Extension of the Validation Basis of Subchanflow by Using Measured Data From the IEA-R1 Research Reactor

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Abstract

The high-fidelity coupled code Serpent2/SCF was recently extended at KIT for the detailed analysis of MTR-cores consisting of fuel assemblies with plate-type fuel and control elements. Serpent 2 and Subchanflow were successfully validated using SPERT IV D12/25 and RA-6 tests. This paper presents an extension of the validation basis of Subchanflow using experimental data measured in the IEA-R1 reactor. The experimental data were measured with an instrumented fuel assembly (IFA) installed inside the reactor core. The IFA consists of 18 plates spaced 2.89 mm from each other, forming 17 internal channels. The thermocouples, type K, are axially distributed on three plates, two external (plates 1 and 18) and one central (plate 9). The sensed values correspond to a series of 2 tests in which a steady-state was achieved. The global measured values correspond to the temperature of the cooling fluid at the outlet of the IFA. The other measurements are focus on the temperature of the cladding of the aluminum plates. The extension of the validation basis of Subchanflow is very important for use of the code within a licensing process. The results show that the difference in coolant temperature between the measured and calculated values are between $\pm$ 0.02 °C. In addition, a maximum cladding temperature difference of 3.9 °C and a minimum of 0.1 °C were obtained, where the central plate had the most significant difference.

1. INTRODUCTION

In contrast to power reactors, which are analyzed with high-fidelity codes combining neutronics and thermal-hydraulics (N/TH), the analysis of reactor cores with plate-type fuel is done with thermal-hydraulic codes adapted with an equivalent plate approach or heuristic methods [1]. This approach has some limitations, as described in [2] and [3]. The Subchanflow code (SCF) has been extended to analyze plate-type fuels. Initially, the MTR-related models were validated using the experimental data from a test facility named RA-6 [4,5]. The results obtained were encouraging and allowed to proceed with the research and development of more sophisticated models. The coupled code Serpent2/Subchanflow developed for LWRs [6] was modified for research reactors with plat-type fuel [7]. To assess the prediction capability of the extended Serpent2/SCF, the IAEA reactor 10 MW was simulated. The results of this novel detailed high-fidelity simulation at the plate/subchannel level were very impressive and they are published in [7] [8]. However, there is a need to extend the validation basis of Subchanflow by performing new validation studies considering additional tests. For this purpose, the data of the experimental IEA-R1 reactor were used, and the main results will be discussed here. The IEA-R1 research reactor is equipped with an Instrument Fuel Assembly (IFA). It contains 18 thin plates, at a distance of 2.89 mm from each other forming 17 internal channels through which the coolant flows downward. The thermocouples are axially distributed on three plates, two lateral (plates 1 and 18) and one central (plate 9). The temperature data of the fluid at the outlet of the IFA and the fuel cladding plates are stored when the tests reach a steady state. The tests performed were presented to the scientific community in the framework of the IAEA CRP project and served as a basis for the validation of different thermal-hydraulic codes [9]. This work consists of five sections. The second section contains a description of the IEA-R1 reactor, geometrical dimensions, and thermophysical properties. The third section contains a brief introduction to the Subchanflow code (SCF), assumptions, and simplifications for the numerical representation of the IFA model. The fourth section contains the global temperature and pressure results as well as the temperature profiles of the three plates used for the SCF validation. Finally, the last section includes the main conclusions of the investigation.

2. IEA-R1 DESCRIPTION

The IEA-R1 is a light water-cooled and moderated research reactor of the pool type with graphite reflection. It was developed by Babcock & Wilcox. It was the first reactor to be commissioned in the Southern Hemisphere on September 16, 1957 [10]. Figure 1 shows the design of the IEA-R1. It can be seen that the small core is immersed in the bottom of the water basin and is surrounded by horizontal irradiation tubes and shielding.

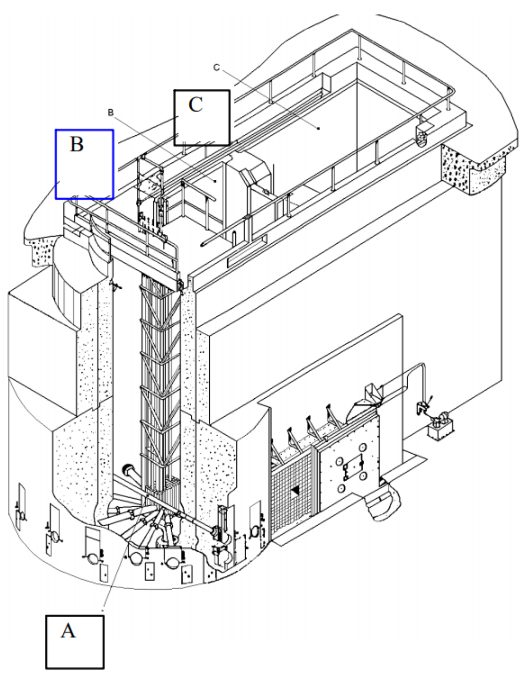

Figure 1. General view of the reactor pool structures, obtained from [10]

Figure 2 illustrates a typical configuration of IEA-R1, including the pneumatic system used for NAA (neutron activation analysis) and the irradiation elements for in-core irradiation. The reactor core, reflector (graphite), and irradiation elements are housed in an 8 × 10 grid plate suspended from a structure connected to a bridge at the top of the pool. Currently, the core consists of a 5 × 5 array of fuel assemblies (FA) with 18 plates of low enrichment uranium oxide (LEU -U3O8Al). The control assemblies are four and are located around the center, and the neutron material absorber is an Ag-In-Cd type.

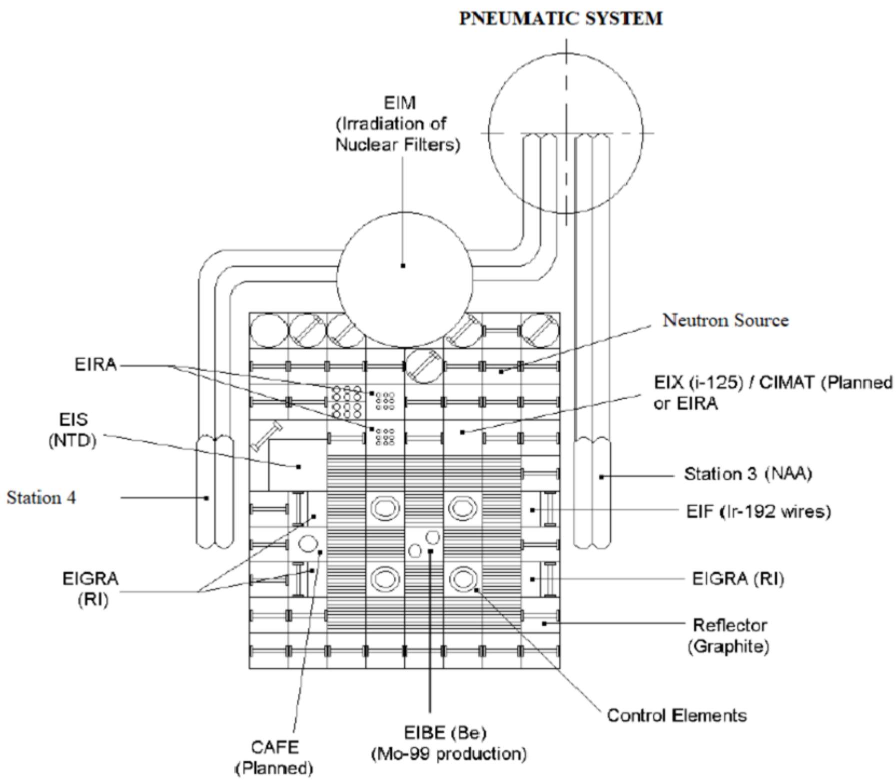

Figure 2. Typical grid configuration of IEA-R1, obtained from [10]

Figure 3 shows the top view of the IAE-R1 core. The location of the fuel assemblies, the irradiation positions, the reflector and the control rods can be observed there. The Instrumented Fuel Assembly (IFA) is placed in position (6, 8). It is built to a scale of 1:1, retaining all geometric dimensions of a Standard Fuel Assembly (SFA). Thermophysical properties are also maintained to obtain a real model inside the reactor core. The main objective of the Instrumented Fuel Assembly (IFA) is to measure the temperature of the coolant and the surface of the fuel plates under full operating conditions [11]. The IFA has 18 fuel plates distributed at a distance of 2.89 mm, forming 17 internal channels and two around whose value corresponds to half a channel. It has 14 thermocouples distributed in three regions: reflective side channel, center channel, and side channel of an adjacent unit. The IFA is equipped with sensors to measure the temperature of the coolant (TF) and the surface of the fuel plates (TC). For this purpose, K-type thermocouples have been installed in three regions: reflector side, central, and FA side. The positions can be seen in Table I and Figure 4.

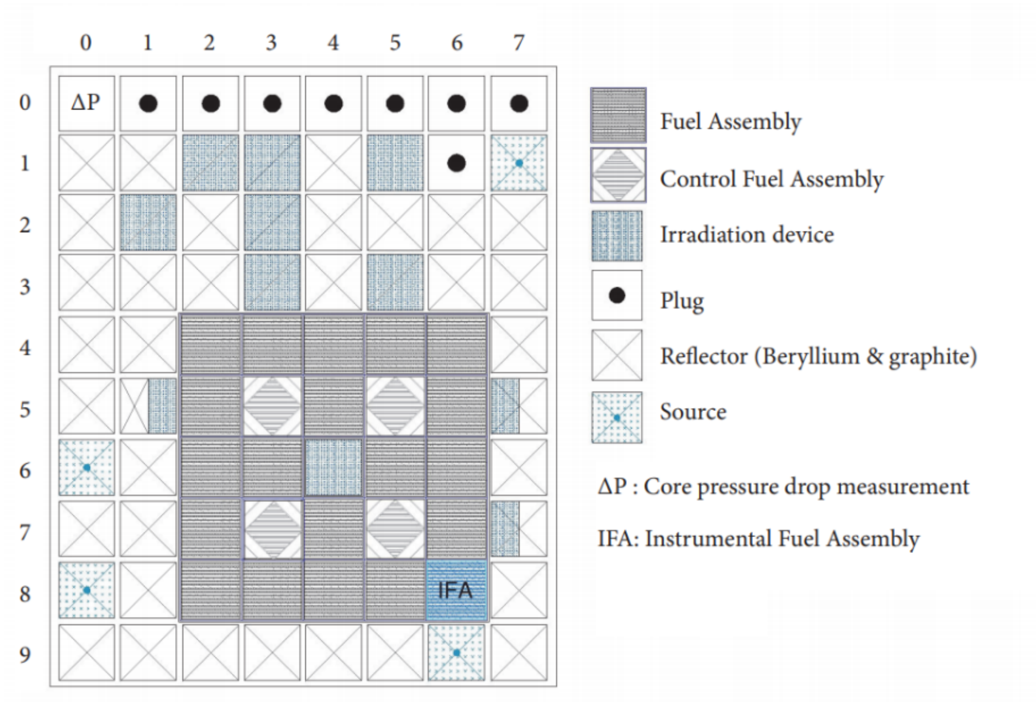

Figure 3. Top view of configuration 243 of the IEA-RA nuclear core, obtained from [11]

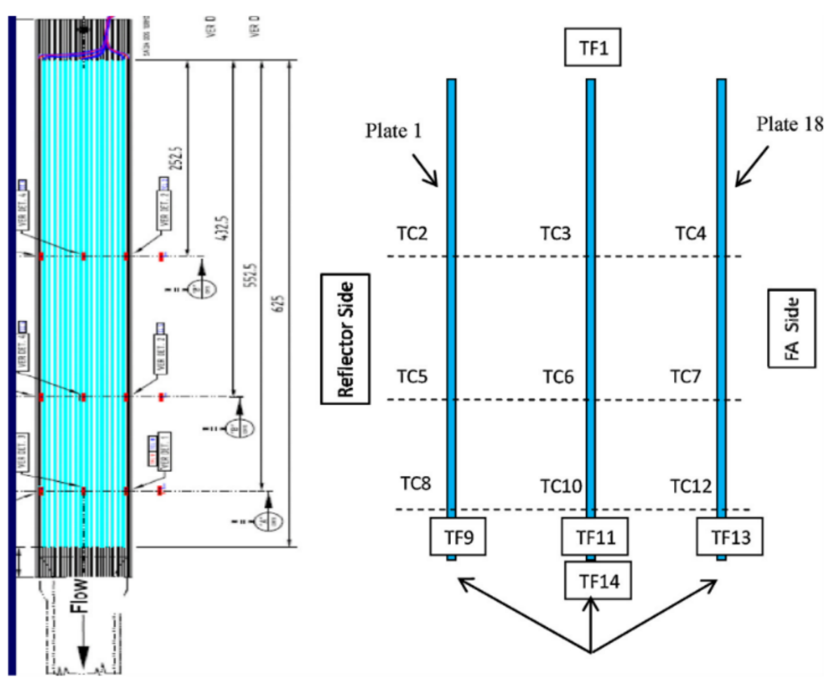

Figure 4. Location of the thermocouples in the IFA, obtained from [11]

Table I. Locations and symbols of IFA thermocouples

Position from top [mm]	Reflector side (plate 1)	Central (plate 9)	FA side (Plate 18)
252.5	TC2	TC3	TC4
432.5	TC5	TC6	TC7
552.5	TC8	TC10	TC12
552.5	TC9	TC11	TC13

2.1. Geometry and Dimensions of the Instrumented Fuel Assembly

The axial and radial dimensions of the fuel assembly instrumentation were taken from the literature reports presented in [12,13,14]. The IFA has 18 plates that form 17 rectangular channels through which the cooling fluid flows downwards. The radial and axial measurements are used to create the SCF model, see Figure 5.

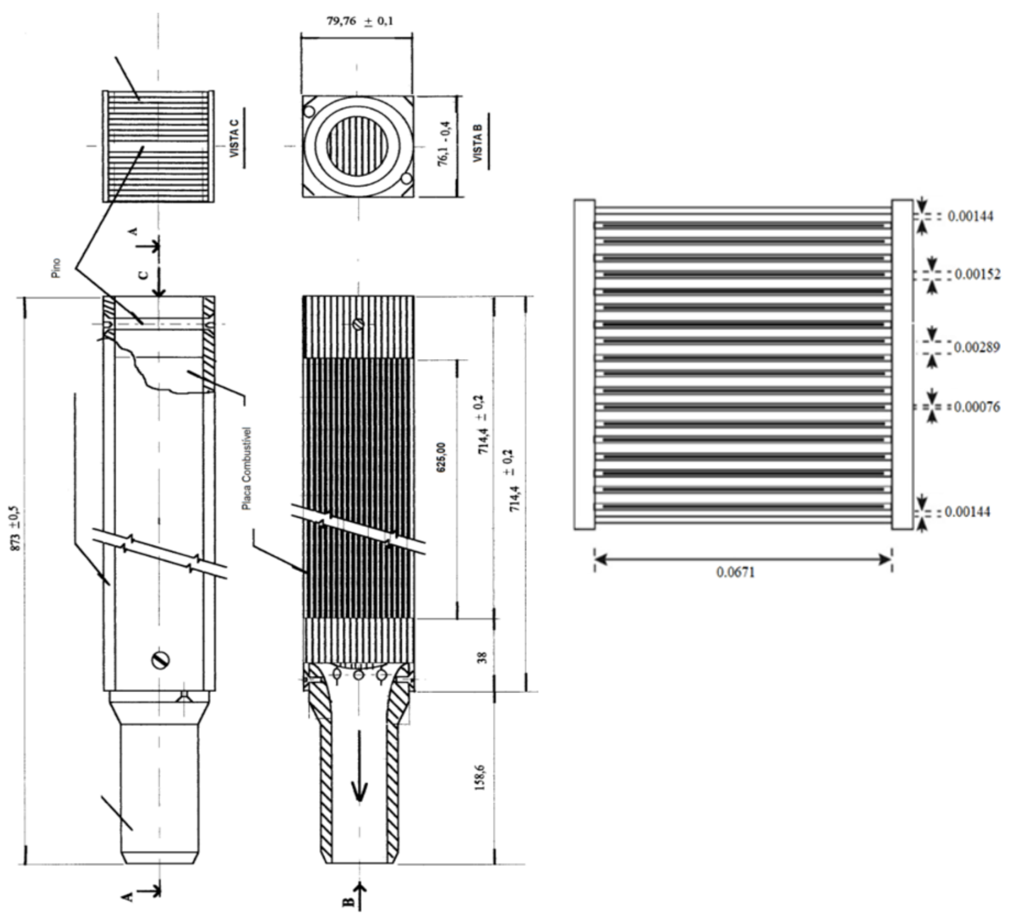

Figure 5. Radial and axial dimensions of Instrumental Fuel Assembly from [12,13,14]

2.2. Material Properties

The MTR-fuel of the IEA-R1 reactor consists of fuel meat and cladding material. The thermo-physical parameters necessary for heat conduction are summarized in Table II, [15,16,17].

Table II. IEA-R1 fuel and cladding properties

Material	Fuel ($U_3O_8 - Al$)	Plate (Al)
Conductivity $\left(\frac{W}{mK}\right)$	11	180
Specific heat $\left(\frac{J}{kgK}\right)$	123.02	892
Density $\left(\frac{kg}{m^3}\right)$	2300	2700

2.3. Tests Performed at the IFA

In this work, the measured temperature values in the steady-state tests are used for validation of SCF [17]. The boundary conditions are given in Table III. The mass flow rate values (6.27 kg/s) for tests have been obtained by interpolation from the experimental measurements presented in [14].

Table III. Operational conditions of IFA tests and of the reactor

Parameter	Units	Test 1	Test 2
Pressure outlet	Pa	162165	162165
Power (IFA)	W	128000	147610
Temperature inlet	^{o}C	32.69	31.67
Inlet flow rate (IFA)	Kg/s	6.27	6.27

3. THERMAL-HYDRAULIC CODE SUBCHANFLOW

Subchanflow (SCF) is a thermal-hydraulic subchannel code developed at the Karlsruhe Institute of Technology [18]. SCF is a very versatile code and has been coupled with various deterministic and Monte Carlo codes for high-fidelity analysis of BWR, PWR, VVER and MTR reactors [6,7,19]. The SCF numerical solution of SCF is based on solving the equations for conservation of mass, momentum, and energy. Special correlations and constitutive equations are used to close the system. The latest version of SCF includes the solution of heat conduction in plate-type fuel, special correlations, and downflow have been included for MTR reactor analysis [5].

3.1. Assumptions and Simplifications

The following assumptions and simplifications are used for the simulation of the IFA model:

- Steady-state power and coolant flow.
- Pressure losses due to acceleration of elevation present in a downward flow are considered negligible due to the scarcity of experimental data.
- Each water channel (17 in total) is represented by a subchannel with 100 axial cells, see Figure 6. The plate as a whole is divided into two sections, the fuel and cladding section. 3 and 2 cells are used in the x-direction, see Figure 7.

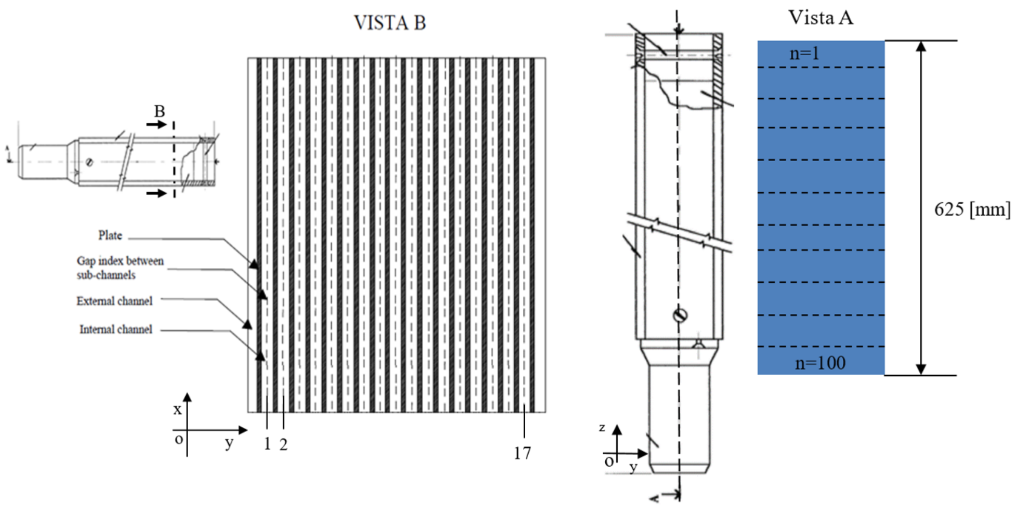

Figure 6. Radial (left) & axial (right) coolant representation, figures adapted from [11,13]

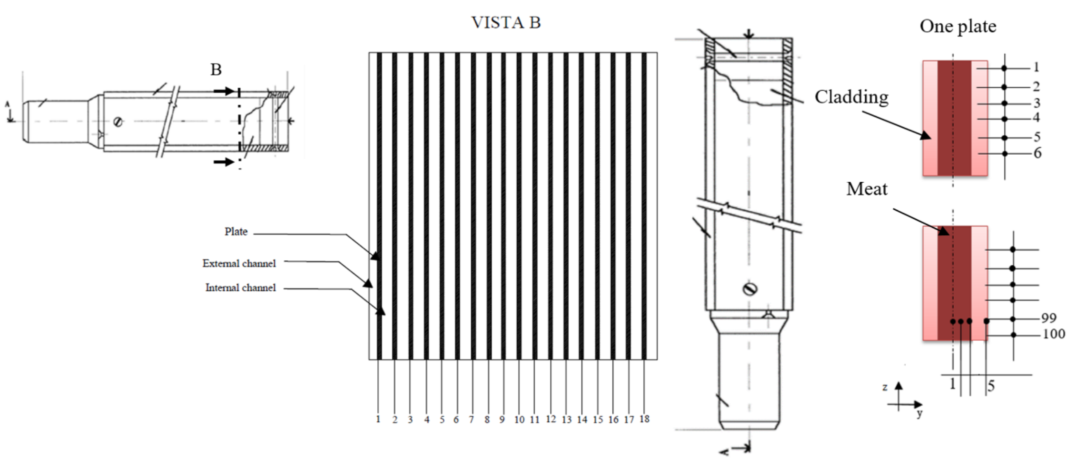

Figure 7. Radial (left) & axial (right) plate representation, figures adapted from [11,13]

The shape of the relative axial heat flux of the IFA obtained from [14] is shown in Figure 8, where it can be seen that it has its lowest values at the inlet of the coolant (upper part of the active length). The highest relative heat flux is almost in the middle; after that, it decreases slightly until the end of the active length (lower part).

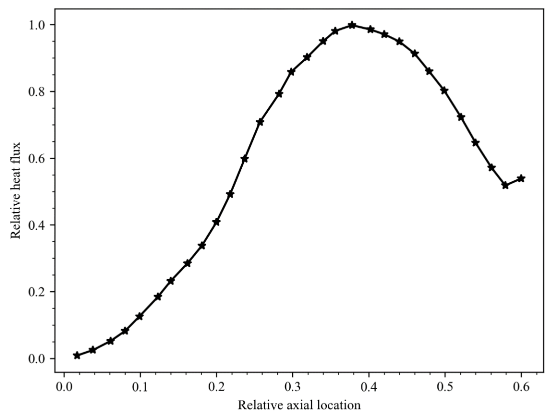

Figure 8. Relative heat flux for the IFA

3.2. Empirical Correlations

The correlations are presented in Table IV. The friction factors recommended in the [20] are used for the pressure drop. The heat transfer correlation developed for the research reactor JRR-3 and proposed by [21] is used.

Table IV. Pressure drop and heat transfer correlations for IEA-R1

Parameter	Correlations
Laminar pressure drop	$f = \frac{0.0791}{Re^{0.25}}$
Turbulent pressure drop	$f = 0.0460Re^{-0.20}$
Heat transfer, Y-Sudo	$Nu = \frac{0.029Re^{0.8}Pr}{[1+1.54Pr^{-\frac{1}{4}}Re^{-0.1}(Pr-1)]}$

4. DISCUSSION OF SELECTED RESULTS

The steady-state simulations of the IFA for the thermal-hydraulic conditions of the two tests were performed with SCF using the specified friction factor relationships and the Y-Sudo correlation for the heat transfer between the plate and the coolant. First, the coolant temperatures measured at the outlet of the IFA during the two tests are compared with those calculated by SCF in Table V. The difference of experimental temperature with that calculated by SCF is in the range of $\pm$ 0.02 °C.

On the other hand, the relative error as a percentage of the predicted pressure drop with respect to the measured one is about -4.43 %, Table VI. The experimental results with respect to the pressure drop were not rigorously studied in this set of tests, but their values can be used as a reference.

Table V. Global parameter predicted by the code SCF, Outlet temperature fluid

Item	Experimental [^{o}C]	SCF [^{o}C]	Difference [^{o}C]
Test 1	37.60	37.58	0.02
Test 2	37.28	37.30	-0.02

Table VI. Global parameter predicted by the code SCF, pressure drop

Item	Experimental [Pa]	SCF [Pa]	Error relative %
Test 1	7835	7488	-4.43
Test 2	7835	7481	-4.522

Figure 9 contains the comparison of the temperature measured at three different positions within the IFA, namely at the surface of the fuel plates located at the Reflector side (plate-1), center (plate-9), and FA side (plate-18), with the ones predicted by SCF. The experimental temperature values are different for the different radial positions of each plate and of course along the elevations. Considering only the experimental results, it can be concluded that the largest temperatures were measured at the Plat-18 of the FA side, followed by plate 1 (reflector side) and the center plate 9.

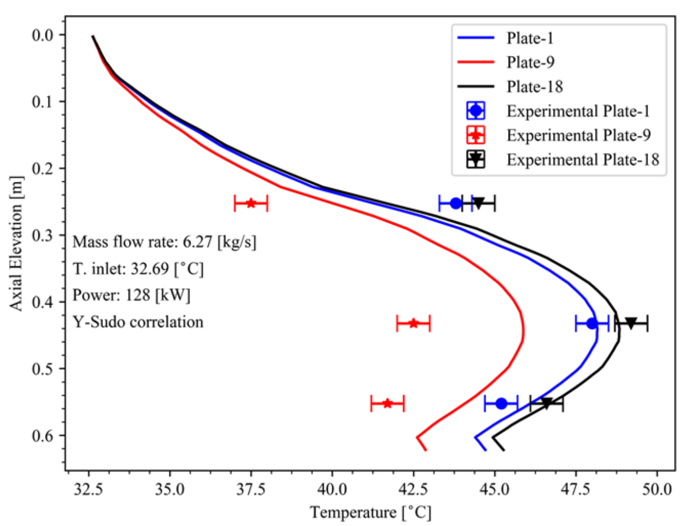

Figure 9. Temperature profiles of the cladding on different plates, test 1.

As shown in Figure 10, the central plate-9 and FA side plate-18 have the highest temperature difference between the experimental and SCF simulated values, with values of -3.3 °C and 2.8 °C respectively.

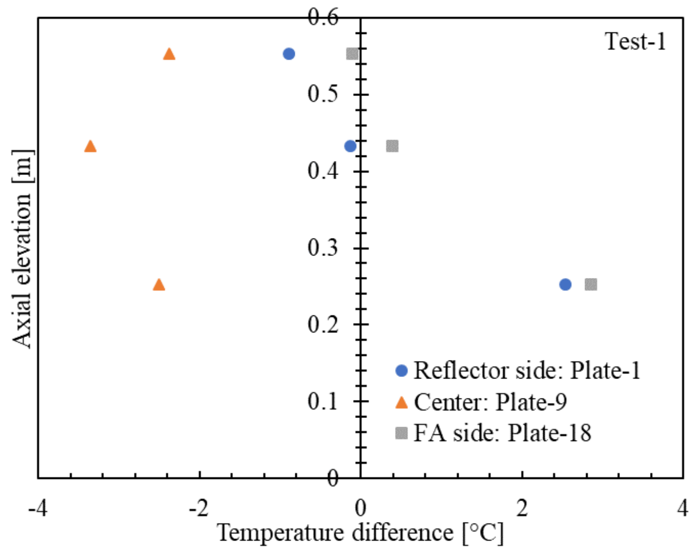

Figure 10. Cladding temperature difference for Test 1.

Figures 11 and 12, belonging to test 2, show the same behavior as in test 1. The central plate has higher temperature difference values, in this case, SCF overestimates all temperature values. Based on the observation of both tests, it is obtained that the temperature values simulated by SCF in the lower part of the fuel assembly present a better temperature difference.

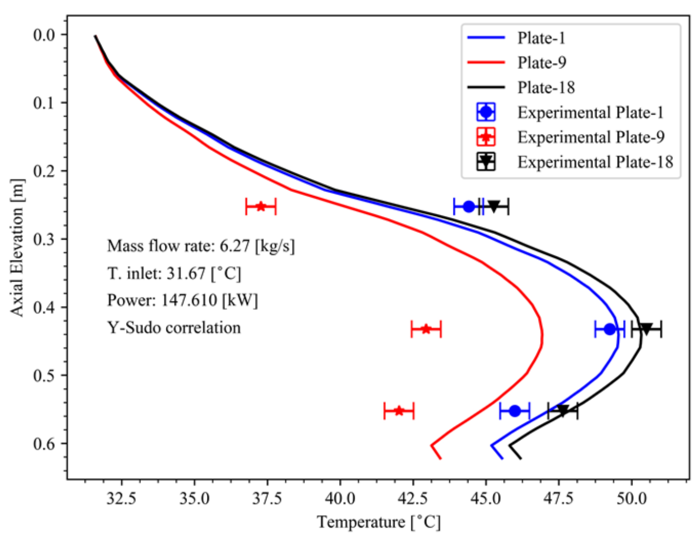

Figure 11. Temperature profiles of the cladding on different plates, test 2.

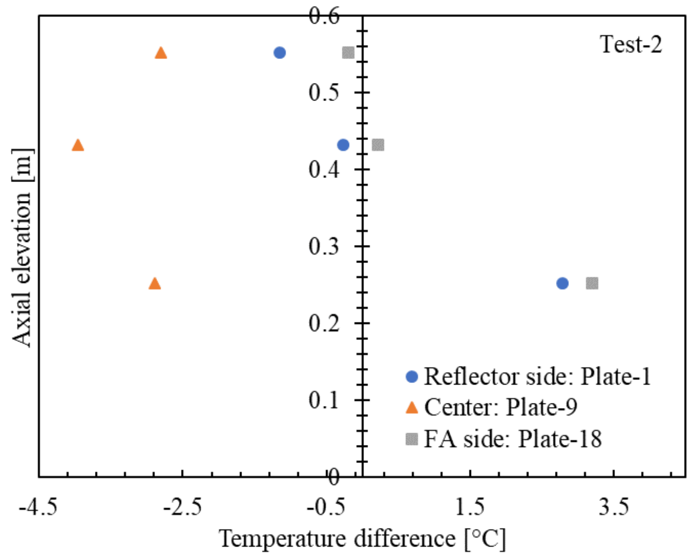

Figure 12. Cladding temperature difference for Test 2

5. CONCLUSIONS

This paper presents and discusses an additional validation of Subchanflow using the data of the tests at the IEA-R1 research reactor. The results obtained by SCF have been compared with the experimental values getting good results: the coolant temperature difference between the measured and predicted values is slight and ranges between ± 0.02 °C. In case of the comparison of the cladding temperature of the plates 1, 9 and 18, the absolute temperature difference between data and predictions is between 3.9 °C and 0.1 °C. The performed investigations confirm the excellent prediction capability of Subchanflow regarding thermal hydraulic parameters of plate-type fuel used in MTR research reactors.

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