U.S. DOE SEVERE ACCIDENT RESEARCH FOLLOWING THE FUKUSHIMA DAIICHI ACCIDENTS

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ABSTRACT

The Department of Energy (DOE) has played a major role in the US response to the events at Fukushima Daiichi. During the first several weeks following the accident, US assistance efforts were guided by results from a significant and diverse set of analyses. In the months that followed, a coordinated analysis activity aimed at gaining a more thorough understanding of the accident sequence was completed using laboratory-developed system-level and best-estimate accident analysis codes, while a parallel analysis was conducted by industry. A comparison of predictions for Unit 1 from these two studies indicated significant differences between MAAP and MELCOR results for key plant parameters, such as in-core hydrogen production. On that basis, a cross-walk was completed to determine the key modeling variations that led to these differences. In parallel with these activities, it became clear that there was a need to perform 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 augmented by insights from Fukushima. In addition, there is growing international recognition that data from Fukushima could significantly reduce uncertainties related to severe accident progression, particularly for BWRs. On these bases, a group of US experts in LWR safety and plant operations was convened by the DOE office of Nuclear Energy (DOE-NE) to complete technology gap analysis and Fukushima forensics data identification activities. Results from these activities were used as the basis for refining DOE-NE's severe accident Research and Development (R&D) plan. This paper provides a high-level review of DOE sponsored R&D efforts in these areas, including planned activities on accident tolerant components and accident analysis methods.

KEYWORDS
Severe accidents, Fukushima Daiichi, US R&D

1. INTRODUCTION

In the aftermath of the March 2011 multi-unit reactor accident at Fukushima Daiichi, the international nuclear community has been reassessing certain safety assumptions about nuclear reactor plant design, operations, and emergency actions, particularly with respect to extreme events that are beyond each plant's current design basis. The DOE has played a major role in the US response to these events. During the weeks following the accident, DOE provided a significant and diverse set of analyses to guide ongoing US assistance efforts during the events at Fukushima [1]. In the months that followed, a coordinated analysis activity aimed at gaining a basic understanding of the accident sequence was completed [2-4] using laboratory-developed system-level and best-estimate accident analysis codes MELCOR [5], MELTSPREAD [6], and CORQUENCH [7], while parallel analysis [8] was completed by industry using their system level severe accident code, MAAP [9]. A comparison of Unit 1 results indicated significant differences between MAAP and MELCOR predictions for key variables such as in-core hydrogen production and melt pour conditions at the time of vessel failure. On that basis, a joint DOE-industry cross-walk activity was launched to determine the key modeling assumptions between the two codes that led to these differences [10].

In parallel with these activities, it became clear that there was a need for 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 recent insights obtained from the Fukushima accident. In addition, there is recognition within the US and internationally that, similar to the TMI-2 experience, data from Fukushima offer the unique means to significantly reduce uncertainties related to severe accident progression, particularly for BWRs. Thus, there was a need to develop a prioritized list of information that could be gathered as D&D work continues at Fukushima that would be beneficial in reducing modeling uncertainties. Based on these technical needs, a group of US experts in LWR safety and plant operations was convened by DOE-NE to complete these two activities [11-12]. The results were used as the technical basis for refining DOE-NE's R&D plan for the Reactor Safety Technology (RST) pathway within the Light Water Reactor Sustainability (LWRS) program. The objective of the RST pathway is to improve understanding of Beyond Design Basis Events (BDBEs) and reduce uncertainty in predicting severe accident progression, phenomenology, and consequences using existing analytical codes and new information gleaned from the Fukushima Daiichi events.

The purpose of this paper is to review DOE-NE's R&D efforts in the area of severe accidents following Fukushima. This review includes a summary of planned R&D going forward in the areas of accident tolerant components and severe accident analysis methodologies.

2. DOE SEVERE ACCIDENT EFFORTS POST-FUKUSHIMA

This section provides a high-level review of DOE-NE's efforts in the area of severe accidents following Fukushima. Areas addressed include the department's initial accident response, follow-on coordinated analyses aimed at gaining a better understanding of the accident sequence, and subsequent technology gap analysis and Fukushima forensics studies that laid the ground work for the department's future severe accident R&D.

2.1. Initial Accident Response

During the first several weeks of the accident, DOE-NE provided a significant and diverse set of analysis to support the events at Fukushima [1]. The response involved a broad set of institutions with over 200 individual contributors from various DOE offices, national laboratories, and universities. The primary mission of these activities was to assess and clarify information for DOE leadership concerning the status of the reactors at the site. The DOE-NE Nuclear Energy Response Team (NERT) organized activities to support various government offices. Analyses covered a variety of areas that included: i) passive cooling assessments, ii) evaluation of alterative means for decay heat removal, iii) thermal analysis of spent fuel pool heatup and boil off, iv) options for water retrieval and treatment, and v) evaluation of the effects of salt water injection on debris coolability and corrosion of structural steels.

2.2. Coordinated Analyses Aimed at Understanding the Accident Sequence

Following the initial accident response, a coordinated analysis activity was initiated with the objective of gaining a better understanding of the accident sequence. Under sponsorship from DOE-NE, analyses were carried out with the MELCOR code as a collaborative effort between Sandia National Laboratories (SNL) and Oak Ridge National Laboratory (ORNL) [2]. The objectives of this project were to: i) collect, verify, and document data on the accidents by developing a web-based information portal; ii) reconstruct the accident progressions using computer models and accident data; and iii) validate the MELCOR code and the Fukushima models using information available from the accidents at that time. As part of this task, Idaho National Laboratory (INL) developed an information portal for Fukushima accident information. SNL developed MELCOR 2.1 models for Units 1, 2, and 3 and the Unit 4 spent fuel pool. ORNL developed a MELCOR 1.8.5 model of the Unit 3 reactor and a TRACE model of the Unit 4 spent fuel pool. Cross-comparisons between the SNL models with data from the plants and the ORNL model results provided additional confidence in the MELCOR and TRACE predictions. The modeling effort also provided insights into future data needs for model development and validation.

Although MELCOR is able to capture a wide range of accident phenomena, the current version of this tool [6] does not contain detailed models for ex-vessel core melt behavior. However, specialized US codes exist for analysis of ex-vessel melt spreading (e.g., MELTSPREAD [6]) and long-term debris coolability (e.g., CORQUENCH [7]). On that basis, additional analysis was performed to evaluate ex-vessel behavior for Fukushima Unit 1 using MELTSPREAD and CORQUENCH [3-4]. Best-estimate melt pour conditions predicted by MELCOR v2.1 and MAAP5 (obtained as part of a parallel industry study to analyze the accidents at Units 1-3 [8]) were used as input. MELTSPREAD was then used to predict the spatially-dependent melt conditions and extent of spreading during relocation from the vessel. This information was then used as input for the long-term debris coolability analysis with CORQUENCH. These results indicated large variations in debris spreading behavior (i.e., extent of floor coverage that is inversely proportional to debris depth) depending upon the melt pour conditions predicted by MAAP and MELCOR. However, in all scenarios, the core-concrete interaction was predicted to be terminated by quenching of the core debris well before penetration of the concrete basemat.

During the enhanced ex-vessel study [3], it became apparent that there were significant differences in the melt pour conditions predicted by MAAP and MELCOR. As discussed below, a recent technology gap evaluation [11] indicates 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. These gaps have led to differences in modeling approaches adopted by MAAP and MELCOR that strongly impact the predicted behavior. A cross-walk activity between the MAAP and MELCOR development teams was then completed to determine the principal modeling differences between the two codes that lead to such a large divergence in predicting in-vessel core melt progression phenomena. The results [10] indicate that the principal phenomenological uncertainty regarding in-core behavior is the extent that core debris is permeable to gas flow during degradation. Namely, impermeable debris (assumed in MAAP) gradually accumulates as a large high temperature in-core melt mass akin to that formed during the TMI-2 accident, while permeable debris (assumed in MELCOR) steadily relocates to the lower head and collects as a debris bed. These modeling differences are being pursued as additional R&D activities that are detailed in Section 2.4.

2.3. Gap Analysis and Fukushima Forensics Activities

As various US and international analyses of the Fukushima accidents moved forward, it became clear that there was a need to evaluate accident tolerant components and severe accident analysis methodologies to identify knowledge and/or data gaps, given the current state of LWR severe accident research and insights from Fukushima. The high-level objective of this activity was to provide the technical basis for refining the RST R&D program plan to focus on knowledge gaps in severe accident behavior that are not currently being addressed by the industry or the US Nuclear Regulatory Commission (NRC). The approach taken incorporated key features of a traditional Phenomena Identification and Ranking Table (PIRT) process that was structured to address generic reactor designs and scenarios to evaluate overall safety characteristics. The process relied on a panel of US experts in LWR operations and safety with representatives from industry, DOE-NE staff, the national laboratories, and universities. The goals were to: i) identify and rank knowledge gaps, and ii) define 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.

Panel deliberations led to the identification of thirteen knowledge gaps on severe accident analysis and accident tolerant components that were deemed to be important to reactor safety and are not being currently addressed by industry, NRC, or DOE-NE [11]. The results are summarized in Table I. 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 conditions, iv) emergency response equipment performance, and v) additional degraded core phenomenology. Planned R&D activities to address the highest priority gaps are summarized in Section 2.4. It is noteworthy that the panel identified two areas related to BDBEs in which gaps are known to exist, but it was concluded that efforts currently underway by industry and the international community could address the gaps. These key areas are: i) Human Factors and Human Reliability Assessment, and ii) Severe Accident Instrumentation.

As part of the analysis, 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. Much is not known about the end-state of core materials and key structures and components within the affected units. However, similar to what occurred after TMI-2, these reactors offer a unique means to obtain prototypic severe accident data from multiple full-scale BWR cores related to fuel heatup, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel materials. In addition, these units may offer data related to the effects of salt water addition, vessel failure, ex-vessel core/concrete interactions, and Mark I drywell liner attack. Information obtained from these units not only offers the potential to reduce uncertainties in severe accident progression, but may also support potential safety enhancements.

As a first step toward documenting data needs from these reactors, a report is under development [12] that details consensus input from US experts for prioritized time-sequenced examination information and supporting R&D activities that could be completed with minimal disruption of planned TEPCO D&D activities. Input for this report was obtained from the same group of experts that participated in the gap analysis [11]. Similar to that activity, experts from the NRC and DOE-NE also attended and informed participants during the meetings on various topics that included on-going regulatory activities and other relevant international research. TEPCO also attended and discussed their D&D efforts.

In addition to identifying information needs, the report [12] describes why certain information is important and how it would be used to benefit the US nuclear enterprise. In many cases, the identified information needs are of interest to other countries, and the information might ultimately be obtained through international programs. It is anticipated that the US will participate in these international

Table I. Summary of identified gaps with associated importance rankings and recommended R&D to address the gaps [11]

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 systems • 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 Phenomena	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.

programs when they are established. By documenting and disseminating the consensus information needs identified by a broad spectrum of US experts, this report provides a basis to ensure that there is no duplication in effort related to examination information from Fukushima.

Table II summarizes activities for addressing information needs from the affected units at Fukushima that were identified by the panel. The table identifies the desired type of examinations and associated region and/or component from which this information would be obtained; additional details for these information needs are provided in [12]. During the discussions, the panel agreed that some information is required for all identified regions and/or components to obtain a complete picture of the events. Hence, the panel concluded that one can only prioritize needs with respect to 'cost' and the logical sequence for obtaining such information. The results of this prioritization indicate (by the number of asterisks shown in Table 2) that the panel placed the most emphasis upon information that could be obtained from visual examinations, such as videos and photographs. The consensus was that such information was more easily obtained and could provide key information as a screening tool as to whether additional examinations were required.

Table II. Summary of proposed forensics examination activities

Region	Examination Information Classification[1]			
	Visual	Near-Proximity	Destructive	Analytical
Reactor Building				
RCIC	****	***	**	
HPCI	****		***	
Building	****	***	**	*
Primary Containment Vessel				
MSL and SRVs	****		***	
DW Area	****	***	**	*
Suppression Chamber	****	***		
Pedestal / RPV-lower head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions & Shroud	****		***	**
Lower Plenum	****		***	**

[1] ***Examination classification examples (importance and timing ranking based on No. of asterisks; **** being highest rank:*** **Visual**– videos, photographs, etc.; **Near-Proximity**– radionuclide survey, seismic inspection, bolt tension inspection, instrumentation calibration evaluation; **Destructive**– system or component disassembly, sampling, etc.; **Analytical**– chemical analysis, metallurgical analysis, gamma scanning, etc.

2.4 R&D Activities Going Forward

The overall objective of the RST R&D pathway is to improve understanding of BDBEs and reduce uncertainty in predicting severe accident progression, phenomenology, and consequences using existing analytical codes and information gleaned from severe accidents, in particular the Fukushima Daiichi events. Information gained from research in this area will be used to aid in development of mitigating strategies and improving severe accident management guidelines for the current LWR fleet. Current R&D in the RST pathway is focused in the following three areas:

1. *Fukushima Forensics and Examinations*: Ongoing work to develop additional insights into the actual severe accident progression at Fukushima through visual examination and data collection of in-situ conditions at the damaged units as well as collection of samples within the reactor systems and structural components as well as associated analyses.
2. *Severe Accident Analyses*: Conduct analyses using existing computer models and capitalize on their ability to provide information and insights into severe accident progression that aid in the development of severe accident management guidelines (SAMG) and/or training operators on these SAMGs; an auxiliary benefit can be an aid to improvements in these models.
3. *Accident Tolerant Components*: Conduct analysis and experiments on hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of beyond-design basis events.

The focus in each of these three areas is on BDBEs (e.g., extended loss of AC power) and corresponding mitigation strategies (e.g., containment venting). Reactor safety technology R&D under BDBE conditions is being conducted only when the DOE laboratories can provide unique expertise and facilities that are needed by industry.

Based on the recently completed gap analysis [11], the RST Pathway R&D plan is currently being updated to address the highest priority gaps, as summarized below.

<u>Fukushima Forensics and Examinations</u>: Continue to interact with TEPCO to extract existing information from data sources in an accessible format and work with US experts to update and evaluate results from Fukushima examinations. This effort could provide substantial lessons-learned on severe accident progression, similar to those gained from TMI-2 examinations.

<u>In-vessel Severe Accident Analysis</u>: Examine past tests or plan appropriately scaled tests for system code (MAAP/MELCOR) analyses aimed at reducing modeling uncertainties related to late-phase in-core melt progression. As a part of this activity, perform code-to-code reactor simulations to aid in SAMG development and/or to use as training tools.

<u>Ex-vessel Severe Accident Analysis</u>: Support industry in the development of an alternate strategy [13] for responding to the severe accident capable vent Order, EA-13-109 [14] by modifying existing models based on ongoing tests to investigate the effect and management of water addition on ex-vessel core debris coolability. As part of this activity, participate in an on-going ex-vessel core debris coolability test program to gather additional data for validation of US severe accident codes.

<u>Accident Tolerant Components</u>: Based on industry input, proceed with the planning for the design and possible operation of a test facility to better determine the actual operating envelope for BWR Reactor Core Isolation Cooling (RCIC) and PWR Auxiliary Feed Water (AFW) Terry Turbine systems under BDBE conditions. As part of this activity, potentially investigate the performance of BWR Safety Relief Valves (SRVs) and PWR Pilot-Operated Relief Valves (PORVs) as appropriate.

3. CONCLUSIONS

DOE has played a major role in the US response to the events at Fukushima Daiichi. During the first several weeks following the accident, a significant and diverse set of analyses supported US assistance efforts. In the months that followed, a coordinated analysis activity aimed at gaining a more thorough understanding of the accident sequence was completed using laboratory-developed system-level, best-estimate accident analysis codes, while a parallel analysis was conducted by industry. A comparison of predictions from these two studies indicated significant differences in MAAP and MELCOR predictions for Unit 1 key plant parameters, such as in-core hydrogen production. On that basis, a cross-walk was carried out to determine the key modeling variations within these two codes that led to these differences.

In parallel with these activities, it became clear that there was a need to perform 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 augmented by insights from Fukushima. In addition, there is growing international recognition that data from Fukushima could significantly reduce uncertainties related to severe accident progression, particularly for BWRs. On this basis, a group of US experts in LWR safety and plant operations completed a technology gap analysis and Fukushima forensics data identification activities. Results from these activities were used to refine and focus the RST pathway R&D plant moving forward to address the highest priority gaps that include: i) ongoing Fukushima forensics and examinations, ii) in-vessel severe accident analysis, iii) ex-vessel severe accident analysis, and finally iv) accident tolerant component performance under BDBE conditions; namely, RCIC/AFW Terry Turbine systems for BWRs and PWRs, respectively.

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REFERENCES

1. J. E. Kelly, "DOE Response to Fukushima Dai-ichi Accident," http://www.scribd.com/doc/96059393/DOE-Response-to-Fukushima-Dai-IchiAccident-John-E-Kelly-June15-2011-Final (2011).
2. R.O. Gauntt et al., "Fukushima Daiichi Accident Study," SAND2012-6173 (2012).
3. K. R. Robb, M. W. Francis, and M. T. Farmer, "Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH," ORNL/TM-2012/455 (2013).
4. K. R. Robb, M. Farmer, and M. W. Francis, "Ex-Vessel Core Melt Modeling Comparison between MELTSPREAD-CORQUENCH and MELCOR 2.1," ORNL/TM-2014/1 (2014).
5. Sandia National Laboratories, "MELCOR Computer Code Manuals," NUREG/CR-6119 (2011).
6. M. T. Farmer, J. J. Sienicki and B. W. Spencer, "The MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading in Containments," *ANS Winter Meeting Session on Thermal Hydraulics of Severe Accidents*, Washington, DC (1990).
7. M. T. Farmer, "The CORQUENCH Code for Modeling of Ex-Vessel Corium Coolability under Top Flooding Conditions, Code Manual – Version3.03," OECD/MCCI-2010-TR03 (2010).
8. D. Luxat and J. Gabor, "Fukushima Technical Evaluation: Phase 1 – MAAP5 Analysis," EPRI Report No. 1025750 (2013).
9. Fauske and Associates, LLC, "Transmittal Document for MAAP5 Code Revision MAAP 5.02," FAI/13-0801 (2013).
10. D. Luxat, J. Hanophy, and D. Kalinich, "Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk, Phase 1 Study," EPRI Report No. 3002004449, December 2014.
11. M. T. Farmer (editor) et al., "Reactor Safety Gap Evaluation of Accident Tolerant Components and Severe Accident Analysis," Final Report, ANL/NE-15/4 (2015).
12. J. L. Rempe (editor) et al., "US Efforts in Support of Examinations at Fukushima Daiichi," ANL/LWRS-15/2 (2015).

13. Nuclear Energy Institute, “Industry Guidance for Compliance with Order EA-13-109,” NEI 13-02, Rev. 2, (ADAMS Accession No. ML13316A853) December 2014.
14. E.J. Leeds, EA-13-109, “Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation under Severe Accident Conditions,” USNRC, June 6, 2013.