Design Status of the Structural Components of the Helical Fusion Reactor FFHR-d1*)

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FFHR-d1 is a conceptual design of the Large Helical Device-type fusion reactor. Several design optimizations for FFHR-d1 have been conducted under a multipath strategy. The structural design began from a radial build design of the components using a system code analysis and considering a shielding/breeding blanket design. The support structure must be sufficiently rigid to hold a constant geometric position of the radial build components. Additionally, large apertures are required for the maintenance work of in-vessel components. The shape of the support structure was carefully chosen and the analytically determined stress level was within the permissible limit. Thus, the basic design of the structural components of FFHR-d1 was established. To accelerate the design activity and achieve a consistent helical reactor system, the high-temperature superconducting joint-winding, liquid metal divertor, a supplemental helical coil addition, a novel divertor structure, and other challenging options are investigated alongside the basic design. This paper describes the structural design status of FFHR-d1 by focusing on the latest radial build design, construction scheme, and effectiveness of the novel divertor structure.

Keywords: helical reactor, superconducting magnet, structural analysis, in-vessel component, cryostat, 3D printing

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1. Introduction

FFHR is a conceptual design of a helical fusion reactor being studied at the National Institute for Fusion Science in Japan. Originally, FFHR was conceptualized as a force-free-like configuration of helical coils (HCs) for reducing the electromagnetic (EM) force. The name originally stands for Force Free Helical Reactor. At the beginning of the design study, three HCs were incorporated and the reduced EM force was expected to simplify the support structure and enlarge the blanket space. After optimizing the size and magnetic field, the number of HCs was reduced to two, as in the Large Helical Device (LHD). In the FFHR2m1 design, the size was increased and the magnetic field was decreased to provide blanket space. Simultaneously, the geometry of the vertical field coils (VFCs) was modified to reduce the magnetic stored energy. We then advanced to FFHR2m2, which has similar coil geometry to LHD's but a larger size and higher current density [1].

During the design study of FFHR2m2, the design window analysis was performed using the system design code HELIOSCOPE [2]. Under the design parameters (magnetic stored energy, W_{mag}, distance between the plasma surface and the bottom of the HC, $\Delta_{\mathrm{c\text{-}p}}$, the beta enhancement factor, f_β, neutron wall load, Γ_{nw}, and edge density limit), the major radius and central magnetic field were surveyed. Figure 1 shows the resulting design window analysis. The design point was chosen from the area satisfying less than 160 GJ stored magnetic energy, and less than 1.5 MW/m^2 neutron wall load. Within this area, we chose the point yielding the widest $\Delta_{\mathrm{c\text{-}p}}$. At this point, the major radius is 15.6 m, magnetic field is 4.7 T, and $\Delta_{\mathrm{c\text{-}p}}$ is 890 mm. These specifications underlie the FFHR-d1 design [3].

The design activity has been progressing through three design rounds. The first round establishes the fundamental specifications and parameters, the second round determines the three-dimensional (3D) structural design details, and the third round plans the construction/maintenance scenario. After establishing the design parameters of FFHR-d1 in the first design round, we introduced a multipath strategy. For this purpose, there are several flexible design options: a basic 3D design with a modified aspect ratio (FFHR-d1A), an increased magnetic-field design that eases the plasma demands (FFHR-d1B), and an optimized vertical field coil configuration that reduces the magnetic stored energy (FFHR-d1C). A sub-ignition version FFHR-c1 is also available for "before demo", compact, and com-

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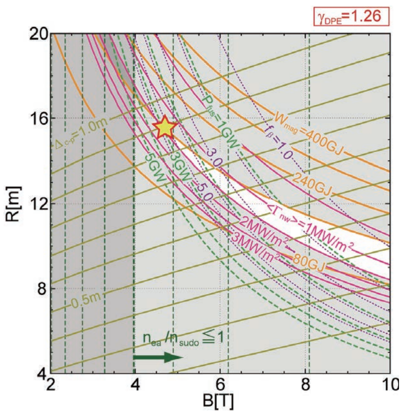

Fig. 1 Design window of the helical reactor [2].

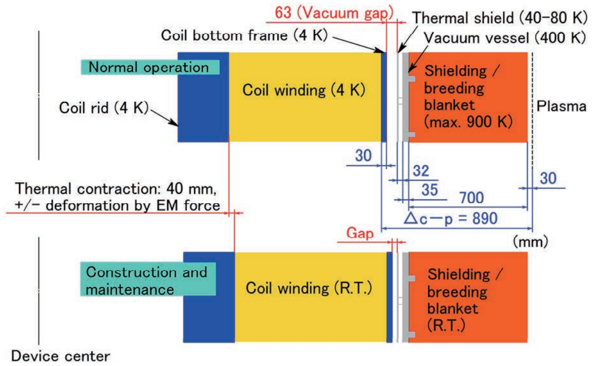

Fig. 2 Radial-build design [6].

ponent tests [4].

Based on the FFHR-d1A specifications, the second round of the design activity (the 3D component design) was started along with research and development activities. However, critical problems remain, such as the winding method of the huge structure, high heat flux and neutron irradiation on the divertor, and narrow radial build clearance. Some of the proposals for solving these challenges are introduced in the present paper.

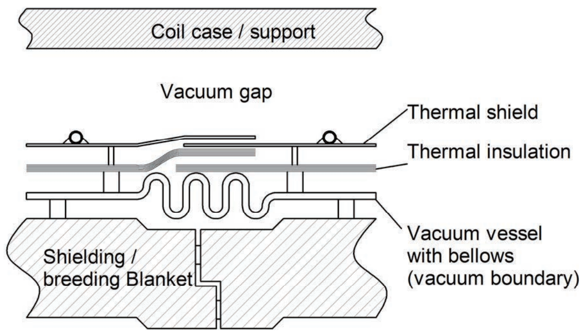

Fig. 3 Schematic of structure between the blanket and the coil in the early design phase [6]. The vacuum vessel is omitted from the latest basic design, as shown in Fig. 4.

2. Structure of FFHR-d1

2.1 Radial build design

The second round of the FFHR-d1 design activity is the 3D structural design of components. The superconducting magnet system of FFHR-d1 comprises one pair of HCs and two sets of VFCs (the LHD has three sets of VFCs). Because FFHR-d1 is a fusion power plant, it requires a blanket system and full divertor system. To expand the space between the plasma boundary and helical coil, which is limited in the LHD-type configuration, the structural design was initiated from a radial build design.

The distance $\Delta_{\text{c-p}}$ between the plasma boundary and the bottom of the HC is limited, especially at the inboard of the torus. In the system code analysis, $\Delta_{\text{c-p}}$ was determined as 890 mm. Of this distance, 20 mm is assigned to the space between the plasma boundary and the first wall. The blanket space, including the shielding and breeding blanket, is estimated to occupy 700 mm with FLiBe and tungsten carbide [5]. The residual 190 mm is distributed among 35 mm of vacuum vessel (VV), a thermal shield of 32 mm, the bottom frame of the coil case (30 mm), and an adiabatic gap of 63 mm, as shown in Fig. 2 [6]. This radial build is an early-design candidate. The VV is attached to the outer surface of the blanket system, and its outer surface is covered by the thermal shield, as shown in Fig. 3 [7]. In this design, a joint section of the VV must be connected by welding within the very narrow space between the VV and the HC case. Furthermore, if a vacuum leak develops in the welding section, the welding will be difficult to repair. In the recent design modification, the shielding blanket plays the role of the VV, i.e., the vacuum condition of the plasma confinement space is realized by the shielding blanket itself. The structural frame of the shielding blanket is a basic strength member in both the early-stage and refined designs. Figure 4 schematizes the relationships among the components. Any gap between the shielding blanket sections can be welded from the inner side (plasma confinement side). In this way, an incomplete welding point can be easily repaired before setting the breeding blanket, which is attached to the inner surface of the shielding blanket.

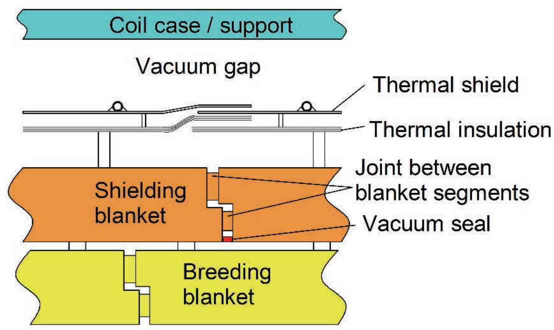

Fig. 4 Schematic of the modified structure between the blanket and the coil.

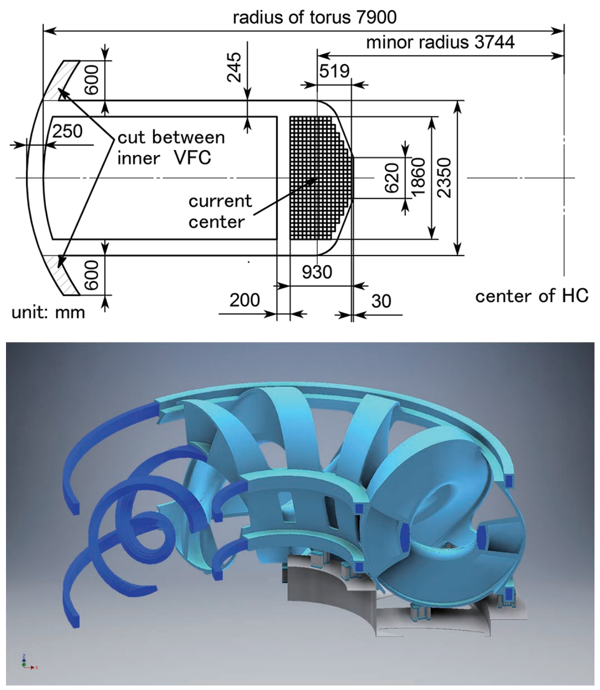

Fig. 5 Cross-section of HC perpendicular to the coil winding direction (upper), and 3D image of the magnet system (lower).

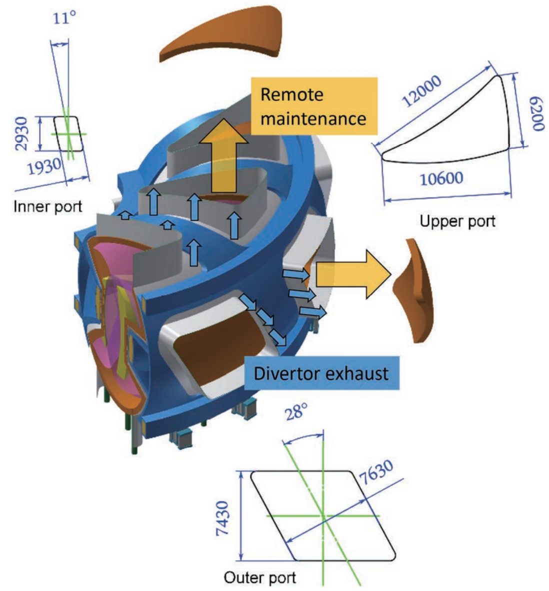

Fig. 6 Schematic of the port conceptualized for maintenance and divertor exhaust.

2.2 Coil support

The superconducting coils, whose stored magnetic energy totals 160 GJ, require a tough support structure. Moreover, the blanket and divertor systems require maintenance and replacement. To maintain the breeding blanket and divertor, the apertures of the coil support structure should be as large as possible, providing a large access port to the shielding blanket, coil support structure, and cryostat. The EM force and the mechanical behaviors of the coil and the support structure with large apertures have been analyzed [7]. To estimate the magnetic field distribution in the coil, we determined the coil cross-sectional shape and the conductor layout, as shown in Fig. 5. According to the EM force calculation, the maximum magnetic field was 12 T at the bottom region of the HC. The maximum overall EM hoop force and overturning force among the cross-sections of the HC were 64 and ± 8 MN/m, respectively.

The results of EM force calculation were then input to a stress analysis. The Young's moduli of the HC, VFC, and coil support structure were 80, 110, and 200, respectively. The Poisson's ratio of all three components was 0.3. Consequently, the maximum von Mises stress was 660 MPa, which was within the permissible limit of stainless steel 316LN [8]. The outer VFC region experienced large deformation (∼ 28 mm). The acceptable deformation level should be carefully considered along with the accuracy of the magnetic field in the plasma confinement region.

In the latest design candidate, the VV role is played by the blanket frame itself, as mentioned in 2.1. The outer surface of the blanket is set 200 mm from the coil support in order to accommodate the thermal shield and gap except at the inboard of the torus, where the radial build clearance is severe. A 750 - 850 mm-wide space is inserted around the port section, of which 500 mm is allotted to the divertor exhaust path [7]. Figure 6 is a schematic of the port with the conceptualized maintenance and divertor exhaust.

The estimated weight of the magnet system is 20,000 tons. The folded multiplate design of the LHD-type gravity support is suitable from both mechanical and thermal viewpoints [7]. Meanwhile, the blanket system weighs an estimated 35,000 tons. The blanket can be supported by legs inserted through the lower port. The adiabatic condition of the magnet system is realized by the cryostat vessel, which covers all structural components as shown in Fig. 7.

2.3 Maintenance and construction

In the third round of the design activity, we begin studying construction and maintenance schemes of the structural components. The blanket system is divided into a permanent part (i.e., the shielding blanket) and a periodically replaceable part (the breeding blanket). The maintenance scenario and a remote handling scheme are being considered under this condition. Each part occupies an approximate volume of 5,000 m^3. Drained of its coolant, the blanket loses 2/3 of its weight. One idea for replacing the breeding blanket is to divide the blanket into modules by the plane of the constant toroidal angle. Most of the modules could then be replaced through large maintenance ports using simple radial/vertical movements [9]. This procedure would simplify the maintenance tools and shorten the replacement time; this will improve the plant availability.

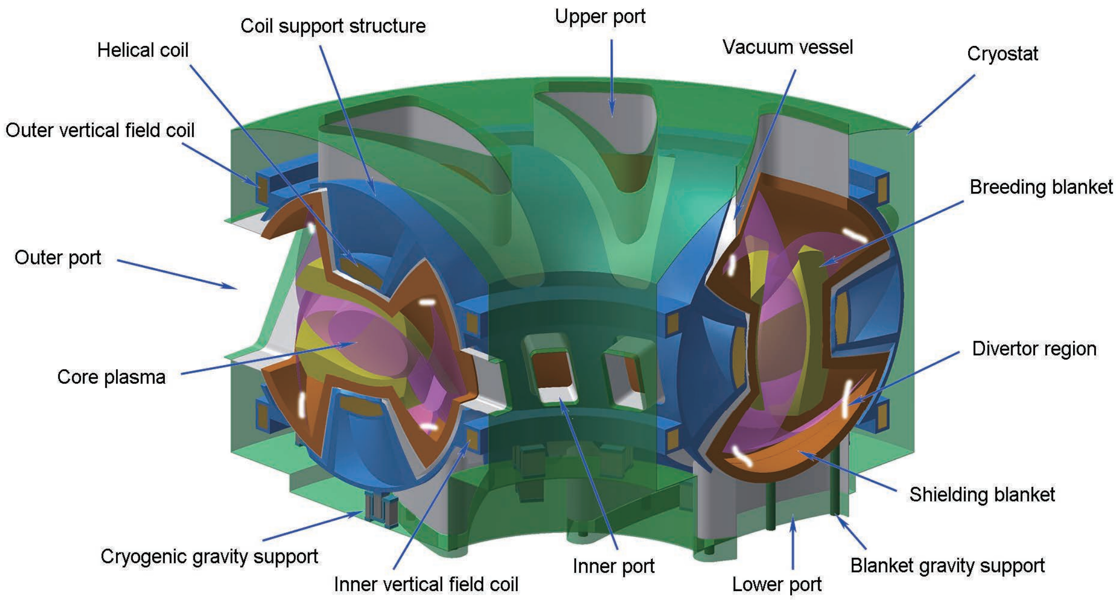

Fig. 7 General assembly of FFHR-d1.

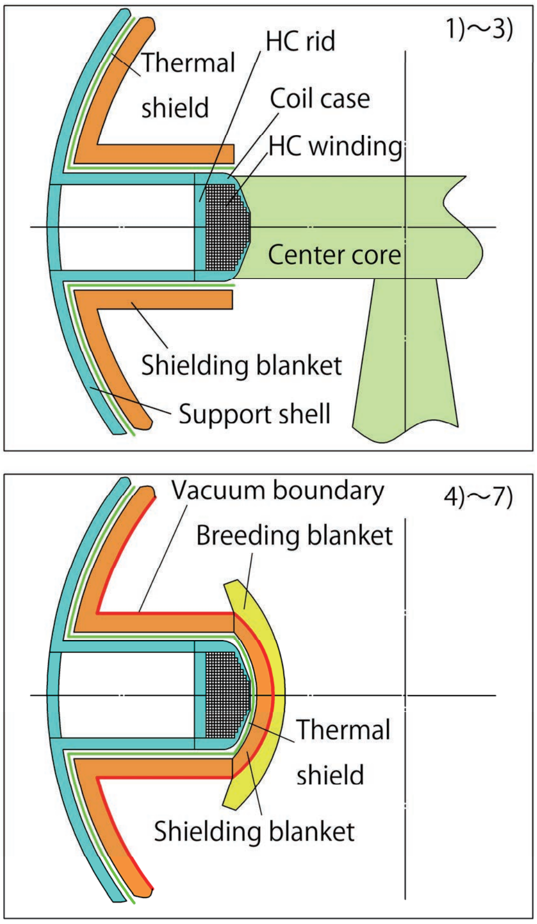

Fig. 8 Construction scheme of the HC and the shielding/breeding blanket.

Finally, the following fundamental construction scheme was proposed: 1) the HC is wound to the coil case, which is sustained by the center core, 2) parts of the shielding blanket (omitting the bottom of the HC) are set, 3) the HC rid and support shell are connected through the full torus, 4) the center core is dismantled, 5) the residual shielding blanket is placed, 6) the gaps between the shielding blanket parts are filled by welding from the plasma side (vacuum boundary), and 7) the breeding blanket is set on the inner surface of the shielding blanket. Schematics of these procedures are shown in Fig. 8.

3. Challenging Options

To solve the remaining issues, several new ideas (or challenging options) have been proposed. For example, the helical fusion reactor might be efficiently constructed using a high temperature superconductor (HTS) with joint-winding. A prototype HTS with a bridge-type mechanical lap joint successfully achieved 100 kA at 20 K and a low-resistance (1.8 nΩ) at 4 K [10, 11]. A detailed welding method for the joint section is also being investigated.

The steady-state heat load in the divertor system will exceed 20 MW/m^2 at its peak. The basic option for the divertor system is a full helical construction from tungsten and copper alloy cooled by water flow. Alternative challenging options are a novel divertor structure that mitigates neutron irradiation to the divertor [12], and a liquid metal divertor using a shower of molten tin [13]. The second design is expected to deliver high divertor maintainability, small amounts of radioactive wastes, and high permissible heat load. The novel divertor is detailed in the next section.

The $\Delta_{\text{c-p}}$ could be enlarged by setting additional HCs, named NITA coils, outside the main helical coils with opposite current flow. The NITA coils are expected to increase the $\Delta_{\text{c-p}}$ to more than 1 m without decreasing the average minor radius of the plasma [14].

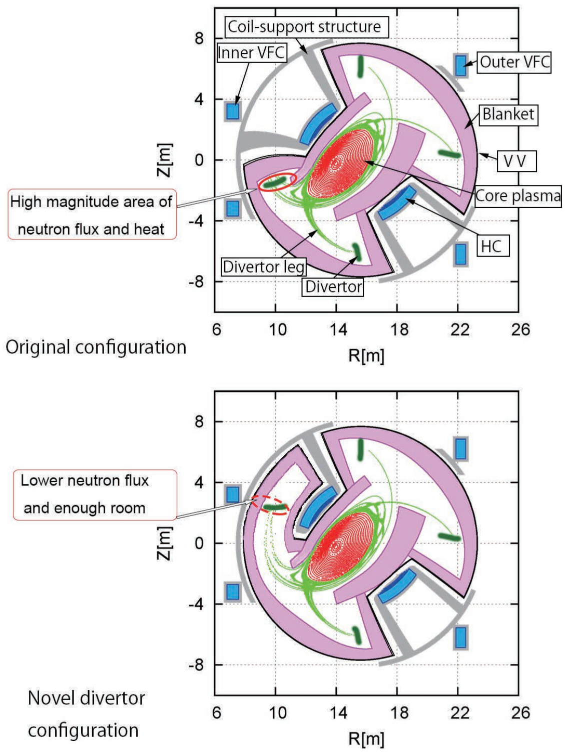

Fig. 9 Plan for relocating the divertor in the inboard region of the torus [12].

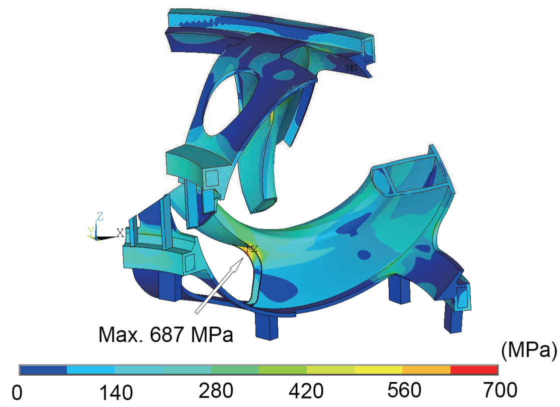

Fig. 10 von Mises stress distribution in the coil-support structure model of the novel divertor structure, determined by structural analysis calculations [12].

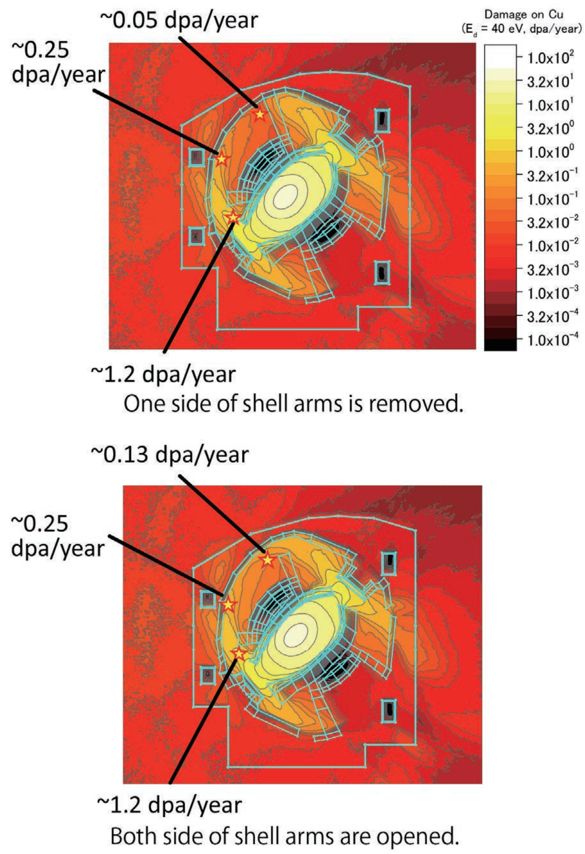

Fig. 11 Suppression of neutron flux at the helical coil backside.

3.1 Novel divertor structure

In an LHD-type fusion reactor, the neutron load on the divertor can be reduced by setting it behind the blanket. However, this design would impart high fast-neutron flux to the inboard side of the torus. The maximum irradiation damage to copper in divertor regions is estimated as 1.6 dpa/year [5]. Although the cooling pipes could be constructed from copper alloy, the irradiation damage would need to be further reduced. Partial removal of the HC arm, allowing relocation of the divertor components, has been proposed. In this plan, the shell arm section and an aperture of the coil support structure are modified from the basic design as shown in Fig. 9. The effectiveness of this novel divertor was evaluated in a stress analysis and neutronics calculation. The EM force was unchanged from that of the original structure. In the stress analysis model, the arms were removed where the HC located in the inboard of the torus, retaining the coil case and the torus shell. Figure 10 shows the resulting von Mises stress distribution. The stress level was within the permissible limit, namely, 700 MPa for FM316LNM in the ITER standard [8]. The bottom of the HC inboard of the torus was deformed by 12 mm [12].

Divertor components could be placed in the provided open space behind the HC. The neutron transport was calculated by the transport code MCNP [15] and data library JENDL-3.3 [16], as in the basic design of FFHR-d1 [5]. In the analytical model, one or both sides of the shell arms were partially removed to estimate the shielding effect. Consequently, the irradiation flux was reduced by 1/5 to 1/10, as shown in Fig. 11. From this result, the divertor lifetime was estimated as six years (assuming a limit of 1 dpa for copper alloy). Divertor parts such as the tungsten at the front, which get damaged by the high heat flux, could be replaced through nearby access ports. The divertor in the other section could be replaced every few years, together with the breeding blanket.

4. Summary

This paper reports the basic design parameters of the helical fusion reactor FFHR-d1A. Details of the structural design and construction method conforming to the specifications of FFHR-d1A are presented. A sufficiently rigid coil support with large apertures is provided for maintaining the in-vessel components, and is a distinct advantage of the LHD-type helical reactor. In parallel with the FFHR-d1A design, challenging options that will accelerate the design activity and achieve a consistent helical reactor system are being investigated. Among these are HTS joint windings, a novel divertor structure, a liquid metal divertor, and additional NITA coils.

Acknowledgements

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