ATHLET-CD/COCOSYS ANALYSES OF SEVERE ACCIDENTS IN FUKUSHIMA (UNITS 2 AND 3) WITHIN THE OECD/NEA BSAF PROJECT, PHASE 1

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ABSTRACT

Consideration and analysis of Beyond Design Basis Accidents (BDBA) of NPPs is an essential component of the defence-in-depth approach used in nuclear safety. Such accidents have very low probabilities, but may have significant consequences if they proceed into a Severe Accident (SA) resulting from the fuel degradation and release of radio nuclides. Analysis of such accidents requires well qualified methods and codes, e.g. ATHLET-CD and COCOSYS developed and applied at GRS.

The GRS participated on behalf of the German Federal Ministry of Economics and Technology (BMWi) in the OECD/NEA project: "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF)". Coupled ATHLET-CD/COCOSYS analyses for the SA progression during the first days for the similar Units 2 and 3 of Fukushima Daiichi have been provided. ATHLET-CD is a detailed SA code based on the thermal hydraulic code ATHLET of GRS to simulate the processes in the reactor circuit before and during core degradation. COCOSYS, developed at GRS as well, is focused on the simulation of accident and SA progression in the containment and the surrounding buildings of NPP.

The objective of the presentation is to share experience gained from code application to the SA scenarios in the BWRs at Fukushima Daiichi site. The GRS results obtained in this OECD/NEA BSAF project, phase 1, are encouraging in terms of capturing essential severe accident signatures. The calculated accident progression of the "best estimate" analyses follows the accident time-line quite closely, what is a prerequisite for reasonable core degradation calculations, as the time window available for core degradation is small. One of the specifics of the analyses is the used detailed torus model allowing modelling of relevant phenomena like thermal stratification and consideration of plant specific details with regard to water/steam injections into the torus. The analyses are useful to identify areas which require further attention, to define information needs to be gained from the decommissioning, and to define further research needs with regard to experiments and code model improvement. The objectives of the GRS analyses as part of the BSAF project, phase 2, which recently started, will be summarized as an outlook at the end.

1. INTRODUCTION

Selected results of the analyses of the accidents in Unit 2 and 3 of the Fukushima Daiichi NPP using the GRS codes ATHLET-CD/COCOSYS will be briefly discussed along with a possible explanation of main parameters with special focus on the Primary Containment Vessel (PCV) pressure transient, the thermal hydraulic conditions within the torus, and the core degradation. The analyses have been performed by GRS within the OECD/NEA BSAF project, phase 1 [1].

2. CODES USED BY GRS FOR FUKUSHIMA DAIICHI ANALYSES

The simulations were performed using the coupled GRS codes ATHLET-CD/ COCOSYS [2-4]. The codes are developed, validated and applied for analyses of a wide spectrum of accidents and SA scenarios in PWRs and BWRs [5-8]. The ATHLET-CD development status is demonstrated e.g. by successful post-test calculations of bundle and integral experiments, such as CORA, QUENCH, LOFT-LP-FP2 and Phébus FP, or the TMI-2 accident [5, 6]. The application range of COCOSYS has been significantly extended in the recent years as well using e.g. results from national and OECD/NEA THAI experimental projects for code validation [7-10].

2.1 ATHLET/ATHLET-CD

The thermal-hydraulic system code ATHLET [2] is being developed by GRS for the analysis of the whole spectrum of leaks and transients in PWRs and BWRs. ATHLET is composed of several basic modules for the simulation of the different phenomena involved in the operation of a light water reactor: thermo-fluid dynamics (5-equation model with a full-range drift-flux model or 6-equation / two-fluid model, simulation of non-condensable gases), heat transfer and heat conduction (one-dimensional model), neutron kinetics (prompt power from fission and decay of short-lived fission products) as well as control and balance-of-plant by the general control simulation module. The code ATHLET-CD [3] has been developed for severe accidents with core damage and is based on the thermohydraulic code ATHLET (Figure 1).
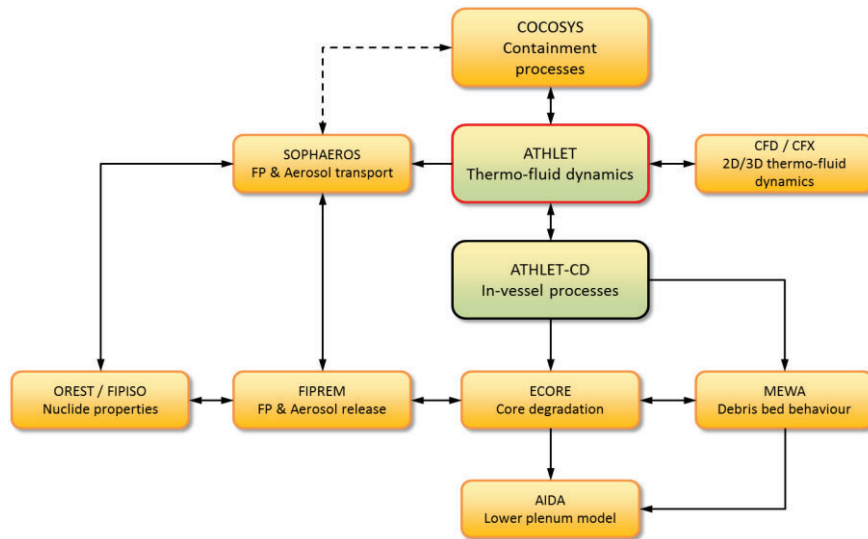

**Figure 1. Structure of ATHLET/ATHLET-CD [3].**

For certain components, e.g. steam dryers, special models are implemented in the code. For the simulation of the reactor safety systems or analyses boundary conditions the general control simulation module was implemented. The user can use basic (logical) components (like function generators) to set up any kind of control or boundary condition. This was used to simulate e.g. Safety Relief Valves (SRV), High Pressure Core Injection (HPCI) or Reactor Core Isolation Cooling (RCIC) system.

The fission product inventory of the core is calculated by OREST using the core power history, its enrichment and the initial conditions. OREST calculates the nuclide inventory by finding the equilibrium for each isotope. The decay heat is also calculated on a per isotope basis using most of the nuclide table (about 3400 different isotopes). Data provided within the OECD/NEA BSAF project have been adopted.

The BWR fuel cladding and canister oxidation and hydrogen generation model in ATHLET-CD is based on an Arrhenius correlation. Steel oxidation of structures linked to the core as well as oxidation of B_4C of the BWR control rods, and of metals in the core debris are not yet calculated. This may explain the lower amounts of hydrogen generated during core degradation compared to other code applications; model improvements are ongoing. The model for fuel clad failure is mainly dependent on the temperature and the thickness of the oxide layer. The melting of BWR control blades, fuel and canisters is mainly temperature dependent and includes eutectic reactions. The release of fission products from the core is simulated using an Arrhenius correlation. Most of the aerosol transport mechanisms are dependent on the thermo-hydraulic properties inside the primary circuit, and the relevant phenomena (deposition, sedimentation, thermophoresis and diffusiophoresis) along the transport are considered. Calculations are terminated at RPV failure as melt transfer to the cavity is not yet treated.

2.2 COCOSYS

The thermal-hydraulic system code COCOSYS [4] is being developed at GRS with the focus on phenomena inside the containment and reactor building of NPPs during accidents and SA. The so-called lumped-parameter approach used in COCOSYS relies on mechanistic models as far as possible for analyzing the physical and chemical processes in containments. Essential interactions between the individual processes, like e.g. between thermal hydraulics, hydrogen recombination and combustion as well as fission product and aerosol behavior, are treated in a comprehensive way. The structure of COCOSYS is shown in Figure 2.
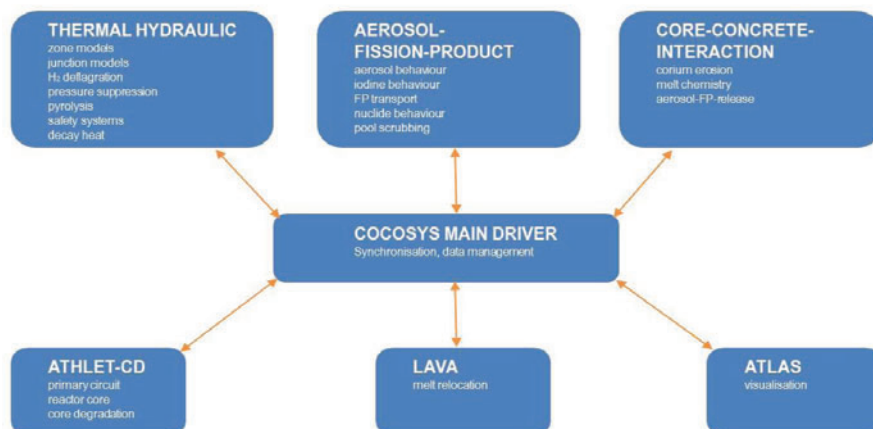
Figure 2. Structure of COCOSYS and links to other tools [4].

Apart from the three main modules (each one is separately executable) dedicated to one specific area of the overall problem: thermal hydraulics, aerosol fission product behavior and core concrete interaction, further programs especially ATHLET-CD are coupled to COCOSYS. The models are based on the fundamental energy and mass conservation, but no momentum equation is modelled in COCOSYS. The pure driving force inside the containment is the pressure difference across an atmospheric junction. In case of zones containing water, the water level is considered for water flows between zones. By this way the simulation of thermal convection between water filled volumes is realized requiring the use of a special sump balance junction type. For the simulation of severe accidents in NPPs it is necessary to take safety systems into account. It is possible to simulate different types of coolers, spray systems, ventilation systems, ice condensers, passive autocatalytic recombiners and containment venting systems. The main modules can be coupled at time-step level, while the extent of the parameters that have to be exchanged is relatively low. To keep the complexity of the data exchange within reasonable limits, any data exchange is only allowed between the main modules and the COCOSYS driver.

3. INPUT DECKS USED BY GRS FOR FUKUSHIMA DAIICHI UNIT 2 AND 3 ANALYSES

3.1. ATHLET/ATHLET-CD Input Deck

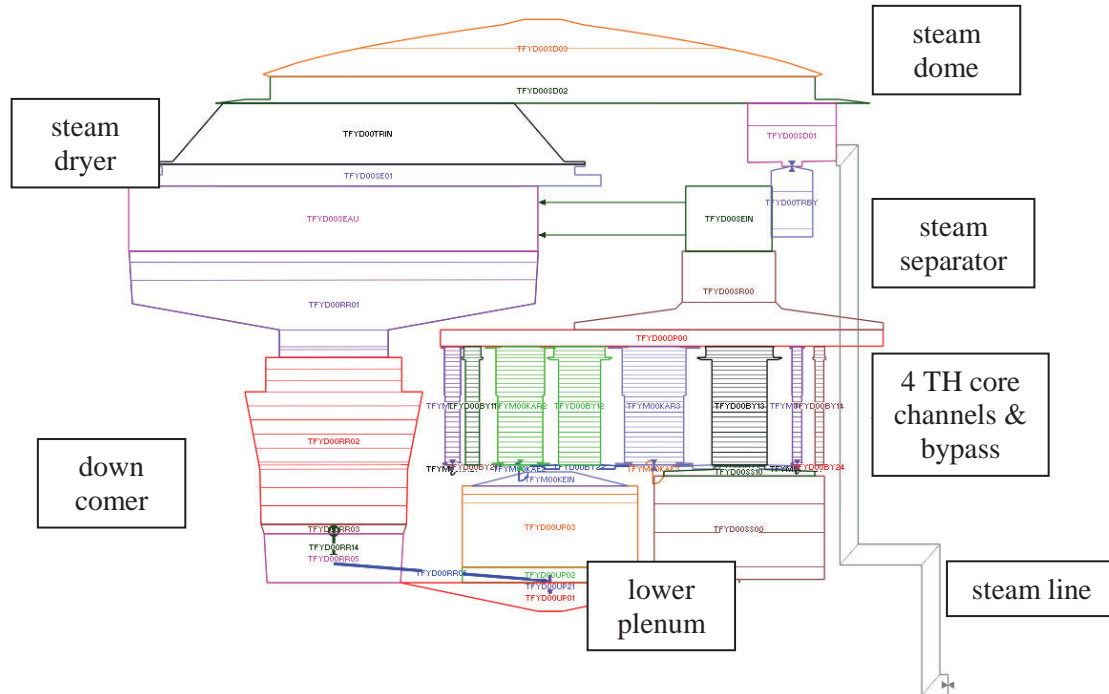The nodalisation used for the Reactor Coolant System (RCS) is illustrated in Figure 3.

Figure 3. ATHLET-CD RCS nodalisation of Fukushima Daiichi, Units 2 and 3.

The RCS was modelled by almost 300 thermo-fluid objects subdivided into many control volumes. The 5-equation model was used in all parts requiring water level treatment, while the 6-equation /two-fluid model was used especially in the core region where counter current flows of water and steam are expected. The core region is split into 4 thermohydraulic channels and 4 thermohydraulic bypasses containing the control blades. The core itself is split into 6 (non-uniform) rings that are coupled to these thermo-fluid elements. In the axial direction a fine nodalisation of the core for the thermo-fluid elements and the core rings with 26 levels is used. Although the different fuel types (MOX, plain UO_2) are considered, the thermohydraulic differences of the different fuel elements could not be taken into account. Below the core region two sets of thermo-fluid elements are used to nodalize the open space between the control blade guide tubes and the space within. In the late phase of the accident melt is relocated into the lower part of the vessel, but melt relocation into the control rod guide tubes is not simulated. The volume of the external recirculation loops is not modelled separately; it is considered as additional part of the lower part of the downcomer volume. For the simulation of the RCIC and HPCI system the water injection rate into the Reactor Pressure Vessel (RPV) and the water/steam extraction rate from the Main Steam Line (MSL) were adjusted to get the best agreement of all measured and simulated data.

3.2. COCOSYS Input Deck

Only the nodalisation of the Primary Containment Vessel (PCV) is shown in Figure 4 (left part), although the whole reactor building was simulated. The drywell (DW) is simulated using six zones (Cont1-Cont6) plus a separate zone for each of the eight vent lines (VL1-VL8). Spray systems injecting into the DW gas space or the wetwell (WW) are modelled as well.
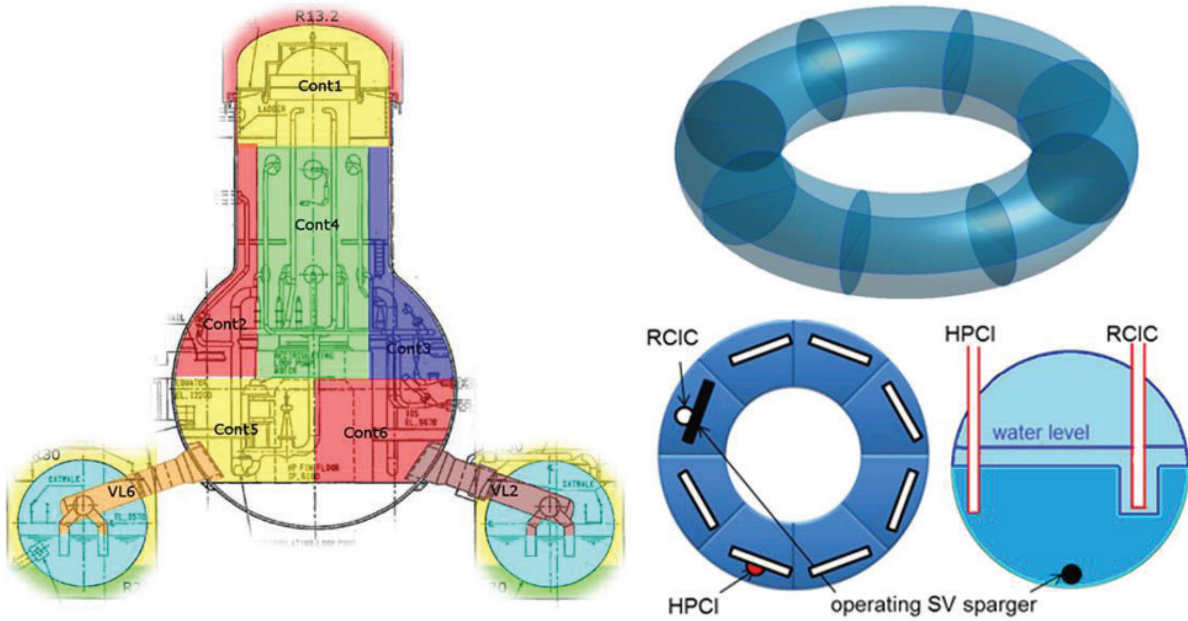
Figure 4. COCOSYS Nodalisation of the containment (left) and of the wetwell/torus (right).

The approach used first to simulate the entire torus or WW as one zone as typically done in BWR plant applications using lumped parameter codes such as COCOSYS, underestimated the measured pressure e.g. in Unit 3 rather drastically (Figure 5).
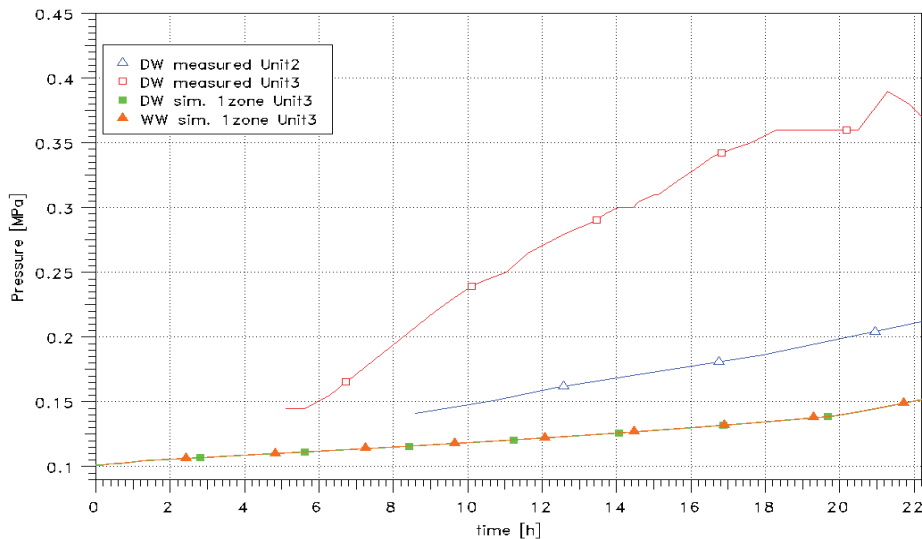
**Figure 5. PCV pressure measured in Unit 2 (red) and 3 (blue)
compared with the calculated results of a 1-zone WW nodalisation for Unit 3.**

To allow a better simulation of processes in the WW expected during the accident progression, like azimuthal and axial water temperature differences (hot water layer above a sub-cooled water layer), the torus was split into two layers (upper ~53 % and lower ~47 % of torus volume) and eight sectors with the initial water level being in the upper layer (upper ~6 % and lower ~94 % of water volume, see Figure 4). This also considers the individual positions of RCIC and HPCI suction and exhaust pipes in the torus and the location of the eight SRV T-spargers, but no special (physical) model exists for the spargers itself. The layers are connected via junctions that are able to simulate a thermal convection beside the water

mass flow caused by changes in the inventory in one zone. To a certain degree, the user has to specify changes in the convective behavior by adjusting the corresponding factors. Some effects like the increase in convection during SRV cycling are considered in the input deck via additional junctions, which are able to simulate convection. Those additional junctions are activated when an SRV opens. The gas flow respectively convection between the upper zones in the torus is simulated by connecting neighboring zones via two atmospheric junctions located at different elevations.

The strong pressure increase measured for Unit 3 can be simulated well with this thermal stratification approach used for the water within the WW, where the energy is mostly accumulated in the upper zones due to the RCIC backflow (see results in chapter 4.2). The difference in the PCV pressure transient between Unit 3 and 2 is obvious from Figure 5 as well. Even this slighter pressure increase cannot be calculated well with a one zone setup (see Figure 5). The most reasonable explanation is an external cooling of the torus caused by water penetrating into the reactor building respectively the torus room of Unit 2 just after the arrival of the tsunami. This was observed in Unit 4, and was modelled for Unit 2 (see results in chapter 4.1) even though it could not be justified for Unit 2 so far. In total ~2475 m³ of water were assumed to flow into the (roughly modelled) torus room within 24 hours, about 2/3 of this volume within ~4 hours after the tsunami, what is important to match the PCV pressure trend. Finally, the water level in the torus room reached ~3.75 m relative to the torus bottom compared to a torus water level of ~4.1 m.

4.   RESULTS OF GRS ANAYLSES FOR FUKUSHIMA DAIICHI UNIT 2 AND 3

4.1. Fukushima Daiichi Unit 2

After the earth quake and the resulting loss of offsite power, reactor scram occurred and heat from the core was removed by steam release through one SRV into the WW. The RCIC system was operated to inject water into the RPV from the condensate storage tank. Even after the arrival of the tsunami and the resulting complete loss of DC power, the RCIC stayed in operation, but in an uncontrolled manner. Therefore, the water level reached the main steam line, resulting in a two phase flow towards the RCIC turbine decreasing its efficiency to feed water from condensate storage tank into the RPV.

The calculated and measured RPV and containment pressure, the calculated steam concentration in the torus zones, and the calculated water temperature in upper and lower torus zones are shown in Figure 6. Special focus was laid in the following on the thermal conditions within the torus and the resulting PCV pressure. As already mentioned, the injection point of the operating SRV through its T-sparger and the backflow of the RCIC turbine through a simple vertical pipe in Unit 2 went into the same sector of the torus, but at different heights (zones WWl7 and WWu7). Due to the lost heat removal from the WW, its temperature was increasing by the steam released through the operating SRV and the backflow of the water /steam mixture from the RCIC turbine. An inhomogeneous temperature distribution in the pool developed in the upper zones, as shown in Figure 6, while there is no water temperature difference in the lower zones due to the additional heat losses to the flooded torus room.

Several phenomena are important in the early accident phase, such as the condensation processes of steam in the RCIC exhaust pipe at low pool temperatures, causing "pressure shocks" mixing the water nearby the injection. With increasing water temperature (larger than ~60 °C) condensation processes get more and more incomplete and steam and hot water exiting the RCIC pipe are flowing directly upwards to the pool surface, while decreasing the water mixing process. The extraction of hot water along with steam from the Main Steam Line (MSL) in combination with the injection of cold water into the RPV is able to reduce the RPV pressure starting at ~6 h, so that steam releases through SRV into zone WWl7 stopped. This led to an enhanced heating-up and steam production in the upper zone WWu7 as shown in Figure 6.
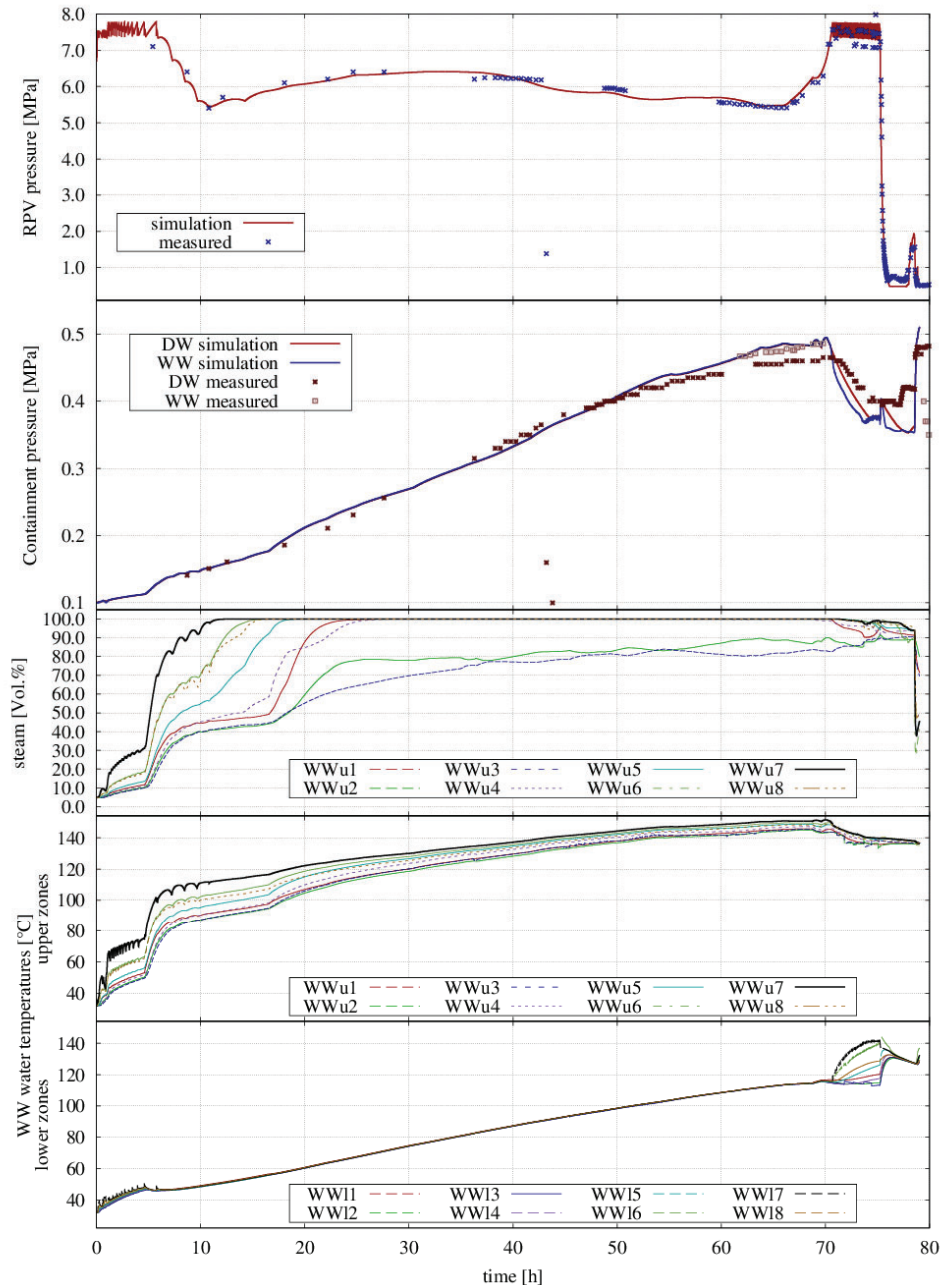
Figure 6. Calculated and measured RPV and containment pressure, and calculated steam concentration and water temperature in WW (Unit 2).

During the first ~12 h the water level drops several times below the MSL, so that the RCIC efficiency was partly increased, resulting in a higher injection of cold water into the RPV. Consequently, a more intense RPV pressure decrease follows. The RPV pressure seemed to settle in a rather stable condition, with the RCIC injection and steam/water extraction rates towards the RCIC turbine being in balance. The RPV pressure minimum at ~11 h, although the cause is unknown, coincides with the calculated last transition from pure steam extraction from MSL to two-phase flow into the RCIC turbine. The adaptation of the RCIC operation done together with the detailed torus nodalisation provides a sound calculation result with physical explanation and a general good agreement with the measured data (Figure 6 above).

The thermal convection in the torus during the RCIC operation was rather limited since no driving force is present, except the suction of the RCIC system from the Zone WWl5, starting at 14.2 h when the suction was switched over from condensate storage tank to WW. The water injected into the RPV from this time on is taken from the lower subcooled region of the WW; in addition, some hot water from upper zone(s) is transferred into the lower zone(s).

At about 68.7 h the RCIC system ceased operation, leading to an increase in the RPV pressure, again up to the actuation limit of the first SRV. The resulting decrease in PCV pressure can be explained only via the water mixing process modelled by the detailed WW nodalisation. At that time the water temperature in the upper zones was above 140 °C, while the temperature in the lower zones was only ~120 °C. The discharge from the SRV through the T-sparger into WWl7 mixes the water in the torus; colder water from the lower zones is pushed into the upper zones of the torus, resulting in a decrease of evaporation and a PCV pressure decrease. The depressurization of the RPV at about 75.3 h by several SRV uncovers the core (see RPV water level in Figure 7) which is starting to heat up. The core heat-up process along with the resulting core damage is shown in Figure 8. It shows the calculated temperature profile of fuel, canister and control blade in each core ring (ROD1-ROD6).

The early phase of the Unit 2 accident progression is typical of a low pressure core melt scenario with steam starved conditions, where no strong zirconium oxidation (cladding and canisters) occurs and not much hydrogen is produced (see Figure 9). The upper two graphs in Figure 8 show a uniform heat-up of the core with temperatures where oxidation would be possible, if steam would be available. At 77.6 h (upper right graph in Figure 8) the control blades start to melt due to eutectic interactions. The operation of mobile fire pumps started at ~77 h but the water did not reach the lower core yet (Figure 7). In the analysis the SRV(s) closed for some time and the following pressure increase at about 78 h prevented any water injections, since the pressure head of the mobile pumps was only about 1 MPa. When a SRV is re-opened, and the pressure dropped, the water injection was successful so that the water level in the RPV reached the core from below. The water evaporated leading to an escalation of core oxidation and heating-up (lower left graph in Figure 8 at 78.7 h). Within ~3/4 of an hour more than 400 kg of H_2 were produced, according to the analyses. Currently at 79.1 h the simulation was stopped, after the core was significantly re-flooded. It has not been discussed yet if a local re-criticality has to be assumed, as the fuel is still almost intact during reflooding, while the control blades are mainly destroyed (lower right graph in Figure 8). Melt relocation into the lower plenum was not predicted yet.
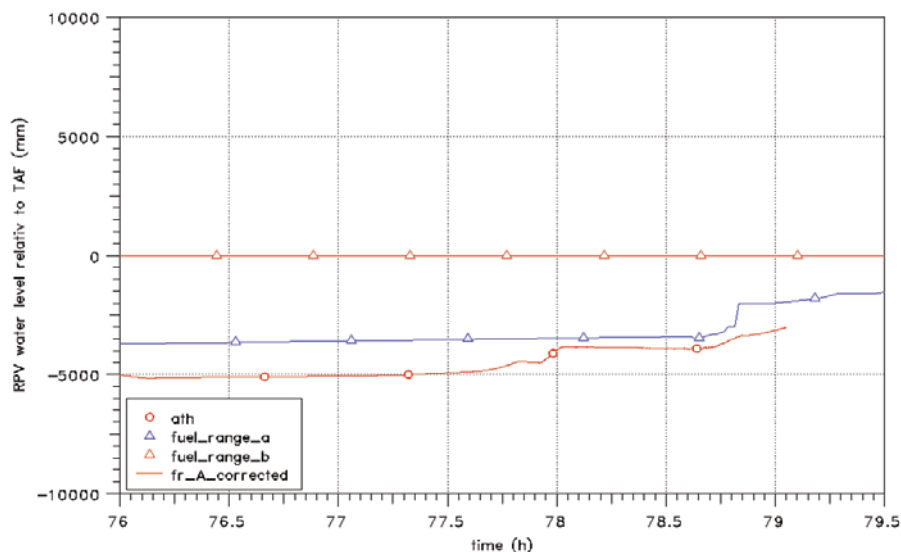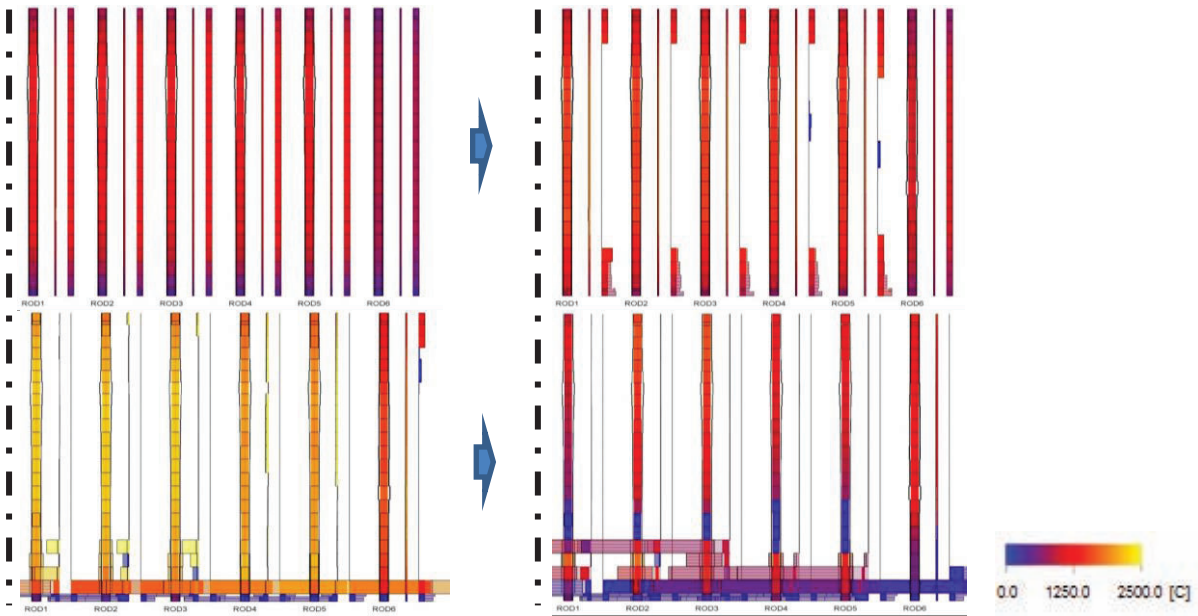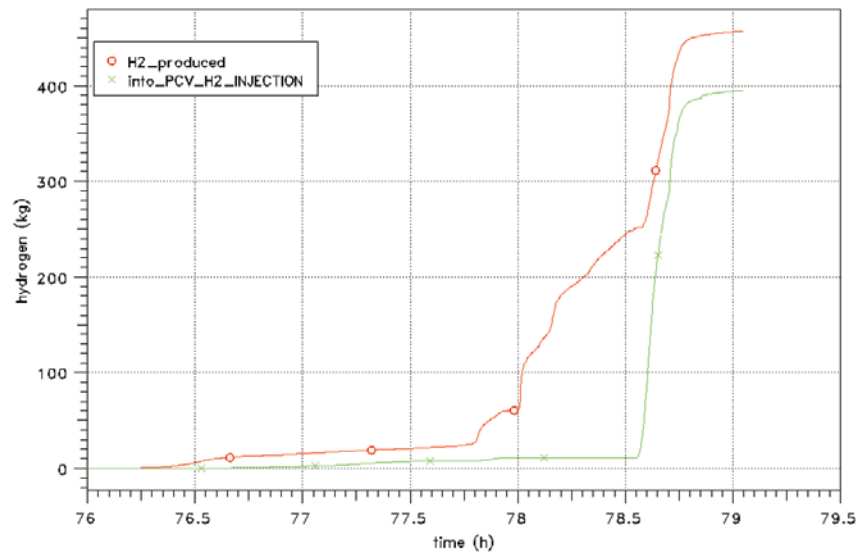

Figure 7. Calculated and measured RPV water level (Unit 2).

Figure 8. Calculated temperature of fuel, canister and control blade in core rings (ROD1-ROD6) during early heat-up of the core at 77.1 h and 77.6 h, during escalation of heating-up and oxidation at 78.7 h, and at 79.1 h during reflooding of the core (Unit 2).


Figure 9. Calculated hydrogen production during core degradation and release into the PCV (Unit 2).

4.2. Fukushima Daiichi Unit 3

The situation in Unit 3 was the same as in Unit 2 for the first ~45 min after the earth quake until arrival of tsunami. Thereafter, since DC power was not completely lost in Unit 3 in contrast to Unit 2, the RCIC system could be operated in a controlled manner, preventing that the water level reached the MSL. During the first 22 h of the accident the RCIC system was operated injecting water from condensate storage tank, while the RPV pressure was limited by one operating SRV. In the simulation a steady decrease in the

injected water mass by the RCIC was assumed, based on information provided that the operators were trying to adjust the RCIC system to prevent a repeated start/stop operation to save DC power [1]. The calculated and measured RPV and containment pressure, the calculated steam concentration in the torus zones and the calculated water temperature in upper and lower torus zones are shown in Figure 10. Special focus is laid again on the thermal conditions within the torus and the resulting PCV pressure.
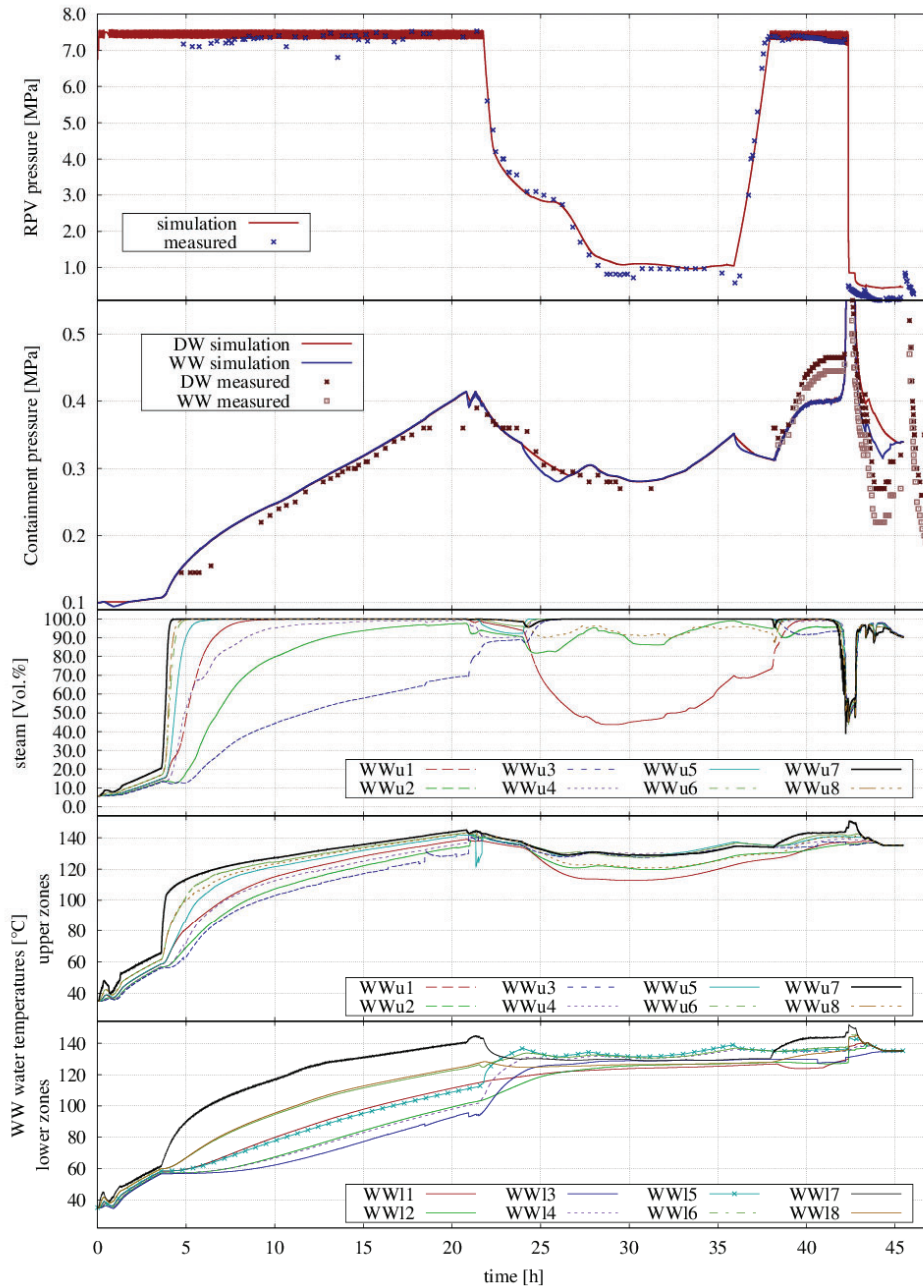
Figure 10. Calculated and measured RPV and containment pressure, and calculated steam concentration and water temperature in WW (Unit 3).

Almost the same nodalisation scheme of the torus is used (except no flooded torus room), with the same injection points of the operating SRV through its T-sparger, and the backflow of the RCIC turbine (zones WWl7 and WWu7). In addition, backflow through the operated (stronger) HPCI system was modelled,

but into a lower zone (WWl5, see Figure 4). In contrast to Unit 2, RCIC and HPCI system use vertical spargers instead of a simple vertical pipe for the backflow of water/steam into the torus, which causes some differences related to the condensation phenomena and the mixing of water.

Due to the parallel steam release through one SRV T-sparger and the sparger of the RCIC system, the WW water temperature was increasing quickly, especially near the release locations in zones WWl7 and WWu7 (Figure 10 above). The phenomena which are important in the early accident phase are similar. A short period of ~3.5 h with almost homogenous temperature increase in upper and lower WW zones is followed by a period of no longer homogeneous temperature distributions in the torus especially in azimuthal direction. As clearly to be seen in Figure 10, with increasing water temperature (larger than ~60 °C, later than ~3.5 h) condensation processes get more and more incomplete and steam and hot water exiting the RCIC sparger are flowing directly upwards to the pool surface, while decreasing the water mixing process. At the latest when the water temperature in the pool locally reaches boiling conditions, the steam released no longer condenses and the steam concentration in the torus gas space increases. Nevertheless, no uniform gas concentration exists in the torus gas space, due to steam condensation on cold water surfaces and releases from the WW into the DW by the operating vacuum breaker(s). Due to minor pressure differences not all of them are operated simultaneously; the number and location of vacuum breakers which were working in the plant are not known.

The RCIC system shut down automatically at ~22 h due to the high WW pressure. Shortly before the HPCI system could be started at ~23 h, mobile pumps were started to spray into the WW in order to decrease the PCV pressure (Figure 10). Even though equal spray water injection into each of the eight upper torus zones was modelled, the condensation effect varies significantly. In some of the zones the steam concentration did not change significantly (Figure 10), especially due to steam produced by boiling water. Due to the enhanced mixing of colder water from the lower torus zone WWl5, after activation of the HPCI system, together with the activated spray in the torus, the PCV pressure decreased (Figure 10).

The RPV pressure decreases sharply due to the large amount of steam taken to operate the HPCI system and the larger amount of water injected into the RPV, compared to RCIC injection before. The decreasing RPV pressure led to a degraded injection rate of the HPCI system and the RPV water level started to decrease. Unfortunately, there are no measured data available for the RPV water level at the time when the HPCI system was stopped at about 35.9 h. The simulated decrease is more pronounced than indicated by the measured values. The simulation was set up, so that the measured and the simulated water levels are equal at the time of the RPV depressurization at 42.4 h (Figure 11). At about 35.9 h in the analyses the water injection into the RPV by the HPCI system and the steam extraction were stopped. The pressure in the RPV went up again until one SRV started operating about two hours later. Than the RPV water level started decreasing (Figure 11) and uncovering the upper regions of the core.

The core heat-up process along with the core damage process is shown in Figure 12 for four different stages. It shows the calculated temperature profile of fuel, canister and control blade in each core ring (ROD1-ROD6). Contrary to the situation in Unit 2 the early heat-up process of the core is typical of a high pressure scenario with a continuously decreasing water level providing enough steam for core oxidation. At about 42 h the lower regions are still cooled by water and steam, while exothermic oxidation reaction of cladding and canisters accelerates the core heat-up even further. The amount of hydrogen produced and transferred into the containment is shown in Figure 13. Within one hour before the RPV depressurization at 42.4 h almost 550 kg of H_2 are generated. The H_2 released into the torus caused a homogenization of the gas concentration and a sharp increase of the PCV pressure (up to 0.66 MPa); the containment pressure was slightly underestimated before. Some H_2 was released through opened vacuum breakers into the DW region. At about 42.7 h the containment venting was initiated by opening a connection to the torus gas space. During the venting the pressure sharply decreased and a large amount of H_2 was released.
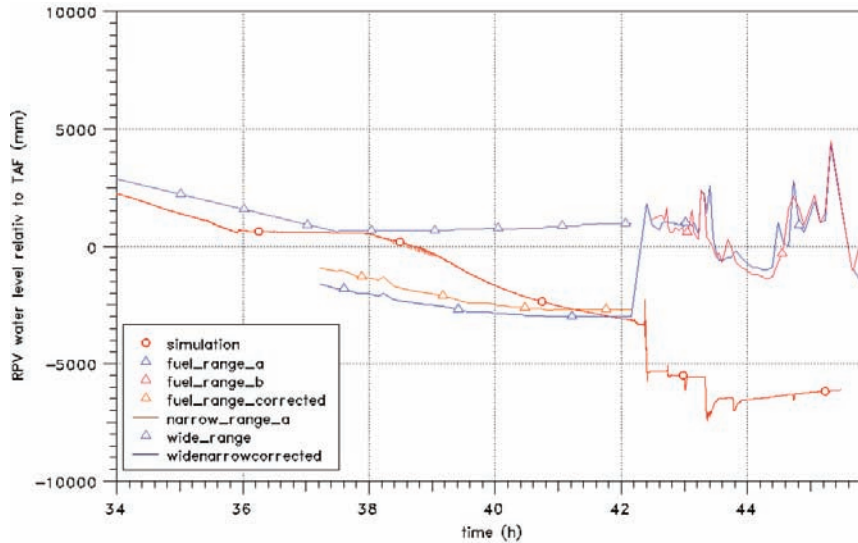
Figure 11. Calculated and measured RPV water level after the stop of HPCI (Unit 3).

At about 43 h a huge amount of ceramic melt is produced in the core in a very short time frame, and collected at the lower core plate (upper right graph in Figure 12). At that time the melt transfer into the lower plenum is started in the ATHLET-CD simulation. It seems reasonable to assume an early material relocation into the lower plenum in this BWR accident, since the number of openings in the core support plate is large; molten pool formation cannot be calculated yet by ATHLET-CD, but would be prevented by intact canisters in the lower core. The transferred melt runs down into the water pool in the lower plenum between the control blade guide tubes and gets in contact with RPV bottom head. The resulting steam production leads to an increase in the zirconium oxidation in the core and a further melting of fuel rods in the inner lower part of the core (lower right graph in Figure 12).
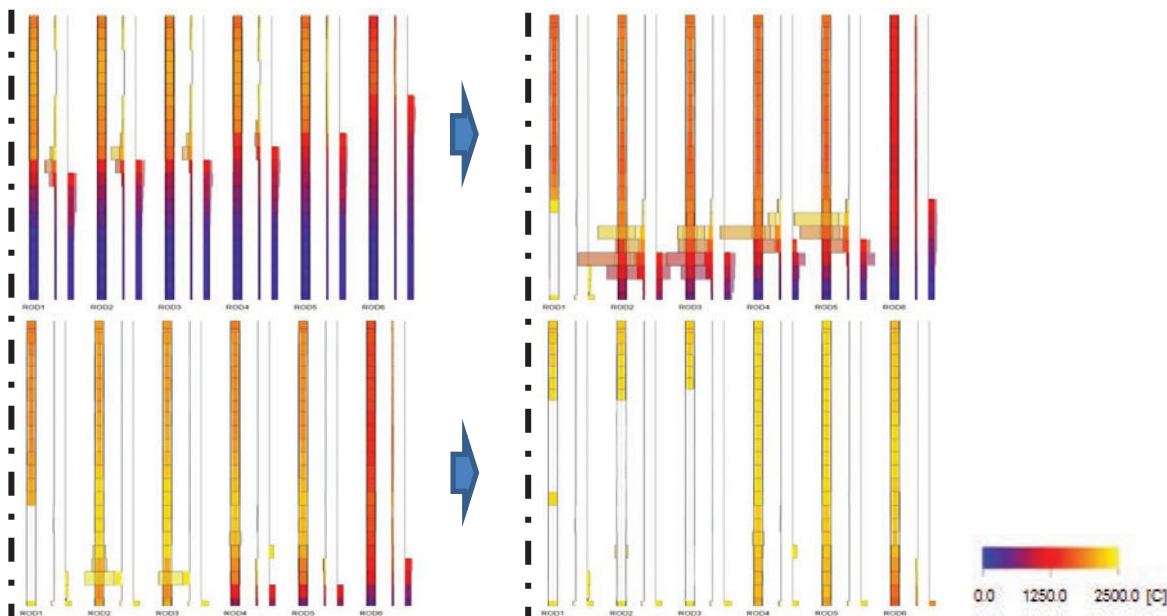
Figure 12. Calculated temperature of fuel, canister and control blade in core rings (ROD1-ROD6) at 42 h during core oxidation, at 43 h before material relocation into lower plenum, and thereafter at 43.9 h and at 45.7 h at RPV failure (Unit 3).
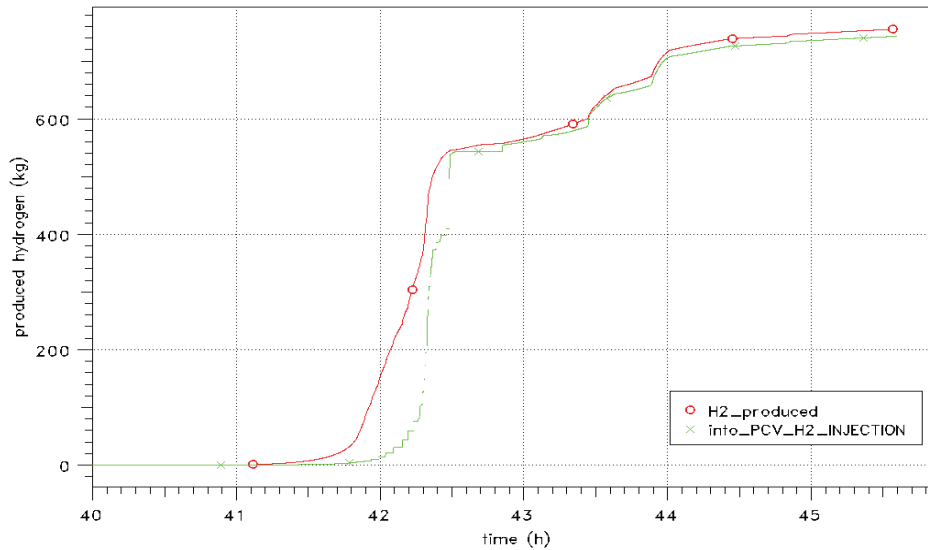
Figure 13. Calculated hydrogen production during core degradation and release into the PCV (Unit 3).

The simulation of Unit 3 would lead to RPV failure at about 46 h and corium discharge into the containment, but it had to be stopped as the transfer of corium from the RPV into the containment cannot be simulated yet with the used coupled ATHLET-CD/COCOSYS version.

5. CONCLUSIONS

At GRS large experience exists in applying the codes ATHLET and COCOSYS for accident analysis of German PWR and BWR [5-8]. The improvement made especially in the code development in the recent years and the availability of the coupled version of ATHLET-CD/COCOSYS allowed its application for analyses of the Fukushima Daiichi accidents in Units 2 and 3. The results obtained in this study are encouraging in terms of capturing essential severe accident signatures and phenomena. As done in other studies by GRS an appropriately detailed nodalisation of all important plant components (RPV, core and PCV) has been applied, with special focus on the phenomena expected in the WW. With almost the same nodalisation used for both Units, the calculation results capture all essential aspects of the accident progression, especially the PCV pressure transient and other important measured plant parameters.

Additional parameters have to be taken into account for the continuation of the analyses, awaiting further code model improvements. The thermohydraulic analysis is limited to a certain degree due to missing data. Especially the water level in unit 3 at the end of HPCI operation has a big impact on the final core status and the open question of the cause for the RPV depressurization. A more pronounced core relocation prediction into the lower plenum would require experimental data or information from the decommissioning process in future.

At this point in the study respectively the OECD/NEA BSAF project, the principal objective was achieved by evaluating the quality and quantity of the analyses in comparison with predicted versus observed data. The extent of core degradation was calculated. From a qualitative standpoint, both GRS analyses of the accidents in Units 2 and 3 using ATHLET-CD/COCOSYS produced accident sequence results that followed the general trends in the TEPCO data and observed events, and that provided results based on appropriate modelling of physical phenomena. The activities will continue in phase 2 of the project and should include analyses of the source term behavior, based on accident progression analyses and utilizing the common information database provided by TEPCO and the vendors.

NOMENCLATURE

ATHLET-CD	- Analysis of the Thermal Hydraulics of LEaks and Transients with Core Degradation in reactor circuit, GRS lumped parameter code
BDBA	- Beyond Design Basis Accidents
BSAF	- Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant
BWR	- Boiling Water Reactor
COCOSYS	- COntainment COde SYStem, GRS lumped parameter code
DC	- Direct Current
DW	- Drywell of the BWR containment
FP	- Fission Product
HPCI	- High Pressure Core Injection system
MSL	- Main Steam Line
OREST	- Code used to calculate the nuclide inventory in a reactor core
PWR	- Pressurized Water Reactor
RCIC	- Reactor Core Isolation Cooling system
RCS	- Reactor Coolant System
RPV	- Reactor Pressure Vessel
PCV	- Primary Containment Vessel
SA	- Severe Accident
SRV	- Safety Relief Valve
TH	- Thermal Hydraulic
WW	- Wetwell or torus of the BWR containment

ACKNOWLEDGMENTS

REFERENCES

1. OECD/NEA, "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project", https://www.oecd-nea.org/jointproj/bsaf.html
2. GRS, "ATHLET short description", http://www.grs.de/en/computer-code-athlet
3. GRS, "ATHLET-CD short description", http://www.grs.de/en/simulation-codes/athlet-cd
4. GRS, "COCOSYS short description", http://www.grs.de/en/content/cocosys
5. C. Bals, C. Köllein, F. Cester, T. Hollands, W. Luther, J.-D. Schubert, S. Weber, (GRS), *„Entwicklung von Kühlkreislaufmodellen zur Spätphase von Kernschmelzunfällen (ATHLET-CD und ASTEC)"*, Abschlussbericht, GRS-A-3646, (März 2012)
6. H. Austregesilo, Ch. Bals, J. Herb, T. Hollands, A. Papukchiev, Ph. Schöffel, S. Weber, (GRS), *„Validierung von Rechenprogrammen zur Simulation von Stör- und Unfällen im Reaktorkühlsystem"*, Abschlussbericht, GRS-A-3706, (Juli 2013)
7. C. Spengler, et al., (GRS) *„Weiterentwicklung der Rechenprogramme COCOSYS und ASTEC"*, GRS-A-3654 (August 2012)
8. W. Klein-Heßling, et al., (GRS), *„Gezielte Validierung von COCOSYS und ASTEC sowie Unsicherheits- und Sensitivitätsanalyse zum Iodverhalten"*, GRS-A-3691, (Dezember 2012)
9. OECD/NEA, *"OECD/NEA THAI Project: Hydrogen and Fission Product Issues Relevant for Containment Safety Assessment under Severe Accident Conditions"*, NEA/CSNI/R(2010)3, (2010)
10. OECD/NEA, "OECD/NEA THAI 2 project", http://www.oecd-nea.org/jointproj/thai2.html