CONCEPTUAL DESIGN OF A COMPACT HELICAL FUSION REACTOR FFHR-C1 FOR THE EARLY DEMONSTRATION OF A YEAR-LONG ELECTRIC POWER GENERATION

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Abstract

Conceptual design of a compact LHD-type helical fusion reactor FFHR-c1 has been conducted. This design focuses on a year-long electric power generation as possible by adopting the operation with auxiliary heating and innovative ideas for the design of superconducting magnet, divertor and blanket system. This design has a potential to satisfy the requirements on the Japanese fusion DEMO (steady-state electricity generation above several hundred MW, tritium fuel self-sufficiency and practical availability). This design ensures the path to helical commercial power plants through the examination of confinement scaling and steady-state operation test of engineering components. Though intensive R&Ds are needed, the innovative ideas provide more options and increase the probability of solving critical issues of fusion reactors: accommodation of high heat and particle load on the divertor, construction and maintenance within a reasonable period.

1. INTRODUCTION

In the past six years, the conceptual design activity of the helical reactor FFHR-d1A has been intensively conducted [1] based on the knowledge from the past design study and the achievements in the experiment of the Large Helical Device (LHD). This activity has shown the design feasibility of the reactor that has a long lifetime of ~30 years with a 3 GW-class fusion output by a combination of the ITER-relevant technology and some degree of improvement in the core plasma confinement from the LHD experiment. In the meantime, several innovative ideas have been proposed, for example, “joint-winding” method of helical coils using the high temperature superconductor (HTS) to shorten the time for the winding [2], blanket space enlargement (~15%) by placing supplementary helical coils (NITA coils) to secure the tritium breeding and neutron shielding performance [3], adoption of liquid metal ergodic limiter/divertor system to enable high heat/particle load accommodation (>100 MW/m²) and easy maintenance [4], and proposal of cartridge-type blanket modules to achieve construction and maintenance without complicated works inside the vacuum vessel [5]. Though these innovative concepts has been proposed to overcome the engineering difficulties in the design of FFHR-d1A, these concepts also enable a compact reactor design if focusing on the production of positive net electric power ($P_{e,net}>0$) by allowing operation with auxiliary heating and a shorter reactor lifetime due to the higher neutron flux to the superconducting coils. Therefore, a new design concept, FFHR-c1, which should be a more cost-effective way for the realisation of the requirements on the Japanese fusion DEMO, has been proposed as a multi-path strategy [6]. In this study, design window analysis to confirm the proposed design point of FFHR-c1 and integrated physics analysis of the core plasma were conducted. The method and result of the design window analysis is given in Section 2. The method and result of the integrated physics analysis is given in Section 3.

2. DESIGN WINDOW ANALYSIS

The candidate design point of FFHR-c1 has been proposed with the major radius of helical coil of $R_c = 10.92$ m and the magnetic field strength at the winding centre of the helical coil of $B_c = 7.3$ T according to a simple scaling of core plasma performance and the stored magnetic energy of the superconducting system [7]. To examine the feasibility of this design point from the viewpoint of plant power balance, design window analysis was conducted using the systems code for LHD-type helical reactors, HELIOSCOPE. Figure 1(a) shows the calculation result with current density of the helical coils of $j_c = 25$ A/mm², which can be achieved by a small extension of the ITER superconducting magnet technology, and the same plasma beta profile as assumed in the past design study, FFHR-d1B [8]. Then contours of plasma performance (fusion power $P_{fus}$ and fusion gain $Q$) and the total stored magnetic energy of the superconducting system, $W_{mag}$, can be plotted. The stored magnetic energy $W_{mag}$ is an index of cost and technological difficulty of the superconducting magnet system. As shown in Fig. 1(a), $W_{mag}$ can be reduced by keeping the same plasma performance by decreasing $R_c$ and increasing $B_c$ at the same time. The cost of most of other plant system depends on its mass or volume. Therefore, the construction cost and the construction period
of the reactor can be reduced if \( R_c \) is decreased by keeping the plasma performance. However, both the decrease of \( R_c \) and the increase of \( B_c \) lead to the decrease in the thickness of the blanket module if \( j_c \) is fixed. The decrease in the thickness of the blanket module cause the increase in the neutron flux to the superconducting magnet system, resulting in high nuclear heat generation. The shaded region in Fig. 1(a) corresponds to the design region with too high nuclear heat (10 mW/cc), which cannot be handled by whatever combination of superconducting material and coolant. If the nuclear heat is smaller than this value, the increase in the nuclear heat leads to the increase of the power consumption of cryogenic system, resulting in the limitation of the design window from the viewpoint of plant power balance. That is because the contour of engineering Q value \( Q_{\text{eng}} \) of 1, which corresponds to the condition of \( P_{\text{e,net}} > 0 \), deviates from the contours of fusion gain in the region of \( R_c < 14 \) m. Because the power consumption of other plant systems is also a function of \( R_c \) and \( B_c \), the design point of FFHR-d1B become a ‘minimum’ \( R_c \) that can achieve the condition of \( Q_{\text{eng}} > 1 \) as a result. However, new design concepts introduced as a challenging option of FFHR-d1A, the adoption of high temperature superconductor and the NITA coil, expand the reachable design window to the region with a smaller \( R_c \). Finally it was found that the originally proposed design point \( (R_c = 10.92 \, \text{m}, B_c = 7.3 \, \text{T}) \) will be a minimum design point that can achieve \( Q_{\text{eng}} > 1 \) if the adoption of NITA coils and the condition of \( j_c = 40 \, \text{A/mm}^2 \) are assumed, as shown in Fig. 1(b). This design point has a fusion power of \( P_{\text{fus}} \sim 400 \, \text{MW} \). The comparison of the primary design parameters of FFHR-c1 and FFHR-d1 is summarised in Table 1.

![Figure 1](image-url)  
**FIG. 1.** Result of the design window analysis with the following assumptions: (a) coil current density of 25 A/mm\(^2\) without NITA coils, (b) coil current density of 40 A/mm\(^2\) with NITA coils. Shaded region corresponds to the region with too high nuclear heating on superconducting coils.

| TABLE 1. COMPARISON OF PRIMARY DESIGN PARAMETERS OF FFHR-C1 AND FFHR-D1 |
| --- | --- | --- |
| Major radius \( R_c \) [m] | Centre text | 15.6 |
| Magnetic filed strength \( B_c \) [T] | Centre text | 4.7 / 5.6 |
| Fusion power \( P_{\text{fus}} \) [MW] | ~400 | ~3000 |
| Peak beat value \( \beta_0 \) | ~3.0 | ~8.0 / 5.6 |
| Net electric output \( P_{\text{e,net}} \) [MW] | >0 | ~1000 |
| Ratio of alpha heating power to auxiliary heating power | 2.52 | \( \infty \) |
| Current density \( j_c \) [A/mm\(^2\)] | ~40 | 25 |
| Stored magnetic energy \( W_{\text{mag}} \) [GJ] | ~150 | ~180 / ~250 |
| Plant lifetime [years] | ~10 | >30 |

3. CORE PLASMA PHYSICS ANALYSIS

3.1. Estimation of achievable fusion gain

To confirm the feasibility of the core plasma design, integrated physics analysis has been conducted using the physics analysis tools developed for the integrated transport analysis suite, TASK3D-a. In this analysis, the same
magnetic configuration as that of the previous study for FFHR-d1B \cite{8} (with a high aspect ratio $A \sim 7$ and inward-shifted magnetic axis position with the ratio of the magnetic axis position $R_{ax}$ to $R_c$ is $3.5/3.9$) was assumed. In this analysis, temperature and density profiles of electrons and ions were determined by the model based on the LHD experimental observation that is given in Ref. \cite{9}. First, the pressure profiles of electrons and ions in the reactor are given as a function of normalised minor radius by

$$p_{e,\text{reactor}}(\rho) = \gamma_{\text{DPE}} \hat{p}(\rho) P_{\text{abs},e}^{0.4} n_{\text{reactor},e}(\rho)^{0.6}, \quad (1)$$

$$p_{i,\text{reactor}}(\rho) = \gamma_{\text{DPE}} \hat{p}(\rho) P_{\text{abs},i}^{0.4} n_{\text{reactor},i}(\rho)^{0.6}. \quad (2)$$

In FFHR-c1, only electron cyclotron heating (ECH) is assumed to be used as an external heating source by considering its small effect on the blanket coverage, efficiency of the core heating and capability of the protection of the device from the direct irradiation of fusion neutrons. Then the terms of the absorbed power in Eqs. (1) and (2) are given as

$$P_{\text{abs},e} = \eta_e P_{\alpha}^t + \eta_{\text{aux},e} P_{\text{aux},e}^t - P_{\text{rad}} - P_{ei}, \quad (3)$$

$$P_{\text{abs},i} = \eta_{\text{aux},i} P_{\text{aux},i}^t + P_{ei}, \quad (4)$$

respectively. In Eqs. (3) and (4), $\eta_e, \eta_{\text{aux},e}$ and $\eta_{\text{aux},i}$ are absorption efficiency of the alpha heating power, auxiliary heating power to electrons and auxiliary heating power to ions, respectively. The terms of the heating power and the power loss $P_X (X = \alpha, \text{aux, rad, ei})$ are calculated from the radial profiles $Q_X(\rho)$:

$$P_X = \int_0^1 Q_X(\rho) \frac{d\rho}{d\rho}. \quad (5)$$

The equipartition power from electrons to ions is calculated by

$$Q_{ei}(\rho) = \frac{1.5 k_B}{T_e^t} \frac{(T_e^t - \rho)}{\tau_e^t(\rho)}, \quad (6)$$

where $k_B$ and $\tau_e^t$ are Boltzmann constant and electron-ion energy relaxation time, respectively.

$$P_e = \gamma_{\text{DPE}} \hat{p}(\rho) P_{\text{abs}}^{0.4} B^{0.8} n_e(\rho)^{0.6}, \quad (7)$$

where $P_{\text{abs}}$, $B$ and $n_e$ are the absorbed power, the magnetic field strength and the electron density, respectively. In Eqs. (1) and (2), $\hat{p}(\rho)$ is the gyro-Bohm normalised pressure profile defined as

$$\hat{p}(\rho) = \frac{P_{\text{exp}}(\rho)}{P_{\text{abs,exp}}^{0.4} P_{\text{exp}}^{0.8} n_{\text{exp}}(\rho)^{0.6}}, \quad (8)$$

where the subscript ‘exp’ denotes that the parameters are obtained from the reference LHD experimental data. In the DPE method, the electron pressure profile of the reactor is estimated by using this normalised pressure profile $\gamma_{\text{DPE}}$ and $\gamma_{\text{DPE}}$ in Eq. (1) and (2) are the confinement improvement factor related to the peakedness of the heating profile \cite{9} for electrons and ions, respectively. The definition is given by

$$\gamma_{\text{DPE},x} = \left( \frac{P_{\text{exp}}/P_{\text{dep}}^{\text{avg},x}}{P_{\text{exp}}/P_{\text{dep}}^{\text{avg,exp}}} \right)^{0.6}, \quad (9)$$

$$\left( P_{\text{exp}}/P_{\text{dep}}^{\text{avg},x} \right) = \int_0^1 \frac{P_{\text{dep},x}(\rho)}{P_{\text{dep},x}(1)} d\rho, \quad (10)$$

$$P_{\text{dep},x}(\rho) = \int_0^\rho P_{\text{abs},x}(\rho') \left( \frac{d\rho'}{d\rho} \right) d\rho', \quad (11)$$

for $x = e, i$. $P_{\text{dep},x}(\rho)$ is the deposition profile of the absorbed power that given by Eqs. (3) and (4).
The reference pressure profile is shown in Fig. 2. This profile was obtained from LHD experimental data with the peak beta value close to its MHD stability limit ($t = 3.533s$). The peak normalised pressure is higher than that in the previous study even the flattening of the pressure profile is observed around the normalised minor radius of $\rho = 0.4$. Here two fittings were conducted. One is the fitting using all data (solid line in Fig. 2) and the other is the fitting using the data with $\rho > 0.7$ (dotted line in Fig. 2), which predicts the peak beta value if the pressure flattening did not take place. The results are given in Fig. 4. The reachable operation region is judged by two constraints just the same in the previous research [8]: Mercier index at the $n/m = 1/1$ rational surface $D_I < 0.3$ and the maximum value over the minor radius of the ratio of neoclassical energy loss to the total absorbed power $(Q_{\text{neoclassical}}/P_{\text{abs}})_{\text{max}} < 0.5$. As shown in Fig. 3, $Q \sim 15$ ($P_{\text{abs}} \sim 370 \text{ MW}$ with an ECH power of $P_{\text{ECH}} \sim 25 \text{ MW}$) can be achieved in the case of with the lower peak normalised pressure, whereas $Q$ remains $\sim 10$ with the higher peak normalised pressure. This is because higher normalised pressure leads to less external heating power amount and the reduction of the total absorbed power especially around the core region, whereas the profile of the neoclassical energy loss is almost uniquely determined by the profiles of electron density and temperature. Therefore, higher normalised pressure does not necessarily lead to higher fusion gain. The peak beta value of the reference profile does not reach 3%. It means feasibility of the $Q \sim 15$ operation point itself is not fully confirmed. Therefore, the approach other than the finding of the reference data with higher normalised pressure may be required to achieve further higher fusion gain.

![FIG. 2. Radial profile of gyro-Bohm normalised pressure of the reference experimental data (circles). The lines are fitting curve used in this analysis using all data (solid) and the data with the normalised minor radius of $\rho > 0.7$ (dotted).](image)

![FIG. 3. Operation region of FFHR-c1 with the pressure profile of (a) solid curve in Fig.3 and (b) dotted curve in Fig. 3. The contours of the fusion gain (solid curve), the peak beta value (dash-dotted curve), the Mercier index (broken curve) and the ratio of the neoclassical energy loss to the total absorbed power (dotted curve) are plotted. The region without the shading corresponds to the operation regime with the physics conditions that have already been confirmed by the LHD experiment.](image)
3.2. Examination of modification of the helical coil winding law

Optimisation of the magnetic configuration is a possible way to achieve further higher fusion gain. In this regard, several sophisticated theoretical works have been conducted using optimisation codes. However, there is a possibility of simultaneous improvement in the MHD stability and neoclassical transport by a minor change in the magnetic configuration, i.e., modification of the winding law of the helical coil. In this regard, sensitivity analysis on two specific parameters of the winding law, helical pitch parameter $\gamma_c$ and pitch modulation parameter $\alpha$, has been conducted. Helical pitch parameter $\gamma_c$ is defined by

$$\gamma_c = \frac{m a c}{\ell R_c}, \quad (12)$$

where $m$, $a_c$, and $\ell$ are toroidal pitch number, minor radius and the number of the helical coil. Here the toroidal pitch and the number of helical coil are set to be the same as those of LHD ($m = 10$ and $\ell = 2$), helical pitch parameter corresponds to the inverse aspect ratio of the helical coil. In the design study of FFHR-d1 and FFHR-c1, $\gamma = 1.2$ was selected because Shafranov shift can be effectively suppressed with this configuration in the LHD experiment. Helical pitch modulation parameter relates the poloidal angle $\theta$ and the toroidal angle $\phi$ of the helical coil through the following formula:

$$\theta = \frac{m}{r} \phi + \alpha \sin \left(\frac{m}{r} \phi\right), \quad (13)$$

The trajectory of the helical coils with different $\alpha$ are plotted in Fig. 4.

In this study, dependence on helical pitch parameter with a fixed pitch modulation ($\alpha = 0.1$) and dependence on pitch modulation with a fixed helical pitch parameter ($\gamma_c = 1.2$) has been examined. Note that a big assumption has been made in this analysis: the shape of the last closed flux surface is kept just as the same as that in the vacuum equilibrium. Figure 5 and 6 show the comparison of the shape of the magnetic flux surface at the poloidal cross-section in which the plasma has a vertically elongated cross-sectional shape. In MHD equilibrium analysis with a finite pressure, the same peak beta value ($\beta_0 = 2.6\%$) was assumed. In this analysis, current of the two pairs of vertical field coils was set so that the magnetic axis position and the ratio of quadrupole field component of the vertical field coils to that of the helical coils are the same as those of the reference case with $\gamma_c = 1.2$ and $\alpha = 0.1$: $R_{av}/R_c = 3.6/3.9$ and $B_q,\text{poloidal}/B_q,\text{helical} = 0.72$. As shown in Fig. 5, cross-sectional shape of the plasma shows almost similar enlargement when $\gamma_c$ increases. As shown in the Fig. 6, cross-sectional shape transforms to ‘D’-shape if $\alpha$ increases. If $\alpha$ becomes negative, the cross-sectional shape becomes to be ‘inverse-D’ shape. The plasma volume has a maximum at $\alpha = 0.0$.

In the case of the finite-beta equilibrium, outward shift of the magnetic axis position becomes larger as the decrease of $\gamma_c$ with fixed $\alpha$ and the decrease of $\alpha$ with fixed $\gamma_c$, respectively. Figure 7 shows the radial profiles of
rotational transform $\theta/2\pi$ and Mercier index $D_I$ for different $\gamma_c$. Figure 8 shows the radial profiles of $\theta/2\pi$ and $D_I$ for different $\alpha$. In the case of fixed $\alpha$, rotational transform slightly decreases over the entire cross-section with the increase of $\gamma_c$, but the change is not so large. In the case of fixed $\gamma_c$, the dependence is not simple but core rotational transform decreases with the decrease of $\alpha$. Regarding $D_I$ at $m/n = 1/1$ rational surface, it slightly decreases with the increases in $\gamma_c$ and decreases with the decrease in $\alpha$. Figure 9 shows the radial profile of neoclassical energy loss. In this analysis, magnetic axis position at the finite-beta equilibrium is not adjusted. Thus, absolute value of the neoclassical power loss exceeds the volume-integrated total absorbed power at some radial position. Therefore, comparison of the absolute value of neoclassical energy loss or fusion gain is meaningless. On the other hand, the radial profile of the absorbed power is similar in all cases because the same radial profiles of electron temperature and density were assumed. Thus, comparison of neoclassical power loss can be an index for the
parameter $(Q_{\text{neo}}/P_{\text{abs}})_{\text{max}}$ in the previous subsection. In this regard, higher $\chi$ and $\alpha$ around −0.1 to 0.0 is favourable. These results indicate there is a possibility of simultaneous improvement in the MHD instability and neoclassical transport by a modification of the winding law, especially slight decrease in $\alpha$. Note that this analysis assumes the realisation of the same LCFS shape as that in the vacuum equilibrium in a finite-beta condition. In LHD experiment, larger outward shift of magnetic axis has been observed in the case of $\chi = 1.25$ compared with 1.2 with the same beta value, which is contradictory to the result in this analysis. In the case of larger $\chi$ and lower $\alpha$, rotational transform in the core region becomes low and $\theta/2\pi = 0.5$ rational surface emerges. Thus, MHD instability in the core region may become problematic. In any case, further detailed finite-beta equilibrium analysis (e.g., by HINT2 code) is required to identify optimum value of $\chi$ and $\alpha$.

4. SUMMARY

Design study of the LHD-type helical fusion reactor which enables a year-long steady-state operation with self-sufficiency of electricity and tritium fuel has been conducted. The proposed design point with $R_c = 10.92$ m and $B_c = 7.3T$ has been confirmed by the design window analysis using the systems code HELIOSCOPE. It was found that this design point can achieve fusion gain of ~15 with a fusion power of ~370 MW by the integrated physics analysis of the core plasma.

There still remains several engineering and physics issues. Especially, design of superconducting coils with a high current density of $j_c > 40$ A/mm$^2$ is a challenging issue. In this regard, innovative concepts including an insulation-
less conductor concept has been being intensively studied. Examination of the equipment layout, construction scenario and maintenance scenario as well as the detailed design analysis of each component (e.g., structural analysis of the coil supporting structure, thermohydraulic and neutronics analysis of the blanket) have been also conducted to ensure the consistent system design. Achievement of further higher peak beta value of ~3% with an inward-shifted configuration is an important physics issue. Regarding this issue, ‘minor’ modification of the winding law of helical coils, especially a slight decrease in helical pitch modulation can be a solution. Consequently, the design feasibility of a compact LHD-type helical reactor that can satisfy the requirements on Japanese fusion DEMO, steady-state electricity generation above several hundred MW, tritium fuel self-sufficiency and practical availability, has been shown.

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REFERENCES