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Testing of Cross Section Libraries for TRIGA Criticality Benchmark

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ABSTRACT

Influence of various up-to-date cross section libraries on the multiplication factor of TRIGA benchmark as well as the influence of fuel composition on the multiplication factor of the system composed of various types of TRIGA fuel elements was investigated. It was observed that k_{eff} calculated by using the ENDF/B VII cross section library is systematically higher than using the ENDF/B-VI cross section library. The main contributions (~220 pcm) are from ^{235}U and Zr.

1 INTRODUCTION

The purpose of this paper is to investigate the influence of various up-to-date cross section libraries on the multiplication factor of TRIGA benchmark [1], as well as the influence of fuel composition on the multiplication factor of the system composed of various types of TRIGA fuel elements.

The calculations are performed with Monte Carlo computer code, MCNP [2], using ENDF/B-VI.8, JEFF 3.1 [3] and in particular the newest ENDF/B-VII cross section library [4]. The uncertainty analysis of the most important contributions to k_{eff} is presented with emphasis on thermal multiplication in TRIGA as affected by special neutron scattering properties of H in Zirconium lattice.

Multiplication factor variations due to different combinations of high and low uranium concentration TRIGA fuel elements in the same multiplying system are calculated. The variations due to the differences in uranium concentration are treated as well. The results are interesting for criticality analysis of storing and shipment operations when fuel elements of low and high uranium content are often mixed in the same multiplying system.

2 TRIGA MARK II RESEARCH REACTOR

TRIGA research reactor at Jozef Stefan Institute is a typical 250-kW TRIGA Mark II. It is a light water reactor with an annular graphite reflector cooled by natural convection [1].

The benchmark experiments were performed as part of the startup test after reconstruction and upgrading in 1991. All core components, with the exception of the graphite reflector around the core were replaced with new ones in the process [1].

The benchmark experiment was performed with standard commercial TRIGA fuel elements of 20 wt. % enrichment and 12 wt. % uranium concentration. The benchmark was thoroughly validated and included in the Handbook of Evaluated Criticality Safety Benchmarks by the ICSBEP [1].

There also exist other types of TRIGA fuel elements of different uranium concentration but same enrichment. The main physical characteristics of different TRIGA fuel element types are presented in Table 1.

Table 1. Main physical characteristics of different TRIGA fuel element types used in the model

COMMON

fuel

material	U ZrHx
inner diameter	0.635 cm
outer diameter	3.645 cm
length	38.10 cm

cladding

material	stainless steel
outer diameter	3.754 cm
thickness	0.0508 cm

SPECIFIC

	standard		LEU	
U concentration. [w/o]	8.5	12	20	30
weight $UZrH_x$ [g]	2235	2318	2462	2500
U enrichment [w/o]	20	20	20	20
H:Zr	1.60	1.60	1.60	1.60
weight ^{235}U [g]	38 .0	55.6	99.0	150.0
Er concentration [w/o]	-	-	0.44	0.60

3 CALCULATION MODEL

The MCNP computer code [2] was used in the k_{eff} calculations. MCNP is a general-purpose, continuous-energy, generalized-geometry Monte Carlo transport code. The calculations reported in this paper were performed with version 5.1.40 of the code and with three different cross-section libraries, i.e. ENDF/B-VI release 8, ENDF/B-VII and JEFF 3.1. At the time of our computations we did not have the thermal scattering files from the ENDF/B-VII library. First we took them from JEFF 3.1. Later we found a large mistake in JEFF3.1 processed thermal scattering file for H in ZrH [4], therefore we decided to use the scattering data for H in ZrH processed at the IAEA [5]. LEU elements contain also erbium isotopes. As there are no cross sections for erbium in ENDF/B-VI, they were taken from JEFF 3.1.

Simplifications of the geometry were done by simplifying the surroundings of the core such that the k_{eff} was not affected significantly as proposed in [1]. The fuel element was modelled exactly, meaning that Zr rod, stainless steel cladding, air gaps and Mo supporting disc were modelled explicitly. The supporting grid, graphite reflector with rotary groove and central irradiation channel in the core were also explicitly modelled.

First the calculations were performed with the benchmark core, in order to verify the computational model. Results of these calculations are presented in section 4. Further calculations of theoretical mixed cores were performed with the reactor core model, that was very similar to the one used for the benchmark evaluation of the TRIGA mark II reactor [1], the main difference being in the presence of the control rods and the number of fuel elements. In order to observe the effect on k_{eff} arising purely from differences in fuel composition and not from other disturbances (e.g. empty positions, control rods, irradiation channels, etc.) the hypothetical core was made as homogeneous as possible by removing the control rods, empty positions and irradiation channels and replacing them with fuel elements. However central position was (as in most TRIGA reactors) filled with the irradiation channel (i.e. empty aluminium tube). All 90 fuel elements in the core were considered to be fresh. Results of these calculations are presented in section 5.

4 BENCHMARK CORE

Two realistic core benchmark configurations were examined [1], denoted as core 132 and core 133 (see Fig. 1). The experimental k_{eff} values with the uncertainties are presented in Table 2. Systematic errors due to geometry simplifications and uncertainties in the material data of the benchmark model [1] are presented in Table 3. The calculated values of k_{eff} for the benchmark model geometry are given in Table 4.

Table 2. Experimental k_{eff} with the uncertainties, [1]

Case	experimental k_{eff}
Core 132	0.99865 ± 0.00015
Core 133	1.00310 ± 0.00015

Table 3. Benchmark model k_{eff}, uncertainties and systematic errors, [1]

Case	Material error	Geoemetry systematic error
Core 132	± 0.0056	± 0.0019 ± 0.0003
Core 133	± 0.0056	± 0.0015 ± 0.0003

Table 4. Calculated values of k_{eff} using different cross-section libraries.

Cross section set → Case ↓	ENDF/B-VI	ENDF/B-VII	JEFF 3.1
Core 132	0.99830 ± 0.00026	1.00448 ± 0.00026	0.98546 ± 0.00024
Core 133	1.00341 ± 0.00026	1.00939 ± 0.00026	0.99041 ± 0.00026

We can observe that k_{eff} calculated by using the newest ENDF/B-VII cross-section library is systematically higher than the one calculated with ENDF/B-VI by ~ 600 pcm. In order to determine the source of the differences between ENDF/B-VI and ENDF/B-VII cross-section libraries, a systematic analysis of the influence of various isotopes from different cross-section libraries on k_{eff} of the benchmark core 132 was performed. First we calculated the k_{eff} for the benchmark core 132 by using cross sections for all isotopes from ENDF/B VI cross section library. Afterwards cross sections for individual isotopes were taken from ENDF/B VII library. It was observed that the highest contributions are from ^{235}U (~ 220 pcm), Zr (~ 220 pcm) (see Table 5).

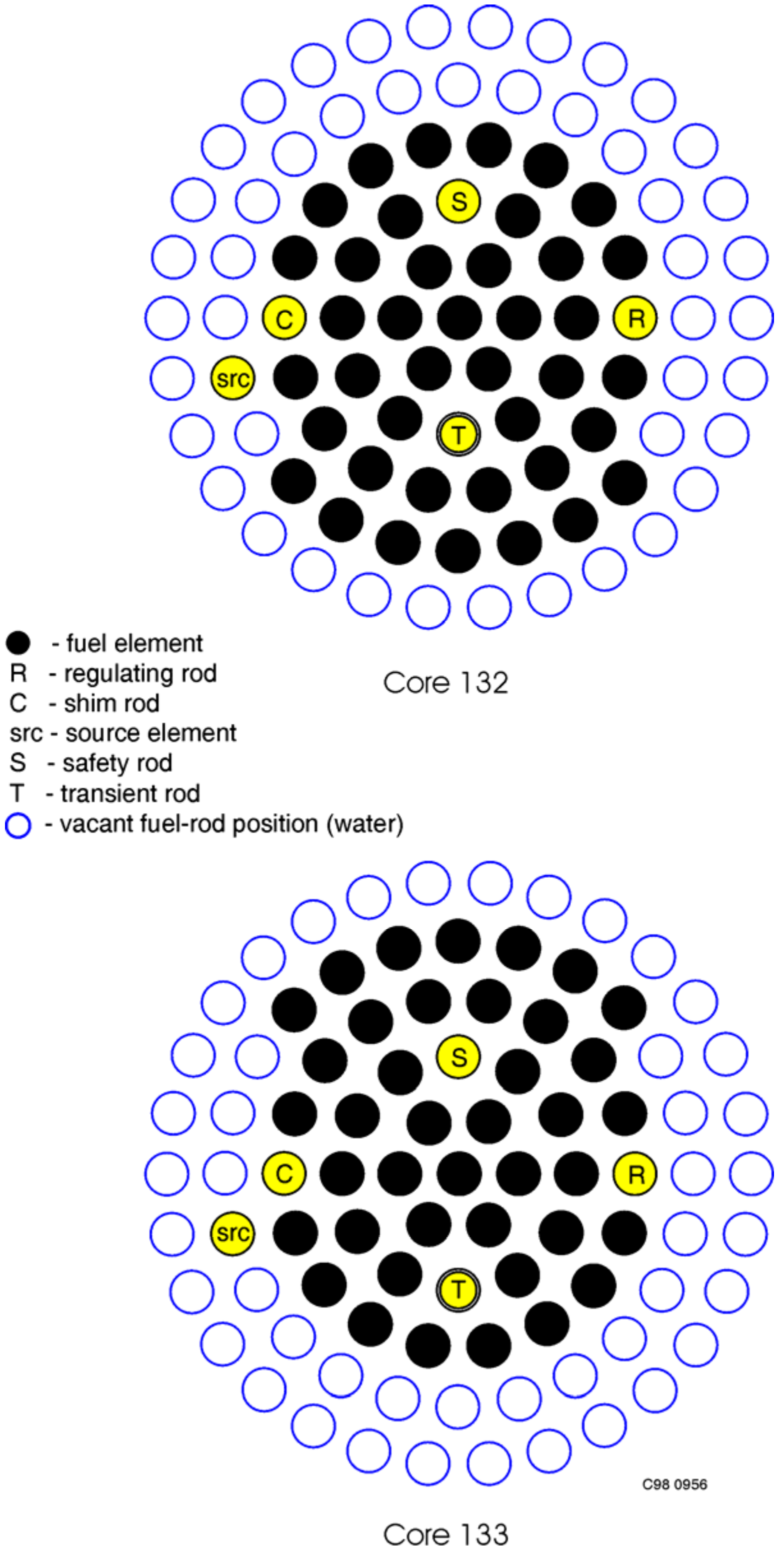

Figure 1. Configurations of the benchmark cores. Top views. [1]

Table 5. Calculated values of k_{eff} for the benchmark core 132 using all isotopes from ENDF/B- VI and some from ENDF/B- VII cross-section libraries. The uncertainties listed in the tables are the statistical errors of the calculations.

Case ↓	k_{eff}
all isotopes from ENDF/B-VI	0.99830 ± 0.00026
^{235}U from ENDF/B-VII	1.00053 ± 0.00027
Zr from ENDF/B-VII	1.00058 ± 0.00028

When JEFF 3.1 cross-section library is used, calculated values of k_{eff} are significantly lower than the ones calculated with both ENDF/B libraries (~ 1300 pcm). In order to determine the source of such large difference among the results, a thorough systematic analysis of the influence of various isotopes from different cross-section libraries on k_{eff} was performed. It was found that the largest difference in k_{eff} originates from H bound in ZrH. A neutron spectrum in the fuel was also calculated and is shown in Fig. 2.

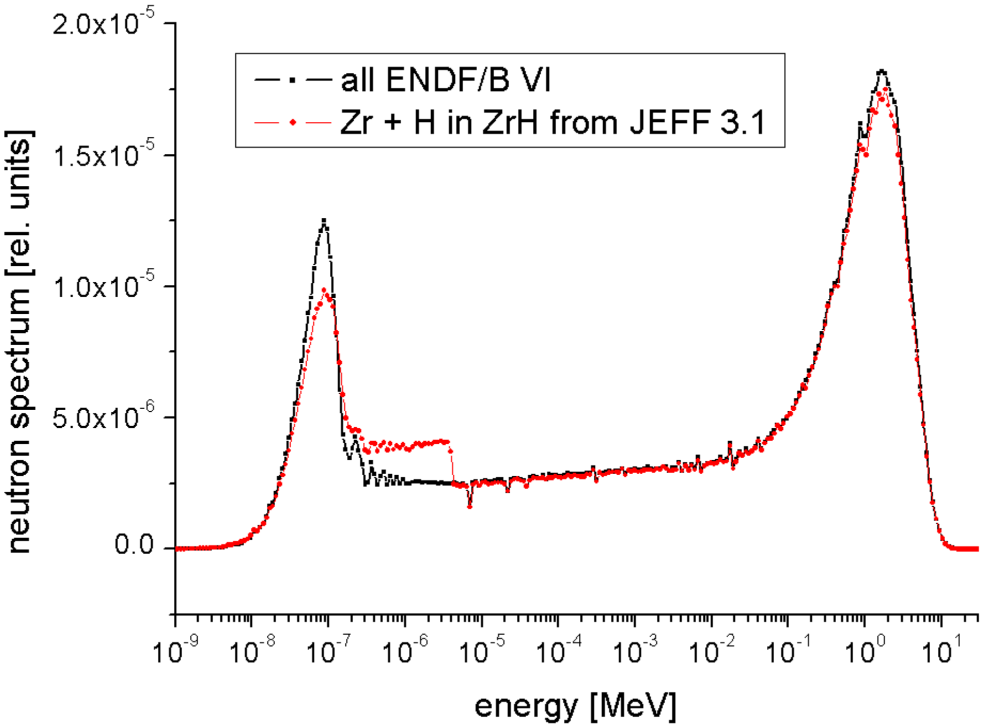

Figure 2. Neutron spectrum in the fuel calculated with MCNP. In the first case (all ENDF/B VI) neutron spectrum was calculated taking all cross sections from ENDF/B VI and in second case Zr and H in ZrH cross sections were taken from JEFF 3.1 [4], other cross sections were from ENDF/B-VI.

A strong anomaly in the thermal energy region of the neutron spectrum is observed when scattering file for H in ZrH from JEFF 3.1 is used. It can be observed that neutron spectrum is unphysically hardened, causing an increase of leakage from the system. As the TRIGA core is relatively small, leakage significantly affects k_{eff}. Checking the library it was found that scattering cross-section for H in ZrH from JEFF 3.1 was exactly two times larger than in ENDF/B-VI. This discrepancy arose from a trivial error in data processing. The error was reported to the NEA Data Bank.

5 MIXED TRIGA CORES

Multiplication factor variations due to different combinations of various TRIGA fuel element types in the same multiplying system are calculated. k_{eff} calculations were performed for several 250 kW TRIGA Mark II reactor core configurations: uniform and mixed core with standard and LEU fuel elements. Four types of fuel elements are considered: standard (8.5 w/o and 12 w/o) and LEU (20 w/o and 30 w/o), all of them 20 % enriched, as presented in Table 1. The results are interesting especially for criticality analysis of storing and shipment operations when fuel elements of low and high uranium content are often mixed in the same multiplying system.

First k_{eff} was calculated for completely uniform core configurations filled with one type of fuel elements (FEs) only. The results are presented in Table 6.

Table 6. Calculated values of k_{eff} for uniform core configurations filled with one type of fuel

Cross section set → Fuel element type ↓	ENDF/B-VI	ENDF/B-VII	JEFF 3.1
8.5 w/o	1.09346 ± 0.00024	1.10015 ± 0.00024	1.08036 ± 0.00034
12 w/o	1.17157 ± 0.00024	1.17766 ± 0.00025	1.15871 ± 0.00037
20 w/o	1.15525 ± 0.00026	1.16092 ± 0.00026	1.13215 ± 0.00038
30 w/o	1.14492 ± 0.00028	1.15041 ± 0.00027	1.12513 ± 0.00038

We observe that k_{eff} calculated using the newest ENDF/B-VII cross-section library is for ~ 600 pcm systematically higher than the one calculated with ENDF/B-VI. We can also see that 12 w/o FEs give the highest value of k_{eff} using all cross section libraries. As already mentioned the k_{eff} values obtained with JEFF 3.1 are significantly lower than the ones obtained with ENDF/B libraries.

Afterwards the completely uniform core configuration with only 8.5 w/o FEs was gradually converted into various mixed cores. Several cases were considered: in the first standard 8.5 w/o FEs in individual rings are replaced by standard 12 w/o FEs, in the second by 20 w/o FEs and in the third with 30 w/o FEs: from one to all elements in the ring. This is done for all rings. The results are presented below.

The trends of k_{eff} are practically the same with all cross section libraries; that is k_{eff} decreases, when 12 w/o FEs are moved towards the core periphery, by ~ 600 pcm. The 12 w/o FEs inserted into the core filled with 8.5 w/o FEs increase the k_{eff} above the uniform core value, however the effect strongly depends on the neutron flux at that position. As the neutron flux decreases towards the core periphery the change of k_{eff} is also decreased. Values of k_{eff} calculated with ENDF/B- VII are for approximately 300 pcm higher than the ones calculated with ENDF/B- VI.

We can see that when 20 w/o and 30 w/o FEs are inserted into the core filled with 8.5 w/o FEs, the trends of k_{eff} are practically the same regardless of the cross section library used. The main difference is only the absolute value of k_{eff}.

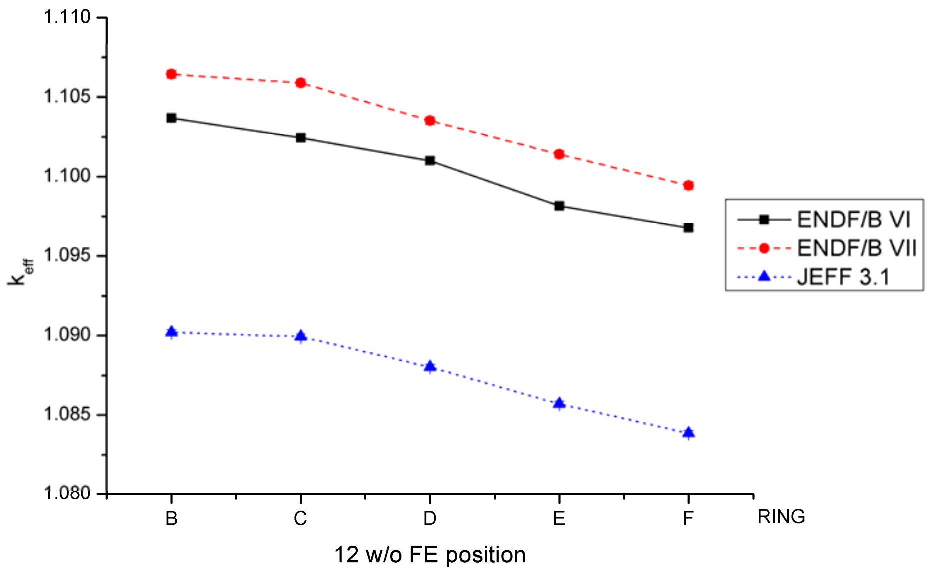

Figure 3. k_{eff} as a function of 12 w/o FEs location in mixed core filled with 8.5 w/o fuel. Note that in each case six 12 w/o FEs are inserted in individual ring.

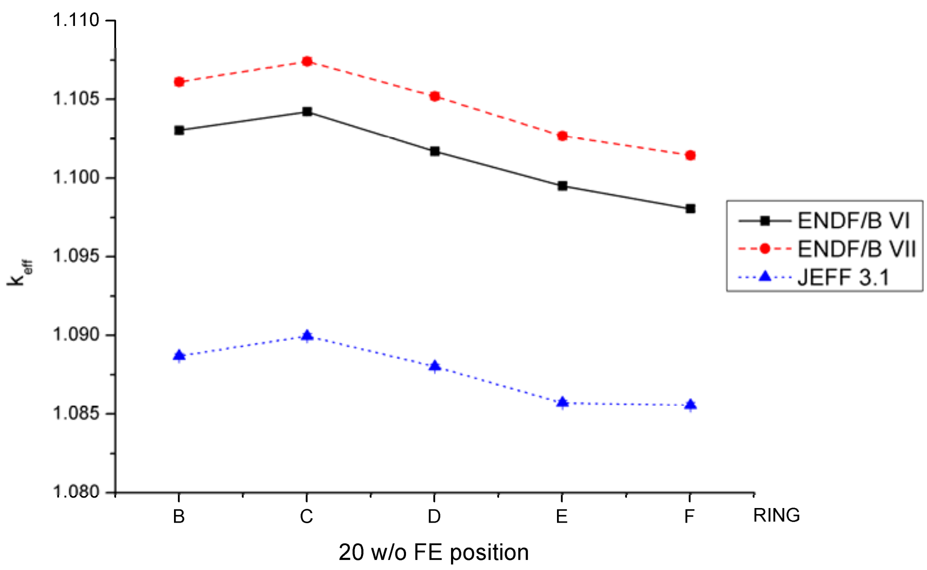

Figure 4. k_{eff} as a function of 20 w/o FEs location in mixed core filled with 8.5 w/o fuel. Note that in each case six 20 w/o FEs are inserted in individual ring.

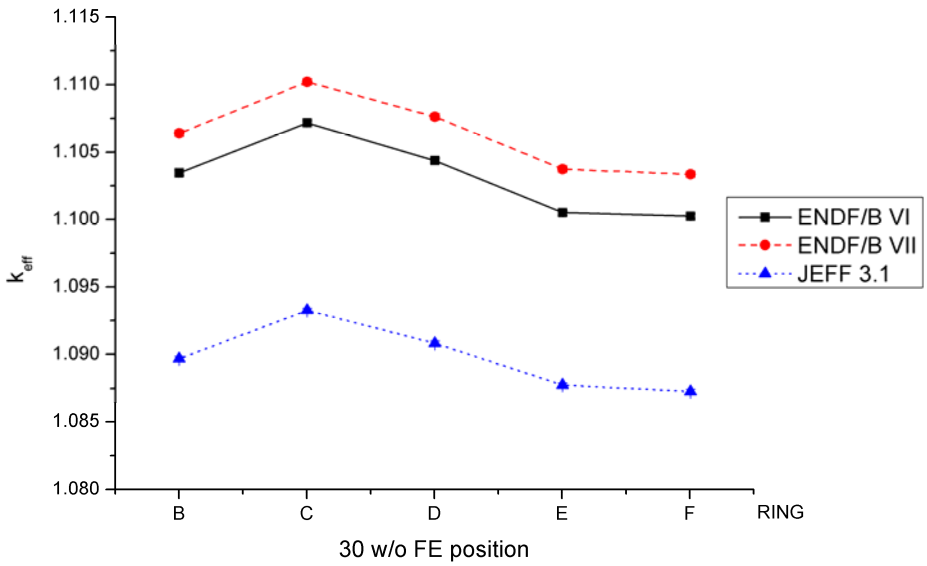

Figure 5. k_{eff} as a function of 30 w/o FEs location in mixed core filled with 8.5 w/o fuel. Note that in each case six 30 w/o FEs are inserted in individual ring.

6 CONCLUSIONS

In TRIGA fuel elements the fuel is homogenously mixed with moderator in form of UZrH causing a large fraction of neutrons to be moderated in the fuel itself. Therefore one should pay special attention to thermal scattering cross sections, especially for H bound in ZrH. As TRIGA fuel is used only in research reactors, smaller attention is paid to the data used in TRIGA fuel calculations. Testing of cross sections libraries for TRIGA fuel is therefore crucial for reliable criticality analysis of TRIGA fuel and consequently safe operation of TRIGA rectors. We have seen that mistake in data processing can have a large influence on neutron spectrum and consequently also on multiplication properties of a multiplying system. We have also observed that k_{eff} calculated by using the ENDF/B VII cross section library is systematically higher than using the ENDF/B-VI cross section library. The main contributions (~220 pcm) are from ^{235}U and Zr.

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