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Neutronics studies for the novel design of lower port in DEMO

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The conceptual design activity of the Demonstration Fusion Power Reactor (DEMO) is in progress in the Power Plant Physics and Technology (PPPT) programme, within the EUROfusion Consortium. In this work neutronics studies, fundamental for the nuclear design of DEMO, are presented for a novel design of the lower port (LP). Two possible configurations of the LP have been investigated: the vacuum pumping port, with the pumping unit located inside the port, and an empty port designed for remote handling. For both configurations 3-dimensional Monte Carlo calculations have been performed with MCNP5 to assess the neutron flux inside and around the port and the nuclear heating in sensitive components, such as the toroidal field coil conductor, the vacuum pumps, the shielding elements and the port closure plate. Different shielding configurations have been considered, by adding shielding blocks at the lower port entrance. Single and double wall port walls and closure plates with different thickness have been studied to reduce nuclear loads and neutron flux. Nuclear quantities under analysis were found to be within the limit for all the components, with the exception of the nuclear heating on the toroidal field coils. The absence of any shield of the divertor cassette pumping duct, as in the DEMO 2017 baseline configuration, is responsible for a huge amount of radiation streaming inside the pumping duct that can cause the excess of heating on the coil conductor. The use of a liner, or of an equivalent shielding component, is proposed, but further improvements are needed to keep the nuclear loads on the coil conductor within the limit.

Keywords: *Lower port, DEMO, nuclear heating, neutronics*

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1. Introduction

The conceptual design activity of the Demonstration Fusion Power Reactor (DEMO) is in progress with the Power Plant Physics and Technology (PPPT) programme, within the EUROfusion Consortium. In this framework, many efforts must be dedicated to the integration of different components. In the current DEMO reference configuration (EU DEMO1 2017) [1] the vacuum vessel (VV) has 16 vertical, horizontal and lower ports. The lower port is of paramount importance because of its structural, operational and maintenance functions [2]. In particular, the lower ports that host the torus vacuum pumping units require an adequate duct to the plasma chamber, while those devoted to maintenance functions are dedicated to the divertor replacement in first place. Previous version of the lower port (DEMO 2015) had a dedicated vertical port for pumps, while the divertor replacement was foreseen by the main lower port, 45° inclined with respect to the horizontal plane. The present baseline model (EU DEMO1 2017) foresees a single 45° lower port, while, in the novel design, the inclination has been reduced and a kink at a radial position close to the toroidal field (TF) coils connects the port to a horizontal duct. This study is devoted to the neutronics analysis of components around the novel design of the lower ports. In particular, two possible configuration of LP have been analyzed: LP for vacuum pumping (VP), a configuration in which the LP hosts a vacuum torus pumping unit (i.e. the metal foil (MF) pump and the linear diffusion (LD) pump) and a configuration in which the LP is dedicated to the remote handling (RH), so it is completely empty. These analyses will provide indication about the neutron flux inside and around the LP up to the closure plate and the nuclear heating of relevant components like the pumping unit and shielding elements. In particular, additional shielding elements have been integrated in the structure to reduce the radiation streaming through the pumping duct that can produce very large nuclear loads on the TF coil conductor.

MCNP [3] calculations have been performed using the the 3-D 11.25° sector 2017 baseline DEMO model developed by KIT and updated by ENEA to have a 22.5° model with the new LP design replacing the baseline lower port. The blanket modules have been represented with a homogeneous mixture representing the latest WCLL Single Module Segment blanket configuration [4]. JEFF 3.2 nuclear data library [5] has been used in the calculations.

Table 1. Main DEMO parameters of EU DEMO1 2017.

Major radius (m)	
Minor radius (m)	
Aspect ratio	
Plasma elongation	
Plasma triangularity	
Fusion power (MW)	

The neutron plasma source was simulated making use of the specially developed user defined source (SDEF card) in MCNP for DEMO baseline, describing a plasma scenario with the parameters reported in table 1.

2. Model description

The MCNP model has been modified in order to replace the standard lower port with the new LP design as shown in fig. 1.

For both VP and RH configurations, several variants have been analyzed. Variants differ for the insertion of several shielding blocks, for the dimension and composition of the port walls and for the thickness of the port plug. In fig. 2 the CAD of the LP is reported.

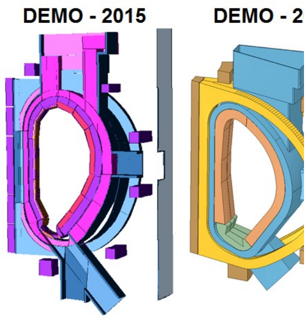

Fig. 1. CAD of the DEMO-2015, DEMO 2017 baseline and DEMO with new LP (from the left to the right).

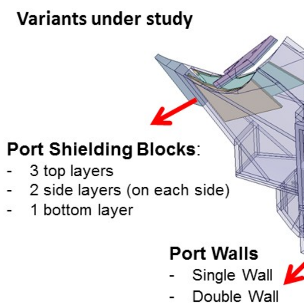

Fig. 2. CAD model of the LP, with variants under analysis.

Closure plate, shielding blocks and port walls are all made of SS-316-LN. Single port walls (SW) are 60 mm thick. Double port walls (DW) are two 60 mm thick walls with 100 mm of water in between them, represented with a homogenized mixture. In table 2 the variants under study are reported.

Table 2. Variants under analysis. CP is closure plate.

Variant	Shielding blocks	Port Walls
VP_REF	*none*	*SW*
VP_V1	*all*	*DW*
VP_V2	*Top and side*	*SW*
RH_REF	*None*	*SW*
RH_V1	*all*	*DW*

3. Results and discussion

Nuclear quantities have been tallied with cell and mesh track length estimators to calculate flux or nuclear heating. Nuclear heating includes the energy deposited by neutrons (generated from both plasma and nuclear reactions) and by photons (generated by nuclear reactions). Meshes have 10x10x10 cm^3 voxels and the relative statistical uncertainties are within 10%, at 1 σ, in all the region of interest (i.e. inside the LP).

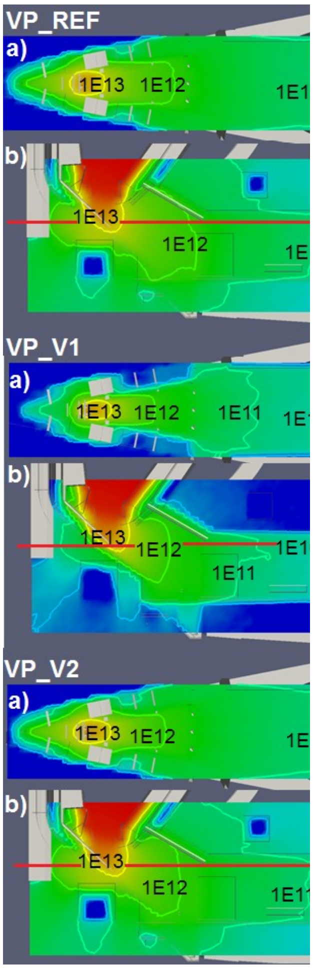

Fig. 3. Neutron flux maps for VP configuration, REF, V1 and V2 variants. Vertical cut along the central axis (b) and horizontal cut along the red line (a).

In figure 3 the neutron flux maps are shown for VP configuration. Neutron flux varies between 10^{13} n/cm^2/s at the VV penetration down to 10^{11} n/cm^2/s at the port closure plate.

Table 3. Neutron flux behind the Closure Plate. Statistical uncertainty is ± 8% in the worst case.

Variant	Neutron flux (n/cm^2/s)
VP_REF	*1.16·10^{10}*
VP_V1	9.1·10^{8}
VP_V2	9.5·10^{9}
RH_REF	2.9·10^{10}
RH_V1	4.0·10^{9}

Behind the port closure plate the neutron flux further decrease of one order of magnitude or more, being 1.16·10^{10} n/cm^2/s for REF, 9.1·10^{8} n/cm^2/s for V1 and 9.5·10^{9} n/cm^2/s for V2, calculated with a cell-based tally These results are reported in table 3. From the comparison of REF and V2 variants is possible to have an idea of the impact of the top and side shields on the radiation streaming. The effect is tiny but still visible in the contour lines: a narrowing of the area enclosed in the 10^{12} n/cm^2/s contour line is observed (due to the lateral shielding block), and a lowering of neutron flux inside the port, visible from the different shape of the 10^{11} n/cm^2/s contour line (effect of the top shielding block). The latter could also be responsible for the lower neutron flux behind the port closure plate, approximately -20%. The picture completely changes when the V1 configuration is considered. The DW port walls have a huge impact on the neutron flux surrounding the port: it is below 10^{10} n/cm^2/s. Also the reduction inside the port is clearly visible: 10^{11} and 10^{12} n/cm^2/s contour lines moves toward the plasma chamber.

In figure 4 the neutron flux maps are shown for RH configuration. In this case only REF and V1 variants have been considered. The same conclusions of VP configuration can be drawn with the exception of a slight increase of neutron flux inside the horizontal part of the LP, due to the absence of the vacuum pumping unit that partially shields some radiation. The effect can be estimated quantitatively if the neutron flux at the closure plate is considered: it is 2.9·10^{10} n/cm^2/s for REF, 2.5 times VP_REF and 4.0·10^{9} n/cm^2/s for V1, 3.4 times VP_V1 (see table 3).

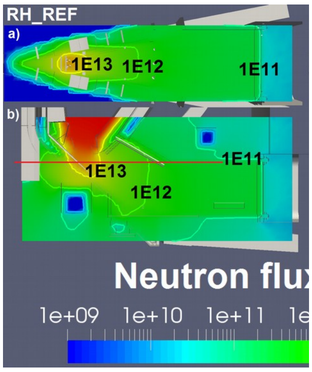

Fig. 4. Neutron flux maps for RH configuration, REF and V1 variants. Vertical cut along the central axis (b) and horizontal cut along the red line (a).

Nuclear heating density has been calculated on the shielding blocks, when present, on the vacuum pumping unit, separately on the LD pump and on the MF pump, and on the conductor of the TF coil in the region around the LP. These results are summarized in table 4, where the average values on the whole components are reported. The maximum nuclear heating density for MF pump is 420 W/m^3 in REF configuration and a small variation is observed among the variants. It is worth to be noted that in V1 a larger nuclear heating is observed with respect V2 variant, despite the DW port walls: this is due to the presence of water inside the port walls that moderate the neutron spectrum, producing more (n,γ) absorption reactions from structural materials and thus a more intense γ field that is the dominant contribution to the nuclear loads in steels.

Much larger nuclear heating densities are observed in the shielding blocks. In this case the nuclear heating does not depend on the configuration or on the variant, because the shielding blocks are mostly subjected to a direct irradiation from the pumping duct.
In the case of the CP we observe a behavior which is similar to that of the neutron flux: the vacuum pumping unit acts as a shield, reducing the loads on the CP.

Table 4. Nuclear heating density on specific components for the LP configurations. *Statistical uncertainties are within 1%.*

Nuclear Heating density (W/m^3)x10^3			
Shielding blocks			CP
Top	Side	Bottom	
-	-	-	0.37
24.3	66	17.7	0.15
24.4	60	-	0.30
-	-	-	1.25
24.1	65.1	17.7	0.59

Nuclear heating density on the TF coil conductor has been calculated as well. The nuclear loads on this component are of paramount importance because the operation of the reactor is dependent from this quantity. In particular, a strict design limit has been fixed at 50 W/m^3 [6]. In order to assess the nuclear loads as a function of the poloidal position, the TF coil conductor has been segmented in 6 cells (see figure 5).

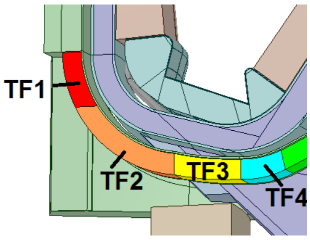

Fig. 5. Segmentation of TF coil conductor.

In table 5 the results of nuclear density on TF coil conductor have been reported. Only the VP configuration has been considered, because the presence of the vacuum pumping unit does not affect the nuclear loads on this component (results in RH configuration are consistent with VP case, within the statistical uncertainties).

Table 5. Nuclear heating density in TF coil conductor segments. Statistical uncertainty is ± 10% in the worst case.

Segment	Nuclear heating density (W/m
	REF
TF1	180
TF2	$4.51 \cdot 10^3$
TF3	$1.54 \cdot 10^4$
TF4	$5.05 \cdot 10^3$
TF5	$1.21 \cdot 10^3$
TF6	345

Nuclear heating density in TF coil conductor results well above the design limit. The most critical part is segment #3 located poloidally in correspondence of the pumping duct with a value 300 times the limit in REF and V1 configuration. The effectiveness of the DW in reducing the load is clearly visible on this segment: it allows a decrease of the nuclear heating density of about a factor 5, i.e. 60 times the limit. The effect of the shielding blocks is also marginal: a reduction of about -40% is observed but only in TF4 and TF5, comparing REF and V2. The segments in which the design limit is respected are TF1 and TF6 in all variants, and TF5, but only in V1 variant. In order to reduce the nuclear loads in the most promising variant, a shielding block has been added above the divertor pumping duct, that otherwise is completely exposed to the direct radiation from the plasma. This element can be considered a dummy liner, i.e. a 136x52x20 cm^3 parallelepiped made of a 40% water 60% SS316L mixture, located 20 cm above the pumping duct of the divertor. This study does not take into account any evaluation about the pumping efficiency or any issue related to the plasma facing, but it is just intended to have an idea about the impact of a liner or a dome covering the pumping duct. This calculation has been performed for V1 variant with a 10 cm and 20 cm thick dummy liner. Results are reported in table 6.

Table 6. Nuclear heating density (W/m^3) in TF coil conductor segments with the additional liner. Statistical uncertainty is ± 10% in the worst case.

Segment	Nuclear heating density (W/m	
	No liner	10 cm thick
TF1	18.7	
TF2	705	
TF3	3300	2060
TF4	386	
TF5	44.2	
TF6	9.38	

With 10 cm thick liner an improvement is observed: the nuclear loads are reduced in all segments and TF2 is now below the limit. However, in the most critical segment, TF3, the nuclear loads are still 40 times the design limit. The improvement passing from 10 to 20 cm thick liner is also marginal.

3. Conclusions

Nuclear analyses on the novel design of DEMO lower port have been performed. Two possible configurations have been examined, varying for each one several shielding elements. Results show that the double wall structure is very effective in lowering the neutron flux thanks to the presence of water inside the walls. Instead the nuclear heating inside the port is not strongly dependent from the wall configuration. The nuclear heating of the shielding blocks is less than 10^5 W/m^3, but, being exposed to direct irradiation from plasma, large gradients are expected, so that active cooling should be considered. On the vacuum pump, the nuclear heating density is below 500 W/m^3 in the worst case. Also the heating of the port closure plate is very low, being 370 W/m^3 at its peak value. Nuclear heating on TF coil conductor has been calculated and it comes out to be a major issue: in the best possible variant a value 60 times the limit is observed. An additional dummy shield has been added just above the divertor pumping duct, that otherwise is subjected to direct irradiation from plasma, but the reduction of nuclear loads is a factor 2 in the best case. Further studies are thus needed to reduce the nuclear heating density of the TF coil conductor in the poloidal region around the DEMO lower port.

Acknowledgments

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