The OECD-NEA Benchmark Study of the accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project – Phase 2

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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 aims of the project are:

- To provide information and analysis results on the Severe Accident (SA) progression, fission products (FP) behaviour, source term estimation and comparison with measured plant data within the first 3 weeks in Fukushima Daiichi Accident in March 2011 to support safe and timely decommissioning at Fukushima Daiichi NPS;
- To contribute to improvement of methods and models of the SA codes applied in each participating organization, in order to reduce uncertainties in SA analysis and validate the SA analysis codes by using data measured through the decommissioning process in Fukushima Daiichi NPS;

The project will help the TEPCO to plan the removal of components from the reactor containment, decontamination, and the final decommissioning. Phase 2 of the project is ongoing. For the Phase 2, the scope of the analysis is extended 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. In addition, the duration of the analysed sequence will be extended to 20 days from the accident initiation, a task which may prove very challenging to the severe accident analysis tools.

The signatory countries from phase 1 continue in phase 2: France, Germany, Japan, Korea, Russia, Spain, Switzerland and United States. Additionally 3 new signatories have joined the project: China, Canada and Finland. The operating agent of the project is The Japan Atomic Energy Agency (JAEA). The project started 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 during phase 2 of the BSAF project. However, the use in the future of other tools which provide a more detailed treatment of hydrogen distribution and fission product behaviour is not excluded. During the present period, some modifications were made to the input; the suppression chamber was split in 3 axial volumes (instead of 2) and the potential release paths from RPV and PCV were updated based on the information of failures and penetrations provided by the operating agent. After the changes a similar sequence to the one from BSAF phase 1 was obtained. Plausible failures of RPV and PCV at various times were evaluated and compared with existing measured data (pressures, dose rates, water levels). One calculation was selected and extended to ca. 10 days, where the increase in pressure between 8–10 days was partially predicted.

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 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 will be 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 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 hence 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, e.g., the reactor building, to evaluate the *hydrogen effects and the source term transport.* For this, it is necessary to improve/develop nodalization for the plant to include the adjacent buildings;
- To improve the methods and models of computer codes in use by each participating country and organisation, to reduce uncertainties in Severe Accidents (SA) analysis and validate SA codes using actual data available from Fukushima Daiichi NPS units 1–3.

The analysis at Paul Scherrer Institute (PSI) will concentrate on Unit 3, as during BSAF Phase 1. During the present period, the main tool for the system level simulation of the sequence was MELCOR 2.1. Use of other tools which provide a more detailed treatment of hydrogen distribution and fission product behaviour is not excluded but were not used in the present period.

Work carried out and results obtained

At the end of BSAF phase-1 project, several issues for the accident sequences in Fukushima remained open [2]. The following issues were addressed in the present period for the sequence in unit 3:

- The RPV depressurisation at ca. 42 h was due to ADS or Main Steam Line (MSL) failure?
- Did the MSL fail, if yes when and where?
- Did the Containment fail, if yes when and where?
- Cause of 3rd pressure drop after depressurisation in unit 3 at ca. 55 h.

During the present period the input deck used to obtain the best estimate calculation for BSAF phase 1 [3] was used as the starting point. In order to address the open issues for BSAF phase 2, the following modifications were carried out:

- The elevations of the possible failures from RPV and PCV were updated in the input deck to be in agreement with the newest information provided by the operating agent.
- The suppression chamber was revised; minor modifications were made to the geometry and the initial water inventory. The nodalization was modified; the suppression chamber was divided in 3 volumes allowing a better representation of the scenario.
- The representation of the spargers in the Wetwell was corrected in the model. In previous calculation they were represented as a single hole (i.e. default option in Melcor). They were modified with the information provided by the operating agent.

Impact of modification in the input

The most significant change to the geometry was the re-nodalization of the suppression chamber. In former calculations the suppression chamber was split in two axial nodes and during HPCI operation it was needed to make the assumption that water was coming from the Cooling Storage Tank to the Wetwell (WW) in order to reproduce the pressure in the Wetwell. Several analytical and experimental studies [4–9] have pointed out that thermal stratification is the most likely cause of the observed behaviour in the suppression chamber; during RCIC operation the steam rate discharged into the suppression chamber is significantly lower than during HPCI operation (2.1 kg/s vs. 14.0–4.1 kg/s). It is plausible that the low steam discharge during RCIC operation caused a «hot» zone to be formed in the middle of the pool; in this scenario the area at the bottom may remain cold forming a stratified pool. In contrast, during HPCI operation the cold area may have mixed with the hotter area due to the mixing produced by the significantly higher amount of steam being discharged into the suppression pool. Therefore, the input was divided in three axial nodes in the present period. In this way, the former assumption that water was coming from the CST during HPCI operation was eliminated. Instead it was attempted to represent thermal stratification following the strategy below;

- The lowest volume represents the cold volume of the suppression chamber which remains cold as long as there is no mixing.
- The middle volume represents the hot part of the suppression chamber where hot gases are being discharged
- The upper part represents the gas phase of the pool
- During the RCIC operation it was assumed that no mixing was taking place; when HPCI starts it is assumed that the significantly higher steam discharge to the suppression pool caused the mixing of the cold area and the hot area. In the model it is represented by a flow path that mixes the low and middle volumes. The rate of water being mixed is proportional to the amount of steam being discharged to the suppression pool. This simplified representation allowed reproducing similar results as in previous calculations, figure 1.

The new calculation (green in figure 1) includes the modification to the spargers in the Wetwell and the changes to the elevations of the possible failures of RPV and PCV. As expected, the changes made no significant impact on the thermo-hydraulic behaviour, however the correct representation of the spargers has a significant impact on the prediction of the fission product (FP) release.

The FP aerosol scrubbing in the WW water is highly dependent on the vent geometries of the tubes discharging gases into WW. The SPARC model in MELCOR which is responsible for the scrubbing calculations reflects correctly on this and would thus predict much higher FP retention in the WW water for the «sparger-type» vent than for a plain tube discharge, see [10].

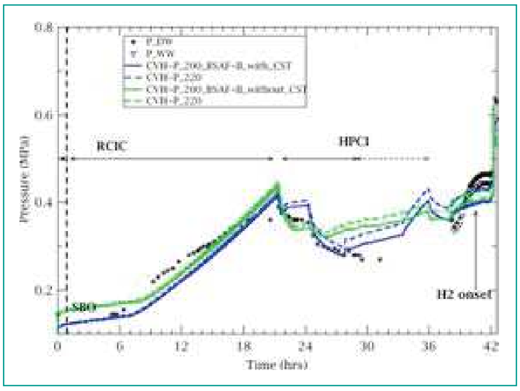


Figure 1:
Containment pressure between 0–42 h after scram

Addressing the open issues for the sequence in Unit 3

In order to answer the open questions for the sequence in Unit 3 it was necessary to integrate the analysis of all the available measured data and to identify assumptions that either support or contradict the observed data. In this way the uncertainties may be reduced. Special attention was given to the data on dose rate and the Containment Atmospheric Monitoring System (CAMS) inside the DW and outside the WW (placed in the WW room).

Several calculations assuming different failures and timings were performed and compared with the various measured data, pressures, liquid levels, dose rates etc. in the present period. Table 1 shows the most representative calculations that were used for the analysis, indicating the failure times that were either assumed or calculated by the code (i.e. DW and MSL leak are assumptions, Relocations to the lower head (LH), and pedestal and penetration failures are predicted by the code).

RPV depressurisation

The Automatic depressurisation System (ADS) actuation is generally believed to be the cause of depressurisation of the RPV in unit 3 at ca. 42 h. However, the possibility of Main Steam Line (MSL)

Fail times	DW leak (h)	MSL Leak (h)	Relocation to LH	Penetration failure	Relocation to Pedestal
SC1	68.2	–	63.4	63.8	–
SC2	42.4	42.4	63.2	66.2	–
SC3	60.3	60.3	61.6	62.4	–

Table 1:
Sensitivity calculation and main failure times

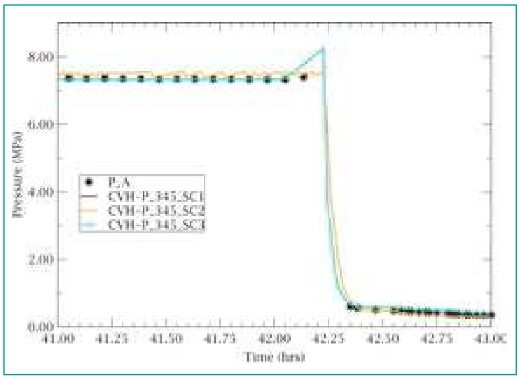


Figure 2:
RPV pressure during depressurisation

failure was pointed out in [3]. Therefore, a sensitivity case was performed (SC2) where the MSL is assumed to fail at ca. 42 h, for this case it is also assumed that DW head flange failure takes place at 42 h in order to avoid the over-prediction of the containment pressures (DW and WW). Figure 2 shows the measured RPV pressure vs. the calculated by MELCOR, the drop in pressure is well predict by either MSL fail (SC2, orange) or ADS (SC1 and SC3).

However, when MSL failure is assumed, the DW leakage would have to decrease at relatively high pressures in order to predict the pressure signature, which is less likely to happen. The analysis was complemented by the observations of the fission product releases to the environment.

Figures 3 (a and b) show a qualitative comparison of the dose rate (µS/h) at the main gate of Fukushima, with the observed containment pressure in the drywell and the calculated pressure and fission product release rates from noble gases, CsOH and CsI (in kg/s) with MELCOR for unit 3. Figure 3(a) focuses in the period between 39 – 59 h and figure 3(b) on the period between 59 – 78 h. During this time frame most of the degradation in unit 3 is taking place.

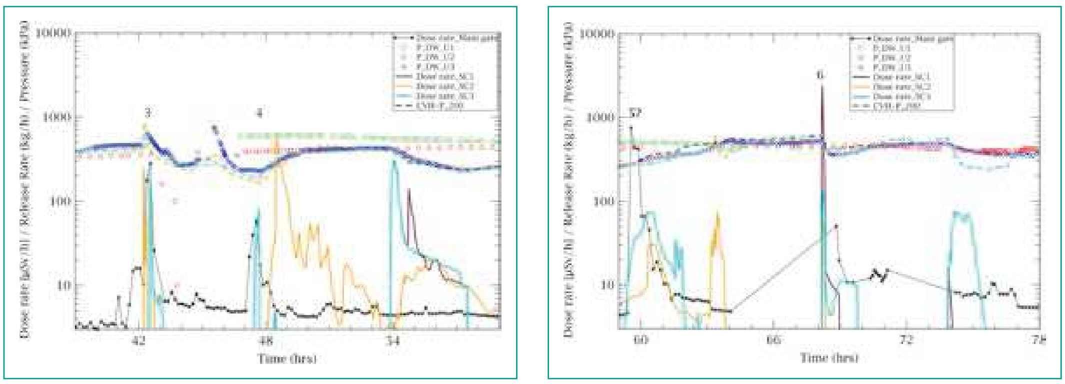
In case MSL failure is assumed (SC2, orange) at ca. 42 h, there is constant release to the environment between 48 – 54 h. This release would follow the path RPV-MSL-DW-Building-Environment. During this time the calculation predicted an overpressure in the building followed by a blow-out panel and releases to the environment. This event didn't take place during the sequence of unit 3. Furthermore, the predicted constant releases from SC2 are not supported by the dose rate measurements at the main gate of Fukushima site (figure 3a).

In contrast the calculations where ADS was assumed (SC1 and SC3) were in a good qualitative agreement with the main releases from the period between 42 – 50 h (figure 3a). Under this scenario, the FP releases from the RPV would go first to the Wetwell, where a large amount of CsOH and CsI would be retained. Therefore the releases to the environment (i.e. by venting the WW) would mainly contain noble gases. This result gives a hint that **MSL was not the cause of depressurisation and supports strongly the theory that ADS was the cause of depressurisation in U3.**

MSL failure at ca. 60 h

An additional sensitivity case was performed where the MSL is assumed to fail at ca. 60 h and DW has to be assumed to leak at ca. the same time. The pressures and the dose rates can be qualitatively reproduced in this scenario. Nevertheless the CAMS measurement doesn't seem to support this theory. Figure 4 and 5 shows a qualitative comparison of the CAMS in the DW (inside) and WW (outside) with the calculated mass of radioactive aerosols and vapour presents in the DW and WW in the calculation respectively.

As mentioned previously, if MSL is assumed to leak at ca. 60 h, DW leakage has to be assumed as well. Under this scenario a reduction in the dose rate (airborne radioactivity) in the drywell may be expected as it was predicted with the SC3 (cyan). The CAMS measurement (figures 4 and 5) seems to indicate that no DW leakage (i.e. at least no large) was taking place between 60 – 68 h. Instead the



Figures 3a + 3b:
Dose rate vs. containment pressure for unit 3

dose rate in the DW increases almost in parallel with the containment pressure increase (see figure 3). At the same time the dose rate in the WW decreases. The reason for the decreasing value of the dose rate in the WW may be that as the WW pressure increases, there is gas flow from the WW to the DW hence causing the DW pressure to increase and the total amount of radioactive vapours and aerosols to decrease in the WW.

It is possible that the increase in pressure in the containment is connected with a major event taking place in the RPV. The case SC1, predicted partial relocation of the debris collected above the core plate to the lower plenum between ca. 61–63 h and had no leakage from the DW before ca. 68 h. These calculations predicted closely the CAMS shape in the DW and WW. **This observation gives a hint that a major degradation event was taking place between 60–68 h and that no major leakage from containment takes place before ca. 68 h.** However these results should be taken only as an indication as the comparison is purely qualitative and there is a lack of data in the hours before and after that period.

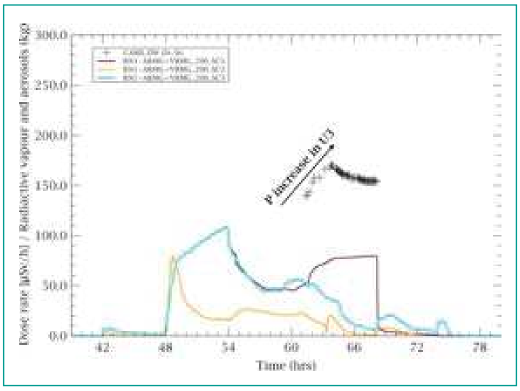


Figure 4:
CAMS in DW vs. calculated FP in drywell

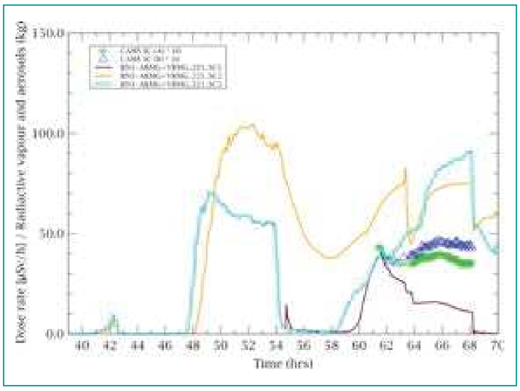


Figure 5:
CAMS in WW vs. calculated FP in wetwellgas space

3rd containment pressure drop in U3 and possible transport of hydrogen to U4

In order to evaluate which reactor is responsible for each release it is necessary to integrate the major events from the accident. The observed releases between 0 h–38 h (i.e. peaks 1 and 2 in the dose rate measurements) take place before the core degradation of Unit 2 and U3 started, therefore it is highly likely that these releases came from U1. Figure 6 shows a qualitative comparison of the dose rate (μS/h) at the main gate of Fukushima, with the observed containment pressure in the drywell for units 1, 2 and 3 and the calculated release rate in U3.

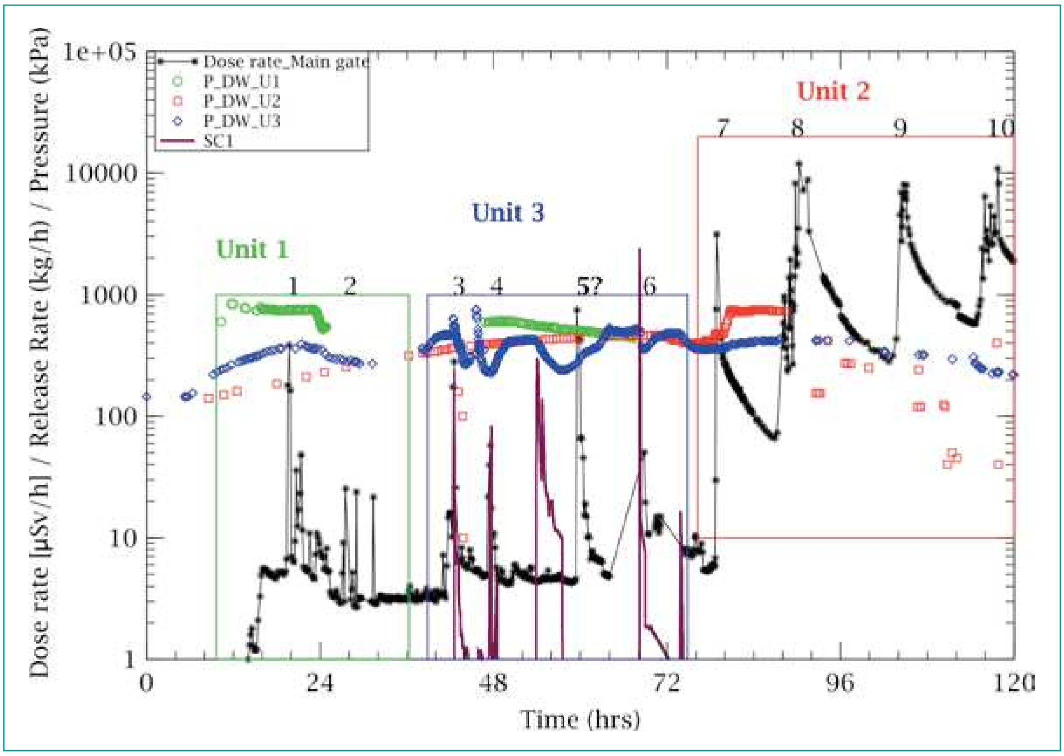


Figure 6:
Dose rate vs. containment pressure and calculated release rates

The observed releases from 39 h to 80 h (i.e. peaks 3 – 6) are very likely to come from the degradation in U3, because during this time frame most of the degradation is taking place in this unit. Both, the pressure decrease in unit 3 and the calculated releases (i.e. including noble gases, CsOH and CsI) are in agreement with the observed releases 3, 4 and 6. However the 5th release peak does not seem to be correlated with the decrease in pressure in U3. The observed release is ca. 6 hours later than start of the PCV depressurisation in U3. The possible explanation for this behaviour is either:

- It took a long time for the fission products to be transported and be detected at the main gate. However, the main gate is just some meters away from Unit 3 and there is no indication that the weather conditions would have transported the fission products in the opposite direction.
- The fission products and hydrogen released from U3 between 54 – 57 h were all transported to the stack but instead of being vented to the environment they found a path to be transported to U4. **Under this scenario it would be possible to explain the build-up of hydrogen in Unit 4 which hours later (at ca. 87 h) had an explosion.** The observed dose rate peak (i.e. number 5) may be in actual fact coming from U1. The alternative water injection was interrupted in U1 ca. 1 hour before the observed release. At that time (ca. 60 hours), the PCV in U1 is leaking (figure 3), the pressure is going down, and if there is any further degradation/relocation which would be taking place at that time the release would be seen in the environment.

The issue remains open as it is not possible at this stage to give firm conclusions. A more detail analysis (e. g. CFD analysis) on the transport of gases from unit 3 to unit 4 as well as the evaluation of the amount of hydrogen needed for the explosions for unit 3 and unit 4 may be necessary to address it.

Releases after 72 h

After 72 h the observed releases may come from unit 3 or unit 2 (figure 6). The releases are orders of magnitude higher that the ones observed previously. This observation may suggest that either a major leakage took place in Unit 3 or the releases belong to another unit. **The scenario presented in the present analysis doesn't support the theory that the releases after 72 h come from unit 3.** During that time frame most of the degradation is taking place in unit 2. Furthermore, the releases (dose rate peaks 7 – 11) seem to be correlated with the decrease in pressure from the U2 containment.

This observation is further supported by the analysis made by [11] where the representative ratio Cs134/Cs137 for each unit was estimated and compared with the measured ratios Cs134/Cs137 found in the soil contamination on the site. This data are only available from **10 days** after the scram, but they suggest that the contamination in the time period between 6 – 10 days may belong to unit 2.

Preliminary long term run

Large variations in the amount of water injection took place between 8 – 9 days. Additionally, a significant increase in the pressure of the containment was observed at ca. 192 h. The pressure in the DW reached a maximum of ca. 0.4 MPa at ca. 216 h. After that time, the DW pressure decreased to ca. atmospheric and never increased again, suggesting a permanent leakage in the DW. On the other hand, when the WW pressure measurement was available again (at ca. 300 h) the pressure in the WW is ca. 0.2 MPa. This pressure corresponds to the hydrostatic head. The fact that the pressure could increase at around 200 hours shows that the PCV was not leaking a high amount of gases before ca. 216 h. This can be explained by drywell head flange, when the pressures are high, the leakages would be higher and as the pressure decreases the leakage decrease substantially or even stops.

At ca. 153 h the water injection decreases which may have caused the remaining rods in the core region to heat up and uncover again. At ca. 201 h the water injection increases significantly and may have caused the RPV to be filled. The steam line may have been flooded increasing the amount of water reaching the suppression chamber. The WW gets also water from the RPV penetration leakage going first to the DW and then to the WW. The combination of all this sources of water may have caused the WW to get full of water.

The SC1 was extended to run in the period from 6 – 10 days and attempts to represent the scenario described above. Figure 7 shows the measured and calculated containment pressure with the SC1. The SC1 scenario predicted that the reactor was partially filled with water at the end of 6th day after scram, which caused the core not being fully cooled down but not hot enough to produce further degradation.

The **DW pressure trend was qualitatively reproduced in the period between days 8–9** which may be an indication that the predicted final state of the core is realistic. The calculated pressure in the WW at ca. 237 h is in good agreement with the observed pressure in the WW (ca. 0.2 MPa) at ca. 300 h. **This pressure corresponds to the hydrostatic head and supports the theory that the suppression chamber was full of water in this time frame.**

It is assumed that at ca. 211 h the MSL penetration seal failed opening a direct release path from DW to building and causing the pressure from the DW to drop. **This type of leakage from the containment would be permanent causing the DW pressure to remain atmospheric.**

The water coming from the RPV through the steam lines would discharge into the WW. Once the WW is full, the water would be transferred to the DW. The DW would get additional water from the water coming from the RPV penetration leakage. The SC1 estimated that at ca. 239 h the water in the Drywell would reach the level of the MSL penetration and radioactive water would start to leak from the containment.

The presented results are preliminary as only the SC1 was included in the period from 6–9 days. It is expected that for the next period more sensitivity cases will be performed, for which the new released version of MELCOR2.1 would be used. This version will allow to make faster runs and to reduce the computational time.

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.

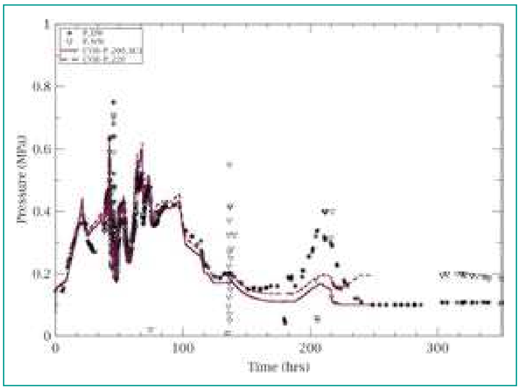


Figure 7:
Containment pressure

Assessment 2016 and Perspectives for 2017

Work at PSI is progressing as planned. The results will be presented in the next BSAF phase 2 meeting in January 2017. For the long term calculations SANDIA has promised that a new version of MELCOR will be released. It is expected that this version will allow faster runs as it is more stable for long term calculations. For the next period additional sensitivity cases will be included in the long term analysis as well as the comparison with the measured temperature at different locations. Special attention will be devoted to the analysis of the transport of fission products and hydrogen as well as the amounts of hydrogen needed for the observed explosions in unit 3 and unit 4.

Publications

M. Pellegrini, K. Dolganov, L. E. Herranz Puebla, H. Bonneville, D. Luxat, M. Sonnenkalb, J. Ishikawa, J. H. Song, R. O. Gauntt, L. Fernandez Moguel, F. Payot and Y. Nishi. Benchmark Study of the Accident at the Fukushima Daiichi NPS: Best-Estimate Case Comparison, Nuclear Technology 2016, doi: https://dx.doi.org/10.13182/NT16-63

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