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The SAFEST project: towards pan-European Lab on Corium Behavior in Severe Accidents. Main Objectives and R&D priorities.

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Abstract

No individual country has sufficient resources to address all severe accident important phenomena within the framework of a national research programme, therefore optimised use of resources and the collaboration at European and international level are very important. Integrating European severe accident research facilities into a pan-European laboratory for severe accident and corium studies and providing resources to other European partners for better understanding of possible accident scenarios and phenomena is necessary in order to improve safety of existing and, in the long-term, of future reactors.

SAFEST (Severe Accident Facilities for European Safety Targets) is a European project networking the European corium experimental laboratories with the objective to establish coordination activities, enabling the development of a common vision and of research roadmaps for the next years, and of the management structure to achieve these goals. One of the main objectives is to address and resolve the variety of the remaining severe accident issues related to accident analysis and corium behaviour. The project is a valuable asset for the fulfilment of the severe accident R&D programmes that are being set up after the Fukushima Daiichi accidents and the subsequent European stress tests, addressing both national and European objectives.

Roadmaps on European severe accident experimental research for water reactors and for GenIV technologies will be drafted. Joint R&D is conducted to improve the excellence of the SAFEST facilities: this includes measurement of corium physical properties, improvement of instrumentation, consensus on scaling law rationales and cross comparison of material analyses.

Joint experimental research is a clear objective in the SAFEST project to provide solutions for stabilisation of severe accident and termination of consequences for the current Gen II and III plants. Consequently, the knowledge obtained in SAFEST shall lead to improved severe accident management measures, which are essential for reactor safety. In addition, it will offer competitive advantages for the nuclear industry and contribute to the long-term sustainability of nuclear energy.

Keywords: LWR, severe accidents, corium, experimental facilities

1. Introduction

After the TMI-2 and Chernobyl-4 accidents R&D programmes have been launched worldwide to study severe accidents and to propose means to mitigate their consequences for the populations and to the environment. In the European Union, these national research efforts have been shared in EC-funded projects starting at least in the 4th Euratom Framework Programme. The SARNET2 network [1] has been co-ordinating research on these issues and is currently pursuing this task as a dedicated Technical Area on severe accidents under Nuclear Generation II and III Association (NUGENIA) [2].

The efficiency of cooling reactor core structures and materials during severe accidents either in the core region, in the vessel lower head or in the reactor cavity is a key issue to limit the progression of the accident. This can be achieved either by ensuring corium retention within the reactor pressure vessel (RPV) or at least by limiting the corium progression and, consequently, the rate of corium release into the reactor cavity in case of RPV failure. These issues are covered within the scope of accident management for existing reactors and within the scope of design and safety evaluation of future reactors. The specific objectives addressed creation of a database on coolability of degraded core, debris formation, debris coolability and corium behaviour in the lower head, development and validation of models and computer codes for simulation of in-vessel debris bed and melt pool behaviour, performance of reactor scale analysis for in-vessel corium coolability and assessment of the influence of severe accident management measures on in-vessel coolability.

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Currently available data on degraded core reflooding are summarised in the reflow map developed at KIT, which is being constantly updated to include findings from previous and currently running programs on the coolability of the reactor core, taking into account available coolant mass flow rates and the core damage state. Substantial knowledge and understanding of the phenomena governing the coolability of intact rod-like reactor core geometry was obtained in numerous national and international projects. However, major gaps in knowledge still exist in the areas of debris and molten pool behaviour.

There is a significant progress on simulation of in-vessel core coolability, with 2D and 3D simulation codes, focusing on the enhanced coolability of debris beds by lateral and/or bottom water inflow [3], [4]. These phenomena are being investigated in a number of experimental programs including study of multi-dimensional effects on debris bed coolability, e.g. PRELUDE/PEARL at IRSN [5] and DEFOR at KTH [6]. Many experiments are currently being performed in this domain in several international projects, e.g. in the frame of SARNET within NUGENIA. They are accompanied by numerous efforts on modelling in simulation codes and benchmarking activities among these codes (e.g. OECD/NEA BSAF 1 and 2 benchmarks on Fukushima-Daiichi accidents).

For the ex-vessel situation, significant progress has been made in R&D activities started after TMI-2 and Chernobyl accidents. Many experiments are currently performed in this domain in international projects, either in SARNET frame or in the PLINIUS experimental platform at CEA. They are accompanied by large efforts on modelling in simulation codes and benchmarking activities among these codes. Other R&D efforts are based on ANL (USA) CCI experiments on molten corium-concrete interaction. Concerning Fuel Coolant Interaction (FCI), the OECD/NEA project SERENA2 has provided a large amount of experimental data on steam explosion, including advanced visualization of premixing (the explosion initial conditions).

As corium molten pool delivers a significant part of the decay heat to the RPV lower head, there is little chance to arrest melt progression without cooling. It is nevertheless necessary to study concrete ablation and molten pool configuration in the cavity, as means to verify that there is no risk of early basement melt-through, and to provide the initial conditions for corium cooling phase or transfer to a core catcher. At present, international R&D is mostly performed in the frame of the NUGENIA TA2/SARNET network, mainly on the basis of the VULCANO (CEA) [7], MOCKA (KIT) [8], HECLA (UJV) [9] and SICOPS (AREVA NP GmbH) experiments [10]. OECD/NEA/CSNI state-of-the-report on molten corium-concrete interaction (MCCI) is also in progress with a significant contribution by the SARNET network. Current 2D oxidic pool MCCI experimental programmes have shown that limestone-rich concretes are almost isotropically ablated while for silica-rich concretes lateral ablation was much larger than vertical ablation. This reproducible behaviour is not yet understood so application of the obtained results to reactor scale or to other concrete compositions (e.g. basaltic) should be done with caution [11]. Furthermore, recent experiments with oxidic and metallic pools have shown phase repartitions which are different from simple-layers assumptions considered in MCCI codes (emulsion or gravity stratification, effect of steel reinforcement).

Globally, an important database exists on the phenomena occurring in non-mitigated ex-vessel situations (steam explosion, direct containment heating, corium spreading and molten core concrete interaction, debris bed formation and coolability) and for the support of commercially available core catchers. Nevertheless, there are some remaining uncertainties on application of the results obtained in experiments to reactor conditions.

Moreover, since most of previous severe accident research topics were focused on PWR design and majority of severe accident codes and models were developed for PWRs, only some special experimental series were focused on BWR-type reactors. It was in agreement with the fact that most of the operating light water reactors (LWR) are of PWR type. However, large number of BWR reactors is operated worldwide and in Europe as well.

The Fukushima Daiichi accident demonstrated that some BWR specific design features can play a very important role in severe accident progression. For example, the penetrations for control rods through the lower head most probably affected release and behaviour of molten core in the reactor cavity; however, this phenomenon has to be confirmed after defueling of Fukushima Daiichi plants.

Additionally, the stress tests of nuclear power plants in the European Union pointed out that some BWR special features may have an impact on severe accidents, e.g. the flooding of lower drywell may stabilise ex-vessel molten corium progression. The planned or implemented safety improvements included also some BWR-specific topics. For example, in Germany the development of accident management measures was proposed to protect the building structure surrounding the spent fuel pool in BWRs, which is located outside the containment, against hydrogen combustions or to prevent them. In Finland, the installation of additional emergency diesel

generators and of diverse and independent ways of pumping water to the RPV via fire-fighting diesel driven pumps is foreseen in BWR units. Other specific features of BWR design which require further detailed analysis include:

- larger vessel dimensions and mass of water, deeper cavity,
- channels and bypasses in the core, separated by the canisters of fuel assemblies which can influence melt relocation and oxidation of Zircaloy,
- control rod and instrumentation guide tubes in the vessel lower head which can influence formation of in-vessel debris beds, RPV failure mode and timing, possible limitation of flow-rate of releasing melt, mass and composition of the releasing corium,
- nitrogen inertisation of primary containment which may have a significant impact on early core degradation.

All these facts emphasise that the severe accident research for BWR needs further efforts aiming at improvement and optimization of BWR-specific models and validation of them against future experiments, which will be one of the main objectives of the SAFEST project.

SAFEST experimental facilities are unique in providing the possibility to perform experiments in specific fields of research on corium behaviour in severe accidents in main types of light water reactors including BWRs. There are less than ten currently operating prototypic corium facilities in the world and three of them are located in Europe and belong to the SAFEST consortium. The facilities are operated by a team of experts who have long and recognised experience in the nuclear safety research. The SAFEST project will contribute significantly to establishment of an integrated pan-European laboratory for severe accident research able to address and successfully resolve the variety of the remaining issues related to the corium behaviour in severe accidents. The way towards the establishment of a pan-European laboratory will clearly depend on conclusions of the roadmap activities planned in the project. This unique consortium will be a valuable asset for the fulfilment of the severe accident R&D programmes that are being set up after Fukushima Daiichi accidents and the subsequent European stress tests, addressing both national and European objectives, in collaboration with major international stakeholders.

In particular, the following topics, all belonging to high-priority issues identified by the SARP group of SARNET [12] and confirmed in the NUGENIA R&D roadmap, are addressed in the SAFEST project, contributing considerably towards understanding and perhaps even closure of these issues:

- Formation and cooling of debris beds and molten corium pools in the reactor core in order to demonstrate effective cooling modes and rates, and coolability limits. The specific R&D objectives are to create and enhance the database on debris formation, debris coolability and corium pool behaviour in the lower head, to develop and validate the models and computer codes for simulation of in-vessel debris bed and melt pool behaviour.
- Influence of control rod and instrumentation guide on debris bed formation and cooling and corium pool coolability for BWRs.
- Though the behaviour of pure oxidic pool in the vessel lower head is quite well understood, the efforts will be focused on scenarios with a large pool of molten corium: heat flux to metal layer in a layered molten pool configuration, including possible 3-layer configuration (observed in OECD/NEA MASCA experiments [13]). An important point is the thickness of the metallic layer. This is of high priority because it can influence strongly the strategy for in-vessel retention concepts in some of the existing (e.g. VVER-440) and new plant designs.
- Corium behaviour inside BWR-type lower heads including corium relocation into a deep lower head filled with water, significant thermo-mechanical loads on the vessel structures during debris re-heating and re-melting, and high fraction of Zr in the core.
- Database on critical heat flux and external cooling conditions in order to evaluate and design SAM measures for external vessel cooling. Influence of BWR lower head penetrations on melt cooling and its influence on the external convection.
- Location of RPV failure for BWRs, failure timing and modes.
- New phenomena related to the in-vessel retention by cavity flooding (pressurized molten corium jets directly entering water, corium concrete interaction starting underwater) that were not considered previously in experimental and analytical R&D programmes.
- Interaction of corium jets with water in the cavity pool (deep pools for BWRs) after a failed in-vessel retention, focusing on jet fragmentation to provide reliable initial conditions for the steam explosion phase and to determine the characteristics of the formed debris bed (particle size distribution, geometrical repartition, chemical state) that are necessary inputs for debris bed coolability issues.

- MCCI for concrete compositions that have not been yet studied earlier (e.g. basaltic concrete mainly used in Japan and in the USA) focusing on 2D convective heat transfer distribution.
- Two-dimensional concrete ablation caused by oxidic melt as well as by stratified oxide-metal melt in presence of steel reinforcing bars, focusing on metal oxidation during MCCI.
- Long-term MCCI (i.e. longer than 1 day after start of interaction) characterised by high concrete fraction and reduced heat fluxes to the concrete interface.
- Criteria and scaling approaches for extrapolation of the results obtained in small to medium scale experiments to reactor case.

To resolve these issues the partners of the SAFEST project perform experiments, develop and validate numerical models aiming at adequate description of the investigated phenomena and upgrade the SAFEST facilities to be able to address the future research needs. As an important part of the project, the SAFEST experimental platforms and facilities are offered for access to external user groups.

2. Project activities

Coordinated by KIT, the SAFEST consortium is small and consists of five research centres (CEA, MTA EK, JRC-ITU, SCK-CEN and UJV), two universities (KIT and KTH), and one industrial partner (AREVA) and forms a balanced mix of expertise in state-of-the-art severe accident management and experimental research. Complementarity and synergy between partners is maximized by identifying overlapping roles in this project and exploiting them by choosing the partner best suited to the identified tasks. The involvement of the industry is beneficial in providing a balanced approach between research and industrial needs. The description of some of the SAFEST activities are given below.

2.1. Development of research roadmaps

The general objective is to develop a common vision and experimental research roadmaps for the next years, and the management structure to achieve these objectives. Safety research roadmap for Gen IV plant safety is also under development taking advantage of the knowledge and expertise obtained for existing reactors as well as on specific safety characteristics of considered Gen IV designs.

The draft of the European corium experimental research roadmap was completed in 2016 with the help of many contributors from the various SAFEST partners [14]. It is based on the research priorities determined by SARNET SARP group as well as those from the NUGENIA Technical Area 2 on severe accidents. It also takes into account issues identified in the analysis of the European stress tests and from the interpretation of the Fukushima-Daiichi accident. The roadmap takes advantage of the current and developing European corium infrastructures and, if necessary, recommends its adaptation.

Following the Fukushima-Daiichi accident, JAEA experts have contacted European counterparts in order to promote joint R&D activities as well as to solicit scientific and technical support from EU to Japan. As a consequence, a series of meetings between the SAFEST consortium and Japanese organizations were held aiming at assessing long-term goals and proposing experimental support needed for proper understanding and interpretation of Fukushima-Daiichi accident, analysis of collected debris in European and Japanese laboratories and even eventual processing of molten cores.

2.2. Experiments in the SAFEST distributed research infrastructure

Being one of the most important activities of the project, the experimental facilities of the SAFEST partners are integrated into a pan-European laboratory for severe accident and corium studies and are offered to external organizations who are interested in performing the tests. The activities of the SAFEST distributed research infrastructure are divided into three groups, each one addressing a specific topic:

- In-vessel corium and debris behaviour;
- Ex-vessel corium and debris behaviour;
- Corium properties.

After publishing the rules of access to the SAFEST facilities in October 2014, two calls for proposals were announced attracting the interested users to specify the experimental requirements and conditions. Following each call, the user selection panel with the help of independent international experts evaluated the proposals and selected a short-list of user groups. Together with facility operators the user groups prepare, perform, analyse and document the experiments. Total of 16 experiments have been selected to be performed in the SAFEST test facilities until the end of the project. Up to now four experiments, MISTEE-AL, MISTEE-OX, DEFOR and

SES, have already been successfully performed. A short summary of objectives of the planned experiments and results obtained up to know is given below.

2.2.1 QUENCH experiment at KIT

The main objective of the test will be study of high temperature degradation of BWR assembly mock-up. The bundle containing the fuel rod simulators and the BWR absorber blade will be heated up to the onset of degradation and flooded with water from the bottom. Data on clad oxidation and hydrogen generation and release, material degradation and relocation (including melting and relocation of control blade materials) will be measured during the test and obtained in the post-test material analysis. These data will be used for severe accident model development and code validation. The bundle cross-section is presented in Figure 1 a). It is based on ¼ of the NPP-specific assembly SVEA-96 OPTIMA2 and includes 21 heated fuel rods, 3 pressurised unheated rods, absorber blade, fuel and water channel boxes. The test procedure (Figure 1b)) will include pre-heat-up, pre-oxidation, transient heat-up and final water quenching phases. The test is scheduled for the second half of 2017.

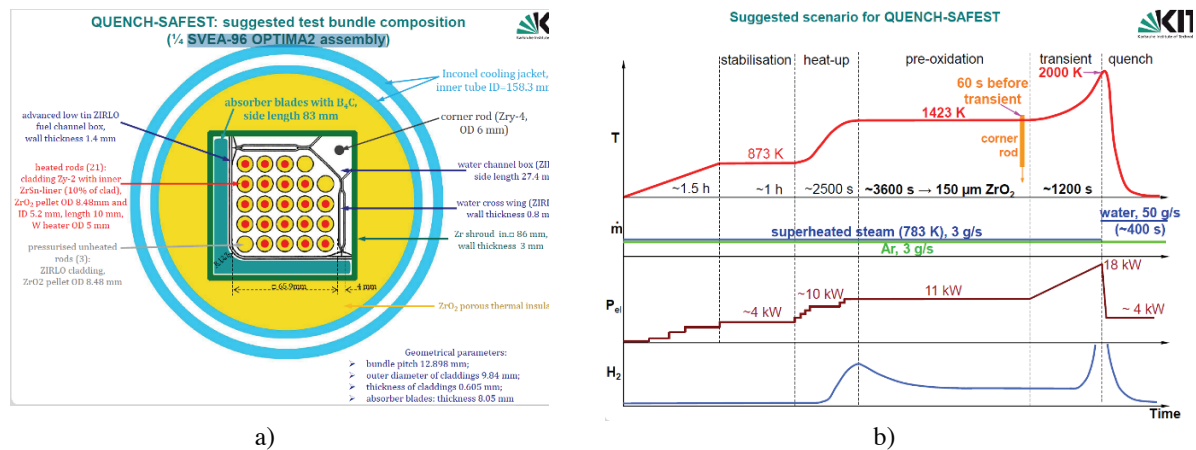


Figure 1 a) Section of QUENCH/BWR bundle and b) Test procedure

2.2.2 LIVE experiment at KIT

The purpose of this test is to assess the characteristic time of convection development in a homogeneous melt pool in the 1:5 scaled 3D RPV lower head after a rapid change in boundary conditions. There are indications that this transient phase may need to be accounted for in the integral codes through some adjustment of the heat flux correlations; this may increase the safety margins of in-vessel melt retention, especially for high power reactors. Meanwhile at the Xi'an Jiatong University in China the COPRA facility has been commissioned and successfully put into operation [4]. The COPRA facility is a 2D 1:1 scaled slice and utilizes the same material to simulate corium melt and same heating system as in the LIVE facility but at a larger scale. The two facilities allow complementary studies of the molten pool behavior under similar conditions for a large range of Ra numbers from 1012 to 1016. A test similar to LIVE is planned also in COPRA in the frame of the ALISA project. Comparison of the results of these two tests will help gaining insight on the influence of geometry (2D/3D), scale (1:1 scale/1:5 scale) and simulant material (molten salt/water) on experimental results, thus improving the extrapolation to prototypic conditions.

2.2.3 DISCO experiment at KIT

The objective of the test will be to analyse the phenomena which occur during an ex-vessel fuel coolant interaction (FCI). The test is focused on the premixing phase of the FCI with no trigger used for explosion phase. The objectives of the test will be to provide data concerning the dispersion of water and melt out of the pit, characterization of the debris and pressurization of the reactor compartments. In previous DISCO tests with water in the reactor pit [5] only small water pools were investigated. The planned experiment will be performed similarly to previous ones but with a larger water pool providing insights on the scaling issues of FCI. Moreover, the pit will have a transparent wall to visualize the premixing phase with high-speed video recording.

2.2.4 MOCKA experiments at KIT

The main objective of the two tests will be study of molten core-concrete interaction (MCCI) in 2D cylindrical crucibles made of concrete typical for Swedish BWRs. To obtain the information on scaling effects, the concrete composition will be close, if not identical, to the composition of SICOPS-BWR experiment at AREVA in Erlangen, Germany, performed at a smaller scale, but with a prototypic corium (section 2.2.16). In MOCKA experiments the corium is simulated by the thermite melt and the volumetric heating of the melt is achieved by addition of thermite and zirconium into the oxidic part of the melt pool. In one of the tests the influence of the steel reinforcement in the concrete (12 wt.%) on the erosion front progression will be studied.

2.2.5 VULCANO experiment at CEA

The test will focus on the influence of crusts on the melt-concrete interface during MCCI. Previous 2D MCCI experiments have shown that silica-rich and limestone-rich concretes exhibit different ablation behaviour. The characteristics of the crust formed at first contact between the melt and the concrete has been considered as a possible explanation of this difference. In the VULCANO experiment the test crucible will be made of a siliceous concrete with a sufficiently thick inner liner made of limestone common sand (LCS) concrete to ensure the global crust failure which is assumed to occur at the beginning of MCCI with LCS concrete. Moreover, the admixture of LCS concrete in the corium melt will additionally decrease the corium solidus temperature that may prevent further formation of crusts on the interface.

2.2.6 VITI experiments at CEA

VITI facility will be used to obtain thermophysical properties (density, surface tension and, if feasible, viscosity) of (U, Zr, Fe) metallic alloys representative of reactor compositions (section 2.2.14). For the first time, it will be possible to have a better information on thermophysical properties of prototypic metallic alloys. The measurements will be carried out in inert atmosphere and around the liquidus point.

2.2.7 MISTEE-AL experiments at KTH

In the MISTEE-AL experiments the material effects related to steam explosion were studied. The objectives of the test series are to check the ability of alumina melt to undergo thermal fragmentation, as a mechanism for spontaneous steam explosions observed in the KROTOS tests at CEA with this material, and to study the void produced by the high temperature drops in a water pool. The main conditions of the tests included different water subcooling and melt superheat:

- High subcooling, between 79 K and 84 K;
- Medium subcooling, between 53 K and 55 K;
- Low subcooling, between 13 K and 26 K;
- Low melt superheat, around 67 K; and
- High melt superheat, at 210 K.

Triggered steam explosions have been observed in the experiments with high and medium water subcooling of the water (Figure 2) and only one spontaneous steam explosion happened at very high melt superheat and high water subcooling (Figure 3).

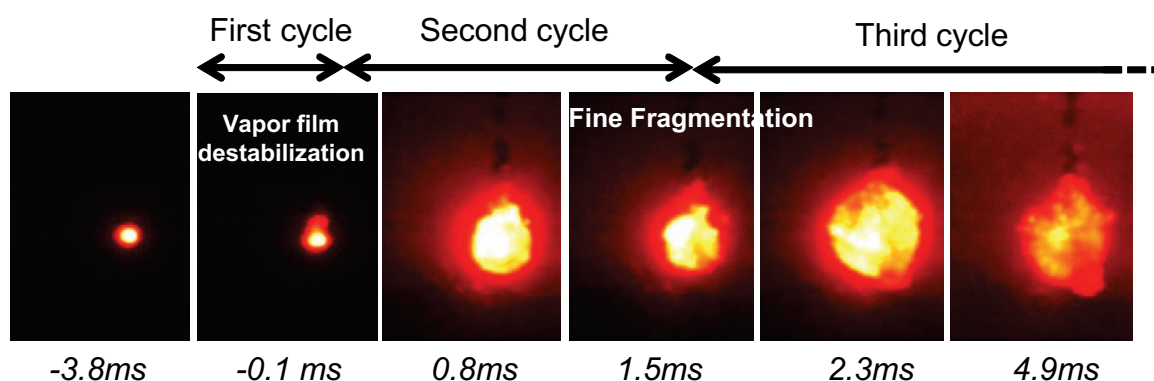


Figure 2 Typical images of triggered steam explosion ($\Delta T_{\text{superheat}} \approx 167$ K, $\Delta T_{\text{subcool}} \approx 82$ K)

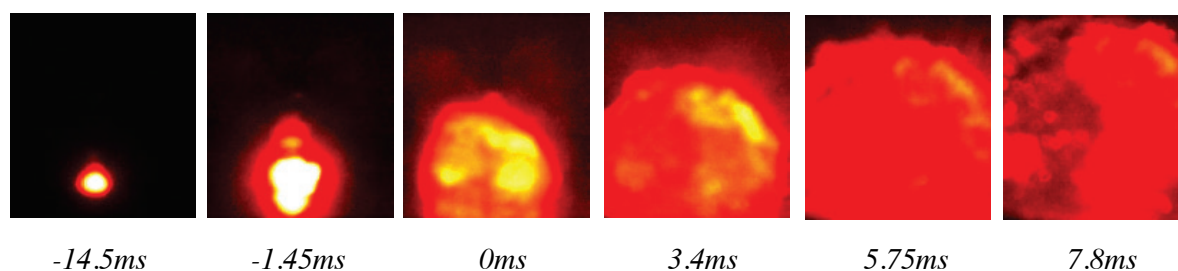


Figure 3 Spontaneous steam explosion observed for the case W5 (high subcooling ≈ 85 °C and high superheat ≈ 215 °C)

The following conclusions are supported by the experimental results:

- The fragmentation characteristics observed for the alumina melt are similar to those observed for materials with a lower melting temperature, such as tin or $\text{WO}_3\text{-CaO}$. Two to three cycles of bubble expansion and collapse appear after a steam explosion which is triggered by a small pressure wave. Due to the higher melt temperature, bigger equivalent bubble diameter and conversion ratio are observed with alumina compared with the other materials previously studied in the MISTEE facility.
- The water subcooling and the melt superheat both play an important role in thermal fragmentation as they directly influence the void production and the melt viscosity. Spontaneous steam explosion observed at high subcooling and superheat is sensibly stronger than the triggered explosions in other experiments.
- A preliminary steam explosion map of alumina single drop in respect to water subcooling and melt superheat has been developed. However, more data are needed in order to provide a complete picture.

2.2.8 MISTEE-OX experiments at KTH

In the MISTEE-OX experiments the oxidation behaviour of Zirconium during FCI is being studied. The main objective of these experiments is to investigate the ability of pure zirconium melt to undergo thermal fragmentation while falling into water and thereby undergoing oxidation. The measurements include high-speed visualization of the Zirconium melt droplets and water interaction, estimation of hydrogen produced during the interaction and estimation of oxygen content in the debris by performing SEM/EDX analysis.

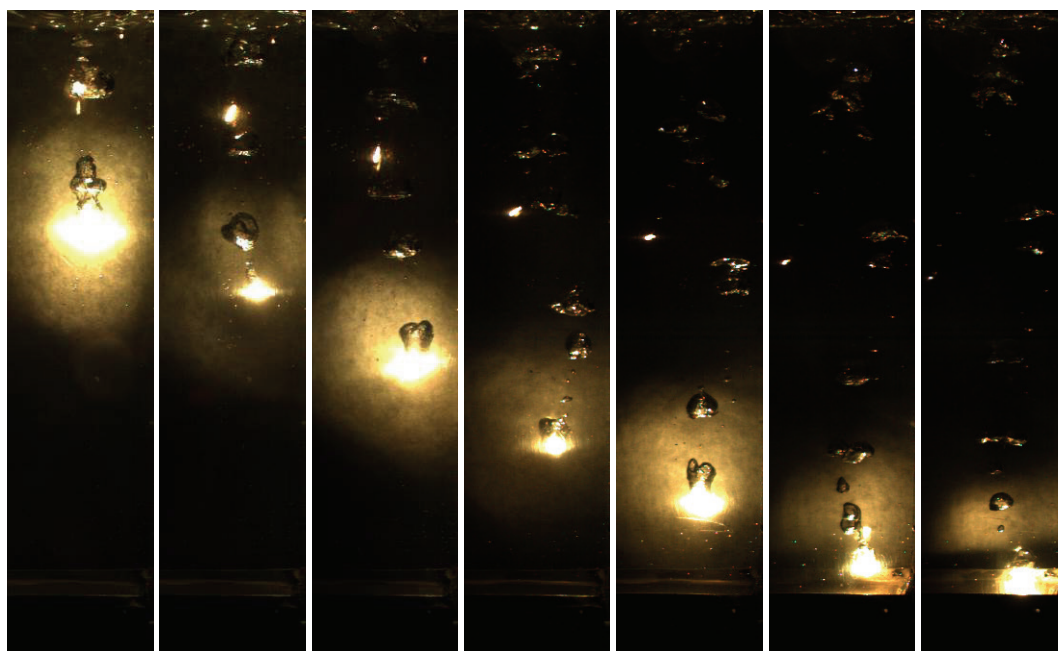


Figure 4 Snapshots of the Zirconium melt and water interaction

The molten droplets of pure Zr metal are discharged from the bottom of the furnace, falling and quenched in a deep water pool. The droplets are allowed to settle in a catcher arranged at a depth of 470 mm inside the water pool. Snapshots of the transient interactions are shown in Figure 4. For the tests performed with pure

zirconium melt at a moderate melt superheat ($\Delta T_{\text{superheat}} \sim 150$ K) under high water subcooling ($\Delta T_{\text{superheat}} \sim 85$ K) no spontaneous steam explosions have been observed. Here the hydrogen volume produced during the interaction is obtained by quantifying the bubbles departing the surface of the water pool, originally detached from the droplets. Based on the preliminary analysis, ~ 0.14 litres of hydrogen production is estimated. An oxidation ratio of up to 22% is estimated based on the total volume estimation of the detached bubbles (which is assumed to be composed of pure hydrogen). The microstructure of the debris revealed formation of an oxygen rich outer layer to be of almost constant thickness along the surface and the oxidation ratio decreases further towards the core, as it is shown in Figure 5.

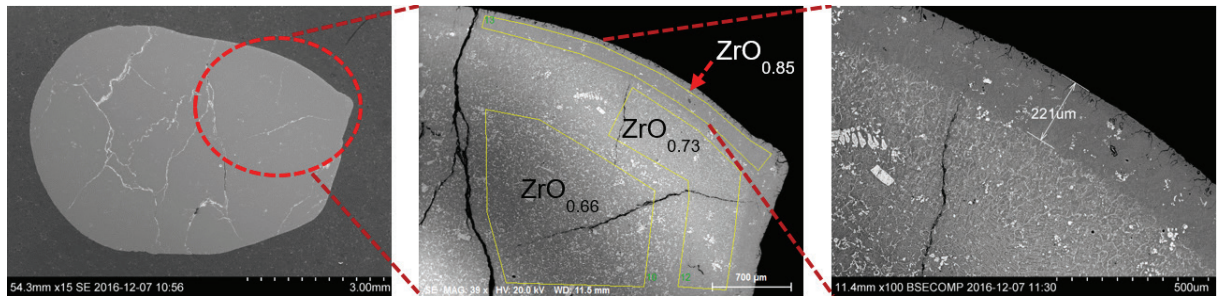


Figure 5 Micro-structure of the debris particle showing stratification of oxidation in layers

2.2.9 DEFOR experiment at KTH

The objective of this test was to investigate the influence of control rod and instrumentation guide tubes on melt fragmentation in water and debris bed formation. As a main difference to previous experiments performed in the DEFOR test facility, a rod connecting the discharge nozzle and the bottom of the test section was installed to simulate the control rod guide tube. The main test parameters were similar to the DEFOR-A4 test which was characterised by a relatively low melt temperature of 1221 K (melt superheat of 78 K). Several cameras were implemented to visualize the melt jet formation and jet impingement to provide visual imaging of transient melt-coolant mixing and debris formation and fragmentation in water. After tilting the furnace, the melt entered the water pool and the fragmentation started (Figure 6). The particles were collected on the debris catchers and the temperature evolution was measured by the thermocouples installed in the catchers while they were filled by the hot particles and cooled by the water.

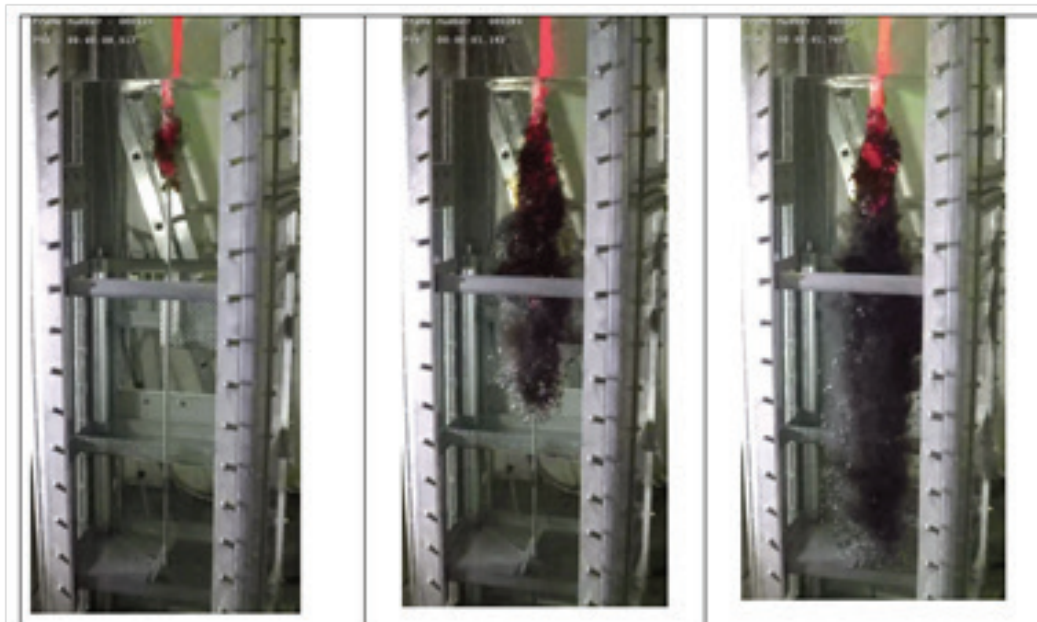


Figure 6 Melt jet fragmentation in water at the beginning, at 0.675 s and at 1.243 s

The cumulative particle size distribution of completely fragmented debris collected in the DEFOR/SAFEST-A22 test in catchers 2-4 is shown in

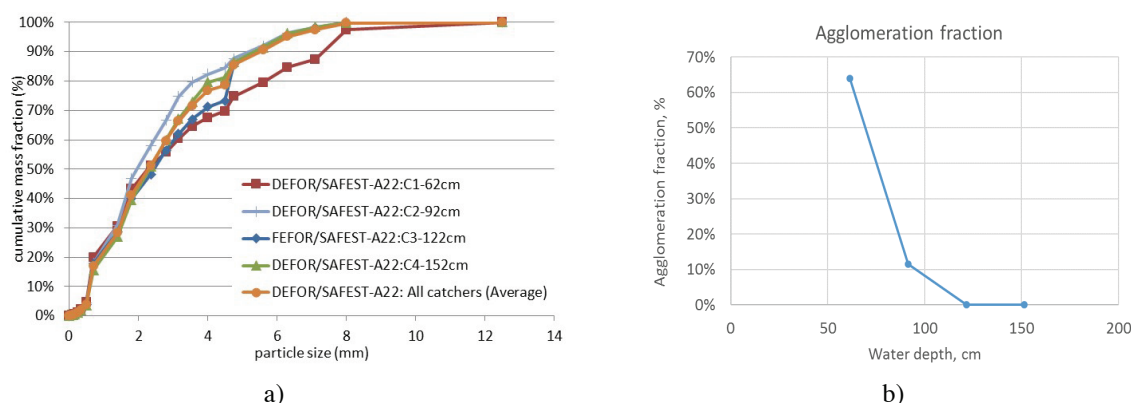


Figure 7 a). The distribution was obtained by sieving the debris on a set of sieves. The dependency of the mass fraction of agglomerated debris on the pool depth is presented in Figure 7 b). Comparison of data on fraction of agglomerated debris from the DEFOR/SAFEST-A22 test also agrees well with the previously obtained results in the DEFOR-A4 and A8 experiments. It is important to note that obtained dependencies are quite consistent with each other and despite the fact that there are some stochastic variations of the mass fraction of debris collected in each catcher in different tests.

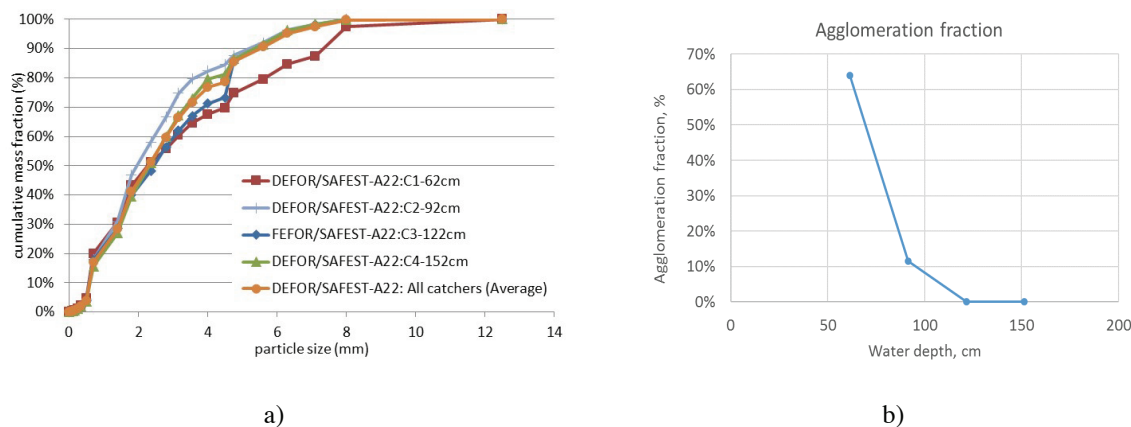


Figure 7 a) Cumulative particle size distribution of fragmented debris and b) Fraction of agglomerated debris as a function of water pool depth in the DEFOR/SAFEST-A22 test

2.2.10 SES experiment at KTH

The objective of the SES test was to measure the energetics of spontaneous steam explosion and investigate the characteristics of the premixing zone in stratified melt under water configuration. 5 litres (39 kg) of molten $\text{Bi}_2\text{O}_3\text{-WO}_3$ eutectic mixture at 132 °C superheat were delivered through an elongated nozzle with outlet located under water above the impact plate. Such configuration for melt delivery was expected to prevent melt jet fragmentation in water and exclude its possible contribution to the spontaneous steam explosion or triggering. 0.6 sec after the onset of melt release from the nozzle a spontaneous steam explosion occurred. Snapshots of the video recording of melt water interaction are provided in Figure 8.

17.8 tons of dynamic load was measured by the dynamic force sensors and resulted in 421 N·s of explosion impulse. Due to early occurrence of the steam explosion and release of aerosols from the nozzle, the observation of the premixing layer was not possible. The steam explosion was followed by the upward displacement of the test section. The impact of the test section and the nozzle caused the splashing of melt from the funnel. Post-test analysis of the debris demonstrates that about 15.5 kg of melt remained in the funnel, 15.6 kg were found on the containment floor and 2.7 kg remained in the test section in the form of a cake and particles.

The early occurrence of the steam explosion compared to previous SES test without the elongated nozzle suggests that triggering of the steam explosion in stratified configuration cannot be attributed to the existence of the jet. This also corresponds to the findings obtained in the previous experiments in the DEFOR facility, where melt was released into deep water pools and in which steam explosion has never occurred.

2.2.11 CERES experiment at MTA EK

In the CERES facility the investigation of the effect of the elevated sump level on the in-vessel melt retention for VVER-440 reactors during the melt retention by flooding of the reactor cavity is planned. One of the main differences to the cooling by the in-vessel water injection is the maximum sump level change during LOCA conditions. Since the varying sump level changes the conditions slightly taking into account the small gap area between the RPV and the cavity wall, it is important to examine its effect on the coolability. The experiment could reveal the cooling conditions with the maximum possible sump level in the containment.

2.2.12 CODEX test at MTA EK

The CODEX test will address the influence of nitrogen on oxidation and degradation of fuel rod claddings in the temperature range of 800-1200 °C. The reference scenario for this test will be a reactor accident with melt through of the vessel lower head and penetration of air-steam mixture from the reactor cavity into the reactor vessel. The experiments will simulate the high temperature interactions of peripheral fuel assemblies in air-steam atmosphere. Slow cool-down will be applied at the end of the test to provide information on bundle state before quench.

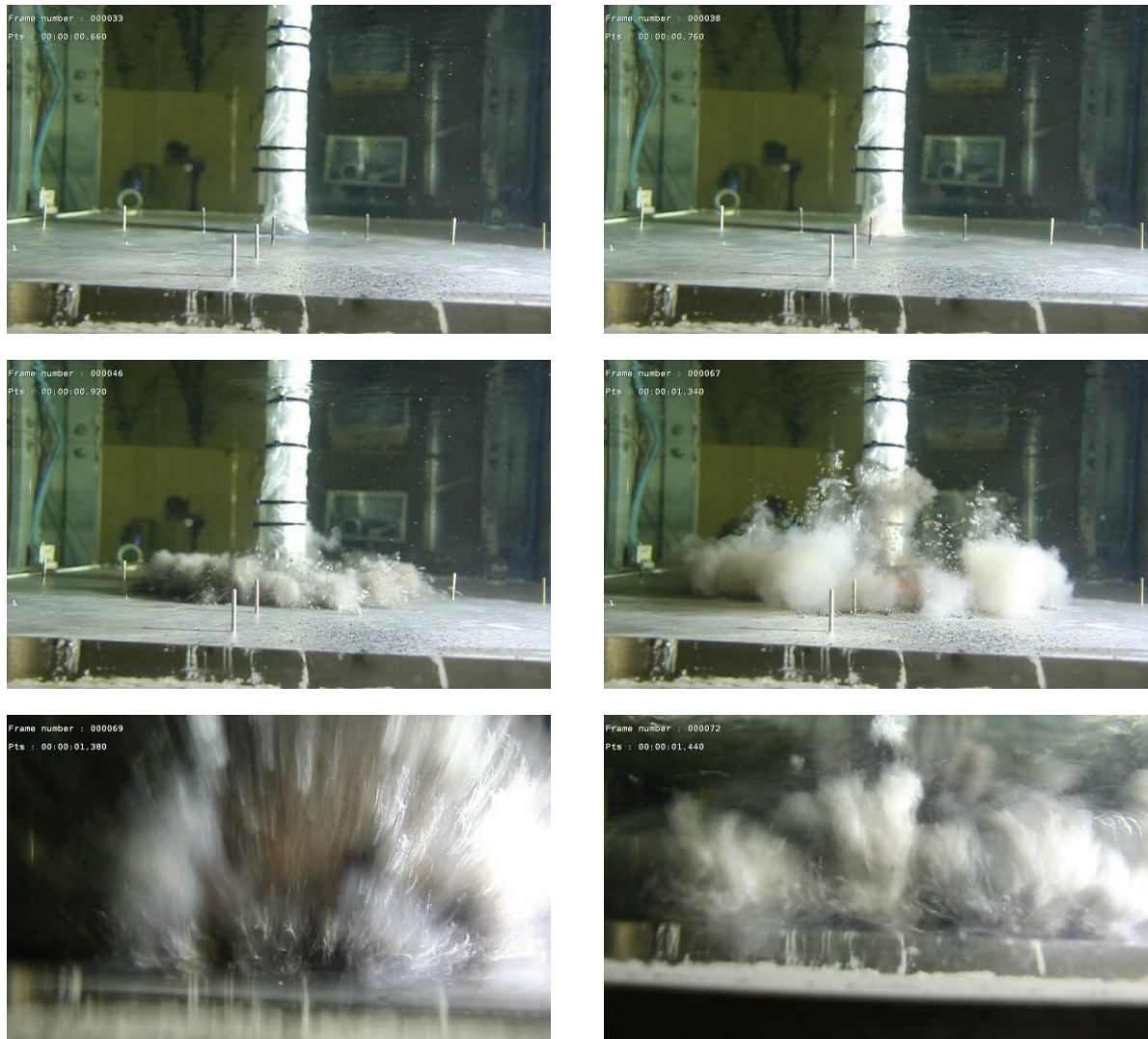


Figure 8 Snapshots of the melt water interaction in SES-S1 test

2.2.13 FLF experiments on U-Zr-O phase diagram at JRC

The objective of these tests is receiving experimental data and improving the understanding of the quasi-binary section of ternary U-Zr-O diagram in the composition domain of BWR-specific corium composition, i.e. at U/Zr atomic ratio close to 0.9 and Zr oxidation index close to 20 - 30%. The method of laser heating combined with fast pyrometry and FLF facility have been intensively used in the JRC Material Research for several decades to study melting and crystallization behaviour of many oxide systems representing different

reactor fuels and to produce phase diagram data, in particular on liquids and solidus temperatures. A novel “Reflected Light Signal” (RLS) method has been applied to diagnose solid/liquid phase transitions from surface reflectivity effects using suited low-power (mW) probe laser. Experiments with volatile substances were carried out in an autoclave under medium or high pressure (a few bars up to a few thousands bars) of an inert gas in order to suppress as much as possible evaporation but also to study systematically the behaviour of phase transition points at different pressures. FLF facility has been also used to study corium phase diagrams, particularly in the ternary U-Zr-O system. Most compositions studied so far were in the system domains representing PWR corium with a specific U/Zr atomic ratio close to 1.2. All the samples foreseen for this project have been prepared by powder metallurgy within the end of 2016 (Figure 9a)). Details of the spectral emissivity analysis are shown in Figure 9b). The solidus point of a 47.2 % UO₂, 10.6 % ZrO₂ and 42.2 % Zr sample was determined to be 2400 K ± 25 K. Similar results will be provided for the other compositions studied in this work. The spectral emissivity at the wavelength of 650 nm of this sample has also been determined (open circles in Figure 9b)), based on the grey body hypothesis. Such hypothesis has been checked to constitute a reasonable approximation for the present compounds. The emissivity at 650 nm changes from a metallic behaviour (emissivity lower than 0.5) to an oxydic ($\epsilon > 0.6$) one upon melting. Emissivity values recorded at low temperature are affected by a large uncertainty due to a high background noise.

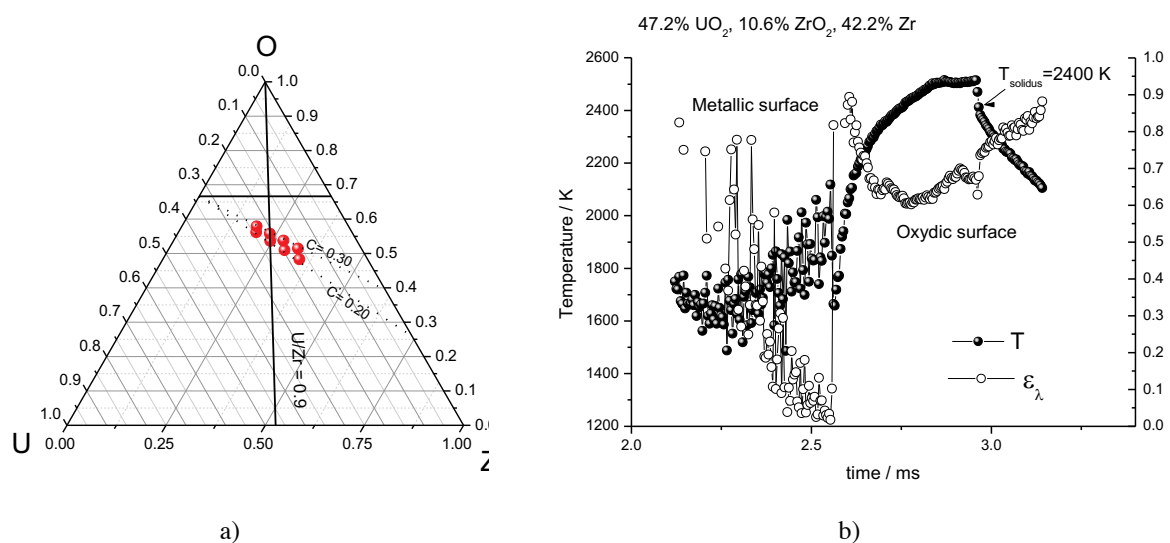


Figure 9 a) Compositions studied in this work plotted in the U-Zr-O ternary diagram, b) Details of the spectral emissivity analysis

2.2.14 FLF experiments on (U, Zr, Fe) emissivity at JRC

The tests aim at improving the knowledge on the emissivity of the upper metallic layer in oxidizing atmosphere. This parameter influences directly the focusing effect phenomena that is very important for in-vessel melt retention strategy. The tests should be performed on pure RPV steel and steel+Zr mixtures, allowing comparisons with the experimental data obtained under inert atmosphere, and a first assessment of the impact of an oxidizing atmosphere. The samples will be manufactured for three domains of the U-Zr-Fe alloy composition: Fe-rich, U-rich and Zr-rich. The results from these tests will be used to reduce the discrepancies in the evaluation of the radiative heat transfer currently observed between various severe accident codes.

2.2.15 KUBI experiments at UJV

The objective of the KUBI test was to study a short-term interaction between UO₂ melt and B₄C pellets with an application to the safety assessment of Sodium-cooled Fast Reactors (SFRs). The experiment has been conducted at UJV in the COMETA facility. The 2 kg of depleted UO₂ were melted in the cold crucible and five B₄C pellets (11 g) were added into the melt from the top. Significant gas release has been observed as interaction started. In the post-test analysis a large porosity, attributed to gaseous releases (mainly carbon monoxide according to equilibrium thermodynamic pre-calculations), has been observed.

2.2.16 SICOPS experiments at AREVA

The SICOPS experiments will address the MCCI for prototypic corium and concrete compositions typical for Nordic BWRs. The test objectives are to understand (a) how significant can be the influence of Swedish

concrete and Nordic BWR specific corium composition on MCCI in the context of possible applications of existing models, which are verified against data of experiments with different melt and concrete compositions, (b) if anisotropy of concrete ablation in radial and axial directions can be expected and (c) if improvements of MCCI models for BWR application are needed. The pre-test series have been completed and the main experiment will be carried out in 2017.

2.3. Guidelines for selecting simulant materials and scaling approaches

Some of the SAFEST experiments are conducted in scaled-down test facilities, and in the majority of cases with simulant materials. For experiments, a complete similitude between test facility and full scale plant conditions cannot be achieved. Consequently, it is necessary to: (1) give rationales to select simulant materials for the experiments; (2) assure that the important processes of interest are well scaled; and (3) assess the effects of distortions (when present) on processes and/or parameters of interest to a nuclear power plant accident scenario.

The draft of recommendations for selecting simulant materials and scaling approaches was prepared by KIT and contains two major parts: (1) Severe accident scaling methodology (SASM) and (2) Application of the scaling approach to SAFEST experimental facilities. As a guideline to describe the scaling of SAFEST experimental facilities an outline of the scaling methodology is given, which was developed by a Technical Program Group (TPG) that was established by the USNRC in 1989. It consists of 11 steps which should be followed in formulating the methodology. They are divided in 3 elements, which are (1) experimental requirements, (2) evaluation and specification for experiments and testing and (3) data base acquisition and documentation. As such these 3 elements and 11 steps are like a roadmap for the development of the SASM and provide a guide to follow for each of the experimental facilities in the SAFEST Project, in order to derive results which could be applied to the plant scale and plant conditions. The 11 steps are described. They may involve some exploratory experiments and testing before the envisaged experiments are performed. The steps also involve identification of similarity criteria and development of models and closure relations and even separate effect tests to improve understanding of key processes (or phenomena).

The document is currently under development. The description of the scaling methodology for the DISCO facility at KIT provides an example of what needs to be done for other experimental facilities in the SAFEST project and for the other experiments that need to be performed on the various facilities. This document, when completed, will be of great benefit to the EU research programs on corium behaviour in severe accidents.

3. Conclusions

Direct outcome from the SAFEST project will be progress towards creation of an integrated pan-European laboratory for study of corium behaviour in severe accidents. Indeed, it encompasses a very large spectrum of nuclear reactors severe accident phenomenology dealing with corium (mainly oriented at LWRs, even though several aspects of Gen IV severe accidents can be studied in some of the SAFEST facilities). By strengthening the links between European corium facility operators, preparing a common roadmap for future EU research and improving the capabilities and performance of experimental facilities, this laboratory is a valuable asset for the fulfilment of severe accident R&D programs which are being set up after Fukushima-Daiichi and the subsequent stress tests both at the national level and at the European level.

Due to the links that are established with other European projects or platforms (e.g. CESAM, IVMR, NUGENIA/SARNET, etc.) the SAFEST project offers a unique opportunity for all parties to get involved in the networks and activities supporting safety of existing and advanced reactors and to get access to large-scale experimental facilities in Europe to enhance understanding of reactor core behaviour under severe accident conditions.

The main result of SAFEST activities will be a better understanding of physical background of severe accidents and prototypic corium behaviour. It will benefit the EU utilities and safety organizations, which will be able to validate (either directly through the access to the SAFEST distributed infrastructure or indirectly through R&D) the hypotheses for severe accident scenarios and propose pertinent procedures for accident mitigation taking into account experimental results. The experimental results will be used for the development and validation of models and their implementation in the severe accident codes such as ASTEC, MELCOR, ATHLET-CD. This helps to capitalize the knowledge obtained in the field of severe accident research in the severe accident codes and scientific databases, thus preserving and diffusing this knowledge to a large number of current and future end-users in Europe.

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