Assessment of Radiation Streaming Effect upon Radiation Loads to Inboard TF-coil of DEMO Utilizing HCLL Blanket

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Abstract. The work presents the results of the detailed analyses performed on the shielding capability of HCLL DEMO-type fusion reactor. The analysis is based on three-dimensional calculations with the Monte Carlo code MCNP-4C and consists in assessment of radiation loads to the super-conducting TF-coil at the inboard side of the reactor. A suitable 20^0-sector model of DEMO reactor, based on the reactor parameters of PPCS model AB and developed by FZK, have been employed in the analyses. The effect of the toroidal and poloidal gaps between blanket modules upon the responses considered has been investigated. Significant impact of the radiation streaming through the gaps on the radiation load has been observed. The poloidal profiles of the responses in respect to the torus mid-plane have been assessed in several variants of the gap widths and compositions of additional shield plugging the gaps. It was found that the design limit of all the responses considered could only be met if the toroidal and poloidal gaps are plugged by LTS material, the later being made of borated steel, WC and water coolant. The dose absorbed in the Epoxy insulator and the radiation damage to the copper stabilizer were found to be the most critical quantities in this regard.

1. Introduction

The Helium-Cooled Lithium-Lead (HCLL) blanket concept, based on the use of Pb-15.8Li eutectic as tritium breeder and neutron multiplier, is one of the European research lines for a DEMO fusion power plant. Its development has been endorsed in 2004 and a Power Plant Conceptual Study (PPCS) model AB has been defined and further developed by CEA [1]. Like model B, based on Helium-Cooled Pebble-Bad blanket and developed by FZK, the model AB is based on near term technological and physical extrapolations. Both blanket concepts share as much as possible common features: the use of high-pressure He gas as a coolant, low activation Eurofer steel as a structural material and radial arrangement of the structure, but each of them makes use of a specific breeder unit inserts.

Three-dimensional radiation transport calculations with the use of the Monte Carlo code MCNP-4C [2] have been performed for the HCLL DEMO-type fusion reactor. Neutronic model, developed by FZK [3] on the basis of PPCS model AB, has been used in the analysis. Shielding capability of the reactor in terms of radiation loads to the super-conducting toroidal field (TF) coil has been assesses at the inboard torus mid-plane, subject to the most intensive radiation. These include the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper stabilizer and the radiation dose absorbed by the Epoxy resin insulator. A 4000 MW fusion power and 40 full power years (fpy) of plan operation have been assumed.

The neutronic model used in calculations represents in details the reactor geometry, i.e. the blanket and shield elements, Vacuum Vessel (VV), TF-coil and 4 cm wide gaps between blanket modules, thus giving the possibility for detailed calculations of reactor shielding performance. The analyses presented in this paper emphasis on the impact of the gaps between blanket modules in toroidal and poloidal direction on the nuclear responses considered. Additional shielding resulting in radiation loads below the tolerable levels, as specified for ITER [4], has been proposed.

2. Brief Description of the Plant Model AB

The PPCS model AB of a fusion reactor DEMO is based on the helium-cooled lithium-lead blanket concept. It uses Pb-15.8Li with 90% ^{6}Li enrichment as breeder, neutron multiplier and tritium carrier, Eurofer as structural material and helium as a coolant with 8 MPa pressure and 300/500^{0}C inlet/outlet temperature.

For good maintenance characteristic, a segmentation of the blanket into large modules is adopted [5]. The HCLL module, developed by CEA, consists of a steel box of about 4 m (poloidal) x 2 m (toroidal) dimensions. The box is reinforced by radial-poloidal and radial-toroidal stiffening plates. The formed internal cells accommodate the breeder units, consisting of five horizontal cooling plates. The Pb-Li breeder slowly flows throughout the box. The box is closed on its back by five parallel plates, constituting the coolant manifolds. Figure 1 shows a view of the blanket module and its components [5].

The in-vessel shield of the reactor is divided into two regions: high temperature shield (HTS) directly behind the blanket, integrating He and PbLi collectors and made of Eurofer steel, and low temperature shield (LTH) attached to the vacuum vessel (VV), made of borated steel, WC and water as a coolant. The layout of the model includes also the divertor and TF coils. It assumes 2mm thick W coating of the first wall and 4 cm wide gaps between blanket segments in toroidal and poloidal direction. The total thickness of blanket and shield amounts to 112.9 cm and 165.4 cm at the inboard and outboard torus mid-plane, respectively, with the breeder zones radial dimensions of 42.5 cm and 78 cm, respectively.

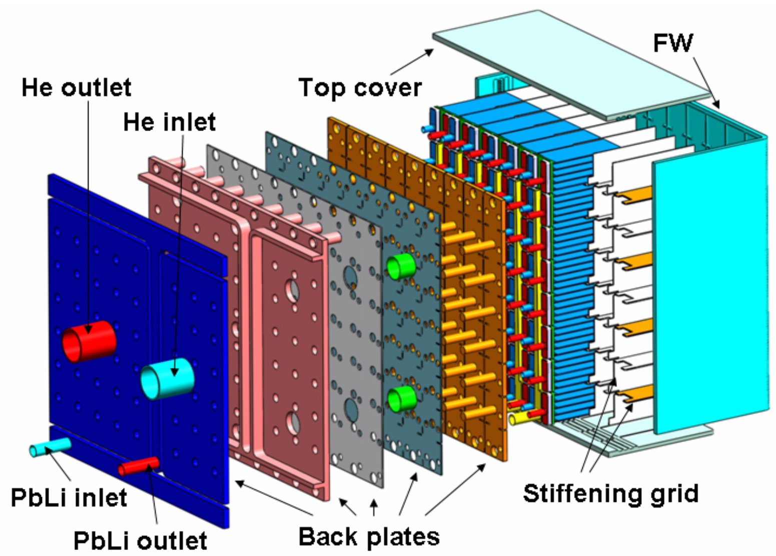

FIG. 1. View of the HCLL blanket module, showing its components.

3. Reactor Modeling and Calculation

Neutron and photon transport calculations have been carried out to predict shielding performance of the HCLL DEMO-type fusion reactor by using the Monte Carlo code MCNP-4C [2] and the continuous energy FENDL-2 [6] and MCPLIB2 [7] cross-section data sets. A 20^0 three-dimensional torus sector model developed by FZK [3] for use with the MCNP code has been used in calculations. The model includes the plasma chamber, inboard and outboard blanket modules, HTS and LTS modules, divertor, vacuum vessel and toroidal field coil. It is based on the reactor parameters of PPCS model AB [4] assuming fusion power of 4000 MW.

The 20^0-sector model is constructed including 3 x (2 x ½) and 3 x (2+1/2) inboard and outboard modules, poloidally arranged around the plasma chamber [3]. The "1/2" means half toroidal extension of the module. The model assumes 4 cm wide gaps between blanket modules in toroidal and poloidal directions. A proper simulation of the spatial distribution of the D-T source neutrons was employed in calculations [3].

The radial-poloidal and radial-toroidal cross sections of the MCNP torus sector model are presented in Figures 2 and 3, respectively, with the essential zones indicated. Due to the toroidal curvature of the VV and to the blanket segments at the inboard side designed as planes, a void space is formed at the bottom of the segment (seen in Fig. 3). For the purpose of the shielding calculations in the model, devised by FZK, the helium and Pb-Li tubes in HTS of the central inboard segment are modelled in details [3], as seen in Fig. 3.

The neutron and gamma ray fields and their responses have been scored according to the aim of the estimate and were accumulated until the desired relative error has been achieved. Since the problems to be solved are deep penetration and radiation steaming ones, a window generator variance reduction technique has been employed in order to increase the efficiency of the calculations and to improve the statistics.

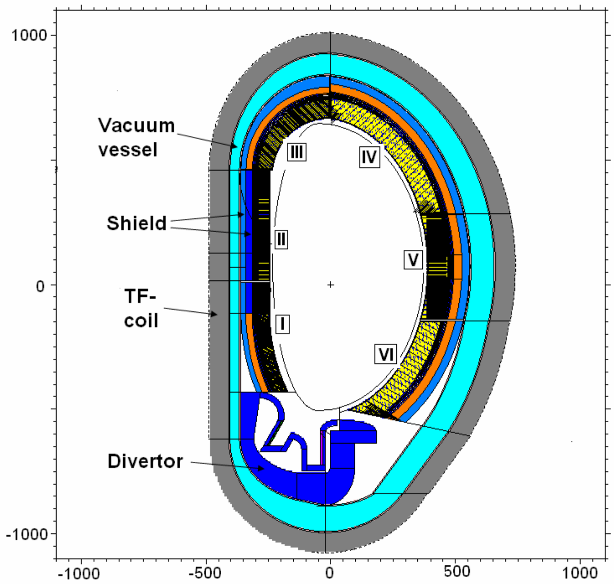

FIG. 2. Radial-poloidal cross-section of the MCNP torus sector model. The numbers from I to III and from IV to VI indicate the inboard and outboard blanket modules, respectively. The dimensions are given in centimetres.

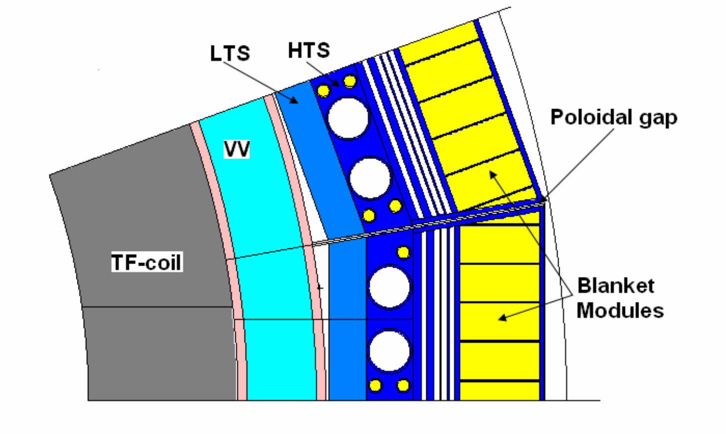

FIG.3. Radial-toroidal cross-section view of the MCNP torus sector model with TF-coil; the vacuum (VV); the low temperature shield (LTS); the high temperature shield (HTS), the inboard blanket modules and the poloidal gap indicated.

4. Radiation Loads to the Super-conducting TF-coils

4.1 Analysis

The integrated shield system protecting the TF-coils from the penetrating radiation consists of the blanket, HTS, LTS and Vacuum Vessel. This system must protect sufficiently the super conducting TF-coils so that the tolerable levels, as specified for ITER [4], are not exceeded. The critical issues in this regard are the fast neutron fluence to the superconductor, the peak nuclear heating in the winding pack, the radiation damage to the copper stabilizer and the radiation dose absorbed by the Epoxy resin insulator.

To assess shielding performance of the reactor the calculations have been performed at the inboard mid-plane where the neutron wall loading shows a peaking value [3]. On the other hand, due to the limited space available for the shield at the inboard side and to the relatively large thickness of the HTS, the LTS is of small radial dimension hindering it to attenuate the neutron flux sufficiently enough. In addition, the HTS integrates Helium and Pb-Li collectors, which deteriorates its shielding efficiency, especially owing to the large size of the He tubes. Due to the 4 cm wide gaps between blanket modules in toroidal and poloidal directions, designed for the maintenance purposes, and to the neutron steaming through them, the shielding efficiency of the shield system is additionally deteriorated. Therefore, the shielding calculations have been performed assuming LTS, made of borated steel (10%), WC (65%) and water coolant (25%), selected as a shield composition revealing excellent shielding capability [3]. The vacuum vessel assumed in calculations consists of 37 cm thick shielding mixture of borated steel (60%) and water (40%) sandwiched between two SS-316 steel plates with thickness of 5 cm each. The radial thicknesses of the HTS, LTS and the VV at the inboard torus mid-plate are 25 cm, 20 cm and 47 cm, respectively. Fusion power of 4000 MW and 40 fpy of fusion plant operation have been assumed.

The poloidal profiles of the above mentioned responses have been calculated in respect to the torus mid-plane and comprise the location of toroidal gap. The spatial locations of the performed estimates have been selected to reflect the most critical shielding conditions, i.e. opposite to the central band of the blanket module and opposite to the poloidal gap including the cross-point of toroidal and poloidal gaps. The poloidal profiles opposite to the central band of the blanket module have been investigated more closely and the most efficient shielding configuration found has been further applied to assess the poloidal gap profile.

The variants considered in these calculations include void gap at the bottom of the blanket segments and that gap filled by SS-316 steel, toroidal gap plugged by steel SS-316 or by the material of LTS, the thickness of the plug being equal to that of LTS and unplugged toroidal gap with width reduced to 2 cm. The poloidal profiles have been obtained opposite to the poloidal gap considering the void gap at the bottom of the blanket segments filled by SS-316 steel and poloidal gap plugged by the material of LTS.

4.2 Results and Discussion

Figs. 4 through 7 present the obtained responses and compare the results with the radiation design limits as specified for ITER.

The poloidal profiles assesses considering the reference design, i. e. 4 cm wide toroidal and poloidal gaps and void space between VV and LTS (see Fig. 3), at location opposite to the central band of the blanket module, show that all radiation loads, with exception of nuclear heating in the winding pack, exceed the design limits in the vicinity of toroidal gap, as shown in Figs 4 through 7. Due to the radiation streaming through the toroidal gap the estimated responses are peaked in that location and the values of some of them become lower than the design limits only beyond the torus mid-plane.

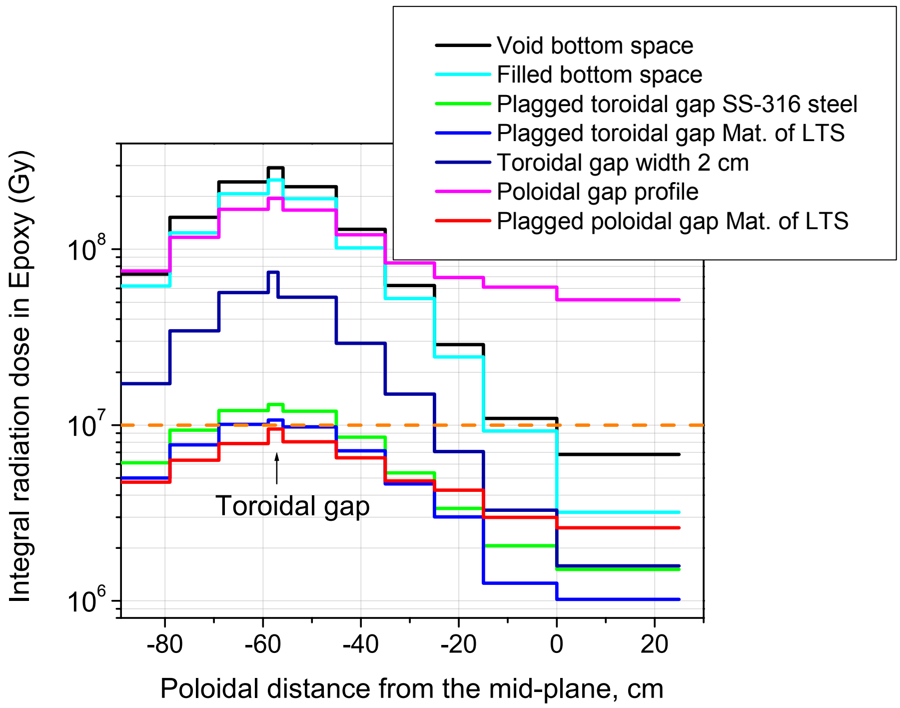

FIG. 4. Poloidal profile of integral radiation dose in insulator (Epoxy). The design limit is shown with dashed line

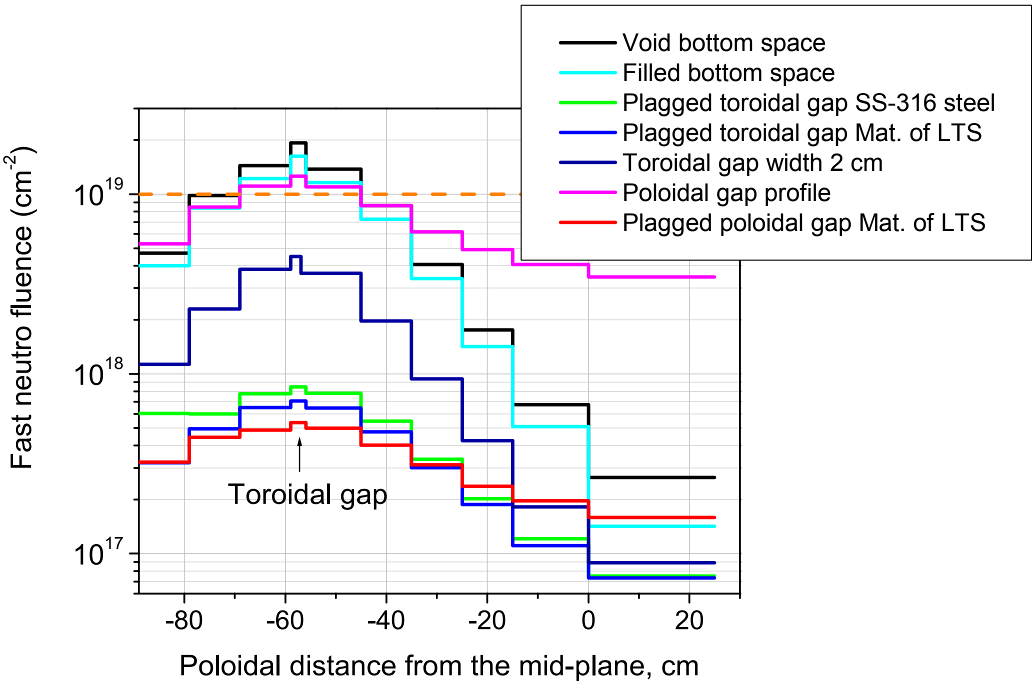

FIG. 5. Poloidal profile of fast neutron fluence (En>0.1 MeV) to the Nb_3Sn Superconductor. The design limit is shown with dashed line.

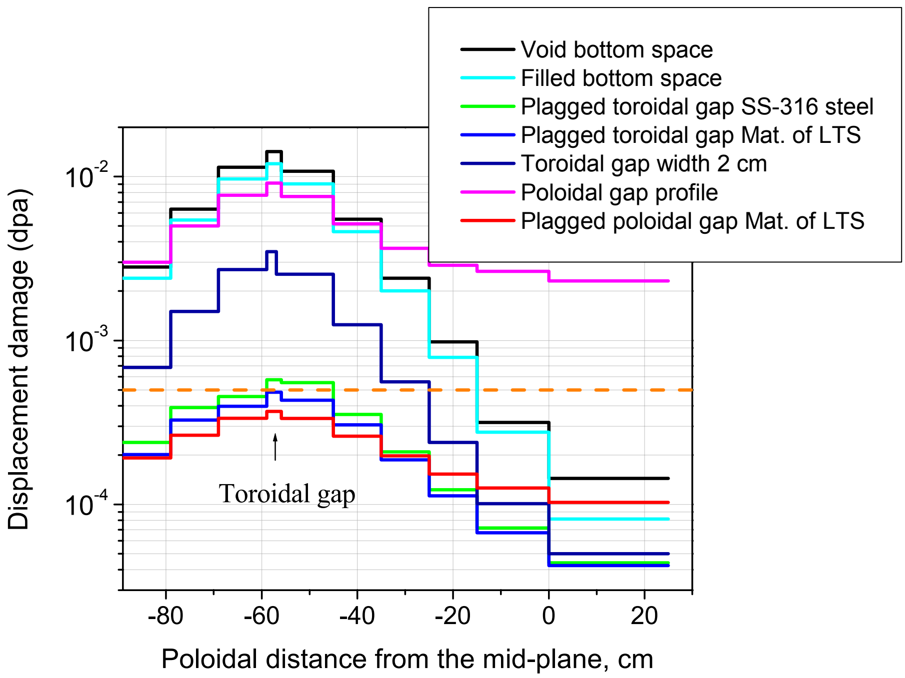

FIG. 6. Poloidal profile of displacement damage to copper stabilizer. The design limit is shown with dashed line.

The filling of the void space between VV and LTS by SS-316 steel as well as the subsequent reduction of the toroidal gap width to 2 cm do not bring the radiation dose absorbed by the Epoxy resin insulator and the radiation damage to the copper stabilizer below the limits in the vicinity of toroidal gap, as seen in Fig. 4 and Fig. 6. The same holds even if the 4 cm wide toroidal gap is plugged by SS-316 steel, the thickness of the plug being equal to that of LTS. The design limits of the above mentioned responses can only be met if the toroidal gap is plugged by LTS material, as seen in Figs. 4 and 6.

The poloidal profiles of the radiation loads calculated opposite to the poloidal gap (the void space between VV and LTS filled by SS-316 steel) are peaked around the cross-point of the toroidal and poloidal gaps, exceeding the specified limits, and tend to become flat, as it should be expected, beyond that location, with values of radiation dose in Epoxy and the displacement damage to the copper stabilizer higher than the design limits. Plugging the poloidal gap by LTS material, the thickness of the plug being equal to that of the LTS, lowers the responses below the specified limits in all positions considered.

The analysis of estimated responses reveals the significant impact of the radiation streaming through the gaps between blanket modules. The most critical quantities among the estimated quantities are found to be the integrated radiation dose in Epoxy insulator and the displacement damage to the copper stabilizer. The radiation dose in Epoxy and the displacement damage to the copper stabilizer, created mostly by the high energy neutrons (E_n > 0.1 MeV), are not attenuated enough by plugging the gaps with SS-316 steel. Using more efficient shielding material, namely WC and water to slow down the neutrons, reduces the fast neutron flux by about 30 times at the locations opposite to the gaps, thus decreasing the critical responses below the design limits.

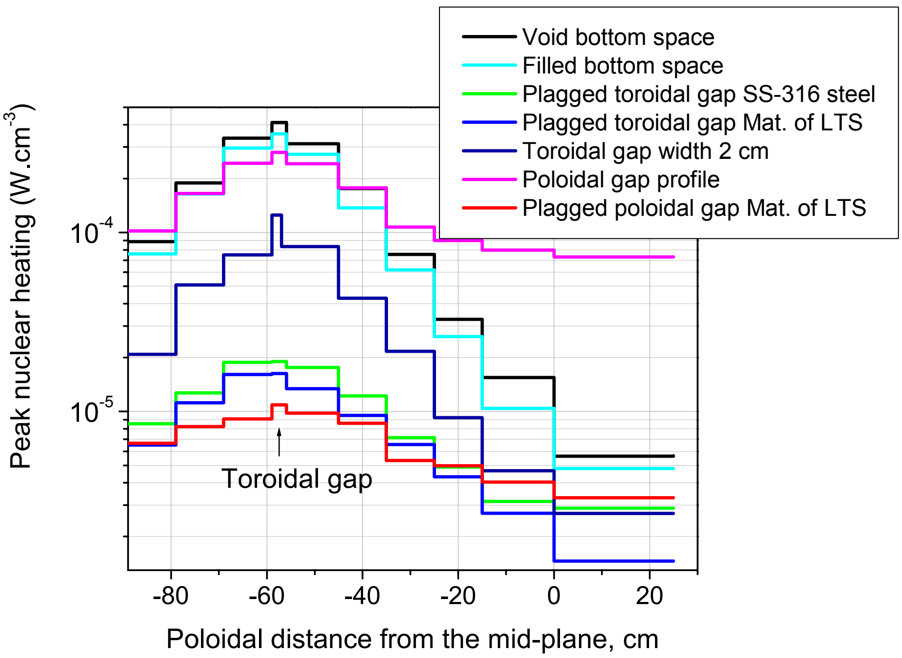

FIG. 7. Poloidal profile of nuclear heating in winding pack.

5. Conclusions

Three-dimensional neutron-photon transport calculations have been performed to assess the shielding performance of PPCS plant model AB for a HCLL DEMO-type fusion reactor. The Monte Carlo radiation transport code MCNP-4C and a suitable 20^0-torus sector model developed by FZK have been used in the analysis. Shielding performance of the reactor has been assessed at the inboard torus mid-plane considering radiation loads to the super-conducting TF-coils.

Poloidal profiles of radiation loads to the superconducting TF-coils, opposite to the toroidal and poloidal gaps between blanket segments, have been estimated in respect to the torus mid-plane. Significant impact of radiation streaming through the gaps on shielding performance of the reactor has been observed. Additional shield plugging the gaps, made of borated steel, WC and water coolant, resulting in radiation loads below the tolerable levels, as specified for ITER, has been proposed.

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