OVERVIEW OF LWR SEVERE ACCIDENT RESEARCH ACTIVITIES AT THE KARLSRUHE INSTITUTE OF TECHNOLOGY

Alexei Miassoedov, alexei.miassoedov@kit.edu
Giancarlo Albrecht, giancarlo.albrecht@kit.edu
Jerzy-Jan Foit, jerzy.foit@kit.edu
Thomas Jordan, thomas.jordan@kit.edu
Martin Steinbrück, martin.steinbrueck@kit.edu
Juri Stuckert, juri.stuckert@kit.edu
Walter Tromm, walter.tromm@kit.edu
Karlsruhe Institute of Technology, Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

Abstract. *The research activities in the light water reactor (LWR) severe accidents domain at Karlsruhe Institute of Technology (KIT) are concentrated on the in- and ex-vessel core melt behavior. The overall objective is to investigate the core melt scenarios from the beginning of core degradation to melt formation and relocation in the vessel, possible melt dispersion to the reactor cavity and to the containment, corium concrete interaction and corium coolability in the reactor cavity, and hydrogen behaviour in reactor systems. The results of the experiments contribute to a better understanding of the core melt sequences and thus improve safety of existing and, in the long-term, of future reactors by severe accident mitigation measures and by safety installations where required. This overview paper describes the experimental facilities used at KIT for severe accident research and gives an overview of the main directions and objectives of the R&D work.*

Keywords*: LWR, reactor safety, severe accidents, corium behavior*

1. INTRODUCTION

Severe accidents can cause significant damage to reactor fuel resulting in a substantial core meltdown and threaten the containment integrity. They are the focus of considerable research, because the release of radioactive products into the environment would have serious consequences. The research priorities on severe accident management were prepared in the framework of the SARNET Network (Van Dorsselaere et al., 2012) by the SARP group (Klein-Heßling et al., 2012) and include the following high priority (further research is considered as necessary) and medium priority issues (should be investigated further as already planned in the different research programs):

- Core coolability during reflood and debris cooling;
- Ex-vessel melt pool configuration during Molten Corium Concrete Interaction (MCCI), ex-vessel corium coolability by top flooding;
- Melt relocation into water, ex-vessel Fuel Coolant Interaction (FCI);
- Hydrogen mixing and combustion in containment;
- Oxidizing impact (Ruthenium oxidizing conditions/air ingress for high burn-up and mixed oxide fuel elements) on source term;
- Iodine chemistry in Reactor Coolant System (RCS) and in containment.
- Hydrogen generation during reflood and melt relocation in vessel;
- Corium coolability in lower head;
- Integrity of Reactor Pressure Vessel (RPV) due to external vessel cooling;
- Direct containment heating (DCH).

These phenomena are extremely complex and generally demand the development of specific research; the research field is too wide to allow investigation of all phenomena by any national program. Therefore the severe accident research activities at KIT are strongly linked to other European facilities and experimental programs and are part of the SARNET2 project of the 7th EU Framework Programme, providing experimental resources to investigate core melt scenarios from the beginning of core degradation to melt formation and relocation in the vessel, possible melt dispersion to the reactor cavity and to the containment, molten corium concrete interaction and finally hydrogen-related phenomena in severe accidents. The main thrust of the experiments is towards large-scale tests under prototypical conditions addressing most of the high and medium priority issues identified by the SARP group of SARNET.

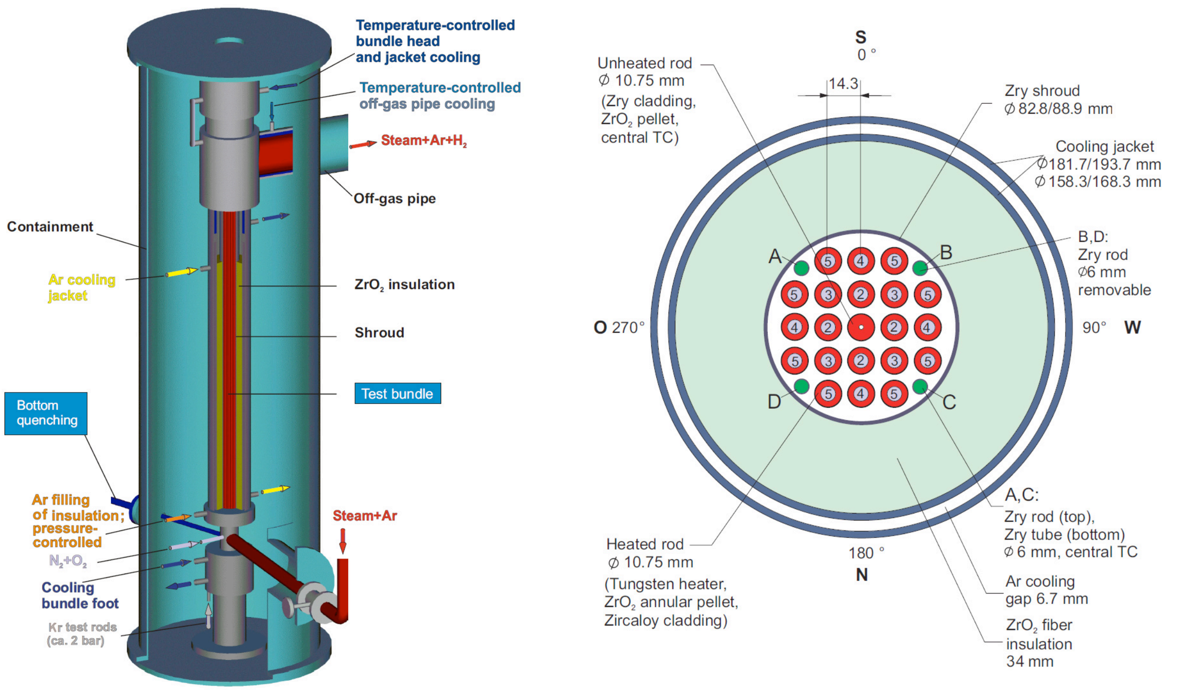
2. SEVERE ACCIDENT RESEARCH FACILITIES AT KIT

The severe accident research experimental platform at KIT includes the following facilities:

2.1 QUENCH – large-scale tests on investigations of the early and late phases of core degradation

Bundle experiments in the QUENCH facility are designed to study the early and late phases of core degradation in prototypic geometry for different reactor designs and different cladding alloys for a proper assessment of the risk posed by quenching of degraded core to full-scale power plants. The QUENCH program (Steinbrück et al., 2010) aims not only to determine the amount of hydrogen released during reflood of a test bundle with genuine core materials as cladding and spacer grids, but also to investigate the related high-temperature interactions of the core materials providing comprehensive data for model development and subsequent implementation into Severe Fuel Damage (SFD) computer codes.

The main component of the QUENCH test facility is the test section with the test bundle (Fig. 1a)). The facility can be operated in two modes: a forced-convection mode (typical for most QUENCH experiments) and a boil-off mode.



a) b)

Figure 1. a) QUENCH containment and test section; b) QUENCH bundle cross section.

The test bundle is approximately 2.5 m long and is made up of 21 fuel rod simulators. Except the central one all rods are heated. Heating is electric by 6 mm diameter tungsten heaters of length 1024 mm installed in the rod center. The rod cladding of the heated and unheated fuel rod simulators is Zircaloy-4 with 10.75 mm outside diameter and 0.725 mm wall thickness. All test rods are filled with Kr at a pressure of approx. 0.22 MPa absolute to observe the first cladding failure as well as a failure progression by detecting the Kr in the off-gas.

The bundle is surrounded by a 3 mm shroud of Zirconium-702 (inner diameter 82.8 mm) with a 34 mm thick ZrO_2 fiber insulation extending from the bottom to the upper end of the heated zone and a double-walled cooling jacket of Inconel (inner tube) and stainless steel (outer tube) over the entire length. The annulus between shroud and cooling jacket is purged (after several cycles of evacuation) and then filled with stagnant argon at 0.22 MPa absolute. The annulus is connected to a flow- and pressure-controlled argon feeding system in order to keep the pressure constant at the target of 0.22 MPa and to prevent an access of steam to the annulus after shroud failure. The 6.7-mm annulus of the cooling jacket is cooled by argon flow from the upper end of the heated zone to the bottom of the bundle and by water in the upper electrode zone. Both the absence of ZrO_2 insulation above the heated region and the water cooling are to avoid too high temperatures of the bundle in that region.

The off-gas including Ar, H_2 and steam is analyzed by a state-of-the-art mass spectrometer Balzers "GAM300" located at the off-gas pipe ~2.7 m downstream the test section. The mass spectrometer allows also to indicate the failure of rod simulators by detection of Kr release.

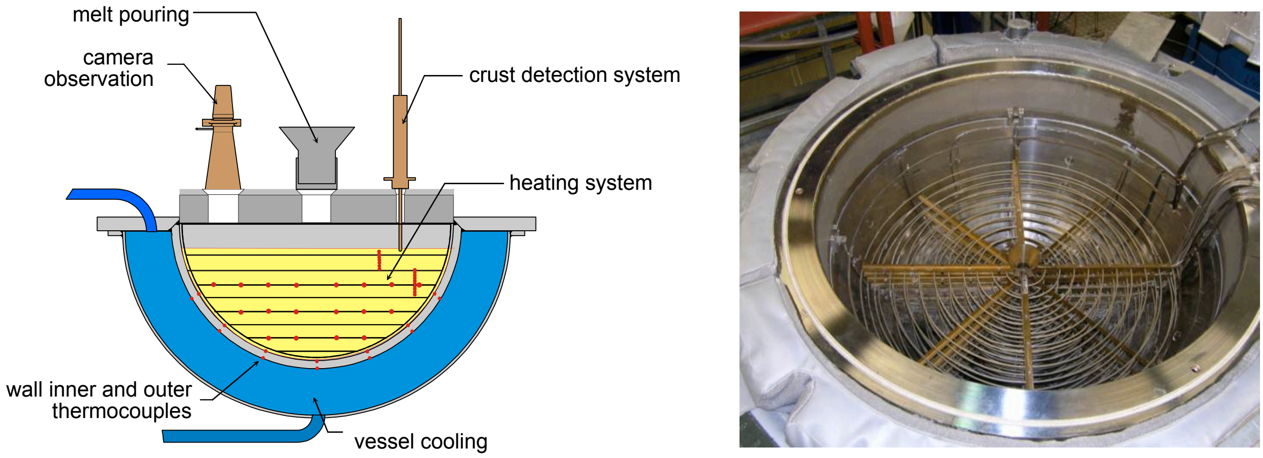
The test bundle, shroud, and cooling jacket are extensively equipped with sheathed thermocouples at different elevations with an axial step of 100 mm. There are 40 high-temperature (W/Re) thermocouples in the upper hot bundle region (elevations between 650 and 1350 mm) and 32 low-temperature (NiCr/Ni) thermocouples in the lower "cold" bundle region (bundle and shroud thermocouples between –250 and 550 mm). At elevations 950 and 850 mm there are two centerline high-temperature thermocouples in the central rod, which are protected from oxidizing influence of the steam. Two thermocouples isolated from steam are installed at the same elevations inside the corner rods A and C. Other bundle thermocouples are attached to the outer surface of the rod cladding. The shroud thermocouples are mounted at the outer surface of Zircaloy-4 shroud. Additionally the test section incorporates pressure gauges, flow meters, and a water level detector. Further details of the QUENCH facility and operation are given by Schanz et al. (2006).

Up to now, 16 bundle tests have been conducted; the main topics investigated include hydrogen source term during reflood, influence of B_4C (Sepold et al., 2006) and Ag-In-Cd control rods (Sepold et al., 2009) on bundle degradation, effect of air ingress (Steinbrück et al., 2006; Birchley et al., 2012) on oxidation and degradation of the core, and specific behavior of VVER bundle geometry and materials during oxidation and reflood (Stuckert et al., 2009). One test was performed with the complete sequence including boil-off phase, pre-oxidation and reflood. In the future the QUENCH experiments will focus on the analysis of the relocation of cladding and fuel and the formation and cooling of in-core debris beds to obtain information on the characteristics of the created debris particles. The main objective of these tests is the investigation of these processes under prototypical boundary conditions for a whole bundle.

2.2 LIVE – large-scale tests on behavior of the corium melt pool

The main objective of the LIVE program at is to study the late in-vessel core melt behavior and core debris coolability both experimentally in large scale 2D and 3D geometry and in supporting separate-effects tests (Miassoedov et al., 2011), and analytically using CFD codes (Palagin et al., 2011) in order to provide a reasonable estimate of the remaining uncertainty band under the aspect of safety assessment. The LIVE-3D test facility is the first facility, which allows the investigation of a melt with simulated internal heat generation in a 3D geometry of a lower plenum of a RPV (Gaus-Liu et al., 2010). The experiments performed in other facilities had only a 2D geometry: BALI (Bernaz et al., 2001), SIMECO (Kolb et al., 2000) and COPO (Kymäläinen et al., 1994) or were performed without heating of the melt like in ACOPO (Theofanous et al., 1997). To compare the experimental results of 3D experiments in LIVE-3D with 2D experiments, the LIVE-2D test facility was also constructed and a series of tests have been performed.

The main part of the LIVE-3D test facility is a 1:5 scaled semi-spherical lower head of a typical pressurized water reactor, as shown in Fig. 2a) and Fig. 2b). The diameter of the test vessel is 1 meter. The top of the test vessel is covered with an insulated lid. The test vessel is enclosed in a cooling vessel to simulate the external cooling. The melt is prepared in an external heating furnace designed to generate 220 l of the simulant melt. The volumetric decay heat is simulated by means of 6 heating planes (Fig. 2b)) providing a maximum power of about 28 kW.



a) b)

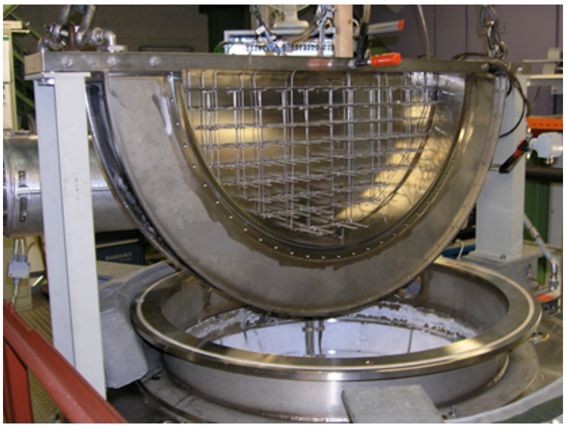
Figure 2. a) scheme of the LIVE-3D test vessel; b) top view of the LIVE-3D vessel with heating coils.

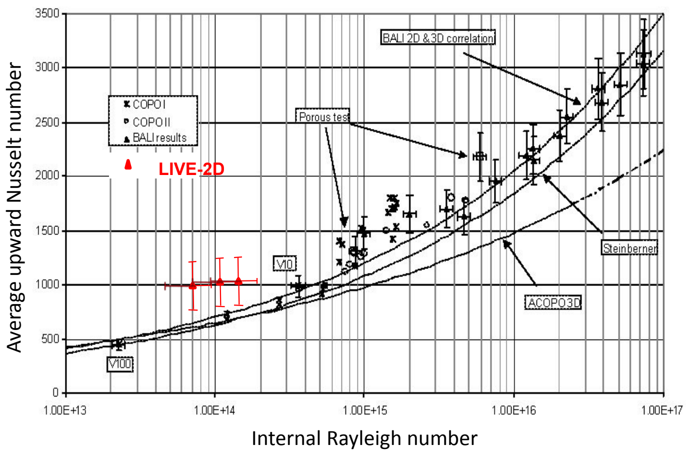
To investigate both the transient and the steady state behavior of the simulated corium melt, an extensive instrumentation of the test vessel is realized. The temperatures of the vessel wall inner and outer surface are measured at 5 latitudes and 4 locations at each latitude. Heat flux distribution through the vessel wall can be evaluated post-test based on the measured temperatures. Additionally, 80 thermocouples are positioned within the vessel to measure the temperature distribution in the melt pool and in the crust formed at the interface between the vessel wall and the melt.

Simulant materials used in the LIVE program should, to the greatest extent possible, represent the real core materials in important physical properties and in thermo-dynamic and thermo-hydraulic behavior. Therefore, the applicability of several binary melt compositions as a simulant for the oxidic part of the corium has been investigated. Important criteria for the selection were that the simulant melt should be a non-eutectic mixture of several components with a distinctive solidification range of about 100 K, and that the simulant melt should have a similar solidification and crust formation behavior as the oxidic corium.

Binary melts composed of KNO_3 and $NaNO_3$ were selected as simulant melts for the experiments both in non-eutectic mixture of 80 mole% KNO_3–20 mole% $NaNO_3$ and in eutectic mixture of 50 mole% KNO_3–50 mole% $NaNO_3$ (eutectic temperature is 225 °C). The solidus temperature and the liquidus temperature of the non-eutectic composition are about 225 °C and 284 °C, respectively (Berg and Kerridgeb, 2004). These melts can be used in a temperature range from 225 °C (solidification) to 380 °C (chemical decomposition). Due to its solubility for water the applicability of such melts is restricted to dry conditions inside the test vessel.

In the LIVE-2D test facility (Fig. 3a)) the melt pool in a lower plenum of a RPV is investigated in the slice having a thickness of 12 cm and a radius of 0.5 m (the same radius as the hemisphere in LIVE-3D). The vessel wall, represented by a ~25 mm thick stainless steel plate is cooled by water. The water flow rate can be regulated and the maximum flow rate is high enough to ensure that the temperature difference of the water between the entrance and the exit is less than 2 K. The front and the back faces of the slice are made of 1 mm thick stainless steel and are well insulated. The LIVE-2D test vessel can be covered at the top by an insulated or by a water-cooled lid. The results of the performed LIVE-2D experiments regarding the upward heat transfer were compared to earlier tests performed in the BALI and SIMECO facilities as well as to correlations developed from those tests. For internal Rayleigh numbers lower than 10^{15} the results of the LIVE-2D tests demonstrated higher heat transfer to the top of the melt, as it is shown in Fig. 3b). One of the objectives of the future tests will be to check if this deviation also persists for higher Rayleigh numbers and to identify the main reasons for this behavior. The comparison with the 3D experiments is still outstanding, because the relevant 3D experiments in the LIVE-3D facility have not been performed yet.





a) b)

Figure 3. a) LIVE-2D test vessel; b) comparison of the upward heat transfer in LIVE-2D tests with earlier experiments.

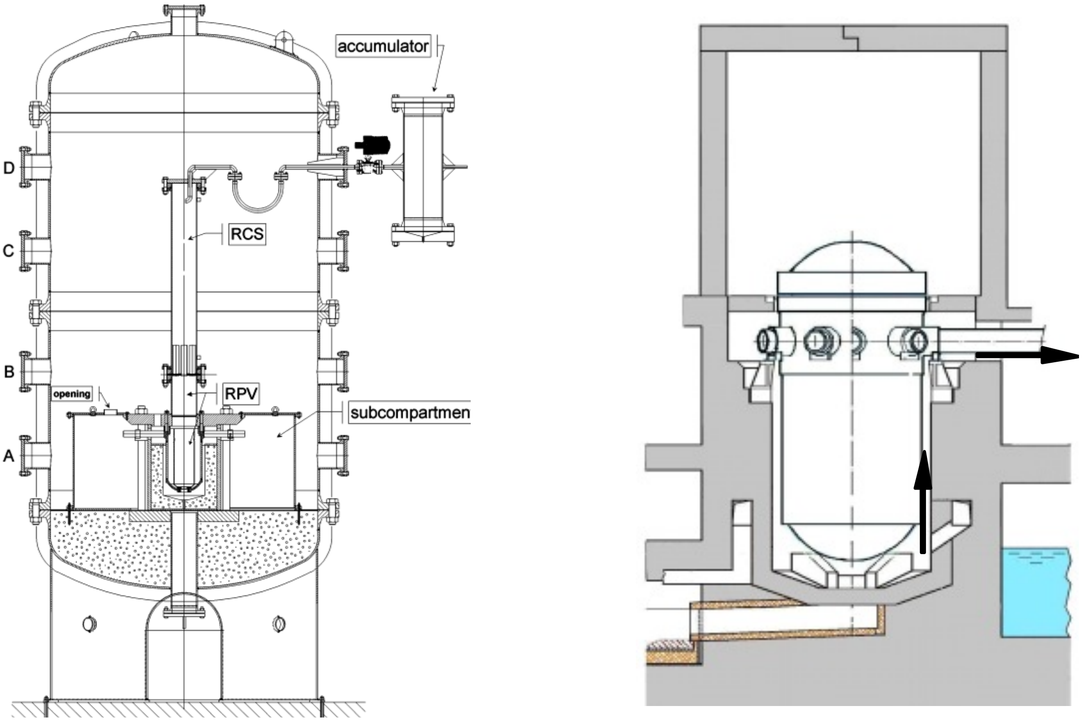
Generally, the information obtained from the LIVE 2D and 3D experiments includes heat flux distribution along the reactor pressure vessel wall in transient and steady state conditions, crust growth velocity and influence of the crust formation on the heat flux distribution along the vessel wall. Complementary to other international programs with real corium, the results of the LIVE experiments provide data for a better understanding of in-core corium pool behavior. They also allow a direct comparison with findings obtained earlier in other experimental programs and are being used for the development and assessment of mechanistic models for description of in-core molten pool behavior and their implementation in the severe accident codes.

2.3 DISCO – large-scale tests on melt dispersion and DCH

The DISCO experiments (Meyer at al., 2009) are designed to investigate the fluid-dynamic, thermal and chemical processes during melt ejection out of a breach in the lower head of a PWR pressure vessel at pressures below 2 MPa with an iron-alumina thermite melt and steam. In the frame of these investigations the following issues are addressed: final location of corium debris, loads on the reactor pit and the containment in respect to pressures and temperatures, and the amount of hydrogen produced and burned during DCH event.

The main components of the facility (Fig. 4a)) are scaled about 1:18 linearly to a large PWR. The model of the containment pressure vessel has a height of 5.80 m and a total volume of 14 m³. The volumes of the reactor cooling system (RCS) and the reactor pressure vessel are modeled by a vertical pipe. The reactor pit is made of concrete and is installed inside a strong steel vessel. The main cooling lines are modeled by eight horizontal steel cylinders with a scaled annular space around each of them, modeling the flow path leading into the equipment rooms. The equipment rooms are modeled according to the reactor design being investigated (Fig. 4 b)).

The containment vessel is closed at atmospheric pressure and room temperature. In most experiments a containment atmosphere was aimed at, as it can be expected during a core melt accident, with steam and a certain hydrogen content. For a period of up to 8 hours steam is filled into the containment vessel additional to the air atmosphere until the pressure reaches 0.2 MPa and the gas temperature is close to 100 °C, while the condensate is frequently drained. At the end of heat-up a metered amount of hydrogen gas (3-6 mol-%) is added through pipes leading into the subcompartment and the upper dome.

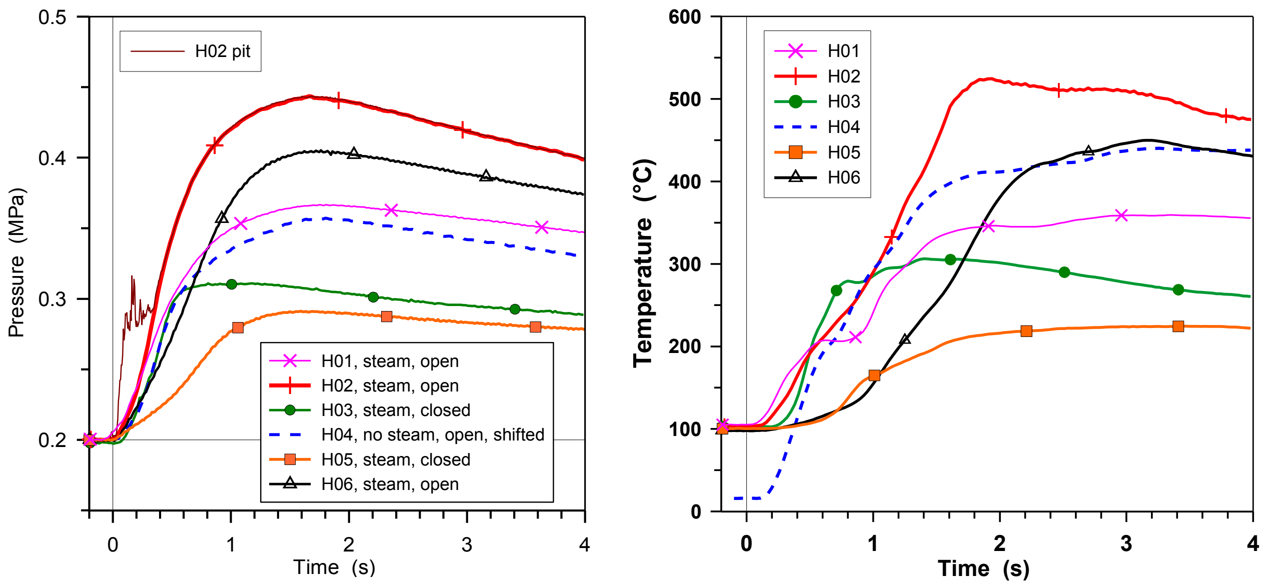


a) b)

Figure 4. a) scheme of the DISCO facility with the model of the EPR cavity; b) configuration of the EPR plant.

The experiment is started by igniting the thermite electro-chemically at the upper surface of the compacted thermite powder. About 3 to 6 seconds after ignition the brass plug at the bottom of the RPV vessel is melted by the 2100 °C hot iron-alumina mixture. That initiates the melt ejection. The melt is driven out of the breach by the steam and is dispersed into the cavity and beyond.

Standard test results include pressure (Fig. 5a)) and temperature (Fig. 5b)) history in the RPV, the cavity, the reactor compartments and the containment vessel, post-test melt fractions in all locations with size distribution of the debris, video film in the subcompartments and containment (timing of melt flow and hydrogen burning), and pre- and post-test gas analysis in the cavity and the containment. The gas analysis allows determining the amount of produced, burned and remaining hydrogen.



a) b)

Figure 5. a) containment pressures in EPR tests; b) containment temperatures in EPR tests.

2.4 MOCKA –large-scale experiments on molten corium concrete interaction

Safety assessment of nuclear power plants requires detailed understanding of the interaction of the corium with the concrete. MCCI is characterized by intense coupling of complex phenomena, such as:

- high temperature behavior of the concrete and its decomposition;
- thermal hydraulics and heat transfer of the corium pool agitated by gas bubbles;
- physical chemistry of the multi-component melt as influenced by the changes in material composition through admixture of molten concrete and oxidation of metals;
- solidification processes and the behavior of interfacial crusts.

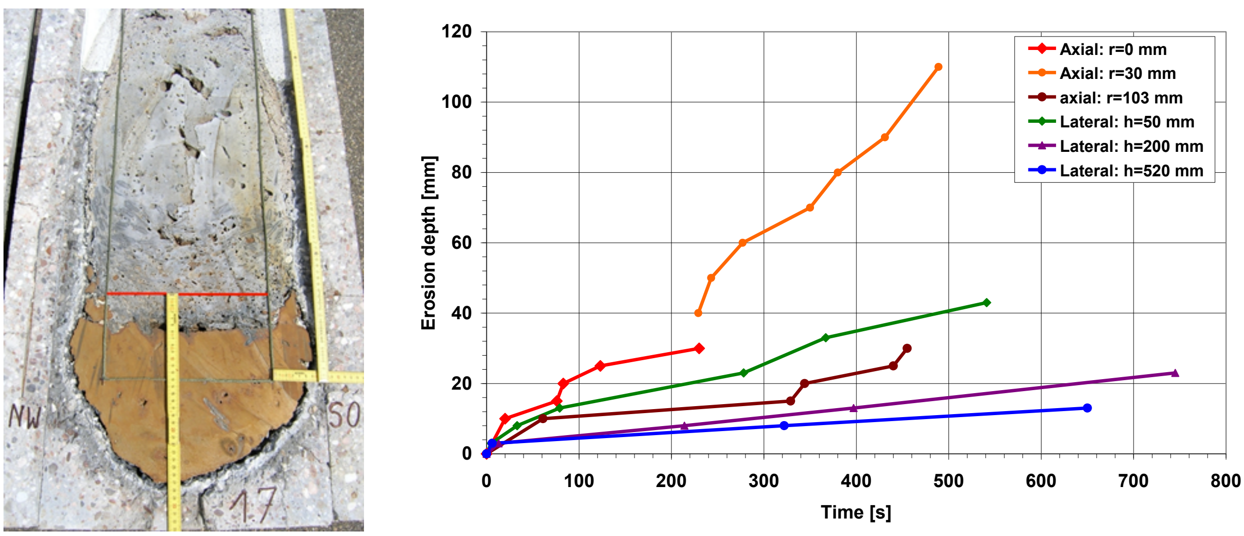
Identification and quantification of these phenomena with respect to their importance for accident analysis requires a variety of integral experiments, separate effect tests and development of models and their verification. Even though extensive research has been undertaken over several years in the area of corium-concrete interaction, several subjects still require further investigations. One of the important issues concerns the distribution of the heat flux to the concrete in the lateral and axial directions during the long-term 2-dimensional concrete erosion by a core melt. The knowledge of this partition is important in the evaluation of the consequences of a severe reactor accident.

The MOCKA facility (Foit et al., 2012) is a new facility which is designed to investigate the corium/concrete interaction in an anticipated core melt accident in LWRs, after the metal melt is layered beneath the oxide melt. The experimental focus is on the cavity formation in the basemat and the risk of a long-term basemat penetration by the metallic part of the melt. Experiments in the MOCKA facility generally consist of the following phases:

- Up to 3200 kg of melt is generated in a concrete crucible by thermite reaction, resulting in steel melt (Fe, Ni) and oxide melt (Al_2O_3, CaO, FeO). Steel melt is at the bottom, oxide on top (corresponding to reactor situation after admixture of eroded concrete).
- In addition to the transient tests, internal heat generation in the oxide phase is also simulated by adding thermite briquettes from the top.
- End of the experiment is defined by maximal concrete erosion.

The melt is generated by ignition of the specific thermite mixture in a concrete crucible. After completion of the chemical reaction the melt separates into the lighter oxide melt on top and the heavier metal melt at the bottom. The initial temperature of the melt is in the range of 1800-1900 °C and the melt has sufficient overheat over the liquidus temperature to show the important effects of concrete erosion also in transient tests. To extend the duration of the interaction with the concrete and to allow for significant concrete erosion by the oxide as well as by the metal melt, a succession of additions of pure thermite and Zr metal from the top into the oxide layer is implemented. The additional enthalpy generated by the thermite reaction and exothermal oxidation reactions of Zr is mainly deposited in the oxide phase. Typical concrete erosion in MOCKA tests is shown in Fig. 6a). Initial size of the crucible is indicated, the red line shows the initial height (13 cm) of the metal melt.

To detect the time dependent erosion front and to control the course of the experiment the crucible is instrumented with NiCr-Ni (Type K) thermocouples embedded in the concrete. In most cases, their failure indicates the arrival of the melt front. It is important that the thermocouples give sufficient information during the test about the actual position of the melt front (Fig. 6b)). No thermocouples are installed to measure the temperature of the melt in the crucible during the erosion, because no stable thermocouples exist for this critical application.



a) b)

Figure 6. a) cross-section of the MOCKA 1.7 crucible; b) position of the melt front as a function of time in the MOCKA 1.7 test.

2.5 HYKA – large-scale tests on hydrogen behavior and mitigation

In the case of a severe accident with and without failure of the reactor pressure vessel, the containment is the ultimate barrier to the environment. The HYKA facility provides unique research capabilities for investigation of hydrogen related phenomena in the containment during severe accidents including the issues of hydrogen distribution, hydrogen combustion and hydrogen mitigation measures (Jordan and Breitung, 2007).

HYKA facilities provide experimental possibilities for containment safety research in Europe through a number of large test vessels which are qualified and approved for operation with hydrogen combustion. The tests can be made under stagnant or under controlled air flow conditions, as well as in horizontal or vertical orientation. Due to the high vessel design pressures test parameters are not restricted by safety considerations. Highly energetic experiments can be performed on the KIT premises with all necessary infrastructure (control rooms, data acquisition, gas preparation and filling systems, workshops). In HYKA it is possible to investigate the whole spectrum of hydrogen phenomena. Research on different hydrogen sources and their distribution behavior can be conducted, as well as experiments with different ignition sources. One of the most attractive features of HYKA is the capability for well-controlled medium to large scale combustion experiments, covering all three combustion regimes, i.e. slow and fast deflagration and detonation. The main technical details of the different HYKA facilities and the phenomena addressed in the tests are summarized in the Tab. 1.

Table 1. Main parameters of the test vessels of the HYKA facility.

Name	Type	Dimensions (m)	Volume (m^3)	Design pressure (bar)	Phenomena
A1	cylindrical vessel	diam. 3.4 length 12.0 (horizontal)	98	100	Large scale tests on turbulent combustion, flame acceleration, detonation, vented explosions
A2	cylindrical vessel	diam 6.0 m length 10.5 m (vertical)	220	10	Large scale tests on turbulent combustion with mixture gradients, standing diffusion flames, vented explosions, interaction of recombiners with containment flows, test of deliberate ignition mitigation schemes
A3	cylindrical vessel	diam. 2.5 length 8.0 (vertical)	33	60	hydrogen distribution, stratification, recombiner and igniter tests, uniform and non-uniform mixtures
A6	cylindrical vessel	diam. 3.3 height 3.1	22	40	as in A1, two large vents (0.8 m), H_2 distribution in closed rooms, integrity of mechanical structures
A8	cylindrical vessel	diam. 1.8 length 3.0	9	100	fast deflagration and detonation at high initial pressure

The HYKA experiments provide information on:
- turbulent combustion, flame acceleration, detonation, vented explosions,
- hydrogen distribution, stratification, recombiner and igniter performance for uniform and non-uniform gas mixtures,
- criteria for prediction of flame acceleration and detonation onset in hydrogen/air mixtures,
- fast deflagration and detonation at high initial pressures,
- vented combustion, hydrogen distribution and local detonations in closed spaces,
- behavior of hydrogen jets in mixed air-steam-hydrogen atmospheres,
- influence of the combustion of hydrogen produced by oxidation during melt ejection from RPV on the containment pressurization.

3. CONCLUSIONS

The severe accident research platform at KIT includes several experimental facilities which are designed to study the remaining severe accident safety issues, including the coolability of a degraded core, corium coolability in the RPV, possible melt dispersion to the reactor cavity, molten corium concrete interaction and hydrogen mixing and combustion in the containment. These facilities are unique in providing experimental programs in specific fields of core damage initiation up to hydrogen behavior and are designed to be complementary to other European facilities and experimental platforms to form a coherent European nuclear experimental network.

The experimental programs are strongly coupled with other European projects, such as PLINIUS and SARNET2, as well as with third countries (Russian Federation, Ukraine, and Kazakhstan) through the ISTC and the STCU. The experimental results are used for the development of models and their implementation in the severe accident codes such as e.g. ASTEC. This helps to capitalize the knowledge obtained in the field of severe accident research in codes and the scientific databases, thus preserving and diffusing this knowledge to a large number of current and future end-users.

4. REFERENCES

Berg, R.W., Kerridgeb, D.H., 2004. The NaNO3/KNO3 system: the position of the solidus and sub-solidus. Dalton Transactions, pp 2224–2229.
Bernaz, L., Bonnet, J.M. and Seiler J.M., 2001. Investigation of natural convection heat transfer to the cooled top boundary of a heated pool. Nuclear Engineering and Design, 204, pp. 413-427.
Birchley, J. et al., 2012. WP5.1: Conduct and Analytical Support to Air Ingress Experiment QUENCH-16. In Proceedings of 5th European Review Meeting on Severe Accident Research (ERMSAR-2012), Cologne (Germany), March 21-23, 2012.
Foit, J.J. et al., 2012. MOCKA Experiments on Concrete Erosion by a Metal and Oxide Melt. In Proceedings of 5th European Review Meeting on Severe Accident Research (ERMSAR-2012), Cologne (Germany), March 21-23, 2012.
Gaus-Liu, X. et al., 2010. In-vessel melt pool coolibility test —Description and results of LIVE experiments. Nuclear Engineering and Design, 240 (2010), pp. 3898–3903.
Jordan, T. and Breitung, W., 2007. FZK methodology for analysis of hydrogen behaviour in containments. In Proceedings of Conference on Numerical Flow Models for Controlled Fusion, Porquerolles, France, April 16- 20, 2007.
Klein-Heßling, W. et al., 2012. Ranking of Severe Accident Research Priorities. In Proceedings of 5th European Review Meeting on Severe Accident Research (ERMSAR-2012), Cologne (Germany), March 21-23, 2012.
Kolb, G., Theerthan, S.A. and Sehgal, B.R., 2000. Experiments on in-vessel melt pool formatin and convection with NaNO3-KNO3 Salt Mixture as melt simulant In Proceedings of ICONE8, Baltimore, MD, USA, April 2-6, 2000.
Kymäläinen, O. et al., 1994. Heat flux distribution from a volumetrically heated pool with high Rayleigh number. Nuclear Engineering and Design, 149, pp. 401-408.
Miassoedov, A. et al., 2011. LIVE experiments on melt behavior in the reactor pressure vessel lower head. In Proceedings of 8th Internat. Conf. on Heat Transfer, Fluid Mechanics and Thermodynamics (HEFAT 2011), Pointe aux Piments, MS, July 11-13, 2011.
Meyer, L. et al., 2009. Direct containment heating integral effects tests in geometries of European nuclear power plants., Nuclear Engineering and Design, 239, Issue 10, pp. 2070-2084.
Palagin, A., Kretzschmar, F. and Chudanov, V., 2011. Application of the CFD CONV code to the simulation of LIVE L-4 test. In Proccedings of the Internat. Congress on Advances in Nuclear Power Plants (ICAPP'11), Nice, F, May 2-5, 2011, Proc. Paper 11096.
Schanz, G. et al., 2006. Results of the QUENCH-10 experiment on air ingress. Scientific Report, FZKA-7087, SAM-LACOMERA-D09, Karlsruhe. http://bibliothek.fzk.de/zb/berichte/FZKA7087.pdf
Sepold, L. et al., 2006. Results of the QUENCH-09 experiment compared to QUENCH-07 with incorporation of B4C absorber. Nuclear Technology, 154. 2006, pp. 107-16.
Sepold, L. et al., 2009. Results of the AgInCd Absorber Rod Experiments QUENCH-13. Forschungszentrum Karlsruhe Report FZKA-7403.
Steinbrück, M. et al., 2006. Experiments on air ingress during severe accidents in LWRs. Nuclear Engineering and Design, 236, 2006, pp. 1709-19.
Steinbrück, M., Große, M., Sepold, L. and Stuckert, J., 2010. Synopsis and outcome of the QUENCH experimental program. Nuclear Engineering and Design, 240, 2010, pp. 1714-1727.
Stuckert, J. et al., 2009. Experimental and post-test calculation results of the integral reflood test QUENCH-12 with a VVER-type bundle. Annals of Nuclear Energy, 36, 2009, pp. 183-92.
Theofanous, T. G. et al., 1997. The first results from the ACOPO experiments. Nuclear Engineering and Design, 169, pp. 49-57.
Van Dorsselaere, J.-P., Fichot, F. and Seiler, J.-M., 2006. Views on R&D needs about in-vessel reflooding issues, with a focus on debris coolability, Nuclear Engineering and Design, 1976–1990 (2006), pp. 236.
Van Dorsselaere, J.-P. et al., 2012. The European Research on Severe Accidents in Generation-II and -III Nuclear Power Plants. Science and Technology of Nuclear Installations, vol. 2012, Article ID 686945, 12 pages, 2012. doi:10.1155/2012/686945.

5. RESPONSIBILITY NOTICE

The authors are the only responsible for the printed material included in this paper.