
Group Activity – I

Monte-Carlo Method: Basics of Neutron Transport Stochastic Simulation

Introduction/ Problem Statement

Use Monte Carlo method to perform neutronics calculations for a 2D slab unit cell with periodic boundary conditions

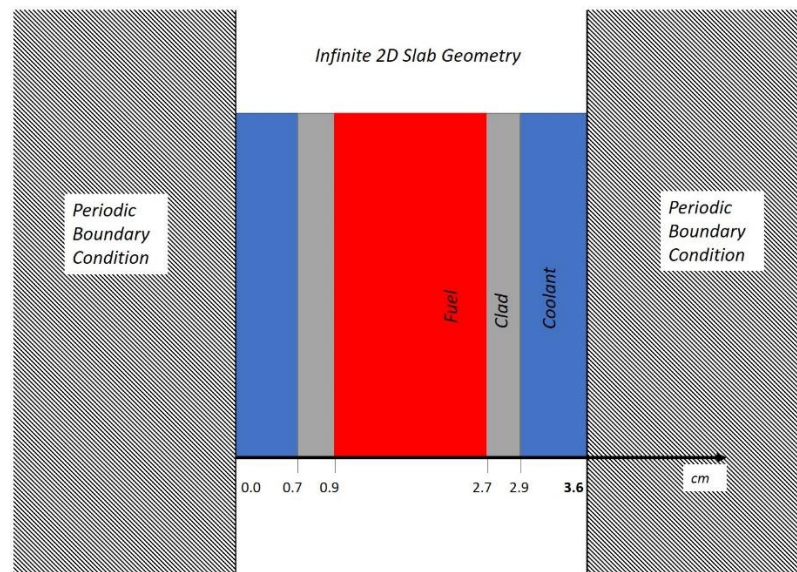


Figure 1 Geometry

Installing and running OCTAVE



For this exercise you will need to install OCTAVE in your computer. However, the exercise can also be done in MATLAB – in case you already have MATLAB, there is no need to install octave.

Windows

Download: <https://www.gnu.org/software/octave/download.html>

Installation instructions : https://wiki.octave.org/Octave_for_Microsoft_Windows

The easiest way is to install it via the executable file

GNU/Linux

sudo apt-get install octave (or any other package installer that you use)

Octave is also available on Ubuntu app store (open app store and search for Octave)

Installation instructions : https://wiki.octave.org/Octave_for_GNU/Linux

Information

The exercise uses the GENDF formatted ADS Nuclear Data Library v2.0, which contains multigroup cross-section data in GENDF format.

<https://www-nds.iaea.org/ads/adsgendf.html>

Following nuclear data files are downloaded, participants are encouraged to download a few files to see the structure:

H_001.GXS	O_016.GXS	B_010.GXS	B_011.GXS
ZR090.GXS	ZR091.GXS	ZR092.GXS	ZR094.GXS
ZR096.GXS	U_235.GXS	U_238.GXS	NA_023.GXS

These files are then converted to MATLAB/OCTAVE format. For every temperature available (293, 600, 900, 1200, 1500, 1800 (depending upon the isotope)), separate MATLAB files are created, which includes:

- atomic weight (amu);
- number of energy groups (=421) and energy group boundaries (eV);
- set of background XSs, sigma-zeros (b), and temperatures (K);
- radiative capture XSs (b) for each sigma-zero;
- (n, α) reaction XSs (b) for each sigma-zero;
- (n,2n) matrix (b) for the first Legendre component;
- scattering matrix (b) for 2 Legendre components (P0 and P1) and each sigma-zero;
- fission cross sections (b) for each sigma-zero;
- nu $\bar{\nu}$ and fission spectrum chi (-);
- total cross sections (b) for each sigma-zero (b).

The microscopic cross-section generated are then used to calculate the Macroscopic cross-sections (with the iterations on sigma-zeros).

$$\Sigma = N\sigma$$
$$\Sigma_{Mix} = \sum_{i=0}^n \sigma_i N_i$$
$$N_i = \rho_i N_A M_i$$

where

N_A = Avagadro's Number

M_i = Molecular weight

Σ = Macroscopic Cross-section

σ = Microscopic Cross-Section

N =Nuclei density

Taking into consideration the constraints of the time, the macroscopic cross-sections have been generated for the required materials. Macroscopic cross-sections are provided for the following in the form of MATLAB/OCTAVE FILES (in the folder named **02.Macro.XS.421g**):

1. Macroscopic cross-section for 3% enriched UO_2 at 900K: *macro421_UO2_03_900K.m*
2. Macroscopic cross-section for 3% enriched UO_2 at 1800K: *macro421_UO2_03_1800K.m*
3. Macroscopic cross-section for 10% enriched UO_2 at 900K: *macro421_UO2_10_900K.m*
4. Macroscopic cross-section for 10% enriched UO_2 at 1800K: *macro421_UO2_10_1800K.m*
5. Macroscopic cross-section for Zircaloy at 600K: *macro421_Zry_600K.m*
6. Macroscopic cross-section for borated water, with boron concentration=4000ppm, water density=0.66238 g/cm³, pressure=16MPa, temperature=600K:
macro421_H2OB_600K_16MPa.m
7. Macroscopic cross-section for very low-density (VOIDED) borated water, with boron concentration=1 ppm, water density=0.00072 g/cm³, pressure=0.2MPa, temperature=600K:
macro421_H2OB_600K_0.2MPa.m
8. Macroscopic cross-section for Sodium at 600 K, with sodium density = 0.8 g/cm³:
macro_Na_600K.m
9. Macroscopic cross-section for very low density (VOIDED) Sodium at 600 K and, with sodium density= 0.001 g/cm³: *macro_Na_600K_VOIDED.m*

Using these nine materials we can construct the following configurations of the slab unit cells:

	Fuel	Clad	Coolant
PWR-1	nominal (mat. 1)	nominal (mat. 5)	nominal (mat. 6)
PWR-2	hot (mat. 2)	nominal (mat. 5)	nominal (mat. 6)
PWR-3	nominal (mat. 1)	nominal (mat. 5)	voided (mat. 7)
PWR-4	hot (mat. 2)	nominal (mat. 5)	voided (mat. 7)
SFR-1	nominal (mat. 3)	nominal (mat. 5)	nominal (mat. 8)
SFR-2	hot (mat. 4)	nominal (mat. 5)	nominal (mat. 8)
SFR-3	nominal (mat. 3)	nominal (mat. 5)	voided (mat. 9)
SFR-4	hot (mat. 4)	nominal (mat. 5)	voided (mat. 9)

Group Exercises

Task 0:

All groups will perform Task 0.

The MATLAB/OCTAVE code *monteCarloPWR.m* is also provided. It will simulate the neutron transport in 2D unit cell using Monte Carlo method. Please open the code and understand the structure. This will also be explained in the lecture.

Note: Please make sure that the path to the macroscopic cross-section is correct in Line 38. Also, if you are using MATLAB, please change the backward slash ('\') to forward slash (/)

```
%-----  
% Path to macroscopic cross section data:  
path(path, '..\02.Macro.XS.421g');
```

0.1 Use the code to calculate the k_{eff} for the PWR type slab reactor under isotropic source approximation using Monte Carlo Method.

0.2 Add detectors to score the thermal, resonance and fast neutron scattering rate shapes along the x axis with the step of 0.1 cm. Calculate and plot the thermal, resonance and fast neutron flux shapes along the x axis.

Group 1:

1.1 Analyse and compare the results between PWR and SFR cases by performing task 0.1 and 0.2 for the SFR slab unit cell.

Group 2:

2.1 Analyse the void effect in a SFR slab unit cell. Does it depend on fuel temperature?

Group 3:

3.1 Analyse the doppler effect with nominal and voided coolant condition.

Group 4:

4.1 Analyse the void effect in a PWR unit cell. Does it depend on fuel temperature?

Additional cross-group tasks:

Compare Doppler effects in SFR and PWR unit cells

Compare void effects in SFR and PWR unit cells