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U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2017 Evaluations

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ABSTRACT

Although 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, Incorporated (TEPCO Holdings) to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. This document, which has been updated to include FY2017 information, summarizes results from U.S. efforts to use information obtained by TEPCO Holdings to enhance the safety of existing and future nuclear power plant designs. This effort, which was initiated in 2014 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 U.S. experts in LWR safety and plant operations that have identified examination needs and are evaluating TEPCO Holdings information from Daiichi that address these needs. Each year, annual reports include examples demonstrating 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.

Highlights in this FY2017 report include new insights with respect to the forces required to produce the observed Daiichi Unit 1 (1F1) shield plug endstate, the observed leakage from 1F1 components, and the amount of combustible gas generation required to produce the observed explosions in Daiichi Units 3 and 4 (1F3 and 1F4). This report contains an appendix with a list of examination needs that was updated after U.S. experts reviewed recently obtained information from examinations at Daiichi. Additional details for higher priority, near-term, examination activities are also provided. This report also includes an appendix with a description of an updated website that has been reformatted to better assist U.S. experts by providing information in an archived retrievable location, as well as an appendix summarizing U.S. Forensics activities to host the TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting that was held in Idaho Falls, ID, on October 10-14, 2016.

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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
CFD	Computational Fluid Dynamics
CLADS	Collaborative Laboratories for Advanced Decommissioning Science
CRD	Control Rod Drive
CNWG	Civil Nuclear Energy and Research Working Group
CRIEPI	Central Research Institute of Electric Power Industry
CS	Core Spray
CSNI	Committee on the Safety of Nuclear Installations
CV	Containment Vessel
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
EPG	Emergency Planning Guideline
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
FY	Fiscal Year[a]
GEH	GE-Hitachi Nuclear Energy, Limited
GPU	General Public Utilities
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HP	Hold Point
HPCI	High Pressure Coolant Injection
IAE	Institute of Applied Energy

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

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
JAEA	Japan Atomic Energy Agency
LIBS	Laser Induced Breakdown Spectroscopy
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
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NAcPs	National Action Plans
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
NSC	Nuclear Science Committee
NTTF	Near Term Task Force
NUGENIA	NUclear GENeration II & III Association
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
PCPL	Primary Containment Pressure Limit
PCV	Primary Containment Vessel
PLR	Primary Loop Recirculation
PM	Plant Maintenance
PORV	Pilot-Operated Relief Valve
PreADES	Preparatory Studies for Fuel Debris Analysis
PRA	Probabilistic Risk Assessments
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 or RBCCW	Reactor Building Closed Cooling Water System
RN	RadioNuclide
RPV	Reactor Pressure Vessel
RPI	Rensselaer Polytechnic Institute
RST	Reactor Safety Technologies

SAG	Severe Accidence Guidance or Severe Accident Guideline
SAMG	Severe Accident Management Guideline
SAREF	SAfety REsearch opportunities post-Fukushima
SARNET	Severe Accident Research NETwork
SAWA	Severe Accident Water Addition
SBO	Station BlackOut
SC or S/C	Suppression Chamber
SEG	Senior Expert Group
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Cooling
SNL	Sandia National Laboratories
SRM	Source Range Monitor
SRV	Safety Relief Valve
SSC	Structures, Systems, and Components
TAG	Technical Advisory Group (for TMI-2 defueling and cleanup)
TAMU	Texas A&M University
TBR	Technical Basis Report
TC	Thermocouple
TCOFF	Thermodynamic Characterization Of Fuel debris and Fission products based on Scenario Analysis for Severe Accident Progression at Fukushima-Daiichi NPS
TEPCO Holdings	Tokyo Electric Power Company Holdings, Inc.
TIP	Traversing In-core Probe
TMI-2	Three Mile Island Unit 2

TVA	Tennessee Valley Authority
U.S.	United States
UW	University of Wisconsin
VIP	Vessel and Internals Program
WW or W/W	Wetwell
WWBX	Willis Walter BiXby (WWBX), LLC
XRD	X-Ray Diffraction
XRF	X-Ray Florescence
1F1	Fukushima Daiichi Unit 1
1F2	Fukushima Daiichi Unit 2
1F3	Fukushima Daiichi Unit 3
1F4	Fukushima Daiichi Unit 4

U.S. Efforts in Support of Examinations at Fukushima Daiichi – 2017 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 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, evaluations show that 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 U.S. efforts to participate in the Daiichi Forensics Evalautions with the following objectives:

- ***Objective 1:*** Develop consensus U.S. 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, Incorporated, (TEPCO Holdings) 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 Holdings 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 U.S. effort. In fact, as discussed in [2,3,4,5] and within this document, the U.S. 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 this U.S. 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 U.S. have the role of implementing institutional measures to ensure prevention of severe accidents. For example, as discussed in Section 2.2.1.2, the U.S. Nuclear Regulatory Commission (U.S. 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 U.S., 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 U.S. 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. As discussed in Section 2.2.2, organizations within Japan are initiating such international efforts. The U.S. Forensics Effort provides a means for U.S. experts to contribute to and benefit from such international efforts.

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, plant staff 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 this U.S. effort are beneficial to the U.S. and to Japan.

For the U.S., this effort provides access to prototypic data from three units with distinctively different accident signatures. U.S. 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.[6] Similarly, the events at Daiichi provide information to better ensure that operators are able 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.[7, 8] 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 U.S. nuclear engineers regarding LWR severe accident behavior.

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

- *Plant Operations* – The U.S. has over 20 operating BWRs, and personnel with considerable experience with respect to BWR operations.
- *Reactor Safety* - Experts involved in this U.S. effort lead development of U.S. severe accident codes and large-scale U.S. experimental programs.
- *TMI-2 Post-Accident Examinations and Cleanup* – Several U.S. experts participating in this program were also involved in TMI-2 post-accident evaluations. During FY2017, the U.S. DOE collaborated with the U.S. NRC to host a U.S.-Japan TMI-2 Knowledge Transfer and Relvance to Fukushima meeting to promote exchange of relevant information to cognizant organizations within Japan.

Unique U.S. expertise provides TEPCO Holdings 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, U.S. 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 U.S. 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. Conversely, additional samples to characterize the interface between relocated debris and the vessel head could help reduce uncertainties in characterizing heat transfer from relocated debris and the potential for vessel failure. Such insights are useful to Japan.

Because of the benefit to global nuclear reactor safety, it is recognized that an international framework may ultimately be needed to support post-accident examinations. If such an international framework is established, it must be led by Japanese organizations. Nevertheless, the U.S. has a vested interest in these examinations. The U.S. 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 U.S. plant designs. Hence, U.S. organizations – both industry and government—are major beneficiaries from any improvements in LWR severe accident knowledge just as the U.S. was a major beneficiary of significant Japanese participation in prior international TMI-2 programs. U.S. collaborative work with the international community in establishing this framework to support our Japanese colleagues is beneficial to the U.S. and Japan, but also offers 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, this is the third year of this effort. Actions taken during FY2015 to complete Objective 1 differed from activities completed during FY2016 and FY2017 to attain Objective 2.

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 U.S. NRC, the U.S. DOE, and TEPCO Holdings also attended and informed participants during these meetings. This effort resulted in a report [5] with a prioritized initial list of information of interest to U.S. stakeholders. In this report, special attention was devoted to identifying why such information is important and how it will be used to benefit the U.S. nuclear enterprise. 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, U.S. 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.
- D&D plans (or activities already completed) address much of the information identified by the U.S. expert panel.
- Maximum benefits from this information requires: reviews by cognizant experts, posting for easy-to-use access, interactions with TEPCO Holdings 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 D&D efforts at Daiichi.
- Important information and data are already available, and more is being gathered at the current time. U.S. 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, 1F3, and 1F4).[b] Although details varied, U.S. experts generally identified 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 [5] tables per location [e.g., the reactor building (RB), the primary containment vessel (PCV), and the reactor pressure vessel (RPV)]. These tables also identify the

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

applicable units for each need and other relevant factors (e.g., how information should be obtained, why it is needed, its expected use or benefits, when it should be obtained, 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 in each unit. Therefore, experts 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. Experts agreed 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 [5]

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	****		***	**

Another important conclusion is that much information is already available. As discussed in Section 1.3.2, Objective 2 activities are underway to evaluate available information (and make additional requests, if required).

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[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.**

1.3.2 Activities to Complete Objective 2

Activities used to complete the second objective are shown in Figure 1. As shown in this figure, activities and products completed by U.S. 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 gray box, severe accident and plant operations experts from U.S. industry, universities, and national laboratories evaluate plant examination information obtained from Daiichi. Objective 2 activities were also informed by experts from the U.S. NRC, U.S. DOE, and TEPCO Holdings that participated in expert panel meetings.

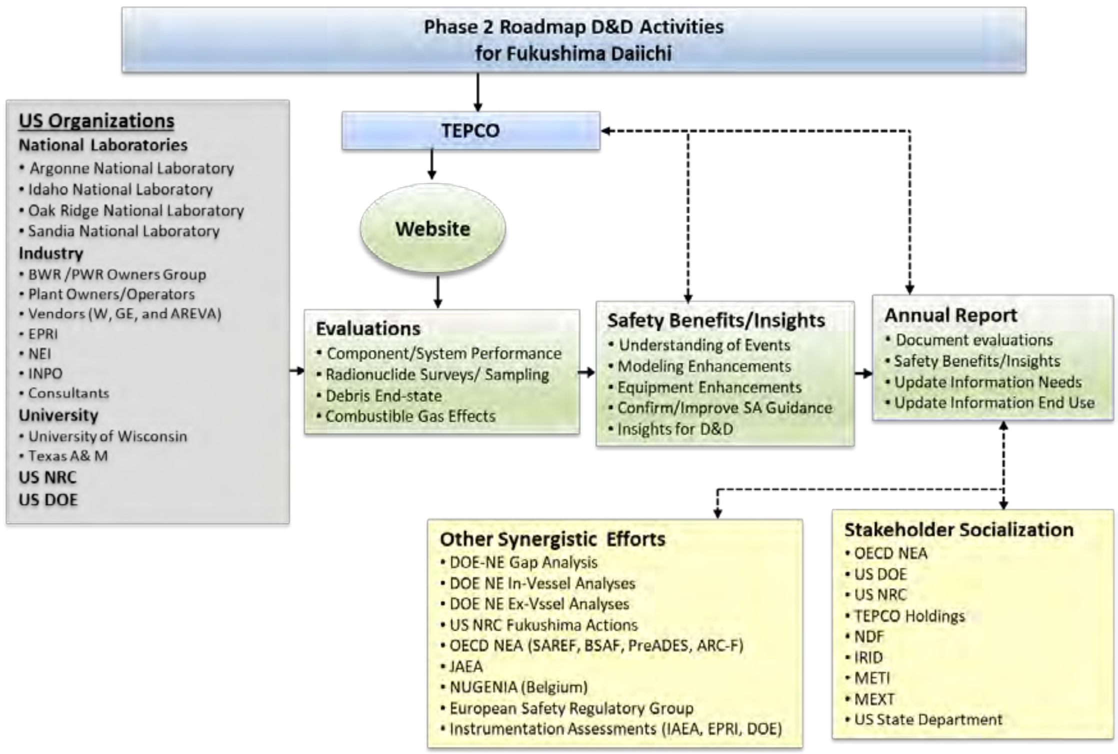

Figure 1. Objective 2 activities[d].

Specific individuals and organizations participating in expert meetings during FY2017 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 U.S. stakeholder input.

Activities and products completed by U.S. organizations are shown in green. Severe accident and plant operations experts evaluated information from four higher priority topic areas identified by the panel. These areas are:

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

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[d] See acronym list.

[e] This fourth area was identified in FY2016.

During the FY2017 meetings, the expert panel agreed that a fifth area, "Plant Operations and Maintence," should be initiated in future years. This area will cover a range of topics of interest to industry, starting with instrumentation survivability information obtained from examinations at Daiichi.

The primary source of information used in U.S. Forensics Effort evaluations is the TEPCO Holdings website.[9] Presentations provided by representatives from TEPCO Holdings,[10 through 21], industry[2, 22], and topic area leads[e.g., 23 through 33]. TEPCO Holdings reports documenting unconfirmed and unresolved issues also received special attention in the forensics effort.[34 through 38] The website created by the Institute of Applied Energy (IAE) [39] is also becoming an important reference for this effort. Appendix B describes the updated effort to archive selected references from TEPCO Holdings and other key sources used by U.S. experts 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 and training for severe accident prevention, mitigation, and emergency planning
- Additional insights beneficial to future D&D activities

As shown in Figure 1, U.S. 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 results from this process. For each area, prioritized questions of interest are identified; available information is reviewed; and insights gained from evaluating this information are provided. Where appropriate, information needs are updated, and a complete list of information needs that includes these updates is provided in Appendix C of this report. In addition, Appendix C includes additional details, such as the benefits, use, and suggested methods for obtaining higher priority, near-term examination activities.

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 this 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 these U.S. activities. In addition, as discussed in Section 2.2, results from this U.S. 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 reports issued in prior years were provided to external stakeholders for comment. To the extent possible, stakeholder comments were addressed in preparing this FY2017 document.

1.4 Report Objectives and Organization

This report summarizes efforts by U.S. experts to evaluate available examination 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 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 FY2017. Appendix B provides a description of efforts underway to archive noteworthy references relied upon in this effort. Appendix C provides updated tables with detailed information needs developed during expert meetings and additional details for high priority, nearer term examination activities. Appendix D contains recent roadmaps produced by the Japanese Government that detail planned D&D activities. Appendix E provides additional information about the "U.S.-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting" that was held as part of the U.S. Forensics Effort during FY2017.

2. Background

As part of this project, experts review important aspects of the TMI-2 evaluation process, synergistic activities underway by other U.S. 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 Evaluations

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. [1,6,40,41] 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.

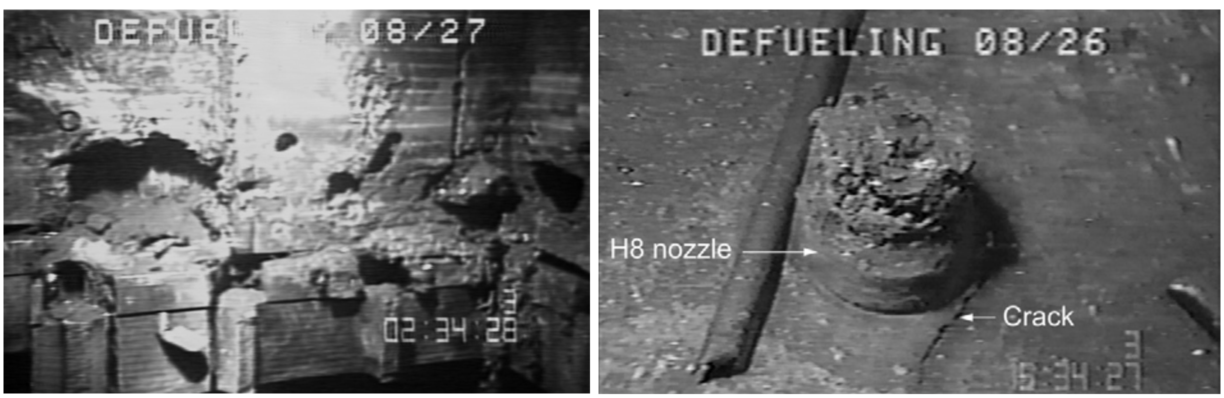


Figure 2. TMI-2 video examinations revealed locations where damage to core barrel and nozzles was more severe. (Courtesy of FirstEnergy)

A systematic investigation of the costs and benefits of TMI-2 inspection information is not available. However, visual inspection information from within the TMI-2 vessel and the containment clearly offered important insights at a lower cost than insights gained from post-accident examinations of radioactive samples. 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 examination information is not available. Such an investigation could provide insights related to the desired number and type of sample measurements, better characterization of 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 U.S. 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).

Although examination information did not reduce regulatory oversight, it enabled General Public Utilities (GPU) to address several technical issues and allowed defueling to proceed. Issues in which examination information was used include:

- *Core Debris Condition* - As documented in [42], initial examination and sampling activities provided evidence that large portions of the core exceeded clad oxidation and melting temperatures and that significant fuel liquefaction and melting occurred. Evidence that large fractions of core material (up to 20,000 kg) relocated to the space between the reactor vessel lower head and the elliptical flow distributor. Evaluations also found that fission product retention was significant within samples of core materials. Such information caused TMI-2 clearnup plans and equipment to be revised.
- *Recriticality* – As documented in [43,44], the allowable fuel mass limit was increased from 70 to 140 kg for TMI-2 during defueling activity and in long-term storage condition evaluations. This limit was developed using more realistic assumptions based on data obtained from debris samples, video inspections, and other defueling data that were unavailable at the time of the original 1985 analysis. These data provided a better understanding of the accident scenario and the actual debris configuration and composition, permitting a refined and more realistic modelling of the fuel debris. Results from calculations performed by independent organizations, in conjunction with available information from examinations showing that the fuel was oxidized and mixed with structural material, were used to select which parameters to monitor during defueling and during shipping.
- *Pyrophoricity* - The TMI-2 Technical Advisory Group (TAG) raised concerns about the potential for pyrophoric reactions from cutting and removal of debris containing metallic zircalloy, U/Zr alloy, and zirconium hydride. As documented in [44, 45], theoretical evaluations, prior experience, and separate effect tests were completed to show that conditions to cause such reactions were limited; and examination information was used to confirm that such conditions were not present at TMI-2.
- *Hydrogen Generation* – There were concerns about radiolytical hydrogen generation associated with debris defueling, transportation, and storage. Examination information related to debris composition and oxidation [44] was used to demonstrate that the offgas system would preclude formation of combustible gas concentrations.
- *Fission Product Gas Release* - There were concerns that chipping of the fuel could lead to release of gasuous fission product (Cs, Sr, etc.). As documented in [42, 44, 46], knowledge of the fuel compositions and fission product retention alleviated such concerns.
- *Plant growth within the RPV* - This issue stopped defueling due to reduced visibility within the reactor vessel.[47] Neither the NRC or the TAG had anticipated that conditions at TMI-2 within the reactor vessel would allow such growth (e.g., radiation levels were high). The decayed plant material in the reactor vessel stopped cleanup operations because visibility was significantly reduced. Underwater cameras within the vessel were unusable. To eliminate such plant growth, hydrogen peroxide was injected into the water to remove the decayed plant material. Concerns about adverse effects (recriticality, chemical reactions, etc.) associated with injecting this hydrogen peroxide were addressed using examination information.

In summary, examination information was essential for TMI-2 cleanup efforts.

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 U.S. NRC, the US DOE, and industry have initiated several efforts in the severe accident area. Furthermore, several new international activities have been initiated. To minimize duplication, it is important that the RST pathway remain cognizant of such activities. Table 2 lists synergistic activities that are deemed to be of special interest to the U.S. 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 U.S. and international organizations.

Table 2. Synergistic activities of special interest[f]

Source	Organization(s)/Countries	Activity/Objective
U.S.	U.S. 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 U.S. 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
	U.S. 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, IAE, IRID, JAEA, TEPCO Holdings	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.
U.S. - Japan	U.S.: U.S. DOE and U.S. NRC JAPAN: METI, MEXT, NDF, JAEA, NRA	CNWG; Collaborative activities related to wide range of research, including examinations, instrumentation, analyses, and workshops, such as the TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting held in Idaho Falls, ID, on October 10-14, 2016.
EU	NUGENIA (includes SARNET)	Prioritization Evaluations; Prioritize 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	NEA Senior Expert Group on SAfety REsearch opportunities post-Fukushima (SAREF)	Senior group of technical and regulatory experts formed to identify activities and research opportunities to address safety research gaps and advance safety knowledge related to the Fukushima Daiichi nuclear accident and to support safe and timely decommissioning in Japan. As an outgrowth, several near-term follow-on projects are proposed (see below).
	Preparatory Study on Analysis of Fuel DEbriS (PreADES)	Proposed near-term follow-on SAREF project that will include: • Fuel debris characterization (analytical, experimental) • Evaluation of radiation doses during fuel debris retrieval operations • Discussion on future long-term projects (fuel debris sampling)
	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF)	Improve severe accident codes by analyzing the accident progression and status of 1F1, 1F2, and 1F3 using a common information database. Phase I analyses covered the first six days of the accidents and focussed on thermal-hydraulics and estimating distribution and composition of degraded core materials. Phase II analyses extended to the first three weeks of the accident and focused on fission product releases and transport within the PCV, the RB, and offsite. Results support safe and timely decommissioning of reactors at Daiichi and identification of data needs for code validation.
	Thermodynamic Characterization Of Fuel Debris and Fission Products based on Scenario Analysis for Severe Accident Progression at Fukushima-Daiichi NPS (TCOFF)	Follow-on Nuclear Science Committee OECD project to perform a detailed thermodynamic characterization for predicting the current status of fuel debris and fission products within units 1, 2 and 3 of Fukushima Daiichi NPS.

[f]See acronym list.

2.2.1 U.S. Efforts

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

2.2.1.1 U.S. DOE

After the initial response to the events at Daiichi, the U.S. DOE [48] 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 during a severe accident.

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 U.S. NRC-developed MELCOR systems analysis codes.[49 through 51] 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.[52,53] The DOE and the Electric Power Research Institute (EPRI) continue to conduct analyses using existing computer models to provide information and insights into severe accident progression.[54] Results from 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 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 U.S. 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 Holdings 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 U.S. industry, U.S. NRC, or U.S. DOE. As discussed in [55, 56], 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 U.S. 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 U.S. 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 U.S. DOE has worked with industry to launch a test program 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) may also be investigated.

The need for reliable instrumentation was recognized after the TMI-2 event,[6] 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 U.S. NRC [57 through 60], several efforts have been sponsored by the U.S. DOE[61,62] and industry groups.[63 through 66]

2.2.1.2 U.S. 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).[67] Results from this effort are documented in a report that contained twelve 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, proposed safety enhancements must be shown to be cost beneficial using established processes.[68, 69] 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.[70, 71, 72, 73, respectively] Currently, it is planned that all these regulatory actions, which address the most important insights from the Fukushima accident, will be provided for in rulemaking. Initially, recommendations related to SAMGs were planned to be addressed in rulemaking.[74] However, in [75], 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). Efforts continue to finalize the rule related to post-Fukushima actions.[76]

As documented in [77,78], the U.S. 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 U.S. 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 [79], NRC severe accident phenomena expertise informed regulatory actions to address post-Fukushima activities.

2.2.1.3 Industry

In response to the events at Daiichi, industry led efforts within the U.S. to take independent steps to develop diverse and flexible coping strategies for BDBEEs, known as FLEX.[80] The focus in the U.S. 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,"[81] 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,[80]
- Staffing and communications recommendations,[82]
- Implementation of new spent fuel pool instrumentation,[83]
- Plant walkdowns to ensure adequate flooding protection, [84]
- Reliable containment venting for Mark I and Mark II BWRs,[85]
- Integration of Accident Management Procedures and Guidelines,[86]
- Enhanced Emergency Response Preparedness,[87, 88]
- Seismic evaluation guidance,[89,90] and
- Plans for enhancing SAG. [88, 91]

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. The strategies rely upon a combination of fixed, in-place and portable equipment protected from BDBEEs. The FLEX concept also involves staffing, communications, 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 [91] 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 the 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.

The basis for each of these insights was drawn from forensic evidence reviewed by the U.S. 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 U.S. 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 U.S. DOE efforts. Section 2.3 of this report provides the organizational structure and current roadmap for completing these activities. For the U.S. efforts to be successful (and to minimize the impact of inspection activities), it is critical that the U.S. remain cognizant of Japanese plans for completing D&D activities and of results from these activities. Furthermore, it is important that the U.S. 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 completed a severe accident gap analysis within Japan [92]. 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 Japan and the U.S. DOE activities to address these gaps. As discussed below, some of these opportunities are covered under existing bilateral agreements between Japan and the U.S. Other opportunities can be pursued through international collaborations in which Japan and the U.S. participate.

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).[93] Formal arrangements have been established covering collaboration in multiple areas including several relevant to LWR safety and post-accident evaluation [94]; 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 the area of reactor examination planning as it relates to informing D&D activities within Japan.

In addition, METI requested that the U.S. hold a workshop to facilitate transfer to Japan of knowledge that the U.S. gained from D&D activities at TMI-2. According, the U.S. DOE and U.S. NRC jointly sponsored the "US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima," in Idaho Falls, Idaho on October 11-12 2016. Over 35 U.S. experts, currently employed by or previously employed by national laboratories, the U.S. NRC, the U.S. DOE, and industry, such as EPRI, First Energy Corporation (formerly Geneal Public Unitilities), and Fauske and Associates, LLC (FAI), attended this event, which featured presenations and panel discussions regarding first-hand knowledge of TMI-2 cleanup activities. Japan sent nearly 35 representatives from METI, Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), TEPCO Holdings, International Research Institute for Nuclear Decommissioning (IRID), JAEA, the Nuclear Regulatory Authority (NRA), and Japan's embassy in Washington, DC. The workshop focused on topics such as an effective safety policy, adequate regulations, and appropriate public communitication in a highly charged political environment after an accident. There was great interest among Japanese participants in the process used to design safe shipping and storage casks for the damaged fuel and what was required to address the possibility of re-criticality accidents. Experts also discussed topics such as the use of robotics versus manual clean up tools and balancing of cost versus safety considerations. Appendix E of this report provides additional information from this meeting.

2.2.2.2 OECD/NEA

The OECD/NEA has been proactive in sponsoring Committee for Nuclear Safety Installations (CSNI) and Nuclear Science Committee (NSC) efforts to ensure that the international community is aware of safety insights from the events at Fukushima.[95,96] Current and proposed new activities of special interest to the U.S. DOE RST pathway are highlighted in this section.

BSAF Analyses

An ongoing analysis activity is the OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) project.[97] The project, which is hosted by the Institue of Applied Energy (IAE) in collaboration with other Japanese organizations such as JAEA, NRA, and CRIEPI, 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 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 informs 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 in the reactor pressure vessels and primary containment vessels for the first six days of the accident, was completed in 2015.[98] A total of 17 organizations from 8 countries participated in this phase of BSAF. The first phase concluded with a comparison of bounding debris endstates and compositions predicted by organizations participating in this effort. Although results differ, these bounding estimates inform D&D activities by Japan. Participants also identified important uncertainties associated with BWR accident progression phenomena, such as the impact of larger zirconium inventories and the effects of boride and carbide inclusions in relocated BWR materials. Phase 2 of the BSAF project, which started in April 2015, is aimed at characterizing release and transport of fission products outside the PCVs and lengthening the time-span for analyses of the accident events to around three weeks. 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[99] 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, which includes organizations from 11 countries, is anticipated to be completed in 2018. In support of these BSAF projects, the IAE has developed a website[39] containing useful information to support Fuksuhima Daicihi accident analysis and Decommissioning activities.

The U.S. DOE and U.S. 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 examination activities inform ongoing BSAF and U.S. DOE funded analyses, and analyses results may lead to revisions in U.S. information needs.

SAREF Research Opportunities from Fukushima

Another noteworthy effort is underway by the NEA's senior expert group (SEG) on SAfety REsearch opportunities post-Fukushima (SAREF). Created in 2013, the objective of this CSNI 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. Organizations from eleven countries are participating in this activity. The NRA of Japan chairs the group. The work scope includes identifying research opportunities that use information from Daiichi that will provide additional safety knowledge of common interest to the member countries. In their report [100], the SEG identified 16 specific topics of interest in three broad areas; namely i) severe accident progression, ii) system, structure, and component (SSC) performance, and iii) recovery phase.

The SEG prioritized information from Daiichi with respect to its importance for making decisions regarding safe execuition of decommisionning activities and with respect to reactor safety. In addition, the SEG considered the 'ease' of obtaining information without adversely impacting decommissioning activities (e.g., visual examinations that contribute to understanding RCIC system or safety relief valve performance). The SEG recommended that focus is placed on the following areas with high safety and decommissioning interest:

Severe Accident Progression

- In-vessel Phenomena
- Ex-vessel Phenomena
- Containment Failure and Venting
- FP Behaviour and Source Term

System, Structure, and Component Performance and Conditions

- Mission Time and System Survivability

Recovery Phase

- Long-term Accident Management and Recovery

The SEG also determined that it would be beneficial to undertake some near-term activities to provide additional information for planning long-term activities. Two near-term proposals are proposed [101]:

- *Preparatory Studies for Fuel Debris Analysis (PreADES)* – The PreADES project aims to summarize knowledge and expertise regarding fuel debris characteristics generated from a severe accident and to optimize methodologies for asessing fuel debris sampling and retrieval. The project will include the following activites:

- Share and update information and expertise on fuel debris generated from severe nuclear accidents;
- Jointly compile and evaluate selected topical issues based on updated data and information, and identify research gaps and priorities; and
- Prepare future collaborative R&D plans on analysis of 1F fuel debris.

- *Thermodynamic Characterization of Fuel Debris and Fission Products based on Scenario Analysis for Severe Accident Progression at Fukushima-Daiichi NPS (TCOFF)* - The TCOEFF project aims to improve the thermodynamic database for fuel debris and fission products by considering available analytical and experimental studies. The improved characterization of fuel debris and fission product behavior will provide valuable input to the decommissioning process already in progress and support the prioritization of appropriate future sampling in the RPV/PCV. Furthermore, if unpredicted materials are discovered within the RPV/PCV, possible scenarios leading to the formation of such materials will be formulated. Project results are also expected to improve the thermodynamic databases for fuel and structural materials used to evaluate fuel assembly degradation and fission product transport.

The U.S. DOE and U.S. NRC participate in the current SAREF and BSAF projects and plan to participate in the proposed PreADES project. It is envisioned that there will be strong interactions between the PreADES, BSAF, and TCOFF projects. It is important for U.S. DOE efforts to be cognizant of and contribute to appropriate international efforts. Ultimately, results from these efforts may lead to the establishment of a potential international examination effort in which the U.S. 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 [102, 103] originate within the Severe Accident Research NETwork of Excellence (SARNET) [NUGENIA Technical Area 2] which has the objectives of:

- Improving knowledge on severe accidents 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 U.S. (e.g., the NRC). Of interest to U.S. DOE RST efforts are results from SARNET efforts to prioritize research programs. As discussed within [104], 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 U.S. DOE strategy, SARNET uses this ranking to ‘harmonize’ and ‘reorient’ existing R&D programs and justify new research topics. Through the NRC, the U.S. 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 EU member states 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 [105] efforts to complete follow-on activities related to "stress tests" on EU nuclear power plants are of interest to U.S. 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 and identify mitigating measures and actions. The operators' reports were first reviewed by the national nuclear regulators. Then, the regulatory agency from each country prepared National Action Plans (NAcPs) that document actions and measures to improve the safety of nuclear power plants. These NAcPs were peer reviewed by an ENSREG panel. As each country implements these actions and measures, the NAcPs are updated and peer reviewed by ENSREG. Observers from the U.S. NRC and other countries participate in these reviews.

2.2.2.4 Summary

In summary, a range of post-Fukushima activities are underway within the U.S., but none duplicate the effort documented in this report. Many international efforts have synergistic objectives to those being performed within the U.S. 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 its benefit.

2.3 Decontamination & Decommissioning Activities

Examination efforts by TEPCO Holdings are primarily focused on obtaining data required to support D&D efforts. However, 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.[106] Furthermore, international participation may be beneficial to Japan because of expertise related to severe accident progression and, in the case of the U.S., because of expertise gained from prior TMI-2 D&D efforts. Although financial constraints and national needs dictate that the primary responsibility of TEPCO Holdings is to obtain information required to support D&D activities at Daiichi, the examination information is being used by the international community to enhance safety (e.g., data for validating severe accident models, source term models, etc.)

Hence, it is important that the U.S. Forensic Effort understands 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 and inputs for key D&D decisions are outlined to emphasize areas where this U.S. effort could use inspection information to benefit on-going D&D efforts in Japan and meet U.S. objectives to enhance reactor safety.

2.3.1 Organization

In 2014, the government of Japan reorganized organizations involved in D&D efforts at Daiichi.[106, 107,108] Major organizations involved in this new structure are shown in Figure 3. The Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) was established to strengthen decommissioning strategies. NDF has the responsibility to develop and issue annual updates to the "Technical Strategic Plan for Decommissioning of the Fukushima Daiichi Plan for Decommissioning of the Fukuhisma Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings" (e.g., the Strategic Plan), which contributes to implementation (and future revisions) of the "Mid- and-Long-term Roadmap towards the Decommissioning of TEPCO's Fukusiham Daiichi Nucler Power Station" (e.g., the "Roadmap") issued by the Government of Japan and for ensuring appropriate and steady conduct of decommissioning at Daiichi. In May 2017, the Nuclear Damage Compensation and Decommissioning Facilitation Corporation Act was amended to require TEPCO to deposit decommissioning funds with the NDF. Under the new system, as the organization responsible for managing and supervising TEPCO decommissioning activities, NDF will: 1) manage the funds for decommissioning in an appropriate manner; 2) manage the implementing structure of the decommissioning process in an appropriate manner; and 3) manage the decommissioning work under the decommissioning fund reserve system.[106] 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 Holdings for developing an "Action Plan" for completing on-site operational activities related to D&D, the International Research Institute for Nuclear Decommissioning (IRID) for overseeing R&D to support technology development for fuel debris retrieval, and JAEA for overseeing required R&D to support decommissioning technologies. 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. To promote collaboration among relevant institutions within Japan, NDF established the Decommission R&D Partnership Council. In addition, JAEA established the Collaborative Laboratories for Advanced Decommissioning Science (CLADS) as a global research and development organization, and JAEA began operation of the Naraha Remote Technology Development Center for development and verification testing of remote operation equipment for D&D equipment.

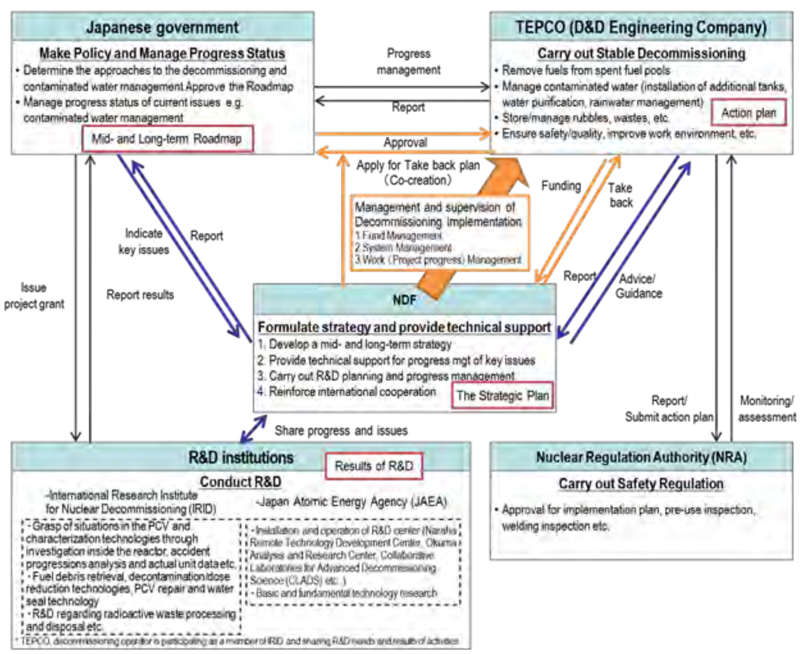


Figure 3. Roles and responsibilities of organizations involved in decommissioning Daiichi. (Courtesy of NDF [106])

2.3.2 Strategic Plan

The 2017 Strategic Plan [106] issued by NDF provides a strong technical basis for implementing the government of Japan's mid-and-long-term roadmap. In addition, it provides strategic proposals for "deciding fuel debris retrieval policies for each unit" and compiles "basic concepts of processing and disposal for solid waste".

There is a high level of uncertainty associated with the characteristics and location of radioactivie materials and the damage to equipment and buildings at Daiichi. High radiation levels preclude direct access to the affected units at Daiichi, making it difficult to reduce this uncertainty. Building upon concepts in earlier strategic plans issued by Japan, the 2017 Strategic Plan emphasizes the need for risk reduction by applying five guiding principles:

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

Activities to reduce the risk from environmental impacts as well as risk to D&D workers at Daiichi are considered. As documented in the strategic plan, the risk levels at Daiichi are based on evaluations of the 'hazard potential' (or consequence, which considers the total amount and form of radioactive materials from various sources) and 'safety management' (or likelihood of release, which considers factors such as the integrity of facilities and containment functions). Figure 4 shows major risk sources and levels based on information available in March 2017. Areas for each risk source represent uncertainty associated with available data. Results from such analyses are used to group and prioritize D&D activities based on the potential for risk reduction.

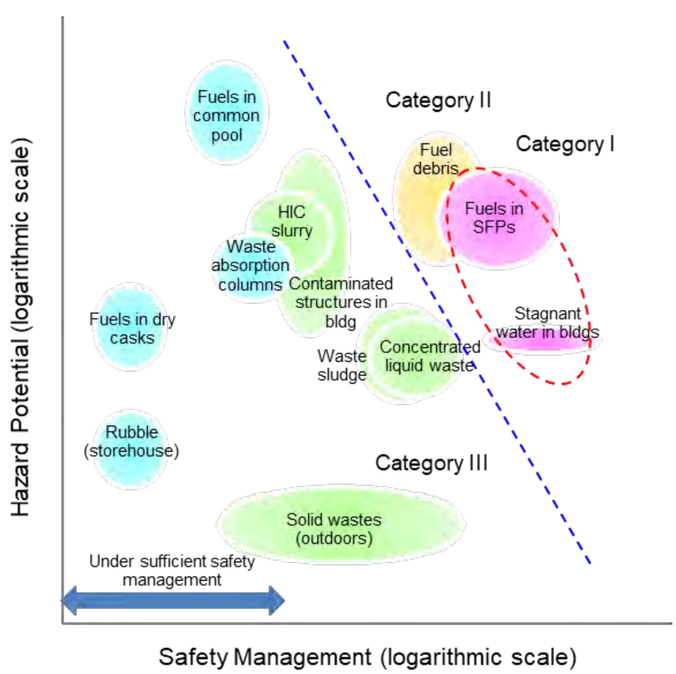


Figure 4. Risk associated with various D&D sources at Daiichi. (Courtesy of NDF [106])

As outlined in the 2017 strategic plan,[106] D&D activities are grouped into three categories based on risk reduction:

- Category I Risk sources to be addressed as soon as possible (the fuel in SFPs and stagnant water shown in the upper right-hand corner in Figure 4);
- Category II Risk sources to be addressed in a safe, effective and careful manner with elaborated preparation and technologies, and bring to a more stable state (the fuel debris shown in Figure 4);
- Cateogry II Risk sources to be systematically addressed to achieve a more stablestate (these sources are focused on reducing the risk for the solid wastes, concentrated liquid wastes, and other wastes shown in blue and green in Figure 4).

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 public or plant workers. D&D activities must be monitored to alleviate concerns about maintaining containment, maintaining cooling, criticality control, PCV and RB structural integrity, occupational radiation exposure, radiation releases, increasing hydrogen concentrations, and non-nuclear industrial accidents.

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 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 initially developed 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.[108] Further revisions will take place as additional information is obtained about the actual conditions within each unit. Periodic updates on D&D progress are provided in a format consistent with activities outlined in the Roadmap (e.g., see [108]).

The roadmap provides U.S. experts general insights about the schedule and types of activities completed and underway by TEPCO Holdings. In addition, results from these activities are posted on TEPCO's website and discussed in periodic updates provided by TEPCO Holdings. 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).

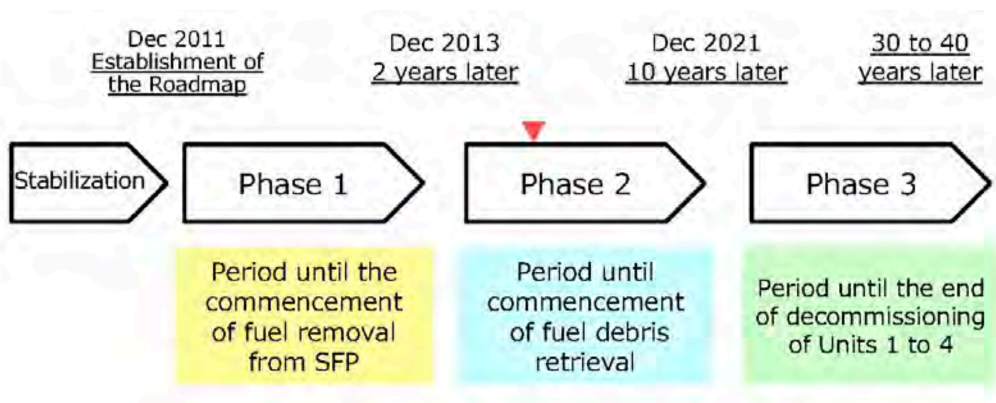


Figure 5. Roadmap phase definition; red pointer designates current status. (Courtesy of NDF [107])

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 other 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 108 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 an overview of the remaining tasks for completing Phase 2 and 3 activities (more detailed figures for selected activities from the 2016 Strategic Plan [107] 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 U.S. Forensics Effort to provide input to these HP evaluations.

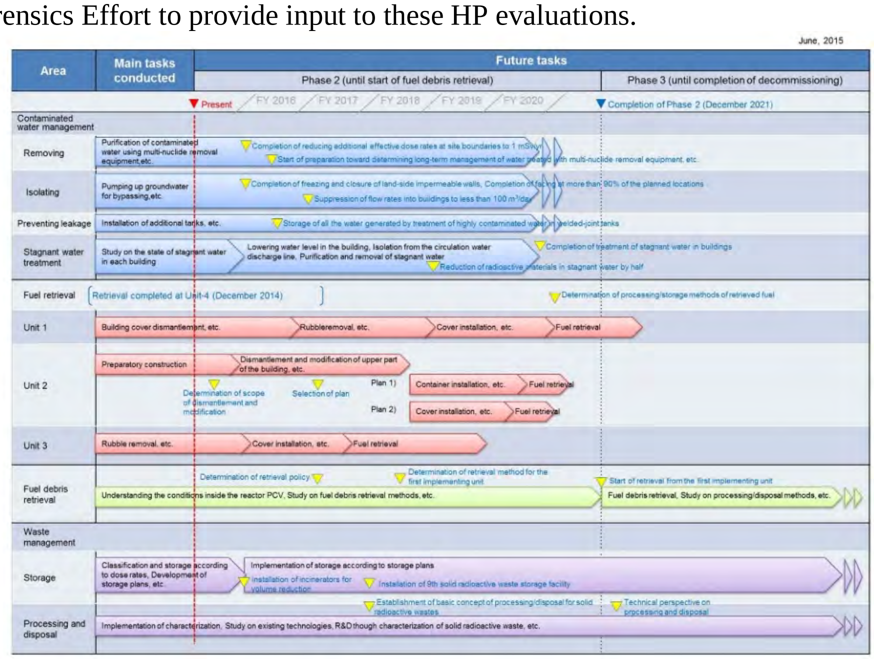


Figure 6. Summary schedule showing remaining roadmap tasks and milestones. (Courtesy of Inter-Ministerial Council for Contaminated Water and Decommissioning Issues [108])

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 3). 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 3. Phase II critical milestones and timing (based on information in [108])

Area	Description	Timing[g]
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 FY2017
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

[g] In Japan, the Fiscal Year (FY) runs from April 1 through March 31.

2.3.4.1 *Fuel Debris Retrieval Methods*

As an example of the potential benefits to Japan from the U.S. forensics effort (and of the information obtained by Japan to the U.S.), it is of interest to consider activities required to complete Milestone 3, "Fuel Debris Retrieval" in a manner that considers risk reduction (as advocated by the Strategic Plan). As described in Section 2.3.3, there is a hold point (HP) for selecting the fuel retrival method at each unit.

Figure 7 shows the three methods on which evaluations are currently focussed. Due to difficulties associated with debris cooling and radiation levels, dry methods are not being considered for evaluations. Several retrieval access methods are under consideration: full and some partial water submersion levels utilizing top entry may require removal of the remnants of major structures, such as the steam dryer, the core plate, the core shroud, etc. The integrity of these structures during the removal process must 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 water level also requires repairs to stop leakage from the PCV. In the case of partial submersion water levels, 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). The 2017 Stragetic Plan [106] indicates that the debris retrieval process is being developed in a flexible manner using a 'step by step' approach based on information gained from the affected reactors. Currently, [106] indicates that efforts will focus on a side-access method with partial submersion that would target any debris relocated to the bottom of the PCV. These efforts will include feasibility evaluations to address issues, such as determining an appropriate water height (to reduce dose levels) and demonstrating the ability to control this water level.

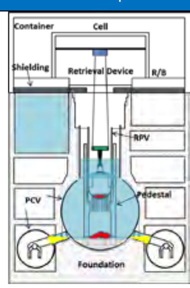
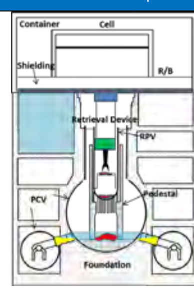
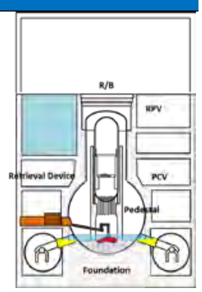
Methods		Submersion-Top Access	Partial Submersion-Top Access	Partial Submersion-Side Acces
Concept Diagram (Image) *Red: potential location of fuel debris *Blue: potential water level *Yellow: Case of bent pipe water sealing				
Major Technical Requirements				
Ensuring Containment Capability	Liquid Phase	• Hard to ensure water seal capability of resisting hydrostatic pressure when being submerged • Hard to ensure capability to remotely fix penetration holes for upper PCV with lots of holes • Emergency water leak prevention measure is required as a large amount of water is supposed to be kept.	• Technical difficulty is slightly lower as hydrostatic pressure is lower than that of submersion case. • Penetration holes on upper PCV to be fixed are limited. • It is possible to prevent water leakage even on emergencies depending on water level settings.	• Technical difficulty is slightly lower as hydrostatic pres that of submersion case. • Penetration holes on upper PCV to be fixed are limitec • It is possible to prevent water leakage even on emer on water level settings.
	Gas Phase	• Although air conditioning system with capability of maintaining negative pressure is required, small scale equipment may be good enough.	• Air conditioning system with capability of maintaining negative pressure for containing alpha-emitting nuclides is necessary. The scale of the equipment will be large, but it can be feasible.	• Air conditioning system with capability of maintaining for containing alpha-emitting nuclides is necessary. equipment will be large, but it can be feasible.
Maintaining Cooling Capability		• Feasible	• Feasible	• Feasible
Criticality Management		• Preventing criticality when reactor core is covered with water is an issue.	• There is a low probability of re-criticality as reactor core will not be covered with water.	• There is a low probability of re-criticality as reactor core with water.
Structural Soundness / Seismic Resistant Features of PCV and R/B		• Although the total weight of coolant in PCV and fuel debris retrieval equipment to be installed at upper R/B increases, required seismic margin will be ensured for major components.	• Although the total weight of fuel debris retrieval equipment to be installed at upper R/B increases, required seismic margin will be ensured for major components.	• Better seismic margin will be ensured as fuel debris r will be installed on the first floor.
Reducing Occupational Radiation Exposure		• Occupational radiation exposure would be of several times of the past annual total exposure when sealing upper PCV as there are lots of penetration holes on the upper PCV.	• Occupational radiation exposure would be less than the past annual total exposure when sealing lower PCV.	• Occupational radiation exposure would be less than th exposure when sealing lower PCV.
Establishing Access Route	Inner RPV	• Scale of work concerning retrieval of fuel debris located in RPV could be significant as inner structures of reactor must be removed.	• Scale of work concerning retrieval of fuel debris located in RPVcould be significant as inner structures of reactor must be removed.	• Building an access route to fuel debris located in RPV is
	PCV Bottom	• Scale of work concerning retrieval of fuel debris located at the bottom of PCV could be more significant than that of side-access method as it is required to bore the bottom of RPV.	• Scale of work concerning retrieval of fuel debris located at the bottom of PCV could be more significant than that of side-access method as it is required tobore the bottom of RPV.	• Scale of work concerning retrieval of fuel debris locat PCV could be less significant than that of top-access m
Conclusions		• Development of technologies for remotely fixing penetration holes for water sealing is difficult. • Total occupational exposure concerning repair work could be enormous.	• It is necessary to continue development of technology for maintaining negative pressure in order to contain alpha-emitting nuclides. • Both top-access and side-access would be required.	• It is necessary to continue development of technolo negative pressure in order to contain alpha-emitting nuc • IBoth side-access and top-access would be required.

Figure 7. Debris retrieval methods under evaluation. (Courtesy of NDF [106])

2.3.4.2 Debris End-state Location and Radionuclide Distribution

As shown in Figure 8, a logic tree has been developed that shows activities used to understand the location, amount, and properties of fuel debris and fission product (FP) distribution. Final selection of the feasibility of a debris retrival method will consider examination information, such as i) plant data, ii) investigations using robots within the PCVs (photos, dose surveys, temperatures), and iii) muon tomography; predictions by severe accident analysis codes; and knowledge obtained from past accidents, and experimental investigations.

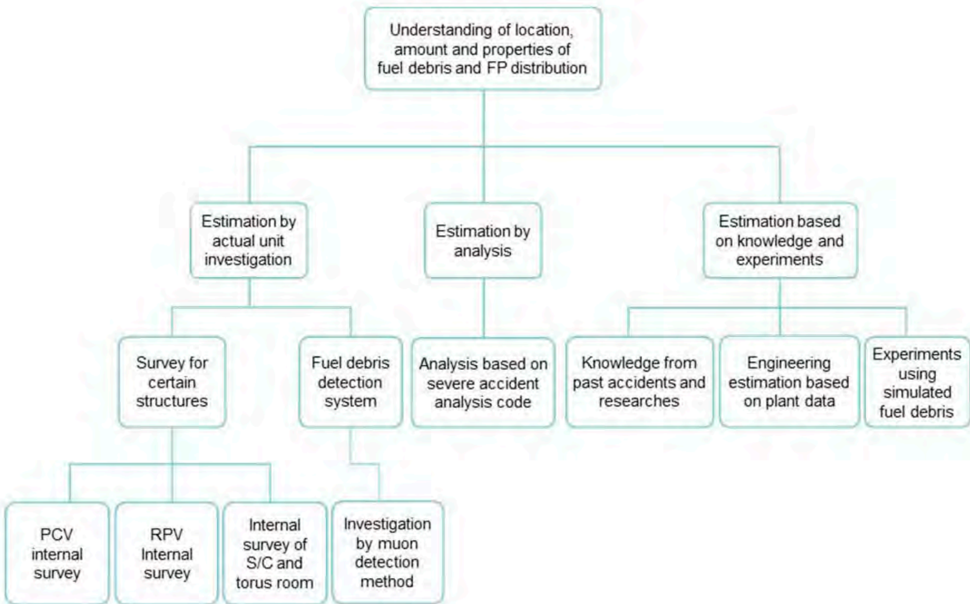
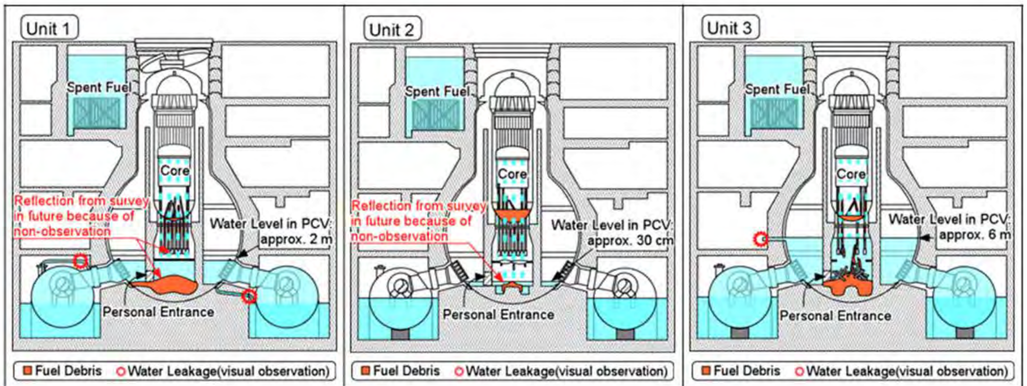


Figure 8. Logic tree identifying activitIes to understand location, amount, and properties of fuel debris and fission product distribution. (Courtesy of NDF [107])

Using available plant instrumentation data, examination information (radiation surveys, images), and results from analyses performed using systems analysis codes (SAMPSON and MAAP), assessments related to debris end-state are summarized below in Figure 9. As shown in this figure, there is considerable uncertainty in such estimates, especially considering 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, and results are updated as additional information becomes available.



Core Region	• Little fuel remains.	• Little fuel remains. • (Stub-shaped fuels might exist in peripheral region.)	• Little fuel remains.
RPV Lower Head	• A small amount of fuel debris is present. • A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing.	• Large amount of fuel debris is present. • A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing.	• Fuel debris remains on the RPV lower head partly. • A small amount of fuel debris is present in the inside and on the outer surface of the CRD housing.
Pedestal Inside	• Most of fuel debris is present.	• A small amount of fuel debris is present.	• Amount of fuel debris in Unit 3 is more than that in Unit 2.
Pedestal Outside	• Fuel debris may have spread on the pedestal outside through the personal entrance.	• The possibility of fuel debris spreading on the pedestal outside through the personal entrance is low.	• Fuel debris may have spread on the pedestal outside through the personal entrance.

* Based on the document provided by IRID and internal survey performed in 2017.

Figure 9. Status of 1F1 through 1F3 status including fuel debris location estimates. (Courtesy of NDF [106])

Timely participation by U.S. experts in evaluating inspection information and results from severe accident analyses provide Japan an independent assessment for selecting retrieval options. U.S. expert opinion may be of special benefit because U.S. researchers, who developed the models in severe accident analysis codes, are aware of model limitations and effects on subsequent source term assessments. Likewise, some U.S. experts were involved in TMI-2 post accident evaluations and relevant testing.

2.4 Summary

As part of their D&D activities, TEPCO Holdings has been and will continue obtaining information of interest to the international community. The government of Japan recognizes that information collected from these reactors is important to Japan and international organizations. Financial constraints and national needs dictate that TEPCO Holdings efforts are primarily focused on obtaining data required to support D&D efforts, but these data are also being used by the international community to enhance safety (e.g., data for validating severe accident models, source term models, etc.).

The DOE has established the U.S. Forensics Effort to work with TEPCO Holdings to learn what information is being obtained and to communicate this information to cognizant U.S. experts that could use this information to enhance safety of the commercial fleet. Appendix C contains lists of consensus information needs identified by U.S. experts. As new inspection information is obtained, U.S. 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. Sections 3 through 6 of this report summarize results from this U.S. 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 seismic events, hydrogen explosions, radiation exposure, and high temperatures can provide insights related to the accident progressions. As observed in Reference 6, component examinations in the TMI-2 containment provided critical evidence of peak temperatures and pressures when instrumentation data were inconsistent.

This section summarizes Fukushima Daiichi 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 Holdings. 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). Note that additional information regarding component and system performance may also be found in Sections 4 through 6.

3.1 Key Questions for Reactor Safety and D&D

Available information was evaluated by U.S. 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?
- Can insights regarding component and structure performance be used to enhance 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 (e.g., containment venting, water addition, and combustible gas mitigation).

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

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 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 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.

3.2 Summary of Information

TEPCO Holdings 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 Holdings has published a large amount of data on its publicly accessible website.[9] In particular, reports documenting unconfirmed and unresolved issues [34 through 38] have received special attention in this forensics effort to evaluate equipment performance.

As discussed in Section 1.3.1, the U.S. Forensics Expert Panel identified information needs during FY2015 that could be obtained from these examinations. An updated list of information needs is included in Appendix C of this report. The information needs address knowledge gaps in severe accident phenomena [55] 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 Holdings. Table 4 through Table 6 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 4. Area 1 information needs from the reactor building.

Item	What/How Obtained[i]	Use[j]	Data Available[k]
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	A
	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 FE-SEM, XRD, etc.	AE, AM, DD	NA
RB-15	Examinations of 1F1 RCW surge tank; water level and additional dose measurements.	AE, AM, DD	A

[i] See list of acronyms.

[j] 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).

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

Table 5. Area 1 information needs from the PCV

Item	What/How Obtained[l]	Use[m]	Data Available[n]
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).[o]	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).[p]	AE, AM, DD	A
	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[p]	AE, AM, DD	A
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 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 TIP tubes and SRV/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

[l] See list of acronyms.

[m] 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).

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

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

[p] Although some images have been obtained; images do not indicate if RPV failed. Photos from 1F2 investigations [See Section 5.2.3] indicate the presence of possible ex-vessel debris, but it has not yet been possible to extract samples for evaluating composition.

Table 6. Area 1 information needs from the RPV

Item	What/How Obtained[q]	Use[r]	Data Available[s]
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-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 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 7 summarizes several findings based on inspections performed by TEPCO Holdings. The table notes the observed status of various penetrations and equipment. In many instances, examination information has not yet been obtained for a unit's equipment. However, TEPCO Holdings 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 Holdings participate in the U.S. expert meetings and review draft versions of this report, there may be publicly available information that is not captured in this table.

[q] See list of acronyms.

[r] 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).

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

Table 7. Results from component and system examinations.[t]

Area	1F1	1F2	1F3
X-100B PCV penetration[u]	Possible melted shielding material [11] No damage observed on outside [109]	TBD	TBD
X-51 PCV penetration[v]	TBD	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [110, 111]	TBD
X-53 HPCI steam supply penetration (1F2/1F3)[w]	High dose rate measured [16]	No damage observed [112]	No damage observed [113]
X-6 PCV penetration (CRD hatch)	TBD	Melted material [114, 115]	No observed damage from inside [116]
Equipment hatch	TBD	TBD	Water puddle [117, 118] unknown source
Personnel hatch and nearby penetrations	No major damage observed [119]	TBD	TBD
HPCI pipe penetration;[x]	No damage observed, but high dose rates measured; traces of flow and white sediment observed [16, 119,120]		
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) [16,121]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [122]	
Wetwell (WW) vacuum breaker line	Leakage on expansion joint of one line (X-5E) [123]	TBD	TBD
DW/WW vent bellows	Water leakage attributed to vacuum line above [123]	No leakage observed [124]	TBD

[t] 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.

[u] X-100B is vacant for 1F1, allowing this penetration to be used for DW investigations.

[v] X-51 is an instrument pipe penetration for measuring differential pressure in 1F2/1F3. The penetration is joined to the Standby Liquic Cooling (SLC) pump injection line in the drywell. This penetration is designated as X-27 in 1F1.

[w] X-53 is vacant for 1F2 and 1F3, allowing these penetrations to be used for DW investigations.

[x] X-53 is the HPCI steam supply penetration and X-54 is the HPCI steam instrument pipe penetration for 1F1. X-11 is the HPCI steam supply penetration for 1F2 and 1F3.

Area	1F1	1F2	1F3
DW sand cushion drain pipe	Leakage [125]	No leakage observed [124]	TBD
SC water level	Almost full [126]	Consistent with torus room water level [126, 127]	Believed 'almost full' but not confirmed [126]
DW Water Level	~2.8 m[126]	~0.3 m[126]	~6.5 m[126]
Torus room	Partially flooded [128,129]	Partially flooded [130]	Partially flooded [130]
	Rusted handrails/equipment [11]	Non-rusted handrails/ equipment [11,131]	Non-rusted handrails/ equipment [11,132]
	TBD	Some room penetrations tested, no leakage observed [133]	TBD
MSIV room	Limited view obtained [16]	Water leakage cannot be observed [134]	Leakage in Line D near bellows [135]
DW head and shield plugs	Reactor well shield plug displaced [136]	Possible leakage [137]	Possible leakage [137]
RCIC or other low SC piping	TBD	Suspected leak location, not confirmed [11]	TBD

U.S. 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 3.2.3 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 Examinations Outside of Primary Containment

A wide range of equipment resides outside of containment that supports plant operation and accident response. In addition, the spent fuel pools and other key facilities (i.e., common spent fuel storage building) also reside onsite. Being outside of containment and generally accessable, several inspections have been performed.

An aerial inspection, intended for radiation inspections, of the shared vent stack (~111 m tall) for 1F1 and 1F2 was conducted in 2016.[138] An object, posited to be a beam, was found located inside the stack approximately 10-20 m below the top of the stack. No other abnormalities were noted. A previous inspection of the outside of the stack in 2013 noted rust and fractures in some of the structures.[139] Subsequent inspections and analyses provided confidence in the structure's capability to withstand additional seismic activity.[140, 141]

The main turbine of 1F4 was inspected.[142]. Examinations found evidence of cracking of the blades, which was attributed to normal operation, and contact traces between the moving and stationary blades and between the oil thrower bearing and the roter, which was attributed to the earthquake. The damage was reported as minimal.

Inspections of the standby gas treatment systems (SGTS) have also been performed. Five valves on the 1F3 SGTS were inspected. All of the inspected valves were found to be in their expected positions for loss of valve power.[143] Based on the investigation of the radiation dose around the rupture disk of 1F2, the rupture disk is not thought to have opened.[144]. The measured doses and locations of the 1F2 SGTS filters suggest there was backflow from 1F1 to 1F2 through the SGTSs and interconnected piping.[144] In addition, the measured dose and locations of the 1F4 SGTS filters suggest there was backflow from 1F3 to 1F4 through the SGTSs and interconnected piping.[145] Gravity dampers installed on the down stream side of the SGTS filters apparently did not prevent this backflow.

Of special interest in U.S. expert panel discussion was the endstate of the 1F1 drywell head shield plug, the reactor building closed cooling water system, the spent fuel pools and shared spent fuel storage building, and the 1F3 and 1F4 spent fuel pool gate.

3.2.1.1 1F1 Drywell Head Shield Plug Relocation

Thermal images taken in October of 2011 of the top of the 1F1 shield plugs, located on the refueling floor above the PCV head, indicated elevated temperatures.[146] Inspection of the 1F1 shield plugs in 2016 and 2017 [147,148] found the top, middle, and bottom layers to be displaced. In addition, higher doses were measured in the shield plug vicinity (see Section 4). Sections of the bottom layer are suspected to be resting on the PCV drywell head. The elevated temperatures and doses, as well as the observed shield plug displacement, support the possibility of drywell head leakage during the accident. This is generally consistent with simulations that predict drywell head flange leaked during the accidient. [49, 50] The following is an extended analysis of the potential causes for the displacement of the 1F1 shield plugs.

In [147], TEPCO Holdings reported that the drywell head shield plugs in 1F1 were lifted and physically displaced during the accident (see Figure 10). For this to occur, a substantial pressure difference would have likely developed between the refueling floor and the space below the shield plugs. The first question to address is what is the pressure force required to lift the plugs. From Figure 10, the combined shield plug mass is evaluated as ~520 MT and the cross-sectional area is ~110 m^2. The required ΔP to lift the plug is thus estimated as ~ 47 kPa (~7 psi). Possible methods by which this pressure difference could have developed include (but are not necessarily limited to) the following:

- A high rate of gas production inside the PCV that caused the drywell head to lift and leak gas into the gap between the head and shield plugs, causing the gap to pressurize, or
- The hydrogen deflagration on the refueling floor that created a temporary sub-atmospheric condition above the shield plugs that is characteristic of an explosive event.

These two scenarios are examined for plausibility below.

<u>High Rate of Gas Production Inside PCV</u>

One possible source for rapid gas production is steam generated from rapid debris cooling. Aside from normal boiling heat transfer to debris spread over the pedestal and drywell floors, possible transient mechanisms include: i) melt jet fragmentation during relocation through water on the drywell floor, and/or ii) rapid water ingress into the core debris in the sump coming from the sump drain line. Another possible source is non-condensable gas produced from MCCI, but calculations indicate that this source is not credible[y].

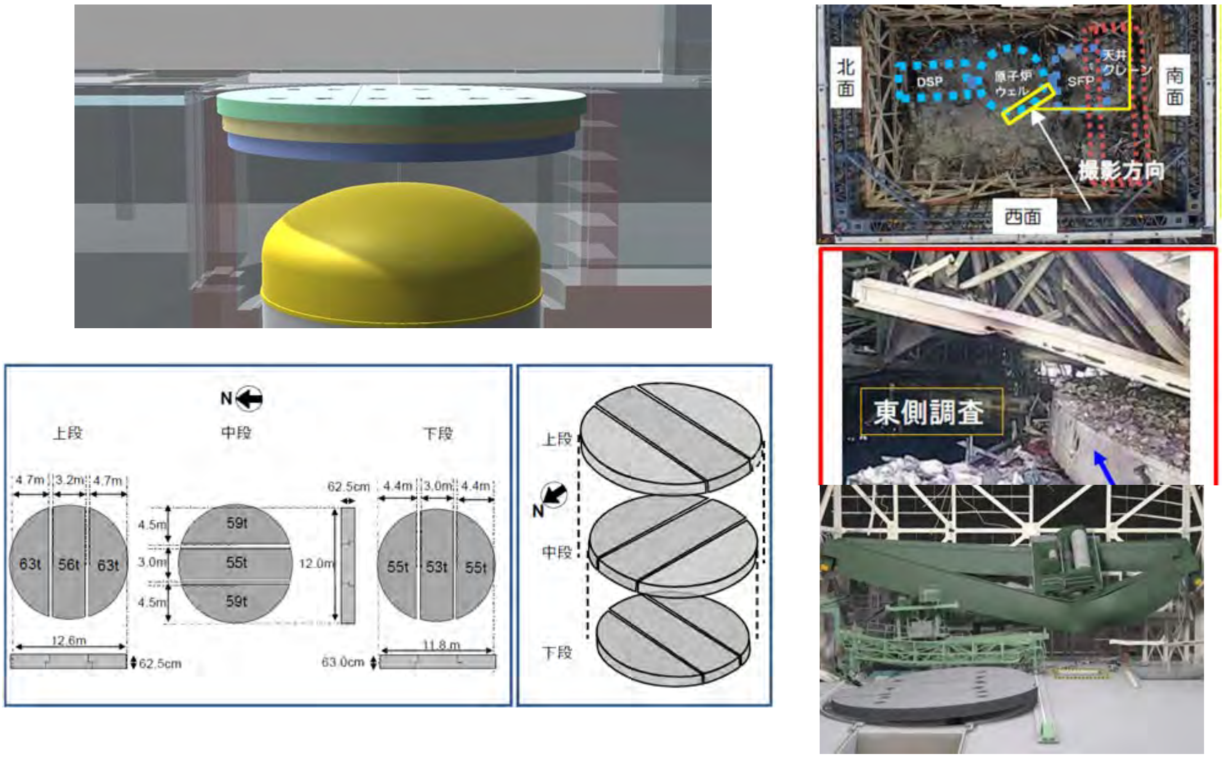

Figure 10. Physical characteristics, photographs, and 3D images based on photographs showing the displacement of the 1F1 drywell head shield plugs. (Courtesy of TEPCO Holdings [147, 149])

As noted above, the pressure difference required to lift the shield plugs is estimated as 47 kPa (7 psig). With a rough estimate of flow area, this ΔP could be used to evaluate mass flowrates for comparison to other data. For instance, if a crack width of ~ 6 mm (1/4") is assumed around the 11.8 m diameter of the shield plug, the flow area would thus be ~0.22 m^2. Treating the opening as an orifice with a loss coefficient (K) equal to 1 and assuming a pressure inside the gap between the drywell head and shield plugs of 100 + 47 = 147 kPa, then a steam flowrate of ~63 kg/sec (73 m^3/sec) would be needed to produce the required lift force.

With respect to evaluating potential steam sources that could produce this type of flowrate, MAAP and MELCOR predict 10 and 55 cm water depths on the drywell floor, respectively, at the time of vessel failure.[150] This corresponds to 1.3 and 1.8 m water depths in the sumps, respectively, at the time of RPV breach since the sumps are 1.2 m deep. The estimated steam production rate needed to lift the plugs (~ 63 kg/sec) is similar to that estimated by MELTSPREAD3 [151] to be produced by melt jet fragmentation and quenching for MAAP and MELCOR pours into the sumps; see Figure 11. However, the argument that jet fragmentation could have occurred in the sump(s) is not straightforward as the sumps are off-center from the RPV in 1F1; see Figure 12.

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[y] For example, at 5 MW decay heat level and 50 % up-down power split, then gas production from MCCI with siliceous concrete would be predominately H_2 at a mass flowrate of ~ 0.2 kg/s, or a volumetric flowrate of ~ 0.3 m^3/s. This is far less than the required gas flowrate to lift the plug, as discussed.

MAAP Pour **MELCOR Pour**

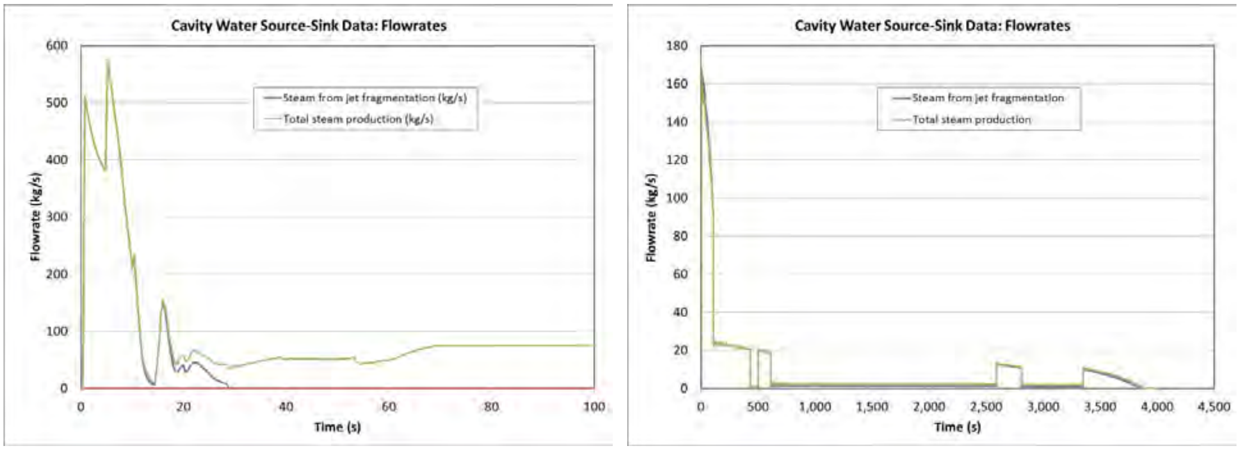


Figure 11. Predicted steam production from melt jet fragmentation during relocation into the sumps for MAAP and MELCOR pour conditions. (Courtesy of ANL [151])

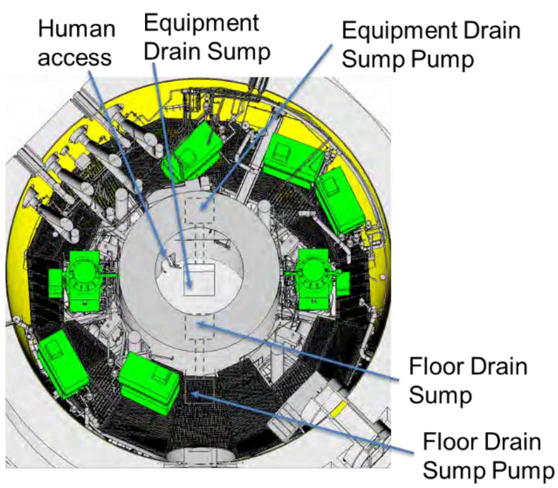


Figure 12. Plan view of 1F1 showing sump locations in pedestal and drywell regions. (Courtesy of TEPCO Holdings [19])

Another possible steam source is rapid cooling of debris in the sump(s) by water ingress through the sump drain piping system. As noted later in this section, the Reactor Building Closed Cooling Water System (RBCCW or RCW) closed loop piping system that provides cooling to the sumps was found to be heavily contaminated, likely indicating breach by corium during the accident. The sump drains in the pedestal and drywell are plumbed together by interconnected piping. Thus, if water was present on the drywell floor at the time of vessel failure as MAAP and MELCOR predict, this water could have flowed back into the pedestal under gravity through the sump drain lines, leading to rapid steam formation by cooling of debris in the sumps. This overall concept has in fact been envisioned as a method to increase the coolability of core melt by providing a pathway to passively inject water from below.[152] In this concept, water injection nozzles are cast into concrete basemat at a recessed location, with the nozzles plumbed to a water source at an elevated location in the plant to provide a static water head. In the event of an accident, the tops of the nozzles are intended to fail by ablation as the melt front arrives, thereby opening a pathway for water to flow into the material under gravity and increase the cooling rate. This approach has been investigated through a series of reactor material tests [152] that focused on examining the effect of water head on the debris cooling rate. The results indicate that this cooling mechanism is very effective, even for very low water heads (e.g., 1 m or less). In fact, it was found that the injection nozzles (2.5 cm or less in diameter or less in the tests) had to be orificed to limit the debris cooling rate to a level that would not disperse the melt due to the rapid steam generation from below. Thus, a second possible explanation for a steam source capable of lifting the drywell head and pressuring the gap between the head and the shielding plugs to the point that the plugs would be lifted is water ingress into relocated core debris through the sump drain lines. If this did occur, there could be some benefits for accident management depending upon melt pour conditions and the degree of spreading. Two examples include:

- Sump drains with access to a water source may provide a means for increasing the coolability of deep melt accumulations by bottom water injection that testing has shown to be very effective. Note that orificing is in fact beneficial to limit the debris cooling rate.
- For cases in which large debris accumulations form in the pedestal (as is predicted for MELCOR-type pours, for instance), then sump drains may provide a way for injecting possibly unused water on the drywell floor into the debris via the sump drain lines. The same would be true for water injected via the drywell sprays, or water from core injection that flows over core debris and out the pedestal door onto the drywell floor.

Vacuum Condition Created by Hydrogen Deflagration

As noted, a transient lift force would have been created on the shield plugs from the hydrogen deflagration on the 1F1 refueling floor due to the backdraft created by an event of this type. As concluded earlier, a net pressure difference of ~ 47 kPa (7 psig) would be able to lift the plugs given their geometry and weight. Thus, any pressure difference exceeding this value would be sufficient to cause the plugs to relocate. In fact, a detailed hydrogen deflagration analysis has been completed for 1F1 [153]. Results indicate that the absolute pressure on the refueling floor would have fallen to ~ 4 kPa for ~ 100 ms. This information allows a rough estimate of the relocation behavior of the plugs to be developed by solving Newton's second law; the results are shown in Table 8. The results indicate that the plugs likely 'jumped' because of the deflagration, but would not have been elevated to the point that the top plug (63 cm thick) would have cleared the surface level of the refueling floor. This rough evaluation that could be further refined if the actual pressure profile during the transient was known.

Table 8. Rough estimate of the relocation behavior of the shield plugs induced by the hydrogen deflagration in 1F1.

Time (ms)	Condition
0	Upper surface of shield plugs suddenly subjected to a vacuum condition on refueling floor. The pressure below the plugs is assumed to be atmospheric during the brief vacuum phase, resulting in an upwards pressure difference of ~ 100 kPa.
100	Vacuum condition ends and the pressure surrounding the plugs returns to atmospheric. Plugs have risen 6 cm at this point and are moving upwards at velocity of 1.2 m/sec.
220	Deaccelerating under the influence of gravity, the plugs reach a maximum displacement height of 13 cm.
380	Falling under the force of gravity, the plugs reseat.

3.2.1.2 Reactor Building Close Cooling Water System

The Reactor Building Closed Cooling Water System (RBCCW or RCW), supplies cooling water to a range of equipment in the reactor building, containment, and other locations. In containment, it supplies water to cool the sumps, drywell coolers, and other equipment. A surge tank is located on the 4th floor of the reactor building and has a vent to the reactor building airspace. A shaft for moving equipment is located next to the surge tank and spans multiple floors.

As noted in [35] and discussed in Section 4.2.1.1, elevated doses were measured in the 1F1 reactor building around the systems fed by the RCW. In particular, high dose rates were observed around the RCW heat exchangers. The possibility for failure of RCW piping in the drywell sumps and the flow of contaminated water into the RCW system was explored by TEPCO Holdings.[35] To summarize, a check valve would prevent backflow into the supply side of the sump coolers. However, if piping failure occurred, contaminated water could possibly flow into the downstream side of the coolers, through the RCW piping, and towards the vented surge tank. Upon lower containment pressures, the contaminated water could flow back towards the drywell following two paths. One path allows the water to flow through the RCW heat exchangers. This was presented as a possible cause for the high doses observed around the RCW heat exchangers and the RCW in general.

Additional observations are as follows. Based on illustrations of the system [35], the static head of water between the drywell sumps and the surge tank on the 4th floor is approximately 0.26 MPa (differential). Comparing this pressure to the available drywell and wetwell data [154] for 1F1, the containment pressure was sufficiently high for an extended period of time that was sufficient to push water into the RCW system (if failure of the sump cooler piping occurred and water was available), see Figure 13. The pressure was also sufficiently high to possibly push water out of the surge tank. Based on images (see Figure 14) from [155, 156], the surge tank appears intact; however, inspection of the tank and nearby piping has not been reported.

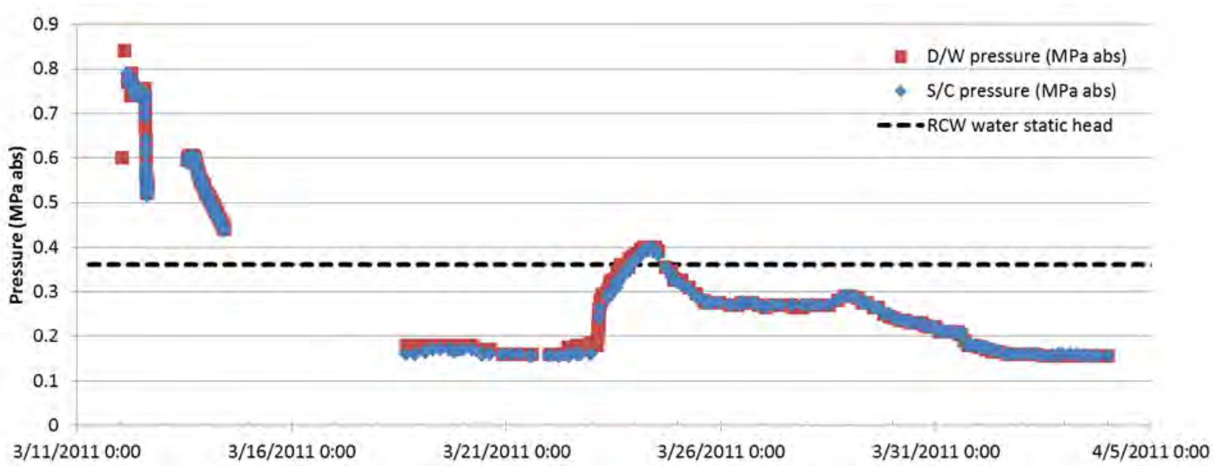


Figure 13. 1F1 containment pressure versus time. (Courtesy of TEPCO Holdings [154])

Figure 14. 1F1 RCW surge tank. (Courtesy of TEPCO Holdings [156])

Design details of the RCW at 1F1 are not readily available. In reviewing the design for a similar BWR4 in a Mark I containment at the Browns Ferry Nuclear Power Plant (Browns Ferry) [157], the pipe size for the drywell equipment drain sump heat exchanger is 3.81 cm (1.5 in). The rest of the RCW piping of relevance is much larger, in the 5-30.5 cm range (4-12 in). The volume of water in the system (relevant to RCW piping, surge tank, and heat exchangers of interest) is not readily known; however, it is estimated to be on the order of several cubic meters (i.e. on the order of 1000 gallons). The RCW piping is designed for 1.14 MPa-gauge (150 psig) at Browns Ferry which exceeds available values recorded for the 1F1 drywell pressure.

Available 1F1 drywell pressure data from 1:00 on March 12 until 10:00 on March 14 are greater than the RCW water static head (Figure 13). To reproduce the 1F1 drywell pressure during the March 12, 7:00-14:00 timeframe, previous simulations modeled a drywell head flange leak.[49, 50] The models require leakage areas (that vary with time) of approximately 4.4 cm^2 [49] and 5 cm^2 [50], which have an equivalent diameter of approximately 2.5 cm (1 in). Interestingly, this corresponds to the approximate size of the RCW piping for the drywell drain sump heat exchangers at Browns Ferry. Thus, a possible alternative to the 1F1 drywell head seal leakage path is through the RCW system during periods of high drywell pressure (i.e., greater than 0.36 MPa-abs.). However, as noted in Section 3.2.1.1, there is evidence the drywell head likely leaked during the events at 1F1. Substantially higher doses have not been measured near the surge tank or on the refueling floor above the surge tank. Thus, to date, the inspection evidence supports the possibility for drywell venting through the drywell head flange. While one explanation for the high doses observed by the RCW heat exchangers is due to contaminated water being pushed up into the system, there is currently a lack of evidence to support the possibility of ‘venting’ the drywell through the RCW piping and surge tank.

Besides a potential leakage path for the drywell, the RCW is a potential source of water. With variations in drywell pressure, water could either flow from the drywell to the RCW or the in opposite direction (assuming there is a failure in the RCW drywell sump heat exchanger piping). As discussed in Section 3.2.1.1, water from RCW has the potential to quench ex-vessel debris in and around the drywell sumps. The efficacy of this cooling method (water injection into the debris from below) is supported by previous testing as discussed in Section 3.2.1.1. It should be noted that previous simulations predict the relocation of approximately 140 tons of debris from the RPV.[150]

In summary, elevated doses are observed around portions of the RCW system in 1F1. Based on available pressure data and analyses, there is a potential that the RCW drywell sump cooler heat exchanger piping failed. Failure of the piping represents a potential leakage path from drywell and a source of cooling water for ex-vessel debris in the drywell. Additional inspections may provide insight as to the role of the RCW during the 1F1 accident progression.

3.2.1.3 Spent Fuel Pools and Shared Spent Fuel Storage Building

The removal of the spent fuel in the 1F1, 1F2, 1F3, and 1F4 spent fuel pools (SFP) is a key activity within the D&D roadmap, and efforts have been underway for several years. In addition to the SFPs of these units, defuelling logistics also involve the large common spent fuel pool, the dry cask storage building, and the newly constructed dry cask temporary storage facility. Insights about the integrity of the spent fuel and supporting structures and facilities during and after the accident are of interest to both TEPCO Holdings and the U.S. nuclear industry.

Six SFPs, the common SFP, and the dry cask storage building were all subjected to severe ground motions due to the earthquake. Early after the accident, no indication of leakage from the SFPs was reported, other than from possible sloshing.[158] To date, no leakage (other than from sloshing) has been reported. There were repeated inspections and analyses of the integrity of the 1F4 SFP and reactor building.[158, 159] The findings indicated the building and SFP had sufficient margin to withstand another large earthquake [158] and confirmed there was no long-term change in structural integrity [159]. The nine spent fuel casks, stored within the dry cask storage building, were found to have remained bolted in their original position after the earthquake(s).[158]

Underwater video inspections of the 1F4 SFP was undertaken in April and May of 2011, March 2012, as well as during defueling. There has been no indication of leakage from the SFP noted. Starting November 2013, the spent fuel was removed from 1F4 spent fuel and transferred to the common spent fuel pool. As noted by TEPCO Holdings, one assembly in the SFP was previously (i.e., before 2011) deformed, and two assemblies were known ‘leakers’.[160] As of December 2014, all spent and new fuel have been removed from the 1F4 spent fuel pool. Inspection of a new assembly, from the fuel pool, was conducted in August 2012. Some rubble and ‘rust’ were noted.[161, 162] Visual inspections of spent fuel assemblies were reported in April 2014. The inspection found there were no problems (i.e., corrosion, deformation, tie rods/plates, etc.) with respect to the soundness of the spent fuel for fuel removal operations [162,163]. Measured oxide layer thicknesses, multiple measurements for each assembly inspected, were within normal ranges for five assemblies with burnup ranging from 37.3-50.5 GWd/t.[164]

Numerous underwater video inspections of the 1F3 spent fuel pool have been undertaken (e.g., April 2012, September 2012, October 2012, February 2013, August 2015, and October 2015). There has been no indication of leakage from the SFP noted. In general, rubble was found on the top of the spent fuel racks. No large-scale deformation of the racks was noted. White deposits were observed on the racks.[164] A deformed control rod blade and a fallen control rod (from the storage hangers on the pool perhiphery) were found. [164] The handles of 4 fuel assemblies were noted to be bent, attributed to the fuel handling machine which was resting in this location [165]. The handles were bent on a couple of additional assemblies that were located under a 2.6-ton concrete hatch that had relocated into the pool.[164]

Underwater video inspections of fuel in the 1F1 and 1F2 pools have not been reported.

The dry cask storage building contained 9 loaded horizontally oriented casks, holding 408 total assemblies, at the time of the events in 2011.[158] This building was partially damaged and flooded during the tsunami.[166] One cask was transferred from the storage facility to the common spent fuel pool facility for inspections. Inspections in 2013 did not note any abnormalities with respect to the cask and structure appearance, fuel assemblies, seal integrity, gas leakage, etc.[167] The fuel from the 9 casks were transferred to the common SFP in 2013. Starting in 2013, fuel from the common spent fuel pool is being transferred to a newly constructed dry cask temporary storage facility onsite.[168]

In general, the observations to date provide confidence in the integrity of the spent fuel, SFPs, storage racks, and storage casks that experienced off-normal conditions (e.g., seismic, water purity, debris) during and after the accident. The responses of the buildings and SFPs provide points of comparison for domestic SFP seismic response assessments.[169,170]

3.2.1.4 Spent Fuel Pool Gate

At 1F3, one of two gates between the SFP and the reactor well was observed to be displaced (i.e., the displaced gate closest to the reactor well). Observations indicated the other gate is maintaining the seal between the spent fuel pool and reactor well. [171]

Visual observation of 1F4 gate did not identify any abnormalities.[172] TEPCO Holdings has postulated that leakage from the reactor well to the spent fuel pool occurred in the days following the accident as the water level in the SFP dropped. [143]

3.2.2 Containment Examinations

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

3.2.2.1 Leakage Locations

Examinations have informed 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 [123], DW sand cushion drain pipe leakage [125]; 1F2: melted material at X-6 PCV penetration [114, 115]; 1F3: main steam line (MSL) D leakage near MSIV [135]; and all three units: possible DW head flange leakage). Although no damage has been detected for many penetrations/lines, there are several 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 [9] 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 15, examinations within 1F1 revealed that a lead shield plate was missing. It is currently unknown whether the plate relocated due to melting or creep. For this lead plate to have melted, gas temperatures inside the drywell exceeded 328 °C/ 622 °F, the melting point for lead.

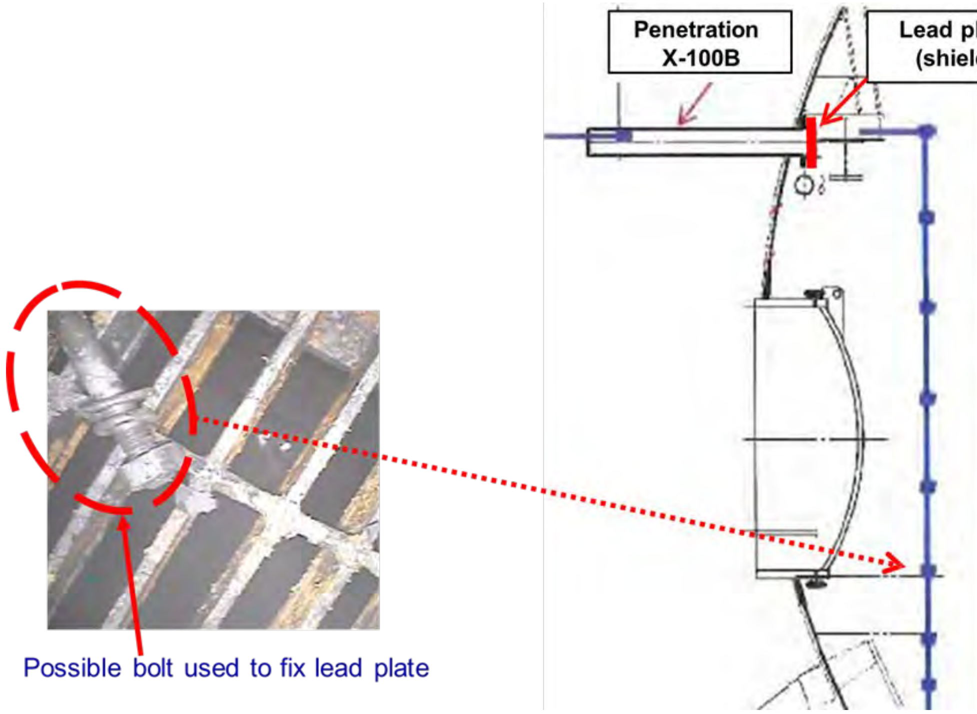


Figure 15. Visual examinations within X-100B penetrations in 1F1 PCV. (Courtesy of TEPCO Holdings [11])

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 16, 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, U.S. 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 [9] 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.

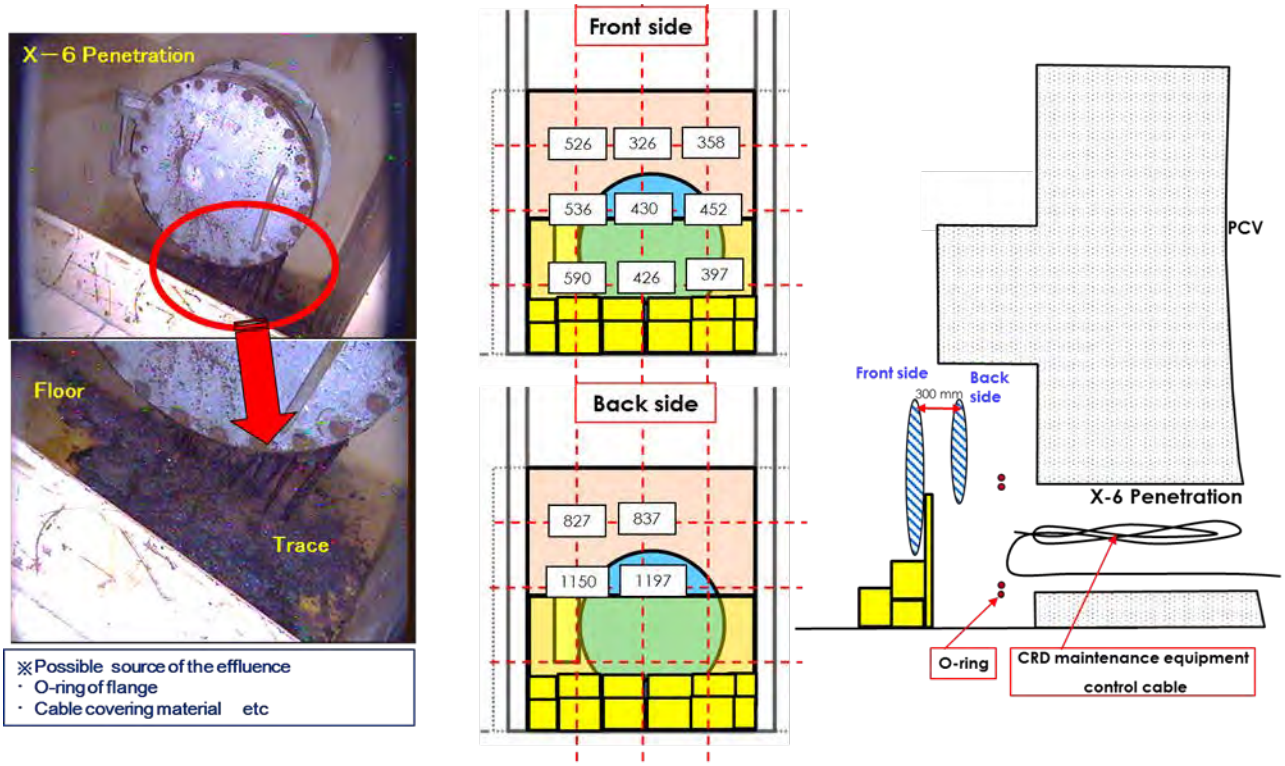


Figure 16. Photographs and radiation surveys (in mSv/hr) near 1F2 X-6 penetration (values measured in 13 locations). (Courtesy of TEPCO Holdings [114, 115])

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 17, steam appears to be escaping at locations near the drywell head, and higher dose rates were measured near this location. Both observations are consistent with a failure of the drywell head, perhaps due to drywell bolt expansion, strain, or seal degradation from high temperatures and pressures within the PCV. Plant data [9] 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.

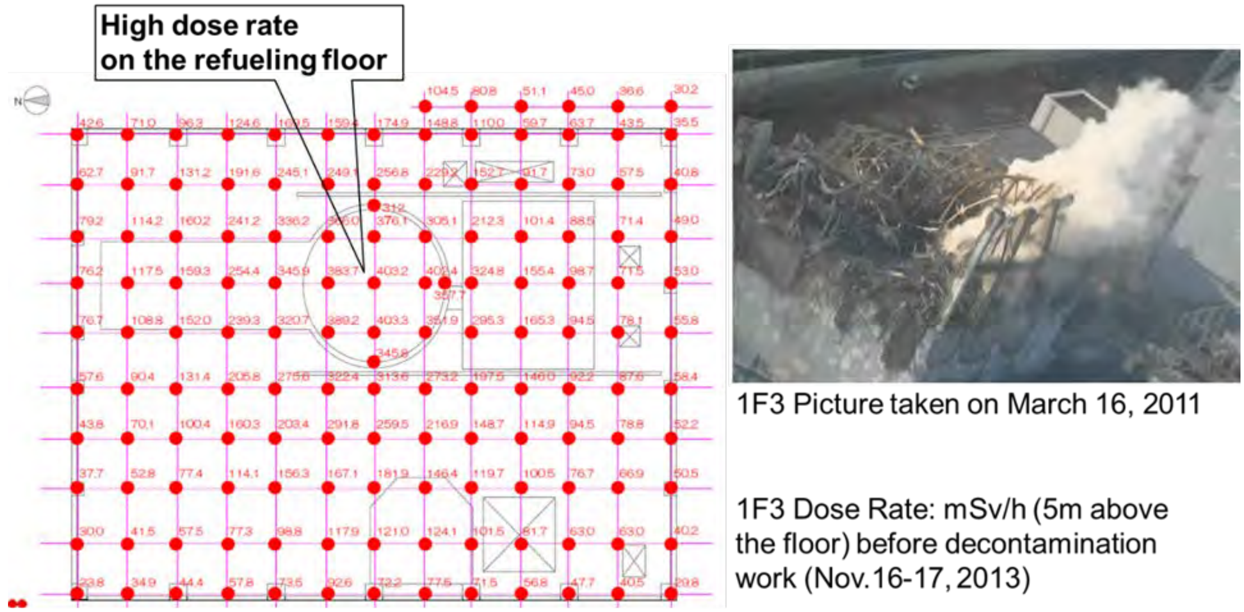


Figure 17. 1F3 radiation survey (value in mSv/hr measured on November 16-17, 2013 at 5 m above the floor at points shown on red grid) and photograph taken on March 16, 2011 (Courtesy of TEPCO Holdings [11,13])

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, including the impact of failure locations and sizes.

The potential for multiple penetrations to fail due to seal degradation is considered by industry in their proposed SAWA strategy. In the U.S., 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 the Nuclear Energy Institute (NEI) report, NEI-13-02, [85] 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 18 shows typical peak temperature and pressure information for BWR Mark I and II PCVs on a figure developed from information in the NEI 13-02 industry guidance for venting. The DW is assumed to begin penetration degradation at a temperature of 285°C/ 545 °F, based on engineering evaluations and testing information available in the literature. Black dotted lines in Figure 18 correspond to peak temperature information available from examinations at 1F1 and 1F2. 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.

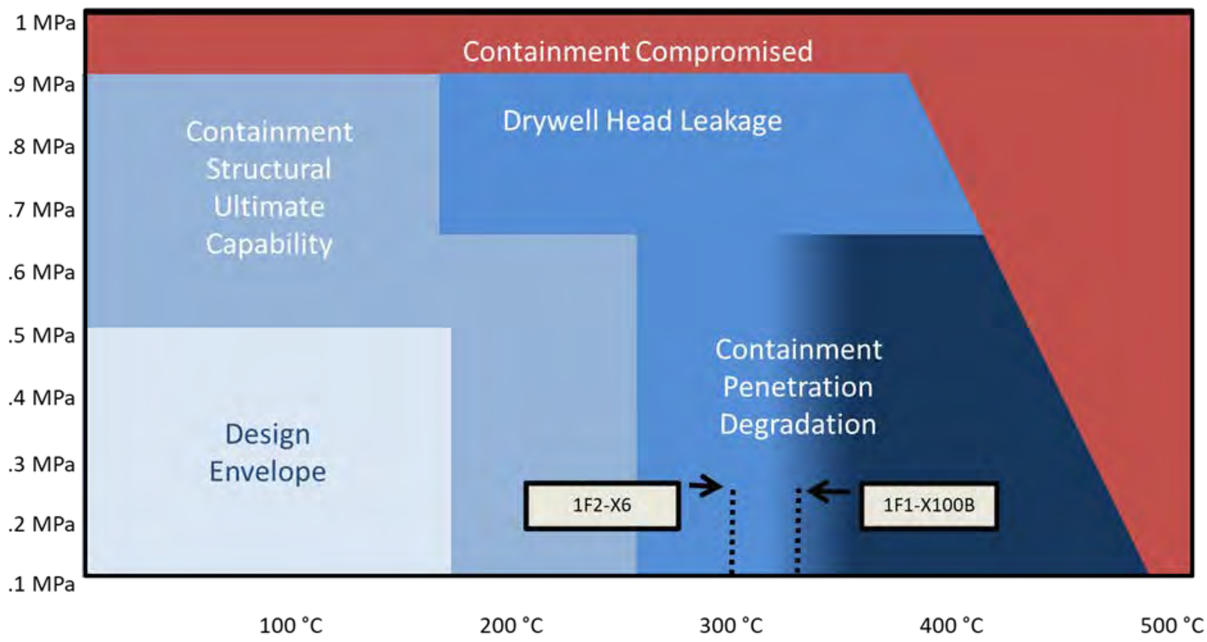


Figure 18. Containment pressure/temperature curve with available 1F1 and 1F2 information. (Graphic courtesy of Nuclear Energy Institute [85] as modified by Jensen Hughes)

3.2.2.2 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 19). 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.[173]

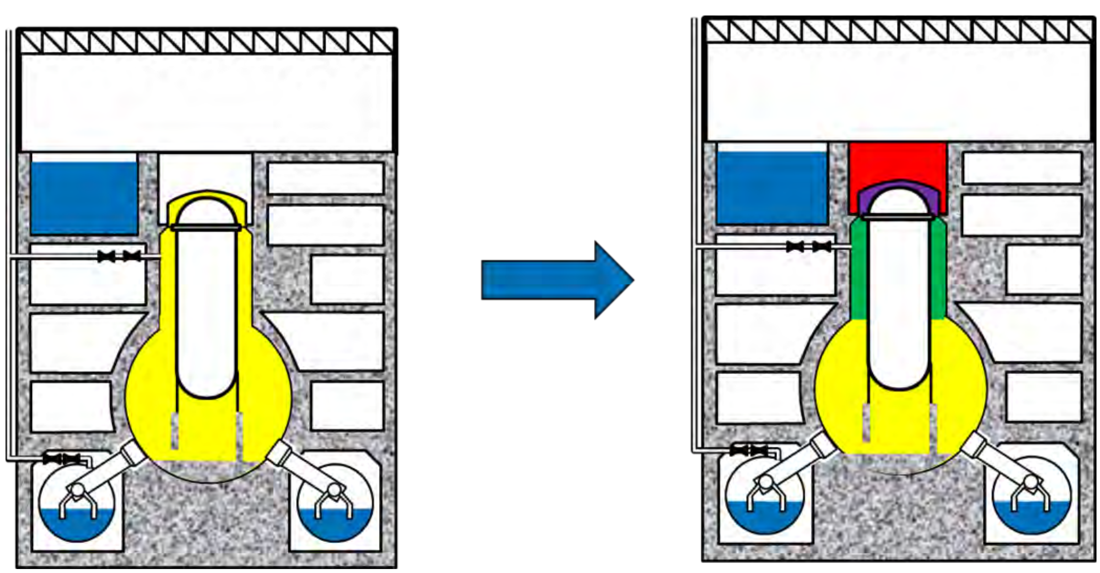


Figure 19. Improved MAAP nodalization. (Courtesy of Jensen Hughes, [23])

In 1F3, the PCV pressure increased more rapidly in the first 20 hours than systems level codes would generally predict.[49,50] Since 2012, there has been discussion and modeling of possible thermal stratification in the SC of 1F3.[17,38,174] 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 3.2.3.

Additional efforts are underway to assess the effects of SRV and RCIC operation on stratification within the containment.[175] 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 U.S. Evaluations with these codes are also useful to Japan as input for D&D Phase II assessments.

3.2.3 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.

Early photos and videos of structures below the bottom head of the 1F2 RPV indicated some of the cabling may still be intact.[11] More recent images (early 2017) show that some cable and structures are intact while other structures, such as a section of the maintenance platform, may have failed (See Section 5.2.3 for additional discussion of this information).

A leak was observed in line D of the 1F3 MSL near the MSIV.[135] 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 Holdings [176] 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., water levels prevent observations of leakage).

The cause for trip of the 1F3 RCIC system was reviewed by TEPCO Holdings.[38] 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. However, it is unclear how much of the injected water was successfully injected into the RPVs during and after the accident. TEPCO Holdings has reviewed the piping networks at the three units used to inject the water to identify possible bypass routes. [38] 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.

The possibility for a recriticality has been discussed and analyzed by multiple organizations, including Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) [177] and EPRI.[178] Discussion of this possibility arose as a possible explaination for radiation measurements near the 1F site main gate. As shown in Figure 20, measurements indicate several times when gamma dose rates increased and neutrons were detected.[37] It has been postulated that these observations could be explained by one or more BWR control rod blades melting prior to the reflooding of the reactor core by firewater injection.

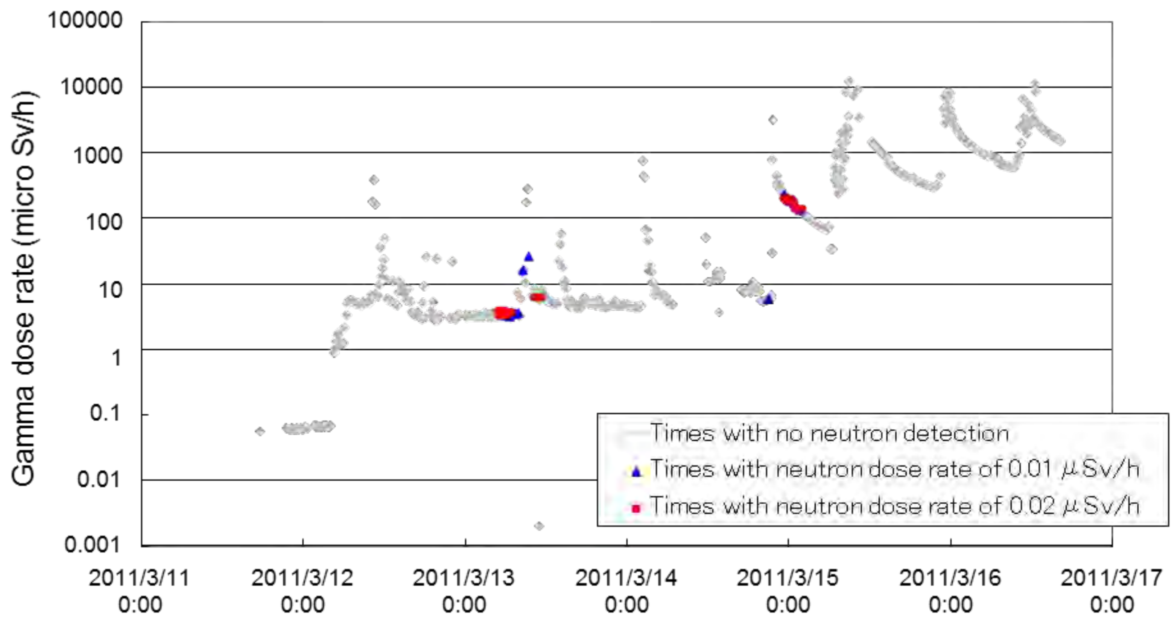


Figure 20. Gamma dose rate measurements at the main gain, indicating timings of coincident neutron dose rate. (Courtesy of TEPCO Holdings [37]*)*

Results from an EPRI-sponsored technical analysis [178] indicate that a recriticality could occur if conservative bounding conditions are assumed. Given the speed of accident progression predicted by MAAP, however, the EPRI study concluded that this was unlikely because the core would have degraded before such a recriticality could have occurred.

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, radiation effects, 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 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 7 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 because of elevated pressures, ultimately causing localized failure of the structure; and
    - weakening of containment boundary seals or structural elements because 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 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 7.

The expert panel concurred that information in Table 7 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 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 contrasts with 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 Holdings. This recommendation is not currently identified as an information need. As more information becomes available, however, the panel will identify specific places.

Many component status observations 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.

Area 1 Recommendation 5:

> The expert panel should consider an exploratory exercise in modeling the reference RPV water level reference leg in codes such as MELCOR and MAAP.

The RPV water level, data and measurement, continue to be of key importance in interpreting the events at Fukushima Daiichi. The potential heatup and boiling of the reference legs of the various RPV water level instrumtation has been discussed by various members of the expert panel (as well as other experts) since early after the accidents. The influences of the reference leg on the indicated RPV water level are important when comparing simulation results to actual field data and when extrapolating simulation results to what operators/response personel may observe. Current MELCOR and MAAP models for Fukushima Daiichi (and BWRs in general) do not model the reference leg. On this basis, it is recommended that an exploratory effort in modeling this component be initiated.

Area 1 Recommendation 6:

> The expert panel should place more emphasis on reviewing available examination information related to spent fuel (i.e., assemblies, pools, casks, etc.) at Fukushima Daiichi. Experts should also consider existing U.S. research that may aid in the decommissioning efforts and identify inspections at Fukushima Daiichi that may benefit ongoing U.S. activities.

To date, U.S. forensics efforts has placed less emphasis on reviewing spent fuel pool examination information from Fukushim Daiichi. The site contains seven spent fuel pools, several existing and a newly built dry storage facilities, and the spent fuel from six operating reactors. The spent fuel and structures experienced 'off-nominal' conditions and remain a key focus area of D&D efforts. The U.S. may also benefit by reviewing information available from these spent fuel D&D efforts. Likewise, the Japanese D&D efforts could benefit from the U.S. experience (The U.S. has a large inventory of spent fuel, with approximately 2500 dry storage casks in operation, and a range of research into related issues, such as spent fuel integrity).

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 7, the expert panel has requested that TEPCO Holdings 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 Holdings and update this table.

Area 1 Suggestion 2:
As discussed in Section 4, 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 1 Suggestion 3:
As dicussed in Sections 3.2.1.1 and 3.2.1.2, the RCW system may have played a role in the 1F1 accident progression. Examination information identified in Appendix C and other information previously obtained by TEPCO Holdings (i.e., dose surveys around the surge tank, system water level, images of system components, etc.) may provide insight into its role during the accident.

The list of information needs in Appendix C was updated to reflect this new information need (e.g., see RB-15 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 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 Holdings. 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 U.S. experts to address the following questions that 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. 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, U.S. 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 Table 9 through Table 11. These tables also note where information is available to address these information needs (see Appendix C).

Table 9. Area 2 information needs from the reactor building

Item	What/How Obtained[z]	Use[aa]	Data Available[bb]
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	A
	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-15	Examinations of 1F1 RCW surge tank; water level and additional dose measurements.	AE, AM, DD	A

Table 10. Area 2 information needs from the PCV

Item	What/How Obtained[cc]	Use[dd]	Data Available[ee]
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).[ff]	AE, AM, DD	A
	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[ff]	AE, AM, DD	A
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

[z] See list of acronyms.

[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].

[cc] See list of acronyms.

[dd] 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).

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

[ff] Although some images have been obtained; images do not indicate if RPV failed. Photos from 1F2 investigations [Section 5.2.3] indicate the presence of possible ex-vessel debris, but it has not yet been possible to extract samples for evaluating composition.

Table 11. Area 2 information needs from the RPV

Item	What/How Obtained[gg]	Use[hh]	Data Available[ii]
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 several 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 provides 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[jj];
- 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 near the MSIV.

[gg] See list of acronyms.

[hh] 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).

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

[jj]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.

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

- 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 to gain 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 Holdings 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 because 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 21.

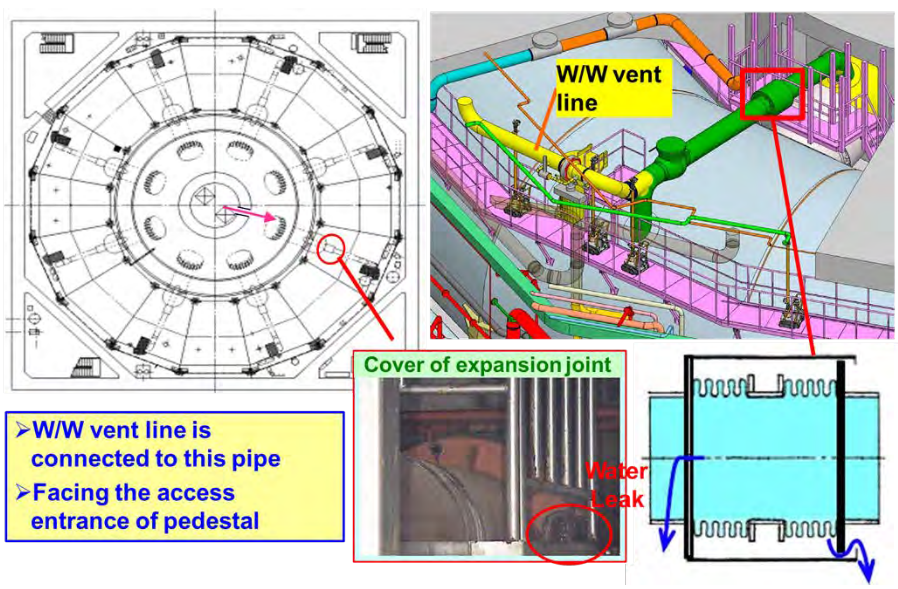


Figure 21. Impairment of 1F1 vacuum breaker line expansion joint. (Courtesy of TEPCO Holdings [179])

- Reactor Building Closed Cooling Water System (RCW) heat exchangers (~1 Sv/h) and associated piping found in the contaminated waste treatment areas (see, for example, Figure 22). 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). Entry into the torus room of 1F1 showed that the sand cushion drain pipes in 1F1 are leaking water (see Figure 23). A potential explanation for this is that the liner of the containment drywell was breached because of melt ablation. Then, as water is injected into the containment it makes its way through the pedestal region out of the hole in the drywell liner and into the sand cushion, eventually leaking out of this area into the torus room.

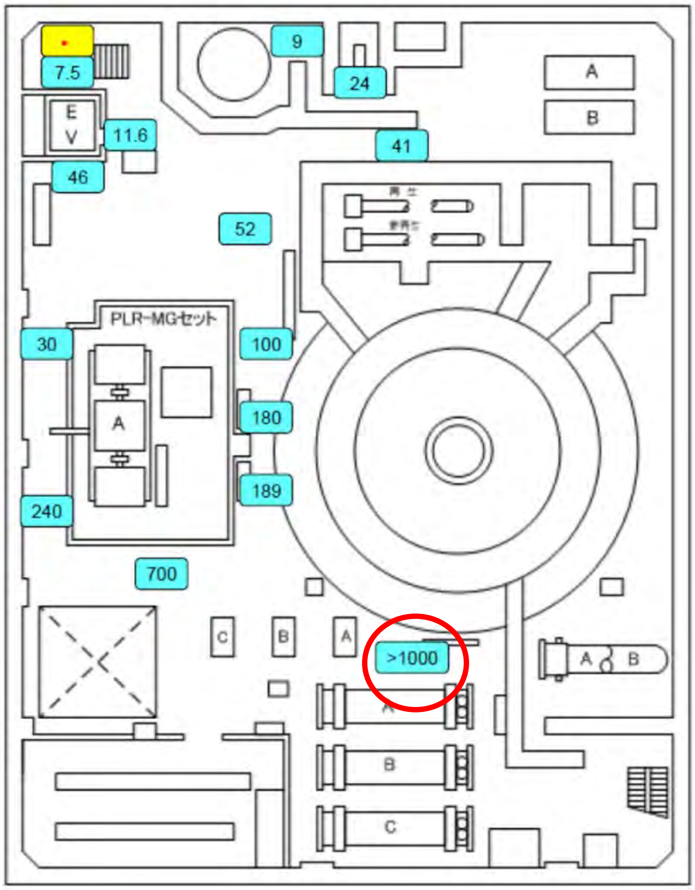


Figure 22. 1F1 reactor building first floor dose rate measurements (in mSv/hr; taken above floor elevation measured on February 14, 2013) illustrating elevated radiological contamination of RCW (circled area). (Courtesy of TEPCO Holdings [180])

Figure 23. Leakage out of the sand cushion drain pipes at 1F1. (Courtesy of TEPCO Holdings [181])

The elevated dose in the torus room of 1F1, compared to 1F2 and 1F3, is a consequence of gas-phase leakage into the torus room occurring at 1F1. Measured doses within the torus room of 1F1 (see Figure 24) are significantly higher than those in 1F2 and 1F3 (see Sections 4.2.1.2 and 4.2.1.3).

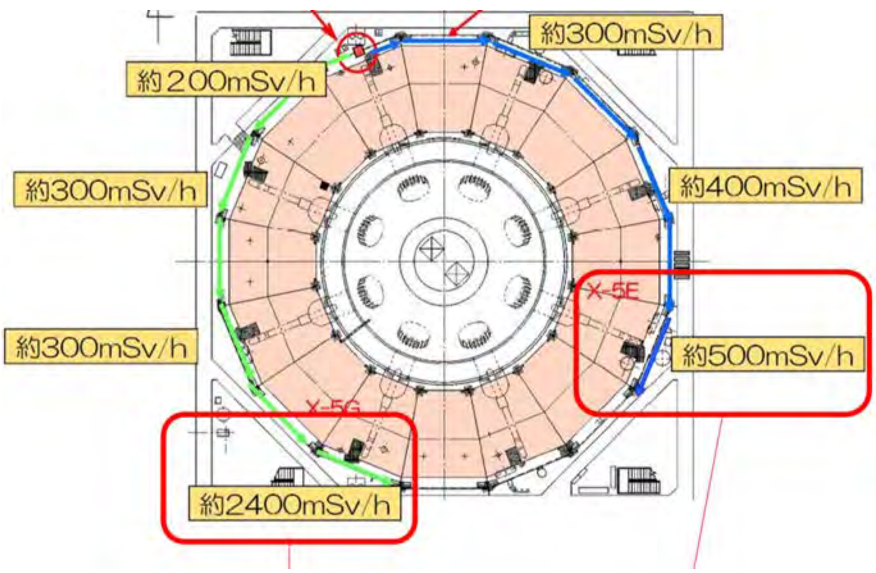


Figure 24. Torus room radiation level readings from entry into 1F1 reactor building measured on May 30, 2014 (in mSv/hr; taken above floor elevation). (Courtesy of TEPCO Holdings [182])

- Although no elevated dose rates were measured in the backside of the 1F1 Traversing In-core Probe (TIP) room, significant dose rates have been measured near the penetrations, see Figure 25 and Figure 26. Gamma camera measurements also indicated relatively high dose rates around the X-31, X-32, and X-33 instrumentation piping from the primary system, but not the X-35 penetration for the TIP tubes. Thus, failures of in-core instrument tubes during core damage progression did not impair the containment, contaminating the entire room.

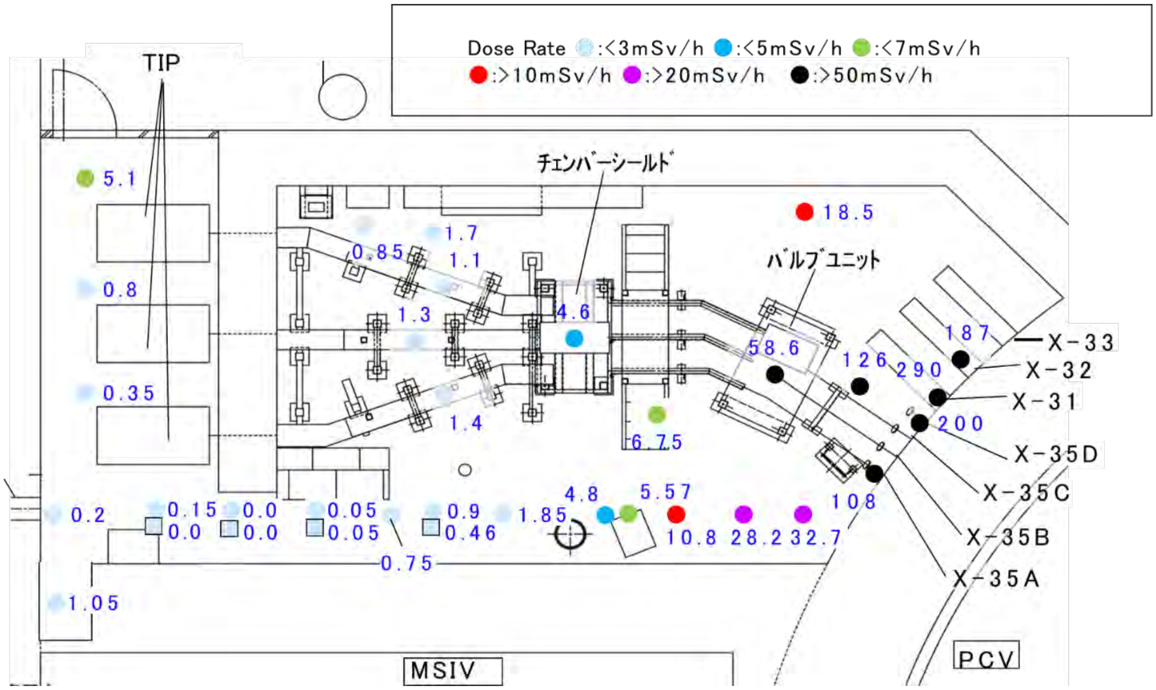


Figure 25. 1F1 reactor building TIP room dose rate measurements (in mSv/hr; taken above floor elevation between September 24 through October 2, 2015). (Courtesy of TEPCO [183])

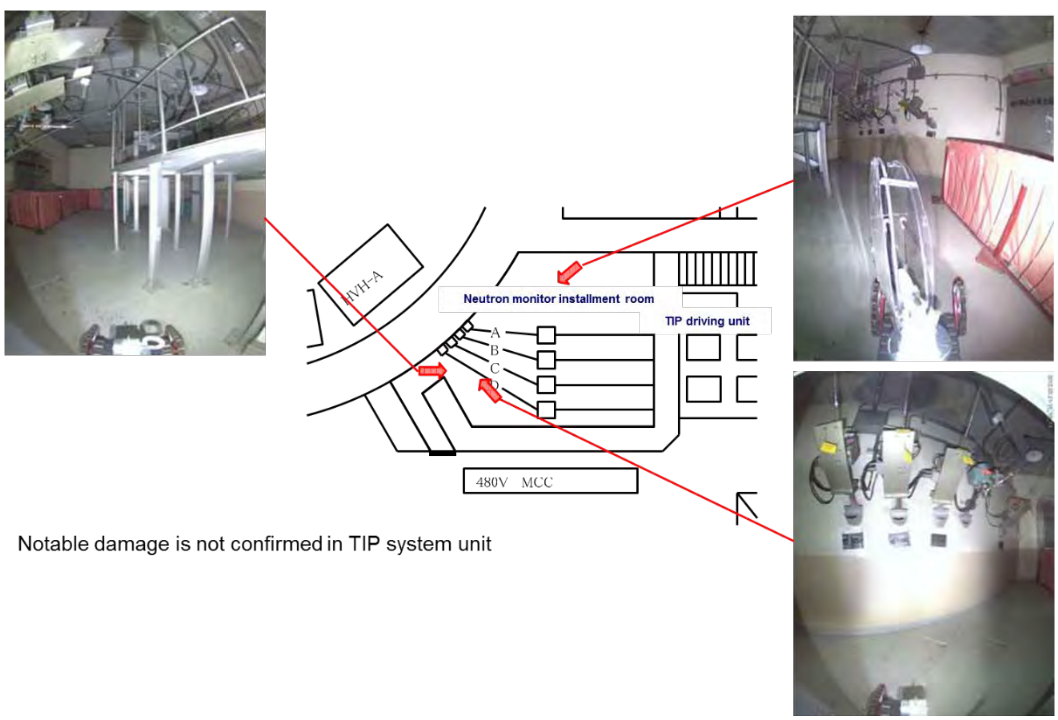


Figure 26. Typical TIP room illustration. (Courtesy of TEPCO Holdings [184])

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 27 shows the open blowout panel in the 1F2 reactor building.

Figure 27. 1F2 reactor building with open blowout panel. (Courtesy of TEPCO Holdings [185])

Torus room radiation dose rates measured in 1F2 (see Figure 28) are lower than values measured in 1F1. Examinations have identified the following areas within the 1F2 reactor building with notably elevated radiation doses:

- *In front of the X-6 penetration pipe flange.* As discussed in Section 3.2.1, high dose rates were measured at this location (see Figure 16). In addition, the rubber material (likely the chloroprene rubber cable sheath material stored inside the penetration for use with the Control Rod Drive (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.* Evalautions have measured dose rates of ~800 mSv/h (Figure 29).

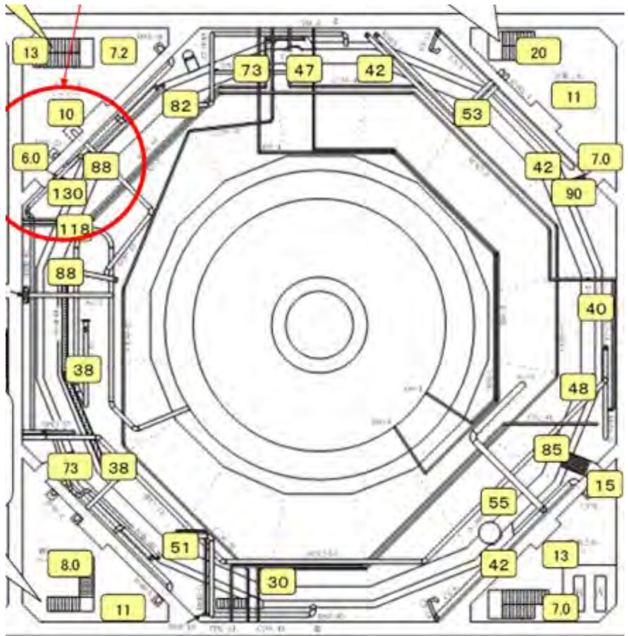


Figure 28. Torus room radiation level readings from entry into 1F2 reactor building (in mSv/hr; taken above floor elevation in April 2011). (Courtesy of TEPCO Holdings [137])

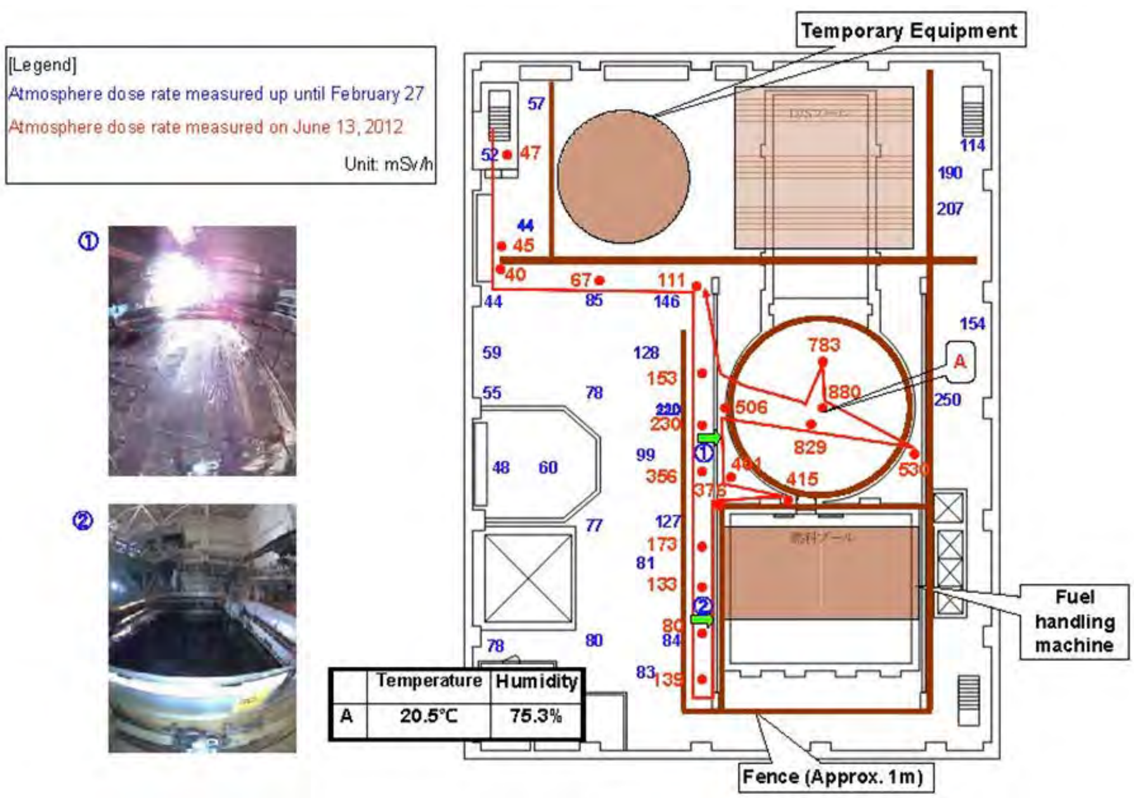


Figure 29. 1F2 reactor building refueling floor visual and dose rate information (in mSv/hr; taken above floor elevation on February 27, 2012 and June 13, 2012). (Courtesy of TEPCO Holdings [186])

Unlike 1F1, the RCW equipment is not contaminated. The cause for this difference is still unknown. It is speculated that the closure system (an explosive closure system) may have been initiated with the return of battery control.

No elevated doses inside the 1F2 TIP room have been identified, neither in the room nor near the penetrations, see Figure 30. Relatively small dose rates were measured near the penetrations. Any failures of in-core instrument tubes during core damage progression did not impair the containment. During panel meetings, U.S. experts questioned if the explosive closure system could have activated and sealed the guide tubes. The "explosive closure valve" is for emergency closure when the isolation valve (motor operated ball valve) cannot be closed during TIP operation (usually this ball valve is closed). However, U.S. experts learned that the TIP was not in operation at the time of the earthquake. Hence, the TIP line was closed and activation of the explosive closure system would not be triggered. Furthermore, it was directly confirmed in December 2012 that the 1F2 TIP ball valves were actually closed, and explosive closure valves were kept open [187]. Dummy TIP cables were inserted into the guide tubes (see [188]). On July 9th of 2013, after the dummy TIP cable was pulled out, attached substances were found on the head with a γ-ray dose rate of 14.0 mSv/h. A similar action was performed on July 19th of the same year, the attached substances yielded a dose rate of 95.0 mSv/h.

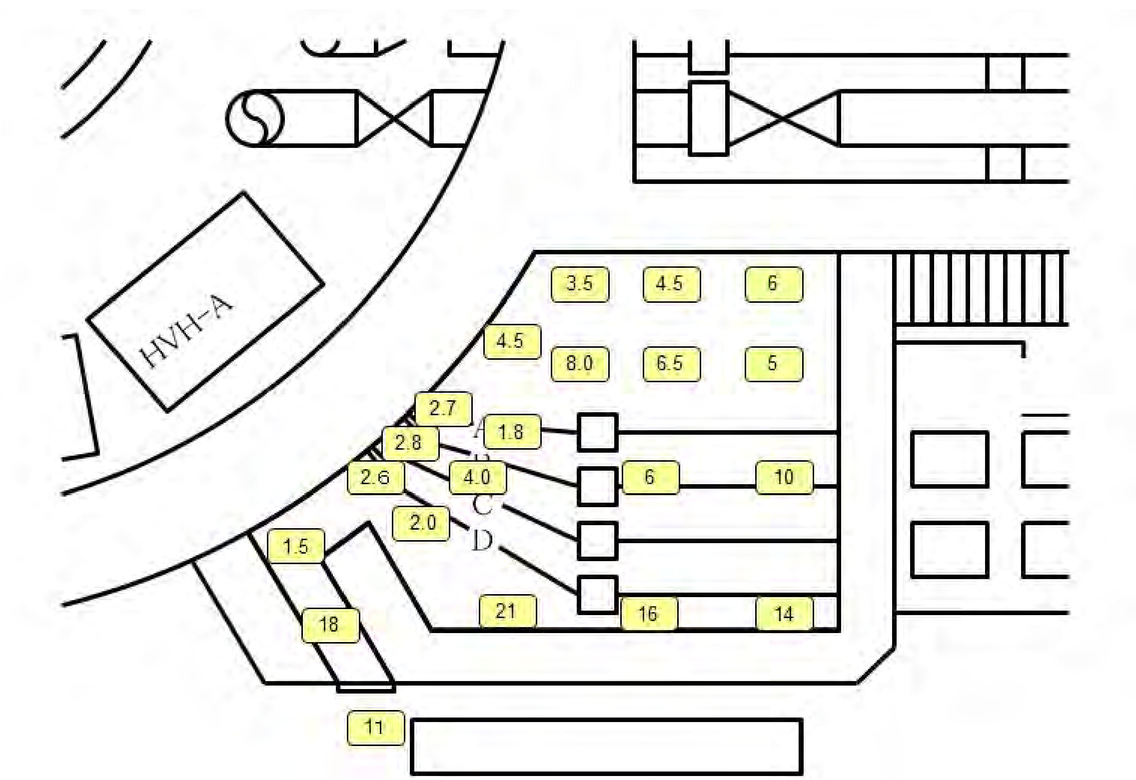


Figure 30. Air dose rates in 1F2 TIP room (in mSv/hr; taken above floor elevation on March 22, 2013). (Courtesy of TEPCO Holdings [189])

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 (see Figure 31 and Figure 32):
  - 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 17).

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

The door leading to the 1F3 TIP room was blown off its hinges (see Figure 33).

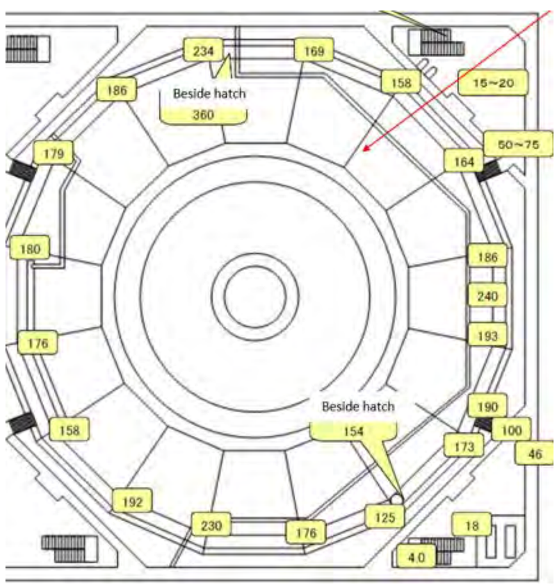


Figure 31. Torus room radiation level readings from entry into 1F3 reactor building (in mSv/hr; taken above floor elevation on April 25, 2011). (Courtesy of TEPCO Holdings [137])

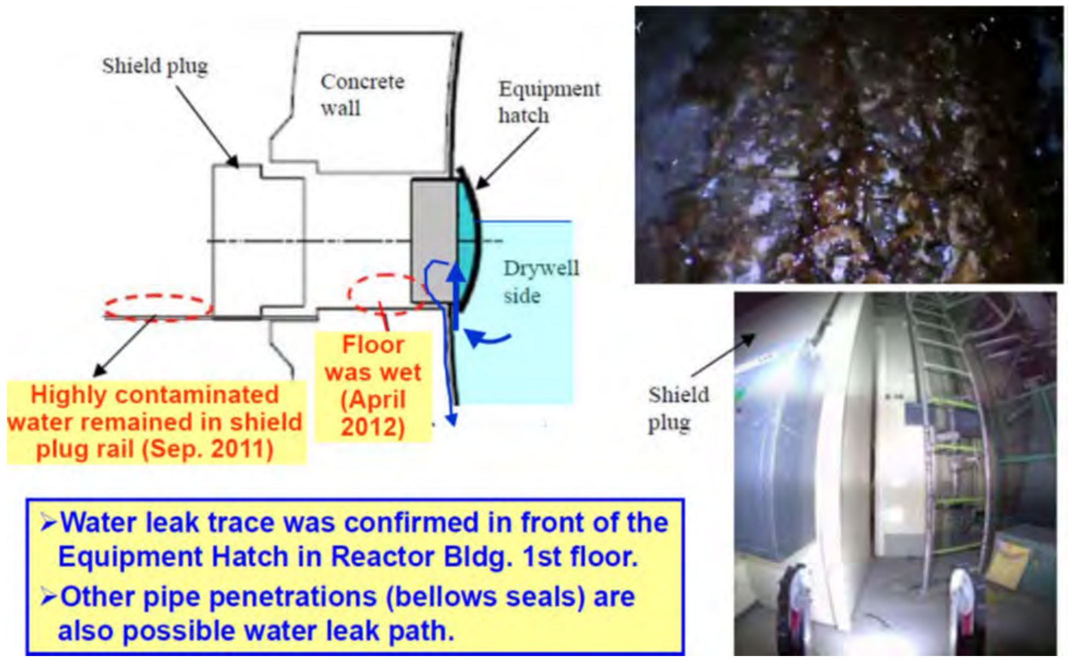


Figure 32. Liquid-phase leakage from 1F3 equipment hatch. (Courtesy of TEPCO Holdings [179])

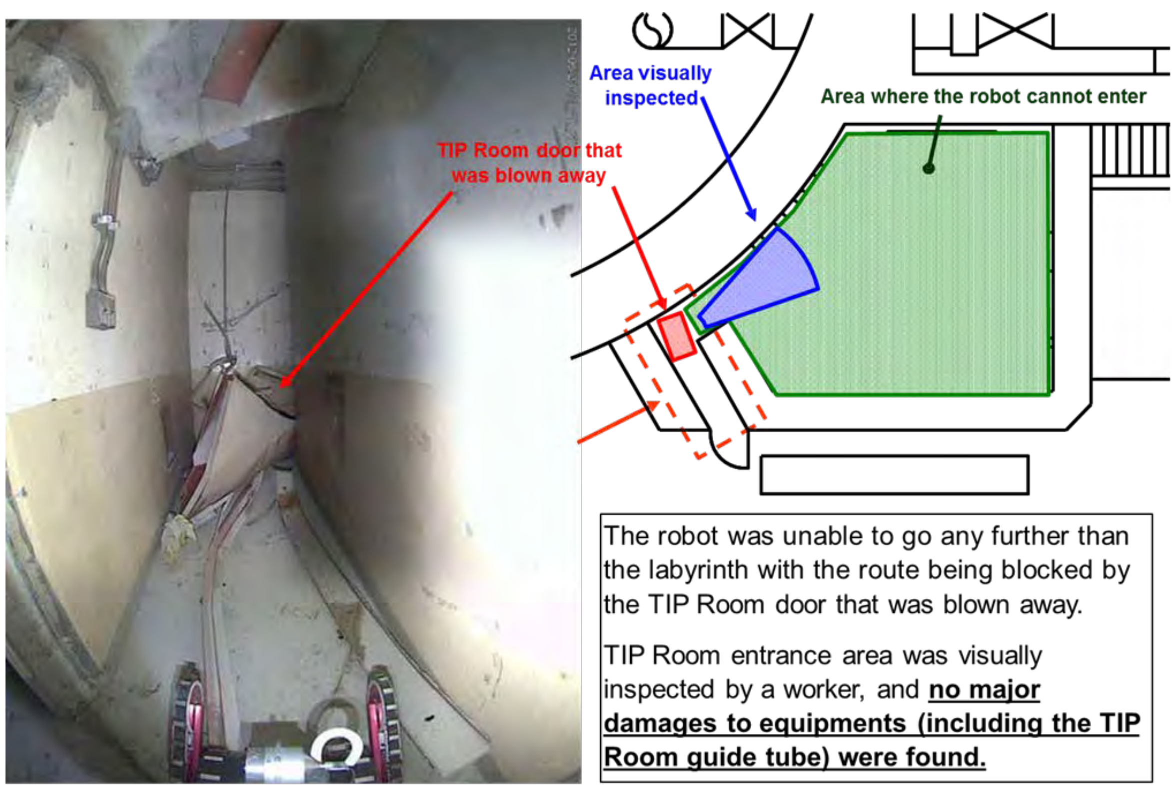


Figure 33. Robot image leading to 1F3 TIP room. (Courtesy of TEPCO Holdings [190])

The damage to the door indicates an overpressure in the room either from blowdown of the vessel or a hydrogen combustion event, suggesting that a thimble tube failed in the core region while the RCS was slightly pressurized. However, the dose rates leading to the labyrinth were small (on the order of 10 to

40 mSv/hr). Inside the TIP room, the dose rates were lower (~2 mSv/hr). Near the penetrations, the dose rates were found to be in the range of 30 to 50 mSv/hr.[190] This implies that some leakage back through the guide tubes may have occurred; however, if a tube had failed, substantial higher dose rates would have been expected.

Thus, it can be conclucluded that any failures of in-core instrument tubes during core damage progression did not result in containment impairment.

4.2.1.4 Insights and Limitations

A summary of notable locations of elevated dose rates are provided in Table 12. 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 12. Locations of elevated dose rate inside reactor buildings[kk]

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 Higher dose rates near the penetrations in the TIP room [183]
	2nd	Very high dose rates around RCW heat exchangers (see [38] 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 [179])
	1st	Elevated dose rates around X-34 penetration (main steam flow rate instrument piping penetration)
	2nd	Elevated dose rates around X-29B/C penetration (RPV water level instrument piping)
	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. Dose rates in the MSIV room slightly higher on the inside of the MSIV room (compared to dose rates in the 2nd floor)
	Refueling Floor	Very high dose rates found at the shield plug, above the drywell head

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

Reactor building dose rate measurements were acquired by TEPCO Holdings in 2012. Most data acquisition was obtained using dosimeters (handheld or attached to robots). Above floor gamma camera measurements were performed to locate radiation sources. 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. 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 that they are supporting.

The focus of the summary in Table 12 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 3.2.2.1, 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 (see Section 3.2.2.1) and the localized flammable gas combustion damage observed on the 1F1 refueling floor (see Section 6). In addition, the 1F1 shield plug relocation and subsequent dose measurements indicate that the plug drywell head flange in this unit likely leaked.
- As discussed in Section 3.2.2.1, degradation of the containment boundary appears to have occurred at all three units through several 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. Thus, 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 50 Sv/h to about 90 Sv/h near the end of the day on March 14, 2011. Although CAM data were fluctuating,[191] this elevated radiation level in the 1F1 drywell persisted for about one day. This coincides with a restoration of water injection to the unit. It also supports the hypothesis that fission products were released 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 such as 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 34 shows radiation measurements obtained from the drywell and wetwell during a period of significant core damage progression at 1F2.

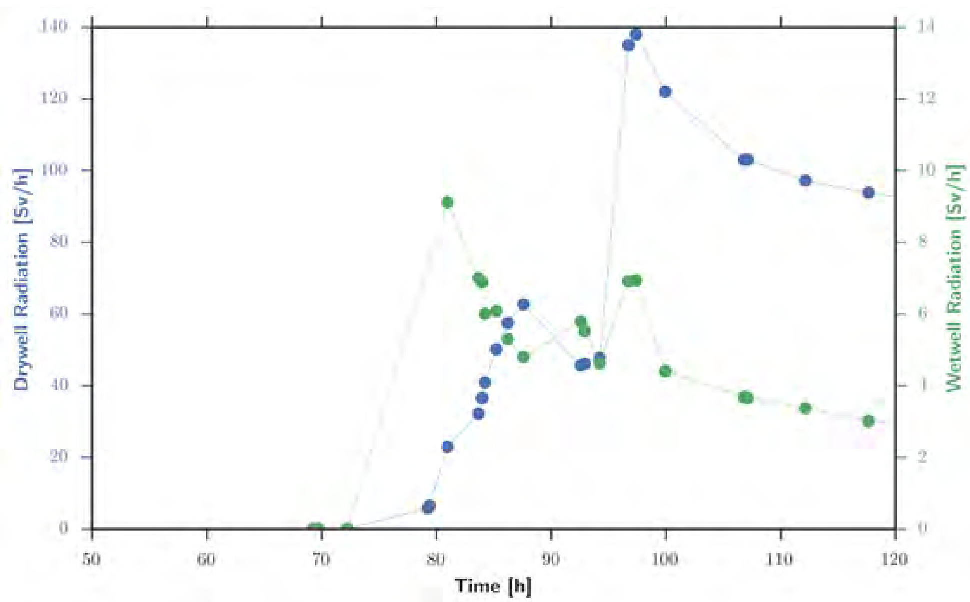


Figure 34. 1F2 drywell and wetwell CAMS readings. (Courtesy of Jensen Hughes based on information in [97])

As shown in Figure 34, 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 that 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.[97] In this effort, comparisions are made to measured values.[192]

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. In addition, it is unknown if the guide tube explosive sealing systems were activated when power was restored. 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. Thus, 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 [51]).

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 Post-Accident Examinations within the Containment

Insights can also be obtained from dose measurements obtained within the containment during examinations conducted with robots.

4.2.3.1 1F1 Examinations

During the March 2017 entry with the self-propelled, shape-changing, “Scorpian” robot through the 1F1 X-100B penetration (see 5.2.2.1), it was found that the dose rates were very similar to those obtained in April 2015. Because the dose is likely driven by long-lived isotopes, this is expected. The dose rates at the grating surface were between 3.8 Sv/h and 12 Sv/h in March 2017 and between 4.7 Sv/h and 9.7 Sv/h in April 2015. In the most recent analysis, a radiation detector was gradually lowered from the grating surface and the dose was found to drop as the detector approached the water’s surface and then drop even further until the floor of the PCV was approached. Here there is a significant amount of accumulated sediment that is likely contributing to the increase in dose. Dose rate measurements for March 2017 can be seen in Figure 35, while similar values for April 2015 can be seen in Figure 36.[193]

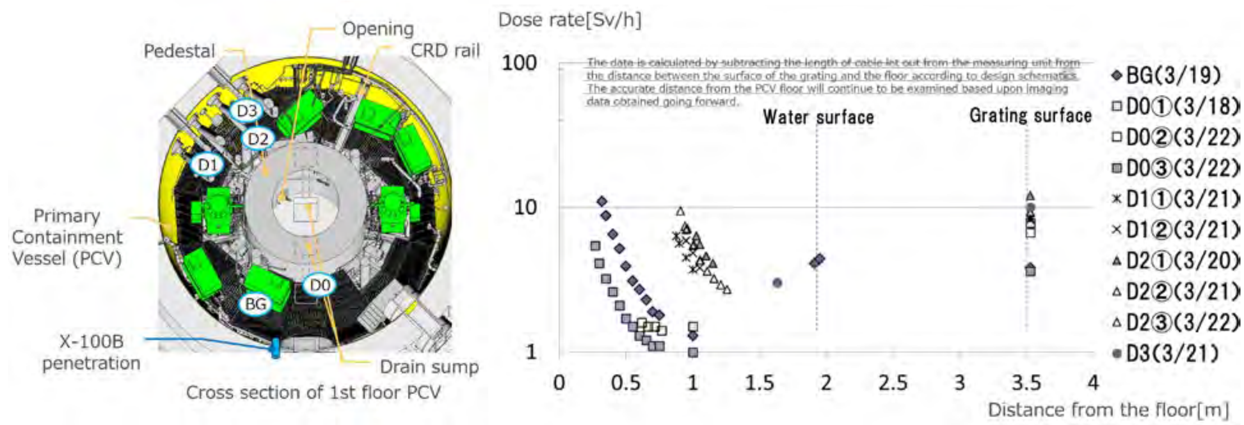
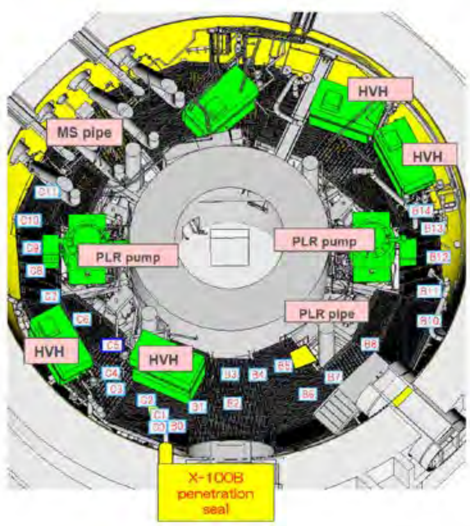


Figure 35. PCV dose measurements on and below the grated floor outside the pedestal region of 1F1, taken in March 2017 (Courtesy of TEPCO Holdings [193])



	Dose rate (Sv/h)
B3	7.4
B4	7.5
B5	8.7
B7	7.4
B11	9.7
B14	7.0
C2	6.7
C5	8.3
C6	7.7
C9	4.7
C10	5.3
C11	6.2

B3~B14 (Measurement date: April 10, 2015)
C2~C6 (Measurement date: April 15, 2015)
C9~C11 (Measurement date: April 16, 2015)

Figure 36. PCV dose measurements on the grated floor outside the pedestal region of 1F1, taken in April 2015 (Courtesy of TEPCO Holdings [193])

4.2.3.2 1F2 Examinations

During the January and February 2017 entries, dose measurements were also obtained with the self-propelled, shape-changing “Scorpian” robot through the 1F2 X6 penetration (see Section 5.2.2.2). Figure 37 shows the dose rates, which had to be corrected due to some initial calibration errors, at selected locations. Values are slightly higher, but similar to values measured for 1F1. It is speculated that variations in dose rate measurements are due to the distribution of debris within the PCV.

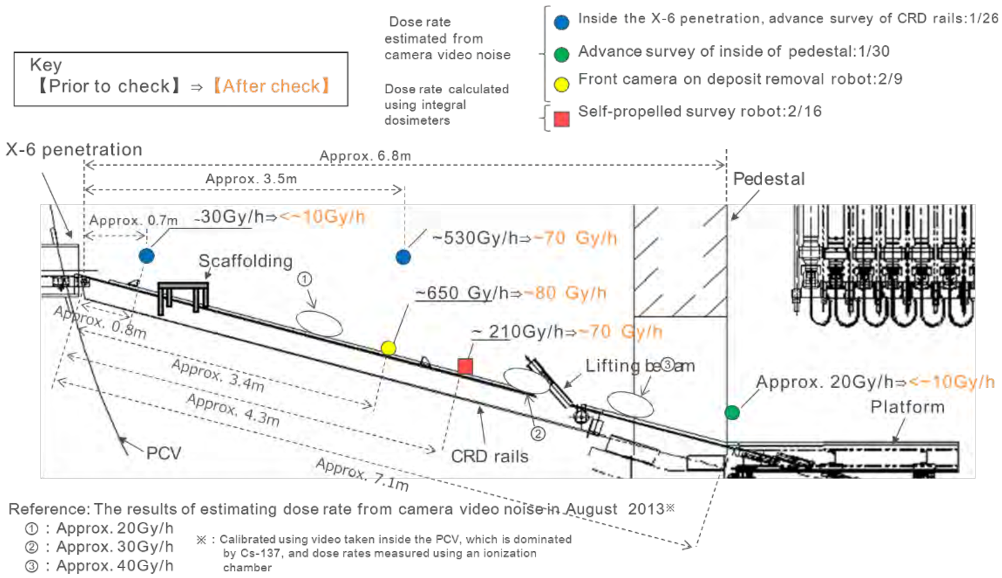


Figure 37. Corrected PCV dose measurements along the 1F2 X-6 penetration, taken January through February 2017 (Courtesy of TEPCO Holdings [194])

4.2.3.3 1F3 Examinations

Dose rates were also obtained during the July 2017 entry with the submersible 'Little Sunfish' robot through the 1F3 X53 penetration (see Section 5.2.2.3). Although measurements were limited to the control system location for this robot, the cumulative dose values were 2 Gy (July 19), 2 Gy (July 21), and 13 Gy (July 22). It is speculated that observed variations are associated with the distribution of debris within the PCV.[191,195]

4.2.4 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 protection of containment during the most active periods of core damage progression can significantly ameliorate the potential for notable off-site consequences. Another issue of interest is whether a core degradation event ultimately progresses 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. Issues of late-phase containment impairment would thus be more relevant from the perspective of accident remediation.

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.

4.2.4.1 Insights from Examinations of Concrete Samples

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. [196]

- 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).

4.2.4.2 Insights from Examinations of Soil Samples

TEPCO Holdings has been collecting and analyzing samples from the 1F site for the past several years and in this collection, the total activity of Cs is of large interest because of its contribution to human dose. In determining the activity of Cs, their sample collecting and examination have also yielded the Cs-134/137 ratios in several key areas around the site. These ratios, which are summarized in Table 13 for 1F1 and 1F2, range from 0.82 to 1.12. These values are consistent with previous samples collected onsite and the average plant-average Cs ratio. These values are also in good agreement from the sample collected within the 1F1 containment and the combined 1F1/1F2 vent stack drain water. This indicates that a significant amount of contamination in this sump is likely from 1F1.[20]

Table 13. Back-calculated cesium activity ratios found in 1F1 and 1F2 samples.[20]

Measurement Location	Cs 134/Cs 137 Ratio
1F2 TIP Tube B-Line	1.01
Sediment from PCV Entry 1F1	0.92
Refueling Floor 1F2 Concrete - Strippable Paint	1.12
Refueling Floor 1F2 Concrete - Waste Cloths	0.82-0.83
1F1/1F2 Vent Stack Drain Water	0.89

Interestingly, two separate peaks of Cs-ratio were found on paint-covered concrete from the 1F2 refueling floor. The first peak centers on 0.82-0.83 and was found in Cs that was easily wipeable from the surface. The higher Cs-ratio peaks were found when the paint itself was stripped off the concrete and examined. Two potential explanations for this are: (1) the influence of 1F1 PCV vent gas flowing backward through SGTS of 1F2 and (2) differing Cs-ratios resulting from the degradation of different burnup fuel assemblies. These two separate peaks can be seen in Figure 38.[20]

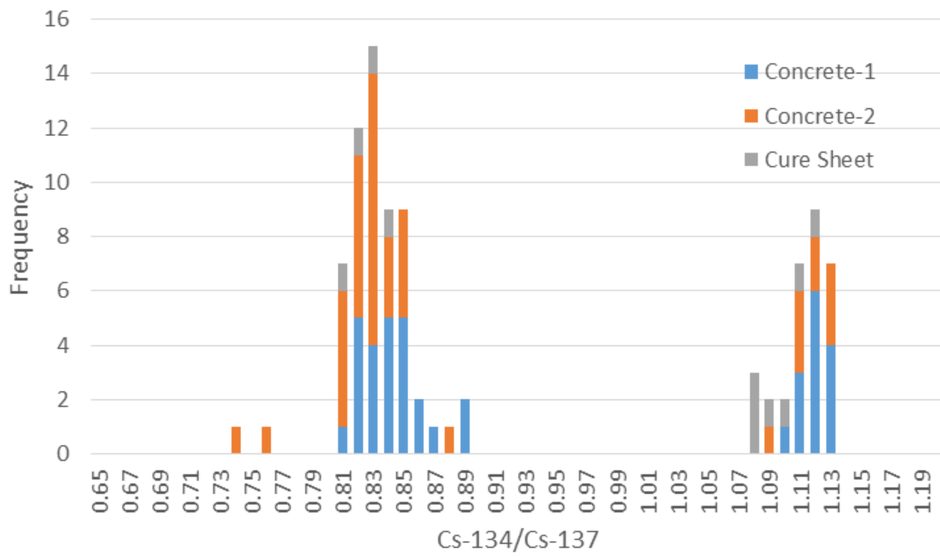


Figure 38. Cs-134/Cs-137 ratios of paint-covered 1F2 concrete. (Courtesy of TEPCO Holdings [20])

Examination of the ratio of ^{134}Cs to ^{137}Cs in the off-site contaminated soil samples indicates that there may also be a statistically relevant contribution from 1F3.[197] This evidence should provide a cautionary warning to avoid excluding 1F1 and 1F3 core damage progression scenarios exhibiting enhanced fission product release from containment to the environment 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. Evaluation results could be used to assess the off-site ^{134}Cs to ^{137}Cs isotopic ratios.

4.2.4.3 Insights based on Initial Burnup

Multiple organizations have explored initial radionuclide inventories at the time of the accident, including the JAEA, TEPCO Holdings, GRS and SNL.[198] These calculations require a 3D representation of the reactor core and subsequent full-core neutronic modeling. Generalized power distributions and burnup distributions for a characteristic BWR can be seen in Figure 39.[198] It can clearly be seen that there is a wide variety in the individual power of any given assembly, as well as the local burnup. The difference in these two values drives both decay heat and the overall radionuclide release. When examining key radionuclides and radionuclide ratios for each unit, it becomes apparent that these values are highly dependent on local conditions within the reactor. The Cs-134 to Cs-137 activity ratio for 1F3 is given in Figure 40 averaged on an assembly by assembly basis.[198] This ratio varies from 0.2 to 1.4, based primarily on assembly burnup. This introduces additional uncertainty in efforts to attribute key events to releases and deposition patterns to specific units at Daiichi.

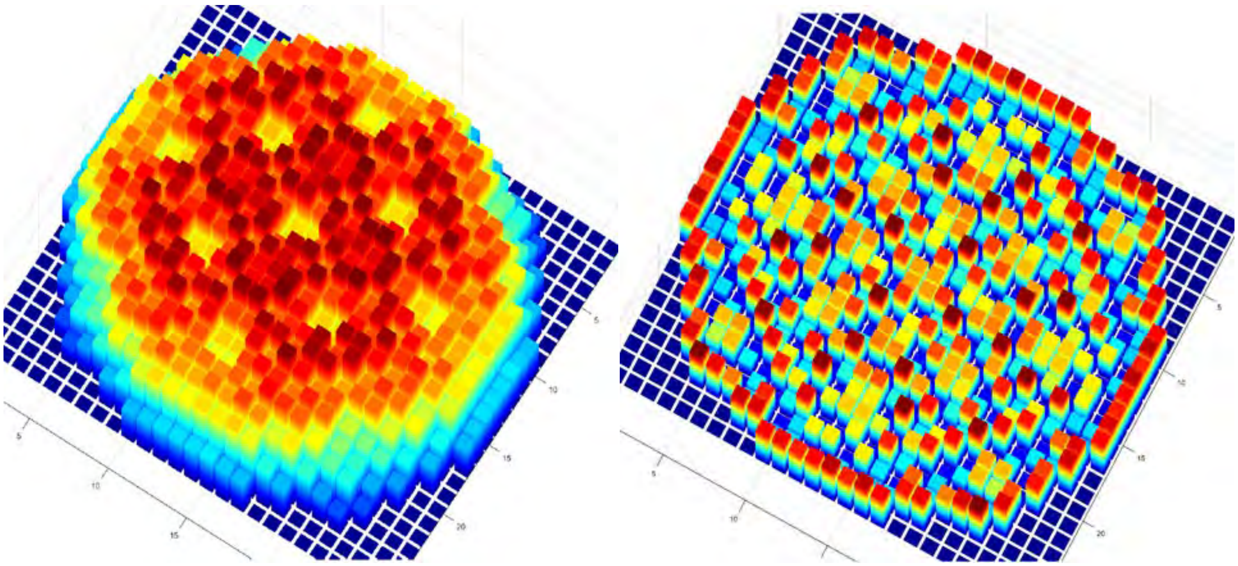


Figure 39. Generalized BWR power distribution (left) and burnup distribution (right) on an assembly by assembly basis, red represents higher power and higher burnup (Courtesy of SNL [198])

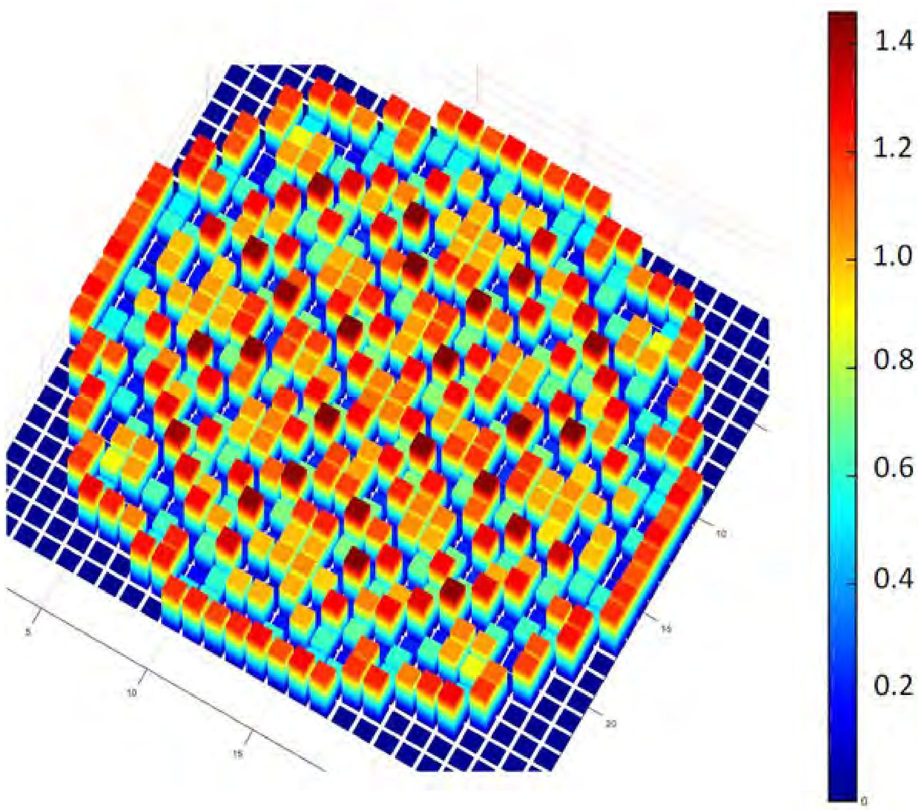


Figure 40. 2D spatial distribution of the Cs-134 to Cs-137 activity ratios in 1F3 (Courtesy of SNL [198])

4.2.4.4 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. These results highlight the insights that could be gained with respect to the release and transport of lower volatility fission products. Likewise, the range of burnups and powers within each fuel assembly of each unit introduces additional uncertainties in efforts to attribute key events, releases and deposition patterns to specific units at Daiichi.

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 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:

Like 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:

Like Area 1 Recommendation 4, the expert panel continues to be interested in examination information of MSIV room components. Specific examination needs to support these evaluations are identified in RB-10 and RB-13 of Appendix C.

The leakage of 1F3 in the MSIV room contrasts with 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. Similarly, the condition of the explosive guide tube sealing valve is desired to gain insights regarding the dose rates measured near the penetrations in the 1F1 TIP room are so much higher than in the other two units. An important component of such an evaluation is determining both the potential point in the accident when failure occurred and the relevance of this failure to gaseous releases from the impairment. As noted in this report, several 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 U.S. Forensics Effort continue to evaluate information obtained from examinations of RPVs within each unit impairment location. Additional visual information would be particularly useful in 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. 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.

If additional samples of concrete are obtained, 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]. Furthermore, the expert panel should continue to monitor any additional off-site ^{134}Cs to ^{137}Cs isotopic ratio data obtained from soil samples. As noted in [199], JAEA and IRID are monitoring the ^{90}Sr to ^{137}Cs isotopic ratio obtained from such samples.

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 [40] 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 interest at Daiichi because comparisons can be made between the multiple units that were affected. In addition, it is desired to gain insights about core debris relocation behavior and morphology within the reactor vessel, debris coolability, the effects of saltwater, and ex-vessel debris spreading and coolability from examinations. As discussed within this section, answers to questions about debris end-state are also required by TEPCO Holdings for successful and safe completion of D&D activities. High radiation levels currently limit the ability to gain direct examination information to address such questions. Hence, indirect as well as emerging direct observations, coupled with analysis model predictions, are providing the basis for preliminary debris removal planning.

This section summarizes findings from the Fukushima Daiichi forensics efforts to date in terms of how 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 insights related to information gained from available forensics information. The section concludes with suggestions to TEPCO Holdings 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 and to what extent has this material interacted with below vessel structure during relocation?
- 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, U.S. 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 Table 14 through Table 16. These tables also note if any information is available to address these information needs (see Appendix C). During FY2017, U.S. experts added several new information needs (PC-17 through PC-21) to Appendix C based on recent examination results. 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.

5.2.1 Thermocouple Measurements

Figure 41 through Figure 43 provide thermocouple measurements [11] 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. 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 Figure 41 through Figure 43) and then flows down along the exterior surface of the core barrel to the core inlet. This led TEPCO Holdings 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 [200]). In such cases, some fraction of the coolant could bypass the core debris; and the material was not fully cooled.

Table 14. Area 3 information needs from the reactor building

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

[a]See list of acronyms.
[b]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).
[c]Some information available [Green]; NA: no information available [Orange].

Table 15. Area 3 information needs from the PCV

Item	What/How Obtained[a]	Use[b]	Data Available[c]
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)[d]	AE, AM, DD	A
	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[d]	AE, AM, DD	A
PC-17	Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1.The upper surface of the sediment is ~ 30 cm above drywell floor	AE, AM, DD	NA
PC-18	Evaluate the nature of the material below the upper surface of the debris at the X-100B penetration location in 1F1 to determine if it is additional sediment or other material such as core debris	AE, AM, DD	NA
PC-19	Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location	AE, AM, DD	NA
PC-20	Chemical analysis of black material on 'existing structure' in 1F1 image at location 'D3'	AE, AM, DD	NA
PC-21	Images from examinations in 1F3 X-53 penetration.	AE, AM, DD	A

[a]See list of acronyms.
[b]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).
[c]Some information available [Green]; NA: no information available [Orange].
[d]Although some images have been obtained; images do not indicate if RPV failed. 1F2 investigations [see Section 5.2.3] indicate the presence of possible ex-vessel debris, but it has not yet been possible to extract samples for evaluating composition.

Table 16. Area 5 information needs from the RPV

Item	What/How Obtained[a]	Use[b]	Data Available[c]
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]See list of acronyms.
[b]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).
[c]Some information available [Green]; NA: no information available [Orange].

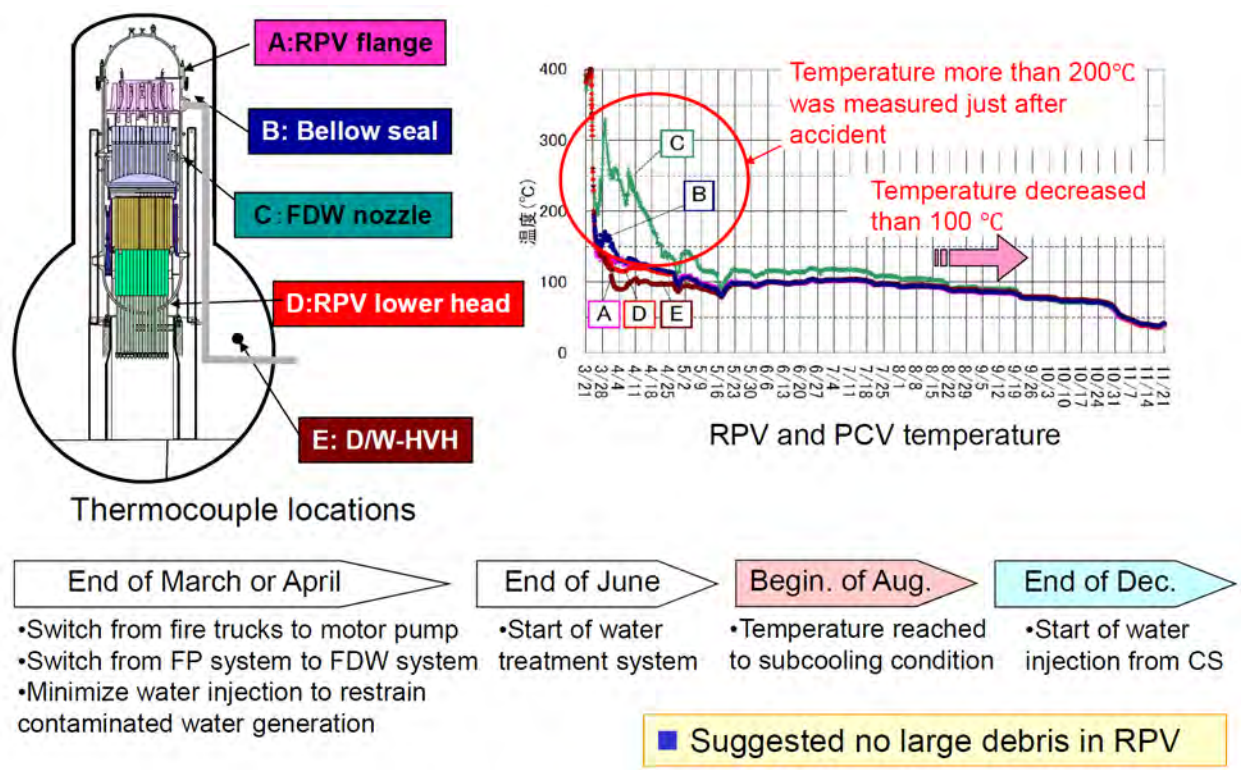


Figure 41. RPV temperature measurements for 1F1 following the accident. (Courtesy of TEPCO Holdings [11])

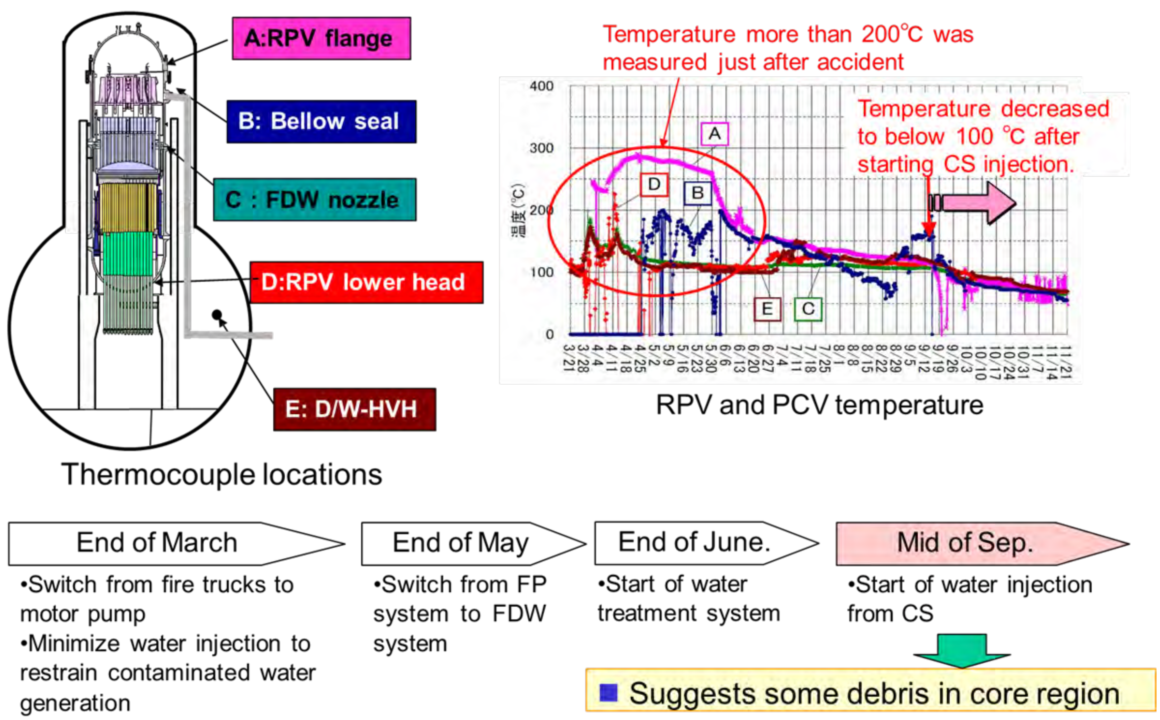


Figure 42. RPV temperature measurements for 1F2 following the accident. (Courtesy of TEPCO Holdings [11])

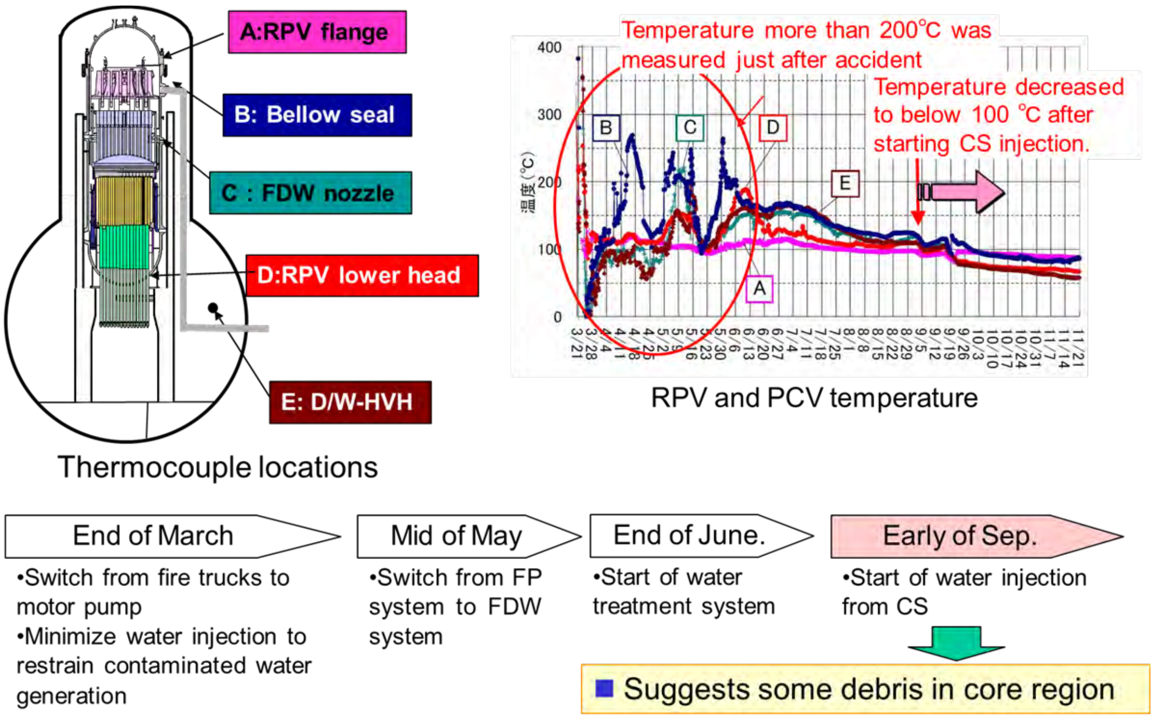


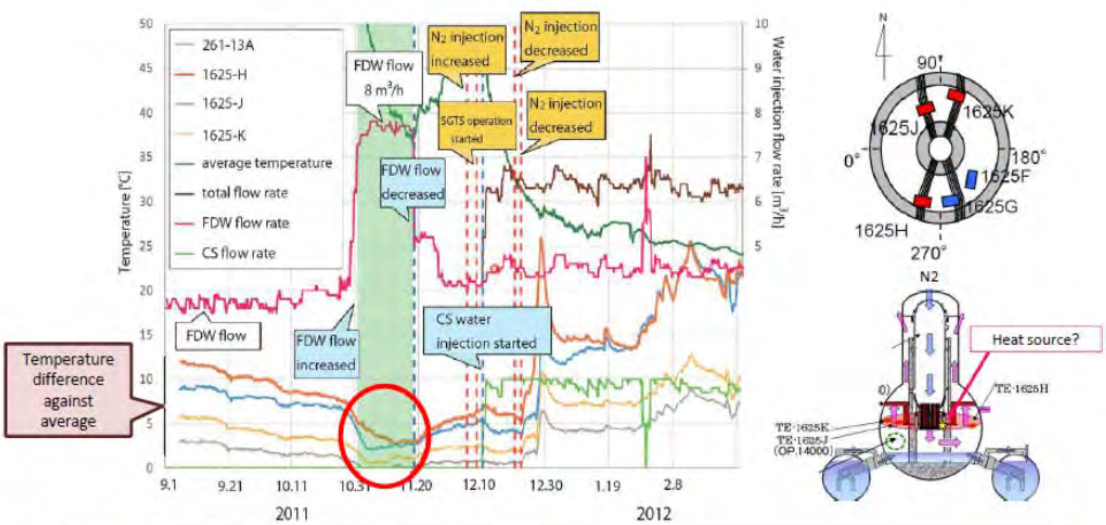
Figure 43. RPV temperature measurements for 1F3 following the accident. (Courtesy of TEPCO Holdings [11])

On this basis, TEPCO Holdings 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 introduces a water spray from directly above the core. As shown in Figure 42 and Figure 43, 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 Holdings 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 U.S. [49,50] as well as international [201] 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 an unmitigated station blackout until ~15 hours into the accident sequence [e.g., an event in which all onsite and offsite alternating current power is lost and in which no successful mitigating actions are taken]. At this point, operators could reflood the core with seawater. However, the predictions are less consistent for 1F2 and 1F3 where operators could 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 Figure 41 through Figure 43) that would allow the actual plant conditions to be ascertained.

Aside from these general observations, it is noteworthy that the TC data in Figure 41 through Figure 43 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 Figure 41 through Figure 43. 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 Figure 41 through Figure 43 may provide further insights on likely debris distributions. This type of analysis is relevant to the ongoing MAAP-MELCOR cross-walk activity;[51] i.e., these two codes predict quite different in-vessel core melt behavior and, thus, 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.

More recently, HVH temperature measurements made by TEPCO Holdings since the accident indicate that there is likely some core debris around the CRDs in the peripheral region below the RPV;[18] see Figure 44. However, based on these measurements it is difficult to tell if the debris is inside or outside the CRD housing. This is based on the observation that the HVH temperatures surrounding the CRDs are higher than the average PCV atmospheric temperature, and this this tendency becomes stronger when the FDW injection flowrate decreases. These inferences are consistent with newly developing information being generated by TEPCO Holdings using robot examinations. As discussed in Section 5.2.2, photographs obtained inside the PCVs for 1F2 and 1F3 indicate extensive damage to CRD structures and adhering material that could be core debris, but at this point chemical analysis has not been completed to verify that the material contains fuel and/or cladding.



・HVH temps. Are higher than the average PCV temperature.
・This tendency becomes strong when FDW injection is low and vise versa (red circle).
⇒It is estimated that debris is located around Control Rod Drive(CRD) at reactor peripheral region.

Figure 44. Relationship between HVH temperature and water injection from the Feedwater System. (Courtesy of TEPCO Holdings [18])

The results of this collection of temperature measurements, as well as the supporting code analyses and observations obtained by robots in all three units (see Section 5.2.2), help to inform D&D activities. The results indicate that TEPCO Holdings 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 all three units. 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 Holdings has obtained other valuable information from within the PCVs for all three units using robotics examinations through containment penetrations.

5.2.2.1 1F1 Examinations

For 1F1, visual information has been obtained by gaining access through the ‘X-100B’ penetration (see Figure 45).[202] 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.

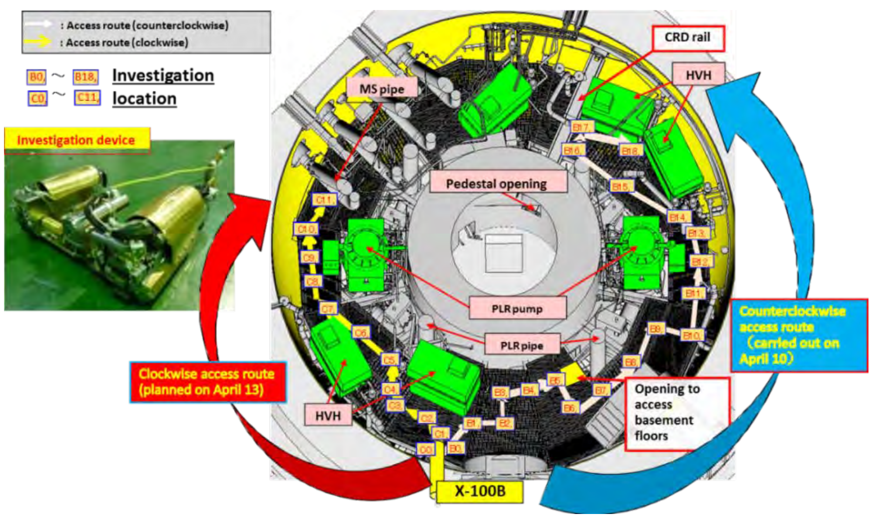


Figure 45.   Location of X-100B in 1F1 PCV and pathways of robotic examinations completed on April 10, 2015. (Courtesy of TEPCO Holdings [202])

Upon gaining access through this penetration, TEPCO Holdings initially lowered a video camera through the catwalk to the drywell floor to measure water level, temperatures, and radiation levels inside the PCV (see Figure 46). The camera views revealed the presence of a fairly deep (i.e., ~30 cm) layer of sediment covering the drywell floor at the penetration location (see Figure 47). The sediment layer deformed when touched by the camera. The blue material on top of the sediment shown in Figure 47 was estimated to be lead, which is consistent with the fact that the penetrations shielding was found to be melted upon gaining access through the penetration (see previous discussion). It was not possible to determine if this 30 cm deep accumulation of material was all sediment, or if the sediment layer was shallower and covered other material which could include core debris that spread from the pedestal opening.

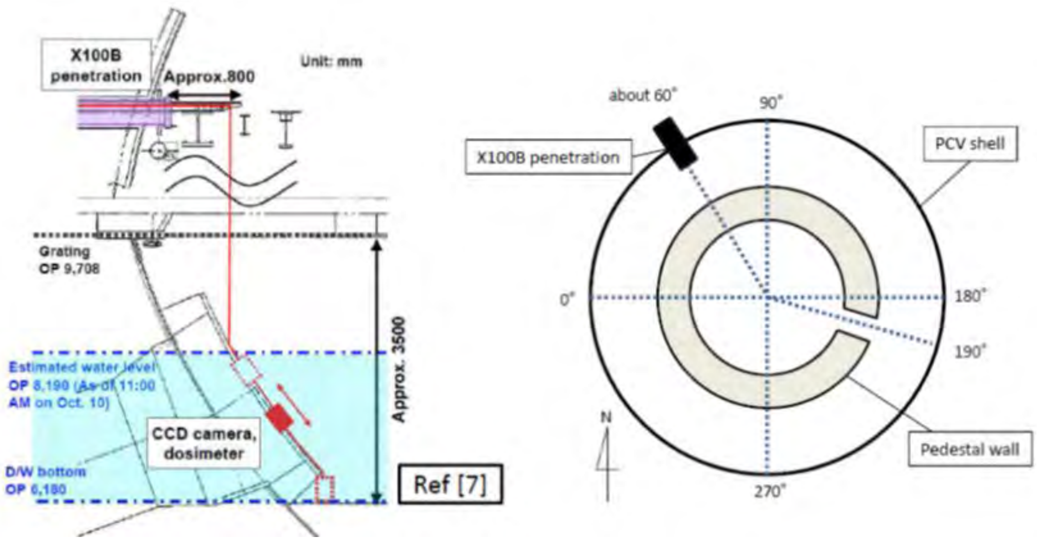


Figure 46. Illustration of CCD camera access through X-100B penetration. (Courtesy of TEPCO Holdings [18])

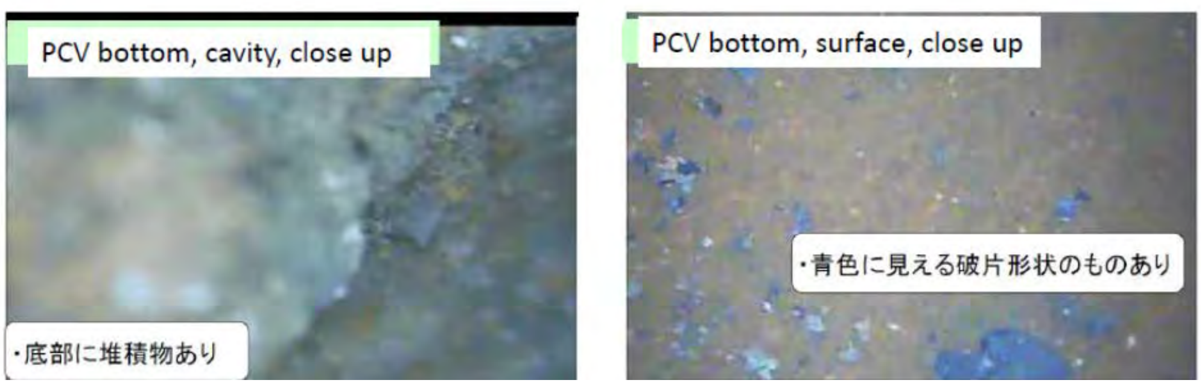


Figure 47. Views of sediment surface covering the drywell floor at the X-100B location. (Courtesy of TEPCO Holdings [18])

Possible materials that could be included in the sediment are paint and rust from the PCV structure, RPV insulation (typically containing silica, calcia, and/or alumina), sand from injected seawater, or concrete aerosols from MCCI.[203] Thus, analysis of this sentiment could provide valuable data on the event sequence (e.g., if core/concrete interaction occurred). Concrete aerosol would be expected to be dominated by SiO_2 since the reactors at Fukushima employee siliceous concrete. To determine if the silica was from insulation versus MCCI, it would be important to compare the silica/calcia and silica/alumina ratios in the sediment to that expected from insulation versus that expected from MCCI. Furthermore, it would be desirable to probe below the sediment surface to determine if hard structure exists, as would be expected if core debris was at the X-100B location.

As noted, due to the presence of the sediment layer, it is not possible to determine if core debris is on the drywell floor at the X-100B penetration, which is ~ 130 degrees from the pedestal doorway (Figure 45). Determining if debris lies below would provide a data point for assessing predictions of ex-vessel core melt spreading based on MAAP and MELCOR pour scenarios as calculated with MELTSPREAD.[52] As is evident from Figure 48, the measurement indicates that if core debris is at the penetration location, then MELTSPREAD prediction of spreading distance based on MAAP pour conditions would be consistent with that finding. However, 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. For this scenario to be correct, the material at the X-100B location would need to consist entirely of sediment. Determining the actual material composition at this location would be useful in reducing the range of possibilities regarding the extent of melt spreading in 1F1.

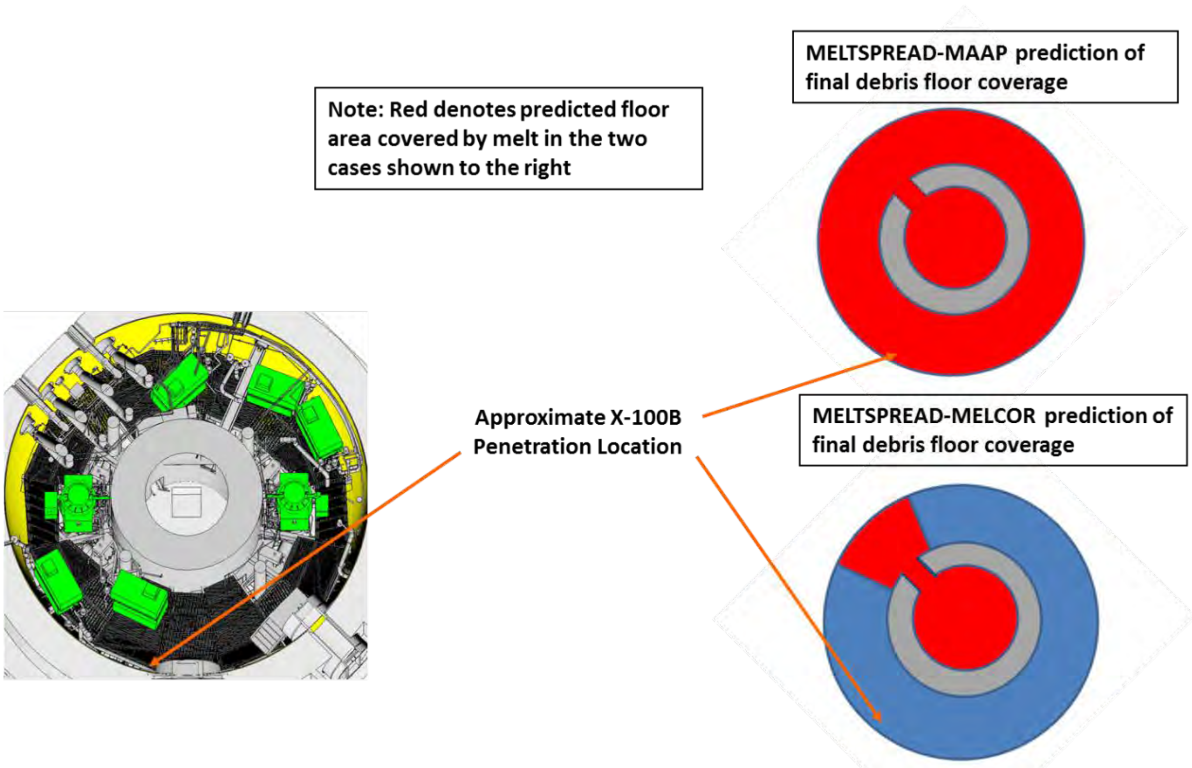


Figure 48. Approximate location of X-100B penetration relative to predictions of core debris spreading in 1F1. (Courtesy of TEPCO Holdings and ORNL [52,202])

Additional examinations in 1F1 near the pedestal doorway and sumps have indicated the presence of foreign material on the drywell floor and walls; see Figure 49.[193] The presence of sediment on horizontal surfaces makes it difficult to determine if this is core debris on the floor in most photos. However, one image (i.e. 'D3' in Figure 49) from outside of the pedestal doorway from a vertical structure suggests the presence splattered core-like debris. If this is splattered core material, two possible deposition methods are: i) melt relocation from the RPV, or ii) splattered from melt eruptions due to MCCI near the structure; see Figure 50 for a comparison with splattered material formed by melt eruptions from previous MCCI tests [204]. Elemental analysis of a splatter sample would be able to distinguish between these two cases based on whether concrete oxides (e.g. SiO_2, CaO, Al_2O_3, MgO) are present in the material, which are indicative of MCCI. This analysis has been highlighted as part of the PC-3 recommended examinations; see Table 13.

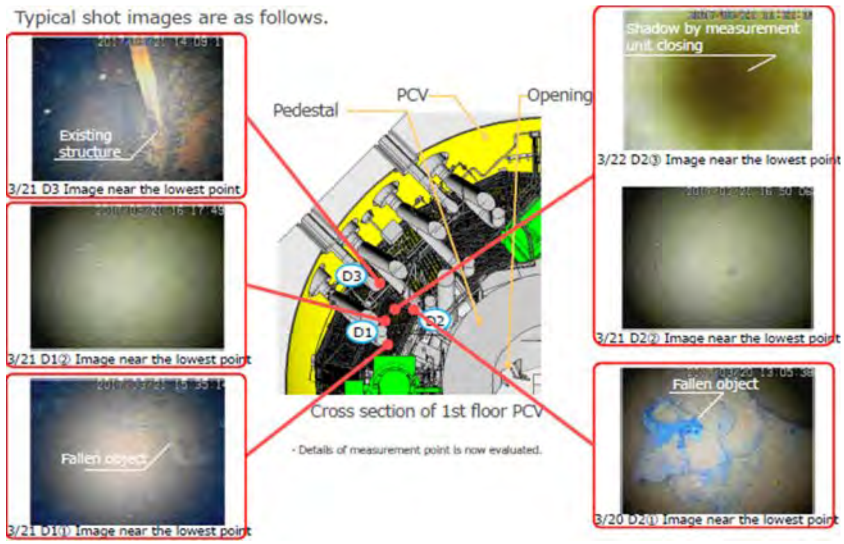


Figure 49. Photographs of structures outside the pedestal doorway in 1F1 (Courtesy of TEPCO Holdings [193])

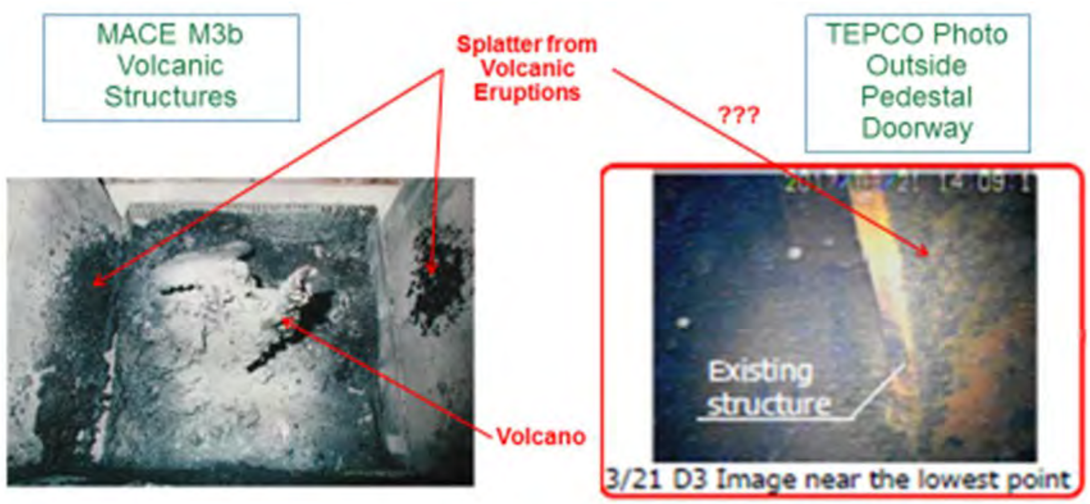


Figure 50. Comparison of photographs of wall splatter material observed in MCCI tests with prototypic reactor material and material observed in 1F1. (Courtesy of ANL[204] and TEPCO Holdings[193])

5.2.2.2 1F2 Examinations

Valuable data are also emerging from 1F2 examinations.[205, 206, 207] Access was gained through the X-6 penetration, allowing robot access to the CRD exchange rail (Figure 51). This initial investigation revealed the presence of black material laying on the surface of the exchange rail near the CRD platform (Figure 52). The composition of the black material has not been determined. Possibilities include melted CRD control cabling insulation or core material. Thus, determining the composition of this material would provide evidence of the temperatures that the below vessel structure experienced during the accident, as well as possibly confirming lower head failure if the material contains a significant amount of core debris.

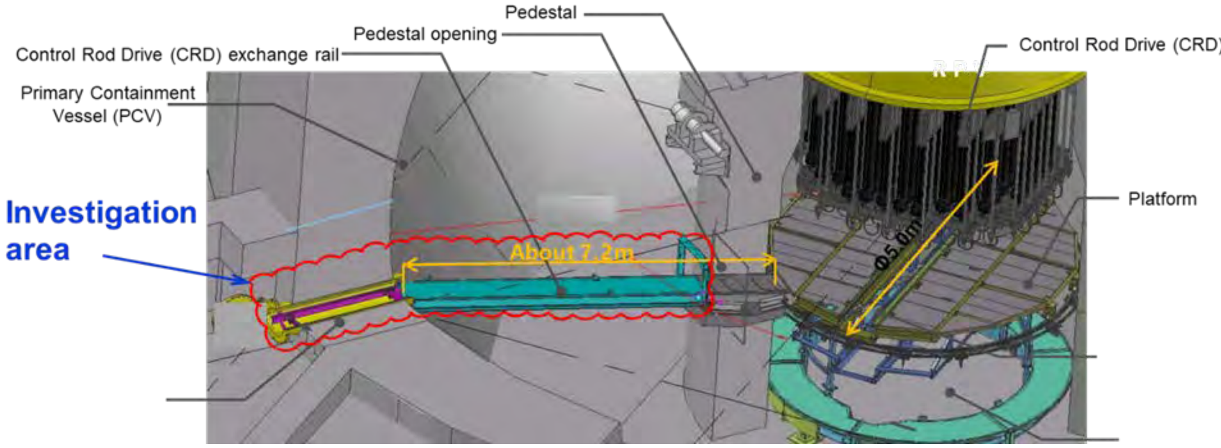


Figure 51. Access path through the X-6 penetration in Unit 1F2 allowing examination along the CRD exchange rail. (Courtesy of TEPCO Holdings[205])

Figure 52. Photograph of CRD exchange rail surface near the CRD platform showing the presence of black material (Courtesy of TEPCO Holdings[205]).

Investigations into the 1F2 containment performed in February 2017 resulted in images of the pedestal region. A combination of the different pictures taken is shown in Figure 53. The metal grating is failed in multiple locations and depressed elsewhere. This figure suggests that a portion of the corium from the

RPV relocated ex-vessel after a vessel failure event. This is further reinforced by the pictures of 1F5 (see Figure 54), which was shut down at the time of the tsunami.[206]

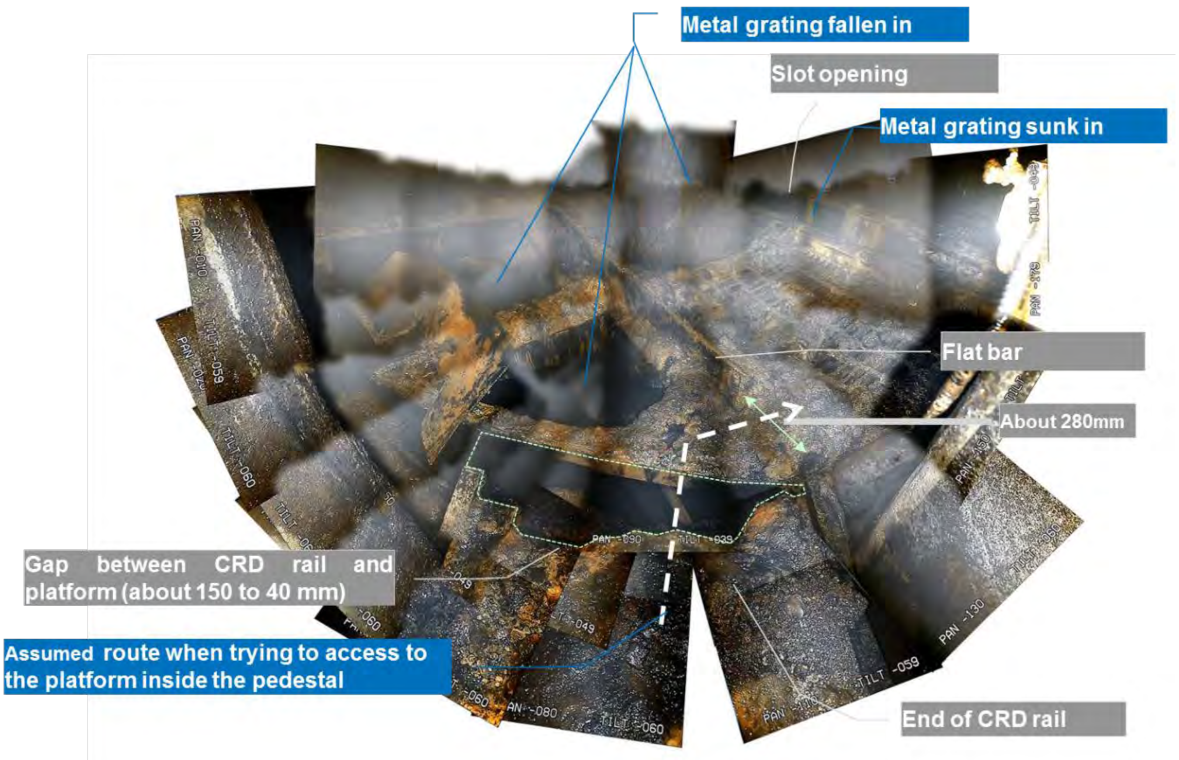


Figure 53. Combined images from entry into 1F2 pedestal region, indicating findings (Courtesy of TEPCO Holdings[206]).

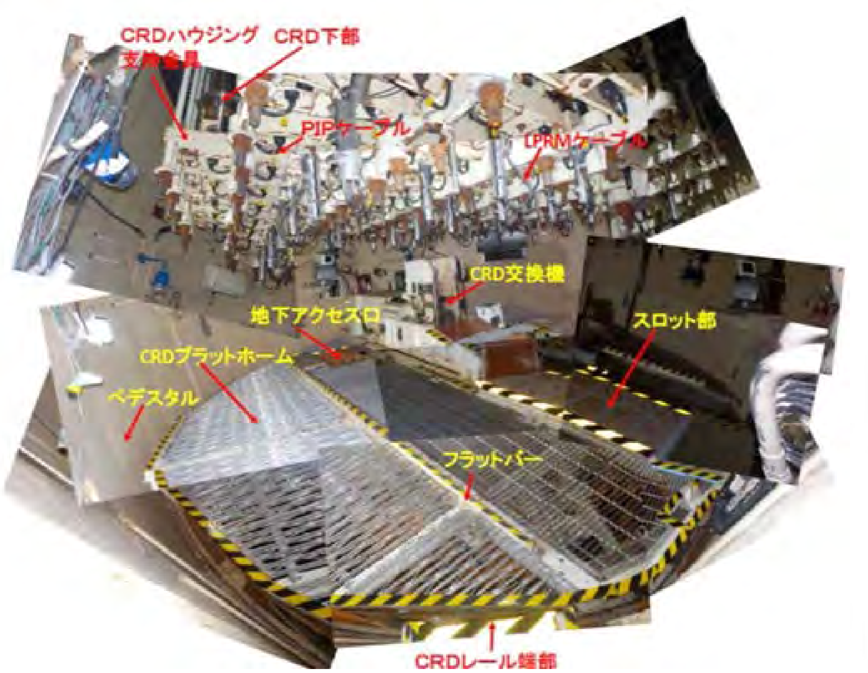
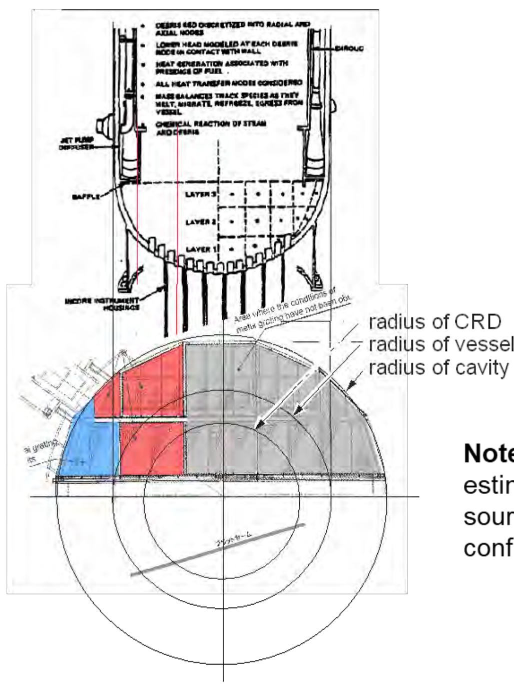


Figure 54. Image of 1F5 pedestal region, included for reference (Courtesy of TEPCO Holdings [206]).

Some of the grating deformations show evidence of gradual plastic deformation in the form of sagging grating, as though a large mass of very hot but somewhat solid corium material came to rest on the grating and caused the grating to gradually deform under the thermal loading and weight associated with this material. Other grating failure show characteristics of a more aggressive thermal attack and failure.[206]

Figure 55 shows a projection of the observed grating damage against the dimension of the vessel CRD penetrations, the vessel outer radius and the cavity wall dimensions to illustrate the potential relocation pathway and possible radial spreading experienced by the core materials that apparently exited a failure in the vessel lower head. It seems plausible that the vessel failure location could have been from one of the CRD or other vessel penetrations around the outer radius of the penetrations in the lower head, which subsequently spread radially outward. This radial spreading might have taken place as a result of the considerable CRD structures that are located just below the vessel, as shown in Figure 56 below where temporary freezing and subsequent lateral spreading could divert the relocating materials from the exit location in the vessel. Subsequent camera investigations should reveal whether the observed grating damage is confined to the location observed so far, which could be explained from a single penetration failure, or whether the relocating melt is more globally observed beneath the vessel. The evidence obtained so far might suggest a penetration failure around the outer radius of CRD penetrations.[206]



Note: Positional relation is estimated based on several sources of information. It is not confirmed by the drawing.

Figure 55. Superposition of grating damage on projection of vessel and cavity dimensions as modified by SNL. (Lower Image Courtesy of TEPCO Holdings [206]; Upper Image Courtesy of ORNL[208]).

Figure 56. Photo showing the inside of 1F2 during a regular inspection. (Courtesy of TEPCO Holdings [207]).

5.2.2.3 1F3 Examinations

Investigations into the 1F3 containment performed in July 2017 have also yielded important information. Numerous images were obtained of relocated material from cameras that were installed on the submersible robot, "Little Sunfish" which was inserted into the X-53 penetration (Figure 57). These images indicate that some relocated materials were deposited on CRD housing and cable structures that remained relatively intact. However, in general, the damage to structures beneath the RPV lower head appear more severe than observed in 1F1 and 1F2 examinations (grating structures were no longer present, some CRD support structures had fallen onto the platform, and a significant amount of previously relocated material was detected on the containment floor within the pedestal region (see Figure 58 through Figure 60). Additional evaluations and examinations will be completed to discern the composition of these relocated materials.

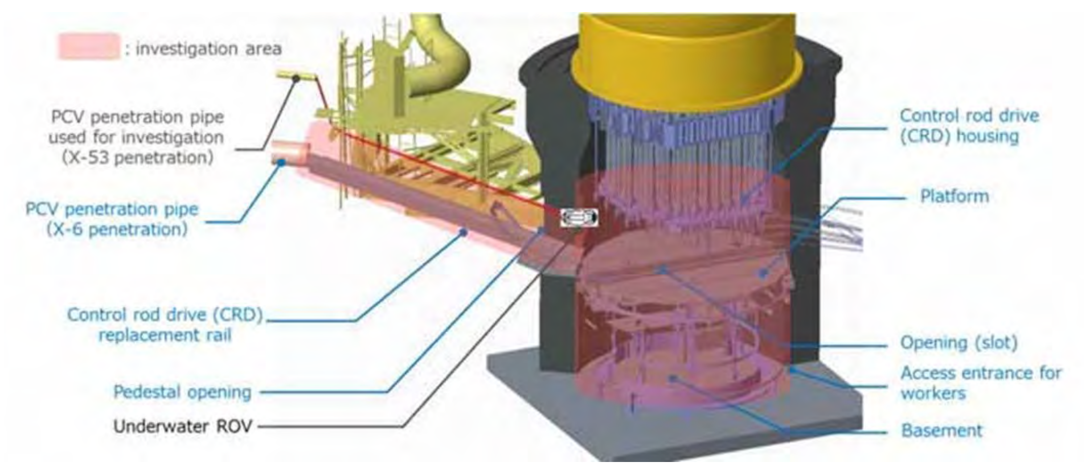


Figure 57. Area explored by the "Little Sunfish" submersible robot within the 1F3 PCV (Courtesy of TEPCO Holdings[209]).

Figure 58. Image of relocated CRD rail on the 1F3 platform with no grating; obtained on July 19, 2017 (Courtesy of TEPCO Holdings [209]).

Figure 59. Image of relocated material adhered to remaining 1F3 CRD housing and cable structures; obtained on July 21, 2017 (Courtesy of TEPCO Holdings [210]).

Figure 60. Image of material that had relocated beneath the 1F3 platform into the pedestal area; obtained on July 22, 2017 (Courtesy of TEPCO Holdings [211]).

5.2.3 Images within the Reactor Building

One important finding regarding ex-vessel behavior is the discovery by TEPCO Holdings that the sand cushion drain line is leaking in 1F1.[125] 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 61) indicate that the liner would not have been ablated through based on either the MAAP low pressure (LP) or MELCOR pour scenarios [49,50,52]; 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.

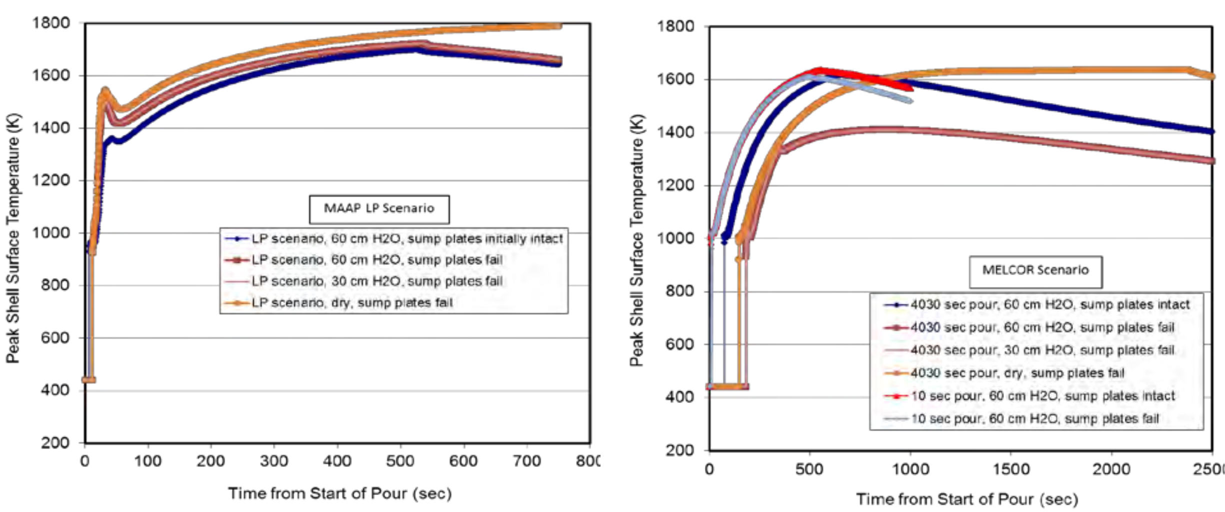


Figure 61. MELTSPREAD predictions of liner heatup due to heat transfer from impinging melt for 1F1 based on a MAAP low pressure (LP) scenario [50] (left) and MELCOR[49] (right) melt pour conditions. (Courtesy of ORNL [52])

5.2.4 Muon Tomography Evaluations

Evaluations have been completed using muon tomography techniques in each of the three affected units.

5.2.4.1 1F1 Examinations

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 62).[15] Using this approach, high density fuel should show up as dark regions in captured images due to muon attenuation. As shown in Figure 62, the core region appears to be essentially devoid of core material. The findings for 1F1 are consistent with previously described system-level code analyses.[49,50]

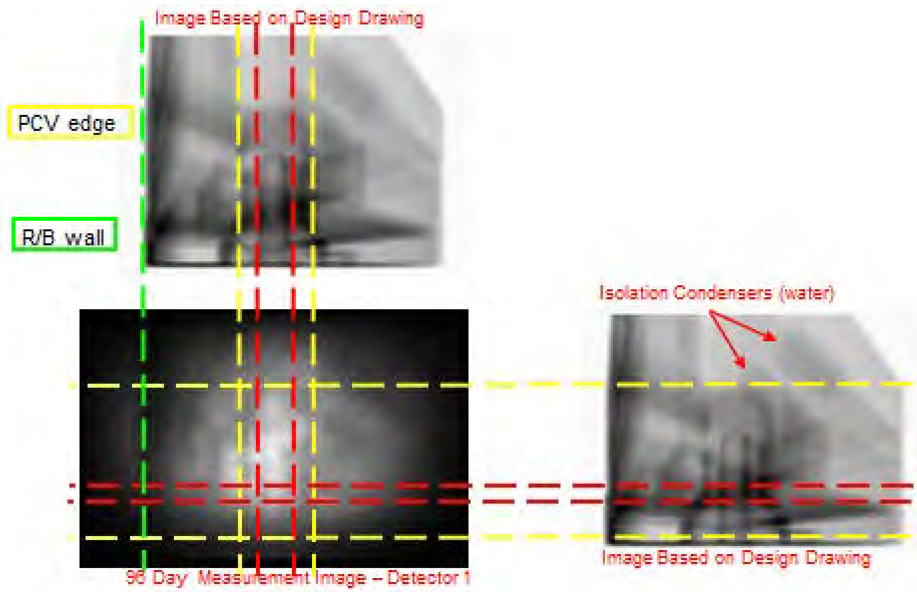


Figure 62. Images of 1F1 obtained using muon tomography with scintillation detectors (The lower left image is measured; the other two images were calculated. Dashed lines are provided to show location of identified geometrical features). (Courtesy of TEPCO Holdings [15])

5.2.4.2 1F2 Examinations

In addition, muon tomography measurements have been obtained from 1F2 [18]; see Figure 63. These measurements indicate the possible presence of core debris located at the bottom of the lower head (due to the dark nature of the material in that region). To a first order, this data can be used to estimate the amount of core material remaining in the RPV. For an assumed core mass of 240 metric tons, the debris volume would be ~34 m^3. Based on the dimensions provided in Figure 63, the volume of the spherical lower head is calculated to be ~33 m^3 (which neglects CRD structures in the lower head volume). The dark area to the right suggests that approximately one third to one half of the lower head is filled with dense core debris. Thus, this difference suggests that one half to two thirds of the core material is ex-vessel.

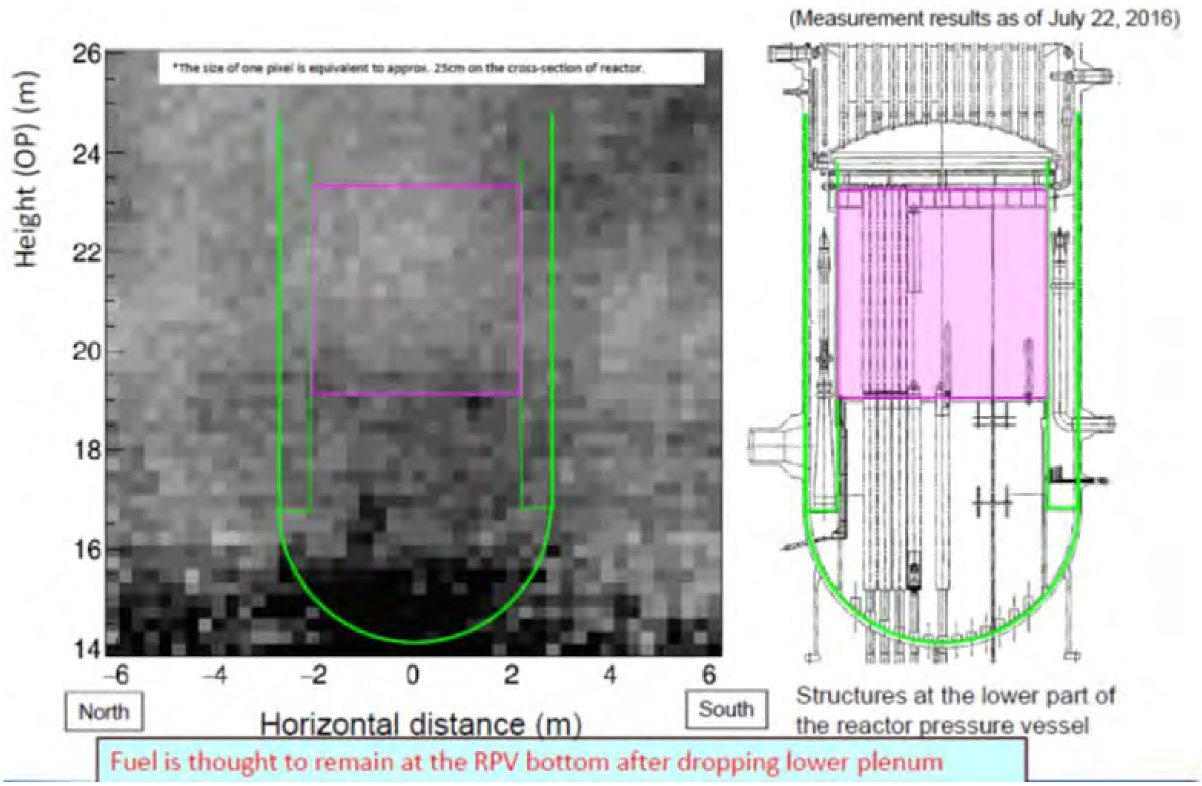


Figure 63. Images of 1F2 obtained using muon tomography with scintillation detectors. (Courtesy of TEPCO Holdings [18])

The muon tomography readings of the bottom of the RPV, in conjunction with qualitative information obtained from the most recent entry into the 1F2 containment indicates that there is most likely a significant amount of corium both within the bottom of the RPV and in the pedestal region of the containment. The results of the muon tomography scan of the bottom of the 1F2 RPV can be seen in Figure 64 indicating the likely presence of debris.[212]

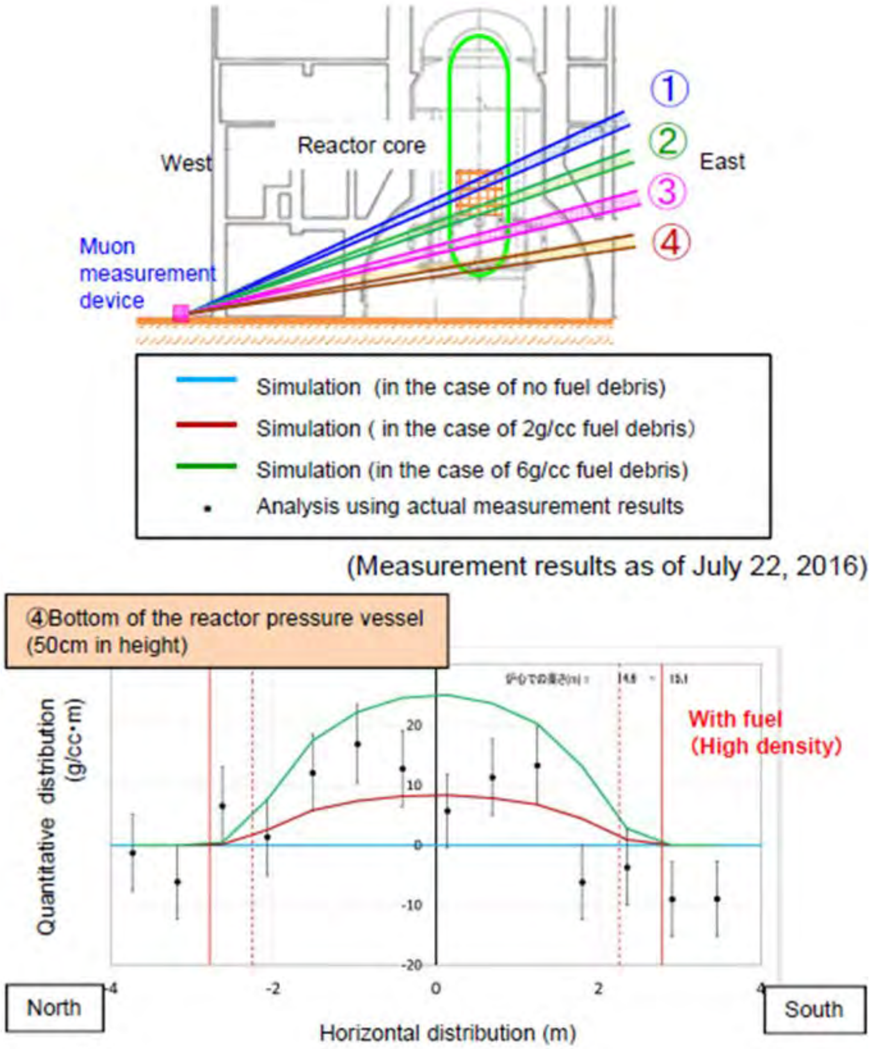


Figure 64. Muon tomography scan of the lower head of the 1F2 RPV (Scan 4). Data points near the upper (green) line indicate the presence of high density debris (between 2.0 and 6.0 g/cc). The presence of high density debris was only found in the muon tomography ray that passed through the lower head. (TEPCO Holdings [212])

The photos taken within the pedestal region of 1F2 during entry to the pedestal region do not show any RPV failures. However, available photos do not preclude RPV failure near the top of the hemispherical bottom head. For example, if the MELCOR debris configuration is assumed (see Figure 65), failing along the side of the RPV would lead to more "molten" (non-particulate) metallic debris to relocate ex-vessel. The metallic debris could form a molten pool when debris agglomerates in the lower plenum, situating itself on top of ceramic debris. The metallic layer has a higher heat flux to the RPV vessel wall, leading to potential failure earlier than areas within the lower plenum that are adjacent to ceramic debris. If this metallic layer were to fail the side of the lower head, then metallic debris and a portion of built-up ceramic debris would relocate ex-vessel, while a significant amount of particulate debris would remain in the bottom of the lower head.

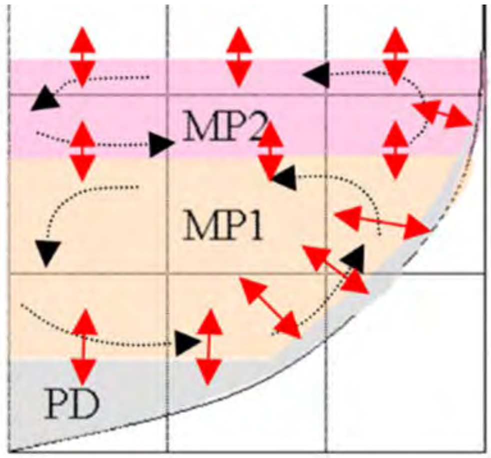


Figure 65. Treatment of molten pools and particulate debris by MELCOR within the lower plenum, the amount of particulate versus molten debris is dependent on the specific accident transient. Molten Pool 2 (MP2) is assumed to contain molten metallic material; Molten Pool 1 (MP1) is assumed to contain molten ceramic material; and Particulate Debris (PD) is assumed to contain particulate material. (Courtesy SNL [8])

5.2.4.3 1F3 Examinations

Starting in May 2017, muon tomography data were collected from 1F3. An interim report [213], issued on July 27, 2017, suggests the possibility that the amount of fuel debris remaining in the core and lower area of the RPV is less than what was observed in 1F2 (see Figure 66).

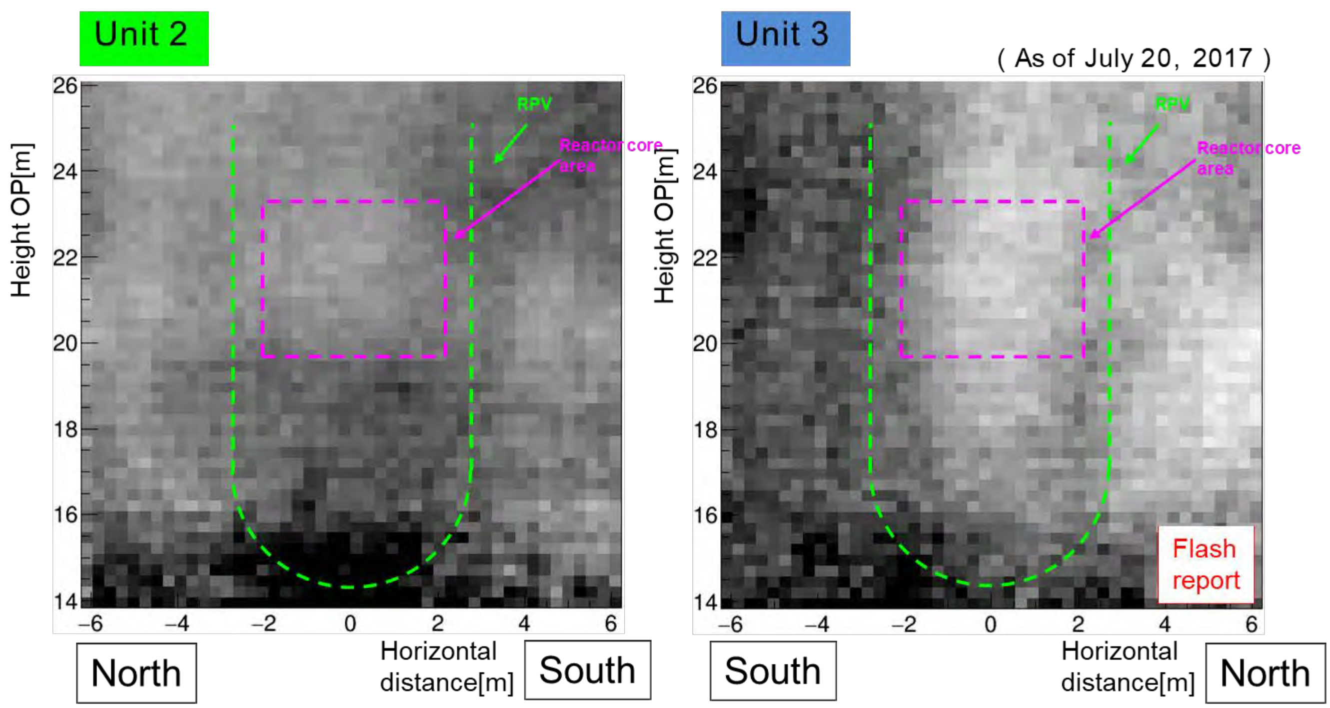


Figure 66. Comparison of muon tomography images obtained for 1F2 and 1F3 using scintillation detectors. (Courtesy of TEPCO Holdings [213])

5.2.5 Gas Cleanup System Measurements

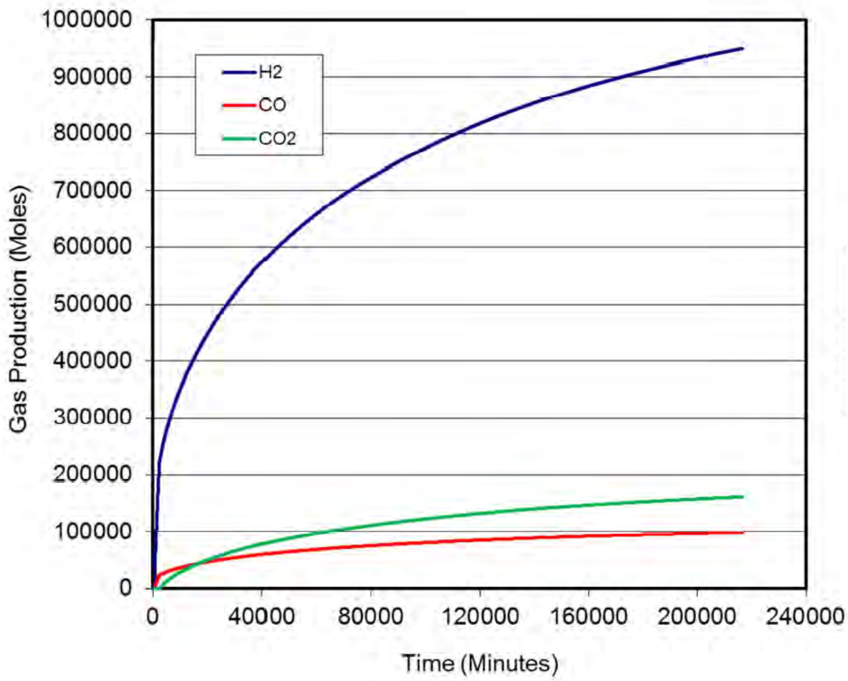
Aside from the three main data sources discussed above, additional data gathered by TEPCO Holdings that has been extremely useful for reactor safety evaluations are the PCV atmospheric composition measurements obtained from 1F1 and 1F2 a few months after the accidents occurred [214] (see Table 17). As discussed earlier, system-level code analyses [49, 50] 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 Holdings used these measurements to determine if H_2, CO, and CO_2 gases produced through MCCI were still present in the containment atmospheres.[202] 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 [214]

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 [33] 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 [50] 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.[33]

The results (Figure 67) 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 67. Expected long-term noncondensable-combustible gas production from ablation within the 1F1 sump if debris had not been quenched and stabilized. (Courtesy of ANL [33])

Given the data in Figure 67, 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 an important observation 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 Holdings 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.[33]

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 Holdings regarding ex-vessel coolability in 1F1 motivated the developers of the MAAP and MELCOR codes to integrate advanced debris coolability models (e.g., see [52]) 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,[16] TEPCO Holdings 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 Holdings found white sediment that was deposited by leakage within the HPCI PCV wall penetration; see Figure 68 and Figure 69.

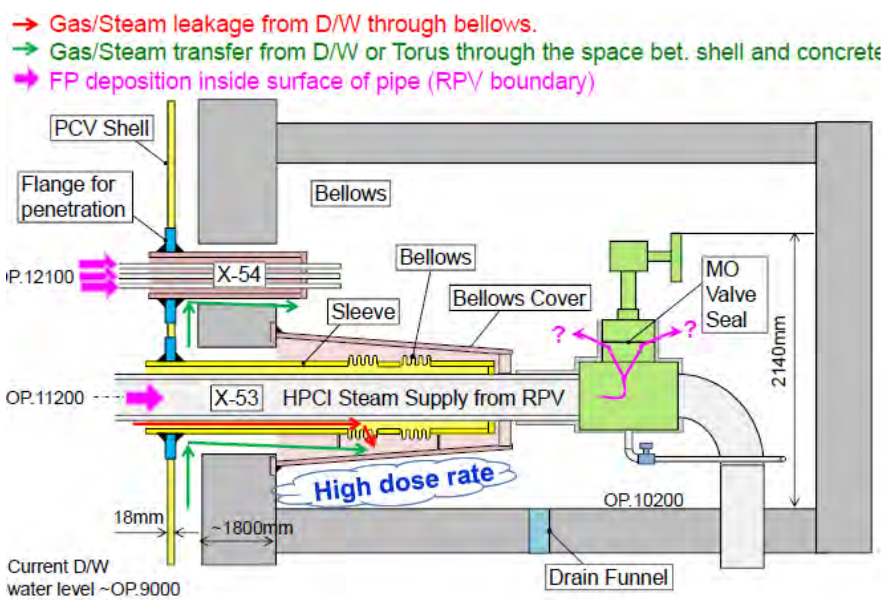


Figure 68. Depiction of HPCI pump steam supply penetration showing location of deposit with high dose rate. (Courtesy of TEPCO [16])

Figure 69. Photographs of leakage location and sediment. (Courtesy of TEPCO Holdings [16])

TEPCO Holdings indicates that the likely leakage point within this penetration is the bellows seal.[16] As shown in Figure 22, 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. As noted previously, this material dominates aerosol produced during core-concrete interaction, [203] particularly for the siliceous concrete type used in Fukushima plant construction. Aside from insulation, 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.

5.2.7 Contamination of the RCW System

Another important finding by TEPCO Holdings that provides valuable clues regarding likely debris locations is the fact that the Reactor Coolant Water (RCW) system is contaminated. As shown in Figure 70, the RCW is a closed-loop system that is intended to cool the water in the 1F1 sumps. The fact that coolant water within this loop is highly contaminated indicates that the piping boundary was likely breached. The presence of core debris within the sumps, coupled with the high PCV pressures experienced during the accident, would likely cause this type of breach and infiltration of fission products into the RCW system. Thus, this finding provides additional information that is consistent with RPV failure and core debris relocation into the sumps, as predicted by various system level code analyses [31,32] for this unit.

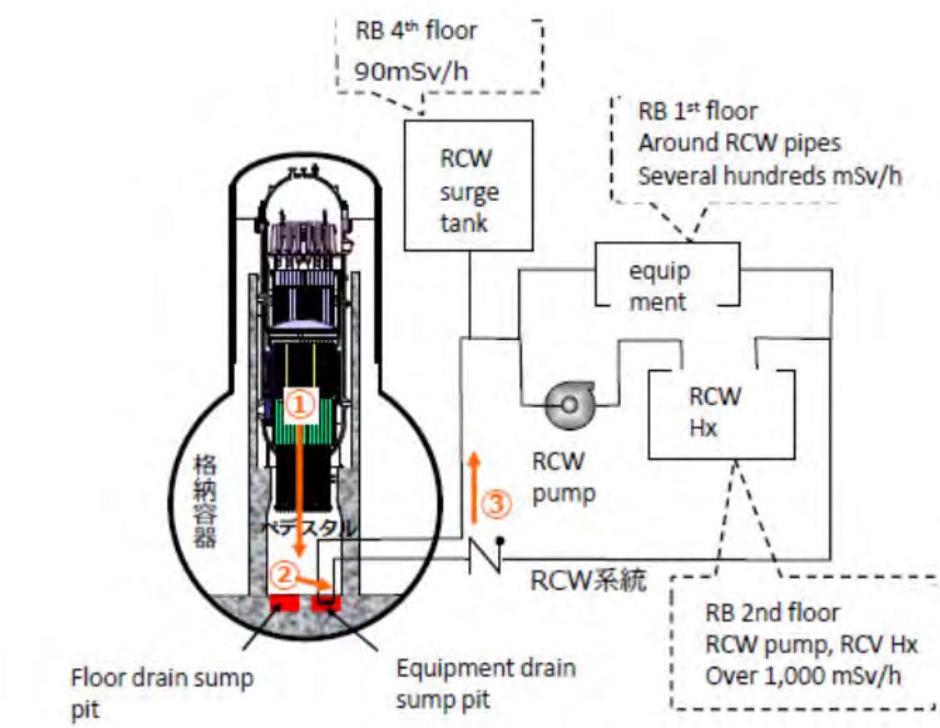


Figure 70. Illustration of 1F1 RCW system showing contamination levels within the system. (Courtesy of TEPCO Holdings [205])

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).
- Ongoing 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 camera inserted into the X-100B penetration found a relatively deep (i.e. ~ 30 cm) layer of sediment on the drywell floor. It is not clear if this material is all sediment or if there is a smaller amount of sediment covering core debris that spread from the pedestal doorway opening. Determining if core debris exists at this location would help to resolve modeling uncertainties related to ex-vessel core debris spreading behavior, in addition to providing insights related to the melt pour conditions that are strongly linked to in-vessel behavior.
- Additional photographs near the pedestal doorway in 1F1 show accumulations of material on the floor and in the sumps. The nature of this material is difficult to determine due to overlying sediment, but splatter material on structures looks similar to core debris splattered by melt eruptions in experiments.
- 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).
- The findings that the closed loop RCW sump cooling system in 1F1 has been breached and is highly contaminated suggest that high temperature core debris entered the sump, causing the system to fail.
- Recent observations in 1F2 indicate the existence of black accumulations on the CRD exchange rail near the CRD scaffolding. If the material is found to consist of core debris, this observation would be consistent with lower head failure as has been predicted by several severe accident codes.

Muon Tomography Investigations:

- Results from muon tomography suggest 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. There have been no direct determinations of the location of the core debris. Observations obtained with remote cameras have provided some clues for all three units. Muon tomography images are also providing data on debris locations, but the resolution of the images is limited. Finally, TEPCO Holdings 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 [49,50] as well as separate effect [52] code analyses have provided tangible predictions for evaluation against the debris end-state information being obtained by TEPCO Holdings. 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 Figure 41 through Figure 43. 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.[51]

Also, the findings from 1F2 and 1F3 suggest that a non-negligible amount of core debris may be held up on structures below the reactor vessel. System analysis codes should be exercised assuming a range of core debris holdup in a situation that is not cooled by water to investigate the impact of heat sources not covered by water on PCV gas phase temperature and pressure.

Area 3 Recommendation 2:

Repeat the MELTSPREAD-CORQUENCH analysis that was originally done for 1F1 [52] 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 [11], 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 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 Holdings has actively and successfully pursued; i.e., muon tomography and robotics inspections. These methods have already proven to be valuable. Based on insights obtained from evaluations of current information, additional suggestions are offered at this time:

Area 3 Suggestions:

- 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.
- Perform chemical analysis of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor.
- Evaluate the nature of the material below the upper surface of the debris at the X-100B penetration location in 1F1 to determine if it is additional sediment or other material such as core debris.
- Perform chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location
- Perform chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'
- Obtain additional images from examinations in 1F3 X-53 penetration.

The list of information needs in Appendix C was updated to reflect these additional items (e.g., see RB-14 and PC-17 through PC-21 in Appendix C). Discussions [215] with TEPCO Holdings indicate that there are plans to implement several of these requests from U.S. experts.

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 Holdings examinations to support these activities are also identified.

6.1 Questions for Reactor Safety and D&D

Available information was evaluated by U.S. 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, which can be used in guidance for severe accident mitigation?
- What are the 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, U.S. 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 Table 19 and Table 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 provide insights about combustible generation, transport, and combustion. In addition, TEPCO Holdings reports evaluating damage associated with explosions at the affected units [216 through 218] and TEPCO Holdings unsolved issues reports [34 through 38] contain important information on this topic. Data from plant instrumentation was 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[ll]	Use[mm]	Data Available[nn]
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	A
	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

[ll] See list of acronyms.

[mm]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).

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

Table 20. Area 4 information needs from the PCV

Item	What/How Obtained[oo]	Use[pp]	Data Available[qq]
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). [rr]	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).[ss]	AE, AM, DD	A
	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[ss]	AE, AM, DD	A
PC-4	Photos/videos of 1F1, 1F2, and 1F3 recirculation lines and pumps	AE, AM	NA

6.2.1 TEPCO Holdings Reports

In 2011, two of the reports issued by TEPCO Holdings [216, 217] evaluated 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 71 for 1F1; Figure 72 through Figure 74 for 1F3; and Figure 75 for 1F4).

[oo]See list of acronyms.

[pp]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).

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

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

[ss]Although some images have been obtained; images do not indicate if RPV failed. 1F2 investigations [see Section 5.3] indicate the presence of possible ex-vessel debris, but it has not yet been possible to extract samples for evaluating composition.

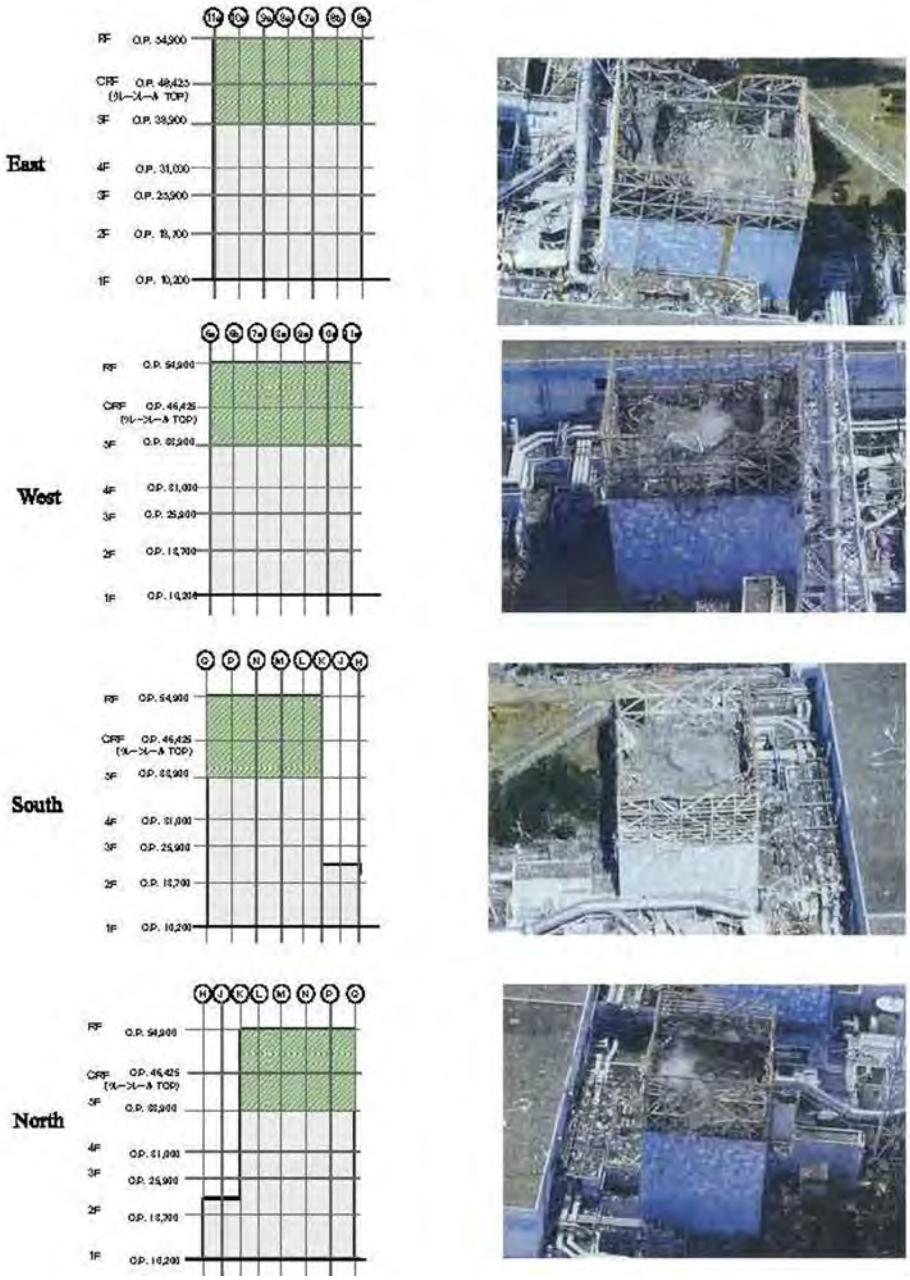


Figure 71. 1F1 reactor building damage following explosion. (Courtesy of TEPCO Holdings [216])

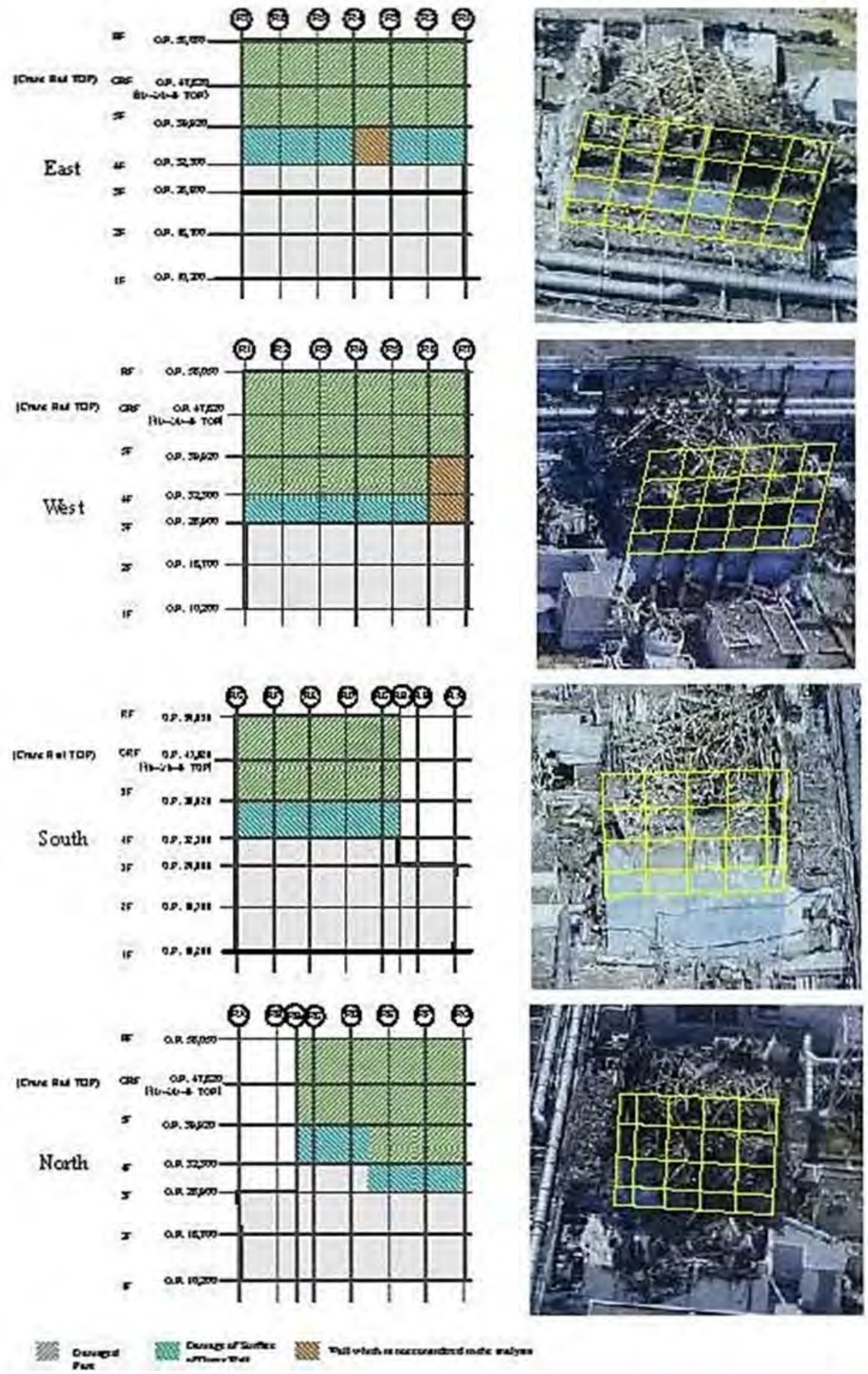


Figure 72. 1F3 reactor building damage following explosion. (Courtesy of TEPCO Holdings [217])

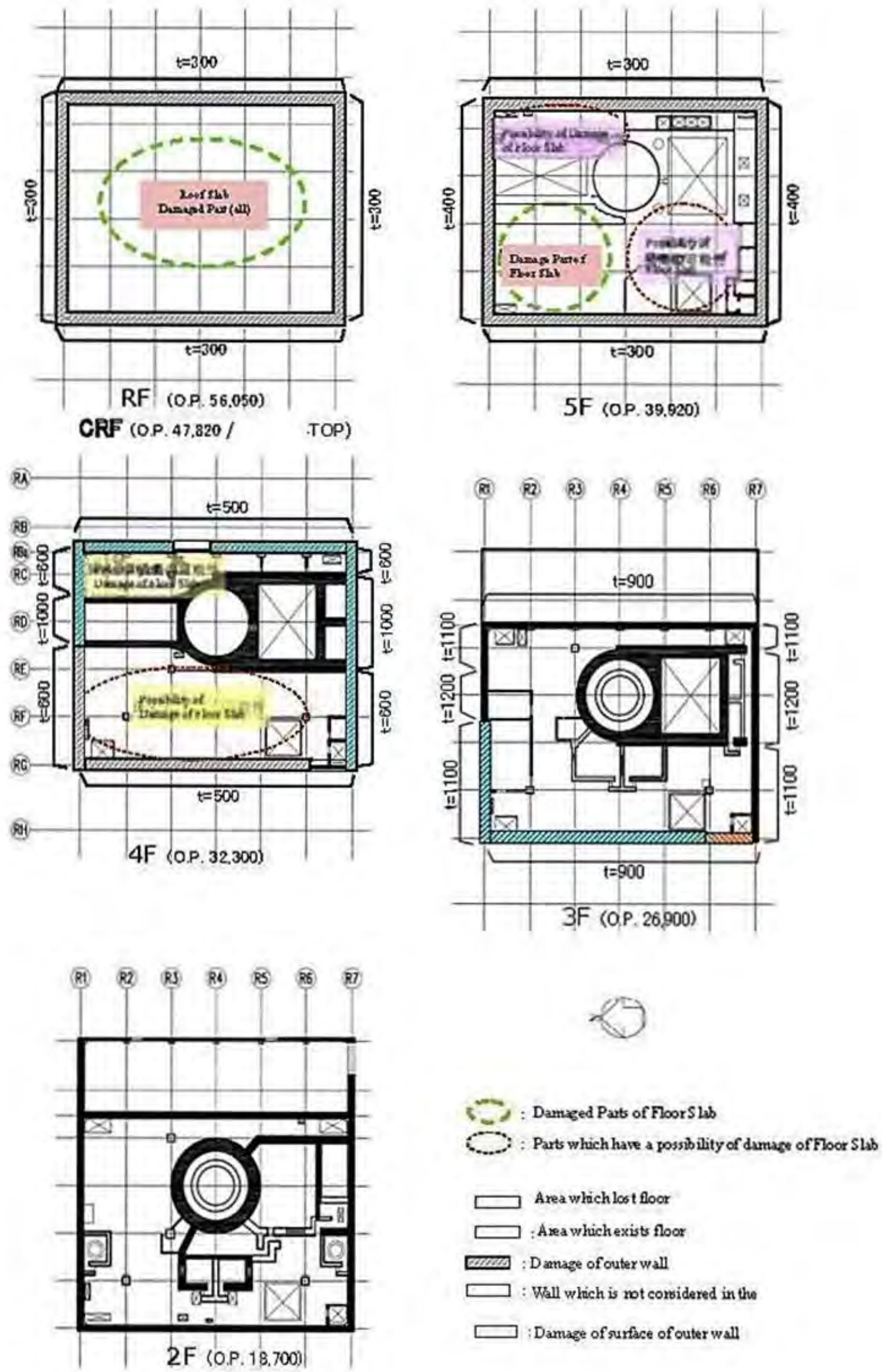


Figure 73. Damaged areas on 1F3 reactor building floor plan. (Courtesy of TEPCO Holdings [217])

Figure 74. Damaged areas on 1F3 reactor building floor. (Courtesy of TEPCO Holdings [16])

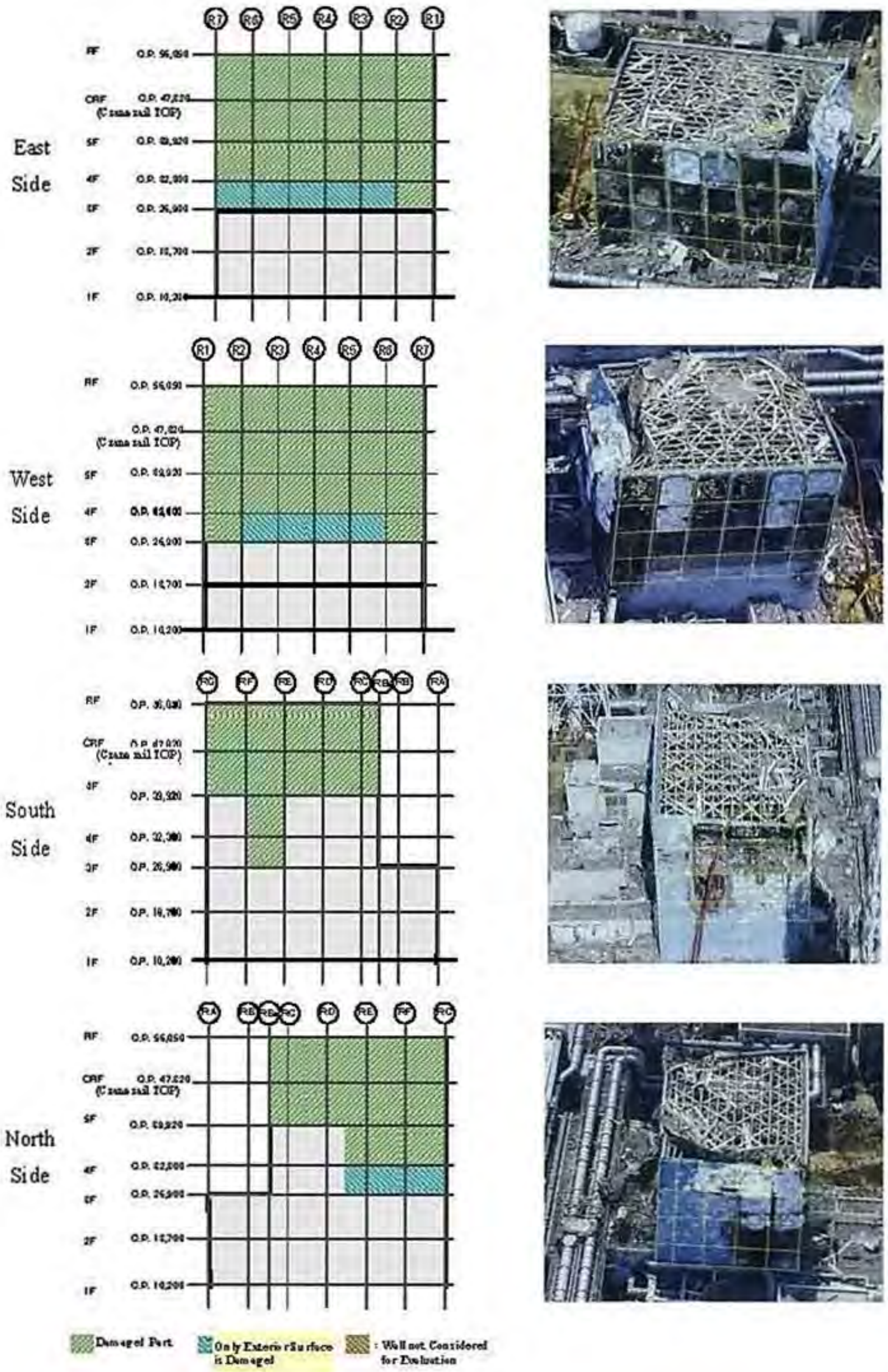


Figure 75. 1F4 reactor building damage following explosion. (Courtesy of TEPCO Holdings [216])

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.[218] 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 significant level (see Figure 76).

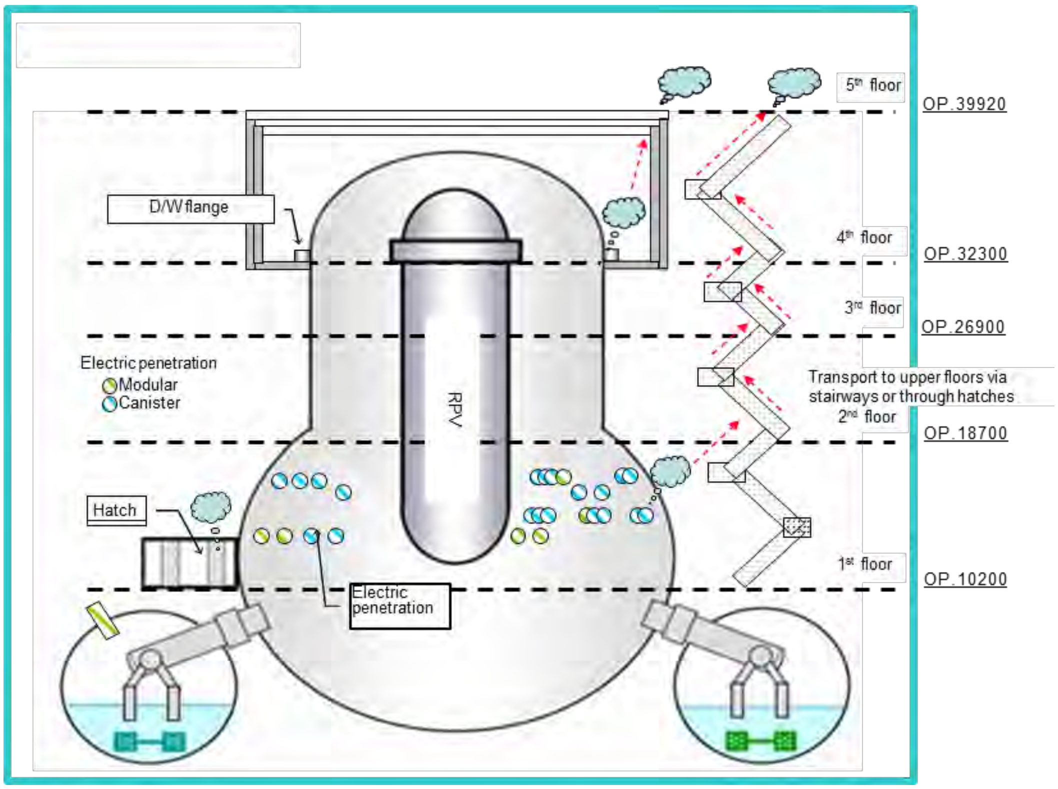


Figure 76. Inferred leakage paths; flow paths differ from 1F1 and 1F3 due to system configuration. (Courtesy of TEPCO Holdings [219])

As documented by TEPCO Holdings,[216] 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.[220] 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.

An analysis of the 1F1 reactor building explosion has been performed by TEPCO Holdings using the Computational Fluid Dynamics (CFD)-based FLACS code to confirm that ignition was not initiated from the 4th floor due to leakage from isolation piping.[21] Two cases were performed. Case 1 assumed an 8.3% H_2 concentration in the 5th floor with ignition initiated from the center of the shield plug. Case 2 assumed an 8.3% H_2 concentration in the 5th floor and a stratified layer of 20-30% H_2 beneath the ceiling of the 4th floor. Ignition was initiated from the stratified layer. The thickness of the stratified layer was about half of the 4th floor height. Based on the speed and direction of the blast generated by the code, it was concluded that the ignition was initiated from 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 27). 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.[218]   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 [217], TEPCO Holdings observed the following damage:

- Collapsed steel framework and concrete were piled up on, and above, the 5th floor (Figure 72);
- 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 73);
- The 4th floor walls 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 74).

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 on 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.[218] 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.[216] Most walls on the 4th floor and some walls on the 3rd floor were damaged (Figure 75).

Evaluations of the explosion at 1F4 have led TEPCO Holdings to conclude that vented gases, including hydrogen, flowed from 1F3 into 1F4 (Figure 77). 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 78).
- ***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 79).
- ***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.[tt]

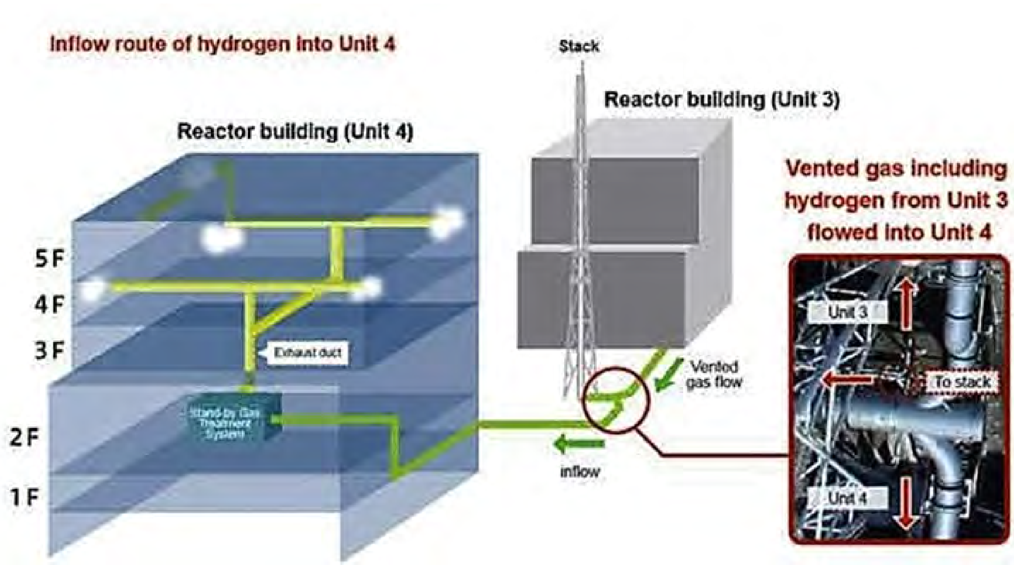


Figure 77. Hydrogen transport path from 1F3 to 1F4. (Courtesy of TEPCO Holdings [218])

[tt] 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.

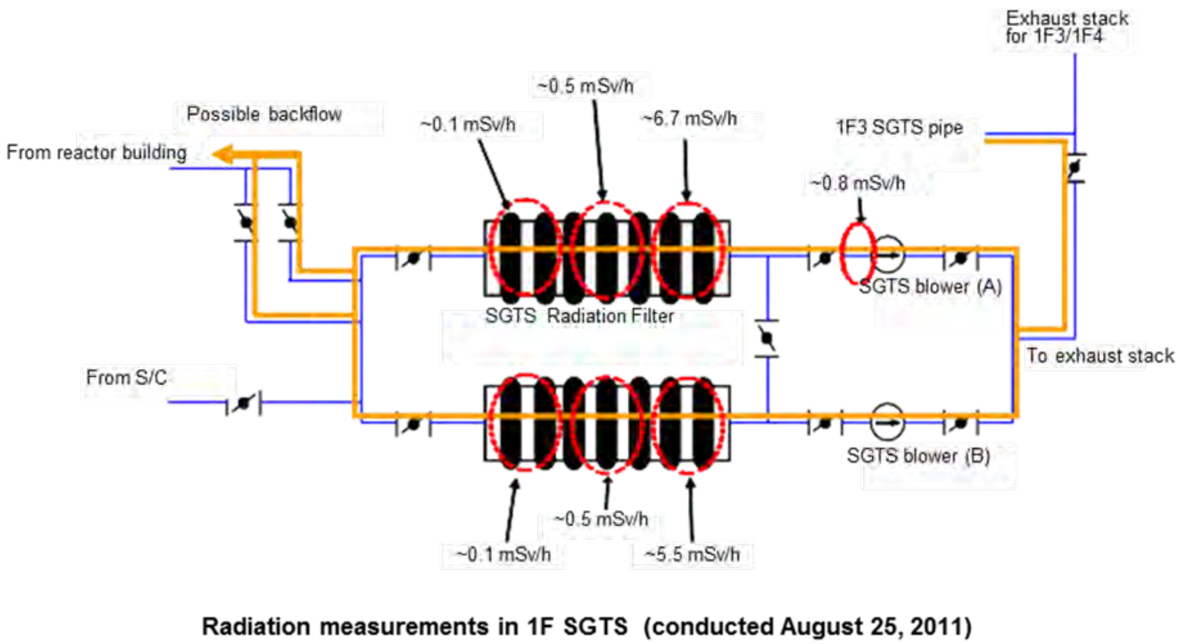


Figure 78. 1F4 SGTS radiation measurement results. (Courtesy of TEPCO Holdings [218])

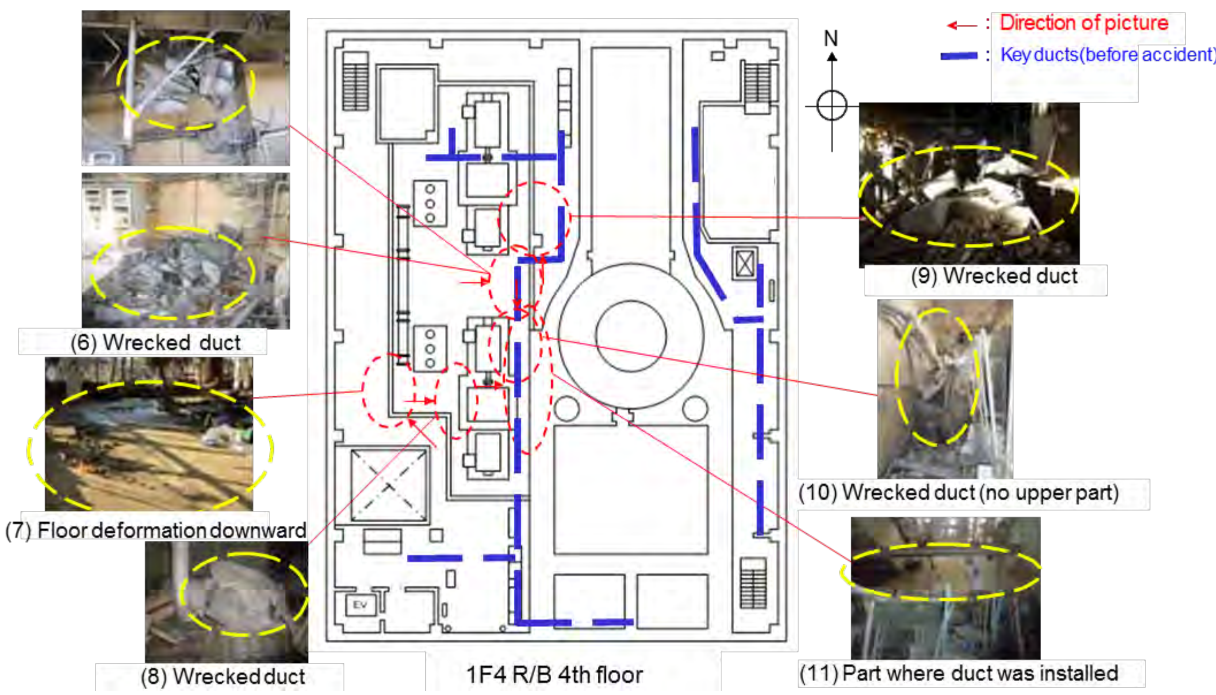


Figure 79. Field investigation of the 1F4 4th floor (Courtesy of TEPCO Holdings [218])

6.2.5 Video Capturing Explosions

Figure 80 shows one-second interval snapshots from videos capturing the 1F3 explosion of 1F3 for the first 9 seconds (about the duration of the explosion). [220] Figure 81 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.

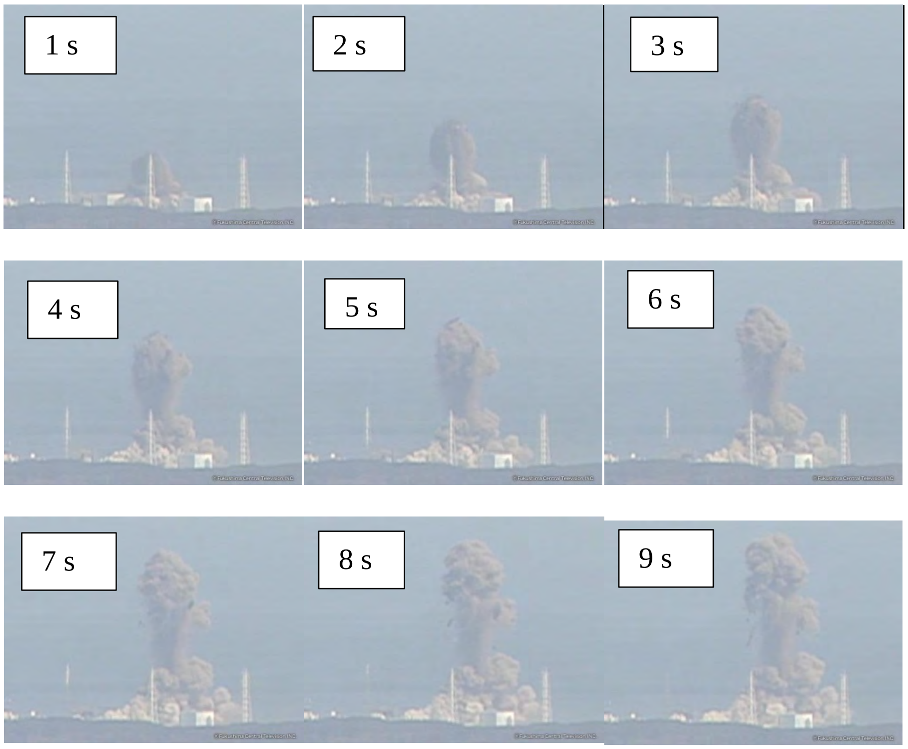


Figure 80. Images of 1F3 explosions at 1-second intervals. (Courtesy of FCT [221])

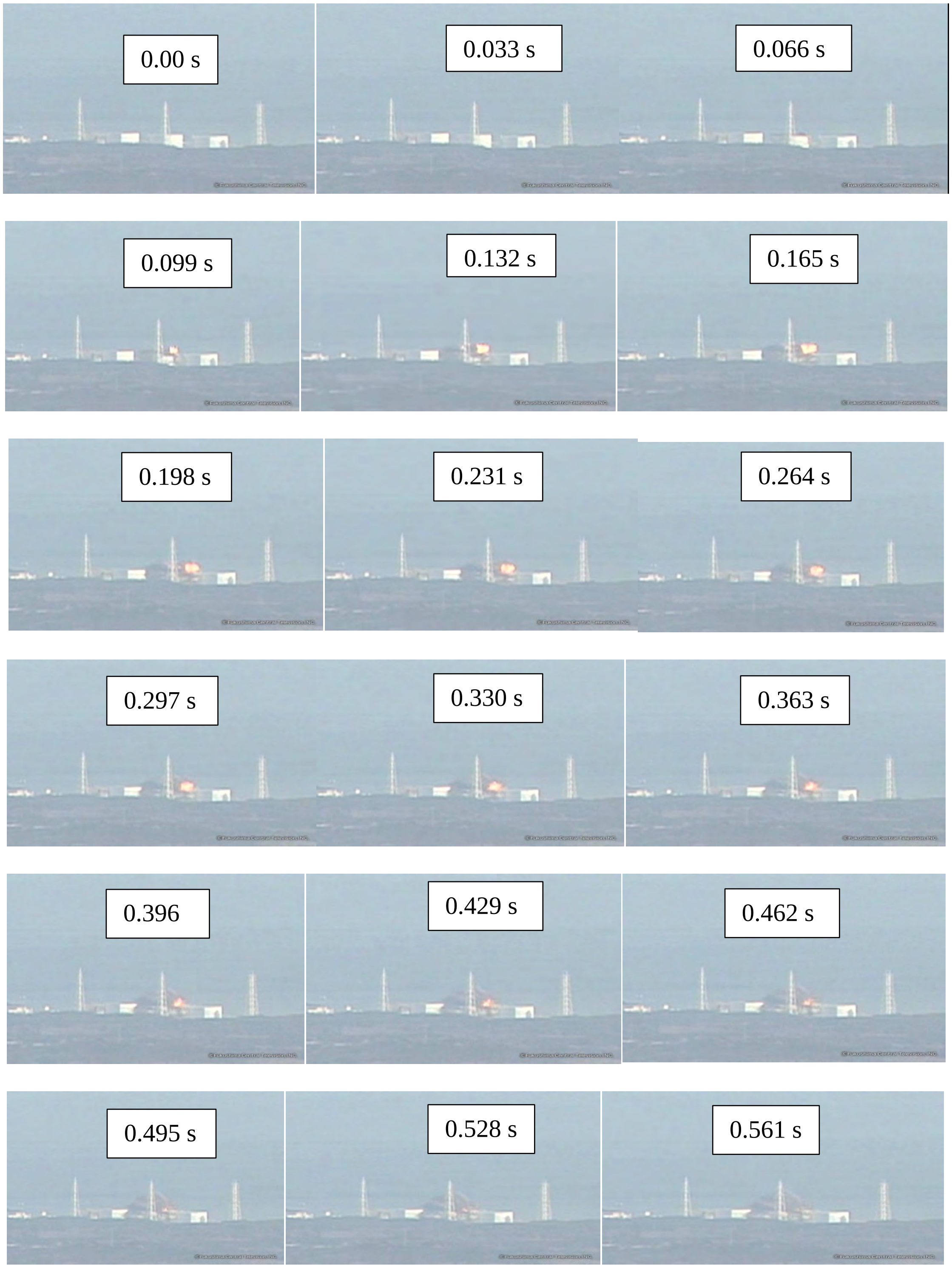


Figure 81. Images of 1F3 explosions during an appearance of a flash fire (Courtesy of FCT [221])

U.S. and Japanese expert evaluations of information related to hydrogen combustion [21,222,223,224] 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.[225] A video of the explosion indicates the presence of a condensation shock wave that was not fast enough to transition to detonation. As shown in Figure 82, 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 80). 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 (5^{th} floor). The flame front apparently propagated to the 4^{th} floor (based on damage seen in Figure 72) and resulted in the deflagration of flammable gaseous mixture accumulated in the 4^{th} 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 2^{nd} explosion appeared in darker color of dust and debris than that of the 1^{st} explosion which appeared white (in the image) and remained at lower elevations close to the building (Figure 80). This is a strong indication that the combustible gases involved in the 1^{st} and 2^{nd} explosions at 1F3 came from different sources as discussed in Section 6.2.6.

1F1 **1F 3**

Figure 82. Images of 1F1 explosion compared with 1F3 explosion. (Courtesy of FCT [221])

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 83). 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 5^{th} 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 5^{th} floor (and 4^{th} 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 gross 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. [Neglecting the presence of rebar, it is noted that it would require 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 the form of a flash fire [visible from the video snapshot at the very beginning (less than 0.5 s) of the 1F3 explosion transient shown in Figure 81]. At the beginning, the combustible gas jet was just forming. 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 from the jet came out, the flash fire disappeared and the energetic second explosion began its process as shown in Figure 80. 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 80). 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).

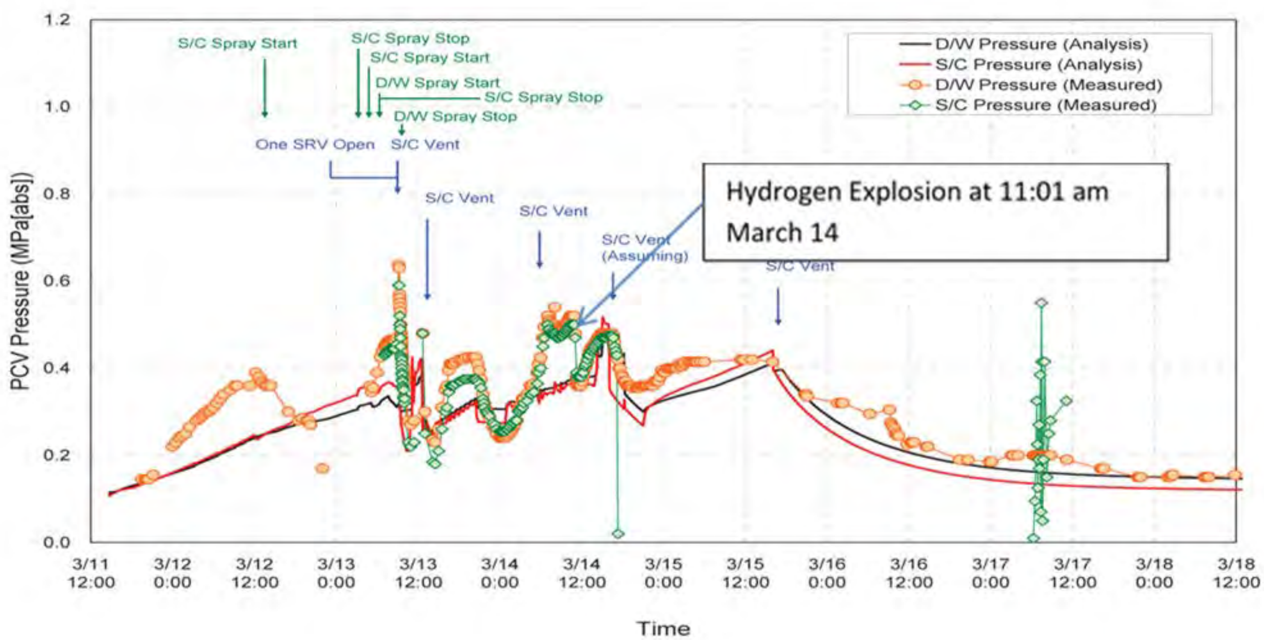


Figure 83. 1F3 PCV pressure - Rapid drop in PCV pressure coincides well with the timing of 1F3 explosion at 11:01 am March 14, 2011 (Courtesy of TEPCO Holdings [225])

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 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 events at Daiichi provide a real-world example of hazard propagation on a multi-unit site.*** As mentioned in Section 6.2.3, the explosion at 1F3 damaged mobile equipment, such as fire engines and hoses, being readied for core cooling of 1F2 to the extent that it could no longer be used. The timing of the 1F3 explosion coincided with the failure of 1F2 RCIC that had been operating for nearly 3 days (Figure 84). Following this event, 1F2 experienced a period of no core cooling for about 6 hours. The meltdown of the 1F2 core might have been avoided if the explosion at 1F3 had occurred at a different time. Thus, these events are a real-world example of hazard propagation on a multi-unit site in that an event at a damaged unit propagated to an adjacent intact unit and resulted in core damage.

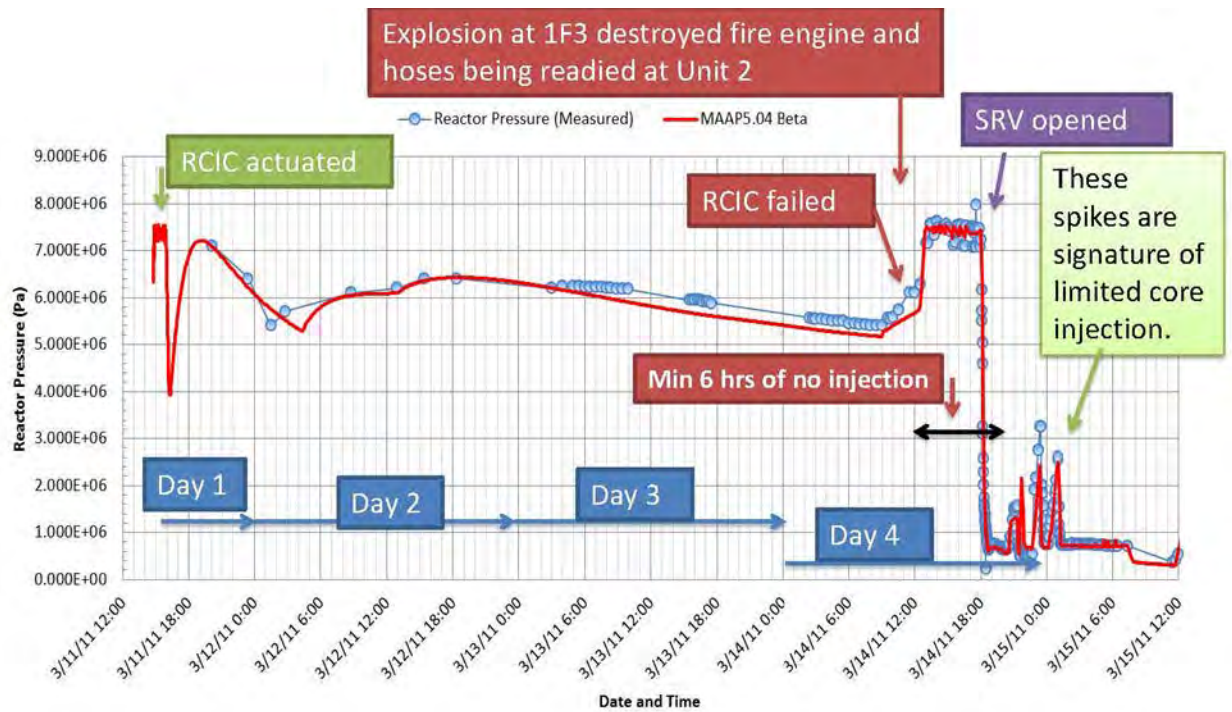


Figure 84. Timing of 1F2 RCIC failure and 1F3 explosion (Courtesy of Fauske and Associates)

- ***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. More 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 Holdings 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 [51] 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.[51] Evaluations with MAAP5 tend to find that this has important consequences for the development of flammable conditions in the 1F1 and 1F3 reactor buildings. Figure 85 and Figure 86 compare results from a MAAP5 uncertainty evaluation of the 1F3 accident.[54] 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.[97]

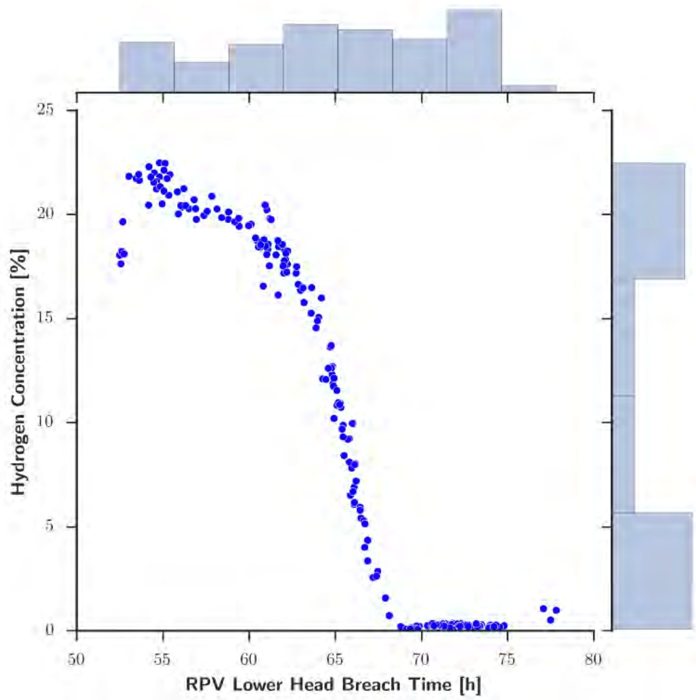


Figure 85. MAAP 1F3 modeling uncertainty evaluation: refueling floor hydrogen build-up at time of 1F3 reactor building explosion. (Courtesy of EPRI [54])

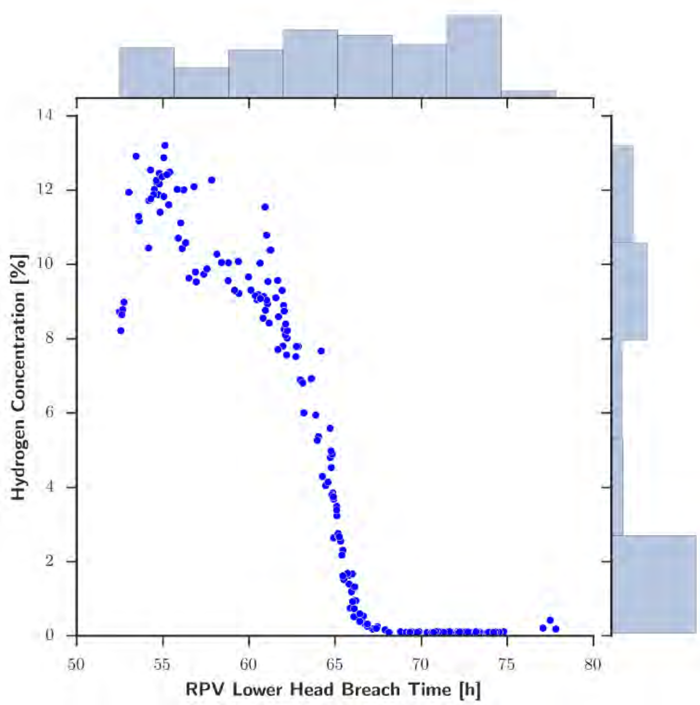


Figure 86. MAAP 1F3 modeling uncertainty evaluation: fourth floor hydrogen build-up at time of 1F3 reactor building explosion (Courtesy of EPRI [54])

From the observed explosions, a rough approximation or 'ballpark estimate' was obtained for the total amount of H_2 that must be generated in 1F3 to support its own explosions and an explosion at 1F4.[226] This estimate provides a constraint on the timing of vessel failure that must be predicted by MAAP5 and MELCOR calculations. Thus, this rough estimate provides additional insights on vessel failure time by making reasonable (but highly uncertain) assumptions on the hydrogen concentrations at the time of the 1F3 and 1F4 explosions. It was also estimated that a venting time of slightly more than an hour is sufficient to transfer necessary amount of hydrogen from 1F3 to 1F4.[226]

Based on the magnitude of explosions at 1F3 and 1F4, a ballpark estimate of the necessary amount of H_2 required in each explosion was estimated by assuming plausible H_2 concentrations at the time that the explosions occurred. Figure 87 depicts the process, and Table 21 lists the ballpark numbers and associated assumptions of H_2 concentrations used to obtain these estimates. Also, the ballpark estimate of the necessary amount of H_2 that must be vented to the environment via the common vent stack and the necessary amount that must be leaked from 1F3 to 1F4 were obtained. The transfer of H_2 from 1F3 to 1F4 during venting is assumed to be subject to a 25% leakage of the vented gas to 1F4 and a 75% discharge to the environment.[224] The final number in Table 21 is the total amount of H_2 that must be generated in-vessel and ex-vessel. This number is then subtracted by the largest typical amount of in-vessel H_2 generation calculated by MAAP5 (550 kg) and MELCOR (2000 kg).[98, 226] This required time is then subtracted from the time when last venting from 1F3 was accomplished prior to the explosion in 1F4, e.g., at 9:04 pm on March 14, 2011, e.g., about 10 hours after the 1F3 explosion time of 68.3 hours.[98] The result is the latest time required for vessel failure. The time required for vessel failure for MAAP5 is shown in Table 22 and for MELCOR in Table 23. Because ex-vessel CO generation in terms of moles is not significant, it is ignored in this estimate.

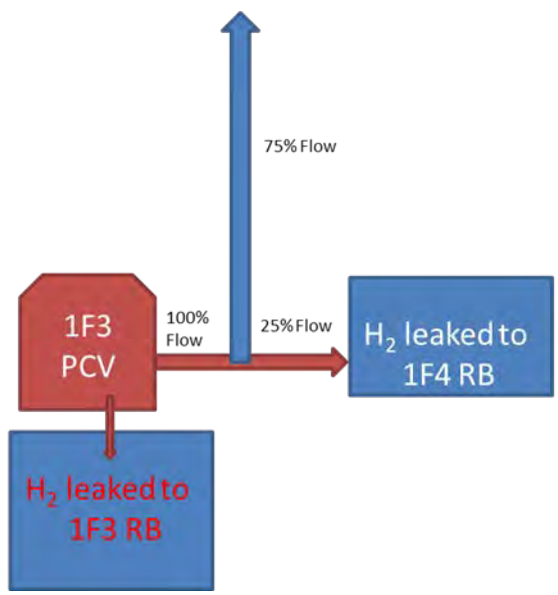


Figure 87. Process used to obtain ballpark estimate of the latest vessel failure time (Courtesy of FAI [226])

Table 21. Ballpark estimate of required hydrogen generation

Source of H_2	Amount of H_2 (kg) based on assumed H_2%	Assumed H_2 Concentrations % at time of Explosion
H_2 in 1F3 RB	370	~20%H_2 in 5F, 15%H_2 in 4F
H_2 in 1F4 RB	370	~15%H_2 in 5F, 4F, 3F
H_2 in 1F3 PCV	640	5 bars, ~30% H_2 in drywell, ~40% H_2 in wetwell
H_2 vented through 1F3-1F4 common stack	3 x H_2 leaked to 1F4 RB =3 x 370= 1110 kg	25% of vent flow from 1F3 leaked to 1F4 RB 75% of vent flow to environment [224]
Total H_2 leaked and vented	370+370 +1110=1850 kg	
Total H_2 that must be generated in-vessel and ex-vessel	1850+640 = 2490 kg	

Table 22. Ballpark estimate of required latest vessel failure time for MAAP5 analysis of 1F3

In-vessel H_2 generation	550 kg (MAAP5.04) [226]
Required ex-vessel H_2 generation mass	2490-550 = 1940 kg
Ex-vessel H_2 generation rate	100 kg/hr [98]
Time required for ex-vessel generation to generate 1940 kg H_2	1940 kg/100 kg/hr = 19.4 hr
Required latest vessel failure time (Assuming vent line was not damaged by 1F3 explosion)	= Time after 1F3 explosion when last venting was accomplished – require ex-vessel time = (68.3+10) hr – 19.4 hr = 59 hr from accident initiation time = 3/14/2011, 2:00 am

Table 23. Ballpark estimate of required latest vessel failure time for MELCOR analysis of 1F3

In-vessel H_2 generation	2000 kg MELCOR [98]
Required ex-vessel H_2 generation mass	2490-2000 = 490 kg
Ex-vessel H_2 generation rate	42 kg/hr [98]
Time required for ex-vessel generation to produce 475 kg H_2	=490 kg/42 kg/hr = 11.7 hr
Latest vessel failure time (vent line is not damaged by 1F3 explosion)	= Time after 1F3 explosion when last venting was accomplished – required ex-vessel time = (68.3+10) – 11.7 = 67 hr from accident initiation time = 3/14/2011, 10:00 am

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 Station Blackout (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 Table 19 and Table 20, there are still information needs that have not yet been met in this area. 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 Holdings will provide these insights.

7. SUMMARY AND INSIGHT IMPLEMENTATION

Information obtained from Daiichi is required to inform D&D activities. This section summarizes the examination information evaluated by the U.S. expert panel and the recommendations formulated from these evaluations. In addition, this section identifies on-going and completed actions to use insights gained from forensics examinations to reduce severe accident modeling uncertainties and confirm severe accident management guidance. Activities to implement insights are beneficial to the U.S. 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 U.S. 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

Significant examination information is already available for evaluation. In the U.S. forensics effort, the expert panel agreed to focus evaluations in the four areas identified in Table 24. This table also lists the types of information available for evaluation in each area. As indicted in Table 24, available information is primarily visual images, data from plant instrumentation, radiation surveys, and isotopic sampling.

Table 24. Evaluation areas and types of evaluated examination information

Area	Types of Examination Information Evaluated
Area 1 - Component Degradation	• Visual information (photos and videos gathered using robotic examinations and stand-off methods such as muon tomography). • Sampling. • Dose rate measurements. • Water level and temperature measurements. • TEPCO Holdings reports documenting unconfirmed 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, of soil, and evaluations of 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 Holdings reports evaluating damage associated with explosions and reports evaluating unconfirmed and unresolved issues. • Data from plant instrumentation.

Forensics evaluations by the expert panel led to the identification of several recommendations for future U.S. RST pathway activities. Table 25 lists the recommendations identified 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 25. Recommendations for future U.S. 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 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 7.
	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 Holdings. This recommendation is not currently identified as an information need. As more information becomes available, however, the panel will identify specific places.
	The expert panel should consider an exploratory exercise in modeling the reference RPV water level reference leg in codes such as MELCOR and MAAP.
	The expert panel should place more emphasis on reviewing available examination information related to spent fuel (i.e., assemblies, pools, casks, etc.) at Fukushima Daiichi. Experts should also consider existing U.S. research that may aid in the decommissioning efforts and identify inspections at Fukushima Daiichi that may benefit ongoing U.S. activities.
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 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.
	Like Area 1 Recommendation 3, concisely compare the predicted conditions by both MAAP and MELCOR at the MSIV (temperature, pressure) for 1F2 and 1F3.
	Like Area 1 Recommendation 4, the expert panel continues to be interested in examination information of MSIV room components. Specific examination needs to support these evaluations are identified in RB-10 and RB-13 of Appendix C.
	The expert panel recommends that the U.S. Forensics Effort continue to evaluate information obtained from examinations of RPVs within each unit impairment location. In particular, additional visual information would be useful in the Area 2 Recommendation 1 sensitivity studies.

Area	Recommendations
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 Figure 41 through Figure 43. 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. Also, the findings from 1F2 and 1F3 suggest that a non-negligible amount of core debris may be held up on structures below the reactor vessel. System analysis codes should be exercised assuming a range of core debris holdup in a situation that is not cooled by water to investigate the impact of heat sources not covered by water on PCV gas phase temperature and pressure.
	Repeat the MELTSPREAD-CORQUENCH analysis that was originally done for 1F1 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 Section 5, 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 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 relocation and debris bed morphology be continued with the goal of reducing uncertainties.
	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.
	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.

The expert panel also developed suggestions for additional examination information. These suggestions are summarized in Table 26. 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 questions of interest. For Area 1, the panel explicitly requested that TEPCO Holdings experts continue to review summary information related to component degradation developed by U.S. experts. In Areas 1 and 3, several new information needs were identified. This item has been added to examination needs as RB-15, PC-17, PC-18, PC-19, and PC-20, and PC-21 in Appendix C. Discussions with representatives from TEPCO Holdings indicate that there are plans to implement several of these requests from U.S. experts.

As discussed in Section 1.3, the expert panel identified a fifth area, “Plant Operations and Maintence,” for evaluation in future years. This area will cover a range of topics of interest to industry, starting with instrumentation survivability information obtained from examinations at Daiichi. To assist U.S. efforts in this area, representatives from TEPCO Holdings agreed to provide additional information regarding instrumentation qualification envelopes and estimated conditions that sensors experienced during the events at Daiichi (see PC-7 and PC-8).

Table 26. Suggestions for additional examinations

Area	Suggestions for Additional Examination Information
Area 1 - Component Degradation	To facilitate updates to Table 7, the expert panel has requested that TEPCO Holdings 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 Holdings 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.
	As dicussed in Sections 3.2.1.1 and 3.2.1.2, the RCW system may have played a role in the 1F1 accident progression. Examination information identified in Appendix C and other information previously obtained by TEPCO Holdings (i.e., dose surveys around the surge tank, system water level, images of system components, etc.) may provide insight into its role during the accident.
Area 2 - Dose Surveys and Isotopic Surveys and Samples	Continue planned additional isotopic evaluations.
Area 3 - Debris Endstate	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.
	Chemical analysis of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor.
	Evaluate the nature of the material below the upper surface of the debris at the X-100B penetration location in 1F1 to determine if it is additional sediment or other material such as core debris.
	Perform chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location
	Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'
	Additional images from examinations in 1F3 X-53 penetration.
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).[2,22,65,81,88,227,228] The Emergency Procedures Committees for the BWROG maintain generic Severe Accident Guidelines (SAGs) and for the PWROG maintain generic Severe Accident Management Guidelines (SAMGs).[uu] New information is continuously

[uu] The BWROG SAGs and PWROG SAMGs are terms used by the owners group to refer to their generic SAG.

evaluated as new information becomes available (e.g., examination information from the Fukushima accidents and from issues identified by member utilities), and the generic SAG is periodically enhanced based on this new information. Plants implement the generic SAG per individual plant specific design features within two refueling cycles or 36 months. Specific examples in which industry guidance is benefitting from the U.S. Fukushima Forensics efforts include:

- ***Primary Containment Venting*** – As discussed in Sections 3, 4, and 6, the three operating units at Daiichi 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 specific temperature and pressure limits as an appropriate strategy.

  Fukushima accident insights lead to improvements in the BWROG containment venting strategies both before and after core damage occurred. The BWROG Emergency Planning Guidelines (EPGs)/ SAGs provide guidance on venting the primary containment when the pressure reaches the Primary Containment Pressure Limit (PCPL). The PCPL is the lesser of:

  - The pressure capability of the primary containment (may exceed design pressure)
  - The maximum primary containment pressure at which vent valves sized to reject all decay heat from the containment can be opened and closed
  - The maximum primary containment pressure at which RPV vent valves can be opened and closed.

  The PCPL is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, RPV vent valve operability, or the structural integrity of the primary containment.

  Containment venting in BWRs is focused on using the suppression pool to prevent or mitigate core damage and minimize radionuclide release. The BWROG SAGs provide guidance on water management to maximize the use of the suppression pool's scrubbing capability during containment venting. The preferred containment vent path is through the wetwell making use of the harden vent path in Mark I and II containment designs.

  The PWROG SAMGs also provide guidance on containment venting and is based on their specific containment design features and equipment performance. Venting is initiated when containment pressure approaches the lower bound of the calculated probabilistic maximum containment pressure capability. Only containment pressure is considered as an indicator of the need for venting because severe accident analyses show that containment temperature limits are not approached before containment pressure limits for credible severe accident scenarios. For US PWRs, there is not a hardened containment vent so that the ability to re-close the vent cannot be guaranteed. However, once containment pressure is restored to design pressure (or half the venting setpoint for Ice Condenser Containments), the PWROG SAMGs direct station personnel to attempt to isolate the vent to minimize the total release. As pressure approaches the containment venting setpoint, actions are taken to prepare for venting including evacuation of personnel from the area around the vent location.

- ***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. (RCIC/HPCI operation, etc.)

  The BWROG SAGs and PWROG SAMGs have always placed a higher priority on injection of water to the reactor vessel compared to the primary containment.

The BWROG SAGs provide guidance on the injection of water to the reactor vessel versus the primary containment. In most conditions, the generic guidance is for vessel injection (this is a function of containment design and other conditions). If the reactor vessel is failed, the injected water is expected to 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. The improvements to water addition/management guidance, based on insights from Fukushima, provides additional strategies that reduce the offsite doses from containment venting and or failure (water is required to cool core debris in all possible locations and scrub fission products; primary containment and in the reactor vessel).

The PWROG SAMGs provide for adding water to the reactor vessel as a higher priority than adding water to containment but provides guidance for adding water to both the reactor vessel and containment if sufficient resources are available. The PWROG SAMG for addition of water to containment is meant to provide a water pool for quenching and cooling core debris entering containment if reactor vessel failure occurs (because injection to the reactor vessel was not effective). Any water addition to containment is limited to prevent flooding of important equipment and instrumentation in the containment.

- ***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. The BWROG and PWROG generic guidance was enhanced 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 the SAG enhancements (BWROG/ PWROG) to include strategies for venting buildings adjacent to the primary containment as an appropriate action when primary containment pressure exceeds a predetermined value based on containment design features. The BWROG SAG and PWROG SAMG include 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 and the normal ventilation system is not running.
- ***Instrumentation*** – As discussed in Sections 3 and 5, instrumentation and control system logic (e.g., RCIC) anomalies contributed to the accident progression at Fukushima, or influenced the decision-making related to accident management. Because of these Fukushima insights, both the BWROG and PWROG Technical Support Guidance (TSG) for instrumentation was enhanced to include additional calculational aids to make better use of the available instrumentation and to better interpret its response. The BWROG is working with the DOE (SNL) to make these calculational aids functional on iPhone, iPad, or Android devices to improve their usability during an accident or training exercise. The enhanced BWROG and PWROG guidance is used to improve the understanding of the expected response trends of instrumentation, tie that response to known severe accident phenomena (Fukushima Case Studies) and compare the instrumentation response from several instruments where possible.
- ***Severe Accident Models*** – The BWROG SAG and PWROG SAMG are symptom based and use insights/first principles from the severe accident phenomena behavior described in the EPRI Technical Basis Report (TBR).[229] The BWROG and PWROG SAGs are largely independent of the severe accident predictions by either severe accident integral codes MAAP or MELCOR. In addition, the BWROG symptom based EPGs/SAGs and the PWROG SAMGs are focused on addressing anything mechanistically possible. For the PWROG SAMGs, MAAP has been used to revise the containment venting setpoint to delay venting as long as possible and reduce offsite releases due to nuclear decay. MAAP has also been used to validate the venting closure setpoint to minimize the total releases.

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 (MAAP and MELCOR). Specific examples for which data are (or are expected to be available) from Fukushima forensic information include the amount of 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.

Insights from the Fukushima Forensics activities particularly for hydrogen generation, temperature conditions at penetrations, and ex-vessel core debris spreading can be used to further improve severe accident code predictions. In addition, these insights may be of benefit to improving the guidance found in the BWROG SAG and PWROG SAMG, as well as Volume 2 of the TBR. The DOE Forensics effort should continue to work with these organizations (BWROG, PWROG, and EPRI) to foster improvements in the guidance and documents when appropriate.

- ***Operation of Turbine Driven Pumps*** – As discussed in Section 3, information from 1F2 and 1F3 provide 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 HPCI 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 provide guidance on the use of turbine driven pumps (RCIC and HPCI for BWRs and AFW for PWRs) to maintain core cooling. 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.

  The DOE is considering further testing to gain additional insights related to operation of turbine driven pumps. Self-regulation of RCIC with water level is a very important accident management strategy that if implemented in the BWR fleet would significantly simplify the management of accidents. Activities in this area should be fostered and supported to the extent possible. Once the technical basis is firmly established, DOE working with the BWROG's emergency procedures committee can make a significant improvement in reducing the core damage frequency in BWRs.

7.2.2 Code Modeling Enhancements

As discussed in this section, analyses of the events at Daicihi and information from forensics examinations are also being used (and will continue to be used) to reduce uncertainties in systems analysis code models.

7.2.2.1 MELCOR 2.2

The analyses performed and insights gained from participating in the OECD/NEA BSAF Phase I Project highlighted several key areas where MELCOR 2.1 could be enhanced by improving model robustness and implementing new dedicated models to better capture phenomenological behavior and key boundary conditions that drive source term. Code improvements have been directed in the following areas to better simulate the Fukushima-Daiichi Power Station response: detailed safety system modeling, ex-vessel behavior, and code performance during core reflood. These improvements to MELCOR 2.1 mark a significant advancement to the MELCOR code resulting in the recent increment in the version number to 2.2. [230]

The accident progressions in both 1F2 and 1F3, where alternative water injection was introduced across various core degradation states, required improving code robustness and performance during reflood. One key set of changes temporally relaxes the rate-of-change of the quench velocity and causes the quench velocity to be smoothly driven to zero within a small distance of the pool level. Several model corrections and numerical improvements to the MELCOR quench model were developed and implemented and have significantly improved the robustness of the code for reflood conditions. [230]

Akin to this, a temporal relaxation model was introduced within the code. Many physical processes in MELCOR are modeled by correlation based relationships developed from steady-state experiments. These models do not represent the time it takes for these processes to respond as conditions change. As a result, temporal “rate-of-change” aspects of MELCOR simulations are not expected to be highly accurate and numerical instabilities can be magnified when sudden changes occur. Temporal relaxation is a simple way to introduce a user-imposed time-scale based model that limits how quickly processes being modeled can change in time. This has made it significantly easier to perform forensic analysis of core oxidation and relocation behavior and improved code robustness. [230]

The necessity of the new MELCOR homologous pump model was highlighted by the behavior of the 1F2 RCIC system. The updated model is similar to that found in RELAP (RELAP5-3D, RELAP4) but with some distinguishing features. A user can fully specify 1) rated pump conditions, 2) single/two-phase pump performance via homologous curve input, 3) pump friction torque as a polynomial, 4) pump inertia as a polynomial, 5) pump speed and motor torque controls, and 6) pump trips. Additionally, pump data from the Semiscale and Loft experiments are included as a “built-in” option with a “universal correlation” taken from the literature. [230]

Recent MELCOR code development has also focused on improvements to the ex-vessel core-melt cooling models available to MELCOR code users. In particular, a water ingression model, a melt eruption model, and a melt spreading model were added. The water ingression model implemented in MELCOR is based on the model by Epstein and generally follows the implementation in CORQUENCH. This model allows water to ingress into crust layer if the upper heat flux is less than the dryout flux. The water ingression model was assessed against the CCI experiments (performed under the OECD MCCI project). The melt eruption model is implemented as a transfer of mass from the melt layer into the debris layer, where the rate of transfer is proportional to the gas sparging rate. This relocated debris has an associated porosity and is therefore more coolable, where the permeability is based on the dryout flux. These models provide a mechanistic approach to determining the interaction between ex-vessel debris and overlaying water. [230]

In the case of core degradation, SNL and the US NRC decided to take a prudent “wait-and-see” approach to changing the phenomenology of core degradation within MELCOR. Future changes to this portion of the code will be highly informed by entry into the reactor pressure vessel and/or the primary containments of each unit. That said, developers are assessing the MELCOR crust formation and molten pool/crust formation modeling with a focus on steam permeability to severely damaged core regions and its effects on hydrogen generation and on sensible heat gain and convective heat loss from such degraded regions. This is partly motivated by recent MELCOR/MAAP crosswalk studies comparing the two code modeling paradigms and also from deep analysis of the Fukushima Daiichi Unit “three peaks” time period where there appears to be evidence of core degradation processes affecting hydrogen generation and PCV pressurization. This investigation could result in further refinement of MELCOR core degradation modeling.[230]

MELCOR 2.2 is a significant official release of the MELCOR code with many new models and model improvements. This section has provided a quick review and characterization of new models added, significant code changes and their impact on analyzing the Fukushima-Daiichi accidents. More detailed information is found in "Quicklook Overview of Model Changes in MELCOR 2.2: Rev 6342 to Rev 9496" [230] as well as the associated User Guide and Reference Manuals for MELCOR.[8] These changes have made it possible for 500-hour long source term calculations of Fukushima-Daiichi Power Station to be performed in under 50 hours of computational time.

7.2.2.2 MAAP

Since the initial evaluations of Fukushima-Daiichi accidents using MAAP, the MAAP5 code has been improved in many areas. These improvements were supported by the Commitment of Decommissioning / Safety Fundamental Technology for Power Reactor by Ministry of Economy, Trade and Industry of Japan in cooperation with the EPRI.

A brief summary of the major improvements to the MAAP5 code are described below.

New BWR reactor coolant system thermal hydraulic model: The MAAP5 BWR thermal hydraulics models were improved to provide the same level of detail as the existing MAAP5 PWR thermal hydraulics models. Enhancements implemented in this release refine water node definitions within the reactor vessel and recirculation system and improve simulation of transients where pressure differences between the RCS nodes significantly influence thermal hydraulic response. MAAP will now model these transients in a manner similar to other contemporary thermal hydraulic codes. Nodalization of the vessel was increased to 17 nodes including two recirculation loops and two steam line nodes. The mass and energy of gas and water are tracked in each node and each node has its own pressure and temperatures. A detailed jet pump model was also added.

Core model improvements: The previous MAAP core model did not explicitly model the details of the BWR specific non-fuel nodes such as fuel support pieces and the core plate. Also, the major relocation path from the core to the lower plenum was based on a TMI-2 like side crust failure. Improvements were made to model detailed BWR-specific geometries explicitly including the fuel support piece, inlet orifice, and external bypass region with the core plate. A potential shroud melt-through relocation path (through the downcomer to the lower plenum) was added. Relocation through this path can be caused by core debris contact with the inner surface of the core shroud. In addition, the upper plenum-to-core natural circulation model was enhanced to include a down flow pathway through the external bypass and an upflow pathway through the in-core bypass when the fuel channel boxes are intact.

Lower plenum debris pool layering model: Previous versions of MAAP model the lower plenum debris pool using a single oxidic debris pool, light metal layer, heavy metal layer, and crusts. The lower plenum debris bed model has been improved to include a new layering model and radially nodalized particulate beds. In the new layering model, the oxidic debris bed is axially nodalized according to the vessel lower head axial nodalization. The material composition and energy in each layer are tracked. Moreover, the material compositions in individual crust components are tracked separately and temperature and solid fraction are calculated for each layer. By layering the material arriving in the lower plenum, the history of different material relocations to the lower plenum can be preserved and used for more accurate assessment of the vessel wall response. For example, during the TMI accident the first material that reached the lower head contained both ceramic and metallic material, which formed a basal crust structure that survived to protect the lower head. With the layering model, a "hot spot" evaluation is also possible. A "hot spot" occurs when molten debris comes in direct contact with the vessel wall, either due to the absence of a protective layer or by disruption of the crust.

CRD tube modeling: CRD tube housings in the lower plenum were previously modeled as one heat sink with five axial nodes. Improvements were made such that the CRD tubes are axially and radially nodalized according to the vessel lower head axial nodalization and according to the core radial nodalization. In addition, the CRD tube model has been extended to model the CRD tubes in the region below the reactor vessel. This modeling includes conduction and radiation heat transfer between the lower head, CRD tubes, pedestal gas and pedestal wall. The possibility of corium melt flowing into collapsed CRD tubes and causing creep rupture of CRD tubes outside the reactor vessel, leading to vessel failure, is now considered. Lastly, ex-vessel CRD tube melting is considered due to interaction with the corium jet exiting the breached reactor vessel after failure.

Vessel Failure Mechanisms: For the most recent versions of MAAP5, models have been developed to represent the influence of increasing core temperatures on the Source Range Monitor (SRM)/Intermediate Range Monitor (IRM) and TIP dry tubes as well as the rate of steam and hydrogen discharge to the containment pedestal and drywell if any of these should fail. Subsequent to the gas and fission product discharges, these core materials would be liquefied, or melted and would flow downward and into the open flow paths. Rapid cooling and freezing of these materials may occur as they flow downward through the core and the lower plenum, which could cause these relatively small flow paths to become plugged by the frozen debris. Once the core material relocates into the lower plenum, melting of the plugged corium in the SRM/IRM could occur resulting in a small pathway leading to a localized failure.

Containment thermal stratification model: Previous versions of MAAP5 had a generalized containment model with lumped nodes and junctions. For a given node, it was assumed that atmospheric conditions are uniformly distributed and each node had one gas temperature and one water temperature. The Fukushima-Daiichi core melt events have identified the importance of fluid stratification effects to determining the overall containment thermal hydraulic response. Improvements were made to the generalized containment model such that the gas region for a given node is divided into two zones (a less dense upper zone and a denser lower zone) and the zonal boundary is calculated dynamically. The new model can calculate thermal and gas (hydrogen) stratification within a node. For a suppression pool, thermal stratification is calculated and one water pool can be sub-divided into many layers with different temperatures.

MCCI related models: 1) In earlier versions of MAAP, the combined pedestal and sump shape was approximated as a single right cylinder, with vertical walls and a horizontal floor. The previous versions of MAAP could not simulate deep sumps in the pedestal region below the RPV. The model was improved to add a combined pedestal and sump shape model for MCCI in the containment. The new model includes a right cylinder for the pedestal with a box-like shape for the sump region to more realistically resemble the true shape of the sump. 2) Previous versions of MAAP used user-defined nominal heat transfer coefficients for sideward and downward core-concrete heat transfer. New mechanistic heat transfer correlations have been developed based on bubble agitation plus natural convection. 3) In some MCCI experiments, stratification between the metallic phase and oxidic phase was observed. The impact of ex-vessel corium pool stratification is investigated by calculating the heat transfer coefficients of the metal and oxide layers independently. Given that a majority of the decay heat will reside in the oxide layer, the metal layer heat transfer coefficients are reformulated to include the interfacial resistance between the layers. The heat transfer coefficient between the corium pool and the floor/walls of the containment node are calculated as contact surface area weighted averages of the oxide and reformulated metal layer heat transfer coefficients.

Debris coolability model: A bulk cooling model is added for the most recent version of the MAAP code. Thus, the latest version of MAAP models all major coolability mechanisms: bulk cooling, melt eruption, and water ingression.

Mechanistic RCIC/HPCI model: The RCIC system of 1F2 operated in an unpowered, unattended mode for the nearly three days. The unpowered state also entailed loss of RPV level control, which resulted in flooding of the main steam line, including the steam extraction line that drives the RCIC steam turbine. The resulting two-phase flow drove the turbine for an extended duration during the accident. A mechanistic model of the RCIC steam extraction line and Terry™ turbine was developed. The new model treats normal single-phase steam operation and the transition to two-phase flow operation after ELAP initiation. The new model demonstrates that the RCIC system, in concert with the connected RPV, yields a self-regulating behavior that gradually adjusts and equilibrates the RCIC pump inflow to the RPV with the RCIC extraction line outflow from the RPV.

7.2.2.3 Possible Future Code Applications and Enhancements

As discussed within this report, several future changes may be implemented based on information obtained from the affected reactors at Daiichi. Selected examples include:

- ***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, including the impact of failure locations and sizes.
- ***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.
- ***MELCOR and MAAP Debris Holdup Studies* -** Findings from 1F2 and 1F3 suggest that a non-negligible amount of core debris may be held up on structures below the reactor vessel. System analysis codes should be exercised assuming a range of core debris holdup in a situation that is not cooled by water to investigate the impact of heat sources not covered by water on PCV gas phase temperature and pressure.
- ***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 Holdings 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 U.S. Forensics Effort is identifying examination needs from the affected units at Daiichi and evaluating information obtained by TEPCO Holdings 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 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

FY2017 Meeting Agendas and Attendee Lists

November 15-16, 2016 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 15-16, 2016

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

Tuesday, November 15th

8:30 AM	Welcome to NEI, Administrative Matters, and Safety Minute	S. Kraft, NEI
8:35 AM	Welcome – RST Lead	M. Farmer, ANL
8:40 AM	Welcome and Overview – DOE-NE DOE NE -New Office of International Nuclear Safety DOE-EM Activities	D. Peko, DOE-NE E. McGinnis, DOE-NE B. Lisann/J. Marra, EM
9:15 AM	NRC Comments	R. Lee, NRC
9:20 AM	Background and Proposed Agenda Status of Action Items from Prior Meeting	J. Rempe, Rempe and Associates, LLC
9:40 AM	TEPCO Update	K. Nozaki, TEPCO
10:45 AM	*Break*	All
11:00	Code Analysis Insights Highlighting Key Uncertainties	N. Andrews, SNL/ J. Gabor, Jensen Hughes /K. Robb, ORNL
12:15	*Working Lunch*	All
	Status of INL Website	P. Humrickhouse, INL (slides presented by D. Peko, DOE)
1:15	LWR Accident Management Insights from Forensics Information	R. Lutz/ K.Klass
1:45 PM	Topic 1 - Component Inspection Updates Related to New Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	K. Robb, ORNL/ J. Gabor, Jensen Hughes
3:00 PM	*Break*	All

November 15-16, 2016 Meeting Agenda (Continued)

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
November 15-16, 2016

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

Tuesday, November 15th (Continued)

3:15 PM	Topic 2 - Dose Measurements and Smears for Isotopic Concentration Evaluations (based on code analysis evaluations, etc.) Updates Related to New Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	R. Gauntt, SNL / /R. Sanders, AREVA
5:00 PM	*Adjourn*	

Wednesday, November 16th

8:30 AM	Topic 3 - Core Debris Location Evaluations Updates based on New Material Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	M. Farmer, ANL/ R. Gauntt, SNL/ M. Plys, FAI
9:45 AM	*Break*	All
10:00 AM	Topic 4 – Combustible Gas Effects Updates based on New Material Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	W. Luandilok, FAI/ N. Andrews, SNL
11:30 AM	Next Steps ▪ Proposed report outline ▪ Action items and schedule ▪ Proposed discussion topics / date for next meeting	J. Rempe, Rempe and Associates, LLC
Noon	*Adjourn*	All

November 15-16, 2016 Meeting Attendees

Number	Name	Organization
1	Phil Amway	Exelon, BWROG
2	Nathan Andrews	SNL
3	Daisuke Baba	Embassy of Japan
4	Mike Corradini	University of Wisconsin
5	Hossein Esmaili	U.S. NRC
6	Mitchell T. Farmer	ANL
7	Terri V. Farthing	GE Hitachi
8	Edward Fuller	U.S. NRC
9	Jeff Gabor	Jensen Hughes
10	Randy Gauntt	SNL
11	Richard Griffith	SNL
12	John Gross	U.S. DOE-NE
13	Takashi Hara	TEPCO Holdings
14	John Kelly	U.S. DOE-NE
15	Ken Klass	Talen Energy, PWROG
16	Stephen Kraft	NEI
17	Richard Lee	U.S. NRC
18	Roy Linthicum	Exelon, PWROG
19	Beth Lisann	U.S. DOE-EM
20	Wison Luangdilok	Fauske and Associates, LLC
21	Robert Lutz	Lutz Nuclear Consulting (by telephone)
22	Donald Marksberry	U.S. NRC
23	John Marra	U.S. DOE
24	Ed McGinnis	U.S. DOE
25	Kenichiro Nozaki	TEPCO Holdings
26	Doug Osbourne	SNL
27	Damian Peko	U.S. DOE
28	Susan Pickering	SNL
29	Marty Plys	Fauske and Associates, LLC
30	Joy Rempe	Rempe and Associates, LLC
31	Kevin Robb	ORNL
32	Michael Salay	U.S. NRC
33	Robert Sanders	AREVA
34	Daichi Yamada	EPRI (Guest Researcher), TEPCO Holdings

May 24-26, 2017 Meeting Agenda

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
May 24-25,2017

Nuclear Energy Institute, 1201 F St., NW, Suite 1100, Washington, DC
Call-in number: +1 719-457-0705 Passcode 498334#

Wednesday, May 24, 2017

8:30 AM	Welcome to NEI, Administrative Matters, and Safety Minute	S. Kraft, NEI
8:35 AM	Welcome and Overview – DOE-NE DOE-EM Activities within Japan	D. Peko, DOE-NE A. Han, DOE-EM
8:45 AM	DOE LWRS RST Pathway Lead Comments - Overview of activities and relationship to Forensics	M. Farmer, ANL
9:00 AM	NRC Comments	R. Lee, NRC
9:15 AM	Objective and Planned Agenda Status of Action Items from Prior Meeting	J. Rempe, Rempe and Associates, LLC
9:30 AM	TEPCO Update	S. Mizokami, TEPCO
10:45 AM	*Break*	All
11:00 AM	Introduction to PreADES Project	I. Sato, JAEA
11:30 AM	Code Analysis Insights Highlighting Key Uncertainties and Impact of Model Updates / Corrections - MAAP Updates -25 minutes - Modeling Below Vessel Structures -15 min. - MELCOR Updates – 20 minutes	C Paik, FAI K. Robb, ORNL R. Gaunt/ N. Andrews, SNL
12:30	*Working Lunch* Status of INL Website	All P. Humrickhouse, INL
1:15	BWR Mark I and II Water Management Implementation in the US per NEI 13-02 Rev 1 MAAP5 Analyses to Support Extended PWROG SAMG Bases LWR Accident Management Insights from Forensics Information – Water Management	P. Amway/R. Bunt, BWROG R. Labarge/R. Linthicum, PWROG R. Lutz, R. Lutz Consulting

May 24-25, 2017 Meeting Agenda (Continued)

Reactor Safety Technology Expert Panel Forensics Meeting

Meeting Agenda
May 24-25,2017

Nuclear Energy Institute, 1201 F St., NW, Suite 1100, Washington, DC
Call-in number: +1 719-457-0705 Passcode 498334#

Wednesday, May 24th (Continued)

2:00 PM	Topic 1 - Component /System Examinations Updates Related to New Material Available Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	K. Robb, ORNL/ J. Gabor, Jensen Hughes
3:30 PM	*Break*	All
3:45 PM	Topic 2 - Dose Measurements /Surveys Updates Related to New Material - 1F1 X-100B exams (dose measurements) - 1F2 X-6 exams (dose measurements) Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	N. Andrews, SNL / /R. Sanders, AREVA/ D. Luxat, Jensen Hughes
5:15	*Adjourn*	

Thursday, May 25th

8:30 AM	Topic 3 - Core Debris Location Evaluations Updates based on New Material - 1F1 X-100B exams (images) - 1F2 X-6 exams (images) - 1F1 Water Addition Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	M. Farmer, ANL/ R. Gauntt or N. Andrews, SNL/ M. Plys, FAI
10:00 AM	*Break*	All
10:15 AM	Topic 4 – Combustible Gas Effects Updates: -Followup on electrochemical source of H_2 -Ballpark estimate of required H_2 for observed explosions Insights/Comments on Information Consistency and Adequacy for Reactor Safety Insights Additional information requests (if needed)	W. Luandilok, FAI/ D. Luxat, Jensen Hughes
11:00 AM	Next Steps ▪ Proposed report outline ▪ Action items and schedule	J. Rempe, Rempe and Associates, LLC
Noon	*Adjourn*	All

May 24-25, 2017 Meeting Attendees

Number	Name	Organization
1	Phil Amway	Exelon, BWROG
2	Nathan Andrews	SNL
3	Daisuke Baba	Embassy of Japan
4	Sud Basu	McGill Engineering Associates
5	Randy Bunt	Southern Nuclear Company
6	Mike Corradini	University of Wisconsin (by telephone)
7	Phillip G. Ellison	GE-Hitachi
8	Hossein Esmaili	U.S. NRC
9	Mitchell T. Farmer	ANL
10	Terri V. Farthing	GE Hitachi
11	Edward Fuller	U.S. NRC
12	Jeff Gabor	Jensen Hughes
13	Randy Gauntt	SNL
14	Craig Gerardi	ANL
15	Takashi Hara	TEPCO Holdings
16	Ana Han	U.S. DOE-EM
17	Paul Humrickhouse	INL
18	John Kelly	U.S. DOE-NE
19	Karen Kirkland	Texas A&M (by telephone)
20	Stephen Kraft	NEI
21	Richard Lee	U.S. NRC
22	Roy Linthicum	Exelon, PWROG
23	Wison Luangdilok	Fauske and Associates, LLC
24	Robert Lutz	Lutz Nuclear Consulting (by telephone)
25	David Luxat	Jensen Hughes
26	Donald Marksberry	U.S. NRC
27	Grace Meikle	U.S. DOE
28	Shinya Mizokami	TEPCO Holdings
29	Chuck Negin	CANegin & Associates
30	Chan Paik	Fauske and Associates, LLC
31	Damian Peko	U.S. DOE
32	Joy Rempe	Rempe and Associates, LLC
33	Kevin Robb	ORNL
34	Michael Salay	U.S. NRC
35	Robert Sanders	AREVA (by telephone)
36	Ilken Sato	JAEA
37	Richard Wachowiak	EPRI
38	Bill T. Williamson	TVA
39	Daichi Yamada	EPRI (Guest Researcher), TEPCO Holdings

Appendix B

Website to Support Forensics Evaluations

Website to Support Forensics Evaluations

Background

During the November 2016 meeting, it was agreed that a website should be developed that provided a searchable location for archived information relied upon by the U.S. Forensics Effort. To ensure that this website addresses needs for this effort, it was agreed to populate an initial website framework and present it at the May 2017 expert panel meeting for review by program participants. To ensure that information in this website would be archived, this framework was developed under the auspices of the INL digital library (https://inldigitallibrary.inl.gov) at https://doeforensics.inl.gov. The INL digital library archives INL reports and publications, the DOE-ID Public Reading Room, and a repository of information related to the cleanup of TMI-2, including the NRC TMI-2 Knowledge Management Library (https://tmi2kml.inl.gov). This appendix describes the new DOE forensics website framework as it has been implemented, along with planned further additions and modifications.

The need to have access to archived information from inspections, analyses, and other relevant sources has been recognized internationally. There are several existing websites that support this need, including the website developed and maintained by TEPCO Holdings (http://www.tepco.co.jp/en/index-e.html and http://photo.tepco.co.jp/en/index-e.html) and the website developed by the Institute for Applied Energy supports to support SAREF Research activities (https://fdada.info/en/home2/). Although this effort relies on information contained in these two websites (and links to these websites may be accessed from the U.S Forensics website), it was agreed that a unique website was needed to archive information relied upon for analyses and evaluations completed within the U.S. Forensics Effort.

Website Design

The primary purpose of the U.S. Forensics Effott website is to display, filter, and search for relevant documents, and the interface is designed with these functions in mind. At the right side of the page, several 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 Figure B-1; in this case all documents presently in the database are listed (15 per page).

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

- **Date**. This can be an arbitrary range of dates entered by the user, or can be set using the checkboxes to view documents added in the last day, week, month, year, or older.
- **Date Added**. This allows users to quickly identify information is new, added within the last week, the last month, the last year, or over a year.
- **Source.** This is the issuing agency. Initial sources include TEPCO Holdings, INL, IRID, and SNL.
- **Media Type**. This includes slides, report, webpage, spreadsheet, etc.

Figure B-1. Default view of the website, with no document filters applied. (Courtesy of INL).

- **Topic**. Site materials are presently organized under four topics:
  - Forensics Investigations
  - Plant Data
  - Unresolved Issues (UI)
  - Accient Analysis

- **Subtopic**. This is a more specific description of the document; examples include: "Combustible Gas", "Debris-Muon" "Debris- Robot", "Debris-Sample, "UI-2nd Report", "Ui-3rd Report", "Water Management", etc. This list will continue to grow as documents are added.
- Location. This can be within a particular unit, such as the PCV, RB, RPV, and Torus, or the Site.
- **Units**. This is can be a particular unit (e.g. 1F2), or multiple units (1, 2, 3, and/or 4).
- **Language**. This is for filtering to Japanese or English documents.

The current options under each filter category are not exhaustive; these attributes are assigned to a document when uploaded by a user, and are not present in the list if they have not yet been so attributed to any document. As documents are added, the number of options in most categories will grow.

Filters are applied by selecting the applicable check box in each category and clicking "Apply." This must be done for each category in which one wishes to filter, which can be any number of categories. Similary removing a filter requires unchecking the corresponding box and clicking "Apply." All checked boxes in each category can be removed simultaneously by clicking "Clear."

In addition to the filters described above, a simple search utility is included that allows users to search for text within the (optionally) filtered document list. This search function acts not just on titles and document descriptions, but also to the entire text of documents (provided, e.g. in the case of PDF files, it contains searchable embedded text and not simply scanned images).

The website is built on the SharePoint platform, and users must have an account to access it. These are available to both internal (INL) and external users. Future development will focus on adding materials and implementing further features identified by the expert panel.

Appendix C

Information Needs

Information Needs

As described in Section 1.1, primary objectives of the U.S. forensics effort are to develop and update, consensus U.S. input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of D&D plans for Daiichi. Initial objectives were developed in 2014. Every year, these objectives are updated. Section C.1 presents the resulting 2017 updates to these information needs. As described in Section 1.3, these information needs are organized into tables for each location (e.g., the reactor building, the PCV, and the RPV).   Several new information requests were added during 2017, PC-17, PC-18, PC-19, PC-20, PC-21, and RB-15. In addition, the status of several U.S. information requests was modified. For example, several information needs have been addressed by recent examinations completed by TEPCO Holdings. In addition, because of renewed U.S. interest in the integrity of instrumentation data (PC-7 and PC-8), TEPCO Holdings will be providing recent results regarding the qualification envelop of sensors and estimates regarding the conditions that these sensors were exposed.

During the May 2017 meeting, it was agreed that more detailed requests should be developed for the following high priorty, nearer term information needs:

- RB-9b
- RB-10
- RB-15
- PC-1
- PC-3a, 3b, 3c, 3d, and 3e
- PC-5
- PC-6
- PC-17, PC-18, PC-19, and PC-20 (combined)
- PC-21
- RPV-1b
- RPV-4 and RPV-5 (combined)

Section C.2 presents additional details have been provided regarding the benefits of obtaining this information, how this information would be used, the methods and/or tools required to obtain this information, the expected schedule for when this information would be available, and any expected preparations to obtain or follow-on research that may be required to use this information.

C.1 Summary 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[vv] 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 Holdings. If torus not drained, requires underwater technology available. If photos or data are obtained as part of D&D activities, please provide (but the U.S. 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 due to seismic events (1F1, 1F2, and 1F3) and due to operation (1F3). • Compare endstate of 1F3 (look for flaws) with the endstate of 1F1 and 1F2. If similar flaws are observed in all three units, it would be useful for assessing impact of the seismic event and of longer term operation.	Impacts AM strategies (equipment utilization).	Currently flooded (requires other alternatives for underwater investigations unless drained).	Not currently considered by TEPCO Holdings; If torus not drained, requires underwater technology. If photos are obtained as part of D&D activities, please provide (but the U.S. recognizes that additional information may not be obtained and that system degradation may be due to long term exposure ro water since the accident).
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 Holdings goes into 1F1 and after debris removal.	TEPCO Holdings 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

[vv] 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-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 Holdings goes into 1F3 and after debris removal.	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 U.S. will investigate the NRA website for images. In particular, if D&D strategy allows 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. Photos after debris removal are also of interest.
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 Holdings goes into 1F4 and after debris removal.	
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 Holdings 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 Holdings 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.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
					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.
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 Holdings is not planning any additional examinations. This item is closed. If additional information become available, please provide.
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 U.S. recognizes that additional information may not 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 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 U.S. recognizes that additional information may not 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 Holdings 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 U.S. recognizes that additional information may not be obtained). US industry agreed to compile a list of high interest locations (based on benefits to maintenance).
	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.[ww]	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 Holdings 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 U.S. recognizes that additional information may not be obtained). US industry agreed to compile a list of high interest locations (based on benefits to maintenance).

[ww] 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.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-10	Photos/videos and dose surveys of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows, penetrations).	• Potential leakage paths for RN and hydrogen release.	Improved AM strategies (Plant improvements for more robustness, training, education); applicable to BWRs and PWRs (which have similar penetration designs). Improved codes. Improved understanding of events; assist D&D efforts.	Now and later.	TEPCO Holdings has considerable information related to this information need.[xx] Now, restoring works for PCV to 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 U.S. recognizes that additional information may not be obtained). US industry agreed to compile a list of high interest locations (based on benefits to maintenance).

[xx] **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.

- 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.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
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.[yy]	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 Holdings; 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 U.S. recognize that additional information may not be obtained).

[yy] 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 Holdings 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 Holdings 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 U.S. recognizes that additional information maynot be obtained).
RB-14	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..	• Presence of Si would indicate MCCI	Assist D&D efforts for determining debris location.	Now	TEPCO Holdings is investigating potential to send white deposits to JAEA for evaluation.

Table C-1. Information Needs from the Reactor Building					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
RB-15	Examinations of 1F1 RCW surge tank; water level and additional dose measurements.	• During events at 1F1, contaminated water may have entered RCW and/or water may have flowed out of RCW into containment	Determine the role of the RCW during 1F1 accident.	Now	TEPCO Holdings has obtained some dose rate measurements in the area around the surge tank.

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[zz] 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 Holdings observed that tensioning is done based on gap requirements; no record available. TEPCO Holdings 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 U.S. would appreciate any additional information (although the U.S. 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 Holdings has some photos (and no damage observed); no RN sampling planned (due to radiation levels). This item has been addressed.

[zz] 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). [aaa]	• Code assessments • Possible model updates for mass, height, composition, morphology (e.g., coolability), topography of debris, spreading, splashing, and salt effects.[bbb]	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 Holdings roadmap).	TEPCO Holdings 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 Holdings roadmap).	TEPCO Holdings has some bellows information and may obtain additional visual information. TEPCO Holdings 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 Holdings 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.

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

[bbb] 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 Holdings has no plans to obtain at this time. TEPCO Holdings 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 Holdings will obtain some information. The U.S. believes this information is very important for benchmarking models. 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 Holdings has some pressure and temperature measurements at Primary Loop Recirculation (PLR) pump inlet since April 2011. No additional inspections planned. The U.S. continues to have interest in this visual information. However, the U.S. recognizes that additional information may not become available.
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 Holdings has not considered photographic exams. TEPCO Holdings has some temperatures around SRV and MSIV recorded since September 2011 for 1F2 and 1F3. The U.S. 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 U.S. 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-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 Holdings has not yet developed plans for such examinations. The U.S. 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 U.S. recognizes that additional information may not become available.
PC-7	Ex-vessel inspections and operability assessments of 1F1, 2, and 1F3 in-vessel sensors and sensor support structures[ccc]	• 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 Holdings completed some examinations and recalibrations and plans to perform more evaluations. Cable integrity examinations by TDR (time domain reflectometry) 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. TEPCO Holdings will provide additional information regarding sensor qualification envelop and conditions exposed to during the accident to address renewed US interest in this topic.

[ccc] 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.

Table C-2. Information Needs from the Primary Containment Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
PC-8	Examinations and operability assessments of 1F1, 1F2, and 1F3 ex-vessel sensors and sensor support structures[ddd]	• 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 Holdings has completed some examinations and recalibrations and plans to perform more evaluations. TEPCO Holdings will provide additional information regarding sensor qualification envelop and conditions exposed to during the accident to address renewed US interest in this topic.
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 Holdings may obtain more data. Please provide additional information when available and consider evaluating presence of coating materials in elemental evalutions of other samples.
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 Holdings 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 U.S. remains very interested in isotopic information from RN surveys/samples for code assessments (but the U.S. recognizes that this information may not become available).

[ddd] 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-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.[eee]	Improved BWR AM strategies (plant improvements). Improved understanding of events. Assist D&D efforts. Potential PWR impacts.[eee]	Now and later. Exams may be completed more easily at Daini.	Not currently considered by TEPCO Holdings; some photos may already be available. The U.S. remains interested in additional photographs from Daiichi or Daini (but the U.S. recognizes that this information may not become available).
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 U.S. continues to have interest in this information. However, the U.S. 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 Holdings; some photos may already be available. The U.S. continues to have interest in this visual information. However, the U.S. recognizes that additional information may not become available.

[eee] 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-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 Holdings has some sample information. The U.S. 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 Holdings 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 is closed.
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 U.S. continues to have interest in this visual information, but recognizes that additional information may not become available.
PC-17	Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1.The upper surface of the sediment is ~ 30 cm above drywell floor.	• Presence of Si would indicate MCCI • Possible model improvements.	Assist D&D efforts for determining debris location and improved accident management strategies.	Now	TEPCO Holdings is also interested in this information.
PC-18	The nature of material belowthe upper surface of debris at the X-100B penetration location in 1F1 to determine if it is additional sediment or other material such as core debris.	• Presence of Si or core material debris would indicate MCCI • Possible model improvements.	Assist D&D efforts for determining debris location and improved accident management strategies.	Now	TEPCO Holdings is also interested in this information.
PC-19	Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location	• Identification of material could provide an indicator of peak structure temperatures and potential for structure failure. • Possible model improvements.	Assist D&D efforts for determining debris location and improved accident management strategies.	Now	TEPCO Holdings is also interested in this information.

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

Item	What/How Obtained	Why	Expected Benefit /Use	When	Status
PC-20	Chemical analysis of black material on 'existing structure' in 1F1 images at location 'D3'	• Presence of Si or core material debris would indicate MCCI • Possible model improvements.	Assist D&D efforts for determining debris location and improved accident management strategies..	Now	TEPCO Holdings is also interested in this information.
PC-21	Images from examinations in 1F3 X-53 penetration.	• Possible model improvements	Assist D&D efforts for determining debris location and improved accident management strategies.	Now	TEPCO Holdings is also interested in this information.

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[fff] 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 Holdings will conduct visual, some metallurgical and fission product exams. The U.S. 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 Holdings has no plans for any such exams. See PC-3 for water leakage information from MSL penetration through PCV. The U.S. 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 Holdings will conduct visual exams and some metallurgical exams. The U.S. remains interested in this information, but recognizes that it may not be available.

[fff] 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 Holdings 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 U.S. 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 Holdings will conduct visual, some metallurgical and fission product deposition exams. The U.S. 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 Holdings 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 U.S. 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 Holdings will conduct visual exams and some metallurgical exams. The U.S. 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 Holdings will conduct visual exams. The U.S. 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 Holdings will conduct visual exams. The U.S. remains interested in this information, but recognizes that some information may not be obtained.

Table C-3. Information Needs from Reactor Pressure Vessel					
Item	**What/How Obtained**	**Why**	**Expected Benefit /Use**	**When**	**Status**
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 Holdings has deployed and provided results from muon tomography examinations.
	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 Holdings 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.

C.2 Detailed Information Needs

■ **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RB-9b: Photos/videos around mechanical seals and hatches and electrical penetration seals **RB-10:** Photos/videos of 1F1 (vacuum breaker), 1F1, 1F2, and 1F3 PCV leakage points (bellows and other penetrations) High-resolution images (photos/videos) of PCV penetrations and other vulnerable areas (i.e., access hatches, piping/electrical penetrations, expansion joints/bellows). Images of similar locations from each unit (1F1, 1F2, 1F3, 1F4) allows for comparison of damage and end state between units. 1F4 photos will provide a good baseline of a vessel not over pressurized. Imaging should be sufficient to estimate whether damage has occurred. External PCV images may be sufficient. Images taken internal to the PCV and of disassembled penetrations (i.e., hatch sealing faces and seal material) are desired if obtained during D&D. History on penetration leakage or repairs correlated to images is also desired.
■ **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Safety - Desired for improving reactor safety analysis models and accident management. Operational - Provides for weak link assessment of penetration capacity. Economic – Provide insight into seal performance capability; could be used to adjust maintenance and inspection D&D - Impacts D&D because of constraints on contaminated water release, airborne radionuclide release path. Can influencs D&D method by identifying where containment is leaking and to what level containment can be flooded.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Locations of PCV failure and leakage can affect the accident progression with respect to timing, accident mitigation actions, venting, and radionuclide and combustible gas releases. This information can be used to validate and/or enhance the current understanding of the conditions required for PCV failure and the locations of such failures. It can also impact operations and maintenance considerstions, such as gasket and seal material selection and replacement. Linkage of repaired or degraded penetrations performance in over design conditions can provide insights to improve realistic estimates of failures and investigate improvements in repair methods.
■ **Methods/Tools Needed to Collect Information or Data**
• High resolution imaging system – external to PCV • Dose survey meter or gamma camera (3D image). • Irradiation resistant high resolution imaging system – internal to PCV • Personnel observations indicating leakage (water dripping, discoloration, puddles)
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and later (continued inspections of containment and identification of leakage points for units 1F1, 1F2 and 1F3). Base line information from 1F4 can be gathered now. History of penetration maintenance and repair can support investigation of radiological releases and flood-up plans
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Identification of actual penetration characteristics (e.g. geometry, seal material) may be needed to apply observations to other units. Prediction of conditions of penetration during accident (i.e., stress, temperature, pressure). Although multiple scenarios may lead to the observed endstate, comparisons between predicted and observed endstates may allow identification of possible scenarios and elimination of other scenarios. U.S. industry should develop a list of high interest penetrations/areas because of maintenance benefits and provide to TEPCO Holdings. Tabletop exercises with operation and reactor safety experts should be conducted to develop potential penetration failure scenario list.

■ **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RB-15: Reactor Building Closed Cooling Water system (RCW) inspection 1F1 Water level measurement of RCW. Dose survey around RCW surge tank. Images of the RCW system inside of containment are desired if obtained during D&D.
■ **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Safety - Desired for understanding 1F1 accident progression and the potential role of the RCW during an accident. D&D - Could influences D&D efforts by identifying where containment is leaking.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
TEPCO Holdings and the U.S. expert panel have identified the potential the failure of the 1F1 RCW sump heat exchanger piping in containment. The RCW system may have influenced the accident progression by allowing releases from containment and/or supplying cooling water to the ex-vessel debris in containment. Understanding the status of the RCW system will aid in determining the role the RCW system had during the accident.
■ **Methods/Tools Needed to Collect Information or Data**
• Dose survey meter or gamma camera (3D image). • Water level may possibly be obtained from gauge on surge tank or a dip stick. If water level is lower than surge tank, alternate assessment methods and locations may be required.
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term, the RCW surge tank and reactor building floors appear accessible. The surge tank inspection could accompany any future investigation of the nearby IC. Long-term, images of the RCW inside of containment (sump heat exchanger piping) may be obtained during D&D or its planning.
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Identifying the designed water volume of the RCW system.

- **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**

PC-1: Photos and/or videos of the drywell head, head seals, and sealing surfaces for 1F1, 1F2, and 1F3.

There is also interest in obtaining the tension, torque, and bolt length records of bolts used to close the drywell head. This information is of interest both prior and during removal.

- Visual – signs of asymmetric lift or leakage paths. Look for thermal deformation due to high temperatures over time.
- RN Swabbing
- Visual inspection of seal
- Visual inspection of the head. Look for evidence of permanent strain in the head flange or bulging of the head hemisphere and for evidence of bending/bowing of the bolts along their length that could result from head flange strain and result in permanent leakage location even after PCV decompression.
- Inspect shield plug – visual inspection of cracks.

- **Benefits (Safety, Operational, Economic, D&D, or other benefits):**

AM Strategies; What happened with respect to the leak path; better simulations for training. Improved understanding of PCV response to overpressure that could inform accident management, especially PCV venting strategies.

- **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**

Determine how head lifted with emphasis on the state of the flange closure gap and any evidence of permanent strain/deformation such that permanent leak paths would persist beyond the simple elastic bolt stretching behavior. Determine peak temperatures. Look for indicators of degradation due to high temperature hydrogen, including hydrogen induced embrittlement.

- **Methods/Tools Needed to Collect Information or Data**

- Mostly photographic

- **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**

When reactor head is opened for decommissioning purposes.

- **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**

None

- **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**

PC-3a: If vessel failed, photos/videos of debris and crust, debris and crust extraction, possible hot cell exams, and possible subsequent testing (1F1, 1F2, and/or 1F3).

High-resolution images (photos/videos) of debris and crust both in the as-found state and during extraction, and chemical analysis to determine composition and oxidation state. Imaging should be sufficient to provide insights into material characteristics (i.e., particle bed versus crust material, and if crust material, the morphology and extent of cracking if possible). A sufficient number of samples should be selected to estimate the spatial variations in composition. Elemental analysis of samples should look for fuel, structural, and concrete components. Evaluations should determine the approximate proportions of U/Zr/SS/Boron from the drive tubes below the vessel head and the corium samples retrieved from the cavity region.

- **Benefits (Safety, Operational, Economic, D&D, or other benefits):**

Structural characteristics of the material are important for supporting tooling design for removal; chemical analysis important for criticality evaluations. These same data are important for improving reactor safety analysis models and accident management.

- **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**

Benchmark and reduce uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. MCCI phenomena are important for assessing combustible gas generation during late phase accident progression, as well as the extent of attack on containment structures. It is important to reduce uncertainty in this phenomenon because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.

- **Methods/Tools Needed to Collect Information or Data**

  - Irradiation resistant high resolution imaging system
  - Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence
  - Ultimately, D&D cutting and removal tools able to extract materials

- **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**

Near-term and/or later (Sample removal possible within next 2 years).

- **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**

Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development.

Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

■	**Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
	PC-3b: If vessel failed, 1F1, 1F2, and 1F3 PCV liner inspections (photos/videos and metallurgical exams) High-resolution images (photos/videos) of PCV liner, with particular emphasis in regions contacted by core debris if that occurred. In areas that were contacted, the imaging should be sufficient to provide insights into the nature/extent of heat transfer and/or thermo-chemical attack on the liner (e.g., distortion/displacement and extent of ablation if that occurred). A sufficient number of samples should be selected in eroded areas to determine if the boundary temperature during erosion was determined by simple melting or by eutectic formation. Evaluations should determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from corium samples retrieved from the cavity region.
■	**Benefits (Safety, Operational, Economic, D&D, or other benefits):**
	For D&D, plugging leaks in the liner will reduce the extent of water leakage from the PCV and determining leakage locations via liner examinations is crucial to this process. These same data are important for improving reactor safety analysis models and accident management.
■	**Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
	Benchmark and reduce uncertainty in models for predicting liner thermal heatup and attack by core debris for ex-vessel accident scenarios. Improved knowledge will be used to enhance accident management strategies.
■	**Methods/Tools Needed to Collect Information or Data**
	• Irradiation resistant high resolution imaging system. • Laser imaging systems to reconstruct liner distortion and/or ablation profiles.
■	**Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
	Near-term and/or later.
■	**Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
	None.

■ **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3c: If vessel failed, photos/video, RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor. High-resolution images (photos/videos), RN surveys, and sampling of 1F1, 1F2, and 1F3 pedestal wall and floor. Imaging should be sufficient to provide insights into structural integrity and/or damage incurred during the accident. A sufficient number of samples should be selected to estimate the RN distribution on the pedestal wall and floor. Evaluations should determine the approximate proportions of U/Zr/SS/Boron from corium samples retrieved from the cavity region.
■ **Benefits (Safety, Operational, Economic, D&D, or other benefits):**
Determining the pedestal wall and floor structural integrity as well as RN distributions is important for safety evaluations of D&D activities.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting structure heatup and degradation during a severe accident. It is important to reduce uncertainties in this area since heat sink inside the PCV can impact predictions of water availability to cool core debris. Improved knowledge will be used to enhance accident management strategies.
■ **Methods/Tools Needed to Collect Information or Data**
• Irradiation resistant high resolution imaging system • Robotic methods for extraction of samples for determining RN distributions.
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

■ **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-3d: If vessel failed, 1F1, 1F2, and 1F3 concrete erosion profiles; photos/videos and sample removal and examination from PCV. High-resolution images (photos/videos) of concrete erosion with possible sample removal and elemental analysis. Imaging should be sufficient to estimate the total volume of relocated core material and the damaged volume of concrete. In addition, imaging should be of sufficient resolution to characterize the morphology (e.g., cracks, gaps, etc.) of the debris and concrete. A sufficient number of samples shall be selected to estimate the spatial variations in composition and oxidation state of relocated materials. Elemental analysis of samples should look for fuel, structural, and concrete components. Evaluations should determine the approximate proportions of U/Zr/SS/Boron from the the corium samples retrieved from the cavity region.
■ **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
Required for D&D facilitate planning for debris removal, and also for evaluation of the mechanical integrity of critical structures such as the reactor pedestal. Desired for improving reactor safety analysis models and accident management.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting molten core concrete interaction (MCCI) phenomena. MCCI is important in assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in MCCI phenomena because it affects strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
■ **Methods/Tools Needed to Collect Information or Data**
• Irradiation resistant high resolution imaging system • Hot cell elemental analysis system • D&D cutting and removal tools able to extract materials
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

■	**Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
	PC-3e: If vessel failed, photos/videos of structures and penetrations beneath 1F1, 1F2, and 1F3 to determine damage and corium hang-up. High-resolution images (photos/videos) of structures and penetrations with retained corium. Imaging should be sufficient to estimate the total volume of relocated core material and the damage to structures and penetrations.
■	**Benefits - Safety, Operational, Economic, D&D, or other benefits:**
	Required for D&D facilitate planning for debris removal and for evaluation of the mechanical integrity of critical structures such as the reactor pedestal. Desired for improving reactor safety analysis models and accident management.
■	**Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
	Benchmark and reduce uncertainty in models for predicting the mass and heat content of material that relocatesfrom the reactor vessel, which in turn, affects PCV gas temperature, PCV pressure, and the potential for MCCI.
■	**Methods/Tools Needed to Collect Information or Data**
	• Irradiation resistant high resolution imaging system • Hot cell elemental analysis system • D&D cutting and removal tools able to extract materials
■	**Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
	Near-term and/or later (Robotic examinations underway).
■	**Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
	Initial findings from from 1F2 and 1F3 suggest that a non-negligible amount of core debris may be held up on structures below the reactor vessel. System analysis codes should be exercised assuming a range of core debris holdup in a situation that is not cooled by water to investigate the impact of heat sources not covered by water on PCV gas phase temperature and pressure.

■	**Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
	PC-5: Photos/videos and temperatures of 1F1, 1F2, and 1F3 main steam lines and ADS lines to end of SRV tailpipes, including instrument lines.
■	**Benefits (Safety, Operational, Economic, D&D, or other benefits):**
	BWR AM Strategies (plant mods, etc.) and better simulations for training; Potential PWR impacts (e.g., modeling, AM strategies, etc.).
■	**Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
	To determine RPV failure mode. Initial examinations should focus on identifying failure mode(s) and location(s). For example, if images indicate that vessel lower head failure occurred, images should be of sufficient resolution to determine if the failure was a gross unzipping or a limited area. If images suggest that vessel depressurization was due to penetration failure, images should be of sufficient resolution to determine the number, type(s) [e.g., control rod drive, instrument tube, and/or drain line], and failure mode(s) [e.g., tube ejection and/or or tube rupture]. Evaluations of MSLs and ADS lines should also focus on identifying failure mode(s) and location(s). Initial images may not be able to detect failure locations. Hence, dose surveys, gamma camera (3D) images, and temperature measurements may be needed to detect where radiation has leaked from the RPV.
■	**Methods/Tools Needed to Collect Information or Data**
	• Irradiation resistant high resolution imaging system (1 mm to 1 cm gaps or cracks). • Dose survey meter or gamma camera (3D image). • Thermal imaging to observe hot spots (> 100 C increases)
■	**Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
	Near-term and/or later.
■	**Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
	None**.**

■ **Name(s) /Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-6: Visual inspections of 1F1, 1F2, and 1F3 SRVs including standpipes in the torus and drywell (interior valve mechanisms)
■ **Benefits - Safety, Operational, Economic, D&D, or other benefits:**
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.).
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
To determine if there was any failure of SRVs and associated piping.
■ **Methods/Tools Needed to Collect Information or Data**
• Irradiation resistant high resolution imaging system
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

■ **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-17: Chemical analysis of upper layer of sediment on drywell floor at the X-100B penetration location in 1F1. The upper surface of the sediment is ~ 30 cm above drywell floor **PC-18**: Evaluate nature of material below the sediment at the X-100B penetration location in 1F1 to determine if core debris is present (1F1) **PC-19**: Chemical analysis (XRF) of black material discovered on CRD exchange rail in 1F2 at X-6 penetration location **PC-20**: Chemical analysis of black material on 'existing vertical wall structure' in 1F1 picture outside pedestal doorway These four information requests are for determining the chemical composition of materials observed at locations in 1F1 (i.e., sediment and underlying material on the drywell floor below the X-100b penetration, and on existing vertical structure near the pedestal doorway), and black material discovered on the CRD exchange rail in 1F2 from the X-6 penetration. Elemental analysis of samples should look for fuel, structural, and concrete components and should also include a measurement of oxygen content if possible.
■ **Benefits (Safety, Operational, Economic, D&D, or other benefits):**
Required for D&D; desired for improving reactor safety analysis models and accident management.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting vessel failure, in-vessel cladding oxidation and hydrogen production, holdup on ex-vessel structures, and MCCI phenomena. Vessel failure, holdup on ex-vessel structures, and MCCI phenomena are important for assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in these phenomena because they affect strategies for venting and water addition. Data from PC-19 and PC-20 evaluations can be used to confirm RPV failure, and additional analysis of PC-19 can be used to assess the extent of in-vessel cladding oxidation. Data from PC-18 evaluations can be used to determine if core debris is present and the X-100B location, thereby providing insights on the extent of core debris location which is also a critical uncertainty impacting accident management strategy. Knowledge gained from these analyses will be used to enhance these strategies. Data from PC-17 can be used to determine if the composition of this sediment varies with height.
■ **Methods/Tools Needed to Collect Information or Data**
• Hot cell elemental analysis system and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence (XRF). • Robotics systems for collecting samples, and for probing the sediment at X-100B location to determine the (loose material) sediment depth.
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Evaluation of this information requires composition information for concrete (to distinguish between sand and concrete), and would benefit from chemical analysis of seashore sand located at the site.

Technical Supplement for PC-18 Evaluation

Examinations at the X-100b location in 1F1 (located ~ 130 degrees counter-clockwise from the pedestal doorway opening) indicate a layer of material covering the drywell floor that is ~ 30 cm deep. This material was identified during the initial entries through the X-100b penetration in 2012, and was reconfirmed during later entries in 2016 that provided additional data on the actual depth of the material. It is known that additional sediment had not accumulated at this location over the intervening four years because unique surface characteristics (i.e. greyish blue material thought to be lead) were still present. The upper surface of the material was determined to be loose sediment. It is not known whether this sediment extends down the entire 30 cm depth, or whether the sediment is a partial layer covering other material such as core debris.

There are a variety of potential sources for this sediment material that may include decomposed/flaked paint, thermal insulation, cable insulation, sand/sediment from low quality seawater injection, aerosol from core concrete interaction, among others. If the material is sand entrained with the seawater that was injected or concrete aerosol from core-concrete interaction, then it may be possible to determine the origin based on the relative proportions of dominant concrete oxides such as SiO_2, CaO, Al_2O_3, and MgO in the sediment. For sand from seawater injection, analysis of a sample of beach sand obtained at the site would provide definitive data for direct comparison with elemental analysis data obtained from a sample of the sediment. In lieu of this information, the composition of sand from 12 different beaches along the east and west coasts of Japan have been reported in the literature.[ggg] The compositions of key compounds varied considerably; i.e., 61.4-99.2 wt% SiO_2, 0.04-5.8 wt% CaO, 1.3-19.0 wt% Al_2O_3, and 0-2.0 wt % MgO. In terms of mass ratios of key elements, the resultant ratio for Si-Al is determined to range from 2.7 to 67 and for Si to Ca is determined to range from 6.9 to 1600.

Fortunately, the composition of concrete from the Daiichi site has also been measured for two samples to provide data for comparison to these ranges; see Table C-4.[hhh] Iron shown in Table C-4 is not considered in the current discussion as it could arise from corrosion (rust) of steel within the PCV, of which there is a massive amount. The corresponding mass ratios for Fukushima Daiichi concrete for the key elements in the two concrete samples are Si/Al: 3.6-4.2, and Si/Ca: 2.7-3.5. The Si/Al ratio for the concrete versus sand samples from around the island of Japan cannot be discriminated. However, the range of Si/Ca ratios does not overlap. In particular, the range boundaries are separated by a factor of ~ 2. Thus, if the Si/Ca ratio is lower and in the range of 2.7-3.5, it is likely concrete aerosol from MCCI. Conversely, if it is higher, ~7 or above, it is likely sand from seawater injection. Aerosol from core-concrete interaction also nominally contains a small amount of fuel (U)[iii] which would also be a discriminating factor.

Table C4. Composition data from analysis of two concrete samples at 1F site.[hhh]

Sample No.	Mass %			
	Al	Ca	Fe	Si
1	7.0 ±1	7.8 ±1	3.6 ±1	25 ±1
3	6.5 ±1	9.1 ±1	3.3 ±1	27 ±1

ggg C. Igarashi and N. Shikazono, "Sound-Producing Sand in Japan: Major Element Composition and Its Minerals Determined by X-ray Diffraction and X-ray Fluorescence," Analytical Sciences, Vol. 19, pp. 1371-1374 (2003).

hhh "Concrete Reference Reactor Building Core Boring Sample Radioactivity Analysis," August 29, Heisei 20, Japan Atomic Energy Agency (in Japanese)

iii J.K. Fink, D. H. Thompson, B. W. Spencer, and B. R. Sehgal, "Aerosol and Melt Chemistry in the ACE Molten Core-Concrete Interaction Experiments," High Temperature and Materials Science, Vol. 33, pp. 51-76 (1995).

■ **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
PC-21 Images from examinations in 1F3 X-53 penetration. High-resolution images (photos/videos) of external surfaces of RPV (especially of vessel failure locations); of material collected on structures beneath vessel (e.g., cables, control rod drives, support structures, gratings; and of concrete erosion on floor of PCV. Imaging should be sufficient to estimate the total volume of relocated core material at each location and the damaged volume of the vessel, any ex-vessel structures, and the concrete. In addition, imaging should be of sufficient resolution to characterize the morphology (e.g., cracks, gaps, etc.) of the debris and concrete. Measurements of dose rates and collection of samples for elemental analysis is desired. Ultimately, a sufficient number of samples shall be selected to be able to estimate the spatial variations in composition. Elemental analysis of samples should look for fuel, structural, and concrete components.
■ **Benefits (Safety, Operational, Economic, D&D, or other benefits):**
Required for D&D; Desired for improving reactor safety analysis models and accident management.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Benchmark and reduce uncertainty in models for predicting vessel failure, holdup on ex-vessel structures, and molten core concrete interaction (MCCI) phenomena. Vessel failure, holdup on ex-vessel structures, and MCCI phenomena are important for assessing combustible gas generation during late phase accident progression. It is important to reduce uncertainty in these phenomena because they affect strategies for venting and water addition. Improved knowledge will be used to enhance accident management strategies.
■ **Methods/Tools Needed to Collect Information or Data**
• Irradiation resistant high resolution imaging system • Hot cell elemental analysis system • Systems to obtain dose rate measurements and collecting fluid or small particles during FY2017 examination (if it is possible). • Ultimately, D&D cutting and removal tools able to extract materials
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (Young's modulus, linear expansion, ultimate strength, hardness, tensile strength, etc.) for cutting tool development and for model development. Evaluation of this information may require composition information for concrete (to distinguish between sand and concrete).

■ **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RPV-1b: Photos/videos, probe inspections, and sample exams of 1F1, 1F2, and 1F3 MSLs. Interior examinations of MSLs at external locations, looking for evidence of thermal/pressure strain and/or rupture, including nature of any ruptures such as fish mouth or more global rupture. Would like to know the approximate size of any rupture failure locations.
■ **Benefits (Safety, Operational, Economic, D&D, or other benefits):**
Improved AM strategies; Improved simulations for training.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
▪ Code assessments and validation of current structural yielding modeling used in codes ▪ Possible model improvements.
■ **Methods/Tools Needed to Collect Information or Data**
Visual inspection
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later.
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
None.

■ **Name(s)/Description(s) - Name (ID #), description of desired information, unit (1F1, 1F2, 1F3), and location from where it should be obtained (PCV, RPV, Reactor Building):**
RPV-4 and RPV-5: a)1F1, 1F2, and 1F3 shroud inspection (between shroud and RPV wall); Photos/videos and sample removal and oxidation testing. b) 1F1, 1F2, and 1F3 shroud head integrity and location (photos/videos, and metallurgical exams) c) Photos/videos of 1F1, 1F2, and 1F3 shroud inspection (from core region) d) Photos/videos of 1F1, 1F2, and 1F3 core plate and associated structures e) Remote mapping of 1F1, 1F2, and 1F3 core through shroud wall from annular gap region (muon tomography and other methods, if needed) f) Mapping of end state of core and structural material (visual, sampling, hot cell exams, etc.) A few samples of core material should be examined to determine the approximate proportions of Uranium/Zirconium/Stainless Steel/Boron from any upper core remnants, core plate accumulations, drive tube accumulations above bottom of vessel, and any accumulations on the lower vessel head region. Results can be used to determine roughly the tendency for spatial separation of lower melting and metallic rich core debris materials from the more ceramic remnants and by implication, the temporal separation of relocation events. The same information is needed for the drive tubes below the vessel head and the corium samples retrieved from the cavity region. This information is needed to validate code assumptions of phase interactions during core degradation.
■ **Benefits (Safety, Operational, Economic, D&D, or other benefits):**
BWR reactor safety analysis models have very significant uncertainties related to in-core damage progression modeling. These inspections can provide information that can help resolve the generally agreed upon largest uncertainties in BWR severe accident modeling. These uncertainties influence the understanding of containment response during a severe accident and are thus relevant to informing accident management.
■ **Use/Motivation - Tie to specific use (code models, maintenance, operations, accident management, etc.) and timeframe when needed:**
Resolve large uncertainties for in-core damage progression at BWR reactor-scale. These inspections are relevant to addressing areas where testing has been unable to reproduce key areas of BWR in-core debris relocation behavior at reactor-scale. The pathways by which debris relocate within the core-region influence the potential for rapid pressurization of containment to occur (e.g., due to rapid steam or hydrogen generation). The acquisition of knowledge to reduce uncertainties in this area can refine severe accident models, enhancing the effectiveness of accident management training.
■ **Methods/Tools Needed to Collect Information or Data**
• Irradiation resistant high resolution imaging system • Hot cell elemental analysis system, and/or in-situ elemental analysis using Laser Induced Breakdown Spectroscopy (LIBS) and/or X-ray Florescence • Ultimately, D&D cutting and removal tools able to extract materials
■ **Roadmap Timeframe - Near-term and/or later; Tie to specific inspections planned for 1F1, 1F2 and 1F3:**
Near-term and/or later (Sample removal possible within next 2 years).
■ **Preparatory or Follow-on Research/Supporting Information (beyond what is obtained from 1F examinations)**
Obtaining /using this information may require additional material property and coolability testing (e.g., oxidation state of in-core debris). Refined understanding of mechanical properties of retrieved in-core debris, however, are of significant benefit to the design and development of cutting tools. Refined understanding of in-core damage progression will require effort to refine analytical models for this phase of a severe accident.

Appendix D

Mid-and-Long-Term Roadmap Phase II Activities

[Figures provided courtesy of NDF; Reference [107]

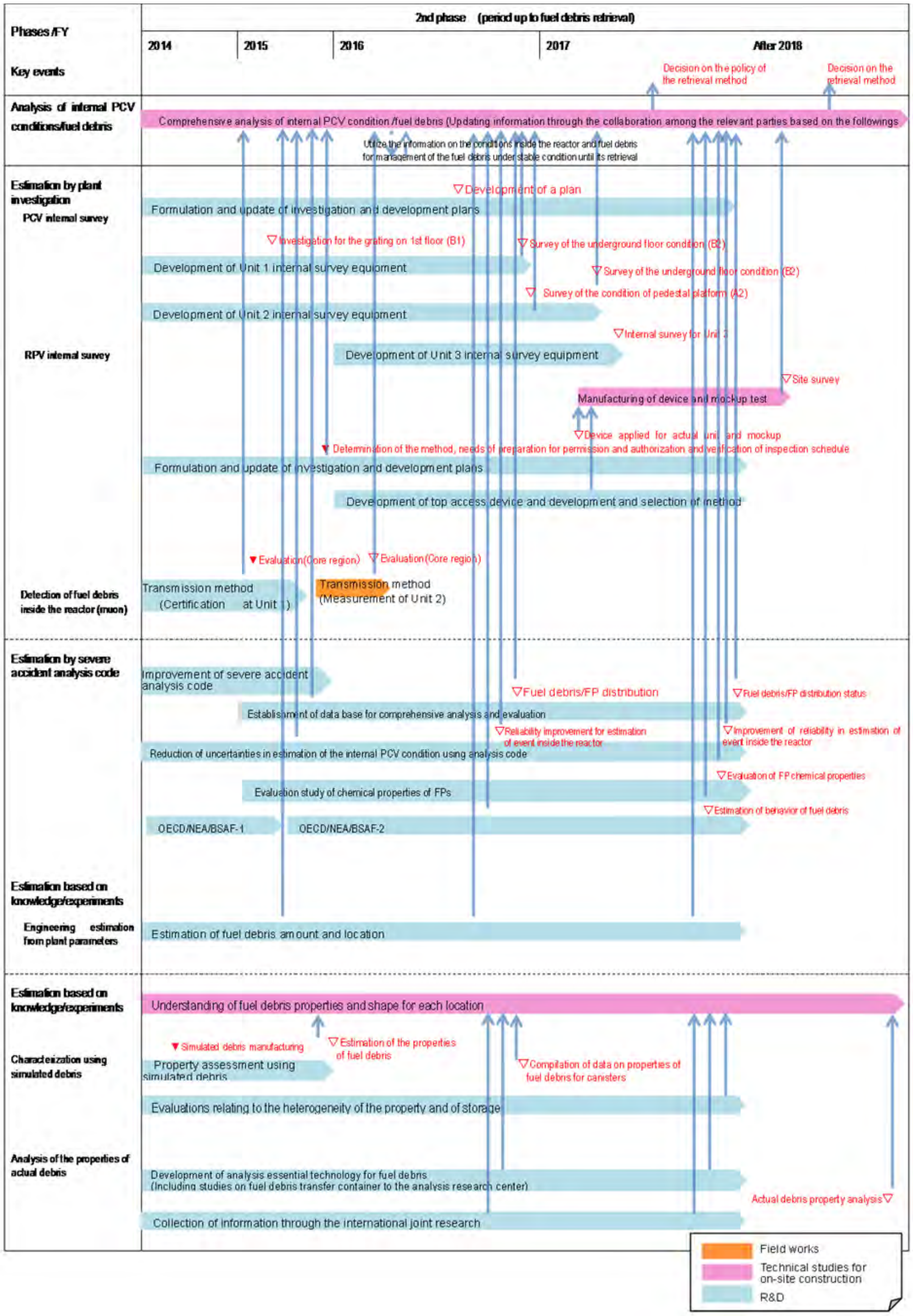

Figure D-1. Phase II actions to characterize conditions within the reactor and of fuel debris. (Courtesy of NDF [107])

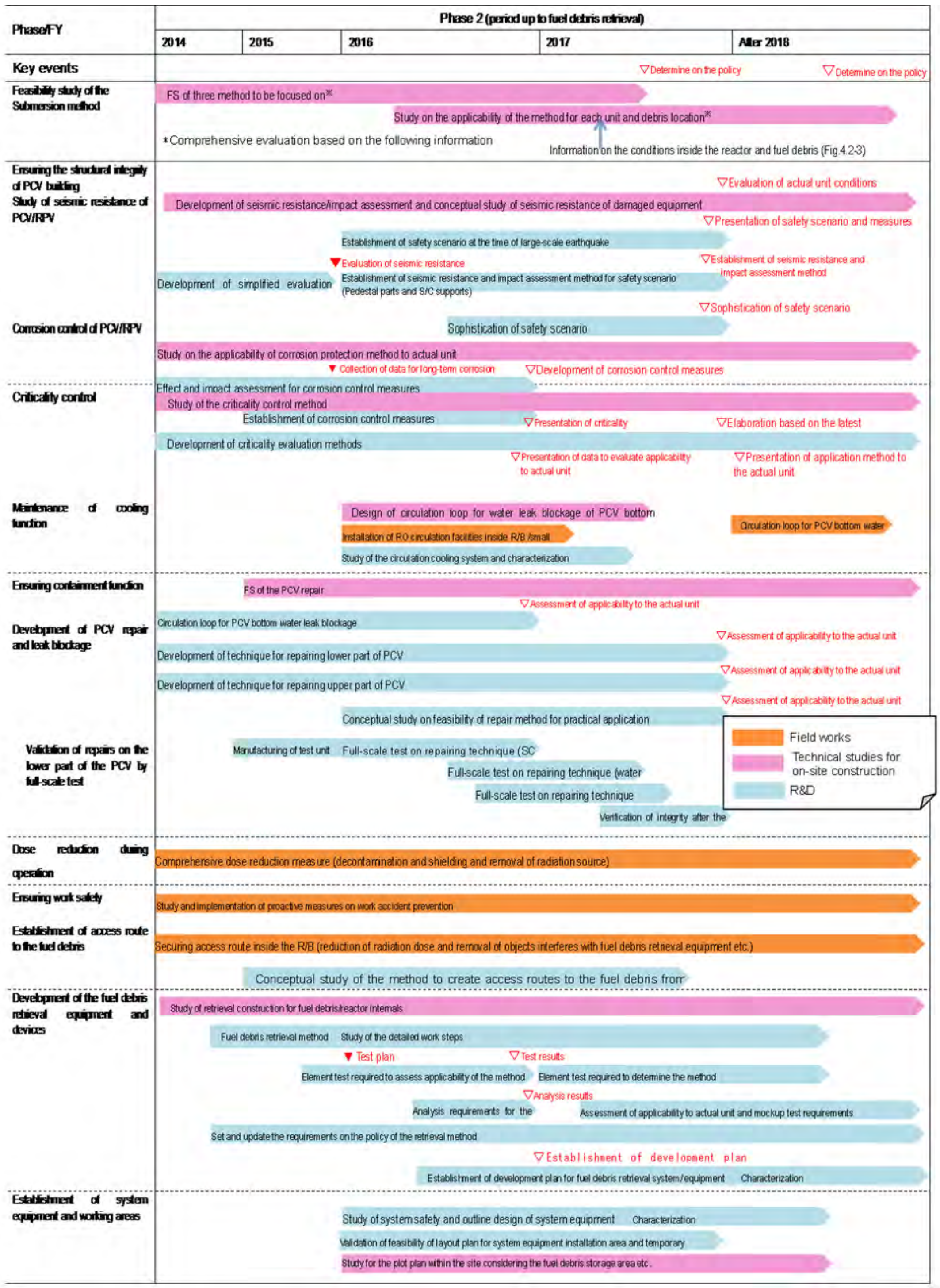

Figure D-2. Phase II actions to complete feasibility study for fuel debris retrieval method. (Courtesy of NDF [107])

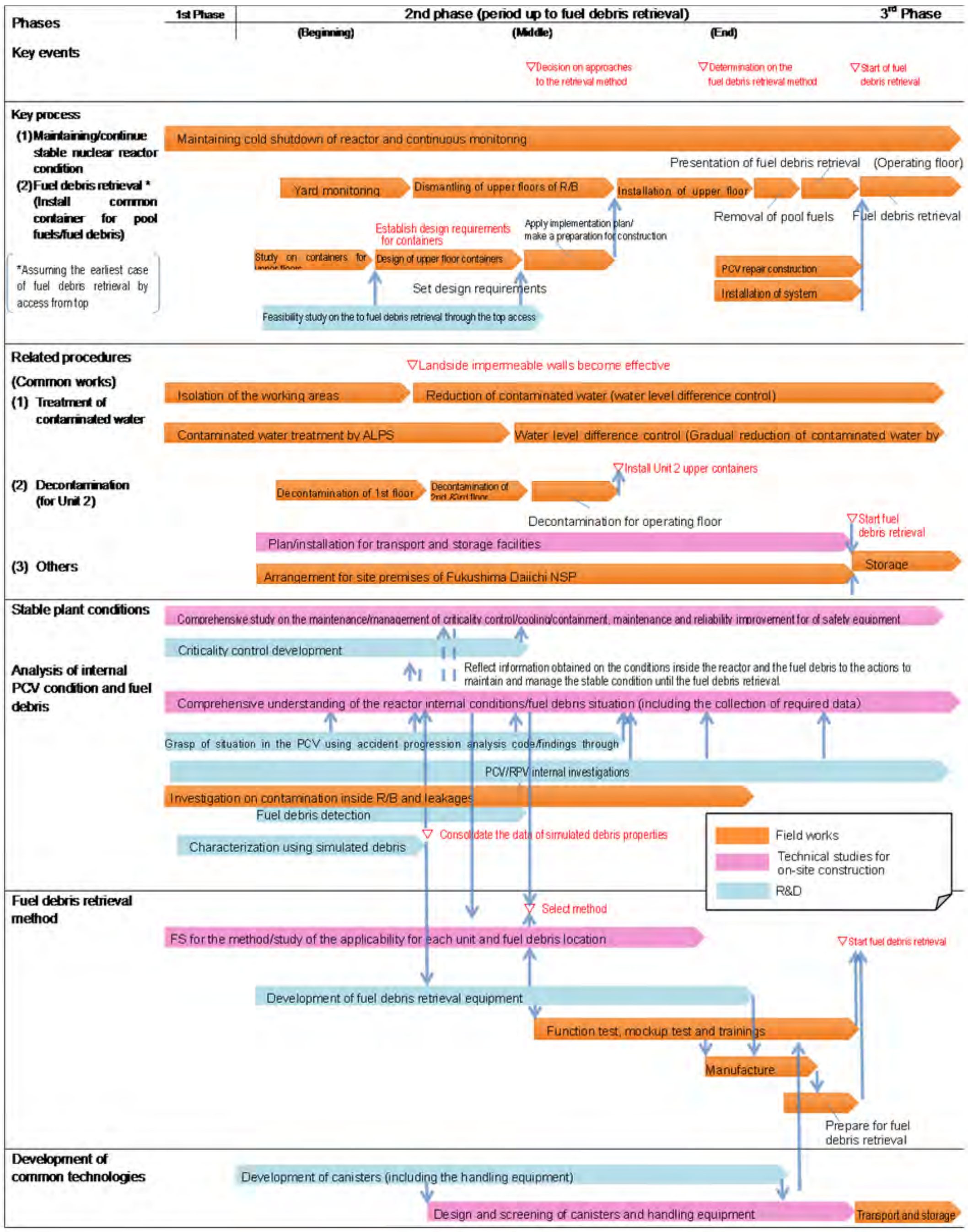

Figure D-3. Phase II actions required to complete entire process of fuel debris retrieval. (Courtesy of NDF, [107])

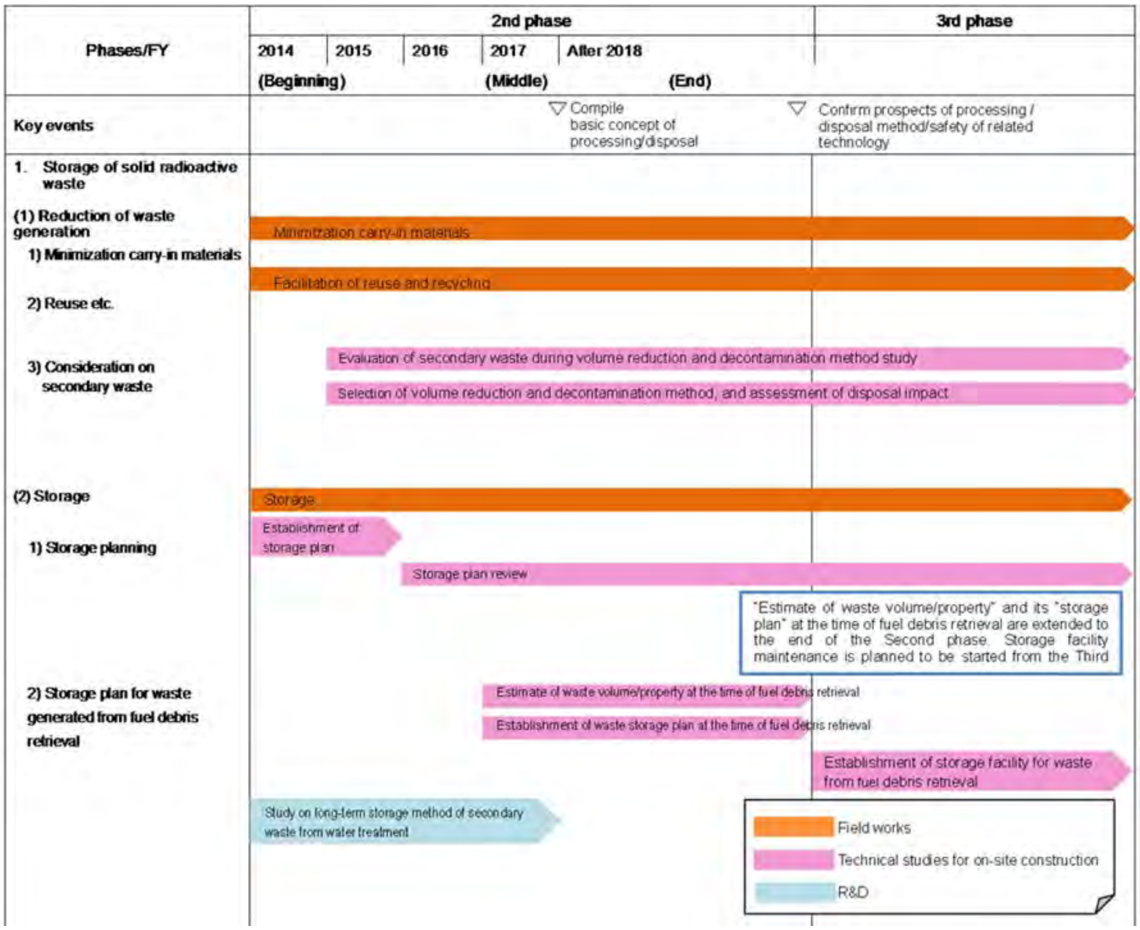

Figure D-4. Phase II and III actions for waste management (Sheet 1 of 2). (Courtesy of NDF [107])

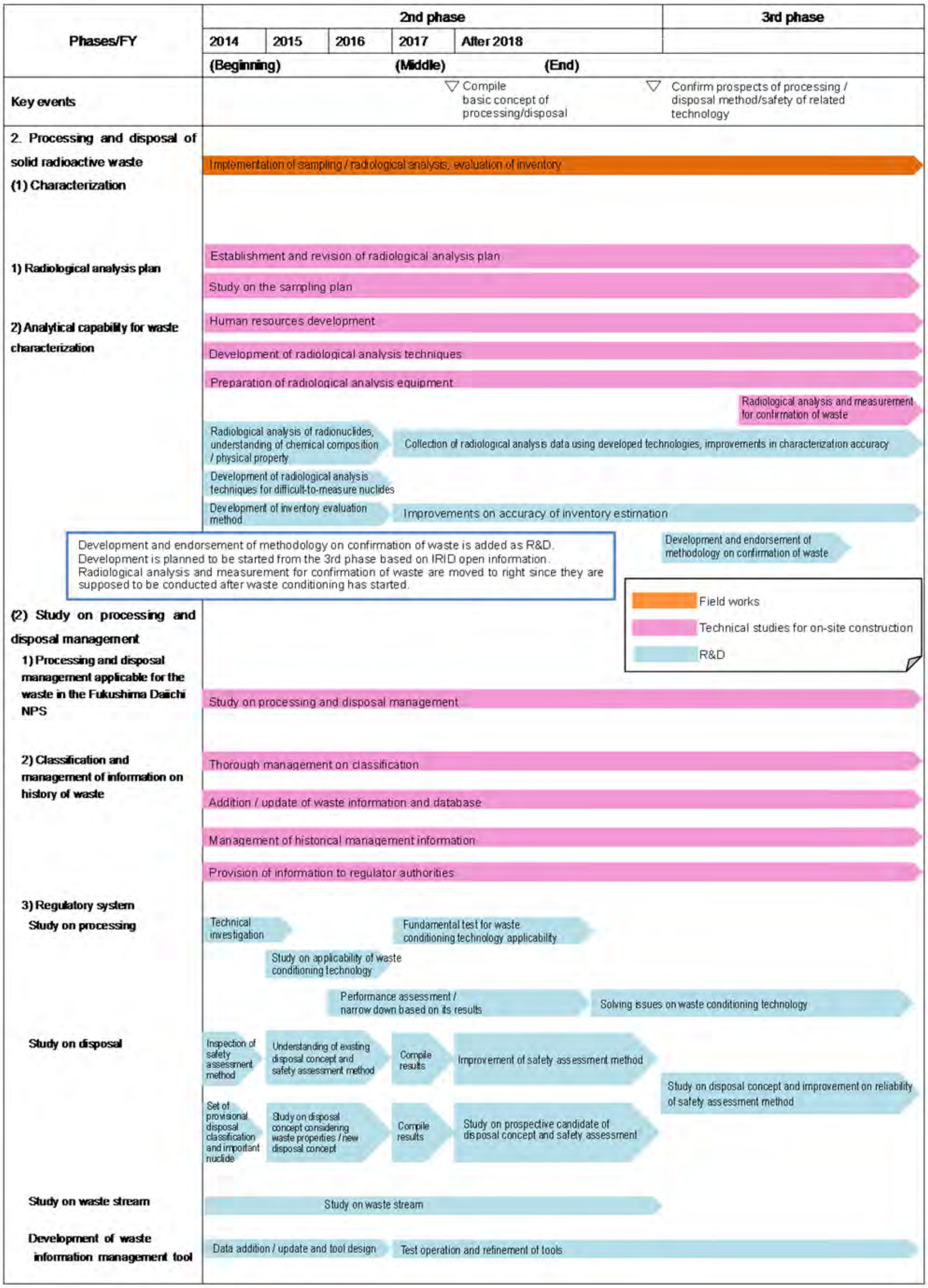

Figure D-5. Phase II and III actions for waste management (Sheet 2 of 2). (Courtesy of NDF, [107])

Appendix E

US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima

The Meeting Agenda is included in this section. Additional information is documented in [231].

US-Japan TMI-2 Knowledge Transfer and Relevance to Fuksuhima Meeting Agenda

US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting

Meeting Agenda
October 11-13, 2016

<u>Tuesday, October 11, 2016</u>

7:15 AM	Pickup at Hotel for Badging and Travel to Conference Center	All J. Carmack, INL
8:00 AM	Meeting Logistics	D. Peko US DOE-NE P. Humrickhouse, INL
8:10 AM	Welcome/Introduction/Opening Remarks	US: J. Kotek, US DOE-NE D. Skeen, US NRC M. Peters, INL Japan: M. Hirai, METI T. Yamada, NRA J. Zimmerman, DOE-EM
8:45 AM	Current Status of Daiichi Estimation on Fuel Debris Location	T. Yamada, NRA S. Mizokami, TEPCO
10:00 AM	*Break*	All
10:15 AM	***<u>Group 1 Category 1: Overview of the Defueling and Cleanup at TMI-2 - Organizational Overview</u>***	Leads: L. Barrett/J. Byrne W. Bixby, US Facilitators: Damian Peko, US M. Nakagawa, NDF, Japan
11:30 pm	*Working Lunch Presentation:* Current Status of TMI-2	All M. Casey and G. Halnon, FENOC Facilitator: Damian Peko, US
12:30 PM	***<u>Group 1 Category 1: Overview of the Defueling and Cleanup at TMI-2 - Key Milestone Categories and Associated Challenges</u> (Case Study - Defueling Safety Case Development and Approval Process Including Criticality Safety)***	Lead: C. Negin Panelists: B. Franz/ D. Mueller/J. Byrne/ W. Austin/J. Tarpinian US Facilitators: Richard Lee, US M. Nakagawa, NDF, Japan
2:15 PM	*Break*	All
2:30 PM	***<u>Group 1 Categories 2 through 4: PEIS, TMIPO, and Environmental and Safety Reviews</u>***	Lead: M. Masnik Panelists: D. Marksberry/ L. Barrett, US Facilitators: Tom Wellock, US M. Nakagawa, NDF, Japan
5:15 PM	*Adjourn*	All
5:30 PM- 7:30 PM	*Hosted Reception (Heavy appetizers and refreshments)* *Location: EIL Meeting Center*	All

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US-Japan TMI-2 Knowledge Transfer and Relevance to Fuksuhima Meeting Agenda (Cont.)

US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting

Meeting Agenda
October 11-13, 2016

Wednesday, October 12, 2016

7:30 AM	Pickup at Hotel	All P. Humrickhouse, INL
8: 00 AM	***Group 2 Category 3: Data Acquisition and Core Damage Characterization***	Lead: W. Bixby/B. Franz Panelists: L. Barrett /J. Byrne US Facilitators: Joy Rempe, US T. Washiya, JAEA, Japan
10:00 AM	*Break*	All
10:15 AM	***Group 2 Category 4: Defueling and Fuel Handling/Shipping Planning***	Lead: C. Negin Panelists: J. Byrne/B. Franz Facilitators: Paul Dickman, US S. Takizawa, TEPCO, Japan
12:15 PM	*Working Lunch Presentations:* NRC and DOE Severe Accident Research	All Richard Lee and Damian Peko
1:15 PM	***Group 2 Category 5: Core Debris Shipping and Storage (at INL)***	Lead: P. Winston Panelists: T. TaylorS. Birk R. Spruill/M. Tyacke US Facilitators: Jon Carmack, US N. Okuzumi, IRID, Japan
3:00 PM	*Break*	All
3:15 PM	***Group 2 Category 6: New Cleanup and Decommissioning Technologies***	Lead: M. Plys Panelists: R. Demmer/ K. Croft/ J. Byrne/ W. Austin US Facilitators: Kevin Kostelnik, US T. Kogai, NDF, Japan
4:30 PM	Closing Session	All
5:30 PM	*Adjourn*	All

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US-Japan TMI-2 Knowledge Transfer and Relevance to Fukushima Meeting

Meeting Agenda
October 11-13, 2016

TMI/Fukushima Knowledge Transfer INL Tour
October 13, 2016

Thursday, October 13, 2016

Le Ritz Hotel, Idaho Falls, ID

7:15 Begin loading tour bus .... P. Humrickhouse

7:30 Depart for INTEC Complex

INTEC Complex........ Mark Stubblefield / Mark Argyle

8:30	Arrive INTEC Guard Gate	
8:45	Group A 11 people – Tour 603 (Group B and C on Bus to ISFSI)	30 min
9:15	Group B 11 people – Tour 603 (Group A and C on Bus to ISFSI)	30 min
9:45	Group C 11 people – Tour 603 (Group A and B on Bus to ISFSI)	30 min

10:15 Depart INTEC

Advanced Test Reactor

10:30 Arrive ATR Guard Gate

11:00 Tour Advanced Test Reactor .... Don Miley

Noon Depart ATR

Materials and Fuels Complex

12:30 Arrive MFC .... Don Miley

12:45 Lunch L&O Conference Room – Presentation Overview of MFC .... John Wagner

1:45 Facility Tours .... Don Miley
- Fuel Conditioning Facility (FCF)
- Irradiated Material Characterization Laboratory (IMCL)

3:30 Depart MFC for Idaho Falls .... Don Miley

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