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Safety insights from forensics evaluations at Daiichi

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A B S T R A C T

Although it is clear that the accident signatures from each affected unit at the Fukushima Daiichi Nuclear Power Station [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 (TEPCO) to characterize potential hazards and to ensure the safety of workers involved with cleanup activities. This paper reports initial results from the US Forensics Effort to utilize examination information obtained by TEPCO to enhance the safety of existing and future nuclear power plant designs. In this paper, three examples are presented in which examination information, such as visual images, dose surveys, sample evaluations, and muon tomography examinations, along with data from plant instrumentation, are used to obtain significant safety insights 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 confirm actions, such as the importance of water addition and containment venting, that are emphasized in updated guidance for severe accident prevention, mitigation, and emergency planning.

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]. Similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2), [1] examinations at Daiichi offer the international community a means to obtain prototypic severe accident data from boiling water reactor (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. Examinations from Daiichi are of special interest because multiple reactors were affected and the accident signature from each reactor appears unique. In addition, these units may offer data related

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Table 1
Prioritization of possible examination activities [16].

Region	Examination information classification[a,b]			
	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 / pressure vessel lower head	****		***	**
-Instrumentation		****	***	
Reactor pressure vessel (RPV)				
-Upper vessel penetrations	****		***	**
-Upper internals	****	***	**	*
-Core regions & shroud	****		***	**
-Lower plenum	****		***	**

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

to the effects of salt water addition, vessel failure, containment failure, and core/concrete interactions after vessel failure. Results of these examinations can be used to update severe accident modeling and accident management practices, thereby enhancing safety of all light water reactors (LWRs) world-wide.

1.1. Objectives

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

- ***Objective 1:*** Develop consensus US input for high priority time-sequenced examination tasks and supporting research activities that can be completed with minimal disruption of Tokyo Electric Power Company (TEPCO) decontamination and decommissioning (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 related to predicting severe accident progression.
- – Gain insights related to severe accident equipment performance.
- – Provide insights beneficial to TEPCO D&D activities.
- – Confirm and, if needed, improve guidance for severe accident prevention, mitigation, and emergency planning.
- – Update and/or refine Objective 1 information requests.

As discussed in Reference [2], the Government of Japan developed a Roadmap to outline activities required for successful D&D of the Daiichi NPS. An important aspect of this US effort is to NOT adversely affect D&D activities outlined in this roadmap. Objective 1 activities were initiated in 2014, and Objective 2 activities were initiated in 2015.

1.2. Approach

Program objectives are completed by a panel of over 30 US experts from industry, universities, and national laboratories. Experts from the US Nuclear Regulatory Commission (NRC), the DOE, and TEPCO also participate. The primary source for examination information is the TEPCO website [3]. Presentations and information provided by representatives from TEPCO [4–11] and in TEPCO reports documenting unconfirmed and unresolved issues received special attention in the forensics effort[12–15]. Results from this effort are documented in annual reports [16,17]. Section 2 of this paper presents representative results and insights from these evaluations, and Section 3 describes how these results and insights are implemented to improve safety of the US operating fleet and reduce uncertainties in modeling severe accident progression.

2. Representative results and insights

2.1. Objective 1

To complete Objective 1, US experts reviewed TEPCO D&D plans and available examination information and developed a list of consensus information needs. Table 1 summarizes, at a high level, the activities identified by the expert panel for addressing information needs from the affected units at Daiichi (see [16] for detailed information needs). The expert panel concluded that some information is needed from all locations to obtain a complete picture of the entire accident progression and conditions that occurred in each unit during these events. Hence, the expert panel prioritized information needs 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 ranks the priority of the information need by the placing more asterisks in boxes for higher priority types of information. As shown in this table, the expert panel typically placed the most emphasis upon information obtained from visual examinations, such as videos and photographs, and near-term proximity exams, such as dose surveys. In general, the consensus was that such information was the easiest to obtain, and could provide critical information related to whether additional examinations were required.

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

- Area 1 – Component/System Performance
- Area 2 – Radionuclide Surveys/Sampling
- Area 3 – Core Debris End-state
- Area 4 – Combustible Gas Effects

Sections 2.3–2.5 present example Objective 2 evaluations with representative insights and results in these higher priority areas. Each example draws upon information from multiple areas.

2.2. Objective 2

Available information in higher priority areas was evaluated using a focused approach that first identifies key questions of interest, such as:

- Is this information consistent with current understanding of severe accident progression?
- Are analytical model improvements needed to predict observed phenomena?
- Can information be used to confirm/improve severe accident guidance?
- Does information provide insights related to D&D (e.g., structure integrity, worker radiation exposure, etc.)?

The evaluation process to address key questions was completed using three steps. First, available information was reviewed by leads in each area. This information included results from calculations completed by DOE contractors and other US and international organizations. Second, initial findings by leads in each area were discussed with the panel of US experts and representatives from DOE, NRC, and TEPCO. Third, conclusions from these reviews, along with insights for industry, were documented in annual reports. To illustrate the process, three representative examples are highlighted in this paper. Additional examples may be found in Reference [17].

2.3. Example 1 – component / system performance (Areas 1 and 2)

Results from TEPCO examinations to support D&D activities are of interest in assessing component and system performance. Of special interest are visual images (i.e., pictures and videos), dose surveys, water level measurements and samples, and temperature information.

2.3.1. Available information

Table 2 summarizes findings by US experts related to the observed status of various penetrations and equipment based on inspections performed by TEPCO. This table indicates notable differences in 1F1, 1F2, and 1F3 degradation. Possible causes for these differences include variations in unit designs, the availability/functionality of backup cooling systems, the ability to externally inject water during the accidents, the ability to vent the primary system and containment during the accidents, and differences in combustible gas effects at each unit.

Available information highlights different leakage points and the possibility for multiple leakage points. Identifying leakage locations, leakage timing, and the conditions causing this leakage was of special interest because of industry efforts to update severe accident guidance. Thus, 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 at each unit.

Table 2
Summary of inspection of various components.[a] References cited in this Table: [6,18–22,25–46].

Area	1F1	1F 2	1F 3
X-100B PCV penetration	Possible melted shielding material [6] No damage observed on outside [18]	TBD	TBD
X-51 PCV penetration	TBD	No damage observed; pressurized water could not penetrate blockage in standby liquid cooling system line [19 , 20]	TBD
X-53 and X-54 PCV penetration (HPCI pipe penetration)	Traces of flow and white sediment noted [11]	No damage observed[21]	No damage observed [22]
X-6 PCV penetration	TBD	Melted material [23, 24]	No observed damage from inside [25]
Equipment hatch	TBD	TBD	Water puddle [26, 27] unknown source
Personnel hatch and nearby penetrations	No major damage observed [28]	TBD	TBD
HPCI pipe penetration	No damage observed, but high dose rates measured [28, 29]		
Traveling In-core Probe 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) [11,30]	Dose surveys do not indicate leakage from PCV through TIP guides. High dose levels in samples of materials from TIP indexer [31]	
Wetwell (WW) vacuum breaker line	Leakage on expansion joint of one line (X-5E) [32]	TBD	TBD
DW/WW vent bellows	Water leakage attributed to vacuum line above [32]	No leakage observed [33]	TBD
DW sand cushion drain pipe	Leakage [34]	No leakage observed [33]	TBD
SC water level	Almost full [35]	Consistent with torus room water level [35, 36]	Believed 'almost full' but not confirmed [35]
DW Water Level	~2.8 m[35]	~0.3 m[35]	~6.5 m[35]
Torus room	Partially flooded [37,38]	Partially flooded [39]	Partially flooded [39]
	Rusted handrails/equipment [6]	Non-rusted handrails/ equipment [6,40]	Non-rusted handrails/ equipment [6,41]
	TBD	Some room penetrations tested, no leakage observed [42]	TBD
MSIV room	Limited view obtained.[10]	Water leakage cannot be observed [43]	Leakage Line D near bellows [44]
DW Head	Reactor well shield plug displaced [45]	Possible leakage [46]	Possible Leakage [46]
RCIC or other low SC piping	TBD	Suspected leak location, not confirmed [6]	TBD

[a] Nomenclature: 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. (For interpretation of the references to colour in this table legend, the reader is referred to the web version of this article.)

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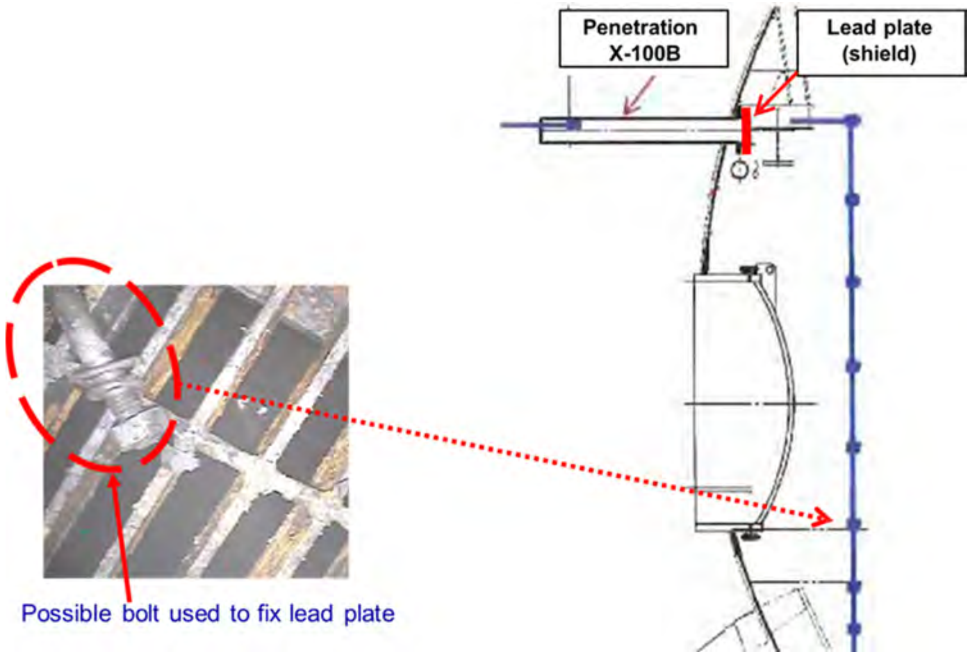


Fig. 1. Visual images within 1F1 X-100B penetration. (Courtesy of TEPCO [6]).

2.3.1.1. 1F1. Pressure data [3] from 1F1 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. The calculated saturation temperature for this measured peak pressure (neglecting any release of high temperature steam or hydrogen from the vessel) is 172 °C/342 °F. However, as shown in Fig. 1, examinations within the X-100B penetration of 1F1 revealed that the lead shield cover plate was missing. In order for this lead plate to have melted, gas temperatures inside the drywell exceeded 328 °C/622 °F, the melting point for lead.

2.3.1.2. 1F2. Insights related to peak temperatures within the 1F2 PCV are available from visual examinations, radiation survey information, and temperature and pressure data. As shown in Fig. 2, visual examinations of material from the X-6 penetration suggest that either the chloroprene cable cover or silicon seal material melted and dribbled out of this penetration indicating peak temperatures greater than 300 °C/572 °F, and the dribbling pattern suggests that relocation occurred at low pressure (rather than a high pressure ejection). Plant data [3] indicate that 1F2 peak pressures reached 0.75 MPa/109 psia on March 15, 2011. Temperature data were not available until March 21, 2011. The calculated saturation temperature for the measured peak pressure (neglecting any release of high temperature steam or hydrogen from the vessel) is 168 °C/334 °F.

2.3.1.3. 1F3. Insights about 1F3 leakage come from photos and dose surveys. As shown in Fig. 3, steam appears to be escaping at locations near the drywell head, and higher dose rates were measured near the drywell head. Both of these observations are consistent with a failure of the drywell head, perhaps due to drywell bolt expansion or seal degradation from high temperatures and pressures within the PCV. Plant data [3] 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. The saturation temperature for the measured peak pressure, neglecting any release of high temperature steam or hydrogen from the vessel, is 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 developed multiple leakage locations.

2.3.2. Insights and limitations

Many of the leakage points identified for 1F1, 1F2, and 1F3 are not routinely modeled in systems level codes for predicting severe accident progression (MAAP [47], MELCOR [48], etc.). Both MAAP and MELCOR simulations predict drywell 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 analysis codes.

The potential for multiple penetrations to fail due to seal degradation is also of interest to industry for accident management strategies. In the US, new BWR Owners Group (BWROG) and PWR Owners Group (PWROG) severe accident management guidance places a high priority on venting the PCV when pressures and temperatures reach prescribed limits. For BWRs, containment conditions can be very close to the PCV design basis pressure and temperature, but guidance in NEI-13-02 [49] 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 Daiichi, available information confirms that maintaining containment below design basis conditions and reducing containment conditions are appropriate strategies.

Fig. 4 shows available peak Daiichi information on a figure provided in NEI 13-02 industry guidance for venting[49]. The DW vent is assumed to have a design temperature of 285 °C/545 °F, and containment penetration degradation temperatures are based on engineering evaluations and testing information in the literature. Black lines in Fig. 4 correspond to peak temperatures inferred from currently-available 1F1 and 1F2 examination information. These values are consistent with the values assumed to cause degradation in NEI 13-02; thus, available Daiichi information support revised industry guidance recommending that operators maintain containments at low pressure.

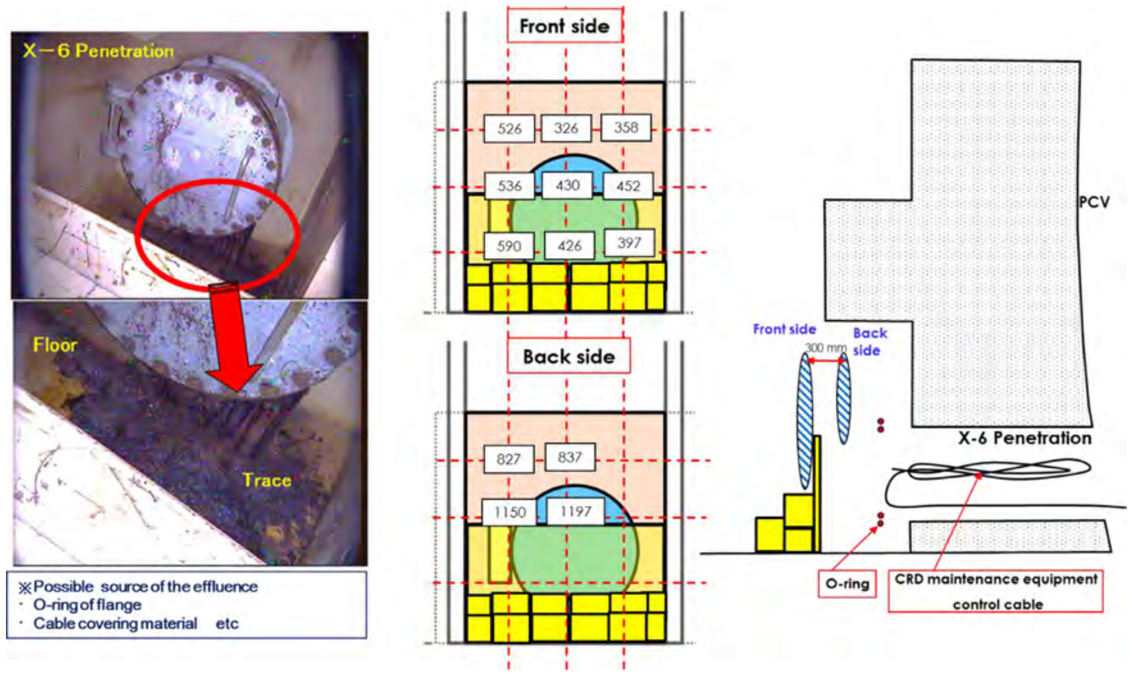


Fig. 2. Photographs and radiation surveys (in mSv/hr) near 1F2 X-6 penetration (values measured in 13 locations). (Courtesy of TEPCO [23,24]).

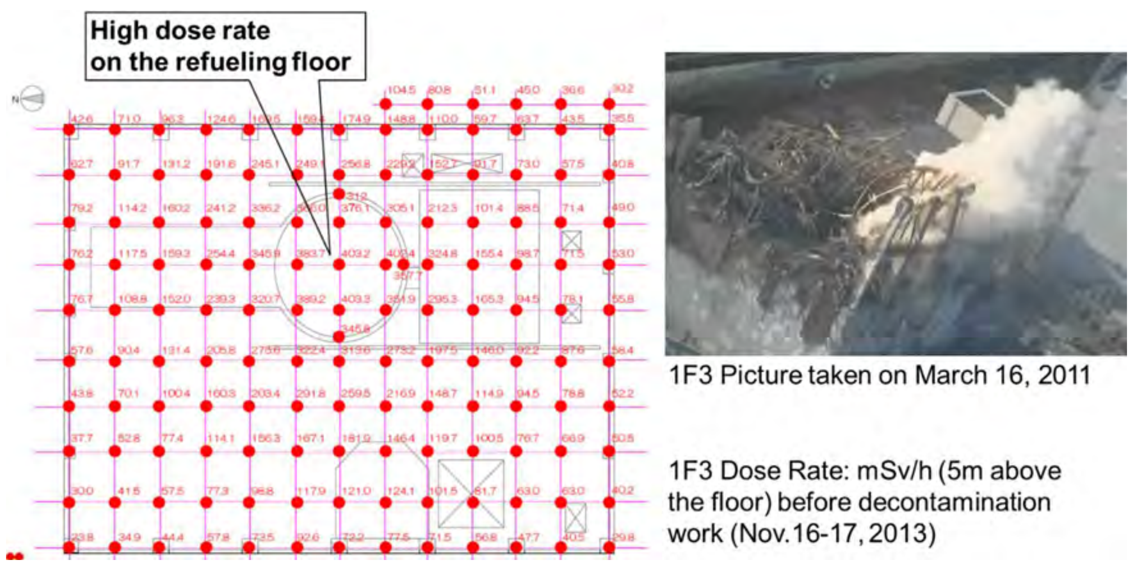


Fig. 3. 1F3 radiation survey (value in mSv/hr measured on November 16-17, 2013 at 5 m above the floor at points shown on grid) and photograph taken on March 16, 2011 (Courtesy of TEPCO [6, 8]).

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

2.3.3. Recommendations

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

Example 1. ***Recommendation 1:***

Perform sensitivity studies with MAAP and MELCOR for a range of containment and primary system conditions on containment failure location and size to evaluate the effects on radiological releases (timing, amount) and the impact on accident progression.

TEPCO examinations indicate that several containment penetrations and components are leaking in the three units. The failure of multiple containment penetrations or even a specific penetration noted in Table 2 is not predicted in best-estimate MAAP or MELCOR simulations of these accidents. Sensitivities for each unit would provide insight into which failure likely caused de-

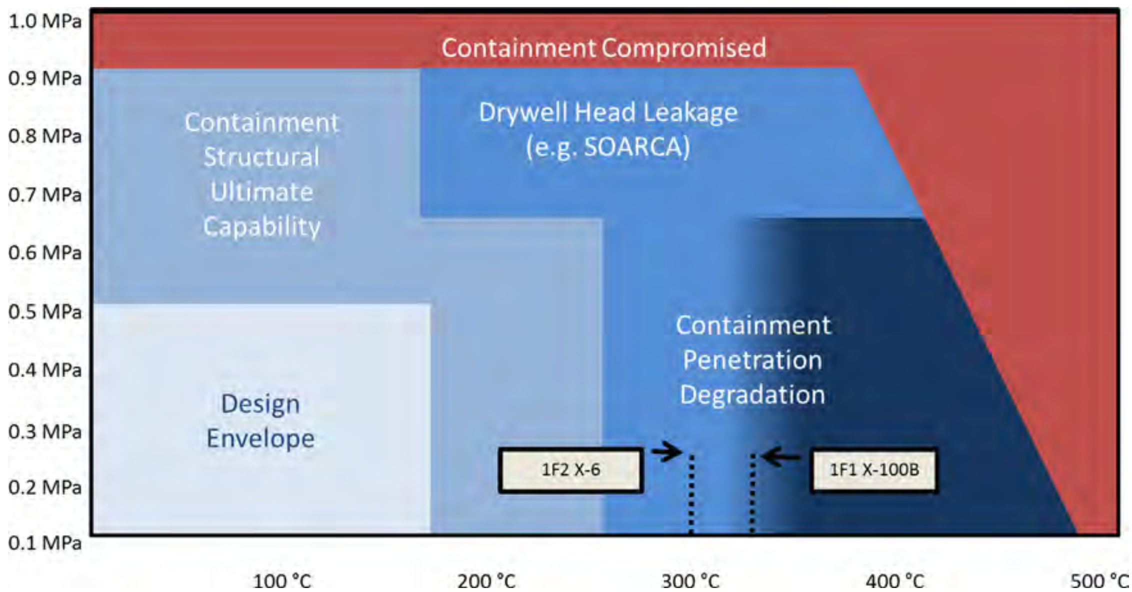


Fig. 4. Containment pressure/temperature curve with available 1F1 and 1F2 information. (Graphic courtesy of NEI [49] as modified by Jensen Hughes).

pressurization, the conditions under which such a failure occurred, and the effect of multiple failures. Severe accident modeling, particularly as it pertains to probabilistic risk assessment, typically does not evaluate containment impairment in a mechanistic manner. In many models, the containment impairments are assumed to develop using the containment structural response evaluations. Hence, continued analytical effort would be of value as part of Daiichi accident simulations to assess the potential for drywell head flange impairment due to high upper drywell temperatures. Additional photographs of the upper drywell structure could help quantify peak drywell dome temperatures that developed.

Example 1. *Recommendation 2:*
Continue to review available information and to update Table 2.

The expert panel concurred that information in Table 2 was useful for summarizing the status of various components within the three units. Information in this table, coupled with code predictions, dose measurements, and available plant instrumentation information, could provide insights related to the timing of failure for various components. Determining whether failures occurred before or after vessel breach is important for predicting and understanding radionuclide transport during an accident and useful in verifying information contained in revised industry severe accident guidance.

2.4. Example 2 – debris end-state (Areas 1 and 3)

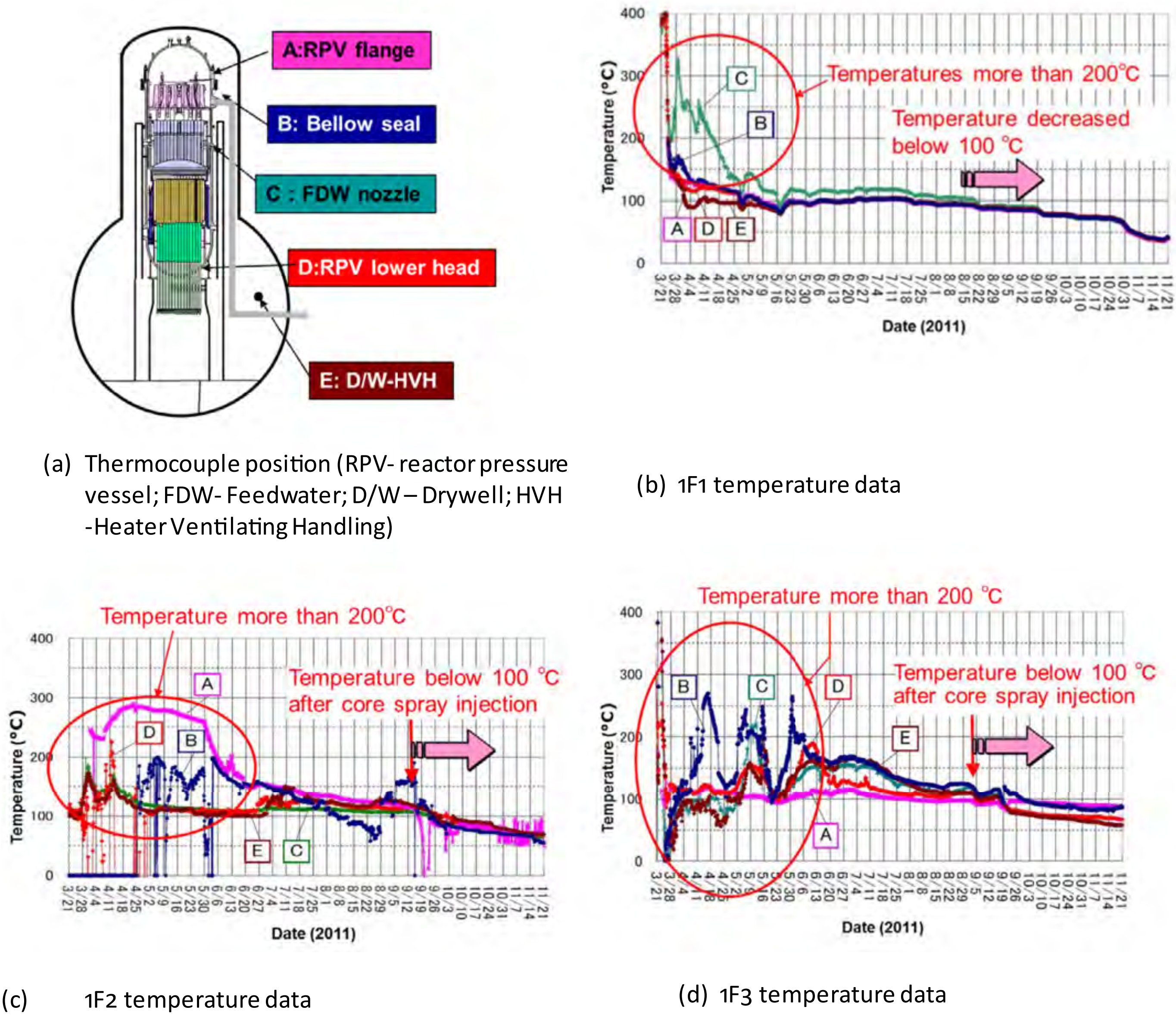
Several types of information are available to indirectly infer the debris end-state location in each unit. There are data from plant instrumentation, such as temperature information obtained during and immediately after the accident, gas concentration data from the gas treatment systems installed at the plant, and neutron and gamma detector data from monitoring systems installed at the plant. In addition, there are visual images and sensor data obtained from inspections conducted by robots within the RB and the PCV. Last, results from muon tomography evaluations provide important insights regarding debris end-state location. This section reviews selected information and associated findings related to debris end-state based on this information.

2.4.1. Thermocouple measurements

Fig. 5 provides thermocouple (TC) measurements [6] obtained from 1F1, 1F2, and 1F3, respectively, for several months following the accident. These measurements provided the first indication of where core debris likely resides, and equally important, where it is not. In particular, water injection was shifted from the fire protection to feedwater (FDW) injection systems for the three units in the April-May timeframe. However, data indicate 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 FDW for a BWR is introduced slightly higher than the center of the RPV (see Fig. 5a) and then flows down along the exterior surface of the core barrel to the core inlet. This, along with indications from water level instrumentation not increasing, led TEPCO and the technical support community to conclude that there may be significant leakage path(s) in the bottom region of the reactor vessel for all three units (e.g., BWR recirculation pumps are known to leak under severe accident conditions [50]). In such cases, some fraction of the coolant was able to bypass the core debris; and the material was not fully cooled.

On this basis, TEPCO changed the water injection from the FDW system to the core spray (CS) system in the September 2011 timeframe for 1F2 and 1F3, while this change was made in late December 2011 for 1F1. This injection method directly introduces a water spray from above the core. As shown in Fig. 5c and d, 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, as shown in Fig. 5b, this change had little if any impact on 1F1; RPV temperatures had already fallen below saturation. This suggests 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. 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 earlier US [51, 52] as well as international [53] code predictions of debris locations for the three units based on modeling conducted relatively soon after the acci-



(a) Thermocouple position (RPV- reactor pressure vessel; FDW- Feedwater; D/W – Drywell; HVH -Heater Ventilating Handling)

(b) 1F1 temperature data

(c) 1F2 temperature data

(d) 1F3 temperature data

Fig. 5. Temperature measurements following the accident. (Graphics courtesy of TEPCO [6]).

dent. Since that time, further refinements of these analyses have not changed these same basic conclusions. The picture is clearest for 1F1, which was essentially a hands-off station blackout (e.g., an event in which all onsite and offsite alternating current (ac) power is lost and in which no successful mitigating actions are taken) until ~15 h into the accident sequence. At this point, operators were able to start reflooding the core with seawater. However, the predictions are less consistent for 1F2 and 1F3 where operators were able to maintain some degree of core cooling by various means for the first several days of the accident. Uncertainties arise about the effectiveness of water injection (due to elevated PCV pressure), as well as the effectiveness and extent of backup cooling system operation under severe accident conditions; this situation was compounded by a general lack of functioning instrumentation (and the fact that surviving instrumentation had in many cases been pushed well outside their qualification envelope).

Aside from these general observations, TC data in Fig. 5 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 the structure that correspond to the locations where the measurements are made in Fig. 5. 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 Fig. 5 may provide further insights on likely debris distributions. This type of analysis is relevant to DOE-sponsored MAAP and MELCOR comparisons [54]; i.e., these two codes predict quite different in-vessel core melt behavior and subsequent primary system failure modes. These modeling differences may be reflected in long-term RPV temperature predictions that could, by comparison with TC data, provide an indication of likely relocation mode(s), one of the key questions being addressed in these comparisons.

The results of these measurements as well as the supporting code analyses help to inform D&D activities. In particular, the results indicate that TEPCO will likely be faced with the need to remove core debris not only from the RPVs for at least two units, but also from the PCV for 1F1. Finally, these measurements have also been very useful in terms of informing severe accident guidance

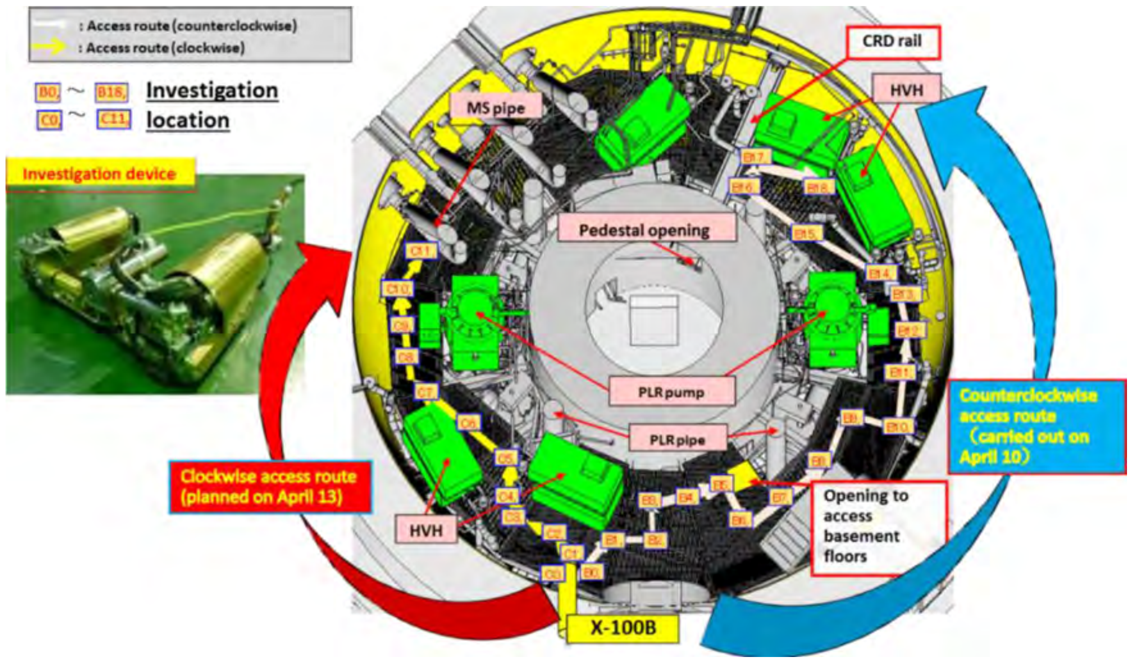


Fig. 6. Location of X-100B in 1F1 PCV and robotic examination paths completed on April 10, 2015. (Courtesy of TEPCO [55]).

(SAG) development. 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.

2.4.2. Visual images from inspections within the PCV

Other valuable information obtained by TEPCO regarding conditions inside the PCV for 1F1 has been obtained by robotics examinations through a containment penetration; i.e., the 'X-100B' penetration; see Fig. 6 [55]. Prior to the accident, this penetration was shielded on the interior of the PCV to reduce the radiation level in the reactor building. As discussed in Section 2.3.1.1, the first piece of information gathered when this penetration was opened was that the lead shielding had melted during the accident. Lead melts at 328 °C/ 622 °F; temperatures this high in the PCV atmosphere are hard to rationalize unless one postulates vessel failure and core debris discharge into the pedestal region.

Upon gaining access through this penetration, TEPCO initially lowered a video camera through the catwalk to the drywell floor to measure water level, temperatures, and radiation levels inside the PCV. These inspections showed that there was no core debris on the drywell floor at this location, which is ~130° from the pedestal doorway (Fig. 6). This finding was important as it provided a data point for assessing predictions of ex-vessel core melt spreading based on MAAP and MELCOR pour scenarios as calculated with MELTSPREAD [56]. As is evident from Fig. 7, the measurement indicates that the MELTSPREAD prediction of spreading distance based on MAAP pour conditions over-predicts the actual spreading distance. Conversely, this single data point is insufficient to gauge the accuracy of the MELTSPREAD-MELCOR prediction as the spreading prediction for that case is limited to the vicinity of the pedestal doorway. Nonetheless, this initial observation through this penetration has been useful in reducing the range of possibilities regarding the extent of melt spreading in 1F1.

2.4.3. Visual images within the reactor building

Another important finding regarding ex-vessel behavior is that the sand cushion drain line is leaking in 1F1[6]. 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. MELTSPREAD analyses [56] of liner heatup (Fig. 8) indicate that the liner would not have ablated based on either the MAAP low pressure (LP) or MELCOR pour scenarios [51,52]. However, analyses predict that the liner would have heated significantly, making it vulnerable to failure by creep rupture due to the elevated containment pressures (~ in excess of 1.5 times the design pressure). Hence, liner failure is consistent with code predictions and with measured radiation levels in the 1F1 reactor building.

2.4.4. Muon tomography evaluations

Muon tomography measurements using scintillation detectors are another information source that has been extremely valuable for evaluating debris end-state conditions for 1F1 (see Fig. 9) [10]. Using this approach, high density fuel should show up as dark regions in the images due to muon attenuation. As shown in Fig. 9, the core region appears to be essentially devoid of core material. The findings for 1F1 are consistent with previously described system-level code analyses. [51,52]

2.4.5. Insights and limitations

Available inspection information and analysis results have led to the following important insights about debris end-state location.

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).

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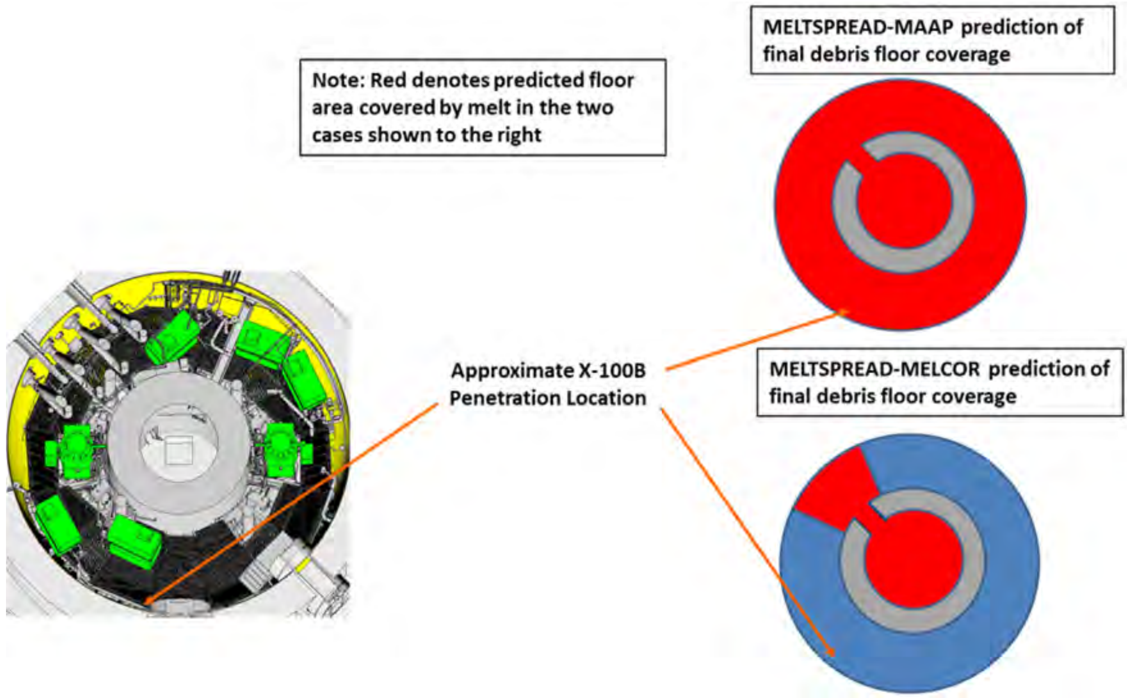

Fig. 7. Approximate Location of X-100B penetration relative to predictions of core debris spreading in 1F1. (Courtesy of TEPCO and ORNL [55, 56]).

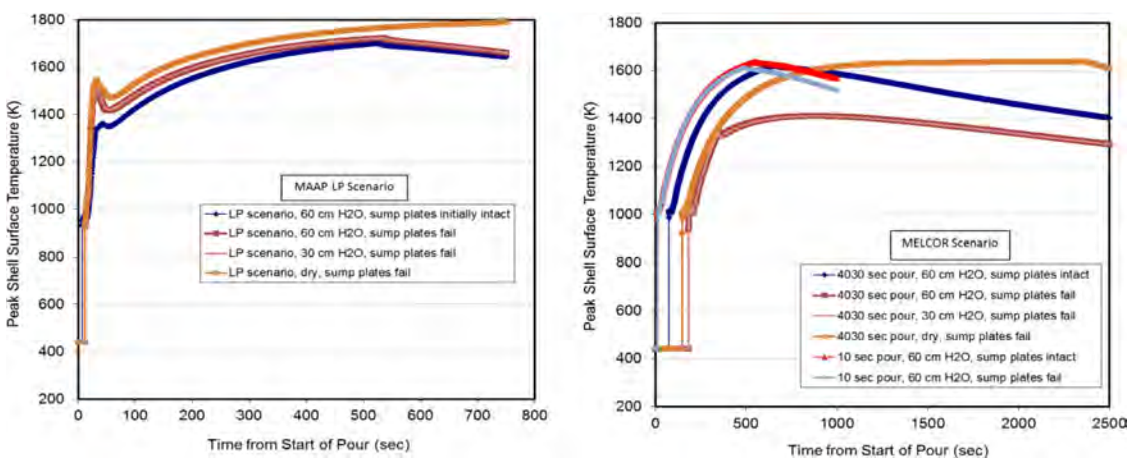

Fig. 8. MELTSPREAD predictions of liner heatup due to heat transfer from impinging melt for 1F1 based on a MAAP low pressure (LP) scenario (left) and MELCOR (right) melt pour conditions. (Courtesy ORNL, [56]).

– Additional analyses of structures and components using refined nodalization schemes may provide more detailed information about the mass, composition, and heat content of relocated debris.

Visual images within PCV and reactor building:

– Images obtained from the 1F1 X-100B penetration indicate peak temperatures of 328°C/ 622°F (due to the absence of lead shielding that melts at this temperature). Such high temperatures support the hypothesis that core debris relocated to the pedestal region.
– The absence of debris in images obtained from a camera inserted into the X-100B penetration suggests that US analyses obtained with MELTSPREAD using results from MAAP are overpredicting spreading of debris in 1F1; however, results obtained with MELCOR are indeterminate at this time.
– Images showing that the 1F1 sand cushion drain line is leaking suggest a failure in the PCV liner. Such failures could be from creep rupture of the PCV liner due to the elevated containment pressures (in excess of 1.5 times the design pressure) at the time of the accidents. Liner failure in 1F1 is consistent with MELTSPREAD code predictions and with measured radiation levels in the 1F1 reactor building.

Muon tomography investigations:

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

Although informative, the amount of information obtained thus far on debris locations is limited. In particular, there have been no direct determinations of the location of the core debris. Observations obtained with remote cameras have shown where the core debris is *not* in the PCV for 1F1, which in itself is valuable information. Muon tomography methods are also providing data on debris locations, but the resolution of the images is limited. Finally, TEPCO

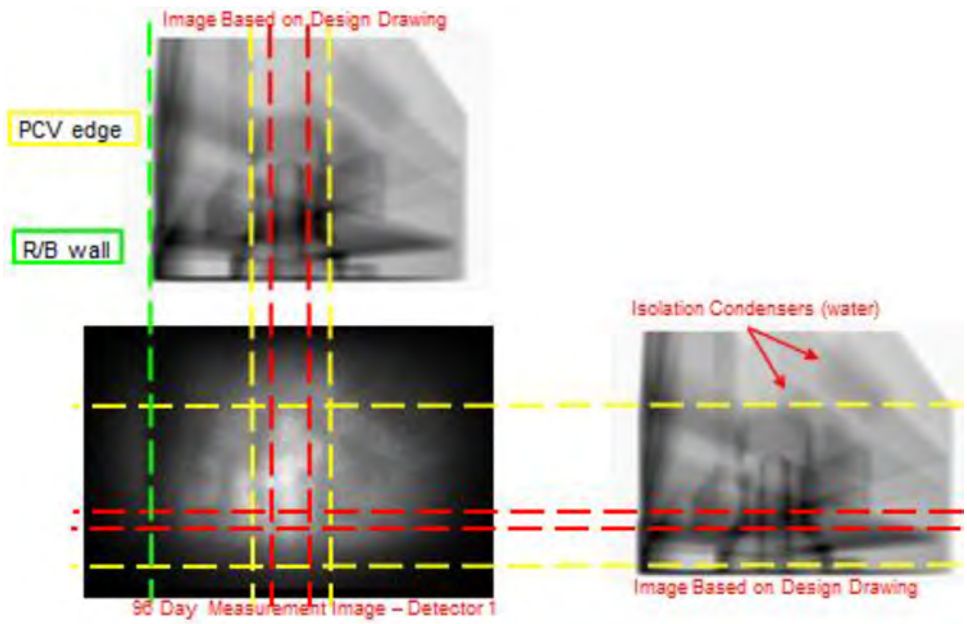

Fig. 9. Images of 1F1 obtained using muon tomography with scintillation detectors (the lower left image is measured; the other two images were calculated. Dashed lines are provided to show location of identified geometrical features). (Courtesy of TEPCO [10]).

has effectively used TC measurements on the RPV, coupled with variations in water injection flowrate and location, to infer debris location. One limitation with this 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 limitations in the available information, the information has nonetheless 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.

2.4.6. Recommendations

System-level (e.g., MAAP and MELCOR) and separate effect (e.g., MELTSPREAD) code analyses have provided tangible predictions for evaluation against the debris end-state information being obtained by TEPCO. In a rough sense, these calculations can be considered to be half-blind benchmarking exercises that are useful in gauging the accuracy and adequacy of the models as additional information on debris end-state conditions becomes available. This initial evaluation has identified several additional analysis activities that would be beneficial in terms of benchmarking the models, reducing modeling uncertainties, and further informing D&D efforts at the site.

Example 2. *Recommendation 1:*
Refine MAAP and MELCOR RPV nodalization schemes with the aim of predicting the measured temperatures shown in Fig. 5.

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 recent MAAP-MELCOR comparisons [54].

Example 2. *Recommendation 2:*
Perform a MELTSPREAD-CORQUENCH analysis for 1F2 that is similar to the 1F1 analysis. [56]

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 h in an unregulated mode. A 1F2 evaluation 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 [6], no evidence of liner failure has been found for 1F2. Hence, the 1F2 analysis would provide a means for rationalizing the finding that the liner in 1F1 has been damaged.

In summary, the forensic analysis activity related to debris end-state conditions has illustrated the intrinsic value of information obtained by TEPCO for providing insights on accident progression, informing SAG development, and validating severe accident codes that are used for plant safety evaluations. However, there is still a need for higher fidelity data related to debris locations. In this early stage of the D&D process, initial insights are being gained on ex-vessel conditions. Due to the high radiation levels, the only practical means for obtaining this data is through stand-off methods which TEPCO has actively and successfully pursued; i.e., muon tomography and robotics inspections. These methods have already proven to be valuable, and efforts are underway by TEPCO to enhance these examination techniques.

2.5. Example 3 – combustible gas generation and transport (Areas 1, 2, and 4)

Several types of information provide insights about combustible gas generation and transport during the events at Daiichi. There are data from plant instrumentation, such as temperature information obtained during and immediately after the accident and seismic acceleration data. In addition, there are visual images, such as videos and photos, taken during the event and during post-accident examinations. Dose surveys also provide insights about gas generation and transport. Important information on this topic was found in TEPCO reports evaluating damage associated with explosions at the affected units [57, 58] and TEPCO reports evaluating unresolved issues [12–15]. This section reviews selected information and associated insights obtained from this information.

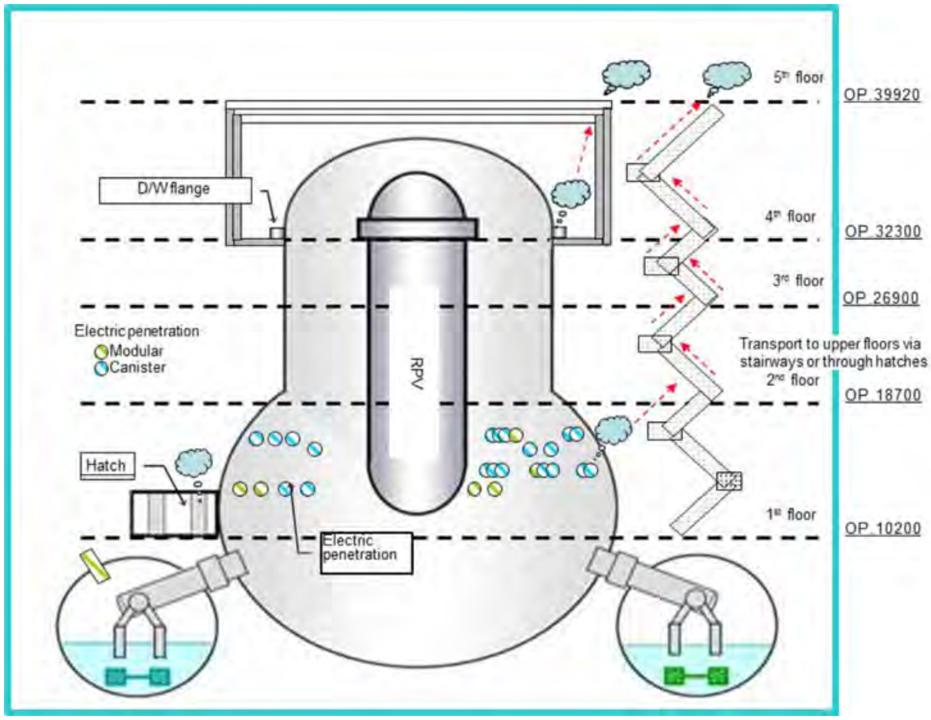

Fig. 10. Inferred hydrogen leakage paths. (Courtesy of TEPCO, [61]).

2.5.1. 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 [59]. It is believed that this hydrogen was primarily due to reactions between steam and fuel Zircaloy cladding. The exact pathway through which the hydrogen flowed is unknown, but available information suggest that it leaked into the building through degraded seals on the head of the PCV, the hatch equipment, and other penetrations (see Fig. 10).

As documented by TEPCO [57], the explosion heavily damaged the 5th floor but did no damage to the floors below except for limited damaged observed near the equipment hatch opening in the southwest corner of the 4th floor [60]. The walls on the 5th floor consisted of a steel framework structure fixed with steel plates and were susceptible to internal pressure increases. The collapse of the walls resulted in a release of inside pressure minimizing any damage to structures below the 5th floor.

The hydrogen explosion at 1F1 significantly hindered 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, it is believed that pressure waves from the 1F1 explosion caused the 1F2 reactor building blowout panel to open. This opening is believed to have averted an explosion in the 1F2 building because it allowed any accumulated hydrogen to vent.

2.5.2. 1F3 explosion

The upper part of the 1F3 reactor building above the refueling floor (the 5th floor) underwent an apparent hydrogen explosion on March 14, 2011 at 11:01 am. [58] Videos show that the explosion and damage were more extensive than at 1F1. In [58], TEPCO identified the following damage:

- Collapsed steel framework and concrete were piled up on, and above, the 5th floor.
- 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.
- The walls on the 4th floor were largely damaged.
- The overhead crane dropped onto the floor of the 5th floor.
- The roof of the turbine building experienced some damage.
- The top of the two-story Radwaste Building 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. [11].

Available information on the damage suggests that extremely high concentrations of combustible gases likely accumulated on both the 4th and 5th floors at the time of the explosion. However, it is not known what caused such combustible gas accumulation on these floors.

2.5.3. 1F4 explosion

The 1F4 explosion in the reactor building is estimated to have occurred on March 15, 2011 at 6:14 am [59]. There were no videos capturing the explosion when it occurred. Unlike 1F1, the structure of 1F4 is a reinforced concrete structure with higher design limits with respect to pressure loadings. Most of the roof slab and walls were blown off leaving only the frame structure consisting of pillars and beams [57]. Most walls on the 4th floor and some walls on the 3rd floor were damaged.

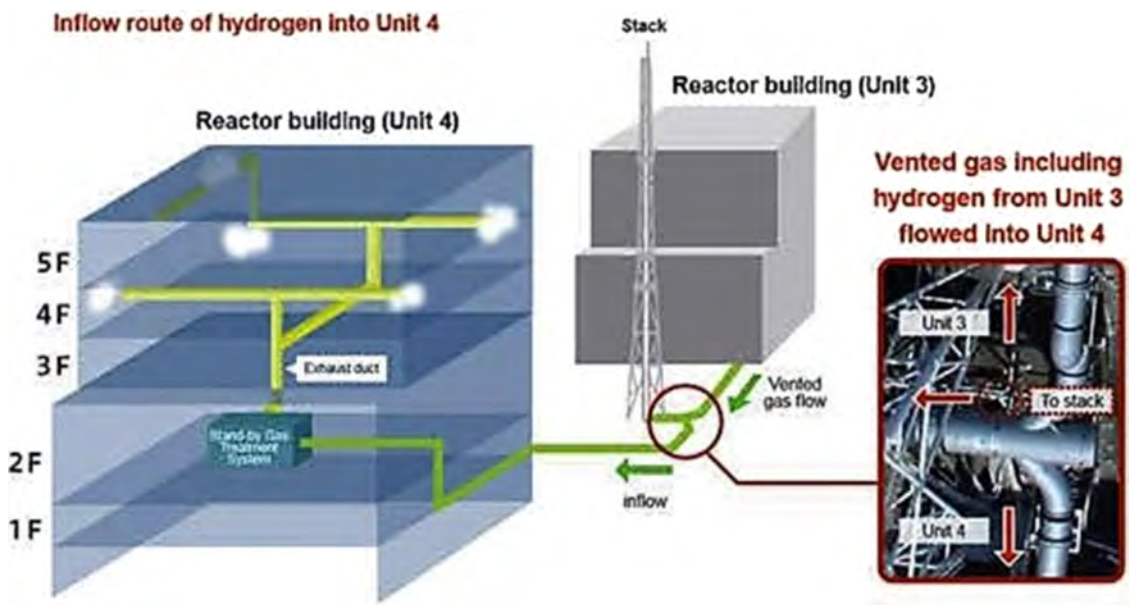

Fig. 11. Hydrogen transport path from 1F3 to 1F4. (Courtesy of TEPCO, [58]).

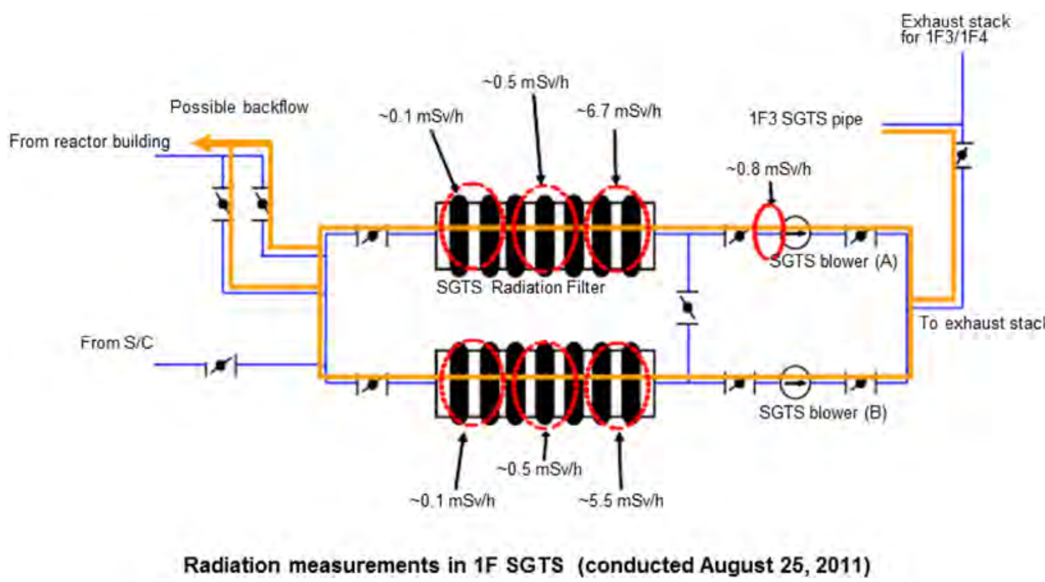

Fig. 12. 1F4 SGTS radiation measurement results. (Courtesy of TEPCO, [61]) .

Evaluations of the explosion at 1F4 have led TEPCO to conclude that vented gases, including hydrogen, flowed from 1F3 into 1F4 (Fig. 11). 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 piping from the outlet, which is reversed from the expected gas flow direction (see Fig. 12).
- ***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 Fig. 13).
- ***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 ventilation 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.[1]

[1] 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 to flow into 1F4 through a SGTS pipe.

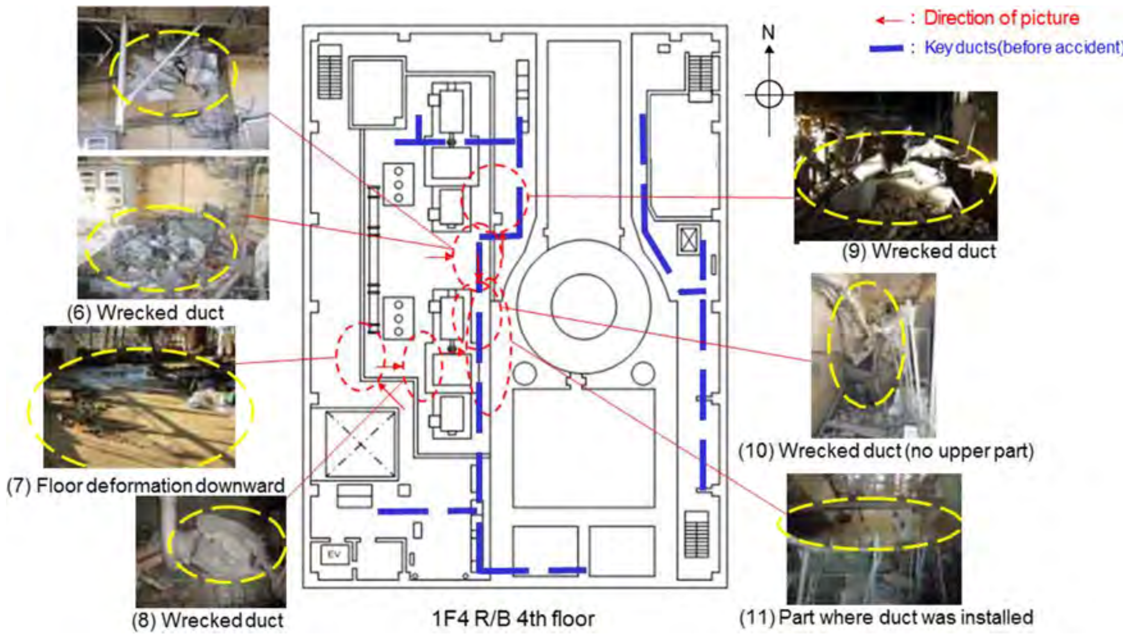

Fig. 13. Field investigation of the 1F4 4th floor (Courtesy of TEPCO, [61]).

Fig. 14. Snapshots comparing 1F1 and 1F3 explosion characteristics (Courtesy of FCT, [66]).

2.5.4. Videos

US and Japan expert evaluations of information related to hydrogen combustion [62,63,64,65], indicate that the hydrogen explosion at the 1F3 reactor building was very different from the explosion of 1F1. As shown in Fig. 14, the explosion at 1F1 was directed horizontally from the top floor of the reactor building. The explosion "smoke" (product gas) was white indicating that the "smoke" was a condensed water vapor which is a by-product of hydrogen combustion. The building roof and side panels were blown away, but concrete pillars remained intact with little damage. The explosions at 1F3 were quite different in appearance and much more energetic. There appeared to be at least two explosions. The first was less energetic and directed horizontally (similar to that of 1F1). The color of the explosion "smoke" appears white and orange. The second was directed vertically with an almost perfect spherical fireball appearing above the building and shooting up very high into the sky (about 3 times the vent stack height). Large chunks of materials appeared to be carried with the fireball. Unlike the explosion at 1F1, available 1F3 images indicate that concrete pillars on the building top floor were highly damaged and the product gas of the explosion appears to be a darker color, raising questions, such as: whether a significant amount of reactor building concrete dust was generated from the explosion; or if dust was generated within the drywell due to Molten Core Concrete Interaction (MCCI).

2.5.5. Plant data

The time of the 1F3 explosion, 11:01 am, March 14, 2011 was near the time when the 1F3 PCV pressure instantaneously dropped from 0.53 to 0.36 MPa (Fig. 15). The instantaneous drop in pressure is believed to correspond to drywell upper head seal failure. This PCV failure would release a hydrogen-steam gaseous mixture into the 5th floor of the reactor building. These hot vent gases could have ignited hydrogen gas that had leaked earlier and accumulated on the 5th floor of the reactor building. Ignition of this hydrogen resulted in the first explosion whose burning mechanism was similar to the 1F1 explosion. Failure of the 1F3 drywell upper head seal then provided a large, continuous supply of hydrogen gas from inside the drywell through the failed head seal. This fuel jet entrained surrounding air as it moved upward and burned as a large fireball emanating from the reactor building into the sky.

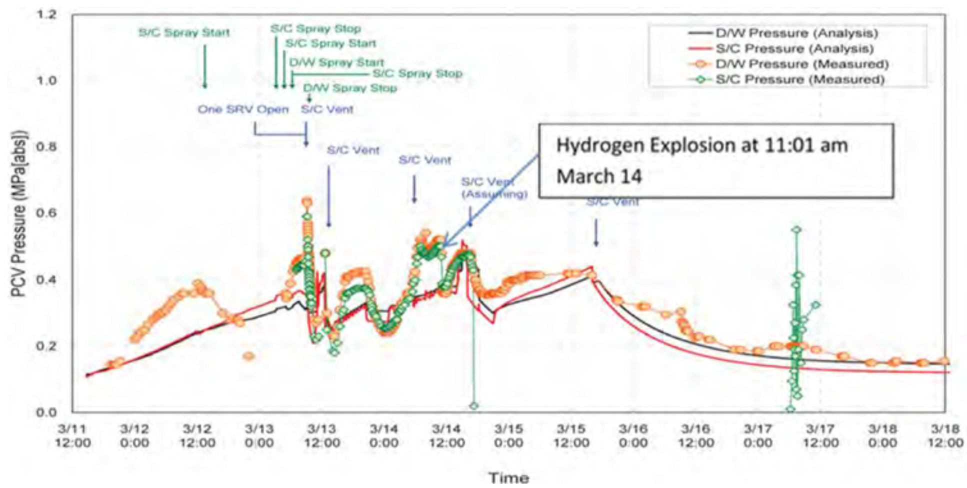

Fig. 15. 1F3 PCV pressure: the rapid drop coincides with 1F3 explosion time (Courtesy of TEPCO, [65]).

2.5.6. Insights and limitations

Available information has led to the following important insights about combustible gas effects.

- ***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 of 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 fireball was a dark color (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.*** Whether or not 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. Further evaluations are needed to determine if this type of explosion can cause containment structural failure at other locations.
- ***The shared vent stack between 1F3 and 1F4 allowed hydrogen 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 implications from this experience.

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

2.5.7. 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 [54] 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 relative to MELCOR [54]. Evaluations with MAAP5 indicate that this modeling uncertainty has important consequences for predicting flammable conditions in the 1F1 and 1F3 reactor buildings. Figs. 16 and 17 compare results from a MAAP5 uncertainty evaluation of the 1F3 accident [67]. 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 h after the earthquake), versus the timing of RPV lower head breach.

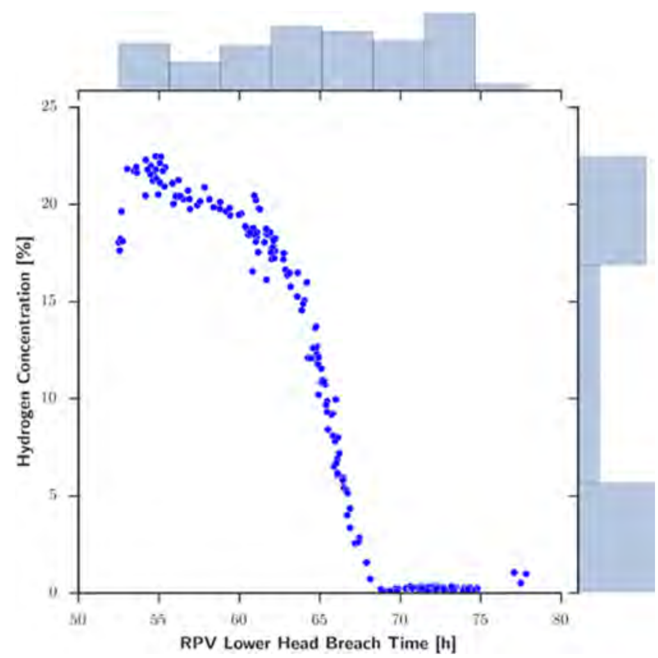

Fig. 16. MAAP 1F3 modeling uncertainty evaluation: refueling floor hydrogen build-up at time of 1F3 RB explosion. (Courtesy of EPRI [67]).

These results illustrate that for simulations predicting RPV lower head breach occurring after ~ 65 h, there is limited potential for flammable conditions to develop on either the 1F3 refueling or 4th floors. That is, MAAP5 simulations of scenarios with

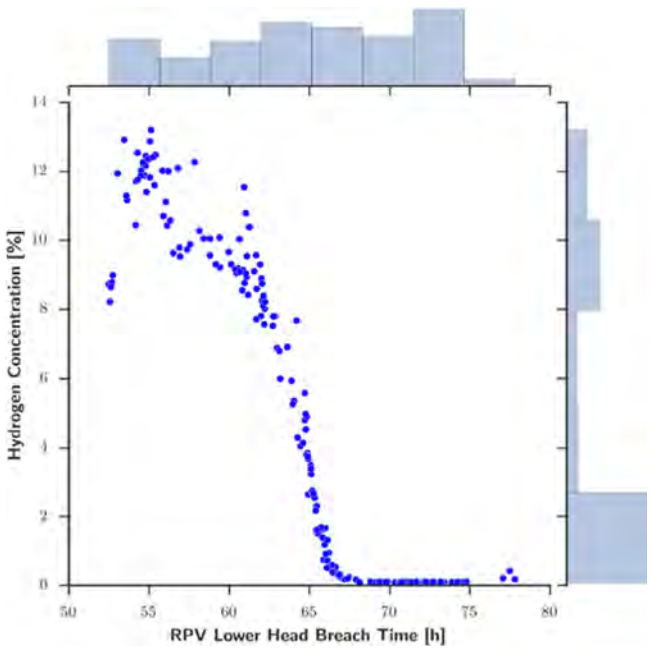

Fig. 17. MAAP 1F3 modeling uncertainty evaluation: fourth floor hydrogen build-up at time of 1F3 RB explosion (Courtesy of EPRI [67]).

in-vessel retention, at least up to the point of the actual 1F3 RB 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 reach the necessary conditions for flammable gas combustion in the RB[68].

Example 3. *Recommendation 1:*

To address this important knowledge gap in severe accident phenomena, evaluations of combustible gas phenomena should be continued to reduce uncertainties in MAAP and MELCOR predictions.

The different levels of damage caused by explosion events at Daiichi 1F1, 1F3, and 1F4 indicate different levels of flammable gas concentrations developed within the RBs. The explosion damage at 1F3 and 1F4 was more severe than 1F1, and high concentrations of combustible gas would be required for these more severe explosions. It is noteworthy that 1F1 was a smaller reactor compared to the other units, but it is not clear at this point if this difference had any impact on the severity of the explosions. Of particular interest is whether the core damage conditions in 1F3 generated sufficient hydrogen for two explosions, one at the 1F3 reactor building and another 14 h later at the 1F4 reactor building, or whether a combination of hydrogen produced from core damage and from MCCI is required to produce the necessary amount of hydrogen. It is recommended that MAAP and MELCOR studies be completed to gain additional insights in this area.

3. Implementation of results

Results from the Forensics Effort are already being used to enhance guidance for PWR and BWR severe accident mitigation and to reduce uncertainties in severe accident code models. Selected implementation related to the examples presented in this paper are discussed in this section.

3.1. Enhanced industry guidance

Industry is enhancing existing SAG to reflect insights gained from the Fukushima accident[69, 70]. Specific examples where industry guidance is benefitting from the US Forensics include:

- ***Primary Containment Venting*** – As discussed in Examples 1 and 3, the three operating units exhibited different patterns of PCV leakage of fission products and hydrogen. The variability introduced by unit to unit differences at Fukushima points to uncertainties in actual leakage locations and confirms that maintaining containment conditions below design basis temperature and pressure limits (and that a high priority is placed on reducing containment conditions when they exceed design basis values) is an appropriate strategy. The revised BWROG and PWROG SAG places a high priority on venting the primary containment when the combination of pressure and temperature reaches a prescribed limit. For BWRs, these conditions can be very close to the containment design basis pressure and temperature.
- ***Water Addition*** – As discussed in Example 2, currently available information from 1F1, 1F2, and 1F3 indicate 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. The revised BWROG and PWROG guidance places a higher priority on injection of water to the reactor vessel compared to the primary containment. If the reactor vessel is failed, the injected water will flow through the reactor vessel breach to the core debris in the primary containment. This ensures that core debris is cooled with injected water (and possibly submerged in water) regardless of its location. Because a large amount of water is required to cool core debris in all possible locations (in the primary containment and in the reactor vessel) for both BWRs and some PWRs, the emphasis on water addition in updated guidance is appropriate. The BWROG also places a high priority on injection of water to the reactor vessel using core spray to assist in more complete cooling of core debris inside the reactor vessel.
- ***Hydrogen Combustion Outside Primary Containment*** – As discussed in Example 3, there were differences in hydrogen accumulation and combustion phenomena for each of the four units. BWROG and PWROG guidance was enhanced immediately after the Fukushima accident to include ventilating 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 thereby confirms that venting buildings adjacent to the primary containment is an appropriate action when primary containment pressure exceeds design basis values. The BWROG and PWROG SAG also includes criteria for ventilating the reactor and auxiliary buildings if normal ventilation is not available. For BWRs, doors at higher elevations within the reactor building are opened on entry to severe accident guidance. Once there is evidence of hydrogen, doors are also opened at lower elevations to promote natural circulation. For PWRs, doors are opened when containment pressure exceeds design basis values.

3.2. Code modeling enhancements

Although results are preliminary, available information has already identified several areas where efforts are needed to reduce uncertainties in severe accident modeling. Specific examples are as follows:

- ***Primary Containment Integrity Challenges*** – As discussed in Examples 1 and 3, the three operating units exhibited differ-

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ent patterns of PCV leakage of fission products and hydrogen. Many of these leakage points are not routinely modeled by systems level severe accident codes (MAAP, MELCOR, 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 analyses codes.

– ***MAAP and MELCOR Nodalization Studies*** – As discussed in Example 2, 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.
– ***1F2 MELTSPREAD-CORQUENCH Analysis*** – As discussed in Example 2, 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 provide useful in rationalizing differences in future observations obtained from 1F1 and 1F2.
– ***Combustible Gas Production, Transport, and Mitigation*** – As discussed in Example 3, 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, 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.

4. Summary

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

Acknowledgements

The US Forensics Effort could not be performed without input and support from multiple individuals and organizations. The US DOE, Office of Nuclear Energy, LWR Sustainability program funded the participation of present and former national laboratory participants as well as the RST pathway leader. In addition, various industry organizations provided substantial in-kind contributions by providing technical experts to participate in this process; these organizations included the Electric Power Research Institute, the BWROG, Exelon Corporation, GE-Hitachi, the Institute of Nuclear Power Operations, Jensen Hughes, the Nuclear Energy Institute, the PWROG, Southern Nuclear, and the Tennessee Valley Authority. Finally, two organizations, the US Nuclear Regulatory Commission and the Tokyo Electric Power Company Holdings, provided technical experts to participate in the panel meetings. In particular, Mr. Takahasi Hara, Dr. Shinya Mizokami, Mr. Kenji Tateiwa, Mr. Daichi Yamada, and Mr. Yasunori Yamanaka from Tokyo Electric Power Company Holdings attended, as well as Dr. Sudhamay Basu and Dr. Richard Lee from the US NRC Office of Nuclear Regulatory Research. These individuals facilitated the overall process by providing key clarifications in various areas as the meetings progressed. These efforts are greatly appreciated.

References

[1] J. Rempe, M. Farmer, M. Corradini, L. Ott, R. Gauntt, D. Powers, Revisiting Insights from three mile island unit 2 post-accident examinations and evaluations in view of the Fukushima Daiichi accident, Nuclear Sci. Eng. 172 (November) (2012) 223–248.
[2] Nuclear Damage Compensation and Decommissioning Facilitation Corporation, “Technical strategic plan 2015 for decommissioning of the Fukushima Daiichi nuclear power station of Tokyo electric power company, towards amendment of the mid-and-long-term roadmap in 2015,” April 2015.
[3] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np, last accessed February 2016.
[4] K. Tateiwa, Fukushima update, Presentation at The First DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory, January 7, 2015.
[5] K. Tateiwa, Dose Map of 1F-3 Reactor Building Refueling Floor email, dated January 8, 2015, to Forensics Expert Panel, 2015 (in Japanese), presentation, dated December 25, 2014, last accessed February 10 http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d141225_11-j.pdf .
[6] Y. Yamanaka, The facts obtained from the investigation activities at Fukushima Daiichi NPS, *Presentation at The First DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory*, January 7, 2015.
[7] Y. Yamanaka, Investigation plan of inside RPV, *Presentation at The First DOE Meeting on Reactor Safety Gap Evaluation and Fukushima Forensics at Argonne National Laboratory*, January 7, 2015.
[8] Y Yamanaka, Outline of the 3rd progress report on the investigation of unresolved issues, *Presentation at the Second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.
[9] Y. Yamanaka, 1F-1 PCV inside inspection results, *Presentation at the Second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.
[10] D. Yamada, Y. Yamanaka, S. Mizokami, Progress for applying reactor imaging technology by cosmic ray Muon in Fukushima Daiichi, *Presentation at the Second DOE Meeting on Fukushima Forensics*, Washington, DC, May 27, 2015.
[11] D. Yamada, T. Futatsugi, S. Mizokami, (1) Dose rate distribution and gamma spectrum analysis on refueling floor at Unit 3, *Presented at DOE Fukushima Forensic Meeting*, Washington DC, April 28-29, 2016.
[12] Tokyo Electric Power Company, Evaluation of the Situation of Cores and Containment Vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and Examination into Unsolved Issues in the Accident Progression, 2013 Progress Report No. 1, December 13.
[13] Tokyo Electric Power Company, Report on the Investigation and Study of Unconfirmed/Unclear Matters in the Fukushima Nuclear Accident, 2014 Progress Report No. 2, August 6.
[14] Tokyo Electric Power Company, Report on the Investigation and Study of Unconfirmed/Unclear Matters in the Fukushima Nuclear Accident, May 2016 Progress Report No. 3.
[15] Tokyo Electric Power Company, The 4th Progress Report on the Investigation and Examination of Unconfirmed and Unresolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident, 2015 December 17.
[16] P. Amway, R. Bunt, M. Corradini, P. Ellison, M. Farmer, M. Francis, J. Gabor, R. Gauntt, C. Henry, D. Kalinich, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, C. Paik, M. Plys, C. Rabiti, J. Rempe, K. Robb, R. Wachowiak, B. Williamson, US Efforts in Support of Examinations at Fukushima Daiichi, in: J.L. Rempe (Ed.), August 2015, ANL/LWRS-15/02.
[17] P. Amway, N. Andrews, W. Bixby, R. Bunt, M. Corradini, P. Ellison, M. Farmer, T. Farthing, M. Francis, J. Gabor, R. Gauntt, C. Henry, P. Humrickhouse, S. Kraft, R. Linthicum, W. Luangdilok, R. Lutz, D. Luxat, J. Maddox, C. Negin, C. Paik, M. Plys, J. Rempe, K. Robb, R. Sanders, R. Wachowiak, B. Williamson, US Efforts in Support of Examinations at Fukushima Daiichi – 2016 Evaluations, in: J.L. Rempe (Ed.), August 2016, ANL/LWRS-16/02.
[18] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120927_02-e.pdf, last accessed Feb. 2016.
[19] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_121003_01-e.pdf, last accessed Feb. 2016.
[20] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/m120730_05-j.pdf, (in Japanese) last accessed March 2016.
[21] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/m111226_08-j.pdf, (in Japanese), last accessed March 2016.
[22] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_151020_01-e.pdf, last accessed February 2016.
[23] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151001_08-j.pdf, last accessed Feb. 2016 .
[24] Tokyo Electric Power Company,http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d150827_08-j.pdf, last accessed Feb. 2016.

[25] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_151022_01-e.pdf, last accessed Feb. 2016.
[26] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120419_03-e.pdf, last accessed February 2016.
[27] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_151127_08-j.pdf, last accessed February 2016.
[28] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130409_10-e.pdf, last accessed Feb. 2016 .
[29] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151224_08-j.pdf, (in Japanese), last accessed March 2016.
[30] Tokyo Electric Power Company, http://www.tepco.co.jp/nu/fukushima-np/roadmap/images/d151029_08-j.pdf , (in Japanese), last accessed March 2016.
[31] Tokyo Electric Power Company, http://photo.tepco.co.jp/en/date/2013/201307-e/130708-02e.html, last accessed March 2016.
[32] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2014/images/handouts_140527_07-e.pdf, last accessed Feb. 2016.
[33] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130315_01-e.pdf, last accessed Feb. 2016.
[34] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_131114_05-e.pdf, last accessed Feb. 2016.
[35] T. Kotaki, The behavior of FP estimated by the investigation of Fukushima Daiichi NPS units 1∼3, *JAEA-Sponsored CLADS Decommissioning Workshop and Seminar (5) FP behavior*, Nov 12, 2015.
[36] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2014/images/handouts_140121_04-e.pdf, last accessed Feb. 2016.
[37] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130221_02-e.pdf, last accessed Feb. 2016.
[38] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120627_02-e.pdf, last accessed Feb. 2016.
[39] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120607_02-e.pdf, last accessed Feb. 2016.
[40] Tokyo Electric Power Company,http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120418_04-e.pdf, last accessed Feb. 2016.
[41] Tokyo Electric Power Company,http://www.tepco.co.jp/en/nu/fukushima-np/images/handouts_120712_03-e.pdf, last accessed Feb. 2016.
[42] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2014/images/handouts_140728_05-e.pdf, last accessed Feb. 2016.
[43] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2013/images/handouts_130416_06-e.pdf, last accessed Feb. 2016.
[44] Tokyo Electric Power Company, http://photo.tepco.co.jp/en/date/2014/201405-e/140515-01e.html, last accessed Feb. 2016.
[45] Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/roadmap/images1/d160128_07-j.pdf, last accessed Feb. 2016.
[46] Tokyo Electric Power Company, "Air dose rates in the buildings," http://www.tepco.co.jp/en/nu/fukushima-np/f1/surveymap/index-e.html, last accessed March 2016.
[47] Fauske and Associates, LLC, Transmittal Document for MAAP5 Code Revision MAAP 5.02, FAI/13-0801, 2013.
[48] Sandia National Laboratories, MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, Version 2.1, September 2011 NUREG/CR-6119, Rev. 3179 (Draft).
[49] NEI, Industry Guidance for Compliance with Order EA-13-109, Nuclear Energy Institute, April 2015 NEI 13-02, Revision 1.
[50] S.K. Shaukat, J.E. Jackson, D.F. Hatcher, Regulatory Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure, April 1991 NUREG-1401.
[51] D. Luxat, J. Gabor, Fukushima Technical Evaluation: Phase 1 – MAAP5 Analysis, 2013 EPRI Report No. 1025750.
[52] R.O. Gauntt, et al., Fukushima Daiichi Accident Study, 2012 SAND2012-6173.
[53] H. Bonneville, A. Luciani, Simulation of the core degradation phase of the Fukushima accidents using the ASTEC code, Nuclear Eng. Design 272 (2012).
[54] D. Luxat, J. Hanophy, D. Kalanich, Electric Power Research Institute, "Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk, Phase 1 Study, 2014 EPRI Report No. 3002004449.
[55] Development of a Technology to Investigate Inside the Reactor Primary Containment Vessel (PCV), Tokyo Electric Power Company presentation, 2015 http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_150430_04-e.pdf April 30 last accessed March 2016.
[56] K.R. Robb, M.W. Francis, M.T. Farmer, Enhanced Ex-Vessel Analysis for Fukushima Daiichi Unit 1: Melt Spreading and Core-Concrete Interaction Analyses with MELTSPREAD and CORQUENCH, 2013 ORNL/TM-2012/455.
[57] Tokyo Electric Power Company, the Report on the Investigation into the Current Seismic Safety and Reinforcement of the Reactors at Fukushima Daiichi Nuclear Station (No. 1), 2011 May 28.
[58] Tokyo Electric Power Company, the Report on the Investigation into the Current Seismic Safety and Reinforcement of the Reactors at Fukushima Daiichi Nuclear Station (No. 2)," July 13, 2011 http://www.tepco.co.jp/en/press/corp-com/release/betu11_e/images/110713e20.pdf, last accessed February 2016.
[59] Tokyo Electric Power Company, The Development of and Lessons from the Fukushima Daiichi Nuclear Accident, http://www.tepco.co.jp/en/decommision/accident/images/outline01.pdf, last accessed February 2016.
[60] Tokyo Electric Power Company, Photos of Unit 1 Reactor Building taken on 2/21/2015, Photos #101-117, http://photo.tepco.co.jp/date/2015/201502-j/150221-04j.html, last accessed March 2016.
[61] Tokyo Electric Power Company, Fukushima Nuclear Accident Analysis Report, 2011 (Interim Report) Summary, December 2.
[62] W. Luangdilok, E. van Heerden, P. McMinn, Calculation of the Probability of DDT During Severe Accidents, NURETH-16, Chicago, August 30 – September 4, 2015.
[63] J.E. Shepherd, The crisis at Fukushima Daiichi nuclear plant, Apr 9, 2011 Presentation to Caltech Community, 2012 updated Apr 22.
[64] M. Ogino, Study on the issues about hydrogen explosion at Fukushima Daiichi NPS, Technical Workshop on the Accident of TEPCO's Fukushima Daiichi NPS, July 2012.
[65] H. Yanagisawa, H. Takeuchi, M. Akinaga, S. Mizokami, T. Honda, M. Watanabe, The accident analysis for unit 3 at Fukushima Dai-ichi nuclear power station, 9th Int. Topical Meeting on Nuclear Thermal Hydraulics, Operation and Safety (NUTHOS-9), Kaohsiung, Taiwan, September 9-13, 2012.
[66] J. Maru, General Manager, News Department, Fukushima Central Television Company, LTD, "Permission for using snap shots", email with photos to J. Rempe, Rempe and Associates, LLC, June 3, 2016.
[67] EPRI, Technical Evaluation of Fukushima Accidents: Phase 2, December 2015 Final Report Number 3002005301.
[68] Nuclear Energy Agency, "NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project," http://www.oecd-nea.org/jointproj/bsaf.html, last accessed June 20, 2015.
[69] NEI, "The way forward", Nuclear Energy Institute, The Institute for Nuclear Power Operations and Electric Power Research Institute, http://www.nei.org/Master-Document-Folder/Backgrounders/White-Papers/The-Way-Forward, last accessed March 13, 2016.
[70] R. Lutz, B. Williamson, Insights from DOE expert panel forensics for LWR accident management, *Presentation at Reactor Safety Technology Expert Panel Forensics Meeting*, Washington, DC, April 28-29, 2016.