Tokamaks with outer X-points: MHD stability

S.Yu. Medvedev[1,2], A.A. Martynov[1,2], B.V. Kuteev[2], A.Yu. Dnestrovskij[3], V.V. Drozdov[3]

[1]*Keldysh Institute of Applied Mathematics RAS, Moscow, Russia*
[2]*NRC «Kurchatov Institute», Moscow, Russia*
[3]*Tokamak Energy Ltd, United Kingdom*

Introduction. Negative triangularity (NT) tokamak plasmas are under active investigation both in the experiments [1,2] and in the studies of core physics [3] as well as the power handling relevant to fusion demonstration reactors [4]. The ideal MHD stability calculations for the NT plasma with X-points shifted to outboard side of the torus [5, 6] confirmed that despite the absence of a magnetic well the stability limits against external kink modes are compatible with the reactor requirements for normalized beta $\beta_N \geq 3$ but a factor of 4 lower pedestal as compared to conventional D-shaped tokamaks with positive triangularity. Therefore, strong ELM crashes critical for large tokamaks can be robustly avoided. However, some core confinement advantages observed in NT experiments, providing insight into the prospects of NT tokamaks as fusion systems, are not explained by plasma theory yet and all projections for reactor scales are very preliminary. The high energy confinement mode (H-mode) in NT diverted plasmas is still to be demonstrated. Vertical axisymmetric instabilities also pose a problem for NT plasmas with elongated cross-sections [5]. There are other engineering challenges to overcome, including the rejection of low-stressed D-shaped toroidal field coils substituted by the coils compatible to the outer divertor with a wider separatrix wetted area at larger major radius. A more general question arises concerning configurations that would allow a greater plasma volume to be pushed into a domain with a lower toroidal field without changing the aspect ratio (especially relevant for spherical tokamaks): is it a necessary handicap for a stationary outboard power exhaust?

The "power-handling-first" paradigm [4] opens a prospect for the outer X-point configurations pioneered by the JT-60 tokamak as reviewed in [7]. The plasma performance was not as good as in the rival D-shaped machines in 1980's and JT-60 machine was completely changed during the 1989 upgrade to lower X-point and elongated cross-section. However, even by that time it was understood that the presence of the X-point at the outboard side of the torus does not contradict to high normalized beta [8]. Indeed, ideal MHD stability calculations for outer X-point configurations demonstrate beta limits $\beta_N \geq 3$. In addition to that, having the outer X-point is an efficient way to avoid the second stability access for ballooning modes in the pedestal. As concerns the axisymmetric stability, the outer X-point plasmas with the elongation close to unity are passively stable against n=0 modes with an exception of localized symmetric (horizontal) n=0 peeling mode driven by high pressure gradient at the separatrix. This is in contrast to the vertical antisymmetric n=0 peeling mode existing only in connected double null configurations with lower/upper X-points [9]. Let us start with the consideration of possible magnetic systems for plasma configurations with outer or external X-points (XX) followed by MHD stability limits estimates and discussion of the XX prospects.

XX: magnetic systems. In this section, some examples of magnetic systems capable of sustaining the XX free boundary equilibria are demonstrated. The poloidal field (PF) coils layout and vacuum vessel for the DEMO-FNS [10] are used for reference. Interestingly, the plasma of the original JT-60 size fits the DEMO-FNS vacuum vessel very well. Additional divertor coils are needed to generate the outer X-point. A standard choice for that is a low net current triplet of divertor coils as in JT-60 in addition to vertical field coils (Fig. 1a). The numerical studies of the free boundary equilibrium using the SPIDER code have shown that vertical field and divertor coils can be combined providing an economical PF layout in terms of the sum of the PF coil current absolute values over the plasma current ratio, $\sum_i |I_{PF,i}|/I_p$, which is just above 1 in the case of the X-point lying close to the divertor coil (Fig. 1b).

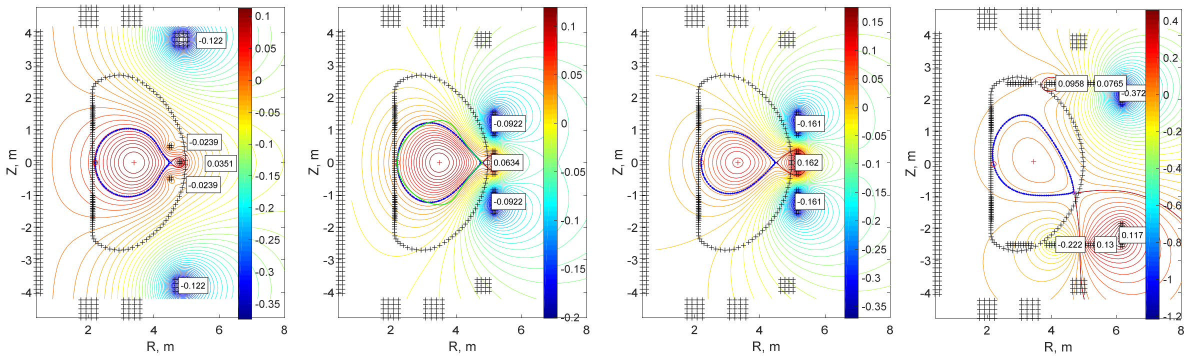

Fig. 1. Poloidal flux contours for free boundary equilibria sustained by the currents in outboard PF coils; a) nearly circular plasma boundary with X-point at the 4.5 m radius, current ratio 2.0; b) X-point radius 4.8 m, current ratio 1.55; c) X-point radius 4.5 m, current ratio 3.0; d) X-point radius 4.7 m, current ratio 6.0. Plasma current 0.16MA, PF coil currents in MA are shown, colorbar for poloidal flux in Wb/(2π) .

The farther the X-point is from the divertor coil, the larger is the $\sum_i |I_{PF,i}|/I_p$ current ratio (Fig. 1c). Such a magnetic system looks quite simplistic and potentially provides a large volume available above and below the plasma equator, but does not seem compatible with neutral beam injectors (NBI). More flexibility and volume would also be desirable for the divertor control. That is why an array of horizontal coils was tried out as an alternative option (Fig. 1d). While being quite flexible and seemingly better NBI compatible, this configuration requires larger PF currents due to divertor coils distanced farther away from the plasma.

XX: beta and pedestal limits. The KINX stability code [11] was used to calculate ideal MHD limits for a series of equilibria with the fixed separatrix from the free boundary equilibrium. Given the plasma boundary (coinciding with the separatrix) and initial parallel current density and pressure profiles, the limiting pressure profile was computed against the ballooning mode stability, with the current density profile kept fixed, and the pressure gradient iteratively adjusted. The resulting plasma profiles are shown in Fig. 2a for the XX configuration from Fig. 1a. The normalized current is defined as $I_N = I(MA)/a(m)/B(T)$, where I, a and B are plasma current, minor radius and vacuum magnetic field at the plasma center respectively, the normalized beta is $\beta_N = (2\mu_0 < p > / B^2) / I_N$. One can note the evidence of access to the second stability region in the core (local marginally stable pressure gradient, shown in the dashed line in the corresponding plot, is higher than the pressure gradient in the core) leading to $\beta_N > 4$ and quite a high bootstrap current fraction $f_b = 0.62$ (bootstrap current density in the collision-less limit is indicated by the dashed line in the parallel current plot in Fig. 2a). The optimized pressure profile was then rescaled and the marginally stable β_N was determined for external kink modes with toroidal mode numbers n from 1 to 5 (Fig. 2b) showing the uniform beta limit for the global modes close to the ballooning limit. The stability limits are very similar for the XX case from Fig. 1b. The ballooning limit β_N =3 is lower for the XX configuration from Fig. 1d, which can be considered as an extreme NT case: no second stability access in the core also with only a weak stabilization from the conducting wall due to strong coupling to internal modes [5].

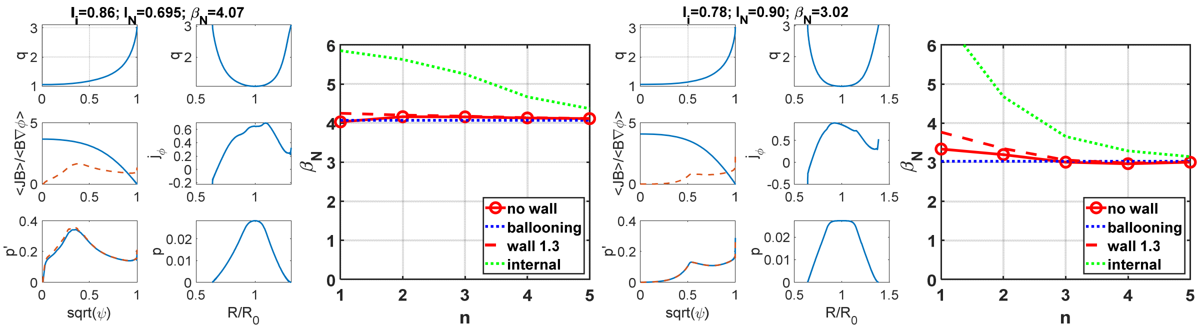

Fig. 2. a) and c): plasma profiles for optimized ballooning mode equilibria with prescribed parallel current density; b) and d) limiting values of normalized β_N for different toroidal mode numbers. Cases where there is no wall, a conformal wall at 1.3*a*, and a wall at the plasma boundary (internal) are compared for XX configurations from Fig. 1a and 1d respectively.

For edge stability calculations the same EPED-like model was used as in [4] with the core pressure gradient rescaled to step back from the global mode stability boundaries: to β_N =2.8 and 2.1, I_p =1.6 and 2.3 MA, B=2T for the cases from Fig. 2a and 2c. The standard hyperbolic tangent pedestal profiles for the plasma temperature and density were introduced with parameters close to JT-60 experiment [12]: $n_{ped} = 2 \cdot 10^{19}$, $n_{sep} / n_{ped} = 0.25$, T_{sep}=75 eV with T_{ped} adjusted to satisfy the DIII-D pedestal width scaling $\Delta = 0.076\beta_{p,ped}^{1/2}$ in the units of normalized poloidal flux. Then changing the width Δ the stability limits are found for the modes with toroidal mode numbers n = 1 – 20. The edge stability limits are set by high-n modes but with diamagnetic stabilization taken into account the limiting mode numbers shift to n = 3 – 5. The pedestal limits are close to each other for both cases and significantly lower compared to the standard D-shaped plasmas: measured in terms of the coefficient C_1 (about 3 for ITER-like plasmas) in the pedestal height scaling $\beta_{p,ped} = C_1 \Delta^{3/4} / I_N^{1/3}$ [13] they correspond to C_1 = 0.6 with the limiting pedestal width and height parameters Δ=0.012, $\beta_{p,ped}$ = 0.02, $\beta_{N,ped}$ = 0.07, T_{ped} = 150 eV. With the diamagnetic stabilization the limiting pedestal width is twice as wide Δ=0.024 (C_1=1.4) with the corresponding increase in the pedestal height $\beta_{p,ped}$ to 0.1, T_{ped} = 600 eV, which are still a factor of 3 lower compared to ITER-like plasmas (Δ=0.04).

Axisymmetric modes n = 0 are expected to be passively stable (without wall stabilization) for nearly circular XX equilibria. This is indeed the case of the equilibria shown in Fig.1a and 1b, with the pressure gradient going to zero at the plasma boundary. However, for the equilibria with rescaled optimized pressure gradient (Fig. 2a), which is finite at the boundary, the horizontal mode localized in the vicinity of X-point is destabilized at quite high beta over the ballooning limit. Symmetric horizontal mode is unstable for the equilibrium in Fig. 1c: with 3cm thick steel vacuum vessel the growth rate is about 10 1/s. Coupled vertical/horizontal mode is unstable for the equilibrium from Fig. 1d (profiles from Fig. 2c): the growth rate is about 30 1/s.

Discussion. In terms of MHD stability, the main advantage of the XX configurations is a robust low pedestal limit set by internal localized modes due to prohibited access to the second ballooning stability region in the pedestal. This opens a possibility of reaching soft edge plasma limits, imposed by an electro-magnetic turbulence or, at least, by much milder ELM crashes. The access to the second stability is still possible in the core of circular or elongated positive triangularity XX plasma, and the pressure-driven external kink limit, β_N, can reach as high as 4 due to this. Considering XX configurations as an alternative divertor solution, one can notice a simple magnetic system and possible NBI compatible flexible configurations with easier outer divertor access and better pumping conductance if the X-point is displaced from the equatorial plane. Anyway, larger outer X-point major radius for separatrix wetted area favors the power exhaust capability combined with a stationary – no large ELMs – confinement. At the same time, the main plasma bulk can reside in a higher B domain leaving the shaping coils in lower magnetic field region, e.g. making NbTi superconductor applicable for the shaping coils. Key remaining questions are related to energy confinement properties of the XX configurations.

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