Novel divertor design to mitigate neutron irradiation in the helical reactor FFHR-d1

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The heat flux at the divertor in a fusion reactor is considered to have a peak of >10 MW/m^2. In a design study of the helical reactor FFHR-d1, the feasibility of employing a copper alloy for divertor cooling pipes was investigated; however, radiation in the divertor area would quickly damage the copper alloy. The neutron load on the divertor can be reduced by a blanket arrangement; nevertheless, in the present divertor structure, irradiation damage of materials on the inboard side of the torus remains relatively high. If the divertor could be moved to an area receiving much less radiation, then the lifetimes of divertor materials should increase. In this paper, a novel divertor structure is introduced in which the coil-support structure is modified to create a region receiving relatively low amounts of radiation without changing the geometry of the helical or vertical field coils. Using this proposed design would increase the lifetime of the copper alloy in divertor components to more than an estimated six years. In addition, the divertor could be accessed from either the upper or lower sides of the device, simplifying maintenance.

Keywords: helical reactor, FFHR, divertor, irradiation, superconducting magnet, structural analysis.

1. Introduction

The FFHR-d1 is a conceptual design for the Large Helical Device-type (LHD-type) fusion reactor being developed at the National Institute for Fusion Science [1]. Several design optimizations for FFHR-d1 have been conducted under a multipath strategy [2]. For the FFHR-d1 series, the base model for three-dimensional (3D) designs has major and minor radii of 15.6 m and 3.744 m, respectively. The superconducting magnet system consists of one pair of helical coils (HCs) and two sets of vertical field coils (VFCs). The current of the single superconductor for the HC is 94 kA, and the magnetic field of the confinement center is 4.7 T. The maximum magnetic field on the HC reaches 12 T. Fig. 1 shows a schematic view of the fundamental design for FFHR-d1.

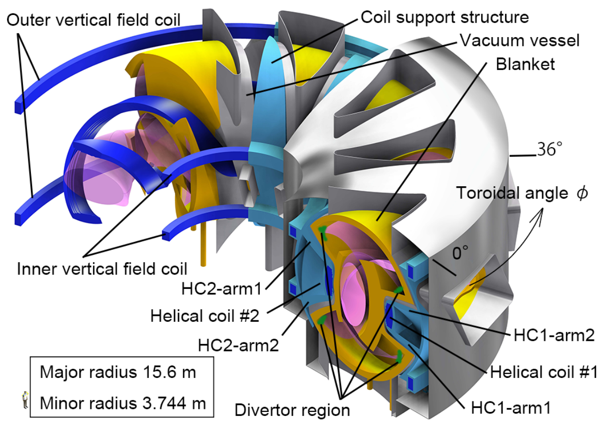

Fig. 1. Schematic of the FFHR-d1 series fusion reactor.

The divertor is a key issue in implementing a fusion reactor. The LHD-type heliotron magnetic configuration is equipped with built-in helical divertors. The divertor heat flux in the FFHR-d1 is anticipated to have a high peak of >10 MW/m^2. Thus, a high-heat-removal divertor needs to be developed. In an LHD-type fusion reactor, the neutron load on the divertor can be reduced by setting it behind a blanket to prevent direct neutron irradiation of the divertor region. This is a unique advantage of the LHD-type configuration. If neutron irradiation were sufficiently reduced, the lifetime of divertor components increases, and employing a copper alloy for cooling pipes would be feasible. This advantage might be realized by moving the divertor to an area that receives a lower level of radiation, such as behind the HC. Gourdon, et al. showed an arrangement in which the divertor was set behind the HCs in a torsatron-type reactor with a triple HC at the point of divertor exhaust [3]. However, there is no precedent for a design study that includes an assessment of mechanical feasibility for a coil-support structure, blanket, and vacuum vessel.

In this paper, we propose a novel divertor location that can mitigate neutron irradiation in the divertor region; to do so, the coil-support structure would be modified without changing the geometry of HC and VFC. The proposed design provides the divertor with enough room for mitigating heat loads, e.g., radiation loss by gas puffing. Another advantage is that, by setting an additional access port, the divertor could be accessed from the upper and lower side of the vacuum vessel; this makes maintenance work easier than having to access the divertor from a lateral side port. This paper presents a 3D design scheme, results from a structural analysis, and a maintenance concept.

2. Condition of divertor in the FFHR-d1

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2.1 Neutronics environment

In the present design of the FFHR-d1 series, divertor regions are placed behind radiation shields (blanket) to suppress irradiation damage, as shown in Fig. 1. Tanaka, et al. calculated neutron transport based on the fundamental design of the FFHR-d1 [4]. From this calculation, the irradiation damage for copper at the divertor region was found to have a displacement per atom (dpa)/yr with a wide span of one order of magnitude, depending on location, as shown in Fig. 2. Since a fundamental design of the helical coil winding in FFHR-d1 has a cyclic symmetry of 5 through the toroidal angle ϕ, and there is a phase shift of 36° between the geometrical position of HC1 and that of HC2, only the result for HC1 in range of $\phi = 0$ to 72° is shown. The trends beside the HC1-arm1 and beside the HC1-arm2 were symmetric about the toroidal angle $\phi =$ 36°. This result can be adapted to the divertor region at HC2 by shifting the data by 36°. Consequently, the irradiation damage would be relatively high when the HC is on the inboard side of the torus. A maximum irradiation damage in the divertor region gave 1.6 dpa/yr. At this irradiation level, use of a copper alloy for divertor cooling pipes is nearly feasible. However, the copper alloy limit is supposedly below 1 dpa assuming that the copper alloy is a kind of oxide dispersion strengthened copper (ODS-Cu) with a temperature of 350 °C [5, 6]. So if a copper material is to be used, a further reduction in irradiation damage or a scheme for easy exchange of parts has to be developed.

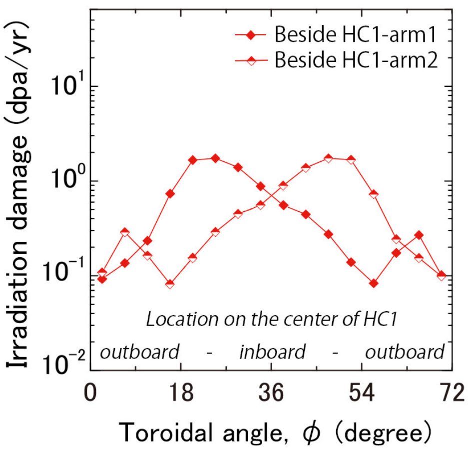

Fig. 2. Irradiation damage for a copper at the divertor region as functions of toroidal angle at 3 GW fusion output power [4]. The locations of HC1-arm1, arm2, and toroidal angle ϕ are shown in Fig. 1.

2.2 Heat flux estimation

The LHD-type heliotron magnetic configuration is equipped with built-in helical divertors protruding from the edge of plasma with clearly defined four legs (not as islands), with which relatively low divertor heat flux is expected on average. For FFHR-d1, the total wetted area of divertor footprints is crudely estimated as ~70 m^2 (~90 m long for the whole four legs each having 80 mm width) and the average heat flux could be as low as 8 MW/m^2 on average for 3 GW fusion power generation without assuming radiation dispersion. Although a detail design of the divertor is under consideration, a fundamental design would be similar to the divertor design in ITER, e.g. tungsten monoblock armor with cooling pipe. However, there is toroidal asymmetry that causes a nonuniform heat flux distribution along the divertor leg. This toroidal asymmetry and the helical divertor footprint have been investigated by Yanagi, et al. [7, 8]. Fig. 3 shows the results of their calculations for the FFHR-d1 configuration, which counts the number of footprints in the magnetic field-line tracing. The figure shows strong asymmetry with toroidal angle and peaks at the inboard side of the torus; $\phi =$ 22° for the divertor beside the HC1-arm1 and at $\phi =$ 50° beside the HC1-arm2. Fig. 3 suggests that the divertor components located at these peaks would experience more than ten times higher heat flux than the average value, which requires continuous realization of plasma detachment to secure high radiation dispersion of heat flux. These results for both neutron transport and heat flux distribution indicate that the divertors located at the inboard side of the torus would be under severe conditions.

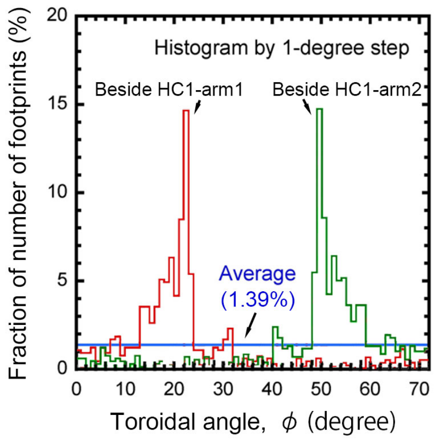

Fig. 3. Histogram for footprints of magnetic field lines entering the divertor region of the FFHR-d1 [8].

3. Modification of coil-support structure

3.1 Design outline

The geometric positions of the coils in the FFHR-d1 are similar to corresponding positions in the LHD. To mitigate not only neutron irradiation but also the high heat load inboard of the torus, we considered a modification of the coil-support structure. Fig. 4 shows a plan for changing the location of the divertor in the region inboard of the torus. By shifting the divertor to an area of lower radiation, the lifetimes of divertor components should increase. However, the vacuum vessel is limited by the coil-support structure, which consists of the coil case, arm, and a torus-shaped shell, as shown in Fig. 5. The arms connect the coil case to the torus-shaped shell. An idea to solve this problem is that the arms would be partially removed to allow divertor components to be moved to an environment having low irradiation of neutrons.

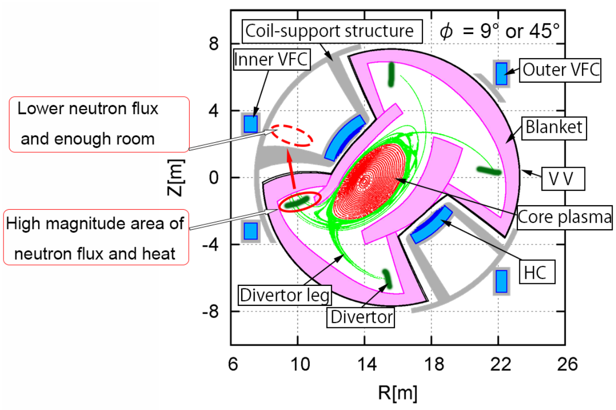

Fig. 4. Plan for changing the divertor location in the region inboard of the torus.

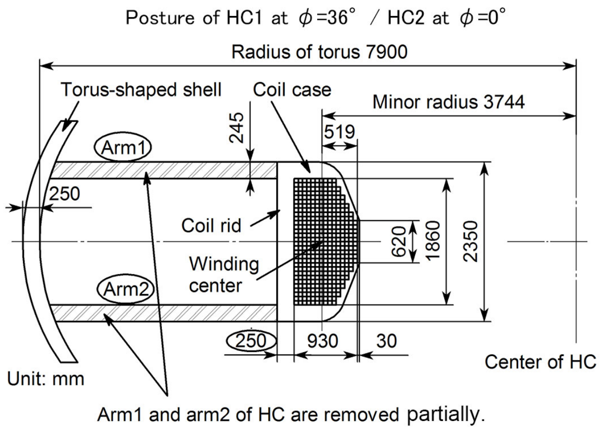

Fig. 5. Cross section of HC perpendicular to the winding direction and removed regions of HC arms.

Based on the results from the neutron transport and heat flux calculations, we made a conceptual model of the coil-support structure; in this model, the arms were removed when the center of the HC winding was in the toroidal angles range of ϕ = 20° to 52° (for HC1-arm1, 2), ϕ = -16° to 16° (for HC2-arm1, 2), and every 72° considering the cyclic symmetry. The coil case and the torus shell remained throughout the torus, the thickness of the coil rid section increased from 200 mm [9] to 250 mm, and support stays were added between the inner VFCs to maintain rigidity.

3.2 Structural analysis

In the modified structure, the electromagnetic (EM) force induced by the HCs and VFCs were the same as those in the fundamental design of the FFHR-d1 [9]. The maximum hoop force and the overturning force for each cross section of the HC were 64 and ±8 MN/m, respectively. A stress analysis was performed on the modified fundamental structure, as shown in Fig. 6. Because of the cyclic symmetry, only a 36° region of the structure was needed for the analytic model. In this analysis, the superconductor was assumed to be made of a high-temperature superconducting (HTS) conductor using a rare-earth barium copper oxide (REBCO) tape [10, 11]. The equivalent physical properties of the HC and VFC winding sections were calculated by a homogenization analysis using the geometry of the cross section and physical properties of the constituent materials [11]. Young's modulus and Poisson's ratio for the stainless steel (SS), which was the structural material for the coil-support structure, were assumed to be 200 GPa and 0.3, respectively. Here only the EM force was taken into account by following reason; (1) there is a thermal radiation shield between the coil-support structure and the vacuum vessel, (2) the coil-support structure is cooled by cooling pipes attached on the surface of the coil-support structure. The heat load to the coil-support structure caused by a radiation load from the thermal radiation shield and nuclear heating could be removed by the cooling pipes. The shape of the coil-support structure was rearranged so that the maximum von Mises stress would be within an acceptable value.

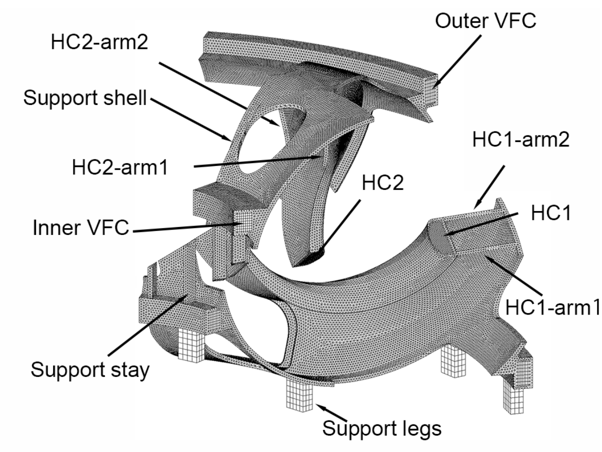

Fig. 6. 3D analytic model of the modified structure, including support legs and support stay.

Fig. 7 shows the resulting von Mises stress distribution and the amount of deformation. A maximum stress of 687 MPa appeared in the removed arm region. Although the maximum stress increased compared to that in the previous design, the stress level was within the permissible limit for the SS, e.g., 700 MPa for FM316LNM in the ITER standard [12]. A maximum deformation of 23.5 mm occurred on the upper part of the removed HC arm region. Deformation of the bottom of the HC inboard of the torus, where the radial build is critical, was 12 mm. The direction of this deformation was opposite to that in the previous fundamental design. In this case, the gap between the bottom of the HC and the surface of the thermal shield during maintenance periods could be 10 mm.

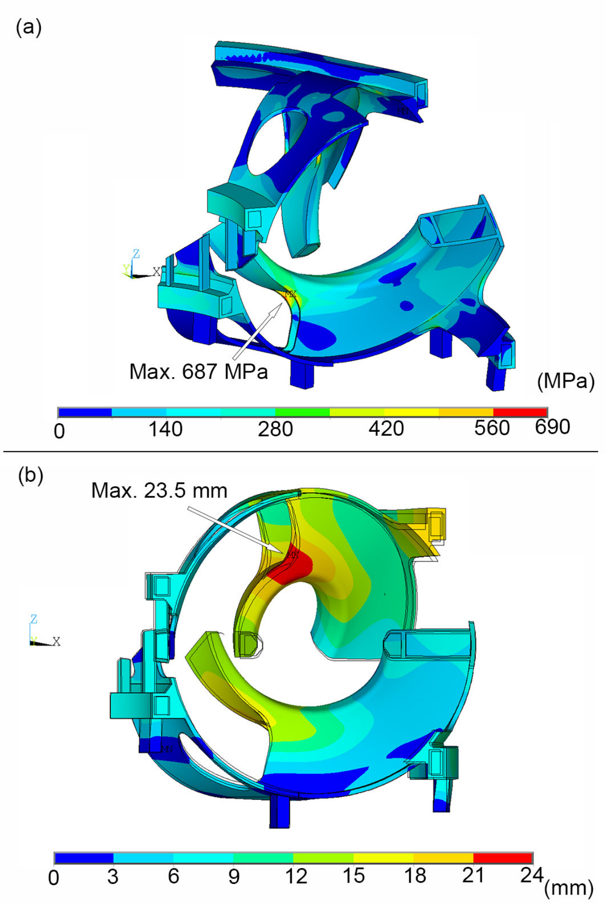

Fig. 7. Results from structural analysis calculations: (a) von Mises stress distribution in the coil-support structure. (b) Amount of deformation in the coil-support structure.

By this modification of the coil-support structure, the vacuum vessel and the blanket could occupy the removed space of the coil-support structure, keeping an adiabatic gap of 200 mm. Divertor components could be placed in this open space.

4. Discussion

In the operation and maintenance of a fusion reactor, the frequency of replacing parts is a very important issue. If a copper alloy is employed for divertor cooling pipes without the modification proposed in this paper, the divertor would have to be replaced every seven months, assuming that the limiting radiation damage for copper alloy is below 1 dpa. However, use of the proposed modification would reduce the damage to divertor. If the divertor at the modified position is sufficiently covered by a neutron shield, the damage will be reduced by one order of magnitude according to the neutron stream distribution analyzed in [4]. This corresponds to a decrease in irradiation damage of copper alloy to less than 0.16 dpa/yr, and the lifetime of copper alloy in the divertor is estimated to be more than six years.

Even if the neutron irradiation issue were resolved, some divertor parts, such as the tungsten plate in front, would be damaged by the high heat flux. An access port should be prepared so that damaged divertor parts could be replaced frequently. Fig. 8 shows a conceptual maintenance scheme using a port for divertor components that are inboard of the torus. These divertor parts could be exchanged through this port directly from the top or bottom of the device. The divertor in the other section could be replaced, together with the first wall or with the breeding blanket, every several years.

We note that the peak divertor heat flux could be flattened by optimizing the vertical field profile (especially the quadrupole component) [8]. However, also in this case, enlargement of the divertor room at the inboard side of the torus is an important issue to be explored.

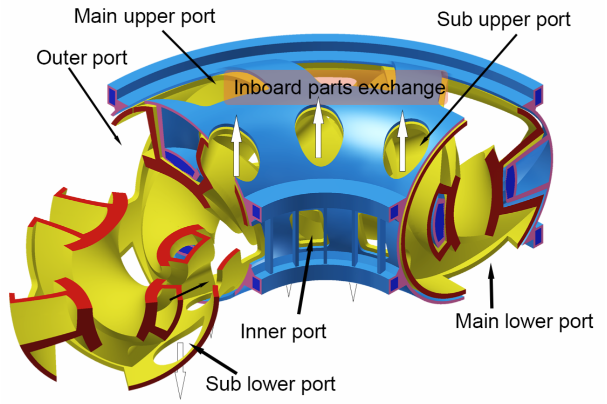

Fig. 8. Cross-section drawing of the blanket with the coil-support structure. Proposed positions of maintenance ports for divertor components inboard of the torus are shown.

5. Conclusions

To mitigate neutron irradiation in the divertor region, we have investigated a modification of the coil-support structure. A novel divertor structure was designed for the FFHR-d1 without changing the geometry of the HC and VFC. The design guarantees a sound coil-support structure and a sound 3D structure, including the blanket and vacuum vessel. In the new design, irradiation damage in divertor material decreases and lifetimes of those materials are expected to be longer. The novel divertor design also provides enough room for mitigation of the heat load and easy access for maintenance.

Acknowledgments

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