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The Divertor Tokamak Test facility proposal: Physical requirements and reference design

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ABSTRACT

The main goal of the Divertor Tokamak Test facility (DTT) is to explore alternative power exhaust solutions for DEMO. The principal objective is to mitigate the risk of a difficult extrapolation to fusion reactor of the conventional divertor based on detached conditions under test on ITER. The task includes several issues, as: (*i*) demonstrating a heat exhaust system capable of withstanding the large load of DEMO in case of inadequate radiated power fraction; (*ii*) closing the gaps in the exhaust area that cannot be addressed by present devices; (*iii*) demonstrating how the possible implemented solutions (e.g., advanced divertor configurations or liquid metals) can be integrated in a DEMO device.

In view of these goals, the basic physical DTT parameters have been selected according to the following guidelines: (*i*) edge conditions as close as possible to DEMO in terms of dimensionless parameters; (*ii*) flexibility to test a wide set of divertor concepts and techniques; (*iii*) compatibility with bulk plasma performance; (*iv*) an upper bound of 500 M€ for the investment costs.

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1. Introduction

One of the major challenges identified by the European Fusion Roadmap [1] is the issue to exhaust the huge amount of heat flowing into the divertor region of a fusion reactor. One possibility, which will be tested on ITER [2], is to realize a detached condition with a large radiative volume in front of the divertor. Due to the uncertainties of this approach, a parallel effort has been envisaged to define and design a Divertor Tokamak Test facility (DTT), to test alternative configurations and materials suitable for DEMO reactor. DTT is expected to operate integrating the most relevant physics and technology issues, with significant power loads, flexible divertors, plasma edge, bulk conditions and pulse length relevant for DEMO and closing the gap of present and near term devices

For the DTT design [3] several different approaches have been proposed, either considering the divertor and SOL (Scrape-Off Layer) regions as completely independent of the bulk plasma, or focusing the interest also on the core. To be DEMO relevant, the key parameters characterizing these two regions should be $P_{SEP}/R \geq 15$MW/m and $\geq P_{SEP}B_T/R$ 110 MWT/m, respectively (P_{SEP} is the power flowing through the plasma boundary, R is the major radius, and B_T is the toroidal field). Previous works showed that, a complete "self-similarity scaled down experiment" cannot be realized, even considering the edge plasma as an insulated region, but it could be approximated by matching a number of dimensionless parameters. An optimization process, which also took into account other constraints on cost and flexibility, led to design a device with the following parameters: R = 2.15 m, aspect ratio 3.1, toroidal field $B_T = 6$ T, plasma current $I_p = 6$ MA and an additional power $P_{add} = 45$ MW, obtained with a combination of ICRH, ECRH and NBI [3]. Particular care was directed to design a divertor able to accommodate different divertor geometries and materials to be tested, in particular targets with a liquid metal like Li and Sn. A set of small internal coils will allow realizing and studying, in addition to single and double null standard configurations, advanced magnetic divertor topologies (at fixed plasma shape), such as snowflake

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and X-divertor, in reactor relevant regimes. It will be possible then to test different divertor materials (tungsten or liquid metals) to a power flow up to 20 MW/m^2, under operational conditions with bulk and edge parameters relevant for DEMO.

2. Basic DTT scaling

In order of priority, the main objectives of the DTT device are:

1) to demonstrate a safe and robust power handling solution to be extrapolated to DEMO;
2) to maintain plasma core and pedestal performances in a plasma regime as close as possible to that of a reactor;
3) to achieve the two previous points by integrating all physics and technological aspects.

Integration is mandatory for any experiment meaningful to DEMO scenario, but it is well recognized that, to simulate all the aspects and the complete behavior of DEMO, the trivial solution should be to realize DEMO itself. To pursue this challenging issue, different approaches have been proposed [4–7], either considering the divertor and the SOL as regions completely independent of the bulk plasma, or focusing the interest also on the core. In any case, a prioritization among the different parameters has to be defined, trying to include all the different aspects, compatibly with available technology and economical resources.

Since the main objective of the DTT facility is the study of the power exhaust issues, the basic concerns are connected with the divertor and the SOL. A key parameter that characterizes these two regions is the ratio P_{SEP}/R. To be relevant for DEMO, the value of this parameter should be 15 MW/m or larger. This high value suggests the use of an ITER-like actively cooled tungsten monoblocks technology, being tungsten a material that can likely comply with the DEMO nuclear constraints. Other two important parameters are the upstream poloidal (q_θ) and parallel ($q_{//}$) power fluxes; the first parameter is expressed as $q_\theta = P_{SEP}/\lambda_q 2\pi R$, where $\lambda_q \propto B_\theta^{-1}$ is the decay length of the mid-plane heat channel and the inverse dependence on the poloidal field B_θ^{-1} comes from Eich's scaling [8,9]. Since the parallel heat transport is dominant, it follows that the latter parameter $q_{//}$ is $\approx q_\theta B_T/B_\theta \propto P_{SEP}B_T/R$ (>110 MWT/m for DEMO), where B_T is the toroidal field. Previous work [6,10] has shown that, even considering the edge plasma as an insulated region, a complete "self- similarity scaled down experiment" cannot be achieved, but that it could be approximated [5,6] by fitting five dimensionless parameters: T_e (with a suitable normalization), $\upsilon^* = L_d/\lambda_{ei}$, Δ_d/λ_0, ρ_i/Δ_d, β, where T_e is the electron temperature, L_d is the divertor field line length, λ_{ei} is the electron-ion collisional mean free path, Δ_d is the SOL thickness, λ_0 is the neutrals mean free path, ρ_i is the ion Larmor radius, β is the plasma pressure normalized to the magnetic one. Some of these parameters are intrinsically linked with the divertor "magnetic topology" and/or with the actual divertor geometry [11]. Therefore, a first strong constraint arises for the DTT design: the necessity of having a very flexible divertor "region/configuration" to study and optimize the role played by the various topologically linked parameters.

The machine dimensions and the plasma bulk performances should guarantee an exhaust solution extrapolating to reactor-graded plasma. It is well known that the plasma physics properties (bulk and edge) are completely determined by the dimensionless parameters υ^* (normalized collisionality), ρ^* (normalized Larmor radius), β and T_e [10,12]. However, it is not possible to simultaneously fix all these quantities. A strategy, which consists in releasing one of these parameters in a controlled way [13], has been proposed to down-scale the main physical properties of a reactor-like experiment (i.e. ITER, DEMO) on a smaller device. Since $\rho^* \propto T^{0.5}/B_T R$, it is practically impossible to preserve this parameter without using machine and plasma parameters with the present achievable values of magnetic fields (at a constant value of ρ^*, B_T would scale with 1/R). Consequently ρ^* is the dimensionless parameter that should be relaxed: $\rho^*_s = \rho^*_r \cdot (R_r/R_s)^\varepsilon$; the subscripts r and s indicate respectively the "reactor" and the "scaled" device, ε is the "controlling" scaling parameter. This choice allows preserving the main physics properties in a scaled experiment dedicated to study the Power Exhaust, i.e., a very flexible divertor region, a meaningful P_{SEP}/R ($\geq$15MW/m) and $P_{SEP}B_T/R$ ($\geq$110MWT/m), and a set of dimensionless parameters as close as possible to DEMO. The typical range for the selection of ε is between 0 and 1: $\varepsilon=0$ would yield preservation of ρ^* (but this would imply $R_s=R_r$), whereas $\varepsilon=1$ would keep the product ρ^*R constant.

The machine dimension also depends on another important constraint, i.e. the cost containment. The cost of a tokamak (without using tritium and not including the additional power) scales as the total energy stored in the toroidal magnetic field. It is possible to show that relaxing ρ^* with the value of the scaling parameter $\varepsilon=0.75$, the cost is nearly proportional to $R^{2.75}$. We assume the cost of the additional heating to be about one third of the total, so the heating cost is about 150M€, since 500M€ is the estimated total costs foreseen for this proposal of DTT. Fig. 1a shows the value of P/R versus R, as obtained by using the mentioned weak scaling, for three different values of the controlling parameter ε. It is immediately clear that if increasing ε above a certain limit, there would be an intrinsic difficulty in achieving a reactor relevant P/R value. On the other hand, for small ε values, the additional heating request would strongly increase. Consequently, an intermediate value of $\varepsilon=0.75$ seems to be appropriate. With this scaling, a rough estimation of the machine cost (not including the heating) can be evaluated as a function of the machine major radius. This feature is shown in Fig. 1b: to stay within a limit of about 350M€, the maximum machine radius is limited as $R_{Max} \leq 2.3$ m. This evaluation provides only a rough estimation and the actual machine cost has been verified by a more accurate analysis based on the design of the various machine components.

When fixing the machine dimension, the P_{SEP}/R criterion determines the minimum necessary additional power and the compatibility with the allocated budget. A meaningful minimum machine radius is not obtainable only by the physics scaling but a useful indication comes from the constraint of having a flexible divertor region and actively cooled plasma facing components. This flexibility will be used (along the time machine life) to "easily" change different divertors, designed to best fit the different magnetic topologies and/or test different materials (tungsten, cooling pipes in copper alloys, liquid metals …). This flexibility will also give the possibility to test different First Wall (FW) materials, and technologies in reactor relevant regimes from both the points of view of plasma bulk performances and power flow. Therefore, the following three different arguments can be used to address the minimum DTT size.

1) To study the physics of quite different divertor magnetic topologies, a small set of internal coils should be introduced to modify the reciprocal position of the main null point and of a secondary poloidal field null. Since the grazing angle at the divertor target is in the order of 1 deg, the local field to be modulated is of the order of a few percent of the toroidal field, i.e. a few tens of mT. Assuming a typical current density value (30÷50MA/m^2) the minimal dimension of each of these internal coils (including the mechanical support) is of the order of 10×10 cm, which implies, for the set of four coils, a radial extension of about 40 cm. With an aspect ratio of 3 a first rough indication suggests $R_{min} > 1.5$ m.
2) In Fig. 1c the injected additional power, normalized to the plasma toroidal surface, is plotted versus the machine radius, assuming P_{SEP}/R=const.= 15MW/m. For comparison, it is shown the power flux to the FW for ASDEX Upgrade (Table 1), together

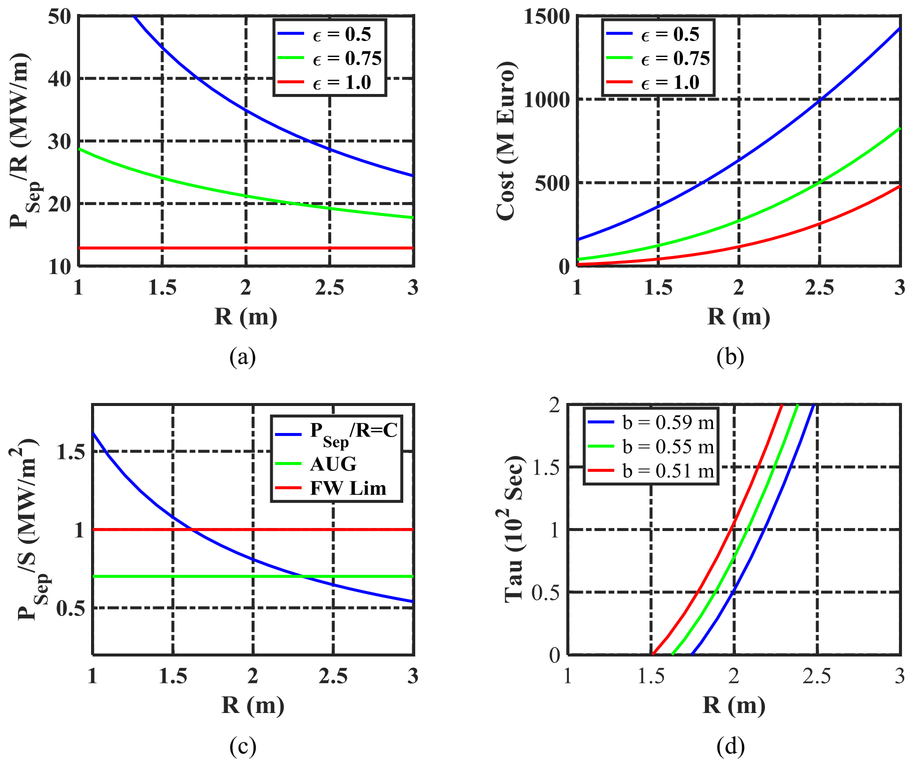

Fig. 1. Dependence various parameters on the major radius R: a) P/R scaling versus R for three different ε parameters: for $\varepsilon = 1$ the P/R value is always too small; b) Load Assembling device cost; for $\varepsilon = 0.5$ the cost is always too high; c) power density versus the machine radius, assuming P/R= 15MW/m; d) Discharge duration, versus the machine radius, at different central solenoid width.

Table 1
Main DTT parameters and comparison with other machines (some figures might be different for other devices in high performance scenarios).

	JET	AUG	EAST	DIII-D	ITER	DEMO	JT-60SA	WEST	TCV	ADX	DTT
R (m)	2.98	1.65	1.7	1.67	6.2	8.77	3.0	2.5	0.88	0.73	2.15
a(m)	0.94	0.5	0.4	0.67	2.0	2.83	1.2	0.5	0.24	0.2	0.70
I_p (MA)	3.5	1.6	1.4	2.0	15	20	5.5	1	0.45	1.5	6.0
B_T (T)	3.2	2.4	3.4	2.1	5.3	5.8	2.3	3.7	1.45	6.5	6.0
V_p (m^3)	82	13	10	19	853	2218	141	15	1.85	0.9	33
<n> ($10^{20} m^{-3}$)	0.9	0.9	1.0	0.85	1.0	0.9	0.9	0.8	1.2	4.5	1.72
<n>/n_G	0.7	0.5	0.4	0.65	0.85	1.1	0.8	0.7	0.5	0.4	0.45
P_{Tot} (MW)	30	25	30	27	120	450	41	16	4.5	14	45
τ_E (s) ($H_{98} = 1$)	0.49	0.07	0.07	0.11	3.6	3.4	0.62	0.05	0.027	0.05	0.47
<T> (KeV)	3.3	2.5	3.3	2.8	8.5	12.6	3.4	2	0.8	1.7	6.2
β_N	1.8	2.4	2.2	2.9	1.6	2.1	2.4	2	2.7	2.2	1.5
υ^* (10^{-2})	8.6	8.4	7.4	4.0	2.3	1.3	4.1	35	65	13.1	2.4
ρ^* (10^{-3})	4.0	8.5	8.5	7.2	2.0	1.6	4.5	5.0	17	7.7	3.7
T_{Ped} (KeV)	1.7	1.3	1.7	1.4	4.3	7.0	1.7	0.5	400	1.3	3.1
n_{Ped} ($10^{20} m^{-3}$)	0.7	0.7	0.9	0.7	0.8	0.7	0.7	0.5	0.9	3.8	1.4
υ^*_{Ped} (10^{-2})	22.6	22	21	10	6.2	2.8	11	92	170	35	6.3
ELMs En. (MJ)	0.45	0.06	0.07	0.13	24	140	1.1	0.2	0.03	0.02	1.2
L-H Pow. (MW)	9.5÷12	3÷4	3.5÷4.5	3.0÷4.0	60÷100	120÷200	10÷12	4÷6	0.6÷0.8	4÷6	16÷22
P_{Sep}/R (MW/m)	7	11	12	11	14	17	9.5	4	3.4	13	15
λ_{int} (mm)	3.2	3.7	2.6	3.6	2.2	2.2	3.7	3	5.5	1.7	1.7
P_{Div} (MW/m^2) (no Rad)	28	44	62	45	55	84	24	25	7.3	110	54
P_{Div} (MW/m^2) (70% Rad)	8.6	13	19	13	27	42	7.4	7.5	2.2	33	27
q// ≈ PTotB/R (MW T/m)	32	44	60	40	100	290	22	23	5	125	125
Pulse Length (s)	≈ 20	≈ 6	??	≈ 6	400	7000	100	1000	5	3	100

with a safe power flux (≈ 1MW/m^2) for a tungsten FW. It appears that reducing machine size below R< 1.5 m, the power density flux increases above the material safety limit. Even considering that approximately only 50% of the power interacts with the FW, a peaking factor of around 2÷3 should also be taken into account, therefore the assumed power flux can be considered still valid.

3) The third point regards the discharge duration time τ_S. The resistive diffusion time τ_R in the proposed DTT is about 6 s (Table 1), leading to a pulse length of at least 20 seconds to reach stationary conditions for the plasma. This must be considered as the "zero" time to study any material thermalization time. Consequently, a plasma current plateau at least twice must be considered; integrating in the plasma duration the rump-up and rump-down phases, a plasma pulse length $\tau_S \approx$ 100 s must be assumed. Fig. 1d shows the discharge duration as a function of the plasma major radius, by using a standard scaling [14] at fixed values of B_T, q_{95} and aspect ratio. The pa-

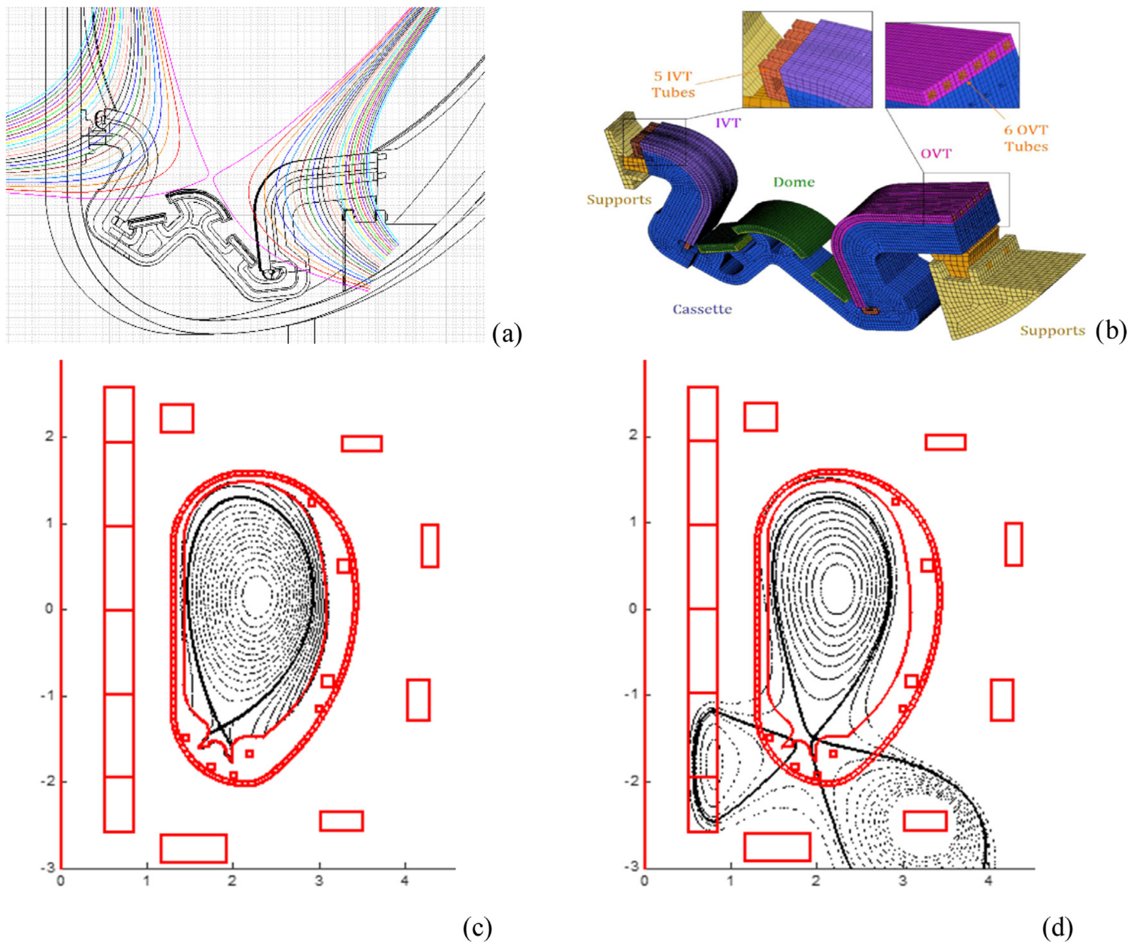

Fig. 2. A possible divertor for DTT compatible with both SN and QSF plasmas: a) cross section; b) 3D view; c) SN configuration; d) SF plasma.

rameter b is the distance between the inner edge plasma radius and the outer radius of the central solenoid. For a given toroidal field and aspect ratio, this distance is roughly fixed. In the case of copper coils, a reduction of b would lead to increase the current density in the toroidal magnet, up to a level where the discharge duration is determined by the magnet coil heating. In our case, with $B_T \approx 6$T and current density around 70MA/m^2, the total distance b would be about 50 cm and the magnet heating would yield $\tau_S \approx 60 \div 70$ s. By using superconductors, the averaged current density should be smaller and the coils must be shielded against the neutron flux. Therefore, also in this case we get b $\approx$ 50÷60 cm leading to $R_{min} > 1.8$ m.

3. Reference DTT configurations

After fixing the main machine parameters, it remains to verify whether the important figure $P_{SEP}/R \geq 15$ MW/m is satisfied together with the possibility to allocate quite different divertor geometries. This volume check cannot be completed in a rigorous way, because the variety of divertors to be tested in the future is not fully known. In Section 4 we show how to achieve some different magnetic topologies with various divertor geometries. Here, to complete the description of the used procedure to fix the machine parameters, we just show the possibility to include a very large divertor capable to cope with standard X point as well as with quasi-Snow Flakes (QSF) [15] configurations. This is shown in Fig. 2, where a QSF plasma impinges on a divertor open on the top part and close on low part. Of course, any “dedicated” divertor geometry would imply a smaller volume.

The estimation of the inter ELMs maximum power load on the DTT outer divertor target (usually the most loaded one), expected in a SN configuration, attached regime, can be done by using the following expression [7,8]:

$$q(s^*) = \frac{q_0}{2}\left(\left(\frac{s}{2\lambda_q}\right)^2 - \frac{s^*}{F\lambda_q}\right) \bullet erfc\left(\frac{s}{2\lambda_q} - \frac{s^*}{F\lambda_q}\right) \qquad (1)$$

where q_0 is the peak heat flux density at the divertor entrance, λ_q is the power e-folding length at the outer midplane and s the width of the Gaussian, convoluted with the exponential profile, taking into account the diffusion in the Private Flux Region (PFR) of the power at the entrance of the divertor, while travelling along the divertor leg. In addition, F is the flux expansion factor calculated at the target, s* is the coordinate along the target surface in the poloidal cross section, with s*=0 at the outer strike point. The power e-folding length at the outer midplane can be evaluated by using the empirical scaling [7,8] $\lambda_q \sim 0.73\ B_T^{-0.8}\ q_{cyl}^{1.2}\ P_{SOL}^{0.1}$, in which $q_{cyl}=(2\pi a^2 B_T(1+k^2))/(2\mu_0 R I_p)$, where a is the minor radius, k the elongation, and μ_0 the magnetic permeability of the vacuums. The width s depends on local plasma parameters and the divertor geometry. Obviously, the final design of the DTT divertor(s) is not yet fixed, but as a first approximation we can use for s the scaling found for the divertor of ASDEX Upgrade with tungsten PFCs [16], the value of s, expressed in mm, is s $\sim 0.09\ n_{e,ped}[10^{19}\ m^{-3}]/B_{pol}$ [T], with $B_{pol}= (\mu_0 I_p/2\pi a)\ ((1+k^2)/2)^{-0.5}$.

The use of (1) in the calculation of the target power leads to a decrease of the peak load (slightly shifted from the position of the outer strike point) and to the definition of the integral power de-

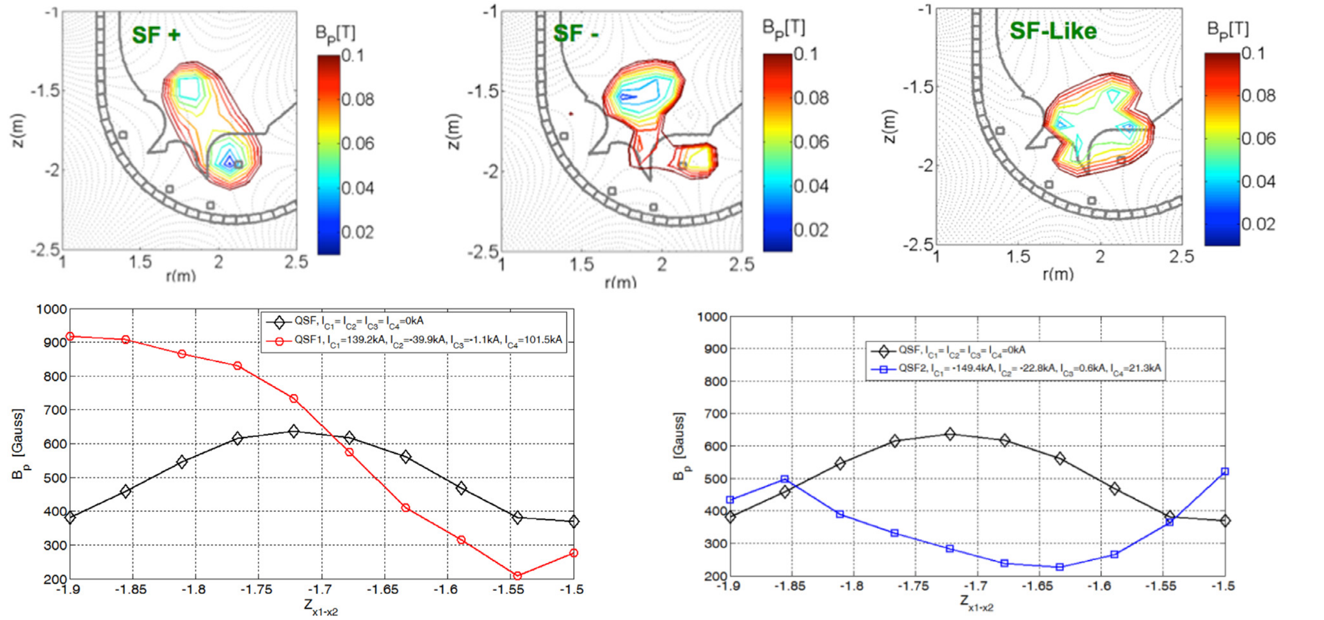

Fig. 3. Top left: QSF (SF$^+$) 5 MA plasma equilibrium obtained by the external coils; top center: the "hill" like field reference configuration has been varied to a monotone slope like field configuration (left bottom); top right: the "hill" like field reference configuration has been varied to a "mirrored" field configuration (right bottom). The abscissa in the bottom figures is the vertical coordinate along the line connecting the active X-point to the external poloidal field null.

cay length $\lambda_{int} \approx \lambda_q + 1.64$ S [17]. The rough effect by approximating the heat load profile on the divertor target with a new exponential profile using λ_{int} instead of λ_q, is to decrease the peak heat load by the factor λ_{int} /λ_q. An average specific power on the outer target (within the first power decay length) of about 28 MW/m^2 can be achieved if using the set of parameters reported in the Table 1: a SN poloidal flux expansion of ~ 4, a conservative ratio between outer and inner divertor loads of 2:1, an incidence angle the field lines on the divertor target surface of 3° (including a tilting angle $\theta = 70°$ of the target in the poloidal plane) and without impurity seeding; we get. This means that the main figure $P_{SEP}/R \geq 15$MW/m is easily achieved, even including the possibility to perform experiments varying the radiation in the different plasma zones (bulk, SOL and divertor region).

The H-mode threshold condition of 16–22 MW at full current is reported in Table 1 and it corresponds to the different P_{LH} scaling laws presently available. By using the most robust and used scaling, we get $P_{LH} = 0.048\ N_e^{0.717}\ B_T^{0.803}\ S^{0.941} = 14.5$MW. Therefore, an initial additional power coupled to the plasma of 25 MW (=>$P_{SEP} \approx 1.25 P_{LH}$) should guarantee to achieve robust H-mode since the very initial operations.

4. Alternative configurations

A strong effort has been made to realize a "divertor" region flexible enough to easily allocate different divertors, with different geometries and materials (Fig. 2), from actively cooled tungsten monoblock up to liquid metal cassette divertor [18].

The external poloidal coil system has been deigned in order to be able to produce the widest possible spectrum of different "alternative" magnetic divertor topologies, including standard SN plasma with the machine target plasma current I_p=6MA; Snow Flake (SF) equilibrium with $I_p = 4$MA, the lower current being constrained by the maximum current density in the central solenoid (CS) and the discharge duration of 100 s; when relaxing the flat top duration the configuration can again be realized with $I_p \approx$ 6MA; QSF configuration and a double null plasma with I_p=5MA. All the equilibria have been studied at the same β_p and the same internal plasma inductance l_i satisfying the following constraints:

a) minimum distance of 0.04 m between the plasma last closed surface and the first wall (the power decay length at 6MA is ~2 mm [7,8]);
b) maximum current density in the poloidal field coils around 25–30 MA/m^2, and maximum field on any PF coils less than 5T and less than 12.5 T for the CS coils;
c) same geometrical plasma features: R = 2.15 m, a = 0.69 m, k≈1.76, <δ>≈0.35.

For feedback control we reserve about 10% capability of outboard PF coils and 5% of PF1, PF6, and CS coils. Within the aforementioned constraints, sufficient flexibility is maintained to allow quite different plasma shapes. Indeed, the PF system should be capable to modify the magnetic configurations in the vicinity of the divertor targets for experiments aimed at exploring the effects of various parameters (e.g., connection length, grazing angle, scrape-off volume, flaring/converging behavior of the plasma channel) on the power exhaust quantities. In the divertor region enough space has been allocated not only to substantially vary the plasma divertor magnetic topology, but also to allow strike point sweeping and to have an efficient pumping capability.

Finally, the feedback control system will be capable to:

- stabilize the vertical position
- keep the shape at steady state within ~ 2 cm from its reference in case of current density profile changes (within $\Delta\beta$ ~ ±0.2 plus Δl_i ~ ±0.1)
- for diverted configuration keep the plasma-wall clearance at least ~ 4 cm at steady state and ~ 1 cm during transients of about ~ 1 s.

The presence of a set of small internal coils around the divertor will allow to locally modify the magnetic topology, provided that a second null is already obtained by the external poloidal coils, without affecting the rest of the plasma boundary. This will allow to perform detailed studies about the role of the divertor magnetic topology in reducing the power flow on the divertor plates, either affecting the local energy transport properties and/or the local radiation. An example of such a possibility is shown in Figs. 3-4. Here the initial configuration is a QSF, obtained by using only the exter-

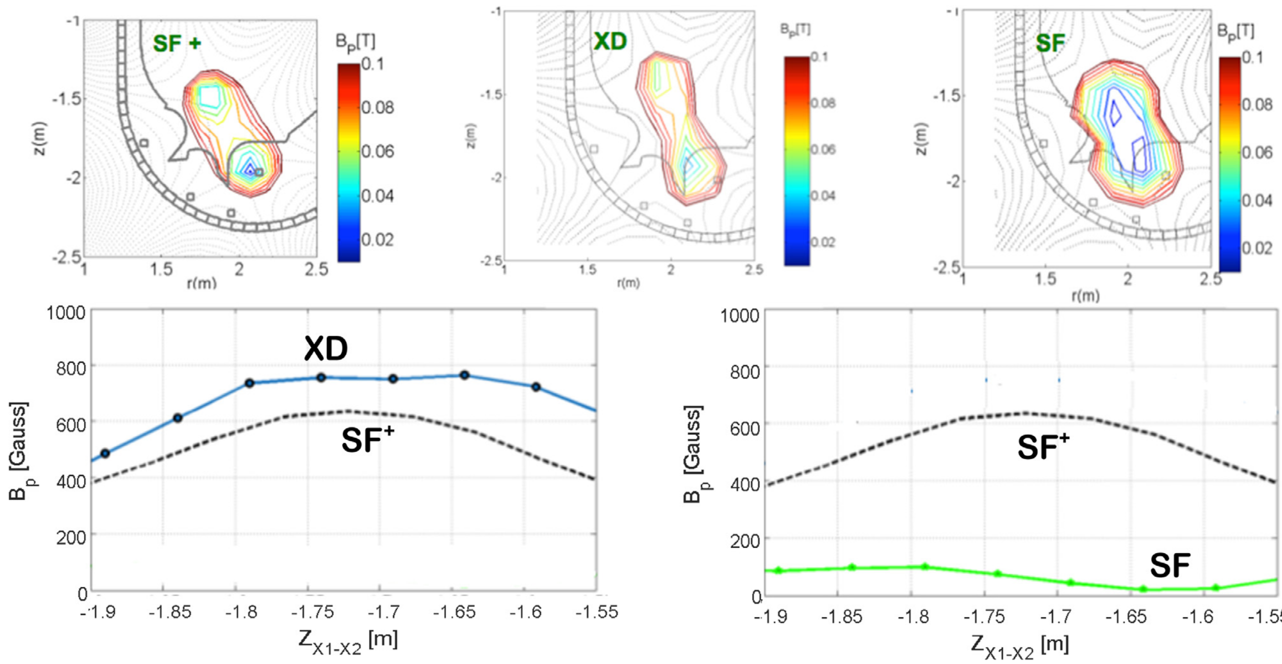

Fig. 4. Top left: QSF (SF$^+$) 5 MA plasma equilibrium obtained by the external coils; top center: using the internal coils, the slope of "hill" like field reference configuration has been varied close to the second null (left bottom); top right: using the internal coils, the "hill" like field reference configuration has been varied to a very flat region (right bottom). The abscissa in the bottom figures is the vertical coordinate along the line connecting the active X-point to the external poloidal field null.

nal poloidal coils. The internal coils allow a great variation of the local magnetic topology (SF$^+$, SF$^-$, pure SF, XD,...) to explore its importance in the power exhaust problem. Further details on the technical features of the proposed DTT machine can be found in [3,19,20].

5. Conclusions

This paper has illustrated and motivated the physical requirements and the reference parameters considered by the Italian proposal for a Divertor Tokamak Test facility [3]. The main aim of the DTT facility is to find out the optimal solution for the power exhaust problem supporting in a unique way the R&D activities required to approach the DEMO fusion power plant design. Since, presently, there is not a definite solution for DEMO that could to be tested in DTT, this facility, along its life, should be able to study as much as possible the different concepts presently under consideration, as well as other possible new ideas. This fact leads to the strong constraint to realize a very flexible machine. The DTT electron density, even lower of the Greenwald limit, is high enough (in DEMO relevant regimes with low collisionality) to sustain radiative scenarios with and without impurity seeding with a FW realized by using reactor relevant materials. The presence of internal coils will allow to test a wide set of different divertor magnetic configurations as well as strike point sweeping and plasma wobbling. Large ports and the wide space available in the bottom and the top of the machine will permit to change the divertor easily, testing different divertor geometries (including an up-down symmetric divertor) and materials (including liquid metals). The different heating schemes are planned to be tested under DEMO relevant conditions. The closeness of the plasma bulk parameters to the DEMO/ITER ones will guarantee that any solution for the power exhaust found and tested on DTT will work in DEMO without degrading the plasma performances.

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