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LWR SEVERE ACCIDENT RESEARCH AT THE KARLSRUHE INSTITUTE OF TECHNOLOGY

Alexei Miassoedov, alexei.miassoedov@kit.edu
Thomas Jordan, thomas.jordan@kit.edu
Martin Steinbrück, martin.steinbrueck@kit.edu
Walter Tromm, walter.tromm@kit.edu
Karlsruhe Institute of Technology, Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

Abstract. *Karlsruhe Institute of Technology (KIT) has a profound knowledge and experience in the R&D of nuclear reactor safety and in national and international co-operation. It has a long experience in performing the large- and small-scale experiments in the specific fields of the core damage initiation and in-vessel and ex-vessel melt progression phenomena. A unique combination of facilities and expertise is available at the KIT. The expertise is maintained by participating in collaborative research with laboratories in the EU and other countries. The KIT is considered as a powerful internationally-recognized centre of reference for nuclear research. The aspects of reactor safety and safety of nuclear waste disposal are coordinated by the Program of Nuclear Safety Research. The research activities in the light water reactor (LWR) severe accidents domain at KIT are concentrated on the in- and ex-vessel core melt behavior. The overall objective is 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, and finally corium concrete interaction and corium coolability in the reactor cavity. The experiments are designed to be complementary to other European facilities and experimental platforms, such as CODEX, VULCANO or PLINIUS to form a coherent European nuclear experimental network. They 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. The experimental programs are part of the SARNET2 and LACOMECO projects of the 7th EU Framework Programme. The results of the experiments are being used for the validation of codes applied for safety assessment and planning of accident mitigation concepts, such as ASTEC. 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 more or less complete 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. This research also reflects a commitment to the defence-in-depth approach.

As stated in the final draft of the Strategic Research Agenda (SRA) (SNETP, 2009) of the Sustainable Nuclear Energy Technology Platform (SNETP), needs for safety research are identified by both regulators and operators, from their respective perspective. As discussed in the SRA, safety research is still needed to support long-term operation of existing LWRs in Europe. Regarding the issues in severe accidents, the SRA refers to the work carried out in the framework of the SARNET Network (Albiol et al., 2010; Van Dorsselaere et al., 2011; Van Dorsselaere et al., 2012) to conclude to a common view on the ranking of the research priorities in the field. The research priorities on severe accident management (Schwinges et al., 2010; Klein-Heßling et al., 2012) were prepared by the SARNET SARP group.

High priority issues (further research is considered as necessary):

- 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.

Medium priority issues (should be investigated further as already planned in the different research programs):

- 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).

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Low priority issues (could be closed after the related activities are finished):

- Corium coolability in core catcher with external cooling;
- Corium release following vessel rupture;
- Crack formation and leakages in concrete containment;
- Aerosol behavior impact on source term (in steam generator tubes (SGT) and containment cracks);
- Core reflooding impact on source term.

Three issues could be closed because of low risk significance and sufficient current state of knowledge:

- Integrity of reactor coolant system and heat distribution;
- Ex-vessel core catcher and corium-ceramics interaction, cooling with water bottom injection;
- FCI including steam explosion in weakened vessel.

The phenomena described above are extremely complex; they generally demand the development of specific research. This research involves very substantial human and financial resources and, in general, the research field is too wide to allow investigation of all phenomena by any national program. To optimize the use of the resources, the collaboration between nuclear utilities, industry groups, research centers and safety authorities, at both national and international levels is very important. Therefore the severe accident research activities at KIT are strongly linked to other European facilities and experimental programs such as CODEX, VULCANO and PLINIUS. The experimental programs are part of the LACOMECO (Miassoedov et al., 2012) and SARNET2 projects of the 7th EU Framework Programme, providing experimental resources by offering the severe accident research platform at KIT for transnational access.

The overall purpose of the research is 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 high and medium priority issues identified by the SARP group of SARNET. These helps to understand the core degradation and coolability, in-vessel and ex-vessel core melt behavior and hydrogen-related phenomena in real reactors in two ways – firstly by scaling-up and secondly by providing data for the improvement and validation of computer codes applied for safety assessment and planning of accident mitigation concepts, such as e.g. ASTEC (Bandini et al., 2010).

2. SEVERE ACCIDENT RESEARCH FACILITIES AT KIT

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

- QUENCH facility is the only operating experimental facility in Europe for investigations of the early and late phases of core degradation in prototypic geometry for different reactor designs and different cladding alloys, incl. analysis of the relocation of cladding and fuel and the formation and cooling of in-core debris beds to gain information on the characteristics of the created particles.
- LIVE facility concentrates on the investigation of the whole evolution of the in-vessel late phase of a severe accident, including e.g. formation and growth of the in-core melt pool, characteristics of corium arrival in the lower head, and molten pool behavior after the debris re-melting in large scale 3D geometry with emphasis on the transient behavior.
- DISCO is the only operating facility available worldwide for integral DCH investigations. It is designed to perform scaled experiments that simulate melt ejection from the RPV to the reactor cavity after the RPV failure under low system pressure during severe accidents in LWRs. These experiments investigate the fluid-dynamic, thermal and chemical processes during melt ejection out of a breach in the lower head of an LWR pressure vessel at pressures below 2 MPa.
- MOCKA facility is a large-scale test facility to study molten corium concrete interaction in stratified geometry with melt mass up to 3200 kg.
- HYKA experimental facilities are among the largest available in the world. In combination with the high static and dynamic pressures the experimental facilities are designed for, a unique experimental centre especially for combustion experiments in confined spaces is available with HYKA. Due to the different orientations and sizes the set of large and strong experimental vessels offers a flexible basis for scientific experimental work on reactive hydrogen mixtures.

Although the LIVE, DISCO and MOCKA facilities can only perform experiments with simulant materials, the tests can be considered as prototypic for the following reasons: the selected materials represent, to the greatest extent possible, the important physical properties of the real core materials, and the large scale used allows extrapolation to the reactor case. Moreover, there is considerable flexibility and variability in the facilities due to the rather simple handling techniques. Pre-tests, parallel separate-effects tests and post tests can be performed in one hand. These tests can be seen as complementary to tests with UO_2 in other research centers.

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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. 1). The facility can be operated in two modes: a forced-convection mode (typical for most QUENCH experiments) and a boil-off mode. QUENCH-16 was conducted in forced-convection mode, in which, superheated steam from the steam generator and superheater together with argon as a carrier gas for off-gas measurements enter the test bundle at the bottom. The system pressure in the test section is around 0.2 MPa absolute. The test section has separate inlets at the bottom to inject water for reflood (bottom quenching) and synthetic air (80% N_2 and 20% O_2) during air ingress phase. The argon, the steam not consumed, and the hydrogen produced in the zirconium-steam reaction flow from the bundle outlet at the top through a water-cooled off-gas pipe to the condenser where the steam is separated from the non-condensable gases. The water cooling circuits for bundle head and off-gas pipe are temperature-controlled to guarantee that the steam/gas temperature is high enough so that condensation at the test section outlet and inside the off-gas pipe is avoided.

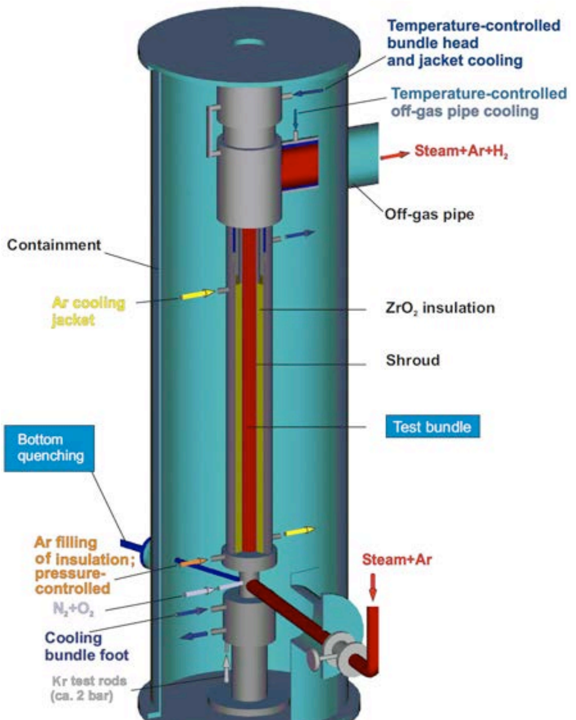

Figure 1. QUENCH containment and test section.

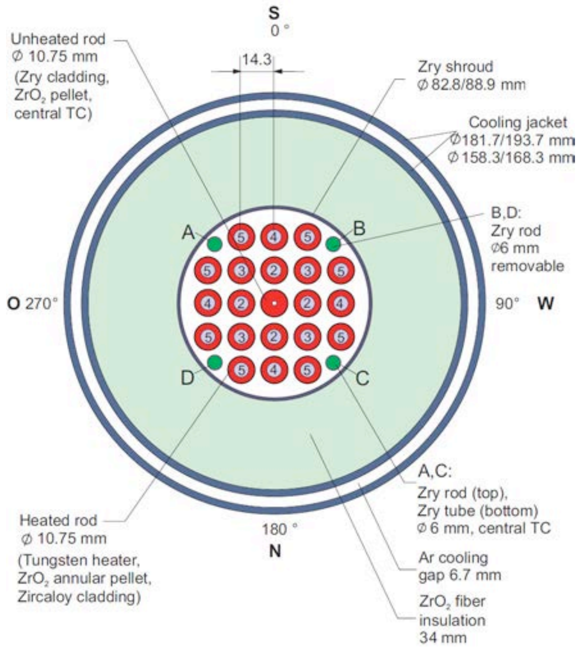

Figure 2. QUENCH bundle cross section.

The test bundle is approximately 2.5 m long and is made up of 21 fuel rod simulators (Fig. 2). The fuel rod simulators are held in position by five grid spacers, four are made of Zircaloy-4 and the one at the bottom of Inconel 718. 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 (lower edge of heaters corresponds to bundle elevation 0 mm). Electrodes of molybdenum (length 300 + 576 mm; Ø 8.6 mm) and copper (length 390 + 190 mm; Ø 8.6 mm) are connected to the tungsten heaters at one end and to the cable leading to the DC electrical power supply at the other end. The measured electrical power includes the heat dissipation at cables. The heating power inside the bundle is distributed between two groups of heated rods. The distribution of the electric power within the two groups is as follows: about 40 % of the power is released into the inner rod circuit consisting of eight fuel rod simulators (in parallel connection) and 60 % in the outer rod circuit (12 fuel rod simulators in parallel connection). The tungsten heaters are surrounded by annular ZrO_2 pellets. 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. The

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rods were connected to a controlled feeding system that compensated minor gas losses and allowed observation of the first cladding failure as well as a failure progression.

There are four corner rods installed in the bundle. Two of them, i.e. rods "A" and "C", are made of a Zircaloy-4 solid rod at the top and a Zircaloy-4 tube at the bottom and are used for thermocouple instrumentation. The other two rods, i.e. rods "B" and "D" (solid Zircaloy-4 rods of 6 mm diameter) are particularly designed to be withdrawn from the bundle to check the amount of ZrO_2 oxidation and hydrogen uptake at specific times. Rod B was pulled out of the bundle before the air ingress phase of the experiment and rod D was pulled before quenching; low part of rod A was removed after test termination.

The test bundle is surrounded by a 3.05 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 ~2.66 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 are: 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 (Stuckert et al., 2009) and materials during oxidation and reflood. One test was performed with the complete sequence including boil-off phase, pre-oxidation and reflood. The QUENCH experiments will focus in the future on the analysis of the relocation of cladding and fuel and the formation and cooling of in-core debris beds to gain 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

A number of studies have already been performed to pursue the understanding of a severe accident with core melting, its course, major critical phases and timing and the influence of these processes on the accident progression (Kymäläinen et al., 1997; Theofanous and Angelini, 2001). Uncertainties in modeling these phenomena and in the application to reactor scale still persist (Van Dorsselaere et al., 2006). These include e.g. formation and growth of the in-core melt pool, relocation of molten material after the failure of the surrounding crust, characteristics of corium arrival in residual water in the lower head, corium stratifications in the lower head after the debris re-melting. These phenomena have a strong impact on a potential termination of a severe accident.

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 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 heated melt in a 3D geometry of a lower plenum of a RPV (Gaus-Liu et al., 2010). Other test 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 (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 the typical pressurized water reactor, as shown in Fig. 3 and Fig. 4. The diameter of the test vessel is 1 meter. The top area of the test vessel is

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covered with an insulated lid. The test vessel is enclosed in a cooling vessel to simulate the external cooling. The cooling water inlet is located at the bottom and the outlet is positioned at the top of the cooling vessel.

The melt is prepared in an external heating furnace designed to generate 220 l of the simulant melt. Therefore it is possible to produce the total amount of the scaled oxide melt mass and additionally the total amount of the scaled metallic part of the simulated corium melt. The maximum temperature that can be reached in the heating furnace is limited to 1100 °C. In addition, the heating furnace is equipped with a vacuum pump; so it is possible to extract the residual melt out of the test vessel back into the heating furnace.

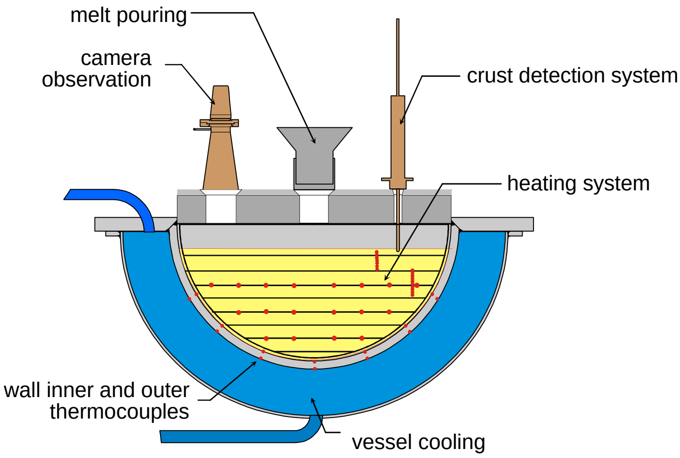

Figure 3. Scheme of the LIVE-3D test vessel.

Figure 4. View of the LIVE-3D vessel with heating coils.

The melt can be poured from the heating furnace into the test vessel either centrally or near the vessel wall. A vacuum pump extracted the residual melt at the end of test back to the heating furnace uncovering the solid crust formed in the experiment.

The volumetric decay heat is simulated by means of 6 heating planes in the test vessel. Each heating plane consists of a spirally formed heating element with a distance of ~40 mm between each winding. The heating elements are shrouded electrical resistance wires and are located in a special cage to ensure the correct positioning. All heating planes together provide a maximum power of about 28 kW. To realize a homogeneous heating of the melt, each plane can be controlled separately. In the case of homogeneous heat distribution the maximum heating power is limited to ~18 kW. The maximum temperature the heating system can provide is limited to 1100 °C.

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 surface and outer surface are measured at 5 latitudes and 4 locations at each latitude. Heat flux distribution through the vessel wall can be calculated based on these temperatures. Additionally, 80 thermocouples are positioned within the vessel to measure the temperature distribution in the melt pool and in the crust.

To get detailed information about the crust formation three thermocouple trees are installed at the vessel wall at different heights. The length of the trees is different to account for different crust thickness and is designed so that part of the tree is located in the liquid phase. A precise crust detection lance can detect the crust front and measure the crust/melt boundary temperature as well as the melt pool vertical temperature profile. Two video cameras and one infrared camera are used to visualize the convection of the melt pool.

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 are that the simulant melt should be a non-eutectic mixture of several components with a distinctive solidus-liquidus area of about 100 K, and that the simulant melt should have a similar solidification and crust formation behavior as the oxidic corium.

Binary melts of KNO_3–$NaNO_3$ are 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 220 °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. 5) the melt pool in a lower plenum of a RPV is modeled by a slice with 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

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mm thick stainless steel plate, can be 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 are made of 1 mm thick stainless steel and are insulated. The LIVE-2D test vessel can be covered at the top by an insulated or a cooled lid.

The melt pool is an 80 mole% KNO_3–20 mole% $NaNO_3$ salt mixture which was also used in the LIVE-3D and SIMECO tests. Due to this the tests in LIVE-2D are especially well comparable with the tests in LIVE-3D and the tests in SIMECO. The volumetric heating system is simulated by 9 planes of heating elements. The maximum design temperature of the heating system is 1100 °C. Each heating plane is formed to several loops with a distance of ~40 mm between each loop. The distance between each heating plane is about 50 mm. To realize a quasi-homogeneous heating of the melt, each plane is controlled separately. The melt can be heated with a power of up to 13 kW.

A set of 13 thermocouples is installed in the melt pool. They deliver information on temperature profiles in the melt during different phases of the experiments. The vessel wall is equipped with 6 thermocouple trees located at the inner side, each consisting of 5 thermocouples in a distance of 5, 10, 15, 20 and 25 mm from the wall. This allows determining a crust profile at the vessel wall. At 10 different positions, thermocouples at the inner and outer side of the vessel wall are installed to determine the heat flux through the vessel wall.

The results of the LIVE-2D experiments regarding the upward heat transfer were compared with the earlier tests performed in the BALI and SIMECO facilities as well as with correlations developed from those tests. For low Rayleigh numbers the results of the LIVE-2D tests demonstrated higher heat transfer to the top of the melt, as it is shown in Fig. 6. 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.

Figure 5. LIVE-2D test vessel.

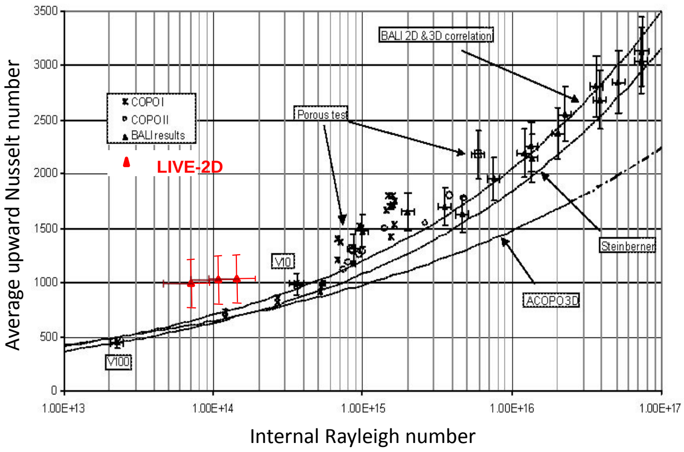

Figure 6. Comparison of the upward heat transfer in LIVE-2D tests with earlier experiments.

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. Supporting post-test analysis contributes to characterization of solidification processes of binary non-eutectic melts. Complementary to other international programs with real corium melts (like METCOR-P, PRECOS, and INVECOR projects of ISTC), 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

In case of a core meltdown accident in a PWR, liquid corium containing metals and oxides may relocate into the lower head of the reactor vessel. If the lower head fails in this condition, with an in-vessel pressure higher than the containment pressure, the molten corium will be forcefully ejected into the reactor pit, finely fragmented and eventually transported outside the reactor pit. The efficient heat transfer from the melt particles to the gas, together with combustion of hydrogen previously released into the reactor building and produced during melt dispersion, will heat-up and pressurize the containment atmosphere. These processes, referred to as direct containment heating, may endanger the integrity of the containment.

The safety philosophy applied to all plants worldwide is to prevent high pressure core melt situations by depressurization of the primary system below a level which would be effective in limiting debris dispersal mainly to the

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reactor pit or adjacent small reactor rooms. This level, sometimes called the DCH cut-off pressure, depends on reactor cavity design and lies between 1.0 and 2.5 MPa, but can be as low as 0.5 MPa for certain plant geometries. If the occurrence of DCH is strictly related with the primary circuit pressure at vessel failure, its consequences depend also on the breach characteristics, on the amount and characteristics of the molten mass and on the layout of the reactor pit and reactor building. Therefore, the evaluation of DCH consequences must be plant dependent. The consequences of DCH are essentially related to the reactor cavity geometry, therefore an experimental database has been established for the plant types EPR, German Konvoi, French PWRs and some of VVERs (Meyer at al., 2009). For other plant specific geometries the experiments must still be conducted. The recent studies indicated that the need to investigate DCH in boiling water reactors may arise. Moreover, other still unresolved issues are the influence of water in the reactor cavity and fission products release during DCH and its impact on the overall source term, which has not sufficiently been investigated before. The DISCO experimental facility is the only one operating worldwide that can address these critical issues.

The DISCO experiments 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 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 pressure and temperature, and the amount of hydrogen produced and burned.

The main components of the facility (Fig. 7) 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 (RPV) are modeled by a vertical pipe. A disk holding 8 pipes (46 mm I.D., 255 mm length) separates the two partial volumes. This arrangement models the main cooling lines with respect to the flow constriction between RCS and RPV. The RPV model, mounted at the lower end of the pipe, serves as crucible for the generation of melt by a thermite reaction between iron oxide and aluminum. The total volume of the RCS/RPV vessel is 0.08 m³.The breach in the lower head is modeled by a graphite annulus at the bottom, which is closed with a brass plate. This plate melts when the thermite reaction reaches the bottom. 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 modelled according to the reactor design being investigated (Fig. 8).

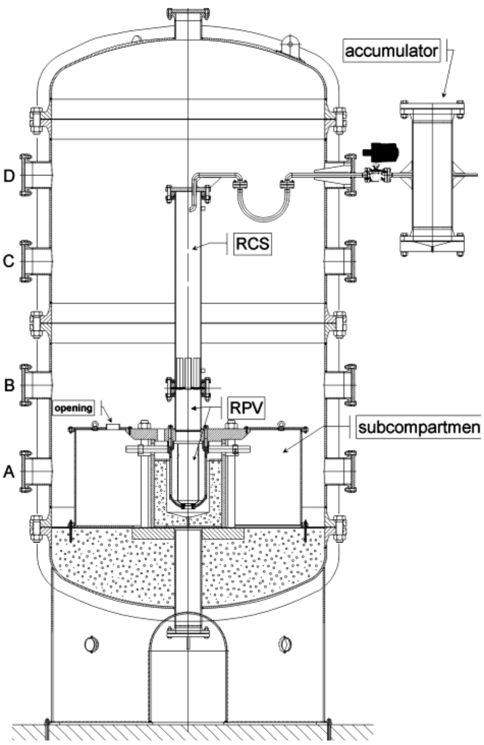

Figure 7. Scheme of the DISCO facility with the model of the EPR cavity.

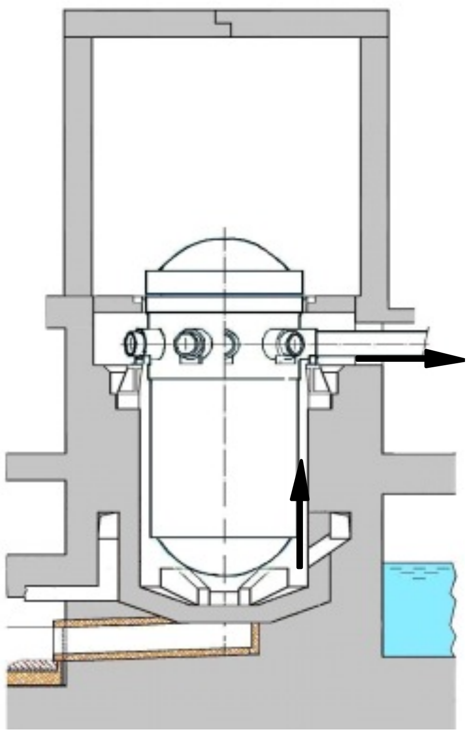

Figure 8. Configuration of the EPR plant.

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

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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. Two fans are running until the blowdown is started, to ensure a well-mixed atmosphere. A gas sample is taken just before the start of the experiment.

The accumulator is pressurized with steam to around twice the planned initial blow down pressure. The model of the RPV contains the aluminum-iron oxide thermite. The experiment is started by igniting the thermite electro-chemically at the upper surface of the compacted thermite powder. When the pressure raise in the RPV-RCS vessel verifies that the thermite reaction has started, the valve in the line connected to the steam accumulator is opened and steam enters the RCS vessel, which is preheated to the saturation temperature of the planned burst pressure. The pressure balance in both vessels is reached quickly and the valve is automatically closed again. 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. 10 second after blowdown the fans are started again and five minutes thereafter post-test gas samples are taken.

Standard test results are: pressure and temperature history in the RPV, the cavity, the reactor compartments and the containment vessel (Fig.9 and Fig. 10), 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.

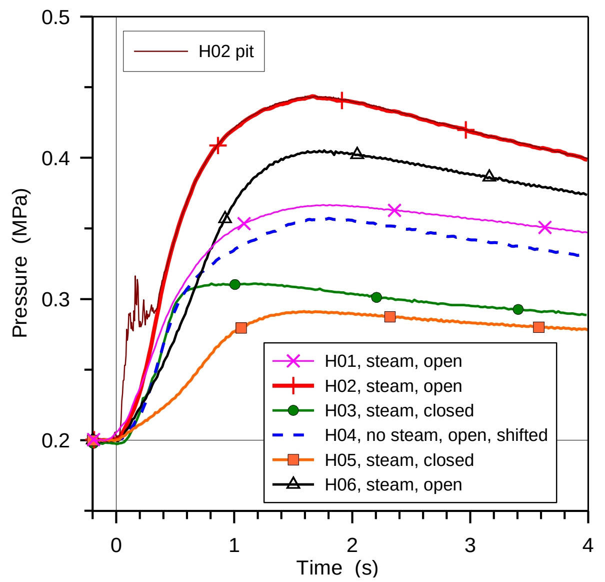

Figure 9. Containment pressures in EPR tests.

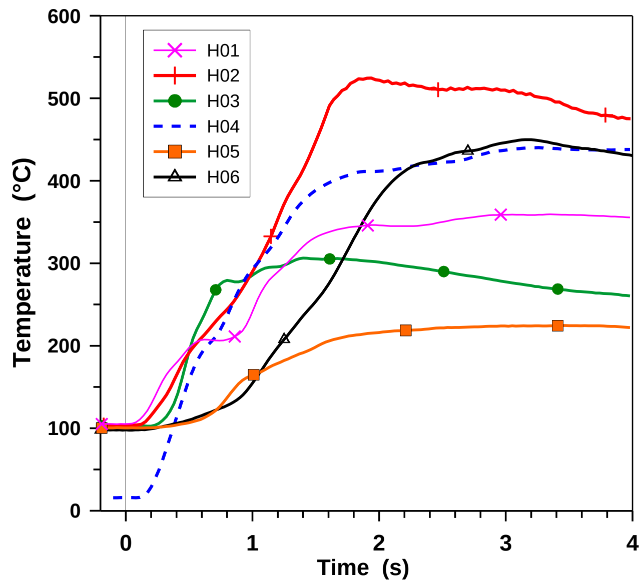

Figure 10. Containment temperatures in EPR tests.

Break wires are placed across the RPV exit hole. The break wires are intended to give timing information on entry of debris into the cavity. The total debris mass dispersed into the DISCO vessel and the debris mass in specific locations are determined by a post-test debris recovery procedure. Loose particles are collected by a vacuum cleaner and weighed separately for each location. The total weight of crusts is determined by pre- and post-test weighing of all parts of the test facility, or if not possible, by removing the crusts from the parts and weighing of them.

A post-test sieve analysis of the debris recovered from different locations is performed for each test. A standard set of 17 sieves is used (10 mm to 40 µm). Except for small pieces, crusts are generally not included in the sieve analysis. The information on the droplet size of the melt that has formed a crust is not available. This has to be taken into account when the results of the sieve analysis are evaluated.

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.

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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 need further investigations. An important issue 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.

Molten corium concrete interaction has been studied since many years, based on small scale laboratory tests with model materials on one hand, and on large-scale experiments with high temperature melts and concrete (Alsmeyer, 1987) on the other. The experimental results obtained in these tests were used to improve the models for heat transfer and material behavior in the computer codes that were developed to predict the ex-vessel accident situation in LWRs. Nevertheless, substantial uncertainties still persist, which are mainly related to the distribution of the heat flux from the melt to the concrete in the lateral and axial directions, respectively, which is also strongly influenced by the onset of partial melt solidification and formation, growth and failure of crusts at the concrete interface. As a consequence, substantial deviations exist in the code predictions of the axial basement penetration time, or in strong lateral erosion, which can undermine containment structures. To overcome this problem, experiments on 2-d concrete erosion by oxide melts are being carried out at ANL, sponsored by the OECD consortium (Farmer et al., 2004) and in VULCANO facility at CEA (Journeau et al., 2009). To complement these data for stratified melts, KIT carries out tests in the MOCKA facility to study the interaction of simulant aluminum/iron oxide thermite melts with concrete (Foit et al., 2012).

The MOCKA facility 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. Layering of steel below the less dense oxide melt is typical for the long-term ex-vessel concrete erosion, as the addition of light oxides which are produced by concrete ablation decreases the density of the oxide layer considerably, so that a layer flip between oxide and metal does occur. 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 can also be simulated by adding thermite briquettes from the top.
- End of the experiment is defined by maximal concrete erosion.

The experiments investigate the two-dimensional concrete erosion in a cylindrical crucible. To allow sufficient time and material for the erosion process, the cavity of the crucible is fabricated as a massive concrete structure. Usually the composition of the concrete is in accordance with reactor specifications: SiO_2: 70.3 w-%, $Ca(OH)_2$: 13.55 w-% , Al_2O_3: 6.58 w-%, $CaCO_3$: 5.46 w-%, free H_2O 4.11 w-%. However other types of concrete can be used in the MOCKA facility: siliceous, siliceous/limestone, limestone, serpentine ($Mg_3Si_2O_7 \cdot 2H_2O$ in East European plants).

The composition and physical properties of the simulant melt are very important because of the required correspondence to the corium melt. As simulant melt a thermite mixture of aluminum with iron oxide and the admixture of further oxide additives is used. The oxide fraction of the produced melt is composed of a mixture of alumina, calcia, and iron oxide. The composition is selected to have a wide freezing range which should be comparable to a corium melt, however at lower temperatures. The wide freezing range is important not only for the growth and stability of the crust, but also for the viscosity of the melt, which substantially increases; when by onset of crystallization in the agitated melt slurry of crystals and liquid may be generated.

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 more than 1800 – 1900 °C and the melt has sufficient overheat over the liquidus temperature to show the important effects also in transient tests. To extend the duration of the interaction with the concrete and 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. 9. 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. 10). 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.

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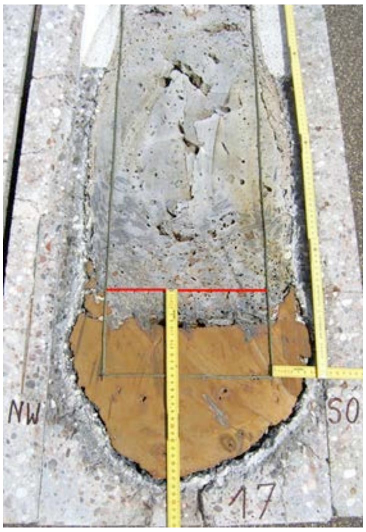

Figure 9. Cross-section of the MOCKA 1.7 crucible.

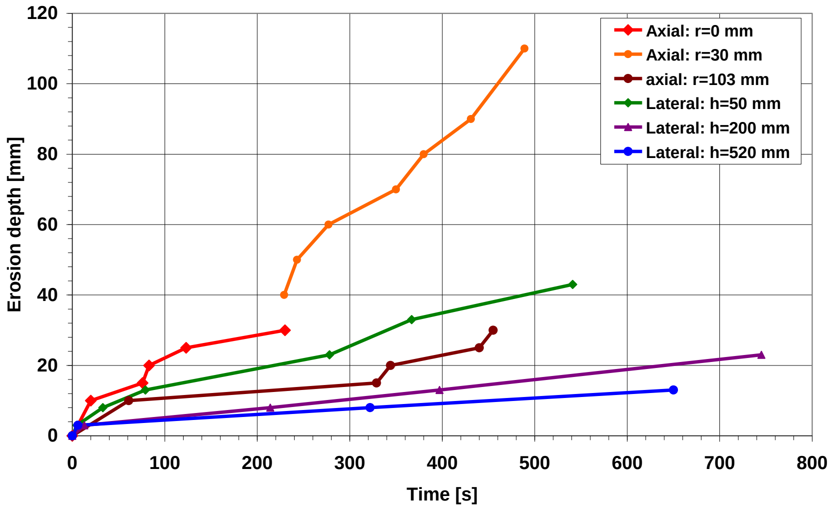

Figure 10. Position of the melt front as a function of time in the MOCKA 1.7 test.

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 containment during severe accidents: hydrogen distribution, hydrogen combustion and hydrogen mitigation measures (Jordan and Breitung, 2007). These phenomena are ranked as high priority issue by the SARP group of SARNET.

HYKA facilities offer 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 near-by (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 (slow and fast deflagration and detonation). The main technical details of the different HYKA facilities 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 A1, two large vents (0.8m) H_2 distribution in closed rooms, integrity of mechanical structures

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A8	cylindrical vessel	diam. 1.8 length 3.0 (horizontal)	9	100	fast deflagrations and detonations at high initial pressures
FTC	rectangular flow test chamber	8.5x5.5x3.3 airflow ≤ 24.000 m^3/h	160	1.07 static 1.7 dynamic	studies on vented combustions (up to 16 g H_2), testing of H_2 local detonations in closed spaces

An important outcome of the research activities in the DCH domain within SARNET was the understanding, that the combustion of hydrogen produced by oxidation during melt ejection from the RPV as well as the hydrogen initially present in the containment can be the dominant phenomenon for containment pressurization (Meyer and Kotchourko, 2007; Meyer et al., 2008). It is now clear that the uncertainty in the combustion rate under these conditions was too large for the assessment of containment integrity for certain reactors. Dedicated combustion codes are presently not capable to reproduce the results obtained in a first series of experiments with hydrogen release conducted in the DISCO facility at KIT. Moreover, the need for hydrogen combustion tests at a scale larger than 1:18 was stressed by the SARNET partners. Without those, the uncertainty in the extrapolation of experiments to reactor scale would still remain too large to assess the containment integrity for certain reactor geometries. This issue will be addressed in the experiments performed in the A2 vessel of the HYKA facility.

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.

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