POLITECNICO DI TORINO

Master's Degree in Sustainable Nuclear Energy

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Neutronic analysis on different configuration of ARC-like tokamaks

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Abstract

The research field for fusion energy is developing rapidly and in addition to the well known mega project (e.g. ITER, DEMO), compact fusion tokamaks are under investigation. One interesting compact reactor is ARC (Affordable, Robust, Compact), a fusion device under development by Commonwealth Fusion Systems, designed for efficient energy production and material testing, which also involves research activities in Italian universities and industry. The presented work is developed in this framework and focuses on a neutronic analysis of an ARC-class fusion reactor. Monte Carlo simulations are carried out by means of OpenMC tool, to estimate Tritium Breeding Ratio (TBR), power generation, and neutron spectrum. First, a CAD-based model of an ARC-class reactor is built, incorporating key components such as the vacuum vessel, the breeding blanket, and the cooling systems. Simulations explore neutron behavior across these components, analyzing how variations in first-wall thickness impacts neutron transport and the efficiency fo the whole system. The study concludes with an uncertainty propagation analysis in which the cross sections of the isotopes are modified to assess the impact on the simulation results of their uncertainties. This evaluation aims to optimize design parameters contributing to the ARC reactor's role as both an energy source and a testing facility for advanced materials.

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Chapter 1

Introduction

With the growing demand for energy and the urgent need to reduce carbon emissions, nuclear fusion has become a focal point of scientific and technological research. However, achieving functional fusion reactors requires innovative solutions to overcome the technical challenges associated with the immense energy needed to initiate and sustain controlled fusion reactions. Globally, fusion reactor projects have sought to balance size, cost, and efficiency. Among these projects, the ARC reactor represents a significant breakthrough, combining a compact design with advanced technologies to improve both the efficiency and practicality of fusion for energy production.
Nuclear fusion is a process where two atomic nuclei collide and merge to form a new nucleus, releasing a large amount of energy due to a slight reduction in mass. For this reaction to occur, two light nuclei must overcome the repulsive Coulomb force with sufficient kinetic energy. Currently, the Deuterium-Tritium (D-T) fusion reaction is the most practical choice due to its relatively low energy threshold. Deuterium, a stable hydrogen isotope, can be sourced from water, while Tritium, an unstable isotope, is rare in nature but can be produced using lithium and a neutron source. In fusion reactors, Tritium production is achieved through lithium blankets that capture neutrons from the reactor's own fusion process. This production is measured by the Tritium Breeding Ratio (TBR). Historically, fusion reactors have been designed for large-scale energy output (around 1000 MWe). However, high costs and extended construction times have led to the development of a more compact, efficient design called the ARC (Affordable, Robust, Compact) reactor, that produce 525 MW of fusion power targeting a 200 MWe output. ARC's design employs advanced REBCO superconducting magnets and a modular structure that allows for flexibility in experimental research and testing of innovative materials. This modular approach allows components to be readily replaced if failures occur, supporting continuous experimentation without compromising the reactor's plasma production. ARC's design combines its function as both a power generation plant

and a nuclear science facility, utilizing high-field superconducting magnets to achieve the necessary plasma confinement and stability within a compact structure[1]. A model of an ARC reactor is represented in Figure 1.1. The reactor's replaceable vacuum vessel is made of Inconel 718, a nickel-based alloy chosen for its strength and heat resistance, though it may face activation challenges in the reactor environment. Surrounding the vessel, a blanket of FLiBe (a molten salt) serves as a coolant, tritium breeder, and neutron moderator. Neutrons from D-T fusion reactions are absorbed in the FLiBe blanket, where they transfer energy and produce tritium fuel, which is then extracted for continued reactor operation. The FLiBe-filled blanket tank functions as a stable, long-term component that also acts as the primary containment boundary, ensuring reactor safety and facilitating efficient tritium production and energy capture.

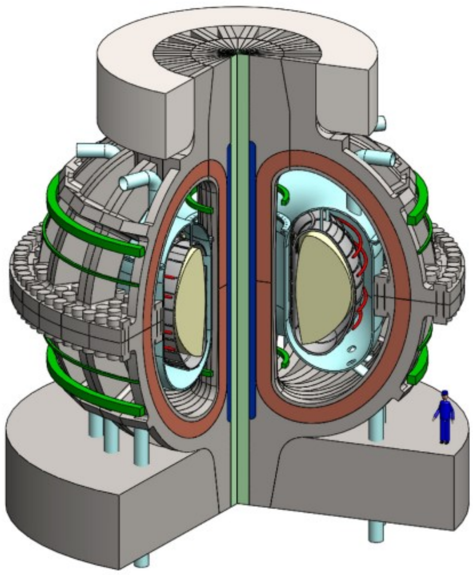

Figure 1.1: ARC Reactor Design with the plasma in yellow and the TF superconducting tape in brown

The aim of this work is to develop a neutronic analysis on a fusion reactor specifically an ARC-like tokamak using the open source code OpenMC after building the geometry and meshing it using Coreform Cubit. This study will concentrate on some characteristics of the neutron transport problem in a fusion reactor such as the Tritium breeding ratio, power densities and neutronic spectra. The model is divided in different volumes one of these is the first wall and a comparison between models with different thickness of tungsten as the first wall will be at the heart of the work.
In chapter two of the thesis the neutron transport problem is analyzed with a

focus on the parameter considered in the study and on the tools used for the calculation, in the third chapter a brief description of the ARC reactor is given also with the characterization of the geometry used, then in chapter four the results of the simulation with the comparison are developed while in chapter five a study of uncertainties propagation of specific parameter using the SANDY tool is done and finally in the last chapter the conclusions and some remarks on the future are listed.

Chapter 2

Neutron Transport Problem

The fusion neutronic transport problem is a challenge in the design and operation of fusion reactors. Neutronic transport deals with the behavior and interaction of neutrons produced by fusion reactions within the reactor environment. These interactions are critical to understand how neutrons impact key reactor functions, such as power generation, fuel breeding, and the integrity of materials used in the reactor. Fusion reactors, particularly those utilizing deuterium-tritium (D-T) fuel, produce high-energy neutrons that present unique challenges, as well as opportunities for optimizing reactor performance. Fusion neutronic transport modeling is essential for optimizing the reactor's design and ensuring it meets key performance criteria, such as the tritium breeding ratio (TBR), power generation efficiency, and neutron spectra management. These parameters must be carefully balanced to achieve a sustainable and economically viable fusion power plant. This chapter explores these aspects, addressing the significance of each in the broader context of fusion reactor design and performance and presents the method and the tools which have been used to obtain and analyze them.

2.1 Tritium breeding ratio

The Tritium breeding ratio is a key parameter in fusion reactors using D-T fuel. Tritium, unlike deuterium, is not naturally abundant and must be generated within the reactor through neutron interactions with lithium in the reactor's blanket. The TBR is the ratio of tritium atoms bred to those consumed in the fusion reaction. For a reactor to operate sustainably, the TBR must exceed 1.0, ensuring that enough tritium is produced to fuel the reactor and compensate for any losses due to leakage or decay. Breeding blankets, typically composed of materials that contain Lithium (in this model FLiBe), are designed to capture high-energy fusion neutrons and generate tritium through nuclear reactions. Optimizing the TBR involves not only selecting appropriate blanket materials but also understanding the transport of neutrons within the blanket to maximize tritium production while minimizing neutron losses. Achieving a sufficient TBR is one of the key challenges in fusion reactor design, as it directly impacts the reactor's fuel sustainability and overall viability. This requires advanced neutronic simulations and careful integration of the blanket with the reactor's overall architecture.

2.2 Power generation

In fusion reactors, power generation relies on capturing the kinetic energy of high-energy neutrons produced in the fusion process. These neutrons carry approximately 80% of the total energy released in D-T fusion reactions, with the remaining 20% being carried by charged particles. The kinetic energy of neutrons is converted into thermal energy when they interact with materials in the reactor's blanket or structural components. This thermal energy is then used to heat a working fluid, typically water or a gas, which drives a turbine connected to an electrical generator, similar to conventional thermal power plants. Neutronic simulations are used to estimate the total energy deposited in the various layers of the reactor due to nuclear interactions, providing a detailed understanding of the heat distribution and helping to optimize the design for efficient power extraction.The efficiency of power generation in fusion reactors is highly dependent on the ability to effectively capture and transfer the energy from neutrons to the working fluid while minimizing energy losses due to radiation, leakage, or absorption by non-breeding materials. Accurate modeling of neutron transport is critical to optimizing power generation. The spatial distribution of neutron flux, energy deposition, and heat transfer within the reactor must be carefully analyzed to maximize efficiency. Furthermore, the choice of materials for the reactor's first wall and blanket plays a significant role in determining the amount of energy that can be harnessed for electricity production.

2.3 Neutron spectra

Neutron spectra refers to the energy distribution, or spectrum, of neutrons within a fusion reactor, and how these neutrons interact with various materials. In fusion reactors, the neutron spectrum is much broader and more energetic than in fission reactors, particularly in D-T fusion where most neutrons are born with an energy of approximately 14.1 MeV. The energy spectrum plays a critical role in defining how neutrons affect power generation, material integrity, and fuel breeding. The interaction of neutrons with materials depends on their energy. High-energy neutrons, such as those produced in D-T fusion, are more likely to cause displacement damage to structural materials, resulting in phenomena like embrittlement and swelling. Lower-energy neutrons, which result from scattering or moderation, are more suitable for breeding reactions with materials like lithium, which are designed to capture neutrons and produce tritium. Optimizing the neutron spectra in a fusion reactor involves managing neutron interactions across a wide range of energies. By tailoring the material composition of the reactor's blanket, first wall, and other components, it is possible to influence how neutrons are moderated and absorbed, thus optimizing both power generation and fuel breeding. Neutronic simulations are essential for predicting the energy distribution of neutrons throughout the reactor, enabling engineers to assess the impact of the spectrum on reactor performance. These simulations inform decisions on material selection, blanket design, and reactor layout to ensure that the neutron spectrum is harnessed effectively. In addition, managing the spectrum neutronic is crucial for mitigating the damage caused by high-energy neutrons, which can degrade materials over time and limit the reactor's operational lifespan.

2.4 Monte Carlo method

For the purpose of this work the Monte Carlo method is used to perform neutronic analysis in an ARC-like reactor. The vast majority of energy output in reactor concepts like ARC comes in the form of neutrons, making the field of neutronic analysis critical to these concepts' design. The Monte Carlo method is a numerical statistical method used for the evaluation of the average value statistical quantities. These quantities are represented as expected values of random variable distributions which has to be defined consistently with the requested results. In this method complex systems are simulated knowing their basic phenomena. Once these basic phenomena are identified, a random variable of use for the quantity to be evaluated is defined and the system behavior is simulated numerically obtaining an estimation of the mean value with its statistical uncertainties. Monte Carlo codes are used in different fields of application such as physics, risk analysis, business and many others.

Neutronic analyses provide critical inputs to almost all other subsystems in the reactor in the form of volumetric heating rates, material damage, and activation. However, neutronic analyses of fusion reactors have three difficulties: the high sensitivity of neutronic to the reactor geometry, the relative scarcity of fusion neutronic analysts to support industry goals, and the challenges faced by users of neutronics software, particularly with respect to access to software and source code, and ease of integration with modern computer-aided design (CAD)[2]. The Monte Carlo code used in this thesis is OpenMC.

2.5 OpenMC

OpenMC[3] is a community-developed MC code proposed by the computational reactor physics group of the Massachusetts Institute of Technology (MIT). It is written in C++, and it has a user-friendly Python API that make use of common Python packages for pre-processing and post-processing. All version control of OpenMC and its documentation are handled through the git-distributed revision control system, and the code is available on GitHub. It implements neutron and photon transport, being particularly devoted to fission reactor applications, like k-eigenvalue or sub-critical multiplication calculations. It supports both Computational Solid Geometry (CSG) and CAD imported geometries, relying for the latter on the Direct Accelerated Geometry Monte Carlo (DAGMC[4]) software, parallelization and both continuous and multigroup transport. The continuous-energy particle interaction data is based on the HDF5 format that can be generated from ENDF files produced by NJOY. For this thesis, ENDF/B-VIII.0[5] has been chosen as the HDF5 library for cross-section data. This library provides the most up to date and accurate evaluated nuclear data, including detailed neutron interaction cross-sections, which are essential for simulating neutron transport and energy deposition in the reactor. The use of ENDF/B-VIII.0 ensures that the simulation reflects the latest nuclear data standards, particularly for fusion reactor applications, where neutron interactions play a crucial role in energy production and material behavior. OpenMC has been applied to many different designs, like PWR, IAEA benchmark Material Test Reactor, MSR and tokamaks[6]. It was also validated against MCNP on an ARC-class tokamak[6].

To set up an OpenMC simulation elements that govern the geometry, materials, physics and output settings must be defined. First of all the materials must be defined specifying their atomic composition and their density, then the geometry, in this case it is inputted as a DAGMC file and then boundaries are initialized: a spherical surface with radius bigger than the geometry that acts as leakage boundary initialized as "vacuum boundary" that means neutrons hitting this surface will escape the system, it functions as a graveyard; two reflective surfaces on the sides of the model to simulate the 360° model with a 10° domain; and another leakage region in a form of a cylinder in the center of the geometry. After the geometry the fusion neutron source has been introduced. This source emits neutrons from a cylindrical surface with a fixed radius and with a uniform angular distribution in the 10° angle of the domain, the emission is isotropic and the energy distribution follows a Muir spectrum that is typical for D-T reactions. The settings are then defined, they determine the physical properties of the simulation, such as the neutron source, the type of calculation (in this case fixed-source), and the number of particles and batches to be run. Finally the quantities that must be calculated are defined as tallies.

2.6 SANDY

SANDY[7], SAmpler of Nuclear Data and uncertaintY, is a sampling tool that can read, write and perform operations on nuclear data files in ENDF-6 format. The primary objective of the code is to produce perturbed nuclear data files containing sampled parameters that represent the information stored in the evaluated nuclear data covariances. Such files can be used to propagate uncertainties through any compatible system. Even though SANDY can be used to propagate different types of nuclear data for the mean of this work only cross sections were studied.
The uncertainty propagation process is carried out using a Fast Monte Carlo (Fast MC) method. This approach is based on the principle of running the same model multiple times, each time with a different set of sampled parameters, drawn from the covariance matrices that characterize the uncertainty in the nuclear data.
The Fast MC method offers several key advantages over traditional Monte Carlo techniques:

- **Optimized sampling**: With respect to traditional MC that uses purely random sampling in the Fast MC some optimizing techniques are used such as importance and adaptive sampling;

- **Faster convergence rate**: Fast MC implements variance reduction techniques that permit the use of fewer samples than regular MC to have similar accuracies;

In the work the SANDY code is launched by giving as input the nuclide of which the cross section must be perturbed the library used that is ENDFB-VIII.0 and the number of perturbation that the code has to produce called samples, for this work it is imposed to 500 samples. So the code with this input information will produce 500 libraries ENDFB-VIII.0 with the cross section of a selected nuclide differently perturbed in each of them.
Once obtained the perturbed library a simple geometry has been built since the focus of this uncertainty propagation is the perturbation on a single tally calculated as function of the different cross sections used and ideally the perturbation on the results is the same independently of the geometries. This simple geometry is a sphere with three regions where the inner and outer most volumes are vacuum and the material is present only in the intermediate one. The chosen tally for this study is the TBR and since the isotope that influence this value the most is Li6 this nuclide has been chosen as input. After the simulation is complete it will produce a statepoint file with the 500 different values of TBR.

Chapter 3

CAD based model

3.1 Geometry and simulation setup

Neutronic simulations are performed on a model of an ARC-class tokamak vacuum vessel. The model is composed by 8 areas: the plasma chamber is modelled as vacuum simulated by very rarefied Hydrogen while the other seven layers are described in table 3.1

Component	**Material**	**Thickness [mm]**
First Wall (FW)	Tungsten	1
Inner vacuum vessel	Inconel 718	10
Cooling channel	FLiBe	15
Neutron multiplier	Beryllium	10
Outer vacuum vessel	Inconel 718	30
Breeding blanket	FLiBe	999.6
Tank wall	Inconel 718	30

Table 3.1: Subdivision of the model

A 3D CAD model has been built with a 10° domain in SolidWorks® using the extrude function in the program on the constructed sketch (Fig. 3.1).

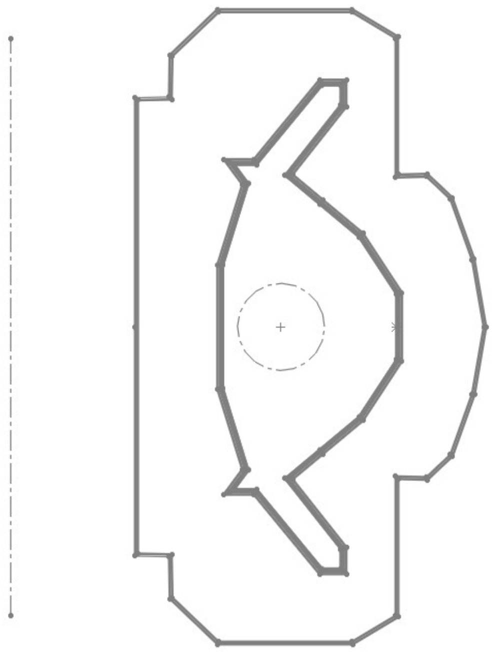

Figure 3.1: Sketch built in SolidWorks®

The model is built with a 10° domain since it has been demonstrated that this simplification using reflective boundaries conditions on the sides simulate a 360° geometry in an accurate way[6].

Simulations were conducted to compare the performance of the same model with two different configurations for the first wall: one with a thickness of 1 mm and the other with a thickness of 1 cm (Figure 3.2), while the other thicknesses remain the same as table 3.1.

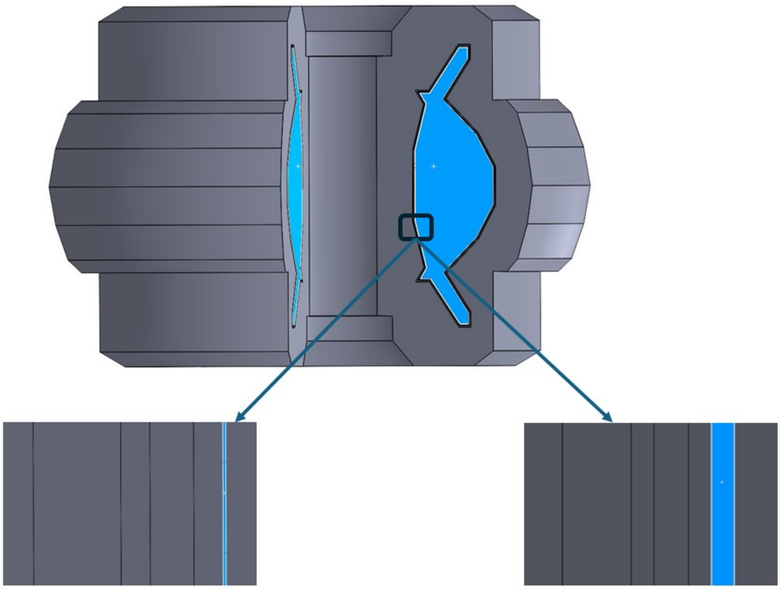

Figure 3.2: Three dimensional representation of the ARC reactor with focus on the two first wall used highlighted in blue in the details: 1 mm (left) 1 cm (right)

The sections are then extruded and exported in the meshing program one by one.

3.2 Meshing setup

The program used for meshing is Coreform Cubit[8] this has been used also for converting the geometry in the DAGMC format file[4]. This format maintains the accuracy needed for physics calculations while optimizing performance by using a triangular mesh for this reason this geometry format is integrated in various Monte Carlo codes including openMC. Furthermore the material assignment in the DAGMC geometry can reference the xml file produced in the openMC simulations via the materials name or ID[9]. This software has been used also for imprinting, merging and faceting the geometry. Faceting is the process of converting continuous geometry (such as curves and surfaces) into a series of discrete flat polygons (often called facets) that approximate the original geometry. This process is essential in computational simulations because computers generally handle complex shapes by breaking them down into simpler, flat surfaces for easier manipulation. Faceted geometry allows simulations to be performed efficiently, as the computer can handle flat surfaces more easily than curved ones, but since it is an approximation, the original smooth geometry is never perfectly captured, the accuracy depends on the resolution of the facets. During faceting the commands imprinting and merging play an important role in ensuring that the geometric model is well-prepared for meshing and are essential for ensuring high-quality, continuous meshes.

- The imprinting command is used to ensure that two or more surfaces, edges, or geometric entities share the same nodes along their intersections. This is crucial to ensure that meshes generated on different objects are consistent with each other where they intersect or touch, imprinting does not change the shape of the objects but adds geometric curves or points on the boundaries of the intersecting surfaces, ensuring that the meshes are aligned along these intersections. Since it divides the geometry along the intersections, it ensure that the faceted mesh will align properly where different surfaces or volumes meet.

- The merging command is used to combine nodes, edges, surfaces, or volumes that are geometrically coincident or very close to each other. This is useful when duplicate or nearly coincident geometries are present and need to be combined to reduce the number of geometric entities. It ensures that nearby geometric entities are united, simplifying the geometry and improving mesh quality finding geometric entities that are within a very small distance and merging them. If merging is not done, faceting can result in duplicated or closely-spaced vertices or edges, which can negatively impact mesh quality and lead to poor simulation results due to mesh discontinuities.

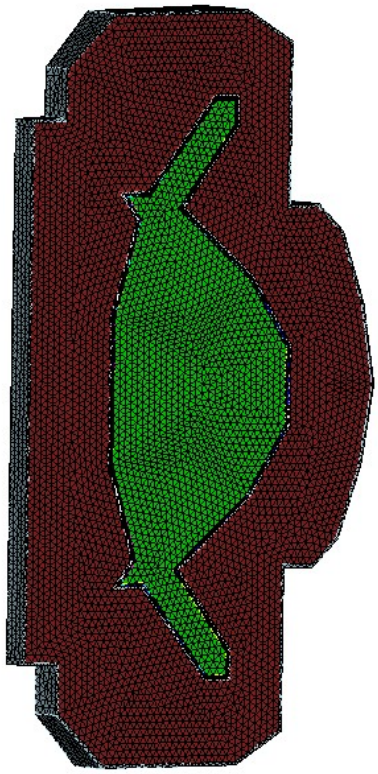

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Figure 3.3: Model meshed in Coreform Cubit

The meshing scheme used is a surface meshing using trimesh. In Coreform Cubit that refers to a triangular meshing scheme. This scheme is used because trimeshes are particularly useful for defining complex surfaces and geometries because triangular elements can easily conform to irregular shapes, allowing for more precise modeling of curved surfaces. The parameters of the mesh used are the default ones so the deviation angle is 15° and the gradation value is 1.3. The final result of a geometry meshed with these settings is represented in figure 3.3.

Chapter 4

Neutronic results

In this chapter the results from the neutronic simulations are analyzed confronting the two geometries considered, the one with FW of 1 mm (reference case) and the one with FW of 1 cm as introduced in paragraph 3.1.

4.1 Neutron flux

Figure 4.1 represent the neutron flux spectrum across the different regions calculated in number of neutron per cm^2 per second. The results are obtained by dividing the results of a flux tally in openMC by the volume of the various regions considered. The volumes are taken from the data of the Solidwork project.
The values for the 1 cm FW are consistently higher than the ones for the 1 mm FW at high energy. Having a bigger flux for higher energy in the thicker FW suggest that having a thickened structure enables scattering of more high energy neutrons due to its increased material volume, decreasing the number of neutrons that leak from the reactor. Instead at low energy the relationship is opposite, with the thinner wall having bigger neutron fluxes. Maybe this is due to the fact that, with a thinner FW there is less reabsorption of low energy neutrons with respect to the 1 cm FW.
The difference in lower energy neutron flux is smaller as the study moves outside of the reactor. In the neutron multiplier and in the outer VV graphs the line for the low energy fluxes overlap each other. This is due to a progressive neutron moderation as neutrons move away from the first wall they pass through equal component in the two cases that slow them down in similar ways.

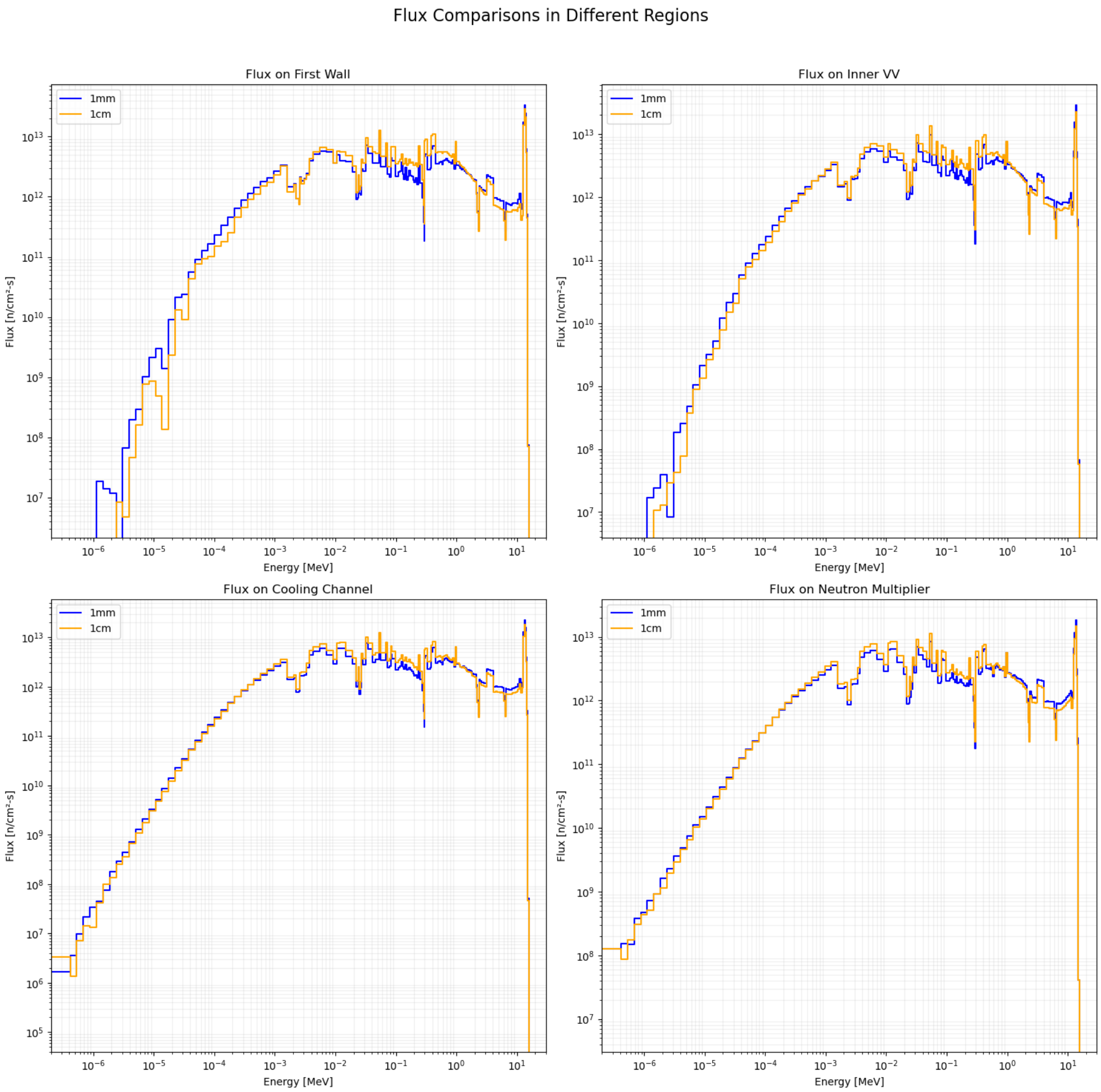

Figure 4.1: Neutron flux spectrum calculated on all the surfaces as a function of energy for first wall thicknesses of 1 mm and 1 cm (continue)

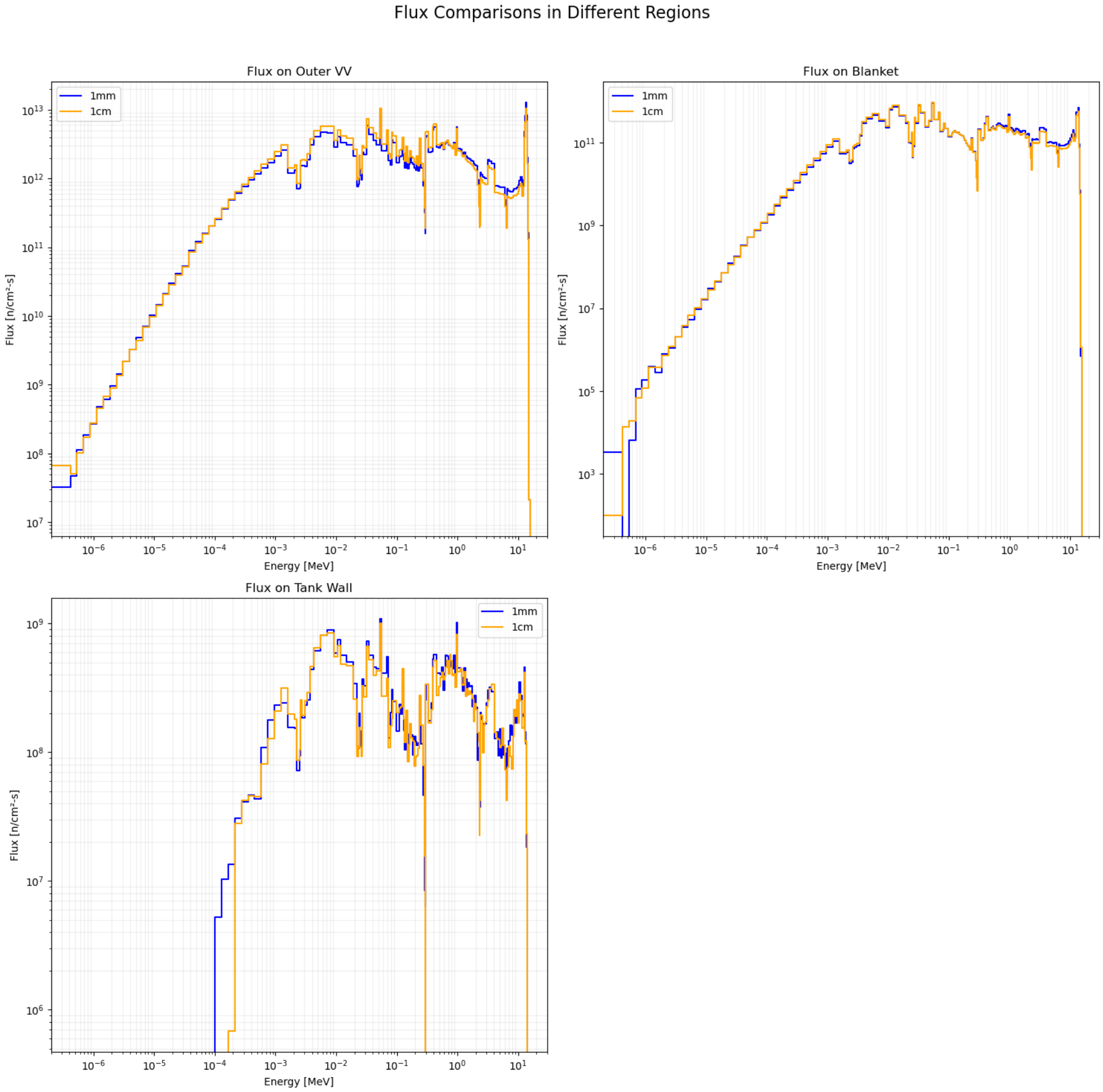

Figura 4.1: Neutron flux spectrum calculated on all the surfaces as a function of energy for first wall thicknesses of 1 mm and 1 cm

4.2 Map of neutron flux

Figures 4.2 and 4.3 present the neutron flux maps with the 1 mm FW and 1 cm FW configurations respectively. The maps are built using a tally using a mesh built in openMC recording the flux on the same mesh and then saving the geometry in vtk format and using Paraview to visualize it. In addition to the maps the model of the reactor has been incorporated.
The notable difference between the two maps is that in the center the light red circle is bigger in the 1 cm FW. This is due to the fact that a thicker FW scatters and reflects more neutrons back into the reactor; this phenomenon is seen also in the comparison of the neutron flux spectra seen in the previous section.

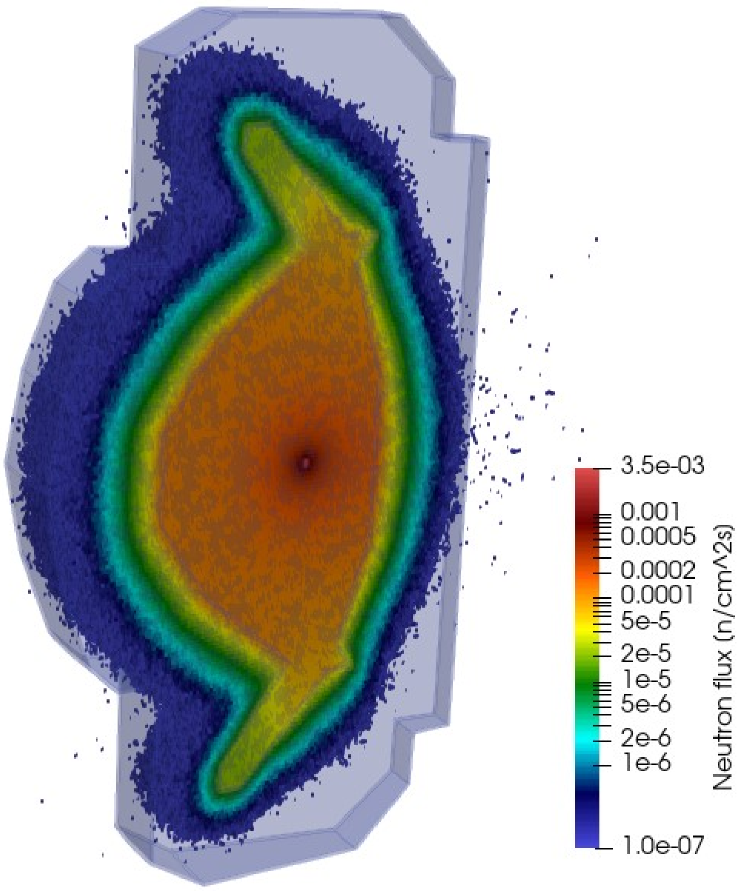

Figure 4.2: Map of neutron flux on surfaces n/cm^{2}s with fw 1 mm

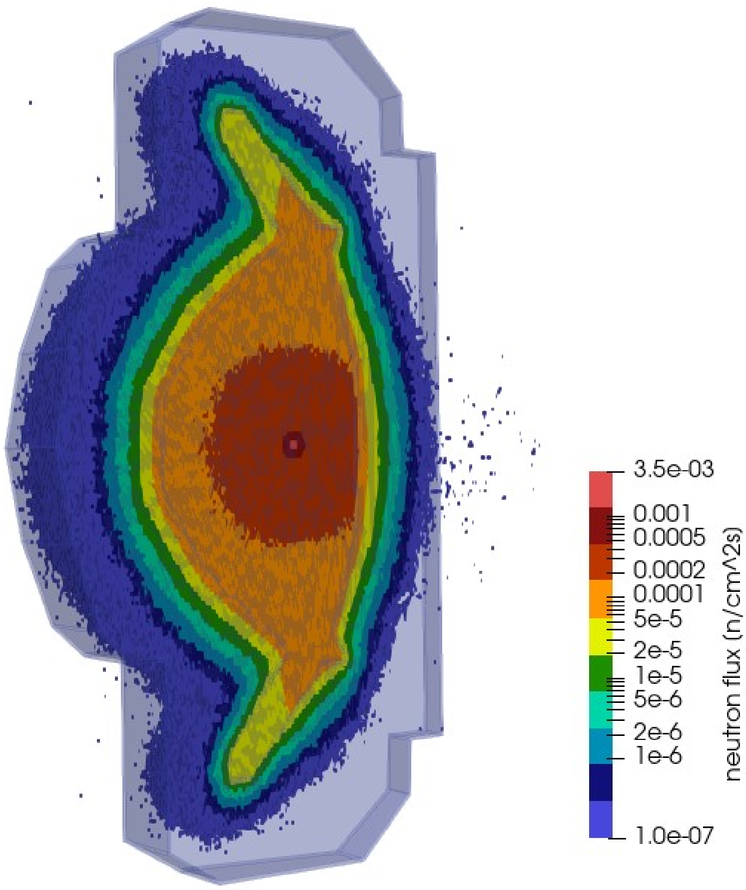

Figure 4.3: Map of neutron flux on surfaces n/cm^{2}s with fw 1 cm

4.3 Volumetric power generation

In this section the heat generation and deposition on the various component is analyzed. In table 4.1 the volumetric power generation divided per volumes and considering the two FW configuration is presented, while in figures 4.4 and 4.5 represent maps of generated heat on surfaces.

Volumetric power generation [MW/m^3]		
Component	**1 mm**	**1 cm**
First wall	24.456	17.121
Inner VV	11.100	9.154
Cooling channel	11.604	11.984
Neutron multiplier	6.503	5.587
Outer VV	3.593	2.964
Blanket	0.700	0.652
Tank wall	0.004	0.003

Table 4.1: Values of volumetric power generated on different components

In table 4.2 the most significant difference is of course in the FW due to the volume being bigger in the second configuration. For the other component the difference arises even though the thickness in the two configurations of the components is unchanged but having a bigger FW volume expands the volumes of all the others component. For the inner VV the thicker FW influence the power deposited, since the higher scattering of neutrons reduces energy deposition on this component. This is due to less high energy neutrons directly reaching the inner VV.
The same scattering that cause the power density to decrease in the inner VV volume is the reason for the increase in the cooling channel volume, since the scattered slowed down neutron have higher probability to interact in the cooling channel. Going further away from the FW, its difference influence less the values of power generated having the 1 cm values smaller than the 1 mm values is due to the bigger volumes in the first configuration.

The heat maps in figure 4.4 and 4.5 shows higher localized heat generation on surfaces near the center of the FW. As for the flux maps also in these the structure of the reactor has been incorporated.
The heat generation peak values for the 1 mm FW are lower compared to the 1 cm configuration. This is due to the thinner FW allowing more neutrons to pass through without significant scattering, leading to a smaller heat deposition in a concentrated area. The concentrated heat areas suggest higher thermal loads on components close to the FW, which could impact material performance over time.
The 1 cm FW configuration shows higher peak heat generation values, especially in regions adjacent to the FW due to the thicker FW's ability to scatter and reflect more neutrons back into the reactor, thereby increasing the local neutron flux and, consequently, the energy deposition in these areas.

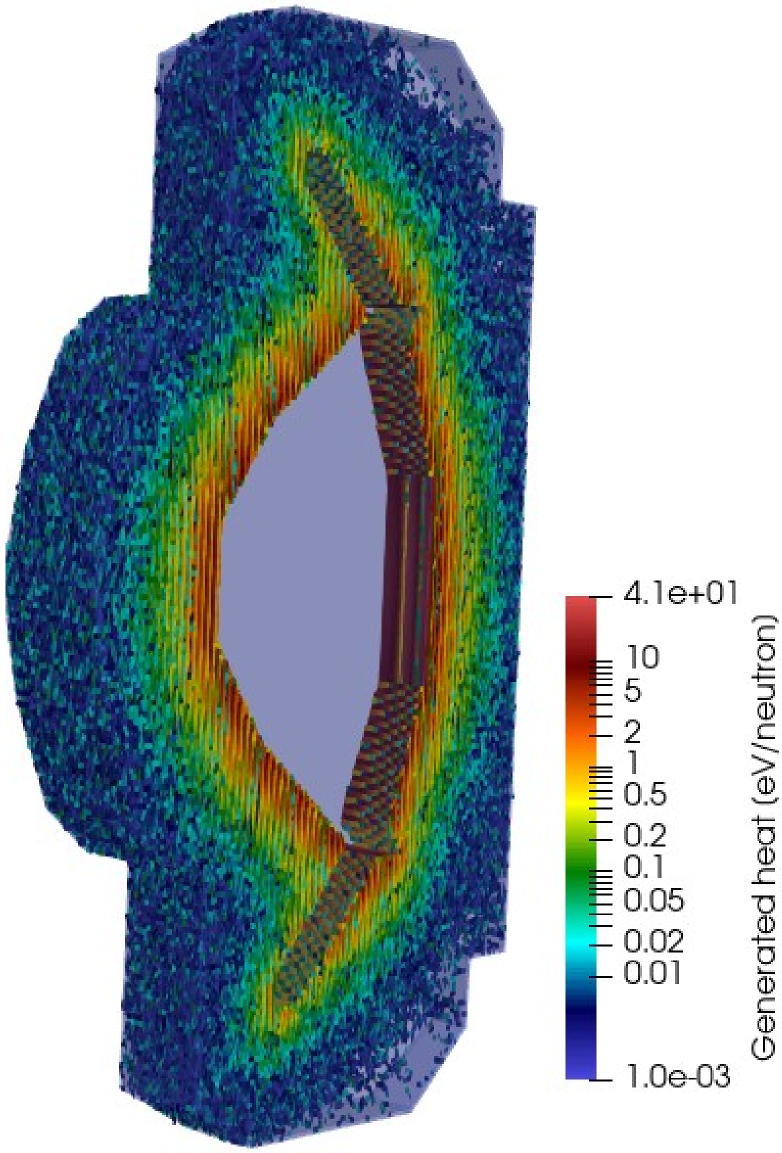

Figure 4.4: Map of generated heat on surfaces in eV/neutron with fw 1 mm

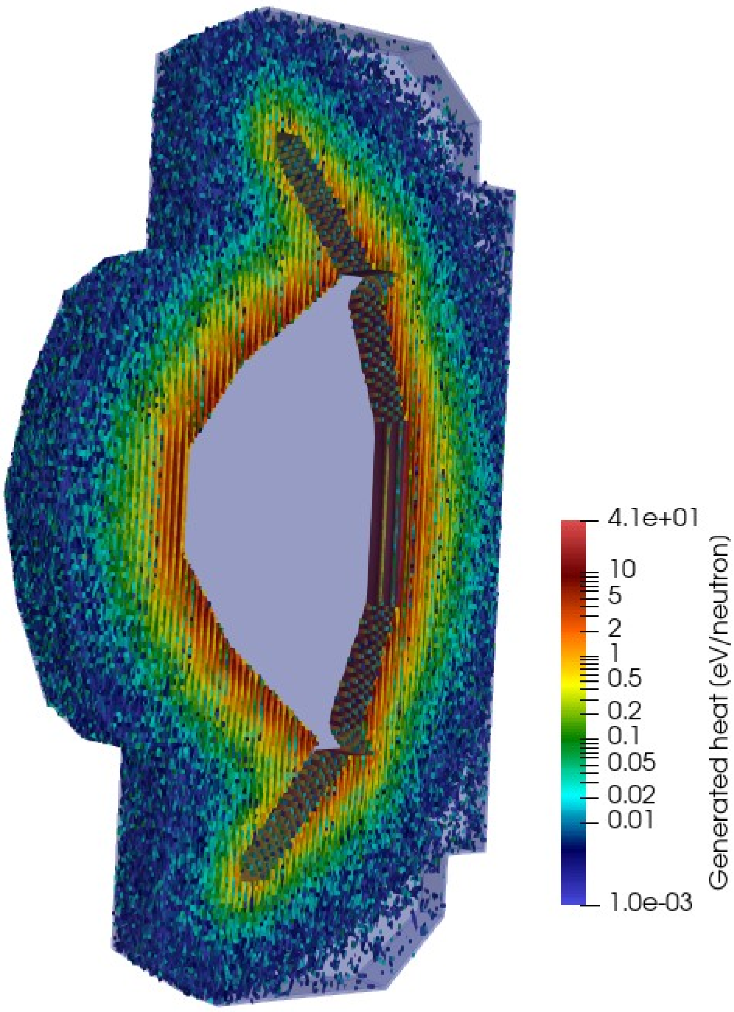

Figure 4.5: Map of generated heat on surfaces in eV/neutron with fw 1 cm

4.4 TBR

TBR [-]				
First Wall	1 mm		1 cm	
	Cooling Channel	Blanket	Cooling Channel	Blanket
Li6	0.1985	0.8602	0.2452	0.8821
Li7	0.0017	0.0073	0.0014	0.0060
Be9	0.0004	0.0012	0.0003	0.0010
F19	0.0011	0.0034	0.0009	0.0028
Total	**1.0763**		**1.1416**	

Table 4.2: Values of TBR for selected isotopes

Table 4.2 presents the TBR values for various isotopes within the reactor's cooling channel and blanket for the 1 mm and 1 cm FW configurations. Li6, a key isotope for tritium breeding, shows an increased TBR contribution in both cooling channel and blanket for the 1 cm FW configuration. This increase is due to the thicker FW. This can be seen in figure 4.6 where it is represented the reaction cross section of the $^{184}W(n,2n)^{183}W$ reaction in function of the energy of the incident neutron. The W184 isotope has been chosen since it is the most present isotope in natural tungsten which is considered in the first wall. In the plot 4.6 it can be seen that the cross section peak at an energy interval near the one of the neutrons produced in the D-T reaction (about 14.1 MeV). Increasing the thickness of the tungsten means that more neutrons will be produced and that more neutrons will interact with the Tritium breeding isotopes in the cooling channel and in the blanket.
Furthermore the thicker FW provides higher neutron moderation and retention, which enables more neutrons to reach the studied regions at optimal energies for tritium production. The enhanced moderation results in a larger portion of neutrons reaching the lithium-rich regions at energies that maximize absorption in Li6, meaning that they are more likely to interact with Lithium supporting more efficient tritium breeding.

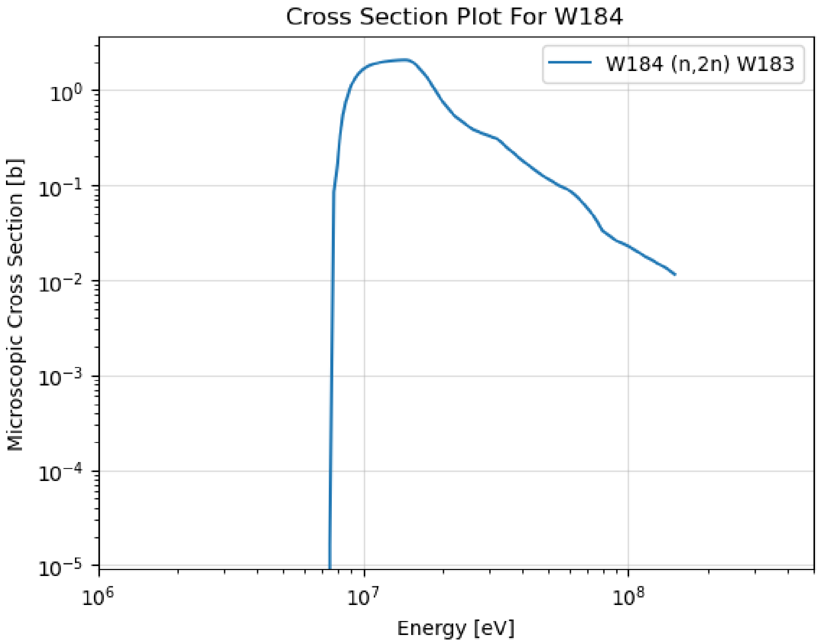

Figure 4.6: Plot of the cross section of the reaction (n, 2n) of W184[10]

Chapter 5

Uncertainties Quantification

This chapter focuses on the propagation of uncertainties in the neutronic analysis using the SANDY tool. Uncertainty assessment is a crucial part of advanced neutronic studies, as it provides a more accurate understanding of the reliability of the obtained results and the potential variations due to uncertainties in the input data. The SANDY tool is a software designed to read, modify, and manipulate nuclear data files in the ENDF-6 format. SANDY was chosen for its capability to generate perturbed files that represent uncertainties in evaluated nuclear data, particularly in cross sections. This choice has several advantages: it is relatively straightforward to implement, and it allows the generation of a wide range of samples based on random perturbations. This approach is especially effective when the goal is to analyze the propagation of uncertainties without the need to calculate adjoint functions, which can be complex and computationally demanding. In this work, the uncertainty propagation focuses on the cross sections of the isotope Li-6, which is critical for tritium breeding in fusion reactors. By perturbing these cross sections, it is possible to assess how uncertainties in the nuclear data influence the results of the TBR calculations. This analysis was performed using 500 samples, each representing a different perturbation of the nuclear data, based on the evaluated uncertainties provided in the ENDF/B-VIII.0 library.

The decision to generate 500 samples was based on the need of having statistical significance but also computational efficiency. Increasing the number of samples generally improves the accuracy of the uncertainty quantification, as it provides a more precise representation of possible variations, however, this also increases computational time. In this case, 500 samples were deemed sufficient to achieve convergence in the results while keeping the simulation time manageable. The Fast Monte Carlo sampling method employed by SANDY ensures that each simulation run is statistically independent, capturing a wide range of possible outcomes for the TBR due to variations in the input data. This approach is advantageous because it can cover the entire input and output space, allowing for a thorough

analysis of potential effects due to uncertainties in the cross sections. The perturbed cross section libraries generated by SANDY were applied to a simplified spherical geometry, specifically designed to isolate the impact of uncertainties. The geometry consisted of three regions: an inner and outer vacuum, with the material of interest located in an intermediate shell. This setup allowed for the evaluation of how variations in the cross-section data directly influenced the TBR calculations and it also reduced computational time with respect to a study on the real model.
Results were analyzed to determine not only the average TBR but also the spread of the values. A detailed investigation was carried out to identify any outliers, which were then assessed to determine if they resulted from statistical anomalies or if they revealed specific sensitivities in the model. Additionally, normality tests were performed to verify if the distribution of TBR values followed a Gaussian pattern, providing insights into the behavior of uncertainties in the data.
By integrating this uncertainty analysis, the study offers a more comprehensive understanding of an ARC-like reactor performance under different conditions, highlighting both the strong points and potential areas for optimization.

5.1 Results of the analysis

Results obtained from 500 perturbed cases for TBR are summarized as follows:

- Mean TBR: 0.5563
- Standard Deviation: 0.0119
- Confidence Interval (95 %): [0.5553, 0.5574]
- Percentiles:
  - 5th Percentile: 0.5360
  - 50th Percentile (Median): 0.5565
  - 95th Percentile: 0.5755

These results indicate a stable TBR value range with respect to nuclear data uncertainties, showing minimal variation as seen in figure 5.1.

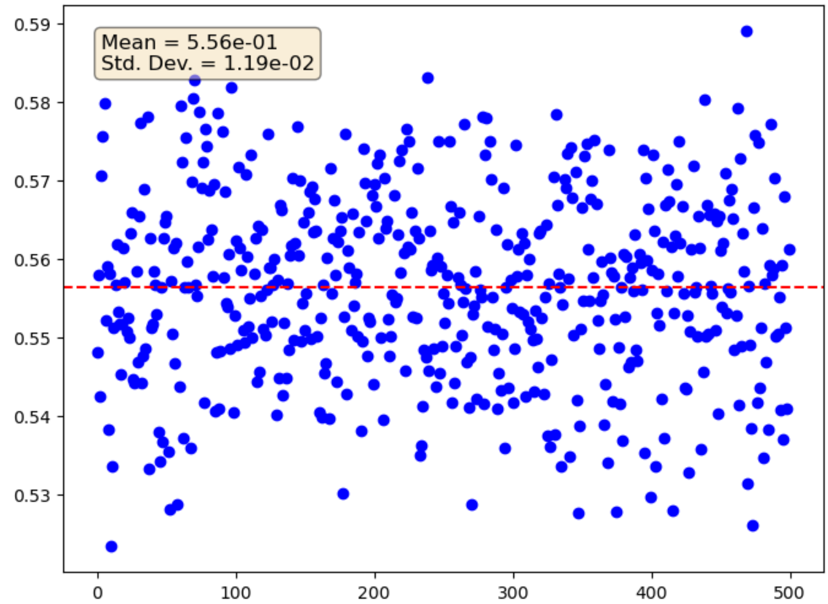

Figure 5.1: Range and spread of TBR values across perturbed samples

The TBR distribution is also visualized in figure 5.2 as a histogram, which displays the frequency distribution of TBR values across the 500 samples.

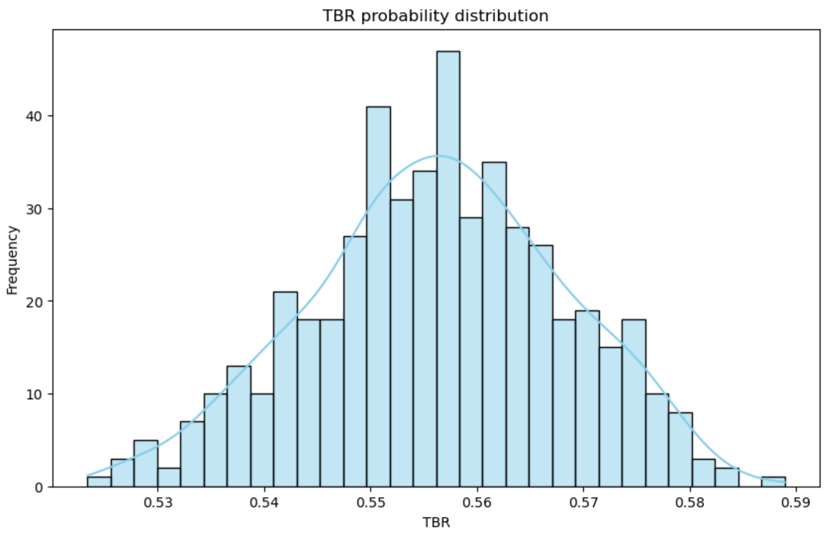

Figure 5.2: TBR probability distribution

5.2 Outlier Analysis

Certain values were identified as outliers during the simulation:

Cell	TBR [-]
10	0.5234
468	0.5890

Table 5.1: Outliers

The outliers written in table 5.1 and represented in the boxplot of figure 5.3 were calculated using the interquartile range method. This procedure consists in calculating the first quartile (Q1) and the third quartile (Q3) and by subracting Q1 to Q3 the interquartile range is obtained. To determine the treshold out of which a value is considered an outlier a lower and an upper bound are calculated as follows:

$$\text{Lower Bound} = Q_1 - 1.5 \times \text{IQR}$$

$$\text{Upper Bound} = Q_3 + 1.5 \times \text{IQR}$$

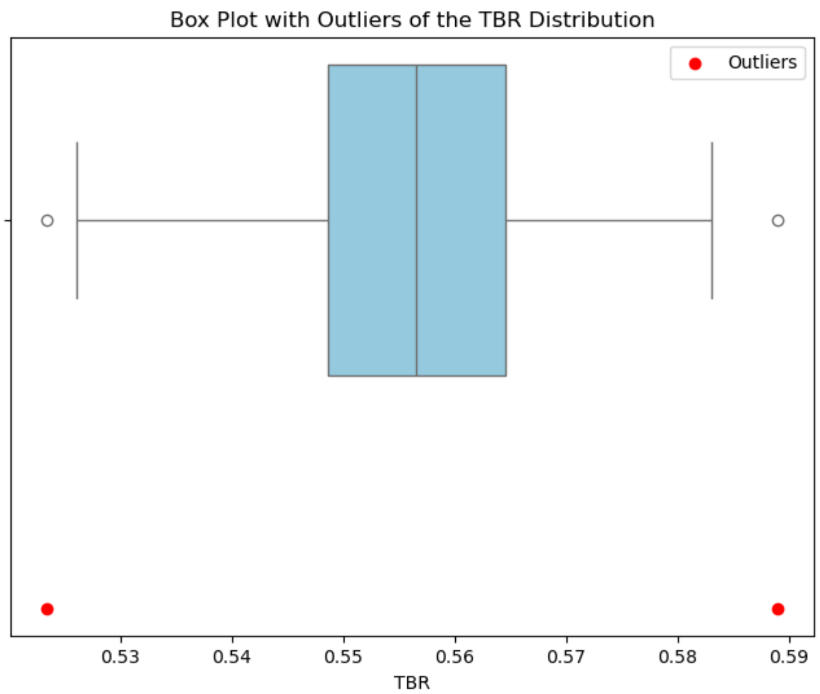

Figure 5.3: Boxplot showing the TBR range and highlighting the identified outliers.

In the boxplot 5.3 the box itself represents the interquartile range, the line inside the box is the median, the whiskers goes from the lower to the upper bound and the red points are the outliers that are detected.

5.3 Normality Testing of TBR Distribution

Normality of the TBR distribution was assessed using the Shapiro-Wilk and Anderson-Darling tests:

- Shapiro-Wilk Test:
  - Statistic: 0.9954
  - p-value: 0.1488
  - The TBR distribution does not significantly deviate from normality.
- Anderson-Darling Test:
  - Statistic: 0.3778
  - Critical Values: [0.571, 0.651, 0.781, 0.911, 1.083] for significance levels of 15%, 10%, 5%, 2.5%, and 1%.
  - At a 5% significance level, the TBR distribution shows minimal deviation from normality.

The results of the normality testing are further visualized in Figure 5.4 using a Q-Q plot, which illustrates the fit of TBR values to a normal distribution. Points closely aligned along the diagonal indicate that the data largely follows a normal distribution.

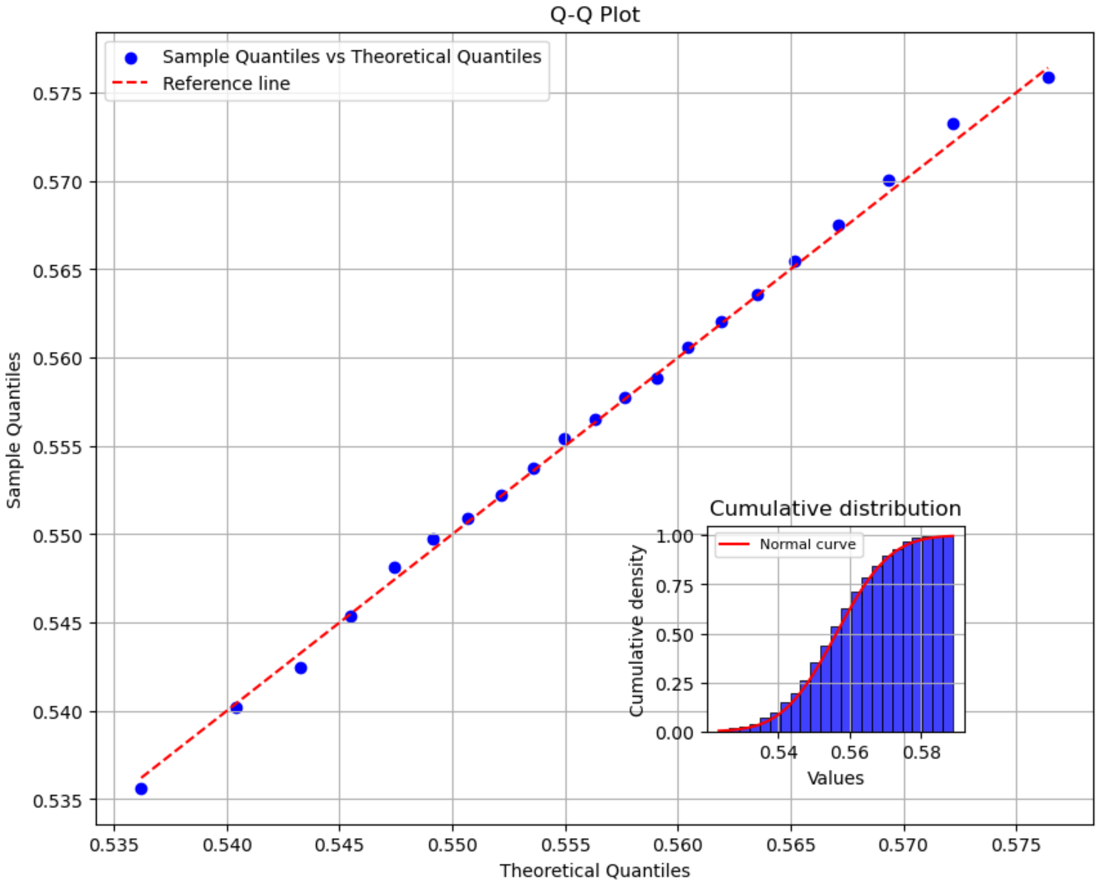

Figure 5.4: Q-Q plot of TBR values, showing the fit of the distribution to a normal distribution.

Chapter 6

Conclusions

This thesis presented a comprehensive neutronic analysis of an ARC-like tokamak fusion reactor model, focusing on the impact of variations in the FW thickness comparing the 1 mm reference case with the cm configuration.
The ARC model incorporate a more modular structure that benefits the reactor's operational flexibility. A modular design allow easier replacement and maintenance of components, enabling ongoing experimentation with advanced materials. This adaptability make the ARC-like reactor not only a reliable power source but also a valuable facility for nuclear research.
The study employed Monte Carlo simulations using OpenMC to evaluate key parameters, TBR, neutron flux, and volumetric power generation across different reactor regions. Additionally, the SANDY tool was utilized to propagate uncertainties in cross-section data for critical isotopes, providing insight into the reliability and stability of tritium production estimates.
The primary findings indicate that an increase in FW thickness positively influences the neutron moderation and retention within the reactor, leading to a higher TBR. This effect is attributed to enhanced scattering and neutron reflection capabilities of a thicker FW, which improve tritium breeding conditions. The uncertainty propagation analysis results also revealed that the reactor maintains relatively stable tritium breeding efficiency despite perturbations in nuclear data, demonstrating the robustness of the design to cross-section uncertainties. Outliers in TBR were identified and analyzed, verifying that they stemmed from statistical variations rather than model inconsistencies.
Future developments in the design of ARC-like reactors could focus on enhancing both the materials and methods used in neutronic analysis to improve efficiency and robustness. One promising direction involves the optimization of first-wall and blanket materials. By exploring alternative materials with better neutron reflection and absorption properties, both energy capture and tritium breeding efficiency could be improved. Additionally, advanced materials might provide

greater resistance to radiation damage, thus extending the operational lifespan of key reactor components.
In parallel, refining uncertainty quantification techniques offers significant potential for improving predictive reliability. Integrating more comprehensive nuclear data with advanced computational methods could lead to more precise analyses, where machine learning and refined perturbation methods may model potential fluctuations in reactor parameters with higher accuracy. With improved tritium breeding, future work could also aim at developing more efficient tritium extraction and recovery processes, crucial for achieving a self-sustaining fuel cycle that minimizes reliance on external tritium sources.
In conclusion, these future advancements hold considerable promise for optimizing ARC-like reactors. By addressing the challenges of neutronic performance and data uncertainty, ARC-like reactors could play a central role in the future of sustainable fusion energy, contributing significantly to the advancement of clean and efficient energy production.

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