# Power Handling and Plasma Protection Aspects that affect the Design of the DEMO Divertor and First Wall

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**Abstract:**

The publication introduces power handling and plasma protection challenges associated with the recent EU DEMO baseline design. Based on this design possible modifications diverging fundamentally from ITER design choices are discussed: A double-null magnetic configuration and the integration of high heat flux limiters at the first wall

## 1   Introduction

The development of a conceptual design for a demonstration fusion power plant (DEMO) is a key priority of the recent European fusion program [1]. The DEMO design and R&D is expected to benefit largely from the experience gained with ITER construction and operation, but there are still outstanding gaps requiring a vigorous physics and technology R&D programme. The constraints coming from specific DEMO requirements (e.g. (1) to maximise machine availability, (2) to select cooling systems and coolant operating conditions for efficient power conversion and electricity production, (3) to enable tritium-breeding to achieve a closed-fuel cycle, (4) to withstand high n-fluence and significant in-vessel radiation damage) bear a strong impact in the design and technology selection process of the components surrounding the plasma. In particular the choice of the divertor

configuration and the first wall design and technology are crucial aspects possibly requiring solutions different from what has been chosen for ITER.

In this publication we discuss two fundamental modifications with respect to the ITER design that might need to be introduced into a DEMO design: (1) A magnetic double-null (DN) configuration and (2) integration of high heat flux limiters at the first wall. We note that there is also a substantial amount of activities towards advanced power handling divertor concepts [2], which is not discussed here.

## 2   The recent DEMO Baseline Design

The reference in this publication is the pulsed 'low extrapolation' DEMO baseline design (EU DEMO1 2015 [3]): $R = 9m$, $I_P = 20MA$, $B_T = 5.7T$, $P_{fus} = 2.0GW$. As ITER it includes a lower single null (LSN) magnetic configuration and a conventional divertor, which is closed (i.e. angle between field line and target plate in the projection on the R-Z-plane $\ll 45°$). As a starting point a first wall geometry has been chosen, which is - similar to ITER - within the limits of the breeding blanket segmentation and manufacturing possibilities aligned to the flux surfaces. The baseline first wall design assumes a few mm of W amour joined onto a EUROFER-97 structure integrating numerous parallel cooling channels a few millimeters below its surface. The options for the coolant are pressurized $H_2O$ ($300°C, 150MPa$) or He ($> 300°C, 80MPa$).

The **divertor protection problem** in DEMO is expected to be more challenging than in ITER. The total heating power and the thermal energy content are each higher by a factor of $\approx 4$, while the major radius is only $\approx 1.5$ times higher. Also it is not clear, if the idea of exhausting most of the power via core radiation fractions of 65% and more is successful. Even if this can be accomplished and a similar divertor performance, expressed as the power crossing the separatrix $P_{sep}$ normalised by the major radius R, as in ITER can be achieved, the enhanced divertor neutron irradiation might lead to reduced material limits and a lower acceptable $P_{sep}/R$ than in ITER. Furthermore, despite ELM control in DEMO being unresolved, it is quite obvious that an increased limit on the divertor ELM energy density capability is desirable.

Also the **first wall load problem** in DEMO should not be underestimated. Currently a static heat load limit of $\approx 1MW/m^2$ is assumed for standard DEMO first wall components [4]. Under DEMO conditions an enhanced SOL e-folding length $\lambda_q$ for the steady state charged particle transport is expected, due to an enhanced level of blob transport. A conservative assumption is that 20% of $P_{sep}$ is distributed with $\lambda_q = 100mm$ [4]. Charged particle wall loads in DEMO have been analysed in 3D based on an initial first wall design employing ITER tools. Using this assumption in combination with $P_{sep,max} = 1.5P_{sep,nom} = 1.5 \times 1.16P_{LH} = 231MW$, where $P_{LH}$ is the H-mode threshold power evaluated with the scaling from [5], the wall heat loads are found to exceed locally the limit of $\approx 1MW/m^2$ by a factor of 5. While the first wall design employed has clear optimisation potential, a penalty factor accounting for all possible deviations from the idealized situation investigated here, which are introduced during design, manufacturing and assembly, additionally has to be included (ITER: up to 2.44).

The effect of thermal transients is expected to have even a stronger impact on the design than the static loads. Hence, the effects of all possible controllable plasma displacements

need to be investigated. In this context in SN in contrast to DN, especially the region on the top of the machine could be at risk.

The plasma elongation, which is limited by **vertical stability**, has a strong effect on the performance of DEMO [3]. While in the DEMO baseline design the elongation is $\kappa_{95} = 1.59$, for ITER $\kappa_{95} \geq 1.7$ has been chosen, which can be understood reflecting the larger distance in DEMO between plasma and toroidal conducting structures relative to the minor radius in DEMO due to the large radial extent of the breeding blanket. Also, due to the plasma and machine up-down asymmetry, specifically SN designs have the intrinsic problem that radial displacements triggered by perturbations in $\beta_{pol}$ or $li$ can lead to significant vertical displacements.

## 3    Aspects of a Double-Null Configuration

Switching from SN to DN introduces significant modification in numerous areas. In this section an initial overview of this range is provided focussing on power handling and plasma protection aspects.

### 3.1    The cyclic motion around the perfect DN configuration

As a perfect DN configuration cannot be controlled, in contrast to SN in DN there is a continuous switching between topologically different magnetic configurations: DN $\rightarrow$ LSN $\rightarrow$ DN $\rightarrow$ USN $\rightarrow$ DN $\rightarrow$ ... The recent state in this cycle can be expressed as the radial distance between the separatrices connected to lower and upper x-point $dr_{sep}$ or in terms of poloidal flux: $\psi_{N,sep2} = (\psi_{x,up} - \psi_{axis})/(\psi_{x,down} - \psi_{axis})$.

The frequency of this cycle depends on the open loop vertical instability growth rate $\gamma$, the control algorithm, the controller gain and the characteristics of the power supplies. Despite most of this information not being available for DEMO, a first estimate for DEMO is made using the JET vertical position control system as an orientation. A simplified model $dz/dt = \gamma z + Kv$ is assumed, where $z$ is the vertical position of the magnetic axis, $v$ is the control voltage and $K$ is an electro-magnetical constant depending on plasma configuration and on the control circuit. As a normalization the minimum voltage needed to recover a VDE of $z_{VDE}$ is used: $v_{min} = \gamma z_{VDE}/K$. For simplicity a single-level bang bang controller without hysteresis is assumed, which operates following a limit cycle.

In a set of simplified DEMO simulations it is assumed that $z_{VDE,DEMO} = 5cm$ [6] and $\gamma_{DEMO} = 9s^{-1}$. The case of $f/\gamma = 0.5$ and $v/v_{MIN} = \pm 0.65$ constitutes one possible option, with $f = 4.5Hz$ and $\Delta z_{cycle} = 1.5cm$. Equilibrium calculations show this vertical amplitude corresponds to $\Delta\psi_{N,sep2} \approx 0.003$. The associated vertical movement of the plasma current centroid can well be described by a triangle waveform.

### 3.2    No ELM/inter-ELM divertor power loads

Due to the additional divertor, DN has the potential to improve the divertor power load situation in phases between ELMs or in no-ELM scenarios. To perform a first DEMO extrapolation, data from MAST L-mode discharges with attached divertors [7] have been used. Figure 1(a) shows the dependency of the peak power flux density at each divertor normalised by the sum of all four peak power flux densities as a function of $\psi_{N,sep2}$. A significant up-down-asymmetry can be observed. For instance, the highest values at the

upper targets are considerably closer in power load than the ones of the lower targets. Despite this, in the following we use this data to develop an initial DEMO extrapolation.

It is assumed that the dependence shown in figure 1(a) holds also for DEMO, which could be the case, if (1) the distribution of power to the four divertor legs is determined by $\psi_{N,sep2}$ and (2) the relative amount of divertor broadening between the four divertor legs does not change between MAST and DEMO. Although the validity of these assumptions is not clear, this first estimate can provide some orientation. Figure 1(b) shows the estimated evolution of the peak power flux density at the four divertor targets, which are obtained by injecting the evolution of $\psi_{N,sep2}$ (subsec. 3.1) into the tanh-fits shown in figure 1(b). It is assumed that of $P_{sep} = 154MW$ a total power of $P_{tar,tot} = 30MW$ is deposited on all four targets, which corresponds to a total radiation fraction of 93%. The wetted area $A_{wet} = 1.4m^2$ has been used [8]. The peak loads of $8.5MW/m^2$ (out) and $4.2MW/m^2$ (in) correspond to a peak load on the outer divertor of $14.3MW/m^2$ for SN assuming 2/3 of the power going to the outer divertor [9]. Albeit the standard DEMO divertor configuration is planned to be detached, $P_{tar,tot}$ is of realistic order and the distribution to the targets is expected to be largely independent of the detachment state.
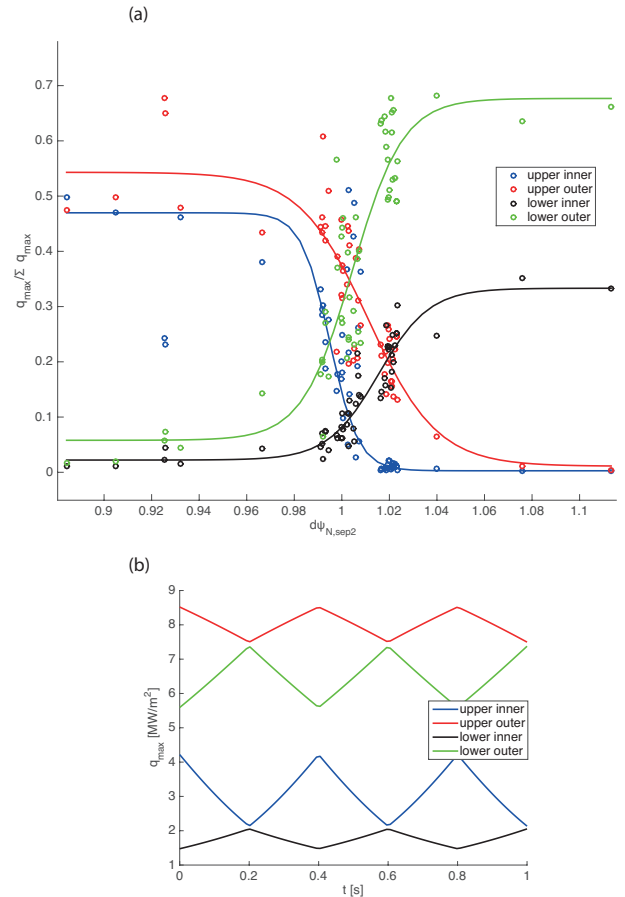


FIG. 1: (a) Circles: Dependency of the peak power flux density at each divertor target normalised by the sum of all four peak power flux densities as a function of $\psi_{N,sep2}$ for L-mode discharges in MAST. Lines: Fits with the function $a + b * tanh\left[c * (\psi_{N,sep2} - d)\right]$
(b) Estimated evolution of the peak power flux density on the four divertor targets in DEMO based on data from MAST, an estimate of the $\psi_{N,sep2}$ evolution (subsection 3.1), $P_{tar,tot} = 30MW$ and $A_{wet} = 1.4m^2$

The impact of the evaluated peak heat flux evolution (DN: upper outer target (fig. 1(b)), SN: outer target with $q_{max} = 14.3MW/m^2$) on an ITER like W monoblock divertor element has been evaluated using `RACLETTE`[1] [10]. Maximum W surface temperatures of $750°C$ (DN) and $1500°C$ (SN)[2] and maximum critical heatflux fractions of 35% (DN) and 63% (SN) have been calculated.

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[1]`RACLETTE` evaluates in 1D with a geometrical 2D correction the thermal response of all components involved in the heat removal process. It includes all key heat transfer processes like evaporation, melting, radiation and water boiling and considers corresponding limits.

[2]W recrystallization starts form $1200°C$.

### 3.3   First wall thermal loads

Moving to a DN configuration can introduce some important modifications of the first wall loads, for which a systematic analysis needs to be carried out.

**Charged particle loads:** In principle power that is transported across the separatrix due to ballooning activity, which happens mainly on the low field side (LFS), is shielded from the high field side (HFS) due to the magnetic topology. It has been found in experiments in Alcator C-Mod, that the pressure e-folding lengths on HFS and LFS SOLs are similar near the separatrix in DN, but the HFS lacks the broad shoulder that is present on the LFS [11]. For SN the broad shoulder is observed on the HFS, whereas the plasma in this region tends to be colder and more dense than on the LFS. The open questions are: (1) For which range of $\psi_{N,sep2}$ is the static charged particle heat flux to the inner wall significantly reduced? (2) Does this lead to a significant advantage in terms of required wall clearance or possible design simplifications?

**Radiation loads:** The main power exhaust strategy for DEMO is to radiate by seeded impurities $\approx 90\%$ of the total heating power $P_\alpha + P_{aux} = 457MW$. It is estimated that for such a plasma in SN between 1/5 to 1/3 of the total radiation power is originating from the x-point region. It is very questionable, if a situation with a second x-point can be controlled in a way that both divertors are detached and in addition both x-points can be strong radiators. However, if this can be achieved, the total radiation capability of the plasma would be significantly increased, which could be important, especially if the radiation capability limitation in JET [12] consolidates. In this context it is also relevant that due to extensive loads on the outer baffle due to x-point radiation and charged particle loads [4] the standard first wall technology may not be applicable in this area.

### 3.4   ELM behaviour

A DN configuration has the advantage to screen loads due to type I edge localized modes (ELMs) efficiently from the inner divertor targets [13]. While this might enable simpler inner target designs, it has been realized that the natural occurring type I ELMs in DEMO are not tolerable due to violation of the divertor surface temperature limit [14]. Hence it is of key importance to demonstrate an ELM mitigation method or a no/small ELM regime, which reduces the ELM loads to an acceptable level.

In MAST RMP ELM suppression can be achieved in SN and DN at similar external perturbation levels [15]. Still of highest importance is the currently investigated question, if coils for efficient ELM mitigation can be implemented in DEMO in a relevant position.

Type II ELMs are an interesting small ELM regime. In ASDEX Upgrade and in MAST at high collisionality $\nu^*$ a transition from large type I to much smaller type II ELMs has been observed when going from SN to DN. For an extrapolation to DEMO the key question is the radial position, at which $\nu^*$ needs to be matched. While there are arguments supporting the collisionality in both the pedestal top region as well as in the separatrix vicinity to be decisive, DEMO has potential to reach a similar $\nu^*$ only close to the separatrix.

### 3.5   Vertical stability

Plasma vertical stability properties are highly relevant for the protection of the plasma facing components. Also, these properties determine the maximum tolerable elongation

of the plasma, to which the performance (i.e. net electric power) of the device has an extreme sensitivity for fixed major radius [3].

A comparison of the vertical stability properties of SN and DN configurations is not simple, as also the machine design has to be optimized to a comparable level. A comparison study starting from a set of system code runs for a number of shapes expressed by $\kappa_{95}$ and $\delta_{95}$ (table I) has been initiated. Based on this, 2D device geometries have been developed. As the DEMO SN baseline design `EU DEMO1 2015` (SN, $\kappa_{95} = 1.59$, $\delta_{95} = 0.33$) is largely based on ITER and hence already optimized in terms of vertical stability, special attention has been given to reducing the distance between toroidal conduction structures and plasma in DN - especially on the LFS.

Table I shows that the flattop stability margins $m_s$ (suggested constraint[3]: $m_s > 0.3$) are significantly higher for SN than for DN with the same shape. An interpretation of this is, that despite geometrical optimisation for the DN, in SN the average distance between plasma and toroidally conducting structures in relevant poloidal regions can be kept lower. For an evaluation of the maximum tolerable elongation this investigation should be repeated for the phase in the pulse,

| Configuration | SN | SN | DN | DN | DN | DN |
|---|---|---|---|---|---|---|
| $\kappa_{95}$ | 1.59 | 1.7 | 1.59 | 1.7 | 1.59 | 1.7 |
| $\delta_{95}$ | 0.33 | 0.33 | 0.33 | 0.33 | 0.45 | 0.45 |
| R [m] | 9.1 | 8.9 | 9.2 | 8.8 | 9 | 8.6 |
| $I_P$ [MA] | 19.6 | 19.2 | 19.8 | 19.3 | 19.5 | 19 |
| $B_T$ [T] | 5.7 | 5.6 | 5.2 | 4.8 | 4.9 | 4.5 |
| $m_s$ at flattop | 1.21 | 0.71 | 0.74 | 0.42 | 0.73 | 0.51 |

TABLE I: PARAMETERS OF THE CONFIGURATIONS OF THE SN-DN VERTICAL STABILITY COMPARISON ANALYSIS: $P_{el,net} = 500MW$, $\tau_{pulse} = 2h$ AND $A = 3.1$

which is most vertical unstable (i.e. during ramp-down).

However, DN has an advantage due to less horizontal-vertical-coupling. In non-linear simulations of ELMs in DEMO ($\Delta\beta_{pol} = -0.1, \Delta li = +0.1$), leading in first place to a horizontal displacement, the vertical displacement of the current centroid immediately after applying the perturbation was $4.8cm$ for SN and only $-0.9cm$ for DN. As the majority of vertical displacements is triggered by horizontal displacements, this means that the trigger rate for VDEs or disruptions would be significantly lower in DN.

## 3.6 Other aspects

- The L-H power threshold[4] $P_{LH}$ in DN is observed in some devices to be lower than in LSN [16]. In LSN in turn it is lower than in USN [17]. The dependence of $P_{LH}$ on $\psi_{N,sep2}$ (tbd) and the amplitude $\Delta\psi_{N,sep2}$ will determine, if this leads to an advantage. Also, recent results showing that divertor configuration optimisation can lead to reduced $P_{LH}$ [18] have to be considered, although it is not clear, if this applies also for closed divertors.
- A comparative investigation employing the EPED1 model [19] revealed that under some circumstances the DEMO pedestal top height in DN can be $> 10\%$ higher than in SN with similar shape.
- Introducing a divertor at the top of the machine could imply stronger W deposition

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[3]It needs to be investigated, if a lower $m_s$ can be accepted for DN.

[4]Ion-$\nabla$B-direction is directed towards the lower divertor.

and dust production in this area. There is concern that more delaminated material or dust falling into the plasma leads to an increased disruption rate.[5]

- A prediction of the tritium breeding ratio (TBR) arrived at TBR=1.12 for a configuration with 2 small divertors and TBR=1.00 with two scaled ITER divertors. TBR needs to be at least 1.05 after the reduction accounting for H&CD systems.
- In DN the control of magnetic configuration and of simultaneous protection of both divertors requires different solutions compared to SN.
- Going from SN to DN implies increasing design complexity, additional constraints for remote handling (to be investigated) and increasing cost of the device.
- The integration of vacuum pumps behind the upper divertor in DEMO seems to be an engineering challenge. At the same time there are concerns regarding the simultaneous control of detachment in case only the lower divertor was pumped.

## 4    Potentials of limiters in DEMO

Next to a change of the magnetic topology, limiters could be an important modification to support power handling and plasma protection. It is assumed here, that these limiters use Cu as structural material and require several remote replacements during the life of DEMO, which need to be relatively uncomplicated and quick, similar e.g. to the ITER port plug maintenance scheme. Three situations, in which dedicated limiters can help, are briefly discussed: (1) Disruptions, (2) static and perturbed situation during flat-top and (3) limited configurations during ramp-up and ramp down

During unmitigated disruptions at full power in DEMO locally the heat impact factor at the first wall can rise above twice the W surface melt limit [4].Despite the fact that DEMO is a point design, which might be optimised in terms of disruptivity of the scenario and redundancy of disruption relevant machine components, a significant number of disruptions during the life time of DEMO need to be assumed. Considering the recent achievements in terms of disruption prediction [20, 21] and the fact that in DEMO these systems have to be trained with the limited set of available diagnostics at a total heating power well below the flattop power, the full avoidance of unmitigated disruptions during the whole life time cannot be assumed. Sacrificial limiters that are designed in a way that they receive the majority of the power released during an unmitigated disruption could help in a way that the remainder of the first wall is not directly affected by such an event. Such limiters would have to protrude with respect to the rest of the first wall.

The static loads to a conformal DEMO wall without limiters have the potential to exceed the technical limits [4]. If this substantiates, limiters might be a tool to deal with charged particle loads, while the first wall could receive most of the radiation loads.

Some initial information on the limited configuration during ramp-up can be found in [4]. Load assessments for this case without and with limiters have started.

As the integration of limiters always means to concentrate the plasma wall interaction to a relatively small wetted area, designed for a higher power handling capability, for all these cases detailed load assessments have to be made. An important requirement related to the introduction of limiters is to keep the TBR high enough.

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[5]The magnetic field can only determine the trajectory of these particles if their size is $\lesssim 0.1\mu m$.

## 5 Summary

As it is not obvious, that design choices relevant for power handling and plasma protection made for ITER are the optimum for DEMO, this publication discusses two alternatives. Switching to a DN magnetic configuration would imply changes in various fields, which are at the moment assessed as follows:[6] Total inter ELM divertor power handling $\uparrow$, $2^{nd}$ x-point radiator $\nearrow$, Type II ELMs $\nearrow$, Pedestal height $\nearrow$, Smaller $\lambda_{q,HFS}$ $\nearrow$, Reduced disruptivity due to VDEs $\nearrow$, Top region dynamic heat fluxes $\nearrow$, L-H threshold $\rightarrow$, Controlability $\rightarrow$, Increased disruptivity due to material from the top entering the plasma $\searrow$, Reduced elongation due to vertical stability $\downarrow$, Design complexity and cost $\downarrow$, Remote handling $\downarrow$, Tritium breeding ratio $\downarrow$, Pumping of upper divertor $\downarrow$. A final conclusion cannot be drawn at the moment, as there is too much missing information that needs to be acquired in future simulations and experiments.

An initial discussion on the potential of limiters to support power handling and plasma protection (during limited configurations, disruptions, static and perturbed flat-top situations) has been presented.

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## References

[1] Romanelli F *et al Fusion Electricity - A roadmap to the realisation of fusion energy* ISBN 978-3-00-040720-8, 2012
[2] Reimerdes H *et al* 2015 *42nd EPS Conf. on Contr. Fusion and Plasma Phys., Lisbon, Portugal*
[3] Wenninger R *et al* 2016 *Nuclear Fusion, accepted*
[4] Wenninger R *et al* 2016 *Nuclear Fusion, submitted*
[5] Martin Y R, Takizuka T, and the ITPA CDBM H-mode Threshold Database Working Group 2008 *Journal of Physics: Conference Series* **123** 012033
[6] Wenninger R *et al* 2015 *Nuclear Fusion* **55** 063003
[7] Temmerman G D *et al* 2010 *Plasma Physics and Controlled Fusion* **52** 095005
[8] Wenninger R *et al* 2014 *Nuclear Fusion* **54** 114003
[9] Eich T *et al* 2007 *Plasma Physics Contr.Fusion* **49** p.573–604
[10] Raffray A and Federici G 1997 *Journal of Nuclear Materials* **244** 85 – 100
[11] LaBombard B *et al* 2016 *PSI conference, Rome, Italy*
[12] Huber A *et al* 2014 *41st EPS Conf. on Contr. Fusion and Plasma Phys., Berlin, Germany*
[13] Petrie T, Watkins J, Lao L, and Snyder P 2003 *Nuclear Fusion* **43** 910
[14] Wenninger R *et al* 2015 *42nd EPS Conf. on Contr. Fusion and Plasma Phys., Lisbon, Portugal*
[15] Kirk A *et al* 2013 *Nuclear Fusion* **53** 043007
[16] Meyer H *et al* 2006 *Nuclear Fusion* **46** 64
[17] Snipes J *et al* 1996 *Nuclear Fusion* **36** 1217
[18] Maggi C *et al* 2014 *Nuclear Fusion* **54** 023007
[19] Snyder *et al* 2009 *Physics of Plasmas* **16** 056118
[20] Vega J *et al* 2014 *Nuclear Fusion* **54** 123001
[21] Moreno R *et al* 2014 *Plasma Physics and Controlled Fusion* **56** 114003

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[6]$\uparrow$ Advantage, $\nearrow$ Potential advantage, $\rightarrow$ Potential advantage or disadvantage, $\searrow$ Potential disadvantage, $\downarrow$ Disadvantage