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# **Towards an Assessment of Alternative Divertor Solutions for DEMO**

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## **Introduction**

The European roadmap for fusion energy [1] has identified heat exhaust as a major challenge towards the realisation of magnetic confinement fusion. Since the scrape off layer (SOL) width is believed to be independent of the size of the device, the exhaust power increases faster than the area where the plasma interacts with the wall. A reactor must, therefore, harness an even greater heat flux than ITER. At the same time the higher particle and neutron fluence in a reactor imposes stronger constraints on the target materials and the admissible erosion. In the current baseline scenario, which foresees a single null magnetic configuration and divertor targets made out of tungsten (W) this must be achieved by primarily increasing the radiation power. It is, however, uncertain whether a higher radiation fraction is compatible with the required energy confinement and whether transients can be sufficiently suppressed to maintain a detached divertor. In addition there are uncertainties in the predicted power decay length, which may modify the exhaust requirements. In order to mitigate the risk that the baseline scenario does not extrapolate from ITER to a DEMO reactor, the EUROfusion consortium is assessing alternative divertor solutions. The assessment focuses on solutions that have a high potential to be fully developed for a DEMO in the 2040s.

In the assessment it is first evaluated whether a proposed alternative could likely be implemented in a DEMO on the foreseen timescale. The assessment then seeks to quantify the potential benefits and costs of the alternatives compared to the baseline solution. The comparison with the baseline solution should decrease the effect of any systematic errors in the extrapolation from present day devices to a reactor. Considered alternatives can be divided into non-standard magnetic configurations, which may decrease the heat and particle flux to the divertor targets, and liquid metal target armor, which may increase their exhaust capabilities.

## **Alternative magnetic configurations**

The main geometric modification of the divertor are the increase of the flux expansion at the target, an increase of the major radius of the target and a reduction of the gradient of the poloidal field in the null point, associated with the *X divertor* (XD) [2], *Super-X divertor* (SXD) [3] and *Snowflake divertor* (SFD) [4] concepts, respectively. All configurations have at least partially

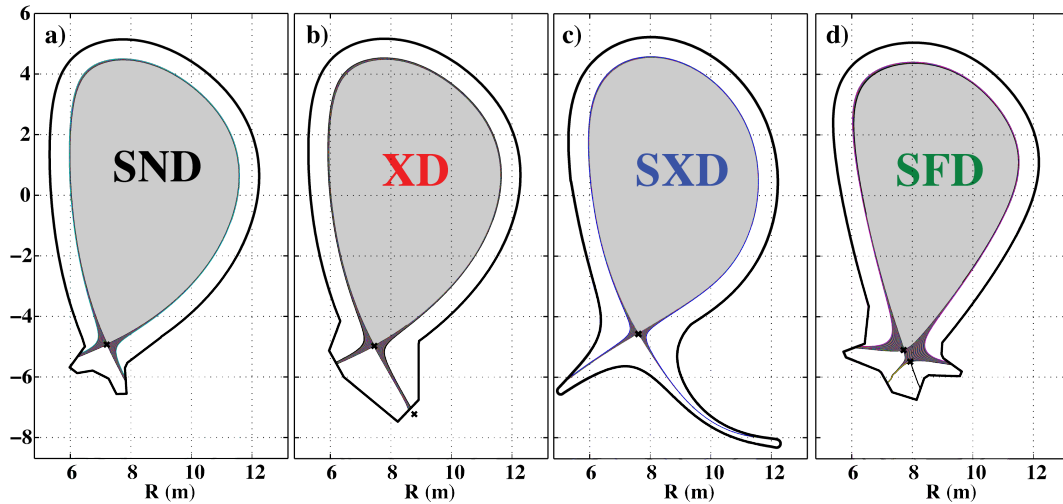


Figure 1: (a) EU A=3.1 DEMO reference configuration (2014) and prototypes for the (b) X divertor, (c) Super-X divertor and (d) Snowflake divertor.

been realised in experiments - in case of the XD even before the concept was named and fully formulated. The SXD is presently only considered as a solution for the outer, usually heavier loaded, divertor leg.

The assessment is carried out in two steps. In the first step a DEMO size prototype of each alternative configuration is generated. The prototype configurations possess the key characteristics of the proposed alternatives, but are not yet optimised, Fig. 1. Core parameters and plasma shape are kept similar to the reference configuration, which has a single null divertor (SND) and an aspect ratio,  $A = 3.1$ , and is designed for an electric power output of  $P_{\text{elec}} = 500$  MW (major radius,  $R = 8.8$  m, toroidal field,  $B_t = 5.8$  T, plasma current,  $I_p = 20$  MA). The axisymmetric targets are inclined for recycling neutrals to reflect towards the separatrix, also referred to as *closed* divertor configurations, but maintain a minimum grazing angle of magnetic field lines of 1.5 Deg.. Coils are always placed allowing for at least 80 cm of neutron shielding. The XD and SXD prototypes only include an alternative solution for the outer divertor leg.

In the second step, which is presently carried out, the prototypes are optimised to reduce their costs and increase the potential benefit using a systematic approach [5]. Considered beneficial geometric quantities are the connection length, the SOL volume, a flaring of the flux surfaces towards the target quantified by the ratio of flux expansion at the target and the minimum flux expansion along the divertor leg and the ratio of the target radius and the null point radius. Considered costs are the total poloidal field coil currents (weighted with their major radius as a proxy for the required conductor volume), the forces on the poloidal field coils, the current of poloidal field coils located inside the toroidal field coils (weighted with their major radius) and the ratio of toroidal field coil volume and the plasma volume.

The extrapolation to DEMO must include a physics model as present day devices are not capable of testing alternative configurations under DEMO relevant divertor conditions. The performance of the alternative configurations is, therefore, evaluated using a set of boundary codes with various degrees of complexity. Technical challenges include the grid generation for the new SFD topologies and an increased run time of the codes due the size of the configura-

tions. So far SFD grids for the SFD configuration have been generated for the EMC3-Eirene and SOLEDGE2D codes. Also the run time of the various codes has been modest, but neither a kinetic descriptions of neutrals nor the inclusion of drifts have so far been attempted.

In order to compare the ability of the proposed alternatives to meet the exhaust challenge several metrics have been proposed: 1. plasma density at the onset of detachment, 2. SOL impurity concentration for the required high divertor power loss and 3. the maximum divertor power loss before the loss of stability.

While various aspects of the selected models have been validated within the fusion community, all of the models include only a diffusive cross-field transport with empirical diffusion coefficient. It is well known that convective transport is important, in particular, in the high density detached operating regimes [6]. It is, furthermore, speculated that the low poloidal field in the SFD increases turbulent cross-field transport. However, models of turbulent transport are presently not capable of addressing novel magnetic configurations and the assessment can, therefore, not be comprehensive.

### **Liquid metal plasma facing components**

A variety of liquid metal target concepts are proposed. They can be distinguished by their principle heat removal mechanism, which can be conduction, convection or evaporation. While concepts that are based on convection or evaporation promise a superior heat removal capability, they also face additional challenges such as the stability of the flowing layer and particle exhaust. The assessment of liquid metals, therefore, focuses on static liquids, whose feasibility is a prerequisite for more advanced concepts. A static liquid can be stabilised in capillary porous system (CPS), made out of a solid material [7]. Considered metals are lithium (Li), due to its compatibility with good plasma performance and tin (Sn), due to its high evaporation temperature. In addition Li/Sn alloys are also investigated as they promise the advantageous properties of both components. Before weighing benefits against costs, the physics and technological feasibility of a liquid metal based concept has to be established.

The heat removal capability is greatly determined by the temperature difference between the target surface and the coolant. The surface temperature is limited by excessive erosion, which is enhanced at higher temperature. In addition to temperature the effective erosion also depends on the surface texture and the plasma pressure. Lastly prompt redeposition can significantly reduce the effective erosion rate. The heat removal capability also depends on the thickness of the structure. Since liquid metals self heal eroded material, the thickness of the armor can be reduced over a solid target. However, structural components and coolant pipes must remain similar. The capability to self heal opens the possibility to tolerate transients that cause some erosion provided that the substrate is not exposed. The tolerance to transient, therefore, depends on the film thickness, which in turn is limited by the surface stability.

A modelling effort has begun to predict the wetting and temperature evolution of a CPS under heat and particle loads. The tolerable increase of the impurity concentration in the SOL is modelled with transport codes with adapted boundary conditions.

With the predictive modelling being limited to various aspects of the CPS based liquid metal solution, the technical feasibility of stationary power and particle exhaust has to be demonstrated in experiments. Effects are investigated separately in laboratory experiments and linear

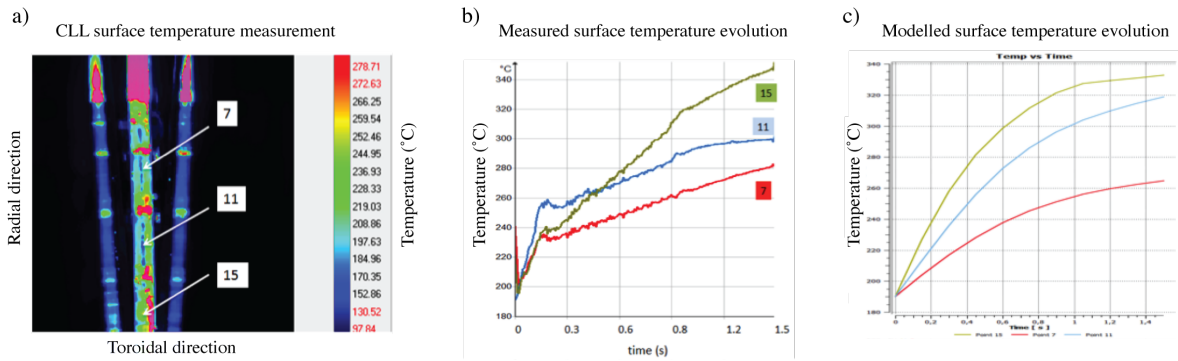


Figure 2: (a) IR measurements of the CLL surface temperature and (b) its evolution are compared with (c) the surface temperature evolution modelled using ANSYS.

plasma devices as well as combined in the FTU tokamak, where experiments with a recently installed cooled liquid Li limiter (CLL) have commenced [8]. Upon conclusion of the CLL experiments it is planned to replace the CLL with a cooled liquid Sn limiter. FTU experiments have shown stability of the CPS system under heat flux of approximately  $2 \text{ MW/m}^2$  in a 1.5 s discharge, but the target temperature and, hence, the heat removal is not stationary, Fig. 2. Present efforts, therefore, aim at an extension of the FTU pulse length to 5 s, which should suffice to demonstrate stationary power exhaust and evaluate the effect on the plasma boundary.

## Conclusion

The EUROfusion consortium is evaluating unconventional power and particle exhaust concepts as an alternative for their DEMO reactor. The assessment has identified the X divertor, Super-X divertor and snowflake divertor configurations as potential alternatives. In addition it is considering a liquid Li, Sn or Li/Sn alloy target based on a capillary porous system. Feasibility, costs and benefits of the alternatives are evaluated and compared to the conventional divertor solution. The assessment result is expected to inform the decision on a European divertor test tokamak facility.

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