Assessment of Plasma Performance in a Magnetic Configuration with Reduced Poloidal Coils for a Helical DEMO Reactor FFHR-d1*)

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The performance of high-density plasmas heated by neutral beam (NB) injection in a vertically elongated configuration using four poloidal coils has been investigated in LHD and shown to be better than that in the normal configuration using six poloidal coils. Reduction of poloidal coils is favorable in designing an LHD-type helical reactor FFHR-d1, from both points of view of cost reduction and realization of large maintenance ports. In the experiment, no clear difference between the two configurations was recognized in the relation between the central density and the central pressure. Meanwhile, a factor C_{exp} used in the direct profile extrapolation (DPE) method is smaller in the vertically elongated configuration, where C_{exp} is proportional to the reactor size, $R_{reactor}$, at a given reactor magnetic field, $B_{reactor}$, i.e., $R_{reactor} \propto C_{exp} B_{reactor}{}^{-4/3}$. The difference in C_{exp} between the two configurations is enhanced to > 10 % as the plasma beta increases. This might be due to the larger plasma volume and/or the mitigated Shafranov shift in the vertically elongated configuration.

Keywords: heliotron, LHD, FFHR-d1, energy confinement scaling, vertical elongation, reactor design

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1. Introduction

In designing a magnetic thermonuclear fusion reactor, it is fairly important to secure large maintenance ports for replacement of in-vessel components of first wall, blanket, and divertor. In the case of a helical DEMO reactor named FFHR-d1 [1], of which the conceptual design activity has been launched at NIFS since 2010, the maintenance ports in the upper and lower side of the torus are maximized by omitting a pair of poloidal coils as shown in Fig. 1. FFHR-d1 is basically a large reproduction of LHD, which is the world's-largest superconducting heliotron [2]. In LHD, a pair of continuously wound helical coils and three pairs of planar poloidal coils generate the nested magnetic surfaces for plasma confinement. In the normal experimental condition in LHD, the toroidally averaged plasma cross section is kept circular by cancelling the quadrupole components of the magnetic field, B_Q, for 100 % using six poloidal-coils. This is called the "circular configuration of B_Q = 100 %". The toroidally averaged plasma cross section becomes vertically elongated when the two of six poloidal coils are not used, while the acceptable size of maintenance port becomes maximized. This is called the "vertically elongated configuration of $B_Q = X$ %", where X denotes the ratio of the cancelled quadrupole component. In the case of a magnetic configuration of R_{ax} = 3.75 m, for example, where R_{ax} is the plasma major radius, X is 53, if a pair of poloidal coils are not used as shown in the upper side of Fig. 1. Reduction of poloidal coils is also effective for saving the construction costs and shortening the term of construction work.

From the point of view of plasma performance, a better energy confinement has been observed in the circular configuration for low-density and low-beta plasmas heated by electron cyclotron heating [3]. This is thought to be a resultant of the neoclassical transport property, which is optimized at B_Q = 100 %. However, the plasma performance is not determined by the neoclassical transport alone. Especially in typical high-density and high-beta plasmas in LHD, the anomalous transport dominates the neoclassical transport. The neoclassical transport is expected to deteriorate the confinement property in the high-temperature (and therefore collisionless) reactor condition. However, it is possible to keep the collisionality in the reactor to the similar level as in LHD by increasing the density. For example, if the temperature in the reactor, of which the devise size is 4 times enlarged from LHD, is 4 times higher than that in LHD, it is possible to keep the collisionalities in the reactor and LHD the same by increasing the density for 4 times. One of the important issues other than the neoclassical transport is the Shafranov shift in the high-beta condition. For instance, the vertical elongation is effective

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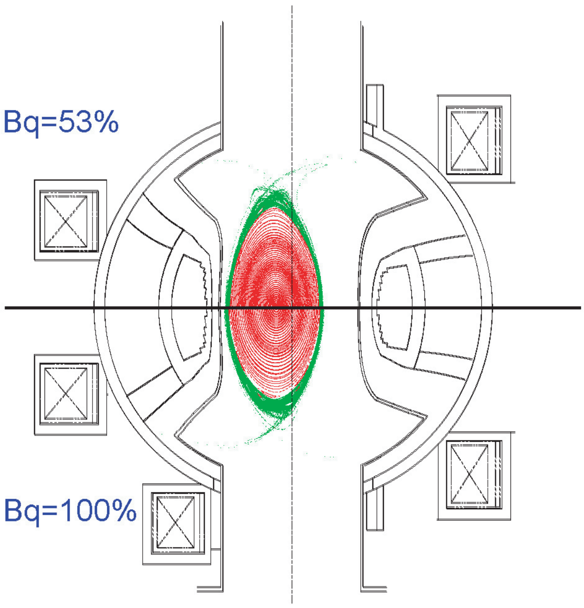

Fig. 1 Comparison of the poloidal coil configuration and nested magnetic surfaces in vacuum with four poloidal coils as in FFHR-d1 (upper) and six poloidal coils as in LHD (lower).

for mitigating the Shafranov shift [4]. This can be a strong merit in the reactor, where the central plasma beta is expected to be as high as ~10 % and a large Shafranov shift is foreseen. A large Shafranov shift should be avoided in the reactor since it is expected to deteriorate the energy confinement property and/or enhance the direct loss of alpha particles. It has been required to establish a new guideline other than the neoclassical transport to make a practical comparison between the circular and vertically elongated configurations.

In this study, the plasma performances in circular and vertically elongated configurations are compared for high-density and high-beta plasmas heated by neutral beam (NB) injection, using the direct profile extrapolation (DPE) method, which has been developed to predict the radial profiles in fusion reactors from the profile data obtained in the experiment [5]. A brief explanation of the DPE method is given in the next section. Experimental results obtained in the circular and vertically elongated configurations in LHD and comparison between them are discussed in Section 3. Finally, conclusions are given in Section 4.

2. The DPE Method

The DPE method has been developed to estimate how large device size will be needed in the reactor with a given magnetic field strength, based on the experimental results [5]. The gyro-Bohm model is used to estimate the heating power needed in the reactor. When the device size is given, on the contrary, one can calculate how large enhancement is needed in the experiments to assure the reactor design. An enhancement factor, f_{X}, is defined as the ratio of a parameter X in the reactor, X_{reactor}, to that in the experiment, X_{exp}, i.e.,

$$f_{\mathrm{X}} = \frac{X_{\mathrm{reactor}}}{X_{\mathrm{exp}}}, \tag{1}$$

where X can be, for example, $T(\rho)$, $n(\rho)$, a, R, P, B, and β, for the temperature profile, the density profile, the plasma minor radius, the plasma major radius, the heating power, the magnetic field strength, and the plasma beta, respectively, and $\rho = r/a$ is the normalized minor radius. For simplicity, some assumptions are adopted in this study as in Ref. [5], e.g., $f_{\mathrm{a}} = f_{\mathrm{R}}$, the temperature and the density of ions are equal to those of electrons, the deuterium (D) to tritium (T) ratio is 50 : 50, no impurity, and so on. According to the gyro-Bohm model, the energy confinement time, τ_{E}, is proportional to ($a^{2.4}$ $R^{0.6}$ $B^{0.8}$ $P^{-0.6}$ $n^{0.6}$) [6–9], i.e., the enhancement factor of τ_{E} is given by

$$f_{\tau} = \gamma_{\mathrm{DPE}}\, f_{\mathrm{a}}{}^{3}\, f_{\mathrm{B}}{}^{0.8} f_{\mathrm{P}}{}^{-0.6} f_{\mathrm{n}}{}^{0.6}, \tag{2}$$

where γ_{DPE} is the confinement enhancement factor. Using definitions of $\tau_{\mathrm{E}} \propto nTa^2R/P$ to eliminate f_{τ}, and $\beta \propto nT/B^2$ to eliminate f_{T}, we obtain

$$f_{\mathrm{P}} = \gamma_{\mathrm{DPE}}{}^{-2.5} f_{\beta}{}^{2.5} f_{\mathrm{B}}{}^{3} f_{\mathrm{n}}{}^{-1.5}, \tag{3}$$

i.e.,

$$P_{\mathrm{reactor}} = \gamma_{\mathrm{DPE}}{}^{-2.5} f_{\beta}{}^{2.5} f_{\mathrm{B}}{}^{3} f_{\mathrm{n}}{}^{-1.5} P_{\mathrm{exp}}. \tag{4}$$

On the other hand, the heating power in the reactor is calculated by the volume-integration as below:

$$P_{\mathrm{reactor}} = f_{\mathrm{a}}{}^{3} f_{\mathrm{n}}{}^{2} \int_{0}^{1} (P'_{\alpha} - P'_{\mathrm{B}})(\mathrm{d}V/\mathrm{d}\rho)_{\mathrm{exp}}\mathrm{d}\rho, \tag{5}$$

where P'_{α} and P'_{B} are the alpha heating power and the Bremsstrahlung loss per unit volume estimated by using $n_{\mathrm{exp}}(\rho)$ and $T_{\mathrm{reactor}}(\rho) = f_{\mathrm{T}}\ T_{\mathrm{exp}}(\rho) = (f_{\beta} f_{\mathrm{n}}{}^{-1} f_{\mathrm{B}}{}^{2})\ T_{\mathrm{exp}}(\rho)$, respectively. The relation of $f_{\mathrm{T}} = f_{\beta}\ f_{\mathrm{n}}{}^{-1} f_{B}{}^{2}$ is deduced from the definition of beta. The plasma volume in the reactor is $f_{\mathrm{a}}{}^{3}$ times larger than that in the experiment of V_{exp}. Since both P'_{α} and P'_{B} are the functions of temperature, the integral part of Eq. (5) can be approximated by $f_{\mathrm{T}}^{\mathrm{Y}}$, i.e.,

$$P_{\mathrm{reactor}} = \mathrm{A} f_{\mathrm{a}}{}^{3} f_{\mathrm{n}}{}^{2} (f_{\beta} f_{\mathrm{n}}{}^{-1} f_{\mathrm{B}}{}^{2})^{\mathrm{Y}}, \tag{6}$$

where A is a factor depending on the used profiles and the MHD equilibrium (i.e., $(\mathrm{d}V/\mathrm{d}\rho)_{\mathrm{exp}}$). Note that the factor A becomes large as the plasma volume determined by $(\mathrm{d}V/\mathrm{d}\rho)_{\mathrm{exp}}$ increases. Deleting P_{reactor} from Eqs. (4) and (6), we obtain:

$$f_{\mathrm{a}}^{3} = \mathrm{A}^{-1} \gamma_{\mathrm{DPE}}{}^{-2.5} f_{\beta}{}^{2.5-\mathrm{Y}} f_{\mathrm{B}}{}^{3-2\mathrm{Y}} f_{\mathrm{n}}{}^{-3.5+\mathrm{Y}} P_{\mathrm{exp}}. \tag{7}$$

The dependence on f_{n} disappears when Y = 3.5, then

$$f_{\mathrm{a}}^{3} = \mathrm{A}^{-1} \gamma_{\mathrm{DPE}}{}^{-2.5} f_{\beta}{}^{-1} f_{\mathrm{B}}{}^{-4} P_{\mathrm{exp}}. \tag{8}$$

From these, the reactor major radius, R_{reactor} (= $f_{\mathrm{R}} R_{\mathrm{exp}}$ = $f_{\mathrm{a}} R_{\mathrm{exp}}$) should be equal to, or larger than

$$R_{\mathrm{reactor}} = \mathrm{C}_{\mathrm{exp}} \gamma_{\mathrm{DPE}}{}^{-5/6} f_{\beta}{}^{-1/3} B_{\mathrm{reactor}}{}^{-4/3}, \tag{9}$$

where C_{exp} is a factor depending on A and experimental parameters as below;

$$C_{exp} = A^{-1/3} R_{exp} B_{exp}{}^{4/3} P_{exp}{}^{1/3}. \tag{10}$$

It should be remembered that C_{exp} becomes smaller if the plasma volume (and therefore, A) increases, even though R_{exp}, B_{exp}, and P_{exp} are unchanged.

In a reactor design study, it is convenient to use the major radius of helical coil, R_c, and the magnetic field strength on the helical coil center, B_c, which are typically fixed to $R_{c,exp}$ = 3.90 m and $B_{c,exp}$ = 2.54 T in LHD, instead of R and B defined at the magnetic axis. In this case, Eq. (9) is modified to

$$R_{c,reactor} = C_{exp^*} \gamma_{DPE}{}^{-5/6} f_\beta{}^{-1/3} B_{c,reactor}{}^{-4/3}, \tag{11}$$

where $R_{c,reactor} = R_{reactor}\ R_{c,exp}/R_{exp}$, $B_{c,reactor} = B_{reactor}\ B_{c,exp}/B_{exp}$, and $C_{exp^*} \gamma_{DPE}{}^{-5/6}\ f_\beta^{-1/3} = C_{reactor^*} = R_{c,reactor}\ B_{c,reactor}{}^{4/3}$, respectively.

To make the fusion reactor compact, C_{exp^*} should be as small as possible. In other words, C_{exp^*} can be a measure of plasma performance like the fusion triple product [10], which is the product of the averaged, or the central, density and temperature, and the energy confinement time. Although the fusion triple product is widely used as a simple and convincing measure of plasma performance, it includes a large ambiguity depending on the assumed profiles. Since C_{exp^*} reflects the whole radial profiles of density and temperature, C_{exp^*} can be a better index than the fusion triple product. In the next section, C_{exp^*} will be used to compare the plasma performance.

3. Experimental Results

To make a comparison of plasma performances between the circular and vertically elongated configurations in LHD, plasma experiments have been carried out in the magnetic configurations of B_Q = 100 % and 53 % at a fixed R_{ax} of 3.75 m. The magnetic field strength was varied from 0.75 T to 2.64 T for B_Q = 100 %, and from 1.0 T to 2.0 T for B_Q = 53 %, to see the beta dependence of the plasma performance. Plasmas were heated by negative-ion based NB injection of which the beam energy is ~180 keV. In the experiments discussed hereinafter, electron heating is dominant as in the reactor sustained by alpha heating, since the beam energy is much higher than the typical plasma temperature of less than 2 keV.

In Fig. 2 (a), shown is the central electron density, n_{e0}, dependence of the central beta, β_0, defined by $(2n_{e0}T_{e0})/(B_0^2/(2\mu_0))$, where n_{e0} and T_{e0} are the central density and temperature of electrons, respectively. The radial profiles of temperature and density was measured by Thomson scattering [11]. In this study, the density and temperature of ions are assumed to be equal to those of electrons, for simplicity. In both cases of the circular and vertically elongated configurations, the upper data of β_0 are on a curve proportional to $n_{e0}^{0.6}$. This is consistent with the gyro-Bohm model, where $\beta_0 \propto n_{e0}^{0.6}$ [5–9]. There observed no clear difference between the two configurations in this figure. A clear difference is recognized in the Shafranov shift as shown in Fig. 2 (b), where β_0 is plotted as a function of the major radius of the magnetic axis, R_0, in the horizontally elongated plasma cross section. As β_0 increases from 0 % to 6 %, R_0 increases from R_{ax} = 3.75 m to over 4.1 m, due to the Shafranov shift. If compared at the same β_0, the Shafranov shift is smaller in the case of vertically elongated configuration of B_Q = 53 % than in the case of circular configuration of B_Q = 100 %. If compared at the same R_0, on the contrary, a higher β_0 can be achieved in B_Q = 53 % than in B_Q = 100 %. This effect of Shafranov shift mitigation in vertically elongated configuration is consistent with the results reported in Ref. [4].

In Fig. 3, C_{exp^*} is plotted as a function of (a) β_0 at

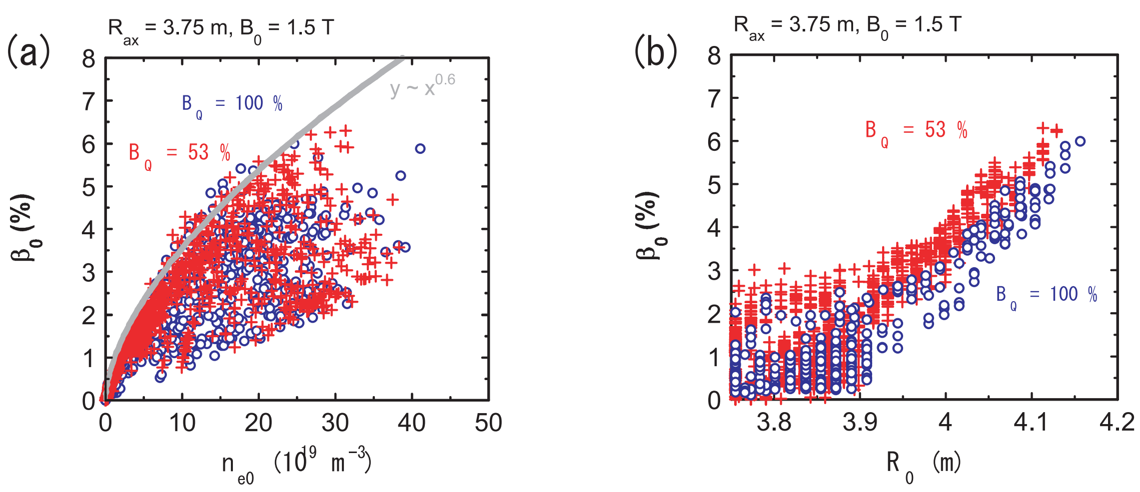

Fig. 2 The central beta, β_0, dependence on (a) the central electron density, n_{e0}, and (b) the plasma major radius, R_0, in the circular (B_Q = 100 %, open circles) and vertically elongated (B_Q = 53 %, crosses) configurations at a fixed condition of R_{ax} = 3.75 m and B_0 = 1.5 T. Thick gray curve in (a) denotes the gyro-Bohm type density dependence of $\beta_0 \propto n_{e0}{}^{0.6}$.

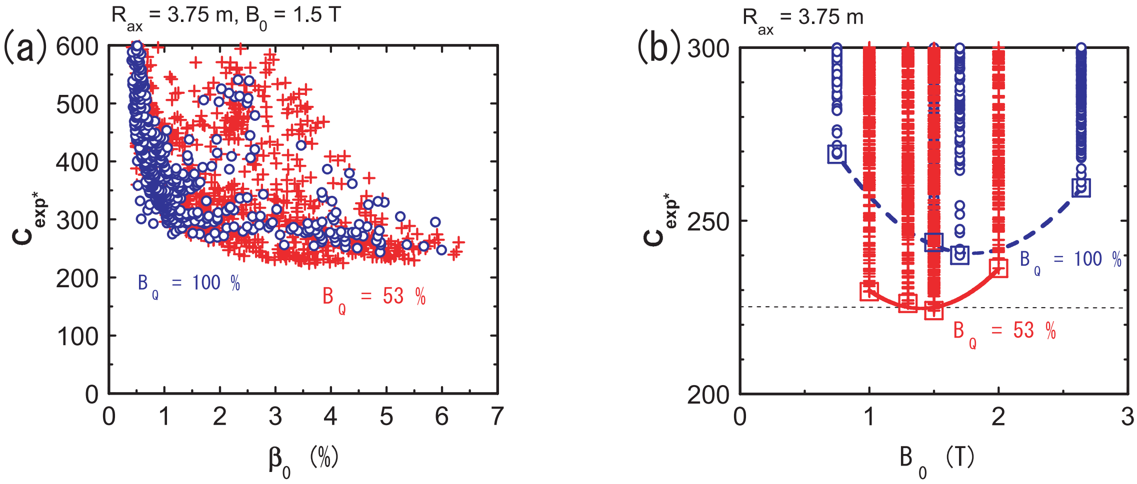

Fig. 3 The factor C_{exp^*} obtained for the circular (B_Q = 100 %, open circles) and vertically elongated (B_Q = 53 %, crosses) configurations, as a function of (a) the central beta, β_0, at a fixed condition of R_{ax} = 3.75 m and B_0 = 1.5 T, or (b) the magnetic field strength on the magnetic axis in vacuum, B_0, at R_{ax} = 3.75 m.

a fixed B_0 of 1.5 T, or (b) B_0, in the circular and vertically elongated configurations at R_{ax} = 3.75 m. As seen in Fig. 3 (a), C_{exp^*} decreases as the beta increases, although it tends to become saturated at high-beta. This saturation is presumably reflecting the degradation of NB heating efficiency due to the increase of direct loss of fast ions supplied by NB injection [12] and/or the degradation of energy confinement property expected in high-beta plasmas with large Shafranov shift [13]. It should be noted that the NB port-through power is used as an upper estimation of the heating power to calculate C_{exp^*} in Fig. 3. It is also recognized in Fig. 3 (a) that the minimum of C_{exp^*} for B_Q = 53 % is smaller than that for B_Q = 100 %. This tendency is independent of B_0 as shown in Fig. 3 (b). The positive B_0 dependence of C_{exp^*} at $B_0 > 1.5$ T is a resultant of the beta dependence, while the negative B_0 dependence might be due to the degradation in the NB heating efficiency and/or the energy confinement property at high-beta. At any B_0, C_{exp^*} obtained in the vertically elongated configuration is smaller than that in the circular configuration.

One of the possible reasons to explain the result obtained above is that the plasma volume, V_p, in the vertically elongated configuration is larger than that in the circular configuration, as is shown in Fig. 4, where V_p inside a_{99} is plotted with respect to β_0. Here, a_{99} is the effective minor radius inside which 99 % of the plasma kinetic energy is confined. Typically, a_{99} roughly corresponds to the effective minor radius of the last-closed-flux-surface (LCFS). Because of the larger V_p, the plasma stored energy, W_p, and τ_E (= $W_p/(P - \mathrm{d}W_p/\mathrm{d}t)$) in the vertically elongated configuration are 10–20 % larger than those in the circular configuration. If compared with the standard energy confinement scalings of ISS04 [14] and ISS95 [15], however, no clear difference between the circular and vertically elongated configurations has been recognized. This means that the improvement in τ_E is mainly due to the enlargement in

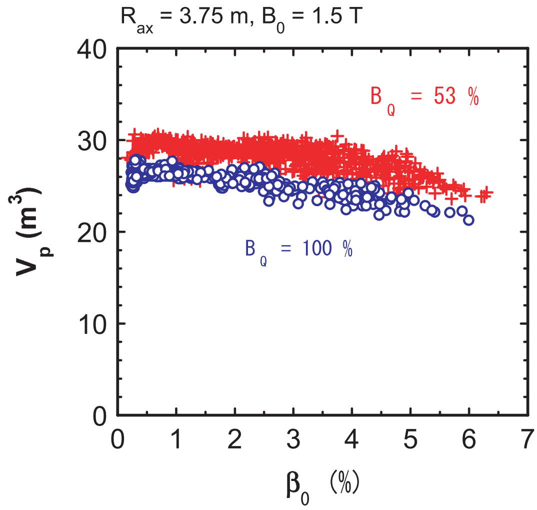

Fig. 4 The central beta, β_0, dependence of the plasma volume inside a_{99} (see text) in the circular (B_Q = 100 %, open circles) and vertically elongated (B_Q = 53 %, crosses) configurations.

V_p (or a). Both ISS04 and ISS95 consist of the gyro-Bohm type parameter dependence [8], where τ_E strongly depends on a. On the other hand, C_{exp^*} is influenced by V_p together with the better NB heating efficiency and energy confinement property, and becomes smaller with the larger V_p that results in the larger A (remind the discussions on Eq. (10)). From the point of view of C_{exp^*}, which is directly related to the device size of fusion reactor, we conclude that the vertically elongated configuration is better than the circular configuration.

4. Summary

The plasma performances in the circular and vertically elongated configurations in LHD are compared using a new index of C_{exp^*} used in the DPE method. In the vertically elongated configuration, Shafranov shift is mitigated and a smaller C_{exp^*} than in the circular configuration can be obtained. According to these results, it has been concluded that the vertical elongation of the plasma cross section due to the reduction of poloidal coils in FFHR-d1 is preferable rather than harmless. It is therefore reasonable to equip large maintenance ports with four poloidal coils in FFHR-d1.

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