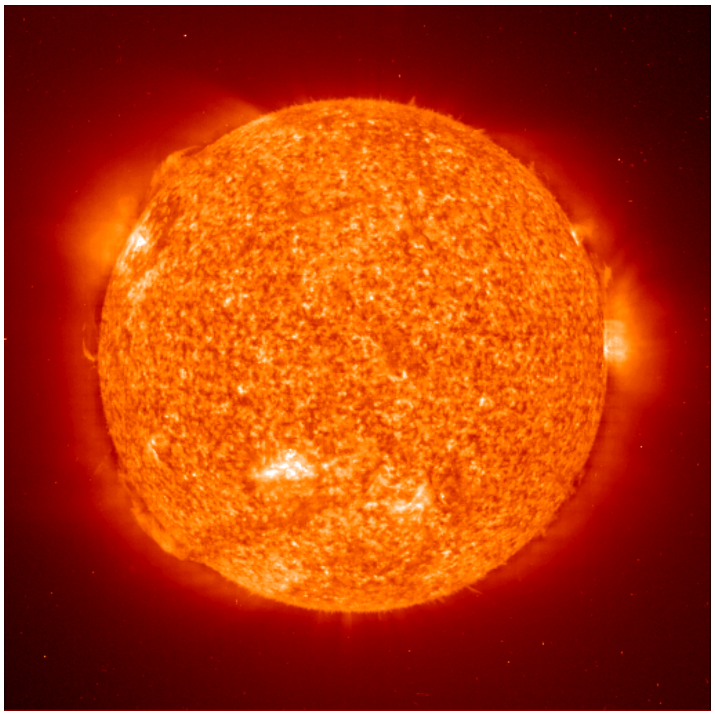
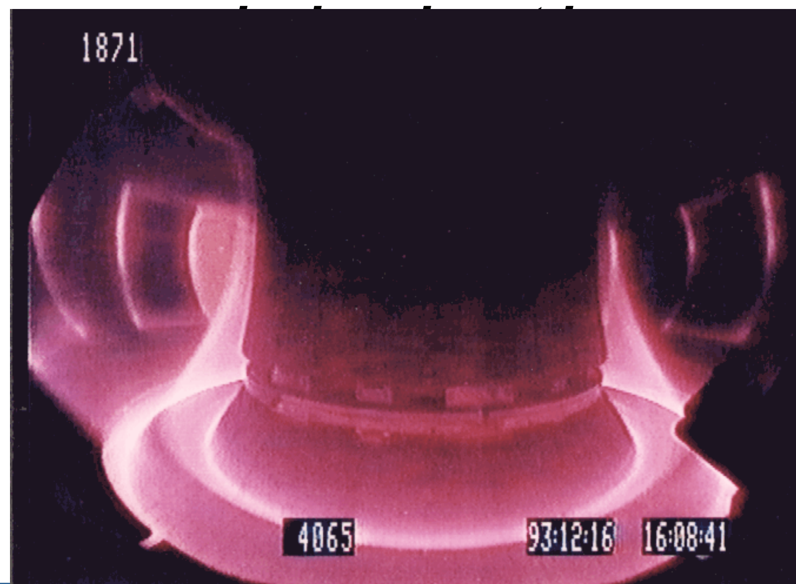
Power Exhaust – Walls – Materials for Tokamaks



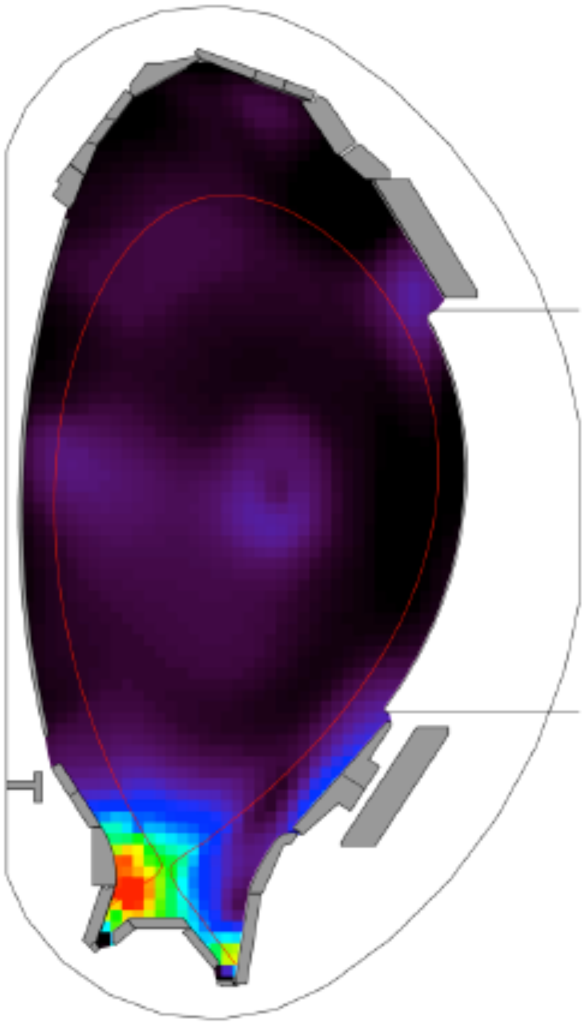
Marco Wischmeier

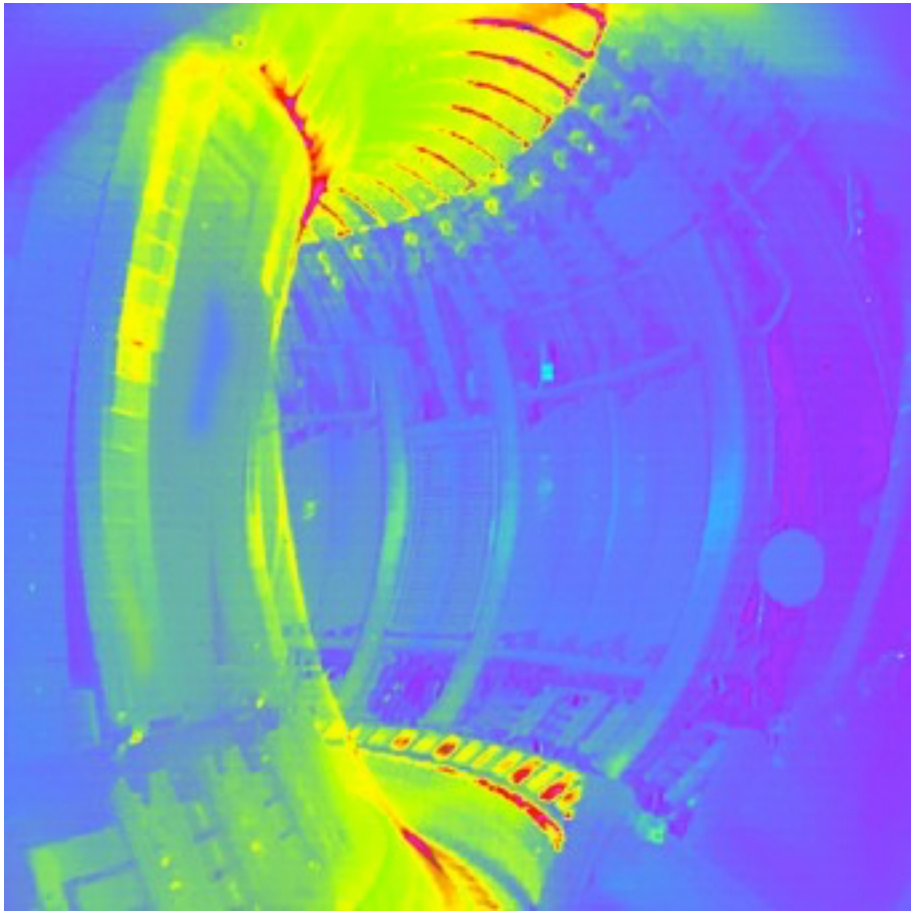
Max-Planck-Institut für Plasmaphysik

85748 Garching



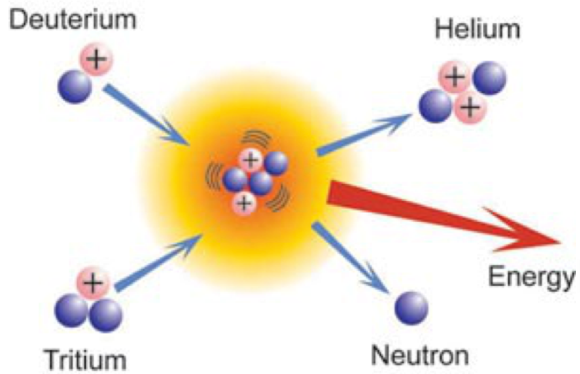
Joint ICTP-IAEA College on Advanced Plasma Physics, Triest, Italy, 2016





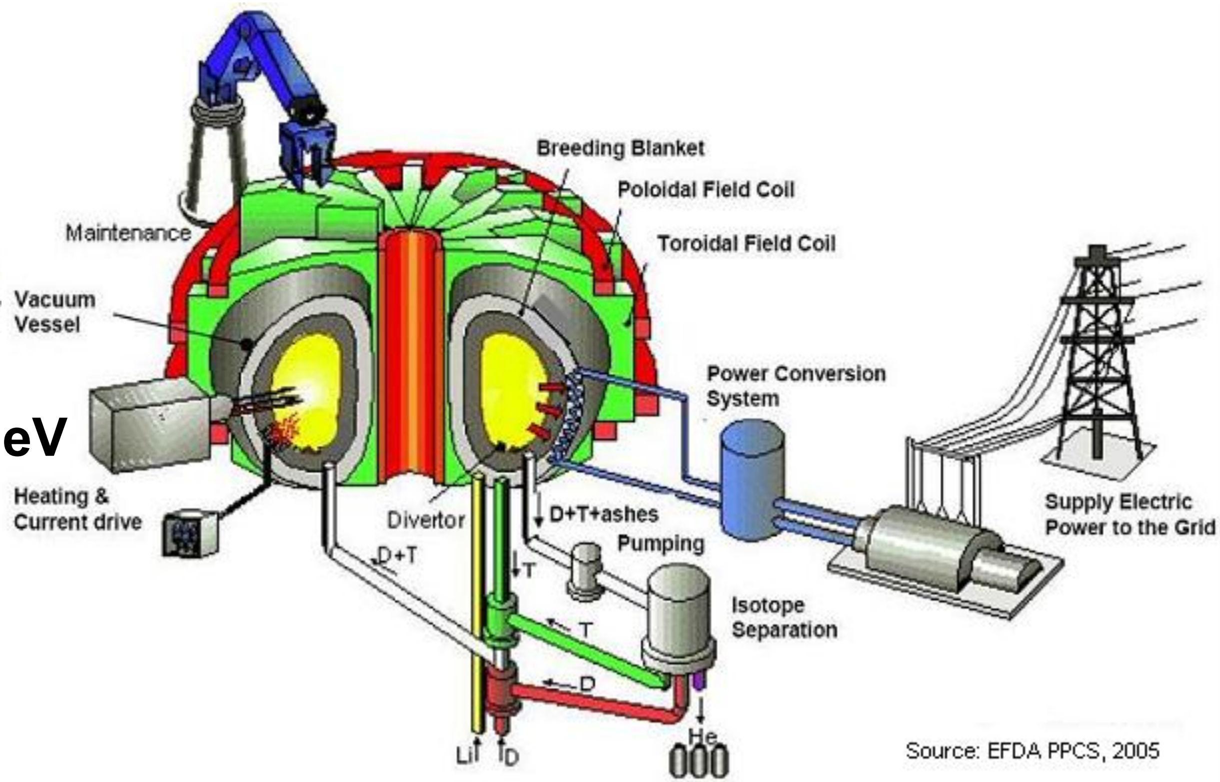
Goal of magnetic confinement research

Neutrons leave plasma into power conversion system ➔ will be used for net energy production



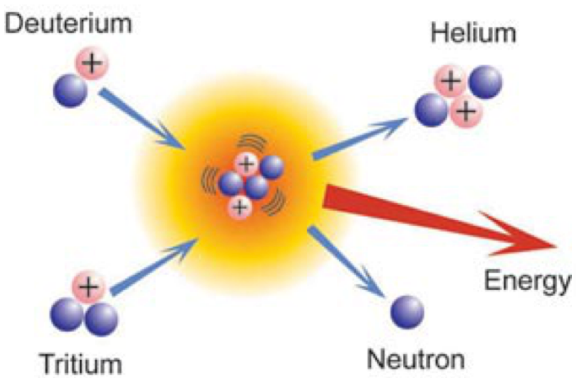
$$D + T \rightarrow He + n + 17.6\ \mathrm{MeV}$$

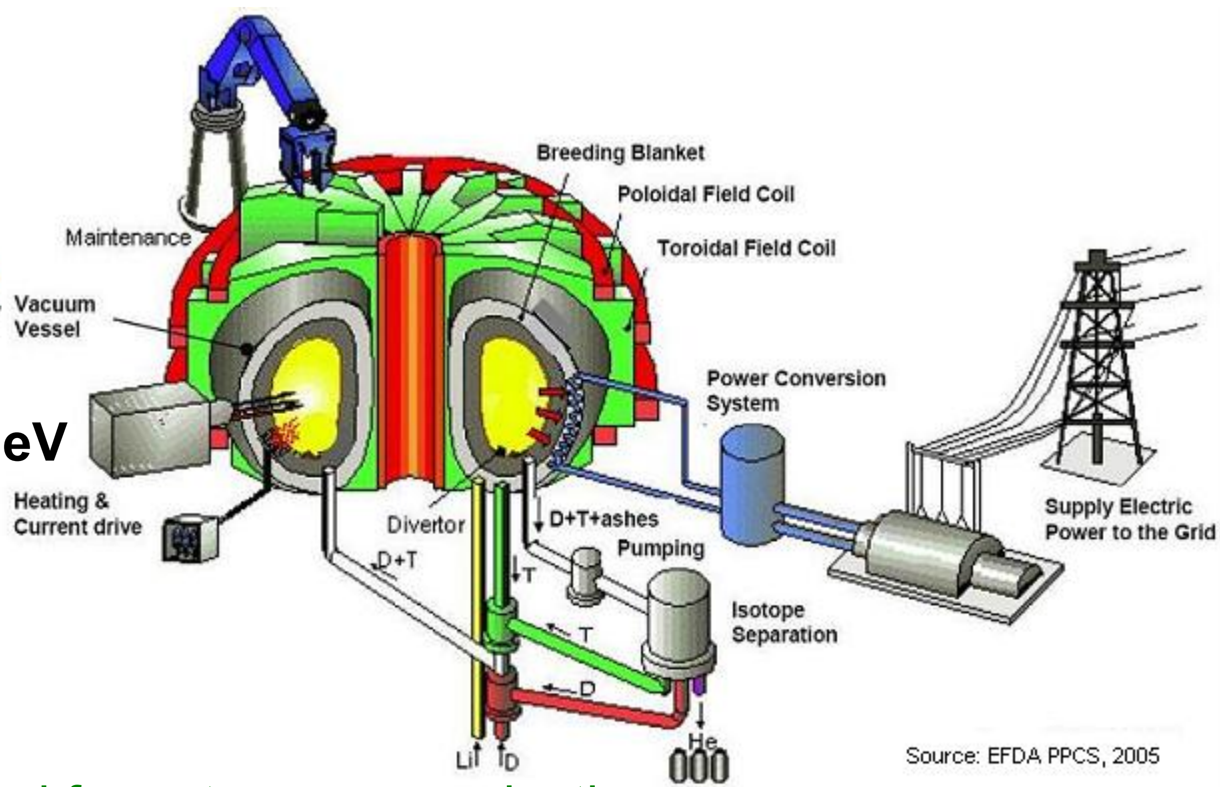
14.1MeV



Goal of magnetic confinement research

He heats plasma ➔ needs to be exhausted





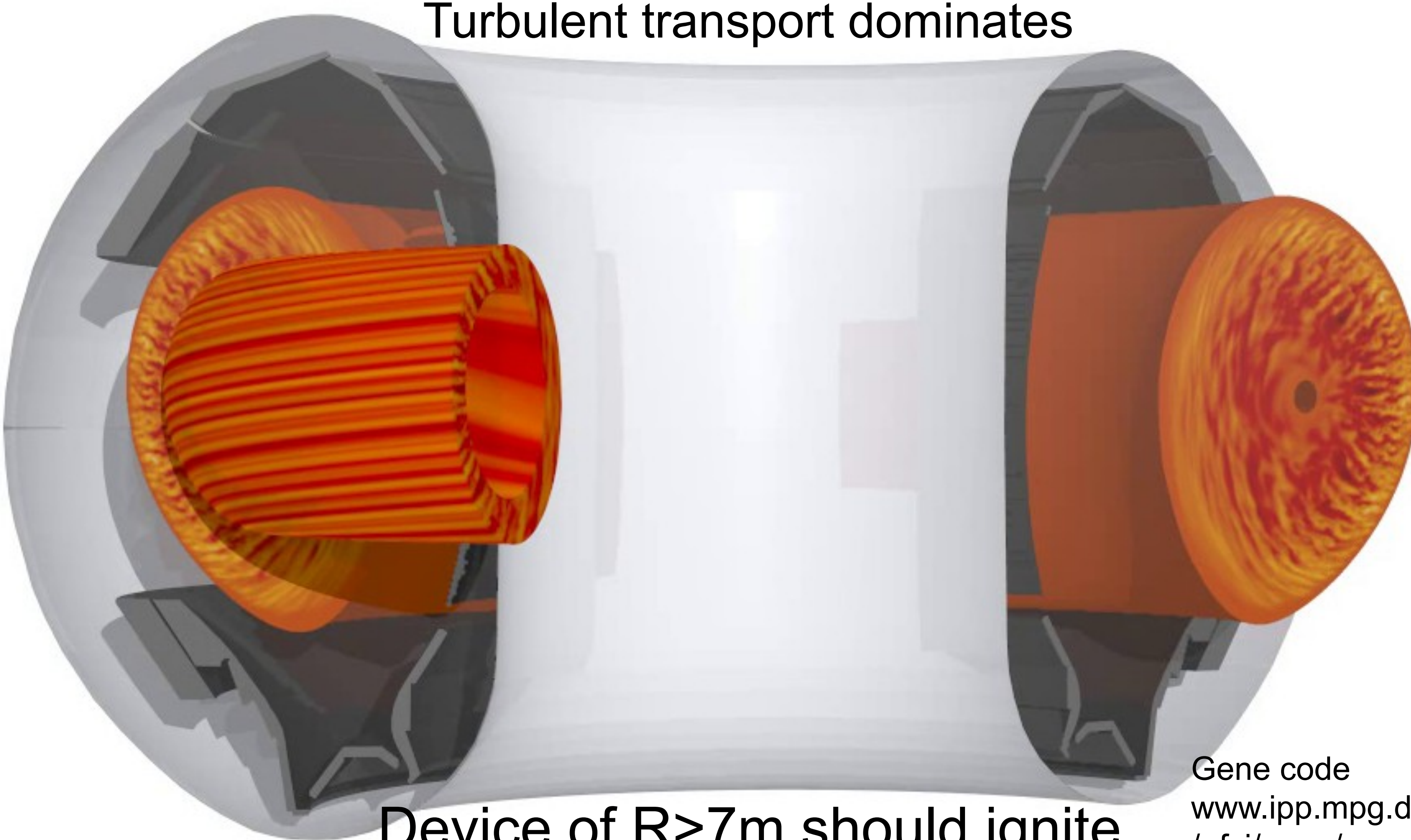
$$\mathrm{D + T \rightarrow He + n + 17.6\ MeV}$$

3.5MeV   14.1MeV

Used for net energy production

Losses perpendicular to magnetic field

Turbulent transport dominates



Device of R>7m should ignite

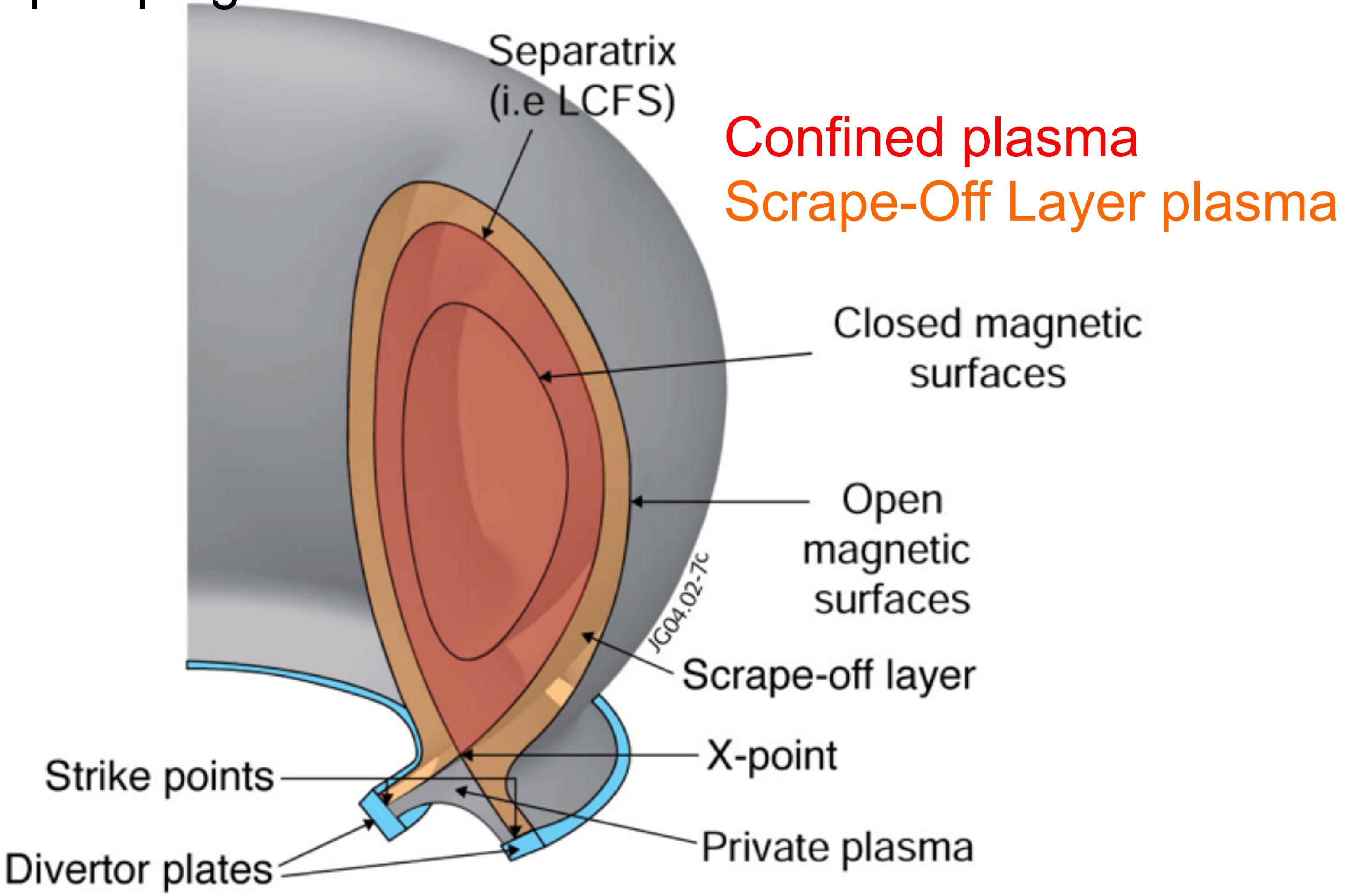
Gene code
www.ipp.mpg.de
/~fsj/gene/

Fusion exhaust must...

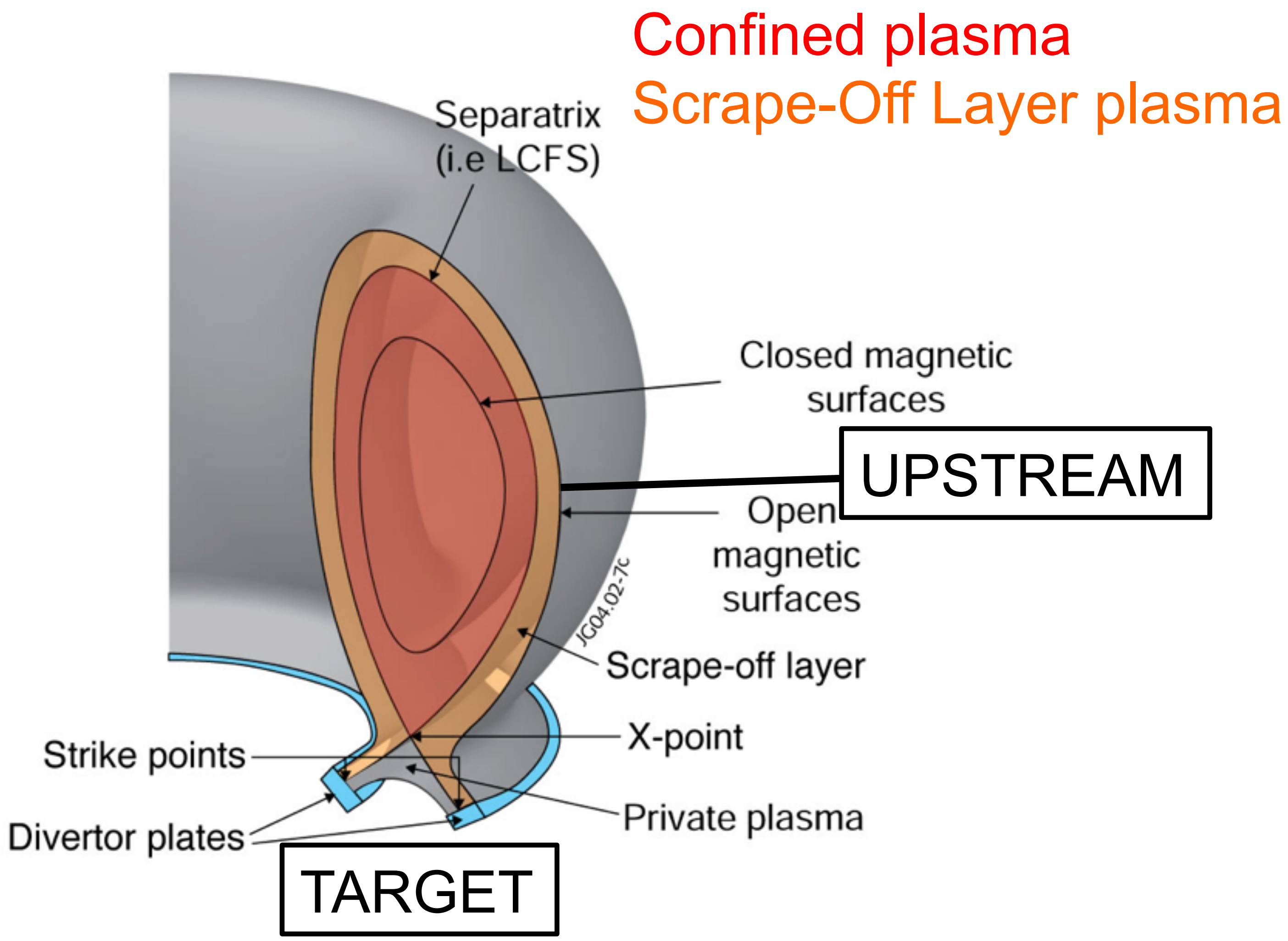
- ❖ Maximize pumping of He ash
- ❖ Provide sufficient pumping of hydrogen fuel
- ❖ Minimize damages to the wall (erosion, melting)

Divertor concept

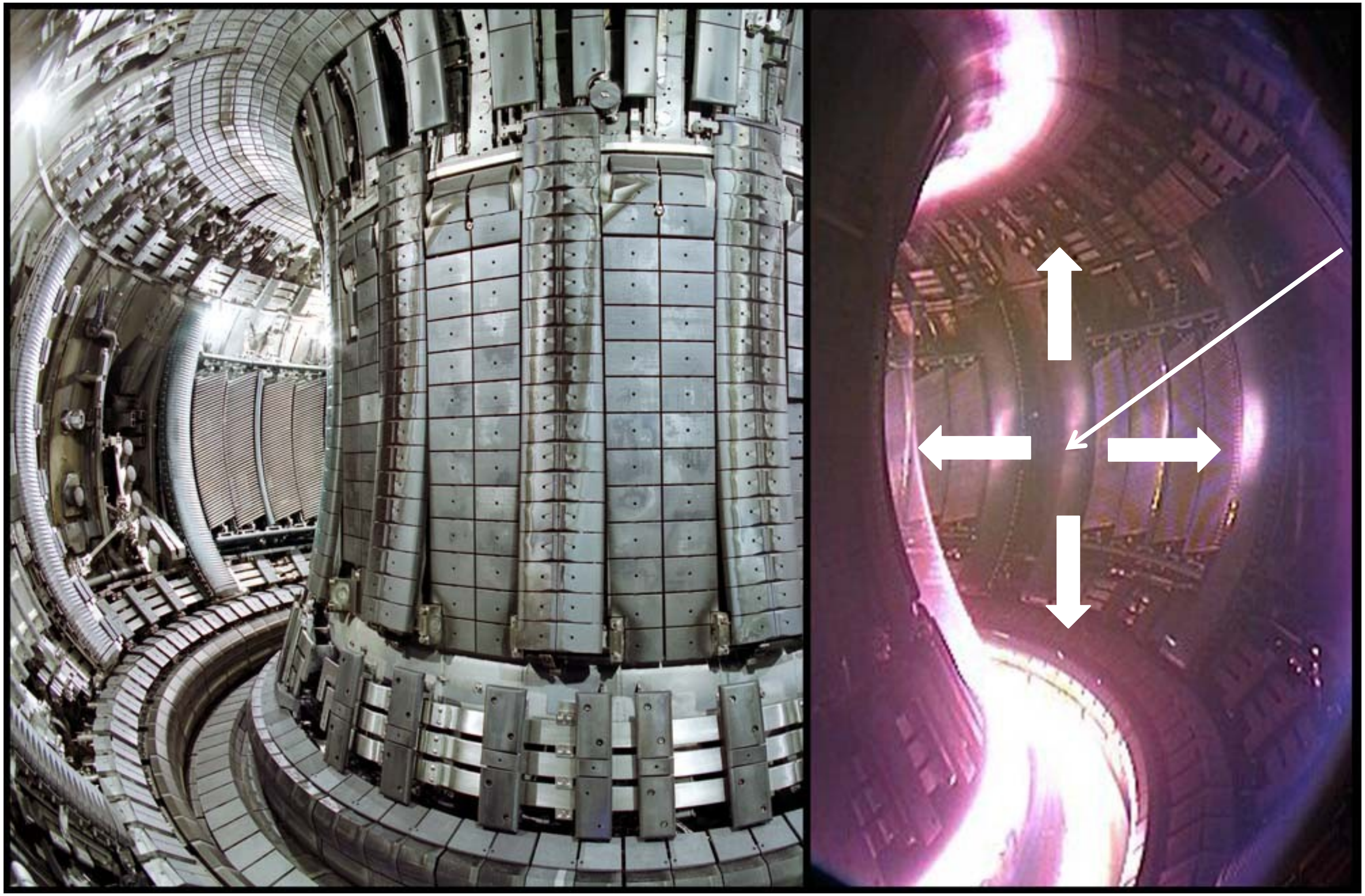
Maximize pumping of He ash and minimize erosion



Divertor concept

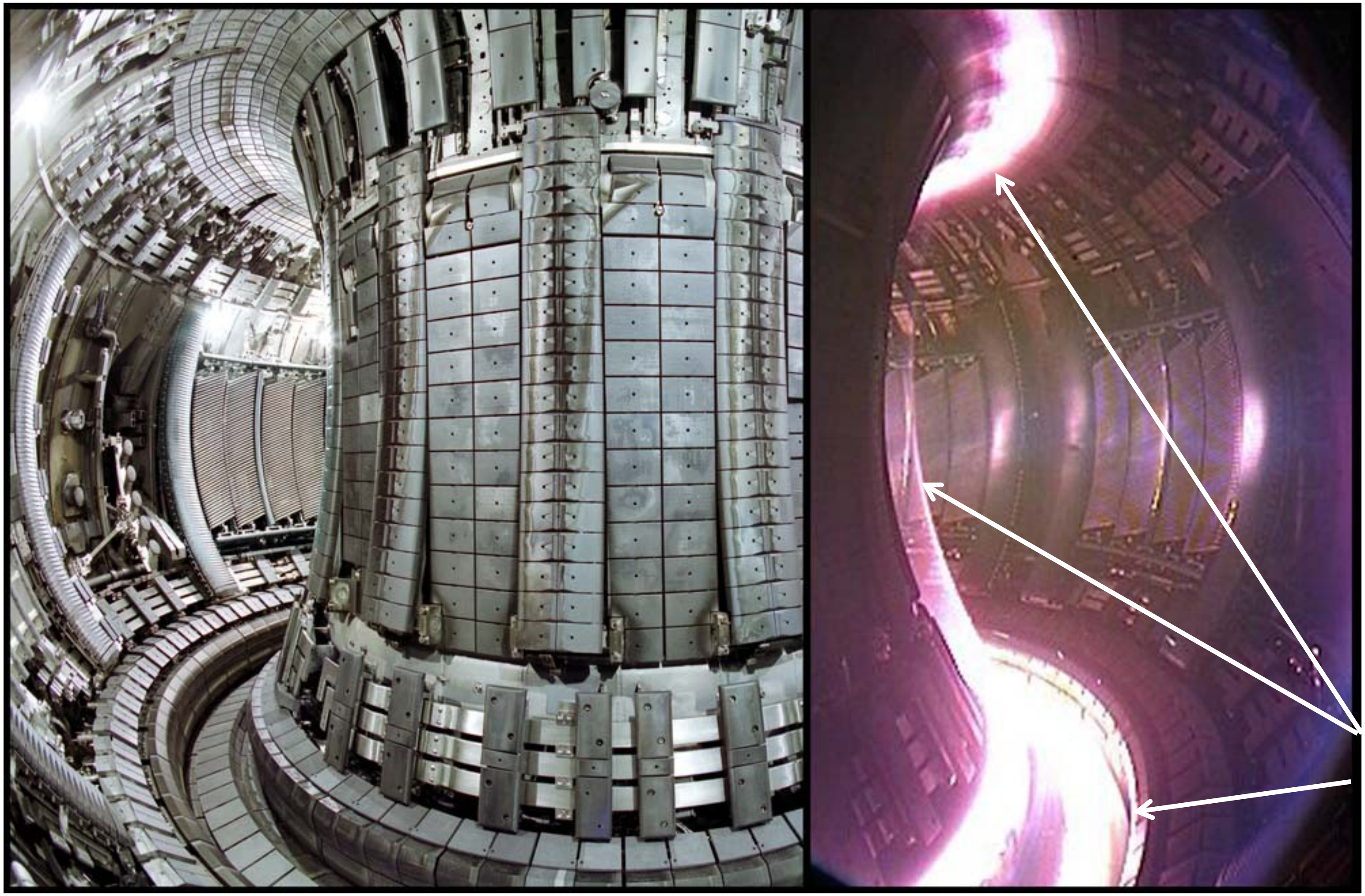


Divertor & Plasma



From JET

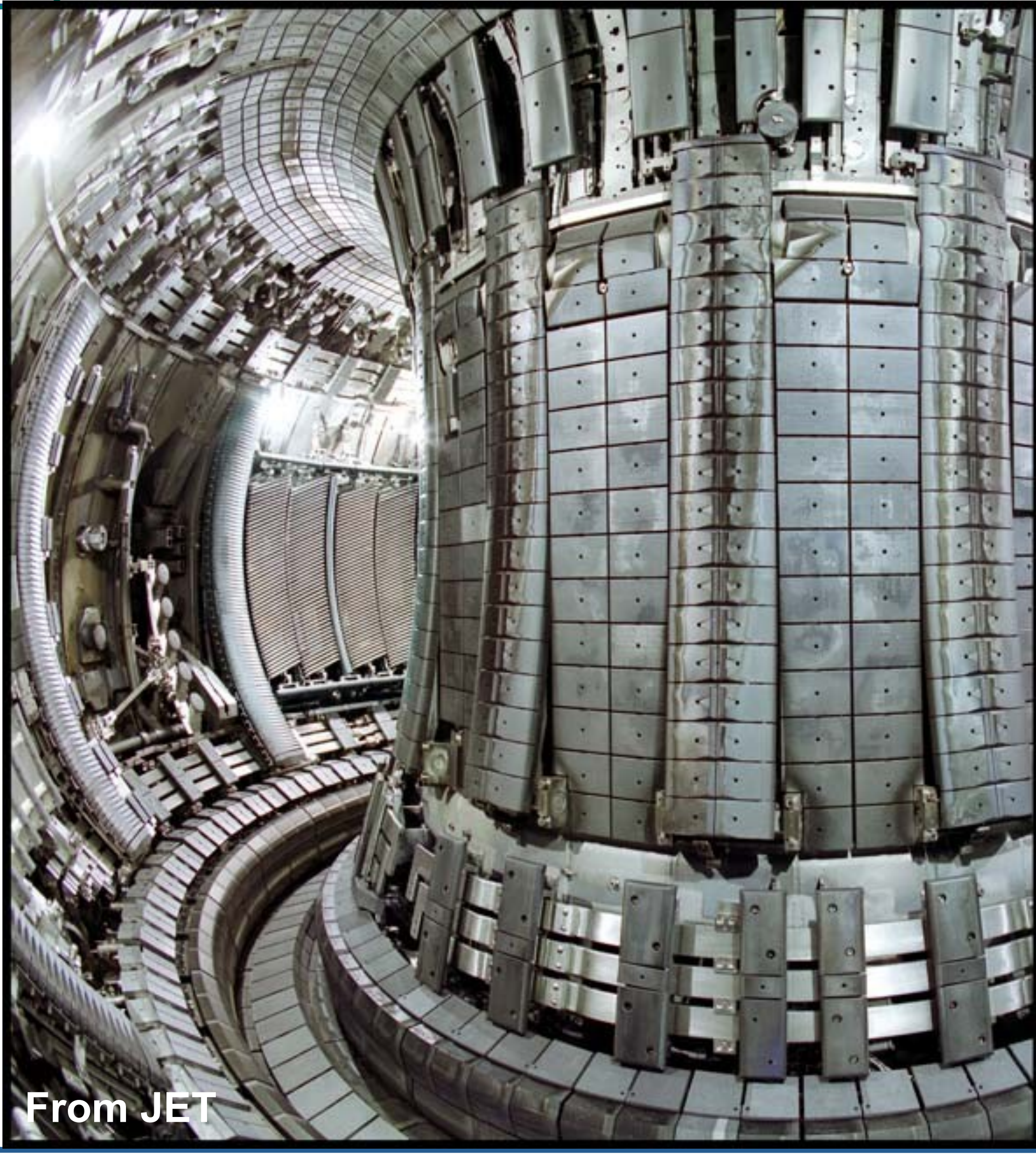
Divertor & Plasma

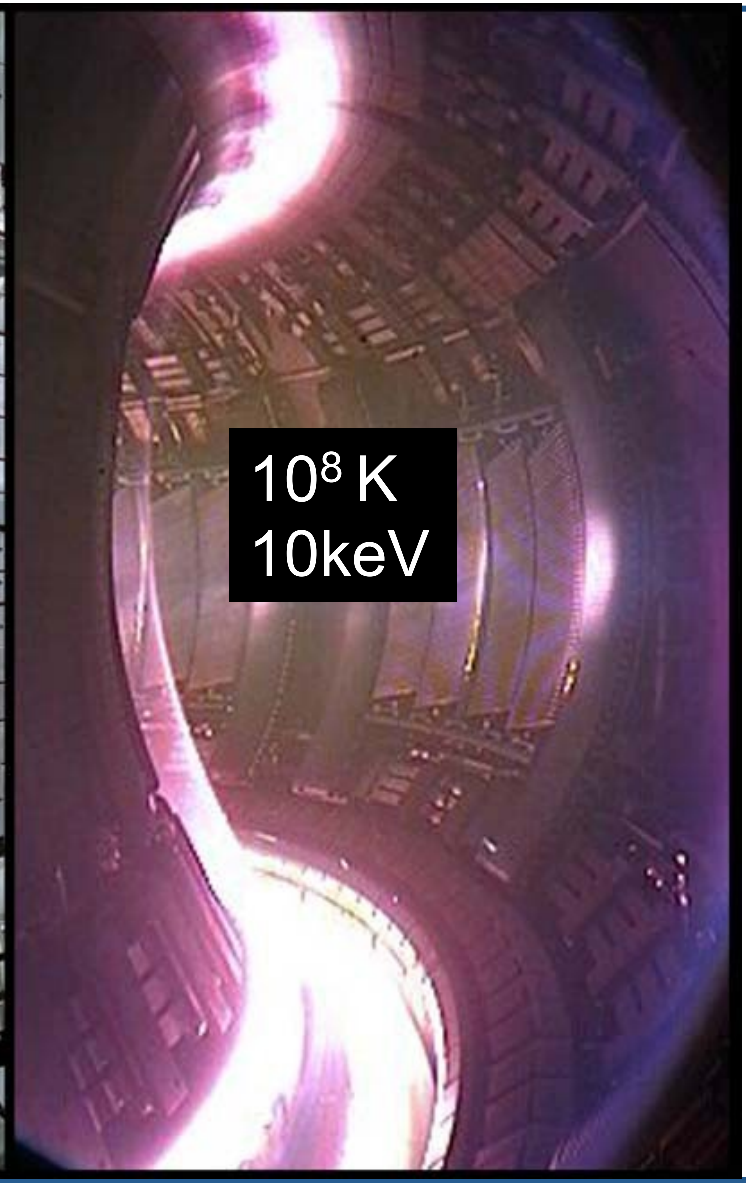


Distributed
Recycling
particles

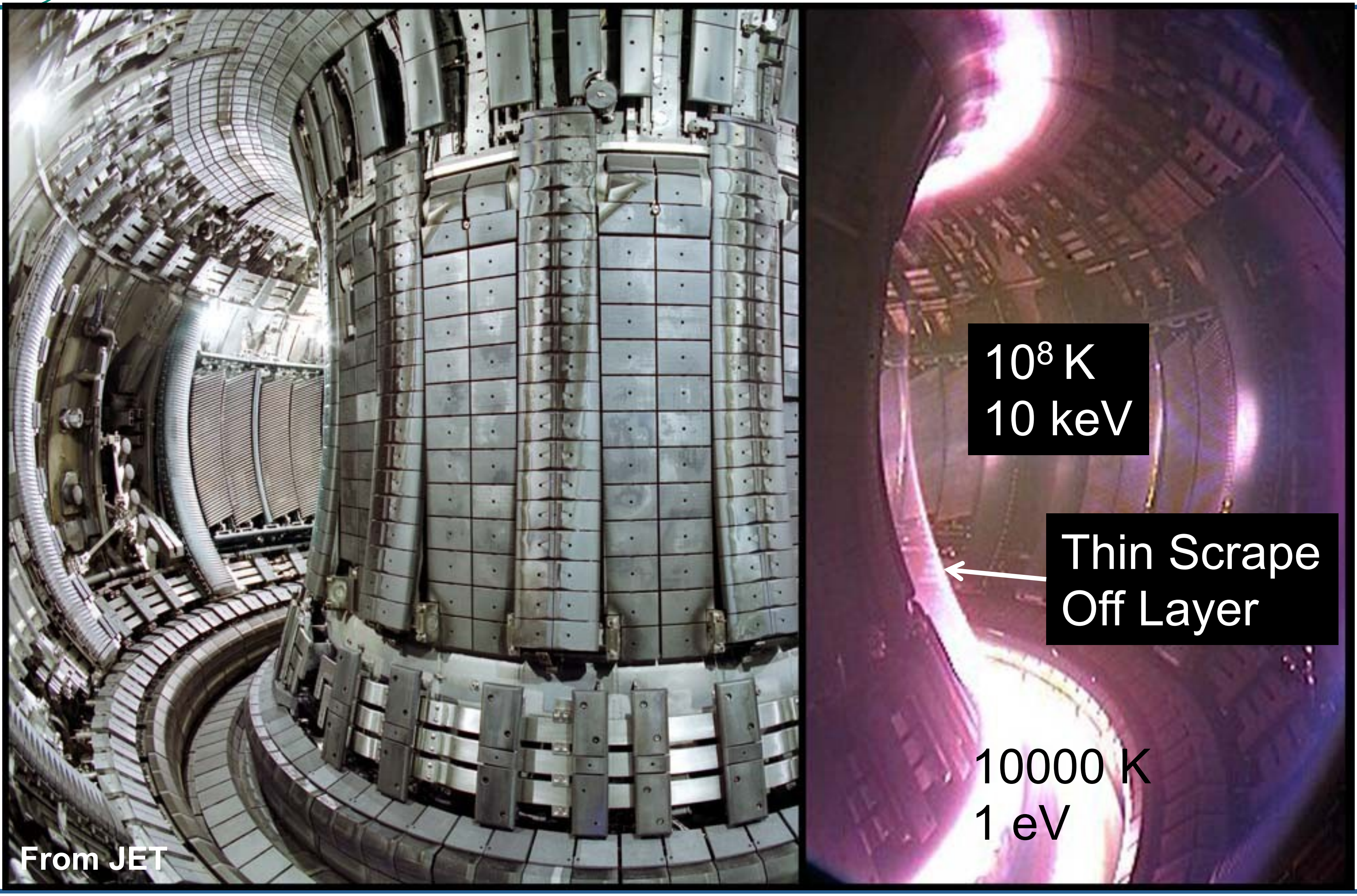
From JET

Divertor & Plasma





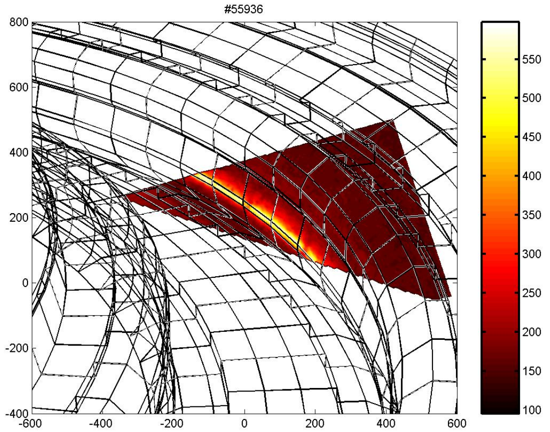
Divertor & Plasma

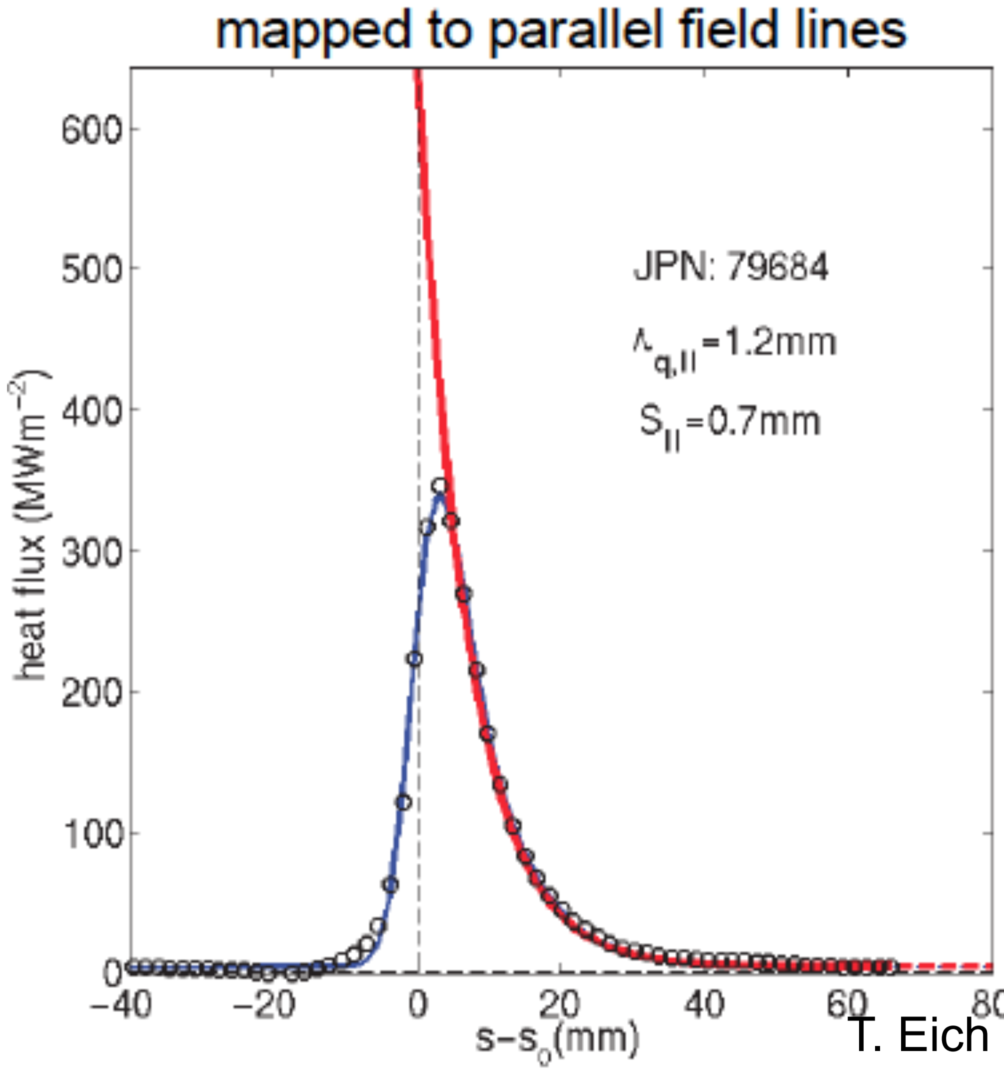


Width of Scrape-Off Layer?
What is the power flux?

Measuring power deposition profile

Infrared image of target

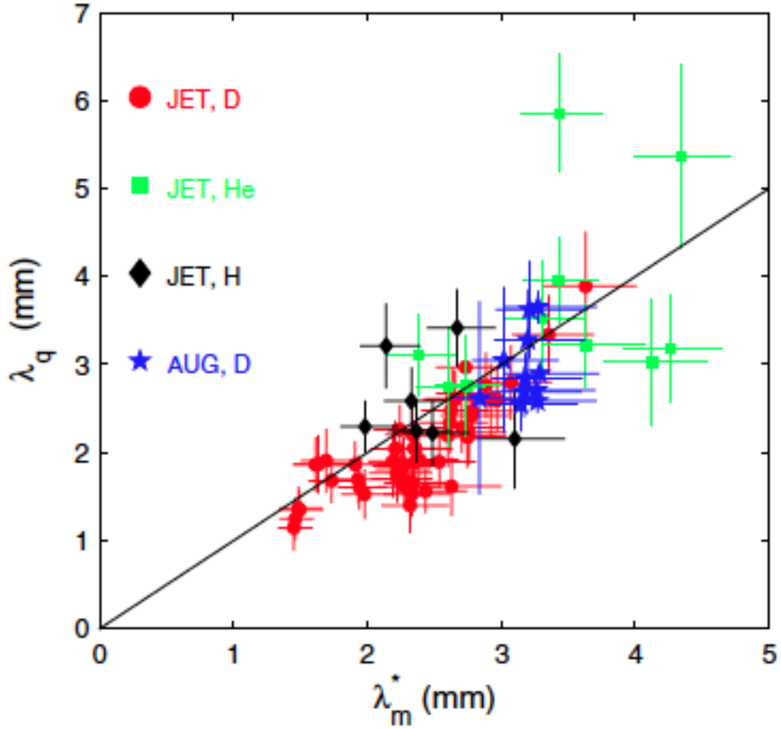




T. Eich PSI 2012

The power decay length λ_q

H-mode (reduced turbulent transport)



T. Eich PRL (2011), T. Eich IAEA FEC 2012, A. Scarabosio PSI 2012

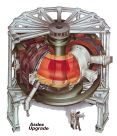
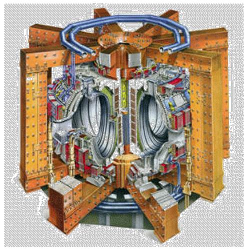
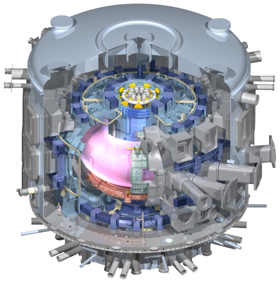
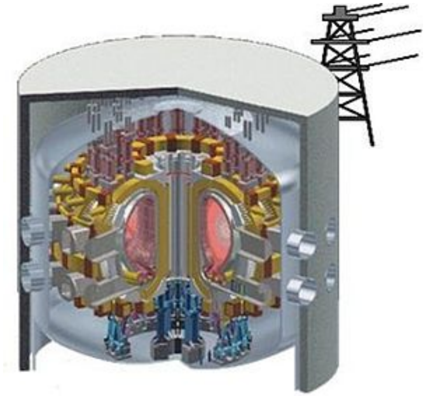
$$\lambda_q = 0.73 \cdot B_{tor}^{-0.78} \cdot q_{cyl}^{1.20} \cdot P_{SOL}^{0.10} \cdot R_{geo}^{0.02}$$

(Carbon divertor, attached conditions, inter ELM)

No dependence on machine size R

What is the power flux density in the SOL?

Importance of tokamak size R

	ASDEX Upgrade (IPP)	*JET (EU)*	*ITER*	*DEMO*
				
Major Radius	*1.65 m*	*3 m*	*6.2 m*	*>7 m*
P_{heat}	*23 MW*	*~ 38 MW*	*~ 100 MW*	*~ 600 MW*

Good energy confinement ➔ large R
(P_{fus} ~R^3 – see yesterday)

P/R as figure of merit

A measure of the severity of the heat flux is

- **P_{heat}/R**

M. Kotschenreuter et al. NF 50 2010
K. Lackner Comm. PPCFusion 15 1994

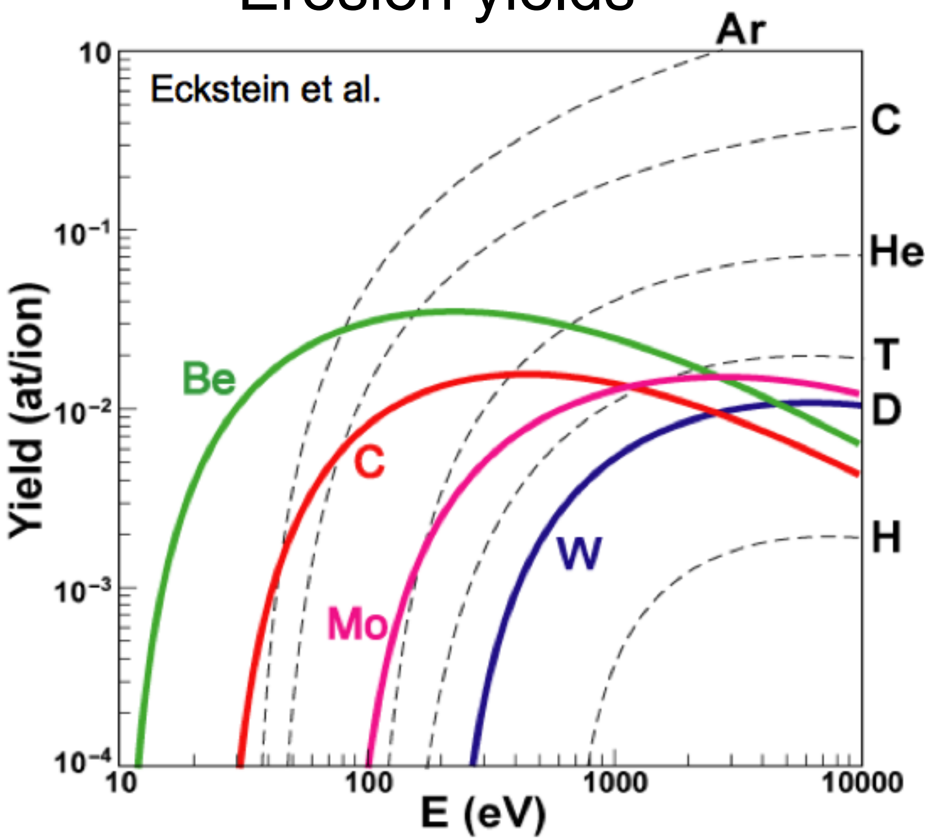
Device	P_{heat}/R	$q_{\|\|}$ upstream
JET	7	20 GW/m^2
ASDEX Upgrade	14	35 GW/m^2
ITER	20	50 GW/m^2
DEMO	80-100	>300 GW/m^2

What are the limitations imposed by wall materials?

Erosion limits maximum Temperature

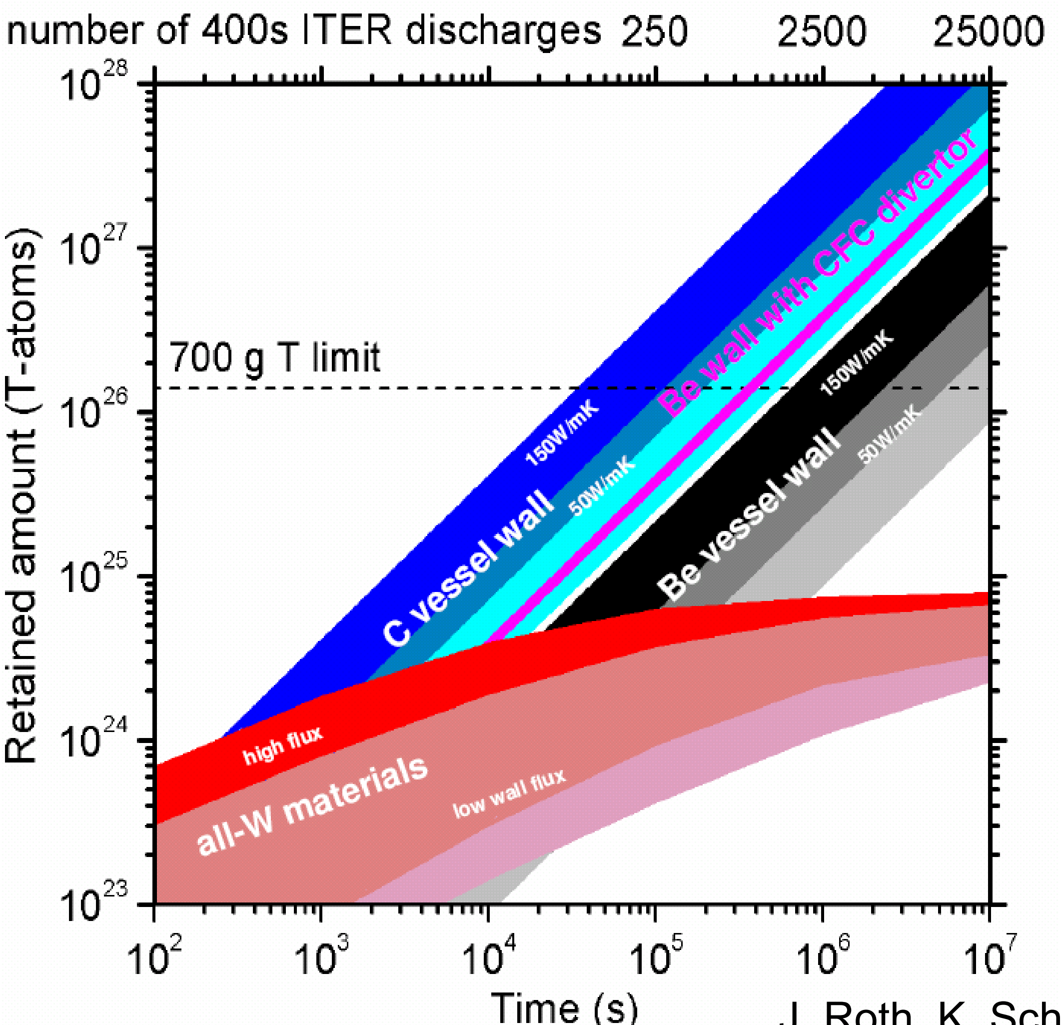
Ions accelerated to energies
~ Z x 3.5 x T_e in electrical field by
sheath potential

Erosion yields



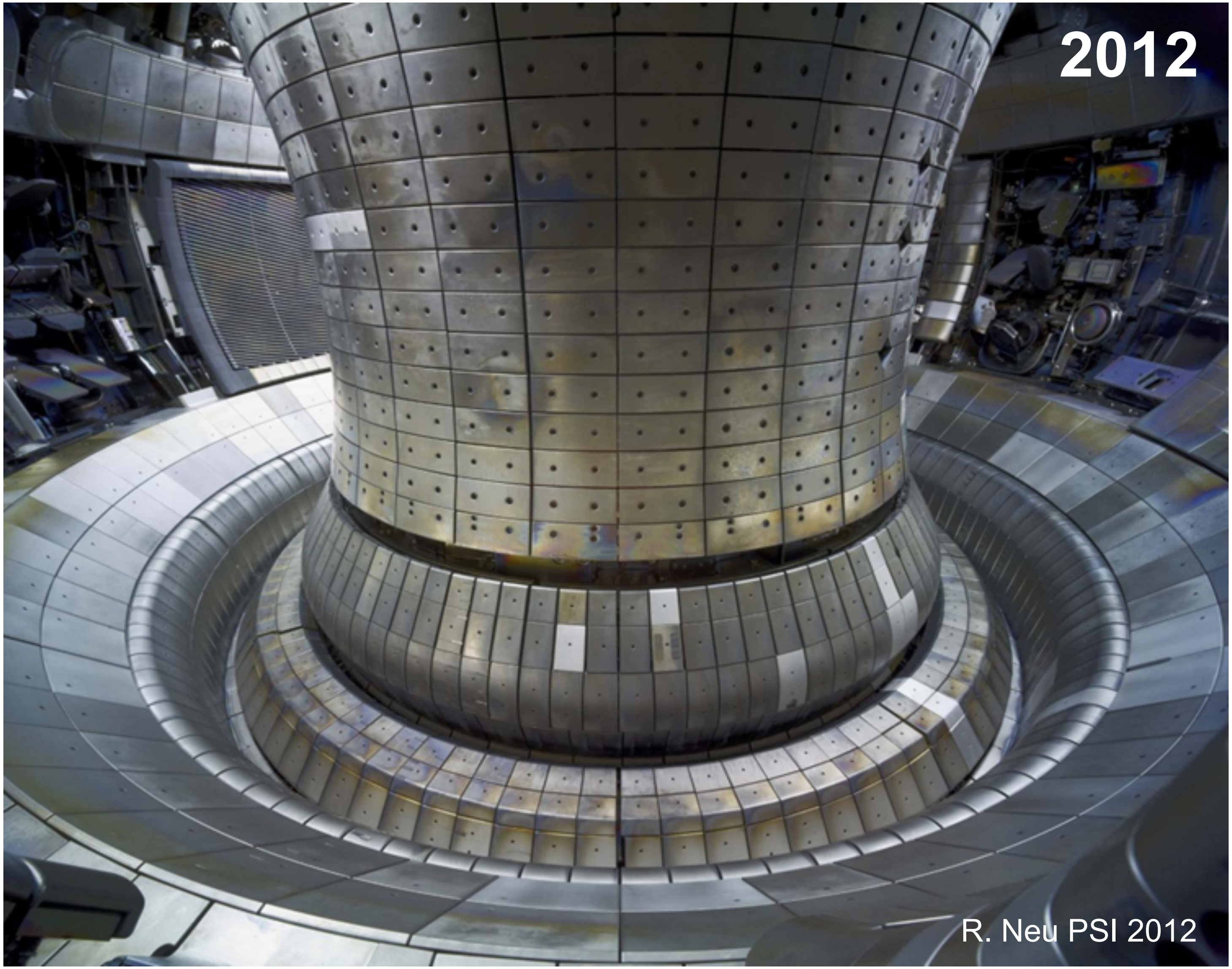
W has low Yield

Tritium retention



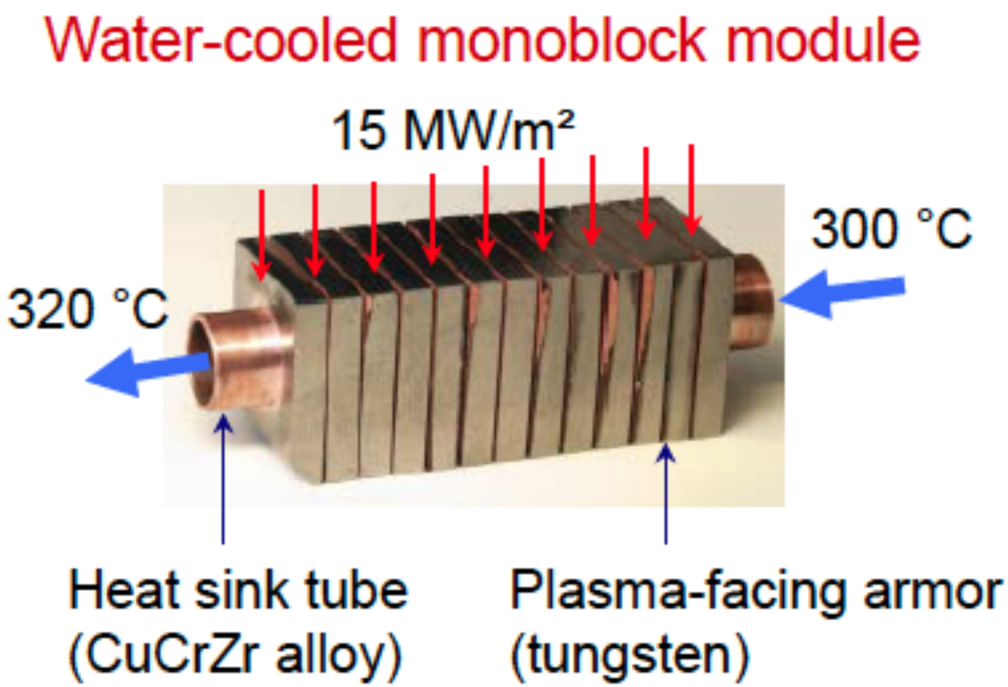
J. Roth, K. Schmid, Phys Scripta 2011

All tungsten plasma facing components in ASDEX Upgrade



Technological limits under neutron irradiation

Integrated approach:
Combination of coolant, structural material of coolant pipe and armour material?

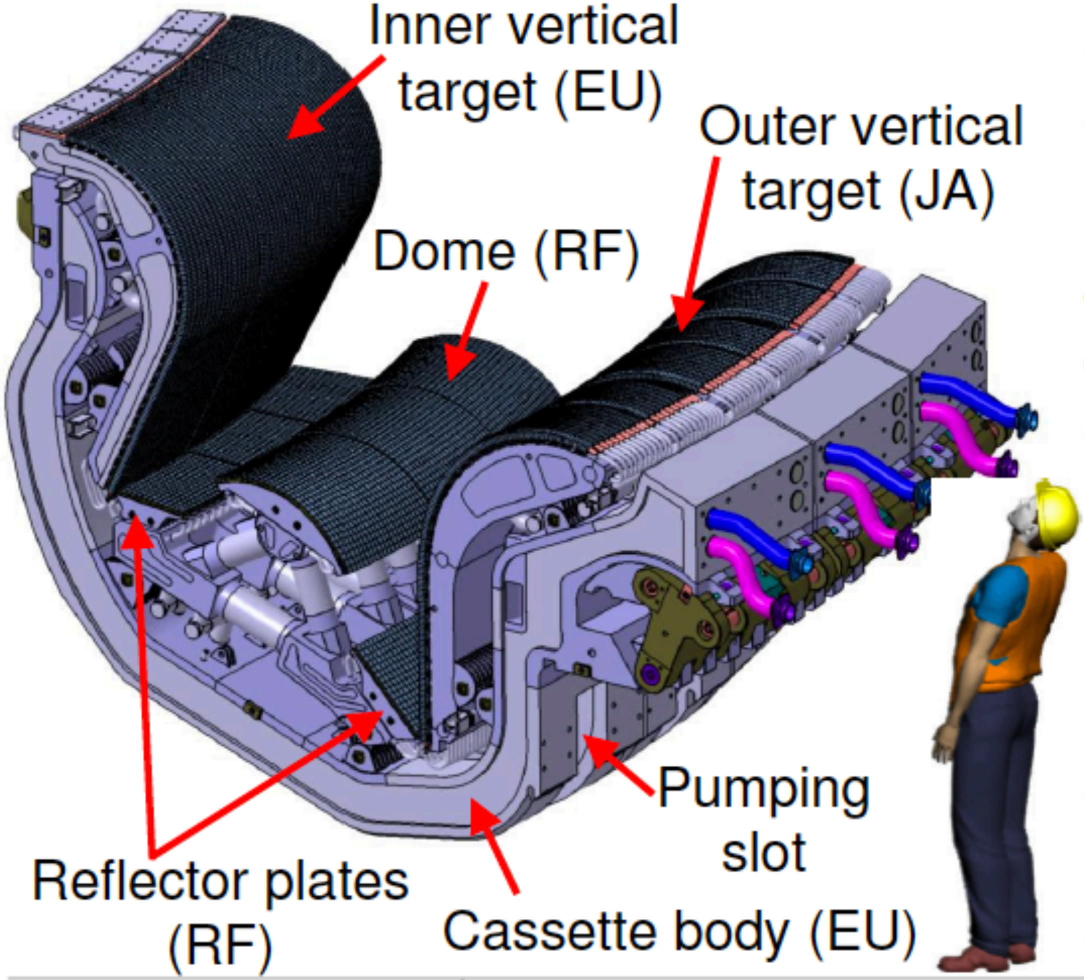


E.U. protoype monoblock

Water cooled divertor segment

10MW/m² to 5MW/m² is the technological limit

Reminder: main divertor characteristics



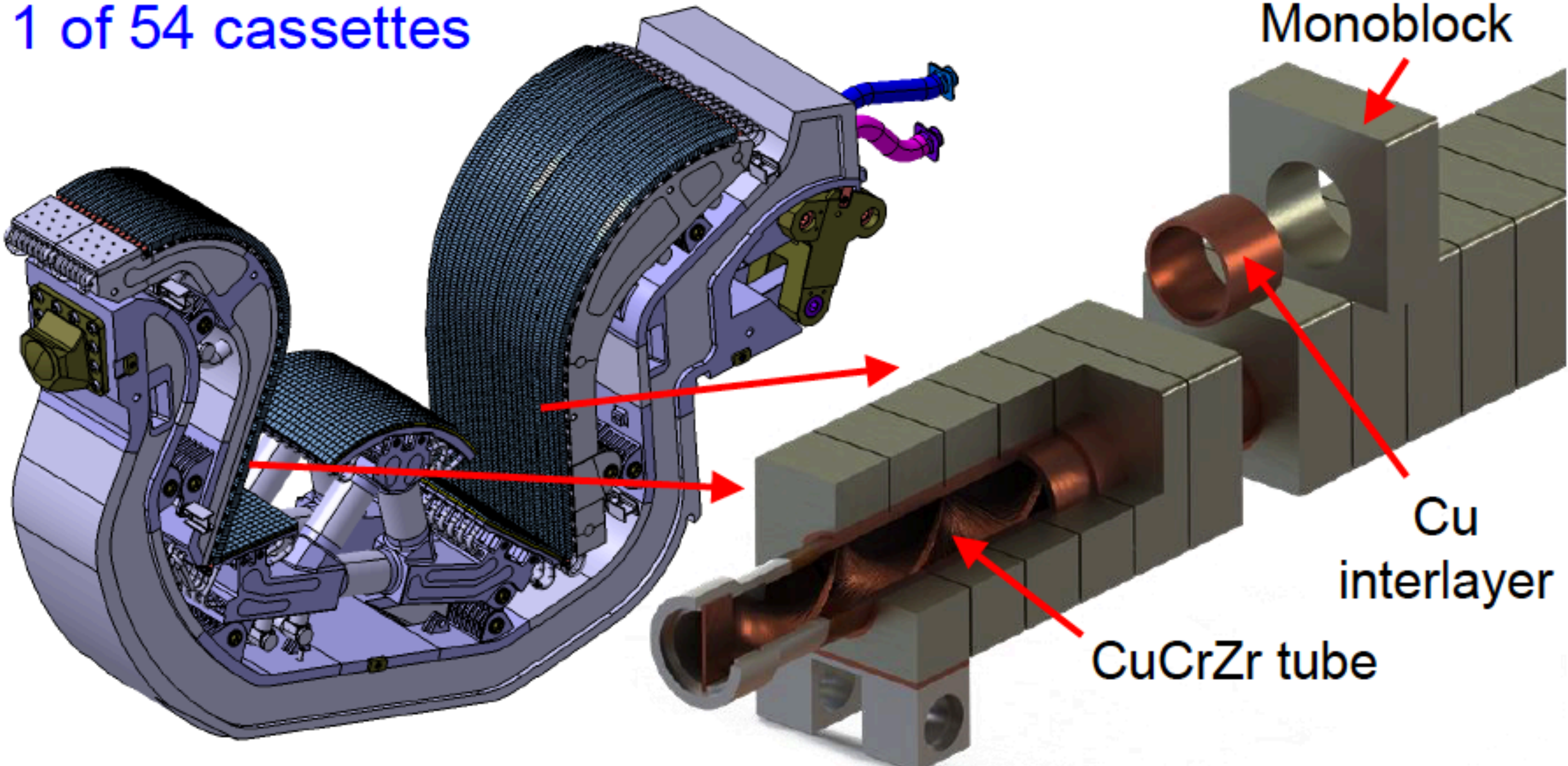
54 divertor assemblies (~9 tonnes each)

4320 actively cooled heat flux elements

Bakeable to 350°C

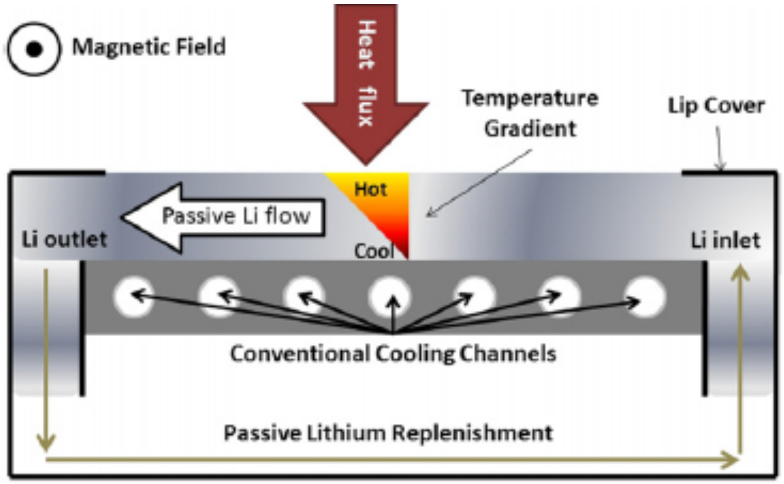
NB: technology limit for steady state power handling on HHF areas: 10 MWm^{-2}

The ITER W divertor

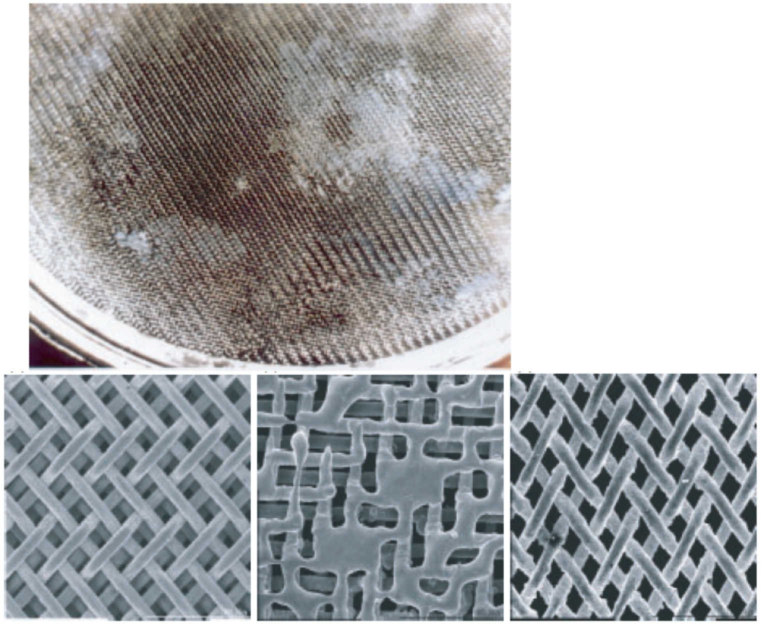


Key outstanding design issue is what shaping to apply to front surface of monoblocks in HHF areas → this talk

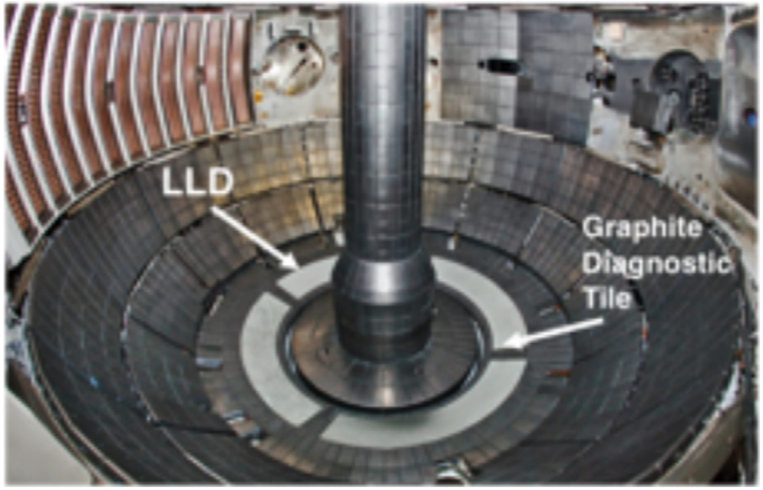
Risk mitigation: Liquid metals as PFC



D. N. Ruzic et al. NF 2011

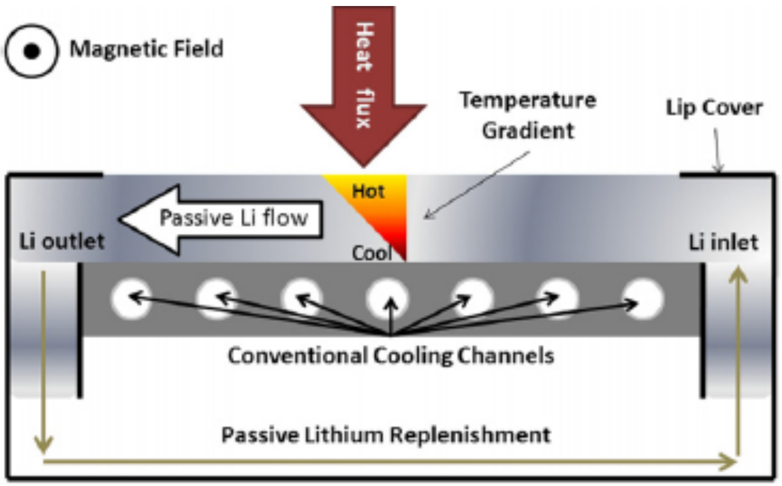


V. A. Evtikhin et al. PPCF 2002

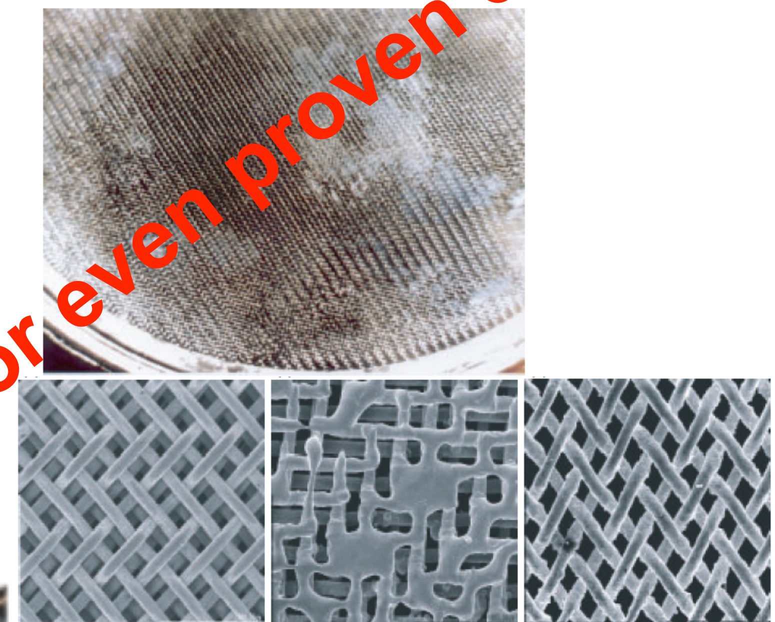


H. W. Kugel et al. Fusion Eng. and Des. 2012
A. G. McLean et al. JNM 2013

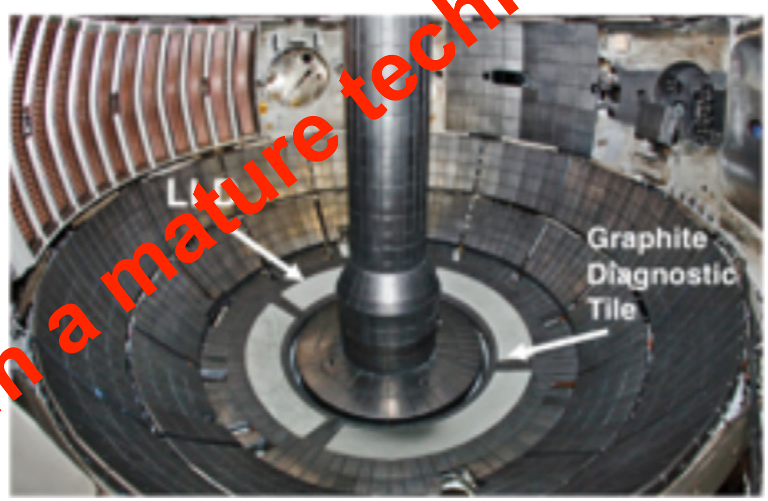
Risk mitigation: Liquid metals as PFC



D. N. Ruzic et al. NF 2011



V. A. Evtikhin et al. PPCF 2002



H. W. Kugel et al. Fusion Eng. and Des. 2012
A. G. McLean et al. JNM 2013

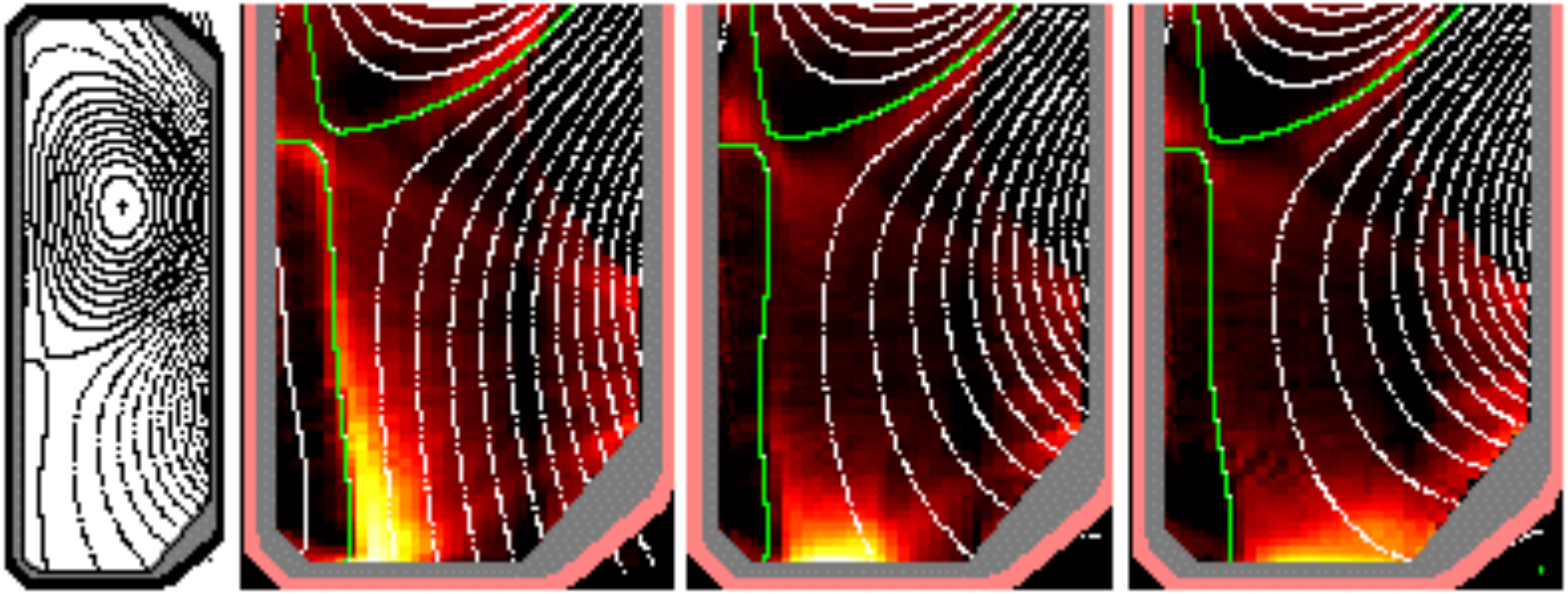
Far from a mature technology or even proven concept

How can we reduce the power load onto the divertor target plates to match the technological limit?

How can we reduce the power load onto the divertor target plates to match the technological limit?

1. GEOMETRY

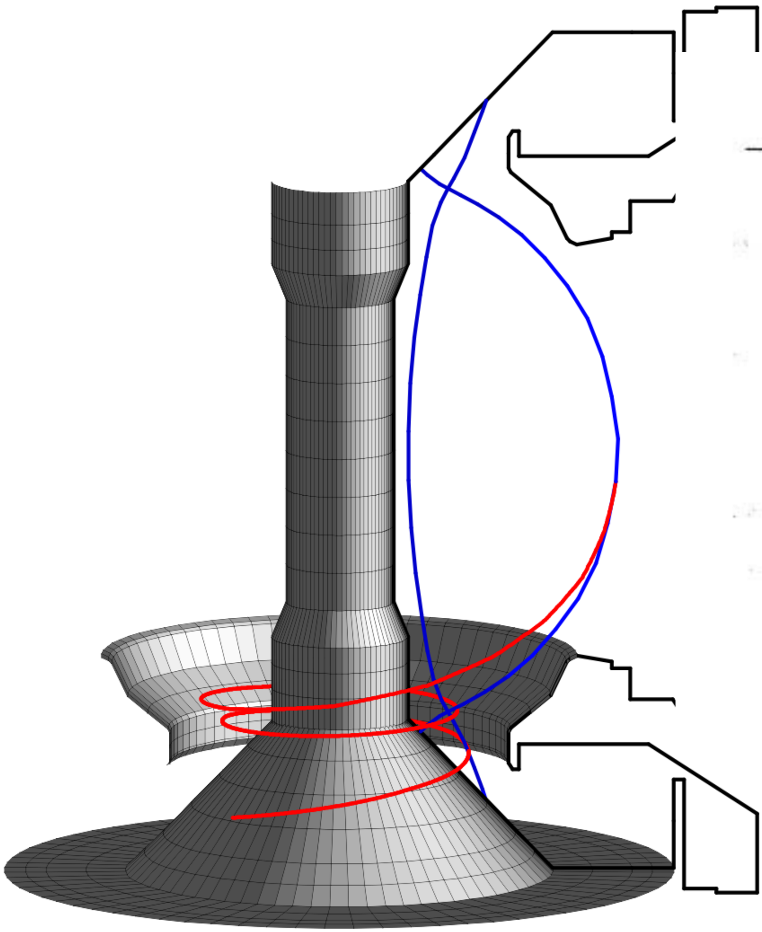
Magnetic Flux Expansion



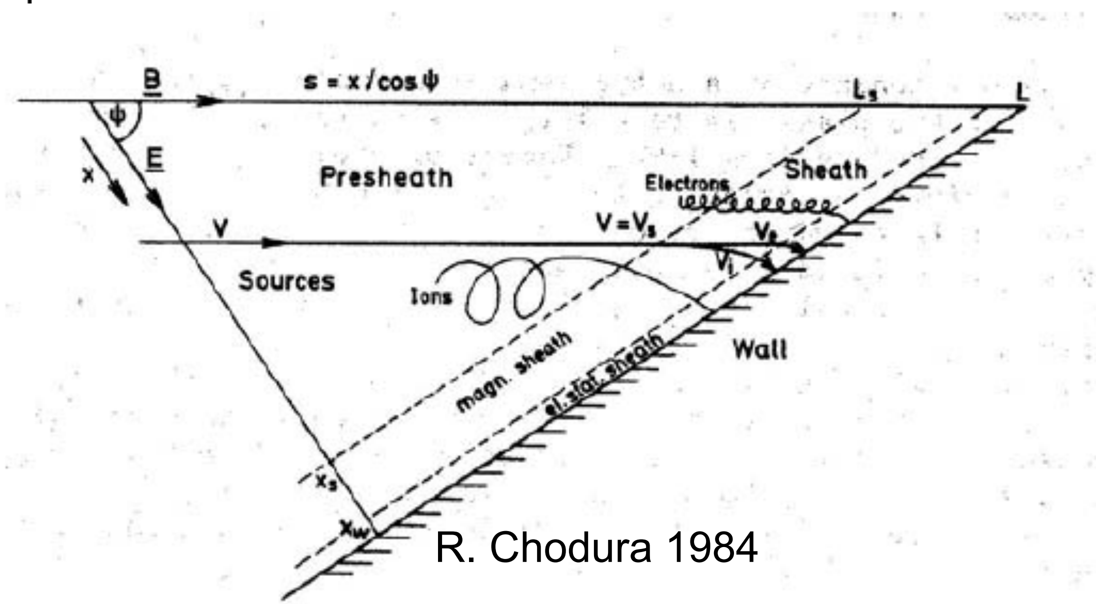
a) #17823 b) #17824, f_e = 2.8 c) #17823, f_e = 6.4 d) #17822, f_e = 9.3

R. Pitts et al., TCV, (CH), PSI 2000

Target inclination



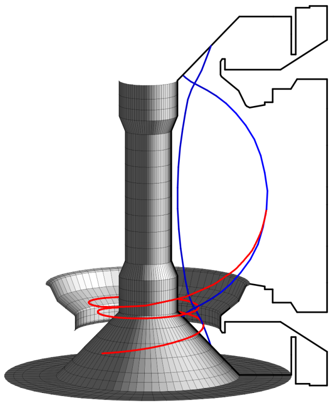
Courtesy H. Meyer, MAST

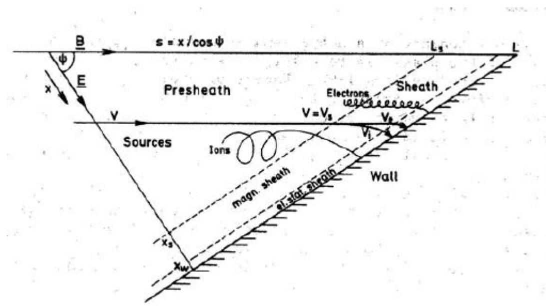


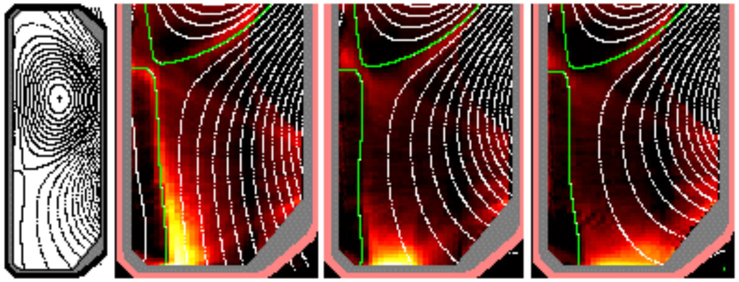
R. Chodura 1984

Impact angles of 1.5 – 3.5 degrees

Power load reduced by geometry





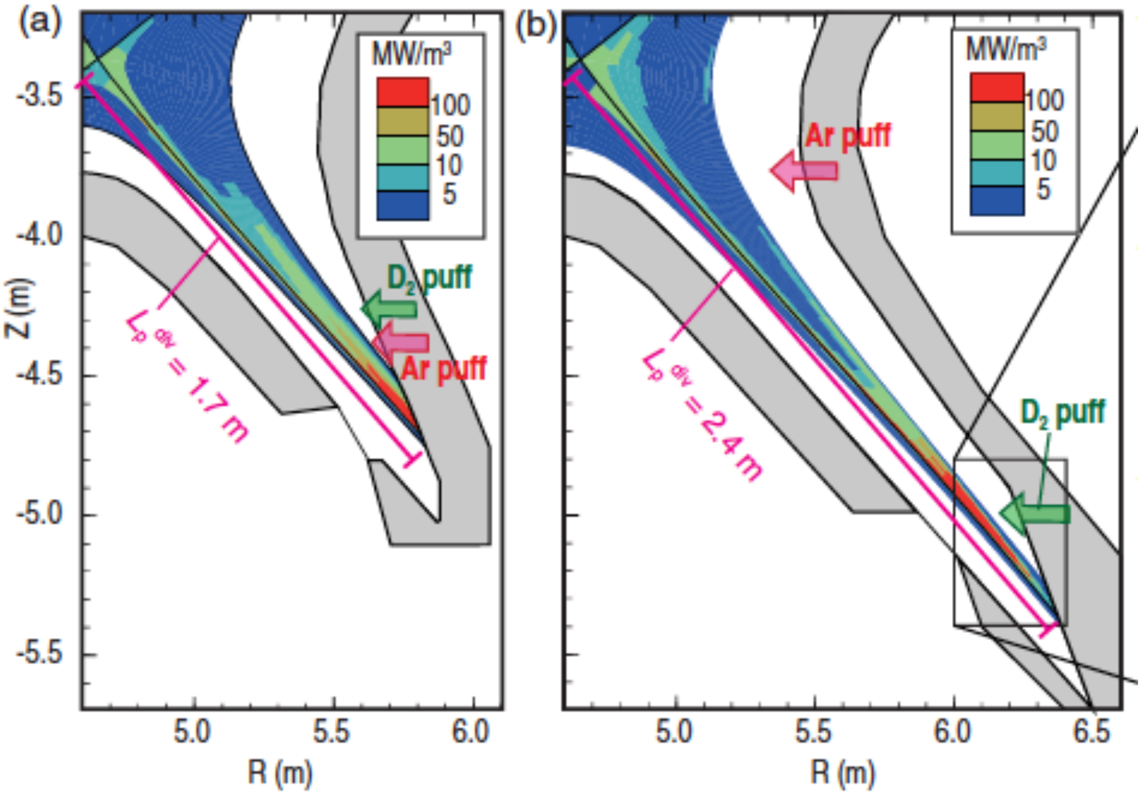


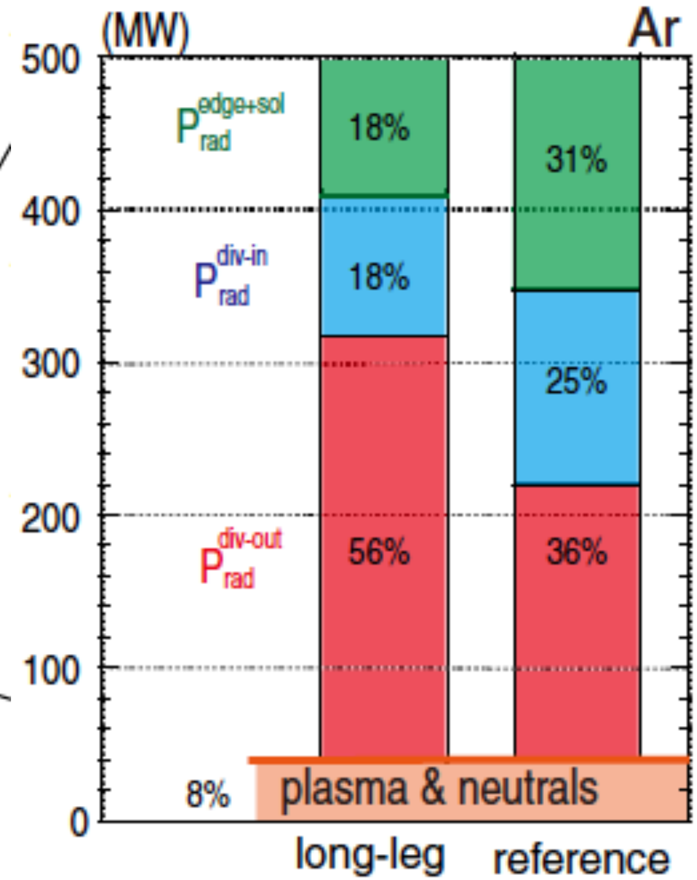
a) #17823 b) #17824, f_e = 2.8 c) #17823, f_e = 6.4 d) #17822, f_e = 9.3

Device	**P_{heat}/R**	**$q_{\|\|}$ upstream**	**q target (geometry)**
JET	7	20 GW/m^2	20 MW/m^2
ASDEX Upgrade	14	35 GW/m^2	35 MW/m^2
ITER	20	50 GW/m^2	50 MW/m^2
DEMO	80-100	>300 GW/m^2	300 MW/m^2

Increase of divertor major R

Numerical simulations of DEMO device: SONIC



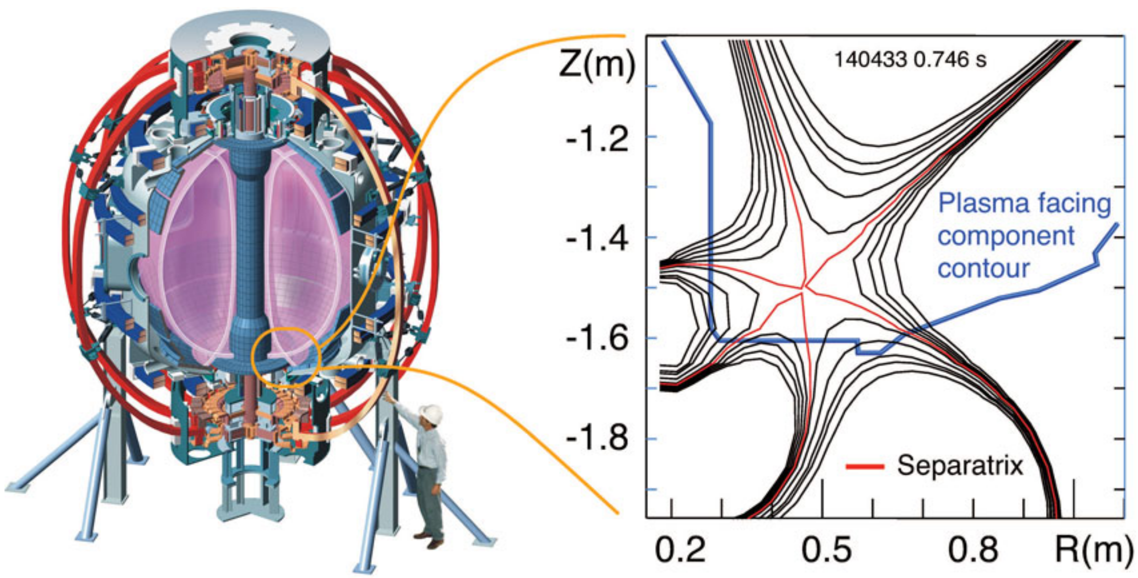


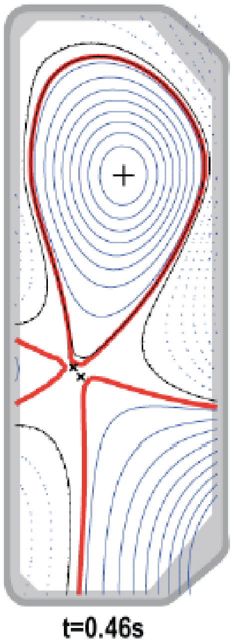
N. Asakura et al. NF 2014

- With long leg – Target at larger R:
- Lower $q_{||}$ and larger A_{wet} (here compensated by lower f)
- Lower core radiation required
- Lower core dilution

N. Asakura et al. P1-103 at PSI2014

Risk mitigation: Advanced divertor configurations (I)

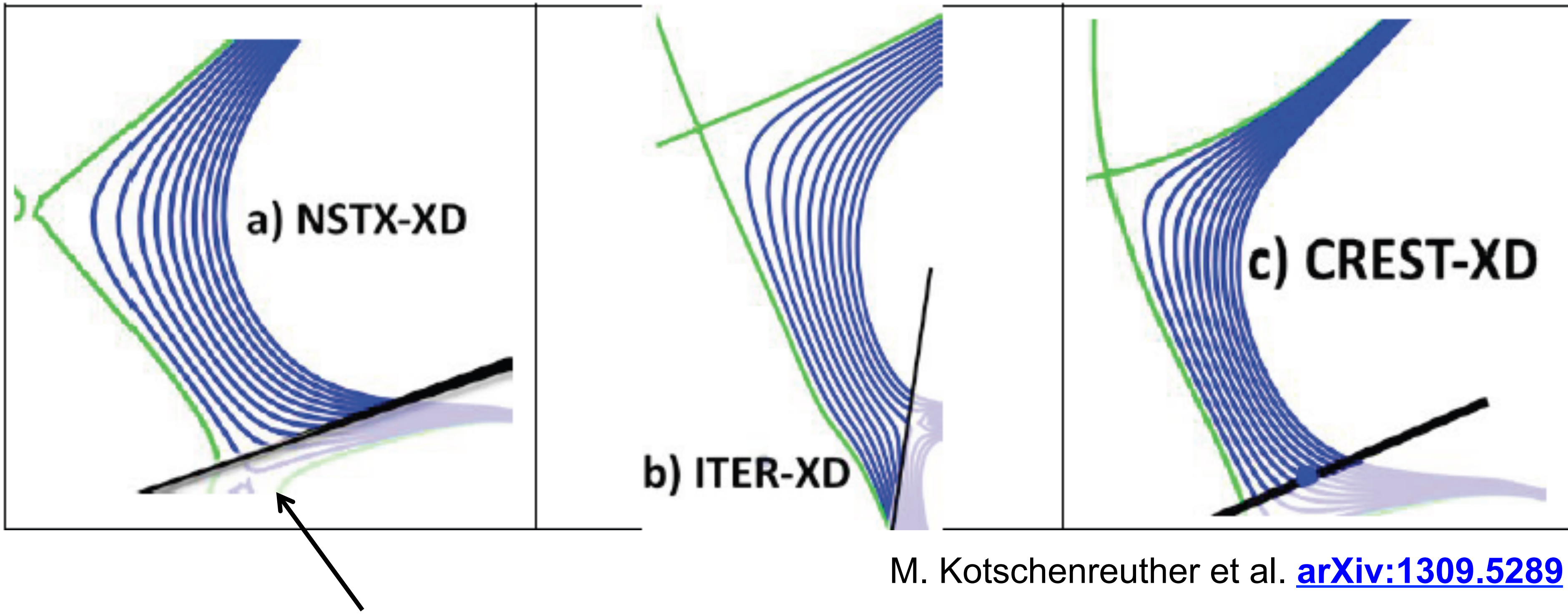




TCV (CH): Snowflake

NSTX (USA): Snowflake

Risk mitigation (II): a (Super-) X divertor



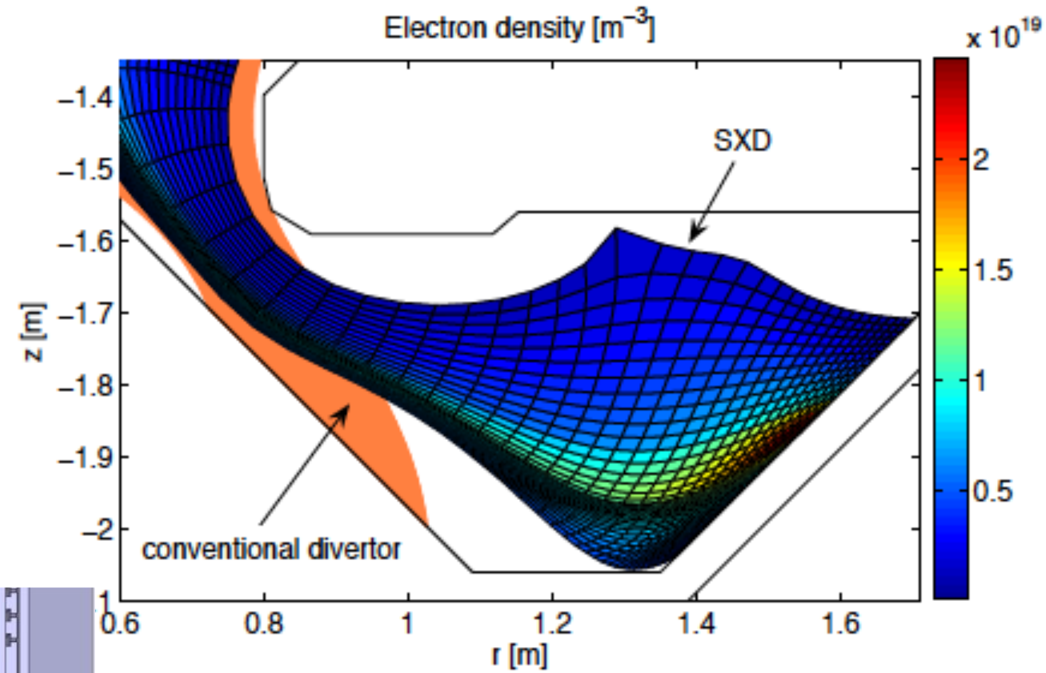
M. Kotschenreuther et al. arXiv:1309.5289

Second X-point ➔ low poloidal B

Caveat of high flux expansion and thus potentially too low impact angles on target plate → but Super-X may reduce issue

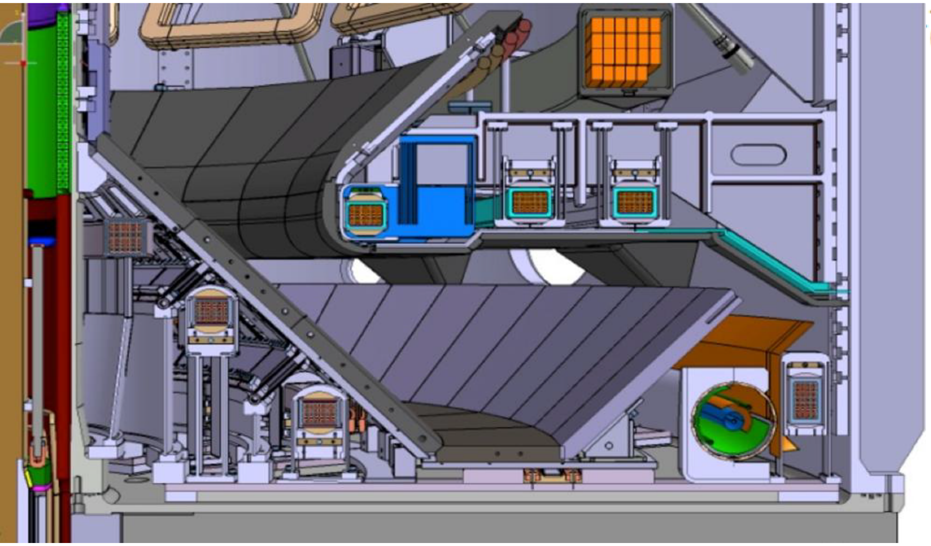
Super-X concept:
Valanju et al. Phys. Plasmas 16, 056110 (2009)

Advanced divertor configurations (III)



MAST Super-X (UK)

E. Havlickova et al., JNM 2013



E. Havlickova et al., PET 2013

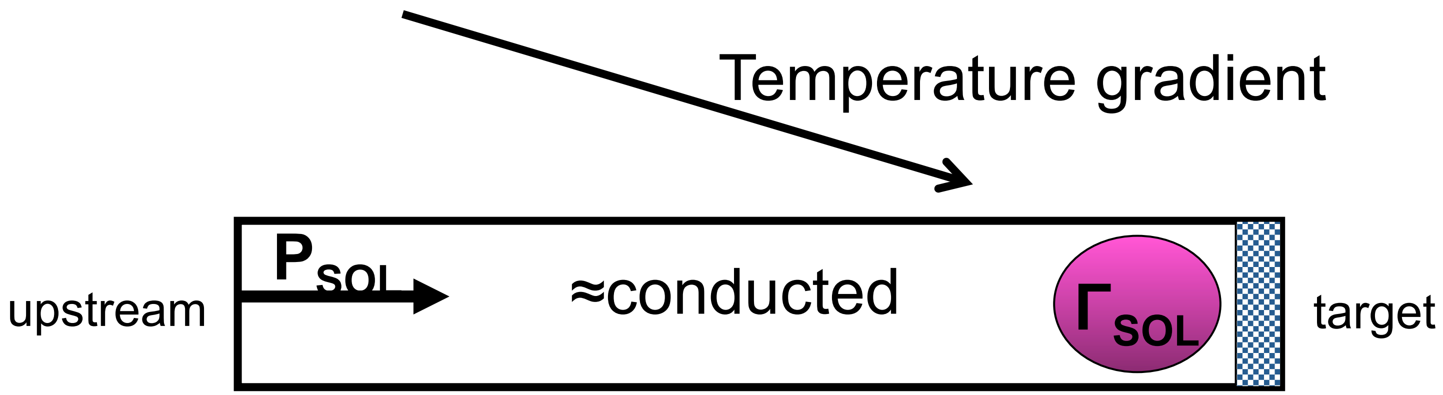
How can we reduce the power load onto the divertor target plates to match the technological limit?

2. Basic SOL Physics

Divertor Regimes: high recycling

Total plasma pressure is constant along magnetic field line

$P_e + P_i$ + dynamic pressure = constant



➔ High recycling regime: low T_e (< 5eV), high n_e

➔ Satisfactory for existing tokamaks

➔ VERY HIGH PARTICLE FLUXES

Do we need to limit the particle flux?

Neglecting power loads on PFCs from radiation

➔ *Total power = (8T + 15.8) 1.602* $10^{-19}\,\Gamma$ *[W] ;* $T_e = T_i = T$ *[eV]*

Power across sheath   Surface recombination of D^+ to D_2

What does < 5 -10 MW/m^2 imply for particle flux?

Neglecting power loads on PFCs from radiation

➔ *Total power = (8T + 15.8) 1.602 $10^{-19}\Gamma$ [W] ; $T_e = T_i = T$ [eV]*

Power across sheath   Surface recombination of D^+ to D_2

❖ *For T_e < 2.5 eV → heat flux similar to power deposited by surface recombination processes*

❖ *Power load via radiation to ~2 MW/m^2*

What does limit of < 5 -10 MW/m^2 mean for the particle flux?

Neglecting power loads on PFCs from radiation

➔ *Total power = (8T + 15.8) 1.602 10^{-19} Γ [W] ; $T_e = T_i = T$ [eV]*

Power across sheath   Surface recombination of D^+ to D_2

- *For T_e < 2.5 eV → heat flux similar to power deposited by surface recombination processes*
- *Power load via radiation of ~2 MW/m^2*
- *5 MW/m^2 with T = 1.5 eV ➔ Γ < 5e23 $m^{-2}s^{-1}$*

What does limit of < 5 -10 MW/m^2 mean for the particle flux?

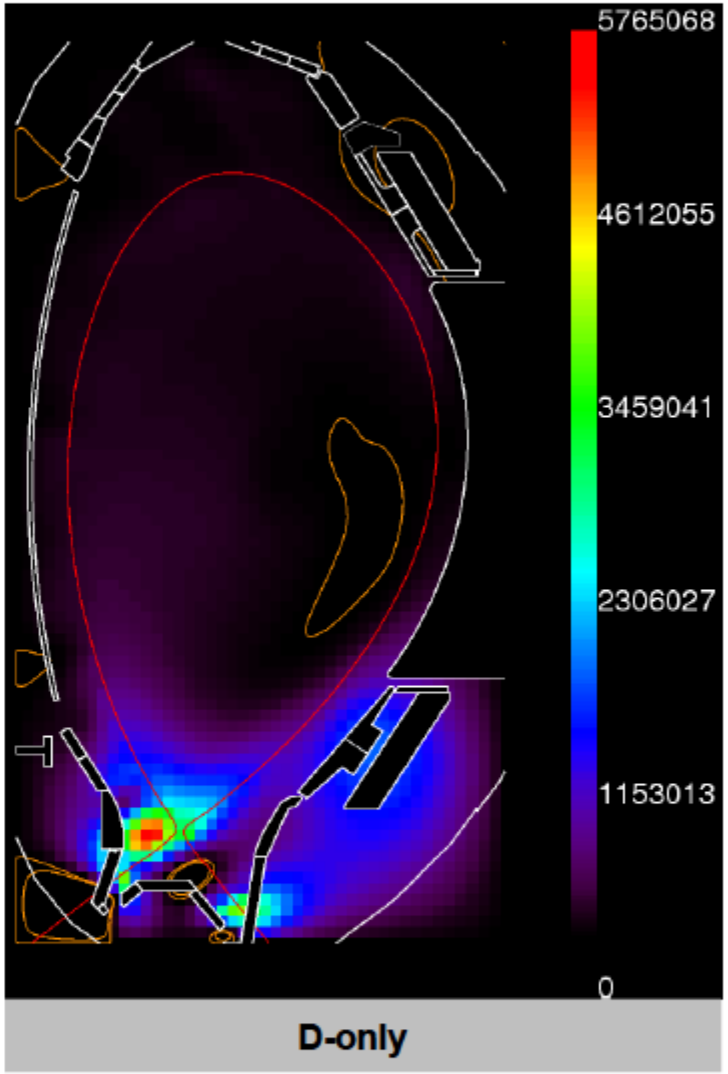
Target power flux < 5 MW/m^2 → ~1GW/m^2 upstream

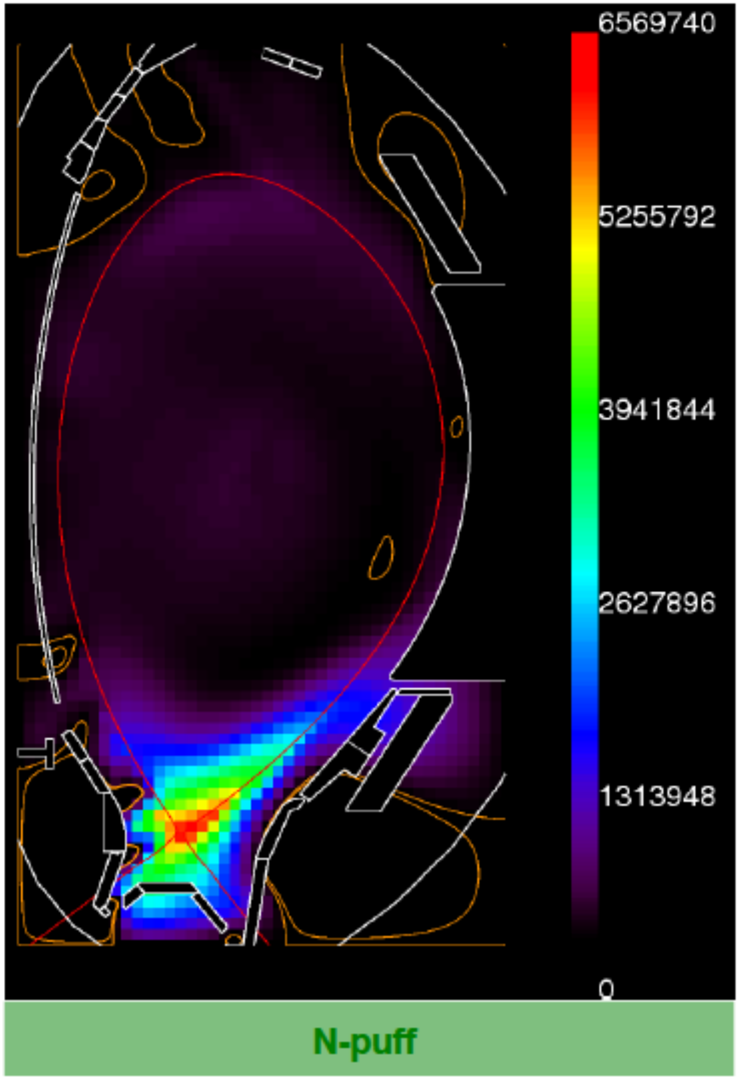
For DEMO: > 95% of power need to be radiated + Ion flux to target reduced to *5 10^{23}m^{-2}s^{-1}*

For ITER (10 MW/m^2 target limit): ~60-80% of power needs to be radiated + Ion flux to target reduced to ~10^{24}m^{-2}s^{-1}

Can we achieve these 5MW/m^2 in existing tokamaks with high P/R?

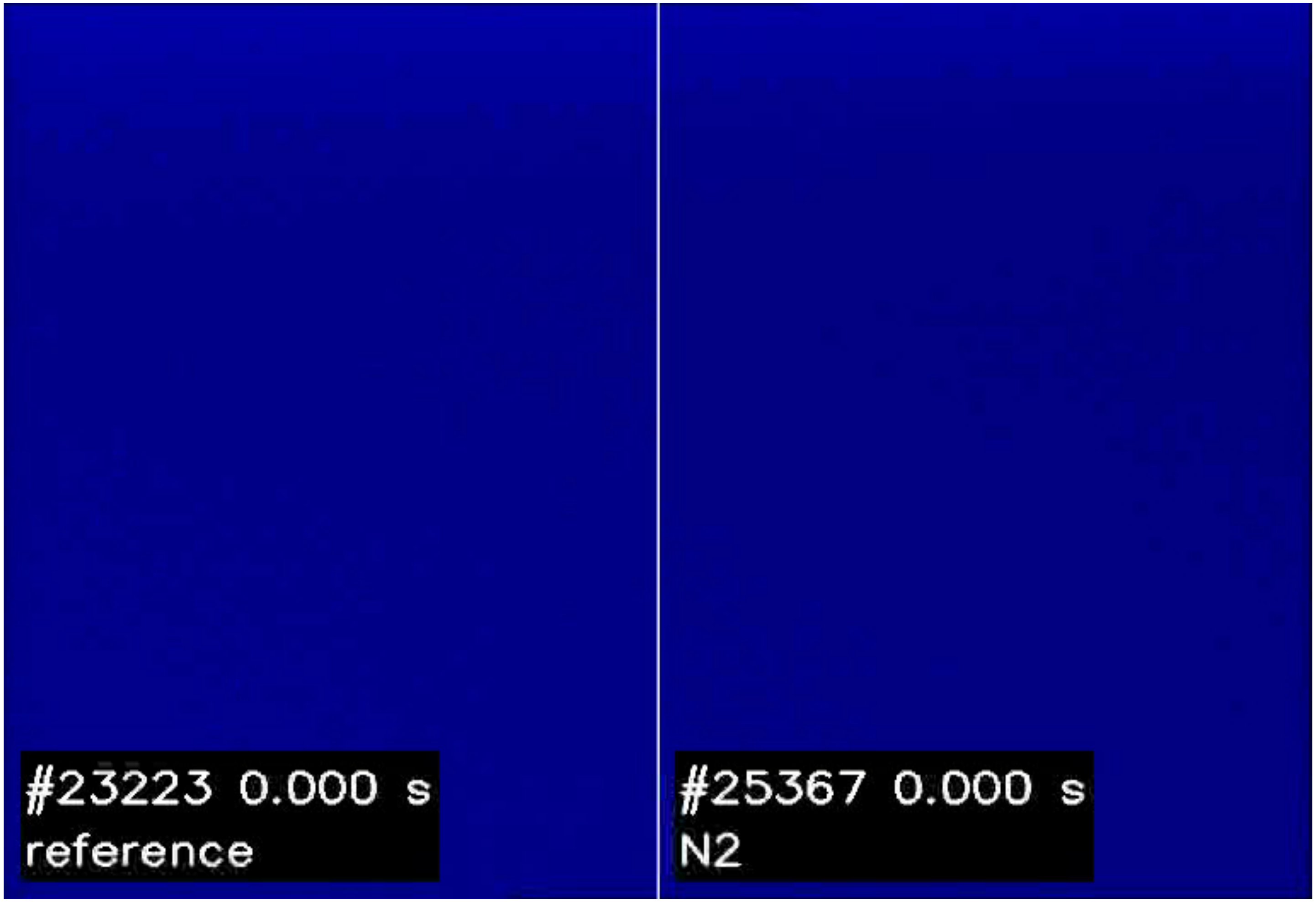
Impurity seeding (I)





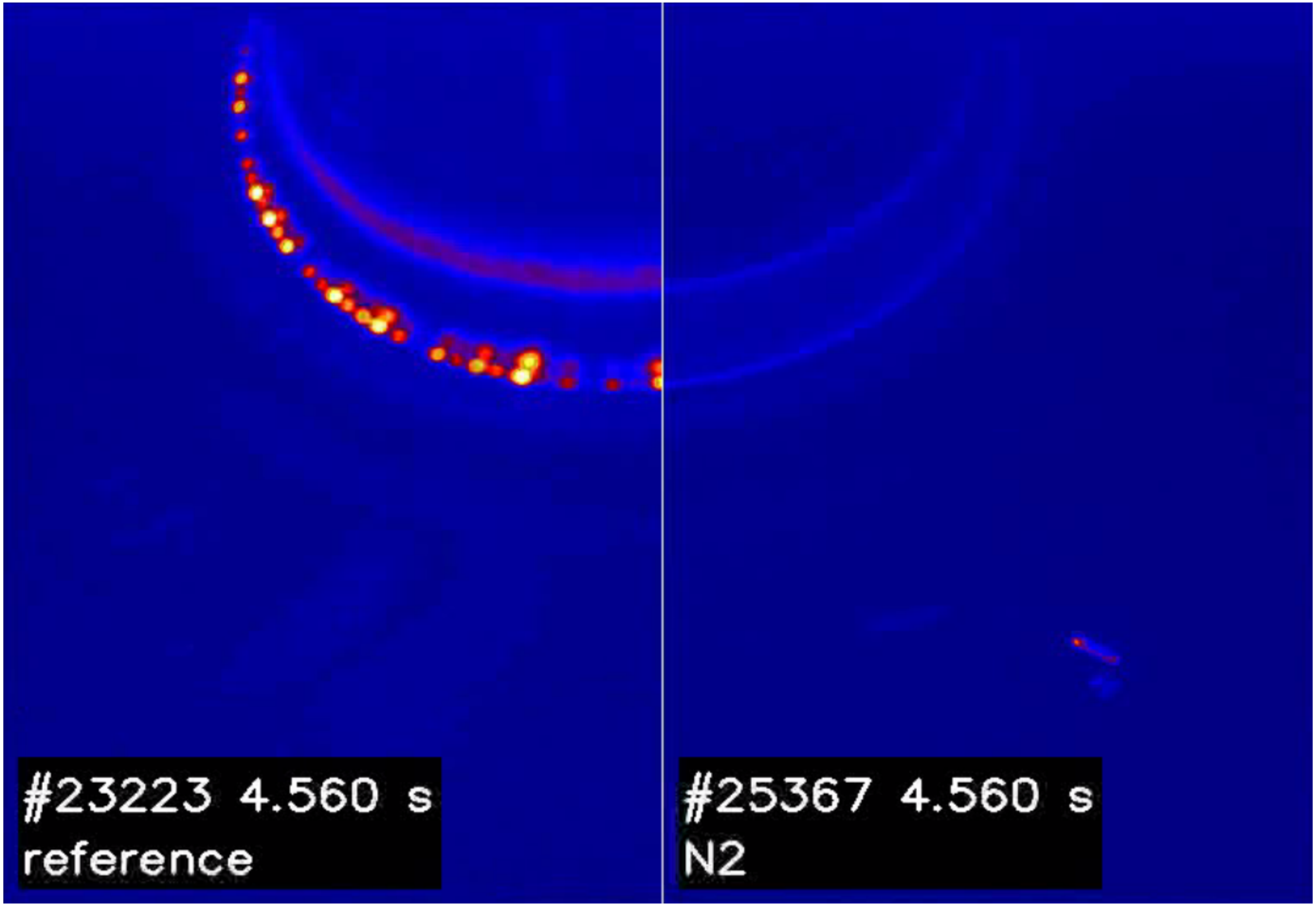
F. Reimold et al. DPG 2012

Impurity seeding (I)

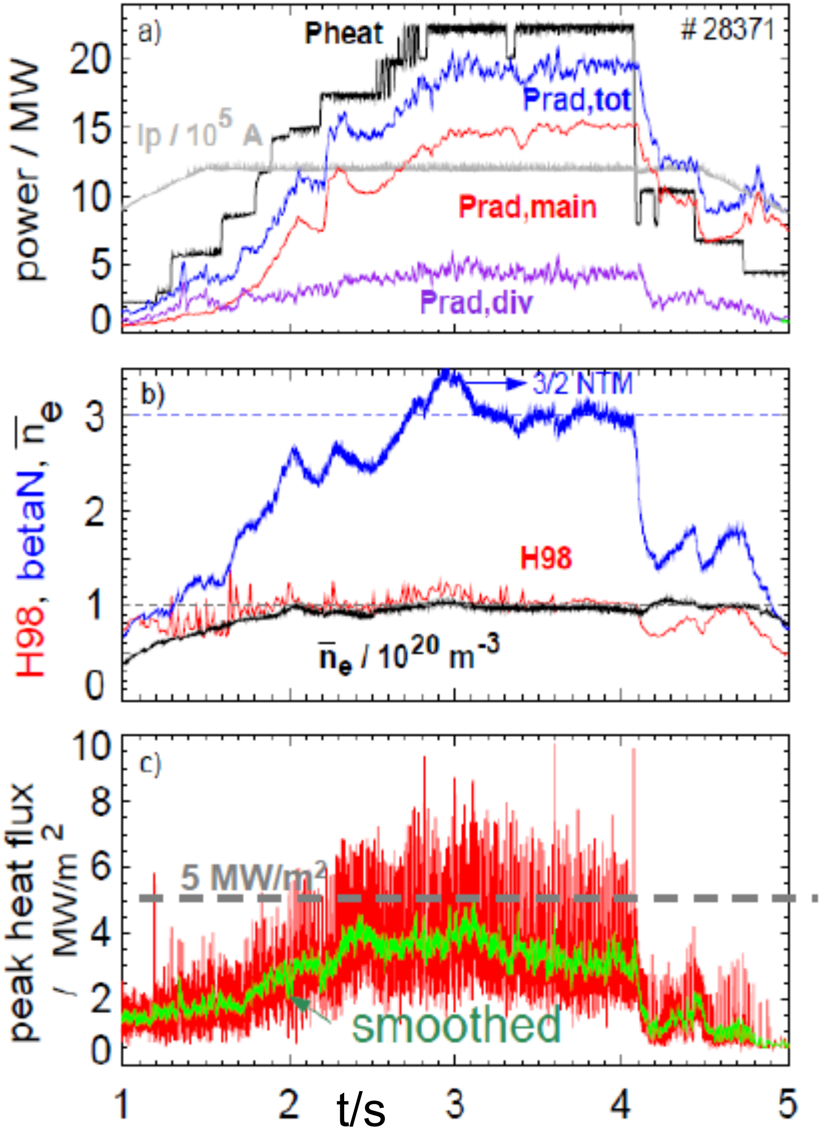


ASDEX Upgrade with all W PFCs

Impurity Seeding (II)



Highly Radiating Plasma (ASDEX Upgrade)



- ❖ P_{heat}= 23 MW
- ❖ P_{heat}/R=14 (world record)
- ❖ Seeding of Ar and N_2
- ❖ Reduction to <5MW/m^2

A. Kallenbach et al. NF 2012

Power flux can be dropped to < 5MW/m^2 in existing devices with high P/R

How is the particle flux limited?

"Modified" two point model as guidance

$$T_t = \frac{q_{\|}^2}{n_u^2}\left(\frac{7q_{\|}L_c}{2\kappa_{0e}}\right)^{-4/7}\frac{2m_i}{\gamma^2 e^2}\frac{(1-f_{pow})^2}{(1-f_{mom})^2(1-f_{conv})^{4/7}}$$

$$n_t = \frac{n_u^3}{q_{\|}^2}\left(\frac{7q_{\|}L_c}{2\kappa_{0e}}\right)^{6/7}\frac{\gamma^2 e^3}{4m_i}\frac{(1-f_{mom})^3(1-f_{conv})^{6/7}}{(1-f_{pow})^2}$$

Particle flux

$$\Gamma_t = \frac{n_u^2}{q_{\|}}\left(\frac{7q_{\|}L_c}{2\kappa_{0e}}\right)^{4/7}\frac{\gamma e^2}{2m_i}\frac{(1-f_{mom})^2(1-f_{conv})^{4/7}}{(1-f_{pow})}$$

f_{pow}: power loss factor (0 – 1) → What is the maximum value?
f_{conv} : 0=no convection; 1= only convection → What is the interplay?
f_{mom}: momentum loss factor (0 – 1) → What is the maximum?

- Value of the loss factors and what interdependence?
- System codes will require scaling laws to define operational regime of DEMO type device

Divertor Regimes: detachment

Prerequisite: Loss of plasma pressure

a) Radiation in the edge of the plasma core

➔ Reduction of upstream plasma pressure

➔ Reduced recycling

At low Te large Complexity of volumetric and surface processes

Reaction
$H + e \rightarrow H^+ + 2e$
$H + H^+ \rightarrow H^+ + H$
$H_2 + e \rightarrow H + H + e$
$H_2 + e \rightarrow H_2^+ + 2e$
$H_2 + e \rightarrow H + H^+ + 2e$
$H^+ + H_2 \rightarrow H^+ + H_2$
$H^+ + H_2 \rightarrow H + H_2^+$
$H_2^+ + e \rightarrow H + H^+ + e$
$H_2^+ + e \rightarrow 2H^+ + e$
$H_2^+ + e \rightarrow 2H + e$
H^+ + electrons(s) $\rightarrow H + h\nu$ or electrons
$C + e \rightarrow C^+ + 2e$
$H^+ + C \rightarrow C^+ + H$

+ seeded processes for impurities…
+ surface interaction physics (reflection, recycling)

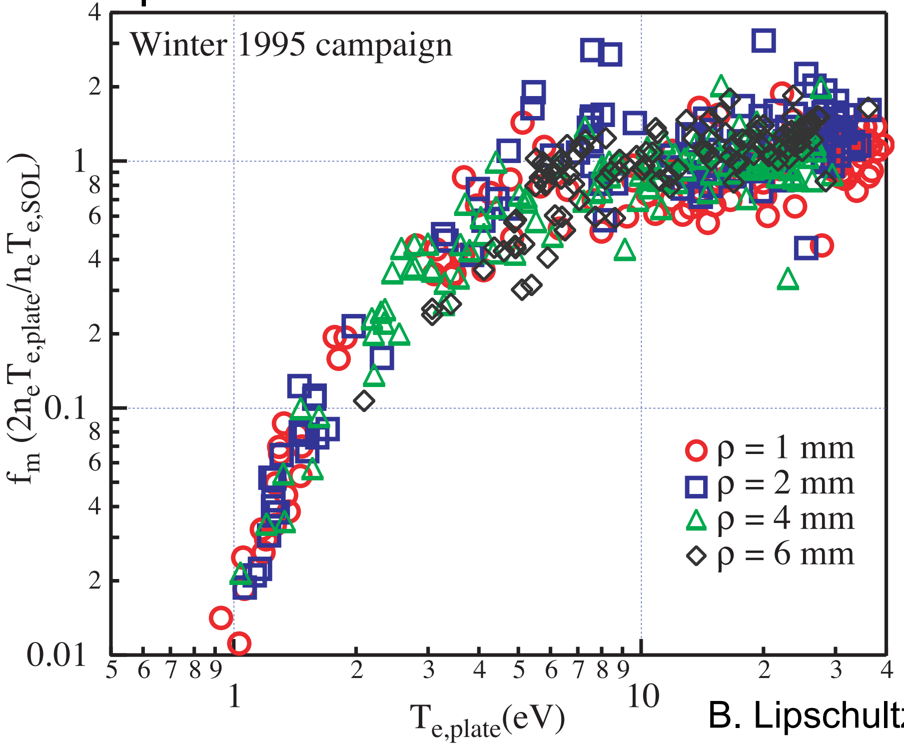
Molecular assisted recombination	MAR	$D_2(v) + D^+ \rightarrow D_2^+ + D$	$D_2^+ + e \rightarrow D + D$
Molecular assisted dissociation	MAD	$D_2(v) + D^+ \rightarrow D_2^+ + D$	$D_2^+ + e \rightarrow D + D^+ + e$
Molecular assisted ionization	MAI	$D_2(v) + D^+ \rightarrow D_2^+ + D$	$D_2^+ + e \rightarrow D^+ + D^+ + 2e$

Divertor Regimes: detachment

b) Pressure loss along field line

- perpendicular transport (independent of T_e)
- CX reaction losses (T_e<5eV)

Ratio of target plasma pressure to upstream pressure for C-Mod



$$D + D^+ \rightarrow D^+ + D$$
$$D^+ + D_2 \rightarrow D_2 + D^+$$

Competes with ionization

B. Lipschultz et al., FST 51 (2007)

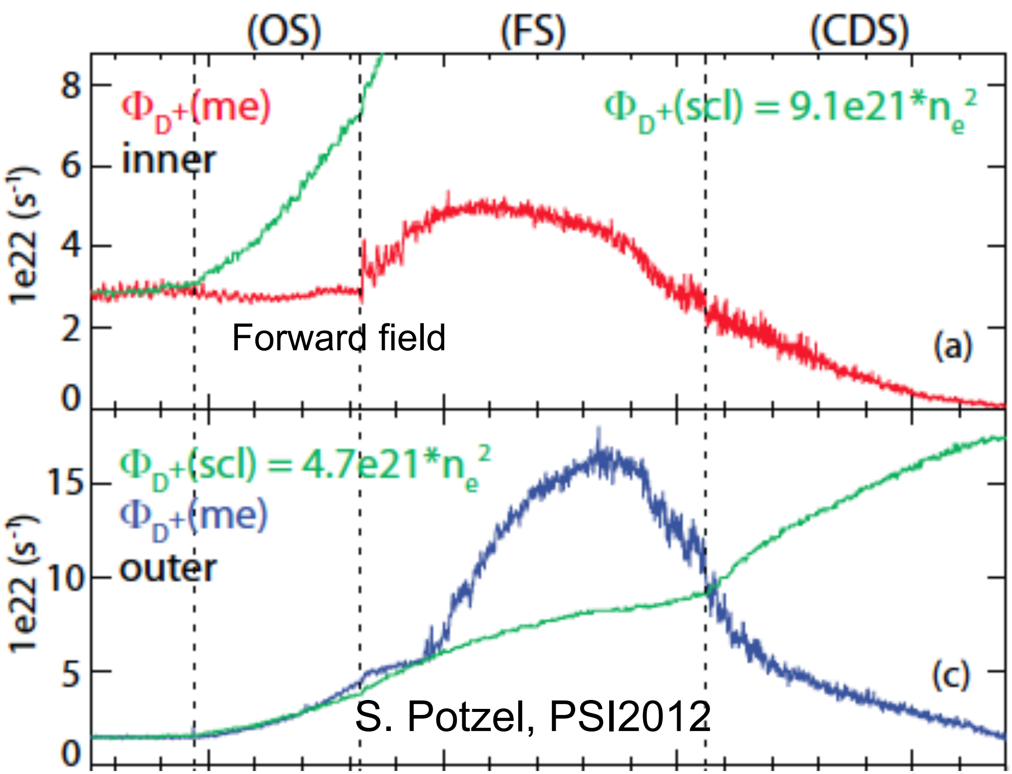
Divertor Regimes: detachment

Prerequisite: Loss of plasma pressure on a field line

b) Pressure loss along field line
- perpendicular transport (independent of T_e)
- CX reaction losses (T_e<5eV)

Density ramp experiments in ASDEX Upgrade

Total ion flux to inner and outer divertor



Time /s = increase of density →

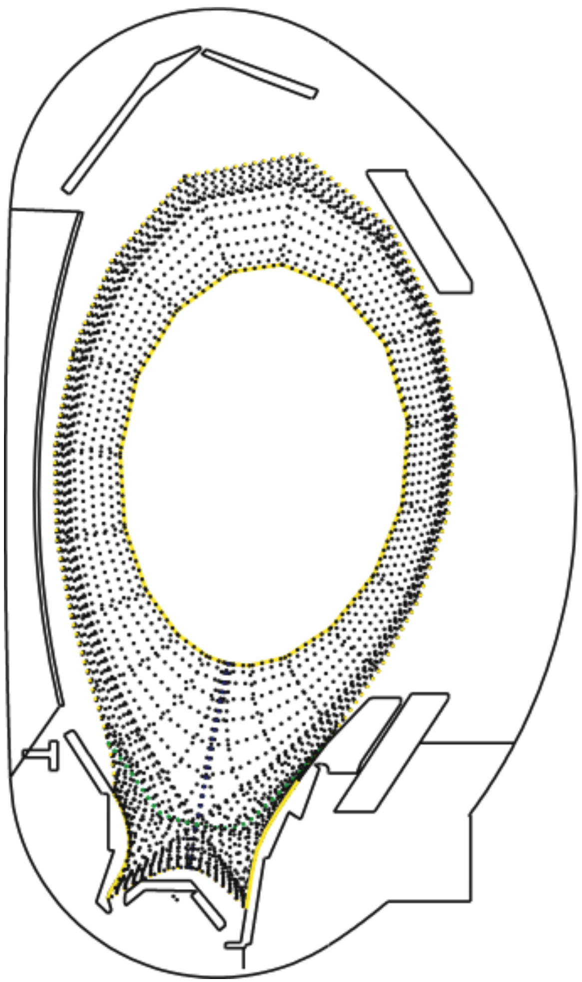
<u>Signature of detachment:</u>

- Volumetric recombination processes (visible in Balmer series)
- reduction of ion flux density on target plates

S. Potzel et al. NF 2014

- Asymmetry of particle fluxes
- Integral 'roll over' at similar time/density for inner and outer

How to extrapolate from exiting devices to future tokamaks?

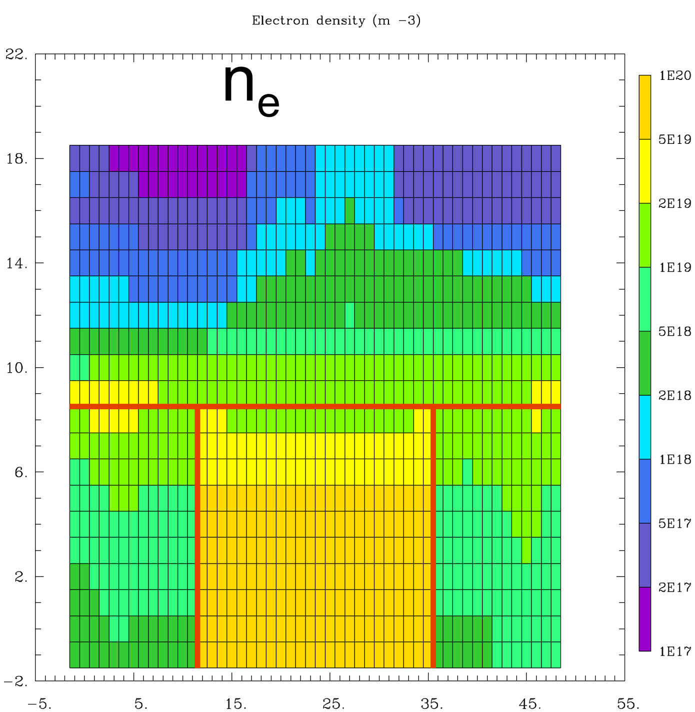


...but requires well established understanding of existing devices first..

Numerical tool

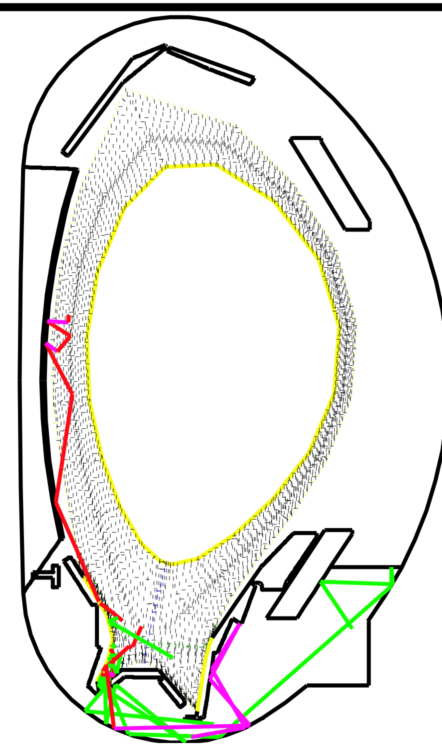
- Multi fluid code B2.5 (2D)
  - Solves modified fluid equations in 2D (Braginskii)
  - Includes fluid treatment of neutrals
  - Kinetic limits

- EIRENE (’96, ’99), 3D
  - Solves time dependant linear transport equations for test particles (photons, neutrals, test ions)

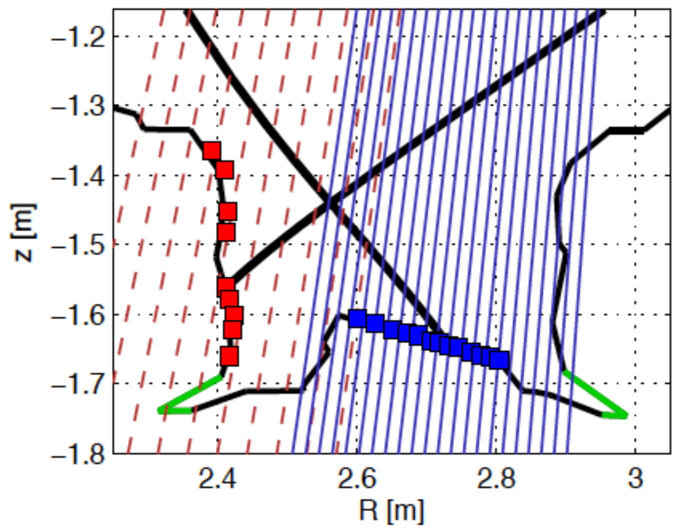


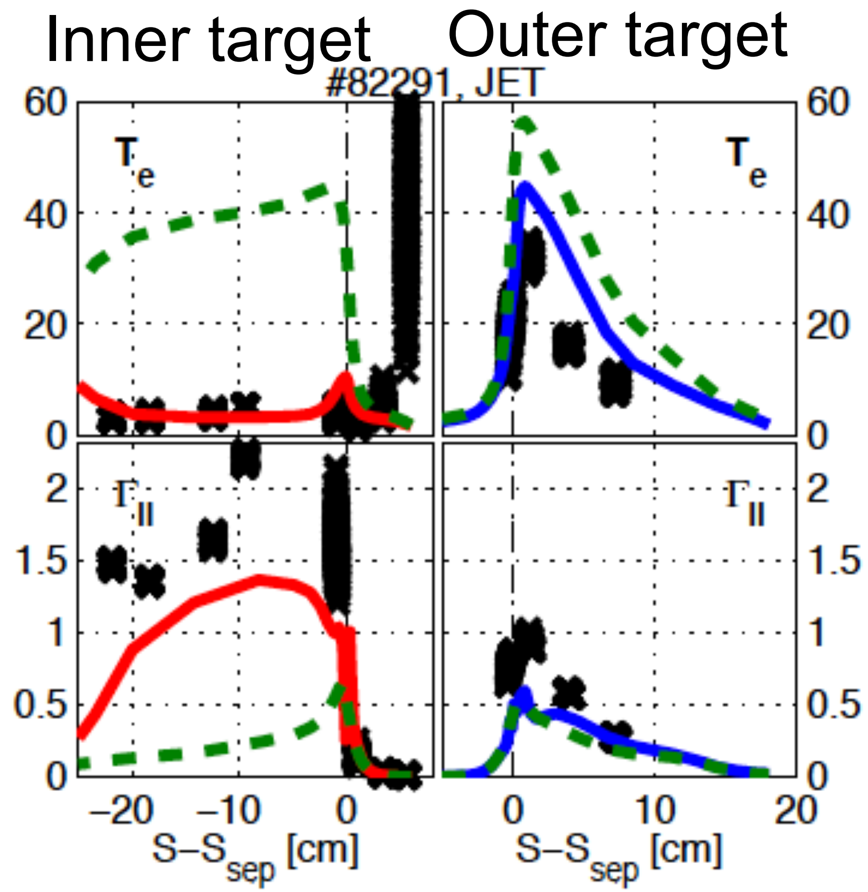
Interface
Fluxes
Sources
Sinks

Can be time dependent



Role of drift terms in simulations for JET



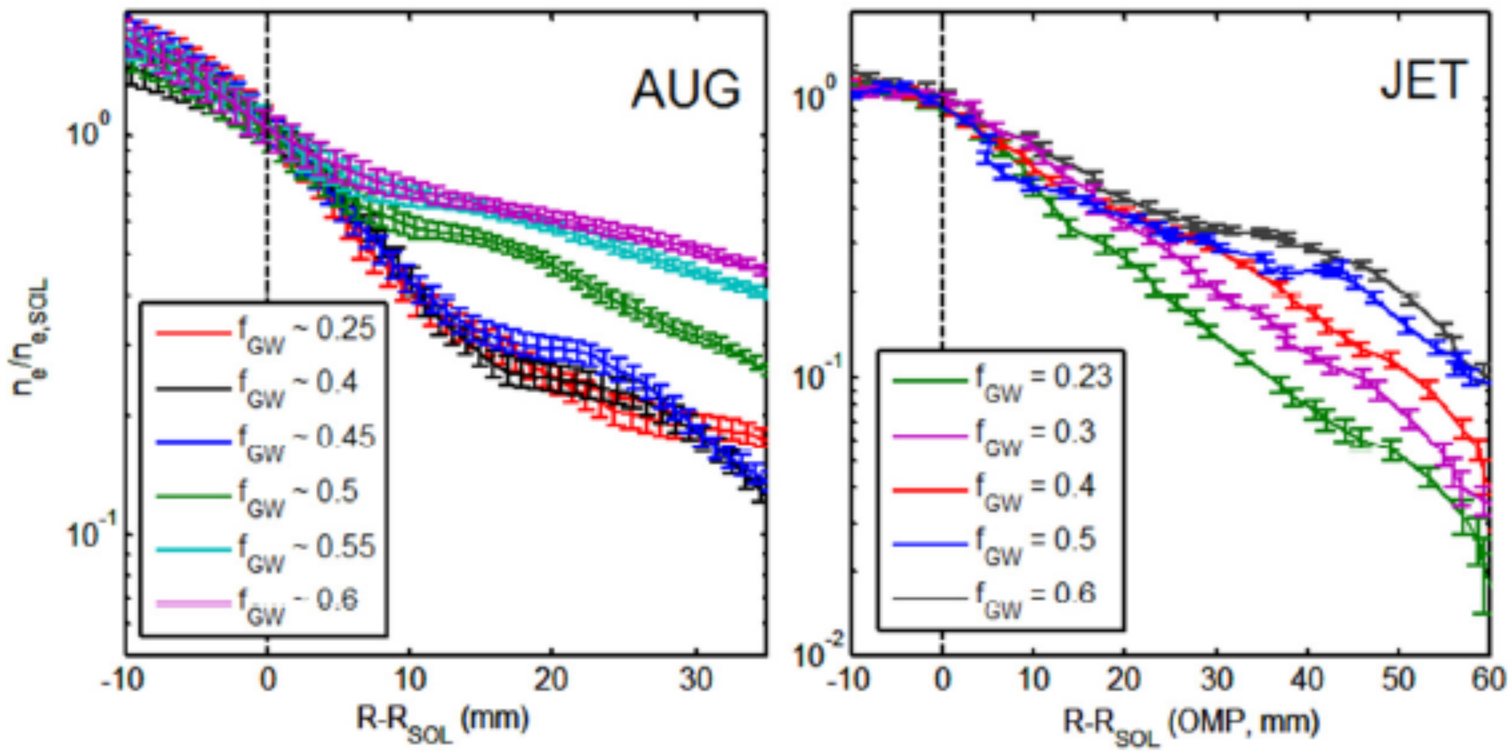


L-mode experiments with semi-horizontal configuration at low density

Exp.
No drifts
Drifts/Drifts

Perpendicular SOL transport: Intermittent transport and main chamber wall interaction

Normalized L-mode SOL density profiles



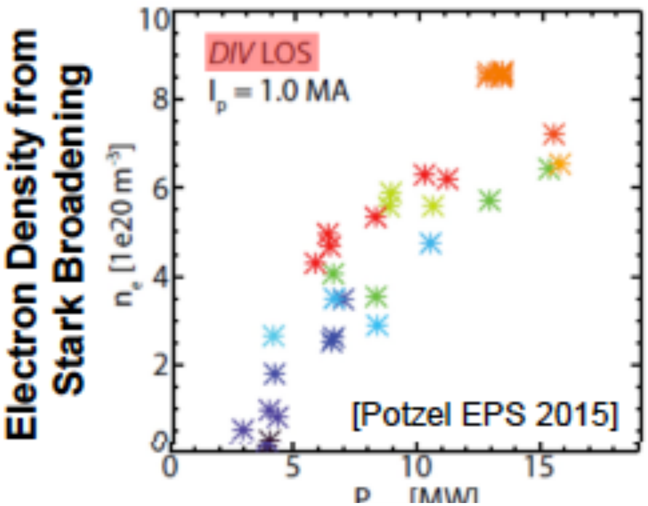
D. Carralero et al. I-16 at PSI14

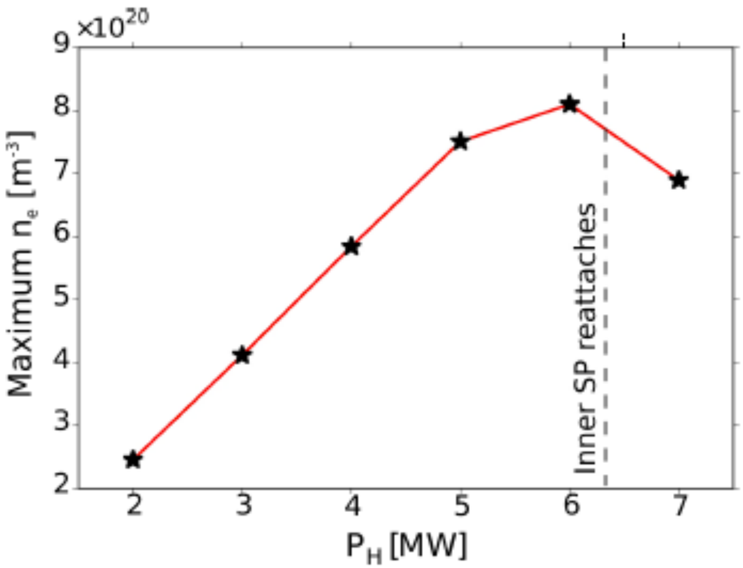
Effective collisionality in divertor determines filamentary perpendicular transport in midplane

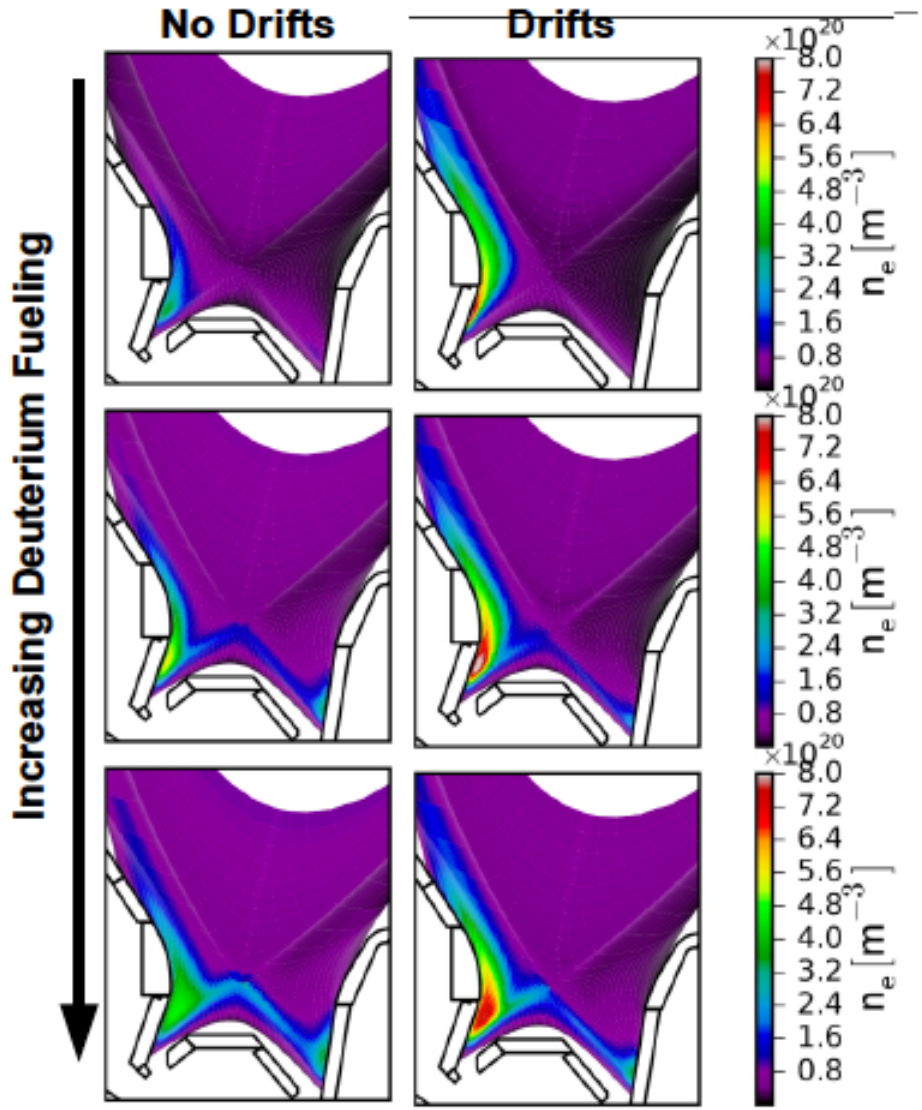
See also:
M. Kocan et al. P3-093 at PSI14
T. Lunt et al. O-23 at PSI14
K. Schmid et al. I-6 at PSI14

B. LaBombard et al. PoP 2001, J. R. Myra et al. PoP 2006

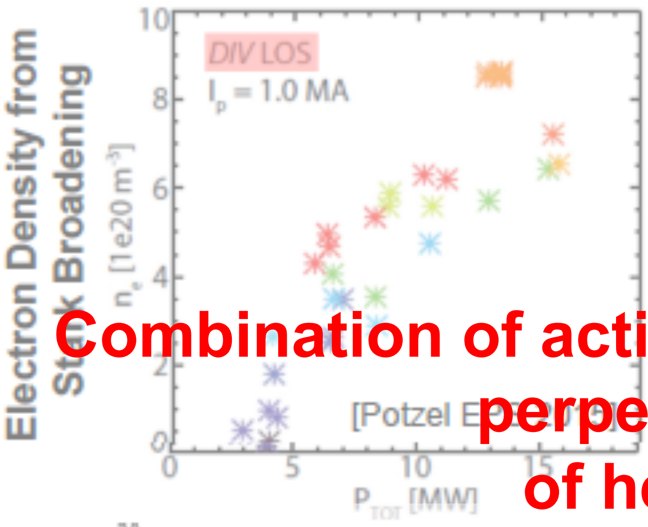
Modeling ASDEX Upgrade and the high field side high density

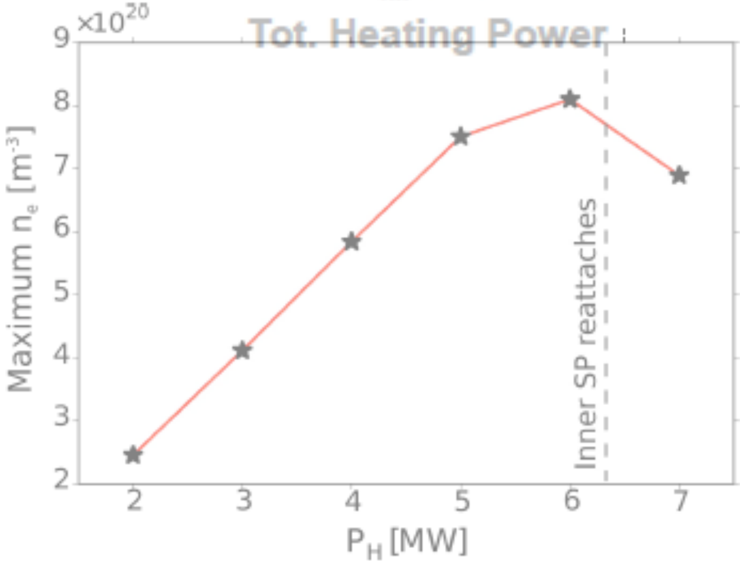


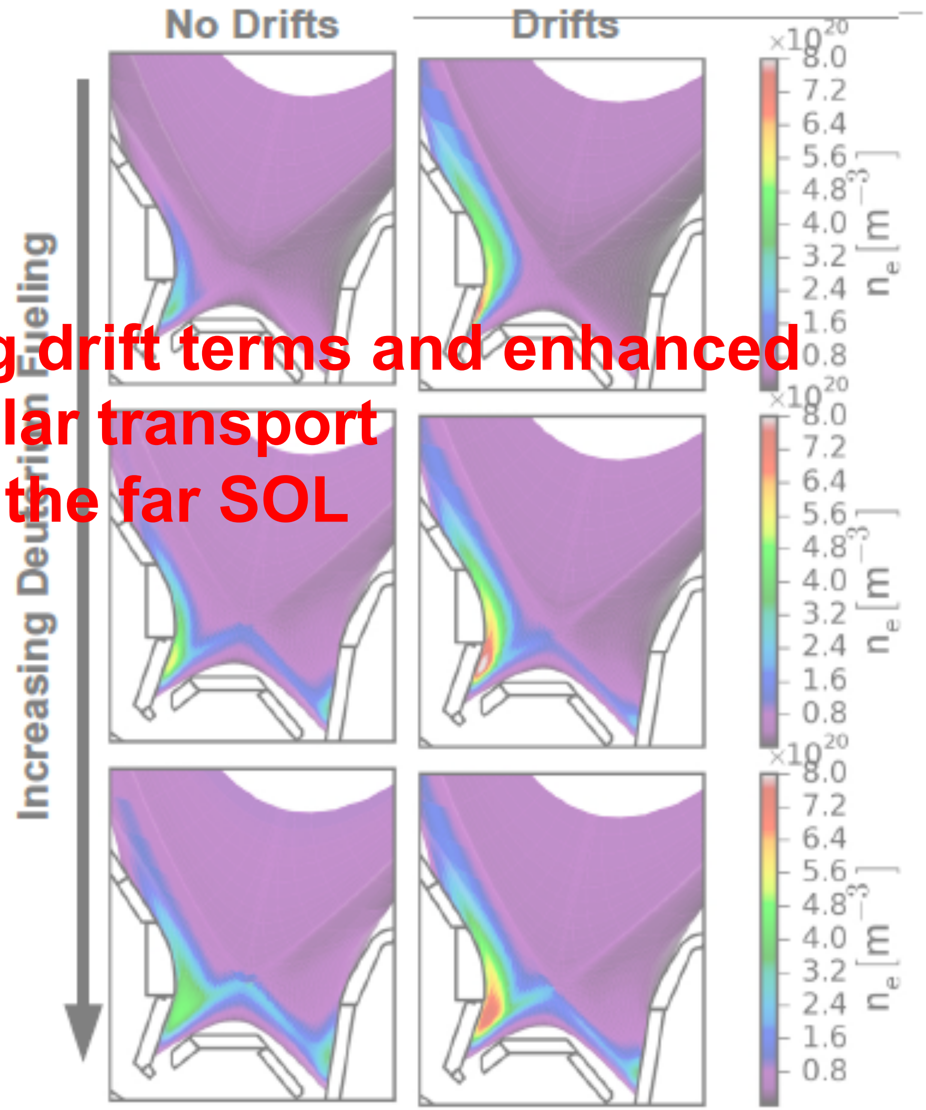




Modeling ASDEX Upgrade and the high field side high density



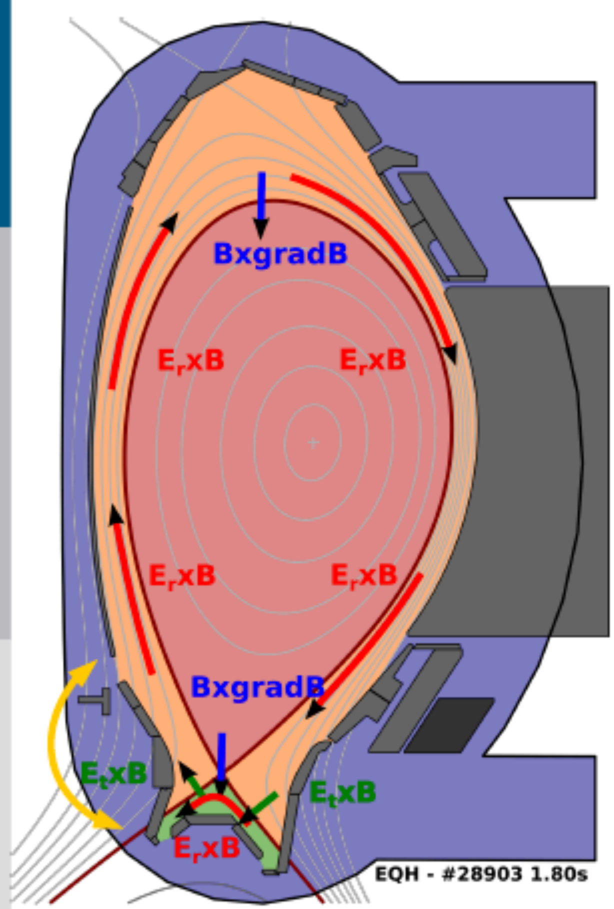




Combination of activating drift terms and enhanced perpendicular transport of heat in the far SOL

Impact of drifts and transport

Drifts change the balance…



In fwd-field drifts are essential to get access:

- ∇B-drift amplifies density in-out asymetry
- ∇B-drift leads to add. perpendicular fluxes [Chankin JNM 1997]

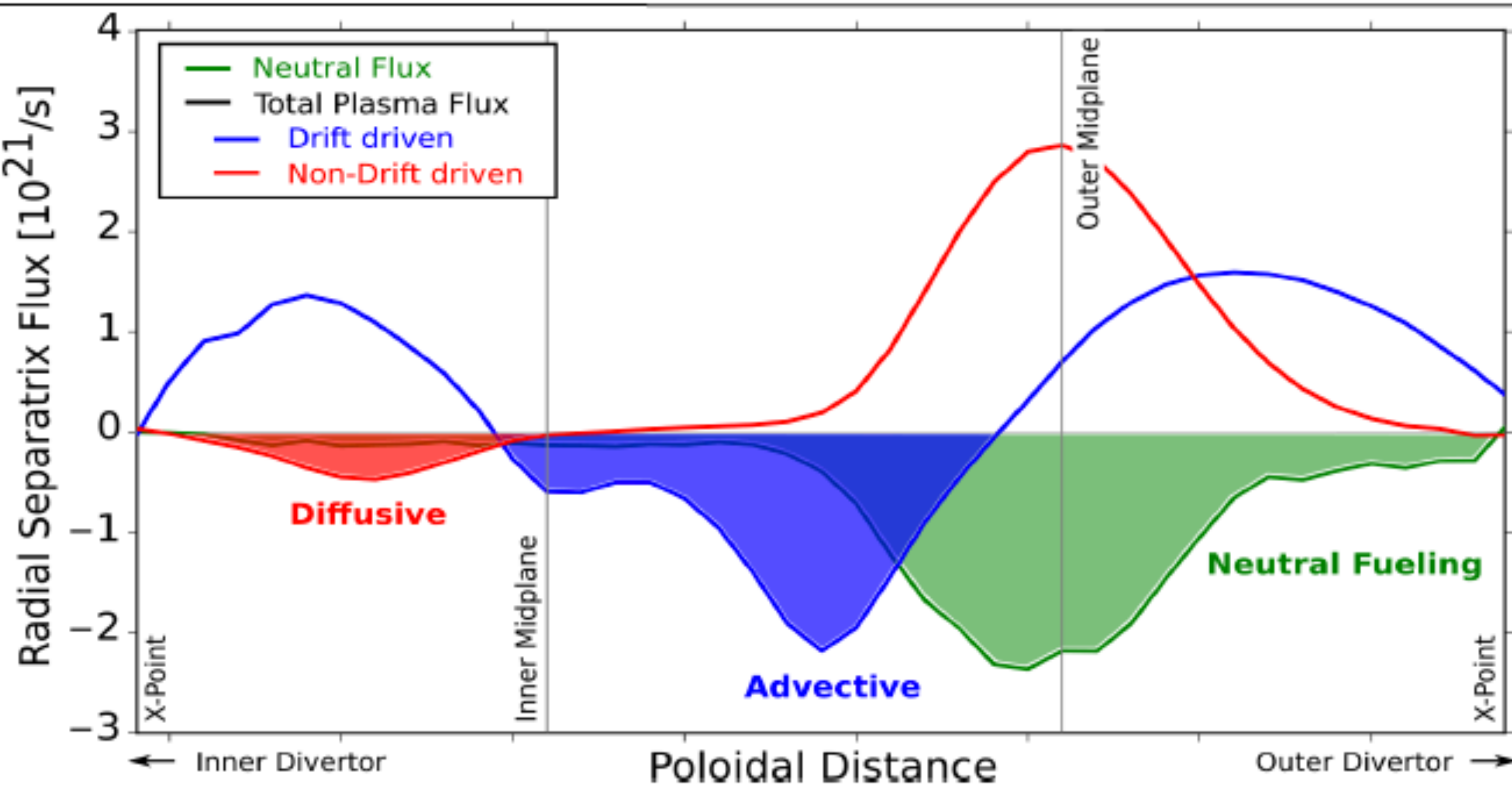
- ExB-drift changes power sharing of divertors
- ExB-drift redistributes particles from outer to inner divertor via PFR [Aho-Mantila EPS (2014)] [Aho-Mantila IAEA (2013)]
- ExB-drift redistributes particles in the perpendicular direction into the farSOL [Reimold Thesis (2015)]

→ Increased recycling above X-point

- Neutral conductances moderate the effect of farSOL (neutral) density (numerically & physically)

→ Effect stand-alone not sufficient

Plasma Fueling of the Core



- Diffusive and advective drift transport of plasma from SOL into confined plasma:
  → Amount of plasma fueling similar to effective neutral influx across separatrix
- Different fueling mechanisms affects different radial regions:
  → Plasma fueling mainly determines separatrix density
  → Neutral fueling mainly determines core boundary density

Summary

- Power exhaust works in existing devices
- Identification of missing physics elements ongoing
- A method for ELM mitigation has been found and fits well with required conditions by power exhaust
- Challenge of power exhaust is increasingly demanding ➔ will determine minimum size of fusion reactor