Conceptual Design of Magnets with CIC Conductors for LHD-type Reactors FFHR2m.

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LHD-type reactors have attractive features for fusion power plants, such as no need for current drive, a wide space between the helical coils for the maintenance of in-vessel components. One disadvantage was considered a necessarily large major radius to attain the self-ignition condition or to secure a sufficient space for blankets. According to recent reactor studies based on the experimental results in LHD, the major radius of plasma is set to 14 to 17 m with the central toroidal field of 6 to 4 T. The magnetic stored energy is estimated at 120 to 130 GJ. Both the major radius and the magnetic energy are about three times as large as ITER. We intend to summarize requirements for superconducting magnets of the LHD-type reactors and to propose a conceptual design of the magnets with cable-in-conduit (CIC) conductors based on the technology for ITER.

Keywords: cable-in-conduit conductor, fusion reactor, helical coil, LHD, superconducting magnet.

1. Introduction

Superconducting magnets for fusion reactors need high mechanical strength, high reliability, and low costs as well as sufficient current densities in the high field. Cable-in-conduit (CIC) conductors have been developed for large pulse coils, and they are adopted for all magnets of ITER [1-3]. Major features of the CIC conductors are a large current up to 100 kA, high strength with thick conduits, small AC losses, and high cryogenic stability. One disadvantage is that they need circulation pumps for forced-flow cooling. 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 heating The CIC conductor will not be the best for magnets of a helical reactor that is operated with a constant current. However, technology related to CIC conductors will be strongly improved through the construction of ITER, especially in a cost and in the winding technique. Then, we study the helical winding with CIC conductors on the engineering base of ITER as a conventional option.

2. Magnet Systems of LHD-type Reactor

A magnet system of an LHD-type reactor consists of a pair of continuous helical coils and more than one pair of poloidal coils [4]. The position of the poloidal coils is determined by the dipole magnetic field, the quadruple magnetic field, the stray field, the position of ports, and so on. At least one set of poloidal coils is necessary to adjust the major radius of the plasma, the quadruple field, and the stray field. In this case, the magnetic field around the machine center is fairly high, and the magnetic stored energy is large. Two pairs of poloidal coils are appropriate, because they can reduce the total weight of supporting structures with reduction of the magnetic stored energy. Also, the position of plasma axis can be controlled without increase of the stray field. The position of the coils is not determined uniquely with the restrictions as above because of the rest of the degrees of freedom. The coil position can be adjusted to attain the space for the blanket, the mechanical support, and ports. In this study, we adopt two additional restrictions that $a_{IV}=a_{OV}$ and $Z_{IV}=Z_{OV}$, where a is the distance from the major radius circle of the helical coil, and Z is the height, as shown in Fig. 1.

The lead angle of the helical coil is defined as the pitch parameter $\gamma = (ma_c)/(lR_c)$, where l, m, R_c, and a_c are the pole number, the pitch number, the coil major radius, and the coil minor radius, respectively. Figure 2 shows the normalized stored energy of the LHD-type reactors with two sets of poloidal coils. Since the poloidal coils cancel the vertical field by the helical coils, the stored energy is the larger with the longer distance of the poloidal coils. Furthermore, it is the larger with the higher γ because the toroidal field area increases.

The coil current density, j, is very important for the design of superconducting magnets. Although the high density is useful to enlarge the space for blankets and for the maintenance, it is restricted by cryogenic stability, mechanical strength, and the maximum field. Considering the space for the structural materials inside the winding, j

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is set approximately 25 A/mm^2 in this study. The highest magnetic field is also important for the superconducting magnets. It depends mainly on the ratio of height of the helical coil to the minor-radius [5]. The highest magnetic field is set less than 13.5 T that is the same as the highest field of the ITER-CS coils.

The necessary magnetic field and size can be determined by the scaling low for the plasma confinement and the necessary space for blankets. The scaling law of ISS04 [6] for the energy confinement time is adopted in this study. At first, we have studied the necessary size under the condition of the enhancement factor of energy confinement of 1.12 to the ISS04 scaling. Since LHD has attained the factor of 0.93, the required further improvement is 1.2 that will be achieved in the near future. Table 1 shows examples of the design parameters of LHD-type reactors, FFHR2m2, under the conditions: the parabolic distribution of both the plasma density and temperature, the minimum space for blankets of 1.1 m, the helical coil current density of 27.5 A/mm^2, helium ash ratio of 3%, oxygen impurities ratio of 0.5%, and alpha particle heating ratio of 90%. The plasma density was set to just the density limit of the Sudo scaling [7]. The central temperature was adjusted to make the average β, the ratio of plasma pressure to the central magnetic pressure, at 5%. The plasma shape is assumed to be similar as that of LHD at the inward shift mode, in which the best plasma confinement has been achieved.

In the conditions as above, the smallest major radius is determined mainly by the space for blankets than the highest magnetic field. The plasma major radius around 15 m is necessary for a reactor similar to LHD. Since the high-density operations elongate the energy confinement time of the ISS04 scaling, the enhancement factor is less with the higher β operation in which the larger fusion power is produced.

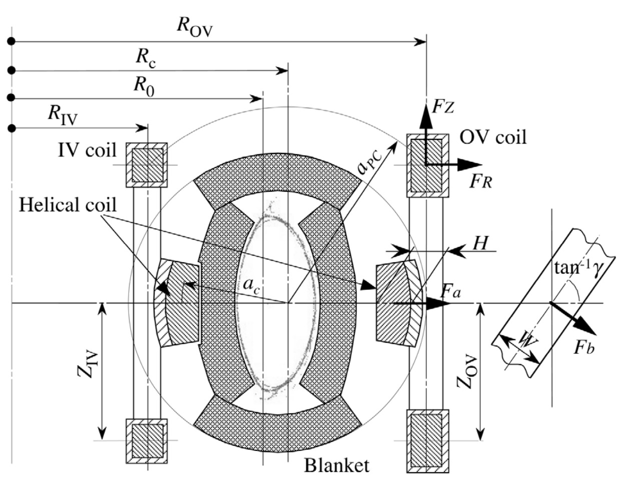

Fig. 1. Magnets and supporting structures of LHD-type fusion reactor.

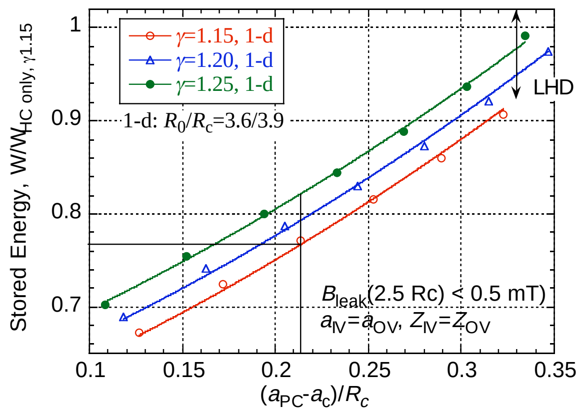

Fig. 2. Stored energy of magnets of LHD-type reactors with two pairs of poloidal coils under the condition of the same height and the same minimum distance from the helical coil in the poloidal cross-section.

Table 1. Case study of LHD-type reactors FFHR2m2-*j*27.5.

	γ1.15	γ1.2	γ1.25
Polarity / Field periods, l/m	2/10	2/10	2/10
Coil pitch parameter, γ	1.15	1.20	1.25
Coil major radius, R_c (m)	15.68	16.52	17.53
Coil minor radius, a_c (m)	3.61	3.97	4.38
Coil center line length (m)	150	162	176
Plasma major radius, R_0 (m)	14.47	15.25	16.18
Plasma minor radius, a_p (m)	1.75	2.24	2.83
Plasma volume, V_p (m^3)	877	1516	2559
Central magnetic field, B_0 (T)	5.53	4.95	4.49
Max. field on coils, B_{max} (T)	12.1	12.1	12.2
Coil current, I (MA)	40.0	37.7	36.3
Coil current density, j (A/mm^2)	27.5	27.5	27.5
Blanket space, Δ_d (m)	1.1	1.1	1.1
Magnetic energy, W (GJ)	128	124	125
Density, $n_e(0)$ (10^{19} m^{-3})	34.9	26.3	20.5
Ion temperature, $T_i(0)$ (keV)	16.9	17.9	18.9
Average beta, $<\beta>$	5	5	5
Fusion power, P_F (GW)	3.9	4.2	4.8
Energy confinement time, τ_E (s)	1.33	1.65	2.01
Enhancement factor to ISS04	1.12	1.12	1.12

3. Helical Coil with CIC Conductors

Main specifications of helical coils for an LHD-type reactor, FFHR2m are as follows: the magnet-motive force of about 40 MA, the magnetic energy of 120 to 130 GJ, a coil center line of 150 to 175 m, and average coil current density of 25 to 30 A/mm^2. Design criteria for CIC conductors based on the ITER magnets are summarized in Table 2. Since the length of the coil center line of the helical coil is five times as long as the TF coil, some ideas are necessary in addition to adopting a large current of almost 100 kA. Parallel winding is a practical solution to shorten the cooling length within about 500 m.

Two types of mechanical structure are known for CIC conductors. One is a thick conduit type, in which rectangular conductors are simply wound with being wrapped by insulating tapes. Fairly high stress is induced in the insulators by summed up forces. The other is an internal plate type, in which the conductor is wound in the grooves of the internal plate. The stress in the insulation is reduced. Besides, the force for winding is relatively small because of thin conduits. Its disadvantage is complicated manufacturing process of the internal plates. However, its technology will be improved through the construction of ITER-TF coils. Internal plates with grooves are suite for parallel winding, because CIC conductors are just put in the grooves as shown in Fig.3. In this concept, react and wind method is preferred to use conventional insulator and to prevent huge thermal stress. Nb_3Al is a candidate for the superconducting strands of the conductor because of its good tolerance against mechanical strain. A method of "react and wind" can be adopted by managing strain during winding within about 0.5%.

The magnetic field in the helical coil is the highest in the first layer, and it is lower in the higher layers. Therefore, the average current density of the superconducting strands can be increased by grading the conductors in the case of layer winding. Non-copper current density of Nb_3Sn is given as [8],

$$j_c = 1/\left(1/j_{c1} + 1/j_{c0}\right) \tag{1}$$

$$j_{c0} = j_0\left(1-(T/T_{c0})^2\right) \tag{2}$$

$$j_{c1} = C_0\left(1-(T/T_{c0})^2\right)^2 B^{-0.5}\left(1-B/B_{c2}\right)^2 \tag{3}$$

$$B_{c2} = B_{c20}\left(1-(T/T_{c0})^2\right)\left(1-T/3T_{c0}\right) \tag{4}$$

$$B_{c20} = B_{c20m}\left(1-a\varepsilon^{1.7}\right) \tag{5}$$

$$T_{c0} = T_{c0m}\left(1-a\varepsilon^{1.7}\right)^{0.333} \tag{6}$$

where j, T, B, and ε are the current density, the temperature, the magnetic field, and the strain, respectively. For ITER conductor j_0=33.51 kA/mm^2, T_{c0m}=18 K, B_{c20m}=28 T, a=1250 for tensile, 900 for compressive, and C_0=1150 [3]. Table 3 shows the non-copper current density of the strands for the various magnetic field at 7 K with the strain of -0.5%. In the case of B_{max} of 12 T, the average current density can be increased by 60% by adopting four grades of conductors.

The typical design parameters of the helical coil are listed in Table 4, compared with the ITER-TF coils. By adopting large conductors of about 90 kA and the parallel winding of five-in-hand, the length of a cooling path is within 530 m including the case of γ=1.25 in Table 1. By increasing the number of quench protection circuits, the maximum discharge voltage can be managed less than 10 kV in spite of the larger inductance and the shorter discharge time constant than ITER-TF coils. Consequently, the helical winding is expected to be realized with small extension of the technology for ITER.

Table 2. Design criteria for CIC conductors based on ITER.

Items	Design criteria	ITER-TF
Max. cooling length (m)	< 550	390
Current (kA)	< 100	68
Maximum field (T)	< 13	11.8
Non-Cu current density (A/mm^2)	< 300	273
Coil current density (A/mm^2)	< 30	20.3
SC material for HC	Nb_3Al (*1)	Nb_3Sn

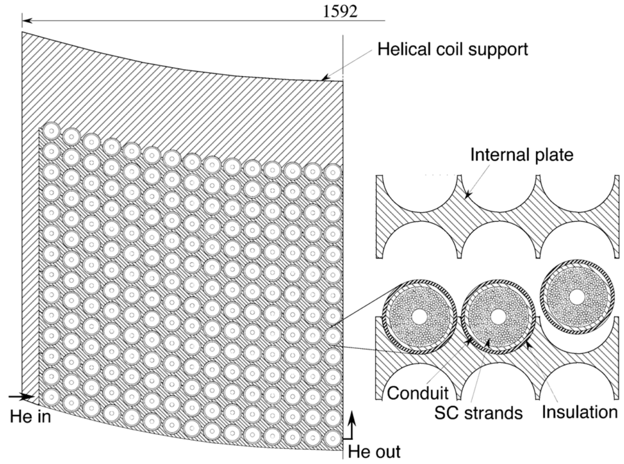

Fig. 3. Concept of helical winding with CIC conductors.

Table 3. Increase of non-copper current densities of SC strands of helical coil conductors by grading.

B_{max} (T)	J_c at B_{max} (A/mm^2)	Av. J_c by grading (A/mm^2) 3 grades	4 grades	5 grades
11.0	361	514	534	543
11.5	304	449	468	477
12.0	254	391	409	417
12.5	209	338	355	363
13.0	170	290	306	314

Table 4. Specification of helical coil with CIC conductors.

	HC_γ1.15	HC_γ1.20	ITER-TF
Maximum field (T)	12.1	12.1	11.8
Magnetic energy (GJ)	128	124	41
Number of coils	2	2	18
Turn number per coil	30*14	30*14	134
Conductor current (kA)	95.2	89.8	68.0
Length of a cooling path (m)	450	486	390
Number of parallel winding	5	5	1
Current density (A/mm^2)	27.5	27.5	20.3
Cu ratio of strand (–)	1	1	1
Non-Cu current density (A/mm^2)	400	400	273.4
Ratio of Cu strands in area (–)	0.452	0.452	0.360
Central tube diameter (mm)	12.0	12.0	8.0
Void fraction (–)	0.34	0.34	0.34
Cable outer diameter (mm)	43.5	42.3	40.2
Conduit outer diameter (mm)	46.7	45.5	43.4
Total length of conductor (km)	126	136	82.2
Total weight of SC strands (ton)	481	490	351
Total weight of Cu strands (ton)	442	450	206
Cu current density (A/mm^2)	151	151	128.7
Discharge time constant (s)	12	12	15
Inductance (H)	28.2	30.6	17.7
Number of coil blocks	35	35	9
Max. voltage (kV)	6.4	6.6	8.9

A winding method is a critical issue for the helical coil. In the case of LHD, a special winding machine was developed. The conductors from a rotating bobbin were plastically formed into the helical shape by the shaping head near the winding guide. This method will not be compatible with the "react and wind" method, because the allowable strain is in the range of 0.5% for Nb_3Al strands. A candidate of winding method is as follows:

(1) Conductors are heated for reaction of Nb_3Al on a bobbin the circumference of which is same as the length of one pitch of the helical coil.

(2) The conductors are transferred to a reel of a winding machine. The reel revolves through the helical coil as shown in Fig. 4.

(3) The conductors are pulled aside by a set of winding guides and wound in grooves of the inner plate with being wrapped with glass tapes.

(4) After winding the whole turns in a layer, the next inner plate are assembled.

The torsion strain $r\theta$ in winding is given as

$$r\theta = \frac{r \cdot tan^{-1}\eta}{2\pi a_c/4} \tag{7}$$

where r and η are the radius of the conductor and the lead angle of the helical coil. In the case of FFHR2m2 in Table 1, the strain is estimated at about 0.3%. Since the effect of the torsion strain on the properties of superconducting strands is not known, the feasibility study is needed.

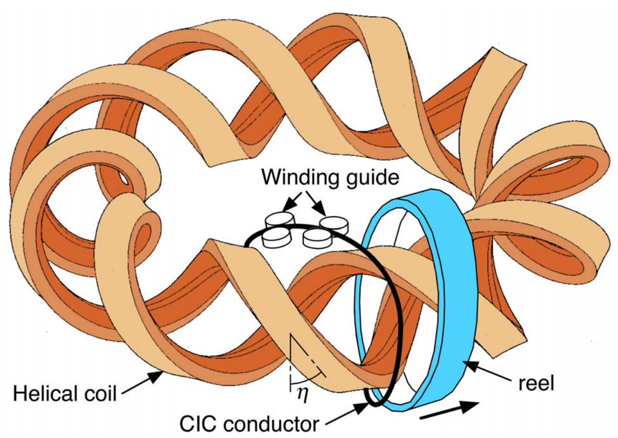

Fig. 4. Concept to wind a helical coil with CIC conductors.

4. Poloidal Coil with CIC Conductors

The poloidal coils of LHD-type reactors are circular, same as the poloidal field coils of ITER. The typical design parameters of the poloidal coil of the FFHR2m2 are listed in Table 5, compared with the largest poloidal field coil of ITER, PF3 coil. Since the radius of the larger coil, OV coil, is almost twice as large as the ITER-PF3 coil, parallel winding is also necessary. Although the coil current is huge, the highest magnetic field can be lowered than 7 T by decreasing the coil current density. Therefore, NbTi strands can be adopted, and these coils are expected to be realized with the same technology for ITER.

Table 5. Specification of poloidal coil with CIC conductors for FFHR2m2_γ1.20_j27.5.

	OV coil	IV coil	ITER-PF3
Radius of coils (m)	21.5	11.5	11.97
Number of coils	2	2	1
Coil current per coil, I (MA)	19.6	12.1	8.46
Current density (A/mm^2)	15.3	16.8	15.26
Maximum field (T)	6.6	6.3	4
Turn number per coil	14*28	10*20	11.75*16
Conductor current (kA)	50.1	60.4	45
Length of a cooling path (m)	473	361	441.9
Number of parallel winding	4	2	2
Cu ratio of strand (–)	2	2	1
Non-Cu current density (A/mm^2)	200	200	230
Ratio of Cu strands in area (–)	0	0	0.24
Central tube diameter (mm)	12.0	12.0	12.0
Void fraction (–)	0.34	0.34	0.34
Cable outer diameter (mm)	40.1	43.7	34.5
Conduit height*width (mm)	55.1*55.1	58*58	52.3*52.3
Total length of conductor (km)	106	28.9	14.1
Total weight of SC strands (ton)	653	215	41.4
Total weight of conductor (ton)	2140	641	258

5. Summary

CIC conductors can be adopted for large helical windings by adopting layer winding and parallel winding method. Since "react & wind" method is preferred for large magnets, Nb_3Al is a candidate for the helical coil conductor because of its good tolerance against mechanical strain. It is necessary to demonstrate the feasibility of its winding method. In addition, structural analyses of the winding area are needed in order to confirm the mechanical feasibility of the helical coil with the high current density of 25 to 30 A/mm^2. This conceptual design is expected to be a conventional option that can be realized by small extension from the ITER technology.

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