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Research Article

Analysis of the triga mark-II benchmark ieu-comp-therm-003 with monte carlo code openmc

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ABSTRACT

Ensuring the reliable use of particle transport computer codes necessitates verification against benchmark experiments. This study aims to verify the Monte Carlo code OpenMC using the criticality benchmark model IEU-COMP-THERM-003 from the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook. The analysis focuses on the TRIGA Mark II reactor cores 132 and 133, employing nuclear cross-section libraries ENDF/B-VIII.0, ENDF/B-VII.1, ENDF/B-VII.0, and ENDF/B-VI.2. Results show that OpenMC provides KEFF values in close agreement with benchmark values, demonstrating its robustness in neutronic simulations. Comparison with MVP code results obtained previously, particularly with JENDL-3.3, shows similar accuracy.

Introduction

Verifying a particle transport code against a benchmark experiment is essential to ensuring its reliable and effective use. This verification process serves two primary purposes: first, to assess the consistency of the physical models and data utilized in the transport code; second, to identify systematic errors arising from the approximate simulation of the experiment, typically due to simplified model geometry.

In this study, the criticality benchmark model IEU-COMP-THERM-003 in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook (Jeraj and Ravnik, 1999) was examined using the Monte Carlo code OpenMC (Romano et al., 2015). The model is derived from benchmark experiments conducted in the steady-state operation of the TRIGA Mark II reactor in Ljubljana, Slovenia. These experiments were part of the startup tests following upgrade and reconstruction in 1991 and utilized completely fresh fuel (Mele et al., 1994a; Mele at al., 1994b). This research reactor has a relatively simple and compact geometry, making it an ideal choice for benchmark calculations. Furthermore, the significant effects of diverse material compositions can be thoroughly examined, allowing for the explicit modeling and separate treatment of specific components. The benchmark presents two core configurations identified as Core 132 and Core 133, which differed only in loading pattern. The fuel is the standard commercial TRIGA fuel element with stainless steel cladding. The active region consists of a homogeneous mixture of uranium and zirconium hydride, with a uranium concentration of 12 wt% and an enrichment level of 20 wt%. The analysis was done with the nuclear cross-section libraries based on ENDF/B-VIII.0 (Brown et al., 2018), ENDF/B-VII.1 (Chadwick et al., 2011), ENDF/B-VII.0 (Chadwick et al., 2006), and ENDF/B-VI.2 (Dunford, 1992). This study is intended to provide a verification of the OpenMC code in the neutronics simulation of small research reactors.

The paper is organized as follows: Section 2 describes the neutronic calculation models, Section 3 presents the simulation results, followed by

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discussions, and Section 4 summarizes the main conclusions of the study.

Description of the benchmark model

There are 91 locations in the core, which are occupied by fuel and control elements. The elements are arranged in six concentric rings about the center position labeled A, B, C, D, E, and F, with 1, 6, 12, 18, 24, and 30 positions, respectively. In Core 132, seven fuel elements are positioned in the E ring on the same side as the transient rod. In contrast, Core 133 places these fuel elements on the opposite side, as illustrated in Fig. 1. The lattice does not have a periodic structure (non-uniform pitch). The core is encircled by a graphite reflector housed within an aluminum casing. An annular groove in the upper part of the reflector body accommodates the carrousel irradiation facility.

The Monte Carlo method is a statistical technique widely used for simulating the behavior of neutrons and photons in complex nuclear systems. It can accurately account for the stochastic nature of particle interactions and capture detailed spatial and energy distributions. Using this method, it is possible to model complex geometries and utilize continuous-energy cross-sections.

Core 132 and 133 were modeled in full three dimensions in OpenMC. OpenMC is an open-source, general-purpose Monte Carlo code designed for continuous-energy, generalized-geometry calculations of stochastic neutron and photon transport. It can perform fixed source, subcritical multiplication, and criticality calculations on three-dimensional models of fuel elements or nuclear reactors using universe-based combinatorial solid geometry (CSG) or computer-aided design (CAD) representations. The code has a versatile and efficient tally system for analyzing various physical quantities. OpenMC is becoming widely regarded for its rich and extensible programming interfaces in Python and C/C++, which allow for a set of features including pre-and post-processing, multigroup cross-section generation, custom source option, material burn-up analysis, multiphysics coupling, geometry, and tally results visualization, as well as workflow automation. OpenMC provides official cross-section data libraries based on several ENDF and JEFF releases in HDF5 format. Multiple temperature evaluations are contained with the same HDF5 file for incident neutron data for each isotope. Additionally, it incorporates a Python-based nuclear data module that allows inspection, modification, and processing of various types of data, including ENDF, ACE, and HDF5 files. Benchmark details were carefully considered in the development of the model to ensure an accurate representation of the geometry and composition. The core, fuel element, control rod, and transient rod modeled in OpenMC are shown in Fig. 2-3.

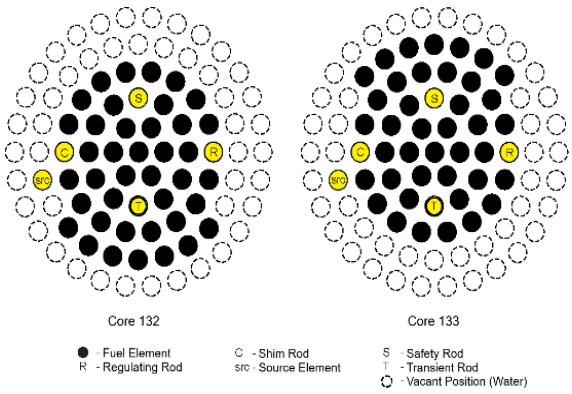
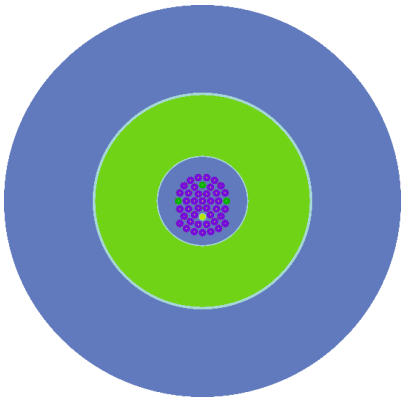
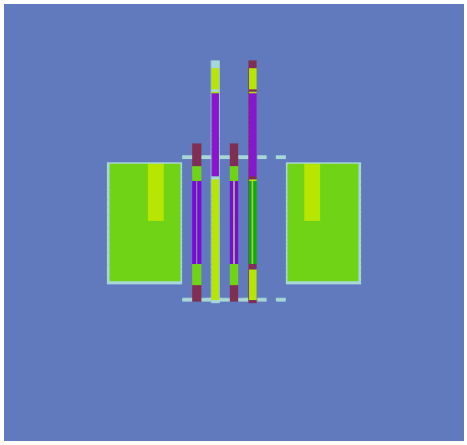


Fig. 1. Configurations of Core 132 and Core 133 (Jeraj and Ravnik, 1999).

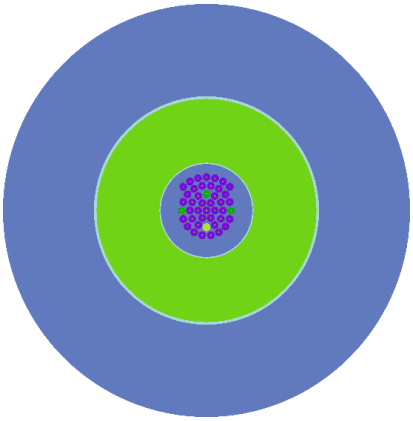


(a)

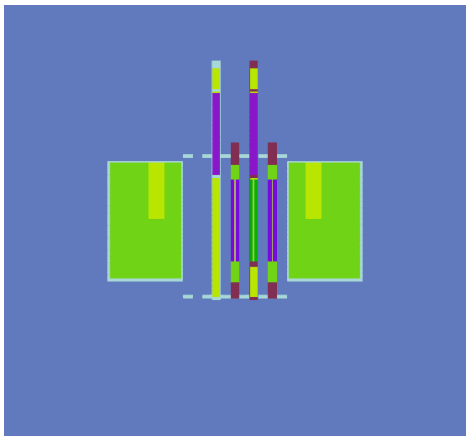


(b)

Fig. 2. (a) Top view and (b) side view of the benchmark model Core 132 in OpenMC.



(a)



(b)

Fig. 3. (a) Top view and (b) side view of the benchmark model Core 133 in OpenMC.

Results and Discussion

The primary objective of modeling and simulation the critical benchmark core configurations was to assess the reliability and validate the OpenMC code for the neutronic calculation of small reactors like the TRIGA Mark II research reactor. This benchmark, involving enriched uranium in a thermal spectrum, is an important test for evaluating accuracy and reliability in criticality calculations. The code's flexible geometry representation allows precise modeling of complex core configurations, such as those in the IEU-COMP-THERM-003 benchmark, ensuring accurate capture of fuel elements, control rods, transient rods, and structural components. Additionally, the Python interface was utilized to generate custom cross-section libraries. Also, an attempt was made to establish a difference between different cross-section libraries. In each of the OpenMC simulations for the criticality benchmark model, a stable fission source spatial distribution was achieved with the first skipping 50 batches and followed by 200 active batches by tracking histories of 100,000 particles per batch. The initial source distribution was taken from seven isotropic point sources.

As mentioned, continuous-energy neutron data were used from four cross-section libraries, namely ENDF/B-VIII.0, ENDF/B-VII.1, ENDF/B-VII.0, and ENDF/B-VI.2 in the criticality simulations. The calculations also included thermal scattering data for light water, graphite, hydrogen, and zirconium in the U-ZrH fuel. This is to compare and investigate the consistency and differences between different neutron cross-section evaluations. Due to the unavailability of certain data from the ENDF/B-VI evaluation, neutron cross-section data for Cr, Fe, and Ni were sourced from the ENDF/B-V library. The scattering data for hydrogen in ZrH, water molecules, and graphite were obtained from the ENDF/B-IV library. All the cross-section data used was for either 293.6 K or 300 K. It is worth noting that using multiple cross-section libraries provides a more comprehensive analysis and a better understanding of the uncertainties and discrepancies in the results.

The comparison was conducted with previous results (Mahmood et al., 2004) obtained using the MVP code (Mori and Nakagawa, 1994), utilizing cross-section data libraries based on JENDL-3.3 (Shibata et al., 2002), JENDL-3.2 (Nakagawa et al., 1995), and ENDF/B-VI.8 (Dunford, 1992). MVP is a general-purpose continuous-energy Monte Carlo code developed utilizing a vectorized algorithm for neutron and photon transport calculations. The contents of the official MVP library are primarily based on JENDL-3.3. Major features of the code include the treatment of scattering models, time-dependent tallies, reaction rate calculations, and flexible source specifications.

The MVP libraries based on JENDL-3.3 included three types of thermal scattering data for ZrH: the free gas model, ENDF/B-VI, and ENDF/B-III. Similarly, the libraries based on JENDL-3.2 featured thermal scattering data from the free gas model and ENDF/B-III. The libraries derived from ENDF/B-VI.8 used the free gas model and ENDF/B-VI for thermal scattering data. All libraries' data above the thermal energy region were consistent with those from the free gas model. The water temperatures for the models were set at 23.5°C (Core 132) and 22.5°C (Core 133), with a water density of 0.9975 g/cm^3, corresponding to the density at 23°C. All the cross-section data were assumed to be at room temperature (27°C). For natural elements, isotopic cross-section data were utilized when available in the libraries. The initial 100 batches were disregarded for the MVP calculations, followed by 2,000 active batches, each tracking 10,000 particle histories.

Several calculations and uncertainty analyses were originally published on this subject (Jeraj et al., 1997); we only compare our model effective multiplication factor, KEFF, with the benchmark values. KEFF is defined as the ratio of the number of neutrons generated by fission in one generation to the number of neutrons lost due to either absorption or leakage in the previous generation. It represents the balance of neutron production and losses in a reactor. The KEFF values for the benchmark core configurations and other calculation results are presented in Table 1-5. The results of the OpenMC presented show the values obtained using the combined collision, track-length, and absorption KEFF estimator. It is to be noted that the source element was not included in either of the two core configurations of the benchmark model, although the source was present in the benchmark experiment.

Table 1. Criticality benchmark KEFF data.

Case	132	133
Benchmark	1.000607 ± 0.00560	1.004607 ± 0.00560
MCNP4 (ENDF/B-VI)	0.9994 ± 0.0002	1.0042 ± 0.0002

Table 2. Results of KEFF for OpenMC Core 132.

Library	KEFF	Relative Error
ENDF/B-VIII.0	1.00252 ± 0.00020	0.19%
ENDF/B-VII.1	1.00191 ± 0.00020	0.13%
ENDF/B-VII.0	1.00377 ± 0.00020	0.32%
ENDF/B-VI.2	0.99720 ± 0.00019	0.34%

Table 3. Results of KEFF for OpenMC Core 133.

Library	KEFF	Relative Error
ENDF/B-VIII.0	1.00716 ± 0.00021	0.65%
ENDF/B-VII.1	1.00595 ± 0.00020	0.53%
ENDF/B-VII.0	1.00816 ± 0.00019	0.75%
ENDF/B-VI.2	1.00151 ± 0.00020	0.09%

Table 4. Results of KEFF for MVP Core 132 (Mahmood et al., 2004).

Library	KEFF
JENDL-3.3	0.99910 ± 0.0002
JENDL-3.2	1.00333 ± 0.0002
ENDF/B-VI.8	0.99831 ± 0.0002

Table 5. Results of KEFF for MVP Core 133 (Mahmood et al., 2004).

Library	KEFF
JENDL-3.3	1.00360 ± 0.0002
JENDL-3.2	1.00752 ± 0.0002
ENDF/B-VI.8	1.00270 ± 0.0002

In both Core 132 and Core 133, the calculated results using OpenMC show reasonable agreement with the benchmark values. The KEFF values for Core 132 were calculated using ENDF/B-VIII.0, ENDF/B-VII.1, and ENDF/B-VII.0 libraries show a slight overestimation compared to the benchmark value, while ENDF/B-VI.2 shows a slight underestimation. The ENDF/B-VIII.0 and ENDF/B-VII.1 libraries provide the closest results to the benchmark, with relative errors within 0.2%.

For Core 133, all libraries overestimate the KEFF value compared to the benchmark value. The ENDF/B-VI.2 library shows results in the smallest relative error at 0.09%, indicating a very close agreement with the benchmark value. The ENDF/B-VIII.0 and ENDF/B-VII.1 libraries also show reasonable agreement, though with slightly higher relative errors.

In both Core 132 and Core 133, the KEFF values obtained using the OpenMC and MVP codes show varying degrees of agreement with the benchmark values. The JENDL-3.3 library for both cores shows good agreement with the benchmark values, while the JENDL-3.2 results tend to overestimate the experimental results by approximately 0.3%. The ENDF/B-VI.8 library results are consistent between OpenMC and MVP, showing no significant differences and suggesting that the choice between these library versions does not significantly impact the KEFF values.

Conclusion

This study aimed to verify the reliability of the OpenMC code for neutronics simulations by analyzing the criticality benchmark model IEU-COMP- THERM-003, as detailed in the ICSBEP Handbook. The analysis focused on two core configurations of the TRIGA Mark II research reactor, using various nuclear cross-section libraries, including ENDF/B-VIII.0, ENDF/B-VII.1, ENDF/B-VII.0, and ENDF/B-VI.2. The simulation results demonstrate that the OpenMC calculations reasonably agree with the benchmark values for Core 132 and Core 133. Overall, this study confirms the robustness of the OpenMC code in simulating small research reactors like the TRIGA Mark II.

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