NTST, A NEGATIVE TRIANGULARITY SPHERICAL TOKAMAK

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1. INTRODUCTION

NTST (Negative Triangularity Spherical Tokamak) is the world's first spherical tokamak with intrinsic negative triangularity under construction by Startorus Fusion. It is designed to explore the performance and application prospects of negative triangularity plasmas in fusion reactors. The device features a specially shaped central column, which resembles an hourglass, conforming to the shape of the NT plasma and freeing up space on the high-field side. Two pairs of PF (poloidal field) coils are positioned on the high-field-side, an unconventional arrangement for an ST (Spherical Tokamak) that enhances plasma control and mitigates engineering challenges associated with the central column. Based on these considerations, the magnetic design of NTST is implemented. The coil positions, the magnetic field null configuration, along with the discharge scenario, including breakdown and plasma phase for both NT (negative triangularity) and PT (positive triangularity) plasmas have been devised. By incorporating small additional in-vessel passive conductors, the VDE (Vertical Displacement Event) of NT plasma can be stabilized, and the high-field-side PF coils are proven to be the optimal choice for VDE control. Overall, the NTST device demonstrates its suitability for NT operations.

2. MOTIVATION

In recent experiments on TCV and DIII-D, it is found that plasmas with negative triangularity (NT, $\delta<0$) have better confinement properties compared with conventional positive triangularity (PT, $\delta>0$) plasmas, it is stated that NT L-mode is comparable to PT H-mode, meanwhile avoiding the problems of H-mode, e.g. ELM [Austin2019]. This discovery arouses massive interest among fusion research community in recent years. At the same time, spherical tokamak (ST) has been believed to be a promising way toward fusion energy, for its compactness, higher beta and lower cost compared with conventional tokamaks. TCV and DIII-D are conventional tokamaks, they have access to H-mode while most STs don't for lack of heating power. Apart from physical consideration, ST with NT divertor configuration also shows attractive engineering advantage, especially for divertor heating load. In NT configuration, the strike points are shifted outward, this would significantly relieve the divertor load and make maintenance easier. One may naturally think: what if we combine NT and ST, will such device exhibit higher performance than any former devices of both kinds? Would it be a competitive candidate for future commercial fusion powerplant?

Such an attempt was not carried out till now for the following reasons. Firstly, early theoretical and experimental research have shown that NT plasma has poorer MHD and VDE stability properties, for NT plasma has shallower magnetic well and lower average magnetic field compared with PT. NT plasma magnetic lines have longer bad curvature portion, the strong ballooning mode located there places a serious restrictions on NT beta limit; NT plasma is more strongly shaped by PF's field, resulting in a lower decay index and thus higher Z displacement growth rate, i.e. bad VDE controllability. So, NT was not a mainstream configuration in tokamak researches. Fortunately, these problems have been relieved by recent progresses in profile tuning [Medvedev2015] and feedback control techniques, making NT a feasible option. Secondly, ST both benefits and suffers from its compactness, the narrow space of central column is packed with solenoid and TF coils, defying the installation of high-field-side PF coils, which are essential for good NT shaping and VDE control. Specific designs may be needed for a ST oriented for NT operation.

3. PHYSICAL AND ENGINEERING DESIGNS

The main objective of NTST device is to explore the combination of ideas of negative triangularity configuration and spherical tokamak, to test whether the advantages of both ideas would sum up, resulting in a compact device with high confinement performance, low cost and easy maintenance. The main parameters of NTST are: major radius (R_0), 0.87 m; minor radius (a), 0.56 m; maximum on-axis toroidal field (B_{T0}), 1.0 T; nominal plasma current

(I_P), 1300 kA; maximum elongation (κ), 2.0; triangularities (δ), -0.8 – 0.6. The wall of the vacuum vessel is made of Inconel 625 stainless steel, thickness = 2 mm on high-field-side, 6 mm on low-field-side. The magnetic field penetration time is 0.5 ms from high-field-side, 5.5ms from low-field side. This means plasma would respond faster to high-field-side coils. Additional reinforcing ribs are attached to the wall. The configuration of poloidal coils is shown in *Fig. 1* left. The engineering design of NTST is shown in *Fig. 1* right. There is a larger Dewar containing the whole vessel and coils, not drawn in the picture.

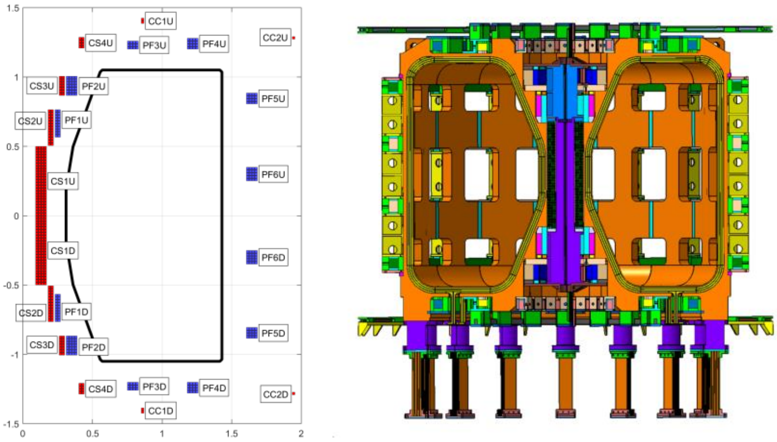

Fig. 1 The poloidal cross-section of NTST (left), the engineering design of NTST (right)

Two example diverted configurations with 1000 kA plasma current are shown in *Fig. 2*, one is NT (δ = -0.8), another is PT (δ = +0.6). Overall properties are kept same: $R_0 = 0.77$ m, a = 0.4 m, A = 1.9, $\kappa = 2.1$, $\beta_p = 0.4$, $l_i = 1.3$, $q_0 = 1.05$, while q_{95} differs: NT is $q_{95} = 3.3$, PT is $q_{95} = 4.4$. This comes from the obvious fact that NT bulk plasma locates further outside than PT, resulting in weaker mean B_T. We can see that the PT one has a less average coil current, because this is a more natural shape for ST since ST plasma tends to be elongated and PT-shaped. We will have to spend more effort to shape the plasma into NT, and such deviation from natural shape shall cause stronger instabilities, as is well known.

We can clearly see the difference in divertor strike points' radius and area, the idea of enlarging strike area by NT is verified. Owing to the proper locations of PF3,4,5 coils, a snowflake divertor readily forms in NT configuration, making the strike area even larger.

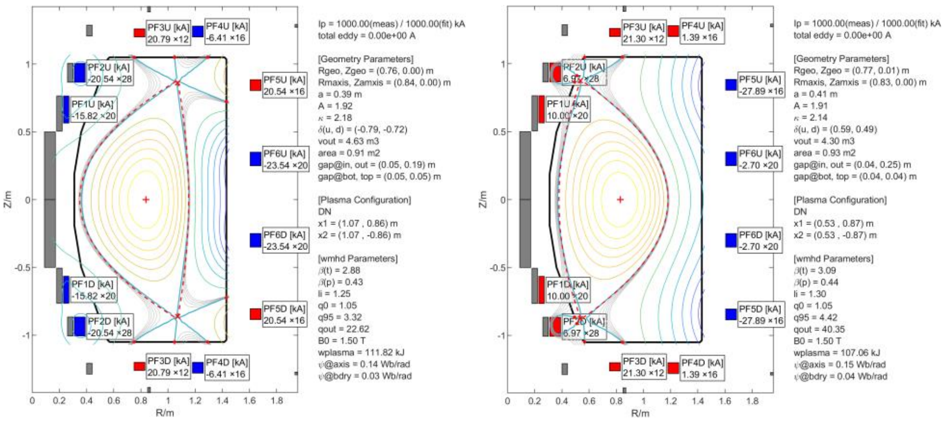

Fig. 2 The NT (left) and PT (right) configurations of NTST

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