FUKUSHIMA DAIICHI UNIT 3 MELCOR INVESTIGATION

Kevin R. Robb, Matthew W. Francis, Larry J. Ott

Oak Ridge National Laboratory: 1 Bethel Valley Road, MS-6167, Oak Ridge, TN, 37831, robbkr@ornl.gov

The Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) jointly sponsored a study of the Fukushima Daiichi accident with collaboration among Oak Ridge (ORNL), Sandia (SNL), and Idaho (INL) national laboratories. The purpose of the study was to compile relevant data, reconstruct the accident progression using computer codes, assess the codes' predictive capabilities, and identify future data needs. The current paper presents extended MELCOR simulation and analysis of the Fukushima Daiichi Unit 3 accident taking into account new knowledge and modeling refinement since the joint DOE/NRC study.

I. INTRODUCTION

I.A. Background

During the 1975–2000 timeframe, ORNL conducted an extensive research program to study boiling water reactor (BWR) accident scenarios and phenomena. During this time, Ott largely developed the BWRSAR code to analyze severe BWR accidents [1]. To consolidate severe accident modeling tools, BWRSAR was provided to Sandia National Laboratories as input into the initial core model development for MELCOR (mid 1980s). During the early development of MELCOR, a model for the Peach Bottom nuclear power plant was developed and exercised by researchers at Brookhaven National Laboratory (late 1980s) [2]. Later, Carbajo expanded and updated the model to MELCOR v1.8.1 [3]. This MELCOR model was then expanded by Dycoda, LLC, updated to MELCOR v1.8.5, and then provided back to ORNL in 2003. Francis then added functions to model additional mitigation measures for long-term station blackouts [4]. Note that over the 30 years of code and model development, the Short-Term and Long-Term Station Blackout (STSBO, LTSBO) accident scenarios for BWRs with Mark I containments, as well as many other scenarios, were investigated [1–8].

Soon after the initiation of the Fukushima Daiichi accident, the DOE set up an emergency response team of subject experts. At that time, the MELCOR model, described in the previous paragraph, was revitalized at ORNL and used in early scoping analyses. Following the emergency response period, the DOE-NE and NRC jointly sponsored the Fukushima Daiichi Accident Study to collect and document data, reconstruct the accident using computer models, and identify future data needs [9]. For that study, the previously described MELCOR model was modified to reflect Unit 3.

A number of reports [9–18] have been released which describe the Fukushima Daiichi accident progression (Table I). Our understanding of the accident continues to evolve as the information is studied and new information is discovered during the decommissioning work and disseminated. Much information used in the joint DOE-NE/NRC Fukushima Daiichi Accident Study was derived from the two reports provided by the government of Japan to the IAEA [10, 11], as well as other information available as of approximately January 2012. Additional information concerning the accident progression was disseminated through TEPCO's Interim Report [12], as well as during a number of technical exchange meetings.

Analysis and simulation of the accident progression have also continued to evolve. During the accident response period, much analysis was based on news reports, very limited data, and using models not necessarily reflective of the Fukushima Daiichi units. In addition, analyses were performed using bounding assumptions and possibilities.

The current paper presents extended analysis of Fukushima Daiichi Unit 3 taking into account new knowledge obtained and additional modeling modifications made since the joint DOE-NE/NRC study [9]. The results are based on the authors' understanding and modeling of the Unit 3 accident as of April 2012.

TABLE I. Key Reports on Fukushima Daiichi Accident

Pub. Date	Author Org.	Ref.
June 2011	IAEA	13
June 2011	Gov. of Japan	10
Sept. 2011	Gov. of Japan	11
Oct. 2011	JANTI	14
Nov. 2011	INPO	15
Dec. 2011	TEPCO	12
March 2012	ANS	16
March 2012	NISA	17
July 2012	NAIIC	18
July 2012	DOE-NE/NRC	9

I.B. Accident Progression, Data, and Observations

The general accident progression and some key details relevant to simulating the accident are summarized below. Most of the information can be found in multiple references. It is stressed that there are many details that are not discussed here. The amount of elapsed time since reactor shutdown is indicated in brackets, {h : min}.

I.B.1. Accident Progression and Operator Actions

At 14:47, on March 11, 2012, {00:00}, Unit 3 tripped and shut down. The reactor shut down successfully and no abnormalities are noted to have occurred. Due to a loss of offsite power, two emergency diesel generators started. At {00:18} the reactor core isolation cooling system (RCIC) was manually activated and automatically tripped at {00:38} due to high water level. Around {00:51} the tsunami caused a loss of AC power at Unit 3. However, DC power from the station batteries remained available.

The RCIC system was manually activated again at {01:16} and later tripped for the final time at {20:49}. During RCIC operation, the RCIC was aligned to take suction from the condensate storage tank (CST). The operators closed the minimum flow line, preventing the pumped liquid from being passed to the suppression chamber (SC). They throttled the test line (to the CST) and used the flow controller to adjust the liquid injection into the primary system [19].

The diesel-driven fire pump (DD-FP) was turned on at {21:19} [20] and used to run containment sprays [19]. From {21:19–36:18} and {38:21–40:56} the SC sprays were active, and from {40:52} to around {41:53–42:23} the dry well (DW) sprays were active [19]. The DD-FP shut off at {55:28} due to fuel depletion [20].

At {21:48} the high-pressure cooling injection (HPCI) system automatically activated. The HPCI system was configured and manually controlled in the same fashion as described for the RCIC. The reactor water level measurement was lost at {29:49} due to battery depletion [19]. A decision was made to inject water into the primary system using the DD-FP in lieu of the HPCI [19]. This decision is described as being based on concerns that the HPCI was no longer injecting water into the primary system [the discharge pressure of the HPCI system approached the reactor pressure vessel (RPV) pressure], and concerns over HPCI damage due to low turbine RPMs [19]. In addition, at some point, "the Emergency Countermeasures Headquarters and the Main Control Room instructed all operators to shift to DD-FP for water injection, in the event water injection by the HPCI becomes unstable" [19]. The HPCI was manually shut off at {35:55} [19].

After the HPCI was shut off, the reactor pressure increased to above the discharge capability of the DD-FP before the switchover could occur. Around this time, actuation of the safety relief valves (SRVs) was attempted in order to lower the RPV pressure below the discharge pressure of the DD-FP; however, the SRVs did not actuate. Thus, the capability to inject water into the RPV was lost. Efforts were made to restart RCIC but were unsuccessful. During the RCIC restoration effort, operators passed through the HPCI room. No abnormalities in the HPCI room were noted [19], that is, no steam-filled room or large leakage.

The vent line valve lineup from the SC to the stack was complete at {41:54} with the rupture disk preventing venting. The RPV was depressurized by manually actuating a SRV around {42:21}. Based on the DW pressure data, the containment is believed to have successfully vented sometime between {42:23–42:37}. As the rupture disk was preventing venting, the depressurization of the RPV likely increased the containment pressure enough to fail the rupture disk. At {44:30} it was confirmed the vent line was closed and was reopened at {45:43} after replacing gas cylinders. Due to difficulties in maintaining the vent line valves open, the vent line intermittently closed and opened over the next couple days.

After the RPV was depressurized, freshwater injection into the primary system via a fire engine started at {42:38} and lasted until the freshwater source was depleted at {45:33}. The fire engine switched over to a different water source and seawater injection commenced {46:25-58:23}. The DD-FP is noted to have been operating during the switch over {45:33–46:25}; however, it is unclear whether the DD-FP was injecting water into the primary system during that period. Due to the decreasing water level of the backwash valve pit (seawater source), water injection was temporarily suspended {58:23–60:33} while the fire engine was repositioned. Seawater injection then recommenced at {60:33}. At {68:14} water injection ceased due to damage caused by the Unit 3 reactor building explosion. Seawater injection resumed at {73:43} and continued through and beyond the explosion of the Unit 4 reactor building at {87:13}.

I.B.2. General Accident Progression Observations

The Unit 3 accident followed a Long-Term Station Blackout scenario. Many previous BWR LTSBO analyses assume battery failure and loss of cooling functions in less than 8 h after SCRAM [2 (6 h), 6 (6 h), 21 (4 h)]. However, for Fukushima Daiichi Unit 3, the RCIC was used to successfully manage the reactor decay heat for 20 h and 49 min. After the RCIC shutdown, there was a 59 minute delay until the HPCI started. The delay is minor given that it would take a few hours for the core to uncover at that point. The HPCI was then used to successfully manage the decay heat until at least 29 h and 49 min into the accident when liquid level indication was lost due to battery depletion. Between {29:49} and {35:55} it is uncertain as to whether the HPCI was successfully injecting water into the reactor to offset the decay heat. After HPCI shut down, it appears there were four major periods of no water injection. The causes for these periods were noted in Section I.B.1. The seven periods of time where water was not being injected into the primary system are summarized in Table II.

TABLE II: Time Periods of No Water Injection into RPV

Period	Time {h:m}–{h:m}	Duration [min]	Timeline Period (between)
1	00:00–00:18	18	Reactor SCRAM RCIC startup
2	00:38–01:16	38	RCIC shutdown RCIC startup
3	20:49–21:48	59	RCIC shutdown HPCI startup
4*	35:55–42:38	386	HPCI shutdown Fire engine inj. - fresh
5**	45:33–46:25	52	Fire engine inj. - fresh Fire engine inj. - sea
6	58:23–60:33	130	Fire engine inj. - sea Fire engine inj. - sea
7	68:14–73:43	329	Fire engine inj. - sea Fire engine inj. - sea

*HPCI may have started injecting little or no water before {35:55}

**DD-FP was possibly injecting water during this time

I.B.3. Data for Simulation Comparisons

The publically available data provided by TEPCO for the RPV, DW and SC pressure and the RPV water level [22] are used for comparison. This data is contained within the Data Portal developed by Idaho National Laboratory (INL) as part of the Fukushima Daiichi Accident Study [9]. Note that there are timeframes where data is unavailable. This is especially true for the RPV temperature data which begins at 6:30 on March 19, 2011 [23]. There are also questions concerning the validity and uncertainty of the data.

There are two other data sources that are not included in the comparison. First, data in the form of strip-chart recordings for the RPV pressure, in addition to other parameters, have been released by TEPCO. Much data is limited to the first 1.5 h; however, some data, such as the RPV pressure, is provided up to 40 h after the accident initiation. The translation, digitization, and verification of the strip-chart data is currently unavailable.

During the ongoing decommissioning work, there have been additional data taken and observations made by TEPCO. A number of surveys have mapped the dosage inside the reactor building. A door to the torus room in the NE corner was found to be bowed outwards (outwards from the direction of containment), March 14, 2012 [24]. During an inspection on May 23, 2012, the door to the traversing in-core probe room (TIP room) was observed to be on the ground and was described as being blown outwards [25]. Inspection of the torus room on June 6, 2012, indicated the torus room and nearby stairwell were partially filled with water up to approximately the midpoint of the suppression chamber [26]. Future examination of these observations may provide additional insight concerning the accident progression.

II. MELCOR MODELING of UNIT 3

II.A. Model Overview

The MELCOR model of Peach Bottom, described in Section I.A, was used as the basis for the Unit 3 model. The model was modified to reflect the systems and the accident progression at Unit 3. The model has been previously described in detail [9]. In summary, RPV volumes and masses, DW volume, SC and CST volume and initial water mass, the core decay power and fuel loading, SRV set-points, and the flow rate capacity for the HPCI and RCIC were modified to reflect Fukushima Daiichi Unit 3.

The timeline events including reactor scram timing, loss of A/C power, RCIC and HPCI availability, SC venting availability, operator action to depressurize the RPV, and the timing and flow rate of the fire injection line were prescribed to the model (Table III). The logic controlling the RCIC and HPCI flows through the minimum flow and system test line was added and modified. Likewise, a flow path and timing for water injection from the fire engines were added.

Section II.A.1 highlights some of the modifications made to the model for use in the joint DOE-NE/NRC Fukushima Daiichi Accident Study [9]. Section II.B discusses additional modifications made since the Fukushima Daiichi Accident Study. Finally, Section II.C notes some of the model deficiencies.

II.A.1. SRV Actuation

In the original, unmodified Peach Bottom MELCOR model, the logic for automatic SRV actuation relied upon the main steam line to suppression chamber differential pressure. However, while the reactor pressure is high, the SRV actuates at a predefined main steam line pressure set-point and does not depend upon the differential pressure. This was corrected for the Unit 3 model.

II.A.2. Suppression Chamber Nodalization

In the unmodified Peach Bottom MELCOR model, the SC was modeled using a single control volume.

Modeling the SC in this fashion is common [21]. However, modeling the torus in this fashion cannot capture localized saturation and thermal stratification [27]. As a stop-gap measure, the torus was divided into eight sections in the circumferential direction as described in [9]. This remains as an area for future model refinement and analysis.

TABLE III: Applied Timeline Summary

Time	Event
00:00	SCRAM
00:18	RCIC available
00:38	RCIC unavailable
00:51	AC power lost
01:16	RCIC available
20:49	RCIC unavailable
21:19	Suppression chamber sprays start
21:48	HPCI starts
29:00	HPCI liquid injection stops (assumed)
35:55	HPCI shutdown, steam flow stops
36:18	Suppression chamber sprays stop
38:21	Suppression chamber sprays start
40:52	Drywell sprays start
40:56	Suppression chamber sprays stop
41:54	Vent lineup complete w/ rupture disk closed
42:08	Drywell sprays stop
42:21	SRV opens, RPV is depressurized
42:38	Fire engine water injection RPV start
44:30	Vent line closes
45:33	Fire engine water injection RPV stop
45:43	Vent line opens
46:25	Fire engine water injection RPV start

TABLE IV: Assumed Water Injection Timing and Rate via the Fire Engines into RPV

Day	Water Injection Timing	Duration	Injection Rate
	{h:min}–{h:min}	min	[kg/s]
3/13	42:38–45:33	175	7.88
	45:33–46:25	52	0
	46:25–57:13	648	7.88
3/14	57:13–58:23	70	5.41
	58:23–60:33	130	0
	60:33–68:14	461	5.41
	68:14–73:43	329	0
	73:43–81:13	450	5.41
3/15	81:13–	-	8.94

II.A.3. Water Injection

The water injection into the primary system RPV from the fire engines was prescribed to the model and the details summarized in Table IV. The water injection timing is based on the timeline presented in [11, 19]. Using this timing, the water flow rate was determined using estimates of the total water injection volume from TEPCO [28]. The amount of water that was injected and made its way to the core region is currently a key uncertainty. In addition, the possible usage of the DD-FP to inject water after RPV depressurization is unknown.

II.B. Model Updates

A few modifications were made to the Unit 3 model since the joint DOE-NE/NRC Fukushima Daiichi Accident Study [9].

II.B.1. Decay Heat

The model previously used the decay heat based on the American Nuclear Society's National Standard for Light Water Reactors (ANSI/ANS-5.1-1979). TEPCO has provided decay heat data, for specific points in time, based on ORIGEN calculations [9]. This decay heat was approximately 10% lower than the ANS standard decay heat curve. Additional data points were estimated between the TEPCO data and implemented to reduce the error introduced through MELCOR's interpolation (Section IV.A.1, Signature 1).

II.B.2. HPCI model

A few methods, each including several cases, of modeling the HPCI have been investigated by the authors including modeling the flow rates as a function of main steam line to SC differential pressure, assuming various values for operator throttling, system degradation with assumed pump/turbine curves, and fixing the flow rates. Uncertainties in the off-nominal performance of the HPCI system, coupled with the lack of detailed descriptions of operator actions (as a function of time) and RPV water level and SC pressure data, complicate modeling the actual HPCI performance.

For the current study, the liquid injection rate into the feedwater line and the steam flow rate through the HPCI were specified, a posteriori (Fig. 1). The purpose was to identify the flow rates needed to reproduce the available reactor pressure and water level data. It is assumed that the use of the flow controller and throttling the test line result in this HPCI performance.

The pumped liquid flow was suctioned from the CST and injected into a feedwater line. It was assumed the HPCI pump was no longer injecting water into the primary system after {29:00}. The TEPCO data indicates the differential pressure between the main steam line and containment fell to 631–901 kPa (92–131 psi) starting at {28:13} [22] (Fig. 2). Although there are gaps in the containment pressure data, the differential pressure likely remained low until the HPCI was shut off at {35:55}. The ability of the HPCI to inject water and provide cooling water to the bearings (approximately 4 kg/s) under the conditions of low reactor pressure and low differential pressure driving the turbine is currently unclear to the authors. HPCI modeling remains a key area for future modeling effort and refinement.

II.B.3. Containment Sprays

Early reports did not describe containment spray usage [10, 11]. Spray usage was previously assumed by the authors for the ORNL Unit 3 analyses presented in the Fukushima Daiichi Accident Study [9]. This assumption was based on information contained in the English translation of the operator log book such as the discharge pressure and fuel level of the diesel-driven fire pump [20]. TEPCO indicated containment sprays were used in the interim report [12, 19]; however, the flow rate remains unknown. A rough analysis suggests the DD-FP may have the capability to inject a couple to a few cubic meters of water per minute [9]. Table V provides the containment spray timing and the assumed flow rate used in the current and previous [9] analyses. The current work assumes the DW sprays shut off at {42:08}.

II.C. Model Limitations

The containment volumes and elevations are representative of Unit 3. However, the elevations/heights of the vessels, piping, and building were not modified. A limited comparison between information for some components of Unit 3 and the model showed the relative elevations were comparable. Specific failure modes, such as head lifting, are generically modeled. More detailed information, such as the containment head flange geometry and the size and pretension of the bolts, and other penetration details, are required to develop quantitative models of the containment failure modes.

The reactor building (outside of containment) was not modified to reflect the Fukushima Daiichi plant. Resources were focused on modeling the in-containment features and accident progression. Previous work notes the complexities in reactor building modeling and indicates that the building can have an important impact on mitigating radionuclide releases [29]. More information and effort are required to accurately model the room volumes, elevations, and flow paths, including the ventilation system.

The vent lines are not modeled in detail. Instead, a flow path is specified that takes fluid from containment directly to the outside environment (or other specified location). This rough modeling approach to venting does not capture cross-flow to Unit 4, radionuclide deposition onto the walls of the vent pipe, the trapping of gasses in-between segments, or the hydraulics of the vent lines.

Due to the model limitations mentioned, the study focuses on the in-containment accident progression.

III. SIMULATION RESULTS AND ANALYSES

The simulation predictions are compared to the available data from TEPCO for the first 48 h of the accident in Fig. 2–7 and Fig. 9. The results are discussed within three timeframes.

III.A. SCRAM to RCIC Shutoff

The reactor pressure and water level are fairly well reproduced (Figs. 2–5). The “saw-tooth” reactor pressure and water level during RCIC operation is an artifact of the RCIC modeling methodology. However, the overall water injected and steam vented from the reactor to the SC is thought to be comparable.

The containment pressure is under-predicted (Fig. 6). The authors believe this is due to the modeling of the SC, which can only approximately capture localized saturation and cannot capture thermal stratification and convection cell phenomena. With the current SC nodalization (eight control volumes), the section where the steam is vented saturates in 50 min, while the coldest section (180° opposite) saturates over 40 h later (Fig. 7). Based on previous work, when the SC is modeled as one large control volume, it takes tens of hours for the SC to saturate [9]. Another possible cause for the discrepancy in containment pressure is leakage through the recirculation pump seals, which is assumed to be zero in the current simulation.

III.B. HPCI Startup to RPV Depressurization

As a result of specifying the HPCI flow rates, the reactor pressure and water level are well reproduced (Figs. 2–5).

After the HPCI steam flow shuts off at {35:55}, the primary system is predicted to repressurize in 153 min, in contrast to the TEPCO data which indicates the system repressurized within 62–138 min [22] and strip-chart data released by TEPCO which suggests approximately 100 min. There are a number of factors that influence the

TABLE V: Containment Spray Details

Location	TEPCO Info. [12, 19]	Previous Study [9]		This Study	
	Timing	Timing	Flow Rate	Timing	Flow Rate
SC	{21:19}–{36:18}	{21:19}–{36:18}	25.24 L/s	{21:19}–{36:18}	12.62 L/s
SC	{38:21}–{40:56}	{38:21}–{40:56}	25.24 L/s	{38:21}–{40:56}	6.31 L/s
DW	{40:52}–{41:53–42:23}	{40:52}–{54:28}	25.24 L/s	{40:52}–{**42:08**}	6.31 L/s

repressurization rate, including the reactor water level, power, mass, and volume. If the reactor mass, volume, and power are accurately modeled, then the repressurization rate can be used to “back-out” the reactor water level. Based on a number of previous simulations, the repressurization period extends with higher initial water level. The simulation results suggest the water level was at or below the top of active fuel (TAF) when the HPCI shut off, supporting the notion that the HPCI water injection was not effectively offsetting the decay heat.

The containment pressurization (Fig. 6), with data starting at {38:13}, can also provide a number of insights. At {38:21} the SC sprays are activated. Both the data and simulation exhibit an approximately 15 kPa drop in the DW pressure around this time. Despite the usage of containment sprays and nearly an additional 20 h for decay heat reduction, the containment pressurization rate based on the TEPCO data (~49 kPa/h) from {38:23}–{40:43} is more than double the containment pressurization rate during the first 18 h (~20 kPa/h). The additional heat and noncondensable gas generation from cladding oxidation during this period could be a cause for the high containment pressurization rate from {38:23}–{40:43}. Despite the simulation predicting cladding oxidation starting at {37:20}, with first gap releases occurring at {37:45}, the containment pressurization is under-predicted during this period. This suggests the current simulation may be under-predicting the amount of cladding oxidation occurring during this time period. Uncertainties related to the containment spray modeling (flow rate, efficiency) may also contribute to the discrepancy.

From {40:43}–{42:08} the TEPCO data indicates the DW pressure plateaued to around 465 kPa abs (67.4 psia). It is unclear why the pressure plateaued. In contrast, the simulation predicts continued increase in containment pressure. One possible cause for the pressure plateau is that the containment began leaking at this pressure. This possibility may be supported by the radiation measurements at the main gate, which indicate an increase in activity around {41:15}, which is approximately an hour before containment venting is noted to occur. This is shown in Fig. 8 by the increased magnitude of the data points just before venting. Another possibility is that the containment spray flow rate was increased during this period, increasing containment cooling and condensation.

III.C. RPV and PCV Depressurization and Beyond

The RPV is depressurized at {42:21} in the simulation, which causes a spike in the containment pressure. While the predicted containment pressure is less than the data indicates, the spike in containment pressure is still large enough to cause the rupture disk to fail and open the vent line. The RPV depressurization causes the water level to drop well below the core in the simulation. However, the data suggests the water level was higher after containment venting. The authors believe the water

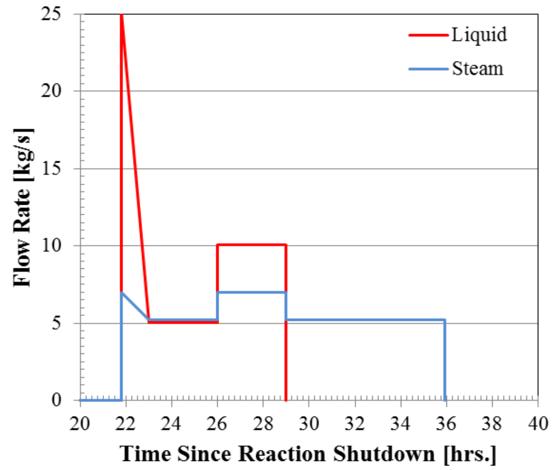

Fig. 1: Assumed HPCI Steam and Liquid Flow

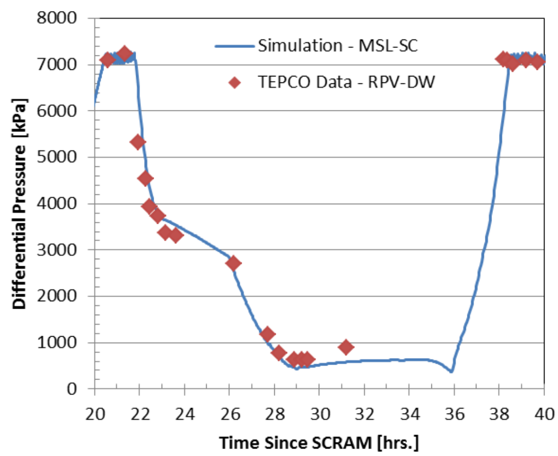

Fig. 2: Main Steam Line to Containment Differential Pressure Data and Prediction

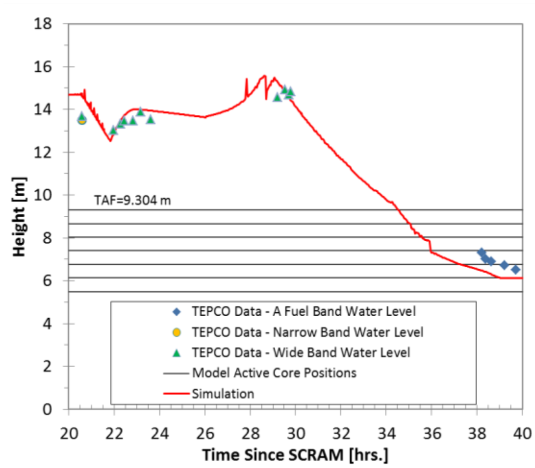

Fig. 3: Reactor Water Level Data and Prediction

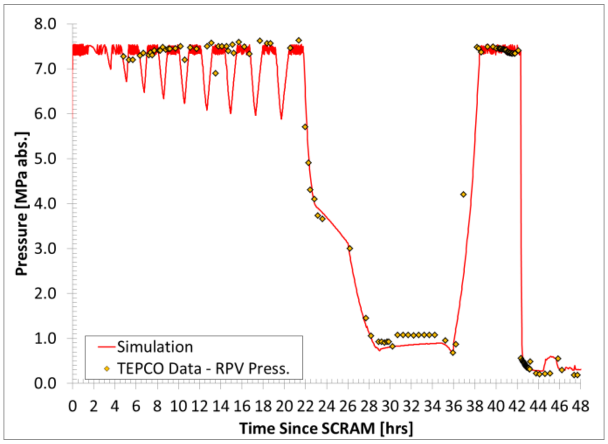

Fig. 4: Pressure in RPV

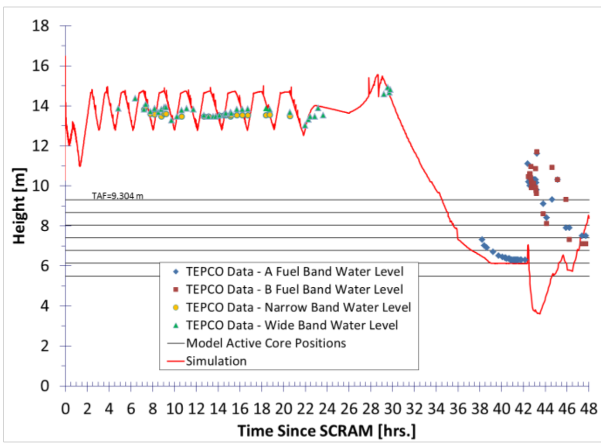

Fig. 5: Water Level in Core Region

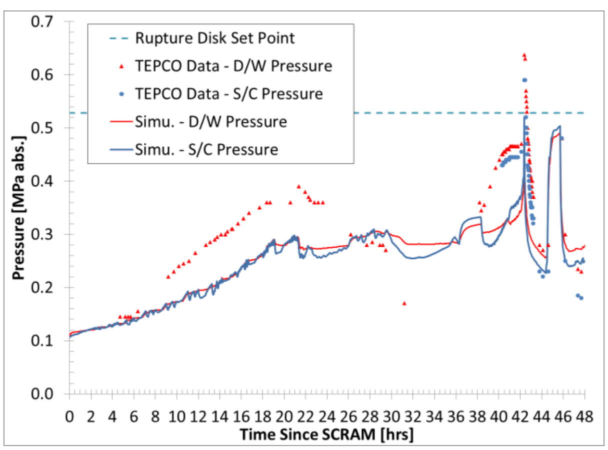

Fig. 6: Pressure in Containment

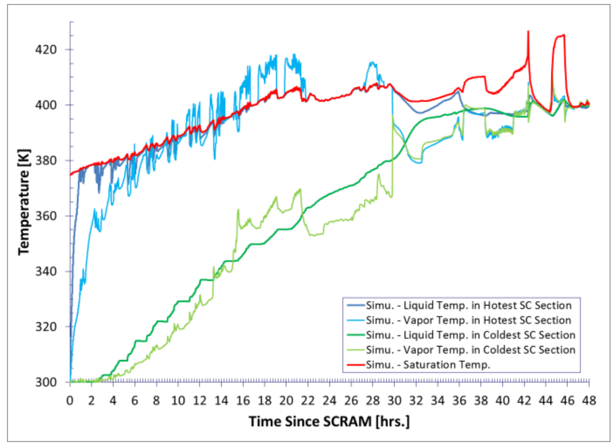

Fig. 7: Suppression Chamber Temperatures

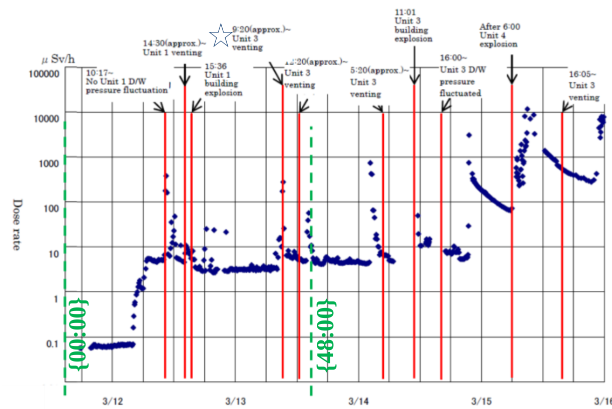

Fig. 8: Dose Data at Main Gate, adapted from [12]

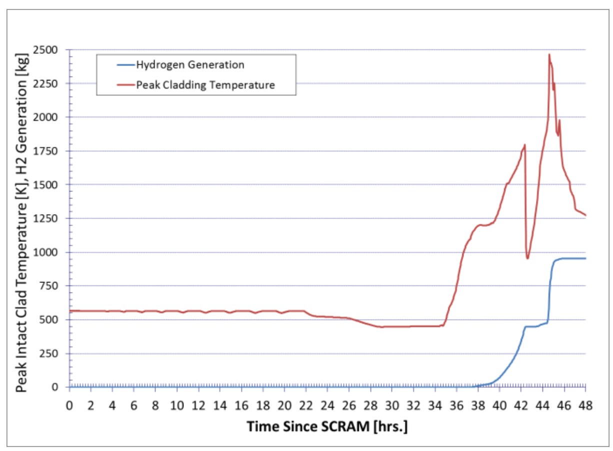

Fig. 9: Peak Clad Temperature and Hydrogen Generation

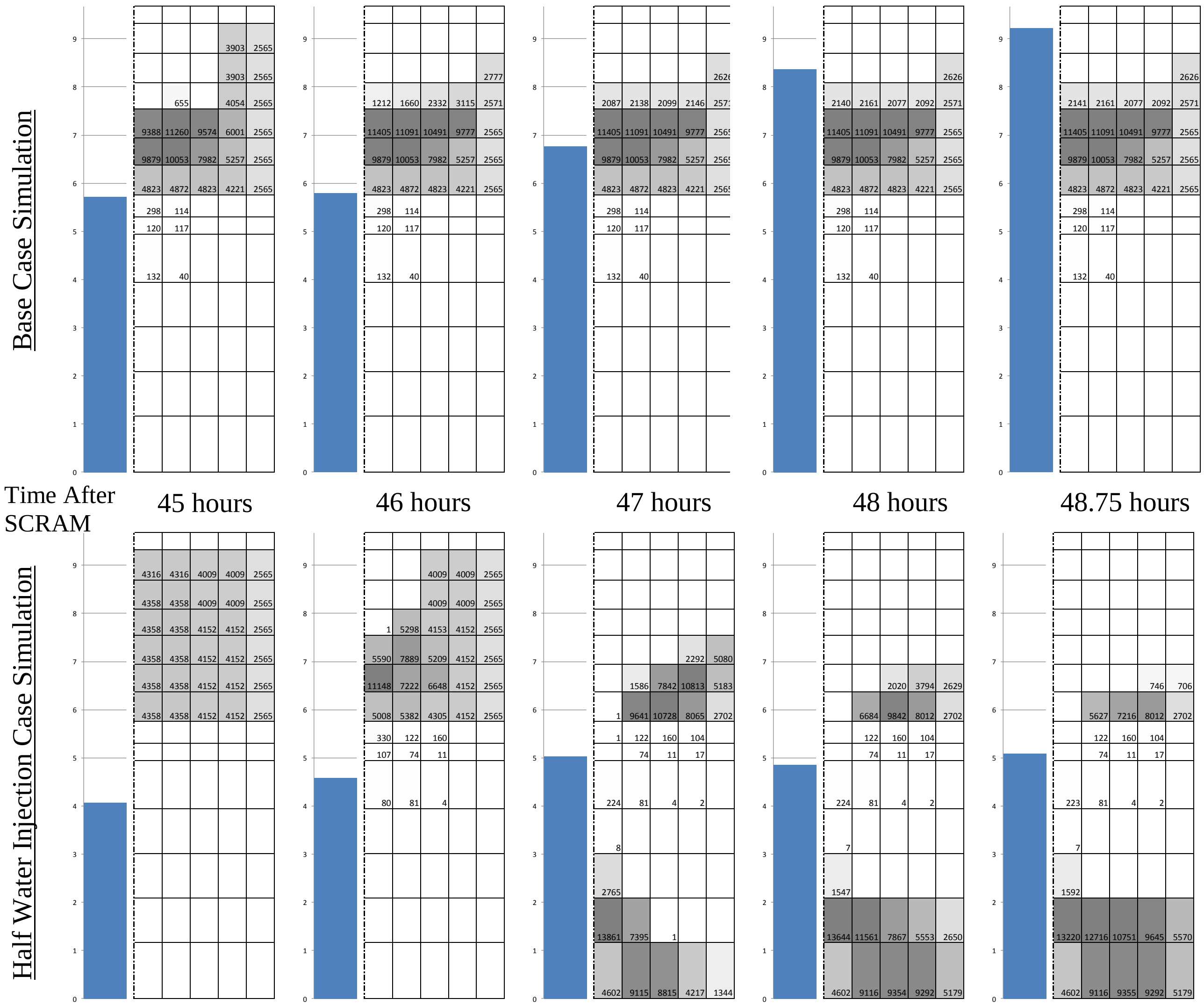

Fig. 10: Fuel Location [kg] and Water Level [m] vs. Time

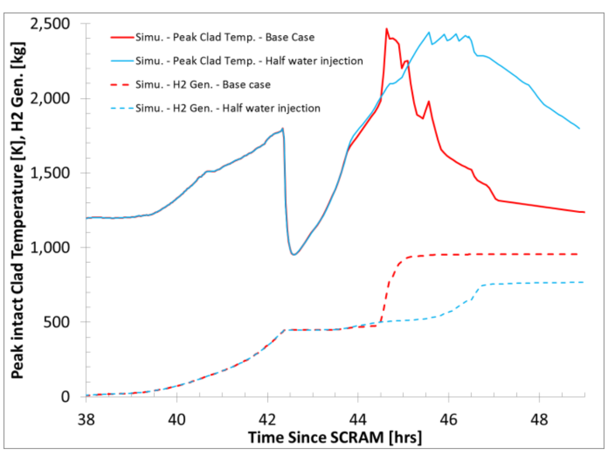

Fig. 11: Predicted Peak Cladding Temperature and Hydrogen Generation

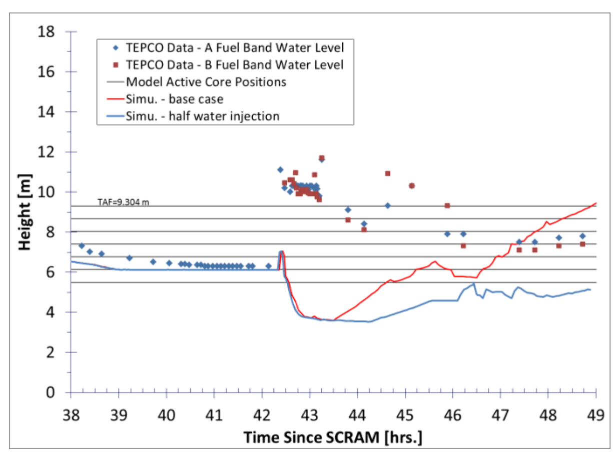

Fig. 12: Water Level Comparison Assuming Decreased Water Injection

level measurements after RPV venting are not accurate, possibly due to flashing or gas entering the measurement reference leg. The lower saturation temperature results in a predicted decrease in cladding temperature (Fig. 9). In addition, reduced steam flow through the core reduces cladding oxidation.

Water injection starts soon after venting at {42:38}. The simulation predicts considerable core damage occurs before the assumed water injection timing and rate are able to offset the decay heat and reflood the core. Most fuel relocation is predicted to occur around {44:40}–{45:38}. By the end of the simulation, approximately two-thirds of the fuel cladding has melted and relocated. Figure 10 illustrates the core degradation progression with the numbers indicating the fuel location.

The water injection rates remain a key uncertainty. TEPCO has provided estimates of the amount of water injected; however, the amount of uncertainty is unknown. In addition, the flow rate as a function of time is also unknown. Valve misalignments or leaks in the piping, connections, and valves provide additional uncertainties. An additional simulation was run where half of the assumed water injection (Table IV) was supplied to the core (Figs. 10–12). As expected, much more core damage occurs. However, interestingly, the decreased water addition results in starving the cladding of steam, which delays and reduces cladding oxidation and heat-up.

Beyond 48 h, the accident progression is primarily dependent upon the rate at which water is injected into the core region. The vent timing has a secondary impact by varying the containment and RPV pressure, and therefore the saturation temperature. The current simulation predicts sufficient hydrogen is generated for explosions in the Unit 3 and 4 reactor buildings.

IV. INSIGHT AND RECOMMENDATIONS

IV.A. Signatures

The following sections describe some explanations for behaviors that can be observed in simulation results. These are provided to aide modelers and those trying to understand modeling methodologies and results.

IV.A.1. Signature 1 – Rapid RPV Water Level Drop

After SCRAM, there is a rapid drop in the RPV water level due to void collapse followed by a relatively high boil-off rate from the initially high decay heat. If the RCIC is not activated, the water level will reach TAF around {1:15}-{1:30}. The following should be taken into consideration to accurately model this time regime.

First, care must be taken to ensure the model is converged and has the correct initial RPV void at the time of SCRAM.

Second, TEPCO indicates the RCIC was manually activated at {0:18} and automatically tripped at {0:38} due to high water level. If the RCIC is not activated during this time, the water level will continue to fall and be near TAF when the RCIC is said to be manually re-started at {1:16}. In the simulation presented earlier, the RCIC activates during this timeframe, however, the water level is not fully replenished before it shuts off at {0:38}.

Third, TEPCO provided six data points for the first 3 h after SCRAM. If this data is directly implemented into MELCOR, the decay heat is overestimated by approximately 5% due to the linear interpolation of the data in MELCOR. As the water is replenished by the RCIC from the CST, the CST capacity is artificially reduced. To prevent this simulation artifact, an additional ten data points were interpolated and incorporated into the current model.

IV.A.2. Signature 2 – Increasing RPV Pressure

If the logic modification to the SRV is not made (Section II.A.1), the actuation point of the SRV will increase with increasing containment pressure. Thus, once the RPV reaches the lowest SRV set-point, the RPV pressure will increase at approximately the same rates as containment pressure. This can be easily observed as an increasing reactor pressure over the time period of RCIC operation. The increased pressure at which the steam is being vented increases the internal energy of the vented steam, reducing the amount of steam that needs to be vented. Note that under-prediction of the containment pressurization, perhaps due to deficiencies in the suppression chamber modeling, can mask this effect.

IV.A.3. Signature 3 – Slow Containment Pressurization

The current and previous [9, 10] simulations under-predict the pressurization of the containment during RCIC operation. Several modeling and simulation issues have been identified as possible contributors for the decreased pressurization rate. The authors believe the primary cause for the discrepancy in the current simulation results is due to the first two issues discussed.

First, if steam is vented through one location in the SC for extended periods, local saturation and thermal stratification can occur. Modeling the SC using one control volume in MELCOR cannot capture these effects as well as other thermal hydraulic phenomena such as local and global recirculation cell formation. The current model simulates the SC using eight control volumes (divided in the circumferential direction). This allows for energy to be deposited into a section of the pool, for example, venting from single SRV, instead of the whole pool. The simulation predicts one section of the pool saturates within an hour of the accident while it takes over 30 h for the coolest section to saturate (Fig. 7). However, this nodalization cannot capture convection cells or thermal stratification within a section. Local saturation, stratification, and recirculation phenomena were previously investigated [27], and it is possible these may have occurred in the Unit 3 SC.

Second, the condensation of submerged/rising steam bubbles in a pool is modeled in MELCOR using the product of two empirical efficiencies. The first efficiency accounts for the bubble rise distance. The second efficiency accounts for the water sub-cooling. The product of the two efficiencies determines the fraction of steam condensed in the pool versus passed through the pool to the atmosphere. The containment pressurization is impacted by the amount of steam that is vented into the SC and not condensed. For the current simulations, the default values for the efficiencies were used. However, the pool sub-cooling efficiency was modified in previous simulations [9 (ORNL Unit 3 results)] using sensitivity coefficient 4405. The water sub-cooling efficiency was set to 100% for pool sub-cooling of 6 K instead of the default 5 K. This had a noticeable impact on the containment pressurization and aided reproducing the TEPCO containment pressure data. Note that deficiencies in capturing localized saturation (see previous paragraph) also impact the predicted steam condensation by the pool.

Third, the recirculation pump seals can degrade and leak if they are not cooled. Modeling leakage of high-temperature steam into containment increases the containment pressurization rate. However, as the recirculation pumps were not in operation during the accident, it is unclear whether the seals were actually compromised, resulting in significant leakage.

Fourth, if the RCIC is modeled as bypassing flow into the SC (through the minimum flow line), the sub-cooled water will decrease the containment pressurization. The RCIC should not be modeled in this fashion as the operators are described to have closed the minimum flow line (to the SC) and instead used the system test line (to the CST) [19]. This same comment applies to the HPCI.

Finally, over-prediction of steam condensation in the DW or under-prediction of the heat transfer from the reactor primary system to the drywell may also result in an under-prediction in the containment pressurization.

IV.A.4. Signature 4 – “Saw-Tooth” Reactor Pressure

As noted in Section III.A., the “saw-tooth,” up-down-up-down reactor pressure during RCIC operation is due to the intermittent injection of water into the RPV. Operators actually throttled the RCIC to smooth-out and reduce such variations. This manner of operation is supported by the strip-chart RPV pressure data released by TEPCO. Additional modeling effort is required to correct this. However, this modeling detail likely has little impact on the overall accident progression as the RCIC was able to successfully offset the decay heat.

IV.A.5. Signature 5 – Long RPV Repressurization

As discussed in Section III.B, the RPV repressurization (Fig. 2), starting after the HPCI was shut down at {35:55}, gives an indication of the water level at {35:55}. Very long (a few hours) repressurization times can be the result of initially high (above TAF) water levels at {35:55}. Other modeling factors that can influence this include low decay heat and excessive reactor mass or primary system volume.

IV.A.6. Signature 6 – Early-Late Depressurization

The timing of containment depressurization around {42:21} can be impacted by the modeling approach. In the current model, the vent line is available at {41:54}, after which time the vent line will open once the containment pressure surpasses the rupture disk set-point (528 kPa abs.). Modeling the vent line in this fashion, instead of specifying a time where the vent line opens, can result in venting occurring any time after {41:54} instead of when venting was noted to occur. This modeling detail depends on the purpose and approach for the simulation.

IV.B. Simulation Results Summary

The simulation results suggest the RCIC was able to successfully manage the decay heat until it shut off at {20:49}. Due to the deficiencies in modeling the SC and/or recirculation pump seal leakage, the containment pressure is under-predicted.

Stemming from uncertainties in the HPCI performance, the HPCI liquid and steam flow rates were specified a-posteriori in this investigation. The purpose was to provide insight into the flow rates which could reproduce the existing plant data. The HPCI system is rated for 268 kg/s of injection capability [10]. For the Hatch Nuclear Power Plant, the steam requirements while the HPCI is injecting 268 kg/s and is suctioning from the CST range from 13.8 to 23.7 kg/s, depending on the reactor and SC pressures. The values for the water injection and steam flow used in the simulation in order to reproduce the reactor pressure and water level are approximately 2–10% and 22–51% of these nominal system values for the liquid and steam flow, respectively, Fig. 1. However, the HPCI was not operating under nominal conditions.

The HPCI likely managed the decay heat until {29:49}; however, after this the HPCI may have not been able to offset the decay heat. A water level below TAF (Fig. 3) at {35:55} is supported by a comparison of the predicted time it took to repressurize the RPV after

{35:55} and the data (Fig. 2). This also provides credibility to the water level measurements data from {38:14} to {42:09}, which indicates the water level was below TAF. Furthermore, the containment pressurization rate from {38:23}–{40:43} (Fig. 6) suggests cladding oxidation was occurring during this time. Finally, one possible explanation for the increased radiation activity around the main gate at {41:15} (Fig. 8) could be from cladding failure (occurring before {41:15}) and containment leakage, possibly supported by the plateau in containment pressure data (Fig. 6), prior to the venting which occurred around {42:21}.

Using the water injection estimate information by TEPCO, simulations [9, 10] generally predict limited core degradation that is later quenched in-vessel. However, the amount of water that made its way to the core region remains a key uncertainty. Decreasing the water injection by half resulted in large-scale core relocation before the end of the simulation. If that simulation were extended, failure of the lower head and melt relocation would likely be predicted.

As noted earlier, due to limitations in time and available information, the simulation does not have Fukushima Daiichi Unit 3–specific models for predicting containment leakage or accurate reactor building modeling outside containment. This deficiency limits the usage of the model in predicting radionuclide and hydrogen transport out of containment.

IV.C. Areas Recommended for Future Work and Consideration

Modeling the ability of the suppression pool to condense steam released through an SRV for an extended period of time needs to be revised in the MELCOR model. The previous one-control-volume approach used in past and recent simulation work as well as the eight-control-volume approach used here both inadequately capture the SC thermal hydraulic phenomena and resulting containment pressure response seen in Unit 3.

The performance of the HPCI is key in predicting and understanding the possibility for core degradation early {29:49-42:38} in the accident. Additional analysis of the available data during this period, and higher fidelity modeling of the HPCI system, may prove fruitful.

The water injection details (timing and rate) by the fire engines, and possibly the DD-FP, remain as a key uncertainty. Higher fidelity modeling of the fire engine pumps, as previously attempted by SNL [9], and flow lines should be pursued. Forensic work to reduce the uncertainty related to the water injection information should also be pursued. The ability to model the accident beyond {42:38}, and predicting the amount of core degradation and final state, is limited by the uncertainties in the water injection.

There are a number of additional areas where modeling could be improved. Information from TEPCO could be used to more accurately model the core assemblies including the power profile, material masses, radionuclide (RN) inventories, etc. These core features have direct impact on the core melt progression, hydrogen generation, and RN release. The failure points of containment must also be modeled in a manner that reflects the Unit 3 containment and physical processes occurring. To the authors' knowledge, no models have predicted containment failure in Unit 3 on an a priori/ phenomenological basis.

Modeling of the reactor building, ventilation system, and vent lines should reflect the Unit 3 details and not Peach Bottom or another plant. The reactor building modeling impacts the hydrogen distribution and deflagration potential, radionuclide transport, and transportation to Unit 4. These are key phenomena and concerns for which MELCOR is designed to analyze.

There are additional areas where the MELCOR code refinements may be required in order to reproduce the accident progression at Unit 3. One example may be BWR core degradation modeling. Much of the previous experimentation and model work has focused on PWRs (e.g., >40 PWR fuel bundle degradation experiments vs. 9 BWR experiments). Another may be the impact of seawater on the accident progression, which is currently unknown and not accounted for in MELCOR.

Finally, the authors have spent much time reading reports, interacting with Japanese representatives, and conducting various analyses over the last year. While the accident provides numerous opportunities to learn from the plant response from a technological point of view, the human element of this accident should be equally understood. Besides the plant hardware, the operators were also faced with beyond design basis operation. The operator actions and observations are key in understanding and modeling the accident progression. The operator interviews provide a picture of the challenges faced and some of the heroic actions of the operators [12, 19].

ACKNOWLEDGMENTS

The foundation for this work was forged during the joint DOE-NE/NRC Fukushima Daiichi Accident Study.

REFERENCES

1. L. J. OTT, “Advanced Severe Accident Response Models for BWR Application,” *Nuclear Engineering and Design*, **115**, 289–303 (1989).

2. I. K. MADNI, “MELCOR Simulation of Long-Term Station Blackout at Peach Bottom,” Water Reactor Safety Research Information Meeting, Gaithersburg, MD, Oct. 22–24, 1990.

3. J. J. CARBAJO, “Severe Accident Source Term Characteristics for Selected Peach Bottom Sequences Predicted by the MELCOR Code,” NUREG/CR-5942, Oak Ridge National Laboratory, Oak Ridge, TN (July 1993).

4. M. W. FRANCIS, “Long-Term Station Blackout Sequence and Mitigation MELCOR Model,” M.S. Thesis, University of Tennessee, Knoxville (May 2006).

5. D. H. COOK, S. R. GREENE, R. M. HARRINGTON, S. A. HODGE, D. D. YUE, “Station Blackout at Browns Ferry Unit One – Accident Sequence Analysis,” NUREG/CR-2182, Vol. I, ORNL/NRUEG/TM-455/VI (Nov. 1981).

6. L. J. OTT, C. F. WEBER, C. R. HYMAN, “Station Blackout Calculations for Browns Ferry,” Water Reactor Safety Research Information Meeting, Gaithersburg, MD, October 22, 1985.

7. S. A. HODGE, L. J. OTT, “BWRSAR Calculations of Reactor Vessel Debris Pours for Peach Bottom Short-Term Station Blackout,” *Nuclear Engineering and Design*, **121**, 327–339 (1990).

8. I. K. MADNI, “Analysis of Long-Term Station Blackout without Automatic Depressurization at Peach Bottom Using MELCOR (Version 1.8),” NUREG/CR-5850, BNL-NUREG-52319 (May 1994).

9. R. O. GAUNTT et al., “Fukushima Daiichi Accident Study (Status as of April 2012),” SAND2012-6173, (July 2012).

10. Nuclear Emergency Response Headquarters, Government of Japan, “Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety – The Accident at TEPCO's Fukushima Nuclear Power Stations” (June 2011).

11. Nuclear Emergency Response Headquarters, Government of Japan, “Additional Report of Japanese Government to the IAEA - The Accident at TEPCO's Fukushima Nuclear Power Stations (Second Report)” (September 2011).

12. TEPCO press release, “Results of Our Investigation of the Accidents and Subsequent Development at Fukushima Daiichi Nuclear Power Station,” report submitted to JNES (December 22, 2011). http://210.250.7.21/en/press/corp-com/release/betu11_e/images/111222e16.pdf (English)

13. International Atomic Energy Agency (IAEA), Division of Nuclear Installation Safety, “IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami” (May 24–June 2, 2011).

14. Japan Nuclear Technology Institute (JANTI), “Review of Accident at Tokyo Electric Power Company Incorporated’s Fukushima Daiichi Nuclear Power Station and Proposed Countermeasures (Draft)” (October 2011).

15. Institute of Nuclear Power Operators (INPO), “Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station,” Revision 0, INPO 11-005 (November 2011).

16. American Nuclear Society, Special Committee on Fukushima, “Fukushima Daiichi: ANS Committee Report” (March 2012).

17. Nuclear and Industrial Safety Agency (NISA), “Technical Knowledge of the Accident at Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Co., Inc. (Provisonal Translation)” (March 2012). http://www.nisa.meti.go.jp/english/press/2012/06/en20120615-1-1.pdf

18. The Fukushima Nuclear Accident Independent Investigation Commission, “The official report of The Fukushima Nuclear Accident Independent Investigation Commission, Executive Summary,” The National Diet of Japan (July 2012).

19. TEPCO press release, “Measures Taken at Fukushima Daiichi Nuclear Power Station and Fukushima Daini Nuclear Power Station (December 2011 Edition) List of Documents” (Dec. 22, 2011). http://www.tepco.co.jp/en/press/corp-com/release/betu11_e/images/111222e18.pdf

20. TEPCO, "Unit 3, 4 Shift Supervisor Task Handover," English translation, accessed December 1, 2011. http://www.tepco.co.jp/en/nu/fukushima-np/plant-data/f1_4_Nisshi3_4.pdf.

21. Sandia National Laboratories, "State-of-the-Art Reactor Consequence Analyses Project, Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110 Vol. 1 (January 2012).

22. TEPCO, "Fukushima Daiichi Nuclear Power Station Unit3 Parameters of Water level and Pressure, accessed December 1, 2011. http://www.tepco.co.jp/en/nu/fukushima-np/f1/images/csv_level_pr_data_3u-e.csv.

23. TEPCO, "Fukushima Daiichi Nuclear Power Station Unit3 Parameters of Temperature," accessed December 1, 2011. http://www.tepco.co.jp/en/nu/fukushima-np/f1/images/csv_temp_data_3u-e.csv.

24. TEPCO, "Preliminary survey in Torus Room of Units 2 and 3, Fukushima Daiichi NPS," Photo and Videos Library, March 14, 2012. http://photo.tepco.co.jp/en/date/2012/201203-e/120314-01e.html

25. TEPCO, "Fukushima Daiichi Nuclear Power Station Unit 3 Reactor Building First Floor TIP Room Environment Investigation Result (May 23, 2012)," Photo and Videos Library, May 23, 2012. http://photo.tepco.co.jp/en/date/2012/201205-e/120524-01e.html

26. TEPCO, "Result of Water Level Measurement at Unit 2-3 Torus Room of Fukushima Daiichi Nuclear Power Station," Photo and Videos Library, June 7, 2012. http://photo.tepco.co.jp/en/date/2012/201206-e/120607_02e.html

27. D. H. COOK, "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, ORNL/TM-8906 (October 1984).

28. TEPCO, "Volume of water injected into the reactor of Unit 1 to Unit 3 of Fukushima Nuclear Power Station," May 31, 2011. http://www.tepco.co.jp/en/nu/fukushima-np/f1/images/110613sanko_table_tyusui-e.pdf.

29. S. R. GREENE, "Role of BWR secondary containments in severe accident mitigation: issues and insights from recent analyses," Workshop on Containment Integrity, Arlington, VA, June 14, 1988.