Concept of Magnet Systems for LHD-type Reactor.

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Abstract. Heliotron reactors have attractive features for fusion power plants, such as no need for current drive and a wide space between the helical coils for the maintenance of in-vessel components. Their main disadvantage was considered the necessarily large size of their magnet systems. According to the recent reactor studies based on the experimental results in the Large Helical Device, the major radius of plasma of 14 to 17 m with a central toroidal field of 6 to 4 T is needed to attain the self-ignition condition with a blanket space thicker than 1.1 m. The stored magnetic energy is estimated at 120 to 140 GJ. Although both the major radius and the magnetic energy are about three times as large as ITER, the maximum magnetic field and mechanical stress can be comparable. In the preliminary structural analysis, the maximum stress intensity including the peak stress is less than 1,000 MPa that is allowed for strengthened stainless steel. Although the length of the helical coil is longer than 150 m that is about five times as long as the ITER TF coil, cable-in-conduit conductors can be adopted with a parallel winding method of five-in-hand. The concept of the parallel winding is proposed. Consequently, the magnet systems for helical reactors can be realized with small extension of the ITER technology.

1. Introduction

Helical fusion reactors have attractive features for power plants, such as being free from current-disruptions, no need for current-drive, and a steady operation with constant coil-currents, owing to inherently net-current less plasma. Their main disadvantage was considered the necessarily large size of their magnet systems. According to recent reactor studies based on the experimental results with the Large Helical Device (LHD)[1], the major radius of plasma is set at 14 to 17 m in order to install breeding and shielding blankets with total thickness more than 1.1 m [2]. The central toroidal field of 6 to 4 T is needed to attain the self-ignition condition. Its major radius is about three times as large as ITER [3]. Also, it is two to three times as large as a recent design of tokamak reactors. From the viewpoint of superconducting magnet technology, the stored magnetic energy and the maximum field are the most important. They almost determine the amount of superconducting wires and necessary supporting structures for the magnets. In the design of the supporting structures, accessibility to in-vessel components is crucial for power plants. An LHD-type reactor has an advantage to secure large ports for maintenance and replacement of blankets, because a space between helical coils is wide. The optimization of the layout and shape of the supporting structures is important to reduce their amount and to attain good maintainability of the blankets. This paper intends to clarify the requirements and design scaling equations for magnets of the LHD-type reactors. In addition, an improved design of their supporting structures and a concept of helical winding are proposed.

2. Requirements for magnet systems

2.1. Main parameters of helical coils

The coordinate sketch of the magnets is shown in Fig. 1. Although a high ratio of width W to height H of the helical coil is useful to enlarge the blanket space, it reduces the area of maintenance ports. The ratio of 2.0 was selected in this study as a moderate value. The trajectory of the current center of the helical coil is defined as

$$\theta = \frac{m}{l}\phi + \alpha^{*} sin\left(\frac{m}{l}\phi\right), \tag{1}$$

where θ, ϕ, l, m, and α^* are the poloidal angle, toroidal angle, pole number, pitch number, and pitch modulation, respectively [4]. α* is set 0.1 same as LHD in this study. The current I and the pitch parameter γ of the helical coil are given by $(2\pi R_0 B_0)/(\mu_0 m)$ and $(ma_c)/(lR_c)$, respectively, where R_0, R_c, a_c, B_0, and μ_0 are the plasma major radius, coil major radius, coil minor radius, central toroidal field, and permeability in vacuum, respectively. Since the current density j of the helical coil is given by $I/(W \cdot H)$, it is in inverse proportion to the major radius in the case of the constant toroidal field and similar geometrical parameters. Consequently, the relative size of the coil cross-section to the plasma becomes smaller in the larger reactor with the constant current density. The coil current density 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 area such as cable-in-conduit (CIC) conductors, j is set approximately 25 MA/m^2 in this study. The highest magnetic field B_{max} is also important for superconducting conductors. Its ratio to B_0 depends mainly on the ratio of height of the helical coil to the minor-radius, as shown in Fig. 2. In the case of W/H=2, its fitting curve is expressed as

$$\frac{B_{max}}{B_0} = 0.4819 + 0.4185\frac{a_c}{H} + 0.006685\left(\frac{a_c}{H}\right)^2 \tag{2}$$

The design criteria for highest field is set less than 13.5 T the same as the ITER-CS coils [3].

The electromagnetic force on the helical coil is divided into minor-radius and overturning components. The former is dominant in a standard operation. Using an equivalent bending radius a_{eq}, the minor-radius hoop force F is given by $F = \langle f_a \rangle \cdot a_{eq} = \langle B_T \rangle \cdot I \cdot a_{eq}$, where f_a and B_T are the minor-radius hoop force per unit length and the transverse field averaged in the cross-section of the coil. As shown in Fig. 3, $\langle B_T \rangle$ normalized with B_0 becomes smaller at smaller γ [5]. Since the hoop force is supported by the tension of the structure, the necessary cross-sectional area of the coil support A_{SS}, can be estimated as F/S_m, where S_m is allowable stress.

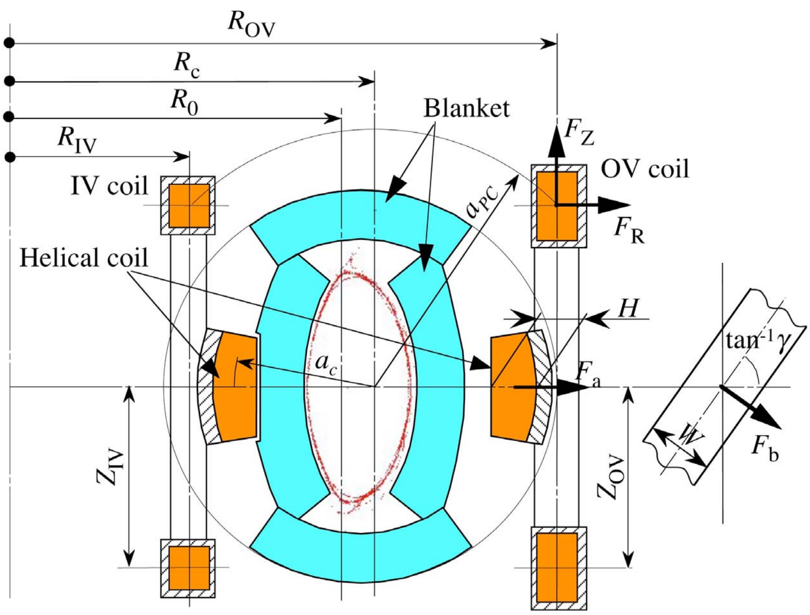

FIG. 1. Magnets and supporting structures of LHD-type fusion reactor.

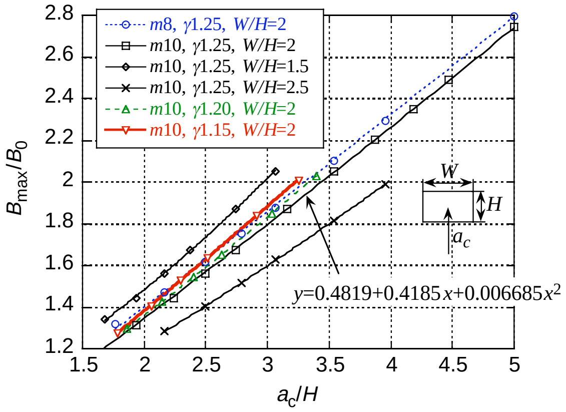

FIG. 2. The highest transverse field in helical coils versus the ratio of coil minor radius to coil height.

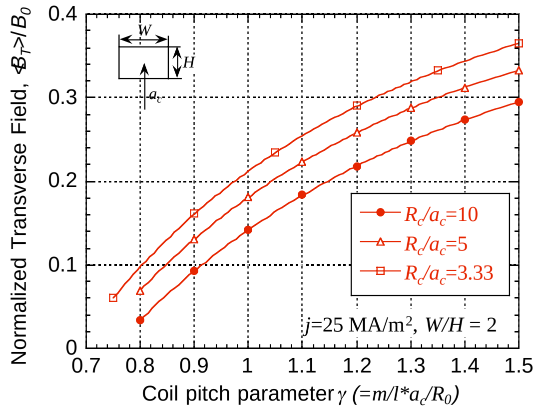

FIG. 3. The averaged transverse field in the helical coil for l=2, j=25 MA/m2, and W/H=2.

2.2. Size and magnetic field

The necessary magnetic field and size can be determined by the scaling law 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. It is expressed as

$$\tau_E^{ISS04} = 0.134 a_p^{2.28} R_0^{0.64} P^{-0.61} \bar{n}_e^{0.54} B_0^{0.84} \iota_{2/3}^{0.41}, \tag{3}$$

where a_p, P, n_e, and $\iota_{2/3}$ are the average plasma minor radius, the plasma heating power, the average electron density, and rotational transform. In the case that n_e, ι, and the heating power per unit volume are constant,

$$\tau_E^{ISS04} \propto a_p^{2.28} R_0^{0.64} \left(a_p^2 R_0\right)^{-0.61} B_0^{0.84} \propto a_p^{1.06} R_0^{0.03} B_0^{0.84}. \tag{4}$$

Under the conditions of the constant τ_E and similar geometrical parameters, the necessary toroidal field is in inverse proportion to the 1.30(=(1.06+0.03)/0.84) power of the major radius. According to the Virial theory, the weight of supporting structures is proportional to the magnetic energy. Considering that the magnetic energy is proportional to B^2a^2R, the weight is proportional to only the 0.40 (=-1.30×2+2+1) power of the major radius under the conditions of the constant energy confinement time and similar geometrical parameters. The influence of the radius on the construction cost of the magnet system will not be strong.

In the above conditions, the τ_E is almost proportional to a_p that varies with a_c, γ, R_0/R_c, and the cross-sectional shape of plasma. In this study, we use the same ratio of a_p/a_c versus γ as LHD, as shown in Fig. 4. It was obtained by numerical calculation of field lines in the vacuum condition. Since the plasma radius is almost independent of the current density of the helical coil, the minimum gap for blanket, Δ_d, is derived as,

$$\Delta_d = a_c - a_{p\text{-in}} - H/2 - 0.1, \tag{5}$$

where $a_{p\text{-in}}$ is the inward radius of the last closed surface at the position where the plasma is vertically elongated and the blanket space is the most narrow. The last term of 0.1 m is a space for thermal shields. The thickness of ergodic layer is included in Δ_d.

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 [6], the required further improvement is 1.2 that will be achieved. Its representative operation conditions are the central electron density of 26.6×10^{19} m^{-3} with parabolic distribution, central temperature of 15.9 keV with parabolic distribution, helium ash ratio of 3%, oxygen impurities ratio of 0.5%, alpha particle heating ratio of 90%, and its heating power of 0.35 MW/m^3. The estimated results are shown in Fig. 5 for three pitch parameters of 1.15, 1.2, and 1.25. The current density j of the helical coil is set 25 MA/m^2. In order to attain the gap for the blanket of 1.1 m, the major radius around 16 m is necessary for a reactor similar to LHD.

According to the ISS04 scaling, the high density operation elongates the energy confinement time. Examples of the design parameters of LHD-type reactors are shown in Table 1, where the density was set to just the density limit of the Sudo scaling, and the temperature was adjusted to make the average plasma pressure at 5% of the magnetic pressure by B_0. FFHR2m1 is the previous design that adopted the concept of outward shift of plasma to enlarge the blanket space at inside. FFHR2m2 is a new design with the concept of inward shift of plasma, in which the best plasma confinement was achieved in the LHD experiments. From the viewpoint of conductor design, FFHR2m2b is more difficult than FFHR2m2a because of the higher current density at the higher field. It, however, has an advantage to attain wider ports for maintenance with the lower magnet costs because of the less magnetic energy.

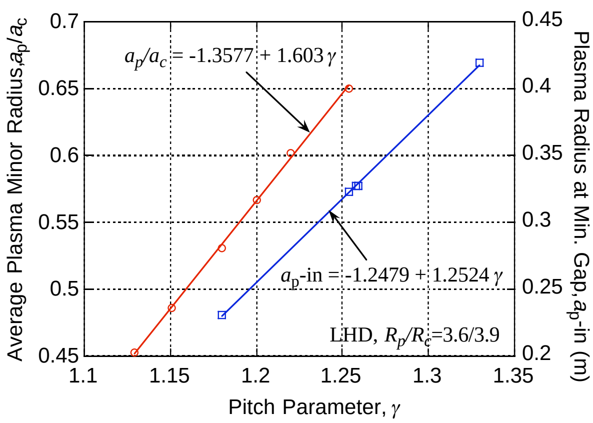

FIG. 4. Plasma average radius of LHD versus the pitch parameter and the plasma radius at the position of the minimum gap for blanket.

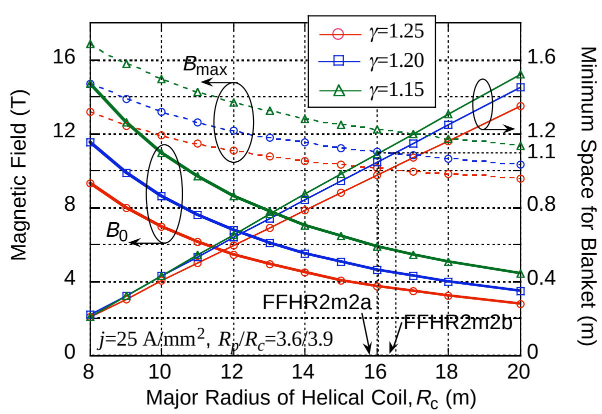

FIG. 5. Scale dependence of the central magnetic field and space for blankets under the conditions of τ_E= const., γ= 1.25, j=25 MA/m^2, and W/H = 2.

2.3. Layout of magnets and magnetic energy

A magnet system of an LHD-type reactor consists of a pair of continuous helical coils and more than one pair of poloidal coils. 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 center cannot be lowered, and the stored magnetic energy is larger. In the case of two sets of poloidal coils, the number of degrees of freedom is six, which makes it possible to reduce the field near the center of the torus. Two pairs of poloidal coils, OV and IV coils, are appropriate, because they can reduce the total weight of supporting structures with reduction of the stored magnetic energy. In addition, independent power sources of the two pairs of coils can control the position of plasma axis without increase of the stray field.

The position of the poloidal coils is not determined uniquely with the above restrictions because of the rest of the degrees of freedom. Additional restrictions must be selected to attain the space for the divertor region, the mechanical support, and ports. Considering the importance of the blanket space, we adopt the two additional restrictions that a_{IV}=a_{OV} and

$Z_{IV}=Z_{OV}$, where a is the distance from the torus center of the helical coil, and Z is the height, as shown in Fig. 1. Design of the divertor will determine a. Figure 6 shows the dependence of the stored magnetic energy to a. Since the poloidal coils cancel the vertical field by the helical coils, the magnetic energy is the larger with the longer distance of the poloidal coils. Furthermore, it is the larger with the higher γ because of the increase of the volume where the toroidal field is induced. The magnetic energy W of an LHD-type reactor can be fitted as

$$\frac{W(I,R_c)}{W_{HConly}(I,R_c)}=0.428+1.6\cdot\left(\frac{a_{PC}-a_c}{R_c}+0.34\right)^{2.5}+0.5(\gamma-1.15)\quad \text{for}\ \frac{R_P}{R_C}=\frac{3.6}{3.9}. \tag{5}$$

When $(a_{PC}-a_c)/R_c = 0.2$,

$$W[\text{GJ}]=163.3\cdot\left(I[\text{MA}]/42\right)^2\left(R_c[\text{m}]/14\right)\left(0.77+0.5(\gamma-1.15)\right). \tag{6}$$

In the case of similar shape of LHD with the enhancement factor of energy confinement of 1.12 to the ISS04 scaling, the necessary central toroidal field are estimated at 4.4 to 5.6 T at a_p/R_0=3.0 m/16.9 m to 1.7 m/14.3 m, and the magnetic energy was estimated at 120 to 140 GJ, as shown in Fig. 7 and Table 1. Higher current densities in the helical coil are preferred to reduce the major radius and the magnetic energy. The higher limit of the current density should be determined by the mechanical stress rather than by the maximum field.

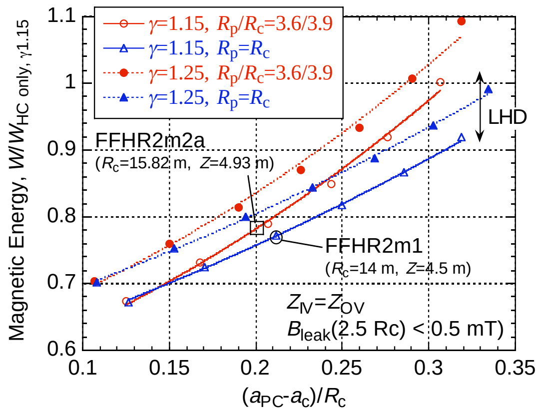

FIG. 6.Magnetic energy of magnets of LHD-type reactors with two pairs of poloidal coils under the condition of the same height.

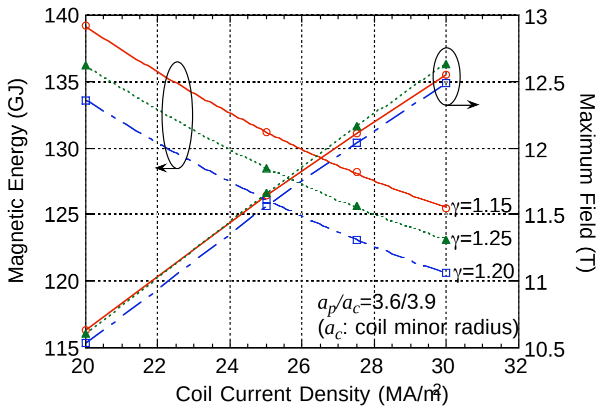

FIG. 7. Magnetic energy and the maximum field of a magnet system of LHD-type reactors.

TABLE I. CASE STUDY OF HELIOTRON-TYPE REACTORS.

			FFHR2m2a	FFHR2m2b	FFHR2m1	LHD
Polarity / Field periods	l/m		2/10	2/10	2/10	2/10
Coil pitch parameter			1.15	1.20	1.15	1.25
Coil major / minor radius	R_c/a_c	m	15.82/3.64	16.36/3.93	14/3.22	3.9/0.975
Plasma major / minor radius	R_0/a_p	m	14.6/1.77	15.1/2.22	14/1.73	3.75/0.61
Plasma volume	V_p	m^3	901	1472	831	27.5
Central magnetic field	B_0	T	5.50	4.96	6.18	3
Max. field on coils		T	11.8	12.4	13.2	7.9
Coil current	I	MA	40.18	37.42	43.26	5.85
Coil current density	j	MA/m^2	25.8	29.5	26.6	40
Min. blanket space		m	1.1	1.1	1.2	0.118
Magnetic energy	W	GJ	130	120	120	0.9
Plasma density	$n_e(0)$	$10^{19}\ m^{-3}$	34.5	26.4	26.6	
Ion temperature	$T_i(0)$	keV	16.9	17.9	15.9	
Average beta	$<\beta>$		5	5	2.88	
Fusion power	P_F	GW	3.9	4.14	1.9	
Energy confinement time	t_E	s	1.34	1.64	1.86	
Enhancement factor to ISS04			1.12	1.12	1.12	

3. Structural analysis

Preliminary structural design for FFHR2m1, the magnetic energy of which is 120 GJ, has been carried out. Since sum of electromagnetic force on all coils is balanced, all coils were supported by each other, as shown in Fig. 8. In considering the maintenance of blanket, large apertures are prepared at top, bottom and outer region. The supports for the hoop force are contained inside each coil. The upper and lower poloidal coils are linked by thick shell structures. The helical coils are fixed to the shell structures at only the horizontal position. Since electromagnetic force on the helical coils is reduced because of their inclined shape, the helical coils can withstand their electromagnetic forces in the minor radius direction. The electromagnetic forces on the coils are shown in Fig. 9. A calculated stress by a FE model is shown in Fig. 10. Used material properties are listed in Table 2. Considering the internal support in the winding region, its average rigidity was set 150 GJ. In the analysis, the highest stress was induced in the outer shell structure near the cross point of the helical coil and the OV coil. The highest stress will be reduced by shifting the OV coils outwards. The maximum stress intensity including peak stress is almost 1,000 MPa that is induced at the corner of the outer support. Except for there, the stress intensity is less than 700 MPa. These values will be allowable for strengthened stainless steel. Total weight of the superconducting magnets and the supporting structure is about 15,500 tons.

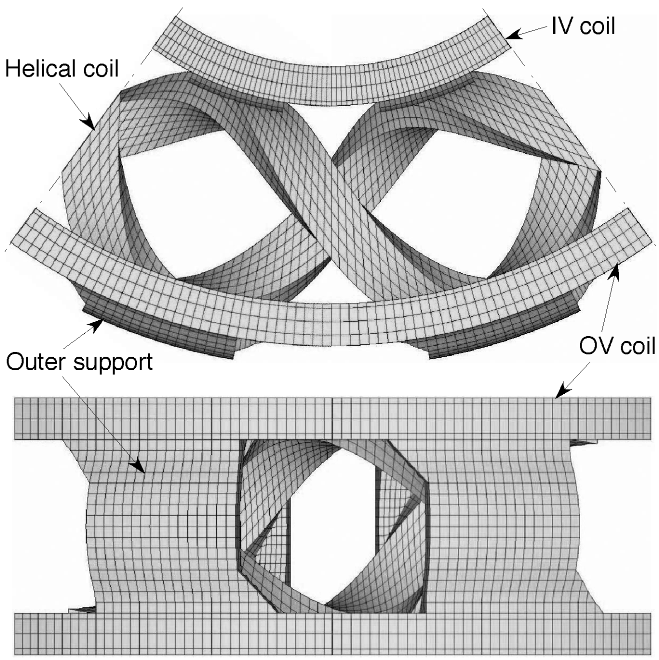

FIG. 8. Concept of supporting structures for magnets of FFHR2m1, where R_0 = 14.0 m, B_0=6.18 T, and γ = 1.15, R_{IV}=9.5m, R_{OV}=17.28 m, Z_{IV}=Z_{OV}=3.6m.

TABLE II. PROPERTIES OF MATERIAL OF THE EF MODEL.

Part	Material	Poisson ratio	Elastic modulus (GPa)
Helical coil winding area	Cable + Stainless steel	0.3	150
Helical coil support	Stainless steel	0.3	200
Poloidal coil winding	Cable + Stainless steel	0.3	150
Poloidal coil case	Stainless steel	0.3	200
Inner support	Stainless steel	0.3	200
Outer support	Stainless steel	0.3	200

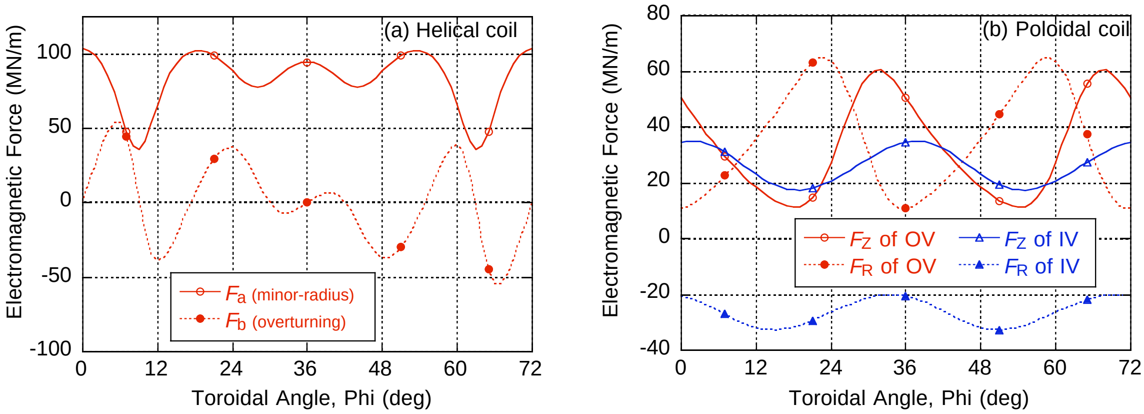

FIG.9. Electromagnetic forces on a helical coil and poloidal coils of FFHR2m1.

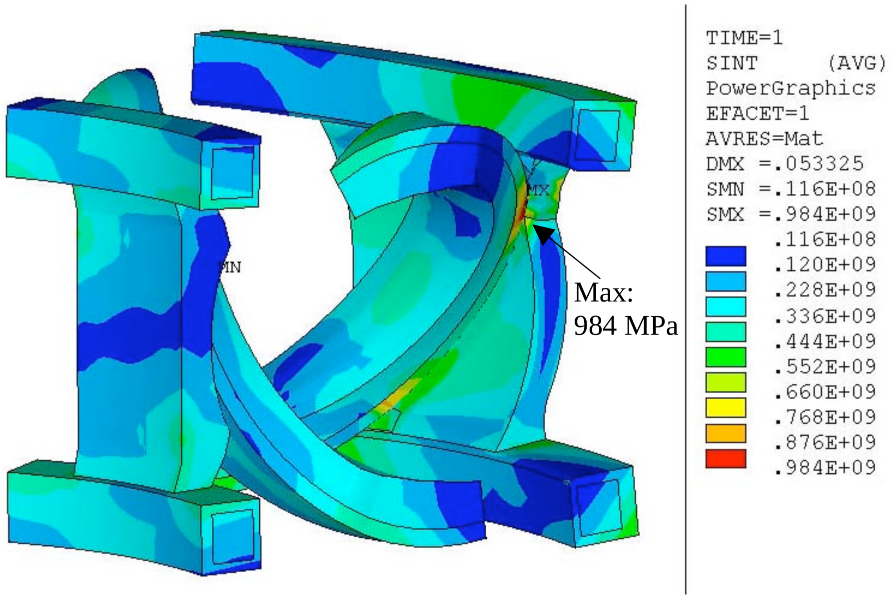

FIG. 10. Finite element model of FFHR and distribution of stress intensity.

4. Winding Technology

It is crucial to establish the winding concept for such a huge helical coil. Cable-in-conduit (CIC) conductors have been developed for large pulse coils, and they are adopted for all magnets of ITER [3]. A CIC conductor will not be the best for the helical coil 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. Further, their feature of high strength and large currents are suitable for large magnets. Therefore, we have studied a helical winding with CIC conductors using ITER technology [7]. The length of the helical coil is longer than 150 m that is about five times as long as the ITER TF coil. By adopting large conductors of class 100 kA and the parallel winding of five-in-hand, the length of a cooling path can be shortened within about 500 m. Then, we can adopt cable-in-conduit (CIC) conductors within reasonable pressure drop of cryogen. Internal plates are useful to increase the mechanical strength and to reduce the stress of the insulator around the conductors. In addition, the grooves of the plates are useful to realize the parallel winding, because the CIC conductors are just put in the grooves, as shown in Fig.10. Nb_3Al is the first candidate of the superconductor because of its good tolerance against mechanical strain.

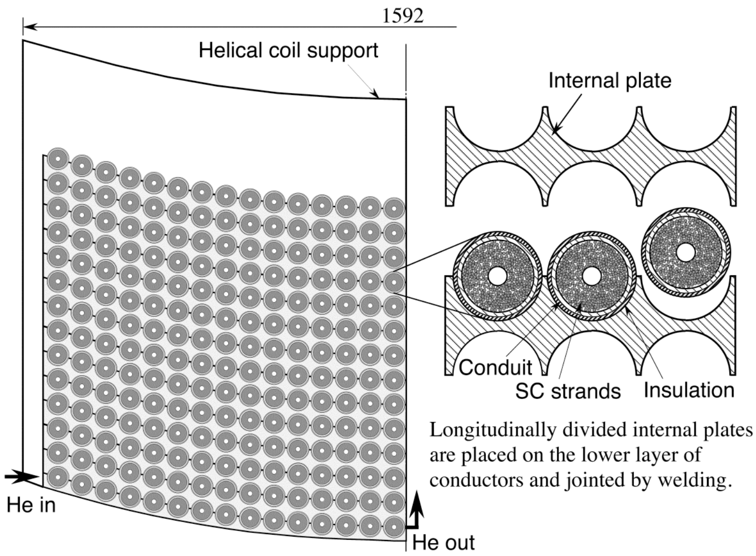

FIG. 11. Cross-sectional drawing of helical winding with CIC conductors.

5. Summary

The necessary size of LHD-type reactors has been investigated under the assumption that the plasma shape is similar to LHD. In order to attain the blankets space more than 1.1 m with the helical coil current density of about 25 MA/m^2 and the confinement enhancement factor of 1.12 to ISS04, the necessary coil major radii were estimated at 15.8 m and 16.4 m for the coil pitch parameter γ of 1.15 and 1.20, respectively. As the results of studies on the position of the poloidal coils, the number of pairs of poloidal coils was set to 2 to reduce the magnetic energy. In this case, the stored magnetic energy was estimated at 120 to 140 GJ for the LHD-type reactors. Next, an improved design of the supporting structures of the magnets is proposed to attain good accessibility to the blankets. The maximum stress intensity of the supporting structures can be suppressed within 1,000 MPa that is the allowable stress of SUS316LN. In addition, the concept of helical winding with CIC conductors is proposed. The magnet systems for helical reactors can be realized with small extension of the ITER technology.

Acknowledgments
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