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Effect of engineering constraints on charged particle wall heat loads in DEMO

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The presently predicted total heating power of the demonstration fusion reactor DEMO is ~3 times the ITER value, while the major radius is only 1.5 times larger. The current DEMO technological wall heat load removal capability is limited to ~1MW/m^2, due to structural material limitations and the tritium breeding requirements, while the ITER first wall (FW) is designed for values up to 4.7MW/m^2. This paper focuses on the evaluation of the effect of the engineering constraints on the required limitation of charged particle heat load.

First, a 2D field-mapping tool is used together with a simple model to take into account the peaking factors present on an engineering 3D wall design. A sensitivity analysis is performed on a set of realistic FW 3D features considering a preliminary estimate of misalignments. The impact on the heat flux to the wall due to different machine geometries, plasma shapes variation, stationary plasma and plasma transients, is presented. An automatic procedure to define the 2D poloidal contour of the FW for minimized particle loads is presented. A subset of the resulting engineering wall design is analyzed using the 3D field line tracing code PFCflux.

Keywords: DEMO, PFC, first wall, power exhaust.

1. Introduction

The design of the demonstration fusion reactor DEMO presents challenges beyond those faced by the ITER project and may require the implementation of different solutions. One of the biggest challenges is managing the heat flux to the main chamber wall. The presently predicted total heating power in DEMO 450 MW, more than 3 times that predicted for ITER value, while the major radius is only 1.5 times larger [1]. Furthermore the present DEMO technological wall constraints limits the maximum wall load to ~1MW/m^2 [2] [5], while the ITER first wall is designed for values up to 4.7MW/m^2 [3]. While both DEMO and ITER consider tungsten as armor material for the first wall, the main difference lays in the heat sink, where in DEMO Eurofer is considered as a structural material, because the copper alloy used in ITER, CuCrZr, would suffers from excessive embrittlement and swelling under high neutron irradiation [4]. In addition CuCrZr has an upper temperature limit (~300-350°C) that is incompatible with the desired higher coolant temperature in DEMO to allow for efficient power conversion, 280-320°C in DEMO in case of water or 300-380°C in case of helium (ITER uses 70°C [6]). The thermal conductivity of Eurofer is ~11 times lower than CuCrZr. This is the main cause of the lower heat flux capability as compared to the ITER FW.

First, a series of optimizations on the plasma and first wall 2D shape is presented, starting from the DEMO single null baseline scenario. A 2D field-mapping tool is employed together with a simple model which takes into account some of the 3D features. A series of sensitivity studies is performed on some realistic 3D feature of an engineering wall and a possible set of misalignments. Finally a series of nonlinear simulations, using the CREATE-NL code is run, to evaluate the plasma shape variations under a list of perturbations. An automatic procedure is developed which calculates a 2D poloidal contour of the FW with heat flux below a prescribed threshold. A subset of the resulting configurations is finally analyzed using the 3D field line tracing code PFCflux, improved to verify the power balance, to verify the heat flux onto a 3D engineering model of the first wall.

2. 2D field mapping and 3D peaking factor model

A simple 2D field line mapping tool was developed to run a large set of FW shapes. Starting from a DEMO 2D equilibria, the tool distributes the power in the scrape-off layer (SOL), P_{SOL}, in the flux lines starting from the plasma boundary at the outer mid-plane, at the z coordinate of the current centroid, with an e-folding length decay function, as in [7]. An example is hereafter reported:

$$¿\vee¿ = \frac{P_{SOL}}{2\cdot\pi\cdot R\cdot\lambda_q} e^{\frac{-d}{\lambda_q}} \quad , \qquad (1)$$

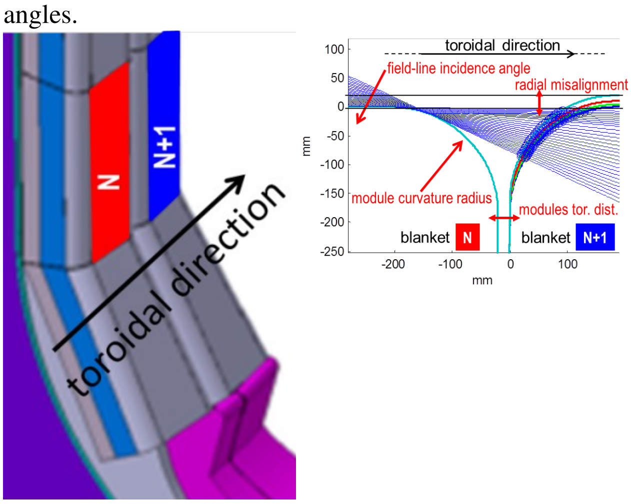
$q_{¿}$

where R represents the plasma major radius, λ_q the e-folding length and d the distance at the outer mid-plane from the plasma boundary up to the wall. A consistency check was enforced in the calculation satisfying the power balance (*i.e.* the total heat flux integrated over the wall must be equivalent to P_{SOL}) within ±10% to allow for numerical errors and wall size resolution. As the main focus of this work is on the heat flux which is deposited on the FW, sensitivity scans were performed on different values of the far-SOL e-folding length, up to 17cm as for ITER [4], and the power that flows in this channel. The resulting conservative set of assumptions, presented in [8], which will be used in the following part of the work, is represented by the simple model which assumes a total power in the far-SOL of 60MW, and λ_q = 10cm.

A tool was developed, based on [9], to evaluate the local thermal peaking factor that occurs on a particular tile due to positional or angular misalignments, or due to the exposure of the tile edge to the particle flux. This calculation was applied to the present design of the DEMO blanket first wall, to link the flux lines toroidal incidence angle to the peaking factor for a series of blanket modules design parameters, and a scan of the manufacturing and installation tolerances, *e.g.*:

- toroidal distance between two adjacent blanket bananas;
- first wall curvature radius in the toroidal direction;
- radial misalignment of the bananas.

An example of the resulting geometrical peaking factor is reported in Fig.1, where the resulting geometrical peaking factor, due to the larger incidence angle on the rounded corner of the tile with respect to the flat face of an ideally axisymmetric FW, span for toroidal incidence flux angle of 1 degree, from factor 15 to 30, from radial misalignment equal to zero and to 20mm respectively, while it became less important for larger incidence angles.



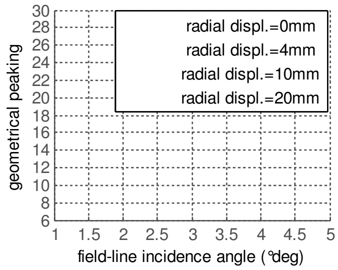


Fig. 1 Example of the application of the peaking factor model to the present blanket design, with: toroidal distance between modules of 2cm, maximum module curvature radius of 18cm, and modules radial displacement of [0 (no displacement), 4, 10, 20] mm.

As in the present design the blanket segments are more than 9m high and due to the integration of complex heat extraction and tritium breeding systems [2] [3] the assumption of a maximum radial displacement of 20mm may be optimistic given manufacturing and installation tolerances and thermal deformation. Nevertheless the tool presented aims at developing the methodology for the automatic design of the 2D first wall with certain geometrical parameters, including the expected tolerances, as soon as they will became available, and to help prescribing as much as possible maximum tolerances and evaluating their impact in terms of heat flux peaking factor. The results of the application of the 2D field mapping tool, together with the peaking factor tool allowed obtaining instantly a rough estimate of the heat flux to the wall, scanning a large set of variables and assumptions, including:

- plasma shape/position, nominal and perturbed;
- power in the SOL and e-folding length;
- different poloidal FW profiles;
- 3D engineering wall features (modules toroidal distance, curvature radius);
- Present assumptions on misalignment tolerances.

The preliminary results, based on initial FW proposals [10], have underlined the zones of the wall which presented the highest heat flux values, namely in the divertor baffles and close to the upper secondary magnetic null, as expectable from the higher flux expansion in such regions. By considering the peaking factor model, for the present preliminary 3D features considered engineering wall, the local heat flux in the upper region resulted in a value above 5MW/m^2, well above the technological limitations of 1MW/m^2, even in the plasma nominal shape case, as shown in Fig.2

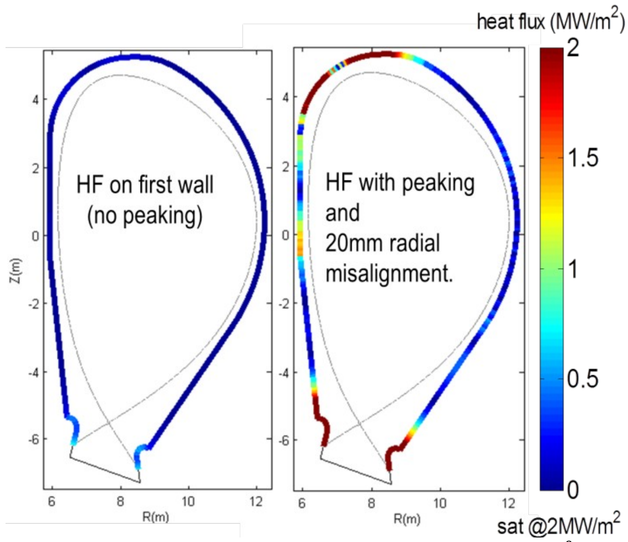


Fig. 2 Computed heat flux (color saturated at 2MW/m^2) for demo baseline equilibrium, with: (plasma parameters) $P_{far\text{-}SOL}$ = 60MW/m^2, λ_q = 10cm, (FW module geometrical parameters) tor. distance 2cm, mod. tor. length 1m, curvature radius 10 cm.

3. Wall heat flux sensitivity to plasma shape optimization and perturbation

The methodology presented in the previous chapter has been employed to evaluate the impact on the heat flux of the plasma shape variation, such as the position of the upper secondary magnetic null and the plasma triangularity, calculated with the CEATE f.e.m. equilibrium code [11]. Using the same plasma and geometrical parameters described in Fig.2, and keeping the FW fixed, the results had shown an increase of the local heat flux up to 10MW/m^2, from 5MW/m^2 of the nominal case, with the upper null closer to the plasma, *i.e.* the null entering in the chamber. In contrary the heat flux decreased to 2.5MW/m^2 in case of the secondary null being further outside the chamber. The results on the plasma with reduced triangularity had also shown a trend to reduce the heat flux to the upper wall to 3.7MW/m^2 for a clockwise movement, and to 2.8MW/m^2 for an even further clockwise movement, see Fig. 3.

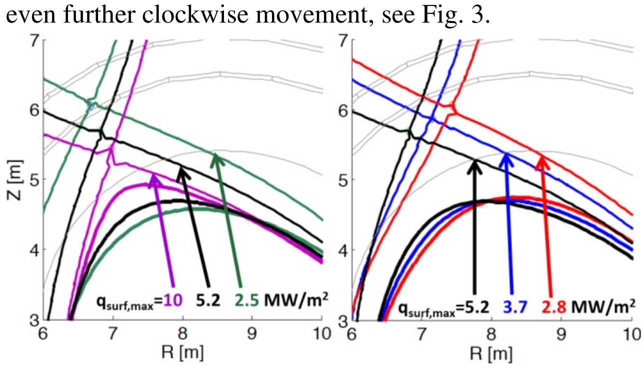


Fig. 3 heat flux to the wall evaluated for the nominal equilibrium (black), upper x-point moving outwards of 40cm (green), and inwards of 35cm (violet), and moving clockwise of 35cm(blue) and of 70cm (red).

Also a perturbation analysis was performed using a preliminary list of disturbances, such as ELMs, minor disruptions, unforeseen H-L transitions. This was obtained running a fully nonlinear control simulation with the CREATE-NL code, including the passive conductive structures, and modelling the disturbances as variations of the plasma internal parameters $\Delta\beta_{pol}$ and Δli, and applying the control action, which counteracts the movement immediately after the disturbances (best achievable performance controller). The modelling of ELM perturbation as $\Delta\beta_{pol}$ and Δli variation has been validated in experimental device, as JET [12] [13], as the one that gives the same plasma shape movement as during the experiment, while for the other disturbances a preliminary model was applied based on experimental variation of such internal plasma parameters. The outcome of this analysis was a set of different maximum plasma movements, due to the various perturbations, which was used to evaluate the different wall heat flux in such off nominal cases. The unforeseen H-L transition was excluded from this set of perturbations, as it resulted in a plasma movement towards the inboard wall in excess of 30cm, resulting in a plasma limiter configuration. Such a case will be revisited as soon as a realistic shape controller and power supply/coil limitations will be available for DEMO.

4. 3D field-line tracing with PFCflux

The PFCflux code [14] has been employed to calculate the heat flux density to the wall for a subset of the wall geometry, plasma perturbations and assumptions on the $P_{far\text{-}SOL}$ and λ_q. For the DEMO FW calculation an important new feature was developed in the code, which is the fulfilment of the power balance within ±10%, as for the 2D tool, including the tile shadowing calculation. The code was employed on the initial DEMO engineering wall proposals, where preliminary simple assumptions were made, such as first wall modules with flat faces extended for ~ 1m in the toroidal direction, and different poloidal segmentation strategies [10] (*e.g.* with 14 and 18 poloidally distributed modules, Fig. 4). A limited number of initial calculations were performed using the nominal plasma shape and some perturbed equilibria, as described in the previous paragraph. A tuning of the code was also performed to account for the –DEMO geometrical parameters and resulting connection lengths. The results had shown a high heat flux on the divertor baffles and the upper FW regions, as reported in Tab.1, in fair agreement with the 2D calculation of paragraph 2.

5. Automatic 2D FW contour design

As the heat fluxes to the wall strongly depend both on the first wall 2D profile and the plasma shape and position, an optimization on the wall shape was performed. An algorithm was implemented, based on the presented models and results, to automatically draw a 2D poloidal FW contour which fulfils a set of desired maximum wall heat flux criteria based on:

- a set of assumptions for $P_{far\text{-}SOL}$ and λ_q,
- a desired limit of heat flux on the FW,
- a list of misalignments assumptions,
- a series of perturbed equilibria,
- a minimum distance plasma-wall.

The last point is introduced as a preliminary constraint coming from simple assumptions on the plasma shape control. The wall design procedure is based on the drawing of a collection of straight lines, centred in the plasma current centroid, and with a predefined angle step α. The first coordinate of the wall design is, for example, chosen at the outer mid-plane at the minimum distance wall prescribed. For each following step the new point is chosen on the adjacent line such as that the heat flux to the wall, calculated as in paragraph 2, including the peaking factor, is smaller than a desired value, ensuring at the same time the prescribed minimum plasma-wall distance. This procedure was repeated for each of the flux maps of the several perturbed equilibria considered, and the resulting final proposal was the convex-hull of each 2D wall profile (one for each equilibrium), with the only limitation that no convex patterns were allowed. The flexibility and speed of the calculation will allow updating any of the criteria, as soon as a new assumption/information is available, and come up with a new proposal.

7. Conclusions

In this work a preliminary analysis of first wall heat flux is presented under the present DEMO baseline plasma and technological assumptions. Due to the limitation of the presently considered technology on the FW heat sink, based on Eurofer, the heat flux density that can be exhausted on the wall is limited to ~1MW/m^2, well below the ITER FW capacity of up to 4.7MW/m^2.

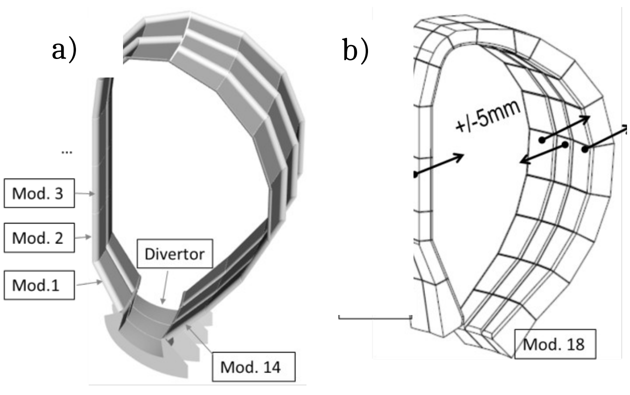


Fig. 4 Examples of preliminary 3D engineering model used for the PFCflux calculations: a) with 14 poloidal modules, b) with 18modules and ±5mm radial misalignment.

Heat flux [MW/m^2]			
a) 14 mod.	b) 18 modules, misalignment:		
	no	10mm	10mm

			inboard	outboard
Mod #1	5.3	7.4	8.4	7.4
Mod #2	0	4.5	6.8	4.5
Mod #3	0	0	0	0
Mod #4	0	0	0	0
Mod #5	0	0	0	0
Mod #6	2.9	0	0	0
Mod #7	2.8	0	0	0
Mod #8	0.7	1.4	1.7	1.4
Mod #9	0.08	1.4	2.3	1.6
Mod #10	0.13	1.3	1.3	1.9
Mod #11	0.29	1.1	1.1	1.2
Mod #12	0.3	0.3	0.3	0.3
Mod #13	0.3	0.3	0.3	0.3
Mod #14	4.9	0.4	0.4	0.4
Mod #15	/	0.2	0.2	0.2
Mod #16	/	0.05	0.05	0.03
Mod #17	/	0.9	0.9	1.6
Mod #18	/	8.9	8.9	12.6

Tab.1 Heat flux calculation for the 14 modules, and with 18 modules with 10mm inner and outer radial misalignment.

The present design foresees large blanket segments with expected relatively large tolerances and deformations. A methodology was developed to perform a sensitivity analysis on the effects on the wall heat flux due to tolerances, on the plasma parameters and machine geometry designs choices. All these different physics and technical aspects have a different level of maturity, and the proposed methodology will help to evaluate and integrate any new assumptions as they become available, and to prescribe the constraints on them. In addition an automatic procedure was developed to design the 2D poloidal profile in such a way that a prescribed maximum desired heat flux density on the wall is not exceeded. The results on the preliminary designs have also shown that the heat flux peaking on the module edges is increased by a factor of more than 10 and hence significantly exceeding the technological limitations. While this can be partially solved by applying roof top tiles for the static plasma configurations, the real challenge remains the transient loads, such as H-L transitions and mini disruptions, when the plasma moves significantly from the nominal position up to the point that it passes to a limiter configuration. For such occurrences the Eurofer technology does not seems to have a solution yet, and some other solution is needed, such as discrete high heat flux limiters in specific zones. The implication on the remote maintainability and the effect on the tritium breeding ratio of such elements is presently being evaluated.

Acknowledgments

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