ANL/LWRS-16/02

Light Water Reactor Sustainability Program

US Efforts in Support of Examinations at Fukushima Daiichi – 2016 Evaluations

August 2016

U.S. Department of Energy

Office of Nuclear Energy

ANL/LWRS-16/02

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August 2016

Prepared for the

U.S. Department of Energy

Office of Nuclear Energy

ABSTRACT

Although it is clear that the accident signatures from each unit at the Fukushima Daiichi Nuclear Power Station (NPS) [Daiichi] differ, much is not known about the end-state of core materials within these units. Some of this uncertainty can be attributed to a lack of information related to cooling system operation and cooling water injection. There is also uncertainty in our understanding of phenomena affecting: a) in-vessel core damage progression during severe accidents in boiling water reactors (BWRs), and b) accident progression after vessel failure (ex-vessel progression) for BWRs and Pressurized Water Reactors (PWRs). These uncertainties arise due to limited full scale prototypic data. Similar to what occurred after the accident at Three Mile Island Unit 2, these Daiichi units offer the international community a means to reduce such uncertainties by obtaining prototypic data from multiple full-scale BWR severe accidents.

Information obtained from Daiichi is required to inform Decontamination and Decommissioning activities, improving the ability of the Tokyo Electric Power Company Holdings (TEPCO) to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. This document reports recent results from the US Forensics Effort to use information obtained by TEPCO to enhance the safety of existing and future nuclear power plant designs. This Forensics Effort, which is sponsored by the Reactor Safety Technologies Pathway of the Department of Energy Office of Nuclear Energy Light Water Reactor (LWR) Sustainability Program, consists of a group of US experts in LWR safety and plant operations that have identified examination needs and are evaluating TEPCO information from Daiichi that address these needs. Examples presented in this report demonstrate that significant safety insights are being obtained in the areas of component performance, fission product release and transport, debris end-state location, and combustible gas generation and transport. In addition to reducing uncertainties related to severe accident modeling progression, these insights are being used to update guidance for severe accident prevention, mitigation, and emergency planning. Furthermore, reduced uncertainties in modeling the events at Daiichi will improve the realism of reactor safety evaluations and inform future D&D activities by improving the capability for characterizing potential hazards to workers involved with cleanup activities.

ACKNOWLEDGEMENTS

Successful preparation of this report required input and support from several individuals and organizations. Financial support for present and former national laboratory panel participants as well as the Reactor Safety Technologies Pathway leader in the US Department of Energy Office of Nuclear Energy Light Water Reactor Sustainability program was provided through that program office. In addition, there were substantial in-kind contributions made by various industry organizations that supported technical experts to participate in this process; these organizations included the Boiling Water Reactor Owners Group, the Electric Power Research Institute, Exelon Corporation, GE-Hitachi, the Institute of Nuclear Power Operations, Jensen Hughes, the Nuclear Energy Institute, the Pressurized Water Reactors Owners Group, Southern Nuclear, and the Tennessee Valley Authority. Finally, the US Nuclear Regulatory Commission (US NRC) and the Tokyo Electric Power Company Holdings (TEPCO) provided technical experts to participate in the panel meetings. In particular, Mr. Takahasi Hara, Dr. Shinya Mizokami, Kenji Tateiwa, Mr. Daichi Yamada, and Mr. Yasunori Yamanaka from Tokyo Electric Power Company Holdings attended, as well as many individuals from the US NRC Office of Nuclear Regulatory Research, such as Dr. Sudhamay Basu and Dr. Richard Lee. These individuals facilitated the overall process by providing key clarifications in various areas as the meetings progressed. These efforts are greatly appreciated.

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ACRONYMS

ADS	Automatic Depressurization System
AE	Accident Evaluation
AFW	Auxiliary Feed Water
AM	Accident Management and Prevention
ANL	Argonne National Laboratory
BDBEE	Beyond Design Basis External Events
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CAM	Containment Air Monitor
CCI	Core Concrete Interactions
CNWG	Civil Nuclear Energy and Research Working Group
CRIEPI	Central Research Institute of Electric Power Industry
CSNI	Committee on the Safety of Nuclear Installations
Daiichi	Fukushima Daiichi Nuclear Power Station
D&D or DD	Decontamination and Decommissioning
DOE	Department Of Energy
DOE-EM	Department of Energy Office of Environmental Management
DOE-NE	Department of Energy Office of Nuclear Energy
DW or D/W	DryWell
ELAP	Extended Loss of AC Power
ENSREG	European Nuclear Safety Regulators Group

EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EU	European Union
FAI	Fauske and Associates, LLC
FCT	Fukushima Central Television Company, LTD
FDW	FeeDWater
FE-SEM	Field Emission Scanning Electron Microscopy
FLEX	Diverse and Flexible Mitigation Capability (for accident mitigation)
FP	Fire Protection
FP	Fission Product
FSG	FLEX Support Guideline
FY	Fiscal Year[a]
GEH	GE-Hitachi Nuclear Energy, Limited
HP	Hold Points
HPCI	High Pressure Coolant Injection
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
INL	Idaho National Laboratory
INPO	Institute of Nuclear Power Operations
IRID	International Research Institute for Nuclear Decommissioning
IRM	Intermediate Range Monitor
JANTI	Japan Nuclear Technology Institute

[a] In the US, the FY runs from Oct 1 through September 30; in Japan, the FY runs from April 1 through March 31

JAEA	Japan Atomic Energy Agency
JNES	Japan Nuclear Energy Safety Organization
LP	Low Pressure
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interactions
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
MHI	Mitsubishi Heavy Industry
MPR	Mandil, Panoff, and Rockwell (MPR Associates, Inc.)
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NPS	Nuclear Power Station
NRA	Nuclear Regulatory Authority (Japan)
NRC	Nuclear Regulatory Commission
NTTF	Near Term Task Force
NUGENIA	NUclear GENeration II & III Association
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PCV	Primary Containment Vessel

PLR	Primary Loop Recirculation
PM	Plant Maintenance
POC	Point-Of-Contact
PORV	Pilot-Operated Relief Valve
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R&D	Research and Development
RB or R/B	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCW	Raw Cooling Water
RN	RadioNuclide
RPV	Reactor Pressure Vessel
RPI	Rensselaer Polytechnic Institute
RST	Reactor Safety Technologies
SAG	Severe Accidence Guidance
SAMG	Severe Accident Management Guideline
SAREF	SAfety REsearch opportunities post-Fukushima
SARNET	Severe Accident Research NETwork
SAWA	Severe Accident Water Addition
SC or S/C	Suppression Chamber
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SNL	Sandia National Laboratory
SRV	Safety Relief Valve

SSC	Structures, Systems, and Components
TAMU	Texas A&M University
TBR	Technical Basis Report
TC	Thermocouple
TEPCO	Tokyo Electric Power Company Holdings
TIP	Traveling In-core Probe
TMI-2	Three Mile Island Unit 2
TVA	Tennessee Valley Authority
US	United States
UW	University of Wisconsin
VIP	Vessel and Internals Program
WWBX	Willis Walter BiXby (WWBX), LLC
XRD	X-Ray Diffraction
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4

US Efforts in Support of Examinations at Fukushima Daiichi – 2016 Evaluation Results and Updated Information Requests

1. INTRODUCTION

The Great East Japan Earthquake of magnitude 9.0 and subsequent tsunami that occurred on March 11, 2011 led to a multi-unit severe accident at the Fukushima Daiichi Nuclear Power Station [Daiichi]. Although it is clear that the accident signatures from each unit at Daiichi differ, much is not known about the end-state of core materials within these units. Some of this uncertainty can be attributed to a lack of information related to cooling system operation and cooling water injection. There is also uncertainty in our understanding of phenomena affecting: a) in-vessel core damage progression during severe accidents in boiling water reactors (BWRs), and b) accident progression after vessel failure (ex-vessel progression) for BWRs and Pressurized Water Reactors (PWRs). These uncertainties arise due to limited full scale prototypic data. Similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2),[1] these Daiichi units offer the international community a means to obtain prototypic severe accident data from multiple full-scale BWR cores related to fuel heatup, cladding and other metallic structure oxidation and associated hydrogen production, fission product release and transport, and fuel/structure interactions from relocating fuel materials. In addition, these units may offer data related to the effects of salt water addition, vessel failure, containment failure, and ex-vessel core/concrete interactions (CCI). As documented in this report, much of the information obtained from these units will not only reduce uncertainties in BWR severe accident progression but also may offer the potential for safety enhancements for BWRs, PWRs, and future nuclear power plant designs. Furthermore, reduced uncertainties in modeling the events at Daiichi will improve the realism of reactor safety evaluations and inform future Decontamination and Decommissioning (D&D) activities by improving the capability for characterizing potential hazards to workers involved with cleanup activities.

1.1 Objectives and Limitations

The Reactor Safety Technologies (RST) Pathway of the Department of Energy Office of Nuclear Energy (DOE-NE) Light Water Reactor (LWR) Sustainability Program is sponsoring the US Forensics Effort with the following objectives:

- ***Objective 1:*** Develop consensus US input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of Tokyo Electric Power Company Holdings (TEPCO) D&D plans for Daiichi.

- ***Objective 2:*** Evaluate obtained information to:
  - Gain a better understanding related to events that occurred in each unit at Daiichi
  - Gain insights to reduce uncertainties in predicting phenomena and equipment performance during severe accidents
  - Provide insights beneficial to TEPCO D&D activities
  - Confirm and, if needed, improve guidance for severe accident prevention, mitigation, and emergency planning
  - Update and/or refine Objective 1 information requests.

As indicated above, there are several potential safety benefits from this Forensics Effort. In fact, as discussed in [2,3] and within this document, the US has already gained significant safety benefit from information obtained from the affected reactors at Daiichi.

Although there are many potential benefits to be obtained from the US Forensics Effort, it is also important to recognize its limitations. As discussed below, other organizations have activities underway to address these limitations.

First, other organizations within the US have the role of implementing institutional measures to ensure prevention of severe accidents. For example, as discussed in Section 2.2.1.2, the US Nuclear Regulatory Commission (US NRC) established the Fukushima Near Term Task Force (NTTF) and Japan Lessons Learned activities to ensure that appropriate near-term regulatory actions were taken after the events at Fukushima. Areas where the Commission concluded that regulatory actions were required, such as the re-evaluation of hazards associated with flooding and seismic events and training of plant and agency personnel, are underway.

Second, within the US, the industry leads the implementation of safety measures in response to insights from Fukushima. For example, as discussed in Section 2.2.1.3, industry has implemented the diverse and flexible coping strategies or FLEX program to address concerns related to events associated with extended loss of AC power (ELAP) conditions.

Third, it is beyond the scope of the US DOE Forensics Effort to develop an international program. However, it is recognized that information gained from Daiichi is of benefit to global nuclear reactor safety. Ultimately, a long-term international framework, led by Japanese organizations, may be needed to support post-accident examinations at Daiichi.

1.2 Motivation

Data, models, and insights from post-accident inspections at Daiichi will inform many aspects of reactor safety, including severe accident modeling and simulation tools, severe accident management guidelines, improved plant training, and new or revised safety requirements in response to Fukushima. Technologies developed and lessons learned from such information can be used to prevent or mitigate future accidents. To increase the benefit from post-accident examinations that support D&D endeavors, an effort is needed to (a) identify data needs to ensure that key information is not lost; (b) identify examination techniques, sample types, and evaluations to address each information need; and (c) when necessary, help finance acquisition of the required data and conduct of the analyses. Results from the US Forensics effort are beneficial to the US and to Japan.

For the US, this effort provides access to prototypic data from three units with distinctively different accident signatures. In particular, US experts are interested in examination information with respect to:

- *Component Performance and System Survivability Assessments* - Examinations provide key information related to the performance of structures, systems, and components at each unit. For example, many improvements were made to plant instrumentation after the TMI-2 accident.[4] However, the events at Daiichi suggest that additional evaluations may be needed to ensure that operators have adequate information to assess the status of the plant and the effects of mitigating actions that may be taken.
- *Enhancements to Accident Progression and Source Term Models* – Similar to the processes that occurred with TMI-2 examinations, knowledge gained from examinations at Daiichi is being used to reduce uncertainties in systems analysis codes, such as the Modular Accident Analysis Program (MAAP) code and the Methods for Estimation of Leakages and Consequences of Releases

(MELCOR) code.[5, 6] These codes are used both domestically and internationally to evaluate the safety of operating plants, as well as new nuclear reactor designs.

- *Accident Management Strategies and Plant Staff Training* – As uncertainties in predicting BWR and PWR accident progression and associated source terms are reduced, strategies for mitigating severe accidents can be improved. Knowledge gained from Daiichi has and will continue to be factored into accident management guidance and staff training to prevent or reduce the consequences of future accidents.
- *Preserving Severe Accident Capabilities* - Examinations provide exciting and important research opportunities that can serve as a springboard for rekindling much needed expertise within the younger generation of US nuclear engineers regarding LWR severe accident behavior.

For Japan, US involvement provides an independent evaluation of inputs to D&D activities. Such evaluations are useful because of US experience with respect to:

- *Plant Operations* – The US has over 20 operating BWRs, and personnel with considerable experience with respect to BWR operations.
- *Reactor Safety* - Leads in the US Forensics Efforts are also leads for development of US severe accident codes, such as MAAP and MELCOR, as well as large-scale US experimental programs.
- *TMI-2 Post-Accident Examinations and Cleanup* – Several US experts participating in this program were also involved in TMI-2 post-accident evaluations.

Unique US expertise provides TEPCO an independent assessment of their progress reports, the adequacy of severe accident analysis code models for evaluations to support their D&D plans, and the adequacy of available examination information and proposed plans for additional examinations. In the latter case, US input focusses on the desired amount of information, the resolution of data required from sampling, and the cost versus the benefit of obtaining such information. As discussed in Section 2.1, the US devoted significant funding for extraction of radioactive samples of core debris from the TMI-2 vessel and evaluating these samples in hot cells. These efforts provided insights about the chemical composition and porosity of core debris, and results were substantiated with separate effects tests. Although such evaluations from the core region improved our understanding of melt progression, it is less clear that results from relocated core debris samples obtain from the lower head were as beneficial. Such insights are useful to Japan.

Because of the benefit to global nuclear reactor safety, it is recognized that an international framework may ultimately need to be established to support post-accident examinations. If such an international framework is established, it must be led by Japanese organizations. Nevertheless, the US has a vested interest in these examinations. The US has the largest number of operating nuclear power plants in the world; there are also a significant number of reactors operating around the world based on US plant designs. Hence, US organizations – both industry and government—are major beneficiaries from any improvements in LWR severe accident knowledge just as the US was a major beneficiary of significant Japanese participation in prior international TMI-2 programs. US collaborative work with the international community in establishing this framework to support our Japanese colleagues would also be beneficial not just to the US and Japan, but would offer the potential to advance reactor safety across the global nuclear energy community.

1.3 Approach

This section describes the approach developed to ensure that objectives outlined in Section 1.1 are achieved. As discussed within this section, actions taken during FY2015 to complete Objective 1 differed from activities initiated during FY2016 to attain Objective 2. Findings and conclusions from activities to meet Objective 1 are also summarized in this section.

1.3.1 Objective 1 Activities and Findings

To complete Objective 1, expert panel meetings were held in 2015 to develop consensus input related to the higher priority time-sequenced examination tasks. Over 30 experts from industry, universities, and national laboratories participated in this process. Experts from the US NRC, the US DOE, and TEPCO also attended and informed participants during these meetings. This effort resulted in a report [7] with a prioritized list of information of interest to US stakeholders. In this report, special attention was devoted to identifying why such information is important and how it will be used to benefit the US nuclear enterprise. In addition, preliminary cost and schedule estimates for near term tasks (i.e., tasks that should be started within the next five years) were included. As discussed in [7], cost and schedule estimates were obtained by working with TEPCO to understand if such information was already available or planned to be obtained as part of their D&D efforts and what additional effort would be needed. This report was vetted by experts contributing to this process, by experts from government agencies observing in this process, and other relevant stakeholders.

During these meetings to complete Objective 1, US experts agreed upon several significant findings:

- Information obtained from the affected reactors at Daiichi offers a unique means to obtain full-scale, prototypic data for enhancing reactor safety (e.g., improved severe accident guidance, possible plant modifications, improved simulation codes for staff training, etc.).
- Insights gained from collecting and comparing similar observations and data from each of the three units are valuable because the accident progression at each unit was unique in many respects.
- This information is important for BWRs and PWRs; i.e., many insights gained from this information are not only applicable to BWRs, but also could have significant impacts for enhancing PWR safety.
- Some information is required for all identified items to obtain a complete picture of the events. It is only meaningful to prioritize data needs with respect to the 'cost' and 'logical sequence' for obtaining such information.
- Information from other units at Daiichi and other plants, such as the Daini plant, also provide valuable insights for forensics, repair, maintenance, and field applications. Critical information from these plants can be more easily obtained at lower cost and with less radiation exposure to personnel.
- TEPCO D&D plans (or activities already completed) address much of the information identified by the US expert panel.
- Maximum benefits from this information requires: reviews by cognizant experts, posting for easy-to-use access (e.g., a website with searchable database features), interactions with TEPCO for added requests and understanding of information available, and interactions with code assessments.
- Ultimately, an international framework should be established to benefit from information obtained during TEPCO's D&D efforts at Daiichi.
- Important information and data are already available, and more is being gathered at the current time. US Forensics Evaluation tasks should be initiated as soon as possible

Most of the information needs identified by the expert panel are related to the affected units at Daiichi Units 1 to 4 (1F1, 1F2, and 1F3).[b] Although details varied, US experts generally identified information needs required to answer fundamental questions related to how the accident progressed in each unit, understand equipment and component survivability, and benchmark severe accident progression and dose assessment codes. These needs are organized in Reference [7] tables according to location [e.g., the reactor building (RB), the primary containment vessel (PCV), and the reactor pressure vessel (RPV)]. These tables also identify the applicable units for each information need and other relevant factors (e.g.,

[b] Because of the hydrogen explosion damage observed at Unit 4 (1F4), this unit is also of interest.

how information should be obtained, why it is needed, its expected use or benefits, when it should be done, and the estimated level of effort).

Table 1 summarizes, at a high level, the activities identified by the expert panel for addressing information needs from the affected units at Daiichi. As indicated above, the expert panel concluded that some information is needed from all locations to obtain a complete picture of the entire accident progression and conditions that occurred in each unit during these events. Therefore, the expert panel concluded that information needs were best prioritized with respect to cost and the logical sequence for obtaining such information. For each location, Table 1 groups the desired examination information by method and specifies the priority of the information need by the number of asterisks in each box. Results indicate that the expert panel typically placed the most emphasis upon information obtained from visual examinations, such as videos and photographs, and near-term proximity exams, such as dose surveys. In general, the consensus was that such information was the easiest to obtain, and could provide critical information related to whether additional examinations were required.

Table 1. Prioritization of possible examination activities

Region	**Examination Information Classification**[c,d]			
	Visual	**Near-Proximity**	**Destructive**	**Analytical**
Reactor Building (RB)				
Reactor Core Isolation Cooling (RCIC)	****	***	**	
High Pressure Coolant Injection (HPCI)	****		***	
Building	****	***	**	*
Primary Containment Vessel (PCV)				
Main Steam Isolation Valves (MSIVs) and Safety Relief Valves (SRVs)	****		***	
Drywell (DW) Area	****	***	**	*
Suppression Chamber (SC)	****	***		
Pedestal / RPV-lower head	****		***	**
Instrumentation		****	***	
Reactor Pressure Vessel (RPV)				
Upper Vessel Penetrations	****		***	**
Upper Internals	****	***	**	*
Core Regions & Shroud	****		***	**
Lower Plenum	****		***	**

[c] ***Examination Classification Examples:***

Visual– Videos, Photographs, etc.

Near-Proximity– Radionuclide Surveys, Seismic Integrity Inspections, Bolt Tension Inspections, and Instrumentation Calibration Evaluations

Destructive– System or Component Disassembly, Sampling, etc.

Analytical– Chemical Analysis, Metallurgical Analysis, Gamma Scanning, etc.

[d] **Prioritization based on number of asterisks, e.g., more asterisks designate a higher priority on this information.**

Other important conclusions are that much information is already available and that efforts should immediately begin to assess if available information is sufficient to address the identified need (and make additional requests, if required). As discussed in Section 1.3.2, US experts are focusing on information related to areas identified as higher priority, the near-term availability of information, and the importance of the information for satisfying Objective 2.

1.3.2 Activities to Complete Objective 2

The activities used to complete the second objective are shown in Figure 1. As shown in this figure, activities and products completed by US organizations are shown in purple and focus on Phase 2 Activities associated with the Mid-and-Long-Term Roadmap for D&D (the blue box; see Section 2.3). As indicated by the orange box, severe accident and plant operations experts from US industry, universities, and national laboratories evaluate plant examination information obtained from Daiichi. Objective 2 activities were also informed by experts from the US NRC, US DOE, and TEPCO that participated in expert panel meetings.

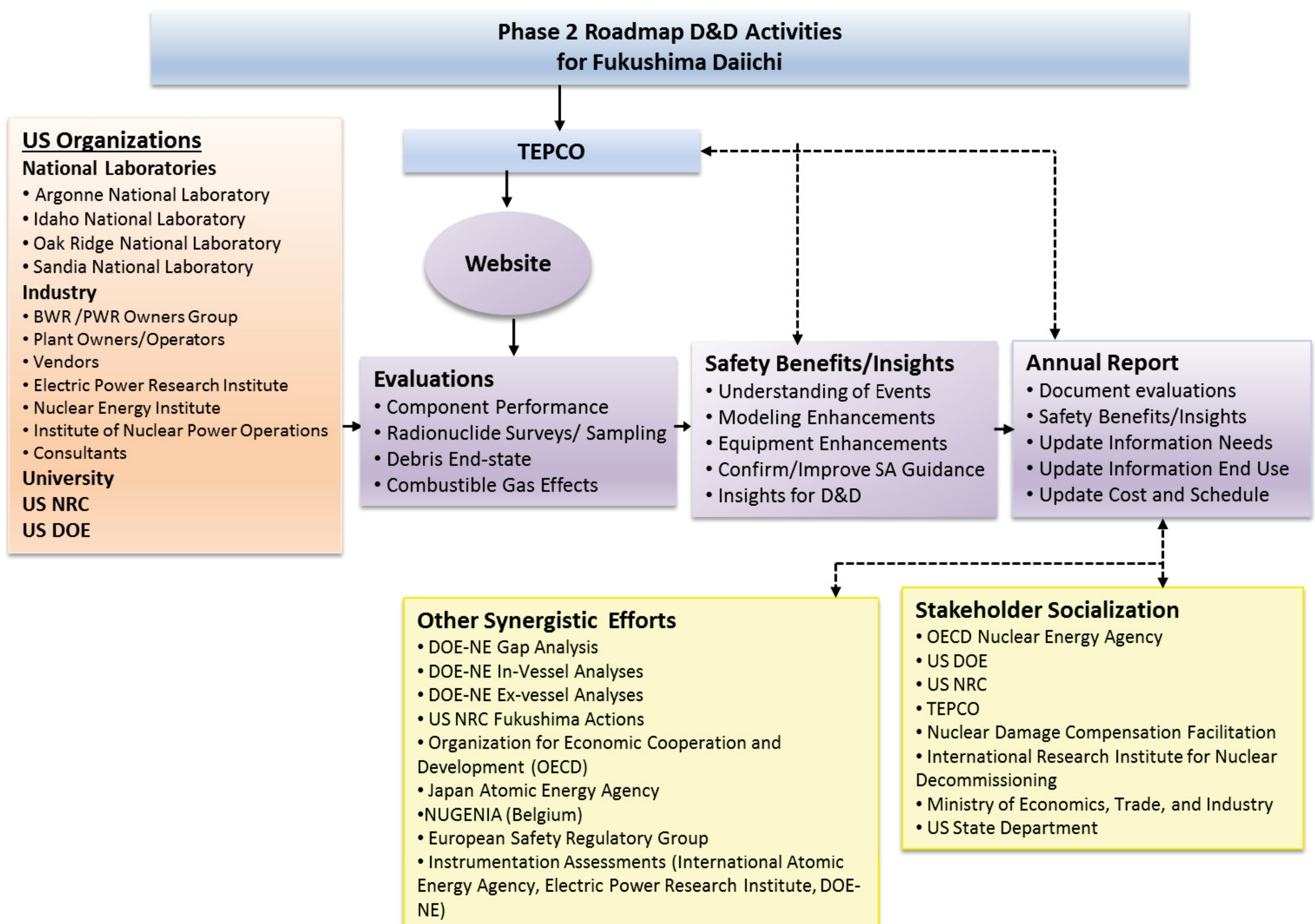

Figure 1. Objective 2 activities.

Table 2 lists specific organizations represented by experts at RST Expert Panel Meetings. Specific individuals participating in expert meeting during FY2016 Forensics Expert Panel meetings are listed in Appendix A of this report. Since its origin, the forensics effort has strived to include a broad spectrum of US stakeholder input.

Table 2. Organizations represented in expert meetings

Type	Organization
Government Agencies	• US Department of Energy Office of Nuclear Energy (DOE-NE)
	• US Department of Energy Office of Environmental Management (DOE-EM)
	• US Nuclear Regulatory Commission (US NRC)
	• Nuclear Regulatory Agency (NRA)
National Laboratories	• Argonne National Laboratory (ANL)
	• Idaho National Laboratory (INL)
	• Oak Ridge National Laboratory (ORNL)
	• Sandia National Laboratory (SNL)
Plant Owner Groups	• BWR Owner's Group (BWROG)
	• PWR Owner's Group (PWROG)
Plant Owners/ Operators	• Exelon Corporation
	• Southern Nuclear Company
	• Tennessee Valley Authority (TVA)
	• Tokyo Electric Power Company Holdings (TEPCO)
Universities	• Rensselaer Polytechnic Institute (RPI) (only participated in November 2013 meeting)
	• Texas A&M University (TAMU) (only participated in November 2013 meeting)
	• University of Wisconsin - Madison (UW)
Vendors	• AREVA
	• GE-Hitachi (GEH)
	• Westinghouse Electric Corporation
Other	• CANegin & Associates
	• Electric Power Research Institute (EPRI)
	• Fauske and Associates, LLC (FAI)
	• Institute of Nuclear Power Operations (INPO)
	• Jensen Hughes (formerly ERIN Engineering and Research, Incorporated)
	• MPR Associates, Inc. (only participated in November 2013 meeting)
	• Nuclear Energy Institute (NEI)
	• Lutz Nuclear Consulting
	• Rempe and Associates, LLC
	• WWBX Consulting, LLC

Evaluations focus on available information related to four higher priority topic areas. Activities and products completed by US organizations are shown in purple. Severe accident and plant operations experts from the US evaluated information from the higher priority topic areas identified by the expert panel. These areas are:

- Component /System Performance
- Radiological Sampling and Surveys
- Core Debris End-state
- Combustible Gas Effects.[d]

In this effort, the primary source of information used in the evaluations was the TEPCO website.[8] Presentations provided by representatives from TEPCO,[9 through 16], industry[2, 17], and topic area leads[e.g., 18 through 20] and TEPCO reports documenting unconfirmed and unresolved issues received special attention in the forensics effort.[21 through 24] A website, containing a searchable database (see Appendix B), is being developed that archives information from TEPCO and other sources used to complete these evaluations.

As previously discussed, these evaluations lead to several types of safety benefits and insights:

- Increased understanding of the events that occurred at each of the affected units at Daiichi
- Enhanced severe accident analysis models (reduced severe accident modeling uncertainties)
- Increased understanding of equipment performance during severe accidents
- Confirmed / improved guidance for severe accident prevention, mitigation, and emergency planning.
- Additional insights beneficial to future D&D activities.

As shown in Figure 1, US experts prepare an annual report documenting results from these evaluations and updates related to information needs, end use, and the updated cost and schedule estimates (if needed) for completing future forensics activities. Sections 3 through 6 of this report provide FY2016 results from this process. For each area, prioritized questions of interest are identified; available information is reviewed; and insights gained from evaluating available information are provided. Where appropriate, information needs have been updated, and a complete list of information needs that includes these updates is provided in Appendix C of this report.

1.3.3 Other Considerations

In completing Objective 2 activities, there are other considerations (shown in yellow boxes in Figure 1). These other considerations are important aspects of the Forensics Effort.

The first consideration relates to other synergistic efforts that are discussed in Section 2.2. These other efforts, including those funded by DOE, those completed by NRC, and those organized by other agencies and other organizations, are considered in all Forensics Effort activities. In addition, as discussed in Section 2.2, results from the Forensic Effort support several aspects of these synergistic efforts.

The second consideration relates to interactions with other stakeholders that affect the feasibility of proposed forensics activities. For example, copies of the FY2015 report were provided to the following individuals for comment:

- Doug Chapin, Principal, MPR Associates, Inc.
- Paul T. Dickman, Senior Policy Fellow, ANL; Chair, International Special Advisor, Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)
- Professor Dale Klein, Associate Vice Chancellor for Research, University of Texas

[d] This fourth area was identified in FY2016.

- William D. Magwood IV, Director-General Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD-NEA)
- Richard Meserve, Chairman, DOE Nuclear Energy Advisory Committee, and Member, DOE Secretary of Energy Advisory Board
- Dana Powers, Retired SNL; Member, Advisory Committee on Reactor Safeguards, US NRC

These stakeholders were contacted because it is recognized that the success of this effort requires information in this report to be discussed with and supported by individuals with their specific expertise and organizational affiliation. Comments regarding our FY2015 report that were received from such individuals were extremely beneficial. To the extent possible, their comments were addressed in preparing this FY2016 document.

1.4 Report Objectives and Organization

This report represents the first of a series of anticipated reports to document efforts by US experts to evaluate available inspection data to address information needs in higher priority areas of interest. The balance of this report is organized as follows. Section 2 provides background information related to prior efforts to obtain similar information from the TMI-2 PWR and provides an overview of other synergistic efforts of interest to this Forensics Effort. Section 2 also reviews the organization and schedule for D&D activities within Japan. Sections 3 through 6 summarize insights from FY2016 efforts to evaluate information in the areas of component /system degradation, dose surveys / isotopic surveys and sampling, debris end-state, and combustible gas effects. Each of these sections identifies key questions of interest and insights gained from the information evaluated. Limitations associated with the insights and recommendations related to future RST program activities and examination information are also provided. Section 7 of this report summarizes key insights and recommendations from this effort. In addition, Section 7 identifies how insights and recommendations from this effort are being implemented. References are listed in Section 8. Appendices to this document provide more detailed information. Specifically, Appendix A provides lists of attendees and agendas from Forensics Effort expert meetings held during FY2016. Appendix B provides a description of the website capabilities developed to support this effort. Appendix C provides tables with detailed information needs developed during expert meetings. Appendix D contains roadmaps produced by the Japanese Government that detail planned D&D activities.

2. Background

As part of this project, experts reviewed important aspects of the TMI-2 evaluation process, synergistic activities underway by other US and international organizations, and D&D plans by Japanese organizations. These reviews ensure that current efforts are cognizant of lessons learned from past inspection programs, avoid duplication of other synergistic activities, and are coordinated with on-going plans to D&D the affected reactors.

2.1 TMI-2 Post-Accident Evaluation Process

Post-accident insights related to what occurred at TMI-2 required an integrated set of information that included post-accident videos, examinations of core debris and vessel structure samples, instrumentation data, calculation results from 'best-estimate' severe accident analysis tools, separate effects laboratory test results, and in some cases, data from large integral tests. [1,4,25,26] Video examinations and ultrasonic scanning surveys were initially used to determine the shape, dimensions, and mass of materials remaining in the reactor vessel and the damage sustained by internal support structures and penetrations (see Figure 2). Several types of samples were removed from the reactor pressure vessel, including fuel, cladding, control rods, fuel support structures, and in-core instrumentation nozzles. Samples from within the primary coolant system and the reactor containment building were also obtained. Analyses to interpret and integrate these information sources were crucial because insufficient data were available from any single source to uniquely define a consistent understanding of the TMI-2 accident scenario.

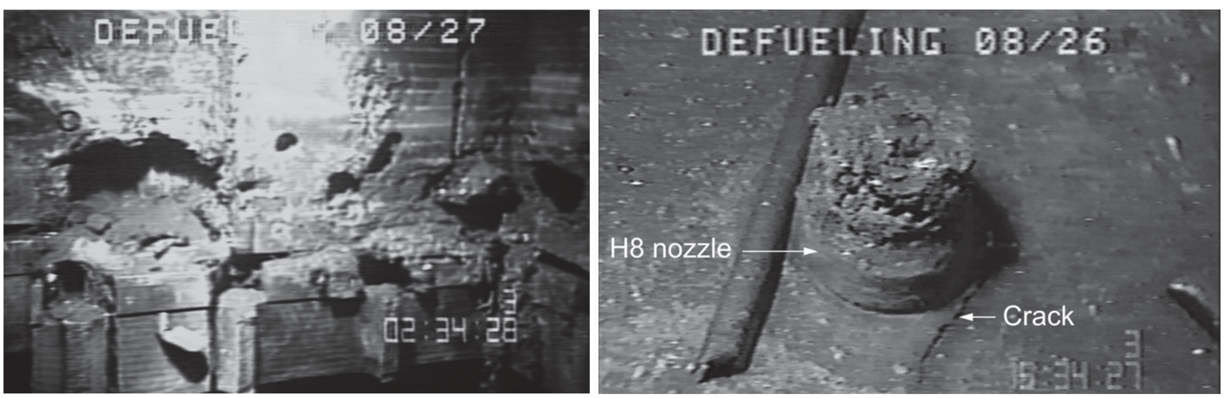

Figure 2. TMI-2 video examinations revealed locations where damage to core barrel and nozzles was more severe. (Courtesy of FirstEnergy)

A systematic investigation of the costs and benefits of TMI-2 inspection information is not available. It is clear that visual inspection information from within the TMI-2 vessel and the containment offered important insights at a lower cost than insights gained from post-accident examinations of radioactive samples. Nevertheless, important insights about the potential for vessel failure were also gained from examinations of vessel steel and nozzles.

Likewise, a systematic investigation of 'lessons learned' from TMI-2 examinations is not available. Such an investigation could provide insights related to the desired number and type of sample measurements, unanticipated hazards associated with D&D activities, the feasibility of advanced sample extraction techniques, and the benefit of separate effects testing. In addition, such evaluations might identify information not obtained from TMI-2 that would be useful to obtain from Daiichi. Nevertheless, TMI-2 experience was applied by US experts in identifying information needs from Daiichi. For example, information needs focused on visual information that could provide important insights at a lower cost (see Section 1.3.1). During FY2017, DOE-NE and the US NRC are co-sponsoring the meeting, "US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting," to transfer knowledge learned from TMI-2 cleanup and recovery activities to Japan.

2.2 Synergistic Efforts

The events at Fukushima have rekindled international interest in LWR severe accident phenomenology. As part of their efforts to address post-Fukushima actions, the US industry and NRC have initiated several efforts in the severe accident area. Furthermore, new activities have been sponsored by the DOE-NE RST Pathway of the LWR Sustainability program. In addition, several international organizations have initiated complementary efforts in this area. To minimize duplication, it is important that the RST pathway remain cognizant of such activities. Table 3 lists synergistic activities that are deemed to be of special interest to the US Forensics Efforts. Section 2.3 describes efforts by Japan to complete D&D activities. This section summarizes the objectives and recent accomplishments of other activities sponsored by US and international organizations.

Table 3. Synergistic activities of special interest[e]

Source	Organization(s)/Countries	Activity/Objective
US	US DOE, Industry, and Universities	Severe Accident Analyses; Complete PWR and BWR severe accident analyses using the industry-developed MAAP code and the NRC-developed MELCOR code. Perform 'crosswalk' to identify differences in predictions for in-vessel and ex-vessel evaluations and root cause for observed differences.
		Gap Analysis; Identify knowledge gaps in experimental data supporting analysis capabilities; prioritize US DOE severe accident research options.
		Accident Tolerant Component Performance; Conduct analysis and experiments on hardware-related issues, including systems, structures and components with the potential to prevent core degradation or mitigate the effects of severe events
	US NRC/Industry	Post-Fukushima Activities; Implement actions to address potential vulnerabilities associated with operating nuclear power plants and associated facilities. Actions informed by MAAP and MELCOR analyses of the affected units at Daiichi and other reactor types.
Japan	NDF, IRID, TEPCO, JAEA	D&D Activities; Complete D&D of affected reactors at Daiichi (see Section 2.3)
	JAEA, MHI, CRIEPI, Universities	Gap Analysis; Identifies gaps in knowledge about the performance of existing safety systems and the need to develop new materials, components, and systems with enhanced performance.
US-Japan	US: US DOE and US NRC JAPAN: METI, MEXT,	CNWG; Collaborative activities related to wide range of research, including examinations, instrumentation, and analyses.
EU	NUGENIA (includes SARNET)	Prioritization Evaluations; Prioritize Research and Development (R&D) topics and use ranking results to 'harmonize' and 'reorient' existing R&D program as well as justify new research topics.
	European Nuclear Safety Regulators (ENSREG)	Stress Tests; Complete reassessments of the safety margins in EU nuclear power plants. Evaluations consider 'extraordinary' external events, such as earthquakes and floods, and the consequences of other initiating events
OECD-NEA	SAfety REsearch opportunities post-Fukushima (SAREF)	Identify Opportunities from Fukushima; Establish process for identifying and following up on research opportunities to address safety research gaps and advance safety knowledge related to the Fukushima Daiichi nuclear accident and support safe and prompt decommissioning in Japan.
	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF)	Severe Accident Analysis; Improve severe accident codes by analyzing the accident progression and current status of 1F1, 1F2, and 1F3; provide useful information for the decommissioning of these units

[e]See acronym list.

2.2.1 US Efforts

As discussed below, synergistic activities performed by the US DOE, the US NRC, and the US industry are of interest to and informed by the US DOE Forensics Effort.

2.2.1.1 US DOE

After the initial response to the events at Daiichi, the US DOE funded high priority safety research activities with the goals of gaining a more thorough understanding of the events that occurred at Daiichi, to identify and reduce in-vessel and ex-vessel severe accident modeling uncertainties, especially with respect to BWR phenomena, and to assist industry in refining guidance to prevent significant core damage and to mitigate source term release.

Severe Accident Analyses

Analyses of events occurring in the affected units at Daiichi indicated notable differences in predictions obtained from the industry-developed MAAP and the US NRC-developed MELCOR systems analysis codes.[27 through 29] A cross-walk activity between the MAAP and MELCOR development teams was then completed to determine the principal modeling differences between the two codes that led to such differences in predicting in-vessel core melt progression phenomena. Results indicate that the principal phenomenological uncertainty relates to the extent that degraded core materials are permeable to gas flow. Namely, impermeable debris (assumed in MAAP) gradually accumulates as a large high temperature in-core melt mass similar to that formed during the TMI-2 accident, while permeable debris (assumed in MELCOR) steadily relocates to the lower head and collects as a debris bed. These in-vessel modeling differences lead to significant differences in subsequent severe accident phenomena, such as hydrogen production, the timing and location of vessel failure, and ex-vessel melt spreading phenomena.[30,31] The DOE and EPRI continue to conduct analyses using existing computer models to provide information and insights into severe accident progression.[32] Results from all of these analyses aid in post-Fukushima enhancements to severe accident guidance (SAG) for BWRs and PWRs and training operators on this guidance. In the case of ex-vessel analyses, an on-going ex-vessel core debris coolability test program is being used to gather additional data for validation of severe accident codes.

Gap Analyses

In parallel with these analyses, the DOE conducted a technology gap evaluation on accident tolerant components and severe accident analysis methodologies. The process relied on a panel of US experts in LWR operations and safety with representatives from industry, DOE-NE staff, the national laboratories, and universities. The goals were to: i) identify and rank knowledge gaps, and ii) define appropriate Research and Development (R&D) actions to close these gaps. Representatives from the NRC and the TEPCO participated as observers in this process. Panel deliberations led to the identification of thirteen knowledge gaps on severe accident analysis and accident tolerant components that were deemed to be important to reactor safety and are not being currently addressed by US industry, US NRC, or US DOE. As discussed in [33], these thirteen gaps were classified into five categories; i.e., i) in-vessel core melt behavior, ii) ex-vessel core debris behavior, iii) containment – reactor building response to degraded conditions, iv) emergency response equipment performance, and v) additional degraded core phenomenology.

Results emphasized the need to address data and knowledge gaps in the existing data base for modeling BWR late-phase in-core fuel and structure degradation and relocation, especially with respect to phenomena that affect multiple assemblies. Results from this evaluation provide a basis for refining US DOE research plans to address key knowledge gaps in severe accident phenomenology that affect reactor safety.

Evaluation results also emphasized that information from the damaged Fukushima reactors provides the potential for key insights that could be used to help address virtually all the identified gaps. Information obtained from these units not only offers the potential to fill these gaps and reduce uncertainties in severe accident progression, but may also inform potential safety enhancements. In recognition of the importance of this information, the DOE sponsored the Forensics Effort that is the subject of this report.

Component and System Analyses

Results from the Gap Analyses also emphasize the need to better characterize the performance of several hardware components and safety systems during severe accidents. To address this, the US DOE is working with industry to develop plans for the design and possible operation of a test facility to better determine the actual operating envelope for BWR Reactor Core Isolation Cooling (RCIC) and PWR Auxiliary Feed Water (AFW) Terry Turbine systems under severe accident conditions. As part of this activity, the performance of BWR Safety Relief Valves (SRVs) and PWR Pilot-Operated Relief Valves (PORVs) would also be investigated.

The need for reliable instrumentation was recognized after the TMI-2 event,[4] and the events at Fukushima have again emphasized the importance of operators having access to critical information from plant instrumentation. To address potential measures under consideration by the US NRC [34 through 37], several efforts have been sponsored by the US DOE[38,39] and industry groups [40 through 43] on this topic.

2.2.1.2 US NRC

In their initial response to the events at Fukushima, the NRC initiated an intensive 90-day effort to document insights (as they were known at that time) and make recommendations for enhancing the plant capability to respond to Beyond Design Basis External Events (BDBEE).[44] The report contained twelve (12) high level recommendations with each having several unique individual recommendations. To address these recommendations, the NRC Commissioners could require safety enhancements through an Order if there was not adequate protection of the health and safety of the public, or the Commissioners could direct the NRC staff to initiate rulemaking to require safety enhancements. In the latter case, the safety enhancements should be cost beneficial as demonstrated using established processes. [45, 46] The Commission issued Orders EA-12-049 (Mitigation Strategies), EA-12-050 (Hardened Vents), and EA-12-051 (Spent Fuel Instrumentation), as well as a request for information letter to licensees concerning resistance to beyond design basis seismic and flooding events.[47, 48, 49, 50, respectively] These regulatory actions addressed the most important insights from the Fukushima accident. Initially, recommendations related to SAMGs were planned to be addressed in rulemaking.[51] However, in [52], the Commission directed the staff to remove requirements imposing SAMGs from this rulemaking. Rather, the Commission instructed the staff to revise their Reactor Oversight Process, such that the staff would periodically review industry's voluntary implementation of updated and revised Severe Accident Management Guidelines (SAMGs).

As documented in [53,54], the US NRC severe accident research program supplies the agency a strong technical foundation for decision-making related to degraded core phenomena identified in probabilistic risk assessments. Recognizing the uncertainties in severe accident phenomena, the agency relies on computational tools developed from the severe accident research program to consider these uncertainties and estimate the margins that exist in light water reactors during severe accidents. Results obtained from these analyses provide the agency essential input for regulatory decisions. The US NRC continues their severe accident research activities to reduce uncertainties in such input and to assess the importance of new phenomena that may need to be considered in such computational evaluations. Participation in international severe accident research programs for evaluating new phenomena leverages the agency's limited resources and maintains staff expertise on emerging issues. As noted by Lee [55] and Uhle [56], NRC severe accident phenomena expertise informed regulatory actions to address post-Fukushima activities, such as EA-13-109, EA-12-049, and EA-12-051.

2.2.1.3 Industry

In response to the events at Daiichi, industry led efforts within the US to take independent steps to develop diverse and flexible coping strategies for BDBEE, known as FLEX.[57] The focus in the US was clearly on enhancements to guarantee continued core, containment, and spent fuel pool cooling in the event of beyond design basis accidents, particularly those resulting from extreme external events. As part of "The Way Forward,"[58] industry is also enhancing existing SAMGs to reflect insights gained from the Fukushima accident.

Industry developed and documented proposed enhancements and submitted them to NRC for endorsement. These enhancements provided guidance for individual plants concerning acceptable methods for satisfying the issues that led to these NRC Post-Fukushima Orders and recommendations. Industry enhancements include:

- Enhanced mitigation capability for BDBEEs,[57]
- Staffing and communications recommendations,[59]
- Implementation of new spent fuel pool instrumentation,[60]
- Plant walkdowns to ensure adequate flooding protection, [61]
- Reliable containment venting for Mark I and Mark II BWRs,[62]
- Integration of Accident Management Procedures and Guidelines,[63]
- Enhanced Emergency Response Preparedness,[64]
- Seismic evaluation guidance,[65,66] and
- Plans for enhancing SAG.[67]

Enhancements for BDBEEs in the U.S. center around the FLEX concept. FLEX involves strategies to maintain core, containment and spent fuel pool cooling for a wide range of BDBEEs that result in the loss of all a.c. power (onsite and offsite) as well as access to the ultimate heat sink for an indefinite period of time. The strategies rely upon a combination of fixed, in-place and portable equipment protected from BDBEEs. The FLEX concept also involves, staffing, communication, procedures and guidelines, and training to assure that strategies are implemented in a timely manner. FLEX defines three phases of response to a BDBEE: 1) initial response using fixed in-place capabilities until portable resources can be implemented, 2) portable onsite resources that are adequate until offsite equipment can be brought to the site and implemented, and 3) portable offsite resources at one of two national centers [67] that can be deployed to a site within 24 hours.

Revisions to BWROG and PWROG severe accident management guidance considered available information from Daiichi. As documented in [2], some of the insights based on events at Daiichi include:

- Hydrogen combustion can occur in structures adjacent to the primary containment,
- Primary containment integrity can be challenged when conditions exceed its design basis,
- Water injection to the reactor vessel should be preferred over injection to the primary containment,
- Primary containment venting will assure long term control of fission product releases, and
- Turbine driven pumps can be operated in extreme beyond design basis conditions.

As discussed within this report, the basis for each of these insights was drawn from forensic evidence reviewed by the US Forensics Expert Panel.

2.2.2 International

The response to the Fukushima accident has been global, resulting in multiple activities by numerous international stakeholders. Post Fukushima-related topics, such as accident mitigation strategies, accident monitoring systems, and overall reactor safety have been the focus of international working groups and meetings sponsored by various agencies, such as the International Atomic Energy Agency (IAEA), and the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD). In addition, associations and groups such as NUclear GENeration II & III Association (NUGENIA) and the European Nuclear Safety Regulators Group (ENSREG) are focusing on the same safety-related areas. To avoid duplication of effort, it is important that the US RST program remain cognizant and informed by these efforts. Selected activities of special interest are summarized below.

2.2.2.1 Japan

Clearly, the D&D activities underway in Japan are of interest to US DOE efforts. Section 2.3 of this report provides the organizational structure and current roadmap for completing these activities. In order for the US efforts to be successful (and to minimize the impact of inspection activities), it is critical that the US remain cognizant of Japanese plans for completing D&D activities and of results from these activities. Furthermore, it is important that the US program provide timely input to Japan related to their experiences from D&D activities completed at TMI-2 and results from safety evaluations.

Gap Analysis

The Atomic Energy Society of Japan [68] recently completed a severe accident gap analysis within Japan. This evaluation focused on quantifying limitations of current systems and identifying research to overcome the limitations of current reactors. Twelve prioritized research topics were selected using input from the Japan Atomic Energy Agency (JAEA), Toshiba, Hitachi-GE Nuclear Energy, Mitsubishi Heavy Industry (MHI), Central Research Institute of Electric Power Industry (CRIEPI), and several universities (University of Tsukuba and Kyoto University). Identified research areas include: development of new reactor materials (e.g., cladding and core catcher); evaluations of the performance of systems, such as the Passive Containment Cooling System, Autocatalytic Recombiners, Hydrogen Removal Systems, and Filter Venting Systems; and development of new instrumentation and measurement devices that can survive severe accident conditions. Clearly, there are opportunities for collaboration between JAEA and the US DOE activities to address these gaps. As discussed below, some of these opportunities are covered under existing bilateral agreements between Japan and the US.

CNWG

A Civil Nuclear Energy Research and Development Working Group (CNWG) 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 (METI) and Ministry of Education, Culture, Sports, Science and Technology (MEXT).[69] Formal arrangements have been established covering collaboration in multiple areas including several relevant to LWR safety and post-accident evaluation [70]; namely, i) severe accident code assessment, ii) accident tolerant fuel, iii) accident tolerant equipment (including instrumentation), and iv) probabilistic risk assessment. Bilateral collaboration is underway in these areas. In 2016, it was agreed to include collaboration in the area of reactor examination planning as it relates to informing D&D activities within Japan.

2.2.2.2 OECD/NEA

The OECD/NEA has been proactive in sponsoring efforts to ensure that the international community is aware of safety insights from the events at Fukushima.[71] Current activities of special interest to the US DOE RST pathway are highlighted in this section.

BSAF Analyses

An ongoing parallel analysis activity is the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) project.[72] The project, which is hosted by JAEA and other Japanese organizations, is an international effort aimed at performing accident reconstruction analyses using a number of severe accident codes, including MELCOR and MAAP. The objective of the project is to improve severe accident codes, to analyze the accident progression and current status of 1F1, 1F2, and 1F3, and provide useful information for the decommissioning of these units. The reconstruction analyses make use of known accident boundary conditions and measurements, such as estimated water injections, operations of emergency equipment [e.g., RCIC, HPCI, etc.], reactor depressurization actions, and containment venting actions. These analyses compare results from a collection of international severe accident analysis codes and provide analytical insights into the estimated damage state of each reactor. Characterization of the damage states includes estimates of the melted core regions, the mass of relocated core materials to the lower head, possible pressure vessel failure locations (e.g., lower head or steam line), and the amount of reactor cavity concrete attack by molten core materials. Hence, results from these analyses can help inform decommissioning activities by providing estimates of core relocation masses and inform data needs that may be addressed during D&D activities. In return, examination and photography of upper reactor vessel internals and steam lines can provide valuable information for validating code estimates of damage in these regions. The first phase of the BSAF project, which focused mainly on the accident progression and core damage phase, has been completed. Phase 2 of the BSAF project, which started in June 2015, is aimed at characterizing release and transport of fission products through the reactor vessel, containment, and reactor building, and ultimately, to the environment. Environmental releases will include both aqueous pathways as well as atmospheric releases. Validation information will be sought from sampling of radiological depositions along these release pathways, including the ground deposition data for cesium in the countryside around the accident site. Jäckel[73] illustrates the type of information from the Daiichi site that will be used in this effort and conclusions that can be obtained from these evaluations. Phase 2 of this project is anticipated to proceed over the next three years. DOE and NRC participate in this NEA project. This participation is important because BSAF analysis results inform on-going DOE activities in evaluating and improving severe accident analysis models. In addition, results from inspection activities inform ongoing BSAF and US DOE funded analyses, and analyses results may lead to revisions in US information needs.

SAREF Research Opportunities from Fukushima

Another noteworthy effort is underway by the NEA's senior expert group on SAfety REsearch opportunities post-Fukushima (SAREF). Created in 2013 by the Committee on the Safety of Nuclear Installations (CSNI), the objective of this effort is to establish a process for identifying and following up on research opportunities to address safety research gaps and advance safety knowledge related to the Fukushima Daiichi nuclear accident and support safe and prompt decommissioning activities in Japan.[74] Organizations from twelve countries are participating in this activity. The work scope includes identifying research opportunities based on information from Daiichi that will provide additional safety knowledge of common interest to the member countries. The SAREF identified 16 specific topics of interest in four main areas; namely i) severe accident progression, ii) structural/material behavior, iii) structure, system, and component (SSC) performance, and iv) accident recovery. Activities are underway to further refine research recommendations for submission to the CSNI by June 2016. DOE and NRC are participants in this NEA project. Clearly, it is important for the US DOE RST effort to be cognizant of and contribute to this effort. Ultimately, results from SAREF may lead to the establishment of a potential international examination effort in which the US will participate.

2.2.2.3 European Union (EU)

NUGENIA/SARNET Research Prioritization

NUGENIA is an association dedicated to the research and development of nuclear fission technologies, with a focus on Generation II and III nuclear plants. Primarily composed of organizations based in Europe, it includes stakeholders from industry, research, and safety organizations. Synergistic activities sponsored by NUGENIA[75] originate within the Severe Accident Research NETwork (SARNET), which has the objectives of:

- Improving knowledge on severe accidents in order to reduce uncertainties on pending issues, thereby enhancing plant safety,
- Coordinating research resources and expertise available in Europe, and
- Preserving the research data and disseminating knowledge.

Participants in SARNET include representatives from 47 organizations; although most organizations are based in Europe, there are organizations from Korea, India, Japan, and the US (e.g., the NRC). Of particular interest to the US DOE Forensics Effort are results from SARNET efforts to prioritize research programs. As discussed within [76], recent SARNET evaluations ranked the six highest priority safety issues as: in-vessel core coolability, molten-core-concrete-interaction (MCCI), fuel-coolant interaction, hydrogen mixing and combustion in containment, impact of oxidizing conditions on source term, and iodine chemistry. Similar to the US DOE strategy, SARNET uses this ranking to 'harmonize' and 'reorient' existing R&D programs and justify new research topics. Through the NRC, the US collaborates on many EU higher priority research projects (see Section 2.2.1).

ENSREG Stress Tests

The European Nuclear Safety Regulators Group (ENSREG) is an independent, authoritative expert body created in 2007 following a decision of the European Commission. It is composed of senior officials from the national nuclear safety, radioactive waste safety or radiation protection regulatory authorities and senior civil servants with competence in these fields from all 28 Member States in the European Union (EU) and representatives of the European Commission. ENSREG's role is to help to establish the conditions for continuous improvement and to reach a common understanding in the areas of nuclear safety and radioactive waste management.

ENSREG [77] efforts to complete follow-on activities related to "stress tests" on EU nuclear power plants are of interest to US efforts to perform severe accident analyses and system performance evaluations. These stress tests, which were requested in March 2011, are targeted reassessments of the safety margins in nuclear power plants. They consider 'extraordinary' external events, such as earthquakes and floods, and the consequences of other initiating events, such as airplane crashes, that have the potential to lead to loss of multiple safety functions. All operators of nuclear power plants in the EU had to review the response of their nuclear plants to those extreme situations. The operators' reports were first reviewed by the national nuclear regulators. The operators then prepared summary national reports that are being reviewed by teams organized by ENSREG. US experts will consider information in the ENSREG reviews when they are available.

2.2.2.4 Summary

In summary, a range of post-Fukushima activities are underway within the US; but none duplicate the effort documented in this report. However, many international efforts have synergistic objectives to those being performed within the US DOE RST pathway. Clearly, it is important that the effort documented in this report benefit from and provide input to other on-going efforts. Future efforts within the DOE RST pathway will continue to be cognizant of and coordinate with other on-going efforts to avoid duplication and to maximize their contribution.

2.3 Decontamination & Decommissioning Activities

Examination efforts by TEPCO are primarily focused on obtaining data required to support D&D efforts, rather than providing data to the international community that could be used to enhance safety (e.g., data for validating severe accident models, source term models, etc.). Nonetheless, the government of Japan recognizes information collected from Daiichi is important to not only Japan for D&D efforts, but also to international organizations for reactor safety.[78] Furthermore, international participation may be beneficial to Japan because of expertise related to severe accident progression and, in the case of the US, because of expertise gained from prior TMI-2 D&D efforts. However, financial constraints and national needs dictate that TEPCO's primary responsibility is to obtain information required to support D&D activities at Daiichi.

Hence, it is important that the US Forensics Effort understand the organization and schedule for D&D activities within Japan. This section highlights key aspects of current D&D activities. The organizational structure for completing D&D is reviewed, and the strategy for prioritizing activities is described. Near-term activities are outlined and inputs for key D&D decisions to emphasize areas where the US Forensics Effort could use inspection information to benefit on-going D&D efforts in Japan and meet US objectives to enhance reactor safety.

2.3.1 Organization

In 2015, the government of Japan reorganized organizations involved in D&D efforts at Daiichi.[79,80, 81] Major organizations involved in this new structure are shown in Figure 3. The Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) has been established to strengthen decommissioning strategies, and work was initiated to revise the strategic plan and medium/long-term roadmap. In this new organizational structure, the NDF plays a major role as a coordinator of decommissioning strategy, and R&D. As depicted in Figure 3, D&D at Daiichi is accomplished as a coordinated effort between the NDF for making strategy- and technology-related decisions, TEPCO for on-site operational activities, the International Research Institute for Nuclear Decommissioning (IRID) for overseeing technology development for fuel debris retrieval, [81,82] and JAEA for overseeing required R&D to support decommissioning technologies.[83] In addition, the Nuclear Regulatory Authority (NRA) oversees D&D activities to ensure that necessary safety measures are taken and that the plant is maintained in a stable condition.

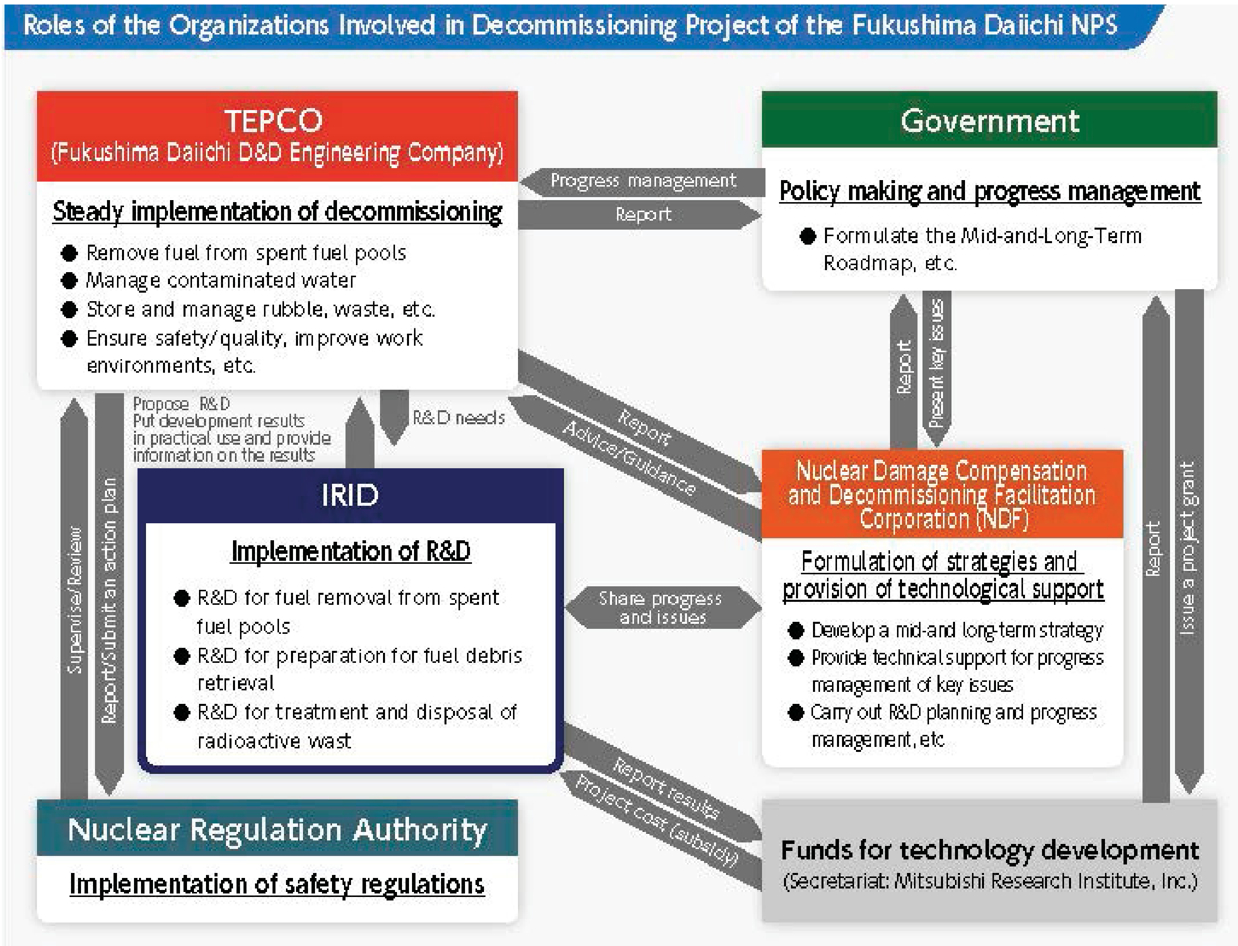

Figure 3. Organizations involved in decommissioning Daiichi. (Courtesy of IRID [81])

2.3.2 Strategic Plan

The NDF developed a strategic plan [79] to provide a strong technical basis for the government of Japan's medium/long-term roadmap. Within this strategic plan, NDF emphasizes the need for risk reduction by applying five guiding principles:

- Principle 1: Safe-reduction of risks posed by radioactive materials and work safety;
- Principle 2: Proven-highly reliable and flexible technologies;
- Principle 3: Efficient-effective utilization of resources (human, physical, financial, space, etc.);
- Principle 4: Timely-awareness of time axis;
- Principle 5: Field-oriented-thorough application of the "Three Actuals" (actual place, actual parts and actual situation.

The strategic plan emphasizes the prioritization of activities to reduce the risk from the disaster-affected area at Daiichi. As documented in the strategic plan, the risk profile of Daiichi is developed based on analytical results for the 'hazard potential' and the 'likelihood of loss of containment function' (see Figure 4). D&D activities are grouped and prioritized based on the need for risk reduction.

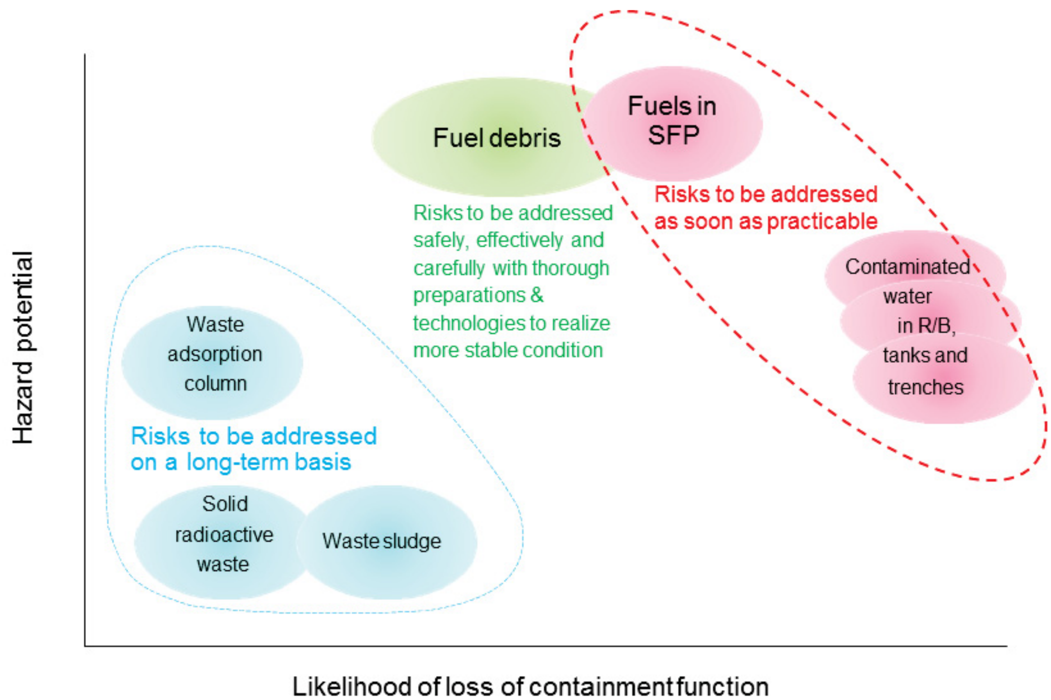

Figure 4. Process to evaluate the risk associated with various D&D hazards. (Courtesy of NDF [79])

As outlined in the strategic plan, D&D activities are grouped into three phases based on risk reduction:

- Phase I actions are taken against the risk sources with comparatively high level of risk as indicated in the upper right hand corner in Figure 4;
- Phase 2 actions are targeted to reduce risk associated with reactor fuel debris in 1F1, 1F2, and 1F3;
- Phase 3 actions are focused on reducing the risk for the stored waste and other waste generated from the actions taken in Phases 1 and 2.

While the primary objective is to complete the D&D efforts as early as possible, D&D efforts must not adversely impact the safety of the general public or plant workers. D&D activities must be monitored to alleviate concerns about recriticality, increasing radiation levels, radiation releases, increasing hydrogen concentrations, increasing temperatures, structural degradation, and non-nuclear industrial accidents. To accomplish all these objectives simultaneously, the risks of proposed D&D work processes have been, and will continue to be, evaluated by TEPCO. An updated strategic plan is expected to be issued in July 2016.[84]

2.3.3 Mid-and-Long-Term D&D Roadmap Activities and Schedule

Because there is uncertainty in many aspects of the plant conditions, especially with respect to the internal conditions of the primary containment vessel (PCV), various approaches are being considered for D&D activities. Current D&D plans are documented in a roadmap, which is updated periodically as new knowledge is gained from the affected reactors at Daiichi. The initial “Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4” (i.e., the Mid-and-Long-Term Roadmap) was finalized in December 2011 at the ‘Government and TEPCO's Mid-to-Long Term Countermeasure Meeting’ to indicate processes to recover from the accident at Daiichi. In June 2013 and June 2015, revised versions of the roadmap were issued. [78, 79] Further revisions will take place based on actual conditions. Periodic updates on D&D progress are provided in a format consistent with activities outlined in the Roadmap (e.g., see [85]).

The roadmap provides US experts general insights about the schedule and types of activities completed and underway by TEPCO. In addition, results from these activities are posted on TEPCO's website and discussed in periodic updates provided by TEPCO. As discussed in Section 2.3.2, the Mid-and-Long-Term Roadmap divides the time until completion of D&D into phases, identifies major tasks to be undertaken onsite, and the associated R&D schedule (see Figure 5).

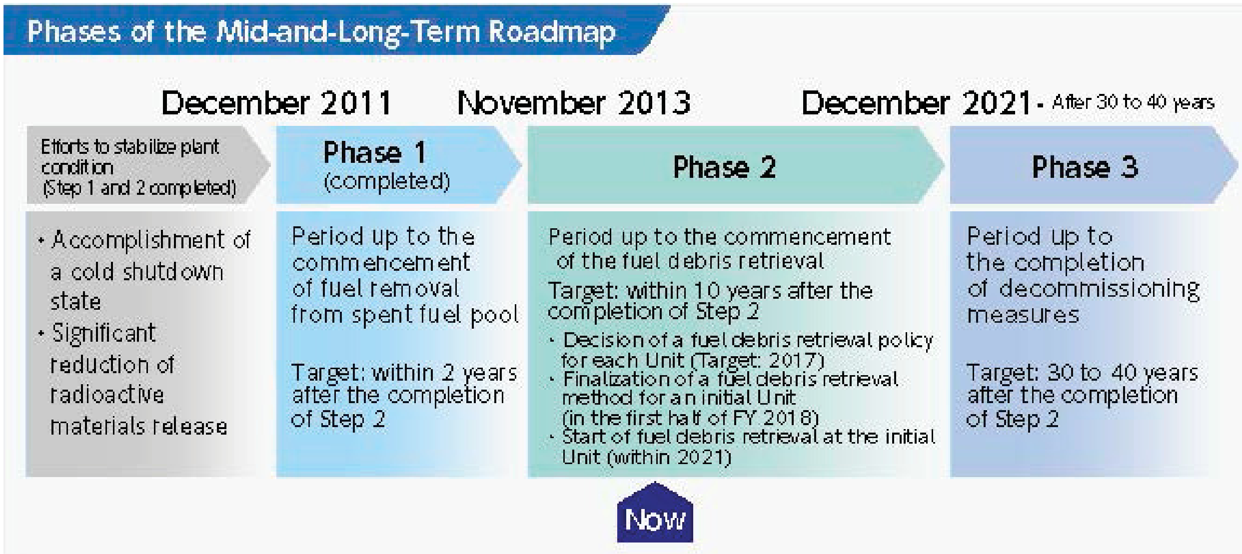

Figure 5. Summary definition of roadmap phases. (Courtesy of IRID [81])

Phase 1 represents the time period between plant stabilization (i.e., when radiation levels were low and releases were minimized) until the time when fuel removal from the spent fuel pool (SFP) begins. Phase 2 started in November 2013 with activities to remove the spent fuel from 1F4 and will continue until fuel is removed from the reactors. Phase 2 includes R&D for fuel removal and PCV repair operations. This includes R&D related to removing fuel from the spent fuel pools, preparing for removal of fuel from the RPV, and processing and disposal of solid radioactive waste. In addition, there is R&D related to alternative options for remote technologies that could reduce the challenges associated with D&D. Reference 78 provides additional details related to the scope and schedule of R&D activities. It is estimated that Phase 2 activities will require approximately 10 years to complete. Phase 3 spans from the completion of Phase 2 until the plant is decommissioned. It is currently estimated that Phase 3 activities will be completed within 30 years (resulting in up to 40 years for the complete D&D of the affected units). The schedule is based on current knowledge of the plants and analyses of differences in conditions of the units. For example, because 1F2 experienced less damage to the reactor building, several D&D activities within the building were completed earlier in this unit. Efforts were made to optimize opportunities to overlap required processes and operations between units. However, schedules may change as additional knowledge is gained.

Figure 6 provides additional details about the remaining tasks for completing Phase 2 and 3 activities (more detailed figures are provided in Appendix D). Major milestones are denoted by yellow triangles in this schedule. Because of the technical challenges associated with Phase 2 and 3 activities, some of these milestones are designated as "holding points" (HPs) or important junctures where decisions will be made regarding the transition to the next step. Such decisions include whether additional R&D is required or selecting one of multiple options for completing a task. As an example, HPs are defined in selecting an option for installing a cover on the reactor building in 1F1, 1F2, and 1F3. Figure 6 also shows 1F1, 1F2, and 1F3 HPs to determine which technology option will be pursued for removing the fuel debris in Phase 2. For example, as discussed in Section 2.3.4, one option under consideration is a 'submersion approach' in which fuel is removed under water to minimize worker exposure. However, the submersion approach requires that equipment be developed that can fit within the PCV and that water leakage from the PCV be stopped. Hence, alternate methods for debris removal are under consideration. Several organizations within Japan are performing activities that will provide input to this HP and other Phase II activities. As discussed in Section 2.3.4, there is the potential for the US Forensics Effort to provide input to these HP evaluations.

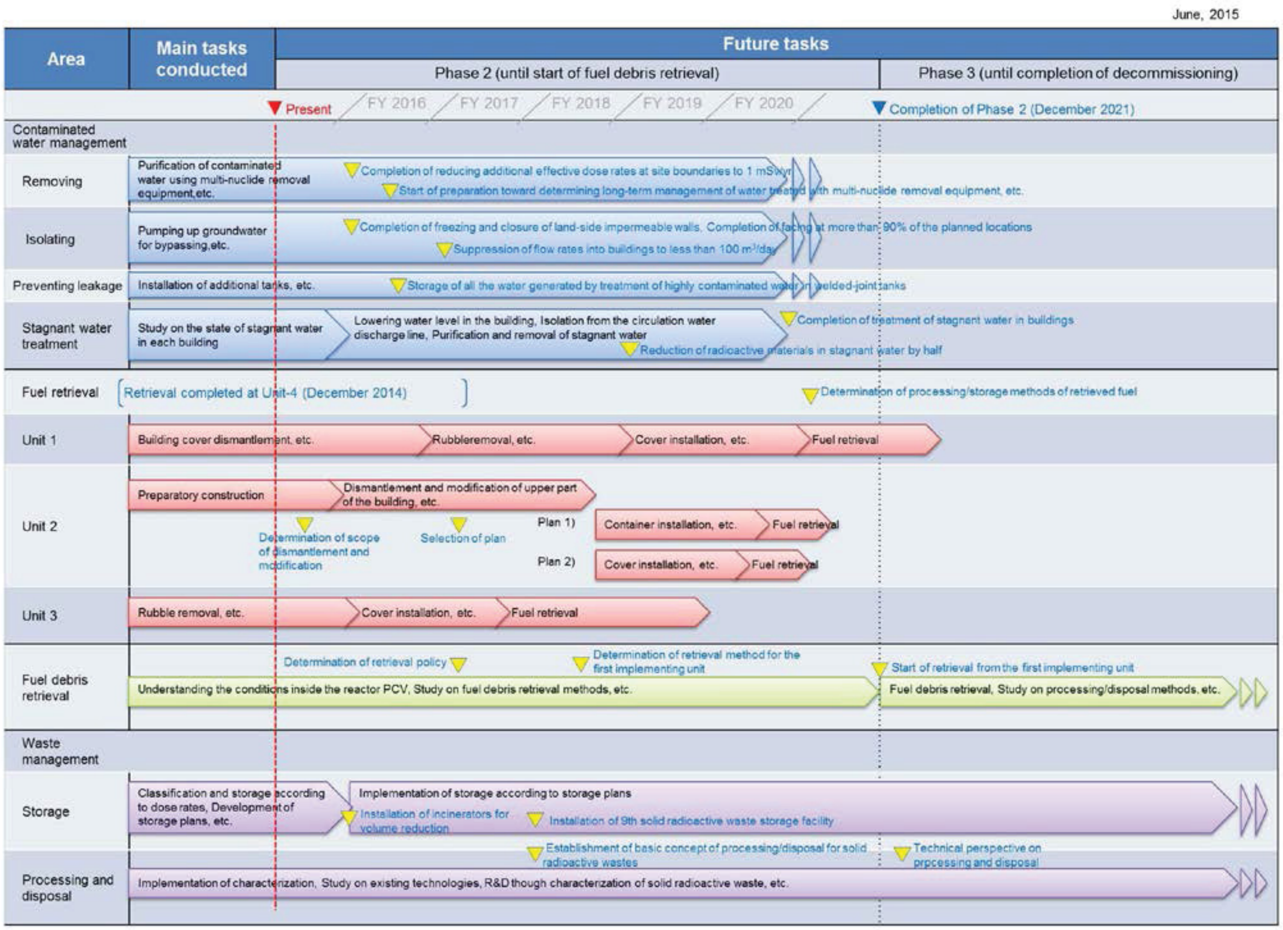

Figure 6. Summary schedule showing remaining roadmap tasks and milestones.[80]

2.3.4 Phase II D&D Activities

Near term D&D activities are associated with completing critical milestones for Phase II of the Roadmap (see Table 4). As discussed in the strategic plan, activities are underway to characterize potential hazards and the ability of tasks to be successfully completed using the five guiding principles outlined in Section 2.3.2. Inspection information and analyses using ‘state-of-the-art’ computational tools are required to complete these evaluations.

Table 4. Phase II critical milestones and timing (Based on information in [80])

Area	Description	Timing[f]
1. Contaminated Water Management		
Removing	Additional treatment using multi-nuclide removal equipment, and completion of reducing additional effective dose rates at the site boundary to 1 mSv/yr	FY 2015
	Start of preparation toward determining the long-term management of water treated with multi-nuclide removal equipment	First half of FY2016
Isolating	Suppression of inflow rates into buildings to less than 100 m^3/day	
Preventing Leakage	Storage of all the water generated by treatment of highly contaminated water in welded-joint tanks	Early FY2016
Completion of Stagnant Water Treatment	1) Separation of a turbine building from a circulation water discharge line.	FY 2015
	2) Reduction of radioactive materials in stagnant water in buildings by half.	FY 2018
	3) Completion of treatment of stagnant water in buildings	By the end of FY 2020
2. Fuel Retrieval from Spent Fuel Pools		
1) Start of fuel retrieval from 1F1		FY 2020
2) Start of fuel retrieval from 1F2		FY 2020
3) Start of fuel retrieval from 1F3		FY 2017
3. Fuel Debris Retrieval		
1) Determination of fuel debris retrieval policies for each unit		Around two years from now[g]
2) Determination of fuel debris retrieval methods for the first implementing unit		First half of FY 2018
3) Start of fuel debris retrieval at the first implementing unit		By the end of 2021
4. Waste Management		
Establishment of basic concept of processing/disposal for solid radioactive wastes		FY 2017

2.3.4.1 Debris Retrieval Characterizations – A Collaboration Opportunity

As an example of the potential benefits to Japan from the US forensics effort (and of the information obtained by Japan to the US), it is of interest to consider activities required to complete Milestone 3,

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[f] In Japan, the FY runs from April 1 through March 31.

[g] Reference [80] issued in 2015.

"Fuel Debris Retrieval". As described in Section 2.3.3, there is a hold point (HP) for selecting the method for retrieving fuel at each unit. As shown in Figure 7, a wide range of activities are underway to provide input to this HP.

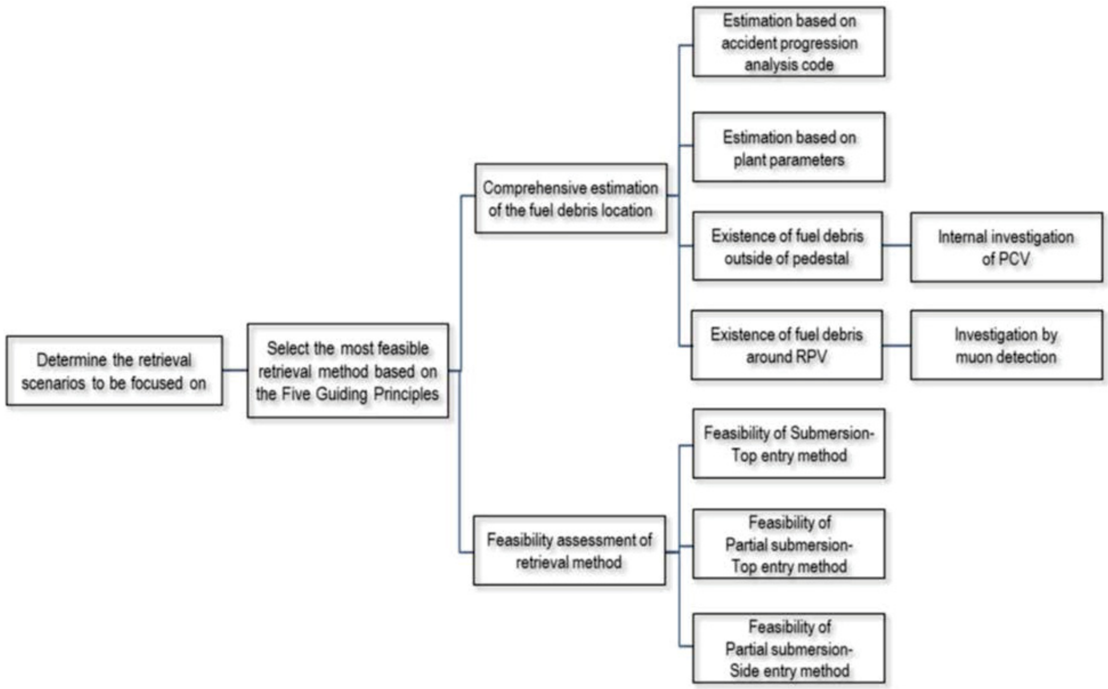

Figure 7. Summary of Phase II activities to select debris retrieval method. (Courtesy of NDF [79])

As indicated in Figure 7, several retrieval options are under consideration. Full and some partial submersion options utilizing top entry may require removal of the remnants of major structures, such as the steam dryer, the core plate, the steam dryer, the core shroud, etc. The integrity of these structures during the removal process must also be considered. A shielded storage area must be installed within the building to contain such structures, and highly contaminated structures will be disposed of with fuel debris. A full submersion option requires repairs to stop leakage from the PCV. In the case of partial submersion options, there is concern about increased radiation and decreased cooling when fuel and structures are lifted above the water. To mitigate such concerns, additional shielding is required (and the weight of such shielding must be considered in evaluating the structural integrity of building structures during D&D removal). An alternate partial submersion method under consideration involves a side-entry for extracting the debris. In any partial submersion method, evaluations must be completed to determine an appropriate water height. Feasibility studies are being performed for all of these options. All options require detailed knowledge of the debris location. However, at this time, debris end-state characterizations rely heavily on predictions by severe accident analysis codes and limited information, such as i) plant thermocouple data, ii) investigations using robots within the PCVs (photos, dose surveys, temperatures), and iii) muon tomography.

2.3.4.2 TEPCO Debris End-state Location and Radionuclide Characterizations

Using available data, examination information, and results from analyses performed using SAMPSON, TEPCO assessments related to debris end-state are summarized below in Table 5, and assessments related to cesium locations at the end of the accident (without considering migration into stagnant water and collection by water treatment) are shown in Figure 8.[79] TEPCO acknowledges that there is considerable uncertainty in such estimates, especially in light of uncertainties with respect to BWR accident progression and of the timing of certain actions and events that occurred in each unit. However, these analyses provide important input to Phase II debris retrieval decisions. TEPCO plans to update these analyses as additional information becomes available.

Table 5. Estimates to support debris end-state characterization (Based on information by NDF in [79])

Unit	Results from Plant Investigation	Estimated Debris Locations and Mass
1	• Water level inside D/W is approximately 3 m from the bottom. • S/C is almost filled with water. • Leakage from sand cushion drain pipe is confirmed. • Leakage from the expansion joint cover of vacuum break line in S/C is confirmed. • High radiation is detected in some areas in southeast on the 1st floor of the R/B (several Sv/h)	Location: Almost all fuel debris fell to the lower plenum, and very little fuel remaining in the core region. Most of fuel debris that fell to the lower plenum fell to the bottom of D/W. Fuel debris scattered outside the RPV pedestal (with possibility of shell attack). Mass: Loaded Uranium: 69 t Estimated Mass of Fuel Debris (including UO_2 and structural materials): 160-180 t
2	Water level inside D/W is approx. 30 cm from the bottom. S/C is about half filled and water level is almost the same as the torus room. No trace of leakage at upper part of torus room. The structure on the lower part of the RPV was confirmed by internal images taken from the opening at RPV pedestal. Damages to the RPV bottom may not be significant.	Location: A part of the fuel debris fell to the lower plenum, or to the bottom of D/W and some remaining in the core region (there may not be any outside the RPV pedestal). Mass: Loaded Uranium: 94 t Estimated Mass of Fuel Debris (including UO_2 and structural materials): 230-240 t
3	Water level inside D/W is approximately 6.5 m from the bottom (estimated from the pressure difference between D/W and S/C). S/C is almost completely filled with water. Leakage around expansion joint of main steam pipe D is confirmed.	Location: A part of the fuel debris fell to the lower plenum or to the bottom of D/W, and some remaining in the core region (there may not be any outside the RPV pedestal). Mass: Loaded Uranium: 94 t Estimated Mass of Fuel Debris (including UO_2 and structural materials): 220-230 t

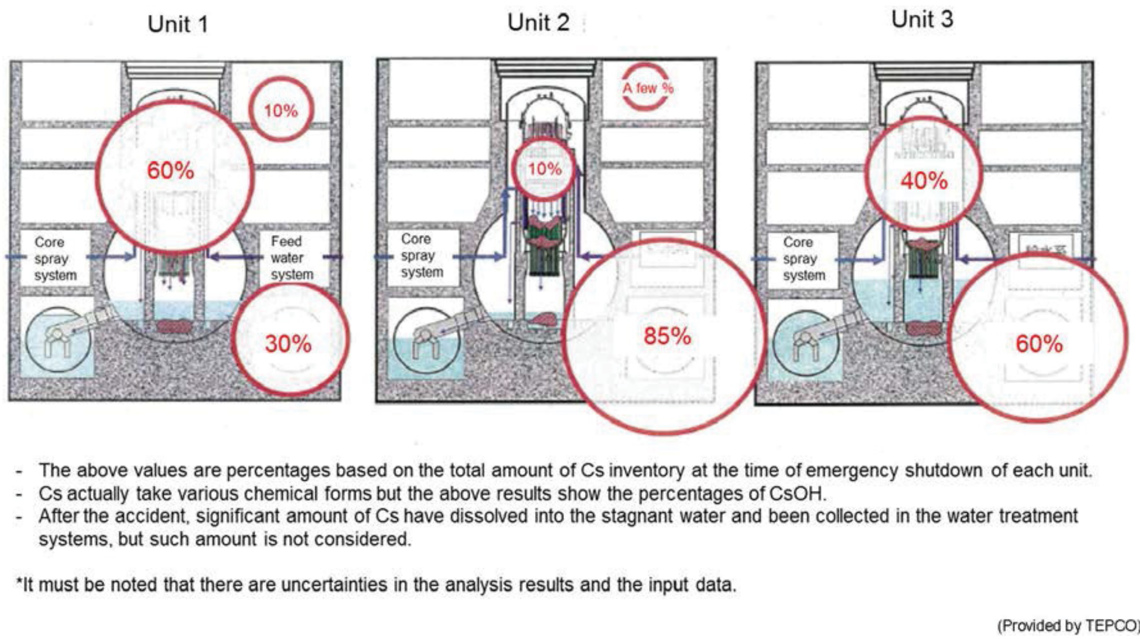

Figure 8. Estimated cesium location at the beginning of Phase II activities. (Courtesy of NDF [79])

Timely participation by US experts in evaluating inspection information obtained by TEPCO and in results from severe accident analyses could provide Japan an independent assessment for selecting retrieval options. US expert opinion may be of particular benefit because US researchers, who developed the models in severe accident analysis codes, are aware of model limitations and effects on subsequent source term assessments. Likewise, some of the participating US experts were involved in TMI-2 post accident evaluations and separate effects and integral tests related to severe accident phenomena of interest.

2.4 Summary

As part of their D&D activities, TEPCO has been and will continue obtaining information of interest to the international community. As noted in Reference 78, the government of Japan recognizes that information collected from these reactors is important to Japan and international organizations. However, financial constraints and national needs dictate that TEPCO efforts are primarily focused on obtaining data required to support D&D efforts, rather than providing data to the international community that could be used to enhance safety (e.g., data for validating severe accident models, source term models, etc.).

The DOE has established the US Forensics Effort to work with TEPCO to learn what information is being obtained and to communicate this information to cognizant US experts that could use this information to enhance safety of the commercial fleet. TEPCO has requested that the US document consensus information needs, along with suggested methods for obtaining the requested information and the intended use of that information. In particular, if there are situations where current D&D plans could preclude TEPCO's future ability to obtain desired information, such situations should be identified.

Activities completed within the US Forensics Effort are designed to benefit the US and Japan. As new inspection information is obtained, US experts can identify where current model predictions may need revision. Such revisions are of interest to Japan for near-term D&D decisions and of interest to the international community with respect to severe accident management and mitigation strategies. Information in Sections 3 through 6 of this report provide initial results from the US Forensics Effort.

3. AREA 1 - COMPONENT /SYSTEM PERFORMANCE

Examinations of components and systems within the RB, PCV, and RPV provide critical information related to their survivability, operability, and peak conditions (e.g., pressure and temperature) they experienced during the accident. Damage incurred from hydrogen explosions, radiation, and temperature can provide insights related to the accident progressions. As observed in Reference 4, component examinations in the TMI-2 containment provided critical evidence of peak temperatures and pressures when instrumentation data were inconsistent.

This section summarizes TEPCO Fukushima Daiichi D&D examination information that provides insights about component and system degradation and how this information can address uncertainties related to equipment performance and modeling uncertainty. To that end, we begin by identifying uses for this information (Section 3.1). Next, a summary of relevant information obtained to date is provided in Section 3.2 with emphasis placed on how these findings relate to reactor safety evaluations and future D&D activities by TEPCO. This is followed by a brief discussion of the limitations of the insights (Section 3.3). We then provide a few recommendations and observations for RST program activities as they relate to optimizing insights and information gained from these forensics studies, (Section 3.4). The section concludes with several questions and suggestions for additional information that would be beneficial regarding future assessments of equipment performance (Section 3.5).

3.1 Key Questions for Reactor Safety and D&D

Available information was evaluated by US experts to address the following questions:

- What visual damage has been observed in components and structures within the RB, PCV, and RPV?
- What plant instrumentation data are available to support component and structure damage assessments?
- What insights can be gained from observed damage with respect to: peak temperatures, peak pressures, radiation levels[h], effect of saltwater, combined effects (e.g., radiation enhanced temperature or mechanical damage, etc.), and multi-unit interactions?
- Should any components and structures be enhanced for reactor safety?
- Can information be used to confirm/improve severe accident guidance?
- Are analysis model improvements needed to predict observed damage?
  - Can information from one unit be used to confirm analysis assumptions, assess model adequacy, and predict conditions in another unit?
  - Can analyses with enhanced models be used to provide insights for future D&D activities (e.g., damaged/deformed structures may be more difficult to remove, etc.)?

Answers to these questions can have significant safety impact, and data from the three units at Daiichi offer the potential to reduce modeling uncertainties. Improvements in modeling capabilities can be used to confirm or enhance, if needed, specific components or systems and to improve accident management strategies with respect to containment venting, water addition, and combustible gas generation.

Answers to the above questions are also of interest to Japan with respect to Phase II D&D activities. Component degradation information provides insights related to decisions for debris retrieval method, development of fuel debris retrieval equipment, and implementation of fuel debris retrieval activities with

[h] Although radiation survey information is primarily discussed in Section 4, the data also provides insights related to component damage.

reduced risks from radioactive materials. In particular, improved models for predicting the timing and mode of vessel failure and the mass, composition, and heat content of material relocated to and from the lower head is of interest in making decisions related to the methods for debris removal and measures needed for worker protection from damaged structures and from radiation.

3.2 Summary of Information

TEPCO has performed a wide range of examinations at Daiichi to support their D&D activities. The outside and inside of the reactor buildings as well as inside the containments have been surveyed by personnel and/or robots. The examination data includes visual (i.e., pictures and videos) as well as limited sampling, dose rate, water level, and temperature information. TEPCO has published a large amount of data on its publicly accessible website.[8] In particular, TEPCO reports documenting unsolved and unresolved issues [21 through 24] have received special attention in the forensics effort to evaluate equipment performance.

As discussed in Section 1.3.1, the US Forensics Expert Panel identified information needs during FY2015 that could be obtained from these examinations. An updated list of information needs, based on FY2016 evaluations, is included as Appendix C of this report. The information needs address knowledge gaps in severe accident phenomena [33] and reduce uncertainties in equipment performance and modeling predictions. Some insights into component degradation and performance can already be ascertained based on observations from the examinations already performed by TEPCO. Tables 6 through 8 summarize the availability of information with respect to the component degradation information needs identified in Appendix C tables. Key aspects of this information are summarized in this section.

Table 6. Area 1 information needs from the reactor building.

Item	What/How Obtained	Use[i]	Data Available[j]
RB-1	Photos/videos of condition of RCIC valve and pump before drain down and after disassembly (1F2 and 1F3)	AE, AM	NA
RB-2	Photos/videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	AM	NA
RB-3a	Photos/videos of damaged walls and structures (1F1)	AE, AM, DD	A
RB-3b	Photos/videos of damaged walls and structures (1F3)	AE, AM, DD	A
RB-3c	Photos/videos of damaged walls and structures (1F4)	AE, AM, DD	A
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	AE, AM, DD	A
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	AE, AM, DD	A
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	AE, AM, DD	A
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	AE, AM, DD	A
RB-8	Photos/videos and inspection of seismic susceptible areas (e.g., bellows, penetrations, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	AE, AM, DD	A
RB-9	DW Concrete Shield Radionuclide surveys (1F1, 1F2, and 1F3 - before debris removed)	AE, AM, DD	A
	DW Concrete Shield Radionuclide surveys (1F1 - after debris removed)	AE, AM, DD	NA
	DW Concrete Shield Radionuclide surveys (1F3 - after debris removed)	AE, AM, DD	A
	Photos/videos around mechanical seals and hatches and electrical penetration seals (as a means to classify whether joints were in compression or tension)	AE, AM, DD	A
RB-10	Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations)	AE, AM, DD	A
RB-11	Photos/videos and available information on 1F1, 1F2, and 1F3 containment hardpipe venting pathway, standby gas treatment system and associated reactor building ventilation system	AE, AM, DD	A
RB-12	Photos/videos at appropriate locations near identified leakage points in 1F1, 1F2, and 1F3.	AM, DD	A
RB-13	Photos/videos of 1F1, 1F2, and 1F3 main steam lines at locations outside the PCV.	AM, DD	A
RB-14	Deposits or particles sampled inside reactor building (1F1, 1F2, 1F3); e.g., white deposits from HPCI room using Field Emission Scanning Electron Microscopy (FE-SEM), X-ray Diffraction (XRD), etc.	AE, AM, DD	NA

[i] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).

[j] Some information available [Green]; NA: no information available [Orange].

Table 7. Area 1 information needs from the PCV

Item	What/How Obtained	Use[k]	Data Available[l]
PC-1	Tension, Torque, and Bolt Length Records (prior and during removal); Photos/videos of head, head seals, and sealing surfaces (1F1, 1F2, and 1F3).[m]	AE, AM, DD	NA
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (1F1).	AE, AM, DD	NA
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3).[n]	AE, AM, DD	NA
	b) If vessel failed, 1F1, 1F2, and 1F3 PCV liner examinations (photos/videos and metallurgical exams).	AE, AM, DD	NA
	c) If vessel failed, photos/video, Radionuclide (RN) surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor.	AE, AM, DD	A
	d) If vessel failed, 1F2, and 1F3 concrete erosion profile; photos/videos and sample removal and examination	AE, AM, DD	NA
	e). If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage corium hang-up	AE, AM, DD	NA
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	AE, AM	NA
PC-5	Photos/videos of 1F1, 1F2, and 1F3 main steam lines and Automatic Depressurization System (ADS) lines to end of SRV tailpipes, including instrument lines	AE, AM	NA
PC-6	Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes (interior valve mechanisms)	AE, AM, DD	NA
PC-7	Ex-vessel inspections and operability assessments of 1F1, 1F2, and 1F3 in-vessel sensors and sensor support structures	AE, AM, DD	A
PC-8	Inspections and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures	AE, AM, DD	A
PC-9	Photos/videos of 1F1, 1F2, and 1F3 PC (SC and DW) coatings	PM	A
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	AE, AM, DD	A
PC-11	Photos/videos of 1F1, 1F2, and 1F3 primary system recirculation pump seal failure and its potential discharge to containment	AE, AM, DD	NA
PC-12	Photos/videos of 1F1, 1F2, and 1F3 Traveling In-Core Probe (TIP) tubes and SRV/Intermediate Range Monitor (IRM) tubes outside the RPV	AE, AM, DD, PM	A
PC-13	Photos/videos of 1F1, 1F2, and 1F3 insulation around piping and the RPV.	AM	NA
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM	NA
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM, DD	A
PC-16	Photos/videos of melted, galvanized, or oxidized 1F1, 1F2, and 1F3 structures.	AE, AM	A

[k] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[l] Some information available [Green]; NA: no information available [Orange].

[m]Available information is limited to the shield plug.

[n]Although some images have been obtained; images do not indicate if RPV failed or show any relocated core debris.

Table 8. Area 1 information needs from the RPV

Item	What/How Obtained	Use[o]	Data Available[p]
RPV-1	1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos with displacement measurements, sample removal and exams for fission product deposition, peak temperature evaluations)	AE, AM, DD	NA
	Photos/videos, probe inspections, and sample exams of 1F1, 1F2, and 1F3 Main Steam Lines (MSLs); Interior examinations of MSLs at external locations	AE, AM, DD	NA
	Photos/videos and metallurgical examinations of upper internals and upper channel guides	AE, AM, DD	NA
RPV-2	Photos/videos of 1F1, 1F2, and 1F3 core spray slip fit nozzle connection, sparger & nozzles	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 feedwater sparger nozzle and injection points	AE, AM, DD, PM	NA
RPV-3	1F1, 1F2, and 1F3 steam separators' integrity and location (photos/videos with displacement measurements, sample removal and exams for Fission Product (FP) deposition, peak temperature evaluations)	AE, AM, DD	NA
RPV-4	1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos and sample removal and oxidation testing.	AE, AM, DD	NA
	1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos, and metallurgical exams)	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region)	AE, AM, DD	NA
	Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures	AE, AM, DD	NA
RPV-5	Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)	AE, AM, DD	A
	Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.)	AE, AM, DD	NA

Table 9 summarizes a number of findings based on inspections performed by TEPCO. The table notes the observed status of various penetrations and equipment. In many instances, examination information has not yet been obtained for a particular unit's equipment. However, TEPCO has released a significant amount of information in the years since the accidents, some of which has not been translated into English. Although representatives from TEPCO participate in the US expert meetings and review draft versions of this report, there may be publicly available information that is not captured in this table.

[o] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[p] Some information available [Green]; NA: no information available [Orange].

Table 9. Results from component and system examinations.[q]

Area	1F1	1F2	1F3
X-100B PCV penetration	Possible melted shielding material [10] No damage observed on outside [86]	TBD	TBD
X-51 PCV penetration	TBD	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [87, 88]	TBD
X-53 and X-54 PCV penetration (HPCI pipe penetration)	Traces of flow and white sediment noted [15]	No damage observed [89]	No damage observed [90]
X-6 PCV penetration (CRD hatch)	TBD	Melted material [91, 92]	No observed damage from inside [93]
Equipment hatch	TBD	TBD	Water puddle [94, 95] unknown source
Personnel hatch and nearby penetrations	No major damage observed [96]	TBD	TBD
HPCI pipe penetration	No damage observed, but high dose rates measured [96,97]		
TIP Room	No leakage observed from PCV through TIP guide penetrations. Relatively high dose rates measured near other primary system instrumentation penetrations (X-31, X-32, X-33) [15,98]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [99]	
Wetwell (WW) Vacuum breaker line	Leakage on expansion joint of one line (X-5E) [100]	TBD	TBD
DW/WW vent bellows	Water leakage attributed to vacuum line above [100]	No leakage observed [101]	TBD
DW sand cushion drain pipe	Leakage [102]	No leakage observed [101]	TBD
SC water level	Almost full[103]	Consistent with torus room water level [103, 104]	Believed 'almost full' but not confirmed [103]
DW Water Level	~2.8 m[103]	~0.3 m[103]	~6.5 m[103]

[q] Nomenclature: [Clear]: TBD (To be determined); no information available; [Red]: available information indicates damage or leakage; [Orange]: available information suggests possible damage; [Green]: available information indicates no damage.

Area	1F1	1F2	1F3
Torus room	Partially flooded [105,106]	Partially flooded [107]	Partially flooded [107]
	Rusted handrails/equipment [10]	Non-rusted handrails/ equipment [10,108]	Non-rusted handrails/ equipment [10,109]
	TBD	Some room penetrations tested, no leakage observed [110]	TBD
MSIV room	Limited view obtained [15]	Water leakage cannot be observed [111]	Leakage in Line D near bellows [112]
DW Head	Reactor well shield plug displaced [113]	Possible leakage [114]	Possible Leakage [114]
RCIC or other low SC piping	TBD	Suspected leak location, not confirmed [10]	TBD

US experts reviewing available information observed notable differences in component degradation between 1F1, 1F2, and 1F3. Possible causes for these differences include unit design differences, the ability to inject water during the accidents, the ability to vent the primary system and containment during the accidents, and differences in the hydrogen explosions (or lack thereof) at each unit.

Sections 3.2.1 and 3.2.2 describe how selected examination information has confirmed revised actions proposed by industry related to water addition strategies to mitigate severe accidents and improve severe accident systems analysis codes.

3.2.1 Containment Examinations

PCV examinations are of interest to TEPCO with respect to D&D. In addition, this information is of interest to US experts with respect to validating revised severe accident management guidance and verifying the adequacy of code models.

3.2.1.1 Leakage Locations

Examinations have informed TEPCO's D&D planning by understanding the ability to floodup containment. Currently, the water level within the DW of each of the units differs, with 1F3 being filled the highest, followed by 1F1, and finally 1F2. This indicates differences in containment failure locations and/or areas. Damage or indication of leakage has been found in at least one location in the containment boundary in each unit (1F1: leakage on expansion joint of one DW-WW vacuum breaker line [100], DW sand cushion drain pipe leakage [102]; 1F2: melted material at X-6 PCV penetration [91, 92]; 1F3: MSL line D leakage near MSIV [112]; and all three units: possible DW head flange leakage). Although no damage has been detected for a number of other penetrations/lines, there are a number of penetrations and locations for which survey information is not available.

The information to date highlights diverse leakage point locations and the possibility for multiple leakage points. Identifying leakage locations, the timing of, and the conditions causing this leakage was of special interest to the expert panel because of industry efforts related to severe accident water addition (SAWA). The expert panel focused on available information that could provide insights related to peak temperatures and pressures within the PCV that would cause such leakage. Expert evaluations of examination information identified relevant, but different, information for each unit.

For 1F1, pressure data [8] indicate that peak PCV pressures were as high as 0.84 MPa/122 psia on March 12, 2011. Temperature data were not available until March 21, 2011. Calculated saturation temperatures for this measured peak pressure, assuming a pure steam environment and neglecting localized hot spots, indicate values as high as 172°C /342°F. However, as shown in Figure 9, examinations within 1F1 revealed that a lead shield plate was missing. It is currently unknown whether the plate relocated due to melting or creep. In order for this lead plate to have melted, gas temperatures inside the drywell exceeded 328 °C/ 622 °F, the melting point for lead.

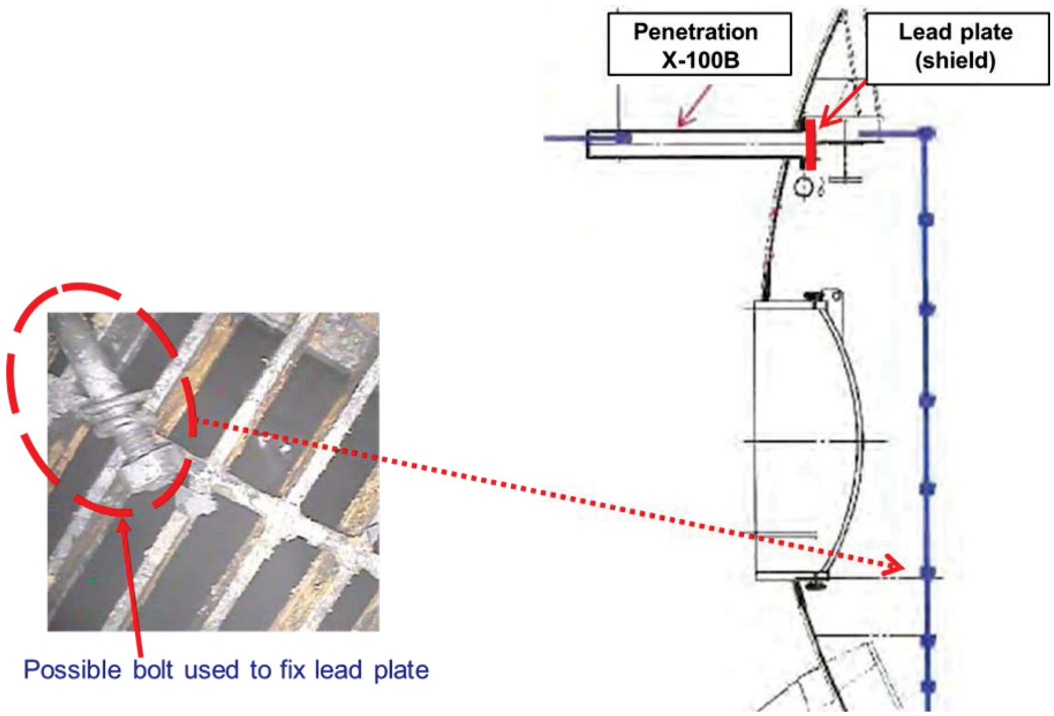

Figure 9. Visual examinations within X-100B penetrations in 1F1 PCV. (Courtesy of TEPCO [10])

For 1F2, insights related to peak temperatures within the PCV are available from visual examinations, radiation survey information, and temperature and pressure data. As shown in Figure 10, visual examinations of material from the X-6 penetration suggest that either the chloroprene cable cover or silicon flange seal material melted and dribbled out of this penetration. In their review, US experts concluded this evidence indicates peak temperatures at this location exceeded 300 °C/572°F and the dribbling pattern suggests that relocation occurred at low pressure (rather than a high pressure ejection of material). Plant data [8] indicate that 1F2 peak pressures were as high as 0.75 MPa/109 psia on March 15, 2011. Temperature data were not available until March 21, 2011. Calculated saturation temperatures for the measured peak pressure, assuming a pure steam environment and neglecting localized hot spots, indicate values as high as 168 °C/334 °F.

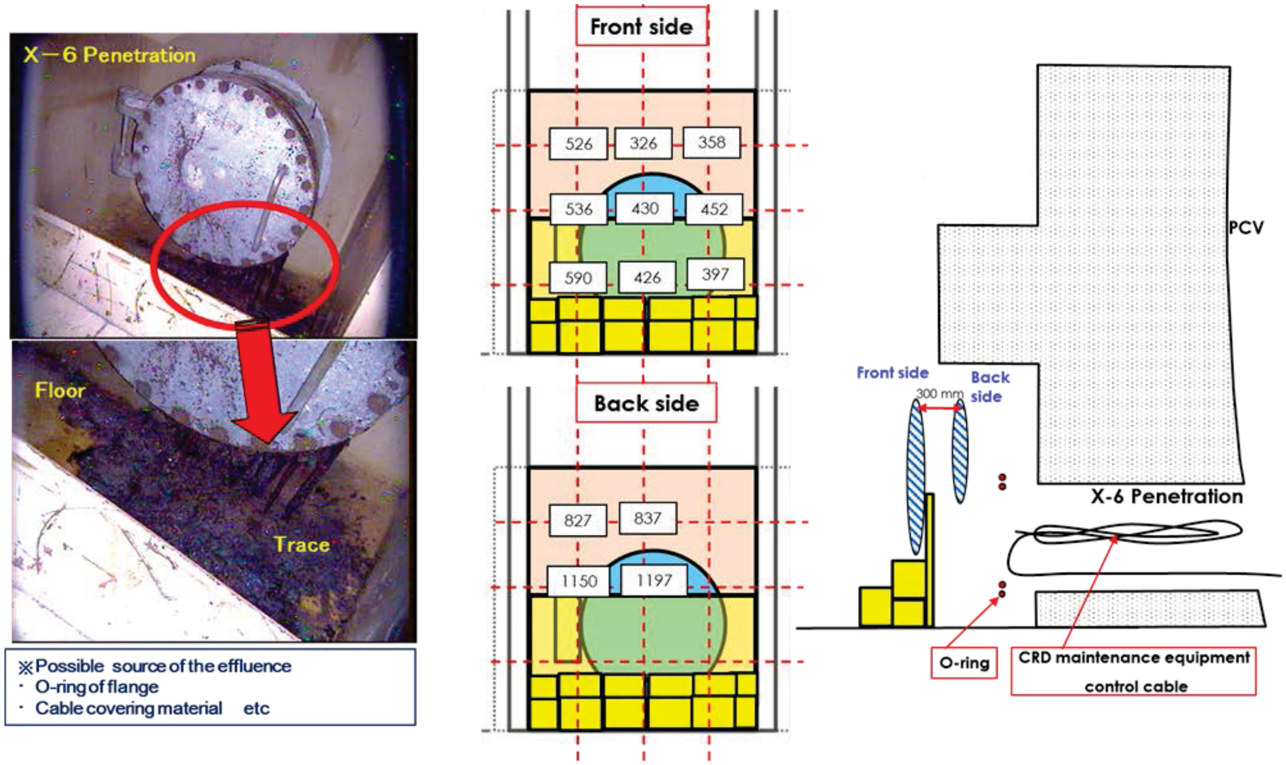

Figure 10. Photographs and radiation surveys (in mSv/hr) near 1F2 X-6 penetration (values measured in 13 locations). (Courtesy of TEPCO [91, 92])

For 1F3, insights about leakage come from photos and data obtained in March 2011 and dose rates obtained in November 2013. As shown in Figure 11, steam appears to be escaping at locations near the drywell head, and higher dose rates were measured near the drywell head. Both of these observations are consistent with a failure of the drywell head, perhaps due to drywell bolt expansion or strain or due to seal degradation from high temperatures and pressures within the PCV. Plant data [8] indicate that 1F3 pressures were as high as 0.75 MPa/109 psia on March 13, 2011. Temperature data were not available until March 20, 2011. Calculated saturation temperatures for the measured peak pressure, assuming a pure steam environment and neglecting localized hot spots, indicate values as high as 168 °C /334 °F. The combined pressure and temperature challenges are postulated to have stretched the drywell head bolts and allowed leakage through that pathway. However, the degree of damage to the head gasket is not known at this time. Photos showing leakage from MSIV expansion joints and radiological surveys from the equipment hatch penetration indicate that 1F3 experienced multiple leakage locations.

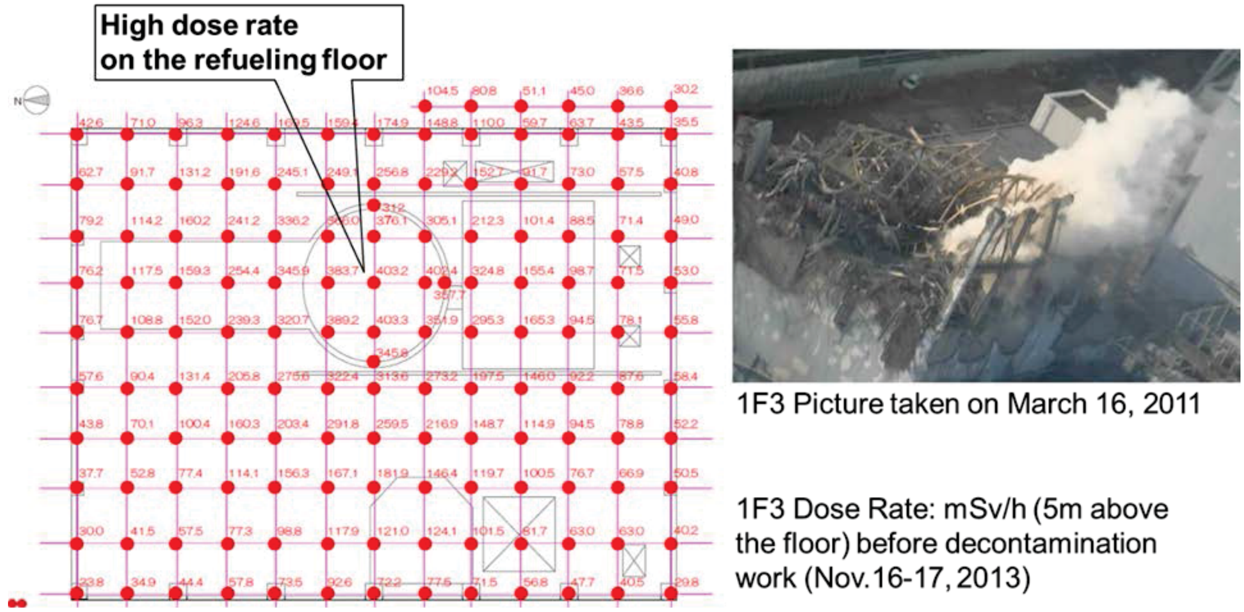

Figure 11. 1F3 radiation survey (value in mSv/hr measured on November 16-17, 2013 at 5 m above the floor at points shown on red grid) and photograph taken on March 16, 2011 (Courtesy of TEPCO [10,12])

Many of the leakage points identified for 1F1, 1F2, and 1F3 are not routinely modeled by systems level severe accident codes (e.g., MELCOR, MAAP, etc.). Both MAAP and MELCOR simulations predict DW head failure for the three units. It is evident that re-consideration of other penetrations/piping failures may be warranted for investigation in these systems analyses codes.

The potential for multiple penetrations to fail due to seal degradation is considered by industry in their proposed SAWA strategy. In the US, the new BWROG and PWROG severe accident management guidance places a high priority on venting the primary containment when the pressures and temperatures reach prescribed limits. For BWRs, these primary containment conditions can be very close to the primary containment design basis pressure and temperature, but guidance documented in NEI-13-02 [62] also considers water addition and water management strategies to enhance the effectiveness of fission product release mitigation during primary containment venting. Although there is variability in information from the units at Fukushima, the available information nonetheless confirms that maintaining containment conditions below the design basis, as well as reducing containment conditions, are appropriate strategies.

Figure 12 shows available peak temperature information on a figure developed from information in the NEI 13-02 industry guidance for venting. The DW vent is assumed to have a design temperature of 285°C/ 545 °F, and containment penetration degradation temperatures in the figure are based on engineering evaluations and testing information available in the literature. Black temperature lines in Figure 12 correspond to 1F1 and 1F2, and peak temperature information available from examinations at Daiichi. These values are consistent with the range of values assumed to cause degradation in NEI 13-02; thus, available information from Daiichi support NEI 13-02 guidance recommending that operators maintain containments at low pressure.

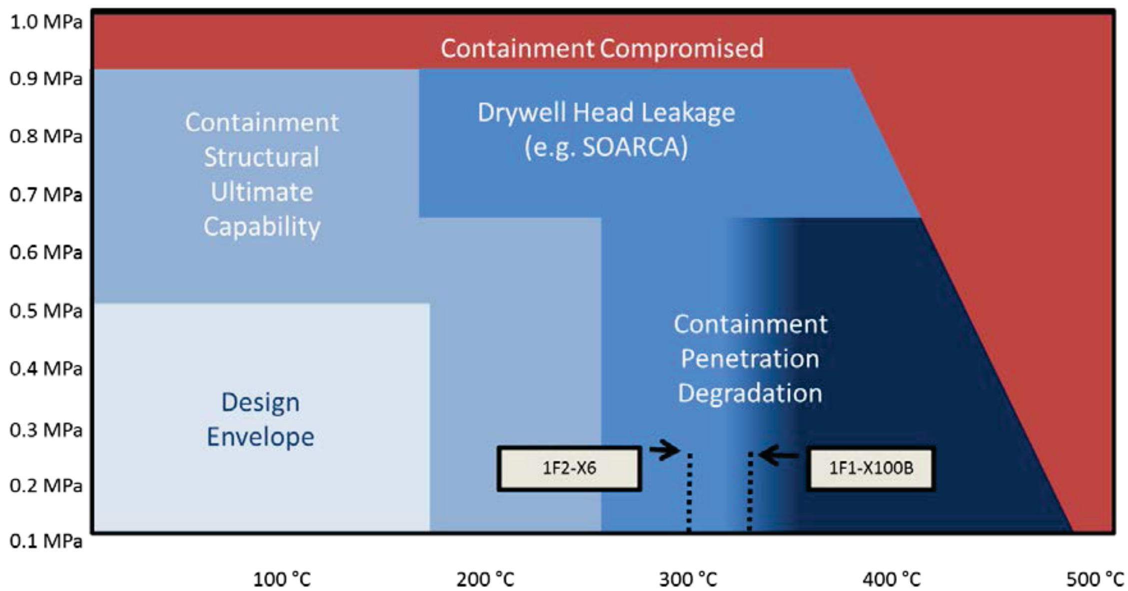

Figure 12. Containment pressure/temperature curve with available 1F1, 1F2, and 1F3 information. (Graphic courtesy of Nuclear Energy Institute [62] as modified by Jensen Hughes)

3.2.1.2 Code Modeling Enhancements

In developing severe accident guidance for water addition (and confirming that the integrity of the drywell head and seals was preserved), there was a desire to confirm the adequacy of the MAAP code to predict temperatures in the drywell. Comparisons between available Daiichi temperature information and analyses results have led to refinements in MAAP containment nodalization. Specifically, the MAAP code has been refined to include three containment volumes and a separate volume for the refueling cavity (see Figure 13). Comparisons of predictions from the MAAP code with available data confirm the adequacy of the revised model to predict the measured temperatures within the drywell. [18]

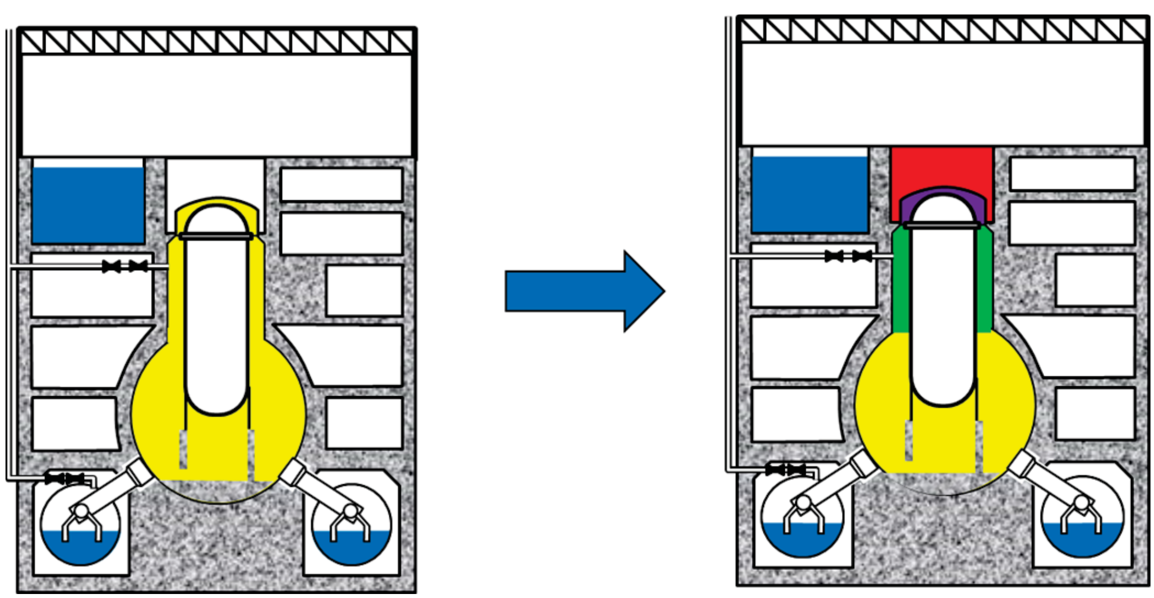

Figure 13. Improved MAAP nodalization. (Courtesy of Jensen Hughes, [18])

In 1F3, the PCV pressure increased more rapidly in the first 20 hours than systems level codes would generally predict [27,28]. Since 2012, there has been discussion and modeling of possible thermal stratification in the SC of 1F3 [16,24,115]. This has led to the development and adoption of refined modeling approaches and is an area of continued investigation. Note, this pressurization possibly caused the trip of the 1F3 RCIC, see Section 3.2.2.

Additional efforts are underway to assess the effects of SRV and RCIC operation on stratification within the containment. Evaluations with the enhanced MAAP models are being applied to predict instrumentation readings available to operators during severe accidents (and identify potential false instrumentation readings). Hence, reduced uncertainties in systems analysis code predictions provide additional confidence in severe accident management guidance in the US. Evaluations with these codes are also useful to Japan as input for D&D Phase II assessments.

3.2.2 Primary System and Water Injection

To date, there is very limited direct information related to the integrity of the primary system. Direct observation of the tailpipes for the SRV, RCIC, HPCI, or the SRVs, MSLs, recirculation piping and pumps, lower head penetrations, etc., have not been made.

Some photos and videos of structures below the bottom head of the 1F2 RPV have been obtained.[10] These images indicate some of the cabling may still be intact. However, the evidence is not conclusive.

A leak was observed in line D of the 1F3 MSL near the MSIV.[112] However, no leakage was observed in the MSIV room of 1F2. This motivates two open questions: What caused line D to leak but not lines A-C in Unit 3? Was there a difference in the accident progression between 1F2 and 1F3 that resulted in a leak in 1F3 and not 1F2? Subsequent correspondence with TEPCO [116] indicate that these differences were not attributed to differences in the SRV setpoints. Rather, available information suggests that differences may be due to differences in the accident progression and water levels within the PCV of these units (e.g., higher water levels prevent observations of leakage).

The cause for trip of the 1F3 RCIC system was reviewed by TEPCO in [24]. The most likely trip mode was identified as high turbine exhaust pressure and not overspeed. This is supported by the available SC pressure data.

During the accident, there were attempts to inject water via fire engines. During and after the accident, it was unclear how much of the injected water was successfully injected into the RPVs. TEPCO has reviewed the piping networks at the three units used to inject the water to identify possible bypass routes [24]. Ten bypass flow lines were identified for 1F1 and four lines were identified in both 1F2 and 1F3. This information has led to revised estimates of water injection into the RPV for 1F1 and could be used to revise estimates for 1F2 and 1F3. In addition, a similar review of bypass lines and check valve locations was performed for the Kashiwazaki-Kariwa Nuclear Power Plant and led to the installation of an addition motor operated valve.

3.3 Insight Summary and Limitations

A primary limitation associated with current insights is that much of the information is based on visual images (e.g., primarily photographs and videos). Distortions in the photographs may be caused by lighting, image resolution, and surface corrosion; such distortions may influence how experts interpret information in these visual images. The initial condition of equipment is also not known either because ‘before’ pictures are unavailable or have not been made available. Some of the observed leaks, peeling paint, and corrosion may not be attributed to accident.

Another limitation is that the timing of the observed damage (leakage, corrosion, etc.) with respect to the accident progression can be difficult to ascertain. The early failure of some components could have contributed to further damage of other components or prevented some components from failing. Also, the long term exposure to post-accident conditions (seawater, elevated temperature and radiation fields, etc.) can obfuscate interpretation of failure timing.

3.4 Recommendations

In reviewing available information for this area, the expert panel formulated several recommendations.

Area 1 Recommendation 1:

Sensitivity studies should be performed on containment failure location and size with respect to radiological releases (timing, amount) and impact on accident progression. These sensitivity studies should be done with both MAAP and MELCOR in order to cover a range of predicted containment and primary system conditions. Sensitivities for each unit would provide insight into which failure likely caused depressurization, the conditions under which such a failure occurred, and the effect of multiple failures. Some previous sensitivity analyses have been performed for failure of the primary system (SRV versus MSL, etc.) and the containment.

As discussed within this section, several containment penetrations and components are leaking in the three units. The failure of multiple containment penetrations, or even a specific penetration, identified in Table 9 is not predicted in best-estimate MAAP or MELCOR simulations of these accidents. Severe accident modeling, particularly as it pertains to probabilistic risk assessment, typically does not evaluate containment impairment in a mechanistic manner. In many models, containment impairments are assumed to develop using the following steps:

- Identify containment boundary locations that tend to exhibit a higher likelihood to become impaired in a severe accident, such as:
  - expansion of the structure at flanges or penetrations beyond the capacity for installed seals to prevent a leakage pathway from developing (e.g., lifting of the drywell head flange at appreciably elevated pressures);
  - development of localized high stresses as a result of elevated pressures, ultimately causing localized failure of the structure; and
  - weakening of containment boundary seals or structural elements as a result of combined mechanical, chemical and thermal loads.
- Define mechanical (pressure) and thermal loading criteria (atmospheric gas or structural temperatures) required to induce failure at the locations identified in the previous step.

As discussed in Section 4, reactor building radiological hotspots provide a means to assess inputs provided to severe accident computer codes, but do not typically facilitate assessment of the actual computer code models. There is, however, one important exception. Namely, mechanical and thermal challenges to the containment boundary predicted by code calculations can be compared with observed locations of impairments. In this regard, continued analytical effort would be of value as part of Fukushima Daiichi accident simulations to assess the potential for drywell head flange impairment due to high pressure and upper drywell temperatures. Photographs of the upper drywell structure could aid in identifying the potential for high upper drywell temperatures.

Area 1 Recommendation 2:

The expert panel should continue to review available information and update Table 9.

The expert panel concurred that information in Table 9 was useful for summarizing the status of various components and for comparing the status of the three units. The information in this table, coupled with code predictions, dose measurements, and available plant instrumentation information, can provide insights related to the timing of failure for various components. Determining whether failures occurred before or after vessel breach is important for predicting radionuclide transport during an accident and is useful for verifying information contained in revised industry guidance for severe accidents guidance.

> ***Area 1 Recommendation 3:***
>
> A concise comparison should be developed for the predicted conditions by both MAAP and MELCOR at the MSIV (temperature, pressure) for 1F2 and 1F3. The expert panel should continue to review any additional inspection information of the MSIV room or MSLs.

The leakage of 1F3 in the MSIV room is in contrast to the observation of no damage in the MSIV rooms for 1F2. As failure in this location bypasses the containment, it would be beneficial to understand why failure occurred in 1F3 but not in 1F2 and why leakage appeared to have occurred in line D and not lines A-C.

> ***Area 1 Recommendation 4:***
>
> The expert panel is interested in 'before' pictures for specific locations from TEPCO. As more information becomes available, the panel will identify specific places.

Many observations of component status are based on photographs or videos. There is a lack of 'before' pictures to compare against the pictures taken after the accident. A potential use is to help discern whether discolored markings on walls near penetrations are due to leakage before or after the accident.

3.5 Suggestions for Additional Information

Evaluations by the expert panel led to several suggestions for this area.

Area 1 Suggestion 1:
To facilitate updates to Table 9, the expert panel has requested that TEPCO continue to review information in this table. In addition, the expert panel will continue to review additional information, such as penetration, component, and system examination results, from TEPCO and update this table.

Area 1 Suggestion 2:
As discussed in Section 4, additional surveys in containment to understand the integrity of the RPV lower head, pedestal, and containment liner are of particular interest. These information needs are identified in Appendix C.

4. AREA 2- DOSE SURVEYS AND ISOTOPIC SURVEYS / SAMPLING

Dose surveys and radionuclide deposition samples collected within the RB, PCV, and SFP are another important data acquisition area to support D&D activities. Samples or swipes are of particular interest because they can provide evidence of fission product release fractions and possibly of fission product speciation.

This section summarizes Fukushima Daiichi D&D dose survey and isotopic survey and sampling information obtained by TEPCO. As discussed within Sections 3, 5, and 6, survey and sampling information provides insights about component and system degradation, debris end-state location, and combustible gas effects. The section concludes with several questions and suggestions for additional information that would be beneficial regarding future assessments of equipment performance.

4.1 Key Questions for Reactor Safety and D&D

Available information was evaluated by US experts to address the following questions which are of international interest for reactor safety and to Japan for completing feasibility studies to support D&D activities:

- How were fission products transported through various structures?
- What compounds were formed?
- Was deposition and transport affected by hydrogen combustion?
- Are there any observed effects from saltwater addition?
- Can 'mass balances' be obtained for the fuel?
- Can released isotopic species be used to estimate the unit from which the release came and peak core temperatures experienced by the unit?
- Can radiation surveys, combined with analysis results, be used to infer a failed component?
- Can analysis provide insights related to worker dose minimization?

Answers to these questions can have an important safety impact. By obtaining prototypic data from each of the units at Daiichi, there is the potential to reduce modeling uncertainties. Improvements in our modeling capabilities can be used to confirm or enhance, if needed, accident management strategies with respect to containment venting, water addition, and combustible gas generation. This information and associated analyses with improved severe accident codes offer the potential for insights that may be beneficial to Japan in their D&D activities. In particular, improved models for predicting the events at Daiichi may provide important insights related to radionuclide transport and deposition, which is important in characterizing worker dose during D&D activities.

4.2 Information Summary

As discussed in Section 1.3.1, US experts identified information needs that could be addressed through examinations at Fukushima Daiichi. Requested information needs from the reactor building and PCV that relate to Area 2 are summarized in Tables 10 through 12. These tables also note if any information is available to address these information needs (see Appendix C).

Table 10. Area 2 information needs from the reactor building

Item	What/How Obtained	Use[r]	Data Available[s]
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	AE, AM, DD	A
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	AE, AM, DD	A
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	AE, AM, DD	A
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	AE, AM, DD	A
RB-9	DW Concrete Shield Radionuclide surveys (1F1, 1F2, and 1F3 - before debris removed)	AE, AM, DD	A
	DW Concrete Shield Radionuclide surveys (1F1 - after debris removed)	AE, AM, DD	NA
	DW Concrete Shield Radionuclide surveys (1F3 - after debris removed)	AE, AM, DD	A
	Photos/videos around mechanical seals and hatches and electrical penetration seals (as a means to classify whether joints were in compression or tension)	AE, AM, DD	A

Table 11. Area 2 information needs from the PCV

Item	What/How Obtained	Use[t]	Data Available[u]
PC-2	Photos/videos and radionuclide surveys/ sampling of Isolation Condenser (IC) (1F1).	AE, AM, DD	NA
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3).[v]	AE, AM, DD	NA
	b) If vessel failed, 1F1, 1F2, and 1F3 PCV liner examinations (photos/videos and metallurgical exams).	AE, AM, DD	NA
	c) If vessel failed, photos/video, RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor.	AE, AM, DD	A
	d) If vessel failed, 1F2, and 1F3 concrete erosion profile; photos/videos and sample removal and examination	AE, AM, DD	NA
	e). If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage corium hang-up	AE, AM, DD	NA
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	AE, AM, DD	A
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM	NA
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys.	AE, AM, DD	A

[r] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).

[s] Some information available [Green]; NA: no information available [Orange].

[t] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[u] Some information available [Green]; NA: no information available [Orange].

[v] Although some images have been obtained; images do not indicate if RPV failed or show any relocated core debris.

Table 12. Area 2 information needs from the RPV

Item	What/How Obtained	Use[w]	Data Available[x]
RPV-1	1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos with displacement measurements, sample removal and exams for fission product deposition, peak temperature evaluations)	AE, AM, DD	NA
	Photos/videos, probe inspections, and sample exams of 1F1, 1F2, and 1F3 MSLs; Interior examinations of MSLs at external locations	AE, AM, DD	NA
	Photos/videos and metallurgical examinations of upper internals and upper channel guides	AE, AM, DD	NA
RPV-3	1F1, 1F2, and 1F3 steam separators' integrity and location (photos/videos with displacement measurements, sample removal and exams for FP deposition, peak temperature evaluations)	AE, AM, DD	NA

Information related to radionuclide release and transport has been acquired from a number of sources during and following the three core melt events at Daiichi. During the accident, sources include:

- Radiation doses encountered by plant personnel entering the reactor buildings;
- Elevated radiation doses that developed in control rooms for the affected units;
- Radiation doses on the plant site due to:
    - Passage of airborne plumes, either forming from containment venting operations or accidental release after the 1F1, 1F2, and 1F3 containments became impaired,
    - Deposition of fission products from these releases onto the site,
    - Dispersal of contaminated structural material over the site due to reactor building explosions when flammable gases combusted inside the 1F1, 1F2, and 1F4 reactor buildings;
- Drywell and wetwell radiation readings from affected units, acquired when operators re-powered containment air monitors (CAMs).

Following the accident, contaminated water in the various reactor buildings provide additional indications of low-elevation leakage from the damaged units. Specific examples include:

- 1F1: Contaminated water leakage was detected in the reactor building basement, and it was speculated that this leakage arose because of damage to the drywell liner by interaction with ex-vessel core debris[y];
- 1F2: Very soon after the event, relatively high levels of radiological contamination were measured in water that accumulated in the reactor building basement;
- 1F3: Contaminated water leakage was detected in the reactor building on the first floor in the vicinity of the MSIV.

Available reactor building and offsite radiological contamination information provide important insights that can be used to:

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[w] Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[x] Some information available [Green]; NA: no information available [Orange].

[y] The nature of this failure is poorly understood at present. No conclusions can be directly drawn related to the long-standing issue of melt-liner attack in a BWR Mark I reactor design.

- Refine understanding of core damage progression and its impact on potential off-site consequences,
- Identify locations at which the containments became impaired to develop insights relevant to enhancing containment protection,
- Understand the isotopic composition of fission product releases for the purpose of enhancing detailed understanding of fission product transport and potential off-site consequences.

4.2.1 Post-Accident Evaluations of Reactor Building Contamination

Post-accident examinations of the reactor buildings provide important information related to likely points of containment impairment. Key results from TEPCO post-accident reactor building inspections are summarized in this section.

4.2.1.1 1F1 Reactor Building Contamination

Access to the 1F1 reactor building is challenging because of damage to the upper floors that occurred as a result of flammable gas combustion at 24.8 hours after the earthquake. The following areas in the 1F1 reactor building have been identified with elevated radiation dose rates:

- First floor area around the penetration between the basement and the first floor providing passage for the wetwell vent line. This has been linked to impairment of the expansion joint on the wetwell vacuum breaker line, as shown in Figure 14.
- Raw Cooling Water (RCW) heat exchangers (~1 Sv/h) and associated piping found in the contaminated waste treatment areas (see, for example, Figure 15). RCW equipment provides an important signature of the possible extent of core damage because ex-vessel core debris could potentially attack the RCW piping present in the drywell sumps.
- Contaminated water running into the 1F1 torus room from the drywell sand pit (suggesting the 1F1 drywell liner is impaired). An obvious explanation for impairment of the drywell liner at the elevation of the 1F1 drywell floor is attack by high temperature material relocating from the core. If such an attack arose early in the event, shortly after RPV lower head breach, it does not appear to have had a governing influence on gas-phase leakage into the 1F1 reactor building. Such leakage would likely have caused a greater build-up of flammable gases at lower elevations in the 1F1 building, promoting combustion at and damage to lower elevations of the 1F1 building. However, as discussed in Section 6, little damage was observed on lower 1F1 building elevations.

No elevated doses inside the 1F1 TIP room have been measured. Thus, failures of in-core instrument tubes during core damage progression did not impair the containment.

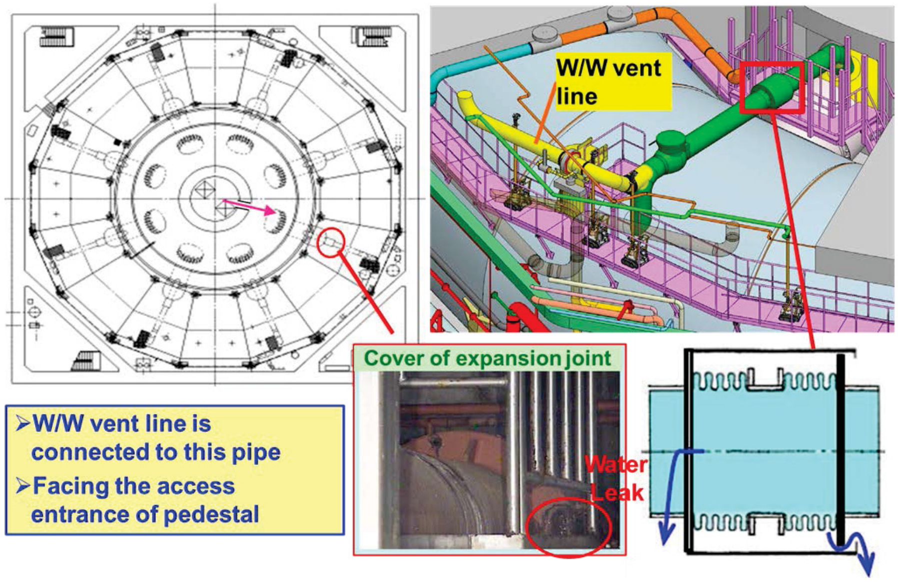


Figure 14. Impairment of 1F1 vacuum breaker line expansion joint. (Courtesy of TEPCO [117])

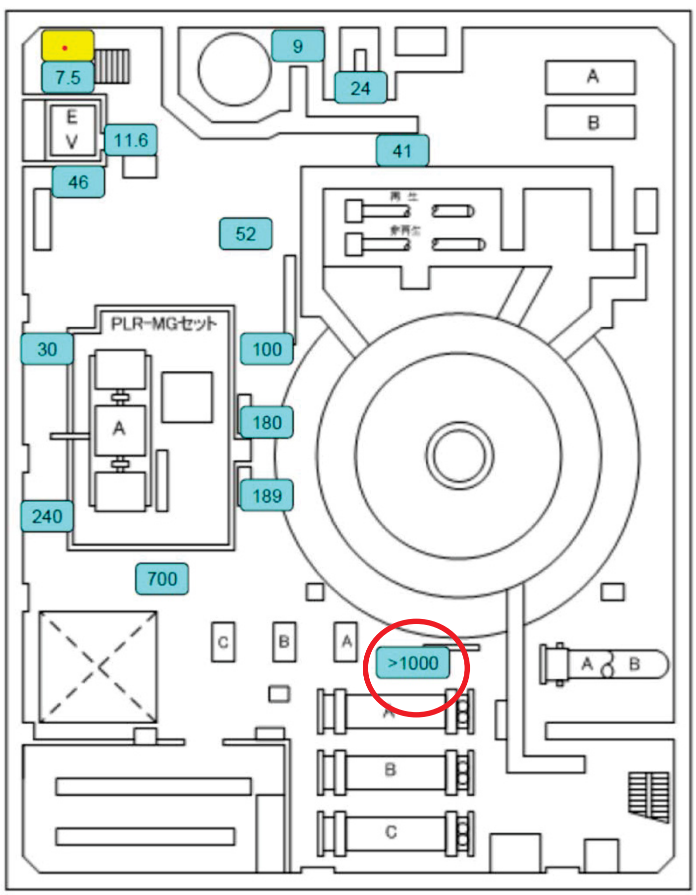


Figure 15. 1F1 reactor building first floor dose rate measurements (in mSv/hr; taken above floor elevation) illustrating elevated radiological contamination of RCW system (circled area). (Courtesy of TEPCO [118])

4.2.1.2 1F2 Reactor Building Contamination

Currently, personnel access to the 1F2 reactor building is less restricted than at 1F1; combustion of any flammable gas leakage from containment impairments did not occur at 1F2. This is likely due to the opening of the reactor building blowout panel on the refueling floor, which occurred due to gas rarefaction following the pressure waves propagating away from the 1F1 reactor building flammable gas explosion. Figure 16 shows the open blowout panel in the 1F2 reactor building.

Figure 16. 1F2 reactor building with open blowout panel. (Courtesy of TEPCO [119])

Inspections after the accident have identified the following areas within the 1F2 reactor building with notably elevated radiation doses:

- In front of the X-6 penetration pipe flange (Figure 10). As discussed in Section 3.2.1, rubber material (likely the chloroprene rubber cable sheath material stored inside the penetration for use with the CRD replacement machine) has apparently melted and led to the formation of organic debris outside the penetration. The presence of this material suggests that high temperature conditions likely occurred inside the penetration leading to ultimate impairment of the silicone rubber O-ring seal and melting of chloroprene rubber cable sheath.

Shield plugs above the drywell head, covering the refueling cavity, have measured dose rates of ~800 mSv/h (Figure 17).

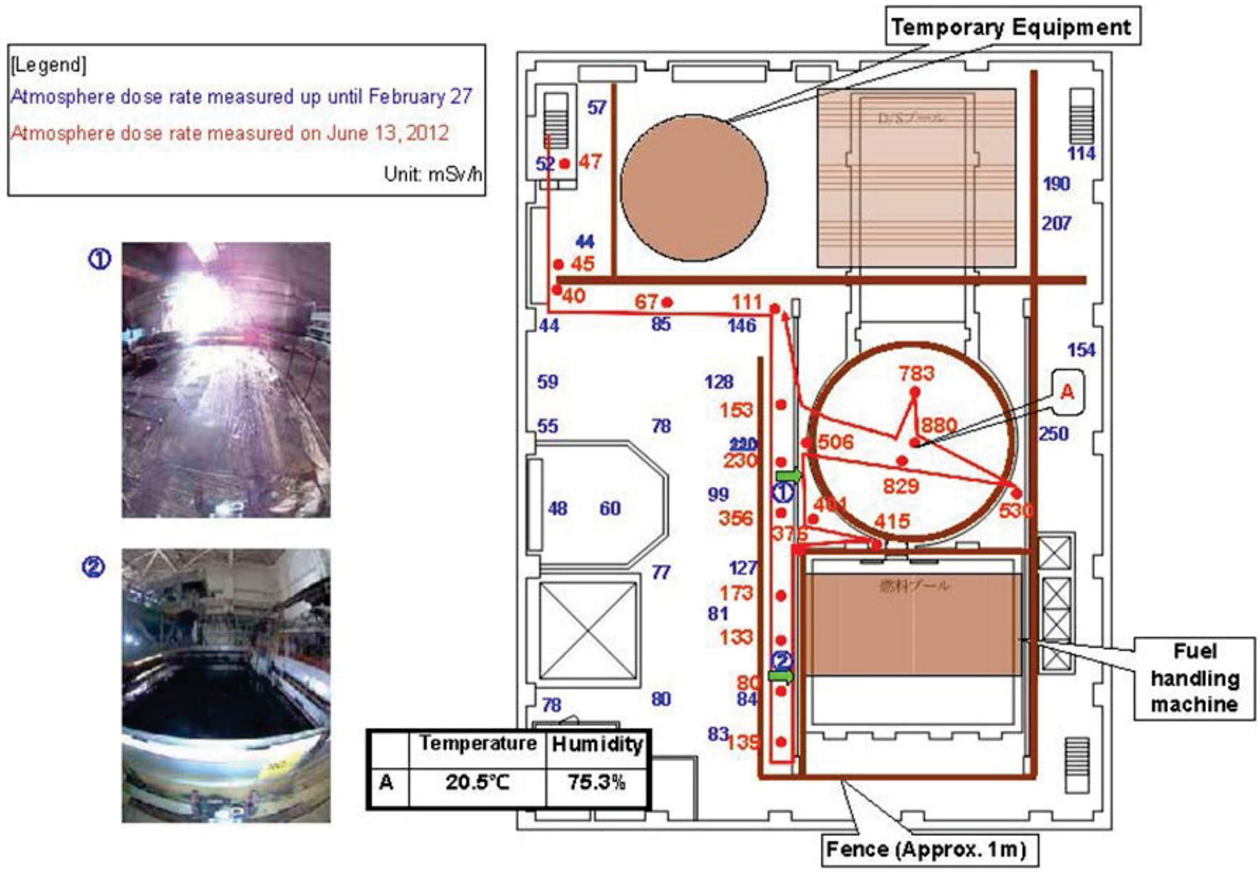


Figure 17. 1F2 reactor building refueling floor visual and dose rate information. (Courtesy of TEPCO [120])

Unlike 1F1, the RCW equipment is not contaminated. The cause for this difference is still unknown.

Similar to 1F1, elevated doses inside the 1F2 TIP room have not been identified. Any failures of in-core instrument tubes during core damage progression did not impair the containment.

4.2.1.3 1F3 Reactor Building Contamination

As with the 1F1 reactor building, access to the 1F3 reactor building is difficult because of damage that occurred when flammable gases combusted at 68.7 hours after the earthquake. Unlike the 1F1 reactor building, more extensive damage occurred to lower elevations of the 1F3 reactor building.

Elevated radiation dose rates have been observed in the 1F3 reactor building at the following areas:

- Equipment hatch on the first floor of the reactor building (Figure 11):
  - High dose rates are restricted to water pools that formed on the floor immediately outside of this hatch;
  - There does not appear to be sufficient evidence to suggest that gas-phase leakage occurred from this location.

- Elevated dose rates inside the MSIV room:
  - Contamination in this region has developed due to leakage of water from containment through an impairment of this penetration;
  - The 1F3 drywell water level does not exceed the elevation of the MSIV penetrations.
- Elevated dose rates above the drywell head have been confirmed at 1F3 (Figure 11).

Unlike 1F1, the RCW piping is not contaminated. The cause for this difference is still unknown.

As with 1F1 and 1F2, elevated doses inside the 1F3 TIP room have not been identified. Any failures of in-core instrument tubes during core damage progression did not result in containment impairment.

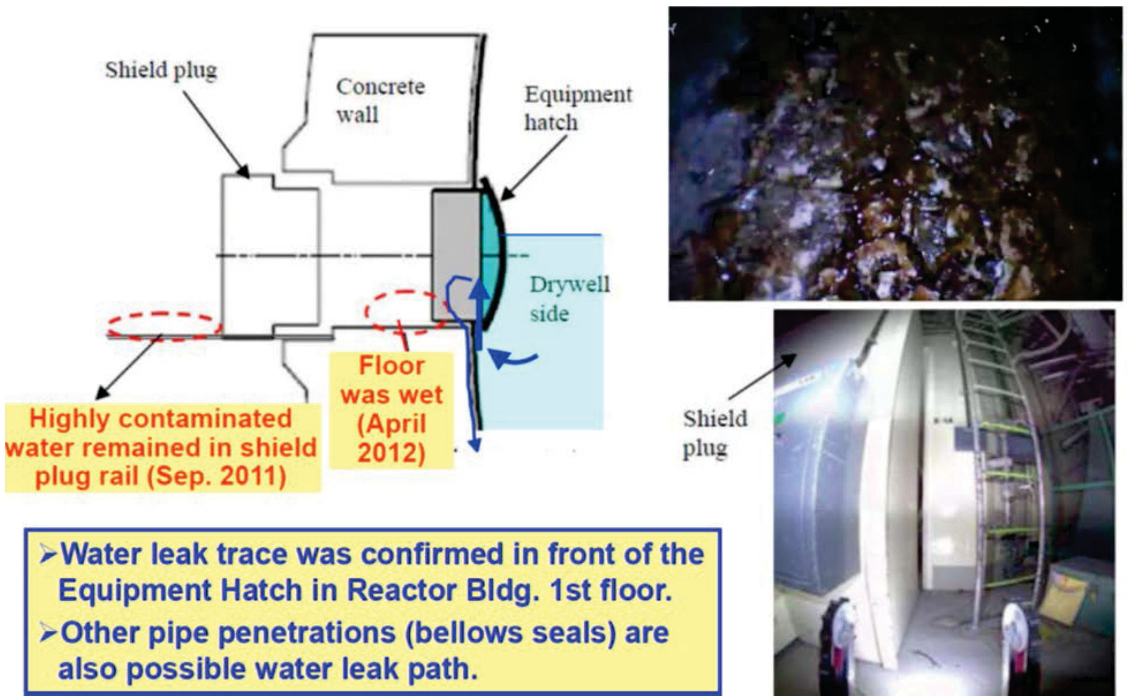


Figure 18. Liquid-phase leakage from 1F3 equipment hatch. (Courtesy of TEPCO [117])

4.2.1.4 Insights and Limitations

A summary of notable locations of elevated dose are provided in Table 13. These measurements were reviewed and categorized for the primary purpose of gaining insights into potential locations where containment integrity may have been lost (or impaired). Of specific interest is identifying containment boundary locations that are more likely to become impaired during an accident. This characterization is also relevant for assessing existing assumptions about BWR Mark I containment vulnerabilities applied in existing safety assessments and Probabilistic Risk Assessments (PRAs).

Table 13. Locations of elevated dose rate inside reactor buildings[z]

Unit	Floor	Note
1F1	1st	Penetration between the basement and the first floor providing passage for the wetwell vent line due to failure of wetwell vent line bellows
	2nd	Very high dose rates around RCW heat exchangers (see [24] for a detailed discussion of the fission product flow path out of containment into the RCW system piping)
	3rd	Elevated dose rates observed on east side of the reactor building underneath a stairwell and associated with a puddle of water
	Refueling Floor	Investigations have not had same access to the refueling floor as 1F2 and 1F3
1F2	Torus	Notable contamination of water in torus room (leakage from damage to RCIC suction piping suspected [117])
	1st	Elevated dose rates around X-34 penetration
	2nd	Elevated dose rates around X-29B/C penetration
	Refueling Floor	Very high dose rates found at the shield plug, above the drywell head
1F3	1st	Elevated dose rates around water pools accumulating outside the equipment hatch. Elevated dose rates in the MSIV room
	Refueling Floor	Very high dose rates found at the shield plug, above the drywell head

Reactor building dose rate measurements were acquired by TEPCO in 2012. The data acquisition was performed with an above floor gamma camera device. While these radiation reading are obviously subject to alteration with time due to radioactive decay and instrumentation uncertainty, the available measurements were observed at similar times using a similar methodology. Given that the primary source of radiation at this point is from long-lived fission products such as Cs-137, the reporting of raw dose rates is reasonable given the qualitative insights which they are supporting.

The focus of the summary in Table 13 is to identify areas of the reactor building where high air dose rates were measured. One exception is noted, however, in reporting of relatively high dose rates in the water of the 1F2 torus room. Unlike 1F1 and 1F3, it has been noted since March of 2011 that water borne radiological release was elevated at 1F2, highlighting a leakage location in the torus or connected piping. The summarized locations do not include detailed discussion of air dose rate readings acquired from within the torus rooms of the different units; these measurements tend to be influenced by shine from inside the gas space of the torus and do not provide an indication of containment integrity.

The following insights can be derived from these dose rates measurements:

- As discussed in Section 3.2.1.1, the drywell head flange at all three damaged units appears to be the primary point at which containment leakage may have first occurred. Radiation surveys from 1F2 and 1F3 indicate the drywell head flange as a point of leakage, and very high dose rates were measured in this region for both units. Radiation survey information from 1F1 is presently not available; however, leakage from the drywell head flange is also suspected at this unit because of temperature information

[z] Nomenclature: [Clear] - No information; [Yellow] - Notable water contamination (< 100mSv/h; [Orange] - Elevated (100 mSv/h to 500 mSv/h));] [Red] - High (> 500 mSv/h).

(see Section 3.2.1.1) and the localized flammable gas combustion damage observed on the 1F1 refueling floor (see Section 6).

- As discussed in Section 3.2.1.1, degradation of the containment boundary appears to have occurred at all three units through a number of additional locations, with each unit having a different set of localized containment impairments. The range of containment impairments observed so far likely reflects features of the accident that are unique to each unit. In several cases, the additional points of containment impairment are localized around a containment penetration, introducing a point at which liquid leakage has occurred. This presents a challenge with respect to decommissioning activities, since it is possible that penetrations higher in the drywell may also be susceptible to liquid leakage that would initiate during any attempt at drywell reflooding.
- In most of these cases, an important limitation is that it is difficult to identify when impairment at these additional locations could have occurred. Late-phase degradation of containment due to persistence of elevated pressure and atmospheric temperature appears to be a likely outcome given the range of observed impairments. While such containment impairments may not be directly relevant to off-site consequences, they have a significant impact on the ability of personnel to access the plant. These longer term containment impairment modes have had a significant impact during the event remediation and cleanup phases, notably through the prolonged contamination of ground water.

4.2.2 Containment Radiation Data Obtained during Event Progression

Radiological data acquired during the event also provide some insight into core damage progression and fission product release to the environment.

4.2.2.1 Overview of Available Radiation Measurements

The periods of most significant core degradation have not been fully captured by the 1F1 and 1F3 drywell and wetwell radiation measurements. Power was not available to the CAM systems at these units to support gathering of this information during the periods when most active core degradation occurred at 1F1 and 1F3.

For 1F1, drywell and wetwell CAM system measurements were not available prior to March 14, 2011; much of the significant core damage progression, including RPV lower head breach, likely occurred prior to March 13, 2011. As a result, distinct signatures showing a change in conditions (i.e., a notable increase in radiation readings) are generally not discernible from the available CAM system data for 1F1. The one exception is that drywell radiation readings appear to increase from about 10 Sv/h to about 90 Sv/h near the end of the day on March 14, 2011. This elevated radiation level in the 1F1 drywell persists for about one day. This coincides with a restoration of water injection to the unit. It also supports the potential for fission product release from 1F1 early on March 15, 2011, coincident with a shift of winds to the southwest of the Daiichi site. Elevated radiation levels to the southwest of the site, at locations like Oono, were identified over this period.

Unlike 1F1, however, there are sparser radiation measurements available from 1F3 during notable periods of core damage progression from March 13, 2011 to March 16, 2011. There are some radiation readings available from the drywell CAMS on March 14, 2011; however, these readings exhibit a relatively constant radiation level. Thus, there is no clear signature suggesting when core damage progression events led to an increase of fission product release to containment.

In contrast, drywell and wetwell radiation readings at 1F2 were restored during a time of active core damage progression (i.e., from about March 15, 2011 to March 16, 2011). The measurements of drywell and wetwell radiation levels obtained during active event progression at 1F2 provide insights into evolving core damage and potential failure of the reactor pressure boundary. Figure 19 shows radiation measurements obtained from the drywell and wetwell during a period of significant core damage progression at 1F2.

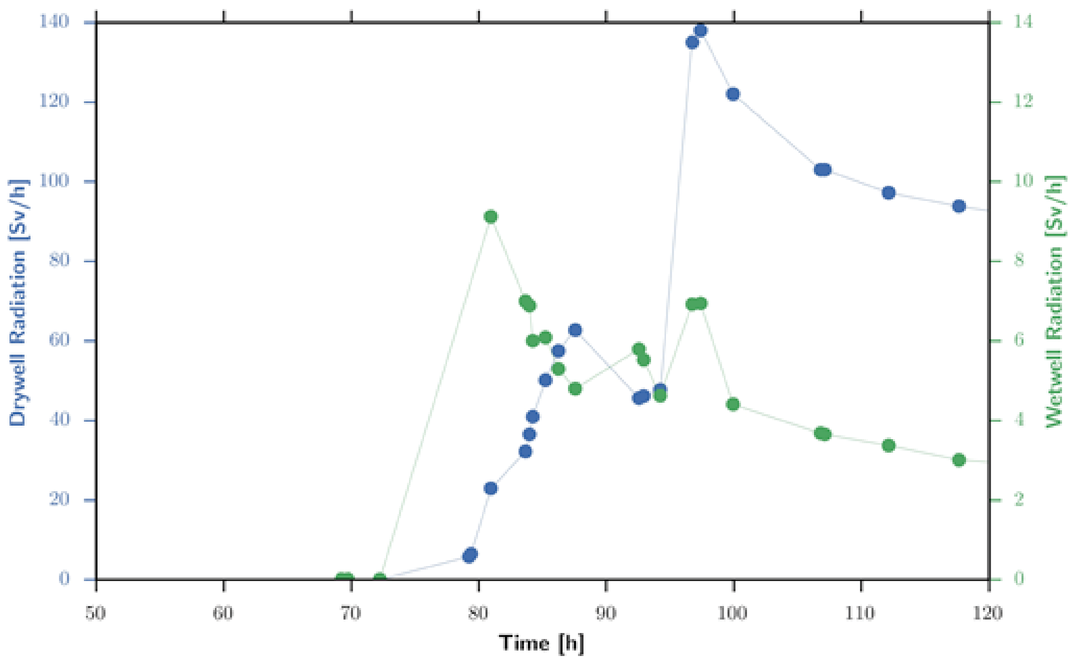


Figure 19. 1F2 drywell and wetwell CAMS readings. (Courtesy of Jensen Hughes based on information in [72])

As shown in Figure 19, drywell radiation readings increased just prior to 80 hours into the event (e.g., drywell radiation measurements increased significantly over a few hours around this time). This increase in drywell radiation readings also corresponds to the measured rapid increase in drywell pressure at around 80 hours into the event.

Beyond 80 hours, wetwell radiation readings began to decrease. Wetwell radiation readings initially provided the leading indicator for release of fission products from damaged fuel around the time of core damage onset (~75 hours). After about 80 hours, drywell radiation readings provide the leading indicator of enhanced fission product release from damaged fuel. Definitive evidence is currently not present, but the shift to the drywell radiation readings as the leading indicator of core damage progression tends to indicate a failure of the RPV pressure boundary directly into the drywell. Note that this does not mean lower head breach; RPV depressurization could be due, for example, to failure of either:

- In-core instrument tubes due to core degradation and relocation, or
- Steam line/tail pipe assembly (including the SRV gasket) impairment due to occurrence of very high temperatures following the onset of core damage.

Between approximately 87 hours and 92 hours, 1F2 containment depressurization occurred, stabilizing drywell containment radiation levels. As discussed above, this is postulated to be due to impairment of the drywell head flange. Beyond 94 hours into the event, the 1F2 containment radiation readings provide an indication of renewed core damage progression. At ~94 hours into the event, a rapid increase in the drywell radiation level occurred with similarly rapid increases in containment and RPV pressures. The surge in containment pressure after 94 hours abated. One possible explanation for the containment radiation readings and pressure measurements around this time is an interaction between molten core debris and water. Such an event could be due to slumping of molten core debris into lower plenum water or relocation of core debris out of a breached lower head. Around this time, there was also a shift in winds to the northwest of the plant coincident with precipitation (i.e., rain and snow). Any releases which occurred from 1F2 during this period would have been transported over a region of Japan that has exhibited the most significant land contamination (as indicated by the long-lived radionuclide, ^{137}Cs).

The implicit assumption that 1F2 was the only contributor to this measured land contamination merits further study; the 1F1 and 1F3 containments were already impaired by this time. However, international code comparison efforts (see Section 2.2.2.2) indicate that some 1F1 and 1F3 accident progression results do not predict significant radionuclide release to the respective containments and the environment at this time [72].

4.2.2.2 Insights and Limitations

The impairment of the RPV pressure boundary prior to breach of the RPV lower head (early RPV pressure boundary impairment) is an important insight captured in the containment radiation measurements. Available data do not provide sufficient means to discriminate between alternative scenarios for the source of the RPV pressure boundary impairment. The specific source of impairment is not ultimately germane to reactor safety considerations—direct discharge from the RPV into the drywell promotes a rapid evolution of gases that present an overpressure challenge to containment integrity.

That said, the rate associated with an overpressure challenge may be slower in the case of in-core instrument tube failures because there is a smaller area available for discharge from the RPV into the drywell. Gross failure of piping in the steam line and tail pipe assembly could result in a more rapid release of energy directly into the drywell and thus a more sudden escalation of containment pressure toward or beyond its design value.

Resolving the likelihood of different early RPV pressure boundary impairments (i.e., prior to RPV lower head breach) is an important limitation associated with available information from the affected units at Daiichi. There is much uncertainty in predicting RPV impairment. Much of this uncertainty stems from the lack of information to uniquely extrapolate smaller scale experiments to reactor scale. As a result, computer codes can exhibit significantly different predictions for the tendency of high temperature conditions to develop above the core following the onset of a severe accident. Previous studies have noted that severe accident models exhibit significantly different predictions of the gas temperature above the core once core damage commences (see, for example, the MAAP-MELCOR crosswalk [29]).

Thus, the containment radiation readings at 1F2 provide an indication that further inspections of its RPV pressure boundary impairment locations will provide information of relevance to assessing fundamental differences in core damage progression models. Furthermore, the drywell pressure measurements obtained from 1F1, though sparse during the first 10 hours of the event, provide additional indication of rapid pressurization of the drywell due to a possible early impairment of the RPV pressure boundary. MELCOR simulations tend to highlight creep failure of a main steam line as the source of early RPV pressure boundary impairment at 1F1. MELCOR modeling of the 1F3 event scenario highlights the potential for conditions to have developed in the 1F3 RPV that would have challenged the integrity of the main steam lines.

Given the strong indications of early RPV pressure boundary impairment, visual data from each of the damaged units relevant to RPV pressure boundary integrity would be of significant value for enhancing the severe accident knowledge base at reactor scale. RPV upper internals and steam line/tail pipe assembly visual data would also be of considerable value.

4.2.3 Other Radiological Measurements

Evaluations of radiological samples from outside the containment buildings provide insights on two important questions:

- Did late-phase fission product releases (i.e., around the 94-hour mark in the event) originate primarily from 1F2, which was still undergoing active core damage progression?
- Did the notable land contamination to the northwest of the Fukushima Daiichi plant primarily arise due to a coincidence of rapid core degradation, impairment of containment, and meteorological conditions?

Both questions pertain to increased understanding that can be gained from these reactor-scale events, whether or not protection of containment during the most active periods of core damage progression can significantly ameliorate the potential for notable off-site consequences. In particular, does a core degradation event ultimately progress to a point where the geometry of the degraded core does not have sufficient surface area to support strong fission product release to the containment and ultimately the environment? Evidence to this effect tends to remove from consideration late-phase impairment of containment due to harsh environments (e.g., temperature and radiation fields) as a meaningful contributor to off-site risk. The issues of late-phase containment impairment would thus be more relevant from the perspective of accident remediation.

4.2.3.1 Other Available Information and Insights

Other radiological information from Fukushima Daiichi primarily consists of concrete samples taken from the reactor buildings of the affected units and evaluations of contaminated soil samples from outside the affected units.

To date, the information available from examinations of concrete samples is insufficient to support a broad-spectrum evaluation of fission product chemistry. However, some trends in the available data are worth noting. [121]

- The overall volatile fission product releases appear to be consistent across all three units. The ^{137}Cs concentration (in units of Bq/g) is high in the concrete samples obtained from all three units.
- The release of fission products having lower volatility appears to be relatively higher at 1F2 compared with 1F1 and 1F3. The concentrations of, for example, Eu, Tc and Sr are generally higher in the concrete samples from 1F2.

Despite these trends, the different locations from which concrete samples were taken prevent any accident progression insights to be developed at this time. The 1F2 information was acquired from a highly contaminated region, the floor concrete in the shield plug area (i.e., above the drywell head).

Examination of the ratio of ^{134}Cs to ^{137}Cs in the off-site contaminated soil samples indicates that there may be a statistically relevant contribution from 1F3 [122]. This evidence should provide a cautionary warning to avoid excluding 1F1 and 1F3 core damage progression scenarios exhibiting enhanced fission product release to containment beyond the 90-hour mark. As discussed above, both the 1F1 and 1F3 containments were impaired well before this time. Additional information on this topic may be obtained by isotopic evaluations of concrete samples taken from the 1F1 and 1F3 refueling floor shield plugs. In particular, evaluation results could be used to assess the off-site ^{134}Cs to ^{137}Cs isotopic ratios.

4.2.3.2 Limitations

As discussed above, evaluations of sources from outside the containment provide a very gross assessment of fission product transport from the degraded fuel and ultimately through an impaired containment to the environment. It is of limited utility in identifying the release and transport of the range of radionuclide species expected to evolve during a severe accident.

The different locations from which concrete samples were taken prevent any accident progression insights to be developed at this time. The 1F2 information was acquired from a highly contaminated region, and the concrete shield plugs were obtained from above the drywell head. Any further information that could be acquired regarding isotopic composition from the shield plugs above the drywell heads at 1F1 and 1F3 would be useful for evaluating fission product release and transport. In particular, these results highlight the insights that could be gained with respect to the release and transport of lower volatility fission products.

In particular, a significant amount of interest has developed in the European severe accident community related to the transport of Ru during a severe accident. It would be especially helpful if it is possible to use insights from the reactor scale events at Fukushima Daiichi to resolve fission product transport issues derived from smaller scale integral tests.

4.3 Recommendations

In reviewing available information for this area, the expert panel formulated several recommendations for future sensitivity studies and evaluations.

Area 2 Recommendation 1:

Similar to Area 1 Recommendation 1, experts agreed that information on this topic suggests that sensitivity studies should be performed on containment failure location and size with respect to radiological releases (timing, amount) and impact on accident progression. These sensitivity studies should be done with both MAAP and MELCOR in order to cover a range of predicted containment and primary system conditions. To compare results from simulations of core damage progression and radiological release to the environment, additional analyses with an environmental radiological transport code, such as MACCS, would be useful. Sensitivities for each unit would provide insight into which failure likely caused depressurization, the conditions under which such a failure occurred, and the effect of multiple failures. Some previous sensitivity analyses have been performed for failure of the primary system (SRV versus MSL, etc.) and the containment. As discussed within this section, reactor building radiological hotspots provide a means to assess inputs provided to severe accident computer codes, but do not typically facilitate assessment of the actual computer code models.

Area 2 Recommendation 2:

Similar to Area 1 Recommendation 3, concisely compare the predicted conditions by both MAAP and MELCOR at the MSIV (temperature, pressure) for 1F2 and 1F3.

Area 2 Recommendation 3:

Similar to Area 1 Recommendation 4, the expert panel continues to be interested in examination information of MSIV room components.

The leakage of 1F3 in the MSIV room is in contrast to the observation of no damage in the MSIV rooms for 1F2. As failure in this location bypasses the containment, it would be beneficial to understand why failure occurred in 1F3 but not in 1F2. An important component of such an evaluation is determining both the potential point in the accident when failure occurred and also the relevance of this failure to gaseous releases from the impairment. As noted in this report, a number of different locations of containment impairment have been identified. It is currently suspected that many of these impairments were of primary relevance either after the containment depressurized or as a location through which aqueous leakage occurred. While such impairments are critically relevant to on-site personnel performing accident management and remediation activities, their influence on off-site radiological contamination is far less significant. In this regard, the impairment to the drywell head flange is believed to be the primary source through which the most significant radiological release occurred with respect to off-site contamination.

Area 2 Recommendation 4:

The expert panel recommends that the US Forensics Effort continue to evaluate information obtained from examinations of RPVs within each unit impairment location. In particular, addition visual information would be useful in the Area 2 Recommendation 1 sensitivity studies.

Given the strong indications of early RPV pressure boundary impairment, visual data from each of the damaged units relevant to RPV pressure boundary integrity would be of significant value for enhancing the severe accident knowledge base at reactor scale. RPV upper internals and steam line/tail pipe assembly visual data would be of considerable value. The occurrence of an early impairment in the RPV (prior to lower head breach) is an important aspect in evaluating CAM response.

4.4 Suggestions for Additional Information

As illustrated within this section, dose survey and isotopic survey and sampling information provides insights about component and system degradation, debris end-state location, and combustible gas effects. The expert panel continues to be interested in this information, as it becomes available. In particular, the expert panel is interested in information obtained from isotopic evaluations from samples of concrete obtained within the reactor building. Based on insights obtained from evaluations of current information, one suggestion is offered at this time:

Area 2 Suggestion:

Continue planned additional isotopic evaluations.

Evaluations of concrete samples extracted from a common location for all three units would be of interest. For example, further information that could be acquired regarding isotopic composition from the shield plugs above the drywell heads at 1F1 and 1F3 would be useful to evaluate fission product release and transport [See Appendix C Information Need RB-7]. In particular, it would be helpful to have additional data against which to assess off-site ^{134}Cs to ^{137}Cs isotopic ratios.

5. AREA 3 – DEBRIS END-STATE

The expert panel also selected debris end-state as an area of emphasis with respect to examination information. Post-accident examinations at TMI-2 [25] demonstrated that the end-state of debris is an important finding from forensics inspections and critical for developing and validating models within severe accident analysis codes. Debris end-state location information is of particular interest at Daiichi because comparisons can be made between the multiple units that were affected. In addition, it is desired to gain insights about debris coolability, the effects of saltwater, and debris spreading from examinations. As discussed within this section, answers to questions about debris end-state are also required by TEPCO for successful and safe completion of D&D activities. High radiation levels limit the ability to gain inspection information to address debris end-state information at this time. Hence, indirect observations coupled with analysis model predictions provide a preliminary basis for debris removal planning.

This section summarizes current findings obtained by TEPCO from the Fukushima Daiichi forensics efforts as they relate to debris end-state configuration and how these findings can be used to address uncertainties in such analyses. To that end, we begin by first providing a summary of relevant information obtained to date, with emphasis placed on how these findings relate to reactor safety evaluations. This is followed by a summary of our preliminary insights and a brief description of the limitations of these insights. We then provide a few recommendations and observations for additional RST program activities that could provide additional insights related to information gained from available forensics information. The section concludes with a suggestion to TEPCO for additional information that would be beneficial regarding debris end-state evaluations.

5.1 Questions for Reactor Safety and D&D

Available information was evaluated to address the following questions which are of international interest for reactor safety and to Japan in making decisions for future D&D activities:

- What is the mass, composition, morphology, and decay heat of materials relocated to the lower head?
- Has vessel lower head failure occurred? What was the timing and mode of such failure (e.g., has global, localized, or penetration failure occurred)?
- What is the mass, composition, decay heat, morphology, and spreading characteristics of material relocated from the lower head?
- Are analysis model improvements needed to predict observed end-state?
    - Are there any observed effects from saltwater addition?
    - Can observed end-states of debris and structures be used to estimate the amount of combustible gas generated during relocation and during molten core concrete interactions (MCCIs)?
    - Can information from one unit be used to confirm analysis models and predict conditions in another unit?
- Can information provide insights about the integrity of structures within the PCV and the reactor building?

Answers to these questions have important safety impacts. By obtaining prototypic data from the three units at Daiichi, there is the potential to reduce modeling uncertainties. Improvements in our modeling capabilities can be used to confirm or enhance, if needed, accident management strategies with respect to containment venting, water addition, and combustible gas generation.

Answers to the above questions also are of interest with respect to Phase II D&D activities. As discussed in Section 2.3.4, debris end-state characterization studies provide key input for decisions related to the debris retrieval approach, development of the fuel debris retrieval equipment, and implementation of fuel debris retrieval activities with reduced risks from radioactive materials. In particular, improved models for predicting the timing and mode of vessel failure and the mass, composition, and decay heat of material relocated to and from the lower head are of interest in making decisions related to the methods for debris removal and measures needed for worker protection from damaged structures and from radiation.

5.2 Information Summary

As discussed in Section 1.3.1, US experts identified information needs that could be addressed through examinations at Fukushima Daiichi. Requested information needs from the reactor building, PCV, and RPV that relate to debris endstate location are summarized in Tables 14 through 16. These tables also note if any information is available to address these information needs (see Appendix C). As these tables indicate, limited direct information has been obtained to date regarding debris endstate location for the affected units. This information has been gathered using robotic examinations and stand-off methods such as muon tomography. Aside from direct information, there are several other data sources available to indirectly infer the debris end-state location in each unit. For all units, there are data from instruments, such as temperature information obtained during and immediately after the accident, gas concentration data from the gas treatment system, and neutron and gamma detector data from subcriticality monitoring systems. This section reviews the available information that provides insights related to debris end-state.

Table 14. Area 3 information needs from the reactor building

Item	What/How Obtained	Use[a]	Data Available[b]
RB-14	Chemical analysis of white deposits found in 1F1 HPCI room using XRD or other methods.	AE, AM, DD	NA

[a]Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[b]Some information available [Green]; NA: no information available [Orange].

Table 15. Area 3 information needs from the PCV

Item	What/How Obtained	Use	Data Available
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3).[c]	AE, AM, DD	NA
	b) If vessel failed, 1F1, 1F2, and 1F3 PCV liner examinations (photos/videos and metallurgical exams).	AE, AM, DD	NA
	c) If vessel failed, photos/video, RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor.	AE, AM, DD	A
	d) If vessel failed, 1F2, and 1F3 concrete erosion profile; photos/videos and sample removal and examination	AE, AM, DD	NA
	e). If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage corium hang-up	AE, AM, DD	NA

[a]Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).
[b]Some information available [Green]; NA: no information available [Orange].
[c]Although some images have been obtained, they do not indicate if RPV failed or show any relocated core debris.

Table 16. Area 5 information needs from the RPV

Item	What/How Obtained	Use[a]	Data Available[b]
RPV-5	Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)	AE, AM, DD	A
	Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.)	AE, AM, DD	NA

[a]Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[b]Some information available [Green]; NA: no information available [Orange].

5.2.1 Thermocouple Measurements

Figures 20 through 22 provide thermocouple measurements [10] obtained from 1F1, 1F2, and 1F3, respectively, for a time period spanning several months following the accident. These measurements provided the first indication of where core debris likely resides, and equally important, where it is not. In particular, water injection was shifted from the fire protection (FP) to feedwater (FDW) injection systems for the three units in the April-May timeframe. However, RPV thermocouple (TC) measurements indicated temperatures well above the coolant saturation temperature after this switch was made, particularly for 1F2 and 1F3. This provided an early indication that all core debris may not have been cooled using the FDW injection pathway. As a reminder, the feedwater for a BWR is introduced near the top of the RPV (see Figures 20 through 22) and then flows down along the exterior surface of the core barrel to the core inlet. This led TEPCO and the technical support community to conclude that there may be significant leakage path(s) in the bottom region of the reactor vessel for all three units (e.g., BWR recirculation pumps are known to leak under severe accident conditions [123]). In such cases, some fraction of the coolant was able to bypass the core debris; and the material was not fully cooled.

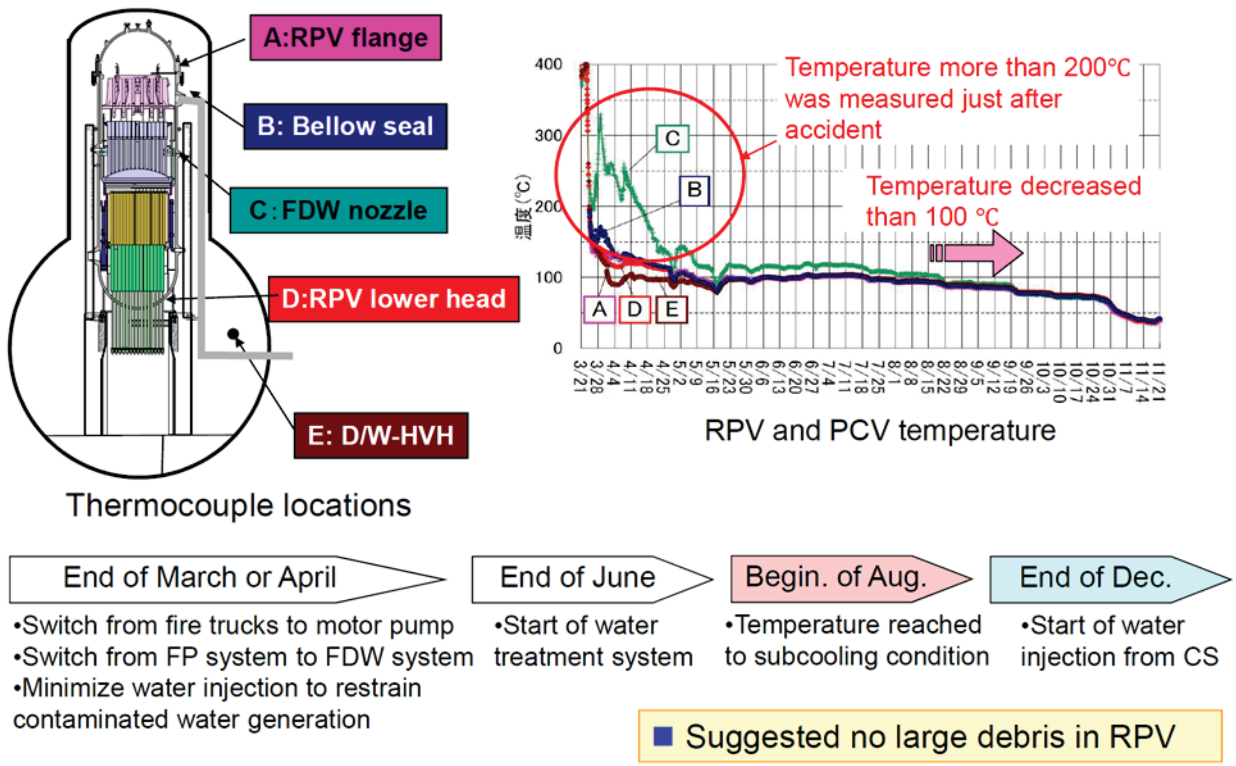


Figure 20. RPV temperature measurements for 1F1 following the accident. (Courtesy of TEPCO [10])

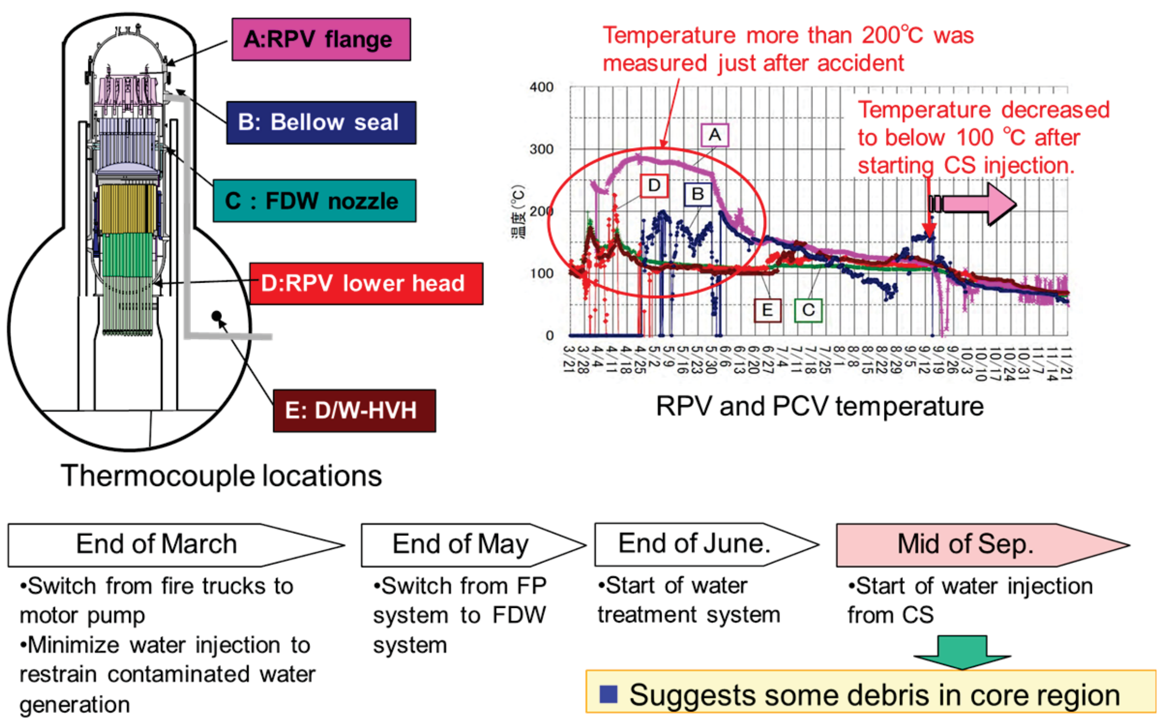


Figure 21. RPV temperature measurements for 1F2 following the accident. (Courtesy of TEPCO [10])

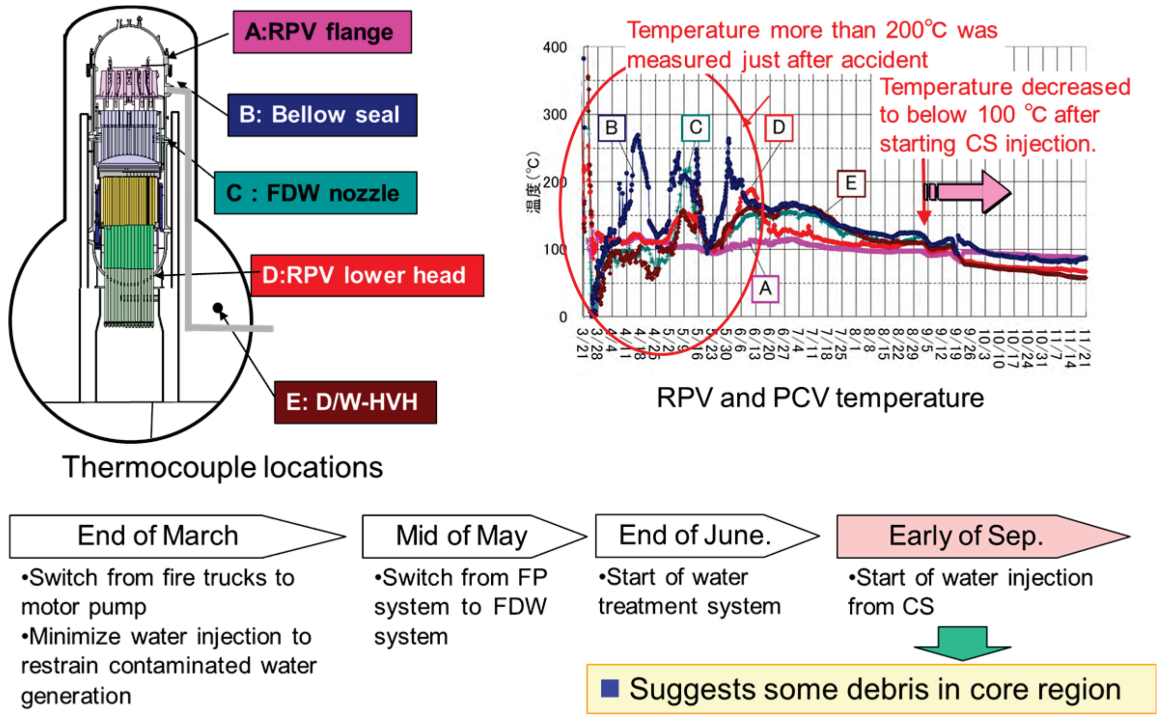


Figure 22. RPV temperature measurements for 1F3 following the accident. (Courtesy of TEPCO [10])

On this basis, TEPCO changed the water injection from the FDW system to the core spray (CS) system in the September 2011 timeframe for 1F2 and 1F3, while this change was made in late December 2011 for 1F1. This injection method directly introduces a water spray from above the core. As shown in Figures 21 and 22, this changed injection point caused the RPV temperatures for 1F2 and 1F3 to be reduced to coolant saturation temperature, which is the expected condition when core debris is covered with water. However, this change had little if any impact for 1F1, for which the RPV temperatures had already fallen below saturation. This, along with indications from water level instrumentation not increasing, led TEPCO and many in the technical support community to conclude that some fraction of fuel remained in the RPV for 1F2 and 1F3, but most of the core debris was likely ex-vessel for 1F1. Note that this information does not rule out the possibility of ex-vessel core debris for 1F2 and 1F3; however, there is likely some fraction of core debris in-vessel that caused elevated temperatures to occur when water was introduced via the FDW system.

This information is consistent with early US [27,28] as well as international [124] code predictions of likely debris locations for the three units based on modeling conducted relatively soon after the accident. Since that time, further refinements of these analyses have not changed these same basic conclusions. The picture is clearest for 1F1 which was essentially a hands-off station blackout until ~15 hours into the accident sequence [e.g., an event in which all onsite and offsite alternating current (ac) power is lost and in which no successful mitigating actions are taken]. At this point, operators were able to reflood the core with seawater. However, the predictions are less consistent for 1F2 and 1F3 where operators were able to maintain some degree of core cooling by various means for the first several days of the accident. The uncertainties arise as to the effectiveness of water injection (due to elevated PCV pressure), and the effectiveness and extent of backup cooling system operation under severe accident conditions; this situation was compounded by a general lack of functioning instrumentation (as well as the fact that surviving instrumentation had in many cases been pushed well outside the normal operating envelope; this statement is true for the TC measurements shown in Figures 20 through 22) that would allow the actual plant conditions to be ascertained.

Aside from these general observations, it is noteworthy that the TC data in Figures 20 through 22 may provide valuable information that could be used to further evaluate likely core debris end-state locations using system-level codes. In particular, these codes have the ability to calculate heatup of the RPV, and through appropriate nodalization, it may be possible to calculate temperatures on structures that correspond to locations where the measurements were obtained in Figures 20 through 22. The core debris distribution calculated by the codes would influence the temperature responses at these locations, and the extent that the codes are able to reproduce the signatures shown in Figures 20 through 22 may provide further insights on likely debris distributions. This type of analysis is relevant to the ongoing MAAP-MELCOR cross-walk activity [29]; i.e., these two codes predict quite different in-vessel core melt behavior and, as a result, RCS failure modes. These modeling differences may be reflected in long-term RPV temperature predictions that could, by comparison with the data, provide an indication of likely relocation mode(s), which is one of the key questions being addressed as part of the crosswalk activity.

The results of these measurements, as well as the supporting code analyses, help to inform D&D activities. In particular, the results indicate that TEPCO will likely be faced with the need to remove core debris not only from the RPVs for at least two units, but also from the PCV for 1F1. Finally, these measurements have also been very useful in terms of informing post-Fukushima enhancements to severe accident guidance (SAG). In particular, the data illustrate the benefit of injecting though core sprays for BWRs; this method optimizes the probability that core debris will be contacted by and cooled with the injected water, even if there are leaks in the pressure vessel.

5.2.2 Images from Inspections within the PCV

TEPCO has also obtained other valuable information from within the 1F1 PCV using robotics examinations through a containment penetration, i.e., the 'X-100B' penetration (see Figure 23).[125] Prior to the accident, this penetration was shielded on the interior of the PCV to reduce the radiation level in the reactor building. The first piece of information gathered when this penetration was opened was that the lead shielding appears to have melted during the accident (see Section 3.2.1). Lead melts at 328 ºC; temperatures this high in the PCV are hard to rationalize unless one postulates vessel failure and core debris discharge into the PCV.

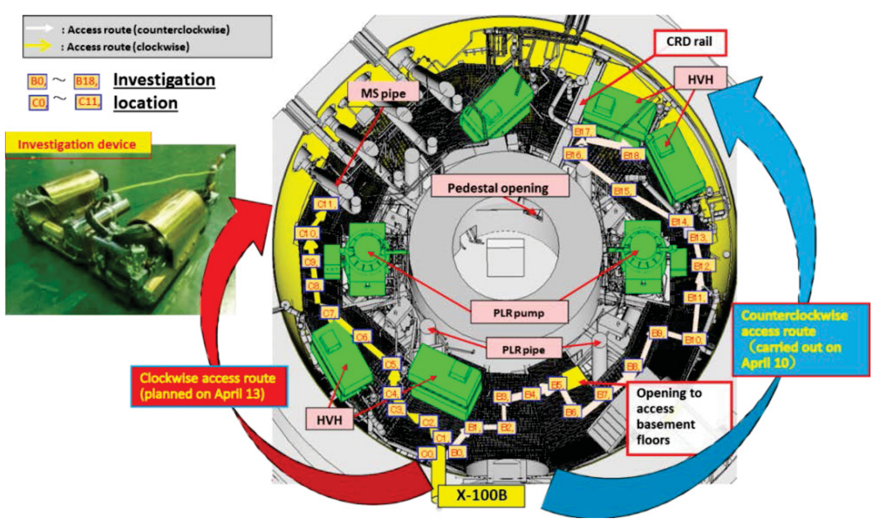


Figure 23. Location of X-100B in 1F1 PCV and pathways of robotic examinations completed on April 10, 2015 (Courtesy of TEPCO [125]).

Upon gaining access through this penetration, TEPCO initially lowered a video camera through the catwalk to the drywell floor to measure water level, temperatures, and radiation levels inside the PCV. These inspections showed that there was no core debris on the drywell floor at this location, which is ~ 130 degrees from the pedestal doorway (Figure 23). This finding was important as it provided a data point for assessing predictions of ex-vessel core melt spreading based on MAAP and MELCOR pour scenarios as calculated with MELTSPREAD [30]. As is evident from Figure 24, the measurement indicates that the MELTSPREAD prediction of spreading distance based on MAAP pour conditions over-predicts the actual spreading distance. Conversely, this single data point is insufficient to gauge the accuracy of the MELTSPREAD-MELCOR prediction as the spreading prediction for that case is limited to the vicinity of the pedestal doorway. Nonetheless, this initial observation through this penetration has been useful in reducing the range of possibilities regarding the extent of melt spreading in 1F1.

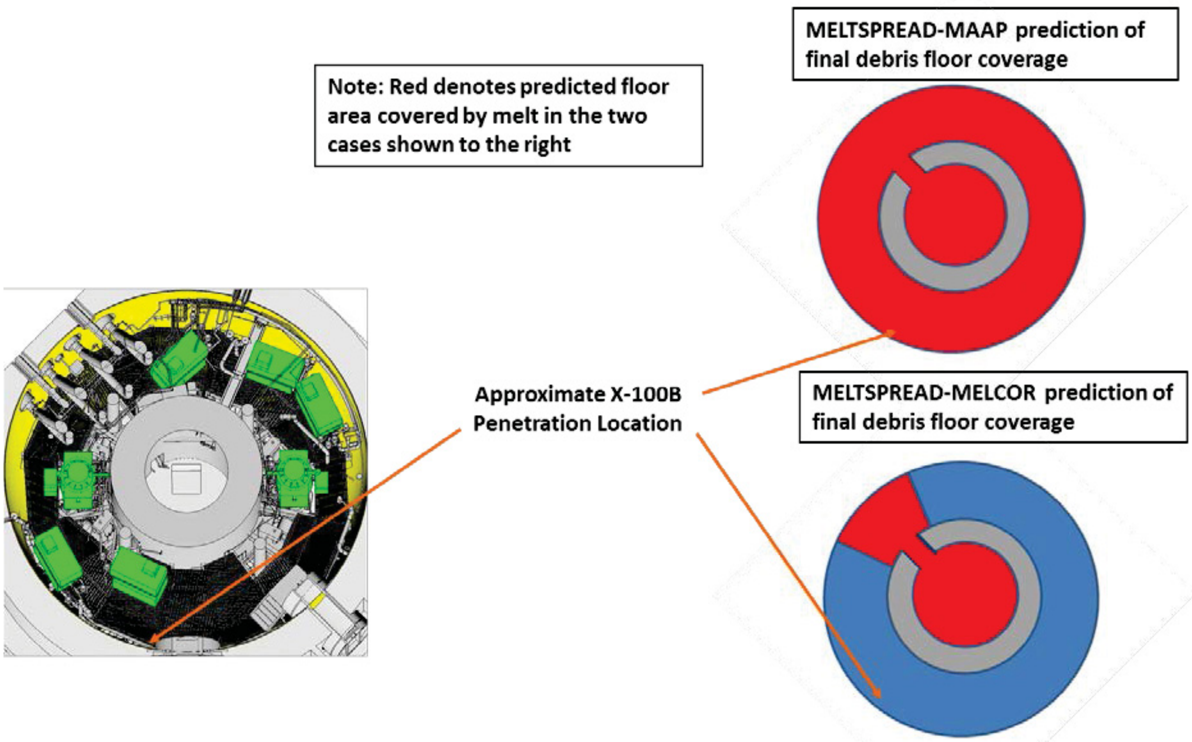


Figure 24. Approximate location of X-100B penetration relative to predictions of core debris spreading in 1F1 (Courtesy of TEPCO and ORNL [30,125].

5.2.3 Visual Images within the Reactor Building

Another important finding regarding ex-vessel behavior is the discovery by TEPCO that the sand cushion drain line is leaking in 1F1.[102] This indicates that there is a leak through the PCV liner. Examinations did not detect water leakage from the bellows on the downcomer, but observations were limited. The MELTSPREAD analyses of liner heatup (Figure 25) indicate that the liner would not have been ablated through based on either the MAAP low pressure (LP) or MELCOR pour scenarios [27,28,30]; however, the liner would have been heated significantly, resulting in a vulnerability to failure by creep rupture due to the elevated containment pressures (~ twice the design pressure) at the time of the accidents. Hence, liner failure is consistent with code predictions and measured radiation levels in the 1F1 reactor building.

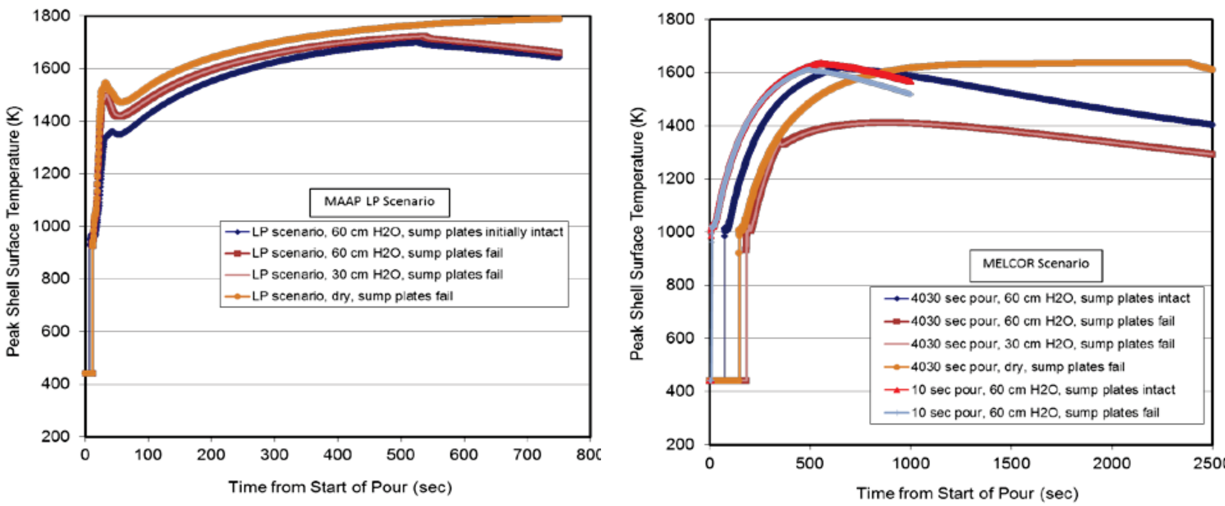


Figure 25. MELTSPREAD predictions of liner heatup due to heat transfer from impinging melt for 1F1 based on a MAAP low pressure (LP) scenario [28] (left) and MELCOR[27] (right) melt pour conditions. (Courtesy of ORNL, [30])

5.2.4 Muon Tomography Evaluations

Muon tomography measurements using scintillation detectors are another information source that has been extremely valuable for evaluating debris end-state conditions for 1F1 (see Figure 26) [14]. Using this approach, high density fuel should show up as dark regions in captured images due to muon attenuation. As shown in Figure 26, the core region appears to be essentially devoid of core material. The findings for 1F1 are consistent with previously described system-level code analyses. [27,28]

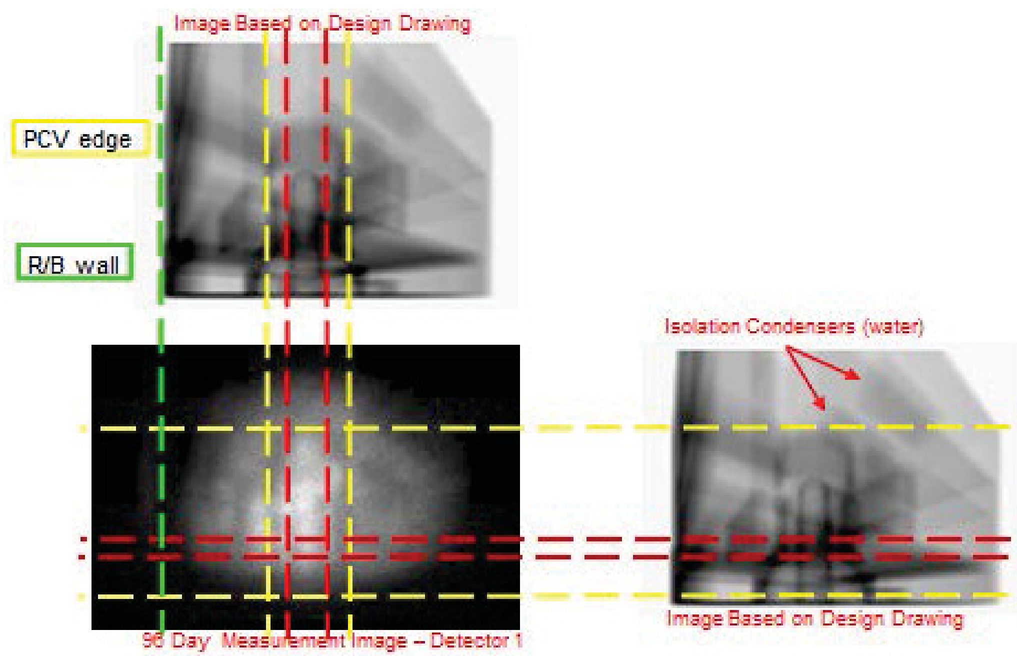


Figure 26. Images of 1F1 obtained using muon tomography with scintillation detectors (The lower left image is measured; the other two images were calculated. Dashed lines are provided to show location of identified geometrical features). (Courtesy of TEPCO [14])

5.2.5 Gas Cleanup System Measurements

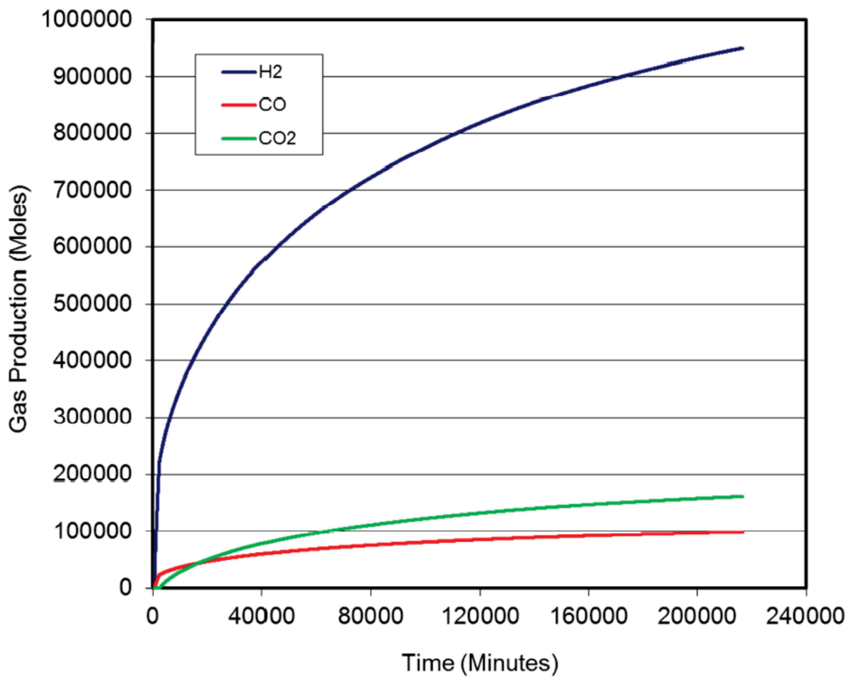
Aside from the three main data sources discussed above, additional data gathered by TEPCO that has been extremely useful for reactor safety evaluations are the PCV atmospheric composition measurements obtained from 1F1 and 1F2 a few months after the accidents occurred [126] (see Table 17). As discussed earlier, system-level code analyses [28, 27] of these accidents indicate that RPV failure with follow-on MCCI was likely, particularly for 1F1. These analyses further indicate that combustible gas production due to MCCI contributed significantly to hydrogen accumulation and eventual combustion on the refueling floor of 1F1. TEPCO used these measurements to determine if H_2, CO, and CO_2 gases produced through MCCI were still present in the containment atmospheres [125]. Based on the low levels of these gases (see Table 17), it was concluded that the core debris was likely quenched and stabilized, thereby terminating MCCI. Note that trace levels of H_2 and CO_2 were still present at the time these samples were taken, but parasitic H_2 production would be expected by dissociation of water caused by radiolysis, whereas CO_2 would be introduced (in dissolved form) from water that is continuously injected to cool the core/core debris.

Table 17. Concentrations (Vol %) of H_2, CO, and CO_2 in the PCV atmospheres of 1F1 and 1F2 measured several months after the accidents [126]

Sample Location	H_2	CO	CO_2
1F1 (September)	0.154	<0.01	0.118
1F1 (September)	0.101	<0.01	0.201
1F1 (September)	0.079	<0.01	0.129
1F2 (August)	0.558	0.014	0.152
1F2 (August)	1.062	0.016	0.150
1F2 (August)	<0.001	<0.01	0.152

The question arises as to what level of MCCI gases would be expected within the containment atmospheres of the affected units if the core debris had not been quenched. In order to evaluate this potential scenario, a CORQUENCH MCCI calculation was performed [19] with debris coolability mechanisms (i.e., melt eruptions and water ingression) disabled to estimate likely gas concentrations at the time the samples were taken (August-September 2011 timeframe). The calculation was run out to 150 days, which corresponds to mid-August. The MAAP prediction of the melt composition at the time of vessel failure was utilized [28] as input into the CORQUENCH simulation. The analysis was limited to the sump volume because this would be the likely location for deep accumulations if the debris was not cooled. Limiting the analysis to the sumps thus represents 26 % of the total core mass available for core-concrete interaction [19].

The results (Figure 27) indicate that if the MCCI had not been stabilized, then ~1.8 Sm^3/hr of noncondensable-combustible gases would still be generated through parasitic, long-term core-concrete interaction at the time the gas samples were drawn. Although cladding and core structural steel initially present in the melt would be oxidized in the first few hours of the accident, long-term production of combustible gases H_2 and CO could occur due to oxidation of iron that is present as rebar in the concrete; reinforcement was assumed to be present at a level of 6 wt %.



Gas Species	Production Rate Mid-August, 2011 (Sm^3/hour)
H_2	1.37
CO	0.14
CO_2	0.30

Figure 27. Expected long-term noncondensable-combustible gas production from ablation within the 1F1 sump if debris had not been quenched and stabilized. (Courtesy of ANL, [19])

Given the data in Figure 27, the expected levels of noncondensable-combustible gases from MCCI in the containment atmosphere for 1F1 at the time the samples were taken are summarized in Table 18. These levels reflect the fact that a N_2 purge into containment has been maintained at a rate of 10-30 Sm^3/hr. The results indicate that if the core debris had not been quenched and stabilized, then the expected gas production would have been easily detectable by gas mass spectroscopy. Thus, if the vessel did fail in 1F1 and MCCI occurred, then the indications are that the ex-vessel core debris was quenched and stabilized. This is a *significant finding for reactor safety*. Ex-vessel debris coolability is one of the key technical issues raised in the wake of the TMI-2 accident. The fact that the debris was stabilized was one of the factors that allowed TEPCO to declare that cold shutdown conditions had been achieved and the accident effectively terminated.

Table 18. Expected atmospheric concentration of MCCI gases in the 1F1 containment atmosphere if MCCI was not terminated [19].

Gas	**Source**	**Expected atmospheric concentration (Vol. %) at N_2 PCV purge rate of:**	
		10 Sm^3/hr	**30 Sm^3/hr**
N_2	PCV purge	84.7	94.4
H_2	MCCI	11.6	4.3
CO_2	MCCI	2.5	0.9
CO	MCCI	1.2	0.4

Aside from providing general insights related to phenomenology and reactor safety evaluations, the preliminary findings from TEPCO regarding ex-vessel coolability in 1F1 motivated the developers of the MAAP and MELCOR codes to integrate advanced debris coolability models (e.g., see [30]) into their modules for calculating ex-vessel MCCI behavior. These updates improve the ability of these codes to realistically reproduce actual severe accident behavior and, thereby, support severe accident mitigation planning for BWRs and PWRs.

5.2.6 Deposits from Leaking Penetrations

As a part of characterizing high dose rate locations within the 1F1 reactor building,[15] TEPCO has discovered a high dose rate deposit that could provide additional insight regarding the accident progression, particularly as it relates to debris location. Namely, TEPCO found white sediment that was deposited by leakage within the HPCI PCV wall penetration; see Figures 15 and 16.

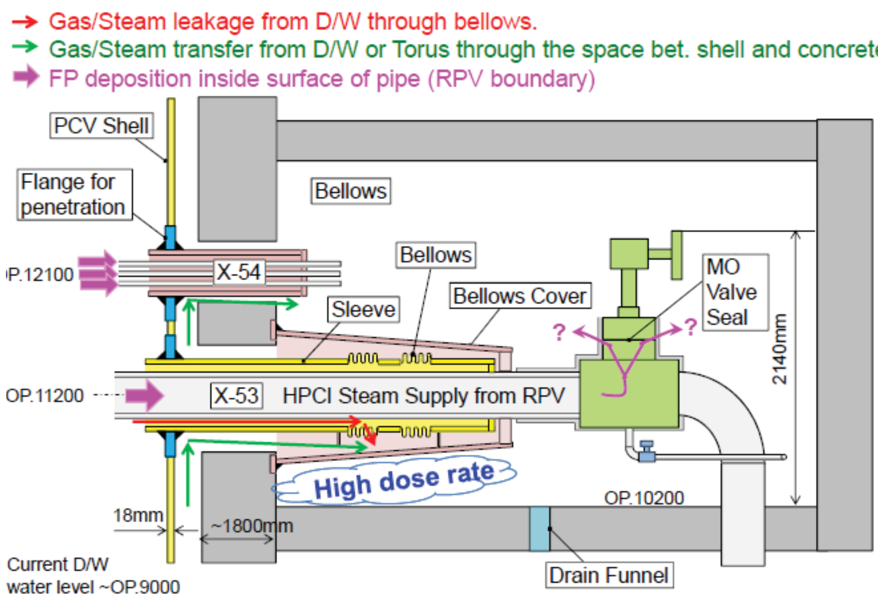


Figure 28. Depiction of HPCI pump steam supply penetration showing location of deposit with high dose rate. (Courtesy of TEPCO [15])

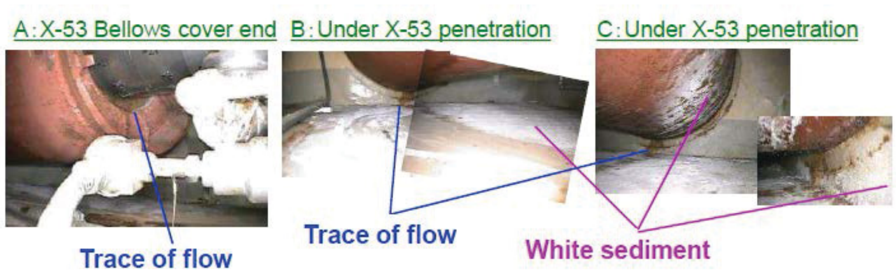


Figure 29. Photographs of leakage location and sediment. (Courtesy of TEPCO [15])

TEPCO indicates that the likely leakage point within this penetration is the bellows seal [15]. As shown in Figure 15, one pathway by which fission products could migrate from the PCV to the bellows location is by leakage through the drywell liner. As noted earlier (Section 5.2.3), there is other evidence indicating probable liner penetration; i.e., sand cushion drain line leakage. Although the chemical composition of this deposit has not yet been determined, one possible candidate is NaCl (i.e., salt). It is well known that extensive seawater injection occurred at 1F1 as operators struggled to cool the damaged core. Salt increases coolant corrosivity which could have been a contributing factor to the development of the leak. However, this is believed to be unlikely because corrosion is a longer term issue, and sea water injection was only maintained for the first few days of the accident progression.

A second possibility regarding the composition of the sediment is amorphous SiO_2 which is also white. This material dominates aerosol produced during core-concrete interaction, [127] particularly for the siliceous concrete type used in Fukushima plant construction. The only significant source of Si within the PCV is in the form of SiO_2 within the concrete. Furthermore, the only credible method by which SiO_2 could be aerosolized is by core-concrete interaction. Thus, if the sediment is found to contain Si, this would be a very strong indication that the RPV failed and that core-concrete interaction ensued as part of the 1F1 accident progression.

5.3 Insight Summary and Limitations

In summary, available inspection information and analysis results have led to several important insights about debris end-state location and associated answers to Section 5.1 questions.

Thermocouple data:

- Available thermocouple data and information about water injection are consistent with analysis results suggesting that vessel failure occurred in 1F1, 1F2, and 1F3.
- Available thermocouple data suggest that most of the debris relocated through the failed 1F1 vessel and that a smaller mass of debris relocated through the failed 1F2 and 1F3 vessels.
- Available thermocouple data confirm the benefit of water addition measures adopted in new SAG (e.g., the benefit of core injection to cool not only any residual core debris remaining in-vessel, but also any core debris that may have relocated ex-vessel. For BWRs, this goal is best achieved by core spray injection).
- Additional analyses, using more refined nodalization, may provide more detailed information about the mass, composition, and decay heat of relocated debris.

Visual Images within PCV and Reactor Building:

- Images obtained from the 1F1 X-100B penetration indicate peak temperatures of 328 ºC (due to the absence of lead shielding that melts at this temperature). Such high gas space temperatures support the hypothesis that core debris relocated ex-vessel during the accident, as these temperatures are very difficult to rationalize otherwise.
- The absence of debris in images obtained from a camera inserted into the X-100B penetration suggests that current US analyses obtained with MELTSPREAD using results from MAAP pour conditions are over-predicting spreading of debris in 1F1, whereas results obtained with MELCOR are indeterminate at this time.
- Images showing that the 1F1 sand cushion drain line is leaking suggest a failure in the PCV liner. Such failures could be from creep rupture due to the elevated containment pressures (~ twice the design pressure) at the time of the accidents. Liner failed in 1F1 is consistent with MELTSPREAD code predictions and with measured radiation levels in the 1F1 reactor building (see Section 4).

Muon Tomography Investigations:

- Results from muon tomography results suggesting that much, if not all, of the fuel debris is absent from the 1F1 core region; this is consistent with results from MELCOR and MAAP analyses.

Gas Cleanup System Measurements:

- Concentration measurements from gas cleanup systems, in conjunction with CORQUENCH analysis results, confirm the conclusion that the debris within 1F1 PCV was quenched and stabilized.

Although very informative, the amount of information obtained thus far on debris locations is limited. In particular, there have been no direct determinations of the location of the core debris. A few observations obtained with remote cameras have shown where the core debris is *not* in the PCV for 1F1, which in itself is valuable information. Muon tomography images are also providing data on debris locations, but the resolution of the images is limited. Finally, TEPCO has effectively used TC measurements on the RPV coupled with variations in water injection flowrate and location to make inferences on debris location. One limitation with this last technique is the fact that many of the TCs on the RPV were pushed well outside their qualification envelop during the accident, which raises questions about calibration as well as potential failures that are difficult to diagnose; e.g., formation of false junctions within the TCs that can provide erroneous indications of temperature at a given location.

Despite these limitations in the available information, it is important to note that the information has provided many insights on accident progression as well as important data for validation of both system-level and separate effect codes that are used for reactor safety evaluations.

5.4 Recommendations

As summarized in Section 5.2, both system-level [27,28] as well as separate effect [30] code analyses have provided tangible predictions for evaluation against the debris end-state information being obtained by TEPCO. In a rough sense, these calculations can be considered to be half-blind benchmarking exercises that are useful in gauging the accuracy and adequacy of the models as additional information on debris end-state conditions becomes available. A few additional analysis activities were identified as part of this initial evaluation that would be beneficial in terms of benchmarking the models, reducing modeling uncertainties, and further informing D&D efforts at the site.

Area 3 Recommendation 1:

As alluded to in Section 5.2, refine the MAAP and MELCOR RPV nodalization schemes for the RPVs of Units 1 through 3 with the aim of predicting the measured temperatures shown in Figures 20 through 22. The post-accident debris locations predicted inside the RPV, coupled with changes in water addition rate and location, may provide a means for assessing the accuracy of the debris end-state predictions. This comparison may also provide insights into appropriate modeling of in-core melt progression that has been identified as a key uncertainty in the MAAP-MELCOR crosswalk exercise [29].

Area 3 Recommendation 2:

Repeat the MELTSPREAD-CORQUENCH analysis that was originally done for 1F1 [30] for 1F2. Various system-level code analyses have shown the potential for vessel failure at this unit also. However, if the vessel did fail, it likely occurred much later in the accident sequence due to the continued operation of RCIC for ~72 hours in an unregulated mode. This study may be useful in showing that it is unlikely that the melt contacted the liner in this late pour scenario, or if it did, that the shell likely remained intact due to reduced thermal loading. As discussed in [10], no evidence of liner failure has been found for 1F2, and this would provide a means for rationalizing that observation relative to the finding that the liner in 1F1 has been damaged.

5.5 Suggestions for Additional Information

The results of this forensic analysis activity related to debris end-state conditions has illustrated the intrinsic value of information obtained by TEPCO for providing insights on accident progression, informing SAG enhancements, and validating severe accident codes that are used for plant safety evaluations.

Regarding additional information needs for this topical area, the primary need is for higher fidelity data on debris locations. In this early stage of the D&D process, initial insights are being gained on ex-vessel conditions. Due to the high radiation levels, the only practical means for obtaining this data is through stand-off methods which TEPCO has actively and successfully pursued; i.e., muon tomography and robotics inspections. These methods have already proven to be valuable, and it is clear that TEPCO is learning as they go in these areas. Based on insights obtained from evaluations of current information, one suggestion is offered at this time:

Area 3 Suggestion:

> Perform chemical analysis of high radiation deposits or particles found inside the reactor building (1F1, 1F2, and 1F3); e.g., the white deposits from the HPCI room using FE-SEM, XRD, etc.

The list of information needs in Appendix C was updated to reflect this additional item (e.g., see RB-14 in Appendix C).

6. AREA 4 – COMBUSTIBLE GAS EFFECTS

During the November 2015 meeting, the expert panel agreed to include the area of combustible gas effects as a fourth topic of investigation. The panel included this topic because it was recognized that damage within the affected units at Daiichi could provide important insights related to the sources for and transport of combustible gas and the ignition point and damage caused by each explosion.

This section summarizes insights with respect to reactor safety and future D&D activities that we hope to gain by reviewing examination information from Daiichi. With this goal in mind, available visual information related to these questions are summarized. This is followed by a summary of our preliminary insights and a brief description of the limitations of these insights. We then provide recommendations and observations for additional RST program activities that could provide further insights related to information gained from forensics examinations. Suggestions for future TEPCO examinations to support these activities are also identified.

6.1 Questions for Reactor Safety and D&D

Available information was evaluated by US experts to address the following questions which are of international interest for reactor safety and to Japan for completing feasibility studies to support D&D activities:

- Where and how did ignition occur, and how did flame propagate from ignition floor to other floors?
- How does combustible gas migrate during a severe accident?
- Can damage to structures provide insights about combustion characteristics, such as ignition location and pre-explosion concentration of combustible gas, that can be used in guidance for severe accident mitigation?
- What are D&D impacts with respect to hydrogen combustion related to the integrity of structures within the RB and PCV and to radiation release and transport?
- What severe accident measures should be implemented to reduce damage associated with combustible gas explosions?
- How much does MCCI contribute to combustible gas generation effects?
- Are analysis model improvements needed for predicting combustible gas generation, migration through degraded seal and penetrations, and accumulation?
- Can analysis provide insights related to D&D worker safety and radiation exposure?

Answers to these questions have important safety impacts. By obtaining prototypic data from the affected units at Daiichi, there is the potential to reduce modeling uncertainties. Improvements in our modeling capabilities can be used to confirm or enhance, if needed, accident management strategies for addressing the consequences of combustible gas phenomena. This information and associated analyses with improved severe accident codes offer the potential for insights that may be beneficial to Japan in their D&D activities. In particular, improved models for predicting the events at Daiichi may provide important insights related to radionuclide transport and deposition, which is important in characterizing worker dose during D&D activities and to structural damage, which is important in assessing hazard potential.

6.2 Information Summary

As discussed in Section 1.3.1, US experts identified information needs that could be addressed through examinations at Fukushima Daiichi. Requested information needs from the reactor building and PCV that relate to combustible gas generation are summarized in Tables 19 and 20. These tables also note if any information is available to address these information needs (see Appendix C). Visual information includes photos and videos taken during and after the explosions. In addition, radiation survey and seismic acceleration data were used to provide insights about combustible generation, transport, and combustion. In addition, TEPCO reports [128 through 130] evaluating damage associated with explosions at the affected units and TEPCO unresolved issues reports [21 through 24] contained important information on this topic. Data from plant instrumentation were also used to provide insights. This section reviews this information and provides insights related to reactor safety and D&D activities.

Table 19. Area 4 information needs from the reactor building

Item	What/How Obtained	Use[aa]	Data Available[bb]
RB-3a	Photos/videos of damaged walls and structures (1F1)	AE, AM, DD	A
RB-3b	Photos/videos of damaged walls and structures (1F3)	AE, AM, DD	A
RB-3c	Photos/videos of damaged walls and structures (1F4)	AE, AM, DD	A
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	AE, AM, DD	A
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	AE, AM, DD	A
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	AE, AM, DD	A
RB-9	DW Concrete Shield Radionuclide surveys (1F1, 1F2, and 1F3 - before debris removed)	AE, AM, DD	A
	DW Concrete Shield Radionuclide surveys (1F1 - after debris removed)	AE, AM, DD	NA
	DW Concrete Shield Radionuclide surveys (1F3 - after debris removed)	AE, AM, DD	A
	Photos/videos around mechanical seals and hatches and electrical penetration seals (as a means to classify whether joints were in compression or tension)	AE, AM, DD	A
RB-10	Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations)	AE, AM, DD	A
RB-11	Photos/videos and available information on 1F1, 1F2, and 1F3 containment hardpipe venting pathway, standby gas treatment system and associated reactor building ventilation system	AE, AM, DD	A
RB-13	Photos/videos of 1F1, 1F2, and 1F3 main steam lines at locations outside the PCV.	AM, DD	A

[aa]Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, and PM – Plant maintenance (see Appendix C for more information).

[bb]Some information available [Green]; NA: no information available [Orange].

Table 20. Area 4 information needs from the PCV

Item	What/How Obtained	Use[cc]	Data Available[dd]
PC-1	Tension, Torque, and Bolt Length Records (prior and during removal); Photos/videos of head, head seals, and sealing surfaces (1F1, 1F2, and 1F3). [ee]	AE, AM, DD	NA
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3).[ff]	AE, AM, DD	NA
	b) If vessel failed, 1F1, 1F2, and 1F3 PCV liner examinations (photos/videos and metallurgical exams).	AE, AM, DD	NA
	c) If vessel failed, photos/video, RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor.	AE, AM, DD	A
	d) If vessel failed, 1F2, and 1F3 concrete erosion profile; photos/videos and sample removal and examination	AE, AM, DD	NA
	e). If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage corium hang-up	AE, AM, DD	NA
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	AE, AM	NA

6.2.1 TEPCO Reports

In 2011, two of the reports issued by TEPCO [128, 129] evaluate damage associated with explosions at the affected units. The purpose of the reports was to find out whether it was necessary to implement urgent measures for seismic reinforcement rather than analyze the cause of the explosions. These reports contain important and useful information, such as reactor building damage surveys in the form of photos and building damage diagrams (see Figure 30 for 1F1; Figures 31 through 33 for 1F3; and Figure 34 for 1F4).

[cc]Use: AE – Accident evaluation (code modeling updates), AM- Accident management and prevention, DD – Decontamination and Decommissioning, PM – Plant maintenance (see Appendix C for more information).

[dd]Some information available [Green]; NA: no information available [Orange].

[ee]Available information is limited to the shield plug.

[ff]Although some images have been obtained; images do not indicate if RPV failed or show any relocated core debris.

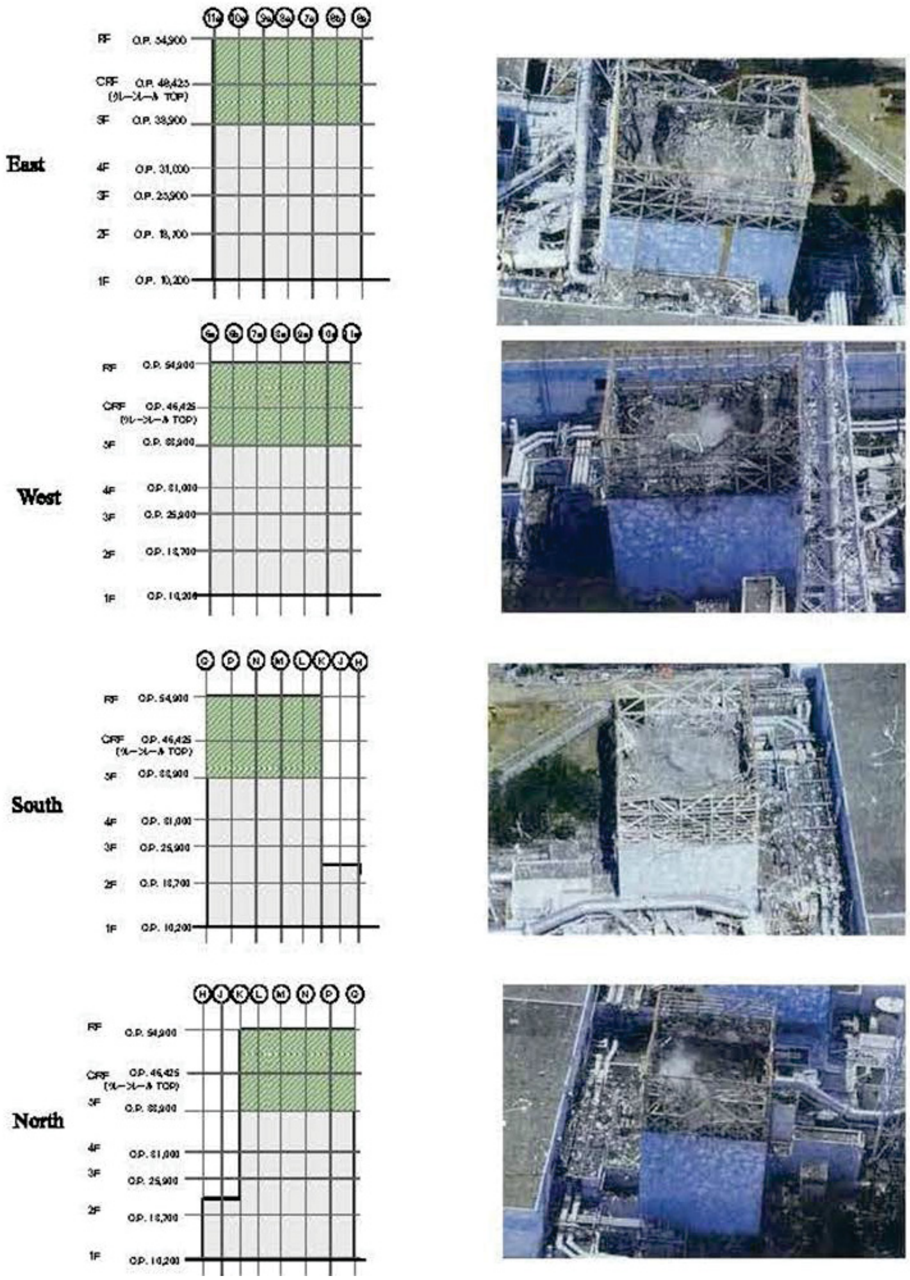


Figure 30. 1F1 reactor building damage following explosion. (Courtesy of TEPCO, [128])

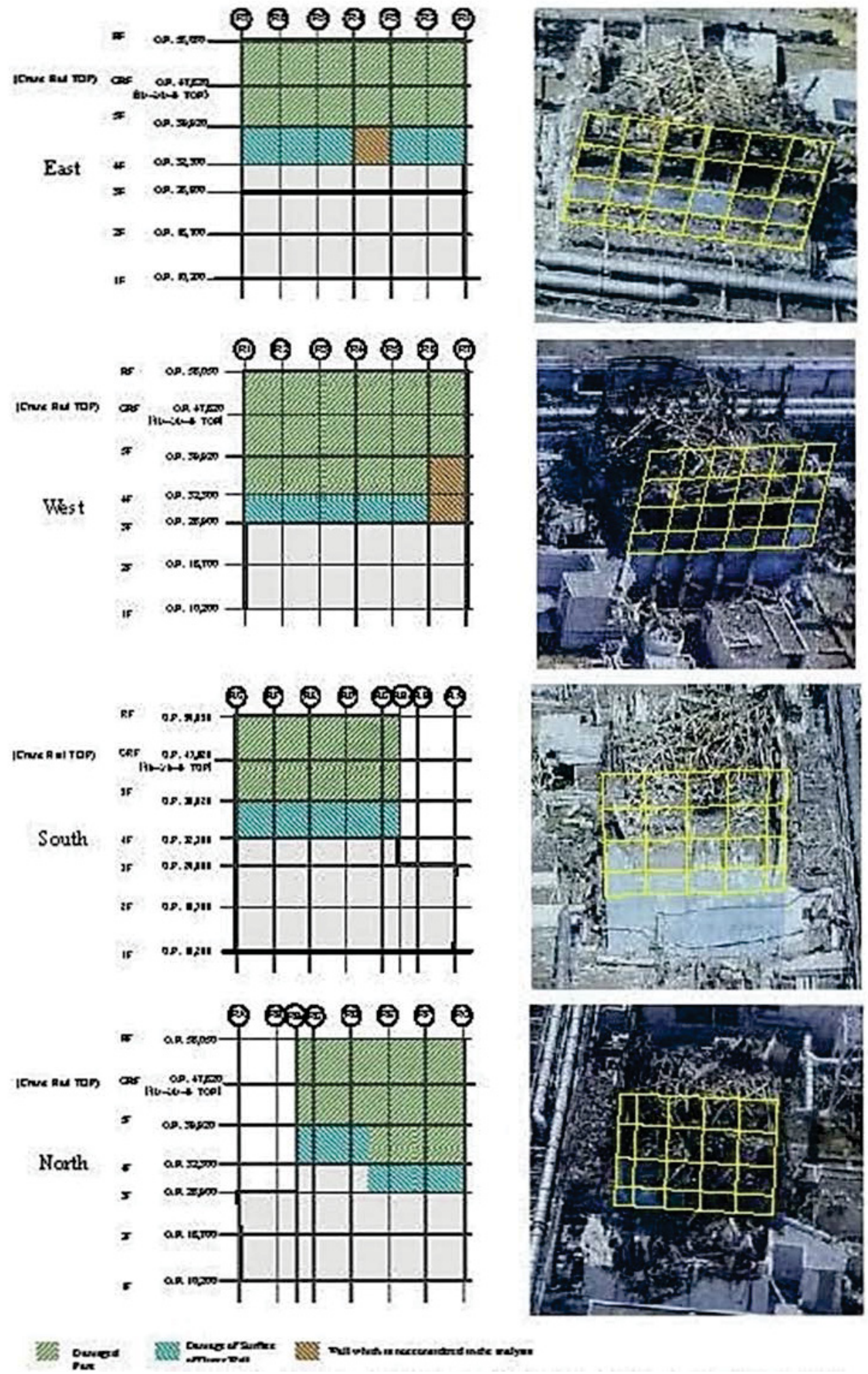


Figure 31. 1F3 reactor building damage following explosion. (Courtesy of TEPCO, [129])

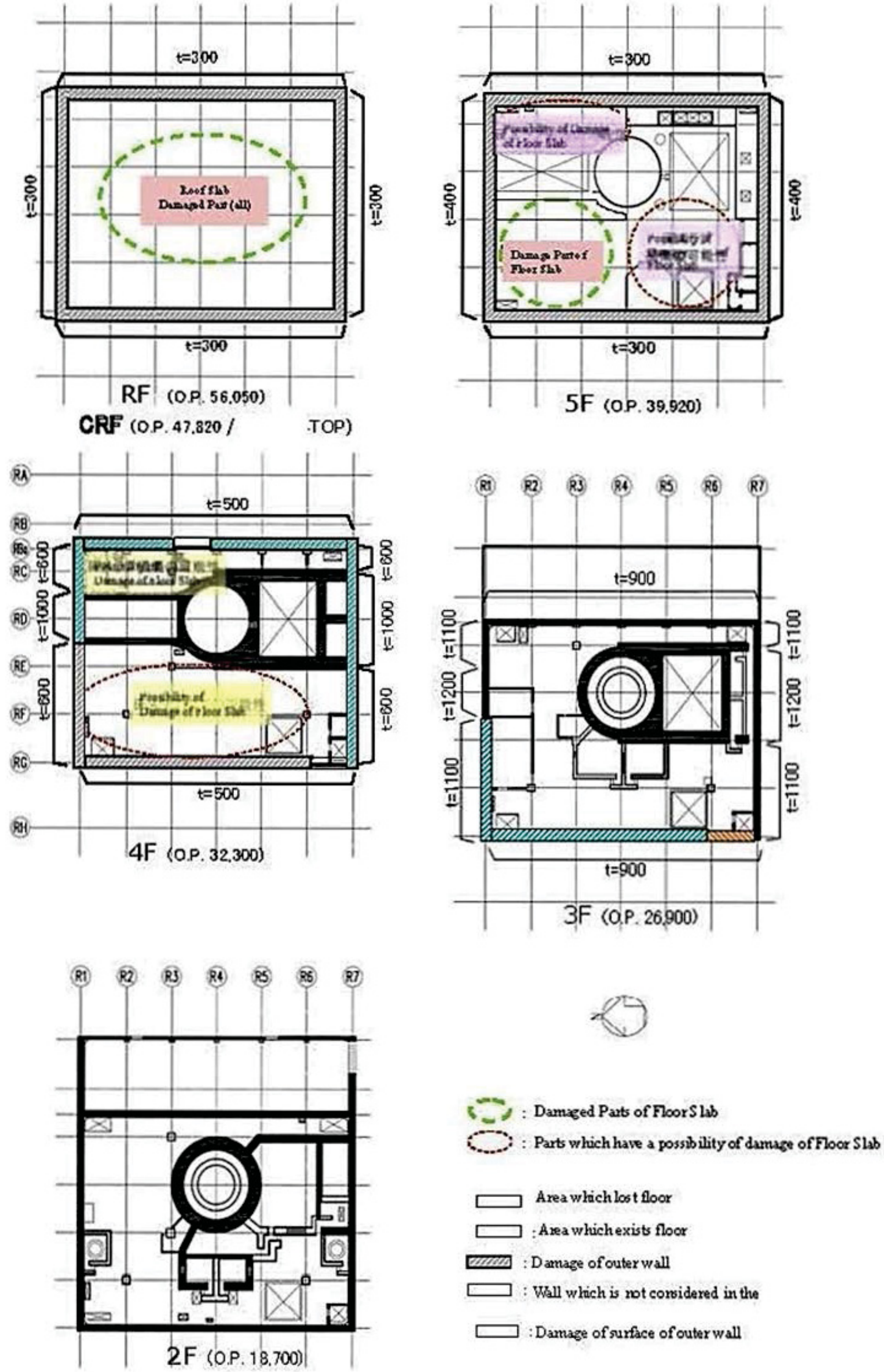


Figure 32. Damaged areas on 1F3 reactor building floor plan. (Courtesy of TEPCO, [129])

Figure 33. Damaged areas on 1F3 reactor building floor plan. (Courtesy of TEPCO, [15])

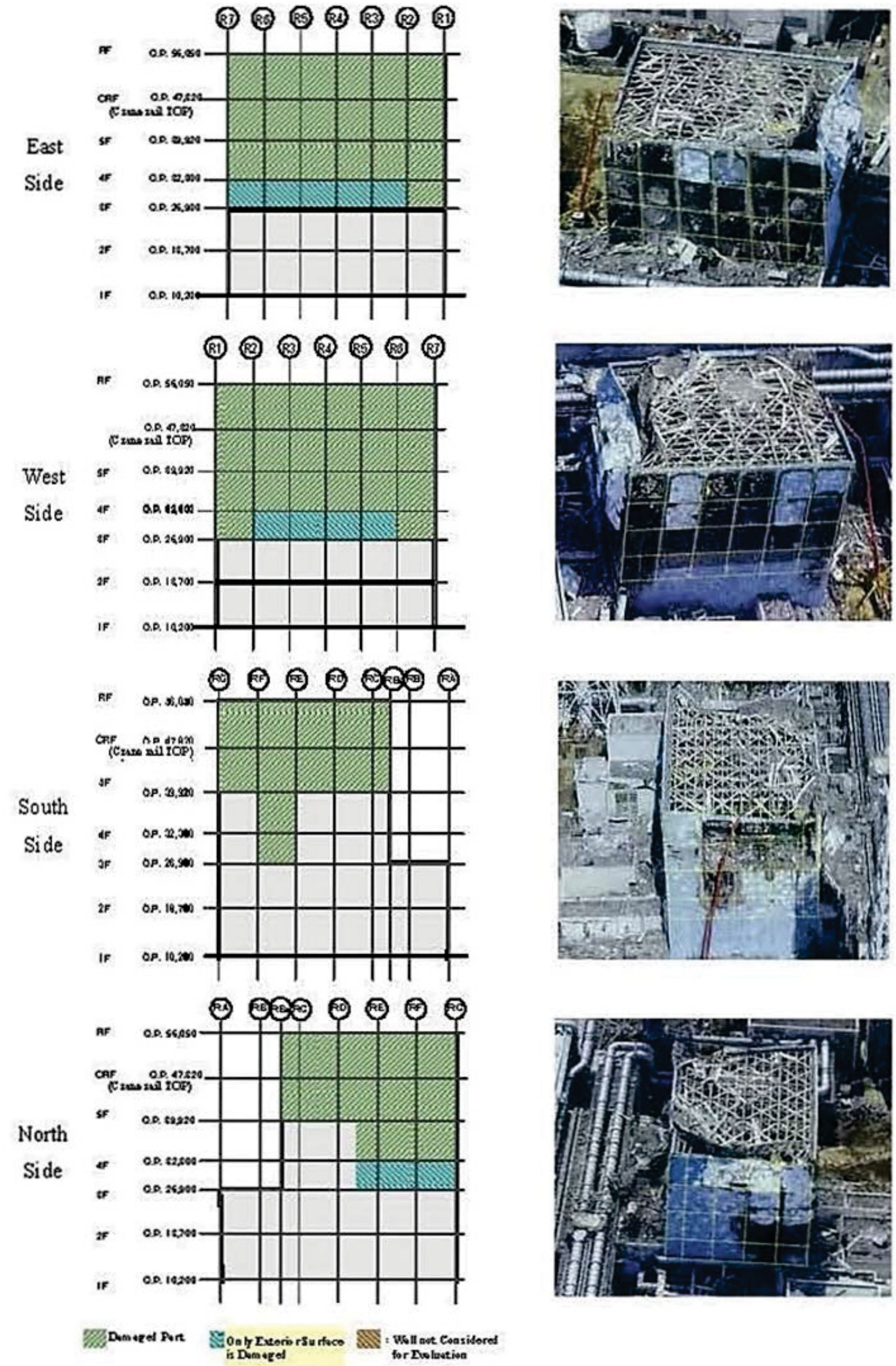


Figure 34. 1F4 reactor building damage following explosion. (Courtesy of TEPCO, [128])

6.2.2 1F1 Explosion

The upper part of the 1F1 reactor building above the operating floor (the 5th floor) experienced an apparent hydrogen explosion on March 12, 2011 at 3:36 pm, approximately 25 hours after the seismic event. [130] It is believed that this hydrogen was primarily due to reactions between steam and the fuel zircaloy cladding. As discussed in Sections 3 and 4, the exact pathway through which the hydrogen flowed is unknown, but available information on the explosion damage suggests that it leaked into the building through degraded seals on the head of the PCV and accumulated in the refueling floor (5th floor) to a significantly high level (see Figure 35).

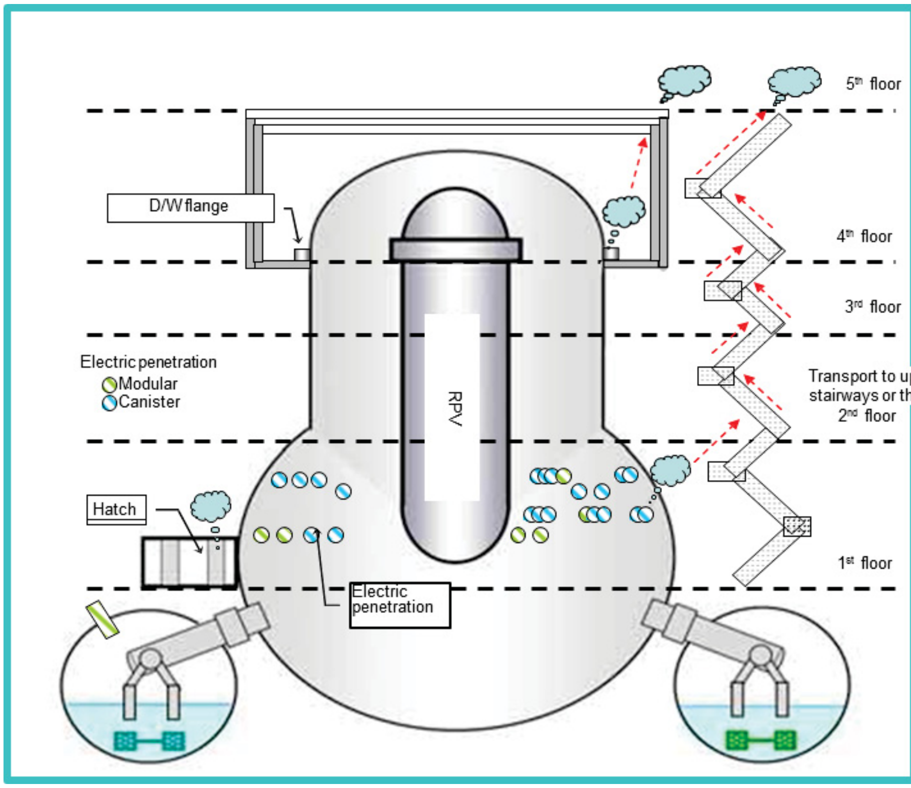


Figure 35. Inferred leakage paths; flow paths differ from 1F1 and 1F3 due to system configuration. (Courtesy of TEPCO, [131])

As documented by TEPCO,[128] the explosion heavily damaged the 5th floor but did no damage to the floors below except for limited damage observed near the equipment hatch opening in the southwest corner of the 4th floor.[132] The walls on the 5th floor consisted of a steel framework structure fixed with steel plates and were susceptible to internal pressure increases. The collapse of the walls resulted in a release of inside pressure minimizing any damage to structures below the 5th floor.

The hydrogen explosion at 1F1 significantly hindered other recovery efforts. Debris from the explosion damaged power lines that had been laid down at 1F2 as well as the power lines being readied at 1F3. This adversely impacted work being done to restore power at both 1F2 and 1F3. In addition, as discussed in Section 4.2.1.2, it is believed that pressure waves from the 1F1 explosion caused the 1F2 reactor building blowout panel to open (see Figure 16). This opening is believed to have averted an explosion in the 1F2 building because it allowed any accumulated hydrogen to vent.

6.2.3 1F3 Explosion

The upper part of the 1F3 reactor building above the refueling floor (the 5th floor) also underwent an apparent hydrogen explosion on March 14, 2011 at 11:01 am. [130] Videos show that the explosion and damage were much more extensive than the 1F1 explosion. In fact, the 1F3 explosion damaged the fire engines and hoses being readied at 1F2 to the extent that they could no longer be used.

In [129], TEPCO observed the following damage:

- Collapsed steel framework and concrete were piled up on, and above, the 5th floor (Figure 31);
- The east side wall was lost on the 5th floor, but the columns survived;
- The west side wall was lost on the 5th and 4th floors; the 3rd floor was partially damaged except for the elevator area on the southwest corner;
- The south side wall was lost on the 5th floor and was partially damaged on the 4th floor;
- The north side wall was lost on the 5th floor and on part of the 4th floor; the columns were lost;
- The north-west part of the floor on the 5th floor was also damaged; part of the collapsed steel framework and concrete accumulated on the 4th floor (Figure 32);
- The walls on the 4th floor were largely damaged;
- The overhead crane dropped onto the floor of the 5th floor;
- The roof of the turbine building experienced some damage;
- The top of the two-story Radwaste Building adjacent to the 1F3 RB also experienced some damage.

More recent photos taken in 2014 after debris removal show that about one fourth of the concrete floor of the 5th floor was severely damaged with big holes through the floor (Figure 33).

Available information on the explosion damage suggests that there was a likely accumulation of extremely high concentrations of combustible gases in both the 4th and 5th floors at the time of the explosion. However, it is unknown at this time how such a level of accumulation occurred in the 4th floor.

6.2.4 1F4 Explosion

The 1F4 explosion in the reactor building is estimated to have occurred on March 15, 2011 at 6:14 am.[130] There were no videos capturing the explosion when it occurred. Unlike 1F1, the structure of 1F4 is a reinforced concrete structure whose wall resistance is supposedly stronger against inside pressure. Most of the roof slab and walls were blown off leaving only the frame structure of the pillar and beams.[128] Most walls on the 4th floor and some walls on the 3rd floor were damaged (Figure 34).

Evaluations of the explosion at 1F4 have led TEPCO to conclude that vented gases, including hydrogen, flowed from 1F3 into 1F4 (Figure 36). This conclusion is based upon the following:

- ***Examinations of the filter train of the standby gas treatment system (SGTS) at 1F4.*** Measurements indicate that the concentration of radioactive materials accumulated at the outlet was higher than at the inlet. This implies that contaminated gas flowed into the 1F4 SGTS pipe from the outlet to the inlet (see Figure 37).
- ***Field investigations near the 1F4 SGTS duct on the 4th floor***. Damage to the 4th floor (along with the floors above and below this floor) and remaining pieces of the SGTS exhaust duct work support the concept that the explosion originated at this location (see Figure 38).

- ***Examinations of the fuel in spent fuel pool for 1F4.*** At the time of the accident, the 1F4 reactor had been completely defueled with the fuel placed in the spent fuel pool for planned work on RPV internals. Thus, the only credible source of hydrogen for this unit during the accident would have been undercooling of the assemblies in the fuel pool. However, all assemblies were subsequently removed from the 1F4 fuel pool, and physical observations made as each assembly was removed indicate no damage (over and above that experienced during normal reactor operation).

These findings are consistent with the hypothesis that the wetwell vent flow from 1F3 travelled into the 2nd floor of 1F4 and then into other areas of the 1F4 reactor building via pipes and the SGTS ducts.[gg]

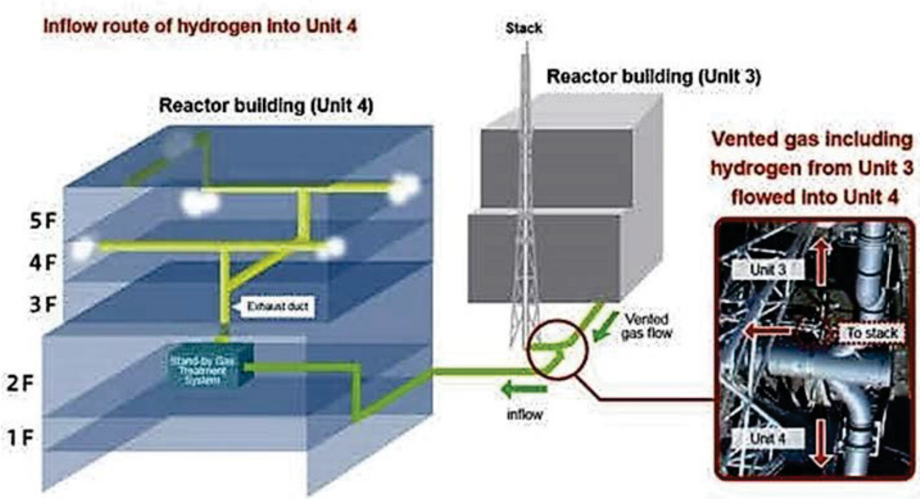


Figure 36. Hydrogen transport path from 1F3 to 1F4. (Courtesy of TEPCO, [130])

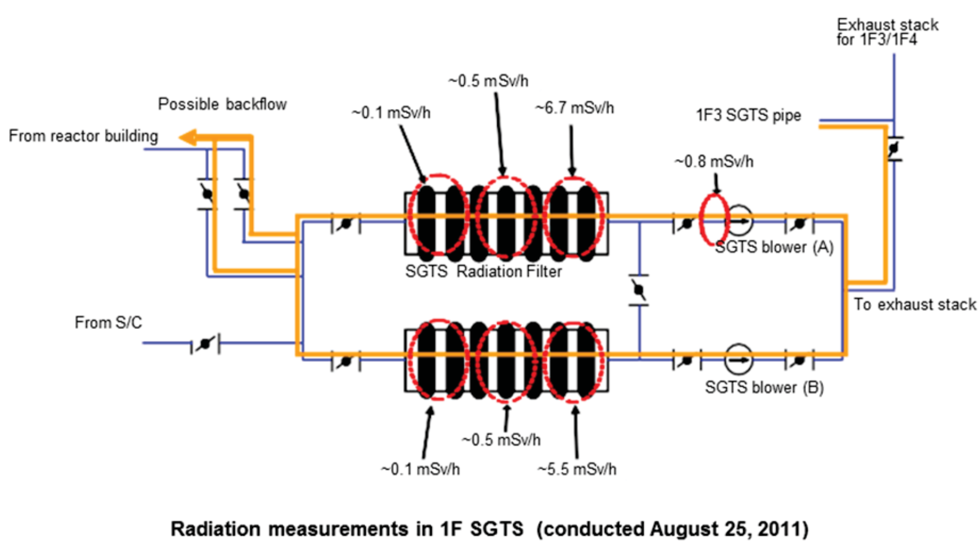


Figure 37. 1F4 SGTS radiation measurement results. (Courtesy of TEPCO, [130])

[gg] Normally, the SGTS is on standby or shut down, and system valves are closed to prevent flow of vented gas between adjacent units. However, venting of the 1F3 PCV was conducted while all AC power sources were lost, and the resulting line configuration allowed vented gas to flow from the 1F3 PCV into 1F4 through a SGTS pipe.

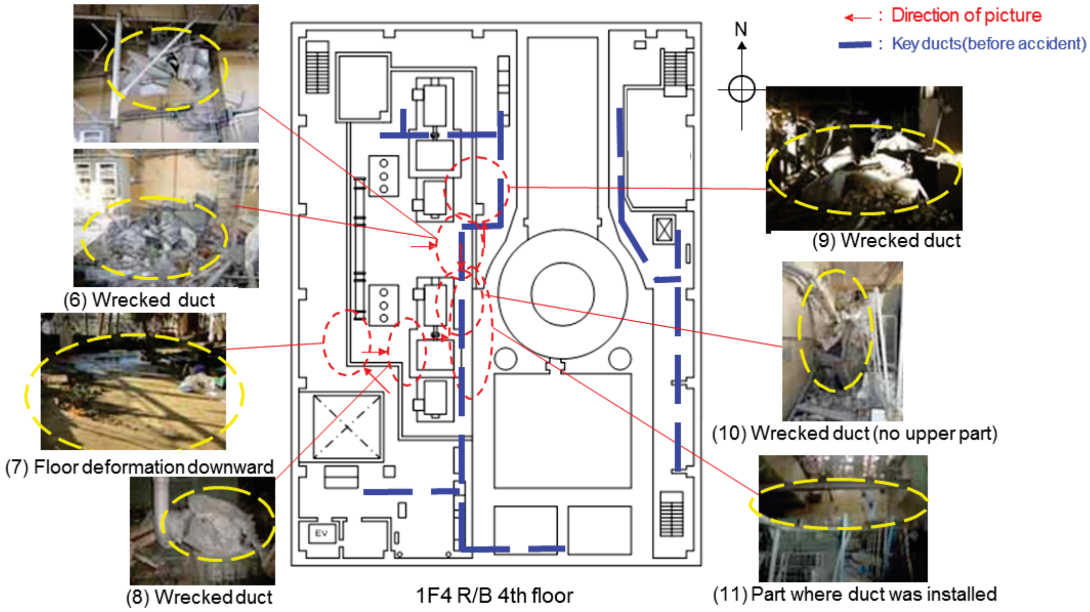


Figure 38. Field investigation of the 1F4 4th floor (Courtesy of TEPCO, [130])

6.2.5 Video Capturing Explosions

Figure 39 shows one-second interval snapshots from videos capturing the 1F3 explosion of 1F3 for the first 9 seconds (about the duration of the explosion). [133] Figure 40 shows millisecond-time scale snapshots of images before and after the appearance of a "flash fire" (an orange flame) which first appeared in the 0.099-s frame and disappeared in the 0.495-s frame.

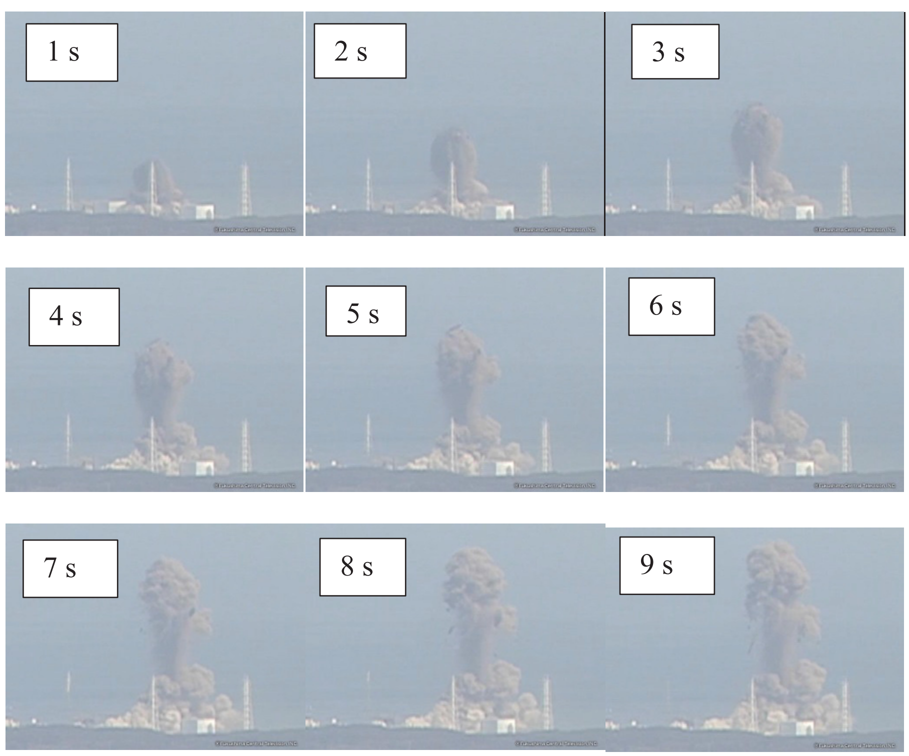


Figure 39. Images of 1F3 explosions at 1-second intervals. (Courtesy of FCT, [133])

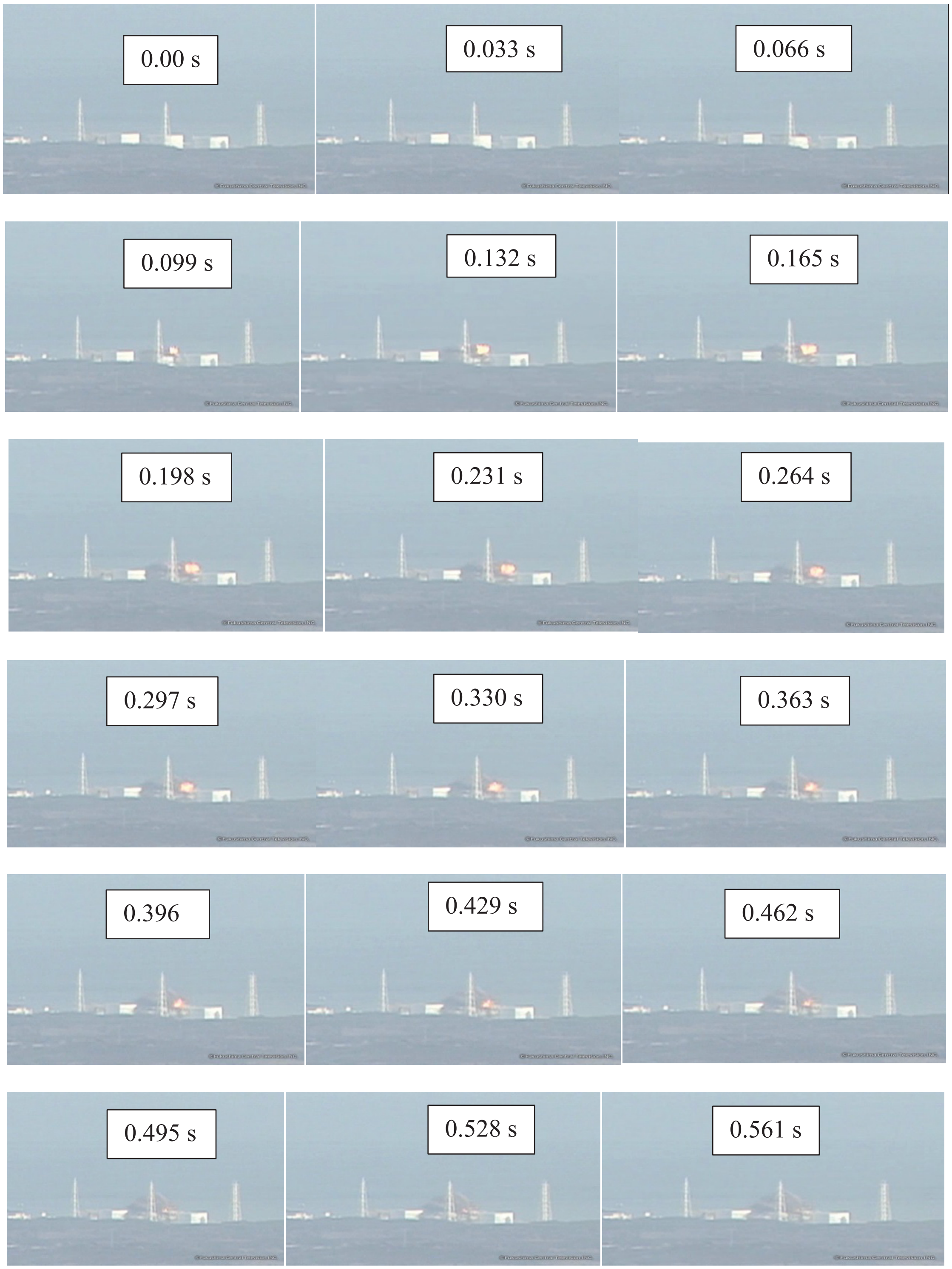


Figure 40. Images of 1F3 Explosions during an appearance of a flash fire (Courtesy of FCT [133])

US and Japanese expert evaluations of information related to hydrogen combustion [133,134,135,136] indicate that the hydrogen explosion at the 1F3 reactor building was very different from the explosion of 1F1. The explosion at 1F1 was a fast deflagration of hydrogen accumulated in the operating bay (5th floor) of the reactor building.[137] A video of the explosion indicates the presence of a condensation shock wave that was not fast enough to reach transition to detonation. As shown in Figure 41, the explosion “smoke” appeared light, suggesting it was primarily dust. The “smoke” was dispersed relatively close to the building in the vertical direction and was directed northward (toward the left in the picture) due to the prevailing wind at that time. The building roof and side panels were blown away by the explosion, but concrete pillars remained intact with little damage. The explosions at 1F3 were quite different in appearance and much more energetic (Figure 39). There appeared to be at least two explosions. The first was similar to the 1F1 explosion which was a deflagration of hydrogen (and possibly CO) accumulated in the operating bay (5th floor). The flame front apparently propagated to the 4th floor (based on damage seen in Figure 31) and resulted in the deflagration of flammable gaseous mixture accumulated in the 4th floor at that time. The second explosion was directed vertically with an almost perfect spherical fireball appearing above the building and shooting up high into the sky (about three times the vent stack height). Large chunks of materials appeared to be carried upward with the fireball. Unlike the explosion at 1F1, available 1F3 images indicate that concrete pillars on the building top floor were highly damaged. The “smoke” resulted from the 2nd explosion appeared in darker color of dust and debris than that of the 1st explosion which appeared white (in the image) and remained at lower elevations close to the building (Figure 39). This is a strong indication that the combustible gases involved in the 1st and 2nd explosions at 1F3 came from different sources as discussed in Section 6.2.6.

1F1 **1F 3**

Figure 41. Images of 1F1 explosion compared with 1F3 explosion. (Courtesy of FCT [133])

6.2.6 Plant Data

The time of the 1F3 explosion, 11:01 am, March 14, 2011, was about the same time when the 1F3 PCV pressure instantaneously dropped from about 0.53 to about 0.36 MPa (Figure 42). The instantaneous drop in pressure is believed to correspond to drywell upper head seal failure. This PCV failure would release a hot hydrogen-steam gaseous mixture into the 5th floor of the reactor building around the drywell plug lifted by pressure buildup below it. It was possible that these hot vented gases (among other random ignition sources) could have ignited hydrogen gas, which leaked earlier and accumulated on the 5th floor (and 4th floor) of the reactor building. Ignition of this hydrogen resulted in the first explosion whose burning mechanism (i.e., deflagration) was the same as the 1F1 explosion. Then, a catastrophic failure of the 1F3 drywell upper head seal would have provided a large, continual supply of hydrogen gas from inside the drywell through the failed head seal. (It is noted that it would require only about 4 psig/0.03 MPa differential pressure (Δp) to lift the drywell shield plug where $\Delta p = \rho g H$, and for the purpose of approximation, ρ= shield plug density ~2330 kg/m^3, g=9.8 m/s^2, and H=shield plug thickness ~ 1.2 m. This magnitude of Δp was achievable with the prevailing DW pressure at the time of the explosion.) This combustible gas jet entrained surrounding air as it moved upward and burned in the form of unconfined "gas cloud explosion" as a large fireball emanating from the reactor building into the sky. The first appearance of combustible gas jet was in a form of the flash fire [visible from the video snapshot at the very beginning (less than 0.5 s) of the 1F3 explosion transient shown in Figure 40]. At the beginning, the combustible gas jet just started to form. The gas cloud was then initially burned as a flash fire. The flash fire anchored at the same location for about 0.4 seconds. When more combustible gas jet came out, the flash fire disappeared and the energetic second explosion began its process as shown in Figure 39. The observed combustion phenomena were the consequence of the PCV blowdown that supplied combustible gas to the second explosion, which was initially visible as a flash fire. The amount of blowdown gases (nitrogen, steam, hydrogen and possibly, carbon monoxide) could be as much as the amount of gases released from the PCV, which experienced a 1.7 bar decrease in pressure at the prevaling temperature, decreasing from 0.53 to about 0.36 MPa in about 9 seconds (as seen from the video snapshot in Figure 39). There is a clear linkage between the PCV blowdown, the second explosion smoke shape and duration, the observed flash fire, and the PCV failure (fast pressure drop).

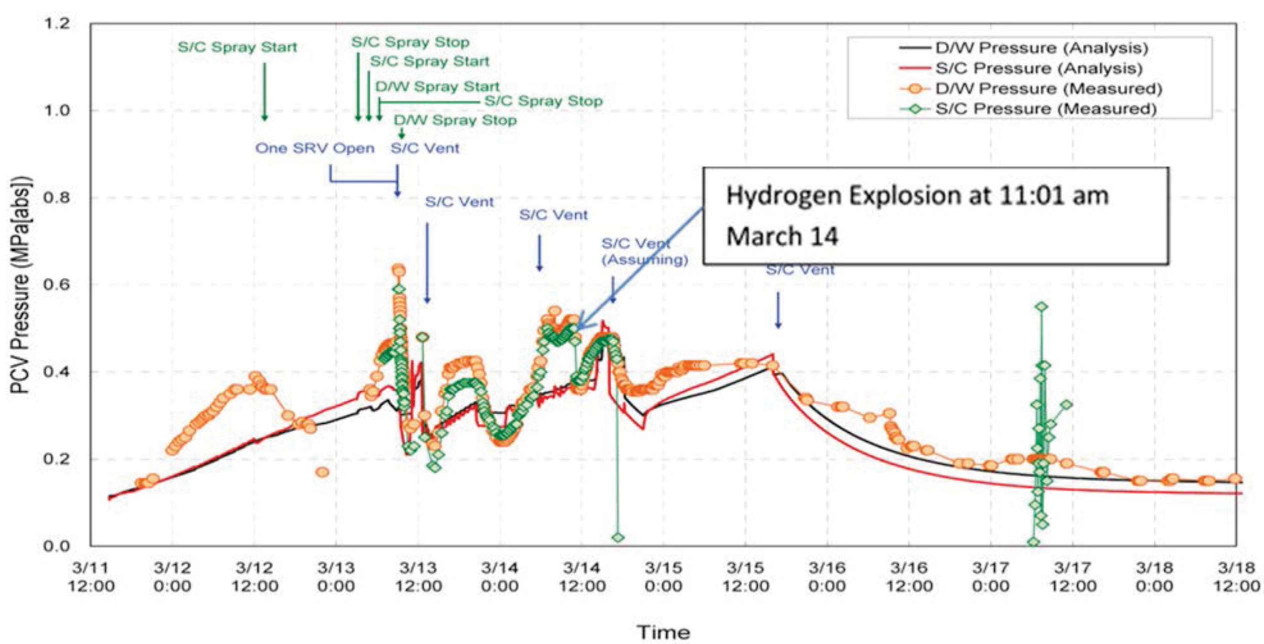


Figure 42. 1F3 PCV pressure - Rapid drop in PCV pressure coincides well with the timing of 1F3 explosion at 11:01 am March 14, 2011 (Courtesy of TEPCO, [137])

6.3 Insight Summary and Limitations

In summary, available inspection examination information and plant data have led to several important insights about combustible gas effects and answers to Section 6.1 questions.

- ***The 1F3 explosion was not a stand-alone randomly occurring event.*** The 1F3 explosion was most likely initiated by failure of the drywell upper head seal when it was at high PCV pressure

(0.53 MPa). The released hot gas was likely the ignition source and became a source of fuel that supplied to the highly energetic fireball burning at and above the building. The hot hydrogen/steam mixture was released as a jet from the periphery of the lifted DW shield plug. The fireball appeared in dark color of dust and debris (rather than the white color of a water vapor condensation cloud). A significant amount of reactor building concrete dust and debris was generated from the explosion.

- ***The damage to the 1F3 building was more extensive compared to damage incurred at 1F1 and 1F4.*** The extent that the damage caused by the energetic explosion was a consequence of drywell head seal failure leading to a PCV blowdown at high pressure and temperature is a question to be answered. Large objects were thrown high into the sky. Big pieces of concrete and equipment were also thrown into the spent fuel pool. Further evaluations are needed to investigate if this type of explosion can cause containment structural failure at other locations.
- ***The shared vent stack between 1F3 and 1F4 allowed hydrogen that was vented from 1F3 to enter the 1F4 reactor building.*** Radionuclide surveys and examination information confirm that the shared vent stack was the reason for the explosion in the 1F4 reactor building. The design of such vent stacks should take into consideration the safety implication of this experience.

In summary, available information has already provided many important insights related to combustible gas generation. However, questions remain in this area. In particular, information is needed to evaluate the contribution of gases generated from MCCI to the observed explosions. This question is, in turn, related to the extent of MCCI following RPV failure as well as the point at which the core debris is quenched and rendered coolable. As D&D activities progress, it is anticipated that planned examinations by TEPCO will address these questions.

6.4 Recommendations

The explosions at Daiichi caused significant damage to the reactor building structures. Assessments of the Fukushima Daiichi event scenarios at each unit highlight the correlation between core damage modeling and the potential for flammable conditions to develop in reactor buildings.

Results from recent studies comparing MAAP5 and MELCOR calculations [29] have identified how limited knowledge regarding in-core damage progression can lead to significant differences in code predictions for hydrogen production. Differences between code predictions stem from a lack of experimental data that would clarify appropriate modeling assumptions regarding in-core melt progression behavior. As a result, the two codes predict different amounts of in-core hydrogen generation, with MAAP5 typically predicting lesser amounts of in-core hydrogen generation relative to MELCOR [29]. Evaluations with MAAP5 tend to find that this has important consequences for the development of flammable conditions in the 1F1 and 1F3 reactor buildings. Figures 43 and 44 compare results from a MAAP5 uncertainty evaluation of the 1F3 accident [32]. These figures show the predicted hydrogen concentrations on the refueling and fourth floors of the 1F3 reactor building, respectively, at the time of the actual 1F3 explosion (68.7 hours after the earthquake), versus the timing of RPV lower head breach.

These results illustrate that for simulations predicting RPV lower head breach occurs after ~ 65 hours, there is limited potential for flammable conditions to develop on either the 1F3 refueling or 4th floors. That is, MAAP5 simulations of scenarios with in-vessel retention, at least up to the point of the actual 1F3 explosion, do not support the necessary conditions for combustion. This is due to relatively low amounts of in-core hydrogen generation being predicted. By contrast, MELCOR simulations can evolve enough hydrogen to support conditions for flammable gas combustion in the reactor building.[72]

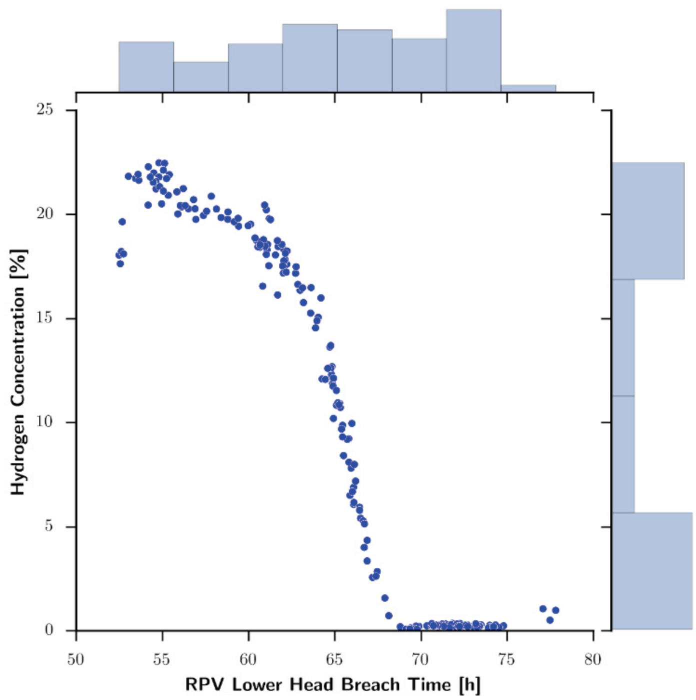


Figure 43. MAAP 1F3 modeling uncertainty evaluation: refueling floor hydrogen build-up at time of 1F3 reactor building explosion. (Courtesy of EPRI [32])

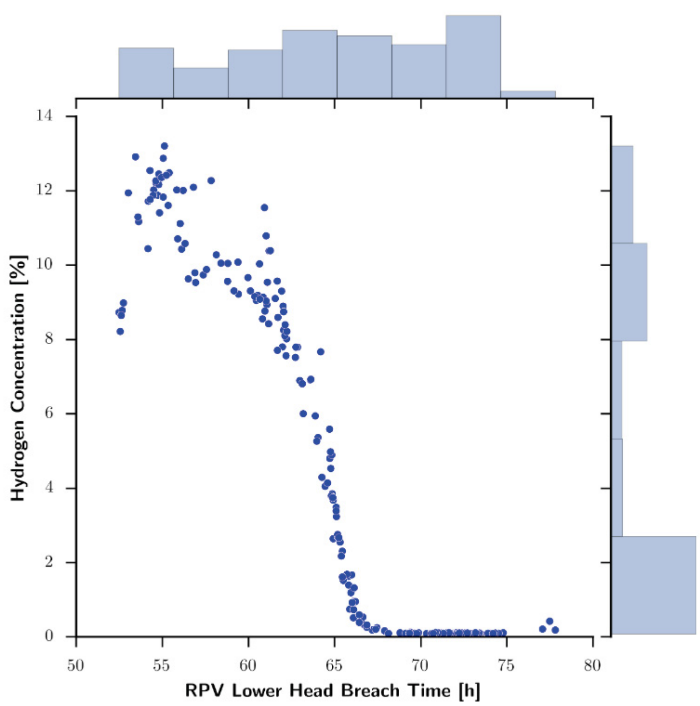


Figure 44. MAAP 1F3 modeling uncertainty evaluation: fourth floor hydrogen build-up at time of 1F3 reactor building explosion (Courtesy of EPRI [32])

In summary, the expert panel formulated several recommendations for this area.

Area 4 Recommendation 1:

To date, the thermal hydraulic and core nodalizations of the reactor pressure vessel in both MAAP and MELCOR have been shown to well represent the physics within the core. However, there are still uncertainties in hydrogen generation driven by modeling of core relocation behavior and debris bed geometry in partially mitigated and unmitigated severe accidents. It is currently unclear if the majority of hydrogen generation in the Fukushima units occurred in-vessel or ex-vessel, with both MAAP and MELCOR indicating different answers. The differences in the two codes in modeling core debris behavior inside the RPV can have significant downstream effects on eventual MCCI and ex-vessel noncondensible gas generation. To address these important gaps in severe accident progression, the expert panel recommends that evaluations of combustible gas generation differences resulting from in-core relocation and debris bed morphology be continued with the goal of reducing uncertainties.

There are also uncertainties with respect to hydrogen migration paths from the PCV to various floors of the RB and ignition sources or the mechanism required to cause ignition within the RB during such an extended SBO. Two recommendations have been developed to gain insights related to this uncertainty.

Area 4 Recommendation 2:

Better knowledge on hydrogen migration paths through degraded seals and penetrations from the PCV to the RB is desirable. The expert panel should continue to review available information for insights.

Area 4 Recommendation 3:

There is little knowledge about ignition sources or the mechanisms that lead to ignition during such an extended SBO for all the explosions at Daiichi; the expert panel should continue to review available information for insights.

6.5 Suggestions for Additional Information

As discussed above, available information has already provided many important insights related to accident management. However, as indicated in Tables 19 and 20, there are still information needs that have not yet been met in this area. In particular, information is needed to evaluate the contribution of gases generated from MCCI to the observed explosions. This question is, in turn, related to the extent of MCCI following RPV failure as well as the point at which the core debris is quenched and rendered coolable (see Section 5). Based on insights obtained from evaluations of current information, one suggestion is offered at this time:

Area 4 Suggestion:

Continue to obtain visual information, radiation surveys, and isotopic evaluations to ascertain the source (e.g., in-vessel, ex-vessel, or both) of combustible gas generation within the affected units.

As D&D activities progress, it is anticipated that planned examinations by TEPCO will provide these insights.

7. SUMMARY AND INSIGHT IMPLEMENTATION

Information obtained from Daiichi is required to inform TEPCO D&D activities. This section summarizes the examination information evaluated by the US expert panel and the recommendations formulated from these evaluations. In addition, this section identifies actions that are already underway to use insights gained from forensics examinations to reduce severe accident modeling uncertainties and confirm severe accident management guidance. These activities to implement insights are beneficial to the US because they provide additional assurance that current severe accident guidance is appropriate (or identify the need for future revisions to such guidance). Activities to reduce uncertainties in modeling severe accident phenomena are also beneficial to the US and Japan for enhancing reactor safety. In addition, reduced uncertainties in severe accident evaluations are beneficial to Japan because improved realism in reactor safety evaluations support D&D activities by improving the capability to characterize reactor component performance during the accident and to estimate post-accident fuel location and fission product deposition and form. This improves the technical basis for characterizing potential hazards to workers involved with cleanup activities.

7.1 Evaluations and Recommendations

As discussed within this document, significant examination information is already available for evaluation. For the US forensics effort, the expert panel agreed to focus evaluations in the four areas identified in Table 21. This table also lists the types of information available for evaluation in each area. As indicted in Table 21, available information is primarily visual, data from plant instrumentation, radiation surveys, and isotopic sampling.

Table 21. Evaluation areas and types of evaluated examination information

Area	Types of Examination Information Evaluated
Area 1 - Component Degradation	• Visual information (photos and videos gathered in post-accident examinations). • Sampling. • Dose rate measurements. • Water level and temperature measurements. • TEPCO reports documenting unsolved and unresolved issues.
Area 2 - Dose Surveys and Isotopic Surveys and Samples	• Radiation doses accumulated by plant personnel during the accident and during post-accident examinations. • Dose rate measurements obtained during and after the accident (including perimeter and adjacent area surveys). • CAM readings in the drywell and wetwell. • Sampling of contaminated water in various reactor buildings and discharge effluents.
Area 3 - Debris Endstate	• Visual information (photos and videos gathered using robotic examinations and stand-off methods such as muon tomography). • Data from plant instrumentation (temperature information obtained during and immediately after the accident, gas concentration data from the gas treatment system, and neutron and gamma detector data from subcriticality monitoring systems).
Area 4 - Combustible Gas Effects	• Visual information (photos and videos taken during and after the explosions). • Radiation survey and seismic acceleration data. • TEPCO reports evaluating damage associated with explosions. • TEPCO unresolved issues reports. • Data from plant instrumentation.

Forensics evaluations by the expert panel led to the identification of several recommendations for future US RST pathway activities. Table 22 lists these recommendations for each area. As indicated in this table, recommendations were primarily related to additional calculations to resolve modeling uncertainties and the need for continued evaluations as additional information becomes available.

Table 22. Recommendations for future US RST pathway activities

Area	Recommendations
Area 1 - Component Degradation	Sensitivity studies should be performed on containment failure location and size with respect to radiological releases (timing, amount) and impact on accident progression. These sensitivity studies should be done with both MAAP and MELCOR in order to cover a range of predicted containment and primary system conditions. Sensitivities for each unit would provide insight into which failure likely caused depressurization, the conditions under which such a failure occurred, and the effect of multiple failures. Some previous sensitivity analyses have been performed for failure of the primary system (SRV versus MSL, etc.) and the containment.
	The expert panel should continue to review available information and update Table 9.
	A concise comparison should be developed for the predicted conditions by both MAAP and MELCOR at the MSIV (temperature, pressure) for 1F2 and 1F3. The expert panel should continue to review any additional inspection information of the MSIV room or MSLs.
	The expert panel is interested in 'before' pictures for specific locations from TEPCO. As more information becomes available, the panel will identify specific places.
Area 2 - Dose Surveys and Isotopic Surveys and Samples	Similar to Area 1 Recommendation 1, experts agreed that information on this topic suggests that sensitivity studies should be performed on containment failure location and size with respect to radiological releases (timing, amount) and impact on accident progression. These sensitivity studies should be done with both MAAP and MELCOR in order to cover a range of predicted containment and primary system conditions. To compare results from simulations of core damage progression and radiological release to the environment, additional analyses with an environmental radiological transport code, such as MACCS, would be useful. Sensitivities for each unit would provide insight into which failure likely caused depressurization, the conditions under which such a failure occurred, and the effect of multiple failures. Some previous sensitivity analyses have been performed for failure of the primary system (SRV versus MSL, etc.) and the containment. As discussed within Section 4, reactor building radiological hotspots provide a means to assess inputs provided to severe accident computer codes, but do not typically facilitate assessment of the actual computer code models.
	Similar to Area 1 Recommendation 3, concisely compare the predicted conditions by both MAAP and MELCOR at the MSIV (temperature, pressure) for 1F2 and 1F3.
	Similar to Area 1 Recommendation 4, the expert panel continues to be interested in examination information of MSIV room components.
	The expert panel recommends that the US Forensics Effort continue to evaluate information obtained from examinations of RPVs within each unit impairment location. In particular, addition visual information would be useful in the Area 2 Recommendation 1 sensitivity studies.
Area 3 - Debris Endstate	As alluded to in Section 5.2, refine the MAAP and MELCOR RPV nodalization schemes for the RPVs of Units 1-3 with the aim of predicting the measured temperatures shown in Figures 20 through 22. The post-accident debris locations predicted inside the RPV, coupled with changes in water addition rate and location, may provide a means for assessing the accuracy of the debris end-state predictions. This comparison may also provide insights into appropriate modeling of in-core melt progression that has been identified as a key uncertainty in the MAAP-MELCOR crosswalk exercise [29].
	Repeat the MELTSPREAD-CORQUENCH analysis that was originally done for 1F1 [30] for 1F2. Various system-level code analyses have shown the potential for vessel failure at this unit also. However, if the vessel did fail, it likely occurred much later in the accident sequence due to the continued operation of RCIC for ~72 hours in an unregulated mode. This study may be useful in showing that it is unlikely that the melt contacted the liner in this late pour scenario, or if it did, that the shell likely remained intact due to reduced thermal loading. As discussed in [10], no evidence of liner failure has been found for 1F2, and this would provide a means for rationalizing that observation relative to the finding that the liner in 1F1 has been damaged.

Area	Recommendations
Area 4 - Combustible Gas Effects	To date, the thermal hydraulic and core nodalizations of the reactor pressure vessel in both MAAP and MELCOR have been shown to well represent the physics within the core. However, there are still uncertainties in hydrogen generation driven by modeling of core relocation behavior and debris bed geometry in partially mitigated and unmitigated severe accidents. It is currently unclear if the majority of hydrogen generation in the Fukushima units occurred in-vessel or ex-vessel, with both MAAP and MELCOR indicating different answers. The differences in the two codes in modeling core debris behavior inside the RPV can have significant downstream effects on eventual MCCI and ex-vessel noncondensible gas generation. To address these important gaps in severe accident progression, the expert panel recommends that evaluations of combustible gas generation differences resulting from in-core debris bed morphology be continued with the goal of reducing uncertainties.
	Better knowledge on hydrogen migration paths from the PCV to the RB is desirable. The expert panel should continue to review available information for insights.
	There is little knowledge about ignition sources for all the explosions at Daiichi; the expert panel should continue to review available information for insights.

The expert panel also developed suggestions for additional examination information. These suggestions are listed in Table 23. As indicated in this table, suggestions were primarily to continue with planned D&D examinations. In several areas, the expert panel requested that planned D&D examinations place additional focus on addressing particular questions of interest. For Area 1, the panel explicitly requested that TEPCO experts continue to review summary information related to component degradation developed by US experts. In Area 3, one new information need was identified. This item has been added to examination needs as RB-14 in Appendix C. Because such chemical analyses could assist in identifying the location of debris in each unit, TEPCO is evaluating the potential to obtain this information.

Table 23. Suggestions for additional examinations

Area	Types of Examination Information Evaluated
Area 1 - Component Degradation	To facilitate updates to Table 9, the expert panel has requested that TEPCO continue to review information in this table. In addition, the expert panel will continue to review additional information, such as penetration, component, and system examination results, from TEPCO and update this table.
	As discussed in Section 4, additional surveys in containment to understand the integrity of the RPV lower head, pedestal, and containment liner are of particular interest. These information needs are identified in Appendix C.
Area 2 - Dose Surveys and Isotopic Surveys and Samples	Continue planned additional isotopic evaluations. As discussed in Section 4, evaluations of concrete samples extracted from a common location for all three units would be of interest. In particular, it would be helpful to have additional data against which to assess off-site ^{134}Cs to ^{137}Cs isotopic ratios.
Area 3 - Debris Endstate	Perform chemical analysis of high radiation deposits or particles found inside the reactor building (1F1, 1F2, and 1F3); e.g., the white deposits from the HPCI room using FE-SEM, XRD, etc.
Area 4 - Combustible Gas Effects	Continue to obtain visual information, radiation surveys, and isotopic evaluations to ascertain the source (e.g., in-vessel, ex-vessel, or both) of combustible gas generation within the affected units.

7.2 Implementation Activities for Forensics Insights

Results from the Forensics Effort are already being used to address many items listed in Objective 2 (Section 1.1); namely, to enhance guidance for PWR and BWR severe accident mitigation and to reduce uncertainties in severe accident code models. Selected implementation activities are discussed below.

7.2.1 Industry Accident Management Guidance

Insights gained from the Fukushima accident have been used, and are continuing to be used, to enhance industry Severe Accident Guidance (SAG). [2,17, 42,58,138,139] This is accomplished through the BWROG and PWROG, who have maintained generic SAG for their member plants since 1998 and periodically provide enhancements as new information becomes available. Plants then implement the generic guidance according to design features of their particular plant. Specific examples where industry guidance is benefitting from the US Forensics include:

- ***Primary Containment Venting*** – As discussed in Sections 3, 4, and 6, the three operating units exhibited different patterns of PCV leakage of fission products and hydrogen. The variability introduced by unit to unit differences at Fukushima points to uncertainties in actual leakage locations and confirms the importance of maintaining containment conditions below design basis temperature and pressure limits (and that a high priority is placed on reducing containment conditions when they exceed design basis values) is an appropriate strategy. The revised BWROG and PWROG SAG places a high priority on venting the primary containment when the combination of pressure and temperature reaches a prescribed limit. For BWRs, these conditions can be very close to the containment design basis pressure and temperature.
- ***Water Addition Pathways*** – As discussed in Section 5, currently available information from 1F1, 1F2, and 1F3 indicates that there are differences in the core debris end-state location. It is believed that these differences are due to differences in the accident progression at each unit, particularly decay heat levels and the timing and rate of periodic water addition prior to stabilizing the core debris. The BWROG and PWROG SAG has always placed a higher priority on injection of water to the reactor vessel compared to the primary containment. If the reactor vessel is failed, the injected water will flow through the reactor vessel breach to the core debris in the primary containment. This ensures that core debris is cooled with injected water (and possibly submerged in water) regardless of its location. Because a large amount of water is required to cool core debris in all possible locations (in the primary containment and in the reactor vessel) for both BWRs and some PWRs, the emphasis on water addition in updated guidance is appropriate. The BWROG also places a high priority on injection of water to the reactor vessel using core spray to assist in more complete cooling of core debris inside the reactor vessel.
- ***Hydrogen Combustion Outside Primary Containment*** – As discussed in Section 6, there were differences in hydrogen accumulation and combustion phenomena for each of the four units. BWROG and PWROG guidance was enhanced immediately after the Fukushima accident to include venting the reactor and auxiliary buildings. The variability in the source of the hydrogen and its accumulation in the reactor building across the damaged units points to uncertainties and confirms recent SAG enhancements by both the BWROG and PWROG to include strategies for venting buildings adjacent to the primary containment as an appropriate action when primary containment pressure exceeds design basis values. The BWROG and PWROG SAG also includes criteria for ventilating the reactor and auxiliary buildings if normal ventilation is not available. For BWRs, doors at higher elevations within the reactor building are opened on entry to severe accident guidance. Once there is evidence of hydrogen, doors are also opened at lower elevations to promote natural circulation. For PWRs, doors are opened when containment pressure exceeds design basis values.
- ***Instrumentation*** – As discussed in Sections 3 and 5, there were several instrumentation anomalies that may have contributed to the severity of the accident, or at the very least slowed the decision-

making during the accident recovery. As a result of these Fukushima forensics insights, both the BWROG and PWROG were recently enhanced to include Technical Support Guidance for Instrumentation. The basis for the enhanced guidance is understanding the expected response trends of instrumentation for each and every potential mitigation action and comparing instrumentation response from several instruments where possible.

- ***Severe Accident Models*** – As discussed in Sections 3, 5, and 6 and summarized in Section 7.2.2, there are certain aspects of the accidents at the Fukushima Daiichi units that are not well modeled by systems analysis codes, such as MAAP and MELCOR. Specific examples that can be confirmed from Fukushima forensic information include the amount hydrogen generation from zirconium water reactions in the late phases of core degradation, environmental conditions near primary containment penetrations, and core debris spreading following reactor vessel failure. These examples illustrate that significant uncertainties still exist in the code predictions that may be due to the limited database for model development. The BWROG and PWROG SAG is, for the most part, based on first principles of severe accident phenomena behavior as described in the EPRI Technical Basis Report (TBR). [140] As a result, the BWROG and PWROG SAG is largely independent on severe accident predictions by either MAAP or MELCOR. However, the BWROG and PWROG SAG, as well as Volume 2 of the TBR, should be reviewed further to determine the reliance of strategies on severe accident code predictions, particularly for hydrogen generation, temperature conditions at penetrations, and ex-vessel core debris spreading. The TBR is important because some plant owners/operators supplement the BWROG and PWROG material with information from the TBR. As new forensic information becomes available, adjustments should be made to the BWROG and the PWROG SAG, as well as the TBR, as appropriate.
- ***Operation of Turbine Driven Pumps*** – As discussed in Section 3, information from 1F2 and 1F3 provide some valuable insights related to operation of turbine driven pumps (RCIC and HPCI) under beyond design basis conditions. Operation of RCIC was critical in delaying core damage for days (almost three days for 1F2) 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. For 1F3, RCIC stopped when a protection signal (dc power was still available) tripped the pump. HPCI auto-started on 'lo lo' reactor vessel water level and ran until the reactor vessel pressure dropped below the operating range of the high pressure turbine; HPCI was operated in 'Test Mode' most of the time with only periodic flow to the reactor vessel which is thought to have resulted in low steam flow to the turbine. BWROG and PWROG accident management strategies place a high priority on the use of turbine driven pumps (RCIC and HPCI for BWRs and AFW for PWRs) to maintain core cooling. With the implementation of FLEX, these pumps are relied upon until the FLEX equipment can be operated as a backup (RCIC, HPCI and AFW would still be used after FLEX equipment is in-place as long as they are operable). As a result of forensic information, the BWROG has provided enhanced guidance on the operation of turbine driven pumps under beyond design basis conditions. The PWROG is considering enhancements to guidance on operating turbine driven pumps under beyond design basis conditions for training and guidance. DOE is considering further testing to gain additional insights related to operation of turbine driven pumps.

7.2.2 Code Modeling Enhancements

Results from the US Forensics Effort are also being used (and will continue to be used) to reduce uncertainties in severe accident code models. Selected examples are discussed in this section.

- ***Primary Containment Integrity Challenges*** – As discussed in Sections 3, 4, and 6, the three operating units exhibited different patterns of PCV leakage of fission products and hydrogen. Many of these leakage points are not routinely modeled by systems level severe accident codes (MELCOR, MAAP, etc.). Both MAAP and MELCOR simulations predict drywell head failure for the three units. It is evident that other penetrations and piping failures should be considered in systems analysis codes.

- ***MELCOR and MAAP Nodalization Studies*** - As discussed in Sections 3, 4, and 5, MAAP and MELCOR RPV nodalization studies to improve temperature predictions could also provide insights related to post-accident debris end-state predictions, as well as provide insights related to modeling of in-core melt progression, , particularly as it pertains to maintaining PCV liner integrity.
- ***1F2 MELTSPREAD-CORQUENCH Analysis*** – As discussed in Section 5, ex-vessel debris spreading analyses have only been performed for 1F1. System-level code analyses indicated that there is the potential for vessel failure to also have occurred at 1F2. An evaluation of 1F2 may prove useful for rationalizing differences in future observations obtained from 1F1 and 1F2.
- ***Combustible Gas Production, Transport, and Mitigation*** – As discussed in Section 6, MAAP core melt progression models do not predict as much in-core hydrogen generation as MELCOR. The ex-vessel combustible gas generation predictions are similar due to modeling of MCCI being similar in MAAP and MELCOR. However, MAAP requires more ex-vessel hydrogen generation from MCCI than MELCOR to predict sufficient accumulation of combustible gas that leads to the large explosions that occurred in 1F1 and 1F3. In addition, as noted above, both MAAP and MELCOR do not predict that seal degradation would occur and allow combustible gas to accumulate within the reactor building. Thus, gas stratification/combustion and seal leakage models in these codes should be reviewed to determine if modeling upgrades are warranted to reduce modeling uncertainties.

7.3 Summary

TEPCO examinations at Daiichi to inform D&D activities improves their ability to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. The US Forensics Effort is identifying examination needs from the affected units at Daiichi and evaluating information obtained by TEPCO to address these needs. Examples presented in this report illustrate the intrinsic value of this information. Significant safety insights are already being obtained in the areas of component performance, fission product release and transport, debris end-state location, and combustible gas effects. In addition to reducing uncertainties related to severe accident modeling progression, these safety insights are actively being used by industry to update and improve PWR and BWR guidance for severe accident prevention, mitigation, and emergency planning.

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Appendix A

FY2016 Meeting Agendas and Attendee Lists

November 9-10, 2015 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 9-10, 2015

Nuclear Energy Institute, 1201 F St., NW, Suite 1100
Washington, DC

Monday, November 9th

8:30 AM	Welcome to NEI, Administrative Matters, and Safety Minute	S. Kraft, NEI
8:40 AM	Welcome and Overview	M. Corradini, UW
8:50 AM	DOE NE / DOE EM Coordination	G. Deleon/H. Nigam, M D. Peko. NE
9:00 AM	NRC Comments	R. Lee, NRC
9:10 AM	Background and Proposed Agenda Resolution of Peer Review Comments	J. Rempe, Rempe and Associates, LLC
9:30 AM	TEPCO Update	T. Hara, TEPCO
10:30 AM	*Break*	All
10:45 AM	Topic 1 - Component Inspection Overview of Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	K. Robb, ORNL/ J. Gabor, ERIN
12:30	*Lunch*	All
1:30	Topic 1 - Continued	
2:00 PM	Topic 2 - Dose Measurements and Smears for Isotopic Concentration Evaluations (based on code analysis evaluations, etc.) Overview of Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	R. Gauntt, SNL / D. Luxat, ERIN

Monday, November 9th (Continued)

4:00 PM	*Break*	All
4:15 PM	Topic 2 - Continued	
4:30 PM	Topic 3 - Core Debris Location Evaluations Overview of Material Available (robot inspections, Muon Tomography, etc.) Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	M. Farmer, ANL/ R. Gauntt, SNL
5:30 PM	*Adjourn*	

Tuesday, November 10th

8:30 AM	Continuation of Topic 3	
10:00 AM	*Break*	All
10:30 AM	November 9th Recap & Next Steps	Leads for Each Topic
11:30 AM	Next Steps ▪ Proposed report outline ▪ Action items and schedule ▪ Proposed discussion topics and date for next meeting	Joy Rempe
Noon	*Adjourn*	All

November 9-10, 2015 Meeting Attendees

Number	Name	Organization
1	Donald Algama	US NRC
2	Nathan Andrews	SNL
3	Mike Corradini	University of Wisconsin
4	Edgardo Deleon	US DOE
5	Phil Ellison	GE Hitachi, BWROG
5	Hossein Esmaili	US NRC
6	Mitch Farmer	ANL
7	Matthew Francis	ORNL
8	Edward Fuller	US NRC
9	Jeff Gabor	Jensen Hughes
10	Randy Gauntt	SNL
11	Takashi Hara	TEPCO
12	Retsu Kojo	NRA (on assignment to US NRC)
13	Stephen Kraft	NEI
14	Richard Lee	US NRC
15	Roy Linthicum	Exelon, PWROG
16	Wison Luangdilok	Fauske and Associates, LLC
17	Robert Lutz	Lutz Nuclear Consulting
18	David Luxat	Jensen Hughes
19	Donald Marksberry	US NRC
20	Hitesh Nigam	US DOE
21	Damian Peko	US DOE
22	Marty Plys	Fauske and Associates, LLC
23	Joy Rempe	Rempe and Associates, LLC
24	Kevin Robb	ORNL
25	Michael Salay	US NRC
26	Kenji Tateiwa	TEPCO (by telephone)
27	Bill Williamson	TVA, BWROG

April 28-29, 2016 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
April 28-29, 2016

Argonne National Laboratory (ANL) Offices
955 L'enfant Plaza, North, SW, Suite 6000
Washington, DC 20024-2168

<u>Day 1, April 28, 2016</u>

8:30 AM	Welcome	M. Farmer, ANL
8:40 AM	Welcome and Overview	M. Corradini, UW
8:50 AM	DOE NE / DOE EM Coordination Update on CNWG and Other International Activities	A. Han/H. Nigam, EM D. Peko. NE
9:10 AM	NRC Comments / Update on International Activities	R. Lee, NRC
9:30 AM	TEPCO Update	S. Mizokami/D. Yamada, TEPCO
10:45 AM	*Break*	All
11:00 AM	Meeting Overview / Discussion of Sections 1, 2, and 7	J. Rempe, Rempe and Associates, LLC
11:30 AM	Insights from DOE Expert Panel Forensics for LWR Accident Management	R. Lutz, Lutz Consulting/ B. Williamson, TVA
12:00 AM	*Working Lunch; Demonstration of Website and Discussion of Website Needs*	P. Humrickhouse, INL
1:00 PM	Area 1 - Component /System Performance Updates based on New Material / Unresolved Issues Reports Resolution of Comments on Section 3 of Draft Report Additional information requests (if needed)	K. Robb, ORNL/ J. Gabor, Jensen Hughes
3:00 PM	*Break*	All

Day 1, April 28, 2016 (Continued)

3:15 PM	Area 2 – Dose Measures/Surveys Updates based on New Material / Unresolved Issues Reports Resolution of Comments on Section 4 of Draft Report Additional information requests (if needed)	R. Gauntt, SNL / D. Luxat, Jensen Hughes
5:15 PM	*Adjourn*	

Day 2, April 29, 2016

8:00 AM	Area 3 - Core Debris Location Evaluations Updates based on New Material / Unresolved Issues Reports Resolution of Comments on Section 5 of Draft Report Additional information requests (if needed)	M. Farmer, ANL/ R. Gauntt, SNL/ M. Plys, FAI
9:45 AM	*Break*	All
10:00 AM	Area 4 – Combustible Gas Effects Updates based on New Material / Unresolved Issues Report Resolution of Comments on Section 6 of Draft Report Additional information requests (if needed)	Wison Luangdilok, FAI/ N. Andrews, SNL/ D. Luxat, Jensen Hughes
11:45 AM	Next Steps ▪ Path for completing FY16 Report ▪ Action items and schedule ▪ Preparation for FY17 activities	J. Rempe, Rempe & Associates, LLC
12:30 PM	*Adjourn*	All

April 28-29, 2016 Meeting Attendees

Number	Name	Organization
1	Donald Algama	US NRC
2	Nathan Andrews	SNL
3	Sudhamay Basu	US NRC
4	Willis Bixby	WWBX Consulting, LLC
5	Randy Bunt	Southern Nuclear, BWROG
6	Mike Corradini	University of Wisconsin
7	Hossein Esmaili	US NRC
8	Phil Ellison	GE Hitachi, BWROG
9	Hossein Esmaili	US NRC
10	Mitch Farmer	ANL
11	Terri V. Farthing	GE Hitachi
12	Edward Fuller	US NRC
13	Jeff Gabor	Jensen Hughes
14	Randy Gauntt	SNL
15	Ana Han	US DOE-EM
16	Takashi Hara	TEPCO
17	Chris Henry	Fauske and Associates, LLC
18	Paul Humrickhouse	INL
19	Retsu Kojo	NRA (on assignment to US NRC)
20	Richard Lee	US NRC
21	Roy Linthicum	Exelon, PWROG
22	Wison Luangdilok	Fauske and Associates, LLC
23	Robert Lutz	Lutz Nuclear Consulting
24	David Luxat	Erin Engineering
25	James Maddox	INPO
26	Donald Marksberry	US NRC
27	Shinya Mizokami	TEPCO
28	Chuck Negin	CANegin & Associates
29	Hitesh Nigam	US DOE
30	Chan Paik	Fauske and Associates, LLC
31	Damian Peko	US DOE
32	Marty Plys	Fauske and Associates, LLC
33	Joy Rempe	Rempe and Associates, LLC
34	Kevin Robb	ORNL
35	Michael Salay	US NRC
36	Robert Sanders	AREVA
37	Kenji Tateiwa	TEPCO (by telephone)
38	Richard Wachowiak	EPRI
39	Bill Williamson	TVA, BWROG
40	Daichi Yamada	TEPCO (EPRI Visiting Researcher)

Appendix B

Website to Support Forensics Evaluations

Website to Support Forensics Evaluations

Background

In the months following the accident, the Fukushima Daiichi Accident Study Information Portal (http://fukushima.inl.gov) was developed at INL to collect and organize information and data. The website was primarily organized around accident *timelines* at each of the reactors. The timelines were made up of a series of *events*, and these events in turn have documents of various types associated with them. This interface, however, did not prove to be a convenient way to locate, search for, or view these documents. The website provides some additional capabilities such as the ability to plot data (temperatures, pressures, and water levels) measured during the progression of the accidents, as well as map subsequent radiation measurements taken in the vicinity.

During the November 2015 meeting, US experts agreed that a reconfigured version of the INL website should be developed to provide a searchable location for information to be archived. To ensure that the reconfigured website meets US needs for the expanded program, it was agreed that INL would develop and populate an initial framework for a website to be presented at the next expert panel meeting for review by program participants. This appendix describes this initial framework as it has now been implemented here, along with further additions and modifications planned for the remainder of FY16 and beyond.

It should be noted that the need to have access to archived information from inspections, analyses, and other relevant sources has also been recognized in on-going international efforts. Preliminary versions of the recommendations being prepared in the SAREF Research Opportunities effort recommend that a website be established to meet the needs of an international forensics investigation. Hence, the US Forensics Effort is developing this website so that it may also address future needs of an international effort.

Website Redesign

In order to provide a more logical interface to display, filter, and search documents, the website has been modified so as to present the user with several options on the top menu bar. The old timeline views (and associated event/artifact views) are preserved under “Timelines”; similarly, plottable temperature/pressure/water level data and radiation maps are available under “Data”. An additional option, “Documents,” has been added; the layout of these options is shown in Figure B-1.

Selecting the “Documents” tab from the top menu bar takes the user to the new interface for filtering and searching documents. At the left side of the page, a number of categories are listed in which one or more filters can be applied; at the center of the page, the list of documents matching the selected filter criteria appears. The default view of this page, in which no filters are applied, is shown in Fig. B-2; in this case all documents presently in the database are listed (10 per page).

There are presently four categories on which to filter the document list:

- Unit (1-6)/Location (on/offsite, other)
- Source (i.e., issuing agency)
- Media type (e.g., documents, photos, videos, etc.)
- Classification (public, protected, or sensitive)

Figure B-1. Homepage with reorganized menu bar, including new "Document" tab at top. (Courtesy of INL)

Figure B-2. Default view of the "Documents" page, with no filters applied. All 71 items in the existing database are shown (10 per page). (Courtesy of INL)

The location may be any of the six reactors, on-site, off-site, or other/NA. Documents in the database may have multiple location entries, i.e., a report on the accident progression at Units 1-3 will be matched when selecting Unit 1, Unit 2, Unit 3, or any of these combined in the location field.

The list of sources is based on the items in the existing database and includes:

- Tokyo Electric Power Company Holdings (TEPCO; most items in the database)
- U.S. Department of Energy (DOE)
- Electric Power Research Institute (EPRI)

- International Atomic Energy Agency (IAEA)
- Institute of Nuclear Power Operators (INPO)
- Japan Nuclear Technology Institute (JANTI)
- Japan Nuclear Energy Safety Organization (JNES)
- U.S. Nuclear Regulatory Commission (NRC)
- Organization for Economic Cooperation and Development (OECD)

Several items are presently classified as "other"; additional sources will continue to be added as required. As the document database grows, some may be eliminated if there are few items attributed to them.

Current media types include:

- Reports
- Photograph/Graphical Images
- Presentations
- Spreadsheets
- URLs
- Videos
- Other

During the April 2016 Meeting, experts recommended that "Citation" be added as a media type. This is to include items in search results that may be useful to users, in cases where the document itself cannot be archived in the database, e.g. if it is proprietary. A link will be provided to such documents, as appropriate.

"Classification" is also provided as a filter category; the database provides "protected" and "sensitive" document categories that would limit access to designated documents to authorized individuals. Presently, all materials are publicly available.

Some additional filter categories are being implemented. These include:

- Date data/information obtained
- Date document published or released
- Date information uploaded to the website
- Component (MSIV, SRV, etc.)

The date filters will allow users to specify date ranges (e.g. with drop-down calendars) for when information was obtained, when the information was published, and when it was uploaded to the website.

The component filter will facilitate searches for information about specific components. Because many (perhaps most) documents will mention many different components in the course of their text, and tagging such documents would prove excessively laborious, it was decided that this filter will apply only to the titles of documents.

In addition to the filters described above, a search utility is to be added. This is present (though not yet functional) on the home page, see Figure B-1; another is present on the documents page (see Figure B-2) that will allow the user to search for text within the filtered results. This search function will apply not just to titles and document descriptions, but also to the entire text of documents (provided, in the case of PDF files, it is embedded text and not scanned images).

Per suggestions at the April 2016 Expert Panel meeting, a number of additional pages have been added though not yet populated. These include a "links" page for links to participating organizations, and a page to list future meeting dates and archive past meeting agendas, presentations, and reports.

Examples illustrating the functionality of selected filters are shown in Figures B-3 to B-5. Figure B-3 shows all items from TEPCO. Note that these take up five pages, i.e., most of the material in the existing database is from TEPCO. If one additionally filters for only spreadsheets under "Media Type," the results are limited to TEPCO spreadsheets only, the entirety of which are listed in one page of results, see Figure B-4. These are the data that are additionally available for plotting on the "Data" tab. Multiple filters can be applied within the same category; filtering for media type "Presentation" in addition to "Spreadsheet" (while maintaining the "TEPCO" source filter) lists all spreadsheets and presentations from TEPCO, see Figure B-5.

Future Work

The following tasks are planned for the remainder of FY16:

1. Implementing functionality described above, including:

    a. Addition of "publication date", "date obtained", and "component type" attributes to the existing database (by July 31, 2016)

    b. Addition of "citation" as a media type (by July 31, 2016)

    c. Addition of dropdown calendars to filter documents by publication date, date obtained, and upload date (by July 31, 2016)

    d. Addition of fully functional search bar to search for any desired text within documents, including filtered lists (by September 30, 2016)

2. Updating attributes of existing documents to match the new database structure. This will include populating the new fields for each document in the database as necessary (component name, publication date, and date obtained) and modifying existing fields where appropriate (e.g., making sure a picture of all four reactors is matched by any or all of location filters Unit 1/2/3/4). Completion date: July 31, 2016.

3. Adding new materials. A considerable amount of material has already been collected and is ready for addition once the database structure is finalized. Completion date: August 31, 2016.

4. Opening of the website to other users, including non-INL users, for initial evaluation. Completion date: Sept. 30, 2016. [It is anticipated that the website will be updated in future years based on user feedback].

Organization

- Unit
- Topic (Dose, Debris Location, etc.)
- Date (Information Obtained, Information Posted)
- Component (MSIV, IC, RCIC, SRV, etc.)
- Type of Information (Policy/Planning, Data, Analysis, Testing, Code-to-Data Comparisons)

Format Needs:

- Data/code results easily exported in an easy-to-edit/import format (e.g., excel file, etc.)
- Figures easily exported (remove protections)
- Auto-generation of citations for information for easy cut and paste.

Appendix C

Information Needs

Table C-1. Information Needs from the Reactor Building

Item	What/How Obtained	Why	Expected Benefit /Use	When	Status
RB-1	Photos/videos[hh] of condition of RCIC valve and pump before drain down and after disassembly (1F2 and 1F3)	• Determine turbine condition. • Gain insights about status of valve and pump at time of failure [PWRs have almost identical pumps for AFW].	Impacts BWR AM strategies (cause of RCIC room flooding). Use to support RCIC testing project (for confirmation of testing results). Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Currently flooded (requires underwater investigations unless drained). Inspections could be completed more easily at Daini.	Not currently considered by TEPCO, but is desired; If torus not drained, requires underwater technology available. If photos or data are obtained as part of D&D activities, please provide (but the US recognizes that additional information may not be obtained).
RB-2	Photos/videos of HPCI System after disassembly (1F1, 1F2, and 1F3)	• Gain insights about degradation.	Impacts AM strategies (equipment utilization).	Currently flooded (requires other alternatives for underwater investigations unless drained).	Not currently considered by TEPCO; If torus not drained, requires underwater technology available. If photos are obtained as part of D&D activities, please provide (but the US recognizes that additional information may not be obtained).
RB-3a	Photos/videos of damaged walls and structures (1F1)	• Determine mode of explosion in 1F1 compared to 1F3.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into 1F1 and after debris removal.	TEPCO has some information (Dose rate distribution measurement around SGTS filter was performed for 1F4 and 1F3. Visual inspection inside R/B was performed from view of integrity of structures for 1F4) If additional images are obtained as part of D&D activities, please include reference length scales (or information about component dimensions). The US will investigate the NRA website for images. In particular, if D&D strategy allows
RB-3b	Photos/videos of damaged walls and structures (1F3)	• Determine mode of explosion in 1F3. • Gain insight about highly energetic explosions in 1F3 compared to 1F1.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into 1F3 and after debris removal.	

[hh] With the exception of general area views, photos and videos should be obtained with a reference length (ruler) at appropriate locations. In particular, it would be extremely useful for RB-1, RB-2, and RB-13; it is required for photos and videos to be most effective for RB-9 and RB-10.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-3c	Photos/videos of damaged walls and structures (1F4)	• Determine mode of explosion in 1F4.	Understanding what happened; assist D&D efforts. Potential BWR improvements; Impacts BWR AM strategies and code models (venting and interconnection between units); Potential PWR impacts (e.g., modeling, AM strategies, etc.).	When TEPCO goes into 1F4 and after debris removal.	additional photos of the shield plugs for all units, include a reference length of damaged components, if possible. If shield plugs are removed, time lapsed videos during removal are requested.
RB-4	Photos/videos of damaged walls and components and radionuclide surveys (1F2)	• Cause of depressurization. • Cause of H_2 generation.	Understanding what happened; assist D&D efforts. Impacts BWR AM strategies (equipment utilization and venting); Improved BWR code simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Completed.	TEPCO has dose distribution information. This item has been addressed.
RB-5	Radionuclide surveys (1F1, 1F2, and 1F3)	• Leakage path identification. • Dose code benchmarks.	Understanding what happened; assist D&D efforts. Improved BWR Accident Management (plant robustness, training, SAMG). Improved BWR code simulations and dose code benchmarks, Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later (as debris is removed).	TEPCO has survey information in 1F1, 1F2, and 1F3 R/B. some concrete samples analyzed to investigate Cs permeation inside concrete floor. Dose rate distribution measurements on 1F2 and 1F3 including top of shield plug. Dose surveys obtained around 1F1, 1F2, and 1F3 pipe penetrations (outside end of penetrations through PCV) in R/B. W/W vent line in 1F1 extremely contaminated such as AC piping in R/B 1st floor, SGTS filter train area, piping connected to stack. Dose rate around rupture disc of 1F2 W/W vent line was performed. No contamination around rupture disc 1F2, but SGTS filter was highly contaminated. If isotopic composition of samples/swipes from drywell head are obtained, data are of interest. In particular, Ru information is of interest. Dose map of 1F1 after cleanup is of interest.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-6	Radionuclide surveys and sampling of ventilation ducts (1F4)	• Isotope concentration could be used for determining source of H_2 production for CCI.	Understanding what happened. Potential BWR plant improvements (hardened vent use, AM strategies, and multi-unit effects, etc.). Potential PWR impacts (e.g., modeling, AM strategies, multi-unit effects).	Completed.	TEPCO is not planning any additional examinations. This item is closed.
RB-7	Isotopic evaluations of obtained concrete samples (1F2)	• Code assessments. • Possible model improvements for building retention assumptions.	Understanding what happened; assist D&D efforts. Improved BWR modeling and emergency planning; cross check of RN surveys. Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now.	JAEA has obtained surface RN concentrations and RN distribution from boring concrete samples. Surface radionuclide concentrations and distribution of radioactive nuclides of boring core samples were obtained. If additional samples or surveys are obtained, isotopic composition is of interest (but the US recognizes that additional information may not it be obtained).
RB-8	Photos/videos and inspection of seismic susceptible areas (e.g., bellows, penetrations, structures, supports, etc. in 1F1, 1F2, 1F3, and 1F4)	• To confirm with data that there were no seismic-induced failures.	Understanding what happened; assist D&D efforts. Improved plant robustness; observed differences between 1F1 and 1F3. Potential PWR impacts (e.g., similar penetrations).	Now and later (as debris is removed); Note that debris currently precludes data from being obtained.	**1F1**: The IC main unit, major pipes, and major valves visually investigated to confirm whether or not there was any damage that could cause reactor to lose coolant. Since inside area of PCV inaccessible, IC, pipes, and valves outside PCV checked. **1F2:** No large abnormality was found in the robot camera's visual inspection. Visual inspection inside PCV performed in 1F1, 1F2, and 1F3 but inspection range limited. If additional information is obtained as part of planned D&D activities, please provide it (but the US recognizes that additional information may not it be obtained).

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-9	DW Concrete Shield Radionuclide surveys (1F1, 1F2, and 1F3 - after debris removed)	• To understand leakage amounts and locations.	Improved AM strategies (Plant improvements, training, and education). Improved codes. Understanding what happened; assist D&D efforts.	Now and later (as debris is removed).	TEPCO has photos and some RN surveys; more will be obtained. If additional informtion is obtained as part of planned D&D activities, please provide (but the US recognizes that additional information may not it be obtained).
	Photos/videos around mechanical seals and hatches and electrical penetration seals (as a means to classify if joints in compression or tension)	• Potential leakage paths for RN and hydrogen release.[ii]	Improved AM strategies (Plant improvements for BWR and PWRS, which have similar seals). Improved codes. Understanding what happened; assist D&D efforts.	Now and later.	TEPCO has photos and some dose survey information (see RB-10). If photos are obtained as part of planned D&D activities, please provide (but the US recognizes that additional information may not be obtained).
RB-10	Photos/videos and dose surveys of 1F1 (vacuum breaker), 1F1,	• Potential leakage paths for RN and hydrogen release.	Improved AM strategies (Plant improvements for more robustness, training, education); applicable to BWRs and PWRs	Now and later.	TEPCO has considerable information related to this information need.[jj] Now, restoring works for PCV to

[ii] For PWR containments, the containment actually grows radially as pressure and temperature are increased so penetrations that may have been in compression (e.g., hatches) may now be in tension.

[jj] **1F1:** Water leaks from a sand cushion drain pipe and an expansion joint (bellows) for vacuum breaker tube observed. The water leak from a sand cushion drain pipe was confirmed since the vinyl chloride pipe (connecting the sand cushion drain tube and drain funnel with an insertion-type joint) had been displaced. Water leaks could not be confirmed at other seven drain pipes, since the drain tubes had not been displaced. However, concrete seams (joints) below sand cushion drain piping were observed to be wet all around on the concrete wall, which indicates that leaked water is filled in the sand cushion area outside of PCV wall. The water leak from bellows of vacuum breaker tube is located in the direction of access opening of pedestal wall in the PCV floor where molten corium might spread out first.
1F2: It was confirmed S/C water level changes together with torus room water level. This indicates water is leaking from the lower position of S/C including suction piping. No water leakage from sand cushion drain pipes or vent pipe was observed. As of now, water leakage is not specified.
1F3: Water leak from near the expansion joint (bellows) of main steam line D in MSIV room was confirmed. The water level in the PCV is estimated at about 2 m above the reactor building first floor by converting the S/C pressure obtained by the existing pressure indicators to water head. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration of MSL.
1F3: Water seeping from equipment hatch is inferred from the following observations.

- Rust was observed along with the hatch interface lower than D/W water level (in November, 2015). Upper part of the interface does not have the rust.

-The increasing dose rate on the floor towards the equipment hatch was observed (in November, 2015), which indicates contaminated water had flown from D/W side.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
	1F2, and 1F3 PCV leakage points (bellows, penetrations).		(which have similar penetration designs). Improved codes. Improved understanding of events; assist D&D efforts.		stop water leakage are prioritized and no plan to scrutinize the damaged area or degree of PCV. If additional photos or information is obtained, please provide (but the US recognizes that additional information may not be obtained).
RB-11	Photos/videos and dose information on 1F1, 1F2, and 1F3 containment hardpipe venting pathway, SGTS and associated reactor building ventilation system	• To assess performance of seals under high temperature and radiation conditions.[kk]	Improved AM strategies (Plant improvements). Improved understanding of events, assist D&D efforts.	Completed.	**1F1:** Dose rate of venting pathway and the point in front of SGTS room. Because of high dose rate, access to SGTS room is difficult. **1F2 and 1F3:** Photos and dose rate of SGTS trains and venting pathway available. This item has been completed.
RB-12	Photos/videos at appropriate locations near identified leakage points in 1F1, 1F2, and 1F3.	• To discern reason for leakage from the reactor building into the turbine building.	Improved BWR AM strategies (Plant improvements); potential PWR impacts, depending on identified leakage path. Assist D&D efforts.	Now.	Not currently considered by TEPCO; some visual information available. This item has been addressed. If additional photos are obtained as part of planned D&D activities, please provide (but the US recognize that additional information may not be obtained).

- Equipment hatch rail was dry in December, 2015. Current D/W water level is lowest since 2011. The D/W water level in 2011 was higher and water seeping from D/W through equipment hatch seal would be higher.

- The observed high dose rate at the rail in front of shield plug for equipment hatch (in September, 2011) would be attributed to water leak through equipment hatch seal.

- Water dripping due to rain fall observed (in November, 2015, rainy day), which might be intruding from refueling floor.

No specific observation regarding gas phase leakage other than dose rate distribution on refueling floor and steam discharging from refueling floor.

[kk] Passage of high temperature gas from venting operations at 1F1 and 1F3 may have affected seals. The effluent vented from Units 1 and 3 would also have subjected these components to high radiation fields. Note that, at present, available evidence indicates that Unit 2 may not have been successfully vented. The high radiation fields in components of the 1F2 reactor building ventilation system appears to have been caused by 1F1 vent effluent bypassing the vent stack shared by 1F1 and 1F2. Many PWRs have safety grade fan cooler units for post-loss of coolant accident containment heat removal; PWRs would be interested if there is anything to learn.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-13	Photos/videos of 1F1, 1F2, and 1F3 main steam lines at locations outside the PCV.	• To determine PCV failure mode.	BWR AM Strategies (plant mods, etc.) and better simulations for training. Assist D&D efforts.	Now and later.	TEPCO has some visual information related to 1F2 MSIV. 1F3: Water leak from near expansion joint (bellows) of MSL D in MSIV room was confirmed. The water level in the PCV is estimated at about 2 m above the reactor building first floor by converting the S/C pressure obtained by the existing pressure indicators to water head, and this was confirmed during first PCV entry investigation. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration of MSL. TEPCO has some temperatures around MSIV recorded since September 2011 for 1F2 and 1F3. Some evidence also on 1F1 and 1F2 provided by Yamada at 4/28/16 meeting. This item has been addressed; However, if more information is obtained as part of planned D&D activiteis, please provide (but the US recognizes that additional information maynot be obtained).
RB-14	Chemical analysis of white deposits found in 1F1 HPCI room using XRD or other methods.	• Presence of Si would indicate MCCI	Assist D&D efforts for determining debris location.	Now	TEPCO is investigating potential to send white deposits to JAEA for evaluation.

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-1	Tension, Torque, and Bolt Length Records (prior and during removal); Photos/videos[ll] of head, head seals, and sealing surfaces (1F1, 1F2, and 1F3).	• Determine how head lifted. • Determine peak temperatures. • Look for indicators of degradation due to high temperature hydrogen, including hydrogen-induced embrittlement.	AM Strategies; What happened with respect to the leak path; better simulations for training. Assist D&D efforts.	Now (initial data and photos) and later (if head removed).	TEPCO observed that tensioning is done based on gap requirements; no record available. TEPCO may have last outage tension records and has obtained photos indicating: **1F1:** Shield plug seems to have moved upward, which was observed by camera's visual inspection in the operating floor. **1F2:** No large abnormality was found in the robot camera's visual inspection in the operating floor. Rubber boots remained standing on the shield plug. **1F3:** Deformation of part of shield plug was observed, which was found in the visual inspection after removing building rubbles. Additional photos may become available. The US would appreciate any additional information (although the US recognizes that this information may not be available). Visual images of deformation and RN samples (with isotopic content) are of particular interest.
PC-2	Photos/videos and radionuclide surveys/ sampling of IC (1F1).	• Evaluate for seismic damage. • Evaluate final valve position. • Gain insights about hydrogen transport.	AM Strategies (plant robustness, use of equipment in limited number of plants with ICs and new passive plants); better simulations for training. Assist D&D efforts.	Completed.	TEPCO has some photos (and no damage observed); no RN sampling planned (due to radiation levels). This item has been addressed.

[ll] With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations. In particular, it would be extremely useful for PC-3(b), PC-3(e), PC-9, PC-12, PC-13.

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-3	a) If vessel failed, photos/videos of debris and crust, debris and crust extraction, hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3). [mm]	• Code assessments • Possible model updates for mass, height, composition, morphology (e.g., coolability), topography of debris, spreading, splashing, and salt effects.[nn]	BWR AM Strategies (plant robustness, use of equipment) and better simulations for training. Potential PWR impacts (e.g., modeling.). Assist D&D efforts.	Now and > 5 years (per TEPCO roadmap).	TEPCO has obtained some samples and some photos from inside of 1F1, 1F2, and 1F3 PCV, more are planned. When additional information is available, please provide.
	b) If vessel failed, 1F1, 1F2, and 1F3 PCV liner examinations (photos/videos and metallurgical exams).	• Code assessments. • Possible model improvements for predicting liner failure and Molten Core Concrete Interactions (MCCI).	AM Strategies (improved plant robustness); better simulations for training. Assist D&D efforts.	Now and > 5 years (per TEPCO roadmap).	TEPCO has some bellows information and may obtain additional visual information. TEPCO may do metallurgical exams (if warranted). When additional information is available, please provide.
	c) If vessel failed, photos/video, RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor.	• For benchmarking code predictions of vessel failure location and area, mass, morphology (e.g., coolability), and composition of ex-vessel debris, and MCCI.	BWR AM Strategies, better simulations, etc. Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO has some information and may obtain additional information later. For 1F1, 1F2, and 1F3, camera and dose rate meter were inserted inside PCV and retained water level in D/W was sampled the water for radioactivity analysis. Sediment was observed in the floor but not debris (For 1F3, the floor was not observed). The inserting location was the opposite side from access opening of pedestal wall where molten corium might spread out first. In 1F2, camera images were taken at the pedestal opening into its inside. Images confirmed the position of the control rod position indicator probe (PIP) cables in the upper part of the pedestal opening, but no clear information was obtained regarding what was in the lower part inside the pedestal. If debris samples are obtained, a collaborative program to evaluate may be possible.

[mm] Although some images have been obtained; images do not indicate if RPV failed or show any relocated core debris.

[nn] Key to applicability for PWRs will be if melt composition does not significantly impact spreading; with different core materials, molten core debris may behave differently. If forensics can confirm basic properties or models, information could be applicable to all LWRs.

Table C-2. Information Needs from the Primary Containment Vessel

Item	What/How Obtained	Why	Expected Benefit /Use	When	Status
	d) If vessel failed, 1F1, 1F2, and 1F3 concrete erosion profile; photos/videos and sample removal and examination	• For benchmarking code predictions of MCCI.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO has no plans to obtain at this time. TEPCO may consider in the future. If end-state is observed, a collaborative program to evaluate may be possible.
	e). If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage and corium hang-up	• Code assessments. • Possible model improvements.	BWR AM Strategies (plant modifications, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Now and later.	TEPCO will obtain some information. See PC-7. The US believes this information is very important for determining vessel failure mode and area. Please provide additional information when available.
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	• To determine PCV failure mode and relocation path.	AM Strategies (plant mods, etc.) and better simulations for training.	Completed.	TEPCO has some pressure and temperature measurements at Primary Loop Recirculation (PLR) pump inlet since April 2011. No additional inspections planned. This item is closed.
PC-5	Photos/videos of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines	• To determine RPV failure mode.	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Now and later.	TEPCO has not considered photographic exams. TEPCO has some temperatures around SRV and MSIV recorded since September 2011 for 1F2 and 1F3. The US continues to have interest in photos to resolve questions regarding SRV failure versus main steam line rupture. In particular, some visual inspection of MSL would be very valuable. However, the US recognizes that additional information may not become available.
PC-6	Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes (interior valve mechanisms)	• To determine if there was any failure of SRVs and associated piping.	BWR AM Strategies (maintenance practices, etc.), SRV functioning in test facility data, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).	Later.	TEPCO has not yet developed plans for such examinations. The US continues to have interest in photos to resolve questions regarding SRV failure versus main steam line rupture. In particular, some visual inspection of MSL would be very valuable. However, the US recognizes that additional information may not become available.

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-7	Ex-vessel inspections and operability assessments of 1F1, 2, and 1F3 in-vessel sensors and sensor support structures[oo]	• Data qualification for code assessment. • Identification of vessel depressurization paths.	Equipment qualification life (1F1 at 40 years; underwater cabling); better simulations for training.	Now and later.	TEPCO has completed some examinations and recalibrations and plans to perform more evaluations. Cable integrity examinations by TDR (time domain reflectrometry) were performed for 1F1, 1F2, and 1F3; and cable damage was confirmed. In 1F2, it was confirmed TIP index tube was stuck. In 1F2, it was found SLC injection tube in RPV was stuck, which indicates blockage by molten core. -New thermocouple was inserted into nearby N-10 nozzle to reinforce RPV temperature monitoring in Oct. 2012. -Beforehand SLC line integrity was confirmed by injecting water and monitoring discharge pressure change. -Pressurized water of about 7MPa could not penetrate SLC line into RPV. This item has been addressed; if additional information is obtained, please provide.
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures[pp]	• Data qualification for code assessment. • Identification of vessel depressurization paths.	BWR and possible PWR equipment qualification life; better qualifications for training.	Now and later.	TEPCO has completed some examinations and recalibrations and plans to perform more evaluations. This item has been addressed; if additional information is obtained, please provide.

[oo] Ex-vessel inspections and evaluations [e.g., continuity checks, calibration evaluations, etc.) of in-vessel sensors [dP cells, water level gauges, TIPs, TCs, etc.] and sensor support structures, cables, removed TIPs, etc.; Requires knowledge of sensor operating envelop.

[pp] Inspections and evaluations (e.g., continuity checks, calibration evaluations, etc.) of suppression pool, PCV, and ex-vessel sensors (e.g., containment air monitors, pressure sensors, TCs, etc.) and sensor support structures and cables; Requires sensors operating envelop knowledge

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-9	Photos/videos of 1F1, 1F2, and 1F3 PCV (SC and DW) coatings	• Assess impact for coating survivability.	BWR and possible PWR maintenance upgrades.	Now and later.	Visual examinations inside PCV performed in 1F1, 1F2, and 1F3, although inspection range limited. TEPCO may obtain more data. Please provide additional information when available.
PC-10	1F1, 1F2, and 1F3 RN surveys in PCV	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out). Assist D&D efforts.	Now and later.	TEPCO has some sample evaluation and survey information and may obtain more data later. Radioactivity data obtained from retained water in basement of each building. Sampling water in D/W was performed for 1F1, 1F2, and 1F3. Sampling drain water and dust of exhaust gas from drywell was performed for 1F1, 1F2, and 1F3. S/C water not evaluated. The US remains very interested in isotopic information from RN surveys/samples for code assessments (but the US recognizes that this information may not become available).
PC-11	Photos/videos of 1F1, 1F2, and 1F3 primary system recirculation pump seal failure and its potential discharge to containment	• To assess performance under high temperature/ high pressure conditions.[qq]	Improved BWR AM strategies (plant improvements). Improved understanding of events. Assist D&D efforts. Potential PWR impacts.[qq]	Now and later. Exams may be completed more easily at Daini.	Not currently considered by TEPCO; some photos may already be available. The US remains interested in additional photographs from Daiichi or Daini (but the US recognizes that this information may not become available).

[qq] Some PWRs have inside containment recirculation systems for Emergency Core Cooling and Containment Spray. BWR recirculation pump seals and PWR reactor coolant pump seals have many material similarities; there may also be some information relevant to reactor coolant pump seals and their ability to function following recovery or provide core cooling with core debris in-vessel.

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-12	Photos/videos of 1F1, 1F2, and 1F3 TIP tubes and SRV/Intermediate Range Monitor (IRM) tubes outside the RPV	• To determine if failure of TIP tubes and SRV/IRM tubes outside the RPV led to depressurization.	BWR AM Strategies and maintenance practices, SRV performance insights, and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.). Assist D&D efforts.	Later.	An attempt was made to insert a fiber optic scope through the 1F2 TIP guide tube. The scope was stuck at the TIP indexer and could not get past that location. 1F2 SLC injection line blockage was confirmed (see PC-7). Also, see item PC-14 for SLC injection line stuck in RPV. The US continues to have interest in this information. However, the US recognizes that additional information may not become available.
PC-13	Photos/videos of 1F1, 1F2, and 1F3 insulation around piping and the RPV.	• To determine potential for adverse effects on long-term cooling due to insulation debris.	Improved BWR and PWR AM strategies (plant improvements).	Now and later.	Not currently considered by TEPCO; some photos may already be available. The US continues to have interest in this visual information. However, the US recognizes that additional information may not become available.
PC-14	Samples of conduit cabling, and paint from 1F1, 1F2, and 1F3 for RN surveys.	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations (plate out).	Now and later.	TEPCO has some sample information. The US continues to have interest in this information, but recognizes that additional information may not become available.
PC-15	Samples of water from 1F1, 1F2, and 1F3 for RN surveys.	• Dose code assessments. • Possible model improvements.	BWR and possible PWR AM strategies/Better simulations. Assist D&D efforts.	Completed.	TEPCO has some sampling information. Sampling water in D/W was performed for 1F1, 1F2, and 1F3. Sampling drain water and dust of exhaust gas from drywell was performed for 1F1, 1F2, and 1F3. This item has been addressed.
PC-16	Photos/videos of melted, galvanized, or oxidized 1F1, 1F2, and 1F3 structures.	• To provide indications of peak temperatures (for possible model improvements)	Improved AM strategies (Plant improvements).	Now and later, this should also be done at Daini.	Some photos may be available. The US continues to have interest in this visual information, but recognizes that additional information may not become available.

Table C-3. Information Needs from Reactor Pressure Vessel

Item	What/How Obtained	Why	Expected Benefit /Use	When	Status
RPV-1	1F1, 1F2, and 1F3 dryer integrity and location evaluations (photos/videos[rr] with displacement measurements, sample removal and exams for fission product deposition, peak temperature evaluations)	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual, some metallurgical and fission product exams. The US remains interested in this information, but recognizes that it may not be available. Laser-Induced Breakdown Spectroscopy methods might reduce costs for chemical evaluations in exams (ongoing R&D at JAEA may make it easier to obtain this information).
	Photos/videos, probe inspections, and sample exams of 1F1, 1F2, and 1F3 MSLs; Interior examinations of MSLs at external locations	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO has no plans for any such exams. See PC-3 for water leakage information from MSL penetration through PCV. The US remains interested in this information, but recognizes that it may not be available.
	Photos/videos and metallurgical examinations of upper internals and upper channel guides	• Code assessments. • Possible model improvements (for predicting peak temperatures, displacement, melting).	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams and some metallurgical exams. The US remains interested in this information, but recognizes that it may not be available.

[rr] With the exception of general area views, photos and videos should be obtained with reference length scales at appropriate locations. In particular, it is required for photos and videos to be most effective for RPV-1(b), RPV- 2(a), RPV-3 and RPV-4(d)

Table C-3. Information Needs from Reactor Pressure Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RPV-2	Photos/videos of 1F1, 1F2, and 1F3 core spray slip fit nozzle connection, sparger & nozzles Photos/videos of 1F1, 1F2, and 1F3 feedwater sparger nozzle and injection points	• Assess operability. • Assess salt water effects (including corrosion). • Applicable to BWRs and PWRs.	Improved AM strategies; Improved simulations for training; Possible use in BWR Vessel and Internals Program (VIP) [depending on plant condition]. Assist D&D efforts.	Now and Later.	TEPCO has some information) and will obtain more data. When water injected through CS line in 1F1, 1F2 and 1F3, it was confirmed that RPV bottom temperature responds. When water injected through FDW line in 1F1, 1F2, and 1F3, it was confirmed that RPV bottom temperature responds. The US remains interested in this information, but recognizes that it may not be available.
RPV-3	1F1, 1F2, and 1F3 steam separators' integrity and location (photos/videos with displacement measurements, sample removal and exams for FP deposition, peak temperature evaluations)	• Code assessments. • Possible model improvements.	Improved AM strategies, Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual, some metallurgical and fission product deposition exams. The US remains interested in this information.

Table C-3. Information Needs from Reactor Pressure Vessel

Item	What/How Obtained	Why	Expected Benefit /Use	When	Status
RPV-4	1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos and sample removal and oxidation testing.	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training. Assist D&D efforts.	Now and later (after 2017 based on current roadmap).	TEPCO has some information and will conduct visual exams. 1F2 PLR pump responded after increasing water flow rate from FDW, indicating a certain amount of water is retained outside shroud. The US remains interested in this information, but recognizes that some information may not be obtained.
	1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos, and metallurgical exams)	• Code assessments. • Possible model improvements.	Improved AM strategies; Improved simulations for training.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams and some metallurgical exams. The US remains interested in this information, but recognizes that some information may not be obtained.
	Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region)	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams. The US remains interested in this information, but recognizes that some information may not be obtained.
	Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures	• Code assessments. • Possible model improvements.	Improved AM strategies; Possible plant modifications; Improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO will conduct visual exams. The US remains interested in this information, but recognizes that some information may not be obtained.
RPV-5	Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed)	• Code assessments. • Possible model improvements.	Improved BWR and potential PWR AM strategies; Improved simulations for training. Assist D&D efforts.	Now and later (after 2017 based on current roadmap).	TEPCO is deploying muon tomography. Now preparing for reactor imaging by cosmic ray muon tomography. TEPCO deployed muon attenuation method to 1F1 and 1F2. TEPCO and IRID plan to apply muon scattering method to 1F2.

Table C-3. Information Needs from Reactor Pressure Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
	Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.)	• Code assessments. • Possible model improvements for predicting debris composition, mass, and morphology (e.g., coolability, topography of debris, spreading, splashing, and salt effects.	Improved BWR and potential PWR AM strategies; plant modifications, and improved simulations for training. Assist D&D efforts.	Later (after 2017 based on current roadmap).	TEPCO has not yet considered but will probably perform, as necessary for defueling and D&D. If samples are obtained, a collaborative program to evaluate may be possible.

Appendix D

Mid-and-Long-Term Roadmap Phase II Activities

[Figures provided courtesy of NDF; Reference 79]

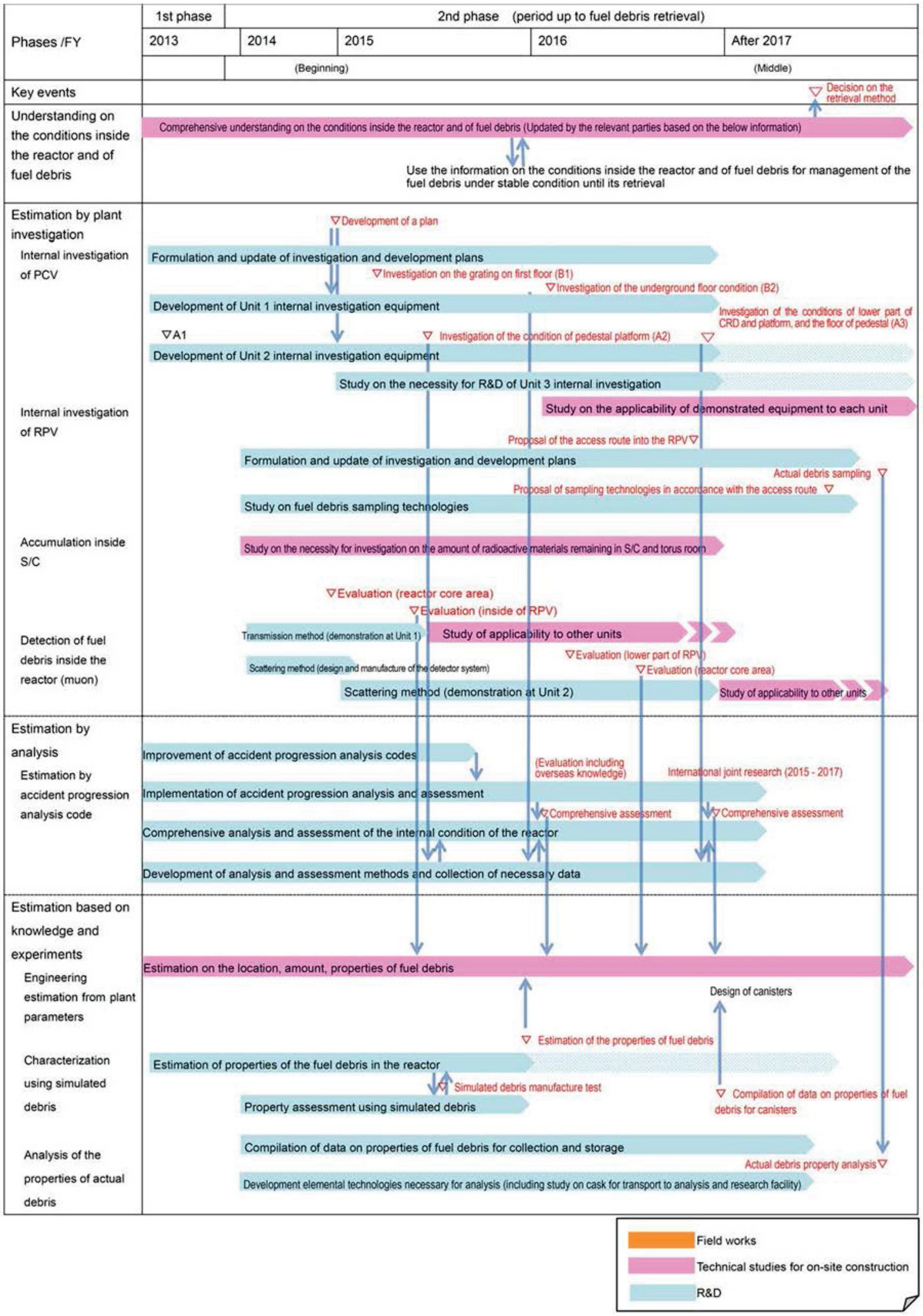

Figure D-1. Phase II actions to characterize reactor and fuel debris conditions to support debris removal. (Courtesy of NDF [79])

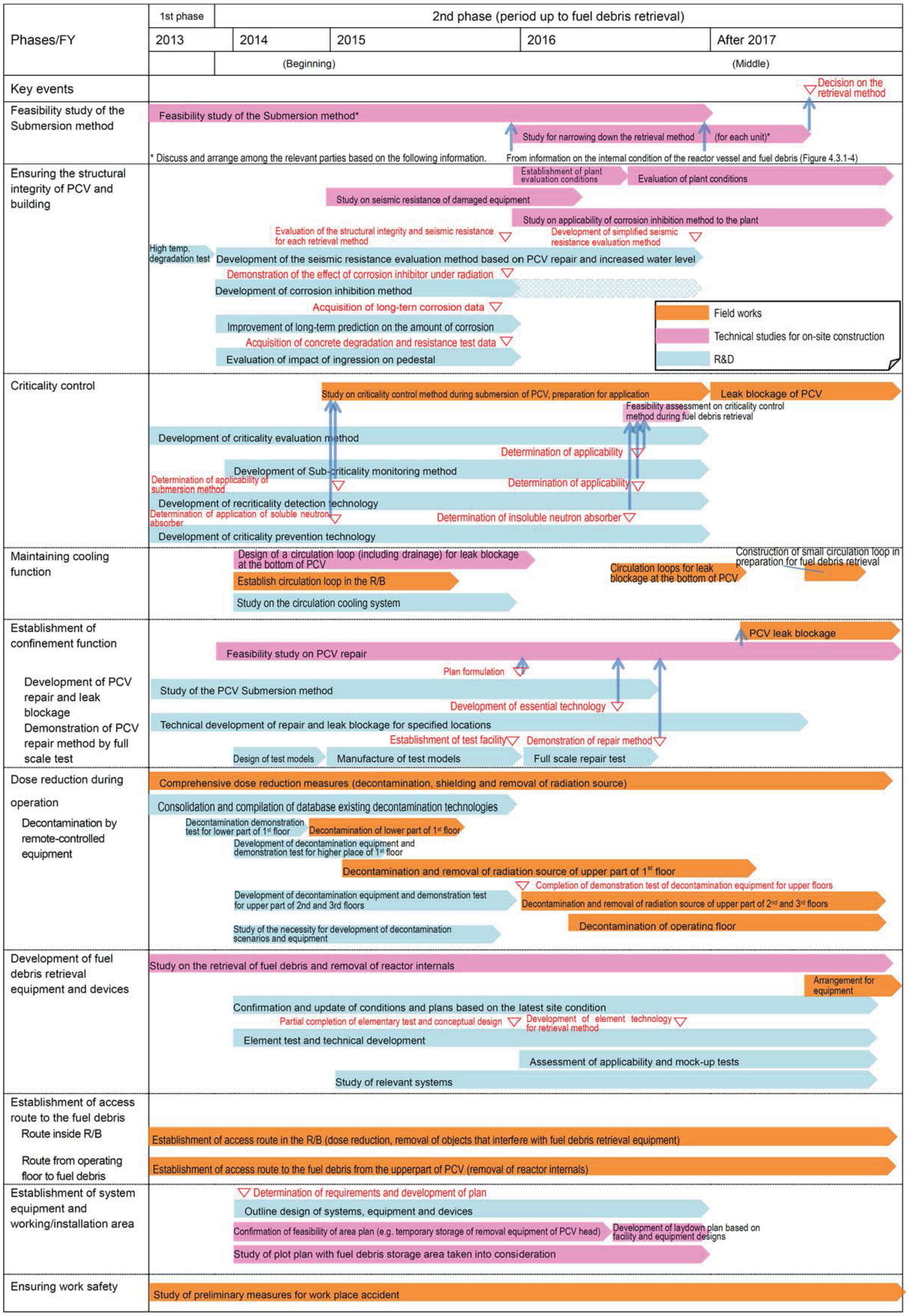

Figure D-2. Phase II actions to complete feasibility study for the full submersion method. (Courtesy of NDF [79])

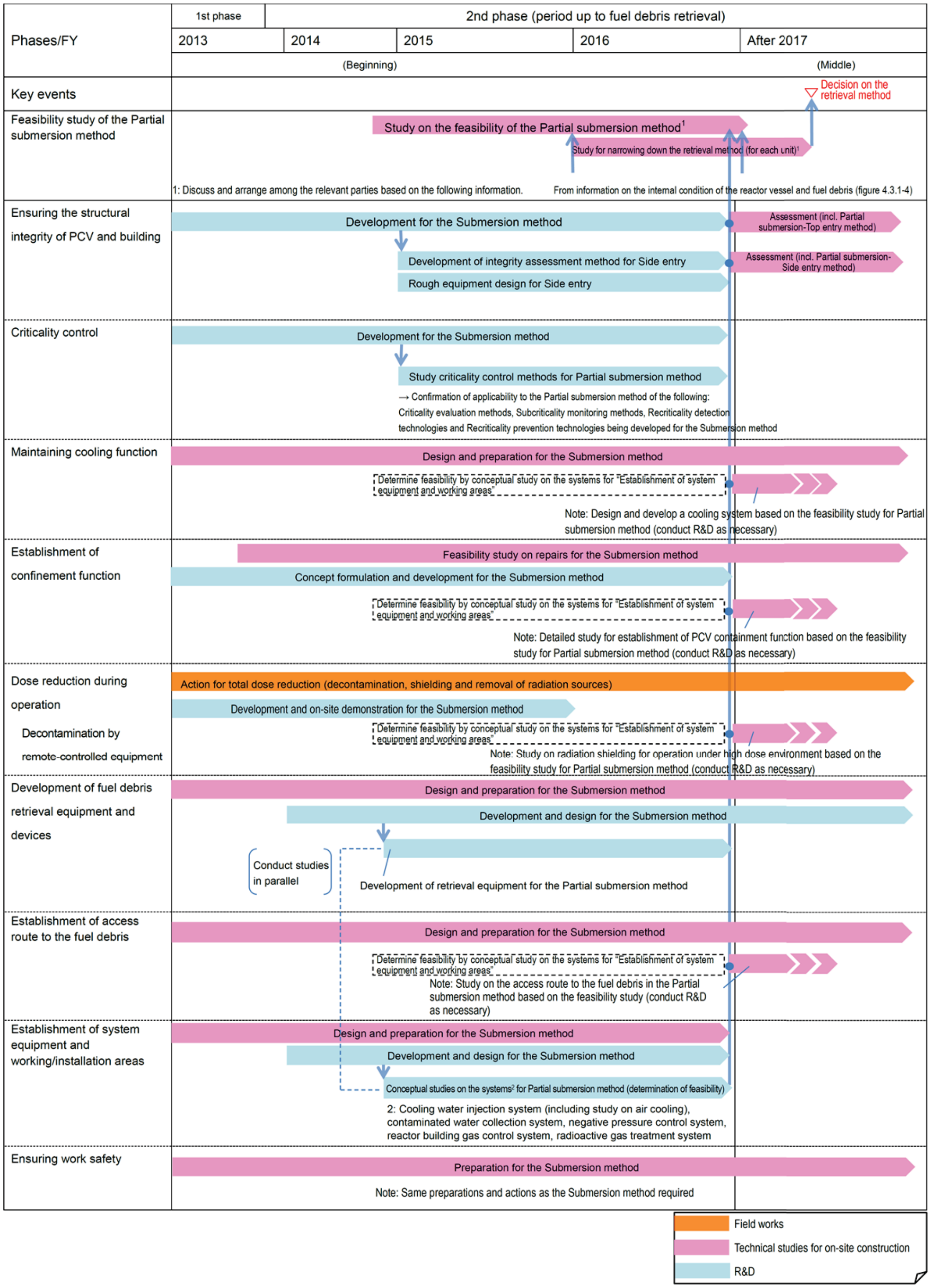

Figure D-3. Phase II actions to complete feasibility study for the partial submersion method. (Courtesy of NDF [79])

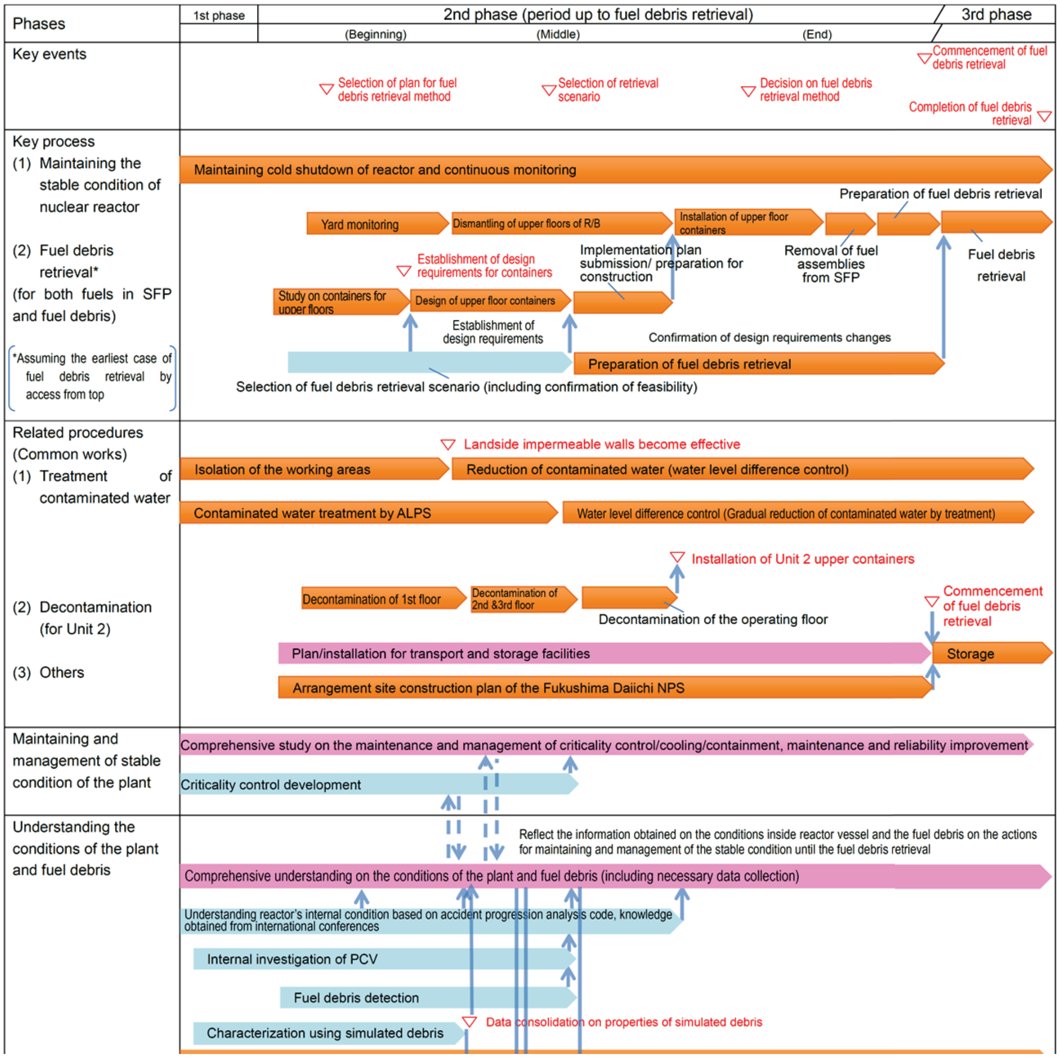

Figure D-4. Phase II actions to complete entire process of fuel debris retrieval. (Courtesy of NDF [79])

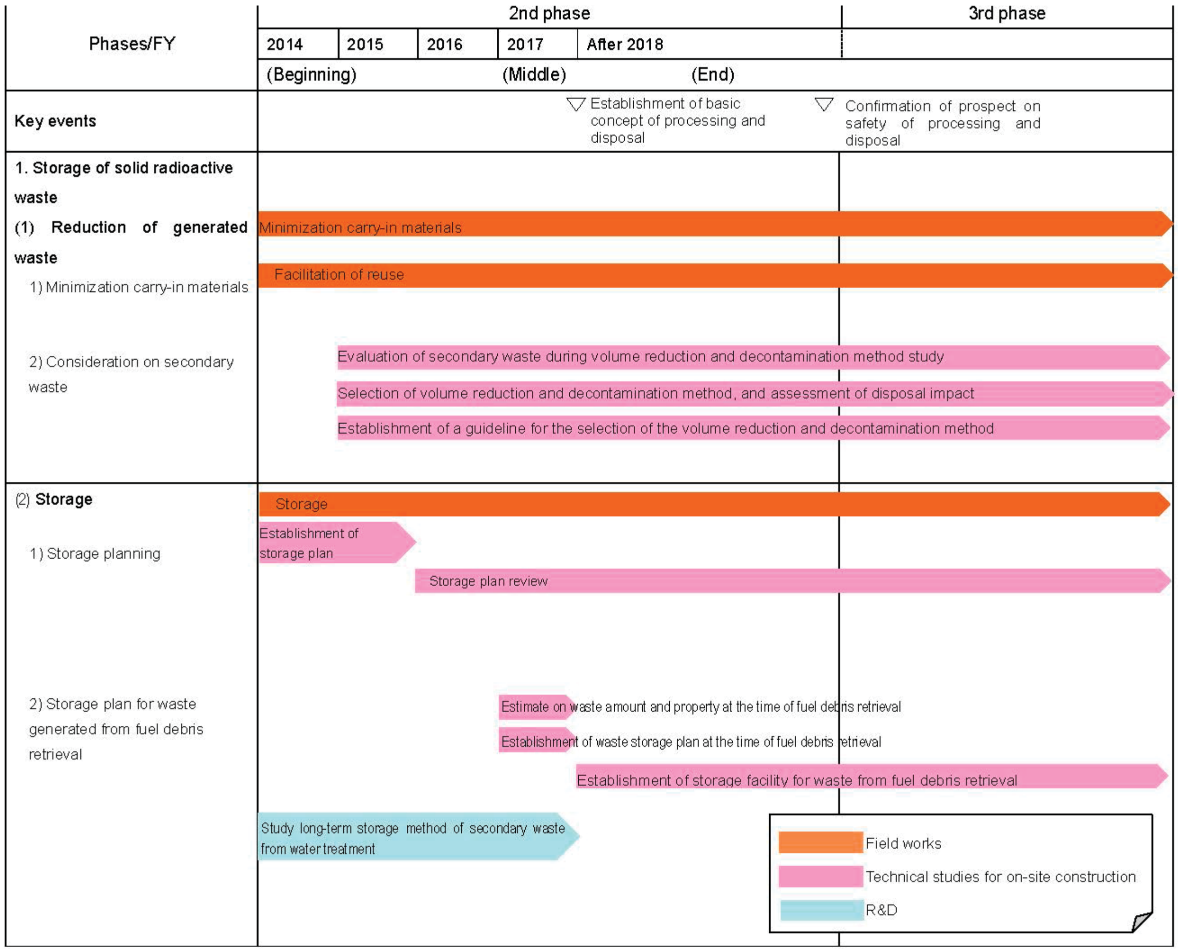

Figure D-5. Phase II and III actions for waste management (Sheet 1 of 2). (Courtesy of NDF [79])

Figure D-6. Phase II and III actions for waste management (Sheet 2 of 2). (Courtesy of NDF [79])