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Progress on in-vessel poloidal field coils optimization design for alternative divertor configuration studies on the EAST tokamak

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Abstract. An upgrade to the lower divertor is currently being planned for EAST superconducting tokamak, aiming at >400s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~10MW/m^2. A new divertor concept for EAST, "Tightly Baffled Divertor", suited to water- cooled W/Cu plasma face components (PFCs) with minimized divertor volume, has been proposed to achieve $T_{e,target}$<5eV 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 (DCs). Preliminary engineering design of in-vessel DCs indicates a maximum current of 8kAt for long-pulse discharges, and 20kAt for the shortest ones. However, flexibility on DCs 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 ~0.95m with a plasma current I_P~400kA, 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.

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 (PFCs) [1]. Thus, it is essential to find plasma solutions that control heat fluxes to keep them within the heat exhaust limitations of the PFCs, i.e, below 10 MWm^{-2} (including both graphite and tungsten), with divertor plasma temperature below 5eV. Experimental Advanced Superconducting Tokamak (EAST) is fully superconducting tokamak capable of long-pulse operations with high power heating to challenge power and particle handling at levels comparable to ITER. 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 [2]. EAST is equipped with 14 superconducting poloidal field coils ($I_{PF,max}$=11kAt 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 [3], are used for fast control of the plasma vertical instability. Presently, PFCs of EAST include ITER-like actively cooled W monoblock upper divertor with up to 10MW m^{-2} heat removing capacity [4], a lower divertor in Carbon material and Molybdenum-tiled vacuum vessel (see Fig.1).

Recently, EAST has been able to achieve ~60sec long-pulse H-mode [5], mainly limited by hot spots on the lower graphite divertor restricting heating power to < 3MW for ~100s long-pulse H-mode operations. Indeed, operations in EAST rely on extensive lithium wall conditioning to reduce

H concentration, recycling and impurity, mainly due to the soft property of graphite. In addition, C tiles divertor, compared to W one, behind the high retention characteristics, presents both limited pumping capability (cooling water speed of 4m/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 >400s long-pulse H-mode operations with a full metal wall and a divertor heat load of ~10MW/m^2. A new divertor concept for EAST, "Tightly Baffled Divertor", suited to water-cooled W/Cu PFCs with minimized divertor volume, has been proposed [6] to achieve $T_{e,target}$<5eV 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 (DC 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.

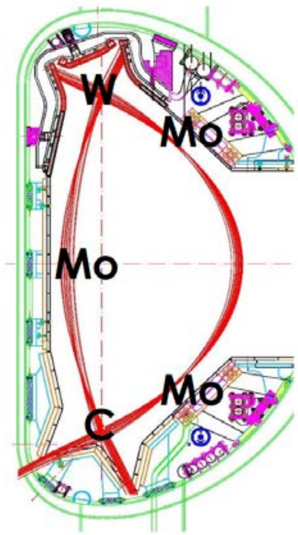


Figure 1. Present 2D EAST geometry schematic view.

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 [6]. The strategies to achieve the aforementioned target have been:

- maximize the divertor power dissipation and pumping efficiency while minimizing divertor volume for maximizing core-plasma performance, that can mainly achievable by increasing the divertor closure;
- be compatible with advanced fully non-inductive core scenarios;
- achieve $T_{e,target}$< 10 eV across entire outer target at a relatively lower separatrix density;
- have enough shape flexibility in configuration;
- accommodate a relatively wide triangularity range, δ = 0.4-0.6; high triangularity should allow access to the small-ELM H-mode regimes and advanced core scenarios [8];
- allow two nearby poloidal divertor nulls (2-NDN) configurations [7] with the assistance of water-cooled internal coils.

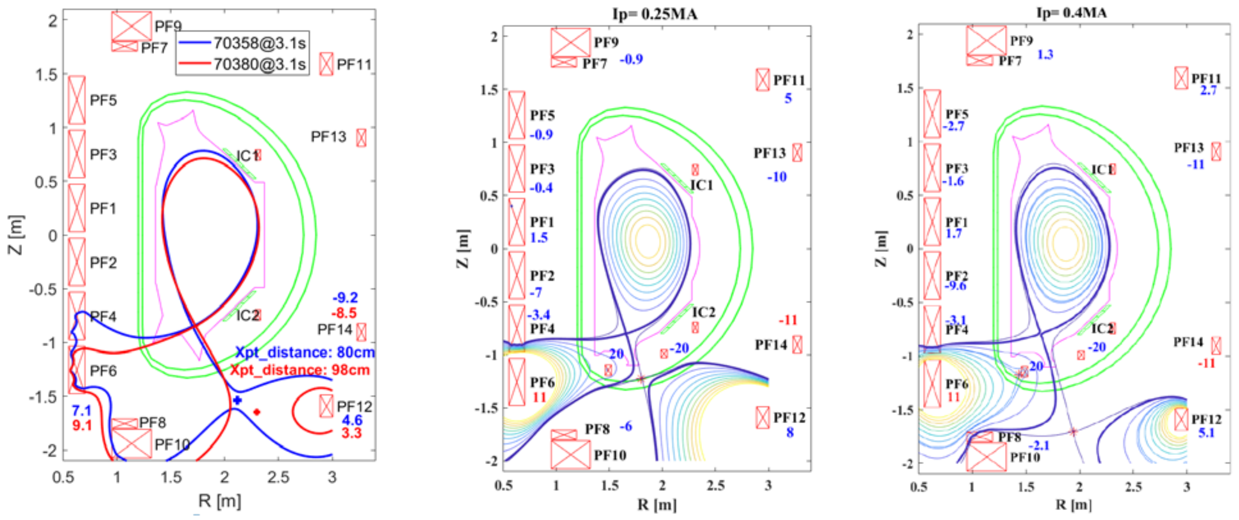
In turn, engineering issues related to water-cooled W/Cu PFCs imposes strong constraints on the divertor structure design (e.g. curvature radius limit, end boxes, etc.) recommending simple geometry to facilitate manufacturing and engineering quality control (e.g. surface alignment, leading edge avoidance, 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 TBD structure are shown in Fig. 2 and summarized as following:

- two kinds of tungsten PFCs 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 ~800 ton/hour, pressure up to ~4MPa and bakeable 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 PFCs 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 ~5MW m^{-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 [6];
- two kind of cryopump have been taken into account: the main one (~3m long) with removal capacity 2x6 m^3/s and the LFS one with a capacity ~75m^3/s;
- two water in-vessel coils (double filled red colour circles in Fig. 2) have been added to achieve a more flexible shaping and will be discussed in the next section.

Figure 2. 2D schematic Tightly Baffled Divertor concept proposed for the new lower divertor on EAST device. The two internal divertor coils are reported as double filled red circles.

3. In-vessel divertor coils 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 [7, 9] and upper divertor [9], in steady state (V_{loop}<0) H-mode plasmas, (H_{98}=1), ELM absent, 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 [7, 8, 10] in the divertor region. This flaring could be then directly translated to the increased wetted surface area and reduced heat flux [10] or on an increase of the total radiated power [11]. Due to the location of PF coils and target plates in EAST, the secondary null could be moved around from the primary one to form a magnetic configuration that features either a contracting [10] or flaring geometry near the plate, the latter being a feature of a single-legged X-divertor [12]. 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 [7, 9]. The configuration has led to an ELMs free regime [9], showing an interaction between the downstream configuration and the upstream features. 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 [7, 13] 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 DCs indicates a maximum current of 8kAt for long-pulse discharges, and 20kAt for the shortest ones. However, flexibility on DCs position optimization is limited to the water cooling system [6]. Using these in-vessel divertor coils, it will be possible to adjust a second null region in snowflake-like configurations [7], obtaining a large area where the magnetic poloidal field B_P and its gradient are close to zero or defining XD-like configuration where the flux flaring at target can be largely varied. As example of the flexibility of a such system, the equilibrium of experimental low 2-NDN discharge H-mode #70358 at 3.1sec (see Fig. 3a), with I_P=250kA, toroidal field B_T=2T, has been considered as starting point in our analysis with no current in the DC coils. Initial plasma equilibrium studies by EFIT code [14], used in combination with CREATE-NL [15] and FIXFREE [16] tools, show that the distance of the two nearby divertor poloidal field nulls D_{xpts}, can be decreased up to 0.464m, with I_{DCmax}=5kA/turn for long plasma discharge, and up to D_{xpts}=0.447m for short one with $I_{DC,max}$=20kAt, by keeping $I_{PF,max}$≈11kAt, leading to a configuration with the secondary x-point located close to the target (see Fig. 3b), with a significant increase of magnetic poloidal flux expansion (of a factor ~1.9) and connection length (of a factor ~1.5) with respect the reference one.



a) b) c)

Figure 3. Use of internal coils DC coils for the modification of the 2-NDN configuration, with I_P=250kA, into a XD-like configuration with a reduced distance between nulls: a) experimental reference discharge #70358 at 3.1s with D_{xpts} = 0.8m; b) 2-NDN, at I_P=250kA, with I_{DCmax}=20kAt and D_{xpts}=0.447m for short-pulse plasma discharge; c) 2-NDN, at 400kA, with I_{DCmax}=20kAt and D_{xpts}=0.963m for short-pulse plasma discharge.

The distance between outer strike point is 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 (11kAt), 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.3c) at I_P=400kA, in long-pulse discharge with $I_{DC,max}$=20kAt, at D_{xpts}=0.963m, as reported in Table I.

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 DC coils.

I_P [kA]	I_{DC1max}, I_{DC2max} [kAt]	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 code as SOLEDGE [17] and EMC3-EIRENE [18] 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 >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 (DCs). 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~400kA, 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.

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