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Doctoral Dissertation

Doctoral Program in Energy Engineering (38th cycle)

Neutronics modeling for fusion reactors

CAD-based workflows, validation, and structural material assessment

By

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Declaration

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Abstract

Accurate modelling of neutron transport is a key aspect in the design and assessment of fusion reactors. This is particularly true for compact concepts, where high neutron fluxes and limited shielding space impose stringent constraints on both materials and system performance.

Over the past decades, significant progress has been achieved in the development of Monte Carlo codes and nuclear data libraries for fusion applications. However, the validation of these tools in complex and realistic geometries remains a critical challenge, particularly when moving from simplified benchmark configurations to reactor-relevant models.

In this context, the capability to reliably simulate neutron interactions in heterogeneous materials and intricate geometrical configurations becomes essential to support design choices and evaluate reactor feasibility. The present work aims to contribute to this effort by improving the state of the art in fusion neutronics validation methodologies.

Monte Carlo simulations performed with OpenMC are employed to investigate neutronic behaviour in fusion-relevant conditions, with particular emphasis on the integration of CAD-based geometries in the frame of neutron transport workflows. The capability of the adopted modelling approach to accurately predict neutron leakage spectra and reproduce experimental benchmark results is first assessed through the Oktavian pulsed sphere experiment, where calculated spectra are compared against experimental data and reference MCNP simulations, showing good overall agreement. The comparison between Constructive Solid Geometry (CSG) and CAD models is performed to assess the impact of geometric representation on the neutronic analysis. This study aims to verify that the transition from idealized CSG descriptions to realistic engineering geometries does not introduce significant

discrepancies in the results, thereby supporting the adoption of advanced geometry workflows for high-fidelity simulations.

The analysis is then extended to an ARC-class reactor model, where a detailed comparison between OpenMC and PHITS (Particle and Heavy Ion Transport code System) is carried out. The two codes provide consistent predictions of neutron flux, energy spectra, and power deposition, with differences mainly attributable to nuclear data and modelling assumptions. The use of reduced geometries is also investigated, showing that sector models can accurately reproduce the neutronic behaviour of the full system while significantly reducing computational cost.

Neutron spectra obtained from OpenMC simulations are subsequently employed as input for activation analyses using FISPACT-II, enabling the evaluation of irradiation-induced effects in structural materials. The comparison between the two approaches confirms the consistency of the methodology and highlights the importance of accurate spectral information for reliable inventory calculations.

Finally, the impact of neutron irradiation on candidate structural materials is assessed, with particular attention to activation, radiation damage, and waste management. Conventional alloys such as Inconel-718 exhibit significant limitations due to long-lived activation products, emphasizing the need for careful material selection in compact fusion systems.

Overall, the integration of CAD-based geometry modelling, Monte Carlo neutron transport, and activation analysis provides a robust framework for neutronic assessment of complex fusion reactor configurations, supporting the development of next-generation fusion energy systems.