

EXPERIMENTAL BENCHMARKS AT JSI THE TRIGA MARK II REACTOR FOR VALIDATION OF COMPUTER CODES AND NUCLEAR DATA

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ABSTRACT

Benchmark experiments performed in an operating research reactor cannot achieve the same level of accuracy as benchmarks in dedicated facilities that are specifically designed for such a purpose. However research reactors offer a great opportunity for benchmark experiments when designed and performed with great caution and accuracy. The paper describes a series of experiments performed at the JSI TRIGA reactor that can serve as benchmark experiments for validation of computer codes and nuclear data. The experiments described are: criticality, self-shielding effect determination for dosimetry analysis, relative $^{197}\text{Au}(n,\gamma)$ and $^{27}\text{Al}(n,\alpha)$ reaction rate measurements in irradiation channels, absolute and relative $^{197}\text{Au}(n,\gamma)$, $^{235}\text{U}(n,f)$ and $^{238}\text{U}(n,f)$ reaction rate measurements in the reactor core, experiments on burnup, kinetic parameters, control rod worth and the isothermal reactivity coefficient. Some of the experiments have already been evaluated and are available to the worldwide community, while the others are yet to be evaluated.

1. Introduction

The need for benchmark experiments has already been identified in the international community resulting in several international projects, some of them coordinated by the OECD Nuclear Energy Agency, i.e. the International Criticality Safety Benchmark Evaluation Project (ICSBEP) [1], the International Reactor Physics Experiment Evaluation (IRPhE) Project [2] and the Shielding Integral Benchmark Archive and Database (SINBAD) [3]. Most of the benchmark experiments described in the above mentioned databases were performed in dedicated facilities and consequently feature relatively small experimental uncertainties. The number of benchmark experiments compiled in the databases is growing continually as new experiments are performed or new information about past experiments is evaluated. In parallel the need for validation of computer codes as well as nuclear data on reliable and well documented experiments is growing together with code and data development.

Benchmark experiments performed in an operating research reactor cannot achieve the same level of accuracy as benchmarks in dedicated facilities that are specifically designed for such a purpose [4]. This is certainly the main reason for relatively small number of evaluated benchmarks on research reactors. However, research reactors offer a great opportunity for benchmark experiments when designed and performed with great caution and accuracy.

The main purpose of this paper is to present experiments performed at the Jožef Stefan institute (JSI) TRIGA Mark II research reactor which can be used as benchmarks. The details about the experiments and the results are available to the worldwide community interested in using the data for the testing of their computer codes. In Section 2 we describe the experiments that have already been performed and evaluated. Section 3 describes experiments that have

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been performed and could be repeated but have not yet been evaluated.

2. Existing benchmark experiments

2.1. Criticality and burnup

A criticality benchmark experiment was performed in 1991, after the reconstruction of the reactor [4]. In 1999 a computational model of the reactor in MCNP [5] was developed, in order to evaluate the experimental uncertainties and to use the model to computationally support experimental campaigns at the reactor. The evaluated criticality benchmark experiment was later published in the International Handbook of Evaluated Criticality Safety Experiments (ICSBEP) [6] under the ICSBEP identifier IEU-COMP-THERM-003. The main results of the benchmark are presented in Table 1. Until recently, this was the only publicly available TRIGA criticality benchmark featuring homogenous mixture of fuel, moderator and Zr. Due to U–ZrH fuel, it is very sensitive to the Zr absorption and scattering cross sections [7][8]. In 2011 criticality benchmark experiments from the NRAD reactor at the Idaho National Laboratory were also evaluated and published in the ICSBEP Handbook [9] under the ICSBEP identifier IEU-COMP-THERM-013.

Tab 1: Experimental, benchmark model and calculated k_{eff} with the uncertainties.

Case	Experimental k_{eff}	Benchmark model k_{eff}	Calculated	
			ENDF/B-VI.6	ENDF/B-VII
Core 132	0.99865 ± 0.00015	1.0006 ± 0.0056	1.0001 ± 0.0001	1.0059 ± 0.0001
Core 133	1.00310 ± 0.00015	1.0046 ± 0.0056	1.0048 ± 0.0001	1.0107 ± 0.0001

After several years of operation, the criticality benchmark was repeated with burned fuel [10]. This benchmark provides useful information for testing of burnup calculation codes and the required nuclear cross-sections, as well as the reactivity effect of fuel burnup. In addition, the burnup of individual fuel elements was measured by reactivity experiments [11][12]. Recently we initiated activities to record the operational history of the reactor thoroughly, together with excess reactivity and control rod worth measurements, which could be used for validation of deterministic core management codes such as TRIGLAV [13] or Monte Carlo codes such as SERPENT [14]. It is important to note that one of the major uncertainties in fuel burnup determination is the uncertainty in the measured reactor power level [15]. The major source of the power level uncertainty in the JSI TRIGA Mark II reactor is the neutron flux redistribution or tilt in the radial and axial direction due to asymmetric control rod insertion. As the reactor power is measured with one detector only, the error in the measured power level at some location in the core can be as much as 20 % - 30 %. This can be corrected by applying corresponding correction factors [16] or measuring the reactor power using multiple detectors [17][18]. The above approaches were verified experimentally and computationally and are described in Section 2.2.

2.2. Reaction rate measurements

The neutron activation method was used to experimentally verify the calculated reaction rates in the irradiation channels of the reactor and in the measurement positions (MPs) between the fuel elements in the reactor core. The reactor core is schematically presented in Figure 1.

In activation foil dosimetry techniques, reaction rate derivations from radioactivity measurements require the evaluation and the correction of the self-shielding effect occurring inside the dosimeter. Indeed, the actual reaction rate is lowered by two potential effects:

- A spatial self-shielding factor associated to the neutron flux depression around and inside the dosimeter (absorbed neutrons are no longer available).
- A resonance self-shielding factor induced by the high reaction rate in the narrow energy domains around the resonance energies in the reaction cross sections.

The JSI and CEA have worked together to evaluate the magnitude of the self-shielding effect on various dosimeter types using measurements performed in EOLE CEA facility and modelling code schemes [19]. For example, resonance self-shielding factors have been evaluated to 0.789 for aluminium foils (2.0mm thickness and 8 mm diameter) and 0.268 for gold foils (0.25 mm thickness and 8 mm diameter) (Table 4 in [19]).

In the first experiment performed in the TRIGA reactor aluminium-gold (Al(99.9 wt. %) - Au(0.1 wt. %)) foils (disks 5 mm in diameter and 0.2 mm thick) were irradiated in 33 locations; 6 irradiation channels in the core (CC, F15, F19, F24, F26) and 27 irradiation channels in the carousel facility in the reflector [20]. The dosimeter geometry and composition have been chosen to reduce as much as possible the magnitude of the above mentioned self-shielding effects. After the irradiation, the activities of the individual samples were measured using a High-Purity Germanium detector (HPGe). The following two activation reactions were considered in the experiment: $^{27}\text{Al}(n,\alpha)$ and $^{197}\text{Au}(n,\gamma)$. Fig 2 and 3 present the comparison between the normalized calculated and measured reaction rates. For the in-core irradiation channels the reaction rates are presented relative to the reaction rates in the central irradiation channel (CC), for the irradiation channels in the carousel facility the reaction rates are normalized to their average value throughout the carousel facility.

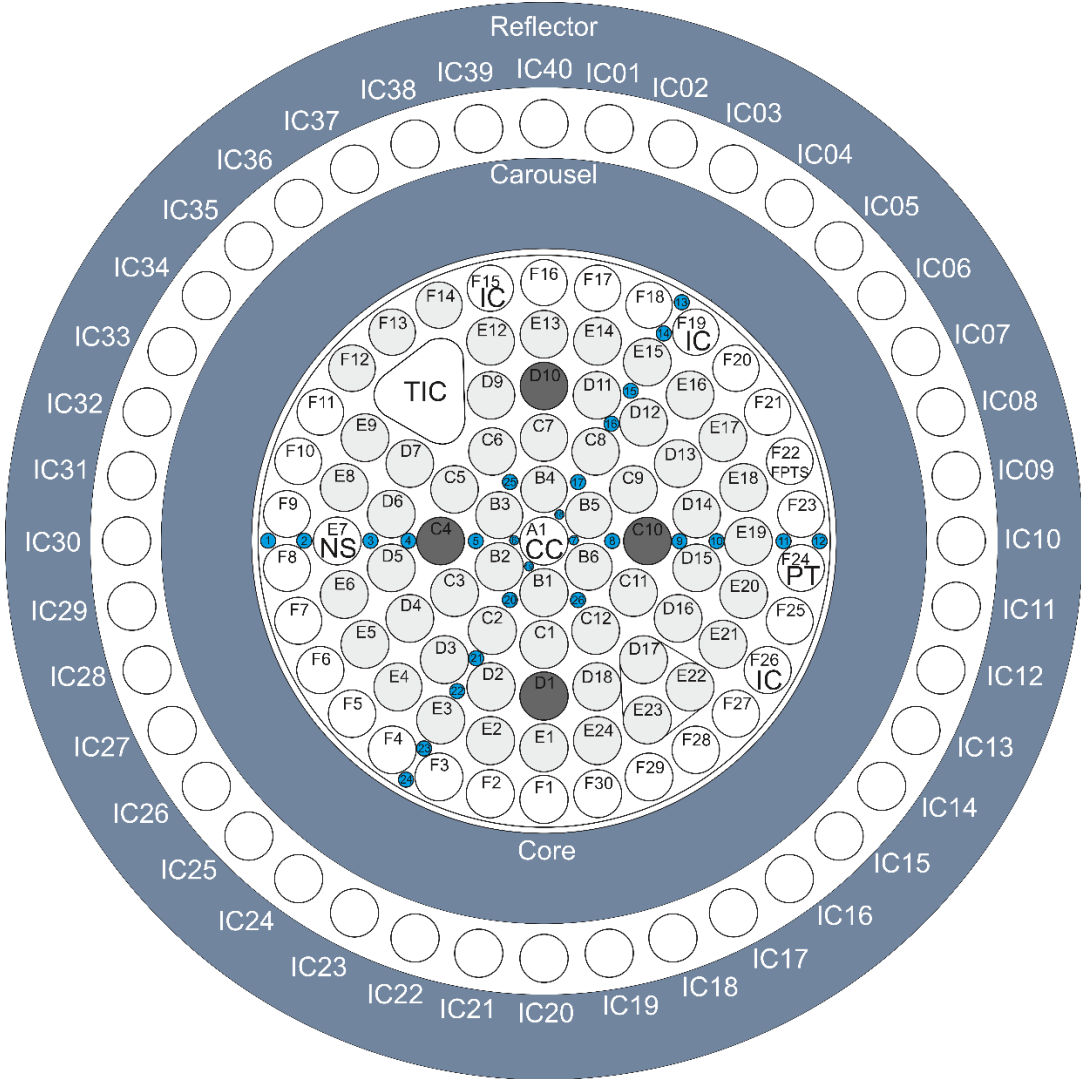


Fig 1. Schematic top view of the TRIGA reactor with marked irradiation positions. The white circles denote either empty positions in the reactor core or irradiation channels: CC, TIC, F15, F19, F22, F24, F26, IC01-IC40. The light grey circles denote the fuel elements and the

dark grey circles denote the control rods. The smaller blue circles denote the measurement positions (MPs) in the core, where axial reaction rate profile measurements were performed.

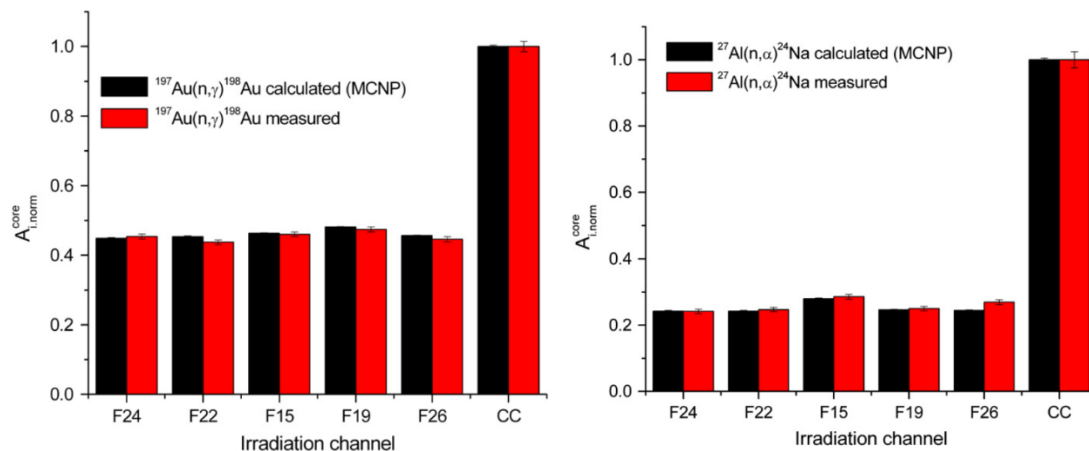


Fig 2. Calculated and measured $^{197}\text{Au}(n,\gamma)$ and $^{27}\text{Al}(n,\alpha)$ reaction rates in the irradiation channels in the reactor core, relative to the values in the central channel (CC). The error bars represent 1- σ experimental uncertainties in the measured results and 1- σ statistical uncertainties in the calculated results.

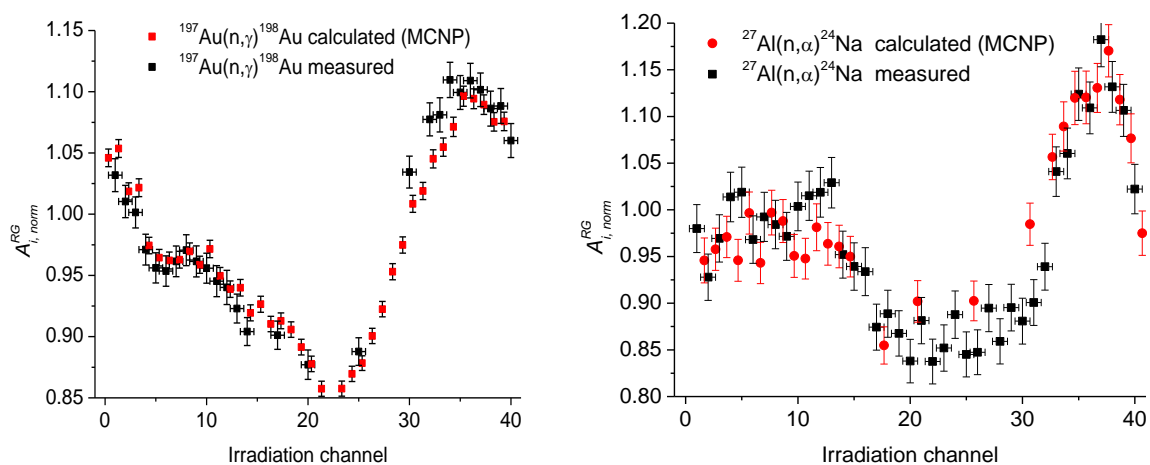


Fig 3. Calculated and measured $^{197}\text{Au}(n,\gamma)$ and $^{27}\text{Al}(n,\alpha)$ reaction rates in the carousel facility, normalized to the average value. The Y represent 1- σ experimental uncertainties in the measured results and 1- σ statistical uncertainties in the calculated results. The X error bars represent the uncertainty in the carousel position during the experiment.

In the second experiment the axial profiles of the $^{197}\text{Au}(n,\gamma)$ reaction rates were measured in four measurement positions in the core at full reactor power [21]. To accomplish this, 5 mm lengths of Al-0.1% Au wire, 1.0 mm in diameter were irradiated inside specially designed aluminium probes. A photograph and a technical drawing of the probes are displayed in Figure 4. The probes consist of a central rod, 5 mm in diameter with 69 through-holes perpendicular to its axis, 1 mm in diameter and spaced every 1 cm. The rod fits into aluminium sleeves, which have several steps of different diameter in the top part. The probes are designed to fit into any measurement position in the reactor core, which are either 8 or 10 mm in diameter.

Four probes were irradiated in the measurement positions MP15, MP16, MP17, MP21. The induced activities were measured using a HPGe detector. Figure 5 (left) presents the comparison between the calculated and experimental reaction rates per target atom for the MP16 position. A fuel element is schematically displayed in Figure 5 for spatial reference. It

was shown that agreement between the calculated and the measured values is mostly within 5% and within the experimental uncertainties.

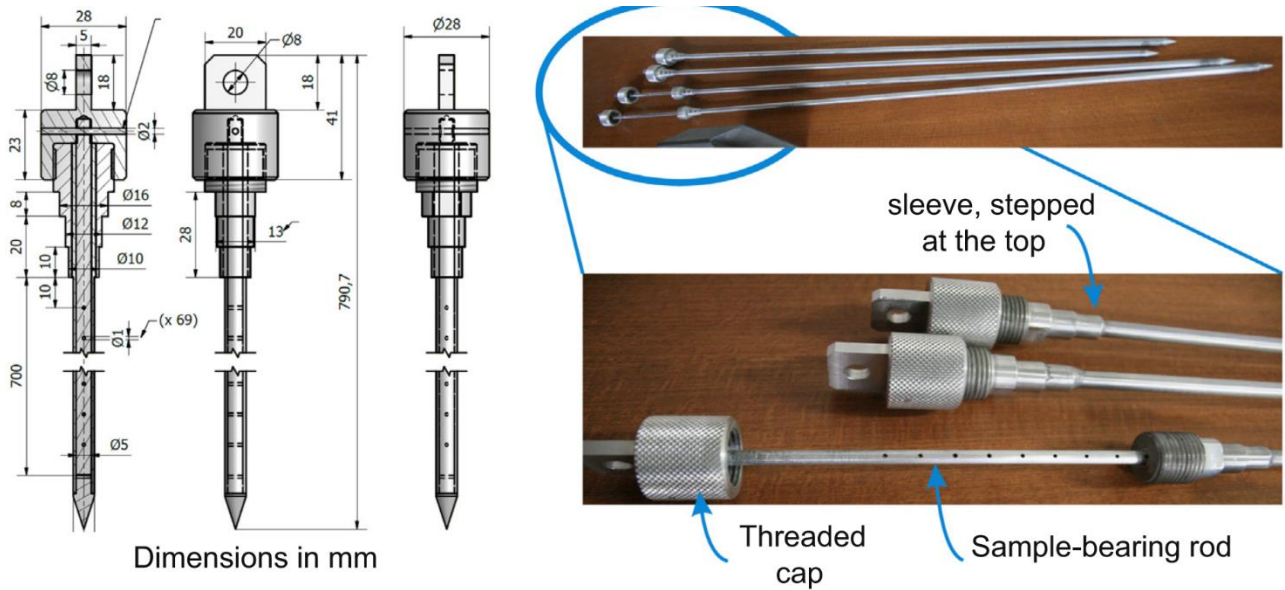


Fig 4. Aluminium probes designed for measurements of axial distributions of reaction rates.

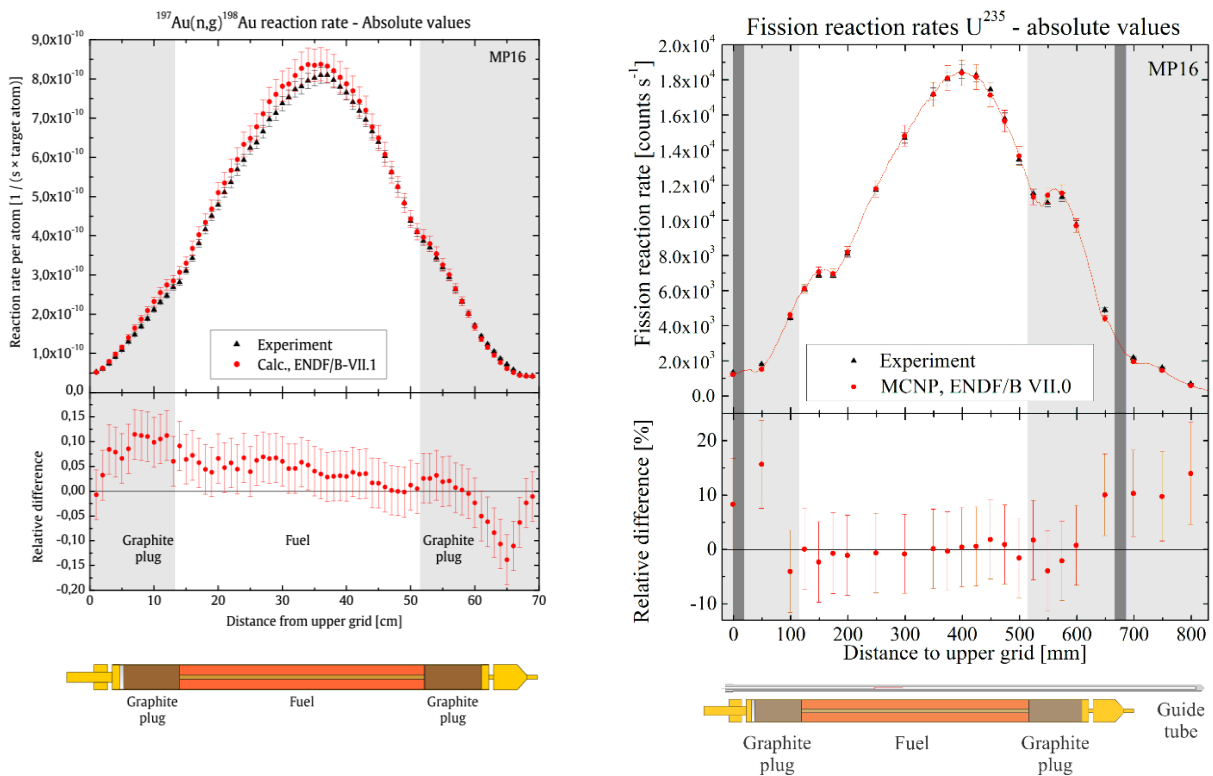


Fig 5. Calculated and measured axial profiles of the $^{197}\text{Au}(n,\gamma)$ reaction (left) and the $^{235}\text{U}(n,f)$ reaction (right). Below the graphs a fuel element is schematically displayed to provide spatial reference.

In 2011 another benchmark experiment was performed where in-core axial distributions of fission rates were measured. In the experiment two absolutely calibrated CEA miniature watertight fission chambers were used [22][23]. The fission chambers were made of stainless steel and filled with a gaseous mixture of argon and nitrogen. They had a cylindrical shape 3

mm in diameter and a 4 mm active part length, as shown in Figure 6. The fission chambers differed in the type and amount of fissionable material deposited on the anode of the active part – one with a fissionable coating composed mainly of ^{235}U and the other mainly of ^{238}U [24][25]. The characteristics of each fission chamber are shown in Table 2.

Tab 2: Isotopic composition of the fissile coating in the fission chambers.

Chamber	^{234}U [at. %]	^{235}U [at. %]	^{236}U [at. %]	^{238}U [at. %]	Total deposit mass [μg]
FC ^{235}U	0.063	98.490	0.038	1.409	8.86
FC ^{238}U	0.0003	0.0359	0	99.964	91.67

Fission chambers were deployed into the reactor core using a specially designed positioning system, composed of hollow aluminium guide tubes, a drive mechanism and data acquisition system. The system installed in the reactor is schematically displayed in Figure 7. A mineral insulated integrated cable connecting the fission chambers to the data acquisition system, was used for inserting and withdrawing the chambers into and out of the reactor core. The fission chamber position was regulated by a commercially available pneumatic drive consisting of a series of valves and pistons, all controlled by a microcontroller. The axial positioning was ensured by an incremental system, which measures the chamber position relative to the reference position at the bottom of the guide tube, and a linear system for absolute position measurements through the use of a magnetized strip. Measurements of the axial profiles of the absolute fission rates over the complete core height were performed with the two fission chambers in 9 radial measurement positions: MP14, MP15, MP16, MP17, MP20, MP21, MP22, MP23 and MP25, which are shown in Fig 1. The experiment was modelled in detail with the Monte Carlo method and a comparison between the measured and calculated fission rates was performed, which can be seen in Figure 5 for measuring position MP16 [26]. The agreement between the absolute reaction rates is very good for both fission chambers and all measuring positions, with the average relative discrepancies being below 5 %. In order to complete the validation process, an exhaustive Monte Carlo model of the reactor has been created and an extensive evaluation of experimental and computational uncertainties has been performed. This included the study of fission chamber positioning uncertainties, material composition perturbation and the evaluation of other uncertainty sources like the use of different nuclear data libraries and core temperature effects.

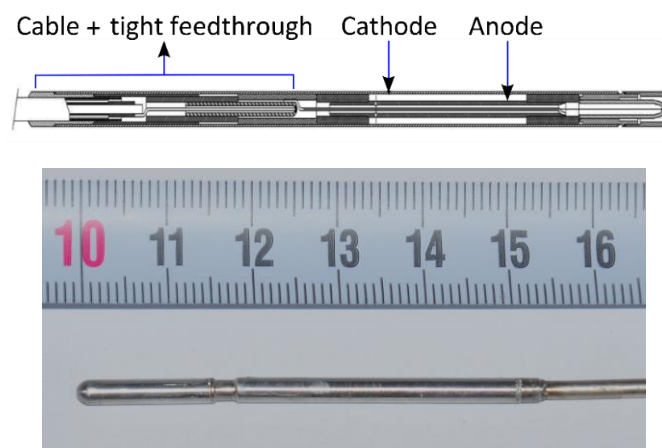


Fig 6. Schematic drawing and photograph of a 3 mm fission chamber used in the experiments.

The above mentioned measurements are used for the verification and validation of the detailed geometric Monte Carlo computational model of the reactor. An interesting feature of the above experiments is that they provide absolute values of the reaction rates, which are normalized to the total reactor power – hence they can also be used for the validation of the Monte Carlo computation normalization coefficient [27]. The benchmark was published in the Handbook of

the International Reactor Physics Experiment Evaluation (IRPhE) project, under the identifier TRIGA-FUND-RESR-002 [28].

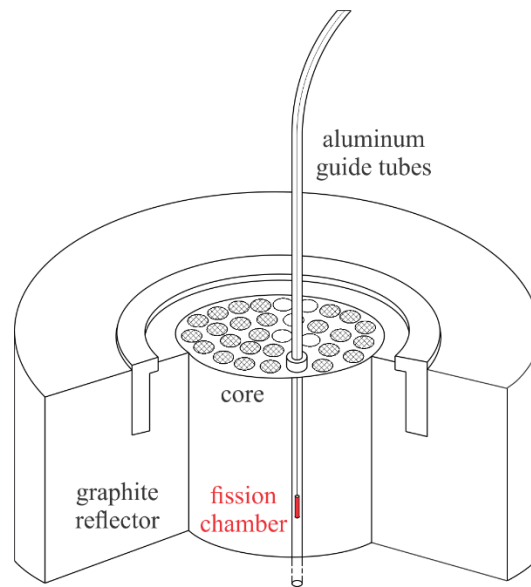


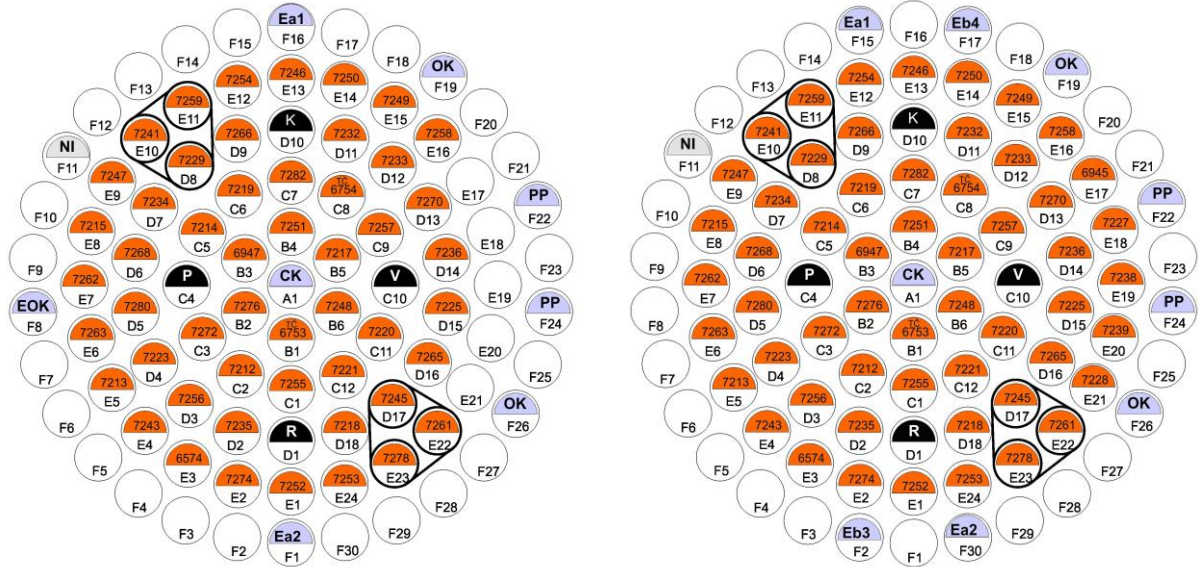
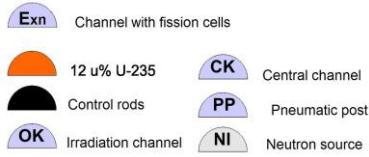
Fig 7. Schematic drawing of the experimental setup for the fission rate measurements using fission chambers inside aluminium guide tubes inserted in the reactor core.

2.3. Reactor kinetics

In 1991 the reactor was equipped for pulse mode operation. In total more than 150 pulses were performed. All of them were later analysed for validation of the so-called Fuchs-Hansen model and its improvement [29]. The pulse experiments also provided valuable information on reactor kinetic parameters such as prompt neutron lifetime l and the effective delayed neutron fraction β_{eff} . These parameters were later used for validation and verification of calculations [30].

In 2013 an experimental campaign has been carried out, in order to assess the mean neutron generation time Λ and effective delayed neutron fraction β_{eff} of the JSI TRIGA reactor [31]. For this purpose, the cross power spectral densities of the signals of two fission chambers have been measured with an Agilent spectrum analyser. The reactor operated with five core configurations corresponding to different control rod and fission chamber positions. A method of comparison between the cross power spectral densities using the Weich statistical test without assuming any kinetic model has been performed: it shows that it is possible to distinguish between the configurations, provided a high enough signal integration time (above 3000 s) is specified. Assuming a point-kinetic model, the parameters Λ and β_{eff} have been robustly derived by the Cohn- α technique.

Λ and β_{eff} were measured for different core configurations, some of which are displayed in Figure 8. The reactor was operating at close to critical conditions in five different core configurations. These were chosen in a way to span between both the minimum and maximum possible excess reactivity, i.e. the control rods were almost withdrawn or significantly inserted. In Figure 8, two of the five measured configurations are presented, namely a) the configuration with the highest excess reactivity (denoted in Table 3 with 201b) and b) with the lowest excess reactivity (denoted with 202).



a)

b)

Fig 8. Two of the configurations for Λ and β_{eff} measurements; a) core configuration 201b with the lowest excess reactivity and b) core configuration 202 with the highest excess reactivity.

The results of the measurements of β_{eff} and Λ for the five configurations [31], including the corresponding integration time of the measurement, are presented in Table 3.

Tab 3: Inferred β_{eff} and Λ for each of the five measured configurations, the corresponding integration time is given [31].

Core	Time	β_{eff} (pcm)	Λ (μs)
201(b)	19448	689 ± 44	38 ± 4
203	5660	686 ± 45	40 ± 5
202	5680	721 ± 46	42 ± 6
202(b)	5620	730 ± 47	42 ± 6
204	5764	736 ± 47	41 ± 5

The neutron generation time is found to be rather constant from one configuration to another, but configurations with different fuel loading and control rod height yield an effective delayed neutron fraction that can be loosely distinguished at 1- σ level. The data are planned to be compiled and examined for their appropriateness to be used as benchmark experiments.

3. Planned benchmark experiments

In addition to the above measurements, we regularly measure the isothermal temperature reactivity coefficient as part of a set of practical exercises for nuclear engineering students. It is interesting to note that the temperature reactivity coefficient is slightly positive at room

temperature, i.e. up to approx. 27 °C [32]. This coefficient becomes negative with the raising of the temperature ensuring the respect of the associated safety criteria.

Control rod worth measurements are also performed on a regular basis using the in-house developed digital reactivity meter DMR-043 [33][34]. Recently, a project was initiated to evaluate the uncertainty in control rod worth by using different methods, i.e. rod swap and rod insertion method. The measurements of the integral and differential reactivity worth were simulated with a static MCNP calculation. Very good agreement was found for the rod swap method, but for the rod insertion method the high importance of the dynamic reactivity effects in the TRIGA reactor was confirmed [35]. Some of the results are displayed in Figure 9. The aforementioned dynamic effects can be reproduced numerically in a 3D dynamic simulation [36]. Using the latest upgrades to our codes [37], a full-scale uncertainty analysis is being performed and improvements of the rod insertion method are sought [38].

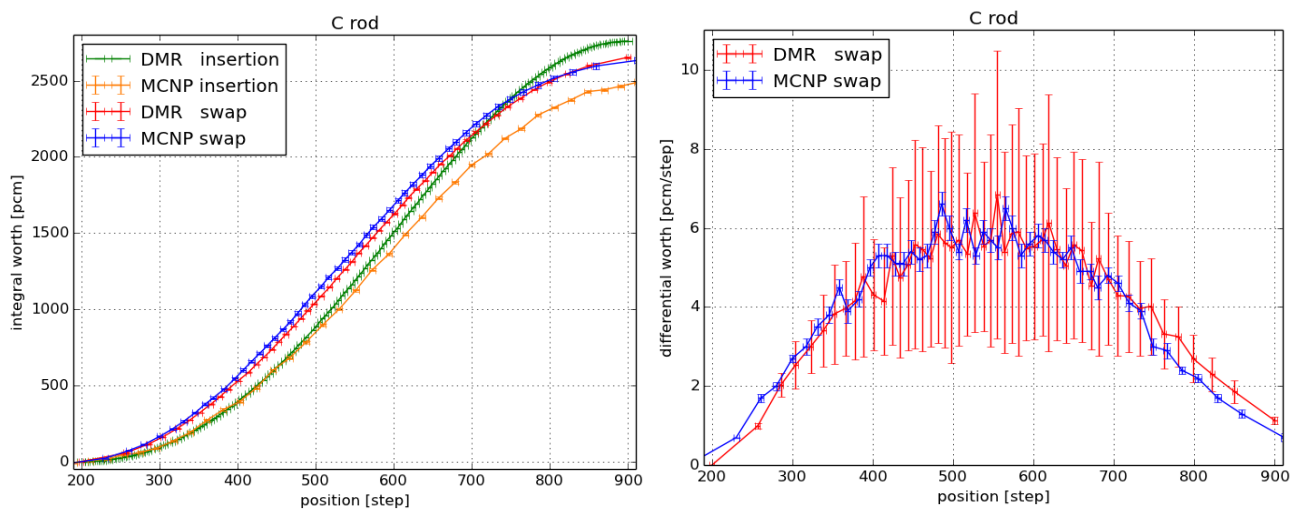


Fig 9. Comparison of measured and calculated integral (left) and differential rod worth (right) for the compensating control rod at the JSI TRIGA reactor using the rod swap and the rod insertion methods. Measurements are labelled with “DMR” and calculations with “MCNP.”

Due to the use of the reactor for radiation hardness studies, several measurements of the photon fields in the irradiation channels were made by using ionization chamber [25][39] as well as radiation sensing field effect transistors (radFETs) in operating and shutdown reactor. Such measurements are very valuable for the validation of photon production methods (during operation) as well as for validation of gamma flux and shutdown dose calculations and development of the so-called rigorous two-step (R2S) methods [40][41], which couple activation and particle transport codes and enable, for example, calculations of dose rates due to neutron induced gamma rays (prompt and delayed) as well as dose rates due to gamma rays emitted from irradiated nuclear fuel or activated materials. In addition, long-lived neutron activation products in the biological shield of the reactor were measured as well [42]. Such measurements are of high importance for safe decommissioning and for validation of activation codes.

4. Conclusions

Recent and current experimental programs performed with various nuclear instrumentation in the JSI TRIGA reactor have greatly improved the knowledge of its intrinsic physical parameters and their uncertainties (kinetic parameters, spatial flux and reaction rate distributions, power level, etc.), and therefore enhanced its experimental capabilities. Completed with a fully validated computational scheme, the presented experimental data sets allows this small and relatively old research reactor, with a rather low neutron flux ($\sim 10^{13}$ n cm⁻² s⁻¹), to efficiently

support both fundamental and applied research. The JSI TRIGA reactor can significantly contribute to the development of new methods and knowledge in reactor physics.

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