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Blanket design using FLiBe in helical-type fusion reactor FFHR

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Abstract

The blanket design for a force-free helical reactor (FFHR) is presented, which is a demo-relevant heliotron-type D–T fusion reactor based on the first all-superconducting-coils device, LHD (large helical device) under construction in NIFS at present. For the goal of a self-ignited reactor of 3 GW thermal output, the design parameters at the first stage for concept definition of FFHR have been investigated. The main feature of FFHR is a force-free-like configuration of helical coils, which makes it possible to simplify the coil supporting structure and to use a high magnetic field instead of high plasma beta. The other feature is the selection of molten-salt FLiBe as a self-cooling tritium breeder for mainly safety reasons owing to the low tritium inventory, low reactivity with air and water, low pressure operation, and low MHD resistance compatible with a high magnetic field. In particular, as common issues in fusion reactors, the FLiBe blanket system in FFHR is expressed in detail by showing engineering possibilities to overcome key issues on tritium permeation, material corrosion, heat transfer, operation pressure, etc. The basic design for maintenance and repair of the blanket is also discussed. © 1997 Elsevier Science B.V.

1. Introduction

The heliotron-type large helical device (LHD) is the first all-superconducting (SC) magnet system, which is now at the 7th year stage of an 8 years project and under construction at Toki, the new site of NIFS [1,2]. Based on the great amount of R&D results on both of physics and engineering in this project, a preliminary attempt on a demo relevant helical-type fusion reactor design has been started to make clear key issues required for power-plant engineering including materials development and to introduce innovative concepts expected to be available in a coming few decades on the supposition of starting construction of the reactor from 2015 and power generation from 2025 [3–6].

Since no plasma current is needed in the helical-type reactor, there are many advantages:

(1) steady-state operation with only external-coils current;

(2) plasma operation with no dangerous current disruptions;

(3) no need to spend a recirculating power to drive plasma current;

(4) natural divertor configuration inherently introduced with helical-coils.

In addition to these, there is one more attractive feature in a continuous-coil system like LHD,

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(5) force-free-like configuration regarding electromagnetic force on helical coils,

which makes it possible to simplify the coil supporting structure [7] or to use high magnetic field instead of high plasma beta. By including the case of high magnetic field as one of the new promising designs in the present work, a wide variety of force-free-like helical reactor (FFHR) designs has been studied at the first stage for concept definition. Cost estimation and design optimization are planned in the next stage in the present study of phase I.

Besides the main feature of the force-free-like coils configuration, the blanket design is the other main feature in FFHR. As preliminary reported in Ref. [3], molten-solt FLiBe has been selected for a D–T reactor blanket material for mainly safety reasons. In this paper, since the blanket design is one of common issues in fusion reactors, the recent progress in the FLiBe blanket and thermofluid system design for FFHR is expressed in detail by showing engineering possibilities to overcome key issues on tritium permeation, material corrosion, heat transfer, operation pressure, etc. The basic design for maintenance and repair of the blanket is also discussed.

2. Reactor parameters in the blanket design

The main specifications of FFHR-1 are listed in Table 1 in comparison with LHD. Ignition conditions were analyzed by using the LHD scaling of plasma energy confinement time τ_{LHD} and density limit [8]. The $l = 3$ system shown in Fig. 1 is adopted to obtain a practical force-free-like coils configuration. Reduction of the magnetic force gives three attractive merits: (1) simplification of coil supporting structures which gives a wide open area for the maintenance of in-vessel components, (2) use of high magnetic fields leading to a some margin in the plasma beta, $\langle \beta \rangle$, and requiring a less-severe enhancement factor for the energy confinement time, and (3) widening of the coil-to-plasma clearance for the blanket and shield space as described below.

Table 1
LHD and FFHR-1 design parameters

	LHD	FFHR-1		
		case A	case B	case C
Plasma parameters				
Number of poles, l	2	3		
Toroidal pitch number, m	10	18		
Major radius, R (m)	3.9	20		
Av. plasma radius, $\langle a_p \rangle$ (m)	< 0.65	2		
Fusion power, P_f (GW)	–	3		
External heating power, P_{ex} (MW)	< 20	100		
Toroidal field on axis, B_0 (T)	4	12	7	5
Average beta, $\langle \beta \rangle$ (%)	> 5	0.7	2.2	4.5
Enhancement factor of τ_{LHD}		1.5	2.25	3.5
Plasma density, $n_e(0)$ (m^{-3})	1×10^{20}	2×10^{20}	1.9×10^{20}	1.5×10^{20}
Plasma temperature, $T_e(0)$ (keV)	> 10	22	24	29
Effective ion charge, Z_{eff}		1.5		
Alpha heating efficiency, η_α	–	0.7		
Alpha density fraction, f_α	–	0.05		
Engineering parameters				
Av. helical coil radius, $\langle a_c \rangle$ (m)	0.975	3.33		
Pitch parameters, $\gamma_c = m\langle a_c \rangle/(lR)$	1.25	1		
Coil modulation, α	+0.1	0		
Coil to plasma clearance, Δ (m)	0.03	1.1	1.25	1.3
Coil current, I_H (MA/coil)	7.8	66.6	38.9	27.8
Coil current density, J (A/mm^2)	(53)	27		
Max. field on coils, B_{max} (T)	(9.2)	16	11.5	10
Stored energy with poloidal coils (GJ)	1.64	1290		
Neutron wall loading, P_n ($MW/^2$)	–	1.5		
Av. heat load on divertor, P_d (MW/m^2)	< 10	1.6		
Blanked material	–	FLiBe (40 vol.%) + Be (40 vol.%)		
Operation temperature	–	inlet 723 K/outlet 823 K		
T breeding ratio (TBR)	–	1.1		
SC material	NbTi	Nb_3Al or $(NbTi)_3Sn$		

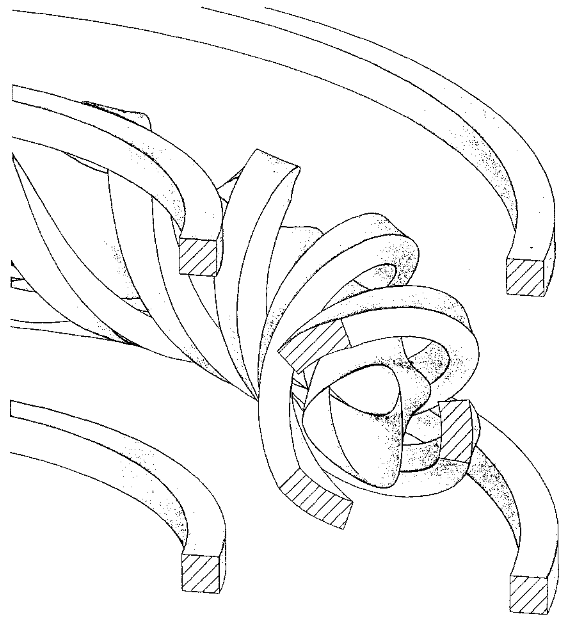

Fig. 1. Coils configuration with the last closed magnetic surface of FFHR.

From the design window shown in Fig. 2, the ignition case A at the magnetic field $B_0 = 12$ T and the major radius $R = 20$ m is almost optimum as far as three constraints are concerned: the maximum perpendicular field $B_{\perp max}$ in helical coils below 15 T, the coil-to-plasma clearance Δ over 1 m needed for blanket and shield, and the enhancement factor h_H for τ_{LHD} less than 2.

The electromagnetic force between continuously winding SC helical coils is reduced by reducing the helical pitch parameter $\gamma_c = (m/l)(a_c/R)$ as shown in Fig. 3, where the averaged minor radius hoop force on helical coils $\langle f_a \rangle$ normalized by B_0 and the coil current I_H in FFHR-1 is reduced to 35% of the value in LHD. At the

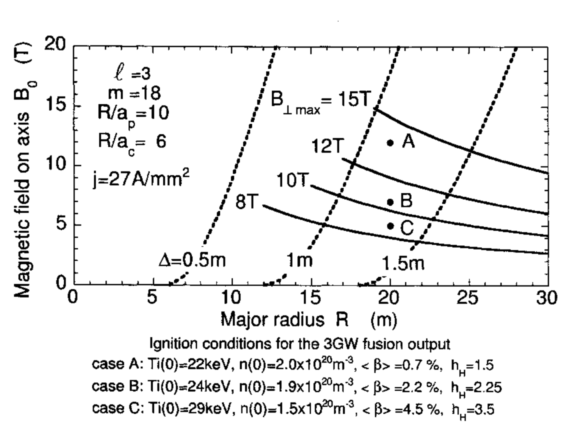

Fig. 2. The design window for B_0 and R under constraints of the maximum perpendicular field in helical coils $B_{\perp max}$, the coil-to-plasma clearance Δ, and ignition conditions.

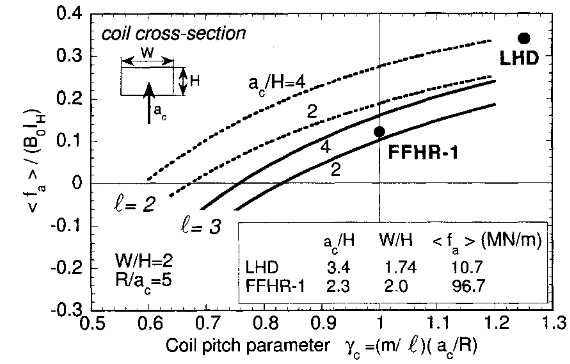

Fig. 3. The averaged minor radius hoop force on helical coils $\langle f_a \rangle$ normalized by $B_0 I_H$ as a function of coil pitch parameter γ_c.

same time, the reduction of γ_c gives rise to shrinkage of plasma minor radius with expansion of the coil-to-plasma clearance. In order to avoid too much shrinkage of plasma minor radius, we selected $\gamma_c = 1$, which gives Δ of about 1 m.

3. LiBe blanket system

3.1. Selection of FLiBe

Molten-salt FLiBe was selected as the self-cooling tritium breeder from the main reason of safety [9]: low tritium inventory, low reactivity with air and water, low pressure operation, and low MHD resistance.

The solubility of tritium in FLiBe is extremely low, which is about eight orders of magnitude lower than that of liquid Li [10], and then, two advantages are given: (1) the disengager system for T_2 gas separation can be quite simple [11], (2) in case of a drain event of coolant, the maximum amount of tritium released together with FLiBe is fairly small, approximately in the order of 10^{-3} kg. The low solubility of T_2 gas, on the other hand, leads to the permeation loss of tritium through the coolant tubing walls. So, the blanket and coolant tubes are double walled and the gap between the walls is filled with flowing He gas in order to sweep out the permeated T_2 and also to monitor drain events of FLiBe itself.

Since the chemical property of FLiBe is quite stable, the reactivity with air and water is quite low, so that the potential hazard of fire is also very small. The main issue associated with FLiBe coolant is the compatibility with structural materials. This will be discussed in a later section.

Low pressure operation is possible with FLiBe due to fairly low vapor pressure in the order of 10^{-3} Pa even at high operation temperatures around 800 K. As for the thermofluid aspect to transfer 3 GW fusion output, the total flow rate of about 7 m^3/s of FLiBe is needed. In case of 18 parallel channels with a flow diameter of 0.2 m, for instance, the pressure drop with a Reynolds number around 10^5 is estimated to be as low as about 10^{-2} MPa/m in each channel as shown in Fig. 4, that is, about 1 MPa for the total length of about 50 m with 5 elbow joints in each channel. This pressure drop requires the total pump power of only 0.8% of the fusion output P_f. Thus, the mechanical stress in the main structural materials can be kept low.

Unlike liquid metal coolants, the MHD loss is almost negligible because of its high electrical resistivity of about 10^{-2} Ω m. This property of FLiBe is quite compatible with the concept of high magnetic fields in FFHR-1, leading to the safety design with respect to operation pressure.

The coolant inlet temperature 723 K was determined from the melting temperature and viscosity of FLiBe. The melting temperatures of peritectic (33 mol% BeF_2) and eutectic (53 mol% BeF_2) FLiBe are 732 K and 637 K, respectively. The viscosity of peritectic FLiBe at 773 K is about 1 order higher than that of water at 273 K but 1 order lower than eutectic FLiBe. So the FLiBe of 40 mol% BeF_2 was selected, where the melting temperature is about 713 K and, even below this temperature, the coolant does not solidify but stays in the phase of liquid with precipitates of Li_2BeF_4.

3.2. Selection of structural materials

Aiming at a maintenance-free helical reactor regarding in-vessel components, the neutron wall loading was reduced to as low as 1.5 MW/m^2 for the 3 GW fusion output by setting the reactor major radius R at as large as 20 m. This concept will be assessed in the light of the total cost of electricity at the next stage of this work. At any rate this wall loading in the reactor lifetime of 30 years leads to the radiation induced material damage of about 450 dpa. At present, within reasonable databases, there is no reliable structural materials for the main in-vessel components up to such a severe damage condition. However, by allowing of maintenance in every 10 years, materials

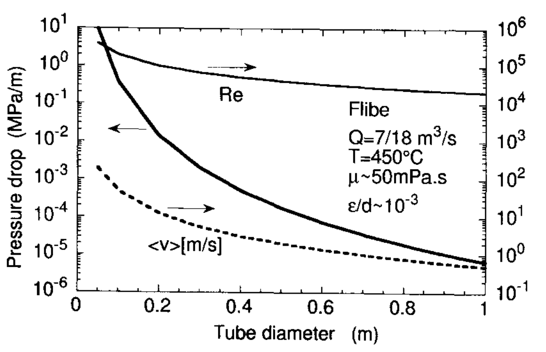

Fig. 4. FLiBe pressure drop, flow velocity and Reynolds number as a function of flow tube diameter.

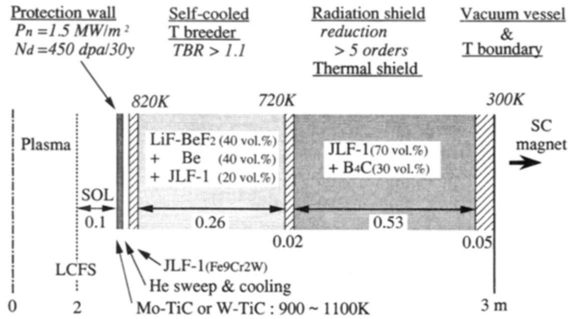

Fig. 5. The blanket and shielding structure in FFHR.

reliable up to 120 dpa are reasonably used. On the other hand, from the highest point of view of public acceptance, low induced radioactivity is the most important aspect in selecting materials. With considering engineering databases and radioactivity, a ferritic steel JLF-1 (Fe_9Cr_2W) was selected as the first candidate. Vanadium alloy ($V_4Cr_4Ti_{0.1}Si$) or ODS steel (oxide dispersion strengthened ferritic steel) are second options [6].

The coolant outlet temperature of FLiBe was determined from creep strength of JLF-1. Under conditions of creep strain less than 0.5% at 100 MPa for the lifetime of 120 dpa, JLF-1 is hopefully used at temperatures around 823 K, because the creep coefficient of this material is relatively low among similar low activation steels [12], and because the FLiBe coolant can be operated at a very low pressure, possibly below 1 MPa as mentioned above.

3.3. Blanket design

Since a large distance of at least 1 m was obtained between the plasma and the helical coils as mentioned in the previous section, this space was divided into 5 zones as shown in Fig. 5: the plasma scrape-off zone, the protection zone, the T breeding zone, the radiation and thermal shielding zone, and the vacuum and T boundary zone. The parameters in the blanket and shielding specified in this figure are optimized within mixed materials compositions used [3]: the volume ratio of Be in FLiBe with the 20 vol.% structural material, the volume ratio of B_4C in the shielding material, and each thickness of blanket and shield with the sum total at 0.9 m. For this optimization, 1-dimensional neutron transport analyses have been performed for an infinite cylindrical system with the P5-S8 ANISN code by using the FUSION-J3 nuclear data set. Nuclear heating calculations also have been performed by using Kerma factors for neutron and gamma-ray heating.

The 0.28 m thick blanket is basically consisted of 40 vol.% FLiBe as self-cooling tritium breeder, 20 vol.% JLF-1 as structural material and 40 vol.% Be as neutron multiplier. This blanket design is determined to give a tritium breeding ratio TBR larger than 1.1 as shown in Fig.

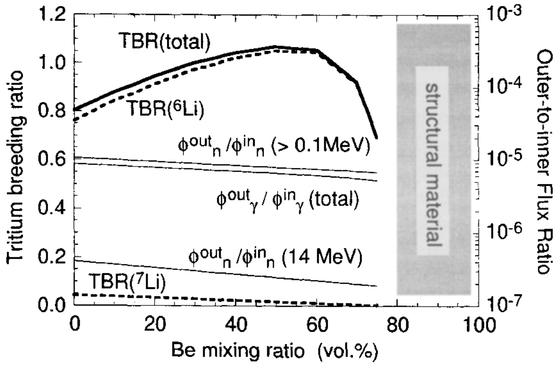

Fig. 6. Be mixing ratio dependence of TBR and radiation shielding efficiency.

6 [3]. The 0.55 m thick radiation shield consists of 70 vol.% JLF-1 and 30 vol.% B_4C. This shield with the 0.05 m thick vacuum vessel of JLF-1 is designed to reduce the fast neutron flux (>0.1 MeV) more than five orders in magnitude at the SC coils as shown in Fig. 6. This reduction means the total neutron fluence in 30 years is lower than $1\times10^{22}/m^2$, or 0.001 dpa, which is required to sufficiently protect SC coils such as Nb_3Sn from deterioration of specific properties including critical current, etc. The boundary position between the blanket and shielding zone is also examined, resulting in a suitable position of about 2.4 m.

The thermal conductivity of FLiBe is not high, about 1 W/mK at 773 K. Therefor the direct deposition of neutron and gamma-ray energy in FLiBe, that is, the volumetric nuclear heating in FLiBe, is strongly desired. Fig. 7 reveals that the nuclear heating in FLiBe is as high as 60% of the fusion output.

The Mo–TiC alloy, which has high resistance against neutron irradiation, is used for the first wall. The W–TiC alloy, which is currently under development by replacing Mo with W in view of the induced radioactivity [6], is the second option. The double walled blanket and transfer tube are covered with He gas to sweep out the permeated T_2 and to monitor drain events as mentioned in the previous section.

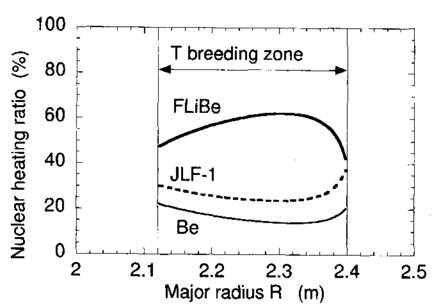

Fig. 7. Nuclear heating ratio in each element of FLiBe, structural material JLF-1 and neutron multiplier Be.

The compatibility between FLiBe and most structural materials has been one of major issues. Since FLiBe itself is very stable (the free energies of formation $\Delta G^{f}_{1000\ K}$ of BeF_2 and LiF are -106.9 and -125.2 kcal/g atom F, respectively), the corrosion is caused by the impurities in FLiBe. In the tritium breeder, there is some possibility that TF can be formed ($\Delta G^{f}_{1000\ K}$ of TF is -66.2 kcal/g atom F), which is very corrosive to most structural materials. The FFHR design has two scenarios to overcome this problem by using Be and MoF_6. The neutron multiplier Be is used as the metal scavenger ($Be + 2TF \rightarrow BeF_2 + T_2$) to reduce the amount of severely corrosive TF molecules. On the other hand, though TF is very corrosive to Fe based and V based alloys, resistance of W and Mo against the TF corrosion is very high. So MoF_6 ($\Delta G^{f}_{1000\ K}$ is -50.2 kcal/g atom F) can be often used to form protection layers of Mo deposited on the surface of the coolant tube with the reaction ($MoF_6 + 3T_2 \rightarrow 6TF + Mo$). Here the produced TF is again reduced by the reaction with Be. Data bases on chemical kinetics in these reactions are strongly desired.

4. Basic design for replacement

Replacement of the blanket might be required at least every 10 years. The basic design for this procedure in FFHR is to use blanket units, which are replaced through maintenance ports by sliding along the continuous helical coils. At this time the total mass of 400 ton of FLiBe is moved to a drain tank. Then, when the blanket consists of the total 180 units for instance, the weight of each blanket unit can be below 5 ton. Radioactive wastes in each replacement are 800 ton of JLF-1, 160 ton of Mo–TiC or 300 ton of W–TiC, which is only 16 m^3 in volume and can be managed, and 350 ton of Be which is the mass of recycling use as well as FLiBe.

5. Conclusion

The high magnetic field force-free helical reactor FFHR is proposed for the reduction of the electromagnetic force by adopting an $l=3$ force-free-like continuous-coil system. In the first stage for concept definition of FFHR, the engineering issues for power-plant reactor are clarified. The molten-salt FLiBe, LiF–BeF_2, is selected in FFHR as the self-cooling tritium breeder from the main reason of safety: low tritium inventory, low reactivity with air and water, low pressure operation, and low MHD resistance which is compatible with the high magnetic field design of FFHR. For the in-vessel structural material the low radioactive ferritic steel JLF-1 is selected as the first candidate. Many possibilities are proposed to overcome FLiBe related issues such as T permeation and TF corrosion of metals.

Further research and development are needed on (1) optimization of the magnetic field configuration to improve the MHD stability of core plasmas, (2) design of divertor pumping systems for He ash, (3) scenario of fabrication of large scale SC helical coils, (4) control of materials corrosion in the FLiBe system, and (5) maintenance and repair techniques.

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