Optimization of the Poloidal Field Coil System by Using an Integrated Design Code for Tokamak Fusion Reactors*)

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A platform for an integrated design code has been developed in which the detailed analysis codes for many components of a nuclear fusion reactor are compiled and incorporated into the system analysis code. Here we have introduced the lasma equilibrium code into the integrated design code in order to optimize the poloidal coil system. This makes it possible to evaluate the capability of the flux supply for ramping up the plasma current and the controllability of an elongated plasma owing to vertical stability. We calculated the margin for vertical stability for various positions of a passive conductor.

Keywords: integrated design code, automatic optimization, plasma equilibrium, vertical instability, passive conductor

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1. Introduction

To design nuclear fusion reactors, we must consider various physical and engineering parameters. Thus, the identification of critical parameters in fusion reactor design is a very important role for researchers. To design a fusion reactor, we use a system code to find the critical parameters. The system code, which includes plasma physics, reactor engineering, and cost estimation, quickly yields a rough one-dimensional (1-D) radial build for a fusion reactor. R. Hiwatari at CRIEPI analyzed many parameters with the system code FUSAC [1, 2], and identified the design points. The ARIES team did similar analysis with its system code [3]. Using the results of the system code, we conduct a detailed design for each component by using sophisticated codes.

We are developing an integrated design code that includes a system code and detailed design codes. The system code shows only a design window with a simplified model, whereas the integrated design code makes it possible to conduct detailed design research for each component of a fusion reactor in one run. This integrated design code consists of a platform based on a system code and detailed design code modules, as shown in Fig. 1 [4], and various types of detailed design code are incorporated into the platform. We have employed the system code FUSAC as a platform.

In this research, we introduced the two-dimensional (2-D) plasma equilibrium code TOSCA [5] into the platform to analyze the optimization of the poloidal field (PF) coil positions. In addition, this 2-D magnetohydrodynamic

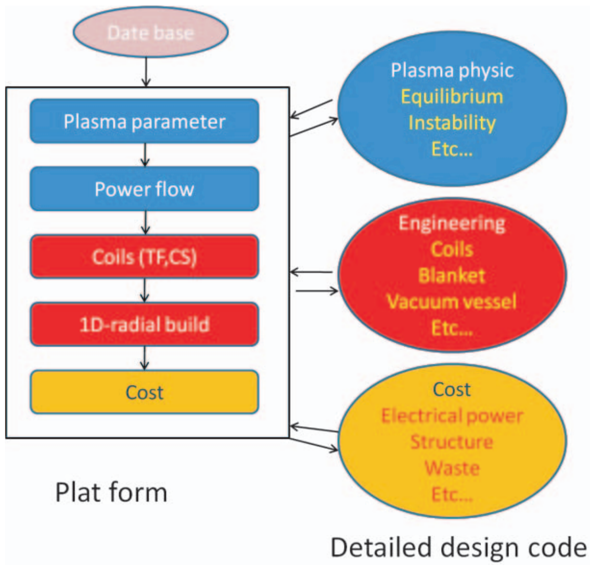

Fig. 1 Integrated design code concept. The system code FUSAC is employed as the platform, and various detailed design codes are integrated as modules.

(MHD) equilibrium analysis code makes it possible to evaluate the margin for the vertical stability of elongated plasmas because the effect of eddy currents due to passive conductors can also be calculated in this code. This new integrated design code takes the 1-D radial-build parameters from the platform to the plasma equilibrium module automatically and optimizes the position of the PF coil in one run. Individual researchers can choose the evaluation function used to determine the optimized position. This equilibrium code also enables us to analyze the plasma's vertical instability and the stabilizing effect of passive con-

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ductor walls.

The integrated design code concept is discussed in Section 2. Section 3 introduces the 2-D equilibrium code. The PF coil position optimization is described in Section 4, and the vertical instability is analyzed in Section 5.

2. Integrated Design Code

The platform of the integrated design code is based on the system code FUSAC, which provides a 1-D radial build of a tokamak fusion reactor, in addition to the overall machine parameters and so on. FUSAC, written in FORTRAN 77, is converted into FORTRAN 90/95, and each subroutine is converted into a module. New detailed design modules can easily be added to the platform, so it can reflect the latest research results. The concept of the integrated design code is shown in Fig. 1. The platform and the detailed design code modules consist of three blocks: plasma physics, engineering, and cost.

The platform first obtains the input data and some physical and mathematical data. Second, it calculates plasma parameters such as fusion power, plasma current, power flow, and neutron flux. After calculating the plasma parameters, engineering parameters such as the toroidal field (TF) coil, central solenoid (CS) coil, backing cylinder, and blanket are calculated. Finally, a self-consistent parameter set is obtained for a reactor design. Since each output parameter is applied to the detailed design code modules automatically, a slight change in the input parameters can easily be reflected in the overall design of the fusion reactor.

Since it is quite important that the detailed design code modules can be easily integrated into the platform, we first established the guiding principle for the architecture of this platform, as described elsewhere [4]. Therefore, the latest research can be incorporated into the integrated design code at any time.

3. Introduction of 2-D Equilibrium Code

To connect the 2-D equilibrium code to the platform, some extension of the 1-D radial build obtained by the system code should be considered. Here, using the design of ITER PF coils as an example, the following guidelines are introduced.

(1) The PF coil system consists of a CS coil and three pairs of PF coils.
(2) The divertor coil, indicated by Nos. 7 and 12 in Fig. 2, is located near the smaller major radius to realize high triangularity. The position must be fixed in order to produce null points.
(3) The vertical field coil, indicated by Nos. 9 and 10 in Fig. 2, is located in the outer region with a sufficiently wide opening for access to the external heating and current drive. The maintenance scenario would also determine this opening width.
(4) The position of the third PF coil, shown by Nos. 8 and 11 in Fig. 2, is adopted to optimize the PF coil system. (The position is determined by the other four coils.)

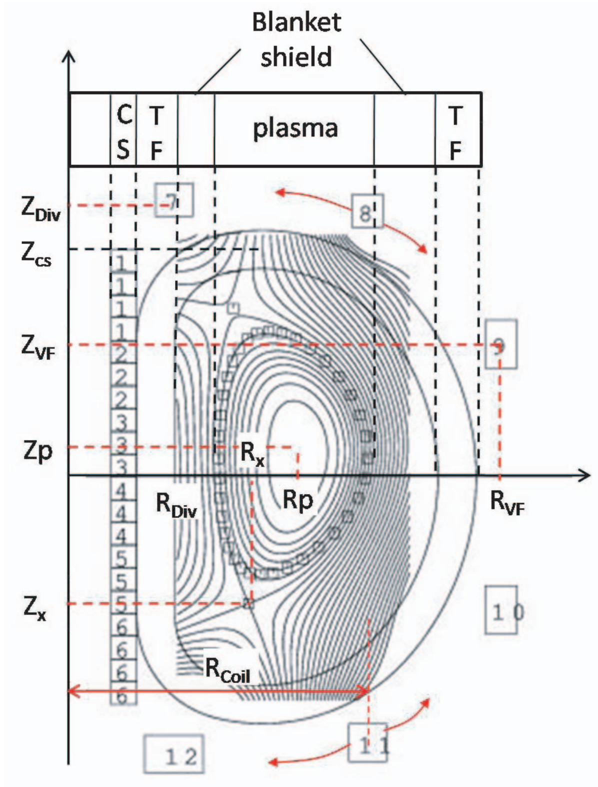

Fig. 2 Model extension from 1-D to 2-D. Model is based on ITER and other tokamak reactors. Small squares represent the plasma boundary and null points.

The positions of the CS coils, divertor coil, and vertical field coil are given by the following simple formulae and are roughly consistent with those of the ITER PF coils [6]. In other design concepts, they can be changed.

CS coil

$$Z_{CS} = Z_{TF},$$

Divertor coil

$$R_{Div} = R_{TF\text{-in}} + \Delta_{TF}/2, Z_{Div} = 1.05Z_{TF} + \Delta_{TF},$$

Vertical field coil

$$R_{TF} = R_{TF\text{-out}} + \Delta_{TF}, Z_{TF} = \kappa a_p,$$

where Z_{TF}, $R_{TF\text{-in}}$, $R_{TF\text{-out}}$, Δ_{TF}, κ and a_p are the height of the TF coil, the major radii of the inner and outer TF legs, the width of the TF coil, the plasma elongation, and the plasma minor radius, respectively.

Since we have considered a single null configuration with a separatrix in the lower region, as shown in Fig. 2, it is necessary to determine the positions of the plasma center and the separatrix automatically. Referring to the design of

ITER and other tokamak reactors, we assume these positions as

Plasma center

$$Z_p = 0.25a_p,$$

Separatrix

$$R_X = R_p - 0.6a_p, Z_p = -\kappa a_p,$$

where R_p is the major radius of the plasma.

4. Optimization of PF Coil Position

As discussed in the previous section, the positions of the CS coil, divertor coil, and vertical field coil are determined automatically, whereas the position of the third coil, located between the divertor and the vertical field coils, is used to optimize the PF coil system. By changing the position of the third coil, i.e., Nos. 8 and 11 in Fig. 2, the plasma equilibrium is calculated. For example, in the case of Fig. 2, coils No. 8 and 11 are moved from $R_{coil} = 4.5$ m to $R_{coil} = 11$ m (where R_{coil} is the radial position of the coils), following the TF coil shape.

Figure 3 shows the calculation results for different positions of the third coil in the steady state. As the third coil, No. 11, moves to a region with a smaller major radius, the divertor coil current, No. 12, clearly increases, accompanied by a decrease in the triangularity. This is caused by cancellation of the coil current between coils No. 11 and 12 because the coil currents between these two coils are opposite in sign. The same phenomenon occurs with the No. 11 coil located near the vertical field coil No. 10. Therefore, the optimized position seems to be located at R_{coil} = 8-9 m.

In addition to optimization of the PF coil current, the flexibility of the plasma shaping should be considered. For example, high triangularity, in addition to plasma elongation, is preferable for high-performance plasmas. As shown in Fig. 3, the triangularity has a broad optimum around $R_{coil} = 9$ m, and these calculations can provide the information needed to determine the PF coil position, such as the coil current and plasma triangularity. By introducing appropriate evaluation functions such as the total stored energy of the PF coil system, researchers can determine the PF coil system.

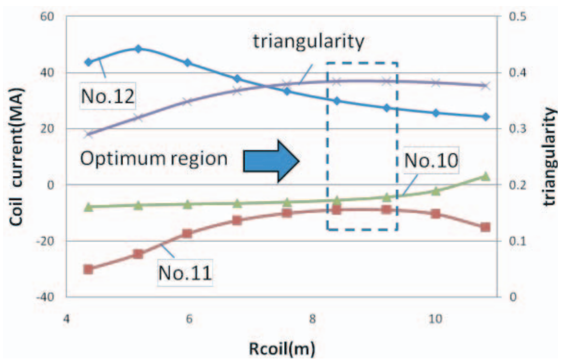

Fig. 3 Coil current as a function of R_{coil}. Green, red, and blue lines represent the currents of the Nos. 10, 11, and 12 coils, respectively; purple line denotes plasma triangularity.

5. Vertical Instability

The vertical stability of an elongated plasma is quite important in the design of a tokamak fusion reactor. A blanket and/or vacuum vessel is sometimes assumed for the stabilizing shell, and a passive conductor is introduced in the inner region of the TF coil. Here, we evaluate the stabilizing effect of these passive conductors by using the equilibrium code, in which the time-dependent circuit equation of the plasma column, PF coils, and structures is solved. This equation is given as

$$\frac{\partial}{\partial t}\left(\sum_{j=1}^{N} M_{ij} I_j\right) + R_i I_i = V_i \quad (i = 1, 2, \cdots N),$$

where M_{ij}, I_i, R_i, and V_i are the inductance matrix, current, resistance, and voltage, respectively, at each component. By using this circuit equation, we can evaluate the instability of toroidally symmetric movement of the plasma column, i.e., the vertical stability. Since the vertical stability is governed by the curvature of the magnetic field produced by the external current, the decay index n [7] is introduced as;

$$n = -\frac{R}{B_Z}\frac{\partial B_{Z(EQ)}}{\partial R},$$

$$n_s = -\frac{R}{B_Z}\frac{\partial B_{Z(shell)}}{\partial R},$$

where n, n_s, $B_{Z(EQ)}$, and $B_{Z(Shell)}$ are the decay index of the equilibrium field, shell stabilizing index due to the passive conductor, vertical fields generated by the PF coils, and vertical field generated by the eddy current of the passive conductor, respectively [8, 9].

The decay index n of the external magnetic field is negative, so a vertically elongated plasma always has vertical instability if no passive conductors are used. When the plasma moves up or down, an eddy current is driven in the passive conductors. Then, the eddy current generates a magnetic field, and the magnetic field pushes the plasma back to the original position. This stabilizing effect is calculated as n_s. When the shell stabilizing index is bigger than the decay index, the plasma is stable against vertical motion. In other words, when $n_s + n > 0$, the plasma is stable.

Here we consider a simple model of a passive conductor wall. We place passive conductors in the upper and lower regions of the plasma column, as shown in Fig. 4.

The stabilization effect of the passive conductors also depends on the growth rate of the vertical stability. First, we considered the case of an infinite growth rate because it is the worst condition.

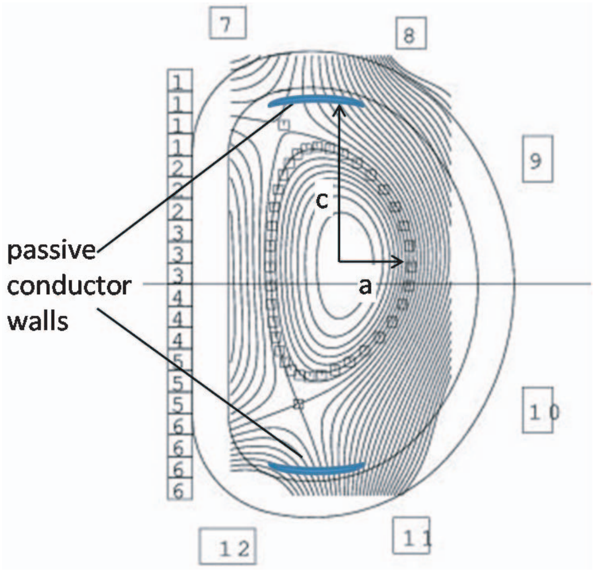

Fig. 4 Passive conductor model. Two passive conductor walls are placed in up-down symmetry. Small squares represent the plasma boundary and null points.

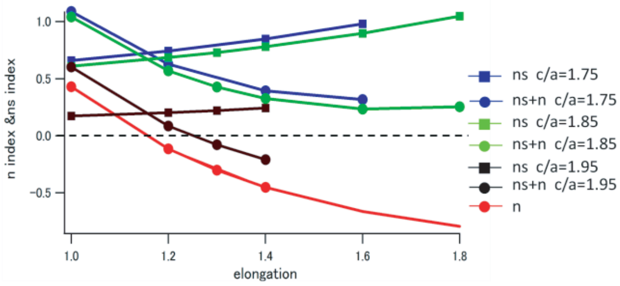

Fig. 5 Indexes n and n_s as a function of elongation. Instability increases as elongation increases.

Figure 5 shows the calculated n and n_s values as a function of the plasma elongation for various positions of the passive conductors, where c is the distance between the plasma center and the upper passive conductor, as shown in Fig. 4. The cross-sectional area of each conductor is $0.5\,\mathrm{m}^2$, and the resistivity is $1.0 \times 10^{-6}\,\Omega{\cdot}\mathrm{m}$. The decay index n depends only on the elongation, and the destabilization becomes larger as the elongation increases, as shown in Fig. 5. Although the shell stabilizing index n_s depends on both the elongation and c/a. the n_s value clearly increases as the elongation increases and decreases as c/a increases. This shows that the stabilizing effect increase as the length between the passive coil and the plasma decreases.

Figure 5 illustrates that the stabilizing effect is near zero when c/a is bigger than 1.85. This shows that when the distance from the passive coil to the plasma is too large, the stabilizing effect is lost.

Next, we considered a finite growth rate. When the growth time of the instability is near the time constant

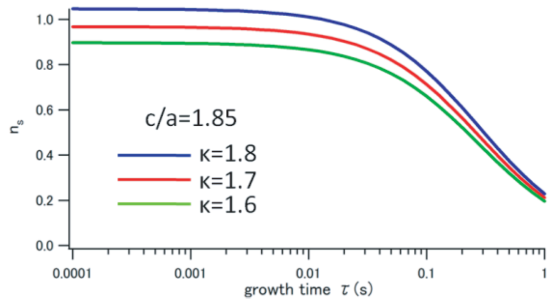

Fig. 6 The n_s index as a function of τ (the growth time of the vertical instability).

(L/R) of the shell, the eddy current decreases, and n_s decreases. The relationship between the value of n_s and the growth time of the instability is shown in Fig. 6.

The value of n_s decreases quickly at a growth time of about 10^{-2} sec. When the growth time is greater than 10^{-2} sec, the stabilization effect decreases, and the plasma becomes unstable. A feedback control system using an external magnetic field should be designed using this time constant.

In this paper, we demonstrated a guideline for integrated design code. Researchers can change the position or size of the passive conductor walls and perform a more detailed analysis with this code.

6. Summary

We are developing an integrated design code for fusion reactors; to date, we have examined the following items.

(1) We have established the platform for the integrated design code, which is based on the system code FUSAC. Various detailed design codes can be integrated into this platform very easily.

(2) We have incorporated the 2-D plasma equilibrium code into the platform. On the basis of the 1-D radial build, the 2-D coil position is automatically generated, and optimization is performed.

(3) We have analyzed the vertical instability and the stabilization effect of the passive conductor coils, so the limitations of plasma elongation and other plasma shaping parameters can be discussed.

[1] R. Hiwatari *et al.*, Nucl. Fusion **44**, 106 (2004).
[2] R. Hiwatari *et al.*, Nucl. Fusion **45**, 96 (2005).
[3] Z. Dragojlovic *et al.*, Fusion Eng. Des. **85**, 243 (2010).
[4] M. Nakamura *et al.*, J. Plasma Fusion Res. SERIES **9**, 186 (2010).
[5] H. Fujieda *et al.*, JAERI-M 08-256 (in Japanese) (1996).
[6] Summary of the ITER Final Design Report (2001).
[7] A. Fukuyama *et al.*, JJAP **14**, 871 (1975).
[8] S. Seki *et al.*, JAERI-M 83-165 (in Japanese) (1983).
[9] G. Kurita *et al.*, Fusion Eng. Des. **38**, 417 (1998).