Reactor Physics and Thermal Hydraulics Analyses for the OECD/NEA MPCMIV Benchmark

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ABSTRACT

In order to complete the Multi-physics Pellet Cladding Mechanical Interaction Validation (MPCMIV) benchmark technical specifications, reactor physics and thermal hydraulics analyses have been performed. The work presented in this paper aims in particular to evaluate some of the missing Boundary and Initial Conditions necessary to complete the technical specifications, and also to perform some of the benchmark exercises connected with thermal hydraulics simulations. A far as the thermal hydraulics area is concerned, the analysis is carried out with the RELAP5 code. It is focused on the modelling of the in-pile loop 1 located inside the R2 reactor core, in which a test fuel rodlet is inserted to perform some power ramp tests. The activity consists in the development of the simulation model of the in-pile tube, the demonstration of the steady state achievement and the transient analysis of the first selected test, validating the simulation results against the benchmark experimental data. Considering the reactor physics area, the Monte Carlo code Serpent 2 is used to perform some single assemblies burn up calculations. The aim is to evaluate the initial composition of the fuel assemblies loaded in the core loadings of interest of the benchmark. Moreover, the temperature values to be used in the Serpent simulations are derived with thermal hydraulics simulations of the single assemblies. Further developments of the work will include the full core cycle analysis to validate the isotopic compositions and the complete model of the main circuit, using the gamma heating from the reactor physics calculations. Finally, the TRANSURANUS fuel performance code will be adopted to compare the results against the available experimental data.

A multi-physics effort is required to carry out the MPCMIV benchmark and appropriate coupling approach will be investigated and tested against the benchmark experimental results.

Keywords: *MPCMIV, Multi-physics, RELAP5, Serpent, Validation*

1 INTRODUCTION

An important branch of the Modelling and Simulation (M&S) codes capabilities, that has shown an increasing potential in the last years, is the high fidelity, multi-physics context. Differently from single physics analysis, multi-physics simulation can provide more accurate responses when evaluating complex system behaviour, capturing feedbacks that are not modelled when code coupling is replaced by boundary conditions. A real important aspect of this improvement is that the increasing fidelity and sophistication of coupled multi physics M&S tools need to be properly validated against experiments of the underpinning models and data, which represents a big issue. In order to address the specific challenges with the validation of high fidelity, multi-physics M&S tools, in 2014 it was created the Expert Group on Multi-physics Experimental Data, Benchmarks and Validation (EGMPEBV) of the OECD NEA. Under the guidance of this

group, the Nuclear and INdustrial Engineering (NINE) company in co-ordination with Studsvik has organized from 2017 the Multi physics Pellet Cladding Mechanical Interaction Validation (MPCMIV) benchmark [1]. The MPCMIV benchmark is based on some ramp test experiments conducted at the Studsvik R2 reactor that require the coupling of Reactor Physics (RP), Thermal hydraulics (TH), and Fuel Performance (FP) analyses to reach high fidelity simulations. A significant effort of the benchmark team went into the initial phase, in order to compensate for the lack of information related to certain areas. Nevertheless, a large number of Boundary and Initial Conditions (BICs) necessary to face the different benchmark exercises is still missing. The work presented in this paper describes the Reactor Physics and Thermal hydraulics analyses that have been conducted over the MPCMIV benchmark to complete the technical specifications through the evaluation of some missing BICs, and also to perform some of the benchmark exercises connected with Thermal hydraulics simulations.

For the thermal hydraulics area, the main goals are: to model the U tube used to perform the power ramp tests; to demonstrate the steady state achievement of the different hydrodynamic configurations of the model; to simulate the transient of the first selected power ramp test. All the thermal hydraulics analyses have been performed through the use of the RELAP5 code [2]. Regarding the reactor physics area the Monte Carlo code Serpent 2 [3] is used to perform some single assembly burn up calculations; the main goal of these calculations is to complete the technical specifications with the initial fuel composition of the different fuel assemblies loaded into the core loadings of interest of the benchmark.

The following sections are subdivided as follow. Section 2 reports the simulation models used for the analyses. The results are presented in Section 3. At the end, in Section 4, some conclusions are drawn, and future works are presented.

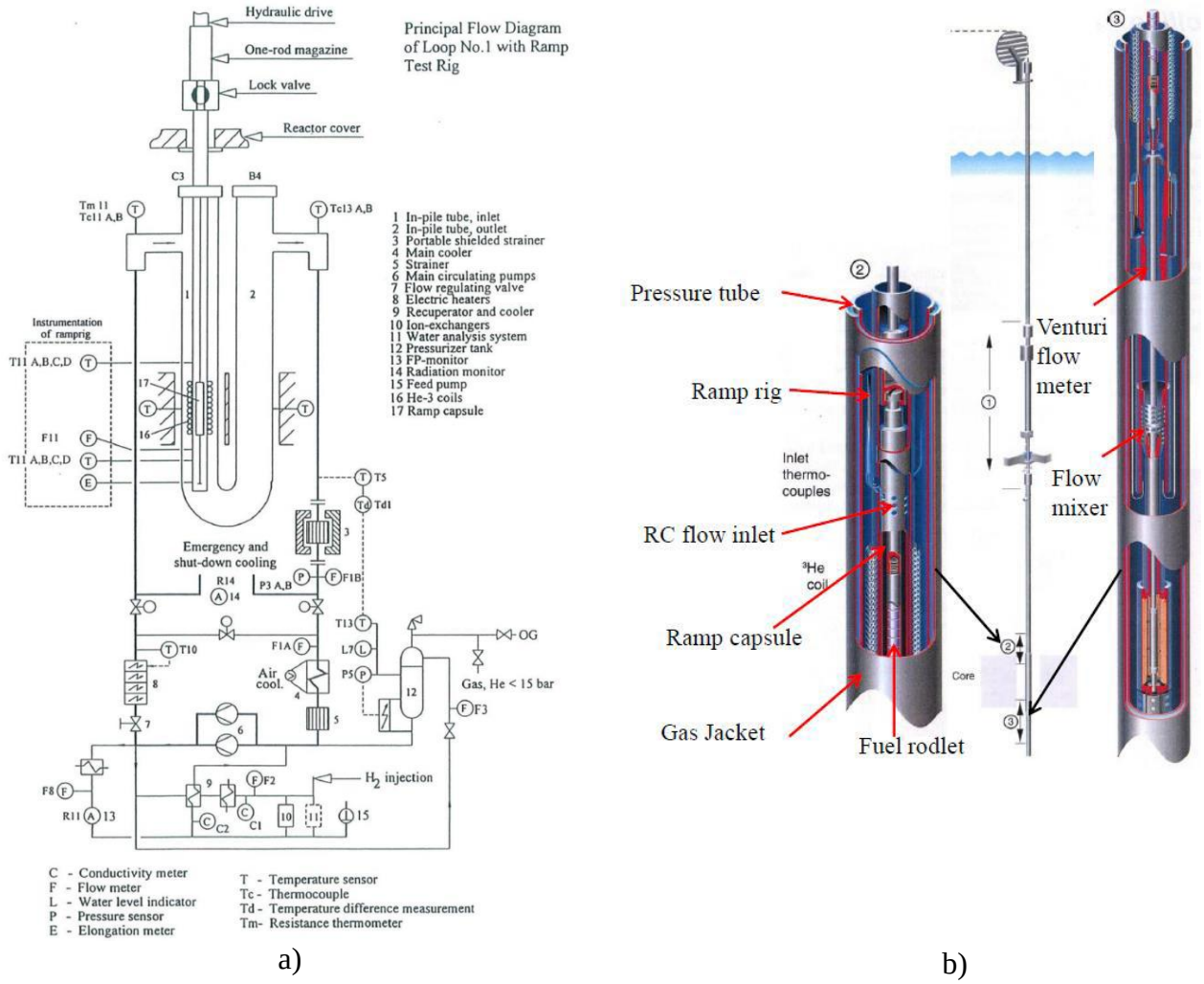
2 SIMULATION MODELS

Both the Simulation Models (SMs) are built following high fidelity modelling criteria, which have the aim to keep high detail level on both geometrical and physical parameters, to reduce the number of approximations. The next subsections report the thermal hydraulics model used for the analysis of the benchmark and the reactor physics model. Moreover, the latter contains a paragraph that describes the thermal hydraulics model of the single assemblies built to support the reactor physics analysis.

2.1 Thermal hydraulics Model

The thermal hydraulics model focuses on the in-pile loop number 1 of the R2 reactor. The loop is able to simulate PWR and BWR normal operation temperature and pressure conditions and it is thermally insulated with respect to the R2 core through a gas jacket containing CO_2. The in-pile loop is used to perform several power ramps tests to study the fuel response at cold critical conditions, i.e. below 100°C. This is done since the cladding mechanical properties and potential failure mechanisms can be very different with respect to those at normal operation.

The experimental facility is reported in Figure 1. A small section of the U tube occupies two adjacent positions (C3 B4) in the 10x8 core lattice, since it is much longer than the core. The inlet and outlet pipes of the U tube are connected to a main circuit which is made by several components and piping between them, as shown in Figure 1 a). The main circuit is not modelled in the present work since too much information is missing about the pressure losses over its components. For this reason, it was decided to focus first on the U tube only and to reach the steady state for it, and then to pass step by step to the completion of the main circuit model. The ramp test facility is inserted in the descending side of the in-pile tube (see Figure 1 b).



a) b)

Figure 1: a) In pile loop 1 flow diagram; b) Detail of the power ramp test facility

It is made by the ramp rig, in which it is inserted the ramp capsule that hosts the test fuel rodlet. The ramp rig is equipped with a Venturi flow meter to measure the mass flow rate in this section and with a flow mixer to enhance the flow mixing just before the temperature measurement of this section through four thermocouples. All these components are cooled by the flow of water which enters inside the ramp rig and the ramp capsule through a series of holes and exits from the open-ended bottom of the structures.

The thermal hydraulics analysis of this system is carried out using the RELAP5 code. The nodalization of the in-pile tube (pipe 101-103) with the inlet and outlet connection pipes (pipe 702-701), the ramp rig (pipe 201-203) and the ramp capsule (pipe 300) has been developed and it is shown in Figure 2. It was performed adopting the "slice technique" and nodalization criteria by NINE. Two possible hydrodynamic configurations are considered during the test:

- Configuration α: computational domain of the ramp rig facility, without the ramp capsule inserted in it (empty ramp rig facility);
- Configuration β or γ: computational domain of the ramp rig facility which is filled with the ramp capsule containing the calibration rod (β) or the fuel rodlet (γ).

The boundary conditions are given in terms of temperature and mass flow rate at the inlet, and pressure at the outlet. These values are taken from the experimental data of the sensor's measurements over the loop.

The system is thermally insulated with some heat structures that simulate the high-pressure tube contained in the gas jacket (HS-101, HS-103). The ramp rig and the ramp capsule are connected to some heat structures to simulate the convective heat exchange between the annular

flows (HS-201, HS-300). Finally, several fictitious heat structures with very high thermal conductivity, very low heat capacity and imposed power, simulate the energy deposition of the gamma rays in the region where the system crosses the core. Outside the core region this gamma heating contribution is assumed equal zero. The gamma heating power is supposed to be derived from a neutronic calculation, but the reactor physics model of the core is currently not available. So, the power contribution in the different sections is derived from a power balance, using the experimental data. First it is computed the power deposited in the whole loop through the formula (1).

(1)

(2)

where is the power deposited in the whole loop, is the mass flow rate circulating in the whole loop, is the specific heat at constant pressure of the loop and and the outlet and inlet temperature of the U-tube, respectively; is the power deposited in the test section only (i.e., along the ramp rig), the mass flow rate fraction passing in the ramp rig, the specific heat at constant pressure in this specific region and and the outlet and inlet temperatures of the ramp rig.

The difference between (1) and (2) gives the power deposition in the descending and ascending side of the high-pressure tube. It is distributed in these two sides proportionally to their flow area. As a result, a fraction of 0.38 goes to the descending side and the remaining 0.62 to the ascending side. Finally, the power of the test section is all given to the ramp rig in case of configuration α, while it is distributed between ramp rig and ramp capsule in case of configuration β and γ. The final distribution is again performed on the base of the flow area: a fraction of 0.25 goes to the ramp rig and 0.75 to the ramp capsule. At the end, the power in each section is uniformly distributed over the 7 axial nodes of the heat structures.

Once that the model is completed, its fidelity with respect to the real system and capability to reproduce the pressure, temperature and mass flow rate conditions of the in-pile tube are demonstrated. In particular, one calibration is needed for each hydrodynamic configuration. The first one is the β or γ configuration.

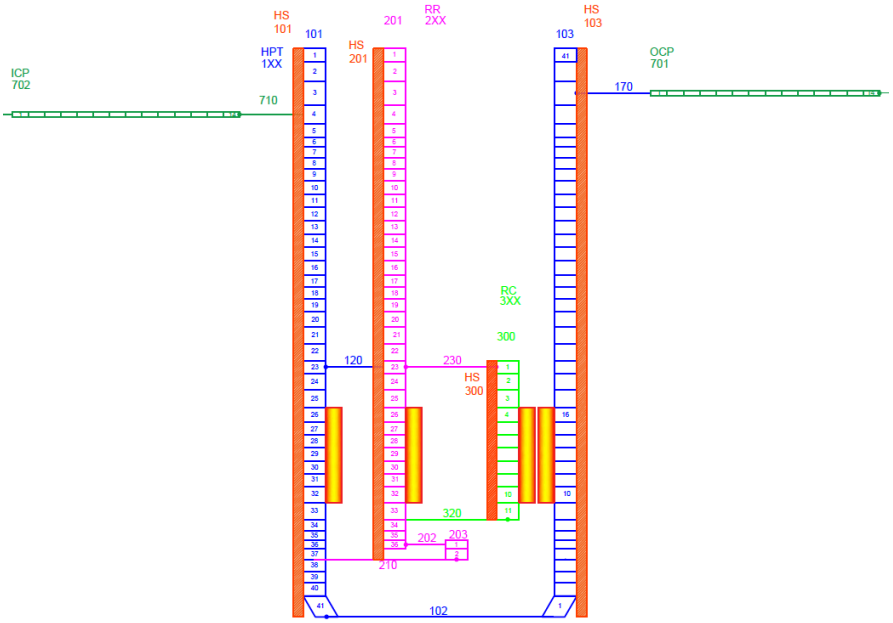


Figure 2: In pile loop 1 nodalization

The reference experimental conditions, against which the model is validated, are those at the beginning of the power calibration period of the test, shown in Figure 3. This graph reports the evolution in time of the reactor power (PR2NEU) and of the ratio of the rig over loop mass FLOw RAte (FLORA). The BICs taken from the experimental data in that instant are imposed. The conditions that can be checked at the end of the steady state simulation are the values of the temperatures in the Control Volumes (CVs) that correspond to the thermocouple location and the mass flow rate distribution inside the ramp rig, that is measured, and inside the ramp capsule, that is not measured, but of which is available at least an indication about its distribution: a fraction of 0.8 of the total test section mass flow rate goes inside the ramp capsule and 0.2 in the annular region inside the ramp rig. A null transient simulation of 1000 s is run to reach the steady state conditions at zero reactor power for this first configuration. Some assumptions are needed in order to properly calibrate the input especially in terms of pressure losses so that the flow distributes in the correct way inside the different components.

Once the first hydrodynamic configuration is validated, the same operation is performed also for the second one, i.e., the configuration α. This time the imposed BICs refers to the start of the pre-ramp 1 period shown in Figure 4, which represent the evolution of power and FLORA until the end of the ramp test 1. Since the only "new" components of this configuration are the ramp capsule and its connections to the rest of the system, only the localized pressure loss coefficients (k losses) associated to these ones can be calibrated, while all the other components are fixed to the value already calibrated. In this way, the steady state conditions are reached also for this configuration.

This preliminary operation is very important since it allows demonstrating the fidelity of the thermal hydraulics Simulation Model (SM) with respect to the real system. If this step is skipped, considering how many assumptions are needed to properly model the complicated geometry of the in-pile tube, it would be impossible to correctly simulate the ramp test.

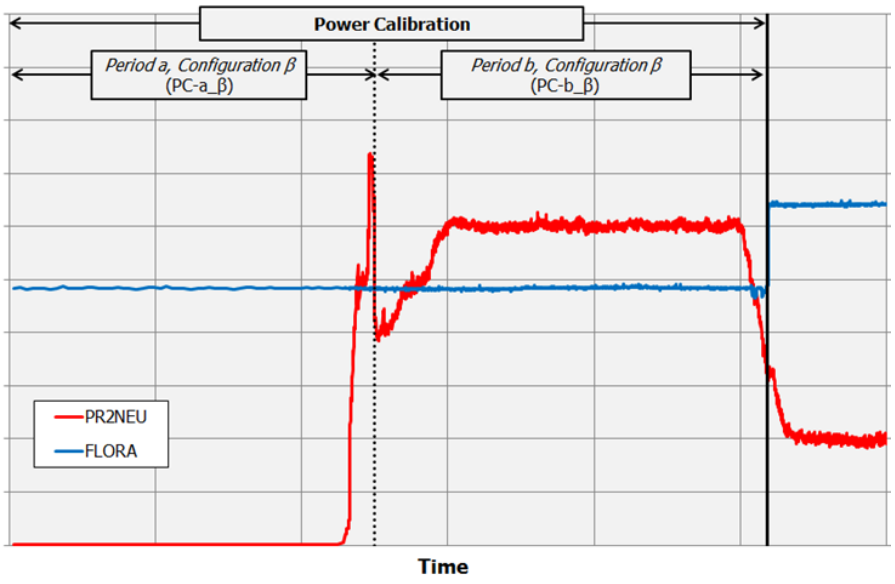


Figure 3: R2 Power and FLORA during power calibration of ramp test 1

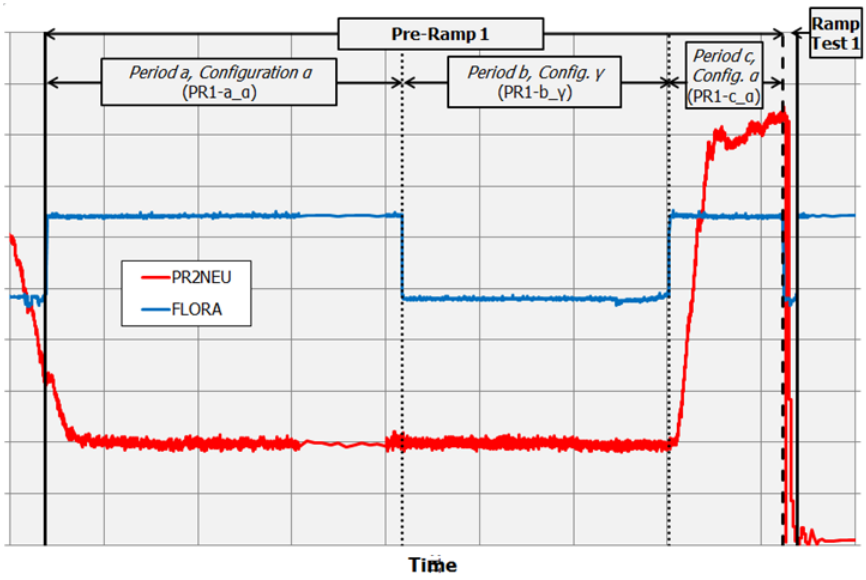


Figure 4: R2 Power and FLORA during pre-ramp 1 and ramp test 1

2.2 Reactor Physics Model

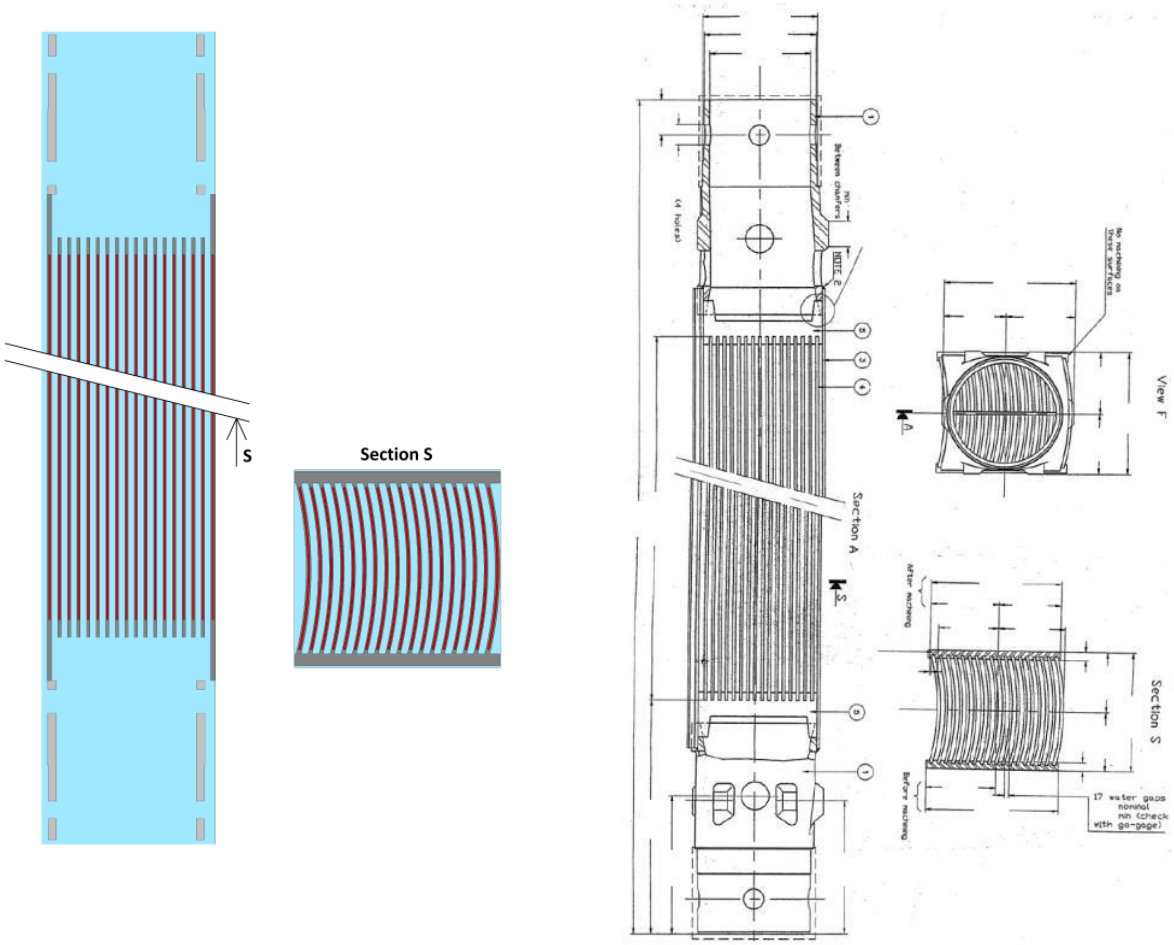
One of the exercises of the MPCMIV Benchmark consists in performing the burn-up simulations of the three core loadings of interest: 1105, 1106 and 1107. In order to complete the benchmark specifications, the isotopic compositions of the fuel assemblies have to be defined. To achieve this goal, an ad hoc strategy has been developed; it is organized into three steps as reported in the following:

1. Infinite lattice depletion calculation for each individual assembly type. Considering the full 3D model of each fuel assembly type (i.e., CA, CF and CAC depending on the U-235 nominal content), the depletion calculation have been performed at several burn-up points to cover the whole life of the assemblies base irradiation;
2. Construction of the isotopic inventory for all the FAs loaded in the core loading 1105. Since the only information available at the beginning of the core loadings of interests is the U-235 mass, starting from the step 1 results, this information is used to find the burn-up point which match the content of U-235; then, the burn-up point is used to find through interpolation the content of the other isotopes. This process is repeated for each assembly loaded in the core.
3. Core loading and depletion calculations, considering the actual history of the three core loadings of interest. The previous steps are repeated for 4 different power levels, considering the corresponding temperatures obtained through the RELAP5 simulation presented in Section 2.2.1; at the end of each full core depletion calculation the isotopic composition of the discharged fuel assemblies will be compared with the values provided in Studsvik documentation. The imposed power that best matches the provided information for each assembly type will be selected, and the corresponding isotopic composition will be adopted in the specifications.

   The procedure is explained in detail in [4].

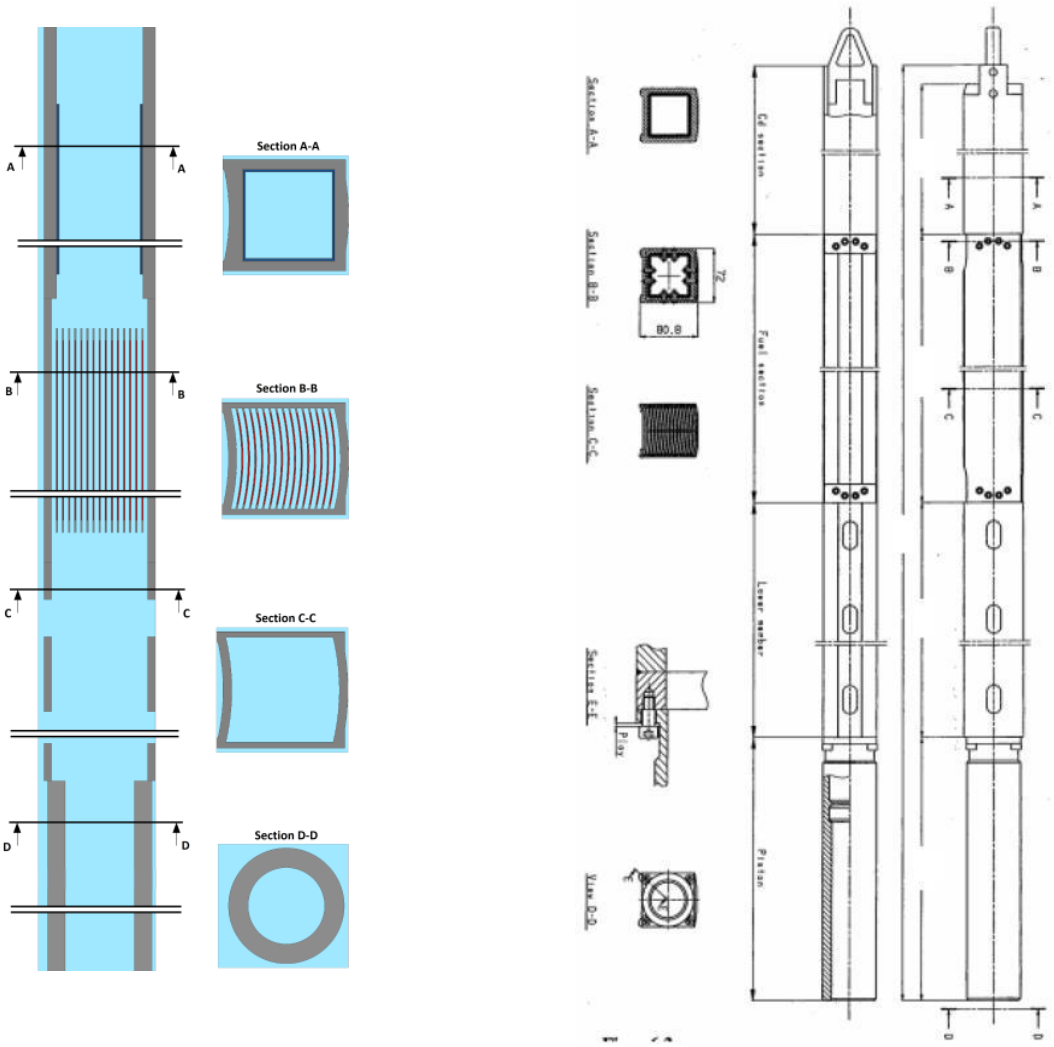
All the calculations are performed for the three FA types. The fuel elements (CF and CA types) are of Low Enriched Uranium (LEU) type and contain 18 curved plates of 19.75% enrichment. The only difference between the two fuel assemblies is the U-235 loading per fuel element. The 16 internal plates are shorter with respect to the 2 outer plates that are of the same length of the side plates. Above and below the fuel section, bottom and top adapters complete the FA. The adapters have two sets of holes for coolant flow, placed at different heights. The control rods elements (namely CAC) are composed of a cadmium poison section and of a fuel section. Upper and lower ends of the fuel section are riveted to the poison section and to the lower member

of the rod assembly. The above region of the cadmium section is an extension of its upper part. The model for the fuel elements is reported in Figure 5. The Serpent 2 code has been used in order to build the model. Figure 6 shows the model of the CAC assembly, namely the control rod. The geometry of the three FA types was taken from the specifications, while some assumptions have been made to cover the lack of information about materials specifications.



a) b)

Figure 5: CF and CA fuel assembly type: a) Serpent model; b) CAD drawing from specifications



a) b)

Figure 6: CF and CA fuel assembly type: a) Serpent model; b) CAD drawing from specifications

2.2.1 Single assembly RELAP5 model

A RELAP5 thermal hydraulics model for each single assembly type (CF, CA and CAC) has been developed and used to compute the temperature distribution starting from a certain power generation. Since the CF and CA elements are geometrically identical only two models are built: one representative of the fuel assemblies and one of the control rods. The RELAP5 model simulates only the fuel active length section, which is 60 cm long, and adopts BICs to simulate the rest of the reactor system.

The nodalization of the model, which is shown in Figure 7, is composed by six hydrodynamic components that simulate the coolant (moderator) and one heat structure that stands for the assembly structure itself. The initial conditions of the coolant are 305.15 K for the temperature, while the average coolant velocity between the fuel plates is equal to 6.2 m/s for the FA model, and 4.9 m/s for the CR model. The outlet pressure condition is set to $2.7 \cdot 10^5$ Pa. The complicated geometry of the water between plates of the assemblies has been simplified as if it was a single pipe. The total length of 60 cm is divided into 6 control volumes of 10 cm each, which preserve the same volume as the real case. In order to take into account the real geometry, it is computed the hydraulic diameter as 4 times the flow area divided by the wetted perimeter. The ORNLANS interphase friction has been applied.

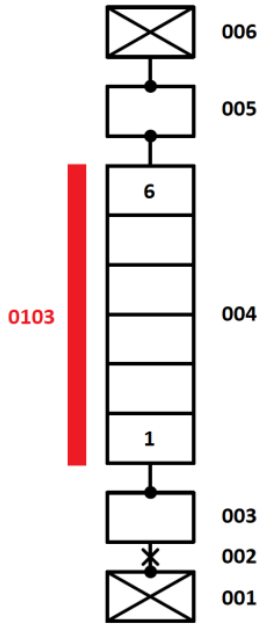


Figure 7: Single assembly RELAP5 model nodalization

The Heat Structure (HS) simulates the fuel plate's structure with 6 axial heat structures of rectangular geometry divided in 9 radial mesh points. 5 radial intervals are occupied by the fuel meat, while the remaining 3 by the cladding. The properties used to describe these two materials are thermal conductivity and volumetric heat capacity. For the cladding they are constant and are respectively 130 W/m K and 2582400 J/m^{3}K [5]. For the fuel meat instead is given a constant thermal conductivity of 59 W/m K, while the volumetric heat capacity depends on the temperature, and it is described by the formula (3) below that gives the result in MJ/ m^{3}K [6].

(3)

where V_P and V_F are respectively volume fractions of porosity and fuel in the meat. The porosity content of the fuel cores produced by a given fabricator remains virtually constant, and it is 4 vol.% for the present one (CERCA). The fuel meat volume fraction is derived from this one with the formula (4).

(4)

is the specific heat capacity of the uranium silicide without the Al powder and is the specific heat capacity of the aluminum, both in J/kg K. They are calculated with the formulas (5) and (6).

(5)

(6)

The HS has an adiabatic condition on the left-hand side, which represents the center of symmetry of each fuel plate, and a convective type boundary condition on the right-hand side which is connected with the pipe 004. The surface area for the heat exchange is calculated as 2 times the area of a single plate face (since there are two sides for each plate facing the hydraulic channel) multiplied by the number of fuel plates (18 for the FAs and 15 for the CR). Finally, the power generation inside the fuel is provided with a constant control variable for each axial node. The axial power distribution is derived following the power profile normalization curve obtained with six detectors put in an infinite lattice Serpent simulation for each specific assembly type. All the BICs

are given constant for a simulation time of 1000 s, during which the system manages to reach the steady state conditions.

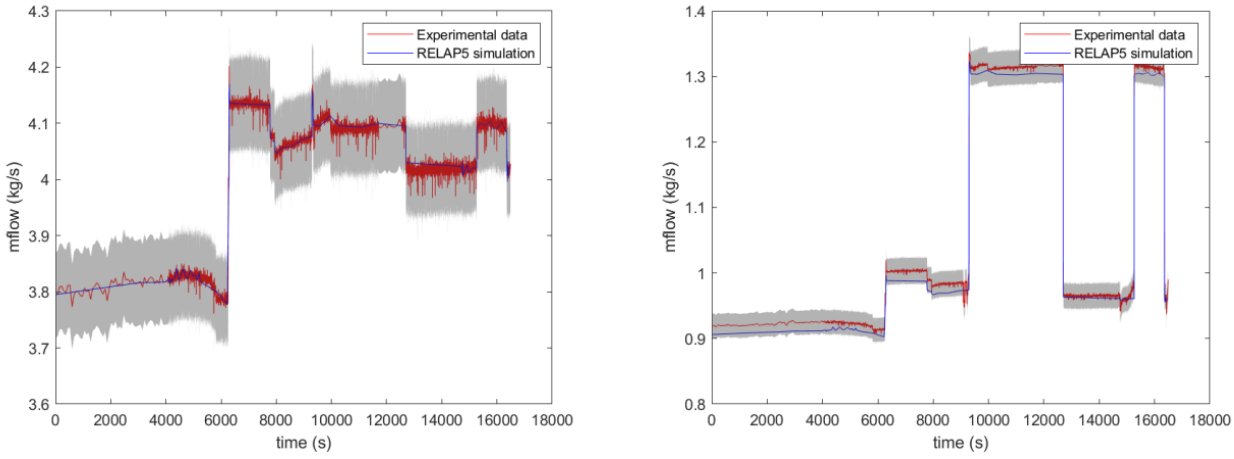
3 SIMULATION RESULTS

The thermal hydraulics model of the in-pile tube is used to perform the simulation of the first power ramp test. Since during the test there are different phases (see Figure 3 and Figure 4) that involve different configurations, the entire test is divided into 5 periods:

- T1: Transient with configuration β simulates both period a and b of the power calibration;
- T2: Transient with configuration α simulates period a of the pre ramp 1;
- T3: Transient with configuration γ simulates period b of the pre ramp 1;
- T4: Transient with configuration α simulates period c of the pre ramp 1;
- T5: Transient with configuration γ simulates the ramp test.

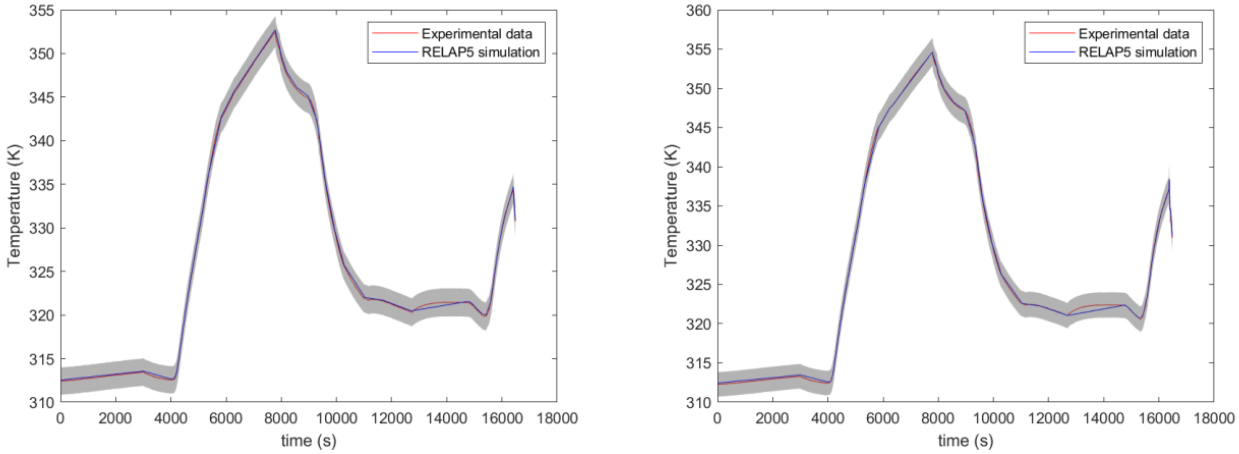
Every time the configuration changes it is necessary to perform a restart in RELAP5 by changing the hydrodynamic configuration. It is very important to initialize correctly the temperature and pressure conditions inside the hydrodynamic components and the heat structures, otherwise the simulation may have some problems to converge. Figure 8 and Figure 9 compare the experimental data with the results of the RELAP5 simulations. The grey region defines the acceptability criteria used by NINE, which is 2% of margin of difference against experimental data for the mass flow rate and 0.5% for the temperature. So, this is not the error due to measurement or uncertainty associated to thermocouples or flow meter that will be investigated in future activities, instead it is a check on the simulation results in order to see if they are in agreement with the experimental data. Figure 10 shows a zoom of the ramp test temperature downstream the test section, from which it is possible to see how it increases following a curve while it passes through the neutron flux and then it rapidly decreases when the reactor is shout down at the end of the test.

All the results of the RELAP5 simulations are in good agreement with the experimental data and fall inside the acceptable region. This means that the model fidelity is demonstrated against validation with experimental data. This good result is clearly also affected by the fact that in the model of the U tube only the BICs are imposed. The next step will proceed with a deeper analysis, first by deriving the gamma power by a neutronic calculation and not from a power balance as it was done for this preliminary work, secondly by substituting the inlet and outlet conditions of the U-tube model with the thermal hydraulics model of the main circuit that is connected to the in pile section. This future analysis will allow to characterize step by step all the main circuit components and to complete the benchmark specifications.



a) b)

Figure 8: Mass flow rate inside the loop a) and inside the ramp rig b)



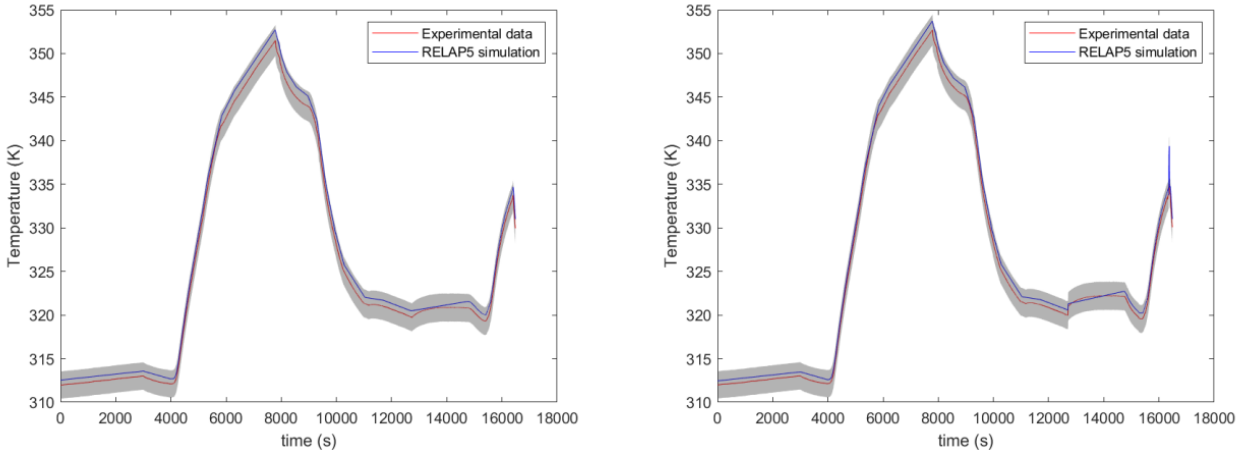


Figure 9: Coolant temperature at the inlet a) and outlet b) of the in pile loop; coolant temperature upstream c) and downstream d) the test section

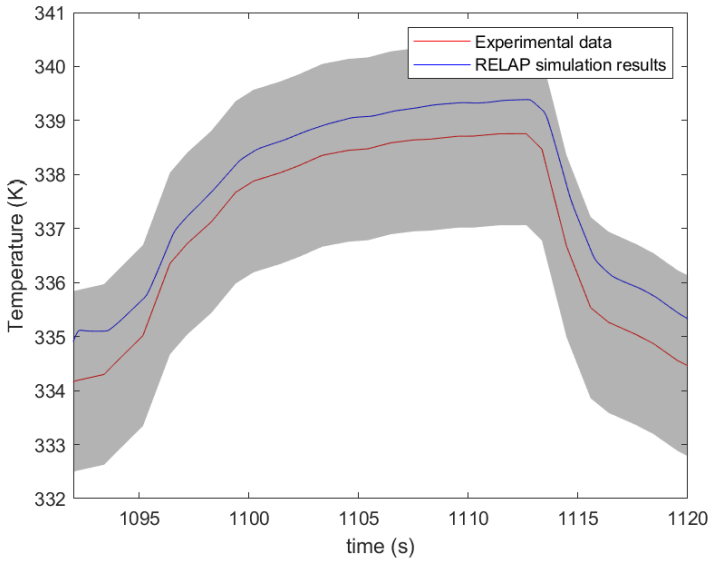
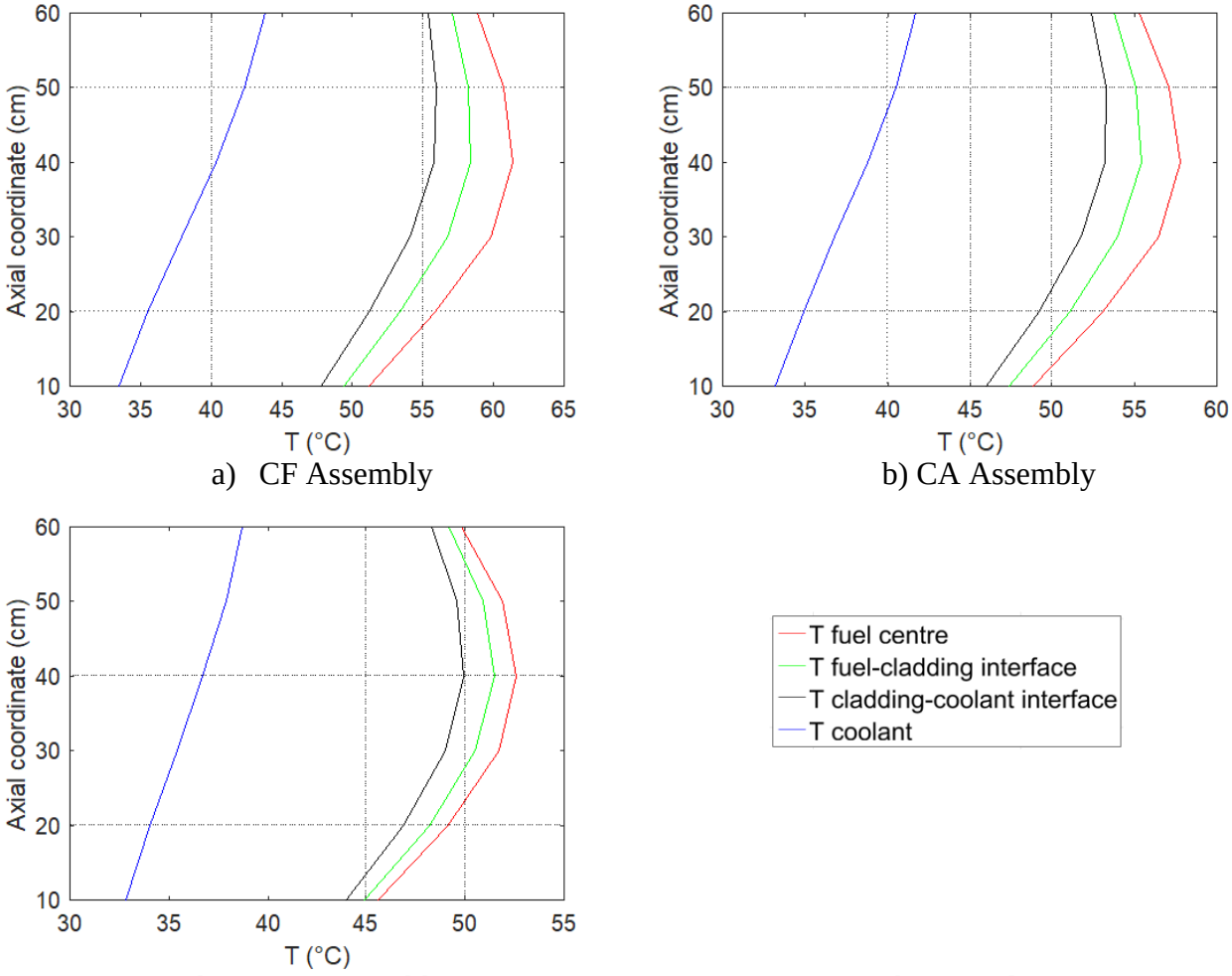


Figure 10: Detail of the power ramp test, temperature distribution downstream the test section

The results of the single assemblies simulations are presented in Figure 11, where it is possible to see the plots of the temperature distributions for the three assembly types along the axial

direction at different radial positions, i.e. at the center of the fuel, at the interface between the fuel and the cladding, at the interface between the cladding and the coolant, and in the coolant. The case considered for the plot is the one with a power value for the assembly correspondent to 50 MW of reactor power (specific power equal to 0.443 kW/kgU). Finally, the average value of the fuel temperature and the average value of the cladding temperature to be used in the Serpent simulations are obtained in two steps. The first step is performed by applying the mean value theorem for integrals over the radial temperature distribution, since radially there is a computed value at each mesh point. In this way an average value for each axial heat structure is obtained. In the second step, the arithmetic average of the values computed in the previous step is evaluated, taking into account that the axial heat structures have the same length. This last step is also applied to compute the average moderator temperature, since also the pipe is axially divided in six control volumes having the same height.



a) CF Assembly

b) CA Assembly

b) CAC Assembly

d) Legend

Figure 11: Temperature distribution inside the three assembly types computed with the RELAP5 single assembly model

Table 1 shows the final values to be set in the Serpent simulations. These values are in agreement with both Studsvik documents [7] and NINE previous experiences with this kind of MTR (Material Testing Reactors), for the OPAL reactor. These values are the one used to perform the infinite lattice depletion calculation of the single assemblies.

Table 1: Temperature Profile Evaluation Approach, Average Temperatures for each Power Value and Assembly Type

	T_{fuel}(°C)	$T_{cladding}$(°C)	$T_{moderator}$(°C)
CF	57.2	54.5	38.9
CA	54.1	51.9	37.6
CAC	49.8	48.6	35.9

a) P=50 MW

	T_{fuel}(°C)	$T_{cladding}$(°C)	$T_{moderator}$(°C)
CF	53.8	51.7	37.5
CA	51.2	49.4	36.5
CAC	47.6	46.6	35.1

a) P=40 MW

	T_{fuel}(°C)	$T_{cladding}$(°C)	$T_{moderator}$(°C)
CF	50.2	48.6	36.1
CA	47.8	46.5	35.4
CAC	44.8	44.1	34.3

a) P=30 MW

4 CONCLUSION

In this work the MPCMIV Benchmark organized by the NINE company has been considered, and some reactor physics and thermal hydraulics analyses have been carried out. The reactor physics analysis has been performed to complete the benchmark specifications through the evaluation of some missing boundary and initial conditions necessary to face the benchmark exercises. Some infinite lattice depletion calculations were performed for each assembly type on different power and temperature conditions to simulate the previous irradiation history of the assemblies, which is not available. A total number of 54 simulations have been completed in this way. These simulations will be used to derive the fuel composition for each assembly at the beginning of core 1105 and to perform the full core burn-up simulation of core 1105. After the full core depletion, the isotopic composition of the fuel elements discharged from the reactor will be compared with some available data given in the benchmark specifications that comes from some Studsvik evaluation and the best initial composition will be individuated among the several possibilities of the elaborated dataset.

As far as the thermal hydraulics analysis is concerned, the in-pile loop 1 has been considered and the U-tube model has been built and validated against experimental data. Both steady state and transient analyses have been performed to demonstrate the fidelity of the model and to carry out the simulation of the first cold power ramp test with the RELAP5 code. The results of the simulation show good agreement with the experimental data. This is probably also influenced by the fact that the BICs has been imposed and the gamma heating has been computed with an energy balance. The future activity will require to evaluate the gamma power with a reactor physics simulation and to check the reliability of the model with a more realistic power distribution. In addition, the model of the main circuit needs to be completed and its components to be characterized.

In conclusion, the analyses carried out in this work have shown good agreement with the experimental data. The models are validated, and the preliminary results are satisfying. Furthermore, the models will be used in the future to perform other simulations, with different conditions and set ups to enhance the quality of the results and conclude the benchmark specifications.

Further activities will include the adoption of the TRANSURANUS [8] fuel performance code for the analysis of the rodlet behaviour during the ramp tests; appropriate coupling approaches will be investigated and tested against the benchmark experimental results.

5 ACKNOWLEDGMENTS

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