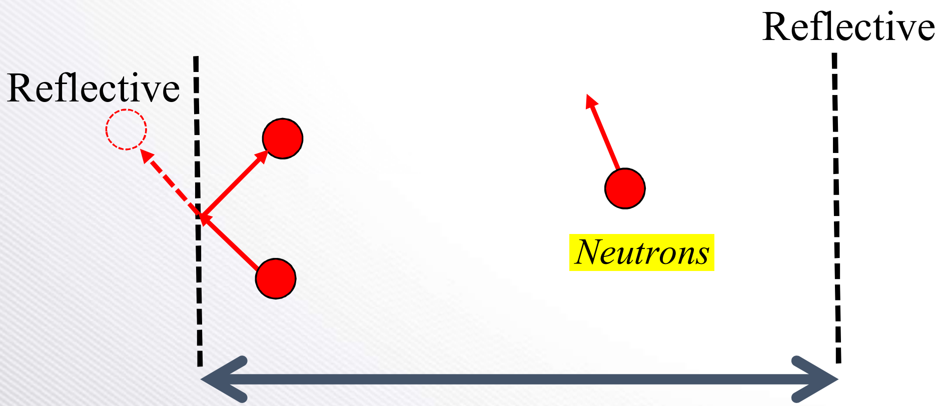
01 Homework #4

» MC Eigenvalue Transport Problem

- Neutron particles are generated by the fission reaction on an infinite homogeneous slab of thickness D=2.0 cm, which is characterized by macroscopic scattering and absorption cross sections. The absorption reaction is divided by the capture and fission reactions.
- Assume that scattering is isotropic in the laboratory system and that our idealized neutrons scatter without loss or gain of energy. The left and right surfaces of the slab have reflective boundary condition (mirror).



Reaction	Notation	Value
Scattering	Σ_s	1.5
Fission	Σ_f	0.3
Capture	Σ_c	0.2
Total	Σ_t	2.0
Nu	ν	2.0

01 Homework #4

» MC Eigenvalue Transport Problem

- Make a MC eigenvalue code for the primitive transport problem .
  - Determine the multiplication factor (k)
  - Applied below schemes :
    ① Stochastic uncertainty estimation
    ② Scaling for MC eigenvalue calculation
    ③ Flux estimation in 10 sub-region (length of sub region = 0.2cm)

- The report should include results, code design, and discussion.
- Upload the report, executable and source code files at the class homepage (e-campus).

- Due date – Dec 15, 2025.