Validating NEUTRO: a deterministic neutron transport solver for fusion applications

E. Goldberg[1], M. Circuns I Duxans[1], A. Soba[1,2], M. J. Mantsinen[1,3]

[1] *Barcelona Supercomputing Center, Barcelona, Spain*

[2] *CONICET, Buenos Aires, Argentina*

[3] *ICREA, Barcelona, Spain*

Neutron reactions on fusion reactor materials are key phenomena to be understood to enable fusion as a feasible energy source for future reactors. We present significant improvements and new capabilities of NEUTRO, a deterministic neutron transport code dedicated to solving the Boltzmann transport equation. The purpose of the development of this module is to enable HPC simulations of multiphysics phenomena involving neutron transport in a fusion reactor by directly coupling NEUTRO with other existing modules at the time-step level within Alya, a massively parallel finite element software package [1]. The solver uses the Discrete Ordinates Method to discretize neutron flight directions, the multigroup method to discretize the energy spectrum and the Finite Element Method (FEM) over unstructured meshes for spatial discretization [2]. We use the NJOY code [3] to calculate the total cross-sections and the respective scattering group matrices of the isotopes considered in each case, from databases in ENDF (evaluated nuclear data file) format.

To consolidate the validation of the code, we have performed a series of classic benchmarks from the literature, testing different aspects of the numerical tools involved in the solver. Special cases of complex materials simulated with effective cross-sections have also been simulated. The solver has been tested using different geometries and materials with varying levels of scattering properties (Ref.1) Results have been compared for classic tests, benchmarks obtained from a Nuclear Energy Agency database, and test cases using other neutron transport codes [4]. We present results for flux distribution in varying domains with different materials, leakage spectra and nuclear heating distribution through the domains. Additionally, we share details regarding our experience with the computational aspect of our simulations in high-performance environments.

References

[1] M. Vazquez, et. al. Alya: towards exascale for engineering simulation codes. arXiv: 1404.4881 [physics.comp-ph], (2014).

[2] A Soba et. al. Fusion Engineering and Design, Vol. 169, (2021).

[3] Kahler III, A. C., Macfarlane, R. NJOY2016. Computer software Aug. (2016).

[4] C.J. Werner, et al., "MCNP6.2 Release Notes", Los Alamos NL, LA-UR-18-20808 (2018)

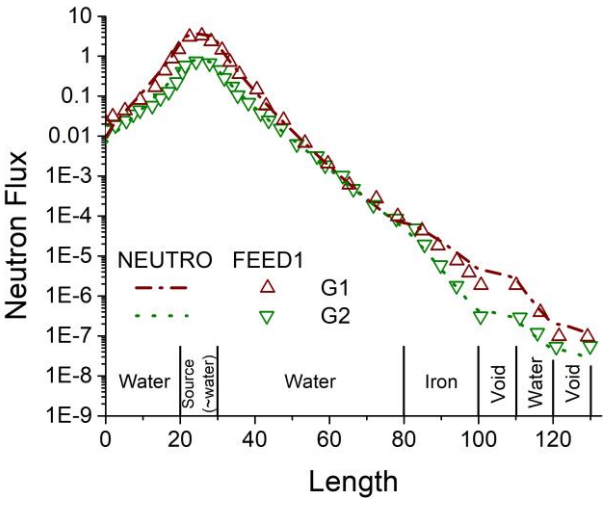

Figure 1 Neutron flux results for two energy groups of a shielding test comparing NEUTRO and the FEED1 code.