## Parametric Neutronic Analysis of HCLL Blanket for DEMO

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**Abstract.** Three-dimensional parametric neutronics calculations using the Monte Carlo code MCNP-4C have been performed for the DEMO-type reactor based on the Helium-Cooled Lithium-Lead blanket. The aim of the analysis is to reduce the radial length of the blanket as much as possible to a level ensuring tritium self-sufficiency and to assess shielding performance of the reactor. It was found that assuming 90% <sup>6</sup>Li enrichment, the reference blanket and that with reduced breeder zone length to 55 cm achieve TBR ensuring tritium self-sufficiency, while blanket with 60% <sup>6</sup>Li enrichment requires breeder zone length of 75 cm to cover the required margin. Taking into account the radiation load to the super-conducting toroidal-field coil it was found that the blanket with reduced radial length to 55 cm utilizing either WC-composite as a shield and vacuum vessel of cryogenic steel or two-component shield of Eurofer and WC and vacuum vessel ITER-FDR design demonstrates the best shielding performance in terms of radiation load below the design limit.

#### **1. Introduction**

The Helium-Cooled Lithium-Lead (HCLL) blanket concept is one of the two research lines considered in the European Fusion Technology Programme for DEMO reactor specification along with the Helium-Cooled Bebble-Bed (HCPB) concept. Both conceptual power plant models are based on near-term technological and physical extrapolations and share as much as possible basic technology, i.e. they use the same modular blanket arrangement and feature the same radial steel structure of low radiation steel Eurofer, utilize Helium at high pressure as a coolant, while the HCLL blanket modules make use of specific breeder unit inserts. The HCLL blanket concept is developed by CEA and FZK on the basis of model B parameters of Power Plant Conceptual Study (PPCS) and a typical breeder blanket module has been designed.

The work presents the parametric neutronic study for the DEMO-type reactor model based on HCLL blanket whose purposes are reducing the blanket radial depth as much as possible without reducing significantly the tritium breeding ratio (TBR) or shielding performances at the inboard torus mid-plane regarding the radiation load to the super-conducting toroidal field coils. The analyses are based on three-dimensional Monte Carlo calculations using torus sector model developed by FZK. Several parametric studies have been performed considering different <sup>6</sup>Li enrichment and radial length of the breeder zone, as well as different composition of vacuum vessel and of the in-vessel shield in the inboard torus mid-plane. The cases provided tritium self-sufficiency and sufficient shielding of the TF-coils have been identified

The results of the analysis will be used for a second iteration of the blanket design. This study is a part of EFDA Technology Work Programme 2004 in the HCLL Breeding Blanket area.

#### 2. Brief Description of HCLL Blanket Module

The HCLL blanket uses reduced activation ferritic steel Eurofer as structural material, Pb-17Li eutectic alloy as breeder and neutron multiplier and Helium as a coolant with inlet/outlet temperature 300/500<sup>°</sup>C and 8Mpa pressure. The blanket module consists of a segment box directly cooled by He and reinforced by radial-toroidal and radial-poloidal stiffening pates, He cooled, to withstand the helium pressure in case of an internal leak. The module box is assembled of first wall with 0.2 cm thick W armour, side walls, top and bottom covers and back plate with Helium inlet/outlet manifolds. The grid of the stiffening plates subdivides the inside of the box in radial cells, in which the Pb-17Li is slowly flowing, cooled by cooling units inserts. The breeder cooling unit consists of five horizontal cooling plates, small steel plate and He inlet/outlet collectors connected to the back plate. The space between collectors is filled with Pb-17Li. The radial cells accommodating the cooling units and Pb-17Li hear after are referred to as breeder units (BU). The blanket module consists of 81 BUs, toroidally-poloidally disposed in 9x9 rows. The dimensions of the modules are 200x200cm in toroidal-poloidal directions. The reference blanket module assumes 90% <sup>6</sup>Li enrichment and is of 100 cm radial length, the breeder unit dimension being 80cm, of which 75cm is the breeder zone (BZ) depth.

#### 3. Modeling and Calculation

The neutronic analyses have been performed utilizing the Monte Carlo three-dimensional radiation transport code MCNP-4C [1]. A  $9^0$ -sector model representing the modular structure of the blanket, developed by FZK [2], has been used in calculations. This model, set up for application with the MCNP code, is based on a DEMO-type neutronic model, derived by CEA and FZK from the European Power Plant Conceptual Study by using the model B parameters. It is developed by integrating into the DEMO model the HCLL blanket module, designed by CEA, and modeled by representing its geometry close to reality. A 3300 MW fusion power has been assumed for the model. A proper simulation of the spatial distribution of the D-T source neutrons was employed in calculations [2].

The 9<sup>0</sup>-sector model includes 5 inboard modules with half their toroidal extension and 7 outboard modules with their full toroidal extension, poloidally arranged around the plasma chamber. The divertor and the vacuum vessel are also included in the model. The toroidal field (TF) coil at the inboard side has been additionally modeled in order to assess the shielding performance of the blanket. Figure 1 shows a poloidal-radial cross-section of the torus sector model. Figure 2 shows the corresponding radial-toroidal cross-section at the inboard torus mid-plane. Radial-poloidal cross section view of the MCNP model of the central outboard blanket module, showing BUs poloidal arrangement and its elements, is presented in Fig.3.

Coupled neutron/gamma calculations have been carried out using the continuous energy FENDL-2.0 [3] cross-section data to obtain the spatial and energy distribution of neutron and gamma-flux, the tritium production rate and other nuclear responses. In order to obtain the peaking values of the responses related to the TF-coil shield they were calculated at front surface of the TF-coil. Tritium production has been calculated with maximum relative error of 0.5% due to the highest accuracy requirement to that neutronic characteristic. In the case of parameters relevant to the TF-coil shielding the statistical error obtained is between 5 and 10%.



Figure 1. Radial-poloidal cross-section view of the MCNP torus sector model.



Figure 2. Radial-toroidal cross-section view of the MCNP torus sector model at the inboard side.



Figure 3. Radial-poloidal cut through MCNP model of the central outboard blanket module.

## 4. Tritium Breeding Capability

One of the primary functions of the breeding blanket is to produce tritium to replace that burnt, in excess by a margin to cover losses and radioactive decay. It was shown [2] that for the conceptual design of HCLL blanket, the TBR, define as the ratio of the tritium production rate to the rate of tritium burning in the plasma, should be  $\geq 1.10$  in order to compensate the losses, to account for the presence of ports and penetrations in the blanket and to add an allowance for uncertainties in neutronic prediction of the TBR arising from methods approximations and errors in nuclear data.

The purpose of the parametric neutronics study performed is to find the blanket systems having the potential to meet the required TBR. Therefore the TBR has been assessed considering different radial depth of the breeder zone of the blanket and assuming two alternative <sup>6</sup>Li enrichments, namely 90% and 60%. The estimated TBR are presented in Table I. The calculations have been continued until a statistical error of 0.5% was achieved.

## TABLE I: TRITIUM BREEDING RATIO IN BLANKET WITH DIFFERENT RADIAL LENGTH AND $^6\mathrm{Li}$ ENRICHMENT.

Radial length of	90% <sup>6</sup> Li enrichment	60% <sup>6</sup> Li enrichment
Breeder Zone, cm		
75	$1.226 (0.005)^{1}$	1.152 (0.005)
60	1.183 (0.005)	1.095 (0.005)
55	1.148 (0.005)	1.065 (0.005)
1)		

<sup>1)</sup> relative error

The results presented in the Table show that the blankets with reduced breeder zone depth to 55 cm and assuming 90% <sup>6</sup>Li enrichment have the potential of achieving tritium self-sufficiency. Among the blankets assuming 60% <sup>6</sup>Li enrichment that of 75 cm breeder zone radial length fulfill the requirement for tritium self-sufficiency, while the blanket with breeder zone radial depth of 60 cm is close to ensure the required TBR.

### 5. Radiation Load to the Super-conducting TF-coils

As it is stated in the previous section the blankets with the three breeder zone lengths considered ensure tritium self-sufficiency. On the other hand, however, the breeding blanket, besides breeding tritium and transforming the nuclear energy into heat serves the function of shielding the super-conducting TF-coils from penetrating radiation. The blanket, in-vessel shield and vacuum vessel should prove sufficient shielding capability in terms of tolerable radiation load to the TF-coils. Therefore it must be shown that 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 insulator do not exceed the tolerable level as specified for ITER [4].

Progressive shielding analyses with respect to the breeder zone length have been performed. Calculations have been carried out for the inboard mid-plane of the reactor where the shielding is most crucial due to the limited space available for the blanket and shielding system, while at the same time the neutron wall loading shows a peaking value due to the concentration of the plasma source around the mid-plane [2].

To keep the total radial thickness of 100 cm (thickness of the reference modules), a 15 cm and 20 cm thick shield behind blanket modules of 60 cm and 55 cm breeder zone radial length has been added. This additional shield is either Eurofer or WC composite depending on the chosen shielding composition between the central inboard modules and vacuum vessel. Therefore in the location of the inboard mid-plane the shield thickness totals 15cm, 30 cm and 35 cm for the blanket with breeder zone depth of 75cm, 60 cm and 55 cm respectfully, the total thickness blanket/shield system in the inboard side amounting to 115 cm.

Tables II and III summarize the responses in the TF-coil based on a 3300 MW fusion power and a 20 years full power operation and compares the results with the radiation design limits as specified for ITER [4].

Two shield/Vacuum Vessel configurations have been assumed in our analyses, namely WC composite (tungsten carbide composed of 50% W and 50% C) as a shield behind central inboard modules and Vacuum Vessel of SS 316-LN cryogenic steel, hereafter referred to as  $VV_{REF}$  and Eurofer shield behind central inboard modules and Vacuum Vessel ITER FDR design, denoted as  $VV_{ITER}$ . The later consists of 30 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 thickness of the vacuum vessel at the inboard side is 40 cm.

The cases employing Eurofer shield and vacuum vessel of cryogenic steel show large excess over the design limit for the four quantities considered and therefore are not included in the discussion.

The results presented in the Tables indicate that the design limit can be met for the following blanket/shield/vacuum vessel configurations:

- Blanket with 55 cm radial length of the breeder zone, 35 cm thick WC shield at the inboard mid-plane and VV<sub>REF</sub> (see Table II);
- Blanket with 55 cm radial length of the breeder zone, two-component shield between the central inboard modules and vacuum vessel composed of Eurofer with thickness of 30 cm followed by 5 cm thick WC shield and ITER FDR design of the vacuum vessel (Table III).

The total and fast ( $E_n > 0.1$  MeV) neutron fluxes across the blanket\shield\vacuum vessel systems have different profiles due to the difference in the material compositions and thickness of the blanket and the shield. Although the fluxes are not at the same level at the TF-coils, the peak fast neutron fluence is below the limit in all cases (see Tables II and III). The radial profiles of fast and total neutron fluxes are presented in Figures 4 and 5 respectively.

The results in Table II indicate that 15 cm thick WC shield behind the central inboard modules in combination with  $VV_{REF}$  is not enough to ensure sufficient shielding of the TF-coils: the integral absorbed dose in the Epoxy and the peak displacement damage to Cu stabilizer exceed the design limit. The results in the two tables show that the most crucial quantity regarding the design limit is the integral absorbed dose in epoxy, which in most cases is over the limit.

	Design	75 cm	60 cm	55 cm
	limits [4]	breeder	breeder	breeder
		zone, 15cm	zone,	zone,
		WC	30cm	35 cm WC
			WC	
Integral radiation dose	$5.0 \times 10^{6}$	$7.2 \times 10^7$	$6.9 \times 10^{6}$	$3.2 \times 10^{6}$
in the insulator (Epoxy)				
[Gy]				
Peak fast neutron	$5.0 \times 10^{18}$	$4.6 \times 10^{18}$	$4.0 \times 10^{17}$	$1.8 \times 10^{17}$
fluence in Nb <sub>3</sub> Sn super				
conductor [cm <sup>-2</sup> ]				
Peak displacement	$2.5 \times 10^{-4}$	$2.3 \times 10^{-3}$	$1.8 \times 10^{-4}$	8.0x10 <sup>-5</sup>
damage to Cu stabilizer				
[dpa]				
Peak nuclear heating in	$1.0 \times 10^{-3}$	$6.6 \times 10^{-4}$	$6.3 \times 10^{-5}$	$3.0 \times 10^{-5}$
winding pack [W cm <sup>-3</sup> ]				

# TABLE II: RADIATION LOADS TO THE INBOARD TF-COIL (20 FULL POWER YERS OPERATION) FOR WC SHIELD AND $\rm VV_{REF}.$

# TABLE III: RADIATION LOADS TO THE INBOARD TF-COIL (20 FULL POWER YERS OPERATION) FOR EUROFER SHIELD AND $\mathrm{VV}_{\mathrm{ITER}}.$

	Design limits [4]	75 cm breeder zone, 15 cm Eurofer	60 cm breeder zone, 30 cm Eurofer	55 cm breeder zone, 35 cm Eurofer	55 cm breeder zone, 30cm Eurofer+
	<b>z</b> o 1 o 6			1 1 1 07	5cm WC
Integral radiation	$5.0 \times 10^{\circ}$	6.9x10°	9.9x10°	1.1x10′	$3.6 \times 10^{\circ}$
(Epoxy) [Gy]					
Peak fast neutron	$5.0 \times 10^{18}$	$3.6 \times 10^{17}$	$2.6 \times 10^{17}$	$2.8 \times 10^{17}$	$6.6 \times 10^{16}$
fluence in Nb <sub>3</sub> Sn					
super conductor					
	4	Λ	5	4	5
Peak displacement	$2.5 \times 10^{-4}$	$1.2 \times 10^{-4}$	$9.5 \times 10^{-5}$	$1.1 \times 10^{-4}$	$2.3 \times 10^{-5}$
damage to Cu					
stabilizer [dpa]					
Peak nuclear heating	$1.0 \times 10^{-3}$	$6.0 \times 10^{-5}$	$1.0 \times 10^{-4}$	$1.1 \times 10^{-4}$	$3.5 \times 10^{-5}$
in winding pack					
$[W \text{ cm}^{-3}]$					



Figure 4. Radial profiles of fast  $(E_n > 0.1 \text{ MeV})$  neutron flux densities at the inboard mid-plane for the three blanket designs, WC shield and vacuum vessel of cryogenic steel (left) and for Eurofer shield and vacuum vessel ITER FDR design (right).



Figure 5. Radial profiles of total neutron flux densities at the inboard mid-plane for the three blanket designs, WC shield and vacuum vessel of cryogenic steel (left) and for Eurofer shield and vacuum vessel ITER FDR design (right).

More than 80% of the integral radiation dose absorbed by the Epoxy insulator and more than 90% of the total nuclear heating in the winding pack are due to the photon absorption for all cases regarded.

For the  $VV_{REF}$ / WC shield configuration the absorbed dose and nuclear heating due to gamma absorption decrease with WC thickness increasing owing to the fact that WC is a good attenuator and absorber of the gamma radiation. Even though only the 35 cm thick WC shield ensures the absorbed dose below the design limit.

For the VV<sub>ITER</sub>/ Eurofer shield configuration with breeder zone depth decreasing, i.e. with Eurofer thickness increasing, the gamma-dose and gamma-heating increase, due to the steel acting as photon emitter. Owing to the high gamma-radiation dose the design limit of the integral radiation dose absorbed in Epoxy is exceeded for the three cases considered, the other responses being below the required limits (See Table III). Therefore the gamma radiation should be decreased in order to reduce the absorbed dose in epoxy. This can be achieved by replacing a part of the steel, at the inboard mid-plane, adjacent to the vacuum vessel, by efficient gamma-radiation absorber, such as WC. Consequently a two-component shield behind the central inboard modules has been applied in the analysis with a blanket of 55 cm radial length, consisting of 30 cm thick Eurofer followed by 5 cm thick WC shield. The estimated responses with the use of the so proposed two-component shield are presented in the last column of Table III. The obtained decrease in the gamma radiation absorbed dose and heating with about a factor of three ensures fulfilling the requirement of the specified limit for the integrated absorbed dose in Epoxy. It is expected that the same would hold, even in larger extend, for blanket designs with breeder zone thickness of 75cm and 60 cm, due to the lower gamma- radiation absorbed dose.

### 6. Conclusions

Comprehensive neutronics study employing the three-dimensional Monte Carlo code MCNP-4C has been performed for the HCLL DEMO-type fusion reactor model. The objectives of the analyses were the estimation of the tritium breeding capabilities for different radial length of the breeder zone and different <sup>6</sup>Li enrichment and the assessment of the shielding performance at the torus mid-plane of the inboard part of the reactor.

It was found that the tritium breeding self-sufficiency can be obtained assuming a blanket with a 90% <sup>6</sup>Li enrichment and a breeder zone radial thickness reduced to 55 cm or with a <sup>6</sup>Li enrichment of 60% and a breeder zone radial depth of 75 cm, the blanket with breeder zone radial length of 60 cm being close to ensure the required TBR.

With respect to the radiation load to the TF-coils the reference blanket meets the limit requirements in the design utilizing Eurofer shield and  $VV_{ITER}$  with exception of the integral absorbed dose in epoxy.

The breeder blanket with 60 cm radial depth demonstrates good performance with respect to the radiation loads to the TF-coil except the integral absorbed dose in epoxy in both designs, namely these using WC shield and  $VV_{REF}$  and Eurofer shield and  $VV_{ITER}$ .

The blanket with radial length of 55 cm shows the best performance by fulfilling the goal of the parametric study to meet the target TBR at reduced radial length and in terms of radiation loads on TF-coils below the design limit. The design limits can be met utilizing

shield/vacuum vessel configuration of WC composite and  $VV_{REF}$  or two-component shield of Eurofer and WC and  $VV_{ITER}$ . It is expected that the later configuration would ensure, even in larger extend, the integral absorbed dose below the required limit for blanket designs with breeder zone thickness of 75cm and 60 cm.

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