A New Monte Carlo Neutron Transport Code at UNIST

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1. Introduction

Monte Carlo neutron transport code named MCS is under development at UNIST for the advanced reactor design and research purpose [1-3]. This MC code can be used for fixed source calculation and criticality calculation. Continuous energy neutron cross section data and multi-group cross section data can be used for the MC calculation.

This paper presents the overview of developed MC code and its calculation results. The calculation capability was tested with famous benchmark problems: International Criticality Safety Benchmark Experimental Problem (ICSBEP), VENUS-2 benchmark, and Hoogenboom-Martin benchmark [4-6]. The real time fixed source calculation ability is also tested in this paper. The calculation results show good agreement with commercial code and experiment.

2. Overview of code

MCS is written in fortran90 that is especially suitable for numerical computation. MCS can simulate 3-dimensional whole core geometry, and it can be easily modeled with lattice function. MCS uses ACE format ENDF-VII continuous energy nuclear data processed by NJOY code. The ACE format data contains continuous energy cross section for the various types of reactions: elastic scattering, inelastic scattering, fission reaction, absorption reaction. Free gas thermal scattering function and S(α,β) thermal scattering reaction for a few nuclides such as hydrogen in water are implemented. MCS estimates criticality by using three types of estimator: collision estimator, track length estimator, and absorption estimator. Fixed source calculation can be performed for both steady state mode and real-time mode. MCS uses collision estimator for the tallies of various parameters such as neutron flux, reaction rate, multi-group micro cross section, multi-group macro cross section, fission counts, and etc. Depletion function and probability table method for the unresolved resonance energy will be implemented.

3. Validation and verification

3.1 ICSBEP Benchmark

The International Criticality Safety Benchmark Experimental Problem (ICSBEP) [4] was used to validate continuous energy MC solver. Fig. 1 show the estimated k_{eff} results, and Table I shows a comparison between experiments and MCS k_{eff} estimate results of 14 selected cases; 7 plutonium base cases, 2 highly enriched uranium base cases, and 7 uranium base cases. MCS uses 100 inactive cycle, 500 active cycle, and 200,000 histories per cycle, and ENDF/B-VII.0 cross section libraries are used.

Table I: k_{eff} of ICSBEP Benchmark

Case	Handbook ID	Benchmark		MCS	
		k_{eff}	uncertainty	k_{eff}	Stdev.
1	PU-MET-FAST-001	1	0.00200	1.00021	0.00007
2	PU-MET-FAST-002	1	0.00200	1.00044	0.00006
3	PU-MET-FAST-022	1	0.00210	0.99859	0.00006
4	PU-SOL-THERM-011, 18-1	1	0.00520	0.99401	0.00009
5	PU-SOL-THERM-011, 18-6	1	0.00520	1.00103	0.00010
6	PU-SOL-THERM-011, 16-5	1	0.00520	1.00889	0.00014
7	PU-SOL-THERM-011, 16-1	1	0.00520	1.00959	0.00014
8	HEU-MET-FAST-028	1	0.00300	1.00304	0.00007
9	U233-MET-FAST-001	1	0.00100	0.99965	0.00006
10	U233-MET-FAST-002	1	0.00100	0.99910	0.00006
11	U233-MET-FAST-003	1	0.00100	0.99884	0.00006
12	U233-MET-FAST-004	1	0.00070	1.00477	0.00007
13	U233-MET-FAST-005	1	0.00300	0.99431	0.00006
14	U233-MET-FAST-006	1	0.00100	0.99968	0.00006

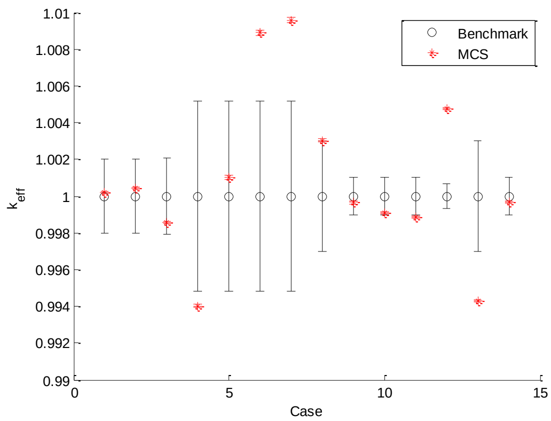

Fig. 1. ICSBEP benchmark results

3.2 VENUS-2 benchmark

The VENUS-2 is widely used MOX benchmark problem for validation [5]. For the verification of MCS, 3 types of pin cell, 2 types of assembly, and 2D whole core were calculated, and it is compared with MCNP result. All calculation was done with ENDF/B-VII.0 libraries.

Table II: k_{eff} of VENUS-2 Benchmark

Benchmark		MCNP	MCS	Diff. (pcm)
VENUS-2 Pin Cell	3.3% UO_2	1.41209	1.41182	27
		0.00008	0.00010	
	4.0% UO_2	1.34236	1.34211	25
		0.00009	0.00011	
	MOX	1.26573	1.26553	20
		0.00009	0.00010	
VENUS-2 Assembly	UOX	1.17862	1.17843	19
		0.00010	0.00010	
	MOX	1.30169	1.30155	14
		0.00009	0.00010	
2D Core		1.08555	1.08526	29
		0.0004	0.00011	

3.3 Hoogenboom-Martin benchmark

Hoogenboom-Martin benchmark is 3-dimensional whole core benchmark problem [6]. MCS calculation result was compared with OpenMC and MCNP results [7]. MCS uses 200 inactive cycle, 450 active cycles, and 100,000 histories per cycle. All calculation was done with ENDF/B-VII.0 libraries. The k_{eff} results of three codes, MCNP, OpenMC, and MCS, matches well.

Table III: k_{eff} of Hoogenboom-Martin Benchmark

Code	k_{eff}	Stdev.
MCNP	1.00023	0.00006
OpenMC	1.00002	0.00006
MCS	1.00090	0.00010

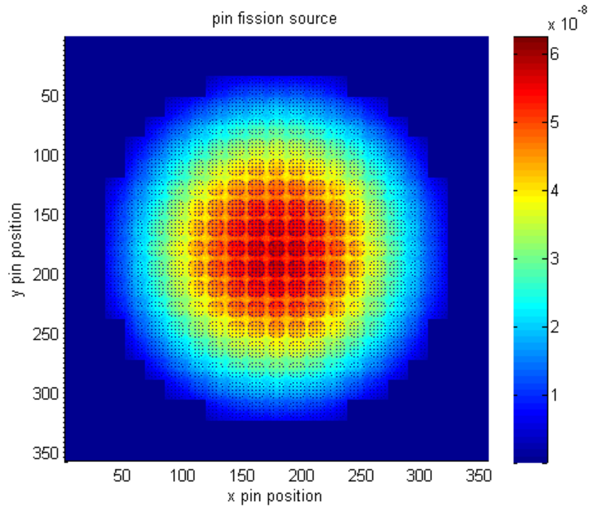

Fig. 2. Fission source distribution of Hoogenboom-Martin benchmark

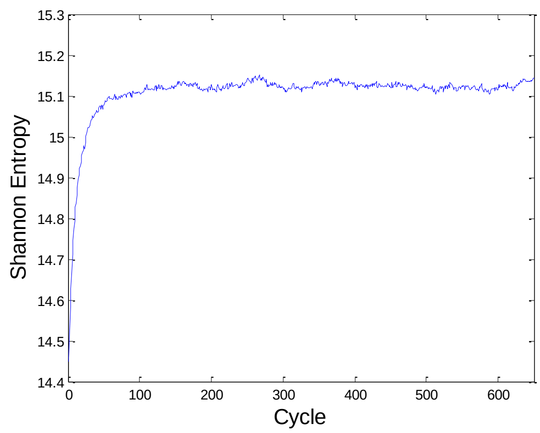

Fig. 3. Shannon entropy of Hoogenboom-Martin benchmark

3.4 Subcriticality measurement

The subcriticality of core can be estimated by using fission counts signal. To test the real-time calculation capability of the MCS, the fission count signals tallied from the MCS were analyzed by the Rossi-alpha that with a modified thermal Godiva problem. Eq. (1) indicates the Rossi-alpha uses auto-correlation of detector counts between adjacent time bins [17], and fig. 4 shows the fission counts signal tallied from the MCS. As shown in fig. 5, Rossi-alpha fitted k_{eff} shows consistent result with k_{eff} from criticality result of MCS

$$P(\tau) = P(k\Delta t) = \frac{1}{N-k}\sum_{i=1}^{N-k} C(i)C(i+k) \quad (1)$$

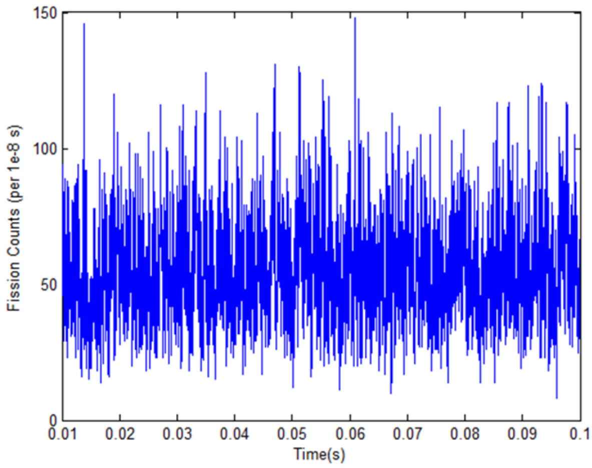

Fig. 4. Fission counts signal

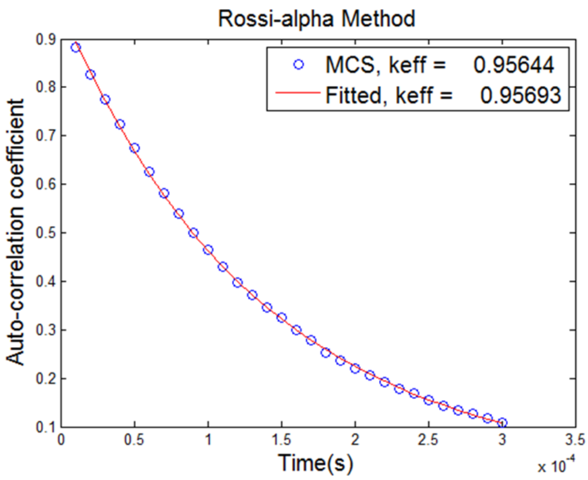

Fig. 5. Rossi-alpha fitting curve

4. Conclusion

A new Monte Carlo neutron transport code is being developed at UNIST. The MC codes are tested with several benchmark problems: ICSBEP, VENUS-2, and Hoogenboom-Martin benchmark. These benchmarks covers pin geometry to 3-dimensional whole core, and results shows good agreement with reference results.

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