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SANDIA NATIONAL LABORATORIES' CONTRIBUTION TO THE OECD/NEA BSAF PHASE II PROJECT

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ABSTRACT

Over the past six years, the United States Nuclear Regulatory Commission (NRC) and Sandia National Laboratories (SNL) participated in both phases of the Organsation for Economic Collaboration and Development Nuclear Energy Agency's Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (OECD/NEA BSAF) project. Involvement in this project has been mutually beneficial to both Japan and the participating organizations. In return for insights that can be used to better decommission the Fukushima-Daiichi site, key information necessary for increased phenomenological understanding and the further development of severe accident analysis tools (e.g., MELCOR and MACCS) were provided. MELCOR code performance and runtime were enhanced, allowing three-week long core-damage simulations to be run in a few days – a significant improvement over previous capabilities.

The NRC and SNL, sponsored by the NRC, have participated in the BSAF project since its inception in 2012. The first phase of this analysis was completed in 2015 and published in the "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project) Phase I Summary Report." This report presented analyses focused on the reactor and containment for the first six days of the accident. However, close examination of the analyses indicated that it was necessary to include the reactor building in the model and expand the calculation duration from six days to three weeks to fully capture the atmospheric dispersion of the radionuclides. This was the focus of BSAF Phase II.

This paper provides highlights of the SNL contribution to the OECD/NEA BSAF Phase II final report, including an overview of all three accident scenarios modeled with MELCOR, resultant MACCS radionuclide dispersion and relevant code updates.

KEYWORDS
MELCOR, BSAF, severe accident analysis, Fukushima Daiichi

1. INTRODUCTION

This paper provides a description of the SNL contribution to the BSAF Project, which is a benchmark of state-of-the-art severe accident analysis software against the three accidents at Fukushima-Daiichi (1F). [1, 2] This paper also presents the results and associated key assumptions that that SNL team used to model the 1F severe accidents, their associated source terms and resultant deposition patterns. The description of any single unit analysis is not intended to be exhaustive, but instead is a highlight of key parameters that impacted the system behavior and resultant source term to the environment. First presented is an overview of the analysis method and models employed by the SNL team. Then each of the three unit analyses is presented, with attention to containment failures and system modeling. Finally, the results of deposition analysis is presented individually for each unit and combined for all three.

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2. ANALYSIS METHOD AND MODELS

2.1. Software Description

MELCOR [3, 4] is a fully integrated, engineering-level computer code with the primary purpose of modeling the progression of accidents in light water reactor nuclear power plants. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. MELCOR is developed by SNL under contract to the U.S. NRC. Analyses performed by SNL use MELCOR 2.2 official builds 10326 for Unit 1, 10721 for Unit 2 and 11085 for Unit 3. MELCOR 2.2 is a significant official release of the MELCOR code with many new models and model improvements. More detailed information is found in “Quicklook Overview of Model Changes in MELCOR 2.2: Rev 6342 to Rev 9496” as well as the User Guide and Reference Manuals. [5] These changes made within MELCOR 2.2, relative to MELCOR 2.1 have made it possible for 500-hour long source term calculations of Fukushima-Daiichi Power Station to be performed in under 50 hours of computational time.

MELCOR Accident Consequence Code System (MACCS) [6] is a fully integrated, engineering-level computer code developed at SNL for the NRC. MACCS simulates the impact of severe accidents at nuclear power plants on the surrounding environment. Currently, the team is working to add the capability of supplementing MACCS with the atmospheric transport and dispersion models available in the National Oceanic and Atmospheric Administration’s (NOAA’s) HYSPLIT (HYbrid Single-Particle Lagrangian Integrated Trajectory) code. [7, 8, 9] This analysis supports this development effort.

2.2. Common Characteristics of Fukushima-Daiichi MELCOR Models

The Fukushima Daiichi Unit 1, Unit 2, and Unit 3 MELCOR models are based upon the models developed from the boiling water reactor (BWR) model BWR/4 Mk-I MELCOR, created for the NRC’s State of the Art Reactor Consequence Analysis (SOARCA) Peach Bottom Analysis project. [10] The models for each of the three Fukushima reactors simulated share the same fundamental design. Therefore, common characteristics of the three models are addressed once; this includes nodalization and connectivity within the reactor building, environment, core, containment and cavity.

The MELCOR reactor building model of each Fukushima unit includes a realistic representation of plant systems within each floor. The model is based on the Peach Bottom SOARCA input deck but takes into account differences in floor geometry and connectivity. [10] These differences are accounted for in the definition of flow paths, heat structures, control volumes and the external environment. There are five floors in the Fukushima reactor buildings. The space on each floor is modeled with large control volumes, as illustrated in Figure 1, which shows a vertical cross section of the reactor building.

The primary containment is subdivided into six distinct control volumes. The representative nodalization of this containment can be seen in Figure 3, which includes representative flow paths used in the model. The drywell floor is sub-divided into three regions for the purposes of modeling molten-core/concrete interactions. The first region (which receives core debris exiting the reactor vessel) corresponds to the reactor pedestal and sump floor areas (CAV 0). Debris that accumulates in the pedestal can flow out into the second region (through an open doorway in the pedestal wall), corresponding to a 90-degree sector of the annular portion of the drywell floor (CAV 1). If sufficient debris accumulates in this region, it can spread further into the third region, which represents the remaining portion of the drywell floor (CAV 2).

MELCOR discretizes the core region, both axially and radially, into different regions. Within the core these regions are modeled with both control volumes and core cells. The control volumes are used to represent thermal-hydraulic phenomena while the core cells represent the fuel, clad, gap, structures (control blades, canisters, guide tubes, tie plates, etc.), and eventually, debris in the RPV. All models represented the reactor core region using five radial rings. In the MELCOR model, there are also five separate stacks of control volumes for each ring. This is considered a best practice when modeling with MELCOR; however, MELCOR can use a higher or lower number of radial rings. Axially, this MELCOR core cell model uses seventeen levels to represent the full reactor core. Within these core cells, component structures that represent both intact and subsequently degraded structures (including fuel, support structures and control blades) are defined by the user. This level of discretization is consistent with the Fukushima models developed by SNL for ongoing Fukushima analysis activities, including BSAF. These models are in turn based on those developed as part of the State-of-the-Art Reactor Consequence Analysis (SOARCA) for the Peach Bottom Atomic Power Station, which is also a BWR with a Mark-1 containment. [1, 10]

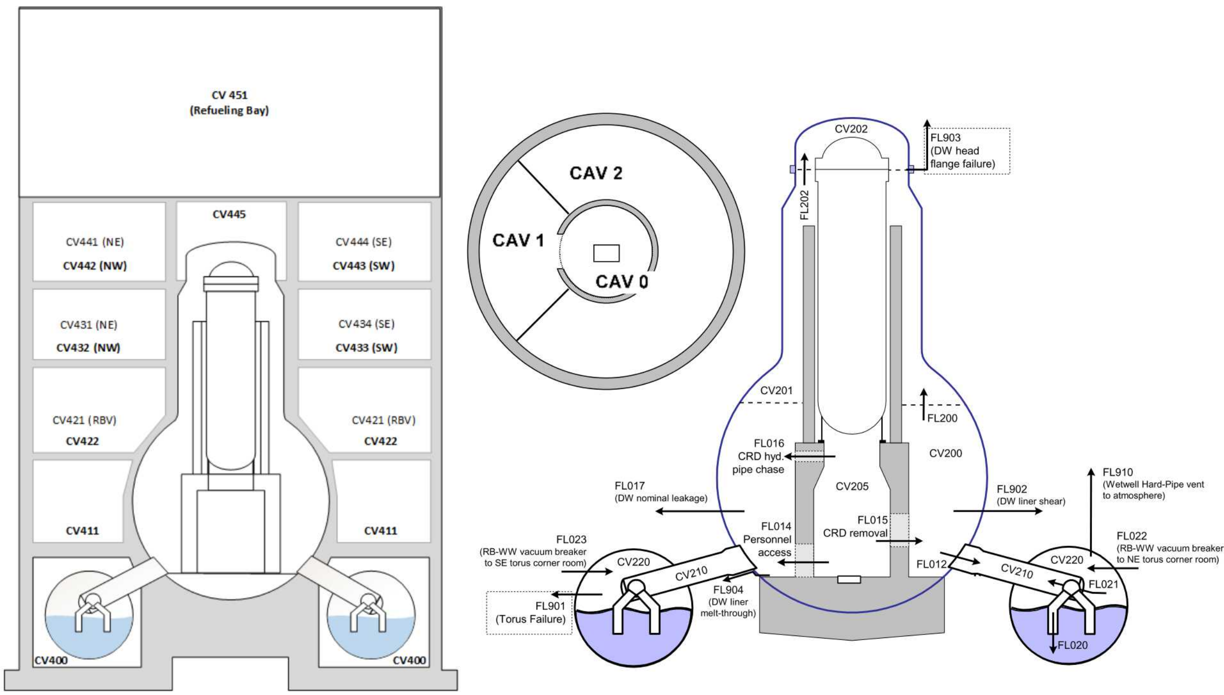

Figure 1. Control Volumes of Reactor Building Model (left) Primary Containment Nodalization (right) and Drywell Floor Regions for Modeling Molten-Core/Concrete Interactions (center)

The reported net injection amounts for each 1F unit sequence were used to create a maximum allowable injection rate. Each of the daily integral injection amounts were used to calculate an average injection flow rate for the model over the same twenty-four-hour period. This flow rate acted as the maximum rate of injection allowed by the model. The rates were bridged between any two consecutive days with a minimum of thirty-seconds to a maximum of three hours of transition time and a linear change in baseline mass flow from the previous day to the next. On top of the baseline injection rates, multipliers between zero and one were applied as deemed necessary and sufficient to follow data. An additional, automatic scaling mechanism that limited the injection rate as a function of RPV pressure is also present. A smooth cubic between two setpoints effected a cutoff to injection above 1.1 MPa and allowed the maximum injection flow rate below 0.6 MPa. This scaling was found necessary to arrest pressure excursions from injection-generated steam during times when injection was physically unreasonable for fire truck injection.

MELCOR's default combustion model is active in all primary and containment control volumes in the model as a function of hydrogen mass fraction. Depending on the orientation of the burn's propagation and moisture content in the air, burns could be induced with a minimum hydrogen mass fraction in a control volume of between 4% and 9%. All reactor building control volumes have burn activation triggered by user-specified time to ensure no burns prior to the known explosion time of the 1F1 and 1F3 refueling bays. At the reported time of the explosion in the 1F3 Reactor building, the MELCOR model has more than 10% of its volume composed of hydrogen by mass. The explosion itself is modelled through a user-specified time and large flow paths opening to the environment on compromised floors.

2.3. Analysis Methodology and Characteristics

In the second phase of the BSAF project, SNL employed what is often called a “forensic approach” and modified boundary conditions and unknown plant parameters to better match plant response and known radionuclide deposition patterns. This led to the high level of agreement seen between the MELCOR simulation and the plant data. Such analysis is very useful in determining what happened during a severe accident when key indicators of progression are known from forensic examination. However, such close agreement between data and simulation would not be obtained for forward MELCOR calculation without such free boundary conditions. In accordance with this forensic approach, deviations were made from typical MELCOR best practices and defaults. Given the overview nature of this paper, these deviations are not discussed for the sake of brevity. Future publications, based on each unit calculation, for the BSAF Phase II analyses will address all such deviations.

3. UNIT 1 ANALYSIS

3.1. PCV and RPV Leakage Pathways

Several leaks between reactor components were modelled to capture the real-world state of the accident. The most important leak paths are the drywell head flange to the undershield plug, undershield plug to the refueling bay, recirculation pumps to the drywell and design leakage from the drywell to Floor 1 of the reactor building. During the initial period when the RPV is at a high pressure, the recirculation pumps have a direct leak to the drywell driven by the pressure difference between the steam dome and the drywell. The leak rate is approximately 10 gallons per minute, and the flashing of the water into the containment acts as a source of pressurization in the drywell. A design leak path is present between the drywell and Floor 1 of the reactor building. The design leak flow area has a nominal value of $5.7 \cdot 10^{-6}$ m^2 and is maintained throughout the 1F1 simulation.

The drywell head flange leakage model is a function of the differential pressure between the drywell and the atmosphere based of the methodology from the Peach Bottom Integrated SOARCA Analysis. [10] The flange begins to open with a pressure differential of 550 kPa between drywell and environment and is considered 100% open at a 1.8 MPa differential. The maximum flow area is taken to be 0.0808 m^2. This regime of drywell head flange leakage is present after the failure of the lower head until the point of containment venting. A second period of drywell head flange leakage then occurs at a lower set point of 0.6 MPa, extending from 30.8 hr to 50.5 hr.

A final leak pathway from the drywell to the torus room is opened at 50.5 hr to simulate ablation of base mat concrete deep enough to reach the drywell liner. This assumption based upon photographic evidence which shows water draining out of the sand cushion below the drywell of 1F1 and the relatively small amount of injected water into 1F1 before 11 days after the accident commenced. This leak pathway is modeled as a valve which is initially sized at 0.0002 m^2 and expands to 0.0003 m^2 at 70.1 hr and then to 0.0008 m^2 at 240.2 hr to simulate the initial breach expanding over time.

3.2. System Behavior

The Isolation Condenser system was operated to maintain RPV pressure below the SRV lifting set point during the first hour of the transient before the arrival of the tsunami and the loss of the system. This was the only engineered safety system in the 1F1 MELCOR model, the heat removal is a fixed boundary condition each assumed to be constant with RPV pressure at 42.4 MW per train.

Between of 5.3 hr to 10.3 hr into the accident, 1F1's RPV experienced a depressurization to the range of the drywell pressure. The drywell experienced a pressurization over the same time frame. This event is modelled in MELCOR through a MSL rupture on steamline A that was opened according to a user-specified time table. This assumption is consistent with the insights from the MELCOR analyses performed in the SOARCA Peach Bottom Analysis, [10] which found that as gasses and steam are heated within the core region they raise the temperature of the steam lines causing them to fail due to creep rupture. The break had a size of 0.10406 m^2 at 6.1 hr into the accident scenario. Primary system depressurization resulted in increased pressure in the drywell, however, the increased pressure was less than the maximum observed at 0.8 MPa, even after the core has collapsed and the lower head has failed. It is believed that this difference in drywell pressurization is due to a decreased total amount of hydrogen generated in-vessel during the simulation.

Based on the forensic evidence obtained from entries into 1F1 by Japanese partners, it was assumed all of the fuel relocated ex-vessel. Accordingly, fuel rings were failed at user-specified times to best match reported pressure data. By doing this a more accurate source term is obtained for use in atmospheric dispersion software. The fuel was relocated at two separate timings. At 9 hr, rings one through three were relocated to the lower plenum. Then at 10 hr, rings four and five were relocated to the lower plenum. These relocations were followed by an enforced failure of the lower head at 10.5 hr. These events were timed within short order of one another to push the pressure of the drywell to near the highest reported values of 0.8 MPa near 12.0 hours into the event.

A single wetwell venting operation was modelled for 1F1 at 23.75 hr into the simulation. The vent was achieved through opening a valve on a flow path from the wetwell to the environment through a vertical stack.

3.3. MELCOR Results

System response behavior for Unit 1 is shown in Figure 2 and Figure 3 for the RPV and DW respectively. Key event timings for all units are presented in Table I.

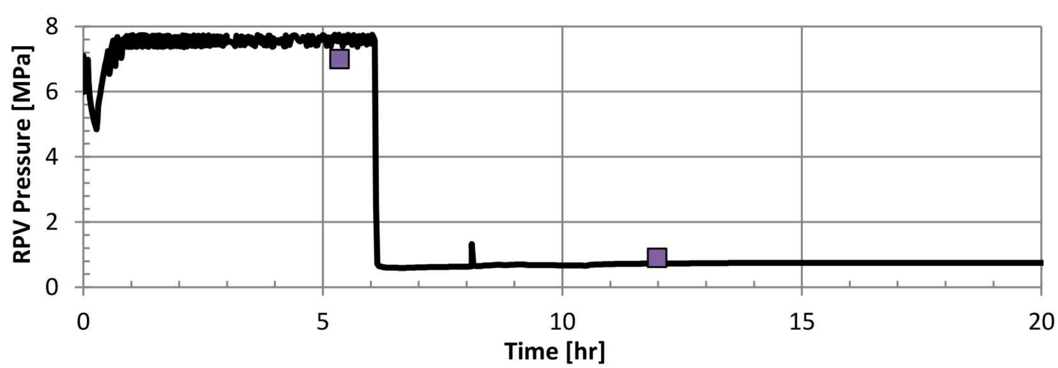

Figure 2: RPV pressure for 1F1 analysis showing TEPCO data and MELCOR results.

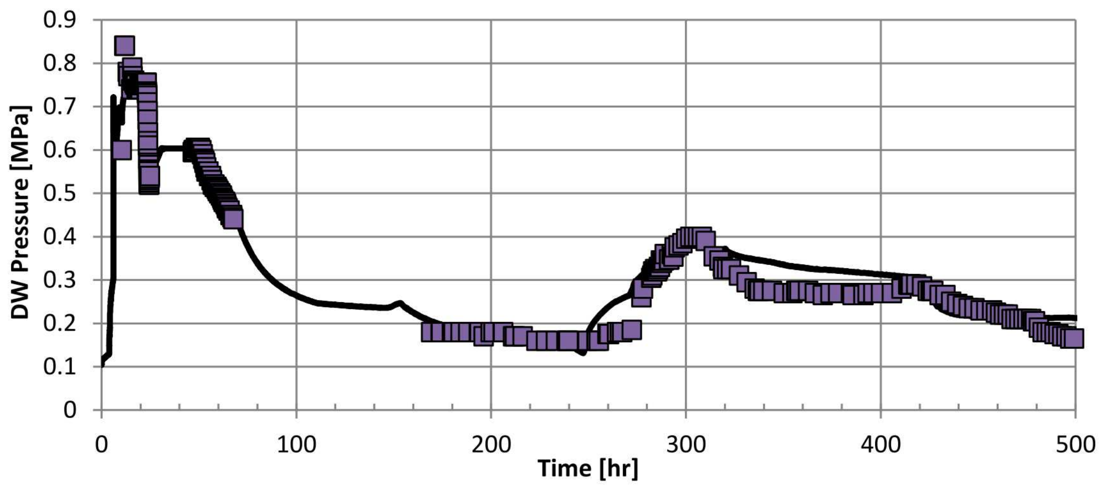

Figure 3: DW pressure for 1F1 analysis showing TEPCO data and MELCOR results

Table 1. Key Event Timing for Fukushima-Daiichi MELCOR Accident Analysis

Event	Time [hours]		
	Unit	Unit 2	Unit 3
First occurrence of water level at TAF	2.5	75.2	35.9
Onset of hydrogen generation	3.4	76.5	41.6
First fuel clad failure time	3.5	76.4	43.2
First control blade failure	3.7	77.4	43.2
First fuel rod failure time	3.9	79.9	43.2
First UO2 relocation to lower head	4.0	79.9	43.2
First lower core plate failure time	8.8	79.9	43.2
First FP release from fuel	3.5	76.6	41.8
First FP release to environment	3.9	79.9	42.1
First RPV pressure boundary failure	6.1	78.51	58.1
Time of lower head failure	10.5	138 (gross failure)	58.1
Time of initiation of MCCI	10.9	96.7 (side penetration)	58.2
Time of initial containment failure	50.5	92.0	42.1
Time of hydrogen burn	24.8	-	68.2

4. UNIT 2 ANALYSIS

4.1. RPV and PCV Leakage Pathways

As in the case of 1F1 several leaks between reactor components were modelled to capture the real-world state of the accident. The most important leak paths are the drywell head flange to Floor 1 of the reactor building following the same path as 1F1, and a small leak from the RPV to the drywell, which opens during core degradation. During RCIC operation, while the reactor is at high pressure, the recirculation pumps have a direct leak to the drywell driven by the pressure difference between the steam dome and the drywell. The leak rate is approximately 14 gallons per minute, and the flashing of the water into the containment acts as the primary pressurization source in the drywell during RCIC operation. The design leak path present between the drywell and Floor 1 of the reactor building has a nominal leak flow area of

$5.7 \cdot 10^{-6}$ m^2. This nominal value is maintained throughout the 1F2 simulation. Leakage from the drywell head flange is modeled as in 1F1. Gross failure of the drywell head flange is modeled as the key containment failure in 1F2, occurring at 92.0 hr, corresponding to the required wind pattern for the large plume that extends to the northwest of the 1F site.

A final leak is opened at seventy-eight hours into the simulation following the re-opening of an SRV during core degradation. This leak path was opened to match trends in the data showing direct pressure communication between the RPV and the drywell with no change in the wetwell pressure. Specifically, there is a jump in the drywell pressure at seventy-eight hours and then a linear increase from eighty to eighty-one hours during the accident. These changes in drywell pressure track well with pressurizations and depressurizations seen in the RPV without significant changes in the wetwell pressure. To model these pressure trends, a leak from Loop A of the MSL to the drywell was opened at 78.51 hours into the simulation with a flow area of $6 \cdot 10^{-6}$ m^2.

4.2. RCIC Modeling

The MELCOR 1F2 RCIC system is a thermodynamic, control volume-based model that relies on vendor data for normal operation conditions with corrections for MSL flooding through calibration to data. Briefly, the model calculates a turbine steam mass flow rate from a choked flow model with governor contractions while power is available to the plant. That mass flow rate determines the maximum available RCIC pump power, and the RCIC pump power is used to determine an injection mass flow rate. The model is calibrated to reproduce expected steam extraction and water injection flow rates when the plant has power available. Following loss of power, the governor contractions are removed, and a flow area equivalent to the estimated governor's flow area at the low pressure operating setpoint is made available to the RCIC. The piping flow area is plant-specific data while the flow area of the governor at the high and low pressure points is estimated from plant-specific turbine flow rates and operating pressures compared to the local sonic steam mass flow rate. The loss of flow restriction increases the extraction and injection flow rates to a value that quickly raises the RPV water level to the MSL, which leads to a two-phase flood of the RCIC turbine.

Flooded operation of the RCIC turbine is handled via a thermodynamic calculation that is presented in Figure 4. The reactor system component abbreviations are as follows: WW is the wetwell, MSL is the main steam line, FW is the feedwater line, and CST is the condensate storage tank. The isentropic efficiency $\eta_T)_s$ is specified and taken to have a nominal value of 0.24 for these results. The void efficiency parameters α_{crit}, η_{crit}, and $\eta_{\min}$ are informed from data and have the values of 0.9, 0.55, and 0.1 for these results, respectively. The pressure loss ΔP_{loss} from the turbine exit to the wetwell exhaust is assumed to be a constant 50 kPa for these results, as well.

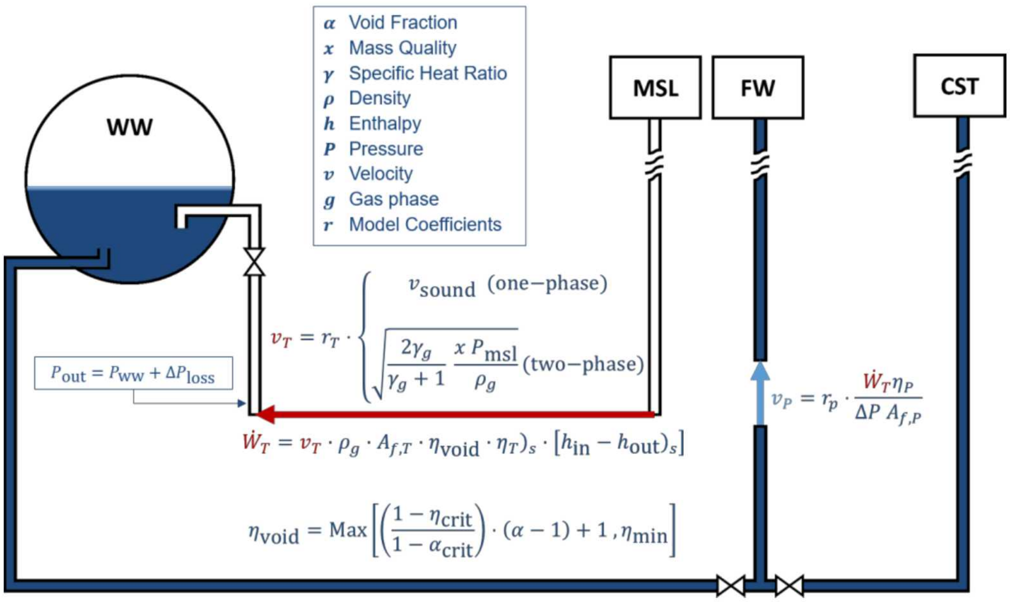

Figure 4: Diagram of the 1F2 RCIC system.

4.3. Three Peaks Modeling

Following the depressurization of 1F2 to facilitate firetruck injection, there were three noticeable peaks in system pressure from about seventy-seven hours to eighty-two hours into the accident. These peaks are modeled as the result of water injection, associated hydrogen generation, fuel collapse, and valve failures that didn't perfectly reseat. The water injection is described earlier, and the hydrogen generation is accounted for by MELCOR's default oxidation model.

Following the system depressurization, the fully open valve was assumed to transition from 100% open to 0.08% open, and there were no more perfect SRV closures for the rest of the simulation. The open area fractions were assumed to respond to evolutions in the system pressure and be dynamic in time but were never taken to be zero. The first peak is the result of accelerated hydrogen production, which cases the system pressure increases more rapidly and is ultimately abated by the operators opening two SRVs, which is modeled as two SRVs fully opening.

For the second peak, the rapid increase in system pressure seen at eighty-one hours is modeled as Ring 1 collapsing and relocating to the lower plenum. The debris in the lower plenum then undergoes quenching, releasing steam. The non-zero valve open fraction allows most of the steam to be driven through the wetwell that creates a modest, linear pressurization in the RPV during the second peak. An SRV that was fully opened by operators and increased injection brought the system pressure back below 1 MPa. The third peak was assumed to be another incomplete valve closure of 1.5% of its maximum flow area. The absence of any fuel collapse resulted in a steady pressurization from steam and oxidation prior to depressurization from operator action.

4.4. Core Degradation and Lower Head Failure

Aside from the collapse of Ring 1 to capture the pressure increase for the second peak, no other fuel collapse was specified. At 83.0 hours, a side penetration was opened on Ring 4 of the lower plenum that allowed particulate debris to relocate ex-vessel to the cavity without gross lower head failure. Without this side penetration, MELCOR predicted lower head failure at ninety-two hours. With this side penetration and relocation of approximately 5% of all core material relocated ex-vessel, lower head failure was still predicted at 138 hours into the accident. While MELCOR still predicts gross failure of the lower head despite evidence that this did not happen in unit 2, further development and investigation into the side penetration model for partial relocation does delay and may even prevent such a failure if enough core material is relocated.

4.5. MELCOR Results

System response behavior for Unit 2 is shown in Figure 5 for the first 100 hours of the simulation. The period of the three peaks is shown in Figure 6. DW response for the full 500 hours of simulation time is shown in Figure 7.

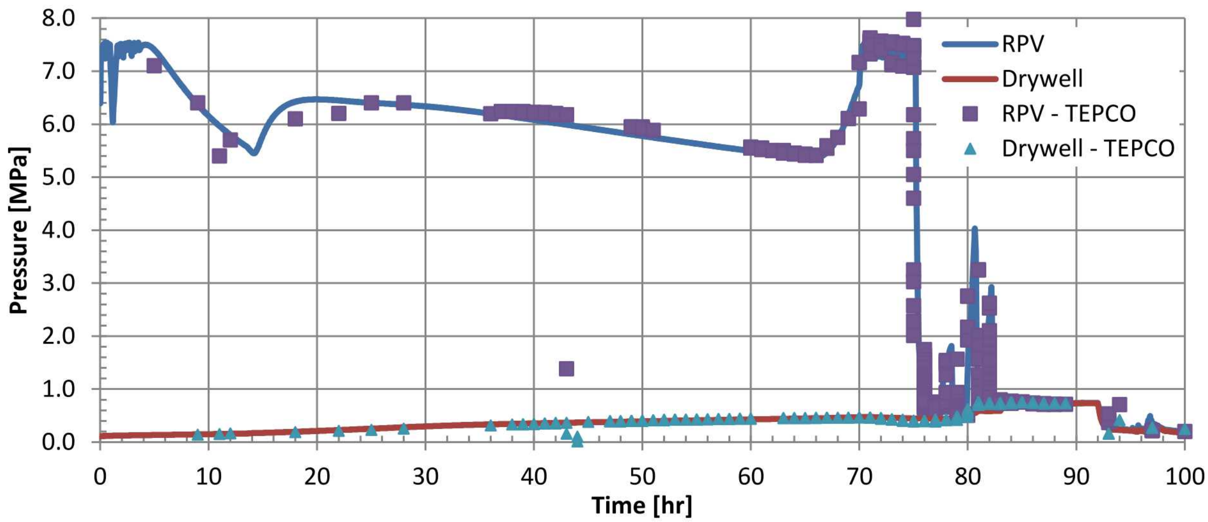

Figure 5. System response behavior for Unit 2 for the first 100 hours of simulation time showing MELCOR results and TEPCO plant data.

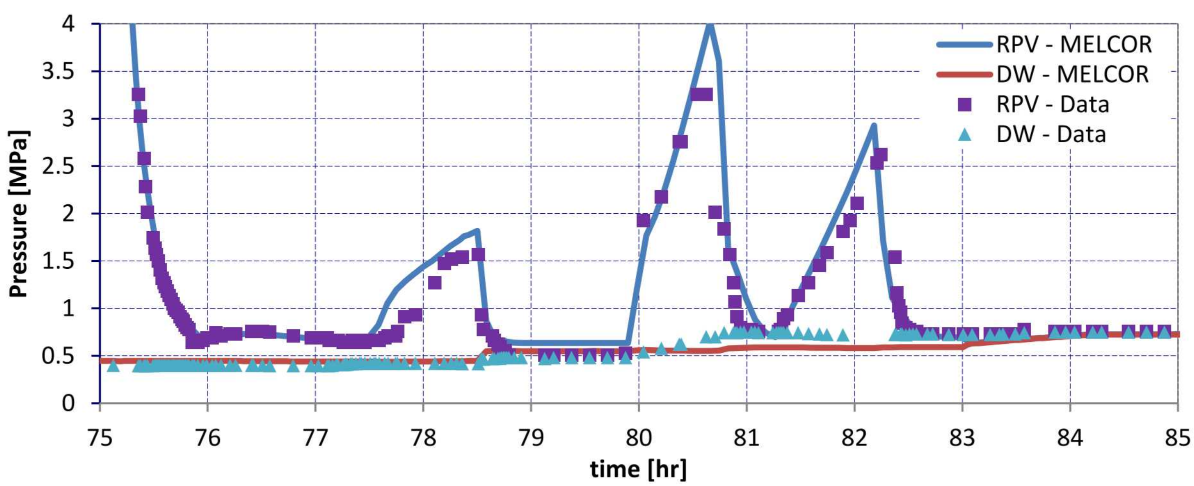

Figure 6. System response behavior for Unit 3 during the "three peaks" time period showing MELCOR results and TEPCO plant data.

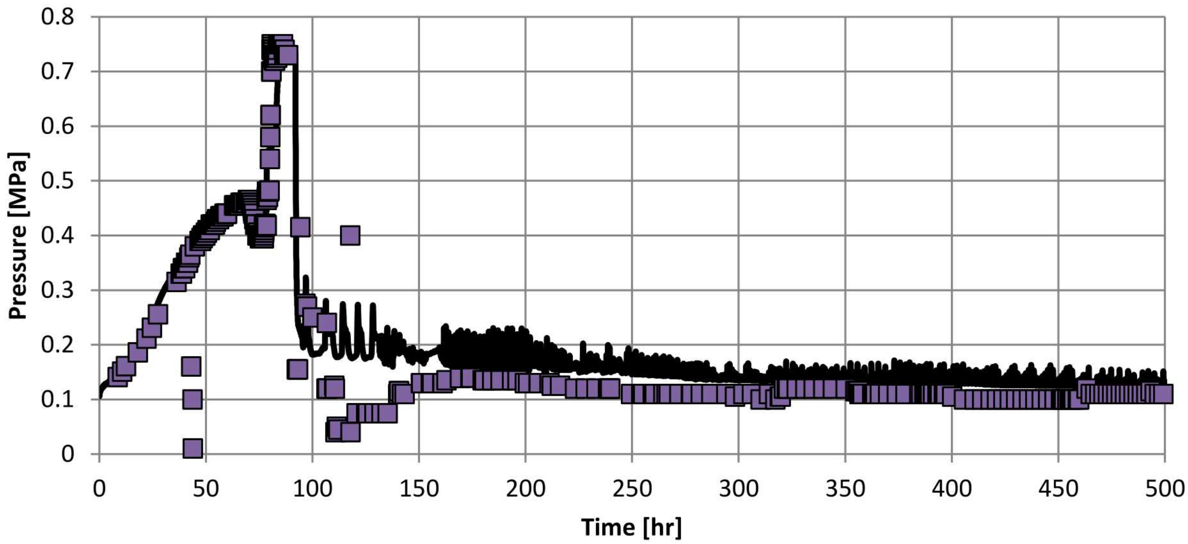

Figure 5. System response behavior for Unit 2 for the full 500 hours of simulation time showing MELCOR results and TEPCO plant data.

5. UNIT 3 ANALYSIS

5.1. RPV and PCV Leakage Pathways

Several leaks between reactor components were modelled to capture the real-world state of the accident. The most important leak paths are the drywell head flange to the undershield plug, undershield plug to the refueling bay, recirculation pumps to the drywell, and drywell to Floor 1 of the reactor building.
The drywell head flange had a leakage model as a function of the differential pressure between the drywell and the atmosphere based of the methodology from the Peach Bottom Integrated SOARCA Analysis. [10] The flange begins to open with a pressure differential of 550 kPa between drywell and environment and is considered 100% open at a 1.8 MPa differential. The maximum flow area is taken to be 0.0808 m^2. Following the MSLB break, a permanent open area of 0.01% of the maximum allowable flow area was added to the default model to approximate damage to the flange from the pressure excursion.

During RCIC operation while the reactor is at high pressure, the recirculation pumps have a direct leak to the drywell driven by the pressure difference between the steam dome and the drywell. The leak rate is approximately 20 gallons per minute, and the flashing of the water into the containment acts as the primary pressurization source in the drywell during RCIC operation. At the start of HPCI operation, the RCIC leak becomes small due to the precipitous drop in reactor pressure, which lowers the leak rate and associated drywell pressurization.

A design leak path is present between the drywell and Floor 1 of the reactor building. Prior to the MSL break, the design leak flow area has a nominal value of $5.7 \cdot 10^{-6}$ m^2. Following the MSL break, the leakage area is increased to $5.7 \cdot 10^{-4}$ m^2 to represent acute damage to the leak area from the event. The leakage area is further increased at 238 hours to 10^{-3} m^2 and at 280 hours to 10^{-2} m^2 to represent long-term damage to the leak areas from temperature and pressure loads in addition to extreme cavity flooding from extended injection.

5.2. System Behavior

From 42 hr to 42.5 hr into the accident, 1F3's RPV experienced rapid depressurization to the range of the drywell pressure. The drywell experienced a rapid pressurization over the same time frame. This event is modelled in MELCOR through a MSL rupture on steamline B that was opened according to a user-specified time table. The break had a maximum allowable size of 0.046 m2 to achieve the depressurization rate seen in the RPV data. However, the break was not opened instantaneously. Assuming the break started small and grew larger in time, the break was augmented with at non-zero, finite time scale. At 41 hours into the accident, the break was opened from zero to 0.01% of its maximum size over thirty seconds, then linearly grew to 0.1% over the following hour, and then to 1% 180 seconds after that. At 42.15 hr into the accident, the break size grew rapidly. Over five consecutive time spans of 3.6 seconds, the break size was increased from 1% to 10%, 50%, 85%, 90%, and 100% of its maximum value.

Two wetwell venting operations were modelled for 1F3: one at 42.3 hr just after the MSL rupture and one at 45.73 hr just after the assumed core ring relocation to the lower plenum. The vent was achieved through opening a valve on a flow path from the wetwell to the environment through a vertical stack. During the second vent, a leakage area of 0.005 m2 was opened to the first floor as a surrogate for non-ideal vents that would leak hydrogen into the reactor building during the vent.

Since 1F3 had on-site power for an appreciable amount of time, the RCIC and HPCI systems were either operated automatically or with direct operator control and did not experience MSL flooding as in the independent RCIC operation of 1F2. Therefore, the models are markedly simpler than in the 1F2 model. The 1F3 RCIC system activates when the RPV swollen water level falls below a low water level setpoint of 4.2 m above TAF. Upon activation, RCIC injects water at a plant-specific maximum flow rate until the RPV swollen water level reaches or exceeds a high-water level setpoint of 4.5 m above TAF. To avoid flooding the MSL, the RCIC injection rate is reduced to a quarter of the maximum flow rate until the water level falls below the low water level setpoint again when injection is reset to the maximum. This injection rate cycling procedure continues for the duration of RCIC operation to maintain system pressure and water level within the windows provided by data. The extraction mass flow rate and associated turbine work is taken from plant-specific data as a function of RPV pressure.

The HPCI system has more detail than the RCIC since it was under manual control, had a diversion of injection through the HPCI test line, and depressurized the RPV over a large six megapascal range. A nominal HPCI extraction and injection rate were obtained from scaling plant-specific RCIC data, both rates and pressures, by the ratio of pipe flow areas of the systems. The scaled extraction rate was used to estimate the necessary governor restrictions on the turbine inlet given a sonic inlet velocity at the operating pressures. This allowed the construction of a table of sonic mass flow rates over the range of pressures observed during HPCI operation to better track the manual corrections made during the depressurization process. For the HPCI turbine power output and pump power, a table of the isentropic enthalpy drop across the turbine from saturated steam to an exhaust pressure of 1 atmosphere over the range of observed pressures was constructed. A user-supplied isentropic efficiency for both the turbine and the pump were used to convert the ideal exhaust state of the turbine and the ideal pump power to non-ideal values. The pump power was used to calculate a nominal injection flow rate. The actual injection rate was a multiple of the nominal flow rate with a scaling factor that reflected manual diversion of injection water through the HPCI test line.

Following the RPV depressurization, the 1F3 data showed one small spike and one large spike in the drywell pressure history at 43.3 hours and 45.4 hours, respectively. The first spike was modelled as a user-specified collapse of ring 1 in the core, and the second spike as a user-specified collapse of rings 2 and 3. Rings 4 and 5 both collapsed at 124 hours into the accident according to MELCOR's default failure rules. The lower head was failed through a user-defined function at 58.1 hr, this point corresponds to a rapid increase in pressure in the reported data.

5.3. MELCOR Results

System response behavior for Unit 3 is show in Figure 8 and Figure 9 for the RPV and DW pressures respectively.

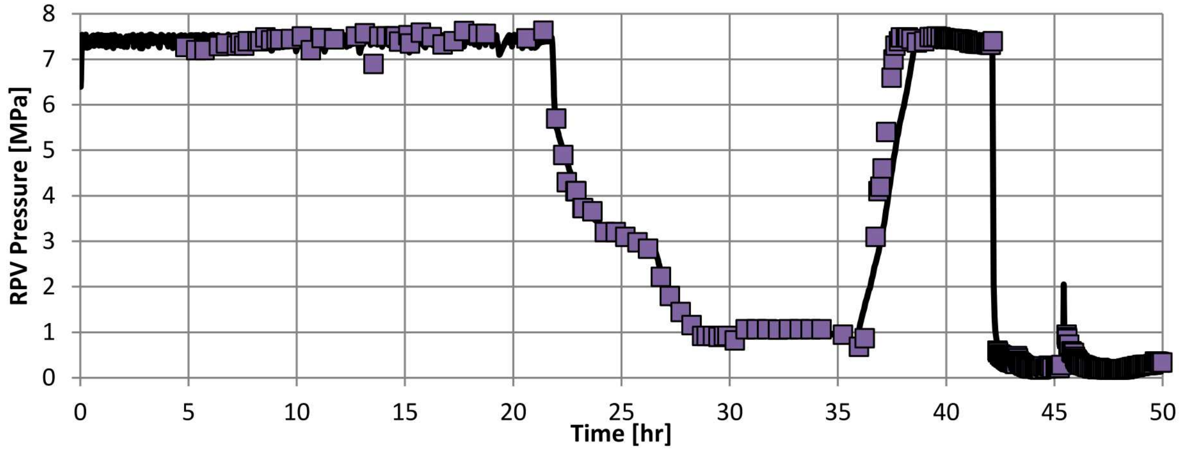

Figure 8. RPV pressure for Unit 3 showing MELCOR results and TEPCO plant data.

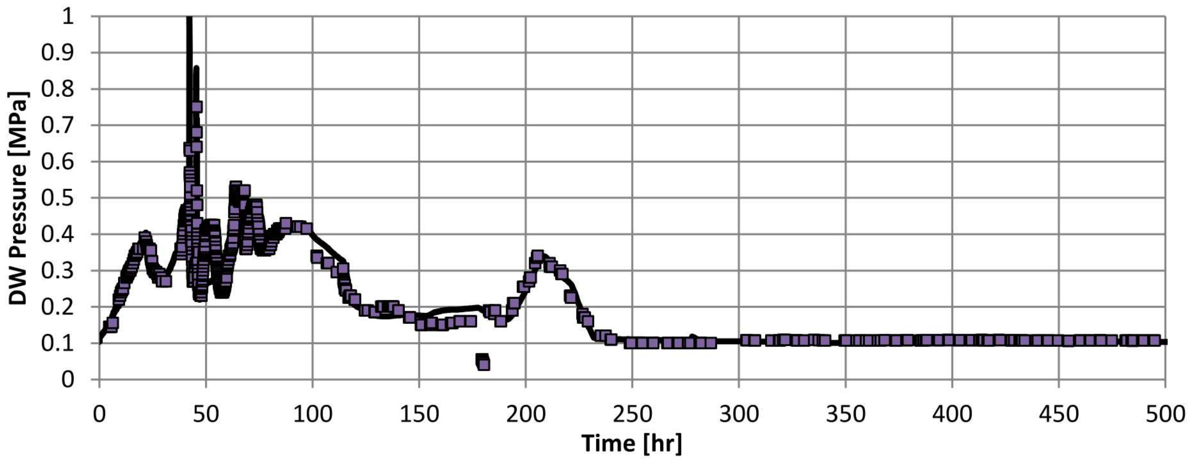

Figure 9. DW Pressure for Unit 3 showing MELCOR results and TEPCO plant data.

6. DEPOSITION ANALYSIS

HYSPLIT [7, 8, 9, 11] was applied to the three unit source terms to treat atmospheric transport and dispersion. The objective was to estimate the deposition patterns that would have resulted from the predicted source terms. The overall deposition pattern can be compared with the observed deposition pattern. Atmospheric transport calculations were performed for a single isotope, Cs-137. It is the primary isotope of concern for long-term contamination, and it is relatively easy to measure the strong gamma signal produced from its short-lived decay product, Ba-137m.

All the atmospheric transport calculations used the actual location of each of the three units; the releases were not presumed to emanate from the same location. Furthermore, plume lofting was considered by accounting for the release energies determined in the analysis. Finally, aerosol size distribution data were considered for purposes of estimating deposition. In some cases, aerosol size distribution can significantly influence deposition patterns. For these analyses, the weather data was taken from the Weather Research and Forecasting (WRF) 2017 dataset provided by NOAA Air Research Laboratory (ARL) of the Fukushima-Daiichi accident. [12] The WRF 2017 dataset was selected as it has been nudged by observational data. The deposition results are shown in Figure 10 and can be compared to the observed deposition data shown in Figure 11. It is shown that Units 2 & 3 contribute more than Unit 1 to observed deposition. The deposition pattern obtained from HYSPLIT has more along the northern coast compared with the observed.

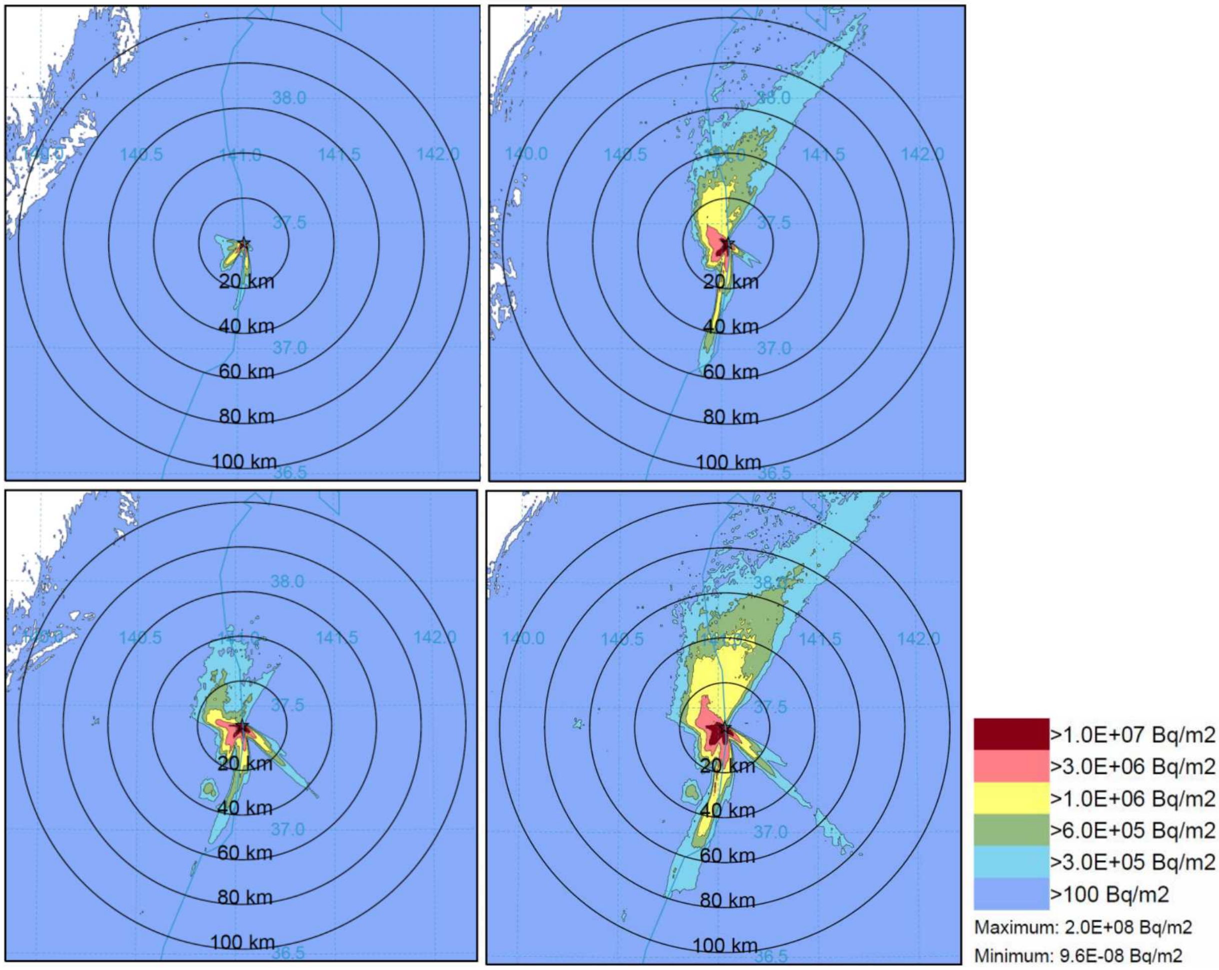

Figure 10: HYSPLIT MACCS deposition results of Unit 1 (top left), Unit 2 (top right), Unit 3 (bottom left) and combined deposition (bottom right). (Bq/m^2, integrated from Mar. 11-21, 2011.)

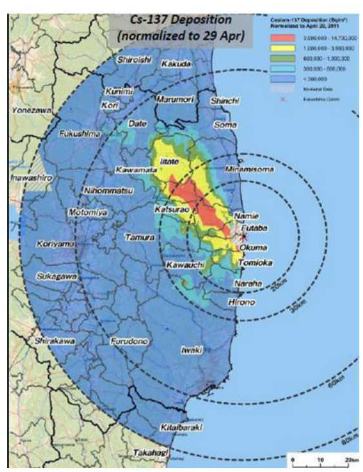

Figure 11. Observed ground deposition pattern for Cs-137

7. SUMMARY AND CONCLUSION

The BSAF project has been the most significant assessment of severe accident code capabilities to date, providing benchmarks of all major severe accident software against three well-documented core damage accidents at the Fukushima-Daiichi site. As information has become available through forensic evaluations and engineering analysis, state-of-the-art of severe accident modeling practices have significantly improved. This can be seen in the development of new phenomenological models and code performance enhancements, which have enabled participants in the BSAF project to model the three weeks following the initiation of the accident, which was not possible at the beginning of the project. The project has led to significant improvement within the MELCOR and MACCS frameworks and has proven to be invaluable in planning for future development activities.

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REFERENCES

1. M. Pellegrini, et al., "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF), Phase I, Final Report, (2015)
2. OECD/NEA, "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project) – Phase I Summary Report, March 2015." NEA/CSNI/R(2015)18 (2016)
3. "MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.2.9541," SAND 2017-0455 O, Sandia National Laboratories, January 2017 (ADAMS Accession No. ML17040A429).
4. "MELCOR Computer Code Manuals, Vol. 2: Reference Manual, Version 2.2.9541," SAND 2017-0876 O, Sandia National Laboratories, January 2017 (ADAMS Accession No. ML17040A420).
5. Humphries, L., Quicklook Overview of Model Changes in MELCOR 2.2 Rev 6342 to Rev 9496," Sandia National Laboratories, Albuquerque, NM, 2017.
6. Chanin, D., and Young, "Code Manual for MACCS2: Volume 1, User's Guide," Sandia National Laboratories, May 1998.
7. Draxler, R.R., and G.D. Hess, 1997: Description of the HYSPLIT_4 modeling system. NOAA Tech. Memo. ERL ARL-224, NOAA Air Resources Laboratory, Silver Spring, MD.
8. Draxler, R.R., and G.D. Hess, 1998: An overview of the HYSPLIT_4 modeling system of trajectories, dispersion, and deposition. Aust. Meteor. Mag., 47, 295-308.
9. Draxler, R.R., 1999: HYSPLIT4 user's guide. NOAA Tech. Memo. ERL ARL-230, NOAA Air Resources Laboratory, Silver Spring, MD.
10. Bixler, N., Gauntt, R., Jones, J. and Leonard, M., "State-of-the-Art Reactor Consequence Analyses Project Volume 1: Peach Bottom Integrated Analysis (NUREG/CR-7110, Volume 1, Revision 1), NUREG/CR-7110, Washington, DC, May 2013.
11. National Oceanic and Atmospheric Administration, Air Resources Laboratory, HYSPLIT User Guide retrieved from: http://www.arl.noaa.gov/documents/reports/hysplit_user_guide.pdf, December 22, 2016.
12. National Oceanic and Atmospheric Administration, Air Resources Laboratory, Weather Data Archive retrieved from: ftp://arlftp.arlhq.noaa.gov/archives/, November 2017.