Plasma Performance Required for a Tokamak Reactor to Generate Net Electric Power

HIWATARI Ryouji, OKANO Kunihiko, ASAOKA Yoshiyuki,
YOSHIDA Tomoaki and TOMABECHI Ken
Central Research Institute of Electric Power Industry, Tokyo 201-8511, Japan

(Received 13 August 2002 / Accepted 2 September 2002)

Using a system analysis code, plasma parameter ranges required for a Tokamak reactor to generate net electric power (P_e^{net}) were investigated. The investigation revealed that in order for a Tokamak to produce a net electric power of 0 MW under foreseeable engineering conditions, it is required to achieve simultaneously the following ranges of parameters, i.e., normalized beta value $1.2 \leq \beta_N \leq 2.7$, confinement improvement factor for H-mode scaling $0.8 \leq HH$, and the ratio of Greenwald density limit $0.4 \leq fn_{GW} \leq 1.1$, for a major radius $R \leq 8.5$ m. It also revealed that a reactor to produce a net electric power of 1,000 MW requires the simultaneous achievement of $\beta_N \geq 3.0$, $HH \geq 0.9$ and $fn_{GW} \geq 0.9$.

Keywords: tokamak fusion reactor, 0D system analysis, net electric power, fast track

Although the following three stages of development are usually assumed for the realization of fusion power plants, (1) demonstration of a fusion reactor operation, (2) demonstration of net electric power generation in a power plant, and (3) demonstration of economic and safe performance, devices are not necessarily required to be built in each of the above stages [1]. For example, the fast track concept recently developed for aiming at the early realization of the fusion power plant suggested a roadmap somewhat different from the usual, i.e., the second and third stages mentioned above, which is usually demo and proto-type reactors stages, would be combined into a single step [2]. If one aims at the early realization of a fusion power plant, careful investigation of plasma parameters for demonstrating net electric power generation, based on the foreseeable physics and engineering advancement through the ITER program, should be worthwhile. Since past studies of fusion power plants such as CREST [3] and A-SSTR2 [4] aimed to make clear the development goal of physics and engineering parameters, these studies inevitably adopted highly advanced plasma parameters. In contrast, a demo plant for the early realization of net electric power generation must be designed with reasonably conservative plasma and engineering parameters.

author's e-mail: hiwatari@criepi.denken.or.jp

Table 1 The parameter ranges of 0D system analysis.

Major radius R (m)	6.0~8.5
Aspect ratio A	3.0~4.0
Plasma elongation κ_{95}	1.5~2.0
Plasma triangularity δ_{95}	0.35~0.45
Plasma temperature $\langle T \rangle$ (keV)	12~20
Plasma surface safety factor q_a	3.0~6.0
Max. magnetic field B_{tmax} (T)	16.0
Thermal efficiency η_e (%)	30
NBI system efficiency η_{NBI} (%)	50

In the present study, extensive 0D analyses were carried out using a system code based on ref. [5]. To show the possible plasma parameter ranges required for net electric power (P_e^{net}) generation under the conditions listed in Table 1, the normalized beta value (β_N), the confinement improvement factor for H-mode scaling (HH), and the ratio of Greenwald density limit (fn_{GW}) were carefully investigated. In these analyses, aiming at the early realization of a fusion power plant, relatively conservative values were assumed for engineering parameters, B_{tmax}, η_e, and η_{NBI}. In addition, plasma current ramp up was provided with the magnetic flux and a torus space for accommodating blanket and shield was sufficiently maintained 1.4 m. Current drive power

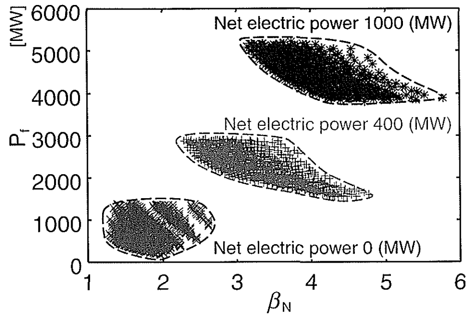

Fig. 1 β_N and P_f for various P_e^{net}.

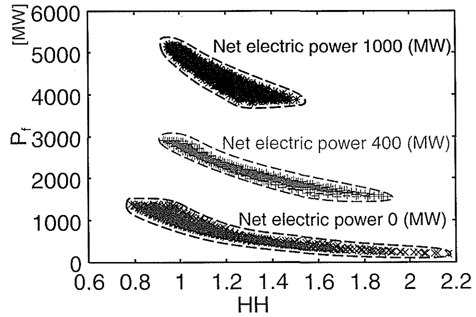

Fig. 2 HH and P_f for various P_e^{net}.

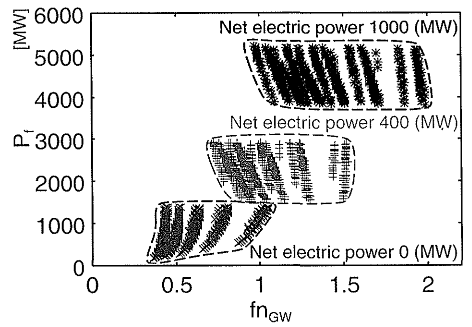

Fig. 3 fn_{GW} and P_f for various P_e^{net}.

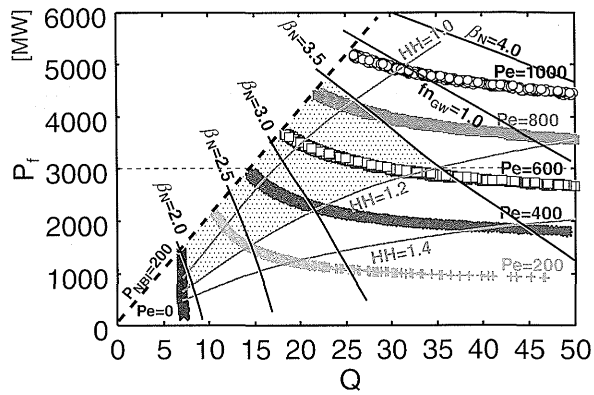

Fig. 4 Q and P_f for various P_e^{net} with β_N, HH, and fn_{GW} for $R = 7.5$ m, $\eta_e = 30\%$ and $P_{NBI} \leq 200$ MW.

P_{NBI} was restricted to 200 MW because of the necessity of a small circulating power in a power plant. It should be noted that all these conditions would be attainable in the near future if the ITER program is implemented as planned. With these conditions and parameters, extensive analyses were conducted for defining the relevant plasma as well as the engineering parameters required for Tokamak reactors, and produced a database covering approximately 100,000 operation points for reactors.

The relationship between plasma performance parameters (β_N, HH, fn_{GW}) and fusion power P_f for P_e^{net} of 0, 400 and 1,000 MW are respectively shown in Figs. 1–3. These results are for $\kappa_{95} = 1.9$, $\delta_{95} = 0.45$, and an effective charge number $Z_{eff} = 1.7$. The figures show that a net electric break-even ($P_e^{net} = 0$) reactor requires the simultaneous achievement of $1.2 \leq \beta_N \leq 2.7$, $0.8 \leq HH$, and $0.3 \leq fn_{GW} \leq 1.1$. It should be noted that the net electric break-even condition is attainable in a relatively moderate case of $\beta_N = 1.8$, $HH = 1.0$, and $fn_{GW} = 0.85$, which correspond to ITER reference operation parameters [6]. Since the improvement of such engineering parameters as η_e, η_{NBI}, and B_{tmax} enhances plasma performance, improvement of those parameters is also important for early demonstration of net electric power.

In Fig. 4, relationship of energy multiplication factor Q and fusion power P_f is shown for the condition of β_N, HH, and fn_{GW} for a major radius $R = 7.5$ m. The plasma parameter lines for β_N, HH, and fn_{GW} shown in Fig. 4 delineate the boundaries of plasma performance. For example, $fn_{GW} > 1.0$ is always needed for the domain above the line of $fn_{GW} = 1.0$, whereas for the domain below the line it is not. Similarly, the other plasma parameter lines shown in Fig. 4 delineate respective boundaries. The shaded region in Fig. 4 is for $\beta_N \leq 3.5$, $HH \leq 1.2$, and $fn_{GW} \leq 1.0$. If fusion power P_f is to be limited to 3,000 MW, a combination of $\beta_N \leq 3.5$, $HH \leq 1.2$ and $fn_{GW} \leq 1.0$ allows the possibility to obtain $P_f \sim 3{,}000$ MW and $P_e^{net} \sim 600$ MW with $R \sim 7.5$ m and $\eta_e = 30\%$. However, if thermal efficiency $\eta_e \sim 40\%$ is attainable, these parameters will roughly allow the generation of $P_e^{net} \sim 1{,}000$ MW.

Based on these results, taking construction cost into consideration, a reactor design for early demonstration of net electric power is underway.

[1] Y. Asaoka *et al.*, CRIEPI report No. T97077, (Central Research Institute of Electric Power Industry, Tokyo, 1997) [in Japanese]

[2] H. Bolt, "Fact Track Concept in the European Fusion Programe", *Report of International Symposium for ITER*, (JAERI, Tokyo, March 2002).

[3] K. Okano *et al.*, Nucl. Fusion **40**, 635 (2000).

[4] S. Nishio *et al.*, *18th IAEA Fusion Energy Conference*, IAEA-CN-77/FTP2/14 (2000).

[5] T. Yoshida *et al.*, CRIEPI report No. T94001 (Central Research Institute of Electric Power Industry, Tokyo, 1994) [in Japanese]

[6] Technical Basis for the ITER Final Design, ITER EDA Documentation Series 22 IAEA, Vienna (2001).