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www: <http://www.drustvo-js.si/port2001/>

e-mail: [PORT2001@ijs.si](mailto:PORT2001@ijs.si)

tel.: + 386 1 588 5247, + 386 1 588 5311

fax: + 386 1 561 2335

Nuclear Society of Slovenia, PORT2001, Jamova 39, SI-1000 Ljubljana, Slovenia



## **MONTE CARLO CHARACTERIZATION OF IRRADIATION FACILITIES IN THE TRIGA REACTOR CORE**

**Robert Jeraj, Tomaž Žagar, Matjaž Ravnik, Bogdan Glumac**

“Jožef Stefan” Institute

Reactor Physics Division

Jamova 39, SI-1000 Ljubljana, Slovenia

[robert.jeraj@ijs.si](mailto:robert.jeraj@ijs.si), [tomaz.zagar@ijs.si](mailto:tomaz.zagar@ijs.si),

[matjaz.ravnik@ijs.si](mailto:matjaz.ravnik@ijs.si), [bogdan.glumac@ijs.si](mailto:bogdan.glumac@ijs.si)

### **ABSTRACT**

Transport calculations can often provide useful information about irradiation field characteristics in addition to the measurements. For this reason Monte Carlo simulations were used to characterise irradiation facilities in the TRIGA Mark II (Ljubljana, Slovenia) reactor core. Several locations in two different core configurations were investigated. Some of the Monte Carlo results were also compared to the measurements. It was found out that the flux as well as its spectral characteristics depend very much on the position of the irradiation channel in the core and can vary significantly between different core configurations. The highest flux is achieved in the central channel of the core, and then decreases to about 10% in the rotary groove. The fast-to-thermal neutron flux ratio also varies significantly and achieves values of about 1.4 in the central channel but only 0.2 in the rotary groove. The core configuration can introduce up to 40% change in the fast-to-thermal neutron ratio, which enables optimisation of the core configuration for different irradiation channels and different irradiation goals. Comparison to the measurements showed that our Monte Carlo model can successfully predict absolute values of the flux as well as its spectral characteristics to within a few percent.

### **1 INTRODUCTION**

Exact characterisation of irradiation fields is very important for successful description of experimental conditions for experiments that are performed in the facilities. So far, mainly measurements were used to determine spectral characteristics of neutron fields in irradiation channels of the TRIGA reactor core [1]. In addition to the measurements, transport calculations can provide useful information of irradiation field characteristics. However, because of the complexity of the TRIGA reactor core geometry, only Monte Carlo transport calculations are accurate enough for reliable characterisation of the in-core irradiation fields. The Monte Carlo transport calculations have also several advantages over the measurements since they enable very fine spectral description (not limited to only few groups), direct calculation of the activation of irradiated materials and provide opportunity for optimisation of irradiation channels. The main goal of this work was to characterise several irradiation

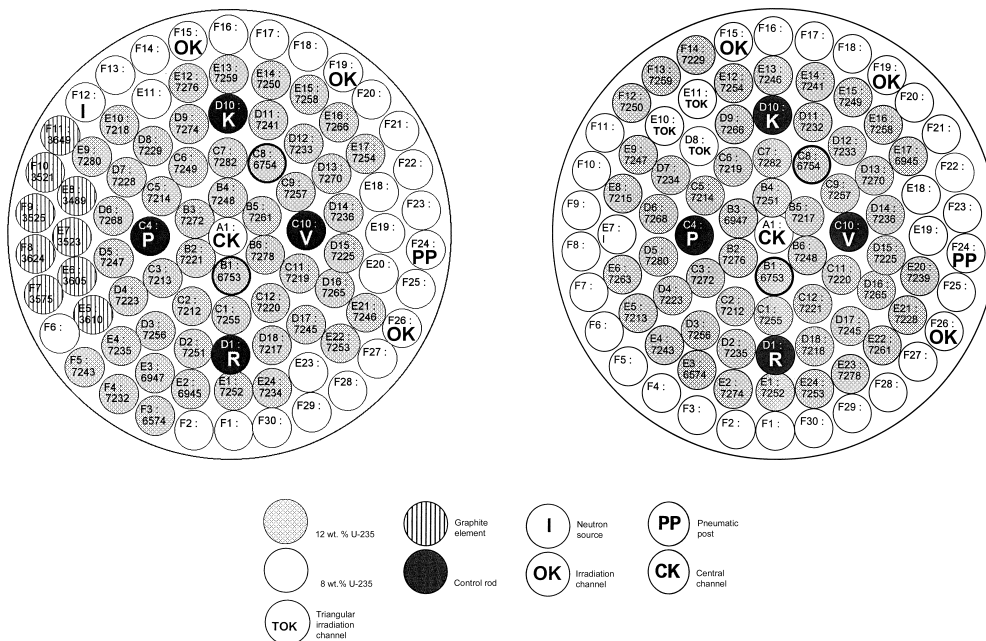
facilities in the TRIGA reactor core and outside, investigate spectral dependencies on the core configuration and compare calculations to the measurements.

## 2 MATERIALS AND METHODS

### 2.1 Monte Carlo simulations

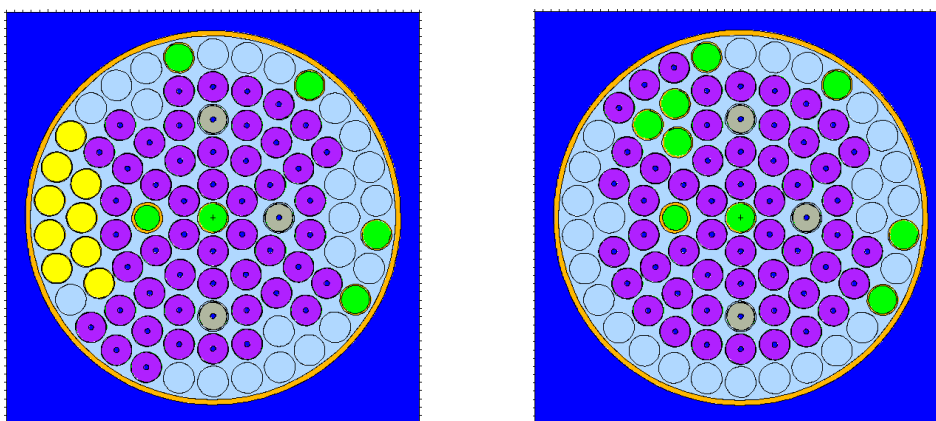
In order to get accurate Monte Carlo transport calculations, a very precise and detailed Monte Carlo model of the TRIGA Mark II reactor core in Ljubljana, Slovenia was constructed [2]. All the irradiation facilities in the core were modelled – the central channel, triangular channel and pneumatic post, together with accurate description of the core loading for any given core. It should be emphasised that the Monte Carlo model has already been successfully benchmarked on the TRIGA criticality benchmark experiment [3]. The TRIGA reactor core is also a rare example of a non-periodic compact core.

For the purpose of this study, two core configurations were considered. The first one was the core configuration labelled 152, which was set up in March 1997, and on which some of the neutron flux measurements were done (Figure 1 left). This core was designed to have a high thermal flux in the thermal column; therefore several graphite elements were inserted. In addition the core had an increased number of fuel elements in the direction of the radial irradiation channel. The second core is the current core labelled 173 (Figure 1 right), which is much more compact and designed for irradiation of the samples in the triangular irradiation channel, which comprises of the locations D8, E10 and E11.



**Figure 1:** Core configurations for two investigated TRIGA Mark II (Ljubljana, Slovenia) cores; core 152 (left) and core 173 (right). Note different core loading for different irradiation purposes. While core 152 was designed to give optimal flux characteristics in the thermal column (graphite elements on the left side of the core) and in the radial irradiation channel (more fuel elements in the bottom-left part of the core, the core 173 was designed to give higher fluxes in the triangular irradiation channel (more fuel elements in the upper-left part).

Based on these core configurations, two Monte Carlo models were constructed, which are shown in Figure 2. The model was not compromised for the accuracy, since only in this way reliable results can be obtained. In our simulations, the MCNP4b [4] Monte Carlo code was used. Only fresh fuel material composition was assumed. Most of the cross section libraries were used from the ENDF/B-VI evaluation, only the thermal cross sections for certain compounds (water, H/Zr, graphite) were based on the ENDF/B-IV evaluation. The flux calculations were performed in several irradiation channels: central channel (CK; location A1), pneumatic post (PP; location A24), irradiation channel F19 (OK, location F19), triangular irradiation channel (TOK; locations D8, E10 and E11) and outside the core in the rotary groove (RG). The flux was scored at midplane in a 10 cm high region on the channel position. Accuracy of the simulations was approximately 0.3 % for the effective multiplication factor calculations, and between 0.2 and 0.6% for the total flux calculations, depending on the location.



**Figure 2:** Monte Carlo models of the cores 152 (left) and 173 (right). Cross section through the midplane is shown. Violet and grey colours indicate fuel meat in fuel elements and control rods, respectively, green is air, yellow graphite, light blue water and dark blue the graphite reflector.

Since the MCNP results are normalised per one source neutron, the results had to be properly scaled in order to get absolute comparison to the measured flux values. The following scaling factor was used

$$C = \frac{\bar{\nu} \cdot P}{1.6022 \cdot 10^{-13} J / MeV \cdot Q_{ave} \cdot k_{eff}} \quad (1)$$

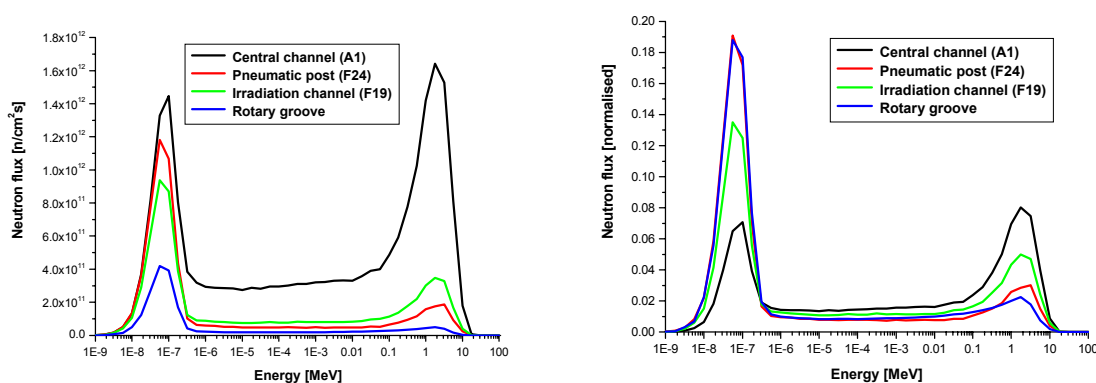
where  $\bar{\nu}$  is the average number of neutrons produced per fission (2.47),  $Q_{ave}$  is the average energy deposited per fission (196 MeV/fission) and  $k_{eff}$  is the effective multiplication factor, which was calculated for the particular core.  $P$  is the power (in Watts) at which the reactor operates. It should be emphasised that the uncertainty in the power determination is rather large and is estimated that could be of the order of 10% [5]. In our study, a slightly less conservative estimation of this error was assumed, and only 5% uncertainty was ascribed to the power determination. Because only fresh fuel was considered in our calculation, the calculated  $k_{eff}$  was different from the real  $k_{eff}$  for that particular core. Nevertheless, the flux calculations should be accurate, except if the burned fuel composition significantly perturbs the shape of the spectrum compared to the fresh fuel.

## 2.2 Measurements

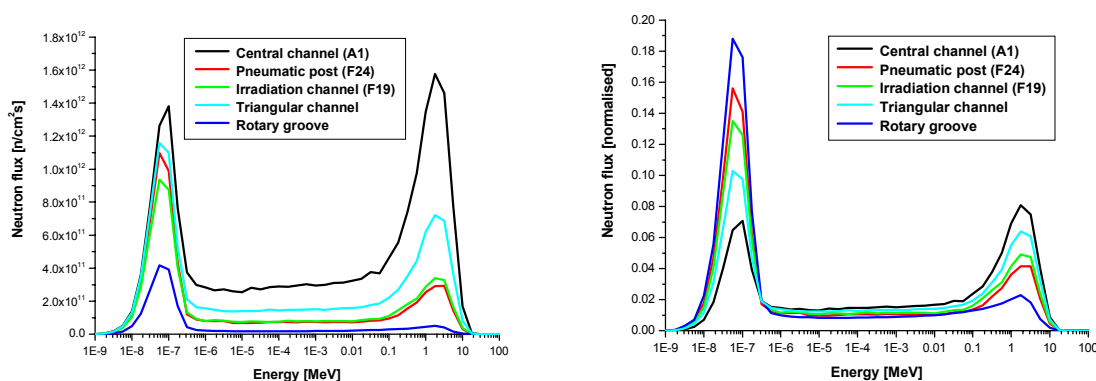
Some of the simulation results were compared to the neutron flux measurements [1], which were performed with core 152. In these experiments the following neutron activation reactions were measured:  $^{197}\text{Au}(n, \gamma)$  with bare diluted gold,  $^{197}\text{Au}(n, \gamma)$  with diluted gold under a cadmium cover,  $^{115}\text{In}(n, \gamma)$  with diluted indium under a cadmium cover,  $^{58}\text{Fe}(n, \gamma)$  under a cadmium cover,  $^{63}\text{Cu}(n, \gamma)$ ,  $^{63}\text{Cu}(n, \gamma)$  under a cadmium cover,  $^{115}\text{In}(n, n')$ ,  $^{27}\text{Al}(n, \alpha)$  and  $^{59}\text{Co}(n, p)$ . All the experiments were performed at the full reactor power of 250 kW. At the end the irradiated samples were analysed with  $\gamma$  spectroscopy. The errors of the measurements were of the order of 5%, which includes also the uncertainty of the irradiated sample position (3%).

## 3 RESULTS

The results of the Monte Carlo calculations of the neutron fluxes for both cores are given in Figures 3 and 4. Both, absolute and normalised neutron flux distributions (which give a nice visual presentation of the flux spectral characteristics) are presented. For easier comparison the flux values in different regions of the spectra and its spectral characteristics are given in Tables 1 and 2, respectively.



**Figure 3:** Neutron flux distribution in several irradiation channels of the core 152 (left) and normalised neutron flux distribution (right). Note that the total flux is plotted (integrated over the energy bin) and not the spectrum. The area under the curve directly represents the flux.



**Figure 4:** Neutron flux distribution in several irradiation channels of the core 173 (left) and normalised neutron flux distribution (right). Note similarity to the flux distribution in the irradiation channels of the core 152, except of the harder spectrum in the pneumatic post.

Neutron flux [ $10^{12}$ n/cm <sup>2</sup> s]	CK		PP		F19		TOK		RG	
	152	173	152	173	152	173	152	173	152	173
Thermal	5.75	5.49	4.19	4.03	3.47	3.50	–	4.47	1.52	1.47
Intermediate	6.77	6.48	1.07	1.36	1.77	1.77	–	3.30	0.44	0.43
Fast	7.95	7.61	0.93	1.25	1.74	1.72	–	3.55	0.26	0.26
Total	20.43	19.58	6.19	6.64	6.98	7.02	–	11.32	2.22	2.16

**Table 1:** Neutron flux at different irradiation sites for cores 152 and 173. The neutron flux spectrum is decomposed into three groups: thermal ( $E < 0.5\text{eV}$ ), intermediate ( $0.5\text{eV} < E < 100\text{keV}$ ) and fast ( $E > 100\text{keV}$ ). The flux values are given in the units of  $10^{12}$  neutrons/cm<sup>2</sup>s. The statistical error of the simulations is approximately 0.5%. The labels for the irradiation sites are the following: CK = central channel, PP = pneumatic post, F19 = irradiation channel F19, TOK = triangular channel, RG = rotary groove.

Spectral characteristics [%]	CK		PP		F19		TOK		RG	
	152	173	152	173	152	173	152	173	152	173
Thermal	28.1	28.0	67.7	60.7	49.8	50.1	–	39.5	68.4	68.3
Intermediate	33.1	33.1	17.3	20.6	25.4	25.3	–	29.1	19.9	19.8
Fast	38.8	38.9	15.0	18.8	24.9	24.6	–	31.4	11.7	11.9
Fast-to-thermal ratio	1.38	1.39	0.22	0.31	0.50	0.49	–	0.80	0.17	0.17

**Table 2:** Spectral characteristics at different irradiation sites for cores 152 and 173. The spectral characteristics are decomposed into three groups: thermal ( $E < 0.5\text{eV}$ ), intermediate ( $0.5\text{eV} < E < 100\text{keV}$ ) and fast ( $E > 100\text{keV}$ ). A fraction of the spectrum in each region is given in %. The statistical error of the simulations is approximately 0.5%. At the bottom the fast-to-thermal flux ratio is given, indicating the hardness of the spectrum. The labels for the irradiation sites are the following: CK = central channel, PP = pneumatic post, F19 = irradiation channel F19, TOK = triangular channel, RG = rotary groove.

Flux	Calculations	Measurements	C/M
Thermal	$3.47 \cdot 10^{12} (1 \pm 0.05)$	$3.19 \cdot 10^{12} (1 \pm 0.05)$	1.09
Intermediate	$1.77 \cdot 10^{12} (1 \pm 0.05)$	$1.53 \cdot 10^{12} (1 \pm 0.04)$	1.15
Fast	$1.74 \cdot 10^{12} (1 \pm 0.05)$	$1.82 \cdot 10^{12} (1 \pm 0.04)$	0.95
Total	$6.98 \cdot 10^{12} (1 \pm 0.05)$	$6.54 \cdot 10^{12} (1 \pm 0.05)$	1.06

**Table 3:** Comparison between the simulations and measurements [1] for the flux determination in the irradiation channel F19 of the core 152. The spectral characteristics are decomposed into three groups: thermal ( $E < 0.5\text{eV}$ ), intermediate ( $0.5\text{eV} < E < 100\text{keV}$ ) and fast ( $E > 100\text{keV}$ ). The error in the simulations includes uncertainty in the power determination of the reactor, which is the prevailing error. The error of the measurements includes only the error of the activation measurements, while the error of the flux deconvolution from the measurements is not accounted for.

The highest total flux values are obtained in the central and triangular irradiation channels. While the total flux is approximately 50% lower in the triangular than in the central channel, the thermal and fast parts of the flux in the triangular channel are approximately 80% and 45% of the corresponding quantities in the central channel. The fluxes in other in-core irradiation sites (pneumatic post and irradiation channel F19) achieve values of approximately 35% of the central channel flux. The rotary groove irradiation sites have significantly lower fluxes, which are only of the order of 10% of the flux in the centre of the core. In general there are no large variations in the flux distribution between different cores; particularly in the rotary groove the flux values are the same. However, local variations can be of the order of 10% for the total flux, depending on the loading pattern in the vicinity of the irradiation channel.

Spectral analyses of the fluxes in different irradiation channels of the core is also very interesting, since it reveals that the hardest spectrum is achieved in the centre of the core (almost 40% of the neutrons are in the fast part of the spectrum, i.e. with the energies above 100 keV). On the other hand, the most thermal spectrum is in the irradiation channels positioned at the outside part of the core and in the rotary groove, where more than 60% of the neutrons are in the thermal region (energies below 0.5eV). The ratio between the fast and thermal neutron flux decreases significantly from 1.5 in the centre of the core to less than 0.2 in the rotary groove. It is interesting to observe that the rearrangement of the core can change the fast-to-thermal neutron flux ratio for up to 40% (compare the pneumatic post ratios). The main reason for this change is the number of the fuel elements that occupy the neighbouring positions to the irradiation site. On the other hand, the rearrangement has only a minor effect on the flux and as its spectral composition in the centre of the core and outside the core. Together with the local flux variations, the strong local spectral dependencies on the core configuration indicate possibilities for core loading optimisation for different irradiation channels and different irradiation goals.

The comparison to the measurements shows that general agreement between the measurements and simulations is rather good and is more or less within the estimated uncertainties. The difference in the total flux is probably mostly due to the error in the thermal power calibration. The small difference in the fast-to-thermal neutron ratio, which is less 20%, can be probably ascribed to the inaccuracy of the deconvolution procedure employed for flux determination from the activation measurements, which can be particularly significant in the fast part of the spectrum.

#### **4 CONCLUSIONS**

It was found out that the flux as well as its spectral characteristics depend very much on the position of the irradiation channel in the TRIGA Mark II reactor core and can vary significantly for different core configurations. The highest flux is achieved in the central channel and then decreases to approximately 50% in the triangular channel, 35% in the pneumatic post and irradiation channel F19 (positioned at the edge of the core) and 10% in the rotary groove. The spectral characteristics of the flux also vary significantly with the position in the core. While there are about 40% more fast neutrons ( $E > 100\text{keV}$ ) than thermal neutrons ( $E < 0.5\text{eV}$ ) in the centre of the core, the ratio between the fast and thermal neutrons decreases to less than 0.2 in the rotary groove. It should be emphasised that the core

configuration can introduce up to 40% change in the fast-to-thermal neutron ratio, which enables optimisation of the core configuration for different irradiation channels and different irradiation goals. Comparison to the measurements showed that our Monte Carlo model can successfully predict absolute values of the flux as well as its spectral characteristics to within a few percent. A small difference observed in the fast-to-thermal neutron ratio is probably due to the inaccuracy of the deconvolution procedure employed for flux determination from the activation measurements.

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