Thermal-hydraulic Experiments Supporting the MYRRHA Fuel Assembly

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Abstract. The development of the LBE-cooled MYRRHA fuel assemblies is supported by an extensive thermal-hydraulic experimental program, as well as numerical studies. For the safety assessment of the reactor, several experimental campaigns considering fuel assembly mockups in representative operating conditions have been completed, and others are ongoing and planned at KIT (Germany) and SCK•CEN (Belgium). These are individually focused on specific issues, such as the heat transfer and pressure drop in nominal conditions, effects of local blockages and their formation, and influences of inter-wrapper flow between neighboring fuel assemblies. Heated tests using LBE, as well as isothermal studies with water as a model fluid, are considered.

This article summarizes the main results of completed projects, highlighting the accuracy of existing correlations, and the relevance of hot spots based on local temperature distribution both at the wall and in the fluid. Moreover, the status of ongoing work is presented and the main open thermal-hydraulic issues for supporting the development of the MYRRHA fuel assembly are identified.

Key Words: fuel assembly, LBE, experiment, MYRRHA

1. Introduction

The MYRRHA fuel assembly (FA) contains a hexagonal bundle of 127 cylindrical fuel pins surrounded by a hexagonal shroud or wrapper. At its ends, this shroud is connected to the inlet and outlet nozzles, guiding the upward lead bismuth eutectic (LBE) coolant flow through the FA. Like many liquid metal fast reactor (LMFR) designs, the MYRRHA FA design uses the helical wire spacer to preserve the spacing between the fuel pins. The main geometrical specifications and reference operating conditions of the MYRRHA bundle are listed in Table I.

TABLE I: MAIN GEOMETRIC SPECIFICATIONS AND NOMINAL OPERATING CONDITIONS OF THE MYRRHA FUEL BUNDLE.

Geometrical parameter	Symbol	Value	Operating condition	Symbol	Value
Number of fuel pins	N	127	Thermal power	Q	1.45 MW
Pin total/heated length	L	1.4/0.6 m	Mass flow rate	$\dot{m}$	71.4 kg/s
Pin pitch	P	8.38 mm	Mean velocity	u	1.9 m/s
Pin diameter	D	6.55 mm	Reynolds number	Re	48000
Wire spacer diameter	d	1.8 mm	Inlet temperature	T_{in}	270°C
Wire spacer pitch	H	262 mm	Outlet temperature	T_{out}	410°C

Compared to other LMFR fuel designs, some of the main parameters are within ranges known from previous experiences in sodium systems, e.g. a pitch-to-diameter ratio of 1.279. The main novelty is the use of LBE as the primary reactor coolant, leading to different thermal-hydraulic scenarios with specific challenges.

From a safety point of view, the FA contributes to three major functions: control of reactivity, heat removal and confinement of the radioactive products. For this reason, the FA behaviour must be thoroughly analysed in normal, anticipated operational transient and accidental conditions. In order to ensure the integrity of the fuel, sufficient cooling performance must be guaranteed in all situations by design. Supporting the licensing stage, experiments and prototype testing must be foreseen, particularly for a first-of-its-kind design.

Many European institutions participate in the safety assessment of MYRRHA in the frame of collaborative projects. This paper describes the experimental activities at KIT in Germany and SCK•CEN in Belgium covering several thermal-hydraulic aspects of the fuel assembly.

2. Experimental facilities for core thermal-hydraulic studies

In general, each test facility is optimized for studying specific phenomena. In particular, loop thermal-hydraulic facilities are designed to provide controlled boundary conditions (flow rate, inlet temperature and pressure and thermal power) to a test section. In this context, specific facilities are used for covering different ranges of these parameters.

The following sections describe the main LBE loop facilities used for the experiments described in this paper. For an overview of other existing LBE experimental facilities, the reader is referred to [1]. In some cases, water is used as a model fluid, see e.g. §4.2.

2.1. COMPLOT loop at SCK•CEN

The COMPLOT facility is a large-scale closed-loop LBE experimental facility, designed to characterize the hydraulic behavior of numerous MYRRHA reactor components at full-scale, in flowing LBE representative of the reactor core conditions.

Its vertical test port is designed to represent a single core channel, allowing for test sections of up to 10 meters in length with upward LBE flow. The loop is isothermal, and its temperature can be varied up to a maximum of 400°C. A wide range of flow rates (2-36 m^3/h) is obtained with a variable speed pump, combined with a throttle valve and bypass. Relevant to this paper, the COMPLOT facility is used to measure the pressure drop (§3.1) and flow induced vibrations (§4.3) in a full-scale 127-pin fuel assembly.

2.2. THESYS and THEADES loops at KIT

The THESYS loop was first built in 2001 in a horizontal arrangement for the development and testing of technologies adapted to an LBE environment, such as electromagnetic pumps (up to 16 m^3/h), instrumentation and electrical heaters. It was later updated to a vertical configuration for benchmarking tests. Currently, it is being updated with three independent power supplies and four parallel flow channels within the SESAME project, see §4.4.

For campaigns requiring higher thermal power and/or flow rates, the THEADES loop can be used, thanks to its larger air cooler (up to 500 kW) and centrifugal pump (up to 47 m^3/h). With these parameters, it is possible to test key components of LBE systems with prototypical dimensions and under reactor-like conditions of operating temperature, velocity and heat-flux density. The maximum hot leg temperature in both loops is 450°C.

3. Selected recently completed studies

The experimental results of two selected campaigns are described in the following sections, covering both isothermal and heated tests.

3.1. Pressure drop in a full-scale 127-rod bundle

Pressure drop measurements of a full-scale mock-up of the MYRRHA 127-rod fuel assembly were performed in the COMPLOT LBE facility at SCK•CEN. At a constant temperature of 200°C, a wide range of flow rates (from 13% to 100% of maximum) was tested in steady state. Further details of the experimental setup can be found in [2].

The differential pressure was measured at a selected edge sub-channel over two distances simultaneously, namely two and three times the wire-wrapper pitch (H). In particular, the measuring positions are located between the axial heights of 1.75H and 3.75H, and between 1.75H and 4.75H, respectively.

These measurements were used to determine an average pressure drop over one wire pitch, which in turn were used to determine a bundle friction factor (f). Figure 1 illustrates the experimentally determined friction factors as a function of the bundle mean Reynolds number, compared with numerous published friction factor correlations, i.e. Rehme [3], Cheng and Todreas [4], and Baxi and Dalle Donne (BDD) [5]. Furthermore, isothermal data from a campaign in THEADES in a smaller bundle (see §3.2) are added for comparison.

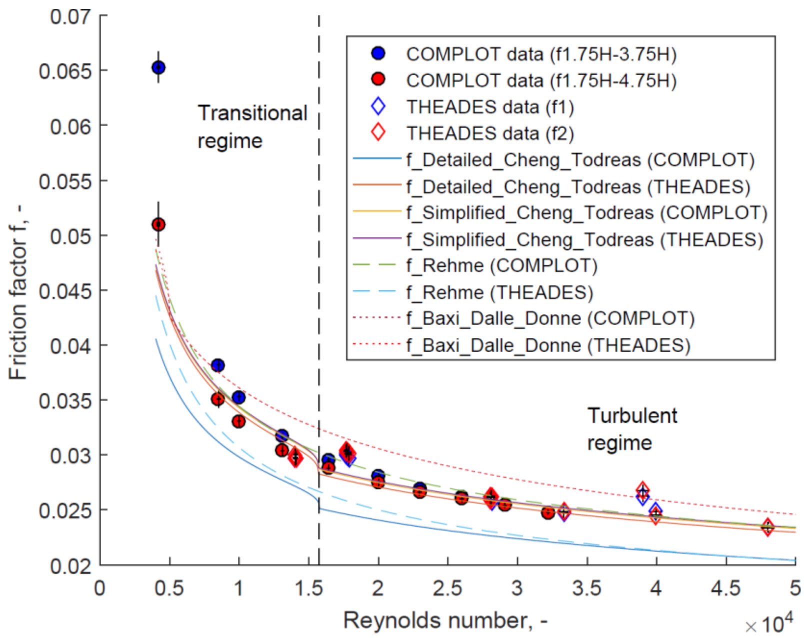

FIG. 1. Measured and predicted pressure drop across the 127-pin MYRRHA fuel bundle

Two flow-regime boundaries were proposed by Cheng and Todreas [4] to define the laminar to transition (Re_L) and the transition to turbulent (Re_T) Reynolds numbers respectively, as shown in Figure 1. In principle, sharp transitions are not found in the experimental data.

In the turbulent regime, particularly for Re>20 000, the experimentally determined friction factors are almost identical for the two measurements across different axial lengths. Within the transition flow regime, however, both datasets diverge and the friction factor in the shorter section is 4-23% greater than in the larger section. This difference is much larger than the experimental uncertainties and it can, arguably, be related to the flow development within the bundle. It should be noted that the onset of turbulence usually reduces the entrance length and thus it could explain why this difference is reduced at larger Re within this range.

A statistical analysis of the prediction performance of each correlation is possible.

- The Rehme [3] correlation shows excellent agreement with the experimental data, slightly over-predicting the friction factor consistently by 1-2%, except for the lowest flow case.
- The Baxi and Dalle Donne [5] model also performs well, over-predicting the friction factor consistently by ~5%, except for the lowest flow case.
- The detailed model by Cheng and Todreas [3] is the worst performing one among the correlations considered, contrary to recent evaluation by Chen et al. [6], where it was consistently ranked first. Better results are obtained with the simplified model.
- Although some models predict large differences between both geometries (127 rods at COMPLOT, 19 rods at THEADES) this is not observed in the experiments.

Investigations are being continued to assess the effect of LBE temperature and LBE conditions on the measured pressure drops.

3.2. Heat transfer in a scaled 19-rod bundle

Within the European project SEARCH, a representative wire-wrapped bundle was tested in the THEADES loop at KIT. For practical reasons these electrically-heated tests considered a lower number of rods and a larger rod diameter (8.2 mm). All other parameters have been scaled accordingly, leading to the same pitch-to-diameter ratio, heat flux density and range of non-dimensional flow parameters, e.g. Reynolds and Péclet numbers.

Detailed instrumentation could be implemented at the heated wall and selected sub-channels, thanks to the use of slightly larger pins. Three measuring levels (MLs) in the heated region are equipped with 23 thermo-couples (TCs) each (69 total). In particular, 18 TCs (0.5 mm) are inserted in grooves in the heater cladding and 5 TCs (0.25 mm) are placed at the center of selected sub-channels, at each ML. Furthermore, differential pressure is measured at two locations; and a Venturi nozzle at the inlet supplements a Vortex flow meter (±0.75%), particularly at low velocities. A sketch of this arrangement is shown in Figure 2.

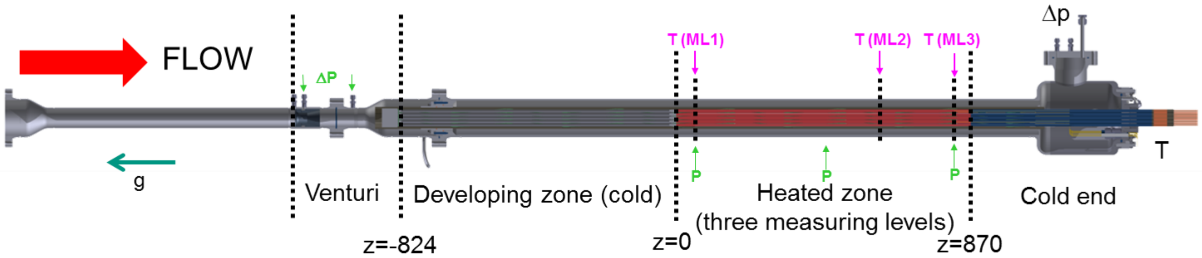

FIG. 2. Side view of the heated 19-rod bundle test section at KIT, indicating the instrumentation [9]

In order to improve the scientific quality of the measurements, several in-situ calibration and correction steps are performed, including the following.

- Differential pressure is corrected by the temperature-dependent hydrostatic contribution
- All TCs are calibrated simultaneously, in isothermal tests with large flow rates, and using the inlet temperature as a reference, leading to a relative precision of ±0.1 K, that is much better than the absolute accuracy reported by the manufacturer (1.5 K).

- As the wall thermo-couples are located inside the cladding, the observed value is larger than the temperature at the outer surface, and this systematic overestimation can be significant for liquid metal experiments, see e.g. [7]. A correction is applied considering the thermal conduction through the wall.

This data-evaluation procedure has also been tested in previous experiences, e.g. in a bundle with grid spacers [8], leading to an improved reliability of the measurements.

Regarding the pressure-drop results, the friction coefficient f(Re) is best predicted by the simplified model of Cheng and Todreas [3], with an RMS error of only 3.3%. This result agrees with a review in [6], but it opposes the observations described in §3.1. Thus, it can be argued that the friction coefficient is slightly affected by the size of the bundle.

At each ML, an experimental Nusselt number (Nu) is derived based on difference between the average measured wall temperature and estimated bulk temperature, and compared with several empirical correlations, as shown in Figure 3 (left).

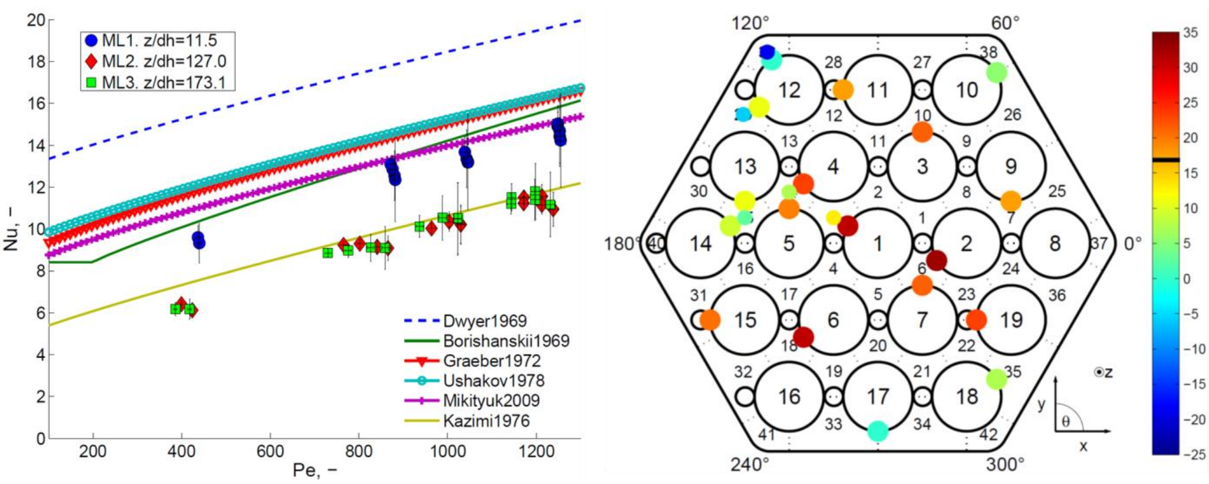

FIG. 3. Experimental heat transfer results in a 19-rod bundle. Left: Nusselt number, compared to several correlations. Right: local temperature above bulk (T-T_b) for a reference case [9]

Disregarding the data at ML1 (not yet fully-developed) best results are obtained with the correlation by Kazimi and Carelli [10], which over-predicts the data by 5.2% on average. This correlation is actually suggested by the authors as a most conservative estimation for bare rod bundle (without spacers), for P/D between 1.25 and 1.30.

The even lower Nu can be attributed as an effect of the wire spacer themselves, analyzing the temperature profiles at a given measuring level, as shown in Figure 3 (right). The wires produce a directional sweeping and affect crossflow between sub-channels. As the inner regions remain relatively hot, the mean wall temperature overheat $T_{wall,mean}$-T_{bulk} (marked as a black line in the color legend) is higher, and Nu is lower than without spacers. Moreover, at each ML large temperature differences are observed. At the hottest spot (central rod), the wall overheat above bulk (T_{wall}-T_{bulk}) is almost twice its mean value ($T_{wall,mean}$-T_{bulk}).

This experimental information is of great value for the design team which must consider temperature thresholds during the irradiation of the fuel assemblies. In addition to confirming the validity of existing empirical correlations for the MYRRHA conditions within a proper uncertainty range, the importance of accounting for local hot spots (also for nominal operating conditions) is highlighted. Considering the limited resolution possible with TCs, it is essential to compare these results with numerical simulations (see [9]) and similar experiments.

Also within the SEARCH project, a 19-rod bundle with smaller rods (6.55 mm) but the same P/D was tested at ENEA in Italy, covering a lower range of Re [11]. Instrumenting different sub-channels, experimental values of Nu laying between the correlation of Kazimi and Carelli [10] and that of Ushakov et al [12], recommended in [1], were obtained. This configuration is being considered for an international CFD benchmark, see e.g. [13].

4. Ongoing work

The safety assessment of the MYRRHA fuel assembly continues within several ongoing European projects, including thermal-hydraulic experiments at different stages of planning, construction, measurements and evaluation. For some of the experimental campaigns described in the sections below, the first results are expected during 2017.

4.1. Heat transfer effect of local blockages

In FAs with wire spacers, solid particles transported by the coolant (e.g. lead oxide is relevant for LBE) can accumulate in some sub-channel and form long, local blockages. Although these can be too small to have a global effect on the pressure drop or flow rate, they can lead to local clad failure, particularly if the foreign material has a low thermal conductivity.

Within the MAXSIMA project, the heat transfer effect of local blockages is investigated, inserting two blockage elements (C1 and E1) in a heated 19-rod bundle setup, based on the setup described in §3.2, as shown in Figure 4.

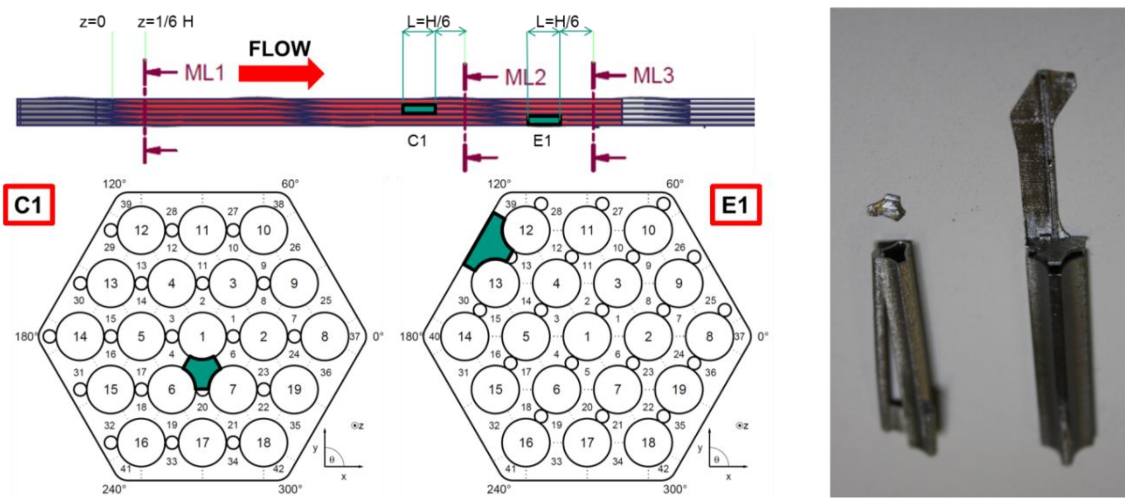

FIG. 4. Local flow blockages in a 19-rod bundle studied at KIT in the MAXSIMA project [14]

Both elements have the same length (H/6), but different shape, they are formed as central and edge sub-channel. In order to simulate worst-case conditions, they are manufactured as a thin-walled (0.2 mm) steel shell, filled with a pourable ceramic of low thermal conductivity (ca. 1.0 W/Km). The instrumentation is focused on the hottest location at the wall, and some additional TCs in the wake region are included for validation of CFD simulations. In a second campaign, a larger central blockage covering six sub-channels (C6) shall be studied.

4.2. Blockage formation and growth

While a fixed blockage size was selected for the heated test described in §4.1, the expected blockage geometry is not yet completely known, and clad failure can presumably occur if they grow large enough. The process of formation and growth of blockages is dominated by the spacer concept, see Figure 5. In FAs with wire spacers, blockages tend to grow axially, but there is no information available regarding the speed and extent of this process [15]. In particular, most materials are lighter than LBE and there is no precipitation as a counter-acting mechanism.

Within the MYRTE project, this process is studied in a water mockup at KIT, with the advantage over LBE that optical instrumentation, as shown in Figure 5, can be used. In order to extrapolate the results to LBE conditions, non-dimensional similarity criteria, considering buoyancy, inertial and viscous forces, are considered and numerical simulations are validated.

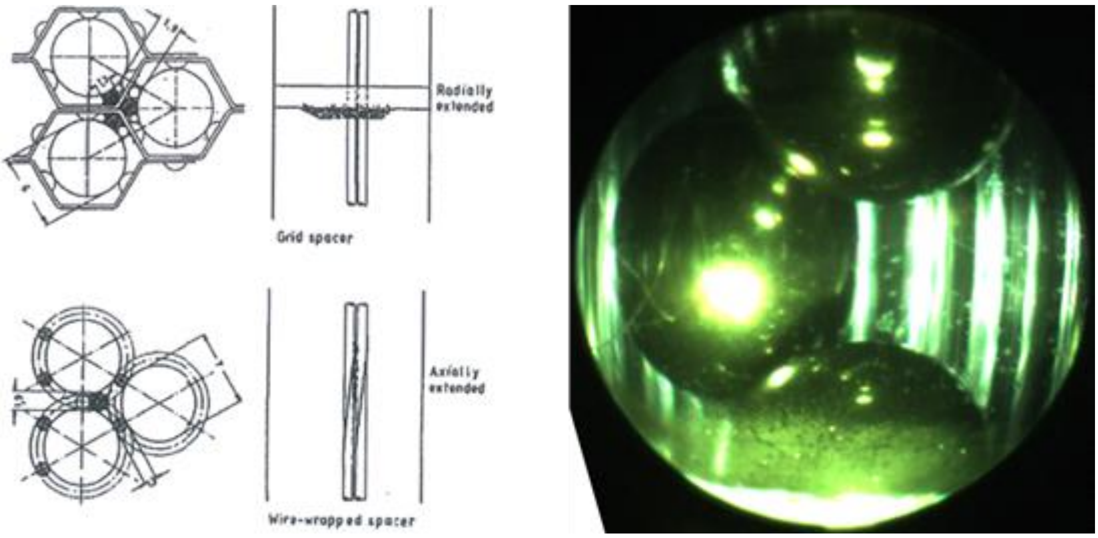

FIG. 5. Process of blockage formation and growth by accumulation of particles. Left: effect of grid and wire spacers [15]. Right: preliminary results of optical instrumentation at KIT

4.3. Flow-induced vibrations

Due to the higher density of LBE, flow-induced vibration can produce larger forces than in water and thus induce larger damage to the cladding. As a first step, towards assessing this damage, the vibration characteristics are investigated experimentally at SCK•CEN. The use of optical fiber Bragg grating (FBG) based sensors was successfully demonstrated at SCK•CEN [16] for vibration measurements in a 7-pin reduced-length mock-up of the MYRRHA fuel assembly, installed in an LBE filled installation, see Figure 6.

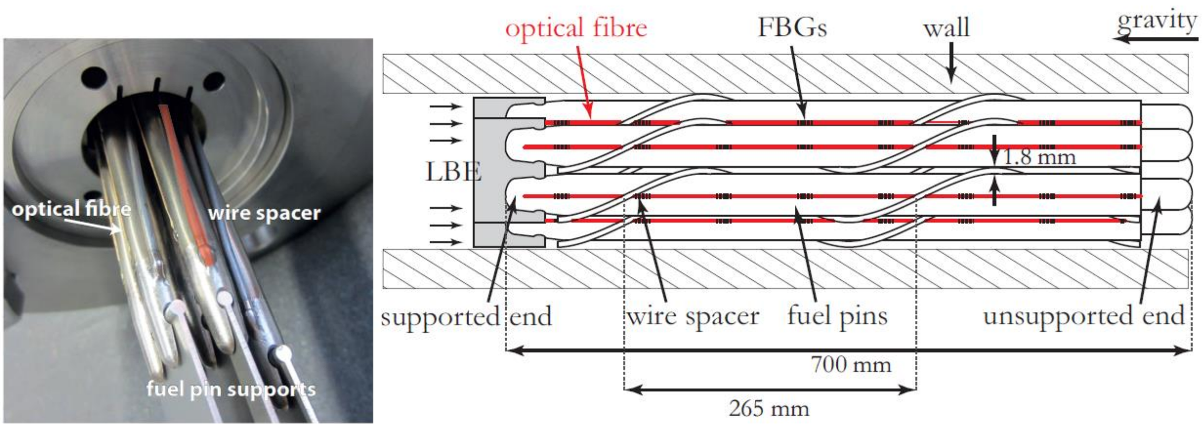

FIG. 6. Seven-pin, reduced-length FA instrumented for flow-induced vibrations at SCK•CEN [16]

The instrumentation was successfully tested in LBE, with minimal intrusion or disturbance to the FA geometry. The fuel pin vibration amplitudes and associated modal characteristics can be extracted for various flow conditions. This 7-pin study will shortly be extended to an experimental flow induced vibration campaign using a full-scale mock-up of the MYRRHA 127-rod FA mounted in the COMPLOT facility (§2.1) as part of the MYRTE project. The knowledge and understanding of these vibrations is necessary to estimate the lifetime of the fuel bundle with regard to structural integrity and mechanical wear.

4.4. Inter-wrapper flow and heat transfer

In most out-of-pile heated tests, single FAs are considered as units isolated from its neighbors and adiabatic boundary conditions are imposed. Although conservative, this assumption can be too restrictive for some scenarios, such as an edge-type blockage. In a reactor core, cooling across the wrapper tube wall is given by a natural-circulation flow between sub-assemblies, i.e. the inter-wrapper flow (IWF), particularly under conditions of passive decay heat removal.

Previous sodium experiments in Japan [17] were focused on the pool dynamics of the IWF, and these results can be used for estimating the expected mean velocity in the gap. In the frame of the SESAME project, the effects of this flow and heat transfer on the temperature distribution shall be studied in a multi-bundle LBE experiment, as sketched in Figure 7.

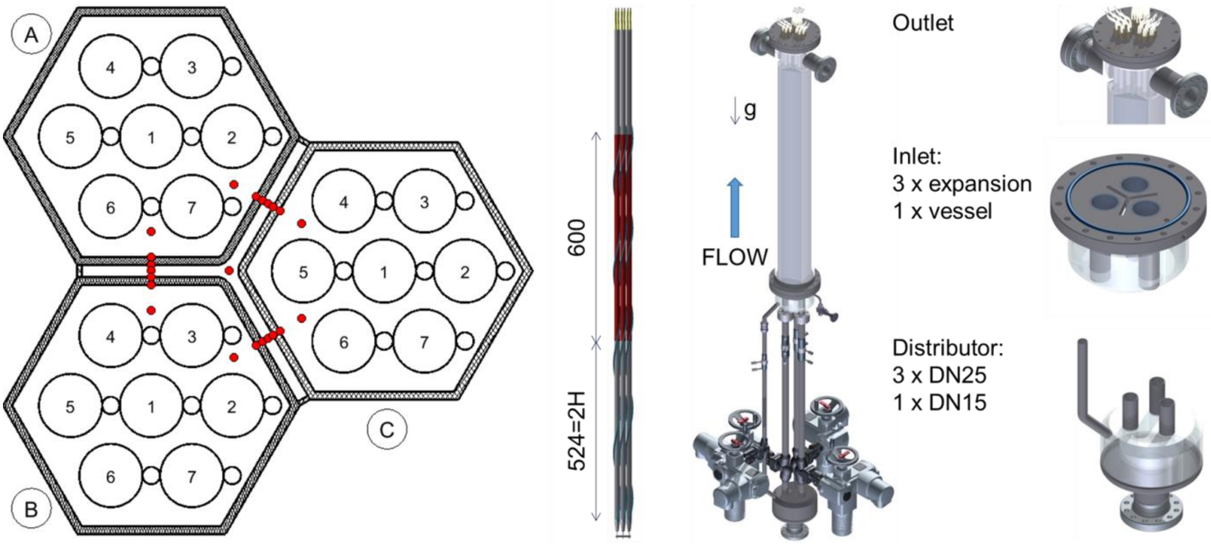

FIG. 7. Inter-wrapper flow test section with three 7-rod bundles at KIT

The main focus is placed on the gap region, which resembles the geometry in the MYRRHA core. On the other hand, the bundles are simplified with 7 pins and they provide appropriate boundary condition to the gap region, where most thermo-couples are installed. For this campaign, the THESYS loop is upgraded to accommodate four parallel flow channels and three independent power supplies, as shown in Figure 7. In this campaign, both symmetric and asymmetric scenarios shall be investigated, with a parametric study of the IWF velocity.

5. Outlook

The development and safety assessment of the MYRRHA reactor is supported by extensive research programs including many international collaborative projects. This article covers the experimental activities at KIT and SCK•CEN studying the fuel assembly thermal-hydraulics.

Such thermal-hydraulic studies play a key role in evaluating the temperature evolution in the core in postulated scenarios. Experimental investigations are a necessary step (supplementary to numerical simulations) for supporting the licensing stage, particularly considering the introduction of innovative systems of components, such as using a heavy liquid metal (LBE) as coolant, as limited lessons can be learned from previous experiences.

In this work, the main lessons learned and selected results are presented. The validity of empirical correlations, developed for other fluids, to the MYRRHA conditions has been confirmed for a nominal geometry. These experiments include the pressure-drop tests of a full-scale 127-rod bundle, and the heat transfer in a scaled 19-rod FA mockup, in both cases at representative conditions of steady-state forced-convective operation. These results provided valuable feedback to the reactor designers.

Ongoing activities are focused in the MAXSIMA, MYRTE and SESAME projects which extend this assessment to non-nominal conditions. Local blockages are a postulated scenario which can lead to clad failure, and this issue is investigated from different perspectives. First, the temperature distribution is measured for selected worst-case blockage conditions in a 19-rod bundle installed in the THEADES loop. Second, the process of blockage formation and growth is studied using water as a model fluid with optical instrumentation, respecting all relevant non-dimensional similarity criteria. Furthermore, flow-induced vibrations are studied in order to assess their effect on the mechanical integrity of the FA. Considering the FAs as a non-isolated part of the core, the contribution of inter-wrapper flow cooling towards reducing hot spots is tested in a setup with three bundles and four parallel flow channels.

The outcome of these ongoing works is expected to greatly contribute to assessing the safety features of the current MYRRHA fuel assembly design and facilitate the licensing process. Some issues remain open and can be the topic of future projects. For example, while most studies have been focused on beginning-of-life conditions, the FA geometry can deform during irradiation, e.g. due to swelling and creep. A sound understanding of the consequences on the heat transfer process is essential for extending the safety evaluation to the complete lifecycle in the core. Furthermore, steady-state correlations are sometimes applied beyond the conditions for which they were developed, for example for transient scenarios or non-uniform power distributions. In some safety-relevant scenarios, it might be necessary to confirm the FA integrity experimentally.

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