FAST: A FUSION ENERGY SYSTEMS INTEGRATION TEST FACILITY

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FAST (Fusion by Advanced Superconducting Tokamak) is a project being proposed as a facility for R&D, testing, and to demonstrate integration of systems necessary for a Deuterium Tritium (DT) fusion energy reactor [1]. The required specifications for FAST are: DT fusion power of $50 - 100$ MW, neutron wall loading of $0.3 - 1\ \mathrm{MW/m^2}$, discharge duration of about 1000 sec, full-power operation time of about 1000 hrs (same order as ITER). These are identified as required and also sufficient for the near-term R&D of the tritium breeding and power extraction blanket to verify the integrity of the fusion system. Since we would like to demonstrate electricity generation technology using the thermal energy extracted from blankets in the 2030's, minimization of the cost is essential, because it determines the necessary funding and the construction period. Integrated fusion fuel cycle and safety features as an energy plant that will fill the technical gap toward net positive energy generation plant is another mission, while the system integration has the highest priority. A quasi-zero-dimensional parameter survey has been carried out to find the parameter region necessary to satisfy the above specifications with the minimum device cost. It was found that a low aspect ratio (A ~ 2.2), compact (major radius ~ 2.0 m) tokamak with high temperature superconductor (HTS) magnets and neutral beam injection (NBI) power of about 50 MW (with the energy of 500 keV) offers a possible design window.

The above required specifications lead to unique features of the device. Since a long full-power operation time over years, very large energy gain, and a high tritium breeding ratio over unity are not mandatory, we can find a reasonably compact and economical design. Figure 1 shows representative profiles obtained by the method described below. The parameters to specify the plasma and the device are: line-averaged density normalized by the Greenwald density f_{GW}, major radius R, elongation κ, aspect ratio A, D-NBI injection power P_{NBI}. We adopt the hybrid scaling proposed in [2], in which an interpolation between the high- and the low-aspect ratio scalings is used, and its enhancement factor is set to $H_{\mathrm{hy}} = 0.9$. Note that $H_{\mathrm{IPB98}} \sim 1.45$ is necessary to reproduce this plasma when we adopt ITER IPB98(y,2) confinement time scaling. The temperature and density profile shapes are fixed (Fig. 1(a)) taken from a result obtained by the METIS transport code [3]. The maximum toroidal field $B_{\max}$ is fixed to be 13.4 T at the position of the inboard toroidal field coil surface located at the major radius $R - a - 0.6$ m. Here, a is the minor radius, and 0.6 m is the sum of thicknesses of shield/blanket, vacuum vessel, SOL, identified in the radial build. The thin radiation shield thickness is acceptable when we consider the limited full-power operation time [4]. Adoption of this thickness and REBCO HTS magnets (with higher critical temperature) enables a compact and low aspect ratio tokamak device. The plasma current is the sum of the bootstrap current I_{BS} and the NBI driven current I_{NBI}, which are calculated from the formulas in textbooks and the local plasma parameters. The driven current density and power deposition profiles are

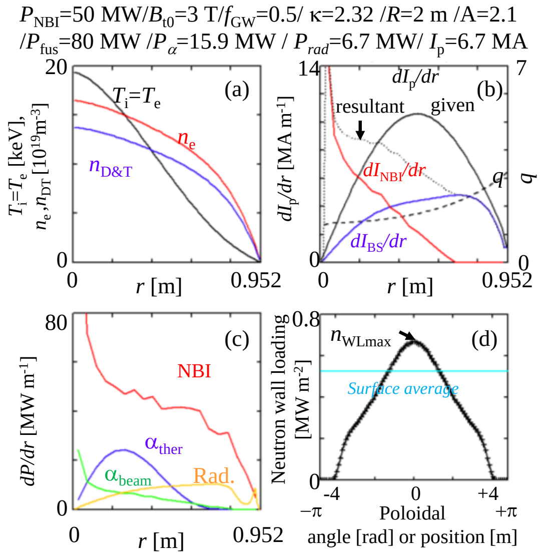

Fig.1 Temperature and density profiles (a), current profiles (b), and power deposition profiles (c). The horizontal axis represents the minor radius at midplane. (d) shows the neutron wall loading as a function of the poloidal angle on a sphere.

calculated (Fig. 1(b) and (c)). Here, α particles generated by thermal DT reactions and beam-thermal reactions are considered. The current density profile shape is fixed, and the bootstrap current is calculated from the poloidal field calculated from the profile. We estimated the prompt orbit loss and shine through, and these are subtracted from the total heating power and the current, although these are negligible in global balances for typical cases. We assume additional 10 % unidentified losses. Similarly, a 10 % loss for alpha-particle heating power P_α is assumed. Thus, the plasma heating power becomes $0.9 \times (P_{\mathrm{NBI}} + P_\alpha) - P_{rad}$, where P_{rad} is the radiation power (at $Z_{\mathrm{eff}} = 2$, with He of 5.7 % and Ar of 0.29%). The device cost is calculated by PEC [6], but it is updated using the data used in [7, 8]. The component costs (coils: 9 M$m^{-3}@70 MAm^{-2}, coil-support, shield, blanket, vessel, base, divertor) and NBI: 7 $W^{-1}, coil-power supply, vacuum pump system etc are included, but BOP (Balance of Plant) is not included. Note that the cost enhancement by a first of kind production is not considered. Neutron wall loading distribution on a sphere (with radius R+a+0.1 m (Fig. 1(d)) is calculated from the local neutron emission. The maximum n_{WLmax} is located on the (outboard) equatorial plane, which is about 1.3 times higher than the surface average (Fig. 1(d)).

Parameter survey has been conducted by randomly chosen parameters in the ranges: $0.4 < f_{\mathrm{GW}} < 1$, $1.8 < R < 2.5$ m, $\kappa_{max} - 0.3 < \kappa < \kappa_{max}$, $1.8 < A < 2.7$ for $P_{\mathrm{NBI}} = 40, \dots, 60$ MW , where $\kappa_{max} = 0.9 \times 1.9 \times (1 + 1/A^{1.4})$ represents the stability limit [5] with 10% margin. Figure 2 shows n_{WLmax} as functions of R, A, f_{GW}, $\Delta\kappa = \kappa_{max} - \kappa$ when $P_{\mathrm{NBI}} = 50$ MW. After removing the cases $\beta_n > \beta_{nmax} = 0.9 \times (3.12 + 3.5/A^{1.7})$ [5], about 34 k cases are plotted in this figure. The green and red symbols show constant costs, which is a strong function of R. The A-dependence shows a maximum at ~2.2, and R-dependence shows a slow increase. These dependences are the results of interpolated energy confinement time scaling and fixed B_{max}. The decreasing dependence on f_{GW} is due to the feature of I_{NBI} and the slow $\Delta\kappa$-dependence reflects the feature of I_{BS}. At a higher P_{NBI} (e.g. 70 MW), n_{WLmax} tends to increase with f_{GW} (, and with high bootstrap current fraction).

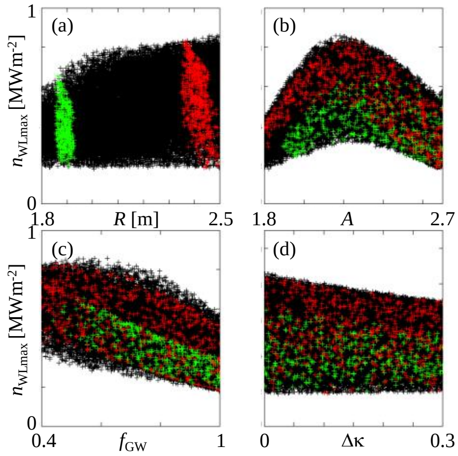

Fig. 2 maximum neutron wall loading n_{WLmax} as functions of R (a), A (b), f_{GW} (c), κ (d) when $P_{\mathrm{NBI}} = 50$ MW. Device costs of ≈ 750 and ≈ 940 M$ are shown by green and red symbols.

Although the largest R in the surveyed range shows the maximum n_{WLmax}, the device cost also becomes the highest. Therefore, we should find the minimum cost which enables the required specifications. Parameter survey at different P_{NBI} has been performed. When we impose $P_{fus} > 70$ MW, $n_{WLmax} > 0.65$ MWm^{-2}, the parameter ranges (mean and standard deviation) for the minimum cost are $f_{\mathrm{GW}} \sim 0.5$, $R \sim 2.0$ m, $\kappa \sim 2.2$, $A \sim 2.3$, $P_{\mathrm{NBI}} \sim 50$ MW, $device\ cost \sim 800$ M$. Here, we assume 3% uncertainty in the device cost to consider statistical error, and $\pm 30\%$ in the relative cost of NBI. Since the cost of NBI can be about 40% of the total cost, the latter uncertainty can affect the obtained parameter ranges. The costs themselves were corrected to estimate those at present (2024) using US Consumer Price Index for necessary components. Note that Fig. 1 is a representative case near the cost minimum.

FAST is a project that provides a test facility for R&D and demonstration of system integration in the DT burning plasma environment. Quasi-zero-dimensional analyses showed an NBI heated, compact, low aspect ratio tokamak can achieve the required specifications. A high P_{NBI}, with a thin inboard shield (i.e., short distance between the coil and the plasma) and HTS magnets make the device very compact and unique. It is a future task to investigate the divertor heat load mitigation scenario and plasma start-up scenario.

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