Introduction

The successful development of ITER and DEMO scenarios requires preparatory activities on devices that are smaller than ITER, sufficiently flexible and capable of investigating the peculiar physics of burning plasma conditions.

The aim of the Fusion Advanced Studies Torus (FAST) proposal [2.1] (formerly FT3 [2.2]) is to show that the preparation of ITER scenarios and the development of new expertise for the DEMO design and R&D can be effectively implemented on a new facility. FAST will a) operate with deuterium plasmas, thereby avoiding problems associated with tritium, and allow investigation of nonlinear dynamics (which are important for understanding alpha–particle behaviour in burning plasmas) by using fast ions accelerated by heating and current drive systems; b) work in a dimensionless parameter range close to that of ITER; c) test technical innovative solutions, such as full-tungsten plasma-facing components and an advanced liquid metal divertor target for the first wall/divertor, directly relevant for ITER and DEMO; d) exploit advanced regimes with a much longer pulse duration than the current diffusion time; e) provide a test bed for ITER and DEMO diagnostics; f) provide an ideal framework for model and numerical code benchmarks, their verification and validation in ITER/ DEMO-relevant plasma conditions.

2.2 Physics

Scientific rationale

The scientific rationale of the FAST conceptual design is its capability to address the peculiar physics issues of burning plasma in the integrated framework of operation scenarios unobtainable in existing or foreseen experimental devices. This can be achieved in a flexible and cost-effective way in the FAST proposal, thanks to the choice of a high equilibrium magnetic field and to the consequent possibility of being able to operate routinely at high plasma current, high plasma density and moderate temperature, while maintaining a significant fusion performance at equivalent fusion power gain Q>1. A typical pulse duration of 25-30 resistive times is foreseen for studying advanced tokamak (AT) plasma scenarios, with the possibility of reaching up to 80 resistive times for extended pulse length, similar to that of ITER. Ion cyclotron resonance auxiliary heating systems are the optimal choice for accelerating minority plasma ions to 0.5 -1.0 MeV energies.

In particular it will be possible to simultaneously achieve:

- Production and confinement of energetic ions in the 0.5-1.0 MeV range, corresponding to dominant electron heating in the range 40-90% of the total collisional power transfer (fusion alphas in ITER will deliver ~70% of their energy to electrons).
- Dimensionless fast-ion and thermal plasma particle orbits close to ITER values, for reproducing micro- and meso-scale fluctuation spectra, their cross-scale couplings and related transport processes.
- Large ratio between heating power and device dimensions at high density and low collisionality for investigating the physics of large heat loads on the divertor plates (ratio of heating power to major radius [P/R] similar to that of ITER), as well as the production and control of ITER- and DEMO-relevant edge localised modes (0.4 MJ/m^2 are envisaged).

Furthermore, optimised performance scenarios at Q=3 can be obtained with plasma current up to 8.5 MA, q=2.6 and a flat-top of 2.5 s.

[2.1] *FAST conceptual study report,* ENEA Internal Report RTI/2007/001 (Dec. 2007)

[2.2] F. Romanelli et al., Fusion Sci. Technol. **45**, 483 (2004)

[2.3] R. Albanese et al., *Unified treatment of forward and inverse problems in the numerical simulation of tokamak plasmas*, Proc. of the 11th Inter. Symposium

Plasma scenarios and equilibrium configurations

The ITER design presently foresees the investigation of three main equilibrium configurations: a) a standard H-mode at plasma current I_p=15 MA with a broad pressure profile (p_0/<p>=2); b) a hybrid mode at I_p=11 MA with a narrower pressure profile (p_0/<p>=3); c) an AT scenario at I_p=9 M with a peaked pressure profile (p_0/<p>=4). FAST equilibrium configurations have been designed to reproduce those of ITER with a scaled plasma current, but still suitable to fulfil plasma conditions for studying burning plasma physics issues in an integrated framework. The plasma parameters obtainable in the various equilibrium configurations are then determined on the basis of a coupled core-edge 0-D code. An overview of the three main configurations (H-mode, hybrid and advanced scenarios) is given in table 2.I. A transient scenario at I_P=8.5 MA and B_T=8 T (q~2.6, Q=3) has also been investigated, assuming an additional 10 MW of negative neutral beam injection power input. The configurations have been designed so that the geometrical plasma features (fig. 2.1 where a FAST H-mode equilibrium is obtained by the CREATE-NL code [2.3]) R=1.82 m, a=0.64 m, k=1.7, <δ>=0.4 remain the same in all the cases.

The discharge duration is always limited by the heating of the toroidal field coils, which are inertially cooled by helium gas at 30 K.

The heating power in all three cases is assumed to be 30 MW provided by the ICRH system (60-80 MHz). However, for the long-pulse AT scenario, 6 MW of LH (3.7 or 5 GHz) have been added to actively control the current profile, whereas 4 MW of ECRH (170 GHz - B_T=6 T) provide enough power for MHD control. In the reference H-mode scenario, one obtains for FAST: $\rho^*_{fast}=2\times10^{-3}$, $\hat{\beta}_{fast}$=1.1% and $\nu^*=8.7\times10^{-2}$ ($\rho^*_{fast,ITER}=1.25\times10^{-3}$, $\hat{\beta}_{fast,ITER}$= 1.1% and $\nu^*_{ITER}=9\times10^{-2}$). The hybrid scenario would allow an equivalent Q of about 1 to be reached, considering an enhanced confinement factor of 1.3×H98. Meanwhile, β_N=2 and n/n_{GW}=0.8. In the AT scenario, the non-inductive/inductive current ratio I_{NI}/I_p is 60% and the pulse length $t_{flat\text{-}top}$ is ≈25 times the resistive time τ_{res}.

Predictive simulations of the above scenarios have been performed by means of the JETTO code, using a semi–empirical mixed Bohm/gyro-Bohm transport model [2.4]. Plasma position and shape control studies are in preparation for the reference scenario.

Table 2.I - FAST plasma parameters (the values obtained by seeding argon impurity into the divertor are given in brackets)

	H-mode	Hybrid	AT
I_p (MA)/q_{95}	7.5/2.9	5/4	3/5
B_T (T)	8.0	7.5	6
H_{98}	1	1.3	1.5
$<n_{20}>$ (m^{-3})	2.8	3	1.3
n/n_{GW}	0.5	0.8	0.5
$P_{th,LH}$ (MW)	17-23	18-23	8.5-12
β_N	1.4	2.0	2.0
$t_{flat\text{-}top}$ (s)	6	15	60
τ_{res} (s)	5	2.8	2.5
τ_E (s)	0.58	0.52	0.25
T_0 (keV)	12	8.5	15
f_{rad} (%)	18 (75)	20 (55)	63
Z_{eff}	1.1 (1.3)	1.0 (1.3)	1.35
Q	1.26	0.9	0.18
$t_{Discharge}$ (s)	14	20	70
I_{NI}/I_p (%)	18	30	60

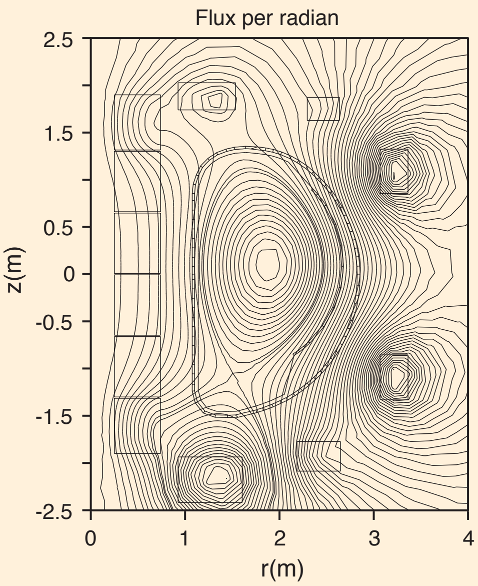

Fig. 2.1 - FAST H-mode equilibrium obtained by CREATE-NL code

on Applied Electromagnetics and Mechanics - ISEM 2003 (Versailles, France, Conference Record) pp 404–405

2.4] G. Cenacchi and A. Taroni, *JETTO: A free boundary plasma transport code (basic version)*, JET Report JET-IR (88) 03 (1988); and G. Vlad et al., Nucl. Fusion **38**, 557-570 (1998)

Minority ion acceleration by ICRH

Minority ions, accelerated by rf waves in the range of MeV energies, predominantly transfer their energy to plasma electrons via collisional slowing down. The use of ICRH in the minority scheme (H or ^{3}He) in D plasmas can indeed produce fast particles that, with an appropriate choice of the minority concentration, rf power, plasma density and temperature, can reproduce the dimensionless parameters ρ^{*}_{fast} and β_{fast} characterising the alpha-particles in ITER. Thus, a device operating with deuterium plasmas in a dimensionless parameter range as close as possible to that of ITER and equipped with ICRH as the main heating scheme would make it possible to address a number of burning plasma physics issues, e.g., fast–ion transport due to collective mode excitations and cross-scale couplings of micro-turbulence with meso-scale fluctuations due to the energetic particles themselves.

A detailed study with reference to the FAST conceptual design was performed to determine the characteristic fast-ion parameters necessary for addressing the above-mentioned burning plasma physics issues and to present a stability analysis of collective modes excited by the ICRH-induced energetic ion tail. The 2D full-wave code TORIC was used, coupled to the SSQLFP code, which solves the quasi-linear Fokker-Planck equation in 2D velocity space. The HMGC hybrid code was also used to investigate the destabilisation and saturation of fast-ion-driven Alfvénic modes below and above the EPM stability threshold, applying as initial velocity space distribution function a bi-Maxwellian distribution for energetic particles, which takes into account the anisotropy in the velocity space ($T_{\perp}>T_{||}$) due to ICRH. A parametric study based on density and temperatures profiles given by the TORIC code was also carried out.

Edge plasma issues

Among the R&D missions for possible new European plasma fusion devices, the FAST project will address the issue of “First wall materials & compatibility with ITER/DEMO-relevant plasmas”. FAST can operate with ITER-relevant values of P/R (up to 22 MW/m, against the 24 MW/m of ITER, inclusive of the alpha-particle power), thanks to its compactness; thus it will be possible to investigate the physics of large heat loads on the divertor plates.

The FAST divertor will be made of bulk tungsten (W) tiles, for basic operations, but also fully toroidal divertor targets made of L-Li are foreseen. Viability tests of such a solution for the DEMO divertor will be carried out as final step of an extended programme started on the FTU with the L-Li limiter.

To have reliable predictions of the thermal loads on the divertor plates and of the core plasma purity, a number of numerical self-consistent simulations has been performed for the H-mode and steady-state scenarios by using the code COREDIV. This code, already validated in the past on experimental data (namely JET, FTU, TEXTOR), is able to describe self-consistently the core and edge plasma in a tokamak device by imposing the continuity of energy and particle fluxes and of particle densities and temperatures at the separatrix.

The overall picture shows that at the low plasma densities typical of steady-state regimes, W is effective in dissipating input power by radiative losses, while Li needs additional impurities (Ar, Ne). In the intermediate and, mainly, in the high-density H-mode scenarios, impurity seeding is needed with either Li or W as target material, but a small (0.08% atomic concentration) amount of Ar, not affecting the core purity, is sufficient to maintain the divertor peak loads below 18 MW/m^{2}, which represents the safety limit for the W monoblock technology presently accepted for the ITER divertor tiles. The impact of ELMs on the divertor in the case of a good H-mode with low pedestal dimensionless collisionality has also been considered, and it has been shown that FAST will reproduce quite closely the ITER edge conditions, so it will be possible to study and optimise them in order to minimise the ELM perturbation.

Diagnostics

One of the key missions of FAST is the study of fast-particle dynamics in H-mode plasmas at high magnetic field (B_{T}). Diagnosing fast particles is not easy due to the high spatial and

[2.5] F.P. Orsitto et al., Nucl. Fusion **47**, 1 (2007)

temporal resolution needed [2.5]: the minimum relevant spatial scale ($\delta R/a$, a=minor radius) for measurements has to be of the order of $\delta R/a$~1/20-1/30, close to the fast particle Larmor radius, and the minimum time scale (τ/τ_A, $\tau_A=R_0/V_A$=Alfvén time ~0.2 μs, R_0=major radius, V_A=Alfvén velocity) is of the order of τ/τ_A~300, close to the relevant time scales for the interaction of Alfvén modes with fast particles [2.5]. The study of the H-mode needs high-resolution kinetic measurements of the pedestal, edge current profile and fine divertor diagnostic systems. At lower B_T values, FAST allows investigation of AT regimes with ITBs: these scenarios require diagnostic systems for the measurement of current profile, plasma turbulence, MHD fluctuations and plasma rotation (toroidal and poloidal) and a very flexible real-time control system.

The diagnostics foreseen for FAST can be subdivided in five groups: 1) burning plasma; 2) kinetic parameters and current profile; 3) magnetics; 4) SOL and divertor; 5) turbulence and emission of radiation. Group 1 (neutron monitors, neutron/gamma camera, neutron spectroscopy, collective Thomson scattering, escaping and confined fast-ion probes) is essential to the mission of FAST and enables the study of burning plasma, fast particles and MHD-related parameters. Group 2 (motional Stark effect and charge exchange recombination spectroscopy, electron cyclotron emission, Thomson scattering, CO_2 interferometry, Bremsstrahlung, spectroscopy and polarimetry) is used for the physics evaluation of plasmas. Group 3 (magnetic sensors) is used for the equilibrium and measurement of MHD activity and for real-time control of the machine (safe operation and scenario realisation). Group 4 (divertor spectroscopy, main chamber reciprocating probes, infrared thermography) is dedicated to determining SOL kinetics, thermal loads on the wall and divertor, erosion, deposition and plasma fluxes. Group 5 (Langmuir probes, reflectometry, bolometry, tomographic gas puff imaging, microwave Thomson scattering) should provide a complete picture of the temporal evolution of electrostatic instabilities (fluctuations in plasma density) related to the ion temperature gradient (ITG) and possibly to the electron temperature gradient (ETG), with reasonable space resolution inside and at the plasma edge.

FAST can be a test bed for the development of the ITER diagnostics that require consistent R&D activity, particularly those for measuring the energy and density distribution of confined and escaping fast particles, for the following reasons: plasma conditions similar to those of ITER (production and confinement of energetic ions accelerated by ICRH delivering a high fraction [40-90%] of their energy to electrons; fast-ion-induced fluctuation spectrum, power load and fast particles); reduced costs and development time of the diagnostics (as FAST works in DD with a tungsten wall, problems and cost related to the use of tritium and beryllium are avoided); flexibility of the device in terms of testing different technical solutions to optimise diagnostic performance; reduced optimisation time due to fewer operational constraints, e.g., in scheduling diagnostic system commissioning.

2.3 Design Description

Load assembly

The FAST load assembly (fig. 2.2) consists of 18 toroidal field coils (TFCs), 6 central solenoid (CS) coils, 6 external poloidal field coils (PFCs), the vacuum vessel (VV) and its internal components and mechanical structure. Resistive coils, maintained at cryogenic temperatures, are adiabatically heated during the plasma pulse.

FAST is a flexible device capable of operating in several scenarios with I_p from 2 MA (around 3-min-long pulse advanced scenario) up to 7.5 MA (H-mode scenario). Since the FAST magnet has to sustain the duration of a long pulse, helium gas at 30 K has been chosen for cooling the oxygen-free copper magnet, as the ratio of copper resistivity to specific heat is minimum at this temperature.

The load assembly is kept under vacuum inside a stainless steel cryostat to provide the thermal insulation of the machine. Several feedthroughs allow the penetrations of bus bars, fluidic pipes and support legs.

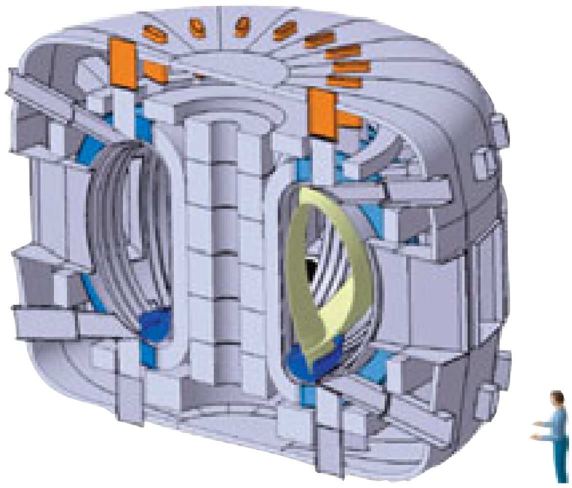

Fig. 2.2 - Axonometric view of FAST load assembly

The cryostat overall dimensions could be thought of as a 7.8-m-diam, 5-m-high circular cylinder. Penetrations are provided for the 18 equatorial ports and the 36 vertical and horizontal ports.

The magnet consists of 18 coils, each one made of 14 copper plates suitably worked in order to have 3 turns in the radial direction. The 42 turns of each coil are welded so that they correspond to the most external region in order to obtain a continuous helix. Each TFC is contained by a stainless steel belt fitted to the outside zone of the coil itself. Two pre-compressed rings situated in the upper and lower zones guarantee structural continuity in a wedged configuration. From the structural standpoint, the magnet section is adequate to self-sustain the forces. The magnet insulation is made of glass-fabric epoxy with a maximum thickness of 0.5 mm between turns and 3 mm to ground. The maximum turn thickness is 30 mm. The plates are tapered at the innermost region to get the needed wedged shape. In this area the filling factor is around 0.9. The inductance and the total magnetic energy of the toroidal magnet during the H-mode plasma pulse are respectively L=332 mH and E_T=1.33 GJ.

After a long current pulse in AT scenarios (i.e., 3 MA – 6 T; flat-top 60 s), the toroidal coils reach a maximum temperature of about 180 K, while the poloidal coil system reaches 60 K as a maximum. Cooling of the toroidal magnet system is guaranteed by a global helium gas flow of about 4 kg/s through suitable channels cut out in the coil turns.

The finite distribution of 18 TFCs can cause significant losses in the confinement of high-energy particles because of their being trapped inside the “ripple” valleys introduced along the field lines by these discontinuities. Hence, the toroidal field ripple in the whole region inside the FW was accurately evaluated, taking into account the real 3D shape of the TF. To limit the TF magnet ripple within acceptable values ferritic inserts have been introduced inside the outboard area of the VV. Different design solutions were compared. At the end of this analysis the ripple on the plasma separatrix was reduced from 2% to 0.3% with optimised Fe inserts.

The main components of the PF system are the CS and the external poloidal coils (3+3 coils). The free-standing CS is subdivided into six coils to increase plasma shaping flexibility, facilitate manufacture and allow cooling. The CS is held in place by means of a central post, which axially constrains it through a spring washer compression stack. Radial grooved plates at the interfaces between coil segments maintain concentricity. The PFCs and busbars are made of hollow copper conductors. The coils are layer wound and have an even number of layers so that the electrical leads can be located on the same side of the coil. The conductors are wrapped with glass fabric and kapton tapes and vacuum impregnated with epoxy resin. The design filling factors is 0.85. The magneto resistive effect was taken into account to calculate the coil temperatures. In the AT scenario with a 3-min-long pulse the final temperature is not expected to ever exceed 130 K in any of the poloidal coils.

The VV is made of 18 D-shaped toroidal sectors welded together by automatic remote equipment. To reduce the start-up flux consumption, the shell is made of Inconel to minimise the VV time constant (about 25 ms); the ports are made of stainless steel. The shell is manufactured by hot forming and welding. The equatorial ports have a 400×1460 mm rectangular area, the vertical ports have a length of 390 mm, and the horizontal are almost triangular with 200 mm max width. The ports are adequate for the required diagnostic system, the ICRH system, remote handling access, etc.

The VV will be supported by the toroidal field magnet system by means of vertical brackets attached to the TFC case through the equatorial ports of the vessel. It has been assumed that the vertical displacement event (VDE) occurs at approximately constant plasma current until the safety factor q at the plasma boundary decreases to a limit value; at this time a low-q-limit disruption occurs and a fast plasma current quench follows at a rate of about 1.5 MA/ms. The disruption events were simulated with the use of the MAXFEA code. The total vertical force produced by the VDE during the 6.5-MA H-mode operating scenario was evaluated with the MAXFEA code and is about 6 MN. The operating temperature of the vessel ranges from room temperature to 100°C.

FAST has to operate with elongated plasmas in single X-point configurations. The first wall (FW) has to withstand thermal loads, both in normal operating conditions and in the case of disruptions, as well as electromagnetic loads due to eddy and halo currents. The FW and the divertor are actively cooled by pressurised water with velocity 5 and 10 m/s, respectively. The FW consists of a bundle of tubes armoured with ~3 mm plasma-spray tungsten. The heat load on the FW is, on average, 1 MW/m^2 with a peak of about 3 MW/m^2. The design has to be remote-handling compatible and maintenance will be carried out from the equatorial ports.

Tungsten and L-Li have been chosen as the divertor-plate material, and argon and neon as the injected impurities to mitigate the thermal loads. As L-Li can be replenished, it would solve the problem of damage and hence might be a solution for the DEMO divertor. Modelling of the coupled edge/SOL-bulk plasma has been started, using the COREDIV code for the H-mode and steady-state scenario. In high-density regimes, $n_e \geq 3\times10^{20}$ m^{-3}, the SOL density is so high as to reduce the sputtered impurity flux from W, and radiation losses due to intrinsic impurities are small. Consequently, almost all the heating power is delivered to the divertor and the average power load on the plates could exceed 18 MW/ m^2, so mitigation with impurity seeding has to be considered. In the case of L-Li as the divertor target, the cooling rate remains too low and prompt re-deposition prevents Li from entering the main plasma, hence requiring the use of an additional impurity. For the steady-state scenario with average density (n_e=1.3×10^{20} m^{-3}) a large concentration of injected impurity ions is required to achieve high radiation fractions, but this leads to an increase in the value of Z_{eff}. The technology adopted for the W divertor is monoblock, which has been extensively tested in the relevant heat flux range. The armour consists of hollow tungsten tiles inserted in a copper tube heat-sink. The impact of the ELMs on the divertor has also been analysed. By assuming the same spatial deposition profile as inter-ELM and a factor of two asymmetry in the in-out ELM energy deposition, the energy density on the inner divertor is expected to be about 0.4 MJ m^{-2}, to be compared with the value of 0.5 quoted as the limit for tolerable W erosion in ITER.

The scheme of maintenance operations is similar to that for ITER, with the frame acting as a carousel all around the machine. The divertor maintenance is based on the development of an ad hoc cassette mover tractor capable of grasping and moving the divertor cassette. An articulated boom plus a front-end manipulator have been considered for the first wall assembly/disassembly.

Heating systems

To achieve burning plasmas conditions in FAST, plasma ions will be accelerated in the half-MeV range through an ICRH system (f=60 –90 MHz) able to couple up to 30 MW of rf to the plasma. The launchers are based on conventional current straps fed by external conjugate-T matching networks to achieve good resilience to fast plasma instabilities. For long-pulse AT scenarios, a 6-MW LH current drive system (f= 3.7 GHz) has been designed to actively control the current profile. Passive active multijunction (PAM) launchers are envisaged for this system to face the harsh plasma conditions expected in FAST. A back-up solution with a more suitable 5-GHz frequency has been studied, the final choice being affected by the availability of suitable rf power generators (500 kW, CW). The third system is a 4-MW ECRH system (f=170 GHz) that will provide enough power for MHD control. The ECRH power is also available for current profile control and electron heating.