HELMHOLTZ RESEARCH FOR GRAND CHALLENGES

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Sun, Coal, Nuclear Fusion: Scientific challenges of the future energy supply

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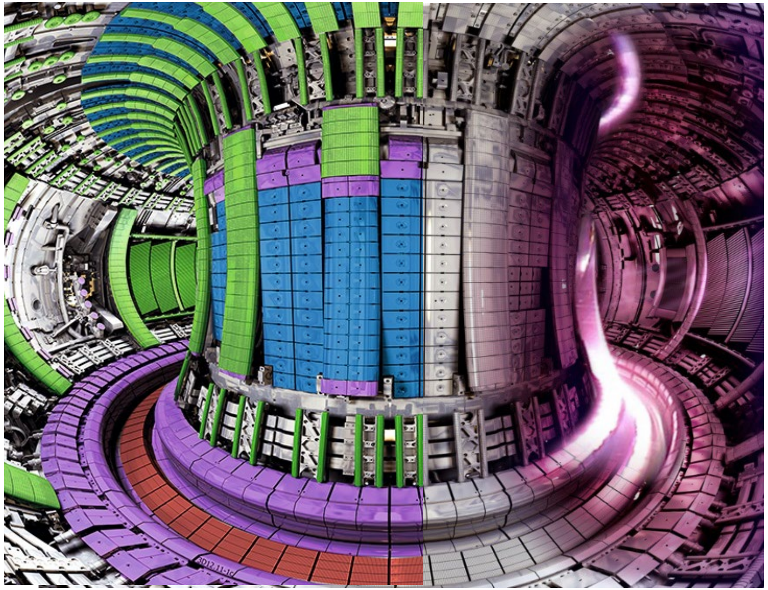
Outline

- Nuclear Fusion
  - Introduction
  - Magnetically Confined Fusion (MCF)
  - Scaling laws and their validity
- JET and ITER
  - Joined European Torus (JET)
  - Plasma-wall interaction
  - Power and particle exhaust challenge
  - International Thermonuclear Experimental Reactor (ITER)
- DEMO and other concepts
  - Challenges
  - SPARC, ARC and tokamaks
  - Wendelstein7-X and HELIAS reactor

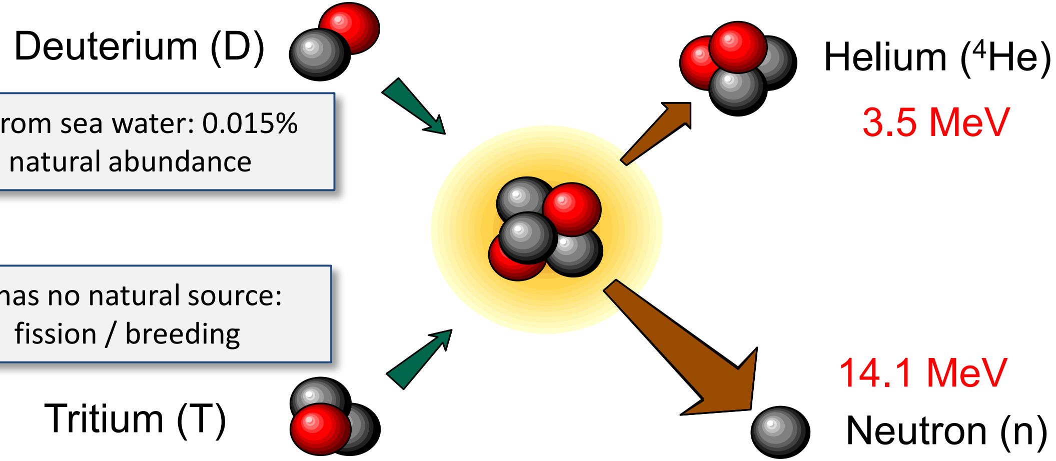
JET tokamak with ITER-like Wall



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

Nuclear Fusion Basics

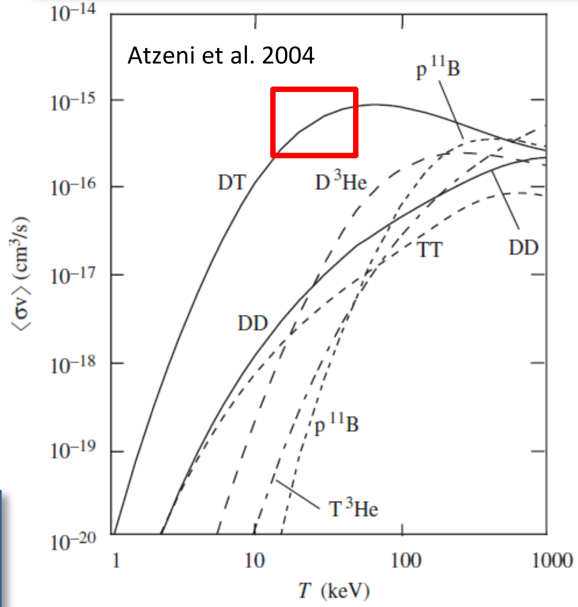
- Thermonuclear fusion with largest reaction rate: DT fusion reaction



D from sea water: 0.015% natural abundance

T has no natural source: fission / breeding

Maxwell-averaged reaction activity vs ion temperatures for fusion reactions



p-p reaction in sun at 10^{-40} cm^{3}s^{-1} !

- Measure for fusion power in a 50:50 DT reaction is the triple product:

  ion density (n_i) x ion temperature (T_i) x confinement time (τ_E)

- Ignition in magnetically confined fusion plasmas achievable if:

  n_i [10^{20} m^{-3}] x T_i [10keV] x t_E [5s] $\geq$ $5\text{x}10^{21}$ keV s m^{-3} (Lawson criterion)

- He used to heat plasma core (20% of required power) before removed

Particles in Magnetised Plasmas

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Lorentz force acts on charged particles in a magnetic field:

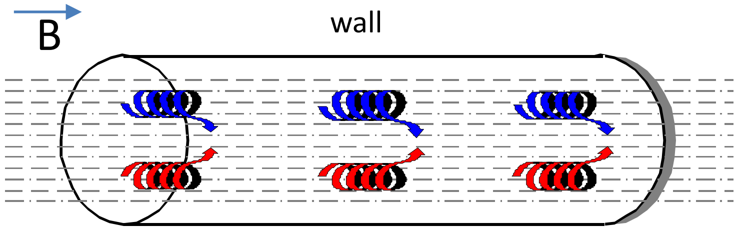
$$\vec{F}_L = q \cdot \left[\vec{\mathrm{v}} \times \vec{B}\right]$$ Lorentz force

Ions and electrons gyrate around magnetic field lines:

$$\left|\vec{r}_L\right| = \frac{\left|\vec{v}_\perp\right|}{\left|\vec{\omega}_L\right|} = \frac{m v_\perp}{qB}$$ Larmor radius

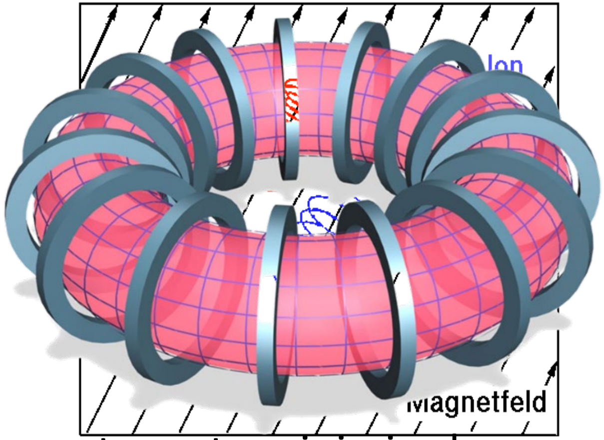
Many gyrations before a collisions occurs in magnetized plasmas:

$$\beta = \frac{\omega_L}{\upsilon_{collision}} >> 1$$ Hall parameter



High mobility along magnetic field lines

Low mobility perpendicular magnetic field lines



Low plasma-wall interactions (PWI) if high magnetization present, full ionization, and no-cross field transport (ideal case).

=> torus to minimize losses

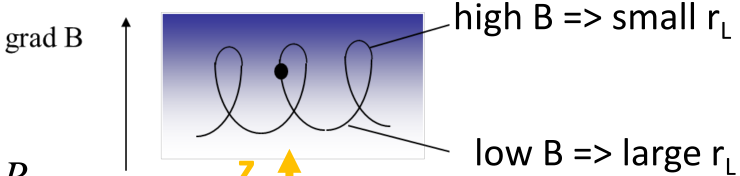
Simple Toroidal Arrangement

Particle drifts due to forces in (inhomogeneous) B-field:

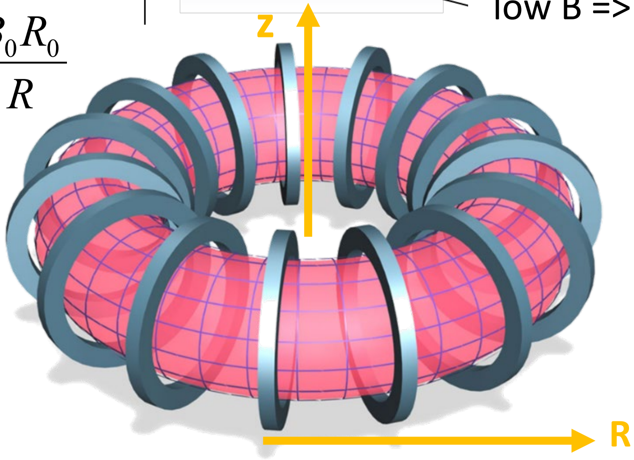
$$\vec{v}_D = \frac{\vec{F}\times\vec{B}}{qB^2}$$

In torus geometry two inherent drifts appear:
∇B drift and curvature drift [so-called "torus drift"]

$$\vec{v}_D = \frac{m}{qB^3}\left(v_\parallel^2 + \frac{1}{2}v_\perp^2\right)\vec{B}\times\nabla B$$

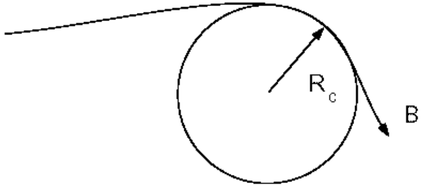


$$B_\phi = \frac{B_0 R_0}{R}$$



$$\nabla\times\vec{B} = 0$$

$$at\ \vec{R}_c:\quad \vec{B}_R = 0$$



centrifugal-like force

Is the simple toroidal arrangement enough for MCF?

Particle drifts due to forces in (inhomogeneous) B-field:

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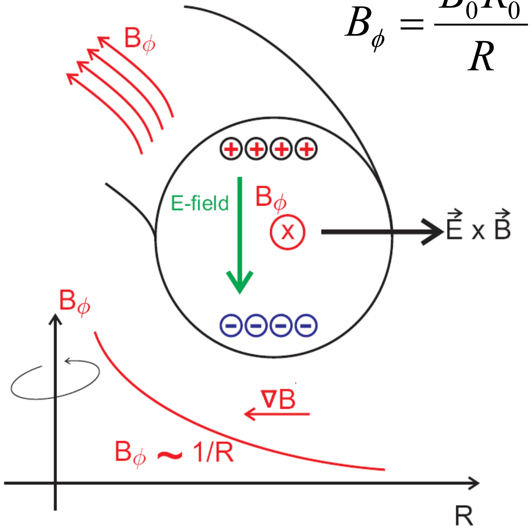
$$\vec{v}_D = \frac{m}{qB^3}\left(v_\parallel^2 + \frac{1}{2}v_\perp^2\right)\vec{B} \times \nabla B$$ Torus drift

$$B_\phi = \frac{B_0 R_0}{R}$$

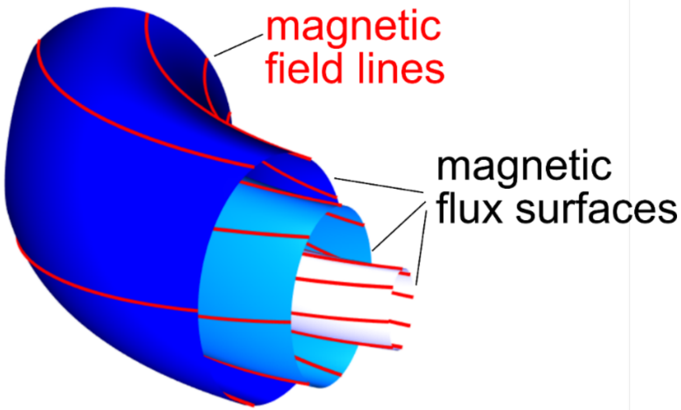


$$\vec{v}_D = \frac{\vec{E} \times \vec{B}}{B^2}$$ ExB drift

- Pure toroidal magnetic field induces charge separation due to "torus drift"
- Charge separation induces vertical electrical field E
- E and magnetic field B induces E x B drift and particle movement outward
- Second magnetic field component required (poloidal direction) => induces screw-like trajectory (helical field lines) and stability

Two options: stellarator or tokamak

Toroidal Magnetic Confined Solutions: Tokamak

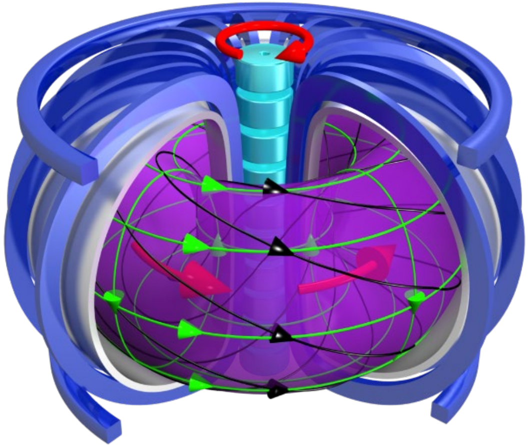


- concepts rely on toroidal magnetic field geometry with nested magnetic flux surfaces
- plasma pressure counteracted by magnetic field force

$$\nabla p = j \times B$$

J current density
B magnetic field

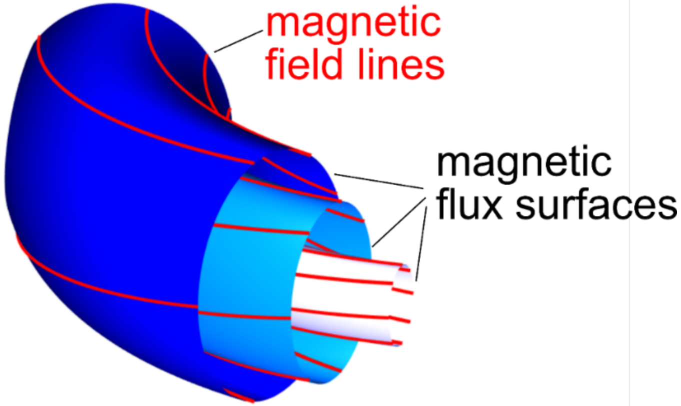
- stability by helical structure requires



$$j \times B = \nabla p$$
$$B \cdot \nabla p = 0$$

- inductively driven toroidal plasma current essential
- no confining magnetic field without plasma
  - ⇨ disruptive plasma density limit
  - ⇨ steady-state operation challenging

Toroidal Magnetic Confined Solutions: Stellarator



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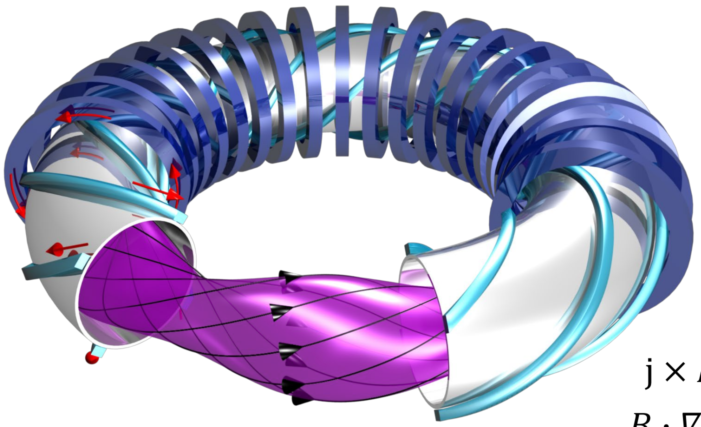


$$j \times B = \nabla p$$

$$B \cdot \nabla p = 0$$

- confining magnetic field entirely generated
- by external magnetic field coils
- variety of different magnetic field geometries possible

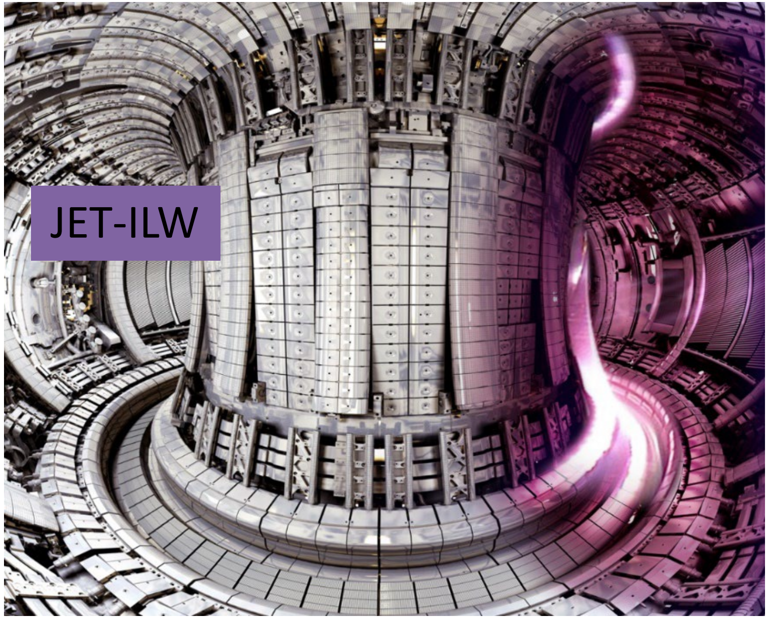
⇨ confining vacuum magnetic field

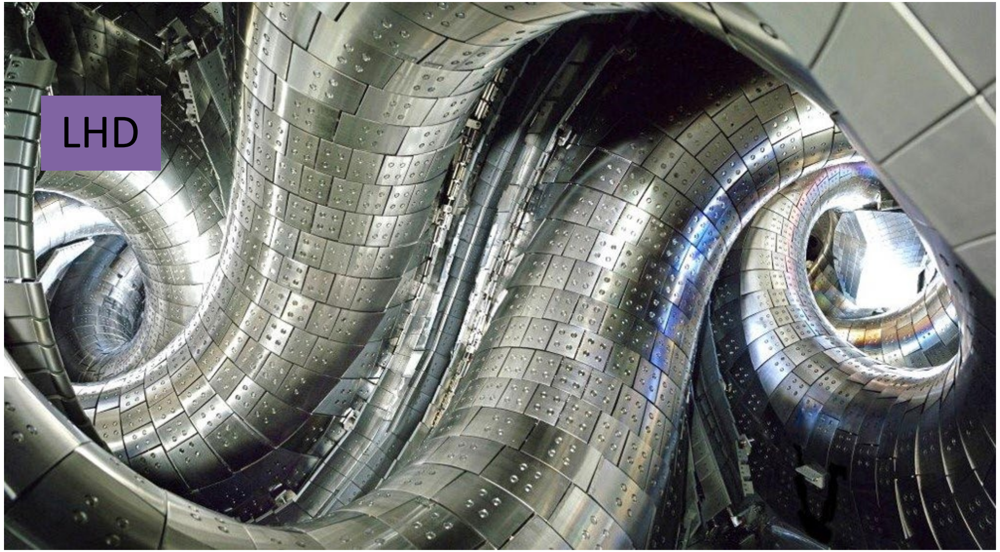
⇨ intrinsic steady-state capability

Magnetically Confined Plasmas: Devices

- inductively driven toroidal plasma current
- plasma required to provide magnetic confinement

- magnetic confinement solely by magnetic coils
- no toroidal plasma current required





- poloidal divertor arrangement
- toroidally symmetric divertor (2D)

- complex divertor geometries (3D)
- helical divertor, island divertor etc.

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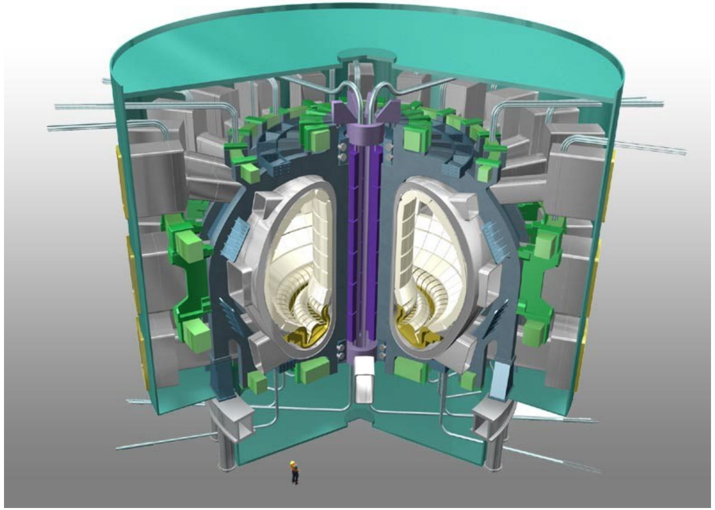
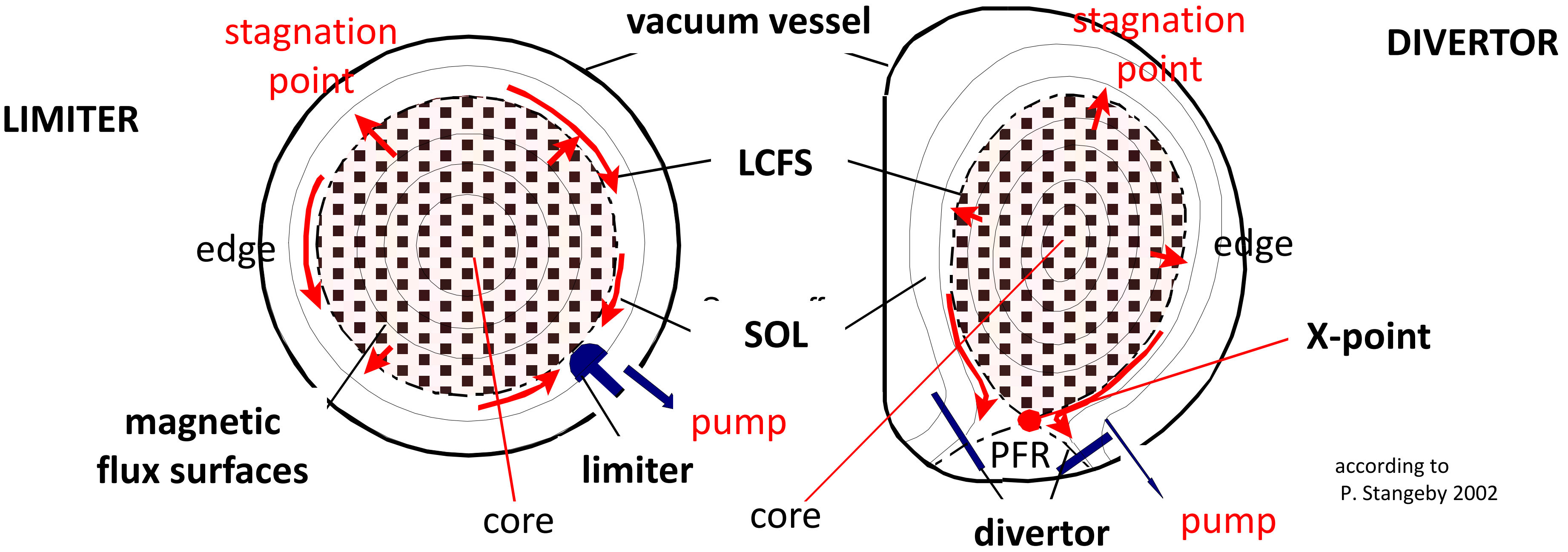


Limiter vs. Divertor Tokamak Configuration

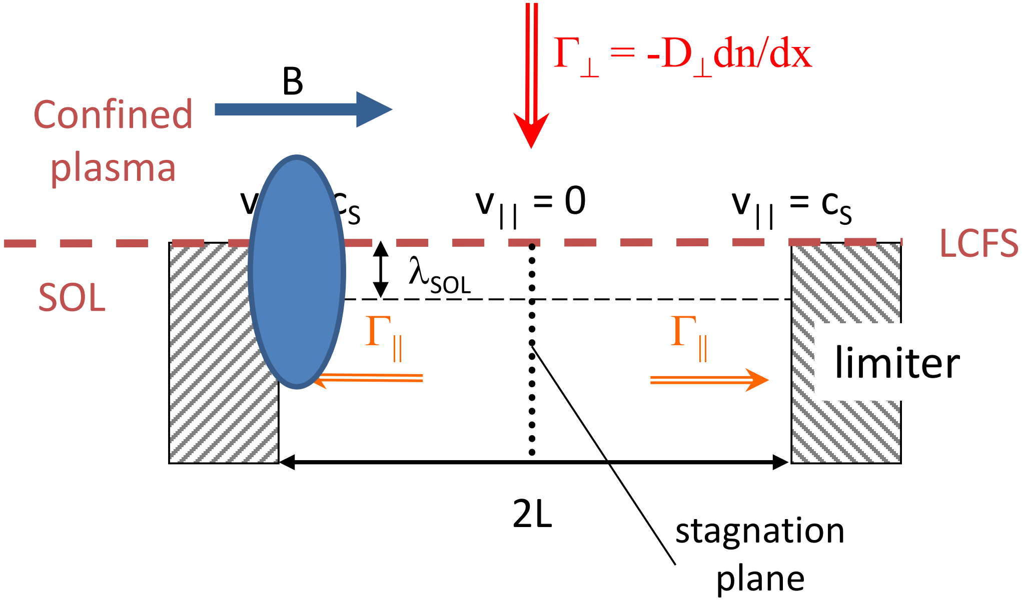


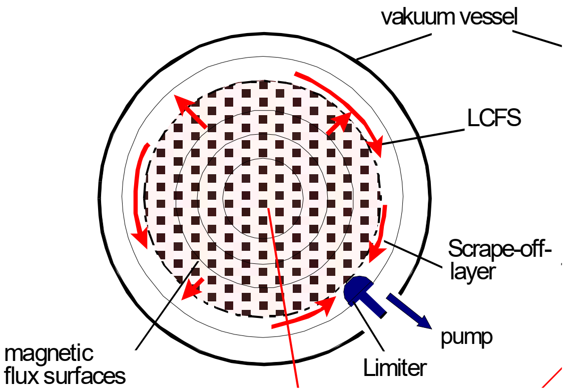
Cold plasma in MCF:
SOL and divertor region

- Wide parameter space at low temperature (close to laboratory plasmas)
  - Electron temperature : **~50eV** down to **<1eV**
  - Electron density: **~10^{17}m^{-3}** up to **~10^{21} m^{-3}**
  - Operational regime: **ionizing** plasma **recombining**

Simple Model for the Edge Plasma

Unfold limiter arrangement



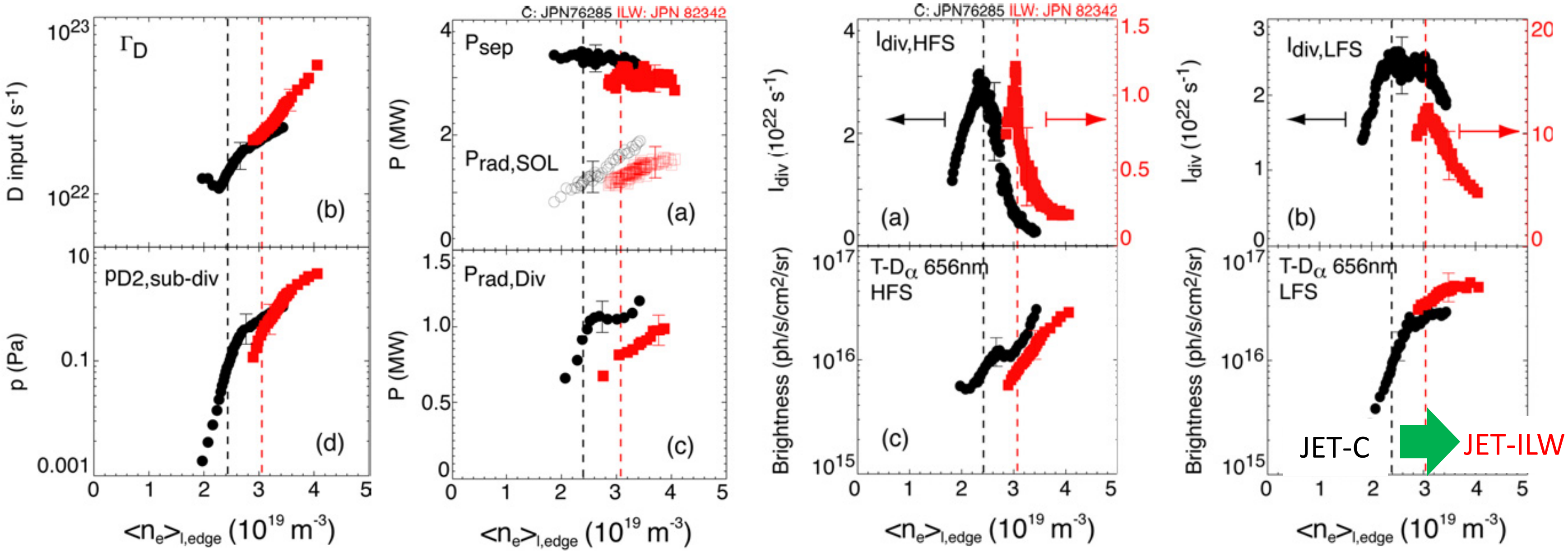


Formation of a neutral cushion in front of the target plate loss of momentum of impinging plasma ions along B: Recycling, + Radiation + Friction + Recombination ...

- Particle flux to edge due to cross field transport $G_\perp = G_{||}$
- Wall is plasma particle sink
- Plasma-wall interaction processes occur e.g.
  - Recycling of fuel species
  - Erosion of fuel materials
- Identical for divertor target plates
- Detachment: neutral cloud before target plate which prevents ions to hit wall

Complete Detachment in Divertor due D_2 Fuelling

- During a fueling ramp in the divertor one passes all steps from recycling via “ion flux roll over” to detachment

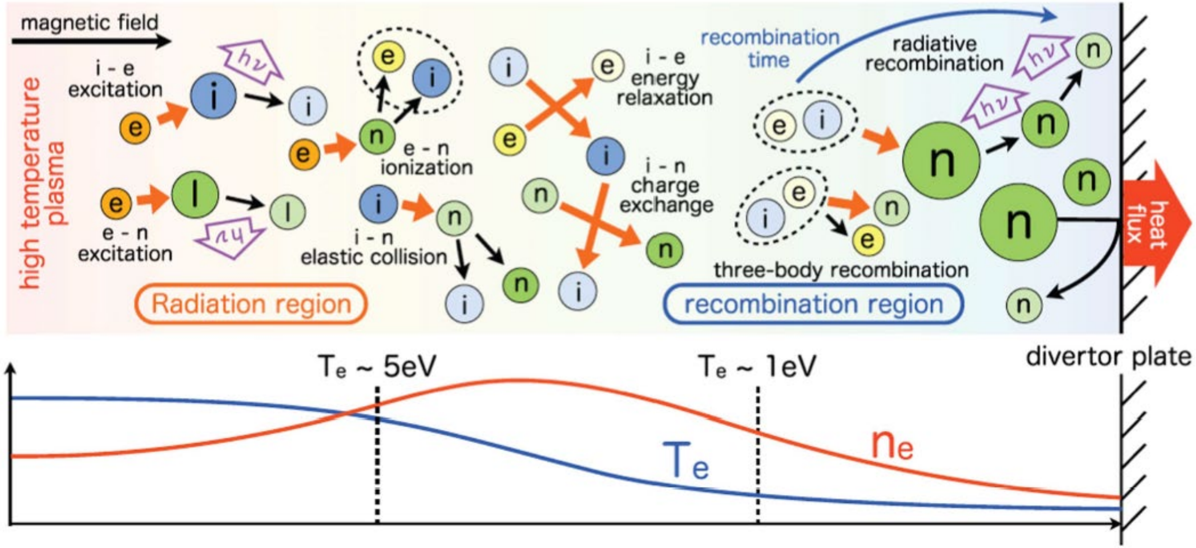


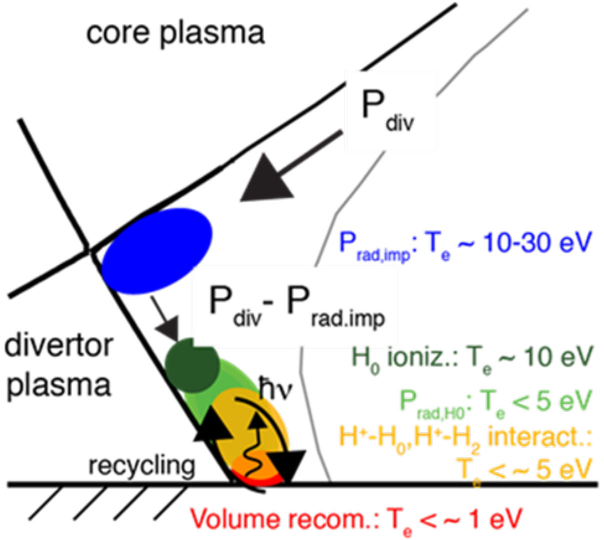
- JET-ILW stable access to highly radiative discharges as required for ITER (f_{rad} ~70%)

M. Groth et al. Nucl Fus. 2013, S. Brezinsek J. Nucl. Mater. 2015

Detachment Process

- Multiple step process including roll-over of ion flux at the target plate and recombination as strongest signature for detachment



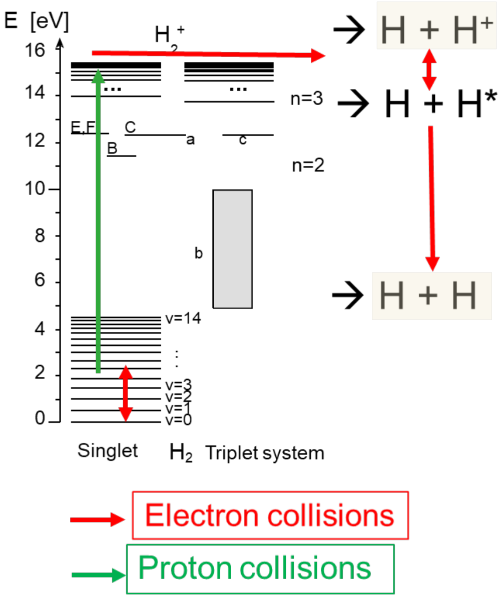


- Strength of detachment described by comparison with upstream plasma conditions
- High recycling / partial detachment / pronounced detachment / complete detachment
- Detachment can be induced at higher power by additional impurity radiation (e.g. Ne and N_2)

Comparison of Relevant Processes in Detachment

Detached

$T_e \approx 1 - 3$ eV



Molecular **A**ssisted **R**ecombination **MAR**

$H_2(v) + H^+ => H_2^+(v) + H$ charges exchange with

$H_2^+(v) + e^- => H + H^*$ dissociative recombination

Molecular **A**ssisted **D**issociation **MAD**

$H_2(v) + H^+ => H_2^+(v) + H$ charges exchange with

$H_2^+(v) + e^- => H + H^+ + e^-$ associated dissociation

n_e~10^{19}-10^{21} m^{-3} and T_e~2-10eV

Electron **I**on **R**ecombination **EIR**

$H^+ + e^- + e^- => H + e^-$ three body recombination or

$H^+ + e^- => H + h\nu$ radiative recombination

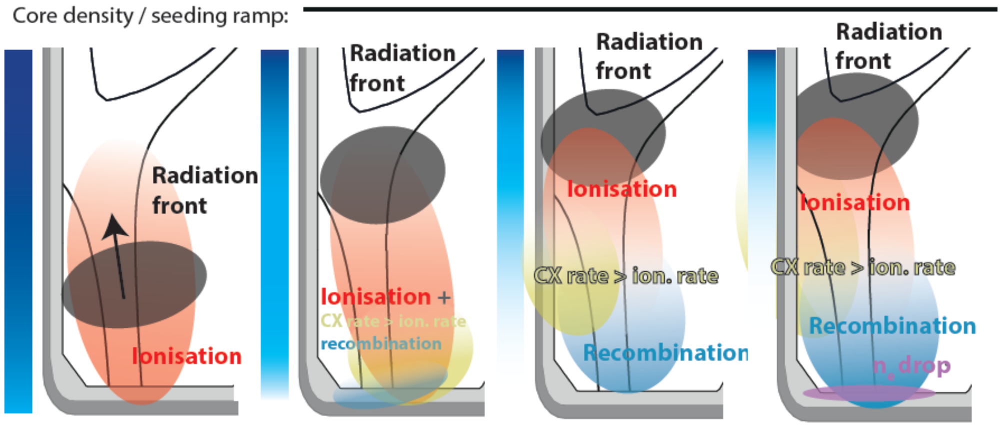
n_e~10^{20}-10^{21} m^{-3} and T_e<1.5eV

See U. Fantz, D. Wünderlich, D. Reiter, R. Janev, K. Sawada etc.

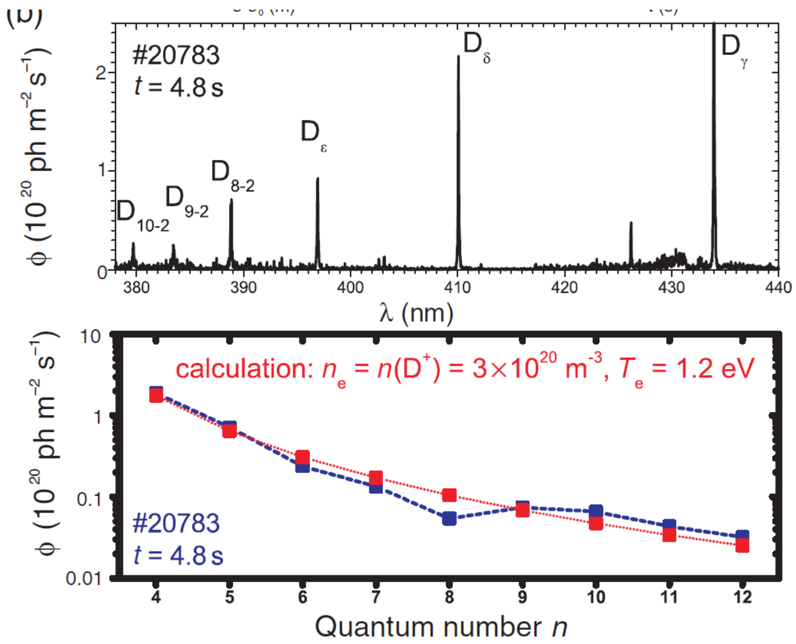
- Competition between different processes sensitive to the plasma conditions => Neutral particle codes like EIRENE

Ionisation and Recombination

Usage of Balmer lines or NII lines (seeding) to locate ionization front

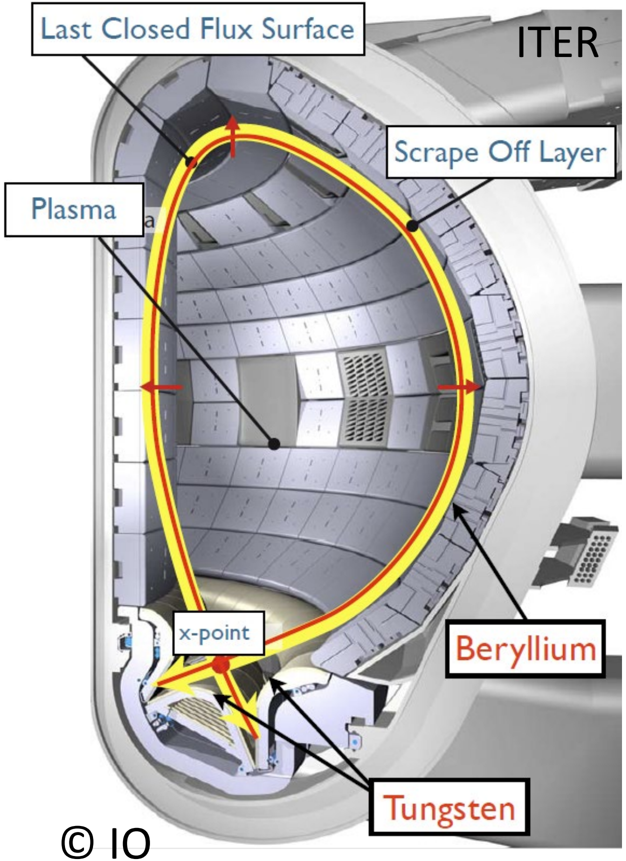


K. Verhaegh (TCV)



Plasma conditions determined by e.g.
line ratios and appropriate CRMs (e.g. YACORA)

Next step device ITER



© IO

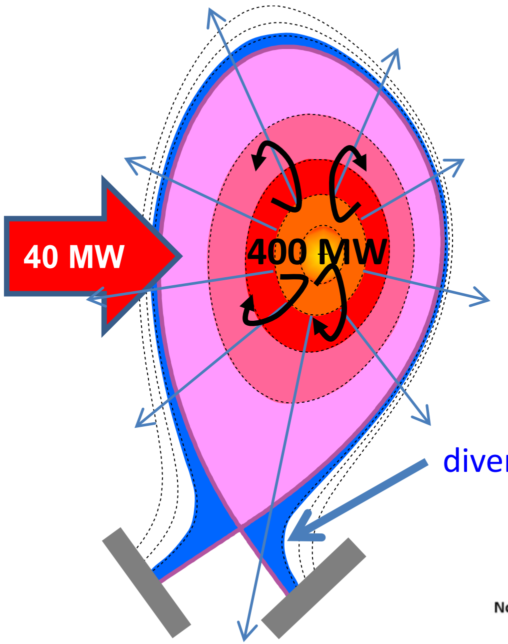
- To demonstrate (i) scientific and (ii) technical /plasma-surface interaction feasibility of fusion
- To achieve extended burn in inductively-driven DT plasma operation with Q=10 (400s)
- To demonstrate readiness of essential fusion technologies (incl. plasma-facing components)

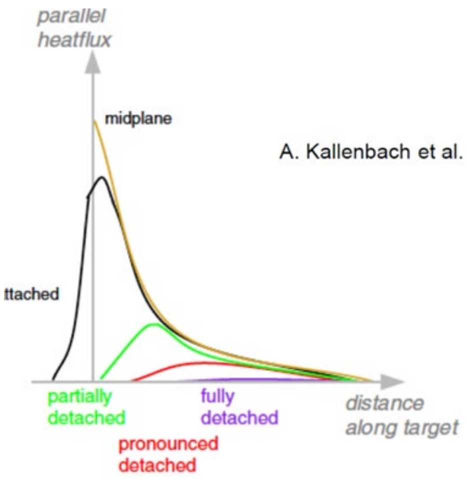
- Neutrons trar … nd blanket modu …
- He-ash (resid … of the plasma on a fa … ement time
- Particles (D^+, … nto the SOL and stream to … ancing incidence

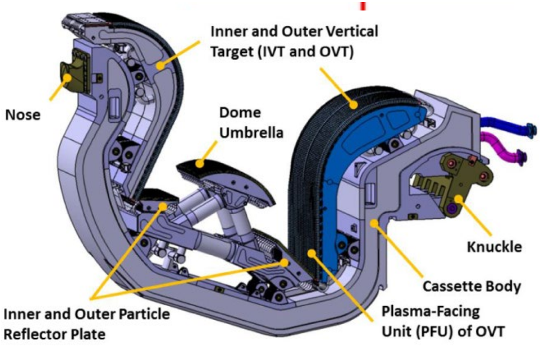
Major Radius: 6.2 m
Minor Radius: 2.0 m
Plasma volume: 840 m^3
Surface area: 260m^2 W and 620m^2 Be
Plasma current: 15 MA
Magnetic field: 5.3 T (12 T)
Energy content: 350 MJ
Auxiliary heating: 70-100 MW
Height: ~25 m and Diameter: ~26 m

The Power and Particle Exhaust Issue for ITER

ITER with Q=10







400 MW fusion power in steady-state

320 MW leave as neutrons
converted to heat in the blanket

80 MW heat plasma (α particles)
40 MW additional heating for control

120 MW have to be transmitted to cooling system towards plasma-facing components:

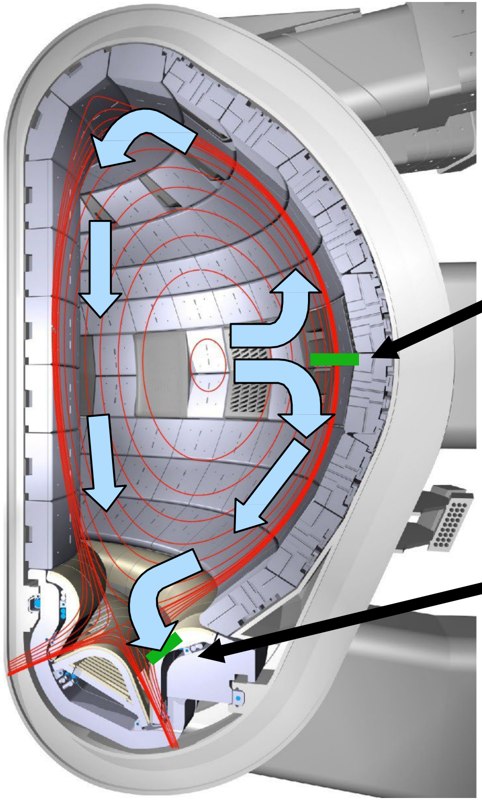
20 MW core radiation and 100 MW to divertor

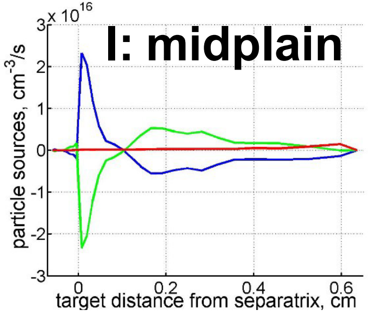
100 MW to divertor over 2.5 m^2 area
=> 40 MWm^{-2} => semi-detachment required

Relative importance of plasma flow forces over chemistry and PWI: I edge region → II divertor

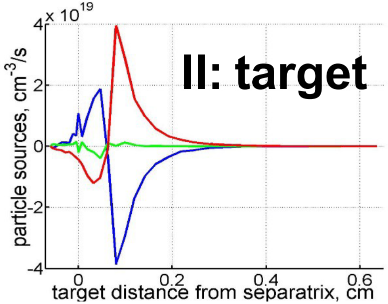
$$\boxed{\vec{\nabla}_{\parallel}(n_i\vec{V}_i)} + \boxed{\frac{\partial}{\partial x}D_{\perp}\frac{\partial n_i}{\partial x}} = \boxed{S_{n_i}}$$

div(nv$_{\parallel}$)+div(nv$_{\perp}$)= ionization/recombination/charge exchange





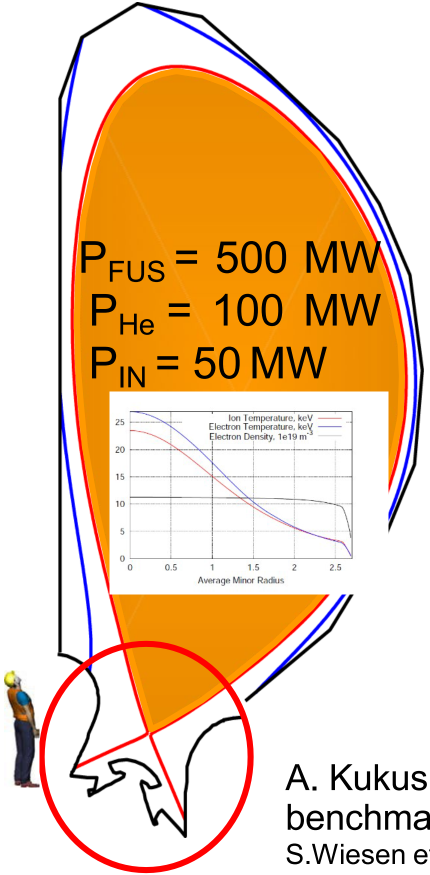
parallel vs. (turbulent) cross field flow



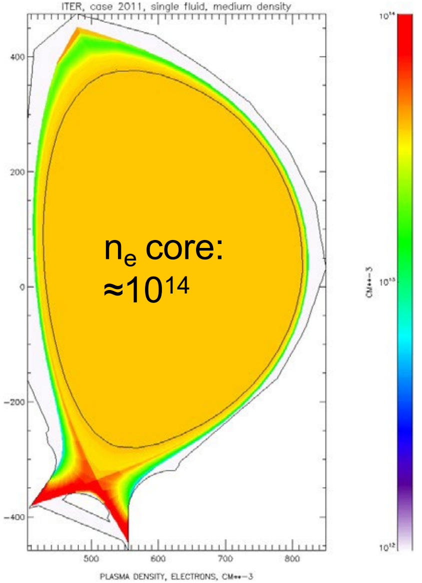
parallel vs. chemistry and PWI driven flow

important process:
friction: p + H_2, recomb., charge exchange } **detachment**

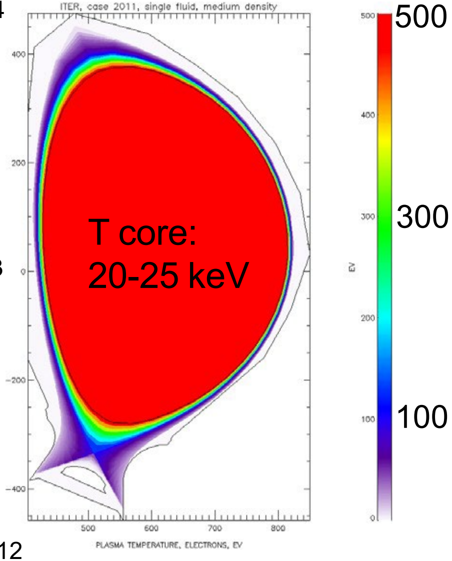
ITER plasma simulation: SOLPS-ITER



Plasma density
cm^{-3}, log scale



Electron temperature
eV, lin. scale

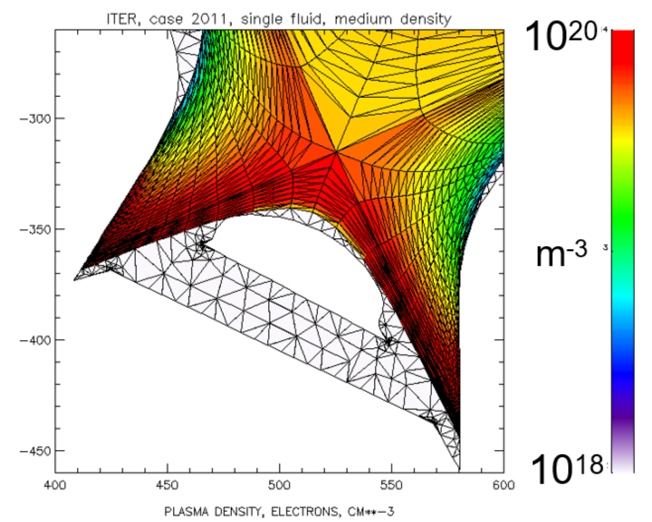


A. Kukushkins SOLPS database Frequently used benchmark e.g. in SOLPS → SOLPS-ITER upgrading
S.Wiesen et al., 21st PSI Kanazawa, J. Nucl. Mat.2014

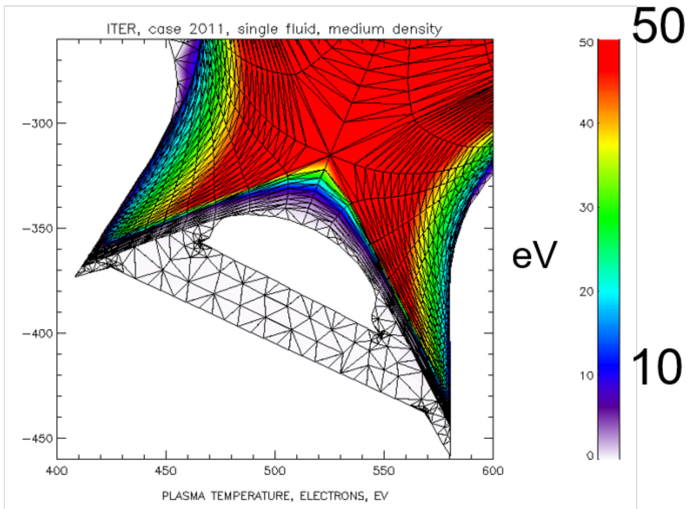
ITER Divertor Plasma Solution: DETACHMENT

- Plasma solution with SOLPS-EIRENE with C exists / Update: SOLPS-ITER for Ne,N_2 and D plasmas
- Self sustained dense, cold plasma layer (≈ 1 – 3 eV) formed in front of divertor components
- Plasma flux drops, despite increased density. Momentum loss predicted.
- ELM burn through not considered / Atomic and molecular data for T, DT not included

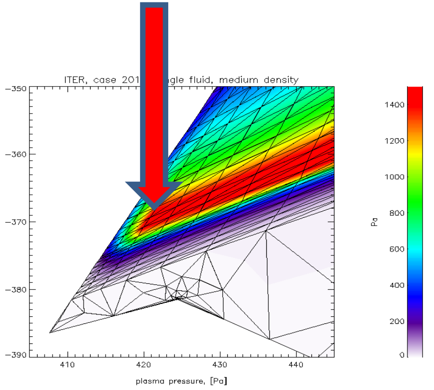
Divertor plasma density
m^{-3}, log scale, 10^{18} - 10^{20}



Divertor electron temperature
eV, lin. scale, 0 – 50 eV

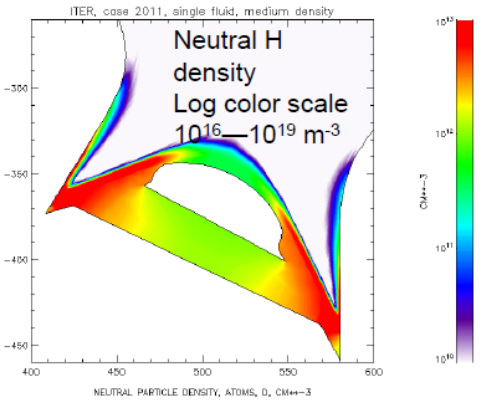


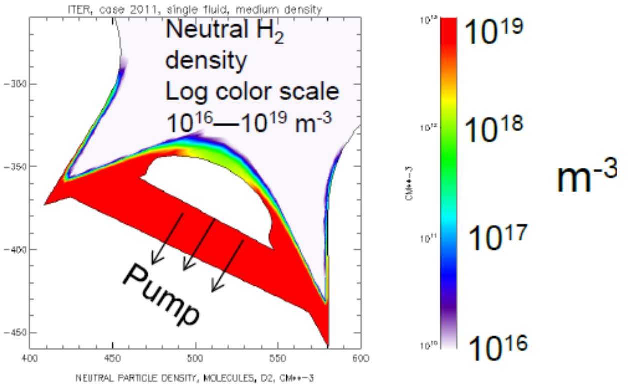
Pressure drop

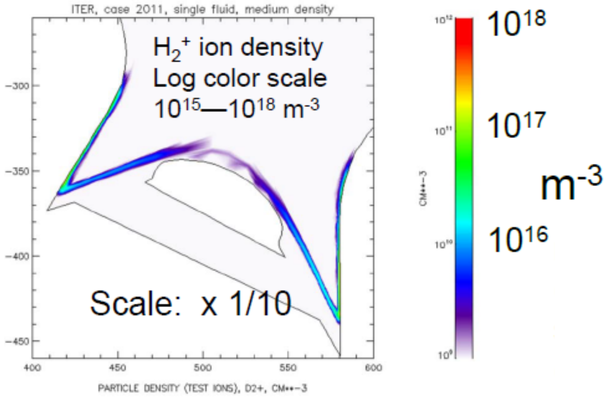


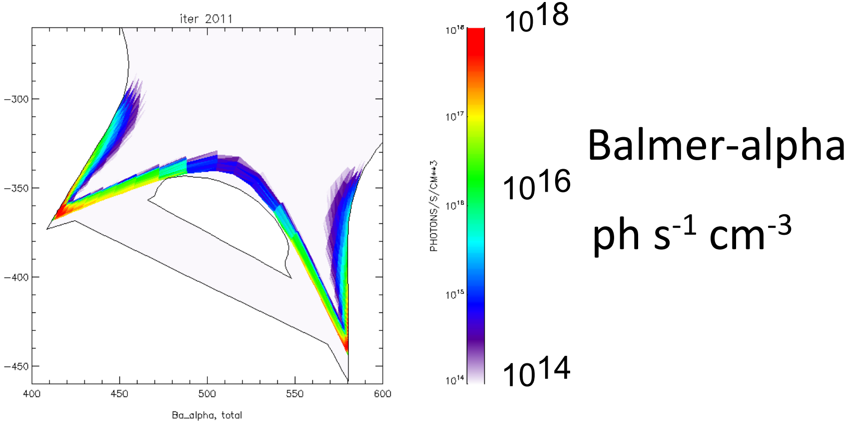
D. Reiter et al. PSI2018

ITER: Self-sustained Neutral Gas Cushion H and H_2



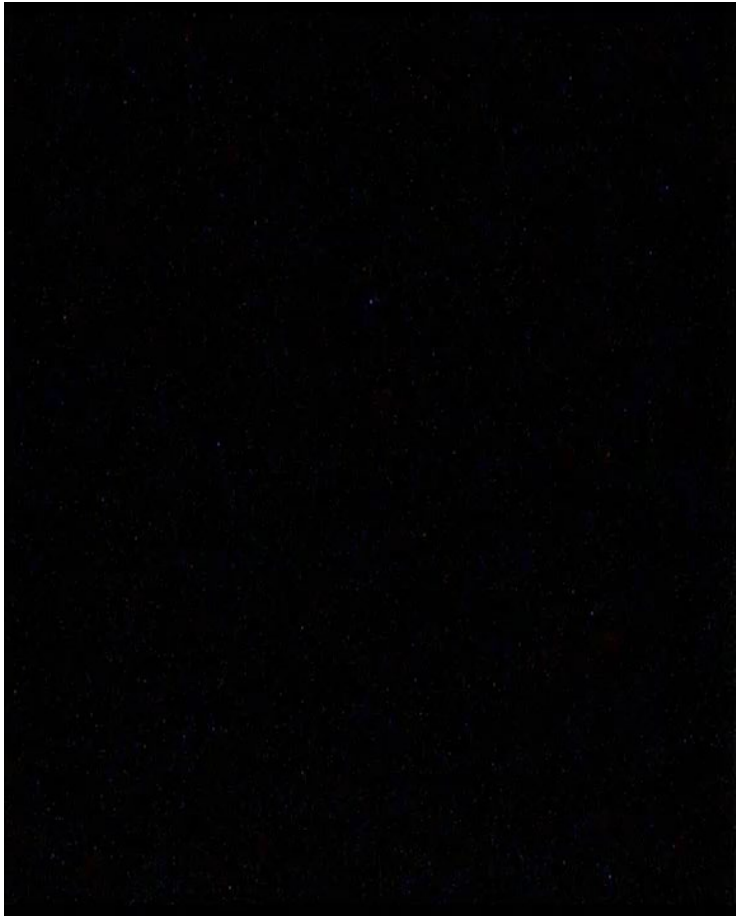


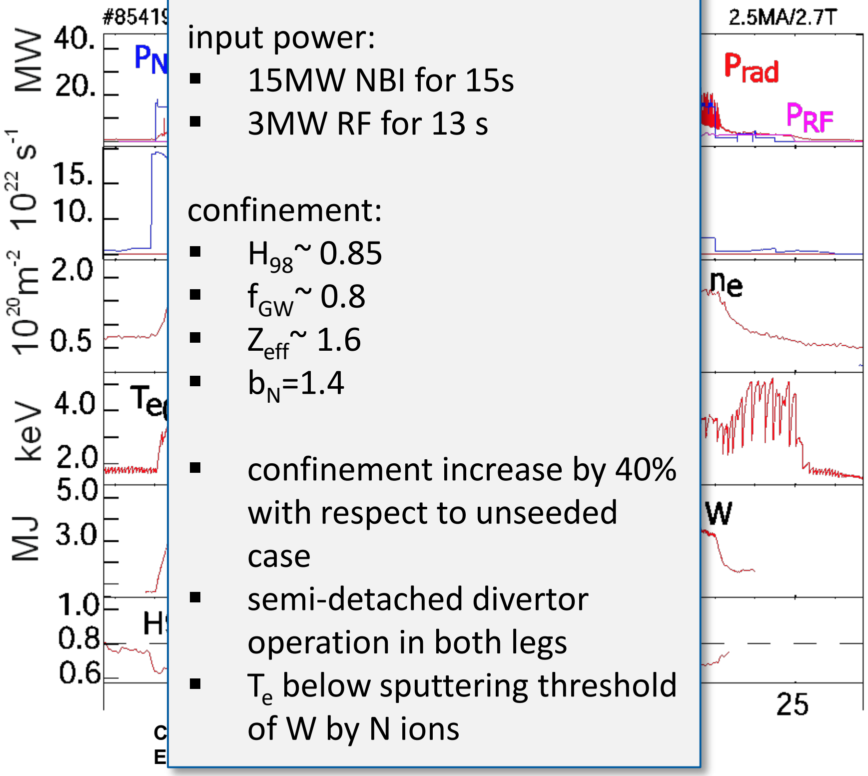




D. Reiter et al. PSI2018

Integrated Scenario with N_2 Seeding





Stationary plasma conditions:

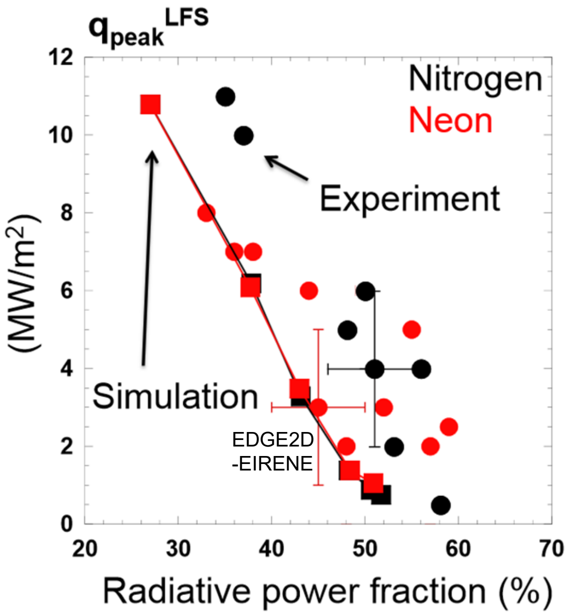
input power:
- 15MW NBI for 15s
- 3MW RF for 13 s

confinement:
- H_{98}~ 0.85
- f_{GW}~ 0.8
- Z_{eff}~ 1.6
- b_N=1.4

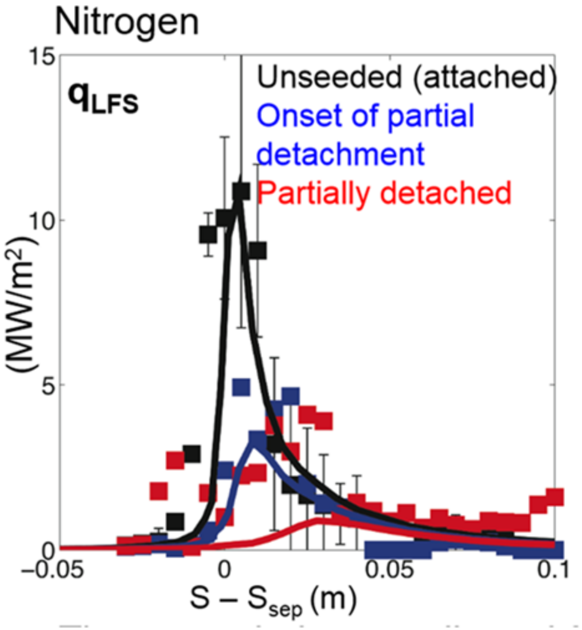
- confinement increase by 40% with respect to unseeded case
- semi-detached divertor operation in both legs
- T_e below sputtering threshold of W by N ions

Divertor Detachement with N_2 Seeding in H-mode

- Full detachment at outer target plate: sequence of power detachment (nitrogen radiation), momentum detachment (D ion-neutral friction) and particle detachment (D volume recombination) reproduced
- Compatibility with power loads and W source for long-pulse operation like in ITER
- Nitrogen radiates predominantly in divertor whereas neon in the edge layer of JET

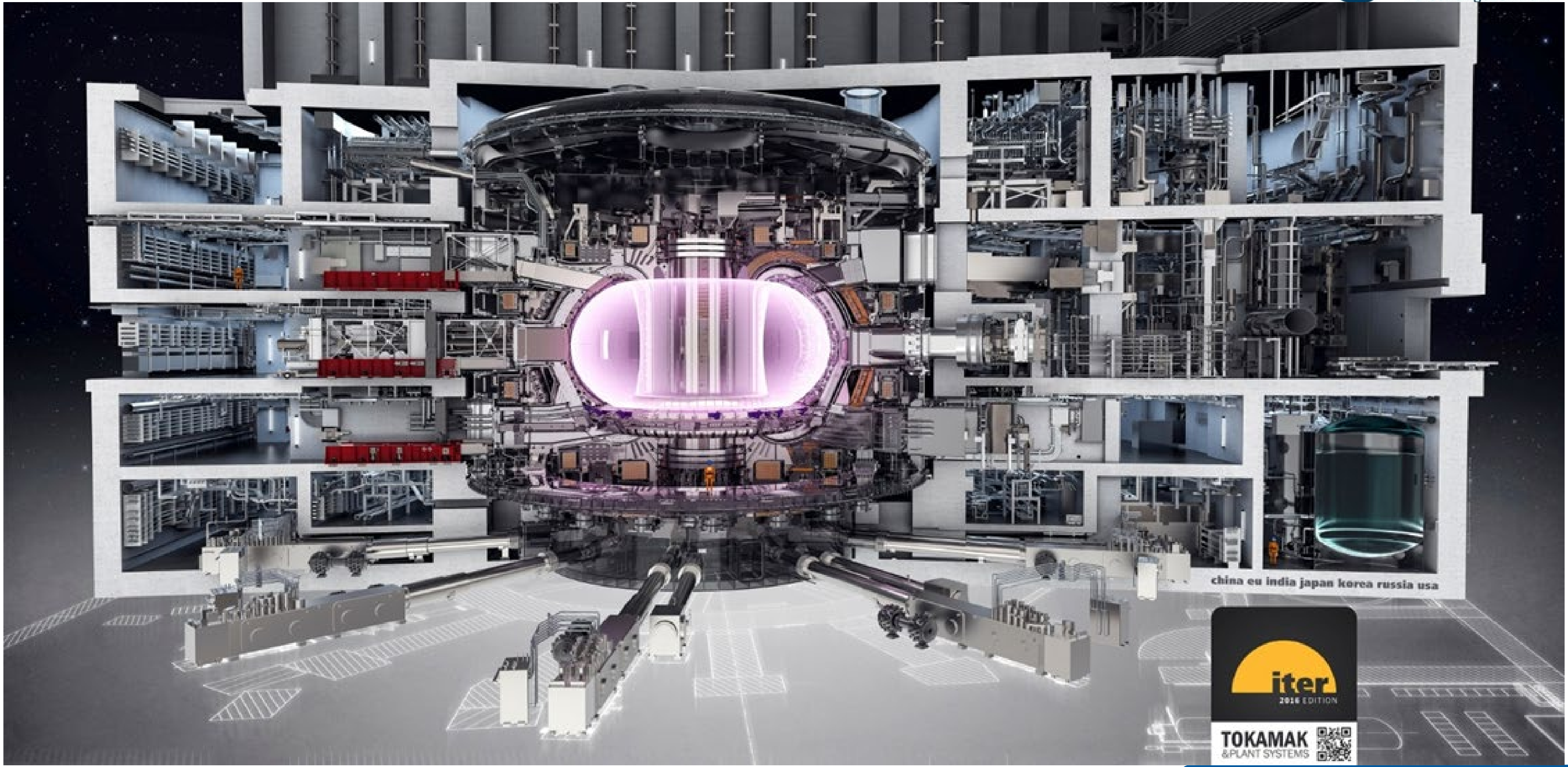


C. Giroud et al.
NF 2014



A. Jarvinen et al.
JNM 2015

ITER Tokamak Building and Infrastructure



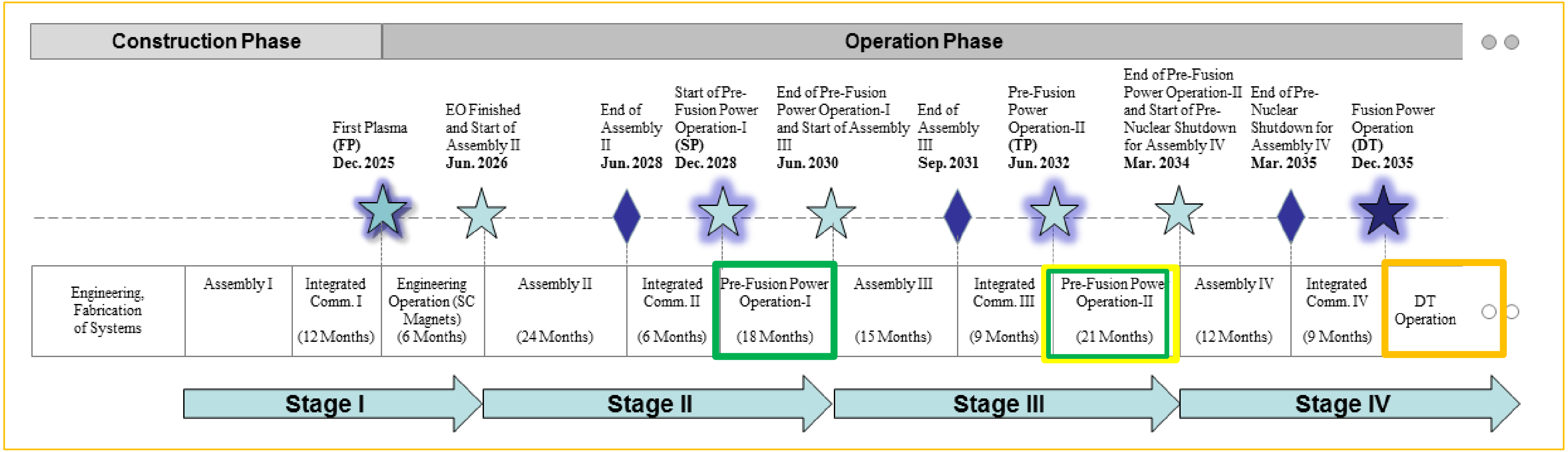
ITER Timeline

Step-wise construction of the tokamak and accompanied exploration program

limiter
plasma (H)

PFPO
divertor plasma (H, He)

FPO
divertor plasma (D, DT)



ITER research plan (2018)
https://www.iter.org/technical-reports

Outline

- Nuclear Fusion
  - Introduction
  - Magnetically Confined Fusion (MCF)
  - Scaling laws and their validity
- JET and ITER
  - Joined European Torus (JET)
  - Plasma-wall interaction
  - Power and particle exhaust challenge
  - International Thermonuclear Experimental Reactor (ITER)
- DEMO and other concepts
  - Challenges
  - SPARC, ARC and tokamaks
  - Wendelstein7-X and HELIAS reactor

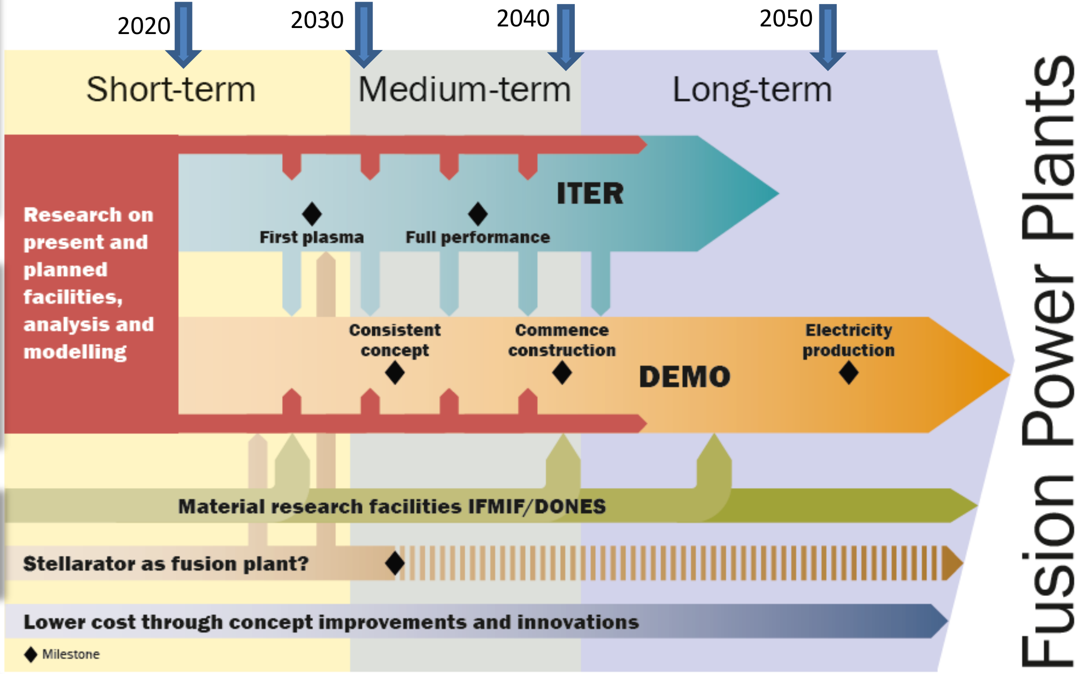
EU Roadmap for Nuclear Fusion

Present goal in Roadmap:
- *"DEMO represents the last step driven by the research community"*
- *"After DEMO, industry will lead fusion power plant production with limited involvement of the research community"*

Two additional goals compared to ITER
- Electricity production
- Tritium self-sufficiency

Not fully addressed in DEMO:
- Neutron-tolerant materials → IFMIF-DONES

Issue:
- Best plasma power and exhaust scenario?



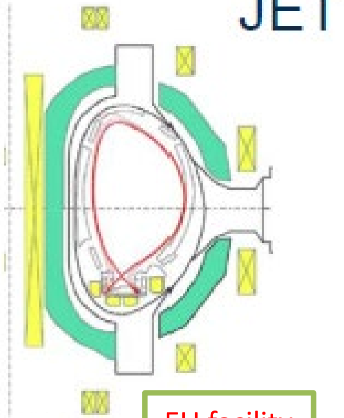
Conservative Step-Ladder-Approach

Same solution: LSN, H-mode, shape, magnets
Critical issues:

- Plasma scenario: disruptions
- Technology: phase transitions
- Funding: capital cost (>ITER)

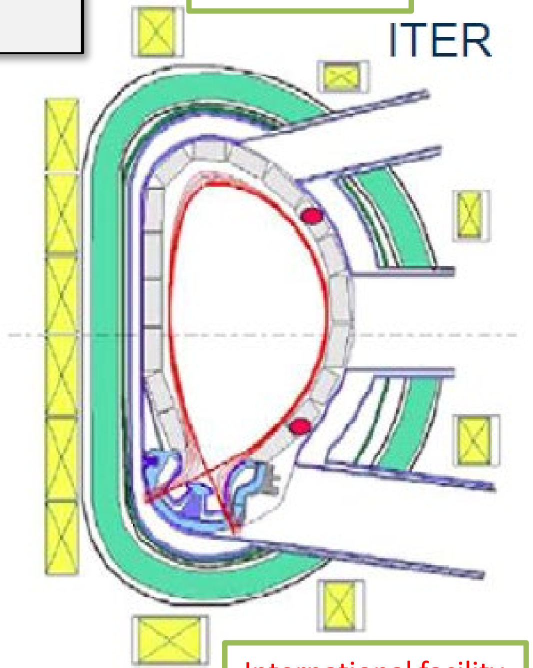
National facility

R_{AUG} x2
P_{fus} 10 MW



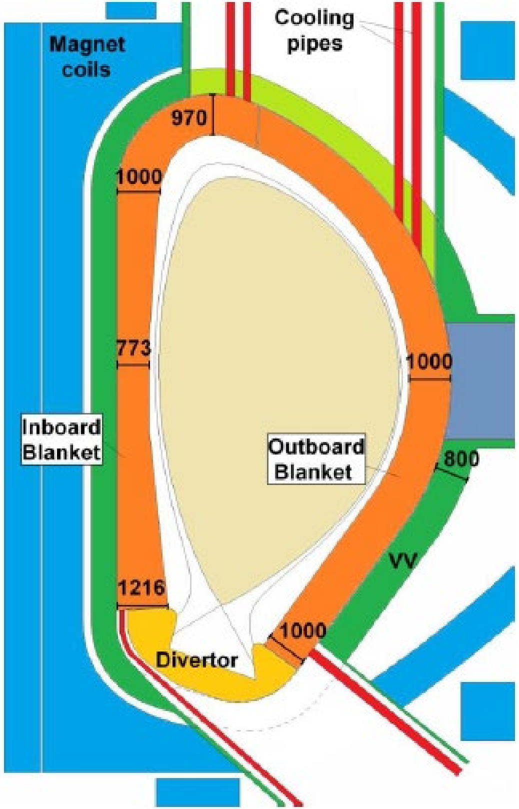
EU facility

R_{JET} x2
P_{fus} 500 MW

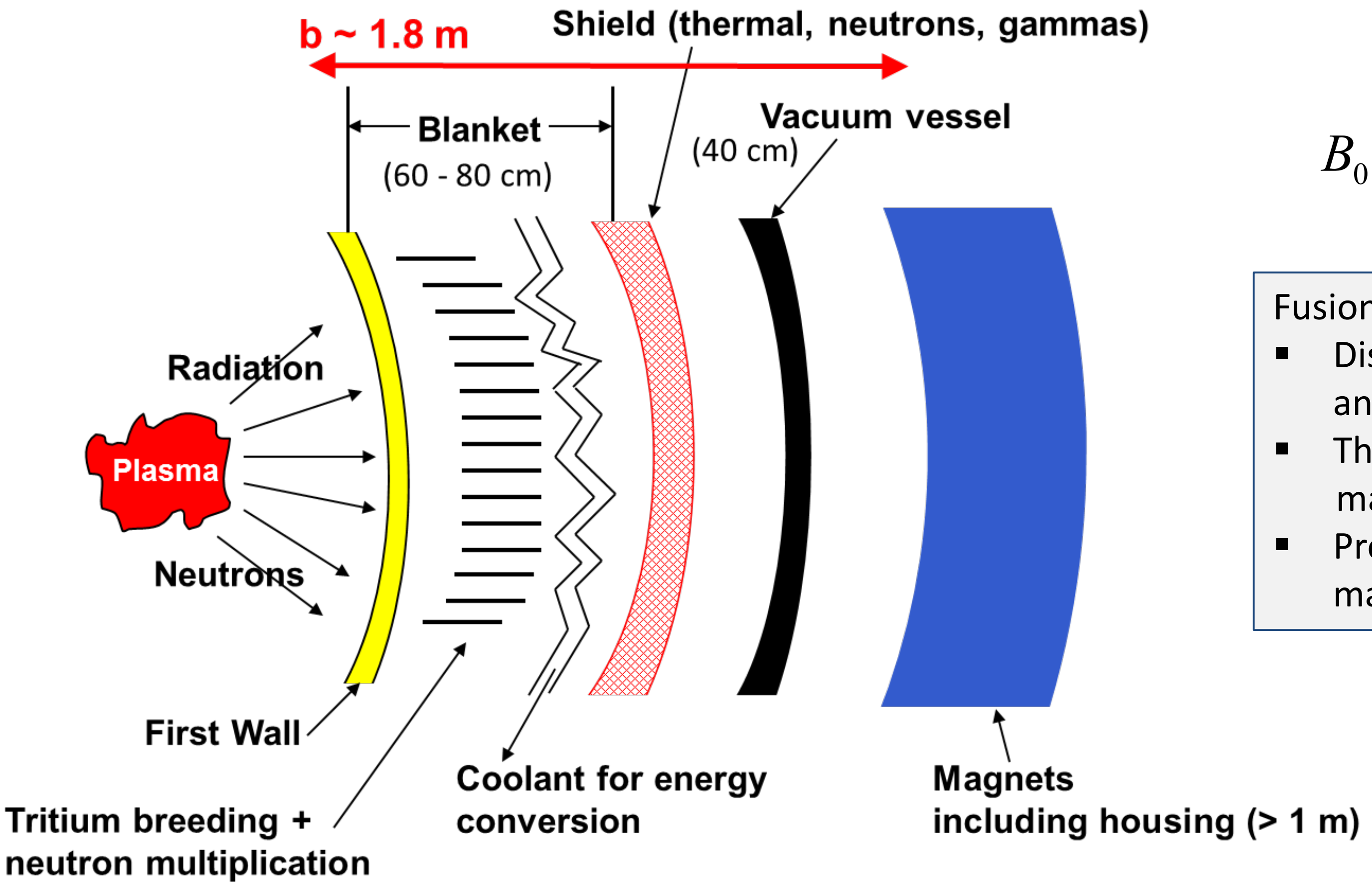


International facility

R_{ITER} x1.5
P_{fus} 2000 MW



“Real“ Wall Dimension with Blanket



$$B_0 = B_{max}\left(1 - \frac{a+b}{R_0}\right)$$

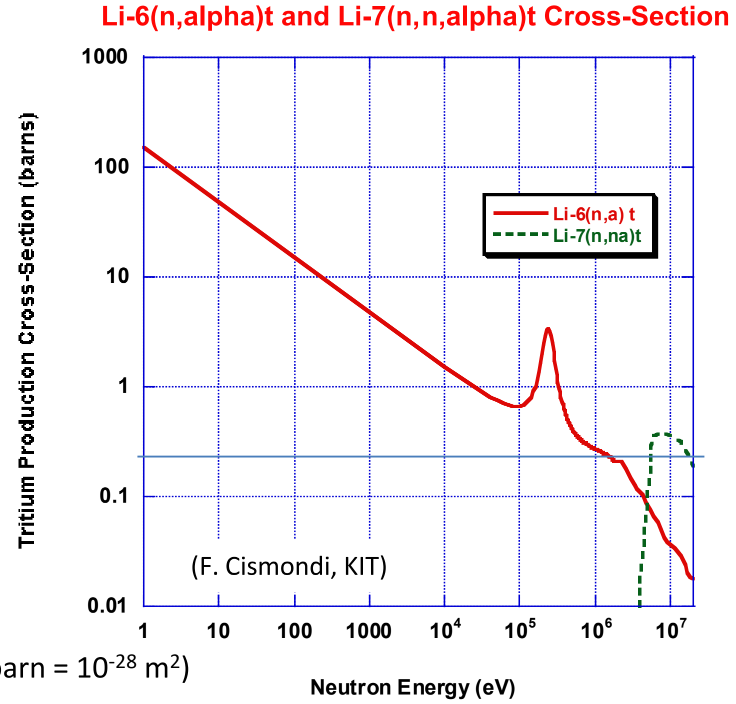
Fusion power plant estimates

- Distance between plasma edge and TF coil casing: **b ~ 1.8 m**
- This value doesn‘t scale with machine size
- Provides envelope for min. machine size for a reactor

F. Cismondi et al.

Tritium Breeding?

- Tritium ($t_{1/2}$ = 12.32 y) currently from fission reactors (external source)
- Tritium to be produced in-situ (internal source = „breeding“)
- Tritium retention in first wall must be minimised as tritium breeding ratio is low



(1 barn = 10^{-28} m^2)

Natural Lithium consists of 7.42% ^{6}Li and 92.58% ^{7}Li.

Possible breeding reactions:

$$^{6}Li + n \rightarrow T + \alpha + 4.78 MeV$$
$$^{7}Li + n \rightarrow T + \alpha + n - 2.47 MeV$$

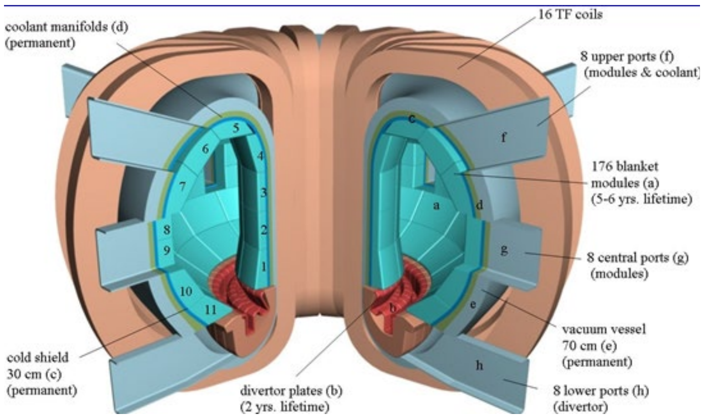
In steady state operation, a fusion reactor should reach a tritium breeding ratio (TBR)~1.2 > 1.

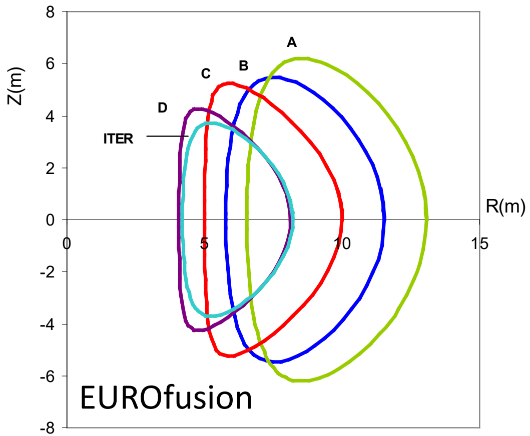
Need of neutron multiplicator: Be

Reactor Design Optimisation (system codes)

Design optimization for a reactor are worldwide ongoing

EUROfusion favours pulsed ITER-like design (up-scaled)





A – simple scaling
B – higher field
C – advanced physics and materials
D – very advanced physics and materials

Safety / Environment

- no need for emergency evacuation, no active systems for safe shut-down, no structure melting following disruption (CO_2 free energy source)
- minimum waste transport, minimum waste to repository (max. 100 years);

Operation

- steady state, ~ 1-2 GWe, base load; Q>30!
- availability 75 ÷ 80 %, with only few unplanned shut-downs/year;

Economics

- public acceptance could be even more important than economics
- economic comparison among equally acceptable energy sources
- construction time ≤ 5 years

Other solutions and ideas speed-up quickly

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Adaption to actual knowledge and society needs

- New physics knowledge => better confinement
- New technical capabilities => higher magnetic field
- Size reduction => reduce impact of disruptions
- Pulses operation => adaption to grid needs

All require information from ITER:

- Alpha particle heating
- Scaling laws / scenarios
- PFC solutions
- T breeding tests

- In line with ITER
  - JT-60SA (Japan)
  - **CFETR** (China)
- Compact tokamaks
  - **SPARC** (USA)
  - **STEP** (UK)
  - DTT (Italy)
  - COMPASS-U (Czech Republic)
- Other initiatives (private)
  - Tokamak Energy (UK)
  - Renaissance fusion (France)
  - Commonwealth Fusion Systems (MIT, USA)

EU-DEMO at ITER side to share infrastrucutre?

https://tracxn.com/d/trending-themes/Startups-in-Nuclear-Fusion

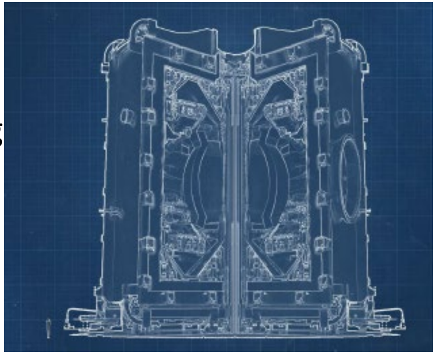
Landscape of reactor studies

- Similar approach but at an intermediate fusion power: CFETR
- Compact approach at a higher magnetic field: SPARC => ARC
- Compact approach with spherical tokamak configuration: STEP

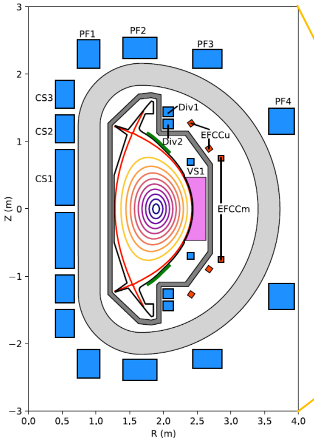
=> ALL DOUBLE NULL and SMALLER in SIZE

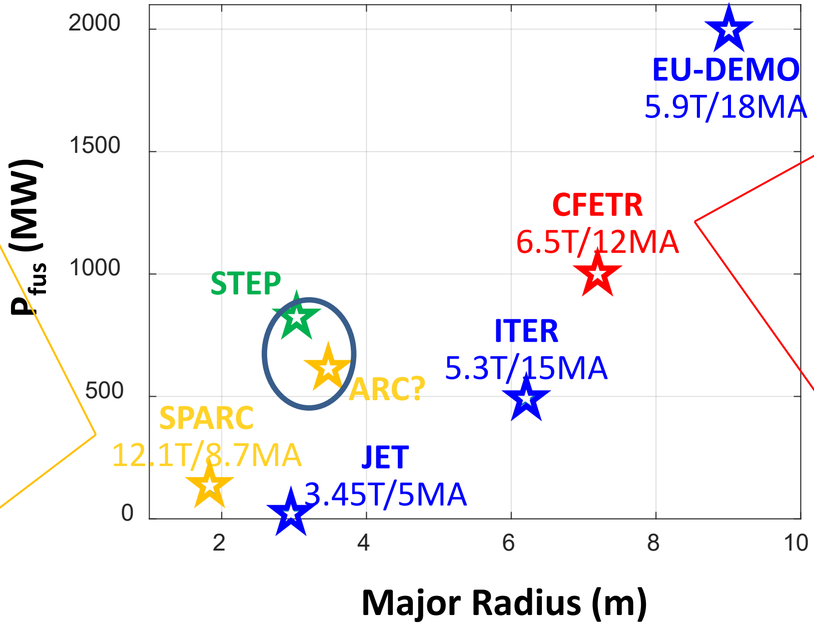
UKAEA + private funding

STEP
UK /UKAEA

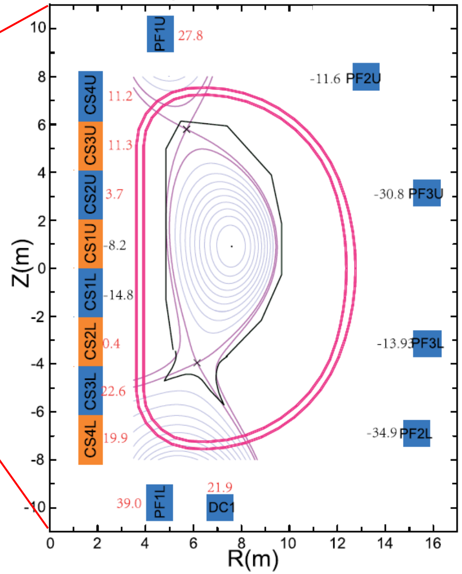


SPARC
MIT / USA





CFETR - CHN



SPARC: MIT + private funding

SPARC Missions and Fast Step Forward

- **Make a plasma with (robustly) break-even fusion energy gain, Q > 2**

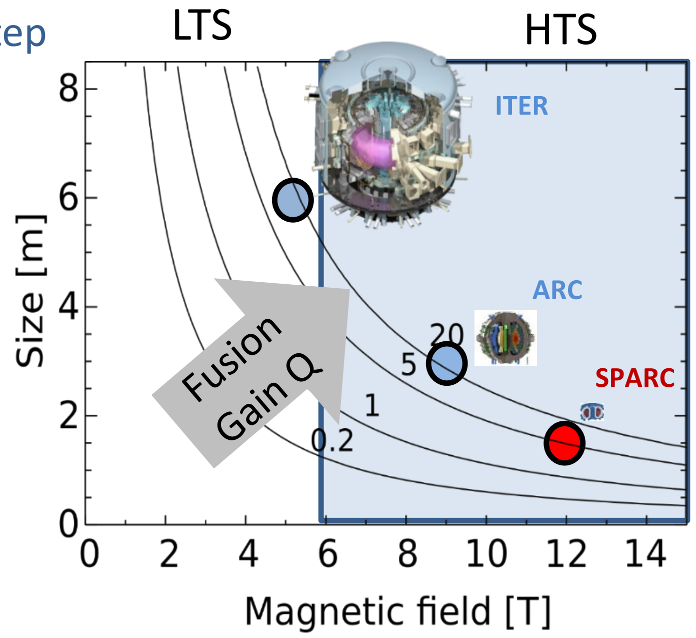
  This would be a sufficient demonstration to put fusion firmly into national energy plans and to attract investments for the next step

- **Demonstrate fusion-relevant HTS magnets at scale**

  Integrated with high-performance tokamak operation

- **Provide the physics basis for high-field pathway**

  Demonstrate high-field, burning fusion plasma scenarios



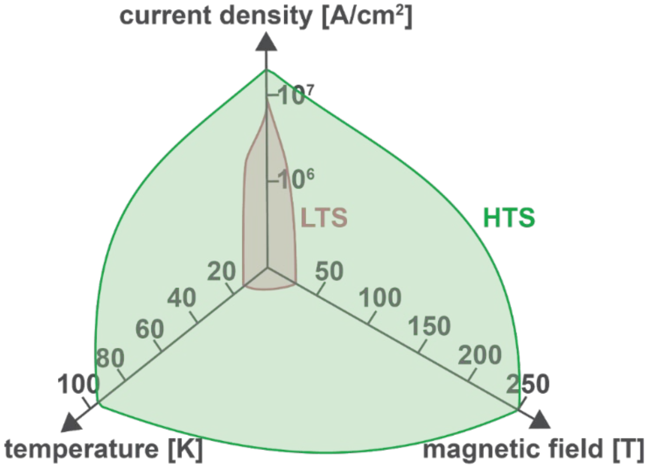
SPARC D. Whyte et al.

- Present limit with LTS (Nb_3Sn) at 12T or so with 6T on torus axis
- Technology jump required …

New Superconducting Materials:
Ideal For Large-Volume, High-Field Fusion Magnets

- Much wider operating space in field, current density and temperature
- Higher current dens
  - Strong structure
- Operation at higher
  - New cryogenic op
  - Better stability
  - Higher heat capa ... nts
- No post-winding reaction process required


Test of 20 T HTS coil in 9/2021



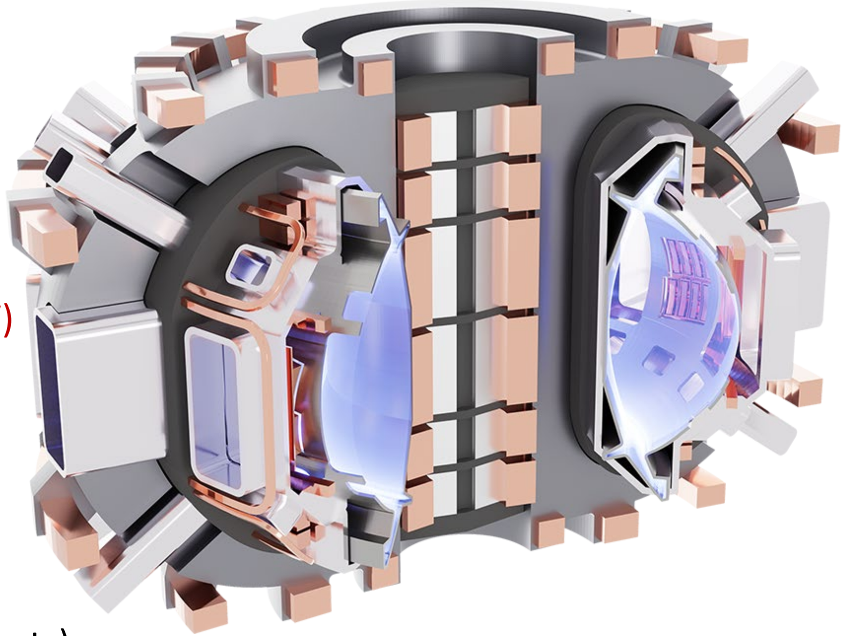
- Key: new High Temperature Superconducting magnets
  - SPARC => Q=2 test at size of AUG, DIII-D, KSTAR
  - ARC => Q=30+ at size of JET

SPARV D. Whyte et al.

SPARC V2: Starting Point for Detailed Engineering Design

SPARC technical requirements:

- Burn D-T fuel
- Fusion Gain: Q > 2
- P_{fusion} > 50MW (up to 140MW)
- Pulsed with 10s flattop burn
- Plasma current: Ip=8.7 MA



Desired schedule:

- R&D: 3yrs (mainly HTS magnets)
- Const.: 4 yrs / Ops.: 5 yrs / Decom.: 4 yrs

Desired construction cost: <$500M

B_0	12.1	T
I_P	8.7	MA
R_0	1.85	m
a	0.57	m
ε	0.31	
κ (area)	1.75	
P_{fus}	50-140	MW
P_{ext}	25	MW

Optimised Stellarator: W7-X (towards HELIAS)

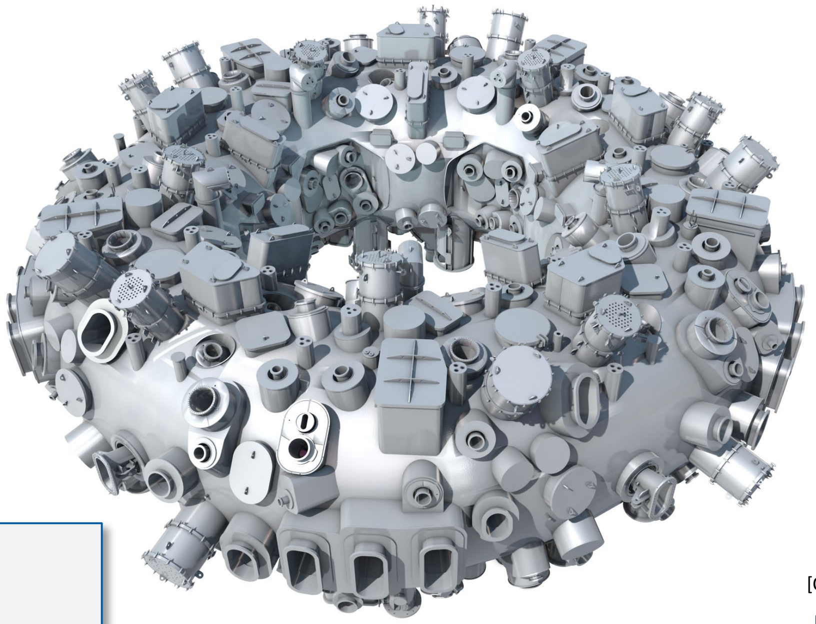
plasma volume: 30 m^3

non-planar NbTi coils: 50

planar NbTi coils: 20

plasma vessel: 80 m^3
in-vessel components: 265 m^2

ports: 254
shapes: 120



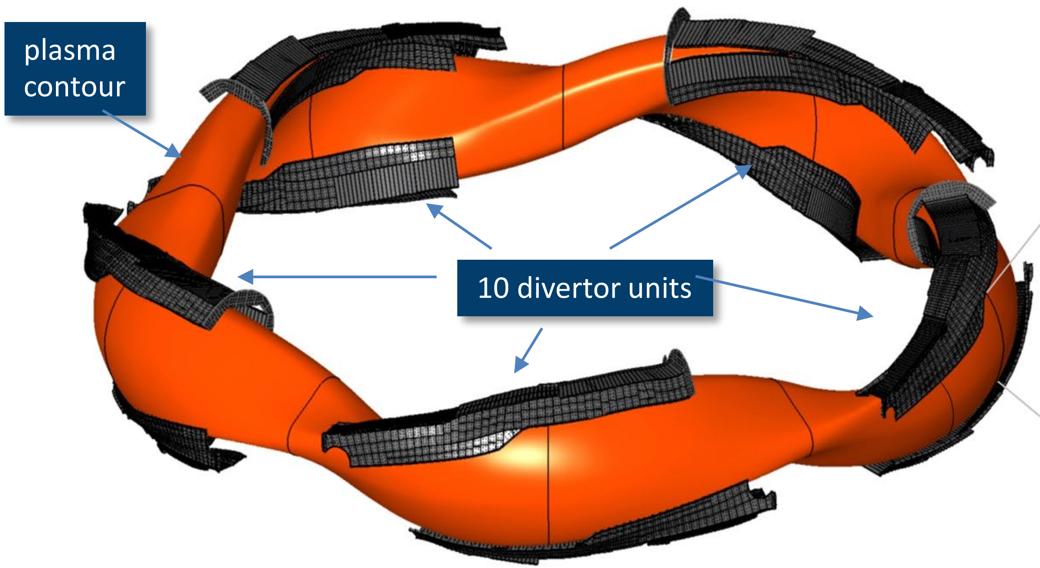
machine height: 4.5 m
machine diameter: 16 m
device mass: 735 t
cold mass at 3.4 K: 435 t

NbTi bus bars: 113
14 HTSC current leads: 14
central support ring elements: 10

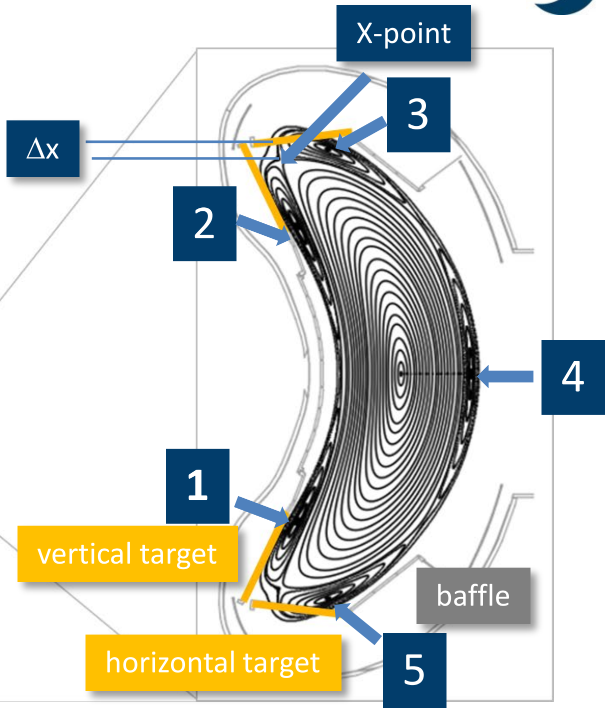
[O. Grulke et al. EPS2019]

cryostat vessel insulation: 420 m^3

Island divertor concept in W7-X



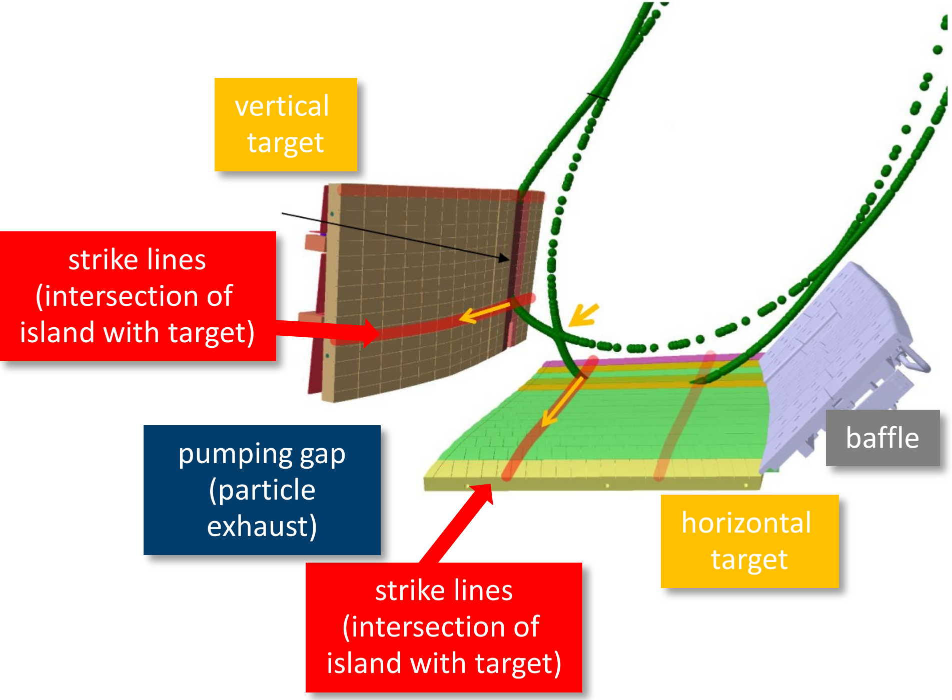
divertor target plates follow the magnetic island geometry in toroidal and poloidal direction

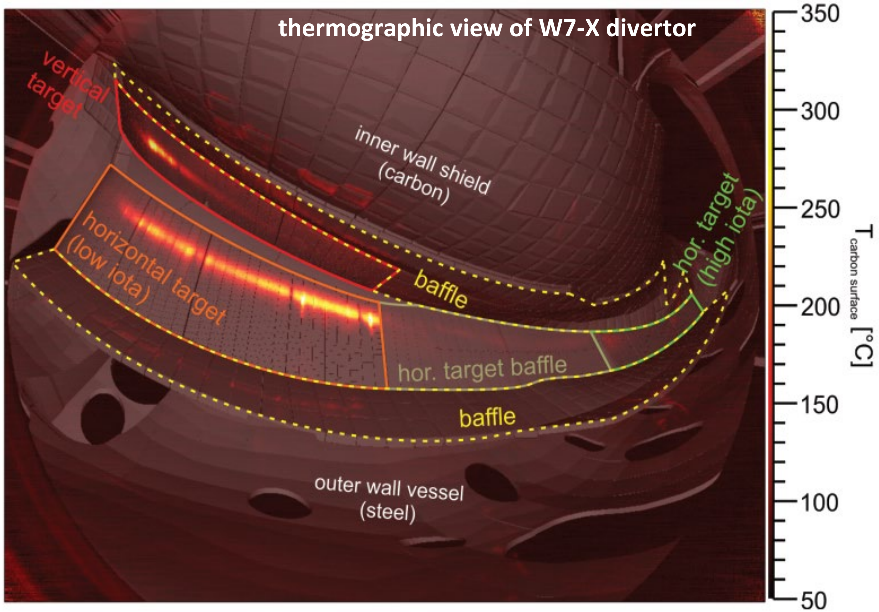


standard divertor solution with 5 magnetic islands

Island divertor concept in W7-X

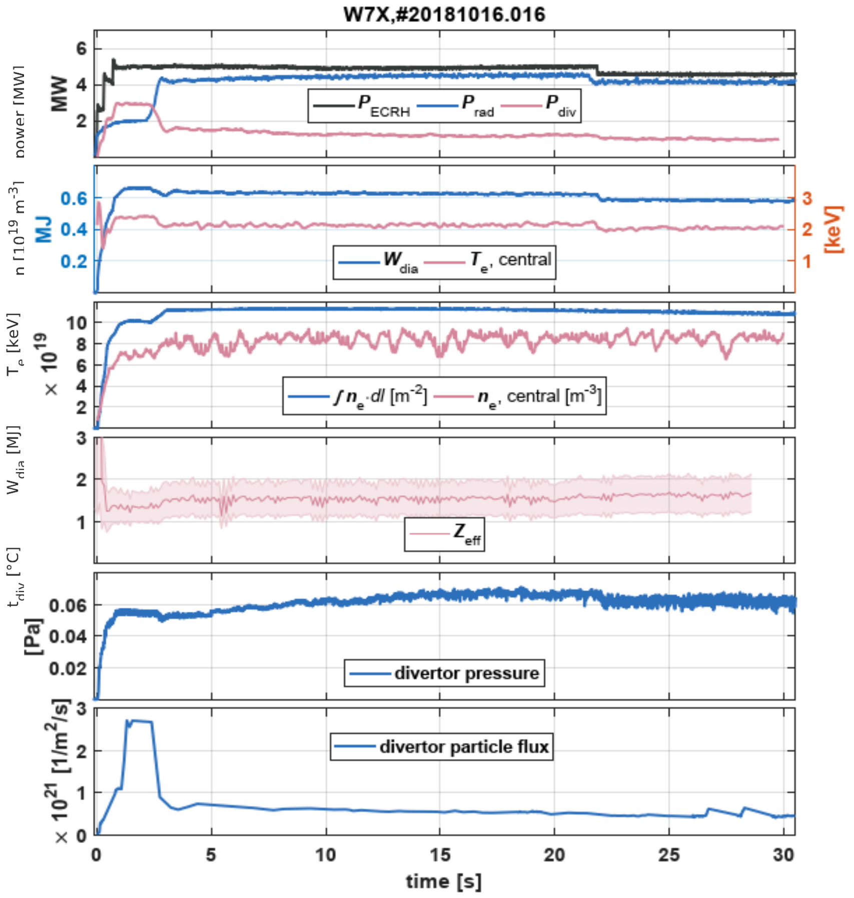
- In standard magnetic configuration two strike lines interact with the target plates
- Visible in heat-flux pattern (power exhaust), imping particle-flux pattern (particle exhaust) and erosion pattern of target plates (impurity production) => main areas of plasma-surface interaction

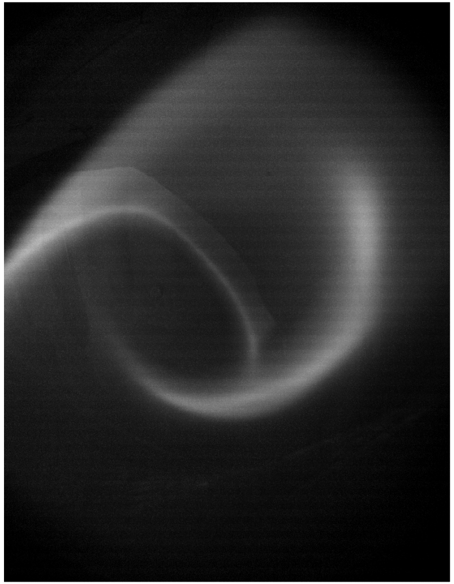


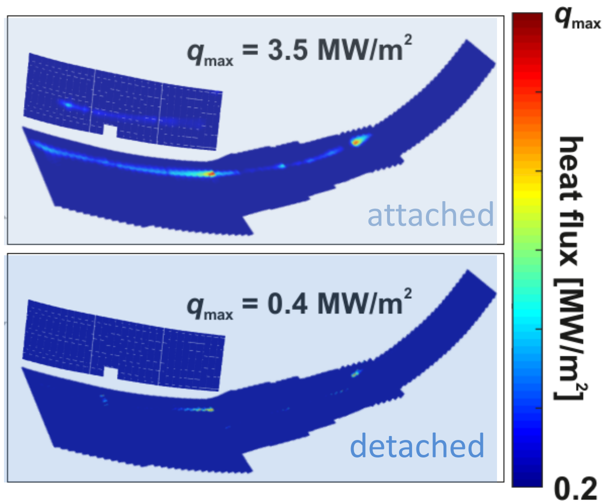


[P. Drewelow et al. PSI 2018]

Stable detachment --- next step actively cooled PFCs



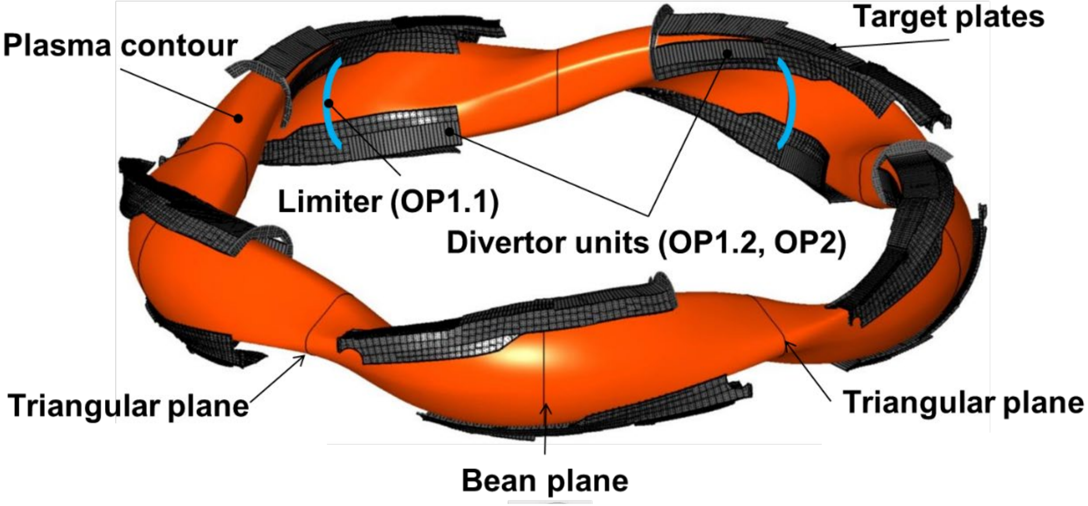




[O. Schmitz et al. , R. König et al. EPS 2019]

- Divertor detachment induced by C+H radiation and H recycling
- Almost no convective heat loads at the target plates
- Symmetric in all five modules with the ten divertor units
- Steady-state operation for 26 s inertially cooled PFCs!

PSI in W7-X with graphite PFCs presented

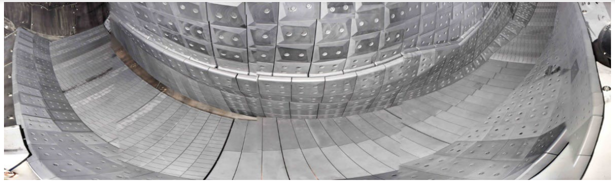


2017 / 2018
- C divertor
- 7.5 MW
- 200 MJ
- 100 s
- H/He

2021/2022
- CFC divertor
- >10 MW
- 18 GJ
- 1800 s
- H, He, D

202X with X > 7
- W divertor
- “optimised” design
- Metallic first wall
- More input power
- H, He, D

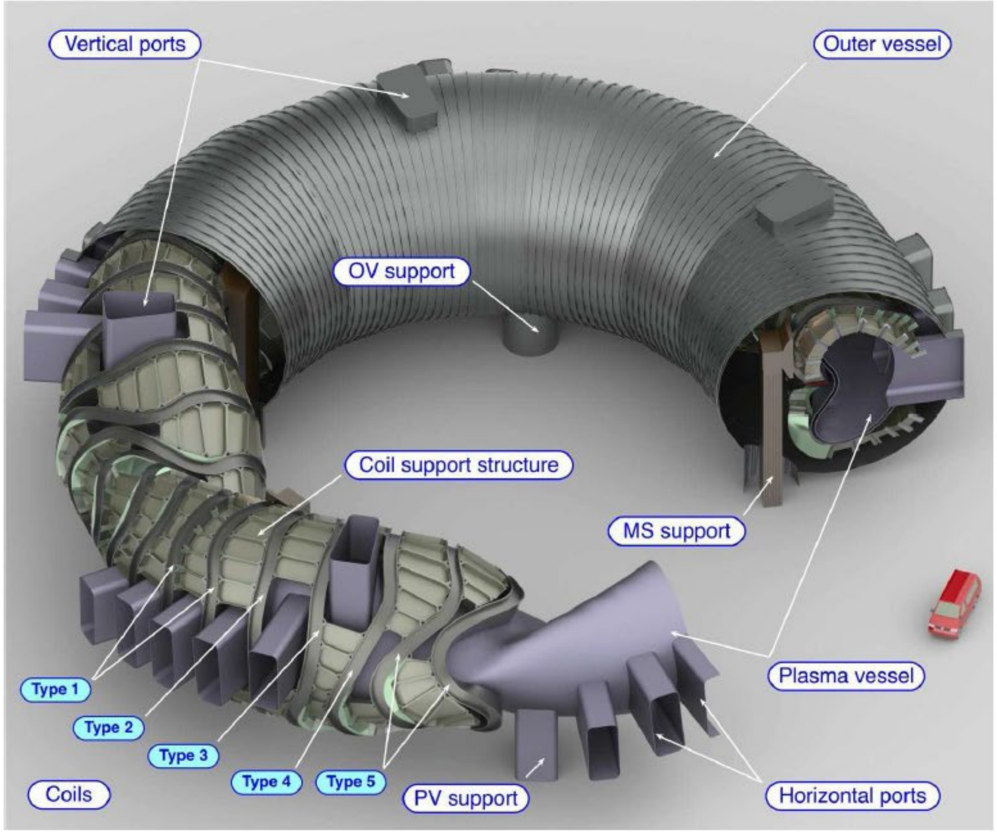
Uncooled graphite divertor (TDU)



Actively cooled CFC divertor

Actively cooled divertor with W PFCs

Stellarator Reactor: Helias 5 b



Symmetry	5 Periods
Coil number	50
Major radius	22 m
Overall diameter	60 m
$B_{avg.}$ on axis	5.9 T
B on coil	12.5 T
Coil current	13.65 MA
Magnetic energy	160 GJ (44.4 MWh)

F. Schauer et al., Fusion Engineering and Design 88 (2013) 1619

Toroidal coil dimensions like in ITER!

Summary and Conclusion

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- Nuclear Fusion with DT reaction works and releases energy
- Present day machines are close to Q=1 (in and out) with metallic plasma-facing materials and relevant scenarios
- Power and particle exhaust with seeding and detachment
- Tritium retention and erosion issues solved for those facilities

- ITER is currently built and simulations/scenarios prepared
- PFPO (H, He) until 2034 / FPO (DT) operation from 2035 on
- ITER is the ONLY facility to demonstrate the full capability of all technologies (incl. breeding) and plasma-facing materials
- EU DEMO developed in parallel and design “ready” by 2030
- Step-ladder approach with size scaling => electricity by ~2055

- Parallel development (including private companies) utilizing the flexibility with HT superconducting magnets
- Permits smaller devices with high Q, less net-power and high flexibility => “fast” track with reactor prototypes by ~2040+

Nuclear fusion can complement renewables by second half of century!

It can replace gas, coal, oil or fission in the energy portfolio on world scale!

Back-up