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Direct extrapolation of radial profile data to a self-ignited fusion reactor based on the gyro-Bohm model

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ABSTRACT

A new method named direct profile extrapolation (DPE) has been developed to estimate the radial profiles of temperature and density in a fusion reactor. This method directly extrapolates the radial profiles observed in present experiments to the fusion reactor condition assuming gyro-Bohm type parameter dependence. The magnetohydrodynamic equilibrium that fits the experimental profile data is used to determine the plasma volume. Four enhancement factors for the magnetic field strength, the density, the plasma beta, and the energy confinement are assumed. Then, the plasma size is determined so as to fulfill the power balance in the reactor plasma. The plasma performance can be measured by an index, C_{exp}, introduced in the DPE method. The minimum magnetic stored energy of the fusion reactor to achieve self-ignition is shown to be proportional to the cube of C_{exp} and inversely proportional to the square of magnetic field strength. Using this method, the design window of a self-ignited fusion reactor that can be extrapolated from recent experimental results in the Large Helical Device (LHD) is considered. Also discussed is how large an enhancement is needed for the LHD experiment to ensure the helical reactor design of FFHR2m2.

1. Introduction

Fusion energy output is one of the most important parameters for thermonuclear fusion reactors. In designing toroidal magnetic fusion reactors, the fusion energy output is calculated by integrating the radial profile of deuterium–tritium (DT) fusion energy output per unit volume that is determined by the temperature and density profiles inside the plasma torus. The DT fusion energy is released in the form of kinetic energies of alpha particles and neutrons. If the plasma is sustained by the former namely the alpha heating alone, *i.e.*, without auxiliary heating, the plasma is in the self-ignition state.

On the other hand, the temperature in a magnetic fusion reactor is determined by the density and the heating power, together with the magnetic field strength, the size of the plasma, and so on, via the transport physics of heat and particles. In a self-ignited fusion reactor, the temperature and density profiles should be consistent with the alpha heating power that is determined by the temperature and density profiles themselves. These profiles together with the plasma current density profile consisting of both driven and self-induced components should be also consistent with the magnetohydrodynamic (MHD) equilibrium. Plasma volumes between adjoining magnetic flux surfaces inside the plasma torus, which are used in the volume-integration to estimate the fusion energy output, are determined by this MHD equilibrium.

In designing fusion reactors [1–5], the temperature and density profiles have been artificially determined by designers. In spite of the strict constraints enforced by MHD equilibrium and transport physics, there still is a large degree of freedom in determining the profiles. This causes a large ambiguity in fusion power estimation. Furthermore, the physics of heat and particle transport dominated by anomalous transport driven by micro-instabilities is not yet fully understood. Even though the models of local transport coefficients governed by anomalous transport have been intensively studied [6–8], it is still difficult to predict entire radial profiles of temperature and density in real situations.

In the new method described in this paper named direct profile extrapolation (DPE), the temperature and density profiles observed in experiment are directly used instead of the assumed profiles. The MHD equilibrium that fits the observed profiles is also used self-similarly. Then, the degree of freedom in determining profiles is largely reduced and therefore the ambiguity in fusion power estimation becomes smaller. The observed profiles are extrapolated to the reactor condition by using a gyro-Bohm type parameter dependence [9,10]. As is widely known, the global energy confinement times in toroidal magnetic confinement systems often show a gyro-

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Bohm type parameter dependence [11]. For example, empirical scalings of the energy confinement time in tokomaks, *e.g.*, H-mode scalings [12], and helical plasmas, *e.g.*, international stellarator scalings (ISS95 [13] and ISS04 [14]), show a gyro-Bohm type parameter dependence. Especially in LHD [15], which is the world's largest heliotron device equipped with superconducting magnetic coils, the gyro-Bohm type parameter dependence is recognized not only in the global energy confinement time, but also in the local relation between the temperature and the density [16,17].

This paper describes the basis of the DPE method and how to apply it in fusion reactor design. Details of the DPE method are described in the next section. An example of extrapolation from LHD to a self-ignited helical reactor is shown in Section 3, where the experimental setup of LHD is also given. A design window for the self-ignited helical fusion reactor, which is extrapolated from experimental results in LHD, is compared with the design of FFHR2m2 in Section 4, where FFHR2m2 is an LHD-type helical fusion reactor characterized by a long-life blanket using molten-salt for tritium breeding [1]. An index, $C_{\rm exp}$, which expresses the accessibility of the experimental data to the reactor, is also introduced in this section. A discussion on the enhancement factors is given in Section 5. Finally, conclusions are presented in Section 6.

2. Direct profile extrapolation method based on the gyro-Bohm type parameter dependence

At first, let us define an enhancement factor, f_X, as the ratio of a parameter X in a fusion reactor, $X_{\rm reactor}$, to that in the experiment, $X_{\rm exp}$:

$$f_X = \frac{X_{\rm reactor}}{X_{\rm exp}}. \qquad (1)$$

For example, the enhancement factors of the temperature, T, the density, n, the heating power, P, and the magnetic field strength, B, are given as

$$f_T = \frac{T_{\rm reactor}(\rho)}{T_{\rm exp}(\rho)}, \qquad (2)$$

$$f_n = \frac{n_{\rm reactor}(\rho)}{n_{\rm exp}(\rho)}, \qquad (3)$$

$$f_P = \frac{P_{\rm reactor}}{P_{\rm exp}}, \qquad (4)$$

$$f_B = \frac{B_{\rm reactor}}{B_{\rm exp}}, \qquad (5)$$

respectively, where ρ (=r/a) is the normalized minor radius and a is the plasma minor radius. It should be noted that P is treated as a scalar here although it also has a radial profile like T and n [18]. This will be discussed later in Section 5. In this study, it is assumed that both the ion density, n_i, and the electron density, n_e, are equal to n, *i.e.*, the effective charge, $Z_{\rm eff}$, is equal to one, for simplicity. It is also assumed that the ion temperature, T_i, and the electron temperature, T_e, are equal to T. The device size is enlarged self-similarly from the experiment to the reactor, *i.e.*, $f_a = f_R$, and therefore:

$$f_{a/R} = 1, \qquad (6)$$

where R is the plasma major radius. From the definition of plasma beta, $\beta \propto nT/B^2$, f_β has a form of

$$f_\beta = f_n f_T f_B^{-2}, \qquad (7)$$

and therefore:

$$f_T = f_\beta f_n^{-1} f_B^2. \qquad (8)$$

Using this, we can calculate the radial profile of alpha heating power per unit volume, $P'_\alpha(\rho) = dP_\alpha(\rho)/dV = Q_\alpha (n/2)^2 \langle\sigma V\rangle_{\rm DT}$, where Q_α = 3.52 MeV and the ratio of D:T is assumed to be 50:50, as long as $T_{\rm exp}(\rho)$, $n_{\rm exp}(\rho)$, f_β, f_n, and f_B are given, since the DT fusion reaction rate, $\langle\sigma V\rangle_{\rm DT}$, is a function of T. According to Ref. [19], $\langle\sigma V\rangle_{\rm DT}$ (in m^3/s) is given by

$$\langle\sigma V\rangle_{\rm DT} = \left[\frac{8.09\times10^{10}}{t^{2/3}}\times\exp\left(\frac{-4.524}{t^{1/3}}-\left(\frac{t}{0.12}\right)^2\right)\times(1+0.092t^{1/3}+1.8t^{2/3} +1.16t+10.52t^{4/3}+17.24t^{5/3})+\frac{8.73\times10^{8}}{t^{2/3}}\times\exp\left(\frac{-0.523}{t}\right)\right] \times\frac{10^{-6}}{6.0222\times10^{23}}, \qquad (9)$$

where $t = T$ (in keV)/86.171. Similarly, the radial profile of Bremsstrahrung loss per unit volume of $P'_B(\rho) = dP_B(\rho)/dV$ ($\propto n^2T^{1/2}$ [20]) is also calculated from $T_{\rm exp}(\rho)$ and $n_{\rm exp}(\rho)$, using Eq. (8). Then, the total heating power in the reactor is given by the volume-integration as below:

$$P_{\rm reactor} = f_a^3 f_{a/R}^{-1} f_n^2 \int_0^1 (\zeta P'_\alpha - P'_B)(dV/d\rho)_{\rm exp} d\rho, \qquad (10)$$

where ζ is the deposition ratio of the alpha heating and $(dV/d\rho)_{\rm exp}$ is defined by the MHD equilibrium that fits $T_{\rm exp}(\rho)$ and $n_{\rm exp}(\rho)$. The auxiliary heating, $P_{\rm aux,}$ is not considered here, since our target is to estimate the parameters needed for self-ignition. If it is necessary to drive plasma current externally, or heat the plasma during the start-up phase, $P_{\rm aux}$ should be included in the right side of Eq. (10). In this study, ζ = 1 is assumed for simplicity. Note that the plasma volume, V, in the reactor is $f_V = f_a^2 f_R = f_a^3 f_{a/R}^{-1}$ times larger than that in the experiment.

On the other hand, the energy confinement time based on the gyro-Bohm model is given by

$$\tau_E^{\rm GB} \propto a^{2.4}R^{0.6}B^{0.8}P^{-0.6}n^{0.6}, \qquad (11)$$

[16]. This well approximates the energy confinement times predicted by ISS95 [13] and ISS04 [14] of

$$\tau_E^{\rm ISS95} = 0.079a^{2.21}R^{0.65}B^{0.83}P^{-0.59}\bar{n}_e^{0.51}\iota_{2/3}^{0.4}, \qquad (12)$$

and

$$\tau_E^{\rm ISS04} = 0.134 f_{\rm ren}a^{2.28}R^{0.64}B^{0.84}P^{-0.61}\bar{n}_e^{054}\iota_{2/3}^{0.41}, \qquad (13)$$

respectively, where $\bar{n}_{\rm e}$ is the line-averaged electron density, $\iota_{2/3}$ is the rotational transform ($\iota = \iota/(2\pi) = 1/q$, and q is the safety factor in tokamaks) at ρ = 2/3. In ISS04, a renormalizing factor of $f_{\rm ren}$ is adopted to compose a unified energy confinement scaling of various helical plasmas. The exponents on a, R, B, P, and n in $\tau_E^{\rm GB}$ are similar to those in $\tau_E^{\rm ISS95}$ and $\tau_E^{\rm ISS04}$. The dependence on the rotational transform is not included in $\tau_E^{\rm GB}$. It should be noted that the energy confinement dependence on the rotational transform is not clearly understood in helical plasmas yet. However, this term is negligible in the DPE method, where the MHD equilibrium that determines the rotational transform profile in the reactor is assumed to be identical to that in the experiment. This is valid as long as f_β = 1 and the MHD equilibrium is determined by the external magnetic field and the plasma pressure profile alone. The bootstrap current, which is negligibly small in typical high-density plasmas in LHD, can change the MHD equilibrium as the driven plasma current does. The quantitative estimation of the change in the MHD equilibrium due to the bootstrap current, which will be not negligible in the reactor with low collisionality, is left for future studies.