SERPENT-2 REACTOR MODEL VALIDATION AT THE VIENNA TRIGA MARK II REACTOR

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ABSTRACT

The Vienna TRIGA Mark II reactor was modelled by means of the Serpent-2 Monte Carlo burnup calculation code. The purpose of the modelling is the determination of core critical parameters and of nuclear fuel composition time evolution.
This paper presents the Serpent-2 reactor model validation process carried out during the last year. Taking into account the irradiation history, the fuel composition was calculated for a certain number of fuel elements in the current core. The fuel elements were selected in order to sample the different core areas, rising from internal B ring to the external part of the core. Subsequently, the calculation results were compared with different fission product activity values measured in the corresponding fuel elements. The measurement campaign was previously performed by means of a fuel gamma-scanning machine installed at the reactor as presented at the RRFM 2016.
The comparison of fission products activity values obtained by Serpent-2 model calculation and by direct measurements are presented in this paper.

1. Introduction

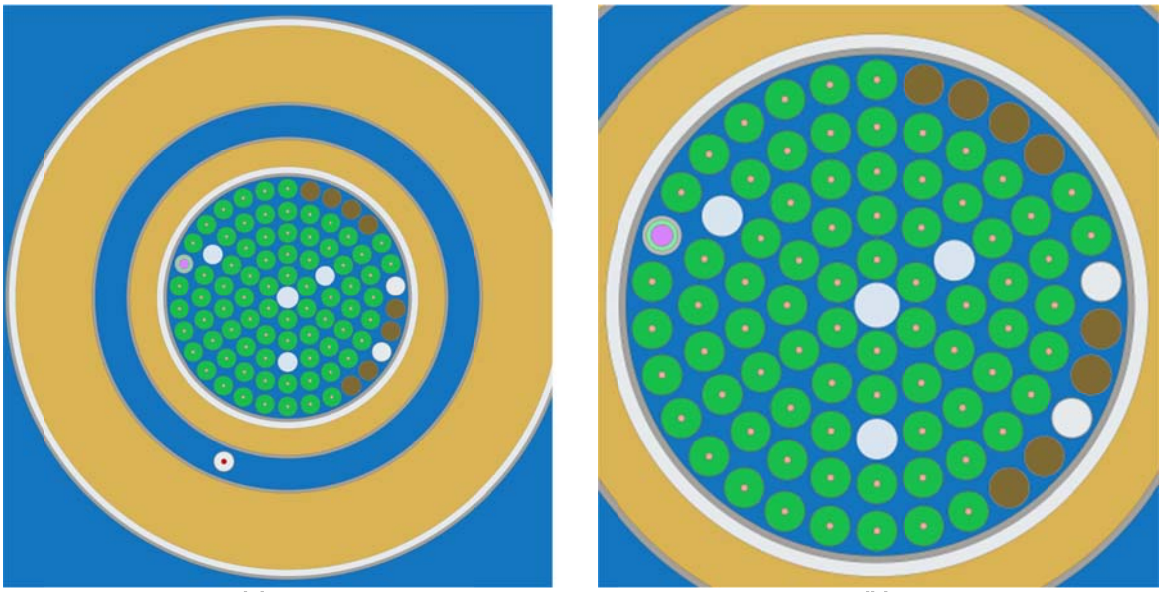
The TRIGA (Training Research and Isotope production General Atomics) MARK II reactor[1] at the Technical University Vienna is a pool-type research reactor moderated and cooled by light water, licensed for 250 kW steady state and up to 250 MW pulse operation. After conversion to a full LEU (Low Enriched Uranium) core, the current core load consists out of 76 stainless steel clad zirconium-hydride fuel elements (8.5%-wt enriched 19.95%-wt in 235U), in a cylindrical geometry.
Recent activity at the TRIGA reactor in Vienna included on one side the verification[2] of the developed MCNP6[3] reactor model against measured neutron flux distribution[4] in different in-core positions; and on the other hand, the measurement of fission product activity values in several irradiated fuel elements.
This work describes the new reactor model implemented by means of the Monte Carlo code Serpent-2[5] and the procedure for its validation. The validation was carried out by benchmark of Serpent-2 results against both MCNP6 calculated results and experimental activity results.

2. Serpent-2 code and the reactor model

Serpent-2 code is a Monte Carlo continuous-energy stochastic transport code for burn up calculations[5,6]. To define a three-dimensional geometric model of fuel element or nuclear reactors, Serpent uses a universe-based combinatorial solid geometry (CSG), likewise to MCNP and other reactor physics code. The geometry is built up of material cells, which are defined by diverse surface types. With this most of the reactor geometry

can be modelled. In the code some additional geometry features are included, to simplify the definition of cylindrical fuel pins and different core layouts.
For burn-up calculation, burnable materials are defined by the user and Serpent selects fission products and actinide daughter nuclides automatically. Burnable materials can be also sub-divided into several depletion zones. This capability was adopted in the present case (see chapter 3 and chapter 4): in fact, a fuel element can be divided along the z-axis; consequently, the outputs for burned material are also divided into smaller areas along the z-axis, instead of being for the total volume.
The output of a burn-up run are given material-wise and in total values for parameters like isotopic composition, activities, spontaneous fission rates and decay heat data. These parameters are printed after each burn up step. Additionally the composition of the burned material can be printed out

A three-dimensional model of the TRIGA reactor in Vienna was developed[7] with Serpent-2 for the first time. This model contains, at the proper level of detail, all essential core components that can affect the evaluation of the neutron flux in-core distribution and the energy spectrum in different in-core positions, including inside the fuel elements; as well as the reactor fuel burn-up and reactor critical parameters.



(a) (b)

Figure 1: The horizontal section of the TRIGA reactor model as obtained with Serpent-2.

The reactor core horizontal section of model obtained by Serpent-2 is shown in Figure 1: Fig.1a shows also the graphite reflector and the circular ring irradiation facility (Lazy Susan) surrounding the core inside the graphite reflector. Fig.1b shows the detail of the current reactor core, the different core component and their current location: the 76 cylindrical FE(s) (green colored), including the visible central Zirconium rod; the neutron source (pink colored, in position F25); the control rods position (in grey, as they are represented completely extracted); the graphite elements (brown colored); the water inside the core (blue colored).
A vertical view of the reactor model including the graphite reflector is shown in Figure 2: main components of the FE(s), such as fuel meat, central Zirconium rod, poison disks, axial graphite reflectors, Al-cladding, were included in the fuel elements modelling.

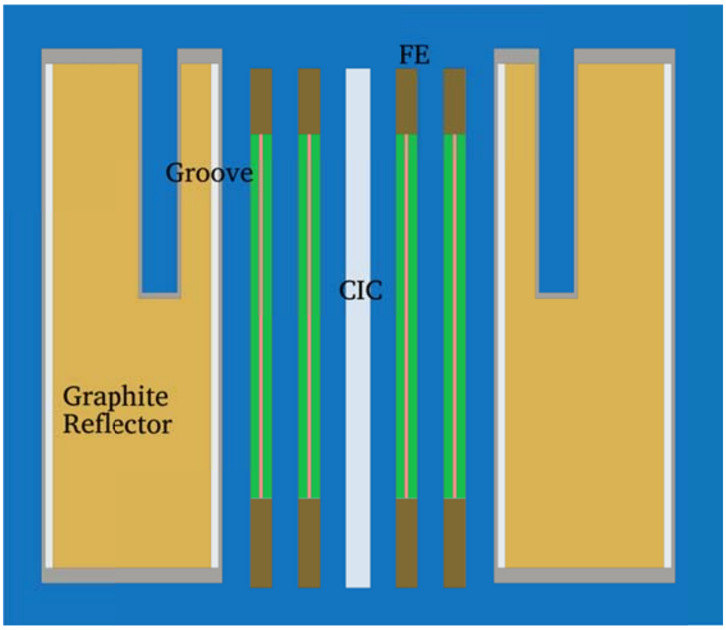


Figure 2: The vertical section of the TRIGA core modeled with Serpent-2.

The neutron reaction data which Serpent uses for transportation calculation are taken from the OECD/NEA Data Bank. Serpent provides libraries based on JEF-2.2, JEFF-3.1, ENDF/BVI. 8 and ENDF/B-VII. For the following simulation the ENDF/B-VII library was used.
A thermal scattering card was used to represent the moderation properties of water, graphite and Zirconium-Hydrogen. These cards were taken from the MCNP6 libraries and copied to the Serpent libraries.

3. Validation of the Serpent-2 model

The first step to validate the Serpent model was to calculate the neutron flux at different position and compare the results with the corresponding values obtained with previously validated MCNP6 reactor model[2].
The neutron flux was calculated with Serpent both along vertical axis and along the radial direction.
Along the vertical axis, 11 positions were calculated in the core Central Irradiation Channel (CIC). The locations of irradiation positions are listed and the exact distances are reported in Table 1; distances are taken from the core equatorial position along the vertical axis (z=0).
Along the radial direction, the neutron flux was calculated in 3 positions at the equatorial level of the core (z=0). For these selected positions (position *b, i, o*) the radial distances from the center of the core (x=0) are also listed in Table 1. Position 6, i.e. core center position, is taken into account as 4^{th} radial position.
At the position of interest, a detector was placed: a cell detector was selected for these calculations.
The calculation with Serpent was performed in such a way to produce results in the form of differential neutron flux over 30 energy groups: the width of the energy groups was chosen same as done in the past for similar MCNP6 calculations. Multiplying the differential value on each group with the width of the energy group, the corresponding integral flux distribution was calculated. As an example, the comparison between Serpent and MCNP6 differential flux over 30 energy points is provided in Figure 3 for the position corresponding to the center of the core (POS 6).
The thermal and total neutron fluxes along the vertical core direction (z axis) obtained both by means of Serpent-2 and MCNP6 are compared as shown in Figure 4.

CIR Irradiation position	Vertical distance along Z axis (cm)
Position 1	20
Position 2	16
Position 3	12
Position 4	8
Position 5	4
Position 6 (EQ)	0
Position 7	-4
Position 8	-8
Position 9	-12
Position 10)	-16
Position 11	-20

RADIAL Irradiation position	Radial distance along X axis (cm)
Position 6 (EQ)	0
Position ***b***	-5
Position ***i***	-13.5
Position ***o***	-22

Table 1: In-core Irradiation positions for flux determination along the vertical axis (Central Irradiation Channel, CIC) and along the X axis (RADIAL Irradiation positions).

The neutron fluxes behavior along the core radial direction (x axis), is displayed in Figures 5: the thermal and total neutron flux component from Serpent-2 and MCNP6 are compared.
The statistic error in Serpent was 1% or lower, while in the MCNP simulation it reaches a maximum of 7% for thermal flux and 3% for total flux.
After comparing MCNP and Serpent, it can be stated that they are in very good agreement. The difference is below 10%, and in general it's clear that the further away from the center the neutron flux is represented, the greater the difference between Serpent and MCNP.

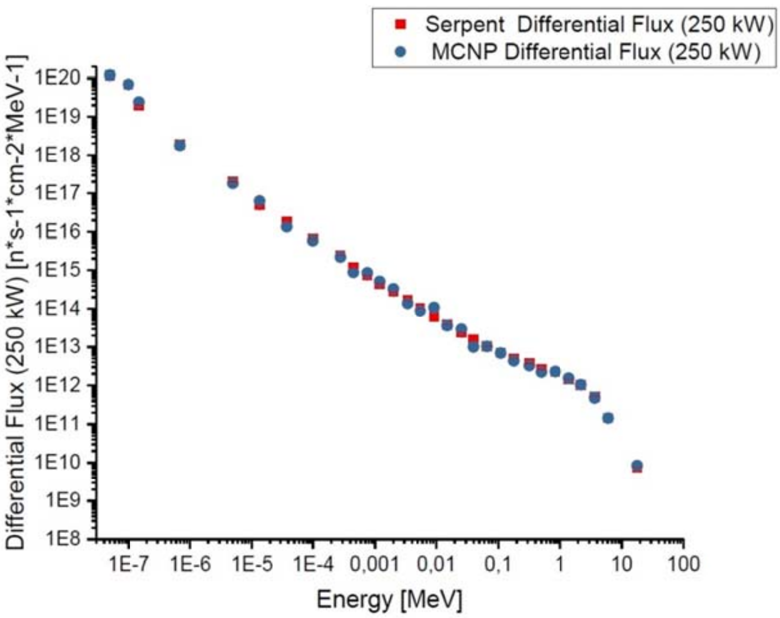
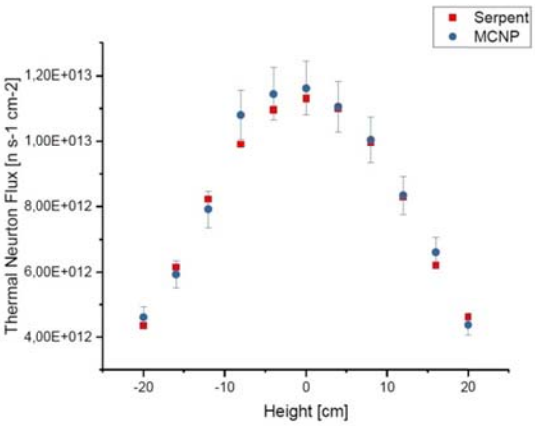
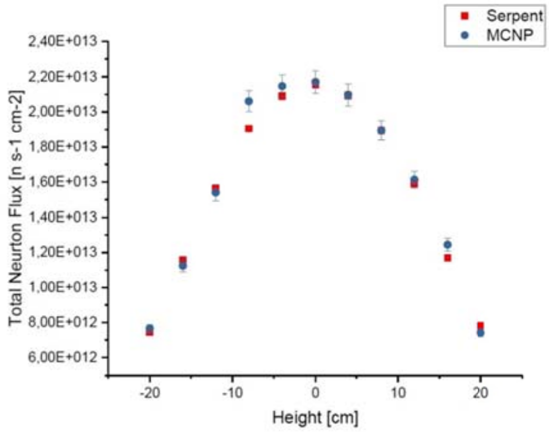


Figure 3: Serpent and MCNP6 Differential Flux in the Central Irradiation Channel (Positions 6, core center).

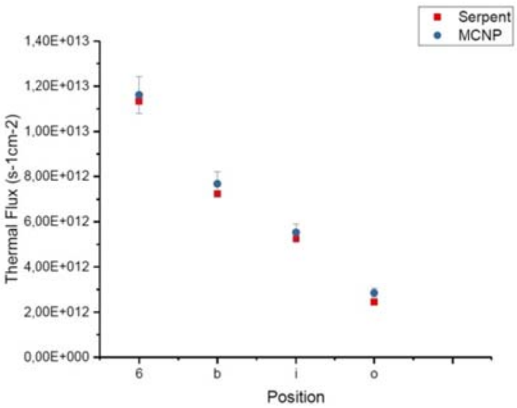


(a) VERTICAL DIRECTION - Thermal neutron flux

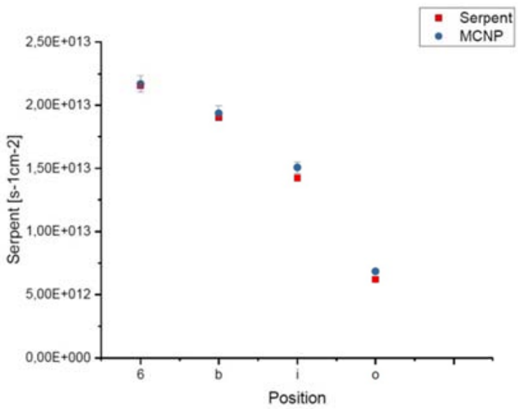


(b) VERTICAL DIRECTION - Total neutron flux

Figure 4: Behaviour of the neutron flux along the vertical core direction (z axis) obtained by means of Serpent-2 and MCNP6.



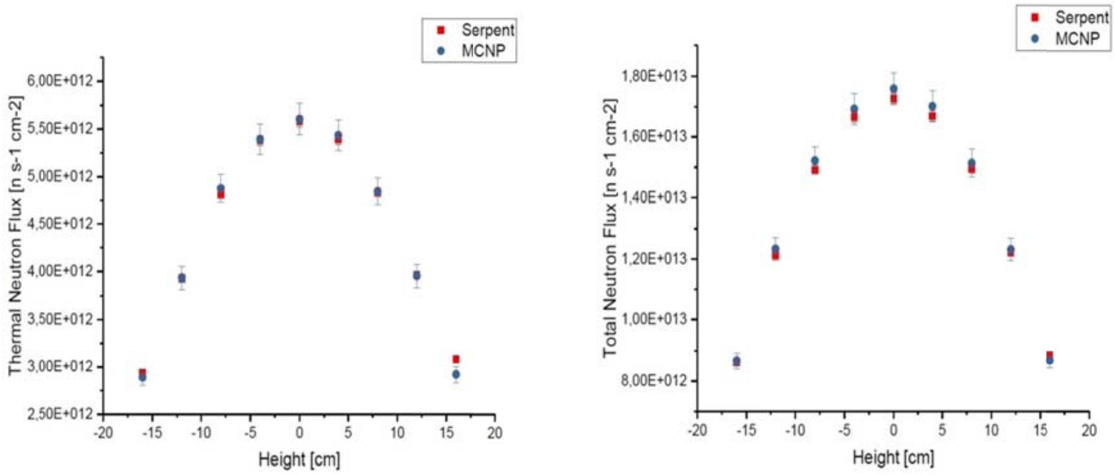
(a) RADIAL DIRECTION - Thermal neutron flux



(b) RADIAL DIRECTION - Total neutron flux

Figure 4: Behaviour of the neutron flux along the radial core direction (x axis) obtained by means of Serpent-2 and MCNP6.

For burn-up calculation the good behavior of the neutron fluxes inside fuel elements is of primary interest. Because of this, the neutron flux inside one sampled fuel element was calculated by Serpent simulation and compared with the correspondent values in the verified MCNP model.
The analysed fuel element was the FE 9213 in Position B2. The fuel meat was divided into nine parts along the z-axis, each of these cells presented an extension of about 4 cm in the z direction and was a cell detector for the neutron flux.
The obtained total and thermal fluxes are shown in Figure 5.
The statistical error in the Serpent simulation is less than 1% and in the MCNP simulation an uncertainty of 3 % was assumed, because of the high neutron flux inside a fuel element. The difference between the MCNP model and the Serpent model is insignificant, the highest is 5%. From this it was concluded that Serpent represents the neutron flux very well inside a fuel element.
As expected, the further one goes outwards, the neutron flux decreases. This can be observed in the Figure 6, where the Serpent neutron flux for five fuel rods in the different rings (B-E) is shown.



(a) B2 fuel element - Thermal neutron flux in the fuel meat.

(b) B2 fuel element - Total neutron flux in the fuel meat.

Figure 5: Neutron flux inside the fuel element in position B2, values along its vertical axis as obtained by means of Serpent-2 and MCNP6.

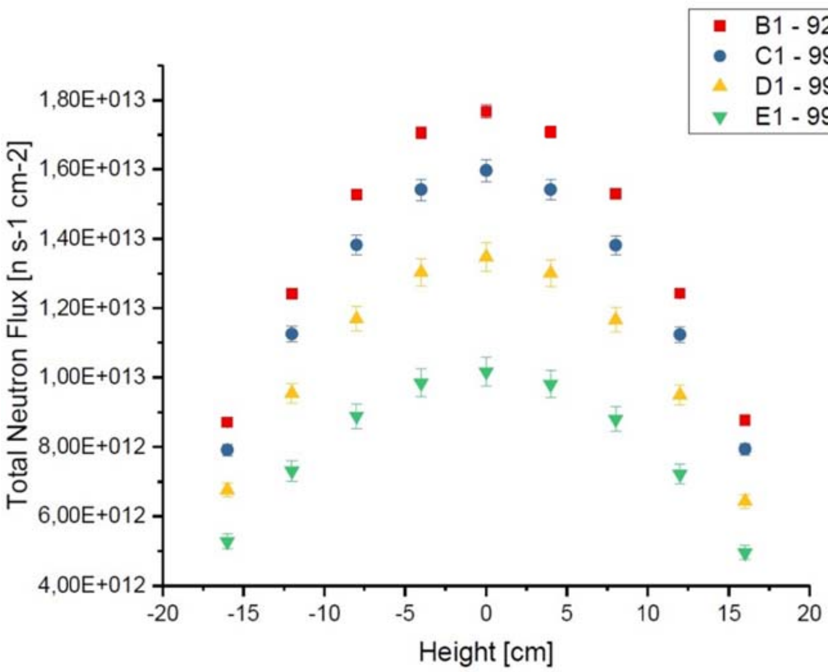


Figure 6: Serpent total neutron flux in the different Fuel Elements positioned in rings B,C,D,E.

4. Fueled experiment Serpent simulation

Following the verification of reactor Serpent model based on in-core flux distribution, Serpent model was used to reproduce a fueled experiment.
In fact, some fuel elements were recently measured by gamma spectroscopy after irradiation in the current core[8]: fission products were detected and their activity values determined. The target of the present Serpent simulation was the verification of the reactor model behavior as for the evaluation of irradiated fuel composition and activity.

Fuel Element	Load to core	Last irradiation	Measurement	Core position
9213	21/01/2013	25/03/2015	01/12/2015	B2
9214	21/01/2013	25/03/2015	02/12/2015	B4
9905	21/01/2013	25/03/2015	03/12/2015	C1
9915	21/01/2013	25/03/2015	03/12/2015	D1
9932	21/01/2013	25/03/2015	03/12/2015	E1

Table 2: fuel elements considered for Serpent verification against measured activity values.

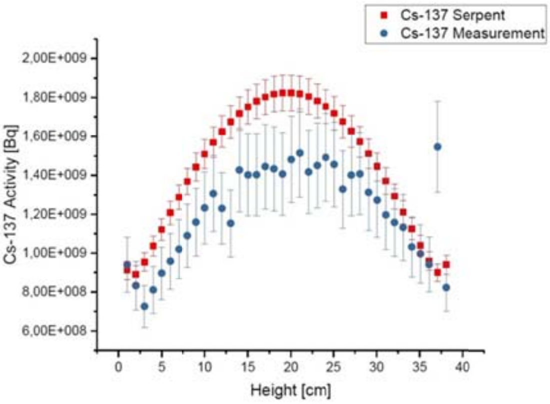
The measured fuel elements that were investigated by Serpent calculation are listed in Table 2: their positions vary in order to check various distances from the core center (from the inner ring B till the E ring); two fuel elements were considered in ring B as one position (B2) lays between the Central Irradiation Channel and one of the control rods. All the fuel elements were loaded in the core in the same date and underwent no reshuffling during the considered period. Considering the limited operation period (see Table 2), the cumulative work of the reactor was also limited and resulted to be:

$$W_{21.01.13\text{-}01.04.25} = 547.841\ MWh$$

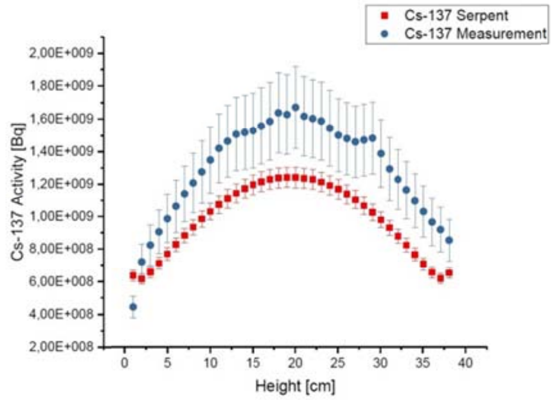
The Serpent burn-up simulation run with 1 million source neutrons per cycle, for a total of 1500 cycles, where the first 70 cycles are skipped. The model starts with the initial guess of k_{eff} = 1 and after 70 cycles the k_{eff} value has an accepted value. The unresolved resonance probability tables are switched on.
As a simplification it was assumed that the reactor operated at full power of 250 kW continuously: that is, for a corresponding period of 91.31 operational days till the cumulative work of the reactor is reached.
Each investigated fuel element was divided in 38 discs along the z-axis, 1 cm extension each, to get the vertical distribution of the activity. In the output, Serpent provided the material composition in each 38 cells, where all isotopes are expressed as atomic density (unit $10^{24}/cm^3$). Multiplied by the volume of the cell, the total number of nuclides (N) was obtained, and then the activity A was calculated as follows:

$$A = dN / dt = N \lambda$$

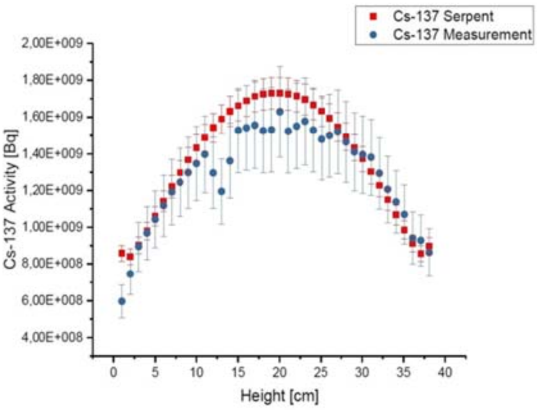
In Figure 7, the Cs-137 activity distribution along the z-axis, in each of the five fuel elements, is shown as obtained by Serpent burn-up calculation. The results are compared with experimental values obtained by gamma spectrometry. Considering the approximation of modelling and calculation, the uncertainty can be evaluate of about 5% for Serpent values.
Serpent calculated results can be considered in fair agreement with the corresponding experimental activity values, demonstrating the capability of the reactor model to reproduce the fission product inventory in irradiated fuel elements.



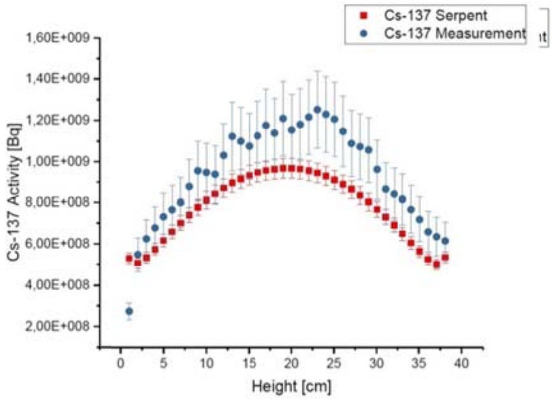
(a) Fuel element B2- 9213



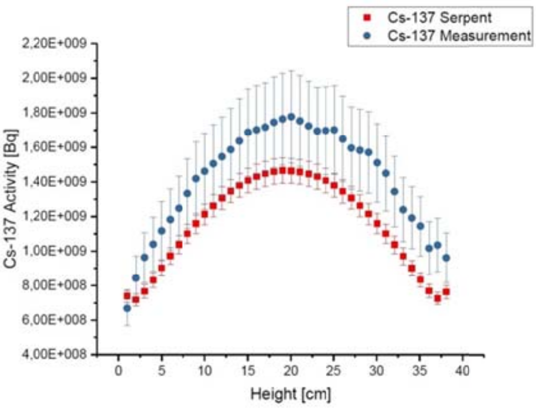
(d) Fuel elemet D1 - 9915



(b) Fuel element B4 - 9214



(e) Fuel element E1 – 9932



(c) Fuel element C1 - 9905

Figure 7: Compared Activities of the isotopes Cs-137 from the measurements (triangle) and the Serpent simulation (squares).

5. Discussion and conclusions

In the framework of recent activity at the TRIGA reactor in Vienna, this paper describes the new reactor model implemented by means of the Monte Carlo code Serpent-2 and its validation. The validation was carried out by benchmark of Serpent-2 results against both MCNP6 calculated results and experimental activity results.
The neutron flux was calculated in several in-core positions with Serpent, both along vertical axis and along the radial direction. Serpent flux values were compared with

corresponding MCNP6 values and they resulted in very good agreement. The difference is always below 10% and it reaches its maximum value in the core external borders.
As for burn-up calculation the good behavior of the neutron fluxes inside fuel elements is of primary interest, the neutron flux inside one sampled fuel element was verified by comparison of Serpent and MCNP6 values. The difference between the MCNP model and the Serpent model resulted below 5%, allowing to conclude that Serpent represents the neutron flux very accurately inside a fuel element.
After Serpent model verification based on in-core flux distribution, Serpent was used to reproduce a fueled experiment. For selected irradiated fuel elements, composition and activity were evaluated. The Serpent Cs-137 activity distribution along the z-axis, in each of the five fuel elements, was compared with experimental values obtained by gamma spectrometry. Serpent results resulted in fair agreement with the experimental activity values, demonstrating the capability of the reactor model to reproduce the fission product inventory in irradiated fuel elements.

6. References

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