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Engineering optimization of heliotron configuration for larger blanket space

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In the conceptual design studies of the heliotron fusion reactor [1], optimization of the magnetic configuration has been investigated both from the physical and engineering viewpoints. From the physical viewpoint, the simultaneous improvement of both plasma confinement and MHD stability is a critical issue, and the helical pitch modulation parameter α of the continuously-wound helical coils is chosen to be 0.0 in recent designs in contrast to 0.1 employed in the former design so that the resultant fusion gain would be enhanced from 10 to 15 [2]. From the engineering viewpoint, acquiring enough blanket space between the helical coils and the plasma, especially at the inboard side of the torus, should be one of the critical issues to ensure a high tritium breeding ratio and sufficient radiation shielding capability to the superconducting coils. For this purpose, various methods have been investigated so far in addition to having a high current density in the helical coils. One of them is to apply the NITA coils which are the sub-helical coils located outside the main helical coils having 10-20 % of the current in the main helical coils in the opposite direction [3, 4]. An example of a NITA coil application is depicted in Fig 1(b) in comparison to the standard case of (a) for the design with a major radius R_c of 7.8 m [5]. Here it is noted that the minor radius of the helical coils, a_c, should be enlarged together with the use of the NITA coils, through the helical pitch parameter

$$\gamma_c = \frac{m}{l}\frac{a_c}{R_c},$$

where m (10) and l (2) are the toroidal pitch and poloidal pole numbers, respectively. The blanket space in (b) is still not sufficient to have a long (>30 years) lifetime for the helical coils due to the intense neutron irradiation. Another difficulty accompanied by the NITA coils is that the main helical coil current should be increased to keep the original toroidal magnetic field. To obtain a larger blanket space, it is thus required to increase the current in the NITA coils much more, which should further increase the helical coil current.

With these conditions, a different set of coils is being explored to obtain a larger blanket space. One such trial is the toroidal field central (TFC) coils, which are the toroidal field coils located in the central region of the torus. An example of magnetic surfaces obtained by the TFC coils, together with a further enlarged γ_c of 1.35, is depicted in Fig. 1(c) which shows a significantly larger blanket space. The TFC coils do not alter the toroidal magnetic field in the plasma region, and thus, there is no need to change the main helical coil current. However, there are still many engineering issues associated with the TFC coils. Presently the current in the TFC coils is as large as that in the main helical coils, which give a high electromagnetic stress in the supporting structure. Thus, a more compact and optimum design of TFC coils is required, and the electromagnetic supporting structure should be optimized, as has been investigated for the former designs [6].

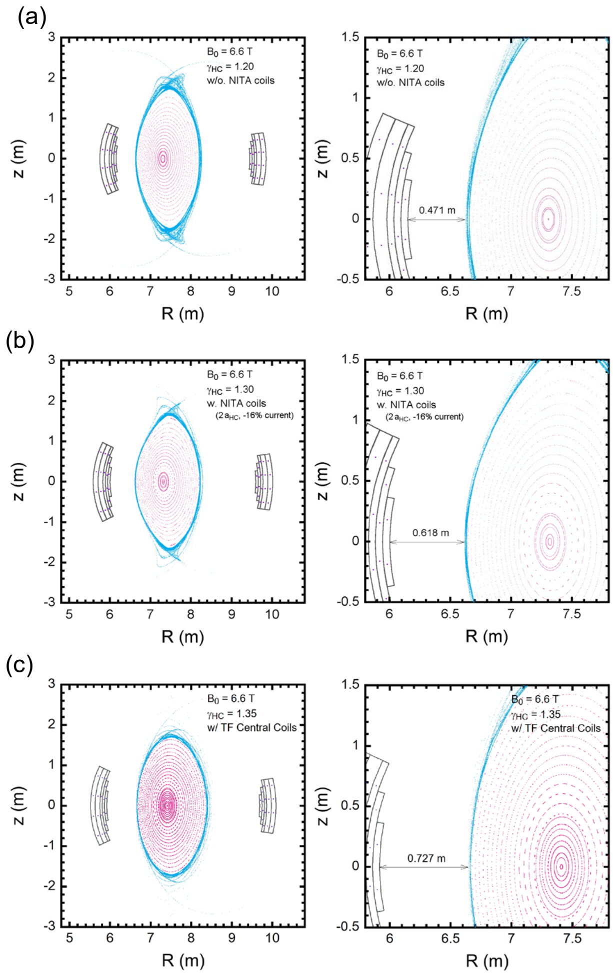

Figure 1 Vacuum magnetic surfaces of the heliotron fusion reactor at the toroidal angle where the helical coils are located on the equatorial plane with (a) the standard configuration of $\gamma_c = 1.25$, (b) $\gamma_c = 1.30$ and NITA coils, and (c) $\gamma_c = 1.35$ and TFC coils. The right-hand figure in each row is an enlarged image of the left-hand figure.

References

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