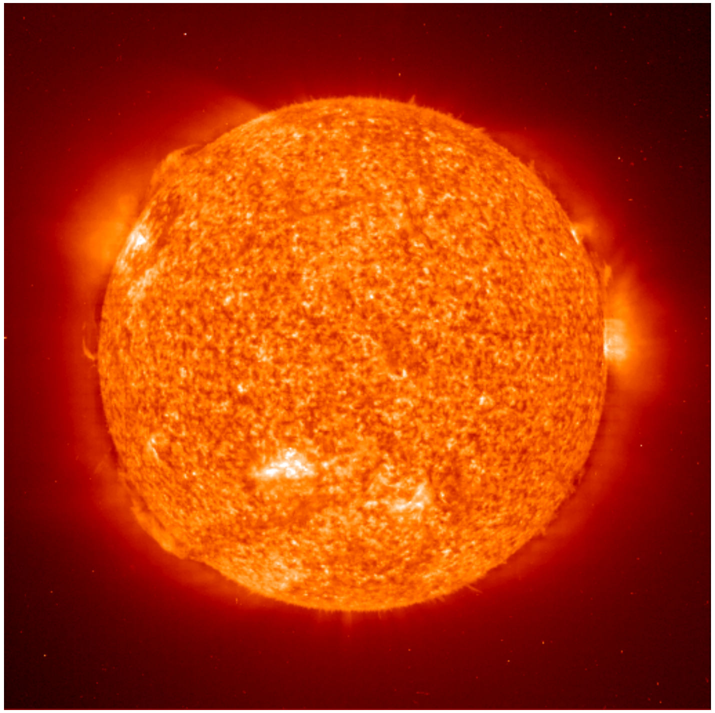
Max-Planck-Institut für Plasmaphysik

Future of Fusion Research: The Way towards Fusion Reactors



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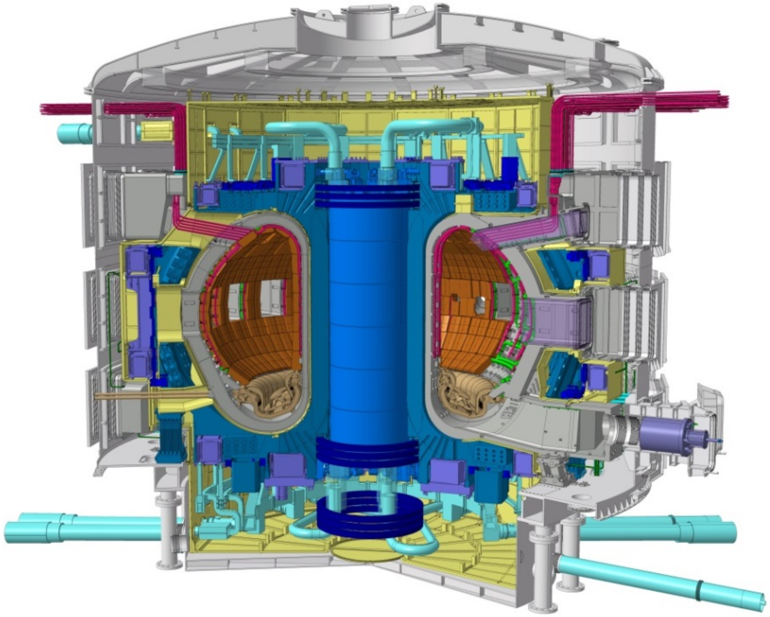
85748 Garching



Hungarian Fusion Winter School
Budapest, 12.02.2017

I. Introduction: Fusion Power Plants in a Nutshell

II. ITER: Demonstrate Burning Plasmas

III. DEMO: Self-sustained generation of Fusion Energy

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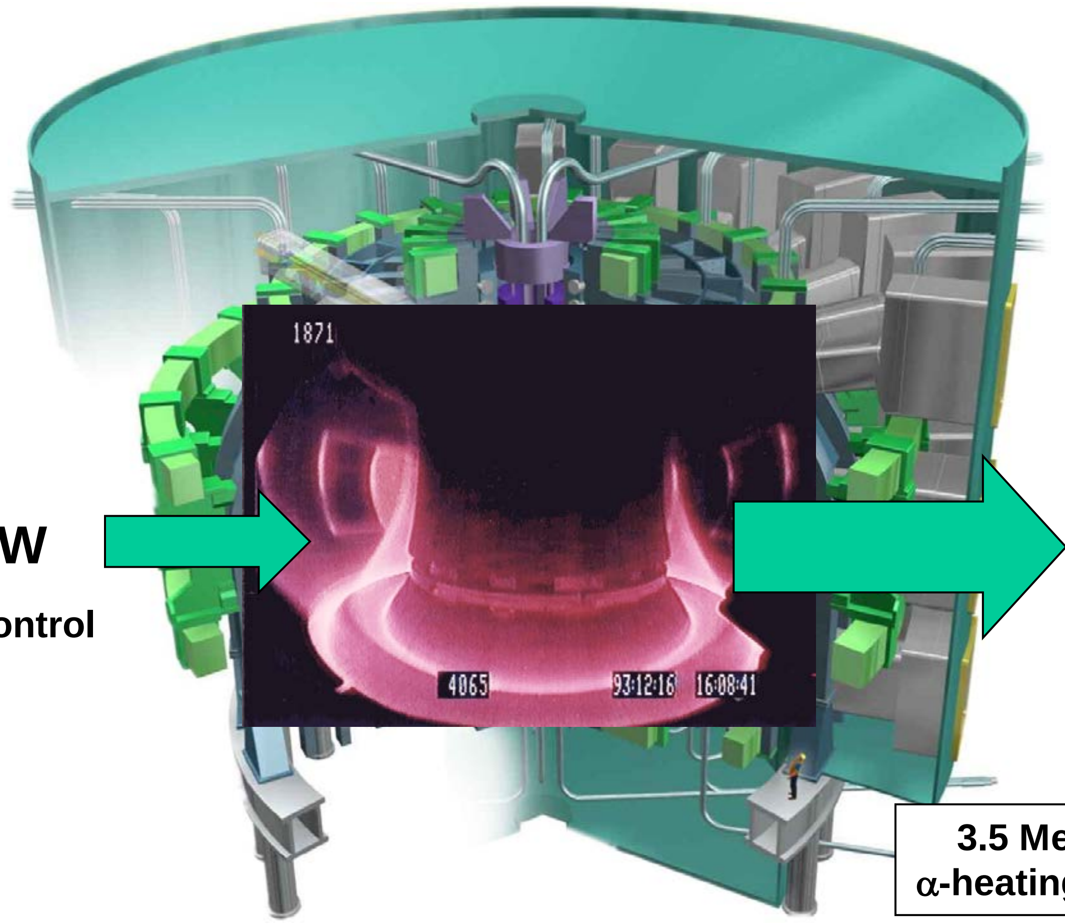
Schematic View of a Fusion Power Plant



P_{in} = 50 MW

(initiate and control burn)

P_{out} = 2-3 GW_{th}

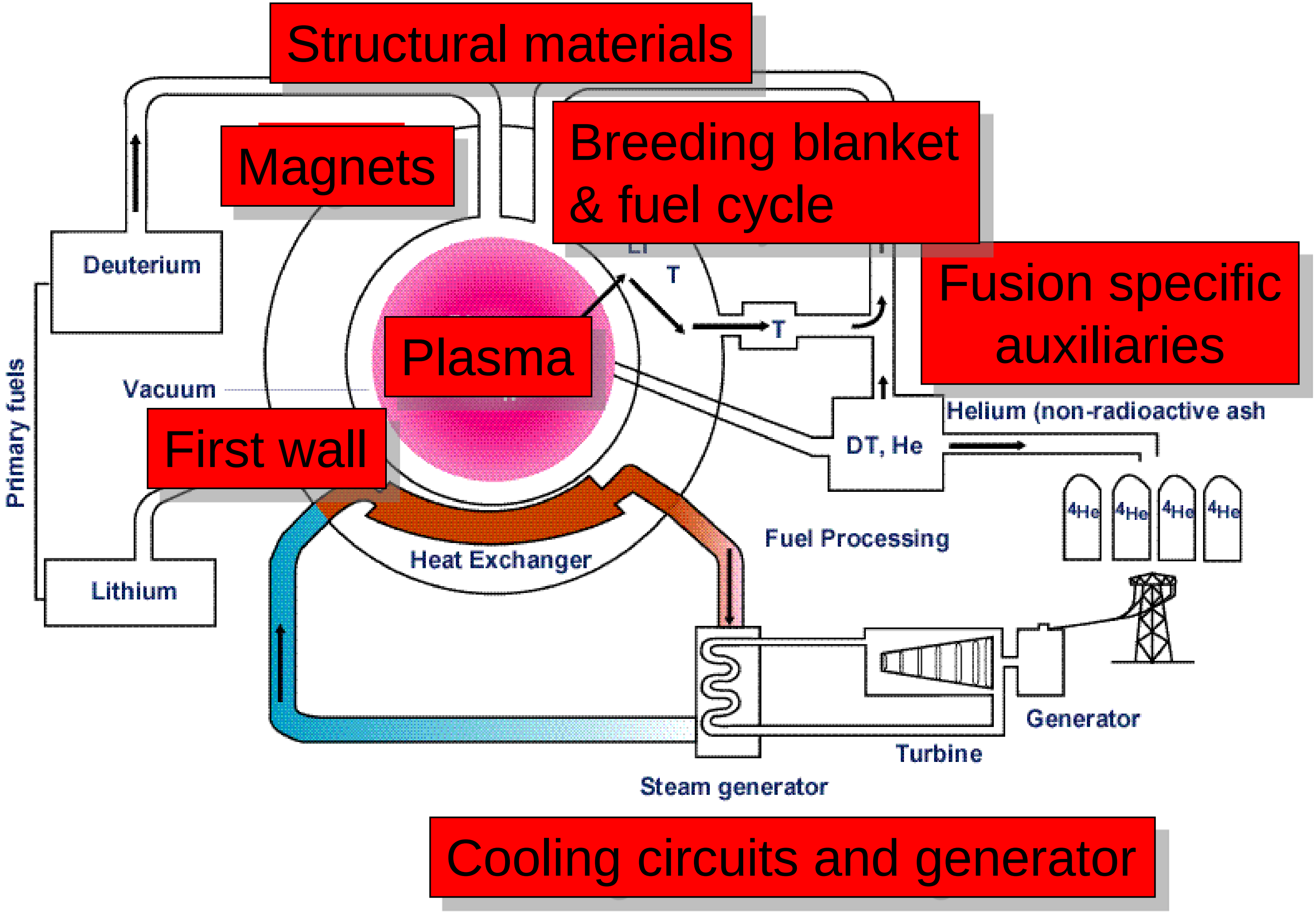
(aiming at 1 GW_e)

3.5 MeV	14.1 MeV
α-heating	wall loading

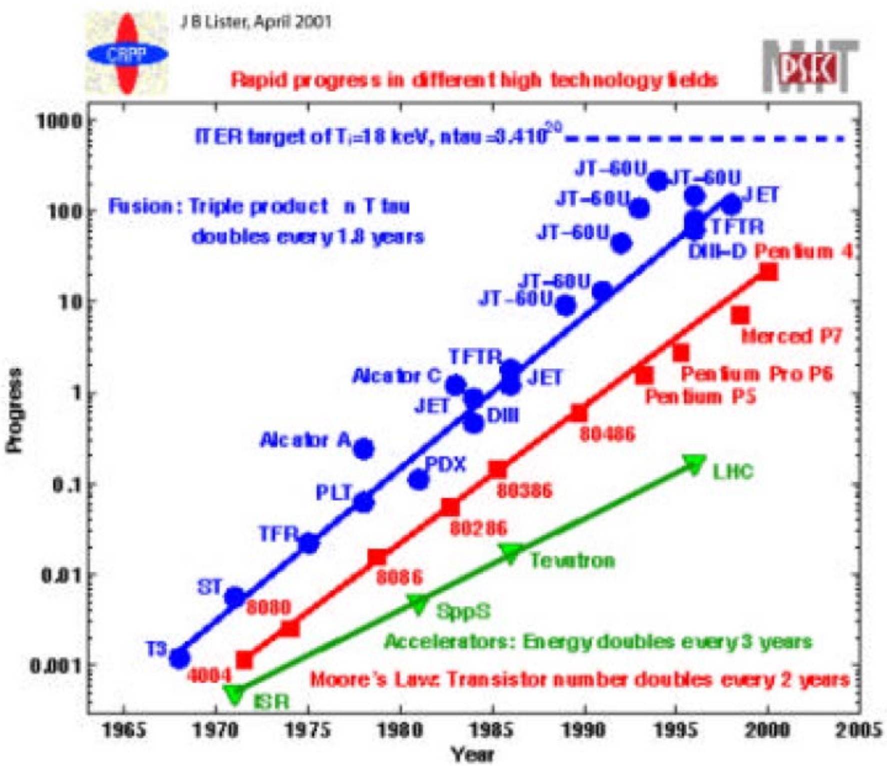
Fusion reactor: magnetically confined plasma, D + T → He + n + 17.6 MeV

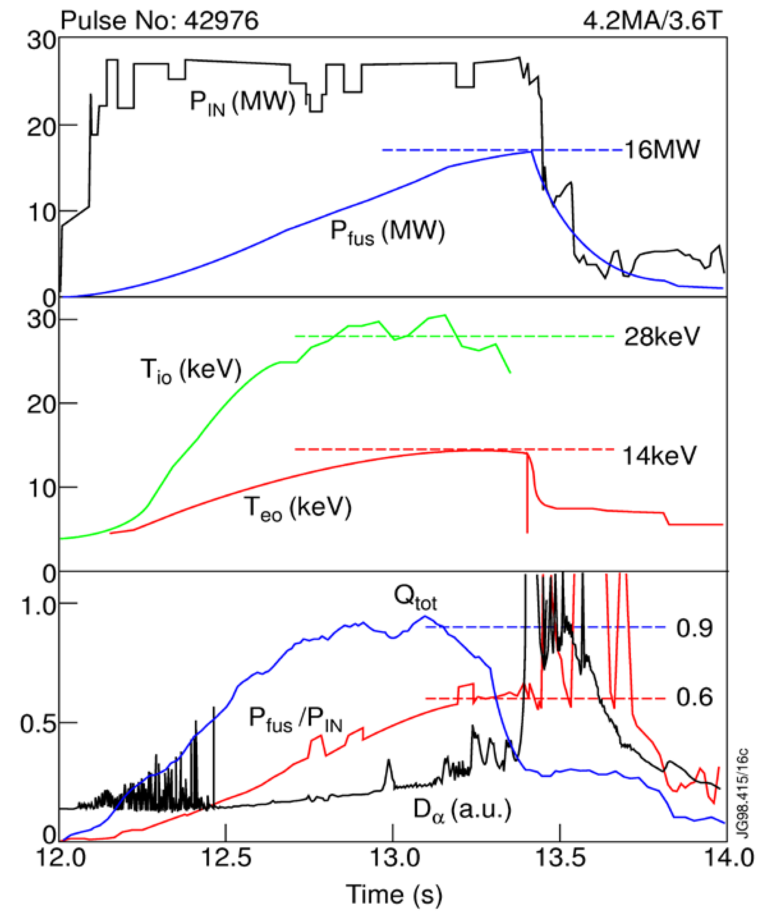
Centre of reactor: T = 250 Mio K, n = 10^{20} m^{-3}, p = 8 bar

Fusion Power Plant - Challenges



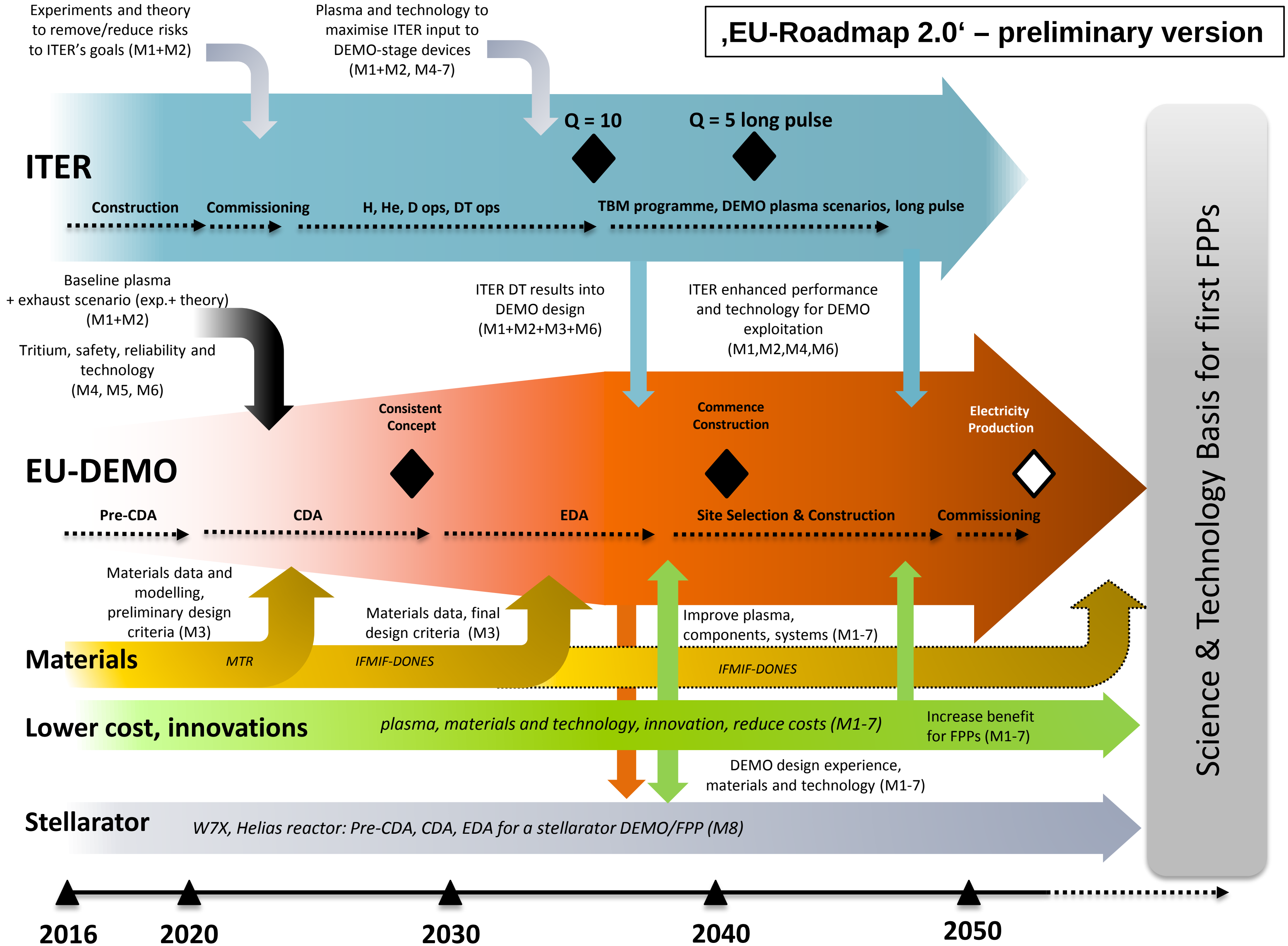
Tokamaks have made Tremendous Progress





- figure of merit $nT\tau_E$ has doubled every 1.8 years
- JET tokamak in Culham (UK) has produced 16 MW of fusion power
- present knowledge has allowed to design a next step tokamak to demonstrate large scale fusion power production: ITER

EU-Roadmap to Fusion Power



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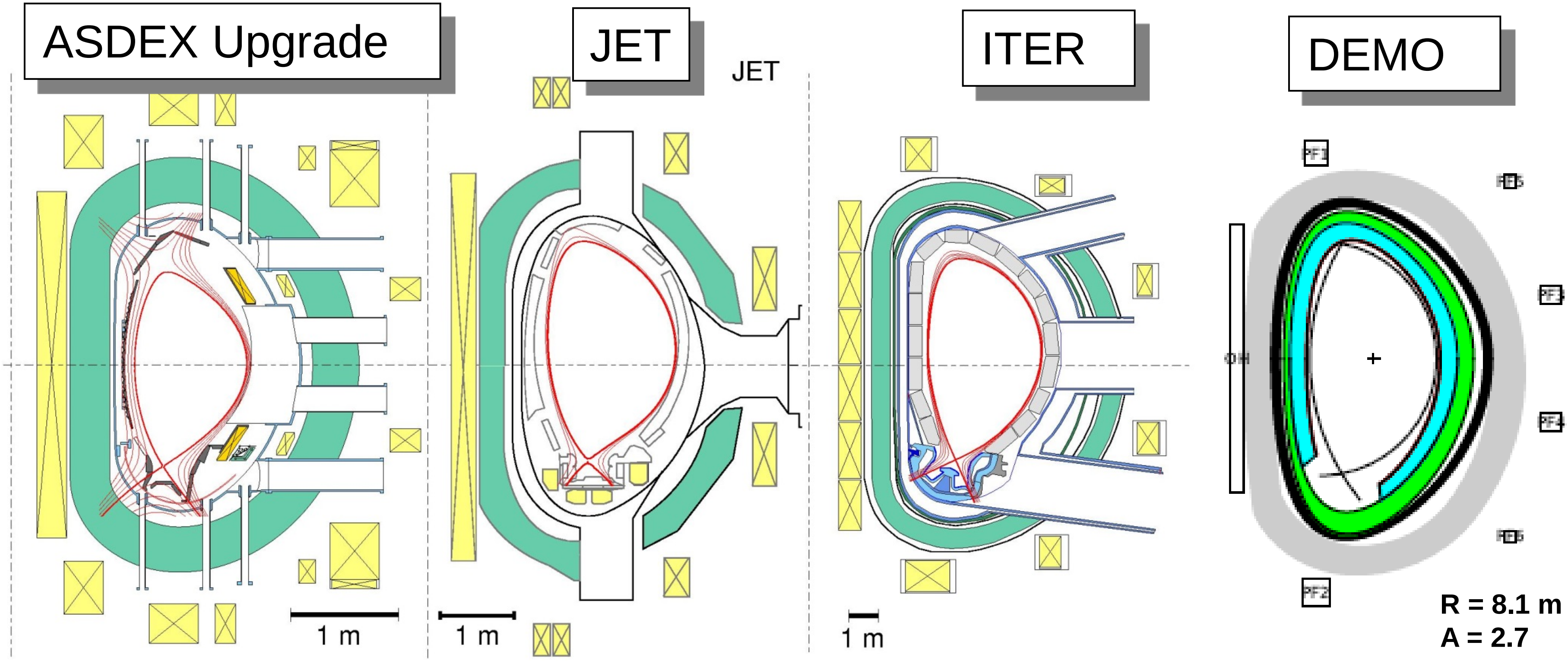
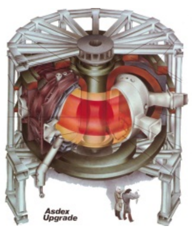
A Stepladder to Fusion Energy

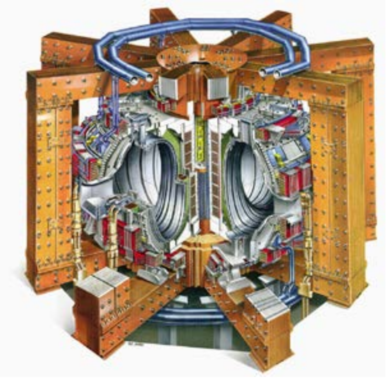


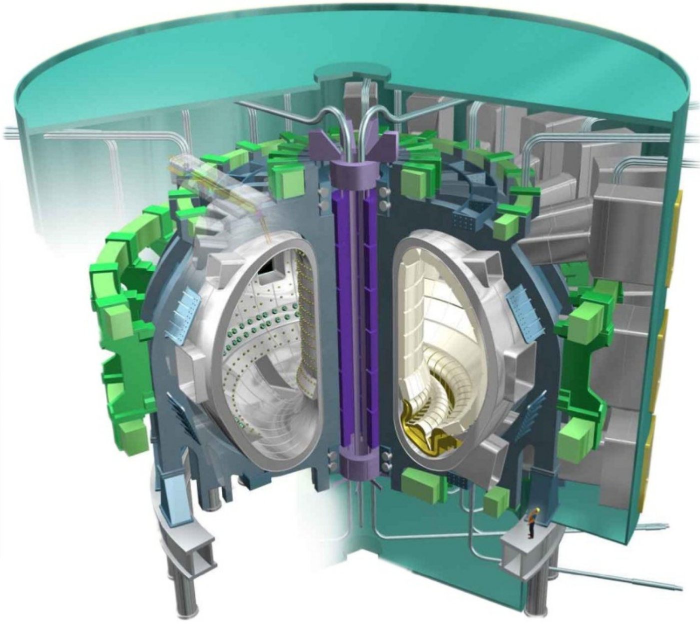
Set of geometrically similar tokamak devices explore physics and technology

- up-scaling requires solid theoretical understanding – fundamental research!

A step-ladder of fusion experiments to ITER

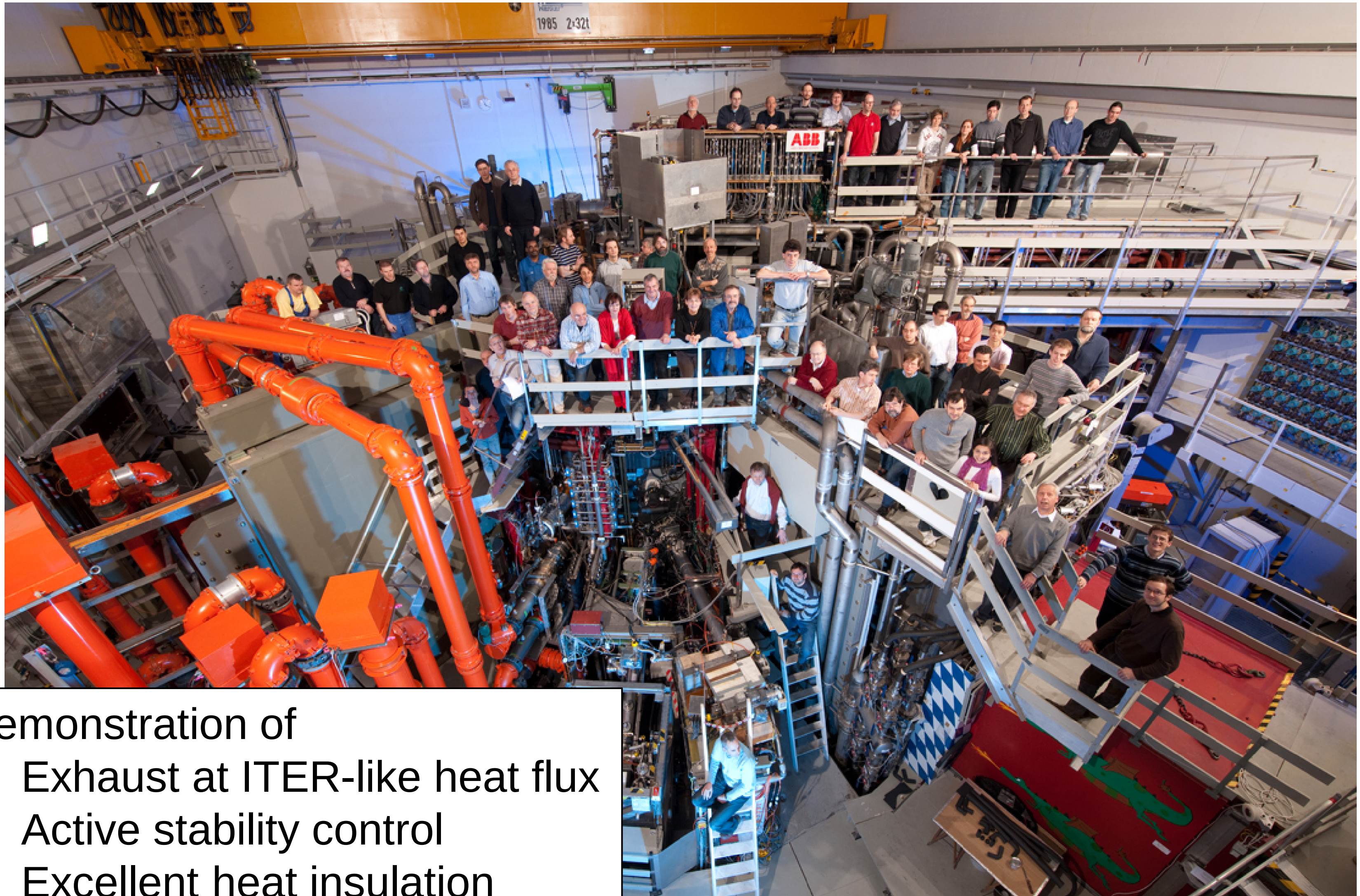






	ASDEX Upgrade (IPP)	***JET (EU)***	***ITER***
Major Radius	*1.65 m*	*3 m*	*6.2 m*
Volume	*14 m^3*	*80 m^3*	*800 m^3*
Fusion Power	*1.5 MW*	*~ 16 MW_{th}*	*~ 400 MW_{th}*
	(D-T equivalent)	*(D-T)*	*(D-T)*

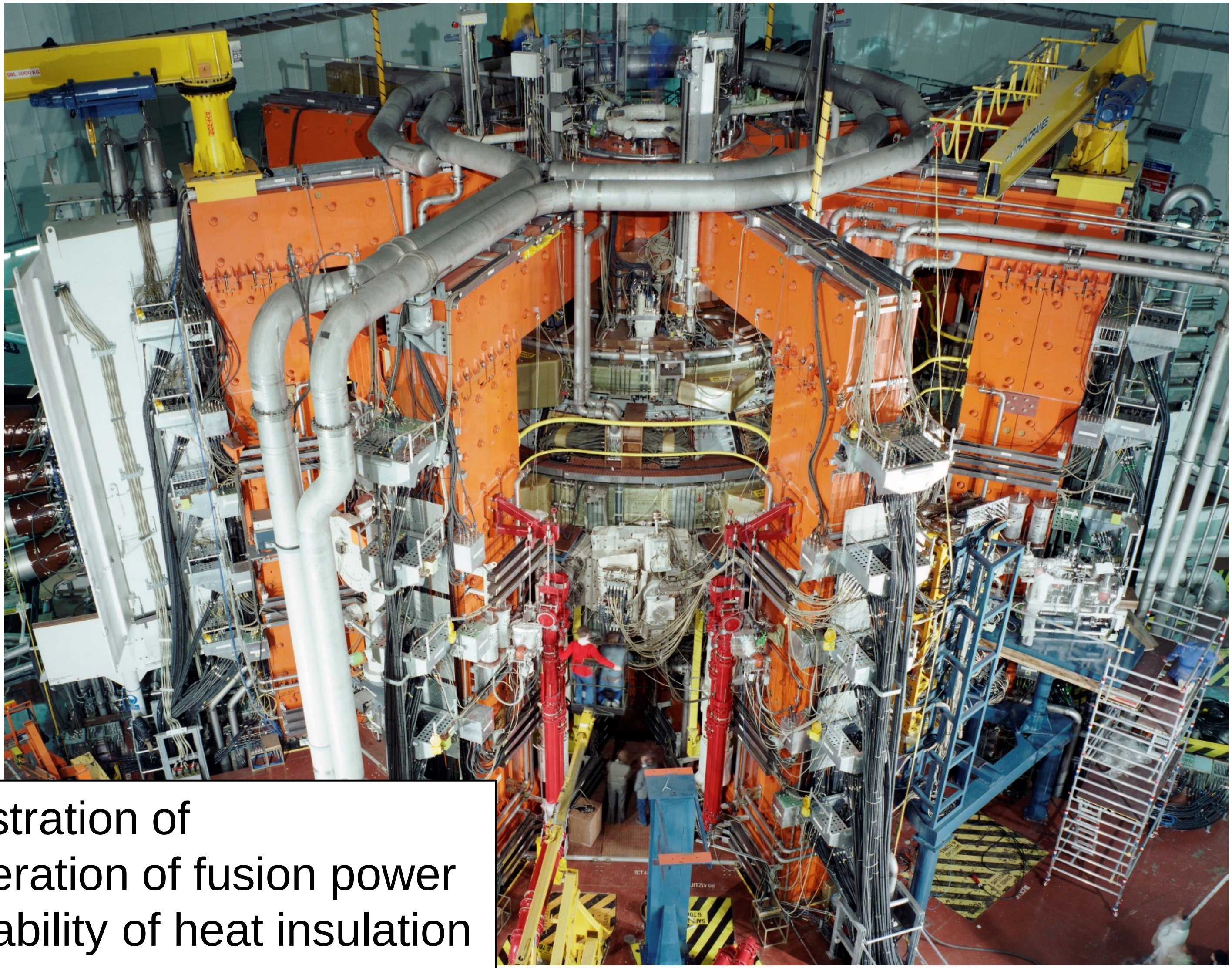
ASDEX Upgrade: ‚Pathfinder‘ for the larger devices



Demonstration of

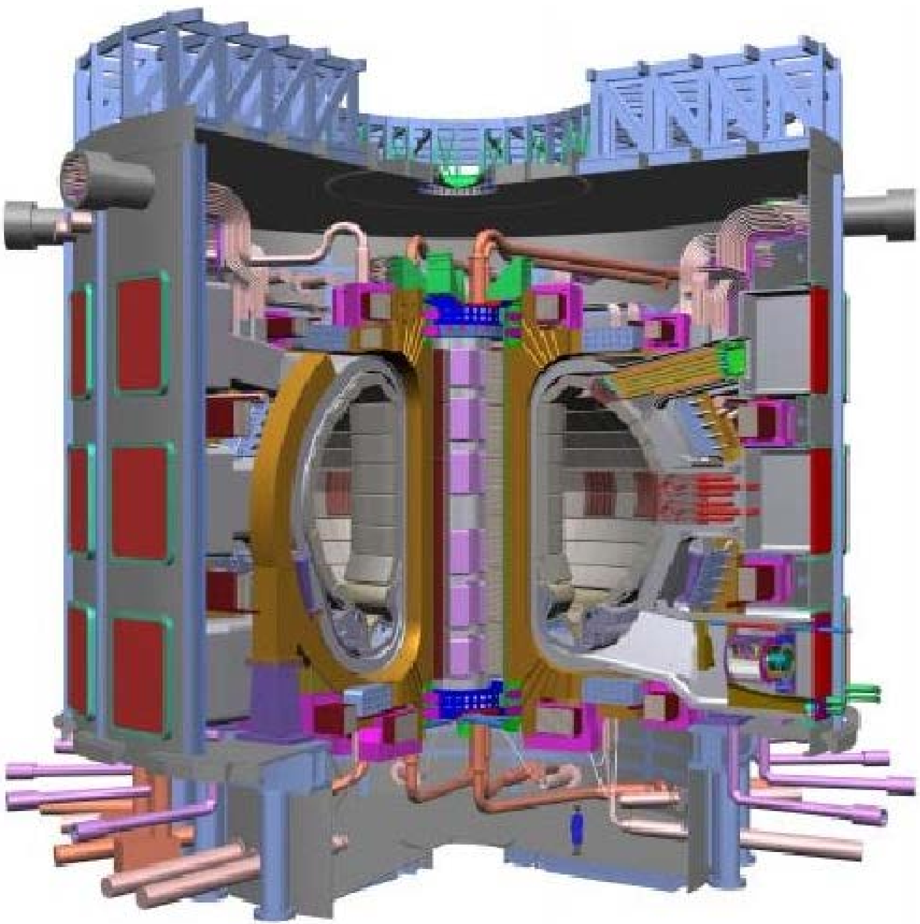
- Exhaust at ITER-like heat flux
- Active stability control
- Excellent heat insulation
- ...

JET: Flagship for tokamak performance



Demonstration of
- Generation of fusion power
- Scalability of heat insulation
- ...

ITER: Demonstration of Fusion Energy



	ITER
Major Radius	6.2 m
Minor Radius	2.0 m
Plasma current	15 MA
Magnetic field	5.3 T (Supercond.)
Power amplification Q	≥ 10
Fusion power	500 MW
Duration of burn	400 (3000) s
External heating	50 (73) MW

Cost: ~ 15 Billion €
Requires world-wide effort

ITER is being built in Cadarache (F) as joint effort – Cn, EU, In, Jp, Ko, RF, US

Partners build ITER through ‚In-kind' contributions

CS with US
(General Atomics)

Pre-compression rings: EU
(EADS CAS)

PF1: RF
(Efremov)

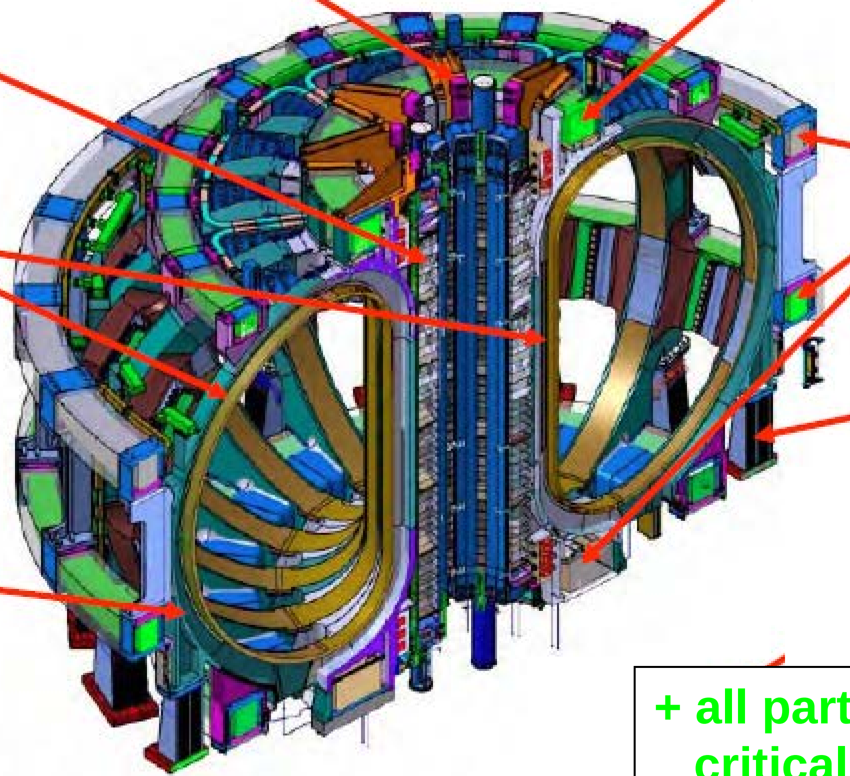
9+1 TF coils with EU
(Iberdrola/ASG/Elytt
Consort.)

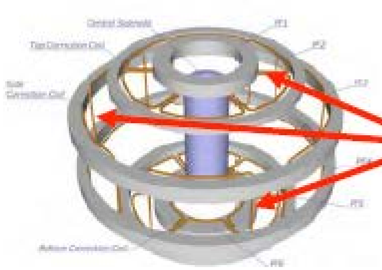
PF2-6: EU
(*not decided*)

9 TF coils with JA
(MHI)

Magnet Supports: CN
(SWIP)

19 TF coil cases: JA
(MHI/Hyundai
Consor



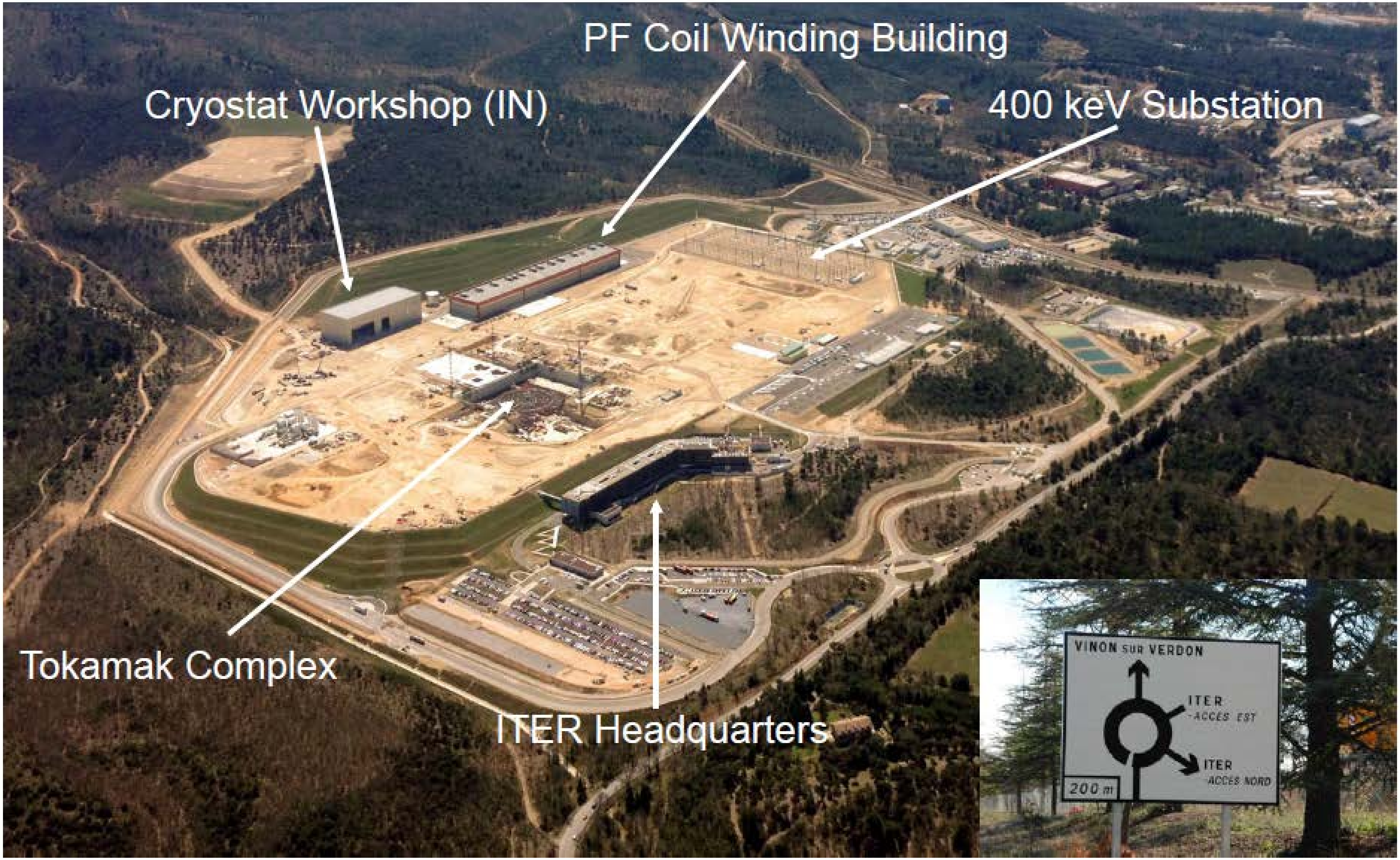


9 pairs of CCs: CN
(ASIPP)

+ all partners acquire the critical technical expertise

- very complex steering of the project

Partners build ITER through ‚In-kind‘ contributions



Partners build ITER through ‚In-kind' contributions

November 2015

Partners build ITER through ‚In-kind‘ contributions

November 2015

Partners build ITER through ‚In-kind' contributions

What will be ‚the physics of ITER‘?

Q=10 operation (self-heated plasma)

- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- exhaust of heat and particles
- self-heating by α-particles

Q=5 steady state operation

- self-sustainment of plasma current

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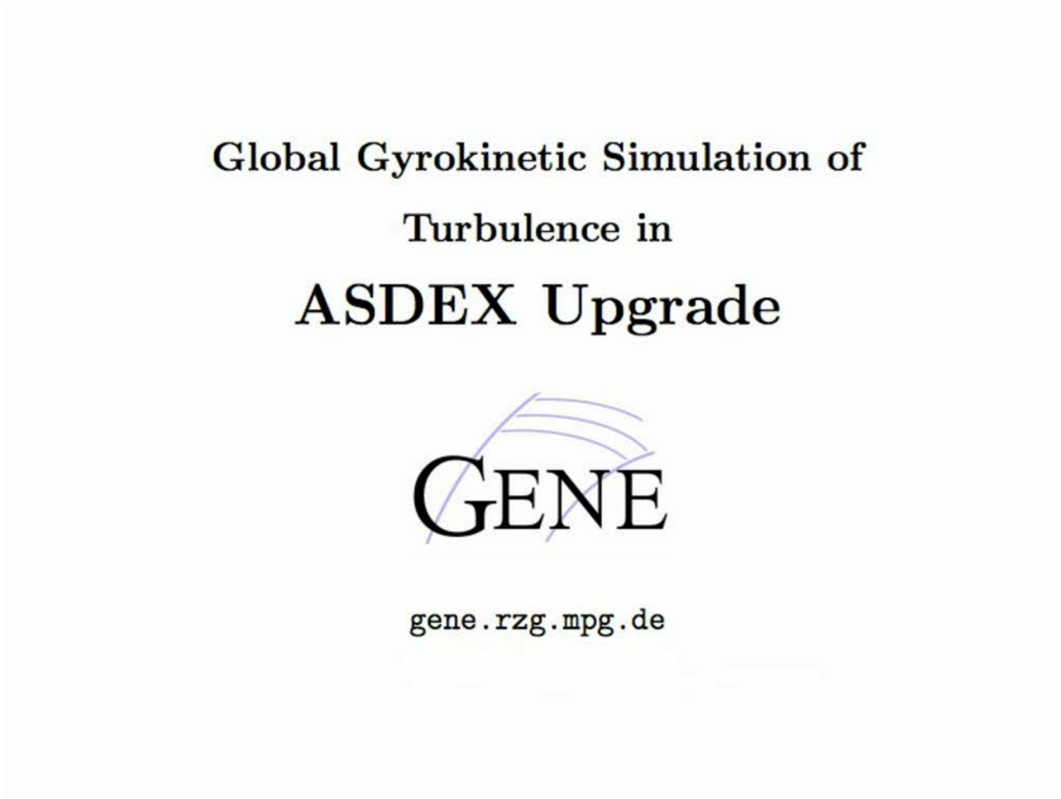
Energy Transport in Fusion Plasmas



Global Gyrokinetic Simulation of

Turbulence in

ASDEX Upgrade

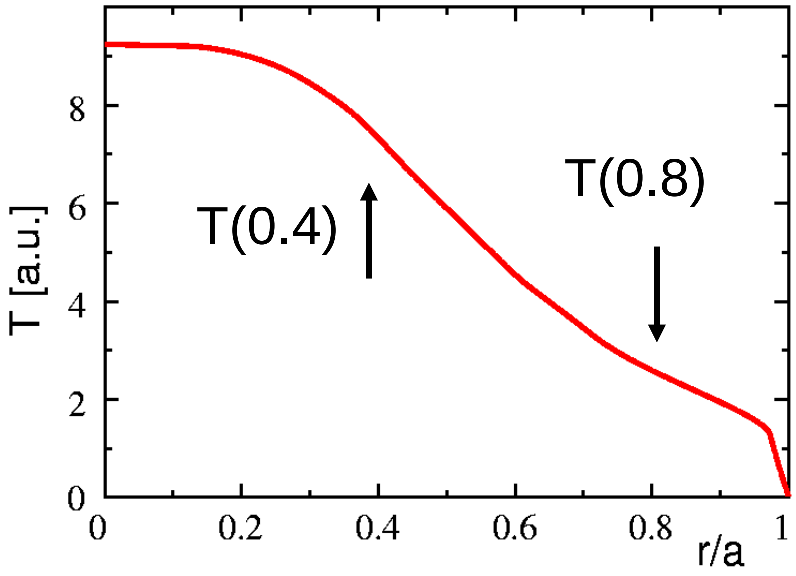
GENE

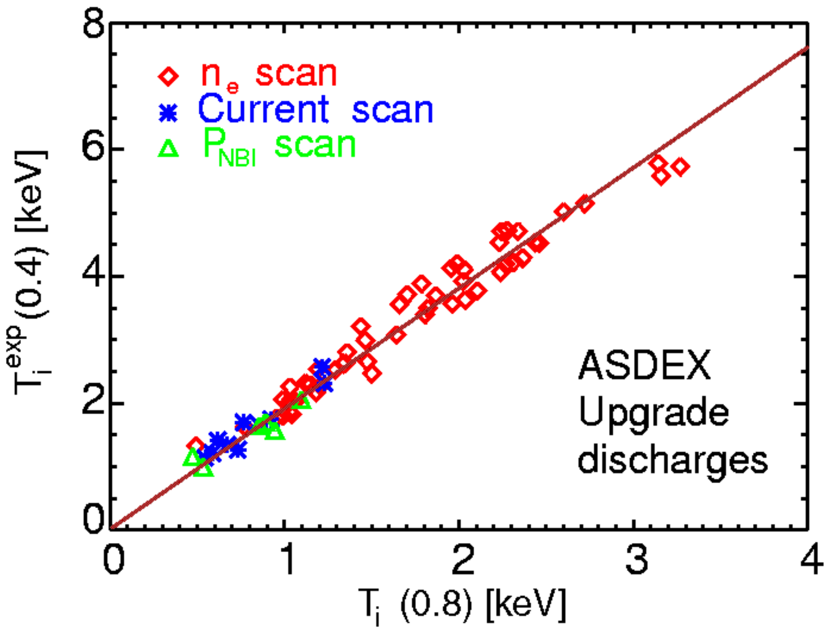
gene.rzg.mpg.de

Anomalous transport determined by gradient driven turbulence

- linear: main microinstabilities giving rise to turbulence identified
- nonlinear: turbulence generates ‘zonal flow’ acting back on eddy size
- (eddy size)^2 / (eddy lifetime) is of the order of experimental χ–values

Energy Transport in Fusion Plasmas





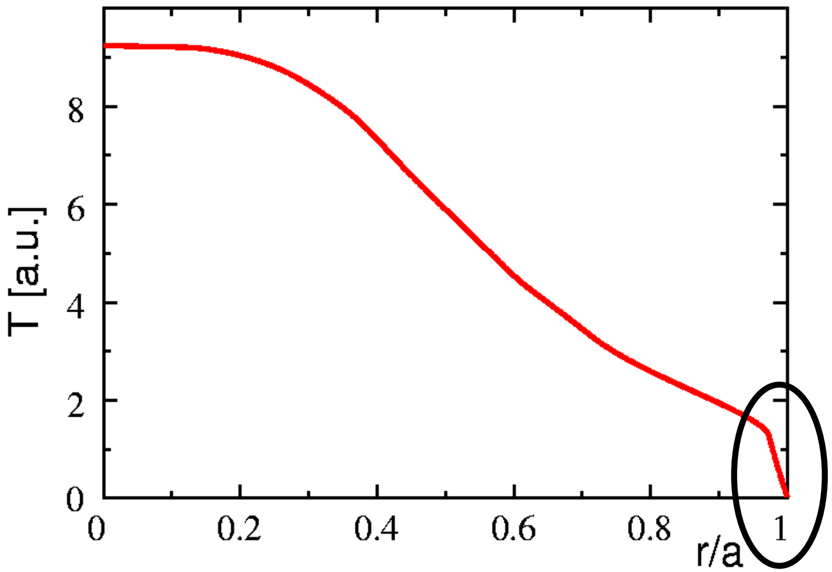
Anomalous transport determined by gradient driven turbulence

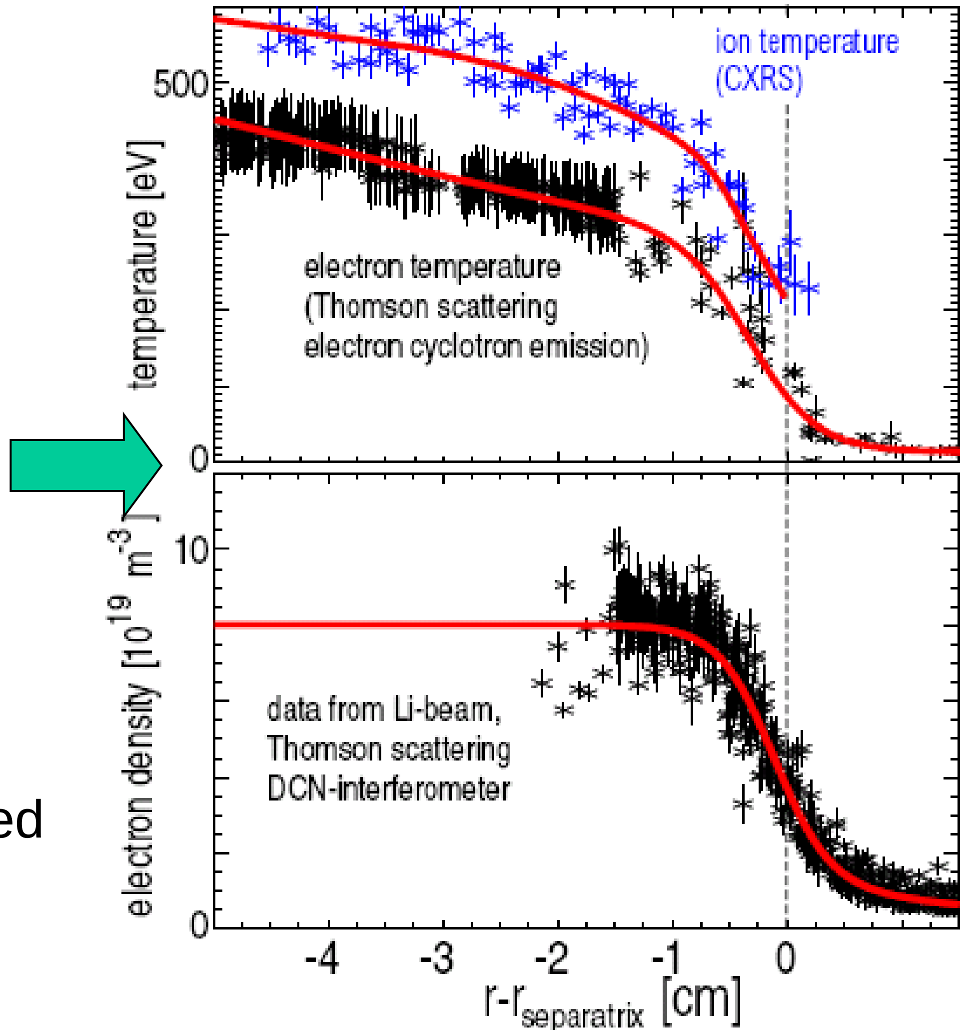
- temperature profiles show a certain ‘stiffness’
- ‘critical gradient’ phenomenon – χ increases with P_{heat} (!)

$\Rightarrow$ increasing machine size will increase central *T* as well as τ_E

N.B.: steep gradient region in the edge governed by different physics!

The H-mode: a transport barrier in the edge



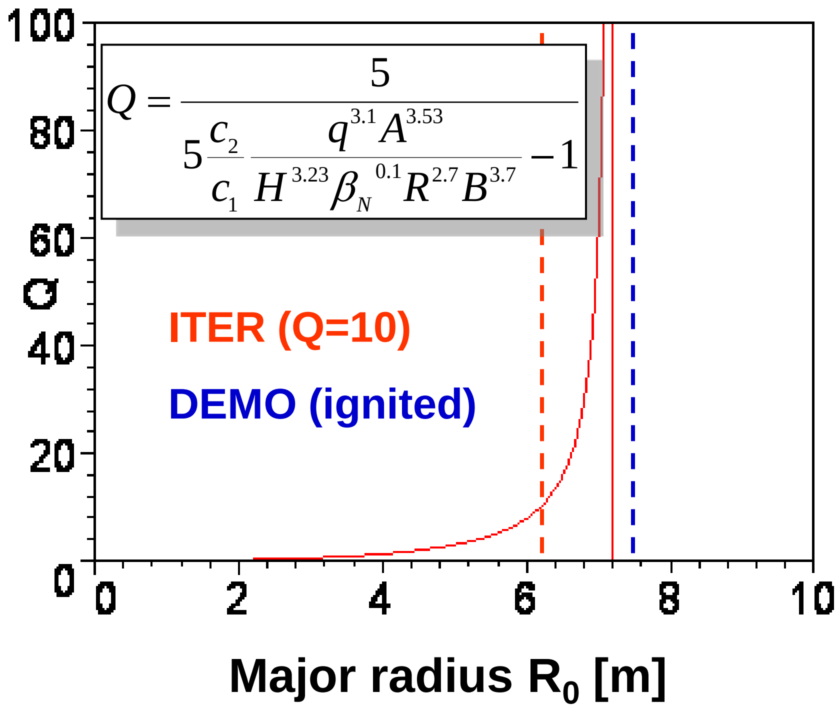


Improved confinement once input power exceeds a ‚threshold' P_{LH}

Turbulence suppression by sheared rotation in the plasma edge

- steep edge gradients of *T* and *n* (‚edge pedestal')
- due to profile stiffness, *T* is higher in whole plasma core

Anomalous transport determines machine size



- ITER Q=10, 500 MW goal crucially depends on H-mode confinment
- will be the ultimate confirmation of the size needed for a reactor

What will be ‚the physics of ITER'?

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Q=5 steady state operation

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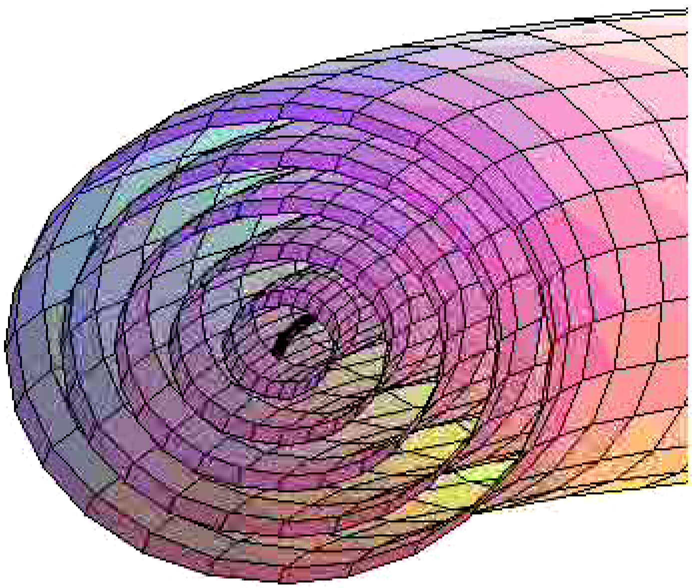
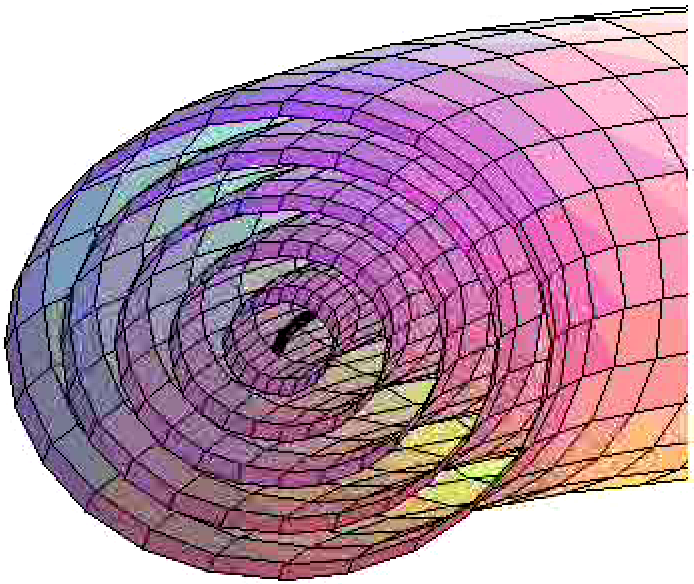
Ideal and resistive MHD instabilities





Ideal MHD: $\eta = 0$

- flux conservation
- topology unchanged

Resistive MHD: $\eta \neq 0$

- reconnection of field lines
- topology changes

Ideal MHD instabilities limit achievable pressure

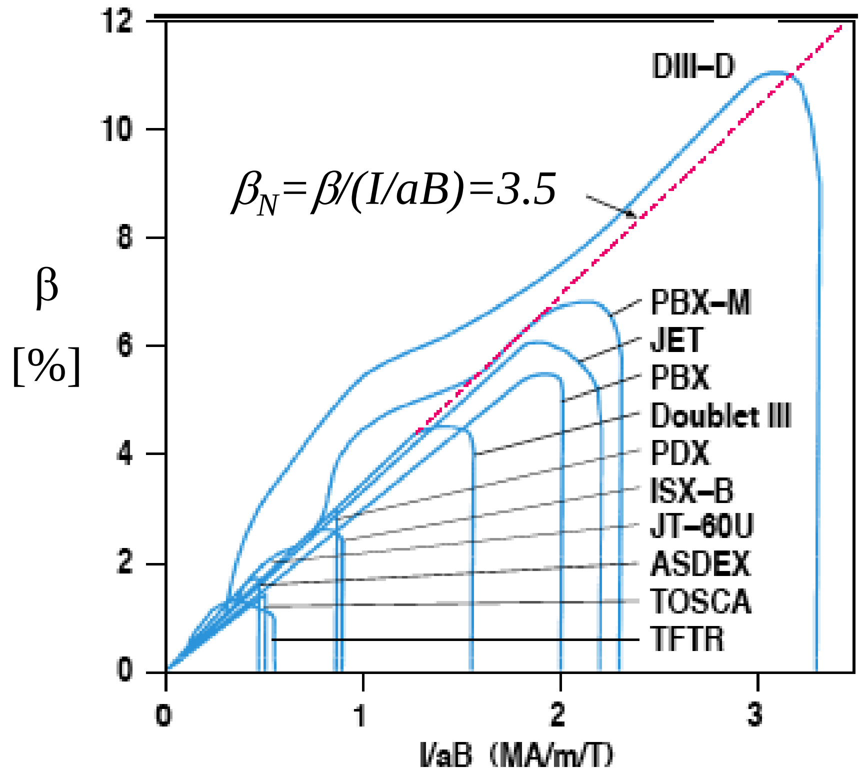
Optimising $nT\tau_E$ means high pressure $p=nT$ and, for given magnetic field, high dimensionless pressure $\beta = 2\mu_0 <p> / B^2$

This quantity is ultimately limited by ideal instabilities

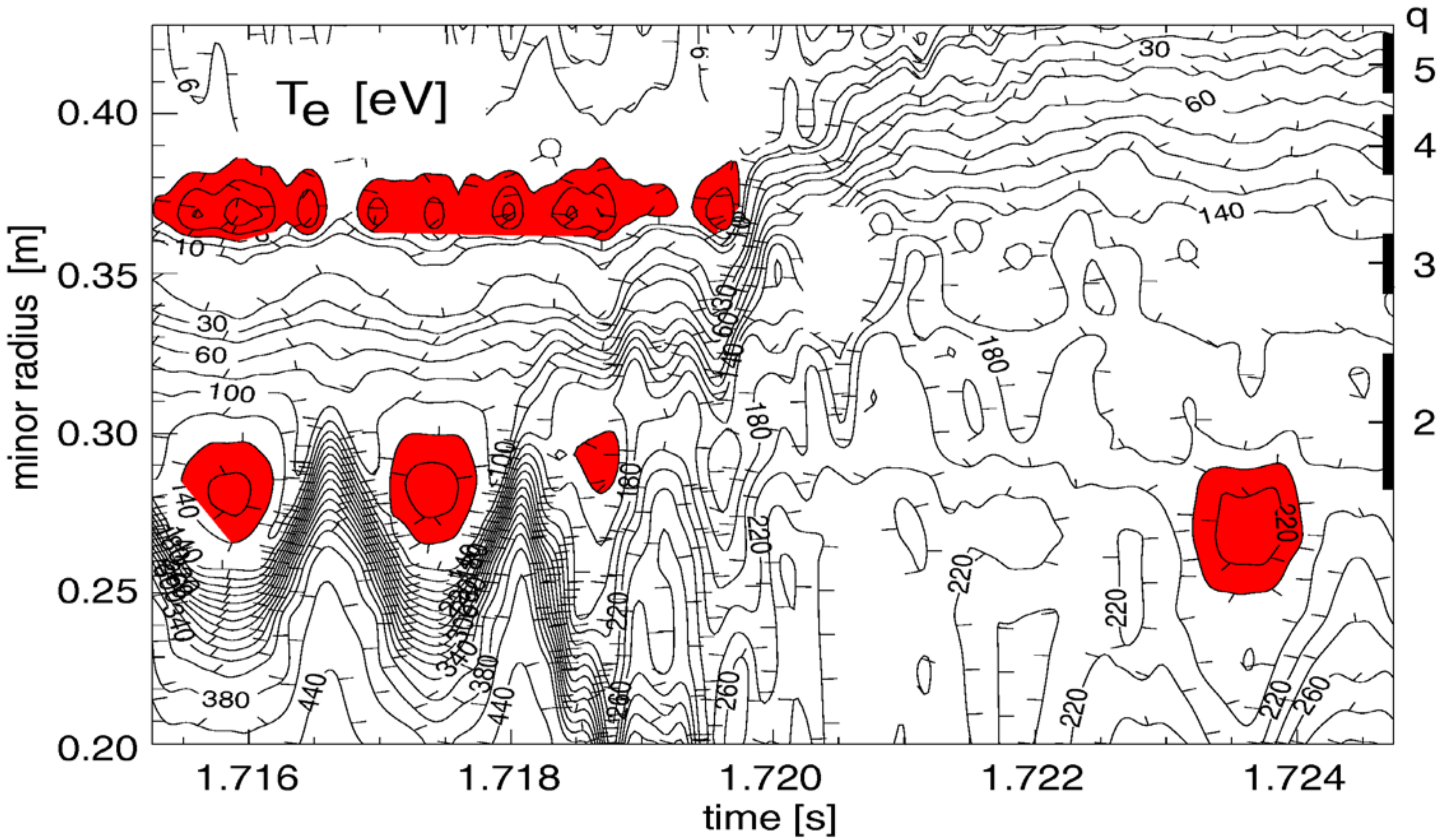
'Ideal' MHD limit (plasma unstable on Alfvén time scale, limited by inertia)

- 'Troyon' limit $\beta_{max} \sim I_p/(aB)$, leads to definition of $\beta_N = \beta/(I_p/(aB))$
- ITER Q=10 operation planned at β_N=1.8, i.e. well below ideal limit

Note: resistive MHD instabilities (‚Neoclassical Tearing Modes') might occur at even these low β-values!



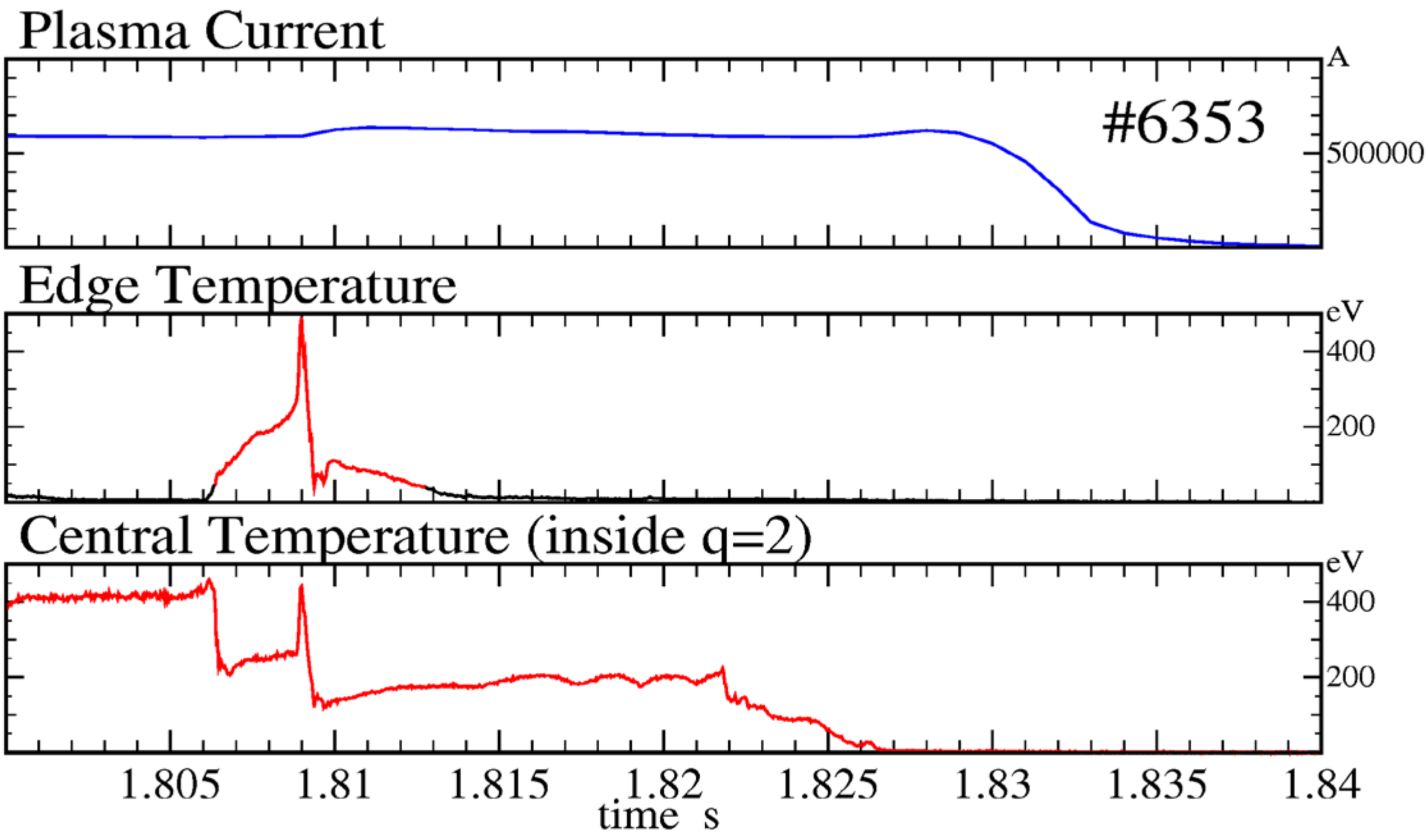
Resistive MHD: magnetic islands in tokamaks



coupling between island chains (possibly stochastic regions)

⇒ sudden loss of heat insulation ('disruptive instability')

Disruptive instability limits achievable density



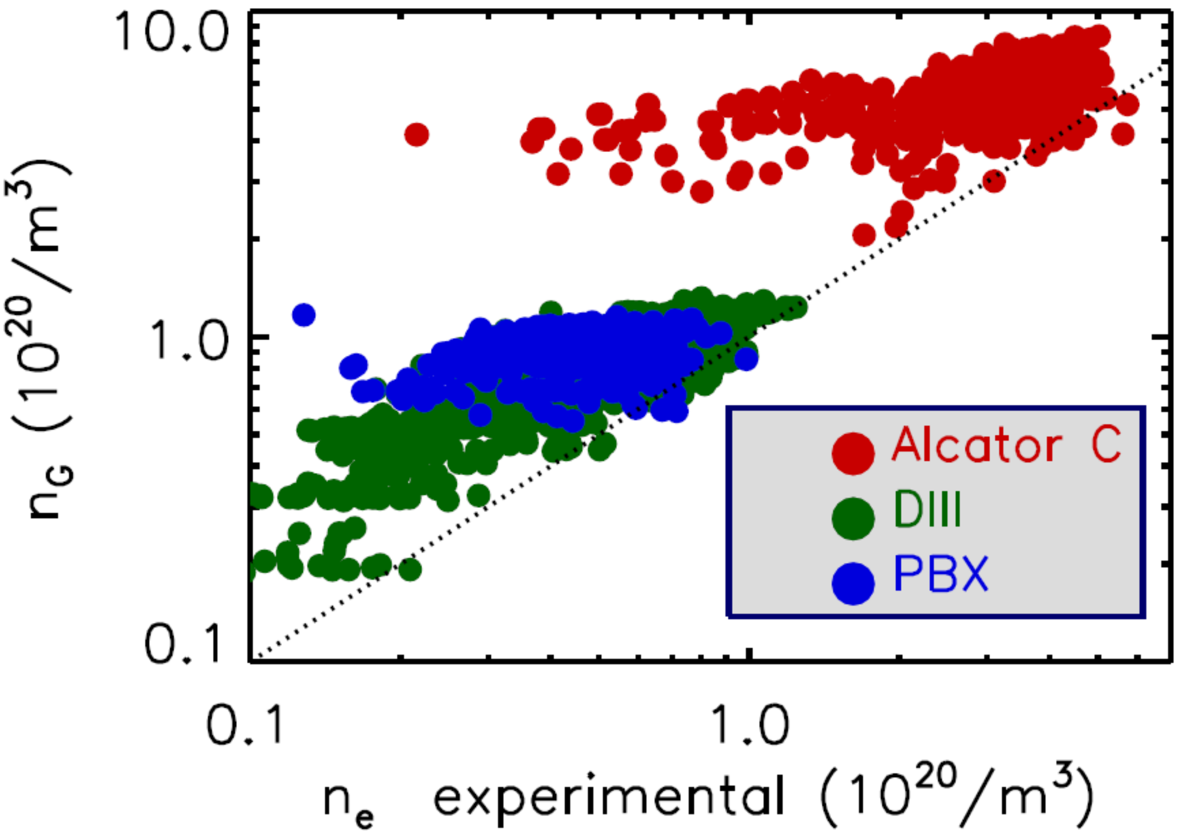
High density clamps current profile and leads to island chains

excessive cooling, current can no longer be sustained

disruptions lead to high thermal and mechanical loads!

Disruptive instability limits achievable density

Disruptive instability limits achievable density

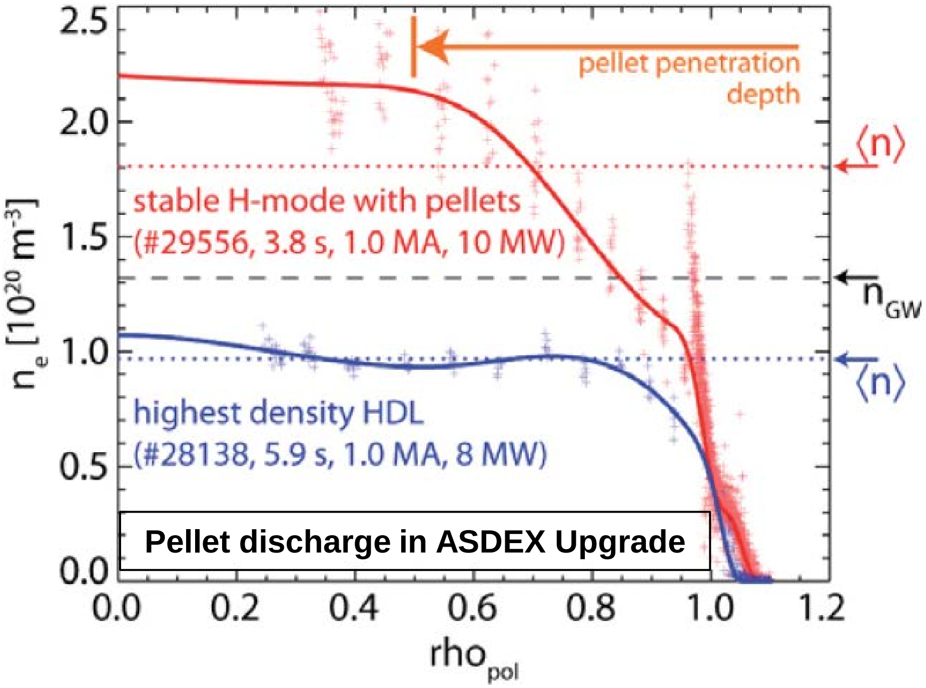


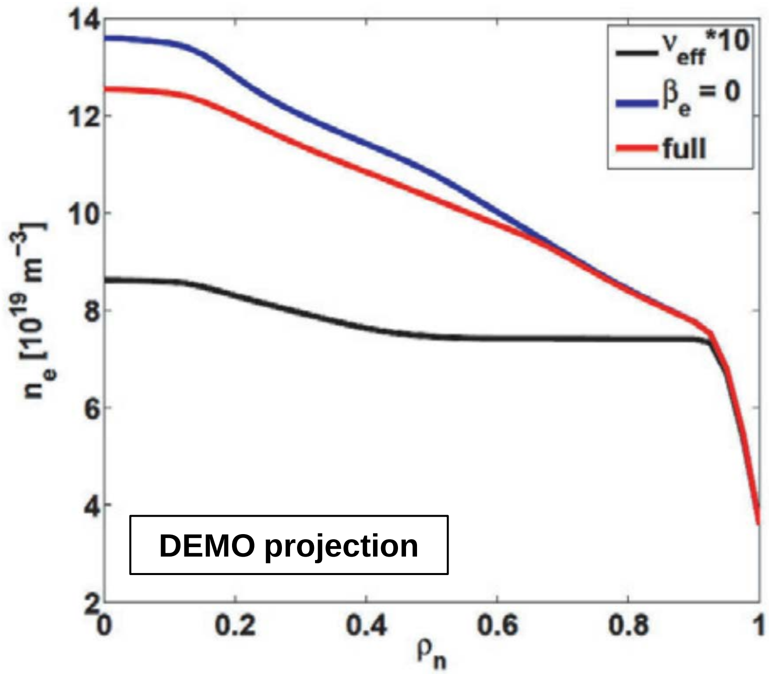
$n_G = I_p/(\pi a^2)$

Empirical ‚Greenwald-limit' describes well maximum density

- seems to be linked to a change in edge transport at $n \sim n_G$
- can be overcome if density profile shape is varied (peaked)

Operation above the Greenwald limit





- present devices: peaking at high n/n_{GW} only for pellet injection
- future devices: expect to see peaked density profiles due to nomalous inward pinch at low collisionality ν^*
- ITER will verify (or not) our assumptions about density peaking in reactors (operation above n_{GW} not absolutely necessary in ITER)

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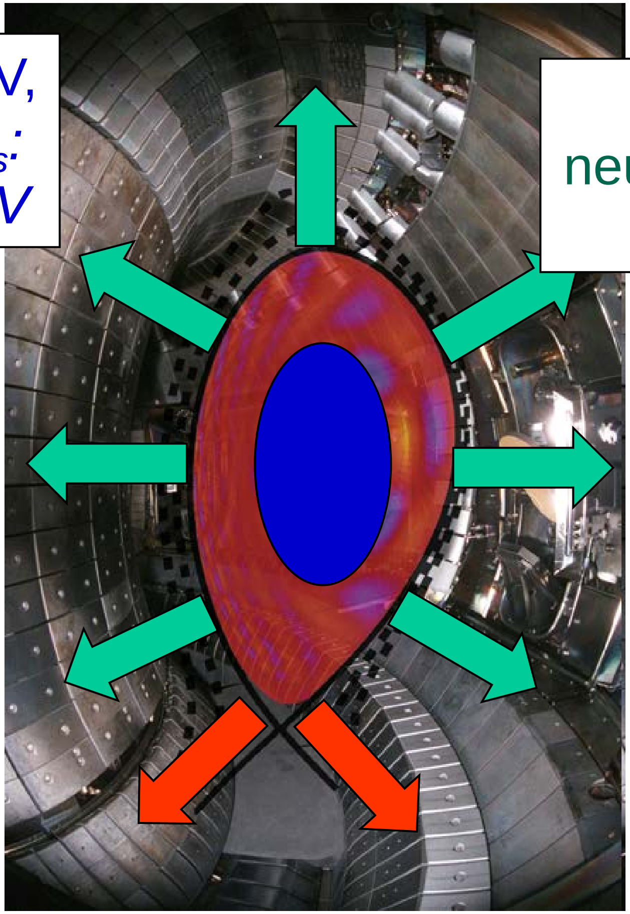
Q=5 steady state operation

- self-sustainment of plasma current

Fusion Reactor Energetics

Core plasma @ T=25 keV, $n=10^{20}$ m^{-3} produces P_{fus}:
$D+T = He + n + 17.6\ MeV$

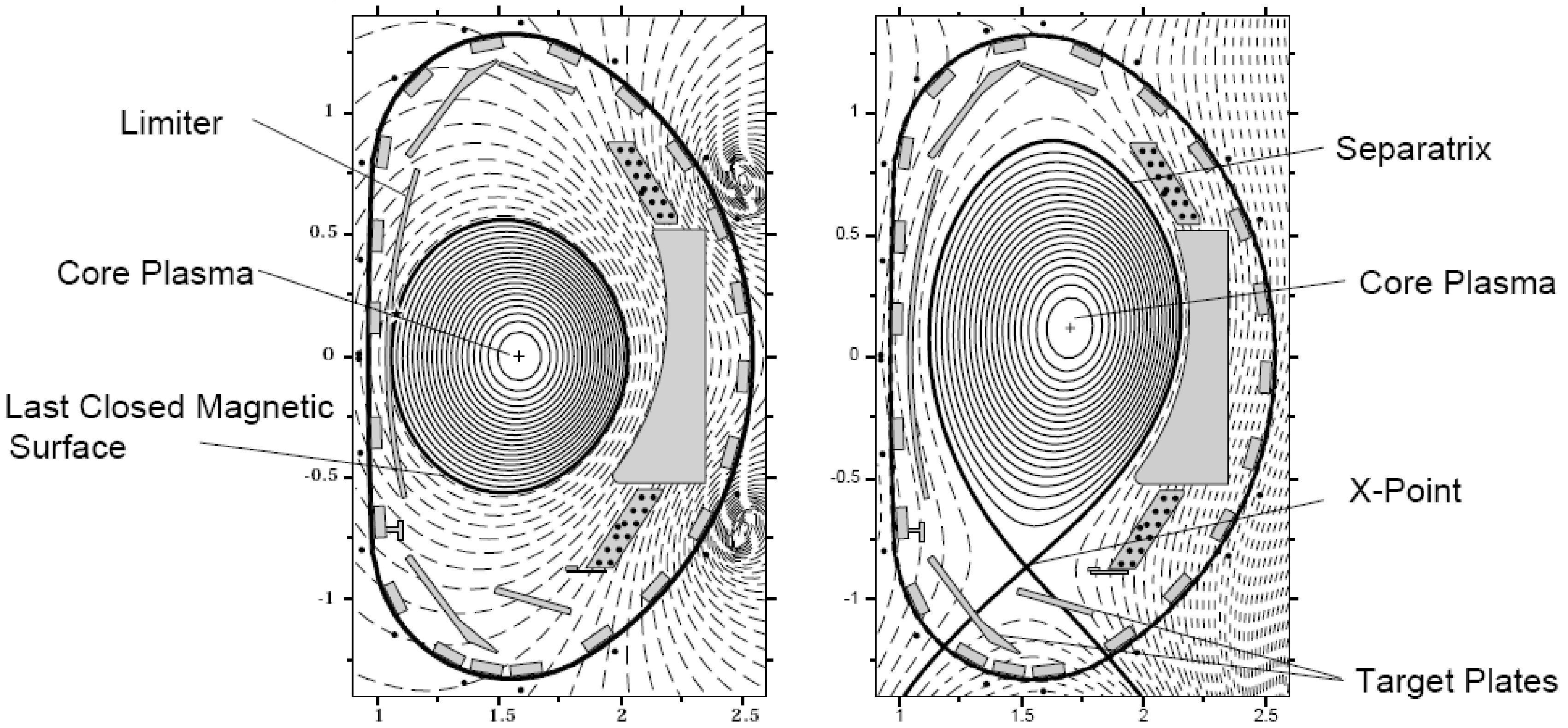


$4/5*P_{fus}$ escape as neutrons and hit the first wall (typically 1 MW/m^2)

$1/5*P_{fus} + P_{ext}$ escape in charged particles along B-field lines and hit the wall in a narrow band (without further mitigation, 100 MW/m^2 !)

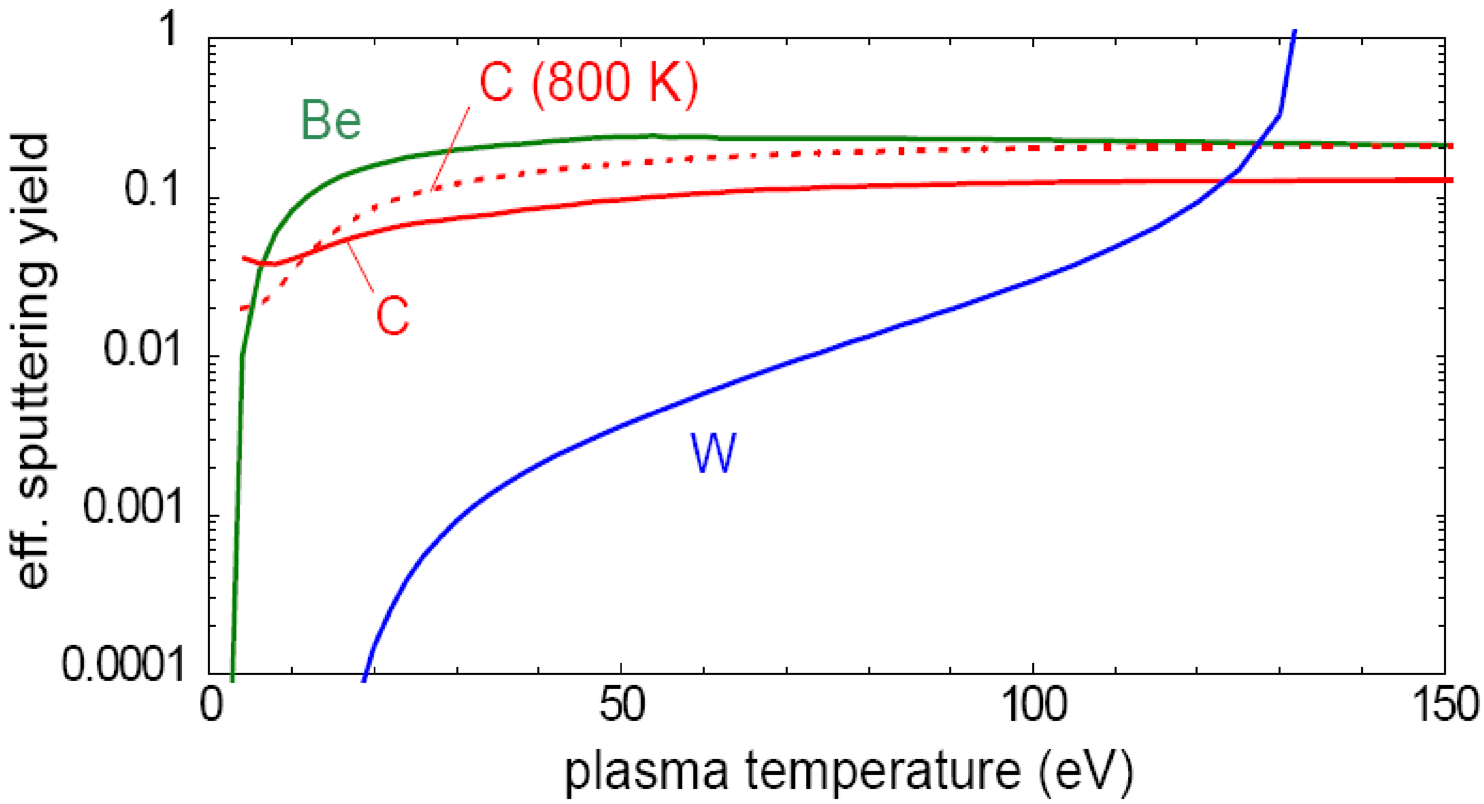
(heat flux on the surface of the sun ~ 60 MW/m^2)

Plasma wall interface – from millions of K to 100s of K



- plasma wall interaction in well defined zone further away from core plasma
- eases particle control (retention of impurities, pumping of He ash)
- along field lines, *T* drops and *n* increases ⇒ decrease particle impact

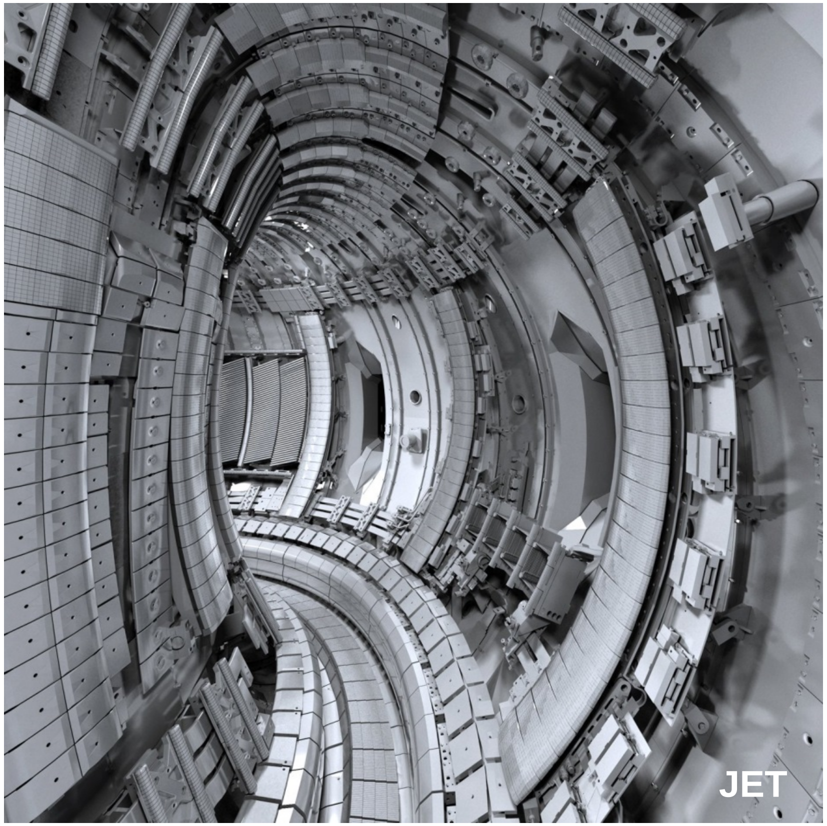
The perfect wall material: Low-Z or High-Z?

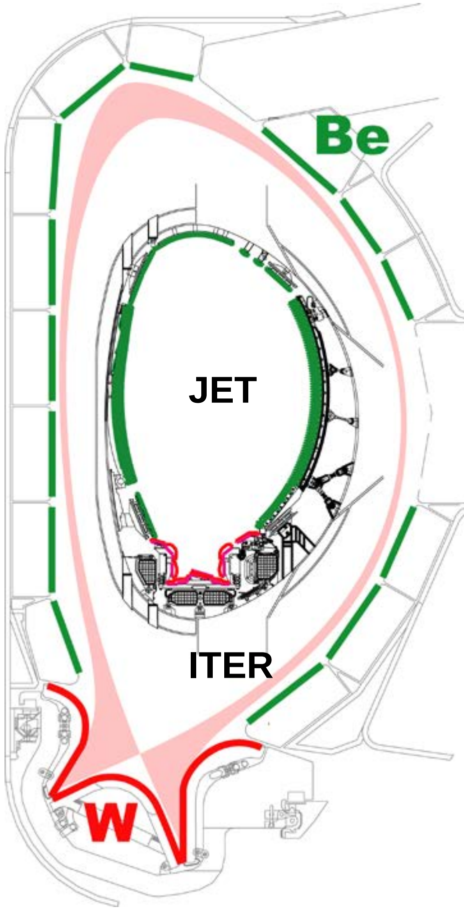


High-Z materials (W, Mo) promise low erosion rates and fuel retention

- if edge temperature is low enough (i.e. below ~ 20 eV)

Tokamak operation with metal wall



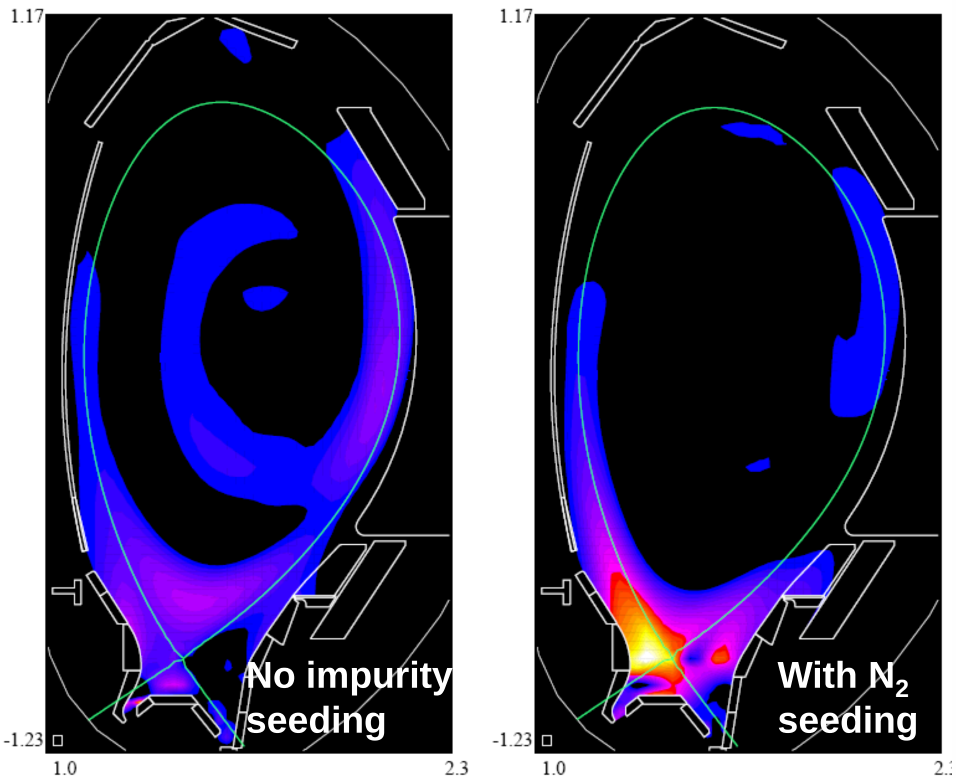


Divertor allows use of W under reactor relevant plasma conditions

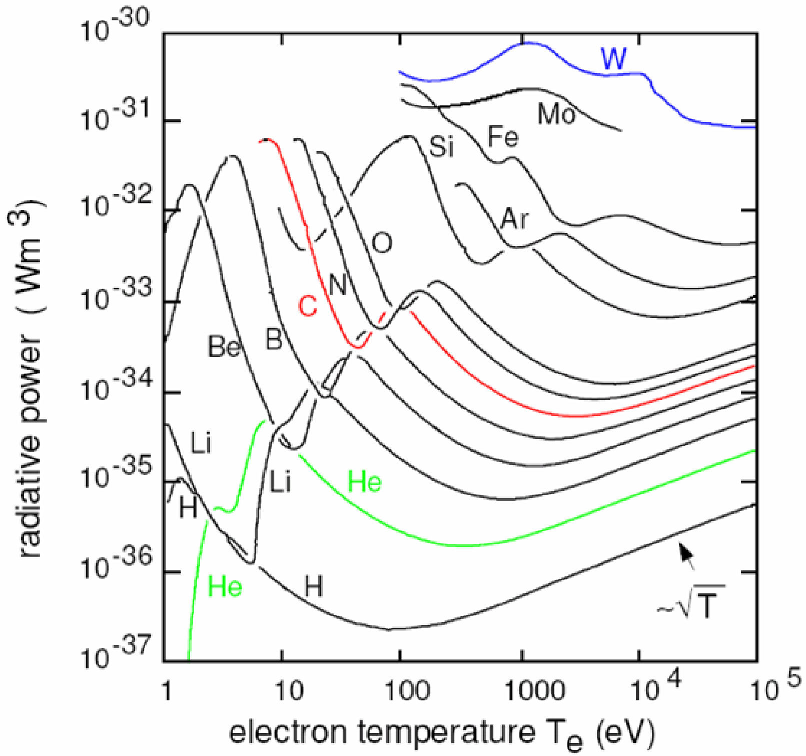
- capitalises on low divertor temperatures that lead to negligible erosion
- needs special care to avoid excessive W content in plasma

Additional cooling by impurity seeding

Bolometry of total radiated power



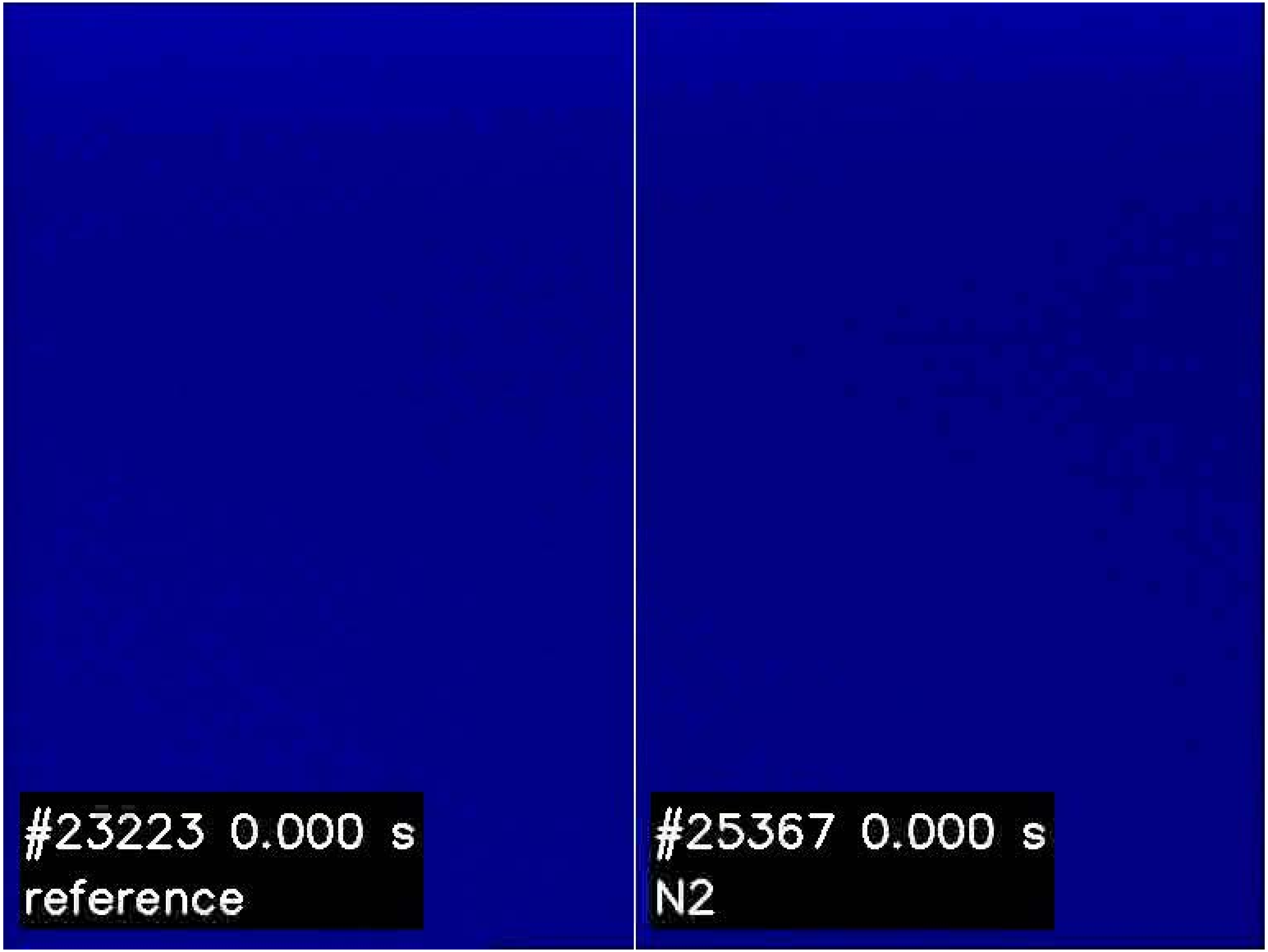
radiation parameter



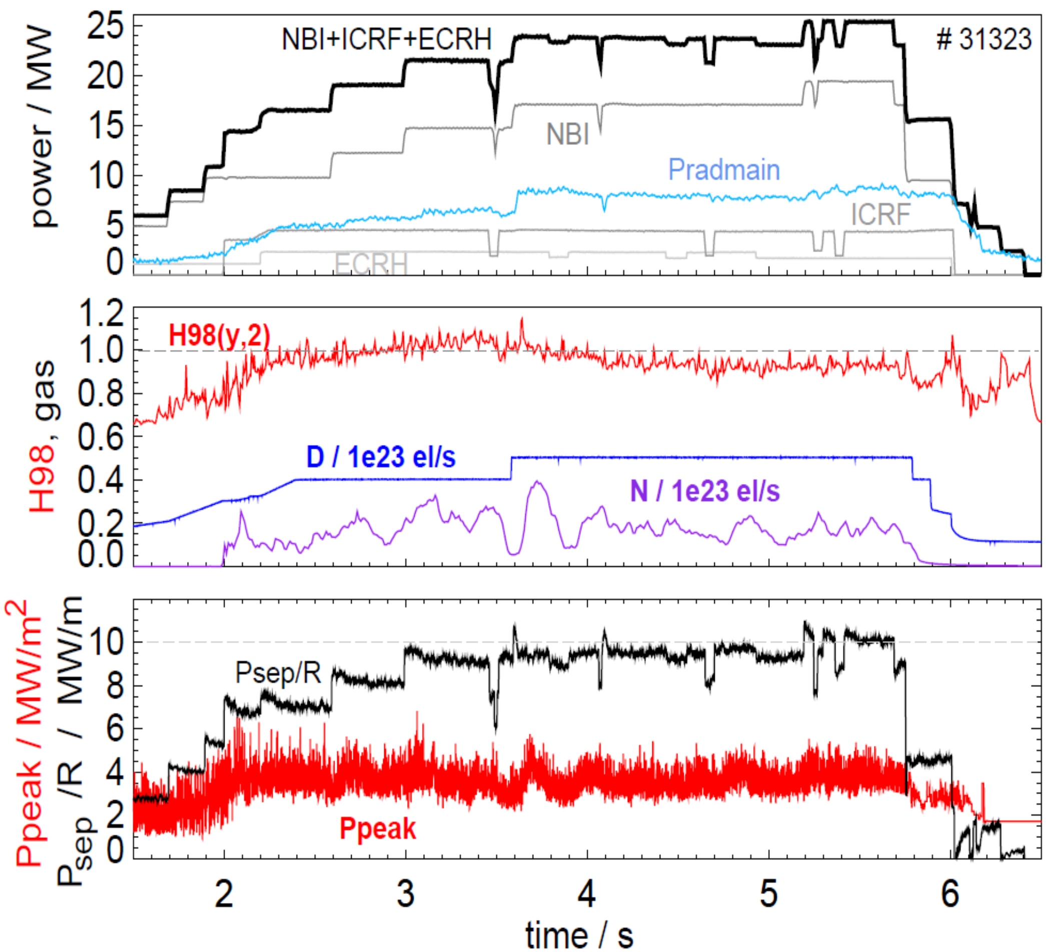
Injecting adequate impurities can significantly reduce divertor heat load

- impurity species has to be ‘tailored’ according to edge temperature
- edge radiation beneficial, but central radiation must be avoided

Additional cooling by impurity seeding



Additional cooling by impurity seeding



ASDEX Upgrade discharge applying N-cooling at 2/3 of Normalised ITER power flux (P_{sep}/R=10 MW/m)

‚Detachment': plasma becomes so cold that it recombines in front of the divertor plates

ITER will verify applicability of the conventional divertor, run in 'detached' mode, for reactor-grade heat fluxes through the plasma boundary

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The next step: studying α-heating

ITER plasma parameters sufficient to generate significant fusion power

- study plasmas with significant self-heating by α-particles
- needs $P_\alpha = 1/5\ P_{fus} >> P_{ext}$, so it necessarily is closer to a reactor

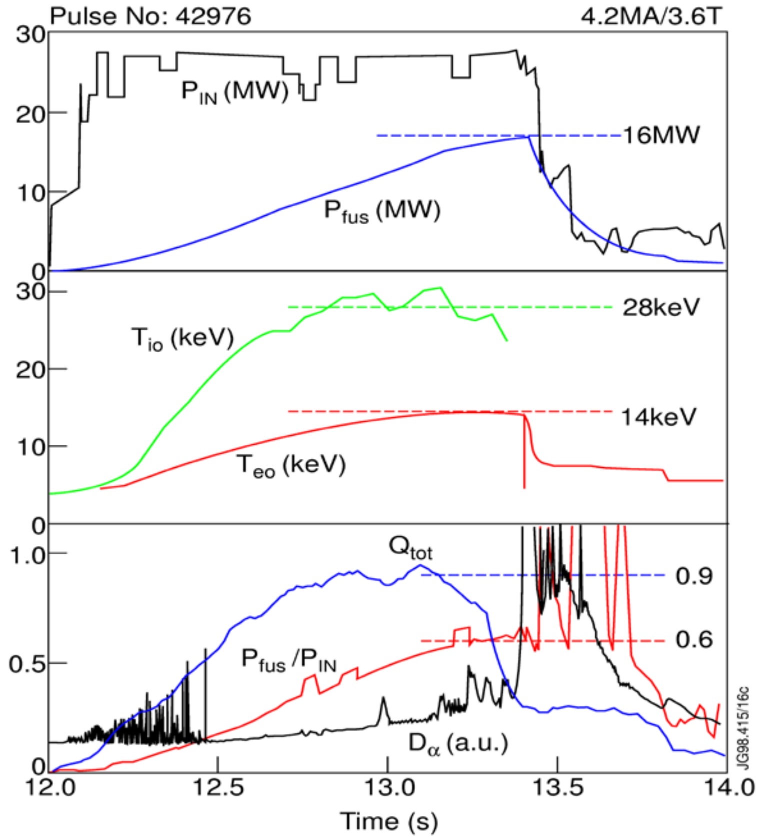
We expect to see qualitative new physics:

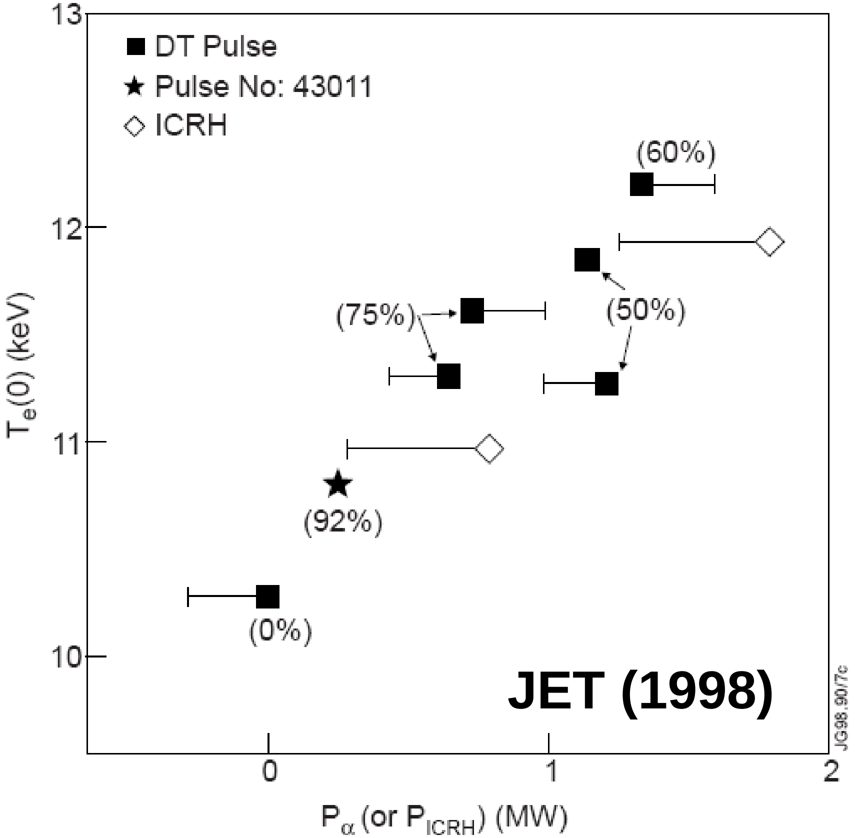
- self-heating nonlinear ($P_{fus} \sim n^2T^\gamma$ *and* $T \sim P_{fus}{}^\delta$) - interesting dynamics
- suprathermal population of α-particles can interact with plasma waves

We can have a ‘preview’ in present day machines

- pilot D-T experiments (JET (EU), TFTR (US))
- suprathermal ions generated by heating systems simulate α-particles

Previous D-T experiments

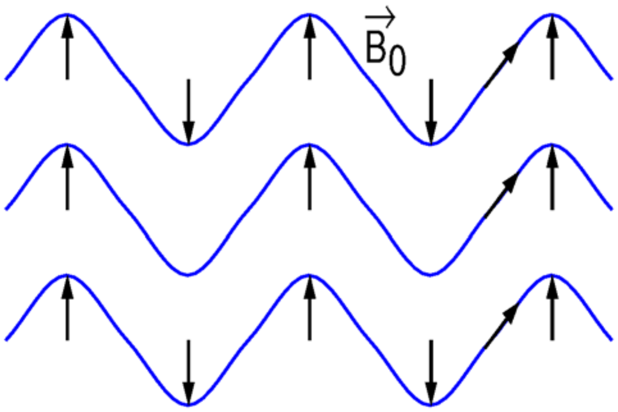




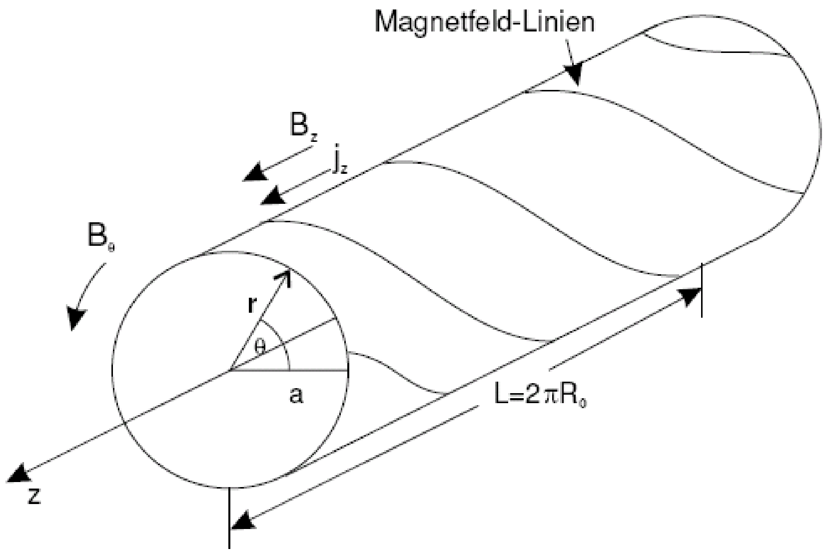
First D-T experiments at low P_α/P_{tot} have demonstrated α-heating

- ‚classical' (=collisional) slowing down would guarantee efficient α-heating
- question: can we expect this also when P_α is the dominant heating?

Alfven waves – continuum and gap modes



torsional (shear) Alfven wave



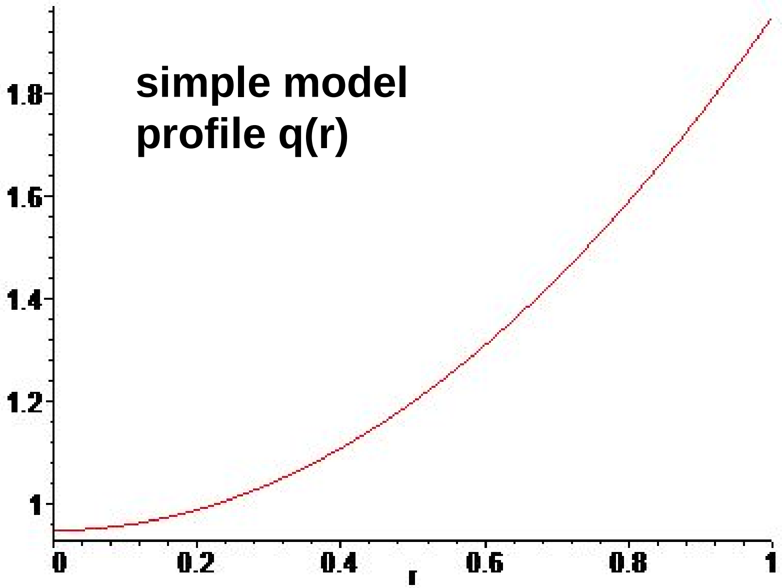
$$\boxed{v_A = \frac{B}{\sqrt{\mu_0 m_i n}}}$$

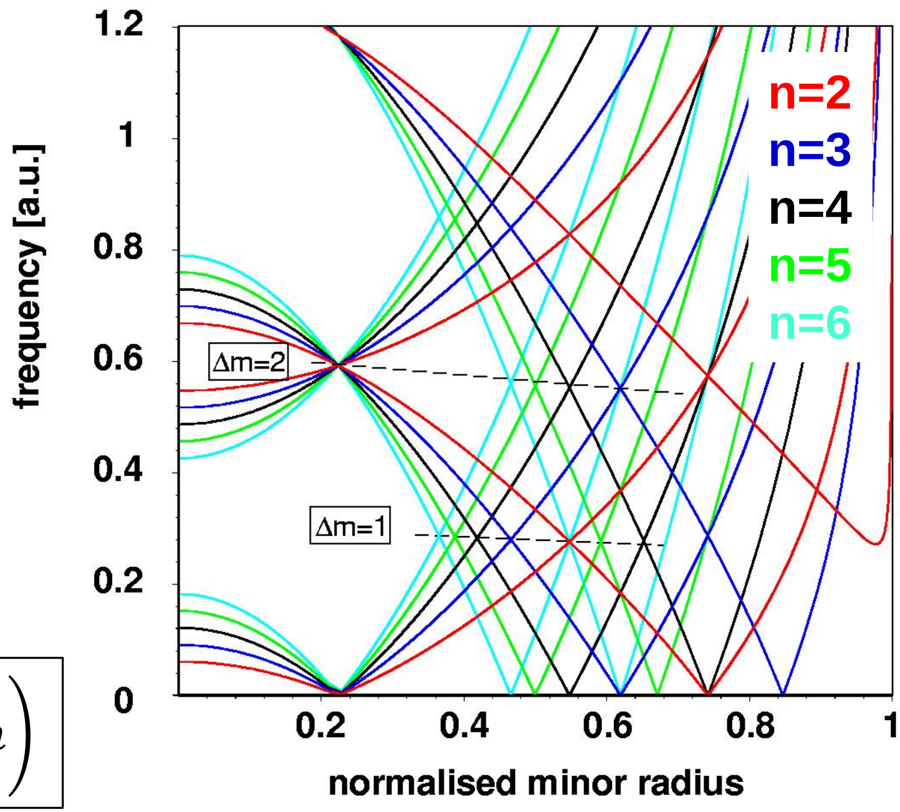
$$\boxed{f(r,\theta,z) = \widehat{f}(r)\exp(i(m\theta - nz/R_0))}$$

B-field lines in a plasma can oscillate like a string of a guitar

- for a a mode with mode numbers *m, n*: $k_{||} = (m/q(r)-n)/R$

Alfven waves – continuum and gap modes





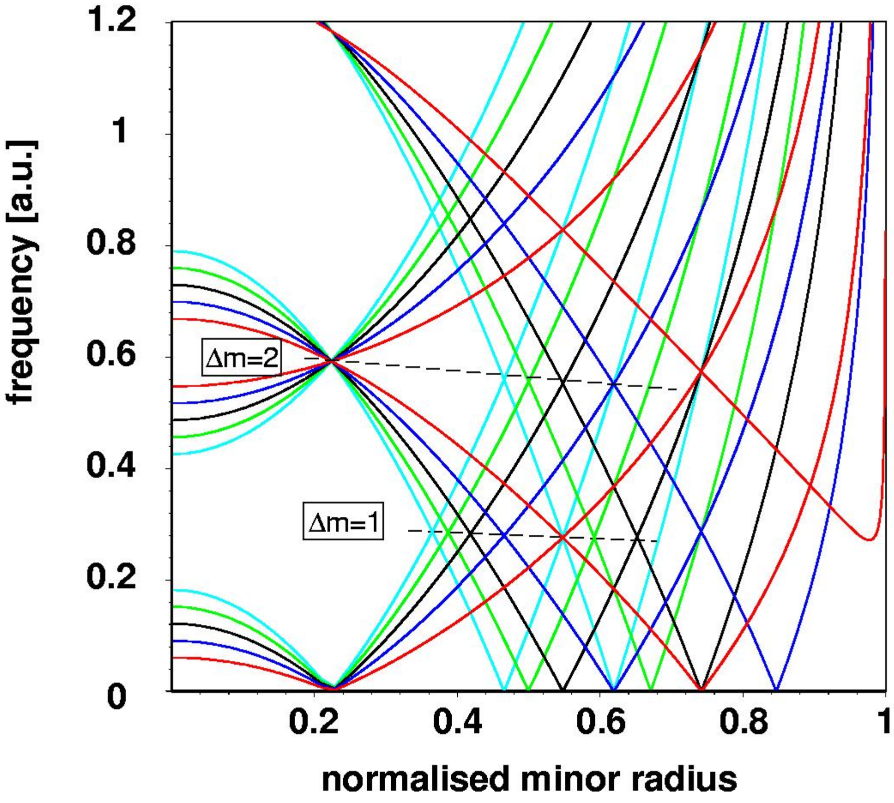
$$v_A = \frac{B}{\sqrt{\mu_0 m_i n}}$$

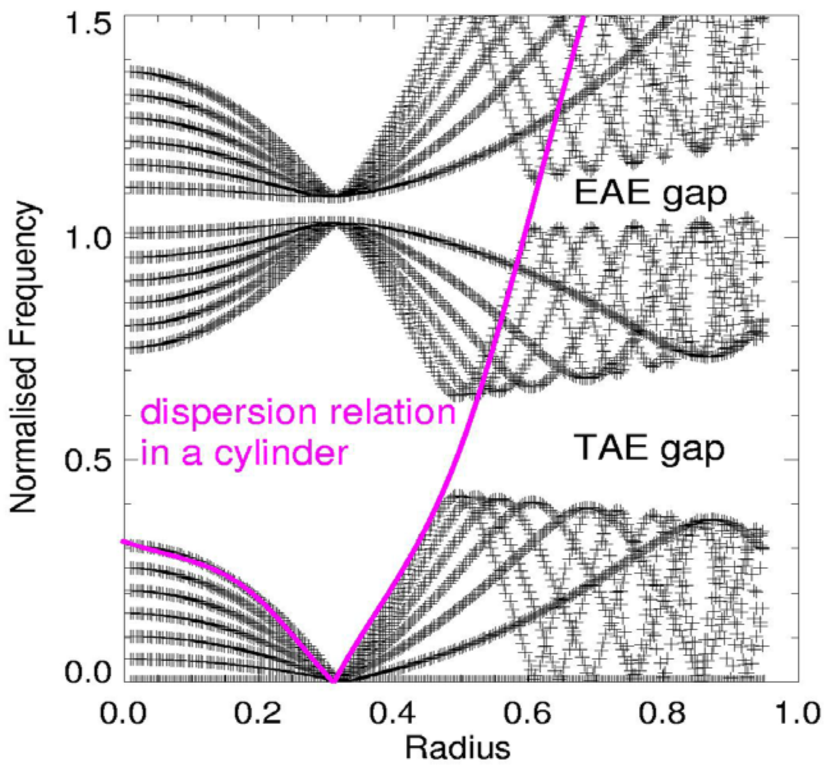
$$k_{||} = \frac{1}{R}\left(\frac{m}{q(r)} - n\right)$$

B-field lines in a plasma can oscillate like a string of a guitar

- double periodic cylinder: $\omega = k_{//} v_A$ gives continuum structure
- this leads to strong damping of the modes (radial variation of ω)

Alfven waves – continuum and gap modes

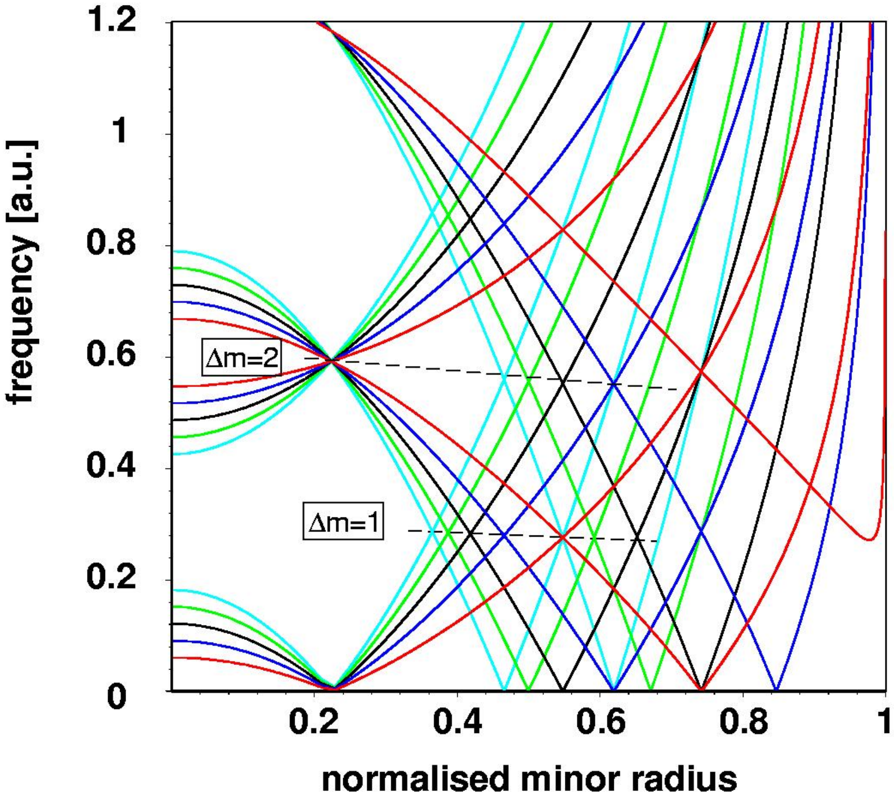


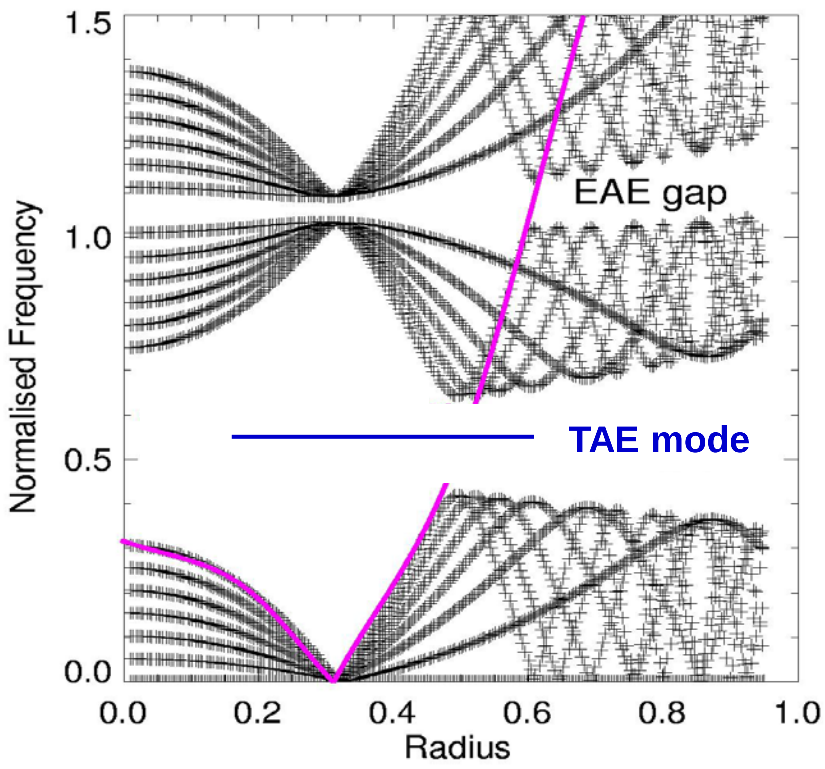


B-field lines in a plasma can oscillate like a string of a guitar

- double periodic cylinder: $\omega = k_{//} v_A$ gives continuum structure
- in a torus, gaps open that allow Alfven resonances to extend over radius

Alfven waves – continuum and gap modes





B-field lines in a plasma can oscillate like a string of a guitar

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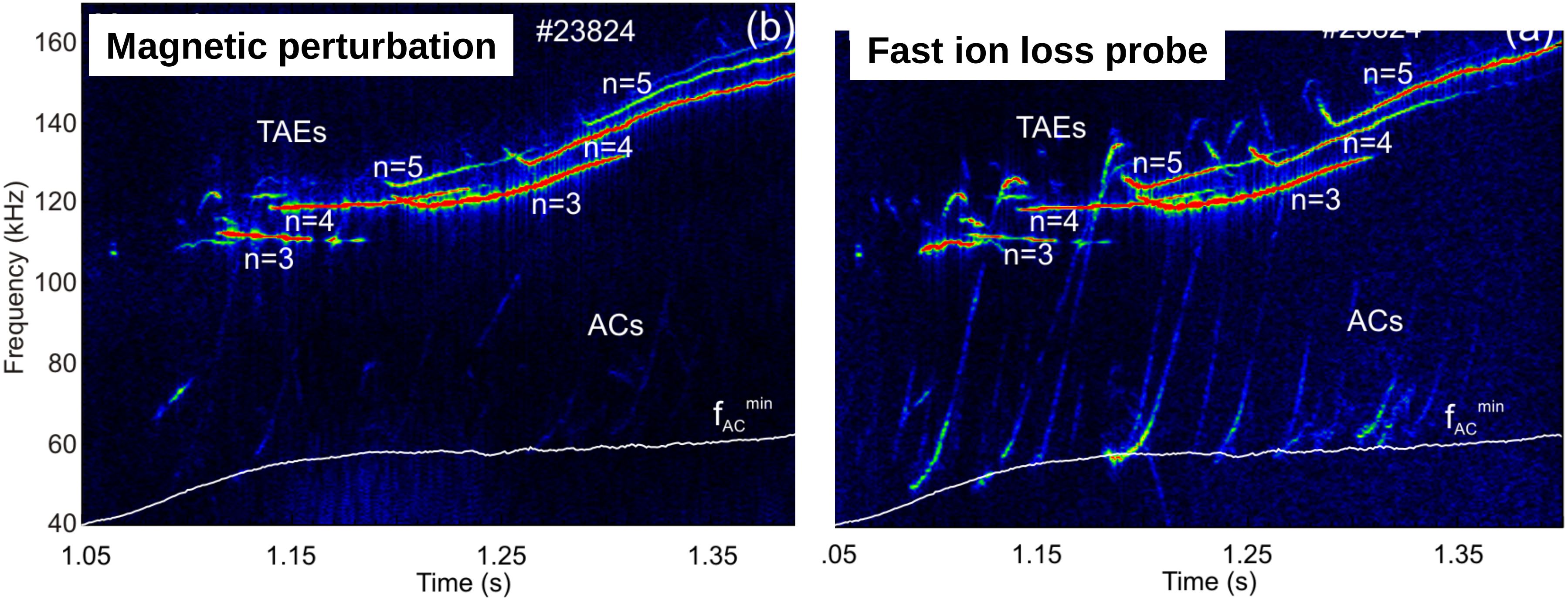
Excitation of Alfvén waves by Fast Particles



Suprathermal ions with $v \approx v_A$ can excite Alfvén waves which expel them

- in present day experiments, these ions come from heating systems
- in future reactors, this could expel α-particles that should heat the plasma!

Study of nonlinear interaction between waves/instabilities and suprathermal particles will be one of the main items of ITER physics

What will be ‚the physics of ITER'?

Q=10 operation (self-heated plasma)

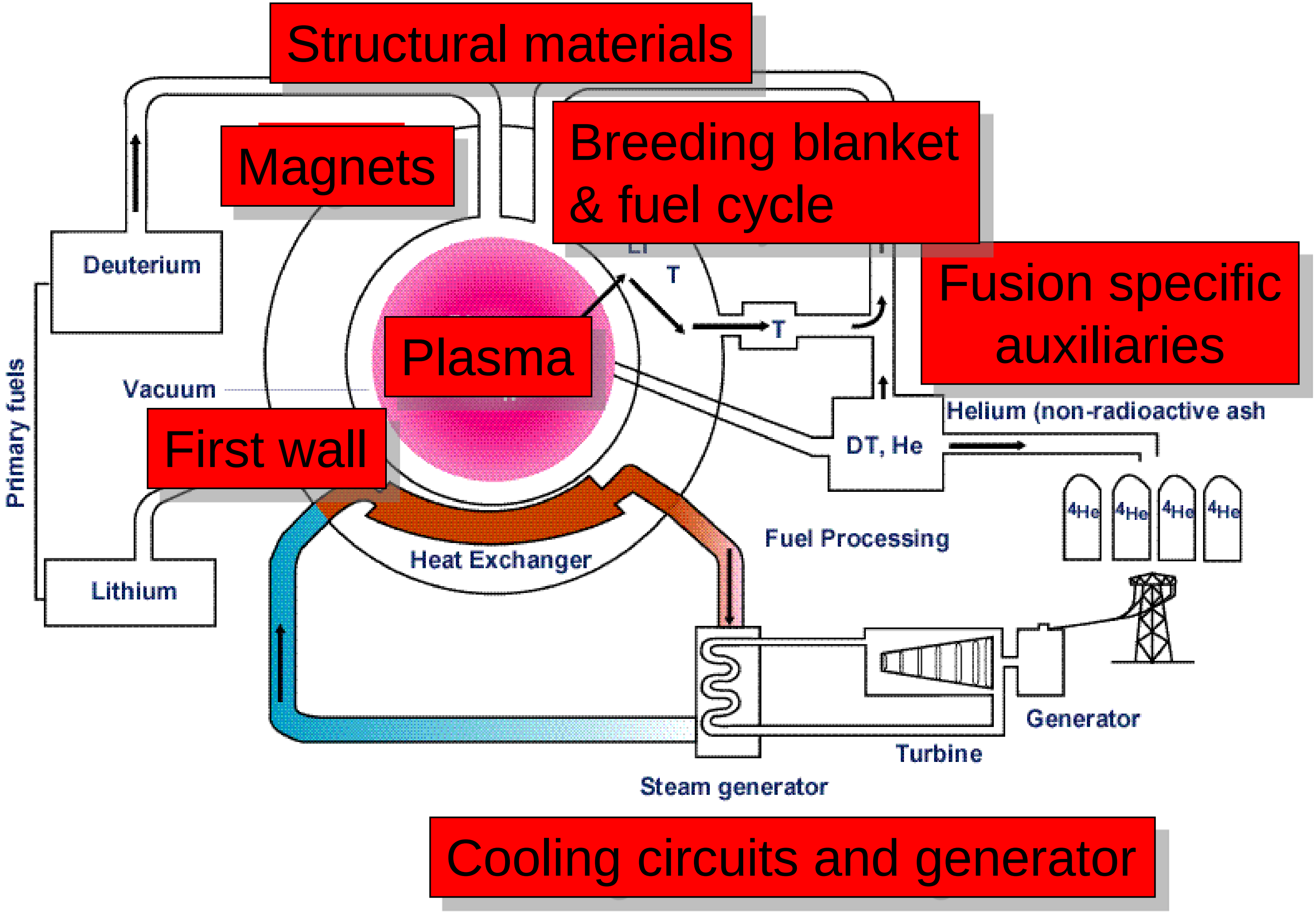
- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- exhaust of heat and particles
- self-heating by α-particles

Q=5 steady state operation

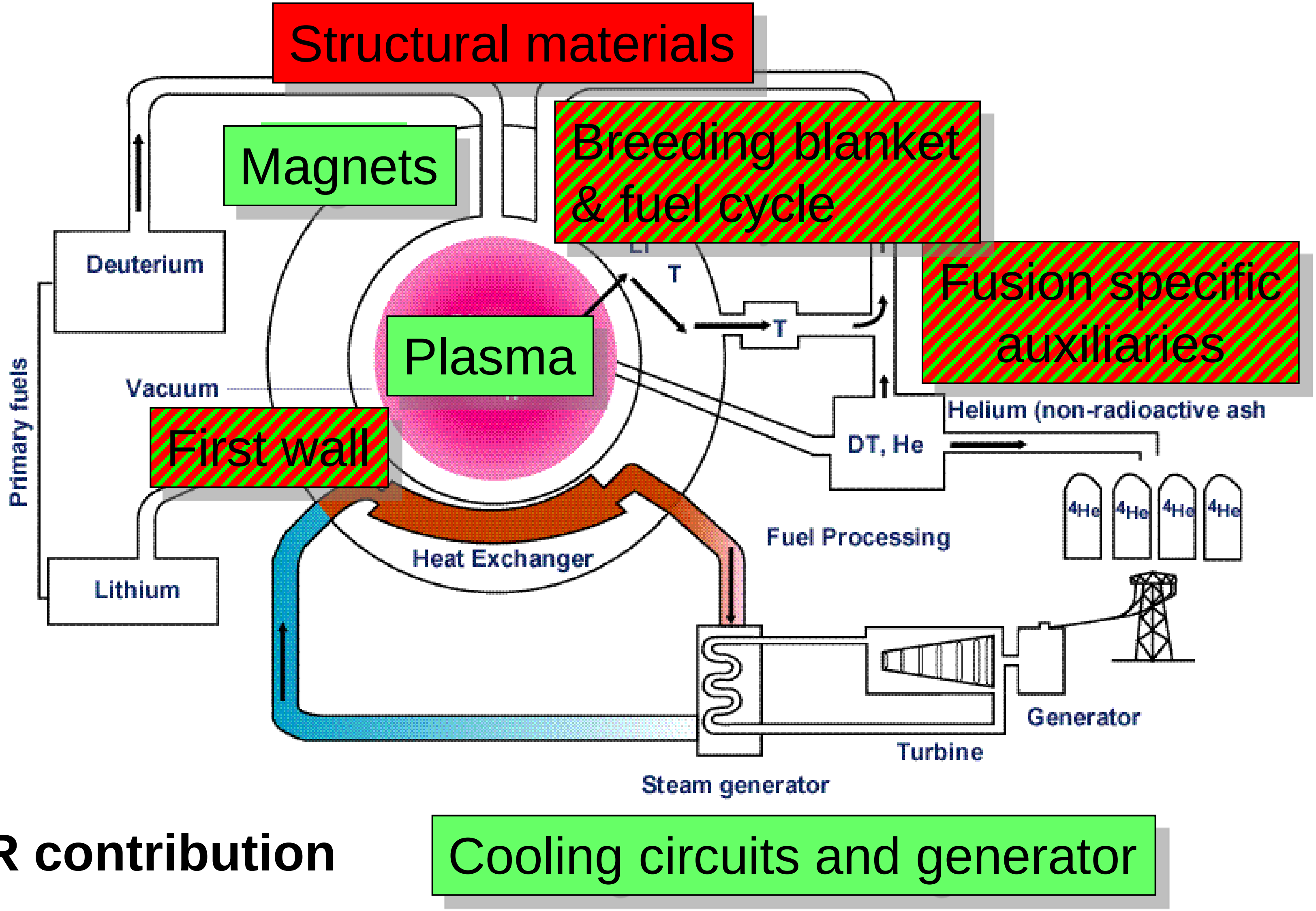
- self-sustainment of plasma current

(we will come back to this when we talk about DEMO)

Fusion Power Plant - Challenges

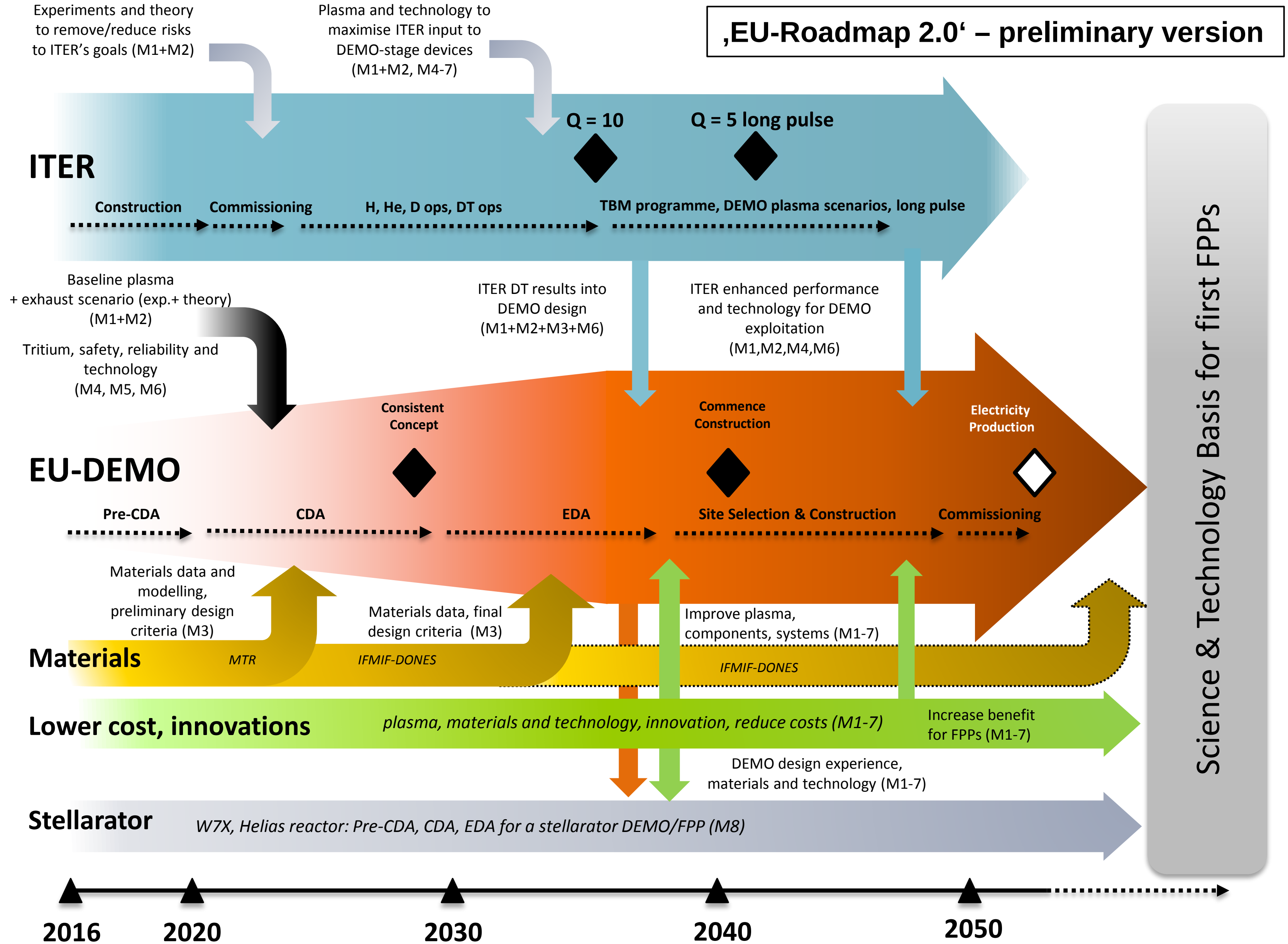


Fusion Power Plant - Challenges



ITER contribution

EU-Roadmap to Fusion Power



I. Introduction: Fusion Power Plants in a Nutshell

II. ITER: Demonstrate Burning Plasmas

III. DEMO: Self-sustained generation of Fusion Energy

- Introduction: what is DEMO?
- DEMO technology challenges
- DEMO physics challenges
- Risk mitigation strategies

- Introduction: what is DEMO?
- DEMO technology challenges
- DEMO physics challenges
- Risk mitigation strategies

What is DEMO?

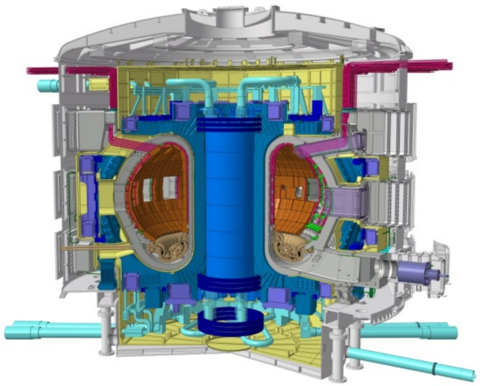
DEMO is the step between ITER and a Fusion Power Plant (FPP)

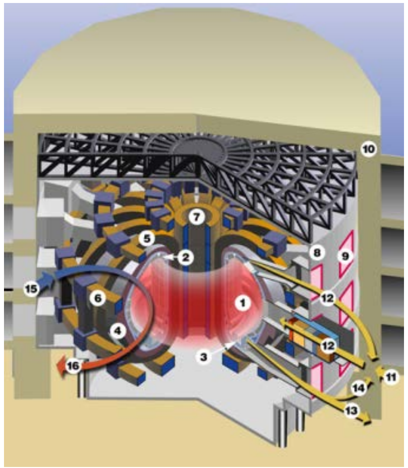
There is no unique definition, but the goals are to demonstrate…

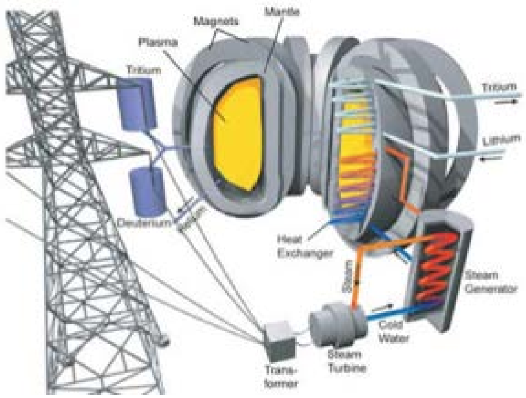
- a workable solution for all physics and technology questions
- large scale (100s of MW) net electricity production
- self-sufficient fuel cycle
- high reliability and availability over a reasonable time span

and allow an assessment of the economic prospects of an FPP

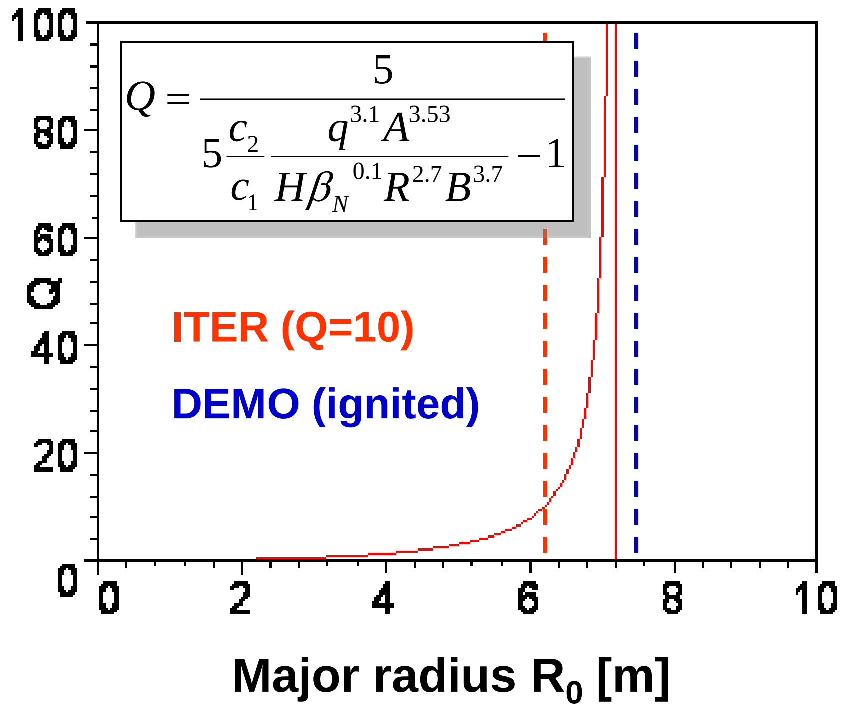
In the EU Roadmap, DEMO is a single step between ITER and an FPP

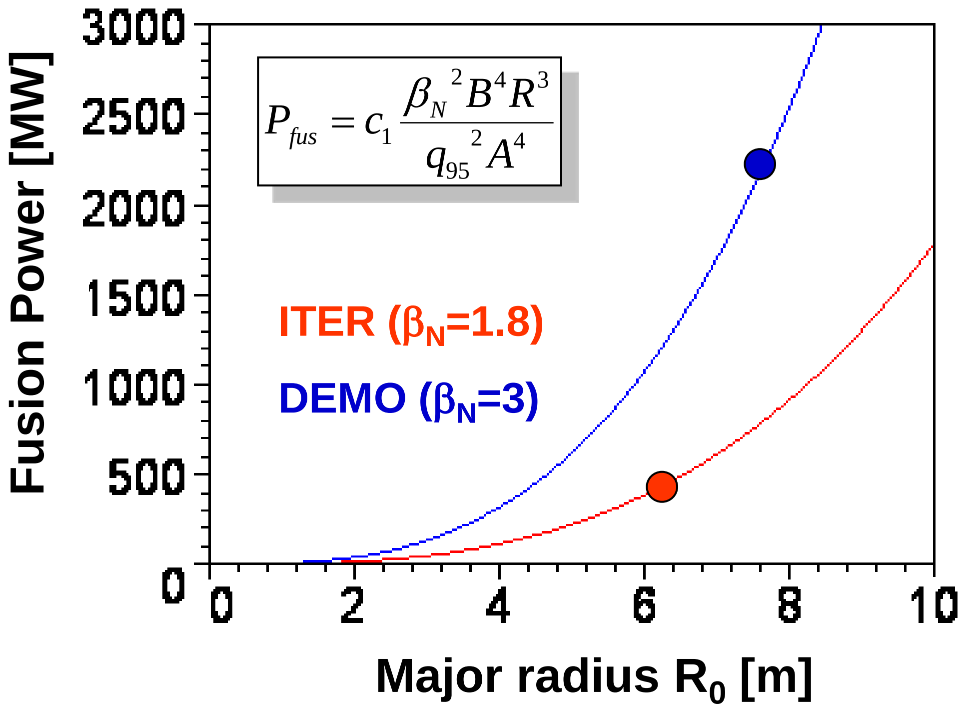






τ_E and β_N determine machine size





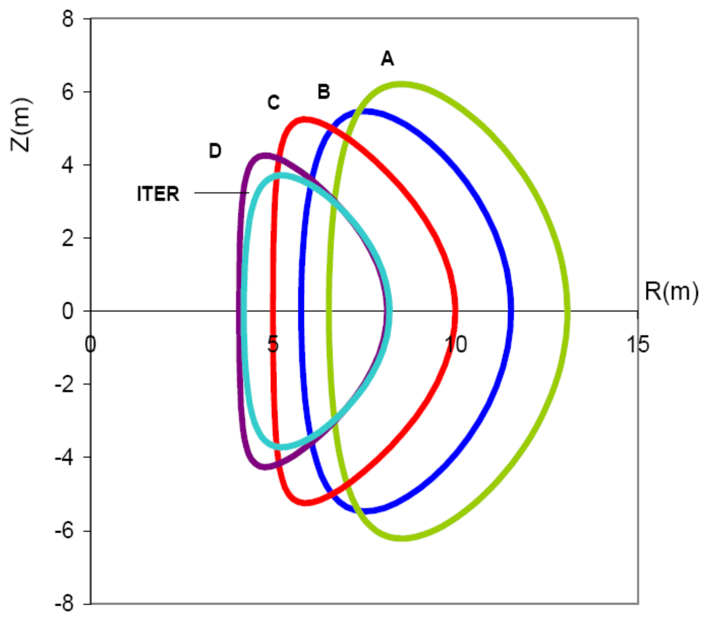
- high τ_E helps to achieve ignition, but does not enter in fusion power
- β_N does almost not enter into Q, but strongly into fusion power

The step from ITER to DEMO

ITER = proof of principle for dominantly α-heated plasmas

DEMO = proof of principle for reliable large scale electricity production with self-sufficient fuel supply

DEMO will be larger: 6.2 m $\Rightarrow$ 8-9 m, 500 MW $\Rightarrow$ ~ 2-3 GW



	A	B	C	D
Electrical power [GW]	1.5	1.3	1.5	1.5
Fusion power [GW]	5.0	3.6	3.4	2.5
Plasma current [MA]	30	28	20	14
Total β_N [% m/MA T]	3.5	3.4	4.0	4.5
coolant	H_20	He	He	LiPb
Efficiency of H&CD [%]	60	60	70	70

First scoping studies indicate that further advances in physics and technology could be very beneificial

- Introduction: what is DEMO?
- DEMO technology challenges
- DEMO physics challenges
- Risk mitigation strategies

DEMO technology challenges

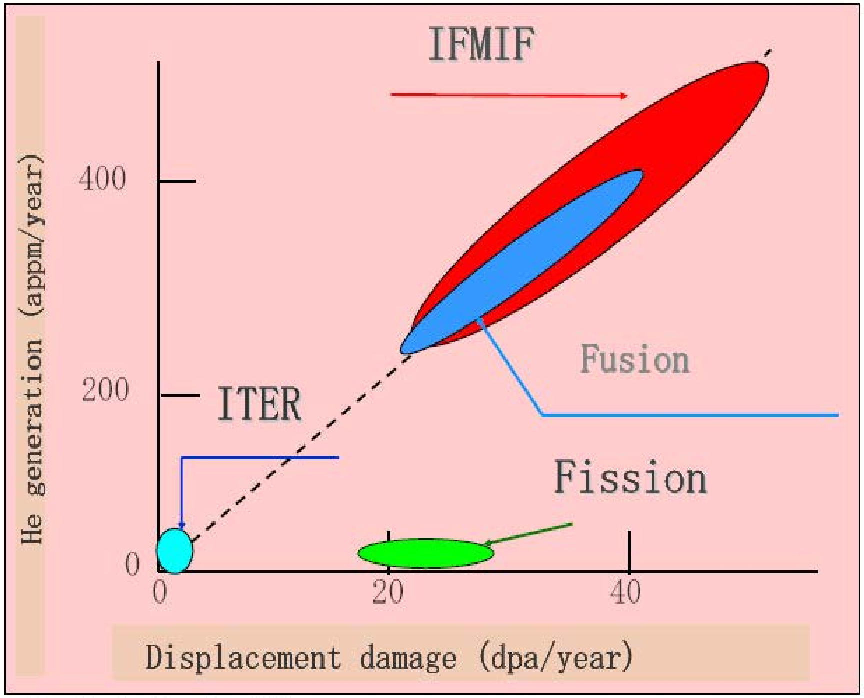
Many solutions can be adopted from ITER – these will not be treated here

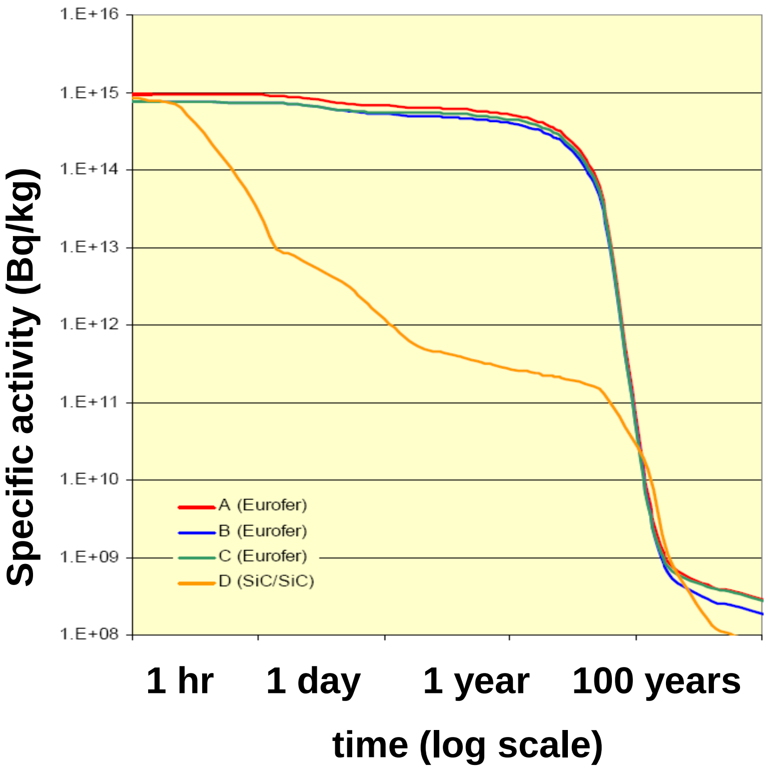
The EU programme has identified the following DEMO technology challenges, i.e. items that will qualitatively go beyond ITER

- Enabling technologies (H&CD, Diagnostics and control, T processing etc.) have to have highest **availability, reliability** and **efficiency**
- **Materials** have to cope with much higher n-fluences at adequate lifetime and, at the same time, low radiological burden
- **T-self sufficiency** has to be guaranteed

‚DEMO is no longer an experiment' – industry should be involved early on!

DEMO Technology Challenges: Structural Materials



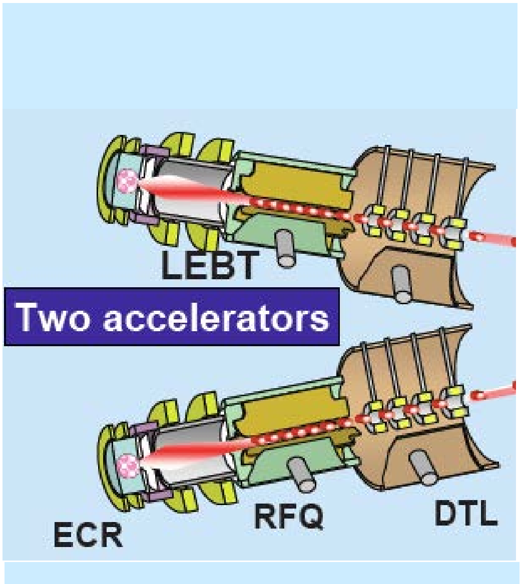


Progress in materials development needed to fully use fusion advantages

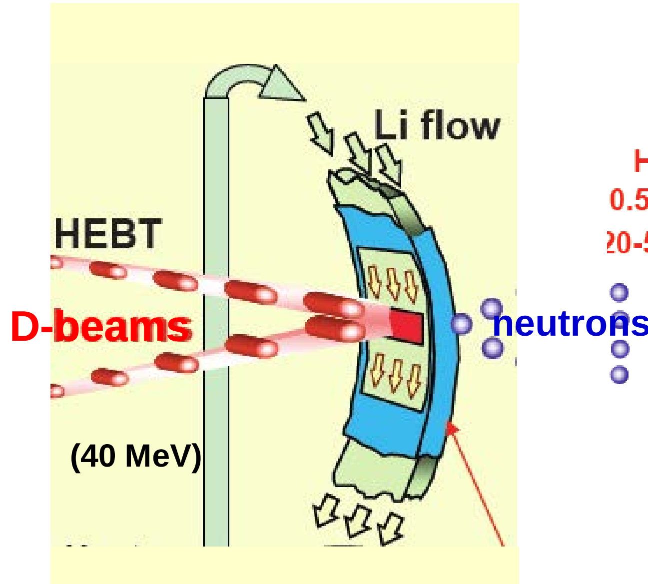
- issues: structural stability at high temperature (Carnot efficiency) and under 14 MeV n-bombardment (rise of Ductile-Brittle Transition Temperature)
- EUROFER steels up to 550º C, better: Oxide Dispersion Strengthened steel
- also reduce waste issues (fuel/burn products itself have short $\tau_{1/2} \leq$ 12 yrs)

Fusion Material Development – IFMIF

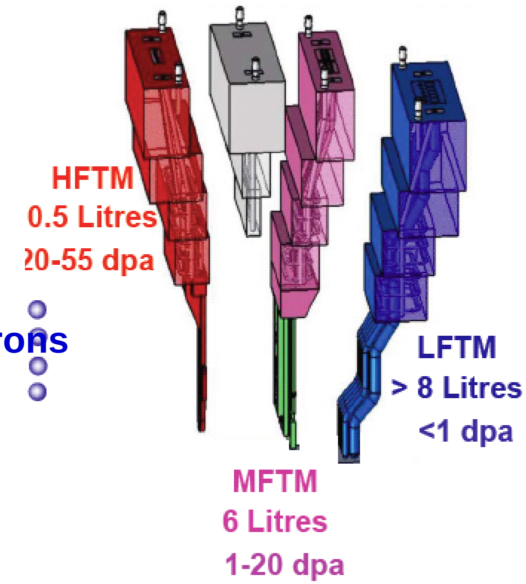
D-Accelerator



Liquid-Li Target



Test cell



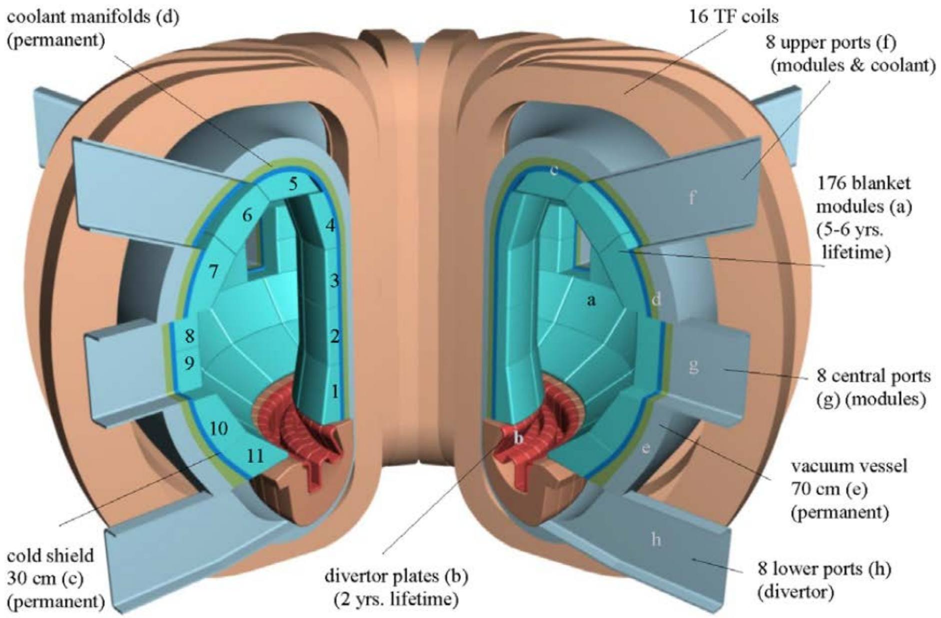
Construction of DEMO as first of a kind requires qualification of materials

- need dedicated facility with high n-fluence of fusion-like spectrum

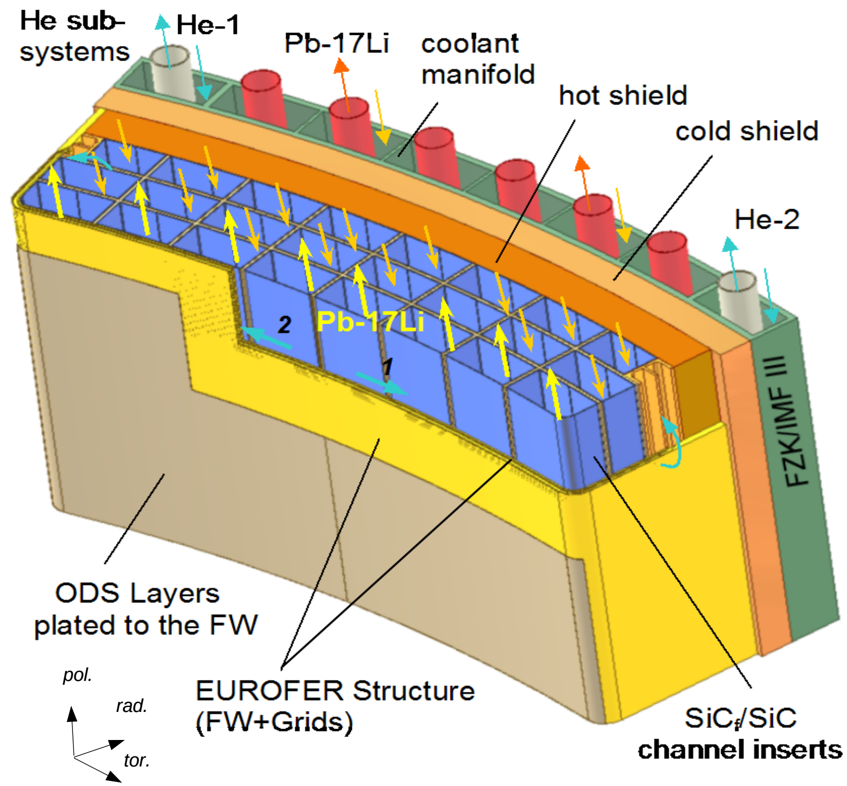
IFMIF can address this and should run several years before DEMO licensing

- important to get IFMIF going if ‚fast track' option should be kept
- present status: ‚EVEDA' (Japan/EU) ~done, EU version (DONES) in 2025?

DEMO Technology Challenges: Blanket



EU Power Plant Conceptual Design Study (PPCS)



"Dual Coolant" He-PbLi LM Blanket Design
$T_{max} \geq 650^\circ$ C, 80-150 dpa in DEMO

Breeding blanket must provide self-sufficient T-supply for fuel cycle

- breeding ratio > 1 needed (1 neutron per fusion reaction)

Blanket also crucial for providing high grade heat (the hotter the better)

- Introduction: what is DEMO?
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The DEMO physics challenges

ITER will address key issues for DEMO beyond present day experiments, the most prominent example being α-heating

The EU programme has identified ‘DEMO physics challenges’ (items not needed for ITER Q=10, but absolutely vital for DEMO and an FPP)

- Steady state tokamak operation at high Q
- High density operation
- Power exhaust (R_{DEMO}/R_{ITER} = 1.2, but P_{DEMO}/P_{ITER} = 4!)
- Disruptions (W_{DEMO}/W_{ITER} > 5!)
- Reliable control with minimum sensors and actuators

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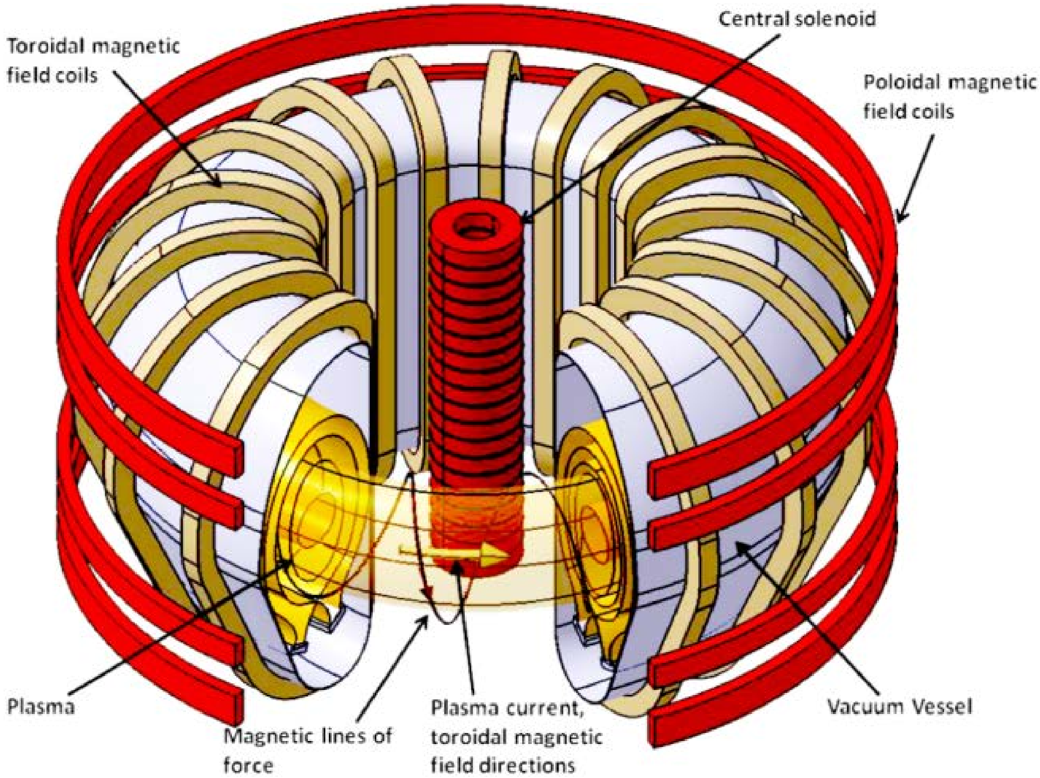
Advanced tokamak – the problem of steady state

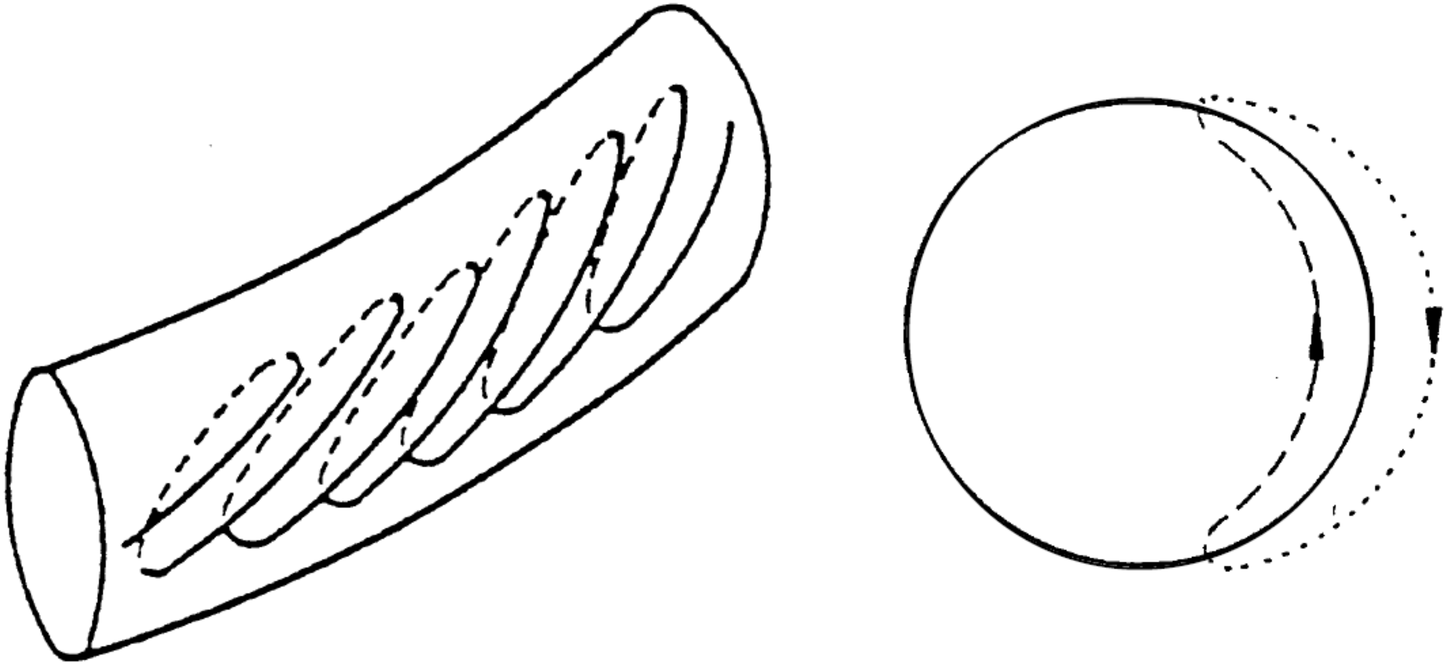


'Conventional': current in transformer (central solenoid) is ramped down continuously to compensate for ohmic losses - discharges inherently pulsed

'Advanced' operation aims at stationary (non-inductive) operation

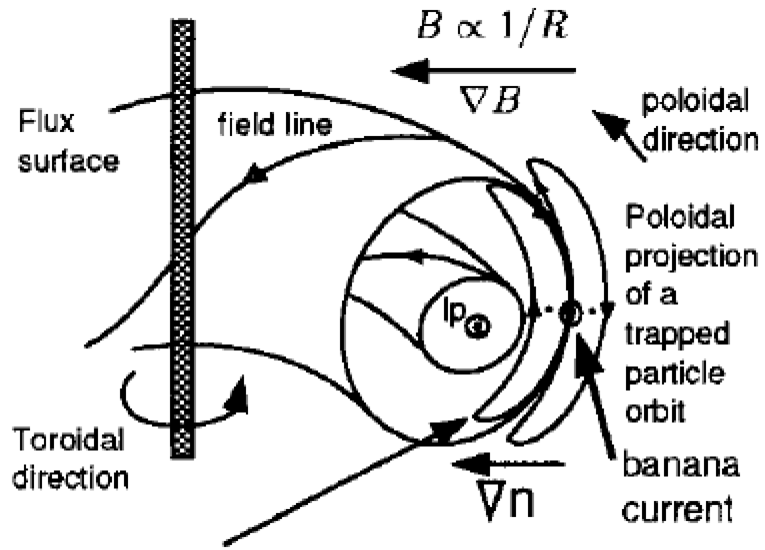
- external CD has low efficiency around 0.1 A per W
- make use of an intrinsic thermo-electrical current - the 'bootstrap' current

Particle orbits: banana orbits in tokamaks

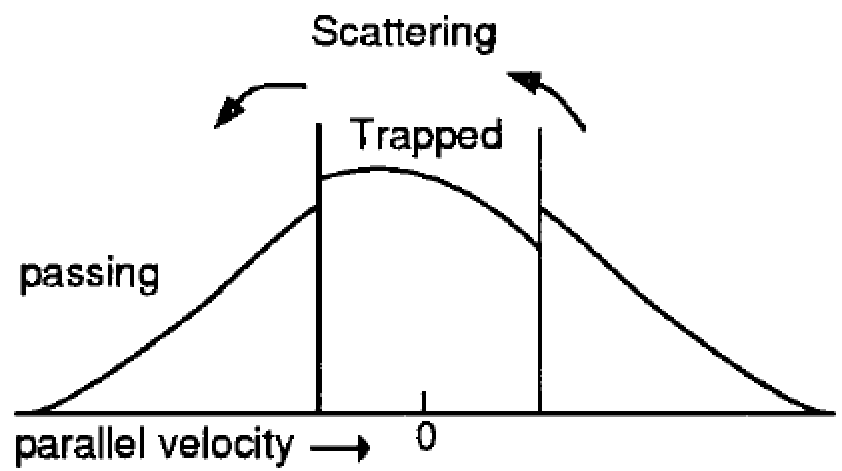


Due to the 1/R decay of a B-field in a torus, there is a magnetic mirror

- particles with low $v_{||}$ are trapped in this mirror, bounce back and forth
- poloidal projection of orbit resembles banana – ‚banana orbit‘



This orbit lies further inside more particles follow this orbit than the orbit further out.



Distribution function as a function of the parallel velocity for constant perpendicular velocity.

For finite pressure gradient, there is a net current of trapped particles along field lines

Distortion of distribution function due to trapped particles leads to a net current of passing electrons

banana current → bootstrap current

Advanced tokamak – the problem of steady state

Advanced scenarios aim at stationary (transformerless) operation

- external CD has low efficiency around 0.1 A per W
- internal bootstrap current high for high $j_{bs} \sim (r/R)^{1/2} \; \nabla p/B_{pol}$

  $\rightarrow f_{NI} \sim I_{bs}/I_p \sim p/B_{pol}^2 \sim \beta_{pol}$

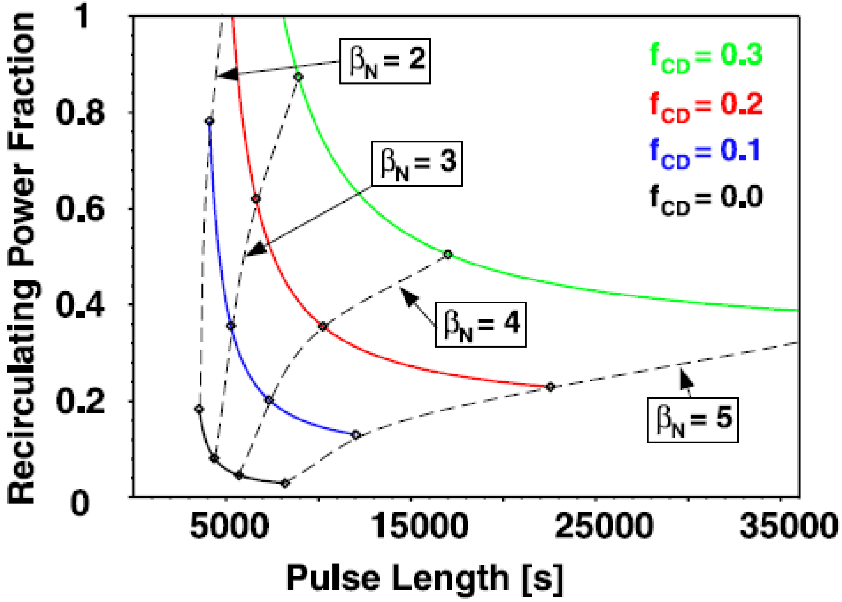
Recipe to obtain high bootstrap fraction:

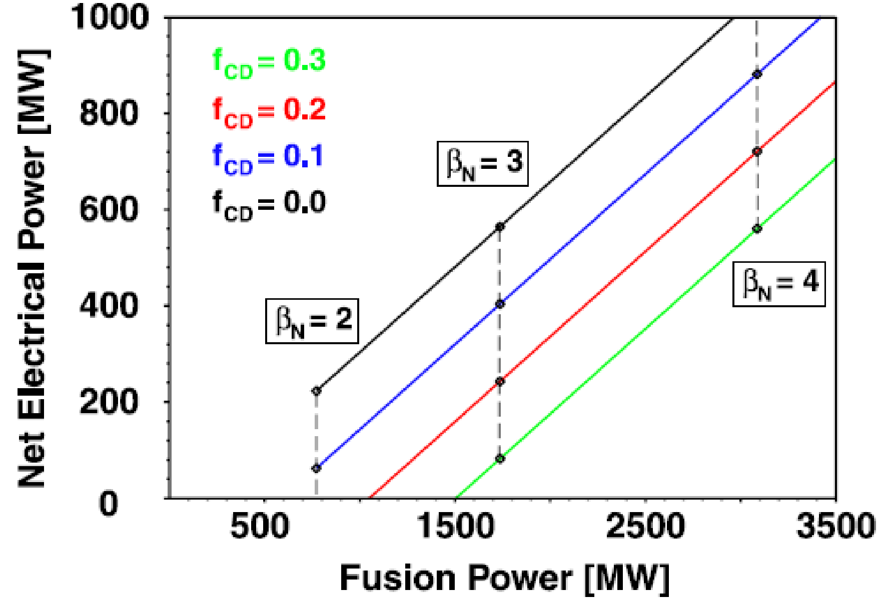
- low B_{pol}, i.e. high q – elevate or reverse q-profile ($q=(r/R)(B_{tor}/B_{pol})$)
- high pressure where B_{pol} is low, i.e. peaked $p(r)$

N.B.: both recipes tend to make discharge ideal MHD (kink) unstable!

The ‘ITER case’ (DEMO with R_0=7.5 m, B_t=5.2 T)

Vary β_N = 2...5 and f_{CD} = 0 (ohmic)…0.3, assume conventional technology





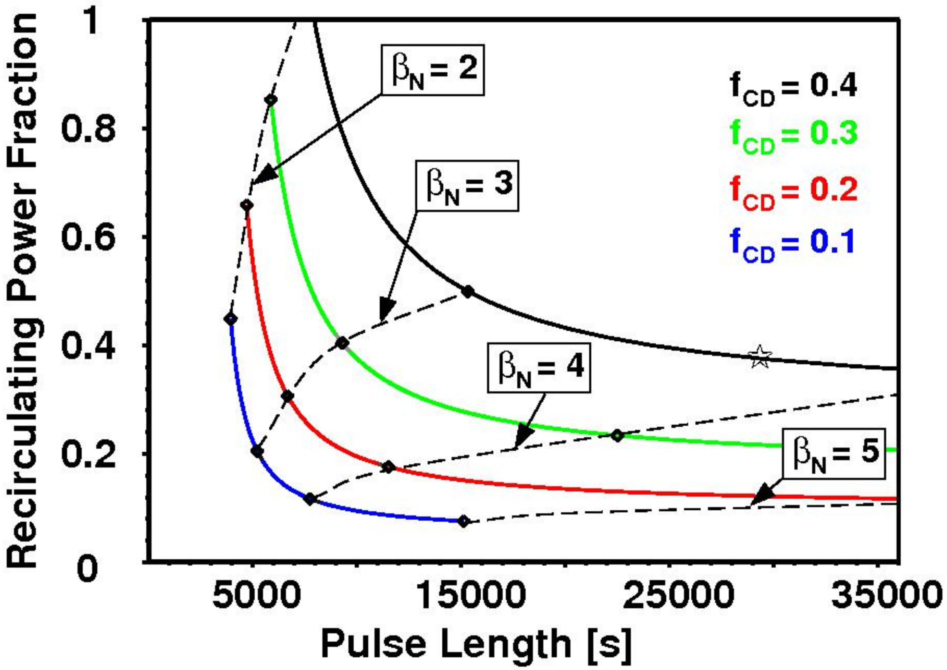
Pulse length is determined by β_N and f_{CD}:

- high normalised pressure β drives intrinsic ‚bootstrap‘ current
- externally driven current $f_{CD}=I_{ext}/I_p$ reduces flux consumption

Increasing the pulse length may severely increase recirculating power

The ‘ITER case’ (DEMO with R_0=7.5 m, B_t=5.2 T)

Assuming improved technology and physics (η_{CD}=0.5, γ_{CD}=0.4, H=1.2), the situation is relaxed w.r.t. power…



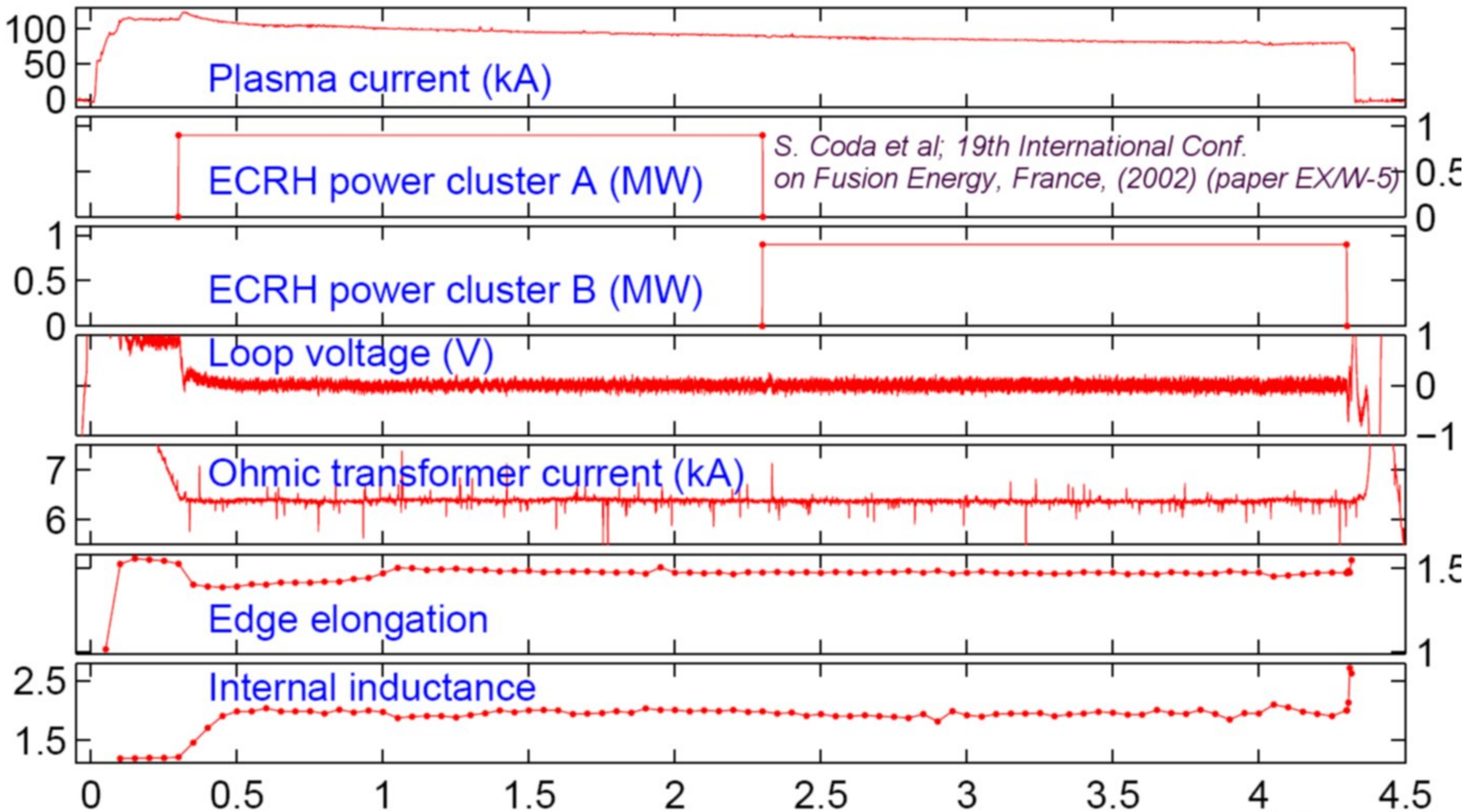
* β_N = 3.5
τ_{pulse} = 8 hrs
f_{rec} = 37%
$P_{el,net}$ = 500 MW

…but achieving steady state is still challenging the stability limits…

⇒ EU: pulsed and steady state options are studied in parallel

⇒ high efficiency of H&CD systems η_{CD} x γ_{CD} becomes crucial

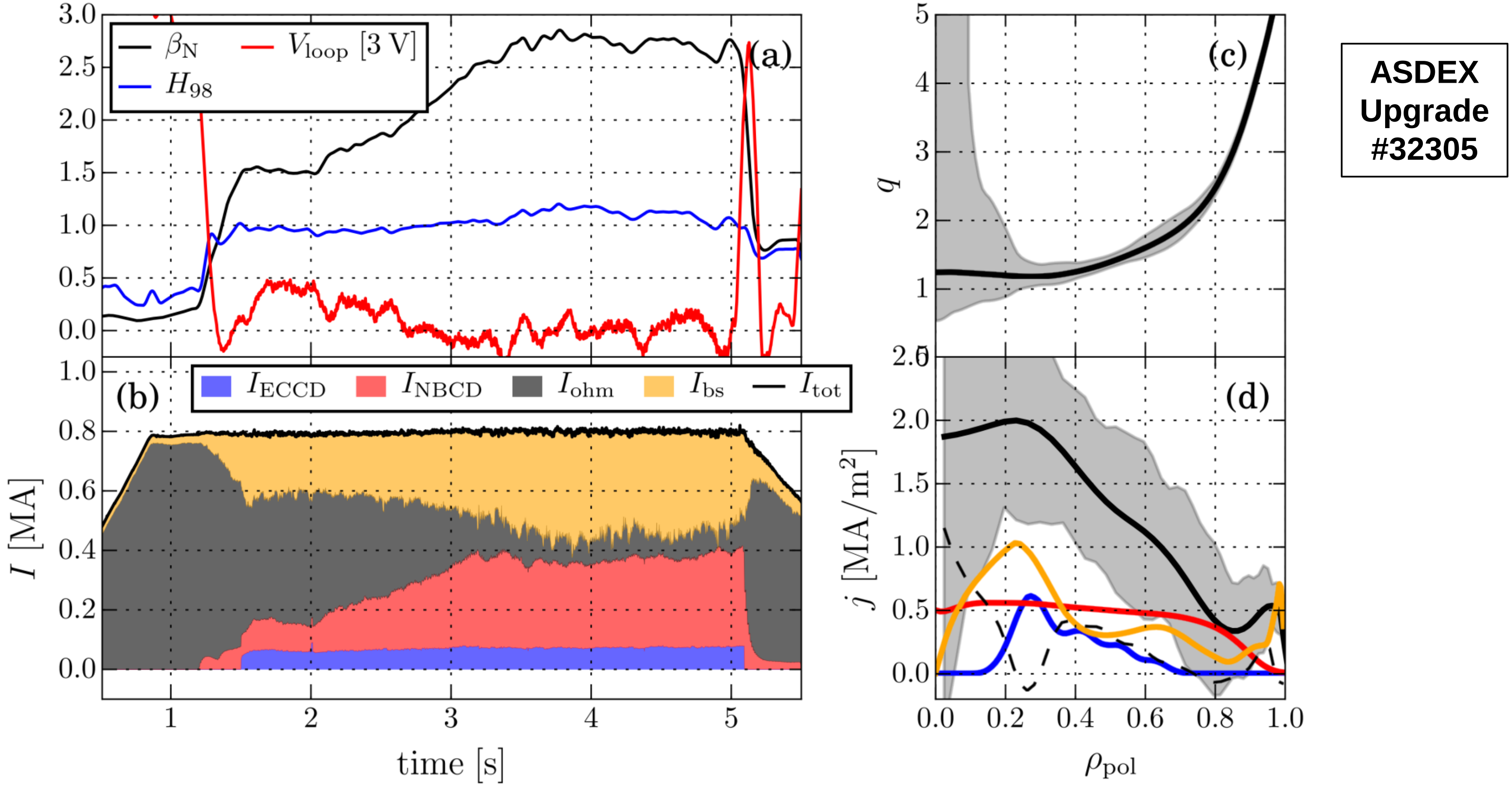
Advanced tokamak – the problem of steady state



TCV discharge 20881 of record length (4.3 s), sustained by 0.9 MW ECCD

A ‘proof of existence’ exists – but a long way to go (q_{95} is above 10!)….

Development of a steady state tokamak scenario



Higher q_{95} (~5.3) than ITER Q=10 for higher bootstrap fraction (~ 50%)

- stationary on current redistribution timescale, approaching fully noninductive
- *q*-profile tailored by off-axis NBCD & ECCD to enable high bootstrap fraction

The DEMO physics challenges

ITER will address key issues for DEMO beyond present day experiments, the most prominent example being α-heating

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- Reliable control with minimum sensors and actuators

Technology sets strict limits for exhaust in DEMO

- Water cooling: (safety, small T_{op}-range (DBTT), < 5 dpa): ≤ 5-**10** MW/m^2
- He cooling: (higher T_{op}-range, but development needed): ≤ **5**-10 MW/m^2
- in addition, $T_{e,div} \leq 4$ eV to limit erosion (consistent lifetime estimate)

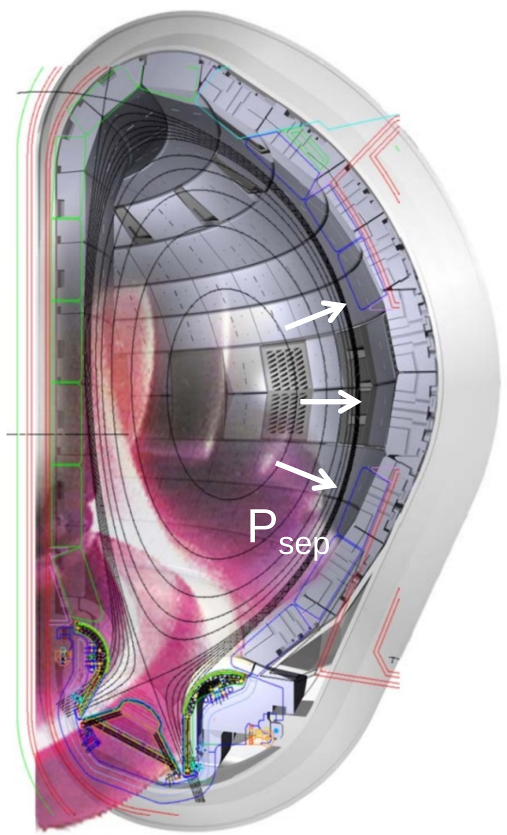
Karlsruher Institut für Technologie

Exhaust problem on DEMO tougher than on ITER

Conservative ansatz: base DEMO exhaust solution on ITER

- solution is a (fully) detached conventional divertor
- divertor challenge qualifier used here: power into SOL area

$$\frac{P_{sep}}{2\pi R\lambda_q} \approx const \quad \Rightarrow \quad \frac{P_{sep}}{R\rho_p} \propto \frac{P_{sep}B}{AqR} = \left(\frac{P_{sep}B}{AqR}\right)_{ITER}$$



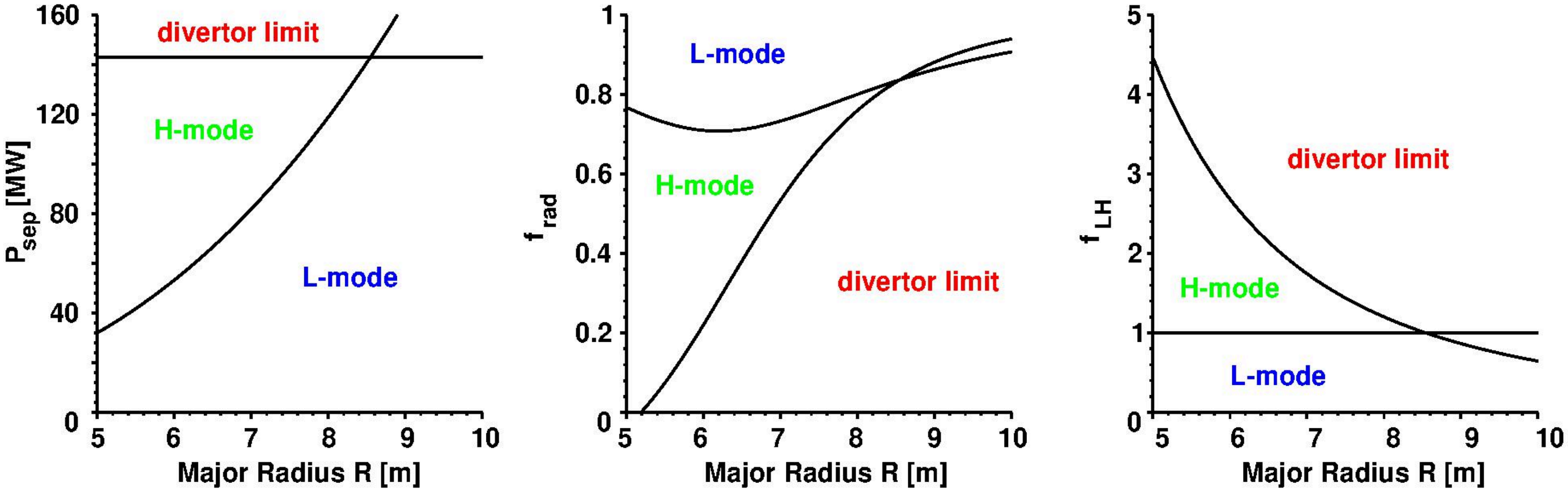
Defines upper limit: $P_{sep} < const.\ R/B$

Lower limit given by $P_{sep} > f_{LH}\,P_{LH} \propto nBR^2$

$\Rightarrow$ P_{sep} window narrows with machine size

Exhaust problem on DEMO tougher than on ITER

Note: in this plot, *B* ~ *R* has been assumed (to keep density similar)

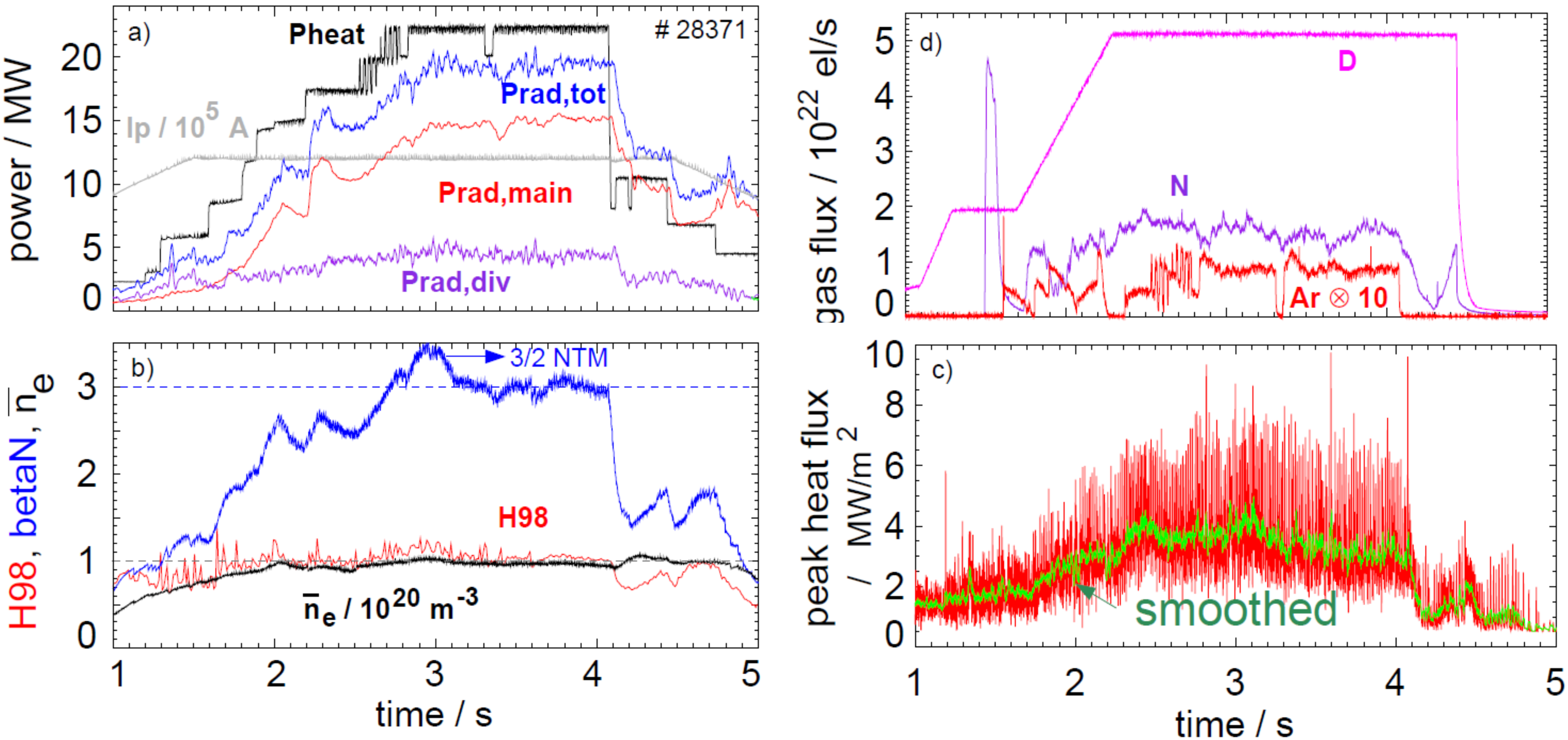


P_{sep} window narrows with size

- core radiation fraction increases with machine size
- allowable power range above LH transition (f_{LH}) narrows as well

Double Radiation Feedback on ASDEX Upgrade

A. Kallenbach et al.,Nucl. Fusion (2013)

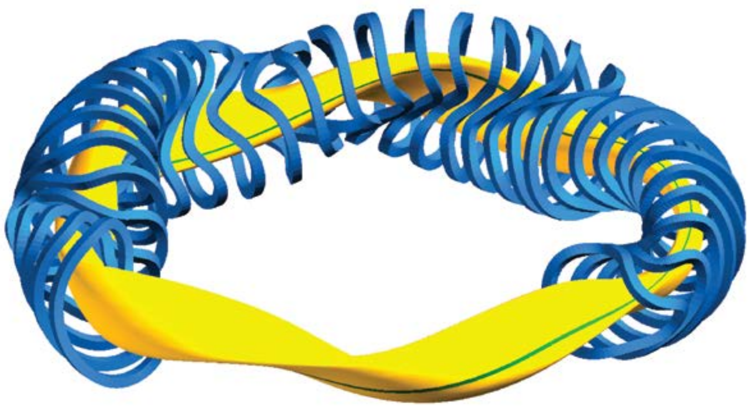


$P_{rad,main}$ controlled by Ar-seeding, $P_{rad,SOL\&Div}$ by N-seeding

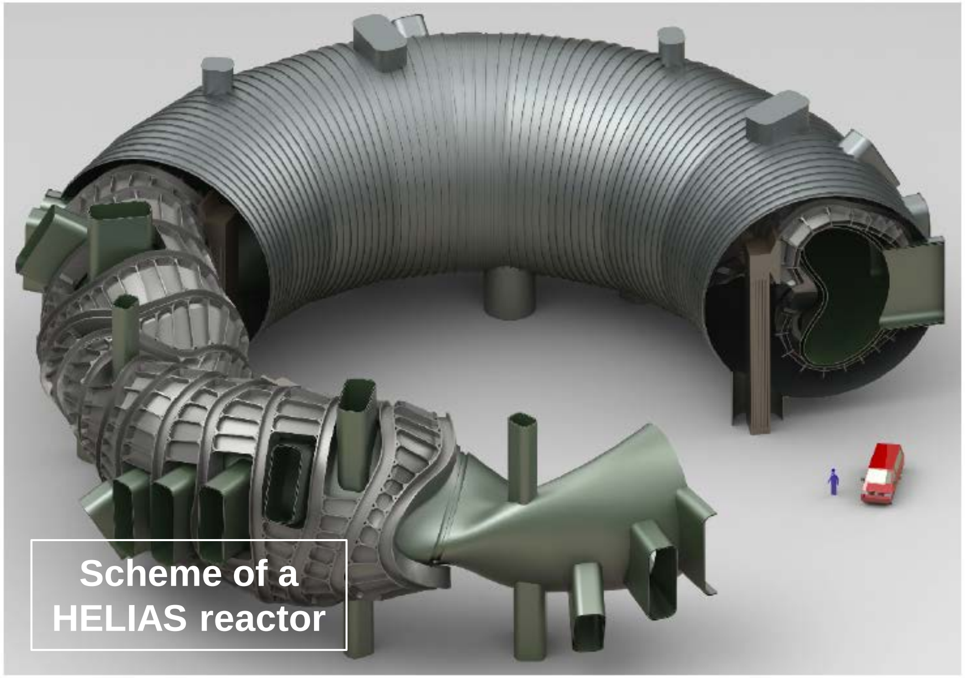
- $P_{heat,tot}$ = 23 MW and $P_{rad,core}$ = 15 MW (67%), $q_{div} < 5$ MW/m^2
- close to P_{LH}, but still $H \geq 1$ and $\beta_N = 3$

- Introduction: what is DEMO?
- DEMO technology challenges
- DEMO physics challenges
- Risk mitigation strategies

Stellarators are intrinsically steady state devices



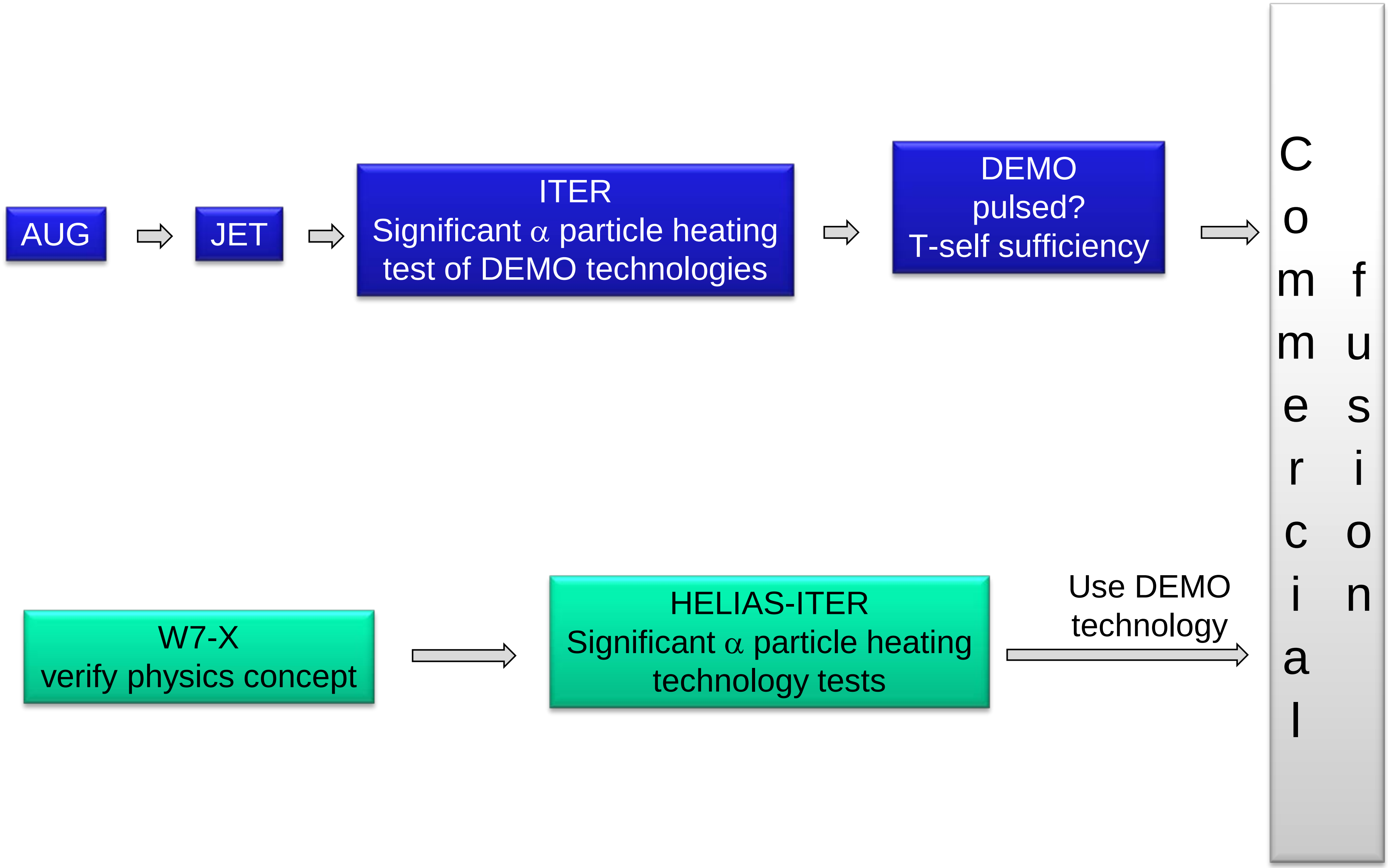
W7-X, in operation since December 2015



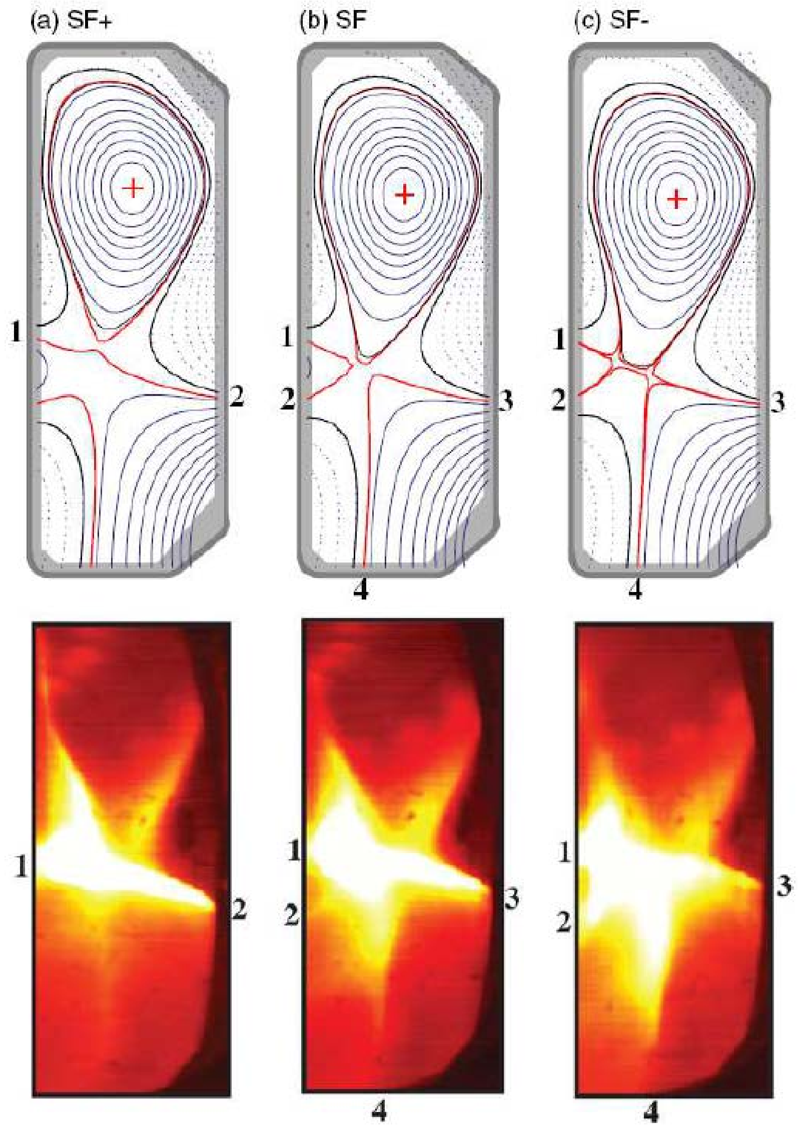
EU programme studies stellarator line as alternative to tokamak

- no internal plasma current – no limitation to steady state plasma operation
- no disruptions since no intrinsic current
- engineering feasibility of stellarator power plant must be assessed early on

Stellarator in a roadmap to a fusion reactor



Optimisation of divertor geometry



The ‘snowflake divertor’ has a higher order magnetic field null

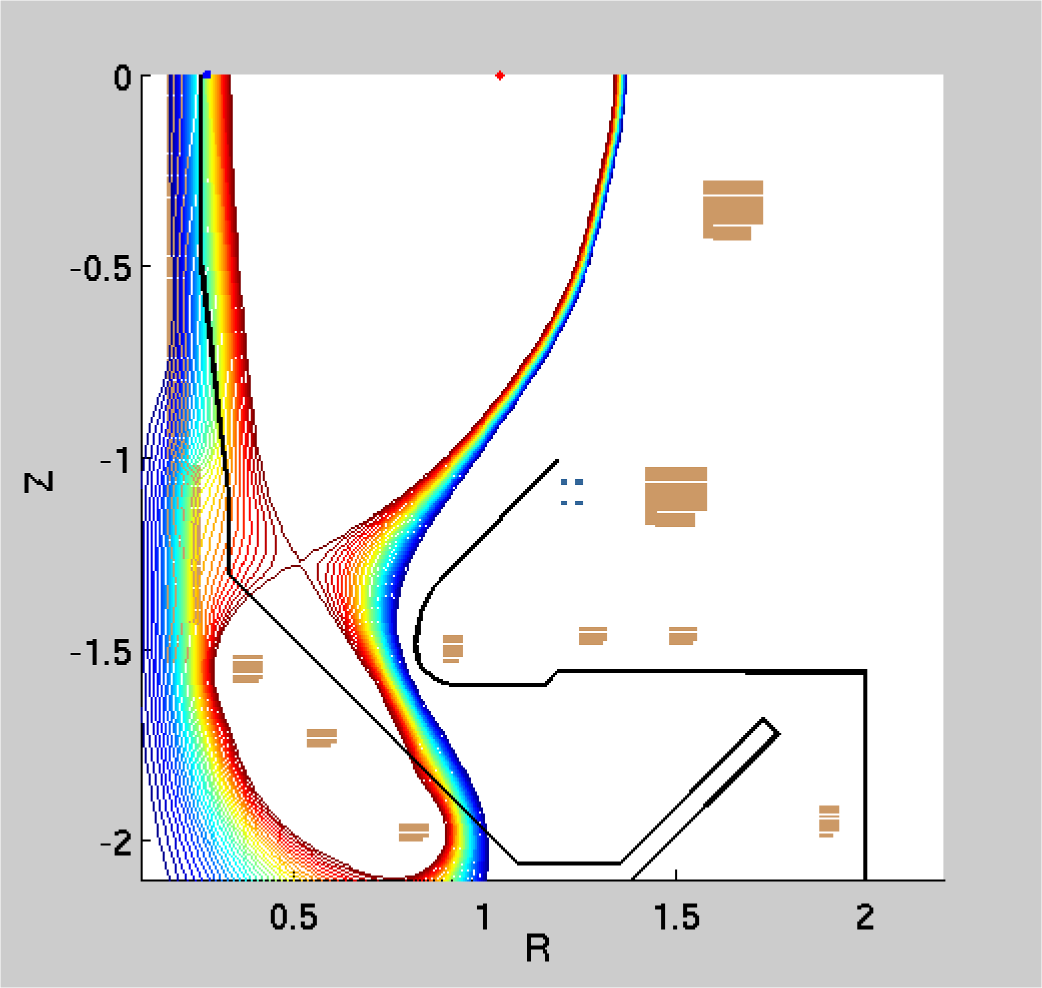
- ‘snowflake’ promises large expansion of magnetic flux and concomitant broadening of wetted area

Technological questions to be studied:

- integration into reactor (higher multipole requires closer coils)
- controllability of configuration

TCV tokamak, Switzerland

Optimisation of divertor geometry



'Super-X divertor' has very long outer divertor leg

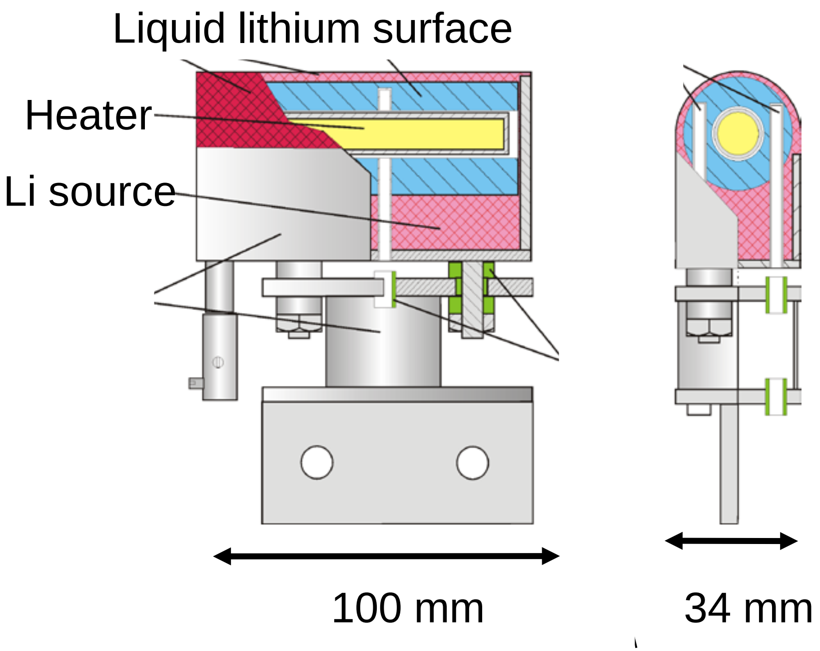
- maximises toroidal magnetic flux expansion
- promises large divertor volume and wetted area

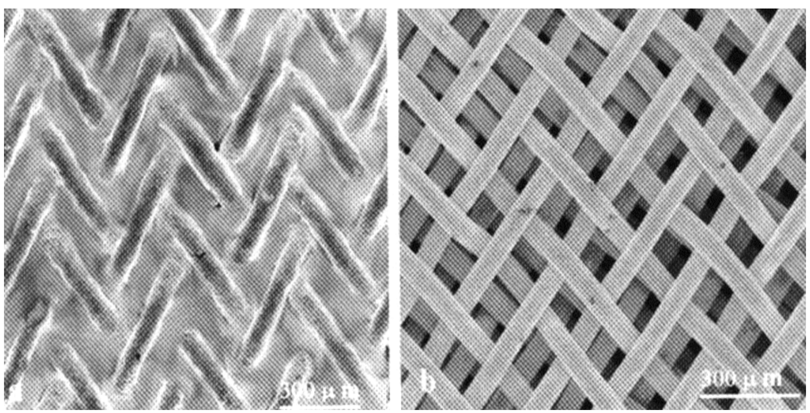
Technological questions:

- integration into a reactor (long divertor leg → coils need to be close to plasma)
- controllability of configuration

MAST tokamak, UK

‚New‘ materials: liquid metals





Experiments with liquid Li show

- good plasma compatibility
- capillary porous system – no droplets

However, technological questions have to be solved before this can be considered a viable candidate

- T retention will be high (use Ga?)
- heat removal concept that does not rely on evaporation heat needed (jxB forces on flowing metal!)
- concept for a closed metal cycle under steady conditions needed

Summary and Conclusions

Fusion energy research has made tremendous progress in recent years

- existing database enabled design of next-step device: ITER

Strategy towards fusion energy comprises 3 major facilities:

- ITER to study burning plasma physics and fusion specific technology
- IFMIF to qualify materials (in parallel to ITER)
- DEMO to demonstrate viability of integrated reactor concept

The stellarator is studied as a viable back-up option and may well be the better concept in the long run

Note: The European fusion research programme is playing an important role in this effort!

Roadmap for Chinese MFE Development