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EXECUTIVE SUMMARY OF THE THESIS

System level study of Fusion-Fission hybrid reactor core components

TESI MAGISTRALE IN NUCLEAR ENGINEERING

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1. Introduction

The growing global demand for sustainable and low-carbon energy sources has renewed interest in nuclear fusion as a long-term solution for large-scale electricity production. Among the proposed configurations, Fusion–Fission Hybrid Reactors (FFHRs) represent an intermediate and potentially more accessible step toward practical fusion energy systems. In this context, the Reversed Field Pinch (RFP) configuration offers attractive features such as high plasma beta, compact geometry, and reduced magnetic field requirements compared to tokamaks.

However, one of the major engineering challenges associated with FFHR systems is the protection of superconducting toroidal field coils from intense neutron irradiation. Fast neutrons generated by fusion reactions can penetrate structural components, causing displacement damage, material activation, and degradation of superconducting properties. An accurate assessment of neutron transport and shielding performance is therefore essential for the reliable design of RFP-based hybrid reactors. While high-fidelity Monte Carlo codes such as OpenMC provide detailed solutions, the development of simplified yet physically transparent transport tools remain valuable for rapid parametric studies and for gaining deeper physical insight into neutron–material interactions.

The present work addresses this need through the development and validation of a custom Monte Carlo neutron transport model aimed at supporting shielding analysis for fusion hybrid systems.

2. Objectives

The main objective of this thesis is the development of a computational framework capable of evaluating neutron attenuation and energy redistribution in candidate shielding materials for RFP-based fusion–fission hybrid reactors.

The specific goals of the work are:

- to analyse the radiation sensitivity of superconducting materials relevant to toroidal field coils;
- to evaluate the neutronic performance of plasma-facing and shielding materials;
- to develop a modular Monte Carlo neutron transport code based on ENDF nuclear data;
- to validate the implemented model against the reference code OpenMC;
- to apply the validated tool to simplified reactor geometries in order to assess shielding effectiveness.

The final aim is to provide a physically consistent and computationally flexible methodology for preliminary shielding design in hybrid fusion systems.

3. Section and subsection

A preliminary analysis was conducted to identify the main radiation damage mechanisms affecting superconducting magnets in fusion environments. Particular attention was devoted to NbTi and Nb3Sn conductors, whose performance degradation is primarily driven by displacement damage induced by fast neutrons.
The damage process was interpreted in terms of primary knock-on atom (PKA) generation and subsequent defect production. The role of the displacement energy threshold and the NRT model was discussed to clarify the dependence of damage accumulation on neutron energy spectrum. This analysis highlighted the importance of reducing fast neutron fluence at the coil location.
The radiation damage capability of a neutron spectrum can be quantified through the displacement energy cross section

$$\langle \sigma T \rangle = \frac{\int \sigma(E)\, T(E)\, \frac{d\phi}{dE}\, dE}{\int \frac{d\phi}{dE}\, dE}$$

from which the damage energy per atom is obtained as

$$E_D = \langle \sigma T \rangle\, \phi t$$

where $\sigma(E)$is the energy-dependent scattering cross section, $T(E)$is the primary recoil energy, $\phi$is the neutron flux, and $t$is the irradiation time. These metrics provide a quantitative link between neutron spectrum characteristics and the expected degradation of superconducting performance.
In parallel, candidate plasma-facing and shielding materials were reviewed. Light materials such as graphite and beryllium exhibit favourable thermal shock resistance and neutron moderation capability, while high-Z materials such as tungsten provide efficient gamma attenuation and structural robustness but limited moderating power. Boron-containing materials were identified as effective neutron absorbers due to the large ^{10}B capture cross section.
This comparative framework guided the subsequent numerical investigations.

4. Montecarlo transport model

The core contribution of this thesis is the development of a custom Monte Carlo neutron transport code implemented in Python. The model was designed to explicitly reproduce the stochastic nature of neutron interactions while maintaining a modular and transparent structure suitable for parametric studies.

4.1. Equations

In the Monte Carlo framework, neutron transport is governed by the macroscopic total cross section, defined as

$$\Sigma_t(E) = \sum_i N_i\, \sigma_i(E)$$

where N_i is the atomic density of isotope $i$and $\sigma_i(E)$ is the corresponding microscopic cross section. The neutron free path is sampled from the exponential attenuation law

$$s = -\frac{\ln \xi}{\Sigma_t(E)}$$

where $\xi$is a uniformly distributed random number in the interval (0,1]. This formulation ensures an unbiased stochastic representation of neutron flight distances within the medium.

The transport algorithm samples neutron free paths from the total macroscopic cross section and determines interaction types according to partial reaction probabilities. The following processes are included:

- elastic scattering;
- inelastic scattering;
- absorption reactions;
- particle-production channels.

Elastic scattering kinematics are treated using two-body collision theory, with angular distributions derived from ENDF-formatted nuclear data. Inelastic events are modelled through energy-dependent cross sections and simplified secondary energy treatments.

4.2. Scattering and nuclear data

Elastic scattering events are treated using two-body collision kinematics. For a target nucleus of mass ratio A, the outgoing neutron energy in the model is given by

$$E' = E\frac{A^2 + 1 + 2A\mu}{(A+1)^2}$$

where E is the incident neutron energy and μ is the cosine of the scattering angle in the center-of-mass frame. This relation correctly captures the limited energy loss per collision for heavy nuclei such as tungsten.
Energy-dependent microscopic cross sections were imported from ENDF-6 formatted libraries and processed into interpolated tables suitable for continuous-energy sampling. Special care was devoted to the consistent treatment of elastic and anelastic reaction channels to preserve the physical meaning of the total cross section.

4.3. Geometry and tallies

The initial validation was performed in slab geometry, allowing direct comparison with analytical attenuation behaviour and with OpenMC results. Particle histories were classified into transmission, reflection, and absorption events, and energy spectra of transmitted neutrons were recorded to evaluate spectral evolution through the material.
Following validation, the model was applied to simplified multilayer configurations representative of RFP hybrid reactor shielding. The simulations highlighted several important physical trends.
Light materials such as graphite exhibited strong elastic scattering and significant neutron reflection, acting primarily as moderators rather than absorbers. Tungsten, due to its large atomic mass, showed limited energy loss per elastic collision but strong attenuation through inelastic channels. Boron-containing materials demonstrated effective neutron removal driven by capture reactions.
The results confirmed that effective shielding requires a combination of moderation, absorption, and high-Z attenuation layers.

5. Model Validation

A structured validation campaign was carried out by comparing the custom Monte Carlo results with equivalent simulations performed using OpenMC.

5.1 Slab benchmarks

Benchmark calculations were performed for several materials, including graphite, tungsten, and boron carbide, over a range of slab thicknesses. The comparison focused on:

- transmission probability;
- reflection fraction;
- absorption rate;
- transmitted energy spectra.

The developed code reproduced the expected exponential attenuation trends and correctly captured the dominant interaction mechanisms for both low-Z and high-Z materials.

5.2 Statistical Reliability

The percentage relative standard deviation (PRSD) of key tallies was monitored as a function of slab thickness. For most configurations the statistical uncertainty remained below 2%, confirming the numerical stability of the Monte Carlo implementation.
Overall, the agreement with OpenMC was found to be satisfactory for the intended level of modelling fidelity, validating the physical consistency of the developed transport approach.

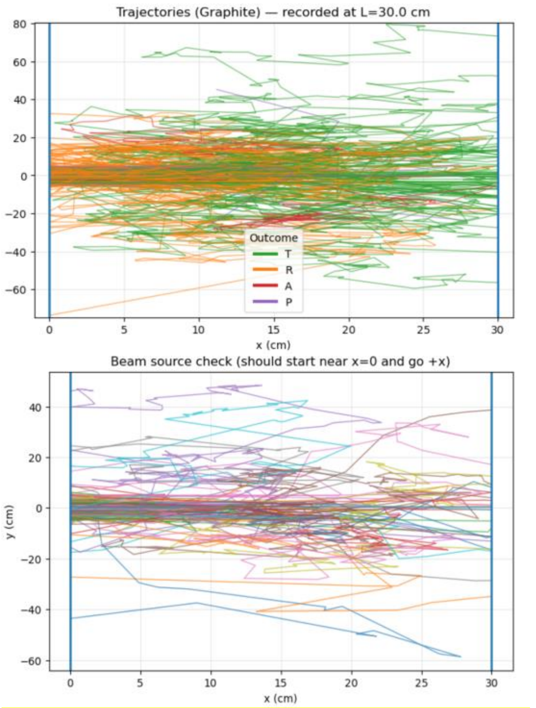

Fig 5.1 Simulated trajectories of 14.1 MeV neutrons through a 30cm thick graphite slab with Neutron transport code (a) and OpenMC(b)

6. Application to Reactor-Relevant Configurations

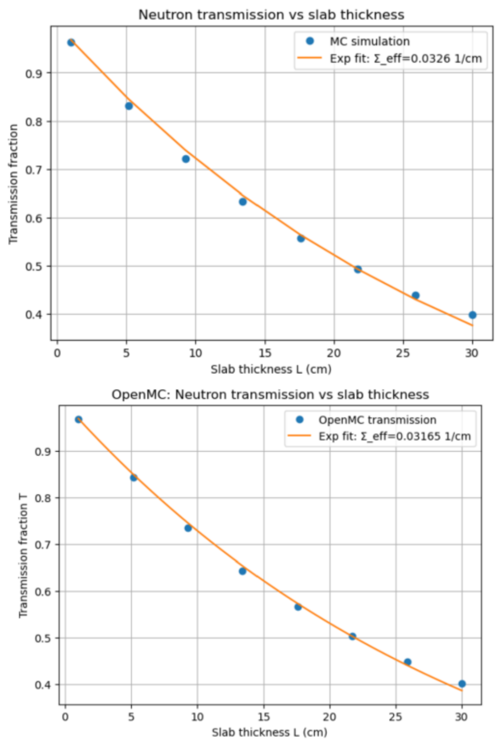

Fig 5.2 Exponential attenuation of 14.1 MeV neutrons in Graphite obtained by the neutron transport code compared with the OpenMC result

Following validation, the developed Monte Carlo model was applied to simplified multilayer configurations representative of shielding layouts in RFP-based fusion–fission hybrid reactors. The objective of this analysis was to translate the previously observed material-level behaviour into design-relevant indications for the protection of superconducting toroidal field coils.

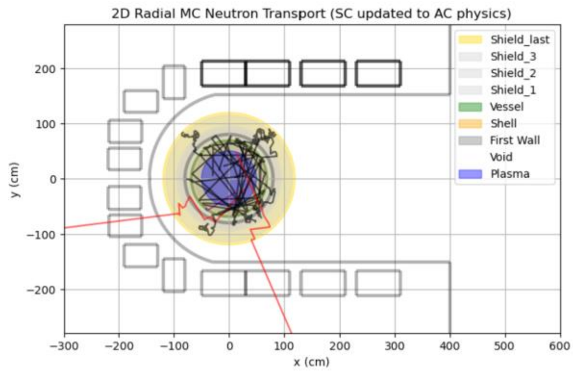

Fig. X Simulated trajectories of ten neutron through the RFX shielding configuration.

The simulations confirmed that different material classes play complementary roles in the attenuation of fusion neutrons. Light materials such as graphite exhibited strong elastic scattering and significant neutron reflection, acting primarily as moderators capable of reducing neutron energy through successive collisions. In contrast, tungsten showed limited energy loss per elastic interaction due to its high atomic mass but provided effective attenuation through inelastic scattering and high-density stopping power. Boron-containing materials demonstrated strong neutron removal driven by capture reactions, confirming their effectiveness as absorptive layers.

From a shielding design perspective, the results indicate that no single material is sufficient to achieve the required reduction of fast-neutron fluence at the superconducting boundary. Instead, an optimized multilayer architecture is required, in which neutron moderation in light materials precedes absorption in boron-rich regions and final attenuation in high-Z structural components. This sequential energy degradation mechanism is essential to efficiently suppress the fast-neutron component responsible for displacement damage in Nb_3Sn conductors.

The analysis further suggests that the relative positioning and thickness of shielding layers have a non-negligible impact on overall performance, particularly in the compact geometry typical of

RFP configurations. These findings support the adoption of integrated shielding strategies that simultaneously address spectral softening, neutron capture, and structural constraints.
Overall, the application study demonstrates that the developed transport tool is capable of providing physically meaningful guidance for early-stage shielding design in fusion hybrid systems, bridging detailed neutron–material interaction modelling with reactor-level engineering requirements.

7. Conclusions

This thesis presented the development and validation of a custom Monte Carlo neutron transport tool aimed at supporting shielding analysis for Reversed Field Pinch fusion–fission hybrid reactors. The study demonstrated that radiation damage in superconducting materials is strongly driven by the fast-neutron component of the spectrum, reinforcing the need for carefully optimized shielding solutions. The comparative assessment of candidate materials clarified the distinct and complementary roles of moderators, absorbers, and high-Z structural materials. The implemented Monte Carlo code successfully reproduces neutron attenuation, angular redistribution, and spectral evolution in representative shielding media. Validation against OpenMC confirmed the physical reliability of the model for preliminary design studies. The results obtained for multilayer configurations indicate that hybrid shielding strategies combining light moderators with strong absorbers provide the most effective reduction of fast-neutron fluence in RFP-based systems. Future work may focus on extending the model to full three-dimensional geometries, improving secondary particle treatment, and coupling the transport solver with detailed radiation damage and activation modules. Overall, the developed framework represents a useful and flexible tool for early stage neutronic assessment of advanced fusion hybrid reactors.

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