Design Windows and Economic Acceptability of the 3GW Low Aspect Ratio Tokamak Power Plant with Superconducting Coils

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Spherical Tokamaks (STs) have attractive characteristics which are natural high elongation, high beta, high bootstrap current fraction, and so on. Since 1990s, novel ST power plant concepts: ARIES-ST [1] and STPP [2] have been proposed using advantages of ST plasma over the conventional tokamak plasmas: extremely high bootstrap current fraction (f_{BS} ~ 95%) and higher β limit (β_t > 50%). However the preceding ST reactor designs assumed challenging reactor engineering conditions, i.e., normal-conducting TF coils, CS-free configuration, and high neutron load (P_{wall} ~ 4MW/m^2) in comparison with conventional tokamak reactors. Thereafter, ST reactors have been mainly focused not on the power plant but on the application to a component test facility such as CTF [3] using its characteristics of high fusion neutron fluxes.

Here we reassess a possibility of the ST as a power plant using the conservative reactor engineering constraints for the first time. We explored new design windows of 3GW superconducting low aspect ratio (A) tokamak power plants which have appropriate reactor size and low-magnetic field. An extensive parameters scan which cover all range of superconducting low-A tokamak reactors was conducted using the system code: TPC code [4] with a new f_{BS} scaling [5] as shown in Table 1. The superconducting TF coils were designed using the SCONE code [6], where Nb_3Al is chosen as the superconducting material and the number of TF coils is set to 12. The maximum toroidal field produced by the superconducting TF coils is shown in Figure 1. Inboard radial build to plasma is consist of the CS coils (R_{CS}), TF coils (R_{TF}), and shield/blanket (Δ_{TF}).

From about 2 million operation points obtained in the above parameters scan, we established five constraints as feasible constraints for fusion power plant in order to determine the optimum and achievable ones as follows.

(ⅰ) Power plant plasma constraint: we choose the operation points which are suitable for advanced reactor plasma with 70% ≦ f_{BS} ≦ 100%, Q ≧ 20, f_{GW} ≦ 1.5, HH_{y2} ≦ 2.0, where Q, f_{GW}, and HH_{y2} are the energy gain, the Greenwald density fraction, and the confinement enhancement factor.

Table 1 Scan parameters, fixed parameters, and calculation modes in the extensive low-A tokamak reactor system study for the TPC code.

Parameter		Range	Points
Scan parameters			
Aspect ratio	A	1.4 - 2.2	5
CS radius	R_{CS} (m)	0.3, 0.5, 0.7	3
TF width	R_{TF} (m)	0.2 - 0.8	7
Gap between TF and plasma	Δ_{TF} (m)	0.8 - 1.4	4
Density profile index	*an*	0.3, 0.5, 0.7	3
Temperature profile index	*at*	1.0, 1.5, 2.0	3
Elongation at 95% magnetic surface	κ_{95}	2.0 - 3.0	11
Safety factor at 95% magnetic surface	q_{95}	3.0 - 8.0	11
Volume-averaged temperature	$\langle T \rangle$ (keV)	10 - 20	11
Fixed parameters			
Fusion output	P_{fus} (GW)	3.0	
Triangularity at 95% magnetic surface	δ_{95}	0.3	
Safety factor at axis	q_0	1.5	
Beam energy	E_{beam} (MeV)	1.5	
Beam injection radius	R_{tang} (m)	$R_p + a_p/2$	
Impurity	(%)	0.1 (He), 0.001 (Ar)	

Calculation mode

・Steady-state (iteratively calculate NBI power)
・Double null diverter configuration
・Fix fusion output

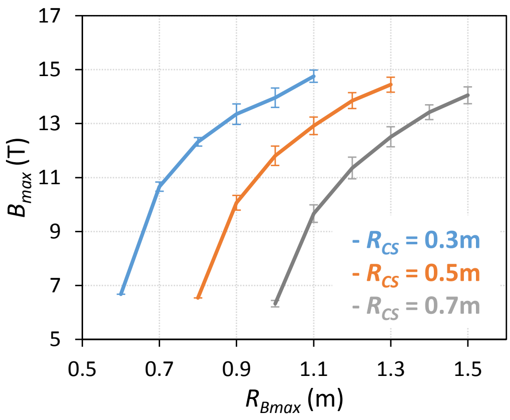


Fig. 1 Maximum toroidal magnetic field produced by the superconducting TF coils calculated by the SCONE code.

(ii) Blanket constraint: local tritium breeding ratio (TBR) is calculated by varying the width of the inboard blanket against the neutron load. Residual space is used for shielding. From the given conditions of the TBR ≧ 1.05 and shield of the TF coil $\Phi_{TFC} \leqq 1.0\times10^{22}$n/m^2, where Φ_{TFC} is the neutron flux to the TF coils, average neutron load with respect to each Δ_{TF} is required as $P_{wall} \leqq$ 0.049, 0.172, 0.600, 2.097MW/m^2 at Δ_{TF} = 0.8, 1.0, 1.2, 1.4m.

(iii) β limit constraint: we deduce a necessary condition for the β_N limit form the Lin-Liu scaling [7] with respect to each aspect ratio A and elongation κ. Feasible κ is expected to be 3.0 assuming the stabilizing shell for the vertical instability.

(iv) Plasma confinement constraint: we line the accomplished confinement as f_{GW}-HH_{y2} boundary of $0.7f_{GW} + HH_{y2} - 2.3 \leqq 0$ from the international global H-mode confinement database [8]. Note the datasets were obtained before 2005, and neither non-inductive shots which are expected in the steady-state operation nor enough ST experiment data (NSTX and MAST) are included.

(v) Diverter heat load constraint: simple scaling calculation in the TPC code (W_{div}) is used for the constraint: $W_{div} \leqq$ 50MW/m^2. This value corresponds to the plasma exhaust power of 450~550MW, and can be reduced to the diverter target heat load below 10MW/m^2 by the gas puff and sophisticated diverter configuration [9].

Using the above five constraints, we obtained 11230 operation points for the 3GW low-A tokamak reactors. Their cost of electricity (COE, $2013) were evaluated using the classical cost model by CRIEPI [10]. Figure 2 shows design point distribution for (a) wall load and (b) plasma elongation and major radius with respect to aspect ratio. Figure 3 shows parameter

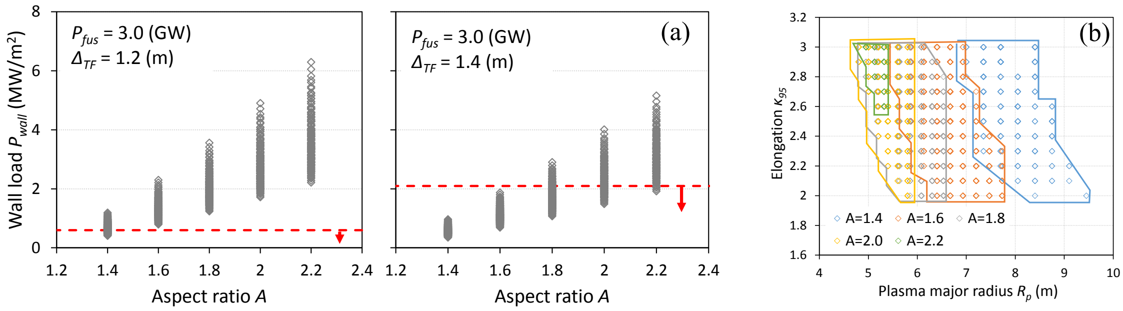


Fig. 2 Design point distribution for (a) wall load and (b) plasma elongation and major radius with respect to aspect ratio.

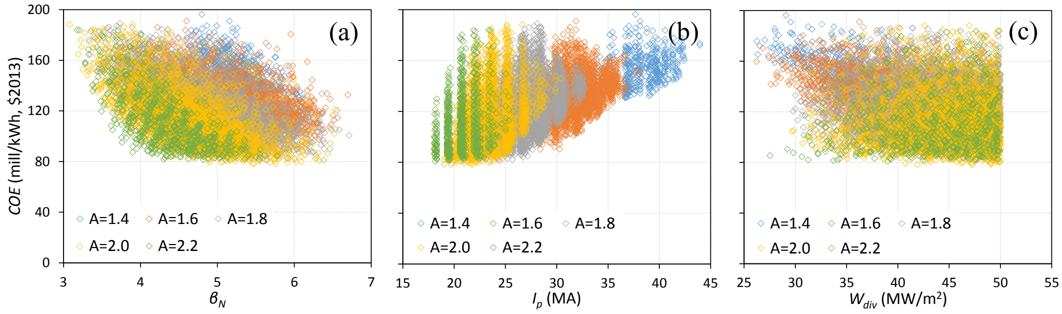
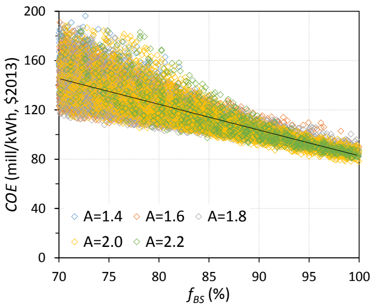


Fig. 3 COE dependence on (a) normalized beta β_N, (b) plasma current I_p, and (c) diverter heat load W_{div}.

dependence on COE. With increasing aspect ratio, plasma parameters required for these plants, i.e. R_p, β_N, and I_p, decrease, but engineering parameters, i.e., P_{wall}, W_{div}, and κ increase. As shown in Figure 4, COEs of these 3GW low-A tokamak plants depend on the f_{BS} values regardless to the aspect ratio. It is because these reactors reduce the coil costs to the limit, and their COEs decisively depend on the net electricity output. Compared with the estimated future energy COEs reported by the Japanese government [11], the breakeven price of the fusion power plant in Japan will be 120mill/kWh, corresponding to f_{BS} of 70 ~ 88% of these low-A tokamak power plants.

In order to choose optimized design point, we considered additional constraints: maintenance scheme and TF ripple. Since we assume inboard blanket to ensure TBR above a unity, horizontal sector transport maintenance is desirable for high plant availability (f_{av}). Also, TF ripple should be small enough to avoid α particle loss. From these two point of view, we selected feasible and optimized design windows as shown in Table 3. Sensitivity analysis for κ_{95} = 2.8 and q_{95} = 6.5 (I_p = 24MA) indicates quantitative trade-off correlation among evaluation-standard parameters such as f_{GW} and HH_{y2} (not shown here). Finally, one of the optimized design point of the 3GW ST power plant was selected from the design windows indicated in Table 3, as shown in Table 4.

In conclusion, we have conducted a series of analysis for the conceptual design of the superconducting ST (low-A) reactors. We established the idea of feasible constraints for the



mill/kWh	f_{av}	2030
Coal	80%	120 ~ 124
Petroleum	80%	278 ~ 311
	50%	293 ~ 327
	10%	454 ~ 489
LNG	80%	127 ~ 133
	50%	136 ~ 143
Fission	70%	104 ~
Hydro	45%	124
Bioenergy	80%	-
Wind	20%	103 ~ 202
	30%	100 ~ 269
Geothermal	80%	107 ~ 135
Solar PV	12%	141 ~ 308

Fig. 4 COEs of the low-*A* tokamak power plants with respect to f_{BS}, and those of the other energy sources in Japan in 2030.

Table 3 Feasible and optimized design windows of the ST power plants.

CS coil radius	R_{CS} (m)	0.5
TF coil width	R_{TF} (m)	0.5
Sheild/Blanket	Δ_{TF} (m)	1.4
Plasma major/minor radius	R_p/a_p (m)	5.4/3.0
Elongation	δ_{95}	0.3
Effective charge	Z_{eff}	1.51
Fusion output	P_{fus} (GW)	3.0
Toroidal field (maximum)	$B_{t,max}$ (T)	~12
Toroidal field (axis)	B_{t0} (T)	~2.2
Triangularity	κ_{95}	2.6~3.0
Safety factor at 95% magnetic surface	q_{95}	5.5~7.5
Plasma current	I_p (MA)	22.1~27.7
Volume-averaged temperature	$<T_e>$ (keV)	11~19
Energy gain	Q	21~128
Normalized toroidal beta	β_N	4.5~6.4
Toroidal beta	β_t (%)	17~27
Poloidal beta	β_p	1.45~2.25
Greenwald density fraction	f_{GW}	1.01~1.50
Confinement enhancement factor	HH_{y2}	1.00~1.56
Bootstrap current fraction	f_{BS} (%)	71~90
NBI power	P_{NBI} (MW)	24~142
Average neutron load	P_{wall} (MW/m^2)	1.45~1.64
Divertor heat load (before radiation)	W_{div} (MW/m^2)	37~50
Power exhaust	P_{div} (MW)	472~539
Net electricity output	P_{net} (MW)	684~893
Cost of Electricity	COE (mill/kWh)	95~171

fusion power plant for the purpose of determining the design windows from the extensive parameters scan by the systems code. Although our design assumes relatively moderate plasma and fusion engineering parameters, attractive and economical ST power plant is achievable thanks to advanced characteristics of ST plasma. These new designs not only use high bootstrap current fraction and low-magnetic field effectively but also apply acceptable wall load by increasing the reactor size appropriately. 2D plasma physics design of the ST reactor is in progress, and further constraints such as plasma ramp-up and plasma control will be considered in order to improve reliability of the conceptual design.

Table 4 Optimized design point of the 3GW ST power plant selected from the design windows indicated in Table 3.

κ_{95}	2.8
q_{95}	6.5
I_p (MA)	24.0
$<T_e>$ (keV)	15
Q	56.4
β_N	6.07
β_t (%)	21.9
β_p	1.98
f_{GW}	1.21
HH_{y2}	1.41
f_{BS} (%)	79.3
P_{NBI} (MW)	53.2
P_{wall} (MW/m^2)	1.54
W_{div} (MW/m^2)	48.3
P_{div} (MW)	503
P_{net} (MW)	841
COE (mill/kWh)	103

References

[1] F. Najmabadi *et al.*, Fusion Eng. Des. **65**, 143 (2003).

[2] R. J. Akers *et al.*, Nucl. Fusion **40**, 1223 (2000).

[3] Y-K. M. Peng *et al.*, Plasma Phys. Control. Fusion **47**, B263 (2005).

[4] H. Fujieda *et al.*, JAERI-M, 92 (1992) (in Japanese).

[5] K. Gi *et al.*, submitted to Fusion Eng. Des.

[6] H. Utoh *et al.*, J. Plasma Fusion Res. **9**, 304 (2010).

[7] Y.R. Lin-Liu *et al.*, Nucl. Fusion **44**, 548 (2004).

[8] The International Global H-mode Confinement Database, http://efdasql.ipp.mpg.de/hmodepublic/.

[9] N. Asakura *et al.*, Nucl. Fusion **53**, 123013 (2013).

[10] T. Yoshida *et al.*, CRIEPI Rep. T94001 (1994) (in Japanese).

[11] National Policy Unit, Cost evaluation committee report (2011) (in Japanese).