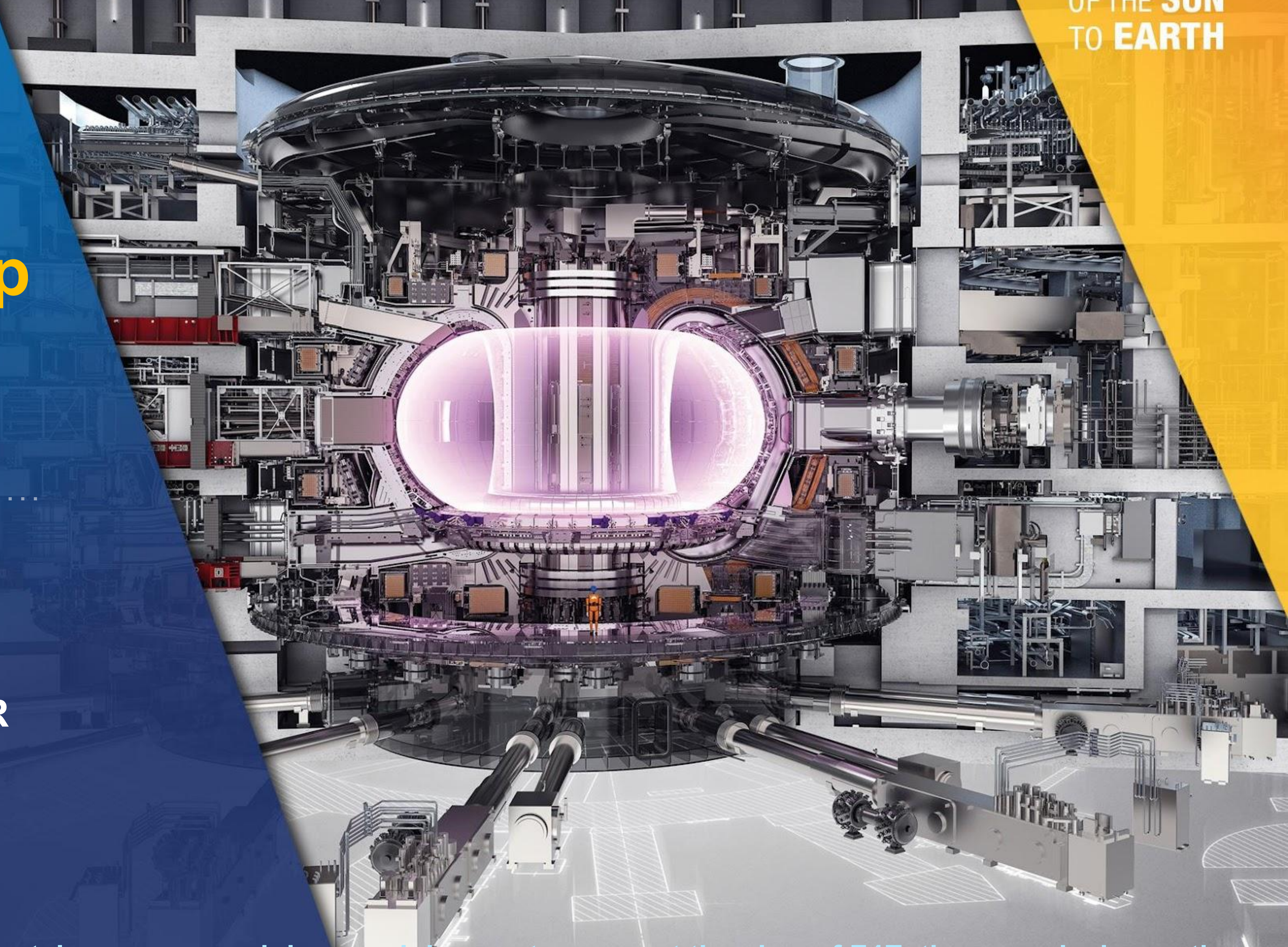
FUSION FOR ENERGY

23 August 2023

BRINGING THE POWER OF THE SUN TO EARTH

ITER, our present step to fusion energy



Gabriella Saibene

With many thanks to A Loarte, R Pitts, N Casal (IO), A Portone, C Harghel , R Sartori (F4E)

This talk contains my own opinions and does not represent the view of F4E, the commission or the IO.

Outline of the talk

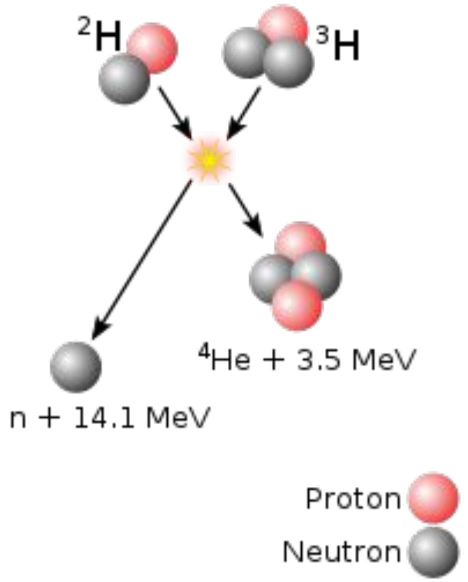
- The ITER project basis, mission and design
- Physics and Technology: the plasma-wall interaction example
- A sneak peak of the ITER site in Cadarache
- Summary

The ITER project basis, mission and design

Fusion on Earth

Fusion reactions between heavier hydrogen atoms are much faster than p-p reactions in the Sun

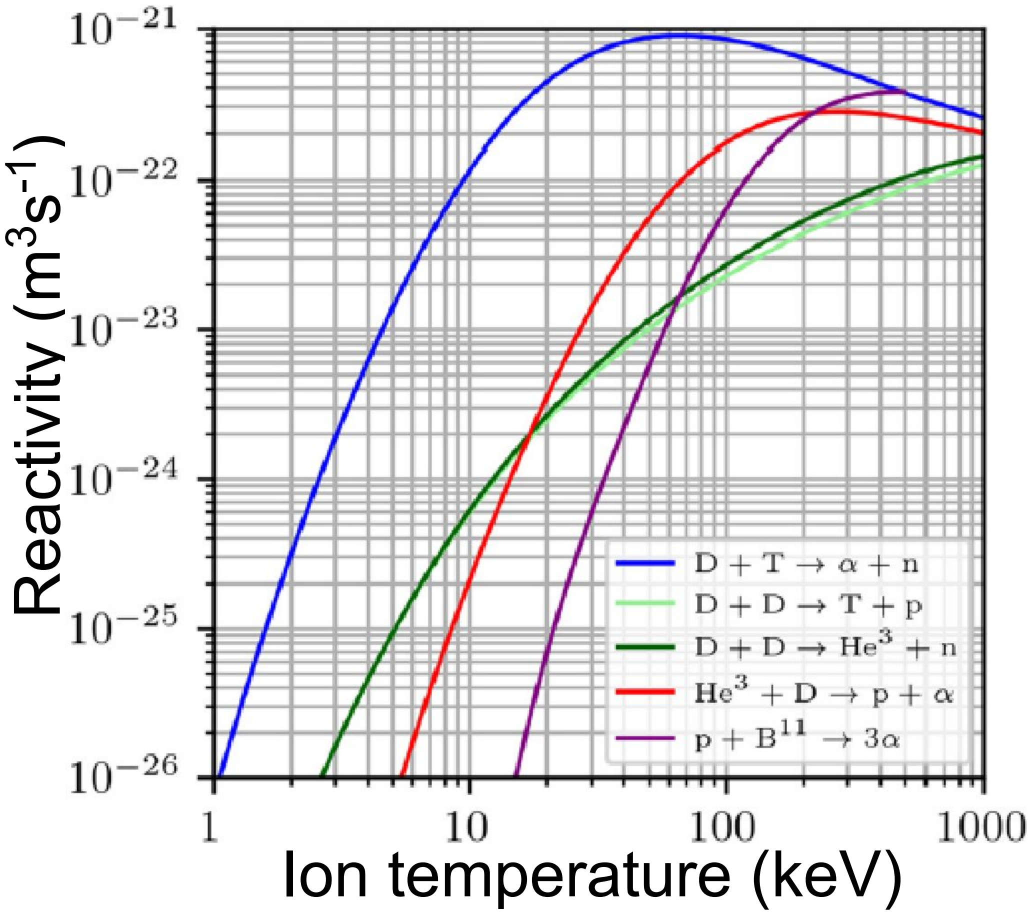
D – T reaction has the highest cross-sections at the lowest energy 10 – 30 keV can be achieved in in man-made fusion reactors (such as ITER).



The mass at rest of the fusion product (He^4) is less that the sum of the mass at rest of D and T

the mass deficit is converted into energy of

- neutron (14.1 MeV)
- and He^4 (α-particle, 3.5 MeV)



ITER basis: size is essential for achieving confinement

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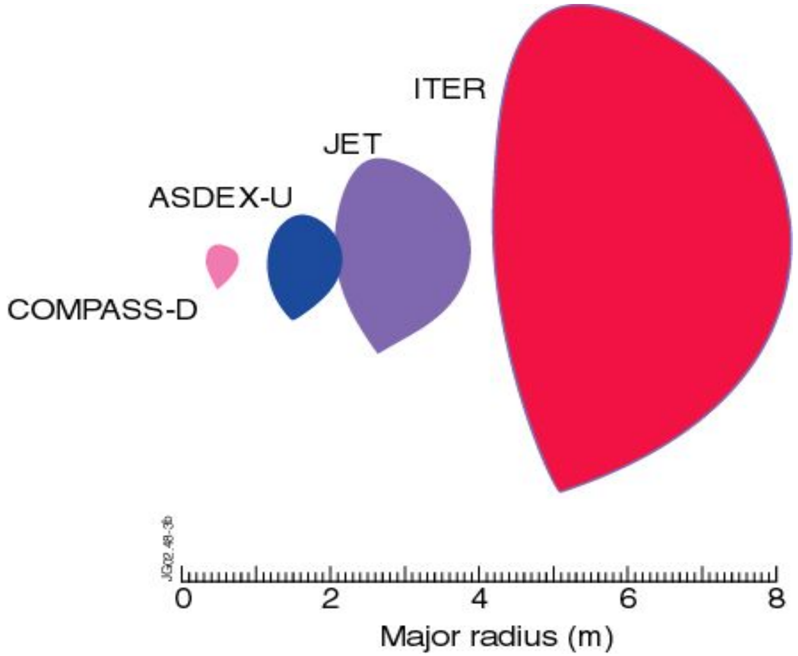
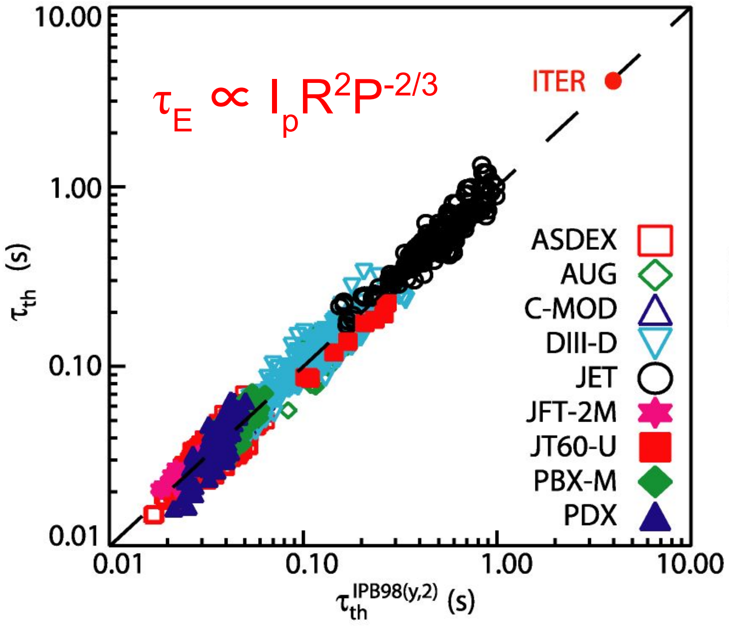
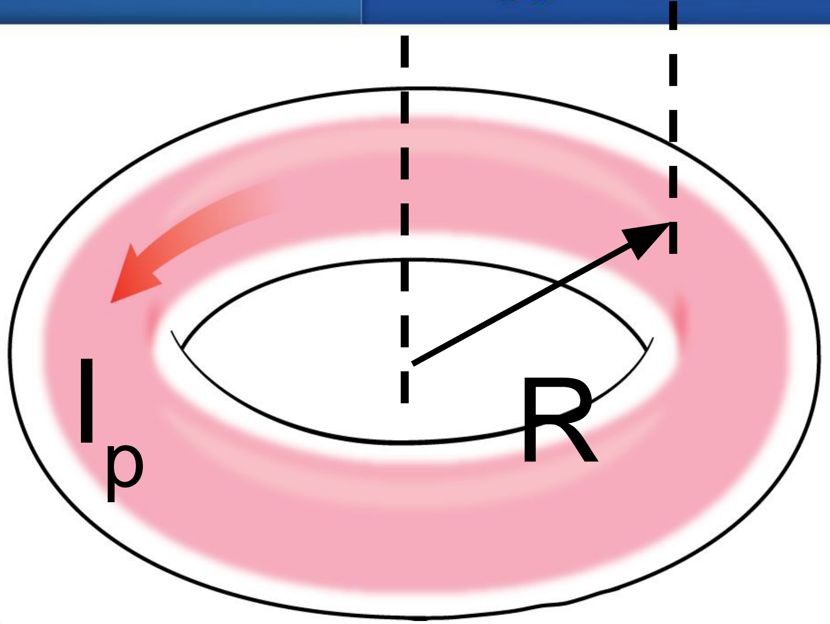
The energy confinement time, τ_E, is defined as

$$W_{plasma}/P_{loss}\ [s]$$

depends very strongly on the size of the tokamak (R), plasma current Ip (and Input power)







The first tokamaks had:

R = 0.6 m, I_p = 40 kA, τ_E ~0.01s

ITER will have:

R = 6.3 m, I_p = 15 MA, τ_E ~3.5 s

ITER basis: DT fusion requires high plasma temperature

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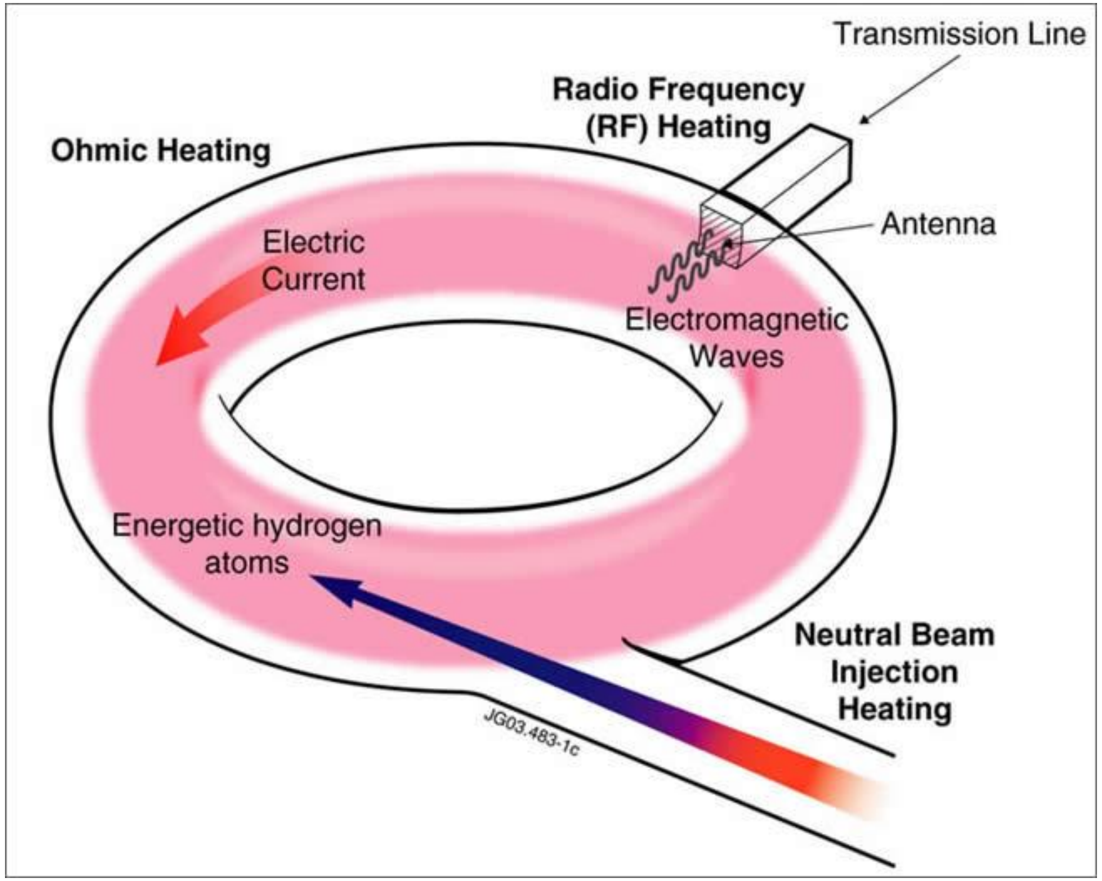
To achieve fusion power production T~ 10 keV + □ heating of plasma is required :

- Ohmic heating = $I_p^2 R_p$; $R_p \sim T^{-3/2}$ □ insufficient
- ITER uses 3 different types of additional heating:
- Radio Frequency Heating (radio and microwave frequencies)
- Injection of energetic H and D atoms (Neutral Beam injection)

For a total of approx. 90 MW auxiliary power



ITER mission plasma dominated by α-heating

 demonstrate the scientific and technological feasibility of fusion power as energy source based on D + T ^{4}He + n (17.6 MeV)

Figure of merit for fusion performance (ignition condition):

$nT\tau_E \geq 3 \times 10^{21}$ keV s m^{-3}

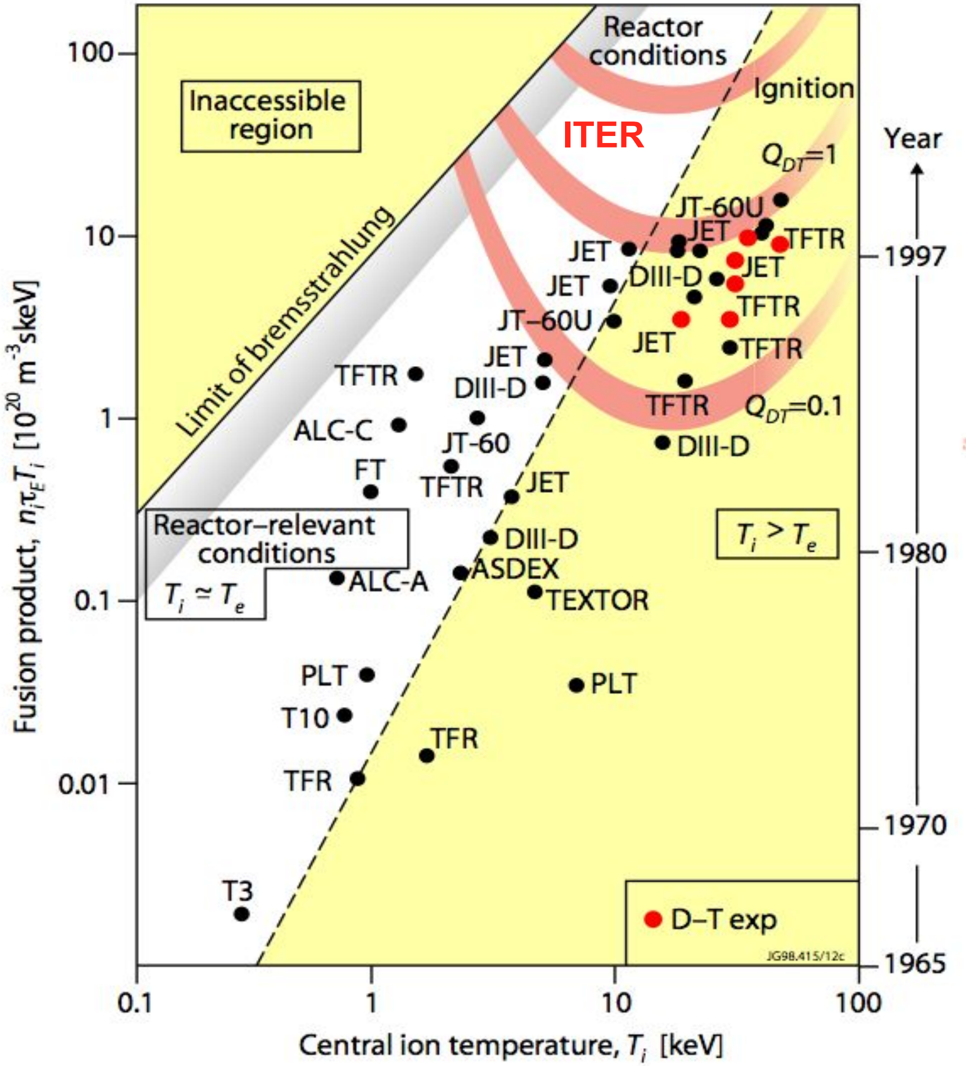
P_{fusion} (^{4}He + n) = $P_\alpha + P_n > P_{external\text{-}heat}$

Fusion gain factor Q = [P_{fusion} (α) + P_{fusion} (n)] / $P_{external\text{-}heat}$

$P_{total\text{-}heat} = P_\alpha (\alpha) + P_{external\text{-}heat}$

$P_\alpha / P_{external\text{-}heat} = Q/5$

ITER will not ignite but will achieve simultaneously values of n, T, τ_E sufficient for high fusion gain (Q=10) plasma dominantly heated by α-particles



Fusion energy production & the Tokamak principle

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To create and sustain a plasma with enough DT particles at high enough temperature/energy

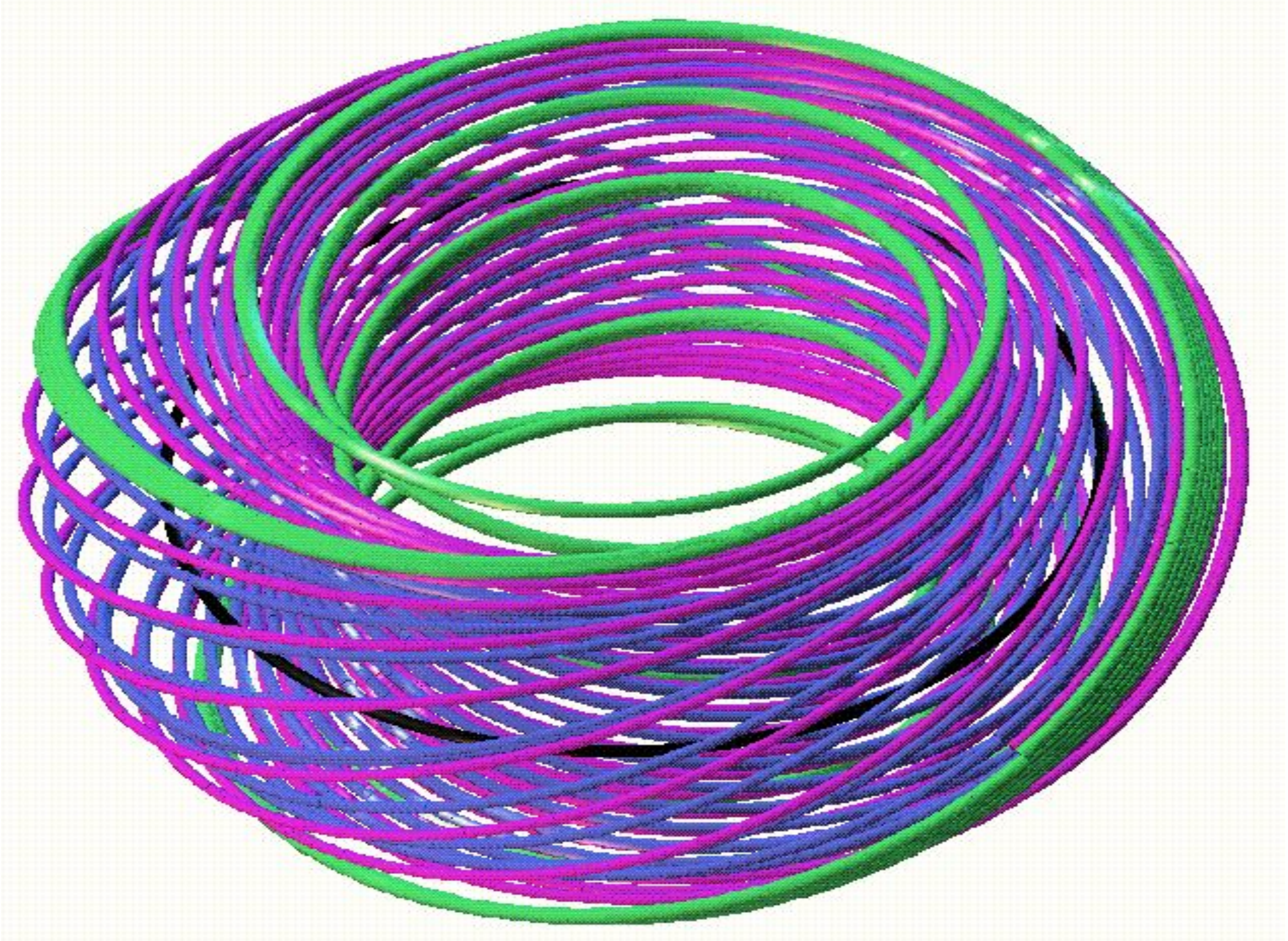
Confine these particles for long enough time to allow fusion reactions to take place

Advanced line of research is the Tokamak

Inject gaseous fuel into a toroidal, high vacuum chamber with a strong toroidal magnetic field (ITER has B_T = 5.3 T)

Induce a toroidal electric field through transformer action, avalanche ionization produces plasma current and poloidal magnetic field

Magnetic fields reduce thermal losses, stabilize the plasma at high pressure, shape the plasma ….



Helicoidal structures – q (safety factor) = n toroidal turns to complete a poloidal turn

ITER magnet system – plasma shape & position control

The magnet system in ITER is more complex then the simple case just shown.

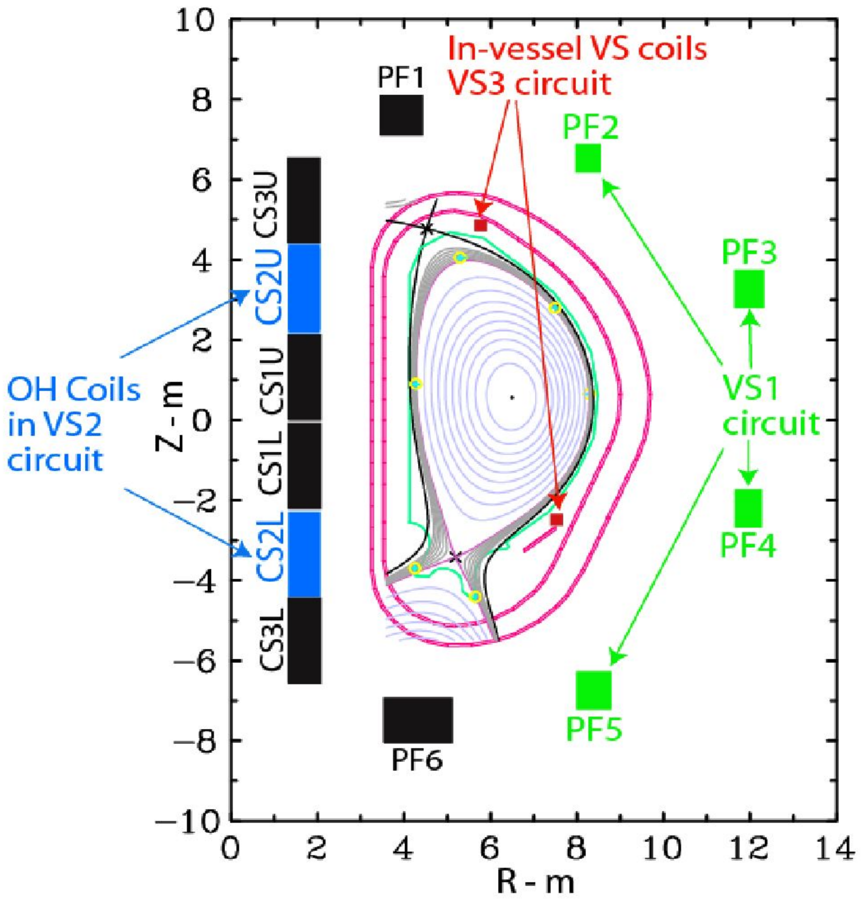
The plasma is elongated and shaped to exploit some technological and physics fact.

TF coils are D shaped (not circular) better mechanical stability + efficient for accommodating the desired plasma shape

Elongation allows to increase the plasma current for the same overall size R of the tokamak (cost!) and magnetic field strength (cost, technology). However elongated plasmas are vertically unstable, so active vertical position stabilization is a must.

The coils circuits allow to shape the plasma, create an x-point configuration (necessary for H-modes), control the divertor geometry (power handling) …

H-mode pedestal properties (such as eight, width, stability, ..) are strongly dependent on the shape of the plasma (triangularity, elongation and squareness) sometimes in subtle ways



ITER "special" features

ITER plasma and the ITER device are essentially different from present-day experiments:

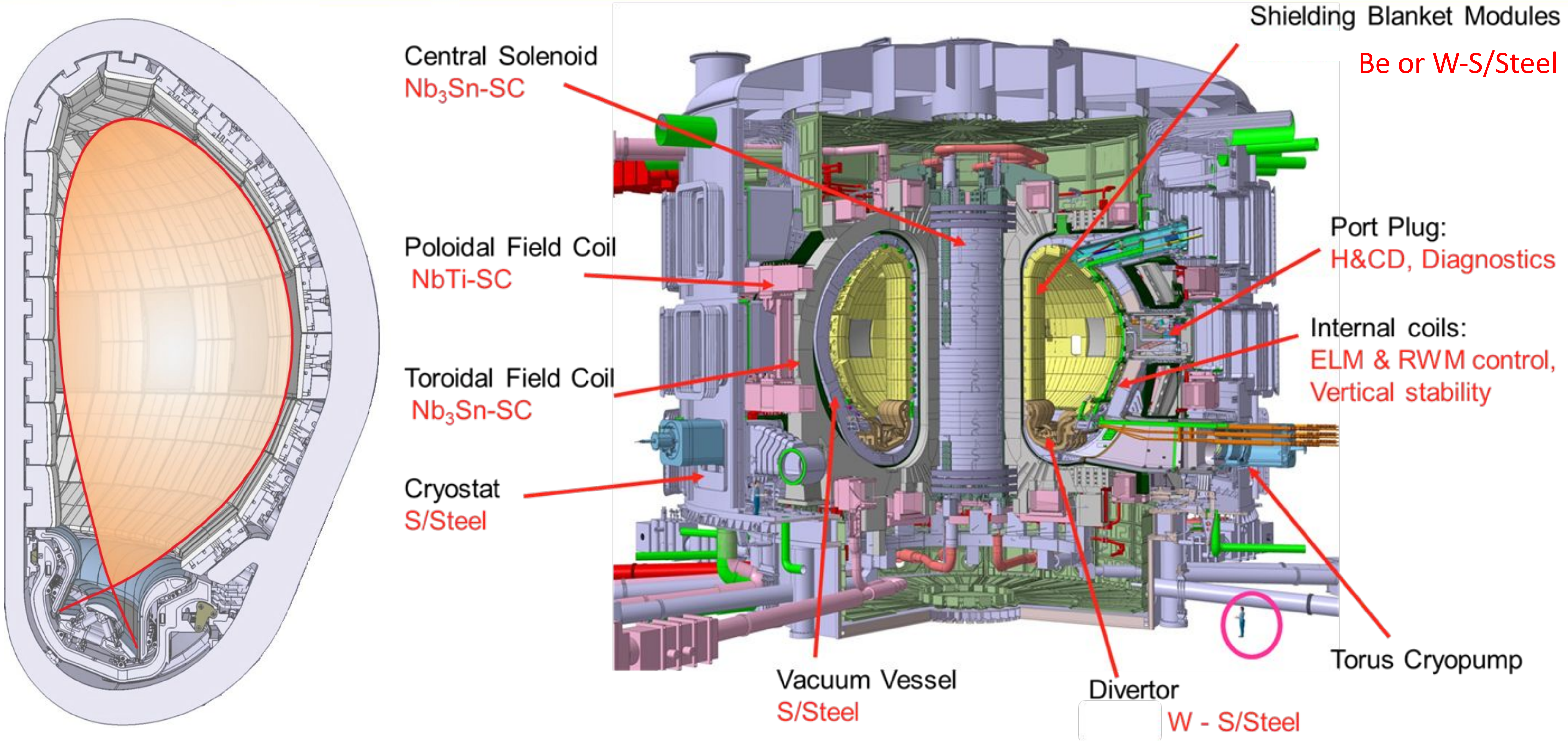
1. Technological challenges:

a. The large size of ITER required for achieving confinement scale of components + very high accuracy
b. The high magnetic fields (5.3T B_T on axis) state-of-the-art superconducting magnets
c. 15 MA plasma current loads on in-vessel components and forces on the vacuum vessel
d. Scale and reliability requirements for heating and current drive systems
e. Regular and extended use of Tritium as fuel – nuclear safety requirements
f. …..

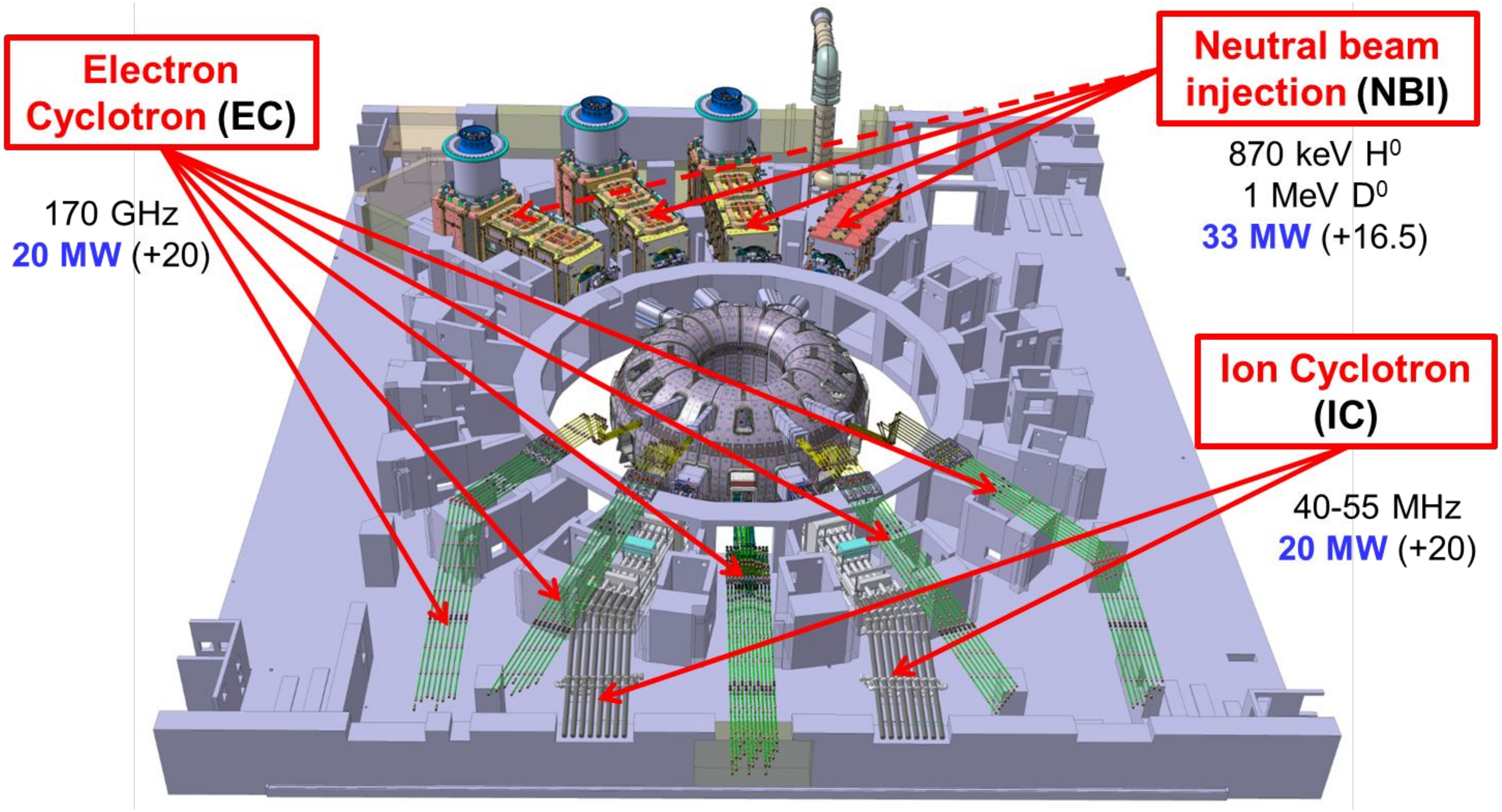
2. Physics challenges (and new knowledge)

a. Plasma behaviour with large energetic ion population (fast ion losses, global plasma stability, effects on turbulent transport, …)
b. Handling of exhaust power from the plasma – walls and divertor
c. Plasma behaviour in long pulses (up to 1h) with high Q and DT mixture control
d. Plasma disruption, generation of run-away electrons and their mitigation and control
e. Mandatory control of edge instabilities (ELMs)
f. …..

ITER Tokamak main components



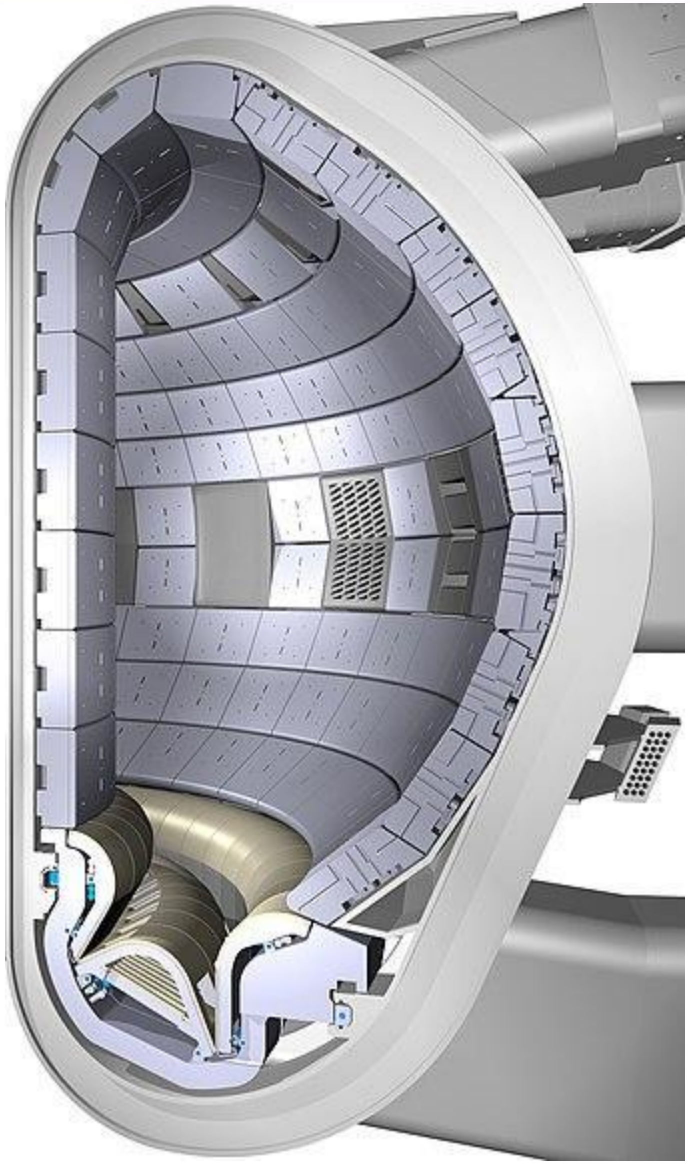
ITER heating and current drive systems (NNB, IC & EC)



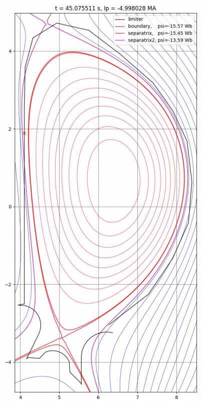
Physics and Technology: the plasma-wall interaction example

Basic starting points

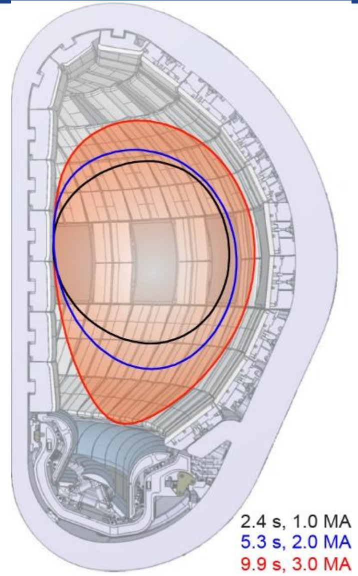
- Functions of the Blanket/Wall/Divertor (PFCs) in ITER:
  1. Protect the vacuum vessel from thermal loads
  2. Neutron shielding for the vacuum vessel, coils and other components
  3. It is the first material surface seen by the plasma (for both limiter and divertor configurations) maintain integrity and minimize plasma contamination
- What is important for function n 3?
  - Power handling in steady state erosion? damage?
  - Power handling during transient events (damage? Type? ..)
  - Compatibility with high-vacuum requirements
  - Tritium retention and recycling
  - Dust production
  - Activation



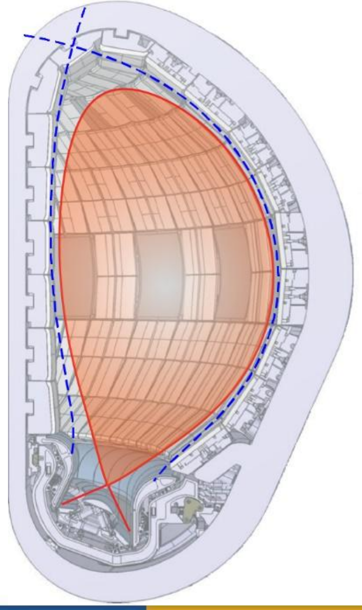
Plasma phases/states and wall requirements – 1



- In normal conditions, the plasma is stable and its magnetic configuration consists of magnetic nested surfaces, closed or open



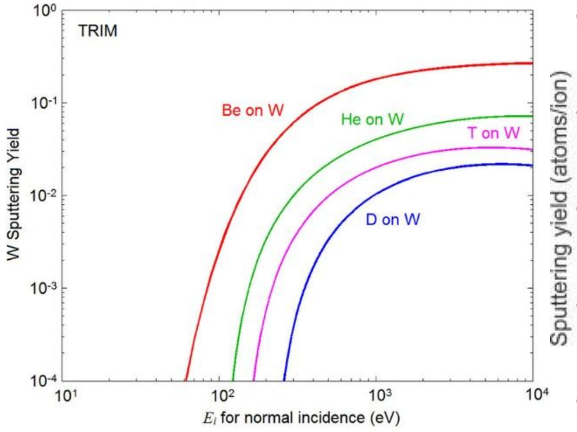
- At the beginning of a plasma pulse, the plasma is in direct contact with the wall
- The magnetic configuration is then changed to x-point to achieve H-mode and high plasma performance (separatrix)
- In steady state, the main mechanisms governing the plasma-wall interaction are:
  1. Radiation (volumetric source) – must be controlled to maintain plasma performance
  2. Heat (and particle) transport along field lines
  3. Charge-exchange neutrals (neutral collisions with more energetic plasma ions)
- N 2 is the main sources of steady-state heat deposition on PFC, and all must be controlled to prevent damage and minimize erosion (plasma contamination and lifetime).

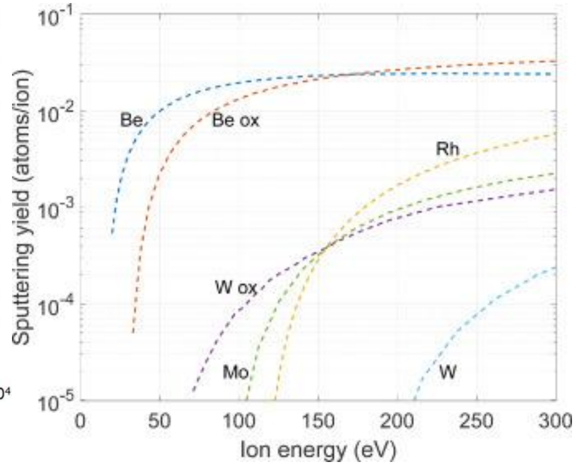


Few words on plasma contamination □ material?

Typical materials used as PFC fusion devices:
- •C (normally CFCs)
- •Molybdenum
- •Beryllium (Z = 4, melting point 1287 °C)
- •Tungsten (Z= 74, melting point 3422 °C)

Physical sputtering depends on target and projectile mass, energy (and target temperature)





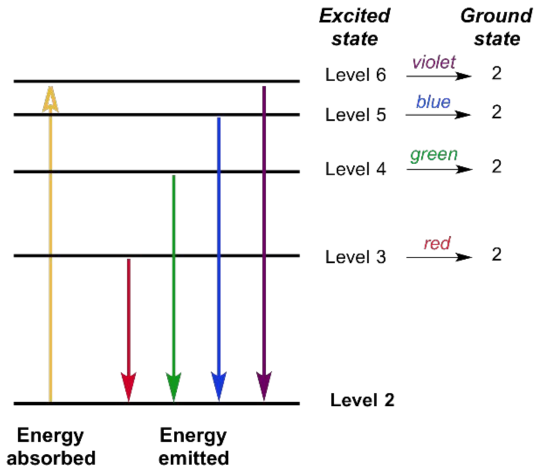
The selection of the PFC material is determined by a balance of the key functions (power handling, T retention, vacuum compatibility, dilution, etc..)

Plasma radiation: is a volumetric source □ loads all PFC

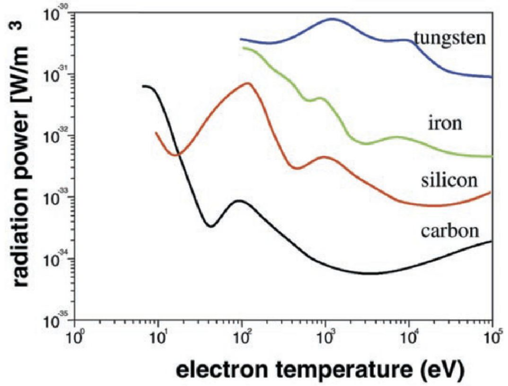
Bremsstrahlung ~ Z^2
Line radiation □

Neutrons

Line radiation origin

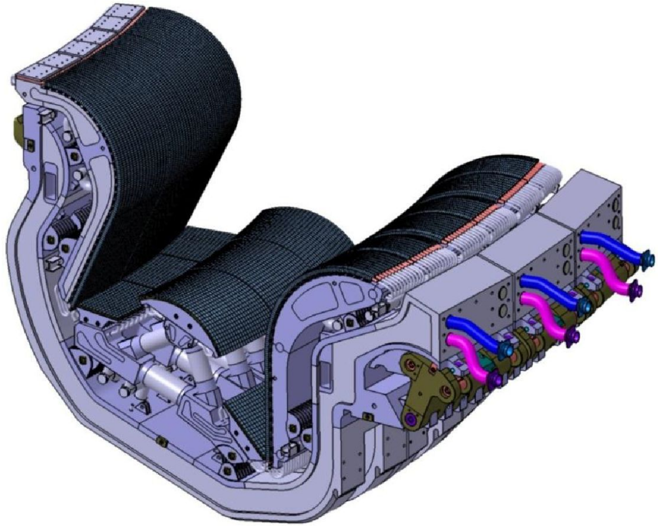


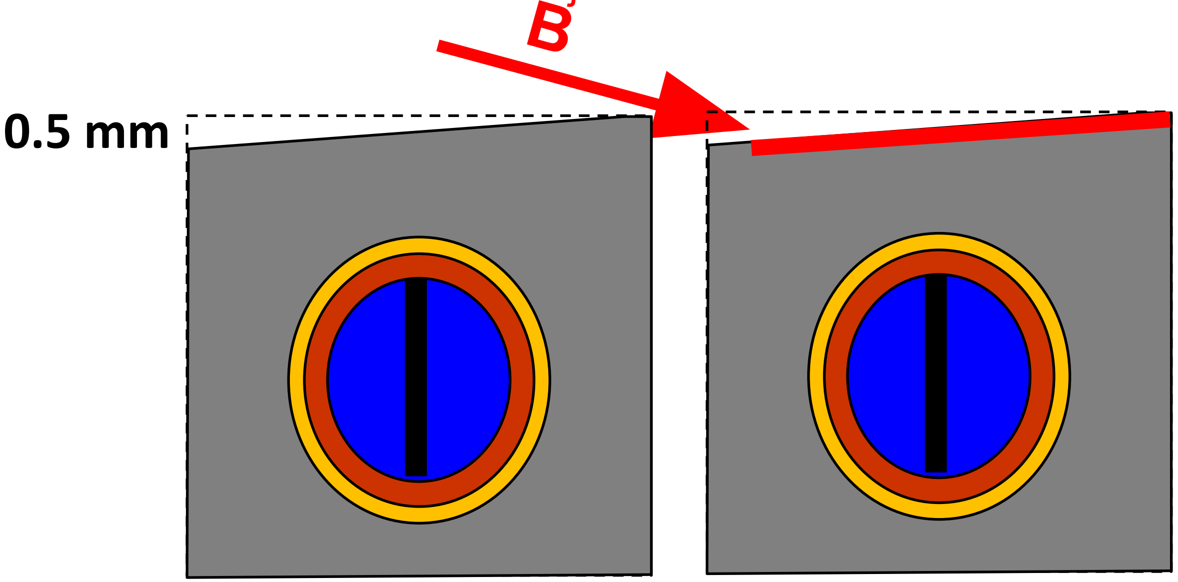
Total radiation power



Coping with heat flux □ alignment & gaps

- Shaping of the PFC is required to prevent or minimize PFC overheating. Failure to do so may lead to deformation, melting, loss of material, changes in the morphology of the PFC material (recrystallization), etc .
- The drive of the shape design are the steady state loads, although edge shading also reduces transient load effect (ELMs).
- The heat deposition patterns are determined by high-resolution line tracing analysis + other codes that can account for near-target effects (such as orbit effects)
- Shaping protects from edge overheating BUT reduces the area for plasma heat deposition □ power flux density goes up
- Melting during transients is not avoided □ ELM mitigation is a must for ITER
- Extensive theory-based modelling and experiments have been carried out to verify if the optical line tracing approach leads to correct results, i.e. that the shaping prescriptions for the ITER PFCs are correct and justified.





Toroidally adjacent divertor monoblocks

Coping with heat flux experiments and simulations

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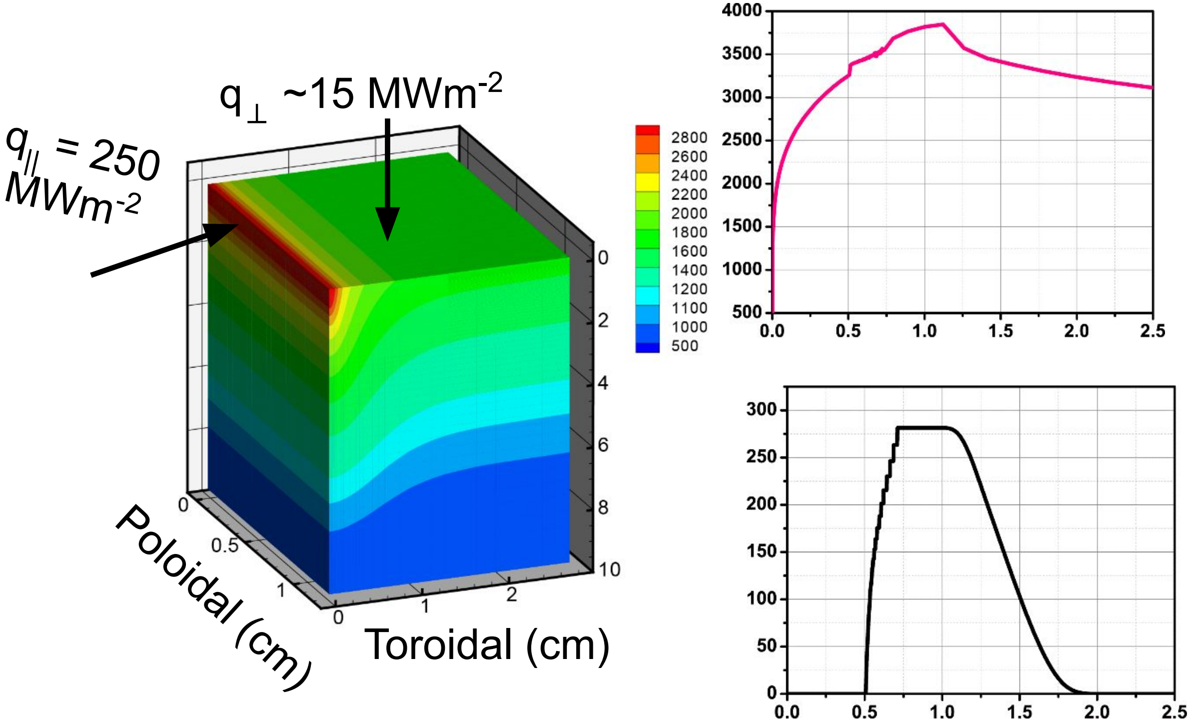
Melt damage observed in JET (W lamella, misaligned).
Multiple ELMs cause melting and melt layers motion.
Modelled with MEMOS code

Simulation of the effects of uncorrected misalignment on a W divertor monoblock in ITER (MEMOS code)



ITER inner wall: same physics and issues

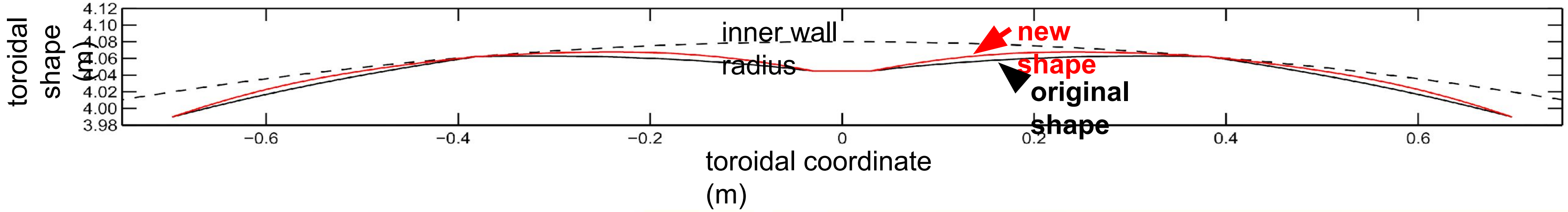
In typical steady-state conditions, the ITER first Wall will experience heat loads << the divertor. The present design values for Be are between 2 and 5 MWm^{-2}, compared to 10s MWm^{-2} for the W divertor.

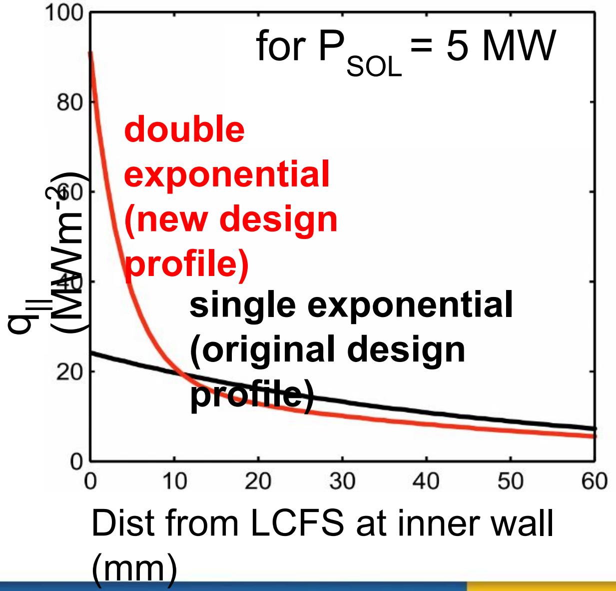
Nonetheless, the same basic principles (and issues) apply, especially the need to hide edges and gaps.

- The FWP need to be shaped toroidally and poloidally, resulting in the present shape
- The curvature of this profile is carefully designed to optimize power spreading for specified λ_q
- The price to pay is a substantial reduction of the "active" area for power handing, to about 10% to 20% o the geometrical FW surface

- New data from experiments indicate the presence of " narrow features" in the power profiles in the SOL the shape of the wall may need changes to cope with steep power decay lengths

Changes to the design may be required/possible if W is used for the wall instead of Be – increased power handling





Transient loads – ELMs and disruptions – 1

The Plasma Facing Components in ITER cannot be designed to handle the thermal loads due to ELMs and disruptions, especially not for the high current, high power/Q scenarios.

Shaping aimed at protecting exposed edges during ELM events is helpful in reducing melt areas and erosion but global power handling cannot be designed in.

What is an ELM?

- ELM = Edge Localized (MHD) Mode
- Instability of the plasma edge (pedestal) in H-modes ELMs are a “disease” of high-performance plasmas
- ELMs expel plasma particles in very short and repetitive bursts
- Most of the power expelled by ELMs ends up in the divertor
- W is of course more resilient to ELMs than Be (hence the choice of divertor armor material)
- For ITER, ELMs have the potential of damaging the PFC for I_p> 7MA
- Effects: Surface melting, dust production, loss of target geometry, …
- ELM suppression is required for ITER Operation (ELM coils)

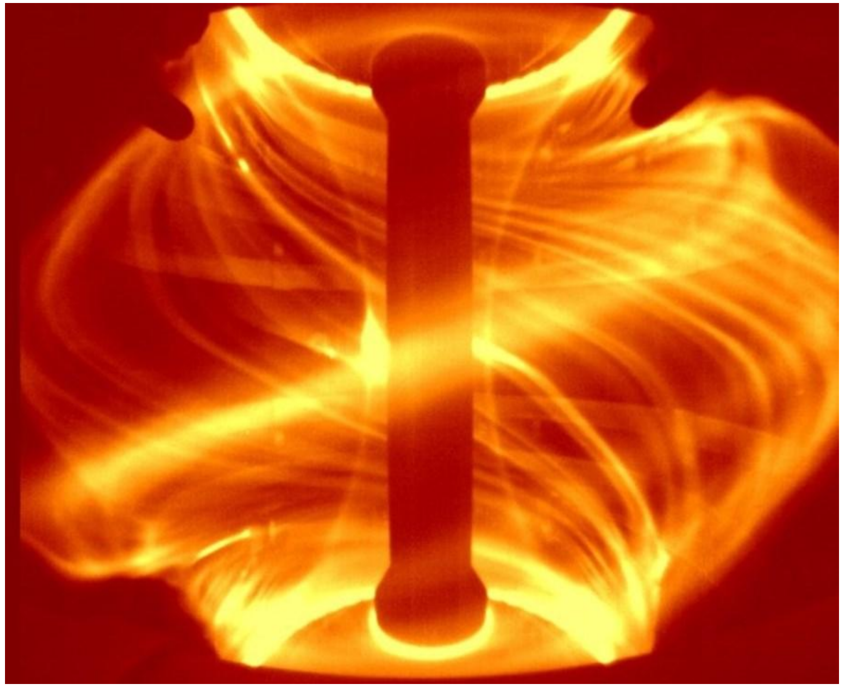


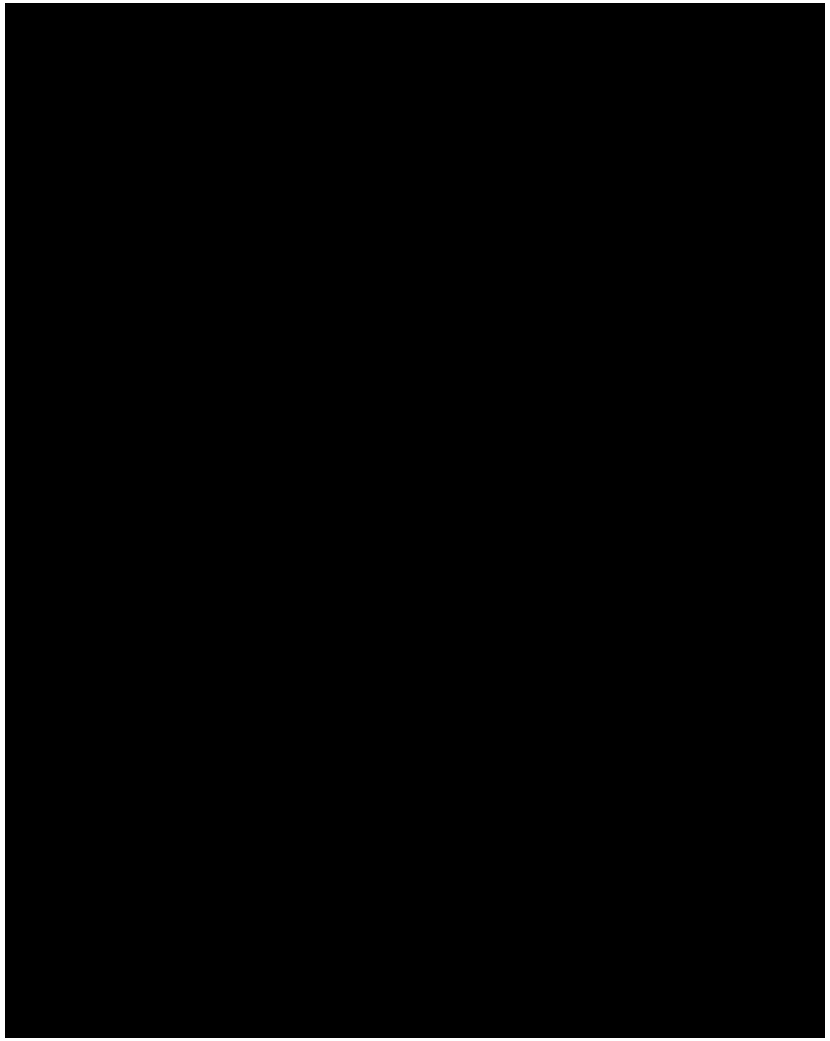
Image from a high-speed video of the MAST plasma at the start of an ELM

An example of disruption from JET

JET Pulse #102336

Deliberate disruption as part of studies of disruption mitigation techniques

This is a very ”benign” disruption

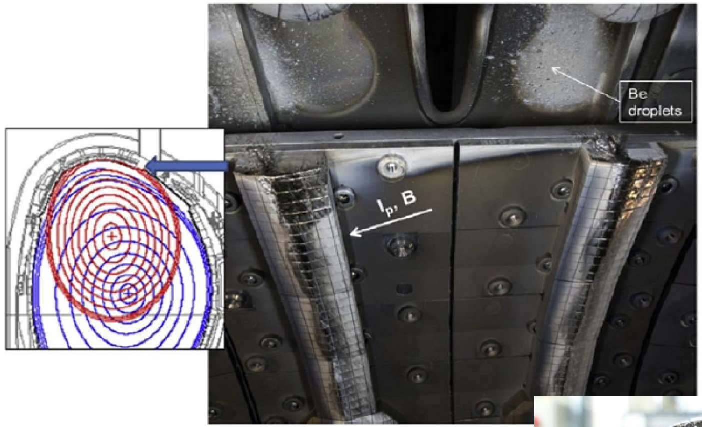


Transient loads – ELMs and disruptions – 2

Disruption: Fast events in tokamak plasmas that lead to the complete loss of the magnetic (CQ) and thermal energy (TQ) stored in the plasma. May be triggered by various events.

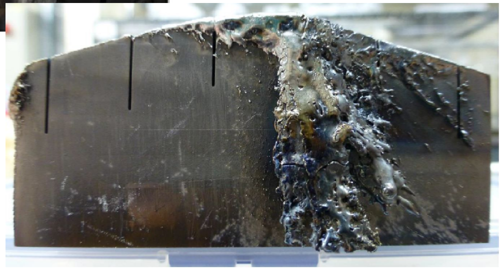
The rapid current quench associated to disruptions is challenging for the mechanical integrity of the FW as well as loading thermally the PFCs (the thermal quench causes surface melting). Shaping aimed at protecting exposed edges is helpful in reducing melt areas and erosion but global power handling cannot be designed in.

Disruptions in ITER have to be avoided, predicted and mitigated, especially at high plasma current

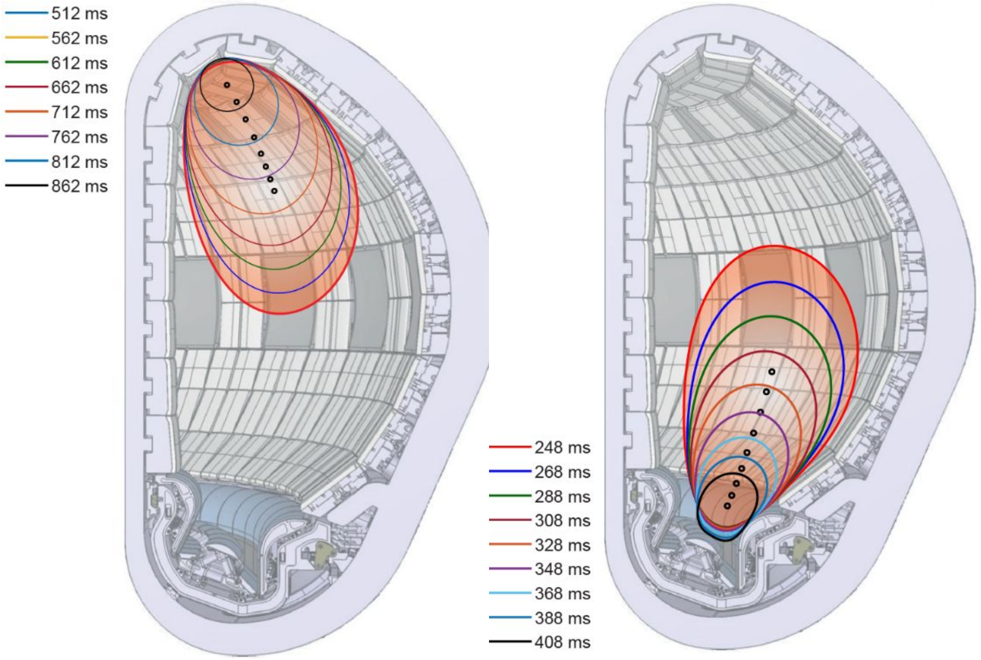


Be droplets

Be “waterfalls”



Be melt in JET caused by a VDE



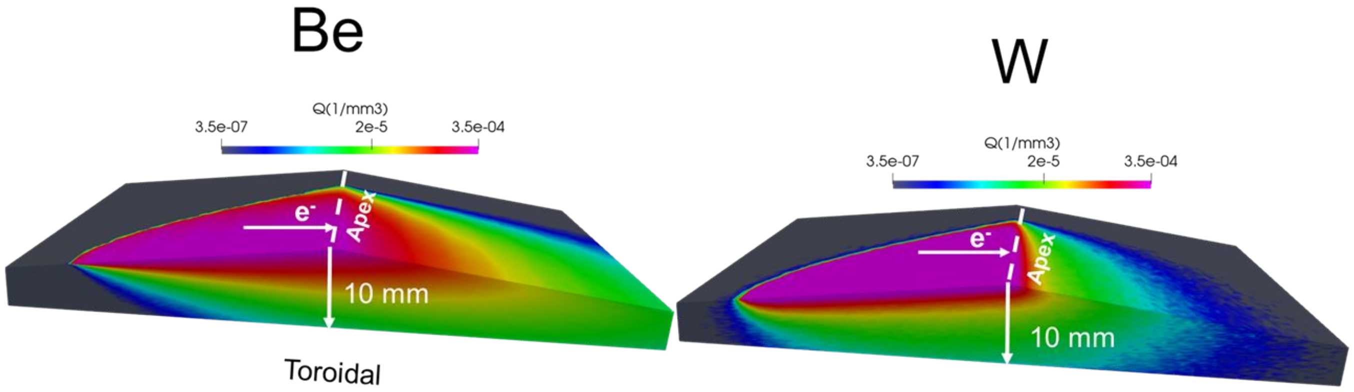
Transient loads: Runaway (RE) effects and damage

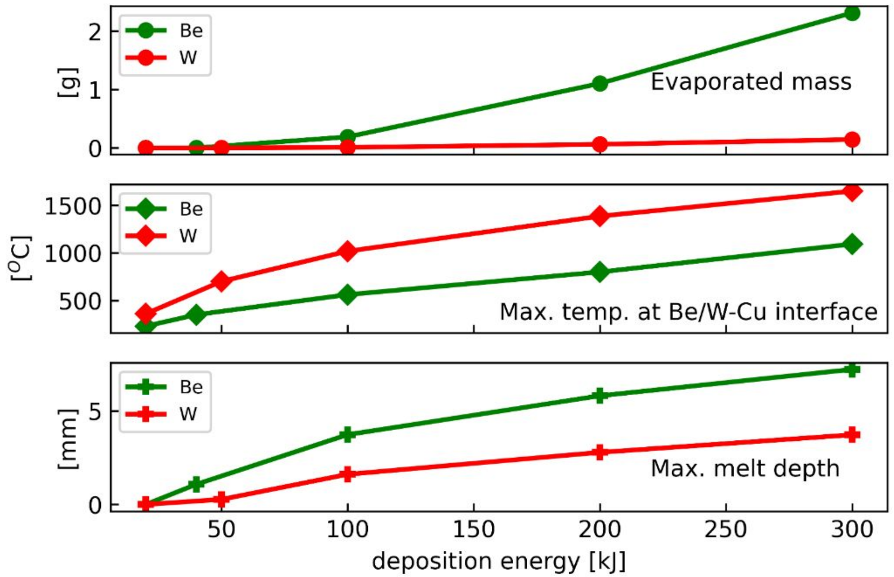
Runaway electrons are beams/columns of high-energy electrons moving toroidally (like the normal plasma current) at relativistic or near-relativistic energies (~MeV). They can be generated typically after disruptions in particular conditions, and require the presence of an high parallel electric field in the vacuum chamber as well as of a “seed“ of fast electrons.

Any PFC hit by a RE beam will be severely damaged, and RE hits have the potential to generate a water leak in one event.

Disruption and RE control/suppression have to be developed for ITER to avoid potentially severe machine downtime

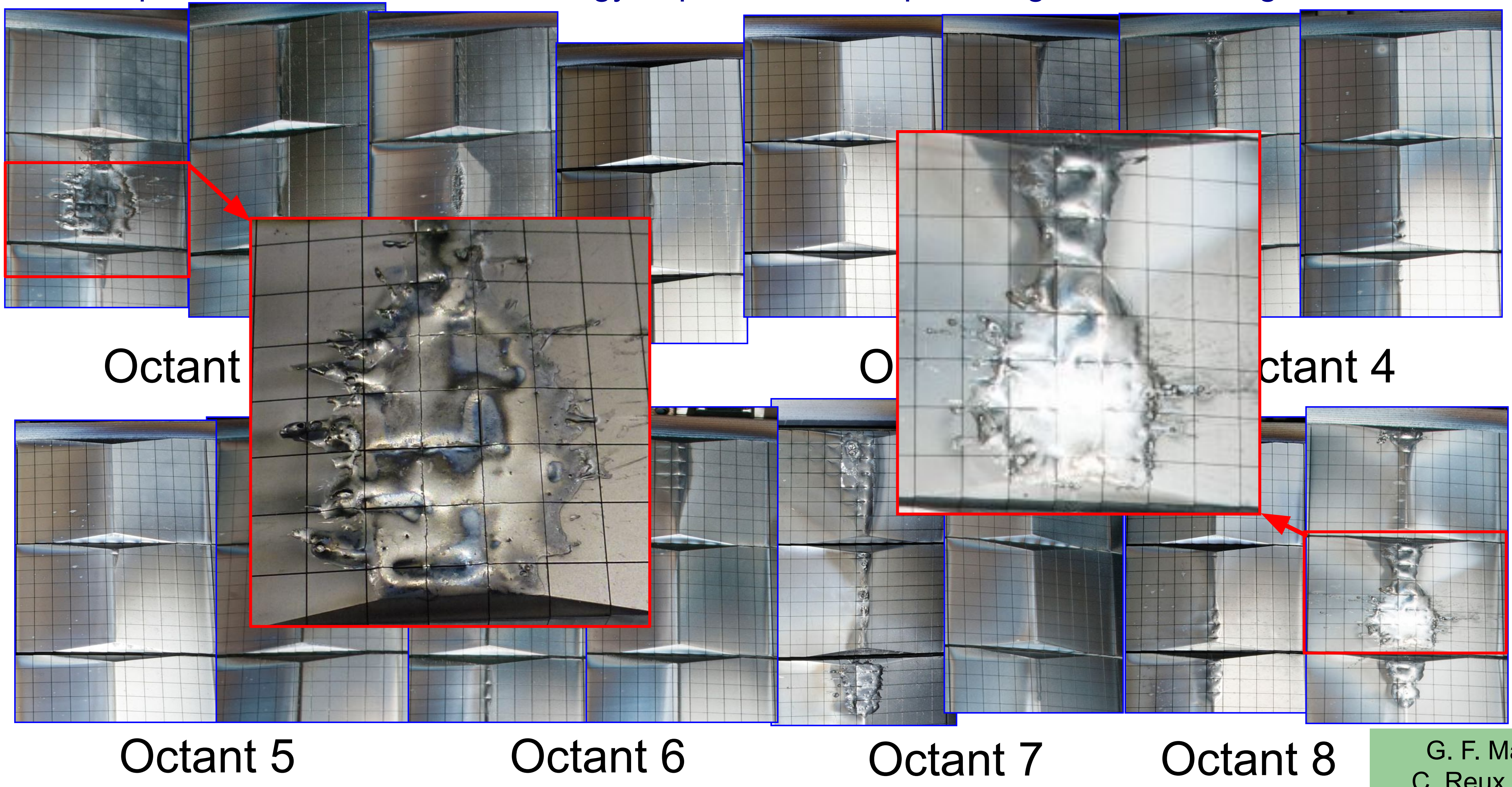
W is less resilient than Be to a RE hit (W has higher e- stopping power, shallower penetration) cooling tubes damage



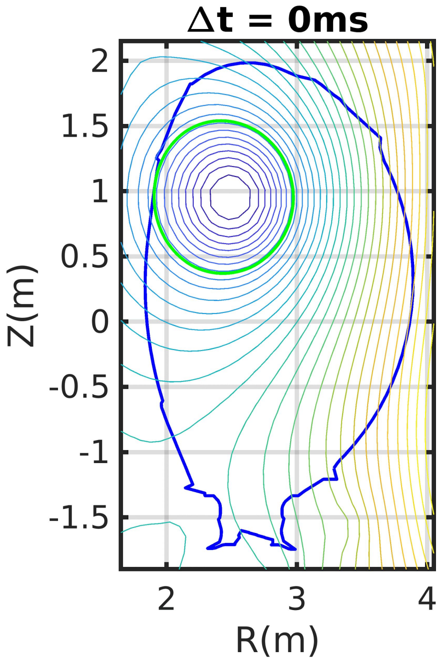


Single RE events can be very damaging

- Example from JET: volumetric energy deposition □ deep melting and Be boiling



JET
EUROfusion



G. F. Matthews et al., PFMC 2015
C. Reux et al., NF **55** (2015) 093013

Erosion and dust production

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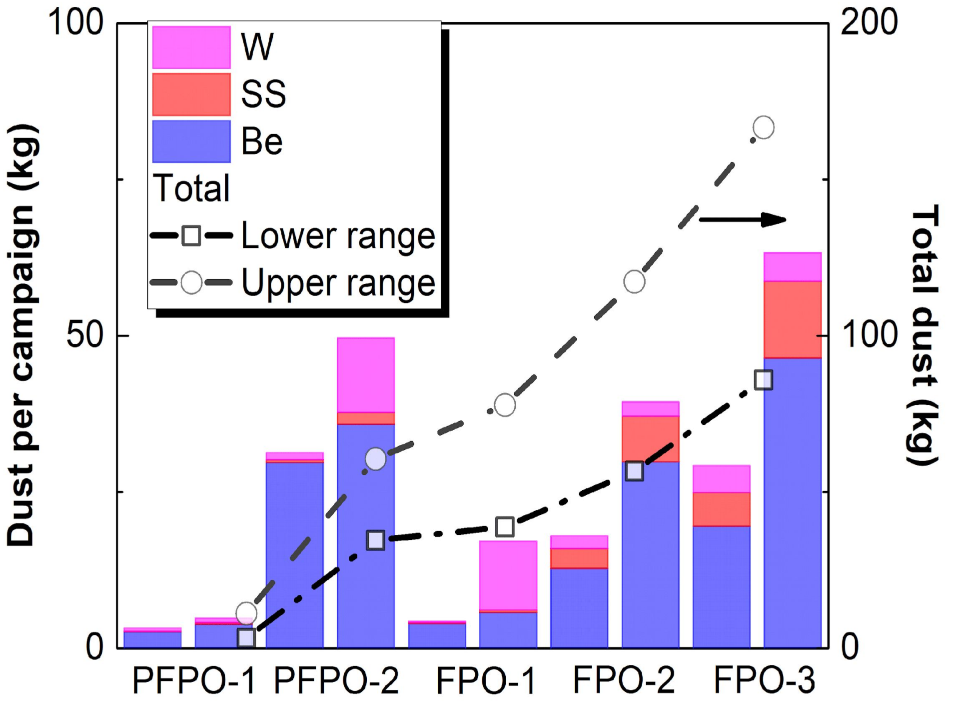
Be: The maximum allowed material loss due to erosion/melting is 4 mm for any first wall panel. Complex mechanisms related to erosion and redeposition of mobilized Be.

Be codeposition with Tritium (and T in the Be dust) are potential issues

For W (and SS) the source of dust is assumed to be just from physical sputtering in steady state and disruption events (melting). W dust does not retain Tritium by codepostion.

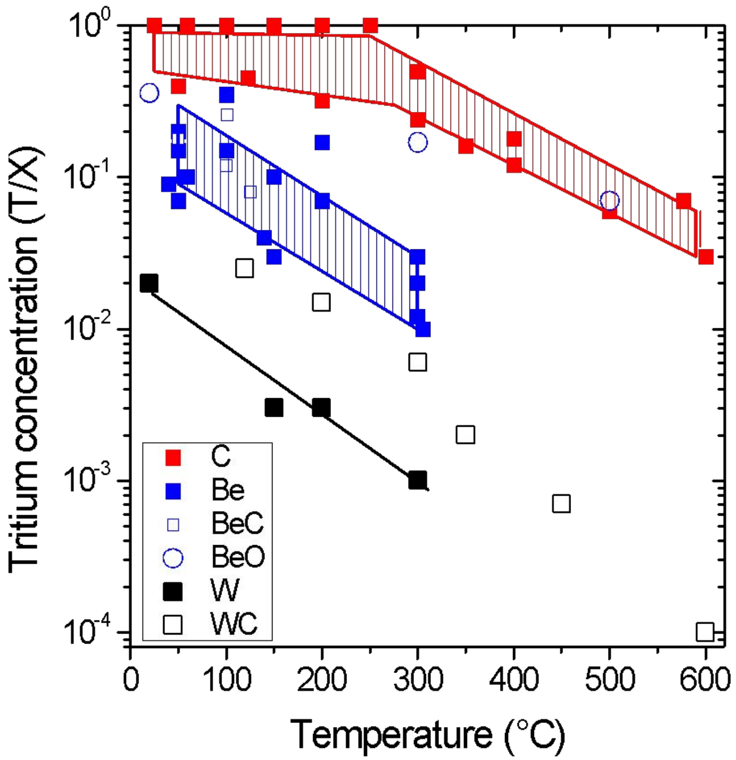
Changing the first wall from Be to W will reduce drastically dust production (see graph) .. And take into account that W atomic mass is ~20 times that of Be.

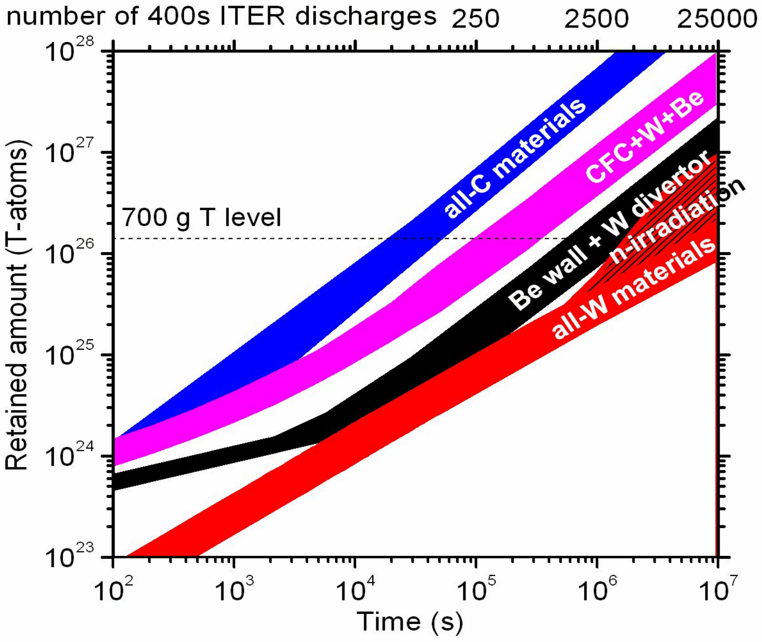


Tritium retention

- A 400 s Q_{DT} = 10 ITER discharge will require ~100 g of tritium fueling (cf. 0.01-0.2 g in today's tokamaks)
- Maximum in-vessel mobilizable T in ITER limited to 1kg
- In practice, administrative limit of ~700 g
  - 120 g in cryopumps
  - 180 g uncertainty

- Be □ T implants and goes in solution in solid Be + co-deposition of T also possible - large potential source of Be from first wall
- W □ most of retention is from implantation □ not thought to constitute a large reservoir
- effects of increased trapping due to neutron irradiation of metals – does not look like an issue from recent results

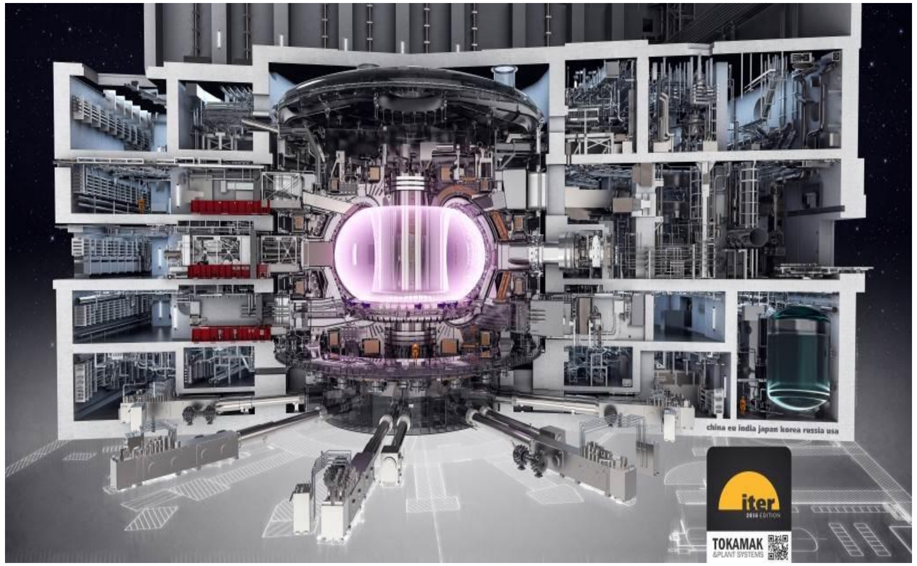




Sneak peak at the ITER site

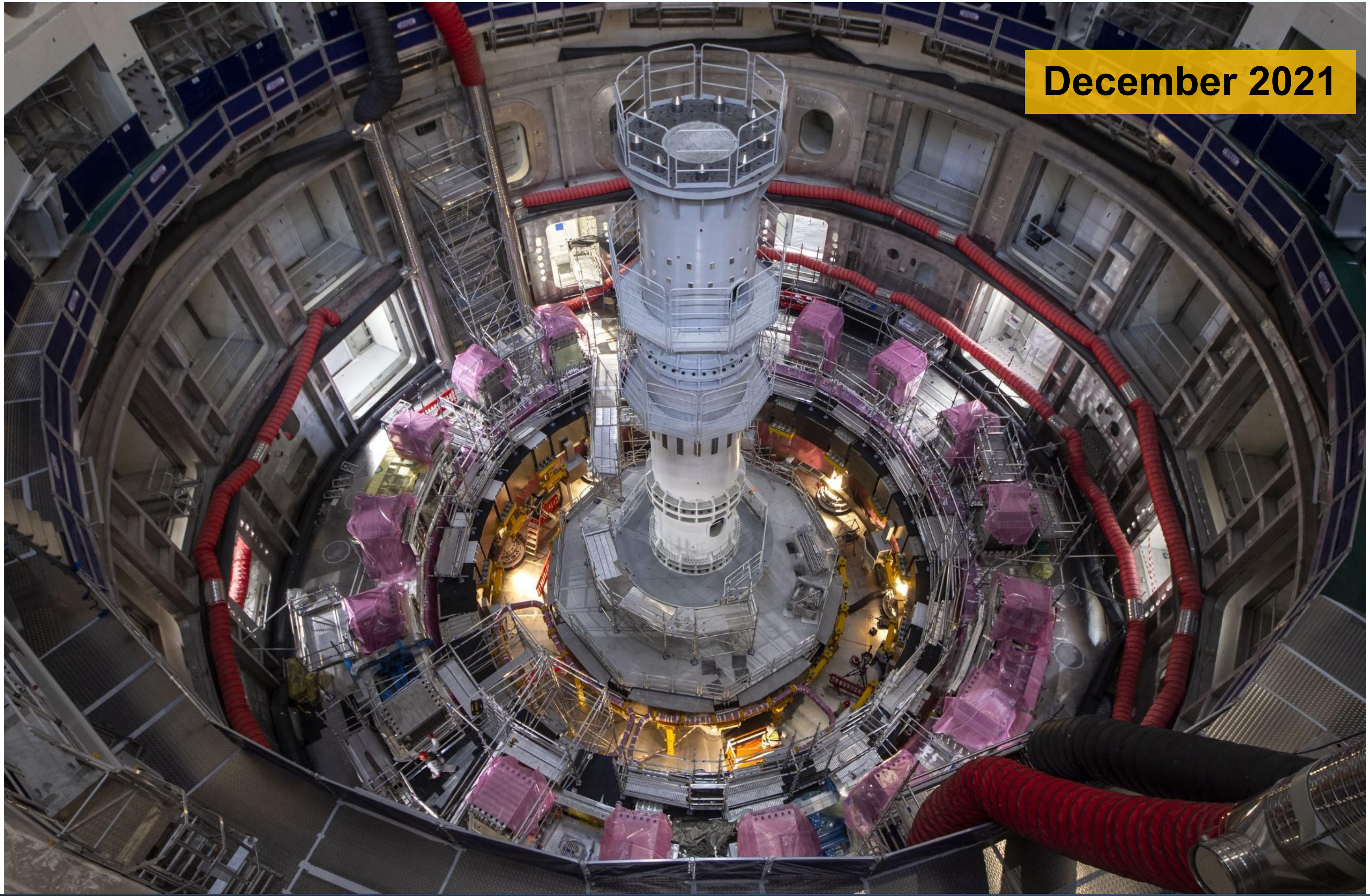
ITER Site today

ITER assembly hall

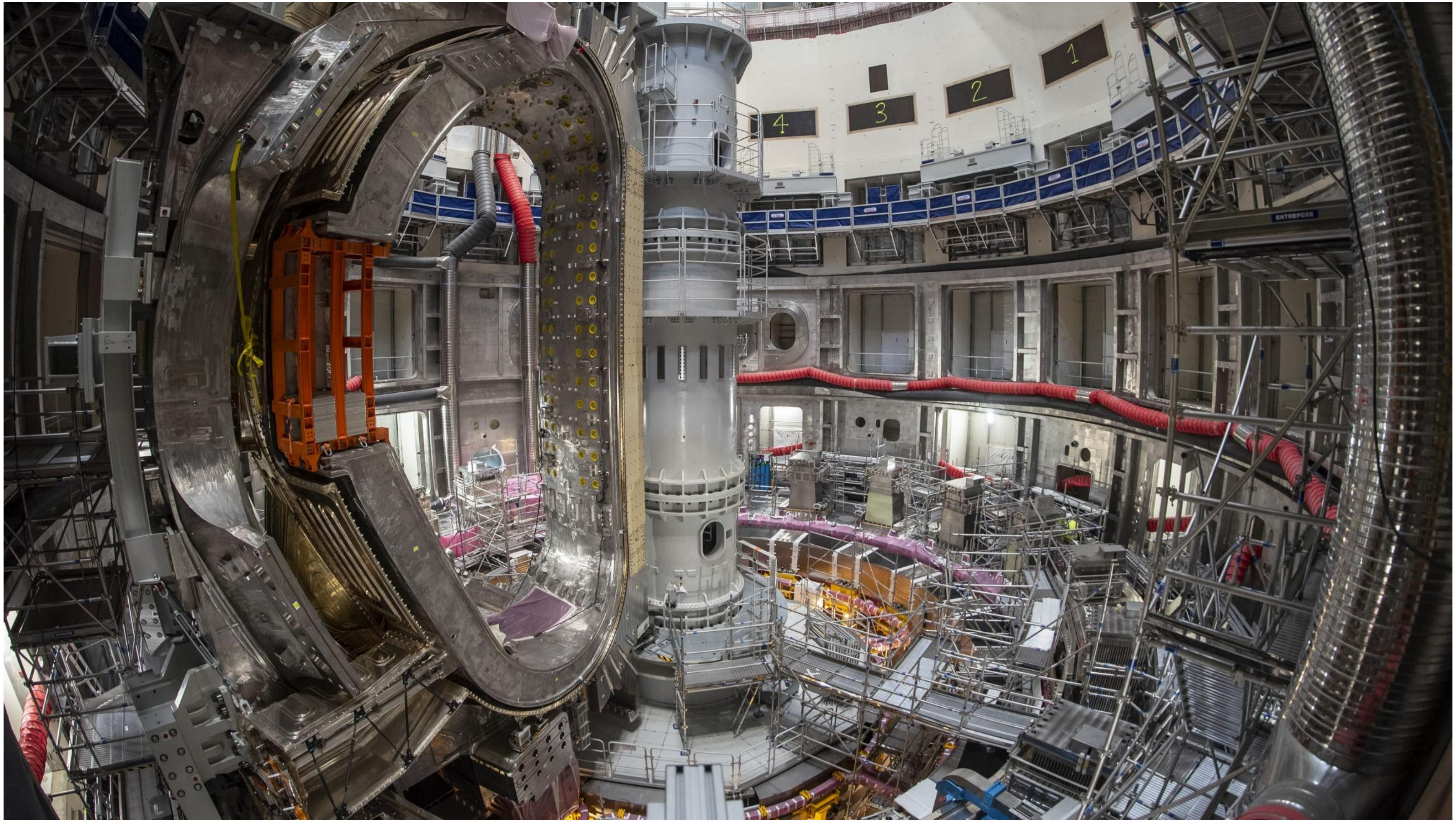


Tokamak components assembled in assembly hall and lifted by cranes into tokamak pit

Tokamak pit ready for the first sector (VV+TF coils)



Sector in place – alignment



In conclusion ….

- ❑ I hope to have given a “flavour” of the complex and very tight relation between the ITER design, the knowledge base in Tokamak physics, as exemplified from the selection of materials and the design of the first wall and divertor.

- ❑ ITER is being built, and design choices have been based on experimental evidence, modelling, technological and cost consideration. This is still ongoing (e.g. Be vs W), and more is needed to prepare for the exploitation phase.

- ❑ I congratulate you for being here, going into an exciting phase of professional career. Fusion is exciting, complex and necessary, and I am looking forward to your new ideas and contributions in the years to come.

Spare slides

ITER additional heating mix

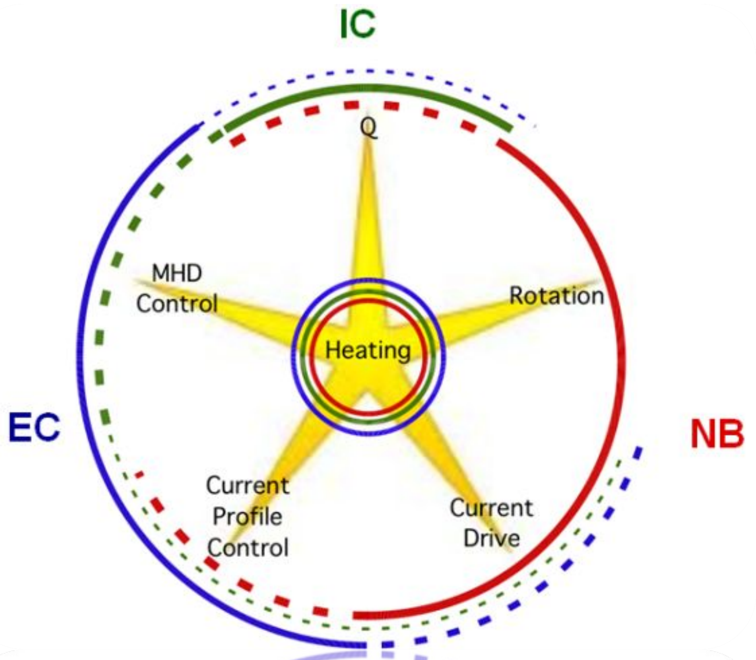
EC system	IC system	NBI system
170 GHz	40-55 MHz	870 keV H^0, 1 MeV D^0
	High fusion gain, ST control, wall cleaning,	Bulk current drive, rotation,
24 gyrotrons (24 x 0.8 MW) 40+ gyrotrons up to 20MW	2 antennas (2 x 10 MW)	2 injectors (2 x 16.5 MW) 3rd injector (16.5 MW)

NB: high-energy neutral injection at tangential angle

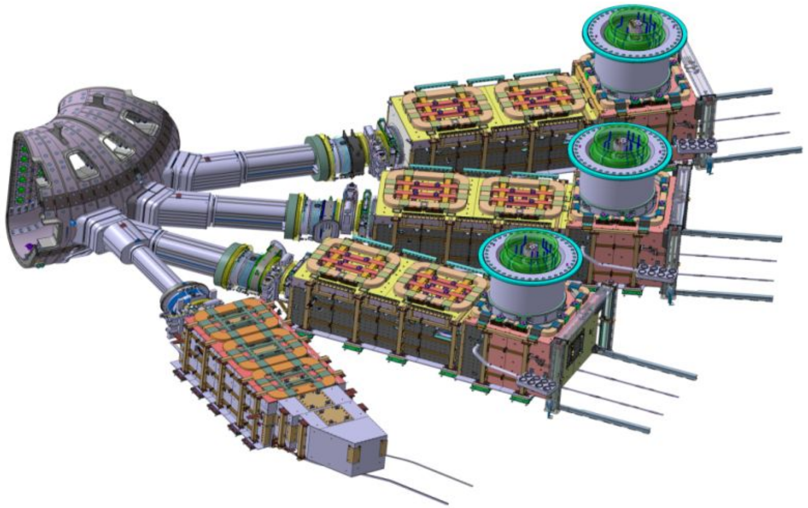
IC: resonance absorption at ion gyrotron frequency

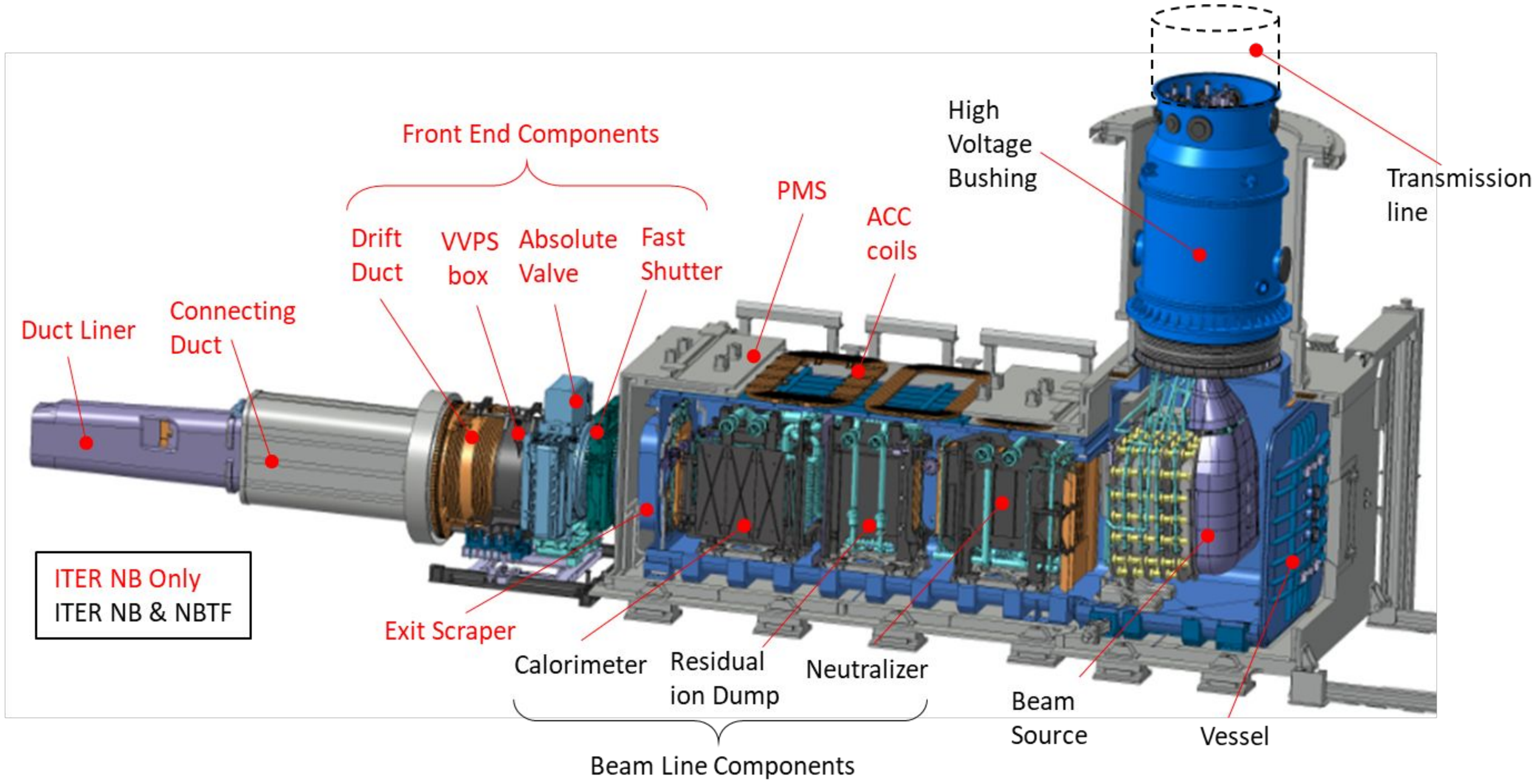
EC: resonance absorption at the electron gyrotron frequency

In addition to providing heating to the plasma, the ITER additional heating system is versatile and fulfills several functions, mostly related to plasma control



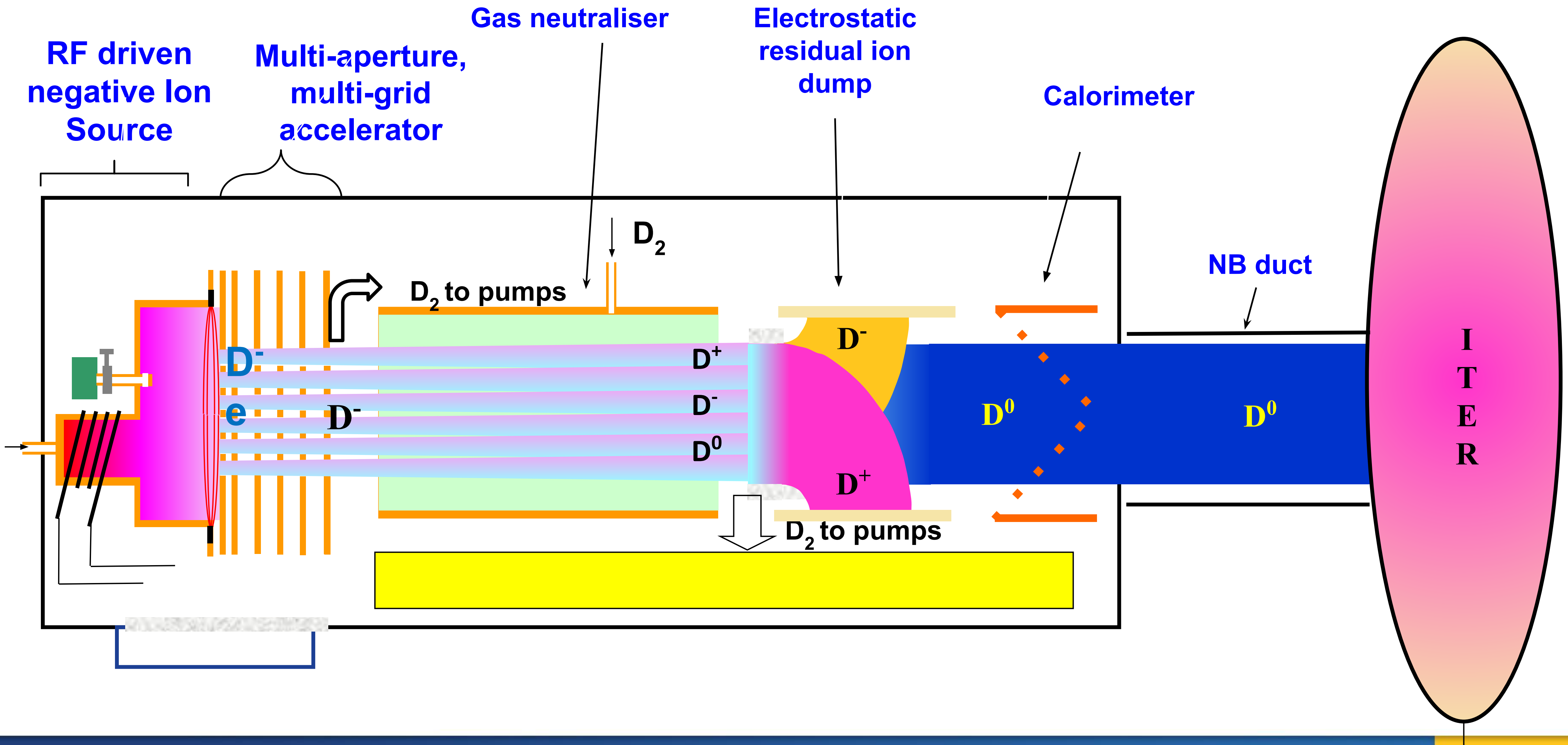
ITER NB components are prototyped 1:1 on the NBTF





NBTF = Neutral Beam Test Facility, Padova (Italy)

How do the ITER NB work?

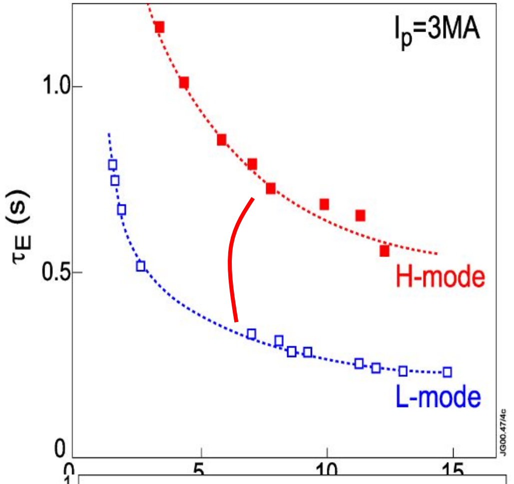


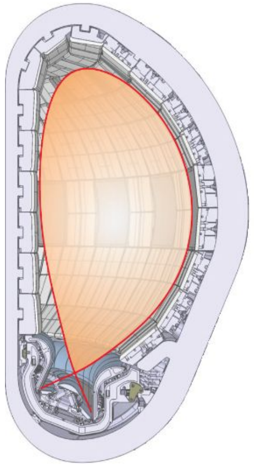
The H-mode plasma

Nature is not kind: when a plasma is heated, the heat loss increases, or τ_E goes down, and the temperature increase in the plasma is less and less efficient – the L-mode regime

But … plasma regimes have been discovered from the 1980s where τ_E improves over the L-mode values □ H-mode(s) □ ITER performance projections are based on the H-mode

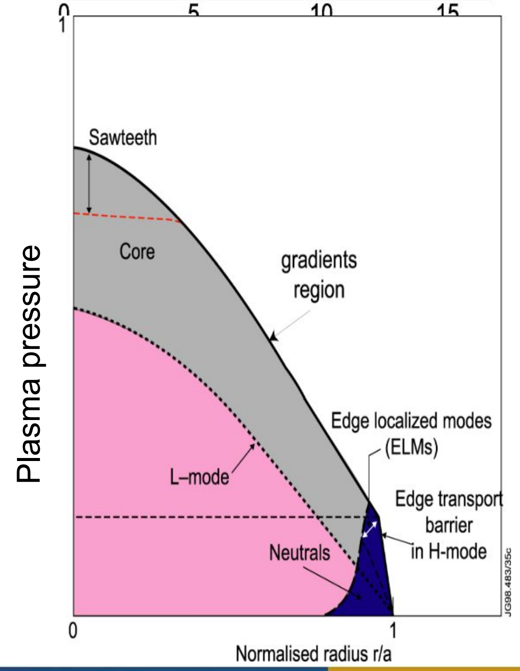
Access to the H-mode regime is facilitated by shaping, in particular by the forming x-point plasmas and require additional heating to achieve the L to H transition.





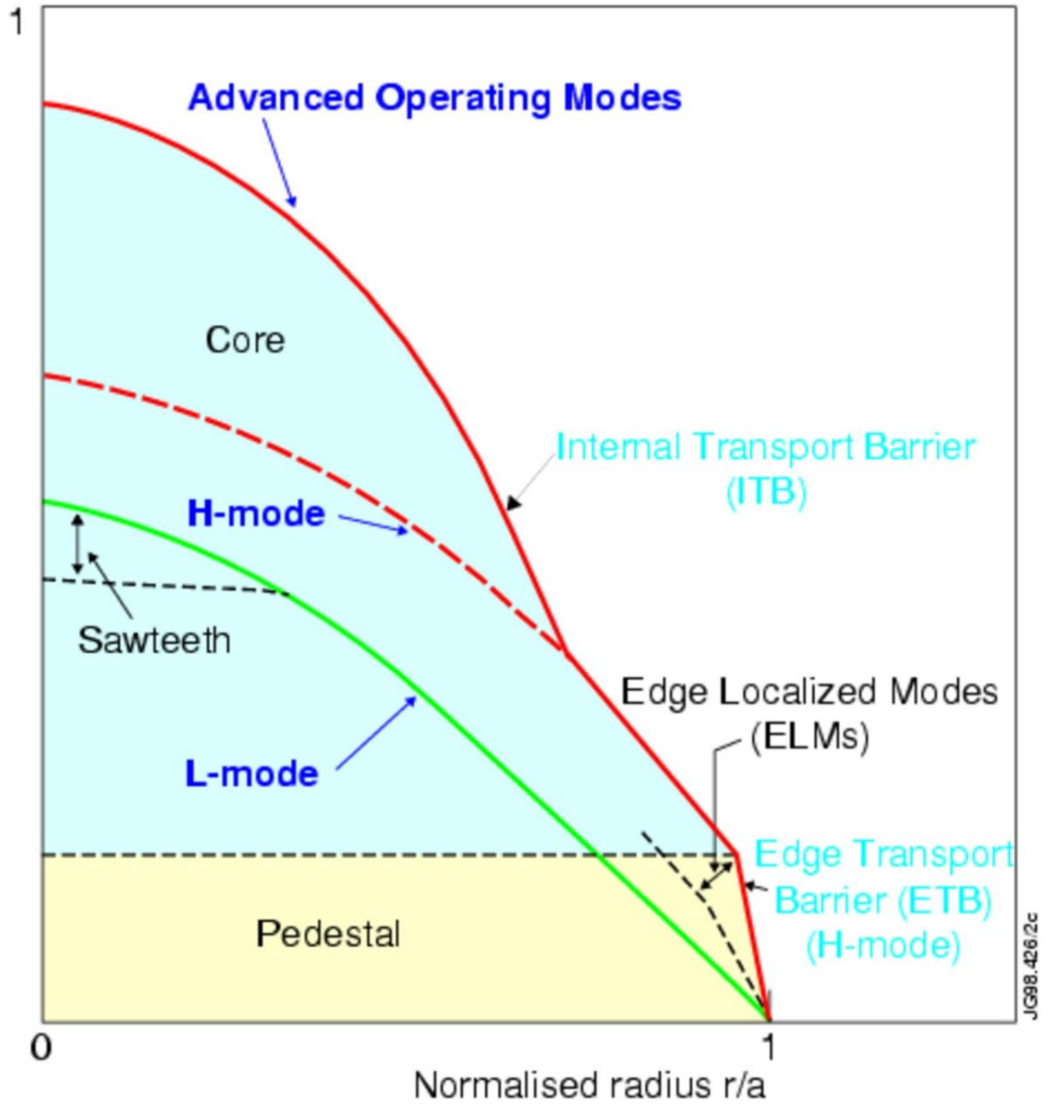
The increased confinement/temperature/density in H-mode is mostly due to improved transport in the edge region of the plasma and the formation of a pedestal.

Pedestal properties (such as eight, width, stability, ..) are strongly dependent on the shape of the plasma (triangularity, elongation and squareness) sometimes in subtle ways.

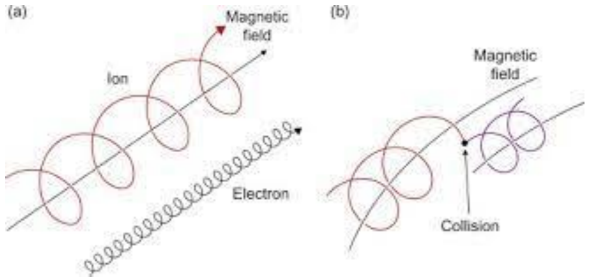


The ITER design includes all the aspects above, pushed to the scale required for achieving a burning plasma

Schematic of plasma profiles in L, H and advanced modes

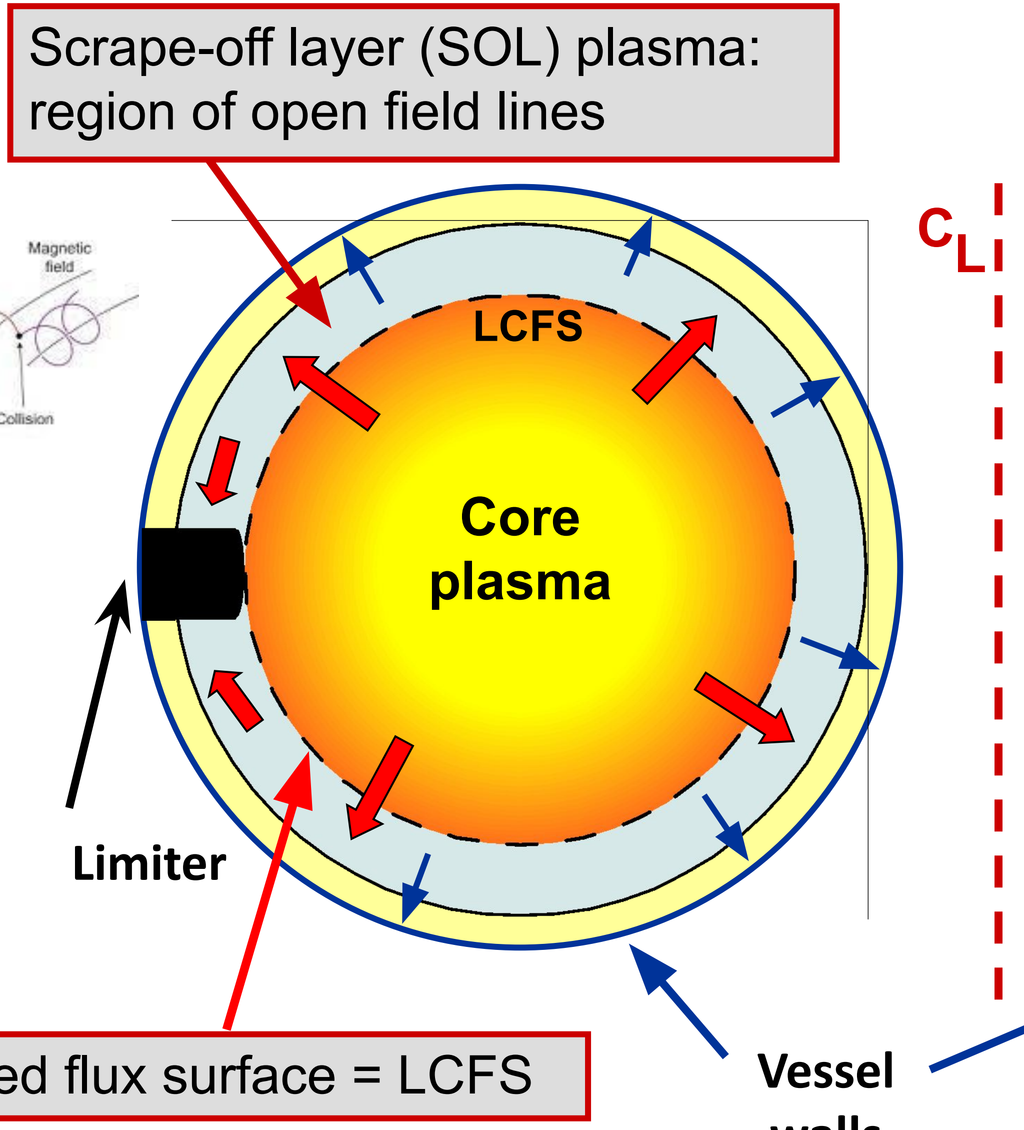


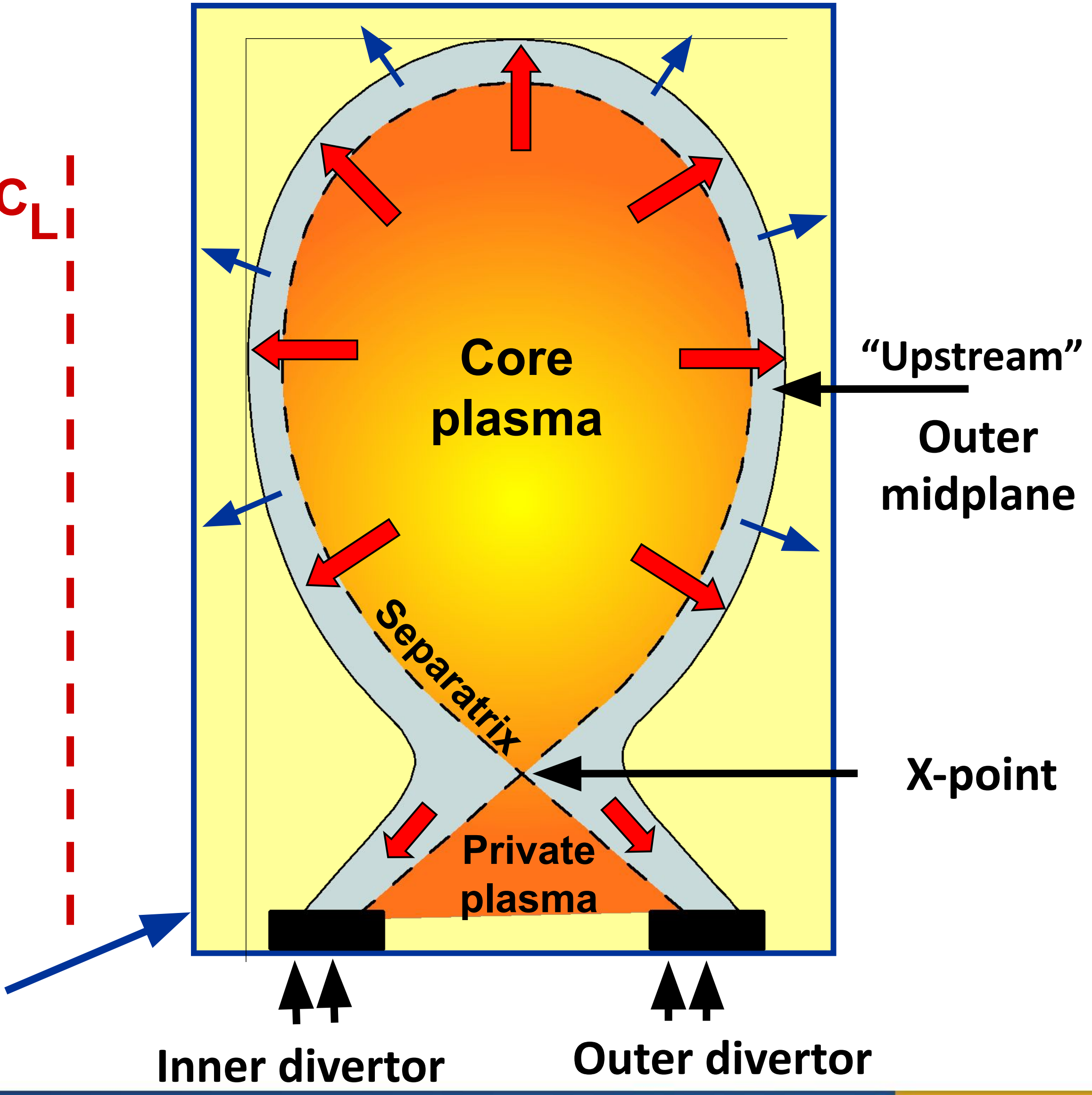
Limiter and divertor configurations – heat fluxes



Classical transport

+ turbulent transport (core and edge)

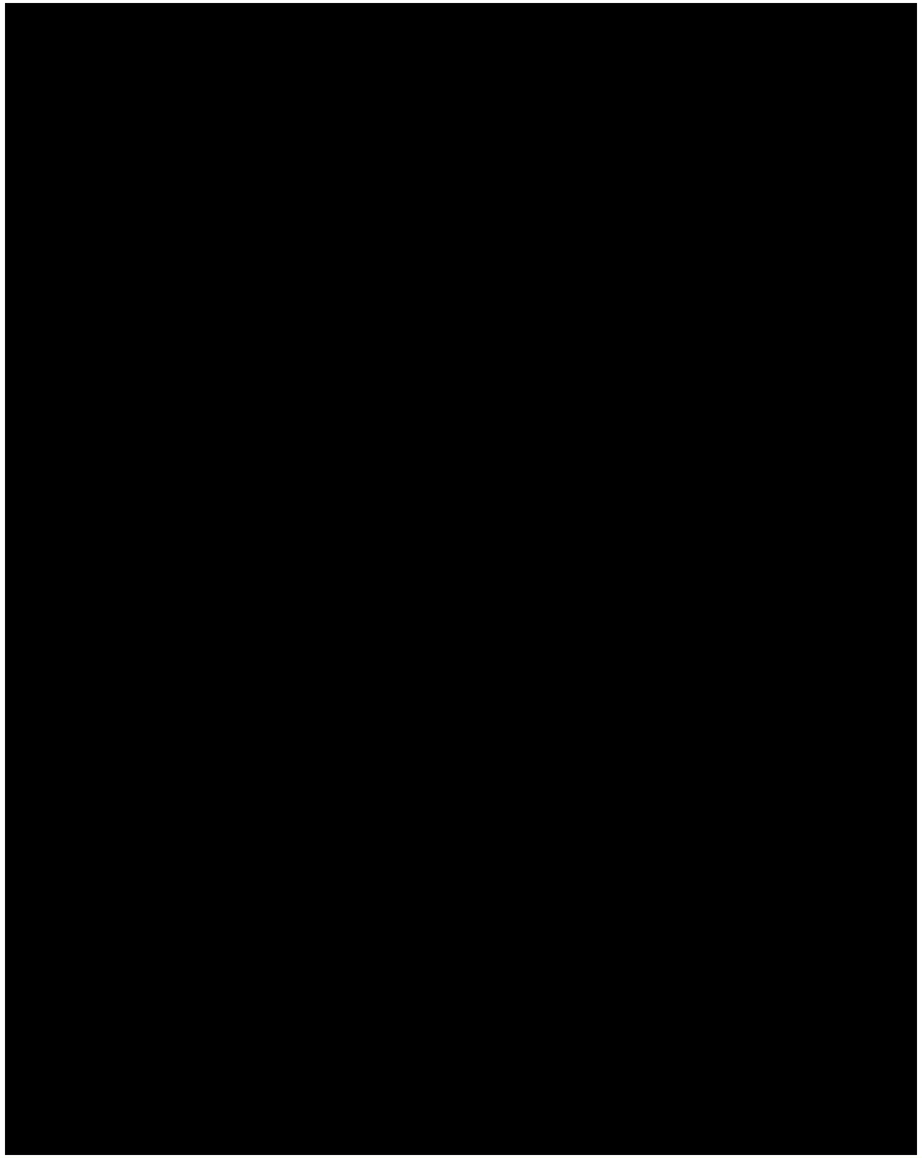




An example of plasma pulse from JET

JET Pulse # 97395
Baseline scenario 3MA/2.85T
27MW NBI + 4 MW ICRH

- Plasma initiation (few 100 ms)
- Limiter phase (~1.5s)
- X-point phase
- Termination (X-point and IW)



Transport creates and moves impurities

FUSION FOR ENERGY

Ions:

Cross-field transport – turbulent driven far SOL ion fluxes

 recycled neutrals

 direct impurity release

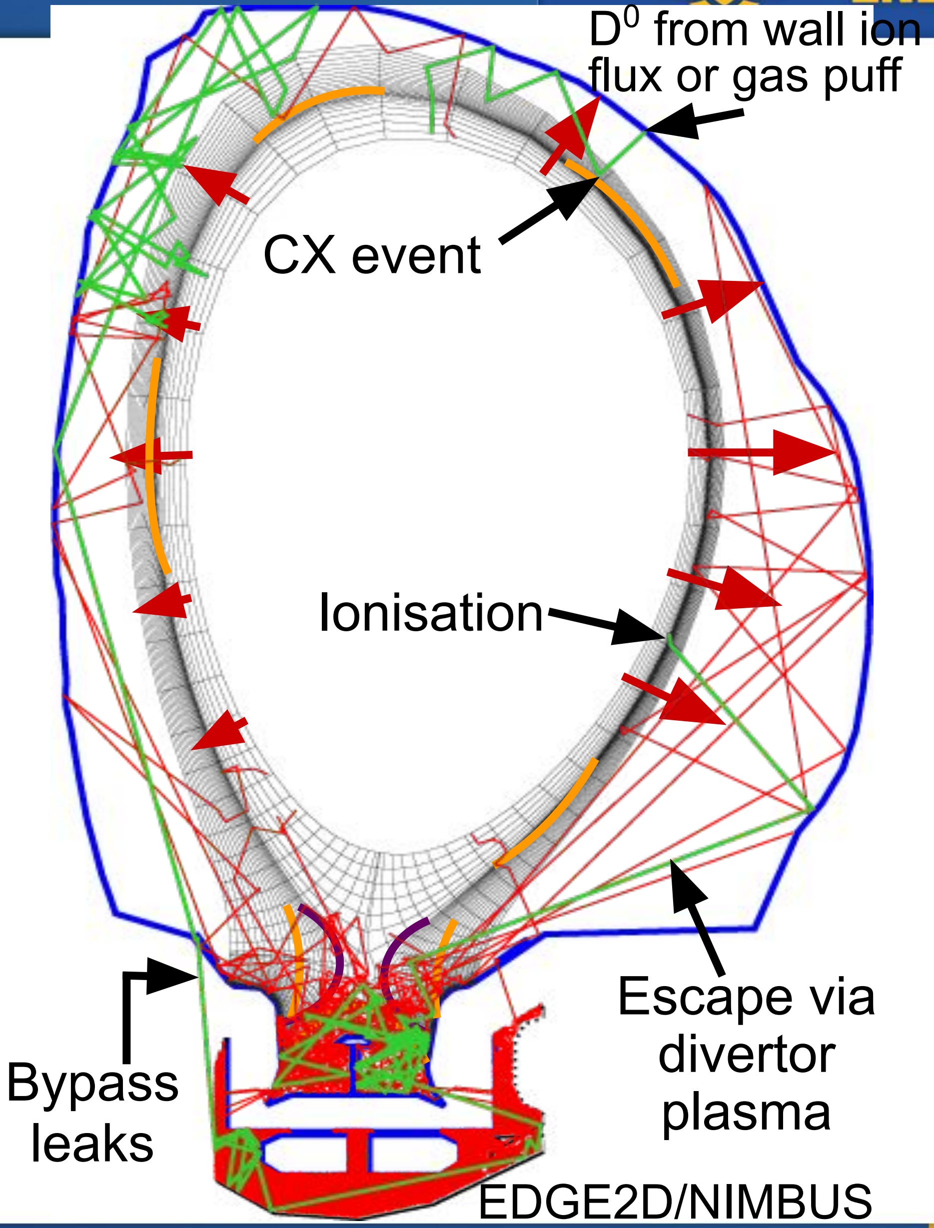
ELMs can also reach first walls

Eroded Impurity ions "leak" out of the divertor (∇T_i forces)

SOL and divertor ion fluid flows can entrain impurities

Neutrals:

- From divertor plasma leakage, gas puffs, bypass leaks low energy CX fluxes wall sputtering
- Lower fluxes of energetic D^0 from deeper in the core plasma



Dust – why worry?

FUSION FOR ENERGY

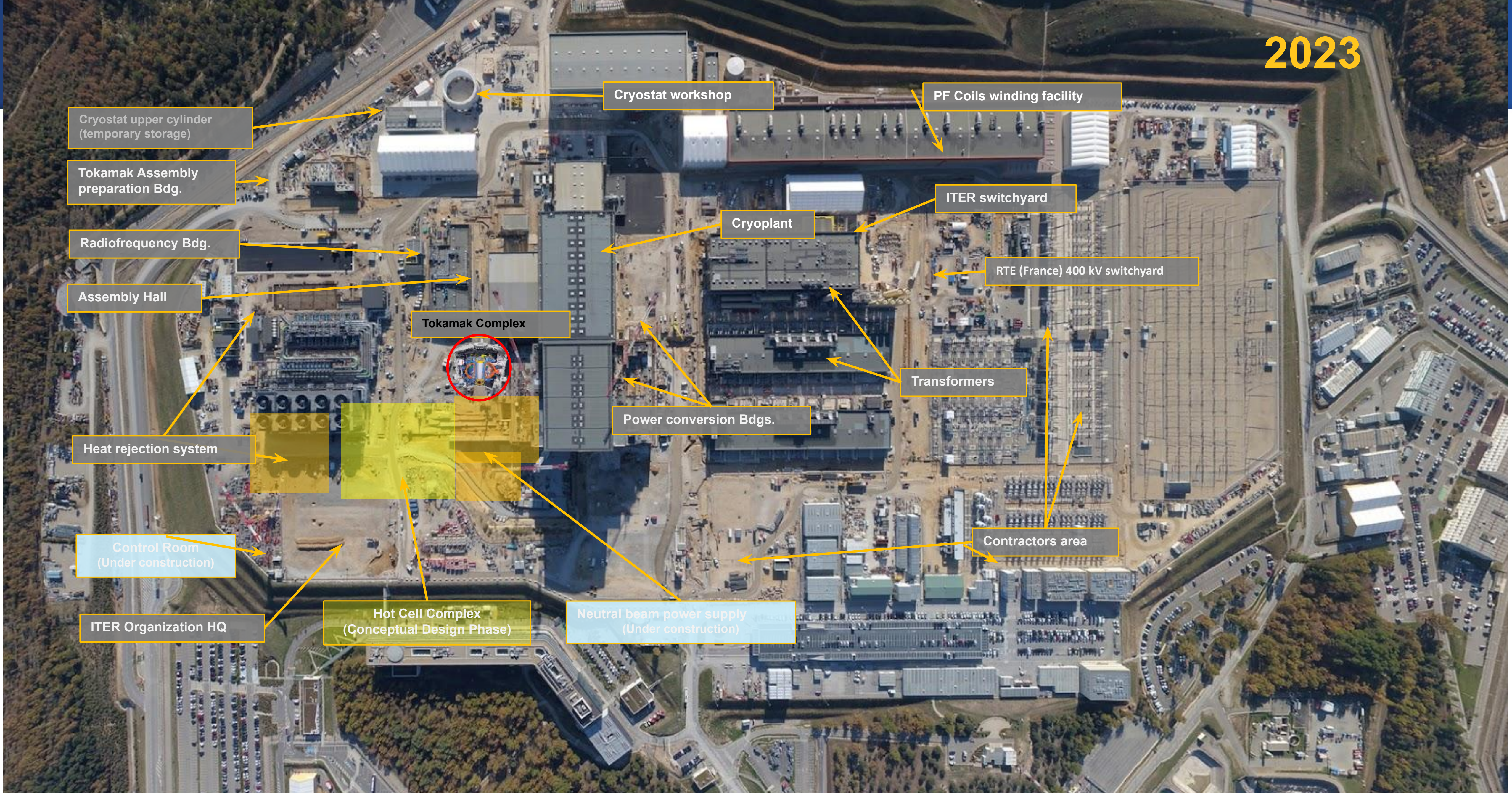
Expectation is that increase in duty cycle and erosion in ITER will lead to large scale-up in quantity of dust particles produced

Like T-retention, dust is a safety issue

- dust particles radioactive (tritium + activated metals)
- potentially toxic (Be)
- potentially responsible for a large fraction of in-VV mobilizable tritium
- chemically reactive with steam or air

Radiological or toxic hazard depends on how well dust is contained in accident scenarios and whether it is small enough to remain airborne and be respirable

- size needs to be <~ 100 mm
- depends on how dust is produced, e.g. crumbling of co-deposited layers or destruction (thermal overload) of tritiated layers during off-normal events
- tritiated dust can levitate in electric fields as a result of self-charging due to emission of beta electrons


2023
Cryostat workshop
PF Coils winding facility
Cryostat upper cylinder
(temporary storage)
Tokamak Assembly
preparation Bdg.
ITER switchyard
Cryoplant
Radiofrequency Bdg.
RTE (France) 400 kV switchyard
Assembly Hall
Tokamak Complex
Transformers
Power conversion Bdgs.
Heat rejection system
Contractors area
Control Room
(Under construction)
Hot Cell Complex
(Conceptual Design Phase)
Neutral beam power supply
(Under construction)
ITER Organization HQ

TF coil completed (coil in case)

Assemby hall – VV sector & TF coils – April 2022