IMPROVEMENT OF ALFRED THERMAL HYDRAULICS THROUGH EXPERIMENTS AND NUMERICAL STUDIES

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Abstract

Research and development on Generation IV concepts cooled by heavy liquid metals have been supported through several national and international funds. Looking at the European community, the technology benefited from approximately 145 M€ of investments, increasing the technology readiness level through various areas including safety and resilience, neutronics and thermal hydraulics. One of the projects currently pursued in Europe and supported by numerous research centers, private companies and member states is ALFRED (Advanced Lead Cooled Fast Reactor European Demonstrator), born around 10 years ago within the LEADER project. Following the FP7 project, Romania declared its support in ALFRED development by proposing as hosting country and the FALCON consortium (Fostering ALFRED Construction) was set out, consisting of Ansaldo Nucleare, ENEA and RATEN-ICN. The consortium has recently revised the reactor concept to solve some thermal hydraulic issues with an increase of technology know-how. The changes were supported by a new deployment strategy based on different power stages, appointing ALFRED as the demonstrator for the future fleet of commercial SMRs cooled by liquid lead. Several experimental campaigns and numerical analyses developed by the consortium and its supporting organizations have been of extreme importance. The areas that benefited mostly from this effort were the thermal hydraulics of the reactor pool, of the core and of the safety systems, thanks to numerical studies made with system codes, computational fluid dynamic tools supported by ENEA technological infrastructure. The paper presents the new plant concept, reporting some of the most important results that have made it possible to solve complex problems such as thermal striping, thermal stratification and freezing of lead by overcooling. Lastly, the paper highlights the next steps for ALFRED development, such as the construction of various experimental facilities that will develop the test campaigns for the safety justification and licensing of the plant.

1. INTRODUCTION

Lead Fast Reactor (LFR) technology is one of the most promising within the Generation IV international forum (GIF) [1] for its potential to achieve the goals for the new fission reactor fleet [2]. Key elements that make the technology attractive are the type of coolant offering a very wide temperature range in the liquid state, high density, good heat capacity, low vapor pressure, absence of chemical reactions with other fluids (water or air) and suitability for fast spectrum [3]. In addition, lead is an excellent shield for ionizing radiation and experimental evidence suggests an innate ability to retain hazardous radioisotopes [3]. These properties result in low-pressure Reactor Coolant Systems (RCS), compact layouts for the nuclear island, direct energy transfer without intermediate circuits and compatibility with passive safety systems adopting natural circulation. In recent years LFR business has seen a rapid escalation of its technology readiness level well above what is usually expected for innovative technologies, ensuring valid design solutions for the construction of future power plants. Focusing on the European panorama, the most important project for the construction of a Small Modular Reactor of intermediate power (over 100 MW(e)) is based on ALFRED design (Advanced Lead Cooled Fast Reactor European Demonstrator) led by the FALCON consortium (Fostering ALFRED Construction) in which Ansaldo Nucleare SpA, ENEA and RATEN-ICN take part. To promote and support the project there are several research centres and leading European companies that through Memoranda of Agreement (MoA) strengthen the design, research and development teams by offering their experimental equipment, specialized analyses and expertise on design and manufacturing of innovative components and systems. The ALFRED project therefore develops within an enlarged consortium where the major European excellences in the field of advanced nuclear power work together for a common goal and where the openness to collaboration is expressed in the full availability of new agreements and research plans for lead fast reactors technology. Over the last few years, several objectives have been jointly achieved, validating various aspects of ALFRED's physics and dynamics and which have similarly inspired changes to the design [4] and the operation strategy of the plant [5]. This article aims to recall ALFRED revised configuration and presents the major R&D activities recently carried out, with an eye to what has been developed on the thermal-hydraulics of the plant, where key issues of advanced reactors such as thermal stratification, pool dynamics, heat transfer performance of the core and general safety of the plant have been addressed.

2. ALFRED RCS DESCRIPTION

ALFRED is a pool type lead fast reactor where all the components of the RCS are arranged inside the Reactor Vessel (RV). FIG. 1 shows an axial cross-section of the RCS: starting from the core (point 1 in FIG. 1), the hot lead rises into the inner vessel to cross the three pump ducts that pull it into the hot pool, i.e., the region where the lead resides at the maximum average temperature. From this region the lead descends into the 3 bayonet-type steam generators to reach the lower cold pool that externally embraces the inner vessel. Before returning to the core, the lead rises up one last time parallel to the RV following the path defined by the Internal Structure (IS). Near the lead free surface, windows in the IS open the lead to the annular region between IS and RV, from which the coolant descends to reach the area immediately below the core.

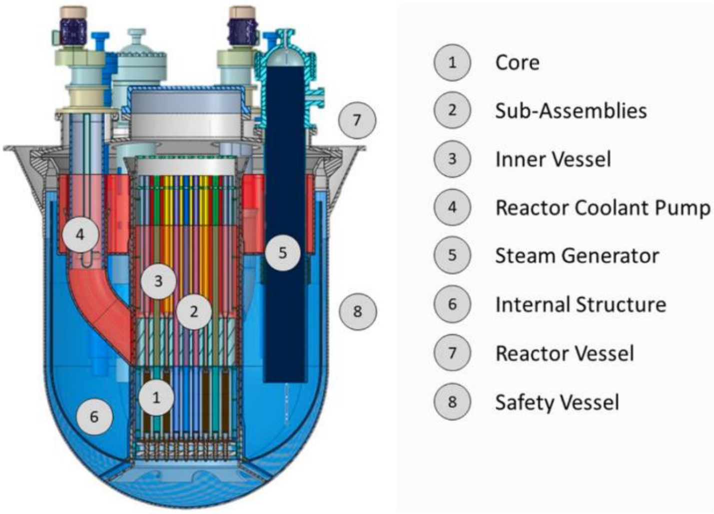

FIG. 1. ALFRED RCS cutaway [4].

The plant configuration is the result of years of improvements to combine the demonstrator concept for the next large reactor fleet with a pilot plant for the Advanced Small Modular Reactors. The main features are a careful thermal-hydraulic design of the RCS allowing to optimize the positioning of the main components, leaving ample space for the insertion of auxiliary systems and In-Pile sections for demonstrating the technology. To date, the design proves to be forgiving, potentially allowing a global reduction of the overall dimensions. The installation of an IS guarantees a clear physical separation between the different areas of the pool, allowing to take both advantages of loop flow predictability and pool coupling of the main components, with a maintenance strategy following the "everything can be extracted" philosophy. The IS also guarantees to set all the lead mass in motion, maximizing the effective heat capacity and avoiding stagnant regions where thermal stratification under accidental conditions can rise. Finally, the direct connection between the SG and the free surface allows to adopt single-wall steam generators where the fluid from a break is conveyed towards the cover gas from which it is then expelled.

The ALFRED operation follows the concept of the staged approach [5], i.e., a strategy that combines fast deployment with the need for operational experience. In this context, it is planned to initially operate the reactor at low power and low lead temperature, to validate its operation in process conditions for which a recognized operational experience is available. With the accumulation of reactor hours and new experimental data, the reactor power is increased, with the maximum lead temperature and the efficiency of the power conversion system. At its last stage, the ALFRED operating conditions allow the full scale demonstration of an SMR plant having the potential to compete on various national electricity markets, with a particular focus on those having high penetration of renewable energy sources (RES).

3. POOL THERMAL HYDRAULICS

At the centre of the efforts to improve the ALFRED configuration lies the fluid dynamics of the pool. In particular, the regions near the core outlet and the lead windows in the IS were investigated numerically. At an experimental level, making use of ENEA's important experimental infrastructure, several experimental activities were carried out in the CIRCE facility as integral experiments to assess the behaviour of a complex pool LFR system with all the components.

Numerical analyses

Lead transport through the IS has been studied by CRS4 through the calculation code STAR-CCM + [6] which, through a coarse mesh and porous media approach for the more complex components, fully simulates the RCS. The model was created with a flexible methodology to allow the study of steady state conditions, natural circulation during decay heat and RV break. The study has made it possible to certify that the most efficient configuration of windows on the IS is the one that uses side openings only in the upper part of the structure. In this way, the onset of thermal stratification is eliminated by design in all operational and accidental conditions. FIG. 2 shows the temperature field in the annulus between IS and RV during an accidental condition at low power, where the temperature difference between the zones is minimized, and with it the associated thermomechanical stresses.

Recently, NRG developed a detailed CFD model using the STAR-CCM+ calculation code [7] which represents 1/3 of the RCS region interposed between the core outlet and the hot pool inlet, including part of the IV and one RCP duct. The aim of the study was to evaluate the velocity and pressure fields, making sure to guarantee acceptable velocity and pressure drops to avoid corrosion/erosion phenomena at high temperature and minimize impact on natural circulation. FIG. 3 shows a detail of the velocity field. It can be noted the different flow distribution between the innermost fuel assemblies and the outermost dummy assemblies. At the outlet of the core, the lead in the innermost part rises to the free surface and travels radially until it enters the RCP channel. More in the periphery, the lead crosses the IV mainly in a radial direction. The analyses allowed to highlight how the flow is the result of a complex combination of vertical motions dictated by the momentum and transverse motions between the rows of the assemblies. The results of the analyses confirm the promising configuration that guarantees both low velocities and negligible pressure drops.

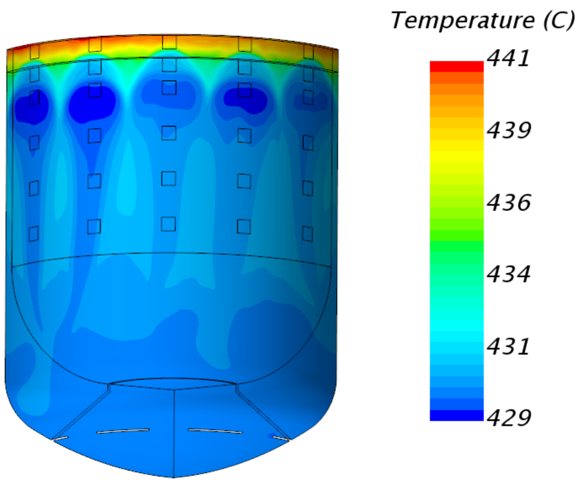

FIG. 2. Temperature field between IS and RV during PLOF at 2% power.

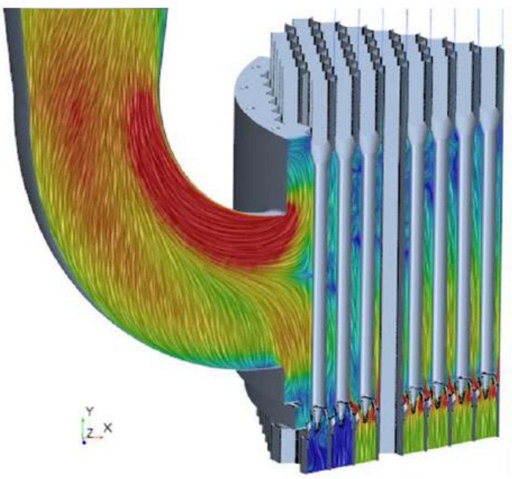

FIG. 3. Velocity field in the IV and the pump duct.

Experiments

A series of experimental campaign were carried out at ENEA Brasimone laboratory in the CIRCE (CIRColazione Eutettico) large pool integral facility to investigate relevant thermal-hydraulic phenomena occurring in normal and abnormal operating conditions. CIRCE is a non-nuclear (electrical) experimental platform in a scale representative of the pool-type LFRs. The main vessel is a cylindrical vessel (1200 mm outer diameter, 8500 mm height and 15 mm thickness) designed to withstand a temperature of 500°C and a pressure of 15 bar. The working primary fluid is lead bismuth eutectic (LBE) and the maximum inventory is 90 tons [8],[9]. In the CIRCE-ICE configuration (see FIG. 4) the heating source of the ICE Test Section is represented by a fuel pin simulator electrically heated composed of 37 pins with a total installed power of 925 kW.

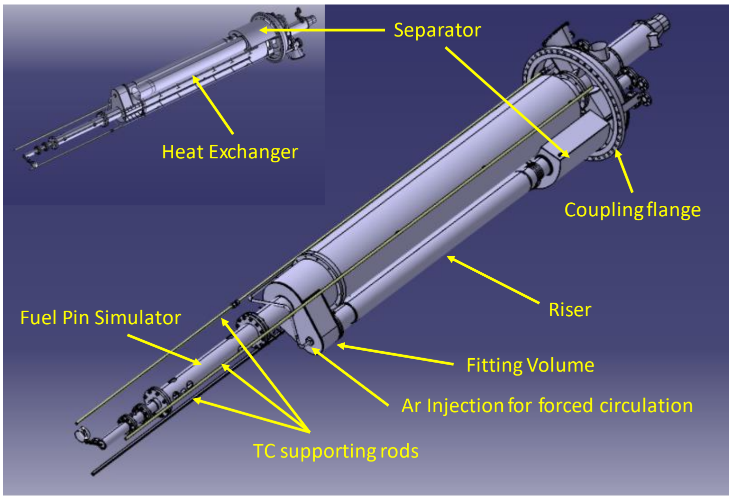

FIG. 4. CIRCE-ICE test section.

Each pin has a diameter of 8.2 mm, an active (heated) length of 1 m and a P/D ratio of 1.8. Such large fuel pin lattice reduces the pressure losses and hence facilitates the establishment of coolant natural circulation for the long term coolability of the fuel assembly (larger p/d values with respect to sodium fast reactors are used in HLM concept thanks to the coolant characteristics of low-moderating medium and low-adsorption cross section). The heated LBE flows upward in the riser up to the separator acting as a hot pool. This component is used also to separate the gas injected at the bottom of the riser for promoting the forced circulation (gas enhanced circulation) and connect the primary system to the heat exchanger (HX). The HX is made of 91 bayonet tubes (bayonet type) consisting of three concentric tubes. The water flows downward in the inner pipes, and then upward in the annulus between the inner and intermediate pipes. Vaporization takes place in the ascending annulus avoiding the formation of flow instabilities. The annulus between the middle and outer pipes is filled by pressurized helium (4.5 bar) and used to detect any tube rupture monitoring the pressure in the common helium gas plenum. The HX is fed by water at ambient temperature and pressure. The decay heat removal system (DHR) acts as heat sink of the system during the simulation of the HX failure (Loss of Flow) for the long term coolability. It is hydraulically de-coupled by the primary system being placed into the downcomer. The DHR heat exchanger is a bayonet tube fed by air at atmospheric pressure designed to have a thermal duty of 40 kW. The simulated accidental scenario was a Protected Loss of Heat Sink with Loss of Flow accident constituted by the total loss of the secondary circuit and the coolant pump trip (simulated stopping the gas enhanced circulation and the water feeding circuit) with the consequent reactor scram (reduction of the electric power supplied to the fuel pin simulator from ~750 kW at full power steady state to ~30 kW) and activation of DHR system to remove the decay heat power (~5% of the nominal value) [10].

The experiment preliminary demonstrate the long term coolability of the fuel bundle under natural circulation condition. Furthermore, the simulated transient revealed the absence of high temperature peaks on the clad of the pins during the transient, before the establishment of the LBE natural circulation (FIG. 5).

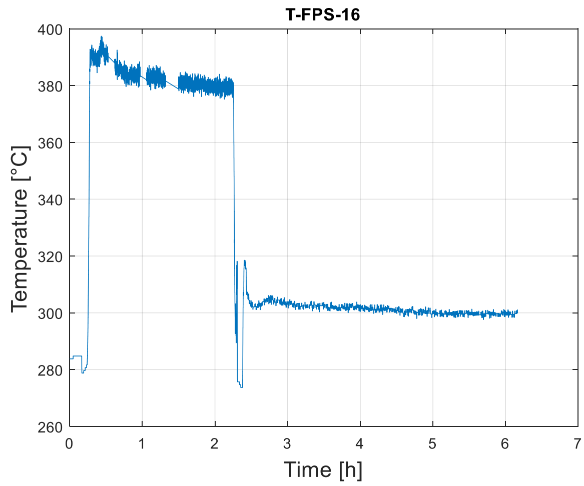

FIG. 5. Clad temperature on the central pin 60 mm upstream of the upper spacer grid

Concerning the thermal stratification inside the pool at the full power steady state, the experiment evidences the same general behaviour independently from the external conditions, characterized by the presence of a thermal gradient of about 40°C in the first 3.5 m starting from the free level (up to the outlet section of the HX). Then between the outlet sections of the HX and the DHR, the slope of the vertical temperature profile increases, with a temperature drop of about 15-20°C in less than 1 m. Moreover, the thermal stratification in the pool is purely vertical with negligible temperature variation on the horizontal planes. After the transition to natural circulation, the vertical temperature profile changes. In the upper and lower part of the vessel LBE temperature is uniform with a layer separating the two zones and where the thermal gradient concentrates. That area, after the transition from forced to natural circulation, moved downwards below the DHR outlet section and the thermal gradient reduces to about 10°C.

During the SESAME (Simulations and Experiments for the Safety Assessment of MEtal cooled reactors) [11] and MYRTE (MYRRHA Research and Transmutation Endeavour) [12] H2020 EU projects, the CIRCE facility has been refurbished replacing the heat exchanger used in the ICE configuration with a new steam generator of type with double wall bayonet tube, named HERO. The Steam Generator Tube Bundle (SGBT) layout was scaled 1:1 in length respect to the steam generators of ALFRED in LEADER configuration [13]. The HERO SGBT (see FIG. 6) consists of a tube bundle of seven bayonet tubes arranged with a triangular pitch in a hexagonal shell, and it is connected to a dedicated secondary loop that provides demineralized water at a high pressure (up to 180 bar) and high temperature (up to 335°C).

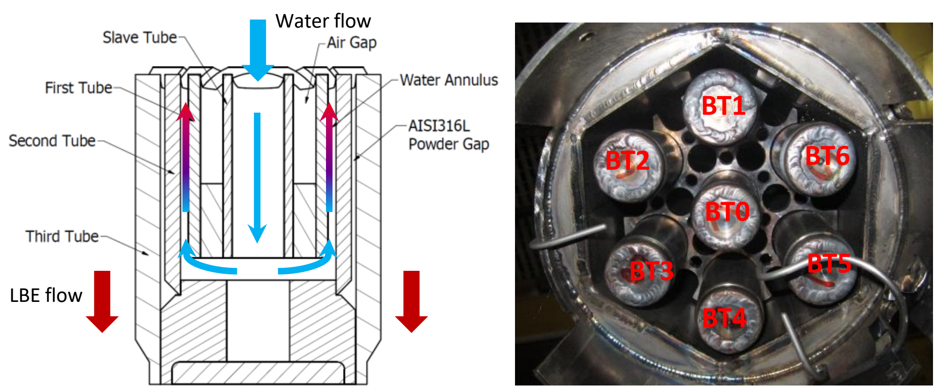

FIG. 6. SGBT layout: tube geometry (left) and bundle arrangement (right).

In HERO configuration, the CIRCE facility has been used within the SESAME H2020 EU project to perform integral tests simulating Protected Loss Of Flow Accident (PLOFAs) scenarios [14],[15], reproducing the main phenomena occurring during the transition from forced to natural circulation scenarios in LFRs (e.g., temperature peaks in the FPS, mixed convection, and stratification in the main pool) and evaluating the performances of SGBT as the main steam generator and decay heat removal (DHR) heat exchanger.

The test matrix consisted in three PLOFA experiments. In all the tests, the heating source power, LBE and water mass flow rate has been changed accordingly with the design conditions reported in [14],[15]. The experimental results proved that, despite the loss of the forced circulation regime in the primary loop, the power transient (decay heat curve) led to a sudden decrease of the LBE and pin clad temperatures along the FPS, avoiding dangerous peaks in the active region and assuring the correct cooling of the heating source (see FIG. 7). Concerning the pool thermal stratification, the tests show that it occurred in a vertical direction only, with uniformity along the horizontal planes (see FIG. 8).

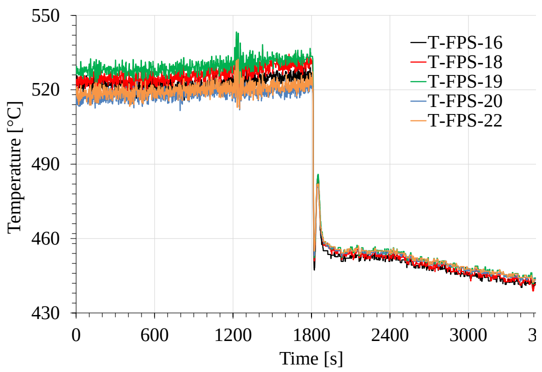

FIG. 7. Pin clad temperature measured by the thermocouples at the FPS outlet section.

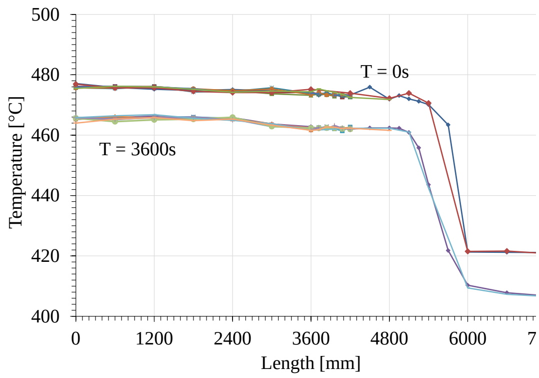

FIG. 8. Pin clad temperature measured by the thermocouples at the FPS outlet section.

Within the MYRTE H2020 EU project, a set of experiments were performed to qualify the SGBT concept in relevant conditions for the MYRRHA primary heat exchanger. For this purpose, an experimental sensitivity analysis has been performed on the main operating parameters (i.e., LBE and water flow rates and temperatures), characterizing the component from a thermal-hydraulic point of view [16]. The tests performed in CIRCE were used to support the qualification and validation process of the numerical tools for heavy liquid metal applications [17]- [19]. A numerical benchmark was also performed for System Thermal-Hydraulic (STH) codes and coupled STH/CFD codes based on the SESAME transient tests [20].

4. CORE THERMAL HYDRAULICS

The core thermal hydraulic is one of the most important topics to be investigated. In the recent years, several experimental and numerical effort were spent at ENEA in this field.

Numerical analyses

The thermal hydraulic analysis of the hottest fuel assemblies was performed by means of subchannel and CFD methods, to retrieve the actual uncertainties associated with the estimations, while capturing local effects perturbing the average thermal fields, at the desired level of confidence (which, for ALFRED, is set at 3σ as set of standard deviations due to its nature of demonstration reactor, the reduction of uncertainties being one of its envisaged achievements).

CFD analyses also provided the pressure fields at different sections of the assemblies, used to optimize the design of the gagging schemes needed for flow zoning, to flatten the coolant outlet temperature both on the fuel assemblies generating different power (though all in the same order of magnitude), and the unpowered subassemblies, as well as the bypass region among them, whose power (thus flow) may differ one the other by about 2 orders of magnitude.

Preliminary evaluations were also performed concerning the anticipated impact of flow blockage, postulating its occurrence in correspondence of a spacer grid supposed to be positioned at the core midplane. Several blocked areas were postulated, differing one the other by location in the cross-section and by extension, to cover a broad range of conditions. The results allowed to retrieve information on the recirculation of the coolant downstream the blockage, the formation of low-flow-to-stagnant regions, the length of the affected region before a normal flow is recovered, and the impact on the temperatures suffered by the cladding (on which temperature limits are established by design).

In parallel with the analytical activities briefly described, efforts were spent in the development of methodologies and tools for the thermal hydraulic analysis of the reactor core, under steady-state operation regimes and single-phase coolant in forced condition. Several initiatives were put forward by ENEA, with the support of some Italian universities. The final objective is to dispose of at least three codes, all based on the subchannel approach, dealing respectively with the thermal hydraulic analysis of a fuel assembly in nominal conditions (ANTEO+), of the bypass region (TIFONE) and of a fuel assembly which has undergone deformations as due to thermal and irradiation-induced swelling gradients (EFIALTE).

concerning ANTEO+, the most mature tool under development, extensive validation campaigns have been performed [21], while activities are ongoing for the preparation of a validation dossier [22], to allow for the qualification of the simulation tool by safety authorities in the anticipated application domain, thus the use of ANTEO+ in support to the licensing of the reactor design.

The development work of TIFONE has recently terminated, and validation is in progress [23]. The early results, although encouraging, pointed out an insufficient number of significant experiments to perform validation, supporting the need for new experimental facilities, as the ones to be realized in Romania in support of ALFRED (see Section 6).

Experiments

A large experimental and numerical work was undertaken at ENEA in the last years on Fuel Pin Simulator (FPS) technology. The work included the design of the experimental test section, the experiments, and the use of numerical tools to predict the instrumentation position and to validate codes. This work concerned mainly the two experimental facilities NACIE (NAtural CIrculation Experiment) and CIRCE (already introduced in Section 3.2). Both wire-spaced and grid-spaced fuel pin bundles were investigated. Experiments were carried out within EU Projects (THINS, SEARCH, and SESAME) and the ADP-PAR National program.

NACIE-UP is a rectangular loop in which there is a difference in height (5 m) between the heat source (FPS) and heat sink (HX), allowing natural circulation. Forced circulation is guaranteed by a gas-injection (Argon) in the riser of the facility – the gas lightens the riser and the hydrostatic unbalance between the two columns will provide a positive pressure head. This method is called gas-lift. A prototypical thermal flow meter measures the mass flow rate. Thermocouples provide temperature distribution in the loop, while the FPS test section is highly instrumented with about 70 TCs.

A 19-pin MYRRHA-like 250 kW wire-spaced FPS was designed, procured, and installed in the NACIE-UP facility, see FIG. 9.

In the first experimental campaign, data on local and overall heat transfer were produced. Uncertainty analysis was performed on the data. For the overall heat transfer, the results showed values between the Carelli–Kazimi and the Mikityuk correlations, and the slope of the experimental trend was very similar to the correlations, see [24] for a complete discussion of the results and FIG. 10 for a representation of the Nusselt number.

With the same test section and the same facility, a few transient experiments were carried out within the SESAME EU project. The three fundamental transient tests concerned transients with gas lift transition, power transition, and a combination of the two (power and gas reduction (PLOFA)), respectively, and focus on the study of the thermal-hydraulic behaviour of the loop and the bundle during power and mass flow. A strong subchannel rank effect emerged from the experimental data, with the central subchannels being hotter than the peripherals ones. Lots of data, both integral and local, were produced and published [25],[26], and were used as reference for an international benchmark.

FIG. 9. *NACIE-UP FPS bundle picture.*

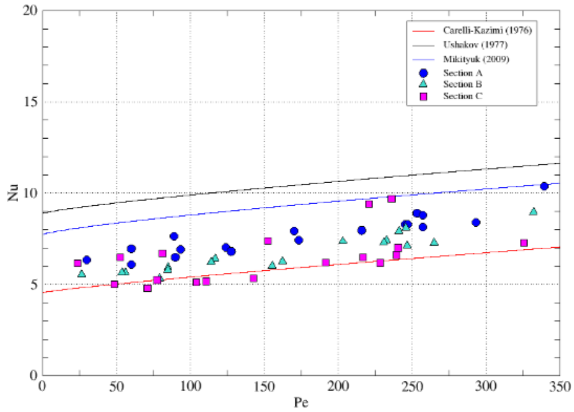

FIG. 10. Section-averaged Nusselt number Nu, for all test cases at sections A, B, C [24].

A lot of experimental data were obtained for the grid-spaced 37-pin fuel pin bundle of the large CIRCE pool facility, already introduced in previous sections, see FIG. 11 for a picture of the CIRCE open Fuel Pin bundle.

The experimental data obtained on the CIRCE pin bundle were compared with numerical simulations, and the results showed a good agreement with the values of the Nusselt number close to the Ushakov correlation [27] and to the CFD analysis, see FIG. 12. Additional data on the CIRCE bundle have been obtained also during integral tests and transients in several conditions and several experimental campaigns, see for example [15],[31],[18]; these data form a unique database for validation and verification of STH, CFD and coupled codes.

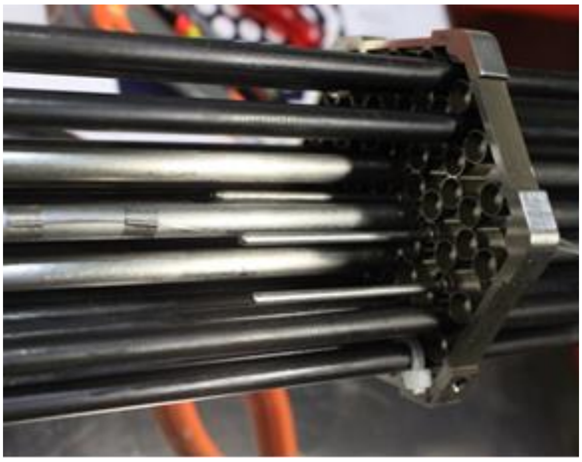

FIG. 11. CIRCE fuel pin bundle.

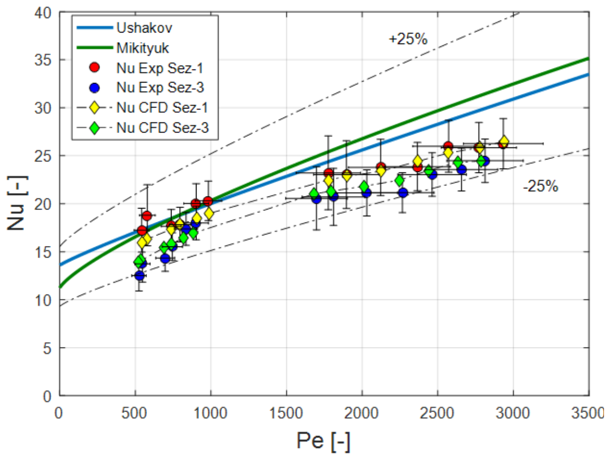

FIG. 12. Experimental and CFD Nu vs. Pe number and comparison with Mikityuk and Ushakov correlations [27].

5. SAFETY SYSTEMS THERMAL HYDRAULICS

Born in the context of R&D projects supported by the European community, ALFRED places safety in all operational and accidental conditions as its fundamental principle. A similar process was recently carried out to support the new revised configuration, where system codes and ad hoc experiments demonstrated the wide coolability safety margins. In addition and to strengthen the research infrastructure, the SIRIO facility is currently in the commissioning phase, devoted to the validation of the passive safety system for the delay of the primary coolant freezing, through a principle patented by Ansaldo Nucleare SpA [28][29].

Numerical analyses

The University of Rome in collaboration with the FALCON Consortium has developed a detailed model of the RCS inclusive of ALFRED's Decay Heat Removal (DHR) system [30]. The DHR consists of a set of isolation condensers that can be connected to the SGs in an accidental condition by opening isolation valves (one per SG). Non-condensable gases in the DHR allow the power transmitted to the external environment to be passively balanced to the decay heat by migrating from-or-towards the isolation condensers. The model, developed for RELAP5-3D [31] software, captures three-dimensional flow phenomena within the RCS while keeping the typical flexibility of system codes. The model was used to evaluate the operation of the DHR in the event of a Station Black Out (SBO) prolonged for a period of 72 hours (system intended grace time, that is, the period of time where no operator action is necessary). The results obtained allowed to validate how the DHR conceived for ALFRED is functional also in the revised configuration adopting single-wall SGs. FIG. 13 summarizes the trend of lead temperatures (hot and cold pools) during the first phase of the transient (about 1 hour) and for its entire duration (72 hours). The cooling phase lasting approximately 10 hours is followed by the maintaining phase where temperatures are kept above the solidification point with margin, thanks to the presence of non-condensables in the DHR system.

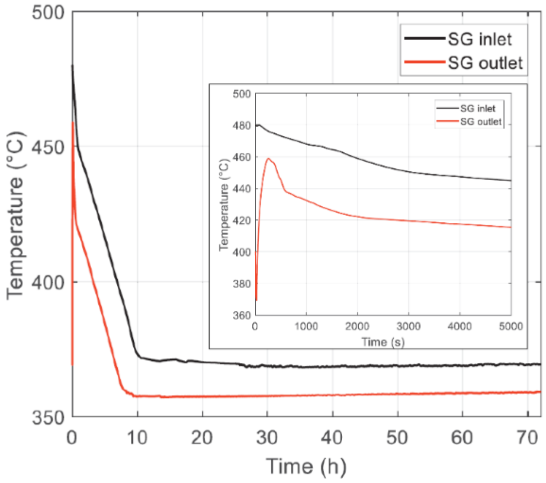

FIG. 13. Lead temperature across the SG during SBO.

Experiments

The blockage FPS (BFPS) test section installed into the NACIE-UP loop facility aimed to carry out suitable experiments to fully investigate the different flow blockage regimes in a 19 fuel pin bundle, providing experimental data in support of the development of ALFRED. A preliminary analysis of this topic on ALFRED FA can be found in [33]. The procurement and commissioning of the test section was enclosed in the framework of the H2020 project SESAME [11] on the thermal-hydraulics of liquid metals.

This fuel pin bundle configuration was relevant for the thermal-hydraulic design of the ALFRED core.

A proper experiment was designed in order to describe the thermal-hydraulic behaviour of a simplified version of FA during an internal flow blockage accident, simulated by blocking some holes of the first spacer grid (at the beginning of the heated length, and located at 1304.85 mm before the end of the test section) with appropriate caps.

The central spacer grid is the key component of the flow blockage experimental campaign. For the flow blockage configuration, several caps were displaced on the different holes of the central grid; those caps were small thin plates of an appropriate shape positioned by moving rods from the bottom to fix a configuration.

Because of the internal flow blockage, the temperature of the pins near the blockage increased due to a lower cooling rate in this region (FIG. 14). The extension and magnitude of these hot regions were strictly related to the blockage type and blocked flow area of the grid.

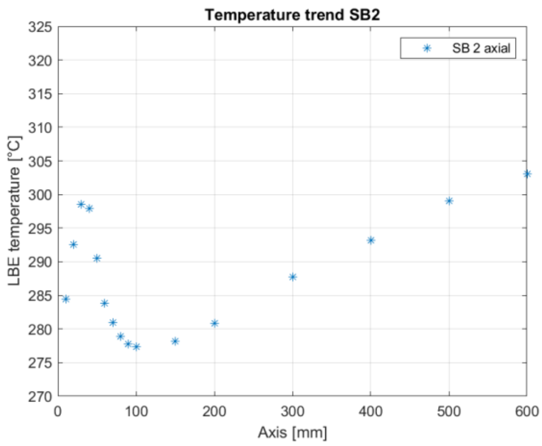

FIG. 14. Axial peak temperature trend in a blocked subchannel.

The experiments investigated the blockage on a spacer grid at the beginning of the active region. A proper pre-test analysis [34] allowed for fixing the bundle instrumentation in an improved way. The results showed a local maximum behind the blockage as seen in FIG. 14: the experimental results were unique and were published together with the associated CFD post-test analysis [35]. CFD simulations with RANS and URANS methods were qualitatively in agreement with experimental data. CFD overestimated the peak temperature, and the LES method is probably needed to correctly capture the flow features. The experiment created a data base to validate numerical methods able to compute internal flow blockage in the ALFRED FA.

In the framework of MAXSIMA project, the experimental campaign investigating Steam Generator Tube Rupture (SGTR) event was carried out in large integral effect test facility CIRCE, at ENEA CR Brasimone [37],[38].

The designed and assembled test section was conceived to perform four runs, investigating water-Lead Bismuth Eutectic (LBE) alloy interaction occurring at two different tube rupture positions (bottom and middle) of MYRRHA Primary Heat eXchanger (PHX). Two runs were carried out for each rupture configuration, for acquiring feedback on test repeatability.

Four full scale portions of MYRRHA PHX tube bundle were hosted in the test section (see FIG. 15), supported by CIRCE cover. This configuration was needed for performing four runs without removing the test section from CIRCE. The section was highly instrumented to acquire high quality data for transient analysis and code development and validation. About 220 signals were acquired for each test and post processed.

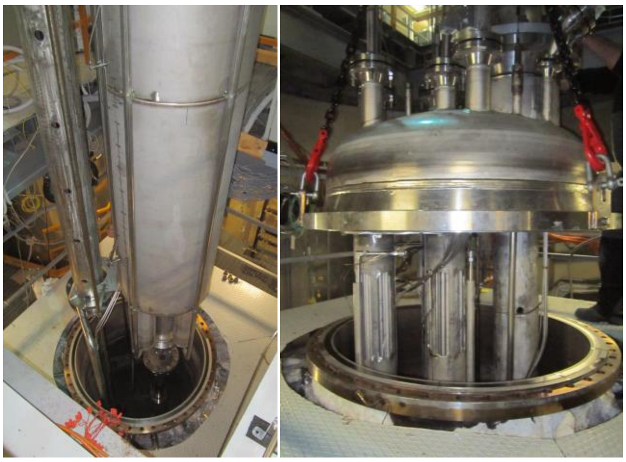

FIG. 15. CIRCE-SGTR test section.

The experimental results showed good test reproducibility. Each tube bundle was equipped with about 40 thermocouples (set on 7 levels) for tracing vapor flow path, which measured a minimum temperature of about 120-140°C in middle and bottom rupture positions, respectively. The maximum pressurization reached in CIRCE main vessel was about 4 bar. The opened rupture discs interrupted the pressurization at about 2.5 bar.

The post-test analysis was carried out by SIMMER-IV code. A 3D Cartesian model was developed conserving height and flowing areas of water and LBE regions. The main components were as much as possible reproduced in the model, compatibly with the coarse mesh character of SIMMER-IV code. The water-LBE interaction simulated in a large calculation domain as CIRCE facility was a very time demanding calculation. For this reason, only CIRCE main vessel was modelled and upstream and downstream components were substituted by experimental pressure time trends imposed as boundary condition in SIMMER-IV calculation. The obtained numerical results showed a good agreement with measured main vessel pressurization and temperature time trends (see FIG. 16).

The validation of ALFRED's main safety system for the residual heat removal under accidental conditions is the purpose of the SIRIO experimental facility, currently in operation at the SIET laboratories in Piacenza, Italy [39]. The system consists of a bayonet SG and an isolation condenser connected by means of piping and valves; both components are scaled with respect to ALFRED. In parallel to the isolation condenser, a bypass heat exchanger brings to the initial operating conditions. SIRIO operates in natural circulation conditions, both for steady state and test conditions. The expected results from this plant are used to validate calculation codes for the characterization of DHR phenomena and to qualify the operation of the patent for passive power control. FIG. 17 shows SIRIO cooling units, the isolation condenser pool (left) and the bypass heat exchanger (right). On the upper part two isolation valves allow to activate the first or the second condenser, while a third is the safety valve of the pressure assembly.

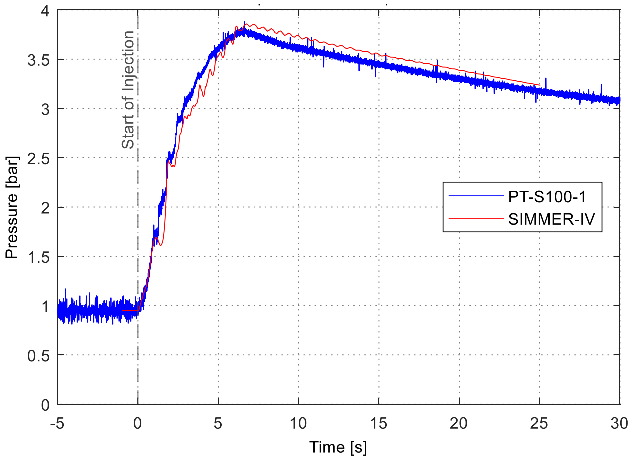

FIG. 16. Comparison of experimental and simulated pressure time trends in CIRCE cover gas (TEST-B).

FIG. 17. SIRIO cooling system.

6. OUTLOOK: A NEW RESEARCH INFRASTRUCTURE

The ALFRED project is heavily supported by the ENEA experimental infrastructure at the Brasimone research centre, to which several installations from supporting organizations are added. To complete the ALFRED validation process, it is necessary to provide additional experimental facilities that allow the collection of evidence to support the licensing of the plant. The FALCON consortium assessed the technological gaps necessary for the realization by defining a series of key facilities to be built at the RATEN-ICN site, Mioveni, and which represent the experimental constellation supporting ALFRED:

ATHENA - the Advanced Thermo-Hydraulics Experiment for Nuclear Application is a 2.21 MW non-isothermal multi-purpose pool representative of an LFR plant. The lead system is enclosed within a 3.2 m diameter and 10.0 m high vessel, to house a core simulator, a pump and a heat exchanger, as well as service and auxiliary systems. The plant has the purpose of testing scaled components for the qualification of calculation codes.

ChemLab - to support the Athena facility, a chemistry laboratory is planned in the same building to carry out interaction analyses between lead and structural materials under prescribed conditions of temperature and oxygen concentration. The laboratory is composed of an experimental section for the tests and a metallographic section for the analysis of the tests.

HELENA-2 - the HEavy Liquid metal Experimental loop for advanced Nuclear Applications (2) is a non-isothermal circuit to test the mock-up of the hottest fuel assembly as well as the absorbed devices provided in ALFRED (i.e. control and safety rods). The aim of the plant is to characterize the thermohydraulic of the fuel assemblies, including flow induced vibration as well as the dynamics of the control and shutdown rods.

ELF - the Electric Long-running Facility is a non-isothermal pool with a 10 MW multi-assembly core simulator, equipped with several steam generators, DHRs and primary pumps. ELF is the main ALFRED electrical demonstrator, designed to investigate thermal-hydraulic conditions under forced and natural flow, testing also the reliability of systems and components associated with the RCS, including oxygen and purification control systems.

Meltin'Pot - it is a hot cell with different experimental sections to test the interaction between the coolant, the fuel and the fission products, including tests for severe accident phenomena. It is designed also to validate the retention capability of fission products by lead.

Hands-ON – it is an isothermal facility to test the handling capacity of fuel assemblies and core shutdown devices, of full scale size compared to ALFRED.

To date, the Athena construction project has begun and the experimental infrastructure is in the detailed design phase, with procurement of the main components already underway.

7. CONCLUSIONS

With a strong impetus dictated by the need for clean, stable energy at a competitive price, there are countless organizations that carry out development projects for new advanced reactors. Of these, LFR technology stands out for its high background resulting from more than 20 years of research. The FALCON consortium is carrying out the development of ALFRED thanks also to several supporting organizations, and the technical activities have achieved to study various aspects of the thermal-hydraulics of liquid metals of precious value for the design of the plant. In this article, some of the most significant examples that have contributed most to define the state of the art have been reported, together with an indication of the future research and development trajectories, focused on the validation of calculation codes and ALFRED licensing, through the definition of a world class experimental infrastructure currently under construction.

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