TEAM MEMBERS

Arkajit Dutta (Graduate Student)

SELECTED SOFTWARE

OpenMC (Open-source Monte Carlo particle transport simulation)

PRESENTATION TITLE

Accelerating Monte Carlo Neutron Transport Simulations Using OpenMC and GPU Parallelization

PROBLEM STATEMENT SUMMARY

Simulate the random transport of neutrons through a material, tracking their movement, interactions, and final state (absorbed, scattered, or fissioned).

High Level Flow of Particle Simulated –

1) The particle (tracking a single neurons journey) starts at a source location.
2) It moves through a geometry made up of different materials.
3) It follows probabilistic behavior dictated by cross-section data.
4) It undergoes collisions that cause scattering, absorption, or fission.
5) Final result: Compute neutron flux, reaction rates, and criticality calculations (The ratio of neutrons produced to neutrons lost, which determines if a reactor is subcritical, critical, or supercritical