研究論文

Parameter Sensitivity of ITER Type Experimental Tokamak Reactors toward Compactness

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Abstract

Dependence of scale of ITER type experimental tokamak reactor on H_L factor (the ratio of energy confinement time τ_E necessary for ignition to $\tau_E^{\mathrm{ITER\ P}}$ of L mode scaling), the ratio ξ of magnetic flux of OH coil to the plasma inductance L_pI_p, elongation ratio κ, the distance Δ between the plasma surface and toroidal field (TF) coil conductor and technically possible maximum fields $B_{t,Mx}$ of TF coil are studied. When the required H_L factor is increased from 2.75 to 3.77 and the distance Δ is decreased from 1.55 m to 1.3 m in the reference case of ITER design parameters, the major radius R is reduced from $R = 8.14$ m to 6.36 m, while keeping $\xi = 1.35$. However the decrease of ξ, while keeping H_L constant, is less effective to make device more compact due to the dependence of H_L factor on the parameters. The increase of κ ratio and the increase of the maximum field $B_{t,Mx}$ is also effective to reduce the scale.

Keywords:
experimental tokamak reactor, ITER, ignition condition, H factor, compactness

1. Introduction

Recent developement of enhanced confinement researches, such as VH mode, high β_p H mode and reversed shear configuration mode, brings the possibility to relax the requirement to H_L factor of energy confinement time in order to design reactors in more compact way. Furthermore it becomes to be realistic to increase the maximum magnetic field of TF coil more than 15 T due to the developement of superconducting coil technology. Therefore it is meaningful to study the sensitivity of the scale of reactors on parameters such as requires H_L factor and the possible maximum magnetic field of TF and OH coils and so on.

2. I_p-A Design Space

When plasma current I_p and aspect ratio A are specified ($A = R/a$, R: major radius, a: minor radius), the other parameters of tokamak device are determined, if the effective safety factor q_{eff} at the plasma boundary, the elongation ratio κ, triangurarity δ of plasma cross section, the distance Δ of plasma boundary and the conductor of TF (toroidal field) coil, the maximum field $B_{t,Mx}$ of TF coil are given [1]. The effective safety factor q_{eff} is given by [2]

$$q_{\mathrm{eff}} = q_I\left(1 + A^{-2}(1 + (\bar{\Lambda}^2/2)\right) \times (1.24 - 0.54\kappa + 0.3(\kappa^2 + \delta^2) + 0.13\delta)\,, \tag{1}$$

where $q_I \equiv (5K^2a[\mathrm{m}]B_t[\mathrm{T}]/AI_p[\mathrm{MA}])$, $\bar{\Lambda} = \beta_p + l_i/2$ and $K^2 \equiv (1 + \kappa^2)/2$. B_t[T] is toroidal field at the plasma (geometrical) center and is equal to

$$B_t = B_{t,Mx}\frac{(R - a - \Delta)}{R}\,, \tag{2}$$

where Δ is the distance of plasma boundary and the position of maximum field of TF coil as is shown in

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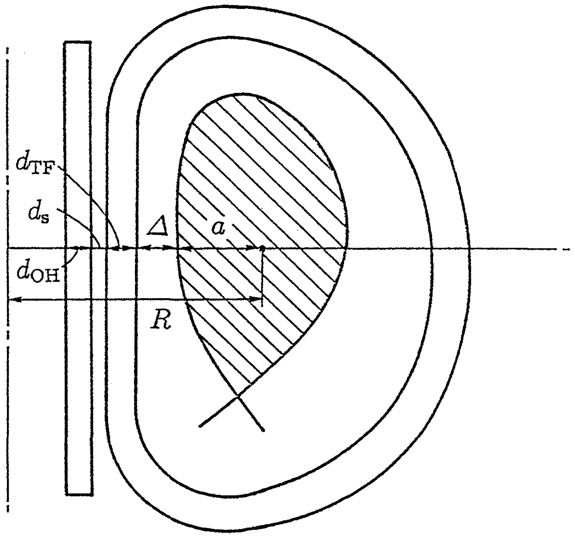

Fig. 1 Geometry of plasma, TF coil and OH coil.

Fig.1. As is clear, R, a and B_t are determined, when I_p and A are specified.

The ratio ξ of the flux swing $\Delta\Phi$ of OH coil and the plasma inductance L_pI_p is given by

$$\xi \equiv \frac{\Delta\Phi}{L_pI_p} = \frac{5B_{\mathrm{OH,Mx}}[\mathrm{T}]\,[(R_{\mathrm{OH}}+d_{\mathrm{OH}}/2)^2+d_{\mathrm{OH}}^2/12]}{I_p[\mathrm{MA}]R[\ln(8A/\kappa^{1/2})+l_i-2]}, \quad (3)$$

where $R_{\mathrm{OH}} = R - a - \Delta - d_{\mathrm{TF}} - d_s - d_{\mathrm{OH}}$, d_{TF} and d_{OH} being the thichness of TF and OH coil conductors and d_s being the distance between TF and OH coil conductors. $B_{\mathrm{OH,Mx}}$ is the maximum field of OH coil. The ratio of electron pressure to that of toroidal field is

$$\beta_e = 4.026\times10^{-2}\frac{\langle n_eT\rangle_{20,\mathrm{keV}}}{B_t^2[\mathrm{T}]}, \quad (4)$$

where $\langle n_eT\rangle_{20,\mathrm{keV}}$ is in unit of 10^{20} m^{-3} keV. The total neutron fusion output power P_n is

$$P_n = \int\frac{n_{\mathrm{DT}}^2}{4}\langle\sigma v\rangle_{\mathrm{DT}}Q_n\mathrm{d}V = \frac{Q_n}{4}V\langle n_{\mathrm{DT}}^2\langle\sigma v\rangle_{\mathrm{DT}}\rangle_{\mathrm{av}},$$

where Q_n = 14.06 MeV and $\langle x\rangle_{\mathrm{av}}$ denotes the volume average of x. The fusion rate $\langle\sigma v\rangle_{\mathrm{DT}}$ near T = 10 keV is approximated by

$$\langle\sigma v\rangle_{\mathrm{DT}} = 1.1\times10^{-24}T_i^2[\mathrm{keV}]\,[\mathrm{m^3/s}],$$

and the following Θ ratio is introduced.

$$\Theta \equiv \frac{\langle n_{\mathrm{DT}}^2\langle\sigma v\rangle_{\mathrm{DT}}\rangle_{\mathrm{av}}[\mathrm{m^3/s}]}{1.1\times10^{-24}\langle n_{\mathrm{DT}}^2T_i^2[\mathrm{keV}]\rangle_{\mathrm{av}}[\mathrm{m^3/s}]}. \quad (5)$$

Then the total neutron fusion output power is

$$P_n = 1.1\times10^{-24}\langle n_{\mathrm{DT}}^2T_i[\mathrm{keV}]^2\rangle_{\mathrm{av}}\Theta\frac{Q_n}{4}V = 0.16f_{\mathrm{DT}}^2\langle n_eT\rangle_{20,\mathrm{keV}}^2Aa^3\kappa\Theta\,[\mathrm{MW}], \quad (6)$$

where $f_{\mathrm{DT}} = n_{\mathrm{DT}}/n_e$. $\langle p^2\rangle_{\mathrm{av}} = 4/3\langle p\rangle_{\mathrm{av}}^2$ is used assuming the parabolic pressure profile. When the magnetic surfaces are elliptic and the temperature profile is parabolic $T(\rho) = 2\langle T\rangle(1-\rho^2)$ (ρ being normalized radial coordinate) and the density profile is constant, Θ is reduced to

$$\Theta(\langle T\rangle) \approx \frac{\int_0^1\langle\sigma v\rangle_{\mathrm{DT}}2\rho\mathrm{d}\rho\,[\mathrm{m^3/s}]}{1.1\times10^{-24}(4/3)\langle T[\mathrm{keV}]\rangle^2[\mathrm{m^3/s}]} \quad (7)$$

and $\Theta(\langle T\rangle)$ is shown in Fig.2 as a function of $\langle T\rangle$. We used the following fitting equation of $\langle\sigma v\rangle_{\mathrm{DT}}$ as the function of T[keV] [3] to estimate $\Theta(\langle T\rangle)$:

$$\langle\sigma v\rangle_{\mathrm{DT}} = \frac{3.7\times10^{-18}}{h(T)}\times T^{-2/3}\exp(-20T^{-1/3})\,[\mathrm{m^3/s}],$$

$$h(T) = \frac{T}{37}+\frac{5.45}{3+T[1+(T/37.5)^{2.8}]}.$$

The accuracy of approximation by Eq.(7) for $\Theta(\langle T\rangle)$ is examined by the calculation of the magnetic flux functions $\Psi(r,z)$ satisfying Grad-Shafranov equation with use of SYSTEQ code [4] for ITER (case I) configuration of Table 1. The following profile of temperature

$$T(r,z) = 2\gamma\langle T\rangle\frac{\Psi(r,z)-\Psi_s}{\Psi_c-\Psi_s}$$

is used, where Ψ_s, Ψ_c are the values of the flux function at separatrix and at the minor axis respectively. γ is normalized factor, that is, $\gamma = 0.5(\Psi_c-\Psi_s)/\langle\Psi(r,z)-\Psi_s\rangle_{\mathrm{av}}$. A uniform density is assumed. The numerically calculated values of $\Theta(\langle T\rangle)$ are compared with analytically obtained results. Difference of both results is

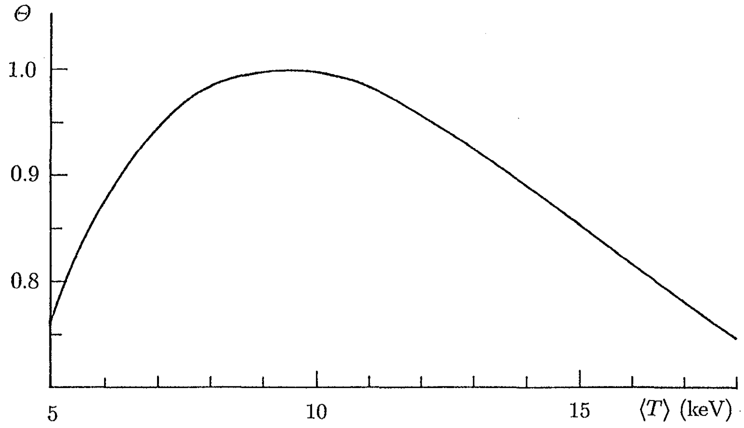

Fig. 2 $\Theta(\langle T\rangle)$ is shown as a function of $\langle T\rangle$.

Table 1. Design parameters of four cases A, B, C, D in the case of [$B_{t,Mx}$ = 12.2 T], $\langle n_e T\rangle = 13\times10^{20}$ m^{-3} keV, $n_e = 1\times10^{20}$ m^{-3}, $\langle T\rangle$ = 13 keV, Θ = 0.928, Δ =1.55 m. The case I is the design parameters of ITER. Δ: distance between plasma and TF coil conductor in m, τ_E in sec, $H_H = \tau_E/\tau_E^{\mathrm{ELMy\ H}}$, β in %, g_{Tr}: Troyon factor, N_G: Greenwald normalized density, B_t: toroidal field at plasma center in T, P_n: total neutron output power in GW, f_n: neutron flux at plasma surface in MW/m^2, d_{TF}: thickness of TF coil conductor. Cases E, F, G are design parameters when Δ is changed to Δ = 1.3 m.

	H_L	ξ	Δ	κ	AI_p	A	I_p	R	a	τ_E	H_H	β	g_{Tr}	N_G
I	2.75	1.35	1.55	1.6	60.9	2.9	21.0	8.14	2.81	6.1	1.08	2.97	2.25	1.18
A	[3.16]	1.35	1.55	1.6	52.2	3.09	16.9	7.49	2.42	6.1	1.22	2.91	2.39	1.09
B	[3.77]	1.35	1.55	1.6	42.8	3.38	12.7	6.78	2.01	6.1	1.35	2.82	2.60	1.0
C	2.75	[1.0]	1.55	1.6	61.0	2.71	22.5	7.94	2.93	6.1	1.13	3.36	2.33	1.2
D	2.75	1.35	1.55	[1.8]	61.5	3.37	18.26	7.31	2.17	6.1	0.97	2.65	1.89	0.81
E	2.75	1.35	[1.3]	1.6	60.2	2.89	20.85	7.68	2.66	6.1	1.07	2.73	2.06	1.06
F	[3.16]	1.35	[1.3]	1.6	51.6	3.08	16.75	7.05	2.29	6.1	1.35	2.66	2.18	0.98
G	[3.77]	1.35	[1.3]	1.6	42.2	3.38	12.49	6.36	1.88	6.1	1.39	2.56	2.35	0.89

	H_L	ξ	Δ	κ	B_t	P_n	f_n	$R+a$	d_{TF}
I	2.75	1.35	1.55	1.6	5.67	1.31	1.09	10.95	0.98
A	[3.16]	1.35	1.55	1.6	5.73	0.9	0.94	9.91	0.99
B	[3.77]	1.35	1.55	1.6	5.80	0.56	0.78	8.79	1.01
C	2.75	[1.0]	1.55	1.6	5.33	1.4	1.14	10.87	1.00
D	2.75	1.35	1.55	[1.8]	5.99	0.79	0.87	9.48	0.99
E	2.75	1.35	[1.3]	1.6	5.90	1.11	1.04	10.39	0.99
F	[3.16]	1.35	[1.3]	1.6	5.99	0.76	0.89	9.34	1.01
G	[3.77]	1.35	[1.3]	1.6	6.10	0.46	0.73	8.24	1.03

within 4% in the range of $\langle T\rangle$ = 10 keV ~ 15 keV and is within 8% in the range of $\langle T\rangle$ = 8 keV ~ 20 keV.

The total α particle fusion output power $P_\alpha = P_n/4$ is reduced to $P_\alpha = 0.0405\, f_{DT}^2 \langle n_e T\rangle_{20,\mathrm{keV}}^2 A a^3 \kappa \Theta$ [MW]. Then the total plasma heating power $P_{\alpha\ \mathrm{heat}}$ corrected for radiation loss is

$$P_{\alpha\ \mathrm{heat}} = f_h (1 - f_R) P_\alpha\,, \qquad (8)$$

where f_h is the efficiency of α particle heating and f_R is the fraction of radiation loss. The thermal energy of plasma W is

$$W = \int (1 + f_{DT} + \sum f_I) n_e T \mathrm{d} V$$
$$= 0.471 (1 + f_{DT} + \sum f_I)$$
$$\times \langle n_e T\rangle_{20,\mathrm{keV}} A a^3 \kappa [\mathrm{MJ}]\,. \qquad (9)$$

The notation f_I is the ratio of impurity density to electron density. Then the energy confinement time necessary for ignition is

$$\tau_E = \frac{W}{P_{\alpha\,\mathrm{heat}}} = \frac{11.6(1+f_{DT}+\sum f_I)}{f_h(1-f_R)f_{DT}^2\,\Theta}\,\frac{1}{\langle n_e T\rangle_{20,\mathrm{keV}}}. \tag{10}$$

The expression (10) does not include the scale parameters of device as is expected. Then H_L factor, the ratio of τ_E to $\tau_E^{\mathrm{ITER\ P}}$ of ITER-P L mode scaling is given by [5,6]

$$H_L = \frac{30.8(1+f_{DT}+\sum f_I)}{f_h^{1/2}(1-f_R)^{1/2}f_{DT}\,\Theta^{1/2}} \times \frac{A^{0.15}}{n_{20}^{0.1}B_t^{0.2}[\mathrm{T}]}\,\frac{1}{(AI_p)^{0.85}}, \tag{11}$$

where $\tau_E^{\mathrm{ITER\ P}}$ is [7]

$$\tau_E^{\mathrm{ITER\ P}} = 0.048 I_p^{0.85} R^{1.2} a^{0.3} n_{20}^{0.1} B_t^{0.2} \times (A_i \kappa / P_{\mathrm{heat}})^{1/2}. \tag{12}$$

The units are of sec, MA, m, 10^{20} m^{-3}, T, MW.

It is noteworthy that H_L factor necessary for ignition depends mainly on (AI_p) and depends on other parameters very weakly. H_H is defined by the ratio of τ_E to $\tau_E^{\mathrm{ELMy\ H}}$ of ITER-H scaling [8] of ELMy H mode as follows,

$$H_H \equiv \frac{\tau_E}{\tau_E^{\mathrm{ELMy\ H}}},$$

$$\tau_E^{\mathrm{ELMy\ H}} \equiv 0.85 \times 0.053 I_p^{1.06} B_t^{0.32} P_{\mathrm{heat}}^{-0.67} \times A_i^{0.41} R^{1.79} n_{20}^{0.17} (R/a)^{-0.11} \kappa^{0.66}. \tag{13}$$

H_H is reduced to

$$H_H = \frac{29.8(1+f_{DT}+\sum f_I)\kappa^{0.01}}{f_h^{0.33}(1-f_R)^{0.33}f_{DT}^{0.66}\,\Theta^{0.33}A_i^{0.41}} \times \frac{\langle n_{20}T\rangle^{0.34}R^{0.22}}{n_{20}^{0.17}B_t^{0.32}(\mathrm{T})A^{0.17}}\,\frac{1}{(AI_p)^{1.06}}, \tag{14}$$

where A_i is atomic number of plasma ion.

3. Effects of H_L factor, ratio ξ, κ ratio and distance $\varDelta$ on Device Parameters

Design parameters of ITER are used as a reference [9]: that is, $q_{\mathrm{eff}} = 3.27$, $\kappa = 1.6$, $\delta = 0.25$, $l_i = 1.0$, ($q_I = 2.32$), $\varDelta = 1.55$ m, $d_{TF} = 0.98$ m, $d_s = 0.13$ m, $d_{OH} = 0.8$ m, $f_{DT} = 0.71$, $f_{He} = 0.09$, $f_{Be} = 0.018$, $f_{Ar} = 0.001$, $f_h = 0.9$, $f_R = 0.37$, $B_{t,Mx} = 12.2$ T, $B_{OH,Mx} = 12.5$ T, $\langle n_e T\rangle = 13 \times 10^{20}$ m^{-3} keV, $\langle n_e\rangle = 1 \times 10^{20}$ m^{-3}, ($\langle T\rangle = 13$ keV), $\Theta(13\ \mathrm{keV}) = 0.928$.

Since H_L depends mainly on AI_p, it is interesting to use (AI_p)-A design space. AI_p is reduced to the ratio of the major radius R to poloidal Lamor radius $\rho_{p,DT}$ of DT ions ($A = 2.5$) with the thermal energy of 10 keV as follows,

$$AI_p = \frac{K}{12.65}\left(\frac{R}{\rho_{p,DT}}\right), \tag{15}$$

where $\rho_{p,DT} = (m_{DT}T)^{1/2}/eB_p$, $B_p[\mathrm{T}] = I_p[\mathrm{MA}]/(5aK)$. Contour plots of H_L = const. and ξ = const. in AI_p-A design space are shown in Fig.3(a) and contour plots of R = const. and a = const. are shown in Fig.3(b). Figure 3(a) shows that H_L depends mainly on AI_p and depends on A weakly. Figure 3(b) shows that $(R + a)$ also depends mainly on AI_p and depends on A weakly. d_{TF} and d_{OH} were determined so that the effective current densities of TF coil and OH coil are kept the same as those of ITER design, that is

$$\frac{(R-a-\varDelta)B_{t,Mx}}{\pi[(R-a-\varDelta)^2-(R-a-\varDelta-d_{TF})^2]} = \left(\frac{(R-a-\varDelta)B_{t,Mx}}{\pi[(R-a-\varDelta)^2-(R-a-\varDelta-d_{TF})^2]}\right)_{\mathrm{ITER}},$$

$$\frac{B_{OH,Mx}}{d_{OH}} = \left(\frac{B_{OH,Mx}}{d_{OH}}\right)_{\mathrm{ITER}}. \tag{16}$$

When H_L factor and ξ ratio are specified instead of I_p, A, the other design parameters are determined. Design parameters of three cases (A, B, C) taking ITER design (case I) as a reference are listed in Table 1. Increase of required H_L factor from 2.75 (ITER) to 3.16 (15% increase in case A) and 3.77 (case B), while keeping $\xi = 1.35$, is effective to reduce the scale of reactors. Figure 4 shows the dependences of R, a and I_p on H_L, while keeping $\xi = 1.35$.

Figure 5 shows the dependences of R, a, $(R + a)$ and I_p on ξ, while keeping $H_L = 2.75$. As is seen in Fig.5 and case C in Table 1, the reduction of ξ, while

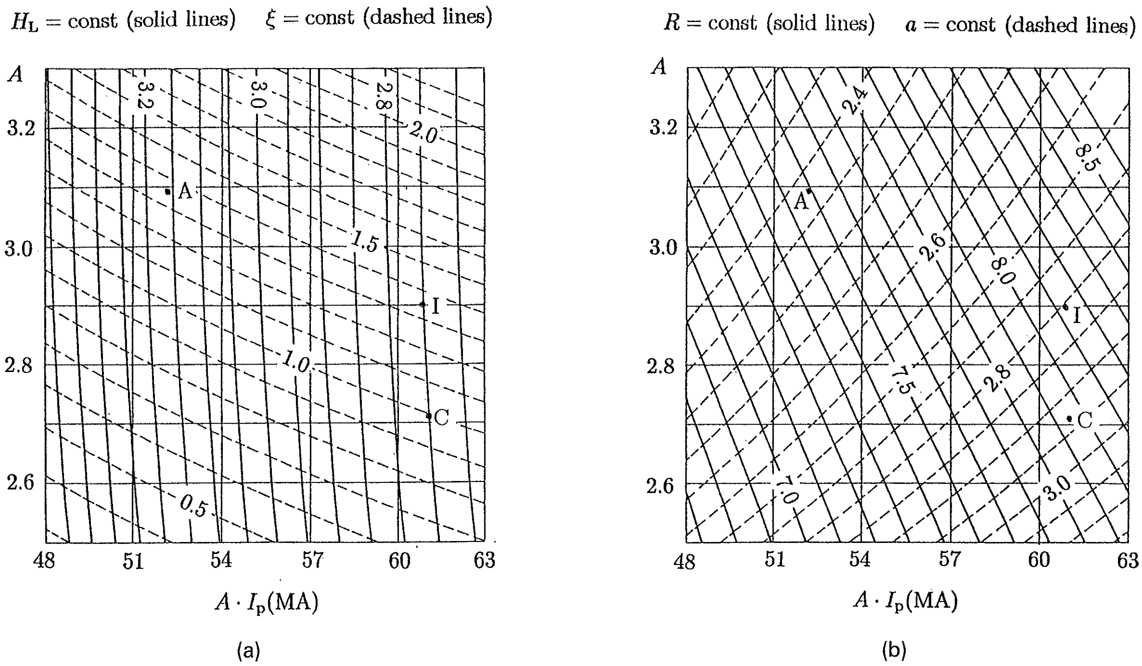

Fig. 3 (a) Contour plots of H_L = const. and ξ = const. in AI_p-A design space. (b) Contour plots of R = const. and a = const. in AI_p-A design space. $B_{t,Mx}$ = 12.2 T, $\langle n_e T\rangle$ = 13 × 10^{20} m^{-3} keV, $\langle n_e\rangle$ = 1.0 × 10^{20} m^{-3} ($\langle T\rangle$ = 13 keV), Δ = 1.55 m. Points I, A and B correspond to cases I, A and B of Table 1 respectively.

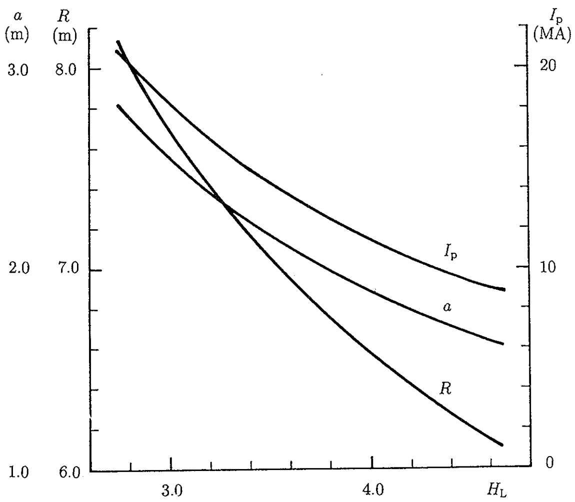

Fig. 4 Dependences of major radius R, plasma radius a and plasma current I_p on H_L factor, while keeping normalized OH coil flux ξ = 1.35.

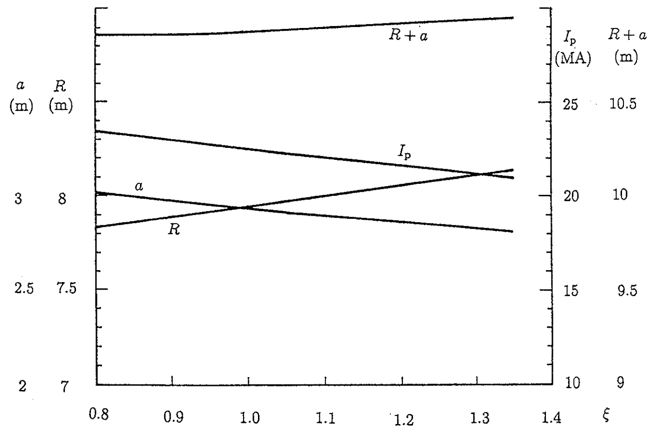

Fig. 5 Dependences of major radius R, plasma radius a, their sum (R + a) and plasma current I_p on normalized OH coil flux ξ, while keeping H_L = 2.75.

keeping H_L factor constant, is less effective to reduce the scale of reactors due to the parameter dependence of H_L factor. The decrease of ξ tends to the decrease of A. Since H_L depends mainly on (AI_p), keeping H_L constant tends to increase of I_p and the both effects cancels with each other.

Figure 6 shows the dependences of R, a, ($R + a$) and I_p on elongation ratio κ, while keeping $H_L = 2.75$ and $\xi = 1.35$. The increase of κ is effective to reduce the scale of reactors, although feedback controle system

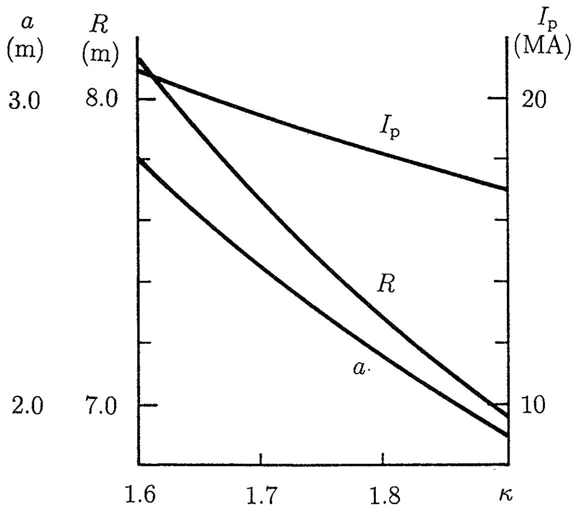

Fig. 6 Dependences of major radius R, plasma radius a and plasma current I_p on κ, while keeping H_L = 2.75 and ξ = 1.35.

Table 2 Q versus specified H_L^{spec} in the case where the attained H_L factor is 2.75 (f_h = 0.9, f_R = 0.37).

H_L^{spec}	2.75	3.16	3.5	3.77	4.57
Q	∞	27	14.2	10.0	5.0

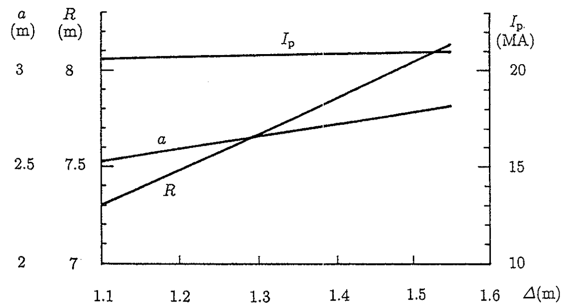

Fig. 7 Dependences of major radius R, plasma radius a and plasma current I_p on Δ: the distance between TF coil conductor and plasma surface, while keeping H_L = 2.75 and ξ = 1.35.

of vertical plasma position becomes more demanding.

Plasmas with parameters determined by a specified H_L factor, say H_L^{spec}, as was described in ch.2, need additional heating to be sustained, if the attained H_L factor is less than H_L^{spec}. The necessary additional heating power $P_{add\ heat}$ to sustain the plasma is

$$P_{add\ heat} = [(H_L^{spec}/H_L)^2 - 1]\, P_{\alpha\ heat}\,,$$

since

$$P_{\alpha\ heat} = \frac{WP_{\alpha\ heat}^{0.5}}{H_L^{spec} \times (\tau_E^{ITER\ P}\ P_{\alpha\ heat}^{0.5})}\,,$$

$$P_{\alpha\ heat} + P_{add\ heat} = \frac{W(P_{\alpha\ heat}^{0.5} + P_{add\ heat})^{0.5}}{H_L \times [\tau_E^{ITER\ P}(P_{\alpha\ heat} + P_{add\ heat})^{0.5}]}\,.$$

Therefore the ratio $Q \equiv (P_n + P_\alpha)/P_{add\ heat}$ is given by

$$Q = \frac{5}{[(H_L^{spec}/H_L)^2 - 1] f_h (1 - f_R)} = \frac{8.82}{[(H_L^{spec}/H_L)^2 - 1]}\,. \qquad (17)$$

The values of Q versus specified H_L^{spec} are given in Table 2 in the case where the attained H_L factor is 2.75.

As is seen in Eq.(11), the machine parameters are determined by the factor of

$$\frac{H_L f_h^{1/2}(1-f_R)^{1/2} f_{DT}}{(1 + f_{DT} + f_{He} + \sum f_i)}$$
$$= \frac{H_L f_h^{1/2}(1-f_R)^{1/2}(1 - 2f_{He} - \sum Z_i f_i)}{[2 - f_{He} - \sum(Z_i - 1) f_i]}$$
$$\approx 0.5 H_L f_h^{1/2}(1-f_R)^{1/2}$$
$$\times [1 - 1.5 f_{He} - 0.5\sum(Z_i + 1) f_i]\,,$$

since $f_{DT} + 2f_{He} + \sum Z_i f_i = 1$ When the fraction f_{He} and the fraction f_R of radiation become large, the scale of reactor becomes large.

Next we examine the sensitivity of the scale on the distance Δ between plasma boundary and TF coil conductor. Figure 7 shows the dependences of R, a and I_p on Δ, while keeping H_L = 2.75 and ξ = 1.35. Decrease of Δ is effective to reduce tha scale of reactors. The values of Δ of reactors designed for power plant ARIES-1 [10], ARIES-RS [10] and SSTR [11] are 1.497 m, 1.42 m and 1.43 m respectively. The values of Δ of experimental reactors FER [12], ITER-CDA [13], NET [14] and ITER-EDA [9] are 1.05–1.16 m, 1.25 m, 1.48 m and 1.55 m respectively. Design parameters of three more cases (E,F,G) with Δ = 1.3 m are also listed in Table 1.

Table 3 Design parameters of two cases E, F in the case of [$B_{t,Mx}$ = 15 T], κ = 1.6, $\langle n_e T \rangle$ = 16.0 × 10^{20} m^{-3} keV, n_e = 1.23 × 10^{20} m^{-3}, $\langle T \rangle$ = 13 keV. τ_E in s, $H_H = \tau_E/\tau_E^{ELMy\ H}$, β in %, g_{Tr}: Troyon factor, N_G: Greenwald normalized density, B_t: toroidal field at plasma center in T, P_n: total neutron output power in GW, f_n: neutron flux at plasma surface in MW/m^{-2}, d_{TF}: thichness of TF coil conductor in m.

	H_L	ξ	Δ	AI_p	A	I_p	R	a	$R+a$	τ_E	H_H	β	g_{Tr}	N_G
H	2.75	1.35	[1.3]	56.6	3.52	16.6	6.91	1.96	8.87	4.97	0.97	1.87	1.80	0.93
J	[3.16]	1.35	[1.3]	48.6	3.82	12.7	6.44	1.69	8.13	4.97	1.08	1.81	1.93	0.86
K	[3.77]	1.35	[1.3]	40.1	4.27	9.39	5.82	1.39	7.21	4.97	1.24	1.39	2.11	0.80

	H_L	ξ	Δ	B_t	P_n	f_n	d_{TF}
H	2.75	1.35	[1.3]	7.92	0.83	1.16	1.27
J	[3.16]	1.35	[1.3]	8.05	0.57	0.99	1.29
K	[3.77]	1.35	[1.3]	8.20	0.36	0.82	1.33

4. Effect of $B_{t,Mx}$ on Design Parameters

It is suggested that there is possibility to increase the maximum field of TF and OH coils by use of superconductor of $(NbTi)_3Sn$ made by tube method [11] for example. Therefore it is meaningful to examine the effect of $B_{t,Mx}$ of TF coil and $B_{OH,Mx}$ of OH coil on design parameters.

Figure 8 shows the dependences of R, a and I_p on $B_{t,Mx}$, while keeping $H_L = 3.16$, $\xi = 1.35$, $\Delta = 1.3$ m and the average temperature $\langle T \rangle = 13$ keV. However the average electron density is increased as $\langle n_e \rangle = 1.0(B_{t,Mx}/12.2) \times 10^{20}$ m^{-3}.

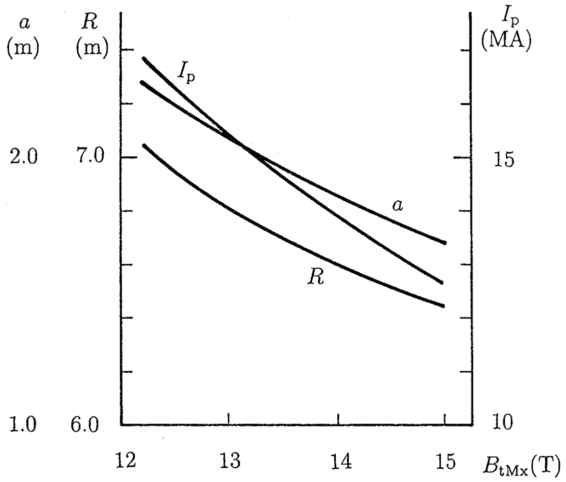

Fig. 8 Dependences of major radius R, plasma radius a and plasma current I_p on the maximun field of TF coil $B_{t,Mx}$, while keeping H_L = 3.16, ξ = 1.35, κ = 1.6, Δ = 1.3 m and average temperature $\langle T \rangle$ = 13 keV. However average electron density is increased as $\langle n_e \rangle$ = 1.0($B_{t,Mx}$/12.2) × 10^{20} m^{-3}.

Design parameters of three cases (H, J, K) with B_M = 15 T are listed in Table 3. Increase of $B_{t,Mx}$ is effective to reduce the scale of reactors.

Conclusion

The specification of H_L factor and the ratio $\xi = \Delta\Phi/L_p I_p$ determines the design parameters of reactors when q_{eff}, κ, δ, Δ, $B_{t,Mx}$ etc. are given. The simple expression of H_L factor necessary for ignition is described. The increase of H_L factor, κ ratio and the maximum field $B_{t,Mx}$ as well as the the decrease of Δ are effective to make reactor more compact, while the decrease of the ratio ξ, while keeping H_L constant, is less effective due to the dependence of H_L factor necessary for ignition on A and I_p.

References

[1] ITER Team, *Plasma Phys. Control. Nucl. Fusion Res.* (Conf. Proc. Washington D.C.) IAEA, Vienna, **3**, 413 (1991).

[2] T.N. Todd, *Proc. 14th Int. Symp. on Heating in Toroidal Plasma*, Rome, **1**, 21 (1984); Y. Kamada *et al.*, Nucl. Fusion **34**, 1605 (1994).

[3] T. Takizuka and M. Yamagiwa, JAERI-M 87-066, Japan Atomic Energy Res. Ins. (1987).

[4] T. Nishino, Y. Shinya, M. Hasegawa and M. Sugihara, JAERI-memo 03-141 May 1991 [in Japanese].

[5] K. Miyamoto, *Fundamentals of Plasma Physics and Controlled Fusion* (Iwanami Book Service Center, Tokyo, 1997).

[6] K. Miyamoto, *Proc. Int. Symp. on Plasma Dynamics in Complex Electromagnetic Fields,* Dec. 1997, Institute of Advanced Energy, Kyoto Univ. Uji, Japan.

[7] P.N. Yushimanov, T. Takizuka, K.S. Riedel, D.J.W. Kardaun, J.G. Cordy, S.M. Kaye and D.E. Post, Nucl. Fusion **30**, 1999 (1990).

[8] ITER H Mode Database Working Group, Nucl. Fusion **34**, 131 (1994).

[9] ITER Team, *Fusion Energy* (Conf. Proc. Montreal) IAEA, Vienna, **2**, 737 (1996).

[10] F. Najmabadi and ARIES Team, *Fusion Energy* (Conf. Proc. Montreal) IAEA, Vienna, **3**, 383 (1996).

[11] Y. Seki, M. Kikuchi, T. Ando, Y. Ohara, S. Nishio *et al.*, *Plasma Phys. Control. Nucl. Fusion Res.* (Conf. Proc. Washington D.C.) IAEA, Vienna, **3**, 473 (1991).

[12] S. Matsuda, H. Tsuji, Y. Akutsu, N. Fujiwara, M. Hasegawa *et al.*, *Plasma Phys. Control. Nucl. Fusion Res.* (Conf. Proc. Washington D.C.) IAEA, Vienna, **3**, 547 (1991).

[13] ITER Team, Plasma Phys. Control. Nucl. Fusion Res. IAEA, Vienna, **3**, 217, 351 (1991).

[14] NET Team, *Plasma Phys. Control. Nucl. Fusion Res.* (Conf. Proc. Wurzburg) IAEA, Vienna, **3**, 229 (1992).