radionuclides released from the fuel and eventually into the environment. The presence of raw water impurities could impact currently modeled chemistry and thermodynamic processes that determine fission product release and transport.
- <u>Water Chemistry, Corrosion, Biotic Growth</u>: Corrosion could compromise the integrity of the RPV, piping, containment boundary (liner, penetrations, etc.) and compromise long-term recovery efforts. Growth of algae, etc. could foul water flow paths and compromise long-term accident stabilization. While the long-term water chemistry could be managed, more moderate-term phenomena such as stress-corrosion cracking or accelerated corrosion due to galvanic couples may warrant consideration.

To date, there has been a limited amount of work investigating the effects of raw water injection on accident management. The EPRI TBR [14] provides an analysis that illustrates the potential impact that seawater could have on debris cooling once significant core material was relocated either to the lower regions of the core or to the lower head. This study indicates that sodium chloride (salt) enhances coolability of core debris once relocation has occurred. The only currently active domestic R&D is being funded by the NRC. The objective of this study is to develop a stand-alone chemistry model that accounts for a number of the raw water chemistry effects in containment [92]. This model is intended to capture key thermochemical effects including radiolysis of iodine species, gas solubility, precipitation, and corrosion. Based on evaluation results, the model may ultimately be integrated into system level codes, such as MELCOR.

In terms of international research in this area, the TBR [14] references water experiments performed by Tuunanen et al. [93] in which precipitated boric acid crystals caused blockage of coolant channels. JAEA is investigating the thermal-hydraulic characteristics of saline solutions in annular tube geometry and is also conducting small-scale cooling tests with simulated core debris (i.e., small debris beds composed of packed beads) [7]. JAEA is also investigating the impacts of salt on the chemical and physical form of solidified $(U,Zr)O_2$ [8]. The NRA is also investigating the thermal-hydraulic characteristics of seawater [94]. As part of this work, they are taking physical property measurements and are conducting short and full length fuel bundle heat transfer and fouling tests. They are also planning experiments to investigate dryout of debris beds under seawater, with a supporting analysis task to evaluate the effectiveness of seawater injection.

Safety Relevance

The top priority during a severe accident is to reestablish and maintain core cooling. Should fresh water sources be exhausted as accident management procedures are being conducted, seawater or other sources of raw water may be used instead [14]. Thus, as with all aspects of accident mitigation, knowledge of the potential impact of deposited material contained in raw water on both short- and long-term core cooling behavior and associated corrosion issues would be useful as supplemental information to support accident management planning.

Knowledge Gaps

Potential impacts associated with the use of raw water to reestablish/maintain core cooling were brought into focus by events at Fukushima. A number of these potential impacts were outlined above, but these are speculative. Thus, the first knowledge gap in this area is the identification of important phenomena associated with raw water addition that affects accident progression, stabilization, and consequences of a severe accident. This gap would best be addressed through scoping studies and potentially bench top experiments that would provide guidance on key phenomenology.

A second knowledge gap at the current time is the lack of an ability to assess the effects of raw water injection on overall accident progression. Questions of this type are usually addressed with system level codes such as MAAP and MELCOR; but as described earlier, these codes currently do not have models that account for the effects of water impurities on accident progression. As basic information becomes available from the scoping studies, this knowledge gap would be addressed by upgrading system level codes to incorporate these findings, and then applying the codes to postulated accident sequences to scope out potential consequences important to core debris cooling and fission product release.

Depending upon findings from the scoping and plant level studies described above, additional research may be warranted to reduce phenomenological uncertainties and/or develop new models that better reflect physical reality.

3.6.2 Fission Product Transport

Background

Accurate modeling of fission product transport phenomena is essential for accurate source term estimates. Significant improvement has been made in our understanding and ability to predict fission product release and transport in the RCS and containment since the TMI-2 accident [24]. These enhanced capabilities have demonstrated that radionuclide retention in the RCS, suppression pools, and containment can significantly affect the potential release of radioactivity from a nuclear power plant during an accident.

During a severe accident, it is possible for fission products, such as elemental cesium and iodine, to be released from the fuel as condensable vapors (see Figure 3-7). Were these vapors to remain gaseous and not react or condense on surfaces within the RCS, it is possible that these elements, along with noble gases such as xenon and krypton, could become a part of the released source term. However, conditions in the RCS, suppression pools and containment are expected to be quite different than in the core. Temperatures are expected to be lower, and oxygen potentials may be higher. These different conditions make it more possible for vapors to form aerosols or deposit on surfaces or on aerosols. This deposition would reduce the source term, but it may only be a temporary reduction within the RCS. Continued heating of such surfaces by natural circulation within the reactor vessel or by decay heat generated within these deposits may cause volatile species to re-vaporize.

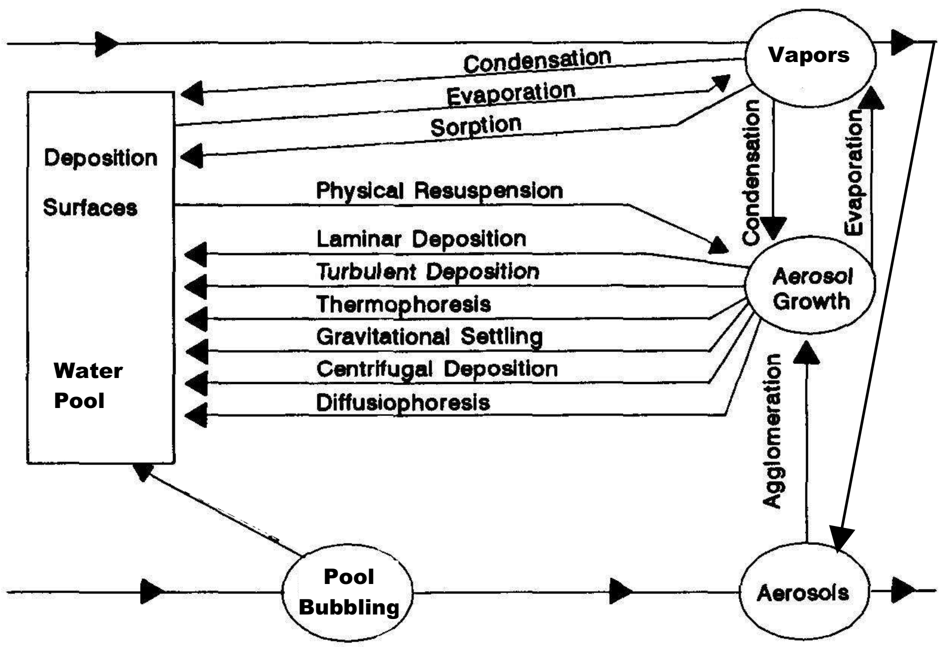


Figure 3-7. Processes Governing In-Vessel Fission Product Transport [95].

The chemical form and concentration of fission product vapors and other RCS conditions determine the point at which condensation can occur to form aerosols. The resulting aerosol chemical species, size distribution, and the aerosol shape affect fission product aerosol processes identified in Figure 3-7. Processes, such as deposition by other fission product vapors and aerosol coagulation or agglomeration, affect aerosol size. Possible aerosol deposition processes include gravitational settling or sedimentation; turbulent, laminar, or centrifugal deposition;

thermophoresis,[5] and diffusiophoresis.[6] However, deposited aerosols may be re-suspended by sudden increases in flow or vibrations associated with a rupture or depressurization of the RCS or containment / suppression pool venting. Decay heat in deposits may raise temperatures to the point that volatile radionuclides vaporize. Collapse of molten core debris into water in the lower plenum of the reactor vessel, penetration of the reactor vessel by core debris, or even operation of SRVs could produce sufficient changes in the turbulent flows near the surface to lift particles off structural surfaces. Additional conditions affecting fission product transport within the RCS are the production of structural aerosols, the temperatures of gas and structures within the RCS, and the presence of engineered safety features such as core sprays. Other engineered features such as containment spray and fan coolers can also affect containment fission product transport.

Regarding data sources, the extremely low radionuclide releases measured during TMI-2 led to significant international efforts to reassess the technical bases for estimating severe accident source terms. It became evident from these assessments that there was a need to better quantify phenomena associated with primary system fission product release and transport, including fission product vapor/aerosol chemistry and vapor/aerosol retention. To address this need, numerous experiments were completed. These experiments range from small-scale, separate-effects tests designed to provide input data to the modeling codes to large-scale, integral tests with prototypic materials to evaluate the coupling between individual models and provide data for validating integral codes [96,106]. Table 3-7 lists several of the larger sources of data. Although TMI-2 provides data from a full scale reactor accident, these data are from one reactor design and one accident scenario.

References [95,106] indicate there are still some gaps in available data for estimating fission product transport. For example, additional data are needed to characterize the thermodynamics of fission product vapor species. Experimental studies do not adequately address regimes where there are simultaneously high temperatures and high partial pressures of steam and hydrogen. High concentrations of steam may stabilize vapor phase hydroxides and hydrates that have been undetectable at the lower steam concentrations in existing experimental studies. Similarly, high partial pressures of hydrogen may create important hydrides not included in available data bases. References [95,106] also note that there are insufficient data to characterize the effects of radiation ionizing gas within the RCS and to characterize vapor interactions with aerosols and with surfaces. In addition, there are no data for evaluating the chemistry effects of raw water addition on fission product transport.

Regarding current regulatory and industry modeling approaches, there are numerous stand-alone codes for predicting fission product vapor and aerosol transport in the RCS [95]. In addition, system codes such as MAAP [34] and MELCOR [35] model severe accident phenomena inside the RCS in an integrated way.

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[5] A Brownian process causing migration of particles toward lower temperatures.
[6] Deposition induced by condensation of water vapor onto structural surfaces.

Table 3-7. Large Scale Fission Product Transport Data Sources.

Source	Description	Phenomena Tested
TMI-2 [95] (USA)	Full scale PWR accident.	Residual fission product levels in ceramic melt indicating that some (2-5% I and 10-15% Cs) remain. AgI deposits found in upper plenum of reactor.
Marvekin [96] (Studsvik, Sweden)	Full-scale PWR primary circuit (including a reactor vessel, simulated internals, pressurizer, relief tank, and filter) with simulant non-radioactive materials.	Transport and deposition behavior of fission products (CsI, CsOH, Te) and structural material aerosols (Ag, Mn) in the primary circuit for code validation.
Loss of Fluid Test (LOFT) [97,98] LP-FP1 and LP-FP-FP2 (INL, USA)	Full-scale fuel bundle severe damage tests conducted in large 1/50th scale PWR facility with two loops and a pressurizer.	Fission product transport tests to investigate fission product transport in just-beyond design basis and severe accident situations
Power Burst Facility Severe Fuel Damage (PBF SFD) [99] (INL, USA)	Large scale PWR fuel bundle severe damage tests with small, continuous flow of coolant.	Fission product release and timing of fuel degradation. Final test included effects of control rod alloy on fission product behavior
LWR Aerosol Containment Experiments (LACE) [100] (HEDL, USA)	LWR containment with a simulant aerosol mixture (soluble CsOH and insoluble MgO) along a complex test pipe 63 mm in diameter, 27 m long with 5-90° bends, 4 horizontal sections, and 2 vertical sections.	Aerosol containment studies with two containment bypass experiments to assess attenuation in the primary circuit pipework and release to the auxiliary building.
PHÉBUS-FPT [101] (CEA, France)	Series of five 1/5000th scale in-pile integral experiments with prototypic fuel.	Tests to measure fission product release and transport through a model of reactor coolant system, and aerosol behavior in model containment.

MELCOR models fission product transport by incorporating modified versions of models used in stand-alone more detailed fission product transport codes, such as MAEROS [102], TRAPMELT-2 [103], SPARC-90 [104], and HECTR [105]. In contrast, MAAP includes an FAI-developed method for modeling fission product transport. Table 3-8 identifies the fission product transport phenomena modeled in these two codes. In order to reduce computational time, both codes have omitted selected phenomena that evaluations indicated were less important. Both codes have an established validation program to ensure that the most important fission product transport processes are simulated.

Boundary conditions for MELCOR fission product transport models are obtained from other MELCOR models that predict the control volume thermal-hydraulics and heat structure temperatures. MELCOR models aerosol agglomeration and deposition processes using techniques based on the MAEROS multi-component aerosol dynamics code. Vapor condensation onto and evaporation from aerosol particles are handled separately using equations from TRAPMELT 2 incorporated into MELCOR. This reduces computational time and ensures consistency with heat transfer and thermal hydraulic predictions by other MELCOR models.

MELCOR fission product transport models were developed in a manner that allows updates to consider additional phenomena. For example, multi-component aerosol features could be activated in MELCOR, but the single component option is the current default based on available experimental data. In addition, there is currently no user input controls available to allow aerosols deposited on the various surfaces to be re-suspended. The decay heat from fission products suspended in the RCS may be transported to the gas phase in any volume or to any surface. Chemistry effects are simulated in MELCOR through user-provided mass and energy transfer data for each stoichiometric reaction. Reversible and irreversible reactions can be used to model adsorption, chemisorption, and chemical reactions. Only fission product vapors undergo chemical reactions.

Table 3-8. Fission Product Transport Processes Considered in MELCOR and MAAP.

Process	Considered in MELCOR	Considered in MAAP5
Vapor processes		
Chemisorption	x	
Vapor condensation /evaporation	x	x
Revaporization	x	x
Aerosol Nucleation	x	x
Aerosol processes		
Vapor deposition on aerosols	x	
Growth by agglomeration, coagulation, and condensation	x	x[7]
Gravitational settling (sedimentation) with agglomeration and hygroscopic effects	x	x
Thermophoresis	x	x
Inertial impaction (e.g., turbulent, laminar, or centrifugal)	x	x[8]
Diffusiophoresis	x	
Leakage or transport between control volumes by bulk fluid flows	x	x
Resuspension		x
Removal by containment sprays, filter trapping, pool scrubbing, etc.	x	x
Chemistry Effects - 'Class transfers' of fission product vapors	x	

In the MAAP code, fission products may exist in up to four states: vapor, aerosol, deposited, and contained within the core. MAAP simulates aerosol and vapor removal rates from the gas phase to surfaces or the re-vaporization rates of deposited material. However, MAAP incorporates an innovative single-component aerosol transport equation model to simplify computational requirements. The calculation procedure for aerosols, with particles growing by agglomeration, uses two correlations for the solutions of the integro-differential equation governing aerosol particle size distribution. Correlations are used for "aging" or settling aerosols and for "new" source-reinforced aerosols. Removal rates are calculated for various phenomena, such as sedimentation, inertial impaction, steam-driven diffusiophoresis, thermophoresis, and particle removal by leakage. When more than one mechanism for aerosol settling is operative, a "combining law" is used to represent the combined effects of the two major mechanisms. Turbulent deposition has not been incorporated in MAAP because analyses suggest that turbulent deposition is less important as other deposition processes. Particle growth is primarily assumed to occur via condensation because FAI evaluations indicate that it occurs much sooner than coagulation.

[7] MAAP only considers condensation, which is considered to be dominant.
[8] Turbulent deposition is not included in MAAP because evaluations indicated it was less important than other mechanisms and it would be computationally intensive.

Safety Relevance

Prevention of fission product releases to the environment is the key goal of nuclear reactor safety. Thus, the ability to characterize fission product release and transport during a severe accident is very important for reactor safety evaluations. On this basis, R&D in this area has been heavily pursued both within the US and internationally.

Knowledge Gaps

In general, adequate data exist for understanding and modeling most fission product transport phenomena that affect source term estimates. Evaluations have identified selected data needs, such as data to characterize: 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 RCS; vapor interactions with aerosols and surfaces; and pool scrubbing efficiency at saturated conditions and elevated pressure. In addition, as discussed in Section 3.6.1, there are no data for evaluating the chemistry effects of raw water addition on fission product transport. On-going analytical research funded by the US NRC [92] may provide some insights on this issue. 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 saturated conditions and elevated pressure. The Japan NRA is funding a series of small and large scale tests that may address this data need [107]. In addition, there is the potential to obtain data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France) [9].

4.0 CONCLUSIONS AND RECOMMENDATIONS

This section summarizes the thirteen safety-relevant knowledge gaps 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 4-1. During the panel deliberations, recommendations on appropriate R&D to address these gaps were also developed; these recommendations are also provided in the table and related discussion.

It is noteworthy that two important areas related to BDBAs were identified by the panel in which gaps are known to exist, but it was concluded that efforts currently underway by industry, NRC, DOE, and the international community were adequate to address the gaps. These areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation. For completeness, these two areas are reviewed in Appendix B.

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.

4.1 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 key 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 severe accident progression codes.

From a reactor safety viewpoint, uncertainty related to in-core melt progression is important as it leads to large variations in the prediction of key outcomes; e.g., in-vessel hydrogen production. 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) of lower head failure, and subsequently 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 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

Table 4-1. 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 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 a 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 the facility and conduct the 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, the latter should be higher priority.

phenomena could improve the ability to train operators on accident management procedures and inform emergency response personnel on the best way to allocate resources.

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. Based on the recent MAAP-MELCOR cross walk activity [5], the principal phenomenological uncertainty regarding in-core behavior is the extent that core debris is permeable to gas flow during degradation; i.e., 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. Note that the full-scale empirical observations and data from TMI-2 are limited to one particular accident scenario. Because of the potential overall benefits to reactor safety, the panel recommends that an integrated Fukushima examination 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 applies to 11 of the 13 gaps identified in Table 4-1.

Although Fukushima data generally offer the best option to aid in 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 specific data needs identified in the crosswalk activity [5]. 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 Table 4-2.

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 [5], reexamine previously conducted tests related to in-core melt progression (documented in Section 3.2.1) to determine if additional insights can be obtained to help reduce the gaps. The panel believed that the chance of gaining additional insights from the existing tests was relatively small, but the amount of effort to complete this effort would be modest and thus worthy of consideration.
- Conduct more detailed discussions, analysis and model development activities between the MAAP and MELCOR development teams to try to create common understanding between the two codes given the current knowledge base in this area. This activity could reduce uncertainties in severe accident key parameters; e.g., hydrogen production. This is a relevant activity for both BWRs and PWRs since differences in code predictions affect the timing of action management functions like containment venting, operator training on accident management, and prioritization of operator actions.

- Consider developing computational tools to inform evaluators and decision makers in the development of accident management guidelines and supporting accident response. Specifically, data processing systems could be used to 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. This activity must consider current uncertainties in core degradation and relocation models, as well as uncertainty in instrumentation indications, to ensure that appropriate decisions are made.

Table 4-2. Design Goals for Experiments to Address Knowledge Gaps Related to Late Phase In-Core Melt Progression.

Experiment Element	Design Objective	Notes/Rationale
Fuel material	UO_2	Experiments targeted at examining morphology-porosity of blockages; thus, prototypic materials needed since porosity/crack formation is strongly dependent on material properties and temperature
Cladding material	Zircaloy	
Decay heat simulation	Incorporate in test design	Needed to simulate long-term relocation behavior
Decay heat level	BWR ~ 3 hours after scram	Typical core uncovery time for an unmitigated accident in a BWR is ~ 3 hours
Composition of material relocating in the assembly	$(U,Zr)O_{2-x}$ plus control material with possible mockup of channel boxes	Experiments targeted at examining late phase assembly degradation including fuel relocation and assembly blockage formation
Number of Pins	Sufficient to resolve sub-channel behavior	Radial heat losses need to be quantified and accounted for in the experiment design
Assembly length	Sufficient to resolve axial melt relocation behavior and blockage formation	May be possible to infer from MAAP-MELCOR analyses; expected length is $\simeq$ 1 meter
Sub-channel coolant flow conditions	Steam	Principle difference between MELCOR & MAAP is that MELCOR permits steam flow through blockages, whereas MAAP assumes that blockages are impervious
Channel reflood	Incorporate in test design	Desirable, but not primary experiment objective
Principal data needs:	Time-dependent in-assembly melt relocation behavior	Possibly use X-ray tomography
	Time-dependent channel blockage formation-porosity	Possibly use specified steam driving pressure; measure assembly outlet flowrate to ascertain the extent of blockage formation; assess debris porosity as part of Post Test Examinations
	Fuel-cladding channel outlet gas temperatures	Possibly use small thermocouple rakes or ultrasonic thermometers to measure cladding and fuel temperatures
	Cladding oxidation rate-chemical heat production	Possibly measure off gas composition and infer flowrate using gas mass spectroscopy

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) 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 could also 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 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 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 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. Again, it is noted that uncertainties in lower head behavior phenomena are overshadowed by uncertainties related to in-core behavior, and how those uncertainties impact melt arrival conditions in the lower head.

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 plant geometrical configurations 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 or reactor cavity 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 an 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 global 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.

4.2 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 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. There is international interest in this area, and there is the potential for collaboration in assessing data from Fukushima and in conducting additional large-scale experiments on this topic.

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.

4.3 Emergency Response Equipment Performance under BDBA Conditions

The ***3rd ranked gap relates to emergency response equipment performance under BDBE conditions***. Specifically, the Reactor Core Isolation Cooling (RCIC) system for BWRs and the Turbine Driven Auxiliary Feedwater (TDAFW) system for PWRs are the key safety systems that are used to remove decay heat from the reactor primary system under a wide-range of conditions, from operational pressures down to lower pressures approaching cold shutdown conditions. Based on events at Fukushima [6], it is known that RCIC operation was critical in delaying core damage for days (almost three days for Fukushima Unit 2). This observation empirically 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 providing more flexibility in emergency mitigation strategies and could greatly increase options for the successful implementation of FLEX [NEI 12-06] (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 in this area 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 carry out 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) carry out the testing necessary to determine the actual operating envelopes for RCIC and TDAFW systems under BDBE conditions. There is international interest in this area, and there is the potential for collaboration in assessing data from Fukushima and in conducting an experimental program on this topic.

Two other gaps were identified by the panel in the category on emergency response equipment performance. In particular, the ***6th ranked gap relates to BWR SRV performance under BDBE conditions,*** while the ***11th ranked gap relates to PWR primary system PORV performance***. In general, data on SRV and PORV performance under 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 AFW 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.

4.4 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 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 [66], and the industry has responded by providing guidance for complying with this order [65]. Both the PWR and BWR Owners Groups have updated their generic SAMGs to reflect lessons learned from the Fukushima accidents. 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 [6] 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 [79]. Thus, the panel recommended that industry leverage these efforts as a basis for developing a H_2-CO containment monitoring system. However, it should be noted that as this report was being finalized, 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 [80]. 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 4-1).

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 [81]. 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 [66], the BWR industry has evaluated [65] available test and engineering evaluation information sources [82-87] 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 group. 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 on reduction of 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 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[9]. However, PARs are used in the Westinghouse AP1000 plants being built in the US and are commonly used in other countries, including 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 [76] available regarding the effectiveness of these devices on reduction of 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.

4.5 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 [6,13]. Current US industry guidance [14] calls for the use of seawater or other sources of raw water

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[9] 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.

(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 steam generator heat transfer surfaces when raw water is used as a feed source for SG 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 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 [92]. In Japan, 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 [107]. 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) [9].

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.

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Appendix A: EVALUATION RANKING TABLE

Phenomenon	Knowledge State: BWR						Knowledge State: PWR						
	Phenomenological understanding (modeling)	Importance	Code Adequacy	Available Validation Data	Feasibility of Getting New Validation Data	Comments	Phenomenological understanding (modeling)	Importance	Code Adequacy	Available Validation Data	Feasibility of Getting New Validation Data	Comments	Idenfitifed Gap
Early Phase In-Vessel Melt Prog.													
Pin-level degradation	high	medium	high	high	on-going	Some data available (DF , CORA, XR, etc.); JAEA has proposed additional tests.	high	medium	high	high	not needed	Several sources of PWR pin, bundle, and fuel assembly data available (CORA, PBF, ST, FLHT, LOFT, and PHEBUS)	None
Assembly/core-level degradation	low	high	low	low	limited	Limited data (XR1 and XR2); some engineering judgement to extrapolate to BWR conditions; Daiichi inspections should offer insights. Currently, codes homogenize plan view fine scale herotgenities; no data to determine if this approach is reasonable or not.	medium	high	medium	medium	not likely nor needed	Some sources of data available (TMI-2, MP, and PHEBUS)	The phenomenology of late-phase in-core melt progression involving multi-assemblies (as modeled, core rings) is a knowledge gap that was reconfirmed in the MAAP-MELCOR cross-walk for open-core BWR geometries.
In-Vessel Melt Progression: Late Phase (Lower head)	medium	high	low	low	possible	Limited data (XR, FARO, MASCA, RASPLAV); inspections at Daiichi needed to resolve uncertainties with respect to debris coolability, hydrogen generation, relocated endstate	medium	high	medium	medium	limited	Some data available (TMI-2,FARO, KROTOS, TROI, RASPLAV, MASCA, JAEA, SULTAN, ACOPO, and KAERI, etc.); inspections at Daiichi could reduce some uncertainties	Timing and mechanisms of melt relocation to the lower head affects the distribution and morphology of the material therein, as well the thermal transient therein. Experiments investigating melt interactions below the core support plate in a BWR are very limited
Vessel Failure	low	high	low	limited	possible	For BWRs the instrument tubes, CRDs, and drain line are potential failure locations. Data limited to CORVIS, FAI, and SNL tests; Daiichi inspections needed to reduce uncertainties.	medium	high	medium	some	limited		The nature and extent of the vessel failure from pontential creep, single penetration failure leading to a large hole, vs. multiple penetrations failing leading to a distrubuted melt relocation mode. The upstream conditions that lead to the core debris initial conditions at the time of failure are critical in determining the fialure mode and pour rate into containment.
Ex-Vessel Behavior													
Dry cavity melt relocation and core-concrete interaction	high	high	medium	medium-high	possible	Spreading in BWR containment is complicated by geometric uncertainties that include large sump(s) under the RPV, pedestal-drywell areas, and containment boundary (Mk I shell)	high	high	medium	med-high	possible	Simpler, smaller cavity for melt spreading to occur.	None
Wet cavity melt relocation and core-concrete interaction	medium	high	medium	low	possible	Affects such as water depth and flooding rate on debris fragmentation and spreading could be addressed analytically. There is little data on particle bed spreading such as would be formed under a PWR with a deep water pool.	medium	high	medium	low	possible	Same comment as for BWRs	Regarding spreading behavior, effect of water pool fragmentation as an initial condition has not been adequately addressed (could be done analalytically); also, multi pour streams have not been addressed experimentally (jprdicted by MELCOR). Regarding core-concrete interaction beahvior, gaps include melt composition effect (i.e. high metal content), presence of rebar, and concrete effects on directional ablation
Containment - RB Response													
Organic seal degradation	high	high	NA	high	possible		medium	high	NA	medium	possible		Understanding seal degrradation under BDBA conditions
Hydrogen and CO monitoring	high	high	N/A	high	possible		high	high	N/A	high	possible		Accounting for condensation in the sampling line is the technical issue with this method in actual plant accidents with loss of ac/dc
PAR performance	high	NA	med	high	possible	PAR removal of CO with aerosols under DBDA conditions	high	NA	med	high	possible		PAR performance for CO/H2 mixtures
H2 stratification and combustion	medium	high	medium	medium	medium		medium	medium	medium	medium	medium		Uncertainty in random ignition sources. Flame front propagation in the containment vent line. Autoignition at high temperatures; this is known for dry air/H2 mixtures, but not H2/H2O/CO.
Emer.Eq. Perf. DBDA Conditions													
RCIC and TDAFW	low	high	N/A	low	possible	No integral testing for extended periods under BDBA conditions.	low	high	N/A	low	possible	Same comment as for BWRs	Performance of RCIC and TDAFW system performance under BDBA conditions.
BWR SRV	medium	high	N/A	Low	possible	No integral testing for extended periods under BDBA conditions.	N/A	N/A	N/A	N/A	N/A	Same comment as for BWRs	BWR SRV performance under severe accident conditions
PWR PORV	N/A	N/A	N/A	N/A	N/A	No integral testing for extended periods under BDBA conditions.	medium	low	N/A	Low	possible	Same comment as for BWRs	PWR PORV performance under severe accident conditions
Raw water	low	med	NA	limited	possible		low	med	NA	limited	possible		Data gap for raw water cooling of molten core debris, on the accident progression in general, and on fission product chemistry.
Fission product transport	medium-high	high	medium-high	medium-high	possible		high	high	high	high	possible		Test data examining fission product scrubbing in H2/H2O mixtures are lacking

APPENDIX B

HUMAN FACTORS AND SEVERE ACCIDENT INSTRUMENTATION EVALUATIONS

As described in Section 3, there are two important areas related to BDBEs in which knowledge gaps are known to exist, but the panel concluded that efforts currently underway by industry, NRC, DOE, and the international community should address these gaps. Specifically, these areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation. For completeness, background material on known gaps and current efforts underway to address these gaps in these two important areas are provided in this appendix. As noted within this appendix, these efforts should be monitored to ensure that existing gaps are addressed.

B.1 Human Factors and Human Reliability Assessment

Background

Nuclear facilities require operator and maintenance personnel. Clearly, understanding how people interact with machines is necessary to ensure safe operations during normal and off-normal operation. On-going industry and regulatory research is addressing issues affecting human performance. As part of these efforts, new tools are being developed for analyzing human performance, and new systems are being developed that will promote improved human performance.

Current NRC human performance research activities address the following goals [108-110]:

- Maintain the infrastructure of expertise, facilities, capabilities, and data
- Ensure that Human Factors (HF)/Human Reliability Assessment (HRA) methods and programs have sound, up-to-date technical bases and guidance
- Improve HF/HRA methods to reduce uncertainty and promote the state of the art
- Expand the HF/HRA infrastructure for new applications (anticipated changes in industry).

Since the earliest days of probabilistic risk assessment (PRA), researchers and practitioners have sought methods for analzying and predicting human performance during accident scenarios. Although numerous methods now exist, none are completely accepted by all interested parties. The NRC is currently evaluating different HRA models in an effort to propose either a single model for the agency to use or to develop guidance on which model(s) should to be used in specific circumstances.

As part of their effort, the NRC participated in the International HRA Empirical Study, a multinational effort co-sponsored by the OECD Halden Reactor Project, the Swiss Federal Nuclear Safety Inspectorate, and EPRI. In this study, different HRA models were used by different teams to analyze and predict operating reactor control room crew performance in responding to certain initiating events. Results are compared to actual operating reactor control room crew performance (e.g., data for crew response to the simulated initiating events were gathered, analyzed, and compared to model predictions for these scenarios). HRA methods considered in this effort included:

- A Technique for Human Event ANAlysis (ATHEANA).
- Standardized Plant Analysis Risk—Human Reliability Analysis Method (SPARH).
- Technique for Human Error Rate Prediction/Accident Sequence Evaluation Program (THERP/ASEP).
- Cause-Based Decision Tree (CBDT).

Data were collected at the Halden Man-Machine Laboratory (HAMMLAB) and a US nuclear power plant.

Experiment results were examined by an international group that developed a number of summary conclusions and a strong recommendation that all HRA analysts first develop a solid and thoroughly documented qualitative analysis to guide later quantification. Based upon these conclusions and recommendations, the NRC is developing a hybrid HRA method that borrows the best aspects of other methods applied in the empirical studies. The goal is to have a single method, or a limited set of alternative methods with a common qualitative underpinning, to apply to specific HRA problems.

The project has produced three reports [111-113]: a volume on the cognitive foundation for HRA, a generic HRA methodology document, and a document for applying this new method to internal events at power. However, this new methodology has not yet been tested on a large-scale application. Limited work on the effects of degraded I&C systems on human performance has been completed.

Safety Relevance

Prediction of human decision making and interactions is of high importance to BDBE scenarios because these require the diagnosis of plant conditions to determine challenges, and future challenges, to fission product boundaries and strategies to address those challenges and return the plant to a safe stable state. There are no automatic actions for these conditions as there might be for some DBA conditions. Additionally, BDBE are likely to result in a high stress environment for the human element which can degrade the ability to choose the appropriate actions in a timely manner.

Knowledge Gaps and Current Efforts Addressing the Gaps

As noted above, research is needed to resolve differences and reduce current uncertainties in HRA modeling for PRA. Of particular importance is human performance when operating in the knowledge based regime such as severe accident management as opposed to a rule-based regime that has been heavily studied in the past. Rule based human reliability is most often applied to prevention of core damage that is guided by Emergency Operating Procedures that use a rule based IF-THEN structure. Severe accident management requires knowledge based decision making because of the many variables and uncertainties that can influence the accident progression. Furthermore, additional research is needed to assess the effects of degraded I&C systems as a result of severe accident environments on human performance.

Current efforts underway by the US NRC, US industry, and the international community indicate that current gaps in HRA modeling have been identified and are being addressed. However, additional efforts may be needed to consider knowledge based human interactions and the effects of degraded I&C systems on human performance.

B.2 Severe Accident Instrumentation

Background

Instrumentation data provide critical information for the operators to diagnose the condition of the plant and assess the impact of any mitigating actions taken during an accident. The need for better instrumentation was recognized after the TMI-2 event, and the events at Fukushima have again emphasized the importance of having a critical set of reliable, rugged, post-accident instrumentation. However, instrumentation system survivability requires knowledge of the environmental conditions that such systems would experience during a wide range of risk-important events.

Regarding US NRC Regulatory Guidance, Appendix A, "General Design Criteria for Nuclear Power Plants" and sections of 10 CFR Part 50," Domestic Licensing of Production and Utilization Facilities" specify several requirements with respect to variables and systems that must be monitored by instrumentation during a DBA and what parameters must be monitored to achieve safe shutdown of the plant and maintain containment integrity. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," contains requirements for new reactor design certification and combined license applications to complete severe accident performance analyses that provide assessments of severe accident equipment needs, predicted environments, and equipment survivability.

Regulatory Guide 1.97 [114,115] provides guidance for instrumentation during and following an accident. Currently, Revision 3 of Regulatory Guide 1.97 [114], which contains a prescriptive list of the minimum number of variables to monitor in BWRs and PWRs, remains in effect for licensees of operating reactors. However, requirements in Regulatory Guide 1.97 are for design-basis events rather than severe accidents. Current reactor and containment instrumentation is not specifically designed to remain functional under severe accident conditions. Revision 4 of Regulatory Guide 1.97 [115] was issued for licensees of new reactor plants. This revision accommodates the increased use of microprocessor-based instrumentation systems in existing and next generation advanced design nuclear power plants. Rather than providing a list of instrument variables to monitor, Regulatory Guide 1.97 Revision 4 provides performance-based criteria for how the variables should be selected. Revision 4 states that licensees should provide instrumentation with expanded ranges and capable of surviving an accident environment (with a source term that considers a damaged core) in which it is located for the length of time its function is required. Revision 4 also endorses (with some clarifying regulatory positions) a standard issued by the Institute of Electrical and Electronics Engineers (IEEE) Standard 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." [116] However, current regulatory guidance has not included a comprehensive evaluation of the instrumentation required for severe accident conditions.

In terms of prior NRC and industry evaluations, the US NRC's Accident Management Research Program funded research in the 1990s to evaluate instrumentation survivability during severe accidents [117-121]. In this effort, the NRC developed a method to identify (a) information needed to understand the status of the plant during a broad range of severe accident conditions including corrective actions, (b) the existing plant measurements which could be used to directly or indirectly supply these information needs, (c) the potential limitations on the capability of these measurements to function properly under the conditions that will be present during a wide range of postulated severe accidents, and (d) the conditions in which information from the measurement systems could mislead plant personnel. Steps were established to identify the severe accidents of interest, the information needed by the operator, the capabilities of the instrumentation system (including transducers, cabling, electronics, and other components), and the severe accident conditions imposed on the sensors. The method was applied to representative PWRs and BWRs for risk-important accident sequences identified in NUREG-1150 [122] using analysis and information available in the early 1990s (e.g., analysis results were limited to available calculations from computer codes, such as MARCH2 and MERGE, which did not consider phenomena, such as natural circulation, that can significantly impact event timing and energy distribution from the core into the upper plenum and regions outside the reactor vessel). Evaluations were completed for five different phases of an accident: (1) initiation; (2) core uncovery; (3) fuel melting and relocation; (4) relocating core accumulation on the vessel lower head and vessel failure; and (5) ex-vessel interactions in the containment. The studies considered selected instrumentation enhancements, such as using existing instrumentation for different applications, extending the operating range of selected sensors, deploying new instrumentation systems, and developing and deploying analysis aids to guide decision-makers during a severe accident.

As part of their response to the accident at TMI-2, the US nuclear industry developed Severe Accident Management Guidelines (SAMGs) for the US nuclear fleet. This guidance encompasses those actions that could be considered to arrest the progression of a core damage accident or to limit the extent of resulting releases of fission products. The original guidance was developed in a logical manner, starting with compiling the best information regarding severe accident phenomena available at that time. This information was, in turn, used to identify candidate high-level actions (CHLAs) that could be taken to manage a severe accident. The CHLAs formed the basis of generic guidance developed by the various owners groups representing the nuclear steam supply system (NSSS) vendors. This generic guidance is ultimately used to assemble the plant-specific guidance for each operating nuclear power plant.

As part of the development of SAMGs, industry sponsored a study in which a systematic process was used to evaluate what types of information might be expected from various types of installed instrumentation during severe accident conditions and the survivability of such instrumentation. The EPRI survivability assessment [123] was completed for two pilot plants (a 4-loop Westinghouse PWR and a Mark II BWR) using plant-specific MAAP computer code analyses performed in the early 1990s. Rather than the NRC approach of focusing on general information needs, the EPRI approach focused on identifying a minimum set of key information needs necessary to support SAMG implementation. Similar to the NRC study, the EPRI method compares the instrumentation operating envelope with conditions predicted to occur for risk-important accidents. EPRI assessment results indicate that existing plant instrumentation can provide the information required during the various phases of severe accidents. Alternatives are available to either directly or indirectly measure the required parameters. For example, rather than identifying sensor enhancements, the EPRI study proposes using operating aides when sensors are not predicted to survive. The EPRI study applied their methodology to two other PWR plants (one CE unit and one B&W unit). Results from this extension suggest that the method can be applied generically with few potential plant-specific differences.

There are on-going efforts by the US industry, NRC, and DOE to address this issue. DOE is sponsoring an updated LWR instrumentation survivability evaluation [126]. The focus of this effort is to determine what key information is needed for severe accident management and mitigation, quantify the environment that instrumentation monitoring this data would have to survive, and identify the gaps in existing instrumentation that would require further research and development. This effort, which includes tasks shown in Figure B-1, draws heavily on successful approaches used in previous evaluations completed by the US NRC and industry.

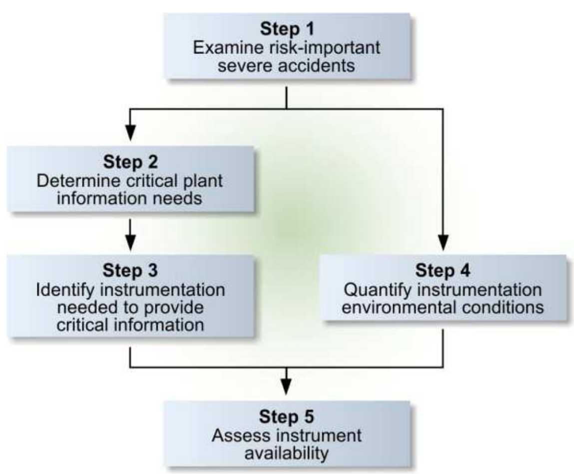

Figure B-1. Instrumentation Survivability Evaluation Approach [126].

Initial studies include evaluating BWR and PWR pilot plants. Because of the availability of severe accident analysis information from recently completed calculations performed in support of the US NRC-sponsored State of the Art Consequence Assessment (SOARCA) program [82,125,126], the pilot plants for this evaluation are the Peach Bottom Atomic Power Station

BWR in Pennsylvania and the Surry Power Station PWR in Virginia. For risk-important accidents identified in the SOARCA studies, critical instrumentation needs are identified based on plant-specific accident management procedures and discussions with plant operators. MELCOR calculation results obtained from the SOARCA studies are then used to quantify the environment to which critical instrumentation systems may be subjected for these events. Then, MELCOR results and instrumentation operating envelops should be compared to assess instrumentation availability. It is currently expected that these pilot plant evaluations will be completed during FY15.

Current NRC efforts to address severe accident instrumentation survivability stem from guidance in Section 4.2 of the Near Term Task Force (NTTF) report [127]. A post-Fukushima action item (Identifier SECY-12-0025, Enclosure 2 [128]) was established to address this concern and to evaluate the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand severe accident conditions. This activity was prioritized as Tier 3 because it requires further staff study and depends on the outcome of other lessons-learned activities. As part of their efforts, NRC staff is reviewing information from previous and ongoing severe accident management research efforts and is monitoring results of the DOE study and international research activities. Some of the questions being addressed by the NRC staff include:

- Is the current instrumentation identified in RG 1.97 adequate to cover the full range of severe accident conditions suggested by the Fukushima event?
- Will the instrumentation qualified to address the guidance of RG 1.97 survive with adequate capability to ensure monitoring of severe accident conditions?

Reference [130] indicates that NRC is considering options, such as dedicated independent power sources for critical plant instrumentation for time periods before FLEX equipment could be installed, analyses and environmental testing that demonstrate that critical instrumentation will survive 'well into the accident progression', and operating procedures that incorporate insights from such analyses and testing. Reference [130] indicates the NRC will make a regulatory determination on this topic by December 2015.

EPRI has formed a Technical Advisory Group (TAG) to address Instrumentation and Control (I&C) for beyond design basis and severe accidents [123,131]. The purpose of the TAG, which consists of representatives from the Institute for Nuclear Power Operations (INPO), EPRI, PWR and BWR Owners Groups, NRC, and DOE, is to promote collaboration and coordination in:

- Addressing the lessons learned from the events in Japan about the required durability and capabilities of I&C systems during severe accident events.
- Identifying the required parameters and ability of reactor and containment I&C systems to withstand severe accident conditions.
- Performing research to determine if the availability of I&C can be improved so that plant data are not lost during beyond DBAs.

Hence, the TAG's role is primarily to facilitate exchange of information and research results.

Both the PWROG and the BWROG are developing Technical Support Guidelines (TSGs) on Instrumentation behavior for severe accidents to complement their post-Fukushima enhanced SAMG [131,132]. These TSGs are based on comparing instrumentation indications with other key information including: alternate instrumentation for the same parameter, assessment of other related, or linked, parameters (such as pressure and temperature), other indications not directly provided by instrumentation and expectations for trending of plant parameters based on the accident progression and mitigation activities. These TSGs provide a basis for the SAMG user to determine the validity of the indications being provided by existing plant instrumentation during a severe accident. The TSGs also provide alternate methods to obtain information when the primary indications for critical SAMG parameters are determined to be unreliable.

Validation activities of enhanced PWROG SAMG incorporating instrumentation TSGs are scheduled for the first half of 2015 at a plant from each of the three PWR reactor vendors (Westinghouse, Combustion Engineering and Babcock and Wilcox). These validation activities will be performed using simulated severe accident scenarios in a table-top mode.

BWROG activities to develop TSGs are currently focused on obtaining insights from detailed evaluations of available TEPCO instrumentation data from Daiichi Units 1, 2, and 3. The BWROG has also performed extensive investigations [133] of the instrument performance at the Fukushima Daiichi units 1-3, including the manner in which differences between indicated and actual values may have influenced actions taken at Fukushima. This included developing principles for use to validate instrument indications as received. These principles were demonstrated on validating RPV water level indications from Daiichi Units 1, 2, and 3, on identifying the presence of metal water reactions using alternate indications (no hydrogen monitors) for Units 2 and 3, and on conflicting indications of RPV pressure on Unit 1 and containment pressures on Unit 2. The purpose behind this was to validate that the SAMG revised to reflect lessons learned from Fukushima could be implemented, with proper training, and utilized with the limited information the operators had at Fukushima.

There have been recent international activities in this area also. The International Atomic Energy Agency (IAEA) established an Action Plan on Nuclear Safety in response to the Fukushima Daiichi events. One of the action items of this plan was to provide guidance to Member States on "Post-accident and severe accident monitoring systems." Reference [134] was prepared in response to this action item to reflect current knowledge, experience and best practices in this area and is based on the results of a series of meetings. It provides a common international technical basis to consider when establishing new criteria for accident monitoring instrumentation to support operation under design basis and design extension conditions in new plant designs and in existing nuclear power plants.

Reference [134] considers monitoring instrumentation and the associated instrumentation support systems for accident prevention and mitigation. The monitoring systems support onsite staff in making decisions for the management of DBAs and Design Extension Conditions

(DECs). Severe accidents are included in DEC. Reference [134] addresses instrumentation that is directly used to implement accident management strategies and instrumentation that may be used to validate or backup the directly used instrumentation. This may include permanently installed instruments that are designated for use in accident monitoring, portable instruments, instruments that are installed but not normally in service, and instruments provided to monitor temporary equipment. Reference [134] recommends that a plant-specific process, similar to the processes used in prior NRC and industry studies, be implemented to ensure that instrumentation with adequate reliability is available for use during a severe accident. At the end of the process, the document advocates that the process should allow a reasonable assessment of existing or contemplated plant capabilities for making decisions related to the adequacy of available instrument, gaps in available instrumentation, and whether additional testing or enhancement of instrumentation systems are needed. The IAEA study emphasizes the importance of considering the following key aspects of instrumentation systems: operating range of conditions, accuracy over the anticipated range, response time, and operating duration. The IAEA study recommends that accident monitoring instrumentation be developed and maintained in accordance with a nuclear quality assurance (QA) program that complies with appropriate guides and to the extent possible, that instrumentation systems be protected and separated from harsh environments (e.g., temperature, pressure, moisture, radiation, shock and vibrations, chemical exposure, electromagnetic fields, voltage surges, etc.).

The Severe Accident - Instrumentation & Monitoring Systems (SA-Keisou) program was established in Japan to develop instrumentation and monitoring systems that could prevent the escalation of an event similar to the accident that occurred at Fukushima Daiichi [134,135]. SA-Keisou emphasizes the need to monitor important variables, such as reactor water level, reactor pressure, and hydrogen concentration, that operators can use to prevent an event from escalating into a severe accident, mitigate the consequences of a severe accident, achieve a safe state for the plant, and confirm the plant continues to be in a safe state over the long term. The SA-Keisou program addresses BWR and PWR instrumentation needs and includes representatives from electric power companies, vendors, and instrumentation manufacturers.

In the SA-Keisou effort, critical parameters or ‘variables’ are selected using somewhat different processes than used in prior US evaluations. Candidate variables are determined using a process that considers: a) required accident management safety functions to prevent damage to the reactor vessel and containment and to suppress offsite radiation release (if the reactor vessel and containment are damaged); b) international guidance; and c) a need for a sequence similar to what occurred at Fukushima. SA Keisou also considers measurement variables required for confirming plant state and equipment operation for various stages of the accident, the required instrumentation accuracy, and the required response time. Recent information indicates that research is underway to provide new instrumentation systems for high priority measurements, and new sensors will be ready for installation in FY2015.

Knowledge Gaps and Current Efforts Addressing the Gaps

Clearly, experiences at TMI-2 and Fukushima Dai-ichi [16,133,136] demonstrate that data from plant instrumentation is critical for plant operators to accurately assess the condition of the plant, diagnose what actions are required, and assess the effects of mitigating actions that are taken. Regulatory requirements are designed to ensure that instrumentation survives DBAs for the existing fleet. Although some regulatory requirements are placed on instrumentation survivability for new reactor designs prior to design certification, additional effort is needed to ensure that reliable assessment of plant conditions necessary for severe accident management can be completed using either instrumentation systems or other means. These methods, in either the existing or new reactors, must be able to withstand severe accident conditions.

As described above, the US DOE is addressing this issue by first completing an instrumentation survivability assessment. For risk important events, BWR and PWR pilot plant assessments are underway to identify critical plant data, determine what environment instrumentation monitoring critical plant data would have to survive, and identify any gaps in existing instrumentation requiring further research and development. It is currently expected that these pilot plant evaluations and efforts to extrapolate results to other plant designs will be completed during FY15. Both the PWROG and BWROG are developing guidance for validating instrument indications to help ensure that appropriate decisions are made considering severe accident effects on instrumentation. In addition, as a NTTF action item, the NRC is evaluating plant instrumentation needs for severe accident conditions. If these efforts identify that selected instrumentation should be enhanced or that additional instrumentation is needed, it is expected that DOE, NRC, and industry will take appropriate actions.

APPENDIX C

EVALUATION OF EMERGENCY RESPONSE EQUIPMENT PERFORMANCE DURING THE FUKUSHIMA DAI-ICHI ACCIDENTS

C.1 Background

Design Basis Accidents (DBAs) are those anticipated, and potentially rare, accidents for which safety systems have been designed to prevent significant fuel damage and to arrest accident progression before a severe accident results that could result in core melting and significant release of radioactivity to the environment. In “design basis space,” systems are included that are designed to prevent significant core uncovering for anticipated events such as large or small break loss of coolant accidents (LOCAs), loss of primary or secondary coolant circulation, and other system transients. The emergency core cooling system (ECCS) components are one example of such systems and include components such as the accumulators in pressurized water reactors (PWRs) and the High Pressure Coolant Injection (HPCI) system in the boiling water reactors (BWRs), whose purpose is to reliably and quickly reflood the reactor core following a loss of coolant accident. These systems generally are classified as “safety grade” systems, subject to strict performance requirements in the “design basis space” as well as to the “single failure” regulatory criteria where failure of any single system, active or passive, cannot result in a severe accident. Typically, multiple system failures are required in order for core damage to result. Other non-safety grade systems, however, may be important in plant response to accident initiators and to whether and when an accident proceeds to core damage. Notable in the recent accidents at the Fukushima Daiichi power station was the role played by the Isolation Condenser (IC) and Reactor Core Isolation Cooling (RCIC) systems, which are non-safety grade systems capable of providing core cooling under emergency situations, as well as the safety grade ECCS HPCI system. The potential utility of these systems under beyond design basis conditions has generally been under-appreciated in severe accident management guidelines and procedures as well as under-credited in probabilistic risk analyses, as will be described in the following.

The accidents at Fukushima Daiichi were well beyond the design basis in a number of ways including extended loss of offsite power, Station Blackout, loss of DC power at Units 1 and 2, and extended isolation from the ultimate heat sink of the Pacific Ocean for all units. The crippled reactors had very little operable equipment to maintain water inventories in the reactor cores and no effective means of rejecting decay heat to the environment. The principal operative systems that factored critically in the damage progression in each reactor were the isolation condensers in Unit 1, and the RCIC and HPCI systems in Unit 2 and 3. Also important was the functioning (or not) of the RPV SRVs, which can function either automatically on RPV overpressure or manually by operator actions involving DC power and pressurized air. Finally, the operator response to the plant parameters, as indicated by instrumentation, [132] is believed to have delayed identification of core damage and hindered accident mitigation.

The following sections describe analyses related to the evaluation of emergency response equipment performance at Fukushima; specifically, the RCS responses for Units 1-3; BWR SRV functioning, and BWR suppression pool behavior.

C.2 Unit 1 RCS Response

The isolation condenser of Unit 1, basically a heat exchanger that extracts heat from the RPV steam and rejects it to the environment is now known to have been operative for only a short time while power was available to Unit 1 before the arrival of the tsunamis that disabled all power systems needed to operate the motor valves controlling the IC functioning. The effect of the intermittent IC operation and its removal of decay heat from the reactor core is reflected in the RPV pressure, shown in Figure C-1.

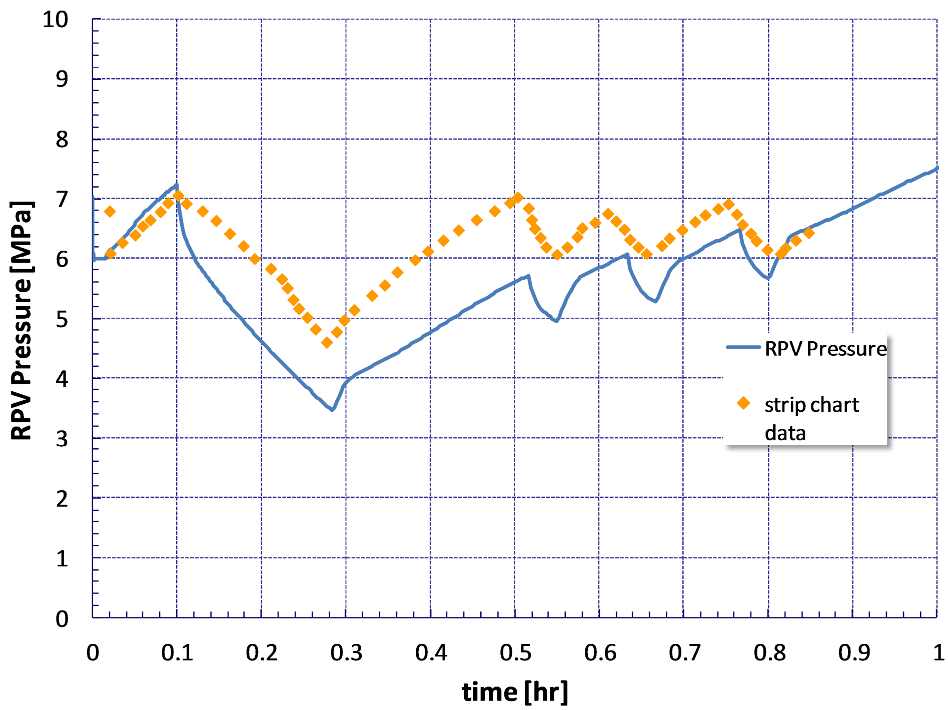

Figure C-1. Fukushima Daiichi Unit 1 RPV Pressure Response as Indicated in Strip Chart Measured Data and Compared with MELCOR Simulation of Isolation Condenser Operation [2].

The HPCI system at Unit 1, potentially capable of injecting water into the reactor vessel, could not be started owing to lack of DC power at Unit 1. With no way to inject water into the RPV and no way to reject decay heat, a boiloff of the reactor inventory commenced following the start of SBO conditions, resulting in core damage, core melt relocation and an assumed breach of the lower head that resulted core materials exiting the vessel and falling to the reactor cavity region below. Based on modeling approaches used in the NRC/SNL SOARCA analysis of SBO in Peach Bottom [2], MELCOR predicted a rupture of the Main Steam Line (MSL) producing an RPV and Drywell containment response as suggested in Figure C-2. Several alternate theories concerning RPV depressurization have been put forth, including one by TEPCO where by failure of the gasket material associated with the cycling safety relief valve flanges (one or two valves), producing a similar RPV and drywell containment response. Other theories include failure of in-core instrument tubes that produce a less rapid RPV depressurization [3]. The actual mode of RPV depressurization in Unit 1 will not be precisely known unless obtained from examinations performed during reactor decommissioning activities. However, it is noteworthy to say that the mode of RPV depressurization can have important effects on both drywell pressure response and on fission product scrubbing (no lack thereof) and ultimately on the magnitude of fission product release from the containment through likely failure locations such as the drywell head flange region. SRV functioning under severe accident conditions could have important impacts on this behavior and will be discussed in greater detail in the following section on Unit 3 accident progression.

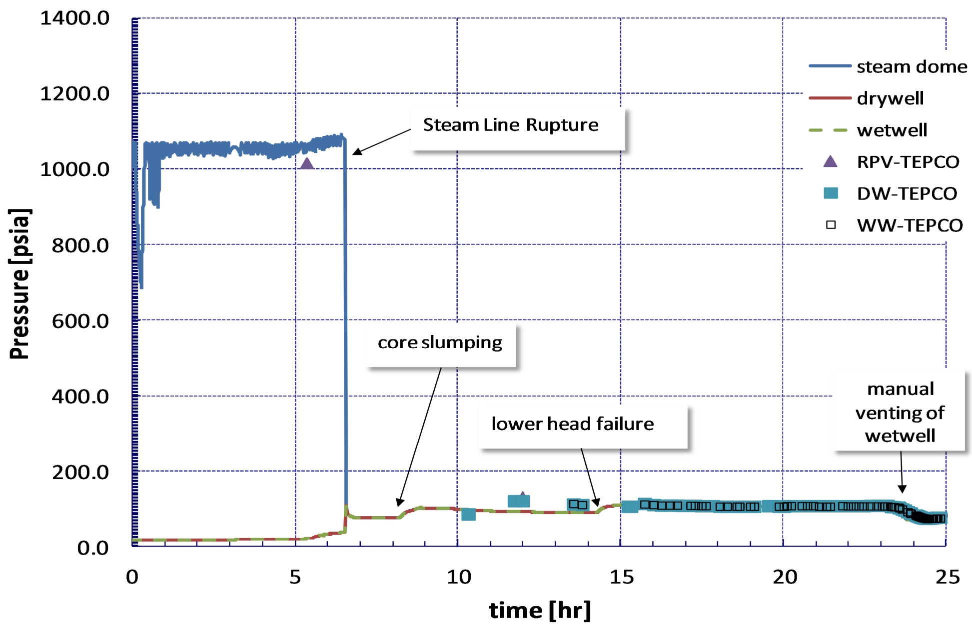

Figure C-2. Fukushima Daiichi Unit 1 RPV Pressure Response are Revealed Through Available Data and Compared to MELCOR Analyses of Unit 1 Accident Progression [2].

C.3 Unit 3 RCS Response

In Unit 3, the next reactor to suffer damage as a result of coolant loss, both the RCIC and HPCI systems were operable because of the availability of DC power. The RCIC system is a simple steam driven turbine that drives a centrifugal pump through a common shaft and is capable of delivering water at full RPV pressure in order to maintain water level as decay heat moves from the RPV to the suppression pool via the RCIC turbine exhaust and cycling SRVs. The RCIC system which was employed initially to inject water into the RPV from either the condensate storage tank or the suppression pool, ran for about 21 hours maintaining RPV water level while the cycling SRV(s) (one or two of 8 were cycling to vent steam and decay heat into the suppression pool, the finite repository for decay energy in the damaged plant (See Figure C-3).

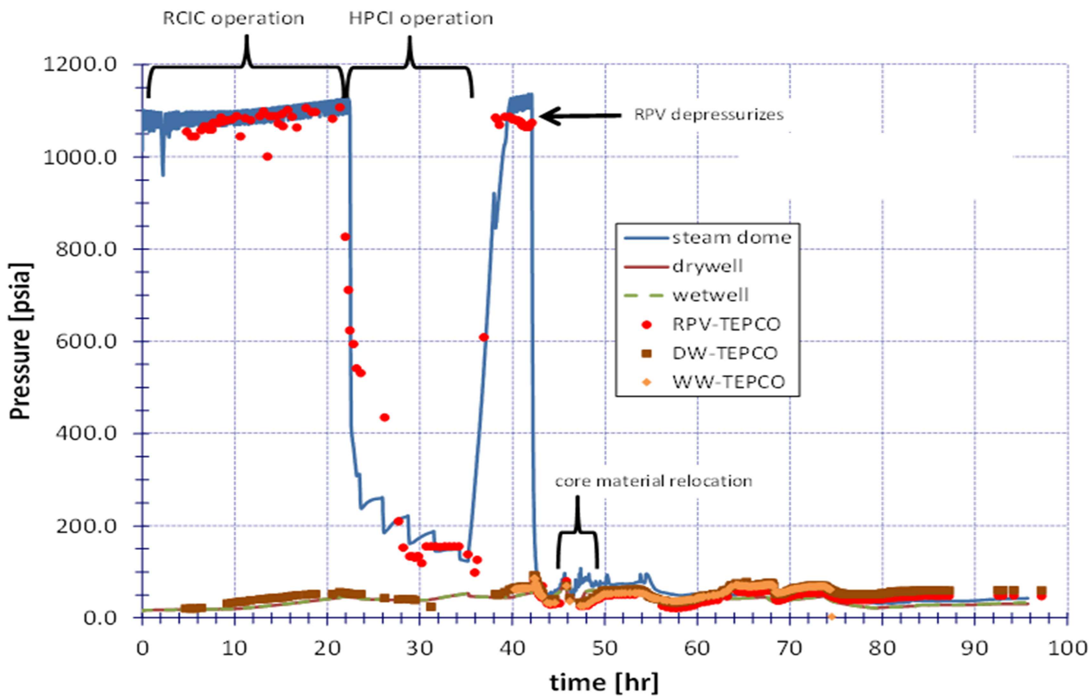

Figure C-3. Fukushima Daiichi Unit 3 RPV Pressure Showing Operation of RCIC and HPCI Systems [2].

The exhaust steam from the operating RCIC and the steam venting through the cycling SRVs, which are directed into the suppression pool, gradually raised the temperature of the water in the pool to the boiling point, thereby gradually raising the pressure of the containment systems comprised by the suppression pool torus and the connected drywell volume. This gradually rising pressure both threatened the integrity of the containment systems as well as the operability of the RCIC system, and eventually, after ~21 hours of operation, the RCIC system shut down automatically on a high turbine exhaust pressure signal, which was only possible because the DC power energizing the shutdown signal was available. The otherwise still-functioning RCIC system was no longer available after that time.

The HPCI system, similar in design to the RCIC system but about ~8-times larger and designed to replace water to the RPV following a postulated LOCA, started up following the RCIC shutdown. This system then re-commenced water injection to the RPV, but because of its larger pumping capacity relative to the RCIC system, the operators, not wanting to be stopping and starting the HPCI system for fear of failure to restart, operated the HPCI system in a continuous mode by recycling much of the water from the pump outlet back to the condensate storage tank and thereby avoiding overfilling the RPV and averting any need to periodically halt HPCI operation. However, this continuous operation had the unfortunate effect of producing a very deep depressurization (See Figure C-3) of the Unit 3 RPV, reducing the RPV pressure so low that the effectiveness of the HPCI steam turbine was *likely* significantly affected, *likely* resulting in little of no water injection after the RPV pressure reached its lowest value of about 150 psig, and loss of RPV water inventory as a result. HPCI operation was terminated at ~35 hrs, resulting in re-pressurization of the RPV to the SRV setpoints and shortly later, core damage is predicted to have occurred.

During the time of MELCOR-predicted core degradation and hydrogen generation, the cycling SRV was observed to stop for some reason as seen in the RPV pressure strip chart data shown in Figure C-4. Thereafter, the RPV dramatically was depressurized at a rate that exceeds the capability of at least 6 simultaneously opened SRVs. This rate of RPV depressurization is consistent with a possible break in the main steam line (MSL) due to high temperature loss of strength, as predicted by MELCOR due to high temperature gases flowing from the RPV to the suppression pool through the cycling SRV. An alternative explanation for the massive depressurization is a possible inadvertent set of signals satisfying the triggering logic for the automatic depressurization system, or ADS, which if indeed operable at the time could explain the opening of 8 SRVs simultaneously and the observed depressurization rate. While some simulations (MELCOR) suggest that core damage was underway prior to the sudden RPV depressurization, evidence of SRV gasket failure, as postulated for Unit 1, does not seem to be operative in Unit 3 based on the steady RPV pressure during this time.

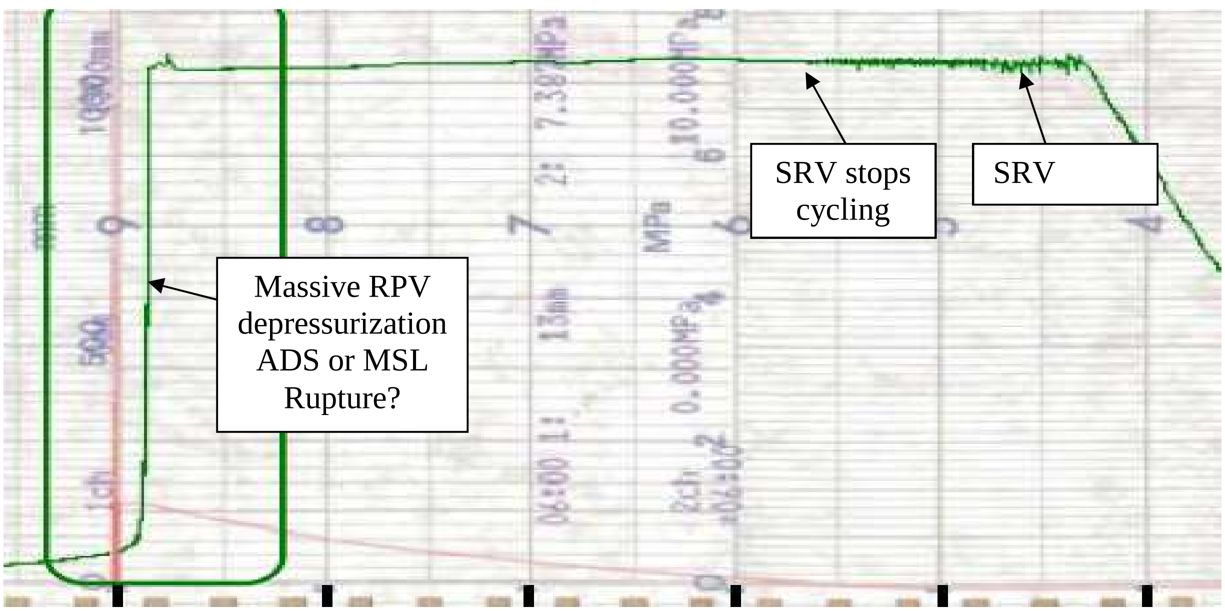

Figure C-4. Strip Chart Data for Fukushima Daiichi Unit 3 RPV Pressure - Time Advancing in Reverse Direction.

Although low pressure water injection in the RPV was commenced following RPV depressurization, core damage is believed to have been extensive with core melting, relocation of debris to the lower head and possibly failure of the lower head, releasing core debris to the concrete cavity below the vessel. The automatic shutdown of the operating RCIC and the degraded operating of the HPCI at low pressure were clear key events in the timeline to RPV water loss and core damage.

Summary observations from Unit 3 include: the functioning RCIC system was disabled by system protection interlock based on high turbine exhaust pressure – However, it is not clear that such protection is warranted under such beyond design basis conditions; HPCI functioning was compromised by too-deep of an RPV depressurization when running in circulation bypass mode – A better understanding of critical component performance under BDBA conditions is needed; and, strange SRV cycling behavior evident from strip chart data suggests accident management can be affected by critical components behavior under severe accident conditions in ways not understood.

C.4 Unit 2 RCS Response

The last reactor to suffer damage at Fukushima Daiichi was Unit 2. Prior to the LOSP and SBO caused by the tsunami that arrived about 45 minutes after the earthquake, the Unit 2 RCIC system had already been started and was running as DC power was lost. With the loss of DC power, the controlling governor valve reverted to a full open position, allowing maximum steam flow to the RCIC turbine and maximum water injection rate into the RPV, well in excess of that required to maintain a steady water level above top of active fuel (TAF). No water level control system was operable to shut down RCIC on high RPV water level. The RCIC system was effectively running uncontrolled after arrival of the tsunami and the resultant loss of DC power. As a result, the RPV water level began to rise steadily until finally reaching the level of the steam lines high in the RPV, whereupon liquid water, instead of pure steam, began spilling over into the steam lines and accumulating at the intake of the RCIC turbine governor valve. Prior to the accidents at Fukushima Daiichi, this condition resulting in water ingestion into the RCIC turbine was assumed to result in failure of the RCIC system and loss of water injection. In the NRC/SNL SOARCA study on Station Blackout in Peach Bottom [82], the assumed depletion of station batteries at roughly 4 hours was assumed to lead quickly lead to RCIC failure due to flooding of the MSL and water ingestion; and in the SOARCA study, core damage ensued quickly following RCIC failure. In contrast, MSL flooding in Fukushima Daiichi 2 apparently did not lead to RCIC failure, which continued to operate uncontrolled for nearly 3 days. RCIC function was ultimately lost at nearly 72 hours due to turbine over-speed interlock shutdown, a mechanical system not requiring DC power and resettable only from the RCIC room. This may have been due to pump cavitations resulting from high suppression pool temperature and development of saturation conditions in the pump.

Analyses now suggest that the Unit 2 RCIC was operating in a self-regulating mode, where water ingestion degrades RCIC functioning temporarily and RPV water injection is reduced or suspended for a period of time as RPV water level drops, followed by periods of RCIC recovery as higher quality steam enters the turbine until the RPV water level recovers to the top of the steam line, and the cycle then repeats itself. This proposed mode of RCIC operation is shown in the MELCOR simulation of Unit 2 RPV pressure response shown in Figure C-5. This reckoned operational mode is consistent with general knowledge that the solid-wheel Terry Turbine in the RCIC system is extremely rugged and has been shown to be unaffected from a damage point of view by short periods of water slug ingestion. This cyclic mode of operation, however, has never been observed in the past. Up to now, water ingestion into the Terry turbine engine has been assumed to lead to RCIC failure in probabilistic risk assessments and in the SOARCA Peach Bottom SBO analyses [82]; and the Terry turbine engine system has not been considered as a recovery system in severe accident management planning to date in spite of its ubiquitous use in both BWR installations (RCIC/HPCI) as well as in PWR auxiliary feedwater systems.

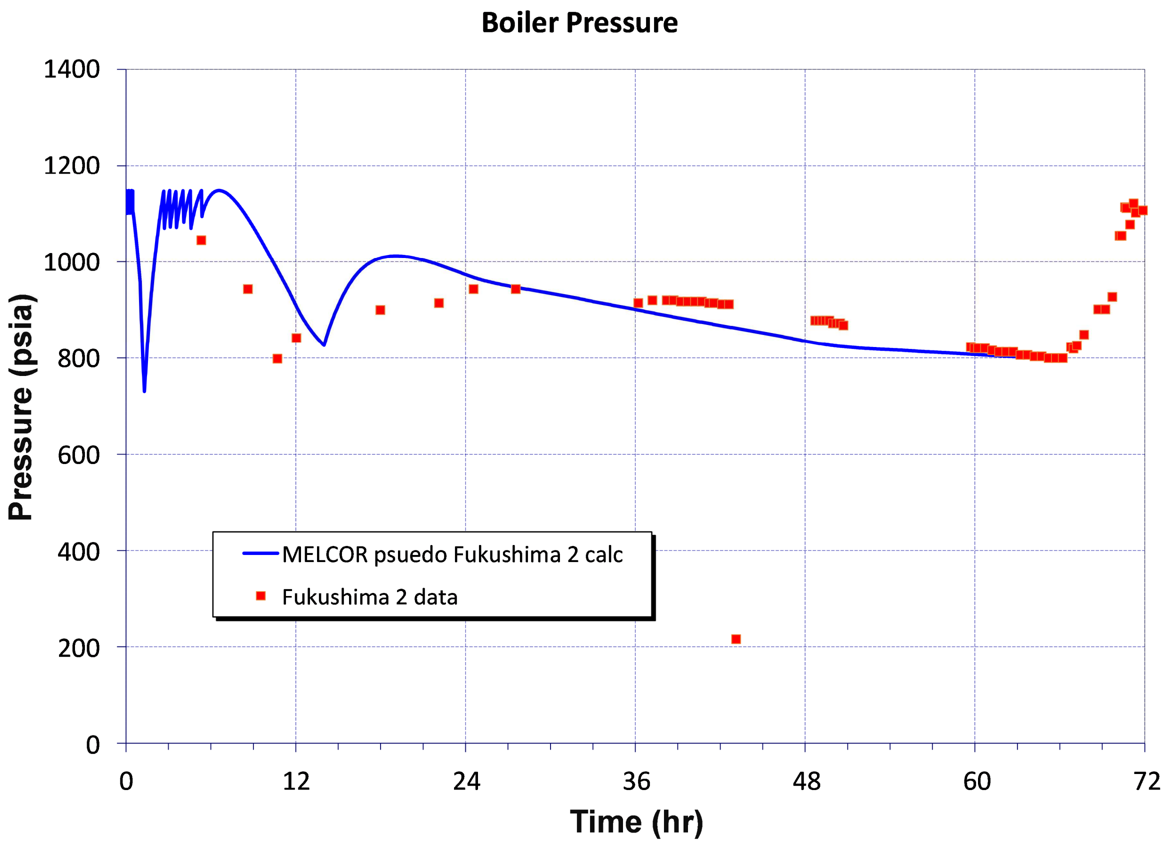

Figure C-5. Fukushima Daiichi Unit 2 Pressure Response Compared to MELCOR Model of Cyclic RCIC Operation with Water Ingestion [2].

This proposed fail-safe use of RCIC under extreme BDBA conditions potentially offers additional time for responders to implement other emergency actions such as FLEX[10] to restore cooling and long term water injection into the reactor vessel. This potential accident mitigation measure has several promising aspects including the fact that Terry turbine driven RCIC and

[10] FLEX is a strategy developed by the U.S. nuclear industry in response to the accidents at Fukushima Daiichi wherein portable equipment such as pumps and generators kept on site or delivered from one of two regional FLEX facilities and used in a “flexible” way to respond to various potential challenges to core cooling and power restoration.

HPCI systems are already installed in all but three BWR installation in the US; the self-regulating mode observed in Fukushima Daiichi Unit 2 requires no operator actions (that we know of), and maintains a maximum RPV water level at the main steamline elevation, providing the greatest heat sink and associated boil-down time should the RCIC system ultimately fail for some reason.

The extraordinary operation of the RCIC system in Daiichi Unit 2 potentially could have prevented core damage, as seemingly ample time should have been available to implement FLEX-like water injection were it not for the plethora of simultaneous multi-unit accident distractions and complications. Ironically, an operational RCIC in Unit 3 was automatically shut down on high turbine exhaust pressure owing to the availability of DC power and the functionality of the system protection features, whereas, in Unit 2, RCIC function was not hampered by operating interlocks such as high RPV water level or high turbine exhaust, and significantly delayed core damage as a result.

The functioning of the Terry turbine driven RCIC and HPCI systems needs to be better understood in beyond design basis conditions, such as loss of DC and turbine control, so that their utility in averting progression to core damage can be realized and factored into severe accident management planning. This includes characterizing their operational characteristics under conditions experienced at Fukushima Dai-ichi Unit 2 where self-regulating behavior was observed to preserve RPV water injection with no operator interactions for nearly three days. Such behavior can then be considered in SAMGs, significantly increasing the success likelihood of accident recovery measures reflected in modern FLEX emergency planning under the most extreme beyond design basis conditions. This is all the more urgent owing to the almost universal use of this system in BWR installations as well as the TDAFW systems that provide emergency feed water to the steam generators in PWRs around the world and the relative low cost of optimizing currently installed equipment.

C.5 BWR SRV Functioning

Little data exist concerning the failure mode and frequency of BWR SRVs under the duress of protracted cycling to relieve RPV overpressure under design basis conditions. Under beyond design basis conditions (i.e., severe accident conditions) there are no data whatever aside from what may eventually be revealed from the Fukushima Daiichi accident decommissioning activities. It is clear from the Fukushima accident recorded data that protracted SRV cycling took place for all three accidents and under the added duress of extreme temperatures cause by core degradation processes. These added stress factors include high temperature hydrogen gas and steam flowing through the steam lines, through the cycling SRVs, down the SRV standpipes and into the suppression pool spargers.

In the SOARCA Peach Bottom study [82], the effect of this high temperature effluent (as high as 1250 K) on the material strength of the steam line piping was estimated through application of Larson-Miller type failure modeling which showed that if RPV effluent temperatures exceed 1200 K that MSL rupture is very likely at the ~1100 psi SRV cycling pressures. The SOARCA study also considered the thermal degradation of the SRV valve itself through thermal distortion of valve stem clearances and the potential for material strain that could lead to SRV seizure in some intermediate position between 100% open to 100% closed. Lacking actual data on valve performance degradation under severe accident conditions, these modeling treatments were somewhat subjective, but based on engineering judgment given typical gap clearances between moving parts and thermal gradients that were expected given the geometry and cycling nature of the valve. Sensitivity studies in the SOARCA Peach Bottom work suggested that if the SRV were to seize open in any position less than 50% open fraction that RPV pressure could not be relieved fast enough to avoid MSL rupture; effluent gas temperatures generally are predicted by MELCOR to exceed 1200K during this phase of the SBO accident. The implications of SRV seizure in severe accident conditions on the potential for MSL rupture are illustrated in Figure C-6. SRV seizure under severe accident conditions at high RPV pressure in stuck-open positions less than 50% could lead to MSL rupture and high airborne fission product concentrations in the drywell with little suppression pool scrubbing benefit.

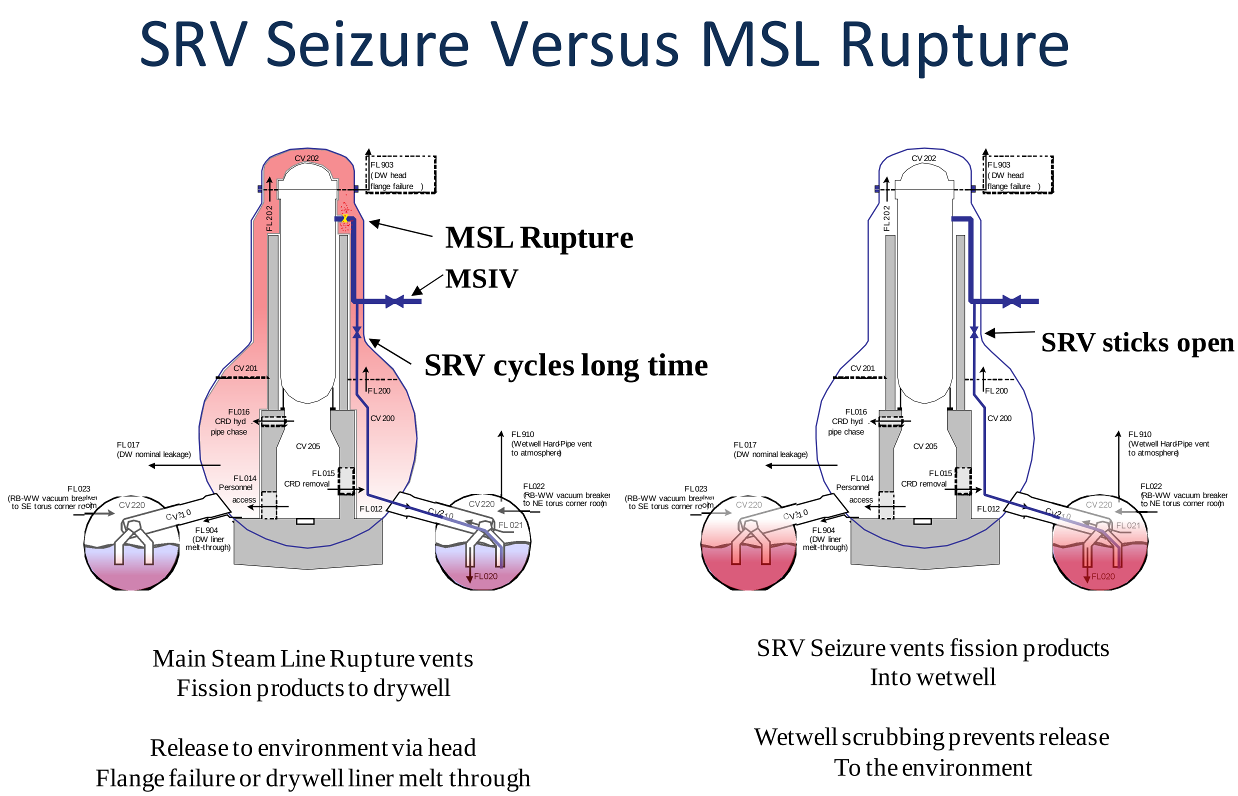

Figure C-6. Effect of SRV Seizure and MSL Rupture on Fission Product Scrubbing and Containment Airborne Concentrations [2].

In order to better understand the possible accident and source term evolution during a BWR severe accident, it is desirable to better understand SRV response to severe accidents, especially the effects of increasing gas temperature on the failure mode of the SRV. This knowledge could allow operators and SAM responders to better manage SRV deployment to control RPV pressure and to minimize potential for MSL rupture that circumvents suppression pool scrubbing.

C.6 Suppression Pool Effects

In the Fukushima Daiichi accidents, the performance of the steam-driven RCIC and HPCI systems has been shown to be highly significant in the progression of the accidents, the timing to core damage, and their potential to arrest the accident by maintaining coolant inventory in the RPV, provided they are operated optimally and are not defeated by common protection systems such as high RPV water level and high suppression pool pressure that are intended to protect systems in a design basis sense.

In the course of post-accident analyses, several unanticipated behaviors in the BWR suppression pool were observed in each of the accidents, notably Unit 2 and Unit 3. The extended operation of the Unit 2 RCIC that prevented core damage for nearly 3 days may be in part assisted by an unanticipated event related to the inundation of the Daiichi site that destroyed the AC/DC power systems and the pumps that served to transport heat to the Pacific Ocean. Based on decay heat considerations alone, the Unit 2 suppression pool pressure should have been considerably higher than was observed. Initially this was thought possibly due to a leak in the containment pressure boundary, but subsequently was postulated to be due to seawater flooding of the torus room. This may have provided an additional pathway for heat rejection from the suppression pool water to the outside environment. In this situation, some fraction of the decay heat accumulating in the suppression pool from RCIC exhaust was evidently being rejected to the water in the flooded torus room. This could have delayed the development of saturation conditions in the suppression pool water and prolonged the operability of the RCIC system.

This discrepancy is shown in Figure C-7 where the expected containment pressure response is shown in the curve indicated “RELAP-5 without leak” and the observed pressure response is shown in the red symbols. (Note it was originally hypothesized that a containment leak must be responsible for the discrepancy).

The water in the suppression pool was apparently being cooled by the seawater in the flooded torus room; however, another curious effect is also evident in the measured data, and that was the likelihood that the suppression pool was also thermally stratified or in some way not well mixed. This means that the observed pressure was driven by the local maximum suppression pool water saturation temperature, or by some uncondensed steam in the drywell or wet well. It is notable that when the SRV begins cycling again after failure of the RCIC system, that the containment pressure drops significantly. This is thought to be due to vigorous mixing and equilibration of the stratified suppression pool water caused by the SRV venting low in the suppression pool. Apparently, the steam entering the suppression pool from the RCIC exhaust prior to SRV self-actuation was insufficient to mix the stratified or localized hot regions.

In contrast to the suppression pool response of Unit 2 where expected pressure was lower than expected, the trend for the Unit 3 suppression pool was just the opposite where observed pressure was larger than expected for a well-mixed suppression pool. This is postulated due either to uncondensed steam venting from the RPV into the wetwell air space or directly into the drywell, or due to thermal stratification or localized saturated conditions in the suppression pool.

The suppression pool response to protracted SBO conditions and isolation from heat sink show that equilibrated pool response may not be the case and that containment overpressure may be strongly affected by non-equilibrium behavior. This can influence containment venting and pressure control strategies as well as the operability of the RCIC and HPCI systems. Understanding of real-world response of this important system (suppression pool) will be important to proper timing of accident management procedures involving containment pressure control and venting and the preservation of capability of systems like RCIC, HPCI and response of SRVs that reference drywell pressure in their opening behavior.

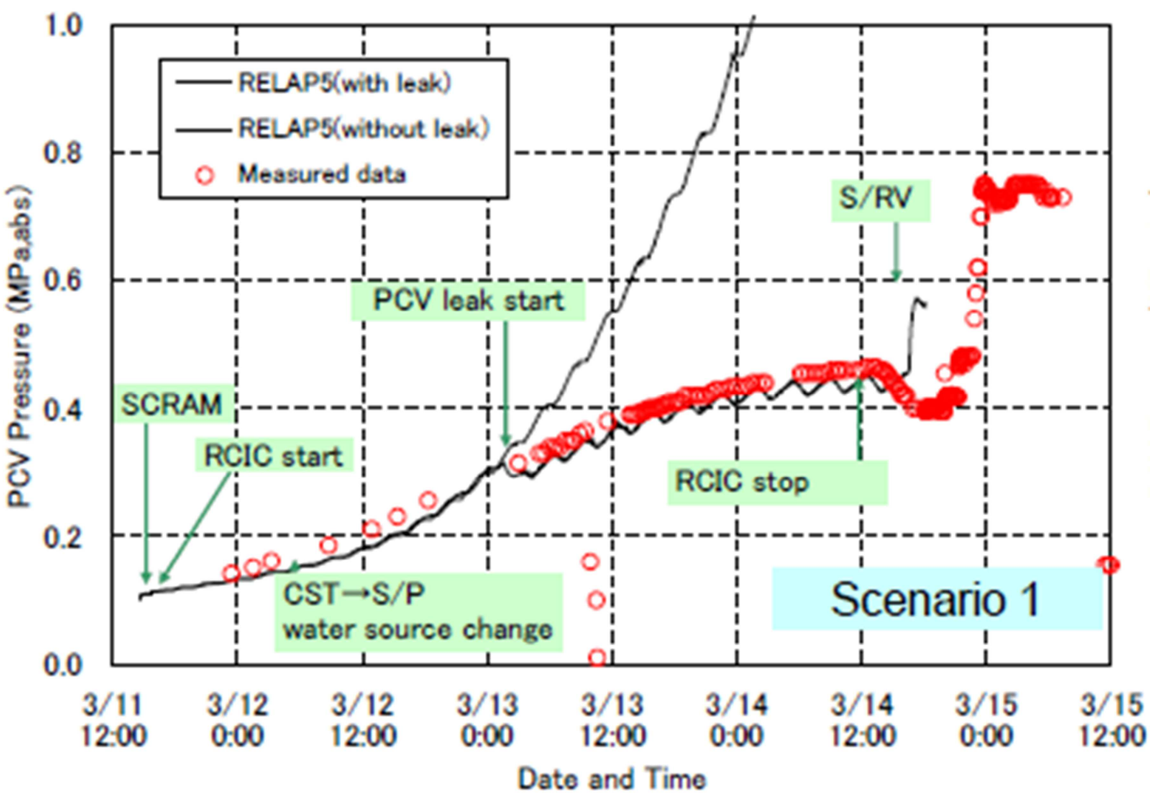

Figure C-7. Fukushima Daiichi Unit 2 Wetwell/Drywell Pressure Response [2].

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