

Preliminary 3-D Modeling and Neutronics Analysis for FESS-FNSF Design

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- Abstract

Several nuclear parameters are of great importance for any fusion device to determine the operational conditions. Among those parameters are the neutron wall loading (NWL), tritium breeding ratio (TBR), dpa, He production, and nuclear heating. Such values reflect on the suitability of the chosen materials composition for the operating conditions, shielding performance, and more importantly, for fuel self-sustainability (tritium self-sufficiency) of the plant. Analysis was performed on the interim FNSF model with material composition that matches those on the pre-final design. DAGMC code was used to couple the solid CAD model with Monte Carlo radiation transport code. The peak NWL values are $1.46 \pm 0.02\% \text{ MW/m}^2$ on OB FW and $1.13 \pm 0.06\% \text{ MW/m}^2$ on IB FW and $0.73 \pm 0.32\% \text{ MW/m}^2$ for the divertor. A total (OB + IB) TBR of $1.054 \pm 0.18\%$ is obtained. Nuclear heating analysis was performed to determine both the radial heating distribution and the total heating in all regions and an energy multiplication of 1.12 is obtained.

I. Introduction

FNSF [1] is one step in the US pathway to fusion energy development with a goal of bridging the technical parameters between ITER [2] and US DEMO [3]. One of the main missions of the FNSF facilities is to establish a database for fusion materials similar to those provided for fission reactors; providing the behavior of different materials in a wide range of operating conditions (e.g. atomic displacements reaching 20-126 dpa, blanket temperatures reaching 500-800 °C) before proceeding to larger size in the DEMO [4].

This study is a first step towards a full 3-D analysis of the new proposed design. The goal of this study is to provide several nuclear parameters using a simplified model. The parameters that are calculated are the neutron wall loading (NWL), tritium breeding ratio (TBR), displacement per atom (dpa), He production, and the nuclear heating due to both neutrons and photons. The NWL is defined as the uncollided energy current density at the first wall of the fusion device which is of importance for material choices, radial build of the following layers, and shielding. The TBR is the ratio of tritium atoms produced in the blanket due to neutron reactions to those consumed in fusion reactions in the plasma. A value >1 is desired for self-sustainability of the fuel. The dpa is a measure of the damage to the materials due to neutron caused atomic displacements. Another form of radiation effect is the nuclear heating which is a measure of the deposited neutrons and photons energy in all regions.

The operating conditions used in this study are close to those of the more recent design; major Radius 4.5 m, minor Radius 1.125 m, fusion Power 450 MW, plant Lifetime ~ 8.5 FPY, availability $\sim 27\%$ on average. The simplified model developed was based on the results of a 1-D shielding analysis using PARTSIN code. The model consists of homogeneous regions resembling those of the newer design but with different thicknesses and without details; no cooling channels in the breeding blanket, combined vacuum vessel (VV) and LT shield, divertor is simulated as one plate, and no magnets or center stack.

II. Methodology

The work flow used for this study is the University of Wisconsin Unified Work flow (UWUW) [5] developed by the Computational Nuclear Engineering Research Group (CNERG) [6] at UW-Madison. The workflow consists of several stages/steps starting from the creation of CAD-based model to a final format ready to be used for transport calculations using transport codes like MCNP, and FLUKA. The workflow used in this study utilizes two main codes; CUBIT [7] and DAGMC [8].

CUBIT [7] is a full-featured software toolkit that enables solid model creation and mesh generation in two and three dimensions developed and released by Sandia National Laboratories. CUBIT has many features that enables both modeling of complex geometries like tokamak in 3-D and coupling with transport codes. Of those features the ability to group volumes/surfaces that belong to different regions of the model which helps in assigning materials, densities, boundary conditions, and define tallies for the transport calculation. Another feature of great importance is the imprint/merge which removes duplicated surfaces in adjacent volumes which might cause problems due to ill-defined model or lost particles later when coupled with transport codes. All those features along with many others and ease of use in creation and manipulation of complex 3-D solid models encouraged the use of CUBIT for this study.

The Direct Accelerated Geometry for Monte Carlo (DAGMC) [8] toolkit is a component of the mesh-oriented database (MOAB) [9] that provides fundamental functions for ray-tracing and related geometry operations of Monte Carlo radiation transport directly on complex CAD-based 3-D solid models. Without translation into the native Monte Carlo input language, DAGMC uses acceleration techniques for ray tracing directly on the CAD-based solid model like high-fidelity faceting combined with hierarchical trees of oriented bounding boxes for those facets. The resulting method is both efficient and enables the modeling of very complex geometries including those with high-order surfaces. DAGMC utilizes the radiation transport code MCNP5 [10]. The material library used is a custom created library of numerous materials composition originally developed for the ARIES project [11].

III. Model and Radial Build

Based on preliminary 1-D simulations using PARTSIN deterministic transport code and knowledge of the results obtained from previous analysis of ARIES-ACT-2 design [12], the 3-D model was developed. Figure 1 shows the inboard (IB) radial build along with all relevant thicknesses of various regions. The outboard was simulated as two regions only; a first wall (FW) integrated with the dual-cooled LiPb (DCLL) breeding blanket with the same material composition as for the IB. The preliminary material compositions are as follows:

First Wall (FW)	34% FS (F82H), 66% He
Breeding Zone (BZ)	76% LiPb (90% Li-6), 13% He/void, 7% FS, 4%SiC
Back Wall (BW)	80% FS, 20% He
Structural Ring (SR)	15% FS, 10% He, 75% WC Filler
Thick Vacuum Vessel (VV)	22% FS, 33% H ₂ O, 45% WC
Coil Case (CC)	95% SS, 5% Liquid He
Winding Pack (WP)	70% SS, 15% Cu, 3% GFF Polyimide, 2% Nb ₃ Sn, 10% Liquid He

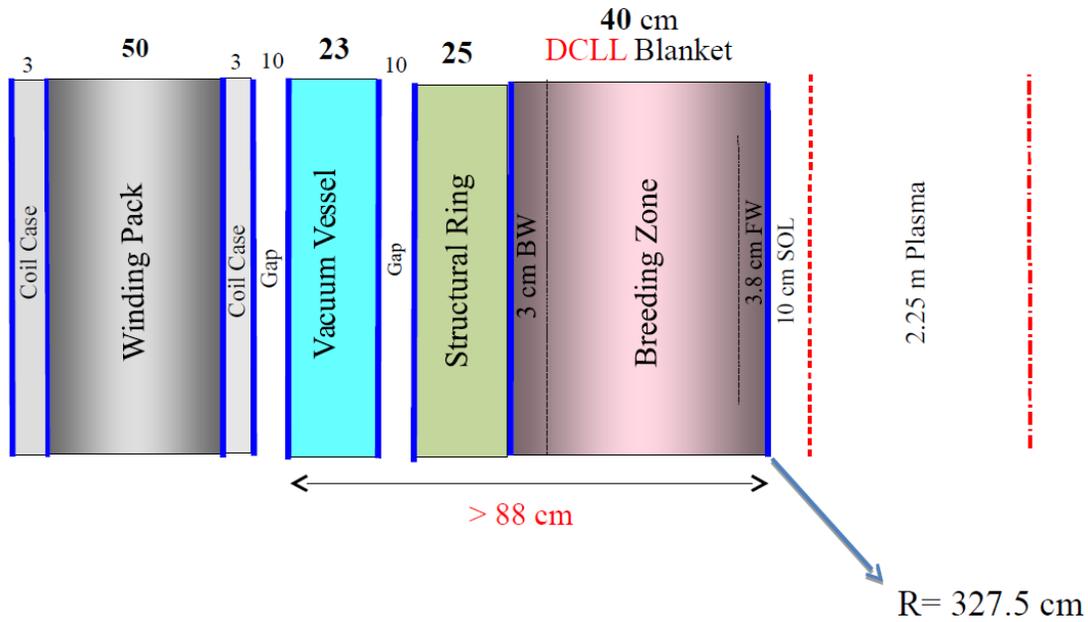


Figure 1. IB Radial build

As stated above the 3-D model is a simplified simulation of the final design for the purpose of obtaining several initial nuclear parameters. Of the approximations used are: simulating both IB and OB breeding zones (BZ) as homogeneous regions rather than including more elaborated details like cooling channels and penetrations, and the divertor plates were simulated as one inclined plate. From previous ARIES-ACT-2 study [12] it was shown that the effect of adding the details to the breeding zone and penetrations on the TBR was a decrease of the value obtained with the homogeneous model. The geometry of the source used is approximated as three-zones, each with different neutron yield depending on the physics of the device. The IB was simulated as co-axial annuli with thicknesses determined by the radial build as in figure 1. An upper and lower shield regions were added with the same material composition as the IB back wall to simulate the back scattering/reflection to IB and OB regions.

Using CUBIT, the 3-D model was created and a vertical cross section is shown in Figure 2. Following UWUW workflow the different regions were assigned materials using the surfaces/volumes grouping capability of CUBIT. A faceted model was then produced with a faceting tolerance of 10^{-4} which is used for DAGMC ray tracing. A material library was then created containing all the isotopic and composition metadata and added to the faceted model and is then used as input for DAGMC.

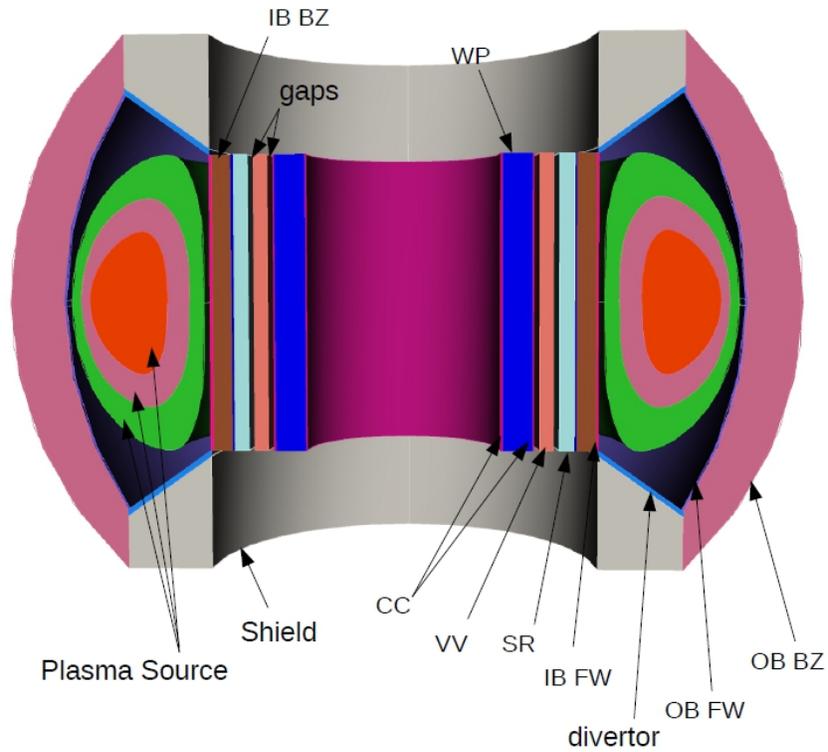


Figure 2. CUBIT solid model with different regions identified

IV. Neutron Wall Loading (NWL)

In order to obtain the poloidal distribution of the neutron wall loading, both the IB and OB FWs were sectioned into smaller surfaces each 10cm high using CUBIT. Using DAGMC, tallies were calculated for each of those surfaces yielding the desired distribution. Figure 3 shows the segmented OB FW and figure 4 shows the segmented IB FW.



Figure 3. Segmented OB FW for NWL calculations

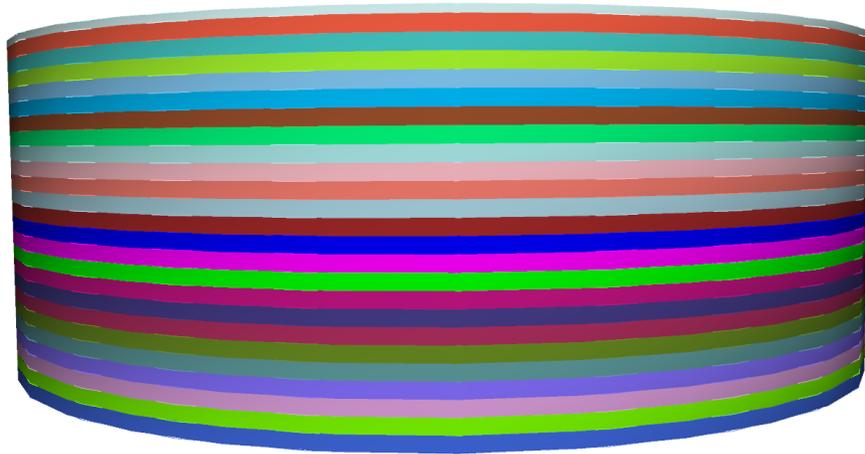


Figure 4. Segmented IB FW for NWL calculations

IV-a. NWL for OB & IB FWs

Results were obtained for the IB and OB FWs using a plasma source with a power of 450 MW. First the theoretical NWL was calculated for a surface 10cm away from the last plasma surface and found to be 1.04 MW/m^2 . The value was also checked against DAGMC using a fictitious surface 10cm away from the last closed plasma surface and a match was found.

Using the segmented FWs shown in figures 3 and 4 and using DAGMC, the NWL peak values for the OB FW is $1.46 \pm 0.02\% \text{ MW/m}^2$ and for the IB FW is $1.13 \pm 0.06\% \text{ MW/m}^2$. Figure 5 shows a plot of the results obtained versus the vertical distance illustrating the expected behavior of decreasing the NWL with the vertical distance. The average values obtained are 0.76 MW/m^2 for the IB FW and 0.96 for the OB FW.

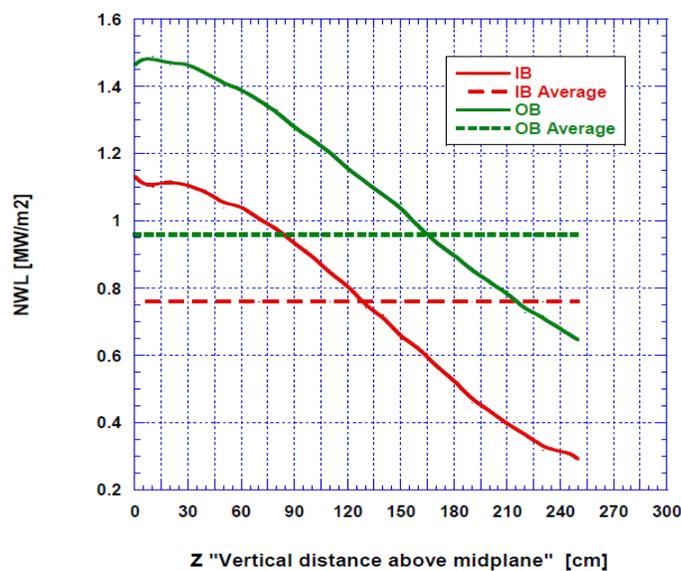


Figure 5. Poloidal distribution of the IB & OB NWL

IV-b. NWL for the divertor

Because of the symmetry of the model, the NWL for the divertor was obtained only for the upper plate. The divertor plate was also segmented as was the case with the OB & IB FWs to obtain the poloidal distribution of the NWL. The peak value obtained is $0.73 \pm 0.32\%$ MW/m^2 and the average value is $0.54 MW/m^2$. Figure 6 shows the poloidal distribution of the divertor NWL and as expected the NWL decreases with the distance from the plasma surface; the larger the distance of the segment from the last plasma surface the lower the NWL.

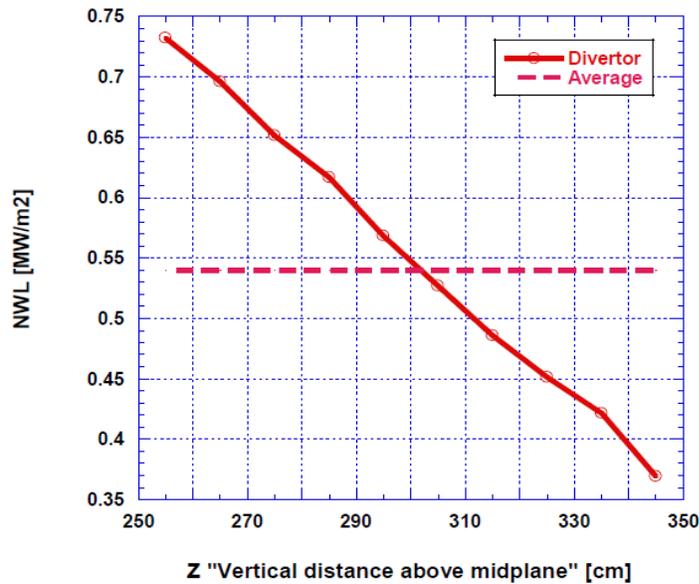


Figure 6. Poloidal distribution of the divertor NWL

V. Tritium breeding ratio

Tritium breeding ratio TBR is obtained for the homogenized IB and OB breeding blankets. Both blankets were assumed to have the same material composition. First, the value of TBR is obtained for the breeding blanket composition as in section III then the LiPb atomic ratios were changed to quantify the effect of a more accurate composition on the TBR. Table (1) shows both TBRs. Li₁₇Pb₈₃ composed of 17 atom percent Li with 90% enrichment in Li-6 and Li_{15.7}Pb_{84.3} is composed of 15.7 atom percent Li with with 90% enrichment in Li-6. The effect of varying the composition with decreasing the Li content is, as expected, a decrease in the TBR.

The values obtained is believed to be the upper bound for the expected TBR for the breeding blankets in the new design. Adding the blanket internals in detail and including the H/CD ports will decrease the breeder volume and hence the obtained TBR [12]. A more detailed study showing the effect of each added detailed is supposed to be carried out for the new design.

Li17Pb83

<u>Li17Pb83</u>	OB	IB	Total
TBR	$0.861 \pm 0.08\%$	$0.215 \pm 0.16\%$	$1.076 \pm 0.18\%$

Li15.7Pb84.3

<u>Li15.7Pb84.3</u>	OB	IB	Total
TBR	$0.846 \pm 0.08\%$	$0.208 \pm 0.16\%$	$1.054 \pm 0.18\%$

Table 1. TBR values for IB & OB breeding blankets for two different LiPb compositions

VI. He production and dpa

The peak damage at the IB and OB FWs was calculated for a 10 cm high surface at the midplane. The obtained values are representative of the peak damage at the midplane at the FW due to neutron displacement of atoms from its lattice sites. The dpa is additive which means that its value at the end of the life of the component is of importance, so in any fusion design it's one of the parameters controlling the lifetime of components. Dpa is often calculated in units of displacements per atom per full power year (dpa/FPY). A value of 14.55 dpa/FPY is obtained for the OB FW and 13.77 dpa/FPY for the IB FW (see table 2).

Several neutron reactions (like (n,α) , $(n,n'\alpha)$, etc) lead eventually to the formation of He atoms. He production in the IB and OB FWs was calculated. A value of 161.18 He appm/FPY is obtained for the OB FW and 152.76 He appm/FPY for the IB FW (see table 2). the HE/dpa ratio is 11 – typical for DCLL blankets.

	OB	IB
dpa/FPY in FS	$14.55 \pm 0.72\%$	$13.77 \pm 1.21\%$
He appm/FPY in FS	$161.18 \pm 0.94\%$	$152.76 \pm 1.77\%$
He / dpa ratio	$11.08 \pm 0.55\%$	$11.09 \pm 0.98\%$

Table 2. He production and dpa to FS of IB & OB FWs

VII. Nuclear heating

Nuclear heating plays an important role in fusion designs from different perspectives; choice of coolants to retrieve deposited energy, choice of materials to withstands the operating conditions, thermal hydraulic analysis, material thermal stresses, and above all energy multiplication. Analysis is done to obtain the peak nuclear heating at the IB & OB FWs, radial distribution in the relevant individual components of the material composition of the different regions, and finally the mapping of nuclear heating in all regions.

VII-a. Peak nuclear heating

Peak nuclear heating values were obtained for FS component of the IB & OB FWs due to both neutrons and photons for a surface 10 cm high at the midplane. Table (3) shows the obtained values.

	OB	IB
Neutron heating in FS (W/cm^3)	$3.02 \pm 0.84\%$	$2.79 \pm 0.73\%$
Gamma heating in FS (W/cm^3)	$5.61 \pm 2.37\%$	$4.65 \pm 1.41\%$
Total (W/cm^3)	$8.63 \pm 2.51\%$	$7.44 \pm 1.6\%$

Table 3. Peak nuclear heating for FS at IB & OB FWs

VII-b. Radial distribution of heating

Radial distribution of heating due to both neutrons and photons is obtained by segmentation of the FW and BZ in the radial direction. The values are obtained for the relevant components of the material composition of both the FW and the BZ, namely FS, SiC, W, and LiPb. Table (4) and figure 7 show the results obtained.

	R [cm]	Neutrons [W/cm^3]	Photons [W/cm^3]	Total [W/cm^3]
W (Peak Value)	327.5	$0.60 \pm 0.62\%$	$24.13 \pm 1.23\%$	$24.73 \pm 1.4\%$
FS (Peak value)	327.5	$2.79 \pm 0.73\%$	$4.65 \pm 1.62\%$	$7.44 \pm 1.8\%$
SiC (Peak value)	323.7	$4.66 \pm 0.63\%$	$1.55 \pm 1.51\%$	$6.21 \pm 1.6\%$
LiPb (Peak value)	323.7	$2.19 \pm 0.46\%$	$10.80 \pm 1.45\%$	$12.99 \pm 1.5\%$

Table 4. Radial distribution of nuclear heating in the IB FW and BZ

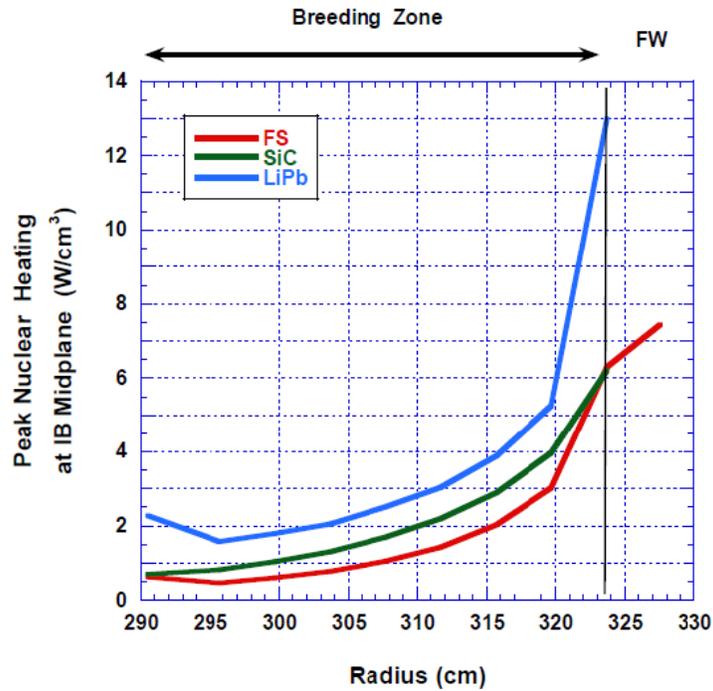


Figure 7. Radial distribution of nuclear heating for the IB FW and BZ

VII-c. Mapping of nuclear heating

As stated before, the plasma source is assumed to have a power of 450 MW. Since the DT reaction is the main source of neutrons, the produced neutron energy is 360 MW. Calculating the energy deposited in all regions considered in the model, an energy multiplication of 1.12 is obtained. Figures 8 and 9 show the mapping of nuclear heating in actual materials due to neutrons and photons respectively. A mapping of the total heating is shown in figure (10).

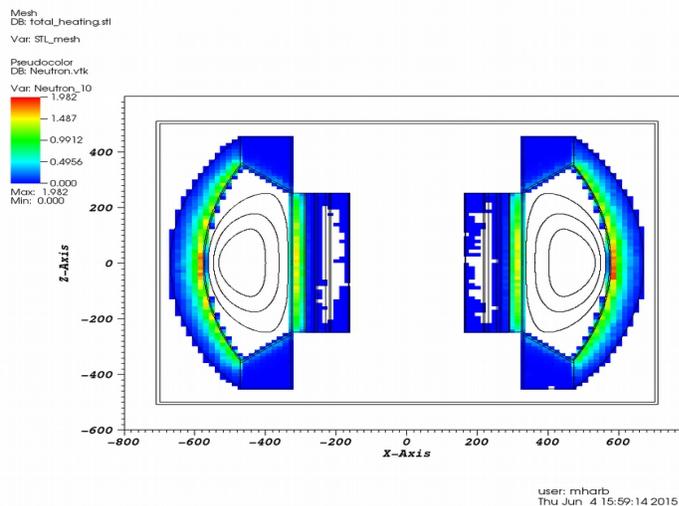
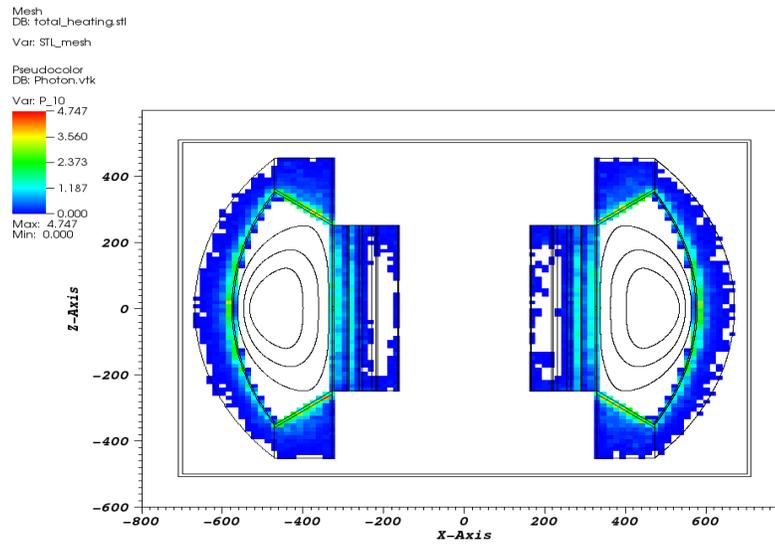
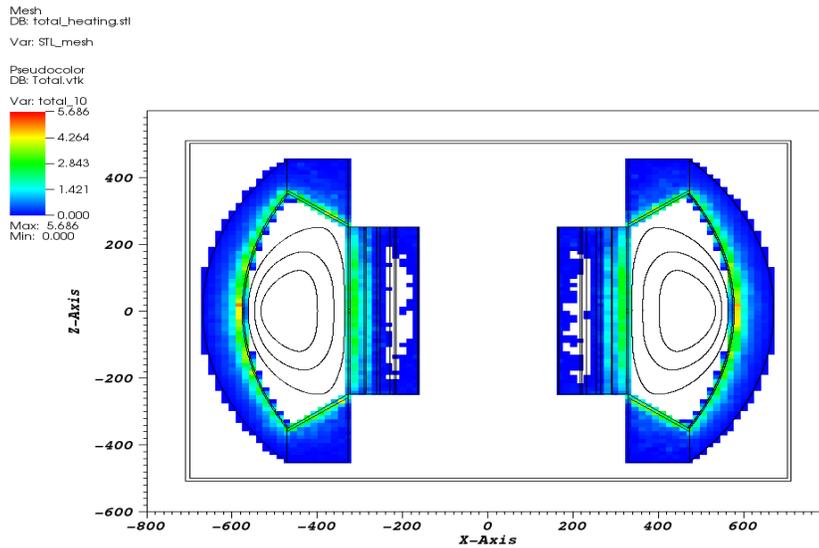


Figure 8. Mapping of nuclear heating due to neutrons in all regions



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Figure 9. Mapping of nuclear heating due to photons in all regions



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Figure 10. Mapping of total nuclear heating (neutrons + photons) in all regions

VIII. Conclusions

In the analysis of the simplified FNSF model, we provided several initial nuclear parameters to promote further design developments. The energy multiplication value obtained (1.12), TBR, peak dpa to FS of IB and OB FWs, and peak nuclear heating of IB and OB FWs are within the expected values for a Tokamak design with 1.5 MW/m^2 OB NWL. Recent developments of software tools for neutron transport calculations made it easier to analyze complex geometries. Coupling of CAD-based softwares with DAGMC proved efficient in the analysis of new designs and allowed for ease of manipulation of the model.

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