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Power plant performance optimization of tokamak fusion reactor with design codes

設計コードによるトカマク型核融合炉の発電プラント性能における最適化

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In Tokamak fusion reactor, plasma and coil performance is in a very close relationship. So far, we have developed design code that can analyze the amount of power in self-consistent parameters. The code calculates plasma physics performance with the toroidal field coils from the radial build on the basis of engineering constraints. Then, analysis was performed with a particular focus on the net electric power at the transmission end, and we have considered an optimal design window for DEMO plants.

1. Introduction

As a next step of the experimental reactor (ITER), the DEMO plant is planned to validate the integrated plant-system to generate net electric power. In this study, possible design window for the DEMO plant has been investigated using a design code based on the expected performance in plasma physics with a magnetic field that is computed by the conditions on the coil engineering expected in near future.

2. Research methods

In the plasma physics code developed by DRIVER of CRIEPI [1], the plasma performance is determined self-consistently based on current distribution, plasma density etc., by using an automatic external current and power source, such as NBI current drive. The current distribution is adjusted to the specified current profile. In the DRIVER code, this profile can be determined by MHD equilibrium / stability analysis, but in this study, the profiles have been given as the input.

The shape of the toroidal field coils is determined under the same condition as the parameter values used in the calculation. Then the magnetic field at the major radius is determined under some engineering constraints based on their shapes [2,3]. Self-consistent analysis of power output from the coil have been done to unify and integrate the code parameters as described above.

3. Design code

The diagram of design code is shown in Fig.1. This code is designed to obtain the self-consistent parameters of the plasma, where the density, temperature and current profile are given as the inputs. The coil design is used to calculate various parameters from plasma physics, and the calculation of the coil dimensions is achieved automatically. It is also possible to estimate the Volt-Sec capacity of central solenoid (CS) and the possible period for inductive (pulsed) operation if the CS capacity exceeds the minimum requirement for initial current ramp-up.

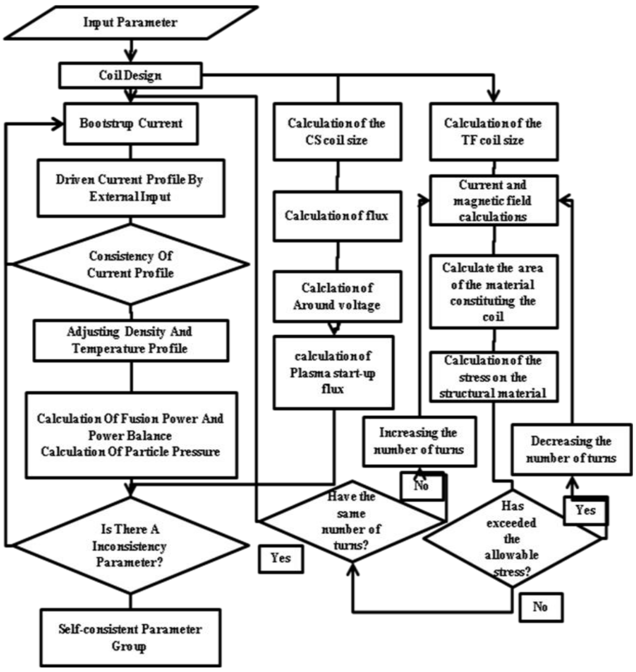

Fig.1. diagram of design code

4. Result

The parameters used here for the TF coil design are equivalent to the SlimCS design by JAEA [4], shown Table I. The size and aspect ratio of a fusion reactor have been scanned and the relationship between the magnitude of the output are obtained (Fig.2). In this calculation, the plasma densities are controlled automatically to keep the conditions $\beta_N < 0.75\beta_{Nmax}$ and $n_e/n_{GW} \leq 1$, where a model of β_{Nmax} formulated in [5] has been adopted. In this formula, higher β_{Nmax} can be expected with lower aspect ratio, as well as with higher elongation. The safety factor q_Ψ is fixed to the specified values in each calculation (3.0 ~ 5.0). The relationship between the major radius and center of the magnetic field strength is shown in Fig.2. With a larger major radius, the maximum magnetic field on coil becomes smaller, and the difference between the maximum magnetic field and the center becomes smaller. Therefore the curves of the magnetic field strength on the plasma center have peaks, as shown in Fig.2.

Table I. Design parameters

Plasma parameters	
Major radius (m)	4.5〜9
A	2〜4
κ_{95}	Determined by the aspect ratio
B_T (T)	Determined by the coil Design
δ_{95}	0.35
Ti(keV)	14,16,18
n_e	Determined by β_N and n_e/n_{GW}
β_N	$0.75\beta_N$limit
I_p (MA)	Determined by $q\Psi$
η_{TD} (%)	35
η_{NBI} (%)	60
TF Coil design criteria	
Super-conductivity material	Nb3Al
Operating temperature (K)	5
Allowable temperature (K)	250
Voltage between terminals(KV)	20
Allowable stress(MPa)	800
Number of coils	15
Thickness of the coil case (m)	0.1
Coil body width (m)	1
Coil body width (m)	1.4
Port tolerance	2
Operating current(kA)	100
Quench delay time (s)	0.5

The relationship between the net electric output and the major radius is shown in Fig.3, where T_i=16keV and q_Ψ=4 have been assumed and aspect ratio of Tokamak, A, has been scanned from 2.0 to 4.0. With A=2, external power is too much and net electric output goes negative value. In the lower yellow area, $H_{H98(y,2)}$ exceeds 1.3. In the upper pink area, the wall load exceeds 3MW. In the left blue area, the capacity of CS coils is less than L_pI_p, and therefore non-inductive current ramp-up (NICR) will be necessary. This NICR will be possible, but no doubt more difficult than the inductive ramp-up. In the right hand side of figure (i.e., larger major radius), the weight of coils naturally increases. Considering those restrictions, reasonable design points will exist in the central area.

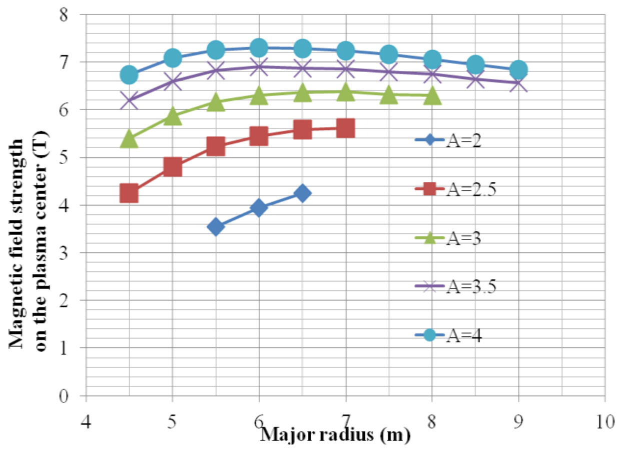

Fig.2. major radius and magnetic field strength on the plasma center

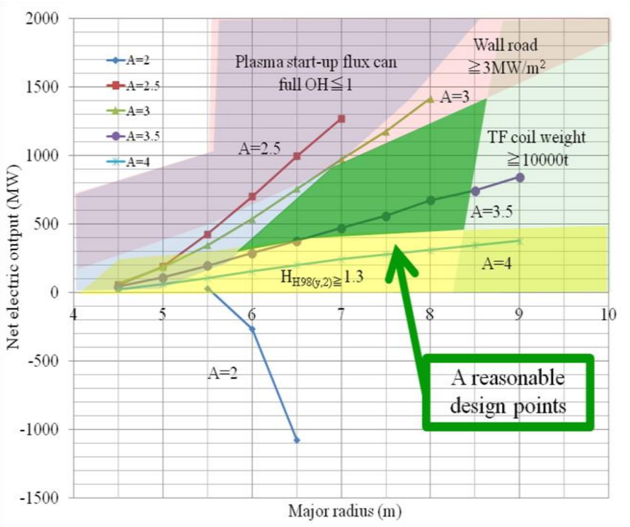

Fig.3. Major radius and net electric output

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