Design Window Analysis for the Helical DEMO Reactor FFHR-d1*)

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Conceptual design activity for the LHD-type helical DEMO reactor FFHR-d1 has been conducted at the National Institute for Fusion Science under the Fusion Engineering Research Project since FY2010. In the first step of the conceptual design process, design window analysis was conducted using the system design code HELIOSCOPE by the "Design Integration Task Group". On the basis of a parametric scan with the core plasma design based on the DPE (Direct Profile Extrapolation) method, a design point having a major radius of 15.6 m and averaged magnetic field strength at the helical coil winding center of 4.7 T was selected as a candidate. The validity of the design was confirmed through the analysis by the related task groups (in-vessel component, blanket, and superconducting magnet).

Keywords: LHD-type heliotron system, DEMO, conceptual design, system design code, design window analysis

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1. Introduction

A helical system with a net current free plasma has suitable properties for a DEMO and a commercial fusion power plant: ease of steady-state operation without disruptive events due to plasma current, low recirculation power, and high plant efficiency. Among helical systems, the heliotron system with two continuous helical coils has recorded several remarkable findings in experiments at the Large Helical Device (LHD) [1]. On the basis of these findings, conceptual design activity for an LHD-type helical reactor FFHR (Force Free Helical Reactor) has advanced since the mid-1990s [2]. The latest design, FFHR-2m2 [3], proposed a commercially attractive reactor that enables long-term (30 full-power years), continuous operation. At the beginning of FY2010, a new conceptual design activity, FFHR-d1, commenced [4]. Here "d" denotes a "DEMO" reactor. As the next-generation reactor after the LHD, FFHR-d1 aims at an early demonstration of maintainability, tritium self-sufficiency, and net electric-power generation. Moreover, FFHR-d1 is designated as a "re-design" of the FFHR for enhanced design robustness, feasibility of construction, and safety. This new conceptual design study has been conducted by a newly-launched research project at the National Institute for Fusion Science (NIFS), the Fusion Engineering Research Project, which consists of 13 task groups and 44 sub-task groups. In the first step of the conceptual design process, design window analysis was conducted by the "Design Integration Task Group" to set the main design parameters (reactor size, magnetic field strength, and fusion power). In the next section, we describe prerequisites for the design. The result of the design window analysis and discussion of the candidate design point are presented in Section 3.

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2. Design Prerequisites of FFHR-d1

As described in the previous section, the past FFHR series was designed as a commercially-attractive fusion power plant capable of long-term operation with high plant availability and a net electric output comparable to that of current large-scale power plants (~1 GWe). Therefore, three engineering constraints were considered. First is suppression of the stored magnetic energy of the coil system, which is an index of the total mass of the structure required to support the electromagnetic forces. Reduction in the stored magnetic energy is also desirable for relaxing the design requirements and quench protection of the superconducting magnets. Second is reduction of the neutron wall load on the first wall, which extends the life-time of in-vessel components, including thermal-hydraulic components with a complex shape. Third is securement of sufficient space for the blanket to enable an adequate tritium breeding ratio (TBR) and simultaneous suppression of fast neutron flux on the helical coils. The third condition in particular is quite important and indispensable for long-term continuous operation because the superconducting magnets cannot be replaced, and the amount of tritium consumed in a ~1 GWe-class fusion power plant is much greater than that available in the market. The simplest method of increasing the blanket space is to enlarge the reactor. This, however, increases the stored magnetic energy. Here one of the important design parameters is the helical pitch parameter $\gamma_c = ma_c/(lR_c)$, where m, l, a_c, and R_c are the toroidal pitch number, poloidal pitch number (the number of helical coils), and minor and major radii of the helical coil, respectively. For an LHD-type heliotron system ($l/m = 2/10$), γ_c is proportional to the inverse aspect ratio of the helical coils. Thus, a_c decreases with decreasing γ_c when R_c is fixed. However, the averaged plasma minor radius decreases more, thereby expanding the space between the helical coils and the plasma. Consequently, a γ_c value smaller than that for the standard LHD configuration (= 1.2) was selected in previous FFHR designs.

In order to begin designing FFHR-d1, the role of a DEMO reactor was reconsidered. A DEMO reactor is the first device to generate high-energy particles and fast fusion neutrons simultaneously. In addition, it is the first device equipped with a full-scale blanket system and other specific components such as high-power heating system and diagnostic tools, which are used in the presence of fusion neutrons. Therefore, a certain period may be required including H-H, D-D and D-T operations, with a phased increase in the heating power for conditioning these systems. Therefore, long-term continuous operation and a 1 GWe-class electric output are not required for a DEMO reactor. Therefore, the design of FFHR-d1 is based on the engineering knowledge base established by the previous FFHR series but focuses more on the certainty of the extrapolation from LHD. Consequently, the magnetic configuration with γ_c = 1.25 and an inward-shifted magnetic axis (the ratio of the magnetic axis position R_{ax} to R_c is 3.6/3.9) was selected as the standard configuration of FFHR-d1. Because many experimental data are available and relatively good confinement properties have been observed with this configuration, design robustness of the core plasma can be enhanced.

In previous FFHR design studies, the position and shape of the helical and poloidal coils were based on the similar extension of those of the LHD. However, a helical-coil current density of j_c = 25 A/mm^2 was selected for FFHR-2m2 by reflecting the technical development for the ITER superconducting magnets. A greater width-to-height ratio of the cross section of the helical coils (W/H = 2, whereas W/H ~1.7 for the LHD) was selected to expand the blanket space. The same helical-coil conditions were selected for FFHR-d1. For FFHR-2m2, the number of pairs of poloidal coils was reduced from three (for the LHD) to two in order to secure a large maintenance space. The position of the poloidal coils was changed from that of the LHD to reduce the stored magnetic energy. In FFHR-d1, the number of poloidal coils is also reduced to two, but their position remains the same as that of the LHD (the IS coils are simply removed), as shown in Fig. 1, to ensure the extrapolation of the LHD experimental data.

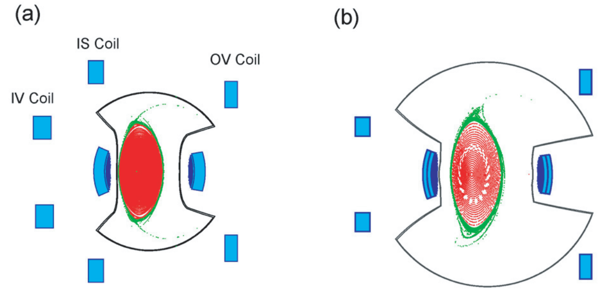

Fig. 1 Comparison of the positions of the poloidal coils of (a) LHD and (b) FFHR-d1.

3. Design Window Analysis

Design window analysis was conducted using the system design code for heliotron reactors HELIOSCOPE [5]. Similar to most system design codes, HELIOSCOPE adopts a simple zero-dimensional power-balance model with volume integral terms for the evaluation of the core plasma performance,

$$dW_p/dt = -W_p/\tau_E + \eta_\alpha P_\alpha - P_{rad} + P_{aux} = 0, \quad (1)$$

where W_p, τ_E, η_α, P_α, P_{rad}, and P_{aux} are the plasma stored energy, energy confinement time, alpha heating efficiency, alpha heating power, radiation loss and auxiliary heating power, respectively. The density and temperature profiles are generally described by the power of the parabolic function of the normalized minor radius ρ

$$n = n_0\left(1-\rho^2\right)^{\alpha_n},\ T = T_0\left(1-\rho^2\right)^{\alpha_T}, \quad (2)$$

because they give a good approximation of typical experimental results, and the volume integral terms in Eq. (1) can be obtained analytically. However, information on the plasma volume is required in such an analysis. Especially for a helical system, the plasma volume is not clearly defined under a finite-beta condition, and another assumption should be made. On the other hand, evaluation results (e.g., the requirements on the confinement improvement) are relatively sensitive to these assumptions.

In order to overcome this problem, Direct Profile Extrapolation (DPE) method [6]—a new method of determining the core plasma profile of a fusion reactor —has been proposed by the “Core Plasma Task Group”. In this method, an enhancement factor f_X is given by the ratio of the parameter X for a fusion reactor $X_{reactor}$ to that in the LHD experiment X_{exp} for temperature T, density n, plasma beta β, reactor size R, and magnetic field strength B. The plasma profile is extrapolated directly on the basis of the gyro-Bohm-type parameter dependence. This is because the dependence is recognized not only in the global energy confinement time but also in the local relationship between the temperature and the density in LHD experiments [7, 8]. Although the profile in the peripheral region is not just described by a gyro-Bohm-type parameter dependence, this extrapolation is useful because the parame-

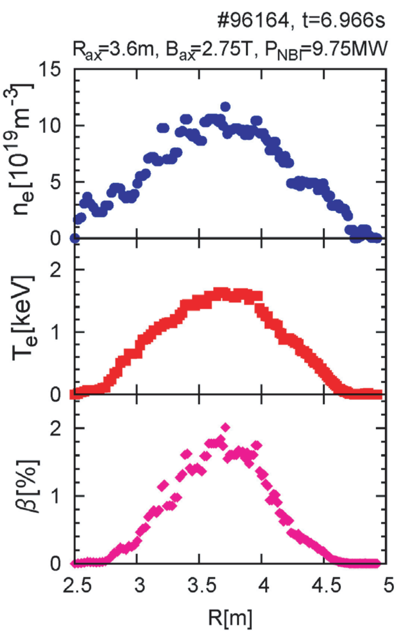

Fig. 2 Profiles of electron density, electron temperature, and plasma beta from LHD experimental data (#96164, 6.966 s) used for extrapolation to FFHR-d1.

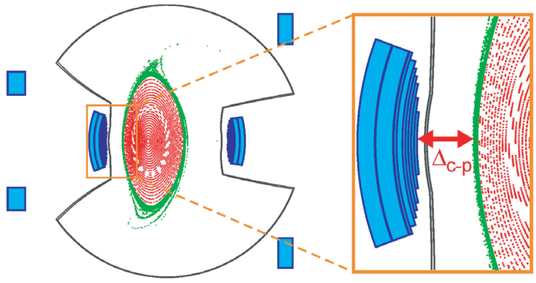

Fig. 3 Definition of the parameter $\Delta_{\text{c-p}}$.

ters in the core region are important in the following analysis. Using this DPE method, uncertainties in the plasma profile and plasma volume can be eliminated, and the existence of a stable equilibrium is also ensured. In the DPE method, the plasma volume that satisfies the self-ignition condition is determined as a function of the magnetic field strength. Especially in case of a similar extension of the plasma shape (the same plasma aspect ratio), the relationship between R and B is given by

$$R = C_{\text{exp}}\gamma_{\text{DPE}}^{-5/6} f_\beta^{-1/3} B^{-4/3}, \tag{3}$$

(see Eq. (18) in Ref. [6]), where C_{exp} is a factor determined by the experimental profile used for extrapolation, and γ_{DPE} is the confinement enhancement factor relative to the experimental results (note that it differs from the confinement enhancement factor relative to empirical scalings). The detailed derivation is available in Ref. [6].

For extrapolation to FFHR-d1, a typical plasma profile obtained by pellet fueling and neutral beam heating with the inner-shifted configuration (shot number #96164, time slice $t = 6.996$ s) was selected. The profiles of electron density, electron temperature, and plasma beta are shown in Fig. 2. The magnetic field on the magnetic axis is 2.75 T. The estimated deposition power is 9.75 MW. Using this profile, a parametric scan for R_c and $B_{t,c}$ (toroidal field at the winding center of the helical coils averaged over one helical pitch) was performed. Because a similar extension of LHD is considered in this study, the distance between the helical coil and the core plasma is proportional to R_c. This distance varies continuously along the toroidal direction and has its minima at the inboard side of the torus on the vertically elongated cross section. Here we define a new design parameter, $\Delta_{\text{c-p}}$, as the minimum distance between the helical coil and the plasma, which includes the ergodic layer (see Fig. 3). Note that not only the blanket materials (tritium breeder, neutron multiplier, and neutron shield) but also the structural materials, vacuum vessel, radiation shield for the helical coils, insulation gap between the radiation shield and the helical-coil bottom plate, and clearance between the plasma and the first wall should be included in this space. Because a constant current density in the helical coils is assumed, $\Delta_{\text{c-p}}$ is also a function of $B_{t,c}$. Another important engineering constraint is the stored magnetic energy W_{mag}. Because W_{mag} is proportional to $R_c^3 B_{t,c}^2$, it also places a boundary on the design window. In DPE method, the reactor size is determined if γ_{DPE}, f_β and B are given by Eq. (3). Conversely, the required value of f_β to realize a design point with specific values of $R = R^*$ and $B = B^*$ is given by

$$f_\beta = C_{\text{exp}}^3 \gamma_{\text{DPE}}^{-5/2} R^{*-3} B^{*-4}. \tag{4}$$

As shown in Ref. [6], the enhancement factor for the temperature f_T that gives the minimum R as a function of B depends only on the density and temperature profiles. Thus, the enhancement factor for the density f_n is also determined by the relation $f_n = f_\beta f_B^2 f_T^{-1}$ if R and B are given. Therefore, these two enhancement factors also place a boundary on the design window. Consequently, the possible design window is determined by restricting these factors. The design study of FFHR-2m1 evaluated that a large-scale helical coil (W_{mag} = 120-140 GJ) could be wound with a small extension of the ITER technology [9]. In the design study of FFHR-2m2, W_{mag} = 160 GJ was given as an indicator of the capability of R&D optimization with the same level of technical base. In this study, the same constraint, $W_{\text{mag}} \le 160$ GJ, is assumed as an engineering constraint. In the LHD experiments, the edge electron density n_{ea} is limited by the Sudo density scaling [10]. In this respect, another boundary of the design window is defined by $n_{\text{ea}}/n_{\text{Sudo}} \le 1$ instead of a specific value of f_n. The remaining constraints are γ_{DPE} and f_β. At present, there is no clear finding to determine the upper limit of these values. Thus here we assume $\gamma_{\text{DPE}} = 1.3$ and $f_\beta \le 5$ (which corresponds to $\beta_0 \le 10\%$ in this case).

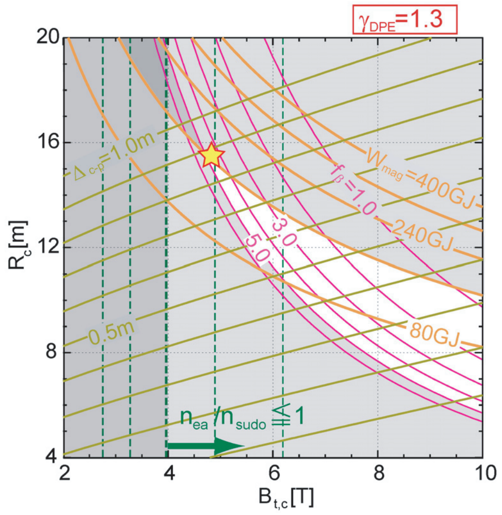

Fig. 4 Result of design window analysis for FFHR-d1. Contours of the stored magnetic energy W_{mag}, minimum distance between coil and plasma $\Delta_{c\text{-}p}$, beta enhancement factor f_β and edge density limit fraction are plotted. Plasma profile was extrapolated from the experimental result in Fig. 2. Star indicates candidate design point for FFHR-d1.

A pure D-T plasma with a D/T ratio of 1:1 (no impurities) and 100% deposition of alpha heating power (no alpha particle loss) is assumed to clarify the parameter dependence although these assumptions may yield an optimistic result. Figure 4 shows the contours of these design parameters. The design window without shading corresponds to the region that satisfies all the design constraints. The design point having the maximum $\Delta_{c\text{-}p}$ within this design window was selected as a candidate for FFHR-d1; R_c = 15.6 m and $B_{t,c}$ = 4.7 T, with $\Delta_{c\text{-}p}$ = 89 cm. In this case, fusion power and average neutron wall load are 3 GWth and 1.5 MW/m^2, respectively.

As discussed in the previous section, the requirement of the blanket thickness could be relaxed for a DEMO reactor. However, the value of $\Delta_{c\text{-}p}$ is ~15 cm smaller than that of FFHR-2m2. Therefore, additional analysis was conducted by related task groups to confirm the validity of this design point. Radial-build design was conducted in order to utilize this limited space to the fullest by the "In-Vessel Component Task Group". The task group concluded that a 70-cm-thick space can be used for the blanket material [11]. Neutronics calculations with an approximate 2-D torus model (at present) were conducted by the "Blanket Task Group". This group found that a sufficient TBR (~1.3 under the fully-covered condition) and maximum nuclear heating on the innermost surface of the helical coils of ~0.5 mW/cc can be achieved by using a 15-cm-thick tritium breeder (FLiBe/Be) and a 55-cm-thick neutron shield (with tungsten carbide) [12]. Thermal-hydraulic analysis on the supercritical helium coolant in the cable-in-conduit conductors was performed by the "Superconducting Magnet Task Group". This group showed that the temperature increase due to nuclear heating of 0.5 mW/cc is acceptable [13]. Moreover, this nuclear heating can be accepted by indirectly-cooled magnet options.

Table 1 Main design parameters of FFHR-d1.

Helical coil major radius R_c [m]	15.6
Plasma major radius R_p [m]	14.4
Helical pitch parameter γ_c	1.25
Plasma volume V_p [m^3]	1878
Helical coil minor radius a_c [m]	3.9
Toroidal field at winding center $B_{t,c}$ [T]	4.7
Magnetic field on axis B_{ax} [T]	5.08
Central electron density n_{e0} [10^{20}m^{-3}]	2.5
Central electron temperature T_{e0} [keV]	10.5
Peak beta value β_0 [%]	10
Fusion power P_{fus} [GW]	3.0
Confinement enhancement factor relative to the experimental data used for DPE γ_{DPE}	1.3
Confinement enhancement factor relative to ISS04v3 scaling $H^{ISS04v3}$	1.19
Helical coil current density j_c [A/mm^2]	25
Maximum magnetic field on helical coil B_{max} [T]	11.9
Minimum distance between the helical coil and the plasma $\Delta_{c\text{-}p}$ [m]	0.89
Average neutron wall load $<\Gamma_{nw}>$ [MW/m^2]	1.5

The main design parameters of FFHR-d1 are summarized in Table 1. Further detailed analysis of both the core plasma (equilibrium, neo-classical transport, alpha particle loss, etc.) and the engineering equipment (design and consideration of a maintenance method for 3-D in-vessel components) are now underway.

4. Summary

Based on the achievements of the LHD experiment and the knowledge base established through past FFHR design activities, conceptual design activity for the helical DEMO reactor FFHR-d1 and related engineering R&D is being performed under the Fusion Engineering Research Project in NIFS. In the first step of the conceptual design process, design window analysis was conducted. Using a new approach to determine the core plasma profile by direct extrapolation from the LHD experimental results, a candidate design point (R_c = 15.6 m, $B_{t,c}$ = 4.7 T) was selected assuming a confinement improvement of 1.3 times. To ensure the design feasibility, further detailed analysis is required. In particular, both design optimization and experimental effort are needed to increase the extrapolation precision of the core plasma and enhance the design robustness. Therefore, the physics that determines the plasma profile in the peripheral region should be carefully examined in future LHD experiments including deuterium discharges. Further detailed analysis, including studies of the 3-D geometry of the in-vessel components, is now underway.

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