Preliminary version – not for citation

Fusion Engineering and Design 146, Part B, 2019

https://doi.org/10.1016/j.fusengdes.2019.03.126

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Abstract. An upgrade to the lower divertor is currently being planned for EAST superconducting tokamak, aiming at reaching over 400 s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~ 10 MW/m^2. A new divertor concept for EAST, "Tightly Baffled Divertor", suited to water- cooled W/Cu plasma face components (PFC) with minimized divertor volume, has been proposed to achieve $T_{e,target} < 5$ eV across entire outer target at lower separatrix plasma density and optimized pumping by a simple closed divertor structure combining horizontal target with inclined baffle, dome and duct. This divertor should allow access to high-triangularity small Edge Localized Mode (ELM) H-mode regimes and also allow achieving advanced magnetic divertor configurations with the assistance of two water-cooled in-vessel divertor coils (Divertor coils). Preliminary engineering design of in-vessel Divertor coils indicates a maximum current of 8 kA for long-pulse discharges, and 20 kA for the shortest ones. However, flexibility on Divertor coils position optimization is limited to the water cooling system. Initial plasma equilibrium studies by EFIT code, used in combination with CREATE-NL and FIXFREE tools, show that the distance of the two nearby divertor poloidal field nulls, can be decreased up to ~ 0.95 m with a plasma current I_P ~ 400 kA, leading to a configuration with the secondary X-point located close to the target, with a significant increase of magnetic poloidal flux expansion and connection length. This may provide a promising divertor solution compatible with advanced steady-state core scenarios.

1. Introduction

One of the major issues facing the design and operation of next-step high-power steady-state fusion devices is the control of heat (and particle) fluxes and erosion of the plasma-facing components (PFC) [1]. Thus, it is essential to find plasma solutions that control heat fluxes to keep them within the heat exhaust limitations of the PFC, i.e, below 10 MWm^{-2} (including both graphite and tungsten), with divertor plasma temperature below 5eV. However, the aforementioned limitations are based on separate restrictions (thermal limits and sputtering of W, respectively) and thus must be simultaneously met. Experimental Advanced Superconducting Tokamak (EAST) is a fully superconducting tokamak capable of long-pulse operations with high power heating to challenge power and particle handling at levels comparable to ITER. The designed power level of auxiliary heating systems and recently achieved injection power [2] on EAST are the following: 2 Lower Hybrid Wave (LHW), the 2.45GHz (4MW, 2.8MW/6s, 1.2MW/410s) and 4.6GHz (6MW, 3.5MW/2s, 1.4MW/101s); Ion Cyclotron Resonance Heating (ICRH) system (12MW, 3.8MW&6s, 0.8MW/60s); Neutral Beam Injection launchers, the first one (4MW, 3.2MW/6s, 0.4MW/100s) and the second one (4MW, 2.6MW/2s); Electron Cyclotron Resonance Heating (ECRH) system (4MW, 0.6MW/5s, 0.5MW/101s). EAST is an up–down symmetric device, with the following main parameters: major radius R = 1.8 m, minor radius a = 0.45 m, toroidal field B_T up to 3.5 T, and plasma current I_p up to 1 MA for highly elongated plasmas with an elongation κ = 1.9. It can be operated in quite flexible plasma shapes with an elongation factor κ = 1.5–2.0 and triangularity δ = 0.3–0.6 for double null (DN) or SN divertor configurations [3]. EAST is equipped with 14 superconducting poloidal field (PF) coils ($I_{PF,max}$=11kA during normal/off-normal operations) for ohmic heating, ohmic current drive, shaping and position control, located outside the toroidal field coils (TFCs). In addition, two 2-turn in-vessel active feedback coils (IC coils), symmetrically located, in the upper and lower part of the vessel and connected in anti-series in order to provide a horizontal field [4], are used for fast control of the plasma vertical instability. Presently, PFC of EAST include ITER-like actively cooled W monoblock upper divertor with up to 10 MWm^{-2} heat removing capacity [5], a lower divertor in Carbon material and Molybdenum-tiled vacuum vessel.

Recently, EAST has been able to achieve ~60 s long-pulse H-mode [2], mainly limited by hot spots on the lower graphite divertor restricting heating power to < 3MW for ~100 s long-pulse H-mode operations. In addition, divertor C tiles, compared to W one, behind the high retention characteristics, presents both limited pumping capability (cooling water speed of 4 m/s) and lower heat removing capacity (~2MWm^{-2}).

Consequently, an upgrade to the lower divertor is currently being planned for EAST device to develop and demonstrate innovative boundary/plasma-material interface (PMI) solutions in EAST, aimed at reaching over 400 s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~10 MW/m^2 [6]. A new divertor concept for EAST, “Tightly Baffled Divertor”, suited to water- cooled W/Cu PFC with minimized divertor volume, has been proposed [7] to achieve $T_{e,target} < 5$eV across entire outer target at lower separatrix plasma density and optimized pumping by a simple closed divertor structure combining horizontal target with inclined baffle, dome and duct. This divertor should allow access to high-triangularity small-ELM H-mode regimes and also allow achieving advanced magnetic divertor configurations with the assistance of two water-cooled in-vessel divertor coils. The paper is organized as follows. Section 2 describes the conceptual design of the new water-cooled W/Cu lower divertor. In section 3 the current and coil position optimazion studies of in-vessel divertor coils are presented. Finally, section 4 contains a summary and an outlook.

2. Innovative divertor concept development for EAST

The overall goal of the new bottom divertor conceptual design in EAST has been to develop and validate a dissipative divertor with low W sputtering and strong divertor pumping for steady-state operations [7]. The strategies to achieve the aforementioned target have been mainly aimed to maximize the divertor power dissipation and pumping efficiency while minimizing divertor volume for maximizing core-plasma performance, that can mainly achieved by increasing the divertor closure and be compatible with advanced fully non-inductive core scenarios. In addition, a $T_{e,target}$<10eV across entire outer target at a relatively lower separatrix density should be achieved. The new divertor should guarantee enough shape flexibility in terms of configurations, by accommodating a relatively wide triangularity range, δ=0.4-0.6 plasmas, allowing access to the small-ELM H-mode regimes and advanced core scenarios [8], and finally, alternative magnetic configuration, as two nearby poloidal divertor nulls (2-NDN) [9], with the assistance of water-cooled internal coils, for power ad particle exhaust studies.

In turn, engineering issues related to water-cooled W/Cu PFC imposes strong constraints on the divertor structure design (e.g. curvature radius limit, end boxes, etc.) recommending the adoption of a simple geometry to facilitate manufacturing and engineering quality control (e.g. surface alignment, leading edge avoidance, etc), reduce costs, increase reliability. To achieve all these goals a novel concept divertor for, Tightly Baffled (TB) divertor, has been proposed. The main features of the TB structure are shown in Fig. 1 and summarized as following:

- two kinds of tungsten PFC with actively water cooling for high/low heat-load areas have been proposed:
  - vertical inner target (VIT) and horizontal outer target (HOT) in ITER-like W monoblock (heat removal capability ~10MW m^{-2}, water flow velocity up to ~ 8 m/s, flow rate in the water main up to ~800m^3/hour, pressure up to ~4MPa and baking possible up to ~250°C;
  - dome structure, VIT baffle, inclined baffle, reflection outer end box plates in flat-type structure (2mm thickness) with a heat remove capability ~5MW m^{-2};
  - it should be noted that ITER-like monoblock and flat-tile PFC have been used in EAST upper divertor for 4 years;
- inner baffle has been introduced to protect against downward strike point excursions whilst the dome structure to improve pumping and, in combination with the inclined outer baffle, to reflect neutral towards private region, increase neutral pressure, facilitate strike-point detachment and protect against transients;
- it should be noted that dome and outer inclined baffle present a curvature radius limit ~90cm and in both the structures the end box surface (flat-type W/Cu PFC with a heat remove capability ~5MWm^{-2}) is nearly parallel to the field lines to avoid direct exposure to heat flux;
- ITER-like neutral communication slot has been considered between the dome and HOT to reduce in-out divertor leg asymmetry; a duct (1.5cm x 5cm) has been added at the low field side (LFS) pump entrance to increase neutral pressure and pumping [7];
- two kind of cryopump have been taken into account: the first (~ 3 m long) with removal capacity 2x6m^3/s and the LFS one with a removal capacity ~75m^3/s;
- two water-cooled in-vessel coils (double filled red colour circles in Fig. 1) have been added to achieve a more flexible shaping and will be discussed in the next section.

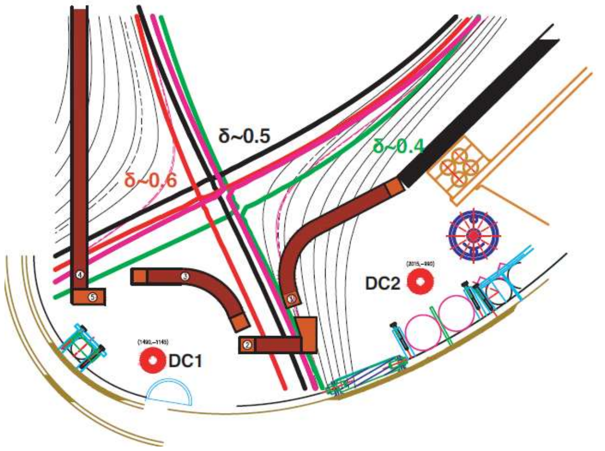

Figure 1. 2D schematic Tightly Baffled Divertor concept proposed for the new lower divertor on EAST device. The two internal divertor coils (DC) are reported as double filled red circles, labelled as DC1 and DC2.

3. In-vessel divertor coil studies

A wide range of alternatives divertor magnetic configurations, aiming at reducing the heat and particle loads at the plasma-material interface, have been developed world-wide (as recently reviewed in, e.g., [8]). In this context, the divertor properties of a 2-NDN configuration have been recently investigated, both on the lower [9, 10] and upper divertor [10], in steady state ($V_{loop} < 0$), H-mode ($H_{98}=1$), ELMs-free plasmas on EAST tokamak. The flaring of the magnetic flux (characterized by the magnetic field gradient) in the primary null is affected by the presence of the secondary null [8, 9, 11, 12] in the divertor region. This flaring could be then directly translated to the increased wetted surface area and reduced heat flux [11] or on an increase of the total radiated power [13]. However, it should be noted that the flaring could be considered as a useful tool for expanding the wetted area in an experimental machine with limited divertor plate angling, on the contrary, if the machine is designed with high plate inclination (e.g. ITER, DEMO) then there is no further benefit to flaring, unless this increases the connection length.

In EAST, the secondary null could be moved around [9] for studying a wide range of divertor/Scrape-Off Layer (SOL) physics (e.g. Snowflake-like [11] or single-legged X-divertor [14] configuration). A reduction of the power flow, on the upper W divertor plates, of about a factor 3 has been observed, of the same order of the increase of the magnetic flux expansion with respect to the standard single null configuration [9, 10]. In addition, the 2-NDN configuration has shown an ELMs-free behavior [10], that could be related to an interaction between the downstream configuration and the upstream features. To better investigate the ELMs-free regime in 2-NDN configuration, showing the possibility to reach such regime over a wide range of plasma parameters, further experiments are needed. However, the role played by the distance between the two nulls is a key physics point of these advanced configurations, and the recently upgraded EAST control capability [9, 15] will allow to vary the nulls distance during the discharge whilst keeping the plasma shape unchanged, in order to study this important physics feature. However, for optimizing the local magnetic configuration and consequently controlling various parameters related to the power exhaust (flux expansion, connection length, and especially the distance between null points, etc.), EAST will be equipped with a set of internal lower divertor coils capable to locally modify the magnetic field in the vicinity of the divertor target. Preliminary engineering design of in-vessel Divertor coils indicates a maximum current of 8 kA for long-pulse discharges, and 20 kA for the shortest ones. However, flexibility on divertor coils position optimization is limited by the water cooling system [7]. Using these in-vessel divertor coils, it will be possible to adjust a second null region in snowflake-like configurations [9], obtaining a large area where the magnetic poloidal field B_P and its gradient are close to zero or defining a XD-like configuration where the flux flaring at target can be largely varied. As example of the flexibility of such system, the equilibrium of the experimental low 2-NDN H-mode discharge #70358 at 3.1s (see Fig. 2), with I_P=250 kA, toroidal field B_T =2T, has been considered as starting point in our analysis with no current in the divertor coils. Initial plasma equilibrium studies by EFIT code [16], used in combination with CREATE-NL [17] and FIXFREE [18] tools, show that the distance of the two nearby divertor poloidal field nulls D_{xpts} can be decreased up to 0.464 m, with I_{DCmax}=5kA for long plasma discharge, and up to D_{xpts}=0.447m for short one with $I_{DC,max}$=20kA, by keeping $I_{PF,max}\approx$11 kA, leading to a configuration with the secondary x-point located close to the target (see Fig. 3a), with a significant increase of the magnetic poloidal flux expansion (of a factor~1.9) and connection length (of a factor ~1.5) with respect to the reference one.

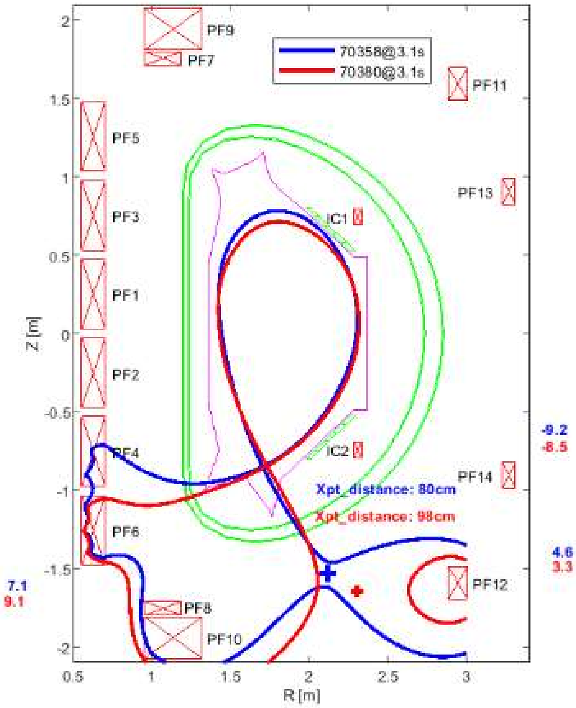

Figure 2. Use of internal coils divertor coils for the modification of the 2-NDN configuration, with I_P =250 kA, into a XD-like configuration with a reduced distance between nulls: experimental reference discharge #70358 at 3.1s with D_{xpts}=0.8m; b) 2-NDN configuration, at I_P =250 kA, with I_{DCmax}=20 kA and D_{xpts}=0.447m for short-pulse plasma discharges; c) 2-NDN configuration, at I_P=400kA, with I_{DCmax}=20kA and D_{xpts}=0.963m for short-pulse plasma discharges.

The distance between outer strike point and the end box is 4~5cm. The divertor magnetic geometric parameters, as outer target poloidal flux expansion $f_{x,OT}$ and connection length L are summarized, for all the studies discussed in this section, in Table I. However, as it can be expected, in order to satisfy the current constraints on PF coils (11kA), when the plasma current I_P is increased, the distance between the active and inactive

X-point will be increased. However, it will be possible to reach a 2-NDN configuration (see Fig.3b) at I_P=400 kA, in long-pulse discharges with $I_{DC,max}$=20 kA, at D_{xpts}=0.963 m, as reported in Table I. It should be noted that the equilibria shown in Fig.3a – b, are identified as modifications, constrained to keep the same elongation, of the reference discharge of Fig. 2. However, the growth rates of the modified 2-NDN equilibria are reasonably close to those of recent EAST experiments, as discussed in [9]. In addition, a solution to decouple the plasma vertical stabilization system from the plasma shape and position controller has been deployed and successfully tested at EAST [19] ensuring reasonable operational space to the feedback control to stabilize advanced magnetic configurations. Further development has been recently investigated by including the possibility to adapt in real-time the controller parameters for the proposed Vertical Stability (VS) discussed in [19] in order to avoid the use of different controller settings for different magnetic configurations, as well as the deployment of an algorithm for integrated control of plasma shape and flux expansion [20].

Table I. Comparison in terms of flux expansion, connection length and distance between two divertor poloidal field nulls at different plasma current with and without in-vessel divertor coils.

I_P [kA]	I_{DC1max}, I_{DC2max} [kA]	D_{xpts} [m]	$f_{x,OT}$	L [m]
250 (exp.#70358 at 3.1s)	0, 0	0.80	5.0	8.7
250	0, -4	0.464	8.1	12.5
250	20, -20	0.447	9.3	13.2
300	5, -5	0.684	5.5	9.7
300	20, -20	0.609	5.3	10.1
350	8, -8	0.812	4.7	8.5
350	20, -20	0.776	5.8	9.2
400	8, -8	1.07	4.5	7.4
400	20, -20	0.963	4.6	7.8

Next studies will be devoted to analyze the heat and particle exhaust properties of the 2-NDN configurations, reported in Table I, by means of edge codes as SOLEDGE [21] and EMC3-EIRENE [22] code, with and without impurity seeding at different electron plasma density and additional power.

4. Conclusions

An upgrade to the lower divertor is currently being planned for EAST superconducting tokamak, aiming at over 400s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~10MW/m^2, and also allow achieving advanced magnetic divertor configurations with the assistance of two water-cooled in-vessel divertor coils. Initial plasma equilibrium optimization studies have shown that the distance of the two nearby divertor poloidal field nulls can be decreased up to ~0.95m with a plasma current I_P~400 kA, leading to a configuration with the secondary X-point located close to the target, with a significant increase of magnetic poloidal flux expansion and connection length.

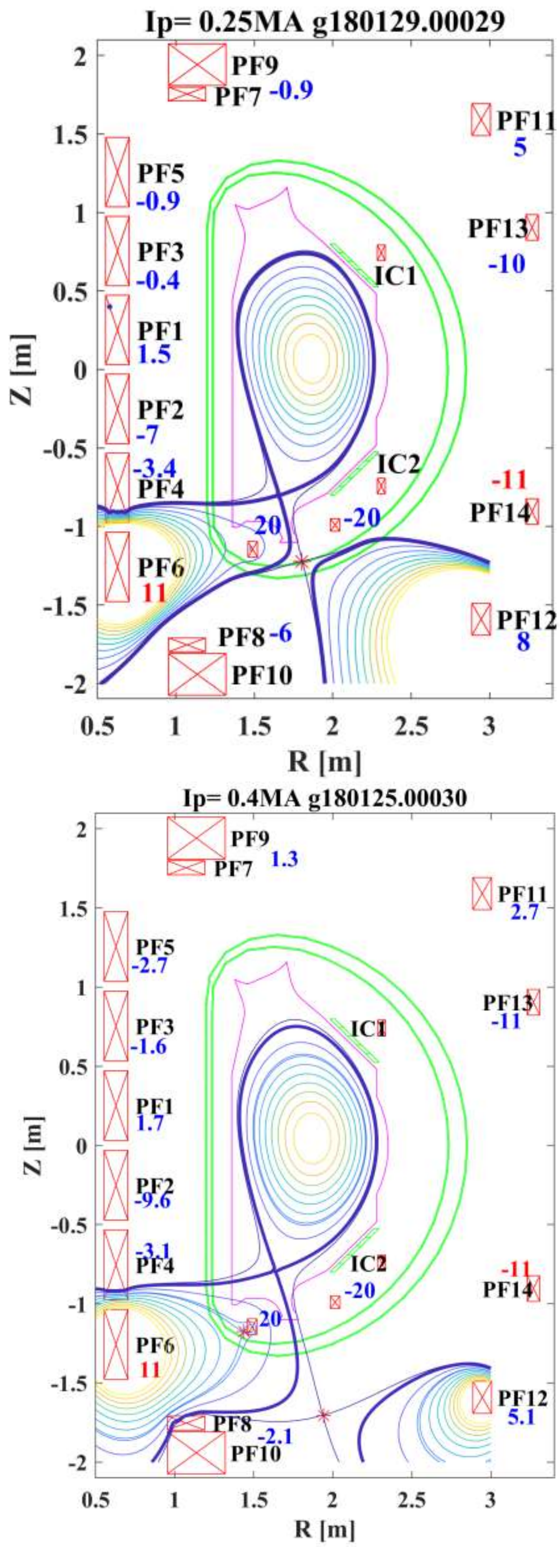

Figure 3. Use of internal coils divertor coils for the modification of the 2-NDN equilibria by means of EFIT code, with I_P=250 kA, into a XD-like configuration with a reduced distance between nulls: a) above, 2-NDN configuration, at I_P =250 kA, with I_{DCmax}=20 kA and D_{xpts}=0.447m for short-pulse plasma discharges; b) below, 2-NDN configuration, at I_P=400kA, with I_{DCmax}=20kA and D_{xpts}=0.963m for short-pulse plasma discharges.

Acknowledgment

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training program 2014-2018 under grant agreement No. 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission. This work was supported by National MCF Energy R&D Program of China, Grant No. 2018YFE0301105 and 2018YFE0302100, the National Nature Science Foundation of China under Contract No. 11875291 and under the EU-China Framework agreement.

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