MAXSIMA

(Contract number 323312)

Final report
Description of the main S & T results/foregrounds

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Description of the main S & T results/foregrounds

WP2: Safety Analysis is support of MYRRHA

The MYRRHA system design proved to be very solid and reliable. No Design Basis Conditions accidental transient will cause the reactor safety criteria to be not fulfilled, provided the reactor shutdown system activation according to the foreseen logics. Such conclusion is obtained from the deterministic analysis as well as from the Uncertainty quantification methodology. A certain attention should be paid towards the primary LBE freezing, by foreseeing suitable design features to delay such occurrence, should it be considered a safety-related issue.

Nevertheless, the extended number of transient analyses performed has provided an extensive feedback, which will prove its full usefulness for the future advancements of the MYRRHA plant design.

The main conclusions achieved by each Task are summarized as follows:

Task 2.1 Neutronic and shielding analysis in support of safety studies

The neutronic analyses were carried out with Monte Carlo and deterministic methods (MCNPX and ERANOS codes, respectively) coupled with the JEFF3.1 nuclear data. In addition to the core characterization in normal operating conditions (e.g. k_{eff}, peaking factors), the reactivity feedbacks and the impact of the main accidental events on the core reactivity were carefully investigated:

- To define the possible working ranges of the MYRRHA cores by providing, at the same time, an important envelope for safety studies
- To provide all neutronic parameters required for transient analyses (T2.2 and T2.3)
- To validate the Neutron Kinetics simulation tools embedded in the codes used for transient analysis

The potential operation range of the MYRRHA facility was defined by identifying the minimum and maximum core configurations that can be arranged in both operational modes (critical and sub-critical). In particular:

- The minimum cores were defined by adopting fresh fuel and without experimental devices (In-Pile Sections, IPS);
- The maximum critical and sub-critical cores were defined by adopting a certain number of IPSs and the fuel compositions at the cycle equilibrium.

The deterministic and Monte Carlo codes were used complementarily by exploiting their peculiarities:

- MCNPX was used to identify the maximum cores at the equilibrium of the fuel cycle. The accurate fuel cycle analyses in critical / sub-critical modes were carried out by envisaging 18 / 12 x 90 day irradiation sub-cycles, respectively.
- ERANOS was used to define the minimum cores and to evaluate the parameters needed for safety analyses. The reactivity coefficients were obtained through a partial derivative approach based on the analysis of three different core conditions: a “theoretical” cold state at 20 °C (to assess safety limits), the zero power level at 200 °C and the normal operating conditions, in which the power level was tuned by assuming 500 °C as maximum clad temperature. The synoptic comparison among all the results represents an important envelope for safety studies: it defines the variation range for the main neutronic core parameters.

The shielding and activation studies were performed with the aim to contribute to the shielding optimization of MYRRHA and to predict the activation levels of the structural materials around the core. The method adopted, based on the combined use of two state-of-the-art Monte Carlo codes (MCNPX and FLUKA), allowed to obtain realistic neutron fields around the core barrel, which are successively used to build complex source terms as input for FLUKA detailed analyses. The (neutron and gamma) radiation fields were evaluated for the system in operation and for the coupled residual radiation.

The work represents an evolution of the analyses carried out in the FP7 Central Design Team (CDT) project. The CDT analyses shown that the radial radiation containment was correctly sized; however, two main design issues remained unsolved:

- The vertical radiation containment function was not guaranteed by the cover plate design, which was correctly sized in the vertical direction, but significant radiation losses were present at its boundary (at cover-vessel connection).
- The choice of the argon as plenum gas posed some activation problems for short and medium term, due to the abundant production of ^{41}Ar ($t_{1/2}$ = 109.61 min) and ^{37}Ar ($t_{1/2}$ = 35.04 d).

These problems imposed new design solutions to be adopted for the MYRRHA reference design configuration.

The first two goals of the FP7-MAXSIMA activities were devoted to test the vertical containment provided by the new cover plate design and the choice of nitrogen as plenum gas. The study was performed in the sub-critical operation mode at the maximum proton current of 4 mA (most conservative case). The new cover plate design resulted very efficient in the vertical radiation containment, including boundaries: the neutrons are effectively absorbed in the cover borated water but, on the other hand, the neutron capture reactions on hydrogen and on boron generate photons in the MeV range, which are not perfectly contained by the steel top cover plate; a thicker external steel plate could be used. The choice of nitrogen as gas plenum did not evidence significant problems, apart the known activation issue due to ^{14}C production (below the commonly adopted regulatory limits).

The shielding analyses final purpose consisted in supplying a database throughout a very accurate and conservative prediction of the activation levels of the structural materials around the core. Such database provides conservative estimate of the activation of 62 representative regions in the structural materials, from the core vessel to the cover plate, including the inventory of the radioactive isotopes, at different decay times (up to 104 years).

Task 2.2 Transient analysis using system codes

In the absence of experimental data to validate the calculation tools employed for MYRRHA transient analysis (mostly System Thermal-Hydraulics codes), the work methodology followed for the Task 2.2 activities has been based on code-to-code comparison.

The agreement between the different codes is overall quite satisfactory, with the limited differences explained in details through discussions between the participants.

However, such differences never resulted to be wide enough to question the fulfilment of the specific safety requirements adopted for the MYRRHA plant.

Thus, the conclusions reached are based on a solid foundation of appropriate and consistent analysis methods.

The results of the safety analysis performed in this framework demonstrate that the MYRRHA facility is a very robust design ascribable to the unique combination of inherent and plant design features. Indeed, for all the Design Basis Conditions accidental scenarios, the core and primary system temperatures remain below the temperature limits established to avoid fuel clad failure. Only in a very limited number of transients, where no reactor shutdown is triggered and thus considered in the category of Design Extension Condition events (namely, Unprotected Loss of Flow, Control Rod Ejection and Unprotected FA inlet section blockages transients), a restricted number of pins could experience failure.

It is important to remind that the analysis of DEC transients is, however, only aimed at finding the reactor system limits, in order to design an adequate reactor shutdown system able to cope safely with similar Initiating Events. DEC events are, in fact, excluded.
For what concerns FA blockage accidents, no clad failure is foreseen for FA section blockages up to 80%. Larger FA flow blockages are already excluded by design, through the implementation of specific features that always allow a LBE flow path to the FA active zone. However, if the accident detection and reactor shutdown is performed within a reasonable time, the systematic clad failure is prevented.

About the system behaviour towards primary LBE freezing, in case of a sudden activation of the secondary/tertiary cooling systems removing maximum power starting from “cold” conditions, the Primary Pumps run is required to increase the primary pool mixing, thus preventing rapid Secondary Cooling System depressurization and delaying local LBE freezing at the PHX outlet.

The DHR-2 system efficiency has been tested for protected and unprotected conditions. The unprotected case results in a very low power level (thanks to strong MYRRHA inherent reactivity feedbacks), comparable to the Decay Heat which represents the power input for the protected event. It has been shown that RVACS is able to remove the core power by cooling the Reactor Vessel through water injected into the reactor pit. The clad temperature is maintained below the failure limits for about 5 hours (when RVACS pressure increase requires purge valve opening).

The application of space Neutron Kinetics coupled with a sub-channel code has been tested on specific transients where local neutronic effects are expected to be important. Such preliminary attempt has proven to be satisfactory, showing evaluations in line with the System Thermal-Hydraulics code predictions. However, strong limitations due to computational time and model detail refinement must be considered when attempting a coupled analysis.

Concerning the sub-critical mode transients (run only by SCK•CEN), the transient analysis has shown, generally, trends qualitatively comparable to the critical mode behaviour. Hence, similar conclusions are drawn for the majority of the transients. A particular attention is given to spurious beam cut-off, which, depending on the cut-off duration, might induce thermal stresses during the beam cut-off and reactivation phases. The impact on the material fatigue requires a deeper mechanic analysis to investigate the consequences of such transients in case of frequent beam trips.

The operational transients, defined according to the procedures of the MYRRHA reactor operation, proves how it is possible to modify the plant state without incurring in situations potentially jeopardizing the reactor safety.

The definition of an enveloping case covering both critical and sub-critical mode is not always straightforward. The most challenging conditions can be met in both operating modes. In particular, transients where the reactivity feedbacks can provide a notable contribution to the accident mitigation are usually more challenging in sub-critical mode, where the power is driven by the beam current and reactivity feedbacks contribution is limited. For such cases, the subcritical conditions must be considered for the definition of the countermeasures to be taken to cope with the transient. On the other hand, in transient events where the reactor protection system is triggered, the Decay Heat (proportional to the power) defines the evolution; in such cases, the critical mode, bearing higher nominal power, is considered representing an enveloping condition.

One of the most interesting features characterizing the Task 2.2 activities is the application, for the first time in MYRRHA safety analysis, of a complete Uncertainty quantification and parameters' propagation Sensitivity (U + S), with the purpose to verify the system compliance with the failure limits by relying on a methodology also including uncertainties in design parameters.

For the application of the U + S analysis to the MYRRHA case, a list of 15 input parameters, associated to a suitable variation range and statistical distribution, has been defined by the Task 2.2 participants, according to the past experience available in fast reactor safety analysis. The propagation of the input parameters variation has been checked in 15 safety-relevant output parameters, with the purpose to investigate the impact of an input variation with respect to the reactor failure limits.

The most important conclusion drawn from the U + S analysis methodology consists in the fact that the system reacts as a good filter for most of the input parameters variations, showing relatively limited safety-relevant parameters' variations.

A good agreement on U + S main findings and conclusions has been found between participants. The relevant input parameter variations have been well identified and their importance properly quantified.

It has been agreed between participants that a considerable effort should be devoted in the identification of a complete list of input parameters associated with a correct quantification of uncertainty bands. This appears to be the key step towards a comprehensive identification of parameters' propagation.

In general, the U + S methodology represents a concrete advancement compared to the pure deterministic approach used in previous FP projects. Such methodology appears to be promising for future applications to any kind of reactor systems, specifically with the aim of including a safety demonstration not solely based on deterministic margins but also relying on statistical evaluations.

Task 2.3 Severe accident analysis

SIMMER-III (2-D) and SIMMER-IV (3-D) calculation models have been set up under Beginning of Cycle conditions for the maximum critical and subcritical cores, respectively. Based on the SIMMER-III 2-D model, neutronic assessment of feedback coefficients has been performed for the critical core, for which also ERANOS calculations were done at KIT, and compared with the evaluations of SIMMER and the results of Task 2.1. Thanks to this comparison, SIMMER-III Doppler and expansion models have been updated for further calculations. UTOP benchmark calculations have been performed and good agreements with other partners in Task 2.2 have been achieved. FA blockage and pin bundle blockage calculations for the critical core with neutronic feedback are made with variation of blockage position and numbers of blocked channels. 3-D simulations, despite not being initially planned, have been preliminary run in view of future plans. Several 3-D models have been developed, in particular with and without special meshes for interwrapper gaps between subassemblies, and applied for the sub-critical case.

The core and the fuel assembly models, used respectively to analyse the protected core blockage scenario and to investigate damage propagation following defective pin failure, have been updated, according to the new design specifications of the MYRRHA core.

In accordance with the new MYRRHA critical and subcritical core designs adopted in the MAXSIMA project, unprotected and protected severe transients were simulated using the SIMMER-III and SIMMER-IV codes at KIT and SCK•CEN. 2D calculation models have been set up for the whole primary circuit with the critical core. A neutronic feedback assessment has been done for the critical core. The calculation of the subcritical core is done under the assumption of constant power for both steady state and the blockage transients.

The main conclusion of transient calculations can be drawn as follows:

- The UTOP calculations of the critical core show that the most severe case, showing a slight prompt criticality and a power excursion of a more than 50 times the nominal value, will not lead to a clad damage, only to a possible fuel melting around the pellet centreline. The UTOP benchmark shows a good agreement between different participants.
- The unprotected FA blockage (UBA) calculations of the critical core show that the most severe FA blockage accident will lead to a clad melting and wrapper wall breakup, but no core damage propagation occurs, while the fuel particles move out of the active core region through the inter-wrapper gap and power is immediately reduced. Two different axial blockage positions and various subchannel ring blockages were investigated. Fuel particle size has an effect on the fuel swept-out process.
- The protected core blockage (PCB) calculations show that the possible highest reactivity insertion due to the clad failure and core damage is 1600 pcm, which is far below the CR shut-down subcriticality margin (11507 pcm), but it is half of the subcriticality margin in the ADS mode (3000 pcm).
- The single pin failure induced blockage (SPB) calculations show that there is no pin failure propagation and coolant temperature increase can be up to 25 °C.

The inter-wrapper gap model developed for the SIMMER-IV code was applied to 3D FA blockage calculations of a heterogeneous arrangement of subassemblies around the blocked FA in the subcritical case. It is observed that a quite mild fuel pin damage takes place, but no wrapper wall breaks up and no further FA damage occurs, and fuel particles are dispersed to the upper part of the FA and the upper coolant plenum. This is mainly due to the 3D simulation of coolant flow in the gaps and to heterogeneous arrangement of surrounding SAs (not only including fuel).

WP3: Core Component Safety

The goal of this work package was to test the integrity of the reactor core under two exceptional operating conditions, first: studying the heat transfer in a partially blocked fuel assembly and second: the reliable movement of the control and safety rods of the reactor core. To address this issue, two experiments were carried out: the measurement of heat transfer in a 19-pin fuel assembly mock up in 4 different configurations (unblocked, one central sub channel blocked, one edge sub channel blocked and 6 central sub channels blocked) at the Karlsruhe heavy liquid metal laboratory of KIT and the measurement of control rod and safety rod movement on a full scale mock up at the COMPLOT LBE facility at SCK•CEN.

Numerical support by partner NRG for the rod bundle and partner CRS4 for the safety and control rod movement was carried out for pre-test and post-test analysis of the experiments.

Task 3.1 Thermal hydraulic fuel assembly blockage experiments

In this task, the thermal-hydraulic effects of flow blockages in a liquid-metal cooled fuel assembly were investigated. In this context, the experimental investigations at KIT-KALLA considered three consecutive campaigns in the THEADES LBE loop. In the first campaign a reference unblocked 19-rod bundle was studied, upon which local, internal blockage elements were later inserted. In the following campaigns at first, the effect of small blockages were studied, and later these were replaced by larger ones.

In all cases, operating conditions representative of the scenarios expected in the MYRRHA reactor were imposed. Similarly the characteristics of the blockage scenarios were selected according to the worst-case postulated events. In a conservative approach, solid blockages were installed, while a certain porosity can be expected if they are formed by accumulation of particles. Moreover, they were constructed using a filling material of low thermal conductivity, instead of being metallic. All blockage elements have the same length of 55mm which is a sixth of the wire pitch but different position and size. In total, three blockages were investigated: one central sub channel, one wall sub channel and six central sub channels.

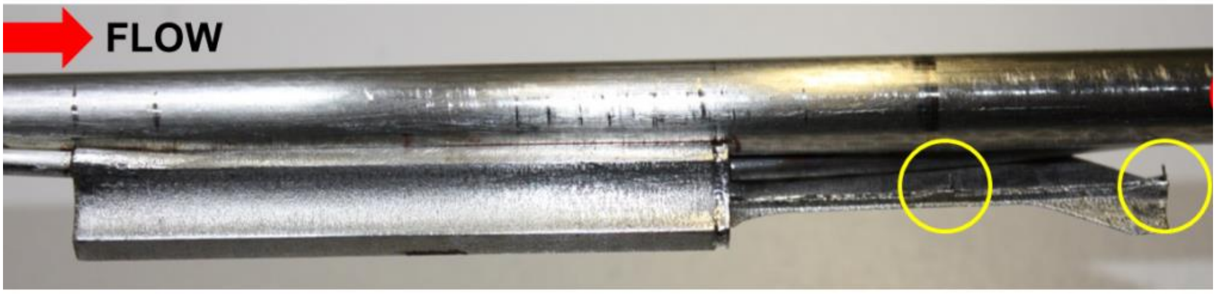

Figure 1: Blockage element for one blocked central sub channel
with two Thermocouples (yellow circles) in the wake regieon

It was observed that the increase in pressure drop due to the blockages is negligible for all cases, indicating that these internal blockage elements cannot be detected during operation as they do not affect the mass flow rate or outlet temperature in the affected fuel assembly. Expressions are developed for the local wall temperature increase as a function of the Reynolds number, heat flux density and blockage thermal conductivity. For the small blockages this increase is acceptable within the safety margins. For the larger blockage, the increase is more significant and the fuel clad integrity might be compromised at full-power conditions.

Task 3.2 Safety rod system tests in Heavy liquid metal

The MYRRHA Control Rod system consists of an absorber bundle within a guide tube filled with LBE. During normal operation the control rods are inserted in the LBE, in the lower part of or below the active core. The high density of the liquid metal coolant allows buoyancy to be the passive driving force for the emergency insertion of the control rods during SCRAM. In this particular case the control rods will also have a safety function. The MYRRHA design requires that in the event of SCRAM, the insertion of the control rods should take less than 1 second. The operation of this CR system within liquid metal is rather different from standard systems.

Figure 2: 19-pin control rod bundle

For the purpose of this task, three experimental test campaigns have been performed on a full-scale mock-up of the MYRRHA control rod in an LBE test section, installed in the COMPLOT LBE facility at SCK•CEN. The tests have characterized the control rod steady state hydraulics in LBE at 200°C under isothermal conditions and established a component system curve for determining the MYRRHA nominal flow rate through the control rods.

Dynamic SCRAM insertion tests were performed at various LBE flow rates, including SCRAM insertion at low flow rates and no flow conditions. The experimental results show that the control rod is fully inserted within the required 1.0 seconds, for the so-called nominal range of LBE flow rates tested. Faster insertion times are measured for higher LBE mass flow rates, as expected due to the increased fluid drag at higher LBE velocity. The no flow SCRAM achieves a 90% insertion in the order of 1.2 seconds for all campaigns.

Task 3.3 Fuel blockage simulation

The pre-test calculations of the 19-pin wire-wrapped rod bundle in the KALLA lab at KIT (task 3.1) were performed in support of the design of the experiment. The influence of the single sub-channel blockages is limited, leading to an acceptable maximum cladding temperature, having a small wake and therefore a fast flow recovery. The blockage covering six sub-channels leads to very high temperatures. Therefore, smaller power inputs are modelled and a linear relation is found between the power input and the maximum cladding temperature. All blockages lead to an increased pressure drop below 8%.

The post-test results of the internal blockages showed that the pressure drop in the bundle is computed within 15% of the experimental results. This is a similar agreement with the experiments as in the unblocked setup in the former FP7 SEARCH [GA295736] project. Also, the temperature trends in the unblocked region are in agreement with the unblocked results, showing a difference in the range of +/- 10 °C. However, the differences in and near the blockages reach up to 95°C.

Sensitivity studies show a sensitivity of a few degrees to: the mesh density, a change in thermo-couple position of 0.1mm, variation the turbulent Prandtl number, changing the turbulence formulation from isotropic to anisotropic and including the conjugate heat transfer in the hexagonal wrapper. Inducing leakage paths between the blockages and the rods or increasing the thermal conductivity of the blockage has a larger effect by highly decreasing the blockage temperatures, though the effects quickly disappear downstream the blockages.

It is recommended that the cause of the differences between the experimental measurements and the simulation results will be examined in more detail in a future collaboration.

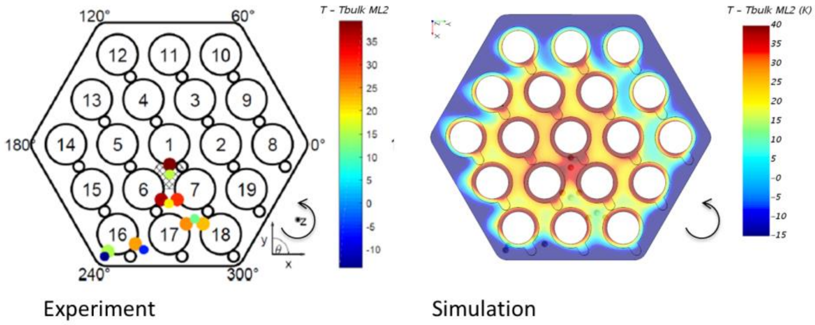

Figure 3: measured and simulated temperature distribution for the blockage of one central channel at nominal operating conditions.

Next to the internal blockages, blockages at the inlet headers of a cluster of 7 complete fuel assemblies were modelled. The central fuel assembly was modelled with two approaches, namely including all rods, grids and wires or replacing them by a porous medium inducing the same pressure drop. The porous medium approach allows future elaboration to full core models within reasonable computational costs. The surrounding fuel assemblies were represented by porous media in both setups.

Three blockage ratios were applied at the inlet of the central fuel assembly upstream the side inlets, blocking 60%, 80% and 100% of the flow area. Both modelling approaches give similar results and show that the 100% blockage only leads to 5% mass flow decrease and 5% temperature increase in the central fuel assembly. This is due to the side inlets in the inlet header, allowing coolant to flow from the adjacent fuel assemblies into the central assembly.

Finally a blockage at the grid located just downstream of the inlet header was modelled, blocking 90% of the flow area as a very extreme case. This lead to a maximum flow reduction of 75% and an average temperature increase of 250% of the original temperature difference between the inlet and outlet and therefore surpass the limits that are defined for accident scenarios.

Task 3.4 CFD simulation of safety / control rods system

A CFD analyses of the control rod dynamics in a numerical model reproducing the test section of the COMPLOT experimental facility was performed. In order to reproduce the movement of the control rod assembly, mesh morphing techniques and automatic optimized re-meshing strategies have been developed and employed, including coupling of the simulation code with Java scripts. These techniques have been applied in a first step for the reproduction of the first part of the movement. For the simulation of the second part of the movement, the approach had to be changed, given the complexity of the flow path, which is not anymore just a guide tube.

For the second part of the movement, the methodology of overlapping grids called overset mesh was used. A good control in a complex flow path configuration has been acquired. Issues like the non-conservation of the mass flow and the oscillations in the fields have been investigated and resolved. The main tools used in this sense consisted in improving the definition of the region enclosing the moving component and optimizing the grid discretization in the overlapping zones. The correct treatment of narrow gaps in the flow path was solved by locally increasing the mesh density and modifying the LBE viscosity.

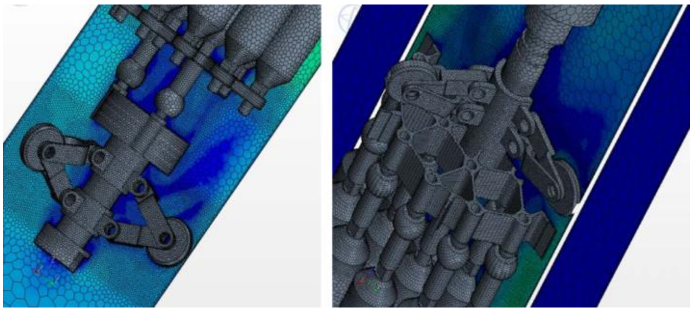

Figure 4: Volume mesh of the 19-pin bundle lower part (left) and upper part (right)

After that, calculations have been upgraded to the conditions of the COMPLOT experimental facility including free surfaces in the buffer tank and in the guide tube representing the cold and the hot plenum of the MYRRHA reactor.

The two-way coupling overset mesh motion, implemented preliminarily in a single phase flow configuration, could be upgraded to a multi-phase flow set-up, with the free surfaces behaving well with the moving component passing through. The Volume of Fluid (VOF) model was used to simulate the insertion of the control rod in the MYRRHA configuration with two different geometries of the 19 absorber pin bundle and allowed the correct transient simulation of the rod insertion. The Principle of Operation of the hydraulic damper was correctly captured numerically. The numerical results generally show a good agreement with the experimental results, within the required 1.0 seconds or slightly above, as in the no-flow configuration. A larger bundle providing higher buoyancy was implemented in the post test simulation for comparison and evaluation of the control rod force differences.

The coherence of the results in the COMPLOT-MYRRHA set-up was fully demonstrated. The control rod insertion time is 10% smaller and confirms COMPLOT as a conservative experimental facility and representative of the MYRRHA conditions.

WP4: Steam Generator and cooling safety

Task 4.1 SGTR Propagation

Concerning the large scale experiment in CIRCE facility (@ENEA Brasimone, Italy), the main objectives of this task were constituted by the experimental and analytical investigations of SGTR event, in a relevant configuration for MYRRHA reactor, and post-test analysis by a suitable code (i.e. SIMMER-IV code).

For this aim, a dedicated test section was designed, constructed, instrumented and implemented in CIRCE pool. The test section was mainly characterized by four full scale portions (four bundles of 31 tubes) of MYRRHA PHX tube array, for performing four SGTR tests (named SGTR-A, SGTR-B, SGTR-C and SGTR-D), one at a time.

A challenge in performing the experiments was related to the correct reproduction of steam/water flow conditions.

Indeed, in the central tube of the bundle in which the test was performed, water flowed upwards and tube rupture was obtained by an external hydraulic system. Water was provided by a pressurized and heated tank upstream CIRCE and discharged in a condenser tank downstream it. Before rupture, LBE flowed downwards shell side of the investigated tube bundle, pushed by a pumping system composed by a centrifugal and jet pump arranged in series. Both water and LBE pressure, temperature and mass flow rate were regulated for reaching, before rupture, initial conditions in agreement with MYRRHA PHX design parameters. Two rupture positions were investigated: bottom and middle. In the two inner tube bundles (named SGTR-B and SGTR-D), of CIRCE main vessel, water injection was performed in bottom location, near the lower tube plate. In the outer two bundles (called SGTR-A and SGTR-C), the rupture was performed at middle position between two spacer grids (second and third one).

The assembling of mentioned components in final test section configuration and its implementation in CIRCE facility, as well as the adopted instrumentation for high quality data acquisition were performed supported by numerical simulations.

The experimental campaign was successfully carried out. The initial and boundary conditions of MYRRHA PHX were reached in agreement with design data for all the test carried out. The critical LBE pumping and LBE sliding valves (aiming to flow LBE only in the SGTR-TS in which test was ongoing, isolating the other three bundles) systems worked properly. Also water line components (water tank, conduct, mass flow rate and pressure regulation valves, isolation valve and discharge tank) fulfilled accurately their tasks. Reached scheduled boundary and initial conditions, the most critical issue was to perform tube rupture simultaneously at precise and accurate DACS operation, being tests not repeatable. The effort made during design, construction, assembling phases and commissioning tests assured the successful execution of the four scheduled tests.

The experimental results highlighted a very satisfactory test repeatability for middle (runs SGTR-A and SGTR-C) and bottom (runs SGTR-B and SGTR-D) rupture scenarios. The couple of runs for each tube rupture position provided pressure (measured in cover gas), temperature and strain time trends very similar, showing high test repeatability. In these tests, being the water injection performed in lower position of LBE pool, sloshing of LBE free surface was identified by oscillations measured in the cover gas pressurization time trends. It was confirmed by numerical analysis performed by SIMMER code.

The test section was highly instrumented with thermocouples (TCs). Seven levels of TCs were set in each tube bundle and were acquired (at 50 Hz) in the tube bundle in which rupture occurred. Therefore, cold water/steam flow path inside tube bundle was well characterized. Also in this case each couple of runs for middle (A and C) and bottom (B and D) rupture scenarios provided very similar results.

For investigating the dynamic effects of water injection on the surrounding tubes, shell and other tube bundles an overall number of 30 strain gages was implemented in the four tube bundles. Sixteen of which were acquired, in different position and mainly in the bundle of tube rupture, during each test at 10 kHz.

The six tubes, composing the first rank of each bundles, surrounding the tube in which water evaporated upwards, were pressurized at about 16 bar before the start of injection, for acquiring engineering feedback on tube rupture propagation (domino effect) in MYRRHA PHX.

All the test showed a final pressure in these tubes equal to the initial one, highlighting the absence of ruptures and leakage due to SGTR event.

The post-test analysis was performed by SIMMER-IV code. The numerical analysis of bottom rupture scenario showed good capabilities of developed model in computing S100 pressurization time trends, both gradients and maximum value. Moreover, SIMMER-IV code was able to predict the right timing and amplitude of pressure oscillations in the cover gas due to sloshing phenomenon in the main LBE pool. By the code it was possible to highlight the correspondence of the pressure and LBE free level oscillations. SIMMER-IV was also helpful for understanding steam flow path in the tube bundle, separator and main pool. It also provided an estimation of water injected. The code was able to predict low-temperature peaks magnitude, timing and amplitude as well as prolonged cooling effect for the entire duration of the tests in agreement with measured data. The LBE solidification temperature reached during the experiments was also computed properly by SIMME-IV code.

The post-test analysis of middle rupture experiments showed a limit of the model developed. Therefore, the model was improved and then also in this case the numerical simulation was valuable for the phenomena interpretation.

SIMMER-IV also in this configuration showed good capability in temperature evolution prediction inside tube bundle.

The numerical simulation of two tests performed in the same rupture scenario (bottom and middle) provided very consistent results (high numerical repeatability). The small differences between two tests of the same rupture scenario, imposed by initial and boundary conditions, were reflected by SIMMER-IV code in numerical results coherent to the experimental ones.

The SIMMER-IV code at the sight of these applications, considering the geometrical simplification due to its coarse mesh nature, and the calculation cell limitation for reducing time consuming, could be considered a valid resource for facing complex phenomena of multiphase multi-fluid nature as water-LBE interaction occurring in simple geometry and pool configuration. Instead, for more complex geometry, as in middle rupture scenario, the fluid-structure interaction with conductive walls should be more in-depth investigated.

In conclusion, a wide database of high quality data measured was formed. Steam evolution in two SGTR configurations was acquired. Domino effect did not occur in any test. Post-test analysis was able to predict pressure and temperature time trends in agreement with experimental data, providing a contribution to code validation for water-LBE interaction scenario in a large pool facility. The performed analysis provided the awareness that a suitable design of depressurization system (e.g. rupture discs) could allow to address postulated SGTR event in heavy liquid metal nuclear systems with confidence and safety.

Task 4.2 SGTR Bubble Characteristics

Concerning the leak-before-break activities, the application of this concept is relevant for improving the safety of a reactor system. In particular, it decreases the probability of the pipe break event.

Therefore, early detection might be applied, if endorsed as a technically justifiable approach, for making the consequences of a postulated accident acceptable, or even for eliminating the accident (i.e. in this case the SGTR scenario) altogether.

This was the goal of Task 4.2 of the project named “SGTR Bubbles Characteristic”. The idea was to implement an experimental activity, supported by the numerical simulations, to characterize the leak rate and bubbles sizing through typical cracks occurring in the pressurized tubes. Basic tests in LIFUS5/Mod3 facility were carried out to correlate the flow rates of the leakage through selected cracks with signals detected by proper transducers. Different crack sizes and geometries were defined and characterized, while the injection pressure and the temperature were kept constant.

Instrumentation able to promptly detect the presence of a crack in the HX's tube, aiming at preventing further propagation which would possibly lead to a full rupture of the tube, has been developed.

The main achievement was the correlation between the flow rates of the leakage through selected cracks with signals detected by proper transducers, through the implementation of suitable instrumentation, such as acoustic devices to detect the bubbles migration through the free level.

Numerical simulations have been carried out supporting the design of the facility and experimental activity.

Task 4.3 Bubble transport validation

Finally, about the experimental data on drag coefficient of gas bubbles travelling through the liquid LBE, these were gained in Sweden @KTH. To achieve this objective a series of the tests have been performed on newly designed and built bubble transport (BUTRA) facility. The result of these tests provided the terminal rise velocity of the bubbles as function of bubble size and LBE flow conditions. Bubble size and rising velocity measurements have been performed with an appropriate accuracy.

In order to achieve the goal and perform such measurements, the development of gas bubble generation and measurement device (BGMD) to controllably inject single submillimeter bubbles into an LBE poo and the development of submillimeter bubble velocity measurement system (BVMS) needed to evaluate the drag coefficients and validation of drag correlations were performed.

In particular, the BGMD method is based on using low pressure gases through solid cavities. The submillimeter size single bubbles can be created from macro-size cavities. The advantages of this methods are: low cost, reusable components, no capillary channel and orifice-bubble interactions that can lead to bubble sticking to solid surfaces, good accuracy in the estimation of the bubble size. The main current difficulty with the method is detection of the time of the bubble departure. The method has been successfully validated in set of exploratory tests with Wood's metal, LBE and Argon bubbles. At present, the uncertainty of the method is estimated ~3-20% in resulting bubble diameter (mainly due to the visualization technique used in this work). An improvement in visualization such as additional high-speed monitoring of the bubble from a side view can help to reduce the uncertainties. According to theoretical estimations, the accuracy better than 15% can be achieved for 0.1-1 mm size bubbles if a digital pressure gauge is used and cavity length is controlled with uncertainty less than 0.25 mm. The time between consecutive bubbles injections can be decreased by introducing automation of the experimental procedure, e.g. computer controlled temperature (melt freezing, and re-melting) combined with automated drilling. Moreover, fabrication of the cavity in inert or vacuum environment may prevent formation of oxidic layer at the cavity surfaces. It is believed that thin film following the rising bubble up to HLM surface are the fragments of that oxide layer. The origin of small satellite bubbles, which were sometimes observed in the tests, has to be clarified in the future work.

Concerning the BVMS, its operational principle is based on resistivity measurements of the liquid LBE. A small resistivity deviation is expected when gas bubble is passed through the measurement volume. If the bubble is rising in 50-70cm high column filled with liquid LBE it should be sufficient to detect bubble passage at two locations for determination of bubble velocity and, thus, deduce drag coefficient.

After the right implementation of the BGMD technique the development of BVMS, tests have been performed in BUTRA facility at stagnant flow conditions where LBE mass flow rate is at low level of natural circulation in the cylindrical Ø38xH700mm column.

Originally, a series of tests have been planned to be performed in different flow regimes i.e. laminar and turbulent. While the developed single bubble generation and its accuracy (size of the produced bubble) are not affected by the presence of the flow, in contrast, the bubble detection technique and BVMS for a single bubble was further developed, and the BUTRA facility and its Ø38xH700mm LBE column was be redesigned.

The employment of electromagnetic pump to sustain a stable LBE flow was unavoidable. However, bubble drag correlations obtained in tests with stagnant flow conditions might be used to derive corresponding correlations for LBE flow conditions.

WP5: Fuel Safety

Task 5.1 Transient testing of MYRRHA fuel for the determination of the pin failure threshold

Rapid power excursion experiments indicate that a threshold for fuel pin failure during transients for high density fuel and moderate cladding temperatures could exist for a relatively low energy deposition. The expected failure mechanism for these conditions is fuel pellet mechanical interaction with the cladding (PCMI). This type of failure could occur during a rapid transient, in which the power produced in the fuel pin is a multiple of the nominal power for only tens of milliseconds. During certain transient scenarios considered for MYRRHA the reactor power evolves as described above. One of these scenarios is an uncontrolled ejection of a control rod from the core, which results in a positive insertion of reactivity and therefore short, but extensive power excursion.

The development of transient testing methodology started with numerical simulation of experimental set-up and core behavior as well fuel performance simulation. Based on preliminary simulation results, design and fabrication requirements for irradiation device and uranium oxide test fuel segments have been completed. Additional neutron moderator materials are used to modify the neutron spectrum in order to increase the fission power density in the test segments during the transient. The design of the test rig and the TRIGA-ACPR pulse characteristics have been engineered in such a way that the power and temperature conditions simulate MYRRHA specific transients. The irradiation device, developed and licensed at ICN, allows the exposure of three test fuel segments in stagnant molten eutectic to a rapid neutron transient. Auxiliary devices for irradiation device preparation and instrumentation have been also developed. The instrumentation is aimed to monitor irradiation capsule physical parameters (neutron flux and temperatures) during the fast transient.

Twenty uranium dioxide test fuel segments have been manufactured at SCK•CEN to be used in the methodology validation tests. Different ^{235}U enrichments, from 7% to 18.5%, have been used to enlarge the energy deposition domain. The un-irradiated test segments have been designed such that they resemble, as much as possible, a MYRRHA fuel pin at high burnup level. This is achieved by reducing the pellet-cladding gap size and a modification of the cladding cold-worked fraction. Fuel performance simulations have been carried out to evaluate expected values of energy deposition producing fuel failure. A simulation of the thermo-mechanical behavior of a UO_2 fuel test segment during the pulse test has been made with the TRANSURANUS fuel performance code. As a result, the fuel pellets inside the pin rapidly heat-up and expand. The fuel thermal expansion can result in Pellet-Cladding Mechanical Interaction (PCMI) depending on the size of the initial pellet-cladding gap. It is noted that during normal operation the pellet-cladding gap closes slowly due to fuel swelling with accumulating fuel burnup level. During the transient the gap closes rapidly with hard contact between pellet outer surface and the cladding inner surface. Large mechanical stresses and cladding plastic deformation are the result, which is known as Pellet-Cladding Mechanical Interaction or PCMI.

The test fuel pin specifications have been issued according the MYRRHA fuel design. The production route for pellets included a pre-compaction/granulation step (to improve powder flowability) followed by pressing, sintering and grinding.

Nuclear grade fuel cladding tubes were produced according to specification established during past fast reactors program. Cladding section were individually selected to allow for the tightest fit possible with the fuel pellets. Radial gap size below 10 µm were achieved for all pins. Circular orbital TIG-welding was performed on the bottom plug before insertion of fuel. Two welds were performed on the top plug to complete the fuel pin fabrication.

The test fuel pins have been exposed in stagnant liquid Pb-Bi eutectic to a neutron flux transient, leading to a desired energy deposition. Eight pulse experiments have been performed according the test matrix. The total energy deposition per test fuel pin has been estimated by fission products gamma activity measurement. It was proved that the planned energy deposition were reached with a margin below 15%. Energy depositions between 350 J/gUO_2 up to 1100 J/gUO_2 have been produced during the transients.

No cladding failure has been observed, the maximum cladding diameter plastic deformation being 1.44% in the high energy deposition domain. In the low energy deposition domain (350 J/g), the plastic deformation is practically negligible.

Nondestructive post-irradiation examination methods have been adapted for energy deposition and fuel pin cladding plastic deformation measurements. Suitable radiation shielding solutions have been applied, without using the hot cell facilities, to avoid fuel pin contamination.

Neutron parameters instrumentation and the fast data acquisition system allowed the pulse relative magnitude knowledge (neutron detector signal area and integral flux on the activation foil detector) and the transient shape.

The temperatures measurement during the fast transient was the biggest challenge of the instrumentation. The LBE temperature measurements using n type sealed thermocouples (TC) show reliable results. As regarding the cladding temperature measurement, the sealed TC's have been excluded. The exposed TC solutions tested during the experimental program did not confirm a reliable instrumentation solution. The cladding temperature measurement in liquid eutectic in fast transient regime remain an open subject for future developments.

The test results support the test methodology and experimental setup validation:

- It was proved that a desired energy deposition in the test fuel pin can be achieved by pulse configuration computation.
- The experiment instrumentation allows the knowledge of neutronic parameters as well the LBE temperature behavior during the transient and after that.
- The technical solutions adopted for internal capsule closure, including the thermocouples passage ensure a safe closure for the high temperature liquid eutectic and test fuel pins.
- Nondestructive post-irradiation examination facilities have been developed on site for energy deposition and fuel pin deformation.

Based on the fuel performance code simulations and test results of the UO_2 fuel segments a neutronic analysis has been made to assess the required enrichment in Pu necessary to deliver the desired energy deposition of 300 – 1000 J/gmox during the transient test. Specifications for MOX fuel test segments, compatible with the irradiation facilities, have been completed.

SCK•CEN has invested in a new infrastructure to fabricate test fuel pins in the frame of MAXSIMA project. The MOX test fuel fabrication poses a number of challenges that have been reviewed. There exits enough knowledge and know-how to handle each of them and produce fuel.

Task 5.2 Fuel coolant compatibility

For the assessment of severe accidents involving a fuel clad failure it is important to address a possible interaction between the fuel and the coolant. Targeted temperatures are between 1000 to 1700 ^{0}C, when cladding material will melt.

Three type of experiments were carried out: UO_2 with cladding and LBE (1), 15-15Ti with and without LBE (2) and 15-15Ti with UO_2 and MOX with LBE and 15-15Ti (3). The most significant results are:

1. The results from the experiment run at 1000°C show similarities with a preparatory run at the same temperature in that the LBE is nowhere to be found in the setup. The final conclusion is that the LBE must have escaped in gaseous form through small openings in the alumina crucible setup. From the SEM/EDX-data few things can be concluded: there is Pb present on both sides of the lid indicating that there has been gaseous lead that has precipitated. Dramatic changes have occurred in the cladding material. The Ti has congregated into grains consisting of titanium carbides instead of being distributed throughout the material.
2. For the setup with just 15-15Ti being heated to 1000°C for 50 hours some changes in the material were observed, however none as dramatic as what was seen originally. For the setup including UO2 but not LBE the changes seen in the experiment including all three parts were observed again. This indicates that the culprit is the UO2 and more specifically carbon impurities in the fuel matrix in combination with the fuel being hyperstochiometric with respect to O.
3. The experiment with MOX posed several problems due to the scarcity of pellets and the high radioactivity. The only assessment made was by observation of the pellet surface which seemed to be visually intact and having maintained the same properties as before the experiment.

Task 5.3 Conceptual design study for experiment in the IGR reactor of NNC (Kazakhstan)
The conceptual design study carried out by NNC is based on the requirements for experimental analysis of MYRRHA fast reactor fuel pins under accident conditions. The study started with a review regarding the IGR reactor pulsing capacities, experimental facilities as well post experiment characterization capacities available on site.

The conceptual design study for experiments with MYRRHA fuel in the IGR pulsed reactor completed by NNC and the results of calculation studies demonstrate the possibility of ensuring the required test conditions for the pellet fragmentation experiment and fuel coolant chemical interaction (FCCI) experiment.

The experiments are foreseen to obtain the representative data of key-licensing phenomena for the MYRRHA fuel (pellets fragmentation and fuel-coolant interaction).

The most important outcomes of the study are:

- Determination of the optimal in-core experimental device design
- Calculations of neutron-physical characteristics in explanation of the construction of the experimental device
- Calculation of the experiment parameters to study fragmentation of fuel pellets and fuel coolant interaction

<u>WP6: Enhanced Safety by Design for HLM reactors</u>
Task 6.1 Safety systems (D6.1)
The work performed in WP6 originates from the fact that the risk of primary coolant solidification is an important issue for HLM-cooled reactors and a peculiarity for this type of power plants. This issue is normally not considered a safety concern, since it is not expected to lead directly to core damage or release of radioactive material. It is in any case advisable for the sake of investment protection that a reactor's DHR systems be endowed with a provision—in the form of a specific process sub-loop or of the underlying physical properties of the working fluid(s)—that is able at least to reduce the heat removed from the system when the primary coolant reaches its cold shutdown temperature. This is especially true if an incident leading to a reactor scram and to the loss of the primary coolant heating mechanism occurs at the early stages of a fuel cycle, when the decay heat is zero or very low.

Starting from the above basic consideration the work was structured with a first phase consisting of a complete overview of different HLM-cooled reactors and their DHR systems as developed in previous EC funded projects in order to analyze their performance and features in terms of systems preventing coolant freezing. Each DHR system, described in detail, was analysed highlighting its major advantages and disadvantages. The safety standards used to assess the performance of such systems are those outlined in the goals of Generation III, III+ and IV nuclear power plants—namely, passivity, compliance with the defence in depth strategy and with the common-cause (i.e., independence, diversity) and single-failure (i.e., redundancy) criteria. Moreover, the importance of primary coolant solidification prevention—an issue peculiar to HLM-cooled reactors—has been stressed.

Task 6.2 Development of innovative passive safety systems
Of the five DHR systems considered, only EFIT's Direct Reactor Cooling system (developed in the frame of EC funded EUROTRANS project) satisfies all the requested features.

In particular, the system has the appealing property of inherently preventing primary coolant solidification, for as long as the reactor's decay heat is greater than the heat losses.

However, its use of diathermic oil as the main working fluid casts some doubts on its applicability, mainly due to its chemical reactivity in air. In any case the characteristics of EFIT's DRC system are considered very attractive and have been assumed as a basis for further development of an enhanced safety system for HLM-cooled reactors to be developed in the following phase of the project. Such basic underlying features of the systems can be summarized as follows:

(a) a primary/secondary coolant heat exchanger immersed in the primary coolant pool, which may or may not coincide with the secondary coolant system;
(b) an air- or water-cooled condenser, dissipating heat to the ultimate heat sink;
(c) a non-condensable gas tank connected to the condenser lower header, which provides fully passive heat exchange degradation by reducing the heat exchange surface whenever a depressurisation—resulting from excessive heat removal—draws non-condensable gas into the condenser tube bundle.

The concept, already developed previously, was successfully adapted to the MYRRHA and ALFRED reactor configurations. In both cases, it was shown through simulations that on the one hand the DHR main function—i.e., removing the decay heat to preserve fuel integrity—is satisfactorily fulfilled by the system, and on the other that the system is able to prevent primary coolant solidification with a margin of more than 20°C for transient times greater than one week after the initiating event.

The demonstration of such system capabilities can be considered as a major breakthrough in terms of scientific and technological innovation, opening a new scenario in the development and use of HLM cooled reactors.

Task 6.3 MYRRHA containment analysis

Another task of the work package was the modelling of the MYRRHA Primary Containment (using MELCOR and CONTAIN codes). Mass and Energy (M&E) releases from the secondary system were calculated (using the RELAP5 code) and provided as input to the two containment simulation codes to carry out a comparison of the main thermal-hydraulic parameters for the identified scenarios. For one scenario (SGTR during maintenance), a conservative estimate of the source term was provided.

Both MELCOR and CONTAIN showed good agreement in predicting the pressure response of the containment to the postulated transients; primary containment temperature distribution showed slight discrepancies between the predictions of the two codes, even though the predicted trends are in good agreement. The assessment of the polonium source term in gaseous form highlighted the strong need for further experimental characterization of irradiated LBE/water interaction and its implications on polonium releases.

Also this task and goal of the work package fulfilled its original objectives, showing the availability of containment system tools to simulate containment transients and by driving future research activities to the specific needs of HLM reactors.

WP7 Education and Training

One objective of this work package is to train young scientists and students. The goal is to give them functional skills and insights pertaining to safe operation of HLM cooled reactors. For this purpose, a virtual simulator environment was developed, in which students can train management of incidents and accidents in HLM cooled systems. Prior to the simulator training the students participated in seminars where lectures on the safety of heavy liquid metal systems will form the basis for student projects on the topic. The work package is also in charge for dissemination of knowledge obtained in the project.

The first MAXSIMA workshop was successfully implemented in Karlsruhe on October 7-10, 2014 in Karlsruhe, including 5 keynote lectures and 41 technical. The workshop was organized jointly with the SEARCH project (GA295736) and external participation from the international community working on heavy-liquid-metal cooled systems was promoted. The following key topics were covered: core thermal-hydraulic and core components, steam generator and cooling safety, coolant chemistry control and heavy-liquid metal corrosion, fuel, fuel safety and safety analysis. With 85 registered participants, the workshop succeeded to enhance the information exchange within the international community working on heavy-liquid-metal systems. Proceedings can be downloaded from http://www.iket.kit.edu/590.php

The MAXSIMA Lecture Series "Lecture notes on safety of HLM systems" was held at the Karlsruhe Institute of Technology from March 24-27, 2015. The main goal was to provide comprehensive background knowledge on safety of HLM systems. Also safety aspects of light water reactors were covered. In total, fourteen lecture notes were presented.

The second MAXSIMA workshop was organized by KTH in Jukkasjärvi (The Icehotel) in Sweden on February 23-26, 2016. In total, nineteen presentations were displayed. During this workshop, a first beta version of an application for mobile platforms, the so-called 'MAXSIMA Simulator', which simulates the transient behaviour of Generation-IV lead-cooled fast reactors for safety-informed pre-design and didactical purposes, was presented. Other topics covered: core thermal-hydraulics and core components, steam generator and cooling safety, coolant chemistry control, fuel, fuel safety and safety analysis.