Design modification of structural components for the helical fusion reactor FFHR-d1 with challenging options

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A conceptual design for a helical fusion reactor is currently being undertaken by the National Institute for Fusion Science, Japan. The coil support structure is designed from the perspective of the allowable stress of the material. A continuous helical coil winding with a low temperature superconductor and a water-cooled divertor made of tungsten and copper alloy have been considered for use in this reactor, and it is defined as the basic option. Several flexible design options have also been proposed; these options solve existing issues in the basic option and they are treated as the challenging options. They can be implemented by modifying the structural components of the basic option. The structural design modifications need to be made in order for the challenging options and the design optimizations consider mechanical soundness were investigated.

Keywords: fusion reactor design, helical fusion reactor, superconducting magnets, structural analysis, electromagnetic force.

1. Introduction

A conceptual design study for a helical-type fusion reactor is being conducted by the National Institute for Fusion Science in Japan. It is developing a Large Helical Device (LHD) type fusion reactor that has been designated the codename FFHR-d1 [1,2]. Fig.1 shows the magnet system of this reactor; it is comprised of one pair of helical coils (HC), two sets of vertical field coils (VFC), and a coil support structure. The major and minor radii of the helical coils are 15.6 m and 3.774 m, respectively. The magnetic field at the center of the plasma is 4.7 T, and the total magnetic stored energy is 160 GJ. The coil support structure has been designed so as to have large apertures that allow for the maintenance of in-vessel components. The structure was also designed with the perspective of the allowable stress of the structural material in mind.

A continuous helical coil winding with a low temperature superconductor (LTS) and a divertor made of tungsten and copper alloy with water cooling are considered for use in the "basic option" of FFHR-d1. A construction scheme has been carried out with this specification in mind, along with research and development into the components that would be used in FFHR-d1. In addition to this basic option, there are several flexible design options that look to solve the issues the basic option has; these issues include the winding method of the huge helical structure, high heat flux and neutron irradiation of the divertor component, and the narrow space between the plasma surface and the helical coil at the inboard side of the torus.

The alternative design proposals that address these issues are treated as "challenging options." For example, a joint coil winding with a high temperature superconductor (HTS), an additional helical coil with a negative current flow that widens the distance between the plasma surface and the main helical coil, and a liquid metal divertor with molten tin, have been proposed.

This paper investigates the structural design modifications that would need to be made in order for the challenging options to be used.

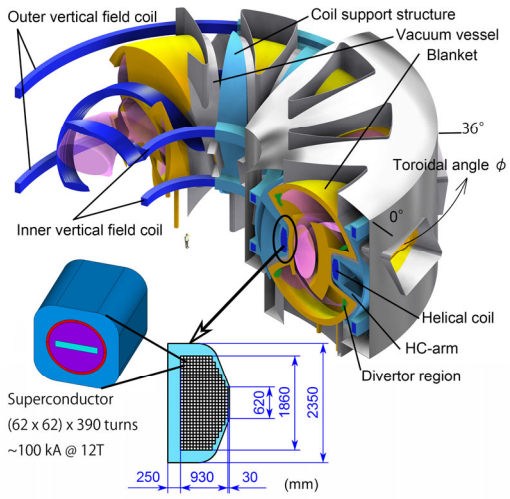

Fig. 1. Schematic of the FFHR-d1, and cross section of the superconducting helical coil.

2. Challenging options and structural components

2.1 Joint-winding of the helical coil with a HTS

Two types of cooling schemes have been proposed for the basic option's helical coil: one is a cable-in-conduit conductor using an LTS [3]; the other is an indirect cooling system with an aluminum alloy-jacketed LTS [4]. Both assume a continuous winding method.

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A challenging idea is the "joint-winding" that efficiently connect segmented HTS [5]. A prototype HTS conductor sample having also a bridge-type mechanical lap joint successfully achieved 100 kA at 20 K, 5.3 T with a low-resistance joint of 1.8 nΩ [6].

There is no difference in the final cross section of the helical coil winding package between the joint-winding and the continuous winding methods. As a result, any candidate superconductor and winding method can be adopted as the spatial design for the coil support structure since the electromagnetic (EM) force induced by the coils is identical.

2.2 Using a NITA coil as a supplementary helical coil

Δc−p is the distance between the plasma surface and the bottom of the HC, and it is estimated to be 890 mm in the basic option of FFHR-d1 [7]. Δc−p could be enlarged by setting an additional HC, named NITA (Newly Installed Twist Adjustment) coils, outside the main helical coils and providing them with an oppositely directed current. This can increase the Δc−p by more than 1 m without decreasing the average minor radius of the plasma [8].

A coil current of HCs and the VFCs are needed to increase to maintain the plasma volume and the geometrical center. Table 1 shows the coil specifications of both the basic option and the option where the NITA coils are added. It can be seen that the EM force distribution can be changed by adding in the NITA coils. As a result, the coil support structure is possibly needed to be re-designed in case of the NITA coil addition.

Table 1. Coil specifications for FFHR-d1 with and without NITA coils.

	Basic option	w/ NITA
HC major radius	15.6 m	←
HC minor radius	3.744 m	3.9 m
HC current	36.66 MA	39.715 MA
NITA minor radius	—	7.488 m
NITA current	—	3.06 MA
Inner VFC current	18.5 MA	22.05 MA
Outer VFC current	−19.88 MA	←

2.3 Liquid metal divertor

The steady state heat load in the divertor system exceeds 20 MW/m^2 at its peak. The basic option for the divertor system is a full helical construction made from a tungsten and copper alloy and cooled by flowing water. The divertor component has to remove an extremely large amount of heat flux, and it is also severely irradiated.

Alternative challenging options include a liquid metal divertor using a shower of molten tin [9] and a novel divertor location [10,11] (which will be discussed in 2.4). The first of these options is expected to deliver high divertor durability, small amounts of radioactive waste, and high permissible heat loads. In order to implement this system into FFHR-d1, an additional aperture has to be made so as to allow vertical access to the liquid metal divertor, which is set at the cross-point of the divertor's legs at the inboard side of the torus. Furthermore, the inner port area has to be enlarged so that a circulation pipe and an exhaustion path can be installed through the inner port.

2.4 Novel divertor location for the solid divertor

The neutron load on the divertor can be reduced by setting it behind the shielding/breeding blanket. However, the maximum irradiation damage done to the copper in the divertor regions is still high It was estimated to be 1.6 dpa/year [12].

In order to address this, a proposal has been made to relocate the divertor components by partially removing the HC support. The effectiveness of this novel divertor location was evaluated, and it was shown that the irradiation flux could be reduced to 1/5 to 1/10 of the basic option's value [11]. By using this result, it was estimated that the divertor lifetime could be six years. A soundness evaluation of the specifically modified coil support structure used for the novel divertor location showed that the stress in the structure was permissible [10].

3. Structural modifications for the challenging options

In the challenging options, the additions of the NITA coils and the liquid metal divertor require the structural components in FFHR-d1 to be modified. Fig. 2 shows an example of a modified design of the coil support structure.

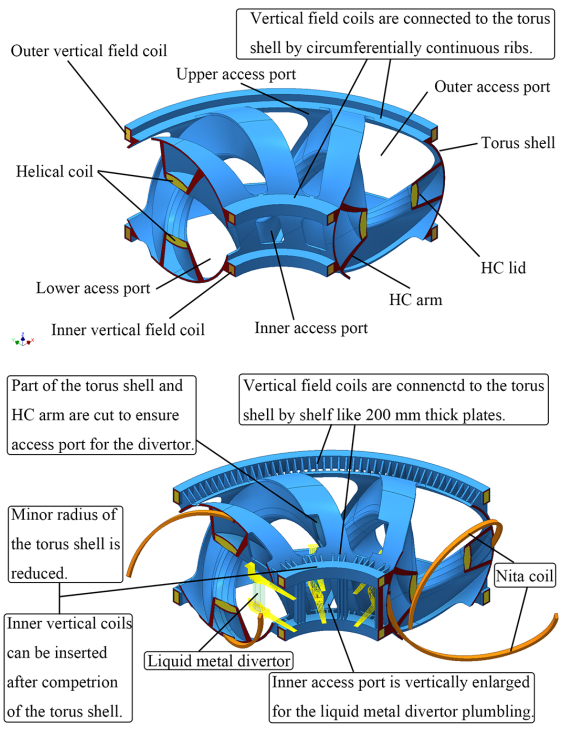

Fig. 2. Fundamental design of the basic option of FFHR-d1 (above), and the modified design involving the challenging options (bottom).

The changes that would need to be made to the fundamental design of the coil support structure in the basic option are the addition of one pair of NITA coils and ensuring an access port (for maintenance and plumbing) for the liquid metal divertor. The design of the modified coil support structure assumes that it is made of 200–250 mm thick stainless steel (S.S.) with full penetration welding. Furthermore, the minor radius of the torus shell section of the support structure was reduced so that it would not interfere with the inner vertical field coils. The inner VFC can be installed after the HC winding and main support structure have been made.

3.1 Magnetic field and electromagnetic force

The magnetic field distribution and EM force in the coil support structure were calculated according to the geometry and current flow in each coil, including in the NITA coils. The maximum magnetic field was 13 T, while the basic option's maximum was 11.6 T. The EM force can be divided into two directions: either corresponding to the hoop and overturning directions with respect to the coil winding direction. The overall EM forces at the hoop and the overturning force for each cross section of the HC are shown in Fig. 3. The maximum EM force for the hoop was 69 MN/m, while the maximum EM force for the overturning was ±8 MN/m.

Fig. 4 shows the EM force on the VFCs. The positive value indicates an expansion or repulsive force. By increasing the coil current, the maximum magnetic field on the HC, the EM force on the hoop in the HC, and both the hoop and up-down forces in the inner VFC increased. The maximum force on the hoop and the inner VFC attractive force reached 79 and 22 MN/m, respectively. The change in the EM force in the outer VFC was relatively small. The magnitude of the EM force in the NITA coils was small, but a rapid change appeared when the NITA coils and VFCs were brought close together, as shown in Fig. 5.

3.2 Structural analysis

A stress and deformation analysis was performed for the modified design of the coil support structure as shown in Fig. 2. An isotropic solid element, with a Young's modulus of 200 GPa and a Poisson's ratio of 0.3, was assigned as the support structure. There are three candidate superconductors, as described in 2.1. A gas-cooled HTS type was used as the coil section in the structural model assuming that the coils were wound by using joint-winding method. The physical properties of the superconductor regions were calculated by a homogenization analysis [13].

Consequently, the maximum von Mises stress in the coil support structure was 764 MPa. The stress distribution and the maximum deformation region are shown in Fig. 6. Although the maximum stress seems to exceed the permissible stress of S.S. (e.g., 316LN) [14], the soundness of it could be guaranteed since it appeared at the root of rib and was considered to be 1st and 2nd order stress. The spatial stress level was found to not exceed 400 MPa, which was low enough for the permissible stress. A maximum deformation of 25 mm appeared in the outer VFC region near the outer port. The normal strain along the coil winding, which is the evaluation criteria of a superconductor, was below 0.16%. The shear stress in the HC, NITA coil, and VFC were 35, 55, and 25 MPa, respectively, as shown in Fig. 7. An insulating material in the NITA coil could be severe for a shear strength of an existing insulating material [13].

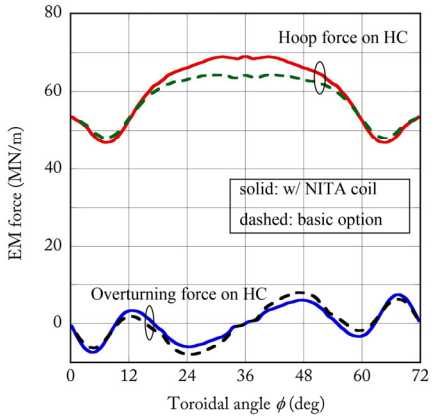

Fig. 3. EM force at each coil cross section of the HC.

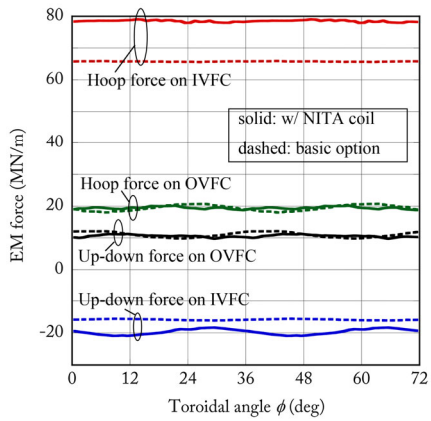

Fig. 4. EM force at each coil cross section of the VFCs.

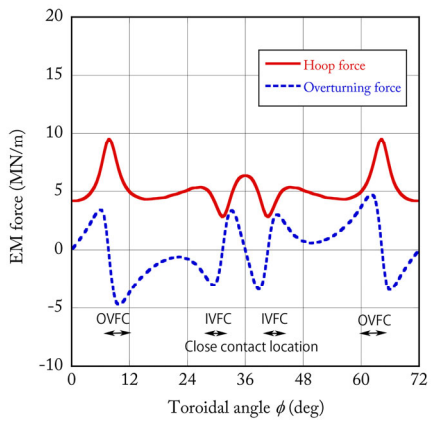

Fig. 5. EM force at each coil cross section of the NITA coil (OVFC: outer VFC, IVFC: inner VFC).

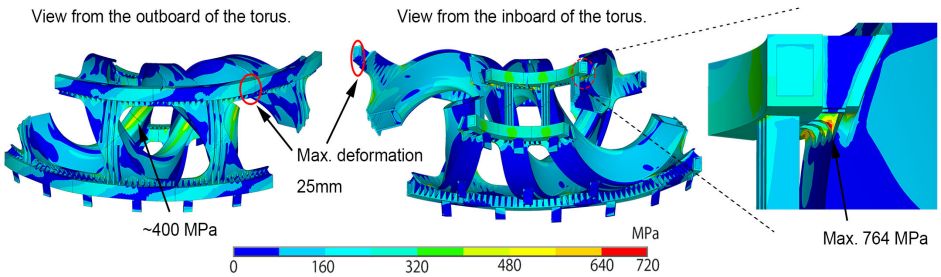

Fig. 6. Von Mises stress distribution in the coil support structure and the maximum deformation region.

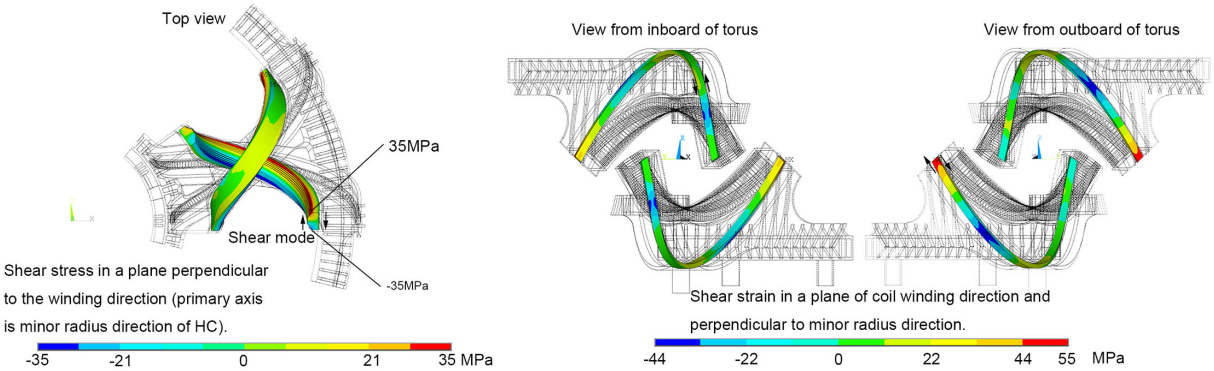

Fig. 7. Shear stress distribution in the HC (left) and the NITA coil (right).

4. Conclusion

In parallel with the FFHR-d1 fundamental design (basic option), several challenging options that has potentials to solve issues in the basic option have been proposed. These options will accelerate the design activity and achieve a consistent helical reactor system. The structural designs of these options were investigated by this study, and the following results were derived: adding a NITA coil pair would lead to an increase in the EM force as it requires the current in the HC and VFCs to be increased. Although the stress level would be severe, it would still be in the acceptable range. By modifying the coil support structure, it would be possible to adopt the challenging options outlined in this paper.

Acknowledgments

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References

[1] A. Sagara, et al., Design activities on helical DEMO reactor FFHR-d1, Fusion Eng. Des. 87 (2012) 594–602.

[2] A. Sagara, et al., Helical reactor design FFHR-d1 and c1 for steady state DEMO, Fusion Eng. Des. 89 (2014) 2114–2120.

[3] S. Imagawa, et al., Concept of magnet system for LHD-type reactor, Nucl. Fusion 49 (2009) 075017.

[4] K. Takahata, et al., A cooling concept for indirectly cooled superconducting magnet for the fusion reactor FFHR, Plasma and Fusion Res. 10 (2015) 3405011.

[5] N. Yanagi, et al., Design and development of high-temperature superconducting magnet system with joint-winding for the helical fusion reactor, Nucl. Fusion 55 (2015) 053021.

[6] S. Ito, et al., Fundamental investigation on tensile characteristics of a mechanical lap joint of REBCO tapes, IEEE Trans. Appl. Supercond., Vol. 25, no. 3 (2015) 4201205.

[7] T. Goto et al., Design Window Analysis for the Helical DEMO Reactor FFHR-d1, Plasma Fusion Res. 7 (2012) 2405084.

[8] N. Yanagi et al., NITA Coil — Innovation for Enlarging the Blanket Space in the Helical Fusion Reactor, Plasma Fusion Res. 11 (2016), 2405034.

[9] J. Miyazawa et al., 1st IAEA Technical Meeting on Divertor Concepts, P-7 (2015) http://www-naweb.iaea.org/napc/physics/meetings/TM49934.html.

[10] H. Tamura et al., Novel divertor design to mitigate neutron irradiation in the helical reactor FFHR-d1, Fusion Eng. Des., vol. 98/99 (2015) 1629–1633.

[11] H. Tamura et al., Design Status of the Structural Components of the Helical Fusion Reactor FFHR-d1, Plasma Fusion Res. 11 (2016) 2405061.

[12] T. Tanaka, et al., Analysis of radiation environment at divertor in helical reactor FFHR-d1, Fusion Eng. Des. 89 (2014) 1939–1943.

[13] H. Tamura, et al., Multiscale Stress Analysis and 3D Fitting Structure of Superconducting Coils for the Helical Fusion Reactor, IEEE Trans. Appl. Supercond. Vol. 26, no. 3 (2016) 4202405.

[14] ITER structural design criteria for magnet components (SDC-MC), Inst. Tech. Educ. Res. (ITER), Naka, Japan, N11 FDR 5001-07-05 R 0.1 (2001).