Validity of the Objectives and Solutions of the Ignitor Program

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From JET to the reactor

P.H. Rebut, Alfvèn Prize Lecture

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http://eps2006.frascati.enea.it/invited/post.htm

In a reactor, the energy produced by fusion reactions only matters, not the record on some of the non dimensional parameters.

The real gain has to be proven in tritium operation.

Having superconducting coils adds to the complexity and the cost of a machine; in my opinion it was premature to do it on ITER on the program leading machine which is still far from a reactor.

Taking into account the **efficiency of the conversion** from heat to electricity, and the **efficiency of the auxiliary heating** and **plasma control**, a **Q of 50** for the fusion reactor is required

To achieve such a Q of 10, ITER must operate in the **H mode**, The H mode appears in presence of a divertor, over a power threshold. It is not possible to **maintain** it for a **long time**.

The X point limiter

P.H. Rebut, cont'd

We may take advantage of the **H mode physics** by installing a limiter in the vicinity of the X point rather than a standard divertor.

The advantages are in a given machine:

- **Better plasma performances**

 a larger plasma volume, a factor 1,3 in the case presented
 an increase of the plasma current by 1,2
 an increase of the plasma pressure by 1,1
 a higher fusion power by a factor 1,5

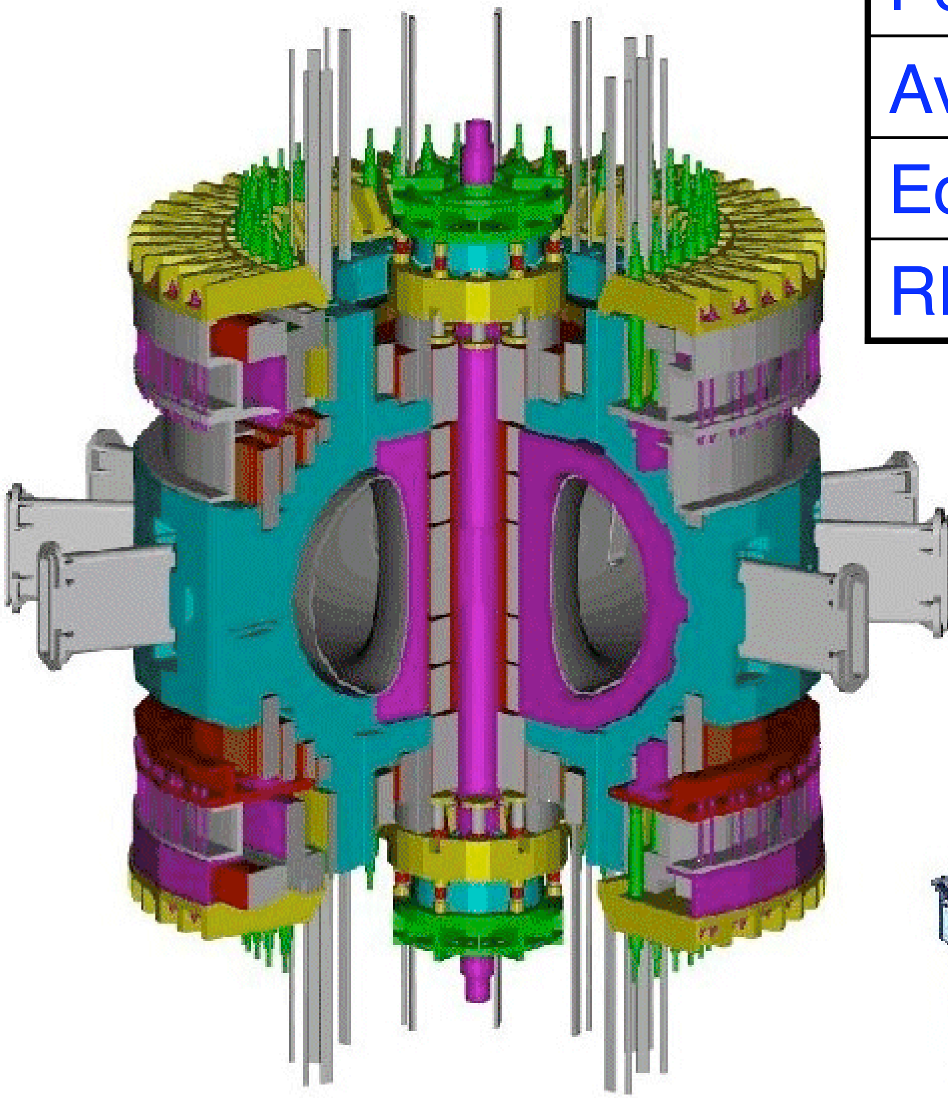
H mode plasma have been possible on JET with the X point pushed into the wall.

- **Better technical solution**

 a simpler technical solution with a larger wetted area
 an easier possibility to sweep and move the contact area

IGNITOR PARAMETERS

Plasma Current I_P	11 MA
Toroidal Field B_T	13 T
Poloidal Current I_θ	8 MA
Average Pol. Field $\langle B_p \rangle$	3.5 T
Edge Safety factor q_ψ	3.5
RF Heating P_{icrh}	<18 MW



R	1.32 m
a	0.47 m
κ	1.83
δ	0.4
V	10 m^3
S	36 m^2
Pulse length	4+4 s

The Ignition Strategy

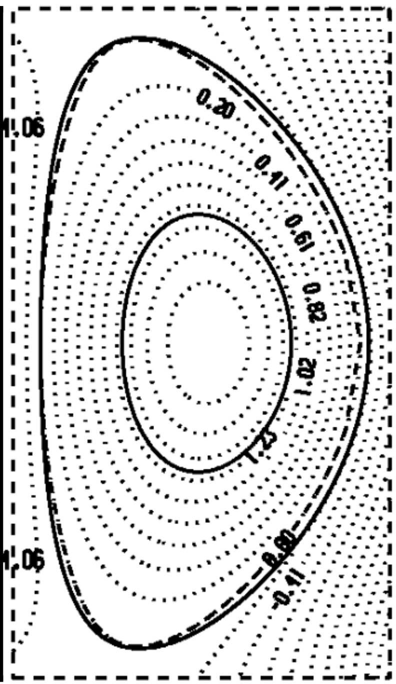
$n\tau T$: high density, moderate τ_E, low temperature

$n/n_{limit} < 0.5$, low $\beta \Rightarrow$ far from empirical operational limits

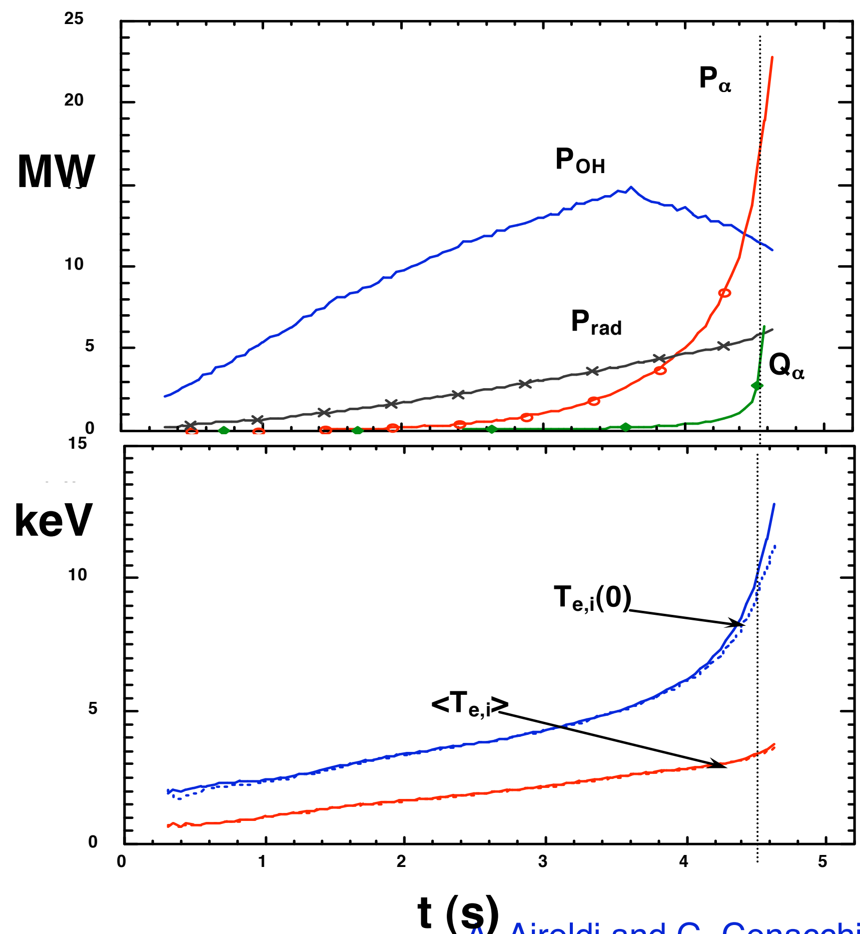
$\tau_{\alpha,sd} \ll \tau_E$, $\tau_{burn} \gg \tau_E$

Typical Parameters at Ignition

T_{e0} , T_{i0}	11.5, 10.5 keV
n_{e0}	10^{21} m^{-3}
$n_{\alpha 0}$	1.2×10^{18} m^{-3}
P_α	19.2 MW
β_{pol}, β	0.2, 1.2%
τ_E	0.62 s
τ_{sd}	0.05 s
Z_{eff}	1.2



13 T, 11 MA



A. Airoldi and G. Cenacchi

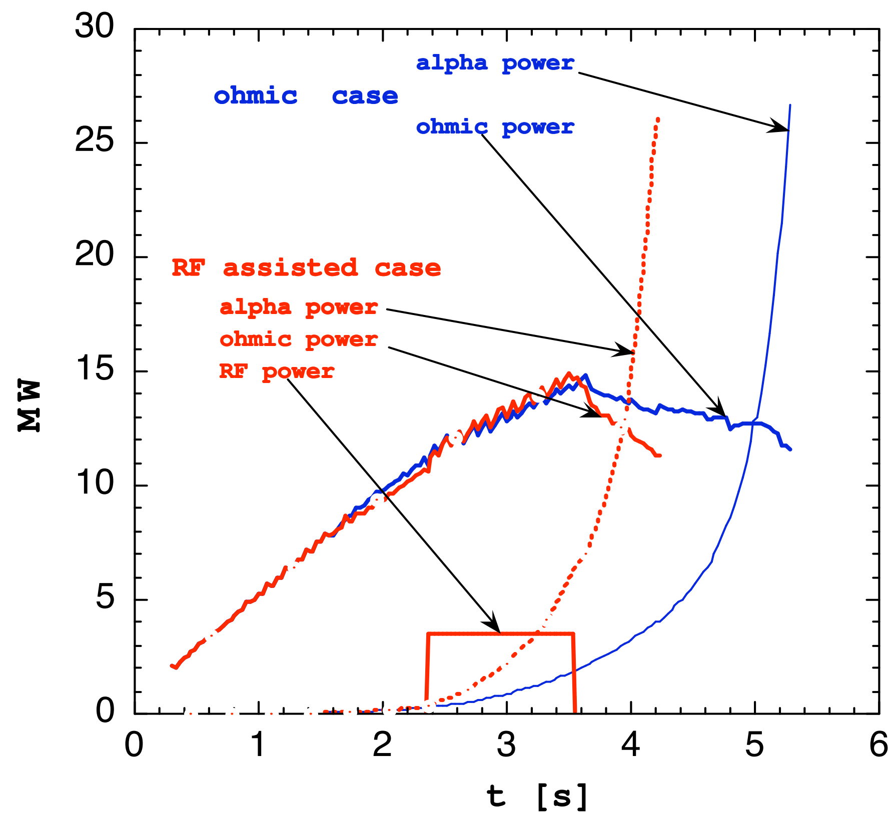
RF Assisted Ignition

Ignition can be accelerated by the application of modest amount of ICRH during the current rise.

The full current flat top is available to study the plasma in burning conditions.

(Note that ignition occurs when ohmic heating only is present)

A. Airoldi and G. Cenacchi



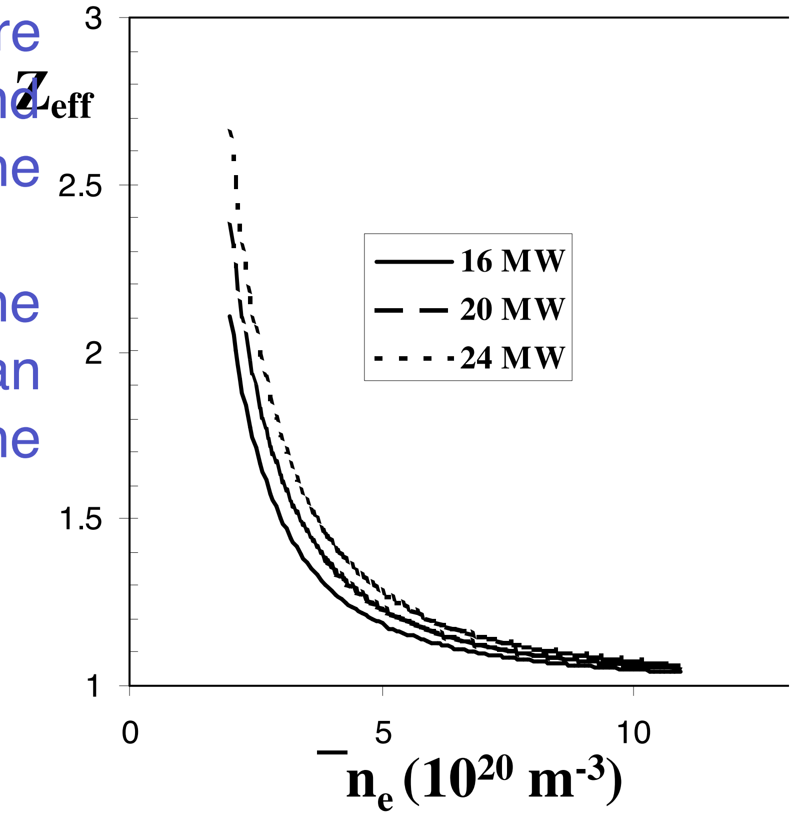
Comparison of Ohmic and RF assisted ignition scenarios (JETTO code).

Divertor: why not?

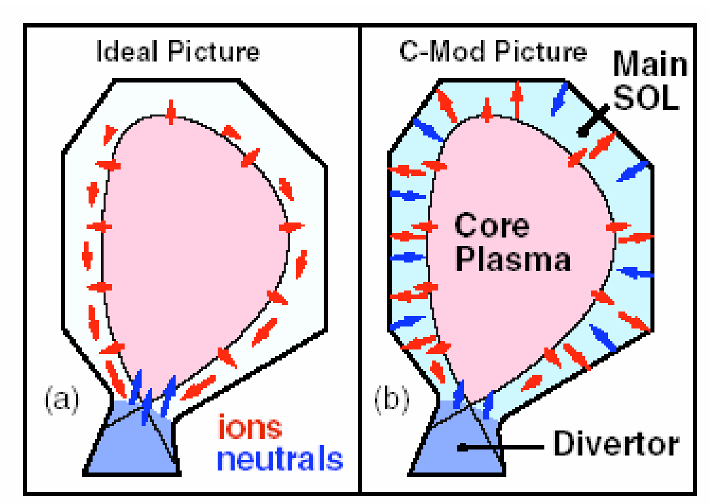
•Divertor machines do not produce “cleaner” plasmas than limiter, high density devices..

•At high density, the low temperature reduces sputtering from the wall and impurities are effectively screened from the main plasma.

•In these regimes, particle recycling from the main chamber and cross- field diffusion can challenge the picture of the divertor as the sole power and particle sink.

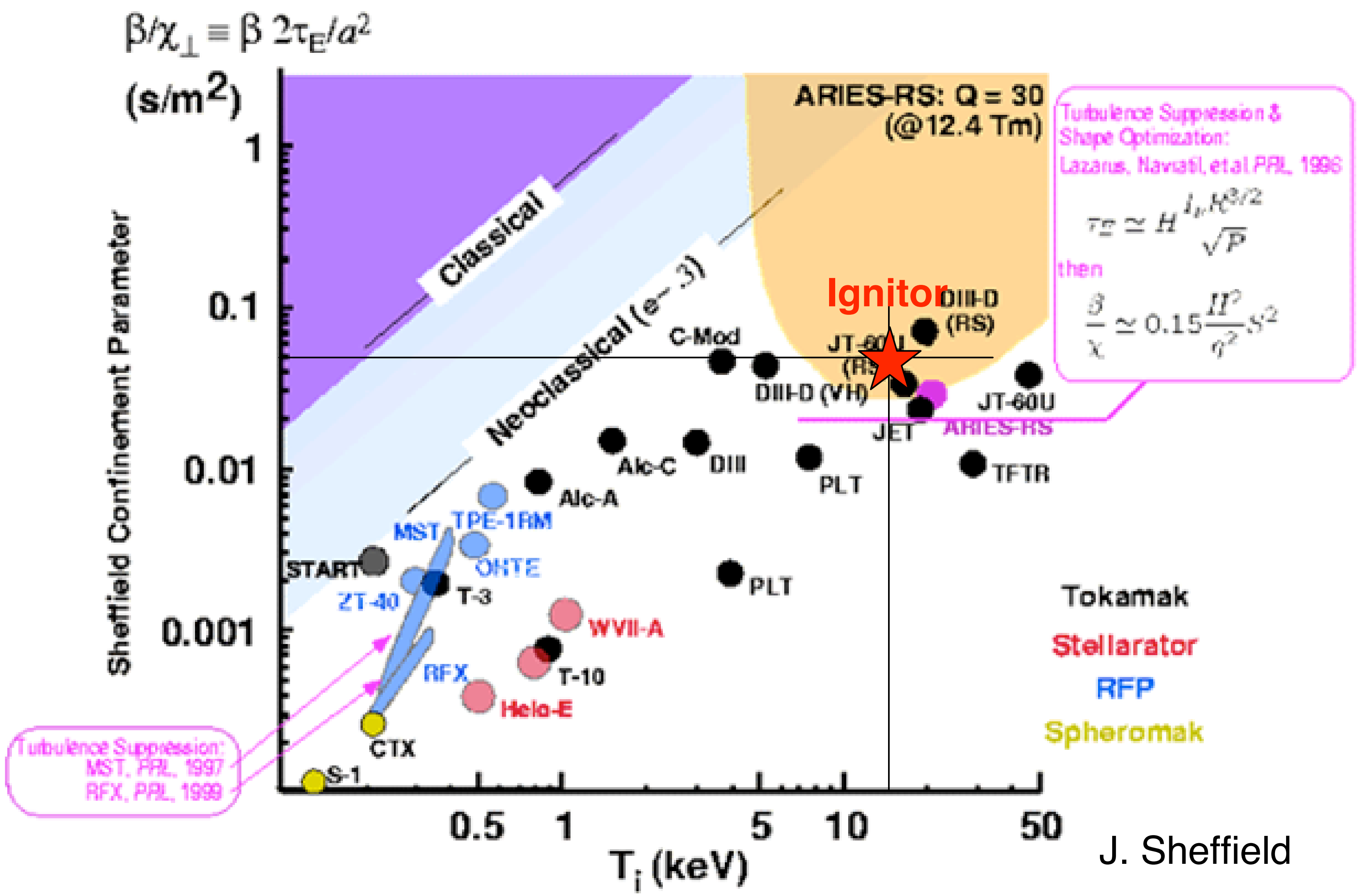


G.F. Matthews, et al., *J. Nuclear Mat*. **241-243**, 450 (1997)



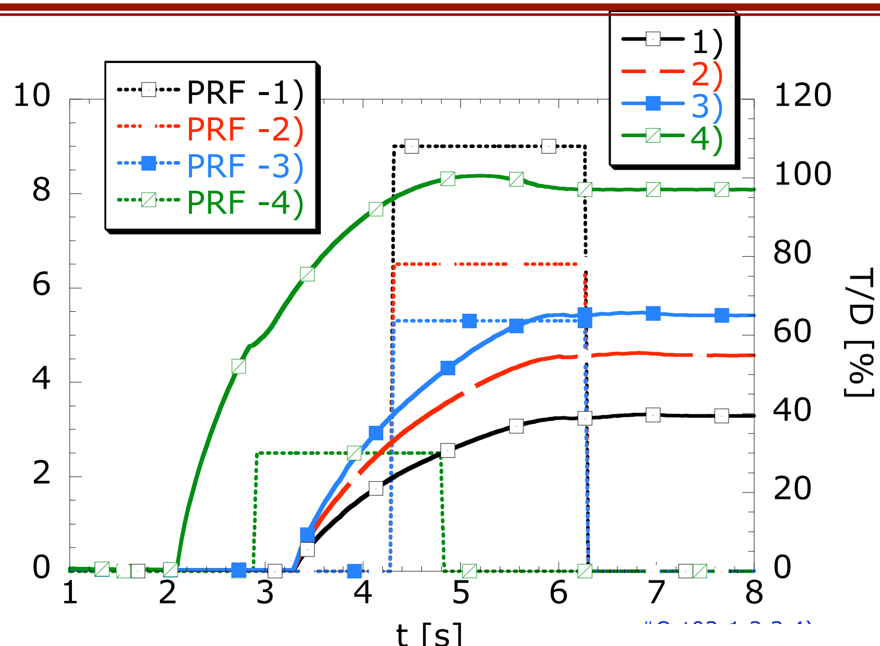
[1]LABOMBARD, et al., *Nucl. Fusion* **40** (2000) 2041.

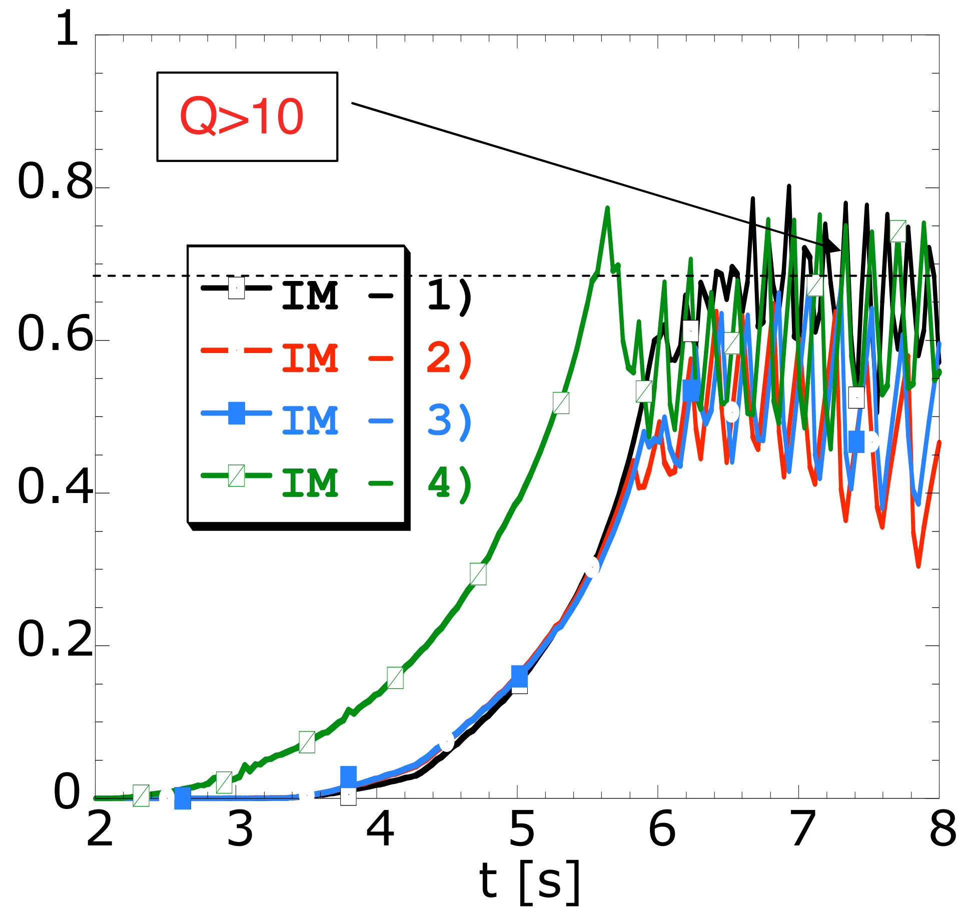
Fusion Energy Relevant Levels of β/χ have been Achieved for Short Pulses

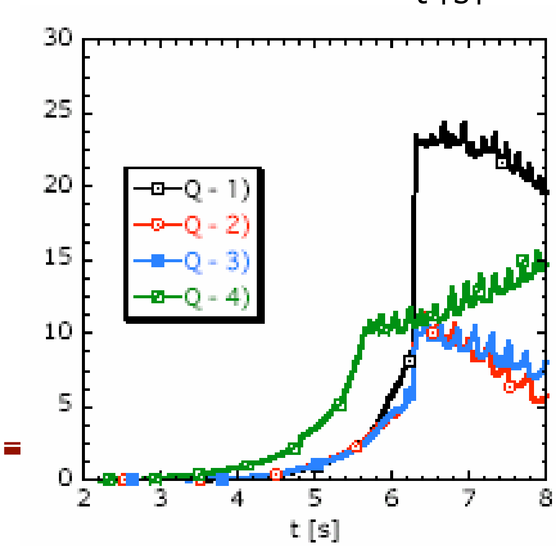


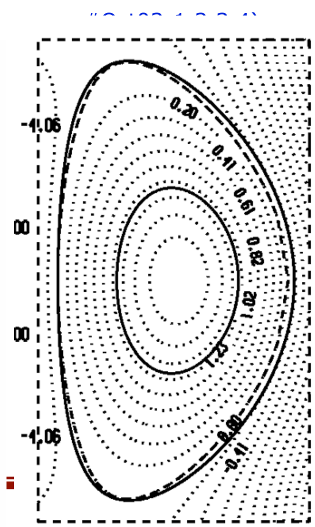
J. Sheffield

Ignition Control by means of Tritium and RF









13 T, 11 MA

With proper timing, the RF power compensates for the unbalanced fuel ratio. As a result, only small differences in the ignition margin are observed.

Plasma Performance in H-mode (See Poster NP1.00003)

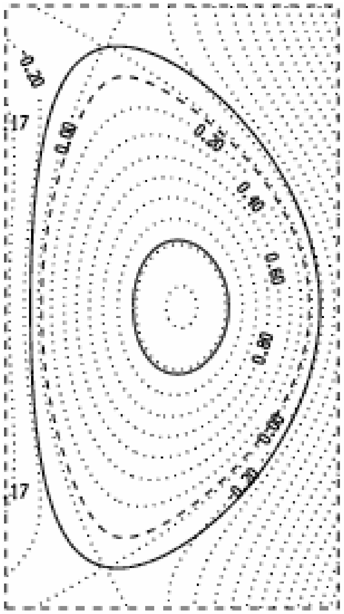
New scalings for the ELMy H-mode confinement have been derived, some having no β dependence. In particular, Eq. 9 of [1]:

$$\tau_{E,scal}^{no-\beta} = 0.0360 \times I_p^{0.85} B_T^{0.17} A^{0.82} R_0^{1.21} a^{-1.25} \bar{n}_e^{0.26} M_{eff}^{0.11} P_{heat}^{-0.45}$$

Similarly, the most recent scaling [2] for the power threshold to access the H-mode regime has a weaker dependence on the magnetic field:

$$P_{L\rightarrow H} = 0.150 \times B_T^{0.58} A^{0.85} \bar{n}_e^{0.56} M_{eff}^{-1}$$

9 MA, 13 T, DN



G. CENACCHI

[1] J.W. Cordey at al., *Nucl. Fusion* **45**, 1078 (2005)

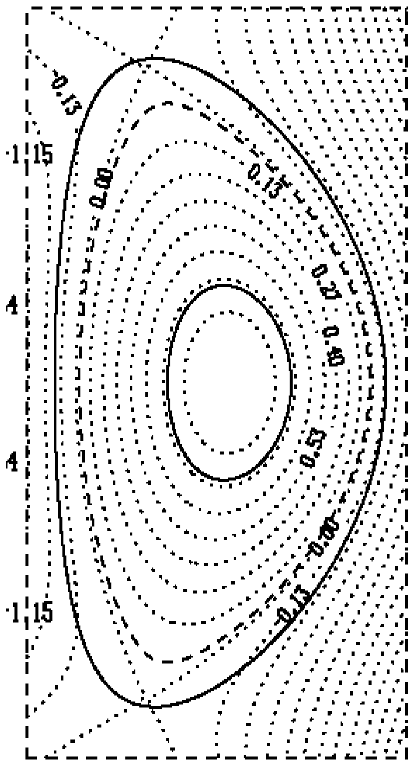
[2] D.C. McDonald et al., *Plasma Phys. Control. Fusion* **48**, A439 (2006)

Scenarios with reduced parameters (See Poster NP1.00004)

Magnetic field up to 9T

Plasma current up to

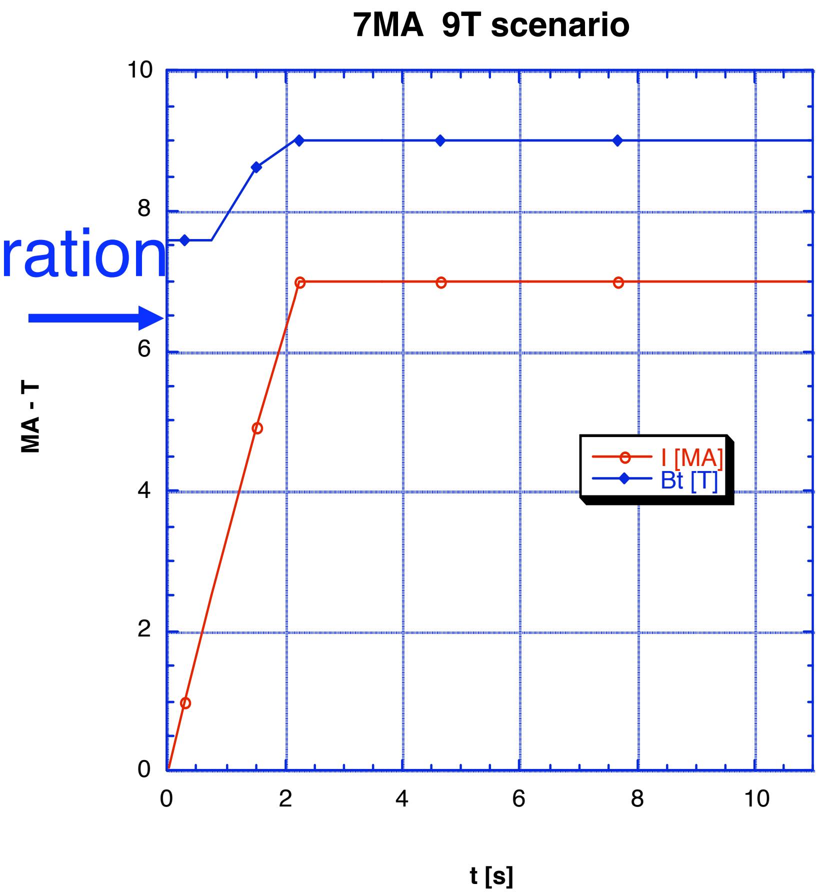
i) 7 MA, “first wall limiter” configuration



⇒ Long pulse

or

(double X-point)



Pulse length consistent with mechanical and thermal requirements of the magnets, and available magnetic flux

The Compact, Multiple Barrel High Speed Pellet Injector

(See Poster NP1.00006)

Both high and low magnetic field experiments have shown that low thermal diffusivities can be produced in the central part of the plasma column as a result of peaked density profiles, such as those resulting from pellet injection.

A new multiple barrel, 4 km/s pellet injector for the Ignitor experiment is being developed jointly by ENEA and ORNL.

In Ignitor good pellet penetration from the low field side can be expected in burning plasma condition

The propelling sub-system is undergoing final testing in Italy before shipping to ORNL for complete integration with the cryogenic system.

Testing with real pellets has begun at ORNL.

ICRH Physics
(See Poster NP1.00005

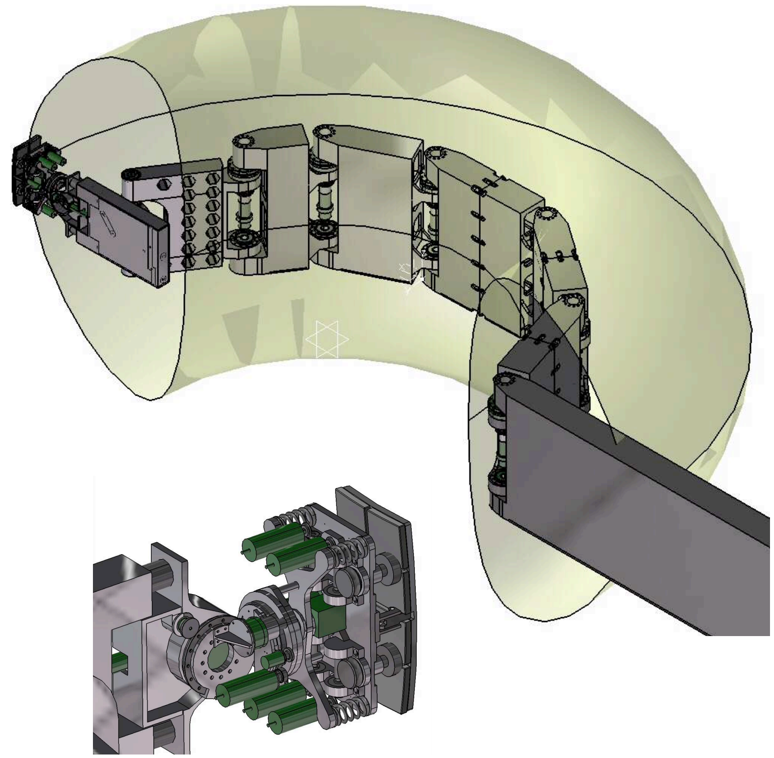
The application of modest amounts of ICRH power (3-6 MW), either during the current rise or the pulse flat-top, can be used to increase the temperature in a range of accessible plasma regimes and provide a safety margin for the attainment of ignition.

The available frequencies of the ICRH system can cover the range of operation at magnetic fields from 9 to 13 T. Different heating scenarios are considered:

B (T)	H/D/T	T/He^3	D
9	1st,2nd,3rd at x-0.5	2nd,1st at x - -0.5	
10	1st,2nd,3rd at x-0.9	2nd,1st at x- -0.25	1st at x- -0.95
11	Out of res	2nd,1st at x--0	1st at x- -0.75
12	Out of res	2nd,1st at x-0.2	1st at x- -0.6
13	Out of res	2nd,1st at x-0.4	1st at x- -0.4

Design Developments

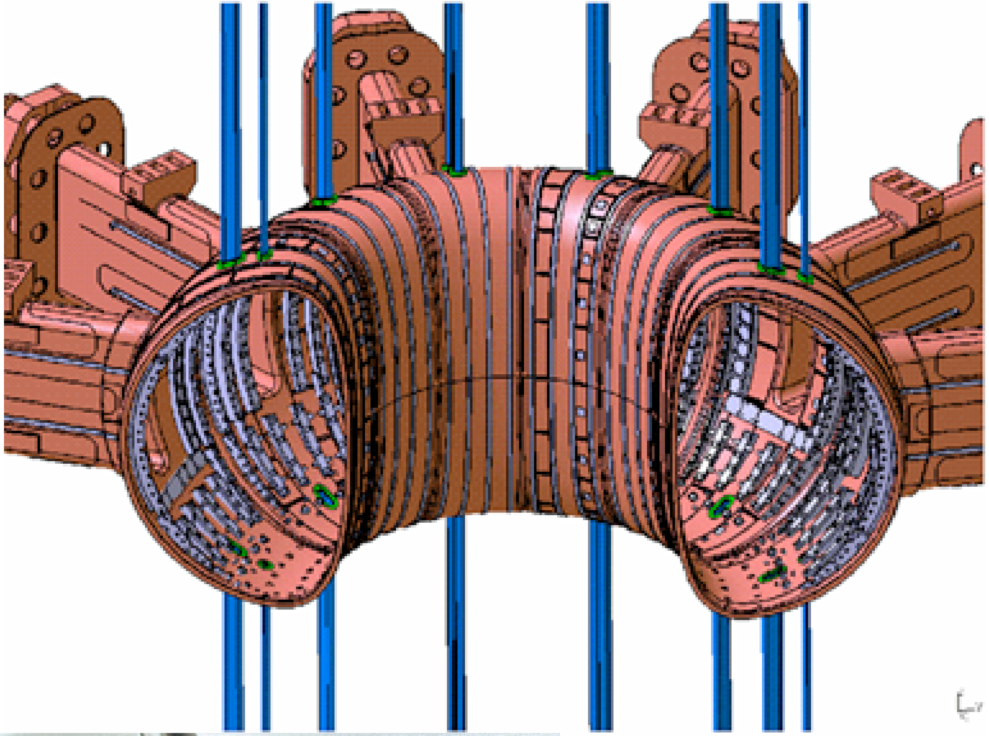
- Non-linear structural analyses of the machine Load Assembly have been performed taking into account the effect of friction coefficients between the significant components.
- Updated plasma disruption conditions for VDE's that result in higher out-of-plane loads have been considered.
- The relevant 3D virtual mockup has allowed for the Remote Handling (RH) analysis of the boom kinematics to cover all positions inside of the Plasma Chamber. The design of the Ignitor RH system is based on that used on FTU to install the Mo toroidal limiter.

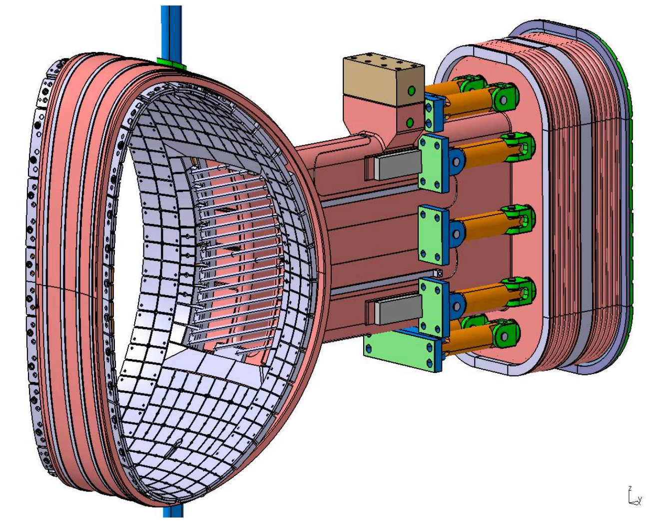


Fully extended boom inside the plasma chamber and end effector for tile carrier

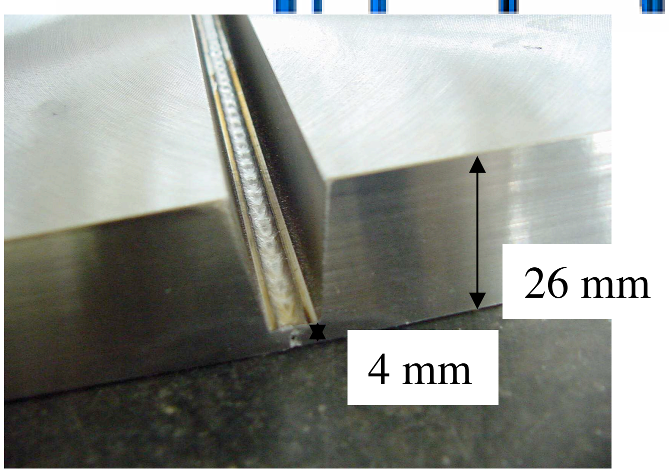
Inconel 625
12 Ð shaped sectors
Variable thickness

Plasma Chamber





One sector of the plasma chamber including the ICRH Faraday shield and first wall.

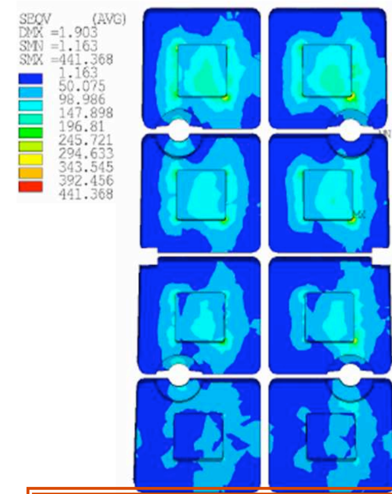


Each sector is joined to the adjacent one by a laser butt welding which ensure vacuum tightness. Once the torus is completed, the welding groove is filled by TIG-NG (Narrow Gap) to strengthen the joint.

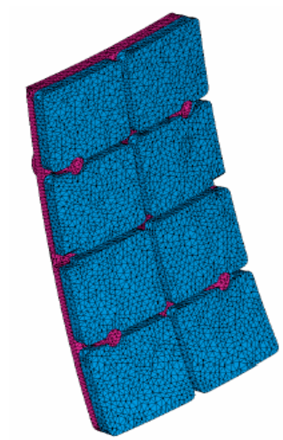
First Wall Limiter

The First Wall Limiter of Ignitor is covered with TZM (Mo) tiles supported by Inconel plates attached to the plasma chamber.

The thermo-structural non-linear analysis of the most stressed tiles under cycled loads shows that temperature increases, stresses, and deformations are within the allowable values.



Maximum VM stress on tile carriers during a VDE disruption



The design of the first wall is optimized to spread the plasma heat loads onto a relatively large area, to reduce the peak values of the thermal wall loads.
The "limiter" geometry is not suitable for numerical codes with magnetic fitted coordinates (developed mainly for single X-point configurations), therefore the development of a new code, ASPOEL [1], was undertaken .

[1]F. Subba, et al., *Proceed. 17th PSI Conf., Hefei Anhui (China), 2006*

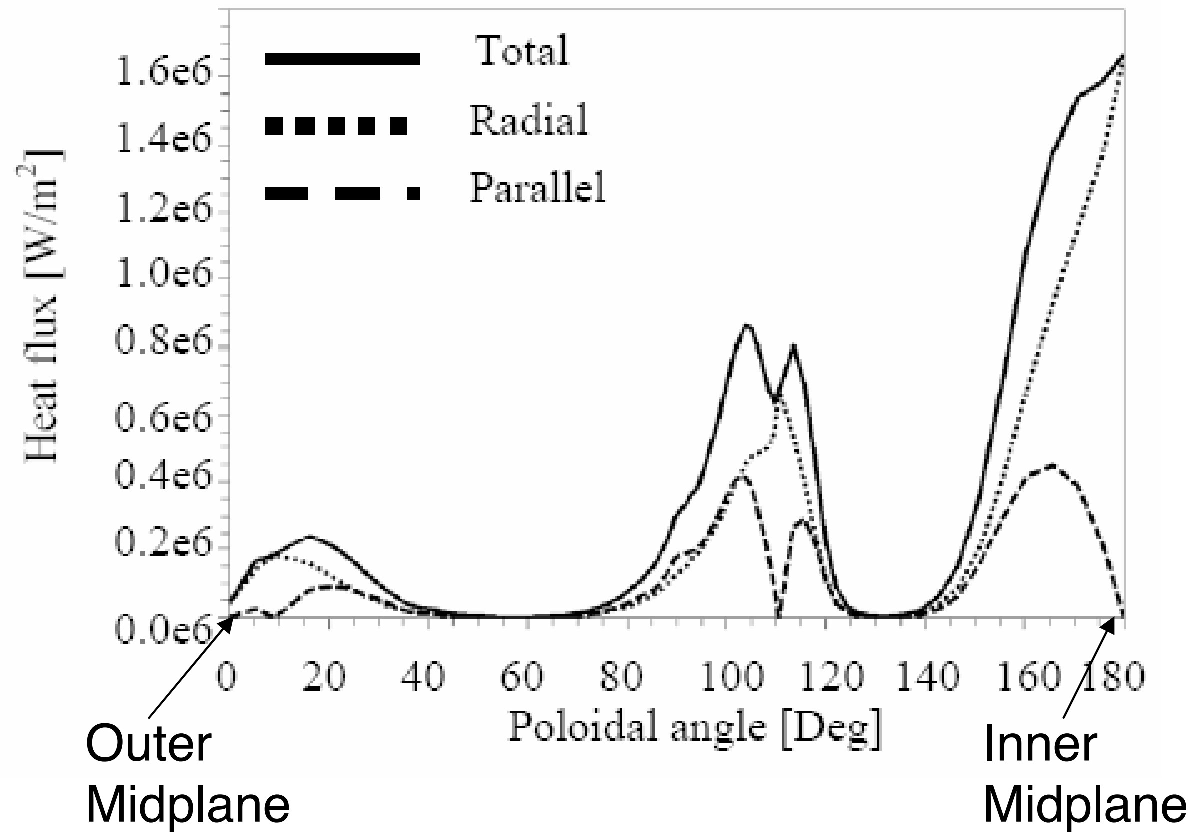
Thermal Wall Loads

The present version of ASPOEL solves a set of single fluid equations, neglecting neutrals and impurities (a given fraction of the input power is assumed to be radiated).

The Control Volume Finite Element (CVFE) technique is applied on a triangular mesh to enforce local energy and particle conservation.

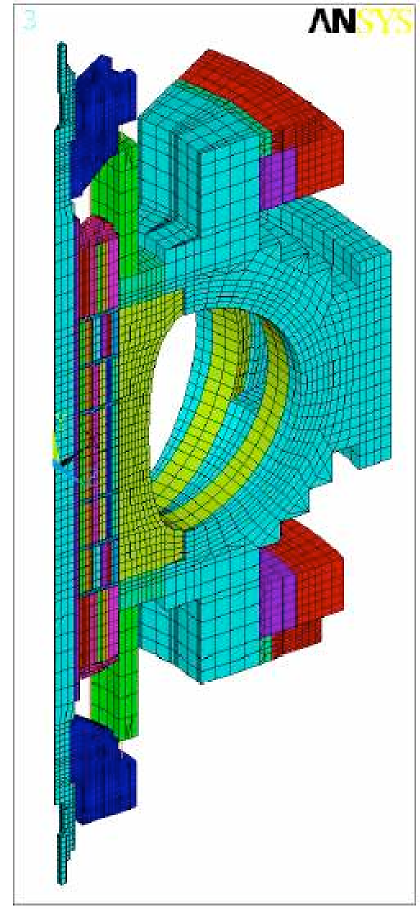
The power conducted/covected to the wall can be separated into parallel and perpendicular contributions.

The peak values computed with the new code are consistent with previous estimates.



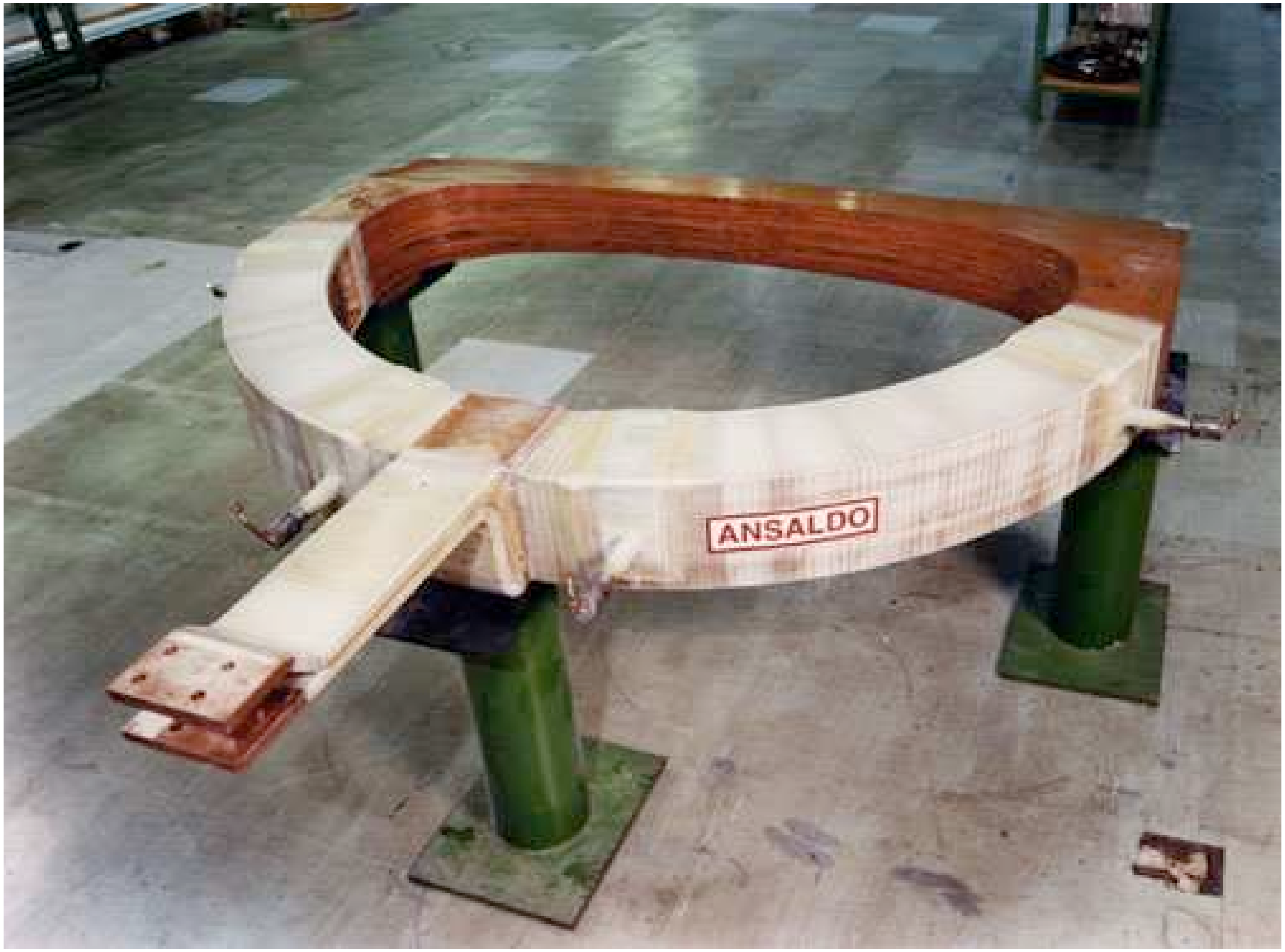
New concept of TFC turns cooling

The Toroidal Field Coil is cooled down to 30 K by gaseous helium. OFHC Copper has been selected for these plates, allowing for an Electron Beam (EB) welding solution of the cooling channels.



The Finite Element ANSYS model of the Load Assembly takes into account friction at the interfaces of significant components.

Full Size Toroidal Magnet Coil

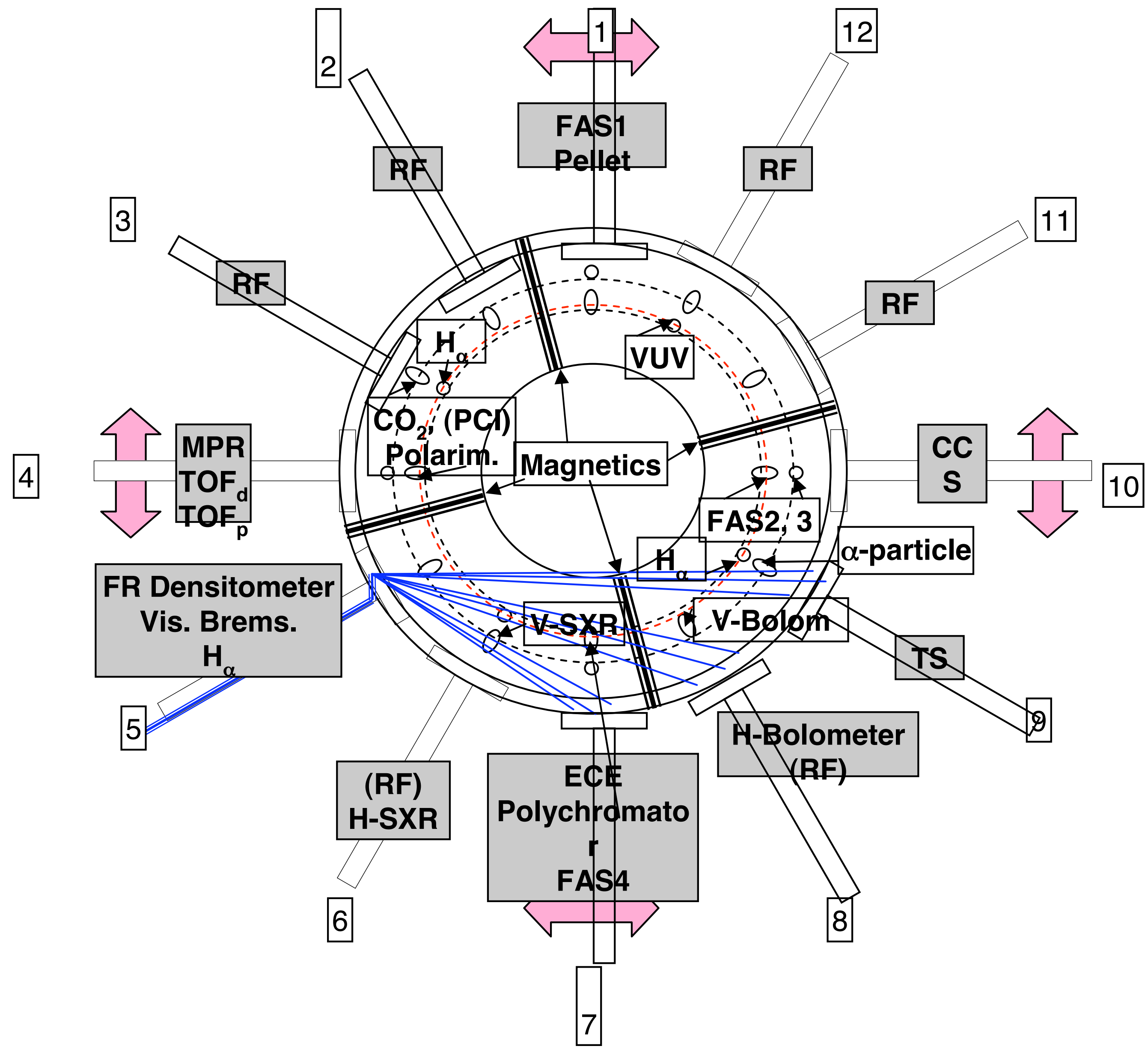


C-Clamp

Central Solenoid Coil

Sector of Plasma Chamber

Diagnostics Lay-out



Site

Ignitor power requirements : $P\text{-}_{max}$ ~ 1.1 GW, E_{EOP} ~ 2.5 GJ.

The Ignitor experiment can be connected to the national power grid at one of major nodes (Rondissone) or at the former nuclear power plant of Caorso . Thus, the need for flywheel generators is avoided and connection costs are minimized.