MELCOR Simulations of the Severe Accident at the Fukushima 1F2 Reactor

Jesse Phillips, Jeffrey Cardoni, Randall Gauntt, Donald Kalinich

Sandia National Laboratories: Albuquerque, NM 87185
Tel: (505) 844-8375; Fax: (505) 844-2829; Email:jncardo@sandia.gov

In response to the accident at the Fukushima Daiichi nuclear power station in Japan, the US Nuclear Regulatory Commission and Department of Energy agreed to jointly sponsor an accident reconstruction study as a means of the assessing severe accident modeling capability of the MELCOR code. Objectives of the project included reconstruction of the accident progressions using computer models and accident data, and validation of the MELCOR code and the Fukushima models against plant data.

I. INTRODUCTION

The MELCOR model for the 1F2 reactor simulates the thermal-hydraulic response of the reactor coolant system, primary containment, and the reactor building; this includes models for operator actions that affect the thermal-hydraulics of the plant (e.g. valve operation and seawater injection). The model calculates the decay heat generation in the core, fuel uncovering and heatup, core oxidation, hydrogen generation, core damage and relocation, and the degradation of support structures in the vessel. If MELCOR predicts vessel lower head failure, models exist to simulate the corium attack on concrete in the containment. Integral in this effort is the capability to calculate the fission product release from the core as it degrades and relocates, and to predict the transport of radionuclides throughout the entire plant. Ultimately, MELCOR estimates the fission product release to the environment from the severe accident at the 1F2 reactor. MELCOR can also predict hydrogen deflagrations that can damage plant structures and increase radionuclide releases to the environment.

The basic plant nodalization for the 1F2 model is adopted from an existing MELCOR model for the Peach Bottom plant, which was used for the NRC's State-of-the-Art Reactor Consequence Analysis (SOARCA). Peach Bottom and 1F2 are similar power plants – both are BWR/4 reactors with a Mark I containment design. However, there exist substantial differences in power rating (3514 MWt vs. 2381 MWt for 1F2), fuel assemblies and core loading, safety systems, and various dimensions of the RCS and containment. Most of the input for the 1F2 model was developed using plant data from TEPCO design documents. The model incorporates decay heat and power distribution information provided by TEPCO neutronics analyses. The model uses the latest known or assumed operator actions during the first four days of the accident.

The MELCOR 1F2 model simulates the first four days of the accident. Compared to the TEPCO pressure data, MELCOR appears to capture the essential features of the pressure response in the vessel and containment.

II. CODE BACKGROUND

MELCOR is a severe accident code developed by Sandia National Laboratories for the NRC. Its primary purpose is to simulate the evolution of accidents in light water nuclear reactors and to generate fission product source terms. MELCOR is composed of several different modules, called packages (which are fully integrated), that model the important phenomena that can occur during severe nuclear accidents. The thermal-hydraulic packages in MELCOR are based on the flexible use of control volumes, flow paths, and heat structures, which are assembled together in an appropriate manner to model the majority of the plant. Special models exist that are nuclear reactor-specific for simulating core damage phenomena, the attack on concrete by hot corium in the containment, and radionuclide behavior.1

Modern MELCOR models are typically comprised of a few hundred control volumes, flow paths, and heat structures. In the reactor vessel, the computational mesh of fuel assemblies and other core structures for the calculation of heat transfer, core damage, oxidation, and material relocation is separate from the hydrodynamic nodalization in MELCOR. The core package (COR) in MELCOR models the core region and the lower plenum in 2-D cylindrical (axisymmetric) geometry. A typical modern core and lower plenum model in MELCOR is represented by 50-200 core cells. A separate hydrodynamic nodalization of the core and lower vessel is coupled to these core cells in order to provide thermal boundary conditions to the COR package. The upper vessel region is usually treated using only the MELCOR hydrodynamic packages; hence without corresponding input in the COR package, the structures in the upper vessel do not oxidize or relocate.

III. MELCOR 2.1 MODEL FOR 1F2

III.A. Spatial nodalization and run-time

As summarized in Table I, the 1F2 model is a relatively high-fidelity MELCOR representation of the core and reactor coolant system (RCS), and it includes the latest best practices models for the primary containment, safety relief valves (SRVs), safety systems, and reactor building for a BWR/4 reactor with a Mark I containment. The core and lower plenum are divided into 5 rings and 17 axial levels, which includes 10 axial levels for the active fuel region (for 50 total fuel cells). There is also a 6th ring in axial levels 2-4 to model the lower plenum region that extends past the radial perimeter of the core region. In total, the 1F2 MELCOR model is comprised of 88 core cells, 146 control volumes, 358 flow paths, and 172 heat structures. Nodalization diagrams of the core, reactor vessel, RCS, safety systems, primary containment, and reactor building can be found in Volume 1 of the SOARCA report.2 The only major nodalization differences between the 1F2 model and the SOARCA model are the steam lines, which are separated into four discrete lines (SOARCA used one discrete line + three lumped lines), and the nodalization of the suppression chamber, which is described in detail later in this paper. Naturally, the dimensions listed in the SOARCA document do not apply to the 1F2 model; all geometric input for the core, vessel, RCS piping, and containment has been modified according to data from TEPCO and GE design documents for 1F2.

Table I. Nodalization summary of 1F2 MELCOR model

MELCOR computational element	Core & RPV	RCS & safety systems	PCV	Reactor building
COR cells	88	NA	NA	NA
control volumes	57	28	21	40
flow paths	105	87	96	70
heat structures	24	37	26	85

The level of spatial discretization described in Table I is relatively coarse compared to higher resolution CFD meshes (several million cells or more), or even to the typical plant nodalizations used in other system-level codes such as RELAP3 or TRACE4, which may consist of several hundred or thousands of elements. But by being a predominately lumped-parameter code, each computational element in MELCOR has several quantities that need to be solved for (e.g. multiple components in a core cell, multiple fluid and radionuclide properties in a control volume, and multiple nodes for a heat structure), and severe accident phenomena for nuclear reactors typically occur on relatively long time scales compared to most plant transients – several days to weeks as exemplified by the Fukushima Daiichi accidents. Such long transients necessitate hundreds of thousands to millions of time-steps in order to properly solve the governing equations in MELCOR. For example, each 96 hour simulation in this work required two- to three-million timesteps to be calculated by MELCOR, for an average timestep size on the order of 0.1 seconds. Each MELCOR simulation of 1F2 entailed five to fifteen days of CPU time on a 3.2 GHz quad-core Intel PC. Thus, although the spatial treatment of the plant may be relatively coarse in MELCOR, such simulations still require considerable CPU time on desktop PCs.

Complicating the modeling effort is the nature of the accident scenarios at the Fukushima Daiichi reactors: core damage is predicted to have occurred long after scram, e.g. 70+ hours for 1F2 and 40+ hours for 1F3, and under conditions of partial cooling from emergency water injection. Core damage and/or vessel breach may have been partially or totally mitigated by water injection into the core using fire engine pumps. These scenario “boundary conditions” complicate and prolong the progression of the core and vessel degradation in MELCOR (i.e. core degradation occurring over *several* hours due to seawater injection and lower decay heat at times long after scram), and this necessitates the use of small time steps over longer time periods in order to precisely simulate the problem.

III.B. Two-Stage Target Rock SRV actuation and operation model

Two-Stage Target Rock SRV actuation and operation logic was implemented through control functions for the 1F2 analysis. A two-stage Target Rock SRV can be opened by two means: 1) meeting the pressure differential set point for passive actuation or 2) through operator action. Operator actuation is available assuming the air supply system is sufficiently pressurized and alignment is available. During passive actuation mode, valve closure occurs when the differential pressure falls below the reclose set point. Re-closure of the valve during operator actuation will result if the air pressure system fails to continually maintain pressure or if the differential pressure between the drywell and SRV falls below 20 psi. Operator actuation cannot occur unless the differential pressure is greater than 50 psi.

Observed pressure data relationship during reported SRV actuation for RPV depressurization demonstrated a 50 psi differential pressure between the drywell and RPV, suggesting Target Rock valves are employed at 1F2.

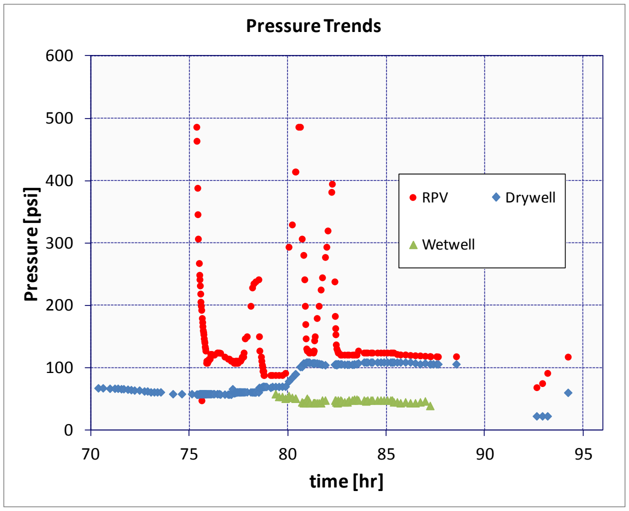

Fig. 1. Depiction of the available pressure data trends around the time of RPV depressurization to illustrate the large differential pressure between containment and the RPV at the time of depressurization.

III.C. Decay power and inventory input

Critical to the calculation of source terms is the accurate input of the initial core radionuclide inventory and overall decay heat. MELCOR has internal inventory data for "reference" BWR and PWR cores, as well as internal decay heat models, but plant-specific source terms require more realistic decay heat and core inventory input that are generated using neutronics codes. For example, SOARCA made use of input data generated by a series of TRITON calculations, along with auxiliary programs and core data from the licensee, for several power histories and three different cycles for the specific fuel assembly design in Peach Bottom.[7] This integrated process generates radioisotope inventories and specific decay heats for each of the 50 core cells that comprise the active fuel region in the MELCOR model. Collapsed 2-D axial and radial decay power distributions calculated by neutronics codes are also important inputs for MELCOR. Variations in 2-D power distributions from plant to plant are usually considered a second order effect for MELCOR calculations of radionuclide releases to the environment. However, these effects may become more significant for accident simulations that involve partial core degradation during extended periods of partial core covering from emergency water injection. According to code predictions and TEPCO plant data for 1F2, RPV water level fluctuated below TAF and above BAF for several hours.

Initially the 1F2 model used scaled-down decay heat and inventories from the Peach Bottom SOARCA model, along with the same axial/radial power distributions, which would tend to increase the uncertainty of the MELCOR calculations. Later in the model development, TEPCO provided neutronics data for each unit at Fukushima Daiichi, which was implemented into each model to decrease model uncertainty. The TEPCO core data was post-processed into a format appropriate for MELCOR input: inventories for hundreds of isotopes and isomers were lumped into MELCOR radionuclide chemical classes, and 3-D power distributions were collapsed into 2-D axial and radial profiles over the 50 active core cells in the 1F2 model.

Fig. 2 shows that the original decay heat curves for the 1F2 model, obtained from scaled Peach Bottom data, overestimated the actual decay power for 1F2 and 1F3 by 5% to 20%, which is enough to significantly impact the thermal-hydraulic accident progression. The red and green curves (generated by TEPCO using ORIGEN2) are the decay powers for 1F2 and 1F3, respectively. It should be noted that the decay powers between these two units are very similar, despite the fact that 6% of the fuel assemblies in the 1F3 core were fuel assemblies containing MOX fuel. The original estimated decay power for 1F1 is very similar to the TEPCO-calculated decay powers. The final MELCOR Fukushima models all incorporate the TEPCO-ORIGEN decay power data.

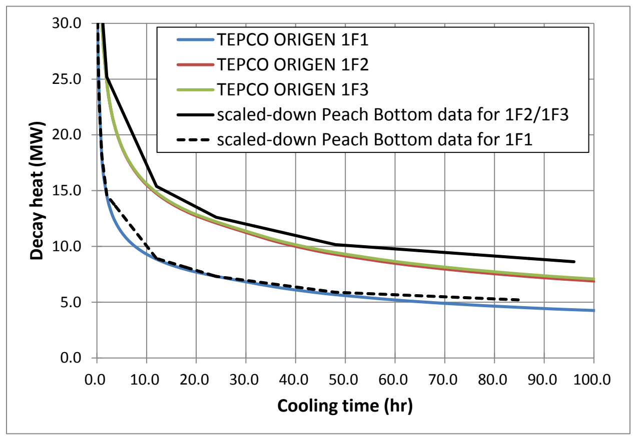

Fig. 2. Decay power vs. time after scram for 1F1, 1F2, and 1F3.

In order to accurately capture the radial distribution of decay power and radionuclides in the core, the size of the radial MELCOR rings must be carefully selected in order for the MELCOR input to be consistent. The number of fuel assemblies, the mass of each radionuclide group, the associated specific decay powers for each radionuclide group, and the radial power fraction in each ring must be consistent with each other and with the total core decay power. (With the radionuclide package active in MELCOR, the decay power distribution is not the same as the fission power distribution, and is determined by the 2-D mass distributions of each radionuclide group throughout the core and their specific decay powers.) The TEPCO-ORIGEN decay power data was used to inform distribution of the fission products throughout the core to recreate the radial and axial power profiles. Table II details the ring nodalization implemented in the 1F2 model. The full power and decay heat power were used to determine an approximate fission product inventory, which is summarized in Table II. In each MELCOR ring, the axially-integrated power fractions are summed and normalized to the number of assemblies in the ring. Radionuclide mass is then distributed in a manner consistent with the radial nodalization and power distribution. Finally, everything is normalized to the total core decay power shown in Fig. 2.

Table II. Radial nodalization of active fuel region

core ring #	Radius (m)	# fuel assemblies
1	0.81	88
2	1.21	108
3	1.57	136
4	1.82	116
5	2.21	100

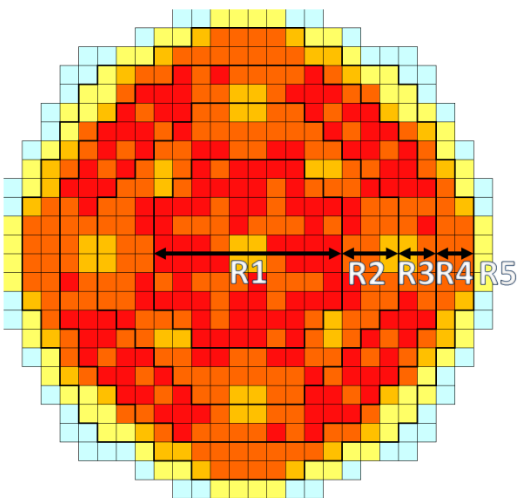

Fig. 3. Axially-integrated, assembly-wise power distribution for 1F2 with MELCOR core rings (post-processed TEPCO data, color contour legend omitted due to proprietary data considerations).

Fifteen MELCOR radionuclide classes are defined for the 1F2 model. MELCOR radionuclide classes are lumped chemically, except for a separate uranium class, or defined as combinations of other radionuclide classes (e.g. CsI and Cs_2MoO_4). A summary of the radionuclide input for the 1F2 MELCOR model is given by Table III. The initial inventory of the iodine class is negligible; it is assumed that nearly all of the iodine is initially in the form of CsI. Immediately after shutdown the decay heat generates 150 MWt, about 6.3% of the full power rating of 2381 MWt.

Table III. MELCOR radionuclide input for 1F2 model

MELCOR radionuclide class	radioactive elements	Initial inventory (kg)	DCH power at shutdown (MW)
Noble gases	Xe, Kr	360.25	11.40
Alkali Metals	Cs, Rb	10.24	0.99
Alkaline Earths	Ba, Sr	159.64	15.41
Halogens	I, Br	6.78E-21	7.93E-21
Chalcogens	Te, Se	33.29	6.61
Platinoids	Ru, Pd, Rh	232.24	2.65
Transition Metals	Mo, Tc, Nb	200.97	18.77
Tetravalents	Ce, Zr, Np	1053.99	11.80
Trivalents	La, Pm, Y, Pr, Nd	1215.40	29.41
Uranium	U	89977.95	2.92
Volatile Main Group	As, Sb	4.47	5.52
Less Volatile Main Group	Sn, Ag	6.53	2.00
Boron	B	4.04E-05	1.20E-05
Cesium Iodide (CsI)	Cs, I	27.66	17.17
Cesium Molybdate (Cs_2MoO_4)	Cs, Mo	264.69	25.34
	total decay power at shutdown =		150.0

III.C. Accident sequence and boundary conditions

The boundary conditions for the accident sequence in the 1F2 MELCOR model are comprised of the (1) TEPCO pressure data, (2) operator actions, and (3) safety system performance. The most reliable boundary conditions are considered the TEPCO pressure data. Hence some boundary conditions for operator actions and equipment behavior were modified according to this more reliable data. Due to the large uncertainty associated with various actions, safety system operations, etc., some assumptions were applied to begin exploring plausible scenarios. Table IV presents the timings utilized as well as related assumptions.

At the time of this writing, there remains considerable uncertainty in the operation of the reactor core isolation cooling system (RCIC), the emergency coolant injection using fire engine pumps, various containment pressure signatures, and safety valve operations. Thus, engineering judgment was applied to investigate these uncertain parameters to explore plausible events and explanations for the observed data trends.

Several assumptions and modeling constraints were implemented with regards to RCIC injection and mass withdraw from the RCS to enforce a specific RPV state, namely, that the RPV at the time of RCIC failure would be representative of the available data. RCIC injection is controlled through level control logic to sustain the RPV level to the observed data. The required steam draw to produce the desired injection, based on an assumed RCIC turbine exit pressure, 5 psi greater than the wetwell pressure, was imposed.

The steam draw required to produce the RCIC injection necessary to maintain level was not sufficient to produce the observed RPV pressure data. The uncertainty regarding the RCIC efficiency, water phases drawn through the RCIC turbine suction line, potential RCS or RCIC leaks, etc. could be potential explanations for failing to match the RPV pressure data. To account for this discrepancy, a stuck open SRV was assumed.

Though, it is best to reduce this assumption to an additional steam loss through a specified open area to the wetwell to account for the inability to reproduce the partial RPV depressurization. The original justification for selecting the SRV is due to the number of SRV cycles which occurred prior to the RPV partial depressurization. Early MELCOR analyses resulted in over 200 SRV cycles by the time the RPV pressure data became available. This lead earlier modeling practices to focus on SRV failure. The SRV "reseat" leak was selected as a 5% SRV open area fraction to simulate reasonable agreement with the containment and RPV data. Due to the inability to maintain this leakage and good agreement after RPV repressurization, the leak is halted at the time of RCIC failure. This, in effect, characterizes the leak as a drastic depreciation in the efficiency of the RCIC system or a leak from the RCIC system that is mistreated by allowing the leak to flow to the wetwell rather than a more likely leak path, such as the through barometric condenser in the reactor building.

An adequate RCIC model has yet to be proposed to address the various and suspected operational states of the RCIC system. Conceptually, the "heavy-handed" approach towards RCIC modeling and SRV leakage was performed to condition the RPV and containment to be in good agreement with available data at the time of the assumed RCIC failure to reduce the uncertainty associate with this time period. These conditional procedures produce an environment whereby the RPV core damage is less sensitive to RCIC operation and more sensitive to the goodness of the available data. Fig. 4 and 5 depict the results of this conditioning phase during RCIC operation.

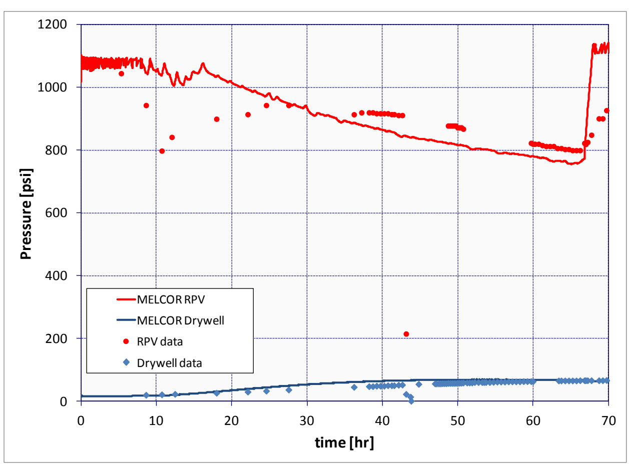

Fig. 4. Demonstrates the good agreement prior to RCIC failure of the system conditioning.

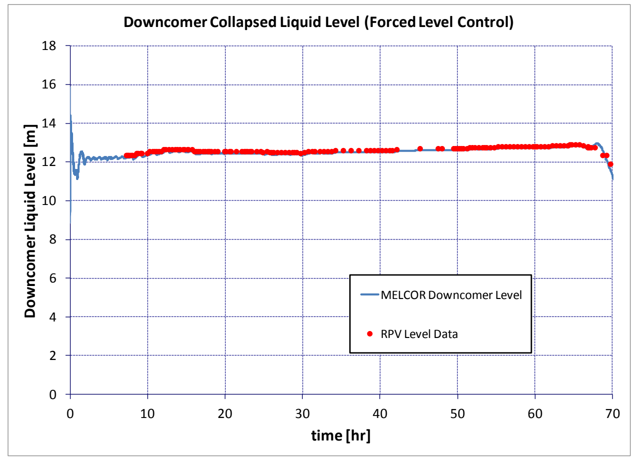

Fig. 5. Level indications are compared to the imposed RCIC controlled level. A zero level is equivalent to the base of the lower head of the RPV.

The method applied to capture the containment pressure trend has been the incorporation of a containment breach. Analyses compare well regardless of an assumed leak while the suppression pool remains subcooled. Once saturated, agreement is not readily achieved. Therefore, containment leakage was imposed during the entirety of the analysis for simplicity. No anticipation regarding its initiation is implied by the authors. A hole with a radius of 1.625 inches was assumed between the wetwell and the torus room.

Table IV. Accident sequence and boundary conditions[6]

System / event	Timing (hr)	Notes / assumptions
Scram	0.0	Scram on seismic event.
RCIC	0.05 – 0.06, 0.25 – 0.68, 0.87 – 66.8	Failure is assumed at 66.8 hrs from the RPV pressure response.
Emergency coolant injection	77.12	Maintained till end of calculation.
Manual venting operation		Assumed to never occur.
SRV opens, RPV depressurized	75.32, 78.55, 95.22	Reported SRV actuation times.
SRV closure	80.22	Reported SRV closure times.
Assumed SRV actuation		
Assumed RPV leakage	5.3-66.8	Modeled as an SRV reseat failure, though treated as a leak associated with RCIC operation.
Containment leak		Assumed

Seawater injection rates were modeled by assuming a pump curve for the fire engine pump as well as applying the INPO reported shutoff head pressure (100 psig).[5] Table V presents the assumed pump curve.

Table V. Assumed fire engine pump curve.

psig	gal/min
0	500
15	500
30	400
70	220
75	200
90	120
95	80
100	0

IV. RESULTS

The MELCOR analysis has been used to produce good agreement with the RPV and containment data during the RCIC operation period. Upon RCIC failure, the RPV is modeled such that the only mass loss occurs through SRV operations, no recirculation pump leakage nor continual steam losses through the RCIC suction line are implemented.

The MELCOR analysis of the RPV response after the system repressurized in response to the RCIC failing is presented in Fig. 6. As seen, the deviation in the observed data and analysis suggest either deterioration of RCIC operation prior to complete failure or that steam leaked from the RCS continued but did not prevent repressurization. Early analyses that maintained the SRV steam leakage during the entire analyses permitted RPV repressurization to the SRV cycling set point. The continual cycling until operator SRV actuation, though, could not be sustained. Potential explanations could include RPV water level error or steam losses were over predicted.

To demonstrate the inclusion of the two-stage Target Rock valve and the assumed SRV operations taken by the operator, four variations are presented, Fig. 7. The first case, where two-stage Target Rock valve logic is not applied, demonstrates a larger depressurization than observed in the data. After the inclusion of the two-stage Target Rock valve logic, better agreement was observed regarding the final depressurization state. Improving RPV pressure should enhance the seawater injection rate prediction and, therefore, improve the final core state prediction.

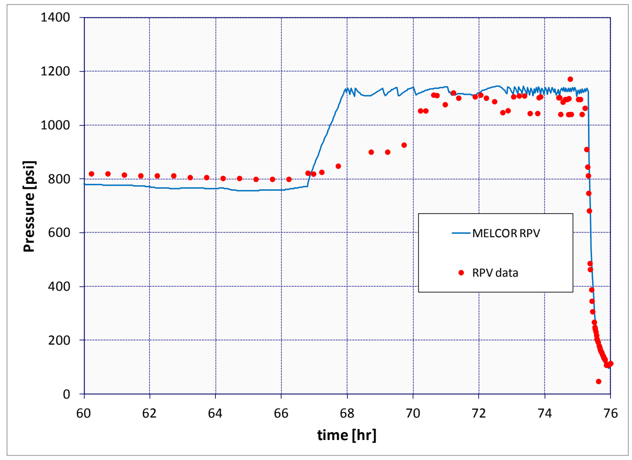

Fig. 6. Comparison of pressure data and the analysis predicted pressure for the RPV around the time of RCIC failure.

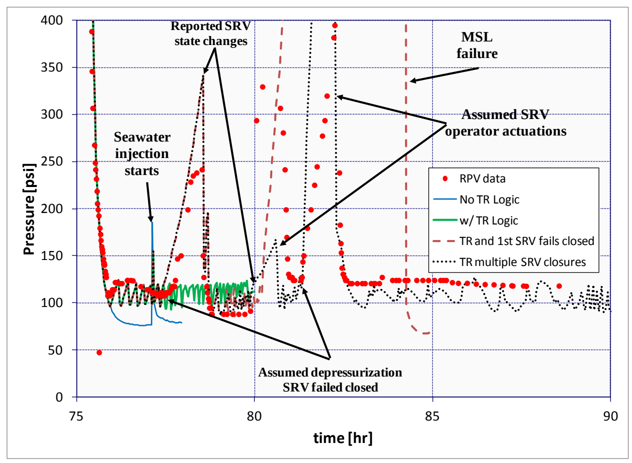

Fig. 7. Model development regarding the assumed SRV failures and operators actuations.

Following depressurization, seawater injection began. In the analysis, the injection of cold water is sporadic, as seen in Fig. 8, due to the frequency with which RPV pressure surpasses the shutoff head of the fire engine pump. The small RPV repressurization observed at ~77.5 hr is not believed to be the result of the seawater injection given the RPV pressure data are nearly at the shutoff head. With the current injection rate predicted and the SRV able to reopen with sufficient differential pressure, this repressurization is not predicted with the MELCOR model as is. Possible explanations of the pressurization event are core material relocation or the SRV failing in the closed position. Given the uncertainty of the surrogate pump curve and the short period of time after depressurization, the pressure spike is not believed to be a product of core degradation with a functional SRV in relief mode. Instead, the SRV utilized to depressurize the RPV was assumed to fail closed. Fig. 9 presents the analysis predicted downcomer level.

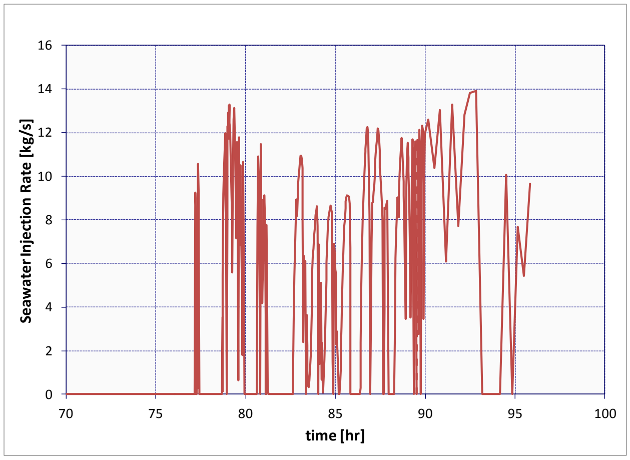

Fig. 8.Analysis predicted seawater injection rates when applying the assumed fire engine pump curve.

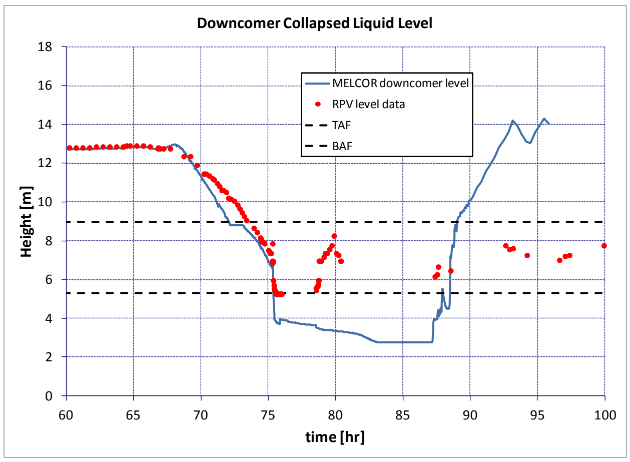

Fig 9. Comparison of the RPV level data and the analysis results for the collapse liquid level in the downcomer.

Due to the repressurization, operators actuated a second SRV to restore low RPV pressure to maintain seawater injection capability. Again, good agreement with the RPV pressure is observed. At 80.22 hr, as specified in the timeline, the second SRV closes producing a second repressurization event. The final analysis, which applied all assumed SRV operations, reduced this timing to 80 hr to demonstrate the sensitivity of the repressurization rate. When compared to the analysis with only the 1st SRV failure closure assumed, which implemented the 80.22 hr timing, the pressurization rates differ greatly. While the analysis captures the pressure impulse, the magnitude does not correspond. Deviation in the steam production rate is coupled to the energy production of the core, zirconium oxidation rate as the largest uncertainty, and the current water level. If either is incorrectly predicted the pressurization rates would fail to agree. The second pressurization rate, as predicted by MELCOR, would not abate without at least one SRV opened. The analysis with only the 1st SRV failure approximation demonstrated that the RPV pressure would rise until reaching the SRV set-point, then, ultimately, the main steam line would fail due to a creep-rupture, given the implemented creep-rupture model.[7]

It seems reasonable to assume that any pressurization event of the reactor, given the lack of power and capability to know system states, would prompt operators to respond under the assumption that the currently open SRV had failed closed, or at the least an additional SRV is required to sustain low RPV pressure. With RPV pressure greater than the seawater injection shutoff head, operates would attempt opening another SRV to restore RPV pressure. Following this logic, the model assumes an SRV was opened and low RPV pressure was restored; when applied, this assumption achieves good agreement with RPV pressure data.

The third pressurization event is once again either a product of SRV closure, zirconium oxidation rates, or core degradation. Following the previously described operator logic, the model assumption applied is to fail closed the open SRV. The difference between the third pressure spike and the second is that the third pressure spike could have been terminated due to RPV pressure boundary failure, SRV operation, or dryout. RPV failure precludes further pressurization events of which no more occur. The analysis up to the third pressure spike has predicted relatively little core damage; furthermore, the seawater injection rate from the surrogate curve model is far less than the TEPCO estimated rates. These seem to support another SRV relief mode operation; this assumption has been applied to the analysis.

The MELCOR analysis of the containment response after the RCIC failure at this time seems far more complex which is compounded by the authors' anticipation of expect trends that are often not observed though predicted with the MELCOR analysis, Fig. 10. At the time of the assumed RCIC failure, wetwell and drywell pressure data were available and in agreement and therefore believed accurate at this time. The observed repressurization of the RPV as a result of some change in the operational state of RCIC does not immediately impact the drywell data, which has an asymptotic behavior suggesting a quasi-steady state was achieved between steam sources and losses as well as energy loss.

Shortly after 70hrs the wetwell pressure data is lost and the drywell pressure begins to descend, suggesting increased losses. The point is exasperated by the observation that the lowest set point SRV begins to cycle at roughly 70 hours. The MELCOR model, though incorporating a containment leak, produces a similar sloped decrease upon RCIC failure by terminating the RCIC turbine exhaust to the wetwell, though this may be a product of chance. But unlike the data, MELCOR predicts a slow repressurization of the containment upon

SRV cycling. A suitable explanation of the differing trends is not proposed at this time.

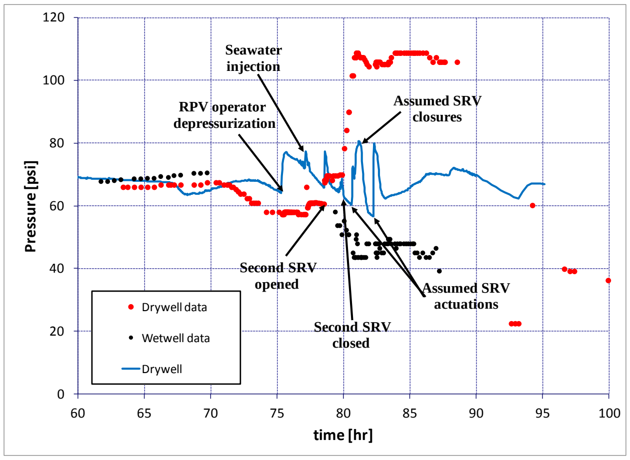

Fig. 10. Comparison between the containment pressure data and the analysis results for containment pressure given the assumed SRV operations.

Containment pressure appears to stabilize shortly before operators actuated a SRV to depressurize the RPV for seawater injection. The depressurization event fails to produce a noticeable change in the containment pressure data. Either the wetwell has achieved a significant, and potentially unlikely, measure of subcooling during the SRV cycling period or the drywell data is suspect.

Around 80hrs, approximately the time of the reported SRV closure, drywell pressure data rapidly rises and stays over 100 psi for a significant duration. During this time period, wetwell pressure data became available for a short time before instrumentation failure. The data are in stark disagreement with a 60 psi difference differentiating the two trends. Note, a rapid and sustained pressurization of the containment is the anticipated trend for significant molten core concrete interaction. Contradicting large lower head failure, the RPV undergoes two repressurization events during the sustained high containment pressure.

V. CONCLUSIONS

The 1F2 analysis is based on SOARCA Peach Bottom model. Early analyses of 1F2 have demonstrated incompatibility between the calculated containment pressure response and the available data of the containment pressure response. To compensate for the conflicting trends, the analysis applied a containment leak from the suppression chamber to the reactor building. Similarly, 1F2 and 1F3 containment response deviated relatively early; this further suggested containment failure had occurred for 1F2. All observed system responses for 1F2 during the first 70hrs of the accident sequence depend upon the RCIC turbine/pump operational state. The available data suggested that the RCIC system maintained water coverage of the core and the combined operation of the RCIC turbine draw and the SRVs maintained the system pressure. Shortly after 5hrs, RPV pressure fell below the SRV set point, while the core level readings remained relatively unchanged until RCIC was believed to have failed.[6]

Due to the uncertainty associated with the RCIC operation, improving the agreement with the containment and RPV states was made a priority to permit a meaningful analysis of the core damage state and progression. Assumptions applied to the analysis regarding the operational state of RCIC were that RCIC maintained the core level at the data values indicated and the deviation between the SRV set point and the decreasing pressure was due to an SRV reseat failure leakage. The pump injection rate was controlled to reproduce the RPV level data and a suitable SRV reseat failure leak open area of 5% was applied to achieve good agreement between RPV and containment pressure just prior to RCIC failing.

When RCIC failed, which was indicated by a decrease in the level data and repressurization of the RPV, the water inventory in the RPV began to boil away. The reason for RCIC failing is unknown at this time, but it is clear that RCIC had been operating outside design basis since the tsunami. To compensate with the continual loss of inventory from the RPV, operators depressurized the RPV to allow firewater injection into the vessel. However, the depressurization event was limited by the relatively high containment pressure. The implementation of two-stage Target Rock valve logic improved agreement following depressurization.

Although the firewater injection was aligned, a re-pressurization event, with a similar signature to a SRV closure event, was observed. The calculated RPV pressure at the onset of the re-pressurization event should have precluded seawater injection from producing the observed re-pressurization based on the INPO reported shutoff head for the fire engine pump. The original SRV utilized to depressurize the RPV was allowed to close in the analysis and the resulting pressure response was similar to the available data. Operators opened a second SRV to reestablish low RPV pressure near the firewater spray injection limit. A second repressurization event occurs shortly after and was documented as an SRV closure.

While RPV repressurization may have resulted from SRV closure, it is only one of several other available explanations, such as core relocation. The pressure trends observed beyond the second SRV closure where investigated by assuming SRV actuations and failures to attempt to reproduce the observed RPV pressure trend. While good agreement with the RPV pressure was achieved, coincidental agreement with the containment pressure was not. Future work will focus on addressing the uncertainty regarding the seawater injection rate if further data becomes available as well as potential core relocation as a product of seawater injection.

ACKNOWLEDGMENTS

The initial development of the 1F2 model used in this work was supported by an accident reconstruction study jointly sponsored by the US Nuclear Regulatory Commission and Department of Energy. Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

ACRONYMS

BAF bottom of active fuel
BWR boiling water reactor
COR MELCOR core package
INPO Institute of Nuclear Power Operators
MOX mixed oxide
PCV primary containment vessel (drywell/wetwell)
RCIC reactor core isolation cooling
RCS reactor coolant system
RN MELCOR radionuclide package
RPV reactor pressure vessel
SOARCA State of the Art Reactor Consequence Analysis
SRV safety relief valve
TAF top of active fuel
TEPCO Tokyo Electric Power Company
TR Target Rock

REFERENCES

1. Sandia National Laboratories, MELCOR Computer Code Manuals, Version 2.1, DRAFT (2010).
2. US. NRC. "State-of-the-Art Reactor Consequence Analysis Project, Volume 1: Peach Bottom Integrated Analysis," NUREG/CR-7110, Volume 1, U.S. Nuclear Regulatory Commission: Washington D.C. (2012).
3. The RELAP5-3D Code Development Team, "RELAP5-3D© Code Manual," Volumes I-V, Idaho National Laboratory, Idaho Falls ID (2005).
4. C. Murray, "Overview of TRACE V5.0," *Proc. Regulatory Information Conference*, Washington D.C., May 14 (2007).
5. Institute of Nuclear Power Operators (INPO), "Special Report of the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station," Revision 0, INPO 11-005, November 2011. http://www.nei.org/filefolder/11_005_Report_on_Fukushima_Daiichi_MASTER_11_08_11_1.pdf
6. R. Gauntt, et al., "Fukushima Daiichi Accident Study," SAND2012-xxxx, Sandia National Laboratories: Albuquerque, NM, DRAFT (2012).
7. US. NRC. "State-of-the-Art Reactor Consequence Analysis (SOARCA) Report," NUREG/CR-1935, U.S. Nuclear Regulatory Commission: Washington D.C. (2012).