Interpretation of TRIGA Experimental Data with SIMMER-III Code for RELAP5 Model Evaluation and Transient Analysis

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ABSTRACT

The experimental data obtained in a test campaign carried out in the TRIGA reactor of ENEA/Casaccia are being used to validate a coupled RELAP5/PARCS numerical model for the thermal-hydraulic and dynamic simulation of the reactor.

The tests conducted 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-III code (CFD two-dimensional code) at different core power levels trying to reproduce the experimental measurements.

The results of SIMMER-III code were used to fit and validate the simplified 1-D model of the RELAP5 code used for thermal-hydraulic transient analysis of TRIGA reactor. Finally, this paper presents the interpretation of some reactivity transients using this improved T/H model with the RELAP5 point-kinetics neutronic model.

1 INTRODUCTION

Some test campaigns were carried out in the TRIGA reactor of ENEA/Casaccia [1] to investigate the reactor behaviour in the nuclear power range 200-1000 kW as a support to proposed sub-critical experiments aimed at the validation of the dynamic behaviour of an Accelerator Driven System. These tests concerned both the thermal-hydraulic characterization of the reactor core and its dynamic behaviour under normal operating conditions.

The experimental data obtained in critical core configuration are being used to validate a numerical model for the thermal-hydraulic and dynamic simulation of the reactor. Such a model based on T/H RELAP5 code and on neutronic 3-D PARCS code will be suitably modified for normal and accident transient analysis of TRIGA in subcritical configurations.

The SIMMER-III CFD code was used for a better interpretation of the experimental data and the evaluation of water mass flow rate through the core at different core power levels.
These results were used to fit and validate the simplified 1-D model of RELAP5 code to be introduced in the coupled RELAP5/PARCS numerical model of TRIGA.

2 TEST CAMPAIGN IN THE TRIGA REACTOR

The TRIGA experimental reactor at the ENEA/Casaccia research centre is a pool-type thermal reactor of 1 Mw power. The scheme of TRIGA is depicted in Fig. 1. The reactor core is at the bottom of a seven meter high water pool which provides both core cooling and shielding. The core power is removed by natural circulation; the hot water is taken from the pool top, cooled by an external circuit, and returned back at the pool bottom in the annular space around the core. Part of cold water (20%) is injected at high velocity above the core to avoid channelling effect and increase the transit time of N-16 radioactive nuclide allowing its decay before reaching the pool surface.

The reactor core (see Fig. 2) is composed of 127 elements, 106 of which are fuelled with a H-Zr-U alloy that provide a very high prompt negative reactivity coefficient with the fuel temperature increase. The water temperature is monitored by several thermocouples located at different radial positions at core inlet and outlet.
Two series of tests (n.54 and n. 55) were carried out in the TRIGA reactor in 2004 to characterize the thermal-hydraulic and dynamic behaviour of the core under normal operating conditions. During the tests the power was increased step-by-step (up to about 1 MW in the first test series and 600 kW in the second one) to investigate steady-state conditions at different power levels. The thermocouple responses during the steady-state revealed that the water temperature in the upper part of the core, just below the upper grid, was continuously oscillating (see Fig. 3). Therefore, a radial map of the temperature across the core was evaluated by the mean values of each thermocouple over a long observation time.

### Fig. 2: TRIGA reactor core

### Fig. 3: Thermocouple data at core outlet for n. 54 test series

#### 3 THE SIMMER-III CODE AND TRIGA REACTOR MODELLING

The SIMMER-III code [2], jointly developed by JNC (J), FZK (D) and CEA (F), is an advanced safety analysis computer code originally developed to investigate postulated core
disruptive accidents in liquid-metal fast reactors. SIMMER-III is a two-dimensional, three-velocity field, multiphase, multicomponent, Eulerian, fluid-dynamics code coupled with a space-dependent neutron kinetics model. By integrating all the original physical models, SIMMER-III is now applicable to a large variety of reactor calculations and other complex multiphase flow problems.

The TRIGA reactor has been modelled in 2-D cylindrical geometry with the SIMMER-III code. The modelling (see Fig. 4) was limited to the region below the reflector zone, because of the high asymmetry of water flow patterns induced by the cold water jet above this zone, which cannot be modelled in cylindrical geometry. The core is represented by six radial fuel rod rings and discretized in fifteen axial nodes. Transverse flows of water across the core induced by radial power distribution are computed by the code. Realistic pressure loss coefficients have been considered at the bottom and top core grid locations. The top grid coefficient was calibrated to best fit the temperature measurements at the highest nuclear power level.

![Fig. 4: Nodalization of the TRIGA reactor with the SIMMER-III code](image)

### 4 INTERPRETATION OF T/H MEASUREMENTS WITH SIMMER-III CODE

The tests conducted at different power levels (n. 54 and 55 test series) provided the temperature measurements at the core inlet and outlet that allow the evaluation of DTs through the core at different radial positions. The water mass flow rate can be then deduced from the average core DT as a function of core power value (Mass flow rate = Power/Cp/average DT). In a first approximation, for the n. 54 test series, the average core DT was calculated as a simple mean value, without taking into account the radial position of thermocouples (Flow_0 curve in Fig. 5). The Flow_A curve in the same Figure represents water mass flow rate values calculated by averaging DTs according to different cross-flow areas related to the thermocouple radial position, and assuming a constant water velocity profile at the core outlet. To better interpret the DT measurements and more precisely evaluate the average core DT, the radial profile of water velocity at the core outlet should be known. For this purpose the SIMMER-III code was then employed.
Fig. 5: Water mass flow rate through the TRIGA core (n. 54 t. s.)

Several calculations have been performed with SIMMER-III at different core power levels trying to reproduce the experimental measurements (DT radial profile through the core). Good agreement was found at the highest power level (960 kW, see Fig. 6) with the SIMMER-III model previously described.

In order to well reproduce the experimental data at lower power level down to 180 kW, the core hydraulic diameter was progressively reduced down to 1/4 of its actual value. This approach could be considered reasonable if we assume that, decreasing the power level and then the mass flow rate through the core, the perturbations induced by the injection of cold water at high velocity just above the core outlet rise in importance. This phenomenon seems to be confirmed by the oscillating behaviour of thermocouple readings at core outlet.

Fig. 6: DTs through the TRIGA reactor core in n. 54 t. s. at various power levels
Although the approach used has many limitations and it is not theoretically funded, the simple assumption on hydraulic diameter value helps to best fit the experimental data. The validity of this assumption is, of course, limited to this specific application only.

The radial DT profiles calculated by SIMMER-III at various power levels are compared with the experimental data in Fig. 6. In order to better compare the results, the data are interpolated by third-order polynomial curves.

The radial velocity profiles calculated by SIMMER-III (see Fig. 7) were used to best evaluate the average core DTs and therefore the water mass flow rates through the core (Flow_AV curve in Fig. 5). The agreement of these last values with the water mass flow rates computed by SIMMER-III (Flow SIMMER curve in Fig. 5) confirms the validity of the approach used in the present analysis.

![Fig. 7: Water velocity radial profiles at the outlet of TRIGA core (n. 54 t. s.)](image)

The same approach was used to interpret the T/H measurements of the n. 55 test series. No calculations with SIMMER-III were carried out, but the radial velocity profiles calculated by SIMMER-III for the n. 54 t. s. were used at corresponding nuclear power levels. The results of the interpretation analysis for this last test series are shown in Fig. 8 (a lower water temperature at core inlet makes the results different from the previous test series at similar power levels).

![Fig. 8: Water mass flow rate through the TRIGA core (n. 55 t. s.)](image)
5 EVALUATION OF THE RELAP5 MODEL

The RELAP5 code [3], developed by INEEL for the US-NRC, is a lumped parameter code used for best-estimate thermal-hydraulic transient analysis in light water reactors. The code is provided with a point-kinetics neutronic model to compute core nuclear power taking into account selected reactivity feedbacks. The MOD3.3 version of thermal-hydraulic RELAP5 code in coupling mode with the neutronic 3-D PARCS code has been proposed for the safety analysis of the TRADE experiment. The present T-H model of the TRIGA reactor is based on RELAP5 1-D modules (Fig. 9); the core is simulated with only one hydraulic channel, therefore, no spatial distribution of water flow velocities and temperatures are calculated by the code.

Fig. 9: RELAP5 model of the TRIGA reactor

The results of the SIMMER-III analysis have been then used to fit and validate this RELAP5 model. Pressure loss coefficients have been defined according to the SIMMER-III model at the bottom and top core grid locations. Following the SIMMER-III approach, a reduced core hydraulic diameter (about 1/3 of the actual value) was fixed in the model to best fit the experimental data in the whole range of power. The Fig. 10 shows a very good agreement of RELAP5 results with SIMMER-III results and interpreted experimental data for both n. 54 and 55 test series.

Fig. 10 – Validation of the RELAP5 TRIGA model

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6 TRANSIENT ANALYSIS WITH RELAP5 CODE

Several reactivity transients have been carried out in the TRIGA reactor, in critical configuration, to investigate the response of the plant when positive reactivity is introduced starting from different reactor operating conditions. The power peak induced by the reactivity insertion is limited by the prompt negative reactivity coefficient introduced following fuel temperature increase. After reaching a maximum, the core power decreases and tends to stabilize at a new power level higher than the original one. Some of these tests have been analysed with the improved thermal-hydraulic RELAP5 model using the internal point-kinetics neutronic model. Concerning the feedback effects, the fuel effect has been assumed to be dominant; therefore a fuel reactivity coefficient has been deduced from previous experimental data [4], whereas any other feedback effect has been neglected. As an example, the results of two analyses (tests n. 3 and n. 23) are presented in this paper in comparison with the experimental data.

In test n. 3, after a steady-state at 180 kW power, a reactivity step of about 175 pcm is introduced in few seconds by partially extracting one control rod. The core power rises up to 310 kW and then decreases and stabilizes at about 250 kW.

The RELAP5 results for this test are presented in Fig. 11 in comparison with the experimental data. The core power trend (Fig. 11-a) is well reproduced by the code thanks to the good prediction of fuel temperature increase (Fig. 11-b) and use of a suitable prompt temperature reactivity coefficient which has been deduced from previous experimental campaigns.

The experimental values of average core DT and mass flow rate through the core shown in Figs. 11-c and 11-d, respectively, have been deduced from core thermocouple readings following the approach described in the previous section. The mean value of oscillating measurements and the relative increase between the initial and final core power levels well match the stable values calculated by RELAP5. The high peak value in mass flow rate
experimental data around 230 s (Fig. 11-d) is not realistic as the interpretation method used is not reliable in transient conditions.

Differently from test n. 3, the test n. 23 starts from a steady-state at 600 kW power and the reactivity introduced is about 285 pcm. The core power reaches a maximum of 925 kW and stabilizes at nearly 800 kW.

The RELAP5 results in comparison with experimental and interpreted data are presented in Fig. 12. In general, the discussion and conclusions from the previous test analysis are still valid for test n. 23. Although the fuel temperature increase is still well predicted, there is a discrepancy between measured and calculated temperature levels. This can be explained both by uncertainty on experimental mean value (very few measurements available at present [5]) and by the fact that no radial power distribution is taken into account in RELAP5 calculation.

![Graphs showing RELAP5 results for n. 23 test](image)

**Fig. 12 – RELAP5 results for n. 23 test**

7 CONCLUSIONS

The SIMMER-III code has been successfully employed in the interpretation of TRIGA experimental data for natural circulation characterization at different power levels. Some model corrections (reduced core hydraulic diameter) have been introduced to well predict water mass flow rate through the core in the whole range of power. More than on a theoretical approach, the work is based on a simple assumption on hydraulic diameter value that helps to best fit the experimental data. The validity of this assumption is, of course, limited to this specific application only.

The thermal-hydraulic RELAP5 model of the TRIGA reactor has been improved and successfully validated following the approach used with the SIMMER-III code. The analysis of some reactivity tests with RELAP5 code has demonstrated, within the limits of our assumptions, the suitability of the improved thermal-hydraulic model to evaluate the reactor behaviour under transient conditions.
REFERENCES


