The OECD-NEA Benchmark Study of the Accident at the Fukushima Daiichi nuclear power station (BSAF) project – Phase 2

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Duration of the Project	2013–2017

ABSTRACT

The OECD-NEA BSAF project is aimed to evaluate and analyse the likely end-state of the reactor core after the accidents at the Fukushima Daiichi nuclear power stations. The main objectives of the project are to assess the status of the reactors and the distribution of the radioactive material inside the plants to support decommissioning and to further develop severe accident analysis methods. Phase 2 of the project is currently on its final stage.

For the Phase 2, the analysis is focused on the **hydrogen generation** and potential for combustion as well as the **source term analysis** and comparison with the measured activities and dose rates at relevant locations at the plant and in the plant vicinity. In addition, the duration of the analysed sequence was extended to 20 days from the accident initiation. Phase 2 of the project started in 2015 and is planned to end in 2018.

PSI is using MELCOR 2.1 as the main tool for the system level simulation of the sequence in unit 3 during phase 2 of the BSAF project. During the present period, some modifications were made to the input concerning the injection point of alternative water injection. In addition sensitivity cases were performed to evaluate the late phase of the accident (8–15 days). One sequence was selected as the best estimate.

The selected sequence predicted remarkably well the main signatures (i.e. pressure in RPV and containment and containment water level). In addition the selected sequence shows similar trends with the measured temperatures for the late phase of the accident. After 14 days it was estimated that ca. 60% of the core materials were debris or molten material, from which 85% remained in the reactor vessel lower head and 15% were expelled to the pedestal.

The estimated source term is consistent with the reverse calculations from Katata et al. (2015) with WSPEEDI where 4.69 PBq (MELCOR 2.1) vs 5.15 PBq (WSPEEDI) of I-131 i.i. were released to the environment in the period between 42–54 h. In addition, the prediction of I-131 and Cs-137 released to the reactor building and turbine building through the Primary Containment Vessel (PCV) water leakage is consistent with the estimation from Hidaka and Ishikawa (2014).

The lack of an iodine chemistry model in MELCOR was identified as a major issue, as possible gas iodine coming from the suppression pool for the Wetwell vent as well as the iodine releases coming from the accumulated water in Wetwell, Drywell, Reactor building and turbine building were neglected.

Project goals

The Project OECD-NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) [1] Phase 2 is intended to extend the scope of the analysis performed in phase 1 [2] to include the **hydrogen generation** and potential for combustion as well as the **source term analysis** and comparison with the measured activities and dose rates at relevant locations at the plant and in the plant vicinity. The following main objectives were addressed:

- To extend the analysis time span from the *6 days* in Phase 1 to until the end of March, 2011, or to approximately *20 days* from the accident initiation (the earthquake).
- To extend the scope of the accident analyses of Fukushima Daiichi units 1–3 to include the amount of *hydrogen generated.*
- To extend the analysis to include the *fission product release* from the core, the retention in and transport through the units (reactor system, containment, reactor auxiliary buildings) and release to the environment, and thereby provide guidance on the level of contamination likely to be encountered during ongoing operations at and in the vicinity of the station. For this, the necessary models/nodalization have to be developed;
- To extend the analysis to consider the buildings adjacent to the containments, such as the reactor building, to evaluate the *hydrogen effects and the source term transport.*

The analysis at Paul Scherrer Institute (PSI) concentrated on Unit 3, as during BSAF Phase 1. The main tool for the system level simulation of the sequence was MELCOR 2.1.

Work carried out and results obtained

During the present period, the input deck used in the 2016 period was modified. The modifications were made concerning the injection point of alternative water injection. In the previous input [3], the Alternative Water Injection (AWI) was injected through the core spray system (i.e. at the top of the core). In the present study, the injection point was corrected to be in the feed water injection as shown in figure 1 (i.e. through the downcomer). This may have an impact in the reflooding progression as it is well known that bottom injection is less effective than top and bottom injection. The considered fission product release paths in the present calculation are represented in figure 1 as follows:

From Reactor Pressure Vesel (RPV) to Drywell:

1. Through the Safety Relief Valves (SRV) discharging to the Suppression Chamber (SC).
2. Lower head RPV penetration failure

From Drywell to Reactor Building (RB) and environment:

3. Venting through the stack
4. Drywell head flange leakage
5. Main Steam Isolation Valve penetration

The main degradation events calculated with MELCOR are presented in table 1 (the events marked with an '*' were assumed to take place at that time). In addition, the sequence was extended to 14 days after the scram. Sensitivity cases were performed to evaluate the late phase of the accident progression, and an estimation of the source term was performed. The results were compared against the

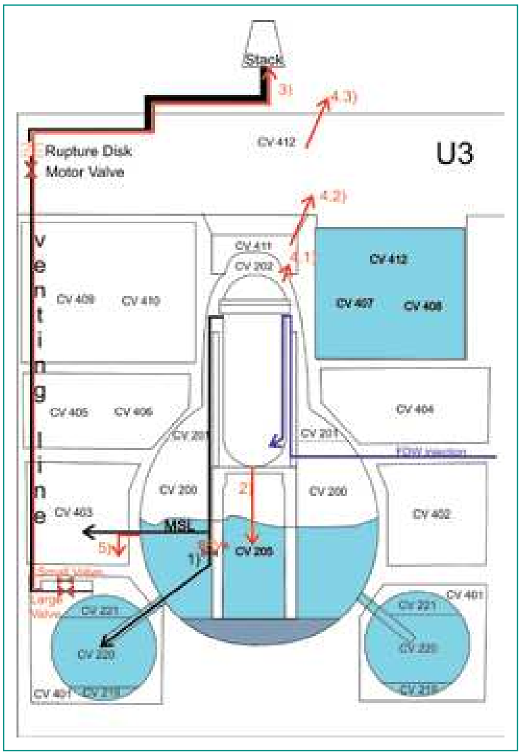

Figure 1: *PCV and RB nodalization and FP releases paths*

Main degradation events	Calculated/ Assumed* t (hr)
H2 generation onset	40.58
DW head flange leakage	68.13*
Debris relocation to lower head	69.52
RPV lower head failure (penetration)	73.03
Debris discharged to containment	82.40
DW MSIV penetration fail	209.18*

Table 1: *Main degradation events*

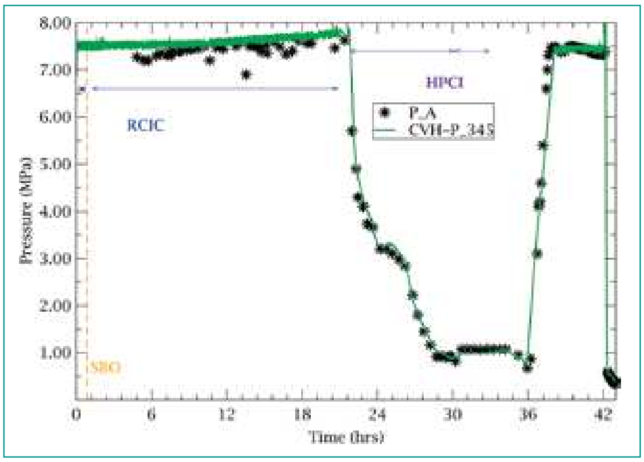

Figure 2: *RPV pressure*

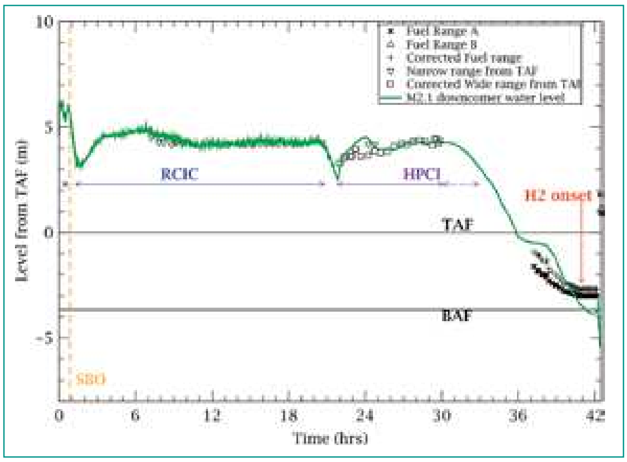

Figure 3: *Downcomer water level*

main signatures, namely RPV and PCV pressures and downcomer water level. For the fission products, the source term was compared with the reverse calculations from [4].

Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (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) was available from 0.18 h until 20.49 h, with an interruption between 0.38–01.16 h. The RPV pressure was maintained within the normal band through the SRV operation which discharged the steam into the Wetwell. The water flow rate to the RPV was manually adjusted in order to gradually increase the downcomer water levels. The High Pressure Coolant Injection (HPCI) started at ca. 21.48 h in the calculations. As with the RCIC, the operators were manually controlling the steam extraction and the water injection to the RPV.

For the calculation, 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 addition it was assumed that the water injection started to degrade at ca. 30.00 h and completely stopped at ca. 34.00 h. In this way, the thermal-hydraulic response during RCIC and the HPCI operation was well reproduced as it is shown in figures 2 and 3.

Core uncovery was predicted to have started after the end of the HPCI operation. The water level in the downcomer reached the Top of Active Fuel (TAF) at ca. 36.00 h, at about the same time the HPCI operation stopped causing the RPV to repressurise. The cladding degradation (i.e. hydrogen onset) started at ca. 40.53 h after scram.

PCV transient after depressurisation until 4 days after SCRAM

Reactor depressurisation was assumed to take place at 42.14 h by Automatic Depressurisation System (ADS). In the input it was manually activated by the opening of Safety Release Valves (SRV's). The SRV's were continuously open for the remaining time of the transient. At the time of ADS initiation, temperatures of ca. 1300 K had been reached and ca. 42 kg of hydrogen had been produced. The predicted Primary Containment Vessel (PCV) pressure is in very good agreement with the measurements as can be seen in figure 4. The water level in the downcomer region is presented in figure 5. This measurement is less reliable as it is possible that some water has flashed in the reference leg during the core uncovery, therefore the comparison is only qualitative and more emphasis is given to reproduce the pressure signatures. Figures 4 and 5 include the alternative water injection (AWI), magenta arrows, and containment vent actions, marron arrows, which were performed by the operators. Although there were a 4th and 5th vent reported by the operators, it is considered that they didn't take place; therefore they were neither considered in the calculation nor shown in the figures. There were several containment pressure responses to the various operator actions in this period. In addition, several dry-out events and subsequent reflood actions have taken place at different times. Figure 6 shows the MELCOR evolution of the core

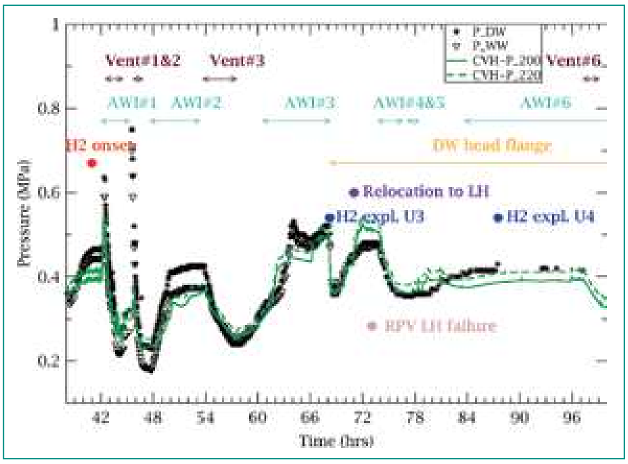

Figure 4: *PCV pressure after depressurization*

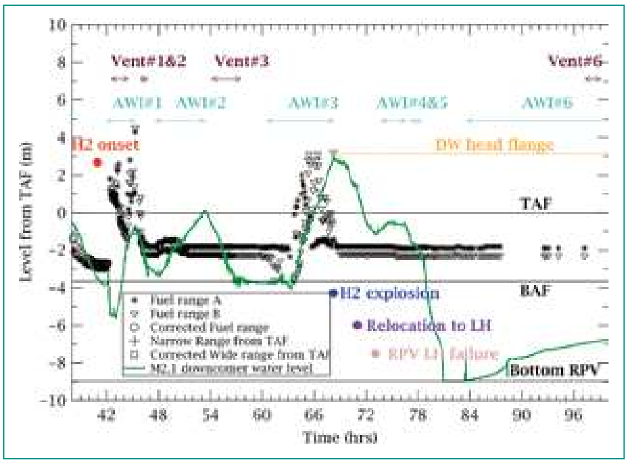

Figure 5: *Water level after depressurization*

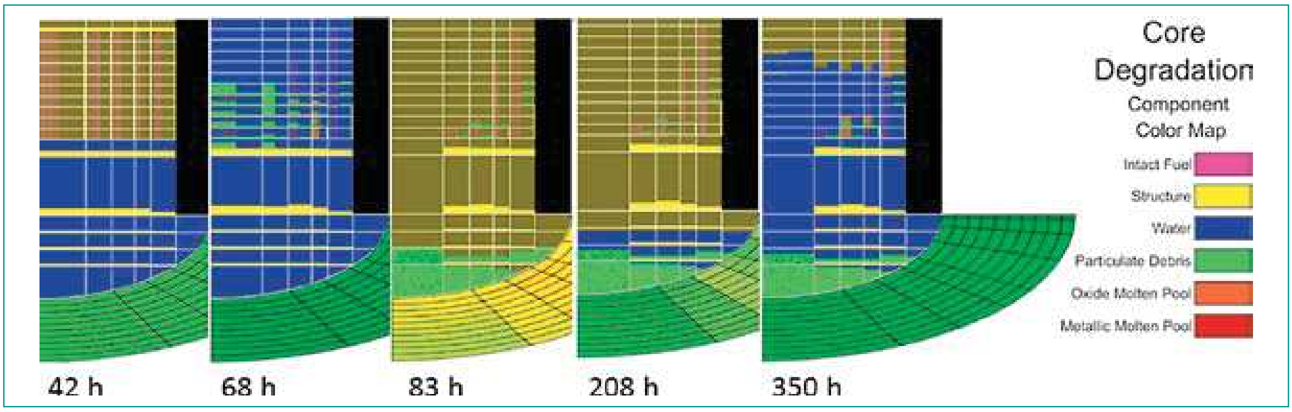

Figure 6: *Evolution of core degradation*

degradation, where the intact rods are represented in pink and the debris are represented in green. Shortly before the explosion in unit 3 (ca. 68 h) the core has suffered major relocation of rods which have formed debris. The debris are being collected on the support plate.

The core support plate was estimated to have failed at 69.52 h causing relocation of debris to the lower head. Penetration failure of the pressure vessel lower head was estimated at 73.03 h causing only water leakage out of the reactor. At ca. 82.40 h 15 % of the debris was predicted to be discharged to the pedestal, but the water present in the cavity prevented the onset of molten core concrete interaction (MCCI).

It is possible that during the accident, major core relocation to the lower head took place earlier than in the calculation. This may have also produced a penetration in the lower head to fail and it may have been the trigger to the DW head flange overpressure failure shortly before the hydrogen explosion. In the calculation it was assumed that DW head flange leakage occurred at ca. 68.13 h, causing the drastic drop in the containment pressure and releasing hydrogen from the drywell to the reactor building. The containment pressure increase between 68.00 and 72.00 h may have been due to the evaporation of the water above the debris in the lower head. After 68.00 h the water level in the downcomer remains always below TAF. Despite the constant AWI after ca. 80 h the water level didn't increase, giving a further indication that the injected water may have been leaking from the RPV.

PCV transient after depressurisation from 4 until 14 days after SCRAM

Figures 7 and 8 show the containment pressure and the reactor downcomer water level, respectively, for the late phase of the accident scenario, i.e., 100–350 h. There was an additional pressure increase in the containment at ca. 6 days after SCRAM. In the calculation, it was assumed that any vent action as well as the drywell head flange leakage stopped.

In addition the AWI was decreased at ca 152 h. Due to this decrease in the coolant flow, the lower head was uncovered. The penetration leakage at the

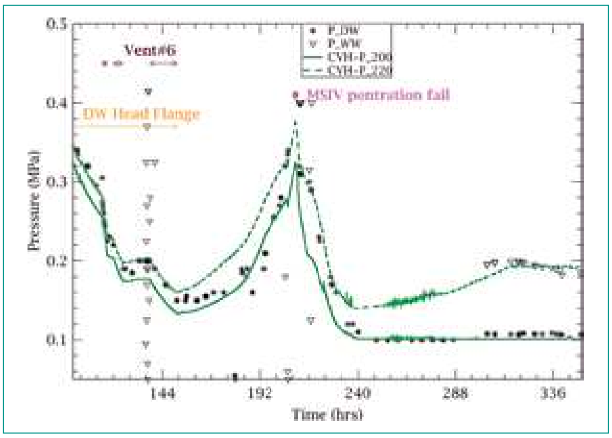

Figure 7: *PCV pressure 4-15 d after scram*

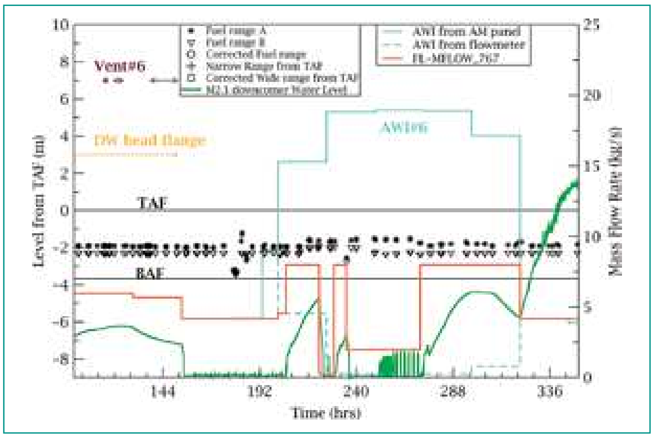

Figure 8: *RPV water level 4-15 d after scram*

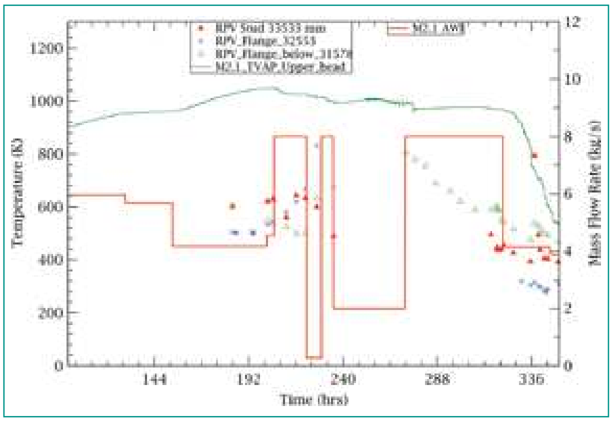

Figure 9: *Measured RPV wall temp. vs calculated steam temp. in M2.1 upper elevations*

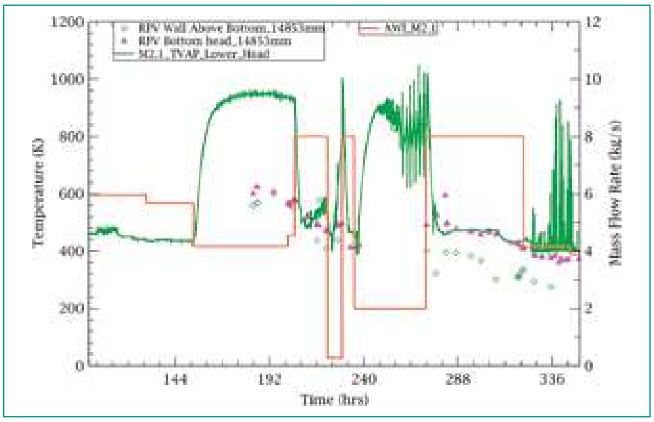

Figure 10: *Measured RPV wall temp. vs calculated steam temp. in M2.1 lower elevations*

bottom of the RPV is shown to be leaking steam from the RPV to the DW in the calculations. In consequence, the DW pressure starts to increase.

There were two measurements to estimate the AWI; the AWI from Accident management panel (i.e. solid turquoise line), and the AWI measured from the flowmeter (i.e. dashed turquoise line). In the period between 192–320 h the AWI differs by orders of magnitude between the different measurements. Therefore the assumed AWI shown in figures 7 and 8 (i.e. red solid line) was chosen based on the measured temperature behaviour inside the RPV and PCV existing only from ca. 190 hrs [5]. Figures 9 and 10 show the comparison between the estimated steam temperatures with MELCOR and the measured values in the RPV upper and lower elevations, respectively. The calculated temperatures with MELCOR follow the trends of the measured values with the assumed amounts of AWI.

Water level in the Drywell and Wetwell

Figure 11 shows the water level in the drywell and the wetwell. The black dashed line indicates the elevation where the Main Steam line isolation Valve

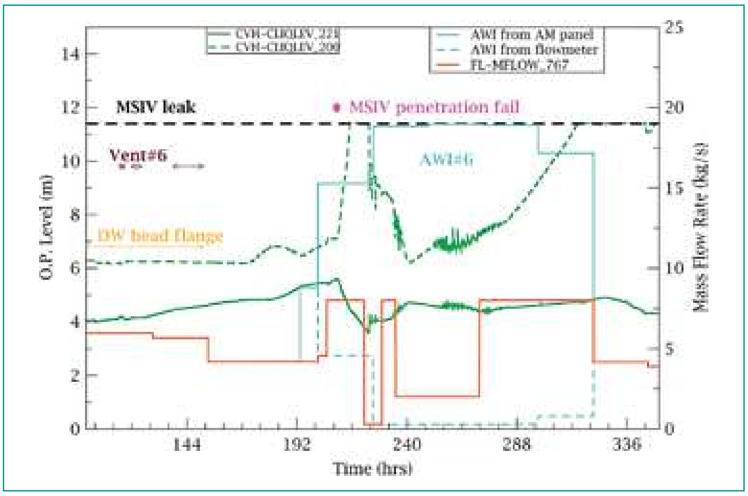

Figure 11: *Wetwell and drywell water level 4-15 days after scram*

(MSIV) containment penetration is located. It has been confirmed that water started to leak through this penetration in the late phase of the accident, however the precise time when this leak started is unknown. During the timeline of the accident published by OECD/NEA [6] it is mentioned that on March 24th (ca. 297–321 h) 3 workers were exposed to high levels of radiation in the unit 3 turbine building. The exposure was likely due to the

Table 2: *Hydrogen production and transport*

	Produced in U3	Leaked to U3 refueling bay	Vented to stack	Leaked to U4 building (35 % of vent)	Leaked to U4 building (20 % of vent)
Before U3 explosion	1188	447	657	230	131
After U3 explosion	102	152	14.5	5.06	2.89
Total	1291	599	672	235	134

highly contaminated water coming from the Unit 3 containment which may have reached the turbine building. Therefore, this event may be an indication when the water leakage started between 297–321 hrs or earlier. In the presented calculation the water was estimated to be leaking from the containment to the reactor building within that time frame.

Hydrogen production and transport

One of the open questions from BSAF-I was the amount of hydrogen needed for the explosions in unit 3 at 68.14 h and in unit 4 at 87.30 h. The hydrogen generated in Unit 3 between 42.00 to 87.30 h is believed to be responsible for both explosions as units 3 and 4 share a vent line discharging to the common stack. It is believed that the hydrogen which was transported from unit 3 to unit 4 was transported through the venting line and caused the explosion in unit 4.

In [7] it was estimated that ca. 130 kg of hydrogen was responsible for the explosion in unit 1. There is not a similar study for unit 3, however, at least the same quantity of hydrogen would have been needed for the explosion in unit 3. Table 2 shows the amount of hydrogen that was calculated to have been generated in Unit 3 in the timeframe relevant for H_2 explosions in units 3 and 4, i.e., between 42 and 87 hours. In unit 3, the assumed drywell head flange leakage allowed ca. 450 kg of hydrogen to reach the refuelling bay at the time of the explosion. This may have been enough to produce the explosion in unit 3 (3.5 times the amount estimated in unit 1). In [8], it was estimated that between 20–35 % of the vented hydrogen could have been diverted to unit 4 reactor building during the vent actions, and this would correspond to 130–230 kg of hydrogen in the present estimation. Therefore, both explosions could be explained with the proposed scenario, giving a certain degree of credibility to the sequence. However, this estimation assumes that a large fraction of hydrogen was in the same room in the reactor building.

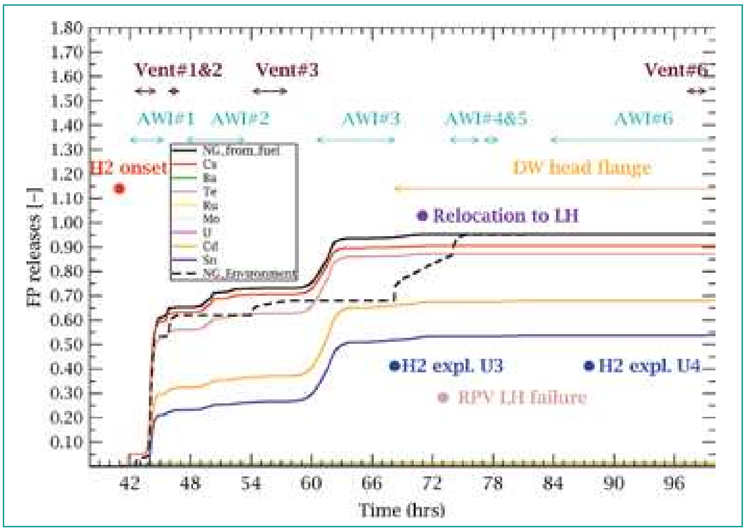

Figure 12: *FP releases representative classes*

Fission product releases until 4 days after SCRAM

Figure 12 shows the fission product releases from fuel for the representative classes in MELCOR. In addition, the figure includes the release of noble gases to the environment (black dash line). There are 3 main releases from fuel which correspond to the time when the water level in the downcomer is below TAF at ca. 42.00, 47.00 and 60.00 h. The main noble gas releases to the environment correspond to the vents at ca. 42 and 45 h and the explosion in unit 3 at 68.14 h.

Figure 13 and 14 shows the releases from fuel and to the environment, respectively, for the 3 different Cs classes, CsOH, CsI and Cs_2MoO_4. The entire fission product released from fuel will be transported to the suppression chamber as long as the RPV remains intact. The steam temperatures in the SRV discharge line before entering the suppression chamber were below 870 K in the time frame where Cs species were released from fuel (i.e. 42.00–62.00 hrs). Under these conditions the Cs injected in the pool was mainly as aerosol form, as observed earlier in [9]. Iodine was considered to be released as aerosol (as CsI), elemental or organic iodine was not considered in the present model.

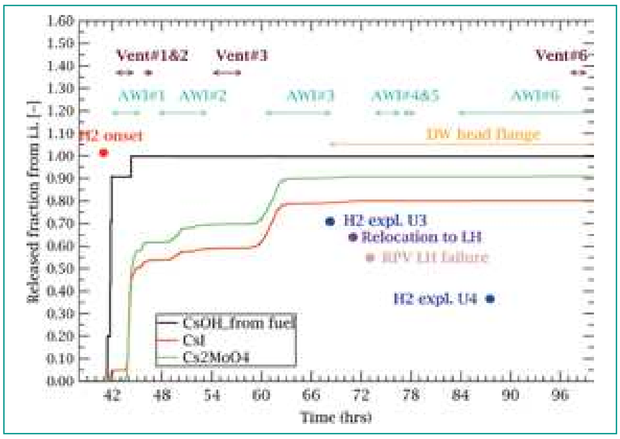

Figure 13: *FP releases Cs classes from fuel*

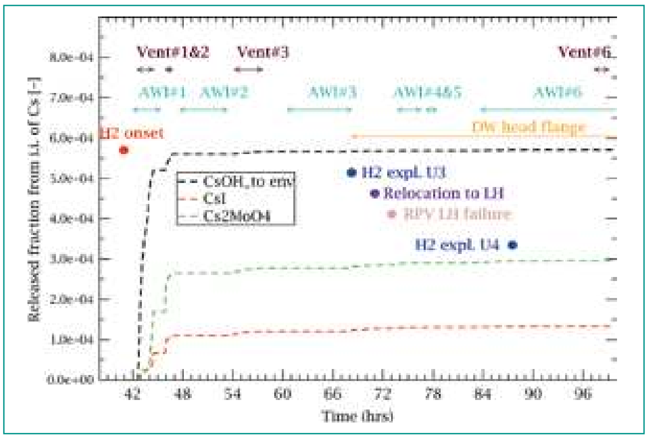

Figure 14: *FP releases Cs classes to environment*

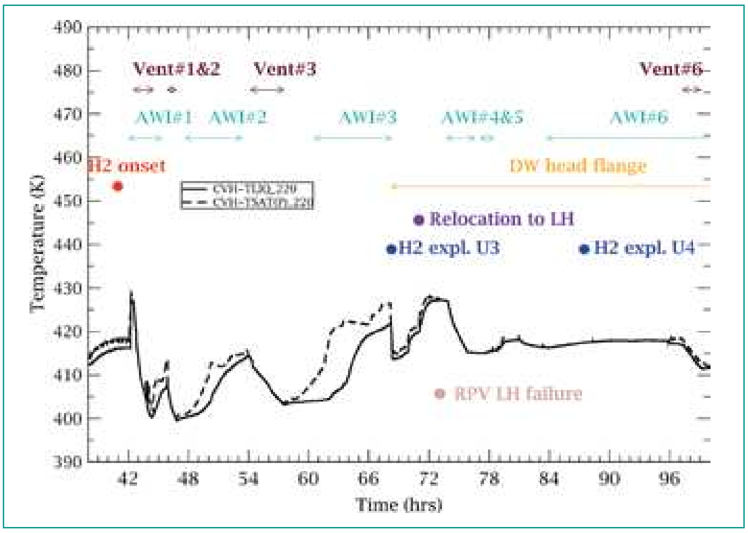

Figure 15: *Wetwell temperature*

The retention in the suppression chamber pool depends on the aerosol particle size but not the composition, therefore, the retention is the same for the different species under the same conditions in the pool (i.e. pressure, temperature and water level) provided that the aerosol particle size is the same. However, the release from fuel for the different species takes place at different times thereby explaining the difference in the retention.

The 1st containment vent from the Wetwell took place between 42.5–44.5 h. The Wetwell is at saturation condition, figure 15, for almost the entire 1st vent duration from 42.5–43.7 h. In this time frame 90% of CsOH, 50% of CsI and 2% of Cs_2Mo are released from fuel.

The majority of the releases for Cs_2MoO_4 from the fuel take place during the time when the containment is not being vented and the suppression chamber is not at saturation conditions. Therefore, the majority of Cs_2MoO_4 aerosols are retained in the pool. The second vent took place between 45.9 h–46.8 h. The pool is boiling for the entire duration of the vent. By this time only an additional 3% of CsI and 1.7% of Cs_2MoO_4 are released from the fuel. At the end of the second vent a total of 0.1% of the total Cs and 0.22% of iodine have been released to the environment.

There is no significant release from the fuel at the time the 3rd vent takes place between 54.00–57.40 h. The water level in the core region is covering the active fuel at 54.00 h as shown in figure 6. The core was further uncovered after the interruption of AWI at ca. 54.00 h reaching the bottom of the active fuel at ca. 58 h. Further releases from fuel took place only after 60 h. The pressure in the suppression chamber increased; as a result the saturation temperature of the suppression chamber is higher causing the pool to be subcooled. In consequence, most of the fission products released from fuel in this time period are retained in the pool.

The drastic PCV pressure drop at ca 68.14 h is believed to have been caused by the drywell head flange leakage thereby potentially providing a direct path for fission product transport from the containment to the reactor building and to the environment. Nevertheless, by this time, most of the Cs and I would have been already retained in the suppression pool (ca. 68% and 74% of i.i., respectively).

Cs-137 and I-131 distribution 14 days after the scram

Late in the accident, there was a water leakage from the containment to the turbine building causing release of Cs and iodine carried with the contaminated water leaking through the MSIV containment penetration as explained previously. Table 3 shows the comparison of the distribution of Cs-137 and I-131 at ca. 350 h with the estimation made by Hidaka and Ishikawa [10] based on measurements of the amount of water in the turbine building, and

Cs and iodine concentrations in the water. The results of the calculations and the estimate based on the measurements seem to be in good agreement noting that the data of Hidaka and Ishikawa are from the end of April 2011 and by this time, further transport of iodine and Cs may have taken place with the coolant water flowing from the unit 3 containment to the turbine building.

Table 3:
CS-137 and I-131 distribution

Releases	MELCOR 2.1 I-131 (PBq)	Katata et al. I-131 (PBq)
1st vent	2.58	0.91
2nd vent	1.68	2.96
3rd vent	0.43	1.28
Total 1st, 2nd and 3rd vent	4.69	5.15
U3 hydrogen explosion	0.27	2.0
Total releases during the accident	–	150.2

Fission product releases from MELCOR vs. WSPEEDI reverse calculation

Table 4 shows the comparison of the activity I-131 obtained with MELCOR 2.1, against the obtained activity with the reverse calculation using WSPEEDI-II [4]. This simulation system calculates air concentration and surface deposition of radionuclides and radiological doses by the successive use of a meteorological prediction model and a Lagrangian particle dispersion model.

One limitation of WSPEEDI is that it estimates average releases over long periods of time (i.e. 0.5–3.5 h). The MELCOR estimation for iodine is in the same order of magnitude compared with the WSPEEDI-II calculations. The releases corresponding to individual vents are different, but the integral value is in very good agreement for the period were vent 1, 2 and 3 took place; providing a good quantitative comparison.

Limitations in the modelling of the Fission product releases

Despite the very good agreement against the RPV and containment pressures, water level and the inverse WSPEEDI-II releases, following limitations with the MELCOR modelling of the fission products were identified:

The reverse WSPEEDI-II calculations have not enough resolution to show the shape of the individual releases. The estimations are based on land and sea measurements over long periods of time and the estimated releases are continuous. Therefore, it is uncertain which integral value should be compared against the main releases with the MELCOR predictions.

Iodine was represented in the MELCOR model only as CsI compound. Therefore it was released from fuel as aerosol and a large fraction was retained in the suppression chamber (59% from i. i.). However, it is possible that a part of it was present as gas phase iodine (I_2 and organic iodides) which could have been released to the atmosphere during the containment vents.

MELCOR doesn't calculate significant releases after 74 h. Although the RPV is calculated to be completely empty in certain time periods (see figure 5), the calculated temperatures of both the remaining fuel and the debris were always below 1500 K. This is below the temperature needed to release fission products according to the model. During the accident, leaching from debris/aerosols may take place, adding additional fission products into the contaminated water which in the late phase of the accident was transported to the reactor building.

Additionally. ca. 10% i.i. of iodine was present in the DW water and ca. 11% was transported with the contaminated water to the reactor building in the late phase of the accident. This represents an additional source of iodine that could have been released to the atmosphere as gaseous iodine. In [11] it was estimated that after March 14th slightly more than half of all of the 131I releases to the atmosphere from the Fukushima accident (for the

Table 4:
U3 atmospheric releases comparison.

Location	Fraction [% i.i]		Hidaka& Ishakawa 2014 [% of i.i.]	
	Cs-137	I-131	Cs-137	I-131
Released from fuel	90.8	80	-	-
In the reactor pressure vessel	12.5	0.009	-	-
In the water in the suppression pool	55.8	59	-	-
In the water in the drywell	9.5	10	-	-
In the water in the auxiliary building	10.8	11	18.0	26
Released to the atmosphere	0.1	0.2	-	-

whole period March 14–31) would have been in a volatile form. These releases are neglected in MELCOR 2.1 as no iodine chemistry is included in the modelling.

Improvements in the MELCOR modelling derived from BSAF-I and -II

One of the objectives of BSAF-I and -II was to further develop the severe accident codes employed in the calculations. Therefore, MELCOR developers implemented several improvements to the code including:

- Improvement of run time which enabled the extension of Fukushima simulation time.
- Fuel rod collapse model. This model was available in MELCOR 1.8.6 but it was not a default option. In MELCOR 2.2 this model is default. In addition the model was improved to eliminate the temperature threshold effect from the previous model.
- To improve the representation of the RCIC operation the Terry turbine model(s) were implemented in MELCOR.
- The quench front velocity has been revised in order to prevent the code from producing unphysical pressure oscillations.
- A debris cooling models and spreading model for the CAV package.

In addition SANDIA is progressing in the development of the following models:

- Vapour condensation/hygroscopic model. Multiple aerosol components (i.e. chemicals or material) can condense or vaporize instead of just one component which is typically water.
- Aerosol re-suspension model. MELCOR will allow for each aerosol component to have user specified material density from a set of material densities. This will significantly affect gravitational settling removal of different aerosol components with different material densities. Algorithm is already developed. Testing, incorporation into MELCOR, and documentation remain on-going.
- Eutectics model. The eutectic model which was not functioning since version 1.8.5 has been reviewed and corrected. The model only applies to conglomerate; the model is almost ready for beta testing.

The changes were implemented in the new version MELCOR 2.2.1 which was promised to be released on December 2017 by SANDIA, together with the updated manual and two additional documentation volumes; volume III containing the assessments of MELCOR and volume IV which will provide a modelling guide.

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 participants are from Japan, Canada, China, Finland, France, Germany, Korea, Russia, Spain, USA, and Switzerland, each cooperate formally with NEA and the OA. There is informal cooperation between the participants.

Assessment 2017 and Perspectives for 2018

Work at PSI is progressing as planned. During the previous period, it had been reported that attempts were made to use the new MELCOR 2.2 code version which presumably allows faster runs and is more stable for long term calculations; however it was not possible to obtain a similar sequence with this code version. The results presented in this report were made only with MELCOR 2.1.

PSI has performed additional sensitivity cases in the long term analysis. The obtained calculation was in good agreement with the existing data. The analysis of the transport of fission products and hydrogen, the comparison with the source term, the amounts of hydrogen needed for the observed explosions in unit 3 and unit 4 and the comparison with the measured temperatures in the RPV were presented as promised in the last period. A plausible accident scenario for unit 3 was proposed.

The BSAF-II project was scheduled to end in March 2018; however the progress of the project has been slower than expected. In consequence, the roadmap for the final meeting has been slightly modified. The results were submitted to the operating agent at the end of October (instead of early September). The final meeting is scheduled at the end of January 2018. PSI will present the results included in the present report. During the meeting it will be discussed if additional results/comparisons are needed and which results are to be reported in the OECD summary report. In addition, it will be decided when the individual reports (i.e. per institution) will be provided, presumably shortly after the meeting.

Publications

Plan to submit the presented results of the final sequence to a journal by the beginning of next year.

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