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Nuclear analysis of the Single Module Segment WCLL DEMO

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The latest design of the Water Cooled Lithium Lead blanket for the European DEMO reactor is based on a Single Module Segment (SMS) concept for the blanket segments: the removal of the gaps between the modules (present in the former Multi Module structure design), with the consequent reduction of the neutron streaming, improves the tritium production and the shielding capabilities. A nuclear analysis has been carried out with the Monte Carlo N-Particle transport code version 5 (MCNP5) and the Joint Evaluated Fission and Fusion nuclear data libraries version 3.2 (JEFF 3.2). A detailed three-dimensional MCNP DEMO model with the Single Module layout has been generated. Three-dimensional neutron and gamma transport simulations have been carried out in order to asses the tritium production, nuclear power deposition in each subcomponent of the blanket and nuclear heating density, neutron flux, neutron damage (dpa) and Helium production, relevant for the design of the system. The results confirm the fulfilment of tritium self-sufficiency and shielding requirements of DEMO. Furthermore, it has been carried out a parametric analysis by varying the thickness of the first wall tungsten layer to evaluate how the blanket performances are influenced by the tungsten layer thickness.

Keywords: DEMO, WCLL, neutronics, nuclear, shielding, TBR, MCNP

1. Introduction

The Water Cooled Lithium Lead (WCLL) blanket is a candidate for the European DEMO nuclear fusion reactor blanket. The structural material is EUROFER, the liquid Lithium-Lead (PbLi) ^{6}Li enriched at 90% is used as breeder, neutron multiplier and tritium carrier. Water is used as coolant at an inlet/outlet temperature of 285 °C and 325 °C, respectively, and pressure of 15.5 MPa [1]. The aim of this paper is to compare the nuclear performances of the new WCLL blanket layout based on a Single Module Segment (SMS) concept with the DEMO blanket requirements [2]. Three-dimensional analyses have been performed with MCNP5 Monte Carlo code [3] and JEFF 3.2 nuclear data libraries [4] to determine the tritium self-sufficiency, shielding capabilities and to assess the nuclear loads on the first wall, breeding blanket, supporting structures and manifolds.

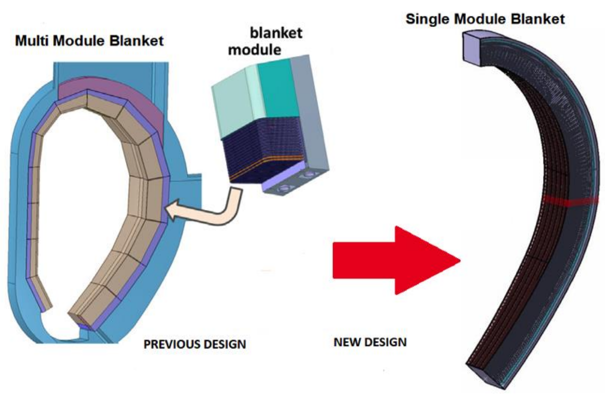

Fig. 1. Comparison between the previous WCLL Blanket layout (Multi Module) and the new one (Single Module).

The new SMS blanket design allows to solve some drawbacks identified in the former Multi Module Segment (MMS) approach [5] (e.g. PbLi drainage and He removal from the breeding zone) as well as to facilitate the manifolds integration, and to improve the TBR and shielding performances. A comparison between the two designs is shown in Fig. 1.

The DEMO reactor layout used in the present studies is the 2015 DEMO1 baseline configuration with plasma parameters shown in Tab. 1 [6].

Table 1. Main parameters of the DEMO baseline configuration.

No. of toroidal field coils	18
Major radius (m)	9.072
Minor radius (m)	2.927
Aspect ratio	3.1
Plasma elongation	1.59
Plasma triangularity	0.33
Fusion power (MW)	2037
Average neutron wall loading (MW/m^2)	1.05
Net electric power (MW)	500

The generic DEMO model [7] has been upgraded, integrating the SMS blanket design and the verification of the following nuclear requirements has been carried out:

- A minimum Tritium Breeding Ratio (TBR) of 1.1;
- A Neutron cumulated damage over 6 full power years (FPY) in the Vacuum Vessel (VV) steel below 2.75 dpa, preventing the degradation of the stainless steel physical and mechanical properties;
- A He production cumulated over 6 FPY below 1 appm in re-weldable zones of the VV; and
- A fast neutron fluence to the superconductor coils and a nuclear heating due to neutrons and gammas deposited on the winding pack below the limits of 10^{18} n/cm^2 and 0.05×10^3 W/m^3, respectively.

Finally, the effect of the First Wall (FW) W-layer thickness variation on the blanket Tritium production and Shielding capabilities has been analysed.

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2. Single Module design and WCLL MCNP DEMO model

The latest DEMO WCLL BB layout, is based on a novel Single Module Segment (SMS) concept where a basic breeding unit (BU) is replicated along the poloidal direction, filling the available space in inboard and outboard (Fig. 2) [8]. The single BU includes the FW and side walls, top and bottom caps, internal stiffening and baffle plates, back supporting structure (BSS), cooling pipes, LiPb manifolds, water manifolds for FW and breeding zone cooling. In Table 2, the reference radial thicknesses for each component of the inboard and outboard BUs are shown. The BU cooling pipes are characterised by a complex helical path in the area close to the FW thus, in order to obtain a reliable equivalent MCNP model, an approach based on the segmentation of the BB into radial sectors with specifically defined material mixture has been chosen [9].

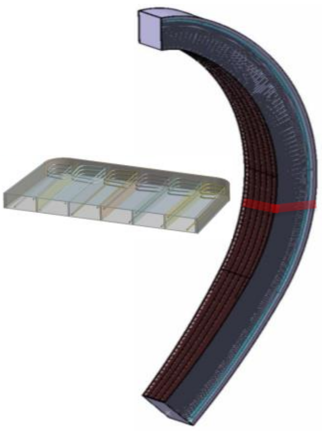

Fig. 2. WCLL SMS: overview of the outboard module and detail of a single breeding unit (highlighted in red).

A detailed MCNP model of the WCLL SMS has been developed and manually integrated into the DEMO 1 generic MCNP model: this procedure required the modification of the inboard and outboard segments, that have been respectively fused in a single 'banana-shaped' breeding zone. The resulting SMS WCLL DEMO MCNP model is shown in Fig. 3: the reduced radial extension of the inboard module has been handled through the removal of the innermost sectors of the breeding zone, thus, ideally, keeping the distance between the FW and the helicoidal cooling channels fixed.

Table 2. Radial thicknesses for each component of the outboard and inboard BUs.

Component	Radial thickness outboard (cm)	Radial thickness inboard (cm)
W - Armour	0.2	0.2
FW	2.5	2.5
BZ	80	47
LiPb Manifolds	14	7.8
BP	3	1.7
Water Manifolds	20	11.2
BSS	10	5.6

3. Nuclear analysis of the SMS blanket

The model described in section 2 has been used to evaluate the WCLL DEMO blanket performance. Results have been normalized to 2037 MW fusion power (neutron yield: 7.2×10^{20} n/s), according to the plasma parameters specified in Table 1. Standard cell-based (F4, F6, F1, F2 tallies) and mesh tallies (FMESH tally) with proper multipliers have been used to calculate the nuclear quantities of interest.

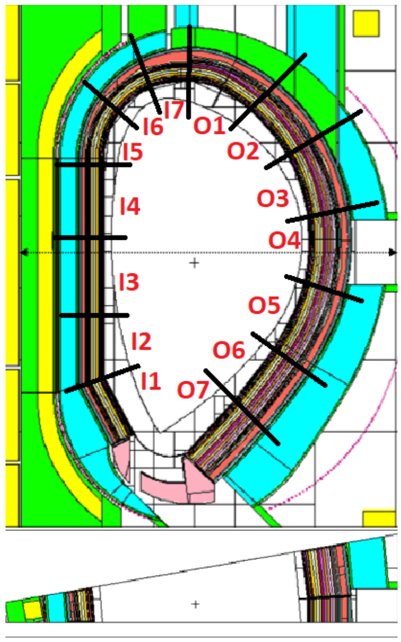

Fig. 3. SMS WCLL DEMO MCNP model: poloidal section showing the inboard and outboard BB (top) and toroidal section along the equatorial plane (bottom). In the poloidal view is shown the partition in sectors used for the analyses about NWL and TBR.

3.1 Neutron Wall Loading

The poloidal distribution of the Neutron Wall Loading (NWL) has been estimated maintaining the same poloidal segmentation used in the WCLL MMS DEMO model [5] (i.e.: subdividing the inboard and outboard breeding blanket with the same surfaces that defined the blanket modules in the Multi Module configuration (Fig. 3), applied to the associated tallies). Results are shown in Fig. 4: the maximum values are in the outboard and inboard equatorial zone (1.35 MW/m^2 and 1.11 MW/m^2 respectively), while the poloidal average is 0.97 MW/m^2.

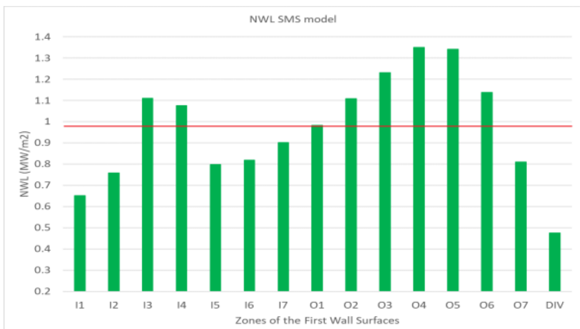

Fig. 4. Neutron Wall Loading poloidal distribution (results are normalized to 2037 MW of fusion power)

3.2 Tritium Breeding Ratio

The TBR has been calculated using track-length (F4 tally) with proper tally multipliers (FM card) for each sector that contributes to the Tritium generation (e.g. the BU sectors and the LiPb manifolds) have been considered, taking into account the neutron capture reactions on both the Li-6 and Li-7 isotopes. The resulting total TBR for this configuration is 1.131 t/n, above the DEMO design target. About 99% of the tritium is produced in the breeder zone, while the residual portion is generated in the manifold zone.

The poloidal distribution of the TBR is shown in Fig. 5, using the same segmentation surfaces as for the NWL in Fig. 3. The highest TBR is obtained in the outboard equatorial zone and the cumulative contributions of outboard and inboard blankets are 71% and 29% respectively.

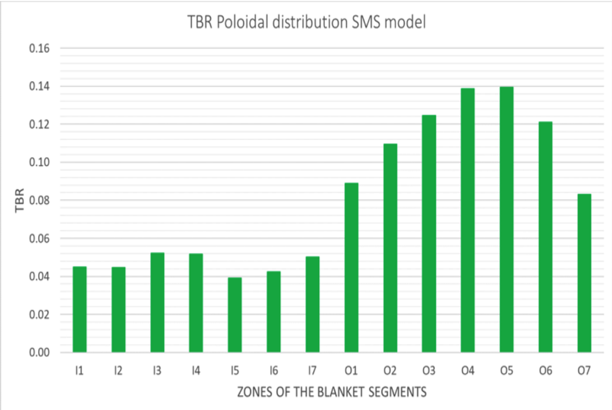

Fig. 5. TBR (t/n) poloidal distribution.

With respect to the MMS concept [5], the total TBR is slightly increased (+0.4%), thus, a dedicated study has been carried out in order to investigate the individual contribution of the BB/manifold layers on the TBR. Figure 6 shows the cumulated TBR for each radial breeder layer of the SMS configuration.

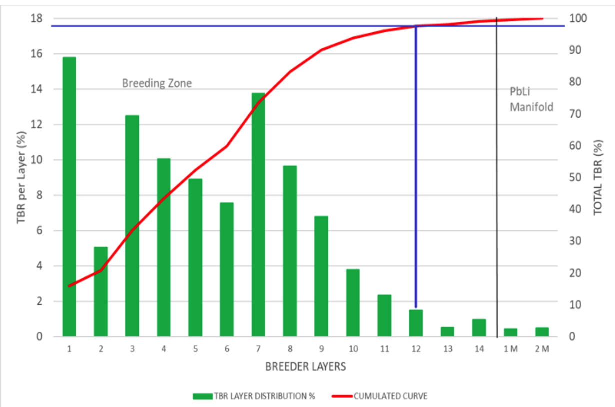

Fig. 6. Comparison between TBR (%) distribution for each breeder layer and the cumulated curve, the horizontal blue line indicates the target DEMO TBR.

The layer that provides the most significant tritium breeding in the outboard region is the first one (16%) and the relative contributions decrease progressively through the outer layers. Typically, 98% of the global TBR is produced in the first twelve breeding layers (about the first 36 cm of the BU). Thus, tritium self-sufficiency could be achieved even reducing the outboard breeding blanket radial extension by about 15 cm: this result is particularly interesting in the frame of the DEMO reactor design development, because its new baseline concept [10], foresees a reduced outboard radial extension for the breeding zone.

3.3 Nuclear Power deposition

The power generation due to neutrons and secondary gammas in the WCLL blanket components has been calculated using energy deposition tally (i.e. F6) on all the system elements. The energy multiplication factor is 1.196, corresponding to 1949 MW. About 93% of the overall nuclear power (1808 MW), is deposited on the breeding blanket/manifold/BSS system. The nuclear power deposited on the outboard segment is about 70% of the total.

3.4 Shielding performances

Neutron flux, nuclear heating, damage in terms of dpa, and He-production in steel components have been calculated along the inboard and outboard mid-plane. The nuclear quantities are averaged on a poloidal extension of 50 cm (from z=10 to z=60 mm), thus considering the detailed breeding blanket description. The FMESH tally feature of MCNP has been applied to evaluate the nuclear responses. The total and fast (E>100 keV) neutron fluxes radial profiles are shown in figures 7 and 8 for the inboard and outboard respectively, while in figures 9 and 10, 3D maps of the total neutron flux in the inboard and outboard equatorial area are reported.

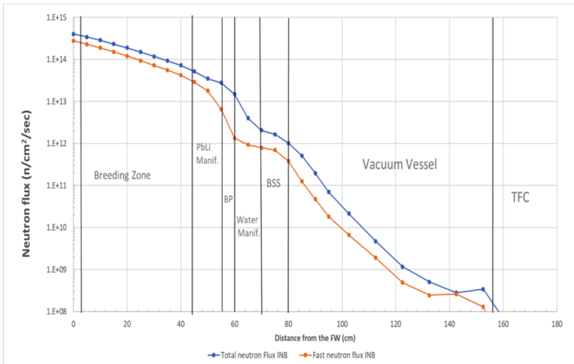

Fig. 7. Inboard radial profile of the total (blue) and fast (brown) neutron flux.

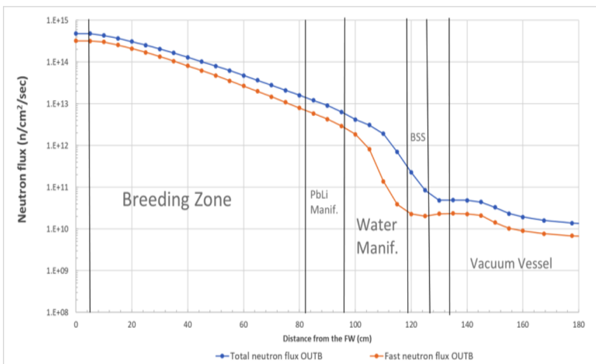

Fig. 8. Outboard radial profile of the total (blue) and fast (brown) neutron flux.

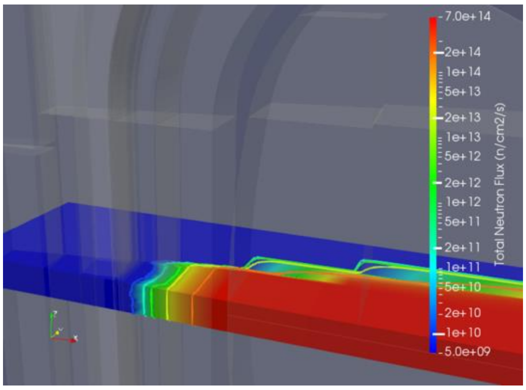

Fig. 9. 3D map of the total neutron flux in the inboard midplane.

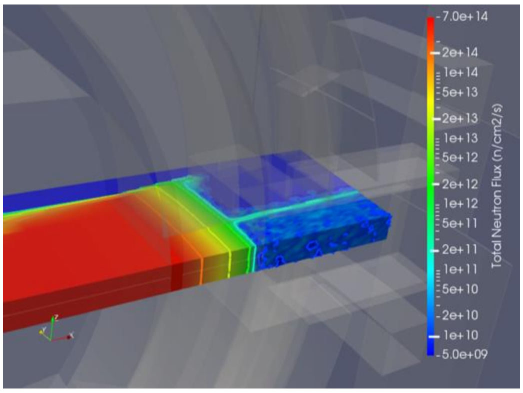

Fig. 10. 3D map of the total neutron flux in the outboard midplane.

The total neutron fluxes at the inboard and outboard FW are 4.1x10^{14} n/cm^2/s and 4.8x10^{14} n/cm^2/s, respectively. The inboard blanket/manifold system reduces the flux to the VV inner shell by more than two orders of magnitude. The neutron flux further decreases by several orders of magnitude across the VV, being 3.4x10^8 n/cm^2/s (total), and 1.3x10^8 n/cm^2/s (fast) on the TF, well below the 10^9 n/cm^2/s design limits. On the FW the fast neutron flux contributes for about 67% to the total, on the VV inner shell this contribution is reduced to about 24% for the inboard and 48% for the outboard. The radial profiles of the nuclear heating density evaluated in Eurofer and LiPb (blanket) and SS316L (Vacuum Vessel and TFC) is shown in figures 11 and 12 for the inboard and outboard respectively.

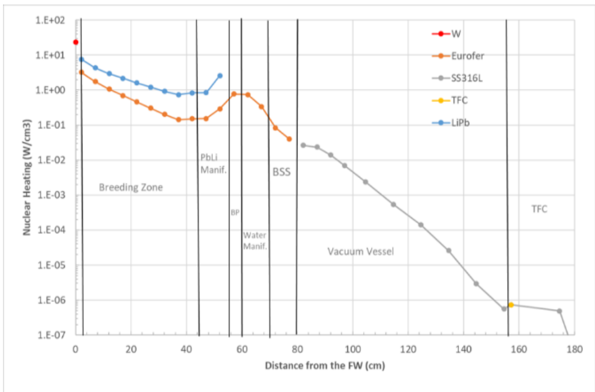

Fig. 11. Inboard nuclear heating radial profile in W-armour, Eurofer, LiPb and SS316L.

For the inboard, the maximum values are 23.4 W/cm^3 on W-Armour and 3.2 W/cm^3 on Eurofer in the FW and 0.026 W/cm^3 on VV inner shell. For the outboard segment, the same values are respectively 26.8 W/cm^3, 8.40 W/cm^3 and 2.40x10^{-3} W/cm^3. In the TF coil inboard leg, the nuclear heating density is 7.3x10^{-7} W/cm^3, well below the design limit of 5x10^{-5} W/cm^3. This result confirms that the shielding capabilities of the BB system ensure sufficient protection of the TF coils.

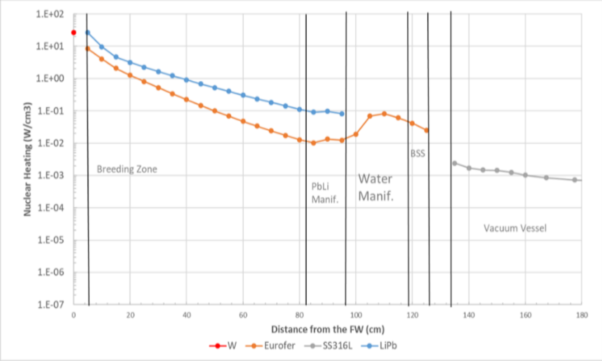

Fig. 12. Outboard nuclear heating radial profile in W-armour, Eurofer, LiPb and SS316L.

The radiation damage has been assessed in terms of dpa per FPY (dpa/FPY) produced in the structural steel of the blanket (Eurofer) and in the SS316L steel of the Vacuum Vessel. The radial profiles in equatorial area are provided in figures 13 and 14 for the inboard and outboard segment respectively.

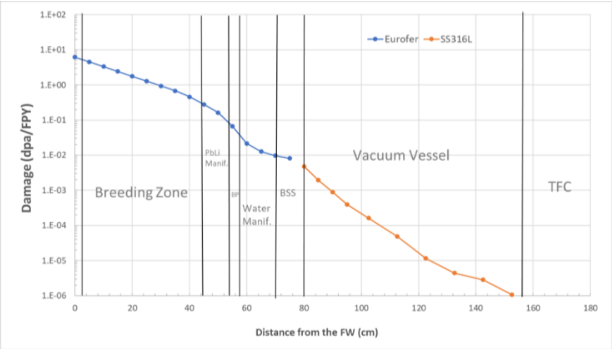

Fig. 13. Radial profile of the damage on Eurofer (up to the manifolds) and SS-316, evaluated in the inboard midplane.

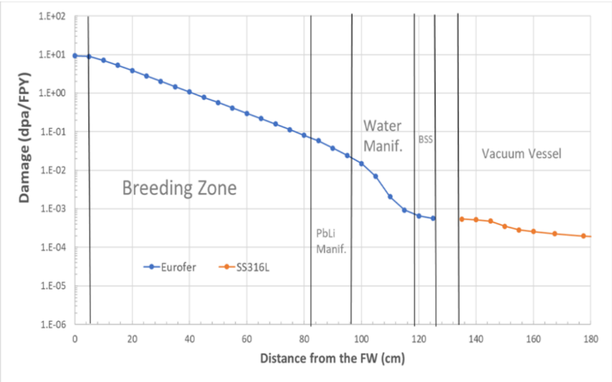

Fig. 14. Radial profile of the damage on Eurofer (up to the manifolds) and SS-316, evaluated in the outboard midplane.

In the inboard, the maximum damage in the Eurofer FW is 6.2 dpa/FPY and it decreases to 4.8x10^{-3} dpa/FPY in

VV inner shell. As far as the outboard segment is concerned, the same maximum values are 9.3 dpa/FPY $5.4x10^{-4}$ dpa/FPY on the FW and on the VV respectively. Considering the design limit of 2.75 dpa integrated over 6 FPY, the cumulated damages over the DEMO lifetime in inboard and outboard VV inner shell are respectively 0.03 dpa and 0.003 dpa: in both cases the target limit is fulfilled with a large margin.

As far as the He-production in steel is concerned (Fig. 15), the estimation performed on the inboard area highlights a reduction of about 2 orders of magnitude from the First Wall (100 appm/FPY) to the Vacuum Vessel (1 appm/FPY). Behind the Vacuum Vessel the He-production drops to values below 10^{-4} appm/FPY. The Helium production is assessed in terms of appm per FPY (appm/FPY) produced in the structural steel of the blanket (Eurofer) and in the SS316L steel of the Vacuum Vessel. The aim of this calculation is to verify that the Helium accumulations at locations where re-welding of steel components is required, is lower than 1 appm at the end of DEMO lifetime. The most critical locations in terms of steel re-weldability are the vessel zones (e.g. close to vessel ports). The He-production radial profiles evaluated in the equatorial area are shown in figures 15 and 16 for the inboard and outboard respectively. In the inboard, the He-production value in the FW Eurofer is 52 appm/FPY and it decreases to $7.6x10^{-2}$ appm/FPY in the VV inner shell, and less than 10^{-6} appm/FPY on TFC. For the outboard module, the maximum in FW Eurofer is 120 appm/FPY and in VV inner shell is $7.5x10^{-3}$ appm/FPY. Considering the VV lifetime of 6 FPY the maximum cumulated He-production is 0.60 appm, the limit of 1 appm is fulfilled. However, it's ought to be noted that this value refers to a zone of the VV where there are no ports or in vessel components penetration. Thus, the analysis should be repeated in VV areas near ports where the shielding could be reduced due to the presence of diagnostics and control systems [11]. Both the radial profiles plots exhibit a peak in back plate zone. This is due to the large presence of water in the manifold layers (with around 93%Vol. of water), which induces a softer neutron spectrum that enhances the He-production in adjacent zone through the reactions with ^{10}B. The same peaking factor can be also observed in the nuclear heating profile (Fig. 12 and Fig. 13), because the softer neutron spectrum causes an increase of the gamma generation in adjacent steel.

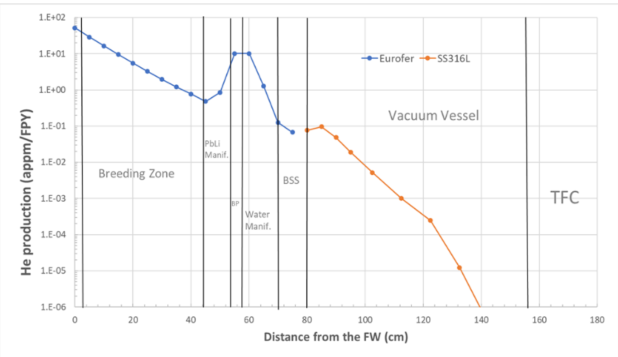

Fig. 15. Radial profile of the Helium production on Eurofer (up to the manifolds) and SS-316, evaluated in the inboard zone.

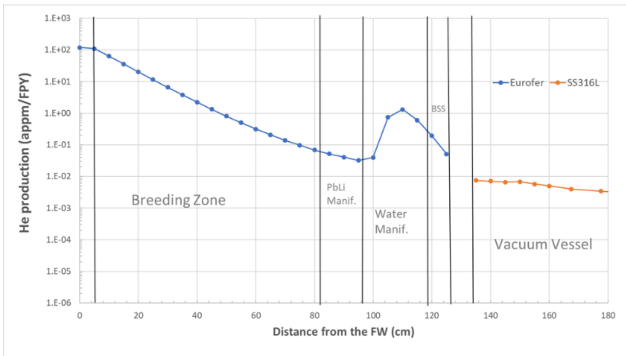

Fig. 16. Radial profile of the Helium production on Eurofer (up to the manifolds) and SS-316, evaluated in the outboard zone.

4. Sensitivity studies of the SMS performance

In this section a sensitivity study of the SMS blanket results is presented. In particular, the effect of the armour thickness on the blanket performances has been assessed. Moreover, additional calculations have been performed in order to understand the difference in the performances of the SMS WCLL DEMO design with previous MMS 2015 [5] configuration due to the difference in structural material/coolant and in the segmentation.

4.1 Effects of the W – Armour thickness

The effect of the tungsten armour layer thickness on the TBR and shielding properties of the blanket has been analysed in order to check that a reasonable compromise between the integrity of the first wall and the tritium breeding can be found on a sufficiently wide domain corresponding to different assumptions on the particle loads on the armour. The actual W-Armour thickness is 0.2 cm: for this analysis five different MCNP DEMO WCLL SMS models have been developed, with five different W-Armour thicknesses (0.001 cm, 0.05 cm, 0.1 cm, 0.7 cm, 1.2 cm), with the purpose to determinate the new TBR values and the shielding parameters. In this study, for each model the blanket layout has not been modified; the variation of the thickness of the W - armour has gone to the detriment of the vacuum chamber, where the plasma is present. The TBR dependence on the thickness is shown in Figure 17. Note that the TBR increases of 2.45% in comparison to the nominal value for the case of no W armour (corresponding to a thickness reduction of 99.5%). On the contrary with a thickness larger than 0.7 cm, the TBR decreases below the DEMO design limit of 1.10, until a TBR reduction of 3.74% with a thickness of 1.2 cm (+500% than the nominal value).

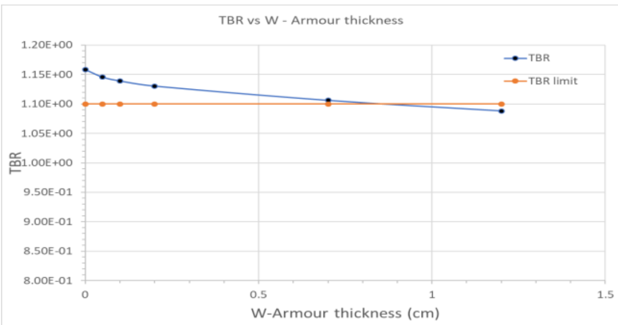

Fig. 17. TBR values versus W – Armour thickness, compared with the TBR limit value of 1.10.

Whit thickness reductions of 75% and 50% the TBR increases respectively of 1.34% and 0.77%; on the other side with a thickness growth of 250% there is a TBR decrease of 2.15%.

The neutron damage dependence on the W thickness is shown in Figures 18 and 19.

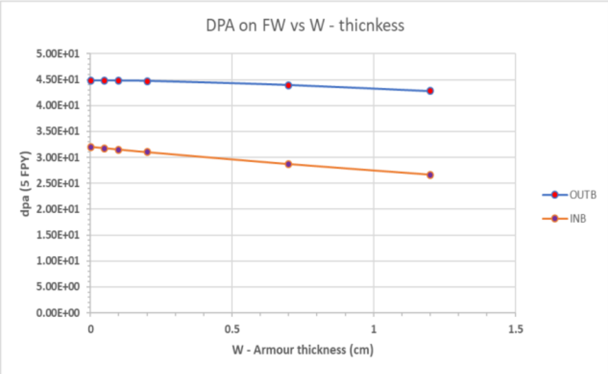

Fig. 18. Neutron damage values (dpa) on FW cumulated on 5 FPY with W – Armour thickness, for outboard and inboard segment.

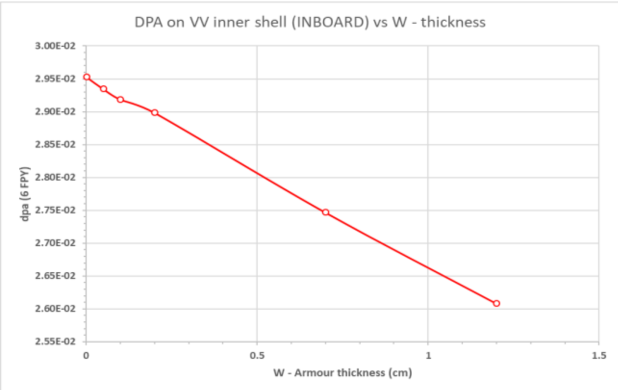

Fig. 19. Neutron damage values (dpa) on VV inner shell cumulated on 6 FPY with W – Armour thickness, only for inboard segment.

The design limit value is 2.75 dpa on austenitic stainless steel for VV structure, integrated respectively on an operational lifetime of 6 FPY for the VV. This limit is broadly always respected for every value of thickness on the VV. Reducing the thickness below the design value does not significantly alter the damage level (for the outboard FW: +1.65%, +2.47%, +3.30% whit thickness reductions respectively of 50%, 75%, 99.5%. For the inboard FW the same percentages are: +0.13%, +0.19%, +0.20%, and for the VV: +0.70%, +1.24%, +1.88%) Increasing the thickness, the neutron damage decreases appreciably, with a reduction with respect to the actual value, of about 7.42% for outboard FW, 1.75% for inboard FW and 5.24% for VV, for a thickness of 0.7 cm and in the same order 14%, 4.4% and 10% for a thickness of 1.2 cm.

Figure 20 shows the Helium production in the re-weldable zones of the VV. The He-production in these areas, cumulated on 6 FPY, must be lower than 1 appm, to allow the re-weldable of the stainless steel, if necessary. This is largely satisfied for every value of thickness. In this case the trend is similar to the damage; whit the thickness reduction, there is a contained increase of the He – production: +2.51% (-50%), +3.91% (-75%), +3.12% (-99.5%). Increasing the thickness, the following reductions have been obtained: -3.12% (+250%), -8.64% (+500%).

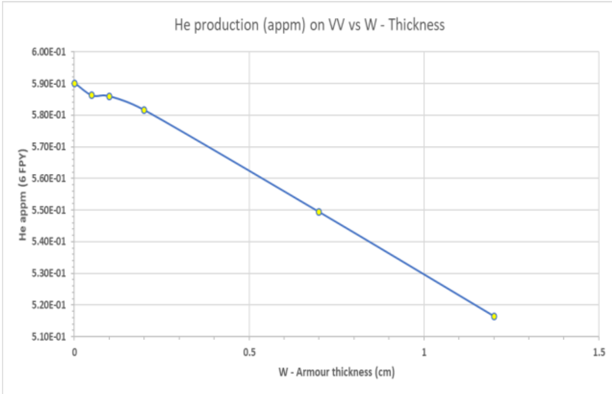

Fig. 20. Helium production (appm) on the VV inner shell cumulated on 6 FPY vs W – Armour thickness.

In conclusion, the analysis shows that an increase of the thickness above 0.7 cm produces a reduction of the TBR, below the design limit of 1.10. The shielding performances are almost independent of the thickness, especially for values below 0.7 cm.

4.2 Comparison between the MMS and the SMS design

In order to compare the new SMS WCLL DEMO design with respect to the previous MMS 2015 configuration [5], three additional configuration of the WCLL BB have been developed:

- "MMS 2015" with a homogeneous model (rather than the layered structure) for each module, as in [8].
- "Full-breeder SMS" where all the breeding layers have been filled with LiPb only; and
- "Homogeneous SMS" where the internal layout of the segment has been filled with the same compounds of the MMS 2015 with the aim of recreating a SMS model with the internal design of the previous MMS model.

The main results of the comparative studies are summarized below.

• *Combined impact of design and segmentation*. The SMS blanket shows a sensible reduction of the neutron flux compared to the MMS 2015 one both on the VV (neutron flux on the VV inner shell: -81%) and on the TFC (nuclear heating: -93%, fast neutron flux: -60%) as shown in Figure 21 The improvement of shielding capabilities in the new model is due to the variation in the blanket design and in particular to the BSS configuration (characterized by a higher content of water in the water manifold) and to the removal of toroidal gaps. The 0.4% increase in the global TBR of the SMS model with respect to the MMS 2015 (Figure 22 and Table 3) is less than what would be expected from the removal of the toroidal gaps only. This modest enhancement is mainly due to the lower amount of LiPb in the SMS [8] layout.

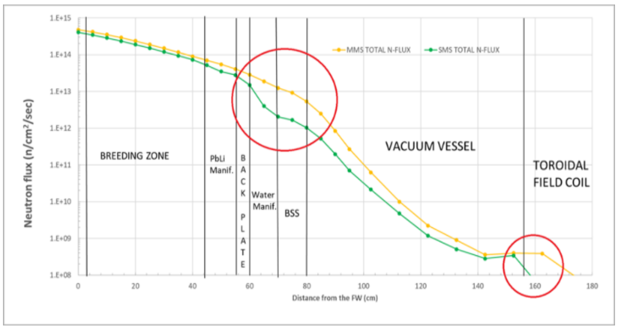

Fig. 21. Radial profiles of the total neutron flux: SMS vs MMS blanket concept.

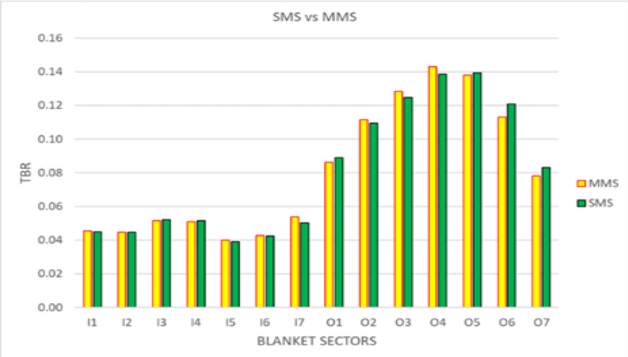

Fig. 22. TBR poloidal distribution: SMS vs MMS WCLL blanket concept.

• *Impact of structural material and coolant on TBR performances*. The global TBR for the "full-breeder" SMS configuration is 1.194 (i.e. only 5.6% larger than the SMS reference model), although the LiPb content is considerably higher (1026 m^3 for the "full-breeder" SMS vs 845 m^3 for the SMS reference design, +21%). This result shows that the lower amount of breeder material is largely compensated by the effect of neutron moderation in Eurofer and water that soften the neutron spectrum increasing the captures on ^6Li. Furthermore, in the SMS reference model most of tritium is produced in the first eight breeding layers (around 83%) and the remaining part is generated in the back layers (Figure 23). The "full breeder" SMS layout presents an opposite behaviour: the reason is attributable to the absence of Eurofer and water again, that reduces the tritium production in the first layers. Instead, in the back layers, the dominant effect is related to the content of LiPb, which is higher in the "full breeder" SMS model than the actual SMS design.

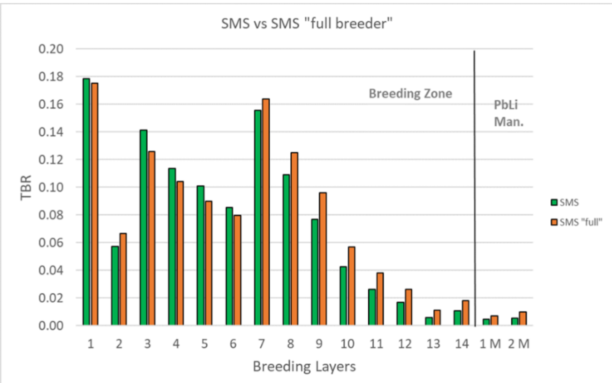

Fig. 23. TBR poloidal distribution: SMS vs "full-breeder" SMS BB.

• *Effect of blanket segmentation*. The comparison between the MMS 2015 and the "homogeneous" SMS layouts, characterized by the same homogeneous configuration but with different segmentation shows a larger TBR (by about 1.7%) for the SMS concept. The additional breeding volume due to the removal of the gaps between the blanket modules is partially compensated by the different shape of the blanket casing in the two layouts, that lead to a reduced LiPb volume in the "homogeneous" SMS (836.08 m^3 vs 852.89 m^3 for the MMS 2015). However, the removal of the gaps in the SMS concept enhance considerably the shielding performances with respect to the MMS: the peak value of the nuclear heating density on the VV decreases of about 30% and a mitigation in both the He-production and damage (-30% and -40% respectively) is also observed.

• *Impact of design configuration*. The comparison of the SMS and the "homogeneous" SMS has been performed to verify the impact of the new blanket/BSS configuration. The results are summarized in Table 3: the tritium production is higher in the "homogeneous" SMS layout (+1.3%) because of a higher LiPb content in the breeding zone (809.09 m^3 vs 800.61 m^3 for the SMS layout) [12] and its different distribution in the BZ due to the materials homogenization.

Table 3. Highlights of the SMS vs "homogeneous" SMS and MMS2015 BB comparative analysis.

Nuclear response	SMS	"homogeneous" SMS	MMS 2015
TBR	1.131	1.146	1.127
Nuclear Heating in TFC (W/cm^3)	$7.3x10^{-7}$	$1x10^{-6}$	$1x10^{-5}$
Damage (dpa) on VV	0.03	0.06	0.1
He production (appm) on VV	0.5	1.4	2

As far as the shielding performances are concerned, the SMS internal layout provides a sensible improvement in all the parameters analysed (-30% in the nuclear heating density in the TF, -50% in the damage and -65% in the He production in the VV).

5. Conclusions

The Single-Module-Segment WCLL DEMO blanket performances have been analyzed using MCNP. A MCNP model has been developed using a quasi-heterogeneous approach: the BB has been segmented into radial sectors with specifically defined material mixture and the obtained model has been integrated in the 2015 generic MCNP DEMO geometry.

The design target for tritium self-sufficiency is fully satisfied with a calculated total TBR of about 1.131. Taking into consideration the future DEMO baseline concept, that foresees a decreased outboard blanket extension, the present analysis highlighted that, with the present WCLL SMS design, the tritium self-sufficiency can be guaranteed even with a 15 cm radial reduction of the outboard breeding zone. The impact of this different design on the WCLL blanket shielding performances has to be investigated in detail: however the results obtained in this study (about 2 orders of magnitude less that the design limit for the heat load on the TF coils) leave ample room for improvement of the blanket/BSS system shielding capabilities.

The combined blanket/manifold/VV system is sufficient to protect the TFC from the radiation streaming: the fast neutron flux evaluated on the TFC winding pack is $1.3x10^8$ n/cm^2/s, an order of magnitude lower than the design target. The evaluated nuclear heating is negligible (0.7 W/m^3 against a design target of 50 W/m^3).

The estimated damage on the VV stainless steel is ~$4.8x10^{-3}$ dpa/FPY for the inboard and $5.44x10^{-4}$ for the outboard. These values are sufficiently low to guarantee the integrity of the VV over the 6 FPY DEMO lifetime (0.03 and 0.003 dpa for the inboard and outboard VV inner shell respectively).

The main nuclear quantities evaluated in Eurofer reach the following peak values at the FW: ~8.4 W/cm^3 nuclear heating density, ~9.3 dpa/FPY damage, 100 appm/FPY He-production.

A comparison has been made with three additional blanket configurations: the MMS 2015 DEMO model with homogeneous blanket description, a “full breeder” SMS layout with the breeding layers filled with LiPb and an “homogeneous” SMS, where the internal layout of the segment has been uniformly filled with the same compounds used to describe the MMS 2015 BB. This analysis allows a quantification of the impact of design configuration (segmentation) and structural material/coolant composition. The comparison shows that the new design provides sensible improvements in shielding performances whereas the impact on the overall nuclear power is marginal.

A sensitivity study of the blanket performance with the W–Armour thickness has been performed.

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References

[1] E. Martelli, et al., Advancements in DEMO WCLL breeding blanket design and integration, Int. J. Energy Res. (2017), in press, doi: 10.1002/er.3750

[2] U. Fischer, Guidelines for Neutronic Analyses, Final Report PMI-5.3-03, EFDA_D_2L8TR9 v1.4

[3] X-5 Monte Carlo Team: MCNP - A General Monte Carlo N-Particle Transport Code, Version 5, LosAlamos National Laboratory, Los Alamos, New Mexico, USA, April 2003.

[4] JEFF3.2 nuclear data library, http://www.oecdnea.org/dbforms/data/eva/evatapes/jeff_32/.

[5] A. Del Nevo et. Al., WCLL breeding blanket design and integration for DEMO 2015: status and perspectives, Fus. Eng. Des. Vol. 124, (2017), 682-686.

[6] U. Fischer et al., Methodological approach for DEMO neutronics in the European PPPT programme: Tools, data and analyses, Fus. Eng. Des. Vol. 123, (2017), 26-31.

[7] DEMO 2015 MCNP model with empty boxes, P. Pereslatstev, private communication, 2015.

[8] E. Martelli et al, Advancements in DEMO WCLL breeding blanket design and integration, Int. J. Energy Res., Vol. 42(1), (2017), 27-52

[9] F. Moro et al., Neutronic analyses in support of the WCLL DEMO design development, Fus. Eng. Des., (2018) in press, doi: 10.1016/j.fusengdes.2018.04.113

[10] C. Bachmann et al, Overview over DEMO design integration challenges and their impact on component design concepts, Fus. Eng. Des., (2018) in press, doi: 10.1016/j.fusengdes.2017.12.040

[11] F. P. Orsitto et al., Diagnostics and control for the steady state and pulsed tokamak DEMO, Nucl. Fusion, Vol. 56, (2016), 026009

[12] A. Del Nevo et al., Final Report on Deliverable BB-3.2.1-T004-D001: WCLL Design Report 2017, EFDA_D_2MYHNE v1.1, 2018