PLASMA SCENARIOS AND MAGNETIC CONTROL IN FAST

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Abstract

The Fusion Advanced Studies Torus (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 of operation in Deuterium plasmas can be obtained from investigations 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. In this paper the plasma scenarios that can be studied on FAST are presented, with emphasis on the aspect of its flexibility in terms of both performance and physics that can be investigated. Plasma position and shape control studies are also discussed. A copper shell inserted inside the vacuum vessel has proposed, aimed at slowing down the growth rate of the vertical instability to a value around $13s^{-1}$. To avoid flux shielding during plasma breakdown, the shell is toroidally segmented. The maximum gap displacement after a minor disruption has been calculated to be less than $8cm$, with a re-settling time of about $2s$. The power required for this stabilization is about $14MW$.

Key words

Tokamak, Plasma Equilibrium Codes, SVD, PID

1 Introduction

FAST has been proposed to help preparation of ITER scenarios [1, 2, 3] and the development of new expertise for DEMO design and R&D in an integrated fashion [4], simultaneously addressing many aspects of non linear dynamics that are relevant for the understanding of alpha particle behaviours in burning plasmas and their interaction with plasma turbulence and turbulent transport, exploiting advanced regimes with long pulse duration with respect to the current diffusion time and up to full non-inductive current driven (NICD), testing technical innovative solutions for the first wall/divertor directly relevant for ITER and DEMO, and providing a test bed for ITER and DEMO diagnostics as well as an ideal framework for model and numerical code benchmarks, verification and validation in ITER and DEMO relevant plasma conditions. The prerequisites to be satisfied, in order to reproduce the physics of ITER relevant plasmas, yield the following set of FAST parameters [5]: 1) plasma current, I_P, from $2MA$ (corresponding to full NICD) up to $8MA$ (corresponding to maximized performance); 2) auxiliary heating systems able to accelerate the plasma ions to energies in the range of $0.5 \div 1MeV$; 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 $170s$ (~ 40 resistive times τ_{res}) at $3MA/3.5T$. 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. The paper is arranged as follows: the FAST load assembly and equilibrium configurations are presented in Section 2, the plasma scenarios are discussed in Section 3 and finally the design of the main parameters of the magnetic control system of FAST is described in Section 4.

2 FAST load assembly and equilibrium configurations

FAST has been designed to achieve a plasma behaviour sufficiently close to ITER together with a significant flexibility in the operation space [5, 6]. The maximum plasma current, I_P, ranges from $2MA$ in the full non inductive current drive scenario to $8MA$ in the high performance H-mode scenario, while the pulse duration can extend up to $160s$ (~ 40 resistive times τ_{res}) in the longest AT (Advanced Tokamak) scenario at $3MA/3.5T$. This features have been accomplished in a compact (major radius $R = 1.82m$, minor radius $a = 0.64m$, triangularity $= 0.4$), cost effective design with a complete set of plasma diagnostics and a full choice of auxiliary heating systems (ICRH - Ion Cyclotron Radio Heating, LH - Lower Hybrid Heating , ECRH - Electron Cyclotron Radio Heating and Negative ion Neutral Beam Injection - NNBI) able to give a dominant electron heating, to drive the plasma current and to accelerate the plasma ions up to energies in the range $0.5 \div 1MeV$. This design permits to address, at the same time and in integrated framework, non linear dynamics effects in the fast particles behaviours [1] typical of the burning plasma physics, plasma-wall interaction under ITER relevant power load [7] and ITER relevant operational issues, including the Advanced Tokamak regimes up to fully non inductive current driven scenarios.

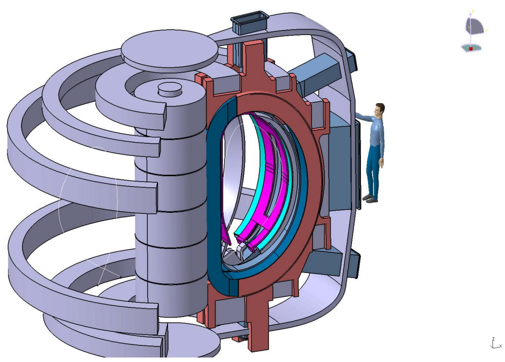

Figure 1. The load assembly of FAST.

FAST is based on a Load Assembly shown in Figure 1 and consisting in a vacuum vessel (VV) with its internal components (First Wall, Divertor, Stabilizing Shell), 18 Toroidal Field Coils (TFC) with the relevant stainless steel supporting belts and pre-compression rings (designed to keep the whole toroidal magnet structure in wedged configuration), 6 External Poloidal Field Coils (PFC) and a Central Solenoid (CS) vertically segmented in 6 coils to increase plasma shaping capability, manufacture easiness and cooling efficiency. Copper has been choose as material for all the coils (TFC, PFC and CS) to narrow the machine cost: the copper resistivity and then the adiabatic heating during the plasma pulse have been minimized keeping all the coils at cryogenic temperatures ($30K$) by mean of helium gas flow before the pulse and limiting the final temperature below 85K for any poloidal coils and $150K$ for the inner legs of the TFCs. The Vacuum Vessel is a single shell 30 to $40mm$ thick, made of Inconel 625 steel to minimise the flux consumption during the plasma start-up and to guarantee the stiffness requested to withstand the loads arising from plasma disruptions. The electrical resistance of the whole vessel is around $60\mu\Omega$, neglecting effects due to ports, at the operating temperature ranging from room temperature to above $100C$ and regulated by a proper water loop. The access to the VV is provided through vertical, oblique and equatorial ports in each 20 degree module. A passive stabilizing shell, consisting of 26 mm thick toroidally segmented copper plates, has been envisaged on the outboard, inside the VV, to make easier the vertical control of the plasma by slowing down the growth rate of the vertical instability around $13s^{-1}$. A set of ferromagnetic inserts [8], located in front of TFC inside the VV on the outboard between the stabilizing shell and the VV itself, have been designed to keep the Toroidal Field Ripple (TFR) below 0.3% on the plasma separatrix. The size and location of these inserts have been optimized to allow operations in the whole toroidal magnetic field range, from $3.5T$ to $8.5T$, with acceptable values of TFR. An alternative active system to reduce TFR well below 0.3% in every toroidal magnetic field configurations has also been evaluated [9]: it is based on the use of small active control coils located in the vessel close to the outer branches of the TFC and fed with a fraction of the TFC current in the opposite direction. The plasma facing components in FAST should withstand very high thermal fluxes [7], relevant to ITER and DEMO Plasma Wall Interactions regimes (normalized power load $P/R \sim 22MW/m$), then the whole vacuum vessel will be covered by a first wall consisting of a bundle of pipes armoured with plasma sprayed tungsten 4 mm thick, able to cope with $3MW/m^2$ expected peak heat flux. The divertor has been designed adopting the tungsten monoblock technology [7], which has been already tested with thermal loads well above the expected $18MW/m^2$ on average. Both the first wall and the divertor will be actively cooled by a proper water loop at $80C$. FAST could offer a full choice of auxiliary heating systems: $30MW$ of Ion Cyclotron Resonant Heating (ICRH) able to accelerate plasma ions in the range $0.7 \div 0.8MeV$, $4MW$ of Electron Cyclotron Resonant Heating (ECRH) for MHD control, heating, current drive at low density and 6 MW of Lower Hybrid (LH) for the current drive and profile control. Moreover an additional 10 MW NNBI system could be used to accelerate plasma ions in the extreme scenarios [1]. The poloidal magnetic field sys-

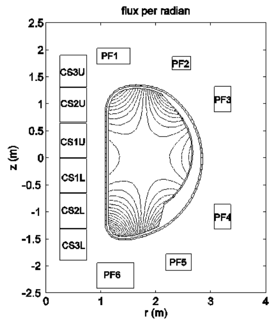

Figure 2. Poloidal Field Coils system and field null region during the plasma breakdown.

Table 1. FAST plasma parameters for the analysed configurations (the highlighted H-mode extreme configuration is foreseen in a second phase of the machine, with the additional NNBI system.

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	2.6	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
$< n20 > (m^{-3})$	2	5	3	1.2	1.1	1
$P_{thH}(MW)$	$14 \div 18$	$22 \div 35$	$18 \div 23$	$8.5 \div 12$	$8.5 \div 12$	$5 \div 7$
β_N	1.3	1.8	2.0	1.9	3.2	3.4
$\tau_E(s)$	0.4	0.65	0.5	0.25	0.18	0.13
$\tau_{res}(s)$	5.5	5	3	3	$5 \div 6$	$2 \div 5$
$T_0(keV)$	13	9	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	140
$t_{flat-top}(s)$	13	2	15	60	160	130
$I_{NI}/I_p(\%)$	15	15	30	60	80	100
$P_{ADD}(MW)$	30	40	30	30	40	40

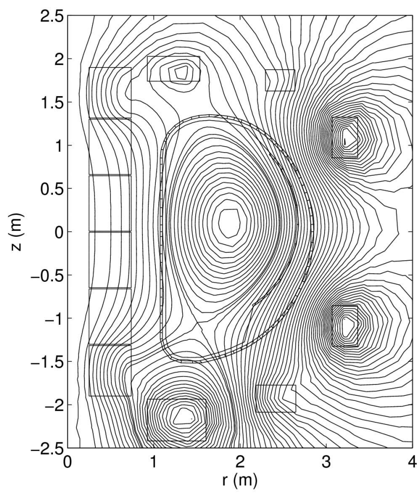

Figure 3. FAST reference H-mode equilibrium.

tem has been optimized, regards to locations and sizes of the coils depicted in Figure 2, to minimize the stored magnetic energy and the copper temperature rise while maximizing the available magnetic flux swing (about $35Wb$ stored) and the extension of the field null during the plasma break-down (a very large central hexapolar region with $Bp/BT < 2 \times 10^{-4}$at low field $B_T = 4T$, as shown in Figure 2). FAST has been designed as a very flexible device able to reproduce, with scaled plasma current, the three main ITER equilibrium configurations: standard H-mode with broad pressure profile, hybrid mode with narrower pressure profile and Advanced Tokamak scenario with peaked pressure profile.

FAST could then work in a dimensionless parameter range close to ITER, with similar equilibrium profiles, dominant electron heating (with $T_e \sim 10keV$) and plasma performance in the fusion parameter space with $Q \geq 1$. An outline of the possible plasma configurations analysed for FAST is given in Table 1. All these plasma configurations have the same geometrical features, as shown in Figure 3: major radius $R = 1.82m$, minor radius $a = 0.64m$, elongation $k = 1.7$, average triangularity $< \delta >= 0.4$, plasma volume $> 20m^3$, minimum distance between plasma and First Wall greater than $3\lambda_E$ (where λ_E is the energy e-folding length assumed to be about 1 cm on the equatorial plane). With a maximum transiently allowable current density in the poloidal field coils around $32MA/m^2$, sufficient flexibility is maintained to allow different plasma shapes, triangularity scan and strike point sweeping. The duration of the plasma flat-top is significantly greater than the resistive diffusion time in all these scenarios, with a ratio $\Delta t_{flat-top}/\tau_{res}$ about half than in ITER. The reference H-mode scenario ($I_P = 6.5MA$, $B_T = 7.5T$) has been designed to be used in the extensive integrated studies and is characterized by a high density ($n = 2 \times 10^{20}m^{-3}$), high current plasma ($I_p = 6.5MA$) with a X-point equilibrium that can be sustained as long as the magnetic flux sweeping allows (13 s of high current flat-top i.e. about two resistive decay time, in a discharge lasting around $22s$ and assuming $32MA/m^2$ as maximum current density in the PFC). Figure 4, 5 and 6 show, respectively, the time evolution of the equilibrium configurations, the PFC currents waveforms and the plasma density current, safety factor, electron and ion temperature profiles during the high β flat-top. The time evolution is characterized by a rise of the plasma current with a circular equilibrium up to $I_p = 2MA$ in $1.5s$ after the break-down, an increase of the elongation while the current keep raising for the next $3s$ when the final X-point shape configuration is achieved with $I_p = 4.5MA$ and then a further increase of the current until the current achieves its target value $I_p = 6.5MA$ at $t = 7s$. The full additional heating, applied at $t = 7.5s$, causes a large increase of the internal kinetic energy (and then of β_N) on a time scale (about $1s$) longer than the plasma energy confinement time: during this increase the plasma boundary shall be preserved by using a technique like the Extreme Shape Controller (XSC) adopted in JET [10]. The extreme H-mode scenario ($I_P = 8MA$, $B_T = 8.5T$ with a safety factor $q_{95} \sim 2.6$) corresponds to the highest achievable performance in terms of Q, by assuming the use of the additional NNBI system. In this transient scenario ($\tau_{flat-top} = 2s$ with $\tau_E \sim 0.7s$ and $\tau_{res} \sim 5s$) a large species coupling (and then $T_e = T_i$) is foreseen to happen due to the high plasma density close to the Greenwald limit.

Several Hybrid and Advanced Tokamak (AT) scenarios can be achieved by FAST, with quite different features: Hybrid ($I_P = 5MA$, $B_T = 7.5T$, able to reach $Q \sim 1$ with $\beta_N \sim 2$ and $n/n_GW = 0.8$), AT (large $B_T = 6T$ with moderate $\beta_N \sim 2$ and plasma current $I_P = 3MA$ driven by LH for 22% and bootstrap for 38%), AT2 ($I_P = 3MA$, lower $B_T = 3.5T$

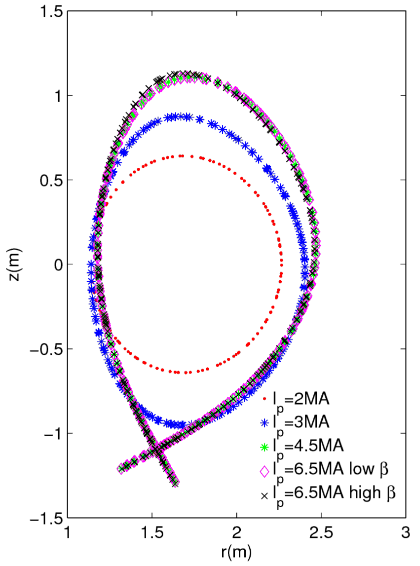

Figure 4. Evolution of the plasma configurations in the reference H-mode scenario.

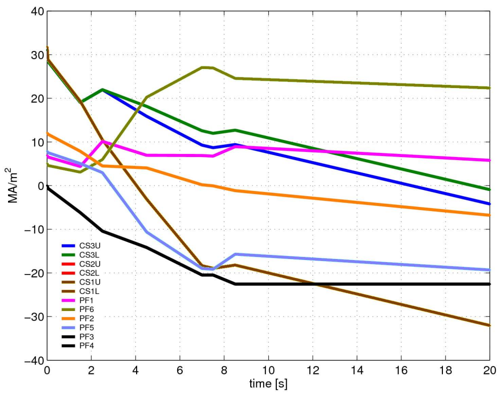

Figure 5. Evolution of the PFC currents in the reference H-mode scenario.

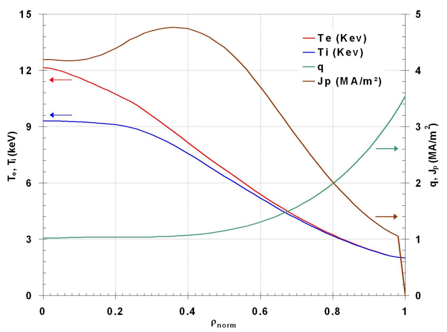

Figure 6. Predictive JETTO [11] simulations: plasma current density (J_p), safety factor (q), electron (T_e) and ion temperature (T_i) profiles during the high β phase.

with $\beta_N \sim 3.2$ greater than the MHD stability), full Not Inductive Current Drive NICD ($B_T = 3.5T$ with very large $\beta_N \sim 3.4$, density $n = 1 \times 10^{20} m^{-3}$ and fully non inductively driven plasma current $I_p = 2MA$ consisting of $60\% \div 70\%$ bootstrap and $30\% \div 40\%$ LH driven fractions). In all these configurations the plasma shape is the same as the reference H-mode. The plasma discharge is sustained in the AT scenarios as long as poloidal flux is available to drive the residual inductive current ($t_{flat-top} \sim 60s$ in AT and $t_{flat-top} \sim 160s$ in AT2 assuming a plasma residual resistivity about $60 \div 100mV$) while in the NICD scenario the duration is constrained by the TFC adiabatic heating ($t_{flat-top} \sim 160s$). In all these cases the discharge lasts quite longer than the resistive time (from 20 up to 40 times).

2.1 3. Plasma position and shape control

The control of the plasma position and shape is a crucial issue as in every compact, elongated and high performance tokamak as FAST. The capability of the Poloidal Field Coil system, as presently designed, to provide an effective vertical stabilization of the plasma has been investigated using the CREATE_NL response model [12, 13], assuming axisymmetric deformable plasma described by few global parameters.

The plasma chamber has been schematized as a Inconel 625 vessel, 25 mm thick, with a resistivity equals to $1.29\mu\Omega m$ at operating temperature: the resulting torus resistance is $62.6\mu\Omega$, neglecting the $3D$ effects of the ports. A stabilizing copper shell inside the vacuum vessel has been designed, optimizing its thickness ($26mm$) and location to provide a slowing down of the growth rate of the vertical instability around $13s^{-1}$ with a safe stability margin equal to 0.97. To avoid flux shielding during plasma breakdown the shell has been toroidally segmented, providing the up-down connection by the poloidal path around the ports, so the net total toroidal current flowing in it is zero. Figure 3 describes the FAST reference equilibrium, modelled by CREATE_NL code, together with the passive stabilizing structures: vessel and copper shell. 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. The structure of the proposed controller (in Figure 7) consists in 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. The two pairs of coils selected for the vertical control will be fed by up-down anti-symmetric currents provided by a dedicated power supply. The voltage for the vertical stabilization system is supplied by a converter driven by simple saddle network controller driven by the plasma vertical velocity. The vertical diagnostics has been modelled as a first order low pass filter with a time constant of $3ms$. To model the plasma, derivation of linear models describing the dynamics of the $n = 0$ plasma displacements around the $6.5MA$ FAST configuration at the flat-top has been carried out by means of the CREATE-NL code [11, 12]. The closed loop system guarantees that, in the presence of a disturbance, the plasma vertical velocity goes to zero, while the plasma vertical position of the current centroid is not recovered. The stability of the vertical velocity loop is guaranteed with a phase margin of about 65^o and a gain margin of $5dB$. The current and shape controller structure uses as controlled variables, besides plasma current, six linear combinations of 39 gaps (between the plasma separatrix and the plasma facing components), strike points and X-point descriptors, obtained using a SVD approach. Therefore a SVD (singular value decomposition) based PID control scheme has been used for controlling the plasma current, shape and position:

$$u_y(t) = K_x x_{pf}(t) + K_p y(t) + K_I \int_0^t (y - y_{ref}) dt \quad (1)$$

where y_{ref} indicates the references for the controlled variables. Consequently the controller also requires the measures of the PF coil currents. The power supply system has been modelled, in a conservative approach as a pure time delay of $10ms$: under this assumption the vertical stabilization controller and the power supplies voltage limits have been designed so as to guarantee a settling time for the plasma velocity of about $800ms$. As far as the current/shape disturbances rejection (recovery of the gaps within $1cm$), the controller has been evaluated simulating the system response to a $1cm$ plasma Vertical Displacement Event, to a $100kA$ step in the plasma current and to a minor disruption characterized by a 20% fall in internal inductance and poloidal beta. In all these cases, the recovery is guaranteed by the current/shape controller as presently designed, with a settling time less than $2s$. The maximum power required for this stabilization is about $14MW$, in the range of the capabilities of the designed PFC system, with the most demanding disturbances, VDE and minor disruption. The voltage and current required for the stabilization, together with the time evolution of the plasma position, velocity, current and gaps, are shown in Figure 8 in the case of the minor disruption simulation.

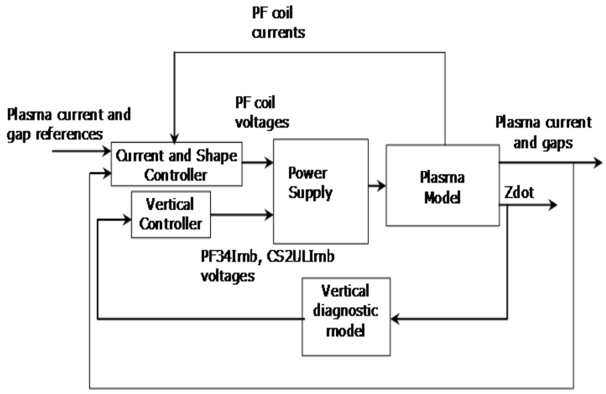

Figure 7. The FAST controller structure.

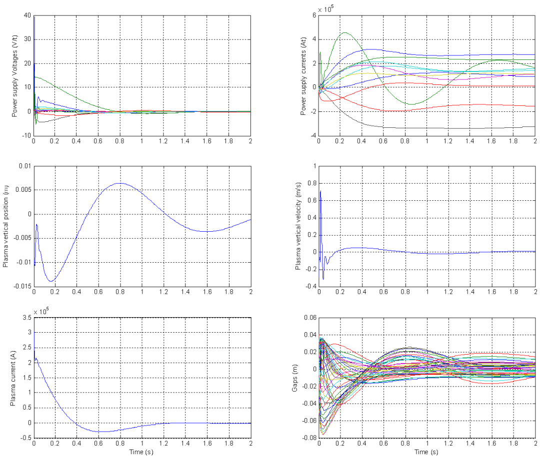

Figure 8. Position and shape controller performances for a minor disruption simulation by CREATE_NL model.

2.2 Conclusions

The Fusion Advanced Studies Torus (FAST) has been designed to provide a European ITER Satellite facility able to explore Fast Particle physics, to investigate ITER relevant Plasma Operations issues, to study the physics and test the technologies required to deal with large heat loads on ITER and DEMO plasma facing components, to investigate long lasting Advanced Tokamak regimes up to fully non inductive scenarios, to validate numerical simulation codes predictions of ITER fusion and burning plasma performance. FAST will be able thus to address most of the EFDA strategic missions and to support the preparation of ITER operation scenarios by using fast ions accelerated by heating and current drive systems, working with deuterium plasmas in a dimensionless parameter range close to that of ITER. FAST equilibrium configurations permit the preparation of ITER scenarios in a compact, cost-effective device still suitable to fulfil plasma conditions required to study burning plasma physics issues in an integrated framework. The FAST flexibility in terms of both performance and physics that can be investigated is emphasized by the variety of plasma scenarios that can be studied, from the extreme high performance H-mode to the full not inductive current driven scenario. The feasibility of a proper plasma position and shape control with the current Poloidal Field system design has been also introduced, showing the possibility of guaranteeing a wide stability region and of rejecting undesired shape modification.

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