MCDP: Development of a code for neutronic calculations for homogeneous reactors

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Introduction

The solution of Neutron Transport equation is incredibly important for any nuclear engineer. And the two fundamental approach for solution are deterministic and stochastic (Monte Carlo methods). There exists a number of computational Monte Carlo particle transport codes which implement Monte Carlo method for criticality safety problems, shielding calculations, dose assessments, burn up calculations.

`MCDP`, a Monte Carlo neutron particle transport code recently developed at PDPU, is a special purpose Monte Carlo D'neutron Particle transport code aimed for the continuous energy ($10^{-5}eV$ to $2 \times 10^{6}eV$) neutronic calculation especially criticality for simple bare geometries. The code is developed in Python programing language. Important features includes variety of initial source, mesh tally for energy wise spatial distribution computations and auto-generated output graphs and files. The validation of the eigenvalue results are performed on some experimental critical systems of MCNP:Neutron Benchmark Problems [1].

Geometry

The available geometries in `MCDP` are spherical, cylindrical, and finite slab with vacuum boundary condition. See Figure(1).

Reactions

ENDF-B/VII.1 pointwise cross section data at $0.0257eV$ is used for various neutron-nuclear reactions such as scattering reactions (n,el) (n,n'), radiative capture (n,γ), neutron producing reactions (n,xn) (x=2,3,4) and fission (n,f). Energy and angular distribution of secondary particles of fission and elastic scattering events are accurately considered. ENDF Law I is used to sample the energy and angular distribution files. The nuclear data (cross sections and distributions) are downloaded from [2].

Neutronic Calculations

The multiplication of the system is estimated using total subsequent method and method of successive generations. The total flux (Integrated to volume) is estimated using collision and track length estimators. The spatial distribution of flux is calculated using collision estimators.[3]

Validation Tests

The validation of results of the code developed for numerical calculations for its practical uses are performed on experimental data (formated as Benchmark Problems). To determine `MCDP`'s ability to calculate multiplication factor of critical configuration, five experimental critical assemblies are analyzed.

1. Lady Godiva 93.71% U-235
2. Jezebel 95.5% Pu-239
3. Jezebel 80% Pu-239
4. Uranium Cylinder 10.9% U-235
5. Uranium Cylinder 14.11% U-235

Godiva and Jezebel are bare spherical critical systems. Each system is modeled in `MCDP` and OpenMC (Open Monte Carlo particle transport code [4]) and the simulation results are compared to the experimental results.

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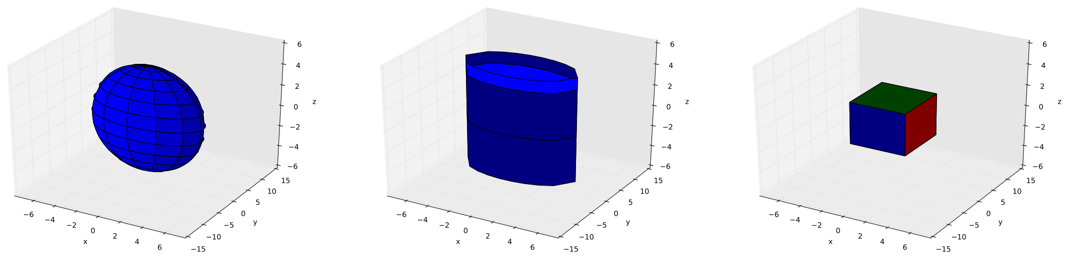

FIG. 1: Generated by MCDP 3D geometry function

TABLE I: MCDP Vs OpenMC for various Neutron Benchmark Tests

Tests No.	True Value	MCNP[1]	N[a]	Gen.[b]	K_{eff} Results of MCDP	OPENMC Results
1	1	0.9976+/-0.0011(%)	1000	500	1.002827 +/- 0.0586(%)	1.00103 +/- 0.145(%)
2	1	0.9986 +/-0.0021(%)	1000	500	1.002503 +/- 0.0540(%)	1.00531 +/- 0.129(%)
3	1	1.0075 +/-0.0012(%)	1000	500	1.002518 +/- 0.0598(%)	1.01097 +/- 0.131(%)
4*	1	1.0024 +/-0.0013(%)	2000	500	0.921623 +/- 0.2272(%)	0.99159 +/- 0.113(%)
5*	1	1.0003 +/-0.0014(%)	1000	500	0.924769 +/- 0.4393(%)	0.99185 +/- 0.153(%)

[a]Number of Initial Particles
[b]Number of Generations
*Systems with uniform distribution of initial source and other systems have point initial source.

Results and Conclusion

From the results of MCDP we conclude following:

For high enriched bare systems, which are smaller in dimensions, the converging result is found for simulation of small number of particles.

For low enriched system, which are large in comparison to high enriched system, the Monte Carlo experiment requires a large number of histories to get converging results.

Monte Carlo simulations depend more on the model of all reactions under consideration.

The multiplication results of MCDP will converge to the true value irrespective of initial spatial distribution of source present in the system.

MCDP can estimate criticality of bare geometries with less error than OpenMC with the same number of particles but time cost is tremendously high in MCDP(of the order of hours) in comparison to OpenMC (in seconds).

References

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