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Insights Gained from Forensic Analysis with MELCOR of the Fukushima-Daiichi Accidents

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Abstract. Since the accidents at Fukushima-Daiichi, Sandia National Laboratories has been modeling these accident scenarios using the severe accident analysis code, MELCOR. MELCOR is a widely used computer code developed at Sandia National Laboratories since ~1982 for the U.S. Nuclear Regulatory Commission. Insights from the modeling of these accidents is being used to better inform future code development and potentially improved accident management. To date, our necessity to better capture in-vessel thermal-hydraulic and ex-vessel melt coolability and concrete interactions has led to the implementation of new models.

The most recent analyses, presented in this paper, have been in support of the of the Organization for Economic Cooperation and Development Nuclear Energy Agency's (OECD/NEA) Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project. The goal of this project is to accurately capture the source term from all three releases and then model the atmospheric dispersion. In order to do this, a forensic approach is being used in which available plant data and release timings is being used to inform the modeled MELCOR accident scenario. For example, containment failures, core slumping events and lower head failure timings are all enforced parameters in these analyses. This approach is fundamentally different from a blind code assessment analysis often used in standard problem exercises. The timings of these events are informed by representative spikes or decreases in plant data.

The combination of improvements to the MELCOR source code resulting from analysis previous accident analysis and this forensic approach has allowed Sandia to generate representative and plausible source terms for all three accidents at Fukushima Daiichi out to three weeks after the accident to capture both early and late releases. In particular, using the source terms developed by MELCOR, the MACCS software code, which models atmospheric dispersion and deposition, we are able to reasonably capture the deposition of radionuclides to the northwest of the reactor site.

Keywords MELCOR, FUKUSHIMA-DAIICHI, INSIGHTS, SEVERE ACCIDENT

1. INTRODUCTION AND BACKGROUND

On March 11, 2011, the Tohuku earthquake struck near the Fukushima Daiichi power station, causing a regional loss of electric power and the operating reactors (Units 1, 2 and 3) to scram. The emergency on-site diesel-powered generators started as designed, and supplied power to the emergency cooling systems needed to keep the reactors cool. Several tsunami waves produced by the earthquake reached the Fukushima Daiichi site roughly an hour later, resulting in the loss of emergency diesel-powered AC generators and causing a loss of AC electrical power. Battery power was also lost at Units 1 and 2 immediately and eventually at Unit 3. Consequently without adequate core cooling, each of the three units subsequently suffered core damage of varying degrees.

In 2012, the first phase of the OECD/NEA BSAF Project was begun. The goal of the international project was to model the primary containment system of all three reactors out to seven days after the initiation of the event. However, it became clear that in order to capture the full extent of the source terms for all three units that this calculation would have to be extended out to three weeks. This three week time frame fully captures releases until the situation at all reactor was stabilized. The second phase of the BSAF project, which was initiated in 2015, was dedicated to modeling and analyzing this three-week time frame. [1]

For this analysis, the severe accident analysis code MELCOR has been used to model the accident progression and subsequent releases for all three accident scenarios. MELCOR is developed by Sandia National Laboratories (SNL) under contract to the U.S. Nuclear Regulatory Commission (USNRC). It has dedicated system models to track radionuclides as they are released from degraded fuel and make their way outside of the reactor pressure vessel, into the containment and eventually into the environment. Key systems and phenomenology captured by MELCOR are shown in Figure 1. Released radionuclides are then taken as inputs to the MELCOR Consequence Code System (MACCS), which models atmospheric dispersions and depositions using weather data available for the time of the accidents. [2]

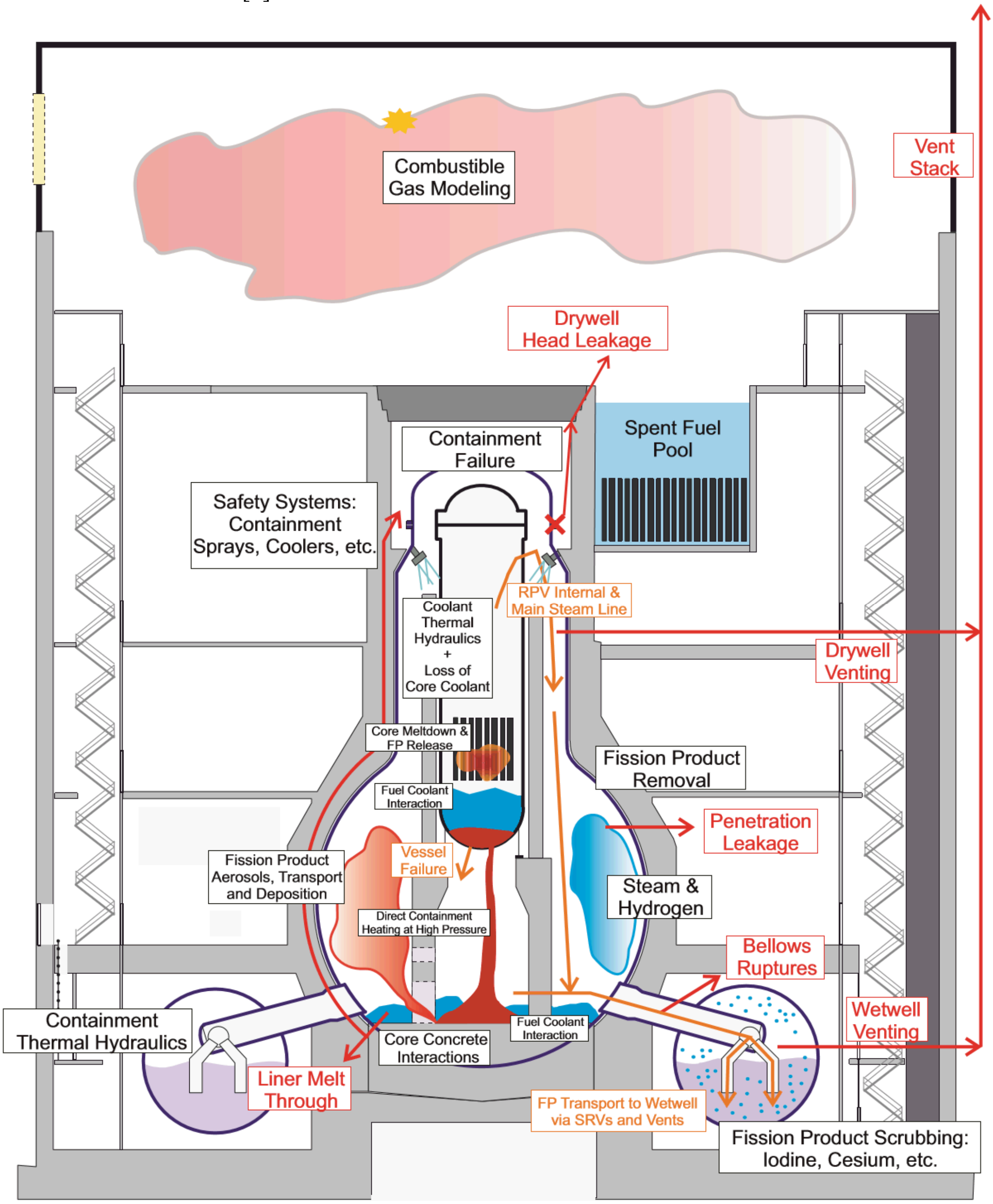

FIG. 1. Reactors systems and phenomenology represented within the severe accident analysis code, indicated in black. Release pathways outside of the reactor pressure vessel are indicated in yellow and those outside of the primary containment are indicated in red.

This paper discusses insights gained from the modeling of each of the three accident scenarios. First, accident progression events and timings are presented. Then each accident scenario is discussed in more detail, highlighting key insights that we have gained from modeling the accidents over the past six years.

It is important to acknowledge that each of the scenarios presented here are single best-guess plausible accident realizations and that alternate scenario variants might also be plausible explanations for the observed trends and behaviors. A full understanding of the events may not be available until the decommissioning of the reactors is complete.

2. FUKUSHIMA DAIICHI UNIT 1 ACCIDENT PROGRESSION

The accident at Fukushima Daiichi Unit 1 (1F1) progressed significantly faster than at the other two units. At the time of the tsunami, the passive cooling system used for management of decay heat, the isolation condenser (IC) was switched off and was not able to be restarted due to loss of DC power following the arrival of the tsunami. This led to the rapid loss of water level in the reactor pressure vessel, with top of active fuel (TAF) estimated as being reached 2.6 hours into the accident and fuel damage occurring at 3.5 hours. By 10.9 hours, the lower head of the reactor pressure vessel is predicted to have failed and at 12.8 hours the first containment breach is modeled. Key timings for the modeled accident scenario are displayed in Table 1. In MELCOR, station blackout scenarios of boiling water reactors, a failure of the main steam line is often predicted. This was highlighted by the State-of-the-Art Reactor Consequence Analysis (SOARCA) Uncertainty Analysis of the Peach Bottom reactor. Just as in the SOARCA analysis, this 1F1 reactor was predicted to have a main steam line break near 6 hours into the accident. [3] This break was enforced to occur at 6.1 hours to best match the plant data provided the Tokyo Electric Power Company (TEPCO).

TABLE 1. KEY EVENT TIMINGS FOR THE ACCIDENT AT FUKUSHIMA DAIICHI UNIT 1

Event	**Time [hours]**
First occurrence of water level at TAF	2.6
Onset of hydrogen generation	3.4
First fuel clad failure time	3.5
First control blade melting/liquefaction time	3.9
First fuel rod failure time (melting or collapse)	3.8
First UO2 relocation to lower head	4.1
First lower core plate failure time	8.5
First FP release from fuel	3.5
First RPV pressure boundary failure (main steam line break)	6.1
Vessel water dryout in lower head	9.1
Lower head failure	10.9
Initiation of MCCI	10.9
Containment failure	12.8
Fresh or sea water injection to RPV and termination of injection	15.0
Hydrogen explosion	24.8
Drywell line breach	48.3

When the predicted pressure signature of the primary containment is examined, shown in Figure 2, it can be seen that there is a large jump in the containment pressure immediately after the failure of the main steam line at 6.1 hours. Immediately after this, additional spikes in containment pressure are seen from assumed core slumping events into the lower plenum. As the pressure of the containment increases, the head flange of the drywell begins to leak. The bolts holding the drywell head in place stretch elastically, leaking combustible gas and radionuclides into the refueling bay. At 48.3 hours into the accident progression, the drywell liner was predicted to be breached by persistent core-concrete erosion.

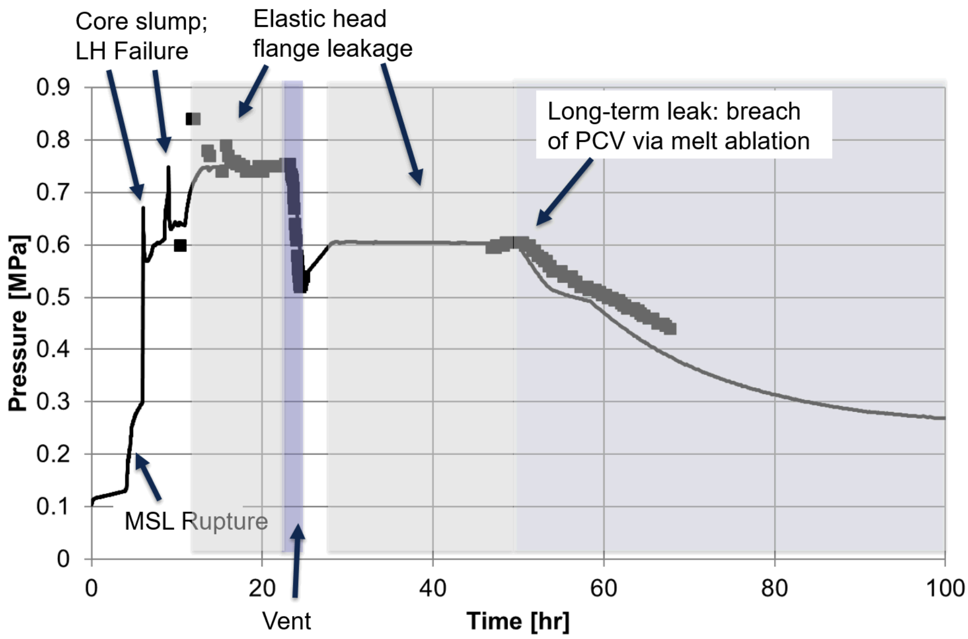

FIG. 2. Reactor pressure vessel pressure for the first 100 hours of the 1F1 accident scenario

2.1 Drywell Liner Breach through Sump Ablation at Fukushima Daiichi Unit 1

In our analysis of this scenario with MELCOR, we predicted a breach of the drywell liner through the drywell sump at 48.3 hours. This can be seen in Figure 3. This is very close to the reported drop in the pressure of the containment that occurs near 50 hours. This has led us to conclude that it is likely that the drywell lined was breached due to ablation of concrete through MCCI in the sump below the reactor pressure vessel. This serves to highlight the importance of having a robust MCCI and corium spreading model. Additionally, it is likely that MCCI and resultant released radionuclides were the leading contributor to late releases that occurred after one week after the earthquake, of radionuclides into the atmosphere.

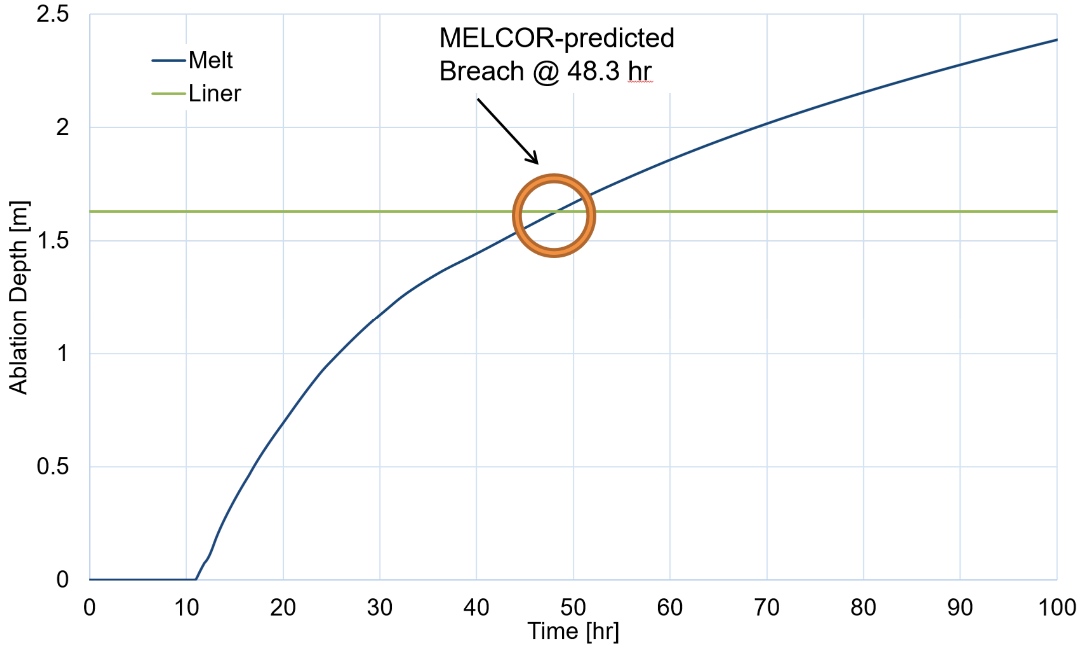

FIG. 3. Ablation depth of the MCCI in 1F1 for the first 100 hours of the accident scenario

2.2 Drywell Head Flange Leakage and Hydrogen Explosion

In our MELCOR analysis, between roughly 12 and 23 hours, steam and hydrogen leak from the drywell head flange and enter the refueling bay through seams in the shield plug. Then the hydrogen, carbon monoxide and steam rise to the rood and spread laterally. Steam present is produced from MCCI and emergency water injection. As the steam remains in the refueling bay, it is gradually condensed out of the hot layer, enriching the local hydrogen concentration. This mixture displaces air from the refueling bay, with the steam mole fraction exceeding 50%, resulting in an inert environment. However, as the scenario progresses, at about 24 hours operators vent the drywell to reduce the pressure. This results in less steam being vented into the refueling bay which up to now has been displacing air from the reactor building and inerting the refueling bay. As the remaining steam content of the refueling bay condenses creating a partial vacuum, air is subsequently drawn into the reactor building from the outside, making the composition of the refueling bay flammable. A representation of this can be seen in Figure 4.

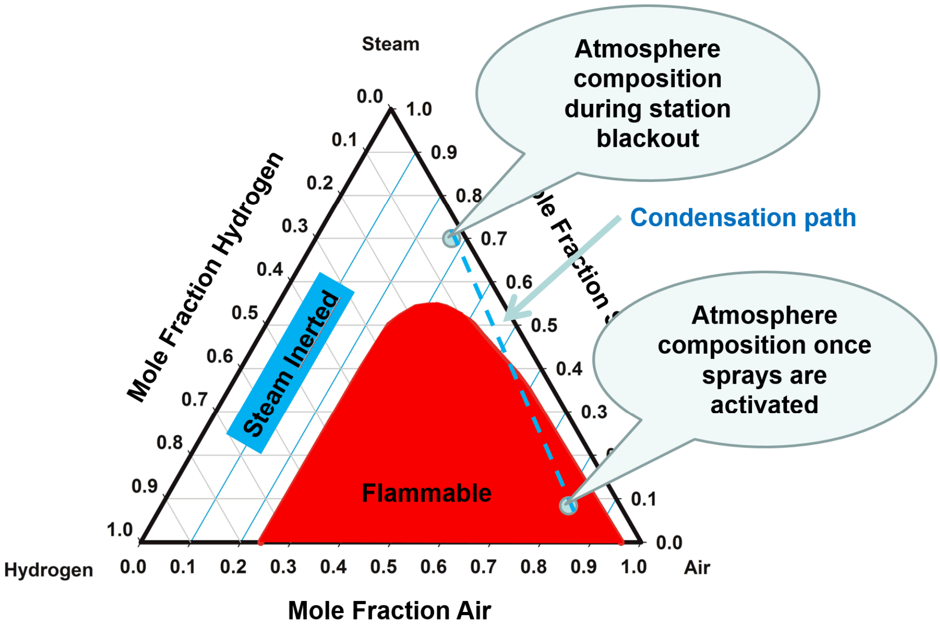

FIG. 4. Shapiro diagram of the atmosphere in the refueling bay of 1F1

3. FUKUSHIMA DAIICHI UNIT 2 ACCIDENT PROGRESSION

Of the accidents at the Fukushima Daiichi site, the 1F2 reactor likely saw the least damage due to the fact that the onset of core damage was delayed significantly by the operation of the reactor core isolation cooling (RCIC) system. This passive system uses steam generated in the reactor pressure vessel to drive a turbine/pump system that takes water from either the condensate storage tank (CST) or the wetwell (WW) and injects it into the reactor pressure vessel. After the termination of this system at near 70 hours the RPV pressure returns to the safety relief valve setpoint while RPV water levels drop and core damage initiates. At 73.8 hours, the operators initiate a manual depressurization of the RPV in order to attempt emergency water injection, but significant core damage has already occurred. Core relocation events are believed to have subsequently occurred leading eventually to lower head breach. These suspected relocation events are described in the following section The containment is failed shortly after this at 92.0 hours, just before the northwesterly wind begins that resulted in the large deposition pattern in this direction. A full list of accident event timings can be seen in Table 2. It can be seen that the first fuel damage occurs here nearly 70 hours after it did in 1F1.

When examining the pressure data reported by TEPCO several interesting phases can be identified. The first of the phases is the period of RCIC operation, which lasts from the time of the earthquake to the 70 hours, when the system failed and the RPV pressure rises to the SRV setpoints. The second period lasts until the reactor pressure vessel blows down due to operator initiated SRV opening. During this period safety relieve valves in the primary system are cycling at their maximum pressure. After the blowdown of the system, a period of core degradation begins, which is often referred to as the “three peaks” period. After this the containment fails and there is a period of long-term leakage and water injection. These four periods can be seen in Figure 5.

TABLE 2: KEY EVENT TIMINGS FOR THE ACCIDENT AT FUKUSHIMA DAIICHI UNIT 2

Event	Time [hours]
Activation and termination of RCIC	0.04/70.0
Manual opening of one SRV for RPV depressurization	73.8
First occurrence of water level at TAF	75.3
Onset of hydrogen generation	76.1
First fuel clad failure time	76.4

First control blade melting/liquefaction time	76.9
First fuel rod failure time (melting or collapse)	77.2
First UO2 relocation to lower head	80.0
First lower core plate failure time	80.0
First FP release from fuel	76.4
First FP release to environment	76.9
Lower head failure	91.4
Initiation of MCCI	92.4
Time of containment failure	92.0

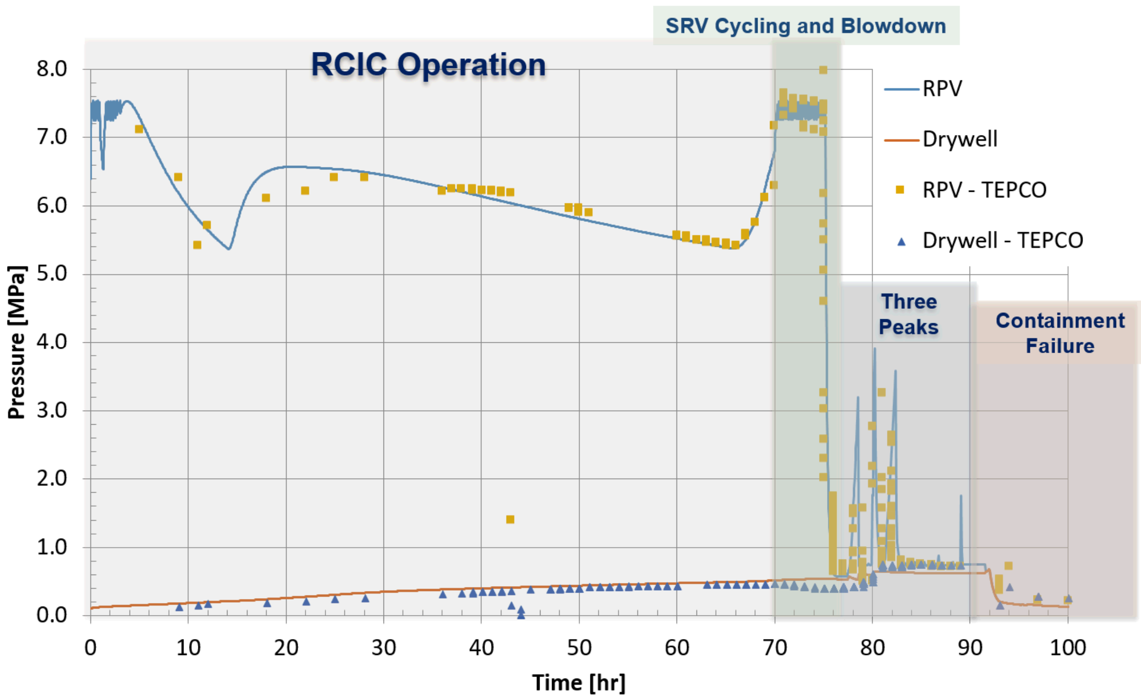

FIG. 5. Reactor pressure vessel and drywell pressure for the first 100 hours of the 1F2 accident scenario

3.1 Simple Mechanistic Model of the RCIC Passive Cooling System

In order to accurately capture the period of RCIC operation, a computationally expedient mechanistic model was developed to accurately capture the feedback from water entering the steam turbine of the RCIC system due to the full open RCIC operation and the overfilling of the RPV to the steam line elevation. In this model, the velocities of the turbine and the pump were specified, assuming the turbine flow is always choked. Prior to PRV vessel overfilling when it is in the since phase region, it is taken to be the sonic velocity; and after the PRV water level rises to the steam line and enters the RCIC inlet when it is two phase, it is taken to be homogenous frozen choked flow. Turbine work is the driver for the pump velocity. Turbine efficiency is degraded as the amount of water in the turbine increases. An algorithmic representation of this model can be seen in Figure 6. This has been shown to reach a steady state operation of the RCIC turbine during two-phase water ingestion where the water returned to the RPV is just enough to maintain the PRV water level at the steam line elevation and a self-regulating operation of the RCIC is attained.

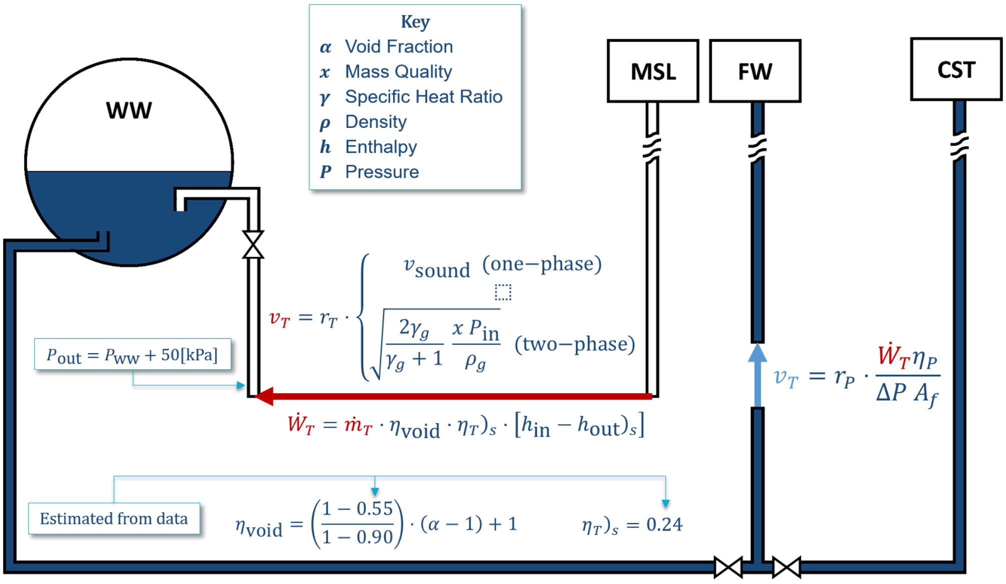

FIG. 6. Algorithmic representation of the RCIC system, with water flows shown in blue and steam flow shown in red

3.2 Torus Room flooding

An accurate representation of the RCIC system itself is not enough to fully capture the behavior of the primary system; it is also necessary to accurately model the amount of flooding in the torus room that is strongly suspected to have occurred from tsunami flooding of the turbine building and subsequent in-flow to the torus room. This boundary conditions influences the temperature of water injected into the RPV from the RCIC system. Flooding of the torus room was initiated in our simulation at the time of the tsunami and continued for one hour to a specified percentage of the total volume of the torus room. Varying percentages of flooding can be seen in Figure 7, with 30% flooding showing the best agreement. The break in agreement between 16 and 30 hours is likely an artifact of how the wetwell is nodalized and modeled in the system. This calculation assumes that wetwell is a single monolithic volume.

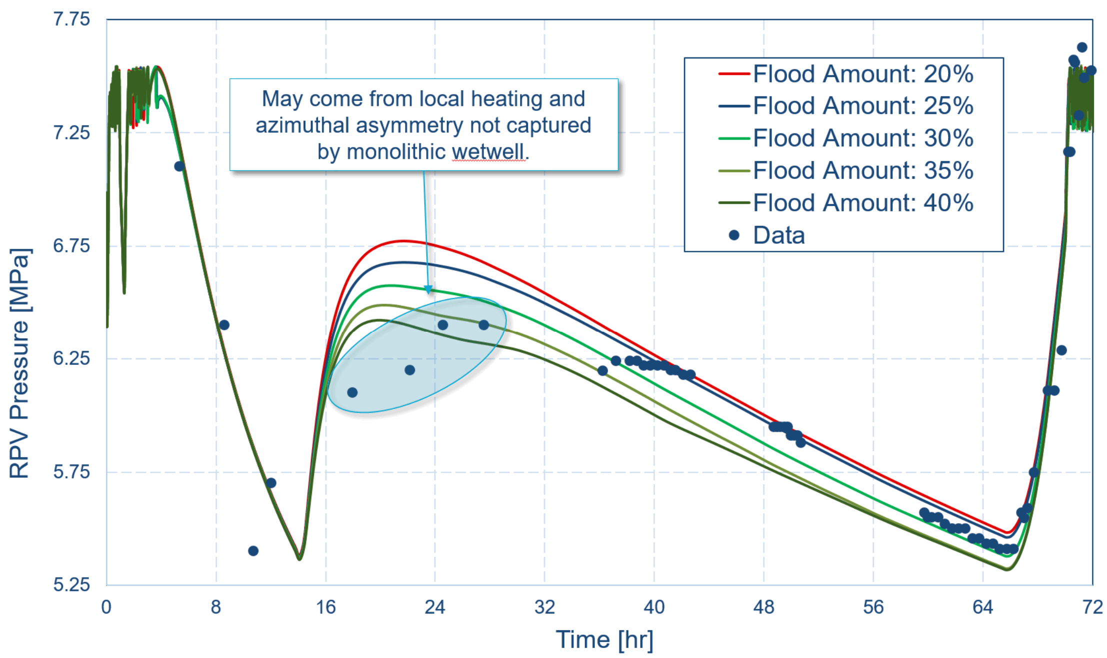

FIG. 7. Influence of torus room flooding percentage on the pressure of the reactor pressure vessel of 1F2

3.3 "Three Peaks" and Insights into Core Degradation

The three peaks period of the accident scenario, where the majority of core degradation occurred, is of key interest to informing the phenomenological modeling of in-vessel core degradation. These peaks are produced from a combination of SRV openings and closures, injection, core degradation and steam generation and reflect the "forensics" approach used in modeling the Fukushima accidents.

It can be seen that for the second two peaks, the increase in pressure is resultant from steam generation due to the quenching of degraded core debris. However, the first peak in our code analyses is primarily dominated by hydrogen generation. This can be seen in Figure 8. In the first peak 64% of the total pressure spike is from hydrogen generation. This hydrogen can only be formed through the oxidation of metals within the core region.

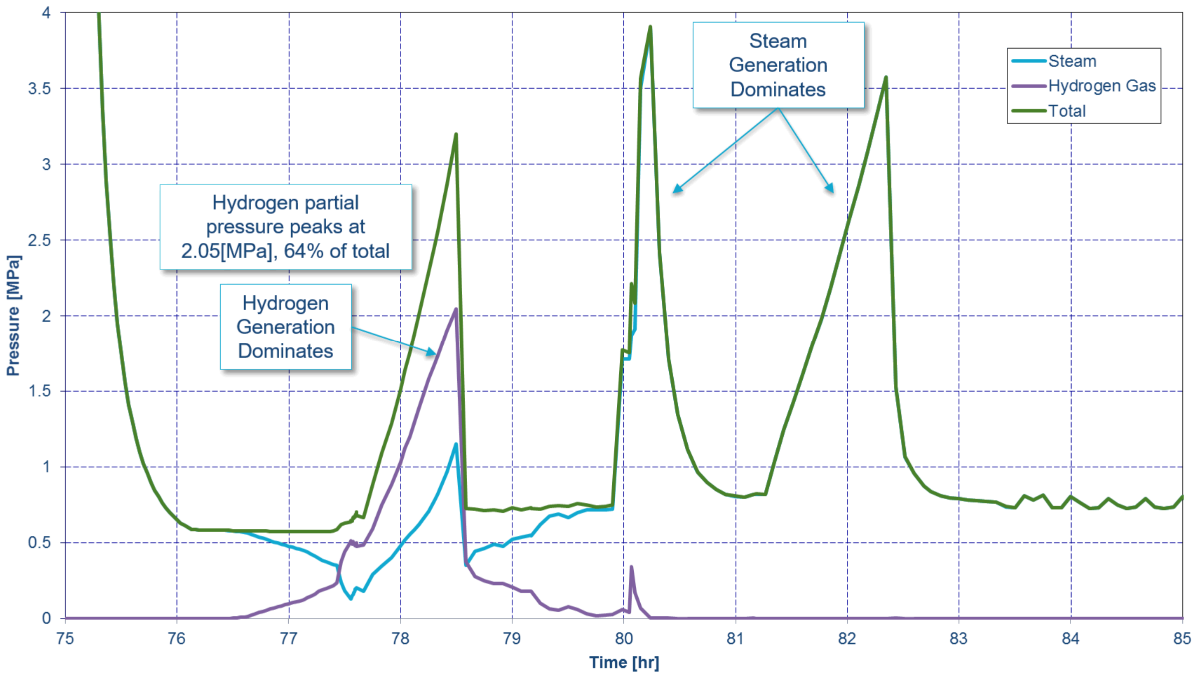

FIG. 8. Partial pressures, with contributions from hydrogen and steam, of the reactor pressure vessel of 1F2

While the agreement between the second and third peak and plant data is quite reasonable, there is divergence in the first peak. Plant data is compared to the MELCOR realization for this period of time in Figure 9. The hydrogen generation during this period leads to an over prediction of the total pressure in the primary system. It is possible that this over prediction is due to the representation and modeling of core degradation in this scenario. In particular, it is possible that a molten pool may have formed in the core region during this time. This is something which is not currently captured in the MELCOR accident scenario. This is an area of current examination and a candidate for future melt progression modeling improvements.

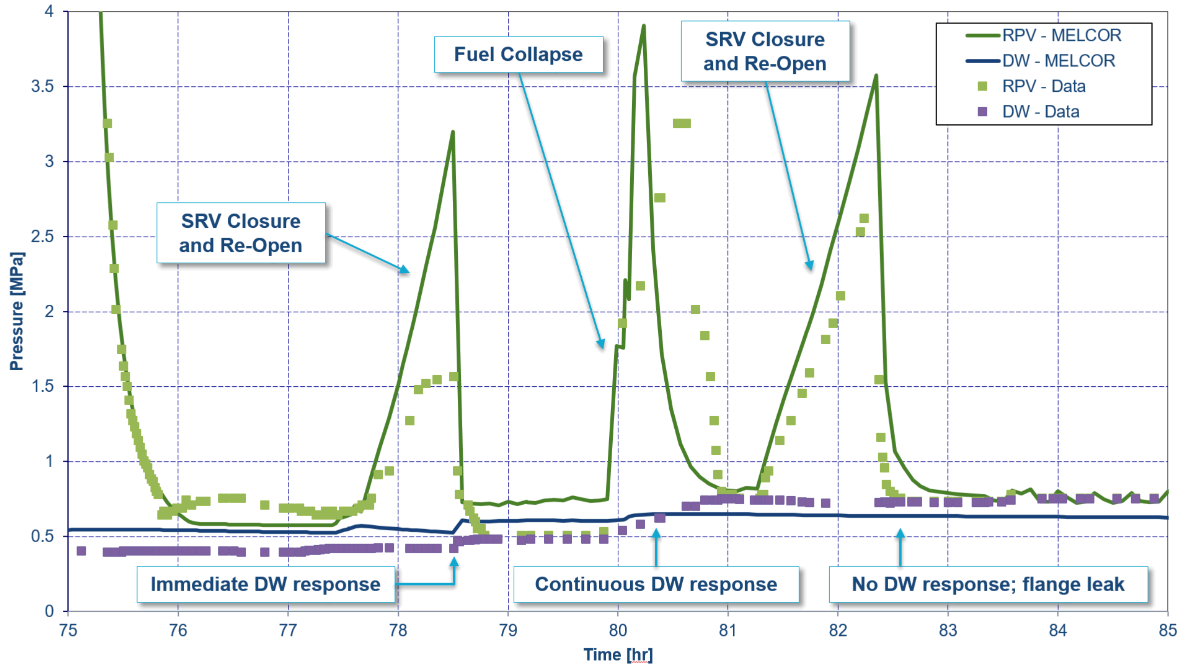

FIG. 9. Reactor pressure vessel pressure of 1F2, showing plant data

3.3 Calculation of Three Week Long Source Term for 1F2

The source term for this scenario out to three weeks for 1F2 is shown in Figure 10. It can be seen that there are large increases in the release fraction at the timings of leakage and containment failure. Additionally, it can be seen that there is an increase in the source term during the period of long term injection after 100 hours into the event. Capturing this is of key importance to properly estimating the radionuclide release. However, there are large uncertainties in long term water injection and MCCI behavior.

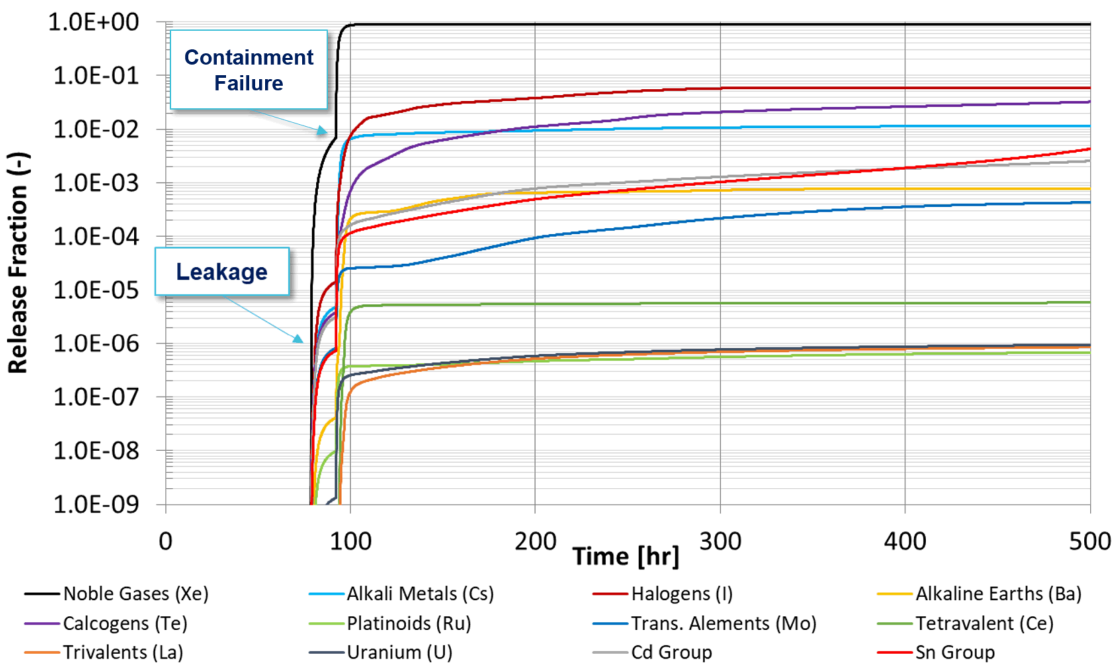

FIG. 10. Long term release of radionuclides in 1F2, showing periods of leakage and the point of containment failure.

4. FUKUSHIMA DAIICHI UNIT 3 ACCIDENT PROGRESSION

The accident at the 1F3 reactor may have resulted in the largest release to the environment. This is a combination of the reactor size, which is 50% more than that of 1F1 and the amount of core damage predicted. The onset of core damage was delayed in the case of 1F3 by the operation of both passive and active systems until 40.8 hours after the earthquake. Key timing for the 1F3 scenario modeled by MELCOR are shown in Table 3.

TABLE 3: KEY EVENT TIMINGS FOR THE ACCIDENT AT FUKUSHIMA DAIICHI UNIT 3

Event	**Time [hours]**
First occurrence of water level at TAF	34.7
Onset of hydrogen generation	37.5
First fuel clad failure time	39.4
First fuel rod failure time (melting or collapse)	40.8
First UO_2 relocation to lower head	41.1
First lower core plate failure time	43.3
First FP release from fuel	39.4
Main steam line rupture	42.1
Vessel water dryout in lower head	58.1
Lower head failure	58.1
Hydrogen burn/explosions	68.2

For the first 21.5 hours of the 1F3 accident, the RCIC system was operating until shut down by the operators. Following this the high pressure coolant injection (HPCI) system was initiated and operated until 36 hours. The significantly larger steam consumption of the HPCI system produced a deep depressurization of the RPV, so deep in fact that the effectiveness of HOCI injection to the RPV at the lowest RPV pressure is believed to have been nearly non-existent resulting in loss of water level in the core. The HPCI system finally fails at about 36 hours. Following the termination of this system, the RPV returns to the SRV setpoint as due to no steam being extracted from the RPV and no water injection returned to the vessel. After core degradation began, MELCOR predicts that the hot gasses venting through the lowest setpoint SRV failed the main steam line associated with that SRV. The timing of this failure, and subsequent core slumping events, were enforced to occur at times corresponding to changes in the TEPCO-provided plant data. Alternative scenarios for the depressurization of the RPV have also been proposed, including an automatic depressurization erroneously initiated. The pressure trends of the RPV and drywell for 1F3 can be seen in Figure 11.

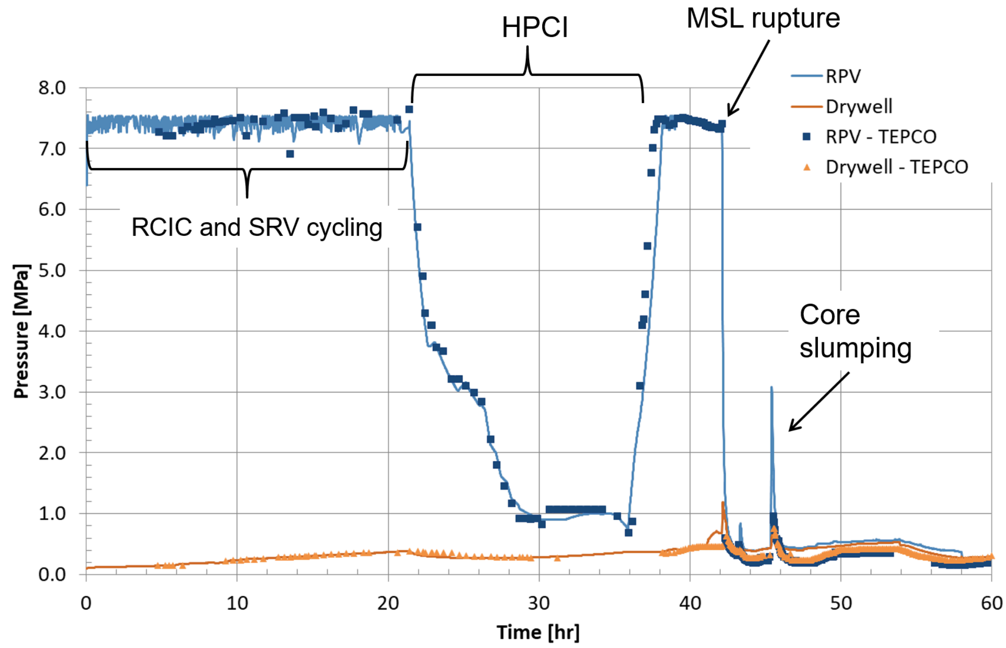

FIG. 11. RPV and drywell pressure for the MELCOR scenario of 1F3, compared to plant data provided by TEPCO

2.1 Core Degradation Modeling in 1F3

The 1F3 accident scenario modeled with MELCOR again clearly demonstrates the forensic approach used in these analyses. It can be seen that there are two spikes in the RPV pressure between 43 and 46 hours into the event; see Figure 12. The first is smaller and occurs just after 43 hours; the second occurs near 45.5 hours. In order to replicate this result, the minor slumping event was assumed to be the failure of the innermost ring of the MELCOR COR package and the larger slump was assumed to be the remainder of the core. It can be seen that the assumption made allow MELCOR to closely represent the spike that occurred in the strip chart data for both peaks.

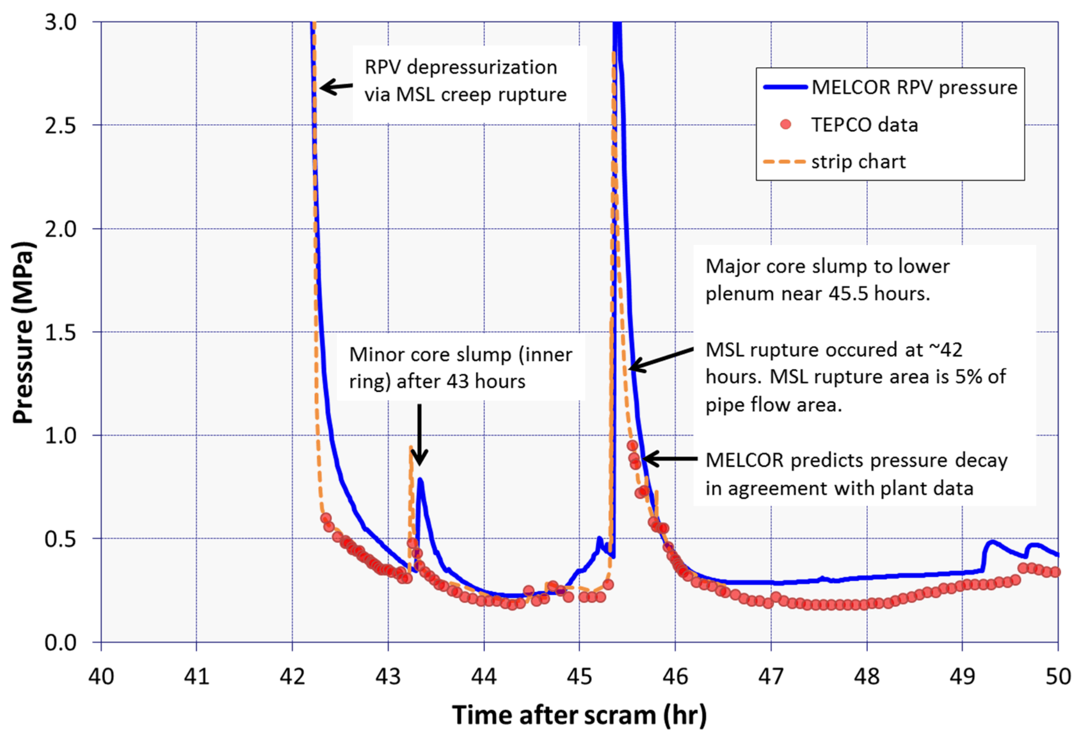

FIG. 12. RPV pressure of 1F3 compared to plant data during the period of core slumping, 40 to 50 hours after the earthquake

5. CONCLUSION

The modeling of the three severe accident at the Fukushima Daiichi reactor site have led to several key insights into how MELCOR represents plant systems, problem boundary conditions and relevant phenomenology. This work has demonstrated how changes in one of these three can significantly impact the other two and thus the overall accident scenario that is being predicted. In particular, we have found that the timing of key events such as core relocation to the lower plenum, the failure of the lower head of the reactor pressure vessel and the failure of the containment itself can be forensically informed, leading to a better prediction of source term. We assert that enforcing such modeled events to occur when observed data suggests preserves the basic core damage phenomena that would be predicted by the code were it calculated in a hands-off manner, but also preserves the overall timing of events leading to a higher fidelity accident replication.

The modeling of the RCIC system was shown to be of key importance to accurately capturing the thermal hydraulic behavior of Unit 2. However, the appropriate modeling of this is complicated by the boundary conditional torus room flooding and how the model represents the wetwell, whether this representation is monolithic or discretized. The self-regulating mode of operation of the RCIC system also demonstrates the robustness of this very common safety system and its potential utility well outside of its intended operational envelope. This work has also highlighted the importance of MCCI in capturing the response of the system for several days after corium relocated ex-vessel. It was predicted that MCCI ablation of the sump lead to a liner breach in 1F1 and long-term MCCI leads to an increase source term.

As MELCOR development continues over the upcoming years, the information gained in this analysis will be used to continually improve the state-of-the-art modeling present within the software. Not only does this work lead to the improvement of MELCOR, but it also informs the decommissioning of the reactors at the Fukushima Daiichi site. In particular, it can provide indication to the location of fuel debris and its chemical composition. This is also true for the location of fission products.

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