Progressi recenti negli esperimenti sulla fusione nucleare e applicazioni

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Associazione Euratom-ENEA sulla Fusione

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Arguments

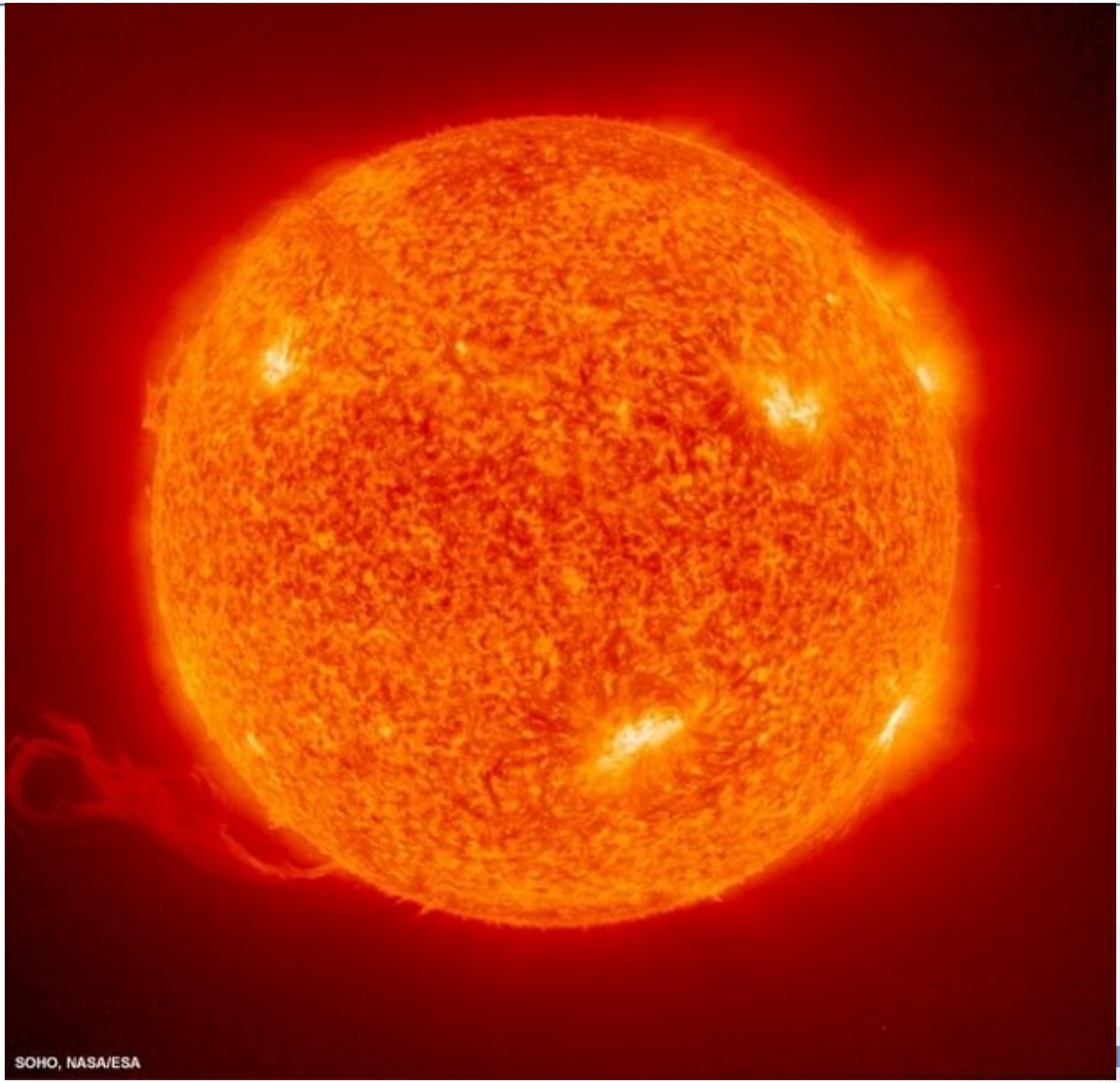
First part : Introducion to Fusion

Second part : main recent results , plasma dagnostics , perspectives in the next decade

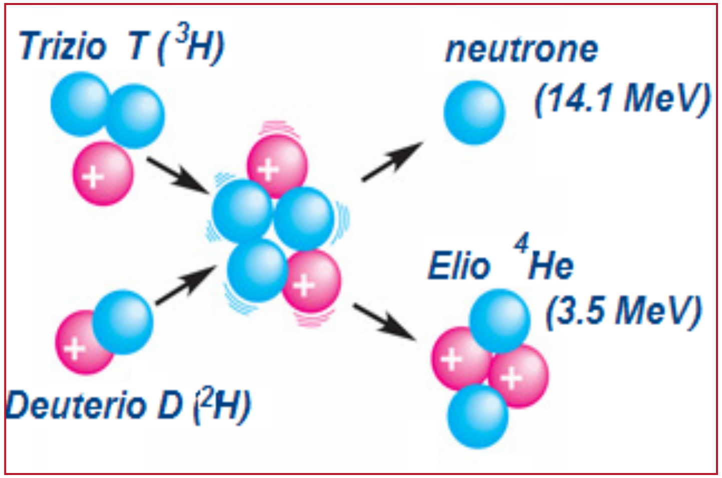
Third part : short term possible application of Fusion , low power neutron sources for Fusion-Fission hybrid reactors

Fourth part : some recent results on Inertial Confinement Fusion

search for ENERGY sources



Per ottenere energia da fusione sulla Terra dobbiamo partire da *isotopi* dell'idrogeno che reagiscono più facilmente:



Energia di reazione >> energia di ionizzazione ⇒ il combustibile è allo stato di ***plasma*** (fluido completamente ionizzato).

Le particelle alfa ($^4He^{++}$) sostengono il plasma (ignizione) se questo è abbastanza caldo, denso e ben confinato.

Temperatura ***T* >10 keV** (100 milioni di gradi); densità *n* e tempo di confinamento τ tali che $\boldsymbol{n \times \tau > 2\times10^{20} sec/m^3}$.

Controlled thermonuclear fusion

Advantages: Deuterium and Litium (which in a reactor is used to regenerate tritium) are available

Safety: i) it is impossible a sudden and uncontrolled increase of radioacivity;

ii)the fuel present in the reactor is small(10g)

iii)the intensity and duration of radioactivity induced by neutron emission on the reactor can be minimized with the development of materials

Organizzation of FUSION in EU

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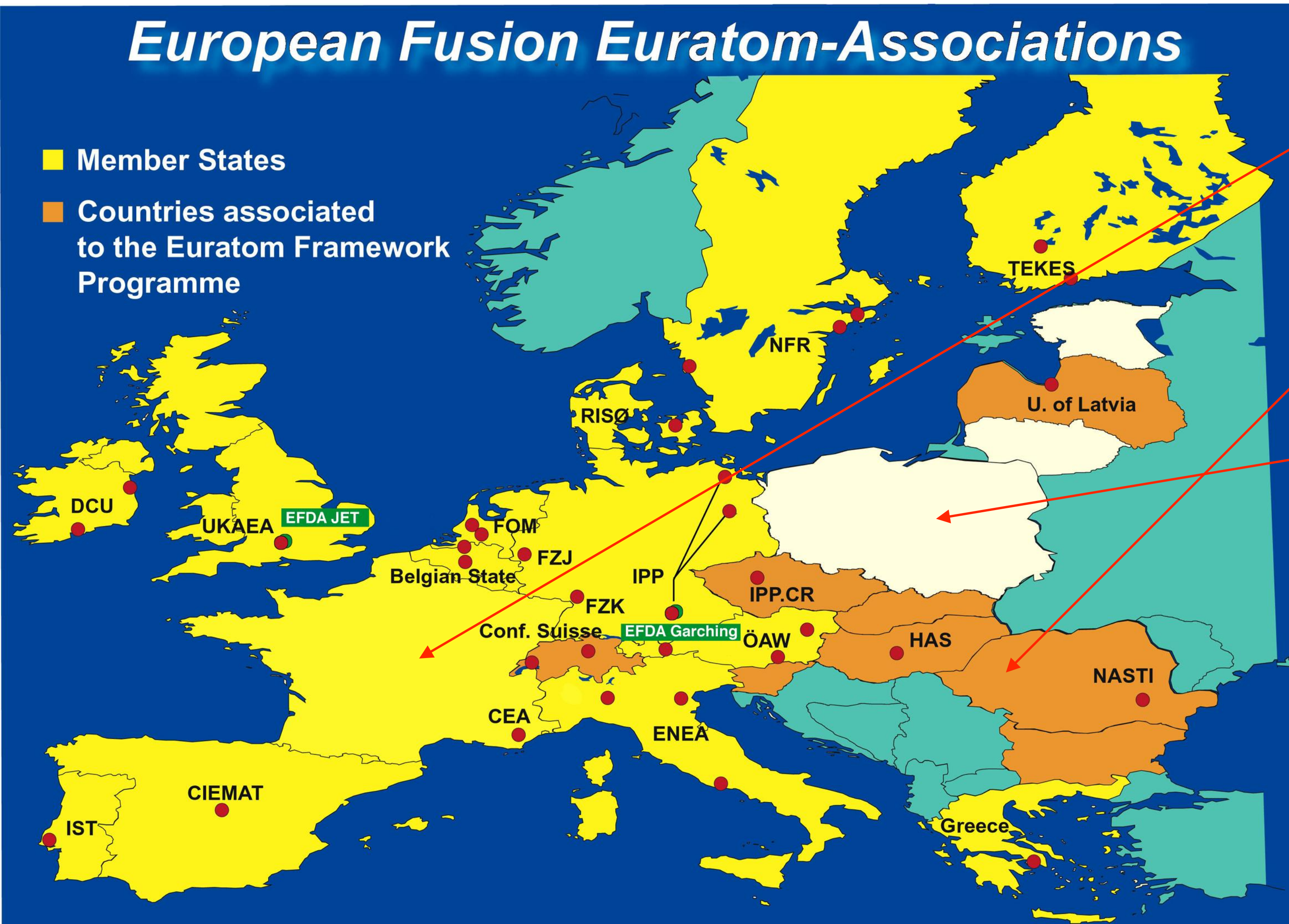
European Fusion Organization

EURO-FUSION Consortium



15 EU Member States

8Associate

3 waiting

m-ENEA sulla Fusione

Fusion : funds and experiments

Fusione: Budget

FP6 (2003-6) 750MEuro,

FP7 (2007-2011) 1950M€ constant approximately in FP8

Esperiments : JET(UK), Tore SUPRA(FR), ASDEX(GE), W7AX(GE), TEXTOR(GE), FTU(IT),RFX(IT), TJII(SP),TCV(Swi)

ITER is starting in Cadarache

About 5000 professionals in EU

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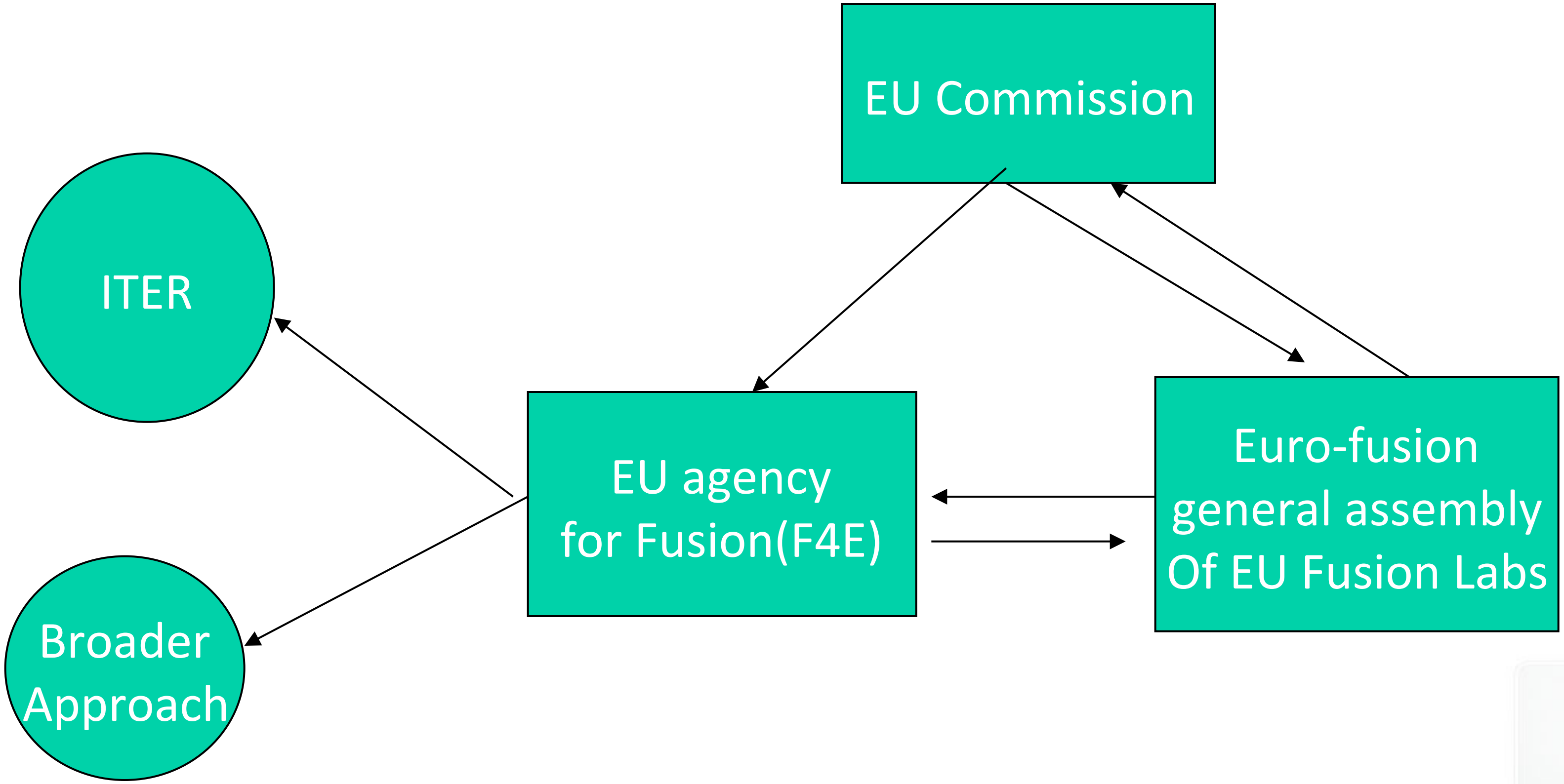
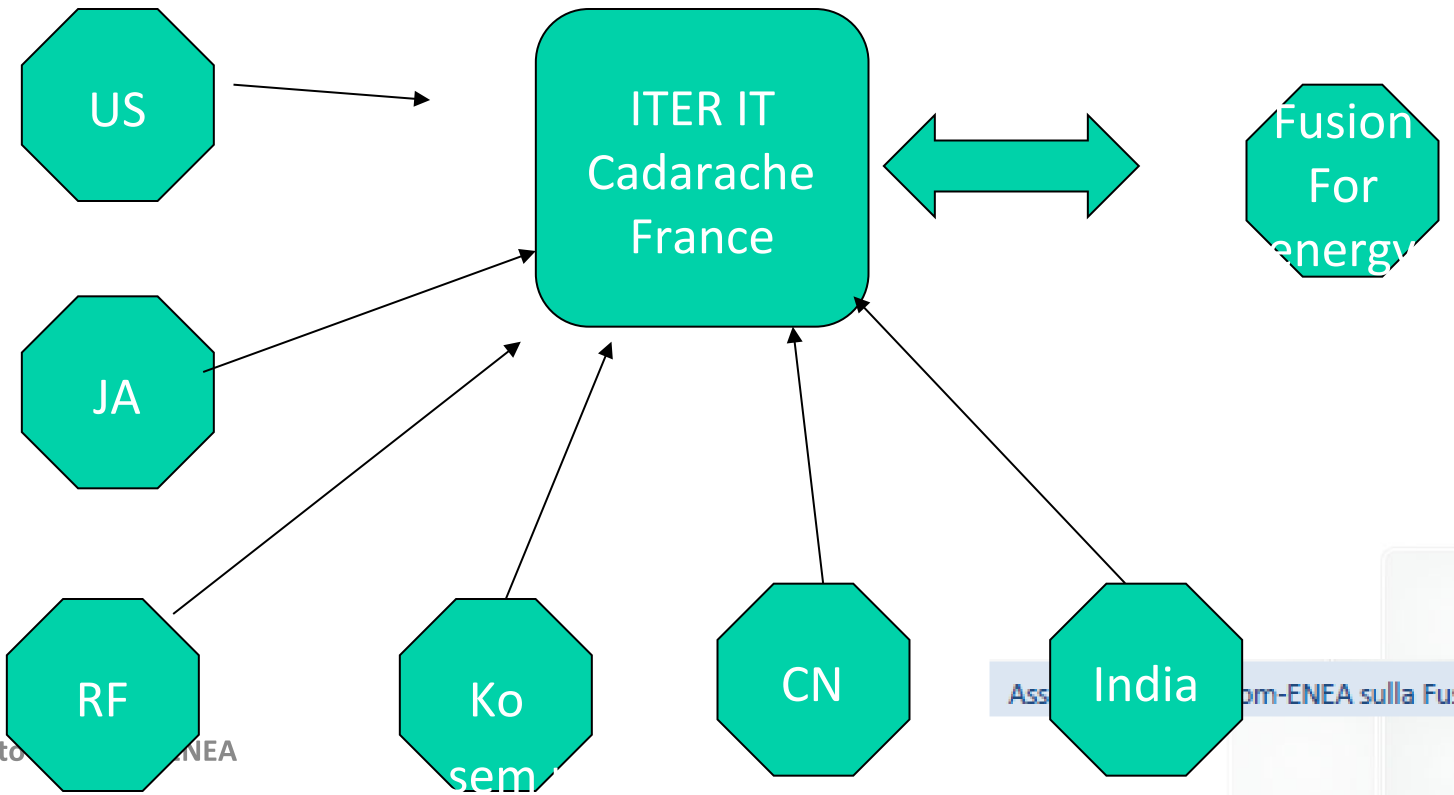
Organization of Fusion in EUROPE

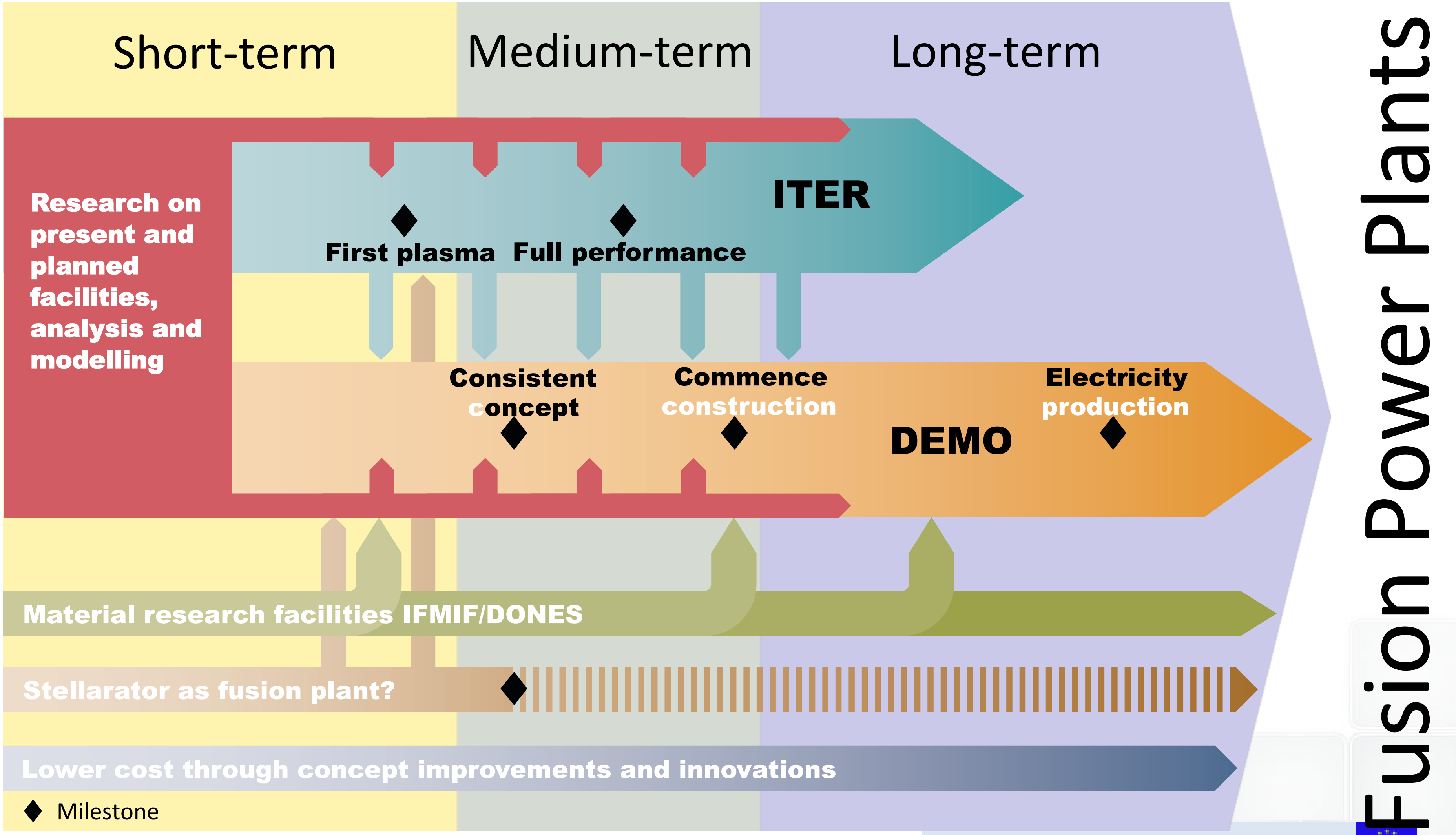


Fusion: ITER IT organization

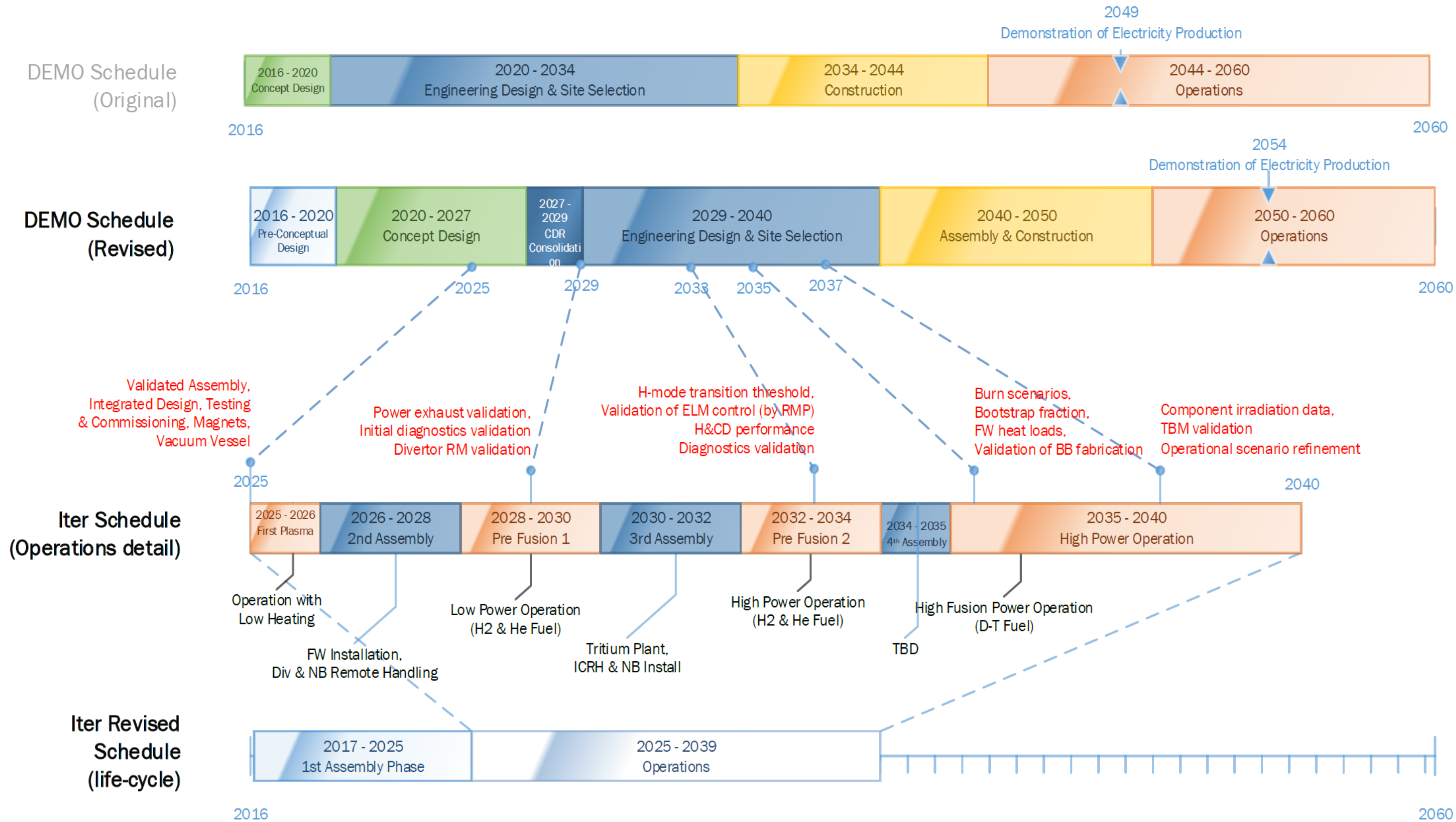
Various domestic agencies contribute to ITER : EU,RF,USA,JA,Ko,IN,CN



FUSION ROADMAP



Phasing of ITER and DEMO operation

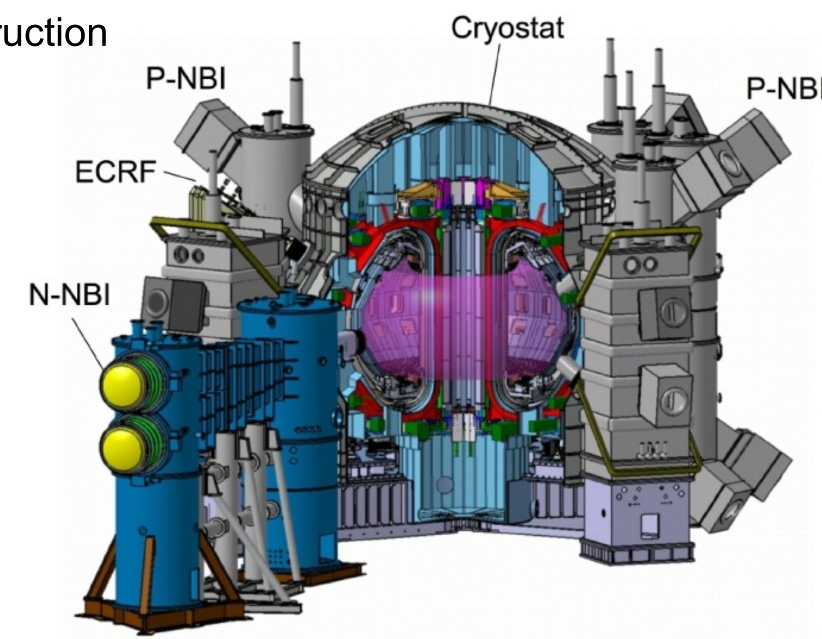


- Revised ITER/DEMO schedule places JT-60SA in an important position as the main support tokamak

Roadmap review and JT-60SA

- **Assumptions on JT-60SA (same as for Roadmap v1)**
  - EU will support JT-60SA operation (25% of total)
  - EU will support upgrades over 2021-5

- **Summary of main points on JT-60SA**
  - JT-60SA is seen as central to the EU strategy, particularly for
    - developing ITER long pulse scenarios
    - ensuring rapid progress to DEMO construction
  - EU supports an early move to a full W wall to ensure rapid progress to DEMO construction



Broader Approach

Includes :

A new tokamak JT60SA in Japan(Naka): ITER satellite

A supercomputing facility (Rokkasho)

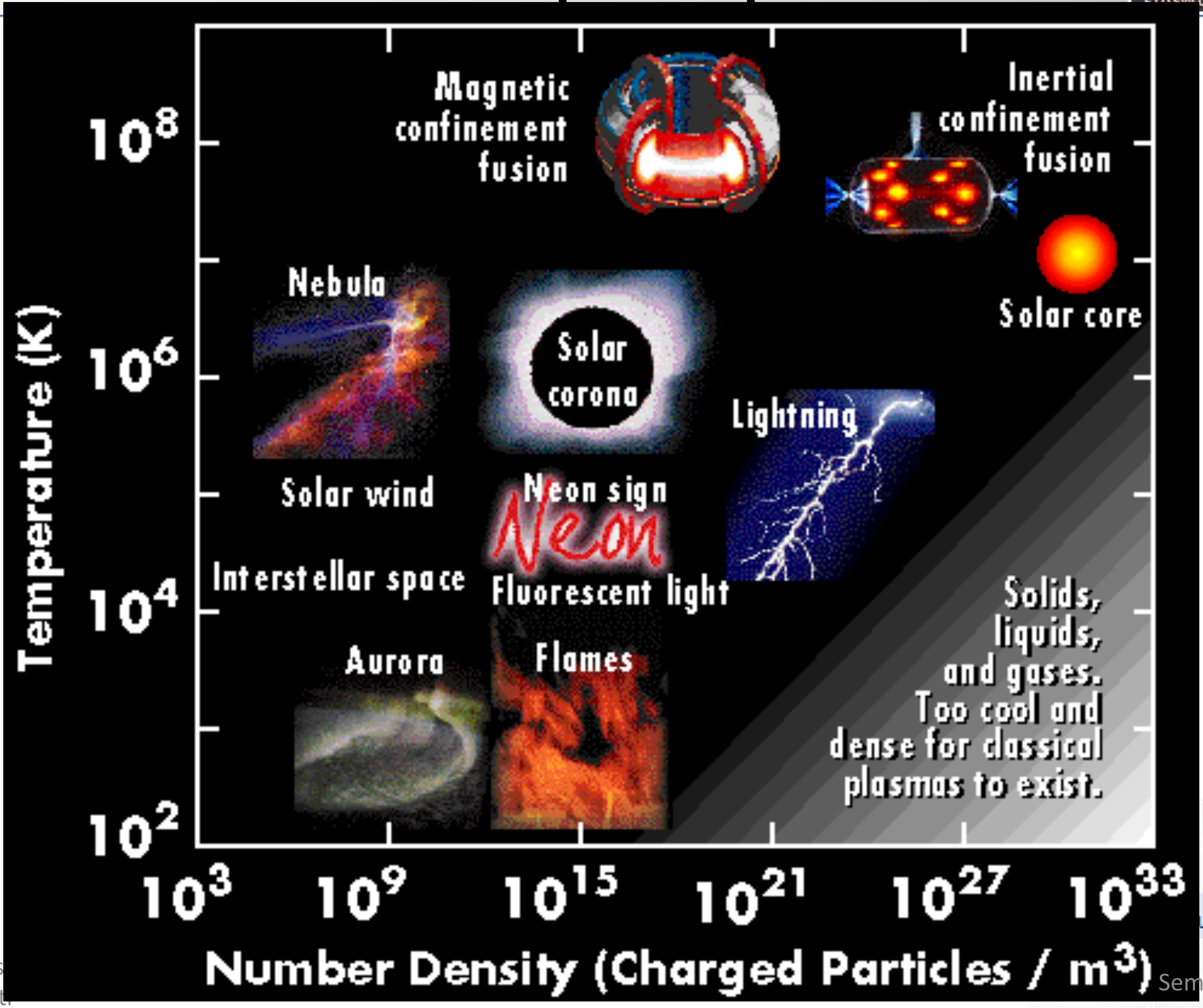
A virtual control room in Japan to run ITER(Rokkasho)

IFMIF – material test facility for high neutron fluxes

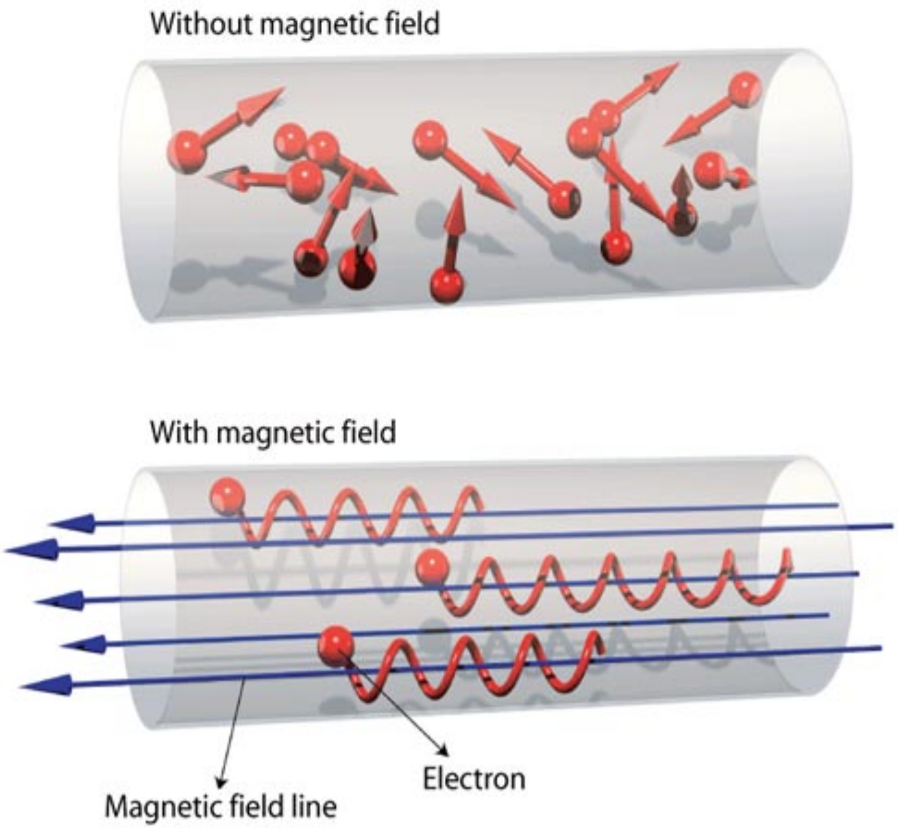
Funds : 50-50 EU/JA

Introduzione : elementi fondamentali

Vari tipi di plasma



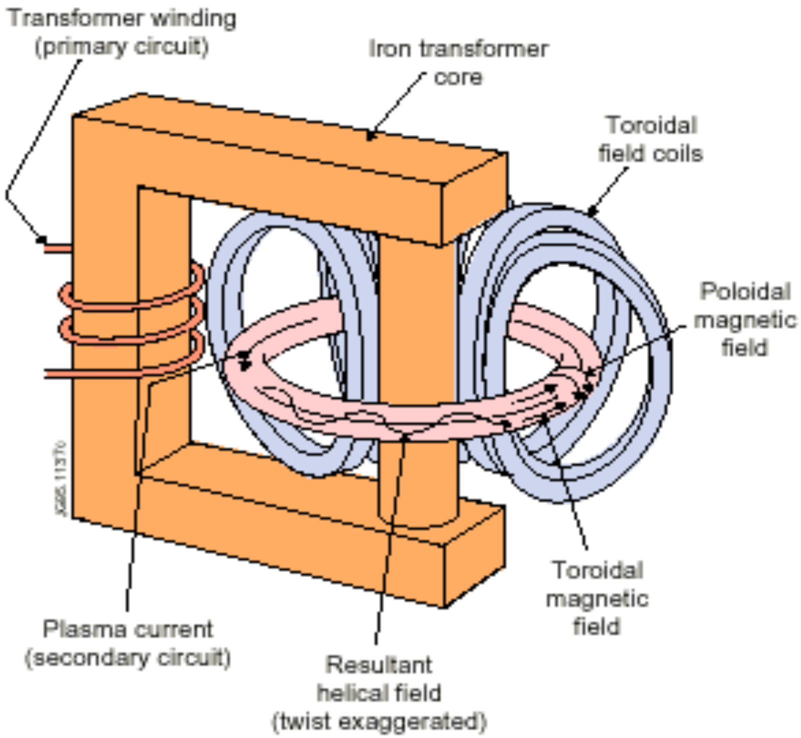
Il confinamento magnetico del



Le particelle cariche seguono orbite a spirale attorno alle linee di campo.

Il plasma si diffonde attraverso il campo per collisioni o effetti di deriva.

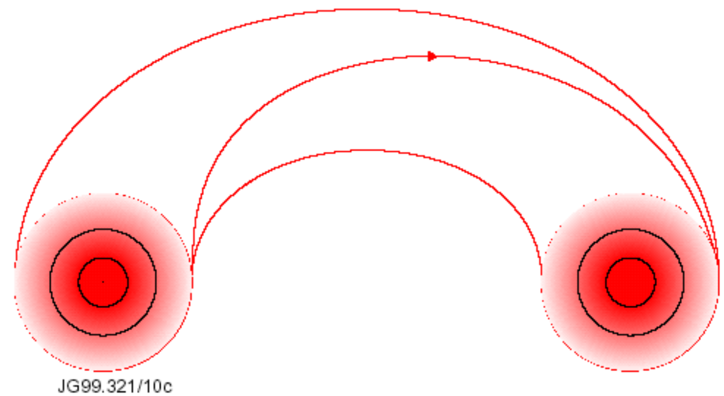
Scheme of a tokamak



$$q = \frac{5a^2 B}{RI}(1 + k^2/2)$$

In a plasma contaned in a toroidal device with axial magnetic field a current is induced by a transformer

A magnetic field results with elical field lines which close after a certain number of turns on surfaces called 'rationale surfaces'



Bpol~ Btoroidal/10;

Safety factor q=(number of toroidal turns /n poloidal turns)=

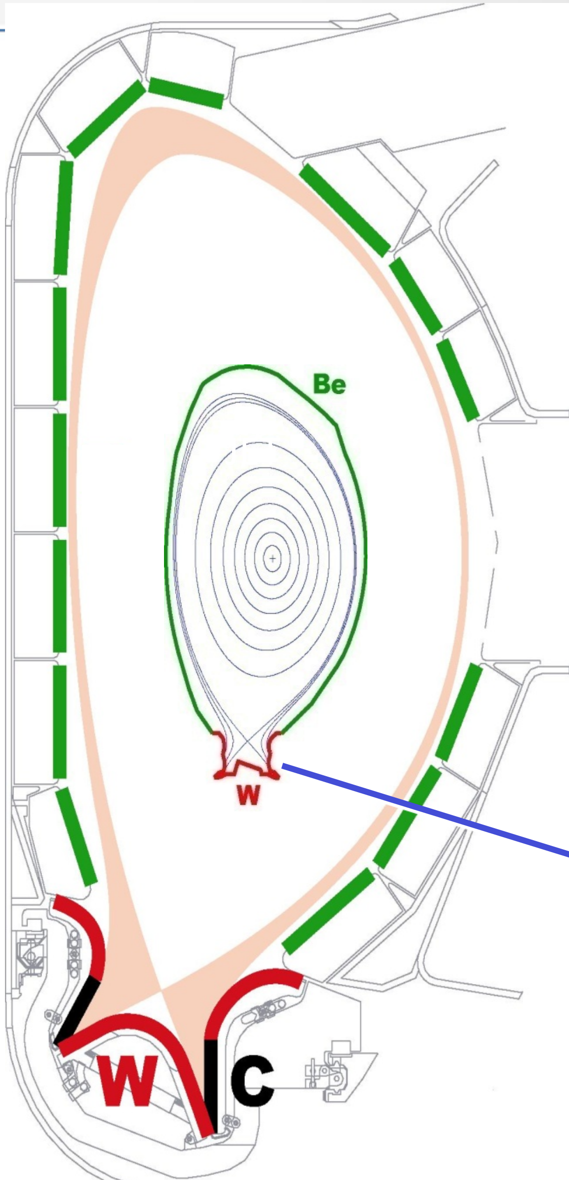
magnetic shear S=(q/r)(dq/dr)

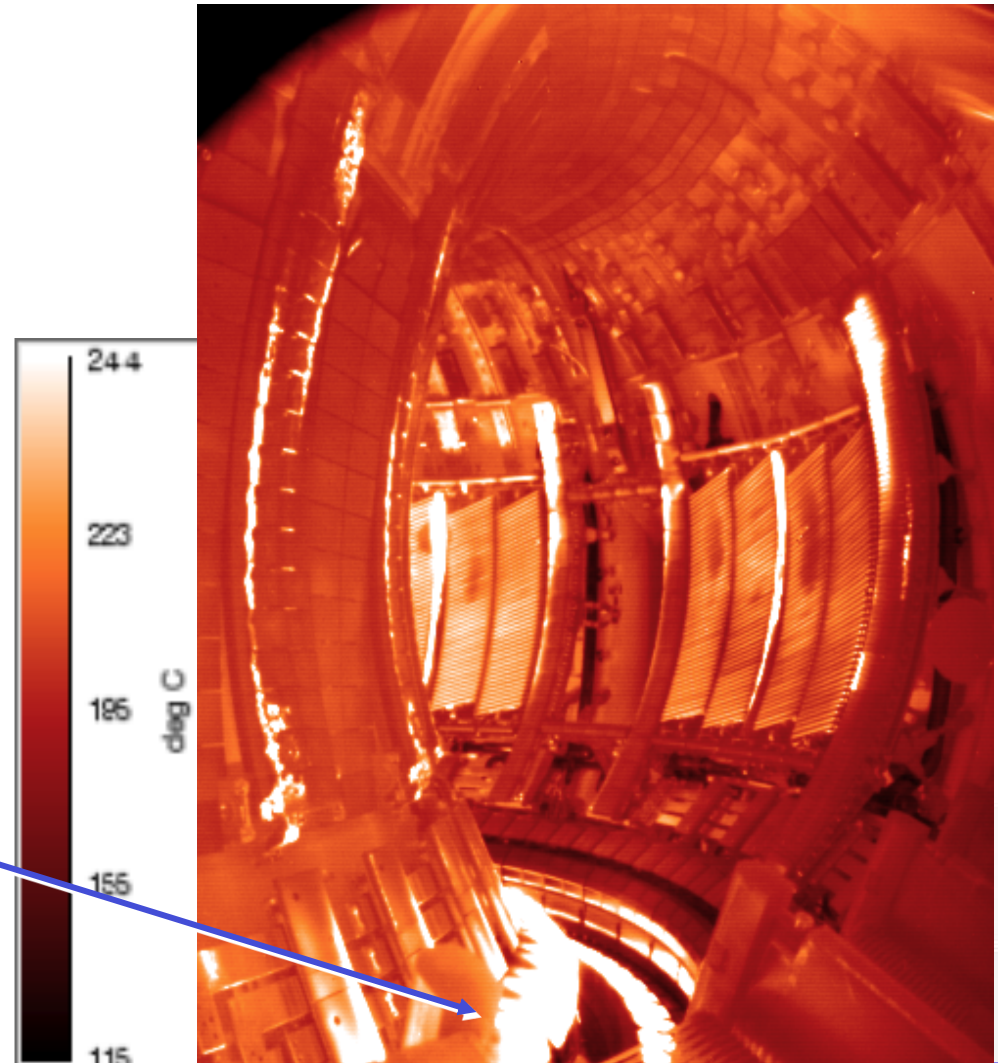
Where we are in MCF (Magnetic confinement fusion – Tokamak)

We gained experience :

- How to build and operate a tokamak pulsed (short pulses of the order of 10s) Q<1 machine , heated with NBI (Neutral beam) and RF (ECRH and ICRH) (~JET(EU))
- How to build a low temperature superconducting device pulsed (of the order of 100s) Q=1 machine , heated with NBI (Neutral beam) and RF (ECRH) (EAST (China) , TORE SUPRA(Fr), JT60SA(JA-EU))
- *We start learning about High Temperature superconducting magnets in the context of MCF : this technology (which will be ready soon, see W H Fietz et al Fus Eng Des 2005, see also ARC project B N Sorbom et al Fus Eng Des 2015) will give access to the high magnetic field fusion neutron sources .*

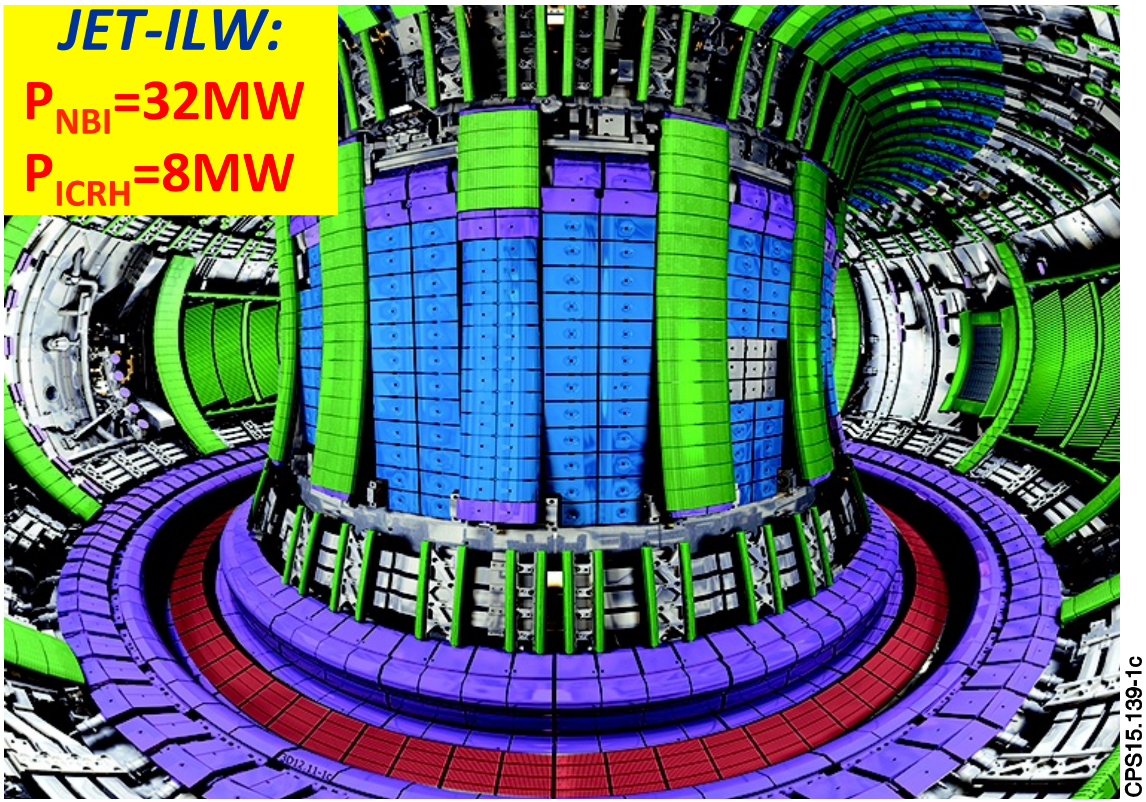
Sezioni non-circolari con “divertore”





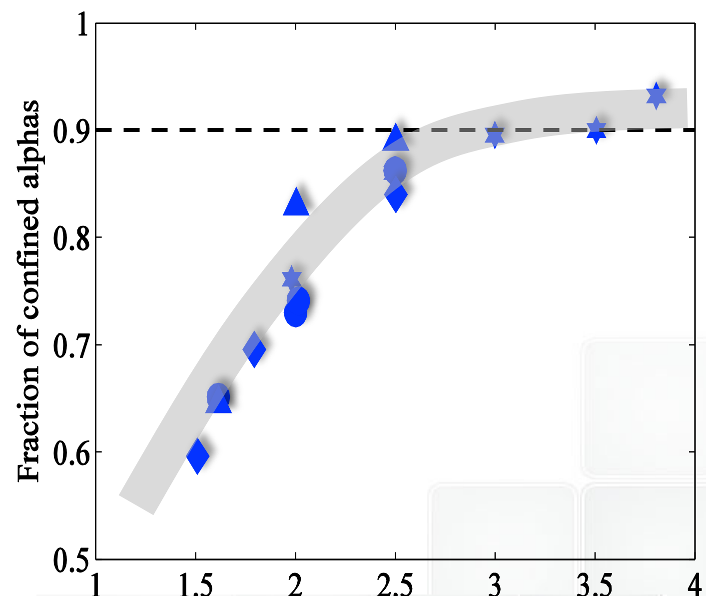
JET is designed for fusion power studies in support of ITER

<u>1975 JET design proposal</u>: *"...describes a large Tokamak experiment, which aims to study plasma behaviour in conditions and dimensions approaching those required in*



Bulk Be PFCs | Be−coated inconel PFCs
Bulk W | W−coated CFC PFCs

JET is capable of confining 90% of alpha particle for Ip above 2.5MA



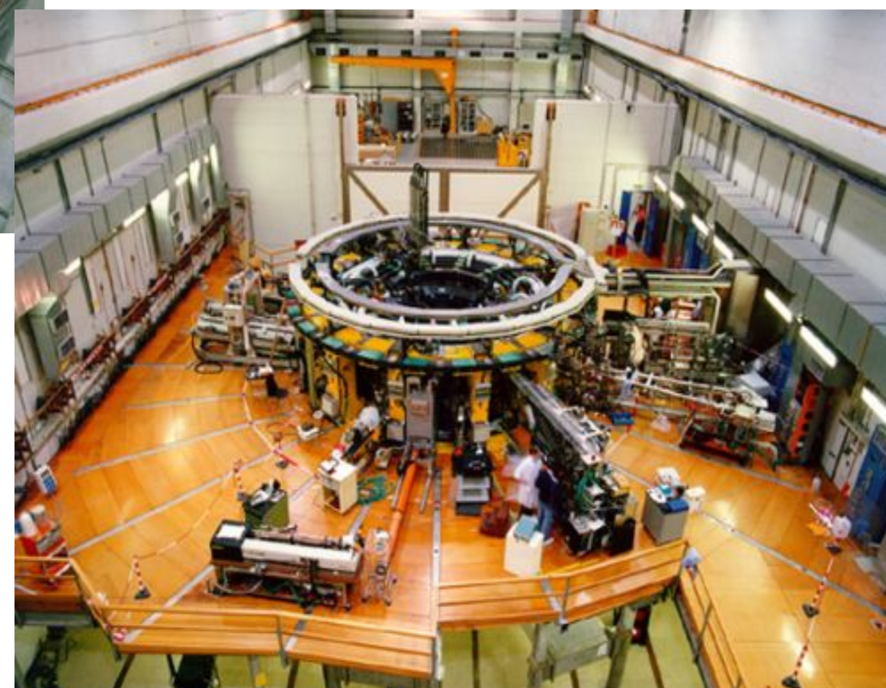
Esperimenti sulla fusione in Italia

FTU (Frascati)

Raggio maggiore R= 0.94 m

Raggio minore a = 0.3 m

Campo magnetico B = 8 T



RFX (Padova)

Raggio maggiore R= 2 m

Raggio minore a = 0.5 m

Campo toroidale B = 0.7 T

Definition of plasma

A system of ions and electrons generally faraway from equilibrium can be defined as a plasma

Approximately described by a maxwellian micro-canonical ensemble

In fusion plasmas a temperature close to 100million degree kelvin (10keV) is approached

Spatial scale

Debye Length

In a plasma electron are attracted by ions and shield its electrostatic field

Similarly an electron at rest repels other electrons and attracts ions .

This effect alters the potential close to a charged particle which results:

$$\phi(r) = \frac{q}{r} e^{-r/\lambda_D}$$

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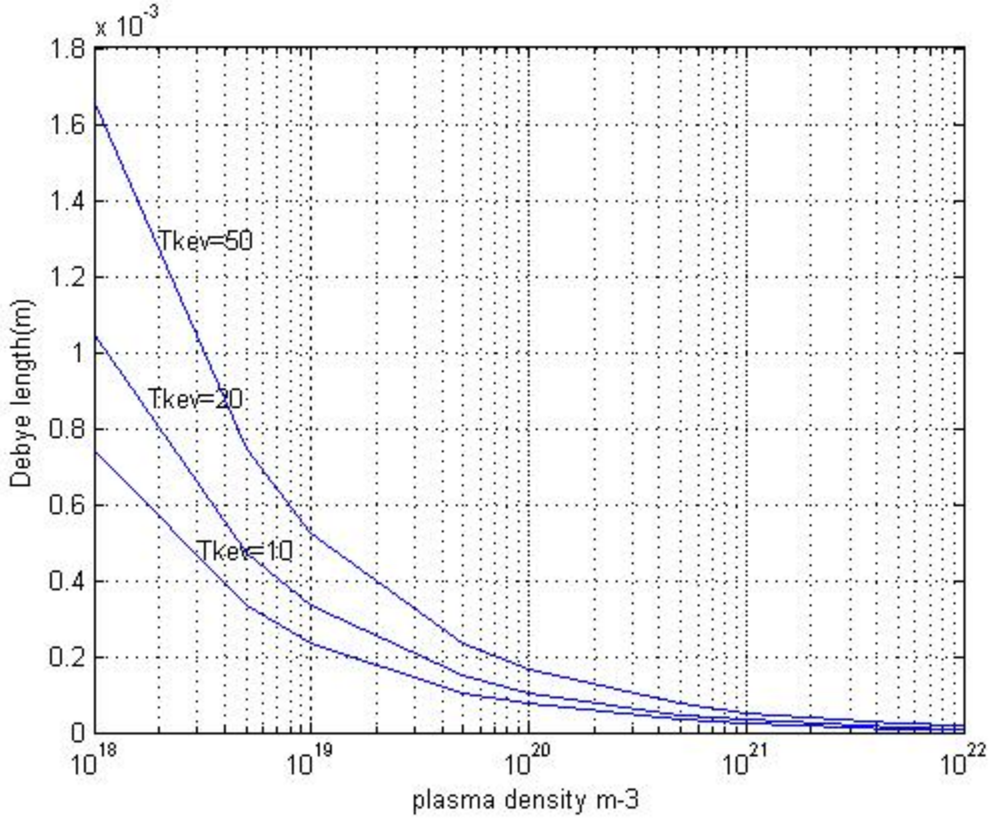
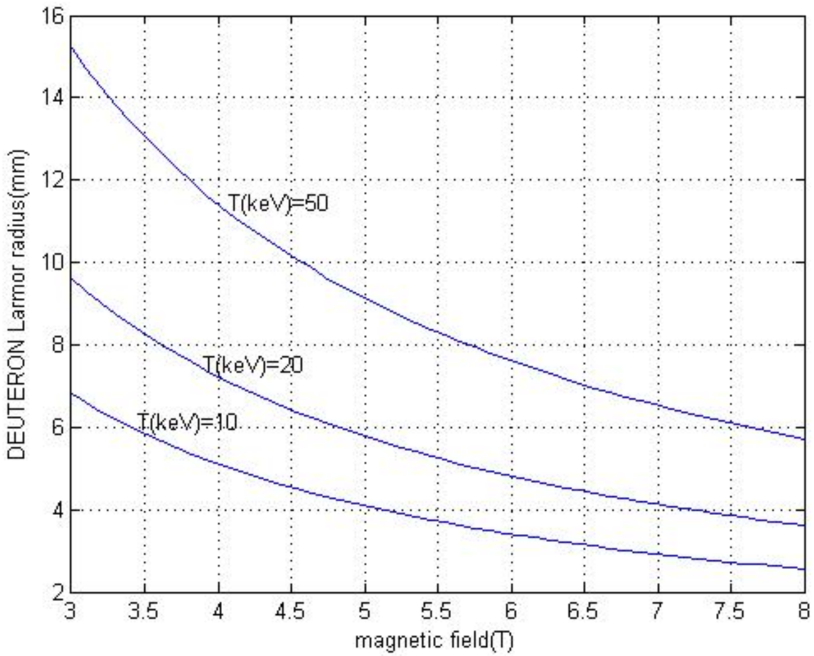
The Debye length

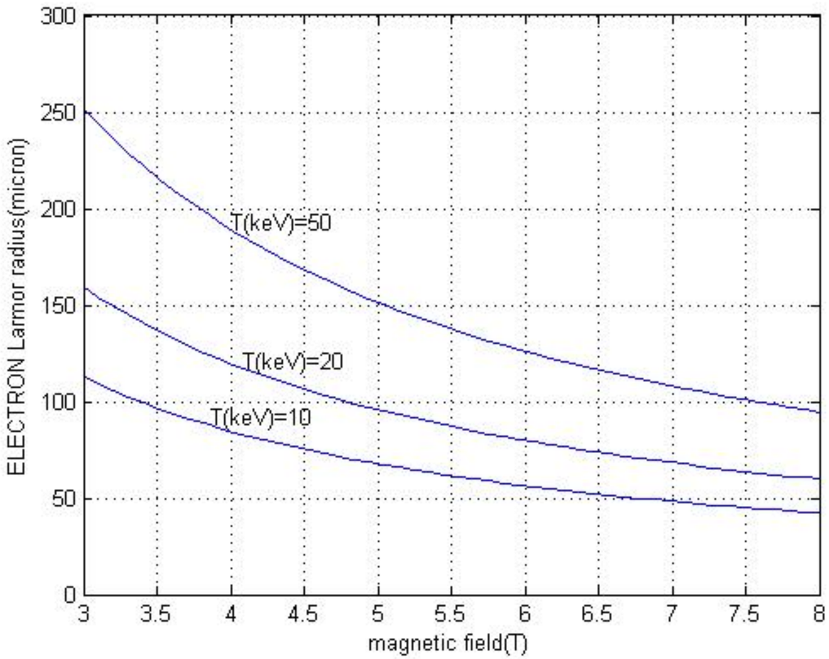
$$\lambda_D = \left(\frac{\varepsilon_0 T}{n_e c^2}\right)^{1/2} =$$

$$2.35\ 10^5\ \left(\frac{T_{ekeV}}{n_e}\right)^{1/2}$$



Spatial scale:Larmor radius





$$\rho e = \frac{v_{\perp e}}{B} = 1.07\ 10^{-4}\ (T_{ekeV}{}^{1/2} / B)\ m$$

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FUSION REACTIONS

Deuterium(1proton, 1neutron) and Tritium(1 proton , 2 neutrons)

Fusion reaction :D+T-> He^4 (3.5MeV, 2 protons, 2 neutrons) + one neutron (14 MeV) + 17.6 MeV energy

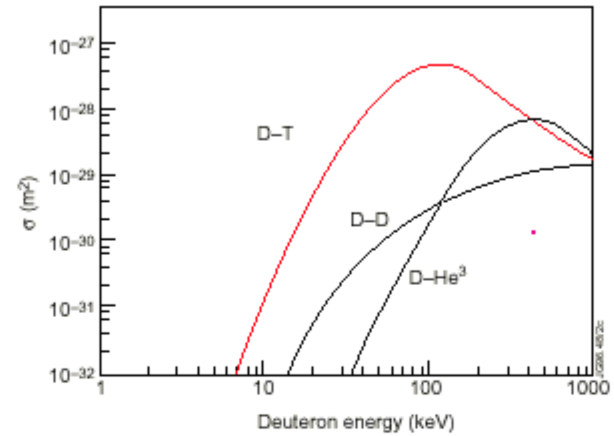
Etot_fus(DT)=17.6MeV

Ealfa/Efus = 1/5

He4 = alpha particles

In a reactor for the regeneration of tritium a ‘blancket’ is used with Litium

Li6 + n -> T + He4 + 4.8Mev



$D + D \rightarrow T$ (1.0MeV) $+ p$ (3.0MeV)

$D + D \rightarrow He^3$ (1.0MeV) $+ n$ (2.4MeV)

$D + T \rightarrow He^4$ (3.5MeV) $+ n$ (14.1MeV) easiest

$D + He^3 \rightarrow He^4$ (3.7MeV) $+ p$ (14.7MeV)

Power from fusion in magnetic confinement(I)

Power Density for fusion D-T(50%-50%) :

$P_{fusion}=1/4 \ \ n_{ion}^{2} <\sigma v> E_{fusion}$

$<\sigma v> \sim 1.1\ 10^{-24}\ T_{keV}^{2}\ m^{3}\ s^{-1}$.(range 10-20keV).

$P_{\alpha} = \ P_{fusion}/5$

$Q= P_{fusion}/P_{external} = P_{fusion}/P_{heating}$

Q=1, 20% of the plasma heating due to alpha particles

BETA LIMIT

Due to the interaction at long range of the coulombian forces the orbits of single particle are only part of the story: Plasma can be modelled as a fluid depending upon spatial scales considered.

The combination of fluid equations and Maxwell equations leads to MHD eqs.

Law of Troyon : $\beta = \beta_N * (I/aB)$

Limit $\beta_{NMAX} \sim 4\, li \sim 3.5$,

li =inductance internal of the discharge

density limit n_{GR}

The density limit in a tokamak is proportional to the plasma current I_P and it depends upon minor radius a:

$n_{GR}(10^{20}m^{-3}) = I_{P(MA)}/(\pi\, a^2)$

For JET : I=2.5MA, a=1m , $n_{GR} \sim 0.8 10^{20} m^{-3}$

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MHD Equilibrium of a plasma

The concept of the magnetic confinement is essentially that : in the fluid approximation

The existence of plasma in a magnetic field of a tokamak is subjected to a balance between the pressure of the plasma and the forces due to the magnetic field $\vec{J} \times \vec{B} = \vec{\nabla} p$

$$J = current\ \ density$$

$$B = magnetic\ \ field$$

$$p = plasma\ \ pressure = nT$$

Consequences of the equilibrium equation

$$\vec{J}\times\vec{B}=\vec{\nabla}p \qquad \Rightarrow \quad \vec{B}\bullet\vec{\nabla}p=0 \quad \textit{there is no pressure gradient along } \vec{B}$$

$$\Rightarrow \quad \vec{J}\bullet\vec{\nabla}p=0 \quad \textit{there is no pressure gradient along } \vec{J}$$

$$\Rightarrow \quad \textit{magnetic surfaces are surfaces with const pressure}$$

$J = \textit{current density}$

$B = \textit{magnetic field}$

$p = \textit{plasma pressure} = nT$

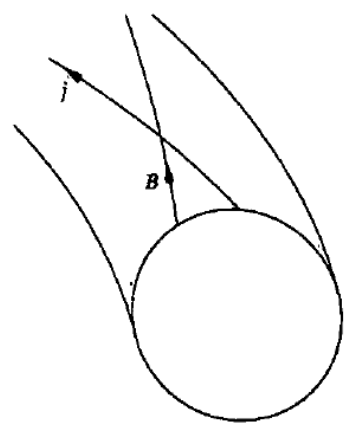


Fig. 3.2.2 Magnetic field lines and current lines lie in magnetic surfaces.

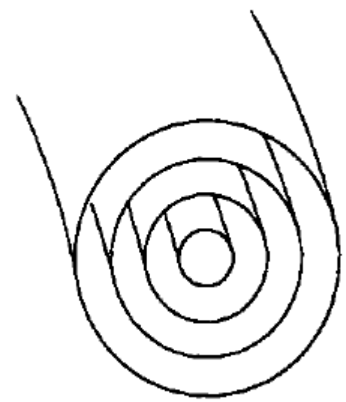
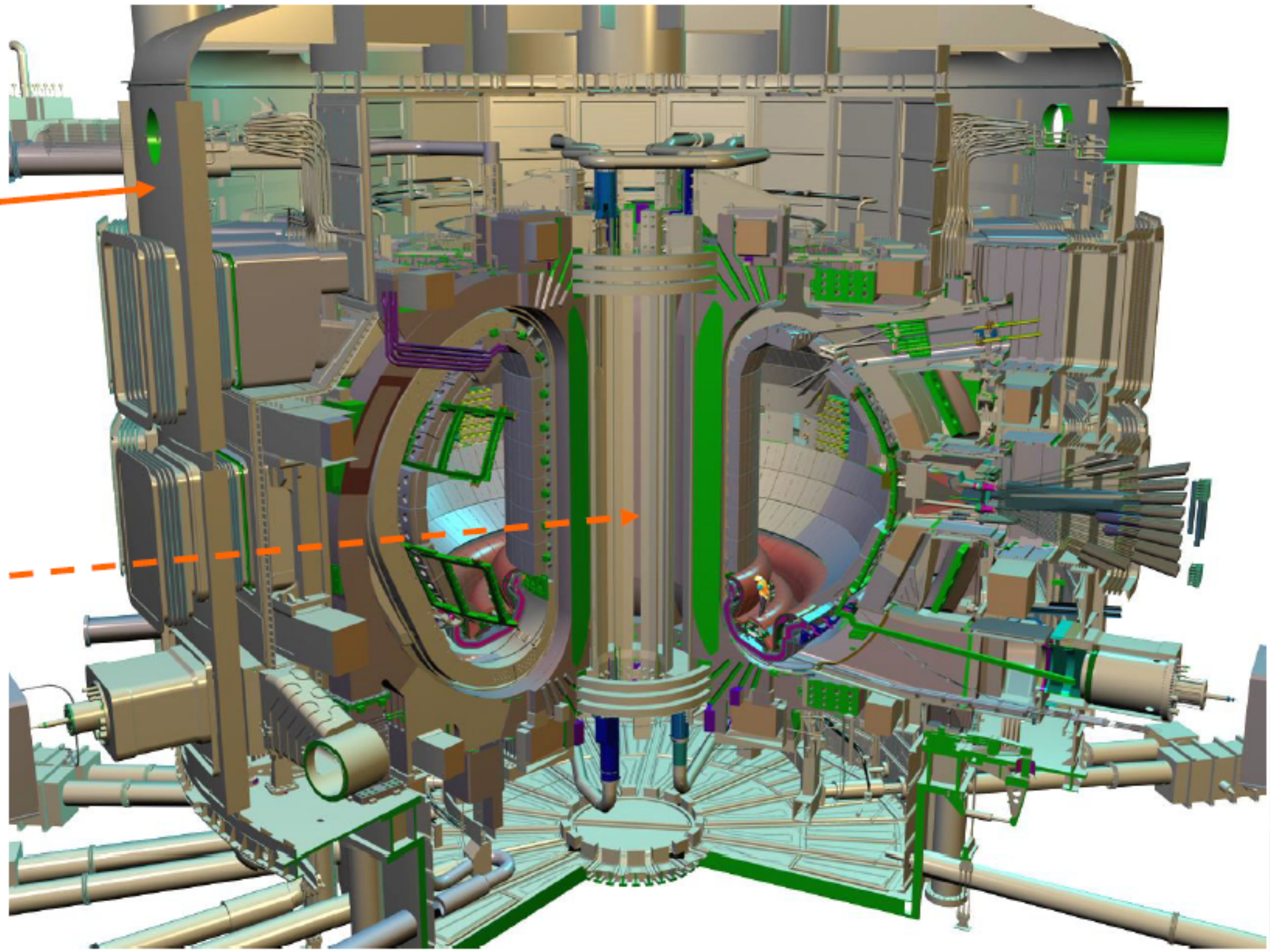


Fig. 3.2.1 Magnetic flux surfaces forming a set of nested toriods.

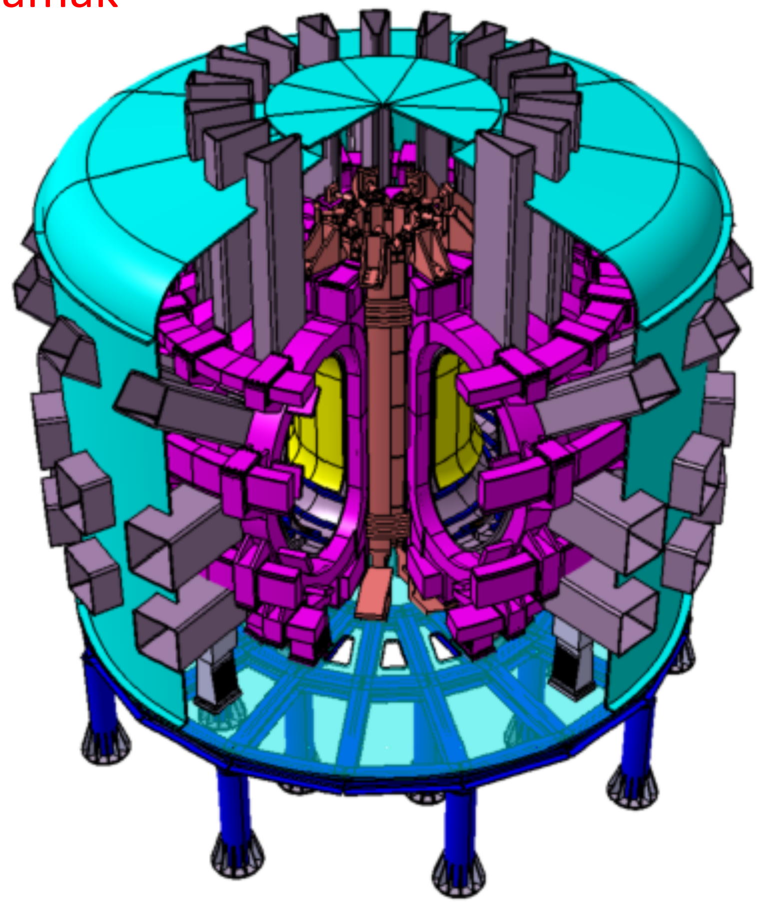
ITER load assembly - 2010

Cryostat

(Solenoid not shown)



DTT Divertor Test Tokamak



Main DTT parameters

R (m)/a(m)	2.11/0.64 SN – 2.14/0.65 DN
A	3.3 SN/DN
Vol (m^3)	≈28
I_p (MA)	5.5
B_T (T)	6 @ R_0
Coil currents margins [% with respect to nominal]	+ 5% on TF and CS +10% on PF and in-vessel coils
Neutron production rate, Sn (n/s)	1.2-1.5 10^17 DD + 1% DT
Maximum dwell time for high performance	3600
Nominal repetition time after disruption (s)	3600
Number of shots per day	5-10
Days of operation per year	100
Years of operation	25
Number of max shots	25000

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Confinement and Plasma scenarios for Fusion machines
Introduction and some results

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Power from fusion in magnetic confinement(I)

Power Density for fusion D-T(50%-50%) :

P_{fusion}=1/4 n_{ion}^2 $<\sigma\ v>$ E_{fusion}

$<\sigma\ v>$~1.1 10^{-24} T_{keV}^2 m^3 s^{-1}.(range 10-20keV).

P_{α} = P_{fusion}/5

Q= P_{fusion}/$P_{external}$ =P_{fusion}/$P_{heating}$

Q=1, 20% of the plasma heating due to alpha particles

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Power from fusion in magnetic confinement(II)

P_{fusion}=1/4 $n_{ion}^2 \langle \sigma v \rangle E_{fusion} \sim (nT)^2$

P_{fusion} = 1.08 $\beta^2 B^4$. MW/m^3.

β=2 nT / ($B^2/2\mu_0$) = [kinetic total pressure(ions+elettrons)] / magnetic pressure

For example. ($B^2/2\mu_0$) = 10000 pascal @ B=0.5T

Typical value of beta β~1-10%

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Comments on Pfus vs (limits of) beta

$P_{fusion} = 1.08\ \beta^2\ B^4.\ MW/m^3.$

$D+T \rightarrow \alpha(3.5MeV) + n(14.1MeV)$

$$Q_{\ fusion\ gain} = \frac{fusion\ \ power}{heating\ power}$$

$$at\ the\ steady\ state \Rightarrow P_{heat} + P_{alpha} = \frac{3nT_e}{\tau_E}$$

$$\tau_E = energy\ confinement\ \ time$$

$$Q = \frac{Pfus}{\frac{3nT_e}{\tau_E} - \frac{Pfus}{5}} = \frac{\overline{Q}}{1-(\overline{Q}/5)}$$

$$\overline{Q} = \frac{Pfus}{Ploss} = \frac{4\beta^2 B^4 \tau_E}{3}$$

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$$\overline{Q} = \frac{P_{fus}}{P_{loss}} = \frac{4\beta^2 B^4 \tau_E}{3}$$

β limited by MHD stability

τE limited by the turbulent transport

Gain Q versus geometry and plasma parameters

$$\tau_E(s) = 0.0562 * I^{0.93} * B^{0.15} * \left(\frac{a}{R}\right)^{0.58} * R^{1.97} * n^{0.41} * P^{-0.69} * M^{0.19} * k_a^{0.78}$$

Scaling of confinement time

$$\beta = \beta_N \frac{I}{aB}; \quad \beta_N \leq 0.035$$

Beta limit

$$\overline{Q} = \frac{4}{3}\beta B^2 \tau_E = \frac{4}{3} * 0.0562 * \beta_N * I^{1.93} B^{1.15} * \left(\frac{R}{a}\right)^{1.39} * a^{0.97} * n^{0.41} * P^{-0.69} * M^{0.19} * k_a^{0.78}$$

$$nGR = \frac{I}{\pi a^2}$$

Density limit

$$\overline{Q} \leq 0.04686 * \beta_N * I^{2.34} * B^{1.15} * \left[\frac{R}{a}\right]^{1.39} * a^{-1.03} * P_{loss}^{-0.69} * M^{0.19} * k_a^{0.78}$$

The gain factor depends upon :

- **Geometry (a(minor radius) and aspect ratio R/a)**
- **Plasma current I,**
- **magnetic field B**
- **beta β_N.**

At a fixed geometry(R/a), magnetic field B and heating power P an increase of β_N and I of 10% → **33%.** increase of Q

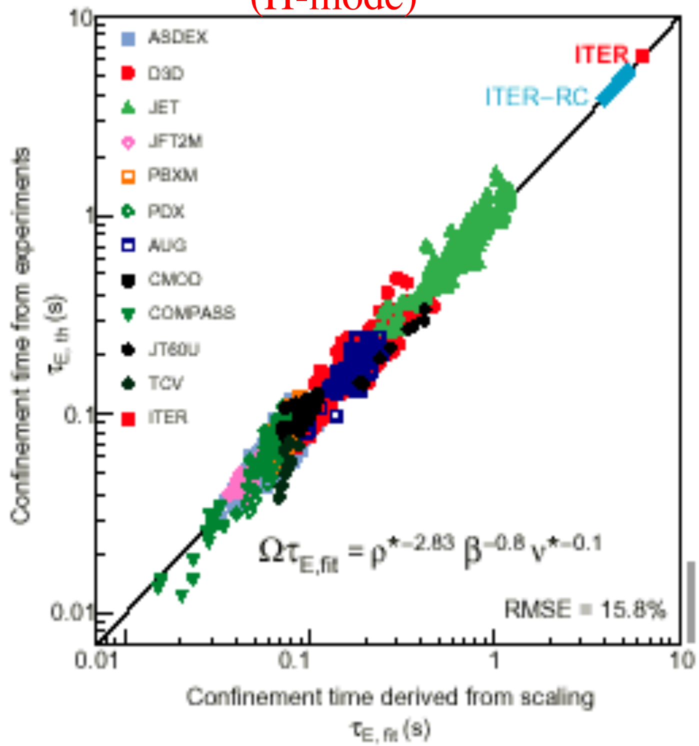
confinement time τ_E

The Time scale (τ_E) of energy loss for thermal conduction is defined by

$P_L = 3nT/\tau_E .= ((3/2)n_e T_e + (3/2)n_i T_i)/\tau_E$

(mean energy/ degree of freedom=1/2 T)

Scaling law confinement time for magnetic confinement tokamak devices (H-mode)



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Physics of confinement

Regimes of confinement are classified in relation to the spatial scales relevant :

i) Regimes where the relevant spatial scale is the plasma dimension are named L-mode (low confinement modes)

ii) Regimes where the Larmor radius is the fundamental relevant scale are named H-modes (High Confinement)

the transition to H-mode is linked to a threshold power

Transition to H mode

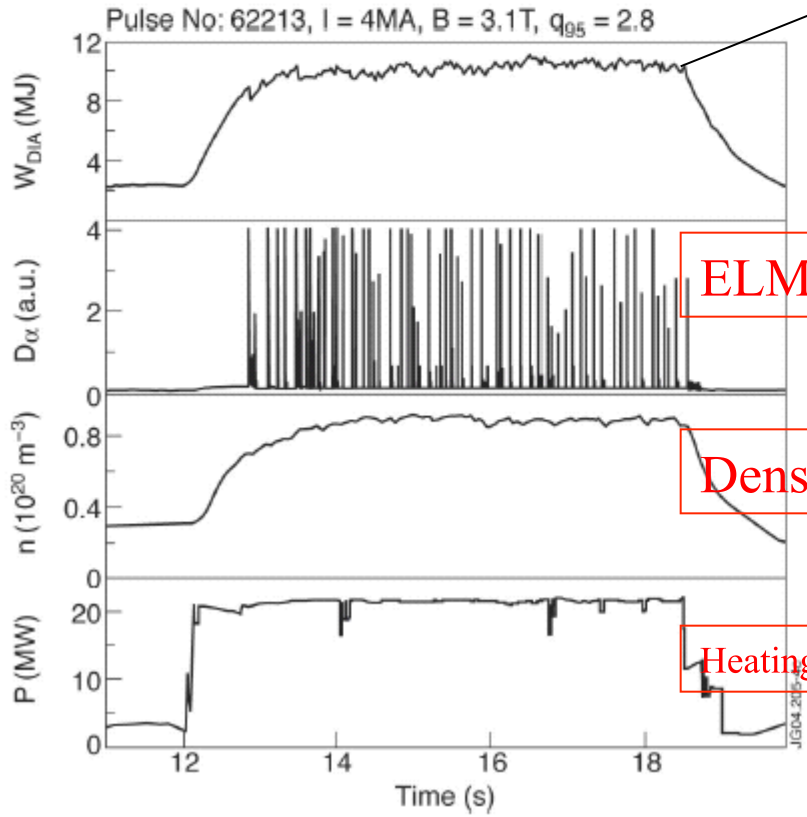
$P_{L\text{-}H} \sim C\ B_T\ n^{0.75}\ R^2.$

For example in JET $P_{L\text{-}H}$ ~8MW

H-mode has important characteristics

Example of a discharge in ELMy H-mode

ELMs (edge localized modes) corrispond to instabilities generated when locally the beta limit is reached



Internal Energy of the discharge

ELMs

Density

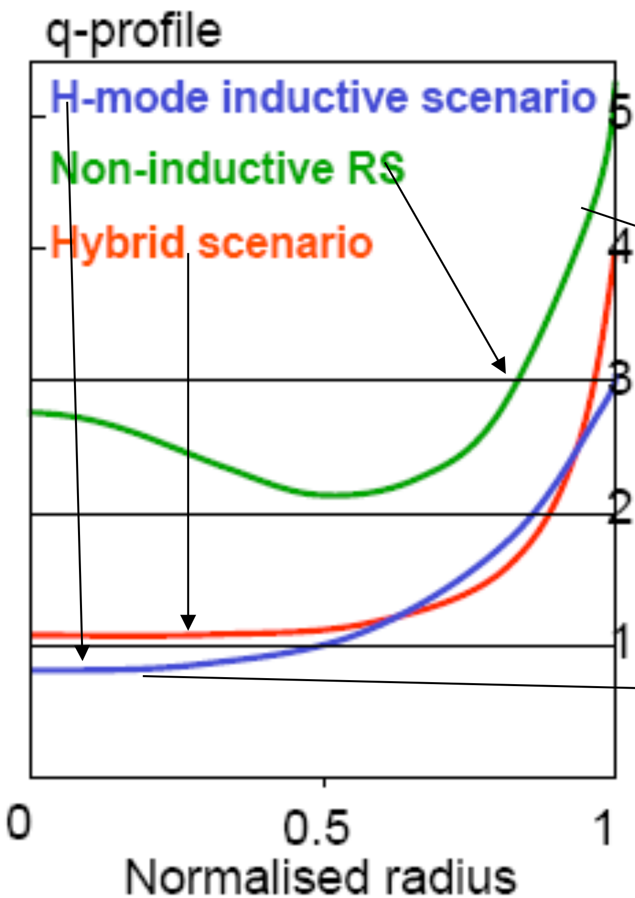
Heating power

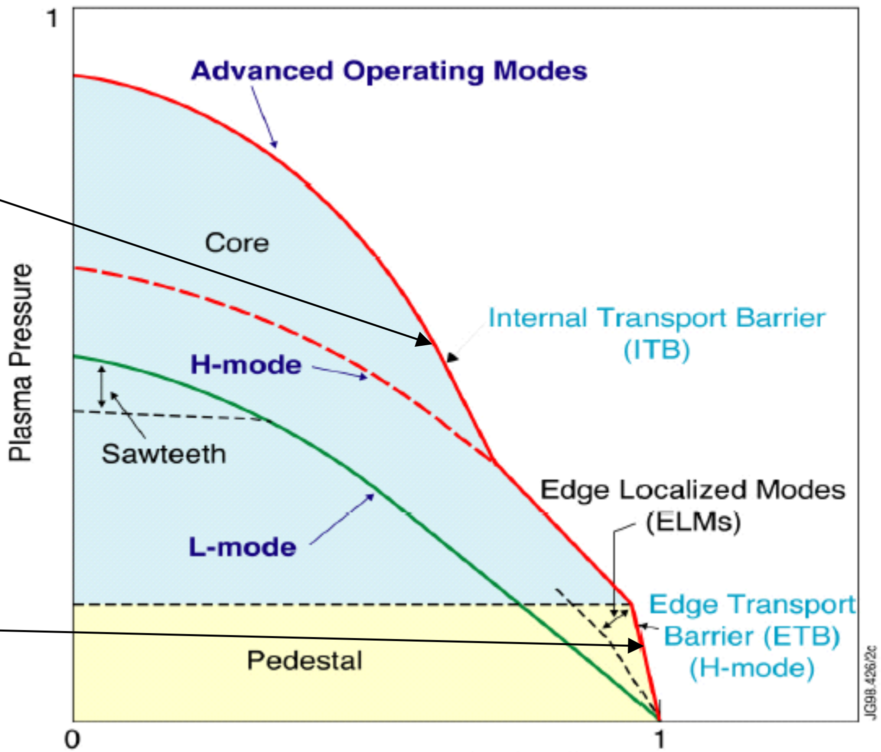
- Max. I_p=4MA, β_N=1.5, Type I ELMy H-Mode
- One of the highest D-D yields achieved on JET (Dec2003)
- ρ^* close to ITER $\rho^*/\rho^*_{ITER} = 1.7$

J Cordey et al. Conf. IAEA 2004

operational regimes in a tokamak:

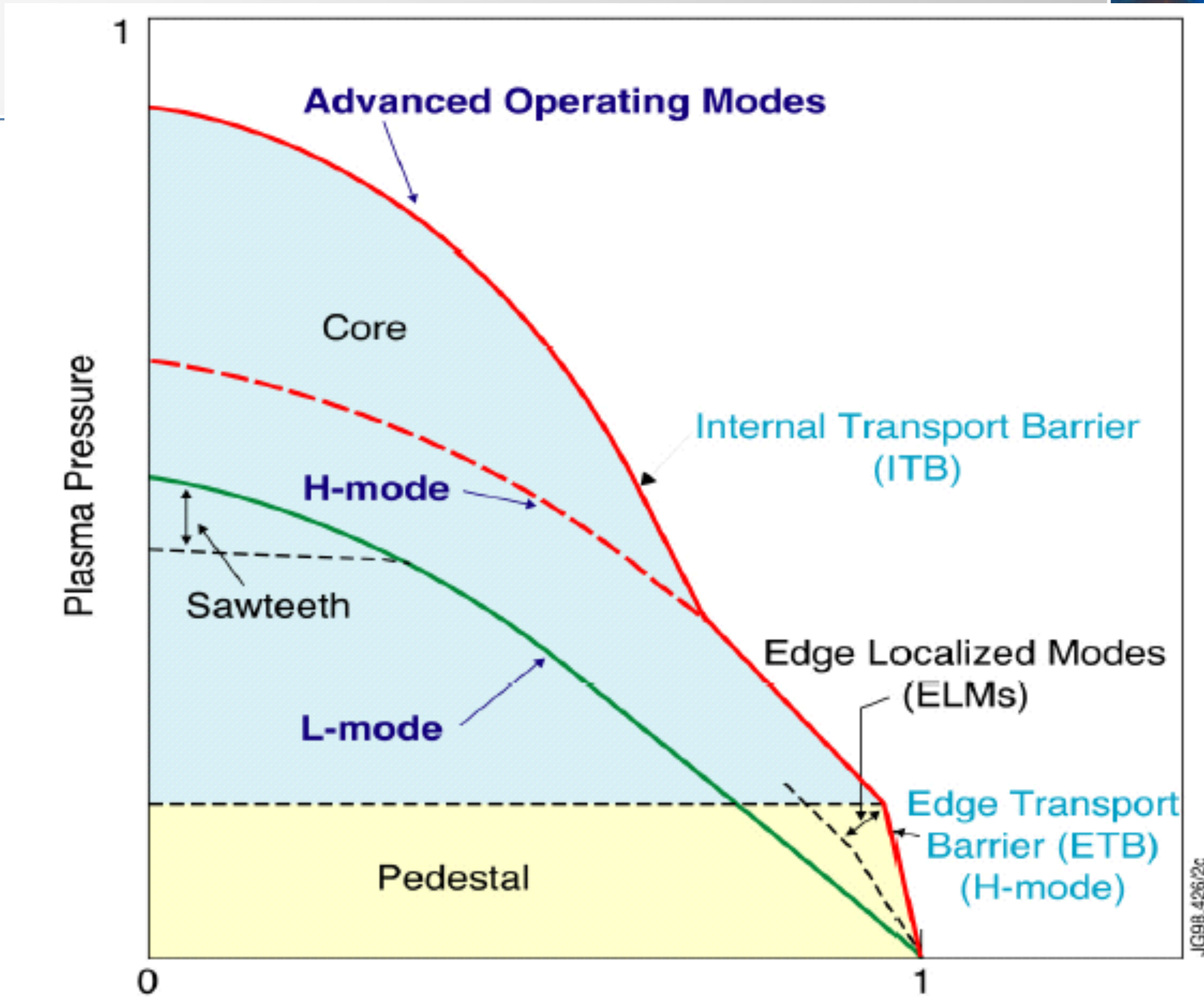
correspondence of current profiles ↔ pressure profiles





E Joffrin and X Garbet Conf IAEA 2004,

T Luce IAEA FEC Conference 2006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Advanced Operating Modes
Core
Plasma Pressure
Internal Transport Barrier
(ITB)
H-mode
Sawteeth
Edge Localized Modes
(ELMs)
L-mode
Edge Transport
Barrier (ETB)
(H-mode)
Pedestal
JG98.426/2c
0
1

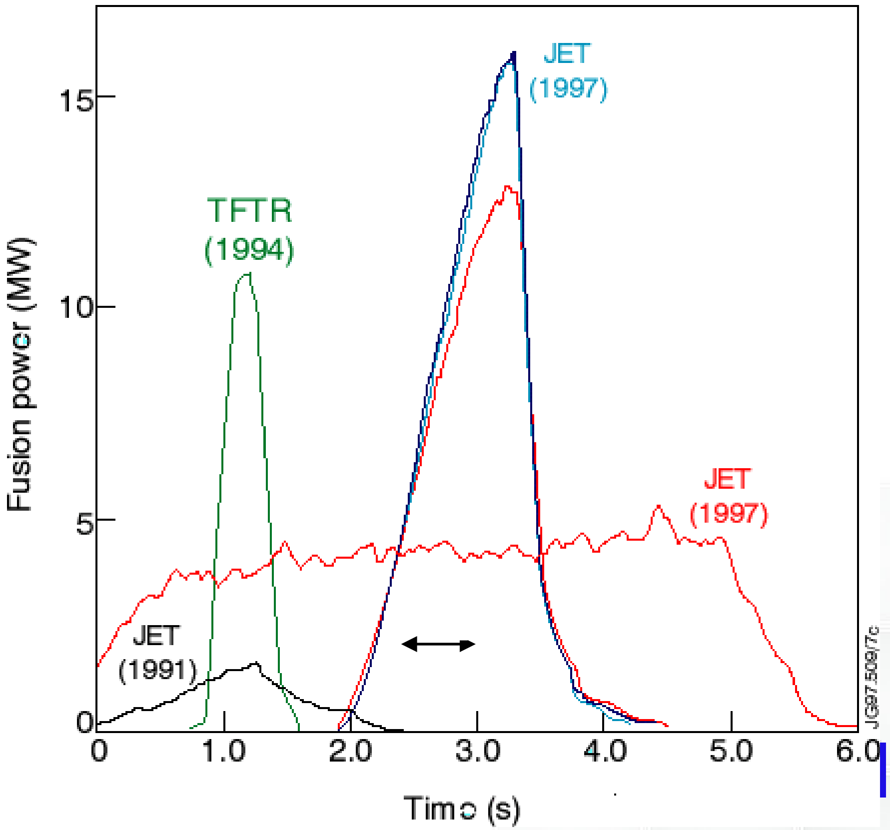
Results obtained at JET in campaign DTE1(1997)

21 MJ fusion energy

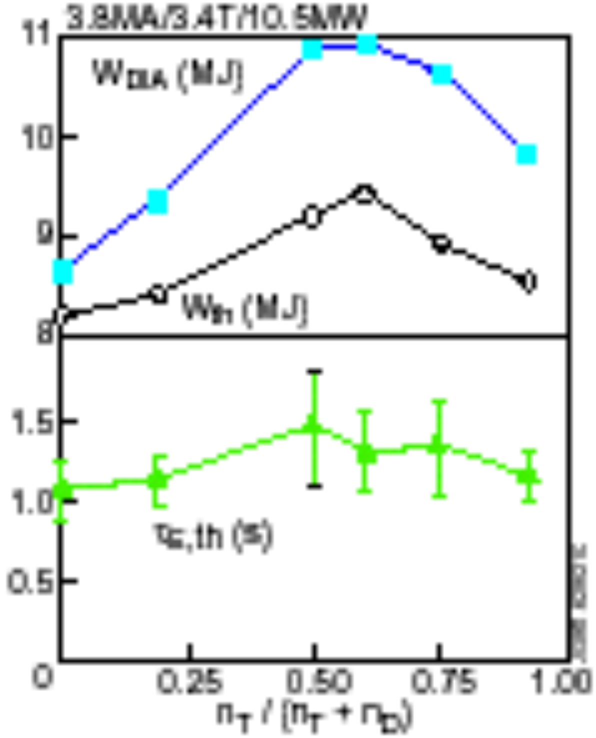
16 MW fusion power

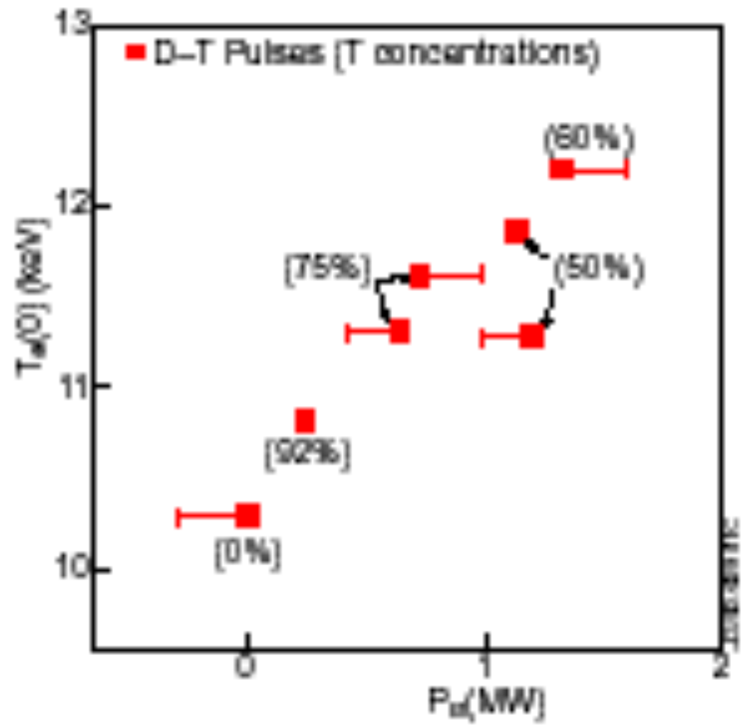
Q=pfus/pinjected

=0.65

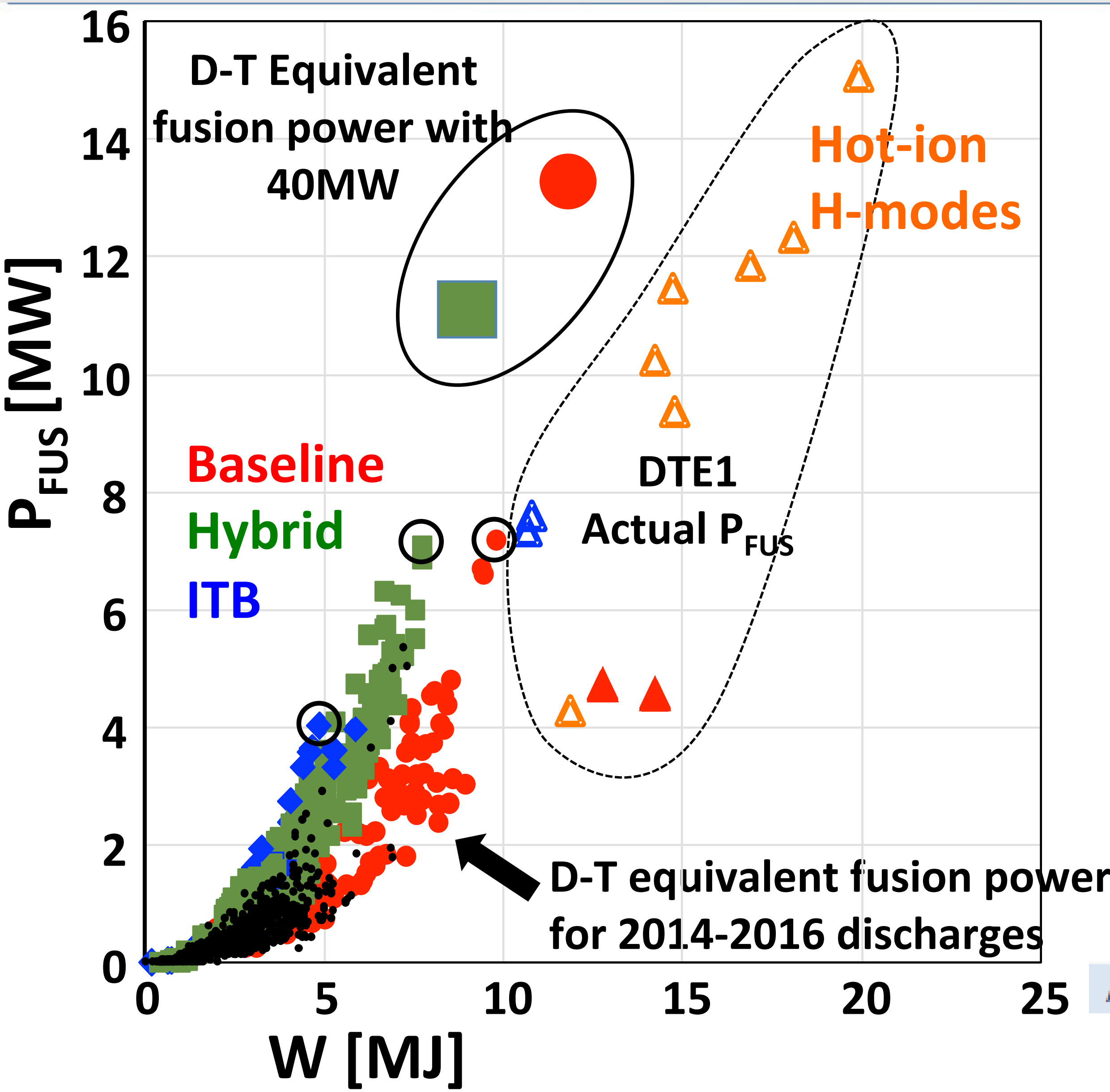


Heating due to alpha particles



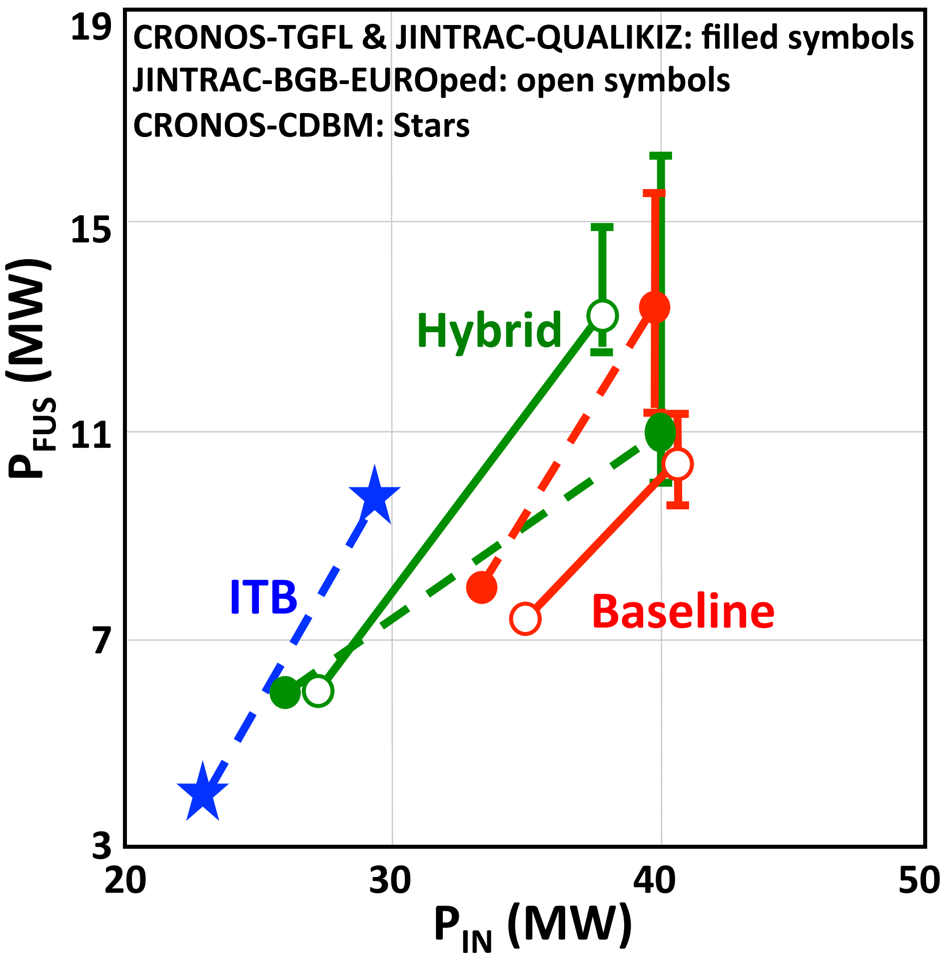


Prospects for 10 to 15MW of fusion power in stationary scenarios in JET-ILW



D-T equivalent fusion power

10-16MW of fusion power also predicted in D-T by first principle based modelling



Taking into account additional effect:

- **Isotope effect from ExB shear stabilisation or energy exchange.**
- **Alpha power contribution**
- **Auto-consistent modelling with core (BGB)-pedestal (EUROped) with no isotope effects.**

Uncertainty in P_{FUS} accounts for different plasma current and bootstrap models

J. Garcia, TH/3-1
F. Casson, TH/3-2
S. Saarelma, PPCF 2017

Associazione Euratom-ENEA sulla Fusione

Neutron sources for Fusion Fission hybrid reactors

Main messages

As is well known , we gained experience in building Q ~1 tokamak devices (TFTR , JET, JT60U) with pulses of the order of 10s
Determining the parameters of a neutron source for Fusion Fission application with FUSION Q~2-3 based on tokamak seems a relatively small extrapolation .

The present talk

- Starts from a *revisited* formulation of scaling laws for Fusion Reactors , which includes the concept of Kasomtsev-Lackner similarity extended to fusion plasmas

- Taking as reference a high performance JET DTE1 discharge (Q=0.55-0.6), the parameters of similar discharges are determined , and then the extrapolation to higher Q useful for FFH is attempted . Parameter sets are determined for tokamak FFH neutron sources.

- The parameters for a reduced performance tokamak useful for a PILOT experiment are defined

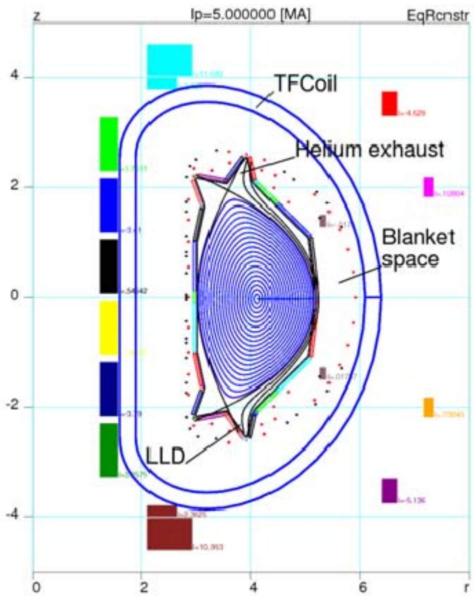
- POSSIBLE Figures of merits for FFH are introduced

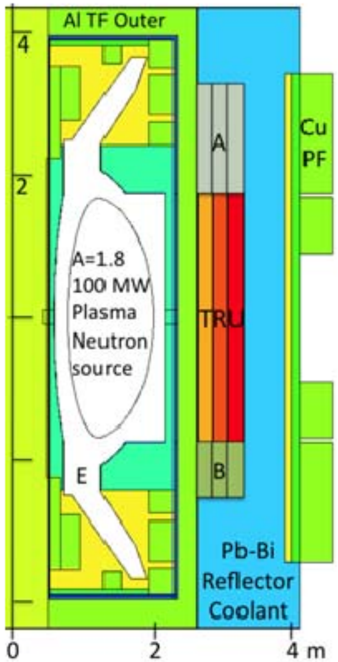
- The Technical Readiness Level of the tokamak subsystems is presented and discussed

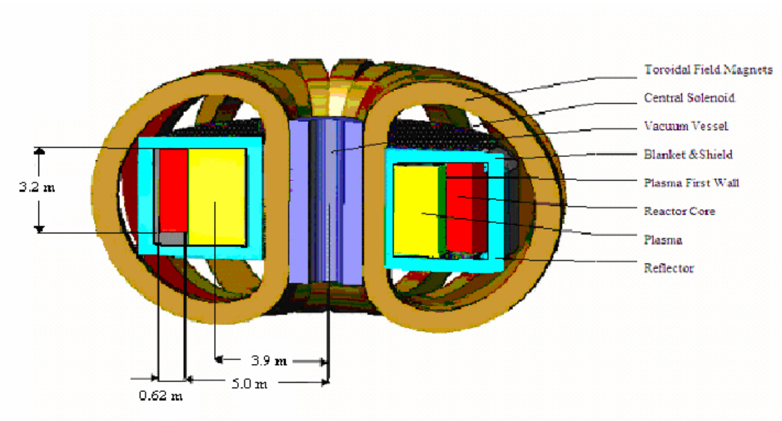
Outline

- 1. Reference TOKAMAK models for FFH : overview of parameters
- 2. Criteria for choosing a model for a MCF (Magnetic confinement fusion) neutron source.(Tokamak, DT)
- 3. Reference Technical Readiness level for FFH
- 4. Possible Figure of merits of a FFH
- 5. Conclusions

Tokamak Hybrid reactor models







(b)

Parameters of Tokamak based FFH Models considered (FDS and SABR)

Tokamak major radius	R0=3.75-4 m
Tokamak **high** aspect ratio	A=3.4-4 (**> ITER A=3**)
Medium magnetic field	B=5-6T (like ITER)
Medium-High current	IP=6-8MA(0.5*IP ITER)
Relatively low norm beta	β_N=2-2.8(like ITER)

HIPB98 =1-1.1

Main scenario : H-mode /Steady State Operation considered in SABR

Pulsed mode /SS

Fusion Power	100-180MW
Q=Pfus/Pheating=3	**(Q ITER =5-10)**

Scaling laws for reactor plasmas

1.$Q=Q_0$ fixed

2.$\tau_{SD} = \Lambda_{SD}\, \tau_E$.($\Lambda_{SD} \leq 1$) (slowing down time of alpha particles ≤ energy confinement time, this
Is true for JET DTE1, ITER, DEMO PPCS and EU-DEMO, Te≤20keV) ;
Λ_{SD}. Is NOT a constant but depends upon the device.

3.$P_\alpha = \Lambda_{LH}\, P_{LH}$ (Λ_{LH} >1.5) the alpha heating is sufficient to keep the plasma in H-mode
4.The energy confinement scaling law is ITER IPBy2 and the scaling for the power threshold for
The transition to the H-mode scaling $P_{LH} \approx A_{lh}\, B\, n^{3/4}\, R^2$.

We find that the scaling parameter linking equivalent fusion plasmas is :

S_{FR} =scaling par. for fusion reactors= $R\ B^{4/3}\ A^{-1}\ Q_0^{\ 1/3}$

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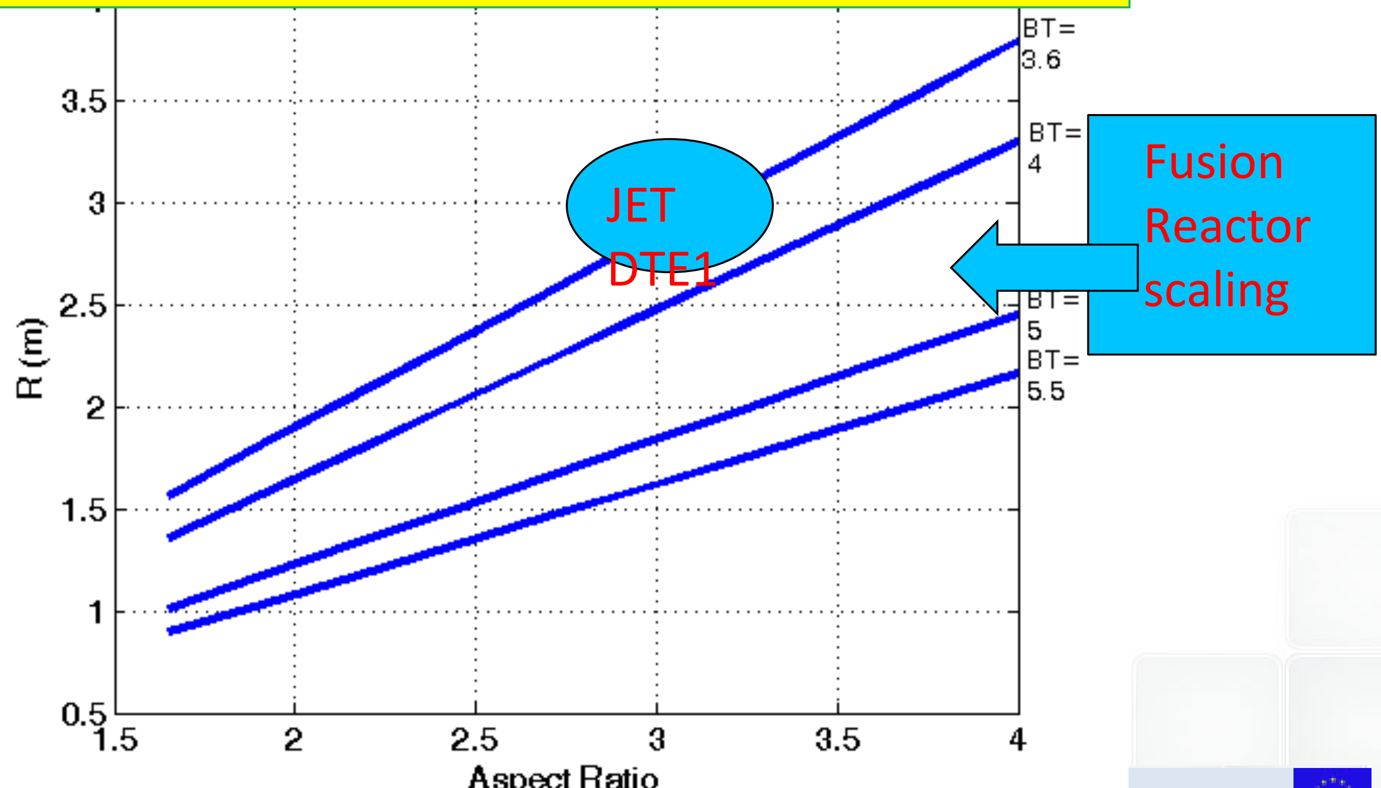
M Romanelli, F Romanelli, F Zonca – 28th EPS Funchal 2001, ECA vol 25 A(2001)697

Geometry of a reactor taking as ref JET DTE1 and Fusion Reactor scaling

$S_{FR} = R\ B^{4/3}\ A^{-1}\ Q_0^{1/3}$

Q=0.55 is achieved at B=5.5 T for A=3 and R≈1.6m
Q=0.55 can be achieved at B=3.6T for A=1.7 and R=1.6m



TRL (Technical Readiness Level)

TRL 1	Basic principles observed and reported.
TRL 2	Technology concept and/or application formulated.
TRL 3	Analytical and experimental critical function and/or characteristic proof-of-concept.
TRL 4	Technology basic validation in a laboratory environment.
TRL 5	Technology basic validation in a relevant environment.
TRL 6	Technology model or prototype demonstration in a relevant environment.
TRL 7	Technology prototype demonstration in an operational environment.
TRL 8	Actual Technology completed and qualified through test and demonstration.
TRL 9	Actual Technology qualified through successful mission operations.

F P Orsitto et al Nuclear Fusion 56(2016) 026009

TRL for Q=2 device 100s pulses

Subsystem	TRL
Superconducting magnets	4 - test needed in the high neutron flux environment
NBI(100keV)	4 - the long term reliability of NBI must be improved
ECRH (1MW gyr)	6 -steady state has been demonstrated for JT60SA and ITER gyrations
ICRH (1MW)	4 -possible problem of impurity 'injection'

TRL for Q=2 device 1000s pulses

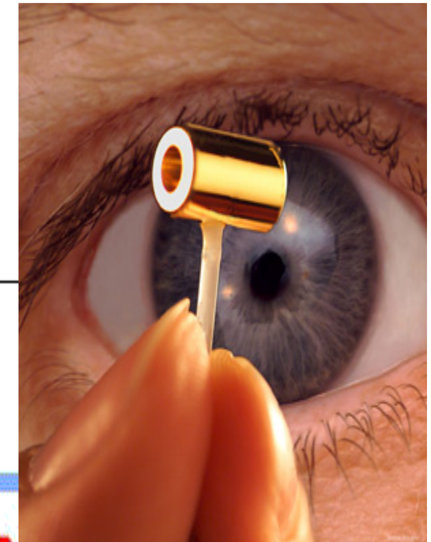
Subsystem	TRL
Superconducting magnets	4
NBI(100keV)	4 problem of long term reliability and neutron effects
ECRH (1MW gyr)	6 The launching systems must be tested in high neutron flux
ICRH (1MW)	4

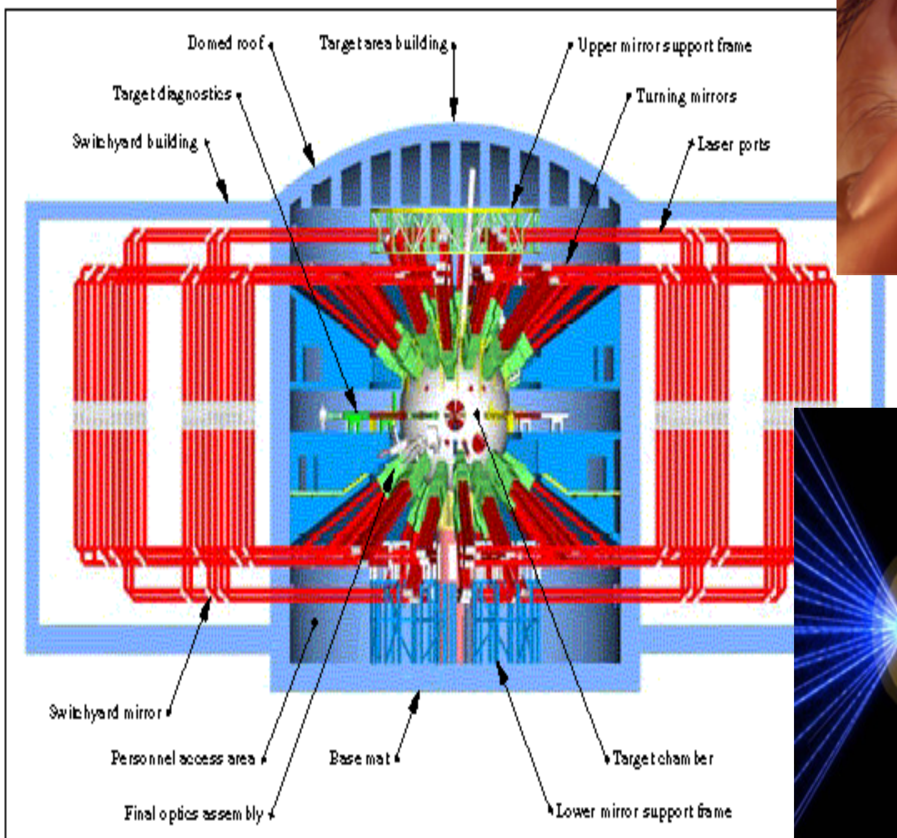
TRL plasma scenario

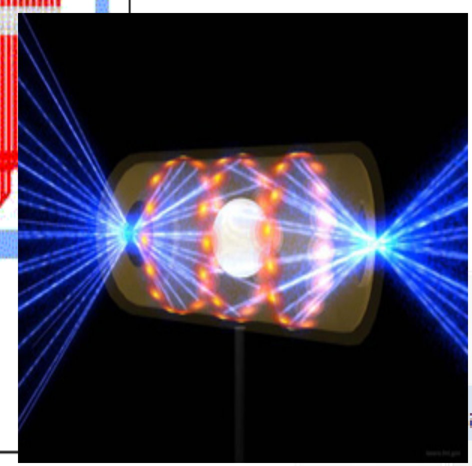
Scenario	TRL
H-mode	6 - H-mode to be demontrated in high power Q=2 device , demonstrated in Q=0.55 discharges in JET DTE1
Hybrid mode	4 – to be demonstrated in relevant environment (say JET DTE2)
Advanced mode	3 -4 to be demonstrated in laboratory

INERTIAL FUSION elements

Lasers - NIF and Megajoule:

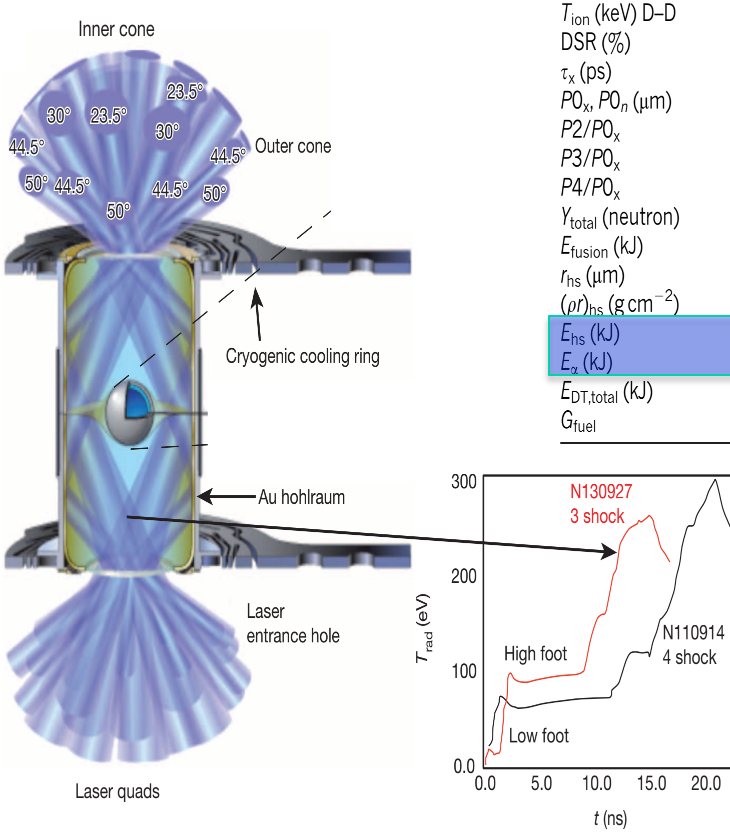






Extremely interesting results after NIC (and bad pubblicity)

Quantity	$N131119^{425\ TW}_{1.9\ MJ}$	$N130927^{390\ TW}_{1.8\ MJ}$
Y_{13-15} (neutron)	$(5.2 \pm 0.097) \times 10^{15}$	$(4.4 \pm 0.11) \times 10^{15}$
T_{ion} (keV) D–T	5.0 ± 0.2	4.63 ± 0.31
T_{ion} (keV) D–D	4.3 ± 0.2	3.77 ± 0.2
DSR (%)	4.0 ± 0.4	3.85 ± 0.41
τ_x (ps)	152.0 ± 33.0	161.0 ± 33.0
$P0_x$, $P0_n$ (μm)	35.8 ± 1.0, 34 ± 4	35.3 ± 1.1, 32 ± 4
$P2/P0_x$	-0.34 ± 0.039	-0.143 ± 0.044
$P3/P0_x$	0.015 ± 0.027	-0.004 ± 0.023
$P4/P0_x$	-0.009 ± 0.039	-0.05 ± 0.023
Y_{total} (neutron)	6.1×10^{15}	5.1×10^{15}
E_{fusion} (kJ)	17.3	14.4
r_{hs} (μm)	36.6	35.5
$(\rho r)_{hs}$ ($g\,cm^{-2}$)	0.12–0.15	0.12–0.18
E_{hs} (kJ)	3.9–4.4	3.5–4.2
E_{α} (kJ)	2.2–2.6	2.0–2.4
$E_{DT,total}$ (kJ)	8.5–9.4	10.2–12.0
G_{fuel}	1.8–2.0	1.2–1.4



This also shows that the MAIN problem towards ignition is REALLY the impact of hydro instabilities related to non-

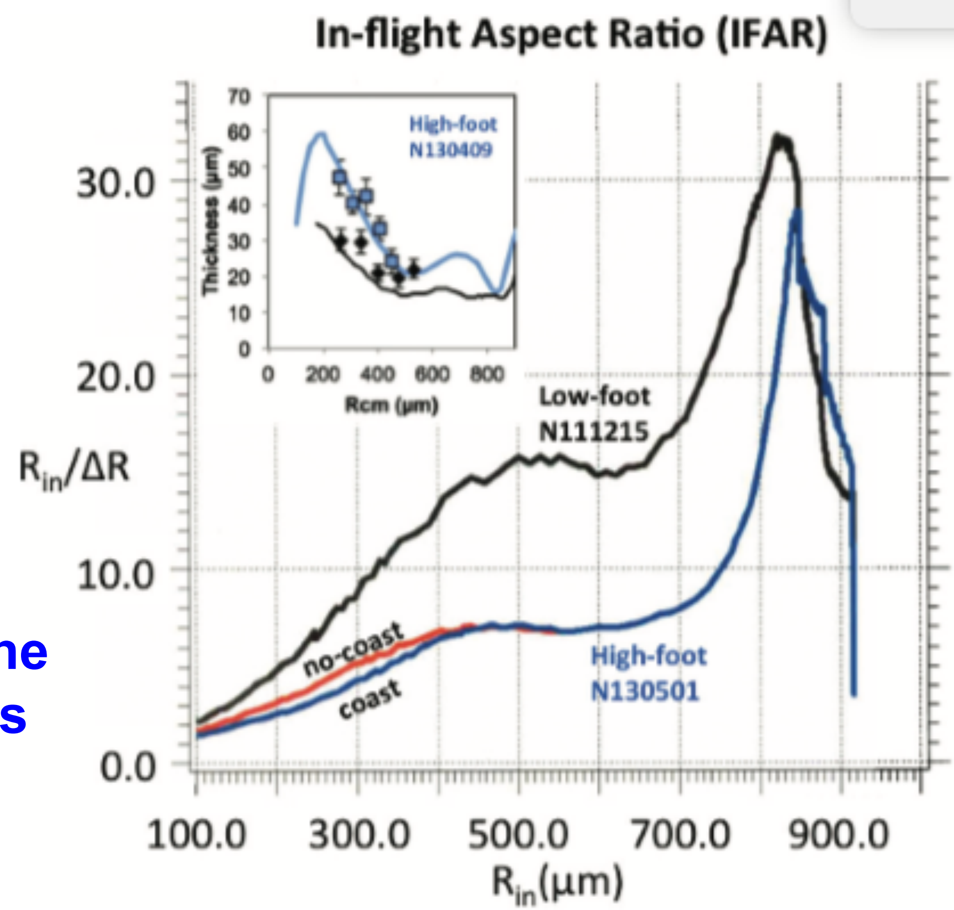
Rayleigh-Taylor stability of high foot shots

$$\gamma_{A-RTI} = \alpha_2(Fr,\nu)\sqrt{\frac{kg}{1+kL_\rho}} - \beta_2(Fr,\nu)kv_a$$

Increases the density gradient L

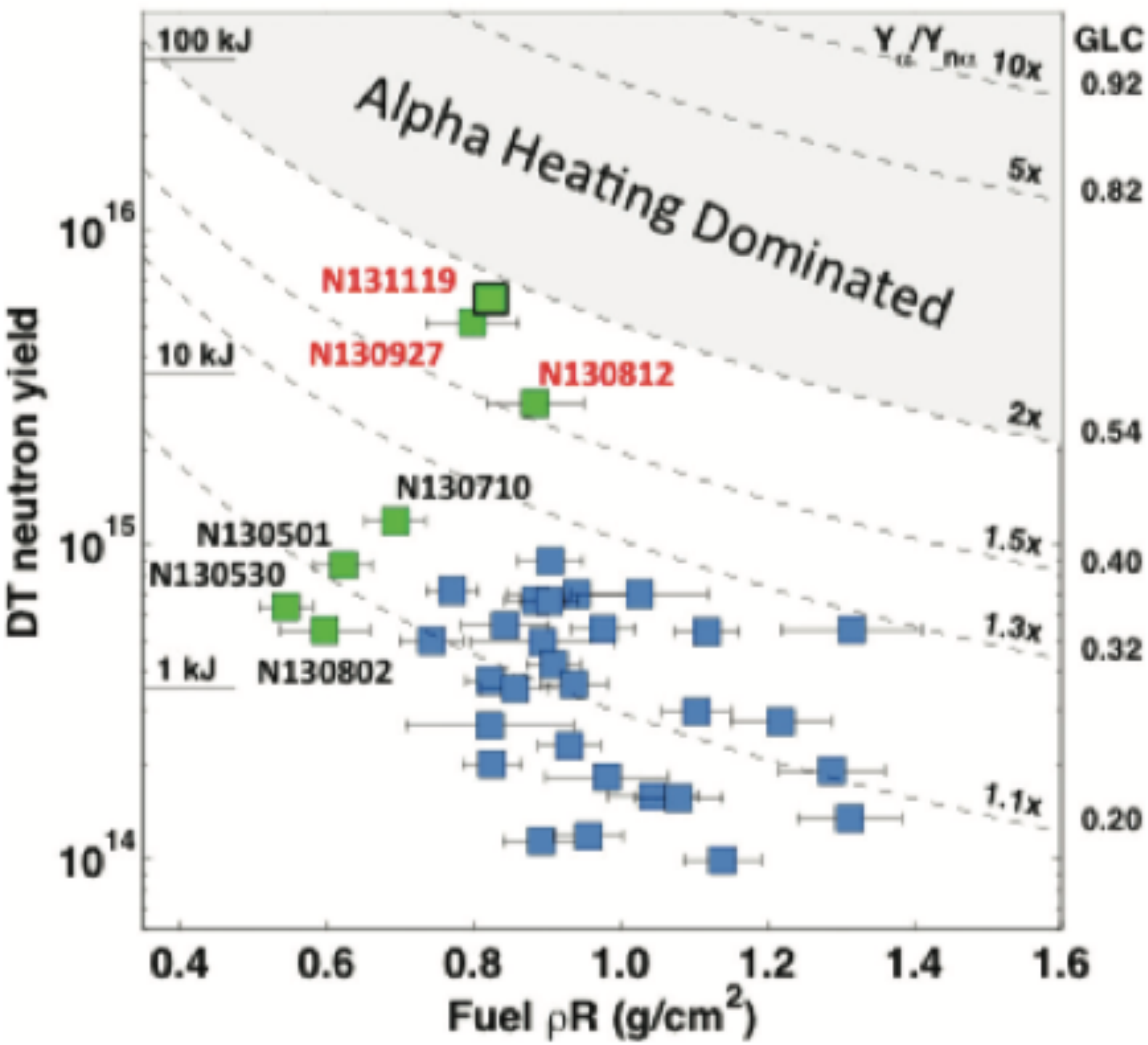
Increases the front velocity

Decreases the growth rate of RT



The best NIF implosions used the High-Foot laser pulse that drives stronger shocks in the "foot"

(to hot-spot)



High foot
Green

Low foot
Grey

056314-3 Hurricane *et al.* Phys. Plasmas **21**, 056314 (2014)

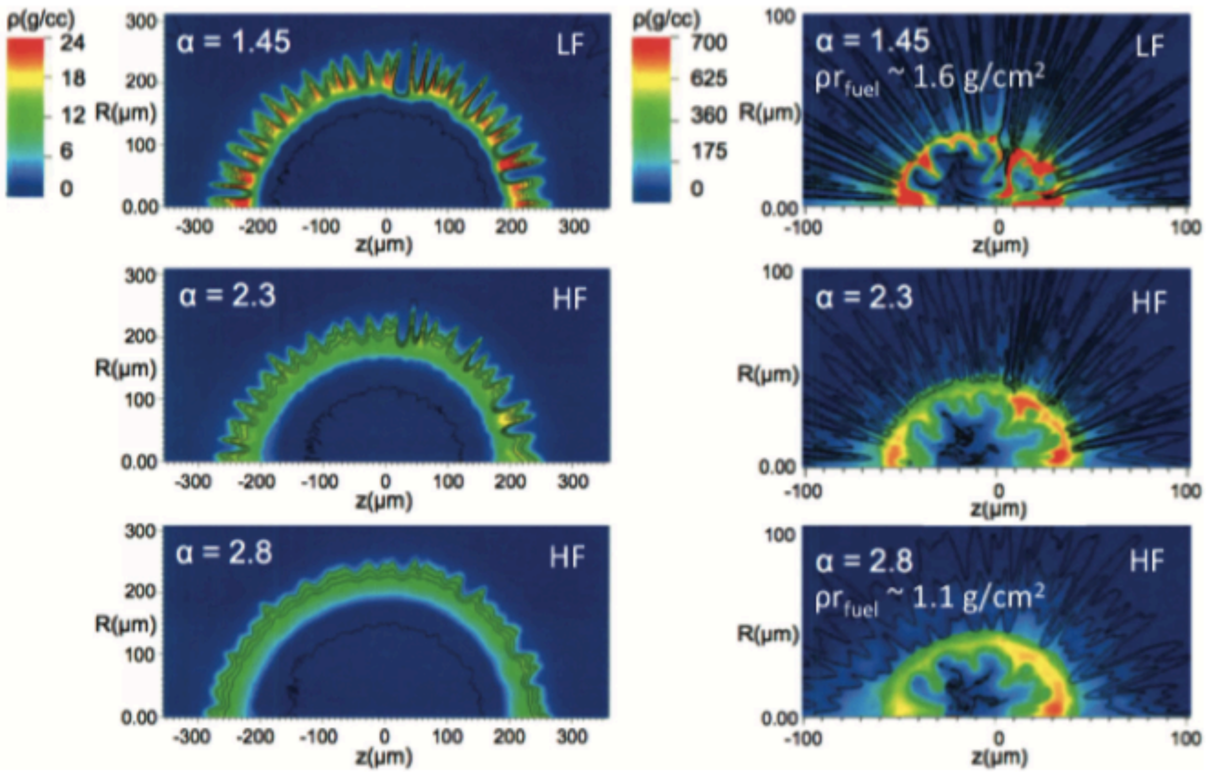
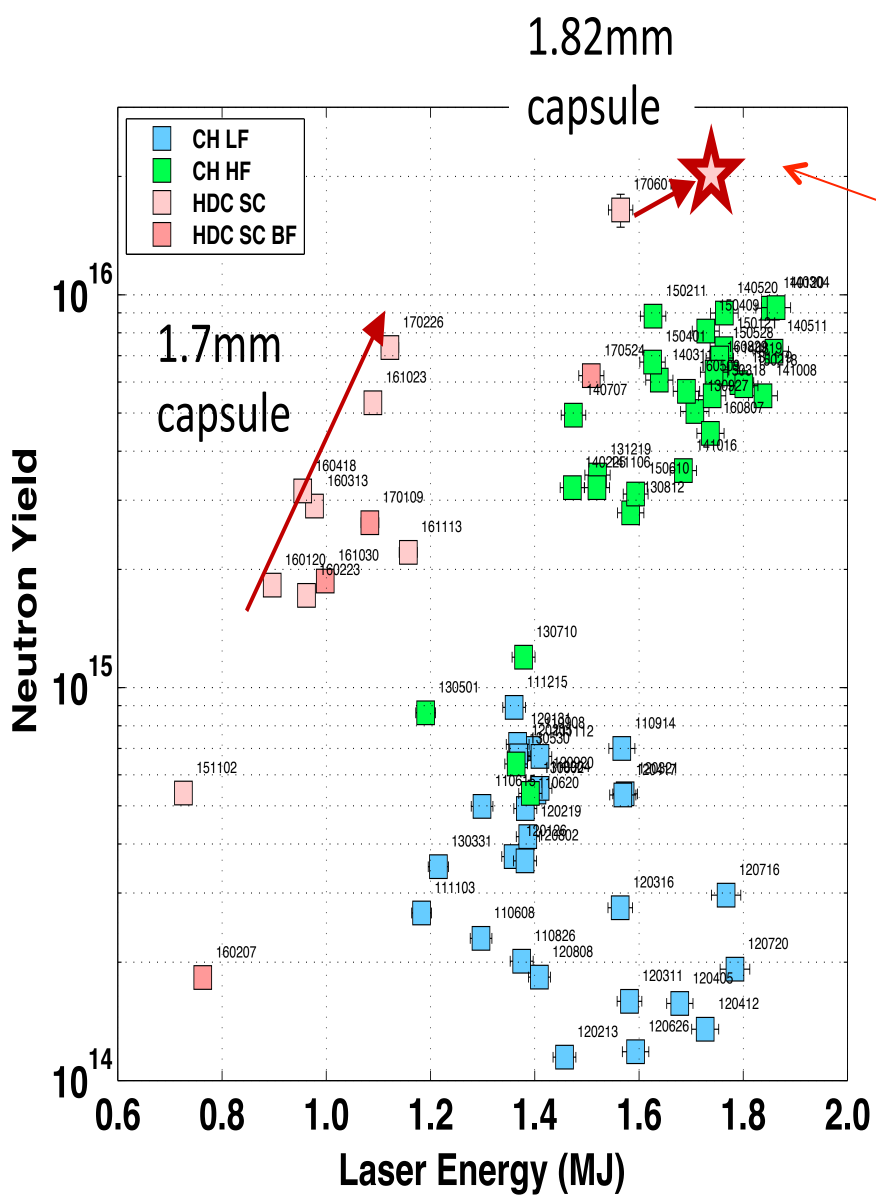


FIG. 2. Capsule-only multimode stability calculations of the low-foot (top row) implosion and two high-foot (second and third rows) implosions are shown. The left column shows the condition of the ablator (on a density color scale) at 200 μm radius which is near peak velocity and the right column shows the condition of the ablator and hot-spot at peak compression. The trade-off between densification and stability are clear. Reprinted with permission from Phys. Rev. Lett. **112**, 055002 (2014). Copyright 2014 AIP Publishing LLC.

Very recent results on NIF



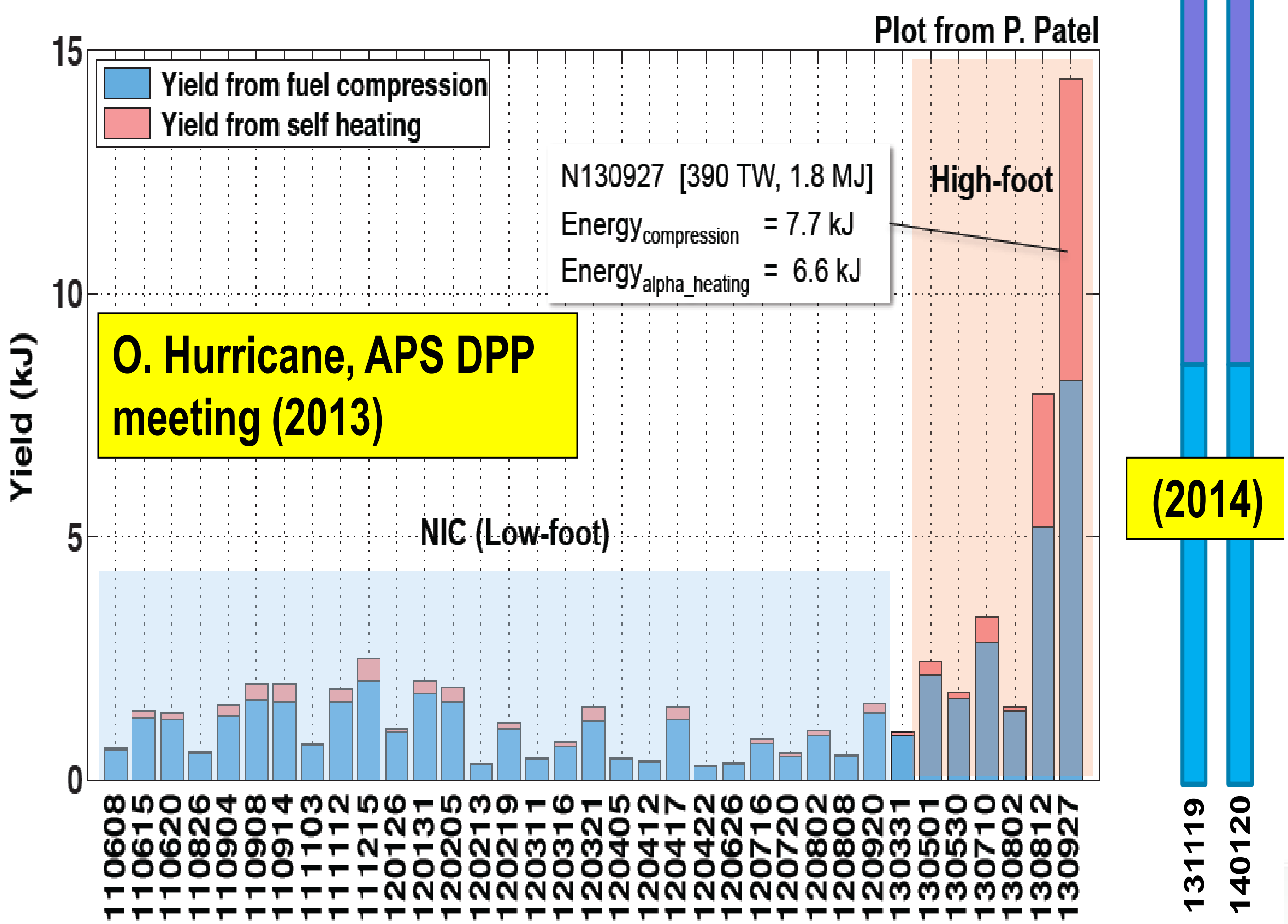
W-doped HDC capsule driven in a low-gas-fill hohlraum 390 km/s, 2e16, 57 kJ of fusion yield, more than 2x α-heating

Near Vacuum Hohlraums reduce Laser-Plasma-Instabilities

HDC (diamond) or Beryllium Ablator have greater hydrodynamic efficiency allowing a more massive (and more stable) shell to be imploded

Rugby holhraum

We finally have an implosion where a large fraction of the total fusion output is from α-particle self-heating



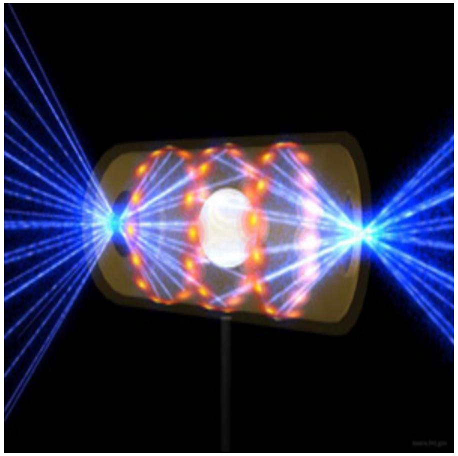
Joint WCI/NIF Team:

D. Callahan, E. Dewald, T. Dittrich,T. Doeppner, D. Hinkel, L. Berzak Hopkins, O. Hurricane, P. Kervin, J. Lee Kline (LANL), S. LePape, T. Ma, J. Milovich, J. Moody, A. Pak, H.-S. Park, B. Remington, H. Robey, J. Salmonson, NIF operations, NIF cryo, NIF targets, GA, LLE, & M.I.T.

NIF and indirect drive

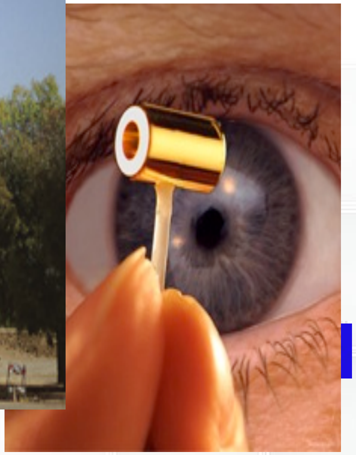
National Ignition Facility

to demonstrate the scientific feasibility of nuclear fusion. This will be an enormous scientific achievement !
However.... NIF is based on INDIRECT DRIVE which does not seem compatible with requirements for fusion reactors:



- Complicated targets
- Massive targets (lot of high-Z material in chamber)
- Above all: intrinsic low gain due to X-ray conversion.

In addition, indirect drive poses "political" problems...
Therefore we need DIRECT DRIVE



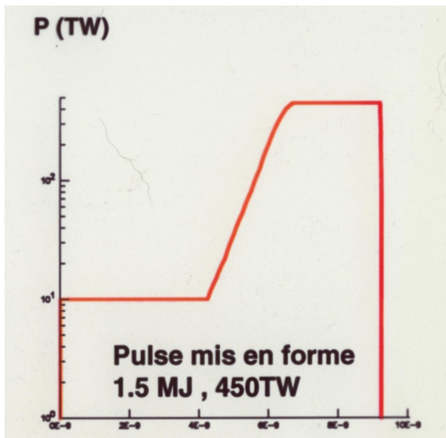
Grazie per la attenzione

Slides per discussione

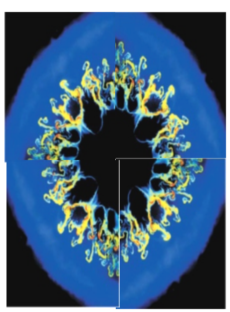
Shock ignition: a final laser spike launches a converging shock

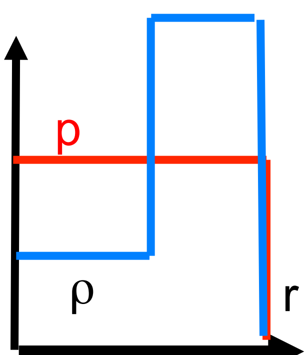
Conventionnal direct drive

450 TW, 1.5MJ pulse



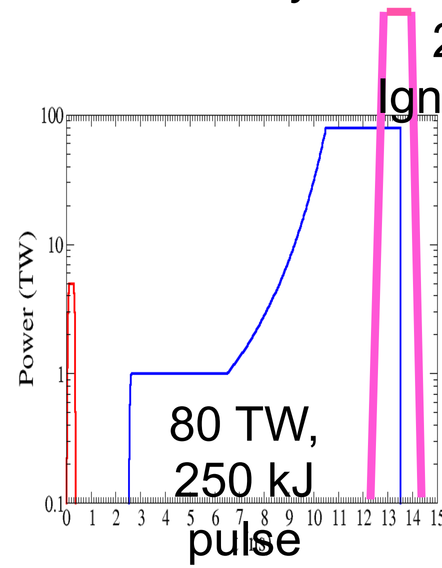
High Aspect ratio target
V ~400 km/s



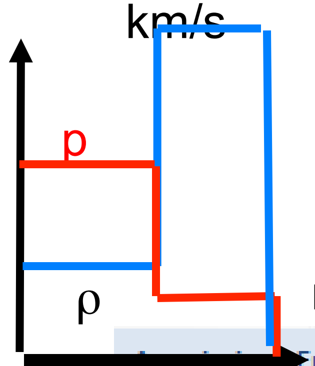


Produces an Isobaric fuel assembly

Low velocity drive



Low AR V~240 km/s

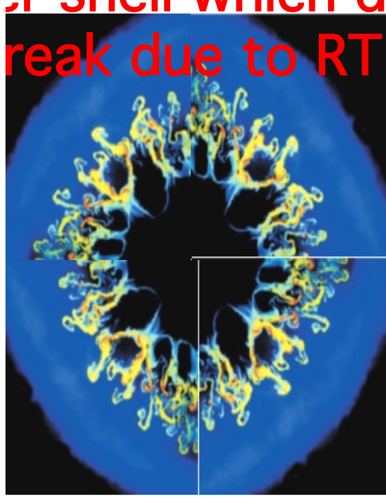


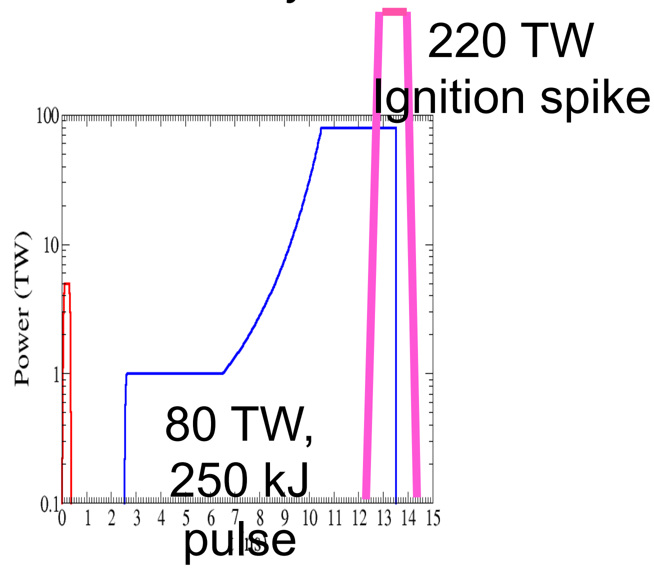
Fuel assembly is non isobaric

Shock ignition is less sensitive to hydro instabilities

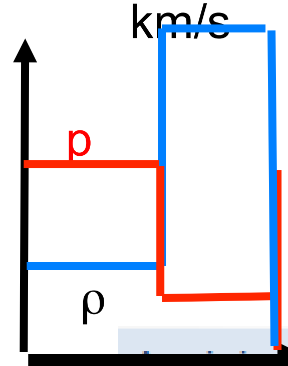
Low velocity drive

In SI, you do not create the hot spot with the “main” compression beam. Hence you do not need such high implosion velocity. Hence you can implode a more massive thicker shell which does not break due to RT





Low AR V~240 km/s



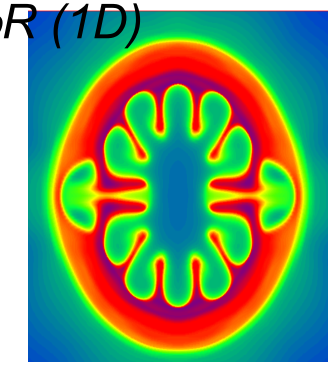
Fuel assembly is non isobaric

The Ignition shock mitigates RT growth at stagnation

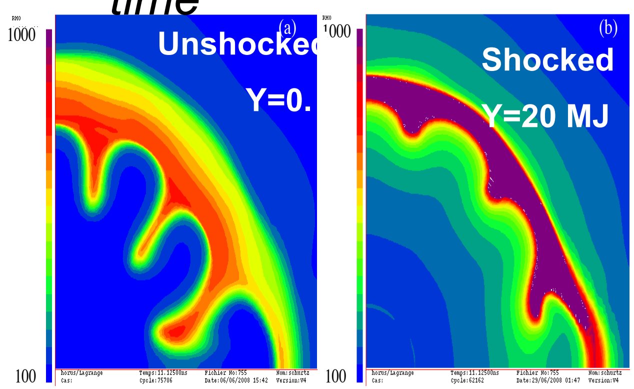
HiPER target at time of maximum ρR (1D)

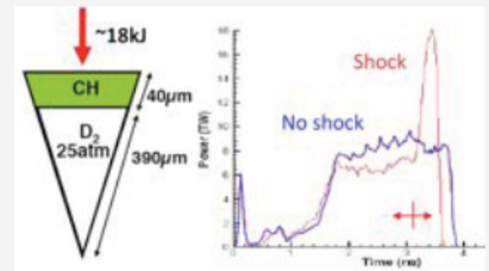
180 kJ

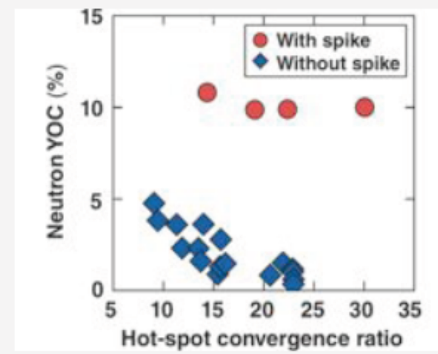
48 beams



HiPER target at ignition time







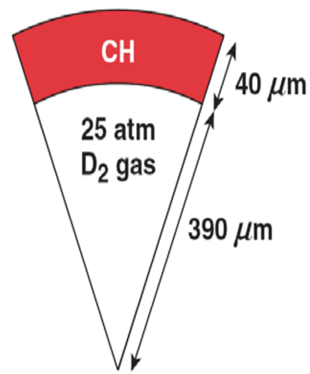
Is this experimental evidence of shock RT mitigation?

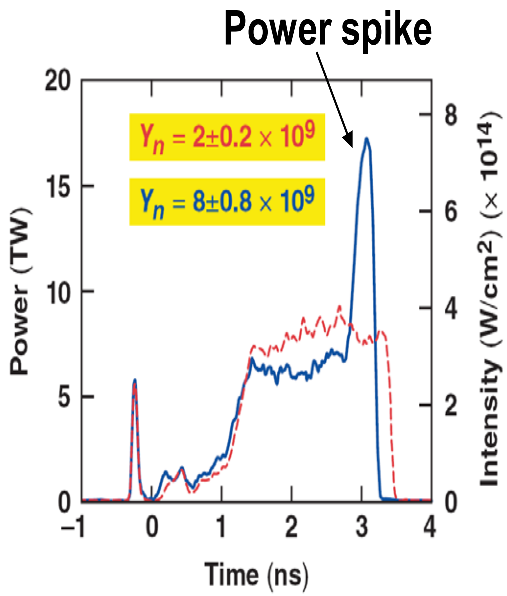
- **OMEGA shock driven target**
- **D2 gas with CH shell**
- **Same total energy: shock, no shock**

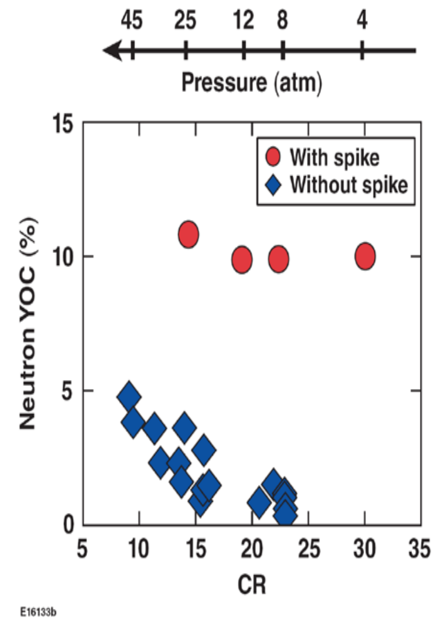
W. Theobald et al.

Shock-ignition experiments on OMEGA have shown improved performance when a shock launching power spike is added at the end of the laser pulse

E_L = 19 kJ, α = 1.3,
V_i = 1.7 × 10^7 cm/s, SSD off







The neutron yield increases considerably when a shock is launched at the end of the pulse.

The measured-to-calculated neutron-yield ratios are close to 10% for a hot-spot convergence ratio of 30.

W. Theobald,et al Phys. Plasmas (2008)

Unknowns of Shock Ignition

- Effect of laser-plasma instabilities at intensities up to ≈ 10^{16} W/cm^2. SRS, SBS and TPD. Do they develop? How much light do they reflect?

- Are there many hot electrons and at what energy? What is their effect? *(usually in ICF hot electrons are dangerous since they preheat the target... Here they came at late times, large fuel ρr, so they could indeed be not harmful or even beneficial, increasing laser-target coupling in presence of a very extended plasma corona...)*

- Are we really able to couple the high-intensity laser beam to the payload through an extended plasma corona? Are we really able to create a strong shock?

- What is the effect of magnetic fields, delocalised transport, delocalised absorption, thermal smoothing in the overdense region on shock generation at high laser intensity?

Experiments done on European Laser Facilities in planar geometry, to study the physics of shock ignition.

HEATING SYSTEMS for Tokamaks

Associazione Euratom-ENEA sulla Fusione

Heating systems

Three heating systems are used in tokamaks

1.ohmic heating

2.neutral beam injection using injection of neutral atoms like hydrogen or deuterium with energy of 80-160keV or higher (0.5-1MeV)

3.radiofrequency heating using fundamental

Resonances of plasmas like :

electron cyclotron , ion cyclotron and hybrid frequencies

Associazione Euratom-ENEA sulla Fusione

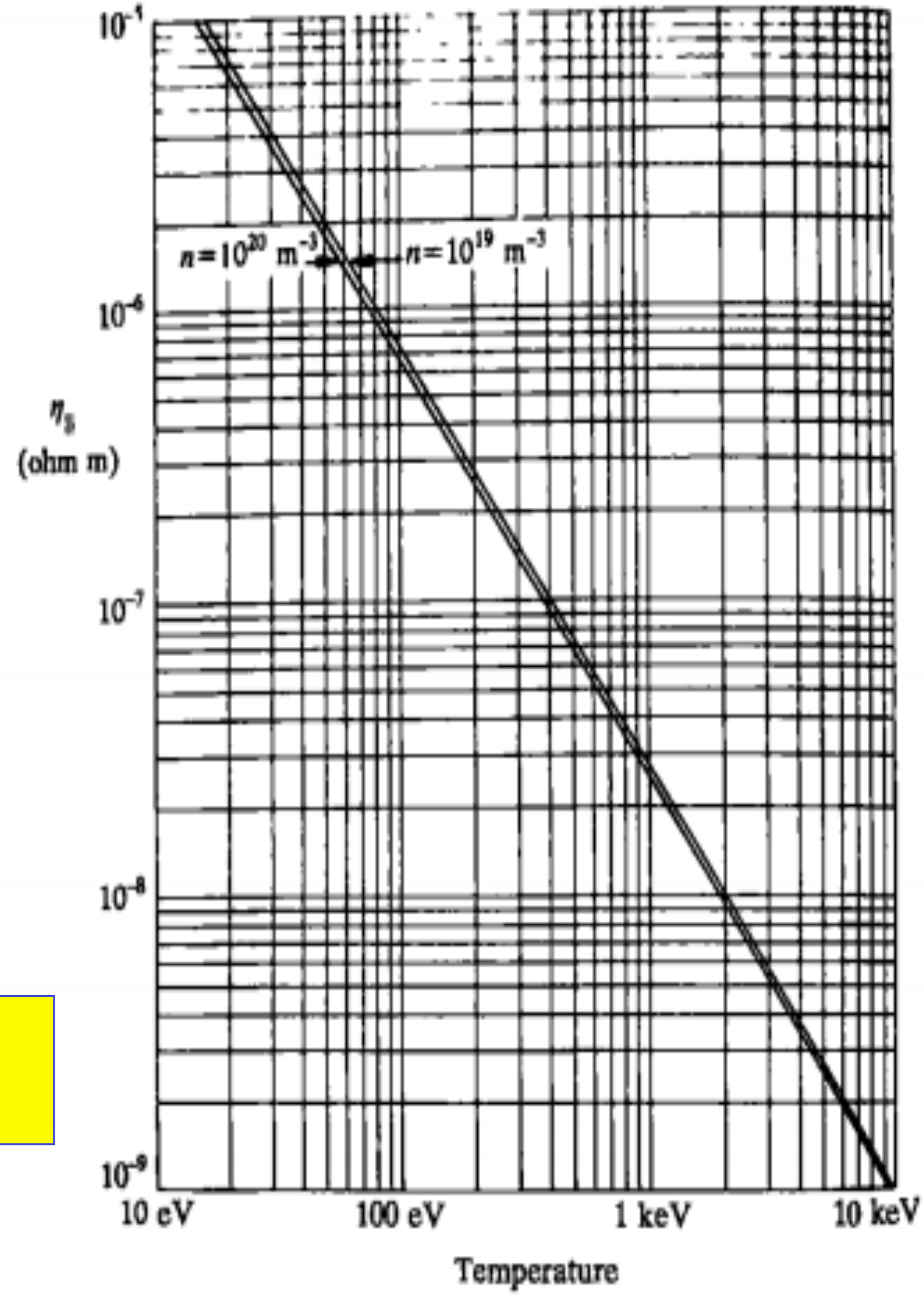
Ohmic heating

$$ohmic\ heating\ power:\ P_\Omega = \eta \left\langle j^2 \right\rangle\ (W/m^3)$$

$$\eta =\ plasma\ resistivity \approx 8\,10^{-8}\ Z_{eff}\ T_{ekeV}^{-3/2}\ ohm*m$$

$$\eta s = spitzer\ resistivity = 1.65\ 10^{-9}\ \ln\Lambda\ T_{ekeV}^{-3/2}\ ohm*m$$

$$j =\ plasma\ current\ density = A/m^2.$$



Spitzer resistivity ohm*m

Ohmic heating

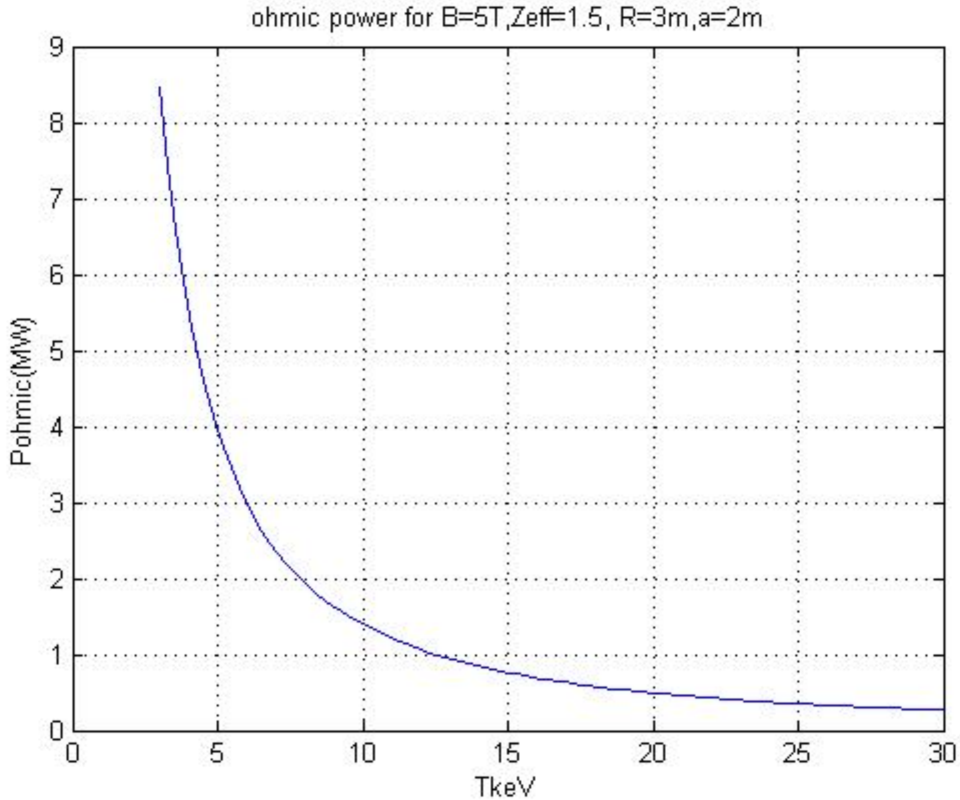
$$ohmic\ heating\ power:\ P_\Omega = \eta \left\langle j^2 \right\rangle\ (W/m^3)$$

$$\eta =\ plasma\ resistivity \approx 8\,10^{-8}\ Z_{eff}\ T_{ekeV}^{\ \ -3/2}\ \ ohm*m$$

$$\eta s = spitzer\ resistivity = 1.65\ 10^{-9}\ \ln\Lambda\ T_{ekeV}^{\ \ -3/2}\ \ ohm*m$$

$$j =\ plasma\ current\ density = A/m^2.$$

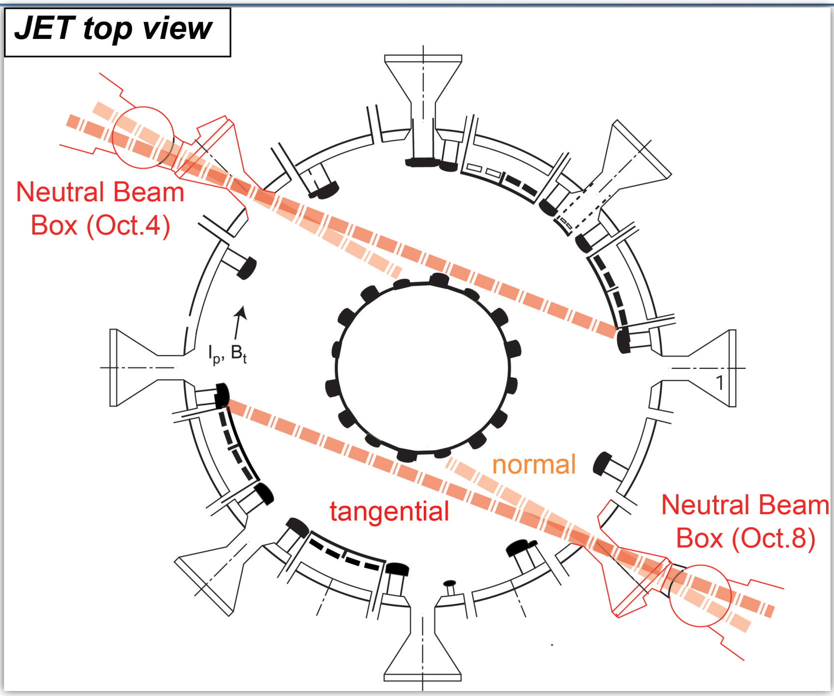
$$\left\langle j^2 \right\rangle = 2*\left(\frac{B_\phi}{\mu_0 * R}\right)^2 * \frac{1}{q_0*(q_a - q_0/2)}$$



Associazione Euratom-ENEA sulla Fusione

JET NBI system
➔ 2 neutral injectors boxes

JET top view



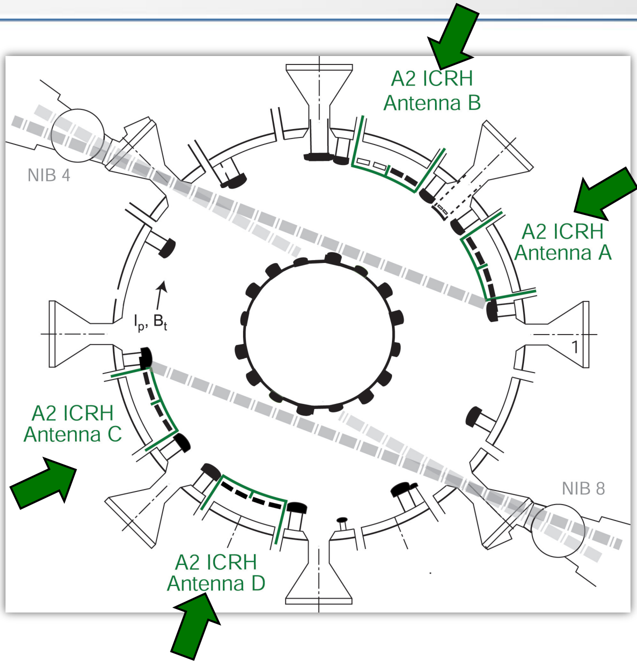
[CIRIC et al., FED, 86, 2011]

- Two neutral beam injector boxes
- Each equipped with 8 Positive Ion Neutral Injectors: **PINIs** ➔ grouped into tangential and normal banks

View of the JET vessel from inside the new

JET ICRF system

➔ 4 antennas with new Be private limiters



4 antennas (A, B, C & D) called the A2s antennas

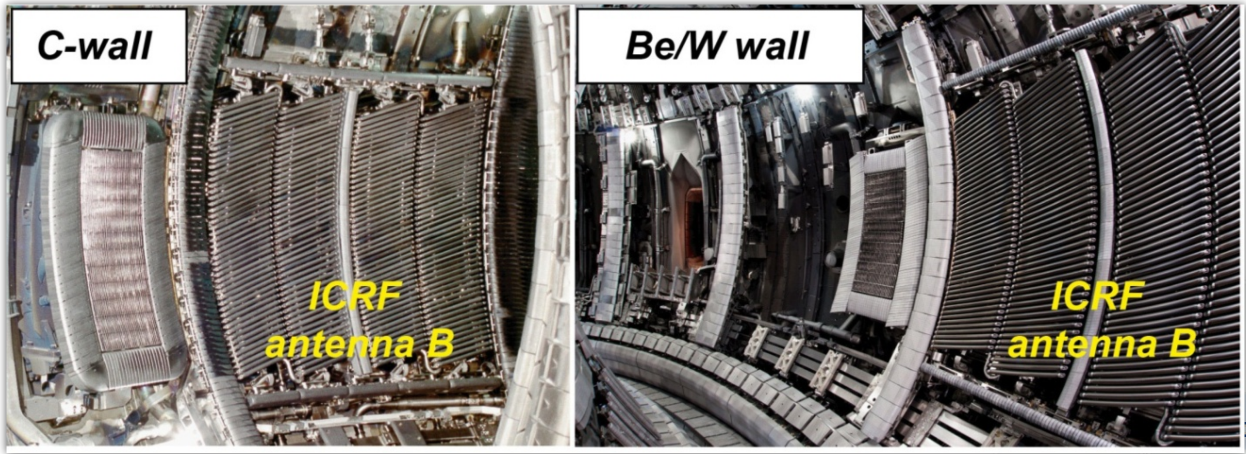
- Frequency range is 23 to 55 MHz
- Wave launched with symmetric spectra ("dipole" phasing) or asymmetric spectra ("± 90°" phasing)
- ICRF system is ELM tolerant (able to couple steady power on ELMs)

ITER-like ICRF antenna not used during last campaign

ITER-like wall related change :

- All private limiters changed from CFC to Be tiles

[GRAHAM M., et al., PPCF (2012)]



ICRF operation in the ILW

➔ No arcing issues, 4MW on type-I ELMs, heat-load within limits

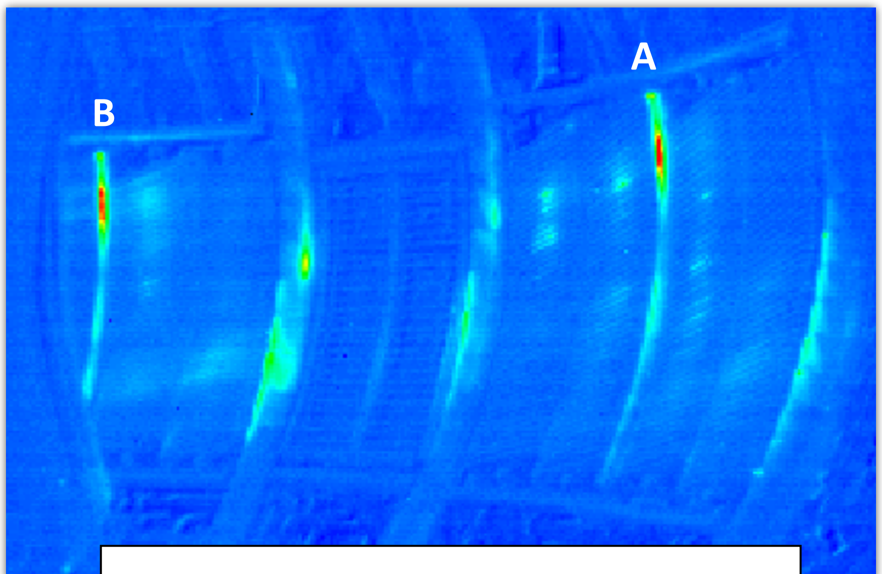
1st possible issue ➔ signs of arcing on new antenna Be private limiters or change in antenna behaviour

No problem

Up to **usual** maximum voltage **(~30kV) reached**

4MW on type I ELMy H-mode (with ¾ of the system)

2nd possible issue ➔ heat loads due to ions accelerated in RF sheath rectified voltages



– Maximum power load (estimated from IR thermography and a thermal model for the ILW Be tiles) was ~ 4.5 MW/m^2

– As design limit for the Be tiles is 6MW/m^2 for 10s ➔ we are safe but monitoring by viewing protection system is still needed.

[JACQUET et al., PSI2012]

Electron cyclotron heating system on ITER

Nucl. Fusion **48** (2008) 054013

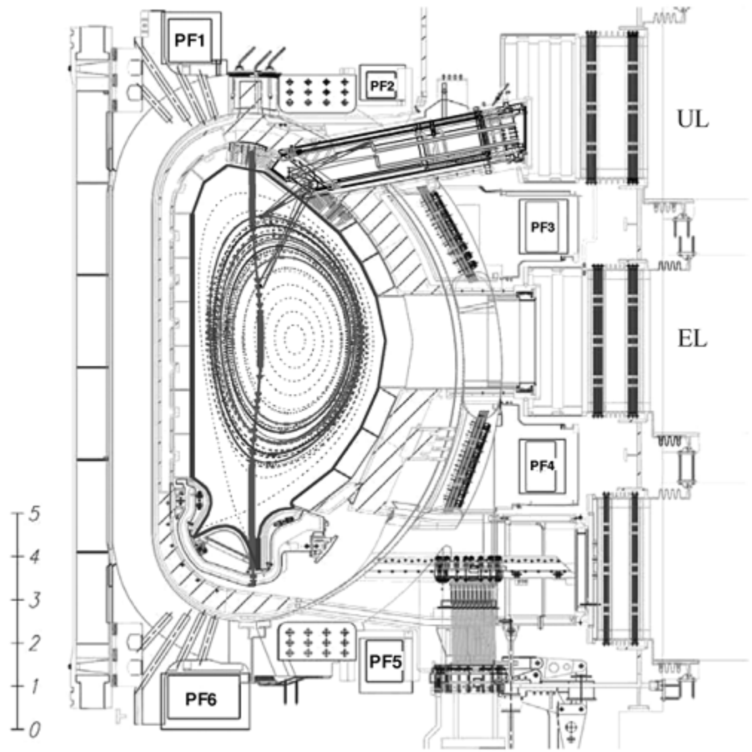


Figure 1. The cross section of ITER with the location of the EL and UL ports identified. A simplified UL is placed in the upper port with the beams aimed at the innermost and outermost expected locations of the NTMs.

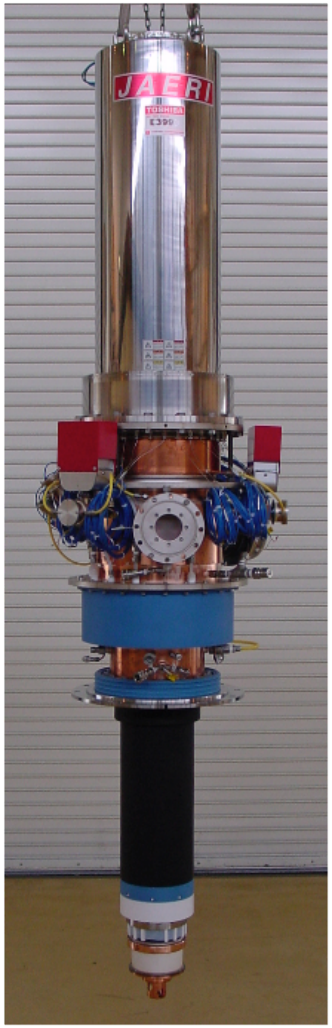
Achievement of ITER relevant parameters with RF gyrotron

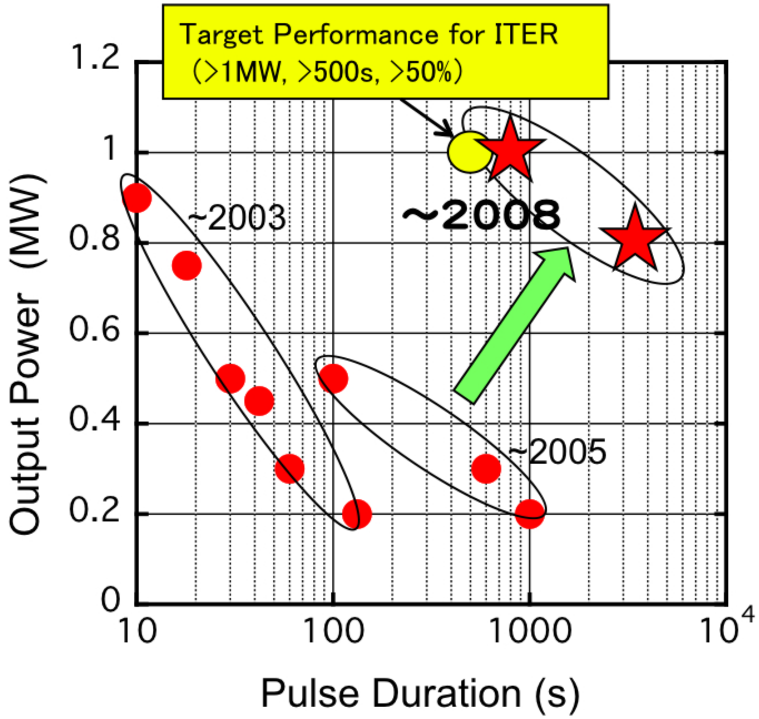
Gyrotron V-11 at the test

Gyrotrons V-10, V-11 were tested in 2010 and 2011 respectively with CRYOMAGNETICS LHe –free magnet.

It is important to note that two last gyrotrons (V-10 and V-11) demonstrate very similar output

170GHz Gyrotron Development in JAEA





Multi-frequency operation, >1.3MW operation

plasma diagnostics : systems and measurements

Classification of diagnostics

Table 7. Assessed measurement capability relative to requirements.

GROUP 1a Measurements for machine protection and basic control	GROUP 1b Additional measurements for control in specific scenarios	GROUP 1c Additional measurements for performance evaluating and physics
Plasma shape and position, separatrix–wall gaps, gap between separatrixes	Neutron and α-source profile	Confined α-particles
Plasma current, $q(a)$, $q(15\%)$	Helium density profile (core)	TAE modes, fishbones
Loop voltage	Plasma rotation (toridal and poloidal)	T_e profile (edge)
Fusion power	Current density profile (q-profile)	n_e, T_e profiles (X-point)
$\beta_N = \beta_{tor}(aB/I)$	Electron temperature profile (core)	T_i in divertor
Line-average electron density	Electron density profile (core and edge)	Plasma flow (divertor)
Impurity and D,T influx (divertor and main plasma)	Ion temperature profile (core)	$n_T/n_D/n_H$ (edge)
Surface temp. (divertor and upper plates)	Radiation power profile (core, X-point and divertor)	$n_T/n_D/n_H$ (divertor)
Surface temperature (first wall)	Z_{eff} profile	T_e fluctuations
Runaway electrons	Helium density (divertor)	n_e fluctuations
'Halo' currents	Heat deposition profile (divertor)	Radial electric field and field fluctuations
Radiated power (main plasma, X-point and divertor)	Ionization front position in divertor	Edge turbulence
Divertor detachment indicator (J_{sat}, n_e, T_e at divertor plate)	Impurity density profiles	MHD activity in plasma core
Disruption precursors (locked modes, $m = 2$)	Neutral density between plasma and first wall	
H/L mode indicator	n_e of divertor plasma	
Z_{eff} (line-averaged)	T_e of divertor plasma	
n_T/n_D in plasma core	Alpha-particle loss	
ELMs	Low m/n MHD activity	
Gas pressure (devertor and duct)	Sawteeth	
Gas composition (divertor and duct)	Net erosion (divertor plate)	
Dust	Neutron fluence	

Note: Expect to meet measurement requirements; performance not yet known; expect not to meet measurement requirements.

Technical specifications examples

Table 2. Continued.

Measurement	Parameter	Condition	Range or Coverage	Resolution: Time or Freq.	Resolution: Spatial or Wave No.	Accuracy
23. Electron temperature profile	Core T_e	$r/a < 0.9$	0.5–40 keV	10 ms	$a/30$	10%
	Edge T_e	$r/a > 0.9$	0.05–10 keV	10 ms	5 mm	10%
24. Electron density profile	Core n_e	$r/a < 0.9$	3×10^{19}–3×10^{20} m^{-3}	10 ms	$a/30$	5%
	Edge n_e	$r/a > 0.9$	5×10^{18}–3×10^{20} m^{-3}	10 ms	5 mm	5%
25. Current profile	$q^{(r)}$	Physics study	0.5–5	10 ms	$a/20$	10%
28. Ion temperature profile	Core T_i	$r/a < 0.9$	0.5–40 keV	100 ms	$a/10$	10%
	Edge T_i	$r/a > 0.9$	0.05–10 keV	100 ms	TBD	10%
30. Confined alphas	Energy spectrum	Energy resolution TBD	(0.1–3.5) MeV	100 ms	$a/10$	20%

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Measurement by Thomson Scattering

A laser beam is injected into the plasma

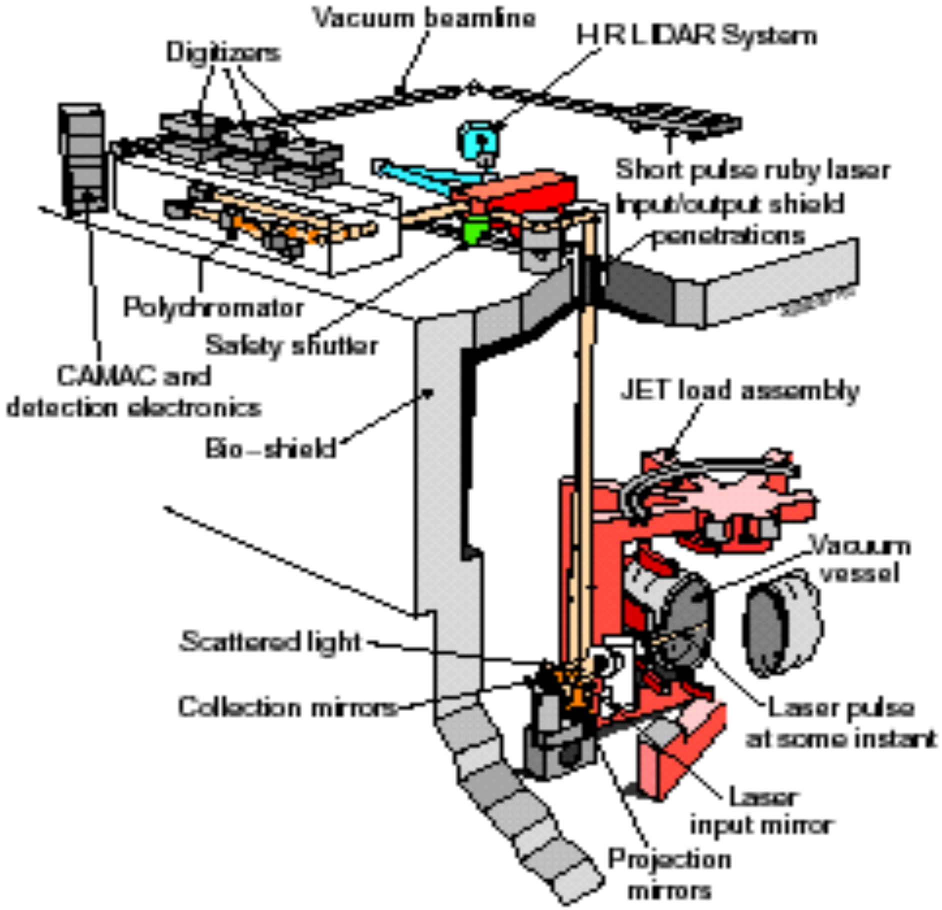
The diffused light is collected at a fixed angle

The spectral width of the scattered radiation is measured

This width measures the electron temperature

The intensity (i.e. the number of photons) is proportional to the eletron density

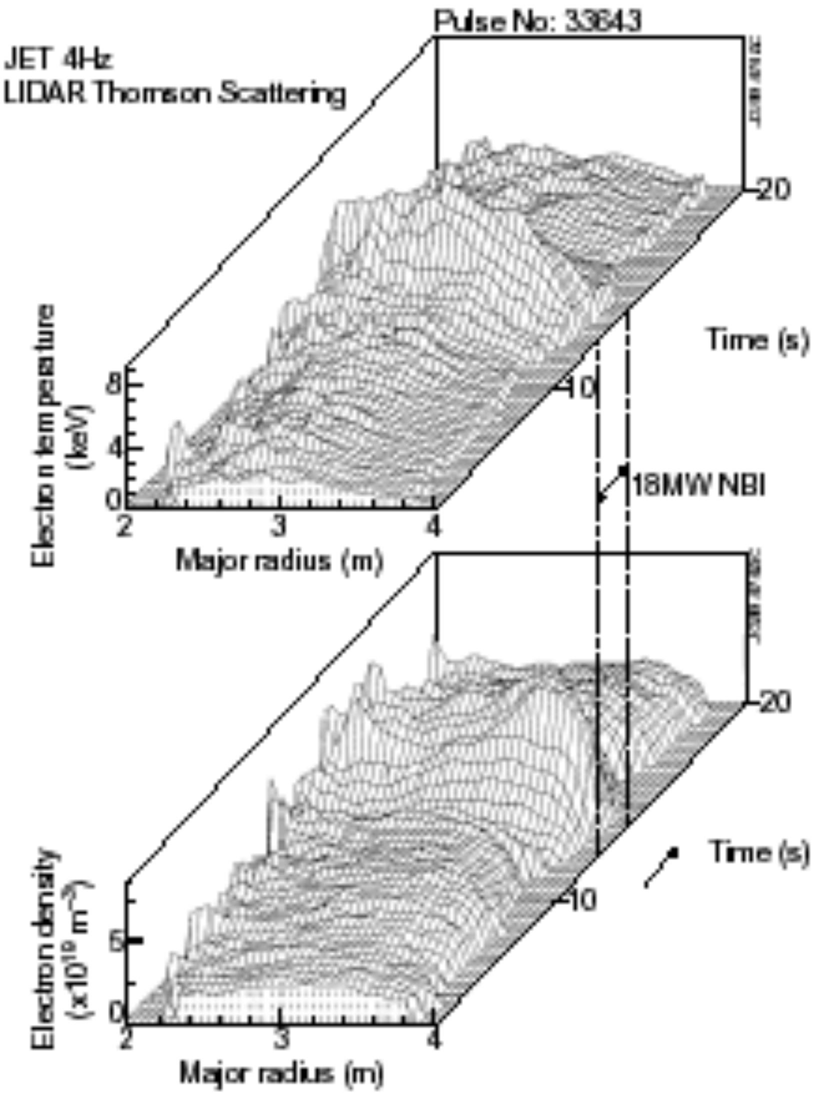
LIght Detection And Ranging
(Time of flight)

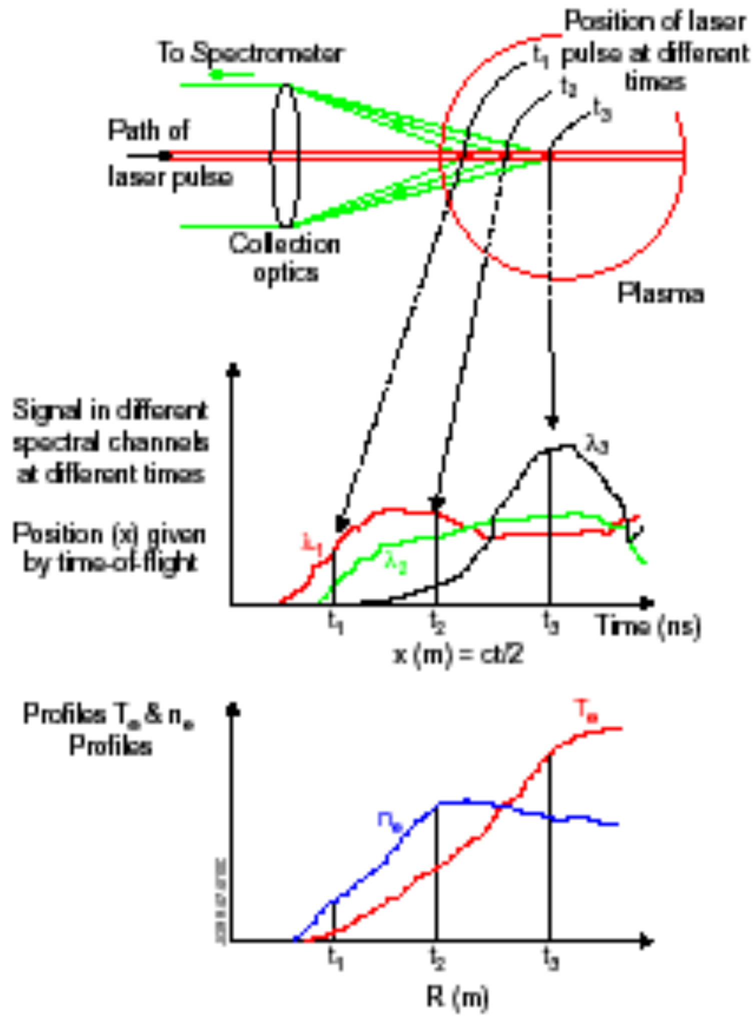


Measures $T_e(R)$, $n_e(R)$

LIDAR Thomson Scattering

Plasma with Neutral Beam Heating



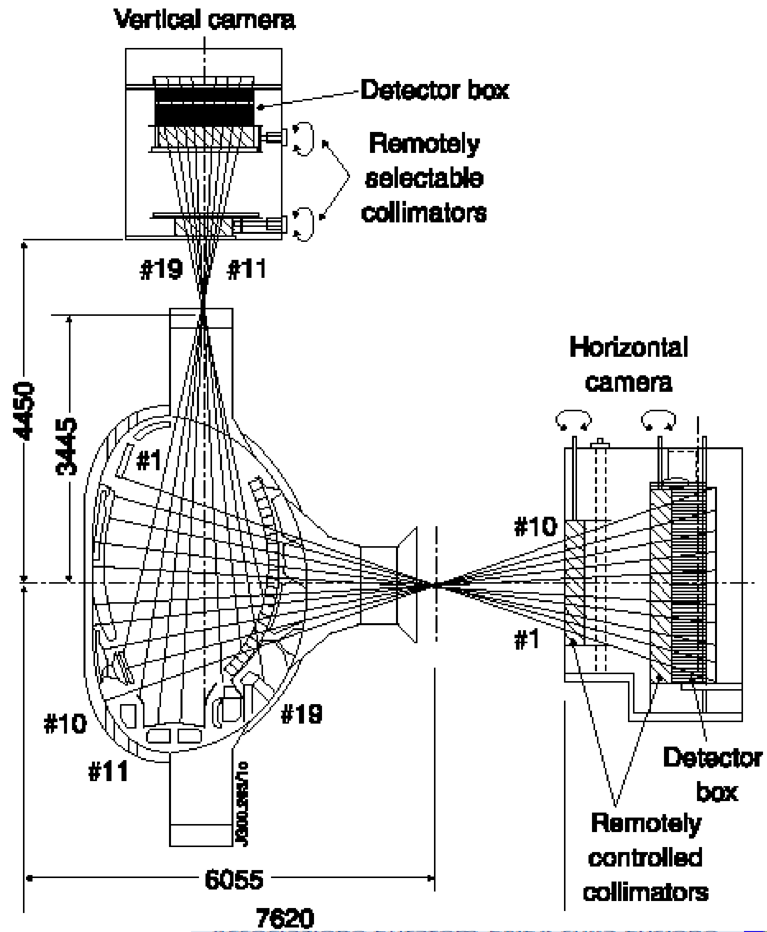


γ - RAY PROFILE MEASUREMENTS:

(Neutron / GAMMA Profile Monitor)

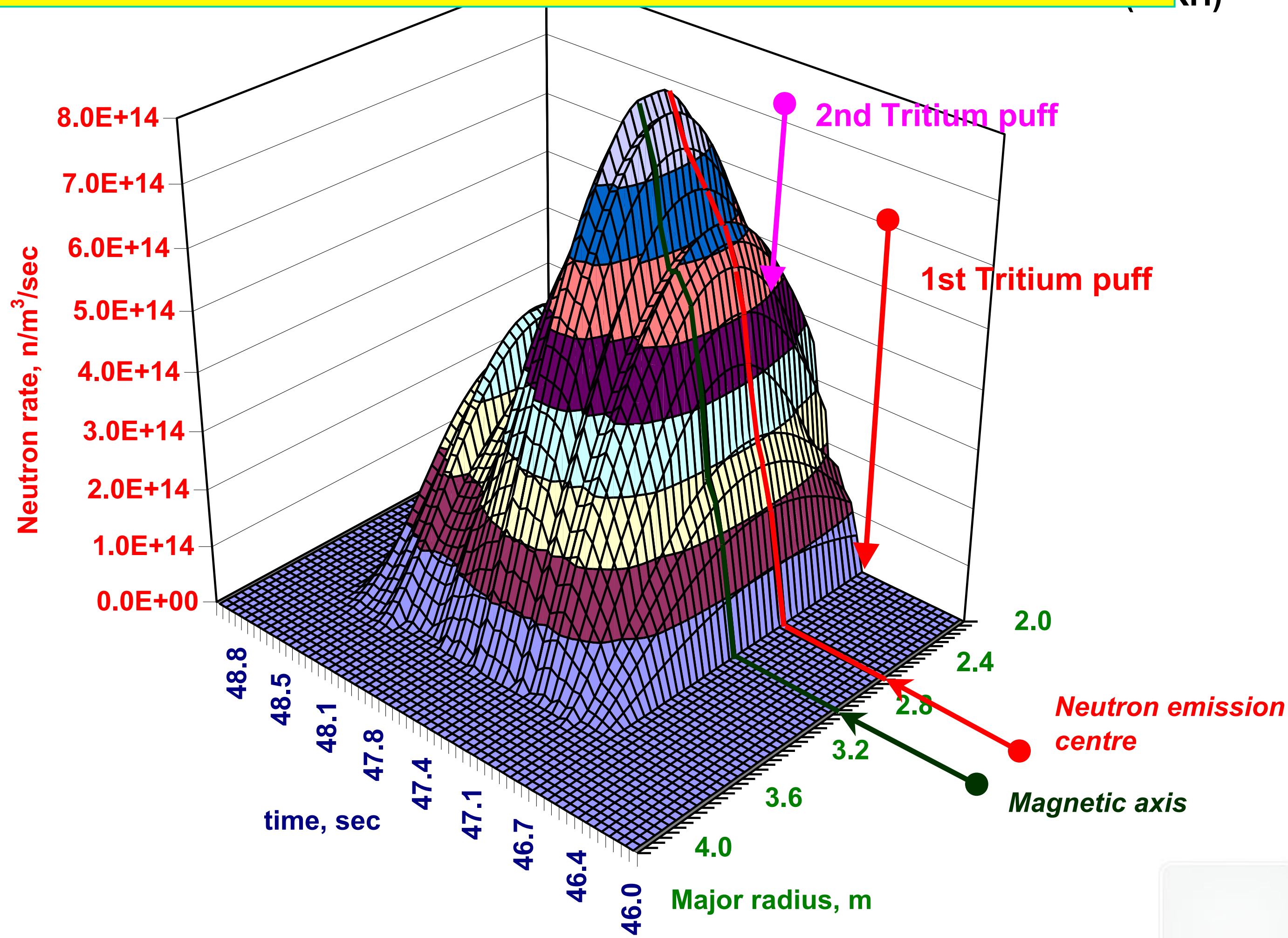
- vertical camera - 9 lines-of-sight
- horizontal camera - 10 lines-of-sight
- Collimators: Ø10 and 21 mm
- Space resolution: 10 cm in centre

✓ CsI-detector array with 4 energy windows is used for the gamma-ray emission profile measurements



Misura di profilo di emissione neutronica in una scarica in cui e' stato iniettato un puff di Trizio (nella campagna TTE ottobre 2003 JET)

RH)



γ-ray profile measurements in ^{3}He-minority RF discharges

1.8MA/3.4T, 37 MHz, 5.6 MW RF:

#57303 - Counter-current wave (-90^{0}),

outward pinch

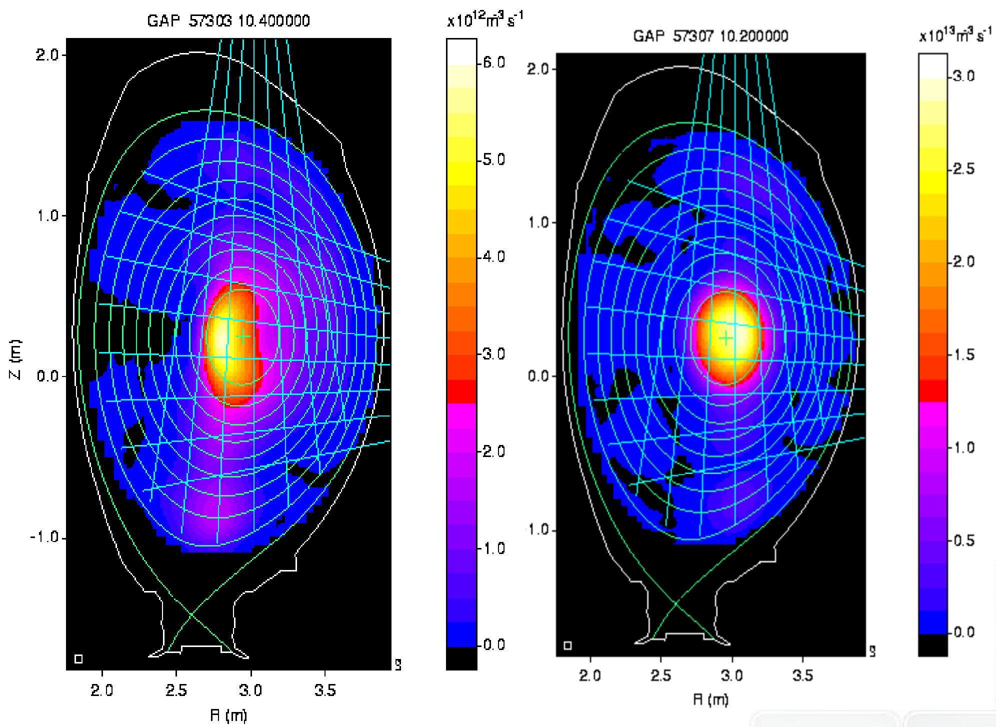
#57307 - Co-current wave (+90^{0}), inward pinch

Results are consistent with our first ^{3}He-pinch observation in 2001

Mantsinen et al PRL 89(2002)115004

- 90^{0} phasing

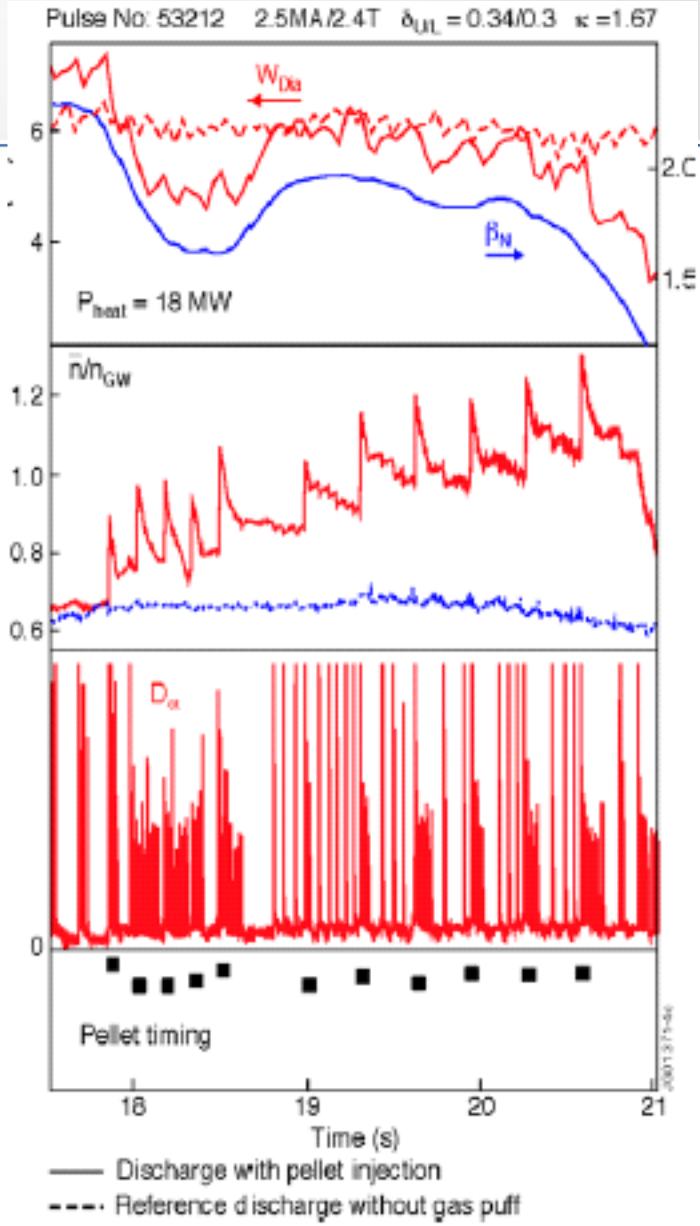
+90^{0} phasing

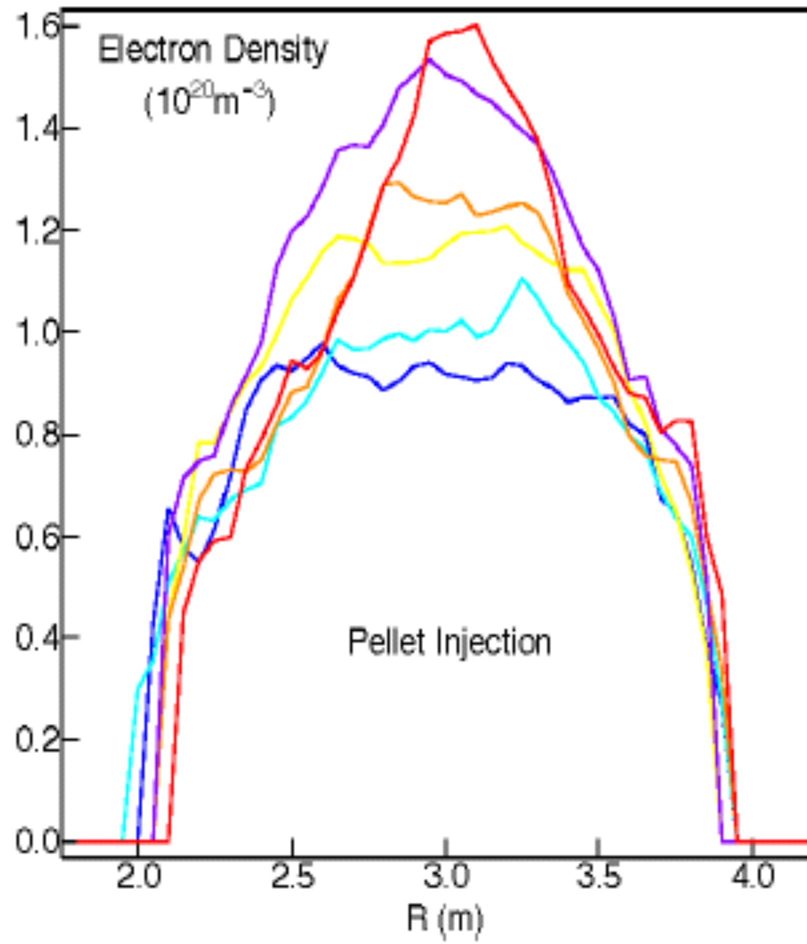


γ-ray emission from the nuclear reactions:
^{9}Be + ^{3}He and ^{12}C + ^{3}He

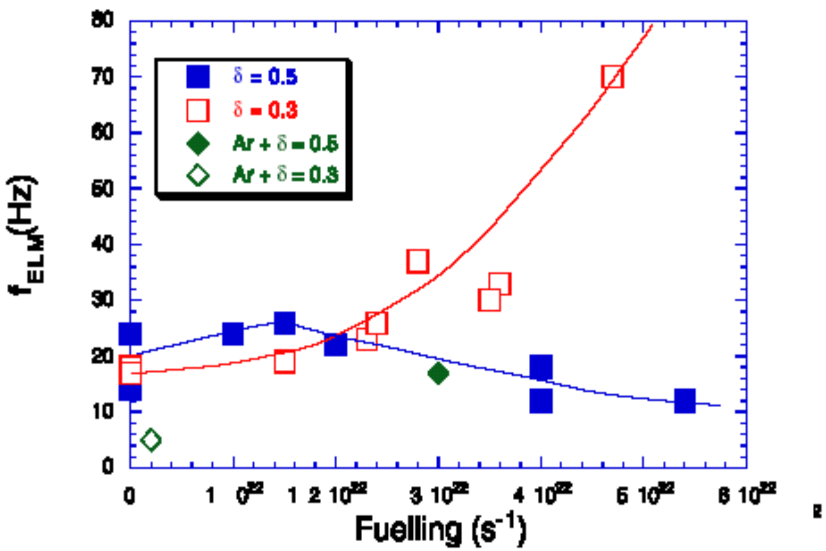
SCENARIO DEVELOPMENT

Fuelling con pellets n/ nGR>1

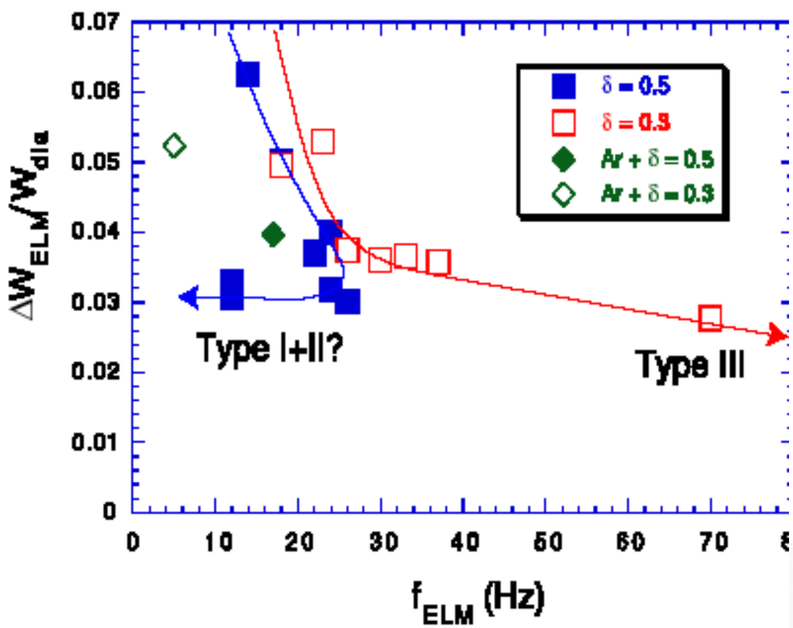




ELMs(Edge Localized Modes)

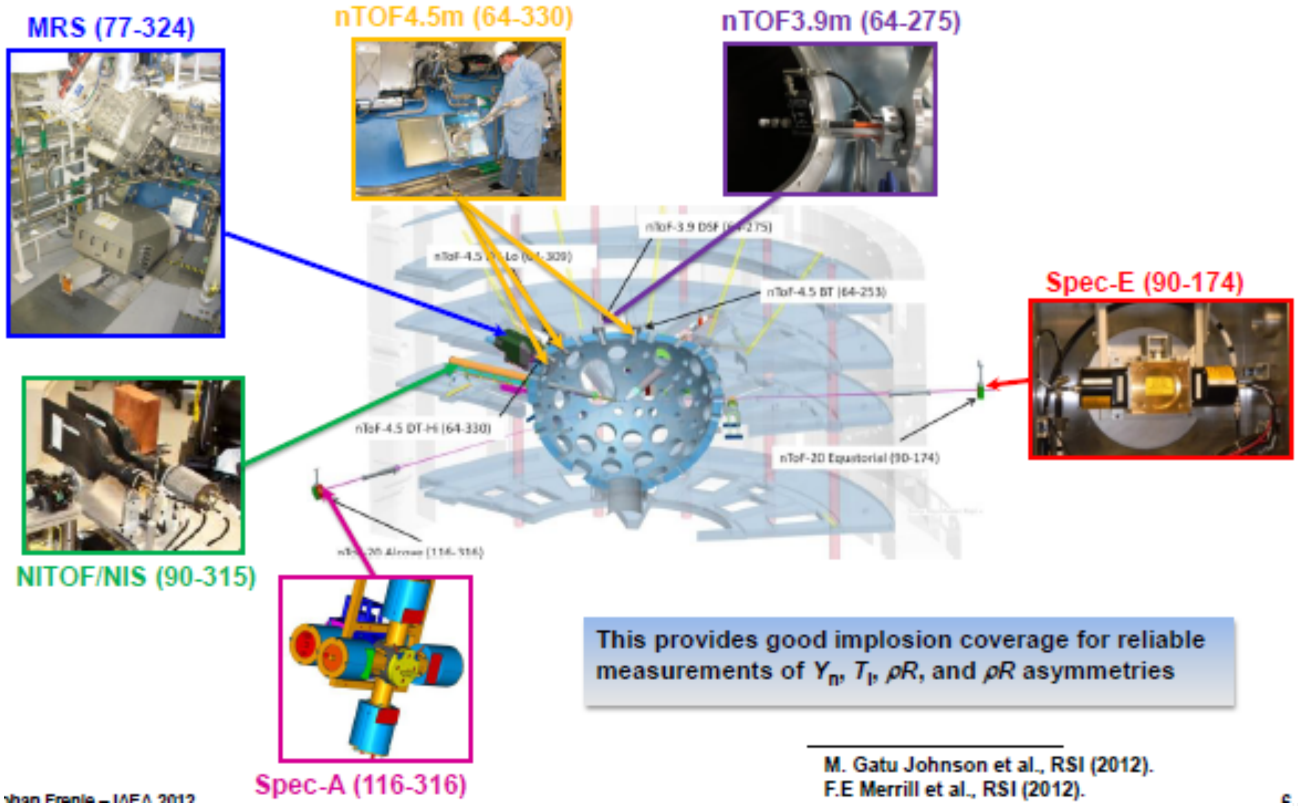


Deviations from f_{ELM} with Gas Fuelling for δ ~ 0.5



New ELM Behaviour breaks $f_{ELM} \leftrightarrow \Delta W_{ELM}$ (for high δ)

Several neutron spectrometers and an imaging system have been fielded at various locations on the NIF



This provides good implosion coverage for reliable measurements of Y_n, T_i, ρR, and ρR asymmetries

M. Gatu Johnson et al., RSI (2012).
F.E Merrill et al., RSI (2012).

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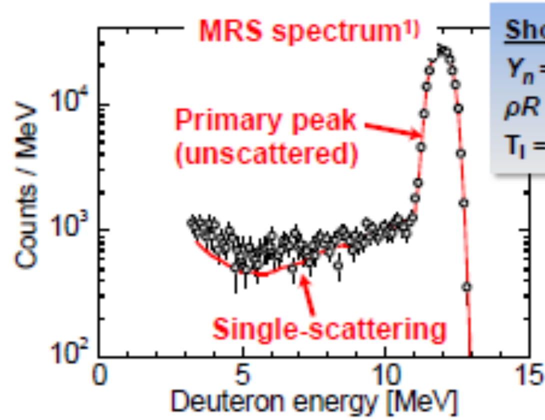
Spectra and images are now measured routinely on the NIF (Example: DT shot N120205)

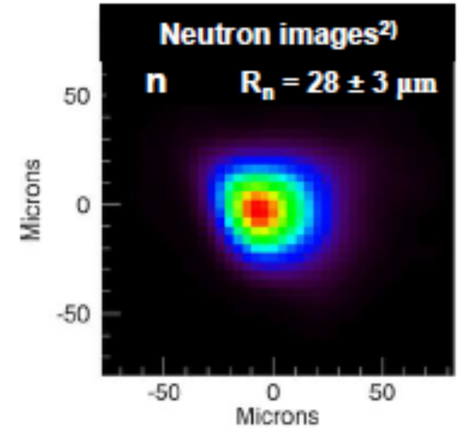


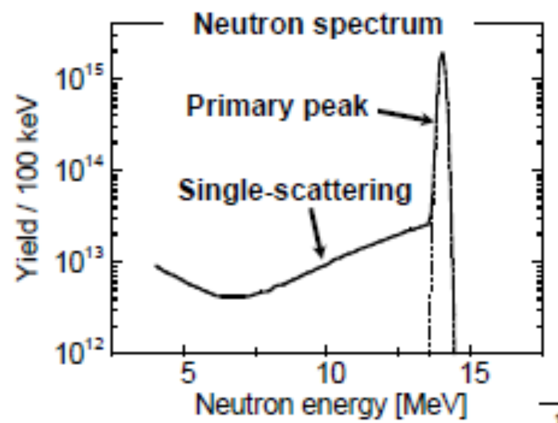
Shot N120205

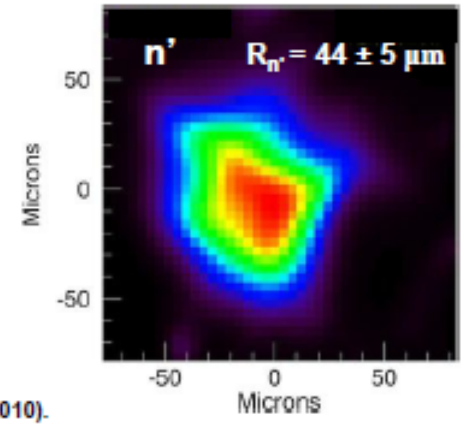
$Y_n = (5.6 \pm 0.2)\times10^{14}$

$\rho R = 900 \pm 40$ mg/cm^2

$T_i = (3.4 \pm 0.1)$ keV







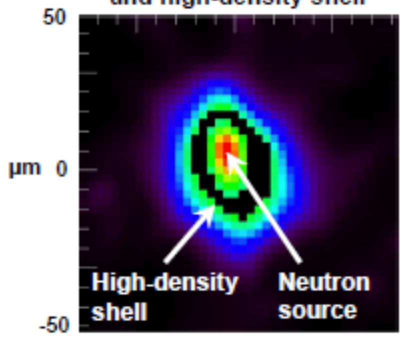
Johan Frenje – IAEA 2012

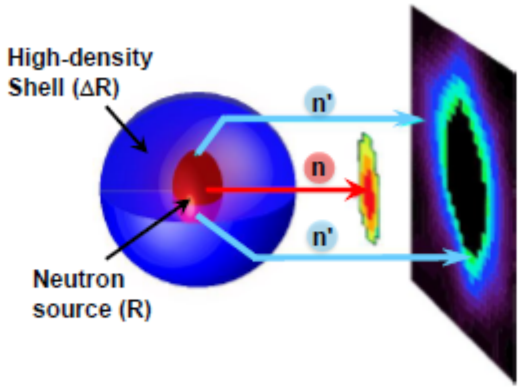
1) J.A. Frenje et al., PoP (2010).
2) G. Grim APS invited, PoP (2012)

Neutron images

Primary and scattered neutrons are imaged to diagnose neutron-source size (R) and thickness of high-density shell (ΔR), resp.

Images of neutron source and high-density shell





Primary neutrons (n):

- *R* of neutron source

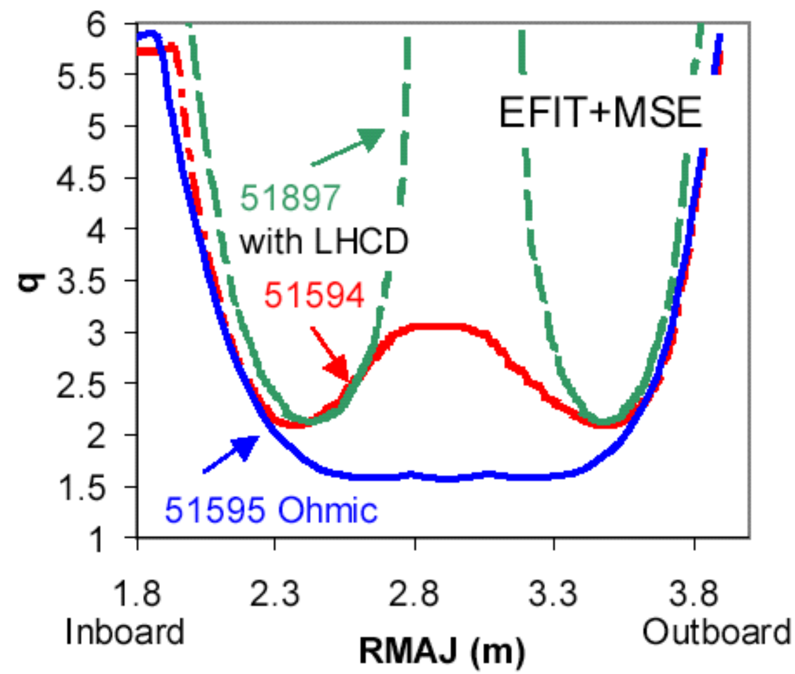
Scattered neutrons (n'):

- *ΔR* of high-density shell

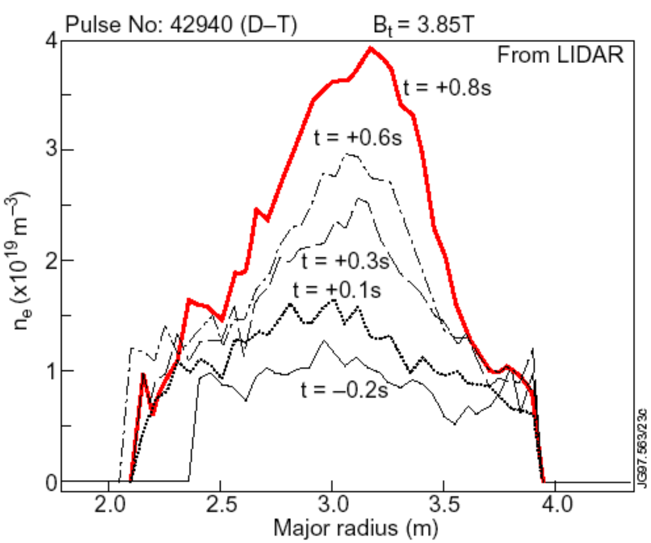
G. Grim et al., APS invited (2012).

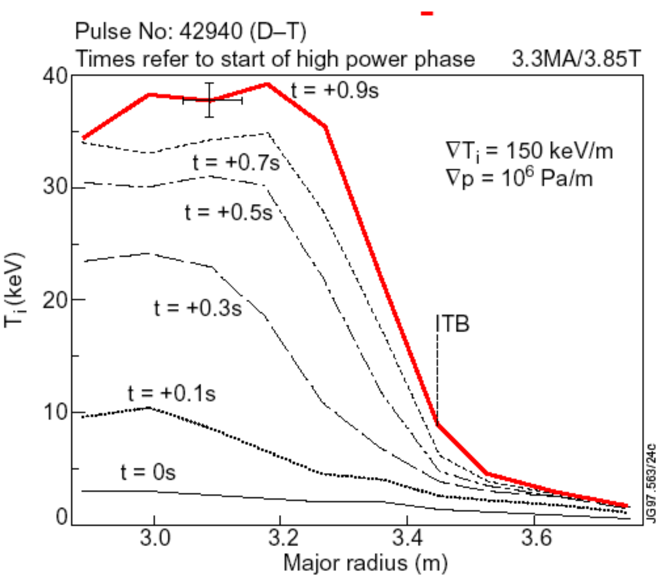
Scenari avanzati: modifica dei profili di corrente e creazione delle barriere di trasporto interne

- **Improved LHCD coupling leads to strong magnetic shear reversal during preheat**
- **strong internal transport barriers**
- **virtually no power threshold when compared to Optimised Shear**



Barriere interne in scariche AT





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MEASUREMENTS OF TURBULENCE IN PLASMAS WITH INTERNAL TRANSPORT BARRIERS

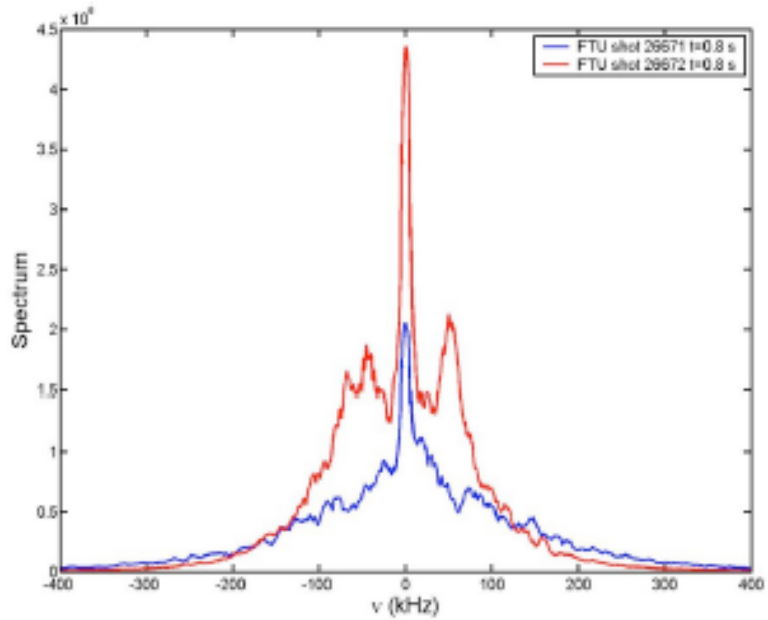


Fig. 4 – Fluctuation power spectra for two discharges with ITB (#26671, blue) and without (#26672, red); $n_e \approx 0.9 \cdot 10^{20}$ m^{-3} for both

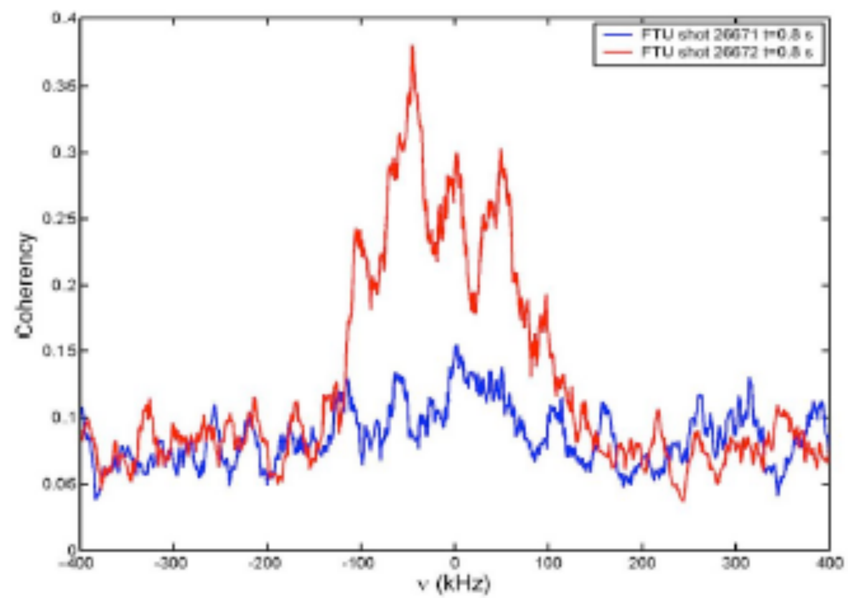
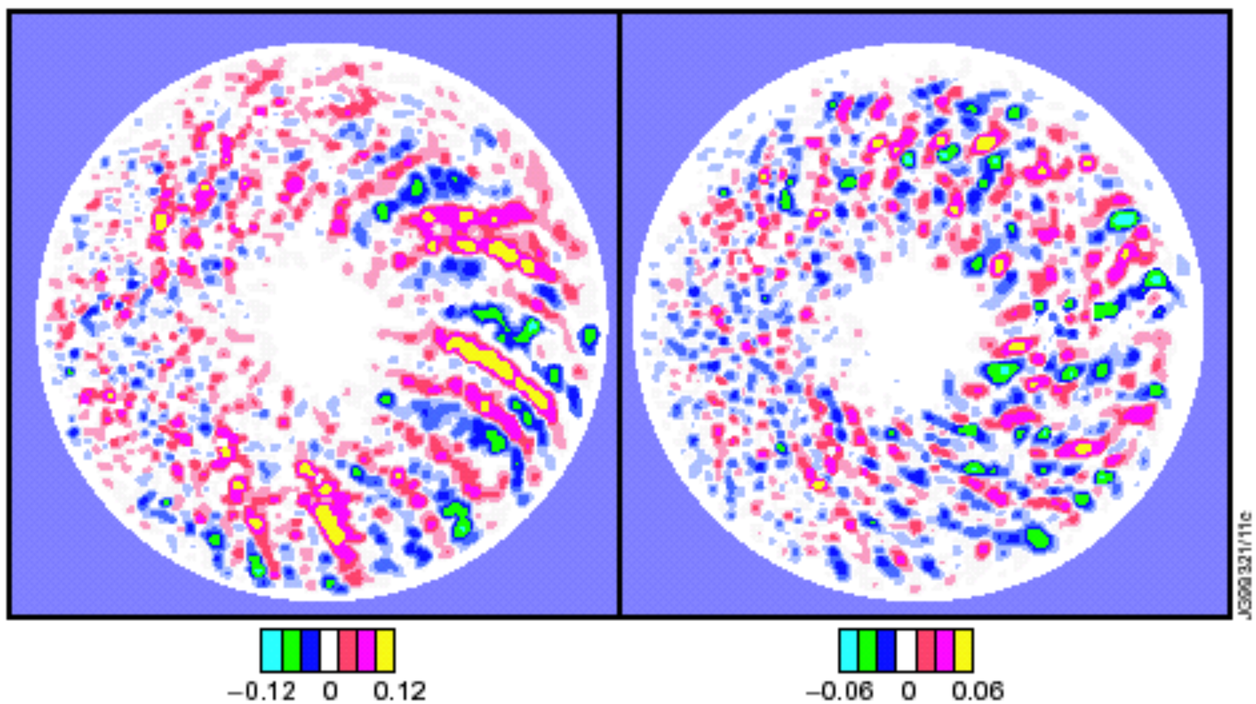


Fig. 5 – Coherence spectra for the same two discharges of Fig. 4

Riduzione della Turbolenza(simulazioni)



Without Sheared Flow   *With* Sheared Flow

Z. Lin, Science 281,1835(1998)

Details of FFH design neutron source

Discharge pulse length time

For the evaluation of the discharge pulse length, the model presented in H Zohm Fus Sci Tech 58(2010)613- On the Minimum size of DEMO Can be used

$$\tau_{pulse} = R_0^2 \frac{c_3 q_{95} A^2 \left(\frac{A-1}{A} - \frac{b}{R_0} \right)^2 - c_4 B_t}{c_5 B_t A^2 (1 - f_{CD} - c_6 0.7 q_{95} \sqrt{A} \beta_N)}$$

b=distance between the inner plasma edge and the inner central solenoid
fCD = current fraction driven by external Current Drive power
A=aspect ratio
R0= major radius
Bt=toroidal magnetic field
q95 = safety factor at 95% of the total flux
βN= normalized beta = β/(I/aB)
c3,c4,c5,c6 are calibration constants obtained to get the ITER value.

Evaluation of the Heating power

An approximate evaluation of the heating power needed can be obtained
If lost power ≈ heating power (Ploss≈Pheat)

By definition of the confinement time:
Ploss = thermal energy of the discharge / confinement time= Wth/ τE

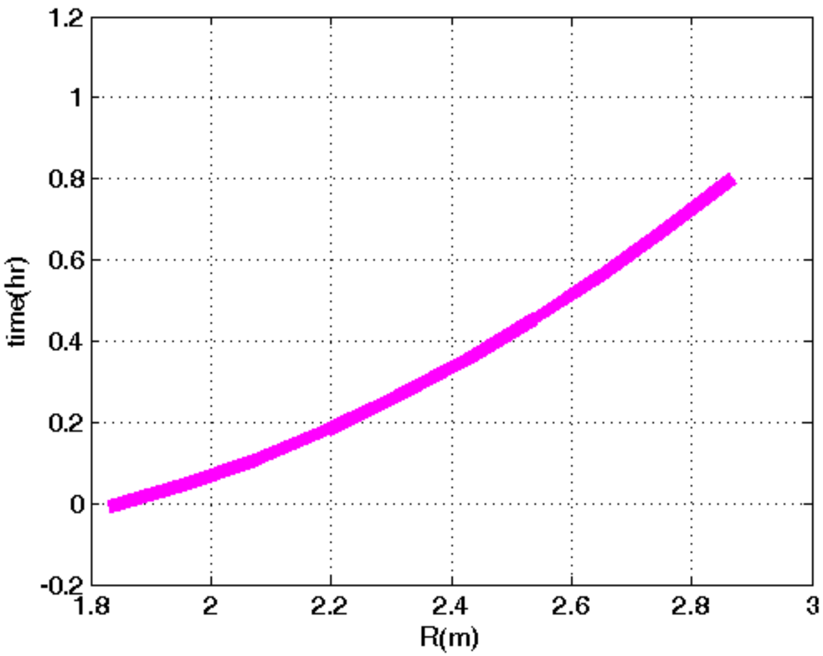
The energy confinement time scaling laws has two options related to the ITER Physics IPBy2 scaling (*)or the JET/DIIID scaling law(**) . These scalings are used freq
to define the expected range of the confinement time .
The JET/DIIID scaling is used in the context of so-called ITER-Like devices.

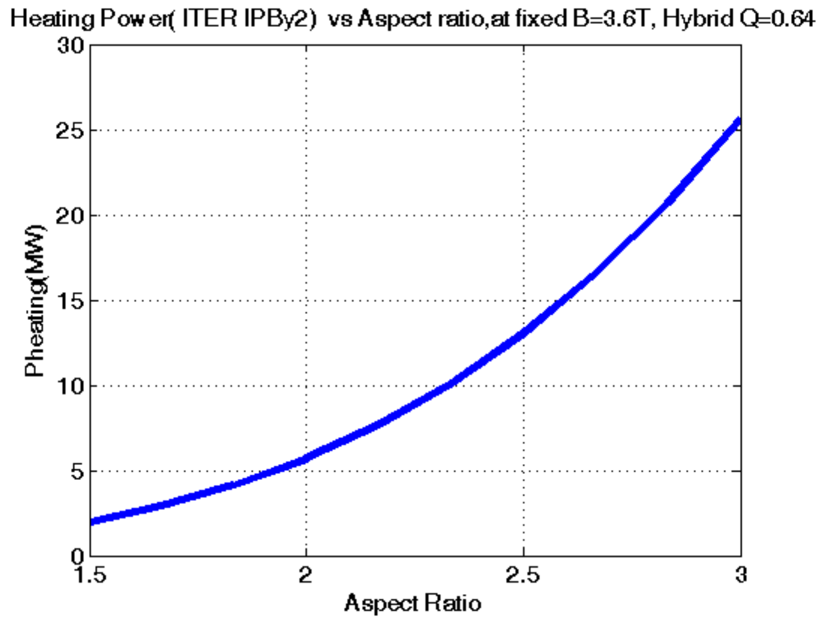
These t

in part

(*) ITER Physics Basis, Nuclear Fusion 39(1999) 2208
(**) Petty CC et al Phys Plasmas 11(2004)2514
McDonald D C et al. Plasma Physics and Controlled Fusion 46(2004)A215

Discharge pulse length and heating power time (Q=0.55)

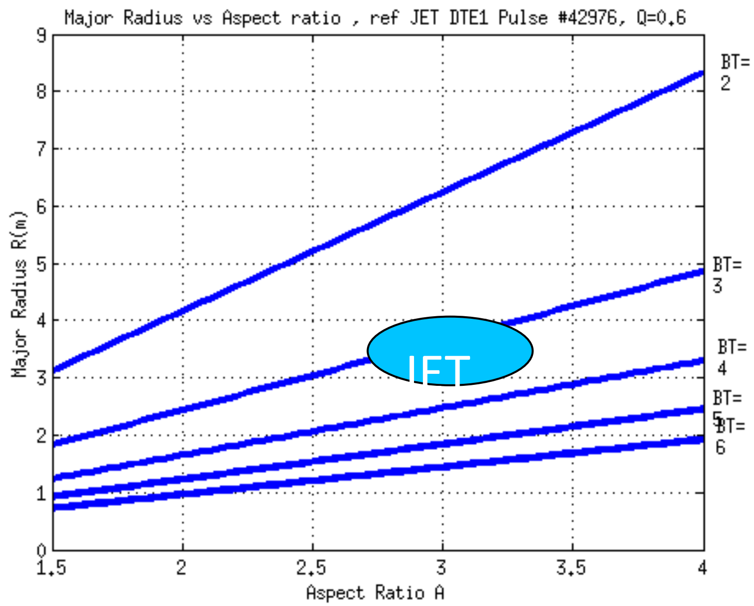




$b=0.2m$, $\beta_N=2.2$, $f_{CD}=0.1$, $B=3.6T$, $q_{95}=3.47$

H Zohm Fus Sci Tech 58(2010)613- On the Minimum size of DEMO

Parameters for Q=0.55 neutron source



Q=0.55 is achieved at B=4 T for A=2.8 and R≈2.2m, tpulse=0.2hr , Pheat(IPBy2)=18MW

Q=0.55 can be achieved at B=3.T for A=1.7 and R=2m,tpulse=0.1hr Pheat(IPBy2)=4MW

Extrapolation for Q>0.55

The choice of aspect ratio A for future machines designs is made difficult because of the paucity of experimental information for any aspect ratio different from A#3.

Recent re-analysis of the H-mode database (Verdoolage IAEA 2018) give the following scaling : [this scaling is different from the ITER IPBy2 and JET/DIIID cited before]

$\tau E \sim Ip^{1.3}\, P_{loss}^{-(0.79\text{-}0.64)}\, R^{(1.2\text{-}1.5)}.A^{(0.32\text{-}0.46)}.n^{(0.13\text{-}0.19)}.$

Taking $\tau E = I_P\ R^{3/2} P_{Loss}^{-1/2}.\ A^{1/3}$. we get :

$Q \sim (I_P\ A^{4/3})^2$. → for Q=2 we need to increase the factor

$Ip\ A^{4/3} \sim BR/(q\ A^{4/3})$ by ~2X starting from the value of Q_JET=0.55.

Q=2 can be obtained with the following parameters :

at B=6.8 T for A=2.5 and R≈2.2m

FFH MCF characteristics

For FFH we need

1.Q~2-3 machine with long pulses (say > 3 hrs)/steady state,DT plasm
PDT~80-100MW , Pin~30MW
1.1. low level of probability of disruptions:
plasma parameters chosen to be away from strong MHD
and density limits (for example with β_N<<3, n/n_{Gr}<0.8)
2.Power on the divertor definitely lower than 5MW/m^2:
in this case the problem of the divertor is easier.
3.A blanket for generation of Tritium and fission
4.A machine with high reliability , working continuously
5.All maintenence by remote handling
6.Modularity (capability of interventions on the divertor)
7.Few, simple diagnostics (the level of complexity of the diagnostics
and controls depend on the plasma scenario and on the physics mode

The scaling laws for tokamak plasmas were introduced by Kadomtsev noting that the Energy confinement is depending upon the dimensionless parameters :

A=major radius / minor radius =R/a

$\beta=nT/B^2$.= kinetic plasma pressure / magnetic pressure

ρ^*=Ion Larmor radius/ machine minor radius=$(MT)^{1/2}$ A /(R B)

ν^*= connection length/(trapped particle mean-free path)= n R T^{-2} q $A^{3/2}$.

q=safety factor = R B A^{-2} I^{-1} k

Where

R= major radius

B=magnetic field

I=plasma current

M=ion mass

n= plasma density

T=plasma temperature

K= plasma elongation

Kadomtsev-Lackner similarity

AGENZIA NAZIONALE PER LE NUOVE TECNOLOGIE, L'ENERGIA E LO SVILUPPO ECONOMICO SOSTENIBILE

Devices with equal (β,ν^*,ρ^*,q) at fixed geometry A exhibit the same confinement properties
This means that equivalent devices (plasmas with similar confinement properties) can be obtained taking fixed the scaling parameter :

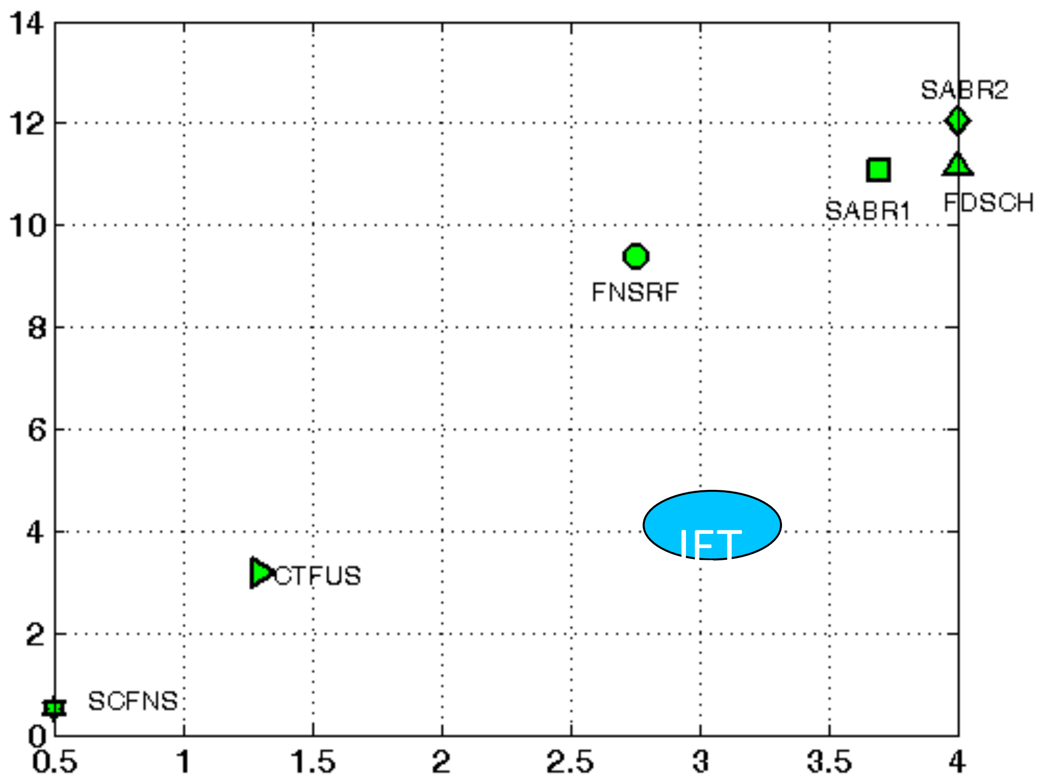
$$SK = R\,B^{4/5}\,A^{-3/2}\,.$$

Scaling laws for reactor plasmas

- For reactor plasmas(Deuterium tritium) the α-particles power (P_{α}) must be introduced as important contribution to plasma heating

- In this case(the reactor plasma) P_{α} , the gain factor Q = Pfus/Pin and the slowing down time of alpha particles(τ_{SD})
must be introduced as parameters defining the plasma state .

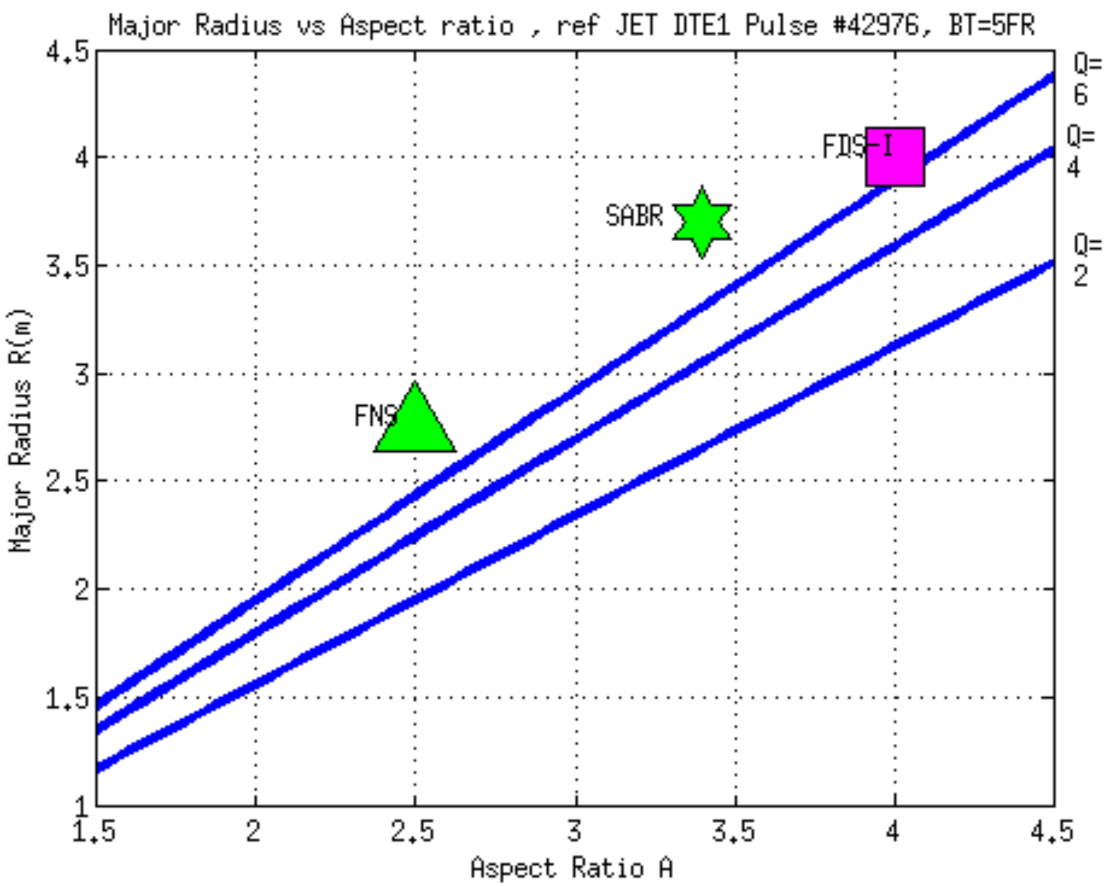
Fusion Fission Hybrid models
S_{FR} vs R



The FFH tokamak models SABR1 and SABR2 (US) FDS(China) and FNS (RF) Similar S_{FR}.

The spherical tokamaks models CTF(US) and SCFNS(super comp Fusion Neutron Source,UK) lowe S_{FR}.

Fusion reactor figures vs MODELS



The blue lines are calculated
Using the $S_{FR} = R\ \ B^{4/3}\ \ A^{-1} * Q_0^{\ 1/}$
Scaling law, fixing the magnetic
Field B =5T, and taking as referen
The plasma parameters of JET DT

The plot includes the points
Representatives of the tokamak
Models (SABR, FDS,FNS) with B≈

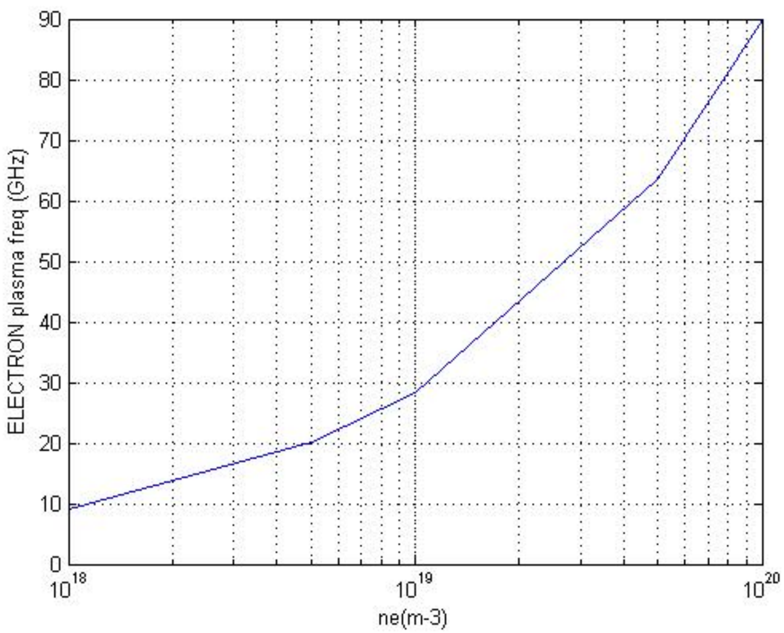
These points (SABR, FDS,FNS)
are close to the lineOf Q=6 ,
while they were calculated
For Q≈3.

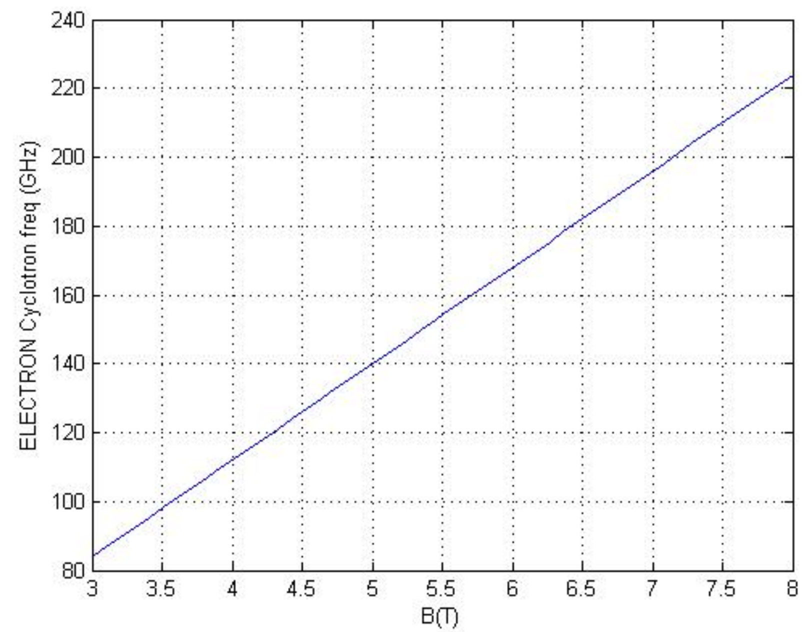
This means that the Fusion Reac
Scaling laws gives a more

Plasma and cyclotron frequencies

$$electron\ \ plasma\ \ frequency\ \ \omega_{p\_electron}/2\pi = 8.98\,(ne)^{1/2}\ \ Hz$$

$$electron\ \ cyclotron\ \ frequency\ \ \omega_{c\,electron}/2\pi = 28 * B\ \ \ GHz$$





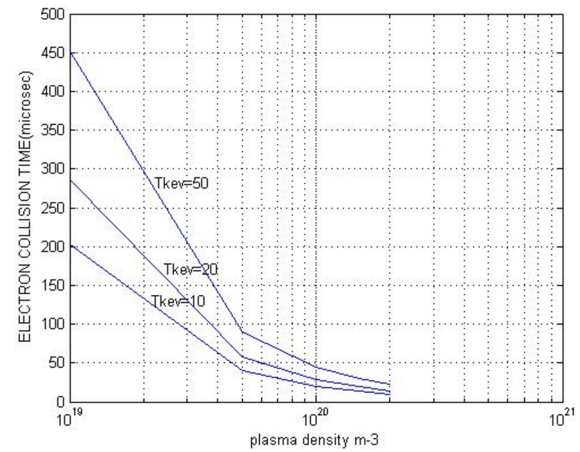
Collision times

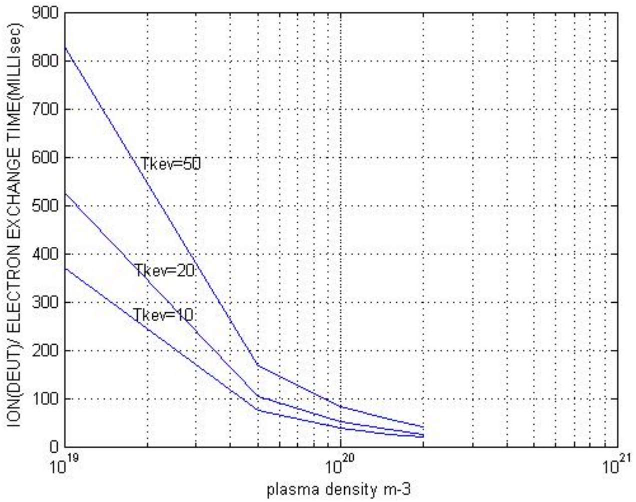
Three important times are related to collisions

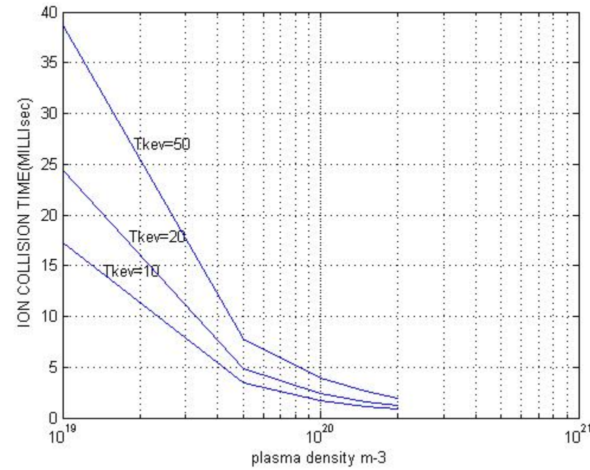
$$\textit{electron collision time} \quad \tau_e = 6.4\ 10^{14}\ \frac{(T_{ekeV})^{1/2}}{ne} \quad s$$

$$\textit{ion collision time} \quad \tau_i = 6.6\ 10^{17} \left[\frac{m_i}{m_p}\right]^{1/2} \frac{(T_{ikeV})^{1/2}}{17 * ne} \quad s$$

$$\textit{energy exchange time} \quad \tau_{ex} = \frac{m_i}{2m_e}\tau_e$$







Drift motion in a plasma

The cyclotron motion is the basic motion of a charged particle in a magnetic field.

If there is also an electric field (E) acts also togethere with a magnetic field(B) a drift of the particles perpendicular to both E and B.

This drift is indipendent of either the mass and charge of the particle

$$\vec{V}_{De} = c\,\frac{\vec{E}\times\vec{B}}{B^2}$$

Associazione Euratom-ENEA sulla Fusione

FUSION ROADMAP – key document

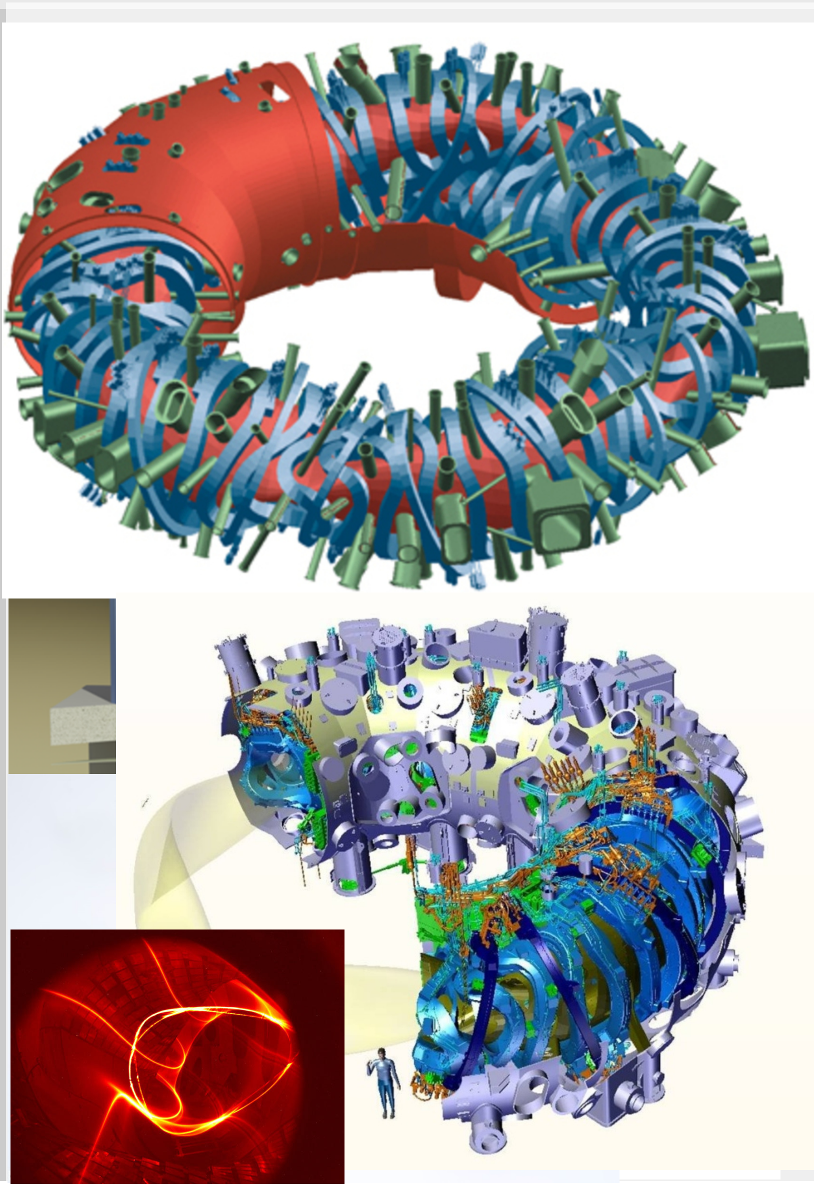
DEMONSTRATE FUSION ELECTRICITY EARLY IN THE SECOND HALF OF THE CENTURY

- Based on a number of technical assessment reports
- Provides coherent EU programme with a clear objective
- Avoids open-ended R&D

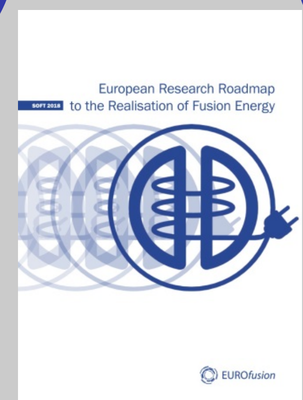
FUSION ROADMAP

- **Several important developments**
  - ITER first plasma: ≈ 2026, Q=10 2035
- **Revision is evolution not revolution, focused on**
  - Faster progress with ITER (operation) + Identify gaps in and support technology R&D programme for ITER (Examine/review relation between ITER Q=10 and various DEMO decisions)
  - Parallel paths – fully exploiting international collaboration
  - Phasing DEMO construction and operation (early start but taking advantage of later developments on ITER and elsewhere)

Roadmap missions



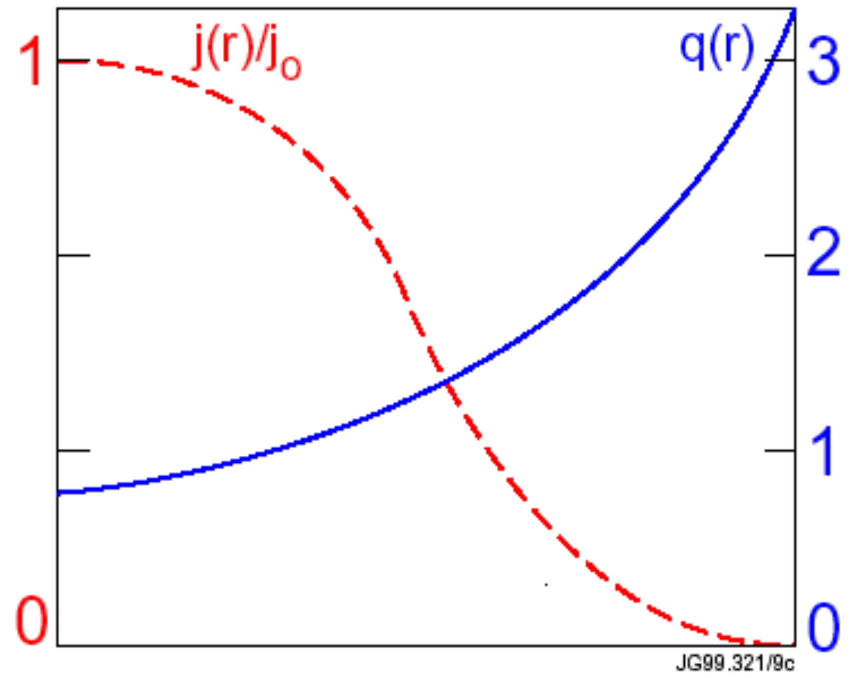
1. Plasma Regimes of Operation
2. Heat-Exhaust Systems
3. Neutron Resistant Materials
4. Tritium Self-Sufficiency
5. Implemen-tation of Intrinsic Safety Features
6. Integrated DEMO design
7. Competi-tive cost of Electricity
8. Stellarator



Safety factor and current profile

The value of q(edge) determine the stability of a discharge for example q_{edge}<2 there is a disruption

rational surfaces have constant magnetic flux where q value is rational



Associazione Euratom-ENEA sulla Fusione

Instabilities of plasma

1. Instab. MHD: at the rational surfaces where q=m/n there are resonant modes

These modes regulate the stability and transport of energy and particles

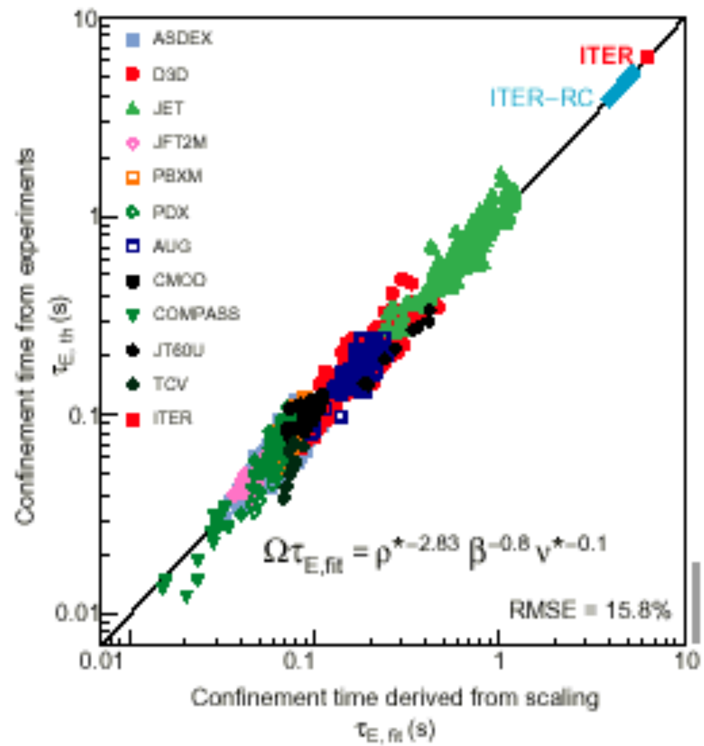
2.Turbulence: a plasma with gradients of density and temperature is unstable. Plasma modes are generated linked to the energy and particle transport

Confinement TIME

The Time scale (τ_E) of energy loss

for thermal conduction is defined by

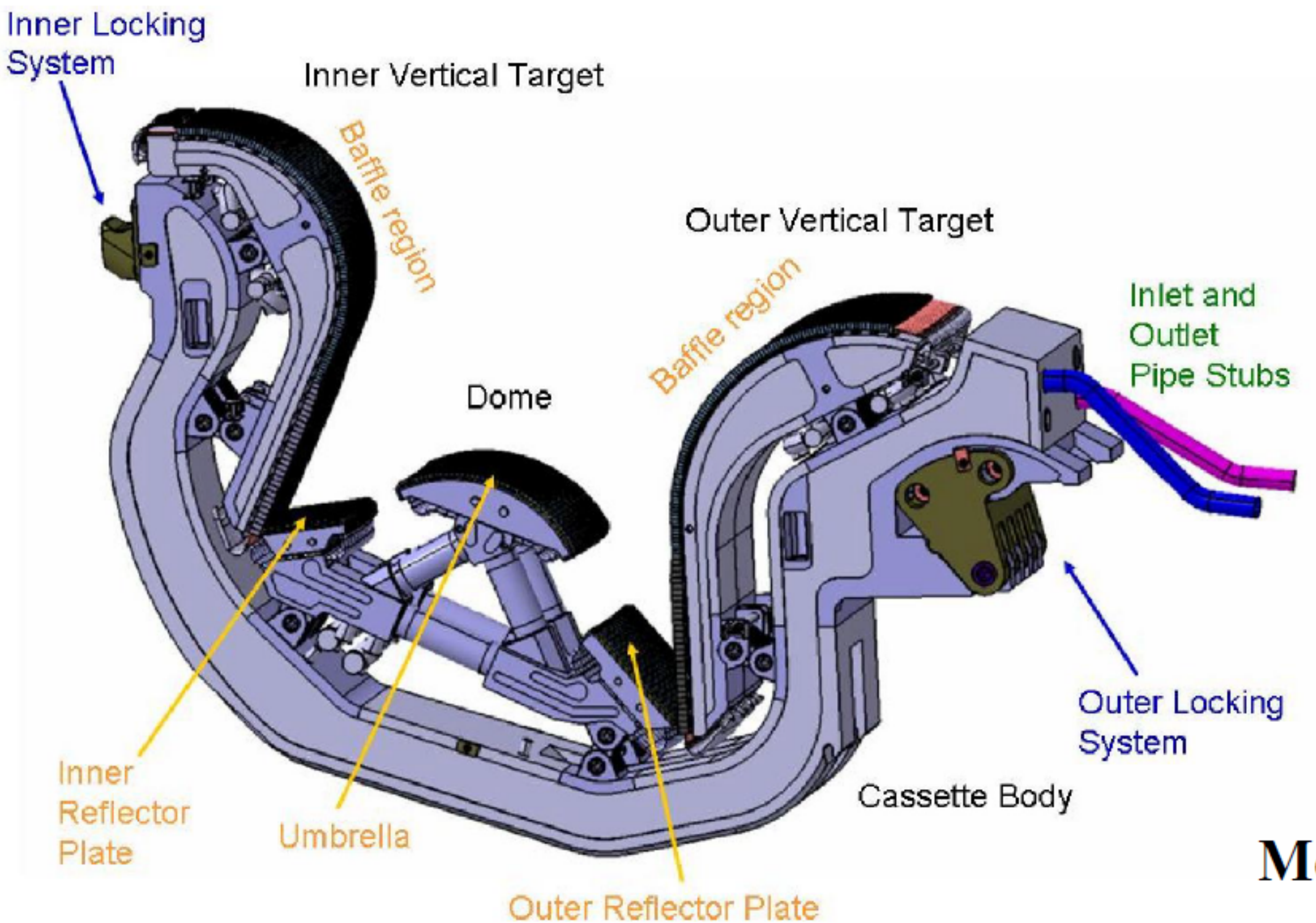
$P_L = 3nT/\tau_E = ((3/2)n_eT_e+(3/2)n_iT_i)/\tau_E$



Experimental Scaling law confinement time for magnetic confinement tokamak devices (H-mode)

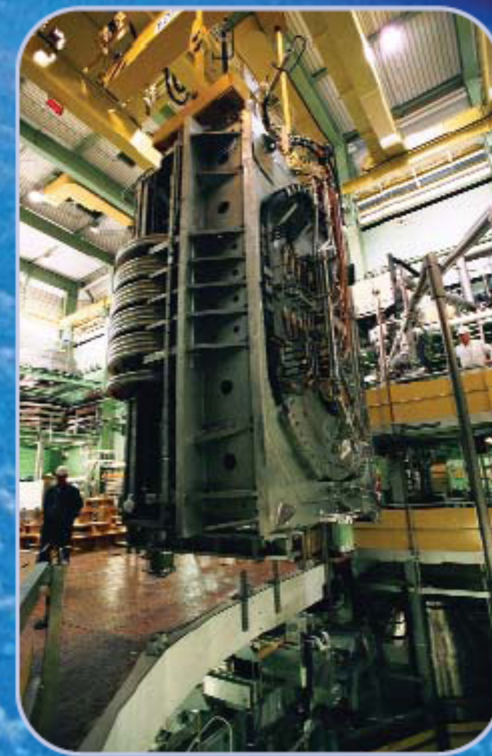
Plasma facing components - divertor cassette

– The gaps beside the dome pass the neutralised particles to the exhaust (cryo)pumps.



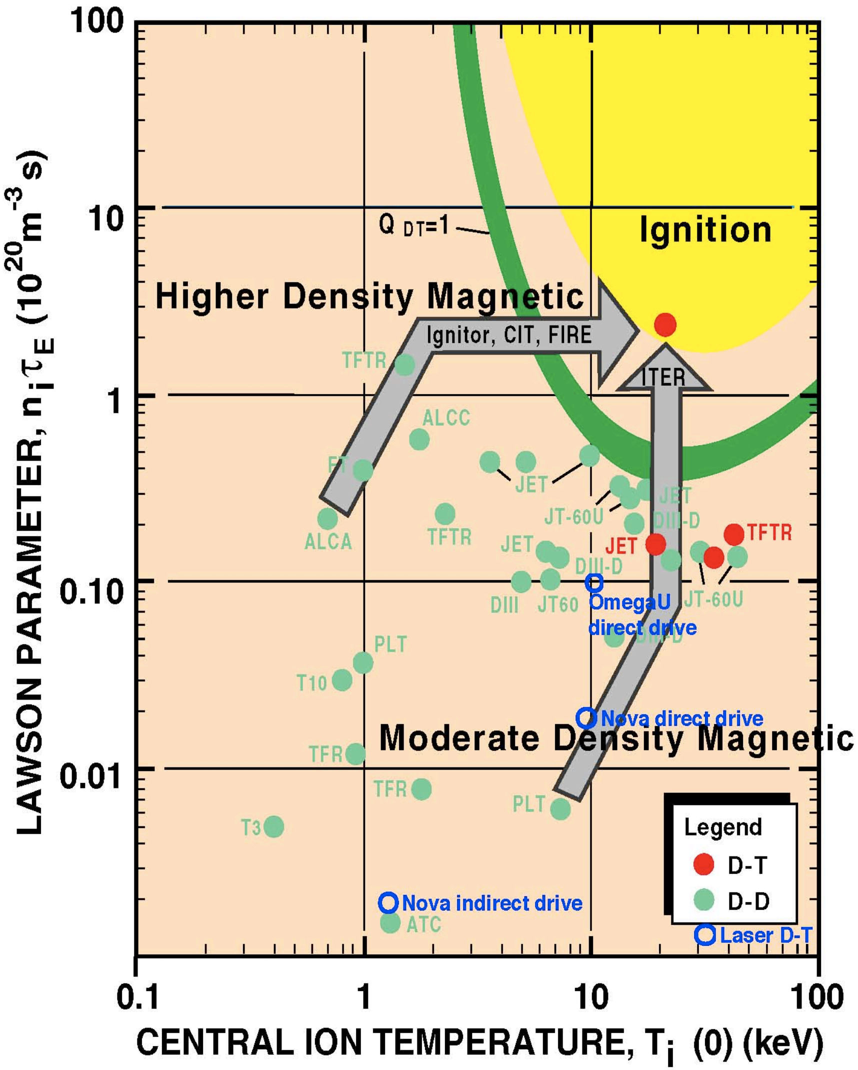
Merola

Divertor Integration Prototype



Toroidal Field Model Coil in the TOSKA test facility (Karlsruhe, Germany)

Avvicinamento all'ignizione



Kadomtsev(1975) e Connor e Taylor(1977)

$$\omega_c \tau_E \propto B\tau_E \propto f(\rho^*, \beta, \nu^*, q, ..)$$

$$\omega_{ce}\tau_E \propto \rho^{*-2.7} \beta^{-0.9} \nu^{*-0.01} \ (scaling IPB98(y,2))$$

$$\omega_{ce}\tau_E \propto \rho^{*-3.0\pm0.3} \beta^{0.0} \nu^{*-0.3}$$

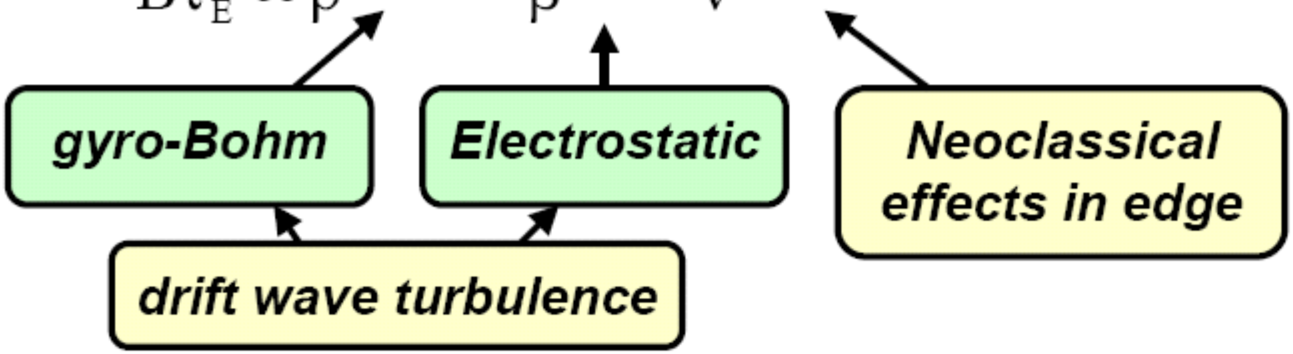
(D McDonalds and J Cordey Conf IAEA 2004,

McDonalds IAEA 2006, Valovic Nuclear Fusion 2006)

Summary of extensive single scan results

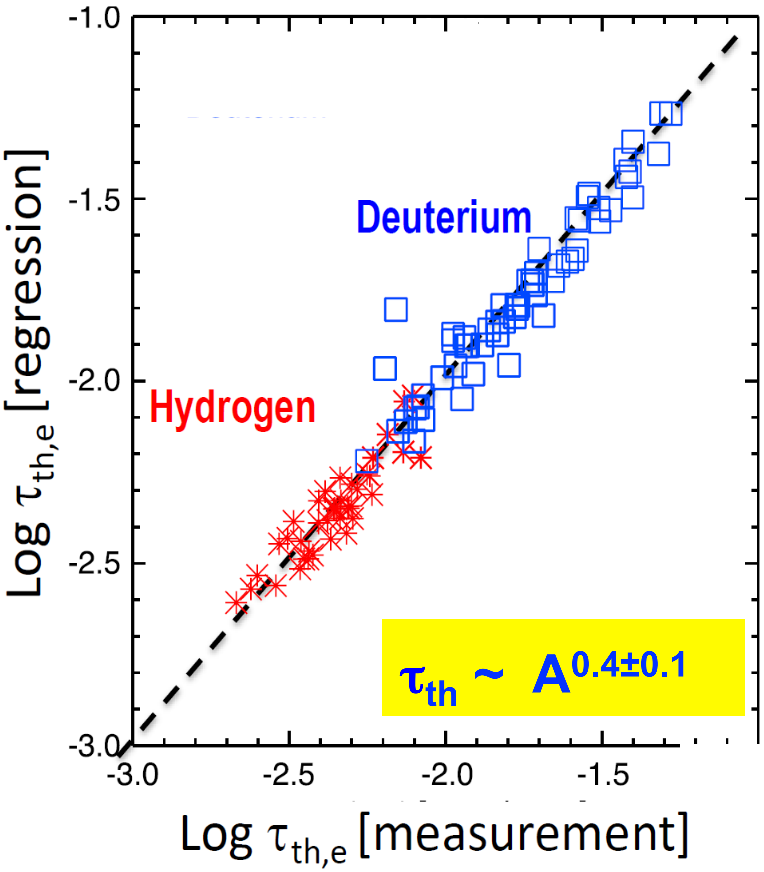
- The ρ^*, β and ν^* results may be summarised as

$$B\tau_E \propto \rho^{*\,-3.1\pm0.5}\;\beta^{\,0.0\pm0.1}\;\nu^{*\,-0.35\pm0.05}$$



- One concern: Do errors in “matched” parameters effect the scaling of the scanned parameter?
- Answer: No, as propagating error in ρ^* only affects the ν^* exponent by ±0.05

Type I ELMy H-mode: strong positive isotope dependence on thermal confinement



- Stronger isotope dependence than in JET-C and IPB98(y,2) ($\sim A^{0.2}$)
- Global momentum $\sim A^{0.5\pm0.15}$
- Global particle $\sim A^{0.5\pm0.06}$

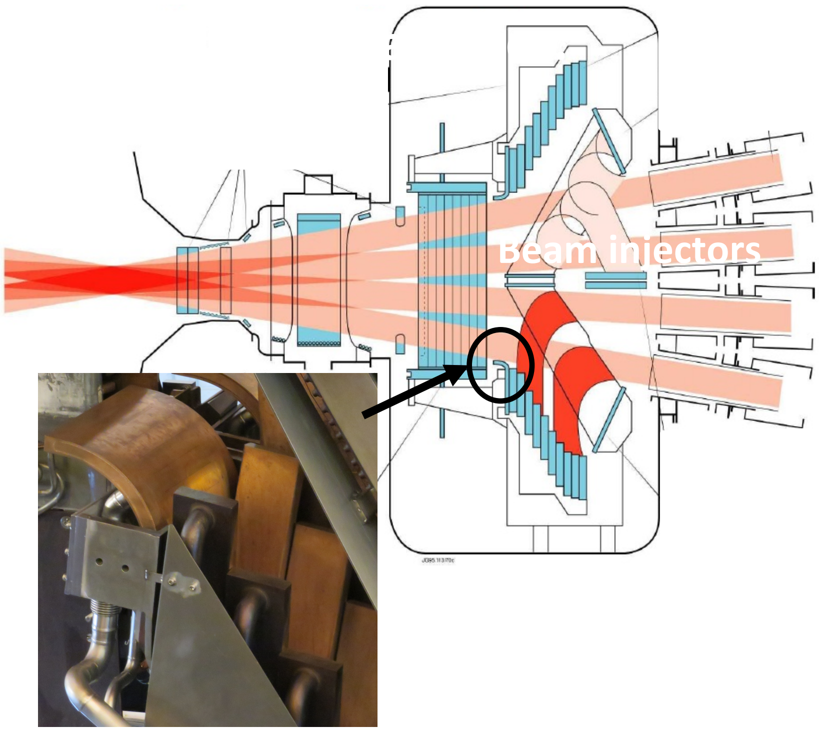
All scalings robust against the set of variables chosen for the regression.

Note: Density systematically lower in H at same external fuelling

H. Weisen, EX/P1-4
C. Maggi, PPCF 2018

The pedestal is an important player in the observed isotope dependence

Upgrades for D-T: NBI and T injection

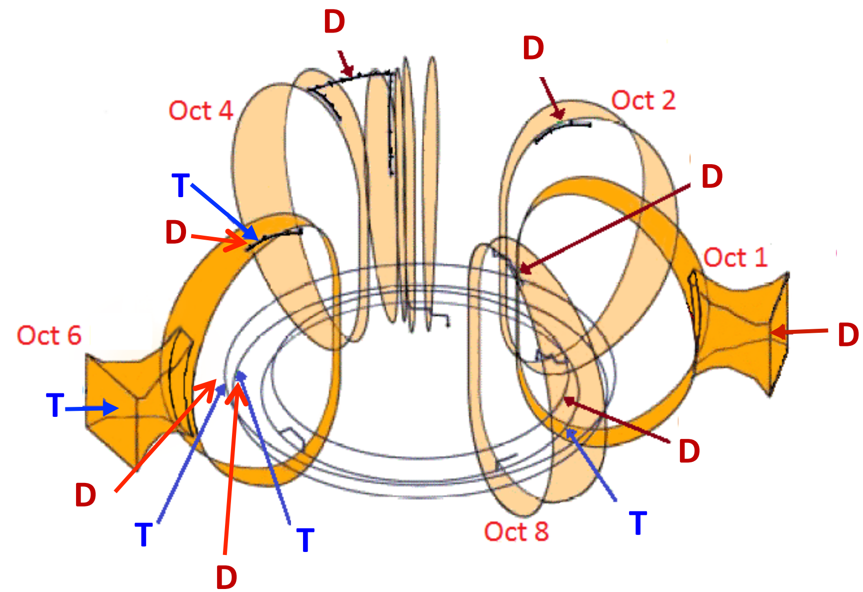


New plates

➔ 34MW in D-T

A. Shepherd, 29th SOFT 2016

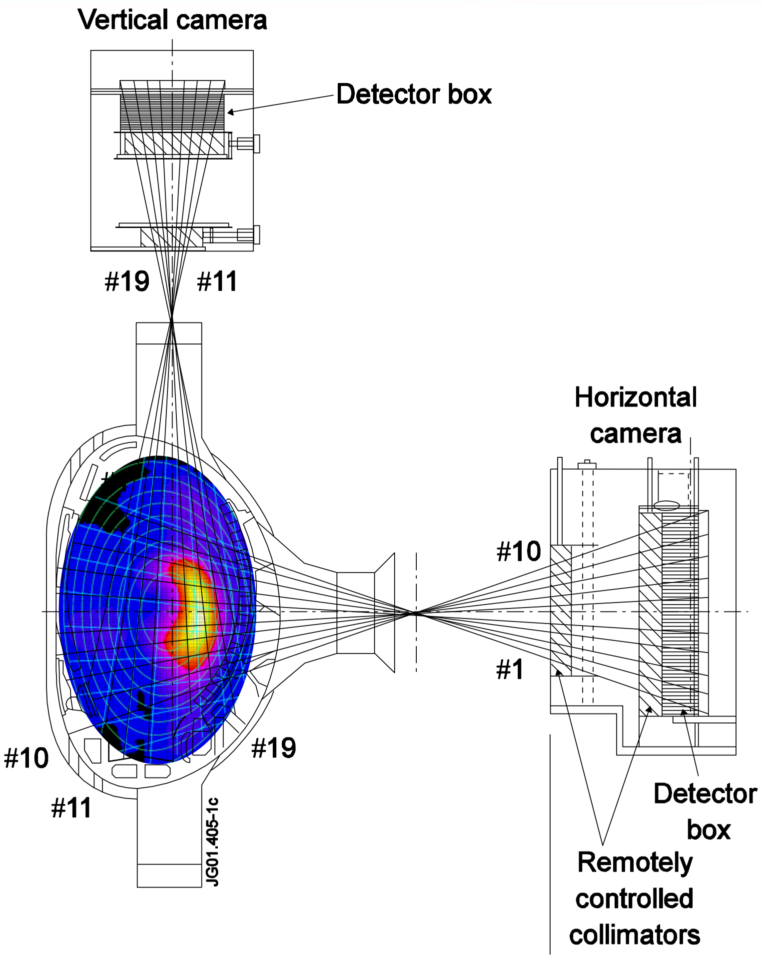
5 new tritium gas injection modules,



I. Carvalho, 29th SOFT 2016

Associazione Euratom-ENEA sulla Fusione

Unique set of new diagnostics in place for alpha physics and burning plasma studies in DTE2



burning plasma

Charge Exchange (T_i)
TAE antenna
Neutron Camera
Vertical Neutron Spectrometer
γ-Ray Camera
Horizontal γ-Ray Spectrometer for alpha-Particle Diagnostic
Upgrade of the scintillator based Fast-Ion Loss Detector (FILD)

Most diagnostics included synthetically in modelling suites for code validation

J. Figueiredo, EX/P7-42
S. Sharapov, EX/P1-28

Misure importanti per caratterizzare un plasma

Profili spaziali di temperatura e densita'   (elettroni e ioni) risolti in tempo ; flussi di neutroni

Profili del safety factor q(r)

MEASUREMENT	PARAMETER	CONDITION	RANGE or COVERAGE	RESOLUTION Time or Freq.	RESOLUTION Spatial or Wave No.	ACCURACY
28. Ion Temperature Profile	Core T_i	r/a < 0.9	0.5 – 40 keV	100 ms	a/10	10 %
	Edge T_i	r/a > 0.9	0.05 – 10 keV	100 ms	TBD	10 %
23. Electron Temperature Profile	Core T_e	r/a < 0.9	0.5 – 40 keV	10 ms	a/30	10 %
	Edge T_e	r/a > 0.9	0.05 – 10 keV	10 ms	5 mm	10 %
24. Electron Density Profile	Core N_e	r/a < 0.9	$3 \cdot 10^{19} - 3 \cdot 10^{20}$ /m^3	10 ms	a/30	5 %
	Edge N_e	r/a > 0.9	$5 \cdot 10^{18} - 3 \cdot 10^{20}$ /m^3	10 ms	5 mm	5 %
25. Current Profile	q(r)	Physics study	0.5 – 5	10 ms	a/20	10 %
			5 – TBD	10 ms	a/20	0.5
	r(q=1.5,2)/a	NTM feedback	0.3 – 0.9	10 ms	–	50 mm/a
	$r(q_{min})/a$	Reverse shear control	0.3 – 0.7	1 s	–	50 mm/a

Associazione Euratom-ENEA sulla Fusione

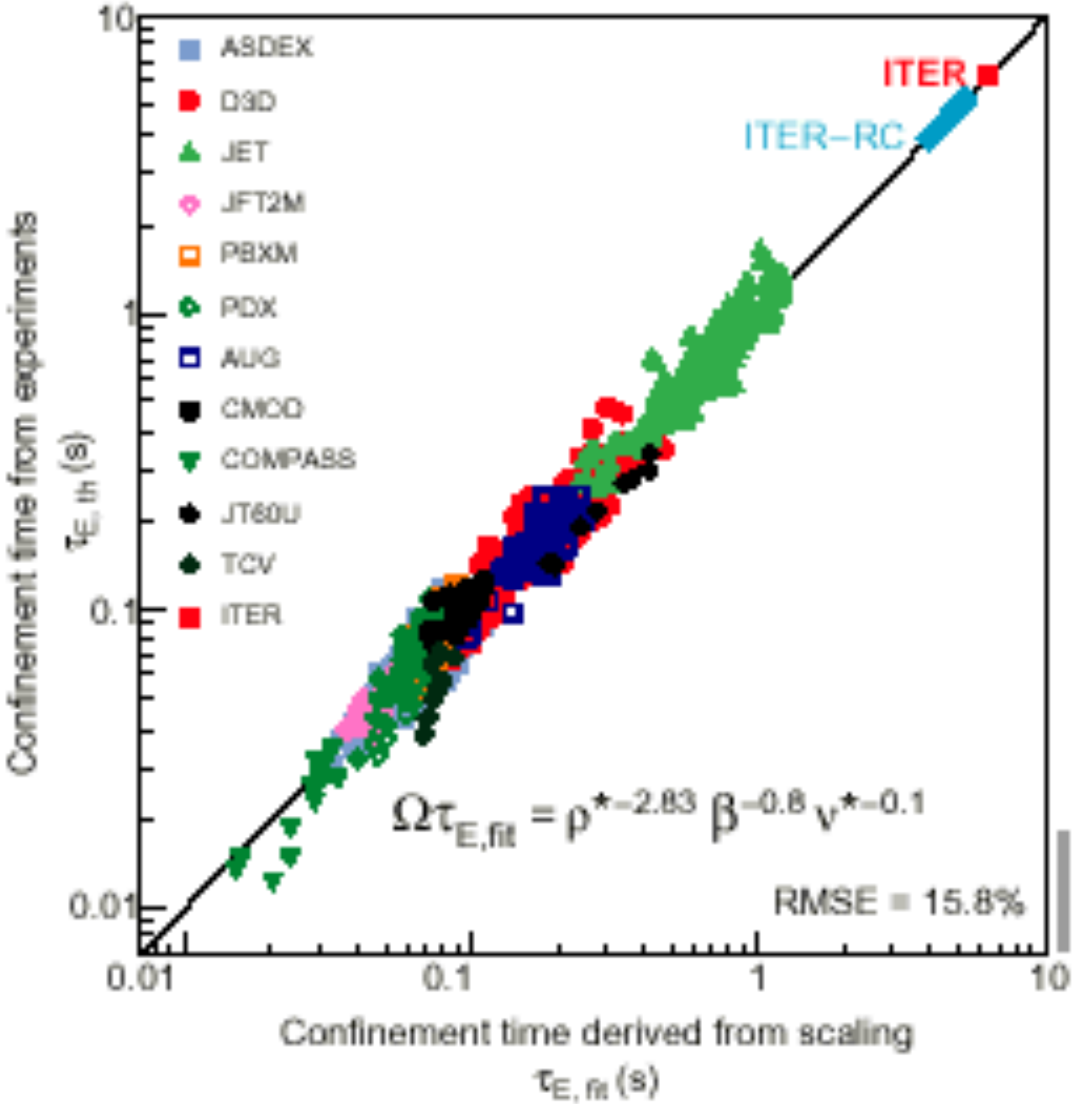
H-mode/Advanced Modes

Nell'ambito dei modi a confinanento 'migliorato' assume un ruolo importante il profilo di corrente e la sua derivata spaziale: per esempio

in scariche con profili monotoni della corrente si crea una barriera di trasporto al bordo;

mentre in scariche con profili di corrente non monotoni si creano anche barriere di trasporto interne al plasma.

H-mode



$\beta = 2nT / (B^2/2\mu_0)$

ρ^*= ion larmor radius normalized to the minor radius=ρ/a

ν^*= frequenza of collisions e-i/ (Cs/R)

Ω= electron cyclotron frequency

confinament is described by adimensional parameters Of simple physical meaning

ELECTRON HEATING by α-particles

And Stabilization of sawtooth by fast He4 ions

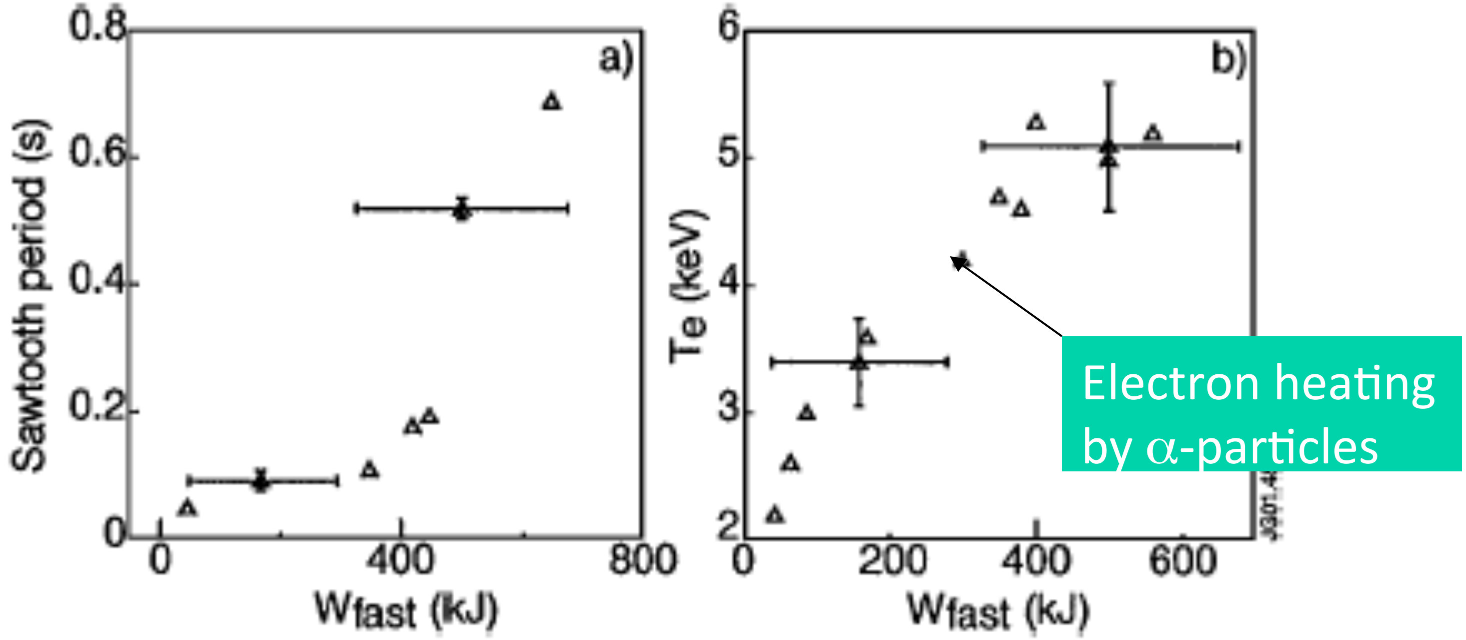


FIG. 4. Sawtooth period (a) and ECE $T_e(r/a \approx 0.25)$ (b) as functions of the fast ion energy $W_{\mathrm{fast}} = 2(W_{\mathrm{DIA}} - W_{\mathrm{th}})/3$ for discharges with $\omega \approx 3\omega_c(^4\mathrm{He})$ and 1.5 MW of 120 keV beams. Here W_{th} is the thermal stored plasma energy deduced from measured plasma densities and temperatures.

Comparison between FFH and ITER and relevance of FFH to DEMO

The Fusion Amplification Q_FFH factor is of the order of 5-7 times that possible on JET at present And about ½- 1/3 of Q-ITER.

Q_FFH = 5-7 Q_JET

Q_FFH = ½-1/3 Q_ITER

Difference with respect to ITER is the availability and pulse length :

i) 75% is supposed for FFH While for ITER presently it is 4%.

ii) Many hours(at least 3hrs) pulse or Steady state for FFH , 400s/1hr pulses for ITER

The FFH Tokamak can be considered a prototype at small scale of DEMO

The FFH availability and pulse length is that hypothesized

for a small (R0=3m) , LOW Q (low performance) DEMO

With a complex blanket :FUSION + FISSION

Criteria for determining the parameters for a MCF (Magnetic confinement fusion) neutron source

AGENZIA NAZIONALE PER LE NUOVE TECNOLOGIE, L'ENERGIA E LO SVILUPPO ECONOMICO SOSTENIBILE

The method followed is :

1. We take as reference a JET DTE1 discharge with high power (DT exp at JET pulse #42976)
2. We apply scaling laws for fusion reactors to determine the main dependences of geometry and plasma parameters compatible with a Q≥1 device

(*)B B Kadomtsev Fiz Plasmi 1(1975)531 [Sov J Plasma Phys 1(1975)295]

K Lackner , Comments in Plasma Physics Contr Fusion 13(1990)163

F P Orsitto, K Lackner, G Giruzzi, T Bolzonella 39° EPS/ICPP 2012 Stockholm paper 2.154,Physics driven scaling laws for similarity experiments

M Romanelli, F Romanelli, F Zonca – 28th EPS Funchal 2001, ECA vol 25 A(2001)697

Comparison Kadomtsev-Lackner and Fusion Reactor scalings

Kadomtsev-Lackner	Fusion Reactor
$S_{K\text{-}L} = R\ B^{4/5}\ A^{-3/2}$	$S_{FR} = R\ B^{4/3}\ A^{-1}$ stronger dependence upon the magnetic field

Possible Figures of Merit for FFH devices

- To breed a fissile isotope (or any specific isotope)

1) (N-intended removed from the machine) / (N-DT reactions in same period)

(NB some burn-up of the intended material will occur, leading to a peak in its inventory and the optimum breeding period will be smaller than that)

2) (Whole life N-intended removed from the machine) / (whole life cost)

(from concept design to decommissioning and radwaste disposal)

- To burn waste isotopes

1a) (N-target isotopes burned)/ (N-new radioactive isotopes created in same period with half-lives >5 years, in whole machine and its radwaste stream)

1b) Reduction of Bq of target isotope / increase in Bq of all other isotopes in and from the whole machine with $t_{1/2}$>1 hour

2) (Whole life change in total Bq with $t_{1/2}$ > 5 years / whole life cost

(and hope the answer is negative!)

Possible Figures of Merit for FFH devices

- To produce electrical power

  1) (Whole life kWhrs sent to the grid) / (kWhrs consumed in whole life cycle of machine including producing its fuel and radwaste disposal)

  2) (Whole life N- fissile removed from the machine) / (whole life cost) (from concept design to decommissioning and radwaste disposal)

- To produce tritium

  The usual TBR: N-tritons bred / N-tritons consumed (preferably both "whole life") (i.e. treat the fissile material as just another neutron multiplier)

- To produce neutrons

  1) (N-neutrons created anywhere in the machine over its whole life) / (whole life cost (design to decommissioning and radwaste disposal))

  2) (N-neutrons created in the machine and absorbed in the intended breeding modules, not the rest of the machine) / (whole life cost)

Possible parameters a pilot FFH experiment

PILOT experiment :
Q_FFH=1 = Qfusion*Qfission
Being Qfission~10 , Qfusion = 1/10

Taking as **ref JET DTE1** , QDT_PILOT=0.06 (1/10 of Q_JET DTE1)
Is correspoding to Pfusion=1.6MW .
This is a neutron source of 0.8 * 1.6 MW/ (2.2431 10^{-12})=
5.7 10^{17} n/s or neutron flux = 3 10^{15} n/ m^2 s (Surf JET~190m^2).
CORRESPONDING TO A FLUENCE OF 0.0067MWa/m^2. (ITER fluence 0.4MWa/m^2) related to an estimated damage of 0.05dpa in 1FPY.

Since B R /(q_{95} $A^{4/3}$)~$Q^{1/2}$.

A MEDIUM SIZE device having the FTU-like parameters (BT=5T, R=1. m , A=3 , q95=3) can reach the figure of 1.6MW fusion power, working at the same plasma parameters(β_N) of JET-DTE1.

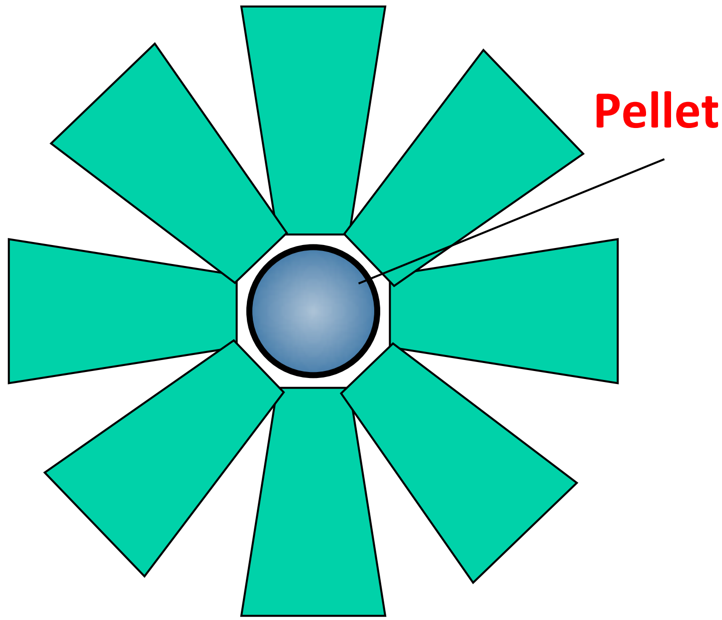
Conclusions

- The tokamak based FFH models (proposed so far) are equivalent from the point of view of Kadomtsev-Lackner similarity theory and also scaling theory of fusion plasmas.
- The application of the scaling theory of fusion plasmas taking as reference a high performance DTE1 JET discharge where

  16MW of fusion power was produced resulting in $Q \approx 0.55-0.6$

  leads to a more favorable (with respect to the magnetic field) design parameters of devices for FFH.
- As a consequence more compact tokamak can be envisaged as neutron sources for FFH application

Symmetric Direct Drive Targets



Laser beams focccused on a spherical target

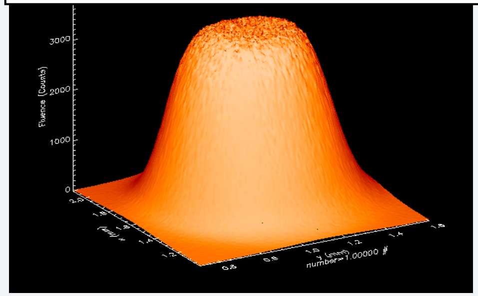
- Efficient illumination geometry…laser directly illuminates target
- Simplest targets to fabricate and recycle
- Easier to understand physics than other approaches

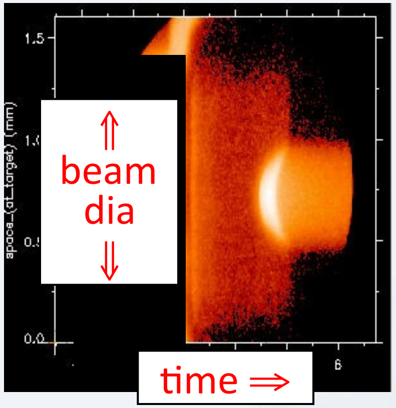
KrF Lasers have inherent advantages for fusion energy

PHYSICS: High Gain

- **Most uniform laser beam**
  Helps achieve smooth implosions
- **Shortest UV (248 vs 351 nm)**
  Better coupling to target
  Higher ablation pressures
  Higher threshold Laser Plasma Instabilities
  ⇒Target can be driven faster
- **"Zoom"** (decrease spot as pellet implodes)

Nike single beam focus



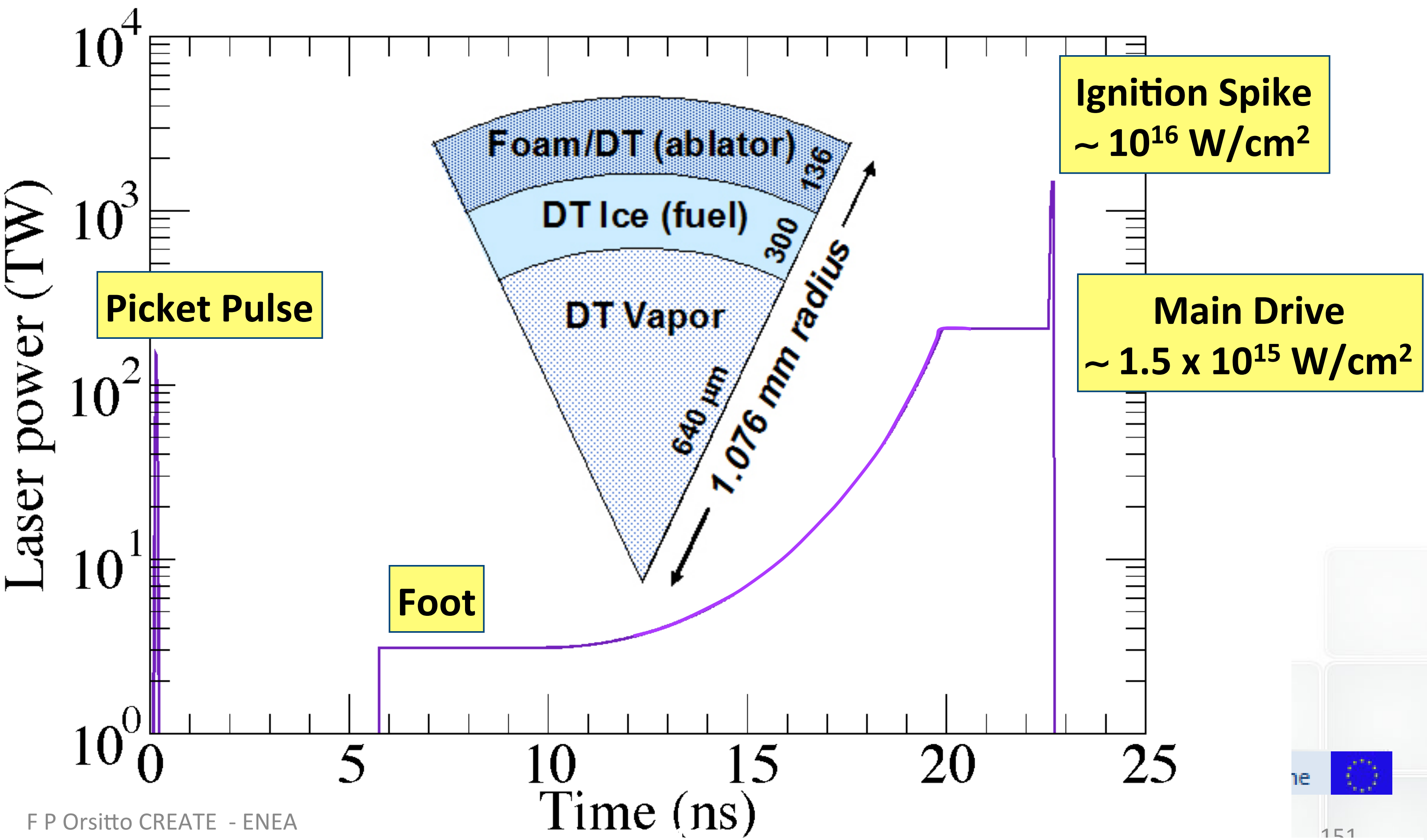


POWER PLANT: Attractive Technology

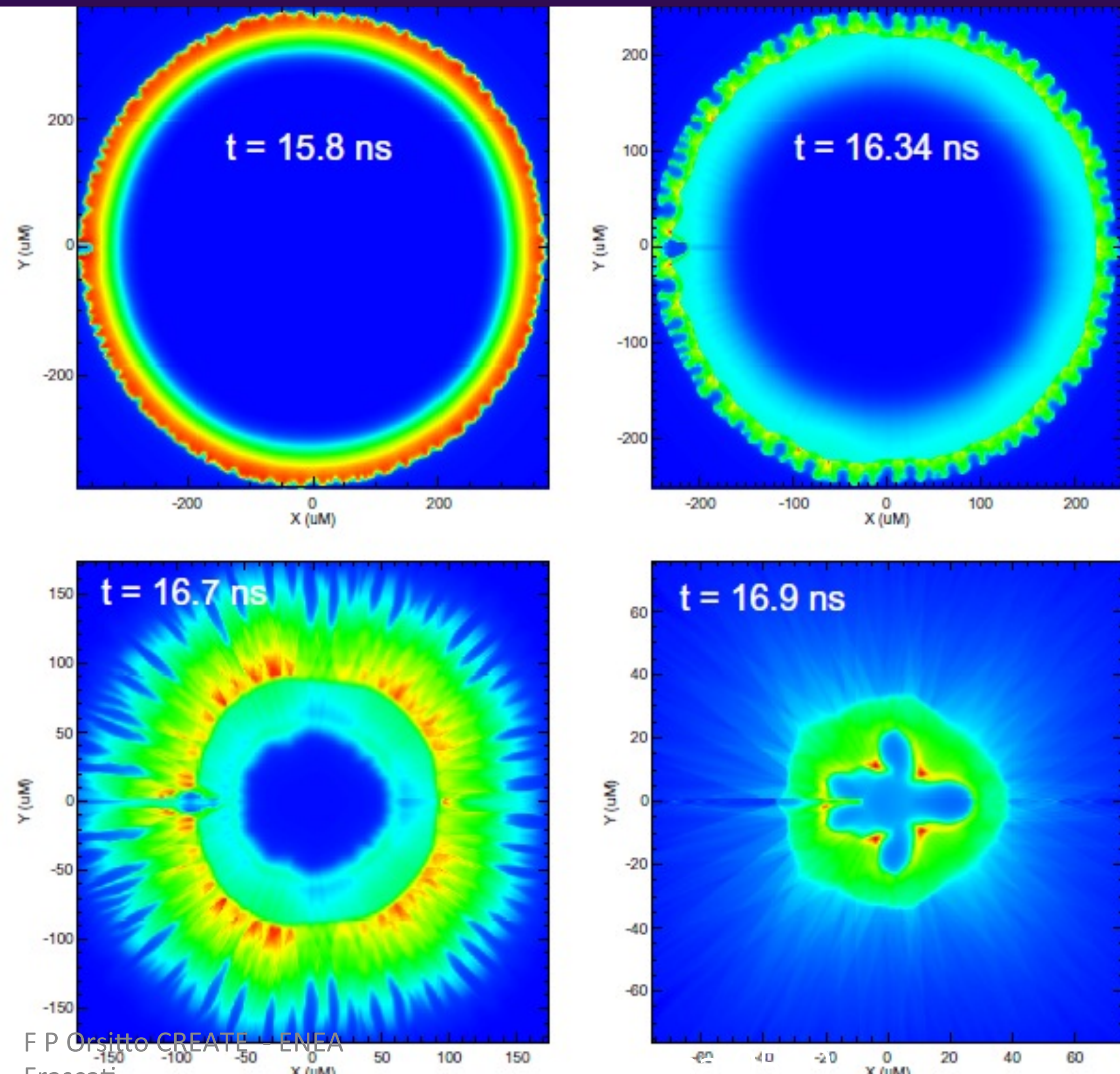
- **Gas Medium...easy to cool, durable**
- **Mostly robust industrial technology**

Shock Ignition:

Shell accelerated to sub-ignition velocity (<300 km/sec), Ignited by converging shock produced by high intensity spike



High resolution 2-D simulations show shock ignition designs robust to hydro instabilities



nominal surface finishes
(0.48 μm on wicked-foam,
1 μm in inner DT ice)
+ 1THz, 300 beam ISI

521 kJ KrF pulse
compression:
110 TW, 1.7 nsec
ignitor spike:
750 TW, 300 psec

<u>521 kJ laser</u>

2D Gain = 102

1D gain = 142