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Control of Fusion Power in a Steady-state Tokamak Reactor

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Abstract

Possibility to control the fusion power in a steady-state tokamak reactor has been studied. Analysis by using 1D self-consistent current drive code (ACCOME) and 1.5D time-dependent transport code (TOPICS) indicates that the operational parameter for controlling the fusion power can be found under the operation constraints such as a fixed confinement improvement factor, the β-limit, the density limit and the self-consistent current drive efficiency. A characteristic time constant for fusion power control depends on a time scale of the current ramp up/down which relates to a capability of the poloidal field coil system. Key issues to be solved to realize the fusion power control are discussed.

Keywords:

fusion power control, steady-state tokamak reactor, A-SSTR, ACCOME code, TOPICS code

1. Introduction

In Japan, the peak daily electric power demand is almost twice of its minimum. For this large change, the power plant based on fossil fuel plays a key role to supply the varying power flexibly. The nuclear power plant supplies the base power because of its large capital cost, although the power control is not impossible in nuclear plants. From a point view of the safety, frequent change in the thermal power is not desirable in nuclear plants. In future where the exhaustion of fossil fuel becomes a realistic problem and/or the discharge of CO_2 is strictly controlled, a new way for matching a large change in the electricity demand should be established. As a future energy source with a reasonable cost and high safety, the fusion power plant must show, at least, that the fusion power control is not impossible. This study is made with focussing this point.

For a fusion power reactor based on a self-ignited plasma, the control of fusion power has been thought to be uneasy, and its controllable range by changing the DT mixture rate and the impurity content is not so wide for a practical use of power control. In a steady-state tokamak reactor, the auxiliary current drive (CD) power is always required to sustain the discharge non-inductively, and results in a finite Q (= fusion power/ auxiliary power) operation. In this case, one can adjust the auxiliary CD power, the density and the operating plasma current for fusion power control.

2. Operation Parameter for Power Control

Constraints to find out the operation parameter for fusion power control are 1) fully non-inductive plasma with a self-consistent current drive efficiency, 2) confinement, 3) stability and 4) operation limit such as the maximum power and current. In this paper, the study has been made for the Advanced Steady-State Tokamak commercial Reactor A-SSTR [1] (P_f = 4.5 GW, β_N =

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4.5, R_p/a_p = 6.0/1.5 m, I_p = 12 MA, B_{TO} = 11 T) where more than 80% of the plasma current is driven by the bootstrap current and the 60 MW of N-NBI (1.5 MeV) provides the remained current. Since the non-inductive current drive with high bootstrap current is one of the important issues in steady-state operation, 1D self-consistent current drive code (ACCOME [2]) is used to find out the operation point for fusion power control.

Figure 1 shows various parameters against the plasma current in A-SSTR where the parabolic density and temperature profiles are assumed. At $I_p \geq 8$ MA, the neutral beam was injected with a power of 60 MW and a tangential radius of 5.5 m, while the beam power was increased linearly with the plasma current for $I_p < 8$ MA; P_{NB}(MW) = 50 I_p (MA)/8 + 10. The beam energy was set to be 1.5 MeV at $I_p \geq 4$ MA and E_B = 0.75 MeV at $I_p < 4$ MA. The electron density was selected so that the energy confinement improvement factor becomes $H_H = \tau_E/\tau_E^{ELMy} = 0.8 - 1.2$ where τ_E^{ELMy} is the ELMy H-mode confinement scaling [3]. Since A-SSTR was designed to minimize the cost of electricity production, aggressive assumptions such as β_N = 4.5 and n_e/n_{GW} = 1.2 are assumed at a full power operation. Figure 1 shows that operation parameters to change the fusion power can be found in a steady-state tokamak reactor at roughly constant confinement improvement factor. Since the CD power and the density are relatively easy to control in a steady-state tokamak reactor, the control of the bootstrap current may be the most important.

Although ACCOME code gives a steady-state solution of current profile at given density and temperature profiles, any dynamic effects are not considered. In order to study dynamic behavior during fusion power control in A-SSTR, numerical simulation by using 1.5 D time-dependent transport code (TOPICS [4]) has been made. Following transport model is employed:

$$\chi_{e/i} = \chi_{e/i}^{\mathrm{ano}} + \chi_{i}^{\mathrm{NG}}$$

$$\chi_{e/i}^{\mathrm{ano}} = C_{e/i1}(1 + C_{e/i2}\rho^2)a^2/\tau_{E}^{\mathrm{ELMy}}$$

$$D = C_d(\chi_{e}^{\mathrm{ano}} + 0.1\,\chi_{i}^{\mathrm{NG}})$$

where $\chi_{e/i}$ and χ_i^{NC} are the electron/ion thermal diffusivity and the ion neo-classical diffusivity, and a is the minor radius. Constants $C_{e/i1}$, $C_{e/i2}$ and C_d are parameters to fit τ_E to the scaling, to produce an appropriate temperature profile, and to make n_{He}/n_e = 8%, respectively.

Figure 2 shows results of TOPICS code simulation. The fusion power can be reduced from 3.8 GW to 1.3 GW for 200 s by changing the plasma current from 12 MA to 9 MA, the density from 2.1 × 10^{20} m^{-3} to 1.3 × 10^{20} m^{-3} and the current drive power from 60 MW to 45 MW, where $\tau_E/\tau_E^{ELMy} \sim 1.0$, and $I_{NB} + I_{BS} > I_p$. The simulation was made by assuming an enough capability of the primary winding circuit to response the change in the inductive flux in order to realize a quick change in the plasma current. Right-hand side in Fig. 2 shows the

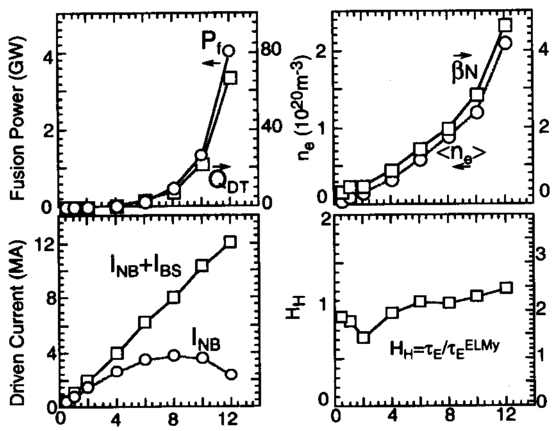

Fig. 1 Operation parameters for fusion power control evaluated by ACCOME code for A-SSTR.

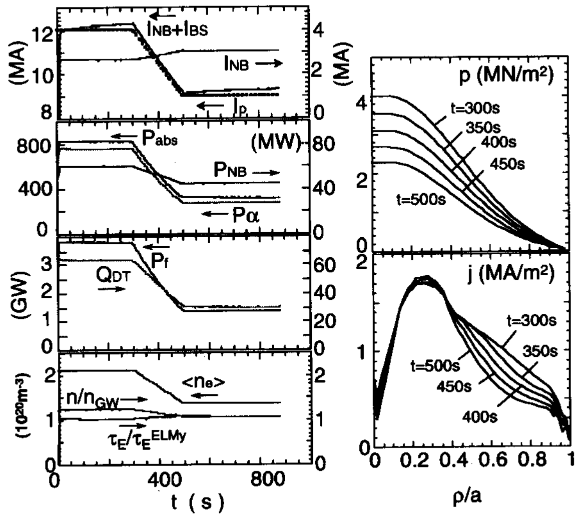

Fig. 2 Transport simulation of fusion power control in A-SSTR evaluated by TOPICS code. Left-hand side: time evolution of non-inductive current, heating powers, fusion power and Q_{DT}, density and H-factor. Right-hand side: Change in the total pressure and total current density during t = 300–500 s.

change in pressure and the profile from 300 s to 500 s. Decrease in plasma current in this case comes from mainly a reduction of the edge bootstrap current according to the decrease in the plasma stored energy, and the core current density does not change in this scenario.

Figure 3 shows similar simulation result for the case to increase fusion power from almost zero output, which corresponds to the ACCOME analysis shown in Fig. 1. The characteristic time of fusion power control in this simulation depends on the current ramp up time, which is determined by the poloidal field coil system. In this simulation, the plasma current is increased from 4 MA to 12 MA during time interval of 600 s with assuming the inductive flux supply from the poloidal coil system. Without a flux supply from the poloidal coil system, the current ramp-up time is estimated by $T = \tau_{L/R} \cdot ln(I_p/I_{p0})/(I_{CD}/I_p - 1)$ in the case of constant value of I_{CD}/I_p where I_{p0} is the initial plasma current, $\tau_{L/R} = L/R$, L is the external inductance and R is the resistance of the plasma. For a typical A-SSTR parameter, $\tau_{L/R}$ becomes ~ 1.3 h. If we drive 10% larger non-inductive current (I_{CD}/I_p = 1.1), about 4 hours is required to increase the plasma current from 9 MA to 12 MA (see Fig. 2) without the external flux. This time scale is comparable with the time scale of the change in the daily power demand. Faster current ramp up/down can be expected within the acceptable AC loss on the superconducting coils if the poloidal coil system has an enough capability. When the flux of ~20 Vs is supplied from the poloidal coil system with I_{CD}/I_p = 1.1, about 2 hours is enough to increase the plasma current from 9 MA to 12 MA.

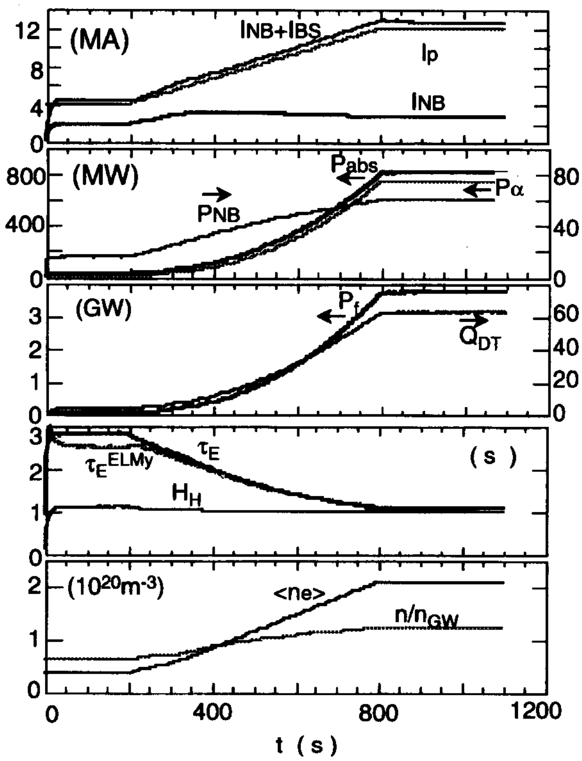

Fig. 3 Fusion power control from almost zero power (TOPICS simulation).

3. Issues to be Solved

The previous section showed that a wide range of the fusion power control is not impossible in a steady-state tokamak reactor. However several important problems should be solved to make the fusion power control more realistic.

3.1 Impact on the design of blanket and energy exchange system

A wide range of the fusion power control means a large change in the thermal output power inside the blanket system. This may affect strongly on the design of the blanket system and the energy exchange system. Careful design of energy exchange system may be required to optimize the energy exchange efficiency through a wide range of thermal output. An essential problem is an impact on the safety of the fusion power plant. Frequent and wide change in the thermal output power may increase a possibility to cause thermal stress on the first wall, the structural material and cooling pipes inside the blanket. Although slow change in fusion power and very careful control of the heat removal system might contribute to keep the same temperature profile inside the blanket, an increase in a possibility of loss of coolant accident due to thermal stress can not be avoided. Detail and precise studies on this point should be made.

3.2 MHD stability and linkage between transport and profile

Since many key parameters including the current and pressure profiles seem to change drastically during a fusion power control, a MHD stability must be one of key physics issues. Figure 4 shows the growth rate of n = 1 ideal-kink mode with and without ideal wall in the case of Fig. 2. With the ideal wall at ρ/a = 1.4, n = 1 mode is stable at t = 300 s. The growth rate decreases with decreasing the fusion power and the plasma current without the ideal wall. Although significant change in

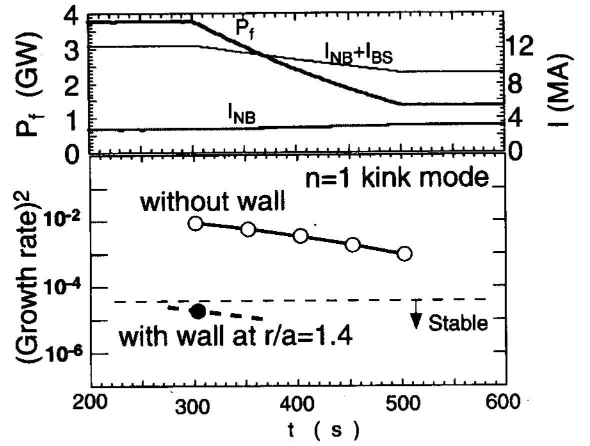

Fig. 4 Time evolution of growth rate of n = 1 ideal kink mode for the case of Fig. 2.

the eigen-function of the mode at t = 300 – 500 s, the kink mode stability is improved with the peaked current profile at low fusion power regime. This analysis suggests that stabilization of low n kink modes by the wall stabilization effect can be expected in the case of Fig. 2. The ballooning modes is stable at t = 300 s and marginally stable at t = 500 s for the profiles in Fig. 2. Profile optimization for the ballooning mode is still open question in the present study.

In the simulation showed in Sec. 2, the transport coefficient is assumed independently of the current profile and MHD activities. However, we can find several experimental results which suggest a correlation between transport and current profile/MHD activities. Figure 5 shows results in JT-60U where the current in the OH primary winding keeps constant from $t > 6$ s, 4.5 MW of NBCD power and 18 MW of NB heating power were injected at t = 6 – 9 s with beam energy of 90 keV. Rapid increase in the plasma current at $t \sim 6$ s is mainly due to the volt-second supply from the vertical field coil corresponding to the increase in β_p. The ion internal transport barrier (ITB) in $r \sim 0.5$ m appears at t = 6.8 s as shown in right-hand side in Fig. 5. From $t \sim 7$ s, ITB starts to shift outward. The plasma current increases rapidly again due to a rapid increase in β_p accompanied by the shift of ITB. At $t \sim 7.4$ s, β_p decreases rapidly after a mini-collapse and closes to a constant value. This result indicates that the plasma current, which is one of the most important factors to the confinement/transport, can be easily affected by the transport itself without strong feedback control. This situation becomes more complicated in the case of strong reversed shear plasma where a steep ITB appears near the pitch minimum

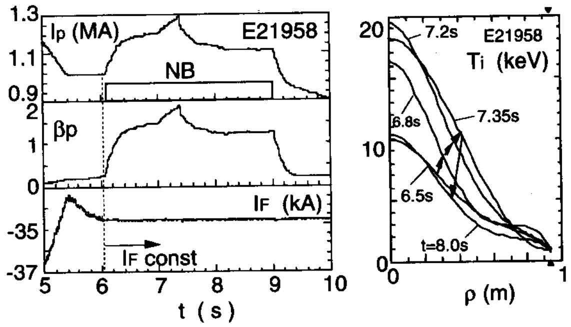

Fig. 5 OH-free non-inductive current drive experiment in JT-60U.

location [5]. The location of ITB will be easily shifted by ITB itself because a large bootstrap current due to a steep ITB modifies the current profile. Transport simulation taking into account a linkage between transport coefficients and current profile in reversed shear configuration has indicated necessity of the plasma current feedback for stable operation [6]. In addition, control of the ITB location and the pressure gradient will be also required.

4. Summary

This paper is summarized as follows:

1. Operation parameter for fusion power control can be found in a steady-state tokamak reactor under given operation constraints.
2. The characteristic time constant of the fusion power control depends on the characteristic time of current ramp. Expected time to change the fusion power without a large flux supply from the poloidal coil system is acceptable from a point view of the daily power demand.
3. Remained issues on engineering design and physics study are emphasized.

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