EUROfusion

Magnetic Confinement Fusion

Part 1. Tokamaks, and plasma physics in Tokamaks
Part 2. ITER for fusion, and JET for ITER

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JET

IAEA-ICTP College on Plasma Physics,
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2) Plasma physics in Tokamaks
    a) Equilibrium
    b) Stability
    c) Transport

Part 2.

1) ITER's goal
2) What we are now doing for ITER
    a) EURO*fusion* Roadmap
    b) Joint European Torus
    c) Fusion research at JET

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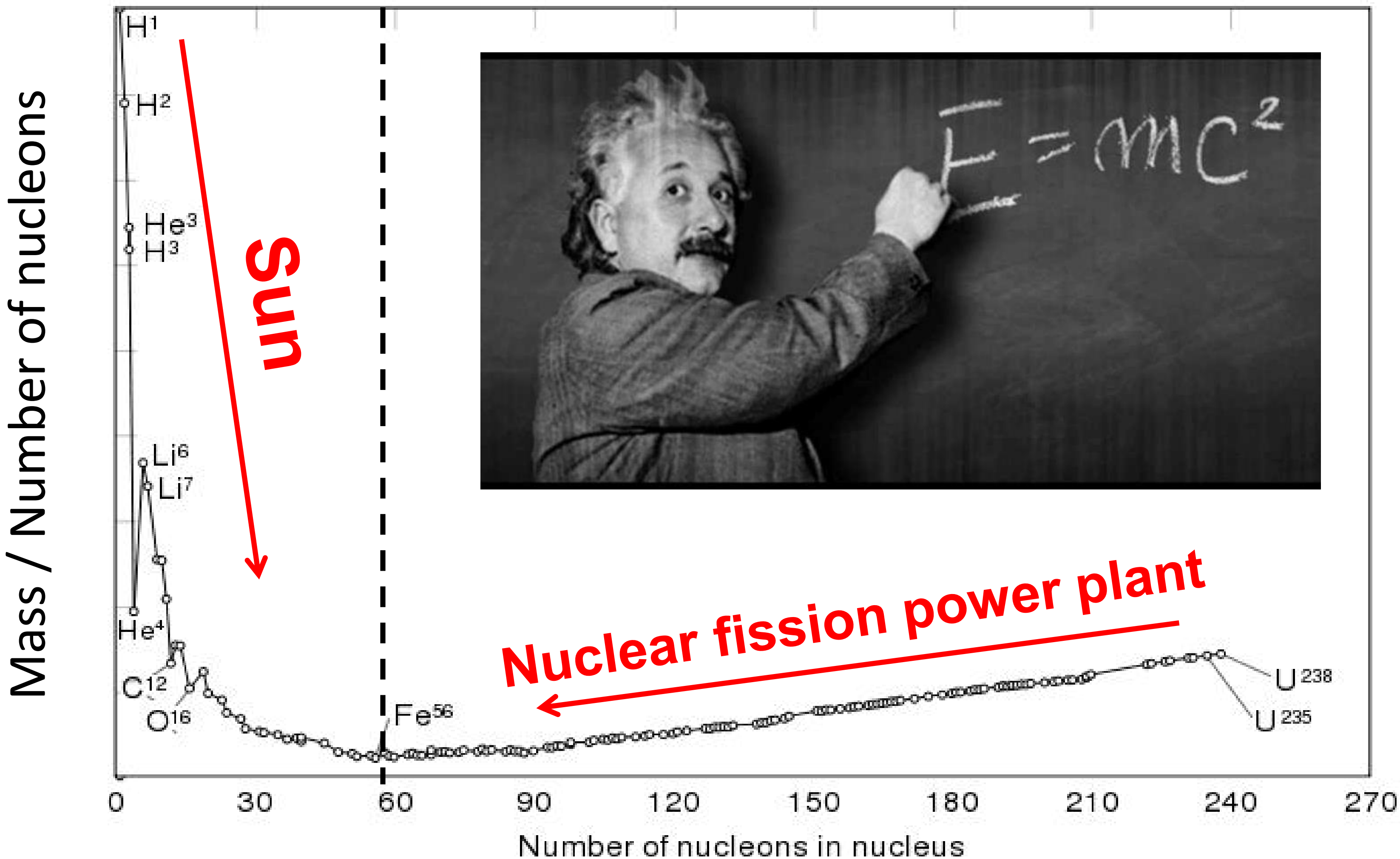
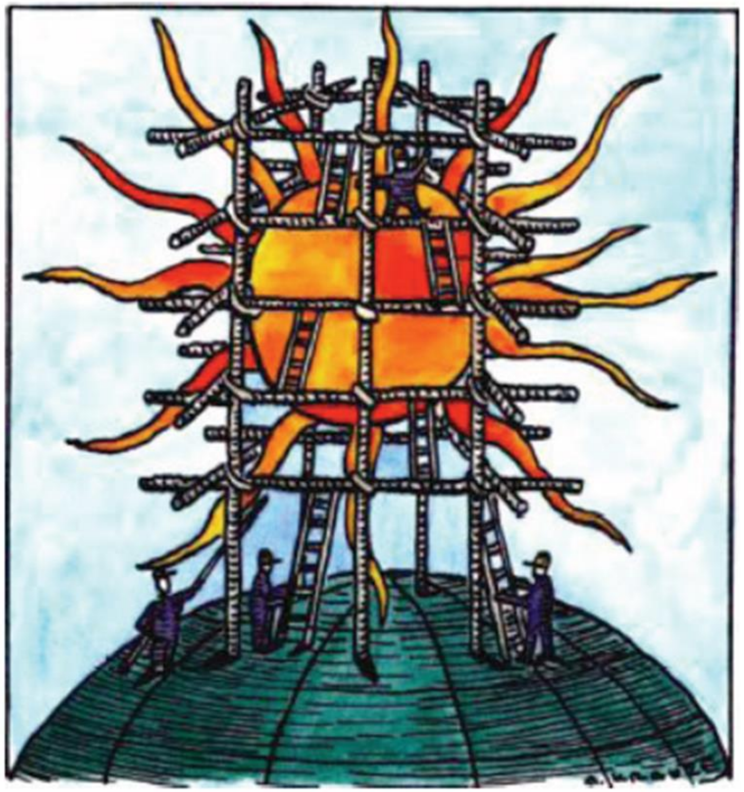
Nuclear fusion, the energy source of the sun

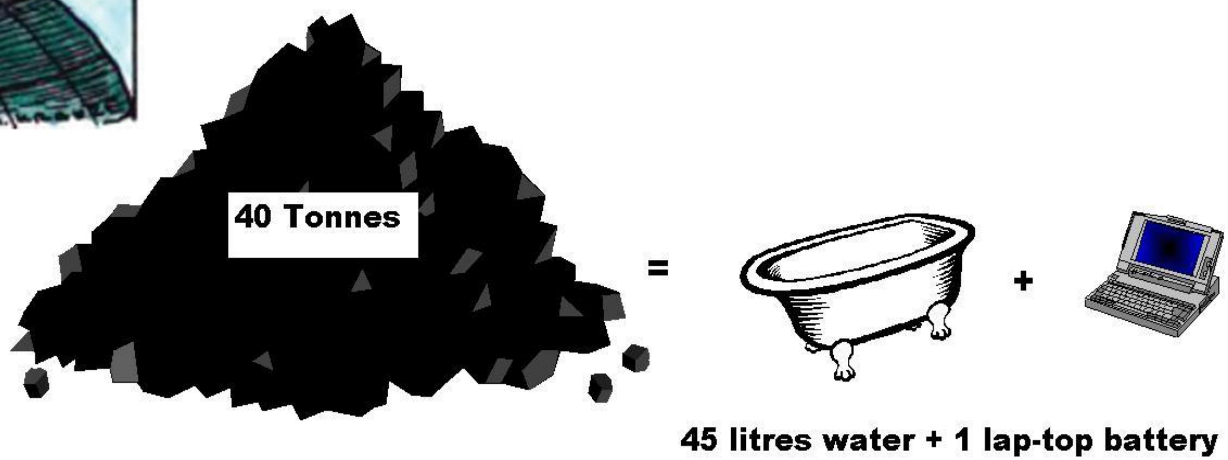


Why fusion energy?

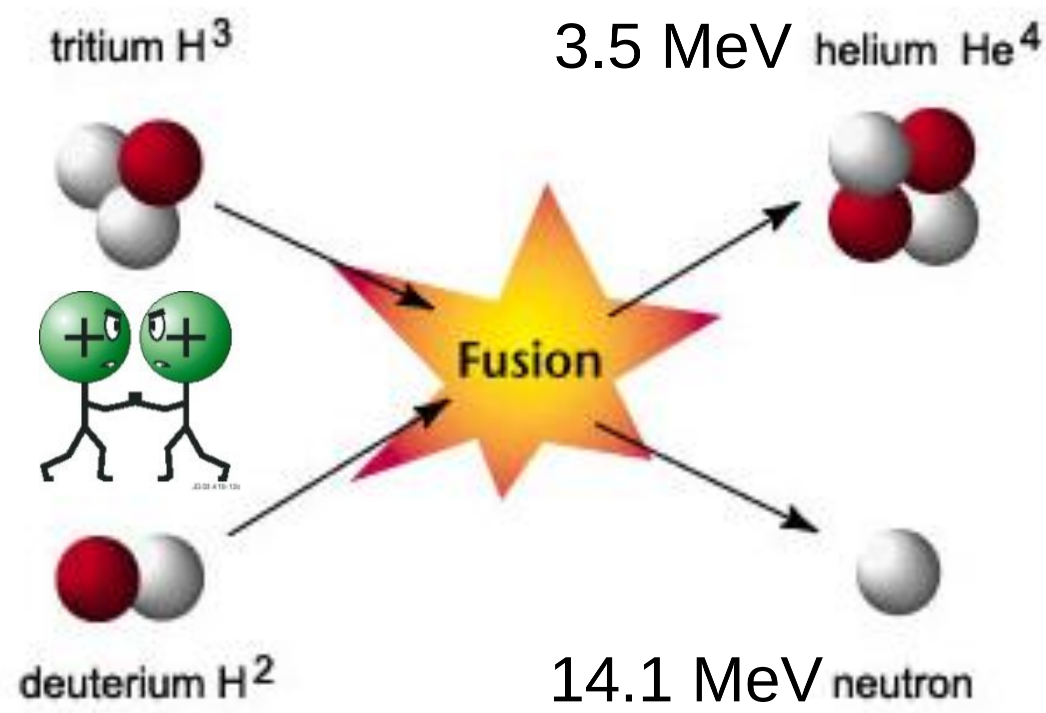


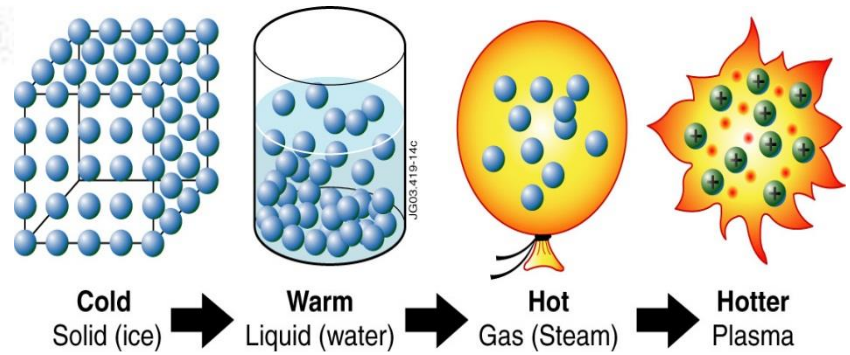
- ✓ No greenhouse emissions
- ✓ No long-lived radioactive waste
- ✓ Intrinsically safe
- ✓ Infinite fuel (>100 million years)

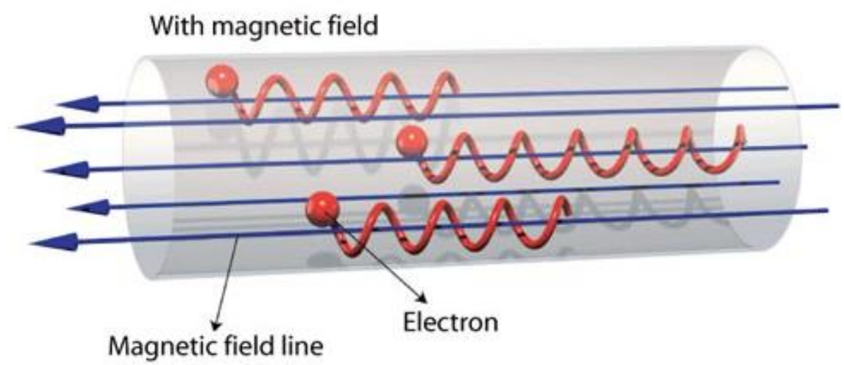
e.g. 40tons Coal = 7g D +10.5g T



How to make fusion happen?

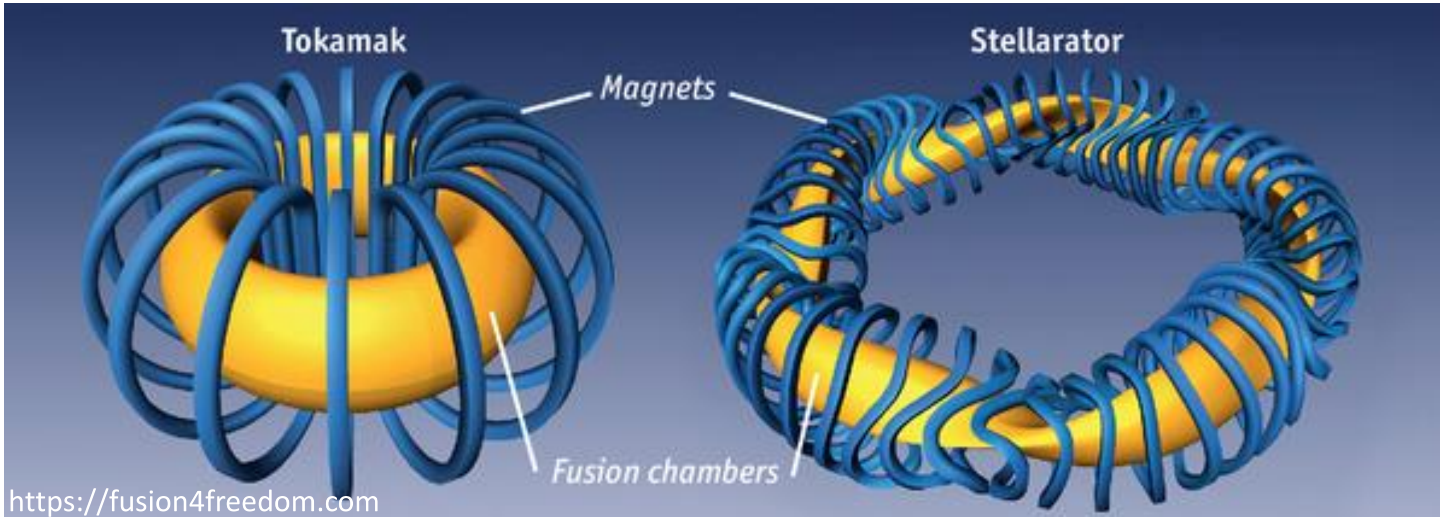




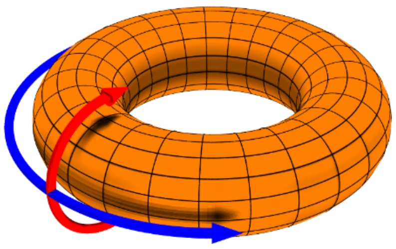


- Electrostatic repulsion between nuclei
- To overcome it, very high temperature (~10^8 °C) and high density required
- Fully ionized at such high temperature for fusion i.e. plasma
- Perpendicular motion of charged particles limited by magnetic fields
- Doughnut-shape magnetic fields needed to prevent parallel particle loss
- Thermonuclear fusion in this way is called **Magnetic Confinement Fusion (MCF).**

Two concepts of MCF devices



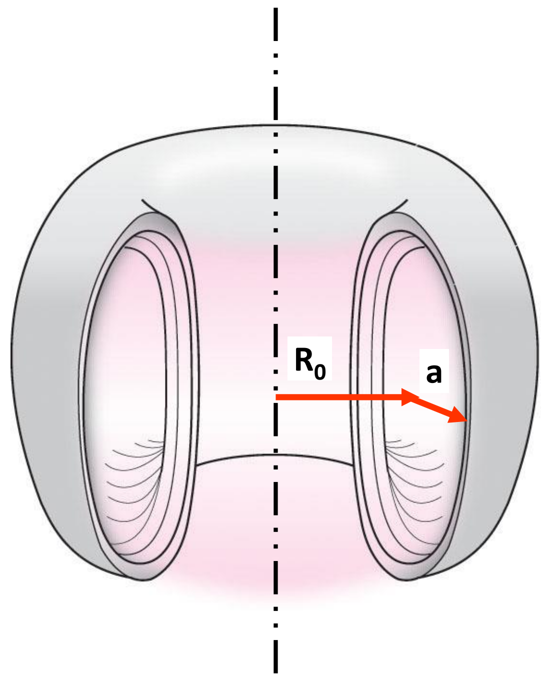
Torodial direction
Poloidal direction



- **Tokamaks** e.g. JET, K-STAR, EAST, WEST, ASDEX-U, TCV, MAST-U
- **Stellarators** e.g. Wendelstein 7-X, LHD
- Main difference: Poloidal magnetic fields are generated by
  - ✓ plasma current in Tokamaks (i.e. pulsed operation), and
  - ✓ electric current in external coils in Stellarators (i.e. steady state operation)
- So far, Tokamaks have shown higher fusion performance than Stellarators. Hence, ITER is also designed as a Tokamak. First fusion power plant is likely to be a Tokamak, although Stellarators can be a long-term alternative.
- **This lecture will focus on Tokamaks.**

Tokamak – Vacuum vessel

Low gas pressure (~5e-8 atm) is needed to make and keep a plasma.



Donut-shaped vacuum vessel

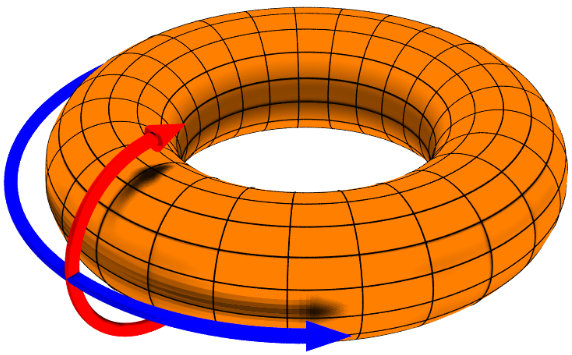
R_0 = 6.2 m, a = 2.0 m in ITER

R_0 = 3.0 m, a = 1.0 m in JET

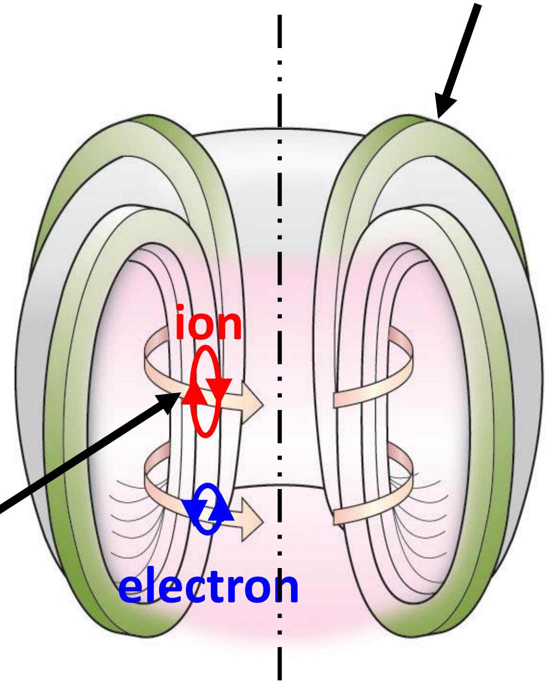
Tokamak – Toroidal Field coils

Torodial direction

Poloidal direction



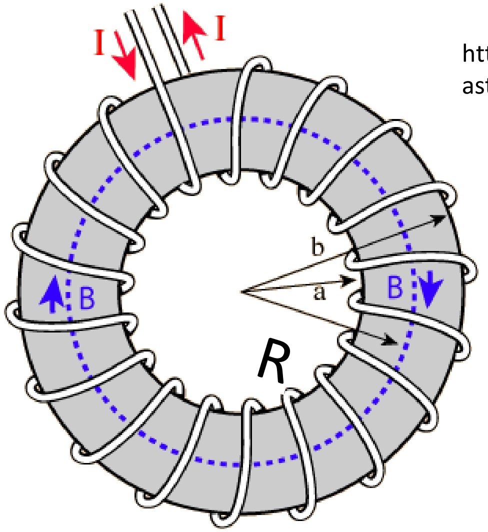
Toroidal Field (TF) coil



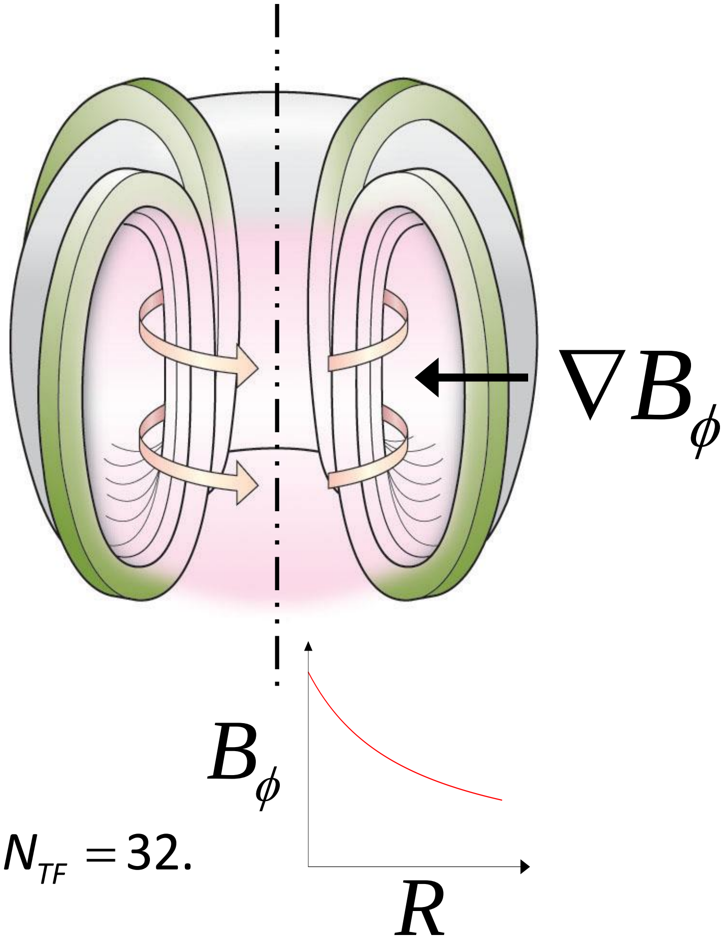
Toroidal Magnetic field

Toroidal magnetic field B_ϕ
5.3 T in ITER and
3.0 T in JET (32 TF coils with 24 turns each)

Tokamak - ∇B_ϕ



http://hyperphysics.phy-astr.gsu.edu/hbase/magnetic/toroid.html



$$\nabla \times \vec{B} = \mu_0 \vec{J}$$

$$\int (\nabla \times \vec{B}) \cdot d\vec{S} = \int (\mu_0 \vec{J}) \cdot d\vec{S}$$

$$\oint \vec{B} \cdot dl = \mu_0 N_{TF} I_{TF}$$

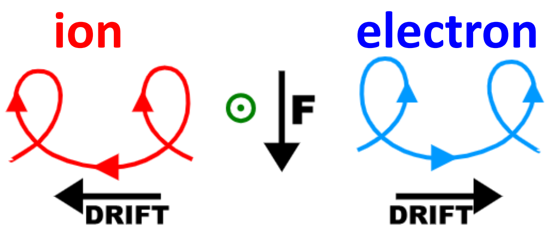
$$B_\phi = \frac{\mu_0 N_{TF} I_{TF}}{2\pi R}$$

$$B_\phi[Tesla] = 6.4 \frac{I_{TF}[MA]}{R[m]}$$, for JET with $N_{TF} = 32$.

Tokamak – Charge seperation

Drift velocity of charged particles

$$\overrightarrow{V_D} = \frac{\vec{F} \times \vec{B}}{qB^2}$$

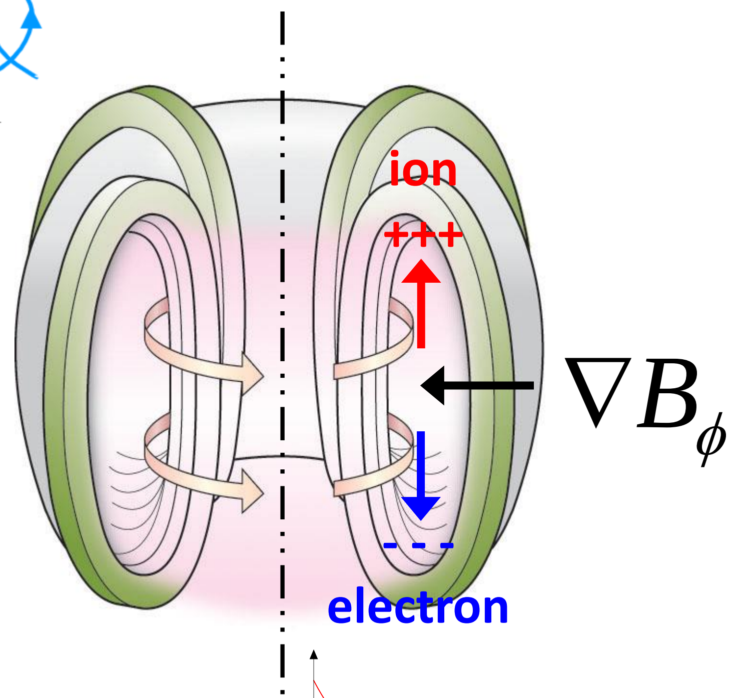


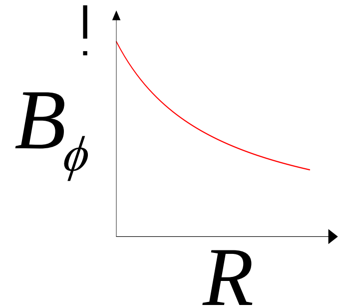
with centrifugal force $\overrightarrow{F_{cf}}$

$$\overrightarrow{V_{D,R}} = \frac{mv_{\parallel}^2}{R}\overrightarrow{e_r} \times \frac{\vec{B}}{qB^2}$$

with ∇B force $\overrightarrow{F_{\nabla B}}$

$$\overrightarrow{V_{D,\nabla B}} = -\mu \nabla B \times \frac{\vec{B}}{qB^2}$$



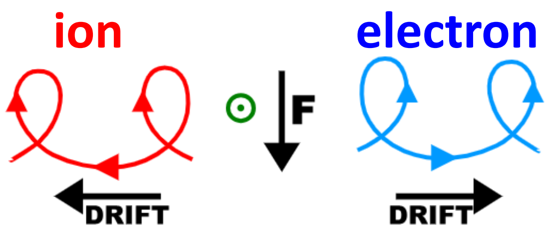


$$B(R) \propto \frac{1}{R}$$

Tokamak – ExB drift loss

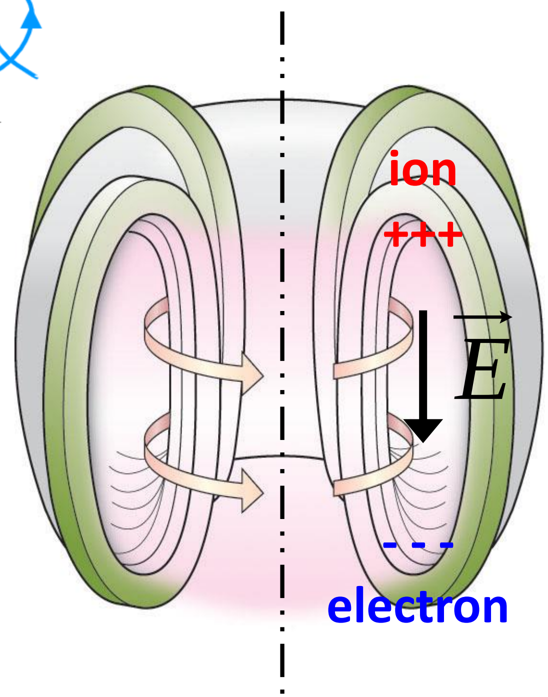
Drift velocity of charged particles

$$\overrightarrow{V_D} = \frac{\vec{F} \times \vec{B}}{qB^2}$$



with electric force $\overrightarrow{F_E}$

$$\overrightarrow{V_{D,R}} = q\vec{E} \times \frac{\vec{B}}{qB^2}$$

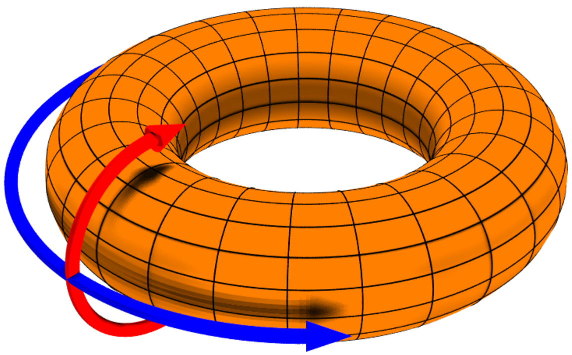


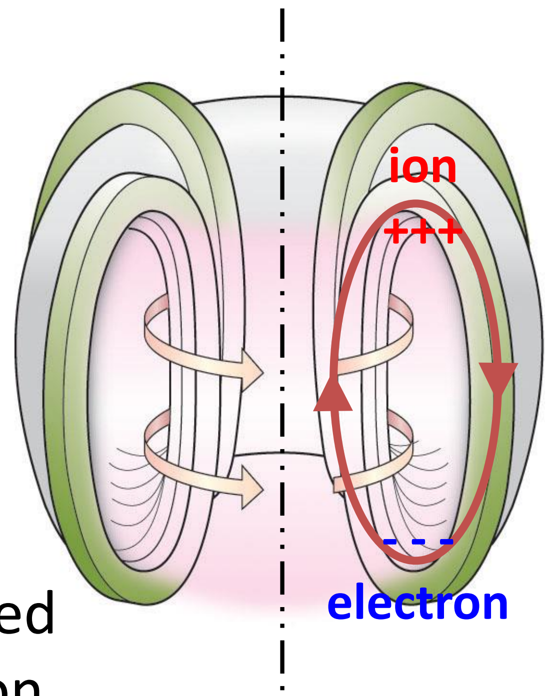
Particle loss
due to ExB drift

Tokamak – ExB drift loss

Torodial direction

Poloidal direction



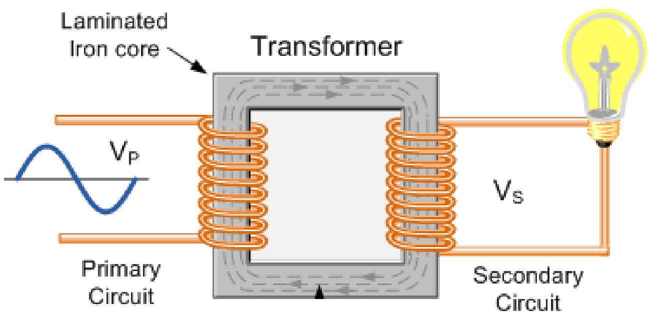


Poloidal magnetic field needed
to avoid the charge seperation
(and to stabilize plasma instabilities)
→ Plasma current needed

JET

Tokamak – Central Solenoid for V_l

http://www.electronics-tutorials.ws/electromagnetism/electromagnetic-induction.html

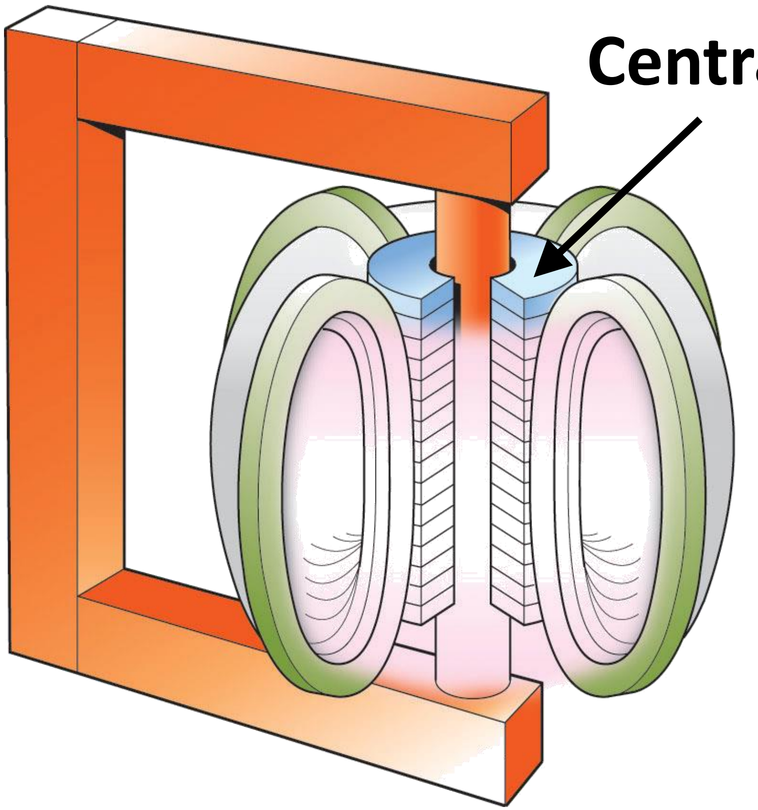


Faraday`s law

$$V_l = -\frac{d\Phi_{cs}}{dt}$$

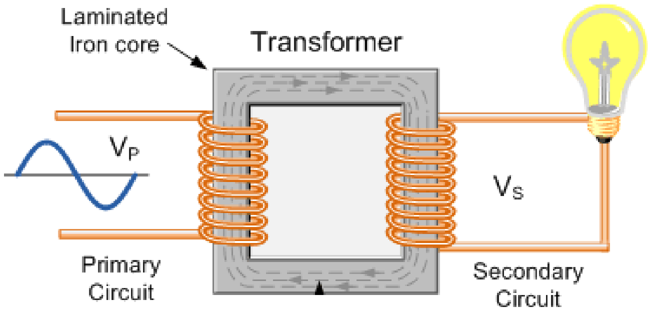
where $\Phi_{cs} \propto I_{cs}$

Central Solenoid (CS)



Tokamak – I_p and B_p

http://www.electronics-tutorials.ws/electromagnetism/electromagnetic-induction.html

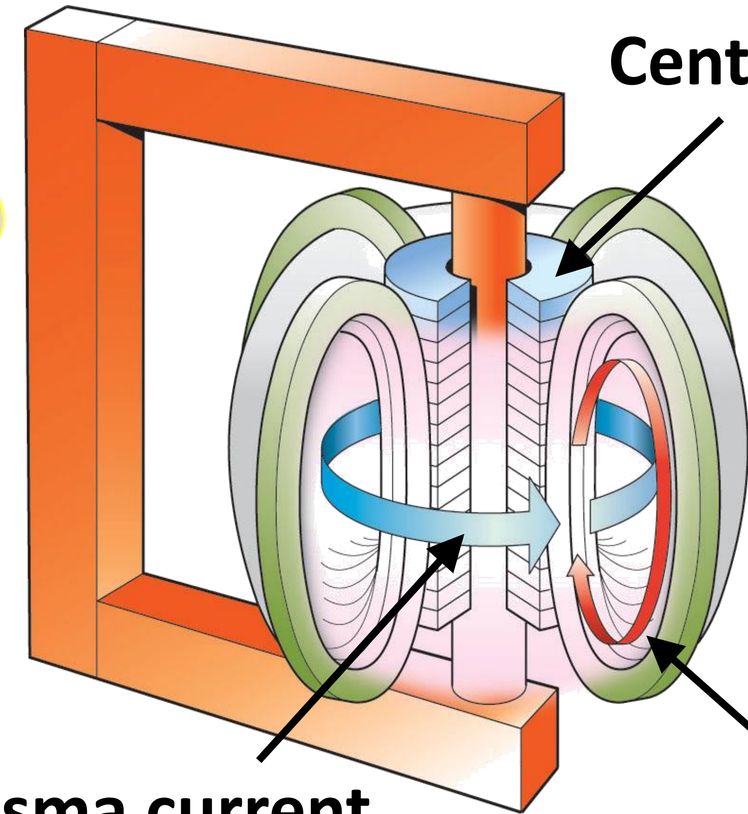


Faraday`s law

$$V_l = -\frac{d\Phi_{cs}}{dt}$$

where $\Phi_{cs} \propto I_{cs}$

Central Solenoid (CS)



Plasma current

$$I_p = \frac{V_l}{R_p}$$

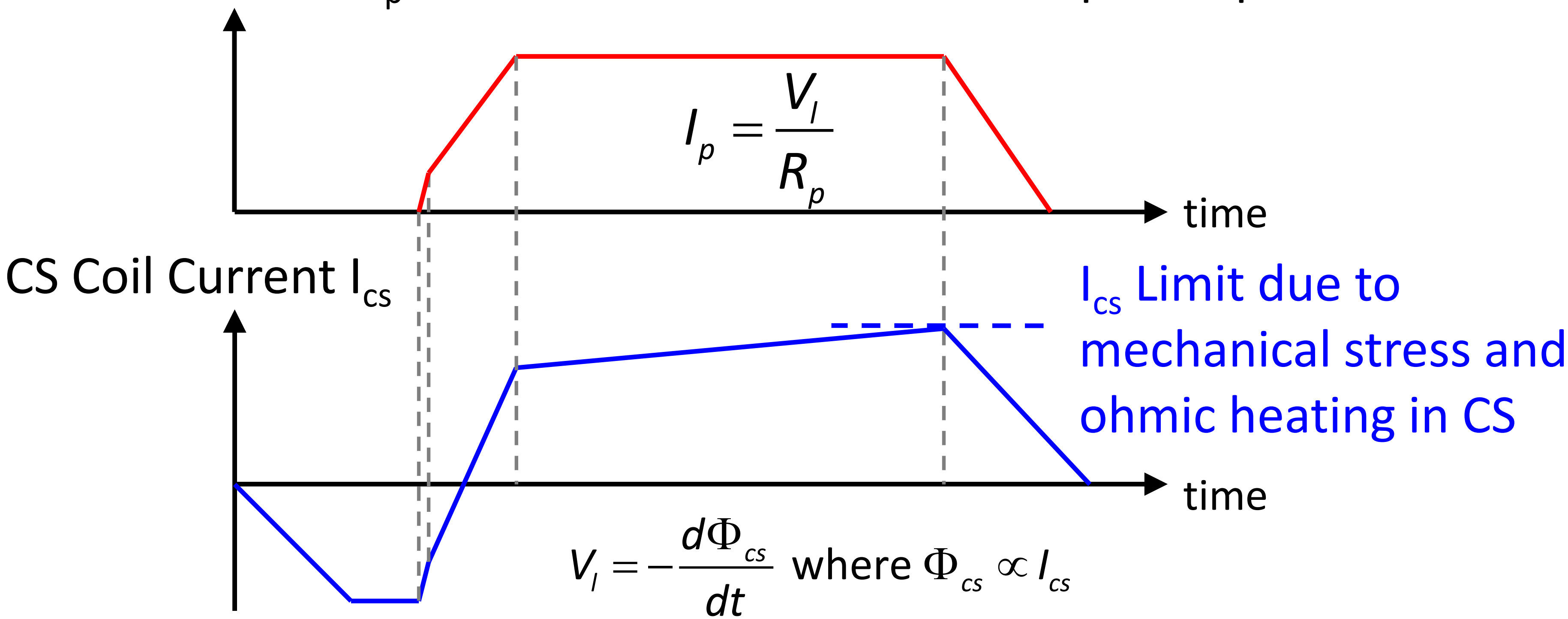
Poloidal magnetic field

$$B_p \propto I_p$$

JET

Tokamak - Pulsed operation

Inductive I_p drive → pulsed operation
Intrinsic problem in tokamaks.
Not desirable for power plant



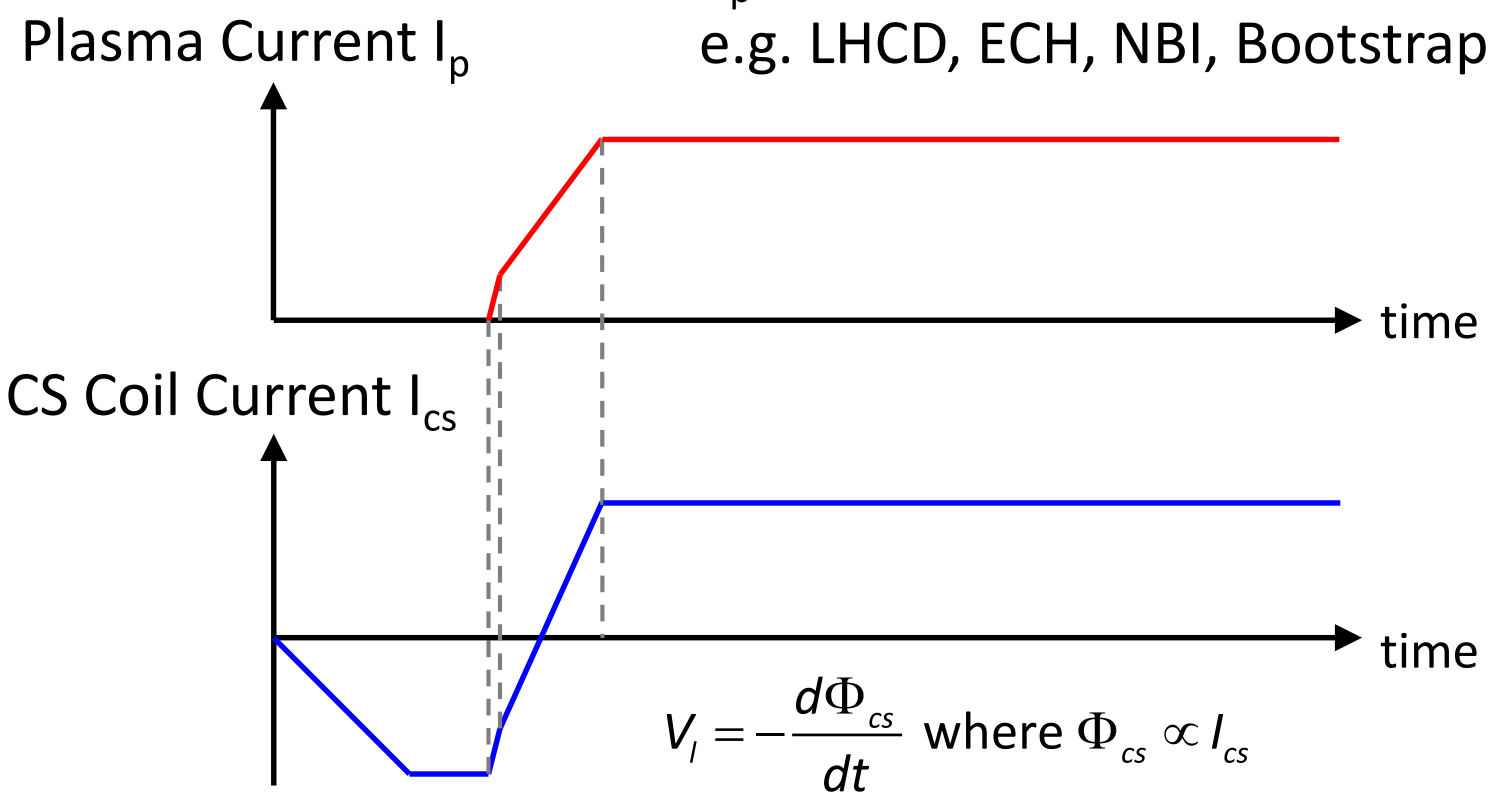
$$V_l = -\frac{d\Phi_{cs}}{dt} \text{ where } \Phi_{cs} \propto I_{cs}$$

$$V_l \propto -\frac{dI_{cs}}{dt}$$

Courtesy of Yong-Su Na (SNU)

Tokamak - Steady state operation

For steady-state operation,
I_p should be driven by *non-inductive* ways
e.g. LHCD, ECH, NBI, Bootstrap current

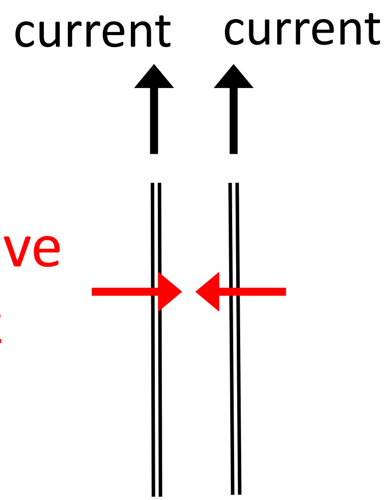


$$V_l = -\frac{d\Phi_{cs}}{dt} \text{ where } \Phi_{cs} \propto I_{cs}$$

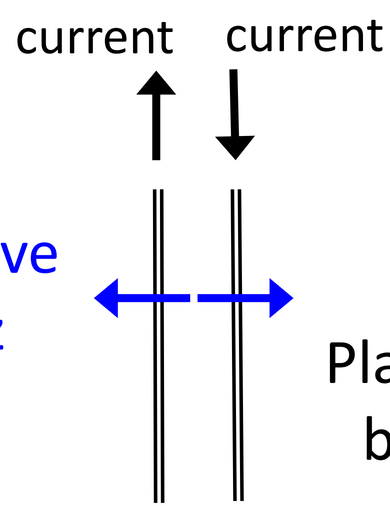
$$V_l \propto -\frac{dI_{cs}}{dt} \Rightarrow V_l = 0$$

Courtesy of Yong-Su Na (SNU)

Tokamak – Poloidal (vertical) field coils

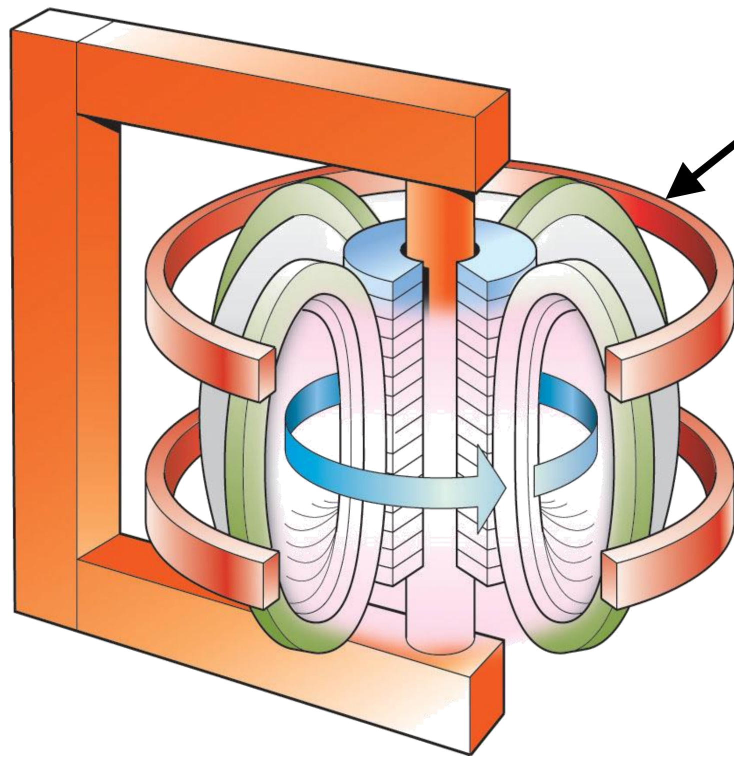


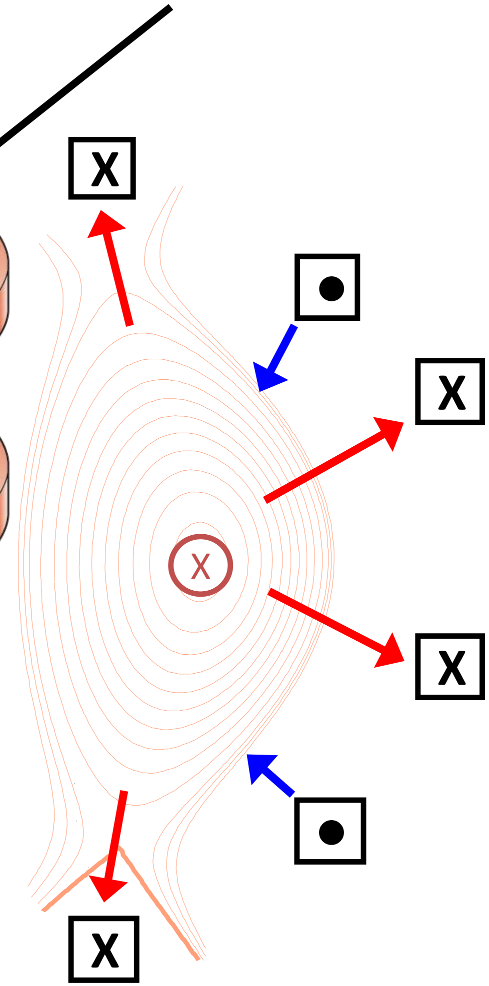
Attractive
Lorentz
Force



Repulsive
Lorentz
Force

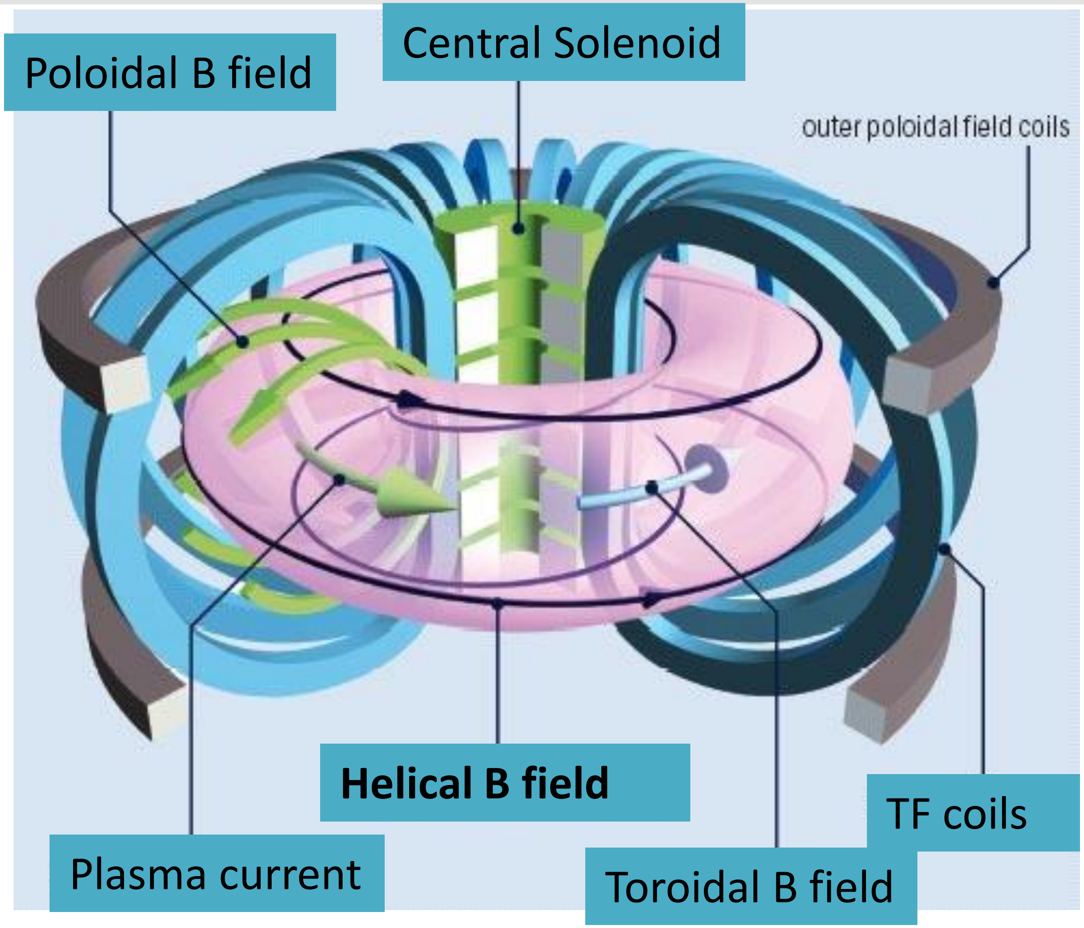
Poloidal Field coils

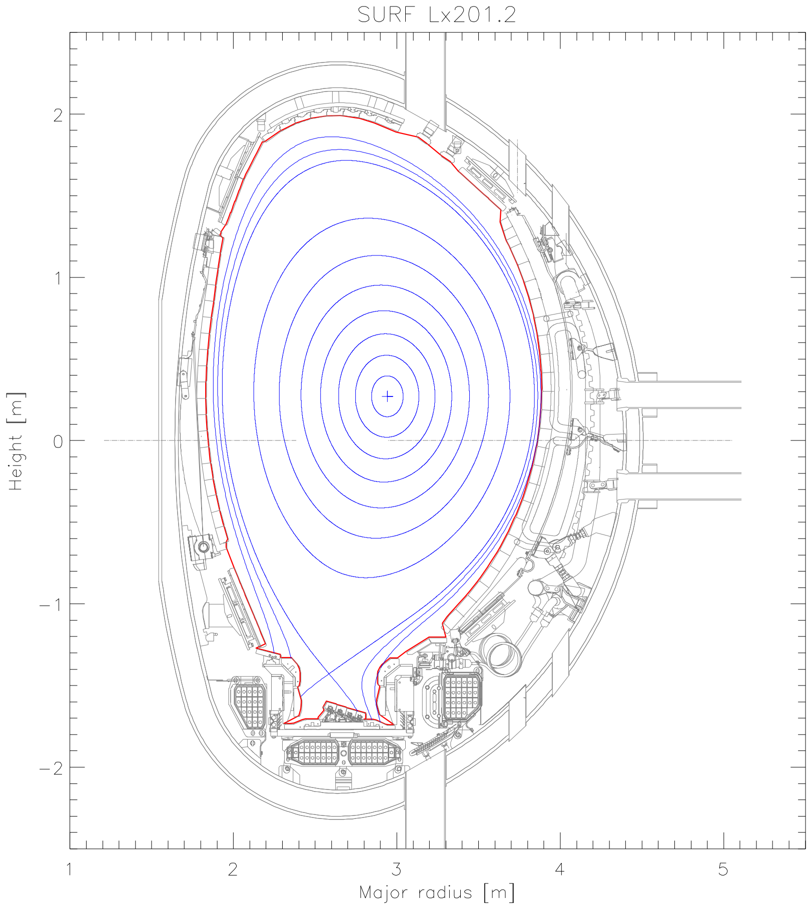




Plasma positioning and shaping
by electric current in PF coils

Tokamak, Magnetic Confinement Fusion Reactor



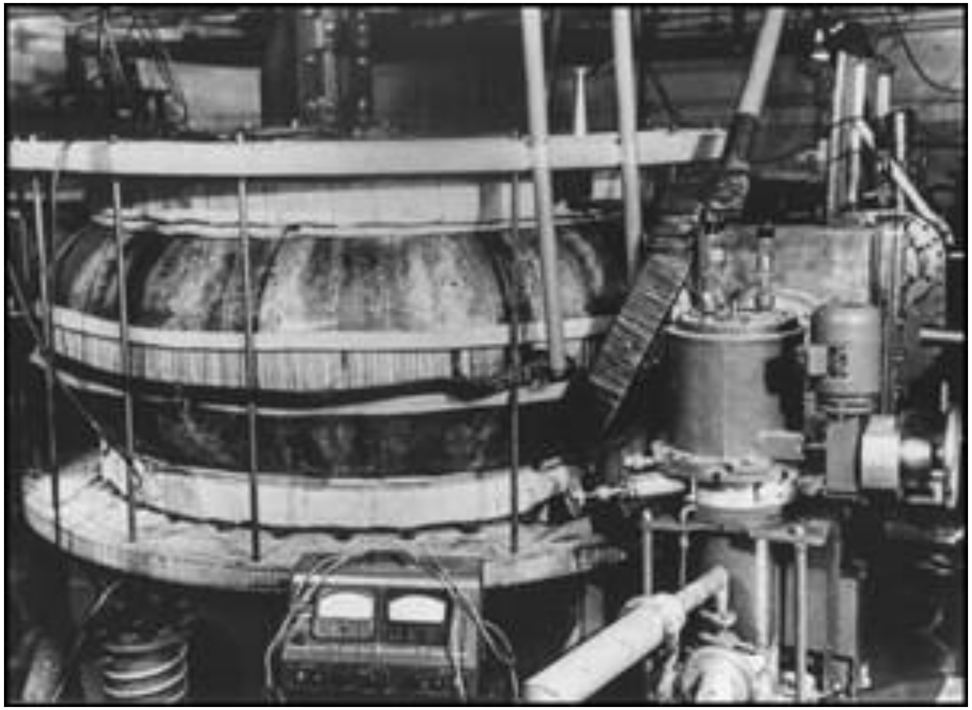


Toroidal magnetic field (by electric current in TF coils)

+ Poloidal magnetic field (by plasma current)

= **Helical magnetic field → Closed magnetic flux surfaces → High confinement**

World’s first Tokamak



https://www.iter.org/sci/BeyondITER

Andrei Sakharov

Igor Tamm

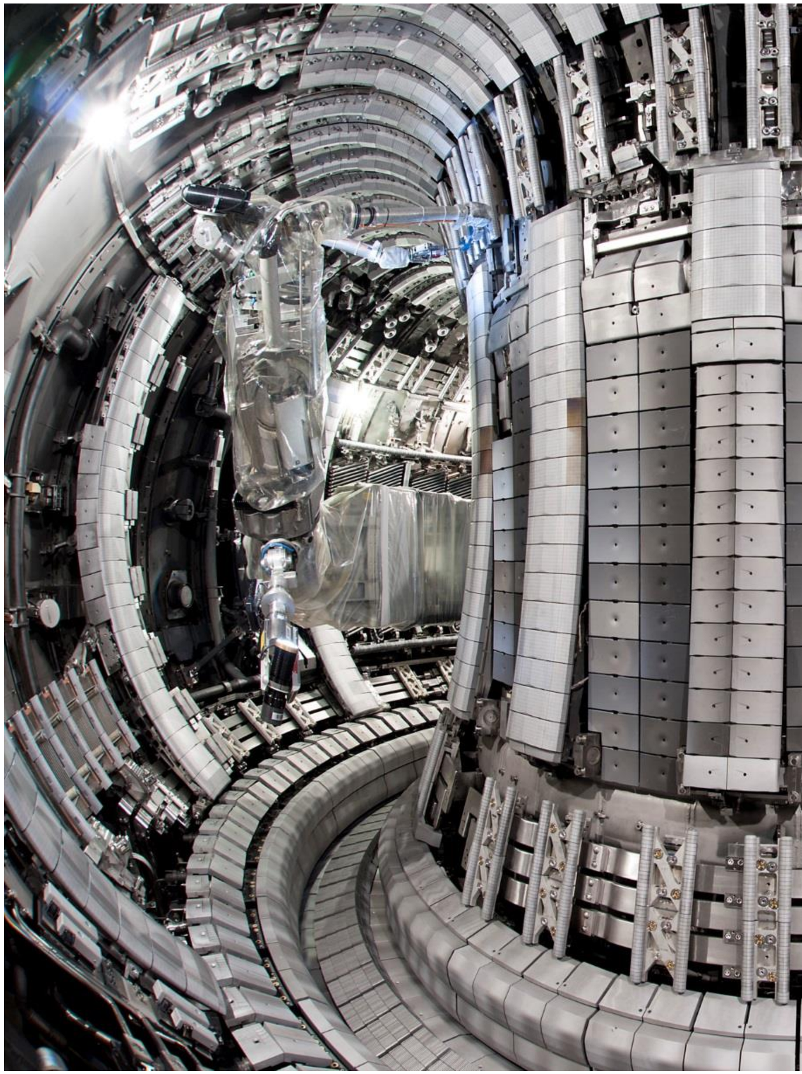
Lev Artsimovich

- T1 tokamak : world’s first tokamak device, operated in 1952 in Kurchatov Institute, Moscow, Russia
- Russian pioneers invented to suppress the kink instability
- 0.4 m^3 Plasmas were produced in its copper vacuum vessel.

- The Russian pioneers named it in Russian, **Tokamak**
- **To**roidalnaja **ka**mera s **ma**gnitnymi **k**atushkami, Toroidal chamber with magnetic coils in English

Tokamak in present days

JET (Joint European Torus): R_0 = 3 m, a = 1 m, operated since 1983, refurbished in 2011



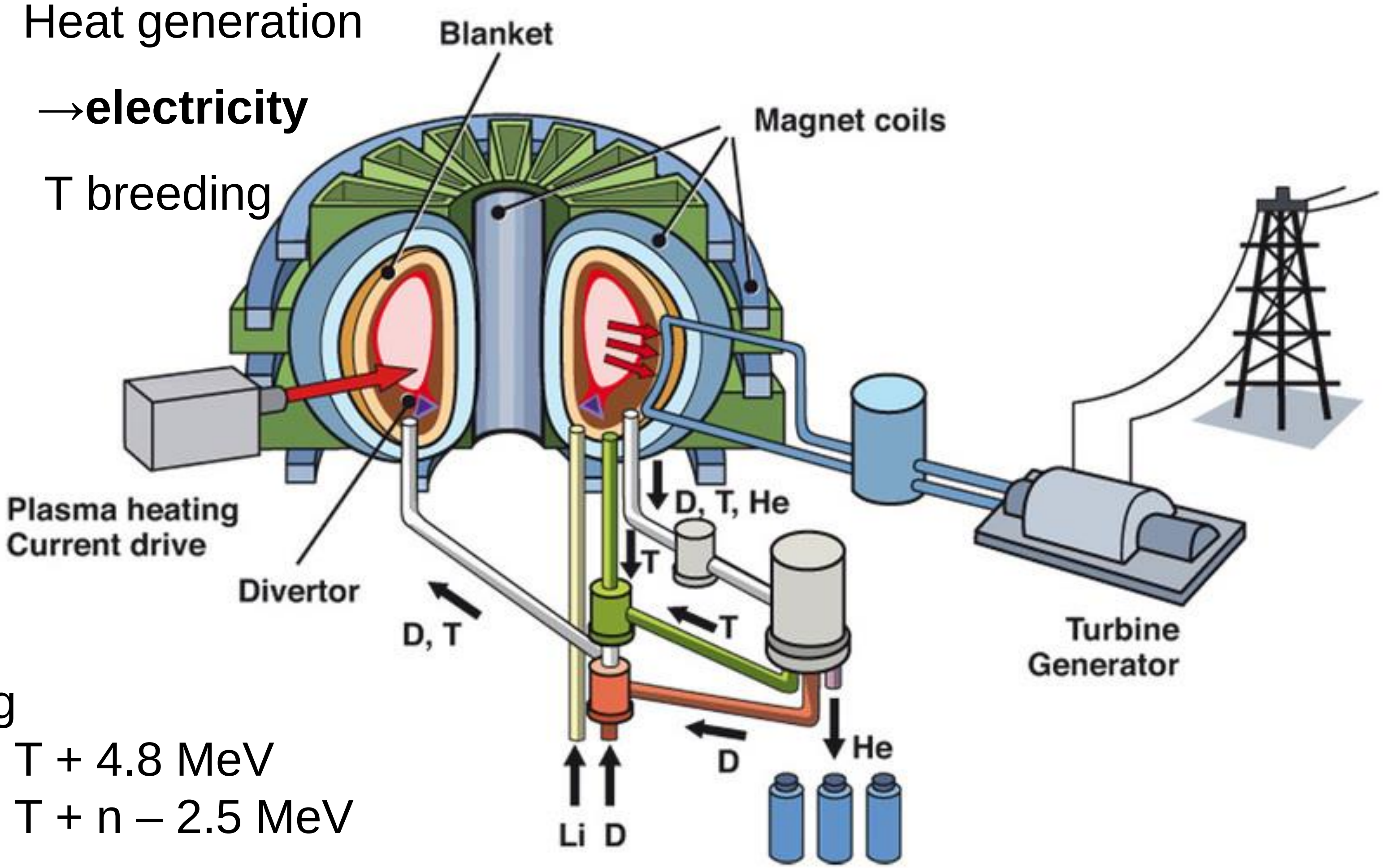

JET

Fusion power plant

D + T → n (14.1MeV) + α (3.5 MeV)

- ➢ Li Blanket
  1. Heat generation

     →**electricity**
  2. T breeding
- ➢ Plasma heating (self-sustaining)

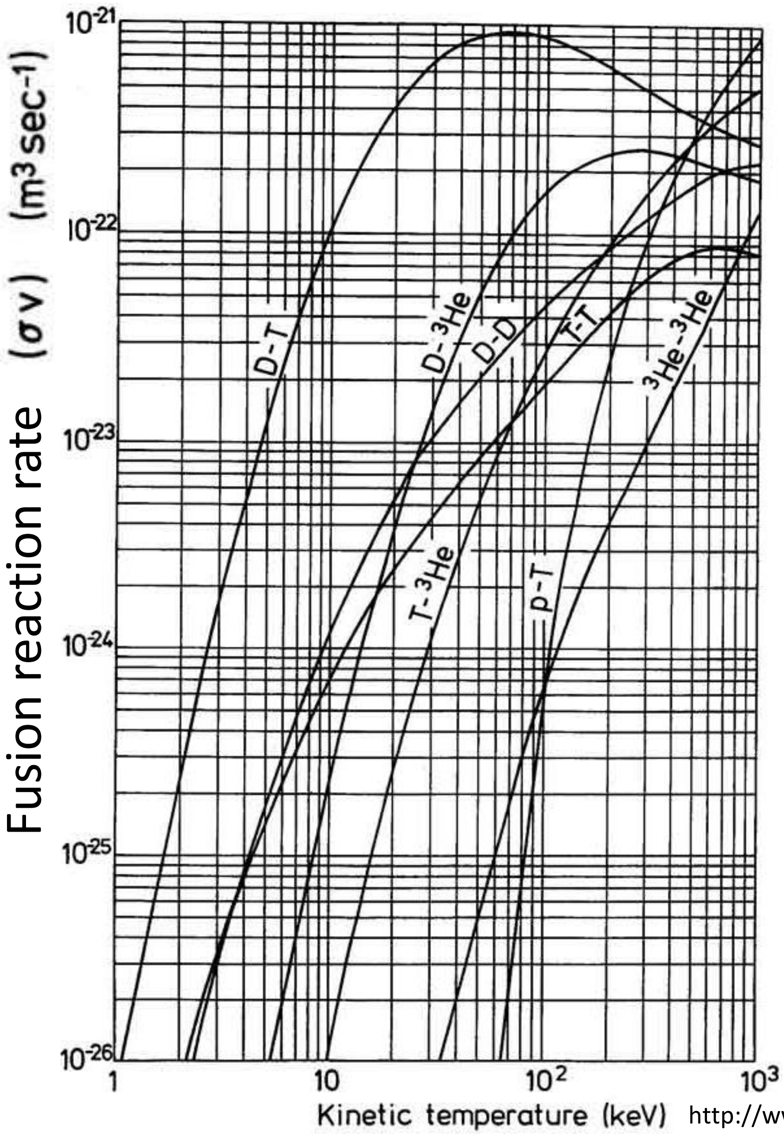


❖ T breeding

$n + Li^6 \rightarrow \alpha + T + 4.8$ MeV

$n + Li^7 \rightarrow \alpha + T + n - 2.5$ MeV

Fusion power



$$P_{\mathrm{fus}} = V_p \times n_D n_T < \sigma\upsilon > \times 17.6 MeV$$

for $n_i = 2n_D = 2n_T$

and $< \sigma\upsilon > \propto T_i^2$ (for T_i = 10-20keV)

$$P_{\mathrm{fus}} \propto n_i^2 T_i^2 = p^2$$

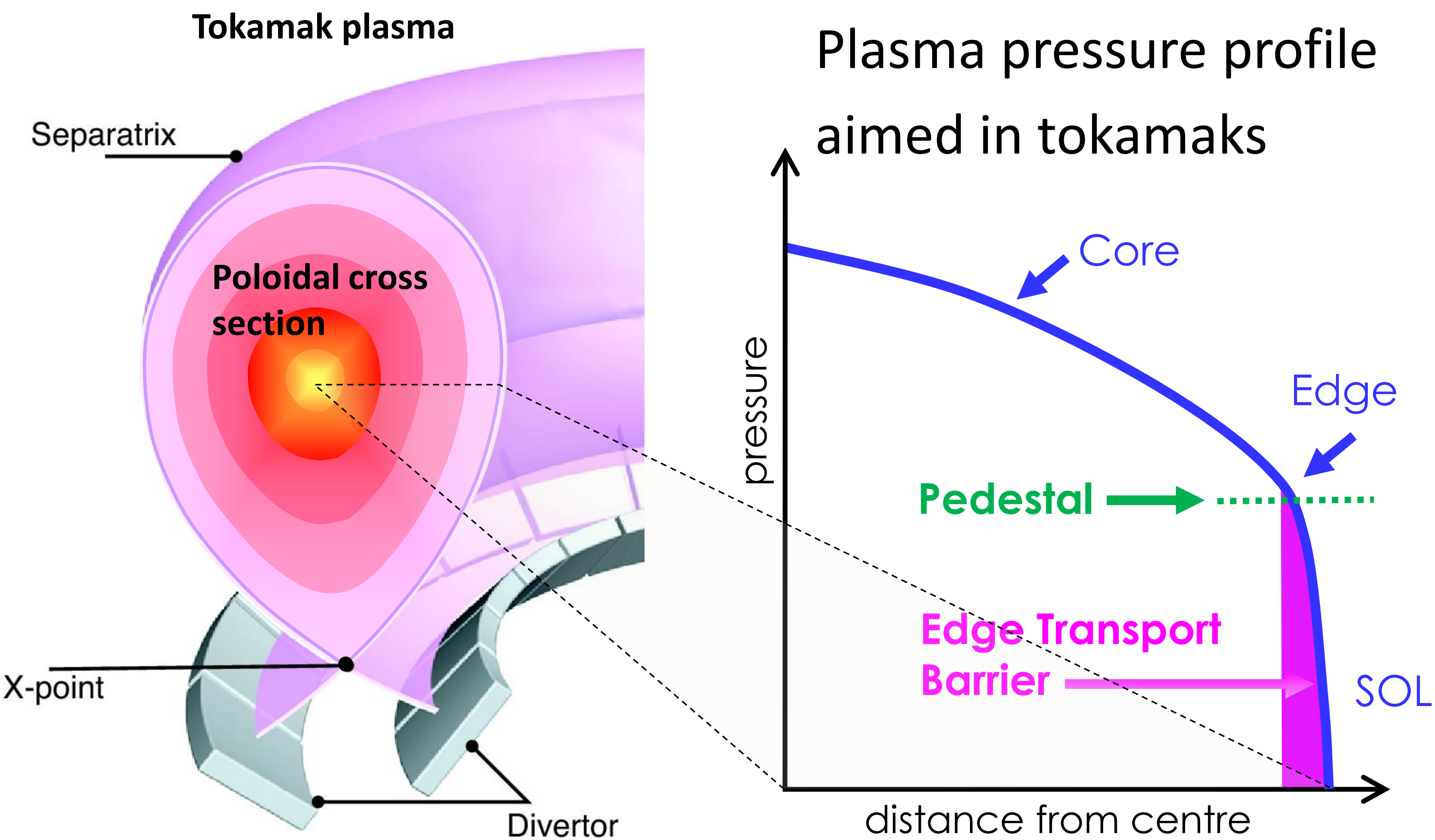
For high fusion power

and self - sustainment,

high plasma pressure is essential.

http://www.scienceall.com/jspJavaPopUp.do?classid=CS000140&articleid=611&bbsid=146&popissue=java

High and stable plasma pressure is necessary for high fusion power



Courtesy of C. Maggi (CCFE)

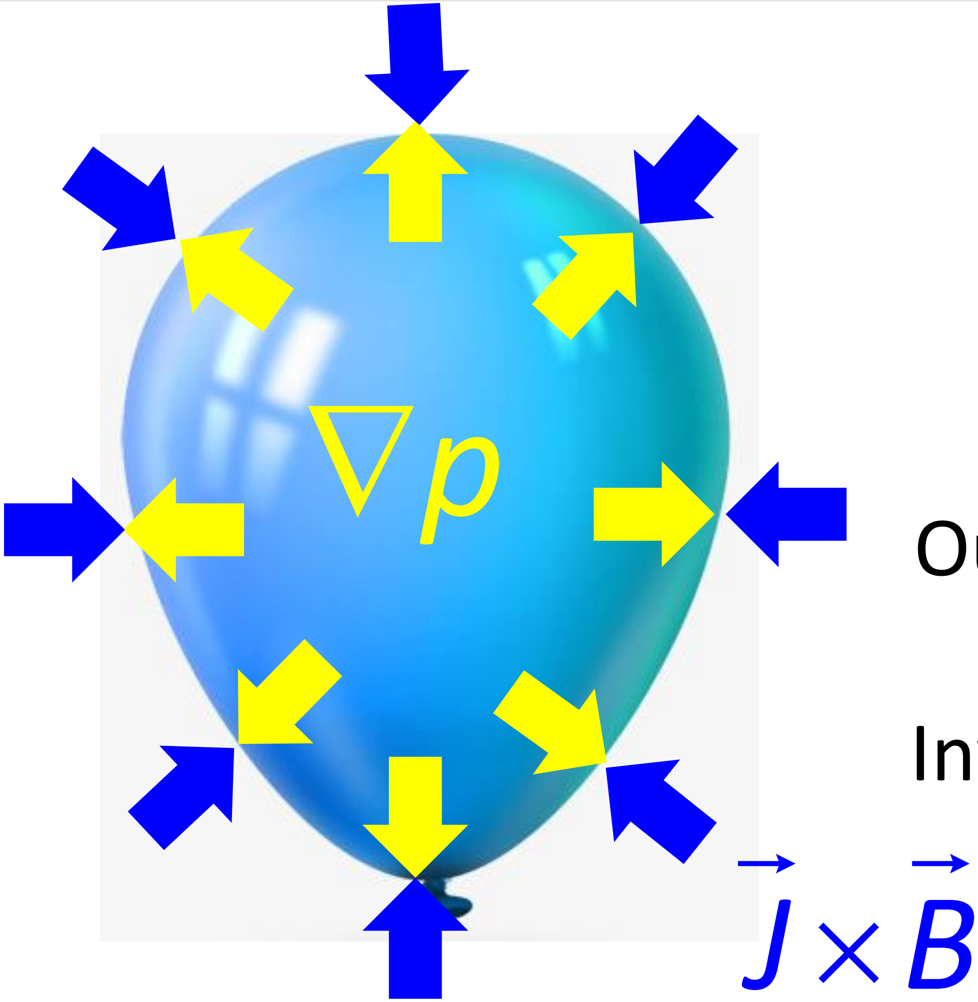
Contents

Plasma physics is essential for high and stable plasma pressure in tokamaks

Transport

Stability

Plasma equilibrium

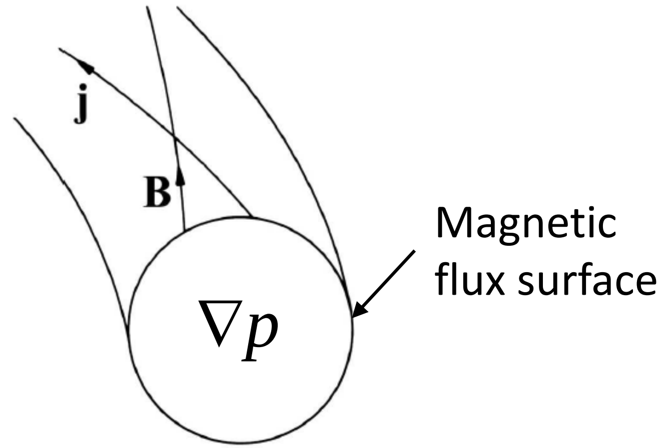


$$\nabla p = \vec{J} \times \vec{B}$$

Force balance
between
Outward plasma pressure
and
Inward magnetic pressure

Plasma equilibrium

$$\nabla p = \vec{J} \times \vec{B} \;\Rightarrow \begin{cases} \vec{B}\cdot\nabla p = 0, \text{ p is constant along } \vec{B} \\ \vec{J}\cdot\nabla p = 0, \text{ p is constant along } \vec{J} \end{cases}$$

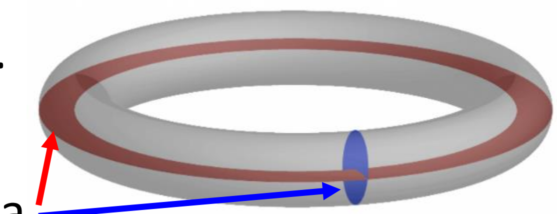


Magnetic flux surfaces are imaginary surfaces on which p is uniform, and $\vec{B}$ and $\vec{J}$ lie (i.e. don't accross the surfaces).

Enclosed magnetic flux is the same everywhere (hence name).

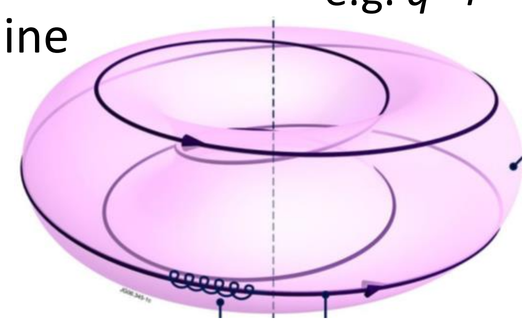
The surfaces are labelled by the enclosed magnetic flux

i.e. toroidal flux ϕ or poloidal flux ψ, crossing the colored area.



Safety factor $q(\psi) \equiv \dfrac{d\phi}{d\psi} \approx \dfrac{rB_\phi}{R_0 B_\psi} = \dfrac{\text{\# of toroidal turn of the field line}}{\text{\# of poloidal turn of the field line}}$

e.g. q=4



i.e. the higher the q, the less twisted the magnetic field line.

q is important for MHD stability (hence name).

In general, more stable at high q.

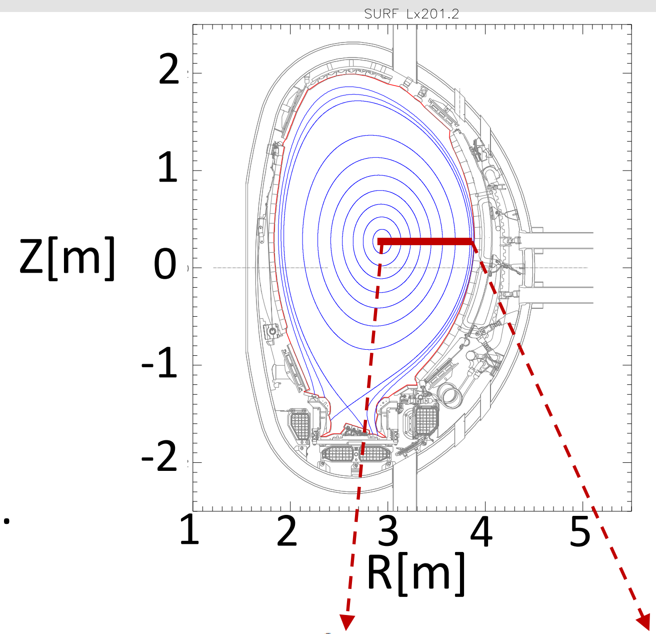
Grad-Shafranov eqn and flux coordinates

$\vec{J} \times \vec{B} = \nabla p$ in cylinderical coordinate

Grad-Shafranov eqn $\Rightarrow \Delta^* \psi \equiv -\mu_0 R^2 \dfrac{dp}{d\psi} - \mu_0{}^2 F \dfrac{dF}{d\psi}$

where $\Delta^* \equiv R \dfrac{\partial}{\partial R}\left(\dfrac{1}{R}\dfrac{\partial}{\partial R}\right) + \dfrac{\partial^2}{\partial Z^2}$ and $\mathrm{F} \equiv \dfrac{RB_\phi}{\mu_0}$

$\psi \equiv \dfrac{1}{2\pi}\int_0^{r(\psi)} B_p dr$ i.e. enclosed poloidal flux in the flux surface.

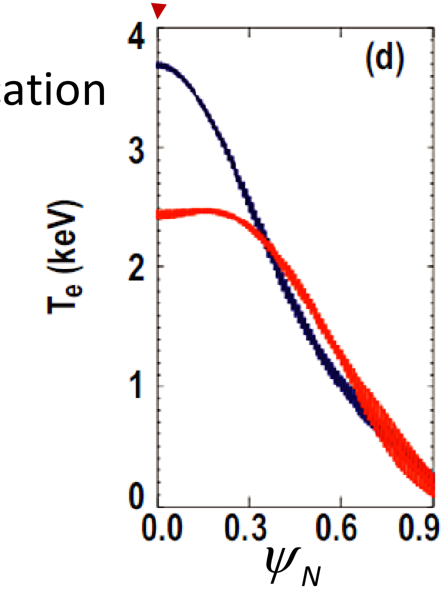


Grad-Shafranov eqn is solved numerically to find the geometrical location of the flux surfaces i.e. $\psi = \psi(R,Z)$.

Assuming plasma parameters are identical on the same flux surface,

normalized ψ_N $(\equiv \dfrac{\psi(r)}{\psi(a)} = 0 \sim 1)$ is used as x-coordinate

of plasma parameter profiles such as temperature e.g. $\mathrm{T_e}(\psi_N)$.



Normalized plasma pressure, plasma beta

$$\nabla p = \vec{J} \times \vec{B} = (\frac{\nabla \times \vec{B}}{\mu_0}) \times \vec{B} = \frac{(\vec{B} \cdot \nabla)\vec{B} - \nabla(B^2 / 2)}{\mu_0}$$ (Ampere's law and vector identity)

$$\nabla(p + \frac{B_2}{2\mu_0}) = \frac{(\vec{B} \cdot \nabla)\vec{B}}{\mu_0} \approx 0$$ (assuming straight field lines)

$p + \dfrac{B_2}{2\mu_0} \approx \text{constant} \Rightarrow$ Sum of kinetic and magnetic energy density is constant.

Plasma beta $\beta \equiv \dfrac{p}{B^2/2\mu_0} = \dfrac{\text{plasma pressure}}{\text{magnetic pressure}} \approx$ 1% for most tokamaks.

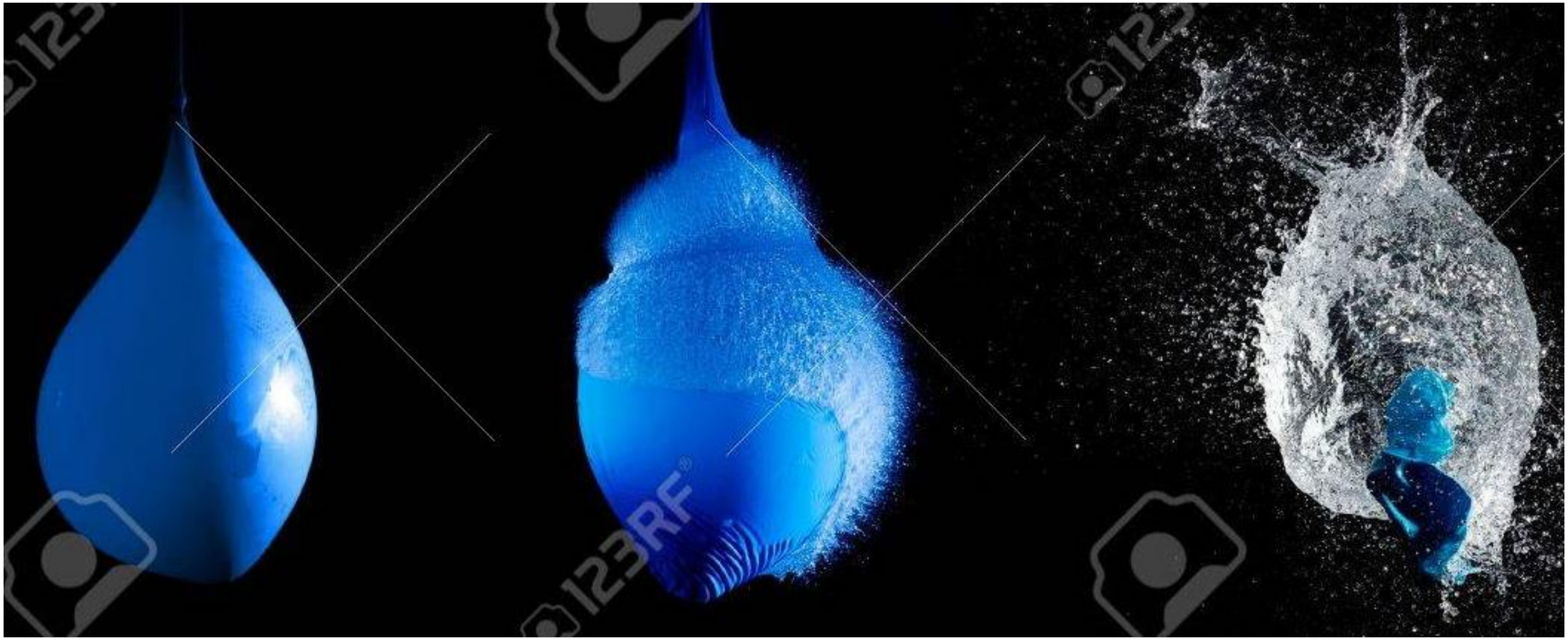
$$\beta_N \equiv \frac{\beta_T}{I_p / aB_\phi}[\%\text{ m Tesla / MA}] \propto \frac{\text{plasma pressure}}{\text{magnetic tension}}$$

Previous numerical calculation predicts the ballooning mode would happen if $\beta_N > 2.8$ (a.k.a Troyon beta limit). Since then, β_N has been conventionally used as a measure of storable plasma pressure, in terms of MHD stability.

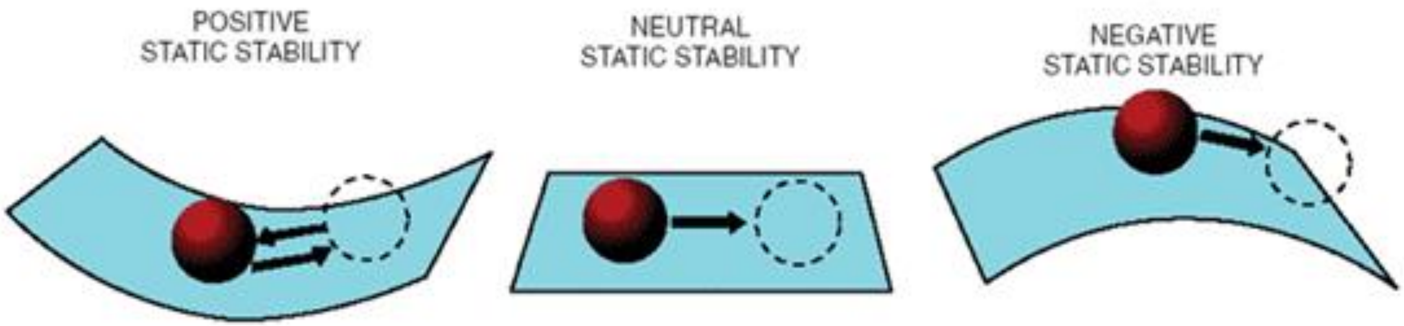
(β_N>2.8 is possible by shaping the plasmas in present devices.)

Plasma stability

Required to keep the equilibrium!



Stability



Linear stability analysis by examining

the time dependence of the small-amplitude perturbation

$Q(\vec{r},t) = Q_0(\vec{r}) + Q_1(\vec{r},t)$ where $\left|Q_0(\vec{r})\right| \gg \left|Q_1(\vec{r},t)\right|$

$$Q_1(\vec{r},t) = Q_1(\vec{r})\exp(-i\omega t) = Q_1(\vec{r})\exp(-i(\omega_r + i\omega_i)t) = Q_1(\vec{r})\exp(-i\omega_r t)\exp(\omega_i t)$$

$\omega_i < 0$: stable as the pertabation decays in time

$\omega_i = 0$: marginally stable

$\omega_i > 0$: unstable as the pertabation exponential grows in time

Linear MHD stability analysis

Whether ω is positive or negative, and how ω can be zero

are examed by solving the linearized momentum conservation eqn,

$$\vec{F}(\vec{\xi}) = \rho \frac{\partial \vec{\xi}^{2}}{\partial^{2} t} = -\omega^{2} \rho \vec{\xi}$$

where

perturbed displacement $\vec{\xi}(\vec{r},t) = \underline{\xi} \exp(\vec{k} \cdot \vec{r} - i\omega t)$

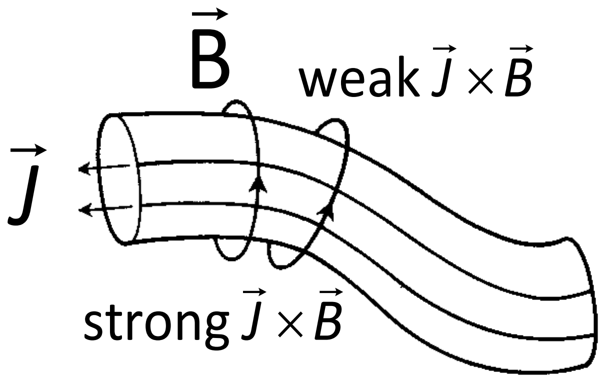
and

force operator in MHD eqns,

$$\vec{F}(\vec{\xi}) = \frac{1}{\mu_0} (\nabla \times \vec{B}) \times [\nabla \times (\vec{\xi} \times \vec{B})] + \frac{1}{\mu_0} \{\nabla \times [\nabla \times (\vec{\xi} \times \vec{B})]\} \times \vec{B} + \nabla (\vec{\xi} \cdot \nabla p + \gamma p \nabla \cdot \vec{\xi})$$

Two main sources of MHD instabilities

1. Plasma current
(kink mode or sausage)



2. Plasma pressure
(interchange instability)

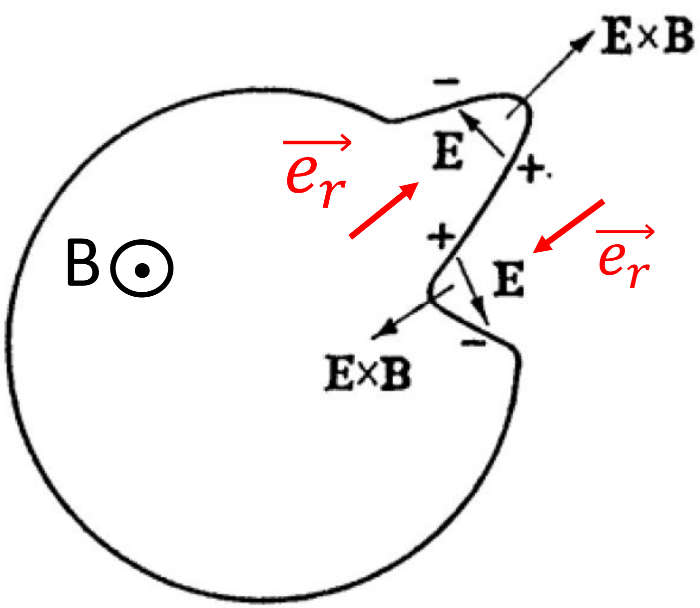
Perturbation in pressure

$\rightarrow$ curvature drift $\overrightarrow{V_{D,R}} = \dfrac{mv_{\parallel}^2}{R}\vec{e_r} \times \dfrac{\vec{B}}{qB^2}$

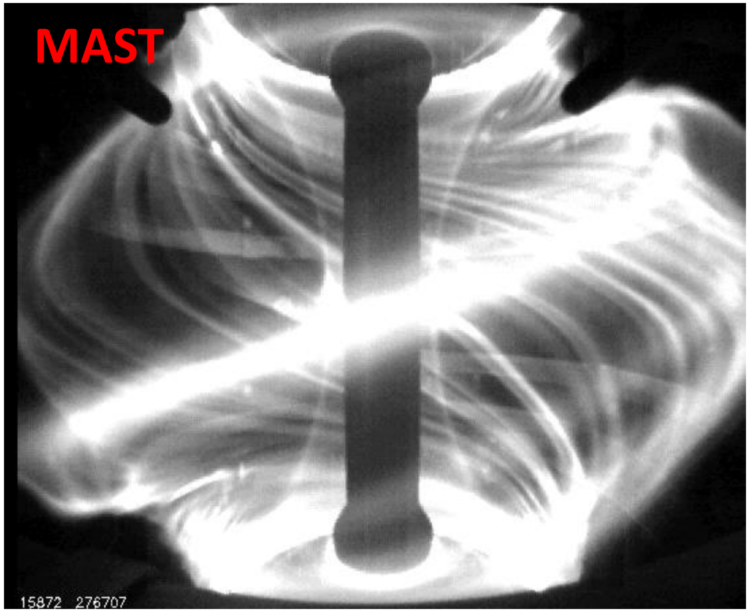
$\rightarrow$ charge seperation

$\rightarrow$ ExB drift $\overrightarrow{V_{D,R}} = q\vec{E} \times \dfrac{\vec{B}}{qB^2}$

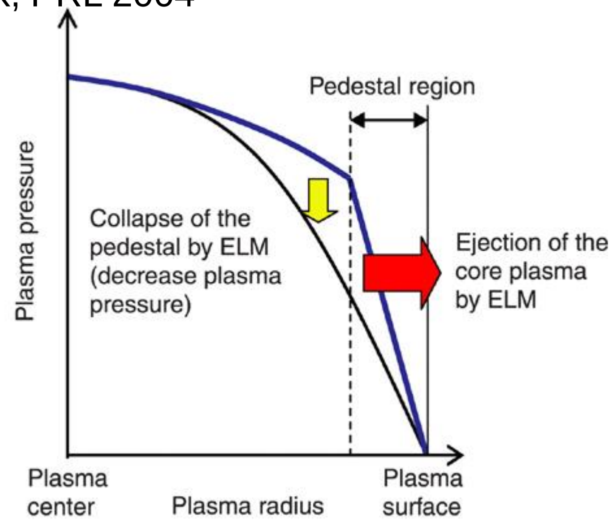
$\rightarrow$ enhance the perturbation



Edge Localised Modes (ELMs)



A. Kirk, PRL 2004

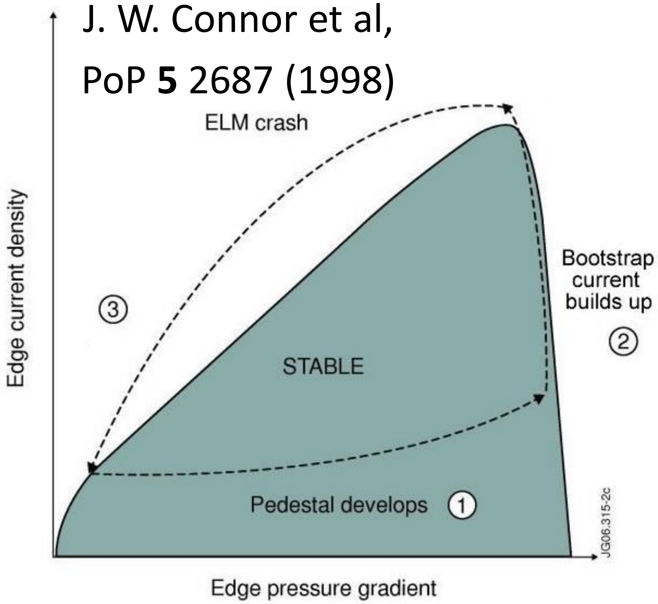


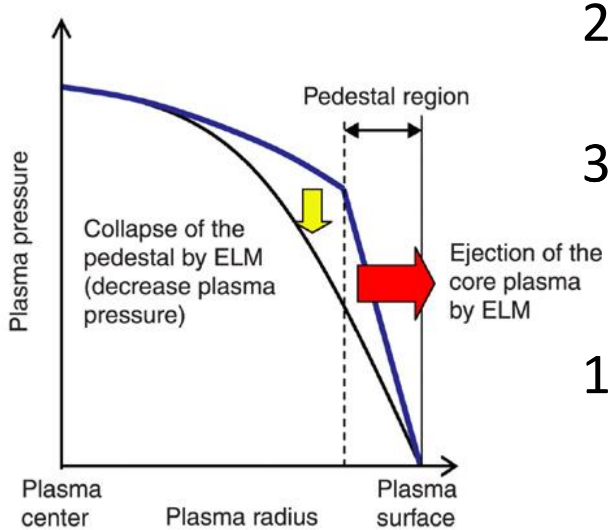
- Periodic MHD instabilities at the plasma edge
- Combined instability of peeling mode (current driven) and ballooning mode (pressure driven)
- Temporary reduction in edge ∇p
- Deposit several % of plasma energy on divertor in short bursts, triggering sputtering i.e. impurity source
- Expel impurities

Peeling-Ballooning model for ELM cycle

J. W. Connor et al,

PoP **5** 2687 (1998)





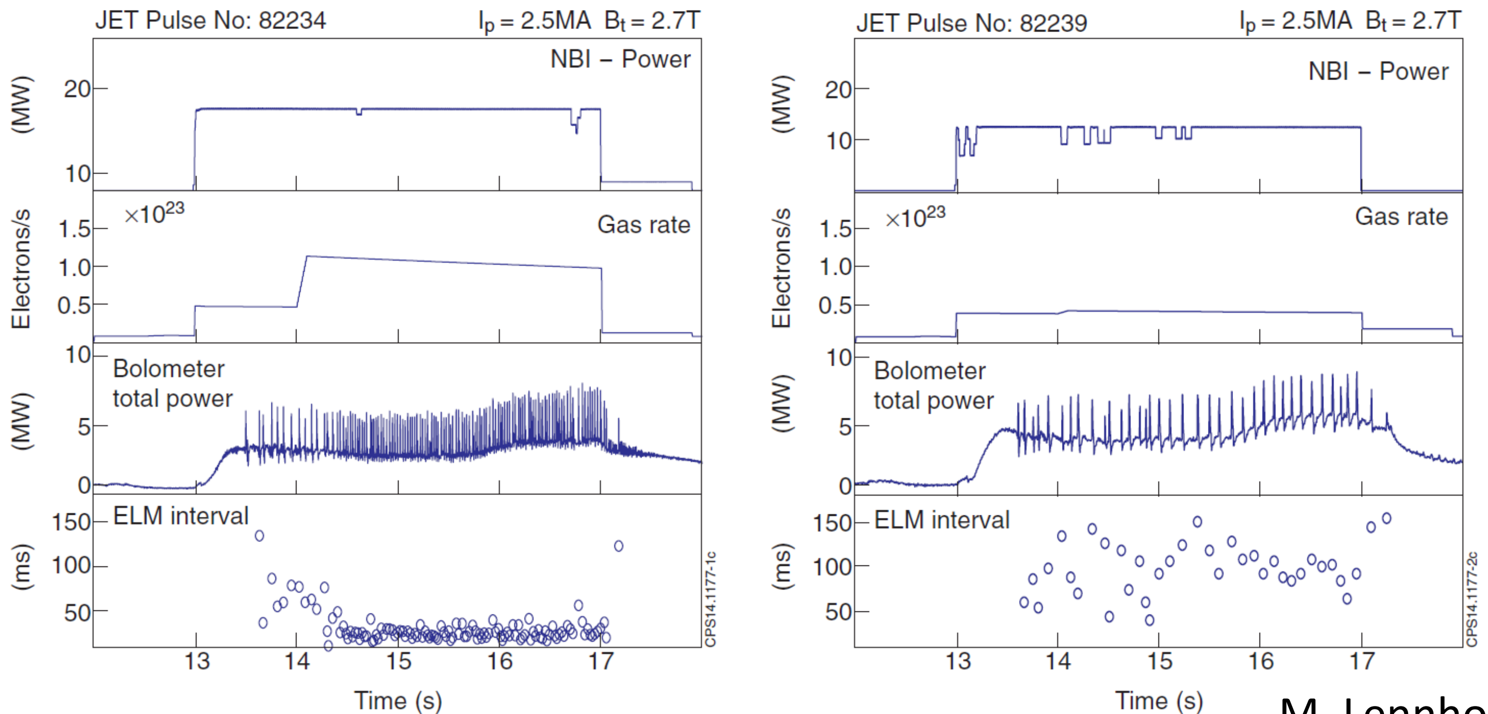
Type-I ELM cycles

Following the previous crash of the edge pressure pedestal,

1) The edge pressure pedestal develops quickly, due to the edge transport barrier The growth of the pedestal stops at “ballooning stability” limit.
2) The bootstrap current starts to grow, due to the high pressure gradient at the edge.
3) Eventually, the bootstrap current triggers ideal peeling mode, which leads to an pedestal crash

1),2), and 3) are repeated.

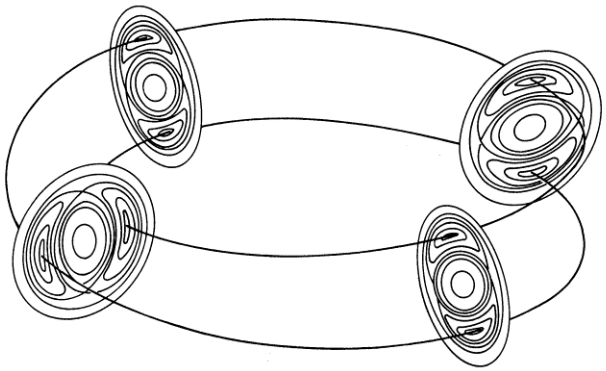
Types of ELMs

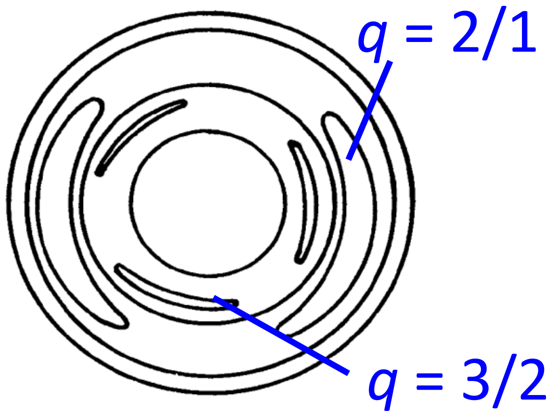


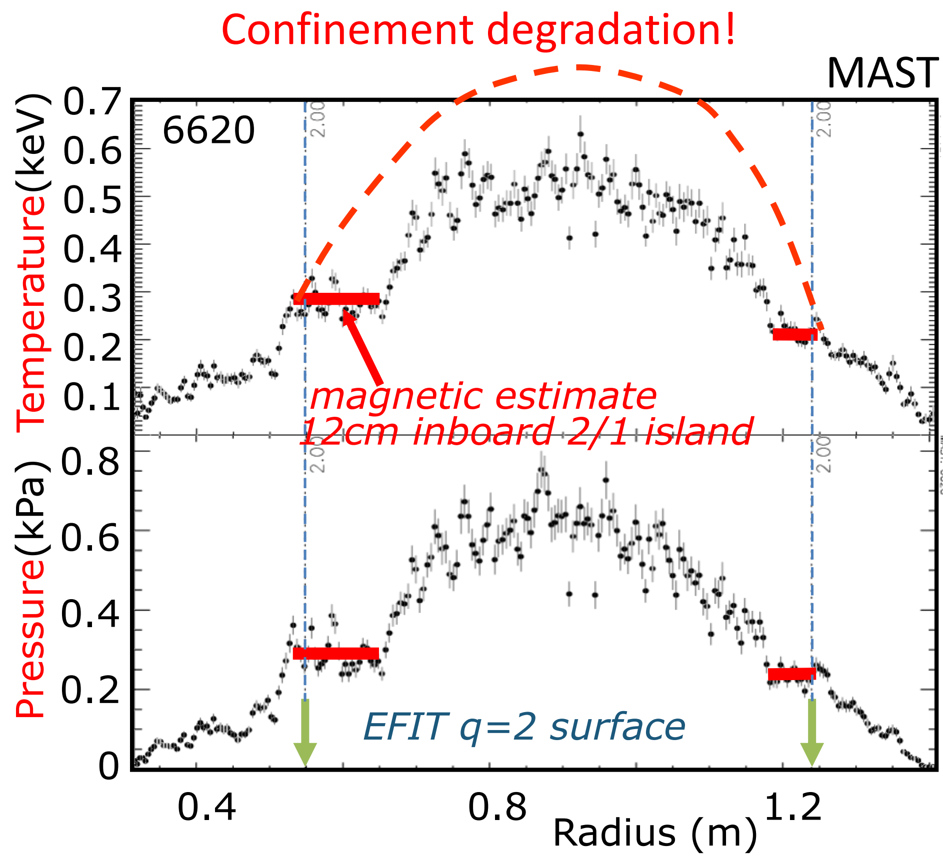
M. Lennholm et al NF 2015

- Peaked radiation power indicates ELM events.
- Gas injection can increase ELM frequency.
- ELM-free : Very high pedestal pressure, but vulnerable to impurity accumulation.
- Type III ELMs : small and continuous bursts. Easy to expel impurities from the plasma, but low pedestal pressure.
- Type I ELMs : large and periodic bursts. High pedestal pressure but risk of damaging the divertor.

Neoclassical Tearing Modes (NTMs)

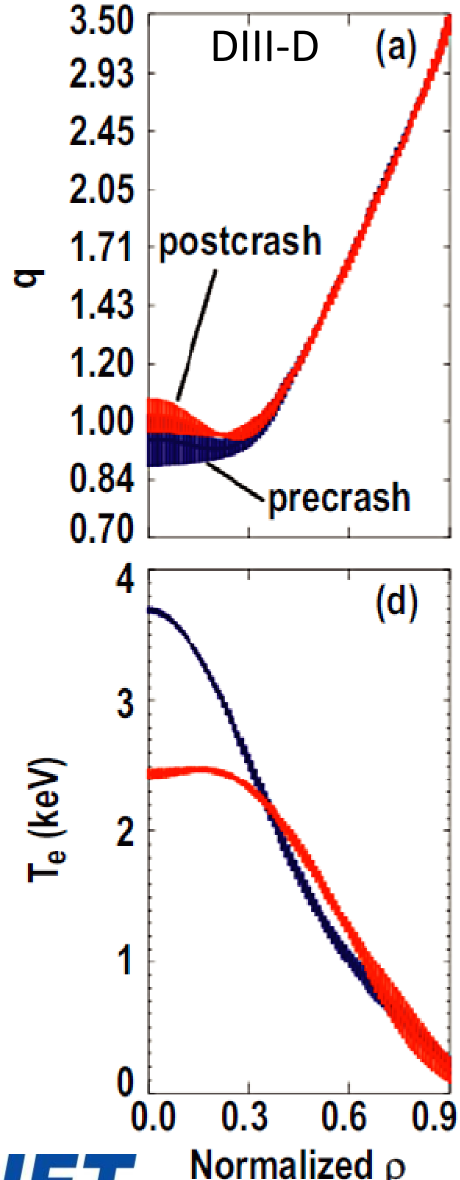


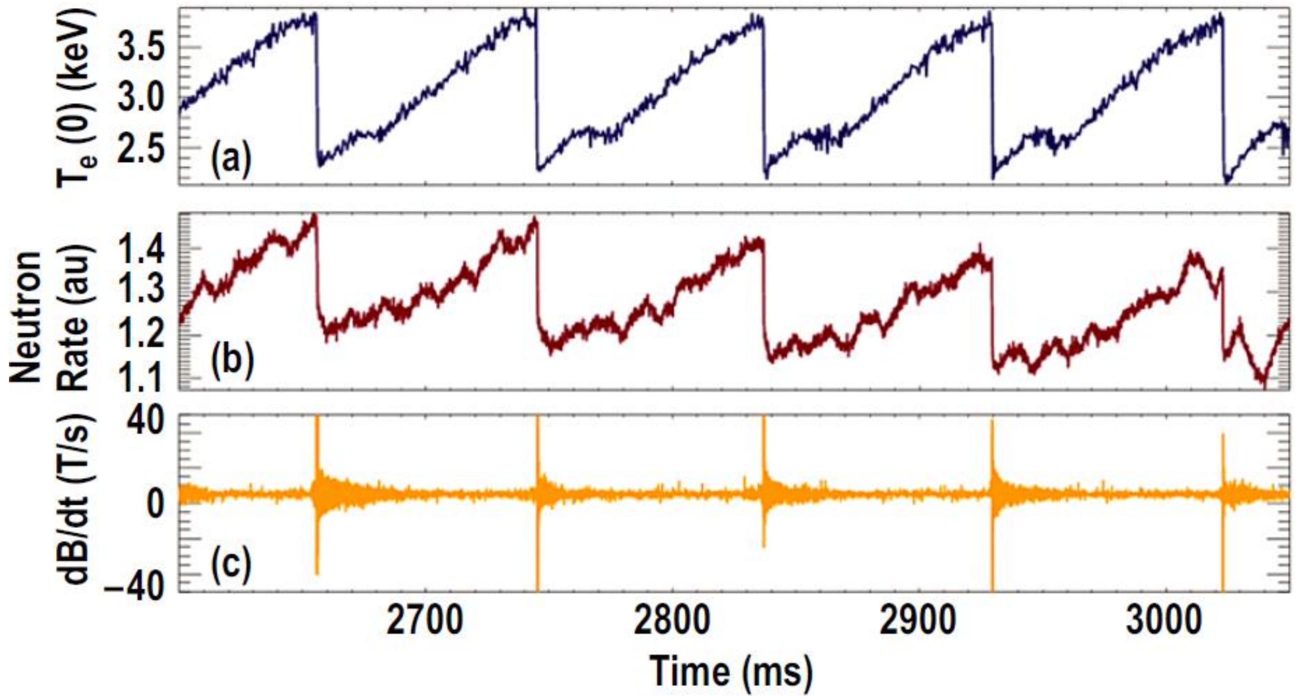




- Magnetic field lines can be reconnected if q is a rational number (e.g. q=2/1 or 3/2), and form magnetic islands at flux surfaces.
- Pressure flattening across magnetic islands due to large transport coefficients along magnetic field lines i.e. confinement degradation.

Sawtooth instability





- Magnetic reconnection occurs if q=1 (Kadomtsev's theory)
- Flattening pressure and fast ion density within q=1 flux surface, leading to degradation in fusion performance.
- q_0 and T_e rise after the collapse, and repeat the rise and collapse (Hence the name Sawtooth instability).

C M Muscatello, PPCF 2012

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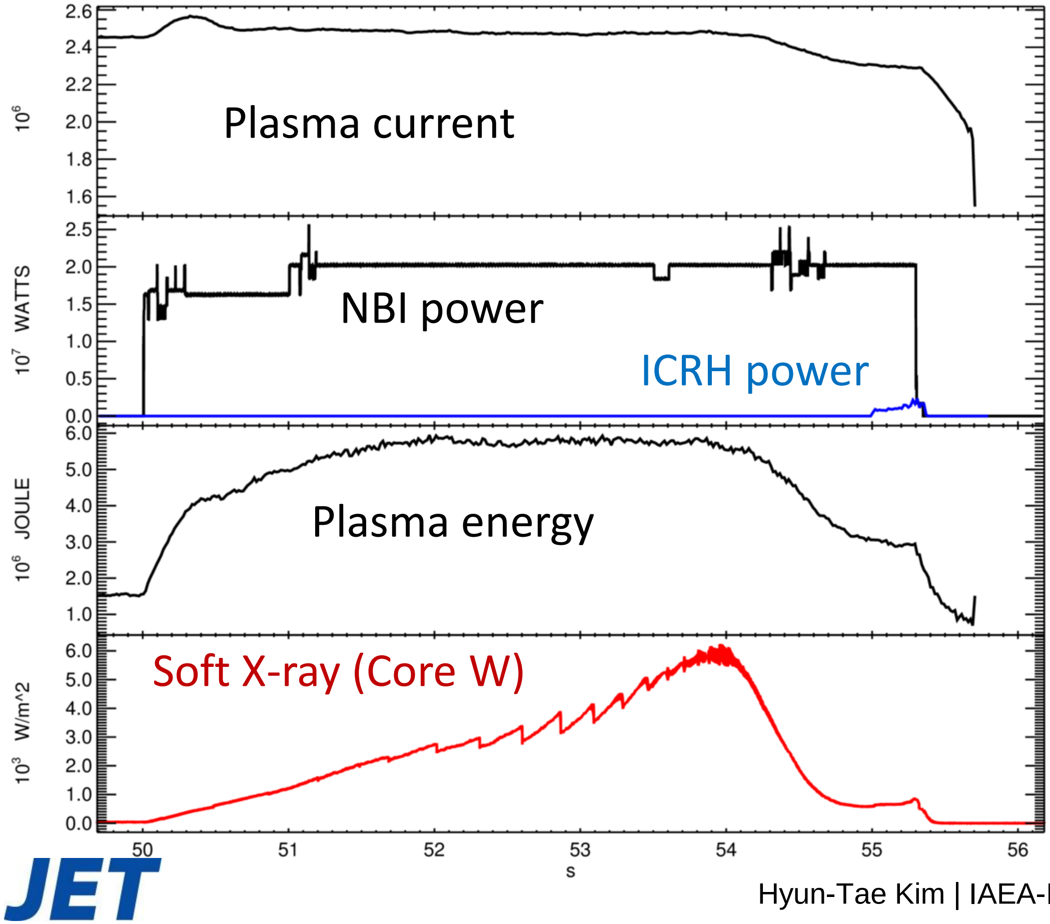
Disruption

Rapidly growing MHD instabilities can result in the plasma disruption, which can seriously damage the surrounding wall by

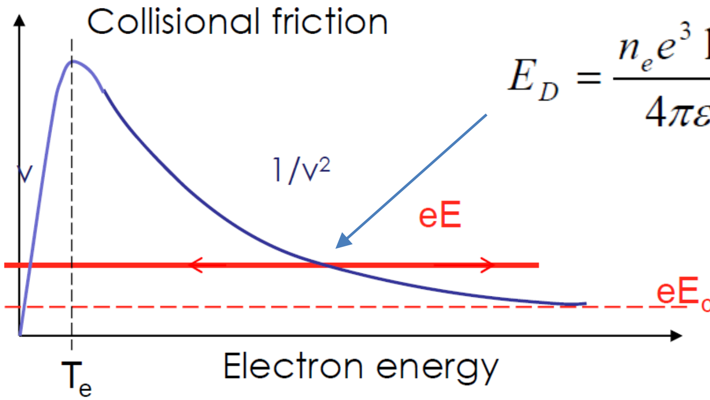
- Melting the first wall due to thermal energy load and
- Electromagnetic force due to the induced halo current on the first wall .



The sudden loss of I_p can also induce E field higher than Dreicer E field, and produce runaway electrons, which can also damage the wall.

Dreicer electric field:

$$E_D = \frac{n_e e^3 \ln(\Lambda)}{4\pi\varepsilon_0^2 T_e}$$



H. Dreicer, PRL 1959

Low transport (i.e. high confinement) required for high pressure!

Classical transport

Transport of charged particles with magnetic field

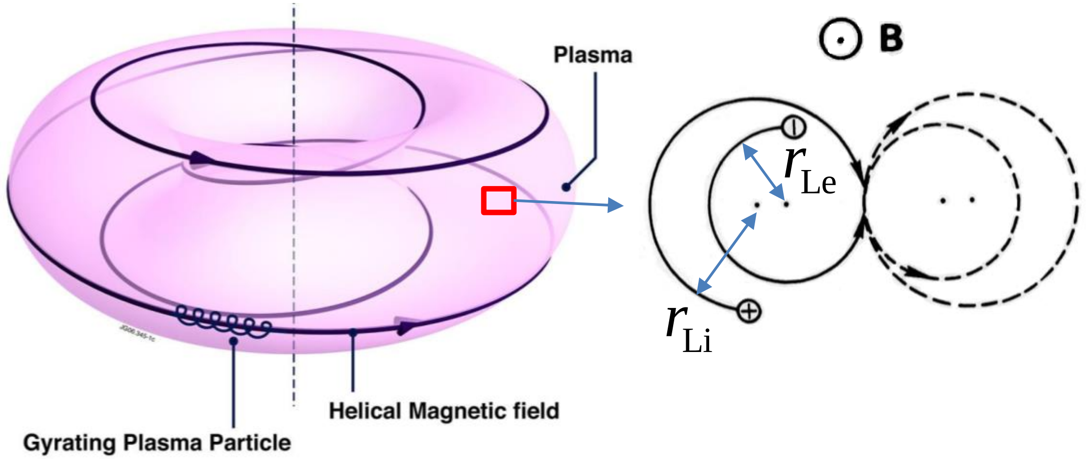
$$\vec{\Gamma}_{\perp} = -D\nabla n$$

$$D_{\perp}^{CL} \approx (r_{\mathrm{Le}})^2 \nu_{ei}$$

$$\vec{q} = -n\chi_{\perp}\nabla T$$

$$\chi_{\perp,i}^{CL} \approx (r_{\mathrm{Li}})^2 \nu_{ii}$$

$$\chi_{\perp,e}^{CL} \approx (r_{\mathrm{Le}})^2 \nu_{ee}$$



- Typical values calculated by the classical transport theory

D ~5x10^{-5} m^2/s, χ_e ~ 5x10^{-5} m^2/s, χ_i ~ 10^{-3} m^2/s

- In experiments, however, D, χ_e , and χ_i ~1 m^2/s

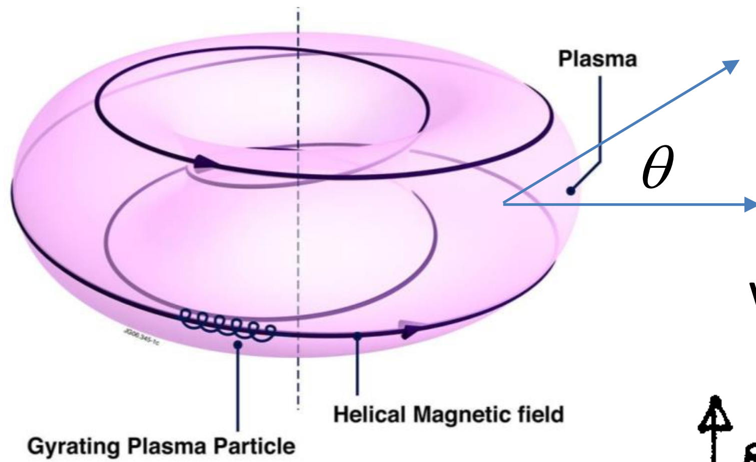
i.e. Transport in experiments >> Classical transport

Neoclassical transport

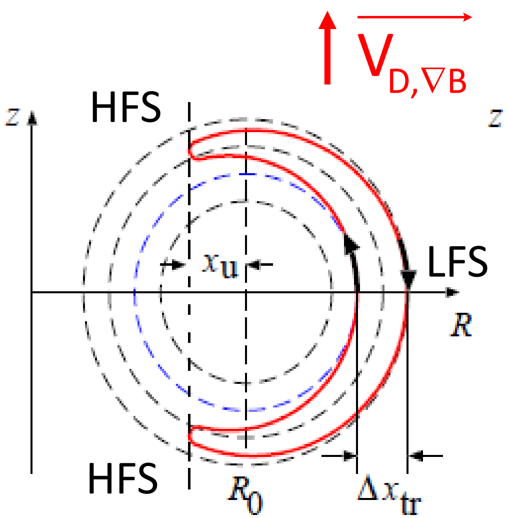
Classical transport + Toroidal geometry

$$\left|B(r,\theta)\right| = \frac{B_0}{1+(a/R_0)\cos\theta}$$

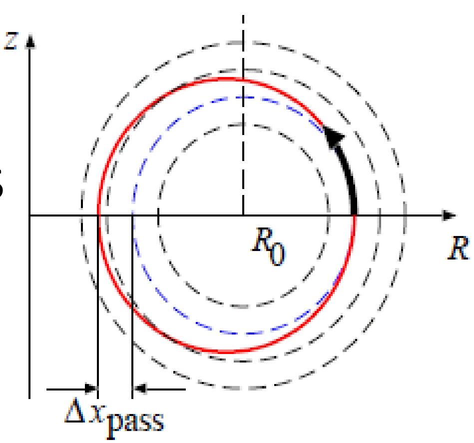
$$\overrightarrow{V_{D,\nabla B}} = -\mu\nabla B \times \frac{\vec{B}}{qB^2}$$



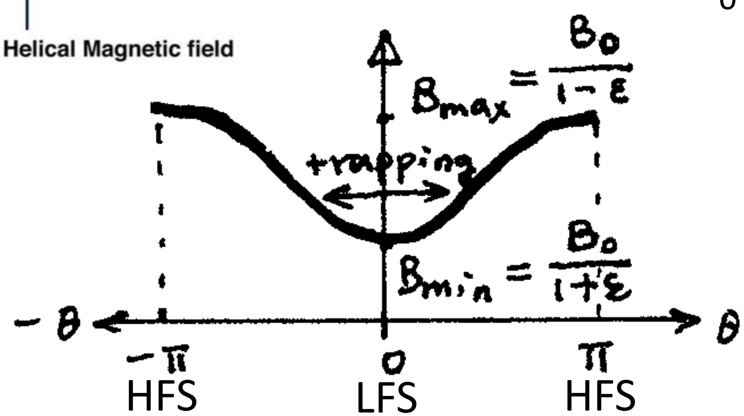
where $\varepsilon \equiv \dfrac{a}{R_0}$



Trapped particles



Passing particles

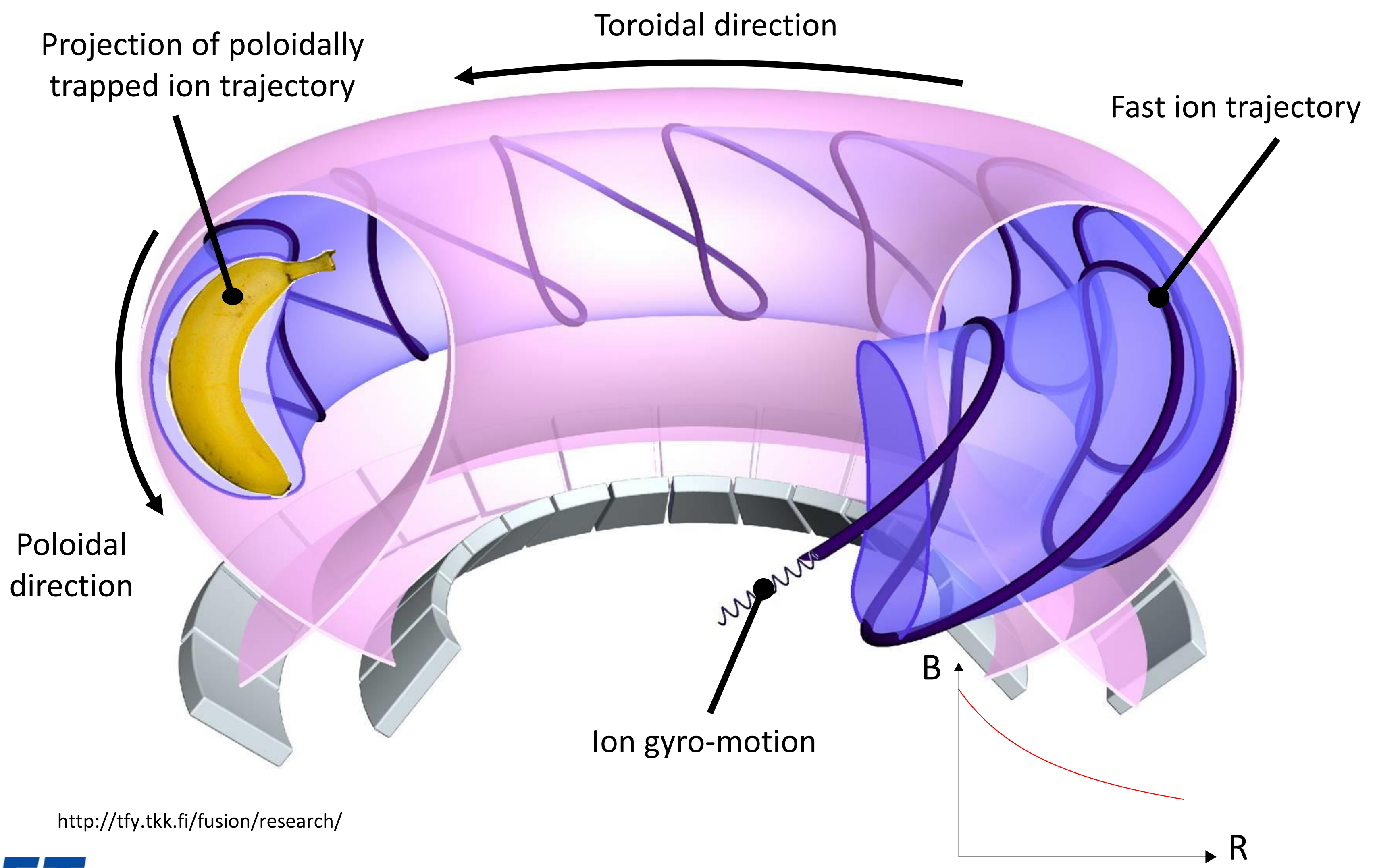


Particles approach HFS along field line.

If $v_{\parallel}/v_{\perp}$ is not high enough, they are bouncing at HFS due to $\overrightarrow{F_{\nabla B}} = -\mu\nabla B$

i.e. trapped particles

Neoclassical transport



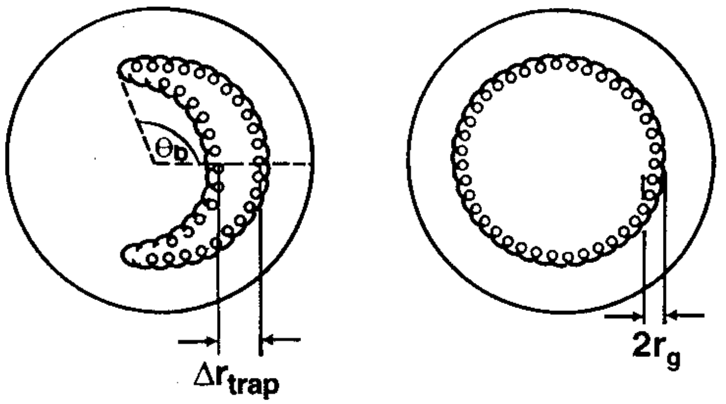
http://tfy.tkk.fi/fusion/research/

JET

Neoclassical transport

Collisional excursion across flux surfaces

- Passing particles: $\Delta x = 2r_g = 2r_{Li}$
- **Trapped particles: $\Delta x = \Delta r_{trap}$ >> $2r_g$** where enhanced radial diffusion across the confining magnetic field



Trapped particles   Passing particles

For example, if $q = 2$ and $R_0 / r = 3$

$$D_\perp^{NC} \approx f(\Delta l)^2 \nu_{eff} \approx (\frac{2r}{R_0})^{1/2}(q^2 \frac{R_0}{r} r_{Li}{}^2)(\frac{R_0}{r}\nu_{90})$$

$$\approx 2.2 q^2 (\frac{R_0}{r})^{3/2} D_\perp^{CL} \approx 0.003\ m^2/s$$

$$\chi_{\perp,e}^{NC} \approx 0.89 q^2 (\frac{R_0}{r})^{3/2} \chi_{\perp,i}^{CL} \approx 0.001\ m^2/s$$

$$\chi_{\perp,i}^{NC} \approx 0.68 q^2 (\frac{R_0}{r})^{3/2} \chi_{\perp,i}^{CL} \approx 0.015\ m^2/s$$

Neoclassical transport increases D, χ up to two orders of magnitude, but still smaller by a few orders of magnitude!

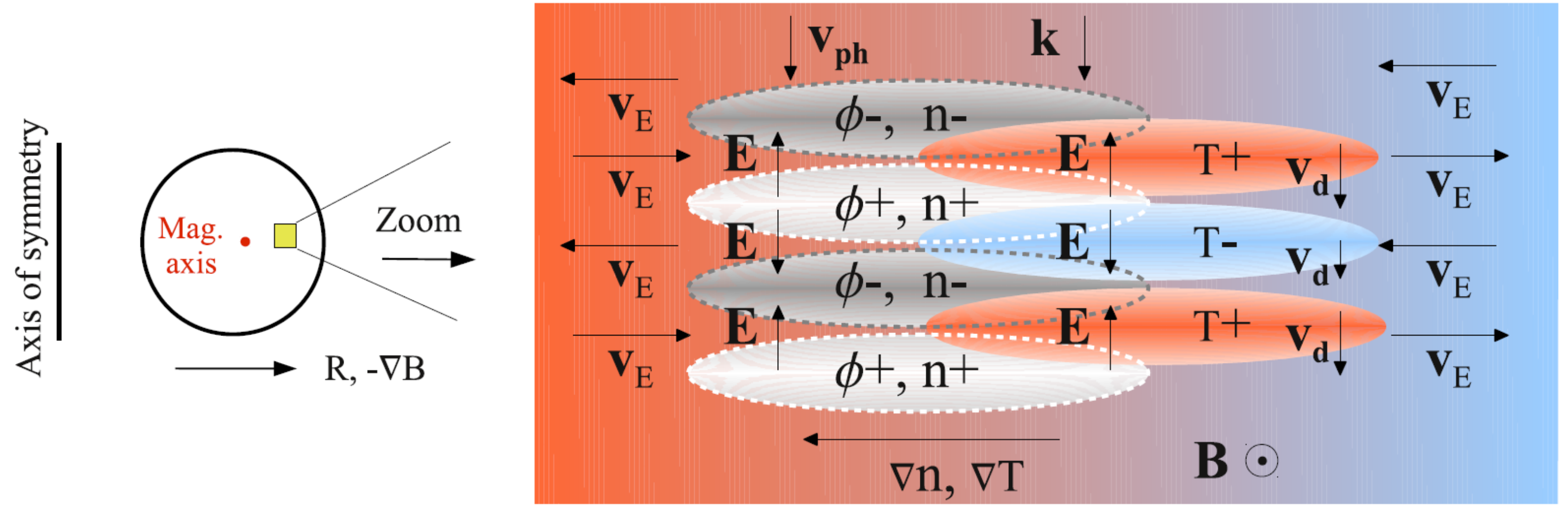
i.e. Transport in Experiments >> Neoclassical transport >> Classical transport

Anomalous transport

Observed in experiments that transport is correlated with turbulent fluctuations of n, ϕ, and B :

- radial extent of turbulent eddy: 1 - 2 cm
- typical lifetime of turbulent eddy: 0.1 - 1 ms

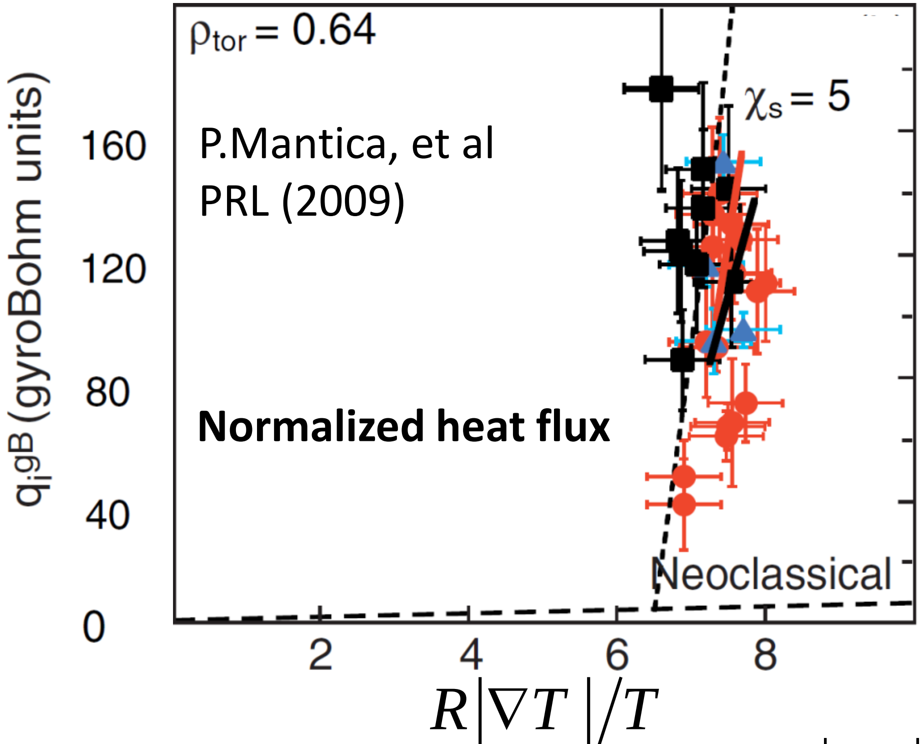
F. Casson, PhD thesis, Univ of Warwick 2011

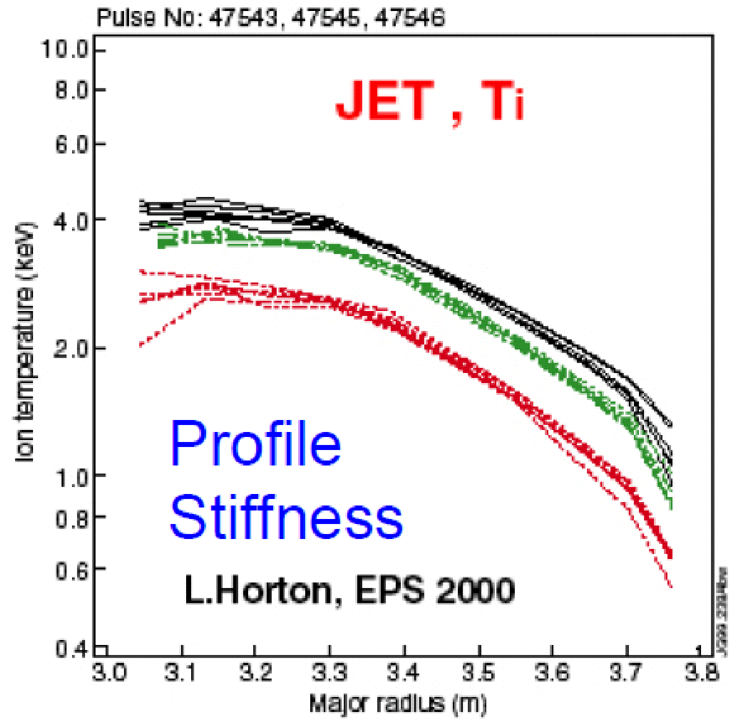


(a) Poloidal cross-section (b) Perturbation in the ions on the outboard side

Transport in Experiments = Anomalous transport + Neoclassical transport ≈ Anomalous transport

Ion Temperature Gradient mode (ITG) and T_i profile stiffness





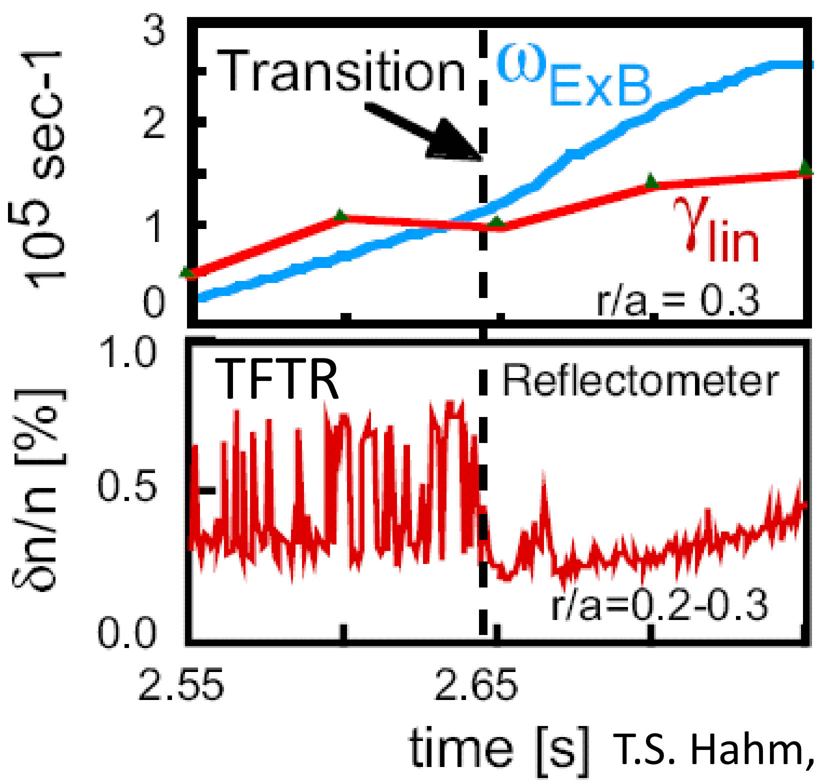
ITG-driven heat flux is triggered if $\dfrac{R|\nabla T|}{T} >$ ITG threshold $\left(\dfrac{R|\nabla T|}{T}\right)_c$

$\Rightarrow$ Heat flux increases strongly with small increase in ∇T above the ITG threshold

$\Rightarrow$ The resultant ∇T is closely tied to the ITG threshold

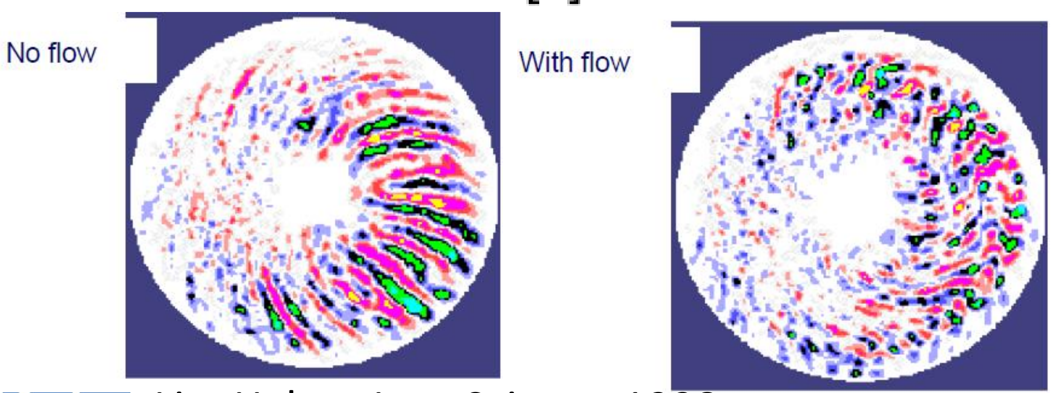
i.e. high stiffness of temperature profile.

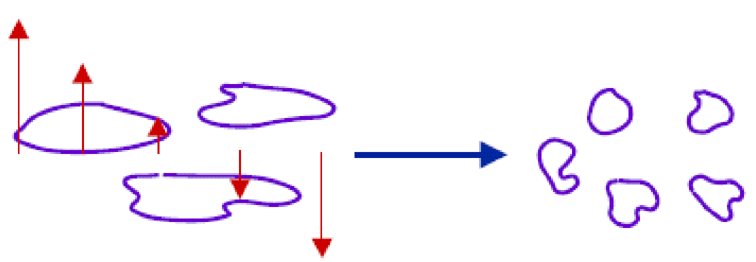
Enhanced confinement correlated with suppression of fluctuation



T.S. Hahm, PoP 1995

- Theory predicts fluctuation suppression when the shearing rate of the E_rxB flow (i.e. ω_{ExB}) exceeds the instability growth rate (i.e. γ_{lin}).
- Suppression accompanied by radial decorrelation of the fluctuation.
- Similar suppression observed on JET
- E_rxB flow shear is also responsible for the suppression of edge turbulence i.e. ETB and H-mode (next slide)



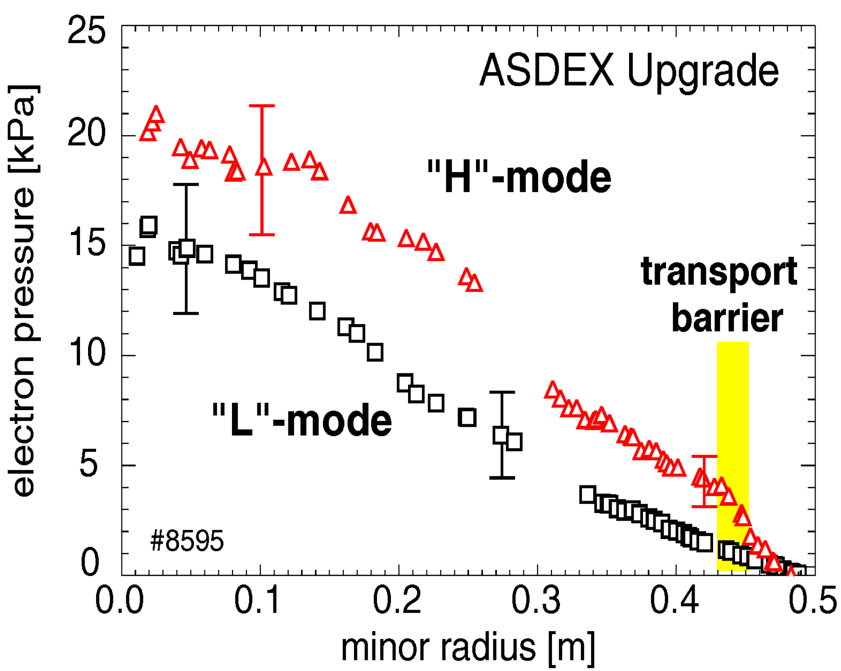


Lin, Hahm, Lee, Science 1998

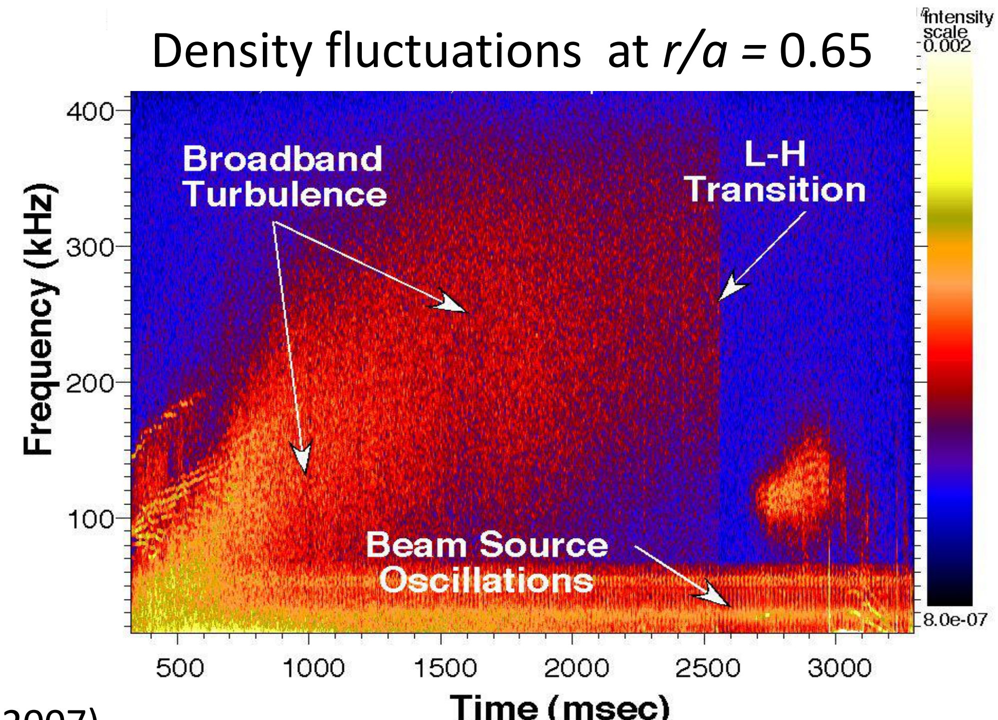
H-mode (High confinement mode)

Transition from L-mode to H-mode (i.e. L-H transition):

- High heating above certain threshold develops steep E_r at plasma edge.
- Increase in E_rxB flow shear rate at plasma edge.
- Reduced fluctuation and turbulent transport i.e. Edge Transport Barrier (ETB)
- Steep pressure gradient at plasma edge
- Core pressure profiles upshifted by profile stiffness → high confinement



Density fluctuations at r/a = 0.65



G.R. McKee, et al. Plasma Fusion Res. (2007)

Summary of Part 1

- Nuclear fusion is an ideal alternative energy source, and Tokamak is a magnetic confinement fusion device
- High and stable plasma pressure is required for high fusion power in a tokamak.
- Understanding plasma physics is therefore essential. Plasma physics in tokamaks are reviewed, in preparation for part 2.
  - ✓ **Equilibrium – force balance between plasma pressure and magnetic pressure.** *Key words: magnetic flux surface, (Normalized) beta, safety factor q, and Grad-Shafranov eqn, flux coordinate*
  - ✓ **Stability – required to keep the equilibrium.** *Key words: Linear MHD stability, Edge Localised Mode, Peel-Ballooning mode, Neoclassical Tearing Mode, Sawtooth instability, Disruption, Run-away electrons*
  - ✓ **Low transport – required to have high plasma pressure.** *Key words, Classical, Neoclassical, and Anomalous transport, Ion Temperature Gradient mode, Temperature profile stiffness, ExB flow shear stabilization, and H-mode*
- Now, we are ready for part 2. So, what scientists are now doing?

Contents

Part 1.

1) Nuclear fusion and Tokamaks
2) Plasma physics in Tokamaks
    a) Equilibrium
    b) Stability
    c) Transport

Part 2.

1) **ITER's goal**
2) **What we are now doing for ITER**
    a) **EURO*fusion* Roadmap**
    b) **Joint European Torus**
    c) **Fusion research at JET**

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Self-sustaining plasma needed for fusion power plant

For alpha heating > total energy loss:

$$\frac{1}{4} n^2 < \sigma \upsilon > \times 3.5\ [MeV]\ >\ \frac{3nT_i}{\tau_E}$$

where

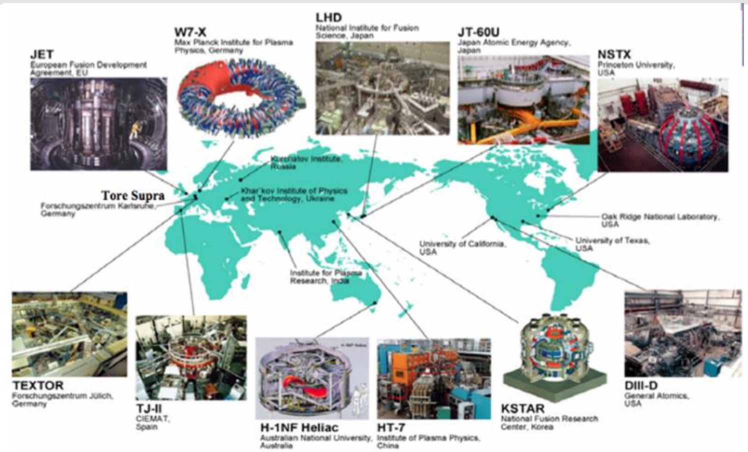
$n = n_e = n_i = 2n_D = 2n_T$, $T_i = T_e$, and

$< \sigma \upsilon > \approx 1.1 \times 10^{-24} T_i^2 [m^3 s^{-1}]$

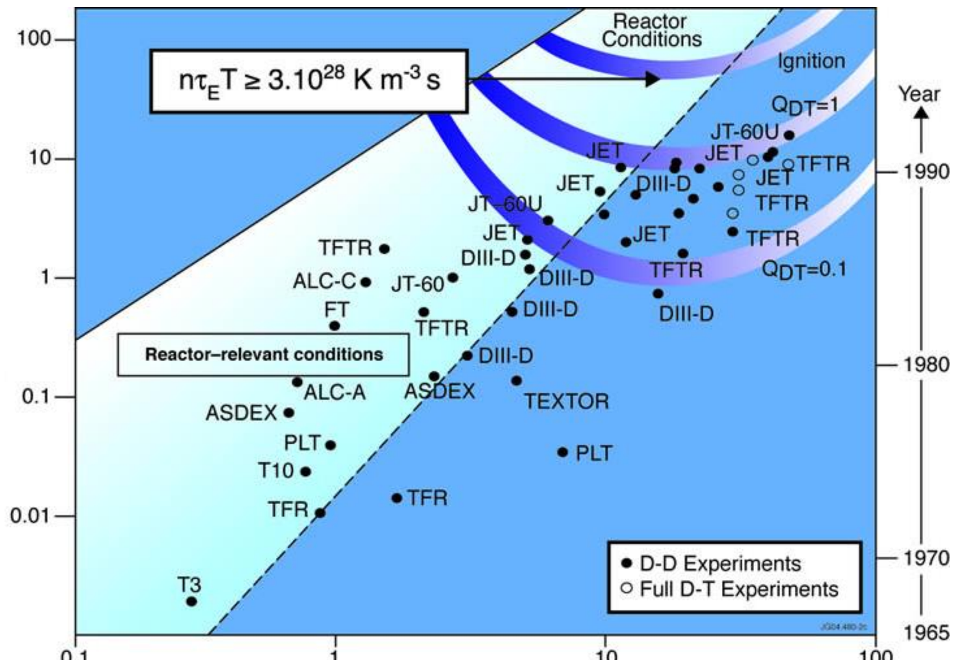
(T_i in keV, for T_i=10~20keV)

$$n \tau_E T_i > 3 \times 10^{21} [m^{-3}\ \mathrm{s}\ keV]$$

1. High T_i (~10 keV)
2. High n (~10^{20}m^{-3})
3. Long τ_E (~ 3s)



$n_i \tau_E T_i [10^{20} m^{-3} s\ keV]$



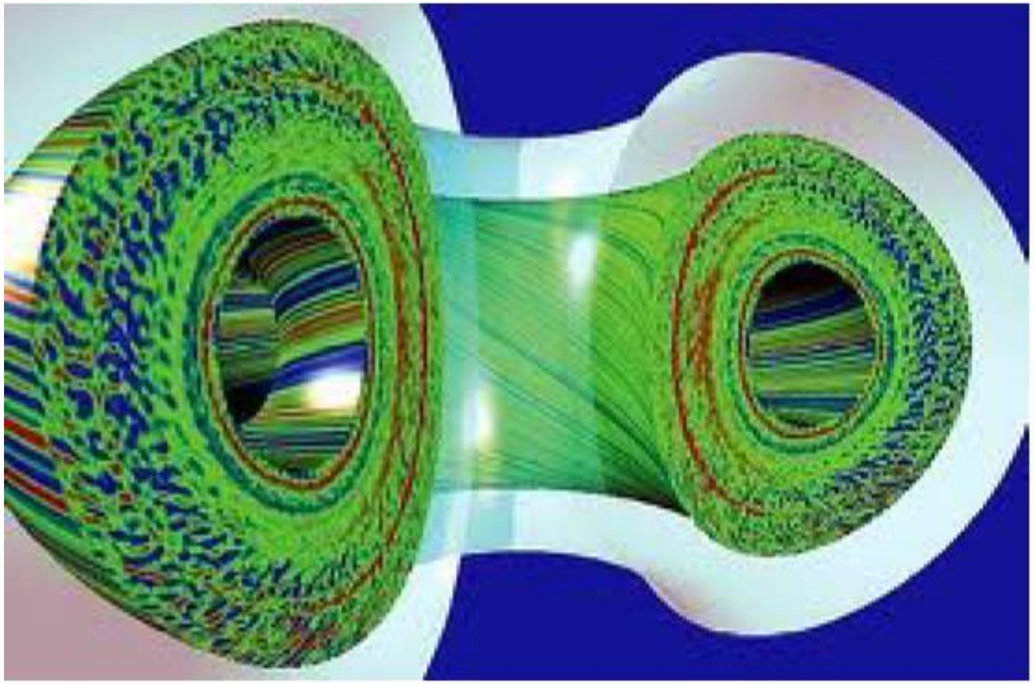
Central ion temperature T_i [keV]

Energy confinement time

Long enough τ_E is required (> 3 s) for self-sustaining plasmas.

However, τ_E measured in present devices is less than 1 seconds.

i.e. $\tau_E \equiv \dfrac{\text{Stored energy in the plasma}}{\text{Rate of energy loss from the plasma}}$



Simulations predict that the larger plasma volume, the longer τ_E :

$\tau_E \propto R^3 B^2 T^{-3/2}$

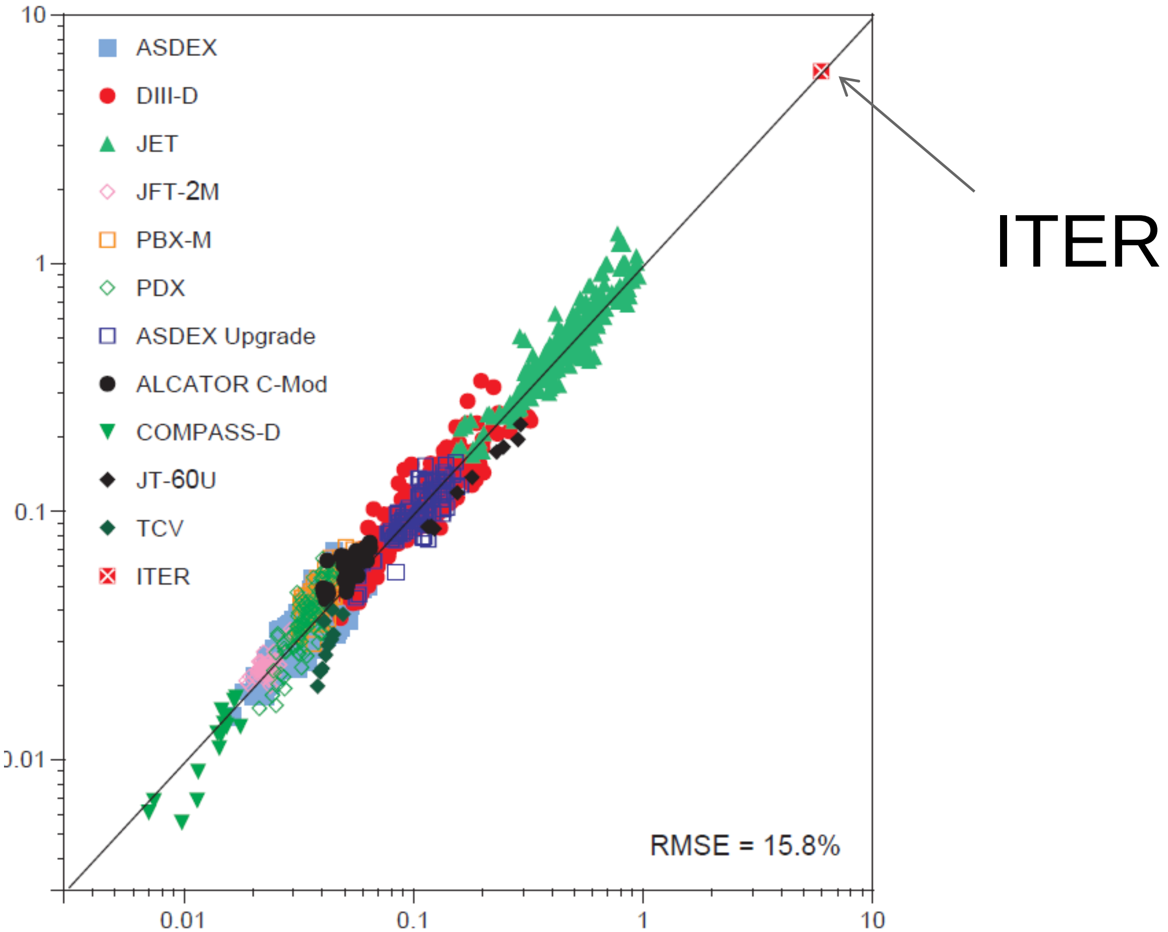
JET

Extrapolation to ITER

Measured value in experiments:

$\tau_E^{\exp}[s]$

$$\text{H factor} \equiv \frac{\tau_E^{\exp}}{\tau_E^{emp}}$$

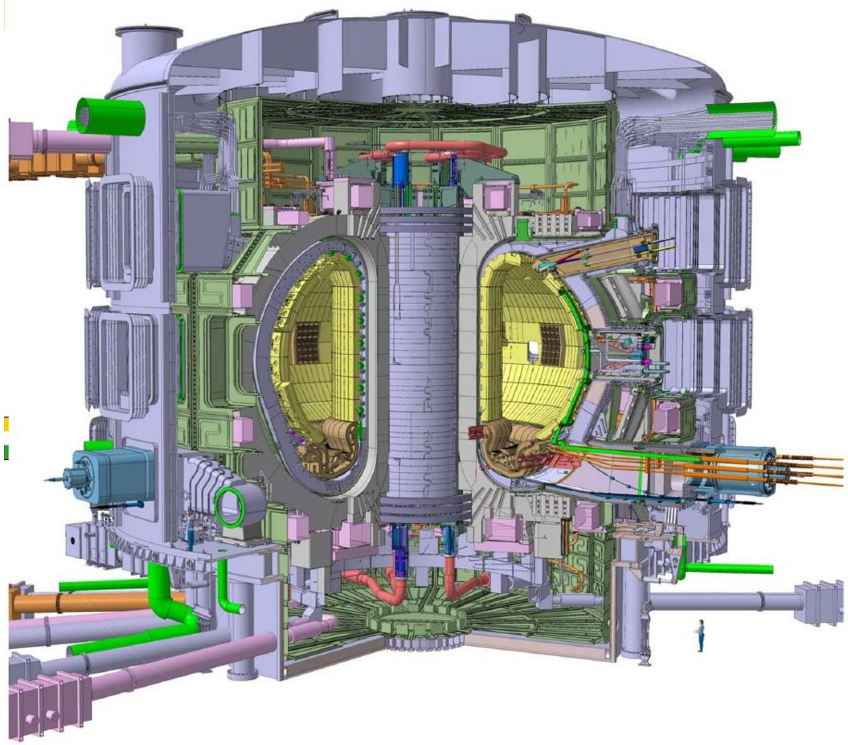


Empirical fit from regression analysis:

$$\tau_E^{emp}[s] = 0.0365 I_p^{0.97} B_\phi^{0.08} P^{-0.063} n^{0.41} M^{0.2} R^{1.93} \varepsilon^{0.23} \kappa^{0.67}$$

International Thermonuclear Experimental Reactor (ITER, "the way“ in Latin)

ITER's goal:
Demonstrate the feasibility of self-sustaining DT fusion plasmas (≈ Q=10), producing 500MW fusion power for 400sec.



- Official agreement between 7 International partners in 2005

- 13 billion euros for construction
- Cadarache in France
- First plasma in 2025
- DT experiments in 2035

Status of ITER construction in 2018


ITER site


TF coil case


Divertor cassette


Flags of ITER partners

Challenges in ITER operation

- Deuterium-Tritium operation
- Beryllium first wall + Tungsten divertor

Lack of experiences in operation

- 15MA of plasma current
- Large volume (R=6.2m, a=2m)

Risk of run-away electrons and disruption, Too high L-H threshold heating power

- High heating power (50MW additional heating+100MW α heating) → Risk of excessive heat load on divertors

ITER operation is very challenging. To ensure achieving ITER's goal, preparation in present devices is essential. European fusion research is therefore now focused on

1. Optimization of ITER operation scenario and technology
2. Mitigation of foreseen operational risk in ITER operation.

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JET

EURO*fusion* consortium

EURO*fusion* is the European consortium of 28 countries working together to achieve the ultimate goal of the EU Fusion Roadmap

- EURO*fusion* research units

EURO*fusion* consortium

EURO*fusion* is the European consortium of 28 countries working together to achieve the ultimate goal of the EU Fusion Roadmap

- EURO*fusion* research units

FUSION FOR ENERGY

F4E is the EU Domestic Agency for ITER: responsible for procurements

iter
china eu india japan korea russia usa

JET

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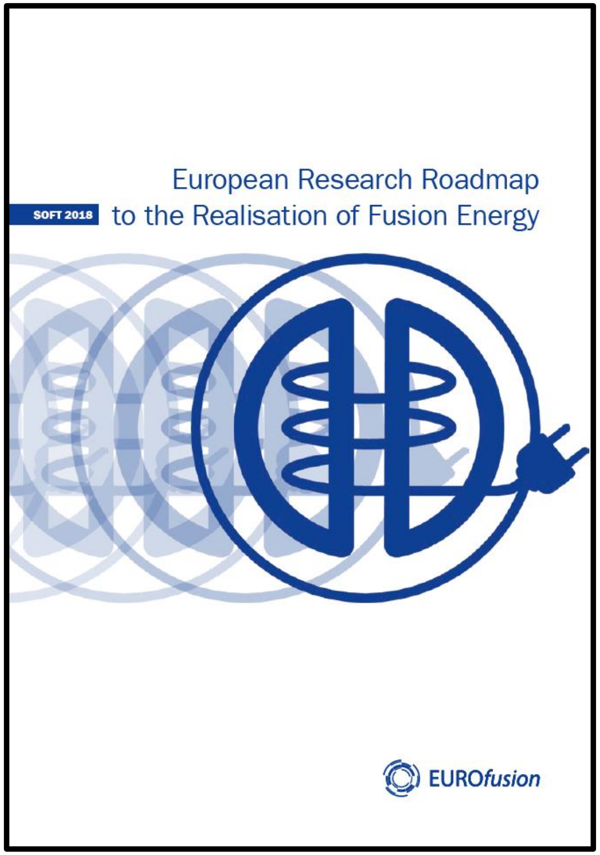
EU Fusion Electricity Roadmap



downloadable from

http://www.euro-fusion.org/

Demonstrate fusion electricity by 2050

- Provides coherent EU fusion programme with a clear objective
- Avoids open-ended R&D
- First issue written in 2012 by EFDA, predecessor of EURO*fusion*
- Revised in 2018, taking into account the ITER research plan announced in 2016
- **EURO*fusion* 'Bible' describing 8 Missions**

Eight missions in EU Roadmap

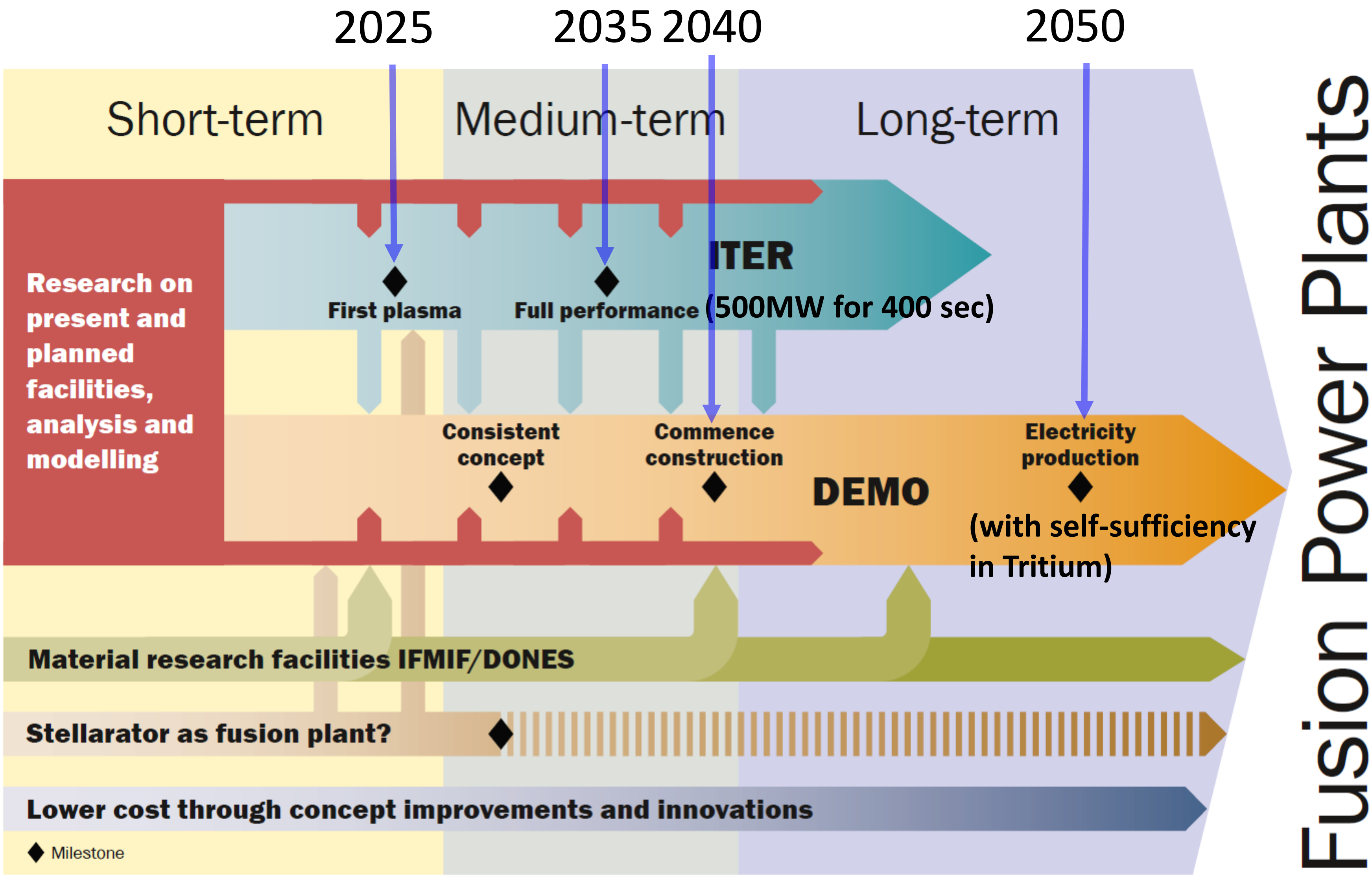
1. Plasma regimes of operation
2. Heat-exhaust systems

JET and MST campaigns ongoing, ITER campaigns after 2025

3. Neutron tolerant materials → IFMIF-DONES
4. Tritium self-sufficiency → Test Blanket Module (TBM) in ITER
5. Safety
6. DEMO → Demonstration of fusion electricity production. Conceptual design in progress. Engineering design with input from ITER and IFMIF-DONES. Commence construction soon after full performance in ITER.
7. Competitive cost of electricity
8. Stellarator → possible long-term alternative, W7X campaigns ongoing

❖ **The eight missions break down into EUROfusion work packages, and all work funded is strictly aligned with the EU Roadmap.**

2018 EU roadmap overview

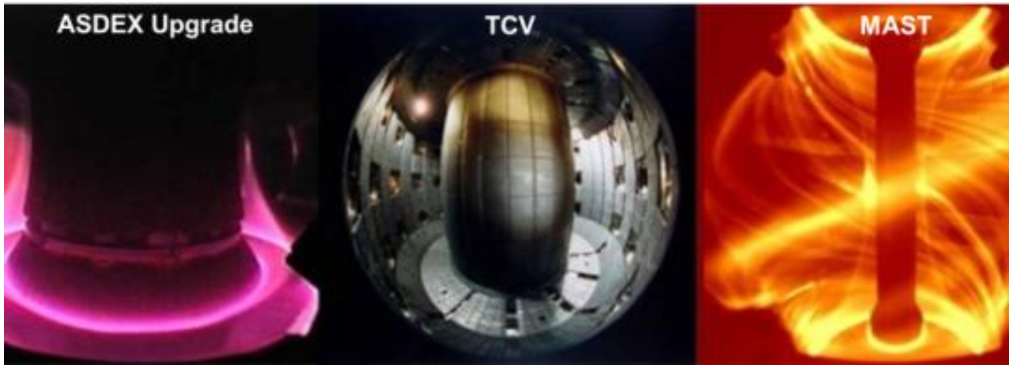


Top Objectives in 2018 – 2019 experimental campaigns in JET and MST

Joint European Torus (JET)

1) Prepare **scenarios for fusion performance** and alpha particle physics.
2) Determine the **isotopes dependence** of H-mode physics, SOL conditions and fuel retention.
3) Quantify the efficacy of **Shattered Pellet Injection** vs Massive Gas Injection on runaway electron and disruption energy dissipation and extrapolate to ITER

Medium Size Tokamak (MST); ASDEX-U, TCV, and MAST-U



1) Demonstrate the compatibility of **small, no/suppressed ELM regimes** for ITER and DEMO
2) Develop and characterize conventional and alternative **divertor configurations** for ITER and DEMO
3) Develop/characterize methods to predict and avoid **disruptions** as well as control/mitigate **runaway electrons** and demonstrate their portability.

JET

Contents

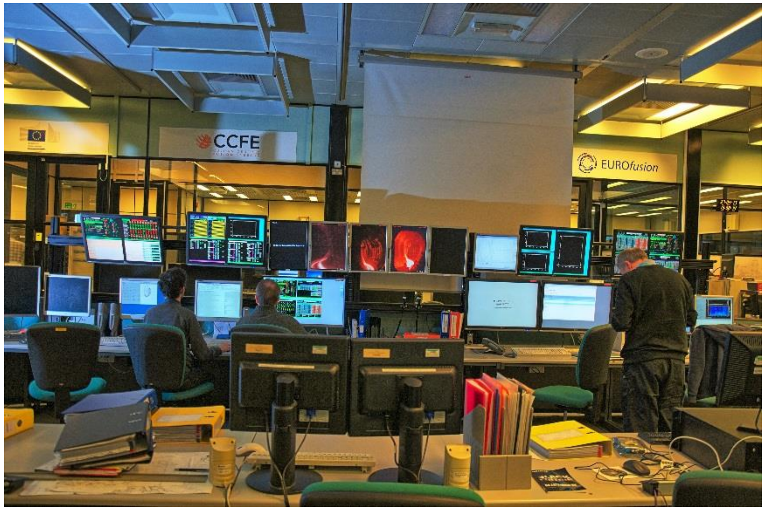
JET, the Joint European Torus

- The world's largest operational tokamak
  - First plasma in 1983
  - 16MW peak DT fusion power in 1997

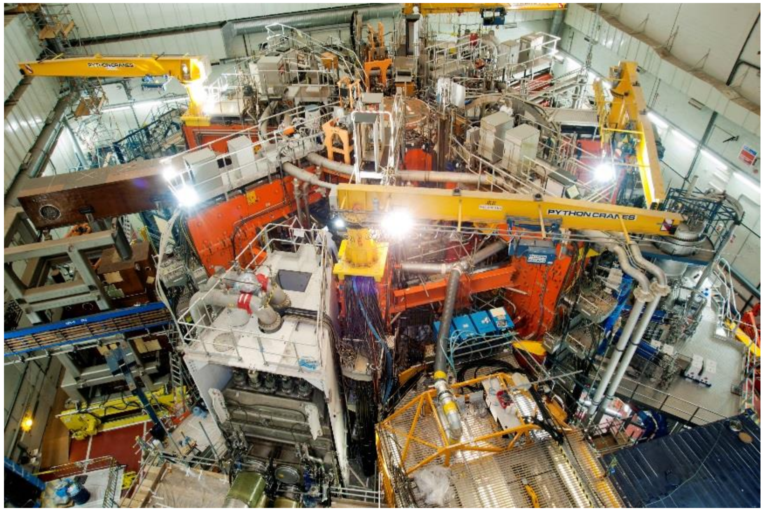
- Operated by the UKAEA (550 full time engineers) under contract from the European Commission
- Exploited by EURO*fusion*, with about 400 scientists from all over Europe.

- Intermediate step towards ITER because of
  - Tritium capability
  - ITER-like wall (beryllium and tungsten)
  - large size (R~3m)
  - High heating power (~40MW)
  - Remote handling

JET Control Room



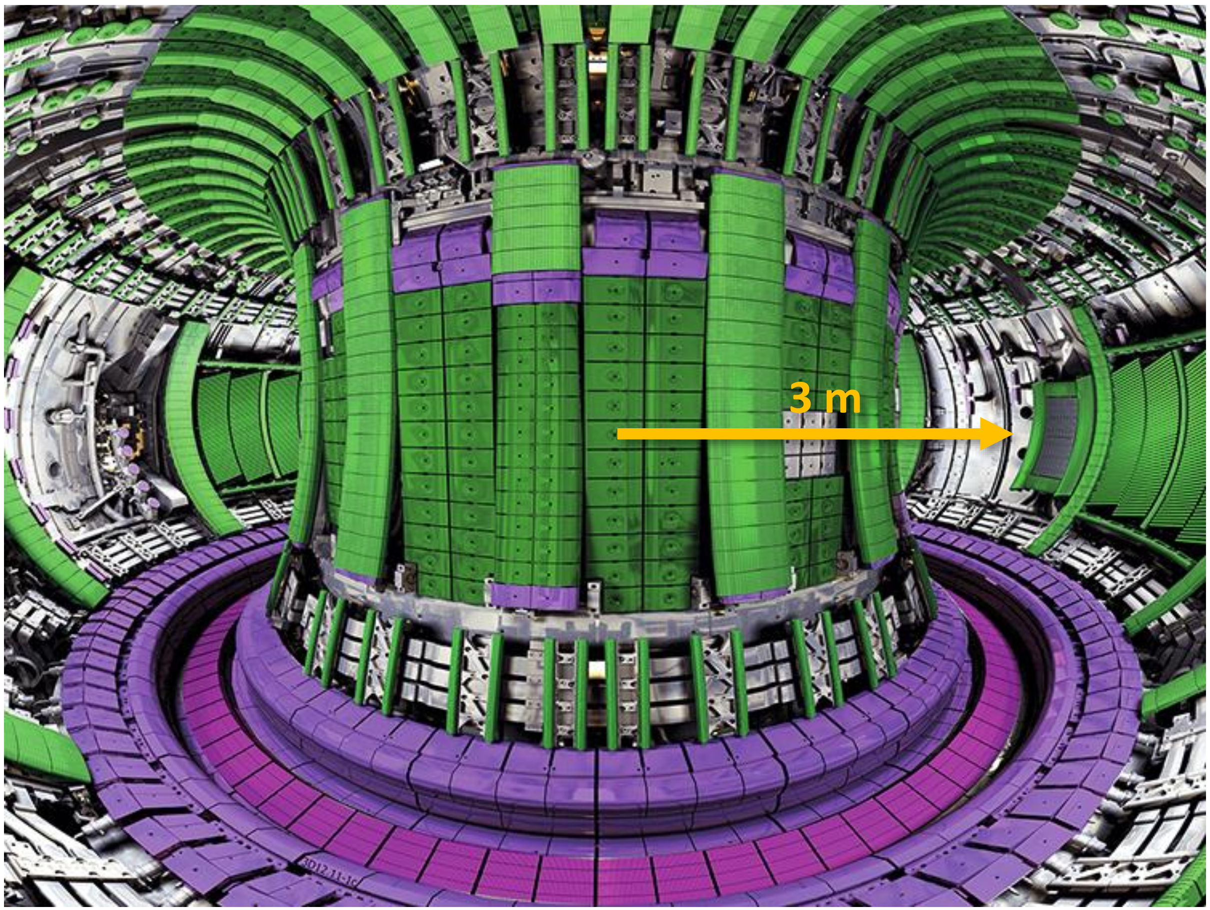
JET Torus Hall



JET

JET's ITER-like Wall

C wall refurbished with W divertor + Be main chamber in 2011

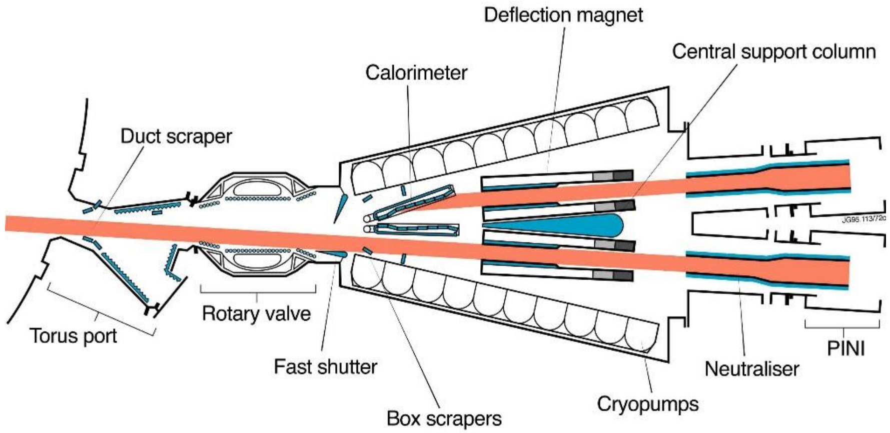


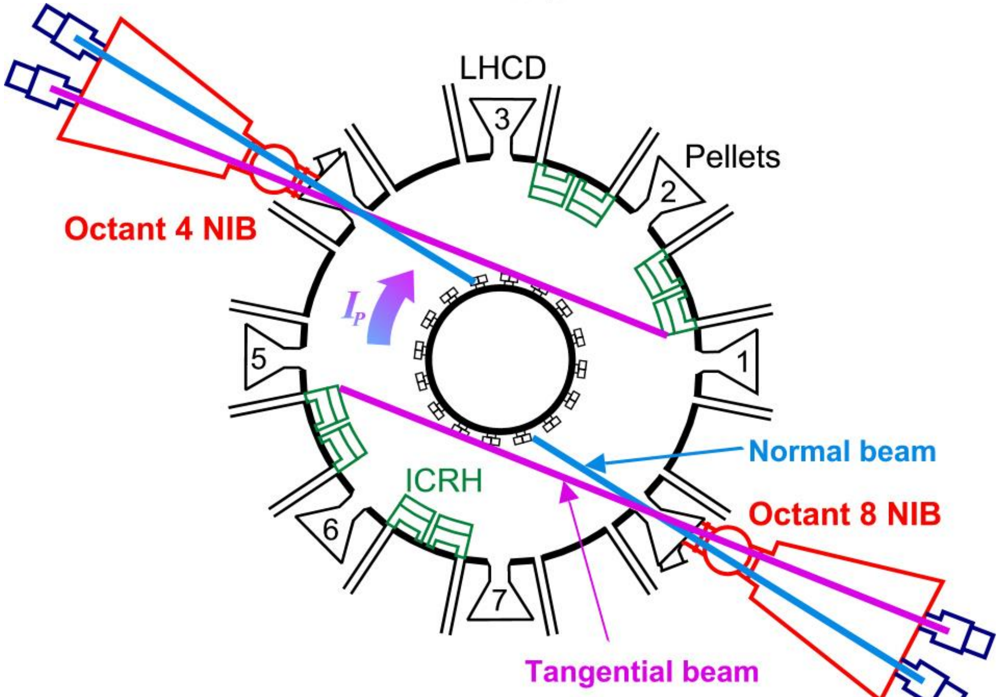
CPS15.139-2c

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

JET

JET Neutral Beam Heating System





Two Neutral Injector Boxes (NIBs) at JET

Fast neutral beam particles (e.g. 125kV) injected into the plasma are ionised by ion or electron impact ionisation or by charge exchange.

Once charged, the fast ions are trapped in the magnetic field, and heat the background ions and electrons by collisions.

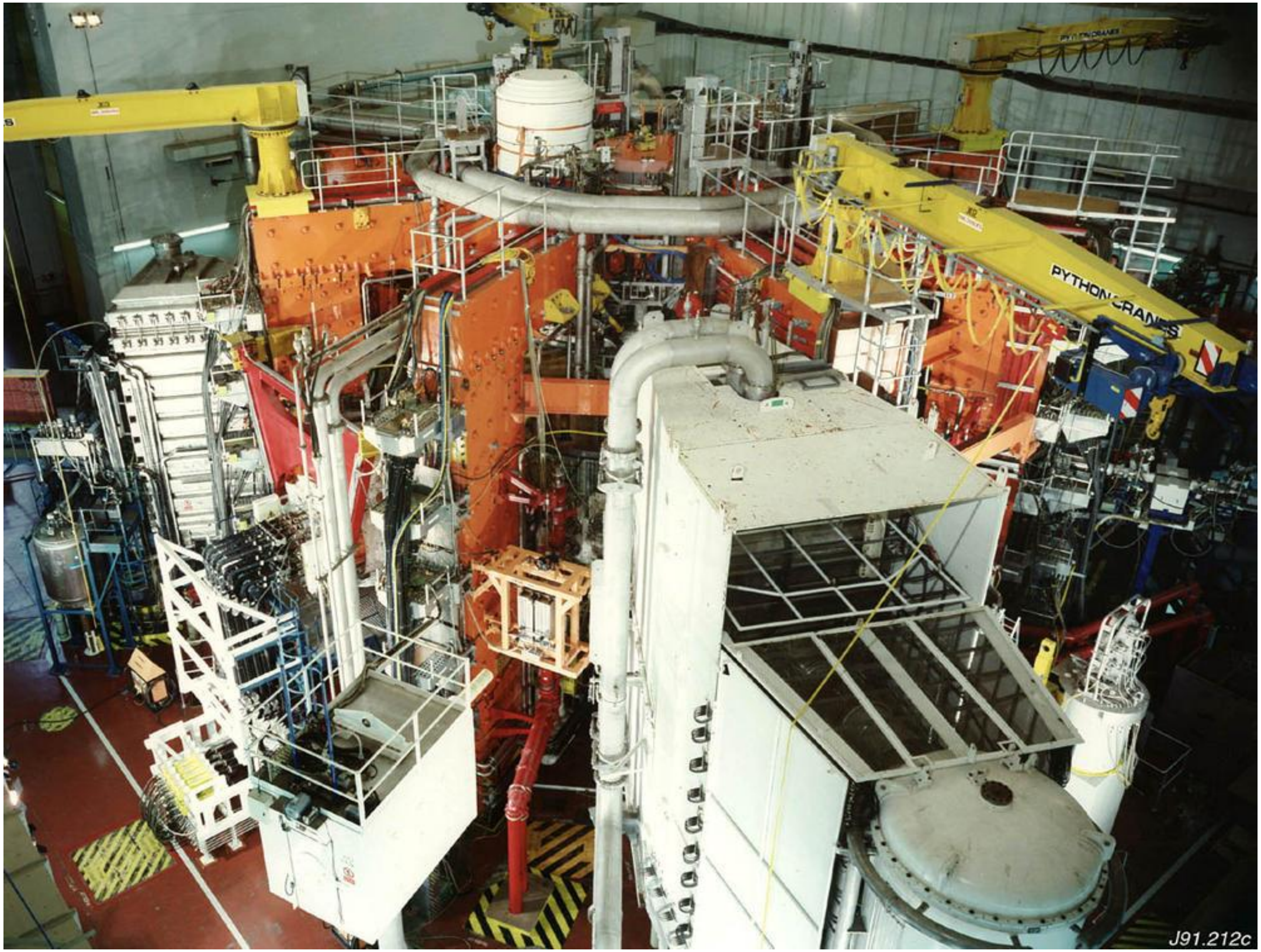
Operational in H, D, T and He.

Designed for 20s pulse duration.

Total deuterium neutral beam power upgraded to **35MW** (maximum) in 2011; previous maximum power was 20MW

JET

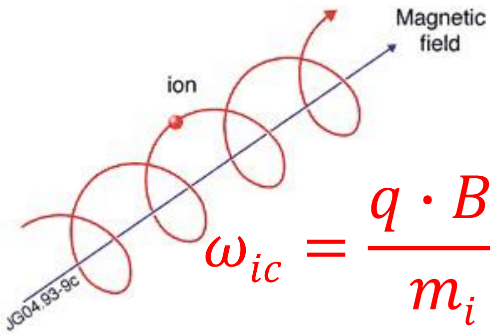
JET Neutral Beam Heating System



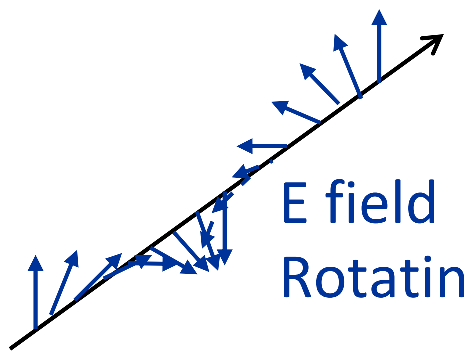
JET machine and Octant 4 Neutral Injector Box

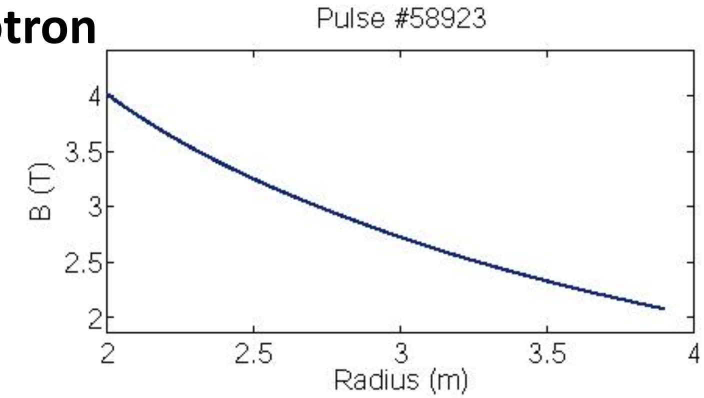
ICRH physics

Ions gyrate around magnetic field lines at an **Ion Cyclotron frequency** – $\boldsymbol{\omega_{ic}}$:



$$\omega_{ic} = \frac{q \cdot Bt}{m_i}$$

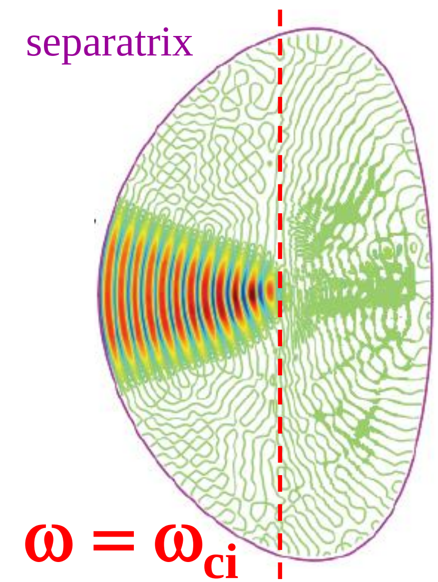




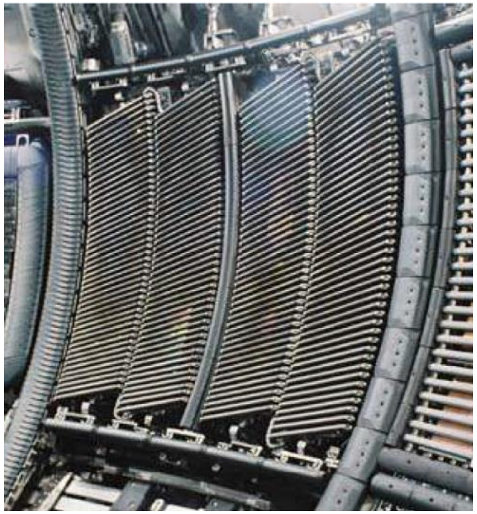
An ICRH antenna launches into the plasma a RF wave with a rotating electric field in the MHz frequency range i.e. ω_{RF}

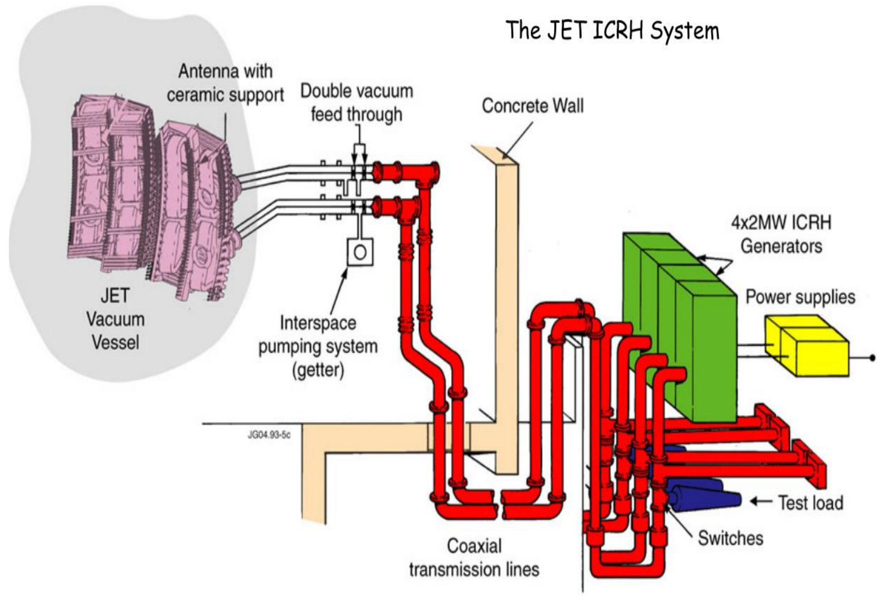
If $\omega_{RF} = \omega_{ic}$, the ions are in **resonance** with the wave, and

- see a constant (coherent) electric field,
- are accelerated → absorbing the energy of the RF wave.
- The energy of the accelerated particles is transferred to the surrounding particles via collisions.
- Local heating at specific R is possible, as ω_{ic} is a function of R

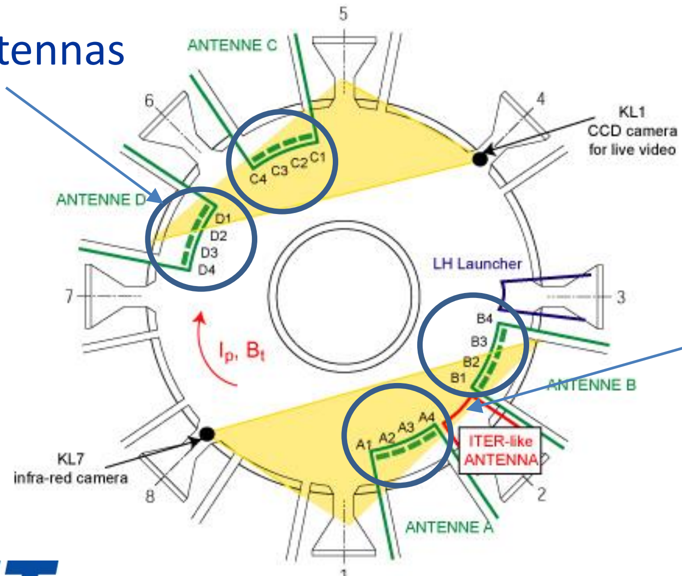


JET ICRH System





A2 antennas



ITER-like antenna

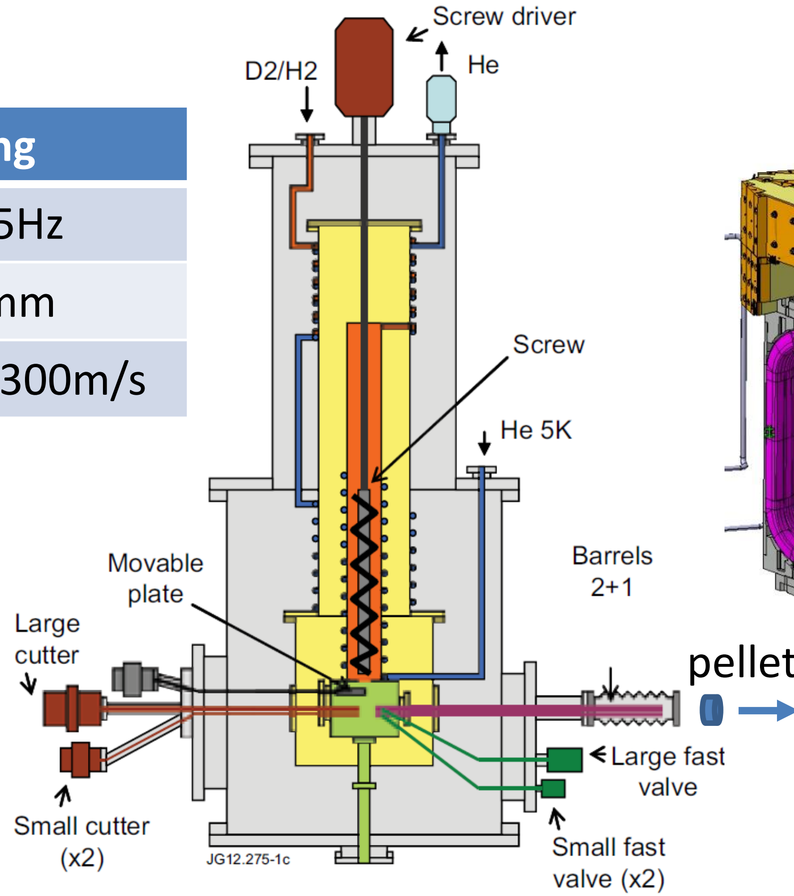
Antenna type	Available power
A2 antennas	4-6 MW at 28-56 MHz
ITER-like antennas	1-2 MW at 33-51 MHz

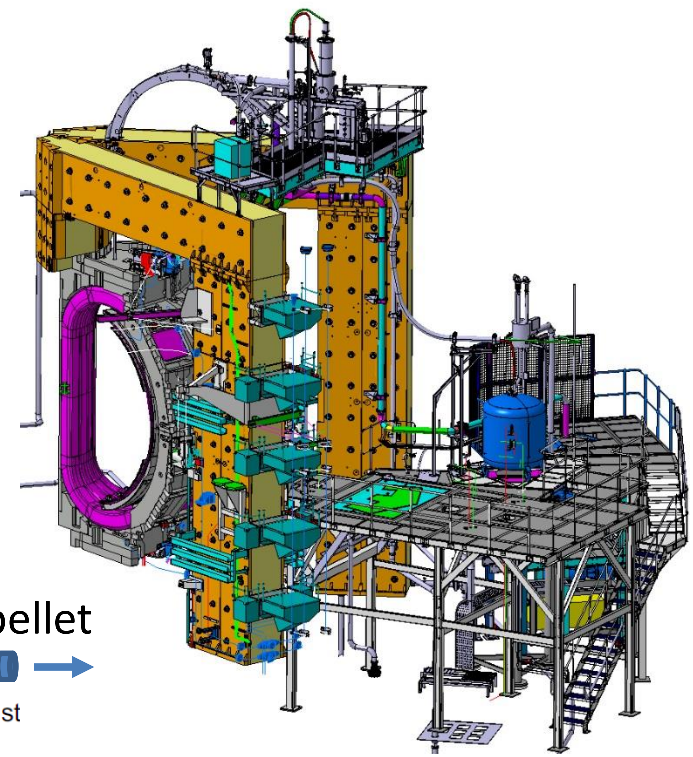
JET

Pellet system

- Standard fuelling of the plasma is to inject H or D gas by opening a gas valve, but particles are deposited just inside the scrape-off layer (i.e. no core fuelling).
- Pellet system is to inject frozen 'pellets' of H or D at high speed.
- Mainly fuelling purpose but it also triggers ELMs i.e. increase in ELM frequency.

Pacing	Fuelling
10 to 50 Hz	0 to 15Hz
D= 2 mm	D= 4 mm
80 - 200m/s	100 – 300m/s





Summary of JET facilities

- R= 3m, a= 1m
- Maximum I_p: 4.5 MA
- Maximum B_t: 3.85 T
- Maximum P_{heat}: 35MW NBI and 8MW ICRH
- Pellet injector for ELM pacing and plasma fuelling
- Pulse duration:10~20 sec flat top
- Main gas species available: H, D, and T
- ITER-like wall: Be first wall and W divertor
- Disruption mitigation system: 2 MGIs + SPI

Contents

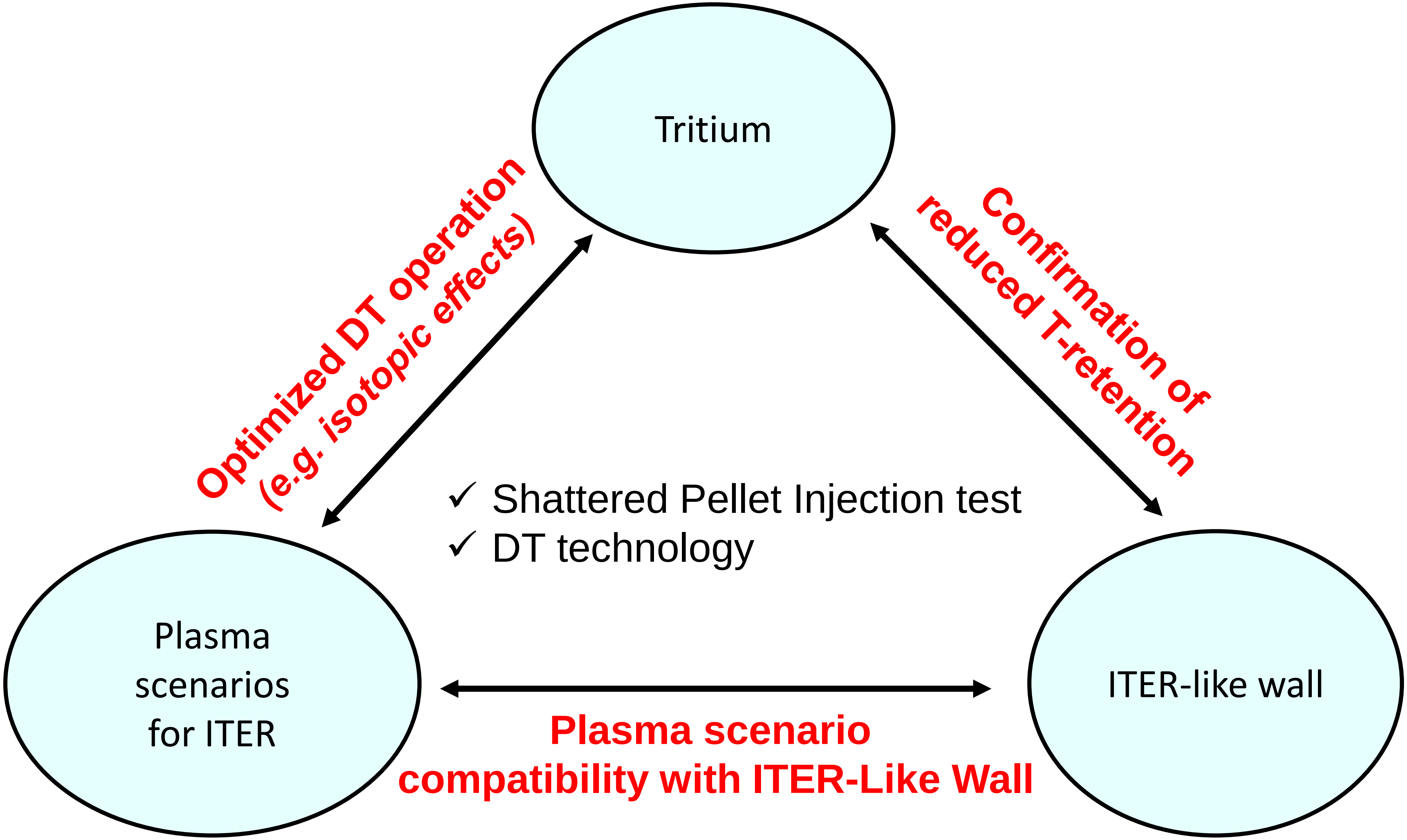
Part 1.

1) Nuclear fusion and Tokamaks
2) Plasma physics in Tokamaks
    a) Equilibrium
    b) Stability
    c) Transport

Part 2.

1) **ITER’s goal**
2) **What we are now doing for ITER**
    a) **EURO*fusion* Roadmap**
    b) **Joint European Torus**
    c) **Fusion research at JET**

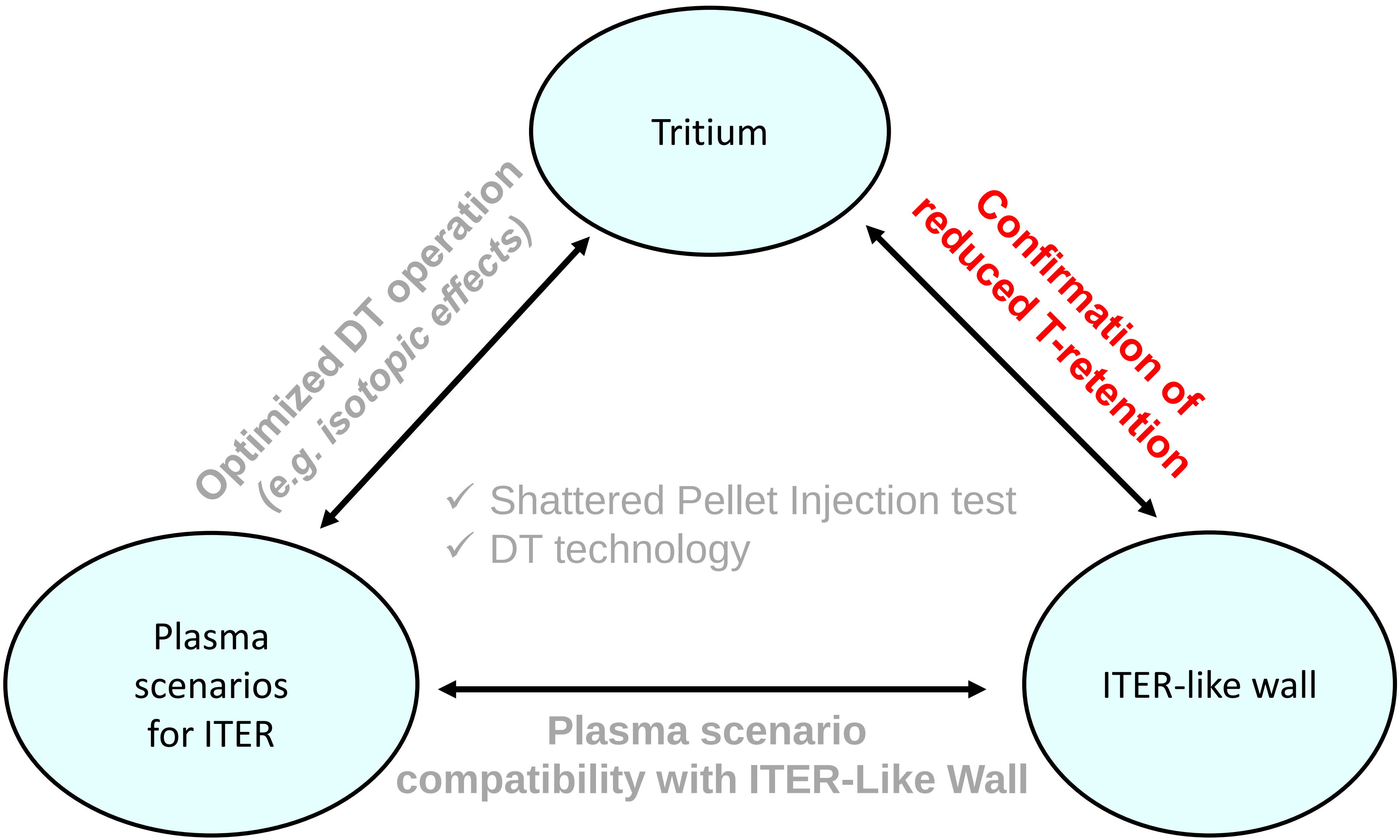
JET Programme in Support of ITER



J. Paméla et al., Fusion Eng. Des. (2007)

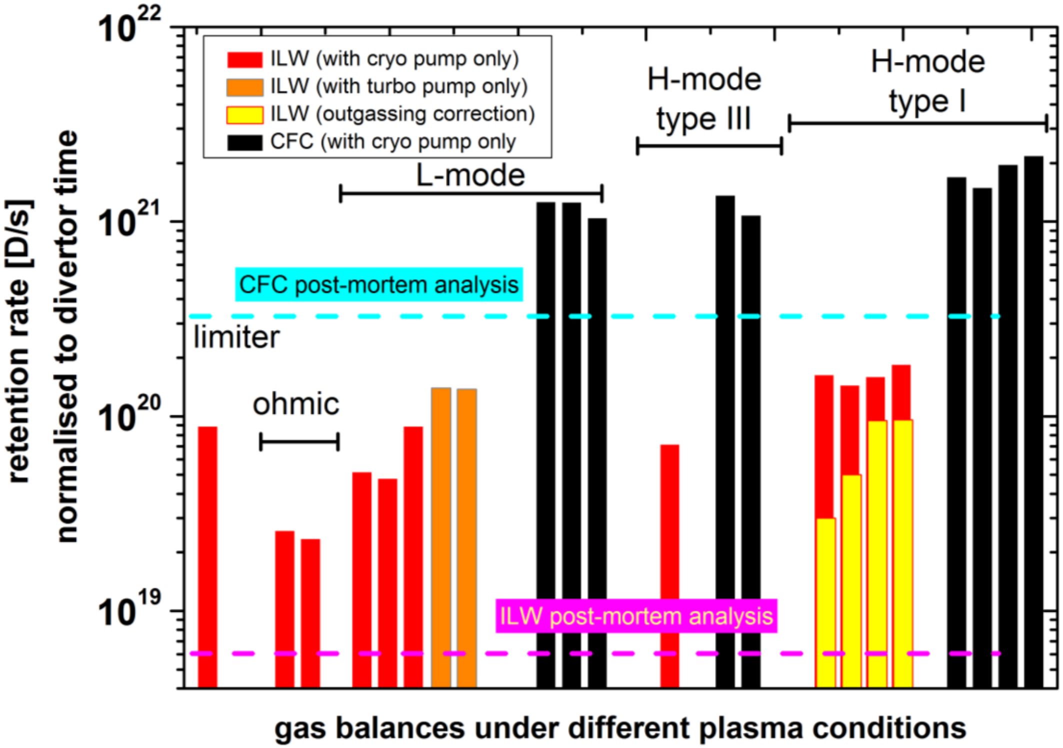
JET

JET Programme in Support of ITER

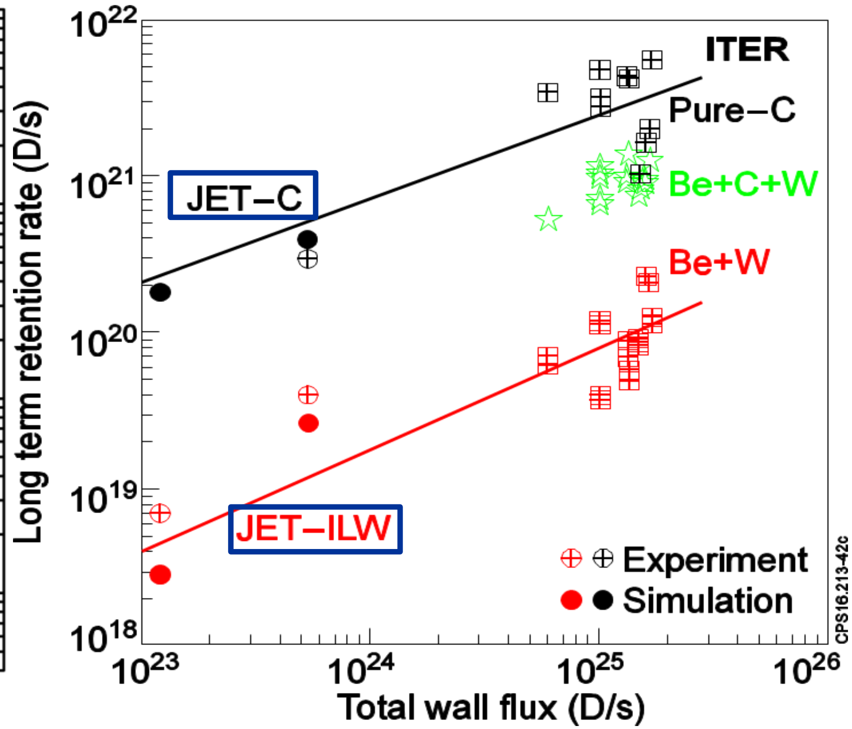


JET

Fuel Retention with the ITER-Like Wall



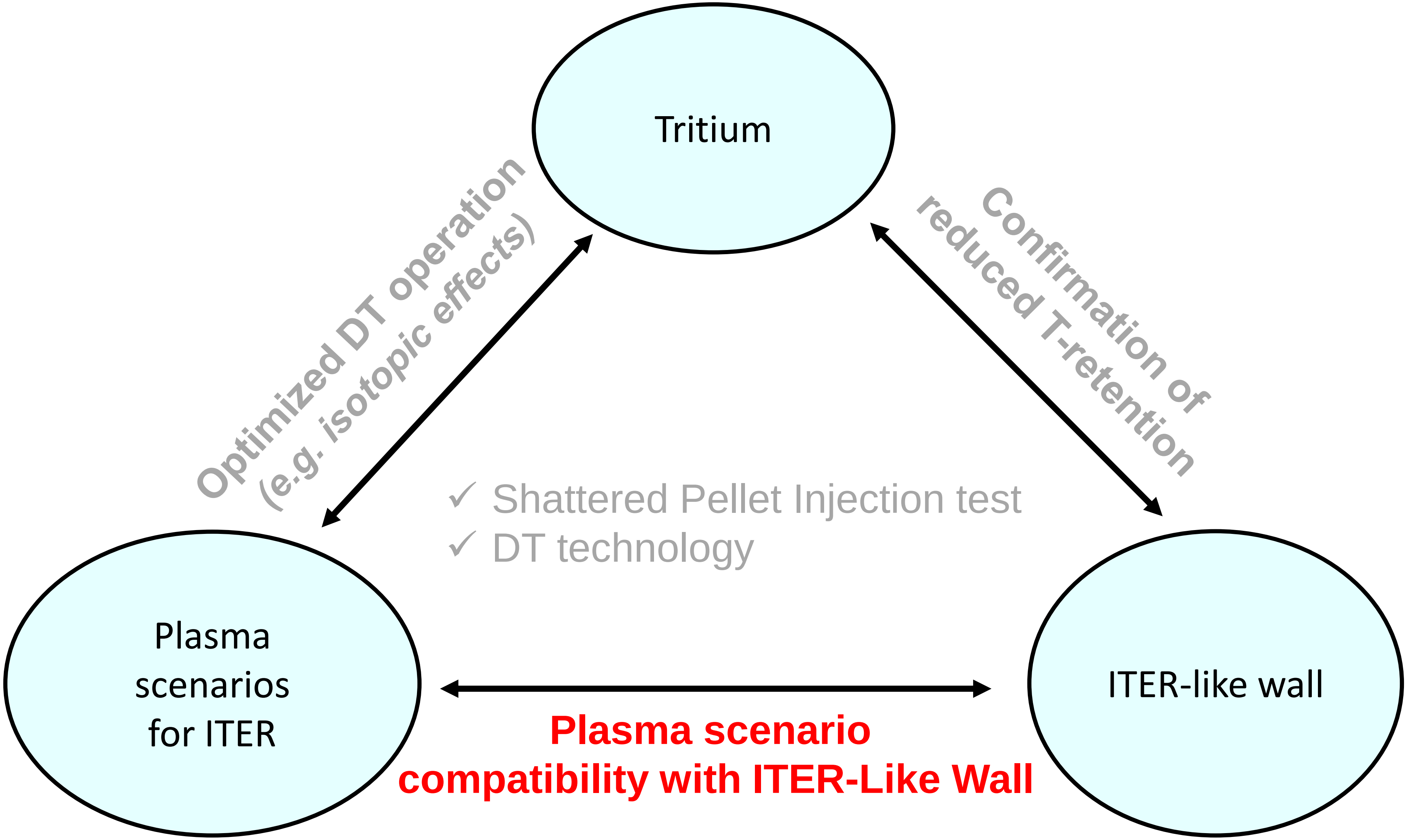
S. Brezinsek et al. NF (2013)



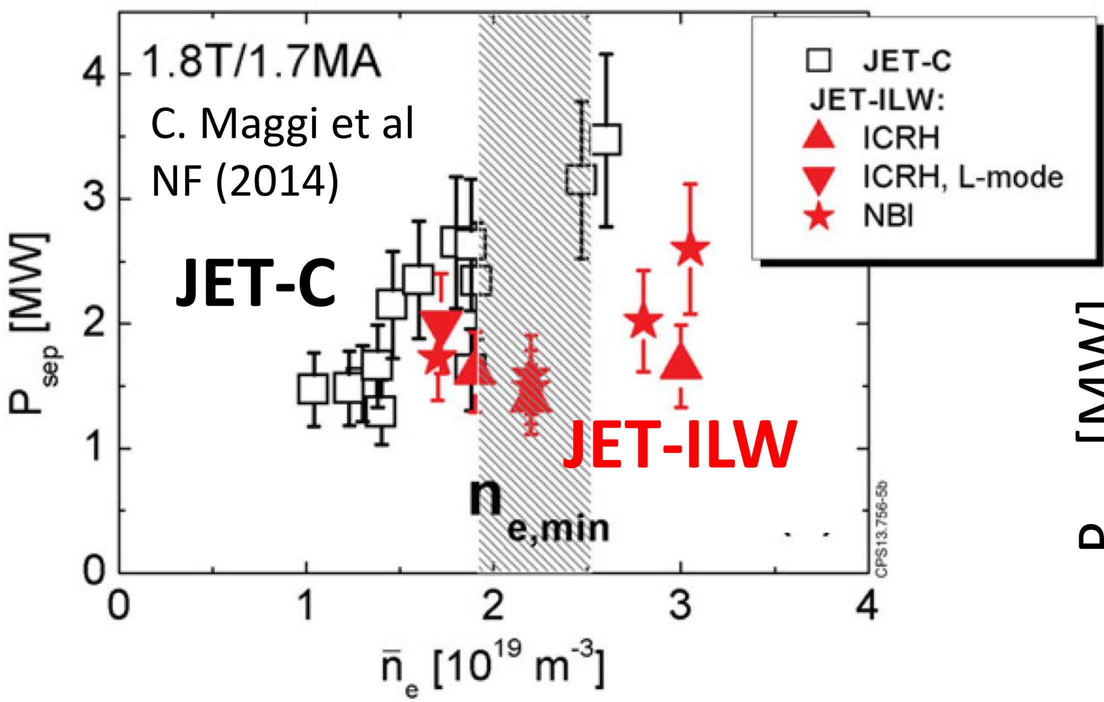
K. Schmid NF (2015)

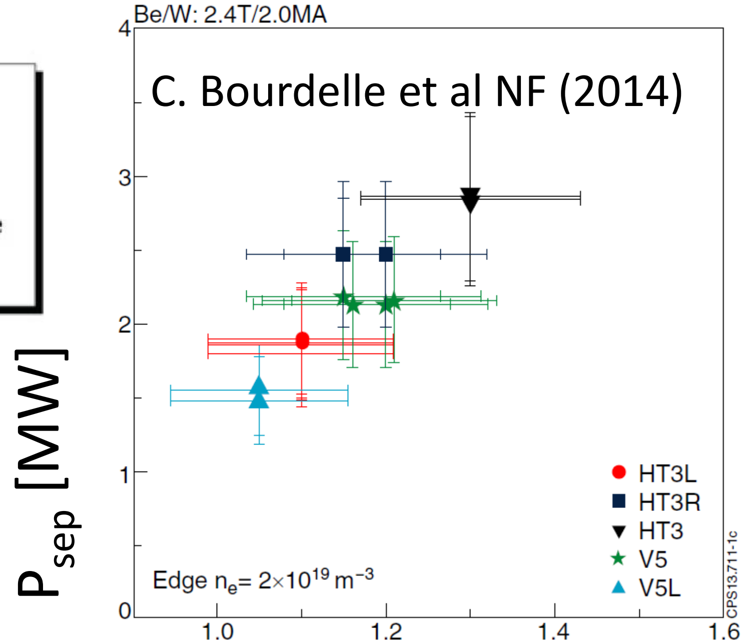
- T retention at carbon wall is unacceptably high for ITER.
- Fuel retention reduced by more than one order of magnitude in ITER-Like Wall
- WallDYN reproduced the fuel retention rate measured in JET-ILW and JET-C
- Simulations extrapolated to ITER predict that 3000-20000 of 400sec discharges would be feasible for Tritium experiments in ITER.

JET Programme in Support of ITER



L-H transition threshold power in ITER-Like wall



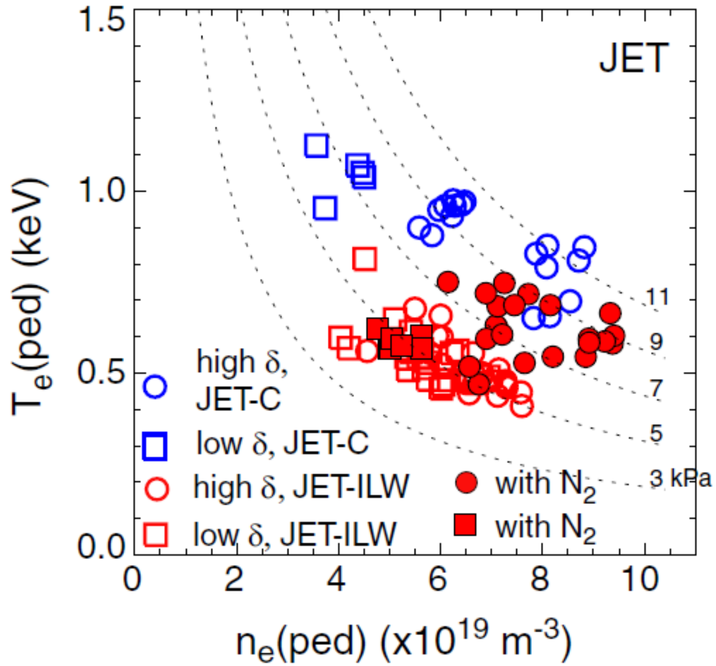


$$Z_{eff} (\equiv \sum_j n_j Z_j^2 \Big/ \sum_j n_j Z_j)$$

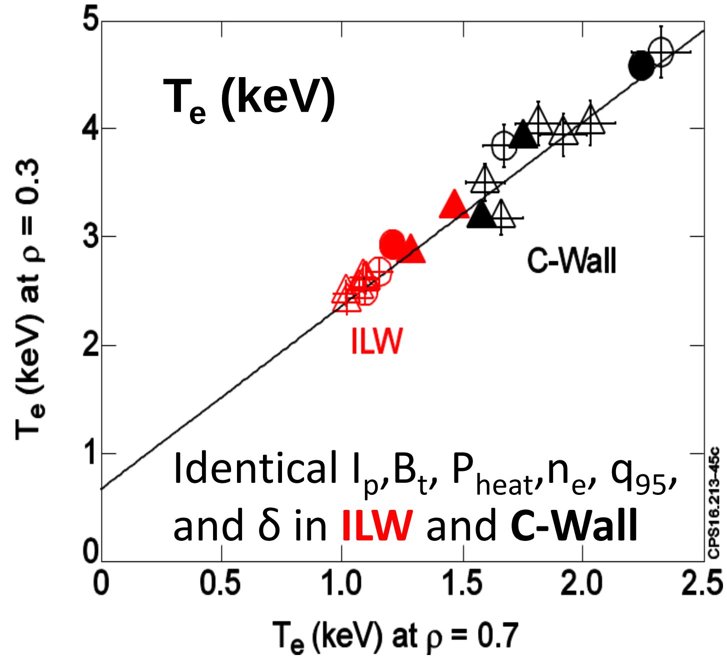
- L-H transition requires heating power above a certain threshold P_{th}.
- P_{th} increases with B_t and plasma size, and is an issue in ITER (P_{th} ~ 50MW predicted by Martin scaling, while 73MW is the maximum heating power in ITER)
- P_{th} was measured by $P_{sep}=P_{aux} + P_{oh} - P_{rad} - dW_{dia}/dt$ at L-H transition
- **P_{th} is reduced by 40% with ITER-Like wall, compared to JET-C wall.**
- **This should be associated with lower Z_{eff} in ILW.**

JET

Confinement with ITER-Like Wall



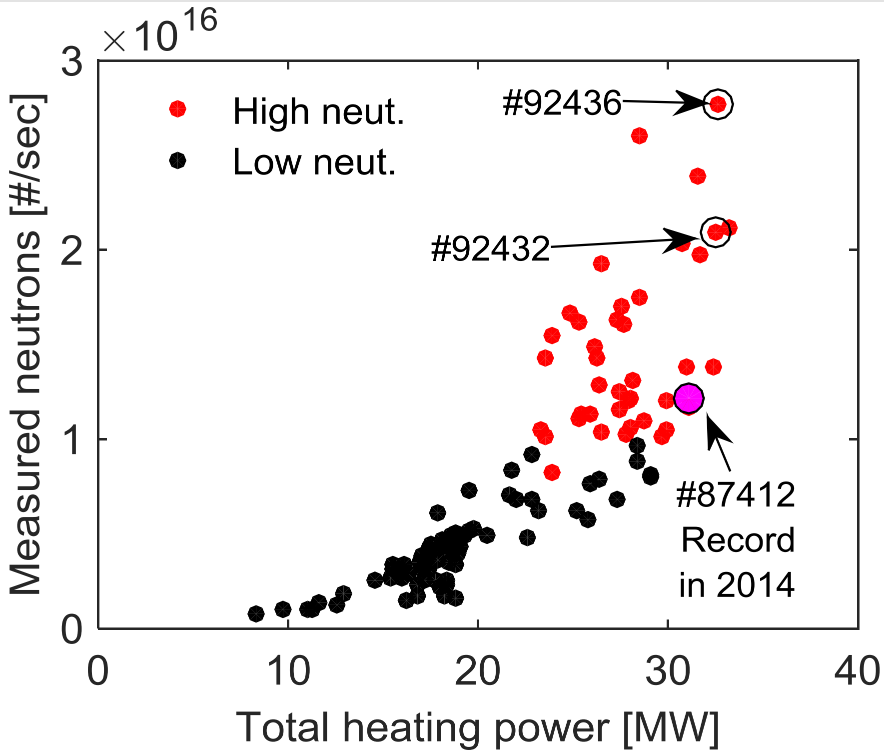
M. Beurskens et al, PPCF (2013)

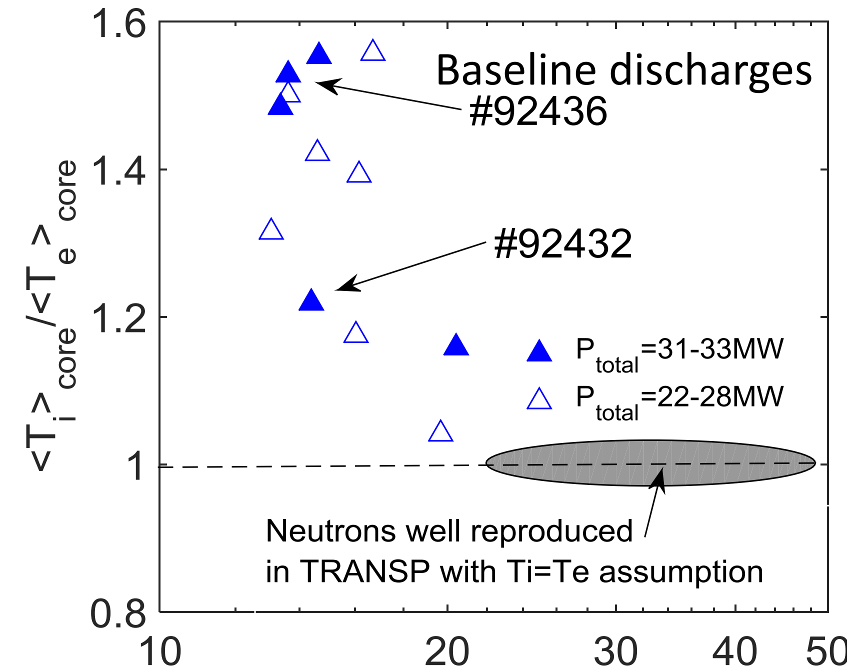


Hyun-Tae Kim et al, PPCF (2015)

- In ILW, high gas fuelling was needed to increase ELM frequency, to avoid W accumulation in the core. This degraded pedestal confinement.
- (N_2 seeding helps partial recovery of pedestal confinement. SOL composition may play a role.)
- Core confinement similar as C-Wall (with identical I_p,B_t, P_{heat},n_e, q_{95}, and δ)
- In 2016, higher heating power and low gas puffing recovered global confinement.

High fusion performance at high T_i/T_e with high heating power and low gas puffing





$$\nu_{ex} \equiv 0.32\times 10^{-14}\,\frac{m_e}{m_i}\,\frac{<n_e[m^{-3}]>}{<T_e[keV]>^{3/2}}$$

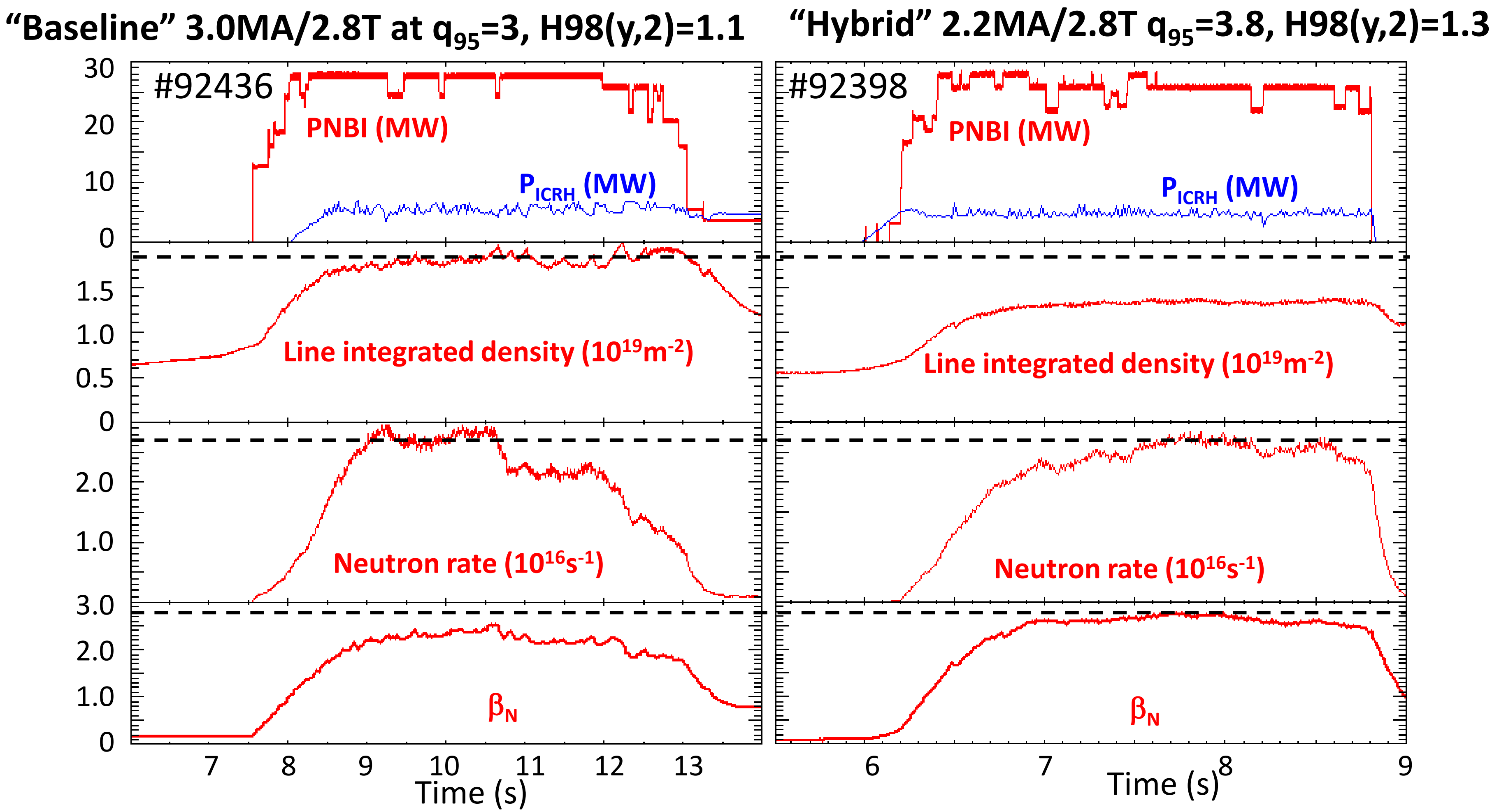
- New DD fusion rate record achieved in 2016.
- Significant increase in thermal neutrons, not beam-target neutrons.
- Attributed to high T_i , exceeding T_e in (high n_e) baseline scenario.

Heating power (>30MW) was high enough to approach low collisionality regime.
Ion heat transport reduced by positive feedback between high T_i/T_e and ITG stabilisation.
High T_i/T_e also correlated to high rotation at low gas puffing, enabled by pellet injection.

Hyun-Tae Kim et al, NF (2018)

JET

Operation scenario with ITER-Like Wall

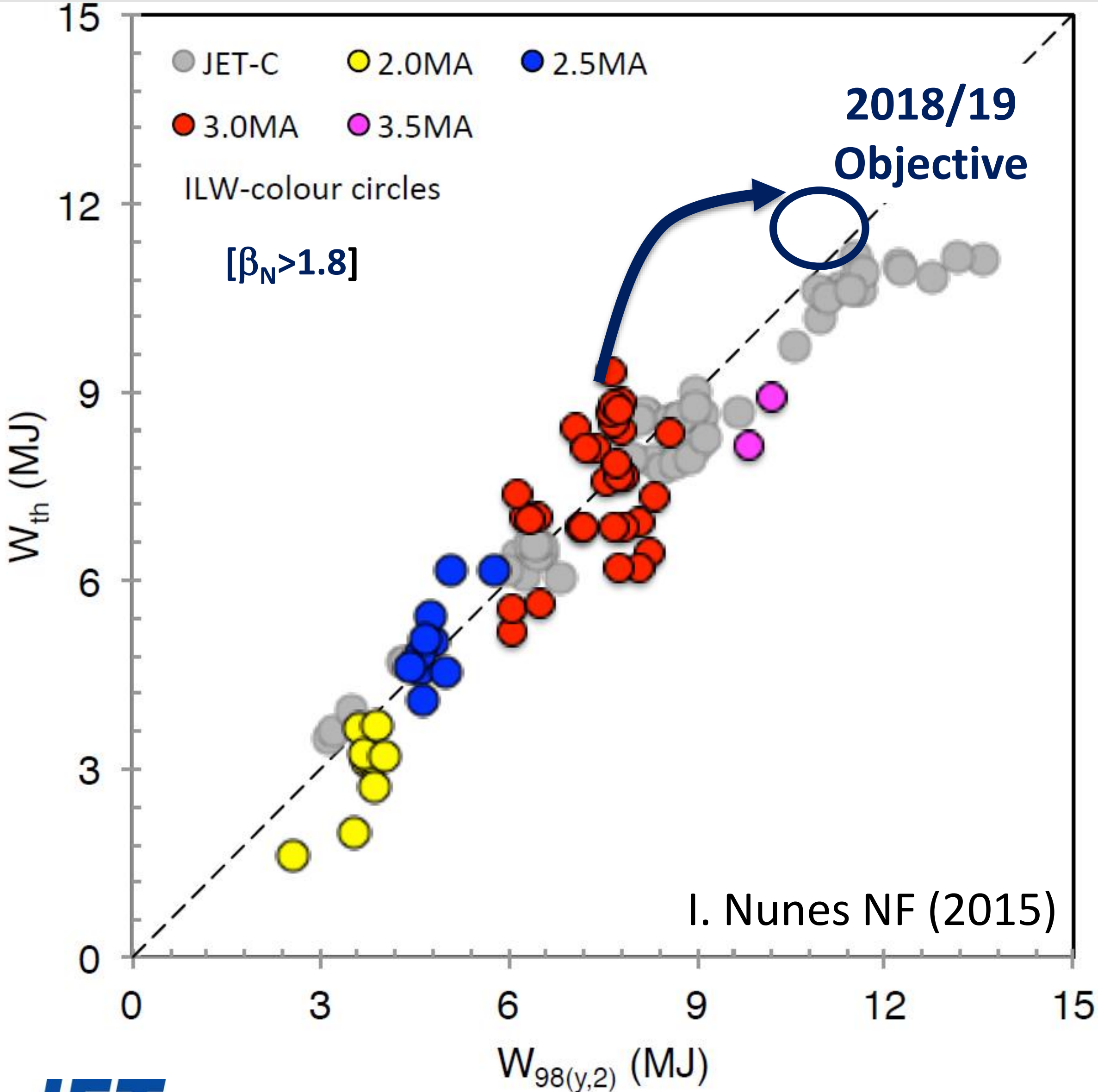


- ITER plasma scenarios demonstrated with ITER-Like Wall
- DT equivalent fusion power is 7~8 MW.

Courtesy of E. Joffrin

JET

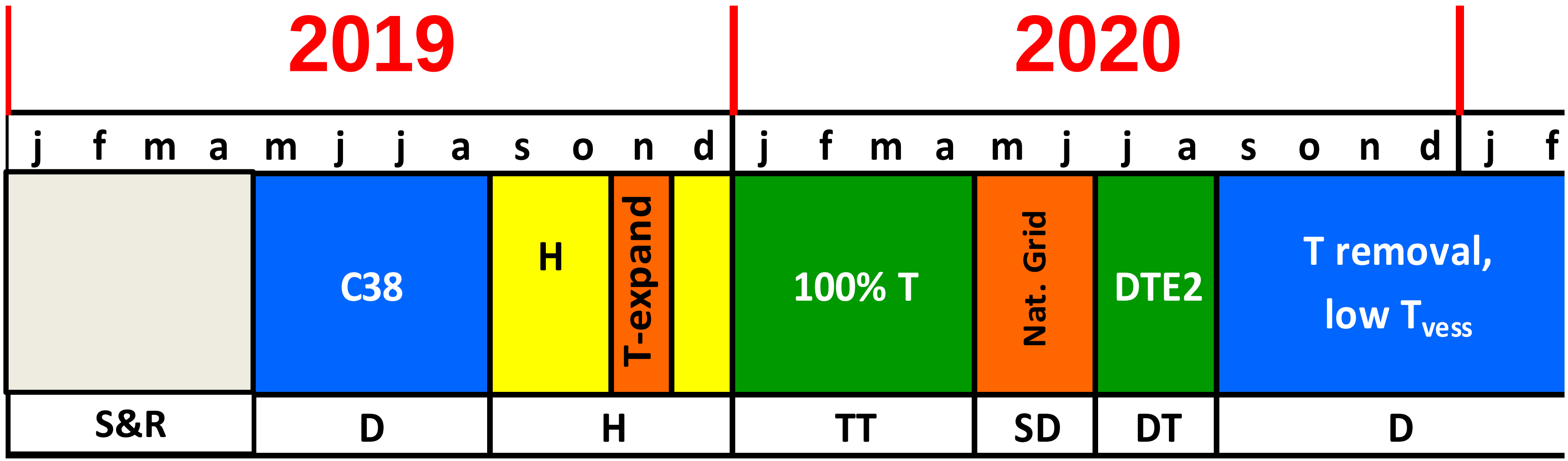
Latest progress: ITER Baseline operation at 30MW 3MA/2.7T



- The remaining challenge is to extend these results to maximum input power (40 MW) and high magnetic field (3.85 T) and high plasma current (4 MA).
- Demonstrate that maximum performance (in D plasmas) is compatible with the ITER-like Wall.

JET

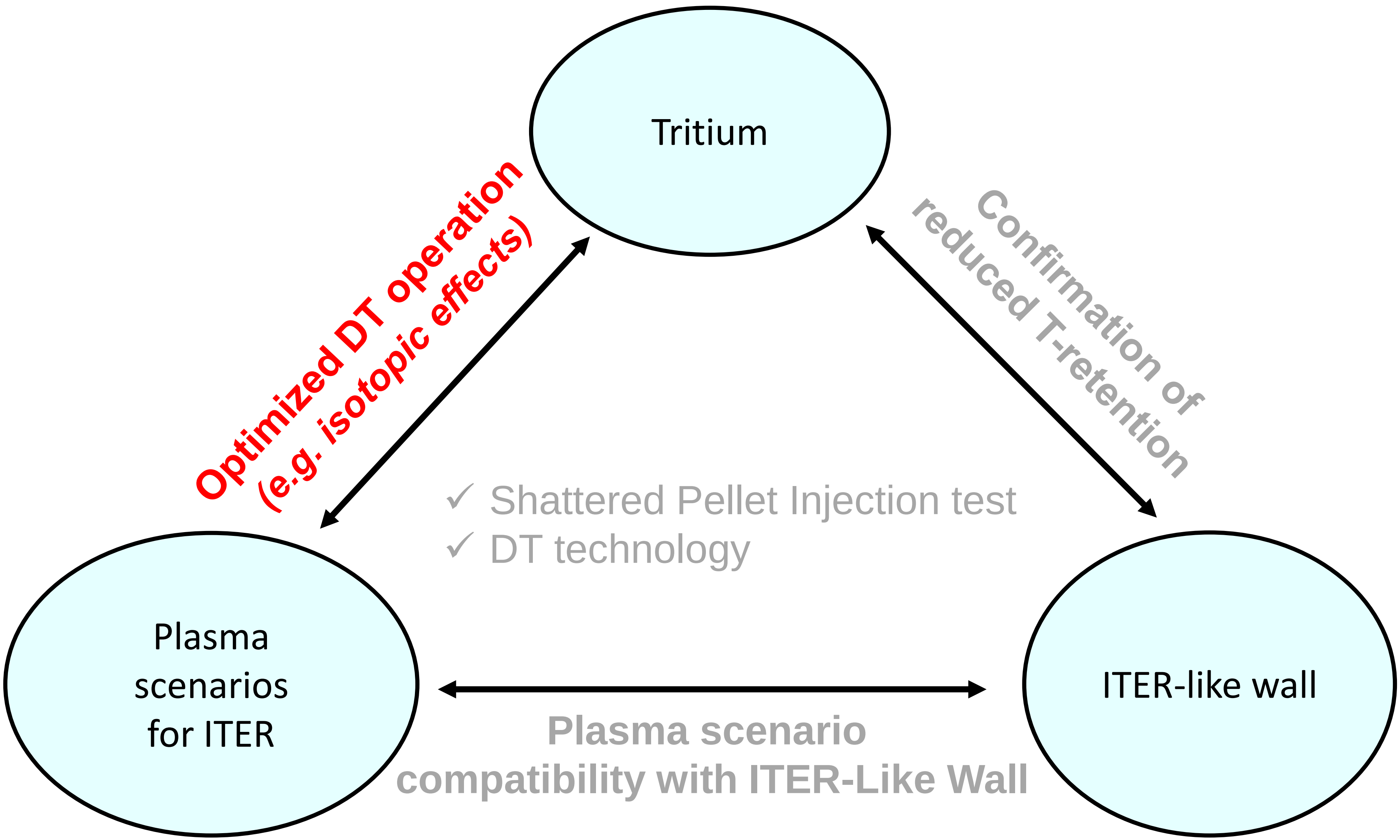
JET campaign plan for 2019 - 2020



1. D experiments with high heating power (~ 40MW) in 2019
2. Isotope studies in H and in T experiments in 2019 - 2020
3. **D-T Experiment 2 (DTE2) in 2020**

*DTE3 in 2024 will be proposed.

JET Programme in Support of ITER

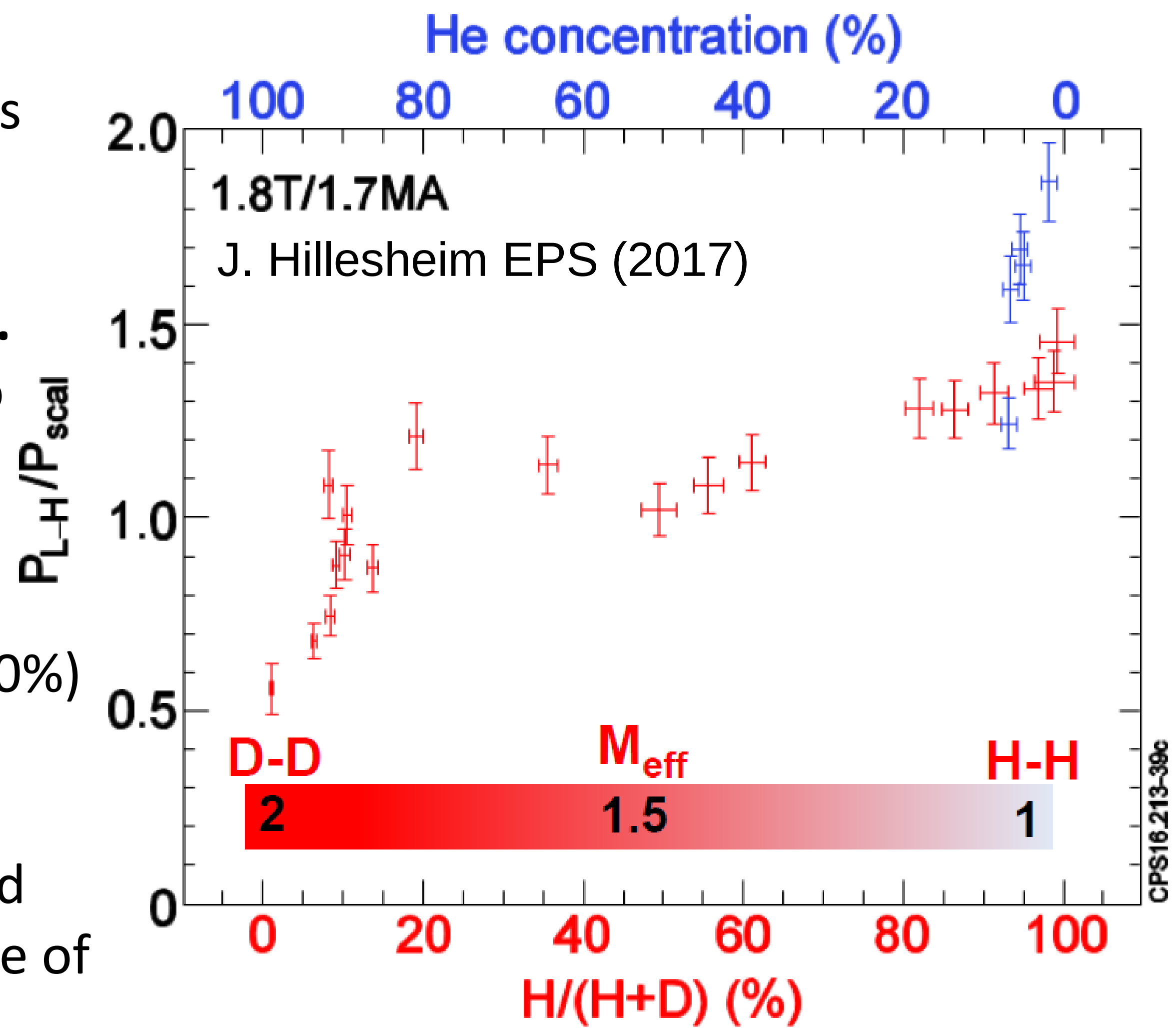


L-H transition threshold power in H and in D

- $P_{L\text{-}H}=P_{aux} + P_{oh} - P_{rad} - dW_{dia}/dt$ at L-H transition
- $P_{scal}= P_{scal}(n_e,B_t,S)$, known as Martin Scaling

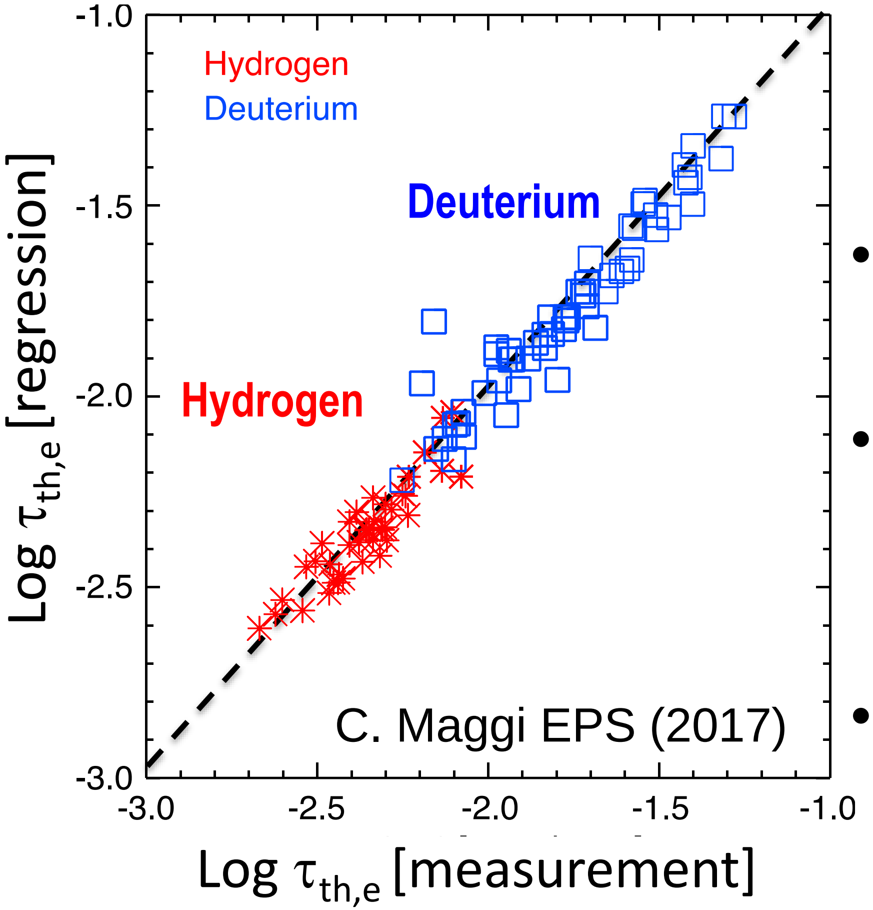
- **$P_{L\text{-}H}$ is lower in D than in H.**
- Little variation and close to Martin scaling in range $0.2 < H/(H+D) < 0.8$

- Small Helium fraction (5~10%) in H plasma shows clear reduction in $P_{L\text{-}H}$
- Helium effect could be used during the non-active phase of ITER operation



JET

Energy confinement time in H and in D



Max H-NBI power = 10MW

H: 1.0MA/1.0T and 1.4MA/1.7T

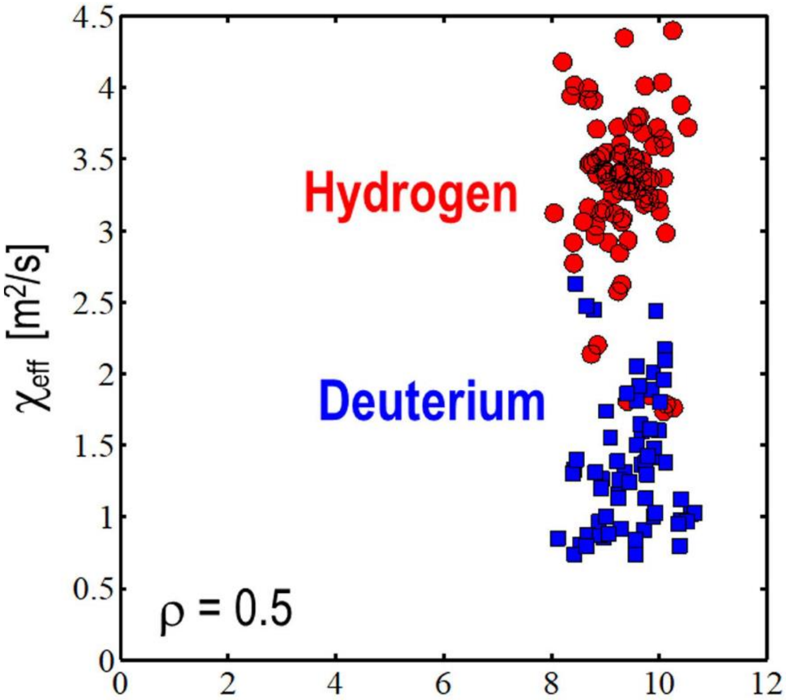
D: 1.0MA/1.0T, 1.4MA/1.7T, 1.7MA/1.7T

- Favourable isotope effect on $\tau_{th,e}$ in type-I ELMy H-modes
- Stronger isotope effect than in ITER Physics Basis published in 1998 ($\tau_{th,IPB98(y,2)} \sim A^{0.2}$)
- Same core confinement, but higher pedestal pressure in D than in H.

$$\tau_{th,e} \sim A^{0.40\pm0.04}\ P_{abs}^{-0.54\pm0.03}\ I_P^{1.48\pm0.17}\ B_T^{-0.19\pm0.09}\ n_e^{-0.09\pm0.10}\ f_{ELM}^{-0.12\pm0.02}$$

Same core transport in H and in D

H: 1.0MA/1.0T and 1.4MA/1.7T .
D: 1.0MA/1.0T, 1.4MA/1.7T, 1.7MA/1.7T
Low plasma triangularity



$$R\frac{\nabla T_i}{T_i} \approx R\frac{\nabla T_e}{T_e}$$

C. Maggi PPCF (2018)

Type-I ELMy H-mode

$T_i \approx T_e$ both in H and in D

Same ITG threshold (i.e. $R\frac{\nabla T_i}{T_i}$)

No isotope effects of core ion heat transport

$$\chi_{eff} \approx \frac{\int P_{heat} dV}{n\nabla T}$$

χ_{eff} in H > χ_{eff} in D, but it is just because

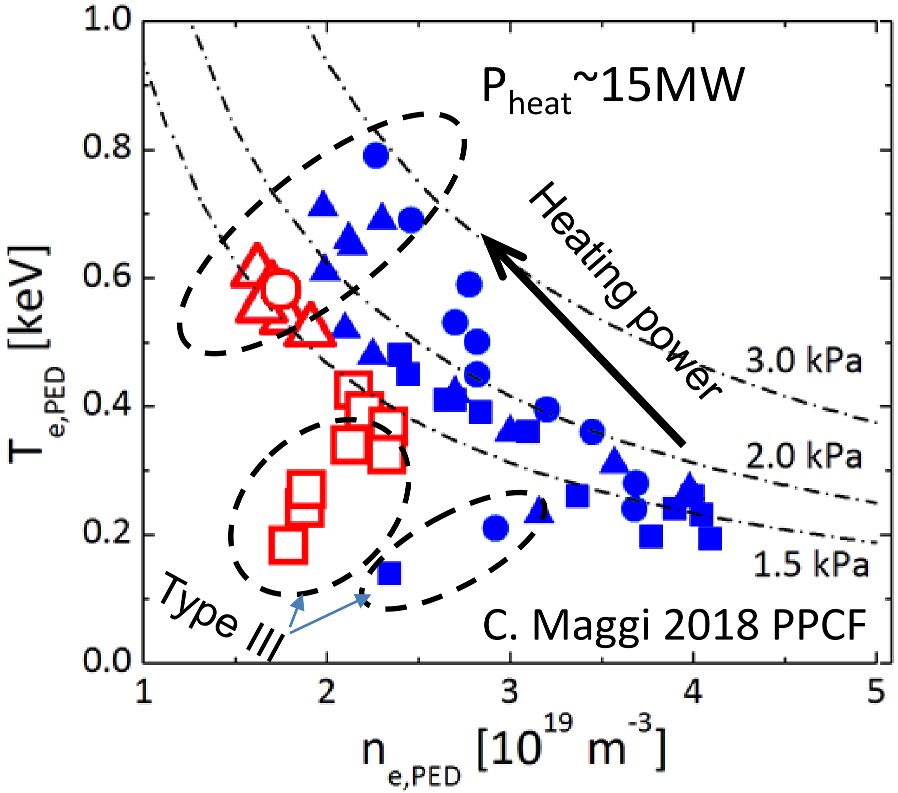
$\int_{\rho=0}^{\rho=0.5} P_{heat} dV$ in H > $\int_{\rho=0}^{\rho=0.5} P_{heat} dV$ in D

in the database.

(H requires higher heating power

for type-I ELMy H-mode access.)

Pedestal pressure in H and in D

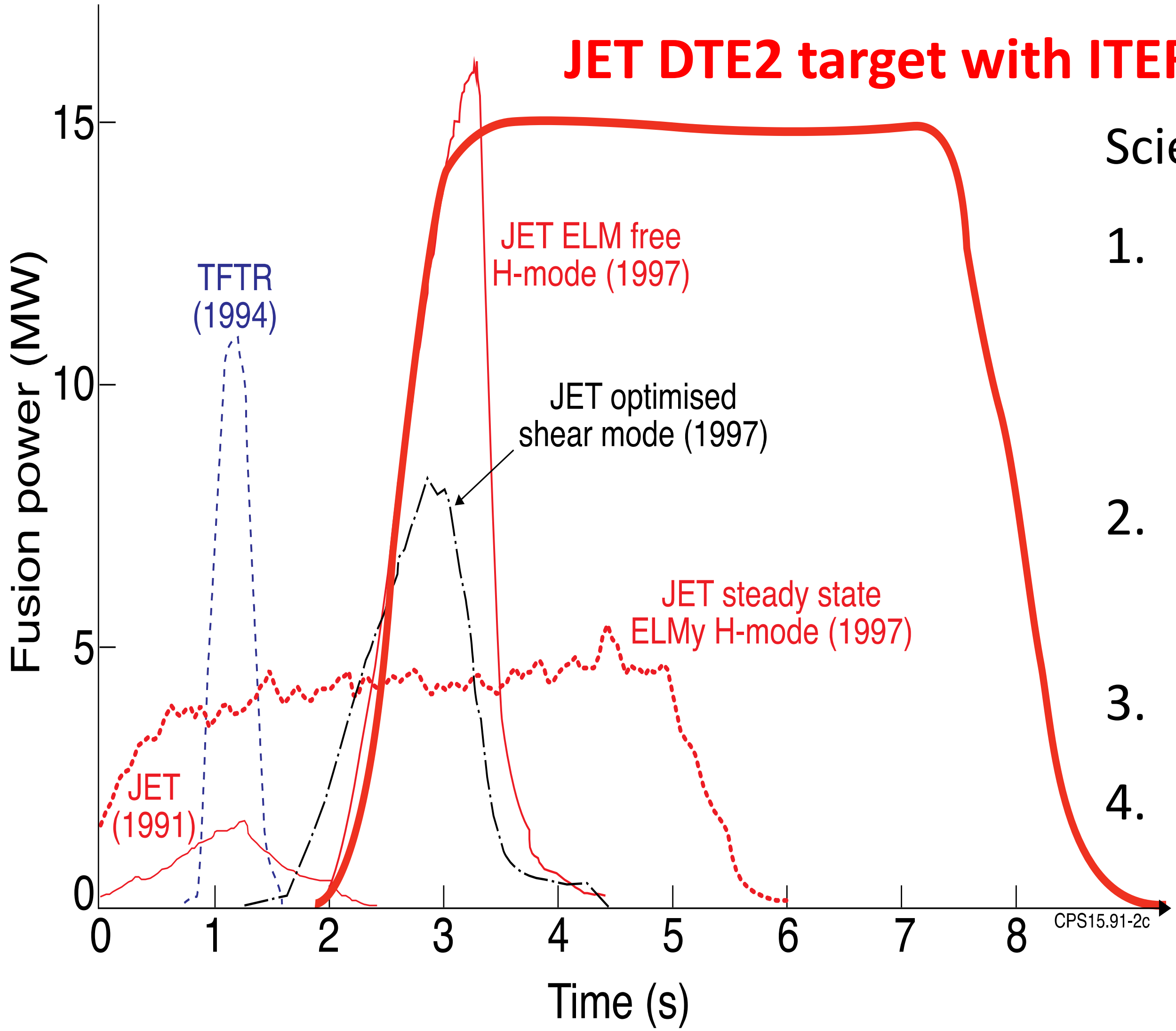


1.4MA/1.7T
Power and gas scans

H	D	Gas puffing rate
○	●	Γ = 3-4 x 10^{21} e/s
△	▲	Γ = 8-9 x 10^{21} e/s
□	■	Γ = 18 x 10^{21} e/s

- Higher $\mathbf{p}_{\mathbf{e,PED}}$ in D than in H at the same heating power
- Low gas puffing increases $\mathbf{p}_{\mathbf{e,PED}}$ in D but not in H.
- Power threshold for type I ELMy H-mode is higher in H.

Objectives of 2020 JET D-T operation: 15MW fusion power for 5 sec stationary state



Scientific objectives in DTE2

1. Demonstration of ITER DT scenarios with ITER-Like wall
2. Isotope effects in DT mixture
3. α-particle physics
4. Fusion technology e.g. T-cycle, Neutronics , Remote-handling, etc

JET

Extrapolation to full heating power (~40MW)

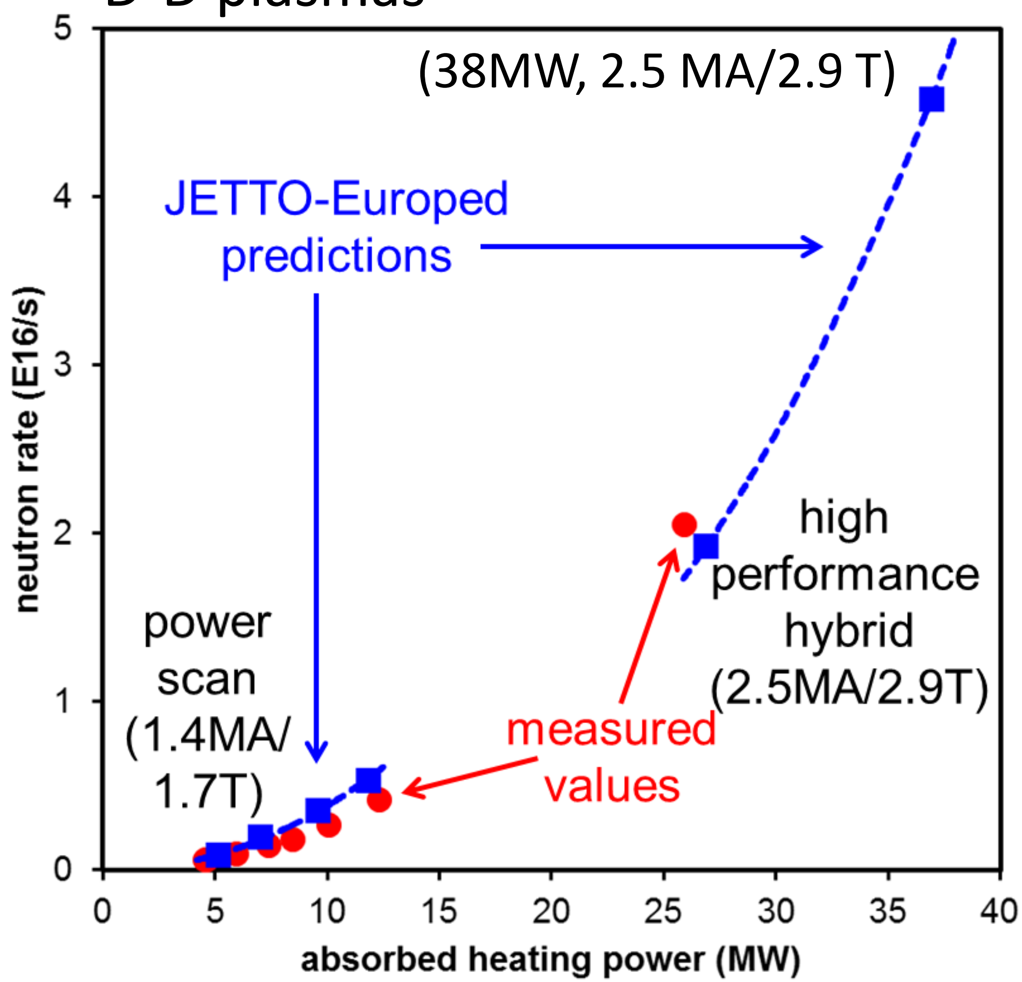
Neutron rates calculated in DD simulations are consistent with measured neutron rates.

- Core prediction: BgB (empirical transport model)
- Pedestal prediction: Europed (physics-based model)

DT equilivalent P_{fus} : ~12.6 MW

- calculated with T_i and n_i profiles predicted in DD simulations at 38 MW heating
- Effects of isotope and alpha particles not included

D-D plasmas



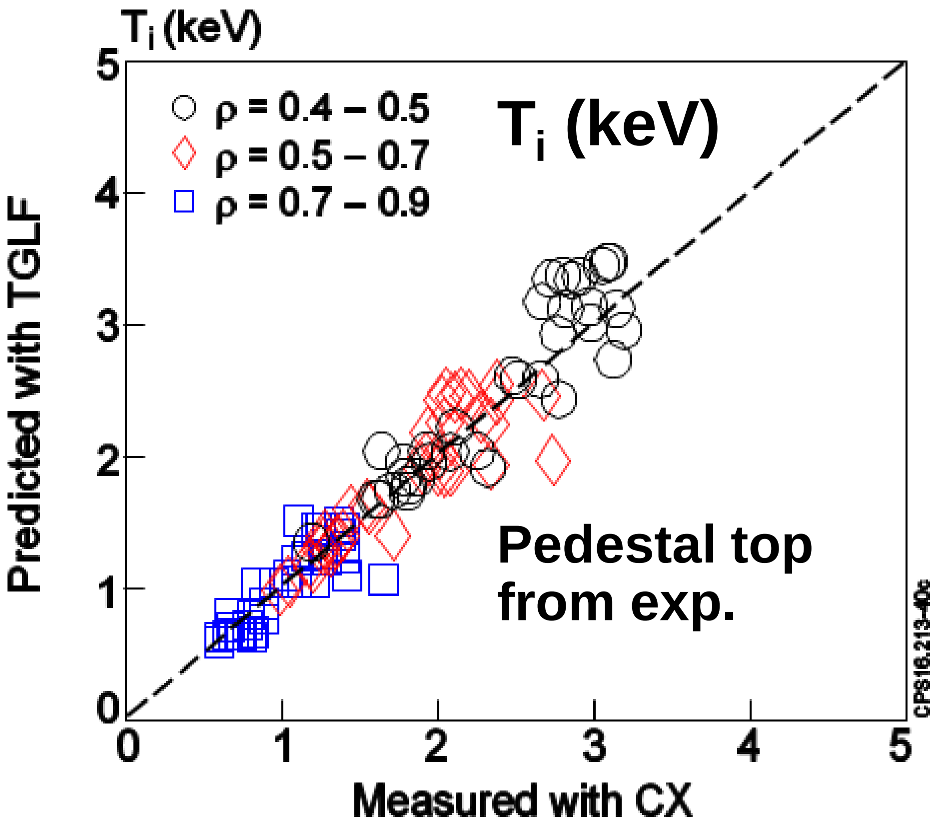
S. Saarelma PPCF (2018)

Prospects for D-T: isotope effect ?

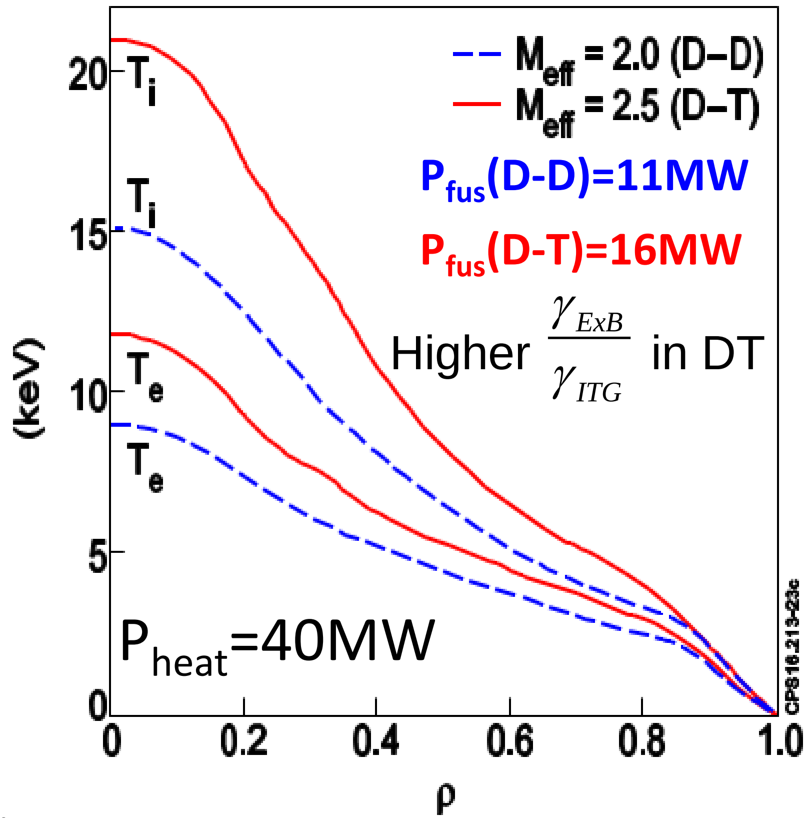
TGLF (physics-based core transport model) predicts significant isotope benefit on performance. To be validated in T-T and D-T experiments

{ Statistical validation of TGLF in D-D
Pedestal pressure assigned

{ Core prediction: TGLF
Pedestal prediction: Cordey scaling



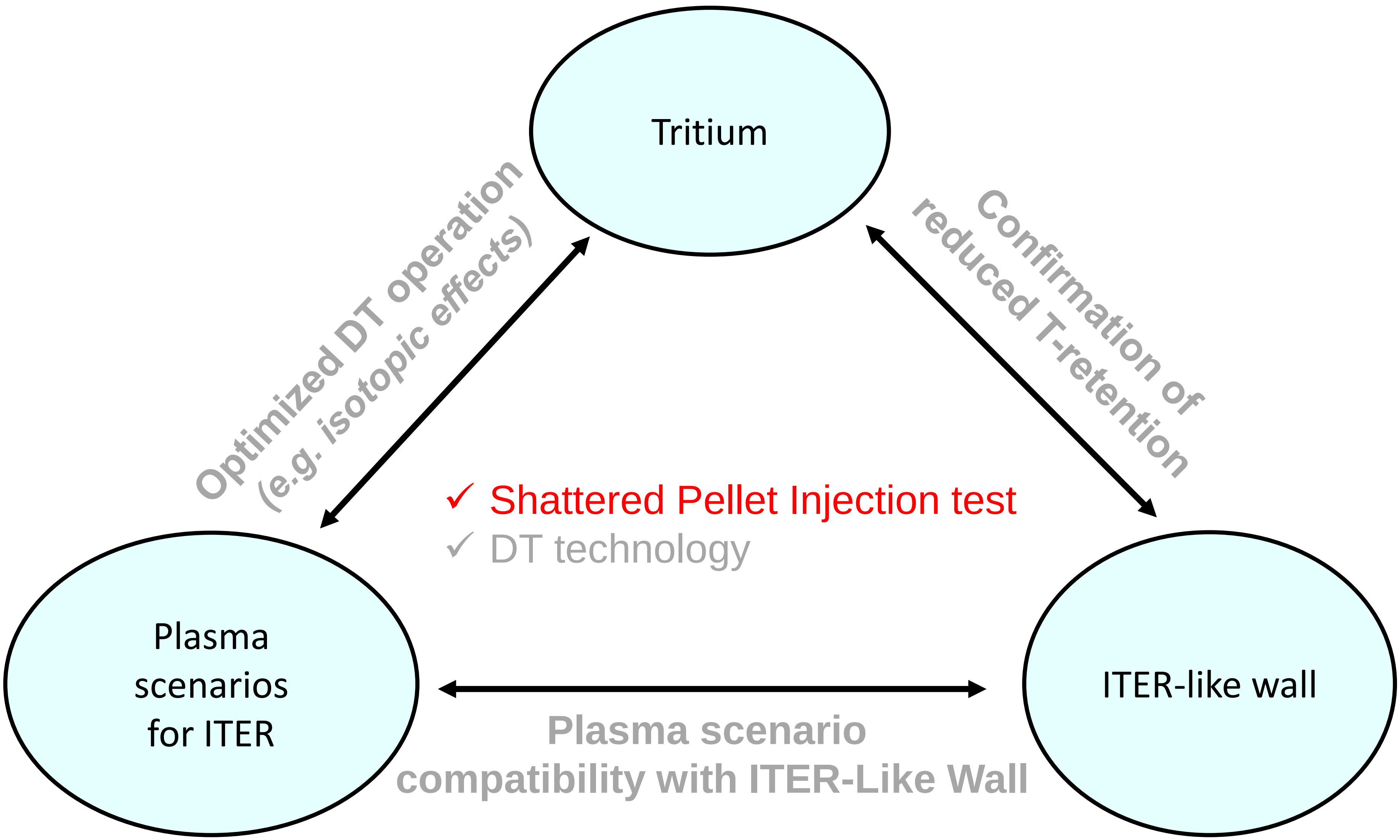
Hyun-Tae Kim et al NF (2017)



J. Garcia et al PPCF (2016)

JET

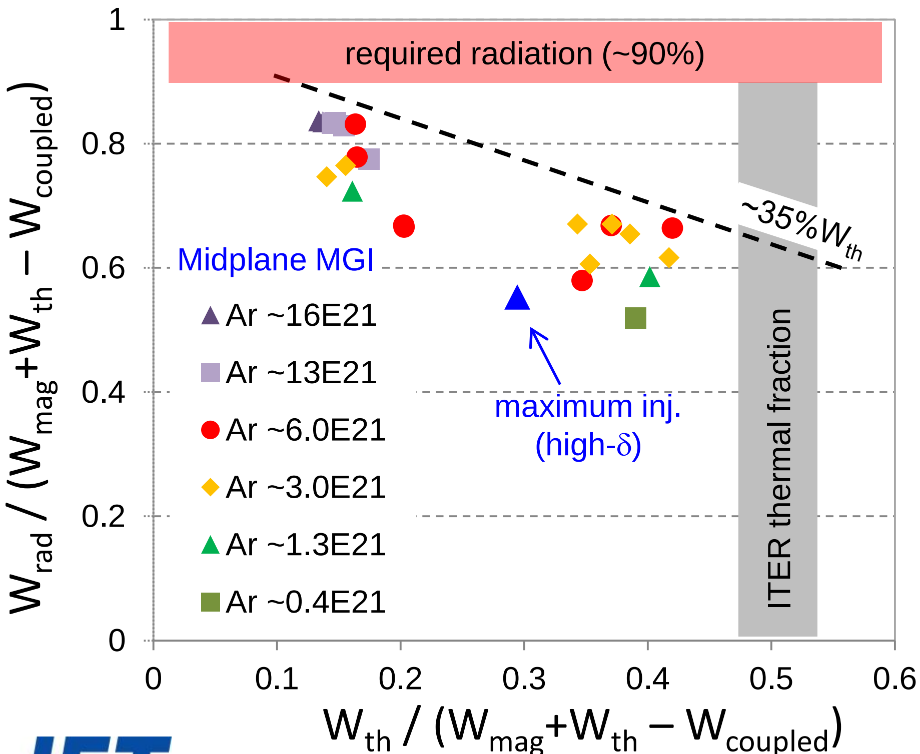
JET Programme in Support of ITER



Disruptions in ITER-Like Wall

Disruption mitigation in JET-ILW :

- Absence of intrinsic impurities (e.g. C) => lower radiation during disruptions
- Slower I_p quench => higher halo currents => larger EM forces on the first wall
- Higher thermal loads => melting Be-tiles
- Three fast Massive Gas Injection (MGI) valves used to mitigate disruption



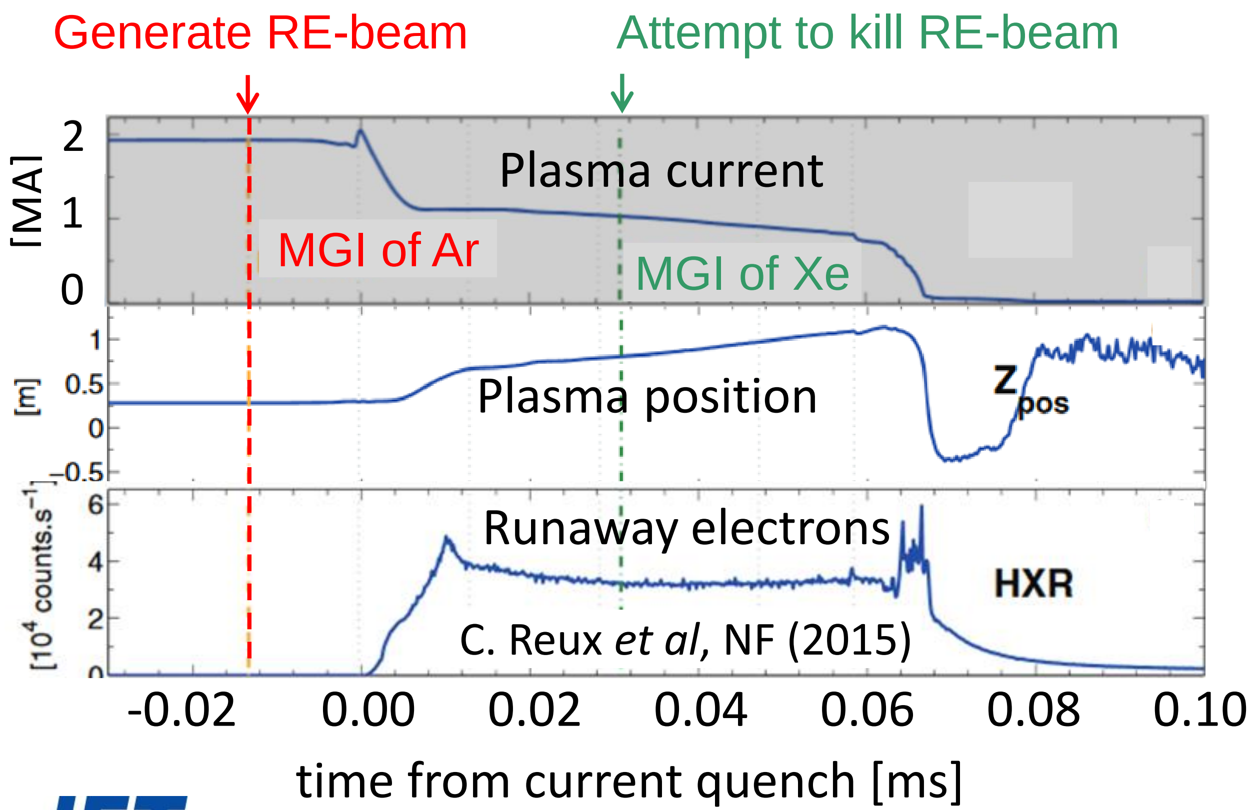
ITER requirements during disruptions:

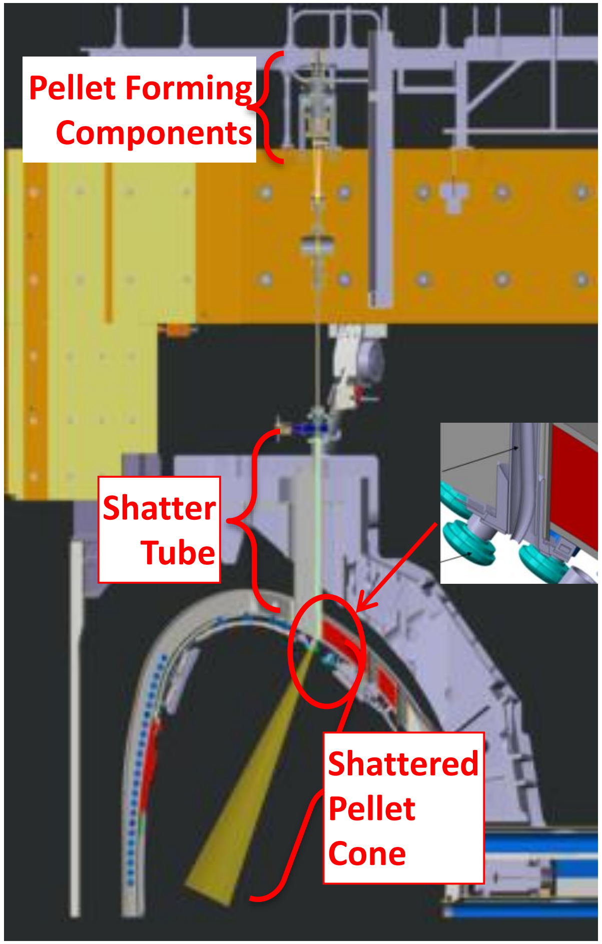
- Thermal load mitigation: 90% of energy needs to be radiated
- Suppression of Runaway Electrons (RE)

S. Jachmich *et al*, PSI, (2016)

Shattered Pellet Injector (SPI)

- MGI is not sufficient to mitigate an existing RE.
- SPI is presently ITER's main strategy for RE suppression
- International project of ITER-IO, US-DOE and EURATOM
- Ne, D_2 and Ar available. Multiple injection possible.
- Experiments to test the efficacy in 2018





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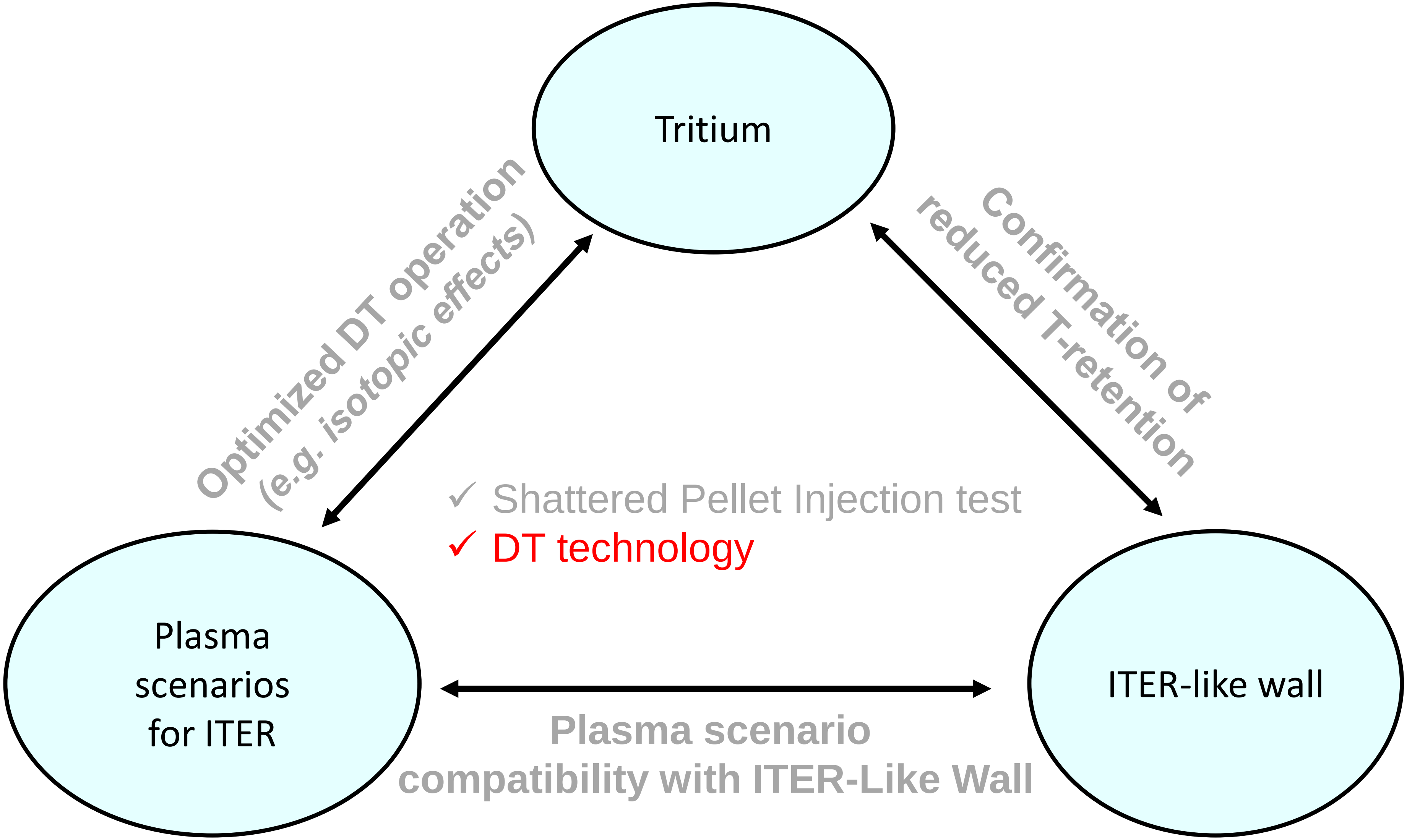
JET Programme in Support of ITER



D-T Technology Programme

T experiments accompanied by an extensive technology programme

- Remote handling of in-vessel equipment
- Beryllium handling
- Management of radioactively activated and T contaminated components including waste processing
- Extensive measurements, simulations, and validation of neutronics and activation codes

14MeV neutrons detector calibrated in 2017

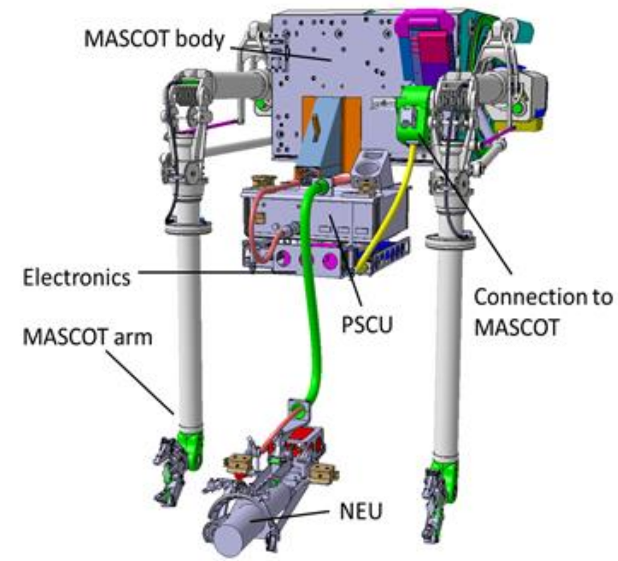
- Needs an accurate calibration to calculate produced fusion power and amount of T burnt
- Calibration procedure envisaged in ITER
- Further details in P. Batistoni, Fus. Eng Design 117, 2017

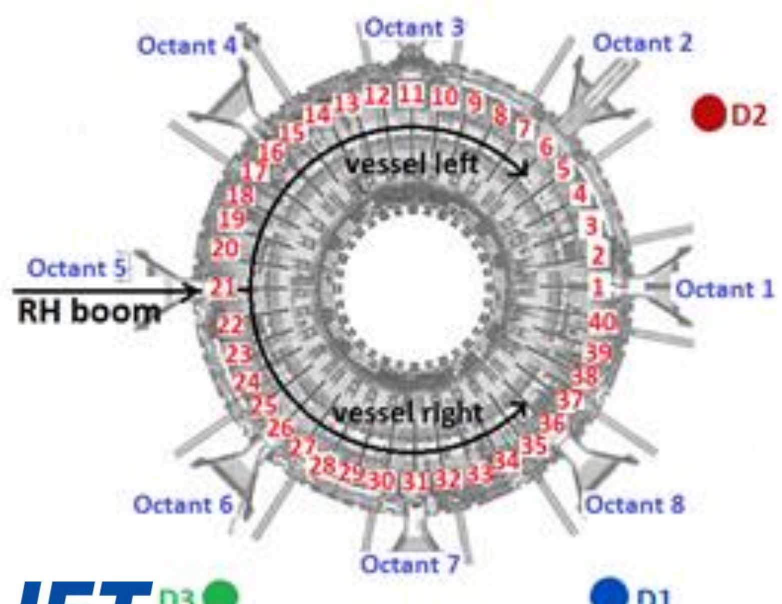
14MeV Neutrons measurements

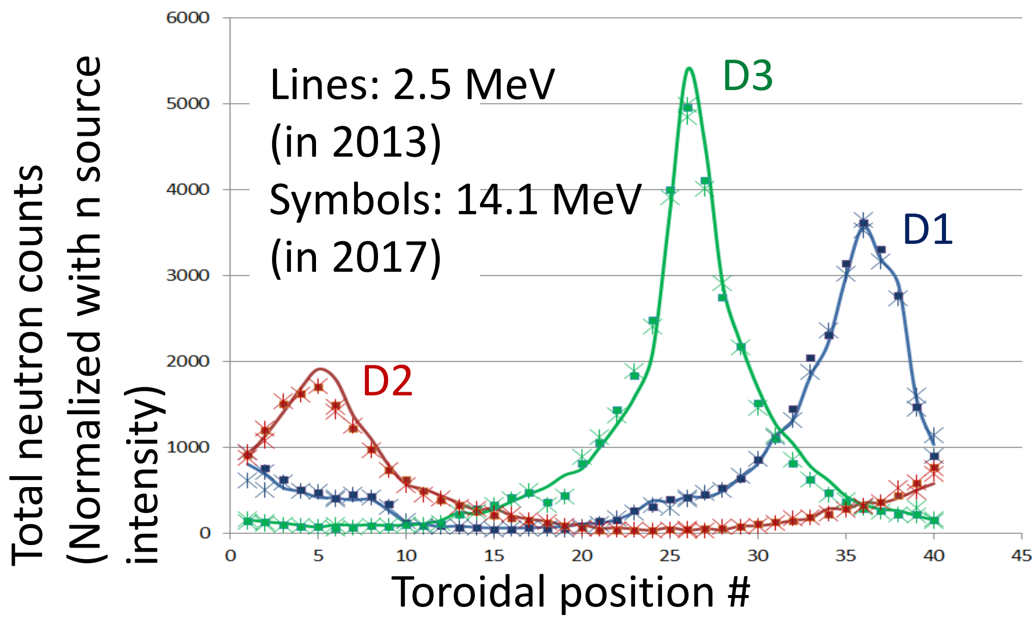
Neutron measurement with ^{235}U Fission chambers at D1, D2, and D3

- 40 toroidal positions of 14MeV Neutron Generator
- 3 radial & 3 vertical positions at each toroidal position
- 277 measurements executed

Fission chamber measurement of 2.5MeV neutrons (in 2013) and 14.1 MeV neutrons (in 2017) are almost identical. **The total uncertainty observed was well within 10%.**







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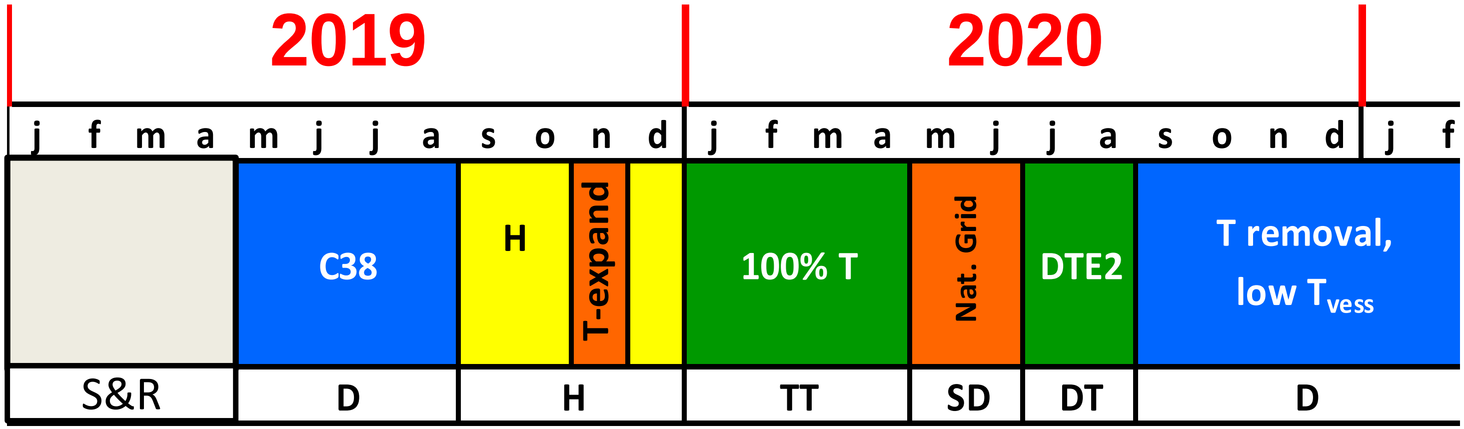
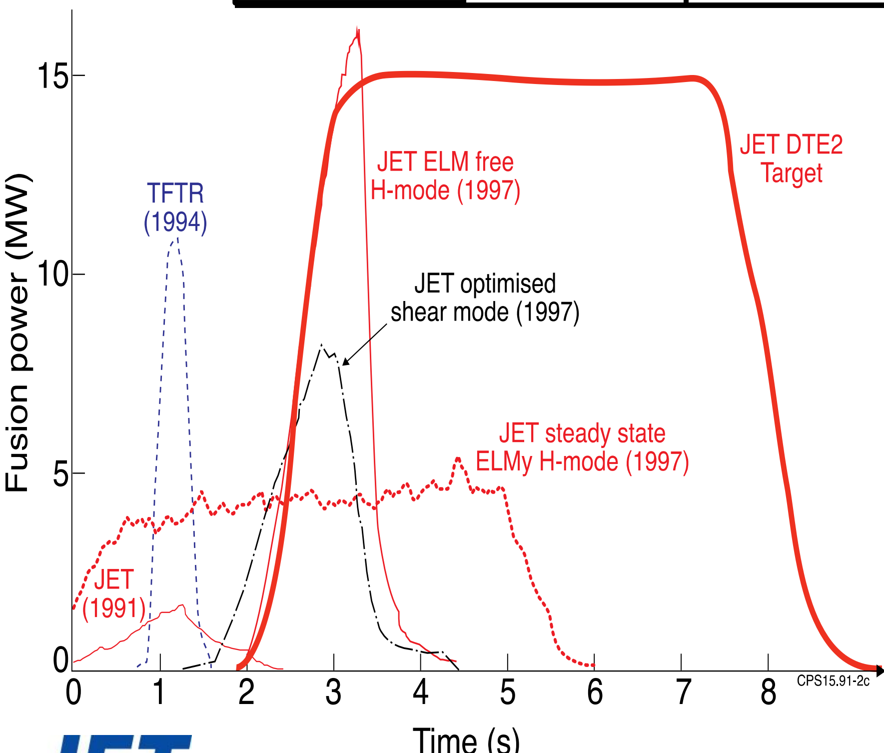
Summary of part 2 – JET for ITER

- ITER's goal: Demonstrate the feasibility of self-sustaining DT fusion plasma with 500MW fusion power for 400sec.
- Preparation in present devices (i.e. optimize operation scenario and mitigate foreseen risks) is essential to ensure achieving ITER's goal.
- EURO*fusion* organizes the European fusion programme along EU Roadmap, and JET is the flagship device in the programme.
- With the unique capabilities of Tritium and ITER-Like Wall, JET research provides the key support for ITER, which are
  - ✓ confirmation of reduced Tritium retention at ITER-Like Wall
  - ✓ plasma scenario compatibility with ITER-Like Wall
  - ✓ optimized DT operation e.g. isotope effects
- Efficacy test of SPI for RE mitigation in 2018
- Extensive DT technology programme also accompanied; successfully completed 14MeV neutron detector calibration using remote handling in 2017

Summary of part 2 – plan at JET





DD campaign with high heating power (~40MW) in 2019, and further exploration on the isotope effects in HH and TT campaigns for 2019~2020

✓ **DT campaign with ITER-like wall in 2020**

Target fusion performance in DTE2:

15MW fusion power for 5 seconds stationary state.

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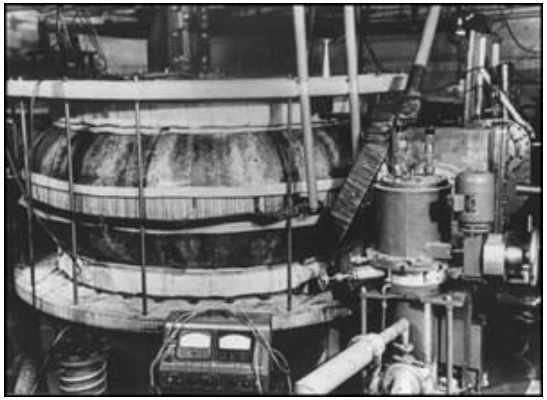
Closing remarks



“Fusion energy will be ready when mankind needs it.”

– Lev Artsimovich, Tokamak pioneer