# ALFRED High priority R&D Needs

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**Abstract**

The identification of the Research and Development (R&D) priorities and needs is a necessary step to complete the development, up to the qualification and demonstration, of the solutions envisaged for Lead Fast Reactor (LFR) technology. In particular, this is of paramount importance to allow the design, licensing and construction of industrial systems. In the present paper, starting from the key scientific aspects (including lead chemistry monitoring and control, thermal hydraulics in large pools, components qualification and integral system operation), the necessary experimental activities are identified in support of the Advanced Lead Fast Reactor European Demonstrator (ALFRED) safety demonstration program. According to the identified R&D needs and considering the current status of the ALFRED conceptual design as well as of the research infrastructures’ implementation in Romania, a prioritization of activities is also proposed, which takes into account also the actual worldwide level of knowledge on the LFR technology. Finally, by taking advantage of the experimental activities, the needs for Verification and Validation (V&V) of the computational tools to be used for safety demonstration are also aligned.

## INTRODUCTION

The aim of Generation IV (GEN-IV) nuclear system development is to excel in safety and reliability, having a very low likelihood and degree of reactor core damage and finally to eliminate the need for offsite emergency response. There is a reasonable expectation to demonstrate that Lead Fast Reactors (LFRs) are able to cover these two fundamental design objectives. Indeed, even if some drawbacks exist – like the presence of high corrosion/erosion effects and the high freezing temperature of Lead with respect to Sodium –, the choice of lead as primary coolant has a number of positive aspects with regard to safety and in simplifying the design [1][2], including but not being limited to the absence of exothermic reactions between lead and water, the characteristics to have high boiling point and heat of evaporation as well as a large retention capability of volatile radionuclides (i.e. I, Cs), and its self-shielding capability against radiation.

The goal of any LFR design is to gain advantage from the use of this innovative coolant. Notwithstanding this, a comprehensive R&D program is necessary because of:

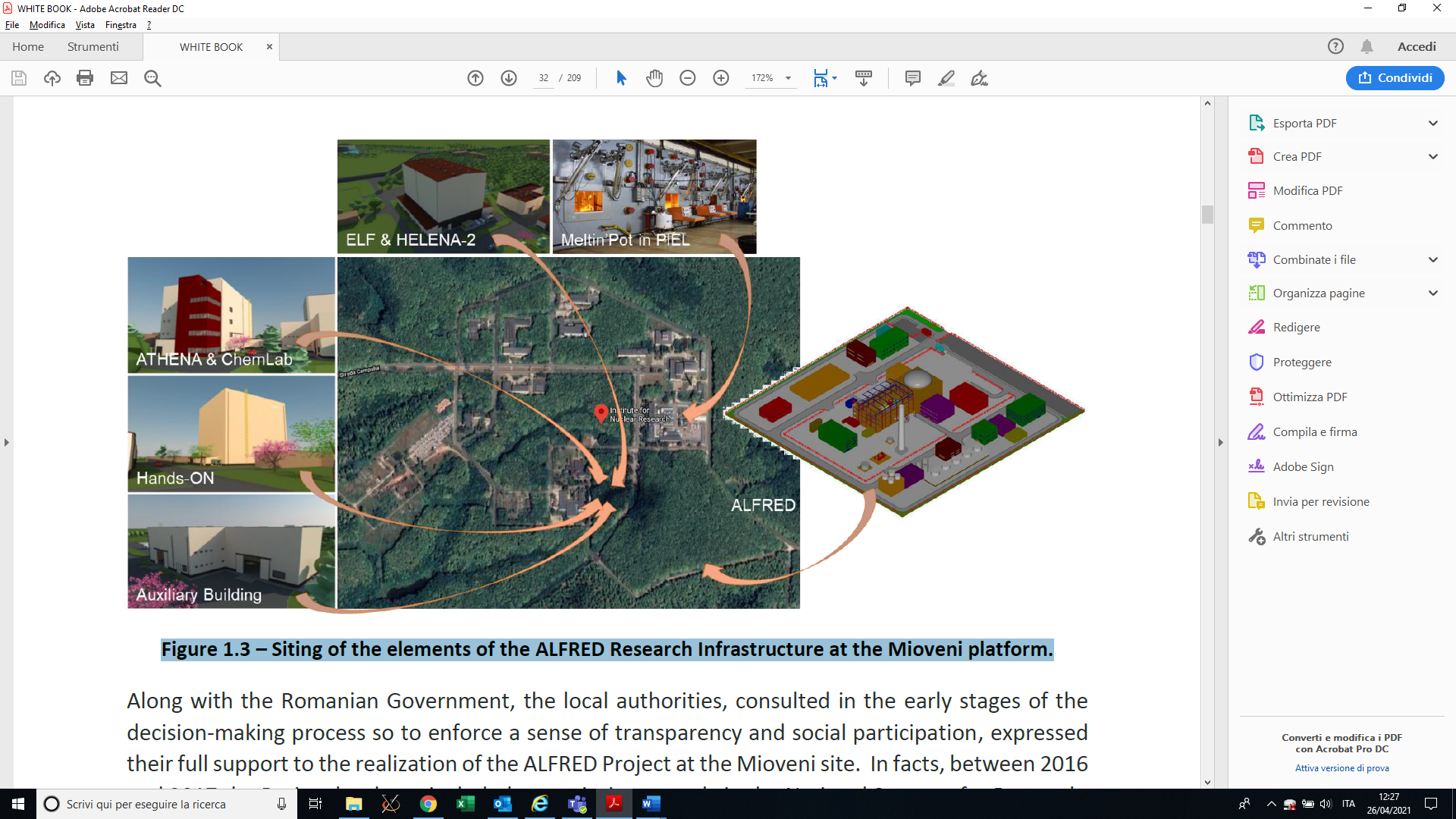
* the use of a new coolant and associated technology, properties, neutronic characteristics, and compatibility with structural materials of the primary system and of the core.
* innovations which require validation programs of new components and systems (the bayonet Heat Exchanger, and its integration inside the reactor vessel, the reactor coolant pump, the passive heat removal system, the fuel handling machine, Oxygen Control System, …)
* the use of advanced fuels (at least in a further stage).

Within this scenario, the Advanced Lead Fast Reactor European Demonstrator (ALFRED) was addressed as mandatory in the European framework to foster the LFR commercial deployment, with the main goal to fill the gap from basic research to market penetration typically suffering from lack of investment. ALFRED was conceived, leveraging on the promising outcomes of previous and ongoing projects devoted to the Heavy Liquid Metal (HLM) technology, to ease the licensing of a first example of the LFR technology in Europe and to gain operational experience in support to the development of an industrial LFR.

With this background, a set of experimental infrastructures, here depicted in Fig. 1, has been defined and scheduled in Romania [1] with the main goal to speed-up the design, safety assessment and licensing process of ALFRED reactor:

* ATHENA (Advanced Thermo-Hydraulics Experiment for Nuclear Application) will be a pool facility for large-scale testing of prototypic components and operating procedures in representative conditions;
* CHEMLAB will be a broad-scope laboratory on the chemistry of HLMs and materials science, for the characterization, qualification and calibration of solutions in support of the Research Infrastructure;
* ELF (Electrical Long-running Facility) will be a pool facility for long-term (endurance) experiments, to characterize the components and systems for use in real reactors from a performance point of view;
* HANDS-ON will be a facility devoted to the testing and qualification of systems and procedures for the handling of core elements from the reactor to the containment and vice versa;
* HELENA2 (Heavy Liquid Metal Experimental Loop for Advanced Nuclear Applications 2) will be a loop facility, capable of operating in forced and natural circulation, for full-scale testing of core elements and their complete thermal-hydraulic characterization;
* MELTIN’POT will be a hot facility to characterize the behaviour of fresh/irradiated fuels and radioactive elements upon interaction with Lead in postulated accidental conditions.

This Research Infrastructure will be then devoted to address the main items of the R&D program: corrosion and irradiation, coolant chemistry, component and systems qualification, Thermal-Hydraulics and Fuel Assembly design. These topics are covered in the following.



*FIG. 1 – Siting of the elements of the ALFRED Research Infrastructure at the Mioveni platform (Romania).*

## THE Research and Development Needs

Extensive R&D efforts are ongoing worldwide (and in Europe notably), addressing issues related to lead technology concepts. Research activities are ongoing and are expected to continue in the future aiming at designing and constructing an LFR prototype.

R&D efforts are necessary for completing the design, support the pre-licensing and starting with the construction of ALFRED, as hereafter explained by topics.

### Material Studies and coolant chemistry

The chemistry of liquid lead is a key topic to be addressed for the development of LFRs. Two main issues have to be considered: (1) coolant oxidation and coolant oxides deposition (mainly PbO) when oxygen is dissolved up to the solubility limit, and (2) corrosion of structural materials. Corrosion is influenced by oxygen dissolved in lead and a sufficient level of oxygen is usually preferred and required to have the formation of a protective oxide layer (i.e. spinel oxide and magnetite) above steel surfaces. According to these issues, oxygen concentration in lead coolant have to be controlled in a proper range to avoid coolant oxidation and minimize steel corrosion via formation of magnetite above steel surface. The strategy for ALFRED about coolant chemistry and oxygen control aims to work with an oxygen concentration (CO) in lead coolant much lower than the saturation at the minimum operating temperature (about 380°C, refuelling temperature). Specifically, as shown in Fig. 2, the CO set for the operation is between 10-6 and 10-8 % wt. to reduce the chance of PbO formation even in case of local heterogeneities and potential deviations [1]. There is no experience in large pools about these operations and their feasibility has to be first assessed in large pool systems in order to define a suitable oxygen control system for ALFRED. For this aims, ELF and ATHENA, supported by CHEMLAB will play a fundamental role.

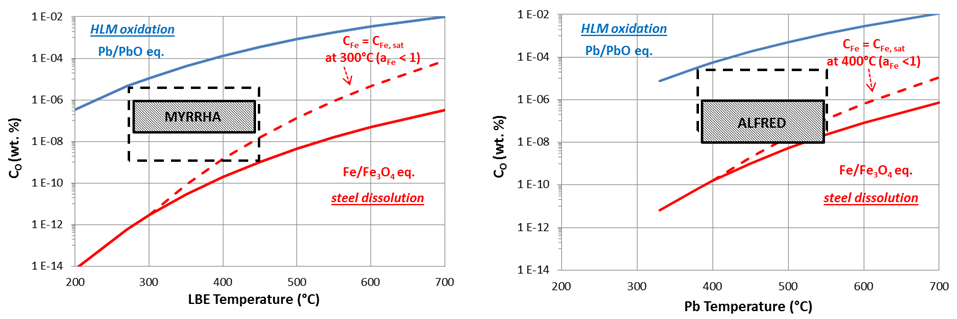
In addition to the corrosion modes, several studies report evidence of Liquid Metal Embrittlement (LME) from testing of ferritic/martensitic (F/M) steels in lead bismuth eutectic (LBE) in the temperature range 200-450°C. For this reason, F/M steels have been excluded from the list of candidate materials for the realization of ALFRED. LME strongly depends on the chemistry and dedicated experiments are needed for an assessment in pure lead, but preliminary data seem to indicate that LME is not an issue for austenitic steels. These experiments are necessary and must be conducted in testing machines where the specimens are exposed to the liquid metal effect, both static and flowing tests. CHEMLAB will serve for this aim, together with a mechanical testing laboratory equipped to perform tests in lead under oxygen control.

Full development of GEN IV programmes also foresees the future increase of reactors operating temperatures (beyond 550 °C). This challenging goal requires, to test “new” materials such as FeCrAl and FeCrAl ODS steels, AFA steels, refractory alloys (Mo), SiC composites, Nb alloys, “MAX” phase materials, as well as coated materials as Al-based surface alloying and Alumina Coatings. The requested testing conditions are 650-800°C lead temperature and 1 – 2 m/s lead velocity. Currently test can be performed by the CHEMLAB with these conditions in stagnant Lead.

The reactor vessel, the structural materials, the internals and the fuel cladding are subjected, to different extent, to several degradation mechanisms such as neutron irradiation, thermal ageing and corrosion. The current knowledge is not complete, and more experimental investigations are needed, providing high quality data on the material behaviour. The following main issues on irradiation performance of candidate materials are of primarily importance for LFR systems development: corrosion in HLM under irradiation (coated and uncoated material), irradiation embrittlement of selected materials, irradiation creep, swelling. The effects of irradiation on materials is a critical issue for LFR system development. The experimental infrastructure needed to address these issues are currently available irradiation machines and research reactors (e.g., BOR-60 in Russia) and they will be investigated also in ALFRED.

It is also essential to control the concentrations of impurities, because of the potential for activation and because of the possible effect on corrosion, mass transfer and scale formation at heat transfer surfaces. Therefore, coolant chemistry control includes oxygen, but also pollution source term studies, mass transport and filtering and capturing techniques. The following specific issues shall be considered: 1) coolant control and purification during operation (i.e. oxygen control, oxygen sensor reliability, coolant filtering, HLM purification, HLM cleaning from components), 2) cover gas control (i.e. radiotoxicity assessment of different elements, migration flow path into cover gas, removal and gettering).

Finally, it is known that the relatively high speed and momentum change between structural material and HLM implies that the pump impellers are subjected to severe corrosion-erosion conditions that might not be sustained in the long term. Tests are planned on specimens of materials, mainly MAX phase or AISI series coated. The materials of the pump impeller have to satisfy demanding requirements which deserve specific experimental installation. The available infrastructure will be CHEMLAB, HELENA2. ATHENA and then ELF, being able to test the capability to withstand to an exposure to high temperature lead (up to 520°C, and higher for long term perspective), the capability to withstand to corrosion/erosion effects due to high relative coolant velocity (10 m/s, and up to 20m/s) as well as to demonstrate the reliability and performances of the pump for a long-term application.



*FIG. 2 – General CO range (dashed black box) and set CO (full black box) for ALFRED.*

### Studies of core integrity

One of the most critical issues in designing fuel assemblies for innovative fast reactors cooled by heavy liquid metals is the correct evaluation and design of spacer grids against Flow Induced Vibration (FIV). Indeed, simulation experience and validation of fluid structure interaction in fuel assemblies are still limited. An engineering experiment is foreseen aiming at providing accurate data set for the validation of numerical approaches, both for design support and safety assessment. For the FIV experiment a not-heated fuel assembly with 61-pins will be designed, instrumented and installed in HELENA2.

Two events represent the loss of the FA coolability: the flow blockage and the deformation of the pin bundle. The flow blockage can happen both internally of the spacer grid and externally, if the blockage is in the spike of the FA. Concerning the loss of coolability induced in a deformed fuel bundle, it can be experimentally studied in a deformed grid spaced rod bundle representing a part of the planned ALFRED FA. To perform the experiment a test section will be designed and manufactured for the HELENA2 loop facility. Using a rotatable deformed central rod, several positions can be experimentally studied, including a deformation with a pin-to pin contact.

Studies on core integrity and main phenomena related to the fuel dispersion after postulated accidental scenarios are carried out also in the MELTIN’POT research infrastructure. One of the foreseen systems aims at studying the chemical interaction between the fuel and the lead coolant. A second system is conceived to investigating the fuel dispersion and relocation in the coolant after a severe accident. Another system is dedicated to the study on the dispersion of fission products. A dedicated system is conceived to inject radioisotopes inside the main vessel, and the main goal is the investigation on retention/dispersion in lead of polonium isotopes.

Finally, the investigation of physical phenomena at relevant scale for the ALFRED reactor is possible thanks to the ATHENA facility, allowing the simulation of the impact of sloshing on the inner structures.

### Steam generator/heat exchanger functionality and safety experimental studies

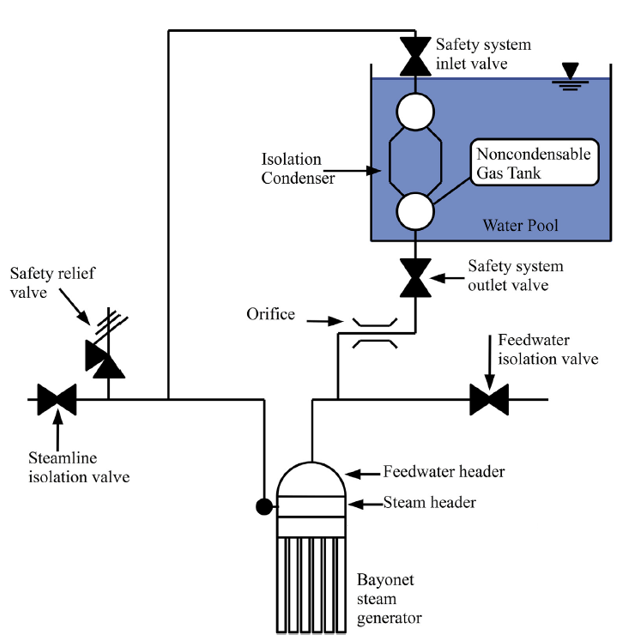
LFR designs are pool type reactors benefitting from the elimination of intermediate circuits (typical of, e.g., Sodium-cooled Fast Reactors), thus having the steam generators directly immersed in the primary coolant inside the reactor vessel. This solution helps the simplification of the design: neither loops in primary system, nor intermediate loops (such as in sodium fast reactor designs).

The rupture of one steam generator (SG) tube constitutes one of the key safety issues of the plant. Such a postulated accidental scenario is referred to as Steam Generator Tube Rupture (SGTR) and could affect the geometry and structural integrity of the plant, as a consequence of vessel pressurization, pressure wave propagation, possible domino effect, sloshing effects, reactivity feedback due to vapor reaching the core and slug/plug formation. The Lead-water interaction on large scale to be simulated in ATHENA facility will permit: to asses a reliable representation of safety parameters at reactor scale; to improve the knowledge of the phenomena / processes, in geometrical and operating conditions more representative of the real reactor; to address the scaling issue in connection with code applications for design and safety analysis purposes.

The SG is of overall importance and deserves accurate studies and evaluations to be qualified. The main qualification studies regard: design validation (unit isolation on demand); pressure drop characteristics; component behavior in normal operation (e.g. forced, mixed and natural convection) as well as operational transients and, as stated above, accident conditions (primarily, the SGTR event and the pressure wave propagation).

The large pool type facility ATHENA is conceived to host a 1:1 scale (in length) mock-up of the ALFRED steam generator. The number of the tubes is then scaled to the electrical power installed in the facility. The experiment will be devoted to the study of the performances in terms of heat removal and pressure drops.

Finally, in ALFRED two separate safety heat removal systems are currently provided, the Decay Heat Removal (DHR) system, here shown in Fig. 3, and the Emergency-DHR (E-DHR) system, whose interfaces with the reactor coolant system (RCS) are made by means of heat exchangers immersed in lead: the SGs or the Dip Coolers (DCs) respectively. One of the main goals of the ATHENA facility is to provide, in a further stage, a full characterization and performance assessment of the IC DHR, providing also experimental data for code validation, and the features of the ALFRED’s E-DHR & Dip-Coolers, together with associated performances, will be tested, addressing their viability and supporting the related licensing process, both with a technological qualification and numerical tools validation.



*FIG. 3 – ALFRED DHR conceptual design.*

### FA Transport System & Spent fuel element transport and cooling system

The HANDS-ON facility is conceived to test the new approach proposed for the fuel handling process of the ALFRED reactor. The concept foresees the use of a transfer cask. The procedure consists in the removal of the Dummy Assemblies (DAs), the introduction of a transfer cask in the core, the displacement of the FA inside it and, finally, the removal of the transfer cask together with the FA from the reactor.

The transfer cask is a cylindrical structure, having a basis able to engage the lower core plate. The upper part of the transfer cask foresees a lateral window, to allow the FA to enter. The lower part of the cask has to be closed for a height sufficient to allow that the spike, the bundle and the outlet are immersed in lead. The cask is sealed before removing it from the reactor.

The new concept of the fuel handling system is characterized by different benefits: being continuously immersed in lead, the FA is safely and reliably cooled by recirculation of lead inside the transfer cask, and the latter through its outer surface by natural circulation of the containment atmosphere; the cask, being stably cooled, allows the elimination of the spent fuel pool in favour of a dry storage area; the cask can be equipped with instrumentation for a continuous monitoring of the FA; the examination of the irradiated FA is one of the goals of ALFRED reactor; the design of the transfer cask allows recovering the stored FA, at any time. If this is required after a very long time, electrical heaters can be integrated to re-melt the lead.

The objective of the HANDS-ON facility is the simulation of the approach described above in order to test and to qualify both the procedures defined and the components and the designed systems. For this purpose, the facility hosts FA mock-ups with the same shape and size of one of ALFRED; moreover, the transfer cask used for the FA handling has the same dimensions of the corresponding item to be used in ALFRED. In such a way, it will be possible to simulate in a reliable way the FA handling, its transport system and the cooling system. Both the FA mock-ups and the cask are instrumented in order to allow the control of FA and cask temperatures, to check the correct cooling during the handling operation (including simulation of accidental scenarios).

### Thermal hydraulics

The investigation about the thermal-hydraulics phenomena occurring in an HLM pool during normal operation, operational transient and accidental scenarios is an important topic for a complete knowledge of an LFR behavior. Nowadays, the only European facility suitable to address this topic of investigation is CIRCE (ENEA, Italy) [4],[5]. Anyway, the refurbishment and upgrade of the CIRCE facility is not sufficient, and new facilities have to be envisaged for pool thermal hydraulics studies. For this aim the construction and operation of experimental infrastructures as ELF is mandatory.

The ELF facility hosts in fact a Core Simulator (CS) which is a representative mock-up 1:1 in height of the ALFRED core. ALFRED will have 3 operating conditions (staged approach) with the inlet and outlet core temperature ranging from 390 to 400 °C and from 430 and 520 °C, respectively, a core thermal power ranging from 100 to 300 MW, a steam pressure ranging from 170 to 180 bar and a steam temperature ranging from 420 to 450 °C. The ALFRED’s core is composed by 134 Fuel Assemblies (FAs); within each FA, a bundle of 126 fuel rods are arranged in a triangular lattice, and the core is completed by 12 control rods, 4 safety devices and 102 Dummy Assemblies (DAs). Considering this scenario of ALFRED, the CS in ELF is composed by a total of 31 Sub-Assemblies (S/As) arranged in a triangular lattice to form a pseudo-cylinder: 16 assemblies electrically heated, representing the Fuel Assemblies (FAs), 1 in-pile section (IPS) and 2 control rods (CRs), which are surrounded by 12 Dummy Assemblies (DAs). The ELF FA reproduces the Spike region, the bottom shroud and the outlet holes. Furthermore, each FA is composed by a central dummy pin and 36 electrical pins with an active length of 810 mm (the same of ALFRED), placed in a hexagonal lattice for a total of 576 electrical pins. The core simulator is surrounded by a baffle following the perimeter of the external assemblies, reproducing in this way the ALFRED core baffle.

Finally, the integral tests represent a key point in the R&D activities dedicated to the development of LFRs. From the execution of such tests it is possible to gather important data concerning the facility operation during steady state normal operation, operational transients and postulated accidental scenarios. ELF is the most suitable facility for the run of integral tests, thanks to its relevant dimensions and the representativeness of the components with respect to the ones of ALFRED, including but not being limited to phenomena and processes of interest at system level and connected with design, safety and operation issues; simulations and analyses of a broad spectrum of accident scenarios; accident management procedures; component testing; scaling issue; generating databases for supporting licensing process; codes assessment and validation.

### HLM pump and corrosion/erosion studies

ALFRED will be equipped with a vertical Lead pump, here illustrated in Fig. 4. In terms of its main data, the nominal head will be 1.5 m, the nominal flow rate will be 1908 m3/h, the lead velocity will be ~10 m/s at maximum and the rotational speed will be ~290 rpm; the bulk material will be AISI316L or AISI321, protected by alumina coating. Thus, corrosion/erosion studies are envisaged in terms of the reactor coolant pump performance, material and long-term assessments. A prototypical vertical pump is to be tested into the ATHENA facility placed inside the riser duct pumping lead from the heater to the steam generator distributor. The experimental test is devoted to characterizing hydraulic performances in terms of pressure head and mass flow rate, addressing efficiency. The tests will be performed as function of the rotating speed. Moreover, an assessment of the pressure drops through the impeller, casing and duct with pump shut down will be investigated as a relevant parameter for the natural circulation characterization. Experiments investigating pump cost down will be performed to support the transient analysis. In terms of design validation, the tests are devoted to investigating a) sealing performances; b) bearing characterization; c) vibrations along the pump shaft and c) characterization of pump impeller materials (coupled with the preliminary results gained by the experiments in CHEMLAB).

An experimental investigation has to be addressed also for the pumps of the primary system which are essential for the reactor reliability. In fact, such components work in a severe environment due to the operation with lead. For this reason, the experimental investigation is mandatory for testing of materials, characterizations of the mechanical parts, performance tests and reliability of the component. In particular, R&D activities are connected with the evaluation of the pump performance as well as their long-term reliability. For this purpose, ELF is equipped with three prototypes of the ALFRED vertical pumps in order to test their reliability and robustness during long run tests and transients.



*FIG. 4 – Lateral (left) and bottom (right) views of the ALFRED reactor coolant pump.*

### Neutronics

The neutronic qualification of a reactor core design by prototypic mockups is an almost impossible task, in facts being practically the purpose of the commissioning tests anticipating the operation of any new reactor. Before these tests, it is responsibility of the neutronic analysts to acquire the soundest possible confidence on the simulation codes, also standing on any available representative real case. Since many codes exist worldwide, focus was oriented to those that have been successfully employed in the past in support to design and licensing of real reactors, so to rely on a sound basis for their full qualification. These codes, however, were applied mostly to LWRs (e.g., MCNP [10]) or SFRs (e.g., ERANOS [9]), so that the extension of their validity to the specific LFR domain remains the main open issue. Additionally, in the perspective of a commercial deployment and of fulfilling sustainability Gen-IV objectives (i.e., MA burning), reduction of nuclear data uncertainties is required.

The specific topics concerning neutronic issues related to LFR development are: Qualification of codes for LFR licensing & operation (validation of critical mass estimation, validation of absorbers worth estimation, validation of flux/power gradients prediction, validation of reactivity effects – Doppler, coolant density, geometry etc. – estimation, validation of neutrons transport prediction in lead, validation of spent fuel composition and decay power prediction, validation of neutron-induced structural damage prediction, validation of shielding design); measurements for nuclear data improvement (uncertainty reduction on main cross sections – Pb, 238U, 241Pu, etc. –, uncertainty reduction on MA cross-sections).

The experimental database for qualification comprises representative integral tests, which are mainly referred to experiments conducted on (zero power) reactors. The International Handbook of Evaluated Reactor Physics Benchmark Experiments (IRPhE) database [6], maintained by the OECD/NEA, is the main source of information about relevant experiments to serve the qualification of neutronic codes. However, among the fast-spectrum experiments, almost all targeted the qualification of SFR designs, so that a gap in representativeness can still be identified for the LFR application domain.

### Fuel Irradiation Testing

For ALFRED development, the material qualification under irradiation and the qualification of advanced fuels are open issues. For the assessment of such issues, material testing reactors, transient testing reactors and hot labs are needed. In the following R&D needs are outlined, considering that for ALFRED the reference fuel is standard FR MOX cladded by 15-15 Ti Steel (AIM-1) with/without coating. It is necessary to complement the database on irradiation performance of candidate materials on the following aspects: coolant-clad interaction (corrosion in lead under irradiation, behavior of degraded cladding, irradiation embrittlement of selected materials – e.g. AFA, 15-15 Ti, AISI316L); irradiation creep (mechanical load, fission gas, in principle covered by the use of proven technology except for the coating); irradiation swelling (in principle covered by the use of proven technology except for the coating); coating coolant interaction; coating integrity.

Moreover, a notional Technology Implementation Plan (TIP) for the development, testing, and qualification of a prototypic fuel element to support design and licensing of an innovative fuel assembly for ALFRED (e.g. UN fuel with AFA cladding) is being currently depicted [7]. This TIP outlines a generic methodology for the progression from non-nuclear out-of-pile (OOP) testing through nuclear in-pile (IP) testing, at operational temperatures, flows, and specific powers, of a FA element in an existing test reactor. Subsequent post-irradiation examination (PIE) will occur in existing radiological facilities. The goals of OOP and IP testing are to provide confidence in the operational performance of fuel system concepts, as well as to provide data for the licensing.

## THE NEEDS FOR CODES’ VERIFICATION AND VALIDATION

Experimental data are fundamental for supporting the development and demonstrating the reliability of computer codes in simulating the behaviour of an NPP, or its systems and components, during normal operation or a postulated accident scenario: in general, this is a regulatory requirement. However, the user always has the responsibility of the appropriate use of such codes. The following main categories of codes are identified as explained hereafter.

### Reactor physics and radiation shielding codes

These codes are used for physics design and safety analysis. The objectives are: to model the core in steady-state conditions and to determine its reactivity, as well as the global flux and power distributions; to determine the reactivity feedbacks and the core response to different perturbations; to define the configuration and features of control and emergency shutdown rods; to analyse the behaviour of the core in transients, from in normal operation to accident conditions; to investigate the core depletion, the refuelling schemes, the discharge burn-up as well as its distribution; to ensure that the radiation-induced damage to the reactor structures remains below acceptable limits; to ensure that the radiation levels in all the premises of the reactor remain below the limits set by the regulation for radiation protection. In case of LFR technology, codes available to fulfil the objectives above are: ERANOS [9] and MCNP [10] (or Serpent [11]) codes for (static) deterministic and stochastic neutron transport calculations; the SCALE package [12] for cross-section generation. It is required to enhance and to validate multi-group and continuous-energy neutron and photon nuclear cross section libraries, dedicated respectively to deterministic and stochastic (e.g. Monte Carlo) transport codes.

### Fuel and Sub-channels behaviour codes

Fuel Performance Codes [13] are developed for predicting the fuel rod behaviour of nuclear power reactors in normal operation and accidental conditions. The issues to be investigated in the LFR technology related to the fuel pin may be preliminary classified as follows [54]: high burn-up performance in normal operation; Fission Gas Release (FGR) and inner pin pressurization in steady state and transient conditions; cladding performance with respect to fuel-cladding mechanical interaction “FCMI” and fuel-cladding chemical interaction “FCCI” phenomena (steady state and transient); fuel and cladding nuclear qualification in lead environment.

Sub-channel analysis is devoted to the prediction of the flow, enthalpy and temperature distribution in the coolant flow channels and on the cladding outer surface as a minimum, and for this represent the most widely acknowledged methods for the thermal-hydraulic analysis of fuel and non-fuel assemblies. In the LFR domain, their development and effectiveness are enhanced with respect to (e.g.) Light Water Reactors (LWRs), being the coolant boiling excluded in the analysis. Nevertheless, their applicability is typically limited to steady-state and some transient conditions, while their use might be challenging if the freezing of the lead is considered. The sub-channel code ANTEO+ [14] could be efficiently used, since it has been specifically developed for this purpose. RELAP5 codes [15] are being up-dated and validated [16] in LBE for different fuel bundle configurations (i.e. ELSY and MYRRHA). Also the CFD codes (see the dedicated Section 3.6), thanks to the computational resources available and the level of detail of these analyses, are expected to play a relevant role as soon as they will be qualified for specific design justification purposes.

### Structural Analysis codes

Commercial FEM codes can be used for the structural mechanics’ analyses. The challenge of the application is connected 1) with the proper evaluation of the loads, connected with the reliability of the working assumptions or the numerical predictions of other codes; 2) with the material properties and performances under irradiation in flowing Lead coolant; and 3) the codes and standards employed to assess the design criteria. In some cases, notably related to the analysis of special components where irradiation-induced effects are definitely not negligible, a different approach was historically followed, and is still pursued by large research organizations. Among the consolidated ad-hoc structural analysis codes, are worth mentioning the Japanese BAMBOO [17] and the French SOLO [18] FA analysis codes, as well as the French HARMONIE [18] and the US NUBOW-3D [19] core restraint codes.

### System thermo-hydraulic codes

System thermo-hydraulic codes such as ATHLET [20], CATHARE2 [21], RELAP5 [22], [23], SASSYS-1/SAS4A [24], TRACE [25],[26], etc. are specifically designed for the system analysis. They model the systems with hundreds or thousands of components and have, in principle, suitable capabilities to support the development of the LFR. These codes have been developed to simulate thermal-hydraulic transients in reactor systems that use light water as the working fluid. Nevertheless, their general structure allows their development to simulate other working fluids. Considering the LFR, these codes have limited validation activities carried out during the last years. Largest part of these activities is carried out with RELAP5/Mod3 and RELAP5-3D. They are mainly based on experiments carried out at the ENEA, KTH, and SCK·CEN research centre. The data available and the validation are more extended, if sodium is also considered. Besides the general approach required to set-up and validate the code models prior the application to the analysis, two well-known phenomena challenging for System thermo-hydraulic codes can be postulated ex-ante. They are: the mixing and the thermal stratification, which cannot be accurately modelled; and the fluid conduction, neglected in the codes. These are evaluated depending upon the geometries and transients during the validation process.

### Severe Accident codes

SIMMER-IV [27] a three-dimensional (3-D), multi-velocity-field, multi-phase, multi-component, Eulerian, fluid dynamics code system coupled with a structure model for fuel pins, hexcans and general structures, plus a space-, angle-, time- and energy-dependent transport theory neutron dynamics model. An elaborate analytical equation of state (EOS) model closes the fluid dynamics conservation equations.

The fluid dynamics portion is interfaced with a structure model through heat and mass transfer at structure surfaces. The neutronics part provides nuclear heat sources based on time-dependent neutron flux distributions consistent with the mass and energy distributions. The SIMMER codes family was primarily developed for mechanistic analyses of transients and accidents in Liquid Metal Fast Reactors (LMFR) and is used as a reference tool for severe accident simulations. Pioneering applications of this code to the SA analysis of HLM reactors is conducted in Ref. [28]. R&D activities are conducted to investigate the fuel-coolant chemical interaction [8].

### Computational Fluid Dynamics codes

Considering the Computational Fluid Dynamics (CFD), the challenges are connected with the pool based LFR design and the features of the HLM (i.e. high density, low Pr). Both commercial (e.g. ANSYS CFX [29], ANSYS Fluent [30], STAR.CCM+ [31]) and open source (e.g. OpenFOAM [32]) can be used for the simulations. The open-source fluid/thermal simulation US code Nek5000 is designed specifically for transitional and turbulent flows in complex domains. Nek5000 is based on the spectral element method (SEM), a high-order weighted residual technique that combines the geometric flexibility of finite elements with the rapid convergence and tensor-product efficiencies of global spectral methods. Nek5000 has been validated extensively both in Direct Numerical Simulation (DNS) and LES mode. CFD simulations can investigate the thermal-hydraulic behaviour of components or local zones in steady state and transient conditions, providing high level three-dimensional detail of the thermal and flow fields. It is expected their applications for simulating, to different extent, the fuel assembly (including sub-channel analysis); the hot and cold pools, the hot pool free level surface, the heat exchanger outlet. Validation activities will play a key role in order to reduce the uncertainty in the prediction, to define procedures for proper code application, thus ensuring the results are not affected by the user, and to demonstrate the reliability and the accuracy of the numerical results.

### Coupled codes

More realistic simulations of complex phenomena and transients may require the coupling of different codes. This coupling techniques include mainly thermal-hydraulic system and reactor kinetics codes, as well as specific codes for the containment thermal-hydraulics, structural mechanics, and CFD. Given the implicit assumption that the codes are qualified for the specific application, the employment of the coupled codes requires the V&V of the coupling technique and of the coupled code, and then the qualification. Different types of coupling tools are postulated relevant for the simulation of the LFR, like SYS-TH/3D-NK [8] (simulating the reactivity-initiated accidents (RIA) and the unprotected transients), SYS-TH/CFD [8] (simulating a system analysis, including integrated, detailed mechanistic calculations of the pool or of the fuel assemblies, or of the steam generator primary system) and SYS-TH/SA [33] (simulating multi-fluid, multi-dimensional system).

## SUMMARY AND CONCLUSIONS

In the present paper the R&D needs for the LFR technology, considering in particular the current status of the ALFRED conceptual design as well as of the research infrastructures’ implementation in Romania, have been presented. A comprehensive summary is here proposed in Table 1, including a prioritization of activities as well as the foreseen facilities of the ALFRED Research Infrastructure being able to address the specific R&D need. The priority level was associated to each R&D need considering the lack of completeness and/or full applicability of the already existing experimental results for the LFR technology as well the key importance role that it is expected to play in support to the licensing phase of ALFRED.

TABLE 1. R&D NEEDS AND ASSOCIATED PRIORITIES

| **Topic** | **R&D Needs** | **Priority Level** | **Research Infrastructure** |
| --- | --- | --- | --- |
| Material Studies and coolant chemistry | Coolant Chemistry and Corrosion in lead | High | CHEMLAB  ATHENA  ELF |
| Embrittlement and degradation of structures by liquid metal | High | CHEMLAB |
| High temperature materials for long term perspectives | Low | CHEMLAB  ELF |
| Irradiation effects on materials | Medium | ALFRED |
| Impeller pump materials | High | CHEMLAB  ATHENA  ELF |
| Studies of Core integrity | Fuel Manipulator Assessment | High | HANDS ON |
| Fuel Assembly Structure and Support | Medium | HELENA2  HANDS ON |
| Fuel Assembly Characterization | Medium | HELENA2 |
| Fuel Assembly Coolability | Medium | HELENA2 |
| Core Arrangement integrity and safety systems | High | MELTIN’POT |
| Control Rods | Medium | ATHENA |
| Steam Generators / Heat Exchanger functionality and safety experimental studies | Lead-Water interaction and SGTR event | Medium | ATHENA |
| Steam generator | Medium | ATHENA  ELF |
| Auxiliary nuclear systems | High | ATHENA  ELF |
| Fuel assembly transport system & spent fuel element transport and cooling system | Remote fuel assembly handling, cask transfer in spent pool | High | HANDS ON |
| Thermal hydraulics | HLM pool thermal hydraulics | High | ATHENA  ELF |
| Core thermal hydraulics | High | ELF |
| Integral tests | High | ELF |
| Long run tests | Medium | ELF |
| Fuel assembly thermal hydraulics | Medium | ATHENA  ELF |
| HLM pump and corrosion/erosion studies | Reactor coolant pump performance and material assessment | High | HELENA2  ATHENA |
| Reactor coolant pump long term assessment | Medium | ELF |
| Neutronics | Measurement for nuclear data improvement | Low | Zero Power EU Reactors (like TRIGA in Pitesti) |
| Fuel Irradiation Testing | Irradiation effects on cladding materials | High | ALFRED |
| Irradiation testing and qualification for innovative nuclear fuel | Low | ALFRED |

Moreover, high priority is assigned to code Verification and Validation. In this regard, a dedicated program plan will be defined in parallel to the development of the ALFRED Research Infrastructure to determine the best physics modeling capabilities needed in system codes and will include assessments for the quantification of uncertainties in code predictions as well as for the scaling issues – the needs for the experiments mentioned above shall also serve to make available experimental database for code validation.

In general, one of the main outcome of the whole experimental campaign will be the support to the deterministic approach for the license of the ALFRED reactor in Romania which for sure has issues, needs and constrains. In fact, the main foreseen need, currently being addressed, is to initiate a formal pre-licensing phase, because of the innovative nature of the LFR technology. This will allow to prepare the subsequent licensing on sound and comprehensive bases, as well as to setup early dialogues with the safety authority. The pre-licensing phase foresees the establishment of the normative framework, the preliminary validation of the reactor design and, finally, the agreement on a “Safety Demonstration Programme” providing the experimental evidence required to justify the safety claims of the proposed design on those areas not covered by the established normative framework. In this regard, the list of prioritized ALFRED R&D needs discussed in the present paper is a key element being now addressed.

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