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Neutronic analyses of the preliminary design of a DCLL blanket for the EUROfusion DEMO power plant

Iole Palermo*, Iván Fernández, David Rapisarda, Angel Ibarra

CIEMAT, Fusion Technology Division, Avda. Complutense 40, 28040-Madrid, SPAIN

In the frame of the newly established EUROfusion WPBB Project for the period 2014-2018, four breeding blanket options are being investigated to be used in the fusion power demonstration plant DEMO. CIEMAT is leading the development of the conceptual design of the Dual Coolant Lithium Lead, DCLL, breeding blanket. The primary role of the blanket is of energy extraction, tritium production, and radiation shielding. With this aim the DCLL uses LiPb as primary coolant, tritium breeder and neutron multiplier and Eurofer as structural material. Focusing on the achievement of the fundamental neutronic responses a preliminary blanket model has been designed. Thus detailed 3D neutronic models of the whole blanket modules have been generated, arranged in a specific DCLL segmentation and integrated in the generic DEMO model. The initial design has been studied to demonstrate its viability. Thus, the neutronic behaviour of the blanket and of the shield systems in terms of tritium breeding capabilities, power generation and shielding efficiency has been assessed in this paper. The results demonstrate that the primary nuclear performances are already satisfactory at this preliminary stage of the design, having obtained the tritium self-sufficiency and an adequate shielding.

Keywords: DCLL, Tritium breeding, Shield, MCNP

1. Introduction

Towards the development of a demonstration power plant DEMO during the design step is crucial the simulation of the fundamental function responses that allow to assess the behaviour of the reactor: tritium breeding ratio (TBR) is essential to determine if the reactor achieves the fuel self-sufficiency; power amplification and power distributions are fundamental to determine the reactor power efficiency and how the thermal load is deposited in the structures to give input for thermal-hydraulics and mechanical assessments; damage responses as helium production, displacement per atom (dpa), fluences and nuclear heating are very important to determine if the components are keeping their structural integrity or their functionality as for example the case of the Toroidal Field (TF) coil superconductivity. The primary nuclear requirements and performances studied in this paper to demonstrate a reliable operation of the DEMO fusion power plant are summarized in table 1.

Table 1. Primary Nuclear Responses under assessment

BB parameters	value
Tritium Breeding Ratio	≥ 1.1 [1]
Energy Multiplication factor	As high as possible in the range 0.9-1.35 [2]
Design limit for TF-coil superconductivity	
Peak nuclear heating in winding pack	≤ 50 W/m^3 [1]

This paper is focused on the neutronic analysis of the Dual-Coolant Lithium Lead (DCLL) Breeding Blanket (BB) System, one of the 4 BB options conceived for the future European Power Plant based on the DEMO 2014 design assumptions [3][4] (i.e. 1572 MW and pulsed scenario).

The DCLL concept is basically characterised by the use of self-cooled breeding zones with the liquid metal LiPb serving as tritium breeder and as coolant for extracting the heat gained from fusion energy. From the first DCLL design [5] others have been conceived among power plant conceptual studies, and the Test Blanket Modules (TBM) ITER Programme. In USA many aspects of the DCLL concept have been studied and developed especially for ARIES and ITER [6] while in Europe, after the EU model C of the Power Plant Conceptual Studies (PPCS) of 2003 [7], it has been not dedicated more efforts to the improvement of this concept. Since 2009, based on the concepts proposed in such model C of the PPCS, a DCLL DEMO design and its Plant auxiliary systems [8] have been developed in Spain by CIEMAT. The main difference respect to the previous DCLL models was the BB segment structure: the Spanish approach consisted in a single continuous BB module instead of a multi-modular segment. Following the experience acquired on DCLL development, CIEMAT is currently leading the development of a DCLL BB among the EUROfusion Programme. The common specification for the 4 different BB systems consists of a Multi-Module-Segment (MMS) structure to facilitate the maintenance procedure.

author's email: iole.palermo@ciemat.es

A DCLL novel design has been developed for the new DEMO 2014 generic design (figure 1a) [3] as described in [9]. The Outboard (OB) equatorial module has been firstly developed in detail (figure 1b). Then, all the DCLL modules have been developed and tested into a specific DCLL segmentation (figure 1c) adapted to the new DEMO 2014 specifications. The details of the neutronic model and the procedure for its development are given in section 2. The results of the neutronic calculations are detailed in section 3.

2. Development and features of the neutronic design

Taking advantage of past experience concerning DEMO developments, a similar procedure has been adopted to obtain a detailed neutronic DEMO DCLL design. For the neutronic purposes, an 11.25º sector has been studied (figure 1a) exploiting the toroidal symmetry of the tokamaks. Each 11.25º sector is composed by 1 inboard (IB) blanket segment and 1 and half outboard (OB) segments. The CAD model of the OB equatorial module (figure 1b) has been simplified to create a detailed 3D neutronic design using MCAM software tools [10], which allows to reduce the complexity of the CAD models to a level compatible with the geometrical capabilities of the Monte Carlo transport code (simplification of sp-lines, elimination of little components and unnecessary details, completion of the model filling the void spaces, among others). The OB equatorial module has been then repeated to the rest of modules (figure 1c) adapting it to the specific features (i.e. dimensions, available space, shape, etc.) of each one. Similarly to the work done for the WCLL development [11], a BB segmentation made by 7 IB entire modules, 8 OB entire modules (7.5º) and 8 OB half modules (3.75º, to complete the 11.25º sector) has been chosen. The modules, adapted to the specific DCLL segmentation have been then introduced into the generic DEMO 2014 (figure 1a) to create a complete DCLL DEMO neutronic model (figure 1d).

The last step before the conversion to MCNP input has been to assign the materials to the components of the model. The components of the generic DEMO have been filled with the following materials:

- Vacuum Vessel/Shield: 80% austenitic steel SS316LN + 18% H_2O + 2%B
- Upper, Equatorial and Lower Ports: austenitic steel
- Divertor: 80% austenitic steel + 20% H_2O
- TF coil: Nb_3Sn + cryogenic steel + epoxy + bronze + Cu + He + vacuum
- Central Solenoid, PF coils: cryogenic steel

The materials compositions for the breeding modules structures are taken from the detailed design and summarized in table 2. For the whole segment, both IB and OB sides, the breeder zones are fully-described (the homogenization concerns only the helium collector and the manifold region or Back Supporting Structure, BSS). The composition for the Manifold/BSS zone is very dense, and should be an efficient shielding system. Furthermore, having an high LiPb content, a benefit is expected on the TBR due to the tritium produced also in this region. The thickness of each component of the BB system is also shown in table 2. The breeder zone occupies 64 cm in the OB side and 30 cm in the IB one.

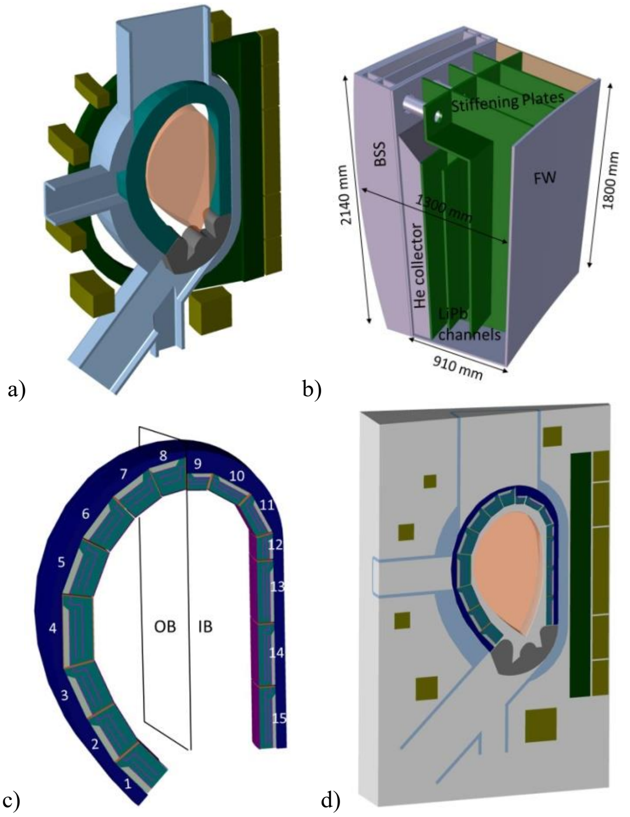


Figure 1: DCLL DEMO model development sequence using MCAM sofware a) generic DEMO model (in dark cyan colour the region available for Blanket and Manifolds); b) detailed OB equatorial module [9]; c) neutronic model of the blankets segment; d) complete DEMO2014 DCLL model

Once filled with material the model is ready to be converted *via* MCAM into the MCNP input. The minor conversions errors are then fixed up to reduce the number of lost particles during the transport (finally ~0.00018% of lost particles has been achieved). Particle transport calculation has been then performed with MCNP5 Monte Carlo code [12] and JEFF 3.1.1 nuclear data library [13]. For tritium production assessment ENDF/B-VII nuclear data library [14] has been also used having the comparison between libraries special relevance in the TBR prediction which usually has to account for different sources of uncertainties, as the case of the nuclear cross-section data uncertainties. Parallel computations have been carried-out in CIEMAT EULER cluster. The plasma neutron source was provided by KIT as a FORTRAN90 subroutine [15], sampling the neutron emission for the DEMO1 plasma according to the new plasma parameters [16]. Direct simulation results have been normalized to 5.581×10^{20} neutrons per seconds [n/s] source, corresponding to the 1572 MW fusion power.

Table 2. Thickness and composition of the components of the BB modules and of the Manifold/BSS

Components		Thickness (cm)	Radial Thickness OB (cm)	Radial Thickness IB (cm)	Composition (% vol)			
					Eurofer	He	LiPb	W
FW	FW armour	0.2	0.2	0.2				100
	FW	1.98	1.98	1.98	85.54	14.46		
Breeder zone + Helium collector	1st, 2nd and 3rd radial stiffening plates	(each one) 2	6	6	91.33	8.67		
	LiPb channels		64	30			100	
	He plena Eurofer walls		17	10	53	47		
Walls	Side walls	2	-		85.54	14.46		
	Top wall	4	-		85.54	14.46		
	Bottom wall	4	-		85.54	14.46		
	Back wall	2	2	2	85.54	14.46		
Total BB Thickness			91	50				
Manifold/BSS			variable thickness		51.29	4.35	44.36	

3. Results

3.1 Tritium production

The tritium production has been primarily evaluated because it represents the essential condition for the reactor viability.

The results, calculated with the ENDF/B-VII library, are presented in table 3, in which the tritium production rate (TPR) density, the local TBR per 11.25º modules, and the total per 360º modules and in manifold are shown.

The total TBR in the breeder modules is 1.041. Due to the relevance of the cross-section data uncertainties in the TBR prediction (among other factors) [1], the total value has been also calculated with the JEFF 3.1.1 cross section data, showing an increment of 0.095% (TBR=1.04165).

Nevertheless, the most important aspect to highlight is that the Manifold which contains 44.36% of LiPb contributes considerably to the TBR of the system. Such contribution amounts to 0.089 T/n in the whole reactor [0.0903 T/n (+1%) using JEFF] which implies an increase of the total TBR to 1.13 [1.13199 (+0.172%) with JEFF] fulfilling the self-sufficiency criterion (TBR $\geq$ 1.1) described in table 1. The specific contribution of the IB and OB sides of the Manifold is 1.79e-3 and 1.03e-3 T/n, respectively (considering an 11.25º sector). It means that the IB represents a 63.45% and the OB a 36.54% of the total tritium in the Manifold zone. This makes evident the relevance of the IB side of the Manifold because the less space occupied by the breeder allows high tritium breeding potential in the zone behind the modules, as shown in figure 2.

In this figure the values of Tritium production outside the breeder regions have not to be taken in consideration. In fact, the "mesh tally" tool of MCNP is able to calculate the nuclear responses only for one material each time. Thus, in this case, the values outside the LiPb breeder zone (i.e. inside the helium collector, the stiffening plates, the walls, etc.) are fictitious values as the components were made by PbLi, but referred to the spectra of the actual materials (e.g. helium, steel, etc.). The same assumption has to be considered also for the "mesh tallies" maps shown in sections 3.4 and 3.5.

Table 3. Tritium production in the BB modules (nº position in figure 1c) and in Manifold cells in terms of local TBR, total TBR and TPR density

Component	nº	T/n in 11.25º	Total in 360º	T/cm^{3}s
OB modules	1	2.34E-03		8.81E+11
	2	3.04E-03		9.53E+11
	3	3.56E-03		9.94E+11
	4	4.82E-03		9.92E+11
	5	3.48E-03		9.59E+11
	6	2.73E-03		9.37E+11
	7	2.03E-03		8.95E+11
	8	1.39E-03	0.749	8.37E+11
IB modules	9	7.99E-04		1.46E+12
	10	1.25E-03		1.44E+12
	11	1.13E-03		1.43E+12
	12	6.66E-04		1.45E+12
	13	1.93E-03		1.62E+12
	14	1.88E-03		1.58E+12
	15	1.48E-03	0.292	1.24E+12
Total BB			1.041/1.042 (ENDF/JEFF)	
Total Manifold (43 Cells)			0.089/0.09 (ENDF/JEFF)	
Total TBR (BB + Manifold)			1.13/1.132 (ENDF/JEFF)	

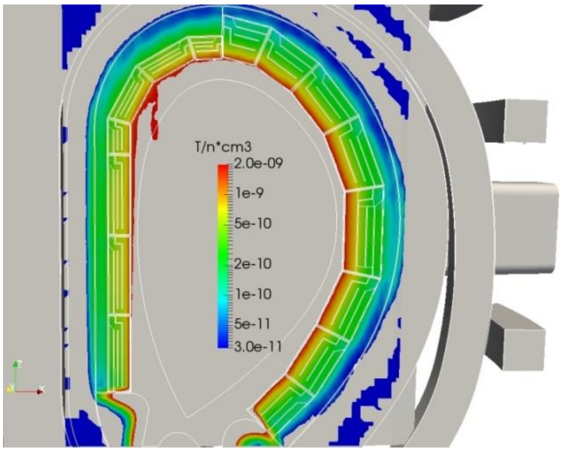


Figure 2. Tritium production as "mesh tally" (in T/n per cm^3)

3.2 Neutron wall loading

Once assured the first neutronic fundamental requirement on the TBR, additional assessments have been performed as the Neutron Wall Loading (NWL) calculation. The Neutron Wall Loading is the rate at which neutrons transfer kinetic energy through the first wall. Its poloidal distribution allows seeing the regions in which a special care for shielding could be considered. Such poloidal distribution of the NWL along the first wall is presented in Figure 3 where a mean value of 1.033 MW/m^2 is also shown. As expected, a strong poloidal variation is observed with two peaks at the equatorial level (modules nº4 and nº13).

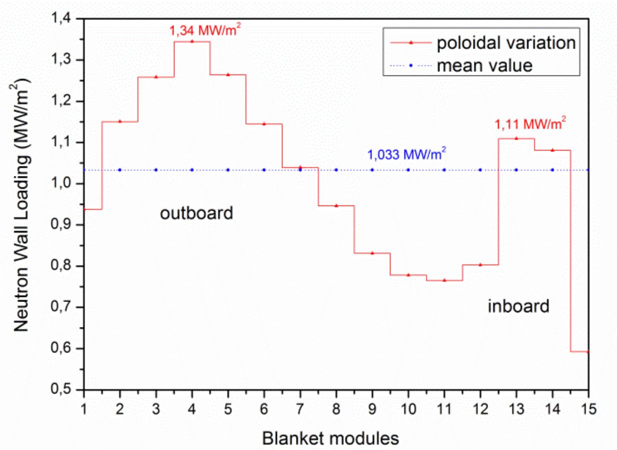


Figure 3. Neutron wall loading poloidal distribution on the FW

3.3 Power generation and Energy multiplication

The power breakdown for the major reactor structures is shown in table 4. Considering the total generated nuclear power of 1503 MW, the obtained energy multiplication factor M_E is 1.195, being M_E the ratio of the total nuclear power over the fusion neutron power (80% of the 1572 MW of fusion power). The result indicates the good potential of the thermal recovery system and hence reduced electric power production costs.

Table 4. Power breakdown along the components of the reactor

Component	Power generated (MW)
BB + Manifold	1229.32
Divertor	262.49
VV + Ports + Coils	11.98
Total	1503.79

3.4 Radial distribution of the Power deposition

The power density has been assessed as radial distribution from the FW to the Manifold, for the OB equatorial module, as shown in table 5, and a greater refinement in the specification of the results has been pursued. In fact, all the plates that constitute the stiffening grid have been singly analyzed (figure 4a) and the power deposited in all the walls and in a large numbers of LiPb positions have been determined (figure 4b).

Table 5. Radial distribution of the nuclear heating (W/cm^3) along the components of the DCLL OB equatorial zone

components OB equatorial module		JEFF 3.1.1 MeV/gr	uncert.	W/cm^3
walls	FW W armour	1.55E-08	0.0006	26.624
	FW Eurofer	1.17E-08	0.0005	6.964
	top wall	1.46E-09	0.0023	0.873
	bottom wall	1.42E-09	0.0023	0.849
	side wall	1.76E-09	0.0021	1.052
	side wall	1.75E-09	0.0018	1.046
LiPb	LiPb radial1	3.58E-09	0.0007	3.044
	LiPb radial2	9.42E-10	0.0018	0.801
	LiPb radial3	4.76E-10	0.0029	0.405
	LiPb up near	7.56E-10	0.0036	0.643
	LiPb middle near	6.74E-10	0.0041	0.573
	LiPb down	6.80E-10	0.0039	0.578
	LiPb up far	2.63E-10	0.0063	0.224
stiffening grid	plate 1	1.43E-09	0.002	0.911
	plate 2	7.91E-10	0.0038	0.504
	plate 3	2.93E-10	0.013	0.187
	plate 4	2.08E-10	0.0076	0.132
	plate 5	1.17E-10	0.0119	0.075
	plate 6	9.94E-11	0.0267	0.063
	plate 7	7.02E-10	0.0012	0.447
	He collector	1.07E-10	0.0047	0.040
	back wall	7.15E-11	0.0075	0.043
	manifold inner wall	2.14E-10	0.0059	0.161
	manifold block	8.20E-11	0.0025	0.062

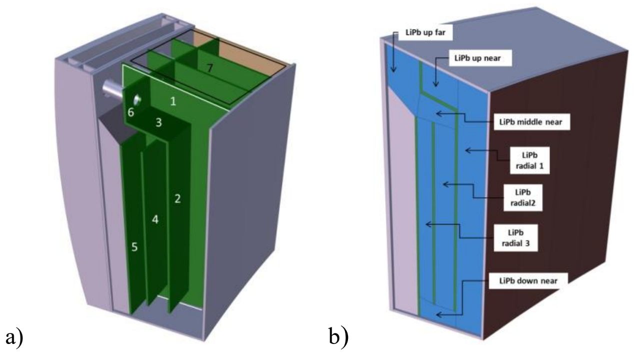
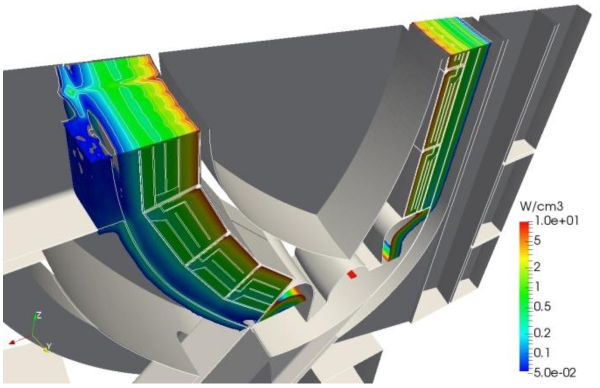
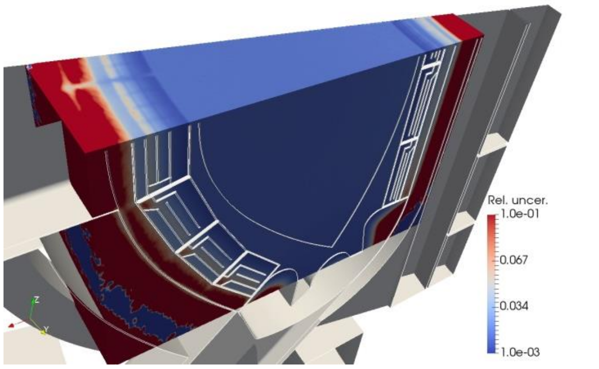


Figure 4. a) Numbers of the stiffening plates and b) names of the LiPb volumes of the OB equatorial module in which the nuclear heating has been calculated, as resumed in table 5

In addition, the power deposition map in LiPb calculated through the "mesh tally" capabilities of MCNP has been also implemented, as shown in figure 5a. In figure 5b the relative uncertainty given by the MCNP stochastic method is presented. All the values of interest have relative uncertainties less than 6%.



a)



b)

Figure 5. “Mesh tally” 3D maps of a) Power density (W/cm^3) in LiPb and b) its uncertainty given as relative values

3.5 Nuclear heating in the TF coil

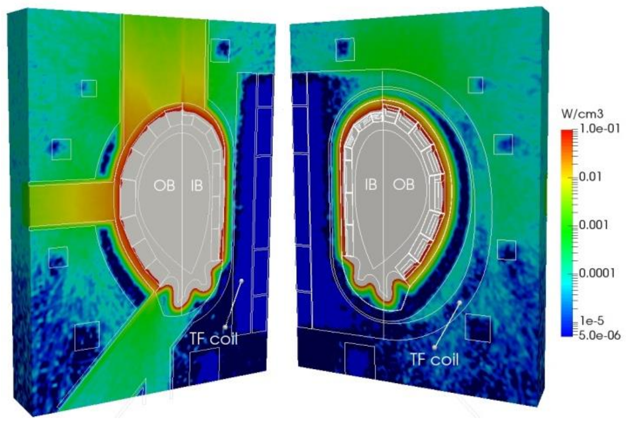
In order to allow a preliminary evaluation of the shielding efficiency of the DEMO radial build, the nuclear heating in the reactor needs to be assessed, paying special attention to the values on the TF conductor at inboard equatorial level.

Table 6. Nuclear heating in TF coil

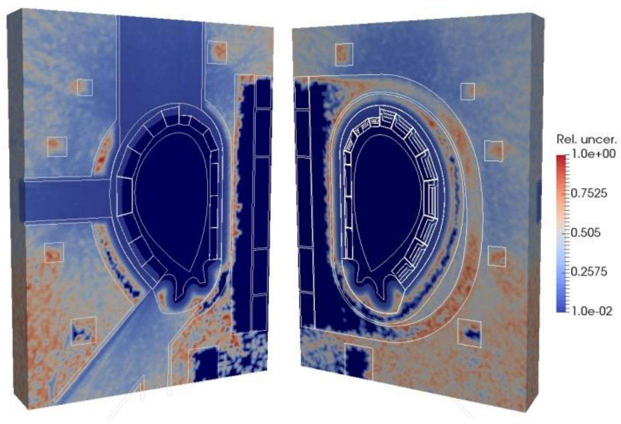
Distance from plane Z=0 (cm)	IB			OB		
	MeV/gr	relative uncert.	W/m^3	MeV/gr	relative uncert.	W/m^3
>160	1.80E-13	0.010	88.57	5.88E-13	0.007	288.83
160:110	9.08E-16	0.200	0.45	1.51E-13	0.053	74.32
110:60	1.42E-15	0.323	0.70	1.99E-13	0.045	97.70
60:10	2.39E-15	0.350	1.17	2.32E-13	0.040	113.73
10:-40	3.32E-15	0.280	1.63	2.51E-13	0.041	123.45
-40:-90	3.15E-15	0.383	1.55	2.27E-13	0.043	111.66
< -90	9.60E-16	0.113	0.47	4.80E-14	0.022	23.57

The IB equatorial values satisfy the recommendation for the nuclear heating currently established (table 1) in 50 W/m^3 (20 times lower than the ITER analogue requirement), except for the global zone above the plane at z=160 cm due to the presence of the Upper port. On the other hand, the limit is not satisfied for the central/upper part of the OB side, due the presence of both the Equatorial and Upper ports. The lack of shield in these zones (both the upper IB and most of the OB) is not of concern because the port plugs have not been already included in the generic DEMO design and it is not a question of the specific blanket design. If the IB equatorial zone is well protected we can assume that the rest of the IB and the OB sides will also be well shielded when the plugs will be included in the generic DEMO design.

Detailed 3D maps are also given (figure 6a) showing the same behaviour explained before: the limit for the TF coil superconductivity is fulfilled for values lower than 5x10^{-6} W/cm^3 (the lower limit of the scale given in blue colour) as occurs in the IB leg of the TF coil (and in darker parts which are out of the scale). Where the limit is not fulfilled (values warmer than blue, as for example in the green regions of the OB central/upper part of the TF coil) it is due to the presence of the open Ports, in fact a strong streaming can be observed in these.



a)



b)

Figure 6. Mesh tally” 3D maps of a) Nuclear heating (W/cm^3) calculated in the material of the TF coil and b) the relative uncertainty

The relative uncertainty given by MCNP for the Nuclear Heating values is shown in figure 6b. High uncertainties can be observed in very much shielded regions far from the plasma because of the very low values obtained in there. Otherwise, uncertainties less than 1% have been obtained in most of the IB side of the TF coil. The MCNP variance reduction technique of importances has been used to pursue this objective.

4. Conclusions

Preliminary neutronic analysis have been performed to support the design of a new DCLL breeding blanket concept, for the development of the newly established pulsed European DEMO reactor. As general results, a TBR of 1.13 has been achieved, thanks to the BSS design, and an average NWL of 1.033 MW/m^2 and a M_E of 1.19 have been obtained. Shielding performances have been also assessed, demonstrating that the DCLL fulfils the current limit of nuclear heating in the TF coils established for DEMO (50 W/m^3). Other results have been also obtained as the radial/poloidal profile of the tritium production and different distributions of the nuclear heating in order to give inputs to the mechanical, thermal, safety and tritium modeling activities needed to upgrade the design. Further analyses are also ongoing [17] to establish if other structural and damage criteria are also observed.

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