

# **FAST - THE FUSION ADVANCED STUDIES TORUS “A PROPOSAL FOR A FACILITY IN SUPPORT OF THE DEVELOPMENT OF FUSION ENERGY”**

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## **Abstract**

*FAST is a new machine proposed to support ITER experimental exploitation as well as to anticipate DEMO relevant physics and technology. FAST is aimed at studying, in burning plasma relevant conditions, fast particle physics, plasma operations and plasma wall interaction in an integrated way. FAST has the capability to approach all the ITER scenarios significantly closer than present day experiments by using Deuterium plasmas. The necessity of achieving ITER relevant performance with a moderate cost has led to conceiving a compact Tokamak ( $R=1.82$  m,  $a=0.64$  m) with high toroidal field (BT up to 8.5 T) and plasma current ( $I_p$  up to 8 MA). In order to study fast particle behaviours in conditions similar to those of ITER, the project has been provided with a dominant Ion Cyclotron Resonance Heating System (ICRH; 30 MW on the plasma). Moreover, the experiment foresees the use of 6 MW of Lower Hybrid (LHCD), essentially for plasma control and for non-inductive Current Drive, and of Electron Cyclotron Resonance Heating (ECRH, 4MW) for localized electron heating and plasma control. The ports have been designed to accommodate up to 10 MW of negative beams (NNBI) in the energy range of 0.5-1 MeV. The total power input will be in the 30-40 MW range in the different plasma scenarios with a wall power load comparable with that of ITER ( $P/R\sim 22$  MW/m). All the ITER scenarios will be studied: from the reference H-mode, with plasma edge and ELMs characteristics similar to the ITER ones ( $Q$  up to  $\approx 2.5$ ), to a full current drive scenario, lasting around 170 s. The first wall as well as the divertor plates will be of Tungsten in order to ensure reactor relevant operation regimes. The divertor itself is designed to be completely removable by remote handling. This will allow studying (in view of DEMO) the behaviour of innovative divertor concepts, such as those based on liquid Lithium. FAST is capable of operations with very long pulses, up to 170 s, despite that it is a copper machine. The magnets initial operation temperature is 30 K, with cooling realised by helium gas. The in vessel components, namely first wall and divertor, are actively cooled by pressurised water at 80 °C. The same water is also used to back up the vacuum vessel. FAST is equipped with ferromagnetic inserts to keep the toroidal field magnet ripple down to 0.3%.*

## **Riassunto**

FAST è un nuovo esperimento proposto in supporto alla sperimentazione di ITER. FAST è stato proposto sia per studiare le fenomenologie tipiche dei plasmi che 'bruciano' sia per operare molto più vicini agli scenari di ITER di tutte le altre macchine realizzate esistenti o in via di realizzazione. FAST utilizza deuterio in modo da mantenere una elevata flessibilità di operazione. La necessità di mantenere l'investimento entro costi moderati ha determinato le dimensioni molto compatte ( $R = 1.82$  m,  $a = 0.64$  m), il campo magnetico alto (BT fino a 8.5 T) e la corrente di plasma fino a 8 MA. Le particelle veloci con cui studiare il comportamento delle particelle alfa prodotte dalle reazioni di fusione grazie sono originate utilizzando potenza a radiofrequenza. Infatti, per studiare i comportamenti delle particelle veloci nelle condizioni simili a quelle di ITER, il progetto prevede un sistema di riscaldamento a radiofrequenza alla di risonanza ciclotronica ionica (ICRH; 30 MW al plasma). Inoltre, l'esperimento prevede l'uso di 6 MW alla frequenza ibrida inferiore (LHCD), essenzialmente per controllo del plasma e per alimentare la corrente in modo non induttivo, e della frequenza ciclotronica elettronica (ECRH, 4MW) per il heating dell'elettrone ed il controllo localizzati del plasma. Gli accessi alla macchina sono stati dimensionati per accogliere fino a 10 MW di fasci negativi (NNBI) nella gamma di energia di 0.5-1 MeV. La potenza ausiliare addizionale sarà di 30-40 MW che origina un carico termico alla parete paragonabile con quello di ITER ( $P/R\sim 22$  MW/m). FAST sarà in grado di studiare tutti gli scenari operativi di ITER: dal modo H di riferimento, con il bordo del plasma e caratteristiche degli Elms simili a ITER ( $Q$  fino a  $\approx 2.5$ ), a scenari steady state che durano fino a 170 S. La prima parete come pure le piastre di divertore sarà di tungsteno per operare in condizioni rilevanti per il reattore. FAST è capace di funzionamenti con gli impulsi molto lunghi, fino a 170 s, malgrado sia una macchina con magneti in rame. La temperatura iniziale di funzionamento dei magneti è 30 K, il raffreddamento realizzato con gas elio. I componenti affacciati al plasma, vale a dire la prima parete e il divertore, sono raffreddati attivamente da acqua pressurizzata a 80°C. FAST è dotato degli inserti ferromagnetici per mantenere l'ondulazione toroidale del magnete di campo giù a 0.3%.

**Parole chiave:** Fusione nucleare, Confinamento magnetico, Tokamak, Particelle veloci, Radiofrequenza, Tungsteno, Litio



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# FAST –THE FUSION ADVANCED STUDIES TORUS “A PROPOSAL FOR A FACILITY IN SUPPORT OF THE DEVELOPMENT OF FUSION ENERGY”

## **Executive summary**

In order for fusion to play a major role in the energy portfolio by the second half of the 21<sup>st</sup> century, requires a rapid exploitation of ITER, the anticipation of DEMO design, and an adequate parallel program of material development (IFMIF).

To meet such an ambitious target, the fusion scientific and technological program must undergo a rapid progress, in particular during the first ten years of operation of ITER, where “DEMO-relevant” regimes of operation have to be understood, designed and realized. The R&D activity for this ambitious purpose is well summarized in the seven “missions” identified in the European Fusion Research Program, recently submitted to this Panel.

Meeting the targets set in that document calls for a dynamic, innovative and diverse accompanying programme. This implies that a new satellite experiment designed to properly solve the issues related with the operation, physics and control of a burning plasma (mission 1,2,3) in an integrated way and in relevant operating conditions is mandatory.

Such a Satellite is a facility which will ready before ITER and operate in parallel with it for an extended time. Its main objectives and justification are:

- shortening the time that will be required for ITER to accomplish its scientific program,
- amplifying the added value of new scientific insights into reactor relevant operation regimes (which are those where various physics are integrated),
- and helping DEMO early design, as foreseen by the European fusion road map.

To achieve these goals, and to investigate ITER- and DEMO-relevant scenarios in a meaningful way, there are stringent requirements on plasma behaviour and power handling capabilities, which calls for demonstration of new physics scenarios and developments of new technologies.

None of the machines presently in operation is capable to satisfy these prerequisites.



Moreover, considering the age of European facilities presently in operation, it is highly advisable to build a new tokamak to be operated as European satellite in order to maintain an adequate high level of scientific and technology skills in the European Community as well as to have the possibility to train scientist on an innovative, integrated and international device, capable to accompany ITER scientific program till its full exploitation.

The aim of the present proposal is to show that preparation of ITER and DEMO scenarios can be effectively implemented on a new facility, named FAST (Fusion Advanced Study Torus) that:

- works with Deuterium plasmas, avoiding the problems associated with the use of Tritium, and investigates non-linear processes that are relevant for the understanding of alpha particle behaviours in burning plasmas, by using fast ions accelerated by heating and current drive systems;
- works in a dimensionless parameter range close to that of ITER;
- tests technical solutions for the first wall/divertor directly relevant for ITER and DEMO such as full-tungsten wall and divertor and advanced divertor liquid-metal devices;
- accesses advanced tokamak (AT) regimes with long pulse duration with respect to the current diffusion time.

FAST can be ready before the ITER operation phase.

Given the ambitious aims of FAST and its importance in the fusion arena, it is clear that such a device will be designed, build and operated with the consensus and collaboration of several Associations. The long-standing expertise of the Italian Association in the fusion engineering and physics guarantees a solid starting point and that design and construction can be realized in the proposed time schedule.

**FAST requires, taking also into account the infrastructures available in Italy, limited investment and operation costs. Moreover, such a device** can also alleviate the financial burden of the program since the total investment cost could be kept around 280 M€ in seven years and the operating cost limited to around 14 M€/y plus personnel. Only 40% of this amount will be on the community budget. This would imply a remarkable improvement with respect the present situation, thus making available additional resources for the EU fusion program.

The conditions to be simultaneously satisfied in a satellite device like FAST to reproduce ITER relevant plasma parameters can be summarized as follows:

1. ITER relevant geometry, i.e. same shape of magnetic surfaces and divertor configuration;

2. Ratio between energy confinement time and electron-ion equipartition time similar to that of ITER;
3. Production and confinement of energetic ions in the half-MeV range (at  $T \sim 10$  keV). This guarantees that they will mainly heat electrons, in the range of 40-90% of the total fast ion power density (taking into account that the fusion alphas in ITER will deliver  $\sim 70\%$  of their energy to electrons);
4. Large ratio of the heating power to the device size, to investigate the physics of large heat loads on divertor plates as well as the production and control of ITER and DEMO relevant Edge Localized Modes (ELM);
5. Pulse duration (normalized to the plasma current diffusion time) similar to that of ITER to study advanced tokamak (AT) plasma scenarios, interesting for high-beta and steady state conditions.

Conditions 1-5 can be met in a compact device like FAST in a cost effective way, thanks to the operation at high magnetic field, high density and relatively low temperature, and to the choices of geometric and plasma parameters as well as of the appropriate heating scheme.

To accelerate plasma ions up to 0.5 MeV methods such as ion acceleration by Ion Cyclotron Resonance Heating (ICRH) or neutral beams produced by accelerating negative ions (NNBI) are employed. The combined use of ICRH and NNBI adds flexibility and richness to the physics issues that can be investigated.

The machine will be initially equipped with 30 MW ICRH, 4 MW ECRH and 6 MW LHCD power. FAST is designed to accommodate further upgrades in auxiliary heating systems, in particular for Neutral Beam injection.

The operating scenarios envisaged to address all the challenges listed above are reported in Table I.

FAST has been conceived to:

- substantially extend the capabilities of present machines on the individual physics issues identified by the programmatic missions,
- integrate these individual physics challenges into a single experiment capable of addressing them simultaneously and in a way complementary to JT60SA.

Thanks to the capability to operate routinely at current greater than 6.5 MA and the unmatched linear power (P/R), FAST can address in a very effective way missions 1, 2, 3 and has a good capability to contribute also to missions 4 and 5.

Table 1- FAST operation scenarios

	H-mode Ref-a	H-mode Ref-b	H-mode "Perf"	H-mode Extreme	Hybrid	AT	AT2
$I_p$ (MA)/ $q_{95}$	6.5/3	6.5/3	7.5/2.8	8/2.6	5/4	3/5	3/3
$B_T$ (T)	7.5	7.5	8.0	8.5	7.5	6	3.5
$H_{98}$	1	1	1	1	1.3	1.5	1.5
$\langle n_{20} \rangle$ ( $m^{-3}$ )	2	4	4	5	3	1.3	1.3
$n/n_{GW}$	0.4	0.8	0.7	0.8	0.8	0.5	.5
$P_{th\_H}$ (MW)	14-18	20-28	22-29	22-35	18-23	8.5-12	8.5-12
$\beta_N$	1.3	1.7	1.6	1.8	2.0	2.0	3.1
$t_{flat-top}$ (s)	13	13	6	~2	15	60	160
$\tau_{res}$ (s)	5.5	3	4	5	2.8	2.5	4
$\tau_E$ (s)	0.43	0.57	0.67	0.68	0.52	0.25	0.25
$T_0$ (keV)	13.0	8.5	9.0	9.0	8.5	15	10
$T_{plate}$ (eV)*	72 (60)	32 (9)	32 (9)	32 (9)	40 (15)	19	19
$f_{rad}$ (%)*	27 (39)	18 (75)	18 (75)	18 (75)	20 (55)	65	65
$Z_{eff}$ *	1.06 (1.55)	1.0 (1.2)	1.0 (1.2)	1.5 (-)	1.0 (1.3)	1.35	1.35
Q	0.65	1.2	1.8	3	0.9	0.18	0.18
$t_{discharge}$ (s)	20	20	14	~10	20	70	170
$I_{NI}/I_p$ (%)	15	20	18	15	30	60	80
P/R	22	22	22	22	22	22	22

It is worthwhile to note that the average ITER cost/shot is more than 50 times higher in than in FAST. This figure indicates that there is also an economical advantage to prepare ITER scenarios in a small device such as FAST. Moreover, also taking into account the fact that this device may well extend its lifetime until the DEMO construction phase and therefore can be used also to test technological issues and study plasma scenarios in preparation for DEMO.

The annual operation costs has been estimated for a typical agenda of 1500 shots (800 of which are performance shots) and 150 operation days: it amounts to 13.5 M€/year (8 M€ for maintenance, 2.5 M€ for consumables, mainly liquid nitrogen, and 3 M€ for electricity).

The FAST construction time is estimated in 6 years - plus one year for commissioning - after the design phase and the placement of the contracts for the long lead items. The realization will involve an average project team of 120 py/y.

It is important to mention how FAST can take advantage from the long experience of many of the laboratories involved in the design, construction and operation of fusion facilities (FTU,

RFX) as well as in the participation to international collaborations. The Italian Association is the main contributor to the JET exploitation.

FAST introduces major innovations thanks to the possibility to take advantages from new technologies as well as the most advanced design tools. This gives a unique opportunity to explore new physics and technology issues.

FAST integrates in a single device many scenarios, which existing devices can only address separately

FAST is therefore an intrinsically “multi-dimensional” device, where key physics issues can be explored and addressed simultaneously. This leads to a broad coverage of the operational space, ideal for “non-linear” physics with many time scales and threshold phenomena.

FAST will be a key facility for international collaborations, for training of young scientist in an ITER-relevant device, and for development of new diagnostics from many Associations.

FAST is relatively low-cost in comparison with existing and planned fusion devices, and its project and realization are triggered by an Association with long-standing records of engineering and physics skills.

## 1. INTRODUCTION

*The road map towards fusion energy production.* Fusion is the most promising energy source since it can satisfy the energy needs in a safe and environmentally responsible way. In order for fusion to play a major role by the second half of the 21<sup>st</sup> century, a rapid exploitation of ITER and an adequate parallel program of material development (IFMIF) are mandatory. Within this approach, the construction of a demonstration/prototype reactor (DEMO) could start before the fully exploitation of ITER. In order to meet such an ambitious time schedule a rapid progress of the ITER exploitation is specifically needed during the first ten years of operation, in order to achieve the demonstration of the DEMO regimes of operation by the start of DEMO construction. This requires a parallel R&D activity on devices that can investigate burning plasma conditions with higher flexibility than a Deuterium-Tritium device such as ITER.

*DEMO regimes of operation to be tested in ITER require a preparation on smaller devices.* The European Power Plant Conceptual Study shows that DEMO regimes must go beyond the regimes developed for ITER. Although the extrapolation in plasma parameters (with respect to ITER) is limited, their demonstration will require a significant exploratory effort. DEMO will indeed operate with a fraction of self-generated (bootstrap) current close to 70%, will use sophisticated methods for plasma control and will need techniques to handle the heat flux on the plasma facing components.

In addition, DEMO will operate with a fraction of self-generated nuclear heating in the excess of 90%, while ITER will reach approximately 70% at peak performance.

All these requirements push the plasma close to the operational limits and imply different technological solutions for plasma facing components and control methods; to perform such an R&D activity directly on a nuclear device such as ITER would clearly be difficult and expensive. Thus, the successful development of the DEMO scenarios, prior to testing them on ITER, requires a preparatory activity on devices smaller than ITER with sufficient flexibility and capable of investigating the peculiar physics of burning plasma conditions.

*A new facility is needed to prepare ITER and DEMO scenarios as well as to explore relevant technologies.* Although many of the existing devices can provide important contributions to the preparation of ITER operation, the requirement that plasma behaviour must be sufficiently close to that of ITER sets stringent constraints to plasma conditions that must be achieved in order to investigate ITER- and DEMO-relevant scenarios in a meaningful way. Furthermore, the power handling capability requires demonstration of physics scenarios and developments of new technologies. The aim of the present proposal is to show that preparation of ITER and DEMO scenarios can be effectively implemented on a new facility that:

- will work with Deuterium plasmas, avoiding the problems associated with the use of Tritium, and will **investigate 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;**

- will work in a **dimensionless parameter range close to that of ITER;**
- will test technical solutions for the **first wall/divertor directly relevant for ITER and DEMO** such as full-tungsten wall and divertor and advanced liquid metal divertor;
- will be able of exploiting **advanced tokamak (AT) regimes** with long pulse duration with respect to the current diffusion time;

Such a facility (FAST) can be ready before the ITER operation phase, and would require, taking into account the infrastructures available in Italy, limited investment and operation costs. FAST design, construction and exploitation will require the collaboration with other Associations. Moreover, FAST will make use of the existing competences in the Italian Association on fusion (ENEA, CNR Milan, Consorzio RFX, CREATE, POLITO, Universities of Roma II and Catania) and will focus the future Italian activities on fusion.

The scientific motivation of the proposal is discussed in Section 2. Section 3 outlines the preliminary design phase of the various components, with the aim of giving a sound basis for the cost estimate. The issues related with the choice of the site are discussed in Section 4. Cost, manpower and construction time schedule are reported in section 5.

## **2. SCIENTIFIC MOTIVATION OF THE PROPOSAL**

### **2.1 - Rationale for the choice of FAST parameters**

The conditions to be satisfied in order to reproduce ITER relevant plasma parameters can be summarized as follows:

- ITER relevant geometry (same shape of magnetic surfaces and divertor configuration);
- Ratio between energy confinement time and electron-ion equipartition time similar to that of ITER;
- Production and confinement of energetic ions in the half-MeV range in order to obtain the presence of dominant electron heating, in the range 40-90% (taking into account that the fusion alphas in ITER will deliver ~70% of their energy to electrons);
- Large ratio between the heating power and the device dimensions to investigate the physics of large heat loads on divertor plates as well as the production and control of ITER and DEMO relevant ELMs;

- Pulse duration (normalized to the plasma current diffusion time) similar to that of ITER to study AT plasma scenarios.

It is possible to show that all these conditions imply the following set of parameters:

- A plasma current  $I_p$  from 3 MA (corresponding to the long pulse advanced scenario) up to 8.5 MA (corresponding to the highest performance scenario);
- Auxiliary heating systems able to accelerate the plasma ions to energies in the range of 0.5 MeV;
- A device major radius dimension of about 1.8m;
- A pulse duration up to 25 resistive time in the advanced scenario.

In order to accelerate plasma ions up to 0.5MeV<sup>1</sup> it is not possible to use neutral beams produced by accelerating positive ions (PNBI), which is the most commonly used heating scheme, because the neutralization efficiency rapidly drops above 140 keV. Other methods such as ion acceleration by Ion Cyclotron Resonance Heating (ICRH) or neutral beams produced by accelerating negative ions (NNBI) must be employed. These two additional heating systems produce different plasma dynamic behaviours mainly due to their different radial deposition profiles rather than to the peculiar velocity space anisotropy of the respective energetic ion distribution functions. Both systems, used separately, can provide useful information on the processes that are relevant for understanding the fundamental dynamics of charged fusion products in burning plasmas. However, the combined use of ICRH and NNBI adds flexibility and richness to the physics issues that can be investigated. FAST scientific rationale is based on the use of ICRH only (see below for further details), although further upgrades including NNBI are not precluded.

The FAST parameters are shown in Table 1 and compared with those of JET, JT60-SA (the proposed upgrade of the JT60-U device, as foreseen by the Broader Approach agreement) and ITER.

The choices of geometric and plasma parameters as well as the heating scheme, which relies on ICRH as dominant heating power, allows FAST to:

- Produce fast ions in the correct energy range (above half-MeV);
- Operate with ITER relevant values of P/R, thanks to its compactness; thus making possible to investigate the physics of large heat loads on divertor plates;
- Have pulse duration long enough for AT scenario exploitation, much longer than that of existing normal conductor magnet machines (see Table 2).

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<sup>1</sup> The 0.5MeV figure refers to the “effective” temperature of ICRH accelerated ions. The injection energy of beam ions can be as high as 1MeV.

Table 1 - FAST, JET, JT60-SA and ITER parameters.

	FAST	JET	JT60-SA	ITER
R(m)/a(m)	1.82/0.64	3.0/1.0	3.06/1.15	6.2/2.1
B(T)	7.5	3.9	2.68	5.3
I <sub>p</sub> (MA)	6.5	3.9	5.5	15
P <sub>ICRH</sub> (MW)	30	12	0	20
P <sub>NNBI</sub> (MW)	0 (10) <sup>2</sup>	0	10	40
P <sub>PNBI</sub> (MW)	0	25	24	0
P <sub>ECRH</sub> (MW)	4	0	7	20
P <sub>LH</sub> (MW)	6	3	0	20?
P/R(MW/m)	22	13	14	24*
t <sub>R</sub> (s)	5–6	7–8	?	80–90
t <sub>flat-top</sub> (s)	13	10	100	400

(\*Accounting for alpha fusion power)

The machine will be initially equipped with ICRH (30MW), ECRH (4MW) and LHCD (6MW) power<sup>3</sup>. Although such a configuration is adequate to investigate the physics issues relevant for the FAST mission, the machine is designed in such a way that, if necessary, further upgrades in auxiliary heating systems (in particular Neutral Beam injector) could be accommodated.

## 2.2 - Plasma parameters and equilibrium configurations

The ITER design presently foresees the investigation of three main equilibrium configurations: a) a standard H-mode at I=15MA with broad pressure profile ( $p_o/\langle p \rangle = 2$ ); b) a hybrid mode at I=11MA with narrower pressure profile ( $p_o/\langle p \rangle = 3$ ); c) an AT scenario at I=9 MA with peaked pressure profile ( $p_o/\langle p \rangle = 4$ ). FAST has been designed with the scope of investigating burning plasma relevant dynamics, associated with the presence of energetic ions and dominant electron heating, as well as advanced plasma operation regimes. For this reason, FAST equilibrium configurations have been designed in order to reproduce those of ITER with scaled plasma current suitable to fulfill the plasma conditions described in paragraph 2 for studying the fast particles dynamics. An overview of some of the possible achievable configurations is given in Table 2. It has to be noted that the extreme scenario at I<sub>p</sub>=8.5 MA ( $Q_{eq} \sim 3$ ;  $q_{95} \sim 2.6$ ) can be achieved, assuming that 10MW of NNBI are installed.

In all cases the configuration has been designed to have always the same geometrical plasma features (see Figure 1): R=1.82 m, a=0.64 m, k=1.7,  $\langle \delta \rangle = 0.4$ . The discharge duration is limited by the heating of the toroidal field coils that are cooled only between discharges.

<sup>2</sup> FAST is designed with the capability to accommodate two NNBI power sources. However, these are not foreseen in the first stage of machine operation.

<sup>3</sup> The power levels are those coupled with the plasma.



Table 2 - FAST Plasma Parameters (the values obtained by seeding Argon impurities into the divertor are given within brackets)

	H-mode Ref-a	H-mode Ref-b	H-mode “Perf”	H-mode Extreme	Hybrid	AT	AT2
$I_p$ (MA)/ $q_{95}$	6.5/3	6.5/3	7.5/2.8	8/2.6	5/4	3/5	3/3
$B_T$ (T)	7.5	7.5	8.0	8.5	7.5	6	3.5
$H_{98}$	1	1	1	1	1.3	1.5	1.5
$\langle n_{20} \rangle$ ( $m^{-3}$ )	2	4	4	5	3	1.3	1.3
$n/n_{GW}$	0.4	0.8	0.7	0.8	0.8	0.5	.5
$P_{th\_H}$ (MW)	14-18	20-28	22-29	22-35	18-23	8.5-12	8.5-12
$\beta_N$	1.3	1.7	1.6	1.8	2.0	2.0	3.1
$t_{flat-top}$ (s)	13	13	6	~2	15	60	160
$\tau_{res}$ (s)	5.5	3	4	5	2.8	2.5	4
$\tau_E$ (s)	0.43	0.57	0.67	0.68	0.52	0.25	0.25
$T_0$ (keV)	13.0	8.5	9.0	9.0	8.5	15	10
$T_{plate}$ (eV)*	72 (60)	32 (9)	32 (9)	32 (9)	40 (15)	19	19
$f_{rad}$ (%)*	27 (39)	18 (75)	18 (75)	18 (75)	20 (55)	65	65
$Z_{eff}$ *	1.06 (1.55)	1.0 (1.2)	1.0 (1.2)	1.5 (-)	1.0 (1.3)	1.35	1.35
Q	0.65	1.2	1.8	3	0.9	0.18	0.18
$t_{discharge}$ (s)	20	20	14	~10	20	70	170
$I_{NI}/I_p$ (%)	15	20	18	15	30	60	80
P/R	22	22	22	22	22	22	22

\*values with impurity seeding are given in bracket

All the plasma equilibria satisfy the following constraints: a) a minimum distance of  $3\lambda_E$  between plasma and first wall to avoid interaction between plasma and main chamber (here,  $\lambda_E$  is the energy e-folding length assumed to be 1cm on the equatorial plane); b) a current density in the poloidal field coils around 30 MA/m<sup>2</sup>. Within these constraints enough flexibility is preserved to allow for different plasma shapes, efficient pumping and, in case of need, strike point sweeping. The location of the poloidal field coils has been optimized in order to minimize the magnetic energy, to produce enough magnetic flux (up to 35 Wb stored) for the formation and sustainment of each scenario and to produce quite a good field null at the plasma break-down ( $B_p/B_T < 2 \times 10^{-4}$  for a “commissioning” toroidal field  $B_T = 4T$ ).

The heating power is assumed in all cases to be 30 MW provided by the ICRH system; however, for the long pulse AT scenario, 6 MW of Lower Hybrid power have been included to have the capability to actively control the current profile, whereas 4 MW of Electron Cyclotron Resonant Heating will provide enough power for MHD control.

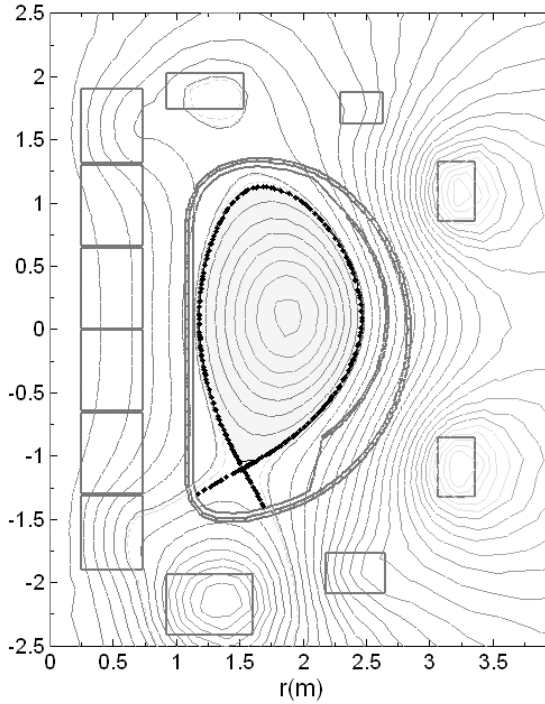


Figure 1 - H-mode equilibrium configuration.

In the “Ref-a” scenario the plasma density is such as to fit  $\rho_{\text{fast}}^* \sim \rho_{\text{fast,ITER}}^*$  and to have a plasma temperature of the order of 10 keV. “Ref-b” scenario is similar to the previous one, but with the density close to the Greenwald limit. “Perf” scenario is aimed to performance optimization.

Moreover, in these scenarios the plasma current has been fixed to have a  $q_{95} \sim 3$  with a discharge flattop  $t_{\text{flat-top}} \approx 2 \div 4.5 \tau_{\text{res}}$ .

The hybrid scenario would allow to reach an equivalent Q of about 1 considering an enhanced confinement factor of  $1.3 \cdot H_{98}$ . The  $\beta_N$  is 2 and  $n/n_{\text{GW}}$  is 0.8.

In the AT scenario the non-inductive/inductive current ratio is 60% and the pulse length is 25 times the resistive time.

The ITER98(y,2) scaling for energy confinement time has been assumed with an enhancement factor  $H_{98}$  shown in Table 2 for the three scenarios.

The access to H-mode conditions has been checked in all the cases on the basis of the most recent ITER scaling laws. The range of variation of the threshold power for H-mode access is also shown in Table 2.

The radiated power fraction and the effective charge  $Z_{\text{eff}}$  are derived from models that allow for the coupling strength between the core and edge plasma. More details are given below in Sec. 2.4. The divertor plate material is assumed tungsten and impurity seeding with Ne or Ar is considered as an option to mitigate the thermal loads. The Ar case is shown in brackets in Table 2.

Plasma position and shape control have also been studied for the reference scenario. The optimization of the copper shell position slows the vertical stability growth time down to 100ms with a comfortable stability margin. No 3D effects associated with the shell and vessel structure have been considered so far. The response of the system to a 1cm Vertical Displacement Event, a minor disruption and a step of 100 kA 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 3 cm with a settling time less than 2s.

### 2.3 - Physics of FAST plasmas

The plasma parameters obtained above have been fully validated in order to determine the amount of absorbed power from the auxiliary heating systems, the fast particle population generated by the auxiliary heating systems and the amount of plasma current driven in a non-inductive way. In FAST, three auxiliary heating methods are foreseen:

- The fast ion population is mainly produced by the acceleration of a minority species by ICRH. Since the minimum FAST magnetic field is  $B=7.5$  T in H-mode scenarios, which are those conceived for fast ion physics studies,  $^3\text{He}$  is used as a reference minority species as it maximizes the power available at sources in the frequency range 70-80 MHz. The power peak of 30 MW at the plasma is considered.
- The control of MHD activity in the AT scenario is made by electron cyclotron resonant heating and current drive (ECRH/ECCD) at 170GHz. The ECRH power is also usable for current profile control and electron heating. The cold resonance location at 6T allows the steering of the power over the entire minor radius for the NTM stabilization at the proper q surface. The amount of power foreseen is 4 MW.
- The generation of plasma current for the AT scenario is made by lower-hybrid current drive at 3.7 GHz. The amount of power foreseen is 6 MW that would guarantee the access, control and sustainment of the required current profiles for AT regimes. An ITER relevant Passive-Active Multijunction (PAM) antenna will be used to couple the power with severe plasma edge conditions (H-mode).

A fourth auxiliary heating system, based on negative neutral beam injection, could be added in a second phase for the production of a nearly-tangential energetic ion population with respect to the equilibrium magnetic field, complementing in this way the physics that can be studied with ICRH produced fast ions with velocity mostly perpendicular to the equilibrium magnetic field.

Ion cyclotron absorption has been estimated using the FELICE and TORIC codes. Both codes solve the integro-differential equation for wave propagation and absorption: FELICE solves the equation in slab geometry using the self-consistent electric field radiated by the antenna; TORIC solves the equation in toroidal geometry employing a spectral method.

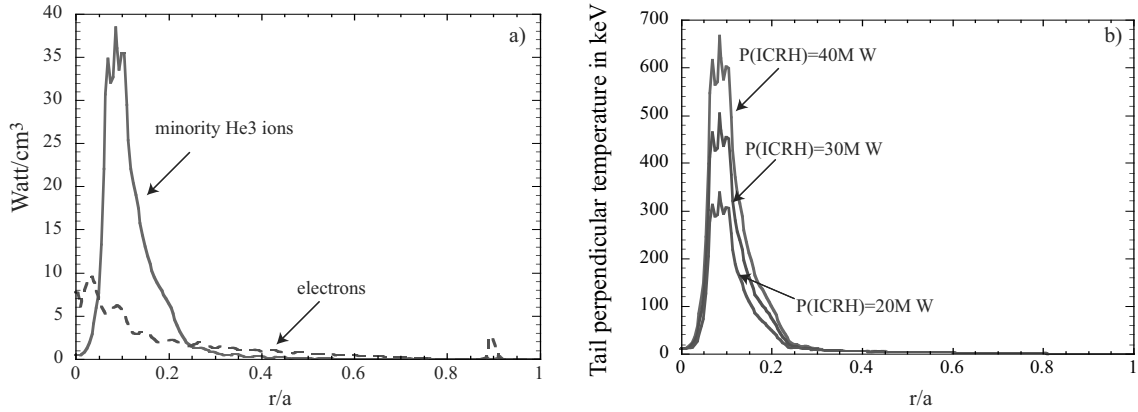


Figure 2a,b - ICRH power deposition profile for the various absorption mechanisms, and for 30 MW of launched power (left) and effective perpendicular temperature of the tail (right) for 20, 30 and 40 MW of launched power. The dominant mechanism is minority absorption (red curve), which produces localized heating in the plasma centre, similar to the alpha particles heating in ITER. The fast ion energy is in the range of 0.5MeV, with  $n_e = 2 \cdot 10^{20} \text{ m}^{-3}$  and 30 MW heating.

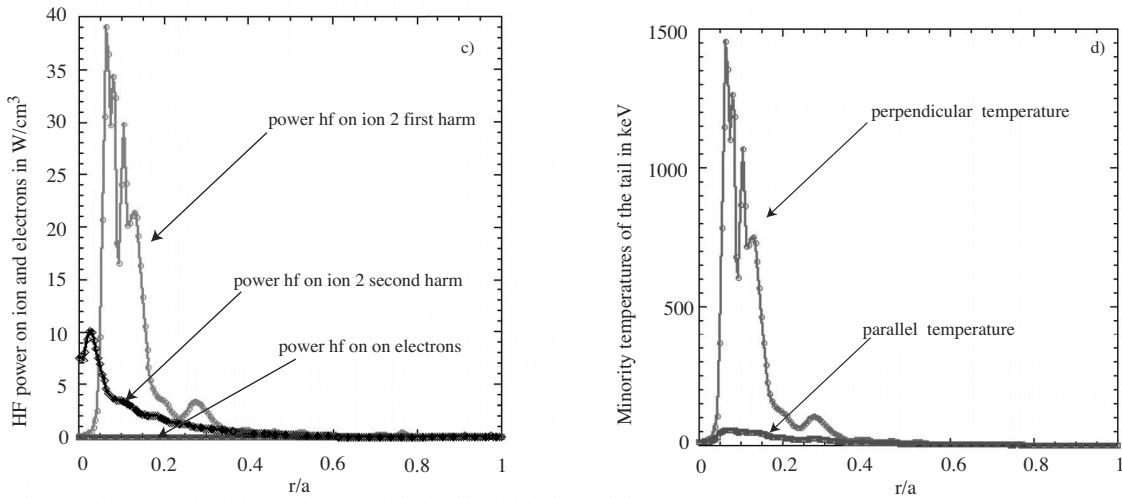


Figure 2c,d - Linear calculation (full wave code TORIC) shows very effective heating of 3He minority near the plasma centre. Only a small power fraction goes to electrons with a broad profile. Effective perpendicular ions tail temperature well above the critical temperature ( $\sim 1\text{MeV}$  with PRF=30MW) from quasi-linear calculations (SSQLFP code coupled to TORIC), ensuring effective collisional electron heating.

Three absorption regimes can possibly play a role: minority absorption (which is the absorption channel to be maximized), electron Landau damping and mode conversion to the ion Bernstein wave. The analysis by the FELICE code tends to give lower minority absorption than that from TORIC. A minority concentration of 2% yields minority absorption of 50% (FELICE) or 70% (TORIC) with a two-strap antenna with  $180^\circ$  relative phase, the remaining power being directly absorbed by the electrons. The power deposition profiles are shown in Figure 2a for the various absorption mechanisms and are related with the H-mode scenario at high density.

To establish how the minority heating produces a tail in the minority distribution function and the effective temperature of the tail, a 2D Fokker-Planck code (SSQLFP) has been run by keeping in input the linear power deposition profile coming from TORIC as input. The ion

tails produced at a power level of 30MW have parameters in agreement with the Stix 1-D theory with the local absorbed power density obtained by the deposition code. In Fig. 2b the perpendicular effective temperature is shown vs. the normalized radius for three different levels of launched power (20, 30, 40 MW). Note that, although the peak power at the plasma is 30 MW, higher power input cases have been analysed to investigate the scaling properties of relevant quantities vs. the values predicted by the Stix 1-D analytical theory. It is possible to notice that the temperature spans from 350 keV (20 MW) to 700 keV (40 MW). The fast particle concentration is about 0.6%. The fast particle population and the associated tail temperature for a coupled power of 30 MW are in the correct range of parameters to simulate the ITER fast particle dynamics. A case with a lower density and higher bulk electron temperature (Ref-a scenario), has shown that the perpendicular temperature of the produced minority ions tail scale correctly with the slowing down time. For the case of 30 MW of injected power, the perpendicular temperature reaches values above 1.0 MeV (Fig. 2c,d).

A preliminary analysis of the global MHD stability for the long pulse AT scenarios has been performed using the MARS code in order to investigate the possibility of stabilizing Resistive Wall Modes (RWM). The no-wall beta limit, for a 6 MA plasma at 6.7T, corresponds to  $\beta_{Nc}=2.8$ , whereas an ideal wall at  $r/a=1.3$  has a beta limit corresponding to  $\beta_{Nc}=3.24$ . 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.

Besides the stabilization of Neoclassical Tearing Modes (NTM) in long pulse AT scenarios, the ECRH system on FAST is used also for electron heating and current drive tasks at densities below  $3.6 \times 10^{20} \text{ m}^{-3}$  which is the cut-off for the ordinary mode for the chosen frequency of 170 GHz (the same as ITER).

As an example of NTM stabilization, the  $m/n=2/1$  island evolution is calculated for the 6T, 3MA long pulse AT scenario and  $\beta_N = 2.1$ . The considered wave is launched from the equatorial port at  $18^\circ$  toroidal angle.

The wave propagation is evaluated with the ECWGB ray-tracing code. The  $m/n=2/1$  island evolution, calculated by the modified Rutherford equation, is shown in Figure 3. With 2 MW  $P_{EC}$  injection the island width is stabilized below 3 cm (about 34% of its saturation value), being the EC current density radial size = 5.5 cm and the ratio  $J_{cd}/J_{bs}=1.15$ . Small increase of EC input power (less than 0.2 MW) leads to a full island stabilization.

The foreseen ECRH power ( $\sim 4$  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, larger than 50% of the  $J_{OH}$  one, can modify the  $q$  profile.

Lower hybrid current drive can be used on FAST to control the current density profile thus allowing the generation and sustainment of AT scenarios in a range of plasma densities

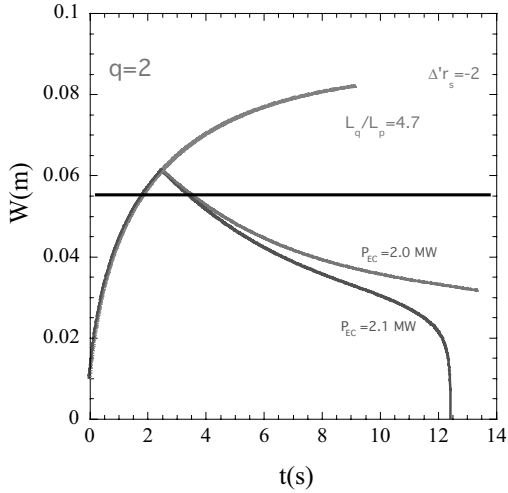


Figure 3 - Evolution of the 2/1 island without (red) and with (green/blue) EC power applied.

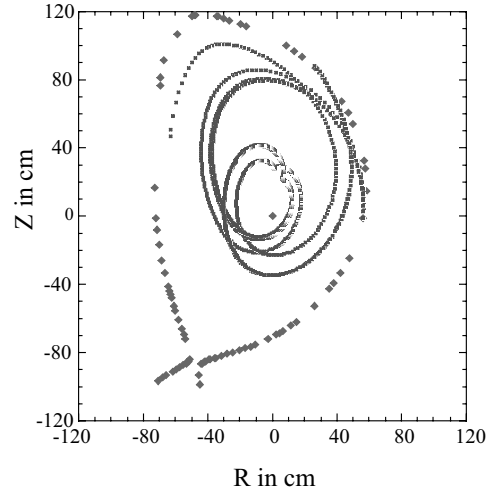


Figure 4 - Lower Hybrid waves trajectories for three values of the parallel refractive index

similar to those envisaged for ITER in scenario 4. A study of the LH penetration and absorption has been performed in a parameter range typical of FAST scenarios. Figure 4 shows the ray trajectories for plasma equilibrium representative of FAST scenarios with an average density  $1 \cdot 10^{20} \text{ m}^{-3}$  and a central temperature of 13 keV. The wave frequency is 3.7 GHz and three values of the parallel refractive index are considered. In all the cases the wave is absorbed around mid radius as requested for this kind of scenario. Current drive efficiencies in the range  $0.25 \cdot 10^{20} \text{ Am}^2/\text{W}$  are predicted. This produces in the reference AT scenario a driven current of  $I_{\text{LHCD}}=0.65 \text{ MA}$ , corresponding to 22% of the total, the remaining 38% ( $I_{\text{NI}}/I_{\text{p}}=60\%$ , see table 2) being driven by the bootstrap current,  $I_{\text{boot}}$ . According to simulations this is enough to produce a negligible evolution of the current profile during the whole discharge. Higher  $I_{\text{LHCD}}$  fraction is possible at lower densities, but at the expense of  $I_{\text{boot}}$ , so that the total non-inductive current remains close to 60%.

## 2.4 - Power Handling in FAST

The crucial aspects of the thermal loads on the divertor plates and of the core plasma purity for the proposed scenarios are governed by the complex relationships binding the core and the plasma edge. To have reliable predictions a series of calculations has been performed in the framework of models capable of describing self-consistently the plasma core and edge in a tokamak device. A two steps route of increasing accuracy and then calculation complexity has been followed. A simple model based on 0D description of the plasma parameters in the core and on a two-point model for the plasma scrape-off layer (SOL) has been used to investigate the wide range of plasma parameters in the FAST tokamak. Then, a more complex code, named COREDIV (1D in the bulk and 2D in the SOL), has been used when some specific questions dealing with plasma performance and/or heat removal from the divertor needed to be addressed in more detail, or when the core-edge coupling has shown so strong to make the simple model predictions poorly reliable. Both models have been validated in the past on experimental data (namely JET, FTU, Textor).

Several numerical self-consistent simulations have been made for the H-mode and AT scenarios. First, the calculations have been carried out for a full W machine, then the option of injecting Ar and Ne to mitigate the thermal loads has been considered and finally the case, relevant for DEMO, with liquid lithium as divertor target has been analysed. All the results are summarized in Table 3.

Table 3 - Self-consistent simulations

<b>Plasma Facing Material</b>	<b>A) - Tungsten only</b>		<b>B) - W + Argon seeding</b>	<b>C) - Liquid lithium + Neon</b>		
Self-consistent model	Simple model	COREDIV	Simple model	COREDIV		
Scenario	H-mode		H-mode		H-mode Ref-b	
	Ref-a	Ref-b	AT	Ref-a		Ref-b
$\bar{n}_e = [10^{20} \text{ m}^{-3}]$	2	4	1.3	2	4	4
$P_{\text{aux}}$ [MW]	30	30	30	30	30	25
$f_{\text{rad}}$ [%]	27	18	63	39	75	73
$P_{\text{div}}$ [MW]	22.3	25.4	11	18.7	8	8
$P_{\text{SOL}}$ [MW]	0.7	1.7	2.1	0.8	2.3	10
$Z_{\text{eff}}$	1.06	1	1.34	1.55	1.2	1.24
$T_0$ [keV]	13	9.1	10.8	13.2	8.7	8.1
$T_{\text{plate}}$ [eV]	72	32	18.9	59.6	9	5.2
$T_{\text{sep}}$ [eV]	102	103	201	99.5	87	150
$N_{\text{sep}}$ [ $10^{19} \text{ m}^{-3}$ ]	0.67	1.33	0.75	0.66	1.33	1.47
$N_{\text{plate}}$ [ $10^{20} \text{ m}^{-3}$ ]	0.47	2.2	3.1	0.55	6.4	10
$\Gamma_{\text{sputt}}$ [ $10^{21} \text{ s}^{-1}$ ]	0.04	0	0.01	0.009	0	11

The simple model is enough accurate at  $n_e=4 \cdot 10^{20} \text{ m}^{-3}$  where the SOL-core coupling remains quite weak because of the low level of W sputtering and hence of its influx. This is due the SOL temperature ( $T_{\text{eSOL}}$ ) that is kept below or close to the W sputtering threshold by the high SOL density ( $n_{\text{eSOL}}$ ). As a consequence, the radiation loss is low (18%), and a large fraction of the heating power is delivered to the divertor plates, where  $T_{\text{e,plate}} > 30 \text{ eV}$ , much higher than that required for plasma detachment. The heat load can exceed the tested safety limit ( $18 \text{ MW/m}^2$ ) for monoblock tungsten tiles in the worst case when a well-closed magnetic configuration is assumed. In this case, the benefit of flux expansion is strongly reduced. The reduction of the effective divertor surface, due to tile shaping needed to avoid edge overheating, is also considered. In the divertor heat load analysis the heat diffusion in the private region has been neglected and a wide range of magnetic field line inclination angles has been considered. It is to be noted that, by simply allowing for only the most probable value of this angle, safe heat load is still retained.

However, in the case of high density H-mode ( $\bar{n}_e = 4 \cdot 10^{20} \text{ m}^{-3}$ ) it is relatively easy with additional impurity to radiate a large fraction of the input power, up to 75%, and then to operate safely with the divertor with a maximum thermal load of 5-7  $\text{MW/m}^2$ , while keeping simultaneously low  $Z_{\text{eff}}$ . Here and later the two power load values correspond to the minimum and maximum values of the assumed magnetic field line inclination angle ( $1^\circ$  and  $3^\circ$ ). The situation is similar for Ar or Ne radiators but the effective charge is lower with Ar injection. On the other side Ne radiates more in the SOL and consequently Ne seeding does

not affect the H-mode power threshold. When  $n_e$  decreases, the increase of  $T_{eSOL}$ , associated with the drop of  $n_{eSOL}$ , raises the sputtering yield of tungsten and hence its concentration in the main plasma. The weight of the W radiative properties grows and becomes dominant for  $n_{eSOL} < 0.5 \cdot 10^{20} \text{ m}^{-3}$  (strong core-SOL coupling), according to simulations. Impurities such as Ar or Ne, with much lower radiative loss rate, slightly affect the overall radiation, if we want maintain their concentration low enough to preserve a high plasma purity. We fixed the acceptable limit to  $Z_{eff} < 1.7$ , that corresponds to 0.08% of Ar atomic concentration. The seeded impurity properties, instead, prevail for  $n_{eSOL} > 0.8 \cdot 10^{20} \text{ m}^{-3}$  (low core-SOL coupling).

The Ref-a case ( $\bar{n}_e = 2 \cdot 10^{20} \text{ m}^{-3}$ ) is in between the two regions. Here, seeding Ar at 0.08% still helps in pushing the heat load on divertors well below the safety limit of  $18 \text{ MW/m}^2$  (maximum =  $11\text{-}17 \text{ MW/m}^2$ ), by increasing the radiated power fraction from about 25% to 40%. The drop of the Ar contribution comes primarily from the lower density, being  $P_{rad} \propto n_e \cdot n_{Ar} \propto n_e^2$  for fixed Ar concentration and is partly compensated by radiation from W atoms.

In the AT scenario, instead, the even lower densities ( $\bar{n}_e \leq 1.3 \cdot 10^{20} \text{ m}^{-3}$ ) raise the W radiation losses high enough to ensure a safe operation for the divertor plates (maximum heat load =  $6\text{-}9 \text{ MW/m}^2$ ), without the assistance of any impurity. COREDIV, which is more reliable than the simple model in this rather strong core-SOL coupling, predicts  $f_{rad} \approx 65\%$  with  $Z_{eff} \approx 1.35$ , against the simple model estimate of  $f_{rad} \approx 42\%$  and  $Z_{eff} \approx 1.2$  (maximum heat load =  $10\text{-}17 \text{ MW/m}^2$ ). This fraction can further increase at lower densities. As pointed out above, the gain in  $f_{rad}$  with Ar at the maximum allowed concentration is quite poor.

However, we must note that an anomalous impurity pinch velocity, not included in the model but possible when internal transport barriers develop, could lead to less optimistic predictions. Ar or Ne seeding remains in this case a valuable option to reduce the W influx, by lowering the self-sputtering processes (table 3).

Finally, simulations with COREDIV code have been made for dominant Li impurity with added noble gas, since FAST is foreseen to test also technical solutions for divertors based on liquid metal PFM. The very low radiative capability of Li implies to seed an impurity in all scenarios to keep the heat load at an acceptable level. The impurity properties always govern the plasma radiation. The results for the high density H-mode case, with Ne seeding at 0.08% of atomic concentration, are reported in the rightmost column of table 3. Li evaporation is neglected in the calculations since it is assumed that the Li modules are actively cooled and maintained at a fixed temperature. The differences with the W+Ar case (second column of table 3) are quite small for the core parameters and are mainly due to the different gas used. This confirms that the target plate material has no influence on the discharge parameters at high density, and that the simple model also well describes the bulk plasma for weak SOL-core coupling. The more accurate SOL description, on the other side, shows that plasma detachment from the divertor plates can be closely approached in this case. At the lower density scenarios the radiation fraction is lower than with the tungsten divertor since no



contribution is given by W atoms, and is in the range of 30%, corresponding to a maximum heat load of 12-19 MW/m<sup>2</sup>. Even though this is well below the limit recently achieved in several test samples (25 MW/m<sup>2</sup>), the response of the liquid divertor to such a high load is one of the objectives of the experiment.

## 2.5 - ELMs Analysis

ELMs are one of the major “concerns” intrinsically connected with the “standard” H Mode scenario. Since the FAST reference scenarios rely on a good quality (H<sub>98</sub>=1) H mode it is natural to imagine that also on FAST there will be a noticeable ELMs activity. Consequently, it is quite important to figure out what type of Elms have to be expected, verify their compatibility with machine PFCs and optimize Plasma Operations to allow as much flexibility as possible to study the ELMs behaviour in a reactor relevant condition.

All the following assumptions have been taken from the present available literature.

In a good quality H Mode the Energy confined in the Plasma Pedestal is proportional to the total Plasma Energy ( $W_{\text{PED}} \sim 40\% W_{\text{TOT}}$ ). The energy released by any single ELM is proportional to the Pedestal energy and the proportionality coefficient depends on the Pedestal Collisionality ( $W_{\text{ELM}}/W_{\text{PED}} = F(v^*_{\text{PED}})$ ). In particular for  $v^*_{\text{PED}} \approx 0.1 \rightarrow W_{\text{ELM}}/W_{\text{PED}} \approx 0.15$ . From the analysis of present data-base, it can be assumed that the power released by the ELMs is of the order  $P_{\text{ELM}} \approx 50\% P_{\text{INP}}$ . Furthermore,  $P_{\text{ELM}} = f_{\text{ELM}} W_{\text{ELM}}$ , where  $f_{\text{ELM}}$  is the ELMs frequency.

Using these assumptions for the scenario at  $I = 7.5$  MA in table 2, but considering lower density of  $n/n_{\text{GW}} \approx 0.3$ , meaning low edge collisionality ( $v^*_{\text{PED}} \approx 0.1$ ), the ELMs energy and frequency are respectively:  $W_{\text{ELM}} \approx 1.5$  MJ and  $f_{\text{ELM}} \approx 10$  Hz.

For a preliminary assessment of divertor heat load by this ELMs energy, considering that for  $W_{\text{ELM}}/W_{\text{PED}} > 0.1$  only about half of  $W_{\text{ELM}}$  reaches the divertor and the fraction of this energy mostly contributing to material damage, i.e. the one deposited in short timescales, is about 40% for low collisionality. By assuming the same spatial deposition profile as inter-ELM and a factor 2 asymmetry in the in-out ELMs energy deposition, the energy density on the inner divertor is expected to be about  $0.4 \text{ MJm}^{-2}$ , to be compared with the recommended threshold for damage ( $0.5 \text{ MJm}^{-2}$ ), adopted by ITER for avoiding too strong W erosion.

It has to be noticed that this low collisionality regime is achieved in a condition of relatively high density ( $n_{\text{EDGE}} \approx 10^{20} \text{ m}^{-3}$ ) comparable with that of ITER, as it is the case for the edge temperature too. Consequently, this experiment will actually have the ITER edge conditions and will allow studying and optimizing the edge conditions in order to minimize the ELMs perturbation.

The large range of achievable densities will also allow operating at higher collisionality with much lower ELMs amplitude. Moreover, the machine will have the possibility to use different divertor materials; in particular it envisages the use of the liquid lithium that could

strongly affect the pedestal parameters. It is foreseen to insert in the machine a variable  $n$  ( $1 \Rightarrow 3$ ) active coils system to control the ELMs amplitude by the edge ergodization.

### 3. PRELIMINARY DESIGN DESCRIPTION

#### 3.1 - Load assembly

The FAST load assembly (Figures 5a,b) consists of the vacuum vessel and its internal components (First Wall, Divertor, passive stabiliser structure), the magnet system and the poloidal field coils, which are adiabatically heated during the plasma pulse. To allow for a

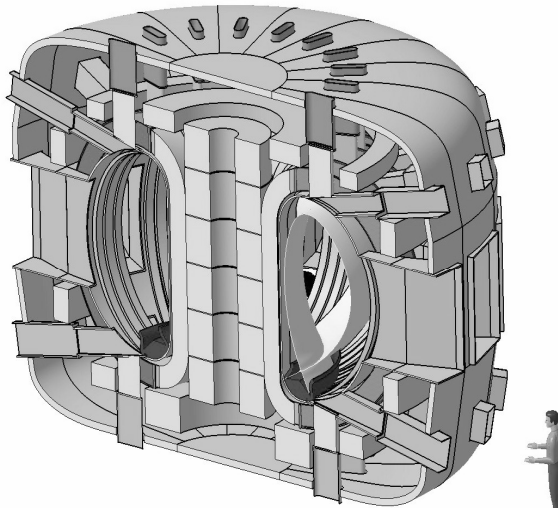


Figure 5a - FAST Load Assembly axonometric view

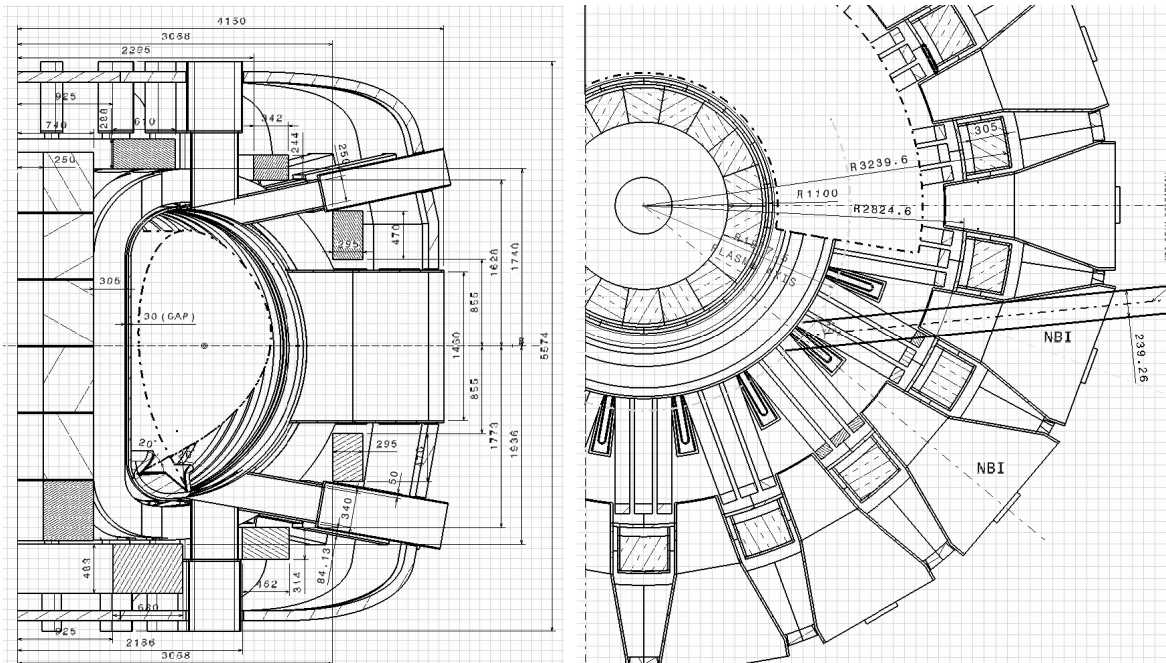


Figure 5b - FAST- Vertical Plant and Views

pulse duration up to 75s for the AT operation (i.e. 6T; 3MA), helium gas at 30K has been chosen for cooling the Oxygen Free copper toroidal and poloidal magnets. In fact at the temperature of 30 K the ratio of the copper resistivity to specific heat is minimized. After a current pulse corresponding to the long pulse AT scenarios, the toroidal coil temperature will reach about 190 K in the leg region and an average of 60 K in the remaining part. The Poloidal Coil System reaches 60 K as a maximum. The cooling of the magnet system is guaranteed by a global helium gas flowing of about 4 kg/s through suitable channels carved in the coil turns obtained from a cryo-cooler. Each turn is fed by He independently from each other (the He flows in parallel in each turn) in order to limit the pressure drop.

### *3.1.1 - Toroidal magnet*

The magnet consists of 18 coils, each of them made of 14 copper plates suitably worked out in order to realise 3 turns in radial direction. The 42 turns of each coil are welded corresponding to the most external region in order to obtain a continuous helix. The maximum turn thickness is 30 mm. The plates are tapered at the innermost region to realise the needed wedged shape; the minimum turn thickness is about 15 mm. The magnet insulation is made of glass-fabric epoxy both for ground and inter-turn insulation.

To limit within acceptable values the TF magnet ripple, which could lead to significant losses of high-energy particles, ferritic inserts with a saturation field  $B_s = 2.2T$  have been introduced. The ferromagnetic inserts are located inside the outboard area of the vacuum vessel, corresponding to the TF coils area. The ripple on the plasma separatrix (near the equatorial port), has been reduced from 2% to 0.3% with optimized Fe inserts.

The coils are brought together by a steel structure, which surrounds them. Two pre-compressed rings situated in the upper-lower zone keep the whole toroidal magnet structure in wedged configuration. The structure is also used to position the poloidal coils, which surround the toroidal magnet, and to fix the vacuum vessel supports. The cooling of the structure of the toroidal magnet is obtained by the contact with the actively cooled components. To enable the operation at 30 K, the whole machine is kept under vacuum by a metallic cryostat.

The magnet dimensions have been determined to limit the coil temperature at the end of the longer pulses. It turns out that from the structural standpoint the magnet section is adequate to self sustain the forces. The first rough evaluation of the stresses indicates the maximum Von Mises to be below 250 MPa.

The fabrication process is based on well assessed technology utilised for FTU and other prototypical components. Therefore no further R&D is required for the construction of the toroidal magnet.

### 3.1.2 - Central Solenoid and Poloidal Coils

The FAST free-standing Central Solenoid (CS) is segmented in 6 coils to allow for plasma shaping flexibility, to ease the manufacture and to allow for cooling. The poloidal field coils and bus bars are made of Copper hollow conductors. They have to withstand both the vertical and radial electromagnetic loads and are free to expand radially.

The CS coils are layer wound and have an even number of layers in order to have the electrical leads located at the same side of the coil. The conductors are wrapped with glass fabric and kapton tapes and vacuum impregnated with epoxy resin. Radial grooved plates at the interfaces between coil segments maintain concentricity.

The poloidal field coils consist of concentric layers, which, as far as their cooling is concerned, are connected in series. The final temperature is never expected to exceed 60 K in any poloidal coil.

### 3.1.3 - Vacuum Vessel

The vacuum vessel is segmented by 20 degree modules. In order to minimise the vacuum vessel time constant, the shell is made of Inconel and the ports of stainless steel. The maximum thickness of the shell is 30 mm while the ports are 20 mm thick. The shell is manufactured by hot forming and welding.

Following the previous experience made on FTU, the vacuum vessel will be supported by the toroidal field magnet system by means of vertical brackets attached to the TF coil case through the vessel equatorial port. According to this constrain scheme; thermal expansion/contraction of the vessel is allowed, while non-symmetric displacements that might appear during disruption or plasma VDE are restrained.

Each vacuum vessel sector is equipped with 5 access ports. The maximum forces during plasma disruptions is about 550 t for a 6.5 MA H-mode operating scenario. The thickness of the wall is adequate to sustain such a load. The vessel time constant is about 25 ms.

The operating temperature of the vessel ranges from room temperature to 100 °C. A suitable water loop is dedicated to regulate the vessel temperatures.

The first-wall and the divertor are actively cooled by pressurized water with velocity respectively 5 and 10 m/s. These components have been designed to exhaust up to a maximum heat power of 50MW during long pulse AT operation.

The vacuum vessel is equipped with a substantial number of ports: 18 equatorial, 36 (18 upper and 18 lower) horizontal and 36 (18 upper and 18 lower) vertical.

### 3.1.4 - Divertor and First Wall

The first-wall surrounds most of the vessel wall. It consists of a bundle of tubes armoured with 3 mm plasma spray tungsten. The heat load impinging the first-wall is, on average, 1 MW/m<sup>2</sup> with a peak of about 3 MW/m<sup>2</sup>. The adopted solution is well suited to resist to these loads, having been tested up to 7MW/m<sup>2</sup>. The first-wall is also adequate to work as a limiter during the start up of the plasma. Its temperature will be kept around 100°C in order to avoid impurities adsorption. The design has to be remote-handling compatible. The first wall maintenance will be made from equatorial and upper ports.

The high power flux in the divertor, as illustrated in the previous section, makes it suitable as target plates only monoblock W tiles, constructed according a recently developed technique. Extensive tests on these tiles have shown that they can withstand without damage a heat load up to 18 MW/m<sup>2</sup> continuously, provided they are actively cooled. Tools to not exceed this limit have been discussed in the previous section.

The armour consists of hollow tungsten tiles inserted in a heat sink copper tube. The heat flux component will be supported by a steel frame, which acts also as cooling circuit. To enhance the critical heat flux, swirl tapes are provided in the most loaded zone. The configuration has to allow easy maintenance operation being likely the possibility to substitute some components. The scheme is like the ITER one with the frame acting as a carousel all around the machine. The maintenance will be done from the lower port.

### 3.1.5 - Neutronics

The neutron emissivity sources (n/m<sup>3</sup>s) and the neutron rates (n/s) foreseen for the H-mode and the AT scenarios have been calculated considering, for each scenario, the equilibrium configuration, the ion temperature and ion density profiles and a maxwellian DD reactivity. The code used (MSST, Measurement Simulation Software Tool), tested on FTU, has shown an agreement with the measured neutron rate within 10%. The calculated neutron rates for FAST are shown in Table 4.

The energy and spatial distribution of the neutron flux and the nuclear heating (neutron and gamma energy deposition) have been evaluated in several positions inside and outside the machine with the MCNP5 Monte Carlo code, using as input the MSST neutron emissivity source and a detailed geometrical model (20-degree machine sector bounded by reflecting planes, based on FAST CAD drawings). In the present model all ports are assumed empty. The flux in the "Perf" H-mode at the equatorial ports is  $\sim 1.4 \times 10^{12}$  n/cm<sup>2</sup>s at 2.9 m from the center of the machine and  $\sim 5 \times 10^{10}$  n/cm<sup>2</sup>s at  $\sim 5$  m (outside the cryostat). The spectra are dominated by the lower energy scattered neutrons: the contribution of the uncollided component ( $E_n > 1.83$  MeV) to the total flux is  $\sim 30\%$  in the port and few % in TF coils. The average nuclear heating decreases from  $\sim 40$  mW/cm<sup>3</sup> on the first wall to  $\sim 6$  mW/cm<sup>3</sup> on the TFCs and it is  $< 1$  mW/cm<sup>3</sup> on the CS and PF coils.

The calculated neutron fluxes on the machine components have been used to perform the activation analysis with the FISPACT code. The radioactivity induced in the components after one year of operation has been calculated assuming an operation agenda of 800 shots/y (with medium performance shots in the last week of operation) distributed as shown in Table 4. The resulting total annual neutron production is  $\sim 1.6 \times 10^{21}$  n/y.

Table 4 - Neutron rates and typical annual agenda of shots

Scenario	Neutron Rate (n/s)	Shot Duration (s)	Number of Shots (shots/y)	Neutrons (n/y)
H- Mode Ref-a	$1.2 \times 10^{17}$	13	500	$7.8 \times 10^{20}$
H-mode Ref-b	$1.9 \times 10^{17}$	13	100	$2.5 \times 10^{20}$
H-mode "Perf"	$2.2 \times 10^{17}$	6	100	$1.3 \times 10^{20}$
AT	$0.7 \times 10^{17}$	60	100	$4.5 \times 10^{20}$
<b>Total</b>				$1.6 \times 10^{21}$

The short/medium term activation is not negligible (for example, the contact dose rate 12 days after shutdown is  $< 10$  mSv/h on the vacuum vessel, but  $\sim 170$  mSv/h on the first wall, due to tungsten activation) and therefore the following recommendations should be considered:

- remote handling is mandatory;
- special precautions or long cooling times are necessary for main repairs;
- shielding optimization should be performed to reduce dose level and to protect diagnostic systems and electronics;
- preparation of a repository is required to store the dismantled activated components.

### 3.1.6 - Remote Handling

A Remote Handling system, similar to the JET TARM, has been conceived for unplanned ex vessel (emergency) operations. Standard maintenance operations shall instead be accomplished by a plug-in design of the diagnostics and the antennae and casked solutions. The ITER Divertor maintenance procedure has been used wherever possible. The procedure is based on the development of an ad hoc cassette mover tractor capable to grasp and move the Divertor cassette. Some of the maintenance tasks of the first wall are similar to those foreseen for the Divertor. Therefore, it is conceivable to standardize pipe sizes in such a way that it will be possible to share cut and weld devices. As far as the first wall assembly and disassembly is concerned, a classical articulated boom plus a front-end manipulator has been considered.

### 3.2 - Heating systems

The FAST auxiliary heating systems have been chosen to be consistent with the present state of the art and do not require additional R&D activity. FAST is equipped with three systems: Ion Cyclotron Resonant Heating (ICRH), Electron Cyclotron Resonant Heating (ECRH) and Lower Hybrid Current Drive (LHCD).

#### 3.2.1 - ICRH heating

A description of the ICRH system is given in Table 5. At a magnetic field of 7.5 T, the use of  $^3\text{He}$  minority requires a frequency up to 80 MHz. In its initial configuration the system will couple 30 MW to the plasma.

Table 5 - ICRH system parameters

Operating frequency range ( MHz)	60-90
Coupled Peak power (MW)	30
Bandwidth (MHz)	$\pm 2\text{MHz}$ (-1db)
Pulse width (s)	up to 75
Time interval between two 75s pulses (hr)	2
Type of antenna	4 rows of 2 straps
Power coupled per antenna (MW)	5
Max radiated power density ( $\text{MW}/\text{m}^2$ )	10
N. of antennae	6
Peak Power per generator (MW)	2
N. of RF generators	24

A possible design of the ICRH antennae could be based on an array of eight (2 toroidal by 4 poloidal) current straps protected by a Faraday Shield made by a set of 30 non tilted elements, having a smooth rectangular cross section. The Faraday Shield (FS) purpose is to suppress the components of the emitted radiation parallel to the local B-field, and to shield the electrically active components from a direct contact with the plasma. All the antenna components (straps and Faraday Shield rods) are water-cooled.

Each antenna could be fed by 4 high power tetrodes “TH 526”, capable of a maximum RF power output of 2 MW in the frequency range 35-80 MHz, routine operations could be foreseen reliably at 1.5 MW or at a higher frequency. Four of such generators are supplied by a 36kV/380A solid-state unit. The front part of the antenna is larger than the access port and must be installed and connected to the coaxial cables from the adjacent ports.

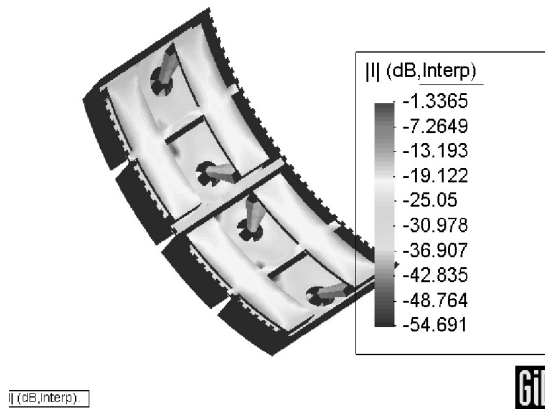


Figure 6 - Distribution of current on the straps

The performance of this antenna has been studied with the TOPICA code on the reference FAST H-mode plasma scenario at 80 MHz with 2%  $^3\text{He}$  minority. Electric current and magnetic current/electric field distribution have been obtained in vacuum and with plasma. The analysis in vacuum of the optimized antenna has shown a very good (low) inter-strap coupling. The analysis with plasma has demonstrated the good performance of the antenna array in terms of power coupled to the plasma for the standard configuration and for a maximum voltage of 45 kV along the coaxial lines. A power of 5 MW can be safely coupled to the plasma by each array at 80 MHz, with a distance between the antenna mouth and the separatrix of 4 cm. In figure 6 the current distribution on the straps is shown, demonstrating the good efficiency obtained with this geometry: the current on the straps has very high absolute value and is almost constant along the entire length of the straps.

### 3.2.2 - ECRH heating

Four identical units compose the ECRH system, each of them with a gyrotron, a transmission line and a launcher. The four launchers are located in the same port. Each gyrotron is fed by an independent power supply, capable of high frequency modulation (up to 10 kHz) and designed to be used as the actuator in the feedback loop for mode suppression. The power is delivered from the source to the launcher by means of an evacuated corrugated waveguide. The reference design for the launcher is based on front steerable mirrors, with real time control of toroidal and poloidal injection angles. No barrier window is considered in the transmission line. The whole system is designed to operate in feedback mode with real-time control of the main parameters (polarization, beam current, mirror steering, power, fault management), addressing in this way technological issues relevant for a system working in a thermonuclear plant. The considered gyrotron is a 170GHz/1MW source, with depressed collector and a pulse length larger than 100s, based on the results of the R&D activity for ITER. Each gyrotron is fed by a 55kV/50A high voltage power supply. The transmission line is an evacuated aluminum corrugated waveguide (i.d. 63.5mm) matched with a gyrotron output beam with an elliptical mirror. Since the power dissipated on the waveguide is small and the pulse length does not exceed 100s, no direct cooling of the waveguide is considered,



while all the other components (mitre-bends, polarizer, dc-break) must be cooled. The launcher under study for the ECRH system of FAST is based on a front steering concept for a major flexibility in terms of beam shaping and injection angles. It is located in a single equatorial port and designed for a vertical scan of the resonant layer. In this way, the intersection of the EC resonance with the  $q=2$  surface, for the relative NTM stabilization, is reached with limited diffraction effects, while the beam spot radius (waist) in the plasma resonant region can be less than 4 cm, able to generate a term  $J_{CD}$  comparable with the local  $J_{OH}$ . The front mirror is real time controlled at a speed compatible with all the control issues assigned to the ECRH system (NTM stabilization, saw-tooth control, disruption mitigation).

The overall estimated losses of the ECRH system (waveguide, mitre-bends, microwave components and launcher) are below 8%, value that can be further reduced with a  $HE_{11}$  to Gaussian beam converter at the beginning and at the end of the waveguide.

### 3.2.3 - Lower Hybrid heating

The LHCD system for FAST is designed to routinely couple a RF power of 6MW to FAST plasmas.

The preliminary design is based on the frequency of 3.7GHz in pulsed regime, with pulse length up to 100s. At this working frequency high power CW sources are available, i.e. the TH 2103 klystron, rated at 500kW/CW and 650 kW/10s, these klystron are the sources of the Tore Supra and JET LHCD systems and they are under further development for higher performance in CW. Table 6 summarises the main parameters of these klystrons. The FAST LHCD system will be equipped with two PAM launchers, located in two independent equatorial port. This will simultaneously allow coupling LH waves in plasma with severe edge conditions and effectively water cool the antennas in long operations and with heavy thermal loads.

The dimensioning of the launcher, given the frequency, is based on the requirement of launching a peak  $n_{//} N_{||peak} = 1.9$  and by the cross section of the FAST ports at the narrower

Table 6 - TH 2103 main parameters

Frequency	3.7 GHz
Bandwidth @ - 1 dB	10 MHz
Output power (CW)	500 kW
Gain	47 dB
Cathode voltage	60 kV
Beam current	20 A
Efficiency	42%
Modulating anode voltage	45 kV
Modulating anode current	50 mA

point. The resulting power density in the active waveguides is limited to  $P_s = 33\text{MW/m}^2$ , which is comparable with the values normally achieved in JET and in Tore Supra.

Taking into account  $\sim 20\%$  RF losses in the transmission lines and in the launcher, a minimum RF power at the generator of 8MW (equivalent to 6MW coupled to the plasma) need to be installed, i.e. 16 klystrons are necessary.

### 3.3 - Diagnostics

Several diagnostics already foreseen for ITER can be developed and exploited in FAST. In particular, the diagnostics dedicated to study the dynamics of fast particles, confined and lost, and the divertor behaviour are in a preliminary phase of development, and they need a dedicated program in order to be ready for ITER. Other systems are still under discussion because the solutions considered for the feasibility are still very primitive (erosion and dust monitors for example).

FAST is a device suitable to develop advanced diagnostics useful for ITER and candidate systems for DEMO, for the following reasons:

1. reduced costs and development time of the diagnostics: as FAST works in DD with W wall, problems and cost related to the use of tritium and beryllium are avoided;
2. flexibility of the device in terms of testing different technical solutions to optimize diagnostic performance;
3. reduced time of optimization due to fewer operational constraints, for example in the scheduling of diagnostic system commissioning.

The diagnostic systems have been conceived to provide measurements for the characterization of fast particle dynamics as well as H-mode and AT scenarios. There are five main diagnostics groups: 1) burning plasma 2) kinetic parameters and current profile 3) magnetic; 4) SOL and divertor 5) turbulence and emission of radiation. Within each group, three levels of importance can be identified:

- standard: Conventional systems which require no or marginal R&D that will be installed at  $t=0$ ;
- first priority: Diagnostics necessary to achieve the FAST mission that can be installed after few years of operation;
- R&D needed: Systems important for the mission of FAST as well as for all the burning plasma experiments, requiring additional R&D and for which FAST can represent an optimal test bed in view of ITER application.

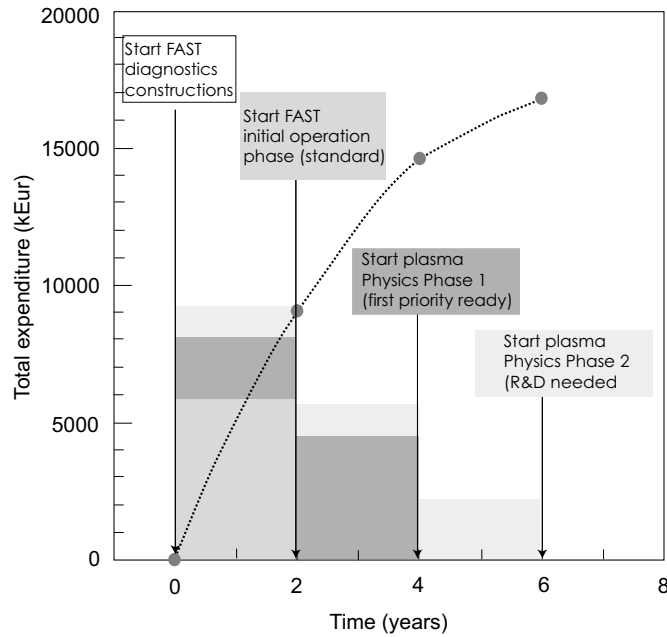


Figure 7 - Time distribution of the diagnostics hardware expenditure: total (red points) and subdivided in subgroups (color histograms). Note that the Physics Phase 2 is expected to last at least 6 years.

Group 1 (neutron monitors, neutron camera, neutron spectroscopy and fast ion diagnostics (CTS, lost ion probes and gamma camera)) is essential for the mission of FAST and enables the study of the burning plasma, fast particles and MHD related parameters. Group 2 can be used for advanced control and physics evaluation of plasmas (MSE and CXRS, ECE, Thomson scattering, CO<sub>2</sub> interferometer, bremsstrahlung, spectroscopy and polarimetry). Group 3 (magnetic sensors) is used for the evaluation of the equilibrium and measurement of MHD activity and for the real time control of the machine (safe operation and scenario realization). Group 4 (e.g.: divertor spectroscopy V and UV, main chamber reciprocating probes, infrared thermography) is dedicated to determine the kinetics of SOL, the thermal loads on the wall and divertor, the erosion and deposition and plasma fluxes. Group 5 (e.g. Langmuir probes, reflectometry, bolometry, tomographic gas puff imaging, microwave Thomson scattering) aims to have a complete picture of the temporal evolution of the electrostatic instabilities (fluctuations of plasma density) related to ITG (Ion Temperature Gradient) and possibly to ETG (Electron Temperature Gradient), with reasonable space resolution inside and at plasma edge.

The number of ports to be fully devoted to diagnostics is roughly 6 horizontal ports (including the diagnostics neutral beam access) and 6 vertical ports (two of them passing, needed for the CO<sub>2</sub> interferometer and Thomson scattering). The port dimensions are compatible with the diagnostics requirements.

The distribution of diagnostic costs along the machine lifetime can be organized considering three FAST phases: FAST Initial Operation Phase (*operation start-up, machine tests, set-up of scenarios*), Plasma Physics Phase 1 (*preparation of scenarios and RF commissioning*) and

Plasma Physics Phase 2 (*full exploitation of scenarios and fast particle dynamics studies*). In this scheme, at the start of FAST Initial Operation the *standard diagnostics* should be available as basic diagnostics (cost ~6000 k€, DAS included). At the beginning of the Plasma Physics Phase 1, the *first priority diagnostics* (cost ~7500 k€, DAS included) must be available and fully commissioned: such diagnostics are crucial for the mission of the device. In order to meet this deadline, the expenditure for the *first priority diagnostics* should start together with the standard diagnostics. The *R&D needed diagnostics* (cost ~ 5000 k€, DAS included) are important for the mission of the device, as several diagnostics in this class are devoted to the measurement of burning plasma parameters, and their expenditure should be distributed during the three phases. A possible distribution of the diagnostics expenditure in time is shown in Figure 7.

FAST will use some of the FTU diagnostic hardware considered to be recoverable: it is important to note that the hardware cost evaluation of diagnostic systems given above is the net additional cost after the use of the available hardware from FTU. The costs for the port integration of systems on the machine are not included in the present evaluation.

### 3.4 - Power supply

The FAST power supply system includes three main subsystems: the 400 kV switchyards, the poloidal field coil power supplies and the toroidal field power supplies. Figure 8 shows the total power for the scenario  $I_p=6.5\text{MA}$ ;  $B_T = 7.5\text{ T}$  also including: a local reactive power compensation (by means of a Static VAR Compensator, SVC) of 80 MVar ; a total heating power of 100 MW (at grid level) and a stationary load of 25 MW. Due to the amount of requested power, connecting to a powerful node of the 400 kV Grid is needed. For this reason, an accurate check by the TERNA (owner of the National Transmission Grid, RTN), including both active and reactive power effects on the specific grid, has been performed providing positive results for two possible ENEA's sites (see Sec. 4).

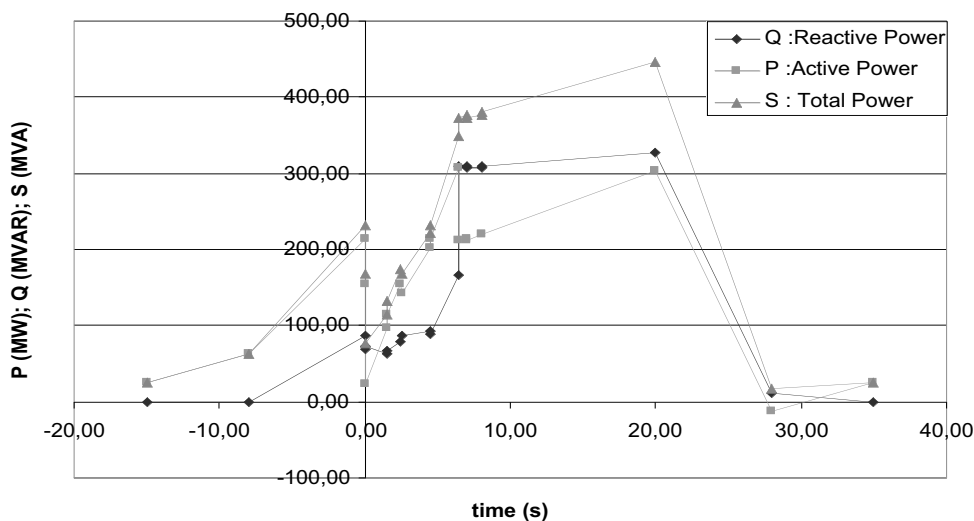


Figure 8 – FT3 Electrical power on 400 kV RTN side

Within the assumed 400 kV reference solution, FAST needs a dedicated switchyard to supply PFC, TFC, Additional Heating Systems and Auxiliaries. All loads are fed by one Main Step Down Transformer (400/36 kV) with three secondary windings: two (225 MVA each) star connected and grounded through a resistor, to supply FAST and one (150 MVA) delta connected to allow free circulation of third harmonic currents. On the basis of the analysis performed by TERNA (the Italian Grid Utility), Active Power Shedding is not requested; in any case, if needed in future, dedicated switched resistors could be connected to the tertiary winding, for this purpose.

Sharing the total power among two secondary windings has the scope to enable the use of 36 kV level on the secondary sides (instead of the more expensive 75 kV level) limiting the rated current and short circuit current within the today breaker capability at this voltage.

Each circuit for the supply of the toroidal field coil (TFC) and the various poloidal field coils (PFC) is generally made by a converter transformer, a thyristor converter unit, a protective crow-bar and high-speed solid state switches for the additional resistance units.

#### 4. SITE ASSESSMENT

An assessment of the generic site requirements has been made. The following elements have been considered: building availability and construction capability, water-cooling and electrical power availability. Two sites have been considered as reported below. The requests in terms of power from the grid, cooling water and building volume are given in Table 7.

Table 7 – Generic Site Requirements

Item	Requirements	Notes
<b>Land</b>	~3 ha (Conventional switchyard)	Building capability -58000 m <sup>3</sup> including a nuclear building of 17000 m <sup>3</sup>
	~2.5ha (Gas-insulated compact switchyard)	
<b>Electrical power</b>	P max= 500 MVA S <sub>cc</sub> = 10 GVA ΔP= ± 100 MW ΔQ = 150 MVar	<ul style="list-style-type: none"> <li>• Present preliminary design includes 100 MVar local Static VAR Compensator.</li> <li>• Active Power Shedding System is not presently foreseen.</li> <li>• Detailed interface conditions between FAST system and 400 kV Grid will be defined in agreement with specific indications by TERNA</li> </ul>
<b>Cooling water</b>	~10 l/s	Mainly due to thyristors converters and transformers

The choice of the FAST site will depend on both strategic and cost issues. In the following we list the costs related with the choice of either one of the two sites.

- Casaccia (ENEA site). To connect the Site at RTN, a 5km/400kV cable link is needed with an expected cost of 12 M€. In addition, the 400 kV gas insulated switchyard cost (which can be covered by preferential support) is 14.4 M€. The required cooling water can be easily achieved. Since none of the available buildings is suitable, a total cost for building of 19.4 M€ must be added. The total site related cost for Casaccia is 45.8 M€ (with 14.4 M€ eligible for preferential support).
- Frascati (ENEA site). This site can reuse most of the existing buildings also to host the 400 kV gas insulated switchyard. For the nuclear building, a new hall could be built adjacent to the present FTU hall. An upper bound to the cost for a second nuclear hall, if required, is 9 M€. The 400 kV line is about 12 km away with a difficult connection path. The cost of the 400 kV cable link is 26 M€. The cost of the 400 kV switchyard is 14.4 M€, as for Casaccia. Cooling water is presently available under the hypotheses of Table 7. Therefore, the corresponding site related costs amount to 49.4 M€ (with 14.4 M€ eligible for preferential support).

## 5. COST, MANPOWER AND TIME SCHEDULE

The total updated investment cost of the machine eligible for preferential support has been estimated 276M€ without including VAT and contingencies (Nov. 2007). In making this estimate it has been assumed that some of the existing heating systems can be re-utilized, i.e. the lower-hybrid system. Diagnostic R&D is included.

The annual operation costs has been estimated for a typical agenda of 1500 shots (800 of which are performance shots see Table 4) and 150 operation days and amount to 13.5 M€/year (8M€ for maintenance, 2.5M€ for consumables, mainly liquid nitrogen, and 3M€ for electricity), plus personnel. The operation team is estimated to be 150 PPY.

It is worthwhile to compare the costs of operation of FAST with that of ITER: the average cost/shot is ~ 60 times higher in ITER than in FAST (~8k€ against ~500 k€). These figures indicate that also from an economical standpoint it is advantageous to prepare ITER scenarios in a small device such as FAST, also taking into account the fact that this device may well extend its lifetime until the DEMO construction phase and therefore can be used also to test technological issues and study plasma scenarios in preparation for DEMO.

The FAST construction time is estimated in 6 years following the design phase and the placement of the contracts for the long lead items. The realization will involve an average project team of 120 py/y.

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