Analysis of the accident in the Fukushima Daiichi nuclear power station

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Duration of the Project	2013–2014

ABSTRACT

In March 2011 a major accident occurred at the Fukushima Daiichi nuclear power station, triggered by an extremely strong earthquake and the subsequent tsunami on the eastern coast of Japan. During the next days, devastation of the local area due to the flooding meant that vital power supplies were unavailable and other services were disrupted. The loss of power meant that vital safety equipment did not function as designed and recovery operations were severely hindered. During the accident, 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, although the extent is as yet unknown. The consequent release of radioactive material meant that a large area surrounding the accident site had to be evacuated.

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 power station 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. Toward the end the results by each of the participants will be discussed at a joint 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 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 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:

- 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 reactor vessel water level measurement is available but it is subject to gaps and uncertainties.

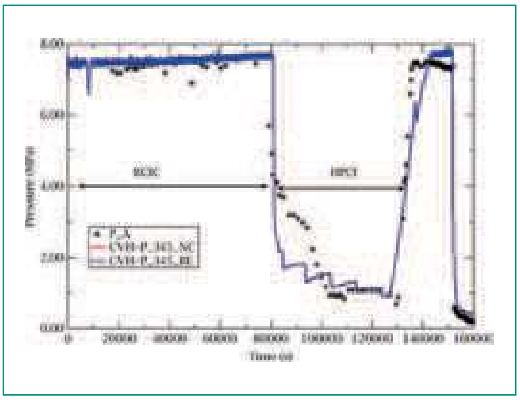
In order to start the analysis, a MELCOR 2.1 generic input model for a Mark 1 BWR similar to Unit 3 was obtained and adjusted to the specifics of Fukushima. The input was imported into the visualisation tool SNAP in order to facilitate overview and manage analysis tasks.

A preliminary calculation has been performed based on nominal accident assumptions (NC) and it has been used as the initial reference case for the study. PSI has steadily worked towards a credible sequence. The effect of the uncertainty in the water injection rates and times; venting times, fraction of opening on the venting valve and venting paths has been studied in detail. More than 50 simulations have been performed in order to obtain a best estimate (BE), namely a sequence that can reproduce the pressure measurements in the RPV and in the WW/SC and the time of the hydrogen explosion.

For this report, the study has been divided in two sections: The first section comprises the actions taken prior to reactor depressurisation, where the Reactor Core Isolation Cooling system (RCIC), High-Pressure Coolant Injection system (HPCI) and the sprays were operating; this section includes as well the depressurization of the RPV. The second section comprises all the actions taken by the operators after depressurisation; this includes the fresh and sea water injection and venting actions.

Accident progression until depressurisation

The RCIC operates by extracting steam from a main steam line to drive a turbine mechanically linked to the injection pump. The exhaust steam from the turbine is transferred to the suppression chamber (S/C) pool. Pump suction is initially aligned to the Condensate Storage Tank CST and may be redirected to the suppression pool when the CST is depleted. For the present analysis an RCIC pump



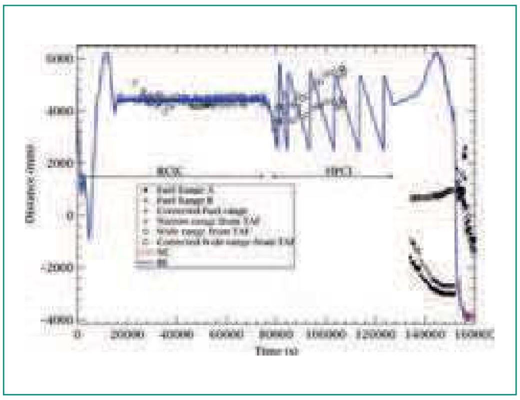
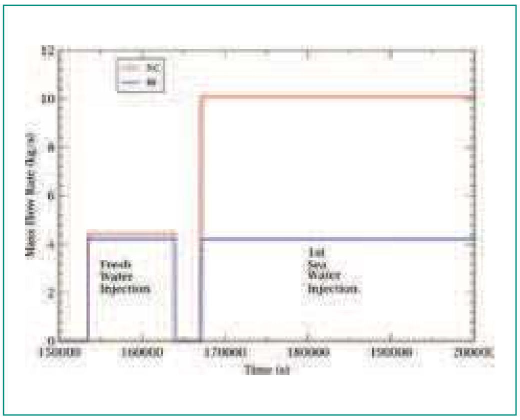


Figure 1 (left):
Pressure in the RPV during RCIC and HPCI operation.

Figure 2 (right):
Water Level in the downcomer during RCIC and HPCI operation.



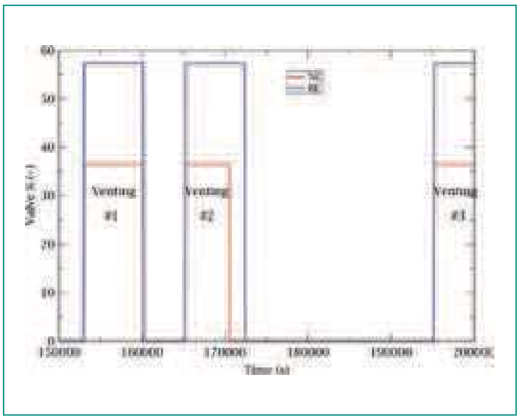


Figure 3 (left):
Water injection during the period before the H_2 explosion.

Figure 4 (right):
Venting valve area fraction during the period before the H_2 explosion.

injection controller was imposed to reproduce the available reactor water level data. This allows the reactor water inventory to be relatively correct prior to subsequent events occurring.

The HPCI is a high-pressure steam-driven pump system. In normal operation, the HPCI turbine continually draws steam from the steam lines and discharges it to the S/C pool. The mass flow rate of the steam through the turbine depends on the pressure in the RPV, the density of steam in the steam lines, and the pressure difference between the RPV and wetwell. The turbine operates continuously in this manner throughout the HPCI operation. The HPCI injection maintains the downcomer water level within an upper and lower range relative water level. Once the downcomer water level falls below the low level, the HPCI injects water at full capacity from the CST into the feedwater lines. If the CST depletes, the HPCI uses the wetwell pool to inject water into the feedwater lines. At full capacity the HPCI injection rapidly fills the downcomer water level to the upper bound cut-off for HPCI injection, where the full HPCI flow is then diverted to the wetwell via a minimum bypass flow line in the model used for this analysis. In this way, the HPCI can simultaneously maintain RPV water level and lower containment pressure (if CST is available).

The RPV pressure and the downcomer level during RCIC and HPCI operation are shown in figure 1 and 2, respectively. The pressure and water level are well reproduced during RCIC operation, whereas during HPCI operation they differ. Depressurization was assumed manual and the predicted pressure in the vessel is in fair agreement with the measured data.

Additional information on the operation of the RCIC and HPCI has been provided during the first review meeting of the OECD BSAF project in October 2013. The RCIC was working at a reduced injection, whereas the HPCI was operated manually and at a reduced rate to avoid the automatic but power expensive switching on and off that would occur during normal operation. This information has not yet been implemented in the present study and it is currently being evaluated and upgraded into our model.

Accident progression after depressurisation

The study performed at PSI has been focused in the events that happened after RPV depressurization. During this time the operators have performed several actions in attempt to stabilize the reactor and to keep the integrity of the containment. The main actions were injection of water using the fire-fighting pumps, and venting of the containment.

Figures 3 and 4 show the boundary conditions assumed for the time before the H_2 explosion. The fresh water injection is similar for the nominal case (NC) and the best estimate (BE) whereas the first sea water injection was reduced to the same amount reported of fresh water (~4.4 kg/s for the BE). The fraction of valve opening during the venting was assumed to be larger for the BE (57.3%) than for the NC (36.4%) and the second venting in the BE took place 30 minutes later than in the NC. Figures 5 and 6 represent the state of the core at 168500s for the NC and the BE respectively. This time correspond to time before the 1st sea water injection. The tables from the left to right represent the cladding and debris temperatures in the different axial and radial locations. The diagram on the right represents the state of the core. The components of the core are represented by different colors (i. e. pink= Intact fuel, yellow =support structure, blue = liquid water, green = particulate debris and red = molten pool). The reduction in the sea water injection resulted in faster heat up of the core in the BE case, as expected, as well as slightly more core degradation.

Figures 7 and 8 show the pressure in the drywell/wetwell (DW/WW) and the global hydrogen production, respectively. The pressure calculated by both cases are very similar, whereas the BE produced ~100 kg more of hydrogen, as expected, due to the higher temperatures reached during the core uncovery. Furthermore 10 kg/s of water injection (NC) seem to have been enough to stop the accident progression. The H_2 production stops after ~180000s whereas in the BE the H_2 production continues, this indicates that the reactor is still hot and cladding oxidation is still taking place.

Figures 9 and 10 show the boundary conditions assumed for the time period when the H_2 explo-

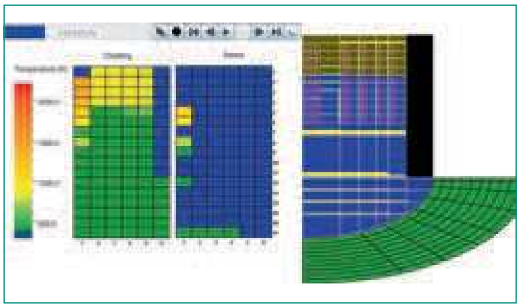


Figure 5 (left): *NC state of the core at 168500s.*

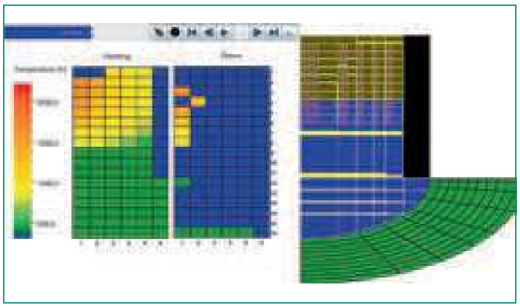


Figure 6 (right): *BE state of the core at 168500s.*

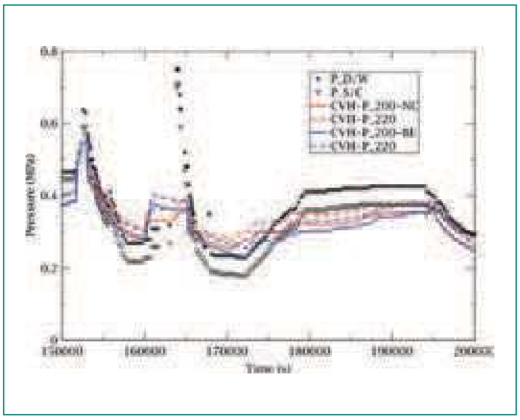


Figure 7 (left): *Pressure in the DW/WW during the period before the H_2 explosion.*

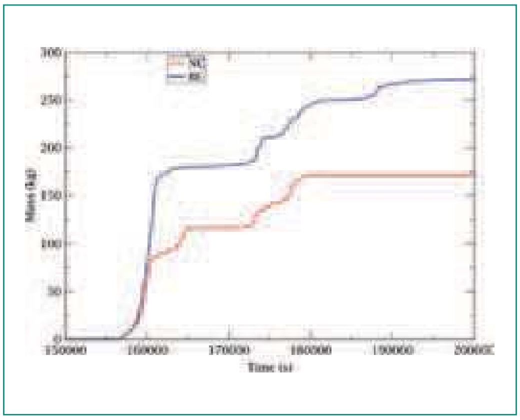


Figure 8 (right): *Global H_2 production during the period before the H_2 explosion.*

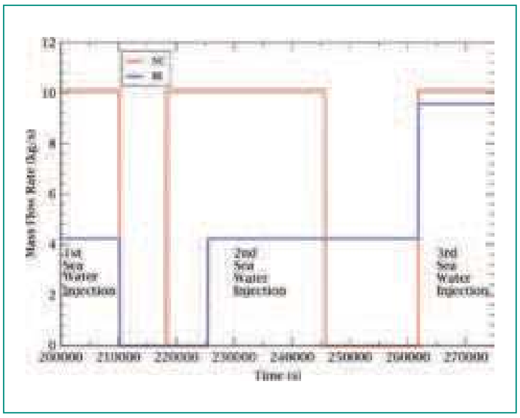


Figure 9 (left): *Water injection during the period when the H_2 explosion was observed.*

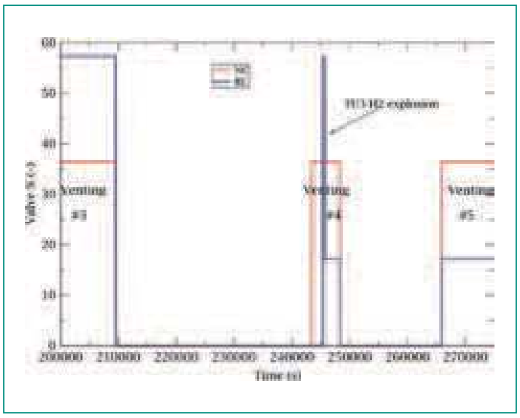
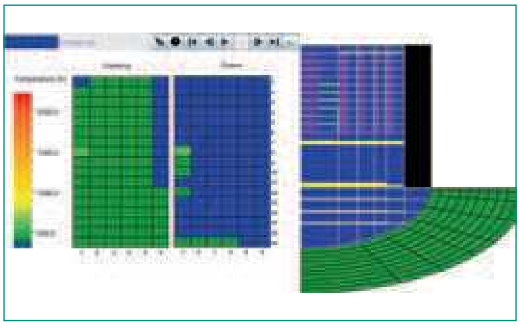


Figure 10 (right): *Venting valve area fraction during the period when the H_2 explosion was observed.*



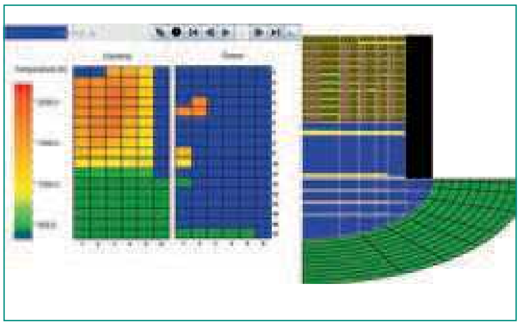
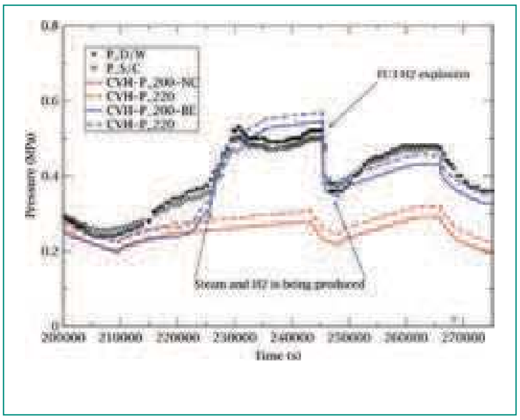


Figure 11 (left): *NC state of the core at 225000s.*

Figure 12 (right): *BE state of the core at 225000s.*



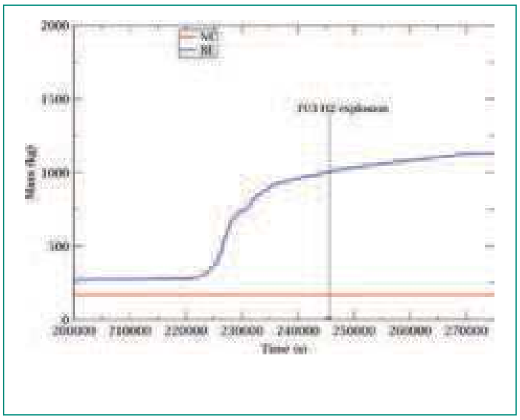
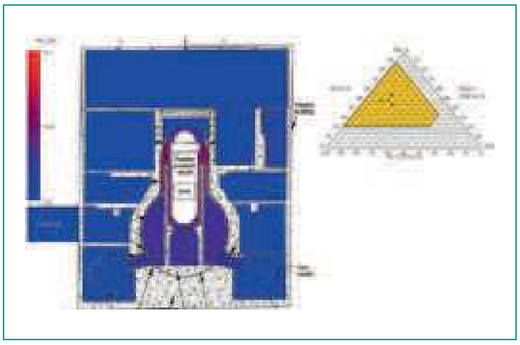


Figure 13 (left): *Pressure in the DW/WW during the period when the H_2 explosion was observed.*

Figure 14 (right): *Global H2 production during the period when the H_2 explosion was observed.*



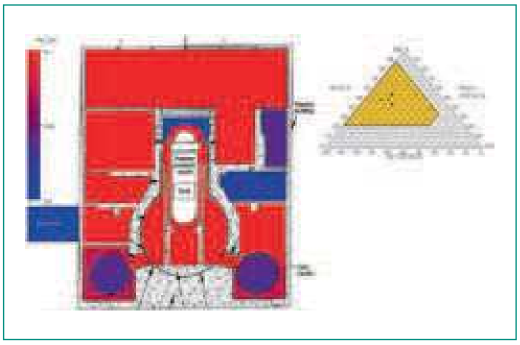


Figure 15 (left): *H_2 concentration in the reactor building for the NC at 245 640s.*

Figure 16 (right): *H_2 concentration in the reactor building for the BE at 245 640s.*

sion took place in unit 3. The second sea water injection initiation for the BE was delayed 105 min and the injection maintained at a reduced rate of ~4.4 kg/s. The injection was then continued until the beginning of the 3rd sea water injection in the BE. At this time the injection rate was increased to near nominal values. For the venting it was assumed for the BE that 4th venting did not take place until 200s before the H_2 explosion. It was also assumed that after the H_2 explosion the valve (or the pipe) may have suffered some damage and the valve opened area was reduced at ~17%. (i.e. the valve may have been damaged and this may have made a blockage for the flow, in the model it is represented as a reduction in the flow area of the valve).

Figures 11 and 12 represent the state of the core at 225000s (near the observed H_2 explosion) for the NC and the BE respectively. For the NC 10 kg/s (nominal) was enough to stop the accident progression, the core is cooled down and only few debris were produced. For the BE, the reduction of the 1st and 2nd sea water injection flow rate to ~4.4 kg/s and the delay of 105 min in the 2nd sea water injection was critical; the core heated-up and uncovered and there was a significantly greater amount of cladding degradation and debris formation.

Figures 13 and 14 show the pressure in the drywell/wetwell (DW/WW) and the global hydrogen production, respectively. The BE has reproduced very closely the pressure in the DW/WW. As a consequence of the core uncovery, very large amounts of steam were being generated, while the temperatures increased to levels where the cladding reacted with the steam and large amounts of hydrogen were produced. The generation of large amounts of steam and hydrogen caused the pressure in the DW/WW to increase.

The opening of the valve shortly before the hydrogen explosion (venting #4) caused the pressure to drop quickly as it was observed during the accident. Furthermore, the continuation of the 2nd sea water injection, the reduction of the valve area after the time of the explosion and the increase of the 3rd sea water injection rate allowed to reproduce the subsequent pressure in the DW/WW. The increase of the rate of water injection seems to have been enough to stop the further progression of the accident.

Figures 15 and 16 represent the H_2 concentration in the reactor building for the NC and BE respectively.

It appears that the venting to the environment was not effective. Instead gas was bypassed into the reactor building. Had venting operation been successful, all the hydrogen released from the wetwell would have been discharged harmlessly into the environment. However, a large explosion was observed in unit 3 reactor building, which implies that a large mass of hydrogen accumulated there to produce an explosive concentration. It is possible that the pipe used for the venting may have been damaged either by an overpressure or the earthquake itself, or that the loss of power prevented normal opening of the vent valves. In any case there would have been a path for gas to leak into the reactor building if impairment of the venting system caused overpressure in the vent line. For the BE the hydrogen accumulation is reproduced by connecting the venting line with the volume at the top of the reactor building, in this way H_2 explosion conditions were calculated at the exact time of the explosion (245 640s) as it can be seen in figure 16, whereas for the NC the H_2 concentration was nowhere near H_2 explosion conditions (figure 15) due to the fact that no hydrogen is being produced during this time.

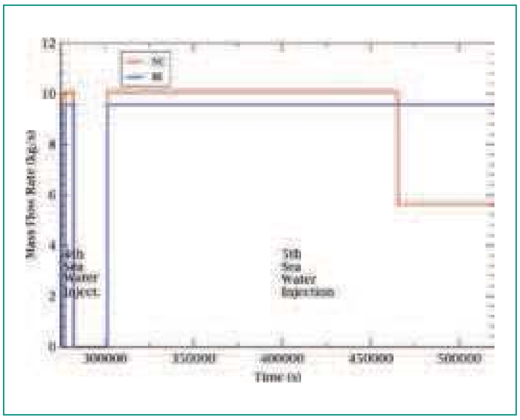


Figure 17 (left): *Water injection during the period after the H_2 explosion.*

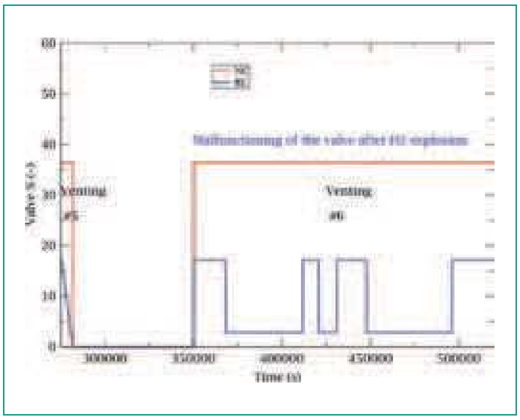


Figure 18 (right): *Venting valve area fraction during the period after the H_2 explosion.*

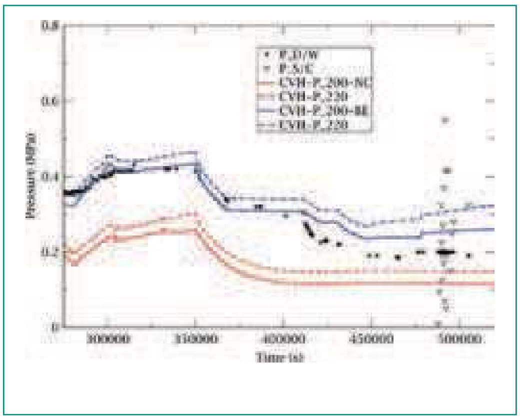


Figure 19 (left): *Pressure in the DW/WW during the period after the H_2 explosion.*

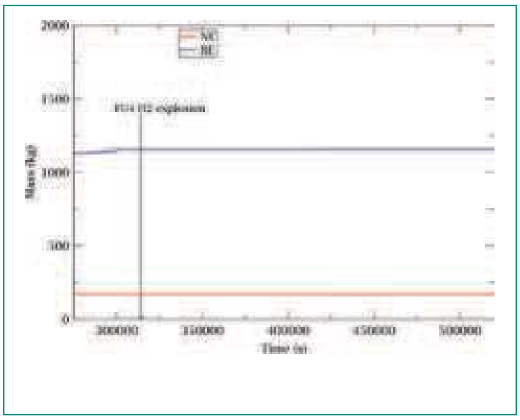


Figure 20 (right): *Global H_2 production during the period after the H_2 explosion.*

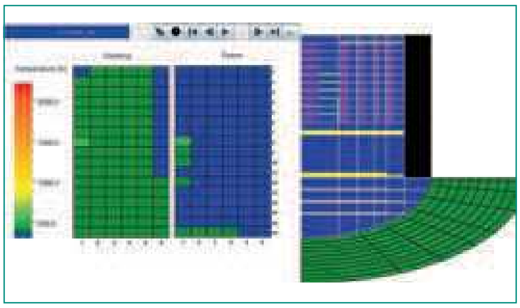


Figure 21 (left): *NC state of the core at 518000s.*

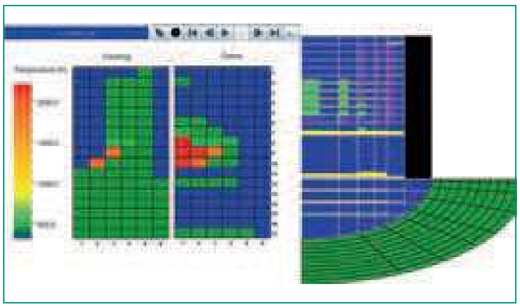


Figure 22 (right): *BE state of the core at 518000s.*

Figures 17 and 18 show the boundary conditions assumed for the time after the H_2 explosion. The water injection mass flows are very similar, for the BE they are assumed near nominal. On the other hand, the valve is considered to be malfunctioning after the H_2 explosion, in the nominal case the venting 6 is constant and the valve does not close again. For the BE a series of openings and closing of the valve were assumed.

Figures 19 and 20 show the pressure in the DW/WW and the global H_2 production. The assumption in the behaviour of the valve allowed to reproduce the pressure in the DW/WW very closely for the BE, whereas in the NC the valve remains open all the time and makes the pressure drop and stay constant after ~400000s. In both cases there is almost no hydrogen production during this period indicating that after the H_2 explosion the operators managed to stabilize the core.

Figures 21 and 22 represent a preliminary statement of the state of the core after 6 days of transient for the NC and the BE respectively. For the NC the core was completely cooled down and there was very little degradation. For the BE there was more degradation but neither relocation nor vessel failure. The intact components in the reactor were cooled down but the debris was still hot at this time.

The NC does not reproduce the most reliable signatures (DW-P, H_2 explosion t). In contrast, the BE reproduces very well the pressure signature and the high hydrogen concentration in the reactor building corresponds to the time of explosion. According to the presented results (BE) the FU3 core seems to have been less damaged as it was believed in a first place. However this is based on the assumptions made for the analysis which are called into doubt by the new information provided during the meeting in October 2013 regarding the RCIC and HPCI operation. The impact of the new information has yet to be evaluated and incorporated in our model; thus they could change the preliminary assessment.

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 2013 and Perspectives for 2014

For the OECD BSAF project, the progress during 2013 has been slower than originally planned due to delays in providing data on the plants and accident conditions. The timeframe of the project was extended until the end of September, 2014. Despite the delay in the OECD BSAF project, PSI work has progressed according to the plan.

Preliminary analyses performed in the first half of 2013 were based on nominal accident assumptions. In addition to the nominal case, several simulations were carried out at PSI to find the best estimate case reproducing the main events of the accident. Revised boundary conditions were proposed at a technical review meeting of the OECD BSAF project in October but they are subject to ongoing discussion between the participants and the OA and are still not finalised. It is expected that an agreed set of accident assumptions for the baseline case will be provided toward the end of 2013. Progress at PSI has continued toward adapting the input model to accommodate the likely conditions.

A definitive baseline calculation, plus best estimate and appropriate sensitivity calculations will be performed in the first months of 2014. Submittal of the baseline simulation is due by end of April 2014, from which the OA will compile a draft report by end of June, for review and finalising by end of September. This will be followed by a wind-up meeting in October or November. The project is redesignated as BSAF Phase 1, in anticipation of a follow-on Phase 2 to address issues not resolved in Phase 1.

Publications

The terms of the agreements with the project impose restrictions on the dissemination to third parties of plant data and the results of the benchmark study. Publication of results and findings will be possible only some time after the end of the project.

References

[1] http://www.oecd-nea.org/jointproj/bsaf.html