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Design Integration of the LHD-type Energy Reactor FFHR2 towards Demo

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Three candidates to secure the blanket space are proposed in the direction of reactor size optimization without deteriorating a-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. 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, conceptual designs of the LHD-type helical reactor FFHR have made continuous progress from 1991 [1, 2]. Those design activities have led many R&D works with international collaborations in broad research areas [3, 4].

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 g 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.

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. 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.

From the point of view of a-heating efficiency over 0.95, the importance of the ergodic layers has been found by collision-less orbits simulation of 3.52MeV alpha

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Table 1. Design parameters of helical reactor

Design parameters			LHD	FFHR2	FFHR2m1	FFHR2m2	SDC
Polarity	l		2	2	2	2	
Field periods	m		10	10	10	10	
Coil pitch parameter	¬		1.25	1.15	1.15	1.20	
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	4.16	
Plasma major radius	R_p	m	3.75	10	14.0	16.0	
Plasma radius	$<a_p>$	m	0.61	1.24	1.73	2.35	
Plasma volume	Vp	m^3	30	303	827	1744	
Blanket space	¬	m	0.12	0.7	1.1	1.05	
Magnetic field	B_0	T	4	10	6.18	4.84	
Max. field on coils	B_{max}	T	9.2	14.8	13.3	11.9	
Coil current density	j	MA/m^2	53	25	26.6	26	
Magnetic energy		GJ	1.64	147	133		
Fusion power	P_F	GW		1	1.9	3	
Neutron wall load	$¬_n$	MW/m^2		1.5	1.5	1.5	
External heating pow	P_{ext}	MW		70	80	43	100
¬ heating efficiency	¬¬			0.7	0.9	0.9	0.9
Density lim.improvement				1	1.5	1.5	7.5
H factor of ISS95				2.40	1.92	1.92	1.60
Effective ion charge	Z_{eff}			1.40	1.34	1.48	1.55
Electron density	$n_e(0)$	10^19 m^{-3}		27.4	26.7	17.9	83.0
Temperature	$T_i(0)$	keV		21	15.8	18	6.33
Plasma beta	¬⇔ℜ	%		1.6	3.0	4.40	3.35
Plasma conduction lo	P_L	MW			290	453	115
Diverter heat load	$¬_{div}$	MW/m^2			1.6	2.3	0.6
Total capital cost		G$(2003)		4.6	5.6	7.0	
COE		mill/kWh		155	106	93	

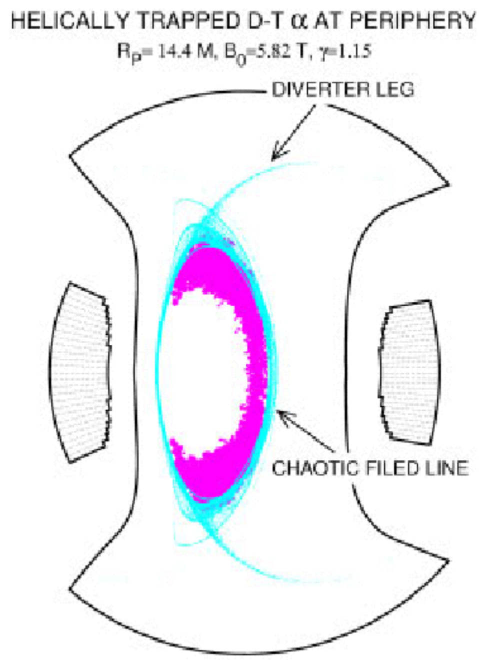

Fig.1 Poincare plot of helically trapped a particles (magenta dots) and the chaotic field lines (sky blue dots).

particles as shown in Fig.1. Therefore, the reactor size is increased. In this case, as shown in Fig.2, it is expected that there is an optimum size around R_c of 15m by taking into account the cost of electricity (COE), the total capital cost, and engineering feasibility on large scaled magnets.

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. Under the averaged neutron wall loading of 1.5 MW/m^2, the maximum loading for the uniform and helical source are 2 MW/m^2 and 1.8 MW/m^2, respectively, at the first wall of blankets on the helical coils. 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 [2, 3] by enhancement of the blanket cover rate to 80%. More increase of the cover rate over 90% is effectively possible by a new proposal of Discrete Pumping with Semi-closed Shield (DPSS) concept as shown in Fig.3, where the helical divertor duct is almost closed with partly opened at only 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.4, 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

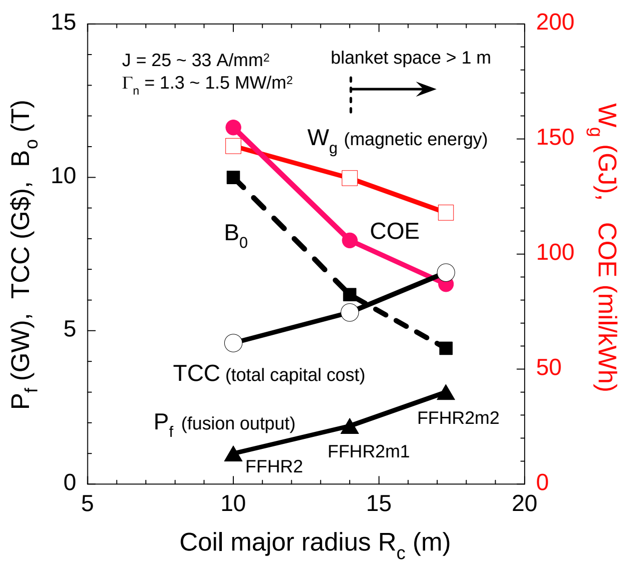

Fig. 2 R_c dependences of the fusion output P_f, the total capital cost (TCC), magnetic field B_0 at the plasma center, cost of electricity (COE) and magnetic energy W_g under almost same conditions on neutron wall loading Γ and current density J on helical coils.

level lower than $1x10^{22}$ 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.

4. Base design of large superconducting magnet system

The base design for the FFHR2m1 superconducting magnet system has been preliminary proposed on the engineering base of ITER-TF coils as a conventional option, where the magnet-motive force of helical coils is about 50 MA and the cable-in conduit conductors (CICC) of current 90 kA with Nb3Al 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, in which the gamma-ray heating is dominant and the maximum is about 200 W/m^3.

The total weight of the coils and the supporting structure exceeds 16,000 tons. As shown in Fig.5, this weight is supported by cryogenic support posts by 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. Gravity per support is 16,000 ton / 30 legs ~ 530 ton, thermal contraction < max. 55 mm, and the total heat load to 4K is about 0.34 kW which is 1/20 of the case of stainless steel post.

The modal and dynamic response analysis using typical earthquake vibrations are the next issue for design optimization.

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

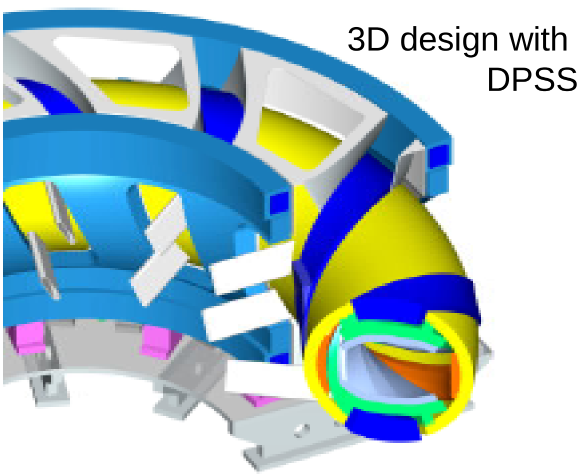

Fig.3 Discrete Pumping with Semi-closed Shield (DPSS) concept, where the helical divertor duct is almost closed with partly opened at only the

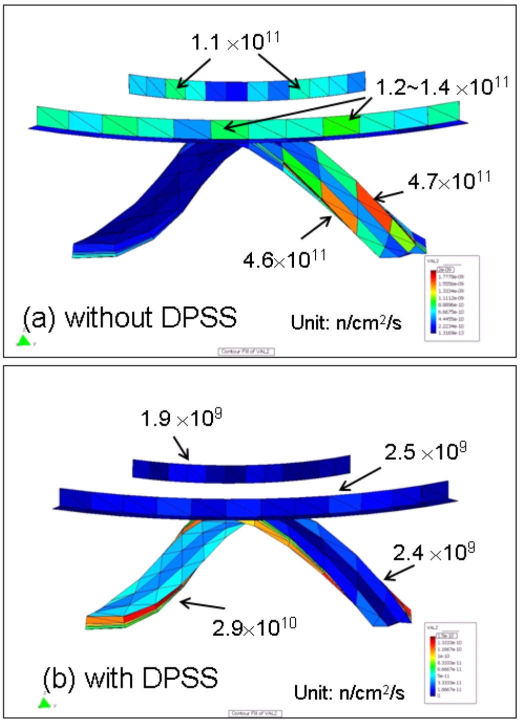

Fig.4 The first neutron fluxes at the poloidal and helical coils (a) without and (b) with the DPSS, where the flux at the rear side of helical coils is high in (a) and one order reduced in (b).

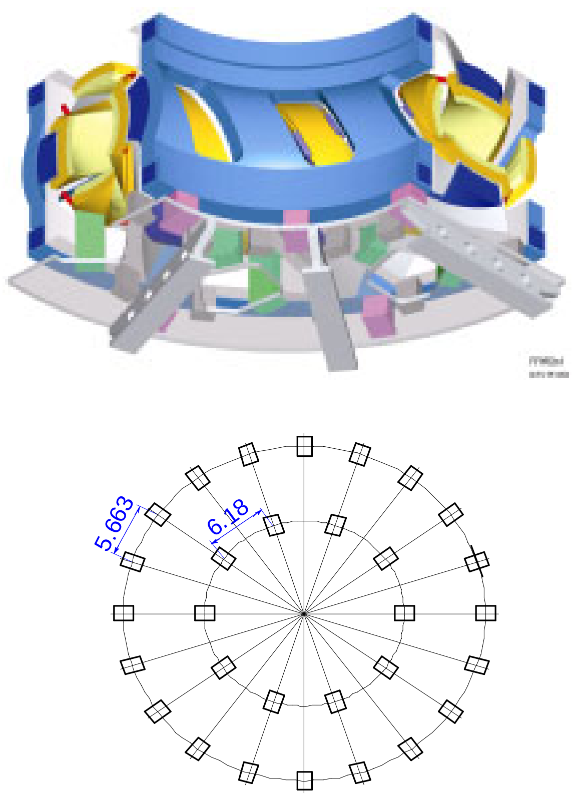

Fig.5 The cryogenic support posts of FFHR2m1.

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 [5].

A new and simple control method of the unstable operating point in FFHR2m1 is proposed for the ignited operation with high-density plasma [6], as demonstrated in Fig.6, where a new proportional-integration-derivative (PID) control of the fueling has been used to obtain the desired fusion power.

6. Summary

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 a-heating efficiency and with taking cost analyses into account.

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 designs of 500 m cooling path and 90 kA with Nb3Al strands and alternative Indirect cooling Nb3Sn conductor designs are proposed with the LHD-type robust design of cryogenic support posts.

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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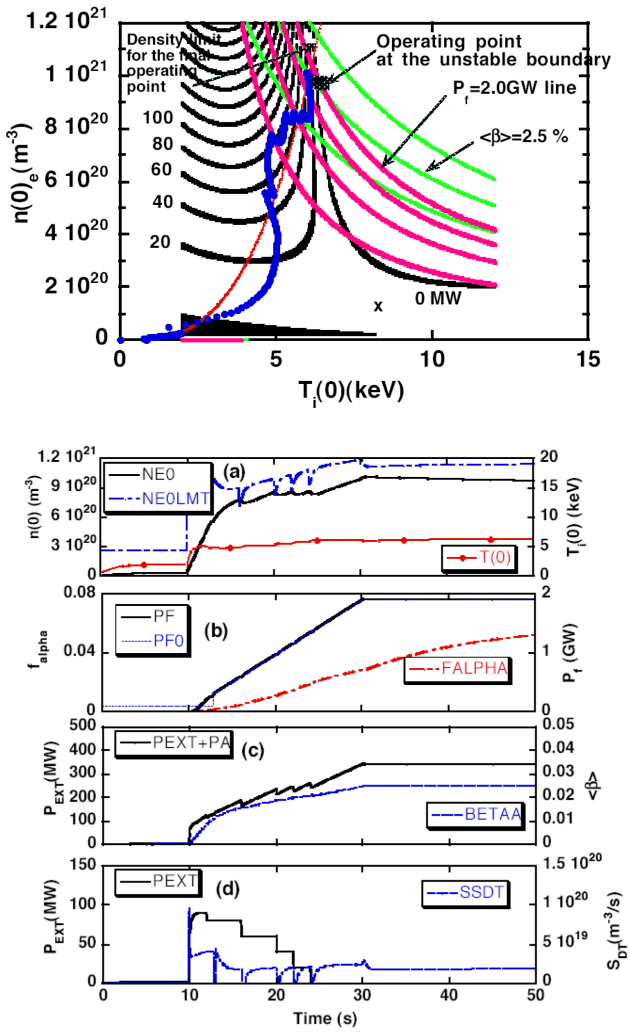

Fig.6 Temporal evolution of plasma parameters in FFHR2m1 at the thermally unstable boundary of high density operations, and the POPCON diagram.