

Helical Coils for the Positional Stability and Elongation of Tokamak Plasma

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Introduction

Tokamaks are the most leading devices of fusion reactors. However, tokamaks have a serious problem: disruptions which cause damages to reactors. Since it is difficult to avoid disruptions completely, we must make countermeasure development to ensure the reactor safety. The process of disruption damage is as follows: As is well known that a good energy confinement and high β can be achieved in elongated plasmas. However, elongated plasmas suffer from vertical instabilities. Usually, we stabilize vertical modes by shell effects and active control using a feedback system. At a disruption, however, large eddy currents are generated in thermal and current quench phases and the feedback system fails to keep the plasma at a desired position. Then a vertical displacement event (VDE) is triggered. The contact of plasma and first wall will lead to the damages of the reactor by high heat flux and induced electromagnetic forces.

Helical coils for the positional stability and elongation

We propose a simple helical coil system which passively stabilizes vertical modes and produces plasma elongation. It avoids an occurrence of VDE. In other words, the coils raise economic efficiency and safety of fusion reactors. Figure 1 shows a schematic illustration of the coil configuration. The windings on the top and bottom sides of the plasma produce finite averaged horizontal field B_R [1] which stabilizes vertical modes. Since averaged fields increase toward the top and bottom sides of plasma. While the windings on the outer side of torus produce finite averaged vertical field B_z .

To confirm this stabilizing and elongation effects, we consider an approximation model. We take x, y, z-direction as horizontal, vertical and toroidal directions, respectively. And we assume the coils with such a dependence as $u(x, z) = u(x + kz)$ on the x-z plane [3]. In vacuum, we have the following equations.

$$\nabla \cdot \mathbf{B} = \nabla^2 \chi = 0, \quad (1)$$

$$\chi = \alpha \exp(-yn\sqrt{1+k^2}) \sin n(x+kz), \quad (2)$$

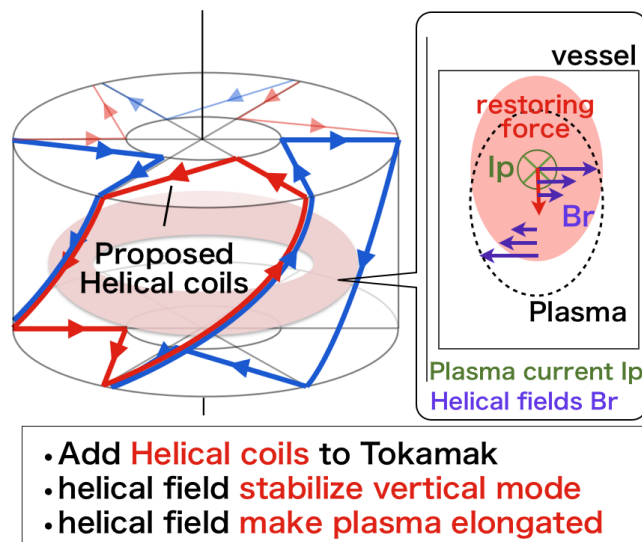


Figure 1: Schematic illustration of the coil configuration. The arrows indicate the direction of coil currents.

where χ is contribution to the scalar potential from helical coils, α is the ratio between toroidal field and helical field strengths. From Eq. (2), we can calculate B_x , B_y .

$$\overline{B_x} = -\frac{1}{2}n^2\alpha^2k\exp(-yn\sqrt{1+k^2}), \quad (3)$$

$$\overline{B_y} = 0. \quad (4)$$

By averaging B_x , B_y over a helical period along the magnetic lines of force, it was confirmed that Eqs. (3)-(4) show produced magnetic field includes only B_x component which depends on the pitch k and y as the distance from the current on the x - z plane. Similar results can be obtained for the outer side of torus coils. Thus we can control the horizontal magnetic fields and vertical magnetic fields independently by varying the pitch values for each part of the coils separately. This characteristics cannot be achieved by axisymmetric coils. Figure 2 shows projection plots of the lines of magnetic force with the currents on the top and bottom of the helical coils. They indicate that the field has only averaged B_r component.

Closed magnetic surface by external coils

Furthermore, the proposed coils can make closed vacuum magnetic surfaces in combination with poloidal field coils. Since usual tokamaks maintain closed magnetic surfaces with the plasma current, the current quench during a disruption will lead to the disappearance of closed magnetic surfaces. On the contrary, our coil system can keep closed magnetic surfaces. Figure 3 shows a disruption control sequence by the coils. In the first phase of a disruption when the plasma current does not drop so much, the plasma position is maintained by the averaged B_r ,

component. In the latter phase, the plasma confinement can be preserved and recovered by heating the plasma up again owing to the closed magnetic surfaces even with small plasma current.

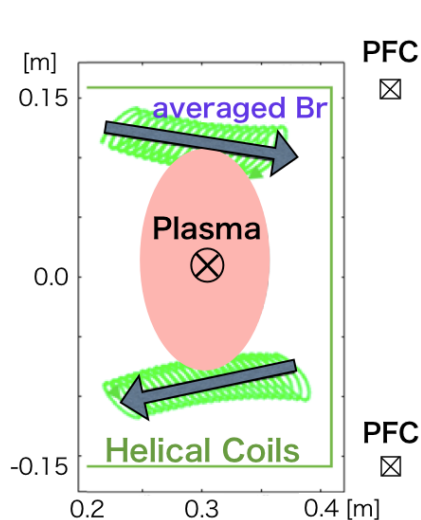


Figure 2: Projection plots of the magnetic lines of force with helical fields generated by top and bottom sides.

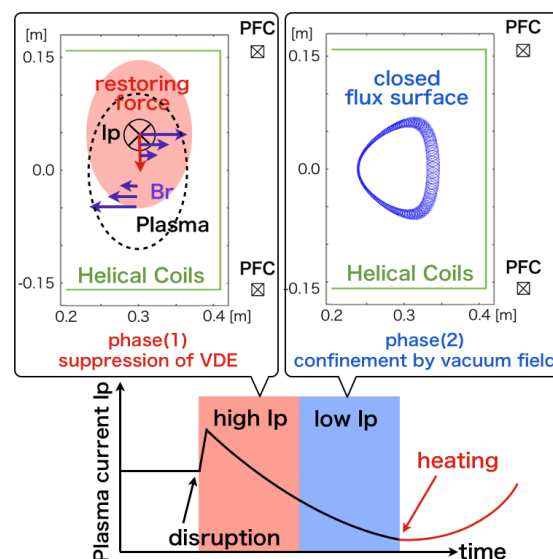


Figure 3: Disruption sequence with helical coils which keeps the plasma position and closed magnetic surfaces.

VMEC Results

We analyzed magnetic flux surfaces with VMEC with free boundary conditions. Figure 4 shows equilibrium configurations with helical fields. Six helical coils are located around the torus except the inner side. The plasma parameters of the configuration plotted in black are $R = 0.3$ m, $a = 0.08$ m, $B_T = 0.5$ T, $I_p = 11$ kA, $\beta = 3\%$. We confirmed that the crosssection is elongated, whose averaged ellipticity κ is 1.6. The q -profile increases with normalized small radius in the same way as conventional tokamaks. Then, we simulated a disruption as by dropping β to zero and the plasma current to a half. The configuration plotted in red shows that the plasma position remains almost unchanged in the horizontal direction.

Summary and future work

We proposed simple helical coils for the position control and plasma elongation and analyzed the magnetic fields by theoretical and numerical methods. We confirmed that the fields stabilize vertical modes and form closed flux surfaces. The calculation with the VMEC code demonstrated that our coil system including proposed simple helical coils produces elongated plasma and that it has a capability of maintaining plasma configuration under a drastic change in parameters. We will evaluate the VDE stabilizing effect further by using the HINT code.

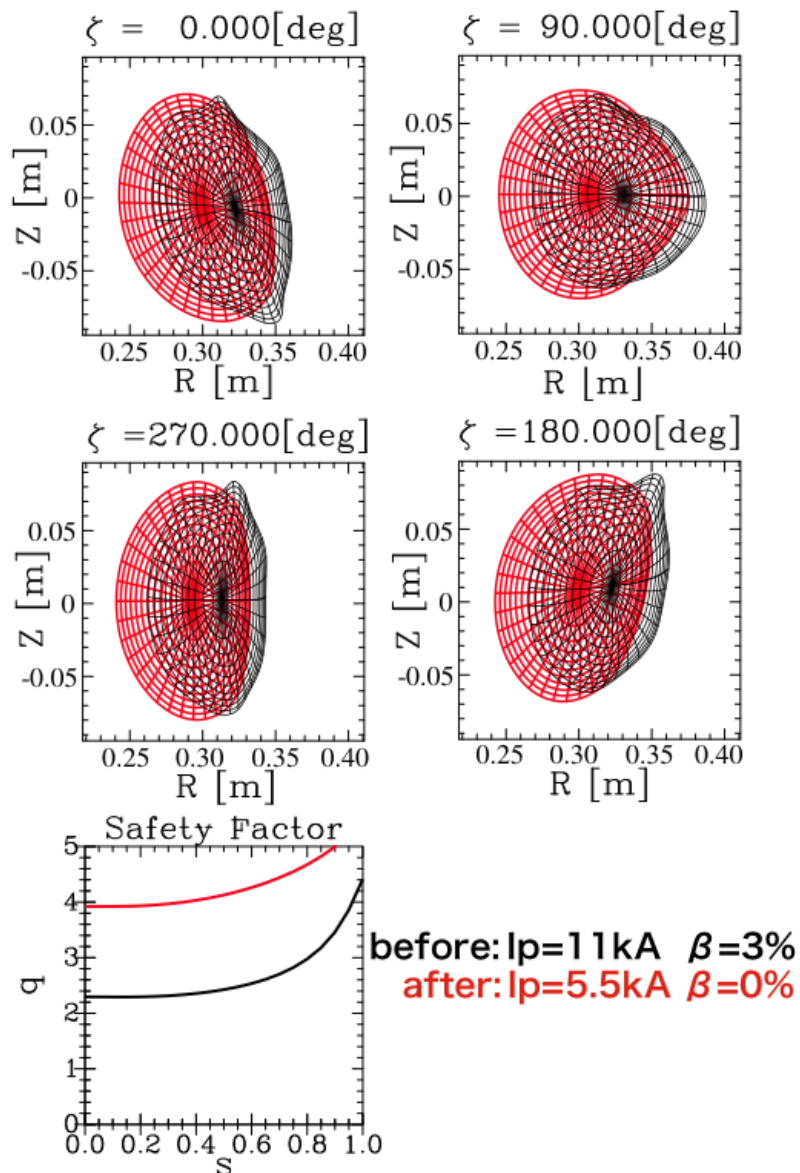


Figure 4: Comparison of equilibrium configurations with helical fields computed with VMEC

References

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