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Burnup Measurements Using Fuel Reactivity Worth Experiments at the JSI TRIGA Research Reactor

Anže Pungerčič[1,2], Alireza Haghighat[3], Luka Snoj[1,2]

[1]Reactor Physics Department, "Jozef Stefan" Institute
Jamova cesta 39, 1000, Ljubljana, Slovenia

[2]Faculty of mathematics and physics, University of Ljubljana
Jadranska ulica 19, 1000, Ljubljana, Slovenia

[3]Nuclear Engineering Program, Virginia Tech
Northern Virginia Center, 7054 Haycock Road, Falls Church, 22043, Virginia USA

anze.pungercic@ijs.si, haghighat@vt.edu, luka.snoj@ijs.si

ABSTRACT

The effect of fuel reactivity of different burned elements in the research reactor JSI TRIGA was measured by two methods. The traditional fuel reactivity worth method was compared with the new fuel reactivity swap method. The changes in core reactivity due to differently burned fuel elements were up to 120 pcm, and a clear relation between the changes in reactivity and fuel burnup was found. The measured data were used to validate the fuel burnup and core criticality calculations. Both experiments were simulated using different computer codes: the deterministic TRIGLAV, the Monte Carlo Serpent-2, and the hybrid RAPID. Great agreement was observed for Serpent-2 and RAPID, indicating that the fuel burnup and criticality calculations are accurate and that reactivity changes due to small burnup differences on the order of 10 pcm can be accurately predicted. In addition, it was shown that due to small differences in burnup, changes in detector response due to fuel-shuffling two fuel elements cannot be detected when using ex-core detectors and large fission chamber in the core periphery.

1 INTRODUCTION

Knowing the physical parameters of the reactor core at all times is essential for safe and reliable operation of fission nuclear reactors. Some physical parameters, such as power distribution in the fuel element, ^{235}U and fission product content, etc., which change with reactor operation, cannot be measured due to operational limits and conditions and physical constraints. Therefore, one must rely on reactor calculations to obtain these parameters. Reactor dynamics are very complex. One of them is the long-term changes in the isotopic composition of the fuel caused by fuel depletion (burnup). Because of the changes in the isotopic composition of the fuel, the Boltzmann neutron transport equation must be coupled with Bateman's depletion equations (Chapter 10 in [1]). The traditional approach has been to use deterministic codes to calculate burnup at the unit cell level and use the calculated parameters to obtain fast solutions for the entire core using diffusion approximations of the transport equation (Chapter 2 in [1]). Such methods do not work very well for smaller heterogeneous reactor cores (e.g., the TRIGA research reactor). In such cases, modern Monte Carlo neutron transport [2] can be used to calculate the fuel burnup, but this can be extremely computationally intensive. Recently, a new method for calculating burnup using hybrid methods has been in development [3], which employs using the Multi-stage Response-function particle Transport (MRT) methodology [4] and the code system RAPID. This method takes advantage of both the accuracy of the Monte Carlo calculation and the speed of the deterministic approximation.

An important step in the process of developing new reactor simulation methods is their validation. In the case of fuel burnup determination, this is usually done by code-to-code comparison, usually using Monte Carlo code results as a reference. However, it has been shown that eigenvalue Monte Carlo particle transport can lead to unrealistic solutions (Chapter 10 in [5]). Therefore, experimental benchmarks that can be used to validate reactor simulation codes are of great importance. In the area of reactor burnup, there are few well-documented burnup benchmark experiments, and there is a need for new ones. Based on the well-documented and analysed operating history [6], we decided to perform burnup measurements at the research reactor JSI TRIGA Mark II. The most accurate and accepted non-destructive method for spent fuel analysis is fuel gamma spectrometry [7], which can be used to determine the composition of individual isotopes, but it requires a complex experimental setup and is also not suitable for reactors with a limited number of fuel elements, since long fuel cooling times are required to perform the experiments safely. It has also been shown that for certain fuel types, gamma transparency [8] or acoustic wave measurements [9] can be used to estimate fuel burnup. However, for TRIGA research reactors, the most practical and accessible method for determining fuel burnup is to perform the measurements of individual fuel element reactivity worth [10]. Due to the limited number of fuel elements that can be removed from the core of JSI TRIGA and still be able to achieve criticality, a new version of the method, called the fuel swap method, is proposed. Performing the two versions of the fuel reactivity measurements, their analysis and cross-comparison is the main purpose of this paper.

The first part of this paper describes the experimental campaign carried out at the research reactor JSI TRIGA Mark II from April 6 to 8, 2022, and analyses the results of the fuel element reactivity worth [10] and the new fuel swap method. The second part of the paper focuses on the reproduction of the experiments with deterministic TRIGLAV [11], Monte Carlo Serpent [2] and hybrid RAPID [12] neutron transport codes.

2 FUEL REACTIVITY WORTH EXPERIMENTS

The experimental campaign was conducted on April 6, 7, and 8, 2022, at the research reactor JSI TRIGA Mark II. The main objectives of the campaign were to measure the fuel burnup of at least 5 fuel elements located in one octant of the core using the fuel reactivity worth method and the fuel swap method. We succeeded in performing both experiments for 7 fuel elements. We also measured the behaviour of fission chambers and compensated ion chambers at different positions to quantify the redistribution of neutron flux that could result from fuel reshuffling.

2.1 JSI TRIGA Mark II Research Reactor

The JSI TRIGA research reactor is a light water reactor (LWR) with annular graphite reflector cooled by natural convection. It reached first criticality on May 31, 1966, and since then has used 300 different fuel elements in more than 240 core configurations. In 1999, all fuel elements, except stainless steel with 12 wt.% of 19.9% enriched uranium in the U-Zr-H mixture were returned to the United States, reducing the amount of available fuel. With the remaining fuel elements, fresh fuel criticality benchmark was performed in 1991 [13], and they have been in operation since then, accumulating average fuel burnup of 20 MWd/kg (HM)[1]. The core lattice has an annular periodic structure. The elements are arranged in six concentric rings, named: A, B, C, D, E, and F, with 1, 6, 12, 18, 24, and 30 locations, respectively, filled with either fuel elements or other components such as control rods, a neutron source, and irradiation channels. The core is surrounded by a 30.5 cm thick graphite reflector containing a so-called carousel with 40 irradiation positions. Fig. 1 shows a core schematic with the loading pattern of core No. 246, which was in operation at the beginning of the experimental campaign.

On the outside of the graphite reflector are five ex-core channels consisting of neutron and gamma detectors used to measure the reactor's thermal power. Safety, linear, logarithmic, and startup channels were used to monitor neutron flux redistribution during fuel reactivity experiments. In addition, an irradiation channel with a large ^{235}U fission chamber was added to the F4 position, right next to the fuel elements that would have been replaced during the experiments. All detectors in the measurement positions used in the experimental campaign are shown in blue in Fig. 1.

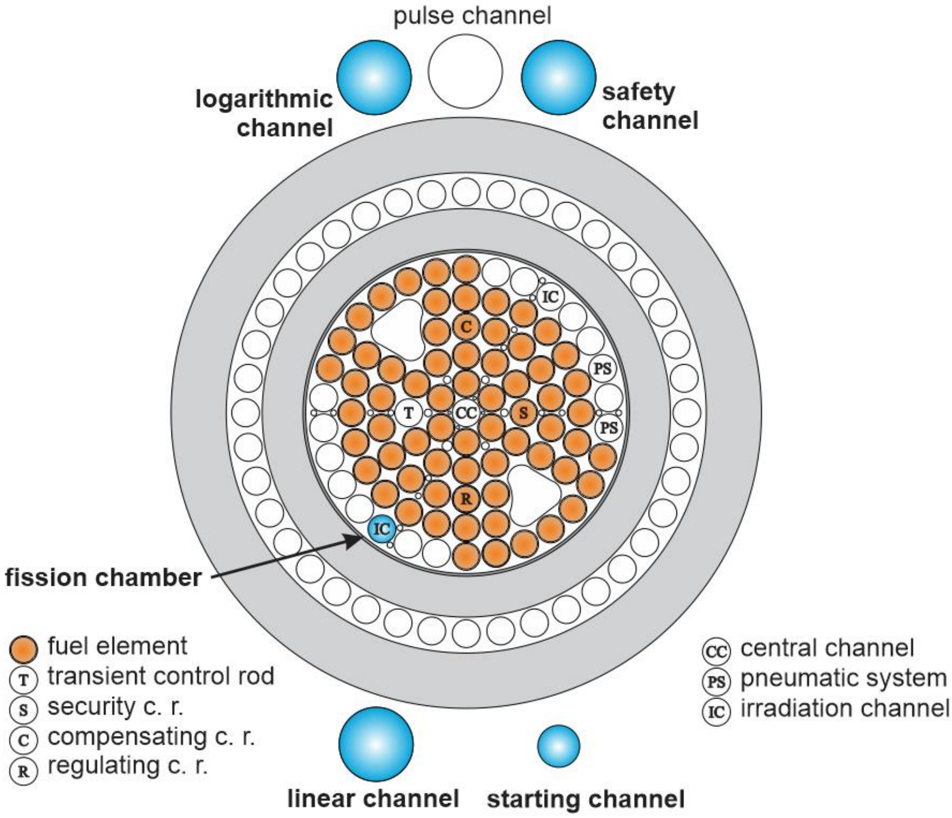

Figure 1: Schematic top view of the reactor upper support grid with denoted fuel elements, control rods, irradiation channels, and ex-core nuclear channels. Core configuration No. 246 (started operating on 23rd July 2021) is depicted. Blue colour depicts detectors, used to measure neutron flux redistribution during the experimental campaign.

In both experiments, reactivity was measured using the Digital Reactivity Meter (DMR) [14], which takes the neutron source measured by the safety channel and directly determines the changes in nuclear reactivity by solving the point kinetic equations. By performing relative reactivity measurements with the same core configurations and the same control rod positions,

[1] MWd/kg(HM) unit of fuel burnup represents energy released per initial mass of heavy materials A ≥ 92

the only uncertainty in the reactivity measurement is due to variations in the safety channel signal, resulting in an uncertainty of ± 3 pcm.

2.2 Fuel Reactivity Worth Experiment

The method for determining the reactivity of fuel, also known as the "Ravnik" method, is described in detail in [10]. In this paper, only the main steps of the method are discussed. Prior to the experiment, the fuel elements to be measured were removed from the core. An approximate critical core was made with all control rods pulled out. A total of 7 fuel elements were removed from the core and placed in a fuel rack on the side of the reactor pool. The fuel elements from ring F were moved to fill in the resulting gap, creating the new core configuration No. 247 shown in Fig. 2. This resulted in the reactor core being critical with all rods pulled out, except for the regulating rod at position 299, which resulted in a ρ_{excess} = 180 pcm. If the 8th fuel assembly were removed, the core could not reach criticality and the fuel reactivity measurements could not be made.

Location C2 with fuel element ID 7212 was selected as the location for all fuel elements to be inserted and reactivity worth measured. For each fuel element, core reactivity was measured at the reference control rod configuration (T - up, C - up, S - up, R - 299) to remove the control rod effect. After the measurement, the reactor was shut down and the measured fuel element was replaced by the next in sequence at the same measurement position C2. The reactor was made critical, the control rods were moved to the reference position, and the change in reactivity $\Delta\rho_{worth}$, with respect to fuel element 7212, was measured by the DMR. The difference between the reactivity values of the measured elements is proportional to the difference in their burnups.

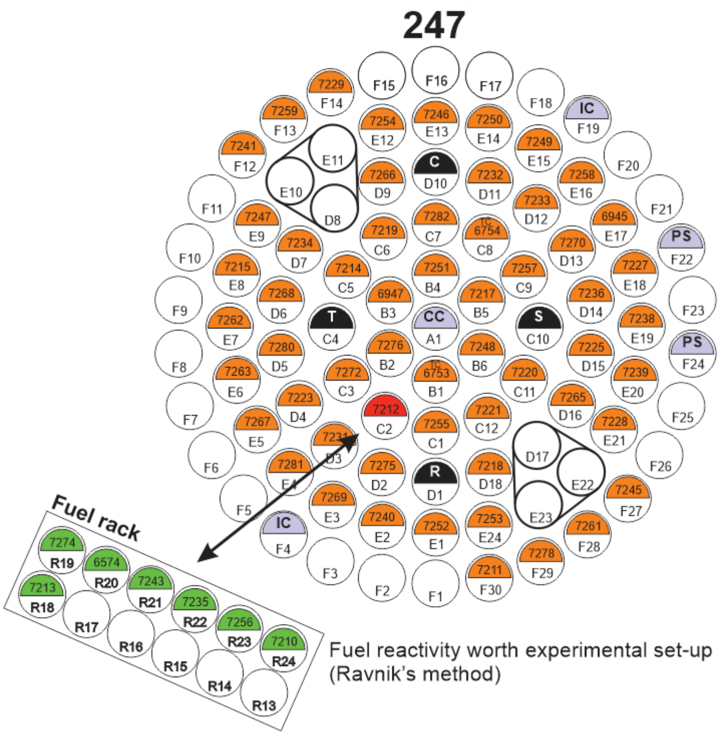

Figure 2: Schematic representation of the fuel reactivity worth experiments performed on core configuration No. 247. Compared to core No. 246, seven fuel elements were removed and placed in the fuel rack located at the edge of the reactor pool and then inserted individually into position C2. All fuel elements measured during the experiment are shown in green in relation to the reference fuel element shown in red.

2.3 Fuel Swap Experiment

Due to the limitations of the fuel reactivity worth experiment, where only 7 fuel elements could be measured, we decided to perform a new type of fuel reactivity measurement by swapping fuel elements in the core itself, which simplifies the experimental setup since the starting core configuration does not need to be changed. For this experiment, the original core configuration No. 246 was chosen. The experiment is performed similarly to the fuel reactivity method. First, the reactor is made critical and the reference control rod position (T - up, C - 536.6, S - up, R - 535) is determined. After the reactor is shut down, the position of the two fuel assemblies is swapped, the reactor is made critical and the control rod is set to the reference position, which removes the effects of the control rod redistribution. The change in reactivity $\Delta\rho_{swap}$ is measured with the DMR. In our case, 7 fuel assemblies were swapped with the fuel element ID 7212 at location C2 and the change in reactivity was measured. The locations of the swapped fuel elements are shown in Fig. 3. The measured difference in reactivity of the core is proportional to the difference in fuel burnup, which is related to the value of the positions the swap occurs (e.g., the fuel element in ring C is worth more reactivity-wise than the same fuel element in ring E).

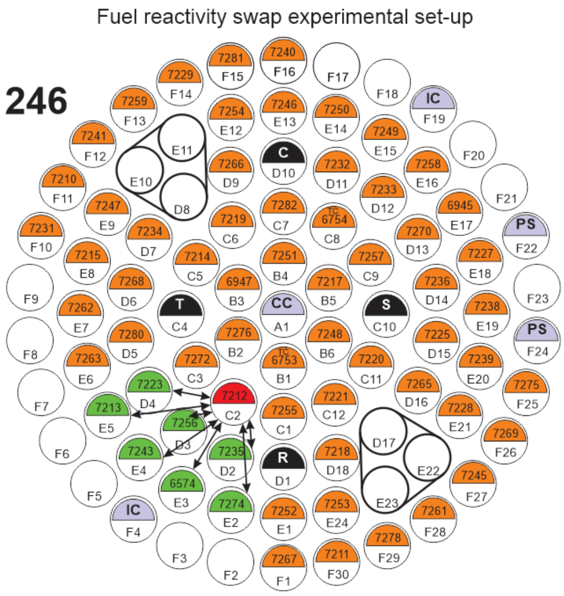

Figure 3: Schematic representation of the fuel reactivity swap experiment performed on core configuration No. 246. The fuel elements depicted in green were swapped with the fuel element in location C2 (shown in red) and the change in reactivity was measured.

When comparing the two methods, it should be noted that the fuel reactivity method limits the number of fuel elements that can be measured, but this is not a problem for reactors with a larger number of available fuel elements. The advantage of the fuel swap method is that no reshuffling is required at the beginning and it is an overall faster method. However, the disadvantage is that the measured difference in reactivity $\Delta\rho swap$ depends on the swap positions and the relation is non-trivial, while the $\Delta\rho worth$ is directly proportional to fuel burnup and a linear relation between the two can be assumed to obtain the burnup of all measured fuel elements. The relation between the $\Delta\rho swap$ and fuel burnup for all different swap positions will be investigated in the future.

3 COMPUTER SIMULATION OF THE EXPERIMENTS

Both fuel reactivity worth and swap experiments were simulated using different neutron transport codes: the deterministic TRIGLAV [11], the Monte Carlo code Serpent-2 [2], and the hybrid code RAPID [12]. All three codes differ in the methodology used to solve the neutron transport equation and thus in the time in which the solution is obtained.

The TRIGLAV code [11] is based on a four-group diffusion equation solved by the finite difference method. All 91 positions in the core are treated separately in the unit cell approximation, and the effective group constants averaged over the unit cell at different burnup are determined using the WIMSD-5B code [15]. The calculated group constants are used in the 2D diffusion approximation. From the diffusion solution, the core power distribution is obtained, and the fuel burnup can be calculated. One core criticality calculation takes about 2 minutes on a PC.

Serpent-2 [2] is a continuous-energy 3D Monte Carlo particle transport code in which the geometry can be defined in detail and which is capable of performing fixed-source and criticality calculations of nuclear system. In addition, it has an automated built-in burnup algorithm capable of calculating detailed 3D burnup, although due to source convergence issues, a higher number of particles must be simulated, increasing the calculation time. A criticality calculation takes 3 hours on the cluster processor[2], while a full burnup simulation of the operating history takes about 14 days.

The RAPID code system [12] is developed based on the MRT methodology [4] and is used for 3D real-time simulation of nuclear systems by pre-calculating response functions/coefficients for a given problem using detailed Monte Carlo calculations. These coefficients are compiled in a database to solve various problems. For the simulation of TRIGA core configuration, the pre-calculation of the coefficients takes ~ 9 hours on the JSI cluster processor[2] and the criticality calculation takes 20 seconds on PC. It should be noted that the coefficients for a given problem only need to be calculated once, making the methodology extremely fast overall.

Fuel burnup calculations were performed according to the method described in [6], which considered the entire operating history of the reactor JSI TRIGA with more than 240 simulated core configurations. The burnup calculations were performed using the TRIGLAV code and the Serpent-2 code. The calculations performed with the RAPID code system included the isotopic composition calculated by Serpent. To date, the RAPID code system with its bRAPID methodology is capable of accurately calculating the burnup of a single core configuration, and the capability to calculate the entire operating history will be developed in the future. The ENDF/B- VII.1 nuclear data library [16], which was also used for the referenced burnup calculations in [6], was chosen for the burnup and criticality calculations.

[2] 2 x Intel Xeon Gold 6240R Processor with 24 physical cores and base frequency of 2.4 GHz

4 COMPARISON OF CALCULATED AND MEASURED REACTIVITY

4.1 Fuel Reactivity Worth Experiment

The Fuel reactivity worth of seven JSI TRIGA fuel elements was measured. The pre-calculated burnup of these fuel elements was in the range of 1 - 16 MWd/kg (HM), therefore different changes in reactivity were expected. The measured $\Delta\rho_{worth}$ reactivity change was simulated using the TRIGLAV, Serpent-2, and RAPID neutron transport codes. For each case, the core multiplication factor was calculated and compared with the reference value of the excess reactivity of core configuration No. 247. The results are shown in Fig. 4. As expected, the $\Delta\rho_{worth}$ reactivity change was highest for the least burned fuel elements and vice versa. Good agreement is found between RAPID, Serpent-2 and the measurements, as all calculations are within 1σ of the uncertainties. A less good agreement is observed for the TRIGLAV code system, especially for the fuel elements that have a lower burnup and consequently a higher $\Delta\rho$ value. The discrepancy is a results from the combination of discrepancies in the burnup calculations and the core criticality calculation. The results show that the method of calculating TRIGA burnup with full operating history analysis [6] is accurate, as the $\Delta\rho_{worth}$ for fuel elements with different burnup and positions agrees with the measurements.

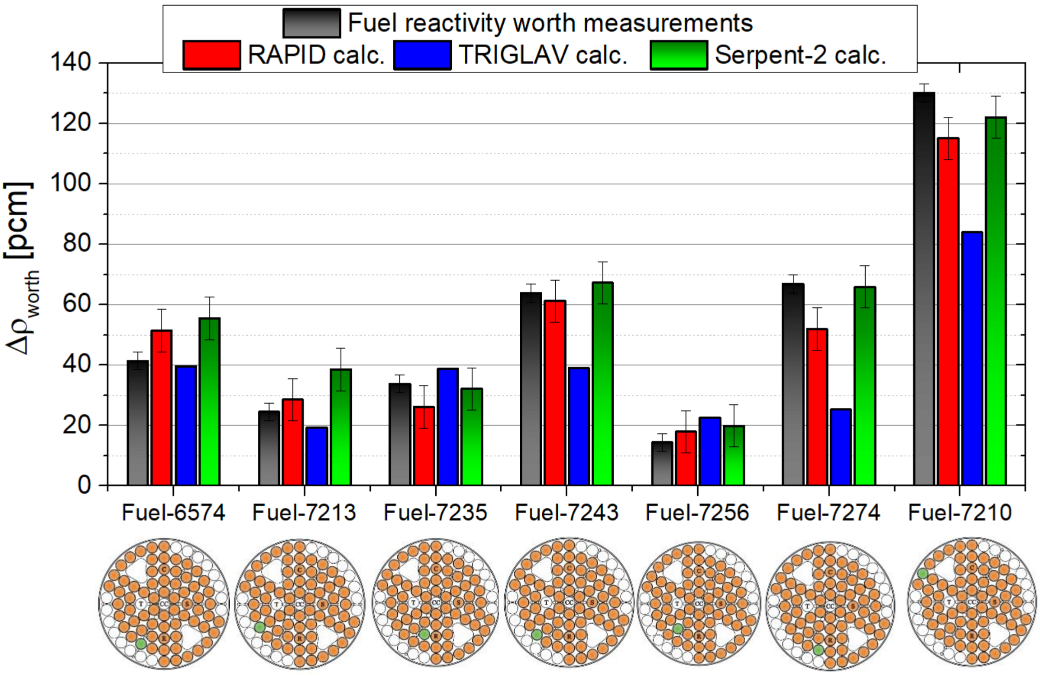

Figure 4: Comparison between calculated and measured $\Delta\rho_{worth}$, using the fuel reactivity worth method [10] for 7 JSI TRIGA fuel elements. Their initial position in the core is indicated on x-axis.

4.2 Fuel Swap Experiment

Changes of core reactivity in the case of fuel swap experiment of seven JSI TRIGA fuel elements was measured. The pre-calculated burnup of these fuel assemblies was in the range of 9 - 16 MWd/kg(HM). Due to operating limits, one fuel element measured in the fuel worth experiment had to be replaced. The measured change in reactivity $\Delta\rho_{swap}$ was simulated using the TRIGLAV, Serpent-2, and RAPID neutron transport codes. For each case, the core multiplication factor was calculated and compared with the reference value of the excess reactivity of core configuration No. 246. The results are shown in Fig. 5. Similar to the fuel worth method, the highest reactivity change $\Delta\rho_{swap}$ was expected for the least spent fuel element and vice versa, but since the swap occurred in the core, a dependence on swap position and lower reactivity changes were expected. The relationship between $\Delta\rho_{swap}$ and fuel burnup for all different swap positions will be investigated in the future. Nevertheless, good agreement is observed between RAPID, Serpent-2 and the measurements, indicating that the calculated burnup and core criticality calculations are accurate. A similar observation to the simulation of the fuel worth experiment can be made for the TRIGLAV code system, as a larger discrepancy is observed for the fuel assemblies that have a lower burnup and consequently a higher $\Delta\rho$ value.

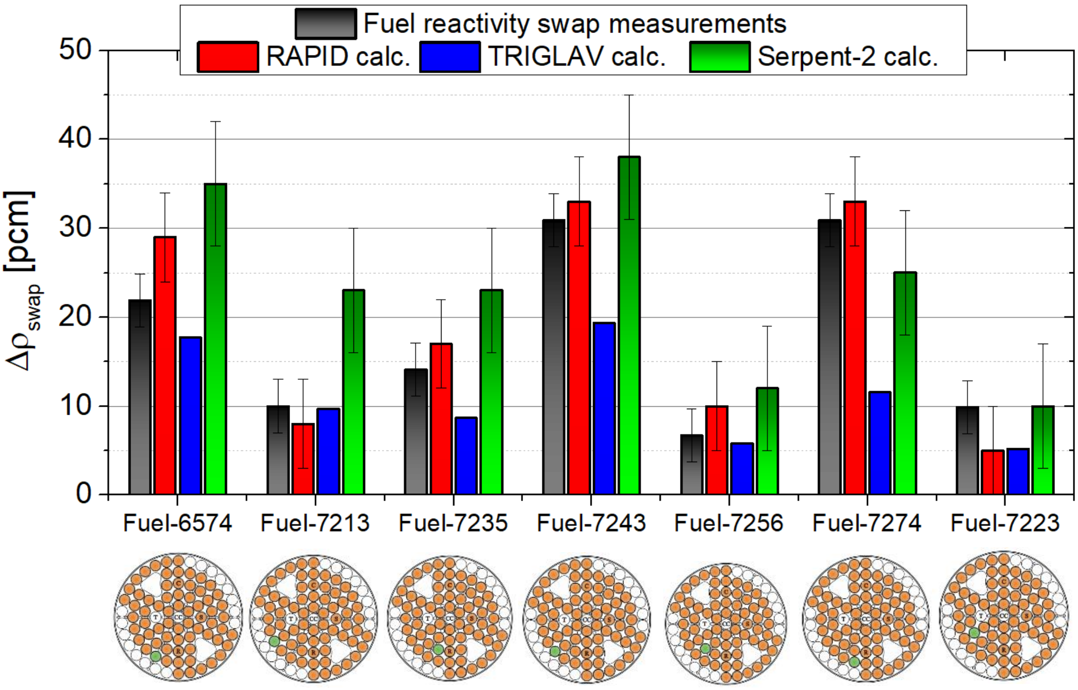

Figure 5: Comparison between calculated and measured $\Delta\rho_{swap}$, using the new fuel reactivity swap method for 7 JSI TRIGA fuel elements. Their initial position in the core is indicated on x-axis.

4.3 Detector response changes

In addition to the measured reactivity changes, the detector response was measured during both experiments in different positions and with different detectors (see Fig. 1). Table 1 shows the detector response changes for the fuel swap experiment, which is shown schematically in Fig. 3. A slight neutron flux tilt was expected as a higher burned fuel element was swapped with a lower burned fuel element in close proximity to the fission chamber at the F4 position. It can be noted that the detector response changes were within ~2 %, which is within signal variation for all detectors and no neutron flux tilt due to the reshuffling was observed. For similar experiments in the future, it is proposed to place miniature fission chambers in measurement positions between the fuel element locations to study the localized changes in neutron flux due to rearrangement of differently burned fuel.

Table 1: Changes of detector response for the fuel swap experiment, where seven fuel elements were swapped. Detector response in 5 locations was analysed: fission chamber in F4 position and compensated ion chamber in ex-core detectors, depicted in Fig.1.

FUEL SWAP	CORE POSITION	F4 FISSION CHAMBER	START-UP CHANNEL	SAFETY CHANNEL	LIN CHANNEL	LOG CHANNEL
FUEL 7274	E2	0.55%	0.23%	-1.50%	-0.30%	-2.28%
FUEL-6574	E3	-0.92%	0.68%	-1.24%	0.19%	-1.31%
FUEL-7243	E4	-0.81%	0.01%	1.59%	-0.29%	-0.01%
FUEL-7213	E5	-0.53%	0.17%	1.62%	-0.21%	0.90%
FUEL 7235	D2	0.67%	0.08%	-1.05%	-0.41%	-0.41%
FUEL-7256	D3	0.09%	0.65%	-0.53%	0.20%	-0.52%
FUEL-7223	D4	-1.20%	-0.26%	0.67%	-0.21%	-0.33%
REFERENCE VALUE	Core 246	13372 counts/s	34208 counts/s	1.743E-10 A	9.9 W	12.6 W

5 CONCLUSION

The fuel reactivity effect of differently burned elements in the research reactor JSI TRIGA was measured by two methods. The conventional fuel reactivity worth method was compared with the new fuel reactivity swap method. The comparison showed that the latter is not limited by the number of available fuel elements and is easier to perform, but the connection between measured reactivity change and fuel burnup is not as trivial as for the reactivity worth method. Both experiments were simulated using three neutron transport codes: the deterministic TRIGLAV, the Monte Carlo Serpent-2, and the hybrid RAPID. Great agreement was observed for Serpent-2 and RAPID, indicating that the fuel burnup and criticality calculations are accurate. It can be concluded that state-of-the-art neutron transport codes can accurately predict small reactivity changes due to small burnup differences on the order of 10 pcm. For the TRIGLAV code, discrepancies were observed for fuel elements with lower burnup and higher reactivity effect. In addition, it was shown that due to the small differences in burnup, changes in detector response due to fuel-shuffling of two fuel elements cannot be detected when using ex-core detectors and large fission chamber in the core periphery. It is proposed to use a miniature fission chamber between the fuel elements in the future.

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