Interpretation of TRIGA Reactivity Transients with RELAP5/PARCS Coupled-Code

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Abstract

In the frame of future experiments to carried out upon TRIGA reactors, which aim to verify the real feasibility of the ADS (Accelerator Driven System) concept, it is essential to build a numerical tool able to simulate the dynamic behaviour of the reactor in subcritical configuration. This model developed to support the design of subcritical experiments and the safety analysis of the reactor, as a first step has to be assessed against the experimental data available for the critical reactor.

To this purpose the thermal-hydraulic/neutronic numerical model based on the RELAP5/PARCS coupled-code is been tested against the experimental reactivity transients conducted on the RC1-TRIGA reactor at the ENEA Casaccia Research Center in forecast of the TRADE (TRiga Accelerator Driven Experiment) subcritical experience.

The results of the calculations already performed show a qualitative good agreement with the experimental data and allow to address the future developments and improvements of the numerical model.

KEYWORDS: TRIGA, RELAP5/PARCS, Coupled-Code.

1. Introduction

The sub-critical experiment TRADE [1], which was aimed at demonstrating the feasibility of the ADS concept, foresaw the coupling of the RC-1 TRIGA pool reactor in subcritical configuration with an external proton accelerator by means of a target located in the central channel of the core. This subcritical experiment seems to be definitely rejected but the interest to validate the generic dynamic behaviour of subcritical reactors is also a goal of the Reactor Accelerator Coupling Experiment (RACE) planned on US TRIGA reactors [2], within the frame of the international collaboration ECATS (Experiment on the Coupling of an Accelerator, a spallation Target and a Sub-critical blanket).

These kind of experiments need powerful numerical models able to simulate the behaviour of the reactor in dynamic conditions to support the experimental design and the safety analyses. These models require to be validated against experimental data that obviously at the time are only available for the critical reactor. To this reason we can consider the experimental campaign conducted at the RC1-TRIGA for the reactor characterization in view of the TRADE experiment. In particular, some reactivity transients at different power level are available to test the thermalhydraulic/neutronic model of TRIGA based on the RELAP5/PARCS code [3].

This paper shows the approach followed to assess the coupled model that start from a

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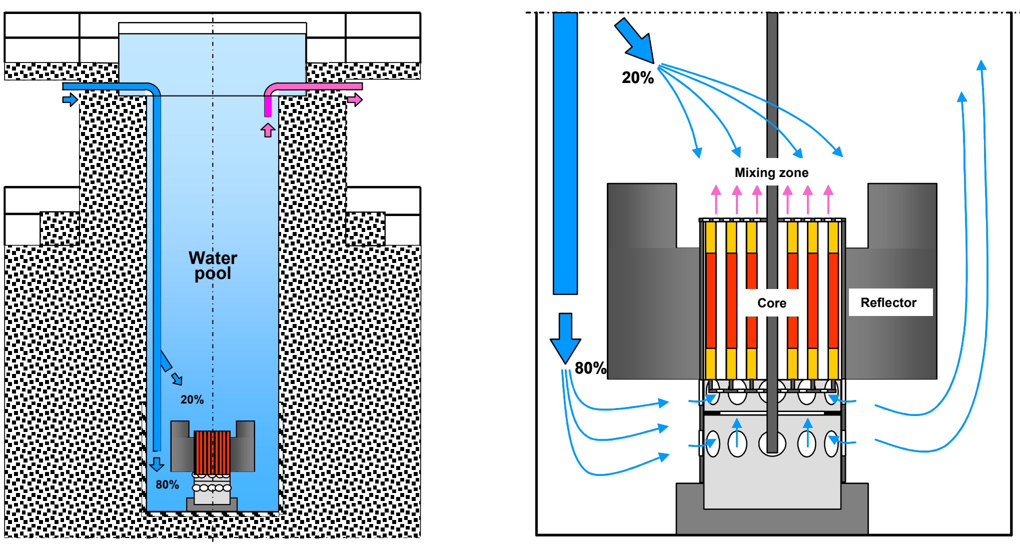
validation concerning the 3D-PARCS code against a simple analytical solution of TRIGA neutronics, validation that was conceived to verify the capability of the code modified to treat the sub-critical systems and tested also in critical configuration. After that, starting from the 1D-RELAP5 nodalization used to support the safety analysis for increasing the MNRC (McClellen Nuclear Radiation Center) reactor power from 1 MW to 2 MW [4], a T/H model of the RC1-TRIGA was developed and assessed on the available experimental data with the help of SIMMER-III, a two-dimensional CFD code. Finally, the coupled model is used to simulate some transients carried out on RC1-TRIGA reactor experimental campaign [5].

2. Development of the T/H-Neutronics Coupled Model

The RC1-TRIGA is an experimental pool-type thermal reactor of 1 MW, cooled in assisted natural circulation with an annular graphite reflector. The core takes place at the bottom of seven meters high water pool that provides both core cooling and shielding (Fig. 1) and it has a cylindrical configuration of seven rings with 127 channels able to host either fuel elements or components like control rods, graphite dummies and various kinds of irradiating and measurement channels [6]. The characteristic TRIGA fuel elements are composed by an homogeneous mixture of zirconium hydride and Uranium 20% enriched in U^{235} with a cylinder of metallic zirconium inside and a stainless steel cladding. This characteristic gives a very high negative prompt temperature coefficient which is the main reason of the high inherent safety behavior of the TRIGA reactors.

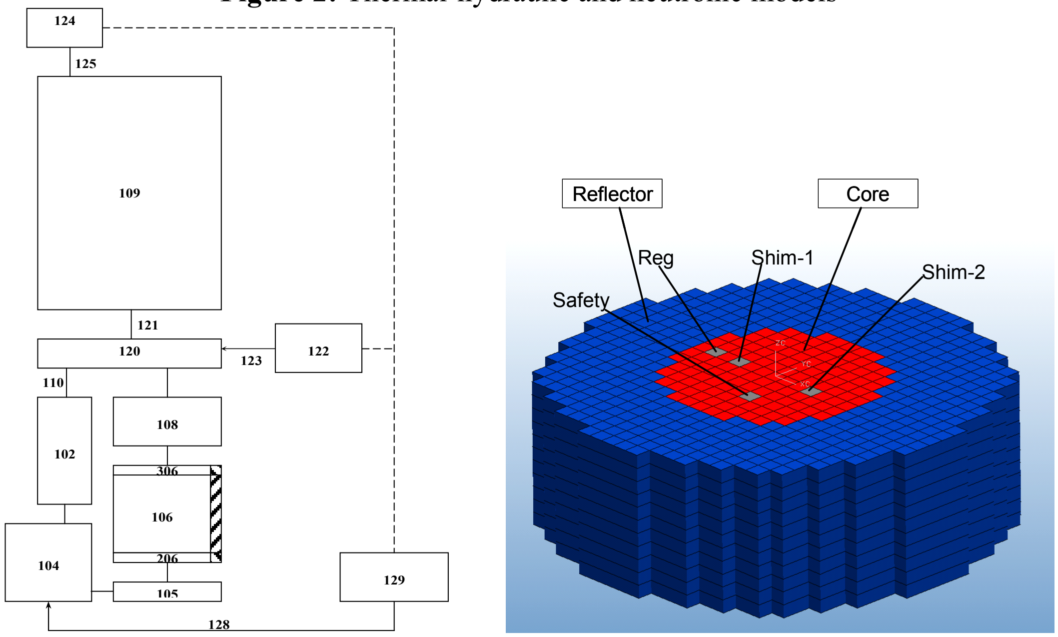
Figure 1: Scheme of the RC1-TRIGA vertical section



The thermal-hydraulic model of the reactor is based on RELAP5, a system code extensively used for the safety analysis of the light water reactors. In the present nodalization (Fig. 2) the core is simulated by one hydraulic channel with associated thermal structure for fuel rods whereas the secondary loop is not considered in this phase due to the fact that almost all the transients are not effected by the behaviour of this loop.

In addition to the RELAP5 neutronic point-kinetic for stand-alone calculations adopted in the phase of improvement of the thermal-hydraulic model, a spatial neutronic simulation has been obtained with the PARCS code which solves the time-dependent, two-group, neutron diffusion equation in 3D Cartesian geometry. The PARCS nodalization also showed in Fig. 2, is considered sufficiently detailed to describe the cylindrical geometry of the core and the radial reflector as well as the control rod positions and their movements during a transient. The axial reflector is not included in the picture even if it is taken into account.

Figure 2: Thermal-hydraulic and neutronic models



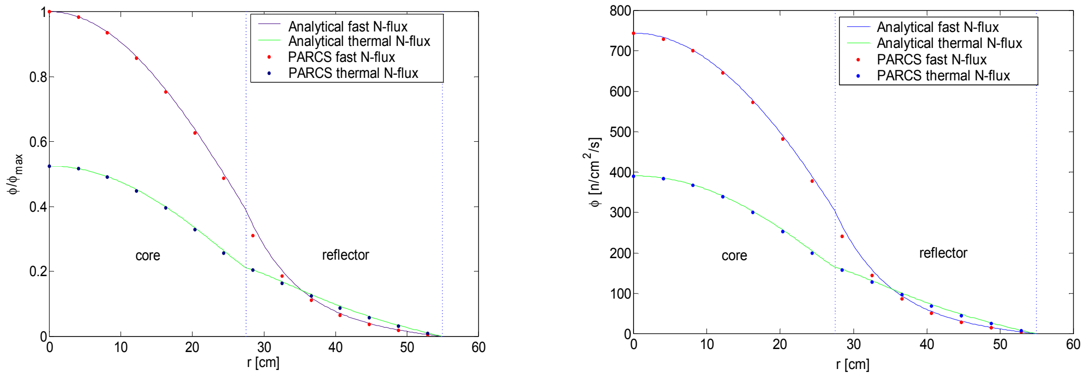
The coupling between RELAP5 and PARCS is performed by means of a General Interface (GI), that is devoted to map the transient evolution and manage the exchange of all thermal-hydraulic and neutronic data, under the PVM (Parallel Virtual Machine) environment. PARCS provide the power densities calculation, than the GI transmits these power densities to the RELAP5 thermal structures coupled with the PARCS nodes. RELAP5 solves the thermal balance equations and returns via GI the temperatures and densities for incorporating the thermal-hydraulic feedback effects in the cross sections calculation.

In view of the TRADE experiment, the PARCS code, modified to treat also an external neutron source, was assessed against an analytical simplified model of the RC1-TRIGA reactor [7]. The analytical model considers a cylindrical annular reactor, axially not reflected (to this target also the PARCS model is lacking of the axial reflector), homogeneous, geometrically characterized by the TRIGA dimensions and it solves the two-groups stationary diffusion equations for multiplication and absorption regions with the pseudo-potential kinetic method [8]. The numerical RELAP5/PARCS model considered is the same reported in Fig. 2 without control rods and axial reflector and the stationary conditions, in sub-critical configuration, are reached after a transient process because stationary calculation are not allowed in presence of the external source.

While in critical conditions the PARCS provides itself to normalize the cross section just to reach the k_{eff}=1, the subcriticality levels desired are obtained modifying the production term of the fast group and the power level desired are obtained by tuning the external neutron source strength.

The benchmark for the critical configuration has provided good results, i.e. analytical k_{eff}=1.08562 vs a numerical k_{eff}=1.08434 with a difference of 128 pcm. In Fig. 3, the left picture shows the normalized radial flux shape compared for critical configuration, while on the right an example of the subcritical k_{eff}=0.97 configuration obtained by an imposed unitary external source is depicted (since the source appears in the analytical method as a linear constant) and where the numerical PARCS results are normalized to the maximum value of the analytical ones. It is possible to see the good agreement between analytical and numerical PARCS results.

Figure 3: Comparison of the neutron fluxes in critical and sub-critical condition (k_{eff}=0.97)

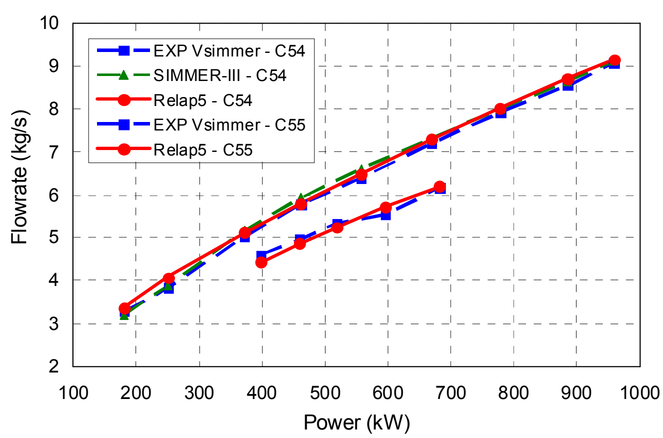


At the same time, the thermal-hydraulic RELAP5 model has been assessed on the dedicated experimental data with the support of SIMMER-III code [9], that is a CFD two-dimensional, three-velocity field, multiphase, multi-component, Eulerian, fluid-dynamics code coupled with a space-dependent neutron kinetics model. By integrating all the original physical models, SIMMER-III is applicable to a large variety of reactor calculations and other complex multiphase flow problems.

The experimental tests campaign at RC1-TRIGA reactor, conducted in critical core configuration at different power levels, provided the temperature measurements at the core inlet and outlet that allow the evaluation of DT through the core at different radial positions. In order to interpret the experimental data for the evaluation of total water mass flow rate through the core in natural circulation, several calculations have been performed with the SIMMER code at different core power levels trying to reproduce the experimental measurements, then the results of SIMMER were used to fit and validate the simplified 1-D model of the RELAP5 code used for thermal-hydraulic transient analysis of TRIGA reactor.

The Fig. 4 show the RELAP5 good prediction of the T/H behaviour of the reactor for a wide range of power that characterize the transients #54 and #55 of the test campaign [5] in comparison with the SIMMER best fitted results.

Figure 4: Validation of TRIGA model RELAP5 code



3. Simulation of TRIGA Transients with RELAP5/PARCS

Several reactivity transients have been performed in the RC1-TRIGA test campaign starting from different insertion of two control rods (i.e. different reactor powers) and different partial extraction of the control rod Shim1 (i.e. different insertion of reactivity) [5]. In Tab.1 there are the essential information about the transients analyzed in the present work. In all transients considered the regulation and safety control rod are always completely extracted.

Table 1: Transients considered

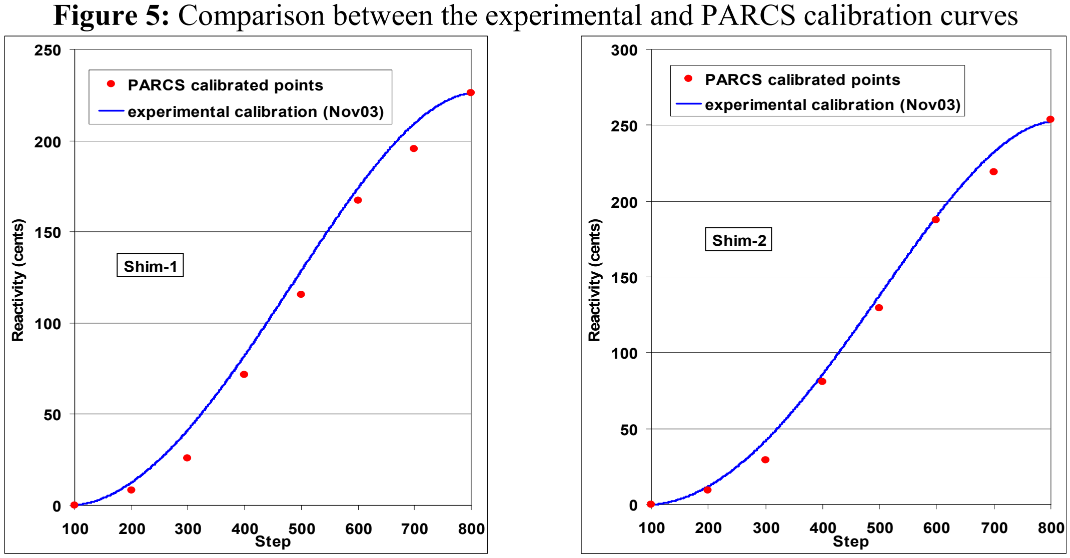
Test #	Steady Power [kW]	Prim. Pumps	Sec. Pumps	Reg Rod	SH1	SH2	SH1 after step	Reaction Step [¢]
3	194	ON	OFF	UP	100	523.5	250,7	25.8
18	293	ON	OFF	UP	100	601.3	300	41.2
24	609	ON	OFF	UP	304	800	404	41
45	215	ON	OFF	UP	100	546.4	355	62
46	306	ON	OFF	UP	100	613	403	83
47	493	ON	OFF	UP	216	800	409.5	70
48	499	ON	OFF	UP	244.2	800	360.7	41
49	610	ON	ON	UP	321.4	800	417.7	41

Before to approach the discussion upon the simulation results it is helpful introduce the scheme of control rods movement. The position of the rods inside the core is detected with a multi-turn potentiometer with a range from 100 step (completely inserted) to 800 step (completely extracted) and a time for the completely extraction requires about 90 seconds.

It is very important to know that the real position of the rods inside the core is affected by 5% error, and consequently the experimental calibration curves are affected by an error estimated in 15%. Moreover, the real material composition of the control rod is affected by a notable uncertainty so the deduction of the cross sections to introduce in PARCS is a very hard work.

Because the first attempt control rod's cross sections, integrated in the neutronic model, caused a not negligible underestimation of the effective control rod worth inserted during the transients, the PARCS model has been fitted on the experimental calibration curves. This is carried out with a tuning on the fast group absorption cross section of the control rods. The good results of the modifications introduced are shown in Fig. 5 for the Shim-1 and Shim-2.

Figure 5: Comparison between the experimental and PARCS calibration curves



The results obtained by the RELAP5/PARCS simulation of the reactivity transients are compared with the experimental curves for fission power and fuel temperature in Fig. 6/13.

Figure 6: Comparison between the experimental and numerical quantities in the test #3

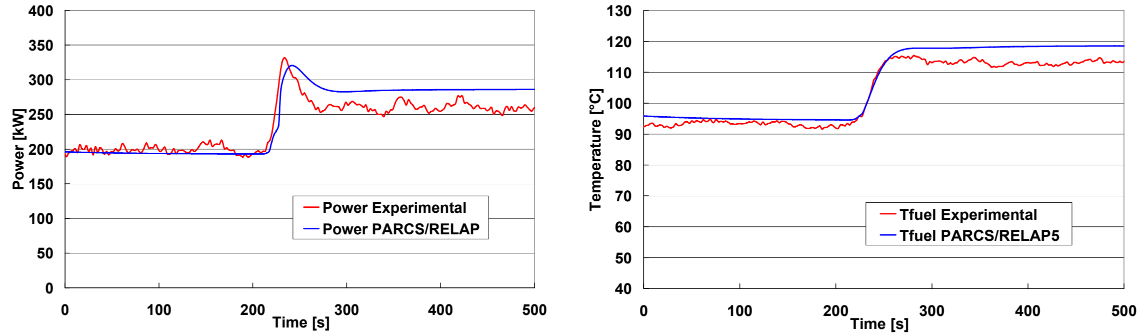


Figure 7: Comparison between the experimental and numerical quantities in the test #18

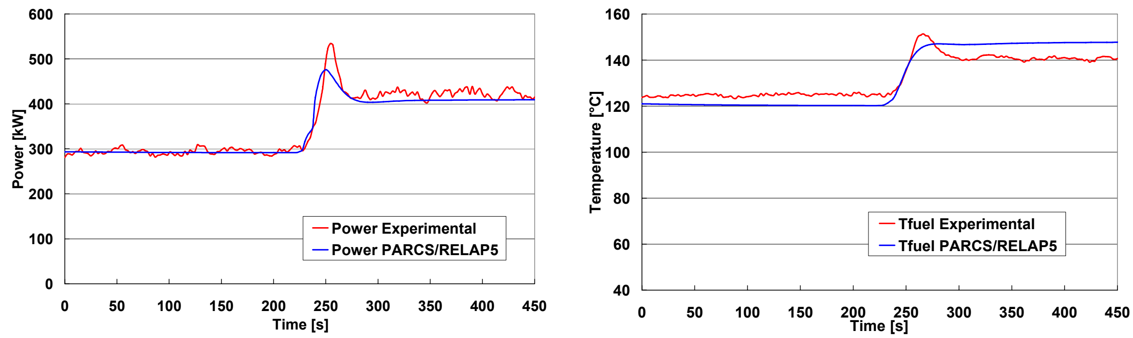


Figure 8: Comparison between the experimental and numerical quantities in the test #24

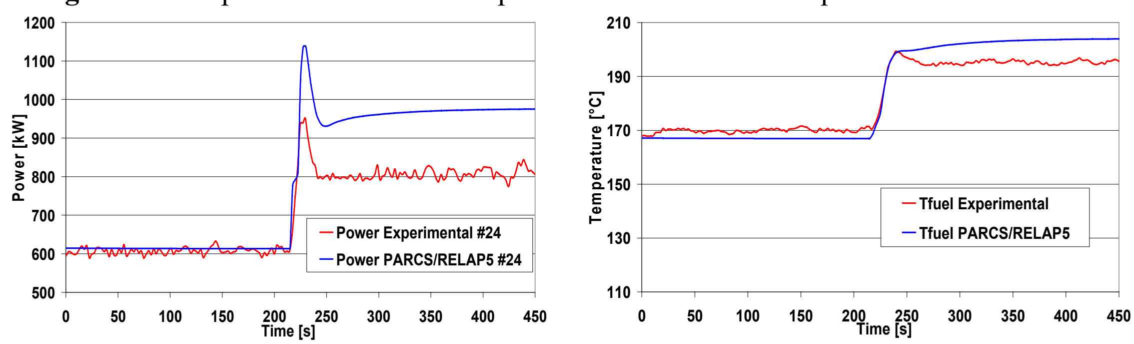


Figure 9: Comparison between the experimental and numerical quantities in the test #45

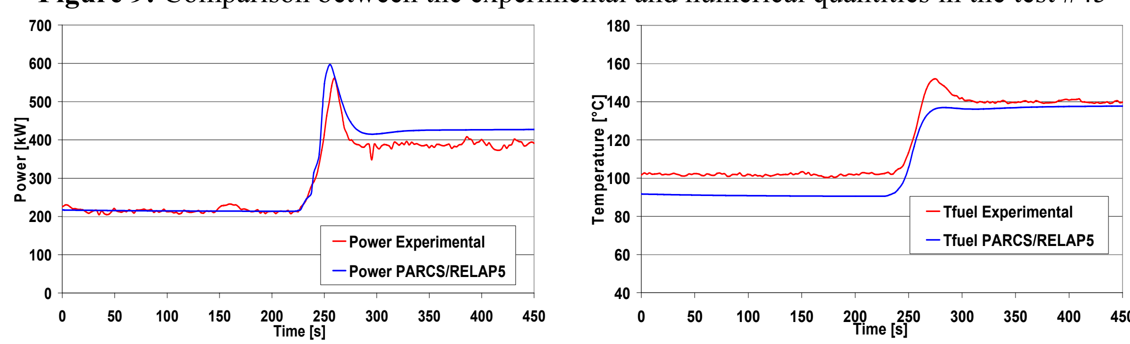


Figure10: Comparison between the experimental and numerical quantities in the test #46

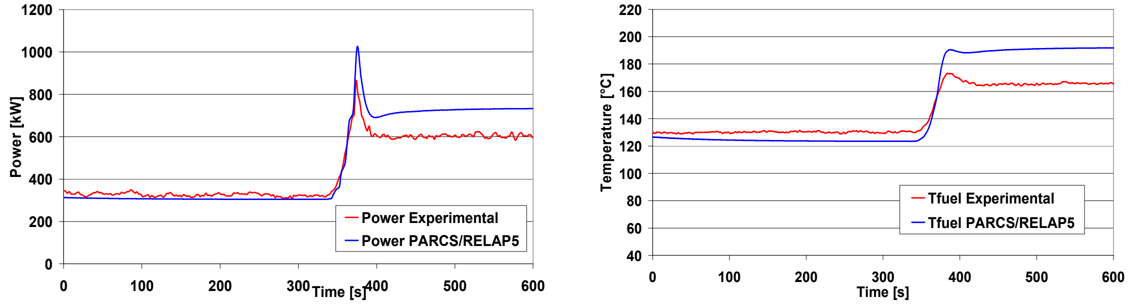


Figure 11: Comparison between the experimental and numerical quantities in the test #47

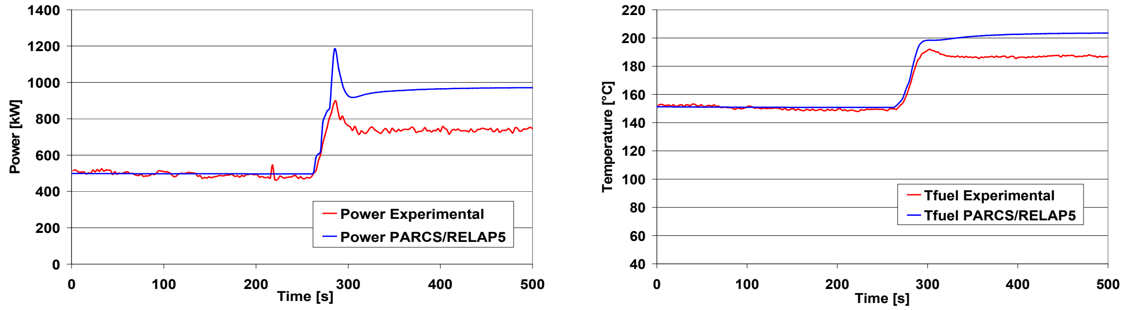


Figure 12: Comparison between the experimental and numerical quantities in the test #48

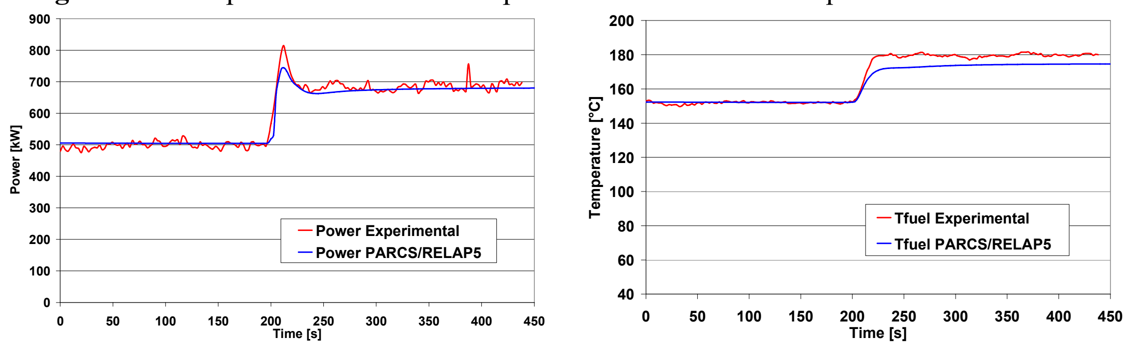
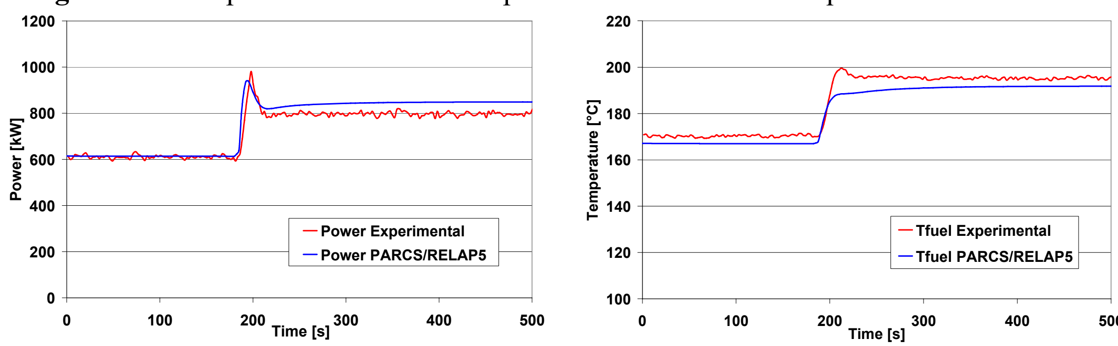
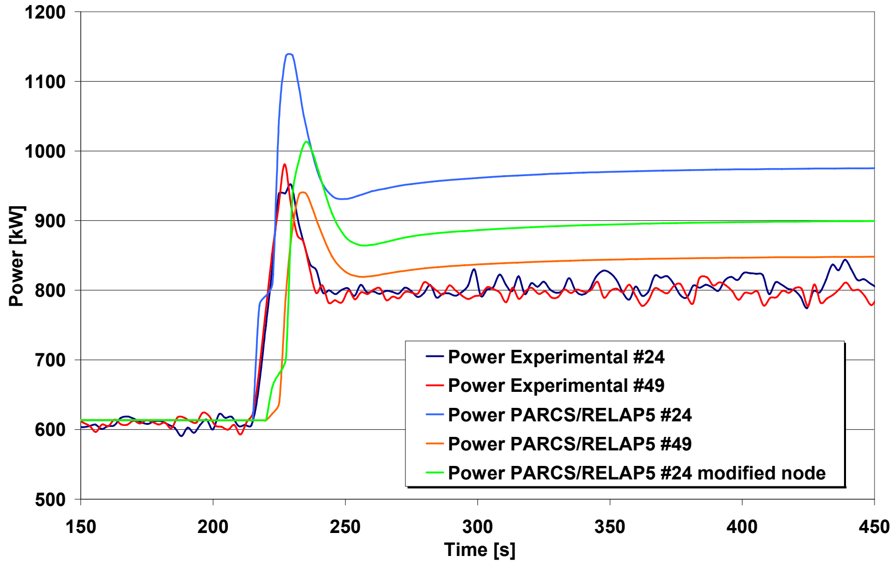


Figure 13: Comparison between the experimental and numerical quantities in the test #49



From these comparisons it is possible to underline that, although in general the calculated power and fuel temperature reproduce quite well the experimental trends, emerges insufficiencies of the numerical model adopted are present. The reason of the major discrepancies could be explained considering in particular the transients #24 and #49 (Fig. 8 and 13), which have very similar boundary and initial conditions as pointed out in Tab. 1. In fact, putting the transients in direct comparison it is possible to denote a very large difference in the simulation results (Fig. 14).

Figure 14: Comparison between the experimental and simulated power in test #24 and #49



The difference is due to a numerical limits in defining the initial control rod position, as in the test #49 the Shim-1 is slightly above an axial node of the coarse mesh adopted, while in the case of test #24 is slightly below the same axial node and this fact essentially affects the reactivity introduced during the transient. In other words, the code is not able to interpret the reactivity introduced by a partial insertion of the control rod between two axial nodes. A re-calculation of the transient #24 with a more detailed axial meshing around the node interested by the initial position of the Shim-1 results in a reduction of this discrepancy, but it also reveals the lack of generality in the model. The same reasons allow to understand the differences respect the experimental power in the tests #46/47, where the initial control rod positions are set slightly above an axial node. The solution at this serious problem is the necessity of use a numerical model able to considered the partial insertion through the nodes of the control rod, for instance the control rod cusping model already implemented in the PARCS code because the version (v1.01) currently employed is not possible to activate this model in coupling with RELAP5.

4. Conclusion

The assessment of the thermal-hydraulic model with the support of SIMMER-III CFD code has allowed to improve its capabilities to reproduce the reactivity experimental transients conducted in the RC1-TRIGA test campaign. Moreover, the benchmark of PARCS code against an analytical simplified problem, based on TRIGA reactor, highlight the effectiveness of the code to treat the critical and the sub-critical reactor.

The thermal-hydraulic/neutronic model developed for the RELAP5/PARCS coupled-code has been tested against the data coming from the experimental campaign of RC1-TRIGA reactor. The first results are encouraging but in the simulation of the reactivity transients they show a strong dependency from the nodalization adopted in the neutronic model.

In the near future will be very important to acquire a more recent PARCS version like the 2.4 version or subsequent, where the cusping model, able to refine the meshing between the axial nodes, works also in coupling calculations.

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