§4. Effect of Outboard Helical Field on Toroidal Plasmas

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The TOKASTAR configuration[1)] is proposed as one of compact tokamak-helical hybrid confinement systems. We first built an N (toroidal mode number) =1 and N = 2 compact coil system C-TOKASTAR[2-3)] (Compact Tokamak/Stellarator Hybrid) without toroidal coil system. This system has several advantages: (1) steady-state operation by helical coils, (2) no current disruption risk by external helical field application, (3) probable high-beta achievement by strong magnetic well, (4) enough divertor space by simple coil configuration, (5) compact economic system by spherical configuration and (6) easy maintenance by simple N =1 or N =2 coil system. Based on the achievement of C-TOKASTAR, a new small device named TOKASTAR-2[4-5)] was designed and constructed. Different from the C-TOKASTAR coil system, the toroidal field coil system is added in the TOKASTAR-2 to generate both tokamak and helical configurations independently. Recently, AHF (Additional Helical Coil) and PVF (Pulsed Vertical Field Coil) were newly installed (Fig. 1).

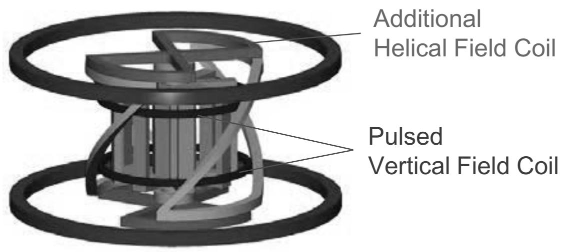

Fig. 1 Coil configuration of TOKASTAR-2. AHF (Additional Helical Field Coil) and PVF (Pulsed Vertical Field Coil) were newly installed

Using OH coils (pulse discharge) and VF coils (static DC power supply), previously we obtained only 90A plasma current. Moreover, the vertical displacement of OH plasma was observed using a fast camera (40500fps) [4-5)]. Then, we installed a conductive shell to increase plasma current and to suppress the vertical plasma displacement. Compared with the values of plasma current without the conductive shell, induced plasma current with the shell increases by 40 A. This increase in plasma current was due to the suppression of the vertical displacement by the conductive shell.

To induce more plasma current and to apply appropriate time-varying vertical field which balances with the hoop force, we installed a new pair of PVF coil (Fig.1) inside the vacuum chamber with pulsed power supply. By this PVF system, about ten times larger plasma current can be achieved (Fig.2).

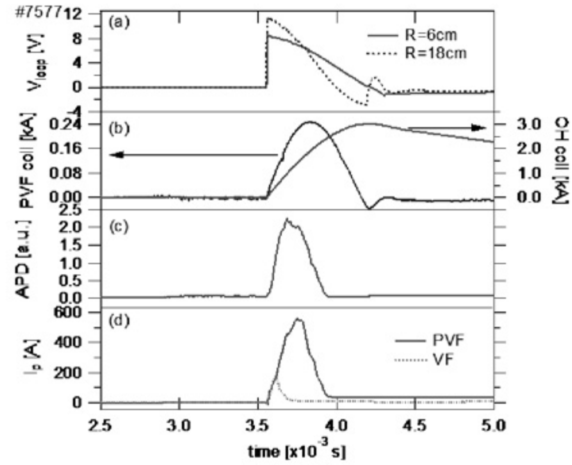

Fig.2 A half kA tokamak operation in Tokastar-2. By applying PVF instead of constant VF, ten times larger plasma current (I_p) can be achieved.

Relevant to helical field application to burning tokamak plasmas, neoclassical tearing mode (NTM) analysis has been carried out using 1.5-dimensional transport code TOTAL, in which the time variation of magnetic island is described by the modified Rutherford equation. In addition to helical field stabilization od NTM, Electron Cyclotron Current Drive (ECCD) stabilization effects are studied[6-7)]. The EC control efficiency depends on EC modulation width and EC injection phase lag from the O-point of magnetic island. NTM in ITER can be stabilized when the EC phase lag is smaller than 10% and the EC modulation width is around 20%, when the time-averaged EC current and power is fixed.

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