

Neutronic analysis of a dual He/LiPb coolant breeding blanket for DEMO

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ABSTRACT

A conceptual design of a DEMO fusion reactor is being developed under the *Spanish Breeding Blanket Technology Programme*: TECNO-FUS based on a He/LiPb dual coolant blanket as reference design option. The following issues have been analyzed to address the demonstration of the neutronic reliability of this conceptual blanket design: power amplification capacity of the blanket, tritium breeding capability for fuel self-sufficiency, power deposition due to nuclear heating in superconducting coils and material damage (dpa, gas production) to estimate the operational life of the steel-made structural components in the blanket and vacuum vessel (VV). In order to optimize the shielding of the coils different combinations of water and steel have been considered for the gap of the VV. The used neutron source is based on an axisymmetric 2D fusion reaction profile for the given plasma equilibrium configuration. MCNPX has been used for transport calculations and ACAB has been used to handle gas production and damage energy cross sections.

1. Introduction

A conceptual blanket design is being developed based on He/LiPb dual-coolant technology for a DEMO fusion reactor within the frame of the Spanish national project TECNO-FUS. The design is initially based on the C model of the European Fusion Power Plant Conceptual Study (PPCS) [1] intending to fully update reactor parameters and plant systems specifications. The average neutron wall loading (NWL) is 2.1 MW/m² for the blanket with 3450 MW of fusion power.

The design has a dually-cooled breeding zone with Pb-15.7 ⁶Li (90% ⁶Li enrichment) serving as breeder and coolant, and with pressurized helium as primary coolant. The reduced activation ferritic-martensitic (RAFM) steel Eurofer-97 is used as structural material in the blanket and 316-LN austenitic steel in the vacuum vessel (VV). The flow channels inserts of the LiPb liquid metal are made of SiC serving as electrical and thermal insulator.

In order to analyze the neutronic behavior of the design, we are focusing on several quantities: (i) tritium breeding ratio (TBR) to address the fuel self-sufficiency; (ii) power deposition due to nuclear heating in both the coils, to ensure its superconductor behavior, and the blanket, to calculate the power amplification factor; and (iii) neutron induced radiation damage to estimate the lifetime of the structural components made of steel in the blan-

ket and the VV. Furthermore, different combinations of water and steel are placed in the gap of the VV in order to optimize the coils shielding.

This work shows 3D radiation transport calculations using MCNPX [2] with ENDF/B-VII nuclear data. We have used a CATIA CAD model and MCAM [3] to convert it into a MCNP input. Nuclear damage cross sections (damage energy and He and H production), processed in the VITAMIN-J 211 format from ENDF/B-VII [4], have been used for damage calculations handled with ACAB [5] using the spatial average flux as input.

2. Methodology

2.1. Calculation procedure

The followed calculation outline consists of four steps linked automatically; (i) Excel/CATIA interface that generate a 3D simplified geometrical model in STEP format, allowing a parametric study, (ii) MCAM to convert CAD geometry (in STEP format) into MCNP geometry, (iii) MCNP for radiation transport calculations, (iv) ACAB for damage calculations.

2.2. Source description

A realistic model for the neutron source has been used in transport calculations. This model is based on the axisymmetric 2D fusion reaction profile for the given plasma equilibrium configuration [6]. The use of a realistic source has a considerable impact in the

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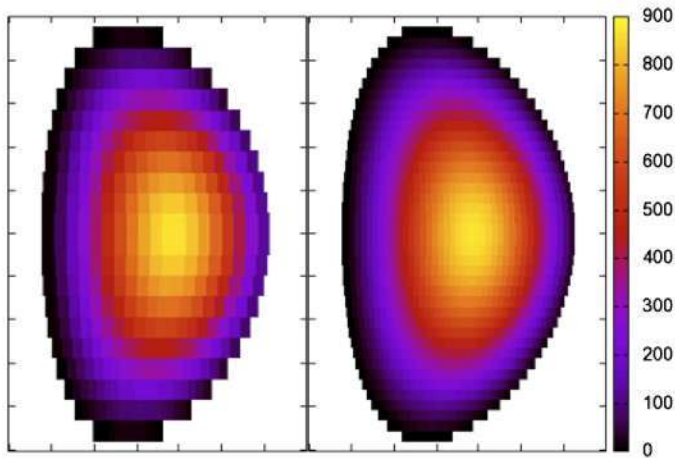


Fig. 1. Different resolution for 2D source profiles.

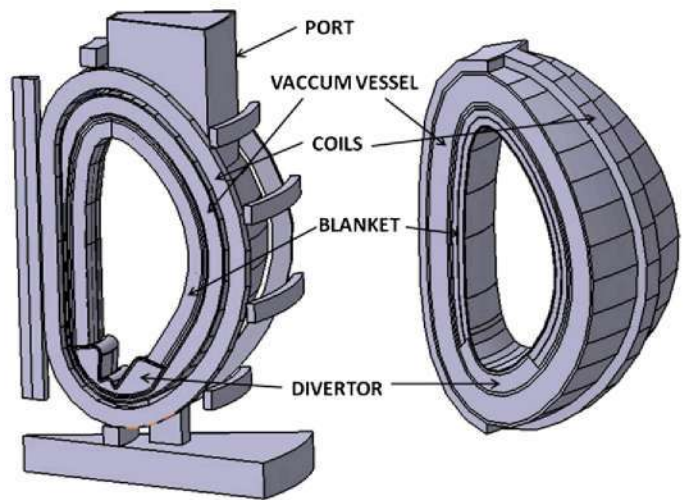


Fig. 3. Reactor geometry models: detailed (reference) geometry model (left) and simplified (neutronic) geometry model (right).

distribution of neutrons between inboard and outboard regions, as well as for accurately calculating the fraction of neutrons crossing the vacuum vessel or divertor regions.

In order to implement this realistic source distribution into MCNPX a discretization process has been applied. In Fig. 1, it can be seen two examples of discretization resolution for the 2D source profile distribution in arbitrary units. The one on the right is used in this work. The source volume has been divided into an inhomogeneous set of annular cells with independent flat source distributions with same integrated intensity as the original source. This method, upon sufficient refinement, produces a source as close to the actual one as required for a given precision. The process has been automated for arbitrary refinement and direct use in MCNP and MCNPX codes.

Table 1

Results of the TBR and the peak power in the coils for the different combinations of water and steel studied in the VV gap.

H ₂ O (cm)	Steel (cm)	Coolant VV	TBR	Peak Power (W/m ³)
0	0	He	1.173	2.09 × 10 ⁵
10	0	He	1.172	7.17 × 10 ³
20	0	He	1.172	3.12 × 10 ³
10	0	H ₂ O	1.170	4.66 × 10 ³
20	0	H ₂ O	1.169	2.49 × 10 ³
10	10	He	1.172	1.01 × 10 ³
10	10	H ₂ O	1.169	6.20 × 10 ²

2.3. Geometry

The used geometry for the particle transport calculations (neutronic model) is a simplification of a 3D detailed geometry (ref-

erence model), see Fig. 3. It is a torus sector of 30° (with reflective boundary conditions), which is layered taking into account that the internal D-shape facing the plasma has the same dimensions as in the reference model. The different materials that compose the blanket have been placed keeping the same volumes as in the reference model. Some of these materials have been homogenized ensuring that the neutrons and gammas cross the same average density in both reference and neutronic models. The divertor zone has been simulated in a very simple way but allow us to keep the original relation between the volumes of the blanket layers and its thickness.

The Fig. 3 shows a scheme of the layers and materials placed in the neutronic model, with its radial distance from the first wall (each box can be formed by one or more layers). The numbered blocks and the numbers beside the Eurofer label in Fig. 3 are the different zones where power deposition, see Table 2, and damage responses, see Table 3, have been calculated. In analyzing the different combinations of water and steel in the gap of the VV, we use 10 cm-thickness layers located just beside the inner wall.

Table 2

Power deposition results in the different blocks of the blanket due to neutrons and gammas.

Blocks	Neutrons (MW)	Gammas (MW)	Total (MW)
I	9.09 × 10 ¹	3.02 · 10 ²	3.93 × 10 ²
II	9.43 × 10 ¹	3.57 · 10 ²	4.51 × 10 ²
III	2.26 × 10 ¹	7.93 · 10 ¹	1.02 × 10 ²
IV	9.19 × 10 ²	5.55 · 10 ²	1.47 × 10 ³
V	1.64	2.35	3.98
VI	1.87 × 10 ²	2.66 · 10 ¹	2.14 × 10 ²
VII	2.22	1.46 · 10 ¹	1.68 × 10 ¹
Total	1.32 × 10 ³	1.34 · 10 ³	2.65 × 10 ³

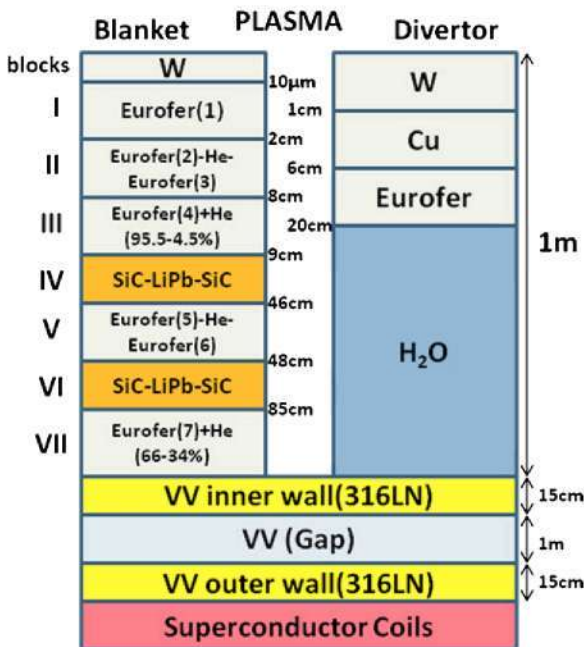


Fig. 2. Different layers of the neutronic model from the plasma to the superconductor coils.

Table 3
DPA and gas production results per year (100% availability).

Zones	DPA/y	H appm/y	He appm/y	He/DPA appm/dpa
1	24.9	884	222	8.92
2	21.0	634	160	7.60
3	18.4	480	121	6.56
4	16.8	386	97	5.77
5	2.15	6.4	1.8	0.84
6	2.07	5.9	1.7	0.81
7	0.17	0.07	0.1	0.60
VV	0.025	0.025	0.11	4.49

The interface between Excel macros and CATIA allow us to change the thickness of the layers in the neutronic model automatically, which is very helpful for neutronic parametric studies.

3. Results and calculations

3.1. Tritium breeding ratio

The tritium breeding ratio has been calculated as the tritium production per neutron crossing the first surface of the blanket. The calculated neutron current in this surface shows that around 85% of the source neutrons cross it. The tritium production due to ⁶Li and ⁷Li (90% ⁶Li enrichment) has been taken into account, but the contribution of ⁷Li is insignificant, <0.1%.

Table 1 shows that the calculated TBR for the different combinations of water and steel in the VV is almost the same (the substitution of He or vacuum for H₂O produces a smaller fluctuation in the value, no more than 1%) and is ~1.17. The calculated TBR is higher than the reference value for tritium self-sufficiency, 1.10 [7], which accounts for the effects of nuclear data uncertainties, the port zones without breeder material and the tritium losses in the fuel cycle.

It is worthwhile mentioning that with this neutronic model we have not considered the gaps between the different blankets units.

3.2. Coils shielding

The kerma due to neutrons and gammas has been calculated for different parts of the toroidal field (TF) coil to estimate the peak power due to nuclear heating deposition. Different combinations of water and steel, located in the VV gap, are considered (the two first columns of Table 1) in order to optimize the VV design. The first column in Table 1 corresponds to the layer just beside the inner wall of the VV and the second column corresponds to the next one in the outer direction.

The results obtained are shown in Table 1. It can be seen that 20 cm of any combination of the considered materials is enough to fulfill the limit referred to the peak of nuclear heating in the winding pack. For a DEMO this limit is $5 \times 10^3 \text{ W/m}^3$, see Ref. [7]. Regarding heat deposition, the best combination is water as coolant of the VV and a combination of materials of 10 cm of water and 10 cm of steel in the gap of VV, obtaining a safety factor ~10. Furthermore, this thickness of 20 cm fulfills the geometry limitation in the inboard gap of the VV. The peak power is located in the equatorial plane of the inboard coils in all cases, and the asterisk in Fig. 4 shows the critical zone.

The contribution to power deposition in the coils due to the neutrons travelling through the port (see Fig. 2 left) has not been considered in this model.

3.3. Power deposition

The power deposition due to nuclear heating has been calculated in the blanket to estimate its power amplification factor. Presented

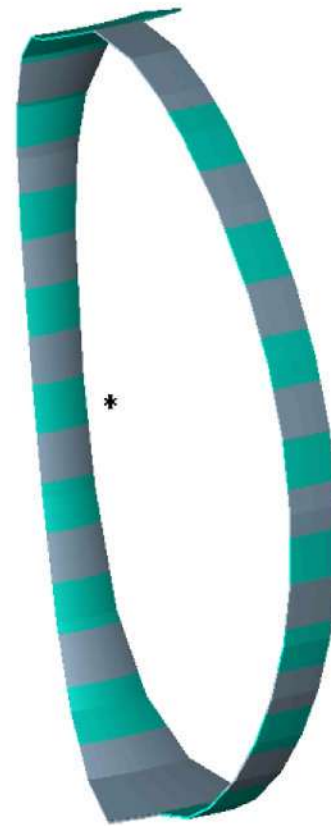


Fig. 4. Different zones used in the heating deposition calculations of the coils. *Peak power position.

calculations in this section are for the case of the VV with a void gap.

The power deposition has been calculated for the different blocks (Fig. 2) of the blanket. Total and break down results are shown in Fig. 5 and Table 2. It can be seen that the total neutron and photon contribution is similar. The most important power deposition is in the two first layers of Eurofer and the first of LiPb.

In calculating the power amplification factor, we have taken into account 85% of the source neutrons (as in TBR calculations) because the divertor zone is not considered. With this considerations and a 2790 MW fusion neutron source, the amplification factor is equal to 1.12, a little bit lower than the value previously reported for the C model, 1.17 [1].

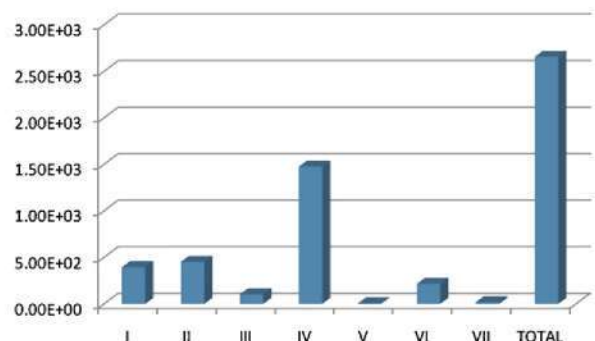


Fig. 5. Power deposition in MW in the different blocks, values in Table 2.

3.4. Damage calculations

Radiation damage produced in the Eurofer and SS316LN has been studied in order to estimate the lifetime of the steel-made structural components of the blanket and VV. Displacement per atom (dpa) and gas production (He and H) have been calculated in the different layers of Eurofer (see numbers in Fig. 3) and for the SS316LN in the VV inner wall, using the flux computed with MCNP and the damage energy cross section from [4] handled with ACAB. The NRT model [8] is used to convert the damage energy into dpa.

The results for the different layers of Eurofer in the outer direction are shown in Table 3. Assuming that ferritic steels (or ODS versions) eventually will be able to operate up to damage levels approaching 150–200 dpa, Ref [9], the structural material of the first wall (zone 1) could have a service life around 6 years. The accumulated damage in the last shielding, located in front of the VV (zone 7), is 6.8 dpa and 4 appm of He after 40 years of irradiation. The accumulated He production is higher than reweldability limit, around 1 appm of He, reported in [10] to be fulfilled for a permanent component.

As for the austenitic steel 316LN of the VV inner wall, the values obtained for 40 years of operation are 0.1 dpa and 4.4 appm of He. The helium production is lower than the 10 appm limit for austenitic steels reweldability [11].

4. Summary and conclusions

The neutronic performance of a conceptual design of a dual LiPb/He coolant breeding blanket has been assessed. Within the assumed neutronic model, the TBR estimation is greater than the required design value and the power amplification factor is around 1.12. The gap of the VV has been optimized to fulfill the limit of the power deposition in the coils, observing the inboard space limitation. The lifetime of the first wall of the blanket is around 6 years and

the last shielding have a He production higher than the reference limit for reweldability. The VV fulfill with all the limits considered for 40 years of operation.

A comprehensive procedure from the excel/CATIA interface to MCNP model via MCAM has been implemented, and is found it very helpful for parametric analysis allowing to change the thickness of the layers automatically. This interface produces neutronic model geometries that are very useful as starting point for blanket neutronic studies.

The next step in the neutronic analysis of the blanket design is to use a more detailed neutronic model where the port, the inboard space limitation, a more realistic divertor model and the gaps between the blanket units are included.

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