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FAST: A European ITER satellite experiment in the view of DEMO

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ABSTRACT

Fusion Advanced Studies Torus (FAST) aims to contribute to the exploitation of ITER and to explore innovative DEMO technology. FAST has been designed to study, in an integrated scenario: (a) relevant plasma-wall interaction problems, with a large power load ($P/R \sim 22$ MW/m; $P/R^2 \sim 12$ MW/m^2) and with a full metallic wall; (b) to tackle operational problems in regimes with relevant fusion parameters; (c) to investigate the non-linear dynamics of fast particles (alpha like) in burning plasmas. FAST will operate on a wide parameters range, namely in high performance H-mode (BT ~ 8.5 T; IP ~ 8 MA) as well as in advanced Tokamak operation up to full non-inductive current scenario (IP ~ 2 MA). The main heating is based on 30 MW ICRH, but the ports have been designed to allocate up to 20 MW of 1 MeV NNBI. Helium gas at 30 K is used for cooling of the full machine, a preliminary analysis shows the possibility of realizing FAST with a complete superconductor set of coils. An innovative active system is under development to reduce and to control the magnetic ripple. Tungsten (W) or liquid lithium (L–Li) has been chosen for the divertor material plates and the code EDGE2D has been used to optimize the divertor geometry.

1. Introduction

It is presently widely accepted that a successful exploitation of ITER and a reliable as well as early design of DEMO will need a strong accompanying program. Along this roadmap, a key role should be played by the so-called "Satellite Experiments": JT-60SA and possibly FAST [1–3]. FAST (Fusion Advanced Studies Torus) aims at studying integrated plasma scenarios, to the broadest possible extent: (a) Plasma Wall problems that ITER will face, with an outlook on possible DEMO relevant scenarios; i.e. very large power load ($P/R \sim 22$ MW/m; $P/R^2 \sim 12$ MW/m^2), with actively cooled Tungsten divertor and First Wall (FW). (b) Highly performing plasma operations with large ELMs and the presence of all the presently foreseen mitigation systems, plus the necessity of completely integrated plasma control tools. (c) Possibility to study burning plasma stability and mutual feedbacks between thermal plasma and energetic particle populations [2], without involving Tritium, but using ICRH and (in a second phase) Negative NBI systems (NNBI). This operational flexibility can be achieved by a careful choice of the machine parameters, aimed at reproducing as much as possible the most important dimensionless physics parameters of ITER (ρ^*, β, ν^*) plus the plasma temperature T [4,5]. Within these constrains, the use of the so called Weak Scaling [2] approach has allowed (as a compromise between approaching $Q \sim 1$ and the machine cost) to select a possible experiment with the following parameters $B_T = 8.5$ T, $I_P = 8$ MA, $P_{ADD} = 40$ MW, $R = 1.82$ m, $R/a \sim 3$, $a = 0.64$ m, elongation $K = 1.7$, triangularity $\delta = 0.64$ and a plasma volume of ~ 23.5 m^{-3}. The flexibility of FAST is elucidated in Table 1, where the features of main plasma scenarios are tabulated. Two different standard H-mode scenarios are shown; in both cases the most common H_{98} scaling [6] has been used to compute the thermal plasma energy confinement time τ_E, as well as the thermal normalized β_N. In the second column it is illustrated the highest performances scenario ($Q_{DT}{}^{eq} \sim 1.5$), with a fairly large average plasma density

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Table 1
Operating scenarios.

FAST	H-mode reference	H-mode extreme	AT	Full NICD
I_p (MA)	6.5	8.0	3.5	2
q_{95}	3	2.6	5	5
B_T (T)	7.5	8.5	6	3.5
H_{98}	1	1	1.5	1.5
$\langle n_{20}\rangle$ (m^{-3})	2	5	1.4	1
β_N	1.3	1.7	2.2	3.4
τ_E (s)	0.4	0.65	0.25	0.13
τ_{Res} (s)	5.5	5	3	2–5
T_0 (keV)	13.0	9.0	12	7.5
Q	0.65	1.5	0.32	0.06
$t_{discharge}$ (s)	20	13	55	170
$t_{flat\text{-}top}$ (s)	13	2	45	160
I_{NI}/I_p (%)	15	15	60	>100
P_{ADD} (MW)	30	40	40	40

$\langle n_e\rangle = 0.8 \times n_{Greewald} = 5 \times 10^{20}\ m^{-3}$ and assuming the availability of 40 MW of additional heating (30 MW ICRH + 10 MW NNBI). In the first column, the so called Reference Scenario it is shown; here, the high performance phase lasts longer than the resistive diffusion time ($t_{FlatTop} \sim 13\ s \sim 2t_{Res}$), which is the longest "physical" time scale. In this case, 30 MW of ICRH heating have been assumed. In the last two columns two extreme Advanced Scenarios are illustrated. The first one is at intermediate beta and at high toroidal field. The last case indicates the possibility of studying extensively a completely steady state scenario at high beta ($\beta_N > 4l_i$) and relatively low toroidal field. Since the dimensionless scaling above quoted is valid under the assumption of similar plasma shaping, the most advanced technology and methodology has been used in the studies performed to guarantee the necessary equilibrium configurations flexibility and the vertical stability control [7,8]. The edge behavior has been analyzed in greater detail by using the EDGE2D-EIRENE code, for studying the main features of the FAST Scrape-Off Layer (SOL) [9]. Simulations suggest that detachment might be attained at the highest density, even without impurity seeding and thus at relatively low radiation losses, while an increase in the SOL and bulk radiation is needed for the intermediate density regimes. The divertor [10] has been designed so as to be easily and remotely replaced. In order to sustain the very large power load, the already tested Tungsten (W) monoblock technology is initially foreseen, but the use of innovative liquid Lithium divertor is planned, as well. The FW will be realized by a bundle of tubes covered by 4 mm of plasma sprayed Tungsten, capable to sustain up to ~7 MW m^{-2} [3,10]. The operating temperature is presently planned to be 100 °C, controlled by a suitable water loop, however it is under analysis the possibility to work at higher temperature (~300 °C) by using a cooling gas system. The machine will be contained within a stainless steel cryostat (Fig. 1). The vacuum vessel will be in Inconel, with ports in stainless steel, and it is designed with a very large number of ports that include the possibility to accommodate up to 20 MW (on two different ports) of tangential (~45° on the magnetic axis) NNBI at 0.7–1.0 MeV [11]. In Fig. 2, it is shown a schematic view of a 10 MW NNBI line, connected to the injection port on FAST. The beam will be able to couple the power on-axis and/or off-axis (around half radius). The main features of the system have been evaluated considering a Beam Source (BS), complete with 1 MV extraction and acceleration grids (Accelerator), a Neutralizer, a RID, a Calorimeter, a Vacuum vessel, a vacuum pumping system and a High Voltage Bushing and Transmission Line. The overall dimensions of the vessel are 14 m and 9.5 m of length and height respectively. The allocated equatorial ports are characterized by a relatively high and rather narrow aperture shape. Therefore the space available for the neutral beam is quite large in the vertical direction but very limited in the horizontal plane. To optimize the available space, and the focusing capability, the neutral beam is realized by two beamlet groups in the horizontal direction and five in the vertical one. The most stringent requirement for NBI realization is the respect of the space and the distances inside the equatorial port assigned to the NBI. Therefore the focusing of the beam has been positioned inside the port itself. Each beamlet group will consist of 16 beamlets in the vertical direction and 5 beamlets in the horizontal one. The Beam Source will be composed by five identical modules; each of the five drivers and the plasma expansion region [12] will supply 1 MV negative ions to a single accelerator segment. Each accelerator segment produces two beamlet groups. A detailed disruption analysis has been carried out to study the largest stress on the machine components [3,7]. The worst analyzed disruption was the 8 MA scenario and due to a Vertical Displacement Event (VDE) with an initial step up of 1 cm; the analysis was performed considering both the eddy and the halo currents [3]. The final result showed an average 250 MPa on the Vacuum Vessel. In the present reference design, 18 copper coils (inertially cooled at 30 K) provide the toroidal field. The discharge duration will always be limited by the heating of these coils and it ranges from ~15 s, at the highest field (B_T = 8.5 T) up to 170 s for the advanced Tokamak scenarios [3].

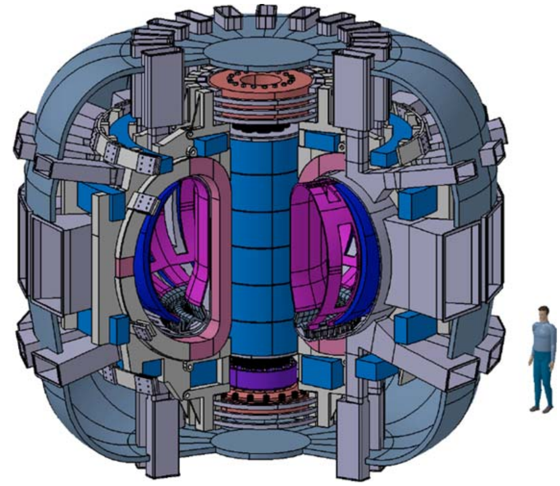

Fig. 1. FAST load assembly view.

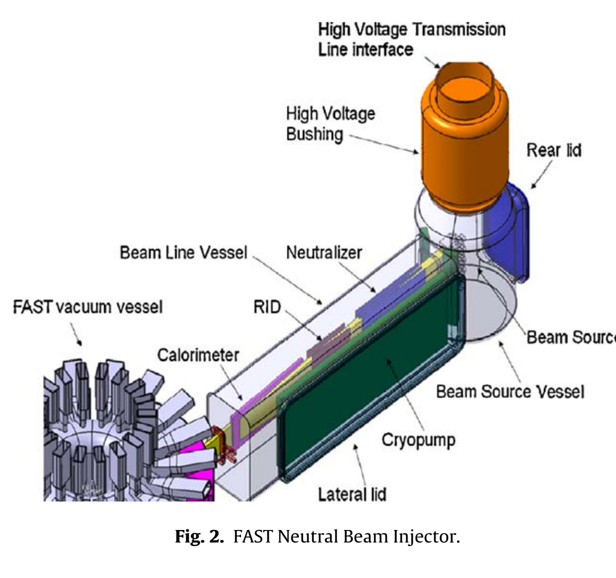

Fig. 2. FAST Neutral Beam Injector.