Overview of Containment Issues and Major Experimental Activities

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SUMMARY

In the CONTAINMENT area of SARNET three issues of severe accident phenomenology are treated which endanger the containment integrity by pressure increase due to fast heat release or transfer: hydrogen combustion, explosive melt water interaction, and direct containment heating. They occur late in severe accidents as consequences of severe core damage. So, their probability of occurrence is extremely small but they involve the risk of a failure of the containment, the final safety enclosure. The pressure load can be quasi-static, i.e. several seconds during direct containment heating (DCH), or highly dynamic as in hydrogen detonation or fuel-coolant interaction (FCI). The related phenomena identified by the EURSAFE project [1] as both important for safety and still lacking sufficient knowledge are investigated experimentally and analytically in a integrated effort by seventeen partner organizations. The paper presents some of the main experimental facilities and results. Concerning the analysis a large number of CFD codes and lumped parameter codes are developed, verified and used, which can only be introduced in brief.

A. INTRODUCTION

The main safety related issues addressed in the CONTAINMENT area are: investigation of Hydrogen Behavior in the Containment (HBC), and investigation of Fast Interactions in the Containment (FIC). They are dealt with in two work packages. Each topic is again subdivided into two subtopics. The general objective of our work is to resolve outstanding issues important for the safety of the containment as identified in the EURSAFE project [1], and to provide ASTEC with appropriate physical modeling. Seventeen partners were involved in the first year of the SARNET Containment topic.

B. WORK PROGRAMME

In **WP12-1** the hydrogen combustion and associated risk mitigation is studied, concentrating on potential combustion modes; slow deflagrations, fast accelerating flames, deflagration to detonation transition (DDT) and detonation. The studies include the reaction kinetics inside catalytic recombiners.

In **WP12-2** the hydrogen distribution in the different parts of the containment is studied, with the objective of assessing the formation of combustible gas mixtures, taking into account the effect of mitigation systems such as sprays or recombiners.

In **WP13-1** the fuel coolant interaction is studied with the objective to increase the knowledge of parameters affecting steam explosion energetics during corium relocation into water, and to develop the tools for determining the risk of vessel or containment failure. This requires e.g. investigation of specific processes like premixing, melt fragmentation and particle heat transfer mode.

In **WP13-2** the processes summarized under the name Direct Containment Heating are studied, which include the dispersion of melt into the various compartments of a reactor, the heat transfer and chemical processes, such as production and combustion of hydrogen.

All work-packages are arranged around the same tasks:

Task 1. Review and selection of available experiments and models for interpretation and modeling activities. Discussion of experimental activities and recommendations for the specification of experiments and programs. Delineate the necessary data to be measured.

Task 2. Synthesis of analyses and interpretations of experiments with existing models or codes. Define important phenomena for which models have to be improved or developed on the basis of the EURSAFE tables.

Task 3. Model synthesis and common proposal of models to be implemented into ASTEC.

C. MAIN ACHIEVEMENTS

C.1 Hydrogen Combustion

During a severe nuclear accident with a possible core melt down large quantities of hydrogen might be produced and released to the reactor containment. As the hydrogen mixes with the containment air the mixture could explode in a violent deflagration event or even detonation depending a large number of factors. Quasi-static and dynamic pressure loadings due to such an explosion could seriously threaten the containment integrity and therefore cause the release of radioactivity into the environment. The possibility of generating missiles is another issue related to hydrogen explosions.

The **H**ydrogen **C**ombustion work package (HC) deals with still open issues identified in the EURSAFE PIRT tables, which are flame acceleration, quenching, deflagration to detonation transition and igniters / catalytic recombiners for mitigation [1]. The experimental as well as the modeling part of this work package is addressing these open issues. Experimental data are obtained from previous experiments as well as from newly performed experiments within SARNET.

The ENACCEF experimental facility operated by IRSN allows studying the influence of hydrogen gradients on flame acceleration and deceleration. The facility is a vertical stainless steel setup, which totalizes a length of 4.9m. It is constituted of two main parts: the acceleration tube (length 3.2 m, inner diameter 154mm), and the dome (length 1.7 m, inner diameter 750 mm). The acceleration tube contains the steam generator mock up and annular obstacles of different sizes dedicated to the flame acceleration. The corresponding blockage ratio varies between 0 and 0.63. Due to the vertical setup it is possible to create defined hydrogen gradients inside the test facility before ignition.

Nine pressure transducers and 16 photomultiplier tubes are placed along the tube. They are used to measure pressure and flame time-of-arrival. Gas velocity ahead of the flame front can be measured with LDV and PIV through viewing windows located in the acceleration tube. More details on the experiments performed in the ENACCEF facility can be found in detail in another contribution to this conference [2].

The second new test facility operated within the SARNET deals with hydrogen mitigation techniques by catalytic recombiners and is operated at Forschungszentrum-Juelich (FZJ). The REKO-3 experimental facility (Figure 1) allows the investigation of catalyst samples inside a vertical flow channel under well-defined conditions such as gas mixture, flow rate and inlet temperature. This is important for model development as separate effects can be studied. The catalyst sheets (stainless steel coated with washcoat/platinum catalyst material) are arranged in parallel forming vertical rectangular flow channels. Such a setup represents a box-type recombiner section, e.g. of Framatome-ANP design. Inside this configuration the distribution of the catalyst temperatures are measured. As new feature it is now for the first time possible to measure the gas concentrations inside the flow channel along the catalyst sheets. The probe head can be introduced at 14 different positions allowing measurement of the hydrogen depletion in flow direction. Several tests have been performed during the reporting period varying initial hydrogen concentrations and flow rates. Some exemplary measurement results are given in Fig. 2.

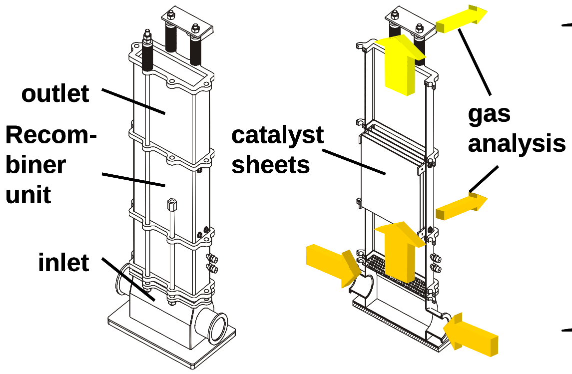

Fig. 1. Sketch of the FZJ REKO-3 test facility for investigation of recombiner sections under well defined conditions

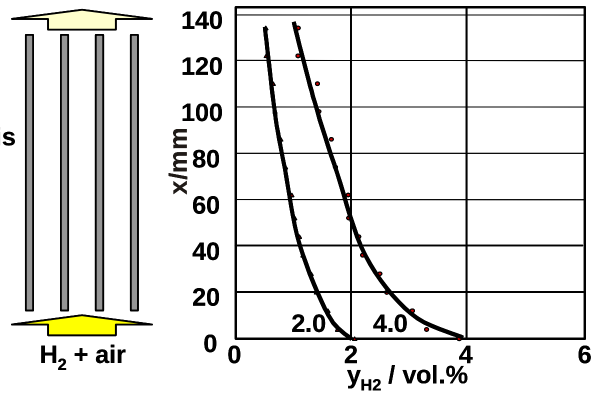

Fig. 2. Transient catalyst temperature profiles (inlet conditions: gas temperature 25 °C, H_2 concentration 2 and 4 vol.%, flow rate 0.8 m/s

These steady-state distributions of the hydrogen concentration were obtained at 2 vol.% and 4 vol.% inlet hydrogen concentration, the inlet gas temperature of 25°C and a flow rate of 0.8 m/s. The symbols represent measuring values while the lines are added for the sake of clearness. The measured values are plotted on the horizontal axis in order to illustrate the vertical arrangement of the catalyst plates. Experimental results obtained have been described in more detail in [3]. For recombiner modeling the code REKO-DIREKT (FZJ) is developed and is validated against experimental data.

Beside experimental work also some modeling is done in HC WP with the purpose of developing new combustion models for the integral ASTEC code when ever possible. In this scope the new PROCO model has been developed and implemented into the ASTEC code by GRS and IRSN. Within the PROCO model two different combustion regimes are modelled: laminar flames and turbulent flames. For the laminar regime a laminar burning velocity is calculated based on an Arrhenius type approach. After the transition to a turbulent flame, the turbulent flame speed is calculated based on the turbulent burning velocity and the isobaric expansion ratio between burned and unburned gases. The expansion ratio is calculated from the gas properties (concentration, heat capacities and heat of release) while turbulent burning velocity is calculated from empirical laws validated against large-scale experiments. The other lumped parameter code been in the HC WP is the GRS code COCOSYS.

Three CFD (Computational Fluid Dynamics) codes are used for hydrogen combustion modeling, which are TONUS-3D (CEA/IRSN), COM-3D (FZK) and REACFLOW (JRC). In comparison to the lumped parameter codes CFD codes model the conservation equation in a transported form. Therefore in addition to the mass and energy equations also three momentum equations (3 space directions) are solved. The benefit of solving the full compressible conservation equations is that dynamic loads due to the multi-dimensional propagation of shock waves generated by the combustion can be assessed much more accurately than in any lumped-parameter formulation. The drawback is that fine grid spacing is needed for CFD. This contributes very much to the much larger calculation time required by CFD codes in comparison to lumped parameter codes. In addition when modelling turbulent flow additional transport equation for turbulent kinetic energy and turbulent kinetic energy dissipation (k and epsilon) have to be solved, which adds to the computational cost.

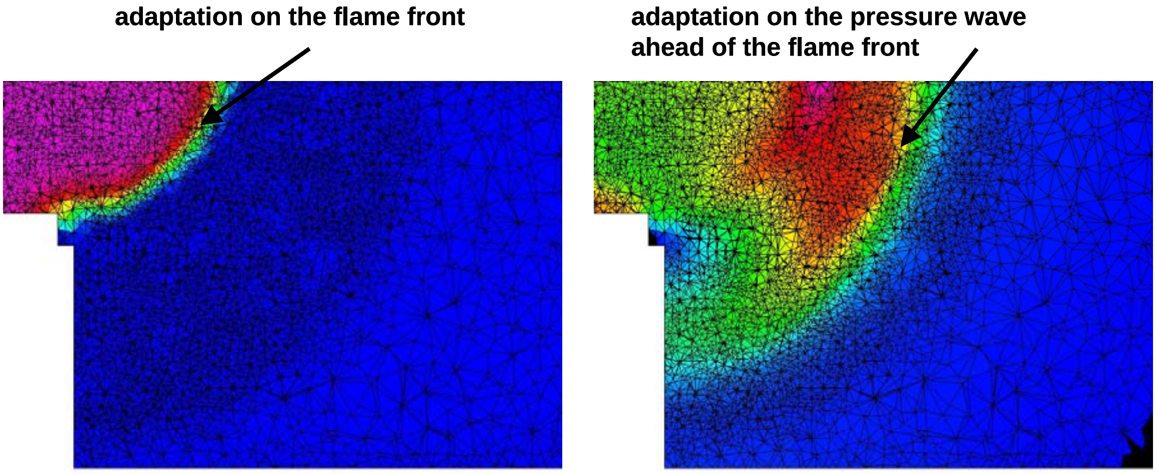

Fig. 3. Hydrogen explosion modeling with the adaptive grid CFD-code REACFLOW. The left picture indicates the flame front. Shown is the H2O concentration field, where the grid in the flame front is refined. The right picture shows the pressure filed, where the grid is also refined in the pressure wave ahead of the flame.

To address the issue of the computational cost due to the use of fine meshes, adaptive grid refinement (and de-refinement) can be used. This has been done for the REACFLOW code, where an adaptive mesh refining method has been developed to refine the grid resolution when needed and reduce the grid resolution when ever possible. Figure 3 shows a detail of a REACFLOW simulation of an explosion modeling in the large-scale RUT test facility. As a new feature, it is now possible to adapt against multiple fields at the same time such as the flame front (see the left picture) and the pressure wave (see right picture). Such decoupling of flame front and pressure wave is very typical for fast turbulent flames, which can threaten the containment integrity almost as severely as a detonation.

Besides developing new combustion models for the ASTEC integral containment code full-scale reactor simulations should be done within SARNET. First results have been published e.g. in [4].

C.2 Containment Atmosphere Mixing

In order to assess the risk of an explosion, hydrogen and steam release and mixing within the reactor containment has to be analyzed and modeled. This release and distribution process is very essential for a possible following combustion event, as they define its initial conditions. Hydrogen distribution issues are addressed in the **C**ontainment **A**tmospheric **M**ixing work package (CAM).

Hydrogen mixing in a reactor containment involves various physical processes. Some of these phenomena are: jet and plumes, steam release and condensation/evaporation on walls, heat transfer to structures, water films and water flow, sump behavior, spray, heat release due to recombiners and effects of volumetric heating by aerosols and fission products.

Two large-scale experimental test facilities are used within SARNET to address open issues in CAM. These facilities are the small scale TOSQAN facility operated by IRSN and the large scale MISTRA facility [5] operated by CEA. Both were recently used to generate data for the OECD/CSNI International Standard Problem No. 47 [6, 7].

Fig. 4. The CEA MISTRA test facility.

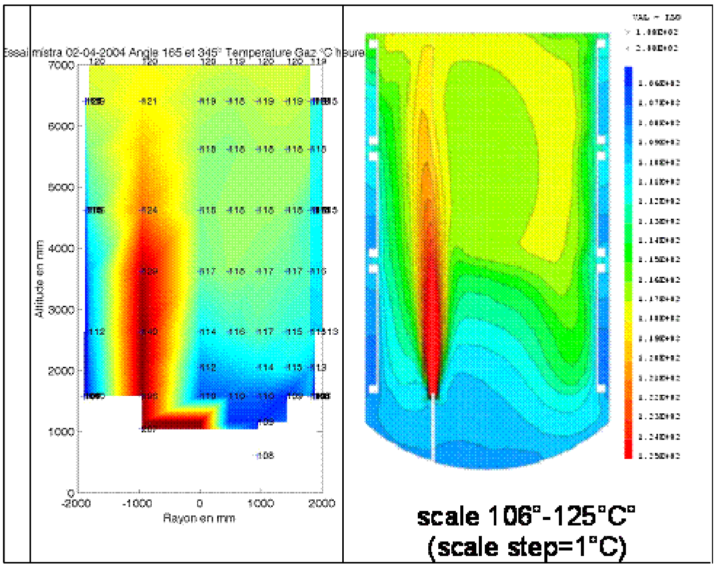

Fig. 5. Steady-state temperature field (experimental, left and numerically simulated, right) corresponding to the air/steam M3 test (off-centered injection, condensation on three condensers)

The TOSQAN test facility is a 7 m^3 (5 m high, 1.5 m diameter) stainless vessel. Wall temperatures can be controlled between 20 and 160 °C, allowing careful studying of all types of condensation effects. Steam and helium (simulates hydrogen) can be released from an injection system with a diameter of 41 mm. The release rates can vary between 0 and 30 g/s for steam and 0 and 3 g/s for helium. An additional spray system allows releasing water at a flowrate between 1 and 40 g/s. An advanced online measurement system with thermocouples, pressure gauges, flow meters, LDV, PIV and RAMAN spectrometry allows recording of all relevant experimental parameters such as temperature, velocity and concentration fields as well as spray size distribution. So fare tests on wall condensation and sprays have been performed.

The second facility used within CAM is the large scale MISTRA test facility operated by CEA. Its vessel (7 m high, 4 m diameter) has a volume of nearly 100m3, and it is equipped with three independent condenser circuits (positioned against the walls and which can operate over a large range of temperatures) which allow to simulate a large range of Grashof numbers – and therefore to study scaling effects, including comparisons with data from the TOSQAN facility. The release rates can vary between 15 and 140 g/s for steam and 1 and 50 g/s for helium with temperatures up to 200 °C. The facility is well-equipped with thermocouples, concentration gauges, flow meters and LDV and PIV for measuring velocities. The MISTRA test vessel can be seen from outside in Figure 4.

In 2004 the M3 experiment (sponsored by IRSN) has been performed in the CEA MISTRA facility where steam was injected at an off-center location. Figure 5 (left side) shows steady state experimental temperature distribution within the MISTRA test facility. The figure indicates clearly the hot temperatures within the release jet and cold temperature close to the walls where condensation occurs. In the right picture the temperature distribution simulated by the TONUS CFD code is plotted for comparison. Overall pictures show good correspondence between experiment and simulation. More details can be found in a paper presented at this conference [8].

As reactor-scale experiments for nuclear accident analysis of CAM is not feasible, numerical modeling is also part of the CAM WP. Both CFD codes as well as lumped parameter codes are used and validated within the SARNET project. The CFD codes used are TONUS (IRSN and CEA), FLUENT (DIMNP), GASFLOW (FZK, NRG and VEIKI), CFX (JSI [9, 10, 12], KTH, NRG, RUB-LEE and UPM) and STAR-CD (NRG). The lumped parameter codes in use are ASTEC (IRSN), TONUS (IRSN and CEA), MELCOR (DIMNP, JSI, NRG and UPM), FUMO (DIMNP [11]), COCOSYS (GRS, RUB-LEE, VEIKI), CONTAIN (JSI and LEI), SPECTRA (NRG) and MAAP4 (NRG). With these modeling tools the following physical aspects are addressed: turbulence, wall condensation, superheated steam and mist, droplet behavior and sprays, sump and heat transfer to and inside structures. Next steps towards full reactor modeling are the development and integration of recombiner models. In particular, the interaction of recombiners with the containment atmosphere and its effect on the hydrogen distribution are being assessed during the 2nd year of the SARNET project.

C.3 Fuel-Coolant Interaction

Fuel-coolant interactions (FCI) and especially their most violent form that is called ‘steam explosions’ are a matter of concern in nuclear reactor safety since long (for more details see the review papers [13-15]). They might occur during a severe accident when molten core materials contact remaining coolant. During recent years, the concern has moved away from the early containment failure as a consequence of an in-vessel steam explosion (the α-mode failure) to other possible consequences of steam explosions, primarily ex-vessel. But, with other concerns than α-mode failure, in-vessel explosions are still considered as well.

The evaluation of possible hazards during severe reactor accidents is largely based on still incomplete and partly contradictory experimental information. One shortcoming of all experiments, one that cannot be overcome, is that experiments can only be performed on a scale that is much smaller than the reactor scale. For extrapolating to the reactor scale and estimating the possible loads on reactor structures, fluid-dynamic codes are developed in several places in the world (see e.g. Ref. [16]). Validation of these codes requires relevant experimental data.

Phenomena identification and ranking performed during the EU-sponsored EURSAFE project has led to selecting ‘FCI incl. steam explosion: melt into water, in-vessel and ex-vessel’ as an item for needed research with the aim of increasing the knowledge of parameters affecting steam explosion energetics during corium relocation into water and determining the risk of vessel or containment failure [1]. As a first consequence, the SERENA research program has been endorsed by the CSNI of OECD/NEA following the proposal of FCI experts from several countries. Phase 1 of this program is essentially finished. Its purpose was to evaluate the available FCI codes with respect to their applicability to reactor accident conditions and to determine whether the available experimental data were sufficient for validating these codes so that they are ‘fit for the purpose’. The results of this work are described in another contribution to this conference [16].

The following FCI codes are being developed by SARNET partners:

Forschungszentrum Karlsruhe :	MATTINA	(essentially for premixing)
IKE (University of Stuttgart):	IKEMIX	(for premixing)
	IDEMO	(for explosion)
IRSN-CEA:	MC3D	(two versions for premixing and explosion, respectively)
JSI (University of Ljubljana)	ESE-2	

More about these codes is described in another contribution to this conference [17]. The present paper is more concerned with experiments.

The only FCI experiment that has been performed until now within the SARNET FCI programme has been the ECO-09, the last of the ECO experiments [18]. These experiments were aimed at really measuring the energy conversion in a steam explosion. To this end, ex-

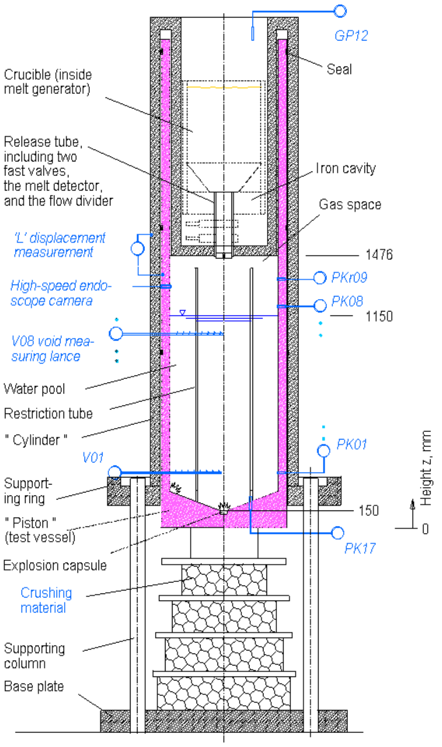

Fig. 6. The ECO test facility. The water pool has a diameter of 0.59 m.

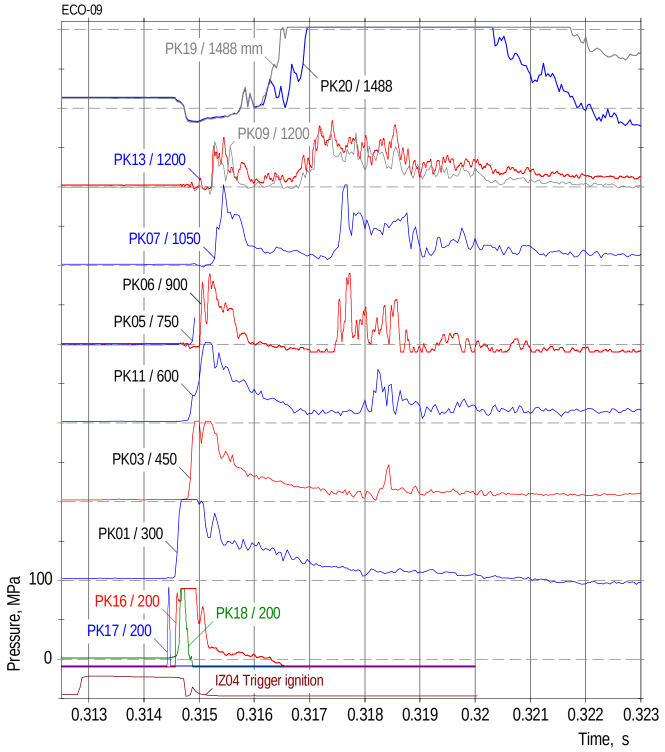

Fig. 7. Test ECO-09: Dynamic pressures after triggering (the numbers give the height of the pressure transducers in millimeters).

plosions have been triggered externally in a test vessel that was contained in a system of cylinder and piston in which the piston could move only by compressing crushing material with known forces. (Compressing the crushing material requires a known and constant force.) In these experiments, the corium melt was simulated by molten alumina (Al_2O_3), so that, from an energetic point of view, the melt masses correspond to threefold masses of corium. The initial temperature was about 2600 K. Figure 6 shows the facility with the stationary cylinder that is fixed to the bottom plate by steel columns and closed on the top by the melt generator in which the molten alumina is produced by a thermite reaction (keeping back the molten iron). The piston which contains the water pool can move downward by compressing the crushing material which is arranged in layers with increasing cross sections (from top to bottom) so that increasing forces are required to compress them (actually the honeycomb structure of the crushing material is oriented vertically). The whole facility is about 4 m high and its mass is about 10^4 kg (10 t).

During the initial tests, melt release started with the melt-through of a steel foil and was terminated (just prior to triggering, to protect the melt generator) by a fast sliding valve. Later, a similar slide valve was used to release the melt (in a more defined way), as shown in Figure 6. The test vessel was equipped with a large number of pressure transducers (most of them flush mounted in the vessel wall) and measuring lances which carry each 8 void probes (conductivity probes) and, close to the vessel center, a thermocouple. These lances were arranged in one plane across the center line and 8 of them formed an array of 8 times 8 void probes on one side of the center while 4 were on the other side. These instruments give useful time resolved information about the extension of the mixing zone during melt release. In some later tests, a steel tube was used inside the test vessel to reduce the amount of water that could interact directly with the melt (c.f. Figure 6).

In the more successful tests 05, 06 and 09, 15…18 kg of alumina were released into the test vessel. During premixing, the pressure in the test vessel rose from the initial 0.25 MPa to

1.3…1.9 MPa. In all cases the explosion pressures that developed after triggering (at the bottom) well exceeded the measuring ranges (45 MPa in ECO-05 and 95 MPa in ECO-06 and 09), see e.g. Fig. 7. Here, the transducers PK16…18 are mounted in the bottom of the test vessel facing upwards and PK19 and 20 are mounted in the roof facing downwards. Why the latter stay at pressures beyond about 100 MPa for such long times (more than 3 ms) is not yet clear. In other locations the peaks exceeding 90…100 MPa are much shorter. In view of the extremely high pressures and their destructive effects on the facility, the work done by the steam explosions on the crushing material is comparatively low so that energy conversion factors of only 0.6 % (ECO-09), 0.8 % (ECO-06) and 2.4 % (ECO-05) have been found. As, on the other hand, the initial conditions for the explosions were not too different, there is no obvious reason for the large variation of these results and we observe (again) that steam explosions are subject to large statistical (or not understood) variations. A report describing these experiments in full detail will appear soon [19]. The data are analyzed by FZK and IRSN.

A new experiment series, KROTOS/Cadarache, is going to start this year. This is a re-installation of the KROTOS facility from JRC Ispra at CEA Cadarache with several improvements, notably a powerful X-ray facility to visualize the melt jet during premixing and a slide valve to initialize the melt release. One of the first experiments will be a test suggested by JSI Lubljana within the PLINIUS programme of the EU. This test aims at studying the effect of fission products that are present in the molten corium. This test may provide important new data:

1. For the very first time, burn-up effects are studied in an FCI test.
2. An important effect of the added material will be a reduction of the solidus temperature of the corium. If the test can be performed with an initial temperature similar to those achieved at Ispra, this test will have a larger difference between initial and solidus temperatures and may thus contribute to understanding the famous ‘material effect’.

The term ‘material effect’ is used for the observation that in the KROTOS test performed at JRC Ispra, the corium did never explode spontaneously (in contrast with alumina) and, if an explosion was triggered, it was much weaker than those with alumina. Until now, the reason(s) for this behavior are not clear. One hypothesis is that the difference between initial and solidus temperatures was so small that most (or all) of the corium was already frozen to such an extent that, after premixing (at the time of triggering), it could no longer be fragmented finely and thus support an explosion. Another suggestion is that the special way in which the melt was released at JRC Ispra caused an important difference in the ways in which the two different materials were initially mixed (premixed) with water. This second argument hopefully will be removed by the release valve installed in the new facility.

Finding the explanation(s) for the ‘material effect’ is an important task of the new KROTOS facility because, if it could be shown that the effect is caused by conditions that are always present in reactor accident situations, evaluations of the risk due to steam explosions could take advantage of this fact. Without such proof it would not be conservative to rely on the material effect. Therefore an important part of the work done on FCI within SARNET will be devoted to analyzing the KROTOS/PLINIUS test and, if available, parallel tests with other materials, e.g. the model corium already used at Ispra.

C.4 Direct Containment Heating

The direct containment heating issue (DCH) has been extensively studied for US-reactor plants in the past, and a large database is available for these geometries. However, different designs of European reactors lead to different conditions at vessel failure and to dissimilar impact of phenomena during blow down. Several phenomena have been identified in the EURSAFE PIRT for which the level of knowledge is still not sufficient to be modeled with the necessary accuracy. Among them the most important are the generation and trapping of particles, which not only govern the direct heating of the atmosphere but also the production and combustion of hydrogen. The pressure rise in the containment is predominantly determined by the concurrence of these two effects.

Only two experimental facilities for the investigation of DCH issues are in operation in Europe, both at Forschungszentrum Karlsruhe (FZK). The facility DISCO-C (Figure 8) is suited for the investigation of fluid dynamic phenomena, such as the two-phase jet, liquid fragmentation, liquid film formation at walls, entrainment and trapping of liquid, and liquid dispersion and deposition. At the other facility, DISCO-H, integral tests can be performed including all relevant DCH phenomena (Figure 9). This test facility is designed to investigate melt dispersion and Direct Containment Heating (DCH) with steam driven iron-alumina melt (2000 °C), in a scenario where the lower head of the pressure vessel fails at moderate system pressure (~ 2 MPa). The position, size and shape of the failure can be varied. The containment is modeled by a pressure vessel with a volume of 14 m^3, rated at 1 MPa. The combined volumes of the reactor pressure vessel and reactor cooling system are modeled by a vessel with a volume of 0.08 m^3. The geometry of the reactor pit and reactor subcompartments is adapted according to the investigated reactor type. The atmosphere in the containment is variable and can be inert, air, steam or a mixture, including hydrogen. Both facilities can model specific reactor geometries in sufficient detail, to take account of the geometry dependence of the processes.

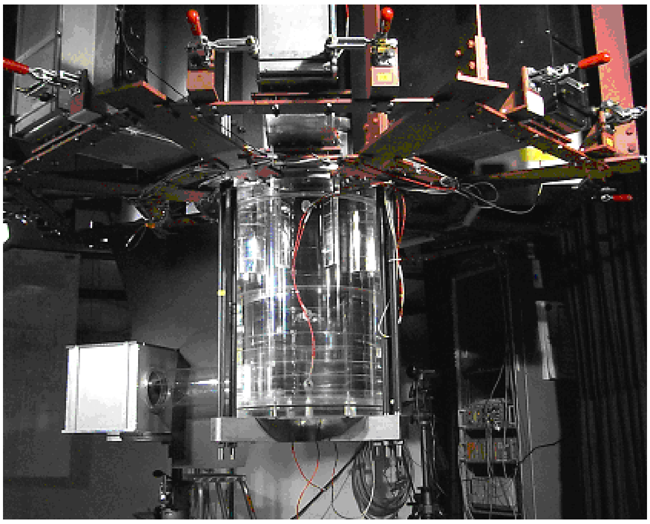

Fig. 8. Plexiglas cavity of DISCO-C

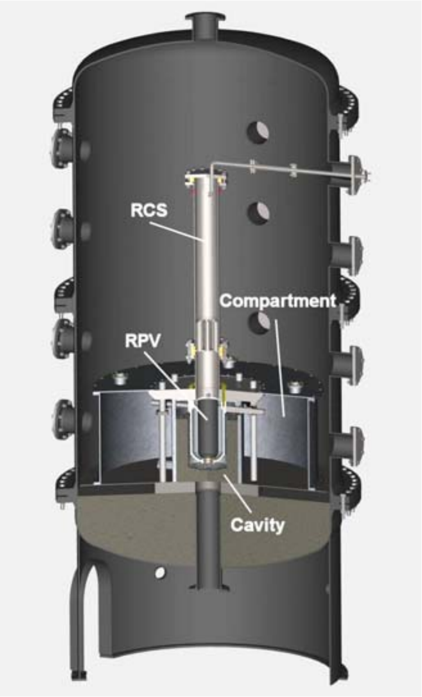

Fig. 9. Scheme of DISCO-H

In the framework of the EC-Program ECOSTAR (5th Framework Program) a test series in EPR geometry (European Pressurized Reactor) was performed in DISCO-C and -H [20-23], and data are available on the STRESA databank. In the year 2004 experiments in both facilities have been performed with a geometry of the French reactor P'4 within a cooperation between IRSN and FZK. One experiment in DISCO-H was conducted in the framework of the LACOMERA program [24]; these data are open to all interested partners.

The experiments have shown the significance of initial conditions, such as size and position of the breach in the lower head of the reactor vessel, but also of the geometry of the reactor cavity and the flow paths into the subcompartments and the containment dome. The contribution of hydrogen combustion to the peak pressure in the containment could be quantified by comparison of experiments with prototypic conditions, i.e. steam driven oxidic and metallic melt into an air-steam atmosphere with preexisting hydrogen, and experiments with inert driving gas and inert atmosphere. While the experiments show the effect of breach size, initial pressure, pit geometry and gas on the containment pressure, the complex interrelationship of those parameters can only be revealed by a detailed code analysis.

Two multi-phase CFD codes are being used for analyzing DCH experiments. The most important achievements have been accomplished with the AFDM code, which is used for this purpose since 1998. Special DCH models have been incorporated into the code including a model for chemical reactions of the metallic melt fraction with oxygen or steam, which involves hydrogen production. The added hydrogen combustion model, however, is a lumped parameter model, that needs to be adjusted with experimental data. Therefore, for extrapolations to reactor scale the use of a hydrogen combustion code is mandatory. The modeling work with AFDM is described in detail in another contribution to this conference [25].

The other CFD code is MC3D, which is used by IRSN only since about one year for the investigation of DCH phenomena. The strength of this code is the complex description of the corium flow. There is however no hydrogen combustion model and the oxidation model is quite parametric. Due to these limitations, the chemical aspects are currently not investigated and the focus is put on dynamical aspects.

Two mechanistic containment codes, CONTAIN and COCOSYS, and two integral code systems, ASTEC and MAAP-4, with their specific modules are applied to DCH problems. In the past code activities related to DCH were performed with data pertaining to US reactors, Zion and Surry. The special issue of Nuclear Engineering and Design, Vol.164 [26] gives a overview of these activities. All the numerical tools used were more or less elaborate lumped parameter codes. CONTAIN was one of them. The modules used in MAAP-4 and ASTEC are also based on the same structures and basic correlations. However, since DCH is highly geometry-dependent, an assessment for each specific reactor type is necessary.

COCOSYS, the containment code system developed by GRS for the analysis of severe accidents, is being extended for the analysis of DCH-phenomena. As a first step the CONTAIN code is used to improve the understanding of relevant phenomena and models. At present GRS uses CONTAIN to select and validate DCH models that are available in the code for the extension of COCOSYS. First evaluations with CONTAIN are focused on the DISCO H02 experiment with EPR geometry. As a result, the important model parameters have been identified and some of them had to be modified by several orders of magnitudes (as the entrainment rate multiplier) in order to fit the DISCO H02 pressures and final debris distributions. The relevant phenomena seem to be identified by the preliminary calculations,. After a first fitting process, the typical influence of thermodynamic parameters could be observed but the approximation achieved is to be improved. Also, the chemistry approach in CONTAIN seems to be correct but the definition of a time-dependent end of the Fe-oxidation as used in the calculation is still not acceptable. Finally, the model settings for trapping have much influence on the combined effect of entrainment/trapping.

MAAP-4 assessment only concerns US reactor plant geometries. No specific interpretation with MAAP-4 of experiments with European-type PWR geometries has been performed up to now.

In ASTEC, the DCH is evaluated within 3 modules: RUPUICUV, CORIUM, CPA. RUPUICUV is the module that calculates all phenomena involved inside the reactor cavity. CPA is the module for the calculation of the containment atmosphere. It is also involved in other phenomena than DCH. CORIUM is simply an interface module. As for MAAP and CONTAIN, the RUPUICUV development was based on experiments with geometries not fit for European reactors. Recently, the calculation of the entire DISCO series with French PWR geometry has been carried out with varying success with the standard parameters of RUPUICUV. In general, the dispersion seems to be overestimated. However, the weaknesses of RUPUICUV are quite well identified and it is understood that improvements are necessary.

The final objective of the code work is the application to reactor scale. The difficulty is the double extrapolation needed to go to the reactor situation: all experiments involve a simulant material and a small length scale. The material extrapolation to prototypical corium is particularly difficult because it does not concern only physical properties but also chemical interactions. Before this can be done for lumped parameter codes it has to be performed with CFD codes, such as AFDM in combination with a hydrogen combustion code.

E. CONCLUSIONS

Several containment issues have been identified as areas of uncertainties by the EURSAFE project [1]. They occur late in severe accidents as consequences of severe core damage. So, their probability of occurrence is extremely small but they involve the risk of a failure of the containment, the final safety enclosure. Therefore a conservative approach is required. On the other hand the damage potentials are such that simple bounding considerations are not useful and there are large uncertainties connected with the initial and boundary conditions – especially in the constitution and the release of the molten core materials involved, the corium. Similarly uncertain are amount as well as mode of release and release location of the hydrogen.

Experiments are being performed and models are developed and refined for the application in CFD codes to the reactor case. They are the basis to provide ASTEC with appropriate physical modeling.

The FCI and DCH studies suffer from the fact that experiments can only be performed on a scale that is much smaller than the reactor scale. In addition, many of the experiments have been performed with simulant materials in place of 'real' corium. This requires a twofold extrapolation in size and material behavior which can be done reasonably with adequate codes only. An essential point of the experiments is to understand the origin and the consequences of the behavior of different materials. Progress in these areas requires both further experimenting and code development and is bound to be slow, given the manpower available.

REFERENCES

[1] Magallon D., *et al.*, European Expert Network for the Reduction of Uncertainties in Severe Accident Safety Issues (EURSAFE), *Nucl. Eng. Des.*, 235, 309-346, 2005

[2] Bentaib A., et. al., Experimental and numerical study of flame propagation with hydrogen gradient in a Vertical Facility: ENACCEF, this conference, Session 3

[3] Reinecke, E.-A., et al., Modelling of catalytic recombiners: comparison of REKO-DIREKT calculations with REKO-3 experiments, Proc. Int. Conf. Nuclear Energy for New Europe, Bled, Slovenia, Sept. 5-8, 2005

[4] Baraldi, D., M. Heitsch, M. and Wilkening, H., Investigation of hydrogen combustion in a simplified EPR containment with CFX and REACFLOW, 11th Int. Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, 2005

[5] Tkatschenko, I., Studer E., Paillère, H., Status of the MISTRA programme for the validation of containment thermal-hydraulic codes, 11th Int. Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, 2005

[6] Tkatschenko, I., Studer, E., and Paillère H., MISTRA facility for containment lumped-parameter and CFD code validation: example of the International Standard Problem ISP47, Proc. Int. Conf. Nuclear Energy for New Europe, Bled, Slovenia, Sept 5-8, 2005

[7] Fischer K., Studer E., Malet, J., Porcheron E., and Vendel, J., Conclusions of the International Standard Problem ISP47 on containment thermal-hydraulics, to be published by OECD/CSNI.

[8] Simon H., Blumenfeld L., Tkatschenko, I., Paillère, H., Bentaib, A., Vendel J., Air steam tests in the MISTRA facility: experimental results and validation of the lumped-parameter/CFD TONUS code, this conference

[9] I.Kljenak, M.Babic, B.Mavko, I.Bajsic, Modelling of atmosphere mixing and stratification in the TOSQAN experimental facility with the CFX code, 11th Int. Top. Meeting Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, France, Oct. 2-6, 2005.

[10] I.Kljenak, M.Babic, B.Mavko, I.Bajsic, Modelling of containment atmosphere mixing and stratification using CFD approach, 13th International Conference on Nuclear Engineering ICONE-13, Beijing, China, May 16-20, 2005.

[11] N. Forgione, S. Paci, Computational Analysis of Vapour Condensation in Presence of Air in the TOSQAN Facility, 11th Int. Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, Oct. 2-6, 2005.

[12] M.Babic, I.Kljenak, B.Mavko, Simulation of ThAI experiment on atmosphere mixing with the CFX code, International Conference "Nuclear Energy for New Europe 2005", Bled, Slovenia, Sept. 5-8, 2005.

[13] Cronenberg, A. W. and Benz, R., Vapor explosion phenomena with respect to nuclear reactor safety assessment, in: Lewins, J. and Becker, M. (Eds.), *Advances in Nuclear Science and Technology*, Vol. 12 pp. 247-338, 1980

[14] Corradini, M. L., Vapor explosions: A review of experiments for accident analysis, *Nuclear Safety*, 32, 337-62, 1991

[15] Berthoud, G., Vapor Explosions, Annual Rev. Fluid Mech. 2000, pp. 573-611

[16] Magallon D., et al., Results of the International Program OECD-SERENA on Fuel-Coolant Interaction, this conference

[17] Meignen R., et al., Comparative Review of FCI Computer Models Used within SARNET, this conference

[18] Cherdron, W., Jacobs, H., Kaiser, A., Schütz, W., Measurement of steam explosion loads: The ECO experiments, 11th Int. Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-11), Avignon, Oct. 2-6, 2005.

[19] Cherdron, W., Huber, F., Kaiser, A., Schütz, W., ECO Steam Explosion Experiments - Documentation and Evaluation of Experimental Data, Forschungszentrum Karlsruhe Report FZKA-7011, to appear 2005

[20] L. Meyer, M. Gargallo, Low Pressure Corium Dispersion Experiments with Simulant Fluids in a Scaled Annular Cavity, *Nuclear Technology*, **141**, pp.257-274, 2003.

[21] L. Meyer et al., Melt dispersion and Direct Containment Heating (DCH) Experiments in the DISCO-H Test Facility, Report FZKA 6988, Forschungszentrum Karlsruhe, 2004.

[22] L. Meyer, G. Albrecht, D. Wilhelm, Direct Containment Heating Investigations for European Pressurized Water Reactors, 6th Int. Conf. On Nuclear Thermal Hydraulics, Operations and Safety (NUTHOS-6), Nara, Japan, October 4-8, 2004.

[23] L. Meyer, G. Albrecht, M. Kirstahler, Corium Dispersion and Direct Containment Heating Experiments at Low System Pressure, 10th Int. Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, October 5-9, 2003

[24] R. Meignen, D. Plet, C. Caroli, L. Meyer, D. Wilhelm, Direct Containment Heating at low primary pressure: Experimental Investigation and multidimensional modeling, (NURETH-11), Avignon, France, October 2-6, 2005.

[25] D. Wilhelm, The challenge of simplifying DCH modeling, this conference, Session 3, paper No. 4

[26] Nucl. Eng. Des., 164, 1996 (Topical Issue on DCH).