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Summary

Power Exhaust is the most challenging gap, between the open Physics and Scenario problems, to be solved before starting the construction of a demonstrative Fusion Power Plant, DEMO. Even under construction International experiment ITER will not be able to satisfactorily tackle the issue. For both ITER and DEMO accompanying programs it should be strongly envisaged an experiment, where all the integrated problems of the power exhaust in burning plasmas are studied and possibly solved. Up to now the no experimental solution exists. In the present machines (AUG. JET,…) highly radiative (up to 80%) scenarios are presently investigated; however they clearly show the intrinsic difficulty (impossibility ?)) to guarantee the good energy and transport properties, necessary for an high gain (Q>20) Reactor, and the total plasma radiation (>90%) necessary to maintain the Power Flow on the divertor tiles within the presently available materials characteristics (≈10MW/m^2n stationary). FAST is a machine proposal designed to study, as main target, all the integrated problems connected with power exhaust. Among the present ongoing experiments EAST is the best one suited to study the Power Exhaust problem in integrated edge-bulk scenario. The actively cooled divertor in Tungsten and the large Power Flux achievable on the tails (up to more than 20MW/m^2) makes EAST and in future FAST the most worldwide relevant experiments to tackle and possibly solve the really challenging Power Exhaust problem in view of ITER and DEMO. The possibility of using, in a future DEMO, highly radiative scenarios is quantitatively analysed, and integration between these scenarios and new magnetic configurations is proposed.

1 Introduction

One of the most challenging gaps in view of DEMO is the power exhaust problem [1, 2]. By using the present available knowledge, and assuming a power plant of around 5 GW thermal, a fractional power around 1 GW must be safely exhausted, without affecting the good bulk plasma quality. Even assuming 80% of radiation, it is clearly impossible to guarantee a power flux to the divertor plates within the present

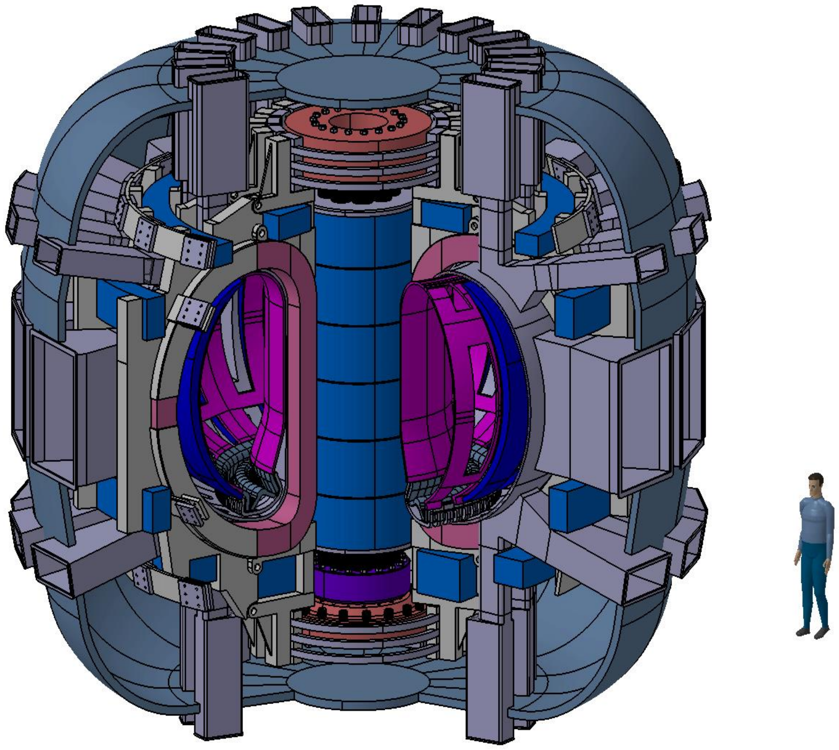

Fig. 1 - FAST schematic view

technological possibilities (<< 20MWm^{-2}). As it is clear from this short summary, the problem is quite serious, also because what it is actually missing is not only a single aspect, but the full integration of very different physics and technical problems in a robust and reliable plasma scenario. FAST [3] (Fig.1, Tab1, here BT is the toroidal field, V_p is the plasma volume, <n> is the volume averaged plasma electron density, H&CD is the total additional power, P/R is the additional power divided the plasma major radius and Q is the Fusion Gain), from the very beginning, has been conceived with the main aim to tackle this problem integrating all the plasma wall interaction aspects. The power density stored within the machine (~1.5MWm-3) and the choice of working in a full W environment (First Wall and divertor) makes, clearly and immediately, the complete FAST relevance to the power exhaust problem in DEMO. FAST will have the unique capability of working at always relatively high density (ne≥10^{20}m^{-3}), although ranging the extremes of the Greenwald limit. This fact will allow fully exploiting the possibility of varying the radiation fraction (between 30% and 80%, by using some slight impurity seeding) in the different plasma regions (bulk, SOL, divertor) trying, at the same time, to maintain very good plasma properties [4]. However, as just previously mentioned, even these specific capabilities could not be sufficient to limit the power deposited on the divertor plates. Since a few years, a "snowflake" (SF) magnetic topology has been suggested, for the divertor region, capable to spread the power flow onto a much wider area [5]. Moreover, recent experimental results have confirmed the possibility of strongly reducing the power flowing to the divertor monoblock taking advantage of a SF geometry [6,7]. Within its reference scenario (Ip=6.5 MA) and by using the normal external poloidal coils, FAST will have the possibility to easily change the magnetic topology

Table 1 – FAST main parameters

Plasma Current (MA)	≤ 8 (*10*)
B_T (T)	≤ 8.5
Major Radius (m)	1.82
Minor Radius (m)	0.64
Elongation k_{95}	1.7
Triangularity δ_{95}	0.4
Safety Factor q_{95}	~ 3 (*2.3*)
V_p (m^3)	23
<n>(m^{-3})	≤ 5.5×10^{20}
Flat-top B_T (s)	15 -> 170
H&CD power (MW)	40
P/R (MW/m)	22
Q	~ 1.5 (*3*)

between the Standard X (SN) point and the SF one, with a "produced" power flow to the divertor plates (with the SN magnetic topology) ranging between 10 and 40 MWm^{-2} [8]. Recent technical developments, in the FAST design, have addressed the possibility of easily replacing, by remote handling, the divertor cassette [9]. This will allow a straightforward comparison between the power loads on the plates, by using standard optimized SN divertor structure, and the power loads by using a fully different divertor geometry, optimized for the SF configuration. Among the existing worldwide experiments, the Chinese EAST [10] Tokamak has practically all the technical facilities that are foreseen on FAST. The key difference will be the quite larger toroidal field (8T against 2T), consequently FAST will be able to guarantee Fusion performances and adimensional plasma parameters (β_p=normalized plasma pressure, ρ*=normalized Larmor radius, ν*=normalised plasma collisionality) closer to the one present on a DEMO machine. However the actively cooled divertor, using the ITER technology of monoblock tungsten, the long plasma impulse due the superconductive coils, the very large planned (up to ≈30MW) additional power, make EAST the best present experiment to advance some of Physics that FAST could/should explore. What it will miss, in view of DEMO will be the complete integration between the possible solution of the Power Exhaust problem and the quality of the plasma confinement in regimes with a volume power density comparable with the one of a power plant.

2 Description of activities and results

2.1 *Power Exhaust: an Integrated Edge-Bulk Problem.*

Assuming DEMO performances ranging between a "relaxed" realistic DEMO-R [11] and a possible Power Plant [1], the total thermal power (P_{Th}) will range between 1.5 ÷ 3.5 GW, with a corresponding alpha power around 0.3 ÷ 0.7 GW. In Table 2 the typical figures, used to characterize the load on the divertor plates (in Fig. 2 it is show a schematic view of the top EAST tungsten divertor), are compared for the Power Plant, the relaxed DEMO-R (assuming for both of them a major radius of 9 m), for ITER, for FAST and for the presently under-construction JT60-SA satellite. P_H indicates the total plasma heating power (external plus self heating), where a gain Q=20 has been assumed for DEMO-R and the Power Plant, while V is the plasma volume. The total power density (P_{Th}/V) characterizes the load on the First Wall (FW) and, in some sense, gives an estimate of the total power load under which the good quality of the plasma bulk must be guaranteed. The figure P_H/R characterizes the load on the divertor plates, assuming the same midplane energy decay length (λ_E), the same flux expansion (F_{EX}), and the same flux lines-divertor plates geometry (sin θ). $A=2\pi RF_{EX}\lambda_E/\sin\theta$ is the equivalent divertor surface, consequently P_{Th}/A is the divertor plates load, when considering no radiation at all and a sharing 1/3, 2/3 between the inner and the outer divertor plates. For an evaluation of A it has been assumed F_{EX}=4, λ_E=0.5cm and θ=20° degrees (typical experimental values for a standard X point configuration). Considering that, with the present available materials and technology, the maximum power load on the divertor plates must be $P_{max} \approx 20MWm^{-2}$, it is evident that even for a machine like the relaxed DEMO-R a fraction around 75% of the total heating power should be radiated, without affecting the plasma performances, in between the bulk plasma, the Scrape Off Layer (SOL) and the private divertor region. The serious problems connected with achieving this challenging task have been already discussed in several papers [2,12]. Here we want to stress the clear point of interconnection of several different aspects, when trying to tackle the problem. In a reactor, the divertor plate material must be able to sustain the 20MW/m^2 power flux and 14Mev neutron flux for a period of several years. So far the only available material seems to be Tungsten. To minimize the sputtering on W we have to keep the plasma temperature in front of it as low as possible (<< 100eV), but at the same time we need a large plasma pedestal temperature. As last, to achieve

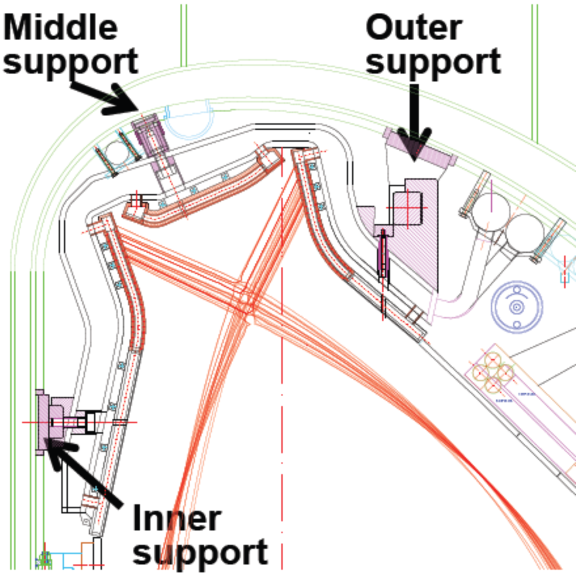

Fig. 2 Schematic view of the new W EAST top divertor

	Power Plant	DEMO-R	ITER	FAST	JT60-SA
R (m)	9	9	6	1.82	3
P_{Th} (MW)	3500	1500	400	40	40
P_H (MW)	975	375	120	40	40
P_{Th}/V (MW/m^3)	1.75	0.75	0.5	1.7	0.31
P_H/R (MW/m)	110	42	20	22	13
P_H/A (MW/m^2)	195	75	36	40	23

Tab. 2 Comparison of Power Load figures (assuming no radiation losses) among the Power Plant, DEMO-R, ITER FAST, and JT60-SA

a radiation fraction ~ 80% we would need some light impurity seeding, again minimizing the interaction with the divertor plates and without affecting the confinement quality. So far no one of the present experiments can even address all these aspects in an integrated way. This fact raises the question of what it is necessary to seriously tackle all these very interconnected and not separable problems, in a single experiment. Clearly we need an experiment with P_H/R and P_{Th}/V not far from reactor relevant values. Moreover, the experiment must have the possibility to allow a large variation of the radiation fraction, while maintaining a high plasma density to minimize the sputtering problems. The SOL atomic physics (that essentially means temperature and collisionality) must be as similar as possible to the reactor one. The divertor should be easily replaceable, to allow testing different materials (for instance liquid Li [13]) and/or different divertor magnetic topologies. Most important, we would need to show that the reactor bulk plasma (characterized by the dimensionless parameters ν *, ρ * and β [14,15]) is not badly affected by the different options. This means that we need a bulk plasma with dimensionless parameters as close as possible to the reactor ones. FAST [3,16] specifically has been designed to accomplish (within an integrated scenario) three different tasks: a) Plasma Wall interaction issues (power exhaust, W divertor, FW, materials…); b) Plasma Operations (ELMs, plasma controls, heating coupling…); c) Burning Plasma Problems (fast particles driven instabilities…). The FAST bulk plasma properties and divertor flexibility have been already discussed in several papers [3,4,17], here we will focus on the complementarity between FAST and EAST, analyzing the possibility of studying, during a single discharge, situation where the a strong radiation is used combined with different magnetic topologies in the divertor regions and analysing the possibility to control the local radiation by the magnetic topology.

2.2 Power Exhaust: "Looking at the radiation".

Within the present technology capabilities the very maximum power flow tolerable by the divertor plates is, for a transient load, ≈ 20MW/m^2 by using actively cooled W monoblocks [18] (in Fig 3 it is show a mock

Fig. 3 Actively cooled tungsten mononblocks mock up

up of the this monoblock technology). But it is not even imaginable to use this "extreme" figure for a steady state regime (or even very long pulse and high repetition rate) in a machine like DEMO or a Reactor. Under this condition we must assume a more safe figure, with a maximum power load < 15MW/m^2. By using these figures and assuming that only the plasma radiation contributes to reduce the power flow on the divertor plates, we can evaluate what should be the total radiation losses for different future reactors or DEMO. Assuming to have a reactor putting ≈ 1GW on the grid (≈3GW thermal), or for a planned DEMO power plant (≈ 0.5GW on the grid and ≈1.5GW thermal), we immediately see that, for machine with major plasma radius of 9 m, it must be radiated a power P_{RAD}≈ 84% (DEMO) ÷ 93% MWm-2 (REACTOR) and for a machine with R=7.5 m P_{RAD}≈ 87% (DEMO) ÷ 94% MWm-2 (REACTOR). AS we note, even for a very large (expensive) DEMO the minimum radiation loss should be, at least, around 84% and we have to wonder if this very large figure is compatible with the necessary good energy and transport particle confinement properties, necessary to guarantee the plasma performances. The best experiments so far realized on an experimental device have been realized on AUG [19]. In Fig. 4 we show the best achieved AUG results. P_{Heat} is the total heating power, P_{rad_tot} is the total radiated power, P_{rad_main} is the fraction radiated in the plasma bulk, P_{rad_div} is the power fraction radiated in the divertor volume; BetaN is the normalised plasma pressure, n_e is the electron plasma density, H98 is a figure of merit describing the plasma confinement, for a reactor it must be H98≥1. Looking at these results it seems that we could have a solution, at least for the large DEMO machine; however looking better at the results the situation is much different. First, when trying to reply these results on larger machine like JET, with higher absolute

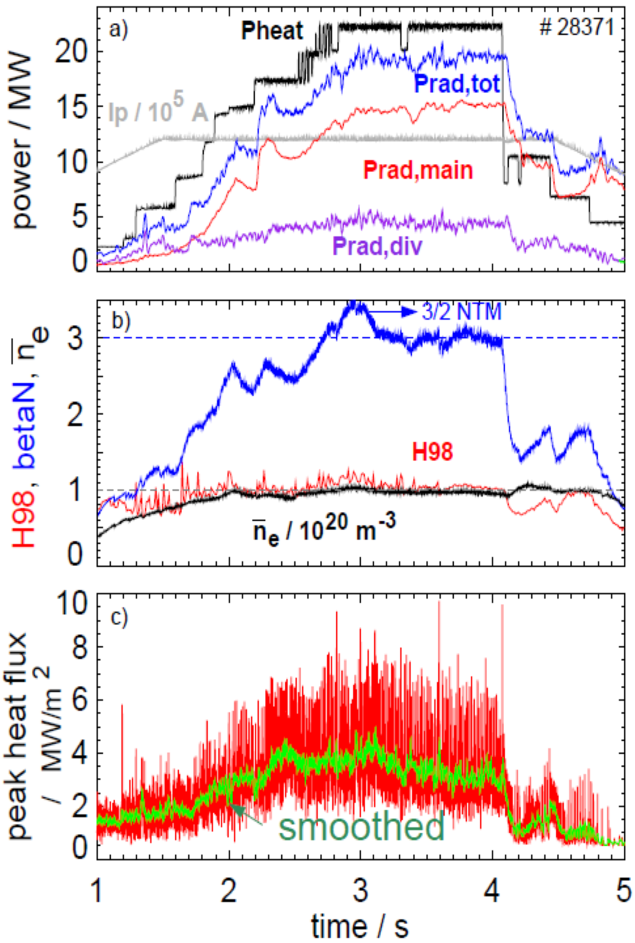

Fig. 4 AUG experiment with 80% of radiated power; a) all the different power in the experiment; b) confinment properties; c) power flux on the divertor

performances and with the adimensional Physics (ν *, ρ *, β) parameters closer to DEMO, the same results are achievable; when the radiation approaches values > 65% the confinement quality goes down up to the very low H98=0.8. But also ignoring this fact and remaining on the AUG data we can see that, assuming that 60% of the total DEMO heating power (≈300MW) is radiated in the plasma bulk, this will means that the power flowing through the plasma boundary and arriving in the divertor volume is $P_{Separatrix}$ ≈ 120 MW. On a machine like DEMO this power will be far to guarantee the necessary power input to the plasma edge to get the so called H mode (i.e. an edge energy transport barrier), that it is the easiest experimental scenario to achieve an high gain experiment. Actually we can think to reverse the problem: what is it the minimum $P_{Separatrix}$ value scenario compatible with a good H mode? Assuming P_{sep_Min} ≈ 200 MW ≈ 66% P_{TOT} it follows that the maximum radiation in the plasma bulk must b $P_{RAD\_BUL\_MAX}$ < 40%. By remembering the previous definitions and with some very easy algebra, it follows that, also assuming the very optimistic AUG data, when trusting only to the radiation to moderate the divertor tiles power flux, it comes out, for the large DEMO, Q_{DIV} ≈ 40 MWm^{-2}, more that a factor two of what sustainable from by using the W nonoblocks technology!

2.3 Power Exhaust: "Looking at the divertor magnetic topology ".

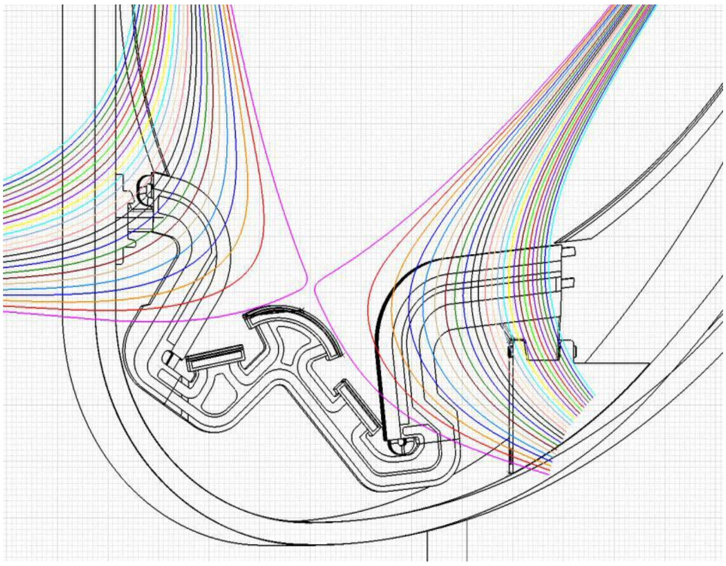

Fig. 5 FAST QSF equilibrium, with divertor designed for such configuration

As previously mentioned on top of the radiation a role solve the Power Exhaust problem can be played by the local magnetic topology in the divertor region. Lt us to recall how we can evaluate the divertor area where it is located the largest power of the exhausting power, A=2 π RF$_{EX}$ λ_E/sin. Here the role played by the magnetic topology is given by F$_{EX}$, that describes as the magnetic flux lines spread in the divertor region. There are several different ideas on how to change the divertor magnetic topology. For the purpose of this contract it is worth to quote the previously mentioned SF configuration. In this case in the X point is not null only the poloidal field, but also its derivative; this fact is practically realized by exactly overlapping two fields nulls, eventually this makes the region with very low poloidal magnetic filed very large. However the exact SF configuration is not viable for a series of reasons, it is intrinsically unstable and, mainly, the magnetic flux line would impinge the divertor plates with an angle too low, whilst, for technological reasons, this angle has to be larger that 3°. For this reason in a real experiment what it is realized is the so called quasi Snow Flake (QSF), where instead of two overlapped nulls the second one only approaches the main one. In Fig. 5 we report QSF equilibrium configuration for FAST, with a divertor in actively cooled monoblocks W, designed for such configuration. Here it is possible observe as the flux line impinge on very large surface of the divertor plates when compared with the standard X point configuration. In Fig. 6 we show the QSF configuration for a DEMO with major radius of 7.5m, here it clear to see the reciprocal role played by the different nulls. Another limitation to the "flexibility" of a QSF configuration comes from the fact that to realize configuration with two nulls close each other, the currents on the poloidal coils increases respect to the necessary ones for standard X point. Eventually considering all the technological limitation just mentioned, in all the mentioned cases (experiments and/or proposals) the highest obtainable gain by the only viable QSF configuration is a factor around 3. With the hypothesis of a conservative (but robust) reduction of a factor 2 we obtain (this time assuming no radiation losses) the following power flux figures: Q$_{DIV_QSF}$≈ 45 MWm^{-2} (DEMO) ÷ 100 MWm^{-2} (REACTOR) for an R=9 m machine and Q$_{DIV_QSF}$≈ 55

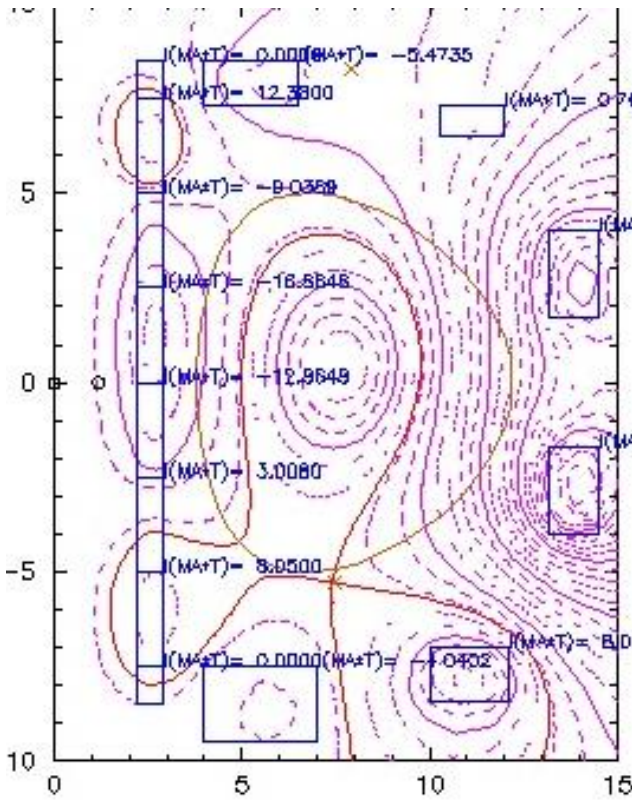

Fig. 6 DEMO (R=7.5m) QSF equilibrium, with the two nulls reciprocal position

MWm^{-2} (DEMO) ÷ 120 MWm^{-2} (REACTOR) for an R=7.5 m machine. Bearing in mind what mentioned for the independent role played, at the best by the radiation, it would come out that only the 9m DEMO could be very marginally within the technologically safe figure of $Q_{DIV} \leq 15$ MWm^{-2}

2.4 Power Exhaust: "Possible Solution ".

Looking at what just mentioned in the previous paragraphs, it would seems impossible to have a solution for the power Exhaust Problem, however there is a way out when considering possible synergies between the radiation and the divertor magnetic topology. Several different experiments [20] have clearly shown that the magnetic topology can strongly affect the local radiation, mainly for configurations like the QSF. Another important point to consider is that the pollution due to the Tungsten monblocks could help in the local radiation, but at the same time could pollute the bulk plasma up to a level to strongly reduce he plasma performances. Tungsten ions can "leave" the monoblocks and enter within the plasma by the sputtering mechanism, that strongly depends from the plasma temperature in front of the divertor plates. Obviously, also the plasma radiation depends from the local temperature and from the atomic Physics of the ions living in the divertor volume. Since all the dynamic of these ions is linked with the magnetic field it comes immediately clear why it could exist a synergy between the local radiation and the local magnetic topology. An idea like this suggests the possibility of using the magnetic topology to "control" and "enhance" the volume divertor losses up the level sufficient to solve the Power Exhaust problem. However, when acting in such away, it must be preserved the magnetic topology of the plasma core, that guarantees the plasma performances. In a recent past it has been developed on JET a Extreme Shape Control System (XSC) [21] capable to control exactly the plasma boundary, regardless of any modification of the internal plasma parameters or of any external perturbation, like the influx of impurities due to the strong interaction plasma-wall. Such a system in principle could do the job requested for controlling separately the plasma boundary from the magnetic topology of the divertor region, and to use this last feature to "control" the local radiation due to the intrinsic tungsten impurity or to some light impurities seeding. In Fig. 7 it is shown the possibility to such a control by a simulation performed on the EAST QSF equilibrium.

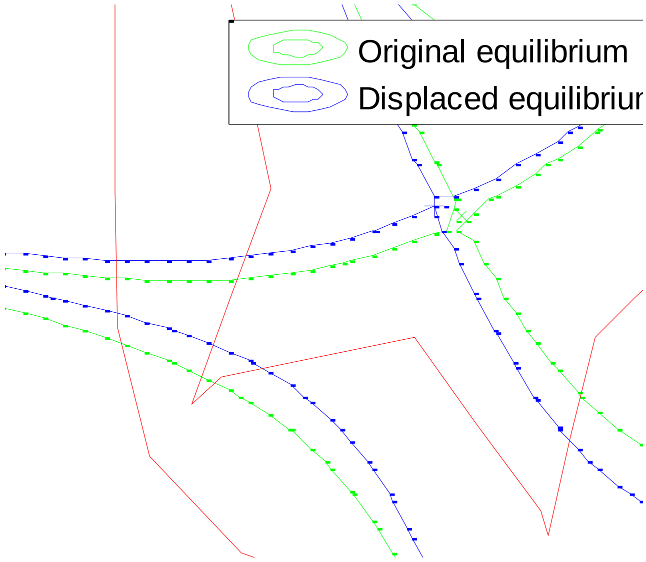

Fig. 7 EAST QSF equilibrium showwing the flexinility in "controlling" the divertor flux lines indipendently from the plasma boundary

3 Conclusions

The Power Exhaust problem has been identified as a potential stopper, on the Fusion Road Map, to realize a demonstrative Power Plant. As clearly highlighted in this research, it is quite obvious that the presently envisaged solutions, by themselves, will not be capable to bring down to the power flow on the divertor tiles up to the level of $Q_{DIV}≈15MWm^{-2}$, compatible with the best available material for realizing these tiles, monoblocks of Tungsten actively cooled. By using only radiation losses, of course approaching very large fraction of losses (≈90%), it could be possible to reduce the exhaust power at any low value, unluckily, when increasing the plasma bulk losses above a certain level it is experimentally demonstrated that the Plasma bulk losses his quality and the total performances strongly downgrade. On the other side, for several technological reasons, by playing only with the local magnetic topology there is no way to reduce the total power flux more than a factor 2÷3, against the necessary factor ≈ 10. Luckily, in principle the synergy between the radiation losses and the control of the magnetic topology, could give, in principle the possibility to achieve the necessary reactor target. Only new experiments like FAST and JT60-SA will have all the necessary edge (very large density power and possibility to test different divertor solutions) and plasma bulk features (adimensional parameters close to the DEMO and/or ITER ones); however the existing Chinese Tokamak EAST has already all the main divertor regions features that will be present on the future DEMO, consequently (even missing the fundamental edge-bulk integration) this experiment cal already well address and prepare the necessary future experiments on FAST and JT60-SA.

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5 Abbreviations and acronyms

3D	three Dimensional
CAD	Computer Aided Design
EM	Electro-Magnetic
FEM	Finite Elements Model
FW	First Wall
MHD	Magneto Hydro Dynamics
PC	Plasma Chamber
PF	Poloidal Field
RH	Remote Handling
SF	Snow Flake
SN	Single Null
SX	Super-X
TF	Toroidal Field
VV	Vacuum Vessel