Tight Aspect Ratio Tokamak Power Reactor with Superconducting TF Coils

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Abstract. Tight aspect ratio tokamak power reactor with super-conducting toroidal field (TF) coils has been proposed. A center solenoid coil system and an inboard blanket were discarded. The key point was how to find the engineering design solution of the TF coil system with the high field and high current density. The coil system with the center post radius of less than 1 m can generate the maximum field of ~ 20 T. This coil system causes a compact reactor concept, where the plasma major and minor radii of 3.75 m and 1.9 m, respectively and the fusion power of 1.8 GW.

1. INTRODUCTION

For the fusion power plant, a realization of a competitive cost of electricity (COE) is the first priority to be adopted by the utility companies. The COE must be low decidedly, but not less important the construction cost (so called capital cost) must be low. The COE's lowering by a scale merit approach is not necessarily received favorably. Tight aspect ratio tokamaks with the aspect ratio A in the range 1.2 ~ 2.0 can offer the possibility of compact fusion reactors in the low capital cost way. This scheme is often called the spherical torus (ST) approach. In the usual ST approach, all non-essential components such as inboard blanket or shield, inboard poloidal coil (PF) systems like a center solenoid (CS) coil system are discarded from the inner side of the plasma. The only customary tokamak component that remains on this side is a single turn copper TF coil center post (CP). In spite of the excellent plasma performances granted by very low aspect ratio less than 1.5, it could not sufficiently compensate the Joule losses in the normal-conducting (NC) TF coil as illustrated in the ARIES-ST power reactor design study [1]. If a super-conducting (SC) TF coil system is used instead, approximately 1 meter of shielding is required between the SC TF coils and the plasma on the inboard side to protect the superconductors from neutron damage and nuclear heating. Consequently, it has been widely recognized that a super-conducting compact tokamak reactor with such the tight aspect ratio would not be feasible. Our recent study, however, opened up the possibility of realizing such a very compact reactor being compatible with the use of SC TF coil system. The key issue is to find the SC TF coil design solution. How high field and how high current density in the coil windings are required for compensating the handicap of the thick shielding?

In section 2, the design guidelines for the tight aspect ratio tokamak reactor are described. The design features of the SC TF coil system are briefly described in section 3. Possible plasma performances within the reliable engineering constraints are described in section 4. The recommendable reactor concept and the discussion for design deepening are described in section 5. The last section consists of design summary.

2. Design Guidelines

The following three guidelines are listed up for compatibility between the tight aspect ratio plasma and the SC TF Coil system.

i) Discard the center solenoid (CS) coil. Non-inductive current generation and sustaining methods such as RF wave or NBI are adopted. In the A-SSTR2 design study, the plasma break-down and current ramp-up scenarios were successfully established without CS coil system by 1.5D simulation code [2]. Furthermore the experimental demonstration by using JT-60 tokamak device was carried out [3].
ii) Discard the inboard blanket. Although the tritium breeding function is limited in the outboard blanket, the tritium breeding ratio requirement of more than 1.05 can be

attainable. Approximately 1 meter of the inboard shielding structure is taken into account.

iii) Increment the TF coil average current density. Not only high magnetic field strength but also high current density are required for the TF coil. This is an essential issue to compensate the "burden" of 1 m shield thickness.

3. TF Coil System

Because of discarding the CS coil system, the TF coil inner legs naturally become a solid-like integrated center post (CP) structure. For compensating ~1 m shield thickness, a slender i.e. high current density CP is indispensable. Fortunately, this TF coil configuration has the structural rigidity and the stored energy is relatively low for its field strength. The SC material is Bi2212/Ag/AgMgSb multi-filament with an operation temperature of 20 K, which is as same as the A-SSTR2 SC material [4]. A calculation method for the material composition optimization among the SC filament, stabilizer, structure (load-carrying) material and cooling channel area has been established. Namely, it is possible to know the maximum achievable field strength under the given CP radius. The design condition is same as the A-SSTR2 [2]. For instance, the SC filament operation current density is 500 A/mm^2 (critical density is 1,000 A/mm^2), the maximum allowable field strength of the SC filament is 23 T, the coil terminal voltage is less than 20 kV, the temperature margin is 3~5 K and the design stress of structure material JJ1 [5] is S_m = 800 Mpa. The maximum field (B_{MAX}) dependence on the CP radius was evaluated by 3D FEM stress calculation. The result is shown in Fig.1.

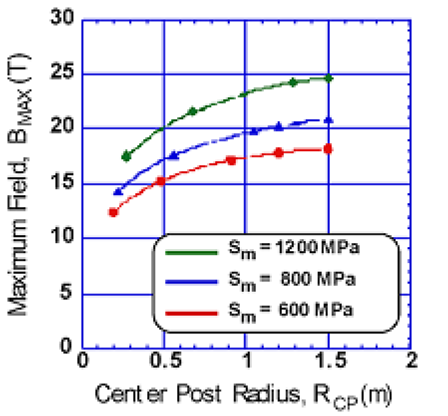

Fig. 1 Maximum field dependence on center post radius. In our study, the design stress of 800 MPa is addopted.

4. Plasma Performance

An acceptable nuclear heating rate of 0.5 mW/cm^3 in the TF coil conductors corresponds to the distance of 95cm between plasma surface and the TFC as to the torus inboard region as shown in Fig.2. An SiC/SiC composite for the power core structure material and VH_2 for the bulk shield material are introduced. A coolant material is liquid lithium. The outboard radial build shown in Fig.2 satisfies the local TBR (tritium breeding ratio) of 1.3, where the liquid lithium is not only the coolant but also the tritium breeding material.

A plasma ellipticity is directly and deeply connected with the reactor concept. While a highly elliptic plasma is preferable for attaining a high beta plasma and for receiving a high plasma current, it become difficult to obtain plasma MHD equilibrium solutions and to control plasma vertical position. From the plasma equilibrium point of view, the high elliptic plasma requires the PF coil position near the equatorial plane of the tokamak machine. The well maintenability is a high priority matter in our reactor design. The power core components are designed for horizontal insertion and withdrawal of entire sectors. Therefore we have no PF coil near the equatorial plane. And the use of low activation material is another high priority matter. A vanadium alloy is adopted to the passive stabilizing shell structure. A electric resistivity of the vanadium alloy is 30 times

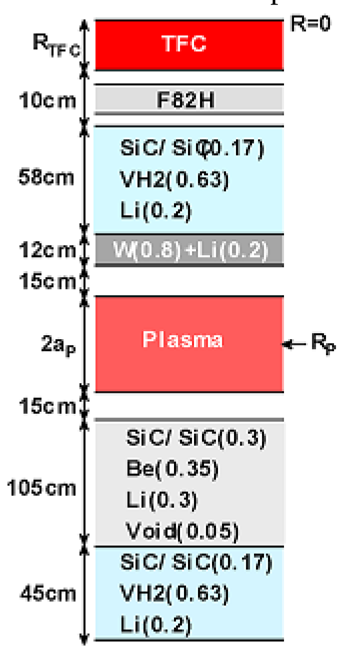

Fig. 2 Radial build based on neutronics (shield and TBR) calculation.

higher than a copper material. For the replaceable torus sector unit, the possible shell structure must be the so called "saddle loop" structure shown in Fig.3. The plasma ellipticity in our design has been decided to be 2.1. The normalized beta in our design is decided with consideration of certain design margin based on the maximum achievable values of Ref. [6]. The plasma performance calculation has been carried out under the physics constraints which are the energy confinement of HH_{y2}=1.8 and the plasma density of n/n_{GW}=1.0. A self-consistent plasma parameter set is calculated for the each given CP radius value. Figure 4 shows the dependences of R_P (plasma major radius), a_P (plasma minor radius), A (plasma aspect ratio), P_F (fusion power), P_n (average neutron wall load), I_P (plasma current), f_{BS} (bootstrap current fraction), P_{NB} (neutral beam power required for plasma current drive) and Q (energy multiplication factor) on the R_{CP} (TF coil center post radius). The plasma average temperature is assumed to be 20 keV.

When the CP radius is in slender region thinner than 60 cm, the field strength on plasma axis is considerably low. This leads the plasma sizes (major and minor radii), the plasma current, and the required current drive power very large. It is not acceptable performance for the power reactor. On the other hand, in the thicker region than 1 m, despite the reduction tendencies of the plasma sizes and current are saturated, the TF coil construction cost must be high. And the high field leads the neutron wall load high. Maximum allowable wall load is limited less than 5 MW/m^2. The maximum wall load of 5 MW/m^2 corresponds to the average value of 3.5 MW/m^2. Therefore, the CP radius of 90 cm is chosen for our new reactor concept named VECTOR (VEry Compact TOkamak Reactor). Major specifications of VECTOR are listed in Table 1.

The plasma equilibria during the ramp-up phase are found within the reasonable ampere-turns less than 100 MAT. In a non-inductive plasma initiation (breakdown) and current ramp-up scheme, the NBI (neutral beam injection) should be replaced by RFW (radio frequency wave) heating and current drive to avoid high shine through and orbital losses in the low density, low current (high

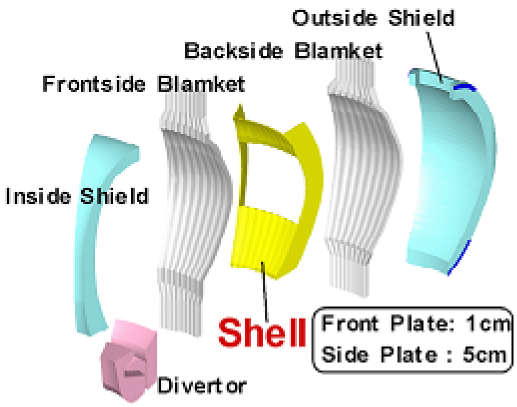

Fig.3 Saddle loop shell structure with the distance of 30-40 cm from plasma surface.

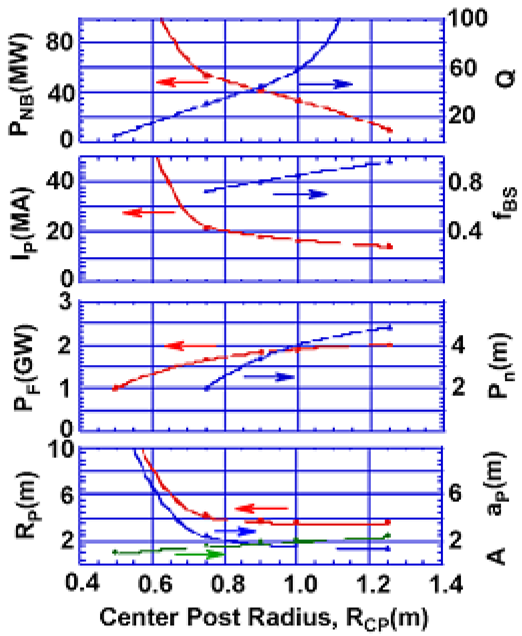

Fig.4 Plasma parameters for given center post radius, which satisfies HH_{y2}=1.8 and n/n_{GW}=1.0. The maximum achievable field strength is uniquely decided as shown in Fig.1.

Table 1 VECTOR Major Prameters

Plasma Major Radius :	R_P = 3.75 m
Plasma Minor Radius :	a_P = 1.9 m
Plasma Ellipticity :	κ = 3.75 m
Plasma Current :	I_P = 18.3 MA
Normalized Beta :	β_N = 3.75
Fusion Power :	P_F = 1.8 GW
Neutron Wall Load :	P_n = 3.5 MW/m2
Maximum Field in TFC :	B_{MAX} = 19.6 T

q) plasma. Here, an EC (Electron Cyclotron) system is considered for heating and current drive.

5. VECTOR Mechanical Configuration

The replacement scheme of the power core component and the vacuum boundary layout are very simple in comparison with the conventional tokamak device. The power core components are designed for horizontal insertion and withdrawal of entire sectors. Since the structure material of the power core components is not a conducting material, the eddy currents are not induced on the components by the plasma disruption event. Therefore, the firm connection (it is an obstacle to the quick replacement) between the adjacent power core sectors and/or the assembling components is not needed. The torus segmentation and replacement scheme are strongly related to the vacuum boundary layout.

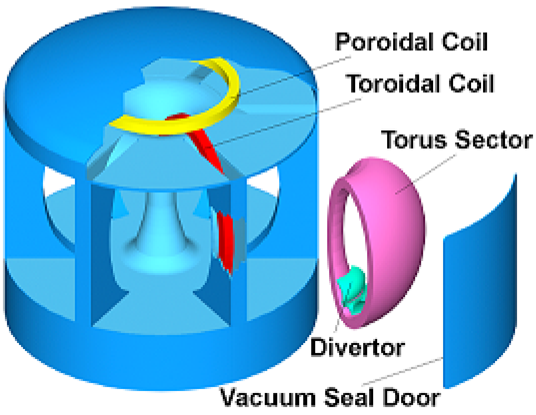

Fig.4 Schematic drawing of vacuum boundary and torus segmentation for VECTOR. The vacuum seal between adjacent torus sectors is needless.

A high degree of vacuum integrity is required for both the plasma chamber and the cryostat for the SC coil system. In the VECTOR configuration, a single-walled containment of the plasma chamber constitutes also the continuous toroidal vacuum-tight containment of the cryostat as schematically shown in Fig.5. The bell-jar envelops all the TF and PF coils. Its outer cylindrical part, provided with certain number of windows for the withdrawal of torus sectors. The bird’s-eye view of VECTOR is shown in Fig.6.

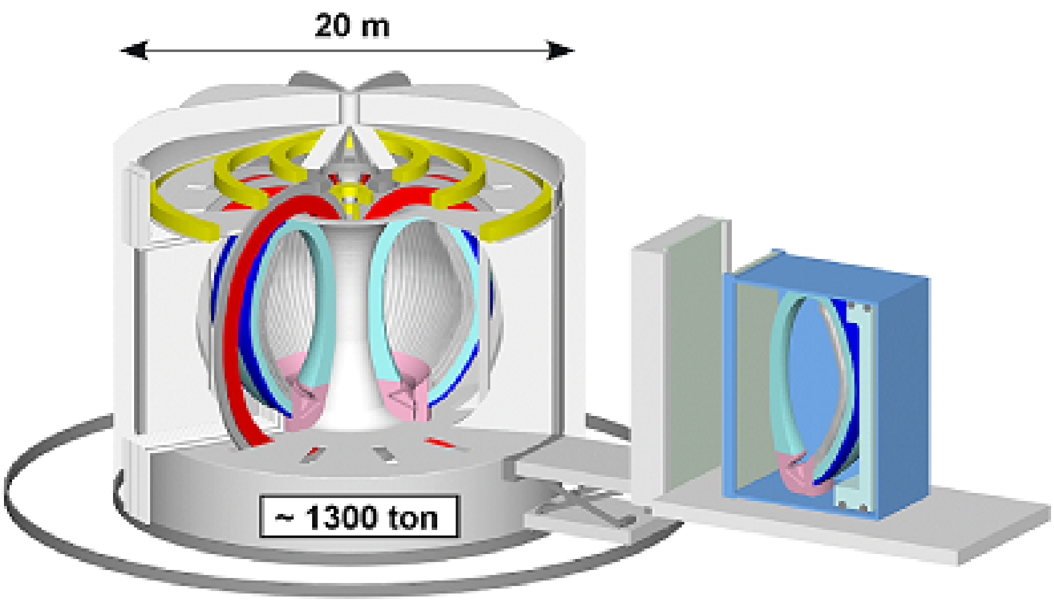

Fig.6 Birds-eye view of VECTOR.

6. Summary

i) Not only low COE (cost of electricity) but also low capital cost (construction cost) are required for the fusion power plant.

ii) A tight aspect ratio tokamak reactor is promising concept answering for the above requirements. But the concept with normal conducting TF coil does not answer for the COE requirement [1].

iii) A key point of the attractive tokamak with a tight aspect ratio is to find the engineering design solution for the super-conducting TF coil with a high field and high current density.
iv) The TF coil system with the center post radius of less than 1 m can generate the maximum field strength of ~ 20 T.
v) Such the TF coil system mentioned above causes a compact reactor concept, where the plasma major and minor radii of 3.75 m and 1.9 m, respectively and the fusion power of 1.8 GW.

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REFERENCES

[1] Najimabadi, F., et al., Proc, of 17th IAEA Intl. Conf. of Fusion Energy, Yokohama, Japan (1998).
[2] Nishio, S., et al., Proc, of 18th IAEA Intl. Conf. of Fusion Energy, IAEA-CN-77/FTP2/14, Sorrento, Italy (2000) or Nishio, S., et al., J. Plasma Fusion Res. Vol.78, No.11 (2002).
[3] Takase, Y., et al., J. Plasma Fusion Res. Vol.78, No.8 (2002) 719-721.
[4] Ando, T., et al., Fusion Eng. and Des. 58-59 (2001) 13-16.
[5] Ishida, K., et al., Advances in Cryogenic Engineering (Materials) **44**, 73-80 (1998).
[6] Miller, R.L., et al., Phys. Plasma **4**(4), April (1997).