DESIGN OF TOKAMAK PLASMA WITH HIGH Tc SUPERCONDUCTING COILS*

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Abstract
This paper presents a design of tokamak plasma in light of how the small ignited tokamak is possible with use of the HTSC coils as plasma stabilizer. The same data base and formulas as ITER are here used and any innovative technology other than the HTSC stabilizing coils is not assumed.

1 Introduction

High Tc superconducting (HTSC) coils have an ability to improve plasma positional instability if they are placed in plasma vicinity. This stabilization method is superior to the conventional one, i.e., active feedback control with PF coils, in two points: (1) faster response to plasma change owing to their quite low resistivity, (2) no need of power supply. This implies highly elongated plasma with HTSC coils becomes a candidate to improve plasma performance, leading to realization of compact ignited reactors.

In order to demonstrate the effect of HTSC coils on the tokamak plasmas, an example of designs with use of them was introduced in light of how the design of smaller ignited tokamak is possible. Assumptions adopted here are to use the same data base and formulas as ITER in principle and not to use any innovative technology other than HTSC stabilizing coils.

2 Determination of major design parameters

Major design parameters are decided with 0-dimensional plasma analysis employing the I-A-B_{tf}-κ analysis methodology which was introduced in the ITER Conceptual Design Activity[1].

Firstly, we determined apriori the basic parameters, the elongation κ=2.3, the maximum magnetic field in TF coils B_{tf}=12.5(T), the safety factor $q_\psi > 3.0$ and the thickness needed for shielding neutrons $t_{\mathrm{shield}} = 1.05$ m with intention of a smaller tokamak of reasonable fusion power. Here, the elongation was chosen as the maximum value in the range that plasma positional stability is secured with HTSC coils after some iterations of plasma stability analysis.

The $I_p - A$ space of HTSC tokamak ($\kappa = 2.3$) is shown in Fig. 1(a) with parameters of major radius determined together with that of ITER (Fig. 1(b)). The hatched region in the figure corresponds to the possible design window which is bounded by consideration of the confinement, the beta limit and the radial build needed to provide the necessary magnetic flux for flat top duration of 1000 sec. It can be easily recognized that major radii in the possible region of HTSC tokamak are much smaller than that of ITER. We determine I_p = 18(MA) for A = 3.5 and R = 5.34(m) as the design point. This is very small compared with the size of the present ITER.

Then, we have to determine plasma parameters so as to secure an ignited condition. For this purpose, a POPCON(Plasma OPeration CONtour) is shown in Fig.2 together with the beta limit based on the Troyon scaling. The scaling law of energy confinement in ITER (ITER-89P, $H = 2.0$) [2] was used to estimate the energy balance of the plasma.

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The digits on the curves in the figure are the additional heating power, P_{aux}, so that domain bounded by two curves, $g = 2.2$ and $P_{aux} = 0$, gives possible (n, T) sets of ignited operation. If we take a fusion power of 500 MW, contour line of 500 MW gives $< n >= 0.9 \times 10^{20} 1/\text{m}^3$ and $< T >= 13$ keV as the cross point with the curve of $P_{aux} = 0$.

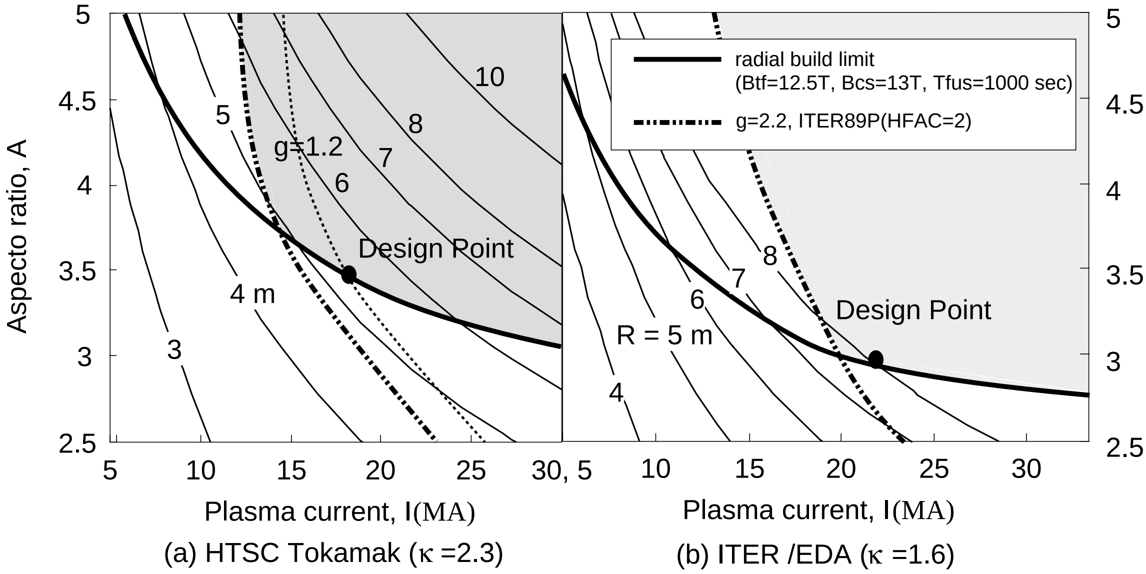


FIG.1: $I_p - A$ space of HTSC tokamak and ITER/EDA.

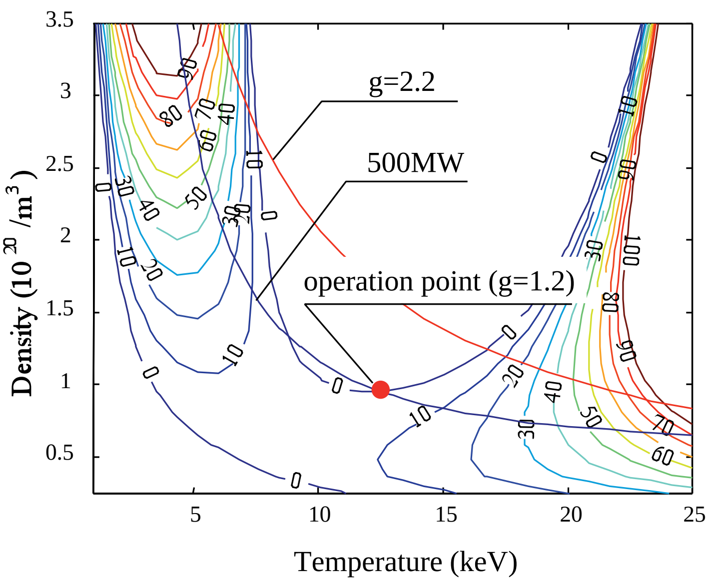


FIG.2: POPCON diagram of the operation point.

3 Analysis of plasma positional stability

The 2-D plasma equilibrium code, SYSTEQ [3], was used to find the location of PF coils with magnitudes of the coil currents. Results are shown in the upper part of Fig.3. Total stored energy(2.71GJ) in the PF coil system is smaller than ITER(47.0GJ). Here, a double null system of magnetic separatrix is adopted to reduce heat flux on the divertor. In the ITER design, a heat flux on the divertor is 5MW/m^2 during a normal operation and proportional evaluation of a heat flux shows about 2 MW/m^2 on the HTSC tokamak divertor.

An elongation of the HTSC tokamak is set a priori as higher than ITER, so that the plasma positional stability should be examined rigorously. The plasma response against the disturbances was computed via the TOFU code, which is based on non-rigid, MHD consistent displacement model [4]. In order to take the consideration of the effect of HTSC coils, TOFU code was coupled by a newly developed code i.e. the HTSC circuit code [5]. Non-linear behavior of HTSC material (Bi-2223) is considered in the form of flux flow and creep model[6]. Stability analyses were conducted for the equilibrium arrangement obtained by SYSTEQ. Electrical resistances are 67.9, 5.63 and 9.41 ($\mu\Omega$) for the first wall, the back plate and the vacuum vessel, respectively, using the data base of the ITER design. In order to improve the plasma positional instability, four pairs of HTSC coils are installed at the back plate of Fig.3 after iteration for optimization of the location and the number of the coils. Here, we estimated the plasma response against the disturbance of $\delta\beta_p = -0.2$. Plasmas move to the in-board side by 16 cm and recover to the original position according to the change of β_p as shown in Fig.4. It does not touch the first wall because an original gap is sufficient, 25 cm. The above result imply that high elongated tokamaks are feasible if HTSC stabilizing coils are applied.

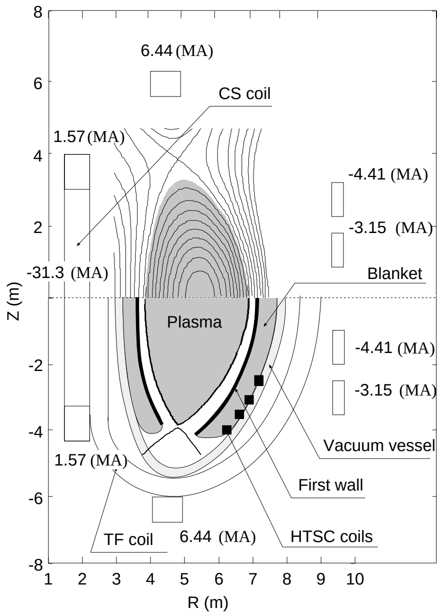


FIG.3: Two kinds of cross section of magnetic flux contour lines and system components.

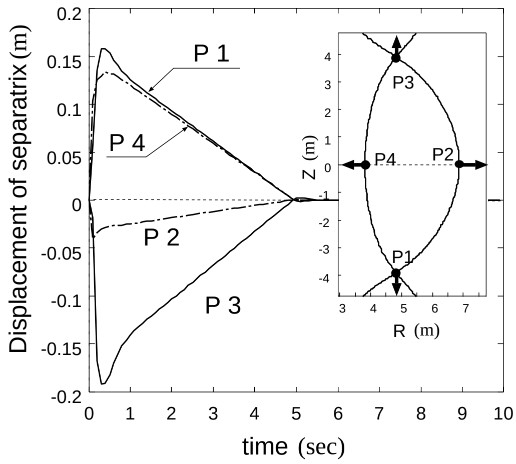
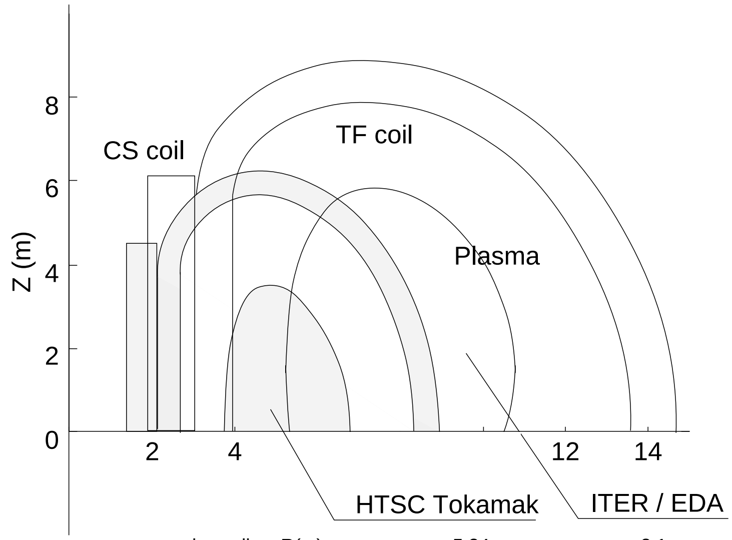


FIG.4: Motion of separatrix during plasma disturbances.

4 Conclusion

To demonstrate how possible small HTSC tokamak is, cross-sections of the two designs are compared in Fig.5 together with primary design parameters. The following remarks are drawn from the figure:

1. Smaller tokamak can be designed with parameters of lower fusion power of 500 MW, reduced stored magnetic energy of coil system, easy handling of heat deposition at the divertor, *etc.*.
2. Additional heating power required for L/H transition can be reduced significantly due to the smaller major radius.
3. Fabrication can be facilitated and remote handling can be accommodated easily.
4. Amount of low level radioactive waste can be reduced due to small scaled structure.
5. Volume of the nuclear island of the HTSC tokamak is roughly one-third of ITER.



major radius, R(m)	5.34	8.1
aspect ratio, A	3.5	2.9
elongation, κ	2.3	1.6
magnetic flux density at plasma center, B (T)	6.5	5.7
plasma current, Ip (MA)	18.0	21.0
normalized beta, β	1.2	2.2
pulse length (sec)	1000	1000
thermal output (MW)	500	1500

FIG.5: Dimensional comparisons of ITER/EDA and HTSC tokamak.

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