NITA Coil — Innovation for Enlarging the Blanket Space in the Helical Fusion Reactor*)

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An innovative idea is proposed for enlarging the blanket space on the inboard side of the torus for the helical fusion reactor FFHR-d1. A set of sub-helical coils, named NITA coils, with opposite-directed current outside the main helical coils, effectively reduces the helical pitch parameter and enlarges the blanket space. Dependence of the blanket space and plasma volume on the effective helical pitch parameter is examined. The obtained magnetic surfaces and their properties are compared with that of the original configuration.

Keywords: helical fusion reactor, FFHR-d1, optimization of heliotron magnetic configuration, blanket space, NITA coil

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1. Introduction

Conceptual design studies of the helical fusion reactor FFHR-d1 are steadfastly progressing at National Institute for Fusion Science (NIFS) taking advantage of the steady-state operation capability of the LHD-type heliotron magnetic confinement, which has no need for current drive and is free from disruption [1]. The present design, FFHR-d1, aims at generating 3 GW of fusion power through self-ignition. The FFHR-d1 has a pair of helical coils with a poloidal pole number $l = 2$ and a toroidal pitch number $m = 10$. The basic magnetic configuration is similar to that of the presently operating Large Helical Device (LHD) [2]. The major radius R is 15.6 m, which is four times that of LHD. A multi-path strategy is pursued to secure the design foundation of FFHR-d1, which presently specifies FFHR-d1A as the base option for promoting three-dimensional engineering design [3]. The FFHR-d1A has the minor radius of the helical coils $a_c = 3.744$ m. Using these values, the helical pitch parameter γ_c is

$$\gamma_c = \frac{m}{l}\frac{a_c}{R_c} = \frac{10}{2} \times \frac{3.744}{15.6} = 1.20. \tag{1}$$

Here γ_c was changed from 1.25 in FFHR-d1 ($a_c = 3.9$ m) to improve the confinement by reducing the Shafranov shift.

One of the difficult issues for the engineering design of the heliotron-type fusion reactor is that the "blanket space" Δ_{cp} on the inboard side of the torus is very small. Here Δ_{cp} is defined as the distance between the innermost layer of the helical coil windings and ergodic layer outside the last closed flux surface (LCFS). In FFHR-d1, Δ_{cp} is evaluated at 890 mm (Fig. 1). Within this blanket space, the following components should be installed [4]: the bottom case of the helical coil, the low-temperature thermal shield, the radiation-shielding blanket (including the vacuum seal) and the breeding blanket. In the present design, the radiation-shielding and breeding blankets have relatively thin thicknesses of 550 mm and 150 mm, respectively. We also note that the gap between the coil case and low-temperature thermal shield only becomes 10 mm

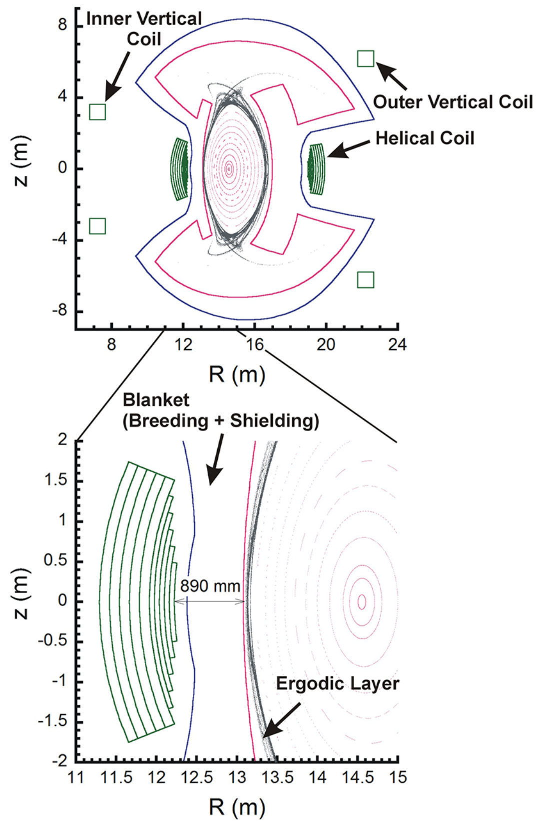

Fig. 1 Cross-sectional image of the FFHR-d1 configuration ($\gamma_c = 1.25$) at the toroidal angle of $\phi = 0°$.

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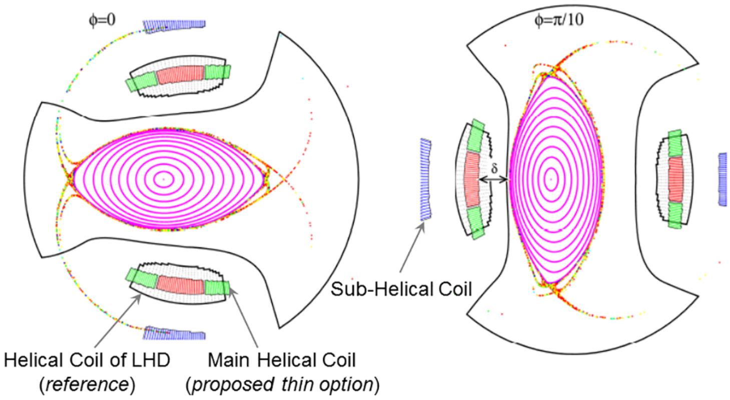

Fig. 2 Original proposal for enlarging the blanket space on the inboard side of the torus by employing a pair of sub-helical coils (presently called NITA coils). In this drawing (by T. Watanabe; slightly modified and coil names added), the main helical coils are divided into three thin blocks. The helical pitch parameter is 1.25. The helical coils of LHD are also shown for comparison. The NITA coils are located at 1.492 times the minor radius of the main helical coils and have an opposite-directed current of −11.59%.

at room temperature during the construction. In the present base option of FFHR-d1A, due to the decrease of γ_c, $\varDelta_{cp}$ increases to 940 mm and the engineering difficulty slightly eases. To further improve the structural integrity, lower the neutron flux and increase the tritium breeding ratio, enlarging $\varDelta_{cp}$ even further is preferable.

Four methods were previously considered for enlarging $\varDelta_{cp}$. The first was to choose a smaller γ_c, such as 1.15 for FFHR-2m1 [5], however, this also results in a smaller LCFS (plasma volume). The second approach was to set a high current density in the helical coil, which intensifies the engineering challenge. The third was to make an outward shift of magnetic surfaces and cut the remnant ergodic layer by target plates on X-points [5, 6]. However, the alpha particle confinement and neoclassical transport deteriorate; as a result, an inward shifted configuration is employed for FFHR-d1. The fourth method included splitting the helical coils on the outboard side of the torus to improve the poloidal symmetry of magnetic surfaces [7], but the outward-shifted configuration is again the problem.

In 2014, another innovative method was proposed by Tsuguhiro Watanabe. The idea was to vary the current density within the helical coil by subdividing the coil windings into multiple blocks [8]. However, the proposed method was later determined technically infeasible due to the extremely high current density required for the helical coil blocks. Then, a modified version based on a similar concept was proposed by the same author to divide the coil blocks in the minor radial direction using a set of sub-helical coils [9]. This method is discussed in this paper.

2. FFHR-d1-TW Configuration with Enlarged Blanket Space by NITA Coils

By dividing the helical coil into multiple blocks with individual currents, the helical pitch parameter of the entire helical coil is given as follows by extending Eq. (1),

$$\gamma_c = \frac{10}{2} \times \frac{1}{R_c} \frac{\sum a_{ci} I_i}{\sum I_i}, \tag{2}$$

where a_{ci} is the minor radius of each block and I_i is the current in that block. Here γ_c is given by the averaged minor radius of all blocks, i.e., the current center. We note that in LHD, the helical coil is divided into three blocks: H-I (inner), H-M (middle), and H-O (outer). By uniformly supplying current to the three blocks, $\gamma_c = 1.25$ in the standard configuration. When the current is only supplied to the H-I blocks, γ_c becomes smallest at 1.12. When only the H-O blocks are energized, γ_c reaches its maximum value 1.38. It should be noted that in LHD, $\varDelta_{cp}$ changes by γ_c since the structure of the entire helical coil does not change.

The new proposal is an extended version of this idea. A set of sub-helical coils is placed outside the main helical coils as shown in Fig. 2 (which is the original proposal by T. Watanabe). By applying an opposite-directed current to these sub-helical coils, γ_c is effectively reduced according to Eq. (2) and $\varDelta_{cp}$ is enlarged as a resultant. Here the sub-helical coils are called NITA (Newly Installed Twist Adjustment) coils.

Figure 3 (a) shows a plan view of the coil system of the presently designed FFHR-d1-TW with a pair of NITA coils located at $a_{NITA} = 2 \times a_{HC}$. An opposite-directed current of −7.692% is supplied to give the effective γ_c of 1.20. The blanket space becomes 1080 mm in this case as shown in Fig. 4. (Here the original blanket for FFHR-d1 is drawn, which should be reexamined for its thickness and position.) Figure 5 shows the dependence of the blanket space on the inboard side of the torus and the averaged minor radius of LCFS on the helical pitch parameter. Using NITA coils, $\varDelta_{cp}$ is effectively enlarged, but the plasma volume does not significantly decrease.

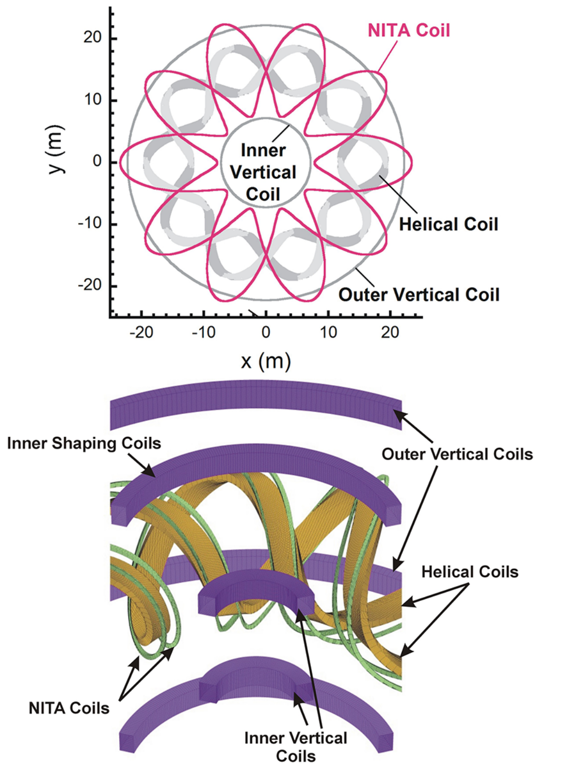

Fig. 3 (a) Plan view of the coil system of FFHR-d1-TW including the NITA coils. (b) Three-dimensional image of the coil system (drawn by T. Watanabe; coil names added). In this case, each NITA coil is divided into two separate coils.

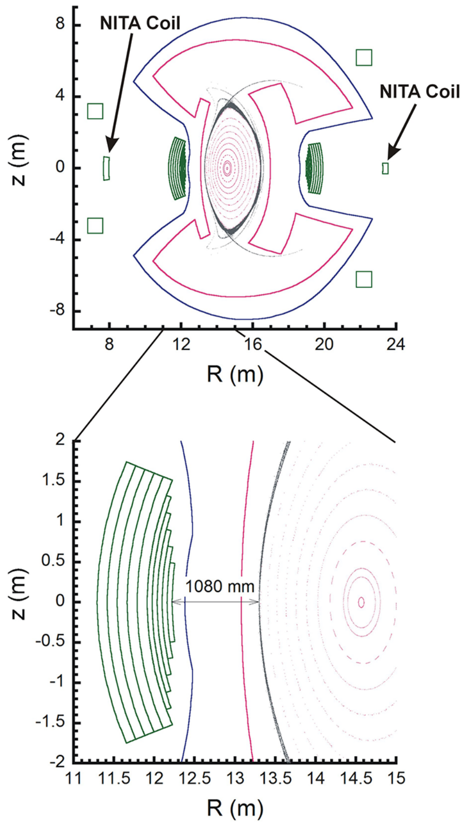

Fig. 4 Cross-sectional image of the FFHR-d1-TW configuration at the toroidal angle of $\phi = 0°$.

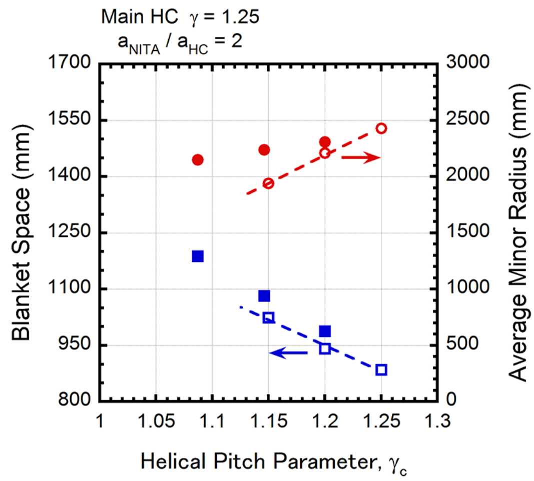

Fig. 5 Dependence of the blanket space on the inboard side of the torus and average minor radius of LCFS on the helical pitch parameter. Open symbols correspond to the case without NITA coils. Closed symbols are with NITA coils.

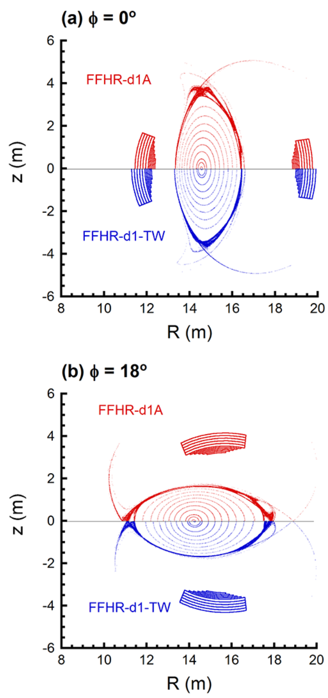

Fig. 6 Comparison of the two configurations, FFHR-d1A (up) and FFHR-d1-TW (down), with regard to the magnetic surfaces at two toroidal cross-sections: (a) $\phi = 0°$ and (b) $18°$.

3. Discussion

Since the basic analyses for plasma transport and MHD stability are presently being carried out for the FFHR-d1A configuration, examining the deviation of the FFHR-d1-TW configuration from that of FFHR-d1A is important. A comparison of the magnetic surfaces is shown in Fig. 6 for the two toroidal cross-sections, which confirms that the core region within the LCFS is almost the same in both cases. To examine the difference in detail, the rotational transform and magnetic well depth are compared in

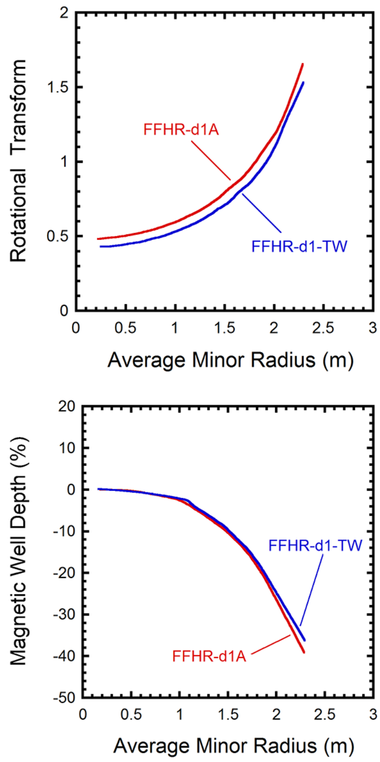

Fig. 7 Comparison of the two configurations, FFHR-d1A and FFHR-d1-TW, with regard to the radial profiles of (a) the rotational transform and (b) magnetic well depth.

Fig. 7. The difference is not significant, which may support that the plasma transport and MHD stability characteristics of FFHR-d1-TW are not very different from those expected for FFHR-d1A. From an engineering viewpoint, the divertor legs protrude from slightly different positions.

With respect to the engineering design, the enlarged blanket space should be effectively distributed to the necessary components. For example, if the radiation-shielding blanket is thickened, the neutron flux could be reduced, which would reduce nuclear heating and enhance the lifetime of the helical coils. Thickening the breeding blanket to increase the tritium breeding ratio is also important. Electromagnetic stress analysis should be carried out to examine the effect of NITA coils, which come close to the Outer Vertical and Inner Vertical coils.

There is also a possibility of promoting a smaller reactor design, FFHR-c1 [1]. Using NITA coils, a blanket space of 900 mm (larger than that for FFHR-d1) is still obtained with $R_c = 13.0$ m.

4. Summary

To enlarge the blanket space on the inboard side of the torus in the helical fusion reactor FFHR-d1, a new configuration FFHR-d1-TW is proposed, which includes two sub-helical coils, the NITA coils, with opposite-directed currents to effectively reduce the helical pitch parameter without changing the position of the main helical coils. The blanket space in the FFHR-d1-TW is enlarged by ~ 200 mm compared to the original design of FFHR-d1 by supplying −7.692% current of the main helical coils with the minor radius of twice as that of the main helical coils. The magnetic surfaces do not significantly change from the basic option of the presently designed FFHR-d1A.

For the engineering design, the enlarged blanket space should be effectively distributed to the necessary components, which is under discussion. In addition, electromagnetic stress analysis will be carried out to examine the effect of NITA coils. There is also a possibility of promoting a smaller reactor design utilizing the enhancement of the blanket space.

Acknowledgments

This paper is dedicated to Tsuguhiro Watanabe, the original proposer of the idea discussed in this paper, who passed away in February, 2015. The sub-helical coils for enlarging the blanket space are now named "NITA" coils according to his nickname. The authors thank the members of Fusion Engineering Research Project at NIFS for fruitful discussions. They would also like to thank Enago (www.enago.jp) for the English language review.

[1] A. Sagara, H. Tamura, T. Tanaka *et al.*, Fusion Eng. Des. **89**, 2114 (2014).
[2] A. Komori *et al.*, Nucl. Fusion **49**, 104015 (2009).
[3] H. Tamura, T. Goto, T. Tanaka *et al.*, Fusion Eng. Des. **89**, 2336 (2014).
[4] H. Tamura, T. Goto, N. Yanagi *et al.*, Fusion Eng. Des. **88**, 2033 (2013).
[5] A. Sagara, O. Mitarai, S. Imagawa *et al.*, Fusion Eng. Des. **81**, 2703 (2006).
[6] T. Morisaki, S. Imagawa, A. Sagara and O. Motojima, Fusion Eng. Des. **81**, 2749 (2006).
[7] N. Yanagi, K. Nishimura, T. Goto *et al.*, Contrib. Plasma Phys. **50**, 661 (2010).
[8] T. Watanabe, N. Yanagi and A. Sagara, Plasma Fusion Res. **9**, 3403089 (2014).
[9] T. Watanabe, private communication (presentation at NIFS Fusion Engineering Research Project meeting, July 2014).