III. MELCOR 2.1 MODEL FOR 1F1[2]

III.A. Spatial nodalization of the 1F1 Power Plant

As summarized in Table I, the 1F1 model is a relatively high-fidelity MELCOR representation of the core and reactor coolant system (RCS), and it includes the latest developer-recommended 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 1F1 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.[3] The only major nodalization differences between the 1F1 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 addition of a model of the Isolation Condenser, which is described in detail later in this paper. Naturally, the dimensions listed in the SOARCA document do not apply to the 1F1 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 1F1.

Table I. Nodalization summary of 1F1 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 spatial nodalization used to represent the reactor core, the reactor pressure vessel (RPV), the containment structures and the reactor building are illustrated in Figures 1 through 4. These models together represent a detailed flow network of the entire reactor plant, allowing for a coupled and integrated analysis of the progression of the accident, accounting for fluid loss, heat generation and heat rejection, and important energy and mass flows throughout the system including release of fission products to the environment.

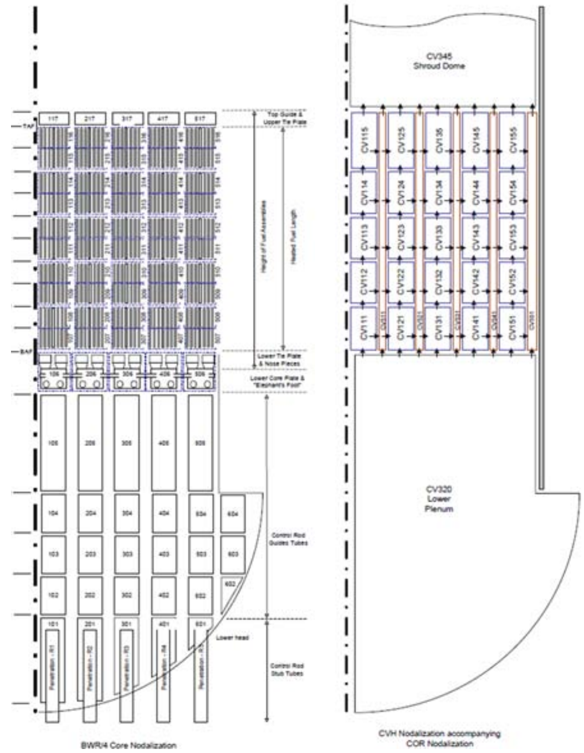

Fig. 1 Core and lower plenum region nodalization used in the MELCOR model for 1F1.

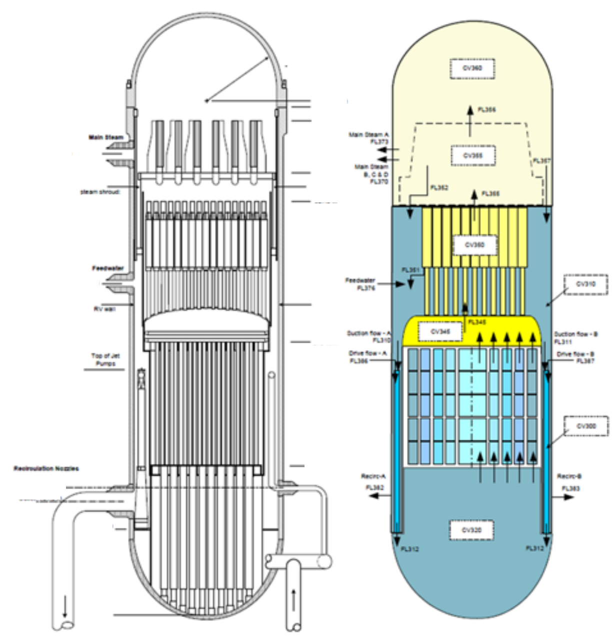

Fig. 2 Reactor pressure vessel (RPV) thermal-hydraulic nodalization used in the MELCOR model for 1F1.

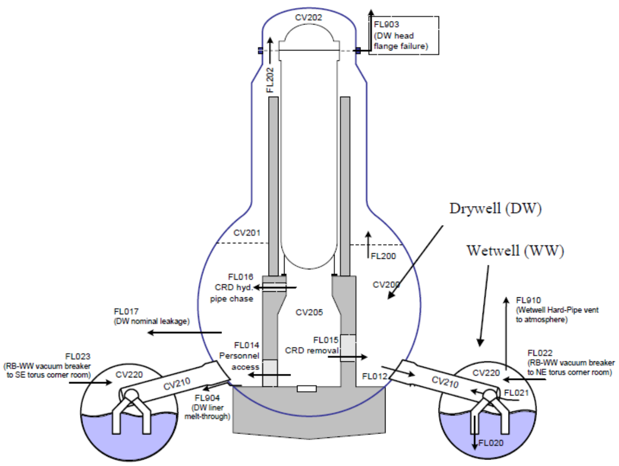

Fig. 3 Containment drywell and wetwell thermal-hydraulic nodalization used in the MELCOR model for 1F1.

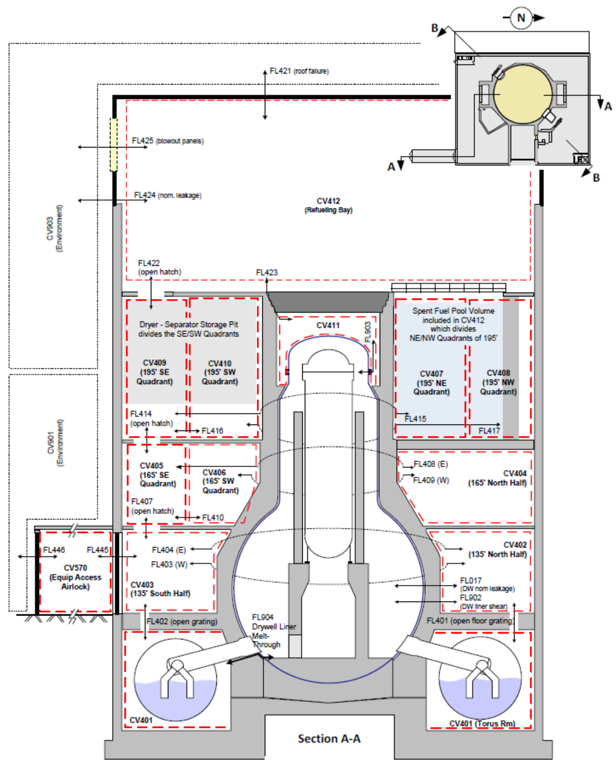

Fig. 4 Reactor building thermal-hydraulic nodalization used in the MELCOR model for 1F1.

III.B. Isolation Condenser Operation and Modeling

The Unit 1 reactor was equipped with isolation condensers (IC) that can in a nearly passive mode (operator action and DC power required to open valves) remove decay heat from the primary system by allowing steam to flow from the RPV into heat exchanges that are located on the upper floors of the reactor building. Each IC has a rated capacity of 42.2 MW. The operational capacity is calculated by multiplying the rated capacity by a pressure-based utility factor (see Table II) to account for the decrease in IC efficiency at lower pressures. In the MELCOR model, a simple negative energy source function (equal to the 84.4 MW times the utility factor) is used to simulate the operation of the ICs.

Table II. IC Utility Factor

Pressure [Pa][1]	Utility Factor [-]
3.26E+06	0.12
4.13E+06	0.33
5.07E+06	0.55
6.15E+06	0.80
7.24E+06	1.00

The IC operation as specified by the Unit 1 timeline (discussed later) was modified. For the last two operating periods (3.53 – 3.65 hr; 6.73 – 20.23 hr), IC operations were not included in the Unit 1 analysis due to the brevity of the first operation period and the model prediction of RPV depressurization prior to the last operation of the IC. Furthermore, the presence of non-condensable gases and aerosols from core degradation (as predicted by the MELCOR model) is assumed to largely disable the ICs' functionality.

III.C. Failure Models for the Primary System

As developed under the SOARCA project, owing to high temperature gases (hydrogen and steam) produced by the Zr-steam oxidation during core degradation, two potential means of unintentional reactor pressure vessel depressurization are modeled in the 1F1 input deck. The first is a so-called high temperature creep rupture of the main steam line as high temperature gases heat the pipe as they flow towards the suppression pool by the cycling of the safety relief valve. Here a Larsen-Miller type analysis is performed to calculate the time of failure of the steam line. On failure, a large double ended break is assumed to open and the RPV will blow down into the drywell. Alternatively, the same high temperature gas may induce a failure of the safety relief valve (SRV) where the valve may seize in an open position resulting in the depressurization of the RPV, only in this case the blow down is directed to the suppression pool. These modeled processes are in competition with each other and generally only one or the other will be predicted, not both. Additionally, the state of modeling in this area is not considered sufficiently developed to be definitively predictive in terms of which mode of RPV depressurization will is more likely, so either is considered equally likely, although, in a deterministic

[1] A utility factor of 0.12 is used for pressures less than 3.26E+06 Pa

sense, one or the other will be produced if high core exit gas temperatures persist for sufficiently long duration. These depressurization modes are illustrated in Fig. 5

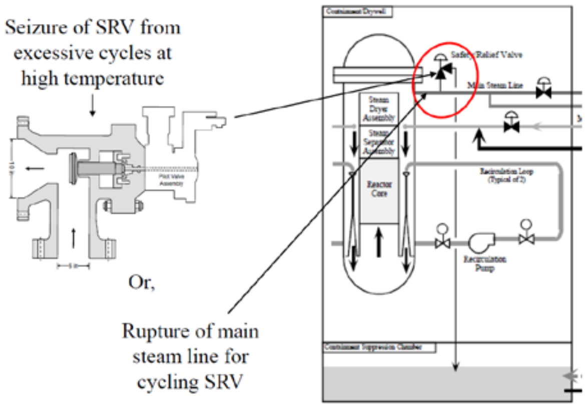

Fig. 5 RPV depressurization modes modeled in the MELCOR model for 1F1.

MELCOR models lower head failure of the pressure vessel using a Larson-Miller parameter for creep rupture, given MELCOR-calculated distributions of temperature and stress in the lower head.[2] The lower head is discretized by a 2-D mesh composed of 8 wall mesh segments, with each having 10 temperature nodes. MELCOR also has a model for penetration failure based on averaged wall temperatures in each ring of the lower head. The penetration failure model is incorporated into the 1F1 model, where penetrations in each lower head ring are assumed to fail once the lumped-temperatures reach the melting point of steel (1273 K). The penetration model however was not exercised in the present study owing to its tendency to predict very early vessel failure.

III.D Containment Failure Modes

Two modes of containment failure are included in the 1F1 model, again adapted from the Peach Bottom SOARCA model. The first model is a containment liner melt-through model that is activated on failure of the RPV lower head with relocation of core materials into the lower cavity. In this situation, molten core-concrete interactions (MCCI) will be considered where chemical heat release as well as concrete decomposition gases are released (H_2, CO and CO_2). In this stage of the accident, depending on the intensity of the MCCI and the temperature of the core materials, the melt may flow across the floor of the cavity and contact the steel liner of the drywell containment pressure boundary. If this occurs, then melting of this liner boundary will be predicted and a release path will be opened up where airborne gases and fission product aerosol can be released into the reactor building, and ultimately to the environment.

The second containment failure mode considered is the stretching of the drywell head bolts at the upper head closure of the containment drywell structure, caused by static over-pressure within the drywell volume, as illustrated in Fig. 6.

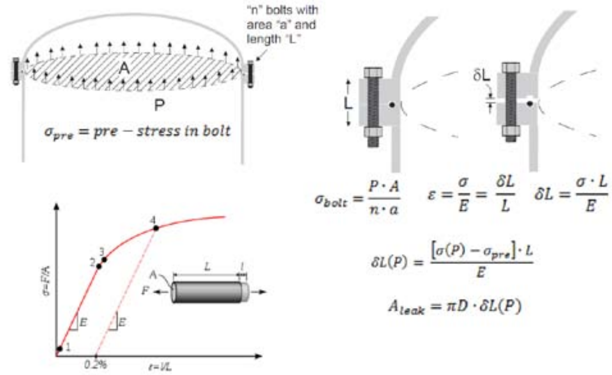

Fig. 6 Drywell head flange leakage model in the 1F1 model allowing containment leakage at high internal pressure.

High pressure in the drywell volume can produce sufficient tensile forces in the closure bolts to overcome the pre-tensioning in the bolts and open a leak path to the refueling bay. Interestingly, in this simple model, the bolt strain remains within the elastic regime of the stress-strain curve, and so if the internal containment pressure is reduced (as by venting actions) then in this simple model the bolts will contract and the leak area will be closed off. The leak area as a function of internal pressure is based on the elastic properties of the bolts (the modulus), but calibrated against observed drywell pressure behavior in the 1F1 pressure history. This basically accounts for the actual effective pre-tension in the bolts and for the actual gasket recovery characteristics when decompressed during bolt straining. In 1F1, good comparison to observation is gained when the model is adjusted to allow leakage to initiate at about 0.7 MPa (110 psid), a slightly higher value than used in the original SOARCA Peach Bottom deck.

In actuality, some permanent strain is considered likely owing to multi-dimensional deformation, and the leak area may not be reduced down to zero as the internal pressure is reduced. Nonetheless, this is the behavior assumed in the 1F1 model, and it will be seen to have interesting implications on the conditions of H_2/CO flammability in the refueling bay later in the 1F1 accident sequence.

III.E. Accident sequence and boundary conditions

The boundary conditions for the MELCOR 1F1 analysis were determined from the TEPCO reported timeline of events which include reactor scram at the time of the earthquake (14:46 Japanese Standard time), intermittent operation of the isolation condensers during the first hour and prior to the arrival of the first of several tsunamis, loss of all DC power after site inundation by the arrival of the tsunamis, operator actions to inject emergency water at about 15 hours and operator actions to vent the containment wetwell just prior to 24 hours. Aside from these events and actions, the 1F1 accident proceeds essentially as a hands-off short term station blackout (classically termed short term because of near immediate loss of DC power). Similar analyses were considered in the SOARCA study for Peach Bottom. The timeline of important events for the Unit 1 accident progression as used to "drive" the MELCOR analysis are summarized below in Table III.

Table III Unit 1 Sequence Timeline[2]

Date and time	**Time after scram [hr]**	**Event**
3/11 14:46	0.00	Earthquake; Reactor scrammed
3/11 14:47	0.02	MSIVs* close due to loss of instrument power, loss of normal heat sink
3/11 14:52	0.10	Isolation Condensers automatically starts (Train A and B)
3/11 15:03	0.28	Isolation Condensers (Train A and B) manually stopped to control cool down rate
3/11 15:07	0.35	Containment A and B spray systems activated
3/11 15:17	0.52	Isolation condenser train A manually started
3/11 15:19	0.55	Isolation condenser train A manually stopped
3/11 15:24	0.63	Isolation condenser train A manually started
3/11 15:26	0.67	Isolation condenser train A manually stopped
3/11 15:27	0.68	First tsunami wave hits
3/11 15:32	0.77	Isolation condenser train A manually started
3/11 15:34	0.80	Isolation condenser train A manually stopped
3/11 15:35	0.82	Second tsunami wave hits
3/11 15:41	0.92	Station Blackout; containment sprays stopped
3/11 18:18	***3.53***	***Isolation condenser train A manually started (not implemented in the model)***
3/11 18:25	***3.65***	***Isolation condenser train A manually stopped (not implemented in the model)***
3/11 21:30	***6.73***	***Isolation Condenser train A manually started (not implemented in the model)***
3/12 5:46	15.00	Fresh water injection from fire water pump starts, 80,000 liters injected by 14:53
3/12 9:05	***18.32***	***Drywell venting attempted (model assumes no venting)***
3/12 11:00	***20.23***	***Isolation condenser train A manually stopped (not implemented in the model)***
3/12 14:30	***23.73***	***Wetwell vented using portable generator and air compressor (model assumed vent value re-closure consistent with TEPCO wetwell pressure data)***
3/12 14:53	24.12	Fresh water injection stopped due to running out of fresh water. 80 ton total injection
3/12 15:36	24.83	Hydrogen explosion in the reactor building
3/12 19:04	28.30	Seawater injection from the firewater system starts
3/14 0:00	57.20	Seawater injection from the firewater system ends
3/15 0:00	81.20	Seawater injection from the firewater system starts
3/15 14:36	96.00	End of simulation

* MSIVs are main steam-line isolation valves

IV. RESULTS

IV.A ACCIDENT INITIATION AND CORE DAMAGE

The first hour of the 1F1 accident sequence began with reactor shutdown and closure of the main steam isolation valves (MSIVs). During this time the operators manually activated the isolation condensers by opening the valves that connect the heat exchanger to the RPV. Initially two IC units were activated during the first hour prior to the arrival of the tsunami, which produced the deep depressurization of the RPV at about 0.3 hours as seen in Fig 7. The ICs are modeled as energy sinks to the downcomer vapor space. The MELCOR predicted RPV pressure is compared to strip chart data available during the first hour following the earthquake.

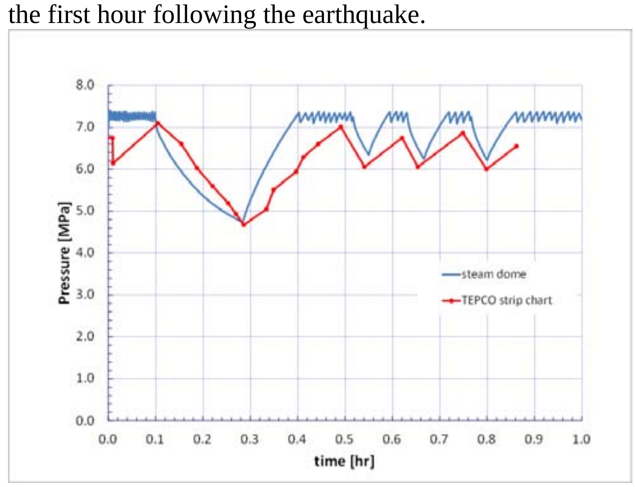

Fig. 7 Predicted and observed RPV pressure signature during the first hour of the Unit 1 accident.

The use of both IC's was reported to have produced and RPV cool-down rate which exceeded the technical specifications for safe cool-down rate, and thereafter only one IC was activated. The MELCOR analysis replicates the essential signature of these actions, but shows some differences that will be corrected in future refined analyses. The strip chart data indicates that SRV cycling was suppressed during the first hour, whereas the MELCOR analysis shows some intermittent SRV cycling.

This deviation is not believed to produce any important later effects on the subsequent sequence of events.

Following the arrival of the tsunami and the development of complete station blackout conditions with the loss of DC power, the accident proceeds in a near hands-off manner as emergency core cooling systems such as the near-passive IC's discussed previously as well as the turbine-driven High Pressure Core Injection (HPCI) pumps could not activated. Neither were there any instruments to indicate RPV pressure or water level until later in the accident when DC power was partially restored to instrumentation at about 10 hours in the accident. By then significant core damage is predicted to have taken place.

Fig. 8 shows the MELCOR-predicted water level in the RPV where uncovering of the top of the active fuel is predicted to have occurred by about 3 hours with full uncovering of the reactor core by 5 hours. Water loss from the core takes place as the safety relief valve(s) cycle to relief RPV pressure by venting steam into the suppression pool. Severe damage to the core is predicted to initiate at about 4 hours as the water level falls in the core, temperatures rise rapidly and significant hydrogen is produced through the Zr-steam oxidation reaction. The high temperatures produced during the Zr-oxidation transient produces significant heating in the steam line associated with the lowest-setpoint cycling SRV, and this steam line is ultimately predicted to fail due to creep rupture at roughly 6.5 hours, where the sudden rapid loss of water level in the lower plenum due to flashing can be seen in Fig. 8. Note that there are no actual data available for comparison to the predictions at this time in the accident.

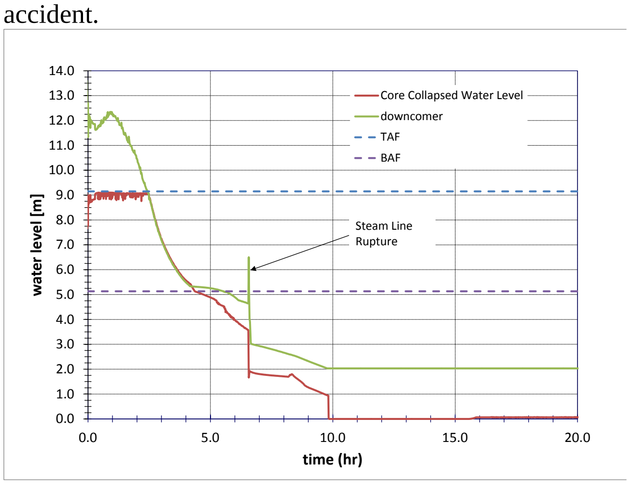

Fig. 8 MELCOR-predicted core water level during the core damage stage of the 1F1 accident.

Core slumping into the lower head predicted at about 9 hours completes the dryout of the lower vessel head and after this time the lower head begins to heat until it fails essentially by melt-through at about 14 hours. As stated earlier, only the global creep failure model for predicting lower head failure was implemented in this analysis. A penetration failure model could have produced an earlier failure of the vessel head, but perhaps not an immediately massive release of core material to the cavity. Future studies will explore these potential sequence variations. The present analysis shows gross failure at about 14 hours when approximately 140,000 kg of core materials was released to the drywell cavity region. Fig. 9 illustrates the predicted core damage map neat the time of vessel head failure. Notice that some peripheral fuel assemblies are predicted to remain standing the core region after lower had failure.

Instrumentation for the reactor was not restored until after 10 hours when operators became aware that the RPV had become depressurized. As of today, there is no clear indications of the actual mode of depressurization, whether main steam line rupture as predicted in this study, SRV seizure in open position, or by other gasket or seal failures in the primary system. That said, the observed pressure signature as seen in Fig 10 is seen to be more consistent with an RPV blow-down in the drywell as opposed to venting into the wetwell where steam condensation and greater pressure suppression would have been predicted. The predicted and observed containment pressures in the 10 to 20 hour time period are more consistent with RPV depressurization into the drywell, where calculation and measured data compare very well.

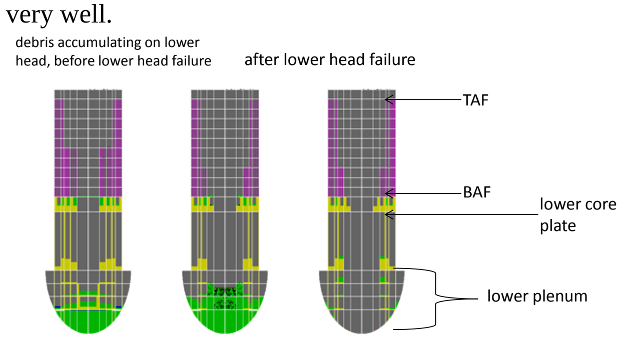

Fig. 9 Core damage map showing predicted damage configuration of the core as relocation to the lower head and lower head failure takes place.

Emergency injection of fresh water was not accomplished until roughly 15 hours (Fig. 11) and after the MELCOR-predicted time of RPV lower head failure. Because of this, the injected water at 15 hours is predicted to have exited the vessel through the failed lower head and dropped onto the MCCI reaction ongoing in the reactor cavity. In the MELCOR analysis, the production of non-condensable gases, H_2, CO and CO_2 as well as release of some hydrated water starting at about 14 hours