



Calculations to Support Absolute Thermal Power Calibration of the Slovenian TRIGA Mark II Reactor

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ABSTRACT

Thermal power calibration of the Slovenian TRIGA research reactor is based on the calorimetric method. In reality, neutron detector signals used for reactor power reading are not necessarily proportional to the integral neutron flux of the reactor core and the reactor power as they are sensitive to changes in the reactor core. Furthermore, calorimetric thermal calibration depends on heat transfer effects, natural convection of water, heat loss and similar effects. In order to reduce the uncertainties in absolute power calibration a project was initiated. In the first part Monte Carlo calculations are performed to understand and predict detector response versus control rod positions. In the second part of the paper, preliminary calculations to support thermal power calibration based on in-core fission chamber measurements are presented. The major source of uncertainty is the mass of uranium in the fission chamber.

1 INTRODUCTION

Accurate reactor thermal power calibration is important for: safe monitoring and evaluation of reactor operation, as the reactor power signal is used in the reactor automatic control system, evaluation of reactor experiments, accurate determination and calculation of fuel element burn-up and isotopic composition of burned fuel, normalisation of calculated neutron fluxes and dose rates.

Presently, thermal power of the TRIGA (‘Training Research Isotopes General Atomics’) Mark II reactor at Jožef Stefan Institute (JSI) is monitored by neutron detectors, absolutely calibrated by the calorimetric method [1]. Five independent detectors located on the outer side of the graphite reflector surrounding the core measure neutron flux and gamma-rays which are proportional to the reactor power.

The relationship between the signal from neutron detectors and the thermal power of the reactor is not simple as it depends on many parameters, most importantly neutron flux distribution, which changes during the reactor operation. Hence the thermal power calibration has to be performed regularly, especially after every change in the core. The uncertainty in absolute power due to uncertainties in calorimetric method can be up to 30 % and the error due to position of the control rods can be up to 15 % [1].

The calibration can be improved by using absolutely calibrated fission chambers (FCs) to measure absolute fission rates at various locations in the reactor core. As in all such

calibrations, neutronic calculations are required to support the physics, safety and engineering efforts. Many are based on Monte Carlo calculations using the advanced Monte Carlo transport codes, such as MCNP [2]. Here a detailed model of TRIGA Mark II reactor at Jožef Stefan Institute [3] was developed and used in order to calculate calibration correction factors and evaluate experimental uncertainties.

In the paper the main activities to improve and support the absolute power calibration of the TRIGA Mark II reactor at JSI [3] are described and preliminary results are presented.

2 DETECTOR SENSITIVITY TO CONTROL ROD POSITIONS

TRIGA Mark II at the JSI is a pool type nuclear research reactor. The General Atomics zirconium-hydride and uranium fuel mixture has unique properties that make the reactor inherently safe and suitable for training, research and isotope production, as suggested by the name. The fuel elements in the reactor core are cooled by natural water convection. A cooling system allows primary pool water to transfer heat to the secondary system.

Reactor power is monitored with five independent neutron sensitive detectors. They are all located on the outer side of a graphite reflector surrounding the reactor core. The detector signals make up the so called starting, linear, logarithmic, safety and pulse channels. The linear, logarithmic and safety channels are used for neutron power monitoring or neutron flux measurements during continuous reactor operation, i.e. up to 300 kW. These detectors are positioned inside separate instrumentation chambers (aluminium tubes) measuring 67.3 cm in height with outer diameter 11.4 cm. The detectors are located at the bottom of the instrumentation chambers, i.e. approximately from -7.95 cm to +25.7 cm around the core mid plane. The current reactor core configuration with detector locations is depicted in Figure 1.

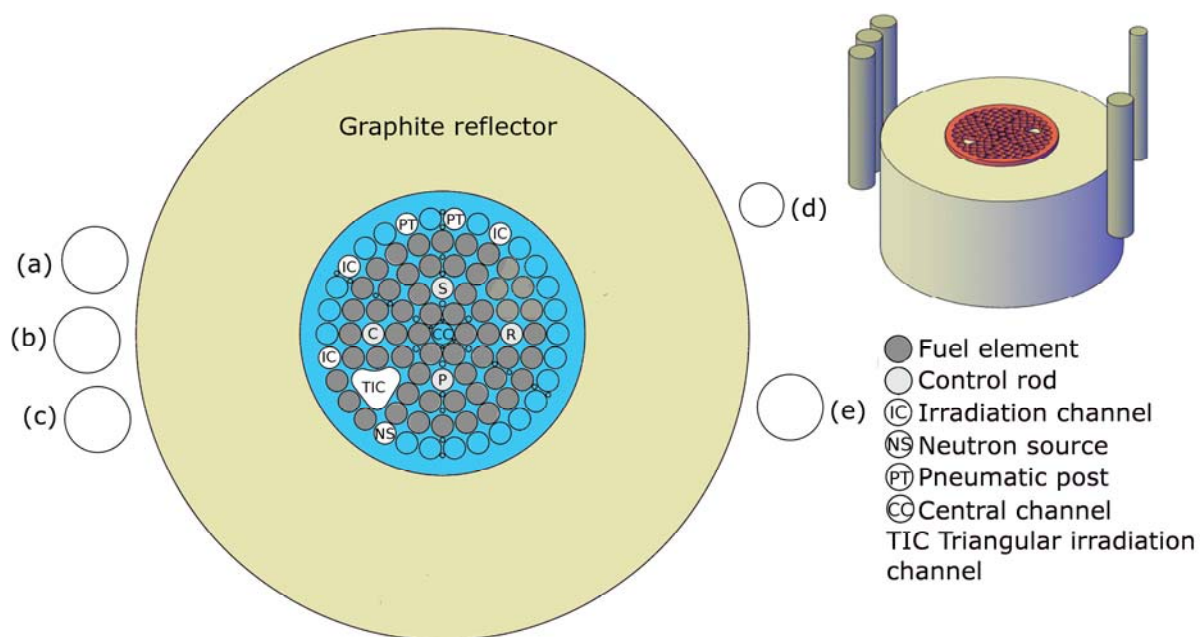


Figure 1: Reactor core configuration with neutron detector locations – safety (a), pulse (b), logarithmic (c), starting (d) and linear channel (e). Control rods are denoted as safety (S), transient (P), compensating (C) and regulating (R).

The detector signal is not always proportional to the average neutron flux of the core and consequently the thermal power. In fact, detectors are highly sensitive to changes in core configuration, especially control rod positions. The changes in detector response due to different control rod positions can be as large as 15 %.

Changes in reactivity during operation for TRIGA type reactors are compensated primarily by changing control rod positions. There are four control rods – safety (S), transient (P), compensating (C) and regulating (R) (Figure 1) in the reactor. Reactivity during continuous operation is usually regulated with compensating and regulating control rod positions, while the safety and transient control rods are completely withdrawn from the reactor core. It is important to note that each control rod is regulated independently from the others; hence the neutron flux distribution can be tilted either in radial or in axial direction by withdrawing or inserting one control rod only.

In order to investigate the effect of control rods on the neutron flux distribution and consequently on the detector response, we performed a series of Monte Carlo calculations. In the calculations the control rod position was varied and the spatial distribution of the neutron flux, Φ , and detector response were investigated. The presentation and the discussion of the results follow below.

In Figure 2 (left) a calculation of radial total neutron flux distribution ($\Phi_{MC,C}$) for fully inserted compensating control rod is presented. The neutron flux is normalized per one fission neutron and is averaged over the height of the core ($z = -19.05$ to 19.05 cm). A large depression in the neutron flux distribution can be observed (Figure 2 (left)) at the position of the compensating rod, as the control rod is made of B_4C , which is a strong thermal neutron absorber.

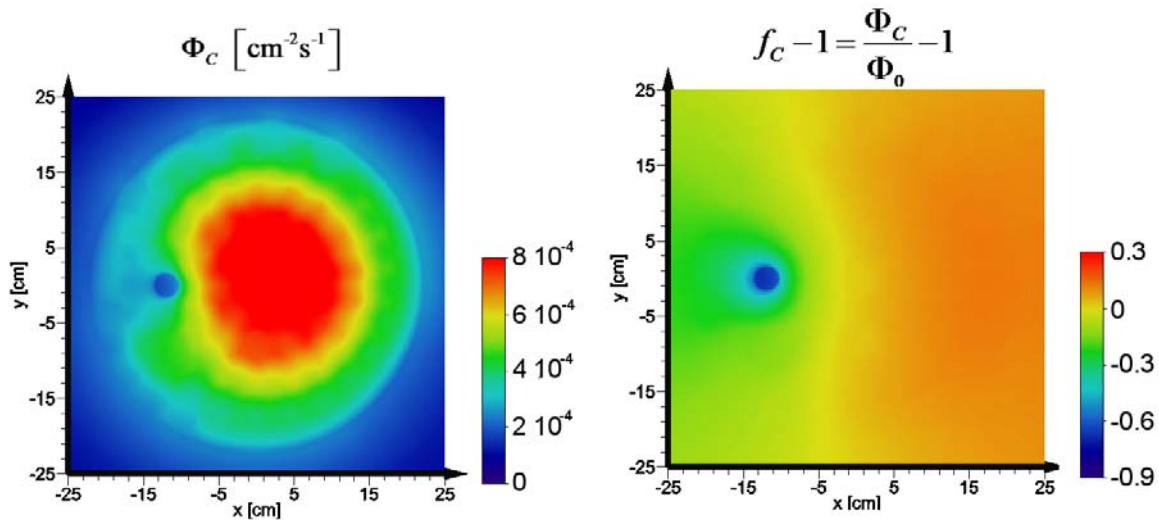


Figure 2: Monte Carlo calculation of neutron flux distribution for fully inserted compensating control rod (left). Neutron flux is normalized per one fission neutron. Relative deviation of total neutron flux distribution for fully inserted compensating rod from unperturbed flux (right).

The effect of control rod position on neutron flux distribution is often described by the so called flux depression factor defined as:

$$f_\gamma(x) = \frac{\Phi_\gamma(x)}{\Phi_0(x)}. \quad (1)$$

It represents the ratio of neutron fluxes measured at a specific location x for fully inserted γ control rod $\Phi_\gamma(x)$ and unperturbed flux $\Phi_0(x)$, where all control rods are withdrawn. Figure 2 (right) shows the flux depression factor for the compensating control rod ($f_C - 1$). Evidently the greatest deviation in neutron flux is inside the control rod itself, seen in Figure 2

as a blue circle. Moreover, the control rod affects neutron flux distribution from the control rod outward. This means that neutron detectors, even though located on the outer side of the graphite reflector, are to some extent sensitive to changes in reactor core composition.

The light green on the left side of the colour field corresponds roughly to 10 – 20 % decrease in neutron flux. Orange coloured regions, on the other hand, experience a slightly increased flux (up to 10 %). The safety (a) and the logarithmic (c) channels are located on the left side of the core, as is the compensating control rod (C). It would seem that they would detect a decrease in neutron flux for inserted compensating rod (C). The linear channel (e) is located on the opposite side of the core and therefore measures a neutron flux that is slightly increased.

A correction of the measured signal S_m is required in order to obtain a value S that can be assumed to be proportional to the integral neutron flux in the core and consequently the reactor power. The correction due to the presence of a control rod can be adequately approximated by [1,4]

$$S(x) = \frac{S_m(x)}{\prod_{\gamma} \left[1 + (f_{\gamma}(x) - 1) g_{\gamma}(l) \right]}, \quad (2)$$

where function $g_{\gamma}(l)$ is an interpolation function that is assumed to have the same shape as the γ -th control rod worth curve. This assumption is supported by measurements [1,4] and calculations (Figure 3). For fully withdrawn rod, $g_{\gamma} = 0$, whereas for fully inserted control rod, $g_{\gamma} = 1$.

The easiest and often the only way to obtain the neutron flux depression factors is by calculations. A geometrically detailed, experimentally verified and validated [5] three dimensional computational model in Monte Carlo neutron transport code, MCNP [2] was used to calculate the neutron flux depression factors for the linear, logarithmic and safety channel for the regulating and compensating control rod. The results are presented in Table 1. These factors will be used in future to correct the signal on the detectors and make the absolute power reading more accurate.

Table 1: Calculated flux depression factors for compensating and regulating control rod in linear, logarithmic and safety channel.

Nuclear channel (x)	f_C	f_R
Linear	1.141	0.851
Logarithmic	0.863	1.154
Safety	0.854	1.149

In addition we verified the analytical function $S(x)$ by comparing it to calculated detector response versus control rod position curve (Figure 3). Position 200 corresponds to completely withdrawn and position 900 to a fully inserted control rod. It can be seen that the correction factors can vary between 1.15 and 0.85, corresponding to a 15 % relative difference of S and S_m . The symbols represent calculated values for flux at detector locations. The solid lines represent the estimation of the measured flux values S_m , using the extreme values Φ_0 and Φ_{γ} that were calculated with a Monte Carlo simulation. We may assume that the Eq. (2) adequately describes the measured signal dependency on control rod positions. By introducing a correction factor, the measured signal can be appropriately weighed in order to achieve proportionality to the integral flux in the core, and consequently to the reactor power.

As mentioned before, the safety (a) and the logarithmic (c) channels are located on the left side of the core, as is the compensating control rod (C). It can be seen from Figure 3:

Monte Carlo neutron fluxes at detector locations for different compensating and regulating rod positions. Error bars represent the standard one sigma statistic deviation of the Monte Carlo flux calculations. The corrections for fully inserted and withdrawn control rod can be as large as 15 %. (left) that the safety and logarithmic channels detect a decreased signal. The linear channel (e) is located on the opposite side of the core and therefore measures a neutron flux that is slightly increased.

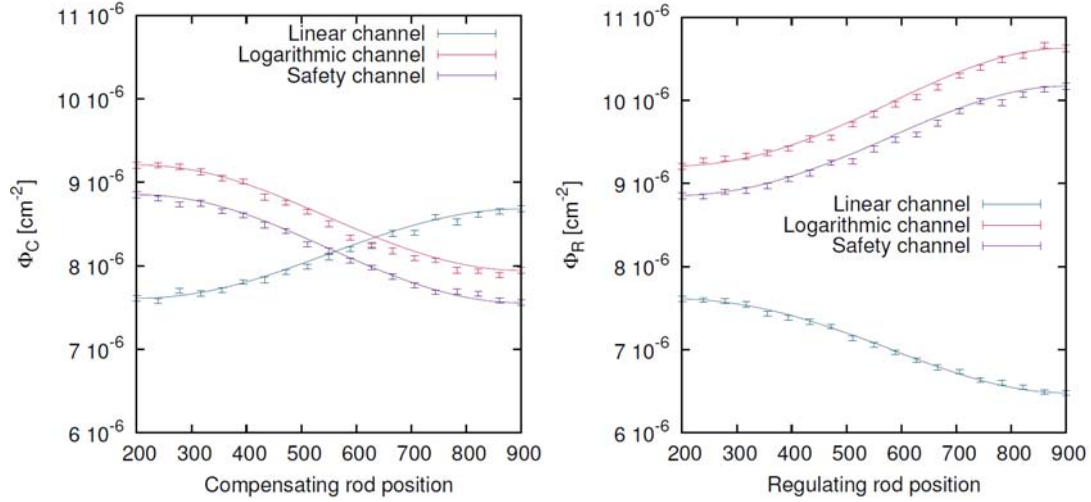


Figure 3: Monte Carlo neutron fluxes at detector locations for different compensating and regulating rod positions. Error bars represent the standard one sigma statistic deviation of the Monte Carlo flux calculations. The corrections for fully inserted and withdrawn control rod can be as large as 15 %.

3 THERMAL POWER CALIBRATION BASED ON IN-CORE FISSION CHAMBER MEASUREMENTS

In addition to the usual calorimetric method and application correction factors to the detector response, the absolute thermal calibration will be verified by inserting absolutely calibrated fission chambers (FCs) at various locations in the reactor core and comparing the measured signal with the calculated one.

Neutron flux distribution calculated with the Monte Carlo method is assumed to be proportional to the actual flux distribution ($\Phi = C\Phi_{MC}$). The scaling factor C for thermal reactors is proportional to the reactor power P and can be approximated by [6]

$$C = \frac{P\bar{\nu}}{w_f k_{eff}}, \quad (3)$$

where $\bar{\nu}$ is the average number of neutrons released per fission and w_f denotes the effective energy released per fission event, k_{eff} is the calculated multiplication factor of the Monte Carlo reactor model and P is reactor thermal power. Monte Carlo codes also allow reaction rate (R) calculations, by multiplying the neutron flux with microscopic cross sections.

Fission chambers can be made very small and suitable for in-core measurements. Measuring fission rate in several locations inside the reactor core and at the same time calculating the fission rate with a Monte Carlo simulation opens up the possibility to experimentally determine the scaling factor C , and together with Eq. (3) the thermal reactor power. This is the basic idea behind the thermal power calibration using an absolutely

calibrated fission chamber. The accuracy of the absolutely calibrated fission chamber is mainly limited by the accuracy in the mass and isotopic composition of the fissile material.

This type of thermal power calibration is scheduled to be performed at the JSI TRIGA reactor in October 2011. Mid-core reaction rate measurements will be performed in measuring positions 13 – 17, 8, 26 and 20 – 24 (Figure 4). Axial reaction rate distributions will be measured in some of these measuring positions. An aluminium tube with 8 mm outer diameter and 1 or 2 mm wall thickness will serve as part of a guiding mechanism for the fission chamber with 3 mm outer diameter.

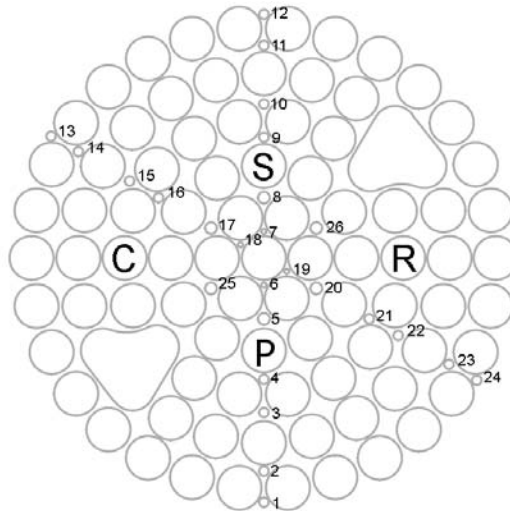


Figure 4: Measuring positions labeled from 1 to 26 for in-core measurements and control rod positions.

Two fission chambers will be used in the experiment – one containing 30 μg of uranium with ^{235}U as the prevailing isotope and the other containing 100 μg of predominately ^{238}U . The accuracy of both fission chambers reaction rate measurement is expected to be better than 3 %, which includes uranium mass uncertainty of approximately 2 %.

We already computationally simulated the experiment by calculating the axial fission rates at several radial measurement positions. In order to estimate the inaccuracy in FC response due to uncertainty in FC axial position we calculate the axial reaction rate gradient as well.

Figure 5 shows calculated absolute reaction rates at various positions along the reactor vertical axis and axial gradients for 1 μg of ^{235}U and ^{238}U . Normalisation by 1 μg of U allows easier correction in case the real mass deviates from the nominal one. Absolute reaction rates were calculated using the scaling factor C from Eq. (3) for 250 kW and $w_f = 193.5$ MeV [7].

Reaction rates of the fission chamber containing ^{238}U are approximately three orders of magnitude smaller than the reaction rates corresponding to the same amount of ^{235}U for the TRIGA-specific neutron spectra. This means that even the smallest fraction of ^{235}U isotope in the uranium located inside the fission chamber will contribute greatly to the measured signal. The axial gradients are on the order of 1 – 2 % per mm. As our FC positioning system will have a positing accuracy of 0.2 mm and reproducibility of 0.4 mm, the uncertainty in the FC response due to uncertainty in axial position will be negligible compared to the uncertainty in the FC absolute calibration.

Preliminary calculations show that the gradients of neutron flux in radial direction are significant. Hence we can expect that even small uncertainty in radial position of the fission chamber will significantly affect the uncertainty in the predicted response. Therefore we estimated the reaction rate uncertainty due to radial position inaccuracy by calculating the radial gradients of the ^{238}U and ^{235}U fission rates. Figure 6 shows the mid-core radial gradient

of fission rate calculations for ^{235}U and ^{238}U . The small white circles in the figure denote the measuring positions selected for mid-core FC measurements. Dark blue colour dominates inside most of the white circles, which corresponds to 1 % uncertainty in reaction rate for 1 mm uncertainty in radial position. It can be assumed that for a fission chamber measuring 3 mm in diameter and for aluminium guiding tubes with inner diameter between 4 and 6 mm the fission chamber reaction rate measurement uncertainty is within 1 %.

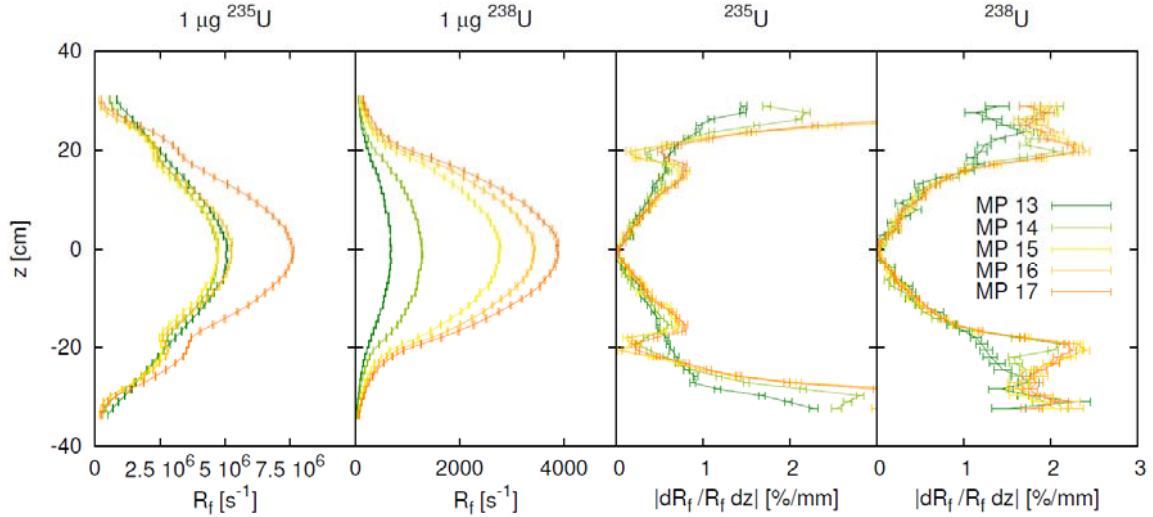


Figure 5: Axial flux and reaction rate distributions and gradients in measuring positions from 13, located at the margin of the reactor core, to 17 that is located close to the centre. Error bars represent the standard statistical one sigma deviation.

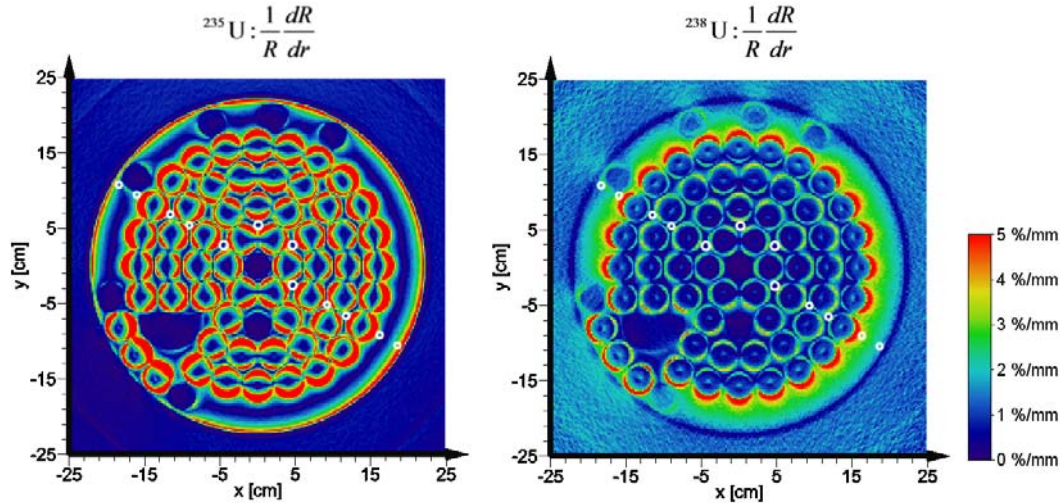


Figure 6: Monte Carlo calculations of mid-core reaction rate gradients for ^{235}U and ^{238}U . The white circles mark the selected measuring positions for mid-core fission chamber measurements.

Another important thing to note is the influence of the aluminium guiding tubes on reaction rate measurements. Due to relatively low absorption cross section of Al the presence of a 1 – 2 mm aluminium tube inside the reactor core has almost no effect on the total flux inside the tube. However, it does affect the neutron spectrum as it displaces water causing a decrease in the thermal neutron flux and an increase of fast neutron flux. This is confirmed by calculations of ^{235}U and ^{238}U fission rates with and without the Al tube.

Relative differences in fission reaction rate due to presence of Al tubes (R_{Al}) and in their absence (R_0) are presented in Table 2. Changes in ^{235}U and ^{238}U reaction rates can be as large as 8 %. It is very important to accurately model the aluminium guiding tube in the Monte Carlo model of the reactor, so that the reaction rate calculations can be assumed to be exact.

Table 2: Relative differences in Monte Carlo reaction rate calculations for the presence of aluminum tubes (R_{Al}) and in their absence (R_0). Reaction rates are averaged for all selected measuring positions.

	1 mm Al wall	2 mm Al wall
$^{235}\text{U } (R_{Al}-R_0)/R_0$	$-4.5 \% \pm 0.5 \%$	$-8.2 \% \pm 0.5 \%$
$^{238}\text{U } (R_{Al}-R_0)/R_0$	$+1.3 \% \pm 0.5 \%$	$+2.7 \% \pm 0.5 \%$

The estimation of the covariance of separate measurements represents the limit for the scaling factor calculation accuracy, and consequently the accuracy of the thermal power calibration. The relative covariance of the calculated scaling factor can be estimated by composing uranium mass uncertainty ($\sim 2 \%$), uncertainty of absolute fission chamber calibration and possible systematic errors of the Monte Carlo reactor model. Reducing thermal power calibration uncertainty to the target 3 % seems achievable, provided that the Monte Carlo calculations have an insignificant uncertainty.

4 CONCLUSIONS

It was observed that the effect of control rod position on neutron detectors' response can be as high as 15 % for fully inserted or fully withdrawn control rods. Despite this strong dependence of nuclear channel signals on control rod positions, accurate power reading is possible by using suitable correction factors described in Eq. (2).

More interestingly, we find that thermal power calibration with an absolutely calibrated fission chamber proves to be very viable. However, knowing the accurate mass of uranium contained inside the fission chamber and accurate calibration of the fission chamber is crucial in correct reactor thermal power calibration. At the same time, it is very important to verify that the Monte Carlo reactor model is sufficiently accurate.

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