FEASIBILITY STUDY FOR VERY HIGH ASPECT RATIO TOKAMAK FUSION REACTOR

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Abstract

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Very high aspect ratio (A) tokamak reactor is proposed as a future power plant. Plasma size and operation performance are surveyed over a wide range of A. To reduce the plasma minor radius, relatively high toroidal magnetic field of 18T is adopted. When the plasma major and minor radii are 12 m and 1.5 m (A=8), steady-state operation with 3 GW of fusion power and large fusion gain (>40) is possible. 90% of bootstrap current fraction is achieved by optimizing the density profile for the given MHD equilibrium. Toroidal field (TF) coil size is estimated by considering the field ripple, NBI beam line and coil case stress. Although overall TF-coil height and width are smaller than those of ITER, cross-sectional area of the TF-coil leg becomes 4 times larger than that of ITER to compensate the vertical force.

1. INTRODUCTION

Aspect ratio A (ratio of plasma major radius R_P to minor radius a_P) is a key parameter to determine the size of a tokamak fusion reactor. A relatively high-A (~4.5) DEMO class reactor was designed from the viewpoint of steady-state operation [1]. On the other hand, an ultra low-A (~1.5) compact reactor was proposed [2]. Recently, a very high-A (=8) reactor with SiC core structure has been proposed and shown to have an advantage from the viewpoint of maintenance feasibility [3]. Very high-A reactor was originally investigated as a core plasma of the second stability tokamak [4]. In this study, however, we concentrate on the first stability plasma (normalized beta β_N~3) and investigate the feasibility of very high-A reactor according to ITER physics guideline [5] and design constraints similar to those of SSTR [1].

2. ASPECT RATIO AND PLASMA PERFORMANCE

To realize a very high-A reactor, the plasma minor radius a_P should be minimized while keeping the major radius R_P sufficiently small. Therefore, high toroidal field B_T at the plasma center is required. In this paper, we assume the maximum toroidal field B_{TMAX}=18 T. The fusion power (P_{FUS}=3 GW), MHD safety factor (q_Ψ~3), plasma elongation (κ~1.4) and inboard shield thickness (= 1.4 m) are fixed to compare the plasma performance. Figure 1 shows the plasma size and performance in steady-state operation for various aspect ratio. Since the plasma current becomes small and bootstrap current fraction increases in a very high-A (>8) plasma, the required current drive power (1.3 MeV NBI is used) decreases rapidly and large fusion gain Q is achieved. If an advanced divertor concept is not considered, the divertor peak heat load is simply estimated by $W_{DIV} = P_{DIV} \sin\theta_{DIV} / (2\pi R_{DIV} \Delta_P)$, where P_{DIV} is the power to the divertor, θ_{DIV} is the plate inclination angle, R_{DIV} is the striking point radius and Δ_P is the heat flux half width, respectively. A plasma with large major radius R_P has a substantial advantage to reduce the peak heat load since the target radius R_{DIV} is approximately proportional to R_P. Furthermore, large R_P is favorable also for the heat diffusion across the field line because of long connection length between plasma and target plates. By using simplified scaling models [6] for Δ_P, the peak heat load is estimated. Figure 1-d) shows the peak heat load as a function of A. The absolute value of the heat load is normalized to 10 MW/m^2 for the parameters same as ITER. In this figure, the current drive power is not included in P_{DIV}. It is seen that the divertor heat load is reduced greatly in a very high-A reactor due to the large plasma surface area, large major radius and strong synchrotron radiation. Adopting radiative divertor (by seeding 0.2% of argon), acceptable divertor condition with low plasma temperature can also be achieved. Finally, we select the plasma with A=8 for the reference reactor design where the minor radius is close to the minimum and the fusion gain is large enough. (If the confinement enhancement factor H≤2 is assumed, the maximum A is also about 8).

3. TOROIDAL COIL SIZE AND REACTOR CONCEPT

In a very high-A reactor, TF-coil design is one of the most important issues since the hoop force of TF-coil becomes very large. Furthermore, outboard space Δ_{TF2} between TF-coil leg and plasma surface should be increased to keep the field ripple sufficiently small. Figure 2 shows the field ripple as a function of Δ_{TF2}. Here, N_{TFC} is the number of TF-coils. The required Δ_{TF2} is about 3.4 m when field ripple is 1% and N_{TFC}=20. In this case, vertical force of TF-coil is about 2.7 times larger than that for SSTR [1]. Critical current density for 18 T magnet is assumed to be 3/4 of that for 16.5T magnet [1]. Therefore, the cross-sectional area of TF-coil case is estimated to be about 5 m^2. Figure 3 shows the plan view of the very high-A reactor. Tangential injection of neutral beam of diameter 1m is possible. Large TF-coil case also increases the distance between the plasma surface and poloidal field (PF) coils. This implies the large PF-coil stored energy and large coil current. Figure 4 shows the results of equilibrium calculation for the very high-A reactor. Stored energy at high β_P (~3.3) operation is about 30 GJ and the maximum coil current is 35 MA (at the divertor coil). Figures 4-b) and 4-c) show the current profiles and MHD safety factor profile corresponding to a). Here, we optimize the bootstrap current profile by calculating the density (n) and temperature (T) profiles which are consistent with the equilibrium pressure P. When the flux average bootstrap current required for the equilibrium is written by $\langle j_{BS}B\rangle=-1.46A^{-0.5}I(\Psi)P(\Psi)(c_N n'/n+c_P P'/P)=f(\Psi)$, the optimized density profile is given by $n(\Psi)=n(\Psi_S)\exp\left[\frac{1}{c_N}\int_{\Psi_S}^{\Psi}\left\{\frac{A^{0.5}f(\Psi)}{1.46I(\Psi)P(\Psi)}+c_P\frac{P'}{P}\right\}d\Psi\right]$. Here, I is the toroidal flux function and s denotes the plasma surface. Figure 4-d) shows the calculated density and temperature profiles. In this case, the bootstrap current fraction is about 90%.

4. DISCUSSIONS

Optimization of the maximum toroidal field B_{TMAX} and TF-coil design is a critical issue in a very high-A reactor. If a plasma with $\beta_N=5$ in reversed shear mode can be achieved, the required B_{TMAX} is reduced from 18 T to 14.5 T for the same fusion power and the design requirements for the TF-coil are mitigated. Increasing the plasma elongation also reduces B_{TMAX}. In this case, however, the bootstrap current decreases while the plasma current increases. Therefore, the steady-state operation capability degrades. Although the confinement margin of very high-A plasma is same level as ITER, there is a large uncertainty in the confinement scaling of a high-A plasma. In a high-A plasma, there is a possibility that the ripple loss of alpha particle is enhanced. In this case, the number of TF-coil should be increased. Relative construction cost considering the amount of structural material with the same design constraints on coil case stress is estimated. Preliminary estimate shows that the total cost is 30% higher than that for ITER. Size reduction of in-vessel components and decrease of replacement frequency of the first wall material could contribute the cost reduction.

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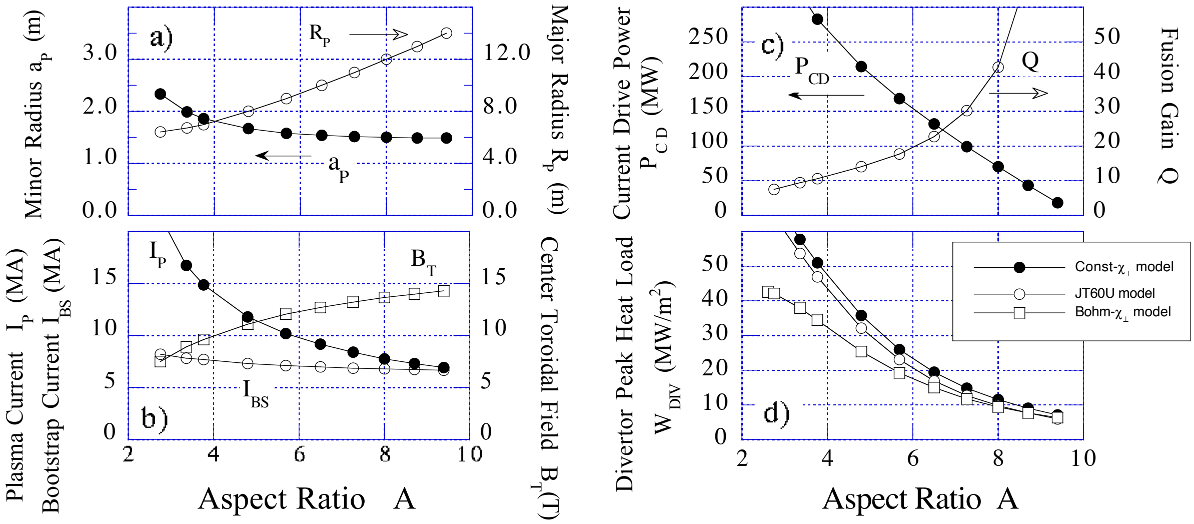

FIG. 1. Aspect ratio A and steady-state operation performance. Here, the fusion power (=3 GW), normalized beta (=3) and the maximum toroidal field (=18 T) are fixed.

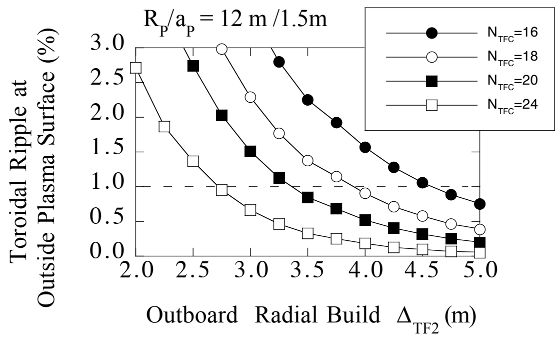

FIG. 2. Outboard radial build Δ_{TF2} and toroidal field ripple at the plasma surface.

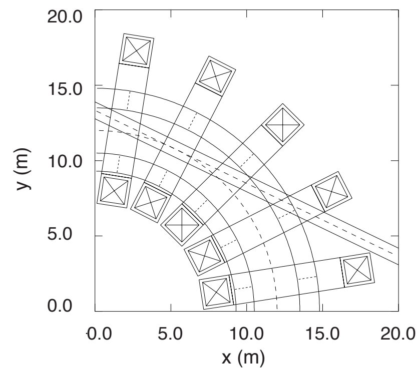

FIG. 3. Toroidal field coils and NBI beam line.

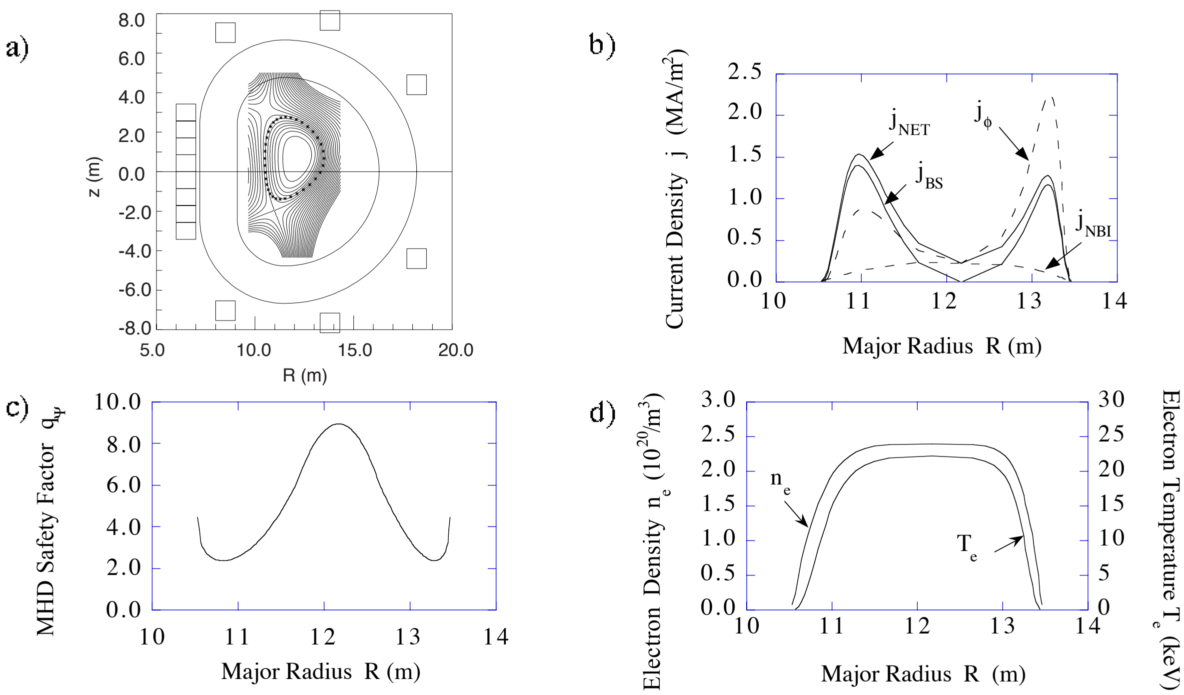

FIG. 4. Total current (j_ϕ), net current (j_{NET}), bootstrap current (j_{BS}), NBI current (j_{NBI}), MHD safety factor (q_Ψ), electron density (n_e) and electron temperature (T_e) profiles.