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Light Water Reactor Sustainability Program

Reactor Safety Technologies Pathway Technical Program Plan

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SUMMARY

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 each plant's current design basis. Because of our significant domestic investment in nuclear reactor technology (99 operating reactors in the fleet of commercial light water reactors (LWRs) with five under construction), the United States has been a major leader internationally in these activities. The U.S. nuclear industry is voluntarily pursuing a number of additional safety initiatives. The U.S. Nuclear Regulatory Commission (NRC) continues to evaluate and, where deemed appropriate, establish new requirements for ensuring adequate protection of public health and safety in the occurrence of low probability events at nuclear power plants; (e.g., mitigation strategies for beyond design basis events initiated by external events like seismic or flooding initiators).

The Department of Energy (DOE) has also played a major role in the U.S. 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 safety concerns emerging from uncertainties about the nature of and the effects from the accident. DOE research and development (R&D) activities are focused on providing scientific and technical insights, data, analyses methods that ultimately support industry efforts to enhance safety. These activities are expected to further enhance the safety performance of currently operating U.S. nuclear power plants as well as better characterize the safety performance of future U.S. plants. In pursuing this area of R&D, 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 this DOE-sponsored work.

The response to the Fukushima accident has been global, and there is a continuing multinational interest in collaborations to better quantify accident consequences and to incorporate lessons learned from the accident. DOE will continue to seek opportunities to facilitate collaborations that are of value to the U.S. industry, particularly where the collaboration provides access to vital data from the accident or otherwise supports or leverages other important R&D work.

The purpose of the Reactor Safety Technologies (RST) Pathway R&D 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 and improving severe accident management guidelines for the current LWR fleet. The RST Pathway's activities have evolved from an initial coordinated international effort to assist in the analysis of the Fukushima accident progression and accident response into the following three areas of current work:

1. Fukushima Forensics and Examinations: This R&D is focused on providing insights into the actual severe accident progression at Fukushima through planning and interpretation of visual examinations and data collection of in-situ conditions of the damaged units as well as collection of samples within the reactor systems and structural components from the damaged reactors as well as associated analyses. This effort could provide substantial lessons-learned on severe accident progression, similar to those gained from Three Mile Island accident examinations.

2. Severe Accident Analyses: This R&D is focused on analyses using existing computer models and 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: This R&D work is focused on analysis or experimental efforts for hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of beyond-design basis events.

In each of these topical areas, the RST Pathway's focus is on beyond design basis events (e.g., extended loss of AC power) and corresponding mitigation strategies (e.g., containment venting). Given the finite resources of the Light Water Reactor Sustainability (LWRS) Program and the need to maintain robust R&D efforts in the other technology pathways, the RST Pathway is expected to engage in reactor safety technology R&D only for beyond design basis event circumstances and when DOE's unique expertise and facilities are needed by industry.

As noted above, many of the activities associated with the RST Pathway represent DOE initiatives that had commenced shortly after the Fukushima accident. Thus, we recognized a need for a more comprehensive review on what the industry has focused upon for beyond design basis event subjects as well as what R&D activities NRC is supporting in this area. In January 2015, a "gap" analysis was completed using a team of reactor safety experts from industry (Electric Power Research Institute [EPRI], Boiling Water Reactor [BWR] and Pressurized Water Reactor [PWR] Owners Groups, U.S. vendors), DOE and its national laboratories as well as observers from the NRC and the Japanese industry. The Gap Analysis results have been critical in informing this updated RST Pathway R&D Technical Program Plan, which is described in the following sections.

The R&D activities that can address the highest priority gaps are summarized below:

- Fukushima Forensics and Examinations: Establish a U.S. point of contact to review available information, interact with Tokyo Electric Power Company (TEPCO), extract existing information from data sources in an accessible format and work with U.S. experts to update and evaluate results from Fukushima examinations;
- In-vessel Severe Accident Analysis: Examine past tests or plan appropriately scaled tests if warranted for system code (MAAP/MELCOR) analyses as well as perform code-to-code reactor simulations to aid in SAMG development and/or to use as training tools;
- Ex-vessel Severe Accident Analysis: Modify existing models based on ongoing tests and investigate the effect and management of water addition on ex-vessel core debris coolability;
- Accident Tolerant Components: Based on industry input, proceed with the planning for the design and possible operation of a facility to better determine actual operating envelope for Reactor Core Isolation Cooling (RCIC)/Auxiliary Feed Water (AFW) Terry Turbine systems and potentially for Safety Relief Valve (SRV)/ Power (or Pilot) Operated Relief Valve (PORV) performance as needed and appropriate.

These four R&D areas of investigations are detailed in the following sections of the Pathway's Technical Program Plan.

ACKNOWLEDGEMENTS

Successful preparation of this report required input and support from several individuals and organizations. To get a clear industry perspective, various nuclear industry organizations provided substantial in-kind contributions by providing technical experts to participate in this process at meeting in October 2014 and January and May 2015. These experts came from a wide range of organizations; the Electric Power Research Institute, Exelon Corporation, Southern Nuclear Company, Tennessee Valley Authority, General Electric-Hitachi, Westinghouse, the Pressurized Water Reactor Owners Group, and the Boiling Water Reactor Owners Group. Finally, two organizations provided technical experts to participate in the meetings throughout the early months of Fiscal Year 2015 as observers to the overall process. In particular, Mr. Yasunori Yamanaka and Mr. Kenji Tateiwa from Tokyo Electric Power Company (TEPCO) attended, as well as Drs. Sudhamay Basu and Richard Lee from the U.S. 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. The U.S. Department of Energy's 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 for these meetings.

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ACRONYMS

AC	Alternating current
AFW	Auxiliary feed water
ANL	Argonne National Laboratory
BDBE	Beyond design basis event
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CCI	Core Concrete Interactions
CFD	Computational Fluid Dynamics
D&D	Decontamination and Decommissioning
DOE	Department of Energy
ECCS	Emergency Core Cooling System
ELAP	Extended loss of alternating current power
EPRI	Electric Power Research Institute
FY	Fiscal Year
INL	Idaho National Laboratory
LWRS	Light Water Reactor Sustainability
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
NEUP	Nuclear Energy University Program
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RCIC	Reactor Core Isolation Cooling
RCS	Reactor coolant system
RPV	Reactor Pressure Vessel
RST	Reactor Safety Technology

SAMG	Severe Accident Management Guideline
SNL	Sandia National Laboratories
SRV	Safety Relief Valve
TDAFW	Turbine driven auxiliary feed water
TEPCO	Tokyo Electric Power Company, Incorporated
TMI-2	Three Mile Island Unit 2
TSG	Technical support guidelines

Reactor Safety Technologies Pathway Technical Program Plan

1. INTRODUCTION

The Reactor Safety Technology Research and Development (R&D) effort was established following the Fukushima Daiichi accident. On October 1, 2014, this effort became a pathway referred to as the Reactor Safety Technologies (RST) Pathway within the Light Water Reactor (LWRS) Program. The focus of this pathway seeks to improve our basic understanding of beyond design basis events, and reduce the associated uncertainty in severe accident progression, associated phenomenology, and key outcomes, by using existing analytical codes and information that has been obtained (or will be obtained) from severe accidents, in particular the Fukushima Daiichi events. The insights gained from these models and analyses and the empirical forensics information will be used with the advice and collaboration of the U.S. nuclear industry to better inform nuclear power plant owner/operators in developing mitigating strategies for accidents that may go beyond the design basis and to aid in the formulation of Severe Accident Management Guidelines (SAMGs) or training on those guidelines for the current light water reactor (LWR) operating fleet. Below we provide a general background and motivation for this pathway plan, as well as discuss the overall organization of the projects. Section 2 provides details of the pathway plan and Section 3 the integrated pathway activities. Section 4 summarizes our path forward.

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 each plant's current design basis [1]. Because of our significant domestic investment in nuclear reactor technology (99 reactors in the operating fleet of commercial LWRs with five under construction), the United States has been a major leader internationally in these activities. The U.S. nuclear industry is voluntarily pursuing a number of additional safety initiatives. The U.S. Nuclear Regulatory Commission (NRC) is still evaluating and, where deemed appropriate, establishing new requirements for ensuring adequate protection of public health and safety in the occurrence of low probability events at a licensed commercial nuclear facility; (e.g., mitigation strategies for beyond design basis events, such as extreme external events such as seismic or flooding initiators).

The Department of Energy (DOE) has also played a major role in the U.S. 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 safety concerns emerging from uncertainties about the nature of and the effects from the accident. DOE R&D activities are focused on providing scientific and technical insights, data, analyses methods that ultimately support industry efforts to enhance safety. These activities are expected to further enhance the safety performance of currently operating U.S. nuclear power plants, as well as better characterize the safety performance of future plants. In pursuing this area of R&D, 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 this DOE-sponsored work.

The response to the Fukushima accident has been global, and there is a continuing multinational interest in collaborations to better quantify accident consequences and to incorporate lessons learned from the accident. DOE will continue to seek opportunities to facilitate collaborations that are of value to the U.S. industry, particularly where the collaboration provides access to vital data from the accident or otherwise supports or leverages other important R&D work.

1.2 Motivation

At the Fukushima Daiichi reactors site, the seismic-induced tsunami and subsequent site flooding disabled internal electrical power systems after the earthquake had cut off external power sources, leaving the plants with only a few hours' worth of battery power. Current nuclear power plants need electrical power on a continuing basis, even when the nuclear reactors are shut down, to be able to operate equipment (e.g., valves, pumps) that are used to cool the reactor core and spent nuclear fuel and remove the decay heat and transfer it to an ultimate heat sink. The U.S. nuclear industry is voluntarily pursuing a number of additional safety initiatives. In addition, the NRC has issued an order (and has begun rulemaking) that requires all U.S. nuclear power plants to implement mitigating strategies that will allow them to remove decay heat and cope with an extreme event without their permanent electrical power sources for an indefinite amount of time. These strategies must keep the reactor core and spent fuel cool, as well as protect the integrity of the containment building that surrounds each reactor. These mitigation strategies are expected to use a combination of currently installed equipment (e.g., steam-powered pumps), additional portable equipment that is stored on-site, and equipment that can be transported into the plant site from regional support centers (i.e., the so-called FLEX approach). In addition, SAMGs are being developed to incorporate these mitigating actions.

The accident at Fukushima Daiichi nuclear power station also reinforced the importance of reliable operation of containment vents for boiling water reactor (BWR) plants with Mark I and Mark II containments. In response to the accident, the NRC issued a set of requirements, which required BWR owner/operators with Mark I and Mark II containments to upgrade or install a reliable hardened containment venting system from the containment wetwell as a first phase to improve containment performance for beyond design basis events. These requirements were intended to increase the reliability of BWR Mark I and II containment venting systems to support decay heat removal from the reactor core and to provide protection against over-pressurization of the primary containments. While developing these requirements, the NRC wanted these venting systems to be capable of operation during severe accident conditions when the core may be degraded and radioactive materials are released from the fuel. For the second phase of the requirement, the industry had suggested (and NRC has approved) the use of water addition to the drywell as a way to halt the severe accident progression outside of the reactor vessel and provide an alternative that precludes the need for any drywell venting systems. Once again severe accident analyses and accident tolerant components will be needed to optimize these mitigating strategies.

1.3 Pathway Plan Objectives

The purpose of the RST Technologies Pathway's R&D 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 and improving severe accident management guidelines for the current LWR fleet. The RST Pathway's activities have evolved from an initial coordinated international effort to assist in the analysis of the Fukushima accident progression into the following areas of current work:

1. Fukushima Forensics and Examinations: This R&D is focused on providing insights into the actual severe accident progression at Fukushima through planning and interpretation of visual examinations and data collection of in-situ conditions of the damaged units as well as collection of samples within the reactor systems and structural components from the damaged reactors as well as associated analyses. This effort could provide substantial lessons-learned on severe accident progression, similar to those from Three Mile Island (TMI) accident examinations [1].

2. Severe Accident Analyses: This R&D is focused on analyses using existing computer models and their ability to provide information and insights into severe accident progression [2] that aid in the

development of SAMGs and/or training operators on these SAMGs; an auxiliary benefit can be an aid for improvements in these models [2-4].

3. Accident Tolerant Components: This R&D work is focused on analysis or experimental efforts for hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of beyond-design basis events [5].

4. Research Partnerships: The R&D work needs to be coordinated with other agencies and international safety programs to gain the full benefit of their insights following Fukushima events.

In each of these topical areas, the RST Pathway focus is on beyond design basis events (e.g., extended loss of AC power) and corresponding mitigation strategies (e.g., containment venting). Given the finite resources of the LWRS Program and the need to avoid overlap with R&D efforts in the other technology pathways, the RST Pathway is expected to engage in reactor safety technology R&D only for beyond design basis event circumstances and only when one or more of the following principles are satisfied:

- DOE and its contractors have unique expertise with the R&D subject matter;
- DOE and its contractors have unique facilities that can support experiments needed for a topic;
- DOE and its contractors have unique ideas/concepts that employ their expertise and/or facilities;
- Contributions to the R&D effort will come from industry (e.g., Electric Power Research Institute [EPRI], LWR Owners' Groups).

As previously discussed, many of the activities associated with the RST Pathway represent DOE initiatives that had commenced shortly after the Fukushima accident. Thus, we recognized a need for a more comprehensive review on what the industry has focused upon for beyond design basis event subjects as well as what R&D activities NRC is supporting for this area. In January 2015, a "gap" analysis was completed using a team of reactor safety experts from industry (EPRI, BWR and Pressurized Water Reactor (PWR) Owners Groups, U.S. vendors), DOE and its national laboratories as well as observers from the NRC and the Japanese industry. The Gap Analysis results have been critical in informing this updated RST Pathway's R&D Technical Program Plan.

The Gap Analysis report is now available [6] and has assisted in guiding project plans beyond 2015.

The results of the Gap Analysis became available in mid-Fiscal Year (FY) 2015 and will inform our R&D activities beyond 2015.

2. REACTOR SAFETY TECHNOLOGIES PATHWAY ACTIVITIES

In this pathway plan, we discuss each of the four areas in greater detail and propose a five-year plan in:

- Fukushima Forensics and Examinations
- Severe Accident Analysis
- Accident Tolerant Components
- Research Partnerships

To prioritize R&D activities in these areas going forward, we summarize the Gap Analysis done in FY 2015.

2.1 Gap Analysis Prioritization

Early in FY 2015 we conducted a Gap Analysis on severe accident analysis and accident tolerant components 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 and early forensics examinations. The objective was to improve the RST Pathway's Technical Program Plan and to address key knowledge gaps in severe accident phenomena and analyses, which affect safety and are not currently being addressed by the industry or the NRC.

In the aftermath of the March 2011 accident at the Fukushima Daiichi nuclear power plant (Fukushima; 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 basis and may lead to fuel rod failure and core degradation.

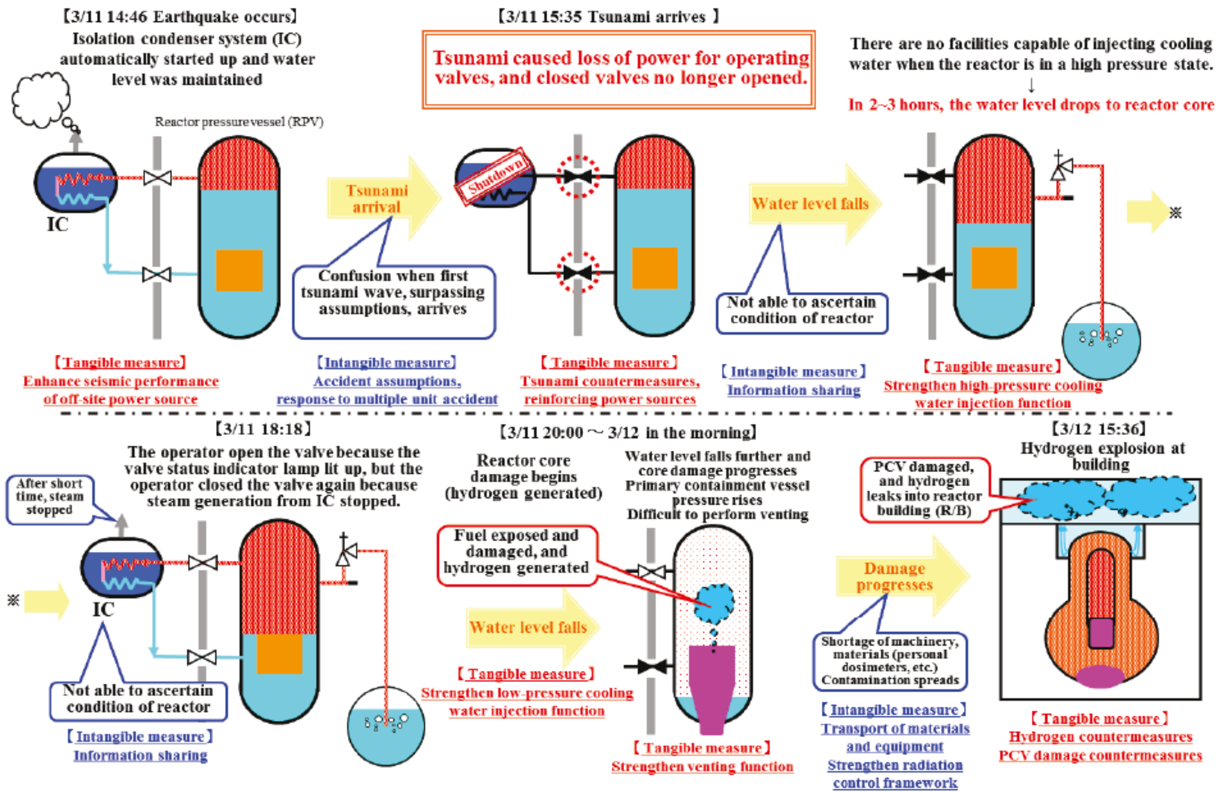

Figure 1. Summary of Accident Progression at Fukushima Daiichi Unit 1 and Necessary Countermeasures (Courtesy of TEPCO).

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 SAMGs that can mitigate challenges to remaining fission product boundaries that could result in a radioactive material release if core damage does occur.

The approach taken to conduct this gap evaluation incorporated key features of a traditional Phenomena Identification and Ranking Technique process. Phenomena Identification and Ranking Techniques 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 analysis, the latter approach was adopted.

The process used a panel of U.S. experts in LWR operations and safety with representatives from the industry [EPRI, boiling water reactor owner's group (BWROG), and pressurized water reactor owner's group (PWROG)], DOE staff, and DOE laboratories [i.e., Argonne National Laboratory (ANL), Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), and Sandia National Laboratories (SNL)]. The goal was 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 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 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. The results are summarized in Table 1, along with recommended R&D actions developed by the panel to address the gaps. 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 (see next Section 2.2 of this plan). Because of this potential the panel recommended that an integrated Fukushima examination plan be developed from the U.S. perspective, which identifies the types and density of data needed from the reactors to address these gaps.

In broad terms, the gap results could be classified into these five categories; i.e., (1) in-vessel core melt behavior, (2) ex-vessel core debris behavior, (3) containment – reactor building response to degraded conditions, (4) emergency response equipment performance, (5) additional degraded core phenomenology.

The R&D activities that can address the highest priority gaps are summarized below:

1. In-vessel: re-analyze past tests or plan appropriately scaled tests, continue MAAP/MELCOR comparison analyses of accident scenarios to aid in SAMG development and/or training tools;
2. Ex-vessel: Modify existing models based on ongoing tests and investigate with appropriate analyses the effect of water addition on ex-vessel mitigation to achieve core debris coolability;
3. Emergency response equipment: Based on industry input, proceed with planning for testing to determine actual operating envelope for Reactor Core Isolation Cooling (RCIC)/Auxiliary Feed Water (AFW) systems a Safety Relief Valve (SRV)/ Power (or Pilot) Operated Relief Valve (PORV) performance. At this time, a decision has not been made to move forward with testing; planning activities and discussions with stakeholders will inform that decision.

It is noteworthy that the panel identified two areas related to beyond design basis accidents 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: (1) Human Factors and Human Reliability Assessment, and (2) Severe Accident Instrumentation. Such topical areas will be reviewed annually.

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 and warranted • 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]	• If a decision to move forward with testing is made, scaled tests addressing melt relocation and vessel wall impingement heat transfer
	Vessel failure	4[a,b]	• If a decision to move forward with testing is made, 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 • If a decision to move forward with testing is made, based on stakeholder input, construct the facility and conduct the testing
	BWR SRVs	6[a]	• If a decision to move forward with testing is made, testing to determine BDBE operating envelope (in RCIC/AFW test facility)
	Primary PORVs	11[a]	• If a decision to move forward with testing is made, 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 core-level degradation; thus, the latter should be higher priority.

2.2 Fukushima Forensics and Examinations

Much is not known about the end-state of core materials and key structures and components wi Units 1, 2, and 3 at the Fukushima Daiichi Nuclear Power Station. However, similar to what occurr after the accident at TMI Unit 2, these Fukushima units offer a unique means to obtain prototypic se 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 rela the effects of salt water addition, vessel failure, ex-vessel core/concrete interactions, and Mark I dry liner attack. Information obtained from these units not only offers the potential to reduce uncertainti severe accident progression, but also may offer the potential for safety enhancements.

Experience from the TMI Unit 2 accident in the United States suggests that critical information be lost if not obtained and documented as soon as feasible during the cleanup and decommissioning process [1]. Experience also suggests that R&D needs must be fully incorporated in cleanup and decommissioning plans early in order to minimize the impact on decommissioning cost and schedul Japan has already begun planning the decommissioning of the damaged Fukushima reactors; theref this is an appropriate time to identify inspection and sampling needs, prioritize them, and sequence most efficiently into the planning process.

As a first step toward ensuring that we obtain the maximum benefit from information available affected units at Fukushima during the TEPCO Decontamination and Decommissioning (D&D), a p was developed that documents consensus input from U.S. experts for prioritized time-sequenced examination information and supporting R&D activities that could be completed with minimal disru of planned TEPCO D&D activities. This plan was developed with input from a broad spectrum of U experts from industry, universities, and national laboratories. Experts from U.S. government organizations (NRC and DOE) also attended and informed participants during the meetings on topic such as on-going regulatory activities and other relevant international research. TEPCO discussed th D&D efforts.

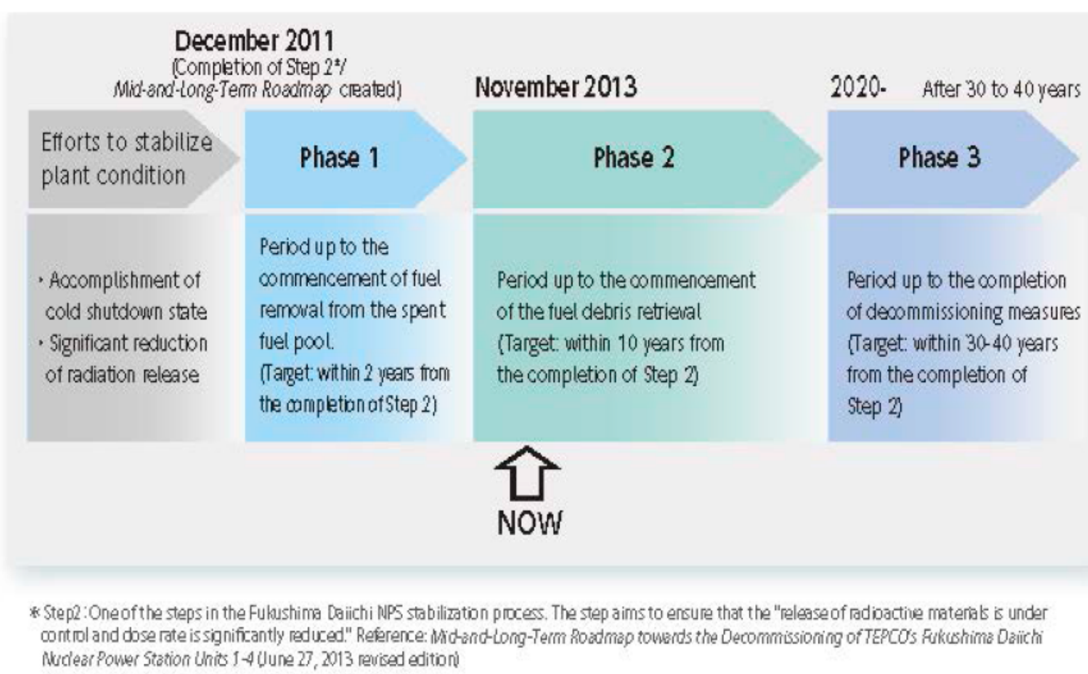

Figure 2. Summary definition of Fukushima D&D roadmap phases.

The plan describes why such information is important and how it would be used to benefit the U.S. nuclear enterprise. In many cases, information needs identified in this plan are of interest to other countries, and the information would likely be obtained through international programs. It is anticipated that the United States will participate in these international programs when they are established. By documenting and disseminating the consensus information needs identified by a broad spectrum of U.S. experts, this plan provides a basis to ensure that there is no duplication of U.S. efforts related to examination information from Fukushima.

Table 2 summarizes activities for addressing information needs from the affected units at Fukushima that were identified by the expert panel. This summary table identifies the desired type of examinations and associated region and/or component from which this information would be obtained. More details regarding information needs are provided in the FY 2015 Forensics Examination Plan [7]. During the FY 2015 discussions, the expert panel agreed that some information is required for all identified regions and/or components to obtain a complete picture of the events. Hence, the expert panel concluded that one could 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 expert 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 2. Summary of proposed Forensics examination activities.

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

a. ***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.

Interactions with representatives with TEPCO during FY 2015 indicate that TEPCO D&D plans (or activities already completed) address much of the information needs identified by the U.S. expert panel. However, TEPCO will be obtaining data to support D&D activities rather than for future safety applications. Hence, current TEPCO plans for data collection, retention, and availability may not meet international nuclear safety needs. Although an international framework should be established that would allow multiple countries to benefit from information obtained during TEPCO's D&D efforts at Fukushima, the expert panel recommended that several tasks be immediately initiated if the U.S. nuclear enterprise is to benefit from this information and future information obtained from TEPCO. These near term tasks (e.g., tasks that should be initiated within the next five years) are outlined below.

Based on the panel input, the high priority tasks that are planned in the next five years are:

- ***Task 1 – U.S. Point of Contact – FY 2015-2020.*** Establish a U.S. Point of Contact to review available TEPCO information, interact with TEPCO, and extract existing information from data sources. Provide in easy-to-read format for U.S. experts to review. Conduct annual program reviews to update information needs (as needed) and issue annual report documenting activities.
- ***Task 2 - Information Evaluations – FY 2016-2020***. Cognizant experts review information for consistency and adequacy, provide additional information requests (if needed), draw reactor safety insights, and post results in easy-to-read format and an easy-to-access location for global access. Selected areas are presented below and activities would be documented in Task 1 annual Report.
  - Component Inspection (based on industry prioritized list and analysis)
  - Dose Measurements for Isotopic Concentration Evaluations (based on analysis evaluations, etc.)
  - Core Debris Location Evaluations
- ***Task 3 - Code Evaluations of Accident Information – On-going – FY 2020***. Review severe accident/dose assessment codes and work with responsible organizations to incorporate new information into code models and provide feedback on recommended forensics (as needed).[b]
- ***Task 4 - Additional Workshops/ Expert Panel Input– FY 2016-2020.*** Conduct new survey/workshops to review results and update information inspection needs by industry with expert input (e.g., instrumentation, structure survivability) Document results in Task 1 annual report.
- ***Task 5 – U.S. Requested Inspections or Technology Deployment- FY 2017- ongoing***. U.S. provides advanced technology to facilitate examinations and sample removal to address information needs or field deployment means of new technology. Document results in Task 1 annual report.

2.3 Severe Accident Analysis

After Fukushima, DOE and other domestic and international groups initiated severe accident analysis efforts aimed at accident reconstruction and analysis of the Fukushima reactor units. While useful insights were gained as to the accident progression, these activities also highlighted where the existing computer system models being used did not always produce consistent results. If such tools were to be used to aid in effective severe accident management guidelines and associated training, further work was needed on identifying the sources of uncertainties and inconsistencies, as well as validating these tools against past and current experimental results, in order to have greater confidence in the use of these tools, as well as inform the need for updated/new tools.

The objective of this R&D activity is to improve understanding of and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing analytical codes; and to use the insights from this improved understanding of the accident to aid in improving severe accident

b. It is anticipated that organizations responsible for development and maintenance of computer codes used in the evaluations of new information would fund these activities and document results separately from this effort.

management guidelines for the current LWR fleet. The information gathered from the application of existing codes to the scenario at Fukushima Daiichi could be used to inform improvements to those codes. At this time, the LWRS Program does not plan to fund major improvement of legacy codes. Rather the objective is to use current tools to develop models that can be used in advanced codes if warranted.

2.3.1 In-Vessel Behavior

Rationale and Objectives:

The first phase of the recently completed crosswalk study [4] identified a number of areas in which MAAP5 and MELCOR have implemented different models of core degradation phenomena inside the reactor pressure vessel (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, these codes adopt 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 (Figure 3).

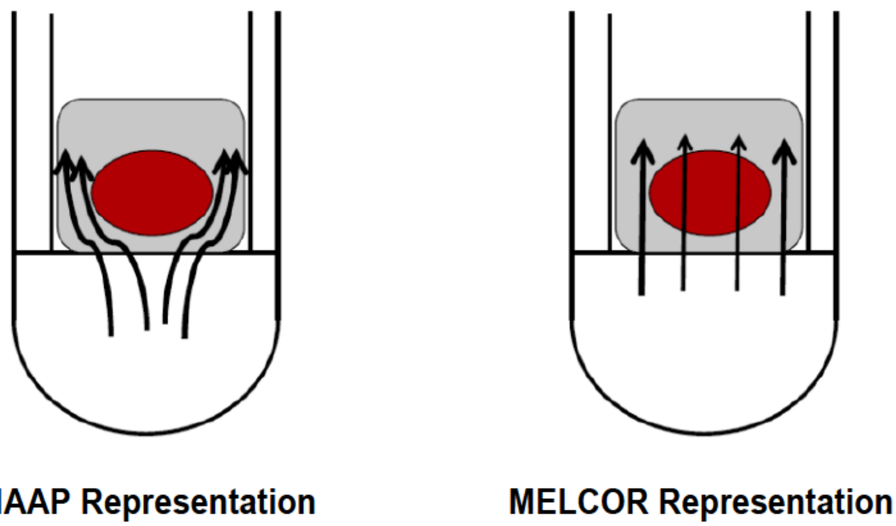

Figure 3. Illustration of differences between MAAP and MELCOR regarding flow through core debris based on different model assumptions regarding debris size and debris permeability

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: (1) formation of a highly superheated molten zone in the core similar to that formed in TMI-2 [3]; and (2) 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 Figures 4 and 5.

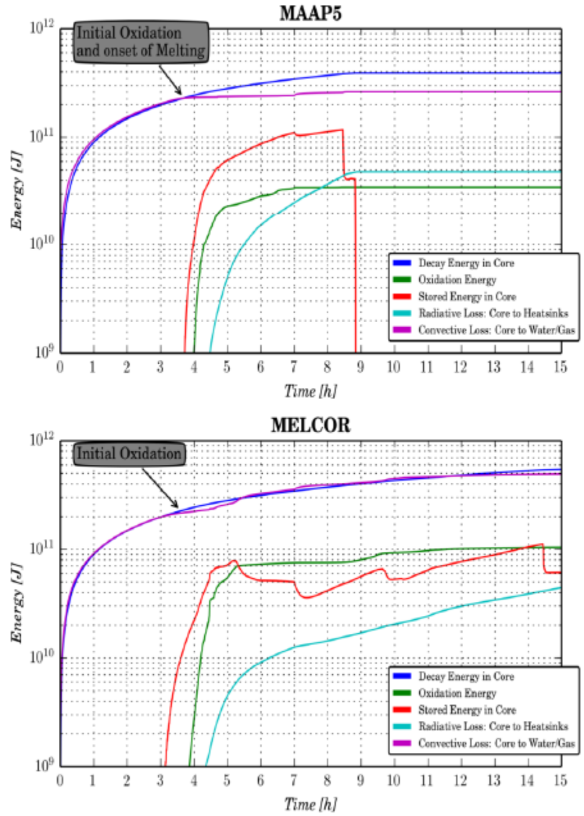

Figure 4. Comparison of Energy Distribution

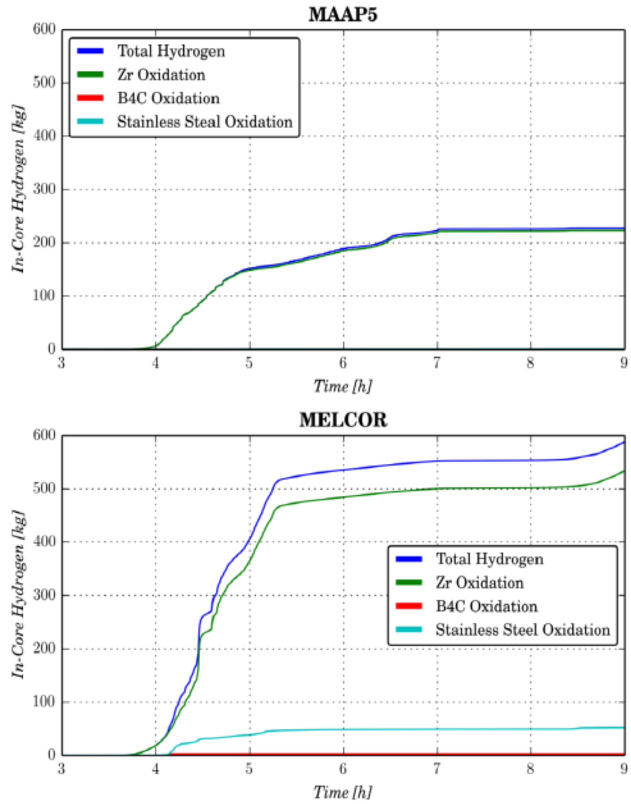

Figure 5. Comparison of Hydrogen Production

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 and core debris compositions and temperature. 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 ultimately 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.

The main objectives of continued severe accident analysis are to: (1) better understand differences in the model physics and eliminate modeling shortcomings, and (2) use these severe accident simulations to help SAMG development and training, with consideration of the inherent uncertainties in the model predictions.

Approach: The following R&D activities are envisioned for In-Vessel Severe Accident Behavior:

MAAP-MELCOR Crosswalk Phase II: This work is a continuation of the Phase I effort in which MAAP and MELCOR were compared against each other for a given set of common inputs for the Fukushima Daiichi Unit 1 in-vessel core melt progression. The Phase II effort will involve a similar comparison of MELCOR and MAAP, but with the Three Mile Island Unit 2 (TMI-2) partial core melt accident scenario as the basis of the comparison. There will be limited uncertainty analyses as part of this effort to investigate the physics in forming the TMI-2 crucible (e.g., eutectic temperature and zircaloy melt breakout temperature). In addition, this second phase of the crosswalk will consider operator actions following the SAMG symptom-based approach during the accident progression.

Confirmation of Symptom-based SAMGs using Severe Accident Analysis with Uncertainties: The purpose of the crosswalk is to identify the model assumptions and limitations in the current severe accident computational tools and by cross-comparison minimize shortcomings in the models and eliminate inconsistencies. Given that the crosswalk efforts have achieved this goal, our expert panel suggested an interesting approach to use severe accident analysis accident signatures to test the SAMG symptom-based approach and confirm that it can address a wide-range of accident signatures and be successful in accident mitigation strategies. The approach is quite straightforward in concept, but promises to be quite powerful, since it will use the severe accident guidance as part of the simulation. The first step is to assemble a subgroup of our expert panel from industry and work with them to identify the key scenarios for BWR and PWR operating plants that may lead to beyond design basis events (e.g., extended station blackout, steam generator tube rupture). The next step would be to simulate the particular scenario and through the symptoms expressed by the simulation, perform the recommended operator actions to mitigate the accident. These accident signatures would take into account potential failures in the operator actions as well as uncertainties in the severe accident simulation. The intent would be to confirm that the SAMGs are robust and will achieve a safe shutdown state even with analysis uncertainties. This would be a multi-year effort and may possibly lead to computer-assisted SAMG development or training.

Development of Software-based Technical Support Guidance: Current industry practice is to develop generic calculations in Technical Support Guidance (TSG) for the Technical Support Center and reactor operators to assist with their SAMGs (e.g., RPV water level and boil down rate given confirmed instrumentation readings). While these analyses can be converted for plant specific calculations, they are not necessarily operator friendly or intuitive. In order for these generic TSG calculations to be intuitive, insightful for severe accident scenarios, and easily transferable between groups (e.g., Technical Support Center, reactor operators, and additional Emergency Operating Procedure/SAMG BWROG support), a new tool needs to be developed. Additionally, a new tool could incorporate more realistic representations of the core, RPV, and containment, and can provide an unbiased and independent validation of the calculations using plant specific MELCOR simulations.

There is proven software, created and maintained as an emergency response tool for the government's Federal Radiological Monitoring and Assessment Center (FRMAC) that provide an intuitive and easily transferable framework necessary to meet a given need. TurboFRAMC, was developed from similar types of calculations over a decade ago. The tool is user-friendly such that any person knowledgeable in plant operations can easily use the code with little instruction. The initial FY 2016 activities will be to work with BWROG Emergency Procedures Committee tool requirements and develop independent verification of current generic analyses. The next step will be to develop an initial software tool for generic BWR TSG analyses. This will be co-funded by the BWROG.

Analysis of Past Experiments with Severe Accident Tools: The Gap Analysis report identified this as an important look-back activity. One option is to do this via the Nuclear Energy University Program (NEUP) to involve university researchers and students. This can assist in documenting test comparisons with new analysts.

Based on the Gap Analysis and input from our panel, the high priority tasks that are planned in the next five years are:

- ***Task 1 – Complete Fukushima Uncertainty Analyses – FY 2015:*** Complete analysis of uncertainties on severe accident progression using the MELCOR systems code as applied to Fukushima Daiichi. This is a follow on to the initial work using general Monte-Carlo analyses.
- ***Task 2 – Complete MAAP-MELCOR Crosswalk Phase 2 – FY 2016-2017:*** This task involves a continued collaboration between EPRI, NRC, DOE laboratories, and possibly with international partners. This second phase of the crosswalk will utilize an accident scenario that is similar to the TMI-2 severe accident as the focus, where operator actions during a severe accident follow SAMG guidance. DOE, NRC, and EPRI will jointly fund this activity during this two-year period.
- ***Task 3 – Confirm SAMG Actions with Severe Accident Analysis and Uncertainties – FY 2016-2020:*** Define a series of key severe accident scenarios (FY 2016), determine the accident signatures from these scenarios (including sensitivities) along with symptom-based operator actions to confirm SAMG mitigating strategy response to severe accident progression (FY 2017-18) and possibly beyond.
- ***Task 4 – Upgrade BWROG Technical Support Guidelines using Severe Accident Analyses – FY 2016-2020:*** Develop software-based tool as part of technical support guidelines for SAMG support using severe accident analyses from MELCOR calculations to inform this software tool.

2.3.2 Ex-Vessel Behavior

Rationale and Objectives:

Specific to the BWR plants, current accident management guidance calls for flooding the drywell to a level above the drywell floor once vessel breach has been determined (Figure 6). While this action can help to submerge the ex-vessel core debris, it can also result in flooding the wetwell and render the wetwell vent path unavailable. An alternate strategy is being developed in the industry guidance [9] for responding to the severe accident capable vent Order, EA-13-109 [10]. The alternate strategy being proposed would attempt to manage the water addition process and throttle the flooding rate to achieve a stable wetwell water level while preserving the wetwell vent path (Figure 7). Ideally, emergency actions would provide adequate water injection to keep the debris covered with water and achieve a quasi-static situation in which the water addition rate matches the core debris cooling rate, thereby maintaining a relatively constant water height over the debris. This approach would achieve the important accident management objectives of keeping the core debris covered while preserving the wetwell vent path.

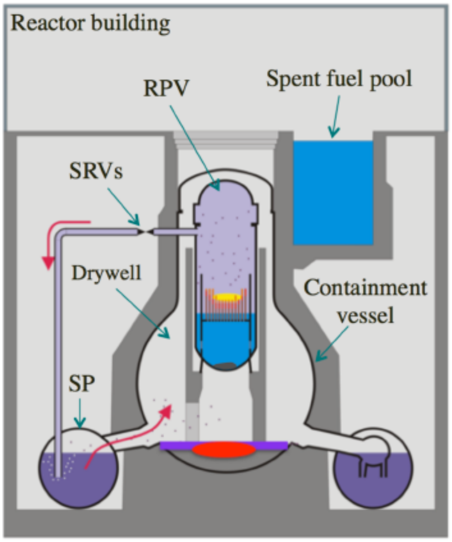

Figure 6. BWR Severe Accident Scenario.

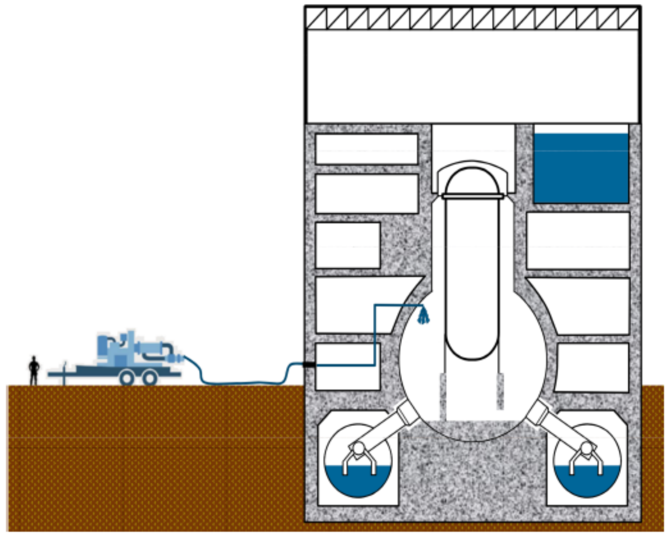

Figure 7. BWR Accident Mitigation by Water Addition.

In support of the development of this alternate strategy, two key questions need to be addressed:

1. What is the minimum water height required to cover the core debris to achieve coolability?
2. Given the extent that the core debris spreads in containment, what is the required water addition rate to match the debris cooling rate?

The answer to the first question relies on two factors; (1) the initial depth of core debris on the drywell floor that is principally determined by the extent that the core debris is able to spread, and (2) local changes in the debris upper surface elevation once the material is immobilized due to natural mechanisms that occur during core-concrete interactions. The extent of spreading is not only determined by the melt pour conditions, but also by melt interactions with below-vessel structure and breakup/quenching in water that may preexist on the drywell floor. Thus, there is a need to employ an ex-vessel spreading model (MELTSPREAD) as well as account for melt-concrete interactions with below vessel structure and water on the drywell floor. Regarding local changes in debris surface elevation due to natural mechanisms during core concrete interaction (CCI), this topic can be informed not only by extending the CORQUENCH debris coolability model [11,12], but also by physical insights and observations gained from core-debris coolability experiments that are ongoing at ANL supported by OECD/NEA.

The answer to the second question relates not only to the extent that the core debris has spread (which determines the melt depth and available surface area for cooling), but also to the longer-term core debris coolability issue [11]; (i.e., the fraction of the overall decay heat in the core debris that is dissipated by boiling to overlying coolant versus ablation of underlying concrete). If the core debris is quenched and rendered permanently coolable, then the minimum water injection rate would be determined by the functional need to remove all the decay heat in the core debris that has relocated. If the debris is not completely quenched, then there will be water accumulation in containment if that same water injection rate is maintained that will accumulate in the wetwell. Finally, responders may not be able to supply the required water injection rate to remove decay heat at all times during the accident due to unforeseen events. Thus, there is a need to extend both spreading and long-term debris cooling models to evaluate situations in which the water addition rate is throttled either intentionally or by unforeseen events.

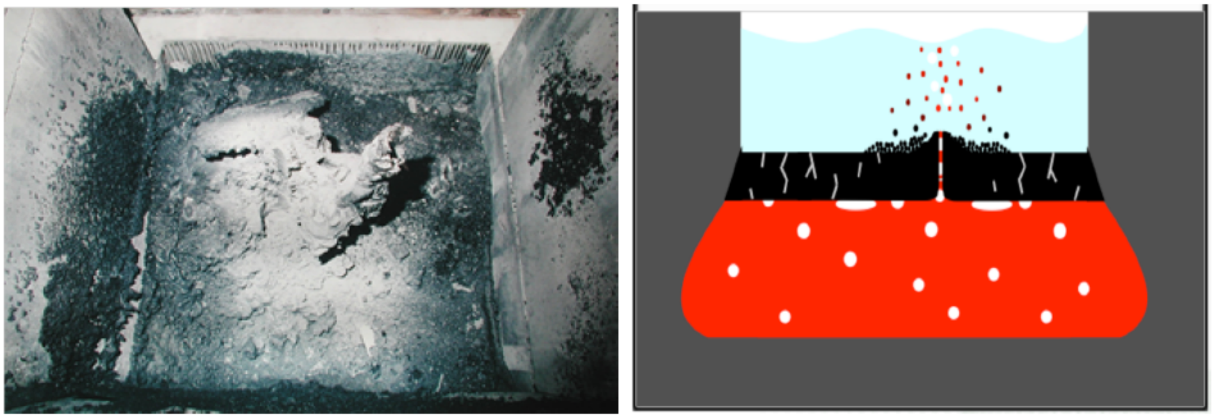

Figure 8. Comparison CCI post-test solidified upper crust and CORQUENCH coolability model.

Note that the work proposed herein is closely related to ongoing core debris coolability experiments at ANL. These tests (Figure 8) give insights into relevant phenomenology, and provide experiment data for code validation; (e.g., data from these tests has been used to validate the CORQUENCH model [12], as well as the nuclear industry MAAP5 code [13]). DOE participation in these experiments is a logical extension of this work as this would provide access to additional data that can be used for validation of upgraded core coolability models proposed as part of this project, as well as other codes such as MAAP5.

With this background, the principle objective of this work is to make modest upgrades to existing analytical tools (i.e., MELTSPREAD and CORQUENCH - which have been used in the Fukushima accident analyses [11]) in order to provide a technical basis for supporting development of water throttling strategies for BWRs that are aimed at keeping ex-vessel core debris covered with water while preserving the wetwell vent path. Specifically, there is currently a gap in analysis capability for evaluating core melt relocation and cooling behavior that accounts for several important factor that include: (1) the influence of below vessel structure and pre-existing water on the containment floor on melt stream breakup and subsequent spreading behvior, and (2) the effect of water throttling on spreading and long term debris coolability. This gap has been identified by the RST Pathway's industry-lab advisory group as a high priority to address [5]. A secondary objective is to participate in ongoing core debris coolability experiments at ANL to provide additional data for validation of these debris coolability models upgraded as part of this task.

Approach: Four separate tasks are needed to accomplish the stated objectives, as outlined below:

Upgrade and document the spreading model: Expand modeling capabilities of an existing spreading code to evaluate: (1) the influence of below vessel structure and pre-existing water on melt stream arrival conditions on the containment floor, and (2) the effect of water throttling rate (including water-starved situations) and water addition location on subsequent spreading behavior. As part of this effort, develop appropriate documentation for these modeling upgrades (as well as numerous other upgrades that have not been fully documented over the years) in the form of a users manual so that the upgraded code can be released to interested users to support industry and R&D activities.

Upgrade and document the core debris coolability model: Incorporate appropriate modeling upgrades into an existing core debris coolability code [12] to evaluate: (1) the effect of water throttling rate (including water-starved situations) on local debris cooling behavior, but more importantly, (2) spatially dependent cavity erosion behavior across the extent of the containment by implementing a multi-nodal scheme that can account for non-uniform cavity geometries such as the Mark I. This latter proposed upgrade, which is considered to be state-of-the art, was carried manually as part of an earlier study [11]; this proposal is to implement software to automate this process for more general applications. As part of this effort, appropriate documentation would be developed for these modeling upgrades in the form of a users manual so that the upgraded code can be released to interested users to support industry R&D activities.

Analysis to support industry efforts to develop an alternate strategy: Utilize the upgraded code suite to carry out a set of parametric calculations to support industry efforts to develop an alternate strategy for responding to the severe accident capable vent Order, EA-13-109 [10]. The initial and boundary conditions for the calculations will be determined through interactions with industry (i.e., EPRI). The overall objective of the analysis will be to define a set of operator actions (e.g., water addition location and flowrate) that will achieve the accident management objective of keeping the debris covered with water while preserving the wetwell vent path. The results of the study will be documented in a technical report.

Conduct debris coolability experiments to validate new debris coolability models: This task involves DOE participation in ongoing experiments being carried out at ANL that are currently sponsored by EdF, IRSN, and NRC. Two large-scale (1 metric ton) tests will be carried out to examine the effect of concrete type and cavity geometry on core debris coolability. Data from these tests would be used for additional validation of the upgraded coolability model (see Task 2). The data would also be available for validation of the U.S. system level severe accident analysis codes MAAP5 and MELCOR. This task involves upgrading the ANL test facility by incorporating new data acquisition capability, as well as modifying the test apparatus to allow longer-term testing, which was identified as a need based on observations from Fukushima.

Based on the Gap Analysis and panel input, the high priority tasks planned in the next five years are:

- ***Task 1 – Upgrade and document the spreading model and debris coolability model – FY 2016-17:*** Complete upgrade and documentation of the spreading model and debris coolability model for industry use.
- ***Task 2 - Analysis to support industry efforts to develop an alternate strategy – FY 2016-17:*** Complete water management severe accident analysis in support of BWR ex-vessel mitigating strategies.
- ***Task 3 – Conduct debris coolability experiments to validate debris coolability models – FY 2016-18:*** These experiments are part of an international exercise with support from NRC, EPRI and international partners. The focus of the experiments is to confirm the range of water addition mitigation strategies that will ensure ex-vessel debris coolability.
- ***Task 4 – Incorporate these models into advanced systems computer codes – FY 2018-20:*** These advanced models would then be appropriate to incorporate into advanced system computer models. This long-term task needs to be re-evaluated after these model upgrades and validation are done.

2.3.3 Source Term Issues

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 remains an important part of reactor safety evaluations. On this basis, R&D in this area has been heavily pursued both within the United States and internationally.

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 reactor coolant system (RCS); vapor interactions with aerosols and surfaces; and pool scrubbing efficiency at saturated conditions and elevated pressure. 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 Nuclear Regulatory Authority is funding a series of small and large-scale tests that may address this data need. In addition, there is the potential to obtain data from experiments conducted in existing facilities located in Europe (e.g., Switzerland, Germany, or France). Most important, as noted in the Gap Analysis [6], there are no data for evaluating the chemistry and associated heat transfer effects of raw water addition on fission product transport. On-going analytical research funded by the NRC may provide some insights on this issue. At this time, we do not see any need for DOE sponsored research. As Fukushima forensics results are collected this judgment may change.

2.4 Accident Tolerant Components

The accidents at TMI-2 and Fukushima demonstrate the importance of accurate, relevant, and timely information on the status of reactor systems during such an accident to help manage the event. While significant progress in these areas has been made since TMI-2, the accident at Fukushima suggests that there may still be some potential for further improvement. Recognizing the significant technical and economic challenges associated with plant modifications, it is important to deploy a systematic approach, which uses state-of-the-art accident analysis tools and plant-specific information to identify critical data needs as well as equipment capable of mitigating the effects of any risk significant accident.

The objective of this R&D activity is to identify opportunities to improve nuclear power plant capability to monitor, analyze, and manage conditions leading to and during a beyond design basis event. Availability of appropriate data and the operator's ability to interpret and apply that data to respond and manage the accident was an issue during the Fukushima accident. The damage associated with the earthquake and flooding inhibited or disabled the proper functioning of the needed safety systems or components.

There are compelling reasons for pursuing this area of R&D both for our domestic reactor fleet; we can also benefit from international collaborations in this area. Results could provide useful information to industry regarding possible post-Fukushima regulatory actions related to sensor and equipment reliability and/or operability [14-16]. Additionally, results and processes developed from this research could benefit Design Certification and Combined Operating License applicants as they are challenged to meet new requirements related to equipment survivability during severe accidents [17]. Finally, analyses and experiments in support of industry initiatives may reveal additional margin in reactor safety systems and components.

Reactor Core Isolation Cooling and Auxiliary Feedwater System Performance

Rationale:

The RCIC for BWRs and AFW 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 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, powering the turbine-pump system injecting water from the condensate storage tank or from the BWR wetwell. For the PWR turbine driven AFW system, steam is drawn from the steam lines upstream of the main steam isolation valves to the turbine-pump with the water source for steam generator injection taken from the condensate storage tank.

Based on events at Fukushima and associated analyses [2], 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.

Except for loss-of-coolant accidents, where the primary system depressurizes down to containment pressure, RCIC and AFW are the major long-term heat removal systems employed under a wide range of transients and accidents for the two reactor types. All probability risk assessment analyses indicate that the dominant accident sequences that are beyond-design basis events (e.g., extended loss of AC power, ELAP) would involve RCIC operation for BWRs and AFW operation for PWRs. Thus, extended performance of RCIC and AFW 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 AFW 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 ELAP events. The importance of the AFW pump has increased in recent years with the installation (or planned installation) of low leakage reactor coolant pump seals in most PWRs. If core damage occurs due to RCS inventory loses, the turbine driven auxiliary feed water (TDAFW) pump also has a high importance in preventing fission product releases from the plant in that it keeps the steam generator tubes submerged and protects them from high temperature creep rupture failures. For extreme external events, AFW can also extend the time at which containment venting might be required.

Implementation of mitigating strategies for BDBEs [9], 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 mitigation 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.

Objectives:

The preceding discussion indicates that there is significant margin in these emergency core cooling systems that has neither been quantified nor qualified with the 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 U.S. 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 extended loss of AC power (ELAP) conditions for both BWR and PWR designs.

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 AFW systems. In effect, there is a need to better determine the actual operating envelope of these components under BDBE conditions. This knowledge can be used in two ways:

- First, it can better inform emergency operating procedures for plant operating staff when RCIC or AFW systems are called upon even under BDBE. This can be a practical aid to expand the time margin before transition to portable systems on-site or brought in from off-site.
- Second, the evaluations could 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. These evaluations could 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.

Approach: This effort has four key elements that are planned over the next few years:

Single-stage Turbine-Pump Model Development: This project would develop a thermodynamically-based analytical model of the single-stage steam-driven turbine-pump system (Terry turbine) operation with mechanistic accounting of liquid water carryover and pump performance degradation. Such a model (Figure 9) could be used for RCIC or AFW models to be included in system codes like RELAP, MELCOR, or MAAP. These insights can provide a basis for test design to operate a RCIC or AFW system under extended uncontrolled operating conditions. Turbine-pump testing of the model would provide an improved understanding and provide the technical basis for improving the reliability of these essential plant systems. Initially, the Fukushima Unit 2 accident reconstruction can be used as the basis for benchmarking this model (Figure 10). 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, if a decision is made to move forward, that would obtain data on RCIC operation under ELAP conditions.

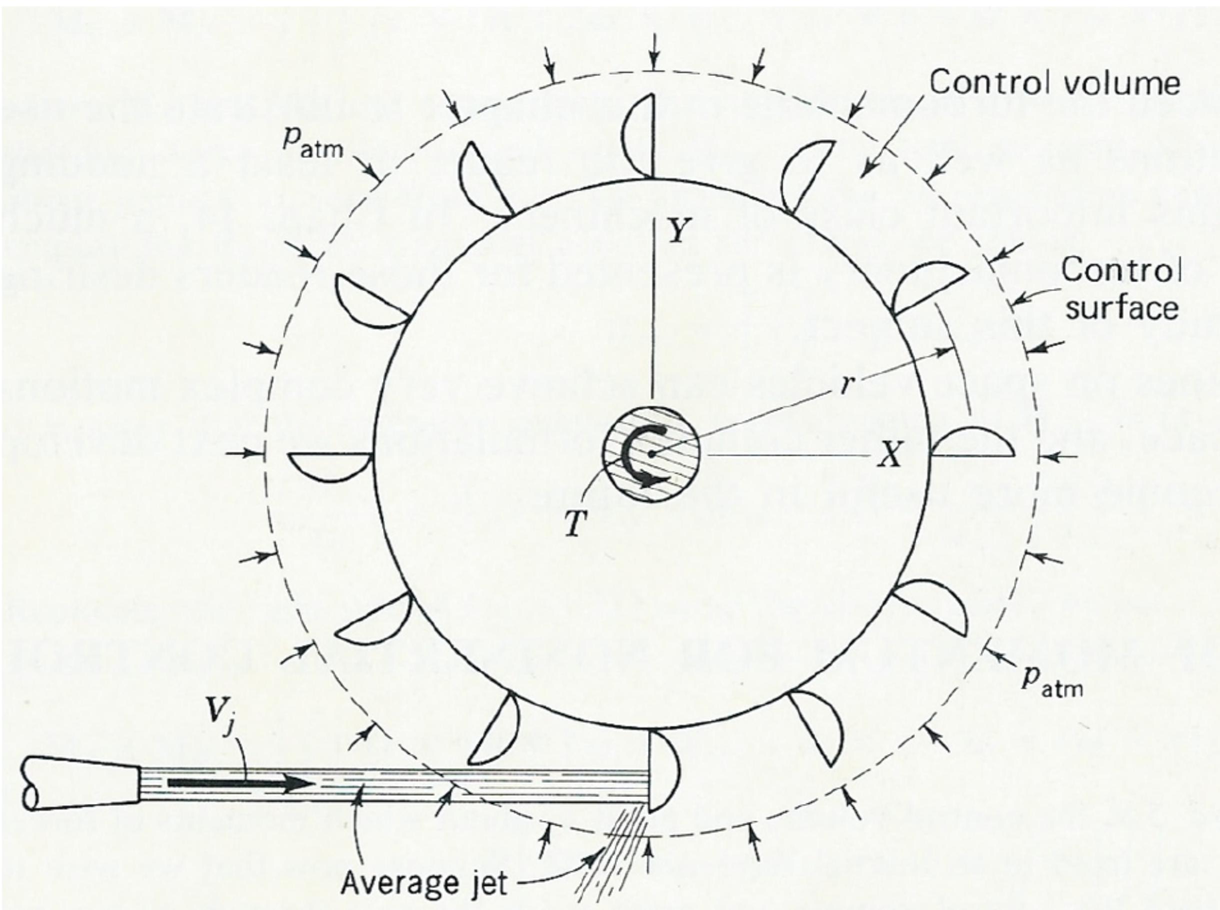

Figure 9. Conceptual Picture of the RCIC Turbine Model.

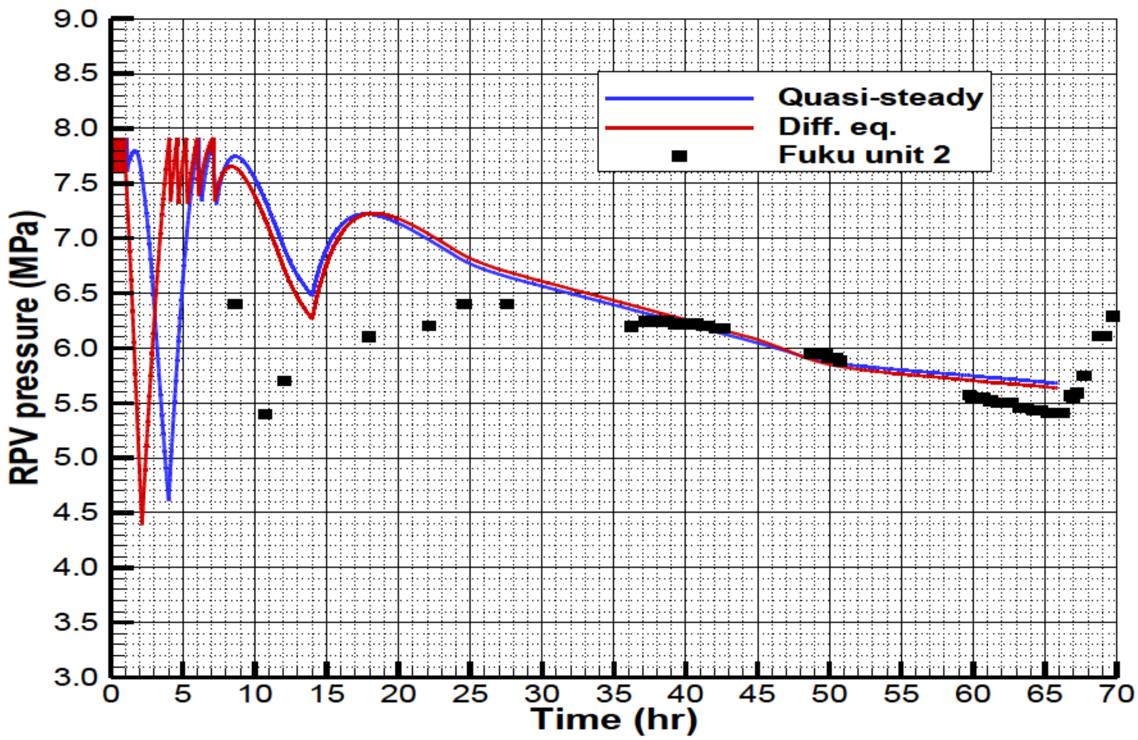

Figure 10. Initial Results of RCIC Turbine-Pump Model simulating Fukushima Unit 2 conditions.

<u>Single Stage Turbine-Pump Experiments</u>: Initial planning of these experiments will begin in FY 2015. Note that at present, a decision hasn't been made as to whether to move forward with testing; that decision will be made based on a cost-benefit analysis. The initial assumption of the experimental program is that testing will be at full-scale pressure, temperature and flow conditions. These initial tests will focus on overall considerations such as test facility size and components, with scoping cost-estimates. To proceed into detailed testing planning will also require industry support and guidance. EPRI will take the leadership role in this supported by Owners Groups. There are a series of issues that need to be clarified before detailed planning and funds are committed.

Because of the scale of these tests, EPRI and stakeholders need to address the following open issues:

- Determination of the value of the testing and the new information on the operating envelope of the RCIC system. It is assumed that if the testing is successful new information would support more confidence in the operational envelope and better values in risk models. It is possible that the testing could support an expanded operating envelope and may translate into changes in the operating practices. While RCIC information in the testing modes is useful, the cost of obtaining this information appears significant and more discussion / analysis is needed in the planning stage for the value proposition to be compelling. Clear and concise project goals and objectives as well as costs are needed.
- There are some important areas of experiments that require more detail such as responsibilities and financial considerations. These details will be important to a successful collaboration. For example, overall project costs, cost sharing proposal, project leadership, contracting, materials procurement as well as others.

- Alternate approaches to full-scale tests (analysis of past tests, scaled tests, subcomponent tests). In addition, based on the Gap Analysis, there may be technical benefit to consider SRV/PORV testing if such a facility is developed.

Based on the Gap Analysis and expert panel input, high priority tasks planned in the next few years are:

- ***Task 1 – Complete the development of a Turbine-Pump (RCIC) model – FY 2015-2016:*** A quasi-steady and transient RCIC model will be developed based on a first-principles technical approach. This model can then be incorporated into various system codes (RELAP or MELCOR or MAAP) for evaluation and comparison to Fukushima integral accident data. Currently researchers at Texas A&M working with SNL are also using computational fluid dynamics (CFD) models (STAR-CCM+ and FLUENT) to validate certain portions of the thermodynamic analytical model for certain aspects (e.g., liquid carryover effects).
- ***Task 2 – Planning for Terry Turbine Experiments – FY 2015-2016:*** Provide initial system planning into possible testing of a single-stage turbine-pump system under beyond design basis conditions. This effort will be in collaboration with industry (EPRI and BWROG) and international partners. EPRI will be the technical lead to determine scope, objectives and program plan if testing is to occur.
- ***Task 3 – Terry Turbine Experiments – FY 2016-2020:*** A decision has not been made yet on whether to move forward with this effort; it will be made with EPRI and industry guidance and a program plan will need to be further developed upon industry consultation.

2.5 Research Partnerships

The RST Pathway interfaces with a number of domestic and international organizations to ensure that the R&D activities of the pathway will be useful to the nuclear community.

Nuclear Industry: Regarding specific RST Pathway R&D activities, EPRI will continue to serve as the primary industry interface. Other important industry interfaces include the reactor vendors (e.g., General Electric-Hitachi on design issues specific to BWRs and Westinghouse for PWRs), individual plant owner/operators participating in plant evaluations, and the PWROG and BWROG regarding licensing and accident management issues as well as Institute of Nuclear Power Operations for operational, management and training matters. In addition to collaboration with these industry groups, the pathway will interface as appropriate with the Nuclear Energy Institute on policy and regulatory matters, and support requests for information on program goals and results.

Nuclear Regulatory Commission (NRC): The interface with NRC is important to the success of this pathway, since NRC regulatory actions and safety priorities can influence DOE's R&D priorities. NRC's Near-Term Task Force report proposed a number of actions for consideration by the Staff and Commission that are now being considered for rulemaking actions. While its primary R&D role is to support research needs that facilitate the deployment and utilization of nuclear energy technologies, DOE has co-sponsored data collection, test programs, code development, and other research topics important to NRC's regulatory oversight role. DOE has a productive working relationship with NRC, based on an effective memorandum of understanding, and a history of successful collaboration on R&D.

International Organizations: The response to the Fukushima accident has been global, resulting in multiple activities by numerous national and international stakeholders. Post Fukushima-related topics, such as accident mitigation strategies, accident monitoring systems, and reactor safety, have already been the focus of international working groups and meetings sponsored by agencies such as the International Atomic Energy Agency and the OECD-NEA. One noteworthy example is the NEA's Senior Expert Group on Safety Research Opportunities Post-Fukushima which is identifying research opportunities that would 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 participant countries. Another example is the NEA's benchmark Study of the Accident at Fukushima Daiichi, which provides information and analysis results on the severe accident progression, fission product behavior, source term estimation of the accident to support safe and timely decommissioning and to contribute to improvement of severe accident codes. Through the RST Pathway, DOE-NE is one of the participants in these multinational projects. These are examples of where a broader experience base and participation can be leveraged to result in more effective, timely and economical responses to enhance the state of safety knowledge. As such, the RST Pathway will continue to seek opportunities for bilateral and multinational collaboration with several international organizations with similar interests and R&D programs.

Bilateral U.S. – Japan Cooperation: A Civil Nuclear Energy Research and Development Working Group has been established under the U.S.-Japan Bilateral Commission on Civil Nuclear Cooperation to enhance coordination of joint civil nuclear R&D efforts between the DOE and Japan's Ministry of Economy, Trade and Industry and Ministry of Education, Culture, Sports, Science and Technology. This Civil Nuclear Energy Research and Development Working Group R&D scope is broader than just reactor safety, but a number of activities, particularly within the LWR R&D sub-working group, such as severe accident code assessment and reactor examination, are closely coupled to the RST Pathway. Further, there are multiple Government and Industry stakeholders in Japan that have been a part of the Fukushima Daiichi response, some of whom are not involved in the Civil Nuclear Energy Research and Development Working Group. DOE periodically meets with Japanese Government and nuclear industry officials from these stakeholder organizations and exchanges views on Japanese policy and activities relevant to the RST pathway.

Universities: University research programs sponsored by DOE represent a potentially important resource for this pathway. The NEUP was established in 2009 to consolidate DOE's university support under one initiative and to better integrate university research with DOE-NE technical programs. The NEUP provides DOE access to a broad array of innovative, cutting edge research and technology within the university system. One current NEUP activity, Multi-Phase Model Development to Assess RCIC System Capabilities under Severe Accident Conditions, directly couples to activities in the RST Pathway while other NEUP activities address issues relevant to safety such as accident tolerant instrumentation. The RST Pathway will continue to seek opportunities for innovative solutions to reactor safety issues within NEUP.

2.6 Summary

The purpose of the RST Pathway R&D is to provide scientific and technical insights, data, analyses and methods that can support industry efforts to enhance nuclear reactor safety during beyond design basis events as we take lessons learned from the Fukushima accident. A listing of the major project plans and associated milestones in the RST Pathway can be found in the Appendix.

3. INTEGRATED REACTOR SAFETY TECHNOLOGY ACTIVITIES

In Section 2, we described the RST Pathway R&D activities based on the Gap Analysis results and the consensus views of our industry panel experts in beyond-design-base accident safety. In this section we summarize the proposed R&D activities and how we integrate these with input from our industry panel.

A summary of our proposed activities is shown in the table below. Note that a decision has not been made yet on whether to move forward with experiments, but all activities are included for completeness.

Table 3. Summary of Reactor Safety Technology R&D Activities: FY 2015 – 2020.

R&D Activity	Fiscal Year					
	FY 2015	FY 2016	FY 2017	FY 2018	FY 2019	FY 2020
Fukushima Forensics & Examinations						
Task 1: Establish U.S. Point of Contact						
Task 2: Accessible Forensics Info.						
Task 4: Expert Panel Evaluation of Info.						
Task 5: U.S.-Japan D&D Technology Trans.						
Severe Accident Analysis						
Complete Uncertainty Report						
Phase-2 MAAP-MELCOR Crosswalk						
SAMG Responses to SA Scenarios						
Develop Software-based TSG w validation						
Severe Accident Water Management Core Coolability Upgrade/Analysis						
Severe Accident Water Management Testing for Core Coolability						
Core Coolability Models for Adv. Codes						
Accident Tolerant Components						
RCIC/AFW Terry Turbine System Model						
Terry Turbine System Test Planning						
Terry Turbine System Testing						

These R&D activities were identified and had consensus approval by a group of U.S. industry panel. To get a clear industry perspective, various nuclear industry organizations provided substantial in-kind contributions by providing technical experts to participate in meetings held in October 2014 and January and May 2015. These experts came a wide range of organizations; the EPRI, Exelon Corporation, Southern Nuclear Company, Tennessee Valley Authority, General Electric-Hitachi, Westinghouse, the PWROG, and the BWROG. In addition, two organizations provided technical experts to participate in the meetings in FY 2015 as observers to the overall process. In particular, Mr. Yasunori Yamanaka and Mr. Kenji Tateiwa from TEPCO attended, as well as Drs. Sudhamay Basu and Richard Lee from the NRC Office of Nuclear Regulatory Research. The RST Pathway intends to convene this group of U.S. industry experts at least on a semi-annual basis to review R&D progress and participate in evaluation of R&D results. In addition, a subset of the panel will provide a formal review of the RST Pathway on an annual basis per the DOE requirement for a peer review.

4. CONCLUSIONS AND RECOMMENDATIONS

The purpose of the RST Pathway's R&D is to improve understanding of BDBE 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 and improving SAMGs for the current LWR fleet. The RST Pathway's activities have evolved from an initial coordinated international effort to assist in the analysis of the Fukushima accident progression and accident response into the following three areas of current work:

- Fukushima Forensics and Examinations: This R&D is focused on providing insights into the actual severe accident progression at Fukushima through visual examination and data collection of in-situ conditions of the damaged units as well as collection of samples within the reactor systems and structural components from the damaged reactors as well as associated analyses. This effort could provide substantial lessons-learned on severe accident progression, similar to those gained from Three Mile Island accident examinations.
- Severe Accident Analyses: This R&D is focused on analyses using existing computer models and their ability to provide information and insights into severe accident progression that aid in the development of SAMG and/or training operators on these SAMGs; an auxiliary benefit can be an aid to improvements in these models.
- Accident Tolerant Components: This R&D work is focused on analysis or experimental efforts for hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of beyond-design basis events.

In each of these topical areas, the RST Pathway focus is on BDBE (e.g., extended loss of AC power) and corresponding mitigation strategies (e.g., containment venting).

Initial activities in the RST Pathway were short term, having commenced shortly after the Fukushima accident. We recognized the need for a more comprehensive review on what the industry has engaged for BDBE subjects as well as what R&D activities NRC is supporting for this area. In January 2015, a Gap Analysis was completed using a team of reactor safety experts from industry (EPRI, BWR and PWR Owners Groups, U.S. vendors), DOE and its national laboratories as well as observers from the NRC and the Japanese industry. The Gap Analysis results have been critical in informing this updated RST Pathway R&D Technical Program Plan, which was described in detail in Section 2.

The R&D activities in our pathway plan, identified by our expert panel, address the high priority gaps:

- Fukushima Forensics and Examinations: Establish a U.S. point of contact to review available information, interact with TEPCO, extract existing information from data sources in an accessible format and work with U.S. experts to update and evaluate results from Fukushima examinations.
- In-vessel Severe Accident Analysis: Examine past tests or plan appropriately scaled tests if warranted for system code (MAAP/MELCOR) analyses as well as perform code-to-code reactor simulations to aid in SAMG development and/or to use as training tools;
- Ex-vessel Severe Accident Analysis: Modify existing models based on ongoing tests and investigate the effect and management of water addition on ex-vessel core debris coolability;
- Accident Tolerant Components: Based on industry input, proceed with the planning for the design and possible operation of a facility to better determine actual operating envelope for RCIC/AFW Terry Turbine systems and potentially for SRV/PORV performance as needed and appropriate.

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APPENDIX

Fukushima Forensics and Examinations

- ***Task 1 – U.S. Point of Contact – FY 2015-2020.*** Establish a U.S. Point of Contact to review available TEPCO information, interact with TEPCO, and extract existing information from data sources. Provide in easy-to-read format for U.S. experts review. Conduct annual program reviews to update information needs (as needed) and issue annual report documenting activities.
- ***Task 2 - Information Evaluations – FY 2016-2020***. Cognizant experts review information for consistency and adequacy, provide additional information requests (if needed), draw reactor safety insights, and post results in easy-to-read format and an easy-to-access location for global access. Selected areas are presented below and activities would be documented in Task 1 annual Report.
  - Component Inspection (based on industry prioritized list and analysis)
  - Dose Measurements for Isotopic Concentration Evaluations (based on analysis evaluations, etc.)
  - Core Debris Location Evaluations
- ***Task 3 - Code Evaluations of Accident Information – On-going – FY 2020***. Review severe accident/dose assessment codes and work with responsible organizations to incorporate new information into code models and provide feedback on recommended forensics (as needed).[c]
- ***Task 4 - Additional Workshops/ Expert Panel Input– FY 2016-2020.*** Conduct new survey/workshops to review results and update information inspection needs by industry with expert input (e.g., instrumentation, structure survivability) Document results in Task 1 annual report.
- ***Task 5 – U.S. Requested Inspections or Technology Deployment- FY 2017-2020***. U.S. provides advanced technology to facilitate examinations and sample removal to address information needs or field deployment means of new technology. Document results in Task 1 annual report.

Severe Accident Analyses: In-vessel

- ***Task 1 – Complete Fukushima Uncertainty Analyses – FY 2015:*** Complete analysis of uncertainties on severe accident progression using the MELCOR systems code as applied to Fukushima Daiichi. This is a follow-on activity to the initial work using general Monte-Carlo analyses.
- ***Task 2 – Complete MAAP-MELCOR Crosswalk Phase2 – FY 2016-2017:*** This involves a continued collaboration between EPRI, NRC, DOE laboratories, and possibly with international partners. This second phase of the crosswalk will utilize an accident scenario that is similar to the TMI-2 severe accident as the focus, where operator actions during a severe accident follow SAMG guidance. DOE, NRC and EPRI will jointly fund this activity during this two-year period.
- ***Task 3 – Confirm SAMG Actions with Severe Accident Analysis and Uncertainties – FY 2016-2020:*** Define a series of key severe accident scenarios (FY 2016), determine the accident signatures from these scenarios (including sensitivities) along with symptom-based operator actions to confirm SAMG mitigating strategy response to severe accident progression (FY 2017-2018) and possibly beyond.
- ***Task 4 – Upgrade BWROG Technical Support Guidelines using Severe Accident Analyses – FY 2016-2020:*** Develop software-based tool as part of technical support guidelines for SAMG support using severe accident analyses from MELCOR calculations to inform this software tool.

c. It is anticipated that organizations responsible for development and maintenance of computer codes used in the evaluations of new information would fund these activities and document results separately from this effort.

Severe Accident Analyses: Ex-vessel

- ***Task 1 – Upgrade and document the spreading model and debris coolability model – FY 2016-2017:*** Complete upgrade and documentation of the spreading model and debris coolability model for industry use.
- ***Task 2 - Analysis to support industry efforts to develop an alternate strategy – FY 2016-2017:*** Complete water management severe accident analysis in support of BWR ex-vessel mitigating strategies.
- ***Task 3 – Conduct debris coolability experiments to validate debris coolability models – FY 2016-2018:*** These experiments are part of an international exercise with support from NRC, EPRI and international partners. The focus of the experiments is to confirm the range of water addition mitigation strategies that will ensure ex-vessel debris coolability.
- ***Task 4 – Incorporate these models into advanced systems computer codes – FY 2018-2020:*** These advanced models would then be appropriate to incorporate into advanced system computer models. This long-term task needs to be re-evaluated after these model upgrades and validation are done.

Accident Tolerant Components

- ***Task 1 – Complete the development of a Turbine-Pump (RCIC) model – FY 2015-2016:*** A quasi-steady and transient RCIC model will be developed based on a first-principles technical approach. This model can then be incorporated into various system codes (RELAP or MELCOR or MAAP) for evaluation and comparison to Fukushima integral accident data. Currently researchers at Texas A&M working with SNL are also using CFD models (STAR-CCM+ and FLUENT) to validate certain portions of the thermodynamic analytical model for certain aspects; e.g., liquid carryover effects.
- ***Task 2 – Planning for Terry Turbine Experiments – FY 2015-2016:*** Provide initial system planning into possible testing of a single-stage turbine-pump system under beyond design basis conditions. This effort will be in collaboration with industry (EPRI and BWROG) and international partners. EPRI will be the technical lead to determine scope, objectives and program plan if testing is to occur.
- ***Task 3 – Terry Turbine Experiments – FY 2016-2020:*** This effort will be determined with EPRI and industry guidance and a program plan will need to be further developed upon industry consultation.