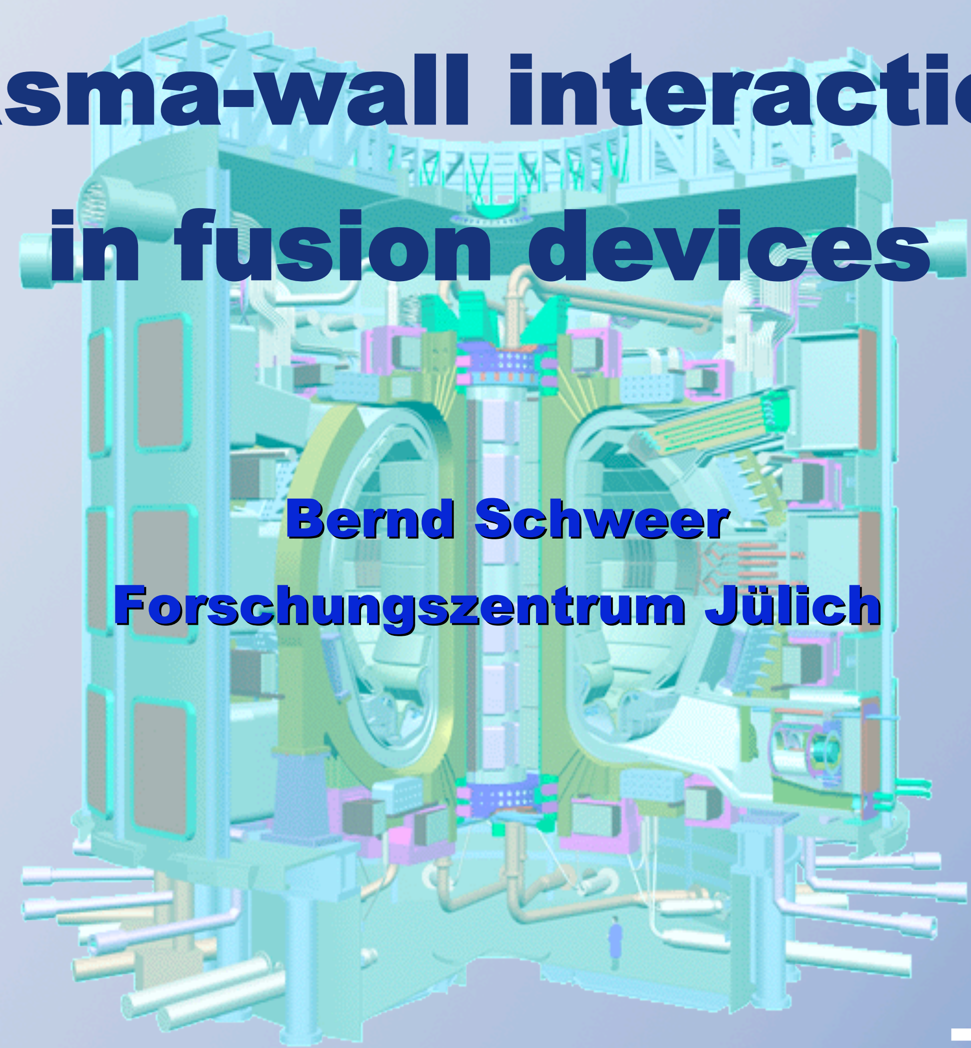
The 5th International Conference on Atomic and Molecular Data and Their Applications
(ICAMDATA) Meudon, France 15-19.10.2006

Plasma-wall interactions in fusion devices

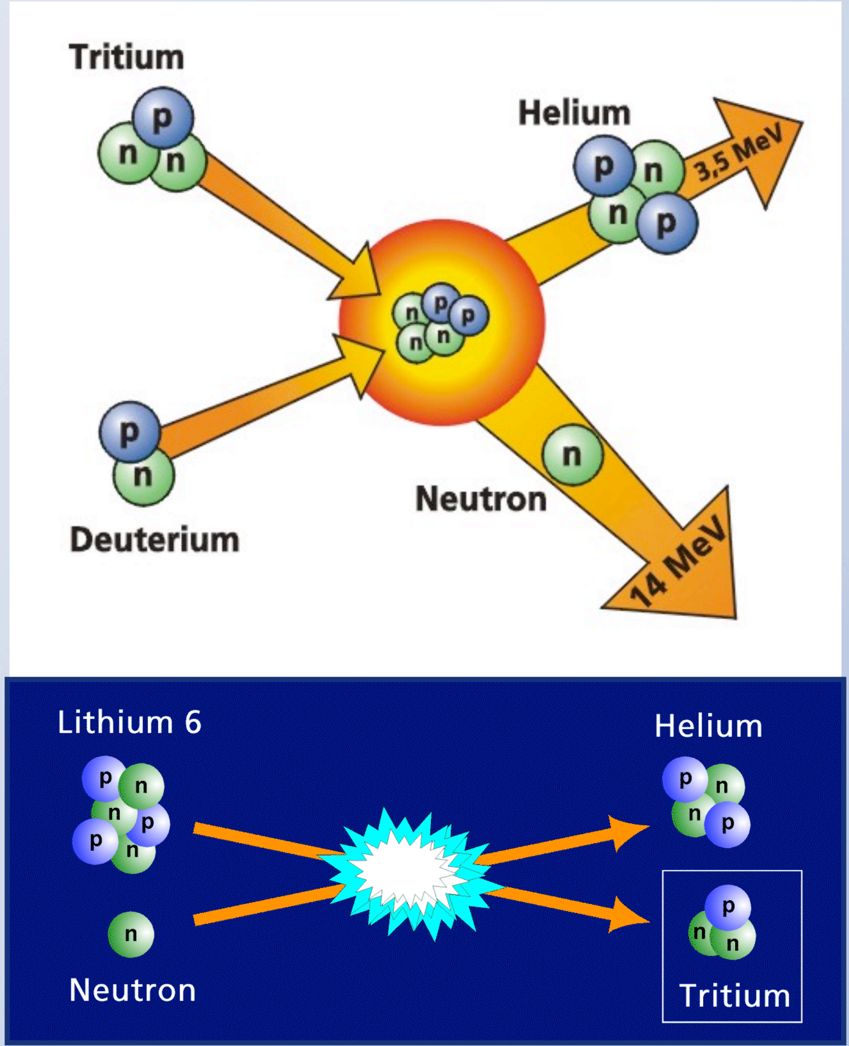


Bernd Schweer
Forschungszentrum Jülich

Trilateral Euregio Cluster
TEC

Forschungszentrum Jülich
in der Helmholtz-Gemeinschaft

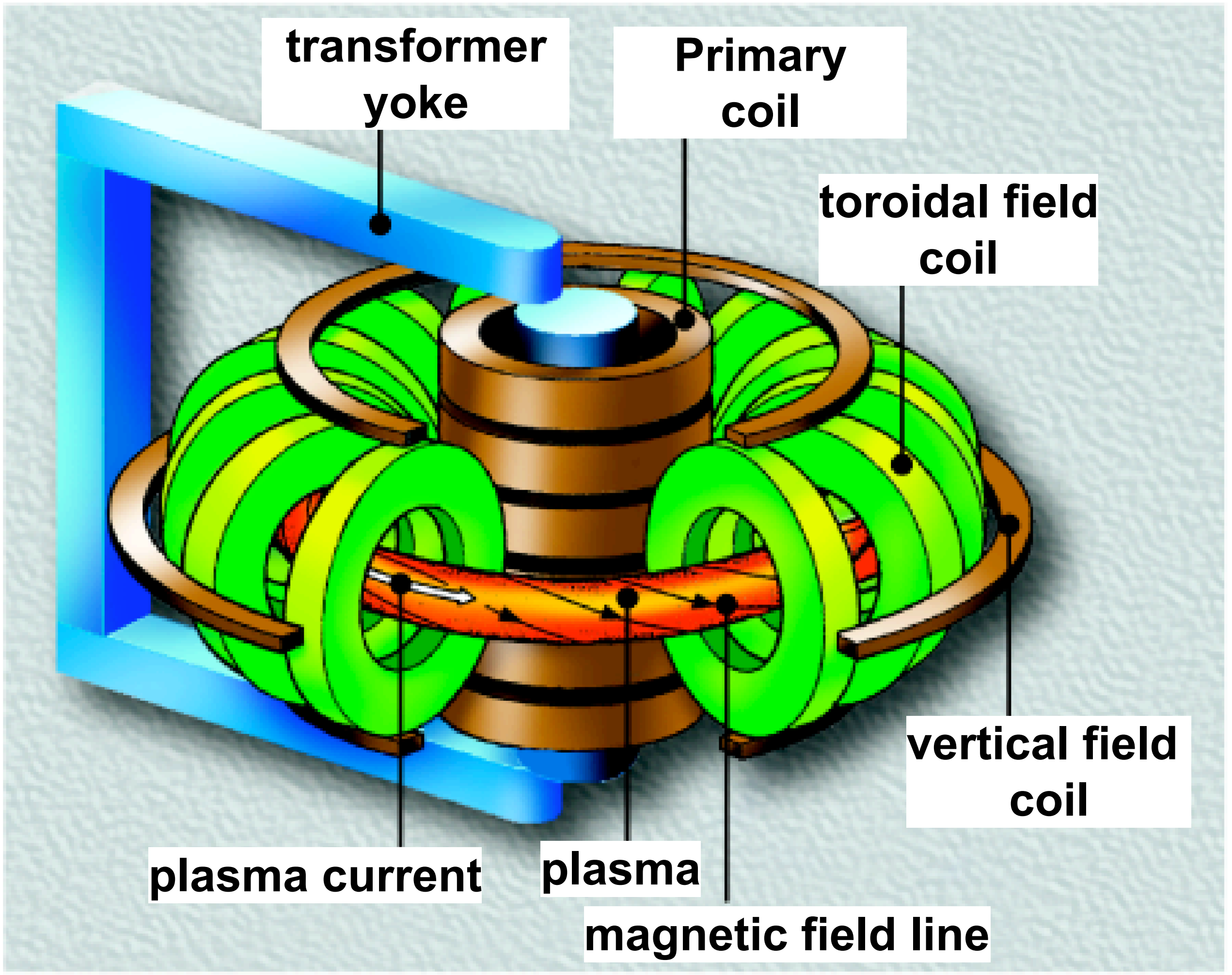
Fusion - a new primary energy source



this contains **75 mg deuterium**
and **225 mg lithium**
as fuel for fusion -
equivalent to 1 000 litres of oil

the fuel is cheap and abundant

Tokamak- principle set-up



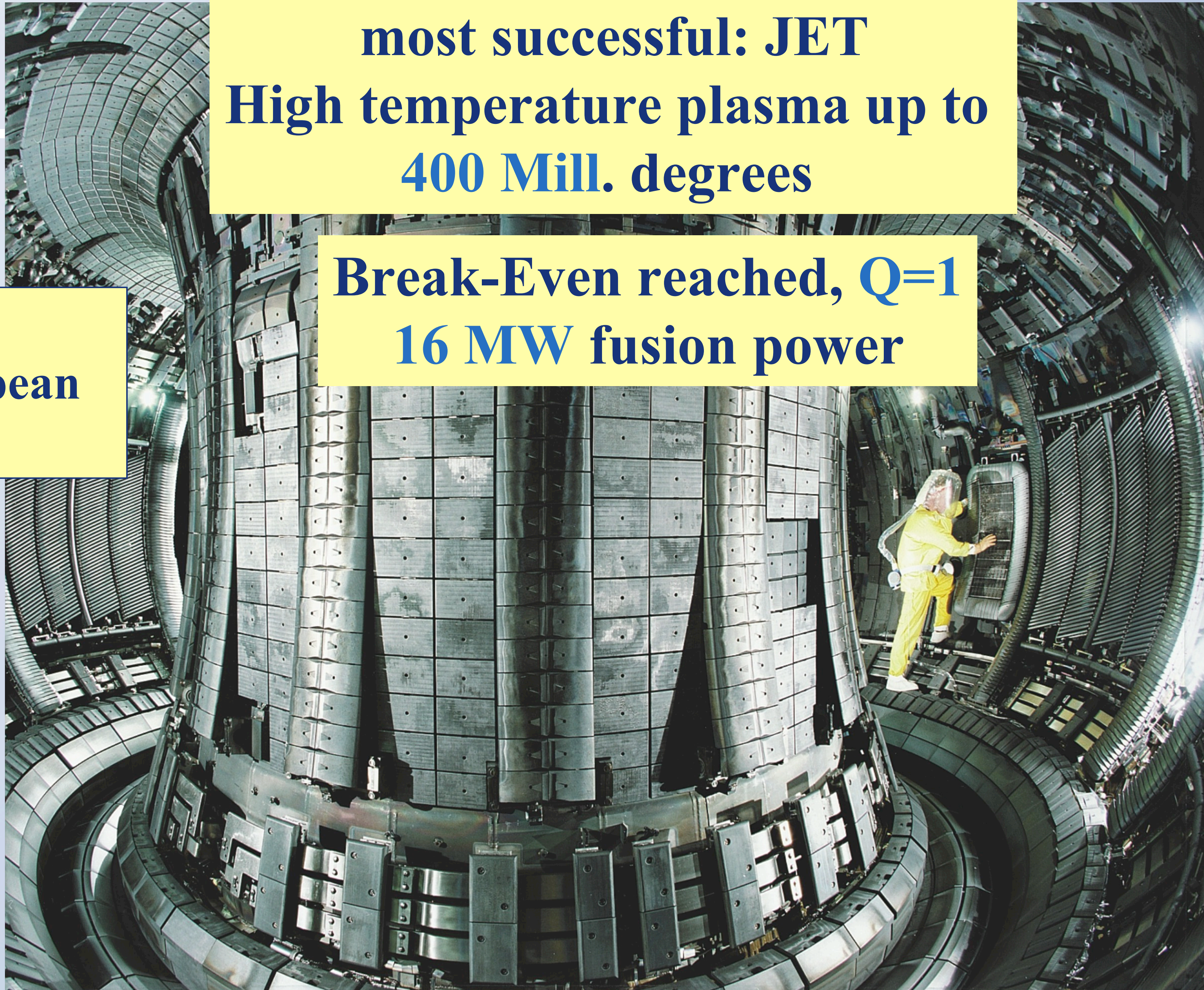
most successful: JET

High temperature plasma up to 400 Mill. degrees

Break-Even reached, Q=1

16 MW fusion power

Joint European Torus



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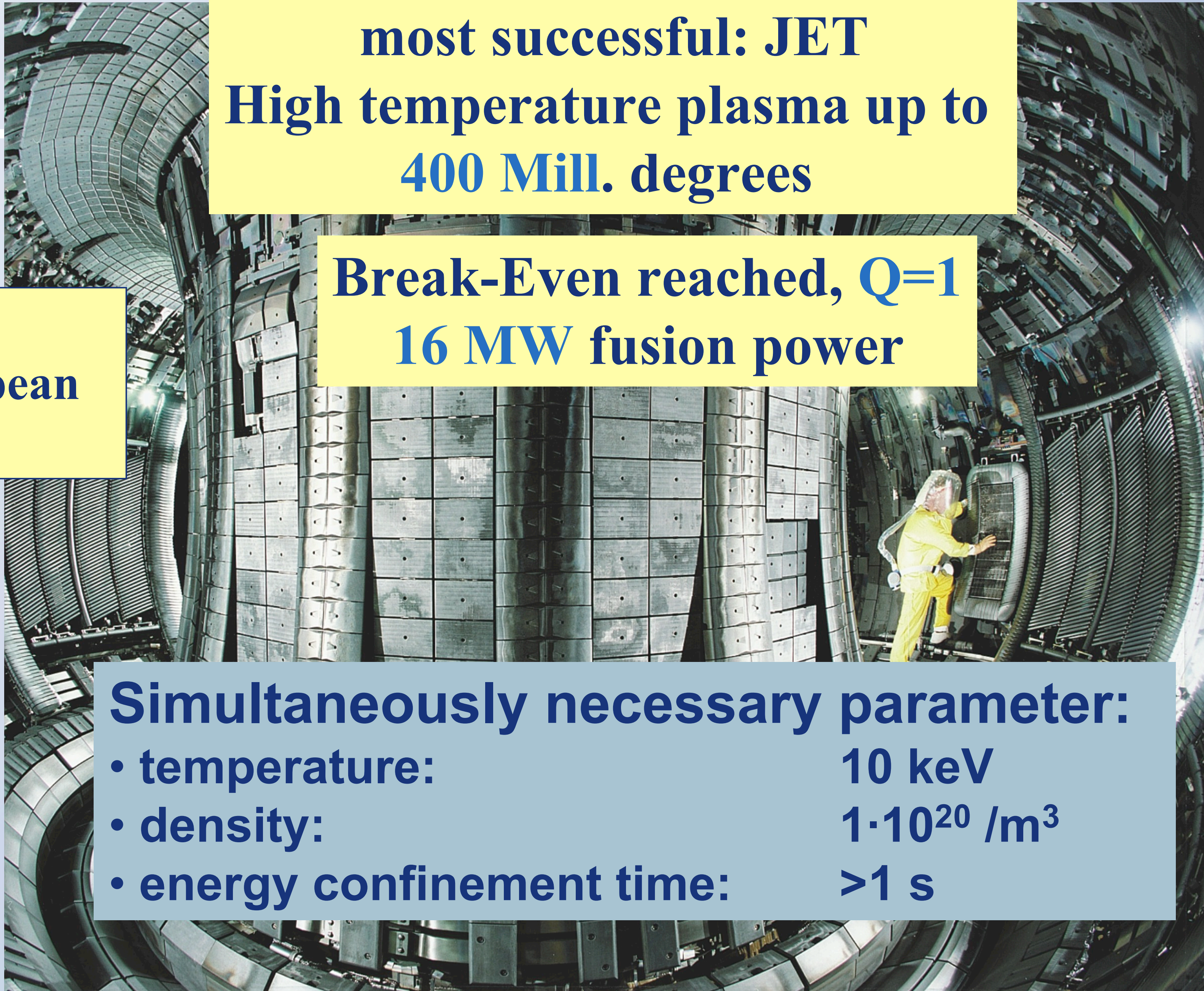


most successful: JET

High temperature plasma up to **400 Mill. degrees**

Break-Even reached, Q=1

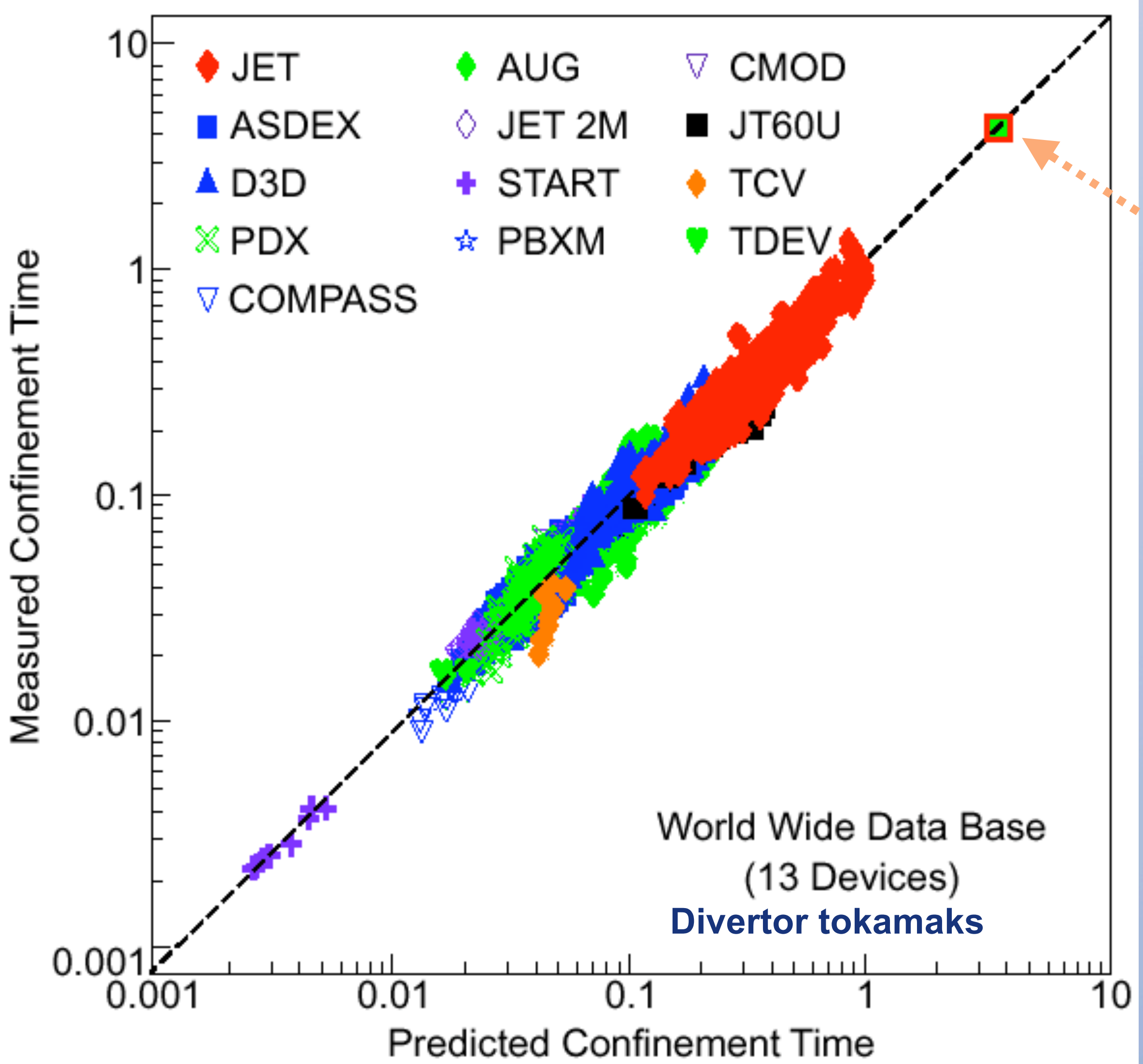
16 MW fusion power

Joint **E**uropean **T**orus

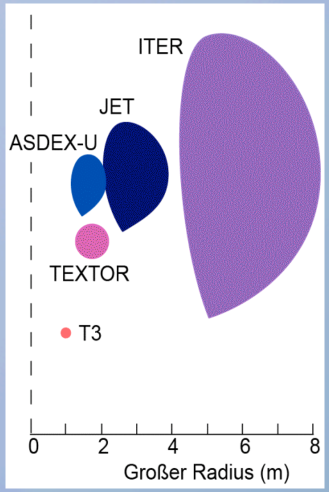
Simultaneously necessary parameter:

- temperature: 10 keV
- density: $1 \cdot 10^{20}$ /m^3
- energy confinement time: >1 s

Robust Confinement Scaling: Confidence about ITER



ITER reference scenario

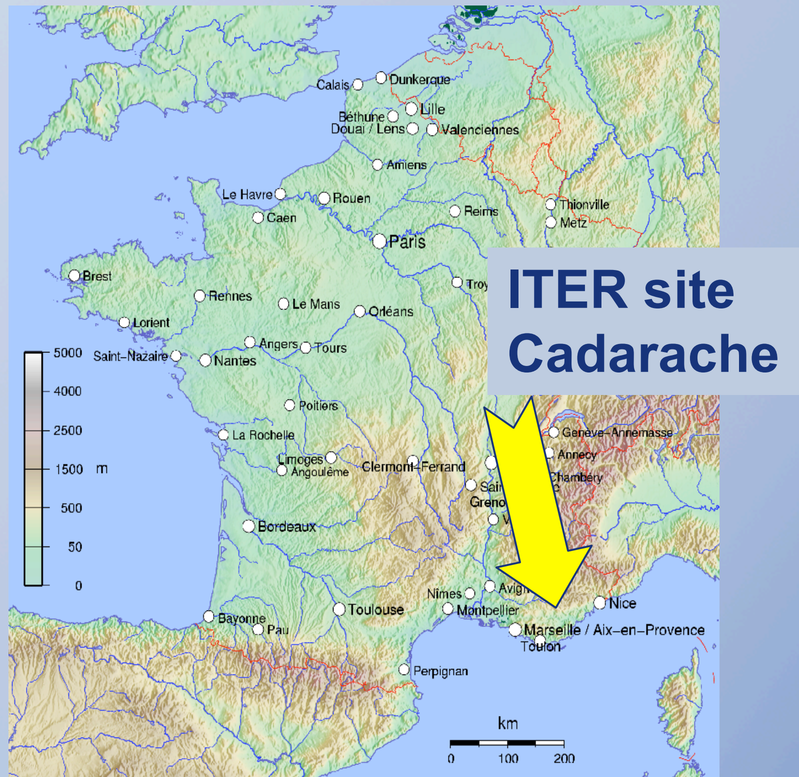


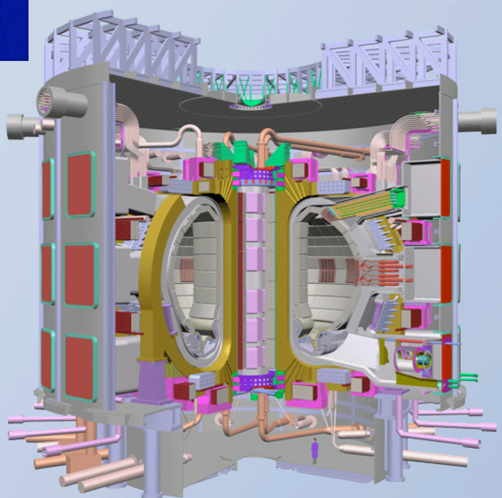
higher magnetic field
larger volume

Next Big Step:
First Burning Fusion Plasma in ITER

A Joint Project of EU, Japan, USA, Russia, South Korea, China, India

contributions „in kind“

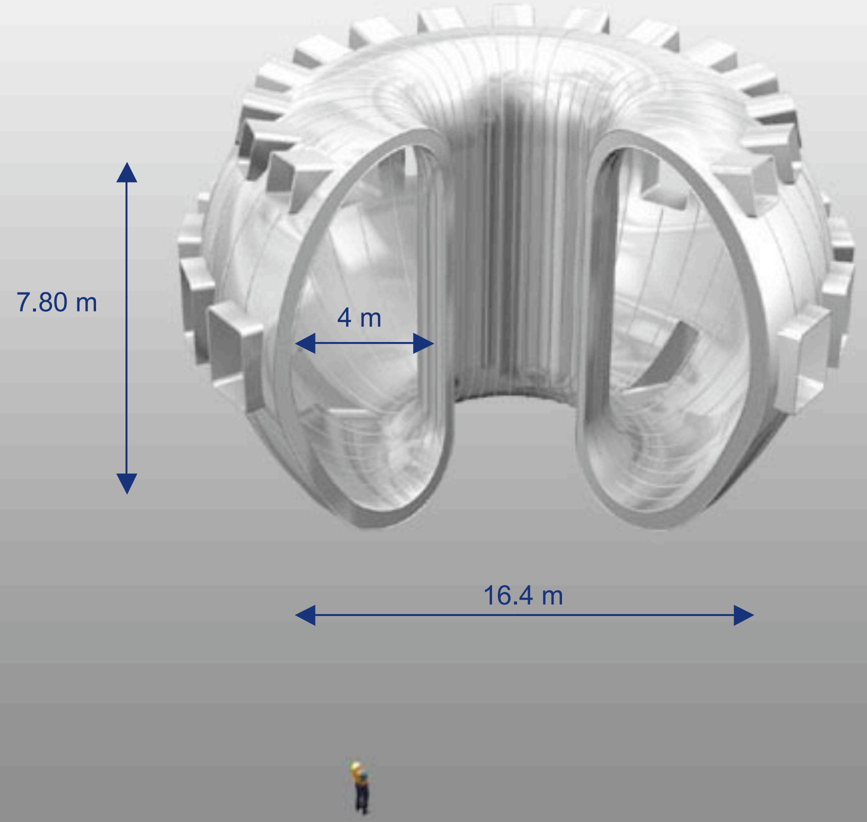




Investment ~ 5 Billion €

24. Mai 2006 Signature of ITER Implementing Agreement

Ratification until end of 2006 / early 2007

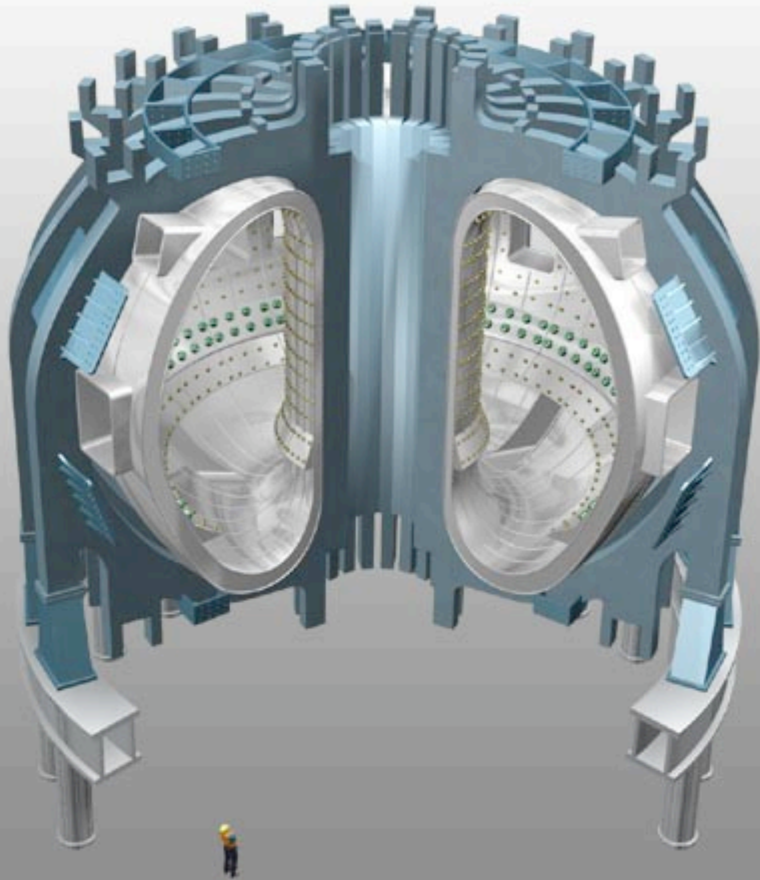


- Vessel
-
-
-
-
-
-

18 coils
superconducting

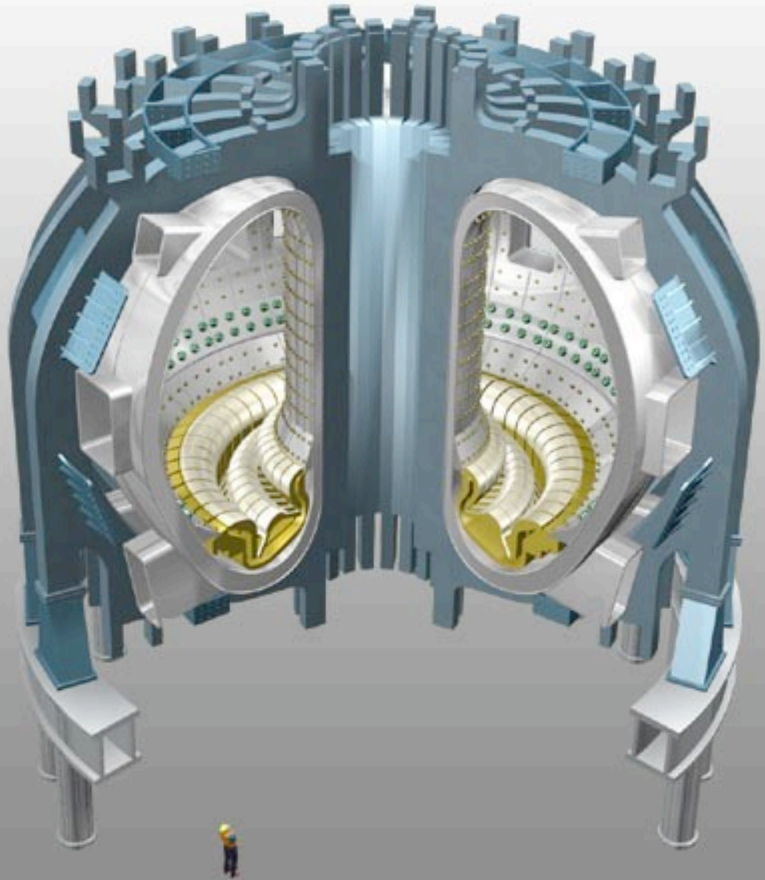
5.3 T

8 700 to

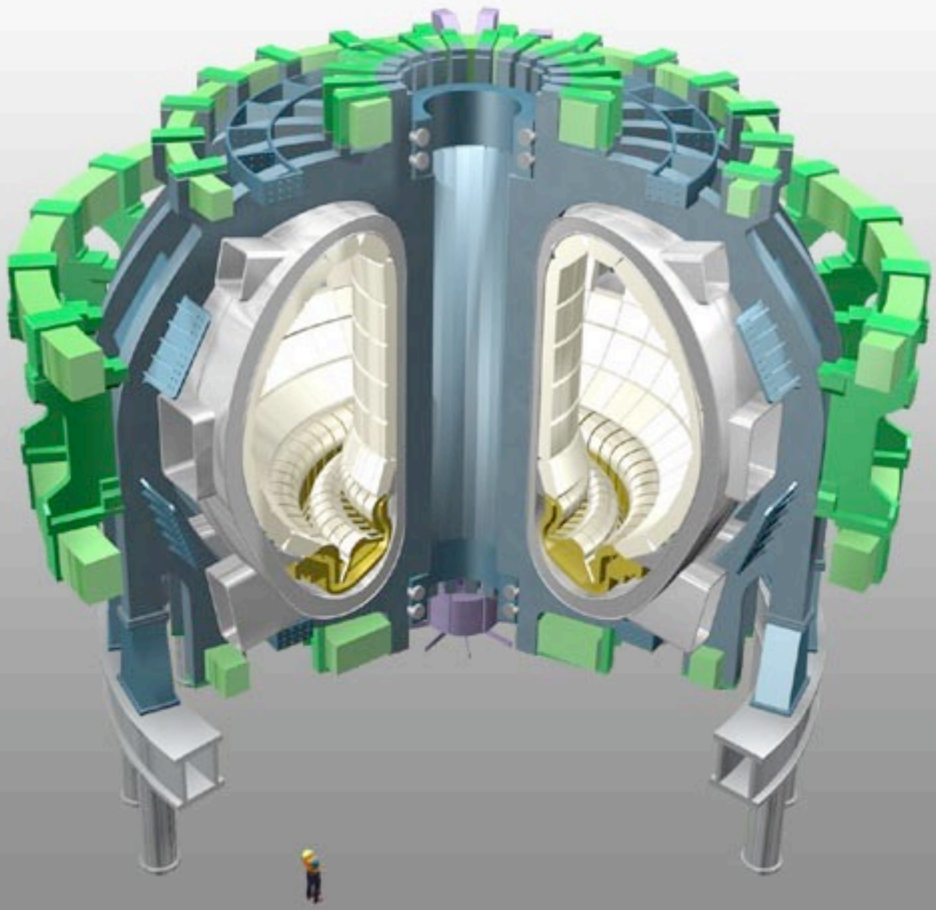


- Vessel
- Toroidal Coils
-
-
-
-
-

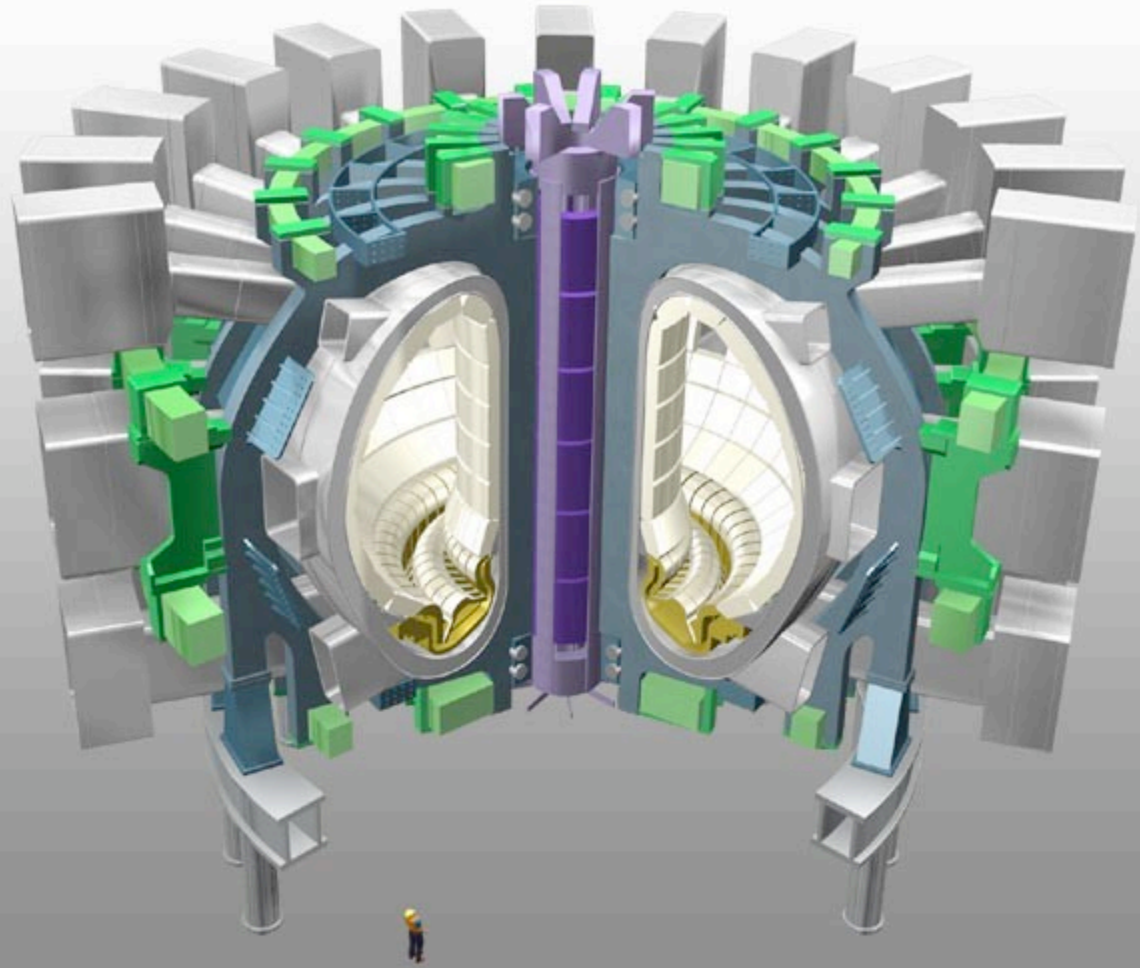
10 MW / m^2 on 8 m^2



- Vessel
- Toroidal Coils
- Divertor
-
-
-
-

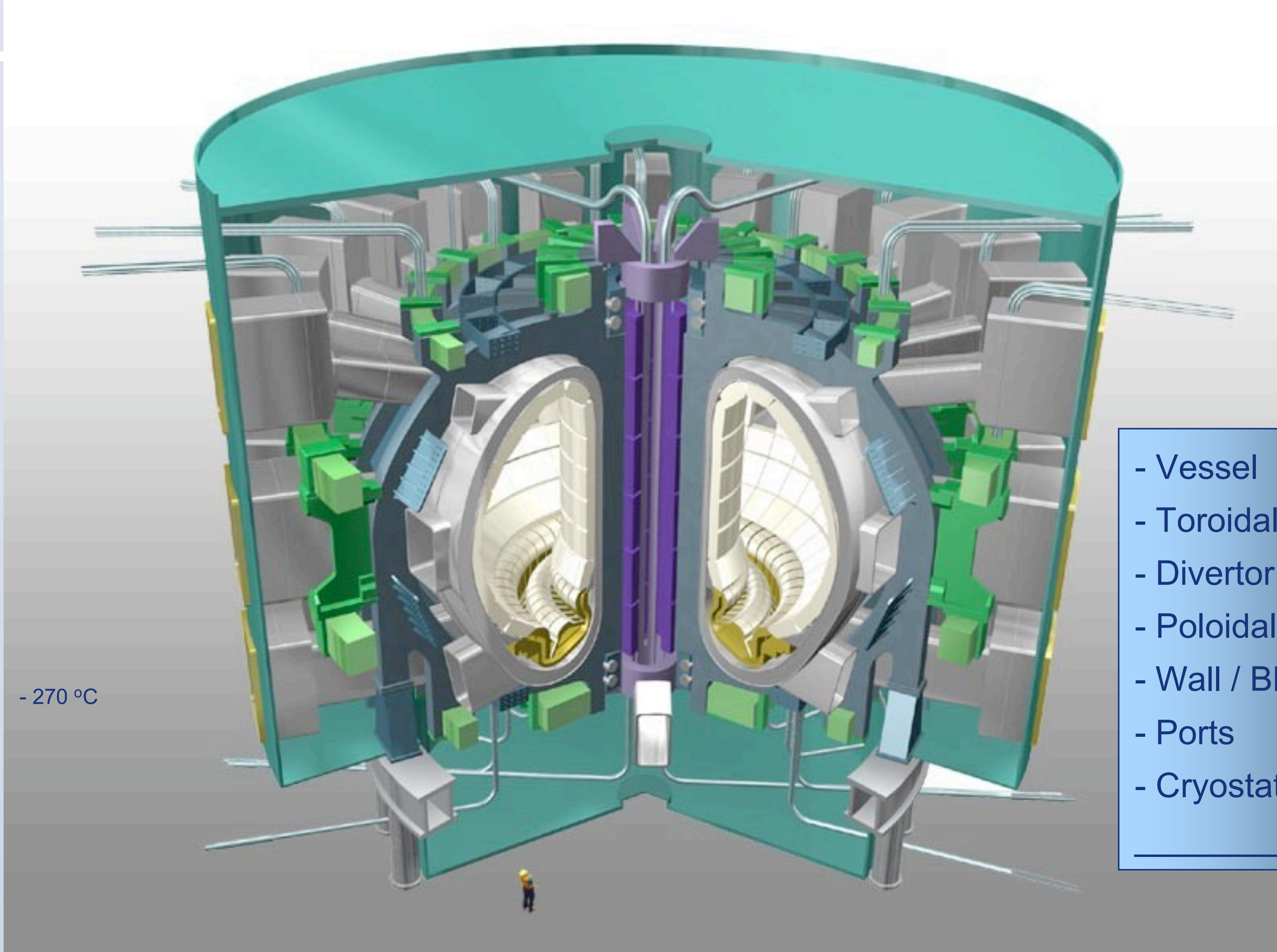


- Vessel
- Toroidal Coils
- Divertor
- Poloidal Coils
- Wall / Blanket
-
-

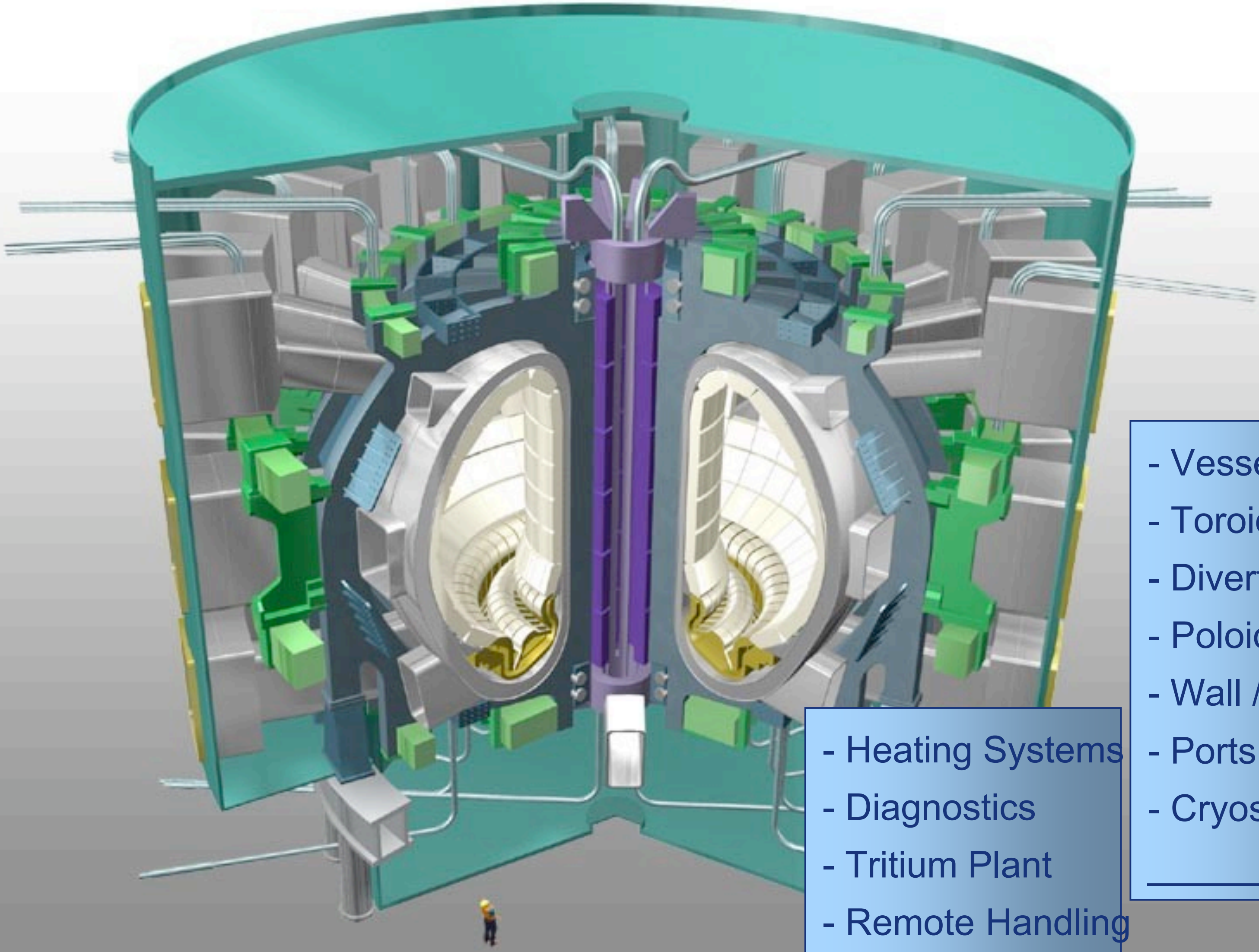


70 MW heating

- Vessel
- Toroidal Coils
- Divertor
- Poloidal Coils
- Wall / Blanket
- Ports
-



- Vessel
- Toroidal Coils
- Divertor
- Poloidal Coils
- Wall / Blanket
- Ports
- Cryostat



- Heating Systems
- Diagnostics
- Tritium Plant
- Remote Handling
- Control & Data

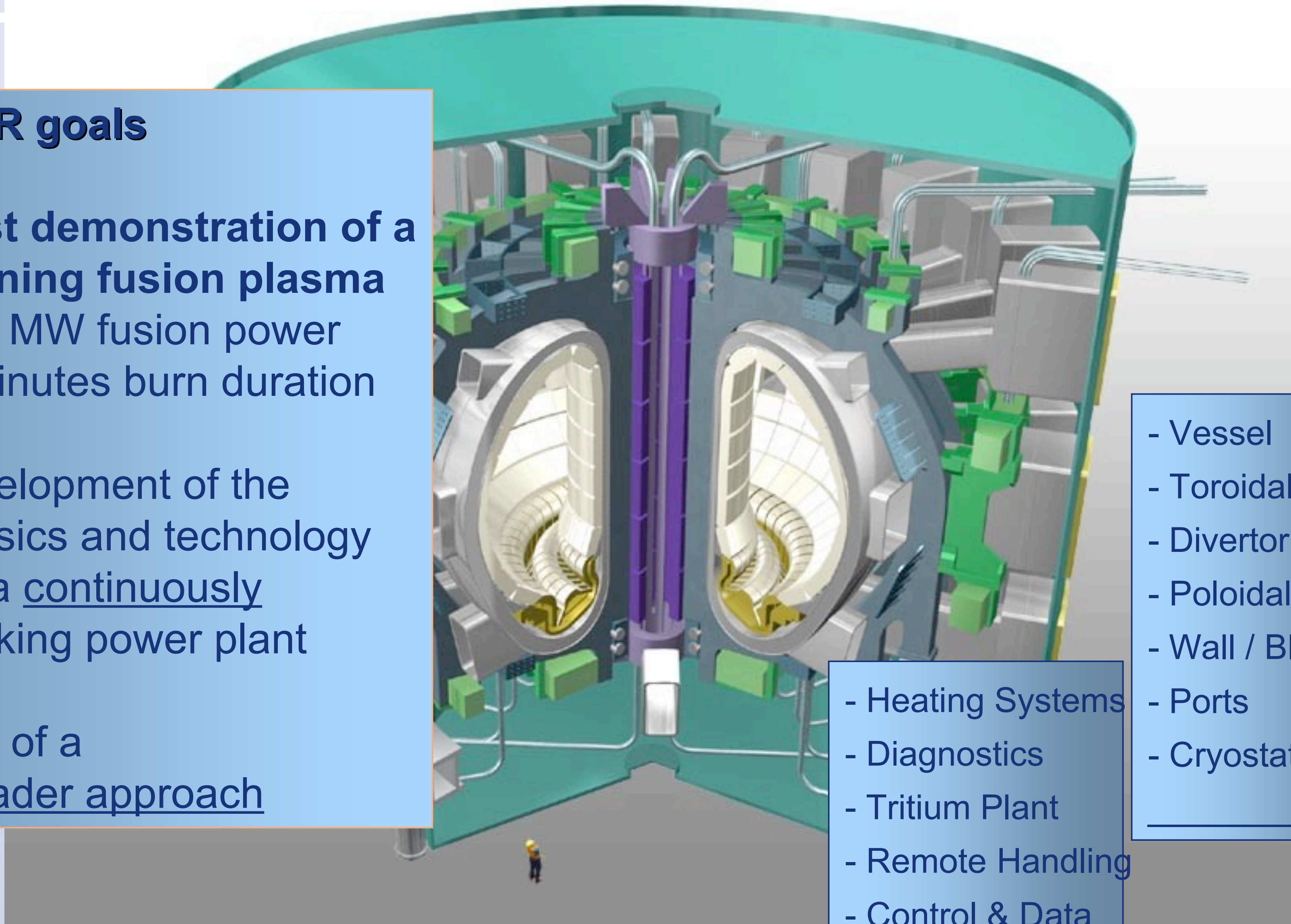
- Vessel
- Toroidal Coils
- Divertor
- Poloidal Coils
- Wall / Blanket
- Ports
- Cryostat

ITER goals

First demonstration of a burning fusion plasma
500 MW fusion power
8 minutes burn duration

development of the physics and technology for a continuously working power plant

part of a broader approach



- Vessel
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- Divertor
- Poloidal Coils
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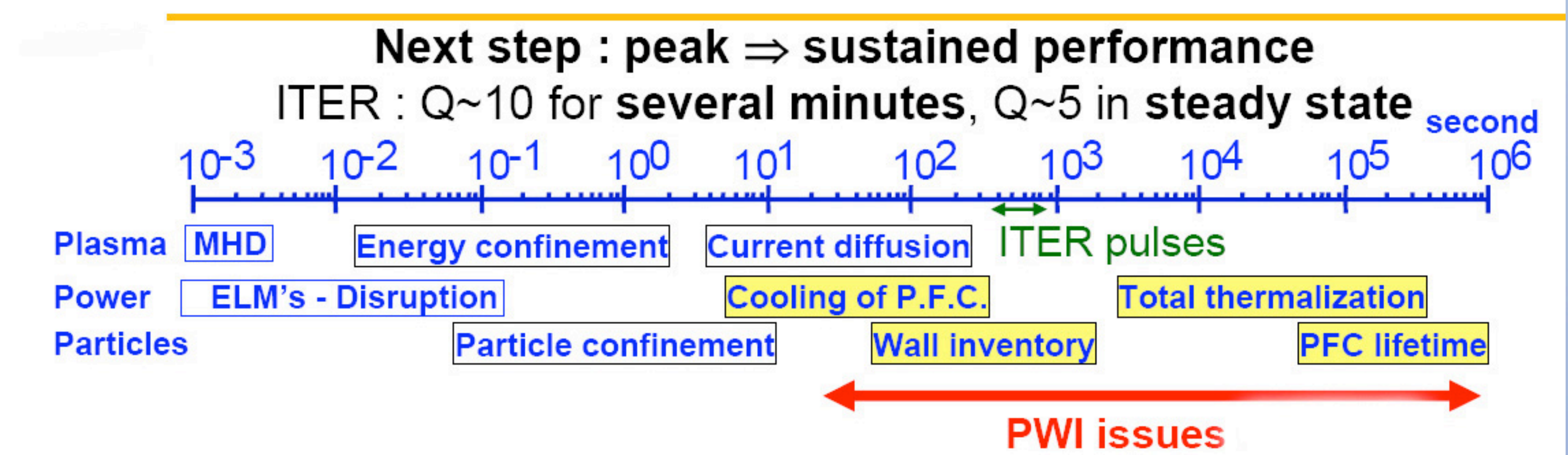
- Heating Systems
- Diagnostics
- Tritium Plant
- Remote Handling
- Control & Data

Long Pulse Operation – a New Challenge to PWI

E. Tsitrone, CEA



tokamaks/stellarators (e.g.): **TEXTOR JET TS ITER W7-X …. DEMO**

new linear devices (e.g.): **MAGNUM-PSI**

Long Pulse Operation – a New Challenge to PWI

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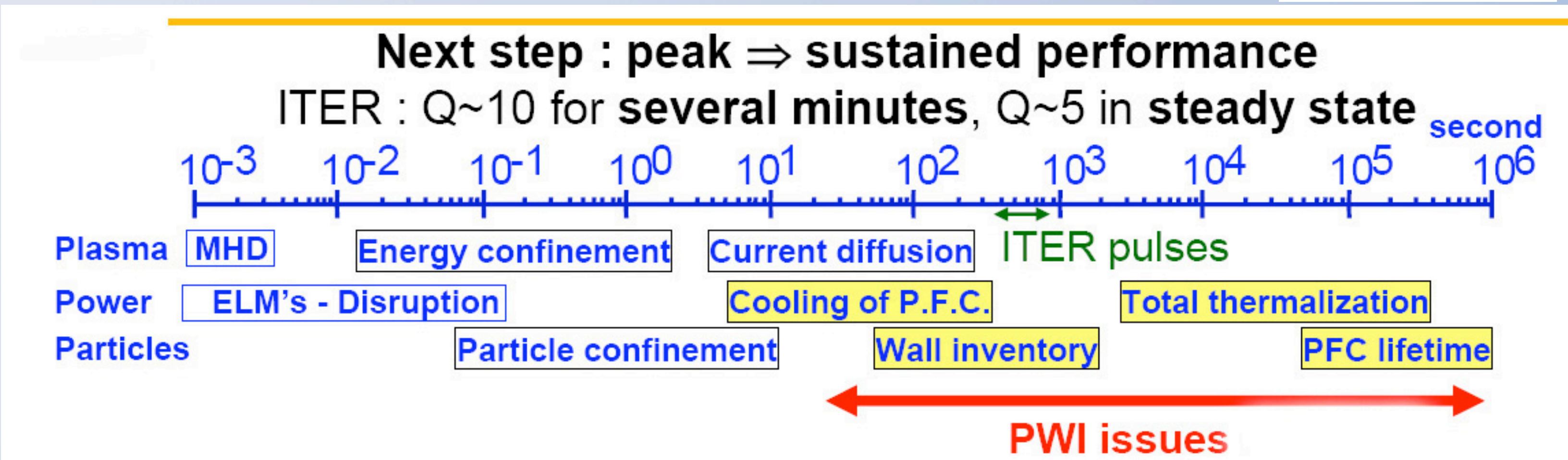
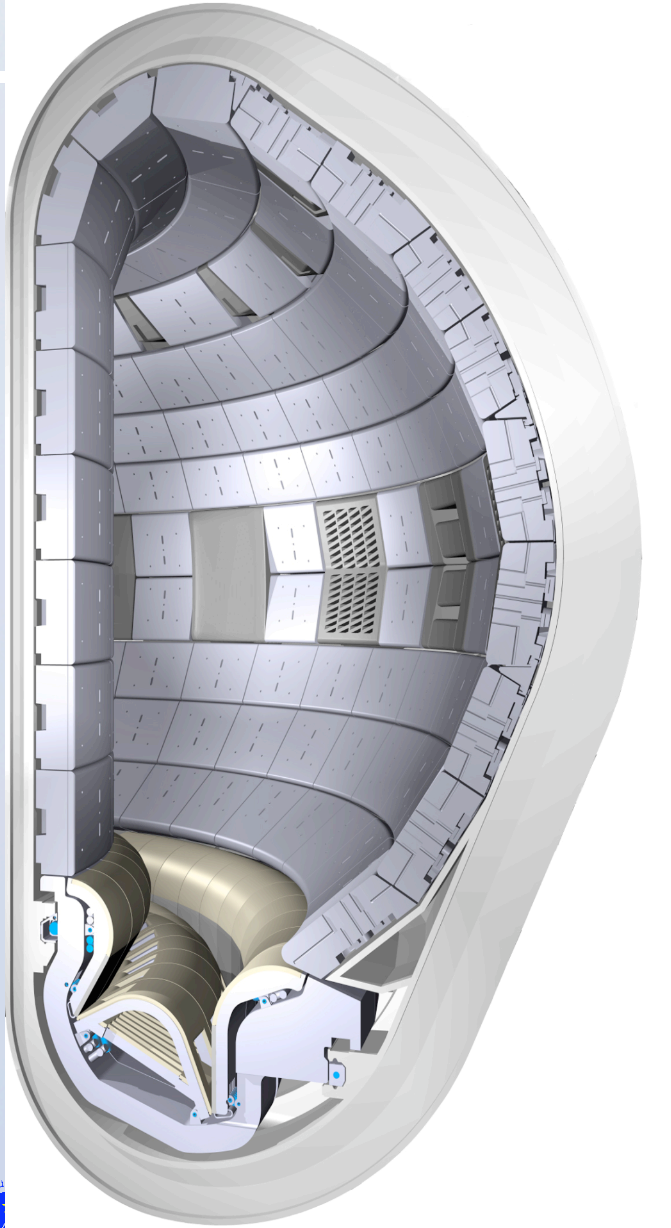


tokamaks/stellarators (e.g.): TEXTOR JET TS ITER W7-X .... DEMO
new linear devices (e.g.): MAGNUM-PSI

Positive energy balance in ITER most likely based on tokamak size
but
Pulse duration and number of discharges depends on PWI

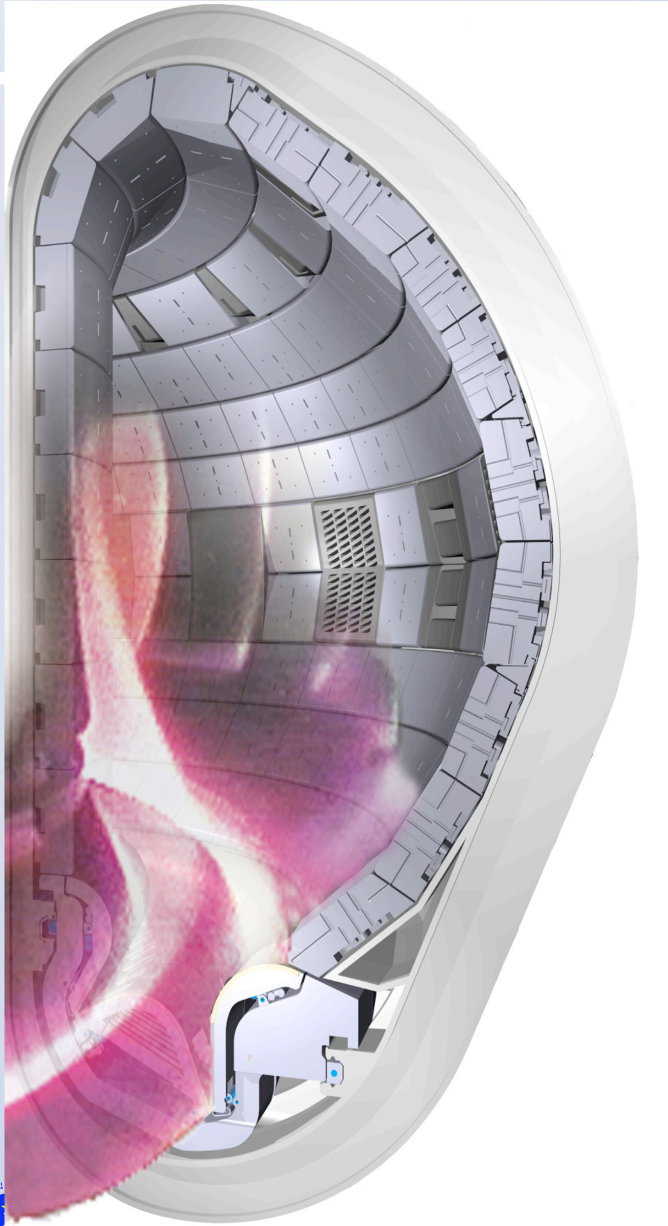


Plasma-Wall Interaction

ITER

Magnetic confinement allows
- 500 MW fusion power
- 8 minutes burn time
- Q = 10

Plasma-Wall Interaction required to
- exhaust 20% of fusion power
- remove impurities (Helium)



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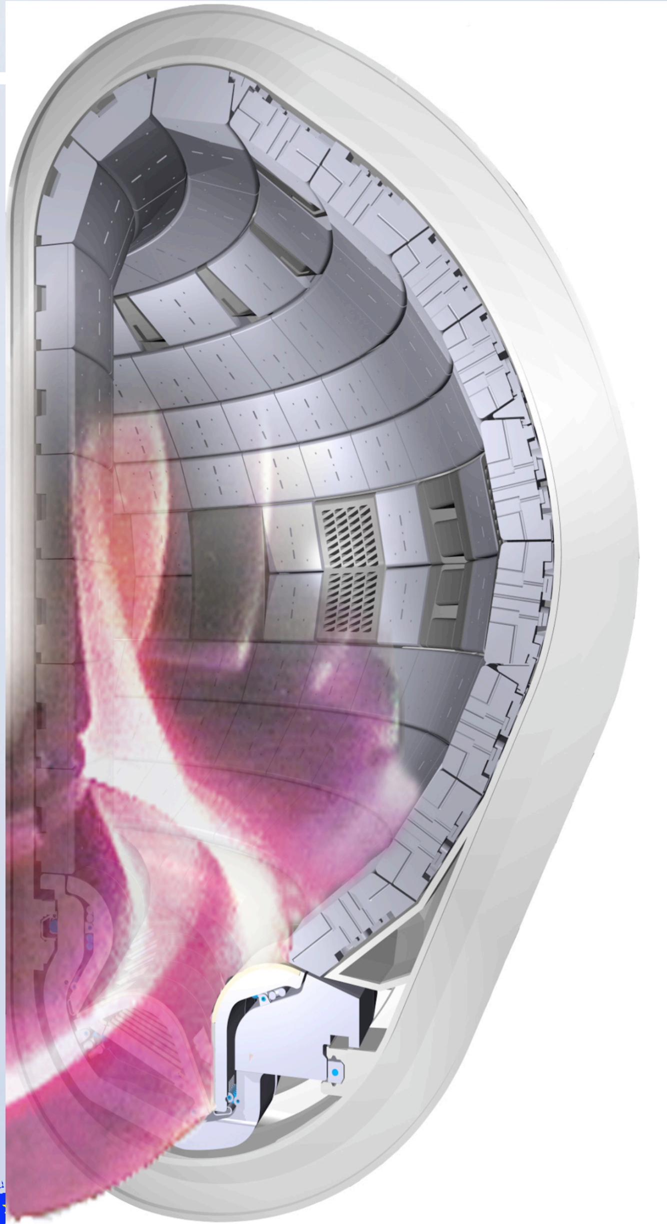


Plasma-Wall Interaction

ITER

Magnetic confinement allows
- 500 MW fusion power
- 8 minutes burn time
- Q = 10

Plasma-Wall Interaction required to
- exhaust 20% of fusion power
- remove impurities (Helium)

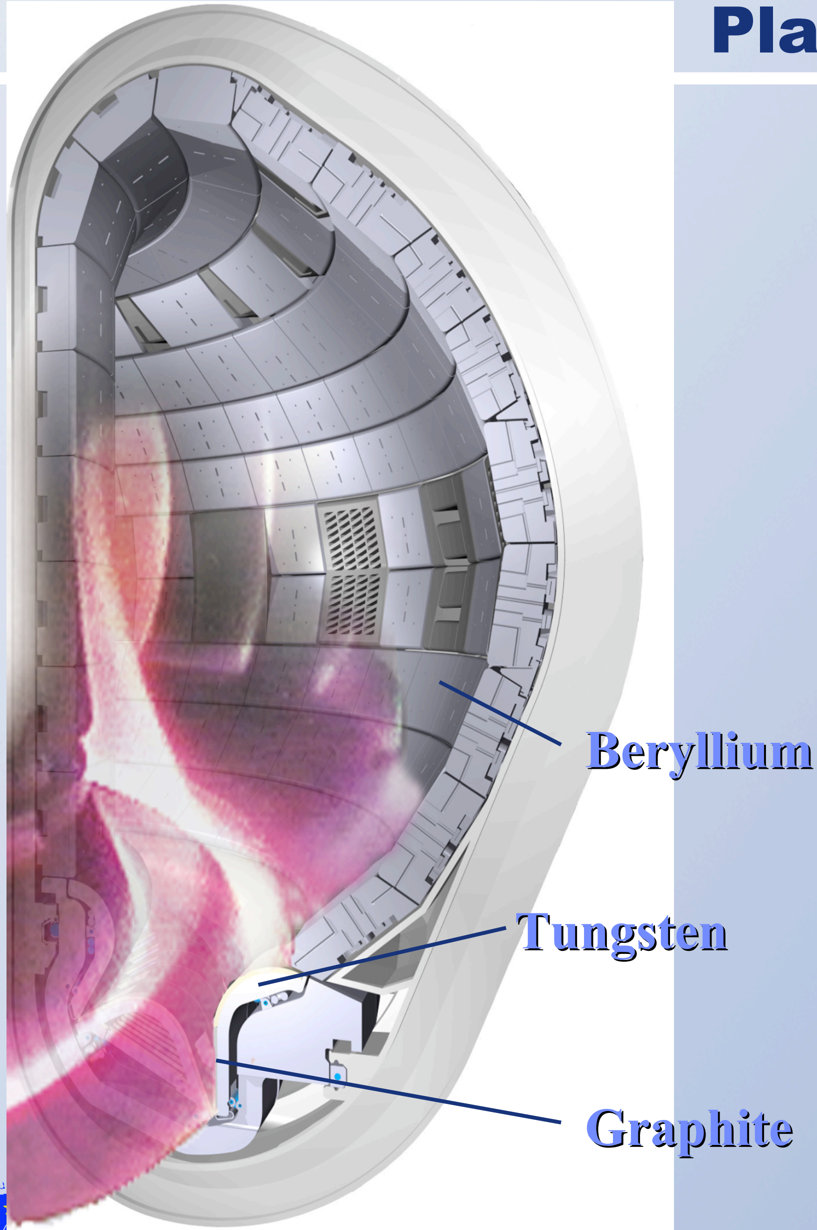
Wall Surface and Power Densities

680 m^2 main chamber: < **0.22** MW/m^2
8 m^2 divertor peak load: < **10** MW/m^2
10^{24} D/T-atoms m^{-2} s^{-1}

Divertor design allows main goals of ITER

Plasma-Wall Interaction

materials for plasma facing components



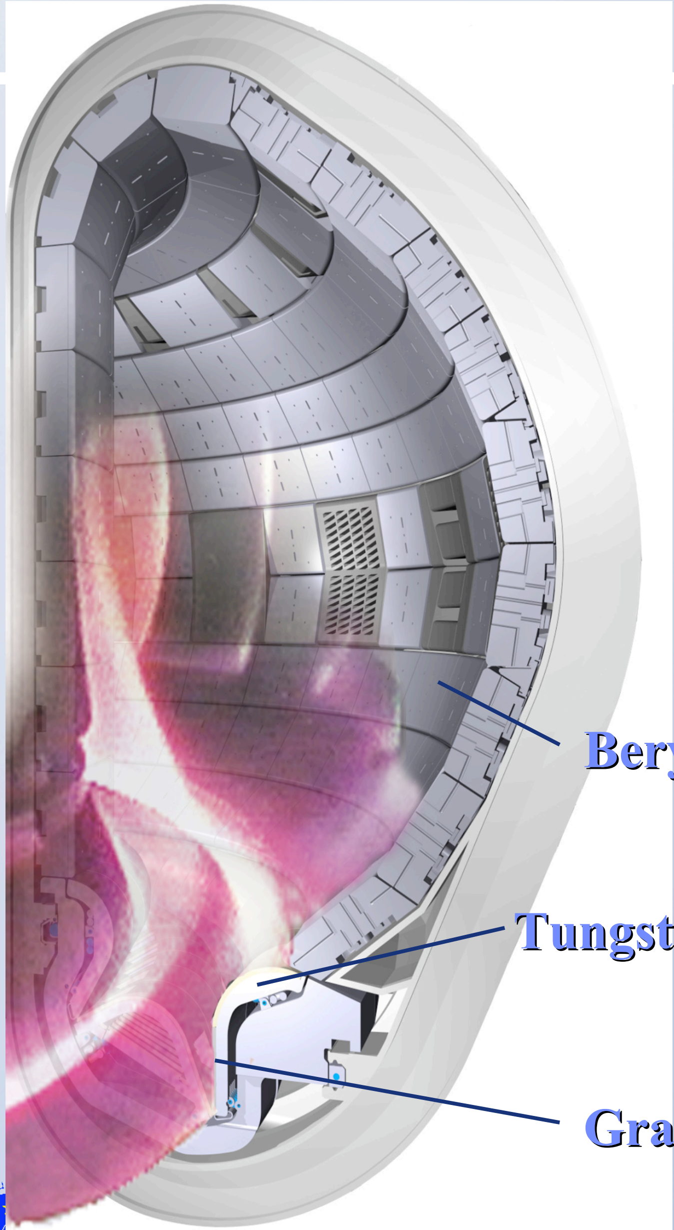
Main criteria:

no radiation from centre
(low-Z) …. **Be**

tolerance with excess loads
(graphite) ….. **C**

low erosion and
high melting point
(high-Z) ….. **W**

Plasma-Wall Interaction



materials for plasma facing components

Main criteria:

no radiation from centre (low-Z) …. **Be**

tolerance with excess loads (graphit…

low eros… high me… (high-Z)…

risks about availability

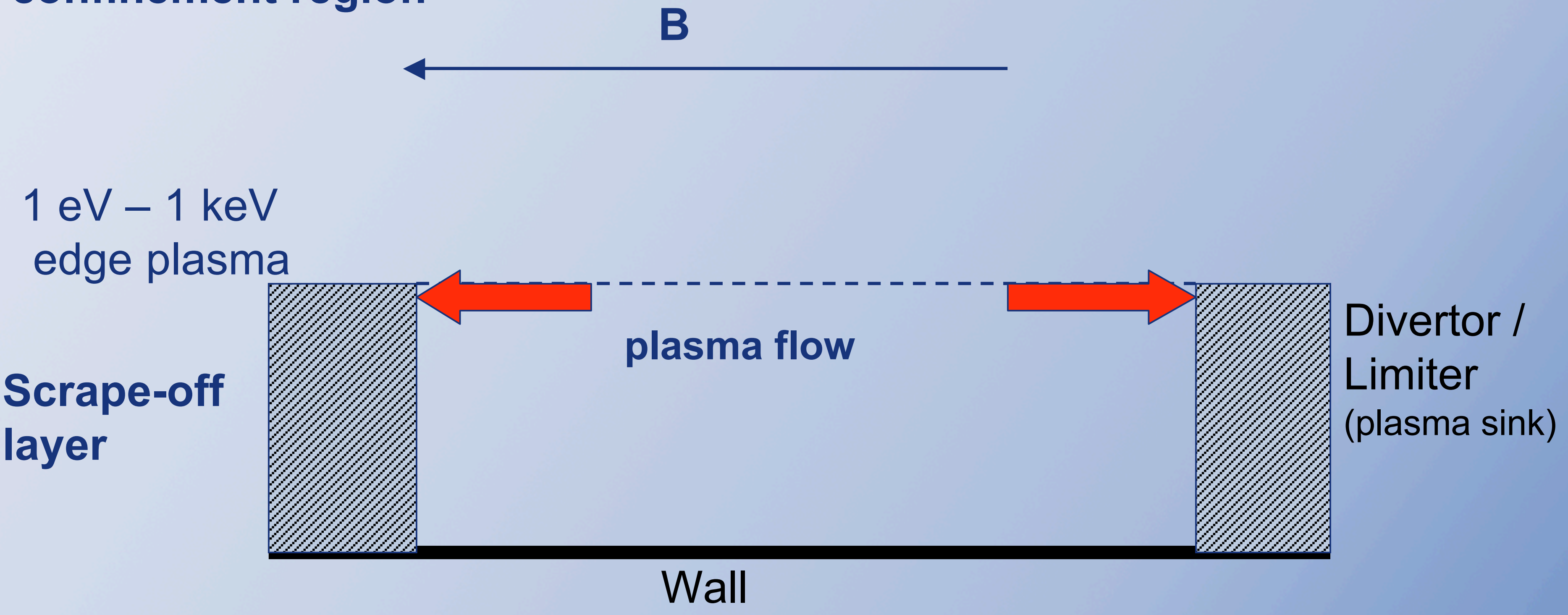
tritium-retention in deposited carbon layers

life time of divertor plates

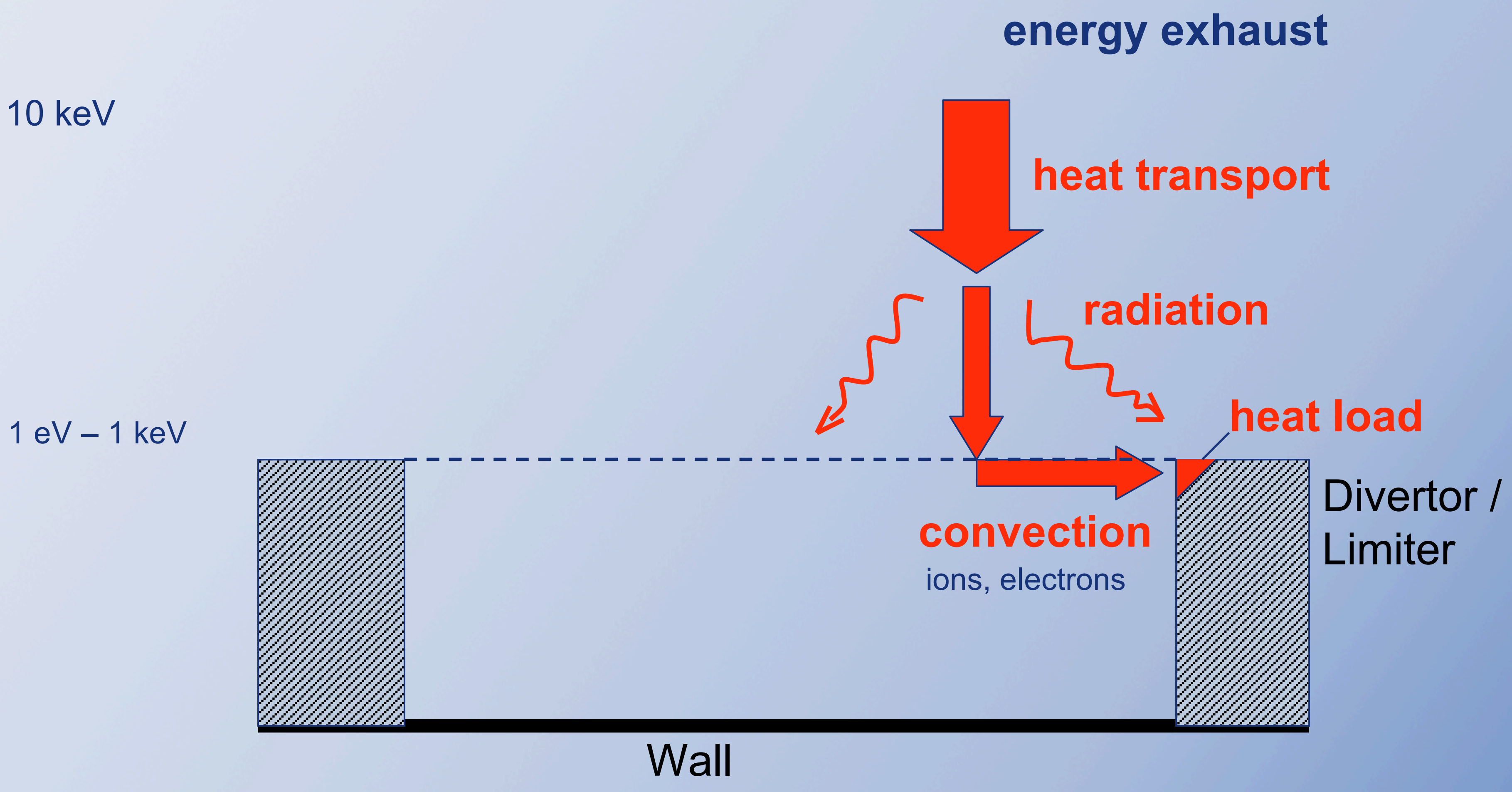
risk minimisation by better understanding of PWI processes

Processes of Plasma-Wall Interaction

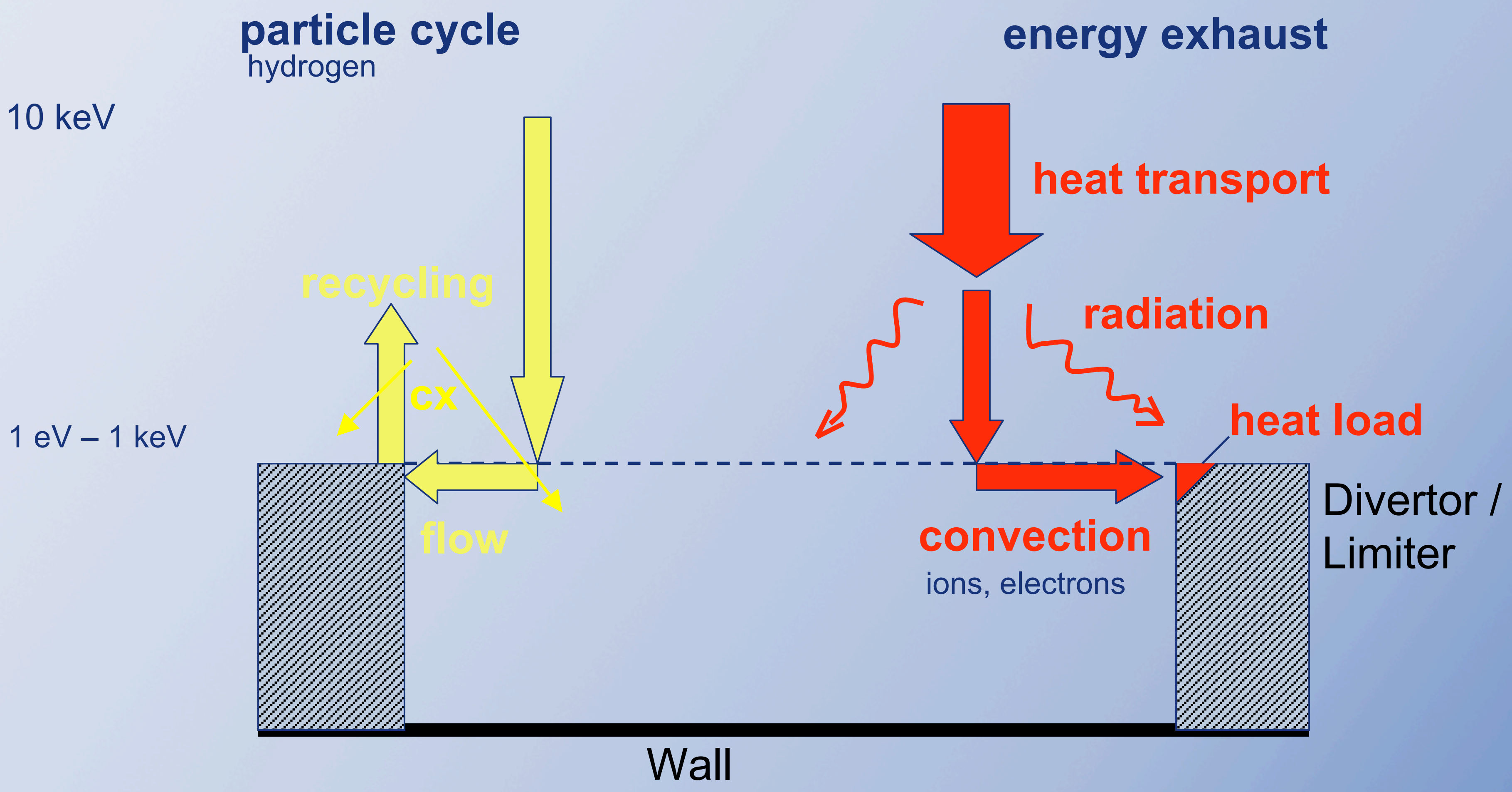
10 keV central plasma
confinement region



Processes of Plasma-Wall Interaction



Processes of Plasma-Wall Interaction



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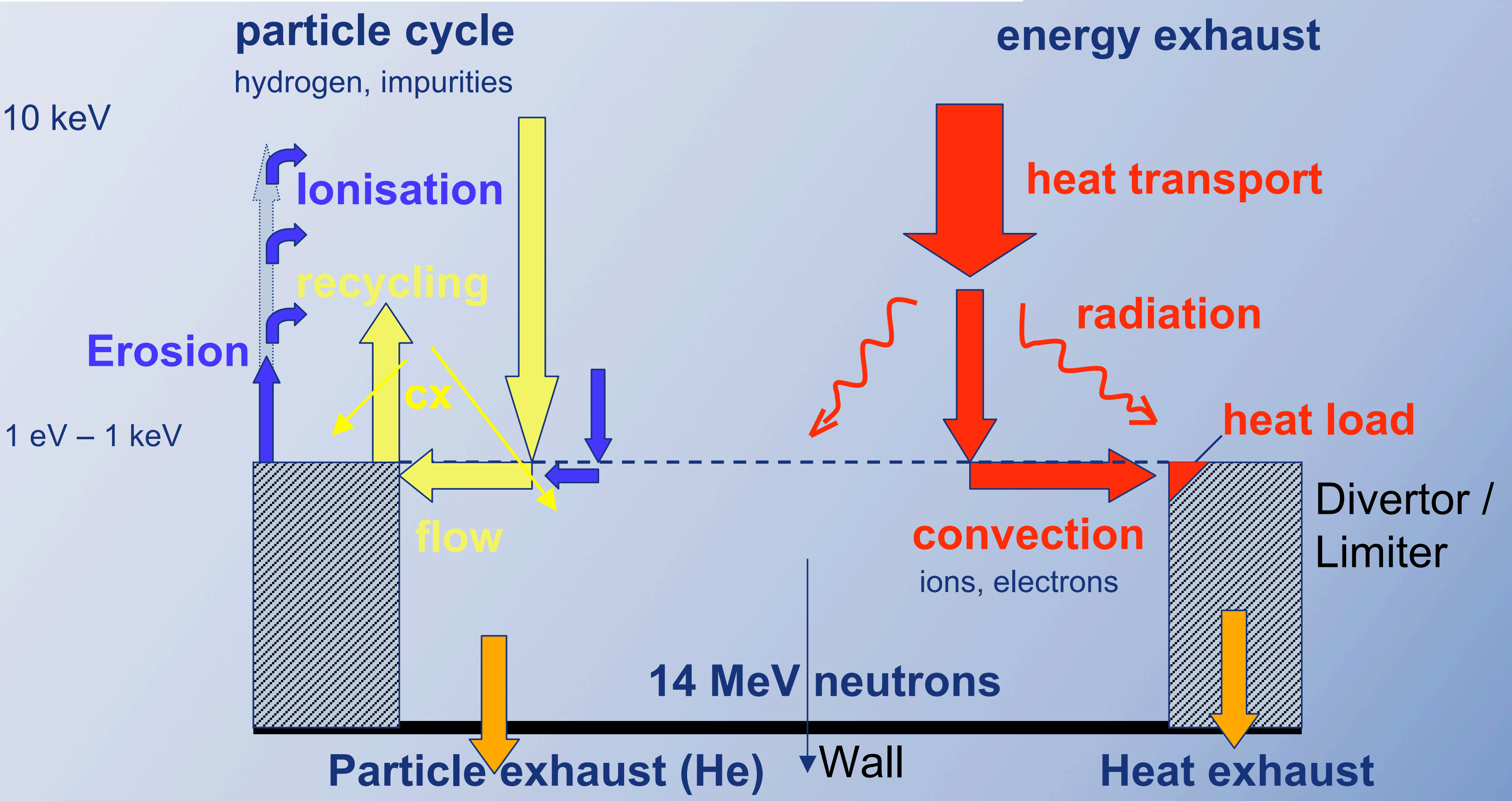


erosion processes - radiation level - plasma parameters - erosion
a non-linear system

Plasma wall interaction tasks

Plasma facing components (PFC)

Material selection:

- Dependence on heat load (melting, sublimation)
- Electro-mechanical properties and their temperature dependence
- Erosion properties (physical and chemical sputter yield)
- Deposition properties (co-deposition, mixed materials)
- Tritium removal
- Dependence on neutron radiation

materials research & engineering of PFCs

Plasma edge:

Optimisation of plasma parameter

- Radiation cooling
- Reduction of transient effects (ELM's, VDE's, disruption)
- Increase of interaction area (ergodisation)
- Influence on transport properties (turbulence, rotation)

Plasma edge physics

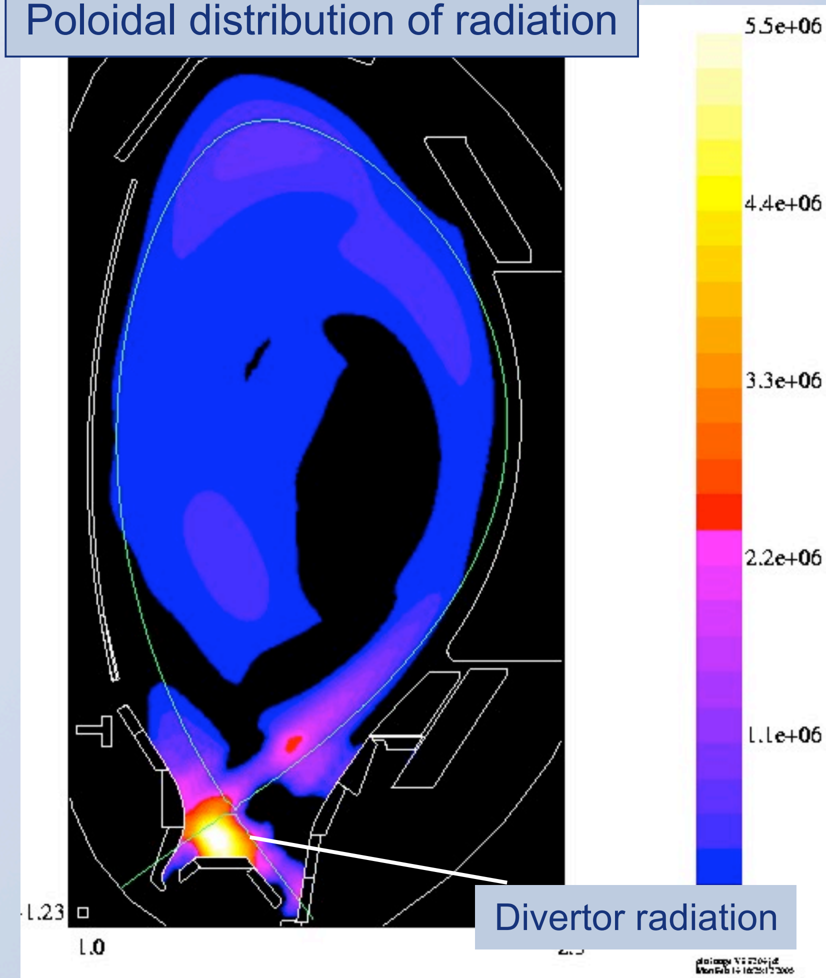
Plasma wall interaction tasks

Diagnostic development:

- **Tritium retention monitor**
  (laser induced desorption,…)
- **erosion and deposition monitor**
  (quartz micro balance, Speckle interferometer, laser ablation, probes,…)
- **flux measurement**
  quantitative spectroscopy (conversion factors S/XB, D/XB)
- **plasma edge parameter** (passive measurement-line of sight)
  spectroscopy, line intensity ratio
- **plasma edge parameter** (active measurement-local)
  atomic beams-BES (collisional-radiative model), -CXRS (line broadening)
  Thomson scattering,..
- **magnetic field, electric field**
  Motional Stark effect, heavy ion beam

Radiation Cooling inside a Divertor

Poloidal distribution of radiation



high radiation level(>70%) from eroded carbon

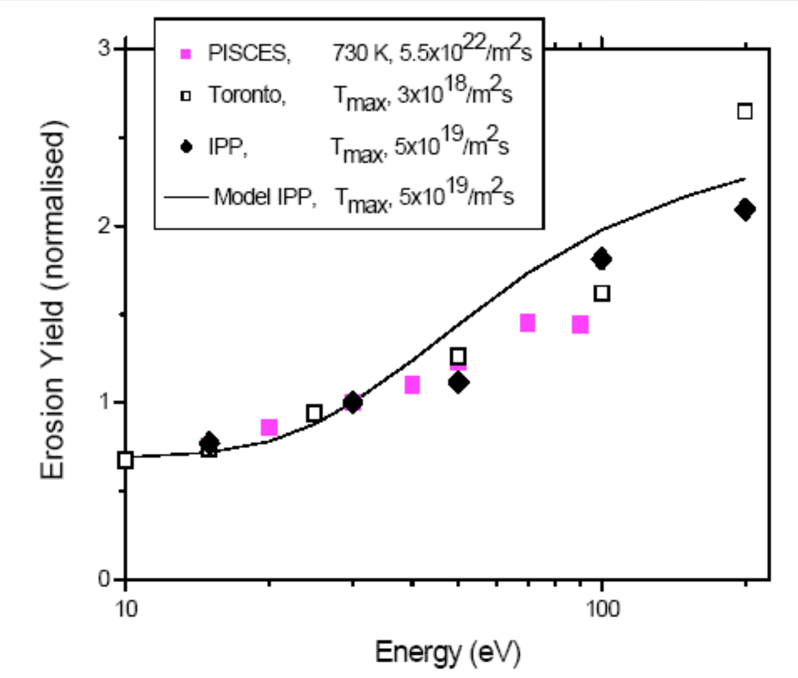
self adjusting to variation of heating power or injection of argon

a self-protection mechanisms of the divertor

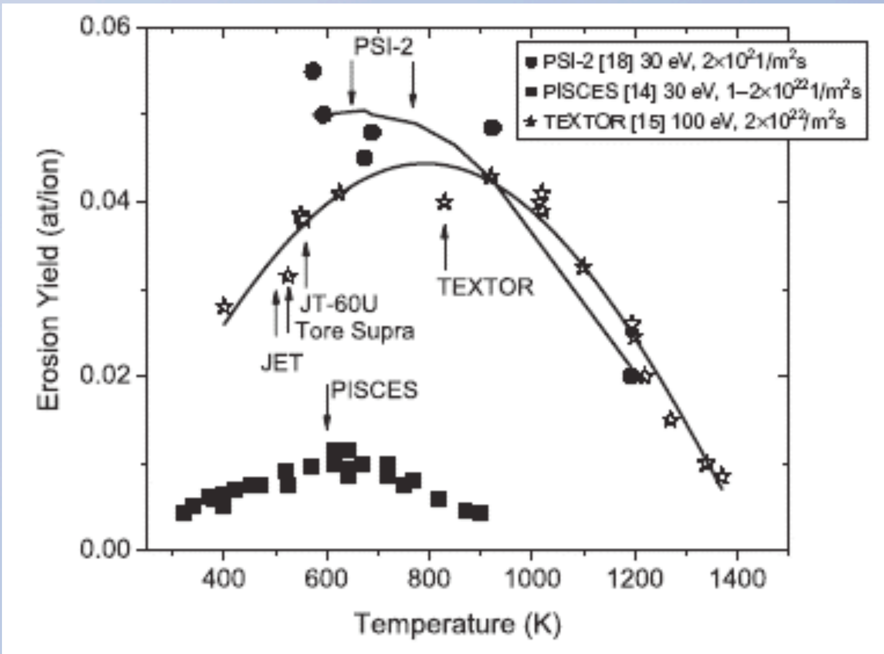
example from ASDEX-upgrade
A. Kallenbach

Chemical erosion yield: $Y=\Gamma_{CxHy}/\Gamma_{ion}$

Energy dependence of Y



Surface temperature dependence of Y



Roth et al.
NF **44** (2004) L21-L25

Ion induced desorption of hydrocarbon radicals

$Y_{chem,low\ \phi}=Y_{surf}+Y_{therm}(1+DY_{Dam})$ ← isotope dependence in D

thermal erosion yield enhanced by radiation damage

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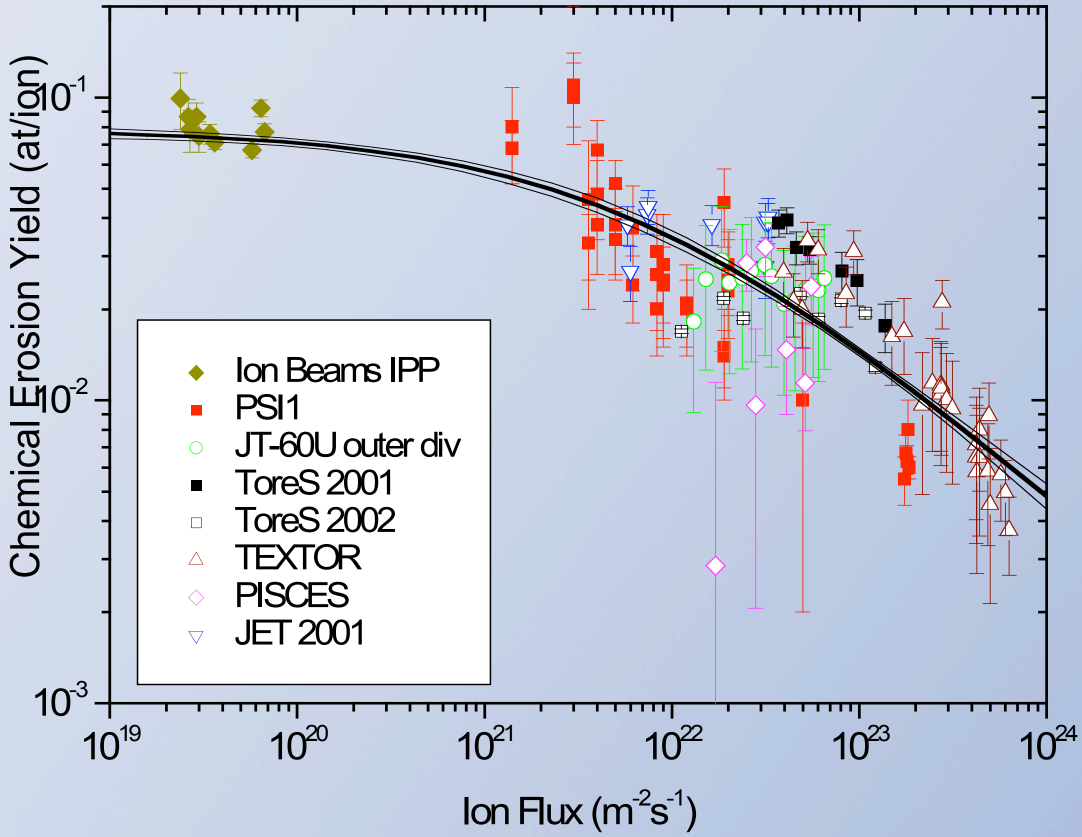
Chemical erosion yield: $Y=\Gamma_{CxHy}/\Gamma_{ion}$

Flux dependence of Y



Description of Y as function of
- Incident ion energy (E)
- Surface temperature (T)
- Ion flux (ϕ)

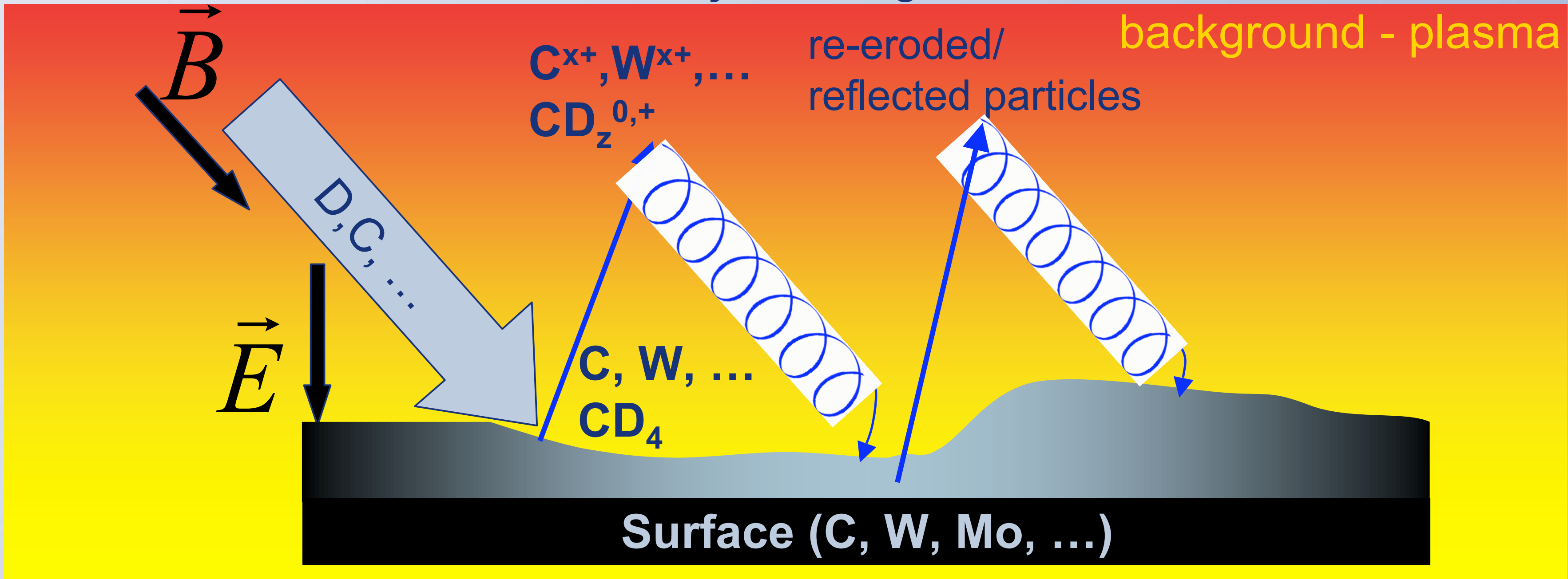
$$Y(E,T,\phi)=\frac{Y_{low}(E,T)}{1+\left[\frac{\phi}{6*10^{21}}\right]^{0.54}}$$

Roth et al.
NF 2004

- In-situ calibration: mass/optical spectroscopy
- Temperature normalisation to T_{max}
- Ion energy normalisation to 30 eV

Erosion, Impurity Transport and Deposition

Lifetime Predictions by Modelling: Monte Carlo Code ERO



plasma-wall interaction:
- **physical sputtering / reflection**
- **chemical erosion (CD_4)**
- **deposition from background plasma**
- **re-deposition of eroded species**

local particle transport:
- **ionisation, dissoziation**
- **friction (Fokker-Planck), thermal force**
- **Lorentz force**
- **diffusion**

Trilateral Euregio Cluster TEC

A. Kirschner, S. Droste, D. Borodin

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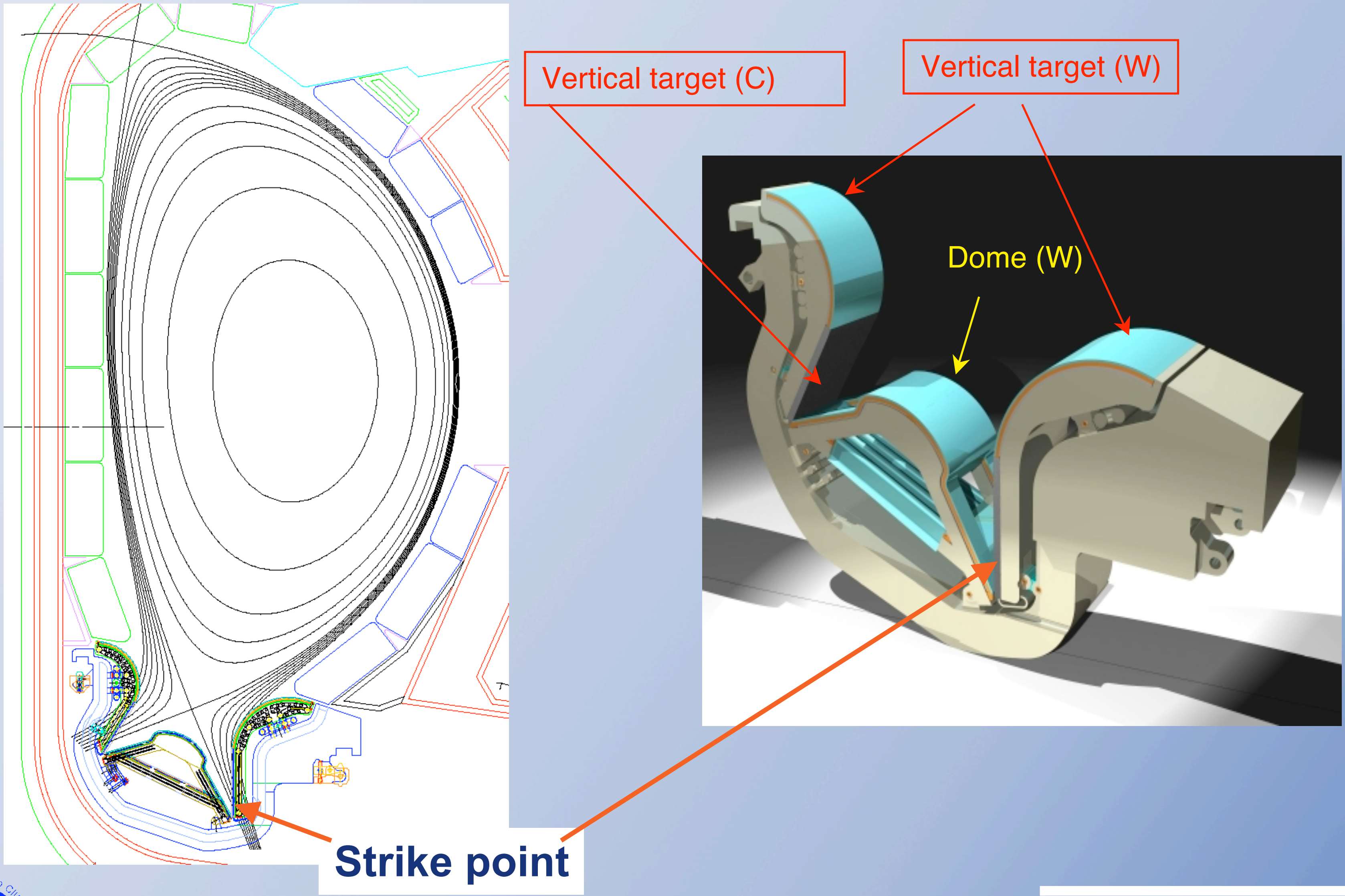
ITER Divertor Design



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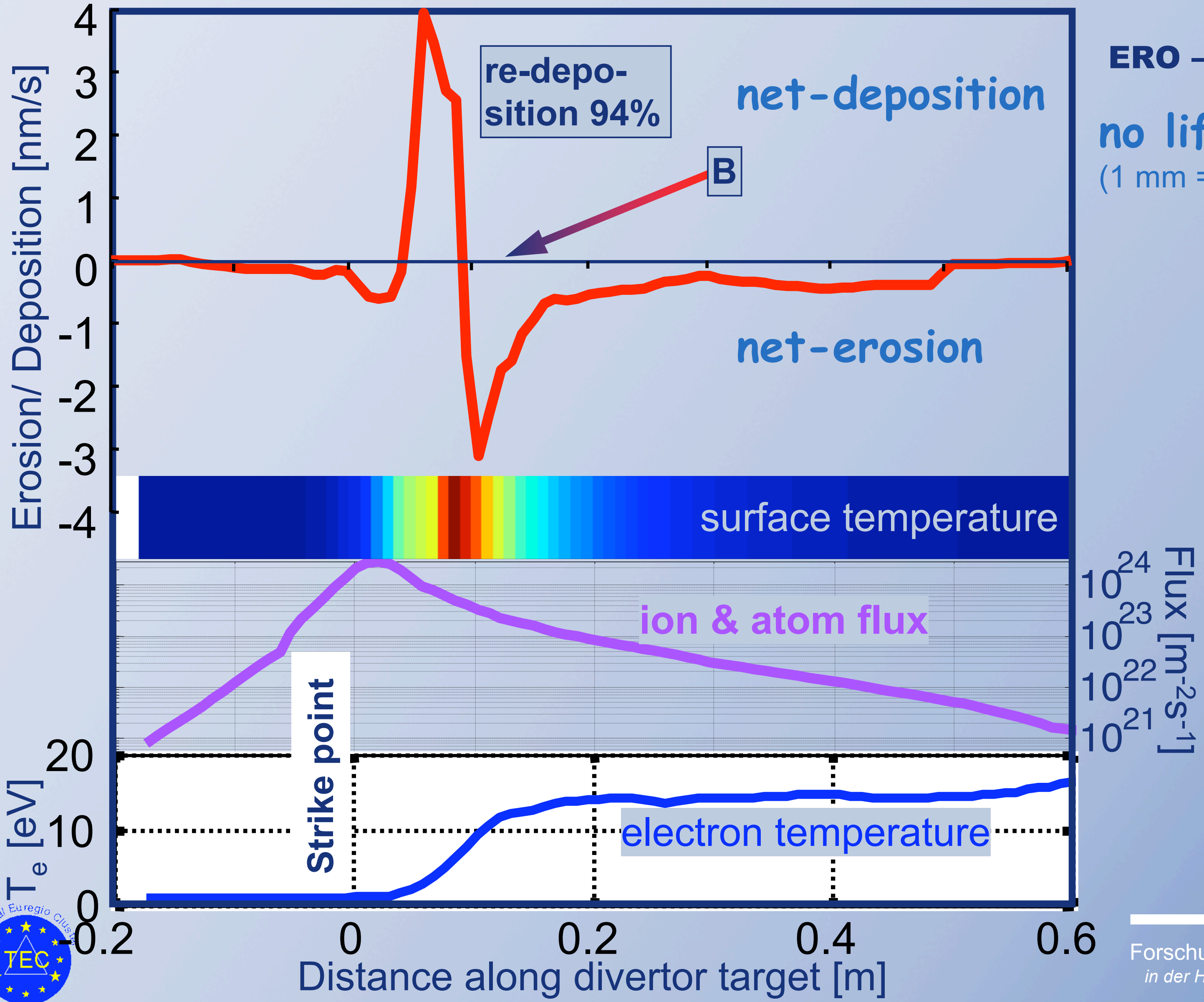
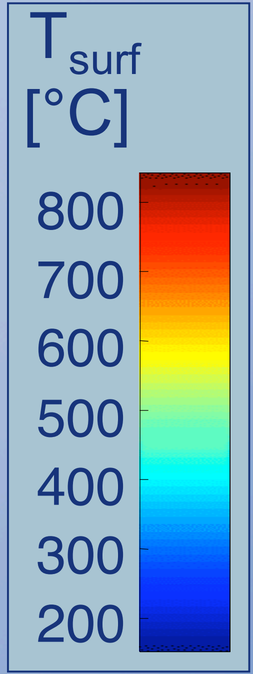
Erosion of ITER Graphite Target Plates



ERO – Modelling for ITER

no lifetime problem

(1 mm = 500 ITER discharges)



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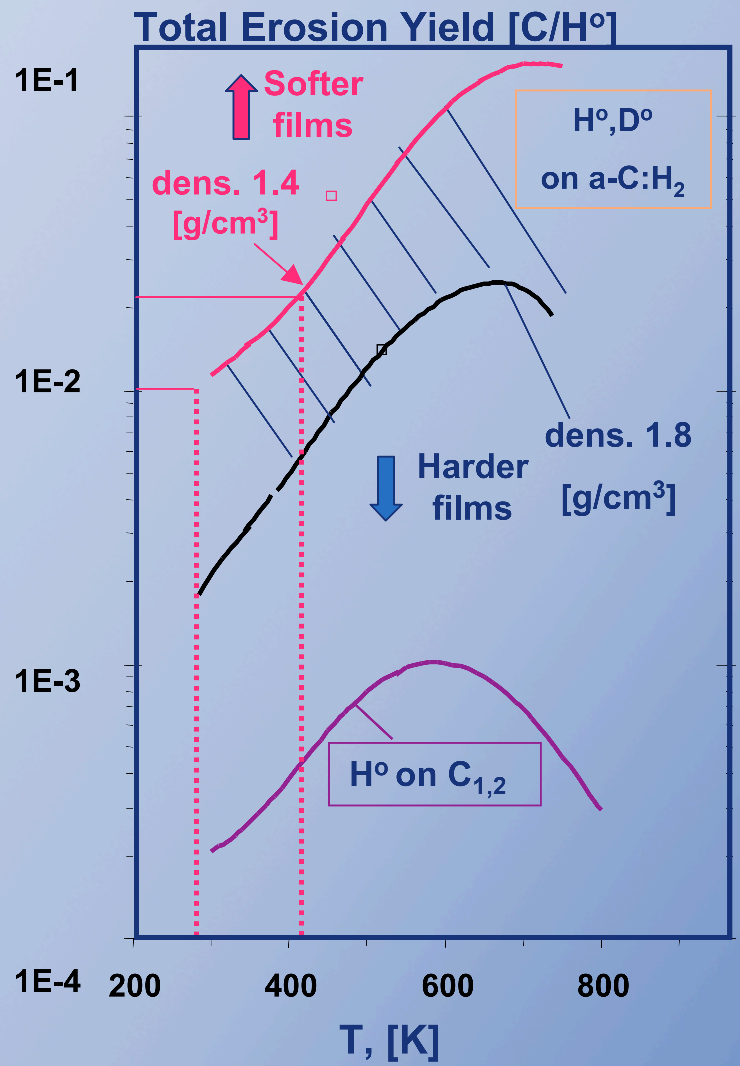
Different erosion yields of “soft” and “hard” carbon layers

Example: erosion of graphite and
a-C:H films by thermal H_0 and D_0 atoms

- chemical erosion shows strong temperature dependence
- soft layers form at low plasma temperature
- enhanced erosion of soft layers

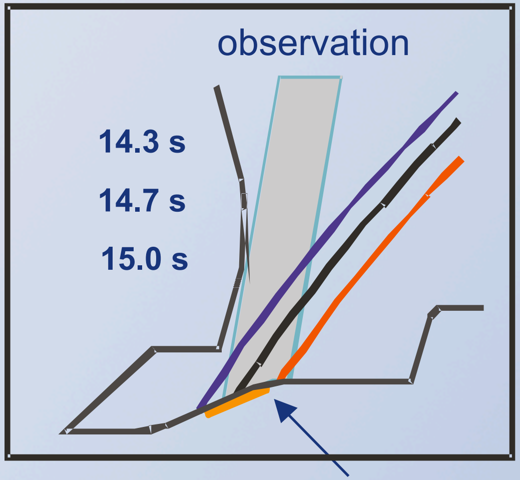


1. J.W. Davis et al, JNM 155-157(1988), 234;

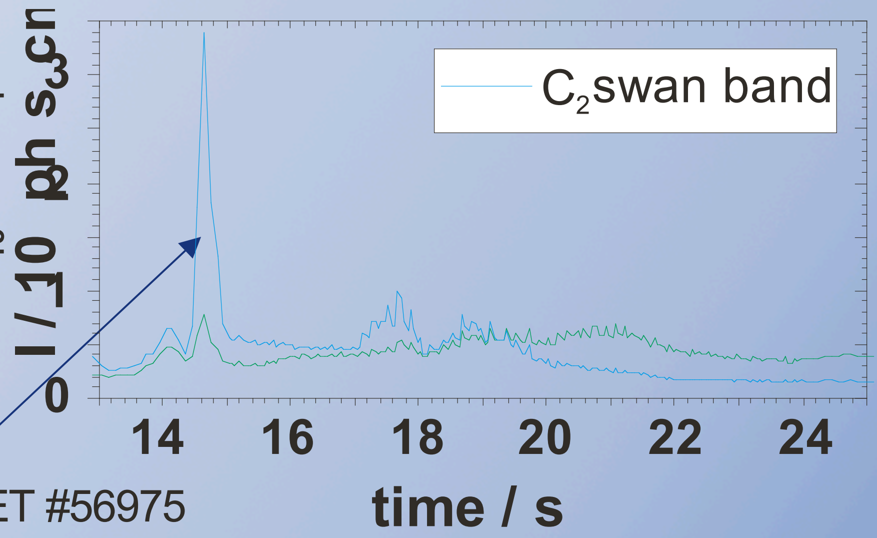
2. E. Vietzke et al. Fus. Technol. 15 (1989), 108.

Hydrocarbon Spectroscopy in JET

- removal of soft hydrocarbon layer identified by C_2 light emission and deposition monitor
- local removal within one discharge with fixed strike-point

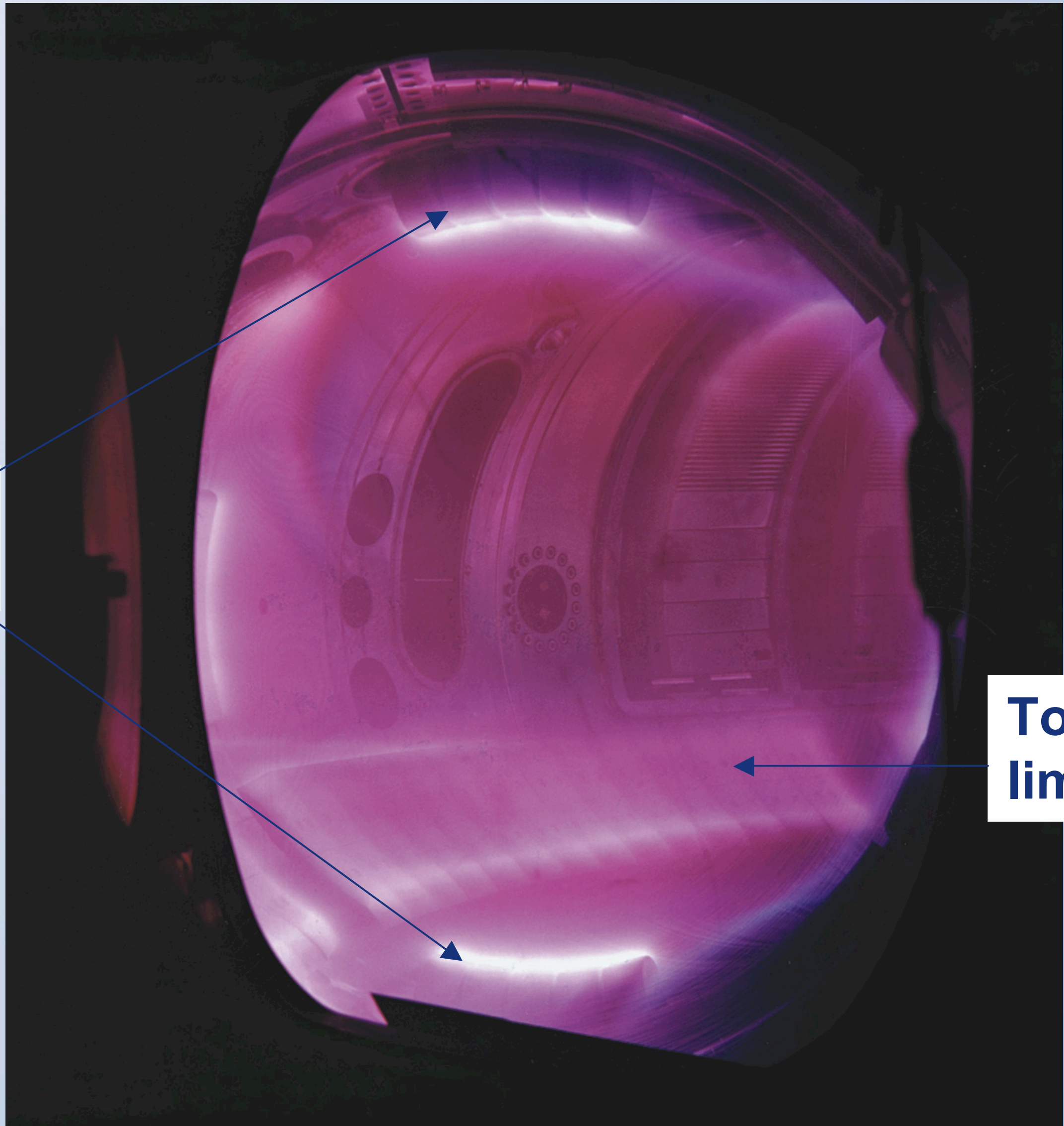


JET divertor
operation with different strike point locations



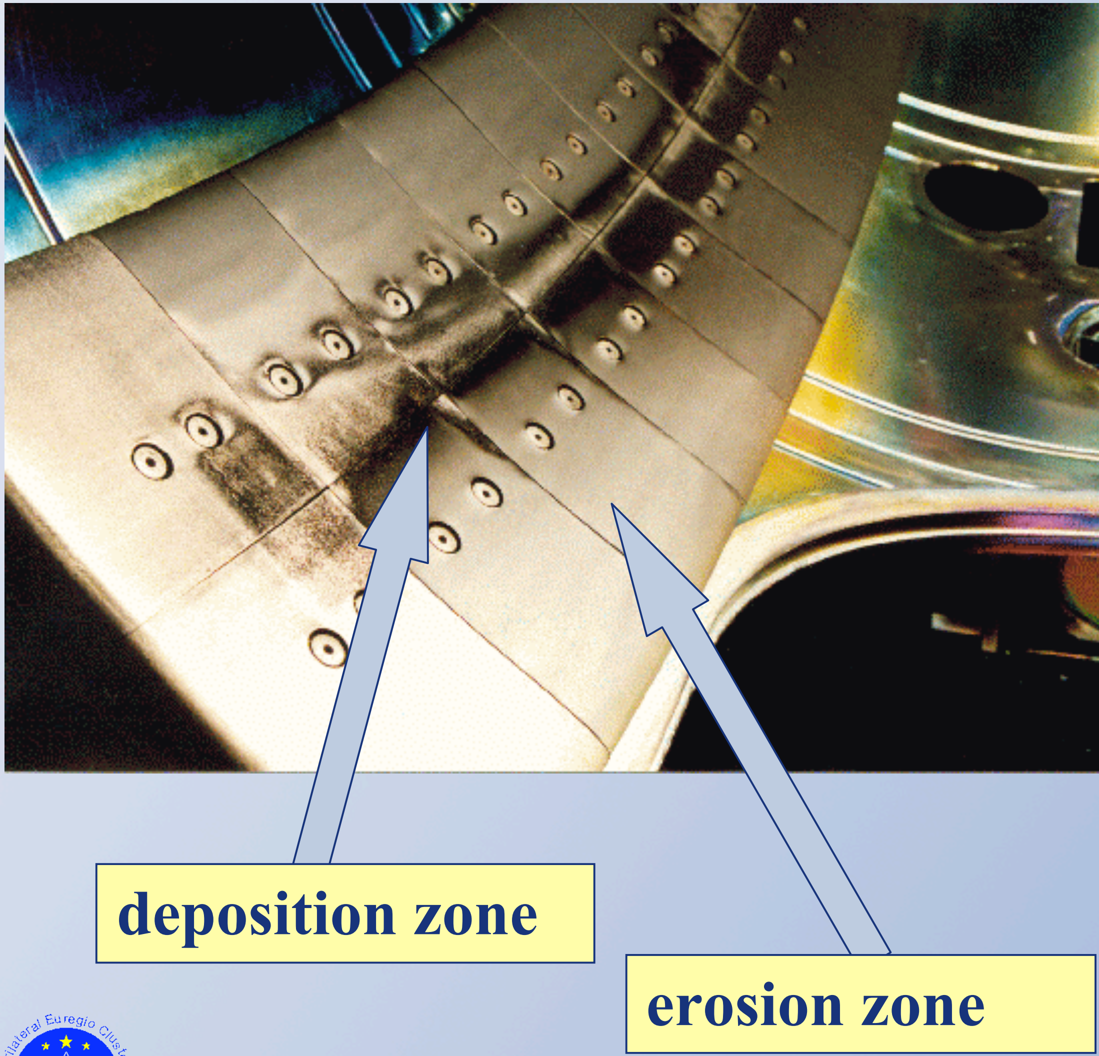
Brezinsek et al.
JNM 2005

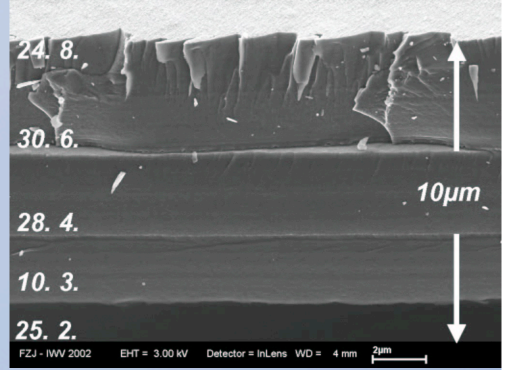
TEXTOR Plasma



Tritium Retention in Carbon Layers

TEXTOR





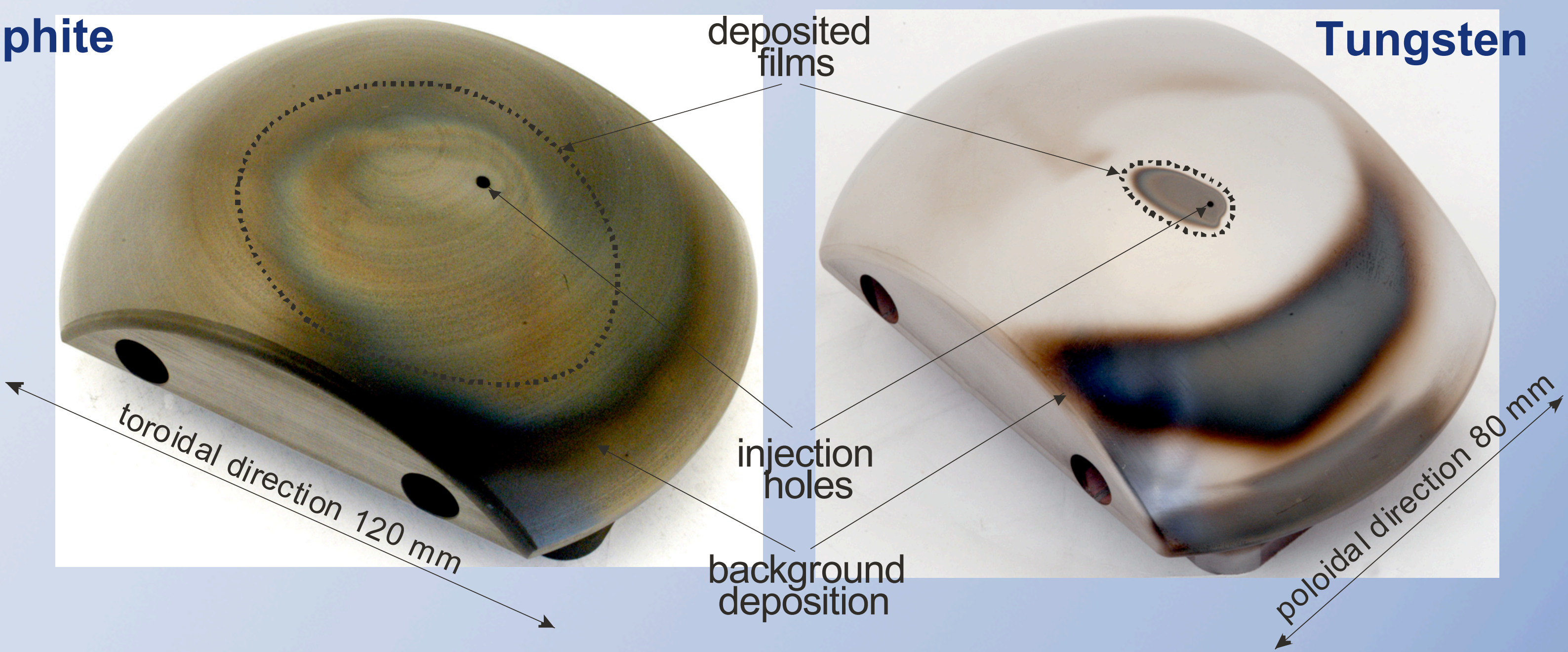
tritium retention by co-deposition;
has to be limited
(safety restrictions)
worst case: reduced availability for ITER

soft and hard layers
soft amorphous layers at E_i<30eV
concentration of hydrogen up to100%

Local Sticking of Carbon on Tungsten and Graphite

$^{13}CH_4$ injection through limiters

TEXTOR



$5.5\cdot10^{20}$ $^{13}CH_4$ puffed
$2.0\cdot10^{19}$ ^{13}C deposited

deposition efficiency 4%

$5.7\cdot10^{20}$ $^{13}CH_4$ puffed
$1.5\cdot10^{18}$ ^{13}C deposited

deposition efficiency 0.3%

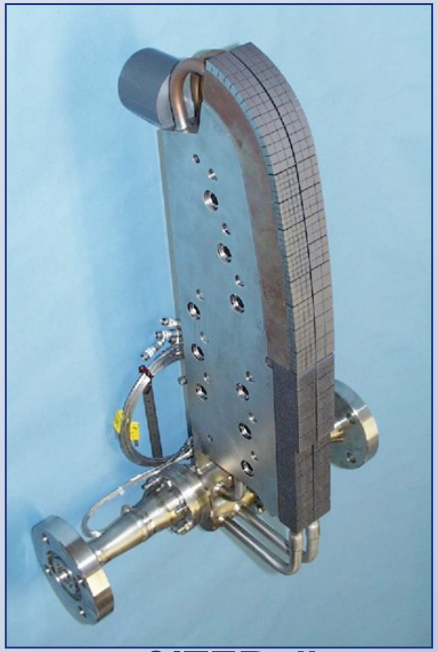
sticking efficiency of CH_x is negligible

A. Kreter *et al.*, Plasma Phys. Control. Fusion 48 (2006) 1401

Material transport in the plasma-shadowed areas

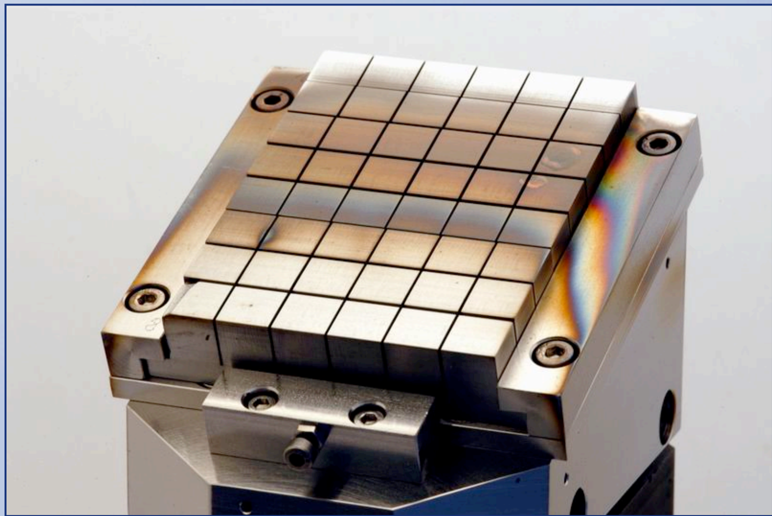
castellation to ensure thermo-mechanical durability of ITER armour

tritium can be accumulated in gaps



vertical target of ITER divertor (EU prototype)

castellated limiters with ITER-like geometry exposed to TEXTOR plasma



A. Litnovsky et al, J. Nucl. Mater. 337-339 (2005), 917.

New challenges:

Enhanced deposition in the shadowed areas
Difficulties to remove the deposits
Complicated particle transport
Materials mixing between gaps

Mixed materials

ITER materials:

Beryllium Tungsten Carbon CFC

Erosion and deposition properties of mixed materials might be changed significantly

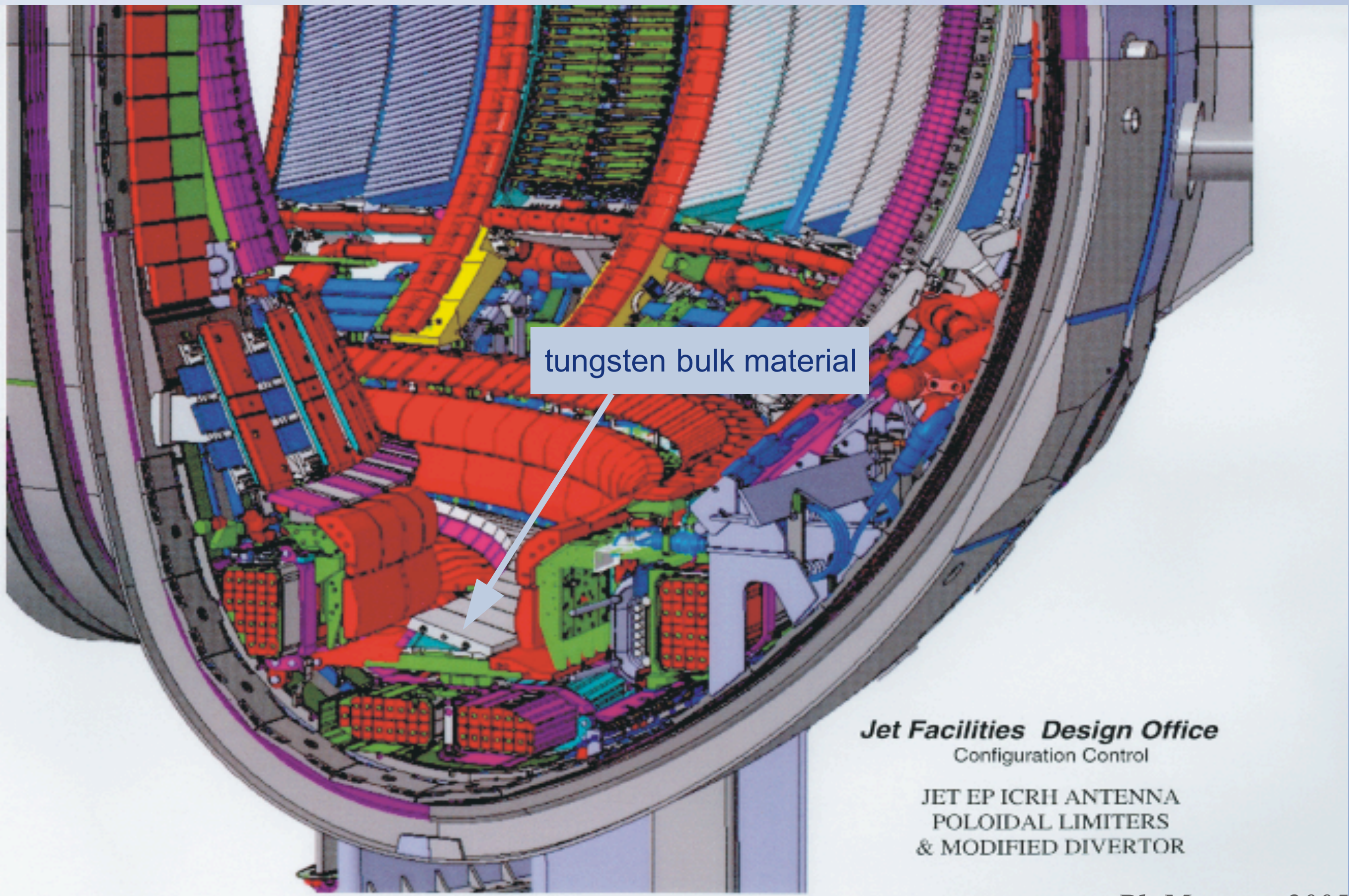
In presence of carbon CFC divertor tiles the problem of tritium retention remains

Replacement of carbon divertor tiles by tungsten tiles???

Full tungsten PFC in AUG (R. Neu)

Beryllium tungsten in JET (2010)

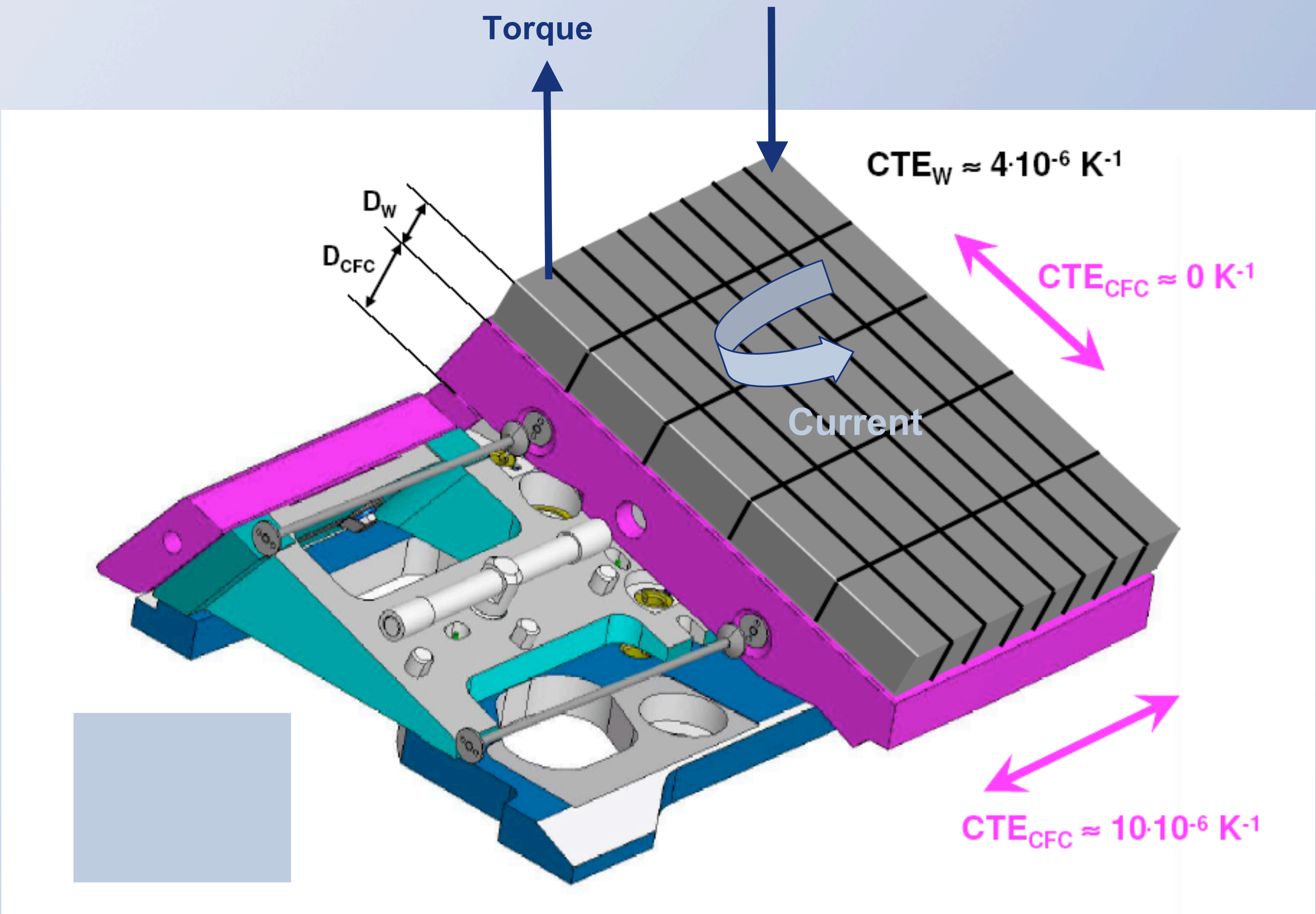
ITER-like wall in JET



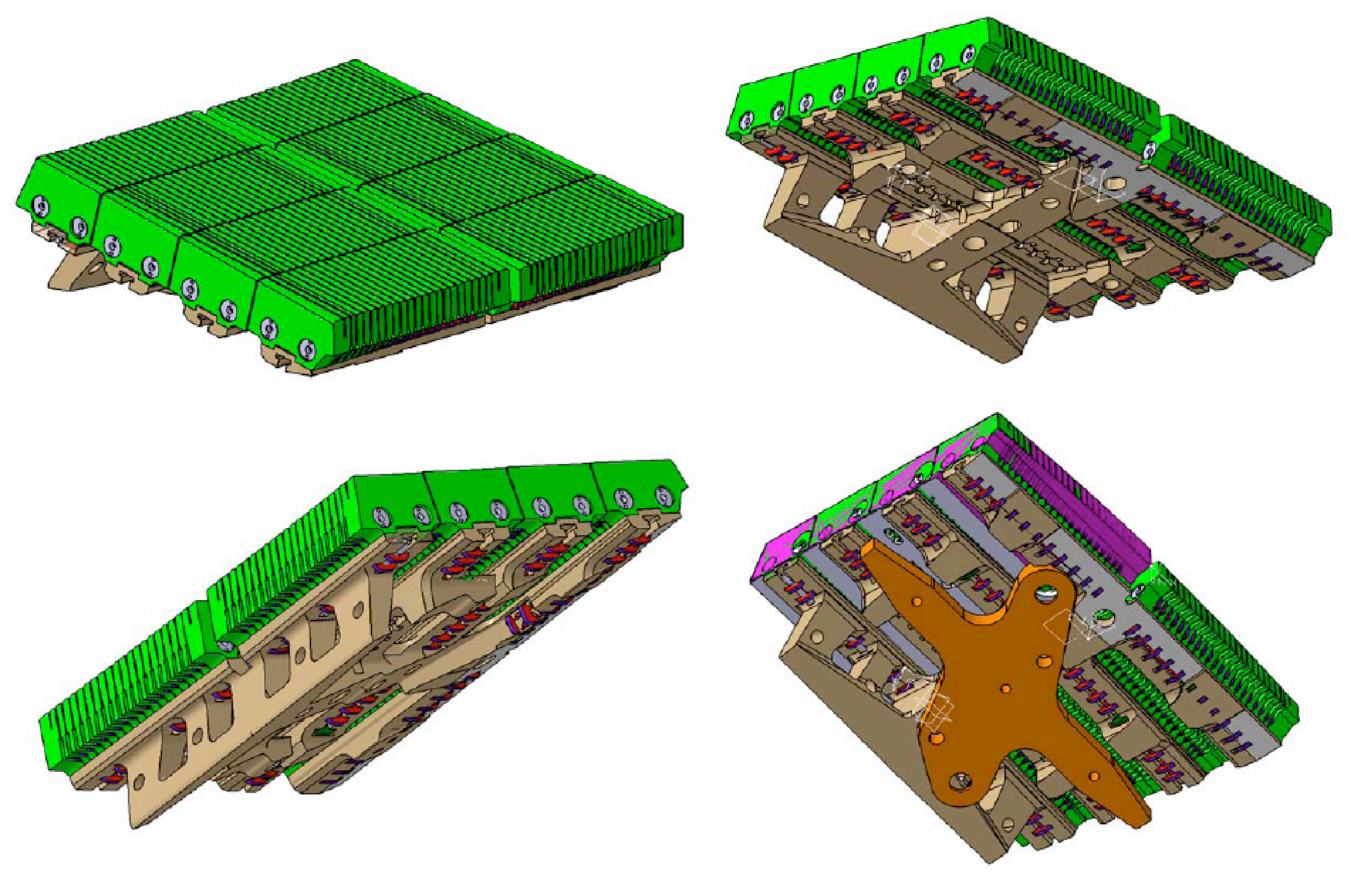
Ph.Mertens, 2005

•conceptual design of various prototypes
• thermo-mechanical and electromagnetic FEM calculations
• definition of tests and manufacturing procedures

Jet tungsten divertor tile



JET tungsten divertor tile



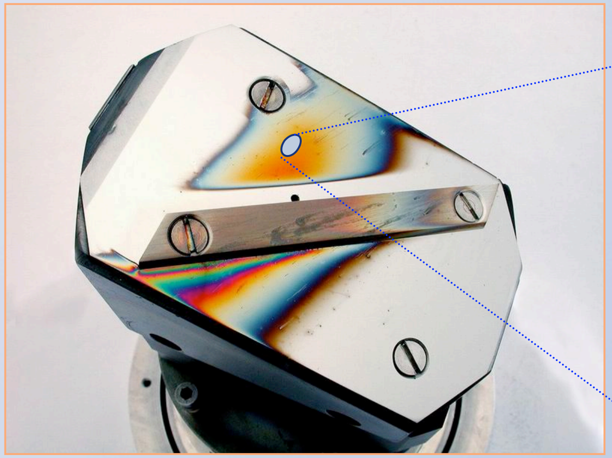
Mirrors for ITER diagnostics

Mirror surfaces will be modified by erosion, deposition and impurity implantation

Impact on all optical diagnostics

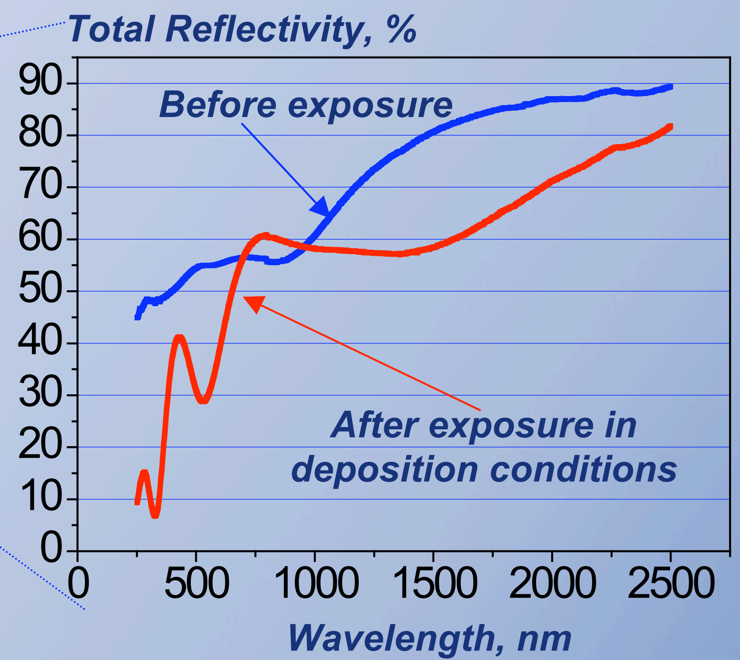
Performance of mirrors under long term plasma operation has to be studied

Mirror tests in present-day machines



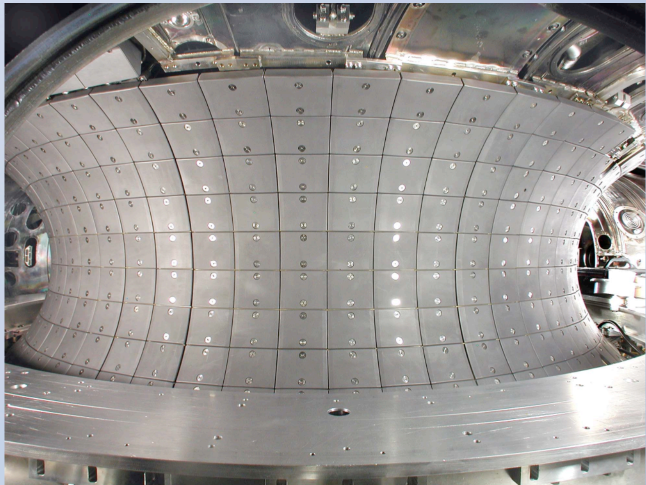
molybdenum mirror system
Carbon deposition can be clearly seen.

P. Wienhold et al, J. Nucl. Mater. 337-339 (2005), 1116.
A. Litnovsky et al, Proc. of 32nd EPS, ECA Vol. 29C, P-4.099 (2005)



TEXTOR

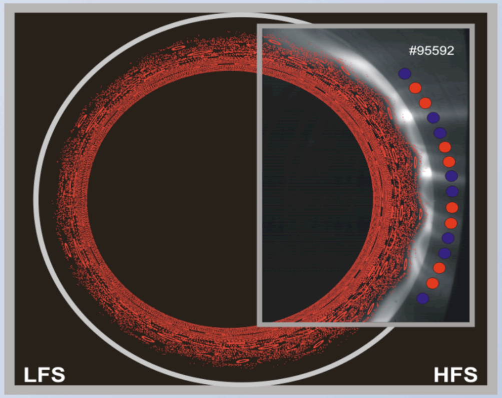
Dynamic Ergodic Divertor (DED) in TEXTOR



resonant helical perturbation

in different modes 3/1, 6/2, 12/4

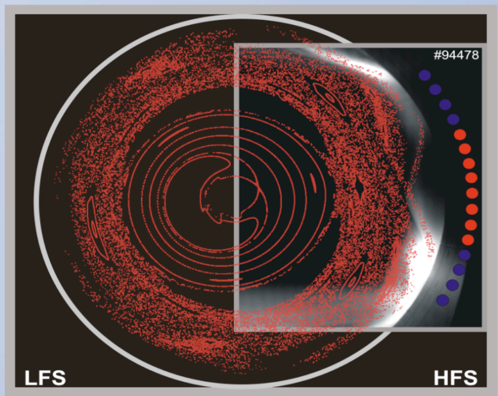
max current 15 kA, DC and AC at max 10 kHz rotation frequency



12/4 (6/2) configuration

helical divertor

stochastic plasma boundary



3/1 configuration

MHD studies (tearing mode stabilization)

Plasma heating

4 MW NBI (balanced injection)

4 MW ICRH

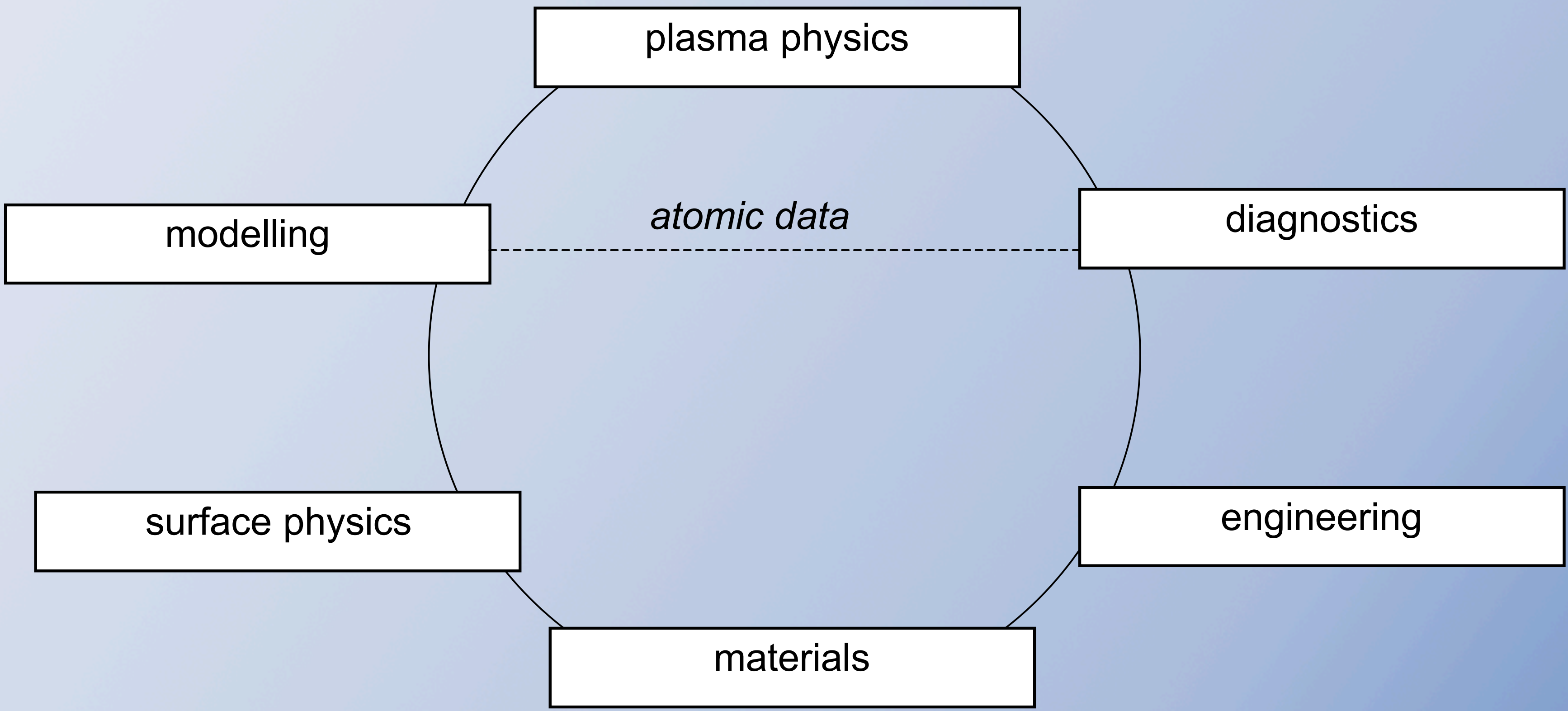
1 MW ECRH

comprehensive edge and core diagnostics

PWI Research Issues and Data Needs for Modelling

- **reflection/sticking coefficients** in particular at at low energies
- behaviour of **mixed material systems**, in particular with beryllium
- erosion properties of **deposited films**
- **tritium retention** and release mechanisms / methods
- **synergy effects** with erosion processes
- migration of materials in **complex geometries** (castellated structures)
- improved **diagnostics** (spectroscopy, surface diagnostics)
- development of **mirror** materials for optical diagnostics

Plasma-Wall Interaction – an interdisciplinary research field



Plasma-Wall Interaction – an interdisciplinary research field

PWI – a key field on the road to economic fusion

plasma physics

modelling

surface physics

cost of electricity

$$coe \propto \left(\frac{1}{A}\right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4} \beta_N^{0.4} N^{0.3}}$$

A - plant availability

η_{th} - thermodynamic efficiency

P_e - net electrical output of the plant

β_N - normalised plasma pressure

N - ratio of the plasma density to the Greenwald density

Power Plant Conceptual Study (PPCS)
D J Ward, I Cook, N P Taylor


Thank you!