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MCNP Simulation of the TRIGA Mark II Benchmark Experiment

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The complete 3D MCNP model of the TRIGA Mark II reactor is presented. It enables precise calculations of some quantities of interest in a steady-state mode of operation. Calculational results are compared to the experimental results gathered during reactor reconstruction in 1992. Since the operating conditions were well defined at that time, the experimental results can be used as a benchmark. It may be noted that this benchmark is one of very few high enrichment benchmarks available.

In our simulations experimental conditions were thoroughly simulated: fuel elements and control rods were precisely modeled as well as entire core configuration and the vicinity of the core. ENDF/B-VI and ENDF/B-V libraries were used. Partial results of benchmark calculations are presented. Excellent agreement of core criticality, excess reactivity and control rod worths can be observed.

1. Introduction

The main requirement for reliable use of a particle transport computer code is its verification on a benchmark experiment. There are two main objectives of such verification. First is to check the consistency of physical models and data used in a transport code and second is to determine systematical errors made due to approximate simulation of the experiment, which is usually caused by simplified model geometry.

So far benchmark calculations have been performed either on relatively simple geometry or on large and periodic reactor cores, but never on small research reactor example. In simple geometry cases sometimes important effects (due to diverse material composition) can not be fully observed and in large, though complex periodic geometries, separate treatment of some parts can not be completely performed. Therefore, relatively small but complex and non-periodic case of a research reactor geometry seems to be a good choice for benchmark calculations.

The scope of this paper is to present the MCNP simulation of the TRIGA Mark II benchmark experiment as described in (1), (2) and (3). To reduce possible systematical errors due to unexact geometry simulation, a very thorough three-dimensional model of the TRIGA reactor was developed. All possible fresh fuel, control and other elements models were prepared in order to be able to simulate any possible fresh core. If need be, the input can be extended to cover burnup of fuel elements as well. The MCNP input was prepared in such a way that a very quick setup of any desired core configuration with adequate position of all regulating rods is possible. Core vicinity was taken from the model of the whole 3D TRIGA model developed for Boron Neutron Capture Therapy applications (4).

All calculations were not yet performed, but goals of our benchmark calculations are to determine:

1. k_{eff} of two critical experiments,
2. difference between ENDF/B-VI and ENDF/B-V neutron cross-section libraries,
3. excess reactivity of the reactor core,
4. control rod worths,
5. fuel element reactivity worth distribution.

2. MCNP Geometry

As already mentioned in the introduction, the detailed description of the TRIGA benchmark experiment can be found in (1), (2) and (3), so we will focus mainly on the MCNP geometry used in our calculations and eventual deviations from the real geometry. The model of the core will be described in more detail. Core vicinity, which does not affect parameters we intend to calculate, is thoroughly described in (4) so no details will be given in this paper.

Our goal in constructing the MCNP model was to prepare a faithful copy of the real TRIGA geometry. The TRIGA core has cylindrical, although not periodic, configuration with 91 locations in the core (Fig. 1). Fuel and control elements are arranged in six concentric rings. Core is surrounded by aluminum-lined graphite reflector. Exact dimensions - used also in our MCNP model - of the core and its vicinity can be found in (1) and (4). The MCNP model of fuel elements completely corresponds to the real geometry except for top and bottom stainless steel plugs which were modelled as cylinders containing apropriate mass of material (Fig. 2a). To estimate error due to this simplification two separate calculations were performed: in first stainless steel was replaced by water and in second no material was put in corresponding MCNP geometry cells. In both cases we observed only a fractional increase in k_{eff}, so entire deviation due to this simplification was estimated to be 10 pcm and was found completely negligible when compared to the overall statistical deviations of subsequent criticality calculations.

In our calculations - as well as in the experiment - only standard stainless steel clad fuel elements with 20% enrichment were used. A generalization was used in material composition of fuel elements and control rods since we used averaged values rather than specific values for each element. The averaged values of all important materials used in our MCNP calculations were obtained from (1) and (3) and are summarized in Table 1.

Our MCNP model of the core enables simulation of any desired fresh core. It is necessary only to specify what type of element (fuel, control or source) occupies the desired position. The axial position of regulating rods can be specified by setting insertion parameters on MCNP transformation specification records in MCNP input file. Furthermore, our model can be extended to cover burnup parameters of fuel elements as well.

Continuous-energy neutron interaction data from ENDF/B-VI cross-section library were used for our calculations. If data from ENDF/B-VI evaluation were not available, data from other libraries were used, see Table 2.

To investigate consistency and differences between different neutron crossection evaluations, all criticality calculations were performed with ENDF/B-V data file also.

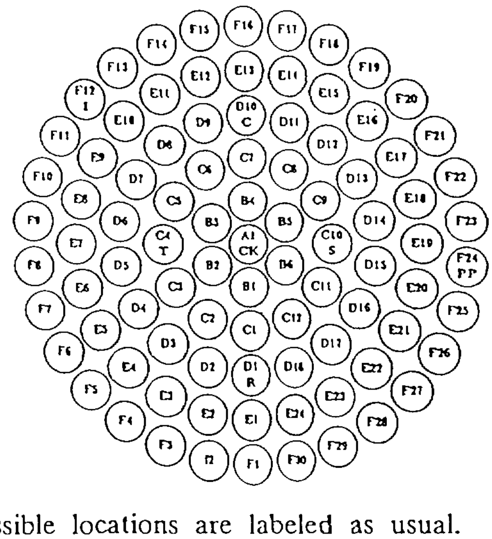

Figure 1: TRIGA core. Possible locations are labeled as usual.

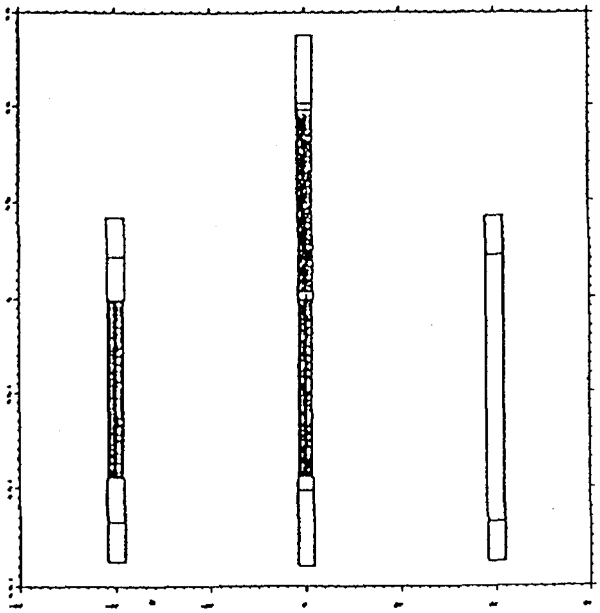

Figure 2: MCNP models of a fuel element (a), a fueled-follower / transient control rod (b) and a source element (c). Side view.

Material	Density [g/cm^3]	Composition	
		Element	Weight. fract.
Fuel (20%,12w/o)	6.122	^{235}U	0.02332
		^{238}U	0.09386
		^{nat}Zr	0.86701
		^{1}H	0.01581
Fuel (20%,8.5w/o)	6.122	^{235}U	0.01690
		^{238}U	0.06801
		^{nat}Zr	0.89863
		^{1}H	0.01646
Zr rod	6.490	^{nat}Zr	1.00000
Graphite reflector	1.670	^{12}C	1.00000
Absorber (B_4C)	2.480	^{10}B	0.13690
		^{11}B	0.64588
		^{12}C	0.21722
SS cladding	7.889	^{nat}Cr	0.19023
		^{nat}Mn	0.02002
		^{nat}Fe	0.69585
		^{nat}Ni	0.09390
Concrete	3.516	^{1}H	0.00383
		^{16}O	0.32833
		^{nat}Mg	0.01848
		^{nat}Si	0.02135
		^{nat}Ca	0.04570
		^{nat}Ti	0.20020
		^{nat}Fe	0.38211

Table 1: Composition of materials used in the MCNP model of TRIGA reactor.

Table	Description
tables from file endf602	
1001.60c	^{1}H from endf-vi.1
5010.60c	^{10}B from endf-vi.1
5011.60c	^{11}B from endf-vi
6000.60c	^{nat}C from endf-vi.1
7014.60c	^{14}N from sp. lanl endf-6 evaluation
8016.60c	^{16}O from endf/b-vi
12000.60c	^{nat}Mg from endf/b-vi
13027.60c	^{27}Al from endf/b-vi
14000.60c	^{nat}Si from endf/b-vi
20000.60c	^{nat}Ca from endf/b-vi
22000.60c	^{nat}Ti from endf/b-vi
25055.60c	^{55}Mn from endf/b-vi
40000.60c	^{nat}Zr from endf-vi.1
92235.60c	^{235}U from lanl proposed endf-vi.2
92238.60c	^{238}U from endf-vi.2

Table	Description
tables from file rmccs2	
24000.50c	^{nat}Cr from njoy
28000.50c	^{nat}Ni from njoy
tables from file endl852	
26000.35c	^{nat}Fe from endl-85
tables from file tmccs2	
lwtr.01t	hydrogen in light water
grph.01t	graphite
h/zr.01t	hydrogen in ZrH
zr/h.01t	zirconium in ZrH

Table 2: Cross-section data used in our calculations.

3. Results

3.1 Critical experiments

The main objective of simulating two critical experiments was to check reliability of our model and to establish a difference between ENDF/B-VI and ENDF/B-V libraries.

We compared our calculation to experimental results of critical experiments described in (1). In the experiment two approximately critical core configurations labeled 132 and 133 were considered. They both had the same number of fuel elements (40) but differed in loading pattern. The core 132 had 7 fuel elements in E ring placed at the side of the transient rod, while the core 133 had them placed at the opposite side, see Figure 3.

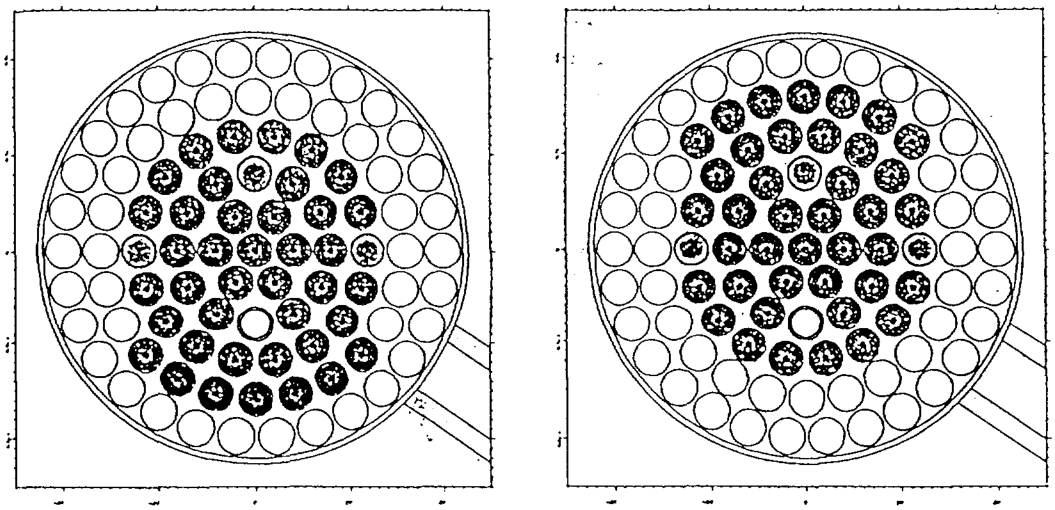

Figure 3: MCNP model of the critical cores labeled 132 and 133. Top view, core midheight.

All three control rods and the transient rod were completely withdrawn. Reactivity of the core was calculated for all different source element locations as quoted in (1) in addition to withdrawn source in core 132. Results of MCNP calculation in Table 3 represent values of the combined collision, absorption and track length k_{eff} estimator. Stable spatial distribution of the fission source was achieved by using 500, 1000 and 2000 particles per cycle in successive MCNP runs for cores without source. In the calculations of different source locations, the initial source distribution was taken from the final source distribution of the core without source element. For the final estimation of the k_{eff} , 4000 cycles of criticality run were computed.

Configuration/ Source Location	Experiment k_{eff}	Estimated $k_{eff} \pm \sigma_{tot}$	MCNP simulation $k_{eff} \pm \sigma_{stat}$
Core 132/Out	not measured	0.99997±0.00032	1.00102±0.00029
Core 132/E12	0.99865	0.99868±0.00032	0.99973±0.00029
Core 133/Out	1.00310	1.00323±0.00032	1.00428±0.00028
Core 133/E12	1.00266	1.00267±0.00032	1.00372±0.00029
Core 133/E2	1.00266	1.00250±0.00032	1.00355±0.00029
Core 133/E7	1.00277	1.00274±0.00032	1.00379±0.00029

Table 3: k_{eff} results for critical cores labeled 132 and 133 with inserted source element on different locations of the core. ENDF/B-VI cross-section library used in MCNP simulation. Experimental results were estimated to have total error of approximately 15 pcm (1).

From the difference between the experimental results and the results of MCNP simulation we estimated systematical error of the MCNP calculation to be

σ_{syst} = +105 ± 13 pcm.

Since all errors due to geometry simplifications fall below statistical errors, we consider this error to be due only to unaccurate neutron cross-section data. So the above oesyst can be attributed to error of ENDF/B-VI data for criticality calculations for TRIGA Mark II reactor. Estimated k_{eff}, obtained from calculated k_{eff} by taking into account the above mentioned systematical error, show perfect agreement with the experiment. Comparison to experimental results ($\sigma_{estimated/experimental}$ = 11 pcm) indicate that estimated experimental error as well as calculated statistical error might be even overestimated.

3.2 Comparison between ENDF/B-VI and ENDF/B-V libraries

Differences between experimental and calculated results using the cross-section data from ENDF/B-V library were much larger, see Table 4. In our opinion this difference is mostly due to better described ^{235}U resonance region in ENDF/B-VI evaluation. For comparison we quote here also results of MCNP simulation of DIMPLE benchmark experiment (5). Again, the ENDF/B-VI library data give better results. Overestimation in TRIGA benchmark calculation and underestimation in DIMPLE benchmark calculation can again be explained with unaccurate treatment of the ^{235}U resonance region. The major difference between both experiments is due to different enrichment: TRIGA reactor fuel is highly enriched (20%), while DIMPLE reactor uses low enrichment fuel (3%).

3.3 Excess reactivity

For calculations of excess reactivity, control rod worths and fuel element reactivity worth distribution, a new core labeled 134 (1) was modeled. It differs from the core 133 in five additional elements in E ring and source location in position E7, see Figure 4.

Experiment	Library errors	
	ENDF/B-VI	ENDF/B-V
TRIGA Benchmark	+105 ± 13 *pcm*	+450 ± 55 *pcm*
DIMPLE Benchmark	−350 *pcm*	−650 *pcm*

Table 4: Comparison between ENDF/B-VI and ENDF/B-V libraries for critical cores.

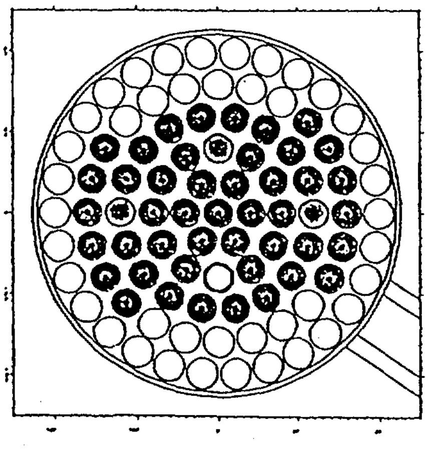

Figure 4: MCNP model of the core 134, used in control rod and fuel element reactivity worth calculations. Top view, core midheight.

Results of the calculated excess reactivity as well as summary of experimental results are given in Table 5. The excess reactivity was determined experimentally from control rod worths critical positions and their calibration curves, measured by the rod-exchange method.

	Experimental ρ_{ex}	MCNP estimate ρ_{ex}
Estimation from control rod critical positions		
R	2144(1±0.10)	2067±100
C	2108(1±0.10)	2185±115
S	1964(1±0.10)	2100±115
T	1872(1±0.10)	2132±115
Average	**2022(1±0.05)**	**2116±55**
Direct calculation		
		2126±32

Table 5: Core excess reactivity for core 134. Experimental results determined from control rod critical positions by the rod-exchange method (2) and calculated results from control rod critical positions. Result of direct excess reactivity calculation result is also given.

3.4 Control rod worth

Control rod worths were calculated as follows: first, core with "all rods out" was modelled and criticality calculation was started, saving the source distribution to a file. After k_{eff} stabilisation and source convergence was reached, the calculation was stopped and source distribution file was preserved.

Now, the selected control rod was moved into the core for a certain length (some 250 pcm in reactivity worth, finer steps were not allowable due to excessive computer times required). k_{eff} calculation was re-initiated and proceeded in the same way as for "bare core". This was repeated until the bottom of the core (control rod completely inserted) was reached.

Table 6 and Figures 5 and 6 give obtained simulation results for regulating and compensating control rods. Again, good agreement between the experiment and MCNP simulation has been observed.

	experiment	MCNP
regulating	2552 pcm	2487 ± 90 pcm
compensating	2430 pcm	2492 ± 90 pcm

Table 6: Comparison of measured and calculated integral worths of regulating and compensating rod.

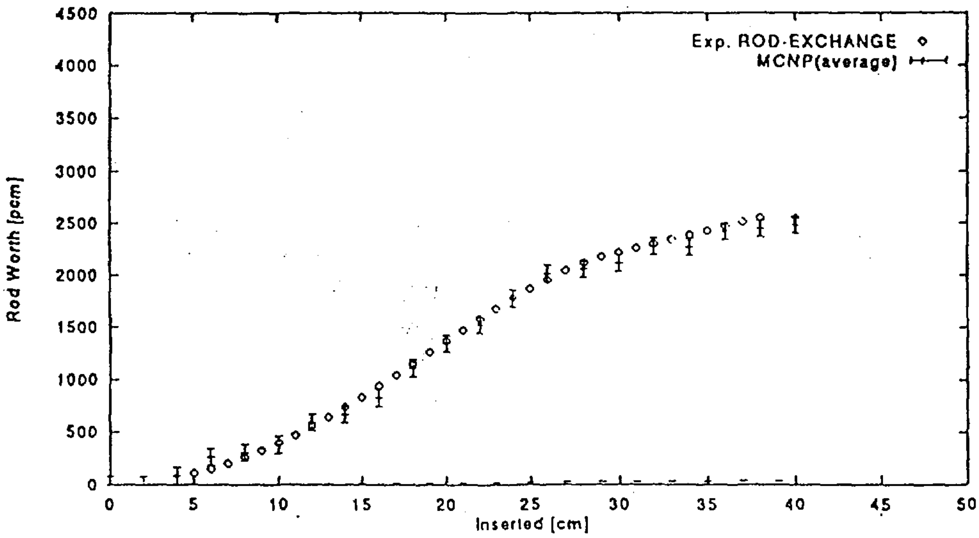

Figure 5: Experimental and calculated integral reactivity curve of the TRIGA regulating rod.

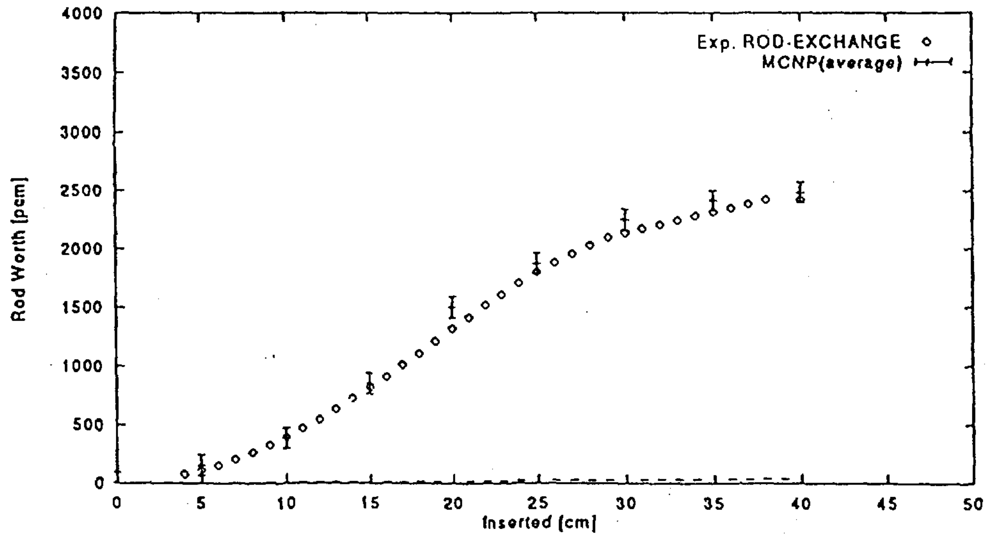

Figure 6: Experimental and calculated integral reactivity curve of the TRIGA compensating rod.

4. Conclusions

The complete 3D MCNP TRIGA Mark II reactor model is presented. ENDF/B-6 neutron crossection library was used for the calculation. Preliminary results of some rector parameters show excellent agreement with the experiments. Further work with this model may lead to a reliable and precise TRIGA benchmark.

5. References

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