Use of an 'Inter-Linked' Central Solenoid for Plasma Current Ramp-Up in a Tokamak Fusion Reactor

Makoto NAKAMURA, Kenji TOBITA and Hiroyasu UTOH

Japan Atomic Energy Agency, Rokkasho, Kamikita, Aomori 039-3212, Japan

(Received 13 July 2012 / Accepted 30 December 2012)

Use of an 'inter-linked' (IL) central solenoid (CS) in a tokamak fusion reactor is proposed for achieving a sufficient amount of the CS magnetic flux swing for the plasma current I_p ramp-up with keeping the reactor size reasonable. It is shown that a large amount of the flux swing by the IL-CS, compared to the conventional (C) CS, is expected for a tokamak fusion reactor with the fusion power $P_{fus} = 2$ GW and the major radius $R_p < 8.0$ m, and that the IL-CS can generate a marginal amount of the magnetic flux swing to ramp up I_p only by the inductive way for a tokamak reactor with $R_p = 6.5$ m while the C-CS cannot a sufficient amount of the flux swing for the I_p ramp-up.

Keywords: inter-linked central solenoid, magnetic flux swing, plasma current ramp-up, fusion reactor design

DOI: 10.1585/pfr.8.1205007

Functions of a central solenoid (CS) in a tokamak fusion reactor are (1) control of the plasma shape and (2) ramp-up and sustainment of the plasma current I_p by an inductive way. While whether these two functions can be performed depends on the CS size, the reactor size also depends on the CS size. If the CS is designed to perform both functions, the CS size will be large to generate a large amount of the magnetic flux swing sufficient for the latter function. This leads to increase in the reactor size and construction cost. On the other hand, if non-inductive current drive methods (for example [1]) take the place of the latter function, the CS and reactor sizes will be reduced like the compact tokamak fusion DEMO reactor SlimCS [2]. Such methods, however, have not been demonstrated extensively in a tokamak plasma with the medium or large aspect ratio and moderate plasma performance. Moreover, such methods are accompanied with a longer current ramp-up duration than the inductive method because of the return current [3].

In this paper we propose use of an 'inter-linked' (IL) CS in a tokamak fusion reactor in order to resolve the above dilemma, that is, to achieve a sufficient amount of the CS magnetic flux swing for the current ramp-up with keeping the reactor size reasonable. Described below are brief description of the IL-CS concept and assessment of the ability of the IL-CS to generate the magnetic flux swing compared with the conventional CS (C-CS).

A basic idea of the IL-CS concept is to wind a CS such that it is *linked in* a set of toroidal field coils (TFCs) in order to achieve a large amount of the magnetic flux swing by increasing the CS cross section. Comparison of the C-CS and IL-CS configurations is shown in Fig. 1. We performed engineering analysis of the IL-CS of which outer radius, height and thickness were 3.0 m, 10 m and 0.4 m, respectively. The averaged magnetic field in the IL-CS of 4.0 T was given. We analyzed applicability of Nb_3Sn and Nb_3Al for a superconducting wire of the IL-CS. It was found that since the critical current density J_c of a Nb_3Sn wire in the high magnetic field environment is smaller than that of a Nb_3Al wire [4], the operation current density J_{op} normalized to J_c of the Nb_3Sn wire is higher than that of the Nb_3Al wire and above the design limit of J_{op}/J_c for the ITER superconducting magnet design. Accordingly, the IL-CS made of Nb_3Sn have to be thicker than of Nb_3Al to decrease J_{op}/J_c. On basis of this analysis, we chose Nb_3Al as superconducting material of the IL-CS. A detailed description of the engineering analysis will be presented elsewhere.

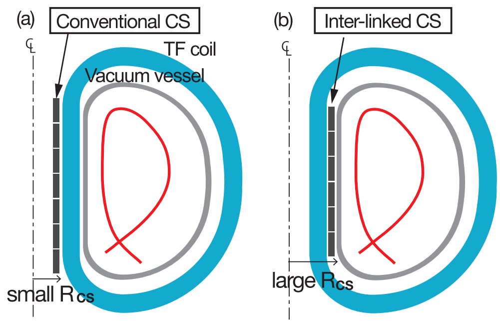

Fig. 1 Key dimensions for tokamak fusion reactors with the (a) conventional and (b) inter-linked central solenoids.

We made comparison of the magnetic flux swings that the IL-CS and C-CS can generate for the I_p ramp-up. For this comparison we considered three cases char-

author's e-mail: nakamura.makoto@jaea.go.jp

Table 1 Design parameters of three cases of fusion reactors with the conventional (C) and inter-linked (IL) central solenoids (CS). R_{cs} : CS outer radius; R_{TF} : outer edge of the TFC inner leg; B_{max} : maximum toroidal magnetic field; R_p : major radius; A : aspect ratio; κ : elongation; q_{95}: safety factor at the 95 % magnetic surface; B_t : on-axis toroidal magnetic field; I_p: plasma current; HH: HH factor.

Case #	1		2		3	
	C	IL	C	IL	C	IL
R_{cs} (m)	1.7	2.9	2.2	3.9	2.7	4.3
R_{TF} (m)	2.9	2.7	3.9	3.7	4.3	4.1
B_{max} (T)	13.3	←	14.0	←	13.1	←
R_p (m)	6.5	←	7.5	←	8.0	←
A	3.0	←	3.5	←	3.5	←
κ	1.8	←	1.6	←	1.6	←
q_{95}	4.70	←	4.46	←	4.40	←
B_t (T)	5.94	5.53	7.35	6.98	7.08	6.75
I_p (MA)	14.7	13.7	13.0	12.3	13.1	12.5
HH	1.30	1.21	1.30	1.23	1.30	1.24

acterized by the reactor size: (1) small size of $R_p = 6.5$ m, (2) medium size of $R_p = 7.5$ m and (3) large size of $R_p = 8.0$ m, where R_p is the major radius. We developed sets of design parameters of the C-CS fusion reactors with the fusion power of 2 GW for the three cases, by using the systems code TOPPER [5]. In this systems code analysis, the (R_{TF}, B_{max}) data developed by the superconducting TFC design code SCONE [6] were used. Here, B_{max} is the maximum toroidal magnetic field and R_{TF} is the outer edge position of the TFC inner leg. Sets of design parameters of the IL-CS reactors were developed so that for each case (i) R_p, A, κ and q_{95} are equal to those of the corresponding C-CS reactor and (ii) the IL-CS outer radius R_{IL-CS} is equal to R_{TF} of the corresponding C-CS reactor, where A and κ are the aspect ratio and elongation, respectively. The sets of the reactor design parameters are summarized in Table 1. In Case # 1 I_p is larger than in the other cases. This is because we focused on the C-CS reactor operation points with $P_{fus} = 2$ GW and because $P_{fus} \propto I_p^2 R_p$ and $I_p \propto R_p/A^2$.

The maximum magnetic flux generated by the C-CS, Φ_{C-CS}, was compared with that required for the I_p ramp-up, Φ_{ramp}. Here,

$$\Phi_{ramp} = (L_p + \mu_0 C_E R_p) I_p, \tag{1}$$

where L_p, $C_E (= 0.45)$, and μ_0 are the plasma inductance, Ejima coefficient, and vacuum permeability, respectively [7]. The averaged and maximum magnetic fields in the C-CS, B_{C-CS} and $B_{C-CS}^{(max)}$, were 12 and 16 T, respectively. The latter value is typical for a Nb_3Al superconducting coil; the former value can be estimated by the relation between B_{C-CS} and $B_{C-CS}^{(max)}$ proposed in [8]. The thickness of the C-CS Δ_{C-CS} of 0.4 m was given. As shown in Fig. 2, $\Phi_{C-CS} <$ Φ_{ramp} in the case of the small size reactor, that is, the C-CS cannot generate a sufficient amount of the magnetic flux for the I_p ramp-up and assistance by a non-inductive way is needed. (Here, the CS was assumed to provide the flux swing in the bipolar mode.) On the other hand, in the other reactor cases with the larger major radius, the C-CS can generate the magnetic flux sufficient for the I_p ramp-up. These results indicate that the reactor size have to be large to ramp I_p up only by the inductive way.

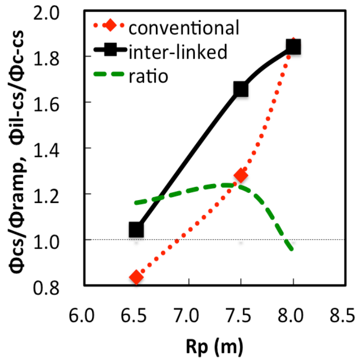

Fig. 2 Ratios of the magnetic flux swings generated by the conventional (dotted line) and inter-linked (solid line) central solenoids, Φ_{C-CS} and Φ_{IL-CS}, to that required for the plasma current ramp-up Φ_{ramp}. The dashed line is Φ_{IL-CS}/Φ_{C-CS}.

We assessed the ability of the IL-CS to generate the magnetic flux swing for the I_p ramp-up. The magnetic field in the IL-CS, B_{IL-CS}, of 4 T was given, which was validated in our engineering analysis. The thickness of the IL-CS Δ_{IL-CS} of 0.2 m is given, which is smaller than Δ_{C-CS} since the fact $B_{IL-CS} < B_{C-CS}$ allows the IL-CS to be slender compared to the C-CS. Comparison of the magnetic flux swings by the C-CS and IL-CS, Φ_{IL-CS} and Φ_{C-CS}, normalized to Φ_{ramp} is shown in Fig. 2. As shown in this figure, a large amount of the flux swing by the IL-CS, compared to the C-CS, can be expected in the cases of the small and medium reactor sizes. In particular, the IL-CS can generate a marginal amount of the magnetic flux swing to ramp up I_p *only by means of the inductive way*, i.e. $\Phi_{IL-CS} \simeq \Phi_{ramp}$, *even* in the case of the small reactor size. This is a remarkable advantage of the IL-CS over the C-CS. On the other hand, when the reactor size becomes larger, this advantage disappears, that is, $\Phi_{C-CS} > \Phi_{IL-CS}$ and $\Phi_{C-CS} > \Phi_{ramp}$. This is because the flux swing generated by a CS is proportional to the product of the averaged magnetic field in the CS and the square of the CS radius, and because $B_{IL-CS} < B_{C-CS}$. Interpolating the calculation results for the three cases, the design window for R_p in which $\Phi_{IL-CS} > \Phi_{C-CS}$ is $R_p < 8.0$ m, and Φ_{IL-CS}/Φ_{C-CS} is maximized at $R_p \simeq 7.5$ m when $P_{fus} = 2$ GW.

Finally, we discuss an impact of use of the IL-CS on the fusion power P_{fus}. The IL-CS configuration can lead to decrease in the on-axis toroidal magnetic field B_t since the

TFC in the IL-CS configuration is distant from the plasma compared to the C-CS configuration. This leads, in turn, to decrease in P_{fus} since $P_{\mathrm{fus}} \propto \beta_{\mathrm{N}}^2 B_{\mathrm{t}}^4/A^4$ where β_{N} is the normalized beta. In the three reactor cases analyzed above, the decrease is estimated to be 18-25 % at constant β_{N} and A. One has to increase β_{N} and/or decrease A to compensate this P_{fus} decrease. It is desirable, therefore, that a fusion reactor with the IL-CS is accompanied with a high performance plasma which, for example, will be demonstrated in JT-60SA [9]. A consistent relationship among the required maximum magnetic field of IL-CS, the current density in the conductor, and the coil cross section (coil width) has to be confirmed even for the transient phase from the limiter configuration to the divertor one. The required magnetic field generally comes from not only supply of the flux swing but also controllability of plasma equilibrium. This paper assesses only the former part. It should be noted that 4.0 T of $B_{\mathrm{IL-CS}}$ in this paper is temporarily determined and is not assured from the viewpoint of the plasma equilibrium. To finalize the required magnetic field of IL-CS, analysis on the plasma equilibrium including the plasma ramp-up phase is required with considering the design consistency between the required maximum magnetic field of IL-CS, the current density in the conductor, and the coil cross section (coil width).

[1] S. Shiraiwa *et al.*, Phys. Rev. Lett. **92**, 035001 (2004).
[2] K. Tobita *et al.*, Nucl. Fusion **47**, 892 (2007).
[3] K. Tobita *et al.*, JAEA-Research 2010-019 (2010) (in Japanese).
[4] T. Takeuchi, IEEE Trans. Appl. Supercond. **12**, 1088 (2002).
[5] K. Tobita *et al.*, Fusion Eng. Des. **81**, 1151 (2006).
[6] H. Utoh *et al.*, J. Plasma Fusion Res. Series **9**, 304 (2010).
[7] S. Ejima *et al.*, Nucl. Fusion **22**, 1313 (1982).
[8] N. Miki *et al.*, JAERI-M 87-153 (1987) (in Japanese).
[9] S. Ishida *et al.*, Fusion Eng. Des. **85**, 2070 (2010).