GPU Acceleration of OpenMC Neutronics for Fusion Applications

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A key engineering challenge in the design of future magnetic-confinement fusion power plants is the necessity to balance shielding requirements against the need to maximise the tritium breeding ratio for sustainable fuel production, and heat production for power generation. To enable a rapid cycle of parameter space exploration for both geometry and material assignments, and yet retain the spatial accuracy typically attributed to Monte Carlo particle transport, it is of paramount importance to maximise the computational performance of these calculations. Fortunately, the particle transport portion of these algorithms is well-suited to massive parallelization, with acceleration upon GPU of the OpenMC code recently demonstrated using the portable programming paradigm, OpenMP Target Offload [1, 2].

In previous work [3], scaling across multiple nodes (on hundreds of GPUs), and performance portability across GPUs from different vendors has been successfully achieved in k-eigenvalue computations of fission-reactor models. For fusion applications, however, the computational load has different characteristics. In particular, the initial neutrons must be efficiently sampled from a known source distribution. Additionally, while fission reactor problems often feature lattice-based geometries whose regularity can be leveraged to accelerate ray tracing routines, fusion reactor problems often feature less regular geometry that make efficient ray tracing more difficult. In this contribution, we address these challenges, and present the first demonstration of OpenMC portable GPU performance in fixed-source mode, applied to a simplified tokamak geometry. We report on scalability and speed-up relative to CPU, and finally discuss the future outlook.

Keywords: Monte Carlo, Neutronics, GPU, Portable-perfomance

References

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[2] `https://github.com/exasmr/openmc`.

[3] Work in preparation.