Design Integration toward Optimization of LHD-type Fusion Reactor FFHR

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Recent activities on optimizing the base design of LHD-type helical reactor FFHR2m1 is presented. Three candidates to secure the blanket space are proposed in the direction of reactor size optimization without deteriorating α-heating efficiency and with taking cost analyses into account. For this direction the key engineering aspects are investigated; on 3D blanket designs, it is shown that the peaking factor of neutron wall loading is 1.2 to 1.3 and the blanket cover rate over 90% is possible by proposing Discrete Pumping with Semi-closed Shield (DPSS) concept. Helical blanket shaping along divertor field lines is a next big issue. On large superconducting magnet system under the maximum nuclear heating of 200W/m^3, CICC and alternative conductor designs are proposed with a robust design of cryogenic support posts. On access to ignited plasmas, new methods are proposed, in which a long rise-up time over 300 s reduces the heating power to 30 MW and a new proportional-integration-derivative (PID) control of the fueling can handle the thermally unstable plasma at high density operations.

Keywords: helical reactor, blanket, COE, nuclear heating, superconducting magnet, ignition

1. Introduction

On the basis of physics and engineering results established in the LHD project [1], conceptual designs of the LHD-type helical reactor FFHR have made continuous progress from 1991 [2-5], aiming at making clear the key issues required for the core plasma physics and the power plant engineering, by introducing innovative concepts expected to be available in this coming decades. Those design activities have led many R&D works with international collaborations in broad research areas [6].

Due to inherent current-less plasma and intrinsic diverter configuration, helical reactors have attractive advantages, such as steady operation and no dangerous current disruption. In particular, in the LHD-type reactor design, the coil pitch parameter γ of continuous helical winding can be adjusted beneficially to reduce the magnetic hoop force (Force Free Helical Reactor: FFHR) while expanding the blanket space, where $\gamma=(ma_c)/(lR_c)$ with a coil major radius R_c, a coil minor radius a_c, a pole number l, and a pitch number m.

As a key feature of helical reactors, the blanket space directly couples to the helical coils configuration as well as the core plasma performances under physics and engineering key constrains. Therefore, as the second step after concept definition of the initial FFHR1 (l=3) design [2], optimization studies have begun on the reactor size, based on the LHD-type (l=2, m=10) compact design FFHR2 (γ=1.15, R_c=10 m) [3]and modified FFHR2m1 (γ=1.15 and outer shifted plasma axis, R_c=14 m) and FFHR2m2 (inward shifted plasma axis, R_c=17 m) [4].

This paper presents recent activities on optimizing FFHR2m1 as a base design to make clear key issues, mainly focusing on blanket space, neutronics performance, large superconducting magnet system, and plasma operation.

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Table 1. Design parameters of helical reactor

Design parameters	Symbol	Unit	LHD	FFHR2	FFHR2m1	FFHR2m2
Polarity	l		2	2	2	2
Field periods	m		10	10	10	10
Coil pitch parameter	γ		1.25	1.15	1.15	1.25
Coil major Radius	R_c	m	3.9	10	14.0	17.3
Coil minor radius	a_c	m	0.98	2.3	3.22	4.33
Plasma major radius	R_p	m	3.75	10	14.0	16.0
Plasma radius	a_p	m	0.61	1.24	1.73	2.80
Plasma volume	Vp	m^3	30	303	827	2471
Blanket space	Δ	m	0.12	0.7	1.1	1.15
Magnetic field	B_0	T	4	10	6.18	4.43
Max. field on coils	B_{max}	T	9.2	14.8	13.3	13.0
Coil current density	j	MA/m^2	53	25	26.6	32.8
Magnetic energy		GJ	1.64	147	133	118
Fusion power	P_F	GW		1	1.9	3
Neutron wall load	Γ_n	MW/m^2		1.5	1.5	1.3
External heating power	P_{ext}	MW		70	80	100
α heating efficiency	η_α			0.7	0.9	0.9
Density lim.improvement				1	1.5	1.5
H factor of ISS95				2.40	1.92	1.76
Effective ion charge	Z_{eff}			1.40	1.34	1.35
Electron density	$n_e(0)$	10^19 m^{-3}		27.4	26.7	19.0
Temperature	$T_i(0)$	keV		21	15.8	16.1
Plasma beta	$\langle\beta\rangle$	%		1.6	3.0	4.1
Plasma conduction loss	P_L	MW			290	463
Diverter heat load	Γ_{div}	MW/m^2			1.6	2.3
Total capital cost		G$(2003)		4.6	5.6	6.9
COE		mill/kWh		155	106	87

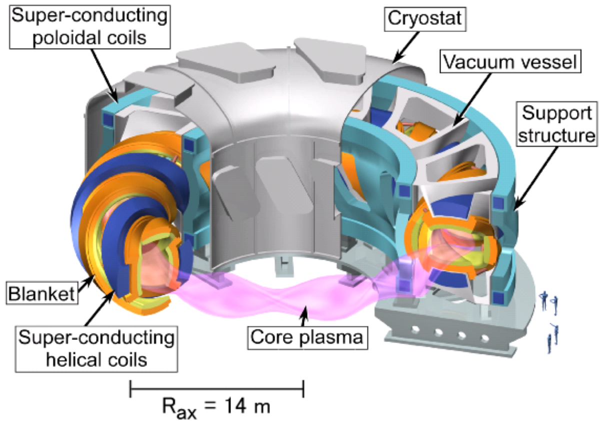

Fig.1. The 3D illustration of the FFHR2m1.

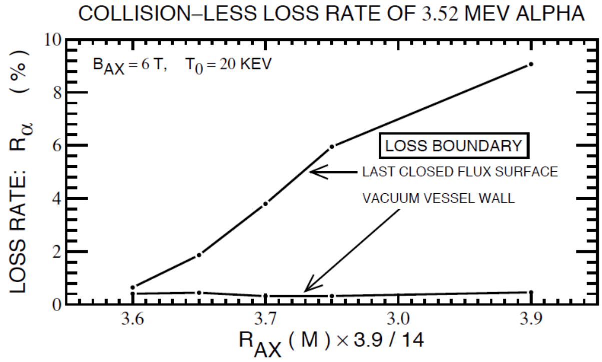

Fig.2 The loss rate of alpha particle in the FFHR2m1, where the two cases on loss boundary are shown as a function of the position of the magnetic axis R_{ax}.

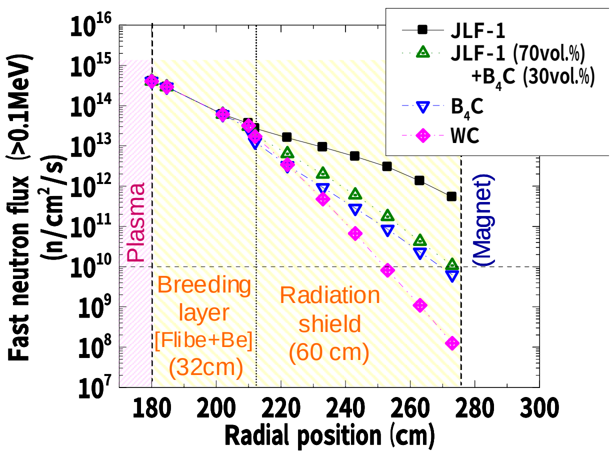

Fig.3 Distributions of first neutron flux (> 0.1 MeV) in the standard FFHR blanket, depending on materials composition of the radiation shield.

2. Candidates to secure the blanket space

The design parameters of FFHR2 are listed in Table 1, which newly includes the recent results of cost evaluation based on the ITER (2003) design [7]. Figure 1 shows the 3D view of the FFHR2m1. In this base design, one of the main issues is the structural compatibility between blanket and divertor configurations. In particular, the blanket space at the inboard side is still insufficient due to the interference between the first walls and the ergodic layers surrounding the last closed flux surface. To overcome this problem, helical x-point divertor (HXD) has been proposed to remove the interference [8]. In this concept, very effective screening of recycling neutrals with 99% ionization is expected according to 3D simulations [9].

From the point of view of a-heating efficiency over 0.95, the importance of the ergodic layers has been found by collisionless orbits simulation of 3.52MeV alpha particles as shown in Fig.2 [10]. Therefore, three alternatives without adopting HXD are considered. One is to reduce the shielding thickness only at the inboard side. Fig. 3 shows that the WC can reduce about 0.2 m in the shield thickness in comparison to the standard FFHR design with B_4C and JLF-1 [11]. The second is to improve the symmetry of magnetic surfaces around the magnetic axis, without shifting the magnetic axis inward, by increasing the current density at the inboard side of the helical coils while decreasing at the outboard side. Modulation of the current density can be practically obtained by splitting the helical coils [12, 13]. The third is to increase the reactor size. In this case, as shown in Fig.4, it is expected that there is an optimum size around R_c of

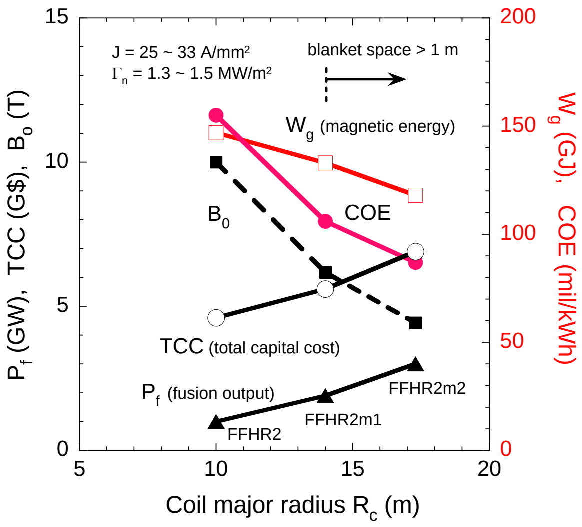

Fig. 4 R_c dependences of the fusion output P_f, the total capital cost (TCC), magnetic field B_0 at the plasma center, cost of electricity (COE) and magnetic energy W_g under almost same conditions on neutron wall loading □ and current density J on helical coils.

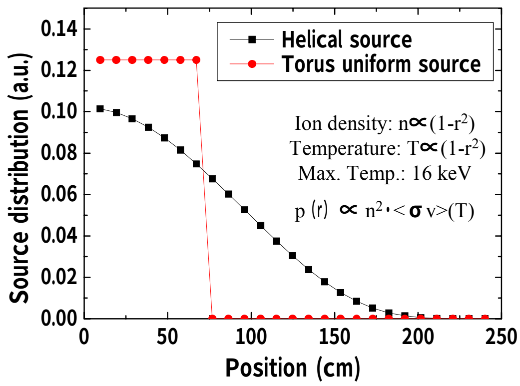

Fig. 5 Neutron source distributions used for neutronics calculations in the FFHR2m1.

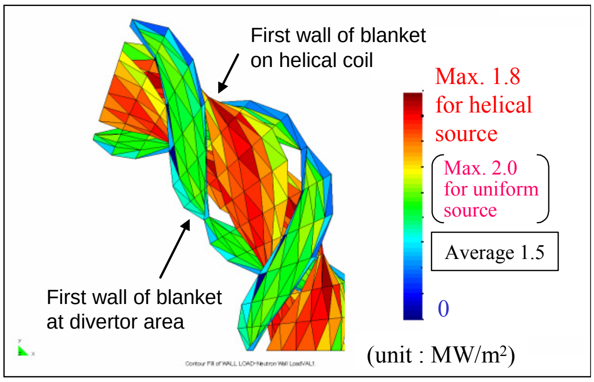

Fig.6 The 3D top view of calculated neutron wall loading distributions. (72° of the torus)

15m by taking into account the cost of electricity (COE), the total capital cost, and engineering feasibility on large scaled magnets. More detailed and integrated optimization, by selecting or mixing those three candidates, is one of key next issues.

3. Progress and issues on 3D blanket designs

In the direction of optimizing neutronics performances, the 3D distribution of neutron wall loading is basically important. Using the recently developed 3D neutronics calculation system for non-axisymmetric helical systems as LHD [14], two cases of neutron sources have been compared as shown in Fig.5: one is a centralized torus uniform source which represents a peaked plasma profile, the other is a helical source which comes from a typical parabolic distributions of plasma density and temperature in the elliptic cross section with the long and short radii of 2.4 m and 1.8 m, respectively. Under the averaged neutron wall loading of 1.5 MW/m^2, the maximum loading for the uniform and helical source to be 2 MW/m^2 and 1.8 MW/m^2, respectively, at the first wall of blankets on the helical coils as shown in Fig.6 [14]. Therefore the peaking factor is estimated to be 1.2 to 1.3.

The FFHR blanket designs have been improved to obtain the total TBR over 1.05 for the standard design of Flibe+Be/JLF-1 and long-life design of Spectral-shifter and Tritium Breeding (STB) blanket [4, 14] by enhancement of the blanket cover rate to 80%. More increase of the cover

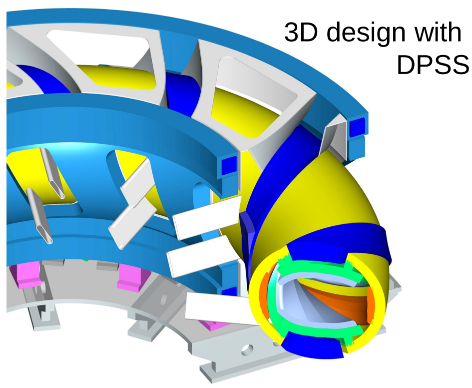

Fig.7 Discrete Pumping with Semi-closed Shield (DPSS) concept, where the helical divertor duct is almost closed with partly opened at only the discrete pumping ports.

rate over 90% will be effectively possible by a new proposal of Discrete Pumping with Semi-closed Shield (DPSS) concept as shown in Fig.7, where the helical divertor duct is almost closed with partly opened at only

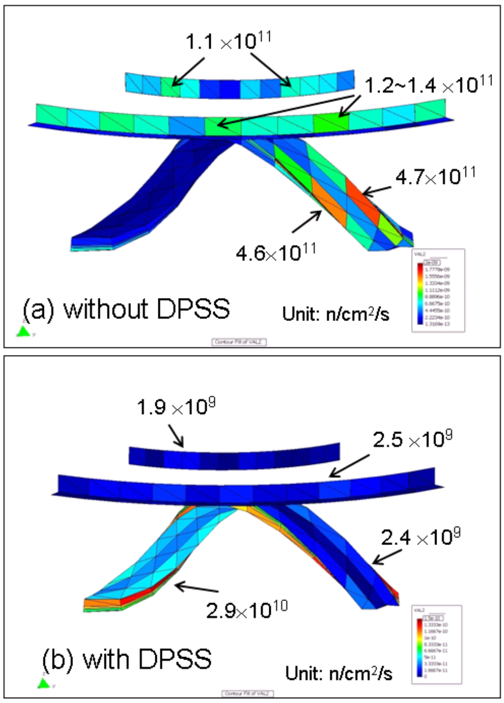

Fig.8 The first neutron fluxes at the poloidal and helical coils (a) without and (b) with the DPSS, where the flux at the rear side of helical coils is high in (a) and one order reduced in (b).

the discrete pumping ports. This DPSS is very important not only to increase the total TBR over 1.2 but also to reduce the radiation effects on magnets. In fact, as shown in Fig.8, the first neutron fluxes at the poloidal coils just out side the divertor duct and at the side of the helical coils are successfully reduced to the acceptable level lower than 1E22 n/m^2 in 40 years. The total nuclear heating is also reduced from 250kW to 40kW, which means the cryogenics power to be about 12MW and acceptable level below 1% of the fusion output.

When the HXD is not adopted as mentioned in the previous section, the blanket design in divertor area should be largely modified, because the intrinsic divertor null point deeply intersects the blanket [8]. Figure 9 shows the tentatively modified design, where the radial position of the divertor area blanket is moved outward. In this case the location of poloidal coils should be also moved in the minor radial direction, resulting in about 11 % increase of the magnetic energy [15]. It is noted in Fig.9 that re-adjustment of helical blanket shaping should be a next big issue by taking into account the blanket space, distribution of wall loading and divertor field lines.

4. Base design of large superconducting magnet system

The base design for the FFHR2m1 superconducting

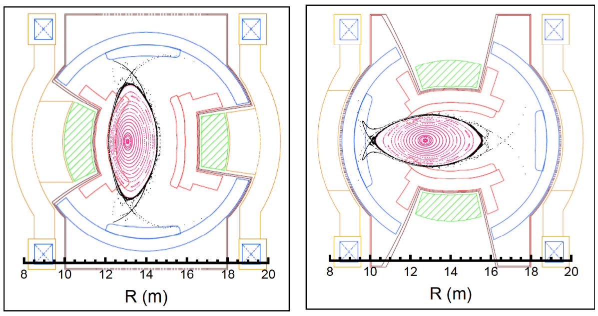

Fig.9 The tentatively modified design to avoid intersection of divertor null points with blankets, where an inward shifted plasma configuration at γ=1.15 is selected as a reference case for optimization.

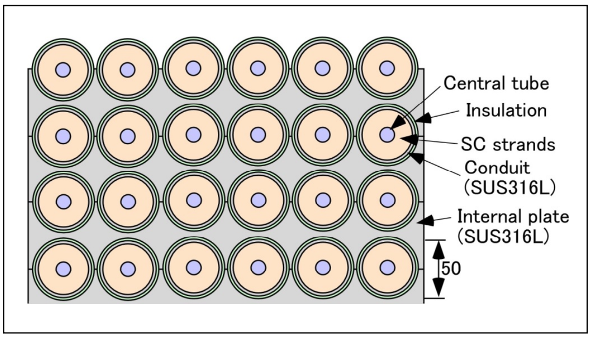

Fig. 10 Concept of helical winding with CIC conductors of current 90 kA with Nb_3Al strands, where the maximum length of a cooling path is about 500 m.

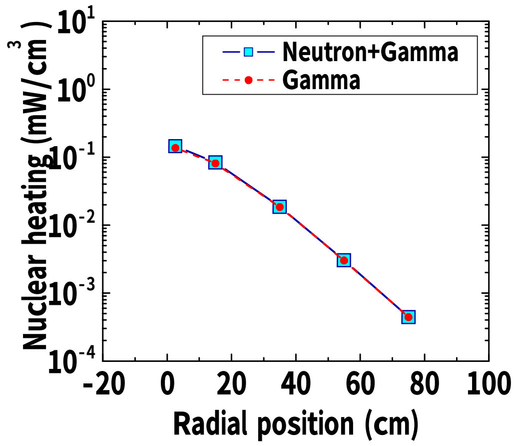

Fig.11 The nuclear heating distribution calculated on the uniformed torus model of FFHR helical coils shown in Fig.10.

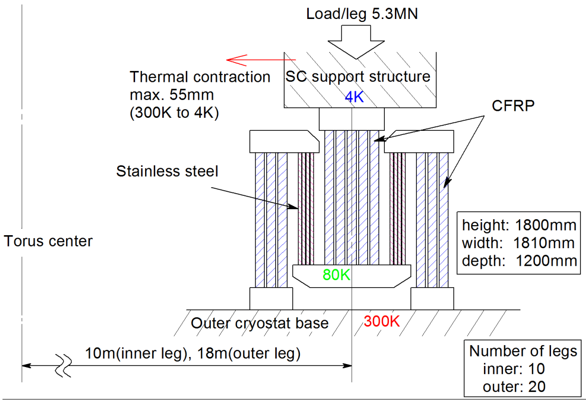

Fig.12 The present design of cryogenic support posts for the FFHR2m1, adopting the same type of the LHD support post.

magnet system has been preliminary proposed [16] on the engineering base of ITER-TF coils as a conventional option. Figure 10 shows the cross sectional structure of continuous helical coil, where the magnet-motive force of helical coils is about 50 MA and the cable-in conduit conductors (CICC) of current 90 kA with Nb_3Al strands are wound in the grooves of the internal plates. In this concept, react and wind method is preferred to use conventional insulator and to prevent huge thermal stress. The maximum length of a cooling path is about 500 m that is determined by the pressure drop for the required mass flow against the nuclear heat of 1000 W/m^3. This value has a 5 times margin of the maximum nuclear heating calculated on the FFHR helical coils as shown in Fig.11, in which the gamma-ray heating is dominant and the maximum is about 200 W/m^3.

Advanced concepts for the FFHR magnet system is of importance as alternative candidates. "Indirect cooling" is promising, because it solves the issue of the pressure drop. The preliminary design using Nb_3Sn has been proposed for the FFHR helical coils, where a conventional quench protection circuit using an external resistor is employed by dividing the coil into several subdivisions [17].

The total weight of the coils and the supporting structure exceeds 16,000 tons. This weight is supported by cryogenic support posts which are set on a base plate of a cryostat vessel. Fig.12 shows the present design of the post [18] adopting the same type of the LHD support post, which is a folded multi plates consisted of Carbon Fiber Reinforcement Plastic (CFRP) and stainless steel plates. The FEM analyses indicate that the LHD-type support post is also valid for the large sized device such as FFHR in mechanical and thermal points of view. The modal and dynamic response analysis using typical earthquake vibrations are the next issue for design optimization.

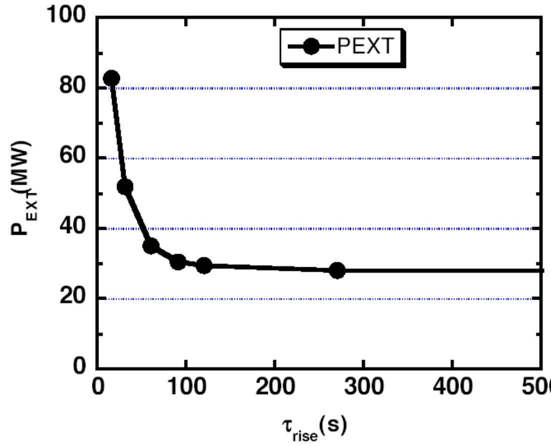

Fig.13 The maximum feedback controlled heating powers to reach self-ignition for various fusion power rise-up times in FFHR2m1.

5. New proposals on access to ignited plasmas

Minimization of the external heating power to access self-ignition is advantageous to increase the reactor design flexibility and to reduce the capital and operating costs of the plasma heating device in a helical reactor. While the fusion power rise-up time in a tokamak depends on the OH transformer flux or the current drive capability, any fusion power rise-up time can be employed in a helical reactor, because the confinement field is generated by the external helical coils. It has been recently found that a lower density limit margin reduces the external heating power, and over 300 s of the fusion power rise-up time can reduce the heating power from such as 100 MW to minimized 30 MW in FFHR2m1 as shown in Fig.13 [19].

A new and simple control method of the unstable operating point in FFHR2m1 is proposed for the ignited operation with high-density plasma [20]. Proportional-integration-derivative (PID) control of the fueling has been used to obtain the desired fusion power with the fusion power error of $e(P_f)=(P_{fo}-P_f)$ in the stable operating point. It has been discovered that in the unstable regime the error of the fusion power with an opposite sign of $e(P_f)= - (P_{fo}-P_f)$ can stabilize the unstable operating point. Around the unstable operating point, excess fusion power supplies fueling and then increases the density and decreases the temperature. Less fusion power in the sub-ignited regime reduces the fueling, decreases the density, and increases the temperature. The operating point approaches the final unstable operating point as oscillation is damped away.

6. Summary

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On access to ignited plasmas, using the advantage of current-less plasma, new methods are proposed, which are a long rise-up time over 300 s to reduce the heating power to 30 MW and a new proportional-integration-derivative (PID) control of the fueling to handle the thermally unstable plasma at high density operations.

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