Analysis of the accident in the Fukushima Daiichi nuclear power station

[redacted]

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. Paul Scherrer Institute (PSI) is taking part in an Organisation for Economic Cooperation and Development (OECD) project, Benchmark Study of the Accident at the Fukushima (BSAF) to reconstruct the events that occurred at the in March 2011. Eleven institutes from eight countries are participating. PSI is performing simulation of Unit 3, using the MELCOR code developed in the USA for simulation of whole plant accidents and made available to PSI via cooperative exchange agreement with the US Nuclear Regulatory Commission. The simulation task is a challenging one because only limited measurement data exist about the conditions inside the reactors

One of the important expected outcomes is an evaluation of the likely end-state of the reactor core which will help the owner of the damaged plant, the Tokyo Electric Power Company (TEPCO) to plan the removal of components from the reactor containment and the final decontamination. The exercise will advance the understanding of severe accident phenomena and contribute to further refinement of the computer models used to perform the simulations. The exercise will continue until September 2014. It is expected that results by each of the participants will be discussed at the final meeting, with a view to formulating a collective view of the accident sequences and reactor end-states.

Project goals

The events at the Fukushima Daiichi station underlined the need for maintaining vigilance in nuclear power operation but also a continued improvement in our understanding of severe accident behaviour and of the modelling tools used for accident analysis. BSAF (OECD/NEA/CSNI, 2014) thus provides an opportunity to exercise our modelling tools and expertise in use. BSAF also focusses attention on issues concerned with reactors with design features in common with the Fukushima Daiichi units.

The generic goals of BSAF are:

To extend the assessment base for code applicability to full scale commercial reactor plants and hence to identify areas for further improvement.

To address severe accident and accident management issues that were identified directly following Fukushima Daiichi.

The specific goals of BSAF are:

To **simulate the accident evolution for the period of six days after the initiating event,** and hence reconstruct as well as possible the event sequence.

To **estimate the likely end-state of the reactor units,** in particular the cores, in order to help plan the future investigation, decontamination and decommissioning operations.

PSI participation is defined by the specific goals of BSAF, **concentrating on Fukushima Daiichi unit 3.**

Work carried out and results obtained

The first step to perform the analysis was to make an extensive review of the available technical data, namely plant design, boundary conditions, accident data and uncertainties. The simulation task is difficult for all participants because so many of the components including measurement devices were not functioning normally, so that much of the plant data are incomplete or uncertain. Nevertheless, the most reliable or/and complete data for Unit 3 were identified. The main data that have been used for the present analysis are (TEPCO, 2014):

- The times at which the hydrogen explosions took place in each unit.
- The pressure history in the reactor (RPV) and in the containment (Drywell/Wetwell, DW/WW) have been identified as fairly complete and reliable data, which is fortunate because this serves 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 reactor itself is uncertain.
- The time when the operators vented the containment to control the pressure and hence avoid catastrophic containment failure, though unfortunately it is uncertain if all the venting operations were successful 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 (SNL, 2008) 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 in order to address the uncertainties.The input was imported into the visualisation tool SNAP in order to facilitate overview and manage analysis tasks.

All participants performed a case using the same set of boundary conditions; this case was designed as Common Case (CC). A progression of

Table 1

Case	HPCI	HPCI CST to WW	AWI	Venting	Forced venting	DW leakage	Penetration failure T	LH leakage
CC	CC	CC	CC	CC	–	–	–	–
C0	working	no	–	–	–	–	–	–
C1	degraded	no	–	–	–	–	–	–
C2	degraded	yes	CC	CC	no	no	–	–
C3	degraded	yes	Adjusted	Nominal	no	no	–	–
C4	degraded	yes	Adjusted	Adjusted	yes	no	–	–
C5	degraded	yes	Adjusted	Adjusted	no	yes	1273	no
C6	degraded	yes	Adjusted	Adjusted	no	yes	950	big
C7	degraded	yes	Adjusted	Adjusted	no	yes	950	small

modified cases (C0–C7) were performed in attempt to obtain the best estimate (BE), named the case that best reproduce the available measurements (e.g. pressure histories of the reactor pressure vessel, dry-well (DW) and wet-well (WW); downcomer (DC) water levels and the observed hydrogen explosion time. The performed cases as well as their main assumptions are shown in table 1.

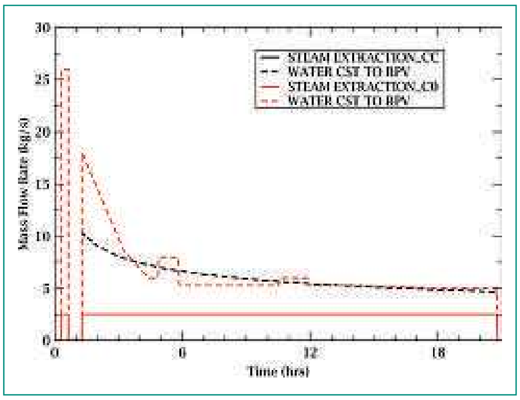
A summary of the main findings during the analysis is presented in the following sections:

RCIC and HPCI operation

The prescribed RCIC and HPCI water flows for the **CC** would be insufficient to recover to the levels measured. They are barely enough to take care of decay heat. A modified case was proposed, where the injected water was tuned manually, meaning the flow rates were adjusted according to the response of the water level, attempting to reproduce in the calculation what the operators did. This case was designated as C0. The flow rates for steam extraction and water injection during RCIC and HPCI operation are presented in figures 1 and 2 respectively. The assumed flow rates simulate how the operators are understood to have used the systems to control the RPV water level, in this way the thermal-hydraulic response during RCIC and early part of 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.

All cases assumed that HPCI operation started 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 calculation results suggest that the sprays were not enough to decrease the pressure in the DW/WW as shown in figure 4 (C0 and C1). The assumption that water was injected in 2 occasions from the CST to the WW, in addition to the sprays, was necessary in order to reproduce the pressure in the DW/WW (C2). 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.

After 29:00 h, the DC water level measurement stopped, the next available measurement was at ca. 37:00 h and it is below the Top of Active Fuel (TAF). In consequence, it is likely that the HPCI water injection stopped at some time after 29:00 h but the exact time when this happened is very uncertain. Had the HPCI continued to inject water



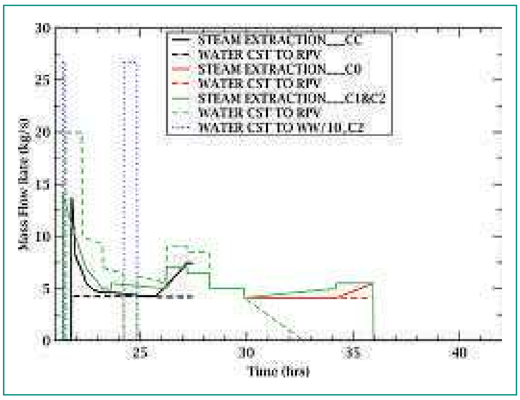
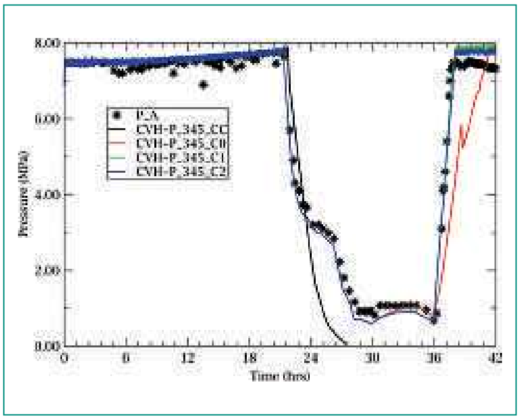


Figure 1 (left): *Steam and water flow rates during RCIC operation*

Figure 2 (right): *Steam and water flow rates during HPCI operation*



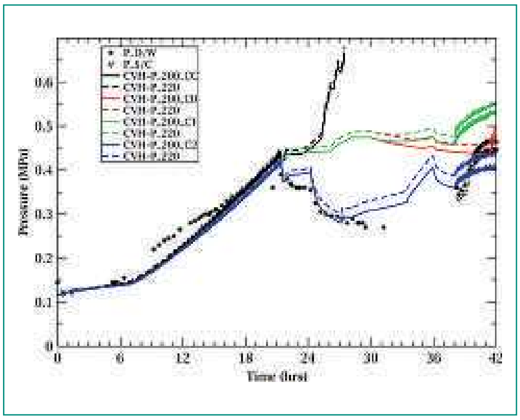


Figure 3 (left): *RPV pressure during RCIC and HPCI operation*

Figure 4 (right): *DW/WW pressure during RCIC and HPCI operation*

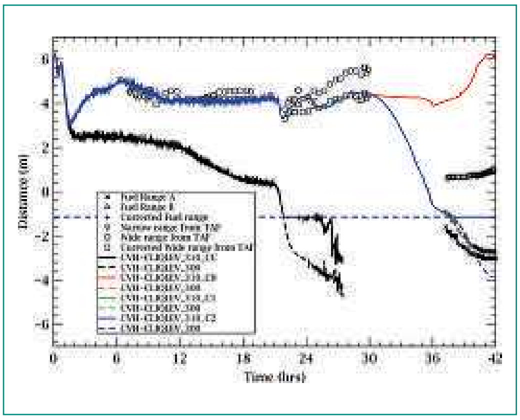


Figure 5 (left): *Downcomer collapsed water level during RCIC and HPCI operation*

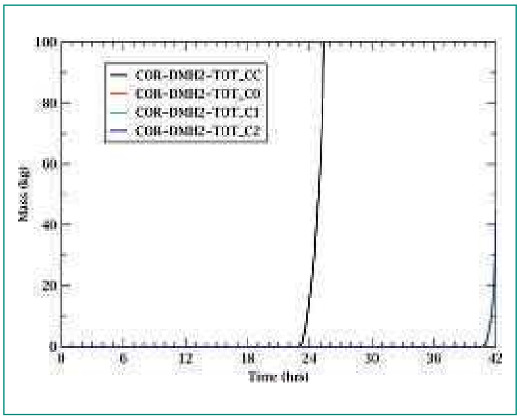


Figure 6 (right): *Hydrogen generation before depressurisation*

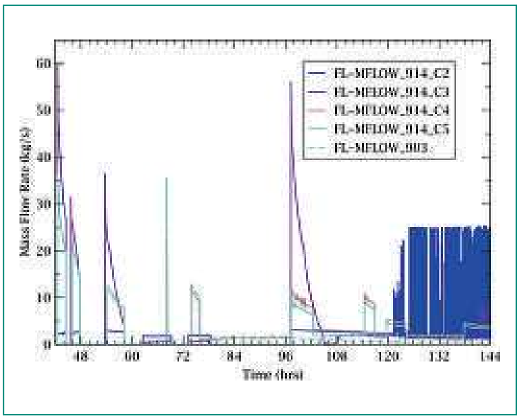


Figure 7 (left): *Vented mass*

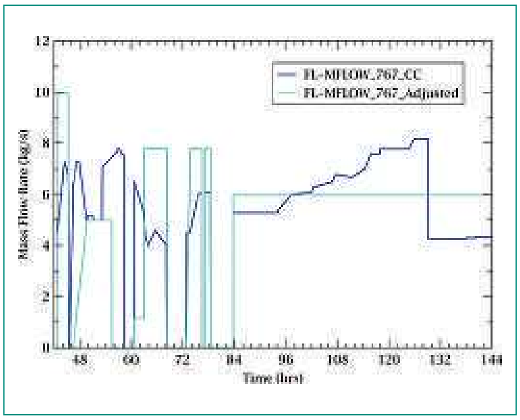


Figure 8 (right): *Alternative water injection*

to the RPV, the DC water level wouldn't have decreased as it was shown with C0 (figure 5). In contrast C1 assumed that HPCI water injection gradually stopped while steam was still extracted. C1 reproduces very closely the observed DC collapsed water level and the pressure in the RPV and supports the theory that the HPCI water injection was degraded after ca. 29:00. The onset of hydrogen generation by cladding oxidation started before depressurisation in the cases which assumed degradation of the HPCI operation (C1, C2). These cases reproduced the RPV pressure and the DC water level very closely to the measurements, reinforcing the theory that the water injection to the RPV stopped while steam was still being extracted during HPCI operation. C2 was able to reproduce very closely the pressure in the RPV and DW as well as the downcomer collapsed water level (figures 5); therefore the continuation of the study will be solely based on C2. However, it is uncertain if the HPCI could have started earlier than reported. Therefore it was identified as one uncertainty that should be address in future analysis.

Depressurization, alternative water injection and venting

According with the calculations, core degradation started at ca. 40:30 h, indicated by the onset of hydrogen generation. Around 45 kg of hydrogen were produced prior to depressurisation. Reactor pressure vessel depressurization was reported at 42:41 h, but would appear from the pressure measurements to have been initiated earlier. In the present analysis depressurisation was assumed to have occurred at 42:08 h, i.e. in order to match the drop of pressure in the RPV.

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. It was therefore assumed that all the venting had leaked to the building by routes not completely identified and that the rupture disk did not burst. C2 used the prescribed valve opening areas for the common case (CC). In this case the fraction of the opening area for motor valve (MO) situated before the rupture disk is only 3.5% and it was assumed that 100% of either the large venting valve (LV) or the small valve (SV) in the venting line where opened according to the reported timeline. For C3, the only difference is that the MO fraction opening was assumed to be larger ca. 60% and for C4 and C5 the venting timeline was used as guidelines, but the exact timing and opening fraction were adjusted by using the measured WW and DW pressure response as a target. Additionally, for C4, it was assumed that the large valve was as well opened in the time that only the small valve was reported to be opened (i.e. referred as forced venting) whereas C5 assumed that the small valve never opened instead DW leakage occur. The mass of steam and hydrogen that reached the top of the building either by venting (FL-MFLOW_914) or DW leakage (FL-MFLOW_903) is presented in figure 7.

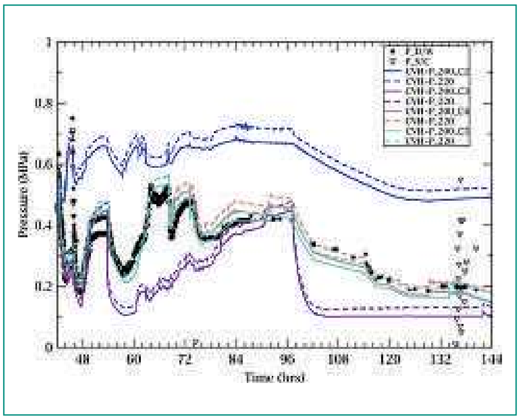
In parallel to the venting, the Alternative Water Injection (AWI) started by means of the fire engines. It is known when the operators reported to have injected water to the RPV, as well as the amount of water that they injected per day, but the actual amount that reached the RPV is uncertain. C2 used the prescribed values from the CC whereas for C3, C4 and C5 the AWI was adjusted following the pressure and the collapsed water level in the DC as guidelines. The AWI is presented in figure 8.

The proposed venting for C2 over predicted the pressure in the DW/WW (figure 9), whereas C3 under predicted it, indicating that a fraction in between 3.5–100% of the MO should have been opened in order to reproduce the pressure data. The pressure was very closely reproduced with C4 and C5 where venting was adjusted.

On the other hand the assumed AWI for C2 was enough to mitigate the accident progression as the collapsed water level in the DC was recovered after ca. 48:00 h (figure 10); however this was not observed in the FU3 sequence and is not consistent with the observed events later. The mismatch with the pressure measurements further confirm that not all the water that was injected reached the RPV. The assumed AWI for C3, C4 and C5 allow to reproduce the observed water level up to 66 h and the pressure signature was best reproduced by C4 and C5 where both venting and AWI were adjusted.

RPV failure, venting vs. DW leakage

The hydrogen generated by C2, C3 C4 and C5 is presented in figure 11. An explosion was observed at U3 building at 68:14 h. which is attributed to hydrogen generated by oxidation of metallic components in the degraded core of unit 3. 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 only the small valve was reported to be open. Although, the cases which considered this venting (C2 and C3), didn't reproduce the increase of pressure in the DW/WW. C4 assumed that initially the small valve didn't open and that shortly



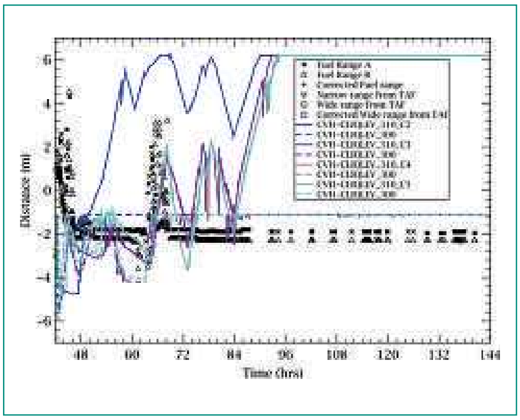
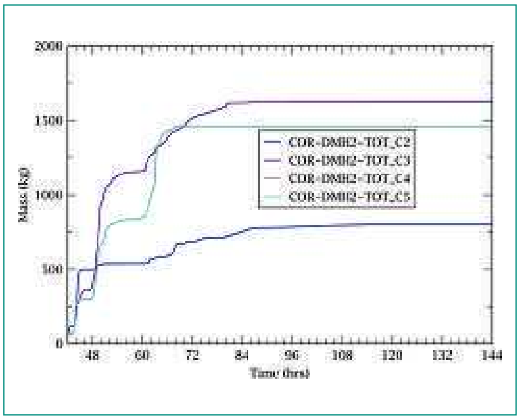


Figure 9 (left):
DW/WW pressure during AWI and venting

Figure 10 (right):
Downcomer collapsed water level during AWI and venting



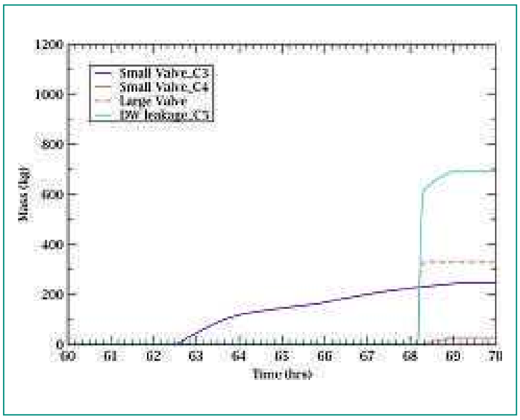


Figure 11 (left):
Hydrogen generation

Figure 12 (right):
Integral hydrogen reaching the building

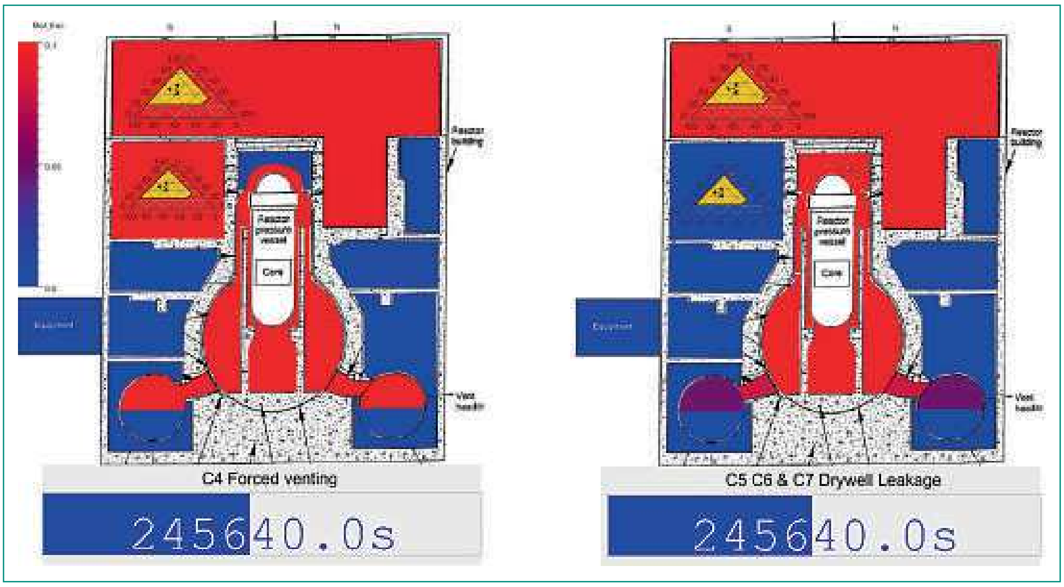


Figure 13: *Explosive conditions calculated in the building*

before the explosion the valve opened. However, the small valve by its own doesn't seem to have been sufficient to predict the large drop in pressure at ca. 68 h. Therefore it was assumed that the large was opened as well.

Another possibility is a leakage from the DW to the top part of the reactor building. This is possible if the internal overpressure is higher than the design pressure of ca. 0.5 MPa to cause the restraint bolts from the head flange to weaken and open a leakage path (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 depressurisation, 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 containment depressurisation can be fully explained by venting 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. C5 is based on this scenario. It is assumed that the first DW leakage took place at ca. 68:11 h and that the leakage was initially equivalent to an area of 0.04 m^2 and then it was reduced as the pressure decreased causing the leak to stop. A second event of DW leakage was assumed to take place at ca. 74:00 h, with an initial leakage equivalent to 0.016 m^2, this second DW failure assumes that the bolts never recovered completely again and that a small leakage of ca. 0.002 m^2 remained for the rest of the transient. The small leakage area is equivalent to the size of the small valve. The integral leaked mass by either venting or DW leakage in the hours before the hydrogen explosion can be seen in figure 12.

Combustible hydrogen conditions were calculated in the reactor building with C4 and C5 (figure 13) at about the time of the observed explosion (ca. 68:14 h.) in FU3; in contrast in the C3, with no DW leakage but venting leakage through the small valve in the venting line, the hydrogen concentrations in the building doesn't seem to have been enough to produce the explosion at the observed time. Moreover, the MELCOR model uses a very coarse nodalisation to calculate the concentration in the building. It may be that locally the concentration was even higher, in the hydrogen detonation regime. Furthermore, the integral amount of hydrogen leaked into the building in C3 was only ca. 350 kg and occurred progressively between 62:00–78:00 h, C4 predicted that ca. 400 kg where released to the building very shortly before the explosion and C5 released very quickly ca. 700 kg of hydrogen at ca. 68:14 h. The previous observations give strong reasons to believe that DW leakage was a major factor in the build-up of hydrogen that led to the explosion. The final part of the analysis will be solely based on C5.

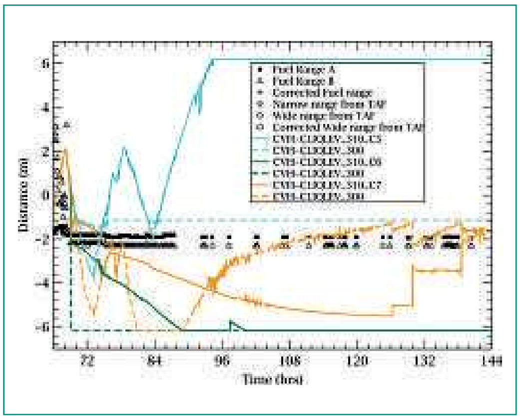
PCV failure, in- vessel vs. ex-vessel

The previous sections were devoted to the analysis of the in-vessel core degradation and the hydrogen explosion. The RPV may have failed, thus despite the continued AWI after 80:00 h, the water level in the DC was never observed to increase. However, the exact time, the extent or mode of the failure (if any) is unknown. The present section makes an attempt to evaluate the possibility of RPV failure by 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 COR radial nodes if certain temperature is reached at the location of penetration. In C5, the MELCOR default penetration failure temperature of 1273 K was assumed, but this case didn't predict any penetration failure. In consequence, the water level started to increase as soon as water injection was again available (ca. 74:00 h) as it can be observed in figure 14. For C6 it was considered that penetration failure occurs at temperatures of 955 K. C7 also considered that penetration failure occur at 955 K but the size of the penetrations were small and the leakage was forced to be just a fraction of ca. 1.0% of the full assembly.

For C6 and C7 penetration failure was predicted in rings 1 and 2 at ca. 68:57 h. In C6, the leakage was big enough to allow all the water injected to go out of the RPV. The measured pressures between 84:00-96:00 h were overestimated (figure 15) and all the debris which were relocated to the lower head (ca. 80 tons) were ejected into the cavity (figure 16). Nonetheless, the C6 is considered a bounding case (i.e. the maximum amount of corium that may have been on the cavity floor). In contrast C7, with the leakage of ca. 4 kg/s, allowed to reproduce the observed level measurement and remarkably close the pressure in the DW/WW (figures 14 and 15, respectively). In this case, the debris remained inside the reactor in the lower head (figure 16).

The assumed area of the penetration leakage as well as the temperature failure criteria was crucial, thus it makes the difference between an in-vessel



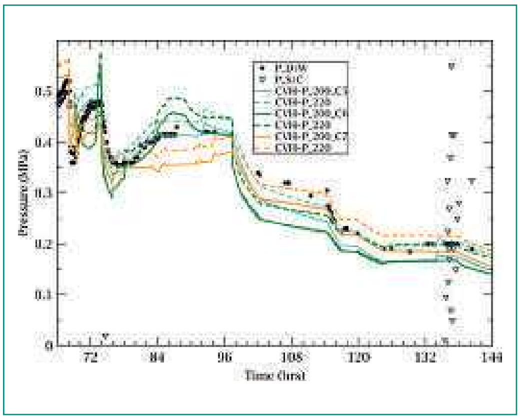


Figure 14 (left):
Downcomer collapsed water level during AWI and venting

Figure 15 (right):
DW/WW pressure during awi and venting

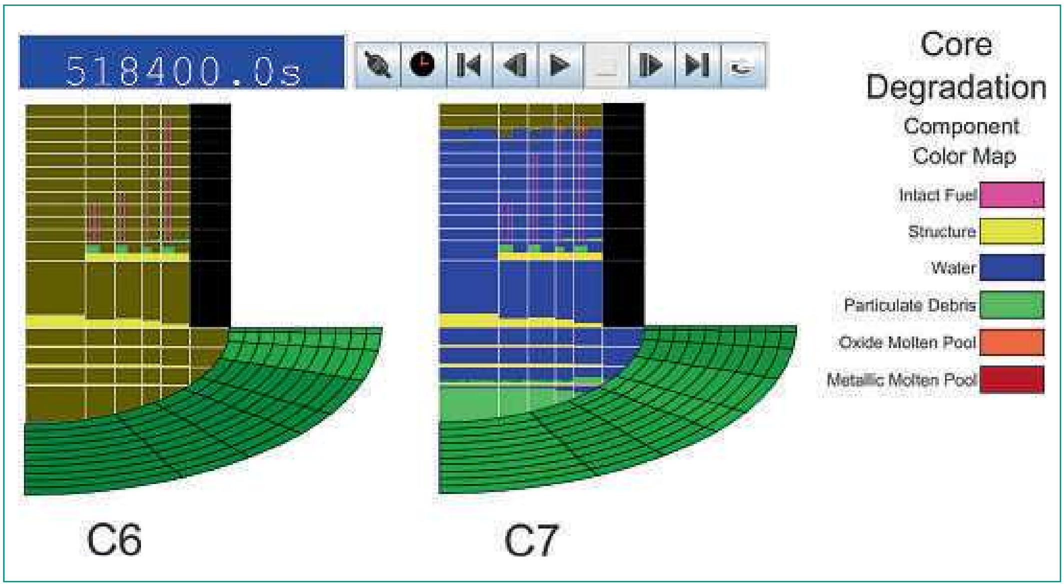


Figure 16:
Downcomer collapsed water level during AWI and venting

(with or without RPV water leakage) or an ex-vessel scenario. It is not certain if any of the debris/molten material were expelled out to the cavity, but the case that predicted the closest the measurements had the debris remaining inside the RPV (C7). Therefore, **C7 was the case submitted for the final report as the Best Estimate (BE).** These results indicate some likelihood that most of the debris remained inside the RPV. However, the calculation of penetration failure is stochastic and its occurrence greatly dependent on the failure parameters assumed by the user in MELCOR (i.e penetration failure temperature, size of the penetration). Furthermore, the predicted state of the core after 6 days of transient is still not fully stable; any possible reduction in the amount of water injected could further damage the core. The prediction or not of RPV failure has been identified as one of the main code limitations thus the assumptions made by the code user influence greatly the results.

Conclusions

The Fukushima Unit 3 sequence was simulated with the severe accident code MELCOR 2.1. The CC failed to reproduce the accident signatures from an early stage. An initial case was performed instead using the reported actions performed by the operators (C0). This case was adjusted step-by-step by means of modified cases in order to obtain one case or a set of cases which best replicate the measurements at the plant, and therefore are expected to best describe the accident sequence in unit 3. The main findings for the FU3 analysis are listed below:

RCIC seems to have operated normally when available, whereas it is very likely that HPCI degraded after ca. 29:00 h. The calculated results indicate that for a period only steam was being extracted and no water was being injected, but the exact time when water was no longer injected to the RPV is uncertain. In consequence hydrogen generation by cladding oxidation is believed to have started before depressurisation of the RPV. Additionally, it is likely that water was injected from CST to WW during the HPCI operation in order to decrease the pressure in the containment, albeit there was no mention of that action in the operator records. According to the calculation, sprays on its own wouldn't have been enough to decrease the pressure. However, due to the model uncertainties, alternative causes of the pressure decrease should be evaluated in future studies.

Best agreement with measured data (i.e. pressure in the RPV and DW/WW as well as collapsed water level in the DC) was achieved by adjusting boundary conditions relative to nominal values. In any case some uncertainty remains concerning the actual values. The calculations with adjusted AWI and venting are the ones that reproduced the best the accident signatures. Delivery of 100% of AWI pumped water to the reactor system would seem highly unlikely. The calculations results point out that only 30–60% of the nominal AWI was reaching the reactor. Furthermore, there is a high indication that the small valve in the venting line didn't open, had it opened the pressure in the DW/WW wouldn't have increased as observed.

The calculations suggest that there were two contributing pathways for hydrogen transport to the reactor building: leakage bypass to the building during venting of the WW and DW leakage. Probably both pathways took place at different times. The assumption of leakage from the DW to the reactor building during a period before the observed explosion gave the best agreement for DW and WW pressure signatures at this time, as well as the large accumulation of hydrogen in the upper compartment of the reactor building. There may have also been a pathway for transport to the reactor building via the venting line, but it do not seem sufficient on its own to explain the DW and WW pressure response.

A large amount of the core in form of debris seems to have been relocated to the lower head. Vessel failure is highly suspected to have taken place, but the results leave uncertainty in the size of the breach and the amount of core material ejected. It is not possible from the present analysis to estimate the exact amount of corium ejected from the RPV. The proposed cases C6 and C7 are believed to be bounding with the actual quantity somewhere in between.

The predicted state of the core after 6 days of transient is still not fully steady; any possible reduction in the amount of water injected could further damage the core.

Although, all the available measurements (i.e. RPV, DW/WW pressure and DC water level) were remarkably well reproduced by C7, there are still remaining uncertainties in some of the boundary conditions assumptions, chosen nodalisation and models as well as the uncertainty of measurements at certain periods of time. Code-to-code comparison analysis as well as comparison with different assumptions made in similar analysis with

MELCOR or with other codes would be required to address the uncertainties and to draw final conclusions on the final state of the core.

The fission product release is not part of the present analysis but conclusions drawn by the present study about the leakage will be the departure point in the evaluation of the fission release in the phase II of the project. Transport via venting of the WW is unlikely to have carried a large quantity of particulate material as that would largely be retained in the liquid. Transport via DW leakage would be expected to have carried any particulate present in the gas and hence a potentially much larger release of aerosol-borne fission products such as cesium to the environment. It is therefore crucial to reach an understanding of the transient from the hydraulic pathways point of view before any detail analysis of the FP can start. The future analysis of the fission product releases may shed additional light on the final state of the reactor and consequently the natural continuation of the present study.

National Cooperation

None.

International Cooperation

The project is coordinated by the OECD Nuclear Energy Agency (NEA). The Operating Agent (OA) is Japan Atomic Energy Agency (JAEA) who is technically supported by the Japan Institute of Applied Energy (IAE). The eleven participants (from Japan, France, Germany. Korea, Russia, Spain, USA, and Switzerland (PSI)), each cooperate formally with NEA and OA. There is informal cooperation between the participants.

Assessment 2014 and Perspectives for 2015

For the OECD BSAF project, the progress during 2014 has been slower than originally planned. The timeframe of the project was extended until the end of 2014. Despite the delay in the OECD BSAF project, PSI work has progressed according to the plan. A definitive common case calculation (CC), plus best estimate and appropriate sensitivity calculations were performed in the first half of 2014. The common case simulation was submitted to IAE on time by end of May 2014. The final best estimate calculation was submitted in August 2014. From these calculations the OA, IAE, compiled a draft report, which was sent to the participant mid-November 2014. The report will be reviewed and finalised by end of 2014. The final meeting for BSAF phase-I have taken place 24–26th November 2014. The final meeting was immediately followed by the kick-off meeting of BSAF Phase-II, from 27–28th November 2014. It is intended that phase-II will address the open issues remaining from the phase-I, special attention will be taken to the transport of the Fission Product Release during this phase of the project.

Publications

Submitted: *Fernandez-Moguel, L. and Birchley, J.* Analysis of the accident in the Fukushima Daiichi nuclear power station Unit 3 with MELCOR_2.1 Annals of Nuclear Energy.

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