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Original Article

Neutronics analysis of JSI TRIGA Mark II reactor benchmark experiments with SuperMC3.3

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ABSTRACT

Jozef Stefan Institute (JSI), TRIGA Mark II reactor employs the homogeneous mixture of uranium and zirconium hydride fuel type. Since its upgrade, a series of fresh fuel steady state experimental benchmarks have been conducted. The benchmark results have provided data for testing computational neutronics codes which are important for reactor design and safety analysis. In this work, we investigated the JSI TRIGA Mark II reactor neutronics characteristics: the effective multiplication factor and two safety parameters, namely the control rod worth and the fuel temperature reactivity coefficient using SuperMC. The modeling and real-time cross section generation methods of SuperMC were evaluated in the investigation. The calculation analysis indicated the following: the effective multiplication factor was influenced by the different cross section data libraries; the control rod worth evaluation was better with Monte Carlo codes; the experimental fuel temperature reactivity coefficient was smaller than calculated results due to change in water temperature. All the results were in good agreement with the experimental values. Hence, SuperMC could be used for the designing and benchmarking of other TRIGA Mark II reactors.

1. Introduction

The Training Research Isotopes and General Atomics (TRIGA) reactor was developed based on the concept of inherent safety and widely researched in the world till now. For example, TRIGA reactor has a very strong prompt negative temperature coefficient due to the uranium-zirconium hydride fuel [1]. A typical 250-kW_{th} (thermal power) TRIGA Mark II reactor was reconstructed and upgraded at the Jozef Stefan Institute (JSI) and a series of experiments was performed [2–4] with well-defined, and well-controlled operating conditions. These experimental records provide benchmark data for TRIGA reactor calculations. Other investigations on these benchmark experiments are still in progress [5–10].

Monte Carlo (MC) method can deal with complex geometries and give accurate calculation results to perform comprehensive neutronics calculations for design and safety evaluations of nuclear systems [11]. In the last decades, a number of computation codes (e.g., MCNP [12] and SuperMC [13]) have been developed.

Super Multi-functional Calculation Program for Nuclear Design and Safety Evaluation (SuperMC) is a large-scale integrated software system which has been used for nuclear energy systems design and safety evaluation [14–18]; it has been used in the field of nuclear technology application [19,20] and the design of nuclear material [21–24].

In this work, three benchmark experiments on the JSI TRIGA Mark II reactor were analyzed and evaluated using the SuperMC code. A detailed analysis was conducted; the calculated results were compared with the benchmark results, experimental results and other available calculated results from different MC codes and deterministic method codes.

2. Benchmark model description of TRIGA Mark II reactor

The benchmark model of the TRIGA Mark II reactor core consists of 91 locations (i.e., for a possible arrangement of fuel elements, control rods, irradiation channel or neutron source) in a cylindrical configuration. The 91 locations were arranged in 6 concentric rings, marked A-F, which can be filled with fuel elements, control rods, neutron source, etc. In the benchmark modeling, the main core

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surrounding structures such as the irradiation channels, the graphite for neutron thermalization, the thermal columns, and the source element were omitted; the end caps of the fuel elements and control rods were also simplified. Three different core configurations are presented in Fig. 1. Both core-132 and core-133 are critical, and each has 40 fuel elements with different loading patterns. The core-132 have seven fuel elements at the E ring (i.e., on the same side of the transient rod-T), while core-133 have the seven fuel elements on the opposite side. Critical calculations were conducted on core-132 and core-133. Fig. 1 (c) (i.e., core-134) has five more fuel elements than the critical cores. Two parameters: control rod worth and fuel temperature reactivity coefficient were calculated from the core-134.

As already indicated, the fuel element is made of the homogeneous mixture of $ZrH_{1.6}$ and 12 wt % uranium of 20 wt % U-235/U enrichment. There are four control rods in the core, namely regulating (R), shim (C), safety (S) and transient (T). The first three (R, C, and S) are the fueled-follower type, and the transient rod is the air-follower type. All the four control rods have B_4C absorber part. The shim and the regulating rods are located at opposite sides in ring D; likewise the safety and the transient rods are located at opposite sides in ring C. The detailed description of the fuel elements and the control rods can be found in Ref. [4].

3. Calculations using SuperMC

Three benchmark experiments in the TRIGA Mark II reactor were analyzed and evaluated using SuperMC. The modeling and real-time cross section generation methods of SuperMC which are commonly used in MC nuclear neutronics calculations were applied to these benchmark experiments. The process application of the methods is briefly introduced and demonstrated in this section.

3.1. Modeling

The TRIGA Mark II reactor models were constructed using the quick modeling method of SuperMC [25]. The geometry parameters and material properties (e.g., density, compositions, etc.) were extracted from the corresponding benchmark model information in the International Handbook of Evaluated Critical Safety Benchmark Experiments (ICSBEP) handbook [4]. The TRIGA Mark II reactor model for core-134 is as presented in Fig. 2 (extracted from the SuperMC modeling interface).

3.2. Real-time cross section generation

The real-time cross section generation in SuperMC is based on a novel on-the-fly Doppler broadening method and use optimal double-exponential formula [26]. The method can generate a neutron cross section rapidly and precisely at desired temperatures. Thus, in the fuel temperature reactivity coefficient calculation, cross sections of the uranium-zirconium hydride fuel at 354 K were generated automatically and immediately by SuperMC for continuing calculations. The thermal neutron scattering S (α,β) cross sections of H in ZrH and Zr in ZrH at 354 K were generated from initial ACE format cross sections data at 294 K and 400 K using Neville interpolation; the cross sections of other nuclides (i.e., uranium) at 354 K were generated from an initial ACE format cross section data at 294 K.

4. Results and discussion

The results for the effective multiplication factor, control rod worth and fuel temperature reactivity coefficient of TRIGA Mark II reactor as calculated by SuperMC are compared with the benchmark results and the results of other codes. The relative differences in results between the codes and experiments are discussed in this section.

4.1. Effective multiplication factor (k_{eff})

To evaluate the impact of various cross section libraries, the ENDF/B-VI.6, ENDF/B-VII and ENDF/B-VII.1 with their corresponding S (α,β) libraries and a mixed library of ENDF/B-VII.1 and ENDF/B-VI.8 S (α,β) were all separately used in the SuperMC calculations. All the calculations were run with 5000 active cycles and 300 skipped cycles, each cycle had 4000 neutrons. The statistical uncertainties of the k_{eff} were within 19 pcm. The benchmark models were originated from the experimental models by geometric simplification and omitting the source element. As indicated earlier, the simplification of the geometry models was in accordance with the experimental description and also the source element was omitted. It has already been identified that geometric simplification of the model core-132 and core-133 contributes bias of about $+0.15\% \pm 0.03\%$ to the benchmark-model k_{eff} in both core-132 and core-133; and omitting the source element in core-132 contributes bias of $+0.04\% \pm 0.01\%$ [4]. The experimental, benchmark and the calculated k_{eff} from SuperMC (i.e., using different data library sets) and their uncertainties followed in bracket (x pcm) are listed in Table 1.

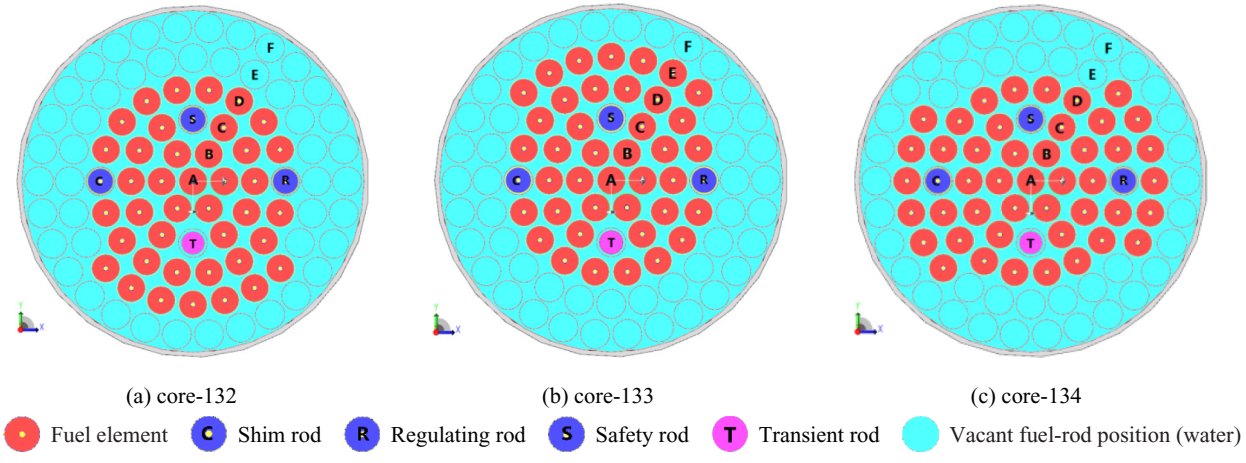


Fig. 1. Top views of three core configurations in SuperMC. The core-132 and core-133 are critical.

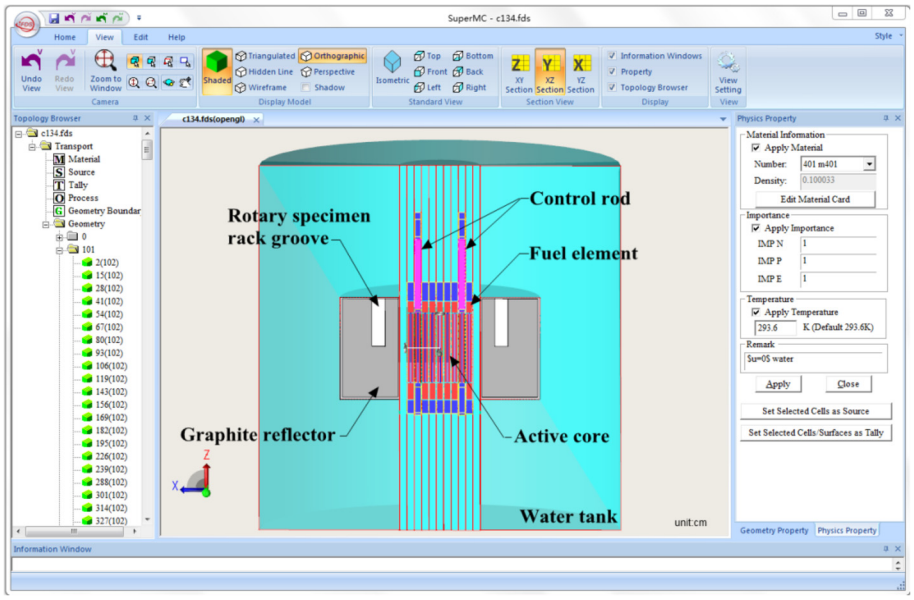


Fig. 2. Side view of core-134 in SuperMC interactive interface.

Table 1
Experimental, benchmark and calculated k_{eff} using different libraries.

Core ID	Experimental	Benchmark	SuperMC				MCNP [9]	TRIPOLI [8]	Serpent [9]
			ENDF/B-VI.6	ENDF/B-VII	ENDF/B-VII.1	mixed library[c]	ENDF/B-VII	ENDF/B-VII	ENDF/B-VII
132[a]	0.99865 (15)	1.0006 (56)	0.99992 (19)	1.00592 (19)	1.00289 (19)	1.00097 (19)	1.00610 (6)	1.00627 (30)	1.00498 (3)
133[b]	1.00310 (15)	1.0046 (56)	1.00455 (19)	1.01055 (19)	1.00755 (19)	1.00579 (19)	1.01062 (6)	1.00997 (30)	1.00915 (3)

Note.
[a] There was a source element in this experiment at the position of E12.
[b] No source element.
[c] Mixed library ENDF/B-VII.1 and ENDF/B-VI.8 S (α,β).

From the table, it can be deduced that the difference between the calculated and benchmark k_{eff} for core-132 and core-133 was below 300 pcm in all the cases of ENDF/B-VI.6, ENDF/B-VII.1 and the mixed library. The maximum difference between the calculated and benchmark k_{eff} using the ENDF/B-VII library were 529 pcm and 586 pcm for core-132 and core-133 respectively. Using the ENDF/B-VII library, the reported results from TRIPOLI [8], MCNP and Serpent [9] all overestimated the benchmark k_{eff} between 436 and 563 pcm and 449–593 pcm for core-132 and core-133 respectively, whereas SuperMC results laid in these ranges. The main contributors to the difference of k_{eff} using different cross section libraries were investigated [26]. The effect of the S (α,β) library on the k_{eff} was obtained by comparing the calculated results with the ENDF/B-VII.1 and the mixed library. The calculated results obtained using the mixed library with the ENDF/B-VI.8 S (α,β) showed approximately 200 pcm better agreement with the benchmark results.

4.2. Control rod worth

Control rod worth is an important parameter for the safety evaluation of reactors. As indicated in Fig. 1, there are four control rods: regulating, shim, safety and transient. The experimental results were measured by the rod-exchange method, where one-rod worth was measured by exchanging with another rod located in the opposite position [2]. In this calculation, the regulating rod and the opposite shim rod were chosen and simulated.

In the simulations, when one rod was selected for calculation, the other rods were completely withdrawn and then the selected rod was inserted into the active core at 5 cm interval for each simulation set. The control rod worth ($\rho_{w,i}$) is determined by the reactivity difference between the reactivity obtained in the condition when all four control rods are withdrawn (k_0) and the reactivity obtained in the condition when one control rod in a certain position (k_i): the relationship is as shown in equation (1):

$$\rho_{w,i} = \rho_0 - \rho_i = \frac{k_0 - 1}{k_0} - \frac{k_i - 1}{k_i} = \frac{1}{k_i} - \frac{1}{k_0} \tag{1}$$

The calculations were run with 5000 active cycles, each cycle had 4000 neutrons, and nuclear data library was the mixed library. As the experimental values in 5 cm were very small, 40000 neutron histories per cycle were used to ensure high accuracy. The calculated rod worths of the regulating and shim rods are shown separately in Tables 2 and 3. Other available results by KENOV.a [6] and PRIDE [10] codes were also involved for comparison. Polynomial fitting was used in the data processing to get smooth integral reactivity worth curves. The integral reactivity worth curves of the calculated and experimental results are shown in Fig. 3. Over 30 cm (critical position), the total worth of each control rod exceeds the available core excess reactivity, so they are vacant in Fig. 3. The rod worth in the fully inserted position was measured by the rod-insertion method [2] with the digital reactivity meter [27].

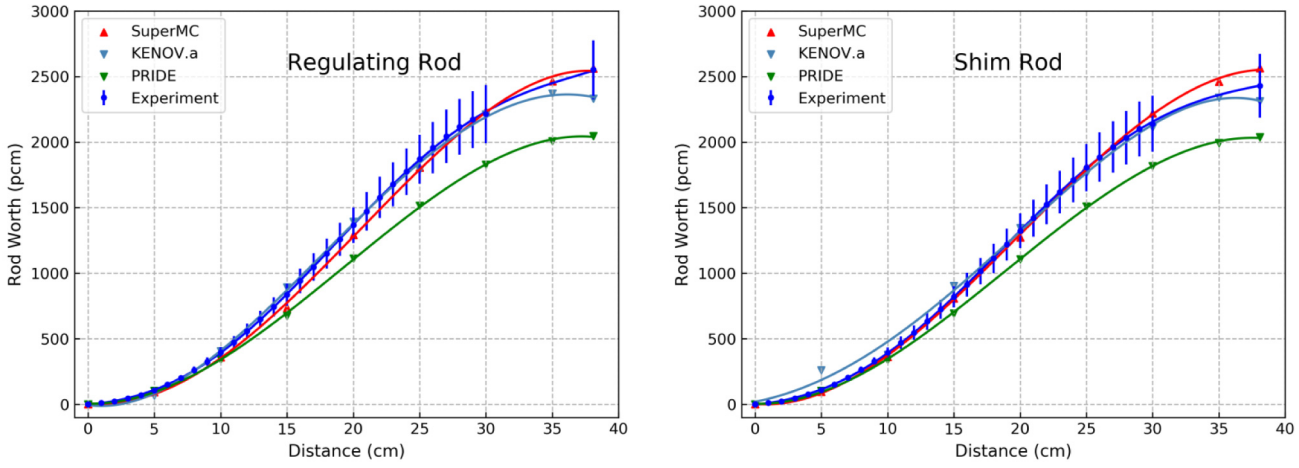
The associated experimental errors of the reactivity curves are approximately 10%, and as can be seen from the figures, SuperMC

Table 2
Regulating rod worth.

Depth (cm)	Experiment (±10% pcm)	SuperMC (pcm)	KENOV.a (±70 pcm)	PRIDE (pcm)
5	107	93 (±7)	67	103
10	396	359 (±27)	405	343
15	836	734 (±27)	892	674
20	1366	1291 (±27)	1393	1112
25	1869	1807 (±27)	1790	1516
30	2213	2237 (±27)	2211	1828
35	2423	2464 (±27)	2372	2007
38.1	2552	2561 (±27)	2331	2047

Table 3
Shim rod worth.

Depth (cm)	Experiment (±10% pcm)	SuperMC (pcm)	KENOV.a (±70 pcm)	PRIDE (pcm)
5	110	96 (±7)	260	103
10	394	361 (±27)	376	342
15	819	808 (±27)	902	693
20	1323	1727 (±27)	1344	1108
25	1805	1811 (±27)	1760	1509
30	2139	2221 (±27)	2110	1818
35	2318	2461 (±27)	2341	1992
38.1	2430	2564 (±27)	2311	2038



(a) Integral control rod worth for the regulating rod.

(b) Integral control rod worth for the shim rod.

Fig. 3. Comparison of the integral control rod worth (regulating and shim rods) from SuperMC calculations to experimental results and the results from KENOV.a [6] and PRIDE [10] codes.
(a) Integral control rod worth for the regulating rod.
(b) Integral control rod worth for the shim rod.

results lie within the limit of the experimental values. In addition, the total worth of each control rod exceeded the available core excess reactivity, which ensured the safety of the reactor. Moreover, SuperMC integral control rod worth results for the shim and regulating rods were almost the same. This was because the two control rods were the same except they were located in an almost symmetric opposite side of core-134.

The calculated results obtained by the multi-group MC code KENOV.a and the deterministic method code PRIDE are also shown in Fig. 3. On the other hand, code to code comparison of the results show that the SuperMC results are consistent with the KENOV.a, while the results from PRIDE are mostly out of the limit of the experimental values: the results were smaller than the MC codes of both SuperMC and KENOV.a in almost the whole range of the two control rods.

4.3. Fuel temperature reactivity coefficient

The TRIGA Mark II reactor fuel was made of uranium mixed with $ZrH_{1.6}$ (the moderator) homogeneously and developed on the concept of inherent safety. The unique safety feature of the reactor is due to the very strong prompt negative temperature coefficient of the fuel. In this study, the averaged fuel temperature $\overline{T}$ over the entire volume of the fuel in the core was calculated from the temperature measured at A ring (T_A). It is expressed as follows:

$$\overline{T} = \frac{1}{\gamma_r \gamma_z}(T_A - T_0) + T_0 \tag{2}$$

where T_0 (294 K) is the initial temperature of fuel at zero power and $\gamma_r = 1.26$, $\gamma_z = 1.25$ [2].

Table 4
Parameters in the fuel temperature reactivity coefficient calculations.

Fuel temperature (K)	Fuel density[a] (g/cm^3)		Water temperature (K)	Water density (g/cm^3)	Regulating rod position (cm)	Structure material
	Fuel element	Fuel follower				
294	6.04495	6.15776	294	0.99790	28.4	Constant
354	6.04494	6.15775	302	0.99582		

Note.
[a] The thermal expansion coefficient of U–$ZrH_{1.6}$ fuel is 7.4×10^{-6} $(1 + 2 \times 10^{-3}$ T (°C))/°C [1].

Table 5
Experimental and calculated results of the fuel temperature reactivity coefficient.

Condition	$\Delta\overline{T}$ (K)	$\Delta\rho$ (pcm)	α_F (pcm/K)
Fuel temperature change	60	−352.5 (±27)	−5.88 (±0.45)
Fuel temperature and density change		−394.6 (±27)	−6.58 (±0.45)
Fuel temperature and water density change		−415.7 (±27)	−6.93 (±0.45)
Fuel temperature, density and water density change		−428.7 (±27)	−7.15 (±0.45)
Experiment (±25%)		−460.6	−7.81

The fuel temperature reactivity coefficient α_F [28] is defined as follows:

$$\alpha_F = \frac{\Delta\rho}{\Delta\overline{T}} = \frac{\rho(354K) - \rho(294K)}{60K} \quad (3)$$

where $\rho(354K)$ and $\rho(294K)$ are the calculated reactivity at 354 K and 294 K; $\Delta\overline{T}$ (60 K) is equal to the change in the averaged fuel temperature $\overline{T}$. The cross section library at 354 K was generated automatically from the existing ACE format cross section data at 294 K by SuperMC during the calculations. In addition, other parameters, determined to be close as possible to the real experimental values, are listed in Table 4. The thermal expansion of the materials was taken into consideration by changing their densities.

As the temperature of the water in the core changed with the fuel temperature in the experiment, the experimental α_F is related to both the fuel and water temperatures. To investigate the effect of each factor, different combinations of fuel temperature change, fuel density change and water density change, were taken into account in the calculations explicitly. The calculated α_F are listed in Table 5.

As indicated in Table 5, the calculated α_F, obtained by considering the fuel temperature and density change was −6.58 pcm/K. The value is bigger than the experimental value of −7.81 pcm/K. To get a result closer to the experimental value, an optimum SuperMC calculated result −7.15 pcm/K was obtained. SuperMC optimum calculation of α_F took into consideration the fuel temperature, fuel density and water density change. It was also verified that the fuel temperature change made the largest contribution to the fuel temperature reactivity coefficient. In summary, all the calculated α_F lie within the limit of the experimental error (±25%). This result is therefore serves as a validation to the cross section libraries and then the efficiency of the real-time cross section generation method in SuperMC.

5. Conclusions

Three important benchmark experiments in the JSI TRIGA Mark II research reactor were calculated and analyzed using the SuperMC code. The k_{eff} of the critical core-132 and core-133 were calculated with four different ENDF cross section libraries. The ENDF/B-VII library showed the maximum difference in terms of k_{eff} parameter as compared to the benchmark results. Similar trends in the comparative results from TRIPOLI, MCNP, and Serpent were observed. In all cases, the difference was below 0.6% between the numerically calculated and benchmark results, and this confirmed the effect of different libraries on the k_{eff} and the accuracy of modeling used in SuperMC. In the case of control rod worth, SuperMC results for the two control rods (shim and regulating) agreed well with the experimental values. Finally, the fuel temperature reactivity coefficient was analyzed in different conditions using the real-time cross section generation method implemented in SuperMC. Moreover, it was concluded that when the calculation model got closer to the real experiment, better agreement with the experimental result was obtained. This work is a preliminary verification of the capability and the accuracy of SuperMC to simulate the uranium-zirconium hydride fuel type reactor: the TRIGA Mark II research reactor.

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