# EUROPEAN OUTLOOK FOR LMFR THERMAL HYDRAULICS

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## ABSTRACT

Liquid metal cooled reactors are envisaged to play an important role in the future of nuclear energy production because of their possibility to use natural resources efficiently and to reduce the volume and lifetime of nuclear waste. Typically, sodium and lead(-alloys) are envisaged as coolants for such reactors. Obviously, in the development of these reactors, thermal-hydraulics is recognized as a key (safety) challenge.

This paper will present recent developments and initiatives with respect to liquid metal cooled reactor thermal-hydraulics. The initiatives to be discussed encompass liquid metal heat transfer, fuel assembly thermal-hydraulics, pool thermal-hydraulics, and system thermal-hydraulics. With respect to liquid metal heat transfer, the purpose is to start from the most promising routes and develop and validate a model which can simultaneously deal with different flow regimes (natural, mixed, and forced convection).

Considering fuel assemblies, the aim will be to take the next steps in liquid metal fast reactor fuel assembly modelling, mainly focusing on validation and evaluation of accidental conditions. With respect to pool thermal-hydraulics, the purpose will be to develop and validate sufficiently accurate methods to model the coolant behavior in a liquid metal reactor pool. When considering the system scale, similarly, the purpose will be to validate and improve system thermal-hydraulics models, but also to further develop and validate multi-scale approaches.

**KEYWORDS**
Liquid Metal, Fast Reactor, Thermal-hydraulics

## 1. INTRODUCTION

Nuclear power plays an important role in power generation and produces about 11% of the total electricity worldwide. The rapidly growing energy demand suggests an important role for nuclear power in the future energy supply, as for example denoted in the projections of the World Energy Outlook 2014 [1]. The most recent IEA/NEA nuclear technology roadmap [2] predicts in its so-called 2DS scenario a stable nuclear energy production in Europe in the coming decades up to 2050. In Europe, the European Commission presented the Vision Report of the Sustainable Nuclear Energy Technology Platform [3] for the role of nuclear fission energy to the European transition towards a low-carbon energy mix by 2050. Within this Vision Report, a large role is attributed to the deployment of fast reactors. In Europe, the European Sustainable Nuclear Industrial Initiative (ESNII) [4] has indicated that the preferred option is the sodium cooled fast reactor with the lead cooled fast reactor as the near-term back-up. This clearly shows the importance of liquid metals in the development of future nuclear energy technologies.

Within the European playing field, a number of liquid metal cooled reactors are under consideration today. ASTRID is the Sodium cooled Fast Reactor (SFR) prototype. The ASTRID industrial prototype shall contribute as a key component to future nuclear systems development, with a first objective to confirm long term innovation options both for the development of the advanced SFR technology, but also for the fuel cycle and waste management [5]. ALFRED is a program targeting the construction of a Lead cooled Fast Reactor (LFR) demonstrator in Central / Eastern Europe [6]. Currently, Romania is proposed as a host country for ALFRED. MYRRHA is a multipurpose fast neutron spectrum irradiation facility proposed to operate as a European large research infrastructure, and to serve as experimental pilot plant for the lead technology [7]. Furthermore, MYRRHA is proposed to serve, as a technological system for waste transmutation demonstration, and as an irradiation facility for material and fuel in support of the liquid metal fast reactor systems. Finally, SEALER is a small lead cooled reactor which is currently under development by the Swedish company LeadCold [8]. This reactor ensures reliable and safe production of power for sites where evacuation can never be an option. This holds for example for a number of arctic communities in Canada, the US, and on Greenland which remain disconnected from national power grids and road networks. Except for the SEALER concept, the reactors under consideration have been described in IAEA [9] and the IAEA booklet on the status of fast reactor designs and concepts [10].

Thermal-hydraulics is recognized as one of the key scientific subjects in the design and safety analysis of liquid metal cooled reactors. In recent years, many related issues have been identified. Summaries can be found in open literature, e.g. [11], [12], [13]. However, in addition, the summary report of the IAEA workshop on priorities in Modelling and Simulation for Fast Neutron Reactors [14] gives a decent summary of the international consensus on the issues. As a result of this workshop, a priority list was assembled by the IAEA international experts. For thermal-hydraulics, the following short-term priorities were identified:

- Liquid Metal Turbulence Heat Transfer;
  - Further validation of the promising approaches using geometrically more complex cases e.g. wire wrapped fuel assemblies, spaced grid fuel assemblies, spiral tubes heat exchangers and flows not bounded by walls e.g. jet flows
  - Further validation of the promising approaches using development of approaches allowing application in all flow regimes simultaneously, and high resolution simulations for realistic Reynolds numbers and geometries to be used as reference data for validation.
- Fuel Assemblies
  - Validation of flow blockages and inter wrapper flow and heat transfer
  - Further development of complete core approaches and validation of low resolution CFD;
- Plenum Thermal-hydraulics in liquid metal cooled reactors;
  - Validation of Flow and heat transfer, oxygen dispersion in pool-type heavy liquid metal cooled reactors, thermal stratification and instability, thermal stripping, and freezing in liquid metals cooled reactors
- System Thermal-hydraulics;
  - Coupling system thermal-hydraulic codes with CFD codes and the validation of these methodologies, including transition from forced to natural convection.
  - Improvement of thermal-hydraulic system codes, including the modelling of 3D effects.

In Europe, once the challenges were identified, feedback was obtained from the respective reactor designers to prioritize the identified topics. According to the reactor designers, priorities should be set to the development of appropriate V&V methodologies, system thermal-hydraulics and multi-scale thermal hydraulics, by coupling system codes with more detailed CFD approaches.

This process has formed the base of two new large activities sponsored by the European Commission. Firstly, the SESAME project which deals with <u>S</u>imulations and <u>E</u>xperiments for the <u>S</u>afety <u>A</u>ssessment of <u>Me</u>tal cooled reactors. This project solely deals with thermal hydraulics issues. Complementary to this project, a large part of the MYRTE (<u>MY</u>RRHA <u>R</u>esearch and <u>T</u>ransmutation <u>E</u>ndeavour) project on the further development of the MYRRHA reactor in Belgium, is devoted to thermal hydraulics. Where SESAME has a focus on pre-normative, fundamental, safety-related, generic challenges, the MYRTE project deals with MYRRHA specific challenges.

Within both projects, (new) analytical and simulation methods will be validated with reference data. If possible and affordable, such reference data will be based on experimental results. However, in some cases it is not possible to create experimental data, or the experimental data will not provide all reference data needed to validate the analytical or simulation method, e.g. because the number of measurement positions in an experiment is limited or because some flow areas are not accessible for measurements. In such cases, the experimental data is complemented or replaced by high fidelity simulation data (typically Direct Numerical Simulations (DNS) or Large Eddy Simulations (LES)).

This paper will present the future European activities in the field of liquid metal thermal hydraulics within the projects SESAME and MYRTE which started in 2015 with a four years duration. The next section will discuss the activities in the field of liquid metal heat transfer. After that, core, pool, and system thermal hydraulics will be discussed in sections 3, 4, and 5. Finally, section 6 will provide a summary.

## 2. LIQUID METAL HEAT TRANSFER

A fundamental issue for the evaluation of liquid metal reactors is the modelling of the turbulent heat flux and thermal fluctuations over the complete range from natural, mixed and convection to forced convection regimes. Current engineering tools apply statistical turbulence closures and adopt the concept of the turbulent Prandtl number based on the Reynolds analogy. This analogy is valid mainly for forced convective flows with a Prandtl number of order of unity. In the particular case of liquid metal, where the Prandtl number is much smaller than 1, the turbulent Prandtl number concept is not applicable, and robust engineering turbulence models are needed. Thus, a model is required which can deal with all flow regimes simultaneously in liquid metal flows. In [15], some promising routes for improvement have been identified and tested on relevant available geometrically simple test cases. Within Europe, consensus was achieved that further development of models should be limited. Main focus should be on the extension of the validation base for mixed and natural convection regimes and for geometrically complex cases. To this purpose, the validation base will be extended by a mixing jets, a flow separation, a mixed convection, and a rod bundle test case. In addition, a model which performs well in all flow regimes simultaneously would provide much added value. Within the new European projects SESAME and MYRTE, some further development of the promising approaches is foreseen. Emphasis is put on the creation of new relevant missing reference data, on more robust testing of the models in geometrically more complex cases, and on implementation of these models in different codes in order to check the code dependency of the models. Some of the selected approaches will provide a more pragmatic computational alternative for the expensive, and for industrial cases often not feasible, high fidelity LES to predict thermal fluctuations for all types of fluid. This work will therefore probably have a positive impact also on the simulation of light water reactors.

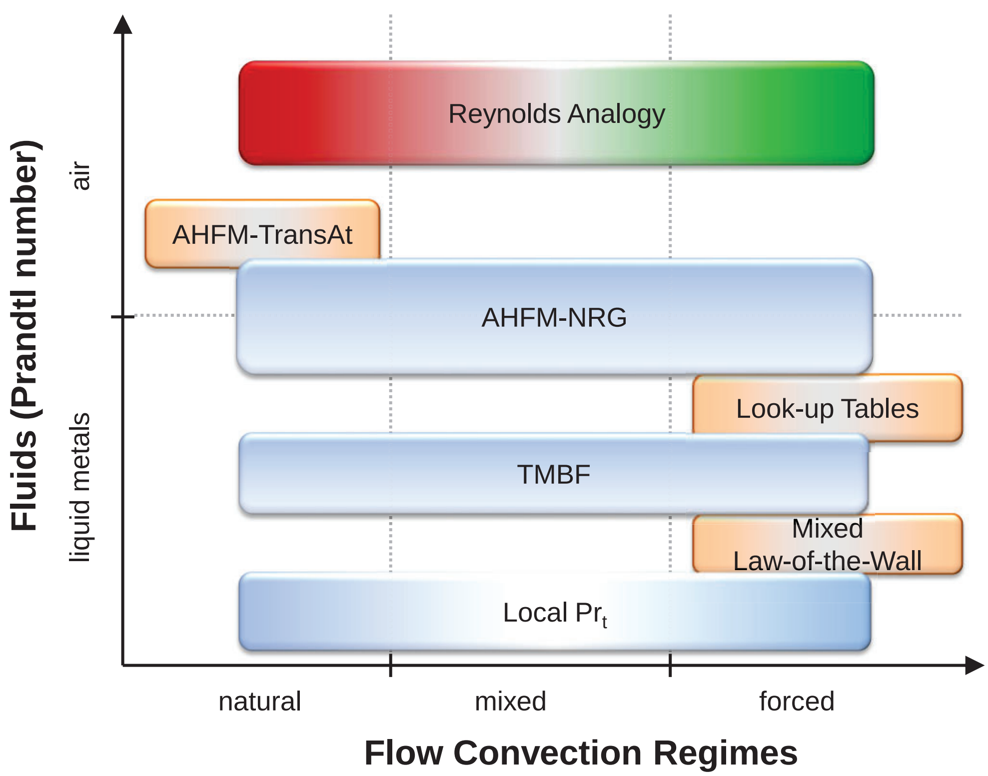


**Figure 1. Liquid metal models identified in [15, 16] and their applicability for different fluids and different flow regimes. The models indicated in blue will be further developed and tested in the SESAME and MYRTE projects.**

The following promising models identified in [15] will be further developed and/or implemented and validated in the SESAME and MYRTE projects:

- AHFM-NRG in the STAR-CCM+, OpenFOAM, and NEK5000 codes
- Local turbulent Prandtl number model in the OpenFOAM code
- The Turbulence Model for Buoyant Flows (TMBF) [16] in TransAT code

## 3. CORE THERMAL HYDRAULICS

Most liquid metal cooled reactors employ wire wraps as a spacer in the fuel assemblies. Although experiments are being carried out in the European context on wire-wrapped fuel assemblies and to a lesser extent on fuel assemblies with grid spacers, the data to be retrieved from those experiments will be limited to the thermal field and will not include detailed information on the flow field. To derive reference data for the flow field in wire wrapped fuel assemblies, ideally a combination of experimental data based on a Matched Index of Refraction (MIR) technique and reference high fidelity numerical simulations is required (figure 2). This is foreseen in the SESAME project which will provide both. Firstly, a reference experiment is foreseen in a 7-pin rod bundle using water as a simulant fluid to obtain validation data for the flow field. In addition to this, numerical simulation data will be generated for a 217 pins and an infinite number of pins rod bundle. These data will allow closing the gap in the validation process of wire wrapped fuel assemblies.

Although blockages at the inlet of the fuel assembly have already been studied, a blockage scenario which has not been studied yet is a scenario in which solid particles flowing into the core can block one or more sub-channels. As they do not affect substantially the outlet conditions with respect to temperature and pressure, they can go unnoticed by the global instrumentation. However, they may lead to local heat transfer degradation and the corresponding increase of the cladding temperature. A thorough understanding of the formation of such blockages is therefore needed. To that purpose, water experiments will be performed and a simulation approach will be developed to model the accumulation of particles in the MYRTE project.

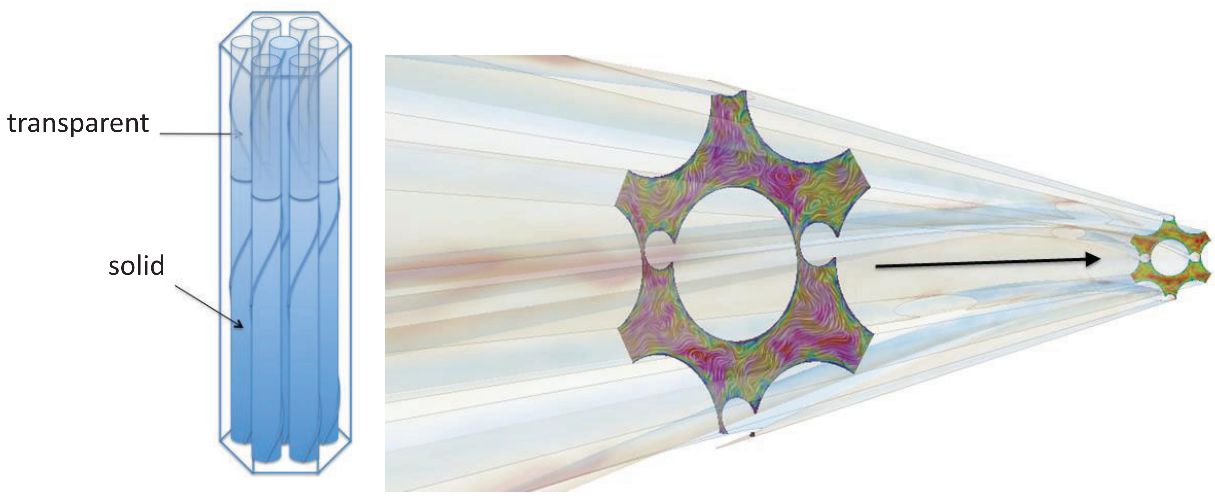


**Figure 2. Sketch of a 7 pin wire wrapped bundle MIR experiment (TUD) and instantaneous snapshot of initial test simulation to obtain high fidelity numerical reference data for an infinite pin bundle (NRG).**

Due to the design specific nature, validation of fuel assemblies using grid spacers is limited. Many publications can be found on grid spacers for water cooled reactors. However, specific geometric details on the grid spacer design are revealed seldom due to IPR issues. Data on the performance of grid spacers for liquid metal cooled reactors are largely missing, although some work has been performed and reported studying the coolant behavior in a lead cooled fuel assembly employing grid spacers [17]. The grid spacer employed in this experiment is not representative for a grid spacer to be employed in a real reactor. Therefore, the SESAME project will provide these missing data by performing experiments in a liquid metal rod bundle. Such experiments will be performed for grid spacers without blockages and with blockages.

Evaluation on the influence of the inter-wrapper flow is an important topic as a step towards a complete core simulation. Some experimental work has been performed in the past on this topic in Japan [18]. However, as experience, both experimentally and computationally, is limited within Europe, it is foreseen to conduct a new experiment in which three reduced scale 7-pin fuel assemblies including their inter-wrapper space are employed as depicted in figure 3. This experiment will provide unique data for model development and validation. As multiple fuel assemblies need to be modelled to evaluate this, reduced or low resolution (coarse grid) CFD approaches are foreseen from a computational point of view [19-21]. Keeping in mind that the end goal will be to increase the modelling capabilities of a complete core, also multi-scale approaches will be tested by e.g., combining sub-channel codes with CFD codes.

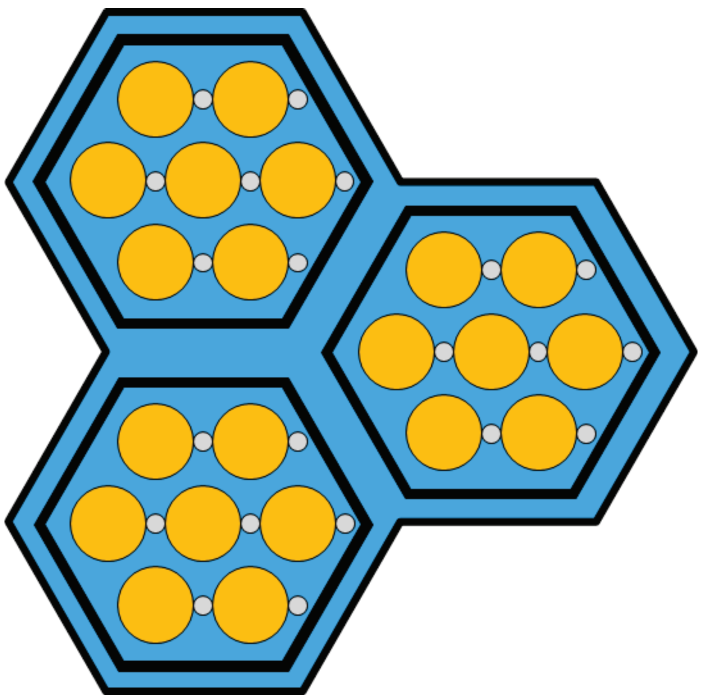

**Figure 3. Outline of the inter-wrapper flow experiment (KIT).**

As a final step and by lack of experimental data, a code to code comparison will be performed employing methods developed and validated using the inter-wrapper flow experiment, to evaluate the applicability for simulation of a complete core. To this purpose, a reference core will be selected which will serve as a basis for this code to code benchmark. Similar approaches are being reported in literature recently for application to light water reactors.

Simulation experience and validation of fluid structure interaction and specifically flow induced vibrations (e.g. in fuel assemblies) are still limited. Some attempts on flow induced vibration modelling are reported in the open literature. However, these attempts are mostly related to rod bundles in cross flow, as taking place in for example heat exchangers and steam generators [22, 23]. In contrast, the flow in a typical nuclear fuel assembly is in axial direction parallel to the rod bundle. Nevertheless, it has been reported, even for water cooled reactors, that these flows may induce vibrations [24]. Because of the high density of liquid metals, such effects are expected to be more pronounced and more important in liquid metal fast reactors. Attempts to analyze such situations are reported in [25,26]. To analyze this, experimental data for validation are missing. Therefore, a fundamental experiment in a seven pin rod bundle is foreseen employing wire wraps as a spacer. This experiment uses water as simulant fluid which will allow obtaining missing validation data both on the structure and on the fluid motion which will support the development and validation of numerical approaches. On the other hand, liquid metal experiments are being performed related to the MYRRHA fuel assemblies. Such experiments are being performed from a small scale (adding to the validation base) up to the scale of a complete MYRRHA fuel assembly (providing important information on the real scale application and the interaction of multiple vibrating pins).

## 4. POOL THERMAL HYDRAULICS

Although it is recognized that pool thermal-hydraulics as such is highly design-dependent, the development and validation of modelling approaches for pool thermal-hydraulics is not. As explained in [27] many experiments typically employing water as simulant fluid have been performed in the past to study the flow behavior in liquid metal reactor pools. However, at that time, CFD capabilities were limited, and such experiments were not equipped to provide the detailed data which is needed for validation of a CFD approach. In relation to the MYRRHA design, a water mock-up is being used to provide first confirmations on the flow phenomena in the MYRRHA pool [28]. Obviously, this facility is also useful in the validation of numerical approaches. Lessons learned from the ongoing efforts will serve as input for simulations of liquid metal mock-ups. Limited data is available from facilities employing liquid metals up to now. In order to improve the validation base, liquid metal experiments are foreseen at different scales. Firstly, an experiment in the TALL-3D facility [29] which will include a small pool in which thermal stratification and mixing phenomena can be studied. In a final step, experiments are foreseen in the larger scale CIRCE [30] and ESCAPE [31] facilities which will be instrumented such that relevant data for thermal stratification and flow patterns can be extracted. In fact, the experiments in ESCAPE (figure 4) should in addition demonstrate the decay heat removal through natural circulation in MYRRHA is feasible. The numerical approaches to be developed can be divided in three variants. Firstly, RANS CFD approaches will be developed and validated using the different steps of available validation data. This will be complemented by further development and validation of a fast approach like the grid free method [32]. Finally, an STH model of E-SCAPE will be developed to identify the possibilities and limitations of this approach.

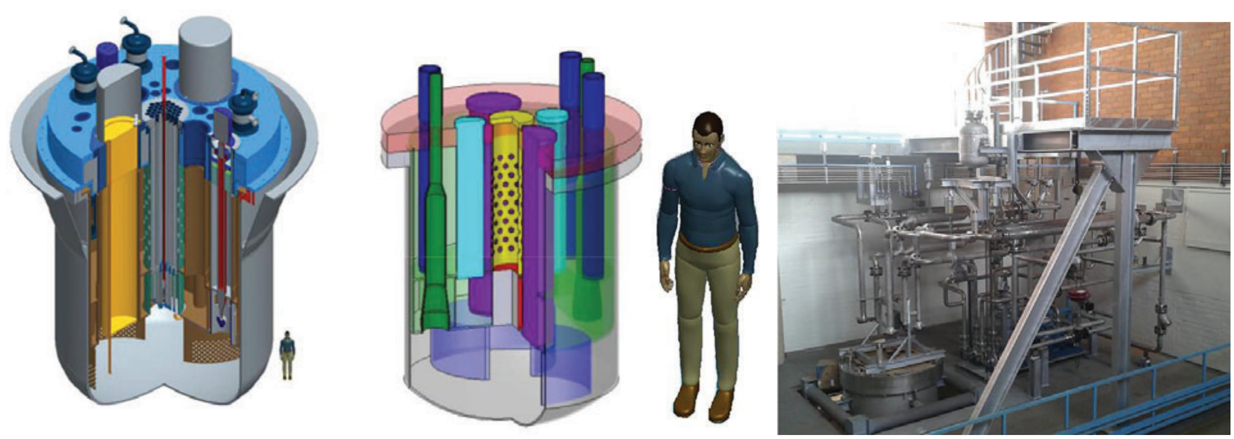

**Figure 4. The MYRRHA reactor and the scaled down ESCAPE facility (SCK●CEN).**

Freezing of the coolant in liquid metal cooled reactors is identified as a potential safety issue. In liquid metal reactors, the highest risk of coolant freezing is thought to occur during shut-down, maintenance, refueling, and whenever an overcooling from the secondary system or the (passive) emergency cooling system is established. However, solidification might be expected even in normal operation in case there will be any flow stagnation zones in the vicinity of cooled surfaces. The local solidification of the liquid metal will affect the natural convection pattern in the pool with possible impairment of the core cooling. Solidification models are often used in the automotive and/or metal molding industries. However, applicability of such models to the solidification of metal in a nuclear reactor, where interaction of solidification interface with local flow conditions is especially important, has not been shown. Therefore, small scale fundamental experiments are scheduled which will provide useful data for model development and validation.

## 5. SYSTEM THERMAL HYDRAULICS

Thermal-hydraulics simulation techniques are essential in order to simulate the behavior of a complete liquid metal fast reactor system, i.e. primary, secondary, and/or energy conversion system. Traditionally, the analysis of such system behavior is performed using system thermal-hydraulics codes. Mostly, such thermal-hydraulic system analyses are validated using integral design specific experiments or reactor data from prototype, test, or demonstration reactors. Specifically for the purpose of application to liquid metal cooled reactors, these codes need to be updated with state-of-the-art algorithms, models and correlations. Furthermore, the validation base should be extended in order to confirm the applicability of such codes for safety analyses. Therefore, the capabilities of the existing system codes to describe a reactor transient involving complex 3D effects are evaluated and validated.

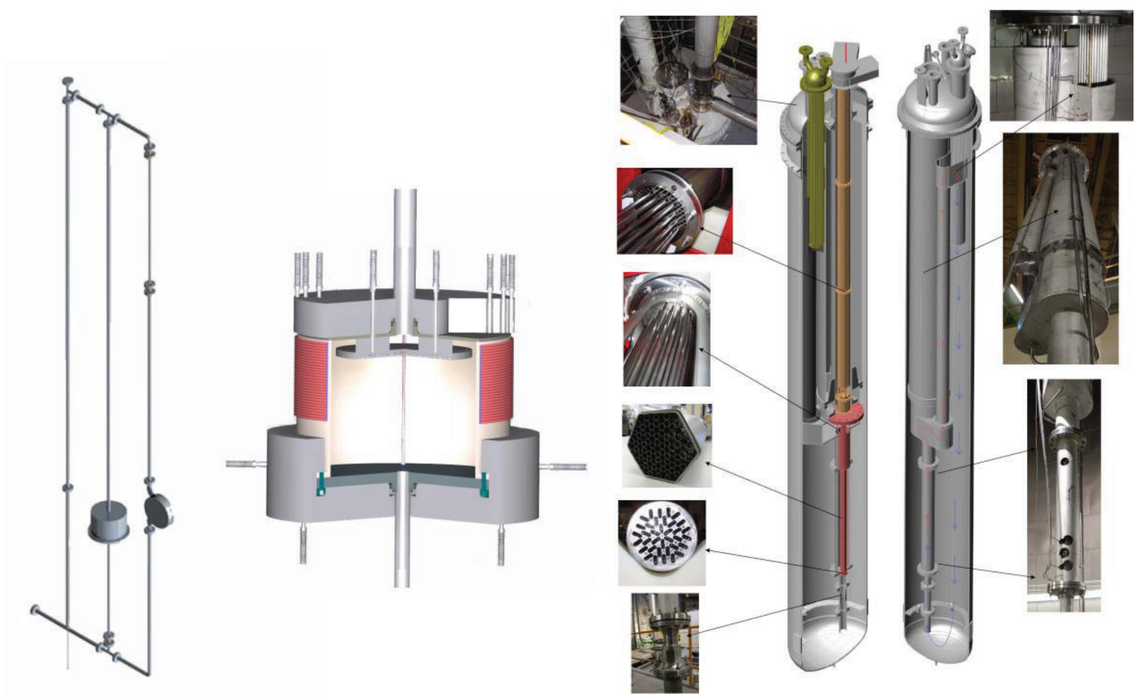

**Figure 5. Loop and 3-dimensional test section of the TALL-3D loop (left) and an overview of the CIRCE test section (right).**

In recent years, the traditional approach of using system thermal-hydraulic codes is supplemented with new multi-scale approaches in which system thermal hydraulics codes are coupled to detailed three dimensional CFD approaches. Development of such approaches is also applicable to light water reactors

[33]. In recent years, such approaches were also developed for liquid metal cooled reactors [34]. However, only a limited set of validation data is available up to now and basically limited to detailed experiments in the TALL-3D test loop, and to the Phénix natural circulation tests. However, the latter data are related to a real reactor in operation. Therefore, the measurement possibilities were limited.

Within Europe, the purpose in the upcoming years will be to extend the validation base by providing reference data at different levels. Furthermore, in addition to the existing numerical development teams, new development teams will contribute extending the experience base in multi-scale modelling. The first level of validation data will be provided again from the Phénix reactor end of life tests. This time, data from the dissymmetric test will be made available. This data will allow validation of the three dimensional effects to a much larger extent than the natural circulation test data which were previously used [35]. This real reactor data will be supplemented by data from four experimental facilities. Firstly, the CIRCE pool will be utilized (figure 5), which is the largest heavy liquid metal pool facility in the world. This pool will allow much more detailed measurements than the Phénix reactor. Furthermore, because of its large size, characteristic flow behavior may be expected to take place in this pool. However, even if this pool allows much more detailed data compared to the Phénix experiments, the measurement data still will have limitations. Therefore, more detailed and fundamental data will be obtained from dissymmetric tests in the TALL-3D facility which was designed for validation of coupled STH and CFD codes (figure 5). Also data from the E-SCAPE facility in nominal and asymmetric operational conditions will be used to determine the best coupling strategy of STH and CFD codes, identifying areas where 3D effects are dominant. Finally, data from the NACIE-UP facility will focus on the multi-scale coupling of the behavior in the fuel assemblies and the loop system.

## 6. SUMMARY

The European outlook for liquid metal fast reactor thermal hydraulics is summarized below in figure 6. This figure shows based on the different components of a typical liquid metal fast reactor which activities have been planned in the SESAME and MYRTE projects which started in 2015.

Within both projects, (new) analytical and simulation methods will be validated with reference data. If possible and affordable, such reference data will be based on experimental results. However, in some cases it is not possible to create experimental data, or the experimental data will not provide all reference data needed to validate the analytical or simulation method, e.g. because the number of measurement positions in an experiment is limited or because some flow areas are not accessible for measurements. In such cases, the experimental data is complemented or replaced by high fidelity simulation data (typically DNS or LES).

As such, within these projects, experiments, high fidelity reference simulations and pragmatic engineering simulation will go hand-in-hand providing not only the international liquid metal fast reactor designers, but also the light water community with valuable new reference data and modelling approaches.

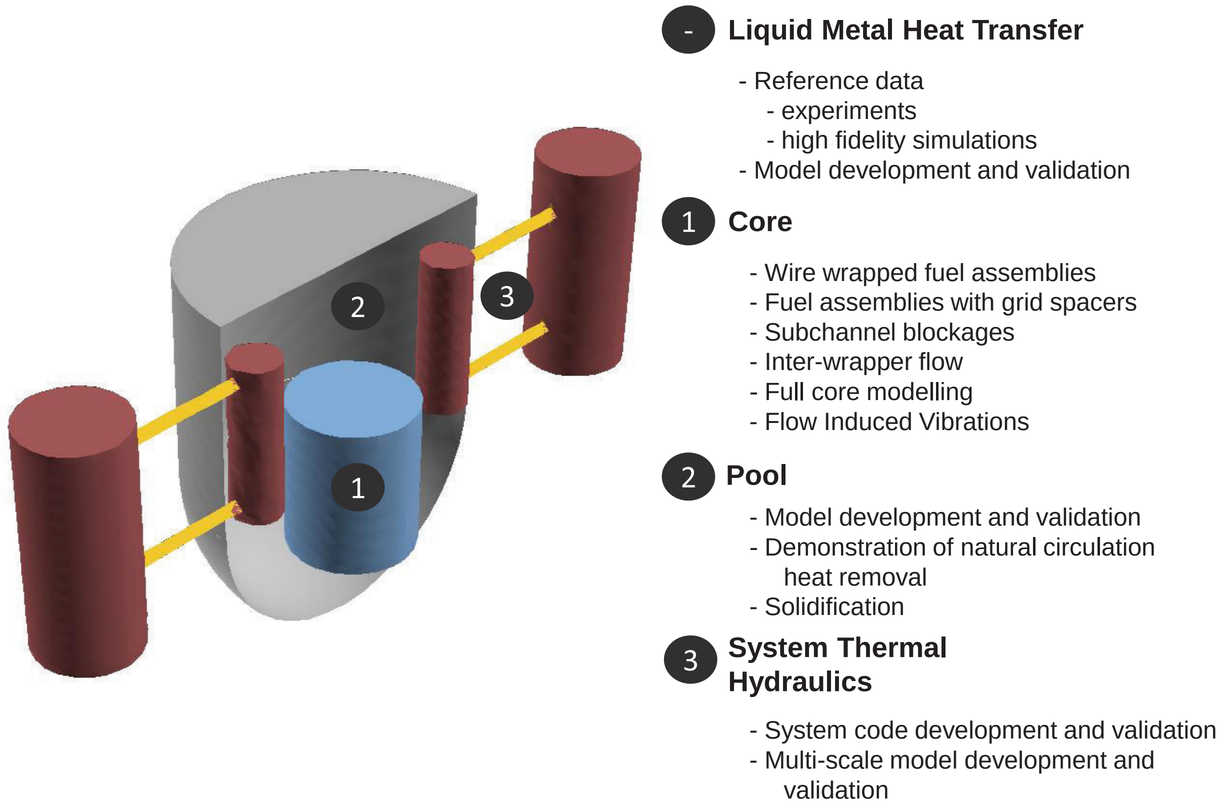


**Figure 6. Summary of the near future liquid metal fast thermal hydraulics activities in Europe.**

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**Figure 7. European projects SESAME and MYRTE.**

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