ANALYSIS OF FUKUSHIMA DAIICHI NPP UNIT 3 WITH MELCOR_2.1

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ABSTRACT

During the major accident occurred at the Fukushima Daiichi nuclear power station in March 2011, three units of the nuclear power plants suffered extensive damage to the reactors and buildings. It is widely believed that all three reactor cores experienced some melting.

In particular for unit 3, there is not a general agreement on the extent of the core degradation. Plausible scenarios varying from in-vessel to ex-vessel have been predicted in previous studies. The present analysis presents a plausible accident sequence for Unit 3. The transient has been analyzed with Melcor_2.1. The main accident signatures were captured in the present analysis and hence give the sequence some degree of credibility. However, there are still remaining questions and uncertainties concerning the possible reactor pressure vessel and primary containment vessel failures which need to be addressed in the future.

Key Words: Fukushima plausible sequence, MELCOR.

1 INTRODUCTION

Paul Scherrer Institute (PSI) is taking part in an Organization for Economic Cooperation and Development (OECD) project, Benchmark Study of the Accident at Fukushima (BSAF) to reconstruct the events that occurred at the Fukushima nuclear power plant in March 2011. Eleven institutes from eight countries have participated in the phase-I of the project. PSI has performed simulations of Unit 3. The analysis was performed using a generic MELCOR 2.1 (SNL, 2008) input model based on Peach Bottom power plant (SNL, 2012), (Carbajo, 1994). The input was adjusted to the specifics of Fukushima.

All the benchmark participants were provided the same set of boundary conditions; which in the presented analysis were considered as a departure point. The remaining questions have been explored by sensitivity studies. Series of sensitivity cases (> 300 calculations so far) were performed progressing systematically between the initial conditions and the set of conditions that best reproduce the available measurements (e.g. pressure histories of the reactor pressure vessel (RPV), dry-well (DW) and wet-well (WW); downcomer water levels and the observed hydrogen explosion time.

The main models employed for the calculation as well as the nodalization will be described in section 2. A plausible sequence of Fukushima unit 3 accidents will be described in section 3. The presented scenario was the result of a detailed analysis presented in (Fernandez-Moguel and Birchley, 2015). The chosen calculation for the present paper reproduced the best the available measurements as well as combustible hydrogen concentrations at the top of the reactor building at the time of the explosion. The proposed transient is considered to be a plausible sequence of the Unit 3 accident, bearing in mind that there are still open questions especially concerning the possible failures of the RPV and the primary containment vessel (PCV) as well as the hydrogen and fission product releases paths.

2 EMPLOYED NODALIZATION AND MAIN MODELS

The system-level MELCOR 2.1_4203 is being used for the present simulations. The general models used by MELCOR are described in the manual (SNL, 2008), only the specific models used in the present analysis will be described in the present section.

The input model encompasses the reactor vessel together with the associated coolant circuits, the containment dry-well (DW) and wet-well (WW) and the reactor building. The vessel noding is shown in figure 1. The active core region is divided into 5 fuel channels each subdivided into 5 axial nodes and 5 bypass channels with a single axial node. A total of 30 hydraulic cells are therefore used for the core. The remainder of the RPV is represented more coarsely with just 7 cells. The feed water and circulation lines are also represented with single cells for each of the main segments from-to the RPV.

The radial core noding is the same as the hydraulic noding but the fuel assemblies, comprising the fuel rods, spacer grids and channel boxes, are divided into 10 axial cells (numbered 7 to 16). The fuel channels are inside the channel boxes and the bypass volumes that contain the control blades are outside the channel boxes. The upper support structure occupies node 17. The vessel lower head, the lower support and other structures extend to the bottom of the vessel and span nodes 1 to 6 which are associated with a single hydraulic node. Penetrations in the RPV lower head structure for the control assembly drive housing and guide tubes are included in each radial node.

The noding of the containment and reactor building is shown in figure 2. The WW is divided in to a lower (normally liquid) and an upper (normally steam and nitrogen) volumes. Four cells are used to model the DW as shown in figure 2. The containment is inerted with nitrogen. Included in the ensemble of steam system and containment nodes are the safety relief valves (SRV) and connections to the WW, the turbine-driven Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) systems and connections back to the vessel, as well as the connections to and from the condensate storage tank (CST). The containment may be vented from either the dry- or wet-well (but in the present scenario only WW venting is actuated). The vent line passes to a rupture disk upstream of the line to the stack, although the line beyond the disk and the stack itself are not modelled. Apart from venting, the DW can release steam and gas to the reactor building if the internal overpressure is high enough to force the head flange to separate from the lower PCV vessel flange due to bolt stretching and hence to open a leak path. In the case gases are released from DW, they are assumed to escape further into the upper part of the reactor building. A total of 12 cells is used for the reactor building which not inerted.

The employed input model adopts the MELCOR default or customary options for most of the physical processes, only the non-default modelling features employed in the calculation will be mentioned in this section. A description of the main employed models was given in Fernandez-Moguel and Birchley (2015). The modelling is described in detail in the code reference manual (SNL, 2008).

The conditions at which fuel rods are likely to collapse to debris is uncertain; in the model this is assumed to occur for fully oxidized rods at 2800 K (non-default) corresponding approximately to the melting temperature of UO_2-ZrO_2 mixture. Fuel rods would collapse at 3100 K (default) regardless of the composition of the cladding.

Vessel breach can occur by penetration failure or by creep damage according to the Larson-Miller parameters (SNL, 2008). In the present model penetration failure is actuated at defined temperature, which is likely to occur before creep failure. By far the largest contribution to the total penetration area is the control rod drive housing. The area of the breach following a single failed penetration is 0.012 m^2, corresponding to the internal flow area of a single control rod drive channel of diameter 123.4 mm, and it is supposed as a bounding case that one such failure might occur in each of the core radial nodes, in the calculation presented in the present analysis. No other mechanism for vessel breach is modelled in the present simulations. Breach of the steam line may have occurred but it is not included in the present analysis.

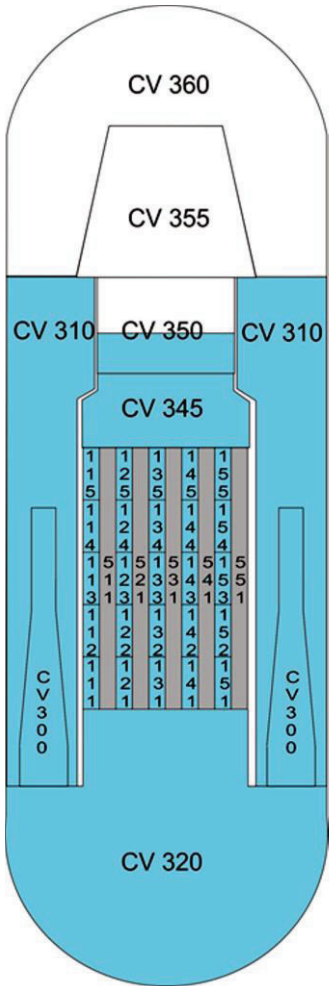

Figure 1: RPV nodalization

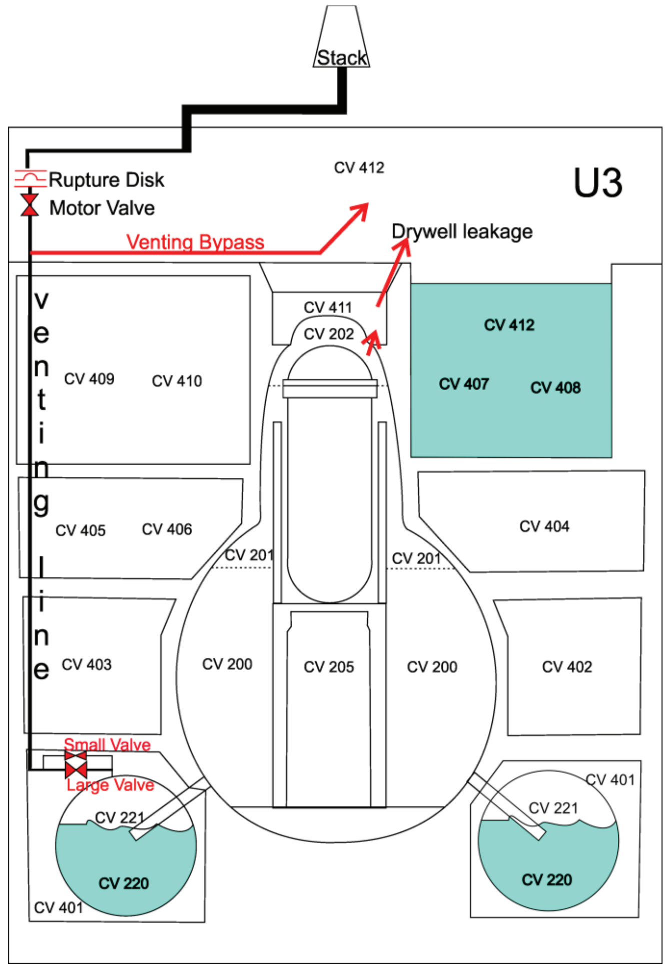

Figure 2: PCV nodalization

3 ACCIDENT SEQUENCE

The present study adopted a step-by-step approach to define the boundary conditions for the various phases of the accident sequence in such a way as to calculate the sequence as well as possible up to 6 days after the SCRAM, while remaining consistent with available records from the plant operation. The period of 6 days corresponds roughly to the unavailability of AC power.

The investigation concentrates on the core uncovery, degradation and relocation, including possibly ejection into the cavity. Fission product release from the fuel is included in the core degradation, but transport through the containment and leakage to the reactor building (i.e. outside the containments) and environment are not addressed in the present study. A credible account of the core degradation and its impact on system integrity are necessary prerequisites to evaluation of the FP release.

The main data that have been used for the present analysis are available in the information portal for the Fukushima Daiichi Accident Analysis and Decommissioning Activities (TEPCO, 2014b).The data used for the analysis are:

- The times at which the hydrogen explosions took place in each unit.
- The pressure history in the reactor pressure vessel (RPV) and in the containment DW/WW have been identified as fairly complete and reliable data, which is fortunate because this serves as a trail of footprints that point to what was happening.
- The times and rates of fresh or sea water injection (by means of fire engine pumps) into the reactor system, though unfortunately the rate of delivery to the RPV itself is uncertain.
- The time when the operators vented the containment to control the pressure, though unfortunately it is uncertain if all the venting operations were successful or had taken place at the reported time and the percentage of the valve opening is unknown.
- The water level measurement is available but it is subject to gaps and uncertainties.

The analysis was performed using a generic MELCOR 2.1 input model based on Peach Bottom power plant (SNL, 2012), (Carbajo, 1994). The input was adjusted to the specifics of Fukushima. An initial calculation was performed and series of sensitivity cases were performed. The detailed analysis was presented in (Fernandez-Moguel, 2015). Only the sequence that represents the best the results will be presented in this study. The following section describes the main operator's events as well as the possible failures of the RPV and PCV following a chronological order.

3.1 RCIC and HPCI operation

The automatic reactor protection system worked as designed and the reactor was SCRAMmed shortly after the earthquake. In the calculations it is assumed that the SCRAM time was on March 11th 2011 at 14:47 and this is considered the time 0:00 h. All the times cited in the present paper refer to hours after the SCRAM.

The Reactor Core Isolation Cooling (RCIC) started to operate automatically due to low level in the RPV at 0:18 h. The pressure was maintained within the normal band through the SRV operation which discharged the steam into the WW. The RCIC injection stopped automatically at 0:38 h due to the high RPV water level; at 0:51 h the tsunami hit the plant causing all AC power to be lost and with the RCIC automatic operation paused at this time. For the next days, only the DC power was partially available.

The downcomer collapsed water level started to decrease and the RCIC was manually restarted at 01:16 h. Operators maintained downcomer water levels by adjusting the water flow rate to the RPV, in order to allow gradual reactor water level changes. This was done using the line configuration where water would pass through both the reactor injection and test lines so that part of the water returned to the condensate storage tank (CST), which would prevent automatic shutdown of the RCIC that would have taken place due to high reactor water levels. In this way the operators avoided excessive battery depletion due to repeated RCIC de-activation and re-activation, and also ensured stable reactor water levels. The RCIC stopped automatically at 20:49 h, presumably due to the high water level in the reactor.

The High Pressure Coolant Injection (HPCI) started at ca. 21:48 h in the calculations. This is 00:25 h before the time reported by TEPCO. This assumption was necessary in order to reproduce the observed pressure drop (figure 3) in the measured data. The operators were manually controlling the steam extraction and the water injection to the RPV. Similar to the RCIC, the operators were able to adjust the water flow rate by splitting the test line flow into the flow going to the RPV (i.e. needed to keep the water level in the reactor), and the flow recirculated to the CST (i.e. the exceeding flow). For the calculation, the injected water to the RPV was tuned manually, meaning the flow rates were adjusted according to the response of the water level calculated in the down-comer. The assumed flow rates for RCIC and HPCI simulate how the operators are understood to have used the systems to control the RPV water level and pressure. In this way, the thermal-hydraulic response during RCIC and the HPCI operation was well reproduced (figures 3 and 4). However, the exact amount of water injected is uncertain and is very sensitive to the calculated thermal-hydraulic RPV conditions (i.e. pressures, temperatures and water inventory) at specific times.

Almost simultaneously with the start of HPCI operation, sprays were started in order to decrease the pressure in the DW/WW. The sprays were injected to the WW from 21:19 to 36:18 h and from 38:21 to 40:56 h to the DW from 40:52 to 42:08 h. The sprays were considered in the calculation. The results suggest that the sprays were not enough to decrease the pressure in the DW/WW. The assumption that water was injected in two occasions from the CST to the WW, in addition to the sprays, was necessary in order to reproduce the pressure in the DW/WW (figure 5). In consequence, it seems likely that this action took place. However, this action was not reported by TEPCO. It is also possible that the lack of spatial resolution in the model for the WW influence the results, thus it is identified as an issue for further study.

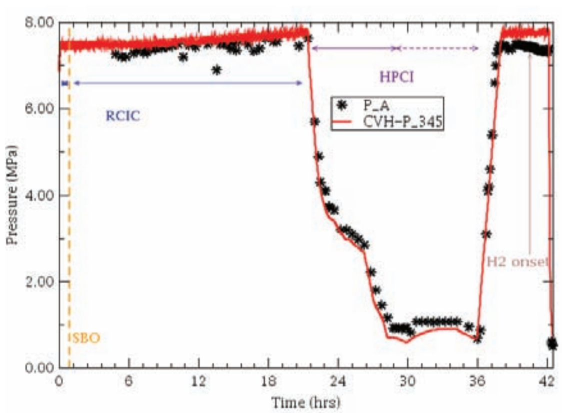

Figure 3: RPV pressure

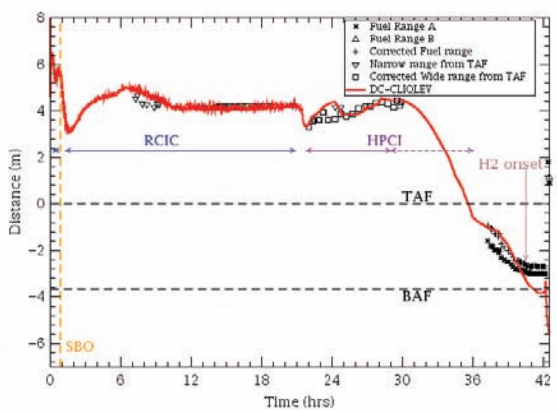

Figure 4: Down-comer collapsed water level

After 29:00 h, the down-comer water level measurement stopped. The next available measurement was at ca. 37:00 h and it is below the Top of Active Fuel (TAF). The operators have been reported to manually having deactivated the HPCI at ca. 35:57 hours. However, based on our analyses, it is likely that the HPCI operation degraded significantly earlier than this. For the calculation, it was assumed that HPCI water injection gradually stopped while steam was still extracted. The calculation reproduces very closely the observed pressure in the RPV (figure 3) and the down-comer collapsed water level (figure 4) and supports the theory that the HPCI water injection was degraded after ca. 29:00.

The water level continued to drop and reached the Bottom of Active Fuel (BAF) at ca. 41:08 h. The hydrogen production started at ca. 40:00 h. The oxidation reaction stopped temporarily, because there was no steam flowing to the core at that time. The temperatures in the core increased. At this stage, cladding temperatures of ca. 1400 K had been reached and ca. 68 kg of hydrogen had been produced. However, the core was still in place and no relocation of materials had taken place, as shown in figure 6.

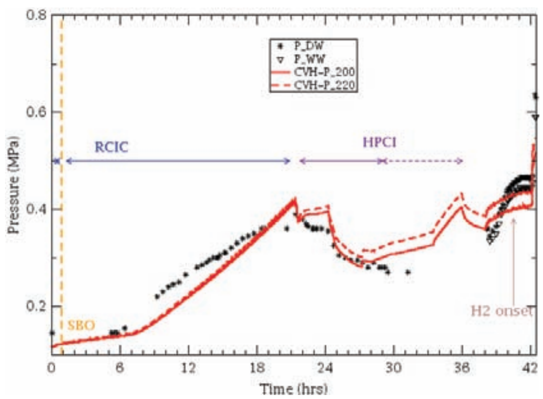

Figure 5: DW/WW pressure

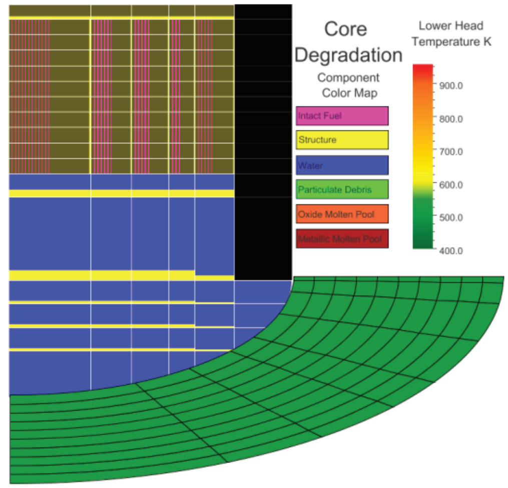

Figure 6: state of the core at 42:07 h

3.2 RPV and PCV behavior after RPV depressurization.

At 42:08 h, the depressurization of the reactor pressure vessel was initiated. It is assumed that this took place by actuation of the automatic depressurization system. However, a failure of the primary pressure boundary, e.g., the main steam line, can not be excluded. The water level in the RPV decreased further during the depressurization of the RPV, reaching levels of ca. 2.2 m below BAF before any water could reach the RPV using alternative water injection (AWI) (at ca. 42:38 h).

After depressurization several venting actions were performed. In principle, the venting should increase the pressure in front of the rupture disk in the vent line and open a path for gases straight to the stack. However, the build-up of H_2 in the upper part of the reactor building points strongly to failure of isolation of the vent line or leaking from PCV to the building. It was therefore assumed that all the successful venting had leaked to the reactor building and that the rupture disk did not burst. In addition the possibility of DW leakage was explored as a possible cause of the hydrogen explosion.

In parallel to the venting, the Alternative Water Injection (AWI) started by means of the fire engines. It is known when the operators reported to have injected either fresh or sea water to the RPV, as well as the amount of water that they injected per day, but the actual amount that reached the RPV is uncertain.

Best agreement with measured data, namely, collapsed water level in the downcomer and pressure in the RPV and DW/WW (figures 7 and 8) was achieved by adjusting boundary conditions (i.e. venting times and magnitude of venting and the amount of alternative water injection reaching the reactor) relative to nominal values. In any case some uncertainty remains concerning the actual values. The calculation results point out that only 30-60% of the nominal AWI was reaching the reactor.

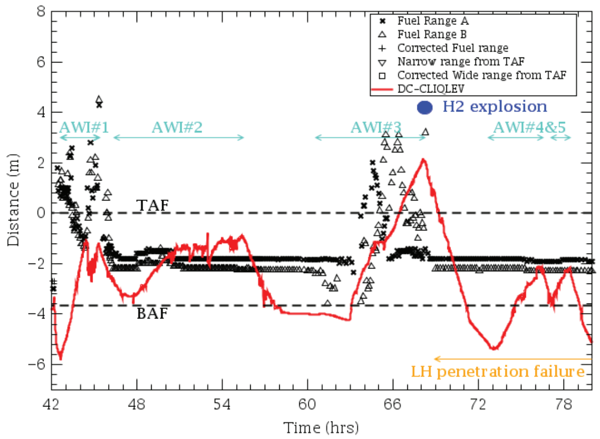

Figure 7: downcomer collapse water level

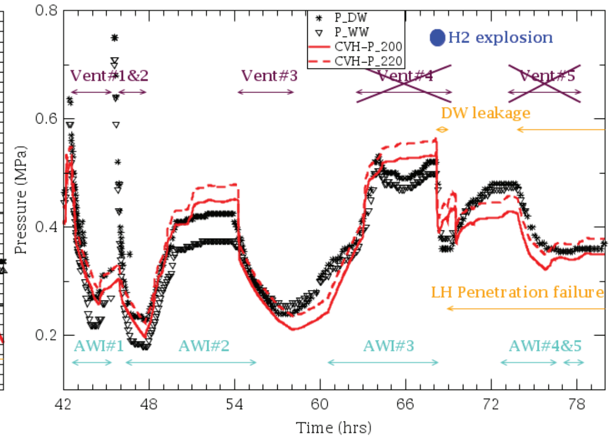

Figure 8: DW/WW pressure

From the calculation, three main hydrogen excursions were identified (as shown in figure 9). The calculations point out that the increase in pressure at ca. 42:00 h, 48:00 h and 60:00 h is likely due to the response of reflooding of an overheated core. Each excursion is closely connected with the start of AWI reported by the operators. Furthermore, shortly before each AWI initiation the water level in the RPV has been below the bottom of the fuel for extended periods of time, causing the core to overheat and be prompt to produce an excursion due to the steam-Zr oxidation reaction. The total amount of hydrogen which was produced in-vessel was ca. 1500 kg, from which ca. 700 kg were generated in the time preceding the hydrogen explosion.

An explosion which is attributed to hydrogen generated by oxidation of metallic components in the degraded core of unit 3 was observed at U3 reactor building at 68:14 h. However it is uncertain how the hydrogen made its way to the reactor building. One possibility is a leakage from the venting line during the time before the explosion, when a venting action was reported to be open (between 62:00-70:00 h). However as it was described in (Fernandez-Moguel and Birchley 2015) the venting action cited by the operators in the time before the accident explosion seems to have been unsuccessful. If the venting action had taken place between 62:00-68:00 h, the increase of pressure in the DW/WW at ca. 60 h wouldn't have taken place. In addition, the amount of hydrogen leaked to the reactor building in this scenario doesn't seem to have been enough to explain the observed hydrogen explosion. In contrast, the increase in pressure in the DW/WW can be very closely reproduced when the venting between 62:00-68:00 h is considered to have been unsuccessful as shown in figure 8.

Since the calculation point out that venting was unsuccessful before the time preceding the hydrogen explosion, DW leakage was assumed as an alternative to explain the hydrogen explosion. The leakage from the DW to the top part of the reactor building is possible if the internal overpressure in the DW is higher than the design pressure of ca. 0.5 MPa to cause the head and vessel flanges to separate due to the stretching of the restraining bolts (Hessheimer and Dameron, 2006). According to the measurements, there are four occasions when the pressure in the containment is higher than or equal to the design pressure of 0.5 MPa. Shortly after depressurization of the RPV, at ca. 42:00 hours, a pressure spike of ca. 0.62 MPa was observed for a short period of time. According to the calculations the depressurization from the PCV can be fully explained by venting from the WW and is in agreement with the time reported by the operators. The second pressure spike was observed at ca. 46:00 hours, but this spike was not captured by any of the calculations and only venting was assumed. The pressure was around 0.5 MPa between 64:00 - 69:00 and 72:00-74:00 h, and it is likely that the long-time operating near or slightly higher than design pressure in addition to the two previous events where the design pressure was exceeded may have caused the bolts to weaken and DW leakage to occur.

It was assumed that the first DW leakage took place at ca. 68:11 h and that the leakage was reduced as the pressure decreased, the stress in the bolts may have been reduced causing the head and vessel flanges to come close together and the gas leakage to stop. The calculation predicted that ca. 700 kg of hydrogen were released to the reactor building very quickly before the explosion (at ca. 68:14 h). The concentrations of gases at 68:14 h (i.e. the time of the observed explosion) at the top of the building were 14.4 mol% of H_2, 35.6 mol% air and 50.0 mol% steam, which were in the combustible regime (figure 10).

A second event of DW leakage was assumed to take place at ca. 74:00 h. This second DW failure assumes that the bolts never recovered completely again and that a small leakage remained for the rest of the transient.

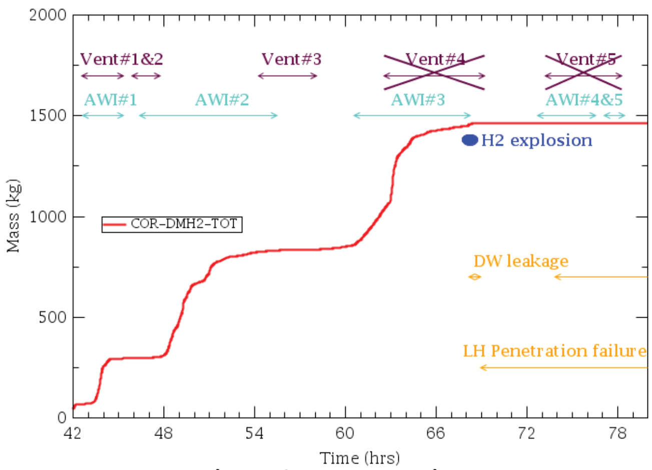

Figure 9: H_2 generation

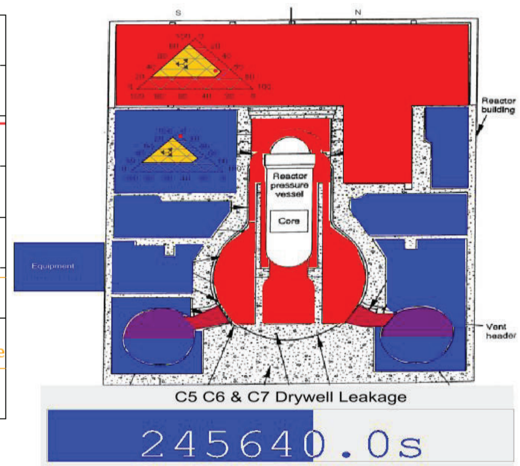

Figure 10: H_2 concentration at the top of the building

In parallel to DW leakage, the RPV lower head (LH) may have experienced penetration failure. The largest contribution to the total penetration area is the control rod drive housing. The area of the breach following ejection from a single failed penetration is 0.012 m^2, corresponding to the internal flow area of a single control rod drive channel of diameter 123.4 mm. In the input model it is supposed that one such failure might occur in each of the radial nodes if certain temperature is reached at the location of penetration. In (Fernandez-Moguel and Birchley, 2015), it was found that a penetration failure area restricted to ca. 1.0% of the full assembly, when the temperature in the penetrations reached 955 K, predicted very closely the pressure in the DW/WW as well as the observed collapsed water level measurement to a certain extent. This case predicted penetration failure failures in rings 1 and 2 at 68:57 h and that all the debris remained in the RPV. In addition, a relative constant water leakage of ca. 4 kg/s was predicted for the rest of the transient.

The assumed area of the penetration leakage as well as the temperature failure criteria is crucial. It influences the prediction of an in-vessel (with all the debris inside the RPV and with or without RPV water leakage) or an ex-vessel (with all the debris that had relocated to the lower plenum being expelled to the cavity) scenario. It is not certain if any of the debris/molten material was expelled out to the cavity, but in (Fernandez-Moguel and birchley, 2015) the case that predicted the measurements closest had the debris remaining inside the RPV. This is the calculation presented in the present analysis. These results indicate some likelihood that most of the debris remained inside the RPV. The prediction of the RPV failure vs. non failure has been identified as one of the main code limitations thus the assumptions made by the code user influence greatly the results.

Due to the water leakage and the interruption of water injection the water level decreased to ca. 1.6 m below BAF at ca. 71:10 hours. Water injection was resumed from 72:43 to 76:33 and from 77:08h to

78:27 h. Despite the water injection interruption, the water level was all the time above the debris during that time.

From 78:27 h until 83:07 h, the AWI was interrupted; in consequence, the water level decreased, until the debris at the bottom of the lower head was uncovered (at 83:07 h). This situation only lasted ca. 38 minutes, which may have been not enough time to heat up and melt the debris. Therefore the debris remained inside the vessel. Water injection was restored at 83:45 h and continued beyond the time of the analysis sequence (144:00 h). After the water injection was resumed, the level in the RPV increased and it stayed at ca. 2 m below TAF from ca. 108:00 h for the rest of the calculated transient (figure 11). At 97:18 venting took place. The pressure in the DW/WW decreased monotonically from 97:18 until 144:00 h (figure 12). At 144:00 the level in the RPV remains ca. 2 m below TAF and the debris remained inside the vessel at the bottom of the lower head, as shown in figure 13.

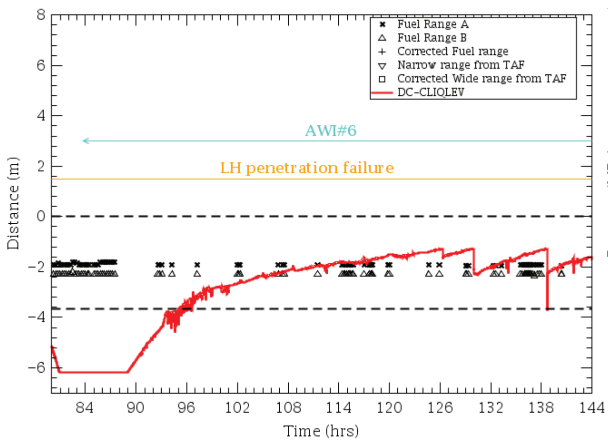

Figure 11: Downcomer collapsed water level

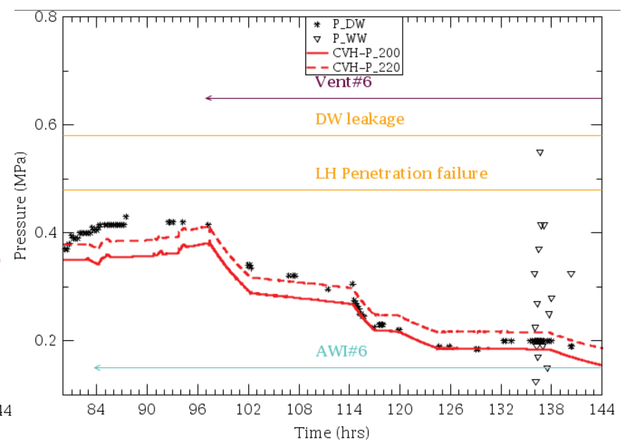

Figure 12: DW/WW pressure

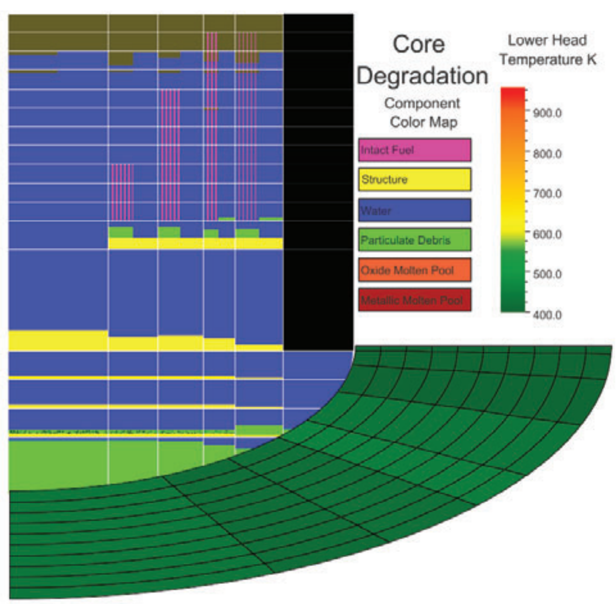

Figure 13: State of the core 6 days after SCRAM

4 CONCLUSIONS

The Fukushima Unit 3 sequence was simulated with the severe accident code MELCOR. A plausible sequence of the accident evolution in Fukushima U3 was presented. From the analysis the following conclusions can be made:

- The reactor was successfully cooled for ca. 29:00 h using RCIC and HPCI systems
- The HPCI operation was degraded before it was manually deactivated by the operators, and it is very likely that for a period only steam was being extracted and no water was being injected.
- Not all the AWI which was reported by the operators reached the RPV. Therefore, the alternative water injection rates and venting times were adjusted in the simulations. These were the calculations which reproduced the best the accident signatures
- The onset of hydrogen generation by cladding oxidation started before depressurization.
- A large amount of the core debris was relocated to the lower head according to these simulations
- The assumption of drywell leakage predicted combustible hydrogen concentrations near the time of the observed H_2 explosion. From these results it seems very likely that DW leakage occurred during the accident, especially shortly before the hydrogen explosion.
- There is a likelihood of a penetration failure, but the extent of the debris which was released from the RPV is uncertain.
- The analysis of the fission product releases may give a closer indication of the final state of the reactor and it is in consequence the natural continuation of the present study. This work is on-going.

The detailed analysis of the fission product release will be performed in the future and may give strong indications as to which are the most likely release path(s) when compared with the radioactivity measurements.

5 ACKNOWLEDGMENTS

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