Massively parallel deterministic neutron transport solver for fusion multiphysics applications

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Neutron damage of fusion reactor materials and tritium self-sufficiency are two major challenges that need to be solved to turn fusion into a viable energy source. Improving our understanding and predictions of these phenomena is of key importance for the development of future fusion reactors. It requires highly demanding simulations of the neutron transport processes and generation of tritium, which involves many technical disciplines. At present, it is tackled with many computational codes such as the Monte Carlo code MCNP [1] for neutron transport, computational fluid dynamics and thermofluid codes for first wall surface temperatures and engineering analysis software for stresses and deformation. Because of the complex geometry of the fusion reactor system, these codes need to be run in three dimensions with a true geometric representation to achieve high-quality predictions. Maintaining consistency in the geometric representation among the codes is challenging. Using the output from a code as an input to a different one currently involves a lot of human effort and computing time. Moreover, it is a potential source of error.

This paper presents a new deterministic neutron transport code (NEUTRO) dedicated to solving the Boltzmann Transport Equation [2] developed as one of the physics modules in the massively parallel finite element general software package Alya [3]. The purpose of the development is to enable HPC simulations of multiphysics phenomena involving neutron transport in a fusion reactor by directly coupling NEUTRO with existing Alya modules at the time-step level. NEUTRO uses the Discrete Ordinates Method on angular coordinates, multi-group for energy discretization and the finite element method (FEM) on unstructured meshes to treat special complex domains. Total cross-sections and the respective group matrix for the elastic cross-section of each material are obtained using the NJOY code [4]. The solver is tested using different geometries, orders of integration for the angular discretization and number of energy groups. Detailed comparisons of the code's results have been carried out against benchmarks obtained from a Nuclear Energy Agency (NEA) database that reported measurements of leakage spectra of several materials. The MCNP package is used to validate those results. Special cases of complex materials simulated with effective cross-sections are presented in this work.

References

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