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Parameter surveys for a fusion energy systems integration test facility FAST

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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 MW/m^2, discharge duration of about 1000 sec, full-power operation time of about 1000 hrs. 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. Construction cost is another important quantity, which determines the necessary funding and the construction period. 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. 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.

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. 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 $f_{\rm GW}$, major radius R, elongation κ, aspect ratio A, D-NBI injection power $P_{\rm NBI}$. The energy of the NBI is 500 keV. We adopted 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 $H_{\rm hy}$ = 0.9. Note that $H_{\rm IPB98}$ of 1.3 is necessary to reproduce this plasma when we adopt ITER IPB98(y,2) confinement time scaling. The maximum toroidal field 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. The plasma current is the sum of the bootstrap current and the NBI driven current, which are calculated from the formulas Here, not only thermal reactions, but also beam-thermal reactions are considered. The prompt orbit loss and shine through of NBI (and a-particles) are negligible in global balances for typical cases. We assume additional 10 % unidentified losses. Similarly, a 10 % loss for alpha-particle heating power is assumed. Thus, the plasma heating power becomes $0.9 \times (P_{\rm NBI} + P_\alpha) - P_{Brems}$, where P_{Brems} is the Bremsstrahlung power (at *Zeff*=2).

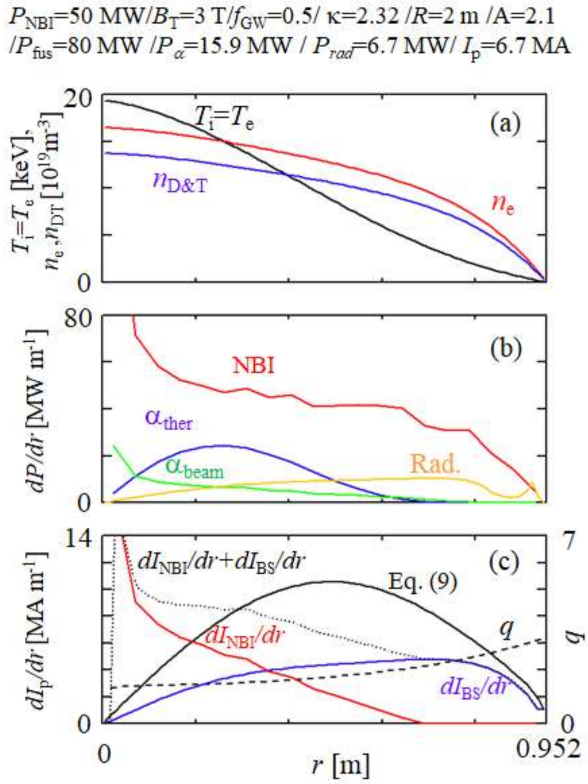

Figure 1. Temperature and density profiles (a), power deposition profiles (b) and current density profiles (c). The horizontal axis is the minor radius on the midplane.

The device cost is calculated by PEC [3], but it is updated using the data used in [4, 5]. In the device cost, 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. Neutron wall loading is calculated, and the maximum is located on the (outboard) equatorial plane. It was found that the ratio between the DT fusion power and the neutron wall loading should be appropriate in terms of the device cost.

References

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