FAST Plasma Scenarios and Equilibrium Configurations

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Abstract. In this paper we present the Fusion Advanced Studies Torus (FAST) plasma scenarios and equilibrium configurations, designed to reproduce the ITER ones (with scaled plasma current) and suitable to fulfil plasma conditions for integrated studies of burning plasma physics, Plasma Wall interaction, ITER relevant operation problems and Steady State scenarios. The attention is focused on FAST flexibility in terms of both performance and physics that can be investigated: operations are foreseen at a wide range of parameters from high performance H-Mode (toroidal field, B_T, up to 8.5 T; plasma current, I_P, up to 8 MA) to advanced tokamak (AT) operation (I_P=3 MA) as well as full non inductive current scenario (I_P=2 MA). The coupled heating power is provided with 30MW delivered by an Ion Cyclotron Resonance Heating (ICRH) system (30-90MHz), 6 MW by a Lower Hybrid (LH) system (3.7 or 5 GHz) for the long pulse AT scenario, 4 MW by an Electron Cyclotron Resonant Heating (ECRH) system (170 GHz – B_T=6T) for MHD and electron heating localized control and, eventually, with 10 MW by a Negative Ion Beam (NNBI), which the ports are designed to accommodate. In the reference H-mode scenario FAST preserves (with respect to ITER) fast ions induced as well as turbulence fluctuation spectra, thus, addressing the cross-scale couplings issue of micro- to meso-scale physics. The non-iductive scenario at I_P=2MA is obtained with 60-70 % of bootstrap and the remaining by LHCD. Predictive simulations of the H-mode scenarios described above have been performed by means of JETTO code, using a semi-empirical mixed Bohm/gyro-Bohm transport model. Plasma position and Shape Control studies are also presented for the reference scenario.

1. Introduction

The FAST conceptual study has been proposed as possible European ITER Satellite facility with the aim of preparing ITER operation scenarios and helping DEMO design and R&D. Insights into ITER regimes can be obtained by experiments for the integrated study of plasma wall interactions with very large power load, of plasma operation open problems (i.e. Edge Localized Modes (ELMs), plasma control) and of non linear dynamics that are relevant for the understanding of alpha particle behaviours in burning plasmas by using fast ions accelerated by heating and current drive systems [1,2]. The use of ICRH in the minority scheme (H or ^{3}He) in D plasmas can produce fast particles that, with an appropriate choice of the minority concentration, of the RF power and of the plasma density and temperature, can reproduce the dimensionless parameters ρ^*_H and β_H characterizing the α-particles in ITER [3]. Here, ρ^*_H is the fast particles Larmor radius in units of the torus minor radius and β_H the fast particle beta. The prerequisites to be satisfied, in order to reproduce the physics of ITER relevant plasmas, yield the following set of FAST parameters: 1) plasma current, I_P, from 2 MA (corresponding to full non inductive current scenario) up to 8 MA (corresponding to maximized performance); 2) auxiliary heating systems able to accelerate the plasma ions to energies in the range of 0.5÷1 MeV; 3) major radius of about 1.8m and minor radius around 0.65m; 4) pulse duration from 20s for the reference H-mode scenario, up to 170 s (~ 40 resistive times τ_{res}) at 3MA/ 3.5T. The ITER design presently foresees the investigation of three main equilibrium configurations: a) standard H-mode at I_P=15MA with broad pressure profile (P_0/<P>=2); b) hybrid mode at I_P=11MA with narrower pressure profile (P_0/<P>=3);

c) Advanced Tokamak (AT) scenario at I_P=9MA with peaked pressure profile (P_0/<P>=4). FAST equilibrium configurations have been designed in order to reproduce those of ITER with scaled plasma current, but still suitable to fulfil plasma conditions for studying operation problems, plasma wall interaction and burning plasma physics issues in an integrated framework. An overview of some of the possible achievable configurations is given in Table I.

TABLE I : FAST PLASMA PARAMETERS.

FAST	**H-mode reference**	**H-mode extreme**	**Hybrid**	**AT**	**AT2**	**AT Full NICD**
I_p (MA)	6.5	8	5	3	3	2
q_{95}	3	2.67	4	5	3	5
B_T (T)	7.5	8.5	7.5	6	3.5	3.5
H_{98}	1	1	1.3	1.5	1.5	1.5
$<n_{20}>$ (m^{-3})	2	5	3	1.2	1.1	1
P_{th_H} (MW)	14 ÷18	22 ÷ 35	18 ÷ 23	8.5 ÷ 12	5 ÷ 7	5 ÷ 7
β_N	1.3	1.8	2.0	1.9	3.2	3.4
τ_E (s)	0.4	0.65	0.5	0.25	0.18	0.13
τ_{res} (s)	5.5	5	3	3	5 ÷ 6	2 ÷ 5
T_0 (keV)	13.0	9.0	8.5	13	13	7.5
Q	0.65	2.5	0.9	0.19	0.14	0.06
$t_{discharge}$ (s)	20	10	20	70	170	170
$t_{flat-top}$ (s)	13	2	15	60	160	160
I_{NI}/I_p (%)	15	15	30	60	80	100
P_{ADD}(MW)	30	40	30	30	40	40

For the H-Mode scenarios two possible cases have been considered. In the reference H-mode scenario (conceived for the extensive integrated studies) $\rho^*_H = 2.6\cdot10^{-2}$, β_H =0.7% and $\nu^* = 5.3\cdot10^{-2}$ ($\rho^*_{H,\ ITER}= 2.5\cdot10^{-2}$, $\beta_{H,\ ITER}$ = 1.1% and $\nu^*_{ITER} = 9\cdot10^{-2}$). Here ν^* is the plasma collisionality. The hybrid scenario would allow to reach an equivalent Q of about 1 at β_N=2 and n/n_{GW}=0.8, considering an enhanced confinement factor of 1.3*H_{98}, meanwhile. For AT scenarios, we considered three quite different cases. A moderate β_N (~2) case, with large B_T (6T) with I_P=3MA and q_{95}=5; a case with β_N (~3.2), B_T=3.5T, I_P=3MA and q_{95}=3 larger than the MHD limit; and a case with very large bootstrap fraction and complete non-inductive driven plasma current (I_P=2MA), with B_T=3.5T, q_{95}=5 and β_N~3.4. In all these cases, the discharge duration is much longer than the resistive current diffusion time (up to 40 times longer). The heating power in all cases is assumed to be the 30MW, provided by the ICRH system for the more standard cases, and 40 MW for the most extreme scenarios. In particular for the AT cases, the 10 extra MW are provided by the LH (6MW) plus the ECRH (4MW, second harmonic) systems. For the extreme H mode scenario, I_P=8MA and B_T=8.5T (q_{95}~2.7, Q=2.5), an additional 10MW NNBI has been assumed. In all cases the configuration has been designed to have always the same geometrical plasma features (see Fig. 1): R=1.82

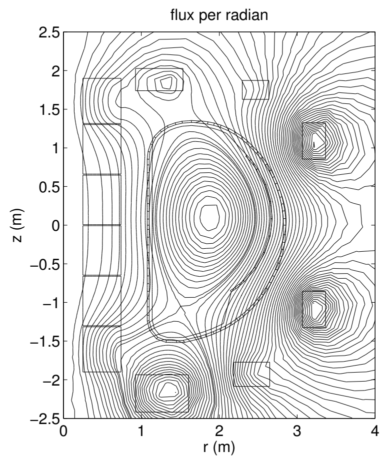

FIG. 1. FAST H-mode equilibrium.

m, a=0.64 m, k=1.7, <δ>=0.4. Helium gas at 30K is used for cooling the copper toroidal and poloidal magnets. The discharge duration is limited by the adiabatic heating of the toroidal field coils [4].

2. FAST equilibrium configuration

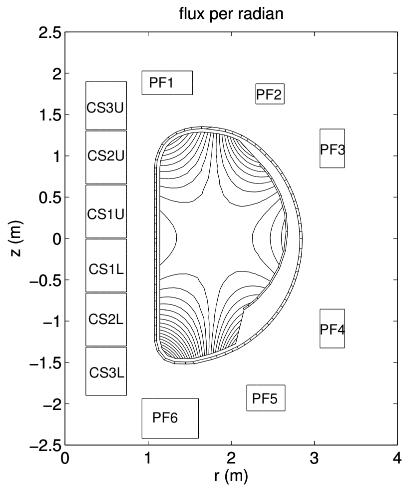

FIG. 2. FAST exapolar field null region during the plasma breakdown.

Coils position (Fig. 2) and size have been optimized with the constrains to minimize magnetic energy, to have sufficient flux to achieve and control selected plasma scenarios and to have a good field null during the plasma break-down. This is also shown in Fig. 2, where $B_p/B_T<2\times10^{-4}$ in the very large central exapolar region even at low field, i.e. B_T = 4T. The toroidal electric field for the break-down has been imposed to be 2V/m for a time lasting at least 40ms. The poloidal circuit has been designed to provide the necessary flux (~35 Wb stored) and to build-up the X-point configuration for the reference scenario with I_P=6.5MA; here the discharge lasts around 20s and the X-point configuration is sustained (at low and/or high beta) for ~13s (much longer than the plasma resistive time), see Sec. 3. The central solenoid (CS) column is split in 6 different coils to allow the largest plasma shaping flexibility. For the same reason, a large number of power supplies have been foreseen, as described in [4]. A large toroidal field ripple is not acceptable in ITER like experiment: it could lead to significant losses of high-energy particles with unpleasant peaking in the heat loads on the First Wall (FW), and/or it could have some negative implication in the H mode quality. To limit the toroidal field (TF) magnet ripple within acceptable values, ferromagnetic inserts have been used. A detailed 3-D evaluation of the TF ripple (TFR) in the whole region inside the vacuum vessel (VV) has been performed [5]. A whole 20° toroidal sector of FAST has been modeled with cyclic symmetry as boundary conditions. Without ferromagnetic inserts, the TFR value exceeds 2% in the outboard plasma region near the equatorial plane. The ferromagnetic inserts are located inside the outboard area of the VV, in correspondence of the TF coils (TFCs). Ripple at the plasma boundary (near the equatorial port) has been reduced from 2% to 0.3% with optimized Fe inserts. A new approach, based on the insertion of active coils between TFCs and the plasma, has been extensively investigated for ITER and FAST devices [5]. This active system would allow reducing TFR to values even smaller than that with the Fe inserts, with the advantage to improve the machine flexibility, by the possibility to perform experiments modulating the toroidal ripple. All the analyses have been carried out with 2D and 3D electromagnetic Ansoft Maxwell FEM code and showed the possibility of reducing the maximum TFR for FAST at the plasma boundary well below 0.3% by using active coils, fed with a current lower than ~(-1/14) of the TFC current, located just in front of the central part of the outer TFCs. All plasma equilibria satisfy the following constraints: a) minimum distance of 0.03m between plasma and first wall to avoid interaction between plasma and main chamber (this value results from multiplying $3\lambda_p$, with λ_p the power flux e-folding length assumed to be 0.005 m on the outer equatorial plane [6], by a factor of 2, taking account for uncertainties in the λ_p estimation; b) maximum current density in the poloidal field coils transiently around 32MA/m^2. Within these constraints, sufficient flexibility is maintained to allow different plasma shapes. In the divertor region enough space to vary substantially the plasma triangularity, to allow strike point sweeping and to have an efficient pumping capability, it has been allocated. The FAST divertor concept is thoroughly described in [4]. Tungsten (W) has been chosen as divertor plate material, however the full divertor can be removed by remote handling to allow tests of more advanced divertor concepts (in view of DEMO) as a divertor based on Liquid Lithium (L-Li). The crucial aspects of thermal loads on the divertor plates and of core plasma purity for the proposed scenarios are discussed in [6], where a slight increasing of the density ($<n_e>\sim 1.3\cdot 10^{20} m^{-3}$) it is proposed for the full NICD scenario in order to attain the acceptable Z_{eff} value of 1.5.

3. H-mode scenarios

The time evolution of coil currents, along with plasma geometrical and physical parameters guaranteeing the sequence of plasma shapes during a pulse, defines a tokamak scenario. The scenario is usually constructed from a finite number of plasma equilibria computed by MHD codes, which determine plasma geometry and current density distribution satisfying force balance in the externally imposed magnetic field. The main parameters of the reference H-mode equilibrium, obtained by using MAXFEA in combination with FIXFREE code, are reported in Table 2. Meanwhile, the time evolution of equilibrium configurations is shown in Fig. 3, whereas the time evolution of the poloidal circuit currents that have been used to achieve the described equilibrium is reported in Fig. 4.

TABLE 2: MAIN PARAMETERS OF THE REFERENCE H-MODE EQUILIBRIUM

I_P (MA)	6.5
β_N	1.3
P_0/<P>	2.35
q_{95}	3
q_{axis}	1.09
Volume (m^3)	22.8

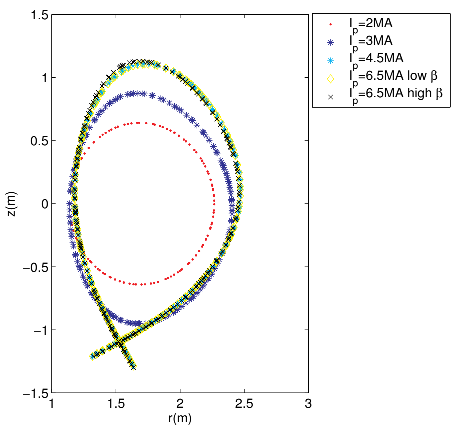

FIG. 3. Time evolution of FAST H-mode equilibrium configurations

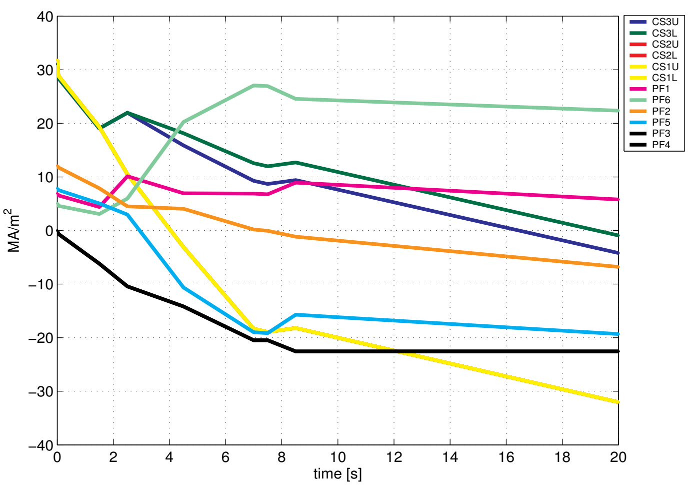

FIG. 4. FAST reference H-mode: PFCs currents evolution.

In this scenario CS2U=CS2L and CS1U=CS1L. After the breakdown, the plasma current rises up to I_p =2.0MA in Δt=1.5s; during this phase, the plasma evolves with a circular shape. Between t=1.5s and t=2.5s the plasma current reaches I_P =3MA, without achieving the X-point, but evolving in shape. Eventually, at t=4.5s the final plasma shape is achieved with I_P=4.5MA. Between t=4.5s and t=7s, the plasma current achieves his target value I_P = 6.5MA, while β_N remains very low. At t=7.5, full additional heating is assumed, causing an increase of the internal kinetic energy on a time scale longer than the plasma energy confinement time. Since it is foreseen to adopt a plasma control technique like the Extreme Shape Controller (XSC) used in JET [9], during this strong β increase the plasma boundary is assumed to remain fixed. After t=8.5s, everything is assumed to remain fixed up to the end of the current plateau at t=20s. This yields an experimental flat-top of about 11.5s (~2 τ_{res}) at the maximum β_N (~1.3) and a plasma current flat-top of approximately 13s. For this scenario, aiming to determine the behavior and to study the transport and confinement of fast (~1 MeV) ions produced by 30 MW ^{3}He (~1%) minority ICRH [2], the plasma parameters obtained above have been fully validated by means of JETTO code, using a semi-empirical mixed Bohm/gyro-Bohm transport model [10]. Thermalization of these fast particles is also providing bulk plasma electron and ion collisional heating. Several iterations between JETTO and TORIC code simulations, in combination with the SSQLFP code (which solves the quasi-linear Fokker-Planck equation in 2D velocity space) to deal with the coupled problem of propagation and quasi-linear absorption of ICRH, have allowed to consistently evaluate the dynamic scenario evolution [3]. The electron and ion temperature profiles obtained during the high β phase are shown in Fig. 5, while the plasma current density and safety factor profiles are plotted in Fig. 6.

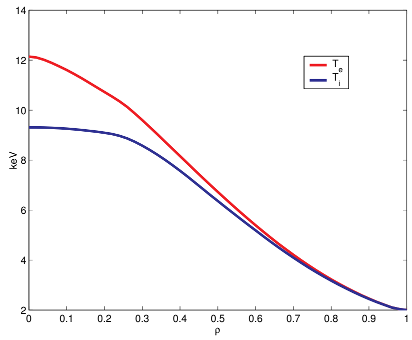

FIG. 5. Predictive JETTO simulations: electron (T_e) and ion temperature (T_i) profiles during the high β phase

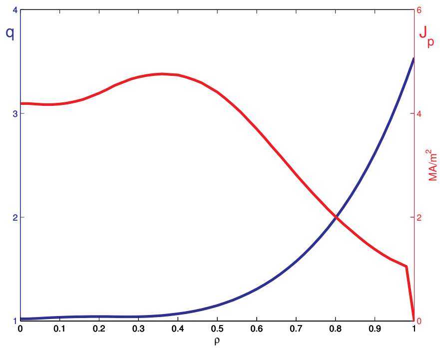

FIG. 6. Predictive JETTO simulations: plasma current density J_p and safety factor q profiles during the high β phase

The central ion and electron temperatures are not the same, despite the strong coupling due to the relatively high density ($\langle n_e \rangle \sim 2\cdot10^{20}m^{-3}$), as a consequence of the dominant electron heating, which characterizes operation scenarios of relevance to burning plasmas. The behaviour of fast ions produced by ICRH has been simulated with the hybrid MHD-gyrokinetic code HMGC [11,12], investigating linear and nonlinear dynamics of Alfvén modes excited by the supra-thermal particle tails and the corresponding fast ion transport [3, 13]. Energetic particle density, perpendicular and parallel temperatures profiles, previously obtained by the combined use of JETTO and TORIC-SSQLFP codes, have been used as input for HMGC. Preliminary results on fast particles behaviours confirm that the Alfvén fluctuation spectrum in FAST will be dominated, as in ITER, by a dense spectrum of modes with characteristic frequencies and radial locations [14,15,16]. The FAST mode number and frequency spectra (normalized to the Alfvén frequency) being self-similar to those of ITER can reproduce both Alfvén Eigenmode (AE) as well as Energetic Particle Mode (EPM) dynamics [14,15,16]. Edge Localized Modes (ELMs) are one of the major "concerns" intrinsically connected with the "standard" H-mode scenario in FAST as in ITER. Since the FAST reference scenarios rely on good quality (H_{98}=1) H-modes, it is natural to expect that in FAST there will be a noticeable ELM activity. Within this scenario, FAST will operate with plasma edge conditions very close to those of ITER, with simultaneously low edge collisionality (ν^*_{Ped}~$1.5\cdot10^{-1}$) and very high edge plasma density (n_{ePED}~$1.5\cdot10^{20}m^{-3}$). Under these conditions, ELMs of comparable size with that of tolerable ITER ELMs (~1.5MJ) are foreseen [6]. A transient (τ_E~0.3τ_{flat_top}<<τ_{res}~5s) H-mode scenario at I_P=8MA and B_T=8.5T (q~2.5, Q=2.5) has also been studied, yielding the highest achievable performance by assuming an additional 10MW NNBI power input. In this scenario, the plasma density has been assumed close the Greenwald limit (n_e~$5\cdot10^{20}m^{-3}$~0.8 n_{eGW}); under these conditions, strong electron-ion coupling is foreseen, yielding $T_e \approx T_i$.

4. AT scenarios

Three different AT scenarios have been considered: AT with moderate β_N (~2) and large toroidal field (B_T=6T), with I_P=3MA and q_{95}=5; AT2 with β_N (~3.2) larger than the $4\cdot l_i$ MHD stability, at lower toroidal field (B_T=3.5T), with I_P=3MA and q_{95}=3; and a case with fully non-inductive plasma current (Ip=2MA) at large β_N (~3.4), with B_T=3.5T and q95=5. In this latter case, a bootstrap fraction of about 60-70% is foreseen, while the remaining current will be driven by an LH system (5GHz), capable of driving 30-40% of the plasma current. For all cases, the plasma boundary shape is essentially the same as that shown in Fig. 1 for the reference H-mode scenario, whereas the q profile is assumed to be slightly reversed, with q_{axis} >2 and q_{min}<2 (at around half radius) and a peaked pressure profile (P_0/<P>~3.5), as expected in advanced tokamak scenarios. For the first two scenarios, in non-full current drive (assuming a residual loop voltage of about 60÷100mV), the poloidal circuit can easily sustain the discharge for a long time ($t_{flat\text{-}top}$ ~ 60s in AT and $t_{flat\text{-}top}$ ~ 160s in AT2 scenarios). Of course, also in the fully non inductive scenario, a maximum $t_{flat\text{-}top}$ ~ 160s is foreseen, the time limit always been given by the toroidal magnet heating. The amount of LH power coupled with the plasma (6MW), is sufficient to guarantee the control and the sustainement of the required current profiles for AT regimes, whereas 4 MW of ECRH (in second harmonic) will provide enough power for MHD control and/or electron heating. A study of LH penetration and absorption has been performed in a parameter range typical of FAST AT scenario. Current drive efficiencies of about $0.25\cdot10^{20}$ Am^2/W have been evaluated. This will produce a driven current I_{LHCD}=0.65 MA in the reference AT scenario, corresponding to ~ 22% of the total, whilst an additional ~ 38% of non-inductive current (I_{NI}/I_p=60%, - Table 1) is driven by the bootstrap. According to simulations, this is sufficient to produce a negligible time evolution of the current profile during the whole discharge. Higher I_{LHCD} fraction is possible at lower densities at the expense of I_{BS}, so that the total non-inductive current remains close to 60%. An analysis of the global MHD stability for long pulse AT scenarios has been performed using the MARS code [18], in order to investigate the possibility of stabilizing Resistive Wall Modes (RWM) [2]. The feedback control analysis show that the use of internal poloidal field sensors can allow the full stabilization of the mode using either internal or external feedback coils, whereas radial field sensors do not allow stabilization. As an example of Neoclassical Tearing Modes (NTM) stabilization, the m/n=2/1 island evolution is calculated for the 6T, 3MA long pulse AT scenario and β_N ~2. The considered wave is launched from the equatorial port with 18° of toroidal angle. The wave propagation is evaluated with the ECWGB ray-tracing code [19]. The m/n=2/1 island evolution, calculated by the modified Rutherford equation, is shown in Fig. 9. With 2 MW

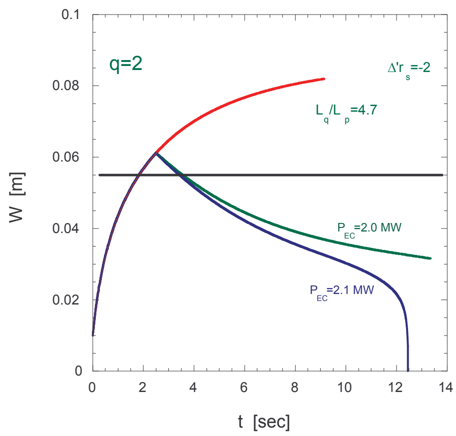

FIG. 9. Evolution of the 2/1 island without (red) and with (green/blue) EC power applied.

P_{EC} (EC power input), the island width is stabilized below 3 cm (about 34% of its saturation value), with 5.5 cm EC current density radial size and J_{CD}/J_{BS}=1.15 (ratio of driven to bootstrap current densities). A small increase of P_{EC} (less than 0.2 MW) leads to full island stabilization. Besides for NTM stabilization in long pulse AT scenarios, the ECRH system on FAST is also used for electron heating and current drive tasks at densities below $3.6{\cdot}10^{20}m^{-3}$, which is the cut-off for the ordinary mode for the chosen frequency of 170 GHz (the same as ITER). The foreseen ECRH power (~ 4 MW at the plasma) is also able to control the current profile in the plasma core (r/a<0.3), where the J_{CD} term, 50% larger than J_{OH} (ohmic), can modify the q profile. Linear and quasi-linear calculations of ICRH wave absorption have been performed for the AT fully non-inductive current-drive (NICD) scenario (B_T=3.5, I_P=2MA and central density n_{e0}~$2.14X10^{20}m^{-3}$) with a ^3He minority-heating scheme (7% concentration) by means of TORIC and SSQLFP codes used in combination. The antenna parameters considered for this study are the following: 1) 30MW coupled ICRH power at frequency f=33MHz (resonant layer r/a≈0.1, close to the plasma centre), 2) power spectrum centered at parallel wave-number n||=9.5. Preliminary results show that the power deposited on minority ions (50%) is completely redistributed to the bulk deuterium ions on the collisional time-scale. In Fig. 10, the effective parallel and perpendicular temperatures of the supra-thermal minority tail are shown vs. the normalized radius. The perpendicular temperature reaches the peak value of 90 keV, well below the critical energy. Assuming 6MW of LH at 5GHz around 30-40% of I_{LHCD} can be driven, well aligned with the foreseen bootstrap current (~60-70%) and with the assumed reversed q profile.

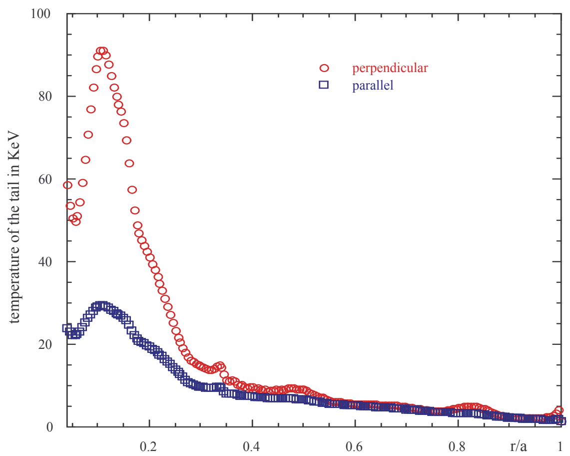

FIG. 10. The effective parallel (blue) and perpendicular (red) temperatures of the supra-thermal ^3He minority tail vs. the normalized radius r/a.

5. Plasma position and shape control

Preliminary analyses have been performed to study the control of the plasma current, shape and position during the flat-top of the reference H-mode plasma scenario [4, 21]. The optimization of a copper shell position inside the vacuum vessel (see Fig.1) slows the vertical stability growth time down to 100ms, with a comfortable stability margin. To avoid flux shielding during plasma breakdown the shell is toroidally segmented. The structure of the foreseen controller (in Fig. 11) consists of a feedback loop, which controls the derivative of the vertical position (using CS2U-CS2L and PF3-PF4 coil pairs), and a slower multivariable feedback loop, which controls the plasma current, shape and position. No 3D

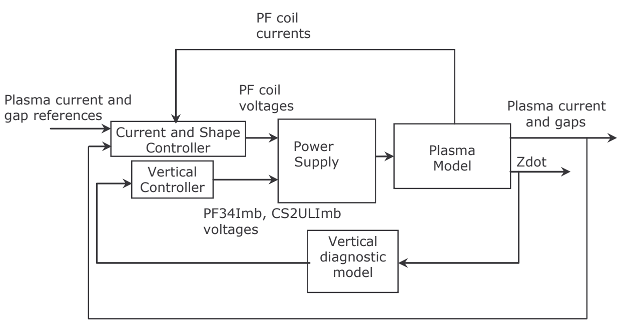

FIG. 11. The FAST controller structure.

effects associated with the shell and vessel structure have been considered so far. The response of the system to a 1cm Vertical Displacement Event (VDE), a minor disruption and a step of 100kA in the plasma current have been simulated by monitoring the plasma-wall distance (gap) at six different poloidal locations. The resulting maximum change in the plasma wall gap is less than 8 cm with a settling time less than 2s. The power required for this stabilization is about 14 MW.

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