Divertor Physics and Technology

9th Course on Technology of Fusion Tokamak Reaktors

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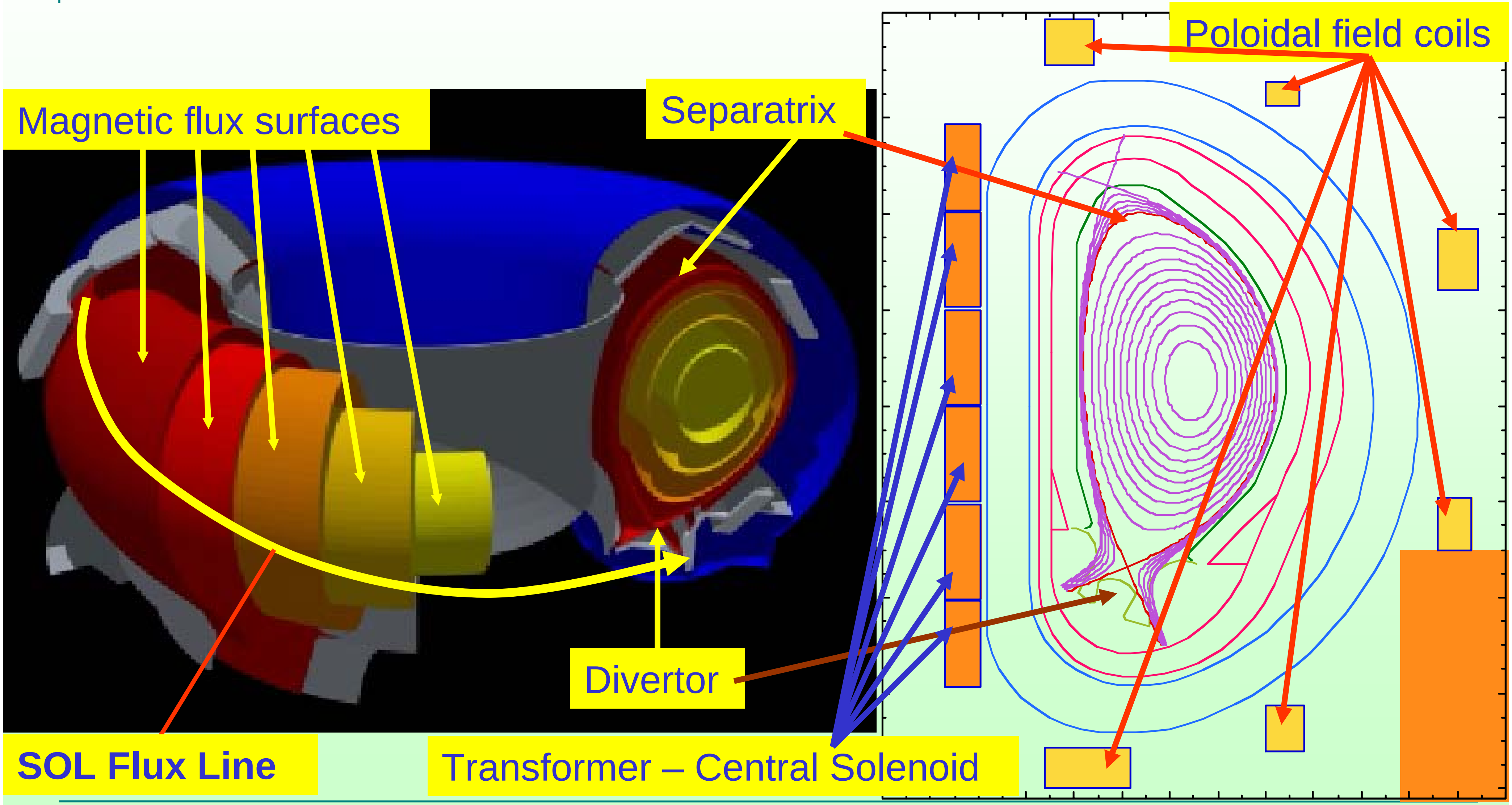
Reference: Stangeby, P.C., "The Plasma Boundary of Magnetic Fusion Devices", IOP Publishing Limited, 2000

Outline

- Scrape Off Layer (SOL) and Divertor Physics
- A water cooled Divertor
- Integration of a Divertor into a Reactor class machine (ITER)
- A He cooled Divertor
- Summary

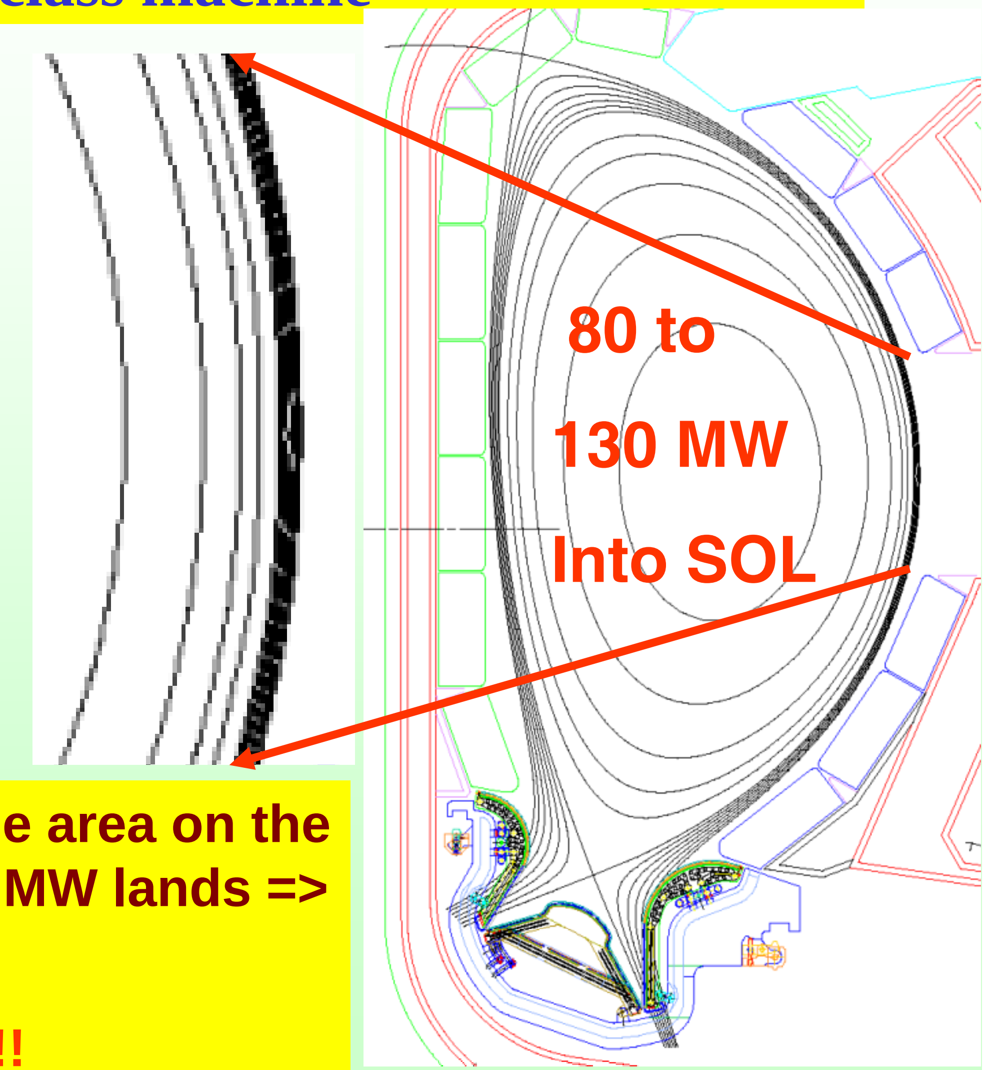
SOL and Divertor Physics

The Plasma of a modern Divertor Tokamak



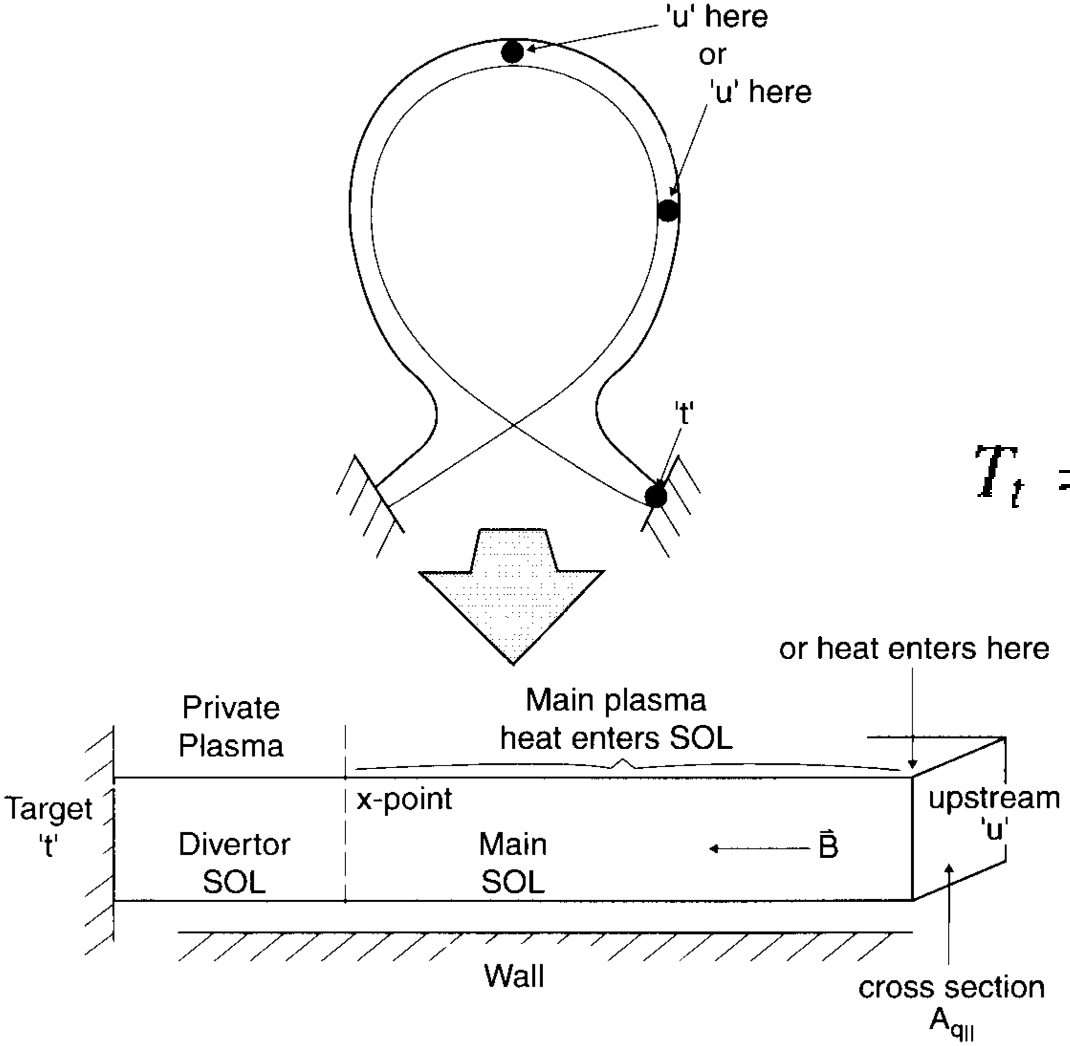
Experimental experience with divertors suggests a problem for a Reactor class machine

- Power decay length in the SOL is ~ 0.5 cm in the midplane
  - => 60% of power flows in a 5 mm thick shell towards the divertor
- Flux expansion in the Divertor ~ 3 to 5
  - 60 % of power in ~ 2.5 cm thick layer
- Poloidal target inclination can bring another factor 2 to 3 widening



- Thus we have in the order of a 5 cm wide area on the target where 60% of the power, i.e. ~ 60 MW lands => peak power > 20 MWm^{-2}
- We must find a better Physics solution !!

The Heat Transport along fieldlines is dominated by conduction



$$q_{\|\mathrm{cond}} = -\kappa_0 T^{5/2} \frac{\mathrm{d}T}{\mathrm{d}s_\|}$$

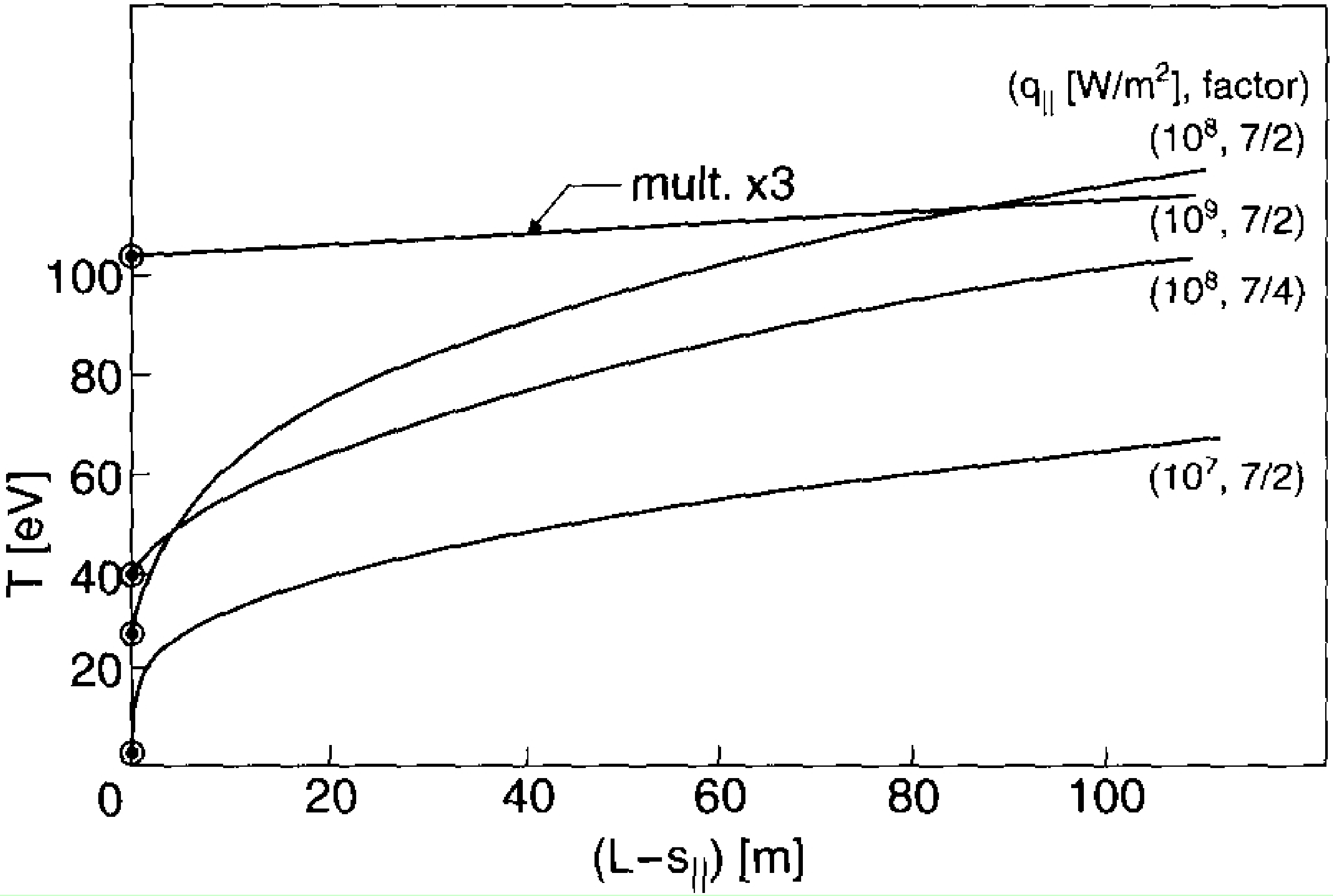
κ_{0e} ~ 2000; κ_{0i} ~ 60

$$T_t = \left[T_u^{7/2} - \frac{7}{2} \frac{(P_{\mathrm{SOL}}/A_{q\|})L}{\kappa_0} \right]^{2/7}.$$

$$T_u \simeq \left(\frac{7}{4} \frac{(P_{\mathrm{SOL}}/A_{q\|})L}{\kappa_0} \right)^{2/7}.$$

$$q_\| = q_t = \gamma n_t k T_t c_{st}$$

Insensitivity of Tu on power and the dependence of a temperature gradient along fieldlines on power can beseen



Cooling by recycling at the plates can increase the temperature gradient

High recycling regime

The SOL and Divertor Plasma in a simple 2 Point Model

$$p + nmv^2 = \text{constant.}$$

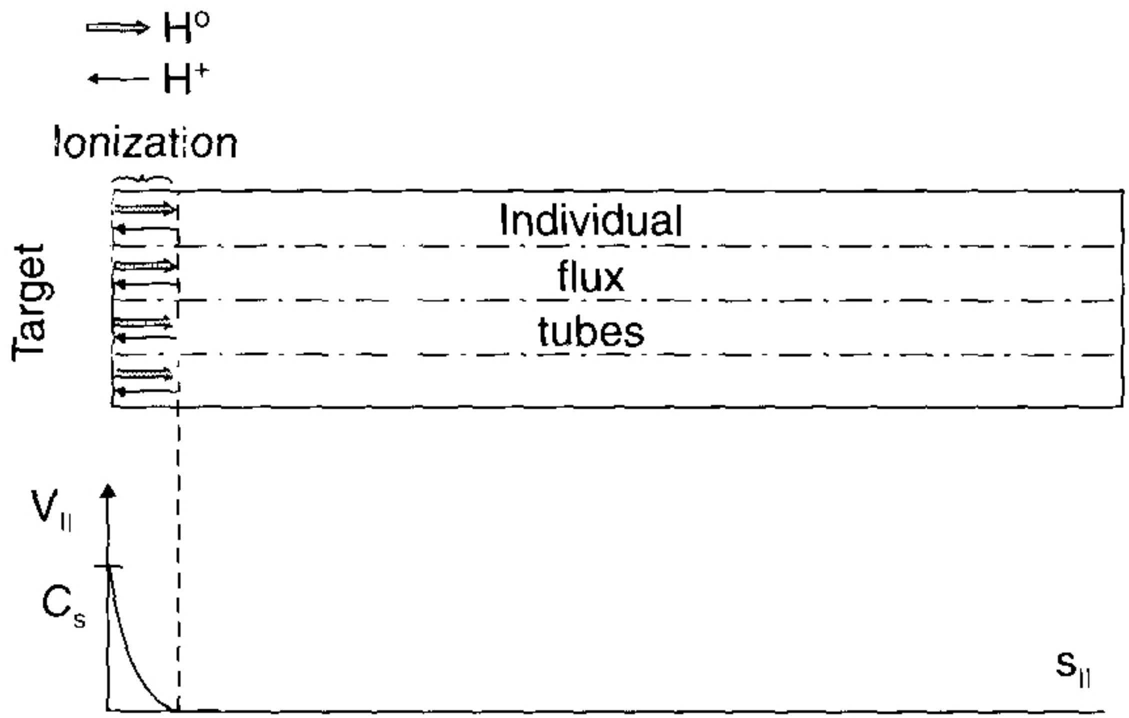
$$n_t(2kT_t + mv_t^2) = 2n_u kT_u$$

two-point model:

$$2n_t T_t = n_u T_u$$

$$T_u^{7/2} = T_t^{7/2} + \frac{7}{2}\frac{q_\parallel L}{\kappa_{0e}}$$

$$q_\parallel = \gamma n_t k T_t c_{st}$$

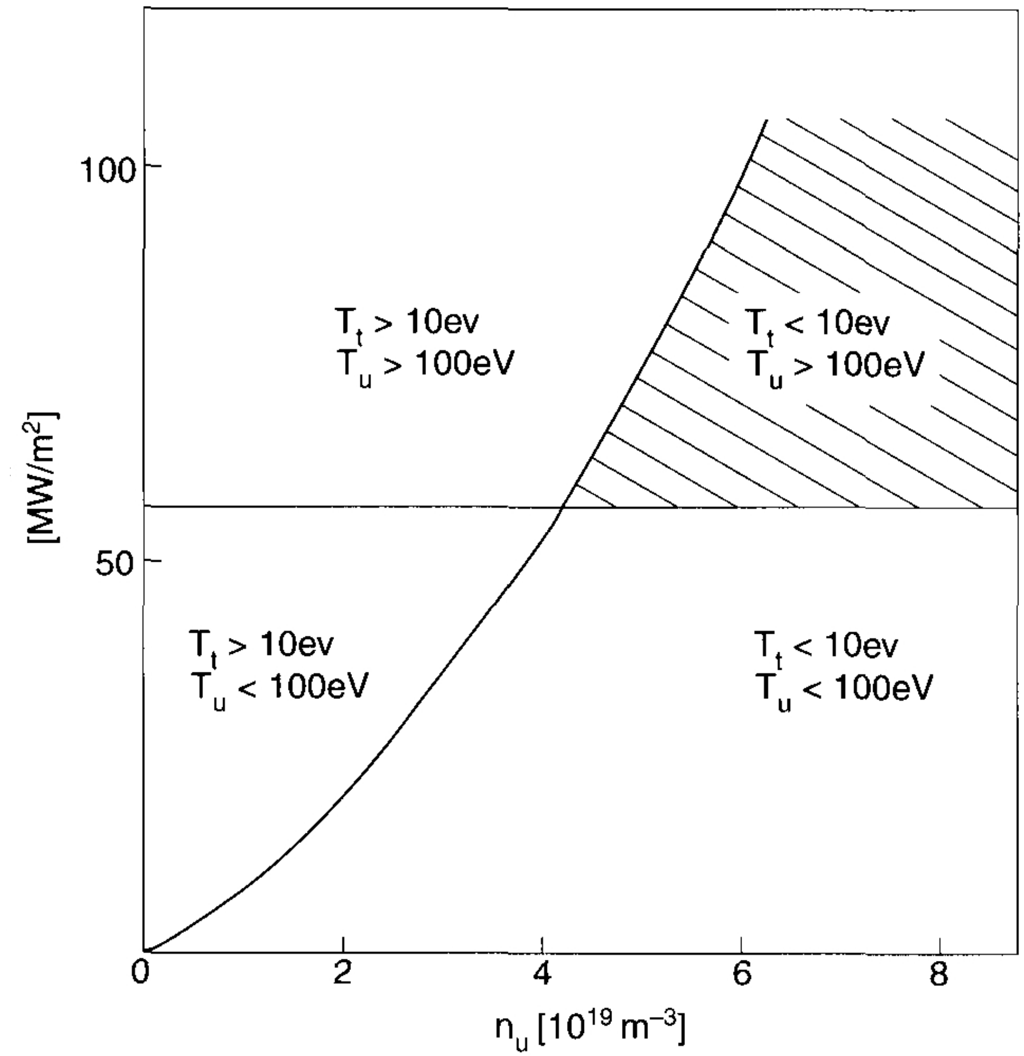


$$T_u \simeq \left(\frac{7}{2}\frac{q_\parallel L}{\kappa_{0e}}\right)^{2/7}$$

Target Temperature, Density and the High Recycling Regime

$$T_t = \frac{m_i}{2e} \frac{4q_\parallel^2 \left(\frac{7}{2} \frac{q_\parallel L}{\kappa_{0e}} \right)^{-4/7}}{\gamma^2 e^2 n_u^2}$$

$$n_t = \frac{n_u^3}{q_\parallel^2} \left(\frac{7}{2} \frac{q_\parallel L}{\kappa_{0e}} \right)^{6/7} \frac{\gamma^2 e^3}{4 m_i}$$



Energy Conservation along Field lines limits the Radiation Losses !!

$$(1 - f_{\mathrm{power}})q_{\parallel} = q_t = \gamma k T_t n_t c_{st}$$

$$P_{SOL} - P_{Imp} - P_{Rec} = P_{Sheath}$$

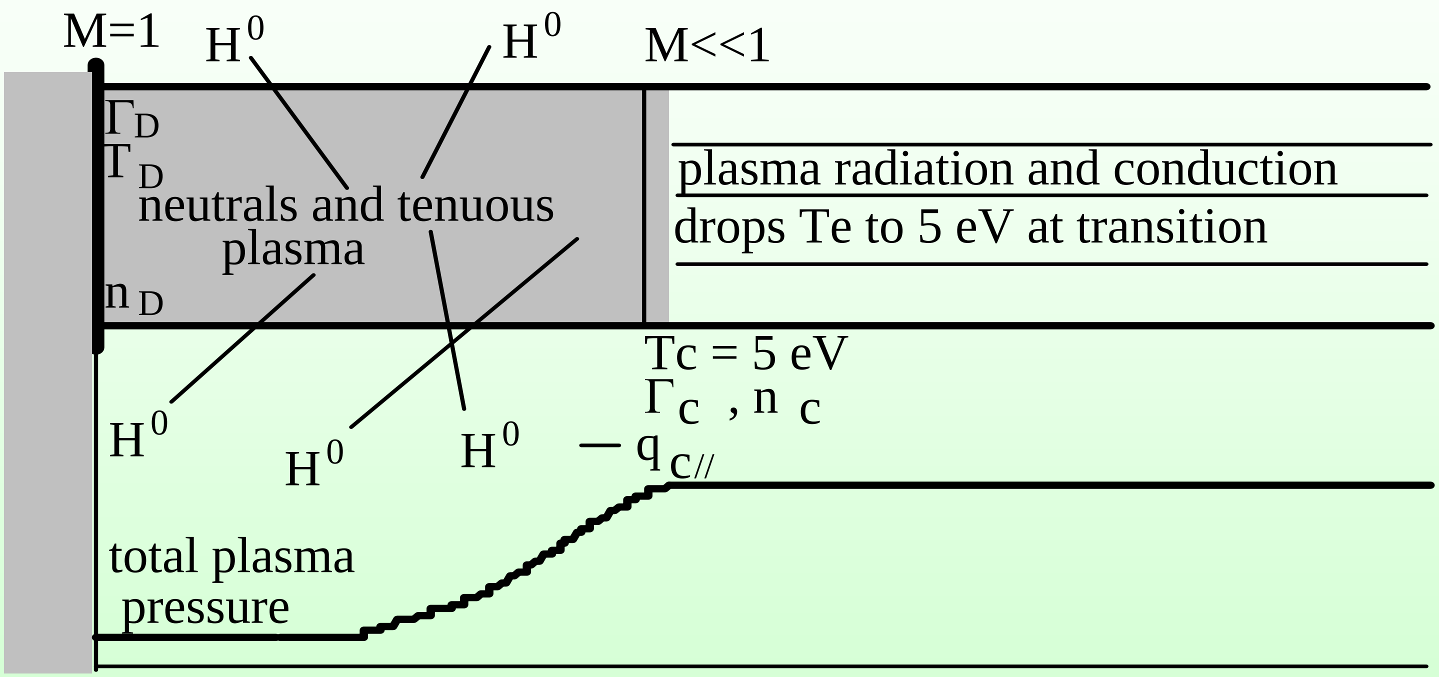
$P_{Rec} = \Gamma_t\ \xi$ where ξ is the ionization energy (13.6 eV)

$$\boxed{\Gamma_t = \frac{n_u^2}{q_{\parallel}} \left(\frac{7}{2} \frac{q_{\parallel} L}{\kappa_{0e}} \right)^{4/7} \frac{\gamma e^2}{2 m_i}}$$

Irradiation by impurities is limited at a given density

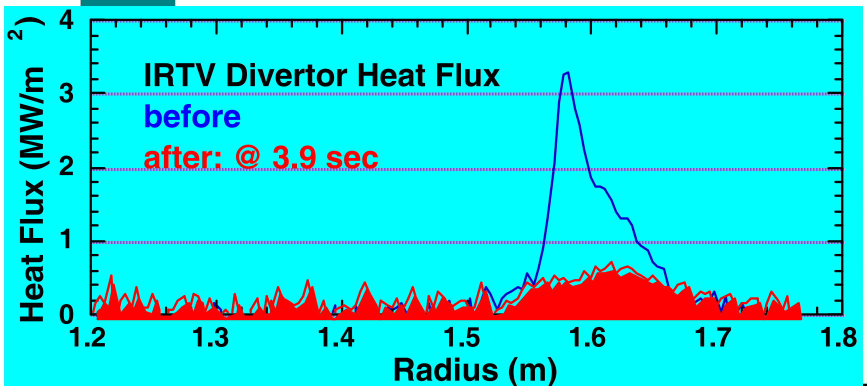
or the upstream density is limited => Edge density limit !!!

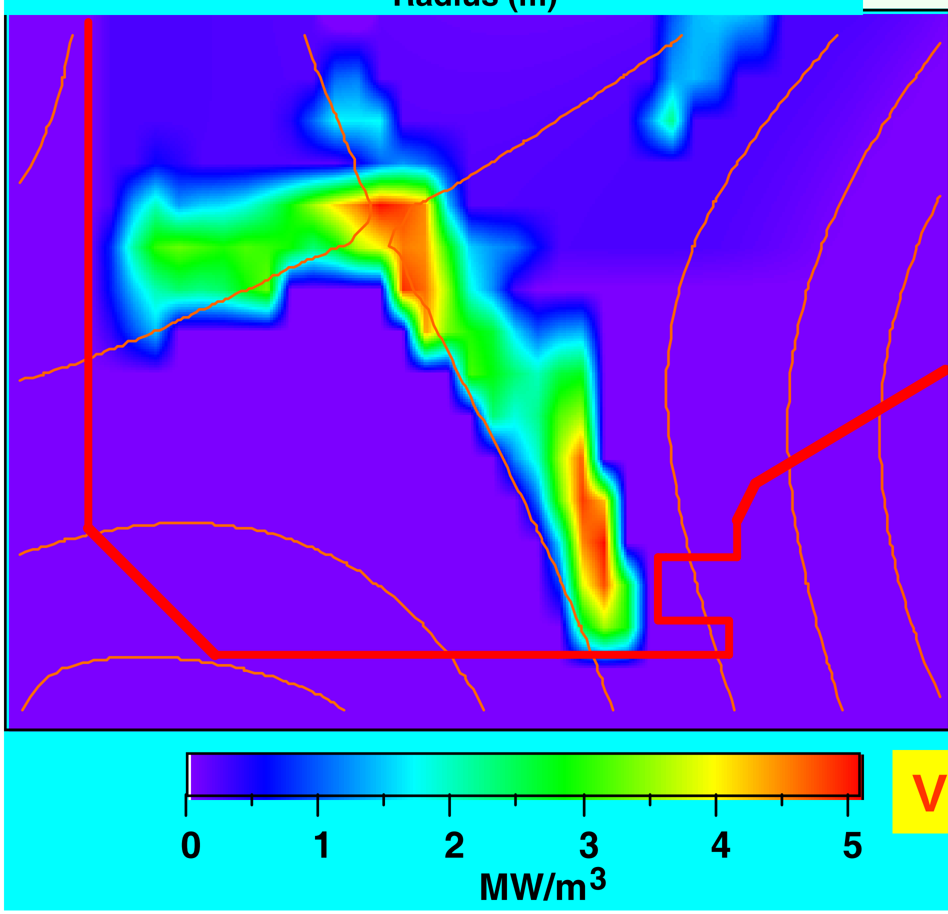
Energy Losses can be increased by breaking the pressure balance, i.e. by momentum losses

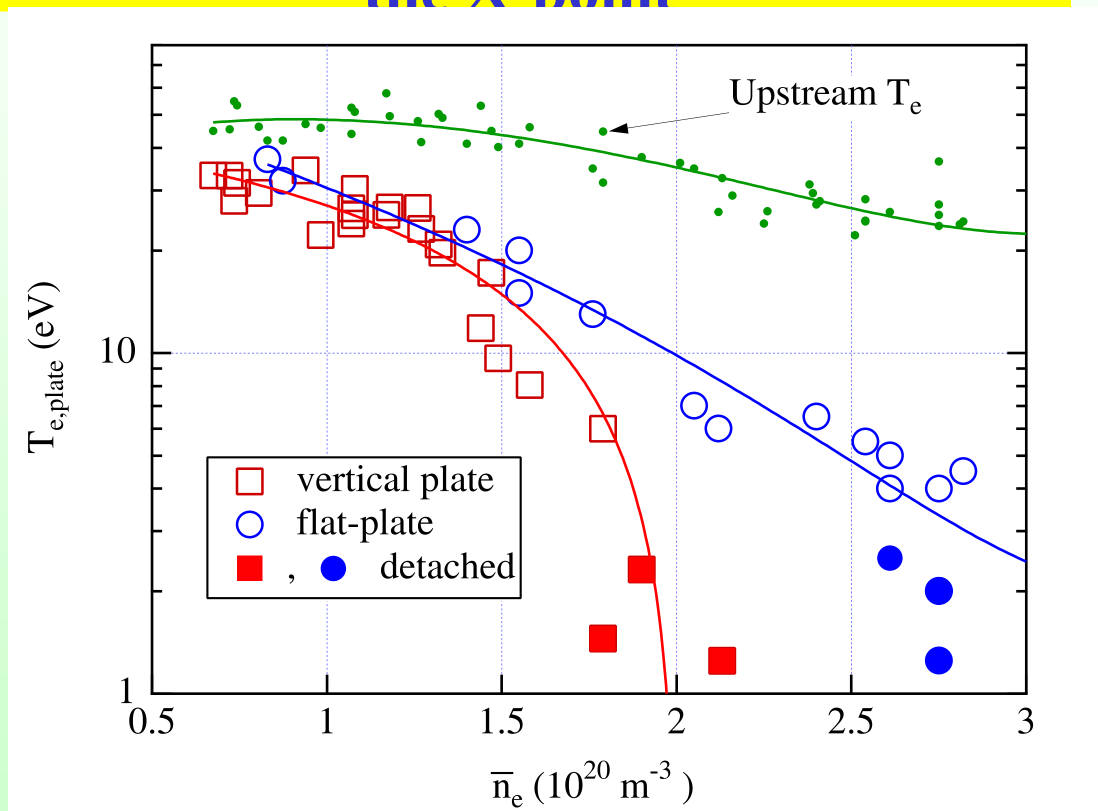


$$\Gamma_t \propto \frac{f_{\rm mom}^2 f_{\rm cond}^{4/7}}{1 - f_{\rm power}},\qquad n_t \propto \frac{f_{\rm mom}^3 f_{\rm cond}^{6/7}}{(1 - f_{\rm power})^2},\qquad T_t \propto \frac{(1 - f_{\rm power})^2}{f_{\rm mom}^2 f_{\rm cond}^{4/7}}.$$

Detachment with targets perpendicular to the flux lines occurs across the SOL and forms Marfe in the X-point

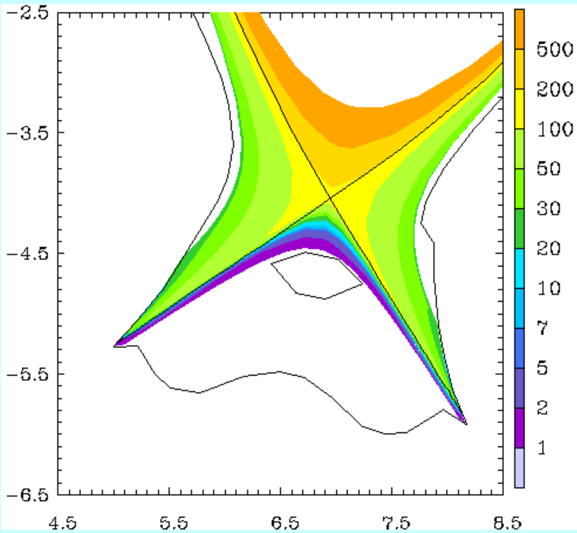




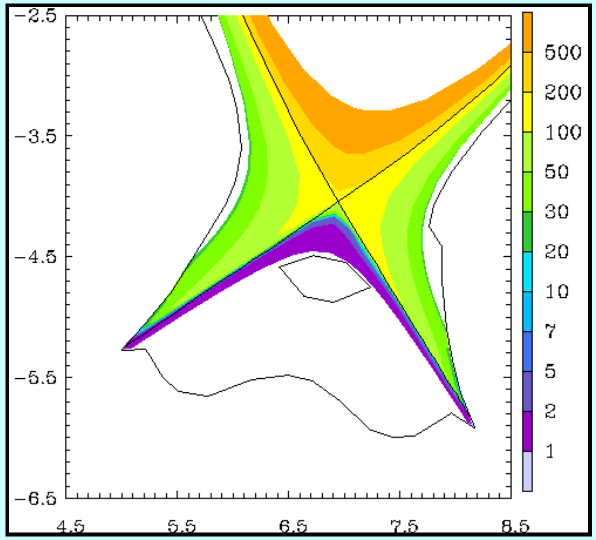


Vertical Target needs less density to detach

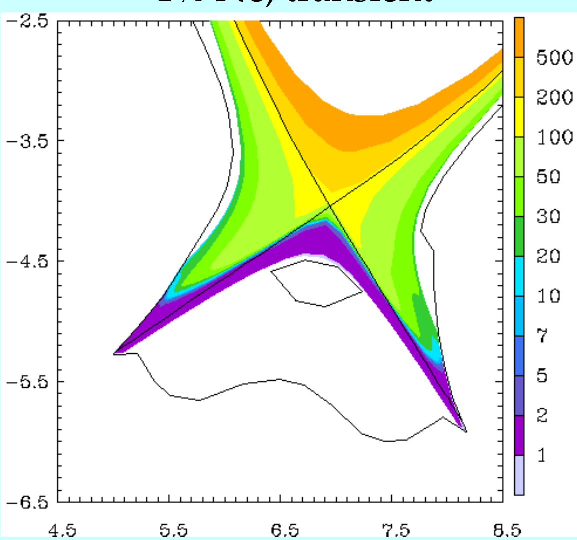
Attached
0.01% Ne



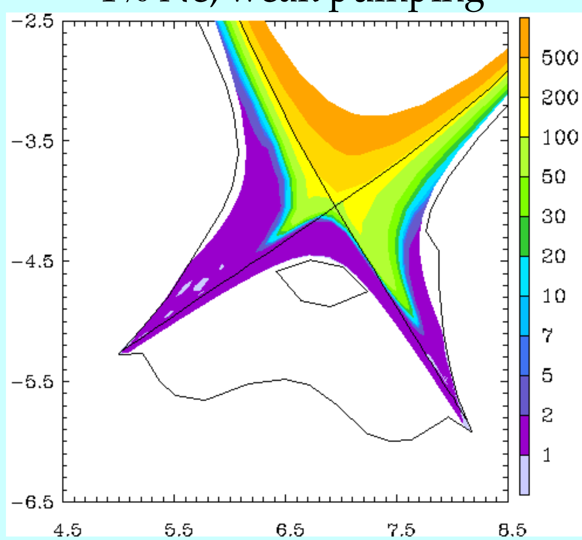
Partially attached
0.7% Ne



Shallow detached (PSI)
1% Ne, transient



Deeply detached
1% Ne, weak pumping



Vertical Target makes partial detachment possible

Partial detachment increasing with impurity radiation

Vertical Target pushes neutrals towards the hot Separatrix

Multiple Charge exchange transports momentum from the hot area to the outer SOL

2-D calculations are essential to understand the behaviour

The 2D B2-EIRENE Model

Model used

B2-Eirene code package, version solps4.0

D, He, C ions and atoms, D_2 molecules

Toroidal geometry

Flux-limited parallel transport

No drifts, no currents

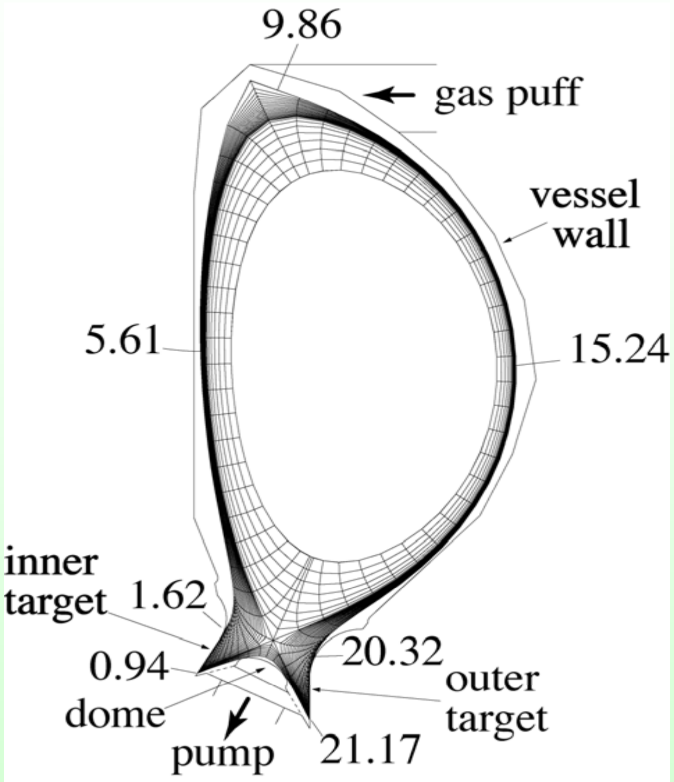
Sheath boundary condition on targets

Constant radial diffusivities in real space

$D_\perp = 0.3\ m^2 s^{-1}$, $\chi_\perp = 1\ m^2 s^{-1}$, No pinch

Decay length of 3 cm for all plasma T & n at the grid edges

Core-edge interface: power and D fluxes, zero C fluxes, no He^+, He^{++} density adjusted to fusion power and pumping speed



100% recycling of D and He everywhere

Albedo at the pumping duct (bottom PFR)

Liners: 56% transparency for toroidal discontinuity

Surface properties - realistic surfaces (RS)

Realistic surface (RS) model:

Carbon at targets

Deposited C can be re-eroded

=>Two surface states, depending on erosion-deposition balance

Erosion-dominated

⇒ metal, no C absorption

Deposition-dominated

⇒ C, no C source

⇒*Determined iteratively*

calculate total C erosion rate at all the surface locations and compare it with carbon deposition there

- specify surface properties accordingly
- iterate until convergence

B2-Eirene setup:
fluxes calculated on different surfaces

Ions on grid edges

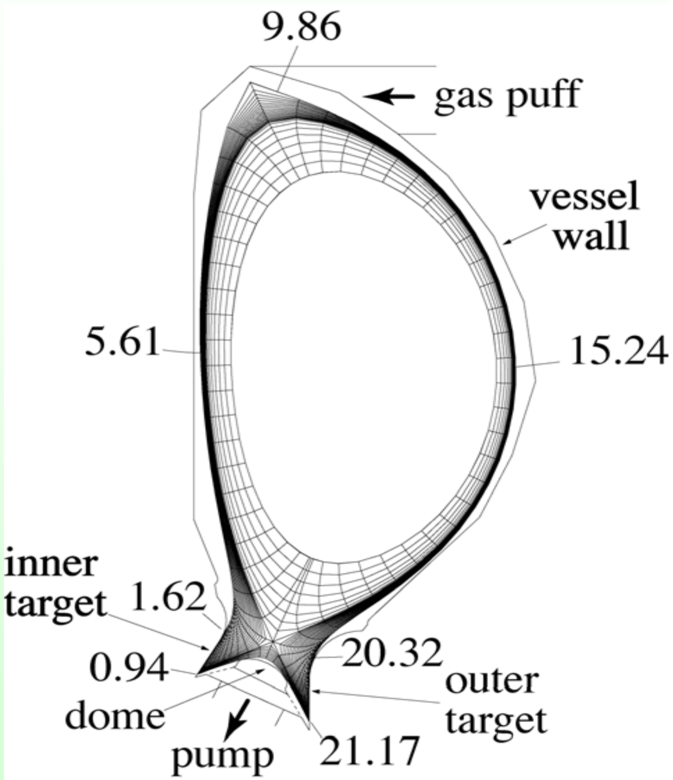
Neutrals on walls

⇒ combine where possible

⇒ only neutrals under dome

Targets: regular C

⇒ the only C source



Sputtering model

Physical sputtering by atoms on the walls
Physical sputtering by all ions, also on the grid edges
Chemical erosion by D atoms and ions => C atoms, 1 eV
Constant $Y_{ch} = 0.01$ as the basic assumption
Vary Y of redeposited carbon on walls

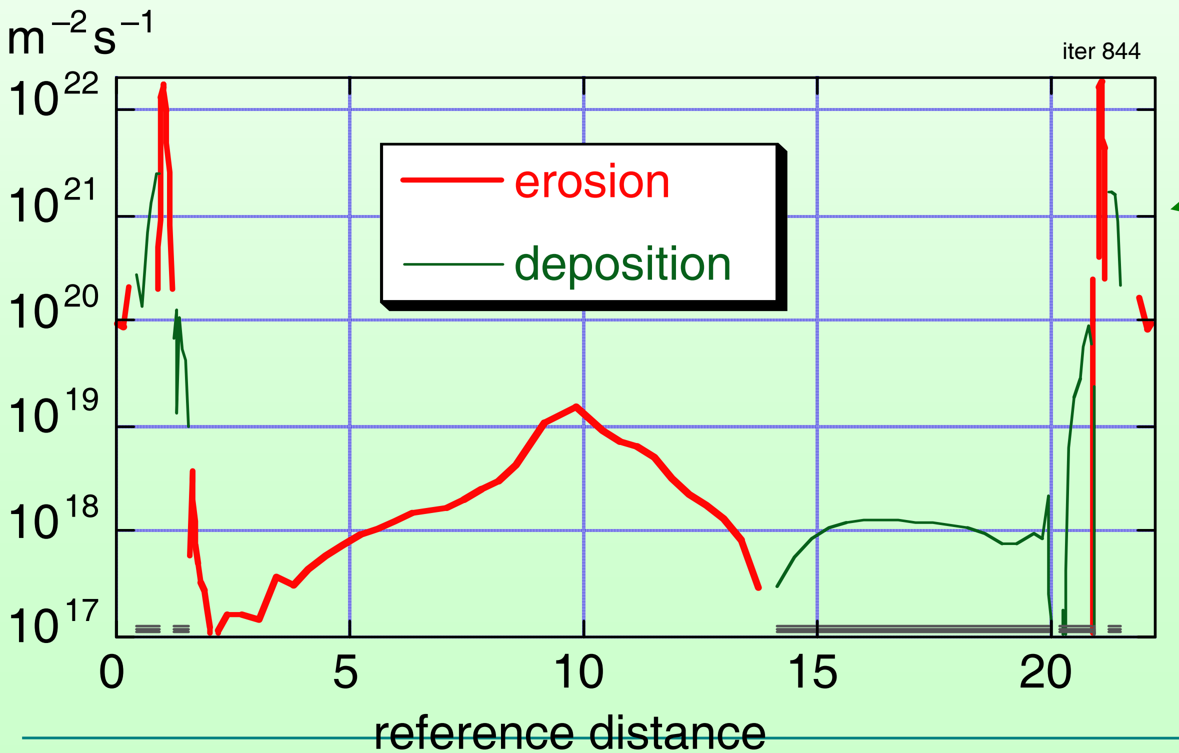
Effect of carbon re-erosion

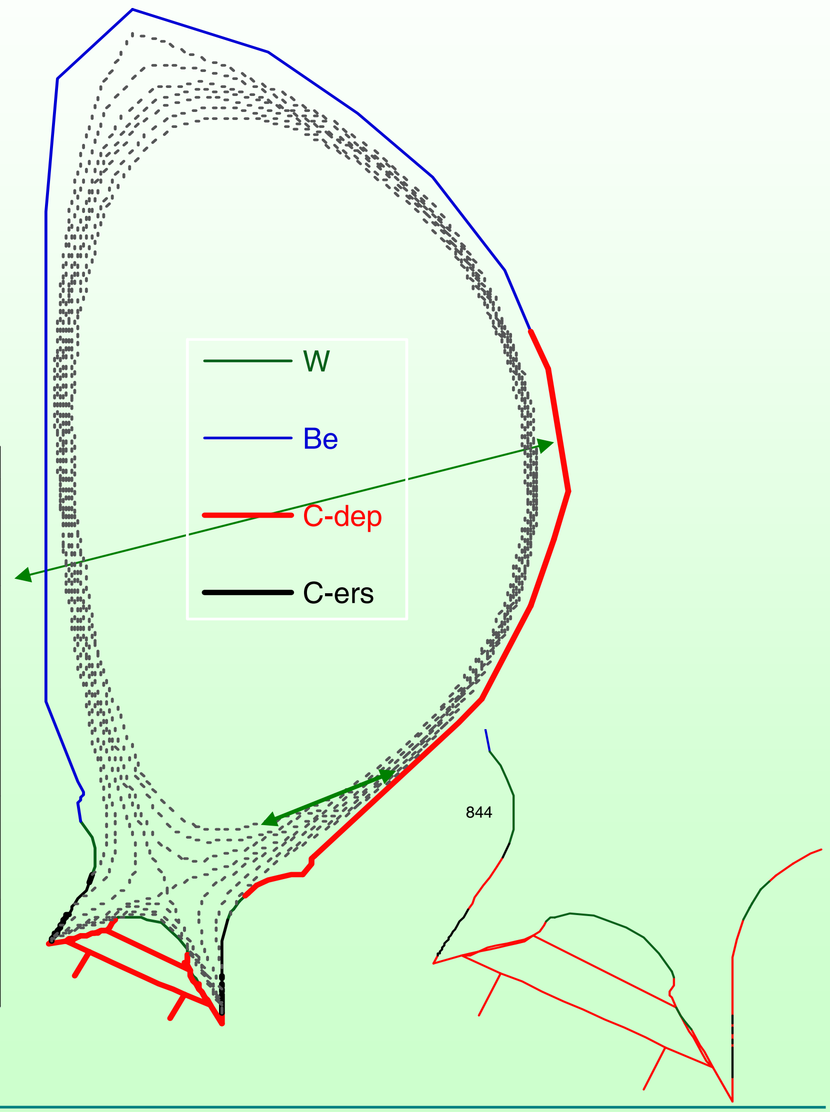
Erosion-deposition profile

RS case: 100 MW, gas puff

⇒Net carbon deposition in the main chamber – *disagreement with JET data?*

⇒Increase of Y for redeposited material reduces deposition zone





Operating window, scaling, detachment

Common feature:

n_s saturation with neutral pressure in private flux region p_{DT}

- two regimes, non-saturated “a” and saturated “b”
- “b”: full detachment of inner divertor => unwanted! The limit of the operating regime is the saturation point

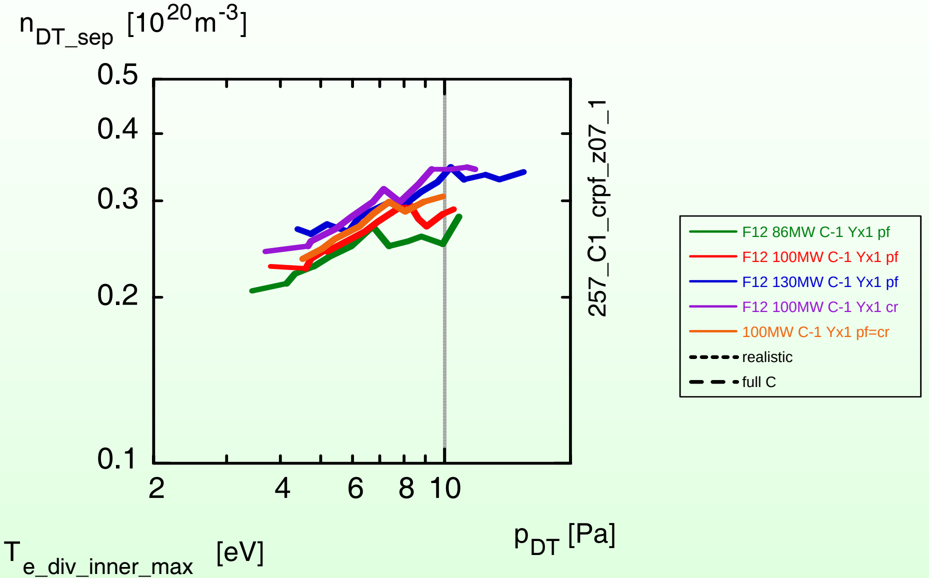
First step: selection of the basic parameters (P_{in}, p_{DT}, f_f, P_i/P_e, S_p, …)

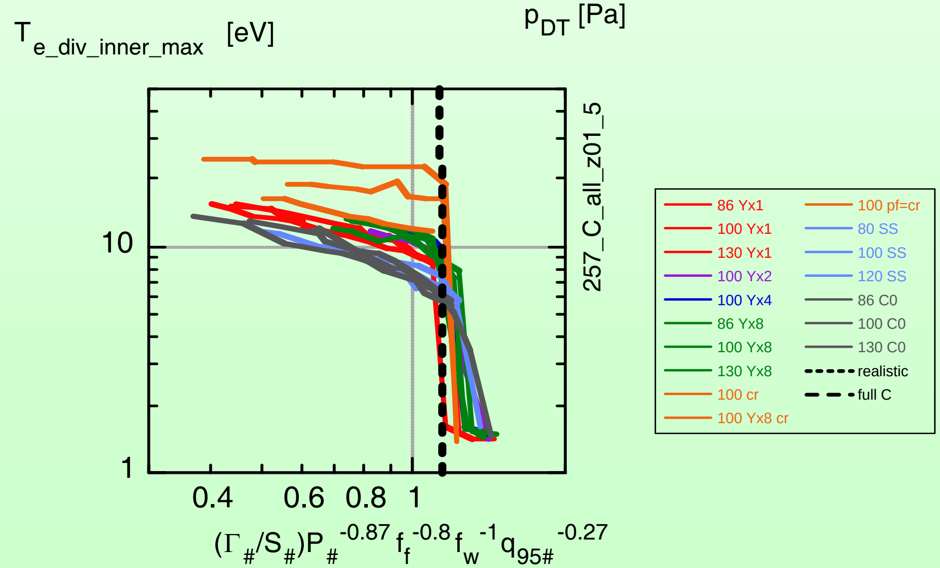
Key parameter: normalised p_{DT}, i.e: (#=normalised)

$$\mu \equiv p_{DT\,\#}\; P_{SOL\,\#}^{-0.87}\; f_f^{-0.8}\; f_w^{-1}\; q_{95\,\#}^{-0.27}$$

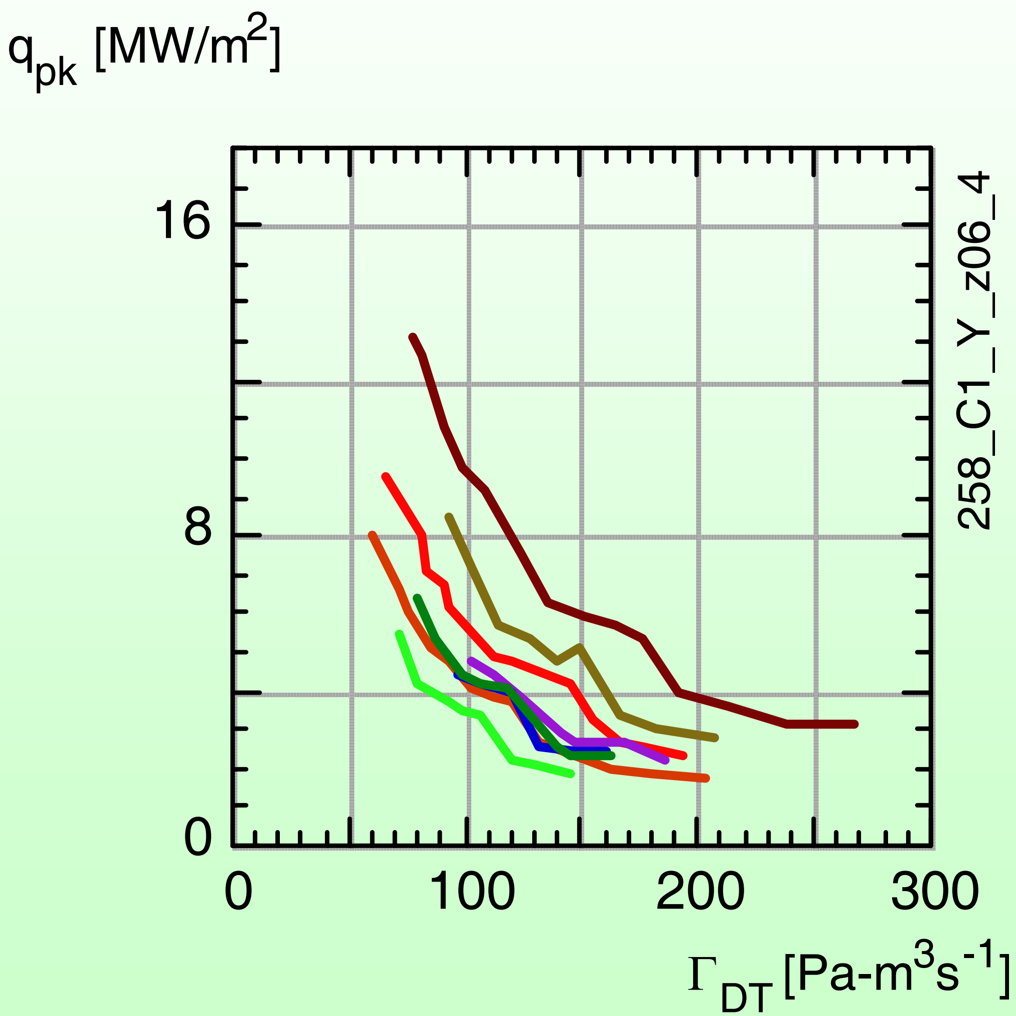
$$p_{DT\,\#} \equiv \Gamma_\# / S_\#$$

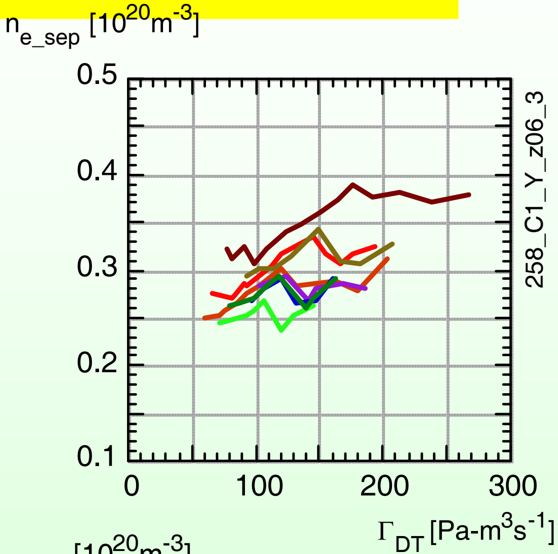
in terms of this parameter, saturation is at 1, full detachment of the inner divertor (max temperature along inner target drops to ~1 eV) is at 1.17

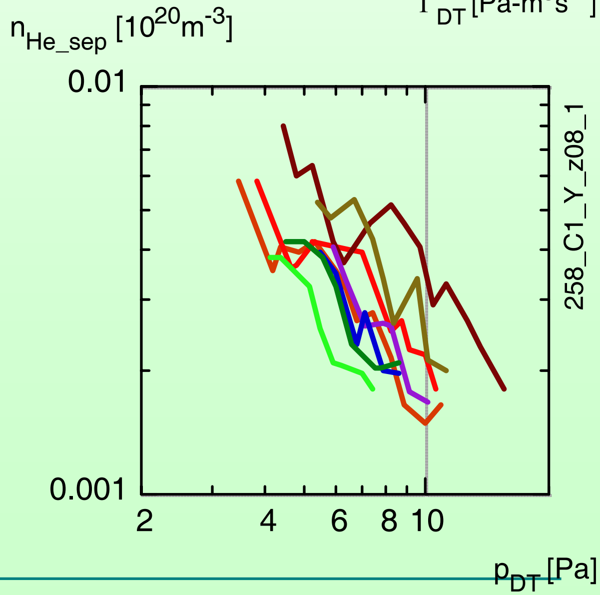




Results from B2-EIRENE for ITER



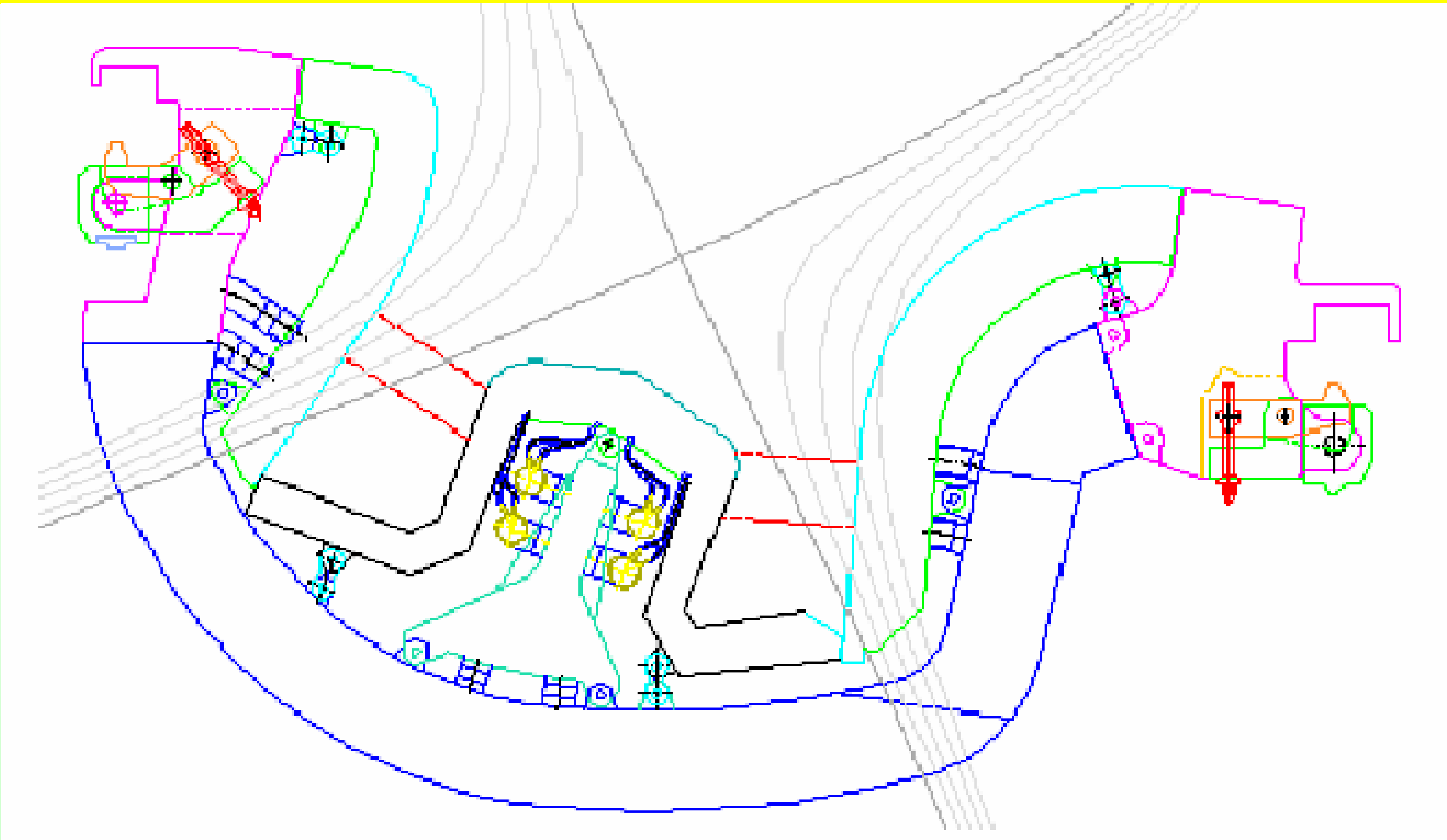




What do we know about divertor configurations ?

- We have learned that Vertical Targets (VT) pushing neutrals towards the private region are optimum
- A so called Domed Target, i.e. a vertical target pointing outward has the opposite effect and would starve the hot region from neutrals
  - The consequence would be very high power loads and low neutral density for pumping
- A target perpendicular to the flux lines pushes neutrals towards the X-point and thus causes uncontrolled full detachment and thus X-point marfes which destroys plasma performance
  - This is also true if the VT is to shallow and reflects neutrals towards the X-point
  - Due to a longer decay length for particles when compared to power the divertor has to be deep i.e. majority of particles pushed below X-point

Simple Rules to Design an optimised Divertor

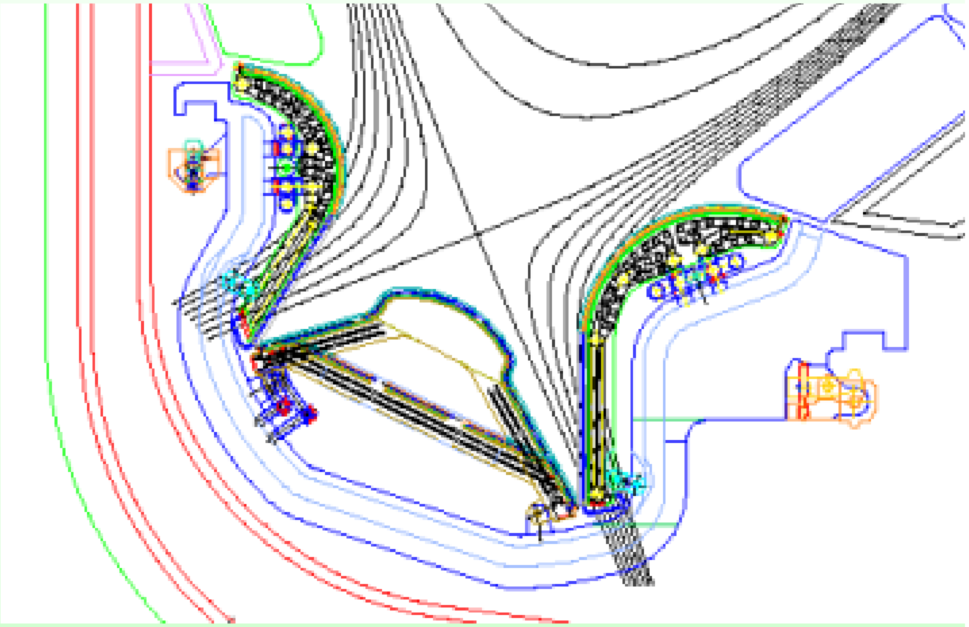


1.) Poloidal Angle such (~ 15 to 25 degrees) that total angle is > 1 degree

2.) Normal to the target where 3 cm flux line intersects points below the X-point

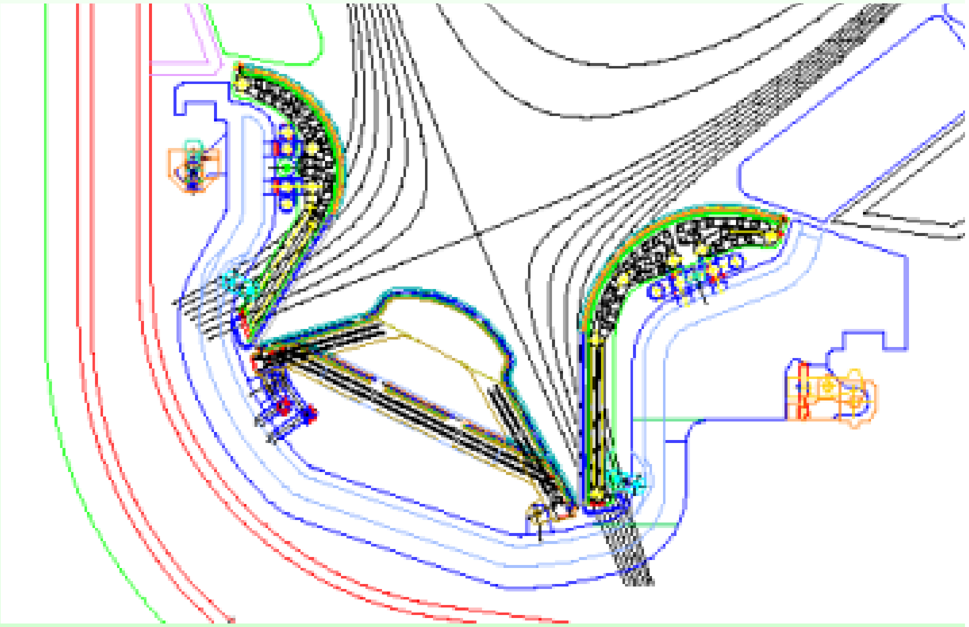
What else do we need on the upper emd of the Divertor Targets ?

- Neutrals have to remain to a large extend in the divertor region
- Due to the fact that the plasma is a very effective pump there is ~ 50% probability for neutrals to get ionised and 50% to do CX
    - CX neutrals leave the plasma again and thus could escape into the main chamber if baffle is too short
- Therefore a baffle following the 5 cm flux line for ~ 0.5 to 1 m poloidal length is required to reduce the neutral flux by ~ 2 orders of magnitude

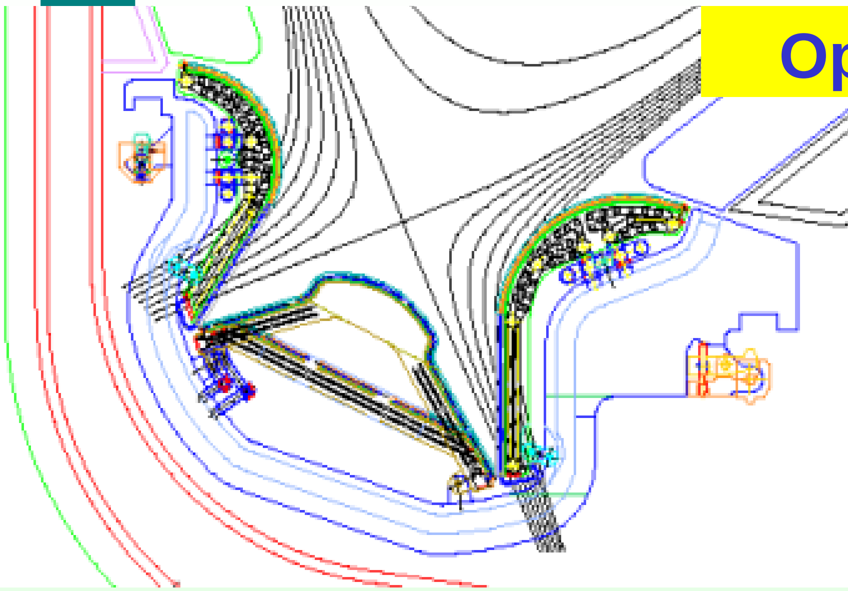


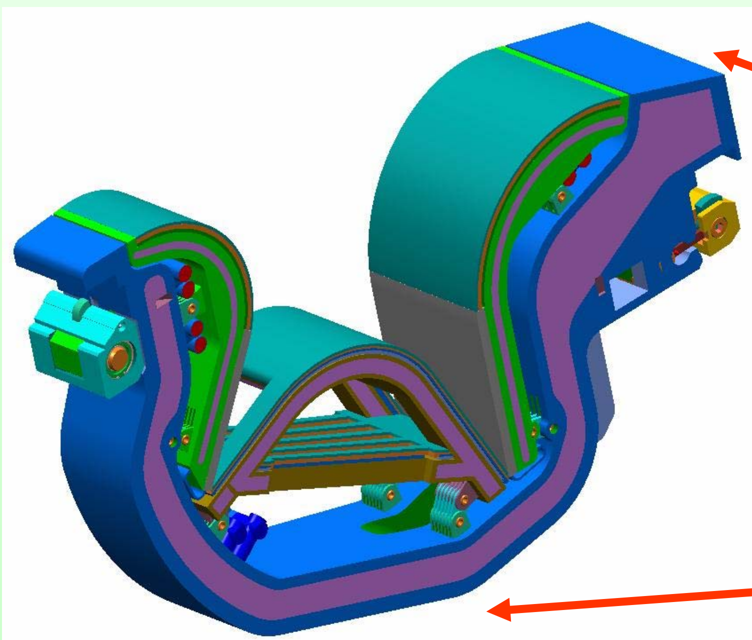
What structure do we need in the private region ?

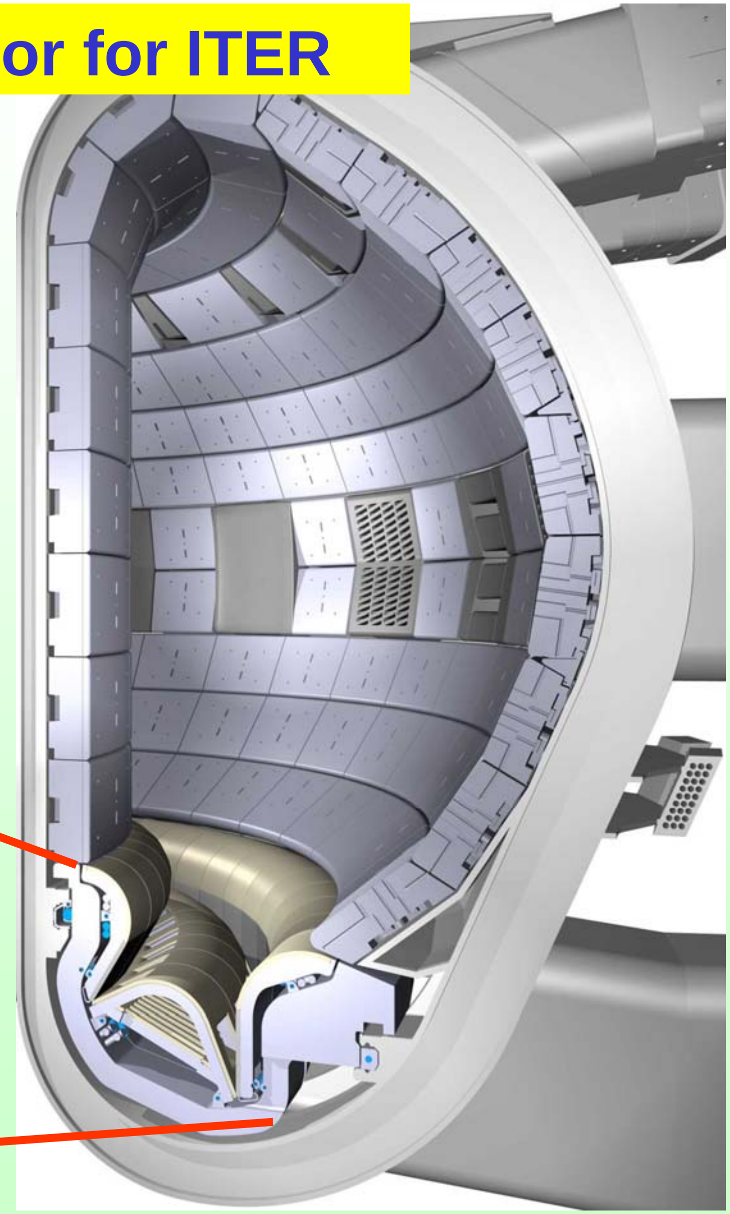
- Due to the high neutral density in the private region an X-point marfe becomes again more likely and thus reduces the operation window in terms of density
    - We need therefore a baffle which shields the X-point from neutrals
    - We also need a large conduction between inner and outer strike zone
- As a consequence a so called Dome Structure is introduced
    - The additional benefit of the Dome is better neutron shielding of VV and coils



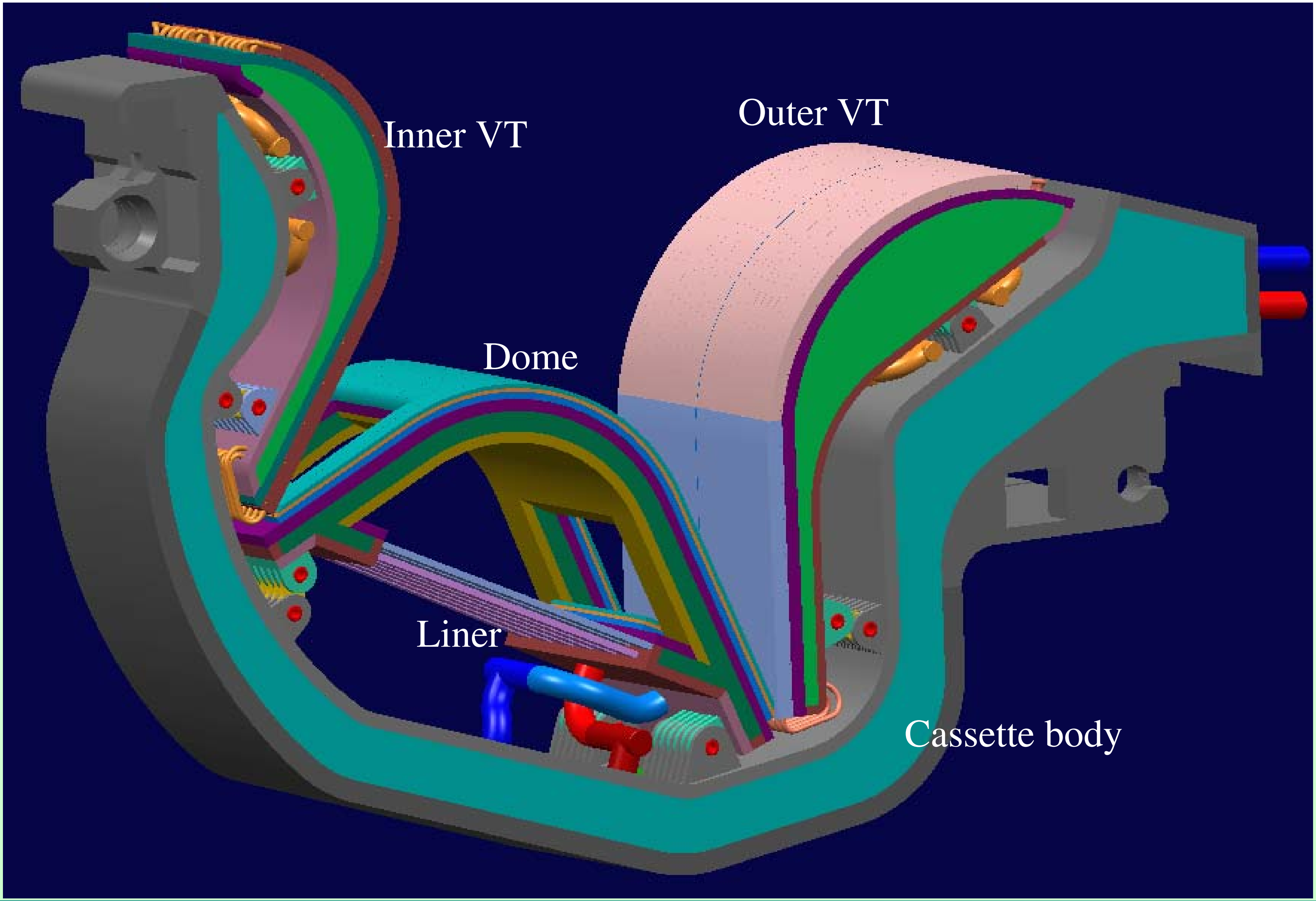
Optimised Divertor for ITER

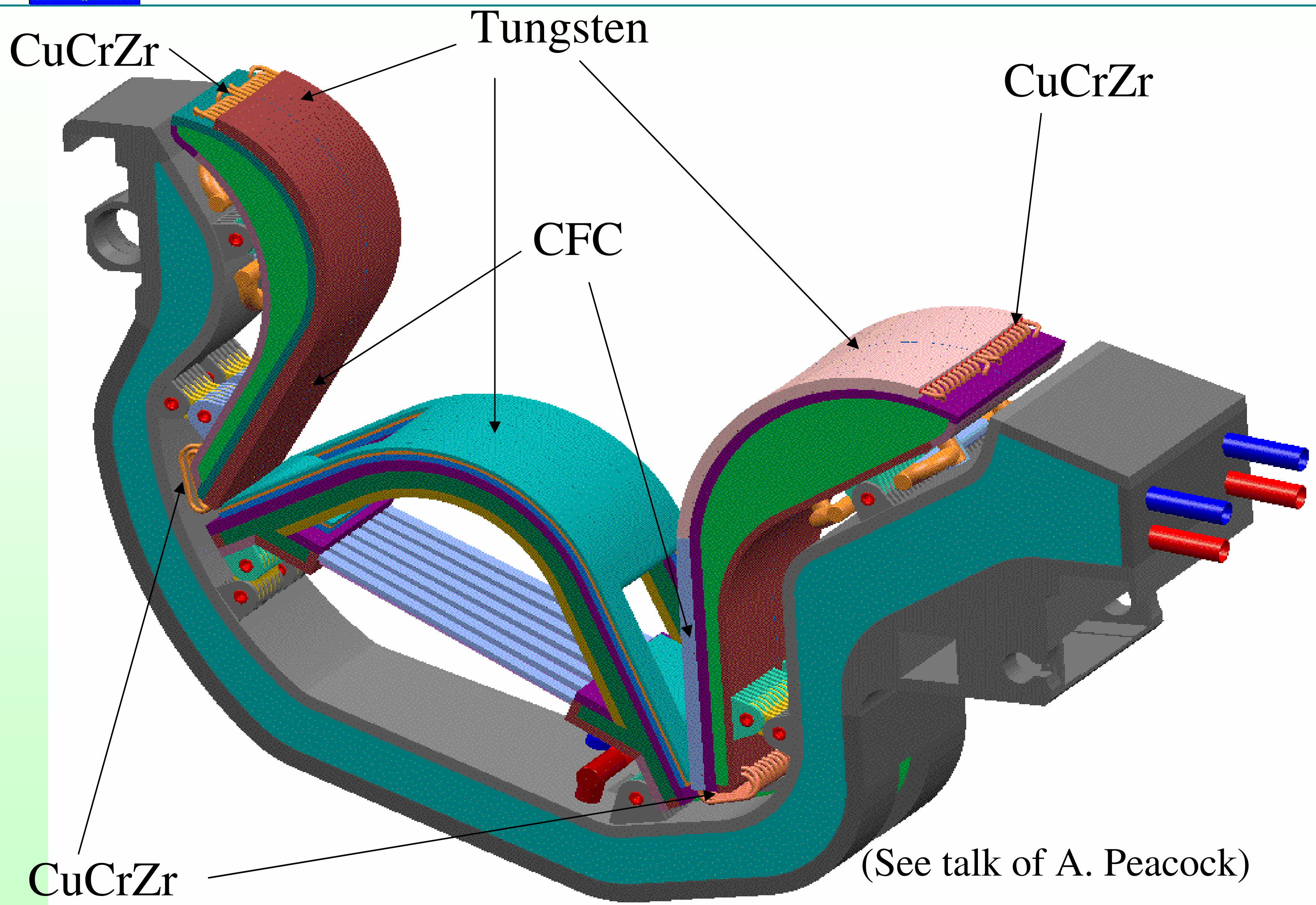






A Water cooled Divertor

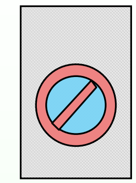

Inner VT
Outer VT
Dome
Liner
Cassette body

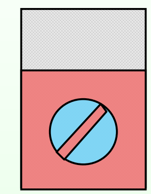


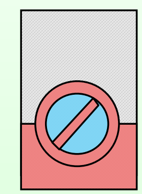
(See talk of A. Peacock)

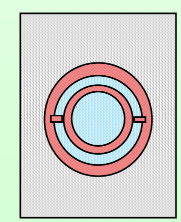
Thermal-hydraulics

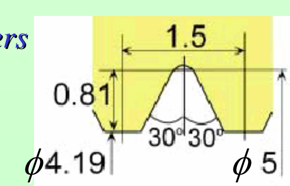
	ICHF Uniform MWm^{-2}	ICHF Peaked* MWm^{-2}	Pressure Drop $MPa\ m^{-1}$	
Smooth tube	17		0.2	
Swirl flow 10 bore, twist ratio, Y=2 As above but CFC †	27 22	34 30	0.6	*Y = number of tube internal diameters per 180° twist.* *Similar results obtained for 12mm bore tube*
Annular flow, 16 OD 11 ID 15 OD 11 ID, Y=3	27 –	– 35	1.0	
Hypervapotron fin 4 high x 3 wide, groove 3	35	43	0.55	*Best suited to flat tile geometry*
Porous coating	> 30			*Ref. Grigoriev et al; March 2000*
Screw tube M10 x 1.5 pitch		43	0.5	*Ref. Ezato et al; Oct 2002* *Values extrapolated to ITER coolant parameters*

Monoblock


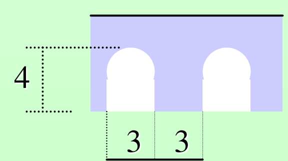
Flat tile monoblock


Saddleblock


Annular flow




Screw tube



Hypervapotron

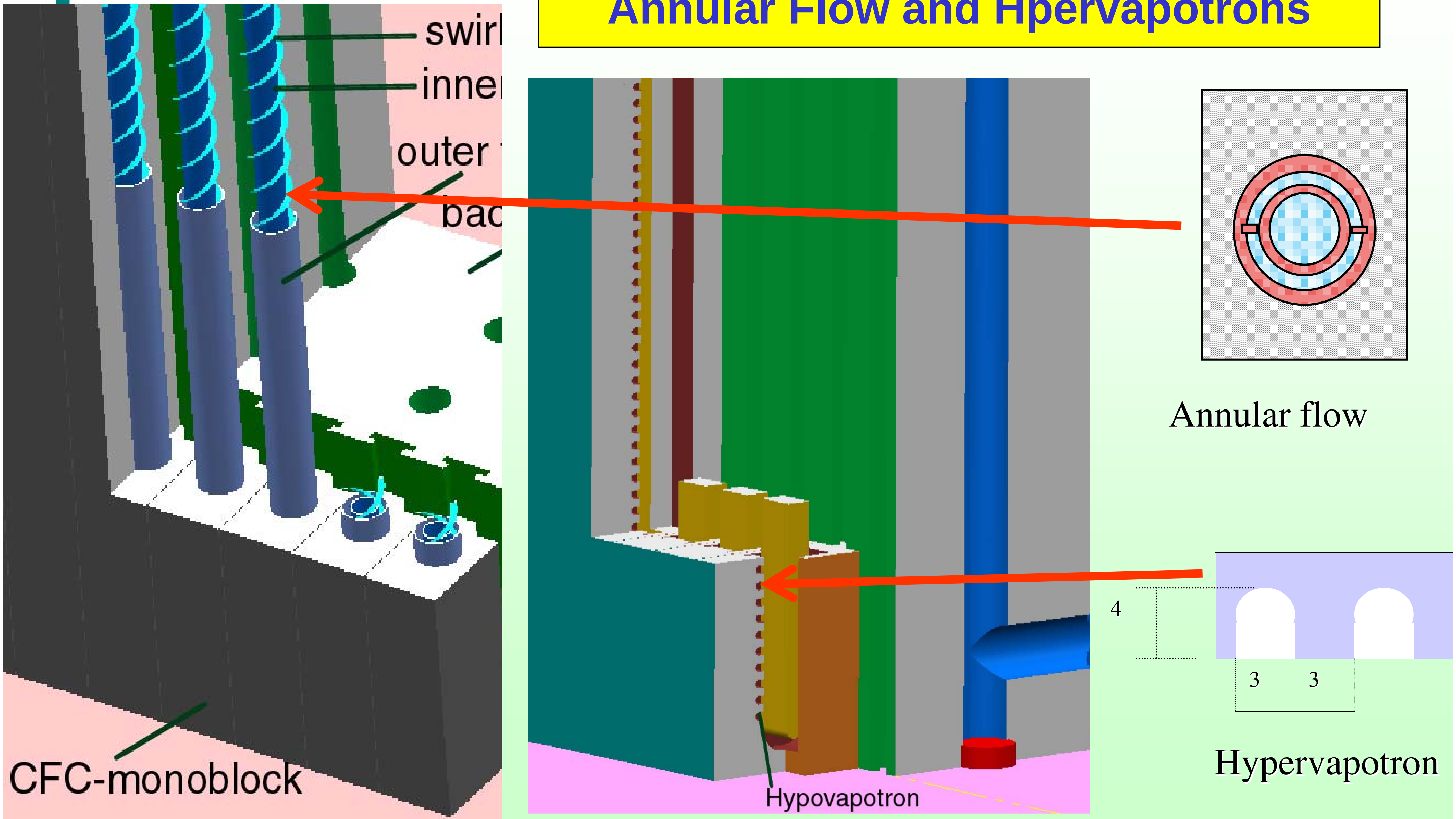
**Peaked heat profile in the tests was $P(x)=P_o\, e^{-x/\lambda}$, where the peak heat flux P_o and λ is ~50mm*

Values extrapolated to ITER coolant conditions tsub 120°C

†Results from all Cu mock-ups except CFC armoured swirl case.

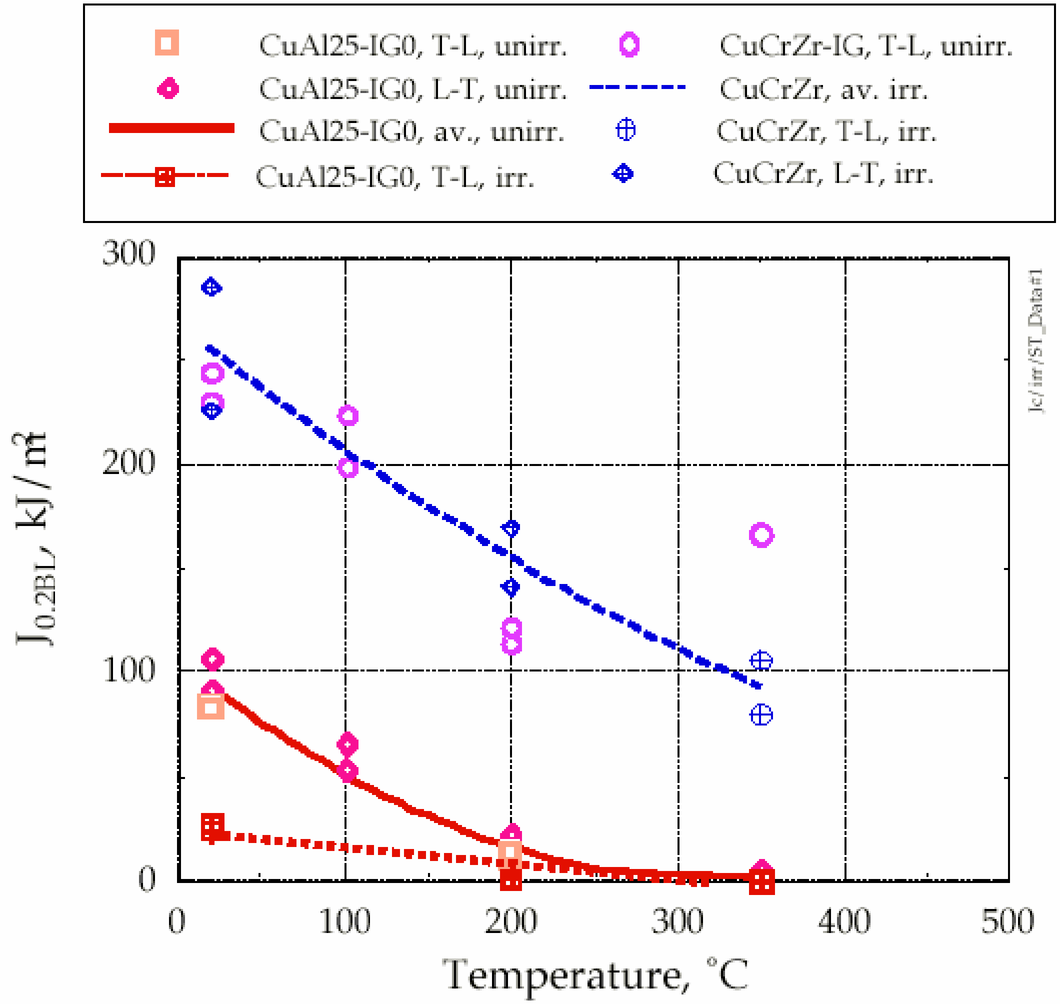
Ref. Except where stated Raffray et al; Fus Eng.&Des 45 (1999) 377-407

Annular Flow and Hpervapotrons



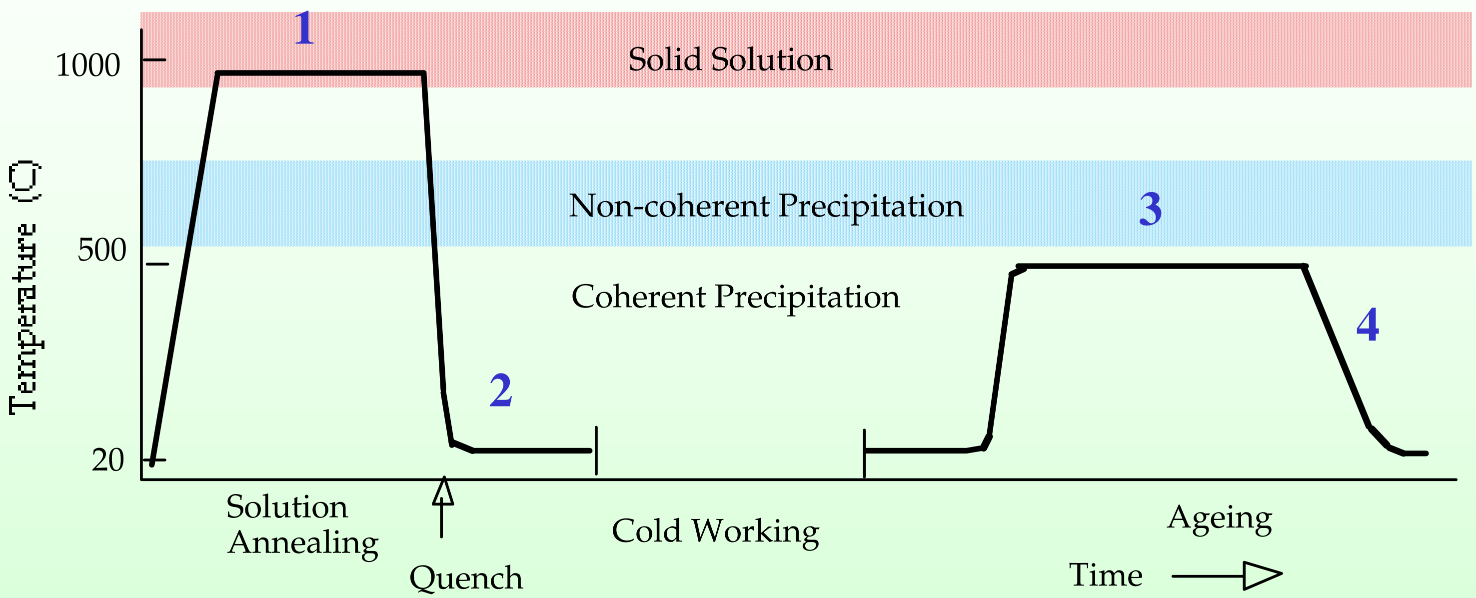
Heat-sink Alloy Selection

- **The precipitation hardened alloy CuCrZr is preferred to dispersion strengthened Cu, because it has much higher fracture toughness.**
- **CuCrZr can be sourced from several suppliers**
- **CuCrZr can be welded.**
- **For CuCrZr the manufacturing process has to be carefully controlled to ensure high thermal conductivity and good mechanical properties are achieved..**



Comparison of CuCrZr & DS Cu Fracture Toughness

Manufacturing cycle compatibility with CuCrZr heat-sink

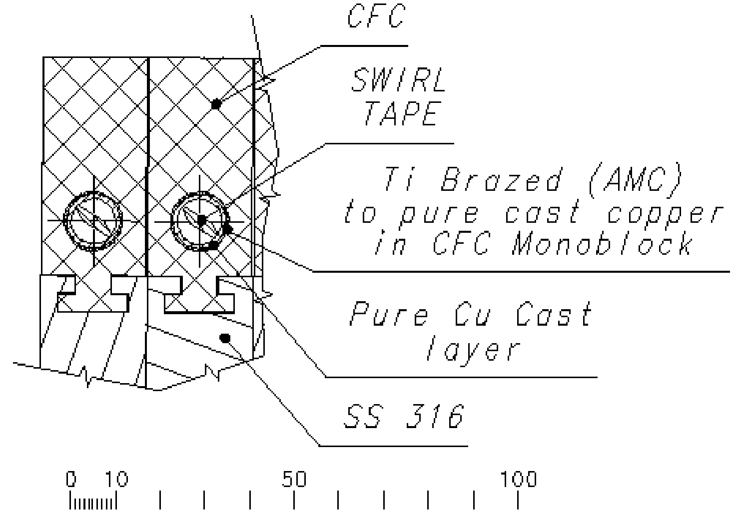


Examples

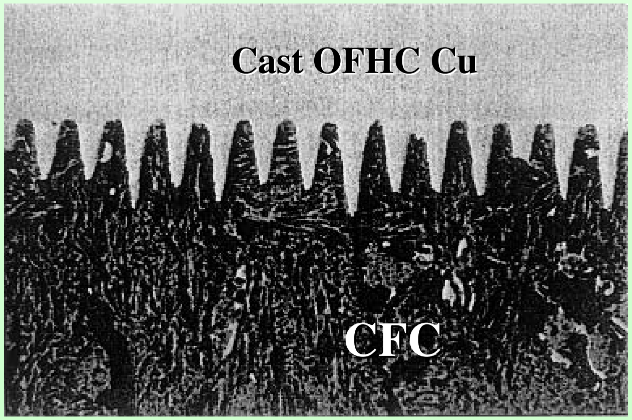
- **RF: W-Cu to CuCrZr joining at stage** 4, **by fast brazing T>500^{0}C, τ<5 min $\rightarrow$ 90% of optimum CuCrZr properties.**
- **EU: W-Cu, CFC-Cu to CuCrZr joining at** 3 **(HIP, 500^{0}C) $\rightarrow$ 100% of optimal CuCrZr properties.**
- **US: joining at** 1 **((HIP, 980^{0}C), followed by quench + ageing $\rightarrow$ 90% of optimal CuCrZr properties.**
- **JA Hot pressing of W rods at** 1 **(900°C 5MPa) followed by quench + ageing**

CFC Armour

- ITER has ~1.0x10^5 carbon-fibre composite tiles.
  - Required to operate @ 10MWm^{-2} for 1000 cycles of 400s, sustain 100 transients of 10s duration & heat flux 20MWm^{-2}.
  - A twisted (swirl) tape inserted in the tube promotes turbulence & provides margin to the critical heat flux of ~1.5.
  - In testing CFC monoblock armoured high heat flux components for ITER have survived >1000 cycles of 20MWm^{-2}.
- CFC to CuCrZr joining using AMC® process :
  - Laser conditioning of CFC;
  - Ti coating is applied to the surface to aid wetting;
  - Vacuum casting of pure Cu layer onto Ti/CFC;
  - Machining of bore to correct size;
  - Monoblocks are HIPed or brazed to CuCrZr tube.



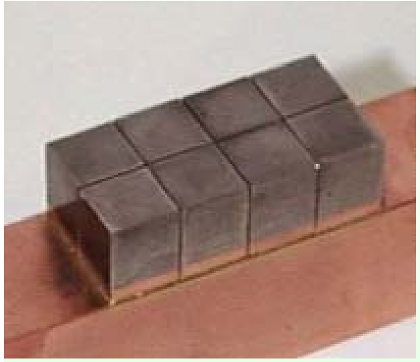
Cross-section CFC monoblock



Detail of AMC ® Joint

Tungsten Armour

ITER has ~$100 m^2$ W armour, say $2.5x10^5$ tungsten tiles ($20mm^L$ x $20mm^W$ x $10mm^H$)
Required to operate up to 1-2 MWm^{-2}. Several options are available:

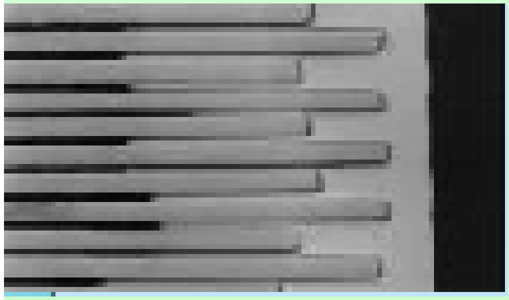


Tungsten tiles brazed on CuCrZr heat sink

Machine (diamond grind) cubes 12 x 12 x 12 mm;
Stack cubes to form tile and cast pure Cu layer;
Chemically etch residual Cu from gaps between cubes.;
Final machine tiles to size.
Survived cycling at $27MWm^{-2}$

Hot pressed tungsten pins (900°C & 5MPa).

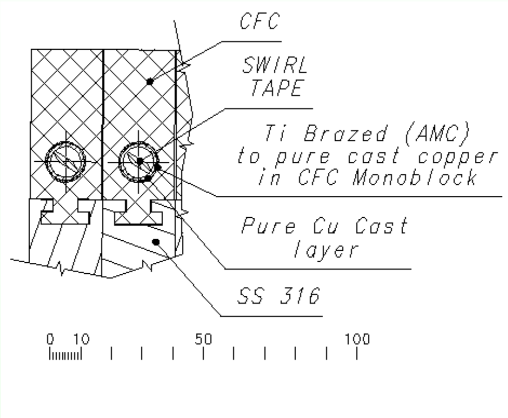
Survived 100 cycles at $15MWm^{-2}$ & 3000 cycles at $10MWm^{-2}$

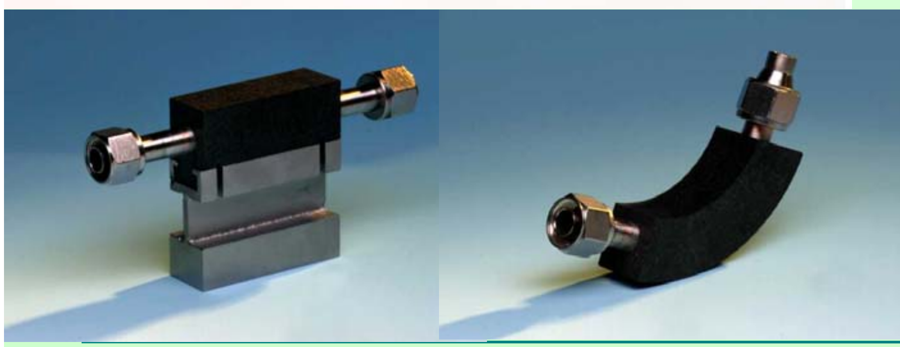


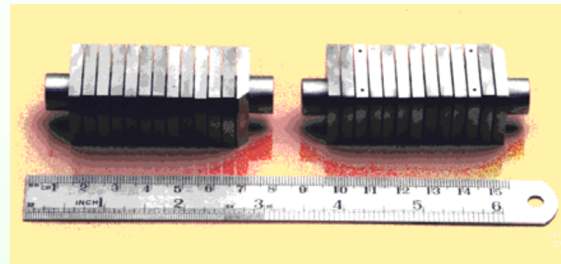
Tungsten lamellae

Cast pure Cu layer, HIP'ed to coolant tube.
Survived cycling to $20MWm^{-2}$

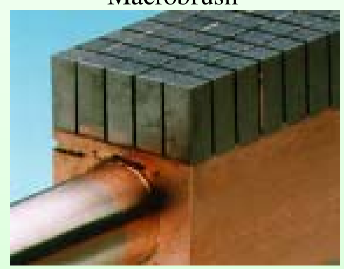
CFC Monoblocks and W brush / Lamella Mock Ups



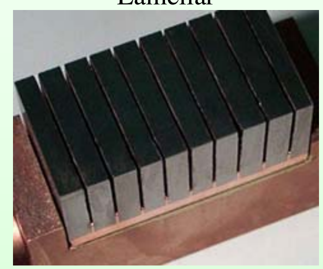




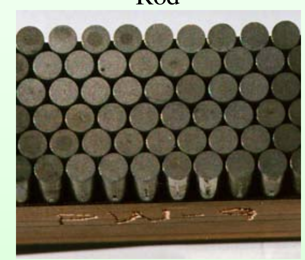
Macrobrush

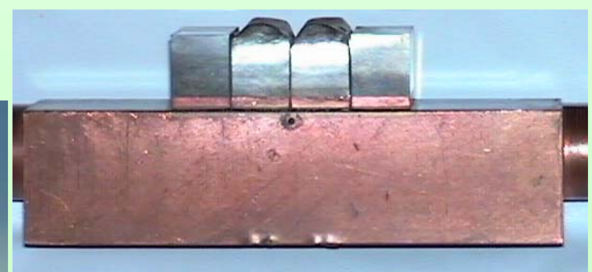


Lamellar



Rod



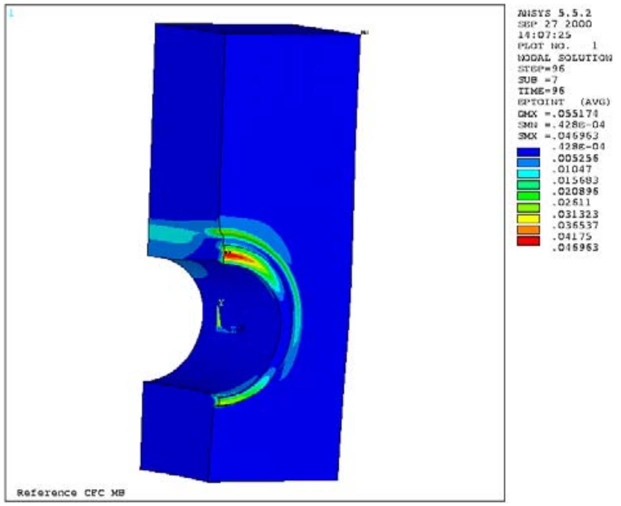


Qabs = 43MW/m2, 10-15s, 2 cycles
Qabs = 27MW/m2, 10s, 1500 cycles

Fatigue damage estimation for reference design CFC on CuCrZr tube

The armour and Cu heat sink performance uses a design by experiment approach.
The design of the CuCrZr tube (10 ID, 12 OD) is supported by fatigue and fracture mechanics analyses.

Type of strain cycle	design number of cycles, n	strain range, $\Delta\varepsilon_{tot}$, %	allowable number of cycles, N_{allow}	Fatigue usage fraction
First 9 cycles 0 => 10 MWm^{-2} => 0	9	0.204	90054	1.0×10^{-4}
First cycle 0 => 20 MWm^{-2} => 10 MWm^{-2} =>0	**1**	**1.219**	**41**	**0.024**
Subsequent cycles 0 => 20 MWm^{-2} => 10 MWm^{-2} =>0	299	0.230	54320	0.002
Subsequent cycles 0 => 10 MWm^{-2} => 0	2691	0.112	1.180×10^{6}	0.006
Cumulative fatigue usage fraction,V				**0.032**

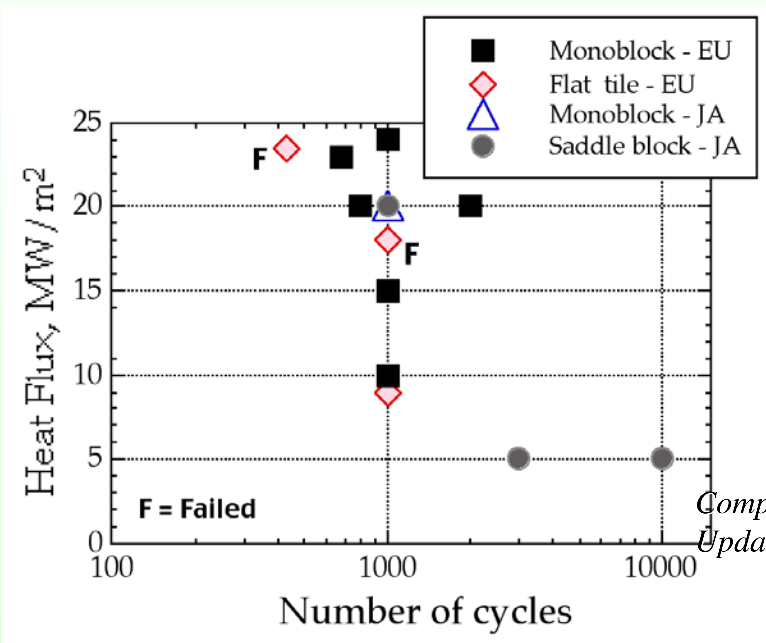


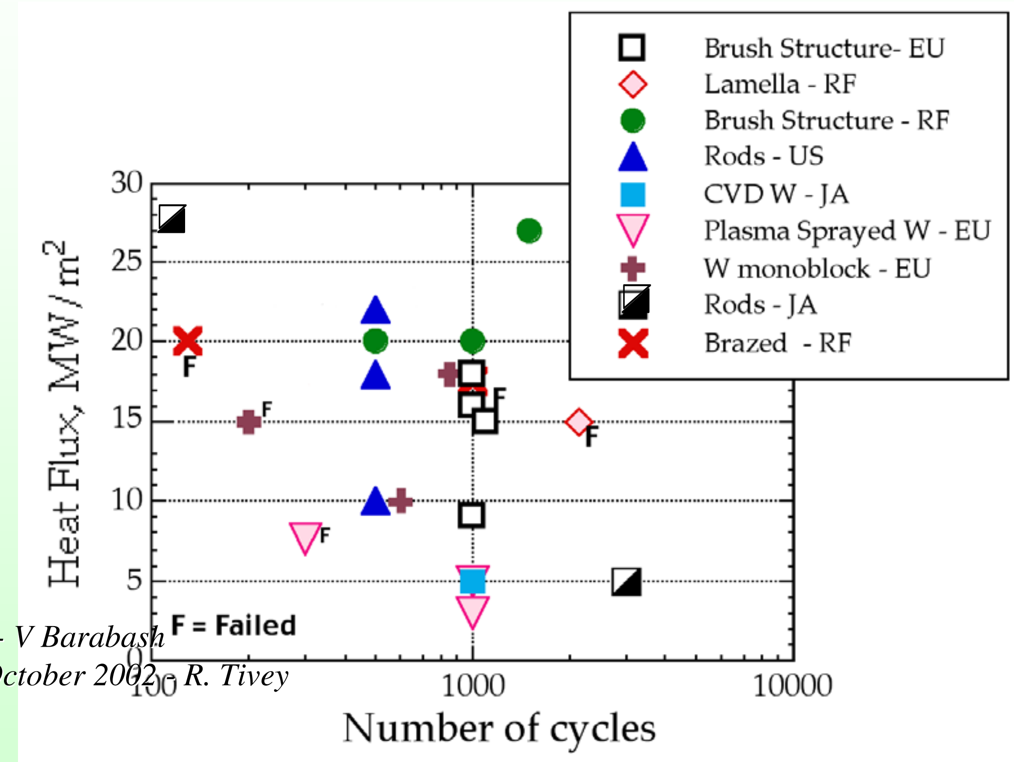
Note usage factor based on 3000 cycles & includes a safety factor of x 20.
*Values are for CuCrZr properties obtained using either HIP-ing (EU) or a fast brazing cycle (RF)***.**

- *Apply low Temperature HIP (480°C, few hours) during the ageing treatment of the CuCrZr*

"Fast brazing" => use CuCrZr in quenched and annealed condition, but heat during joining up to ~ 800°C for short time (total time at T > 600°C ~ 5 min). Some over ageing occurs, however the mechanical properties are in the acceptable level (~90% of optimum).

High Heat Flux Test Summary





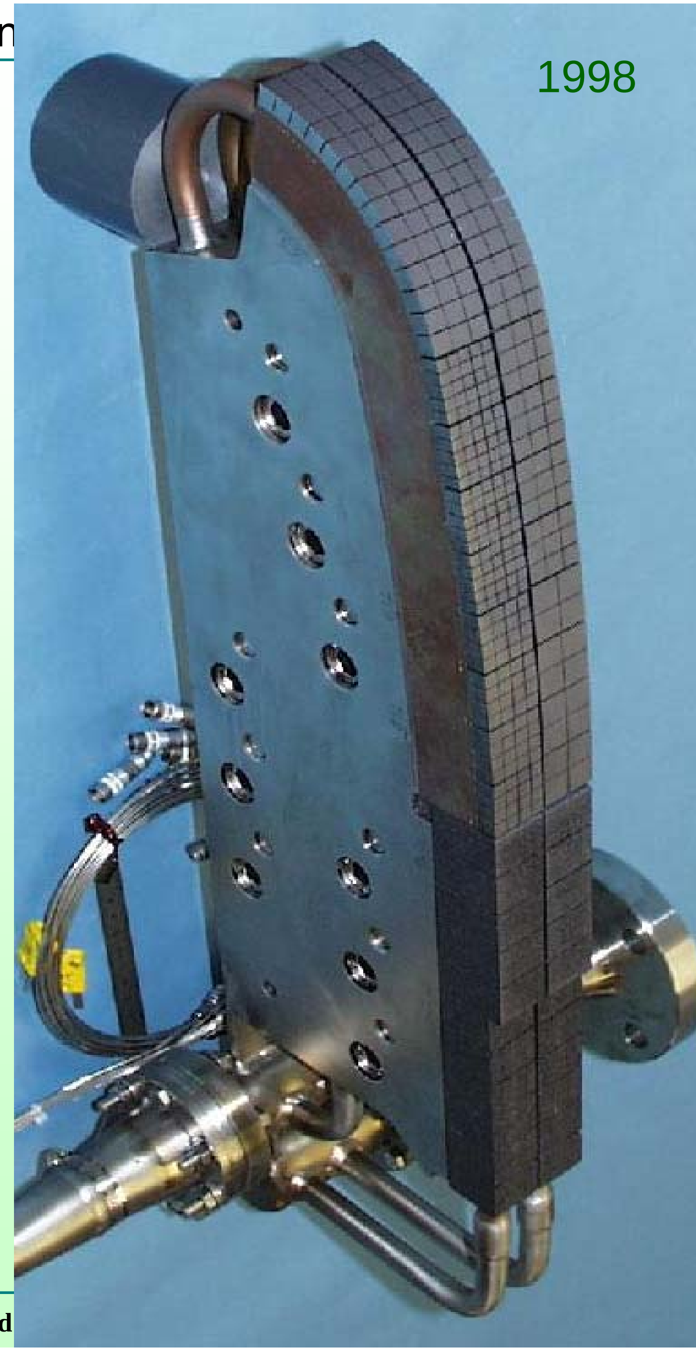
Compiled - V Barabash
Updated October 2002 - R. Tivey

CFC/Cu **W/Cu**

Note that the best results have been obtained using pure Cu as a compliant layer between armour and heat sink to accommodate differences in coefficient of thermal expansion for both between CFC and W armoured components. OFHC Cu has very high thermal creep rate & this is very useful for the stress relaxation.

Vertical Target
Medium Scale Prototype

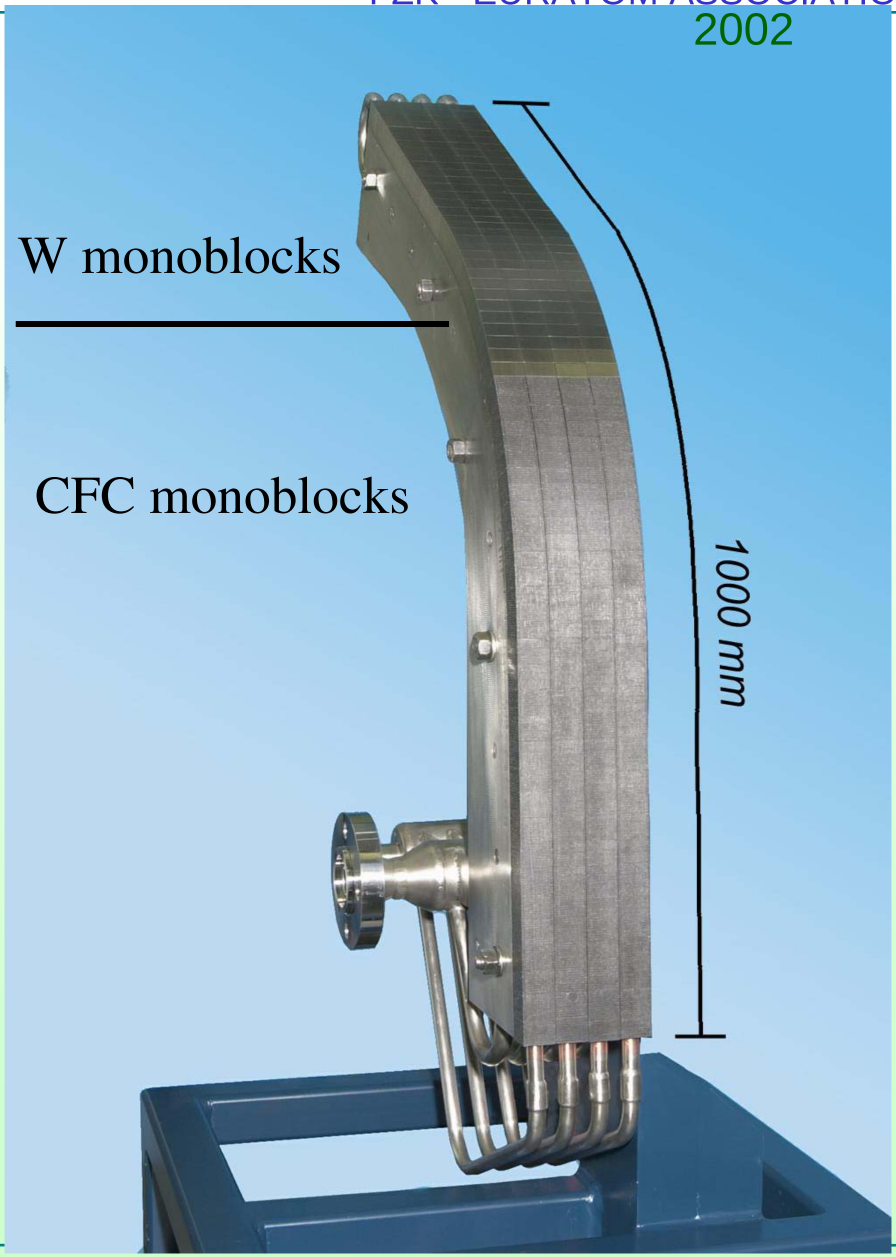
- W macrobrush:
  15 MW/m^2 x 1000 cycles
- CFC monoblock
  20 MW/m^2 x 2000 cycles
- CHF test > 30 MW/m^2



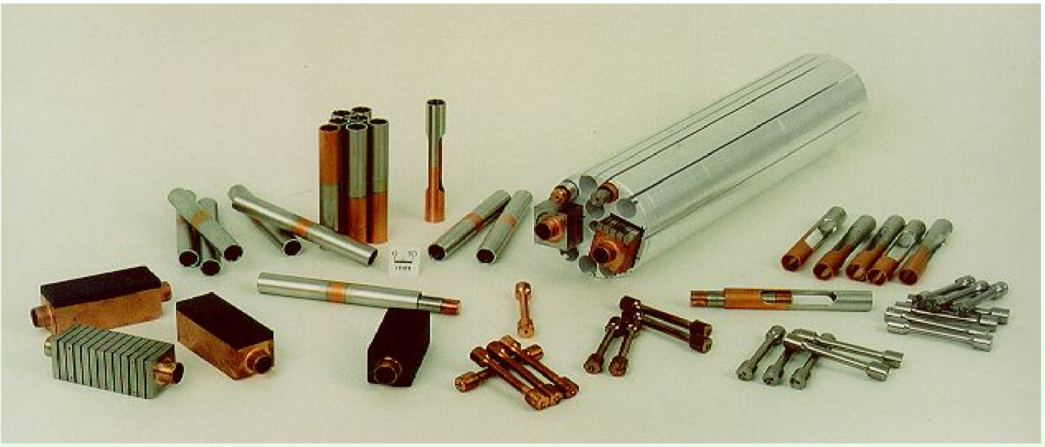
500 mm

Vertical Target
Full-Scale Prototype

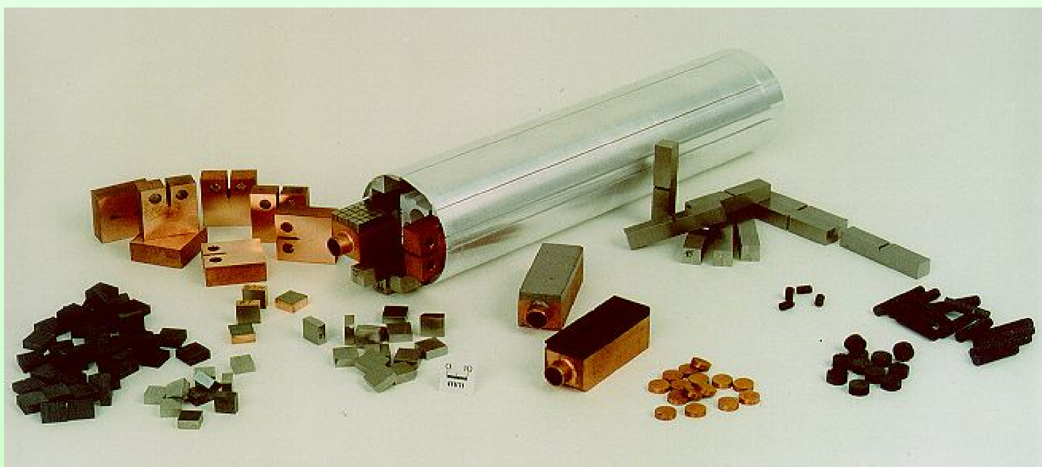
- Test in progress in FE200 facility at Le Creusot (F) (see talk F. Escourbiac)
- W monoblocks:
  10 MW/m^2 x 700 cycles
- CFC monoblock
  10 MW/m^2 x 1000 cycles
  20 MW/m^2 x 1000 cycles



Neutron-Irradiation Experiments PARIDE 1 - 4



(See talk of J. Linke)



PARIDE 1:
- temperature: 350°C
- target fluence: 0.5 dpa

PARIDE 2:
- temperature: 700°C
- target fluence: 0.5 dpa

PARIDE 3:
- temperature: 200°C
- target fluence: 0.2 dpa

PARIDE 4:
- temperature: 200°C
- target fluence: 1 dpa

Testing of W monoblock Mock-Ups

unrirradiated

- 1000 cycles x 10-14 MW/m^2– no failure
- 1000 cycles x 18 MW/m^2 – no failure, several mock-ups with different technologies
- 1000 cycles x 20 MW/m^2 – no failure

200°C, 0.1 dpa (in W)

- 1000 cycles x 10 MW/m^2– no failure
- 1000 cycles x 14 MW/m^2 – no failure
- 1000 cycles x 18 MW/m^2 – no failure

200°C, 0.5 dpa (in W)

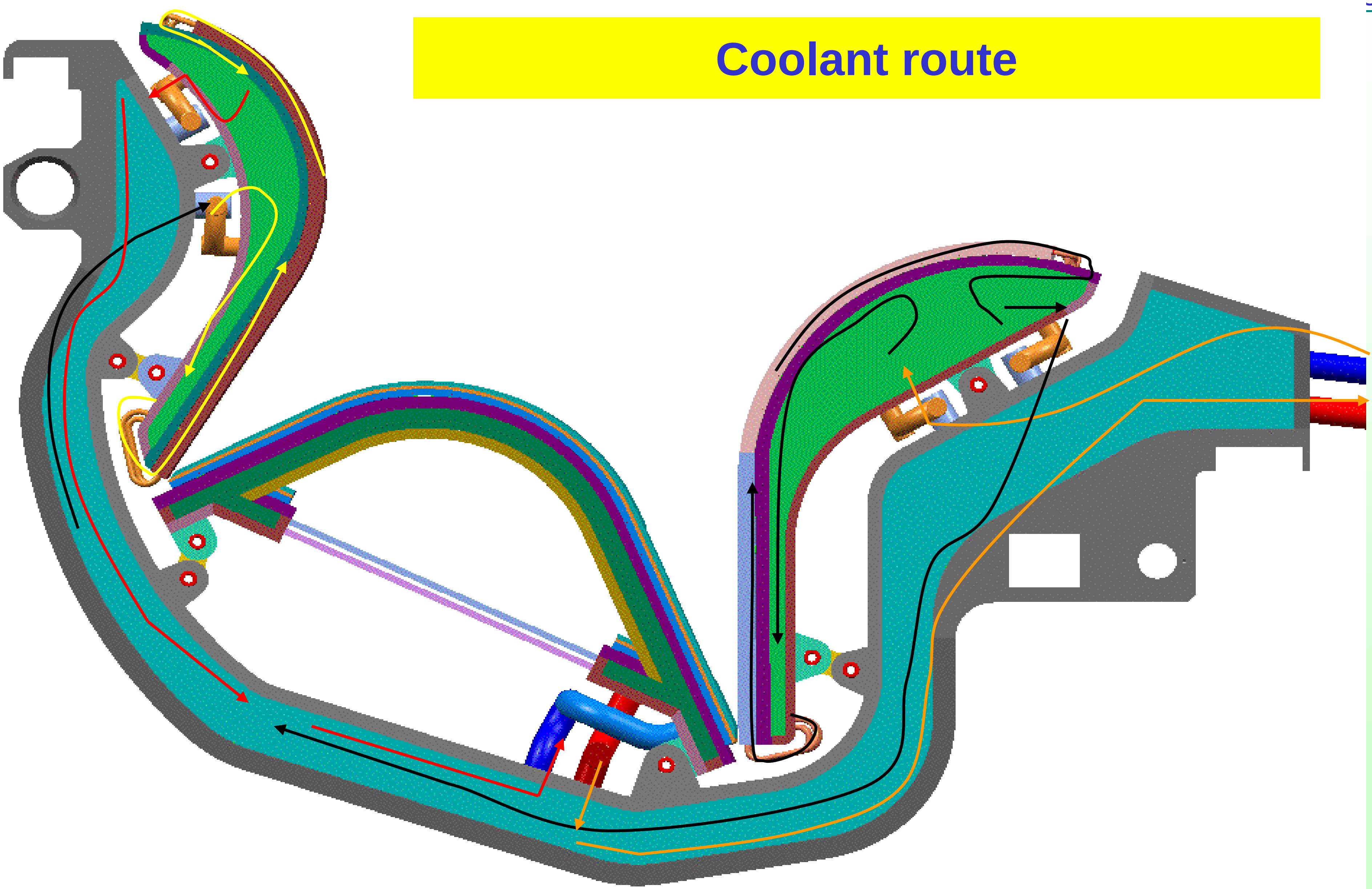
- 1000 cycles x 10 MW/m^2– no failure
- 1000 cycles x 14 MW/m^2 – no failure
- 1000 cycles x 18 MW/m^2 – no failure

The irradiated pure Cu interlayer leads to a reduction of the high heat flux performances in a flat tile geometry below the ITER requirements for the strike point region.

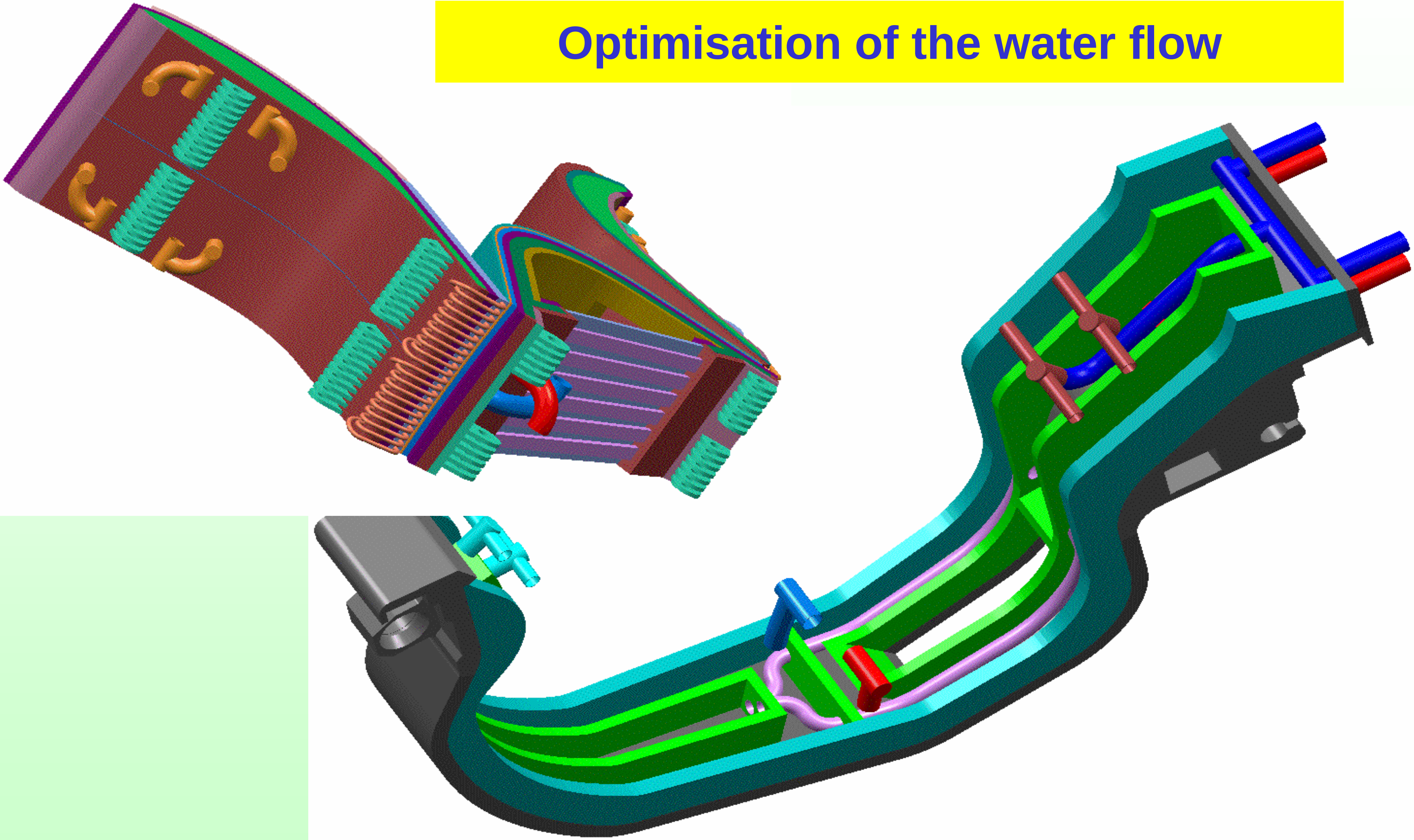
The monoblock solution appears therefore mandatory for this region.

Integration of a Divertor into a Reactor Class Machine

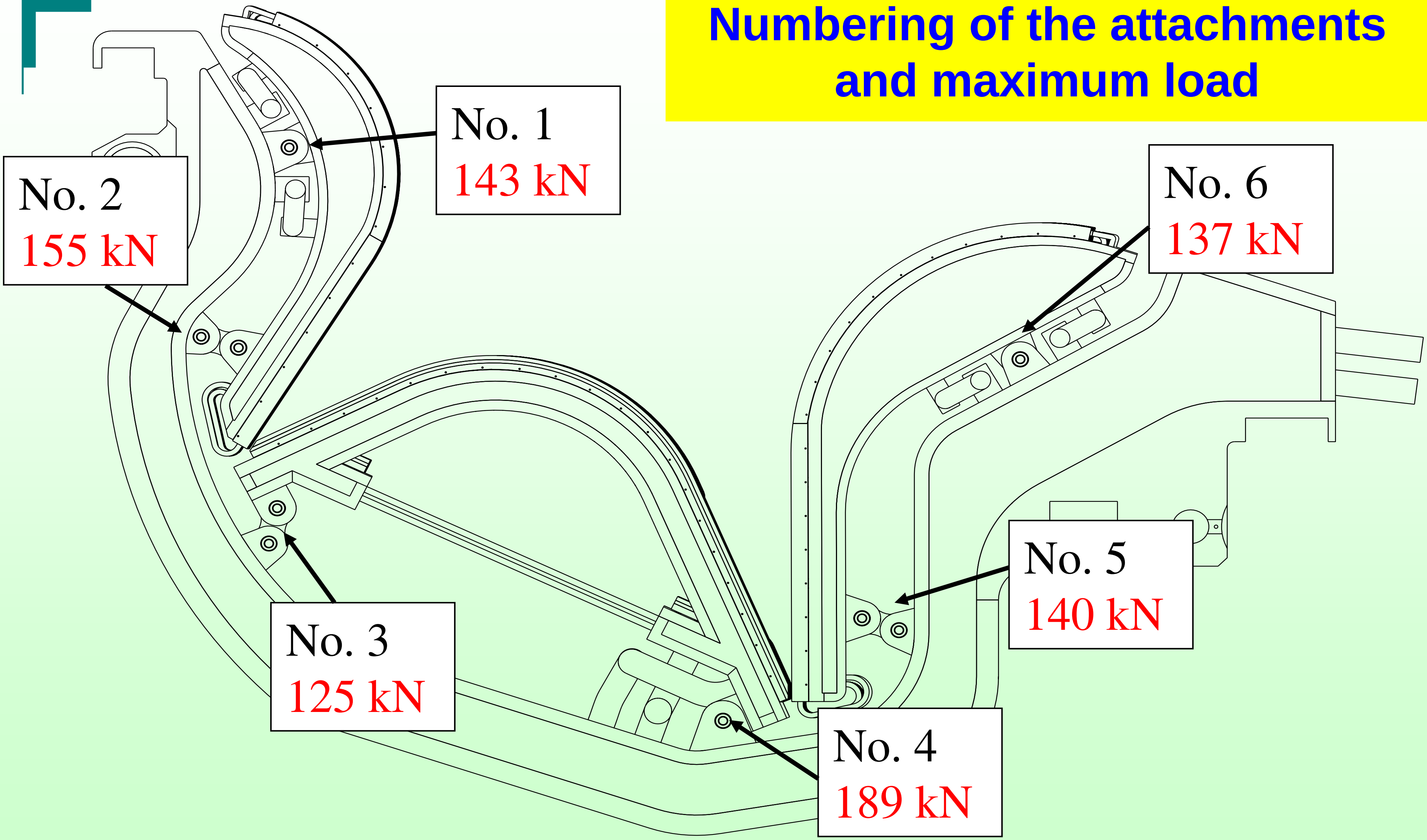
Coolant route



Optimisation of the water flow

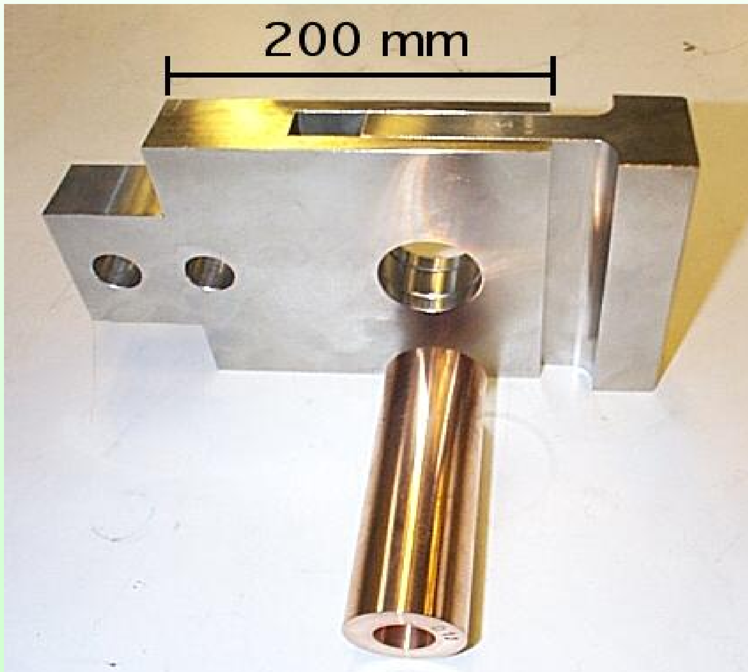


Numbering of the attachments and maximum load

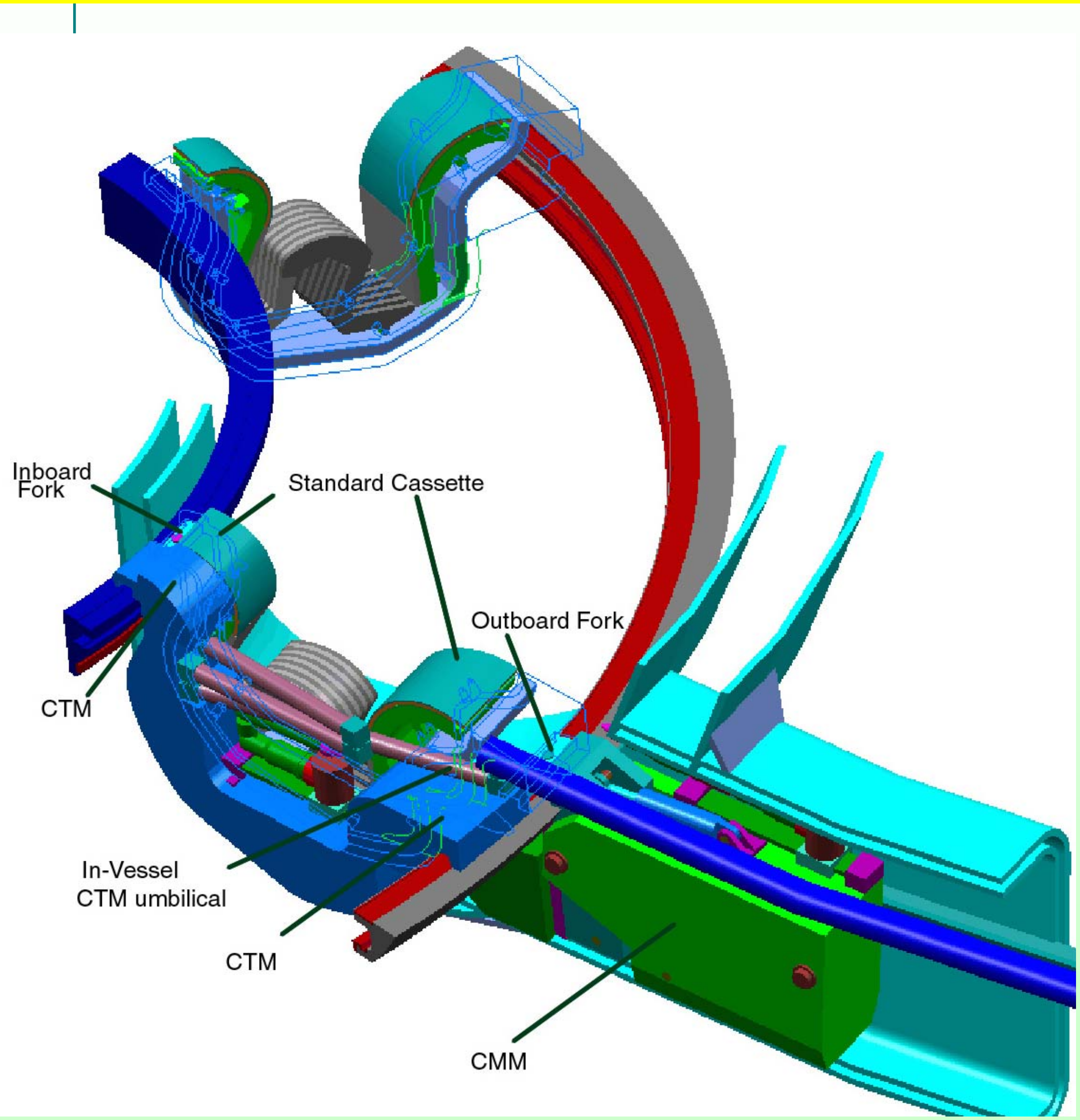


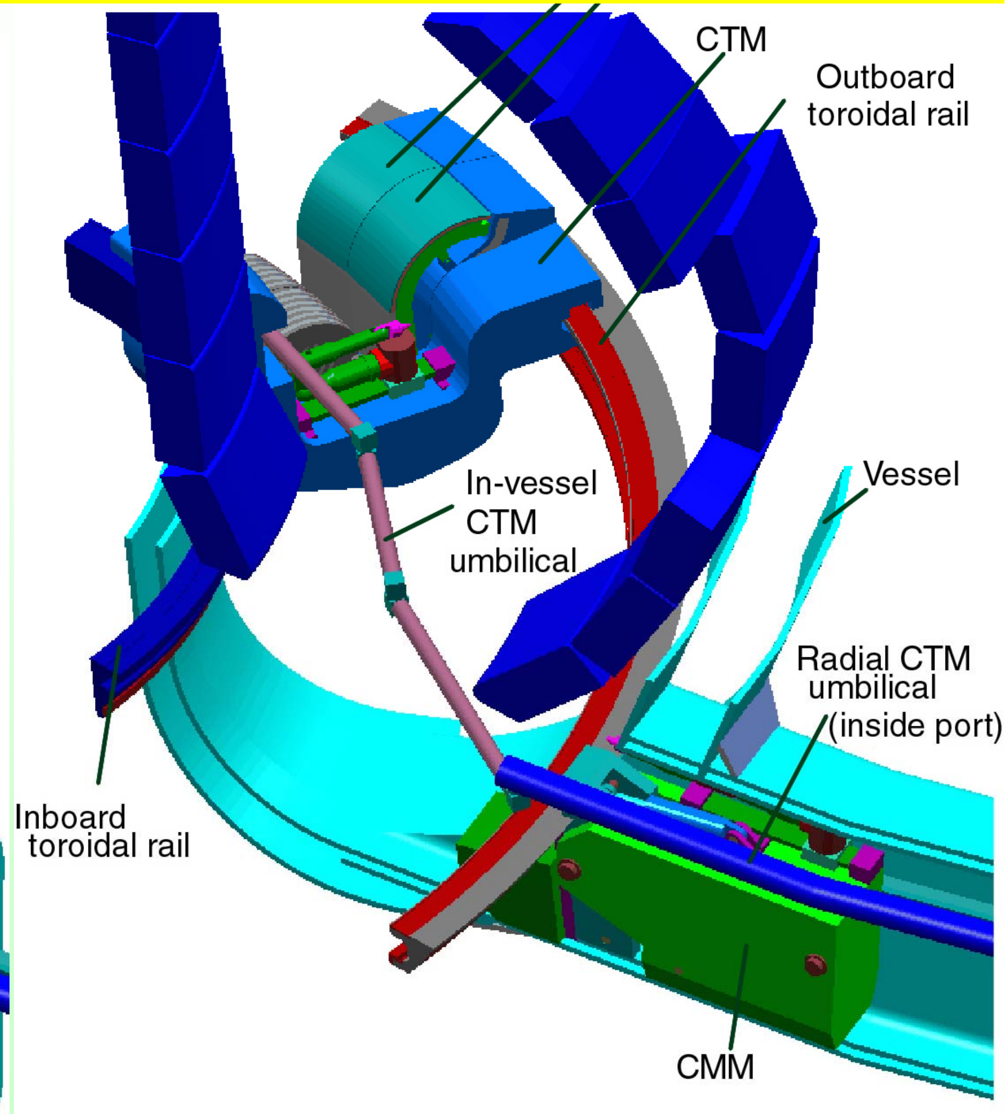
PFCs multilink Attachment

Mandrill to extrude pin

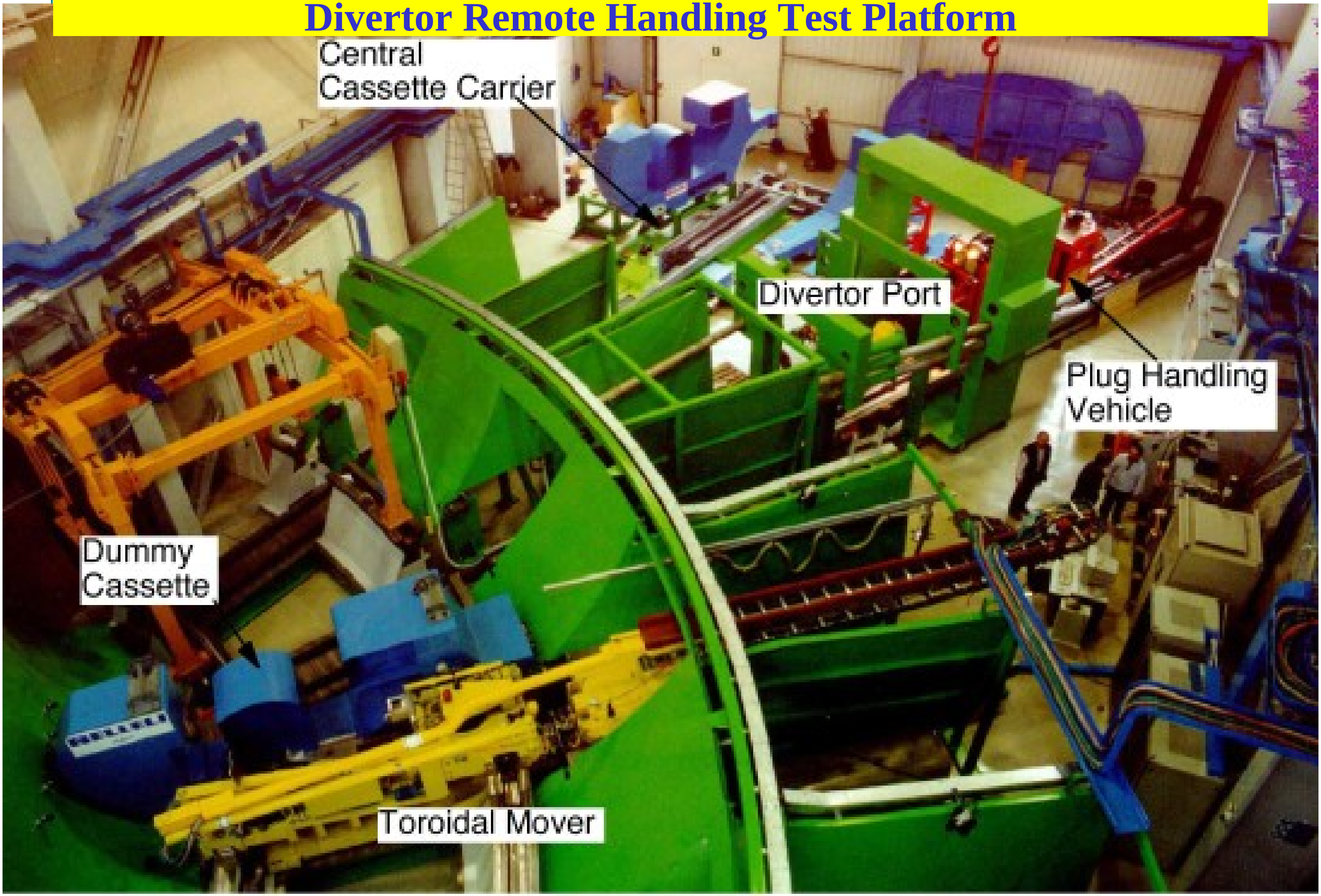


Remote Maintenance of the Divertor - cassette toroidal and radial mover

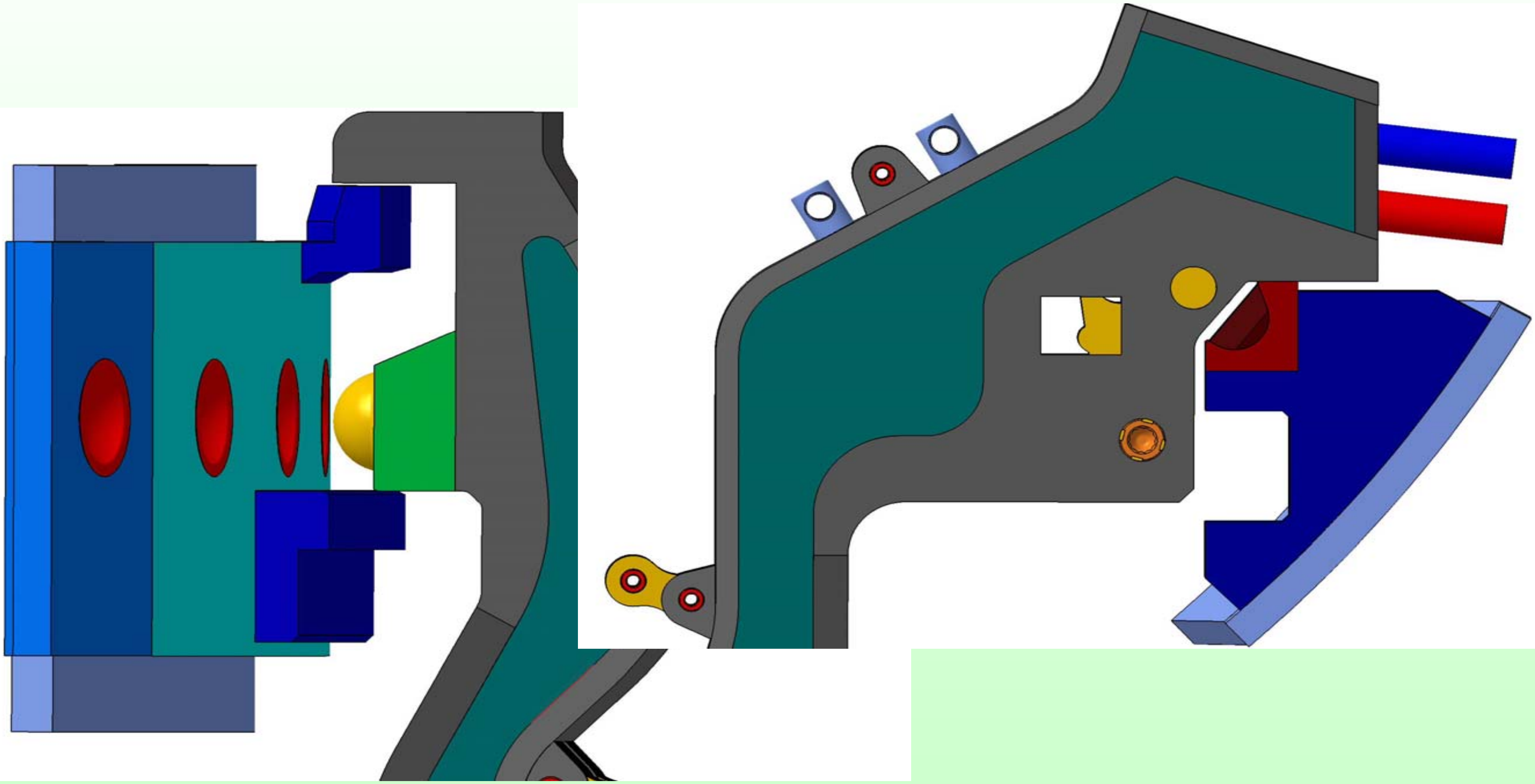




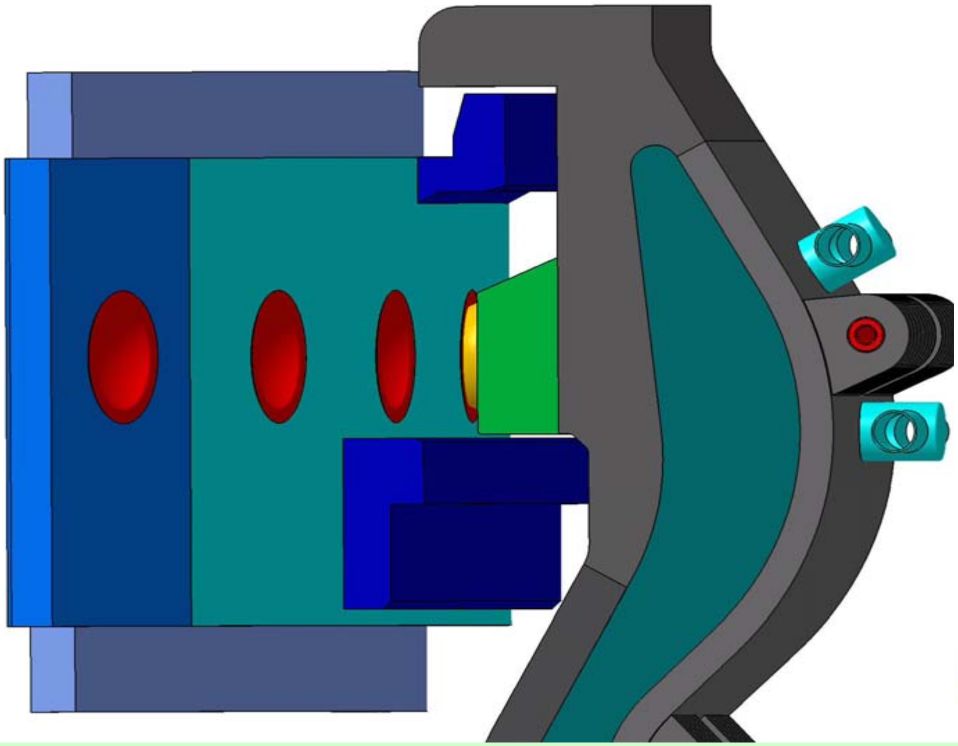
Divertor Remote Handling Test Platform

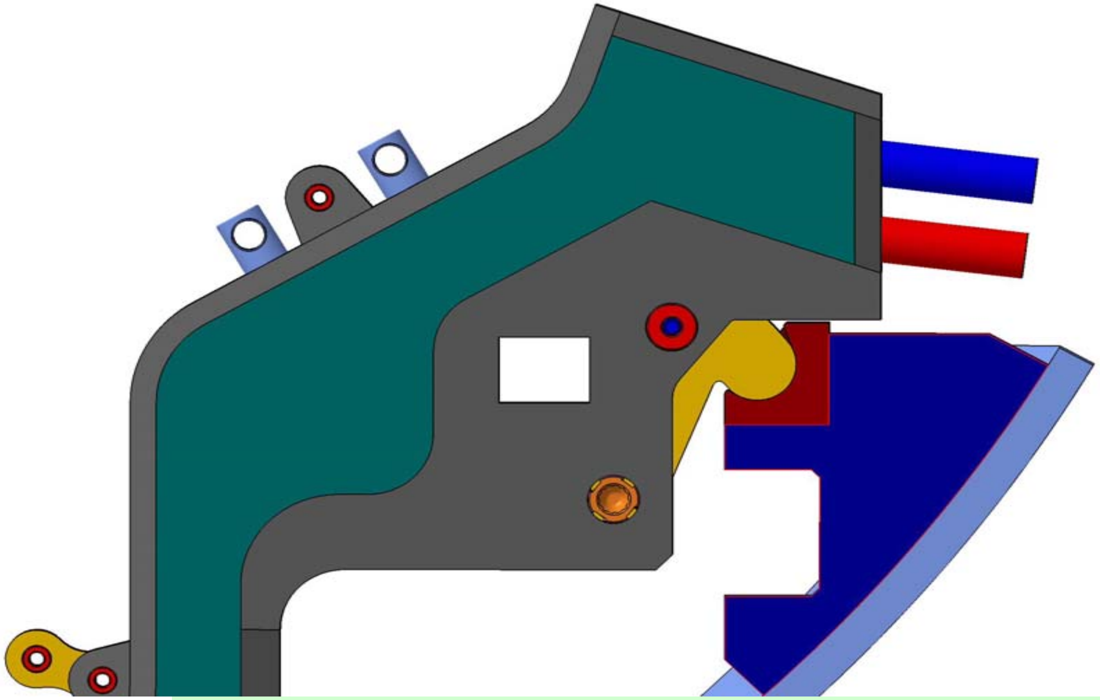


Step 1:
The CB is located in the toroidal position



Step 3:
The CB is preloaded

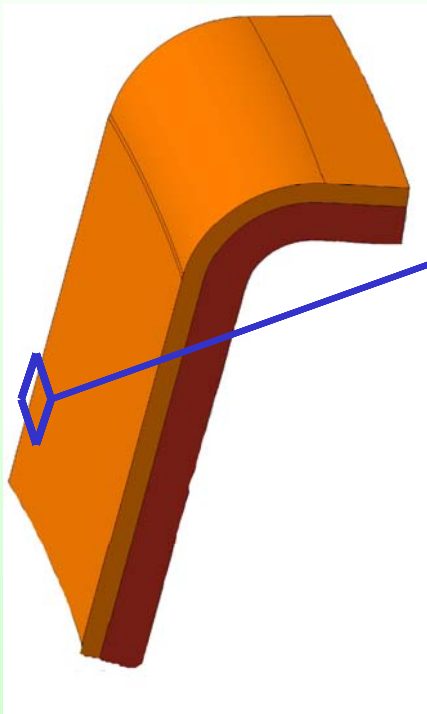




A He cooled Divertor

Target plates made of W-Armor Segments

Target Plate

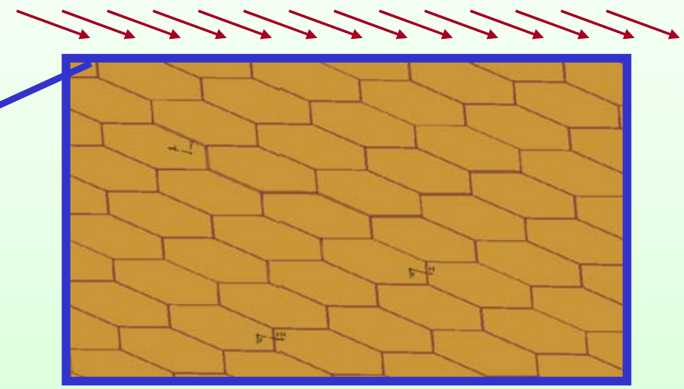


up to 4000
W-ARMOR
Segments

(for acceptable Thermal Stress in W-Armors & Cooling System)

Castellated Armor Layer

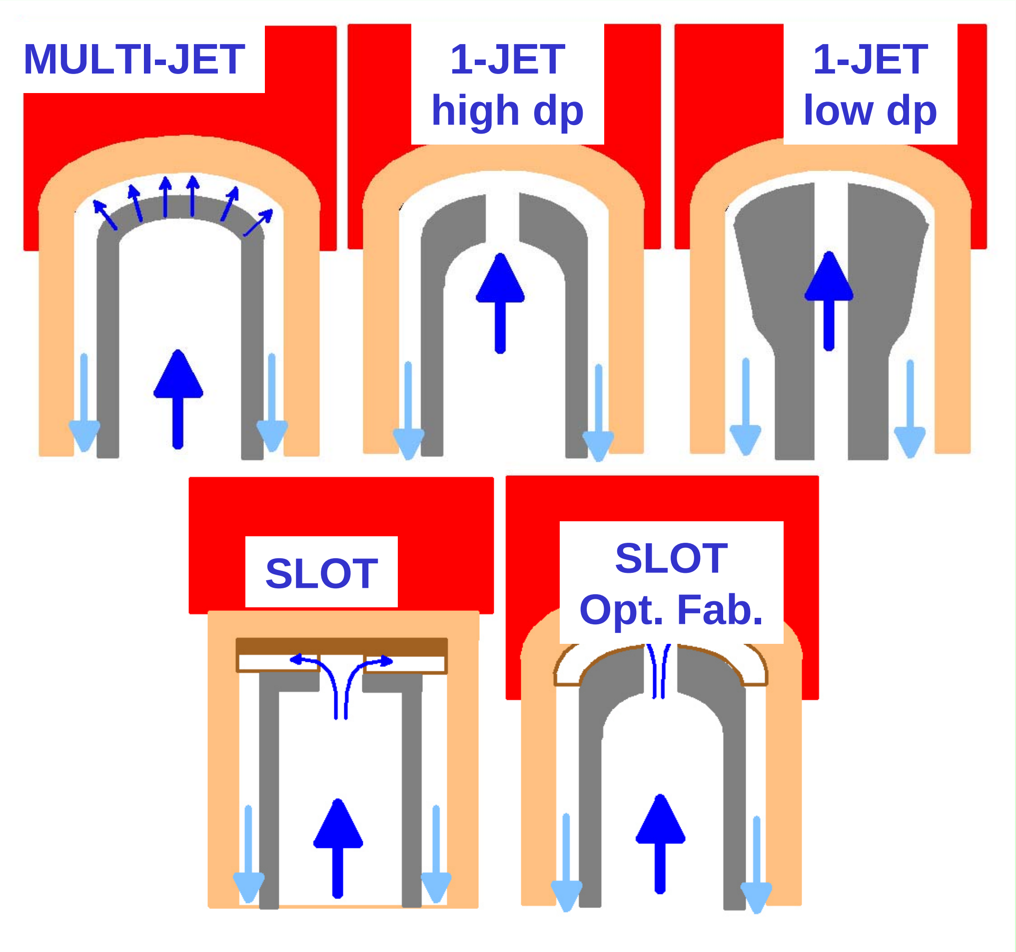
High Heat Load + Erosive Particle Flux



W-Armor:
- \+ High Melting Point
- \+ High Heat Conductivity
- \+ low Sputtering Rate
- \- need of low Impurity of W in Plasma (due to high atomic number of W)

HE-COOLING TECHNIQUE ?

Helium Cooling Systems



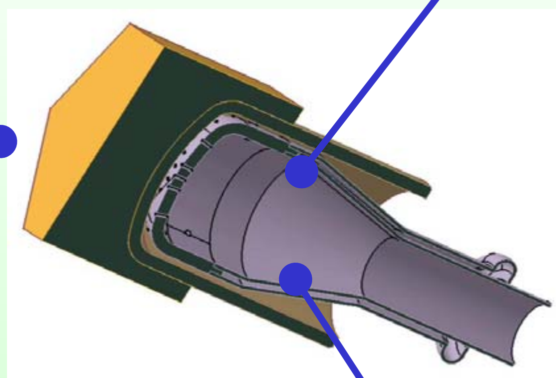
Jet Cooling

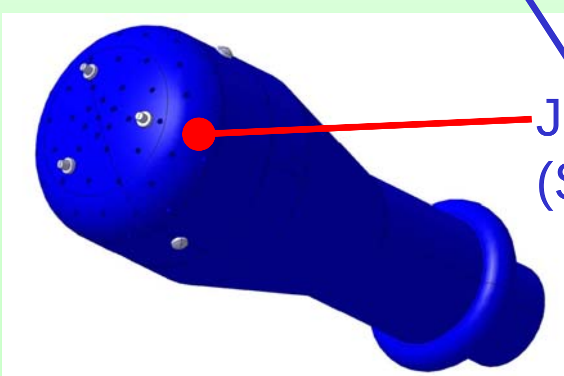
Design: Cooling Fingers for DEMO Divertor

Multiple-jet-cooling-system

Armor (W)

Cap (WLa10)

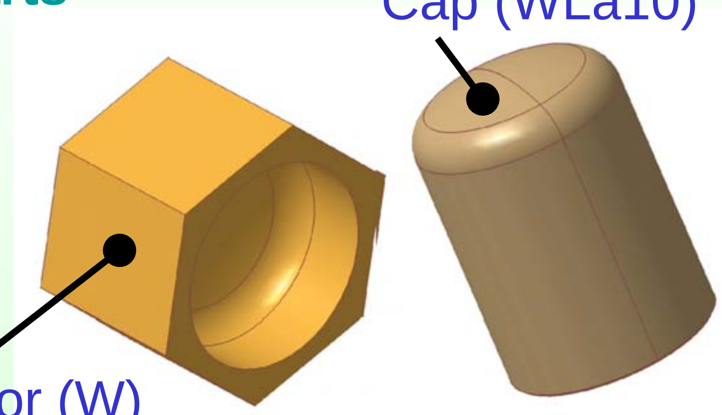




Jet Cartridge (Steel)

W-Parts

Cap (WLa10)



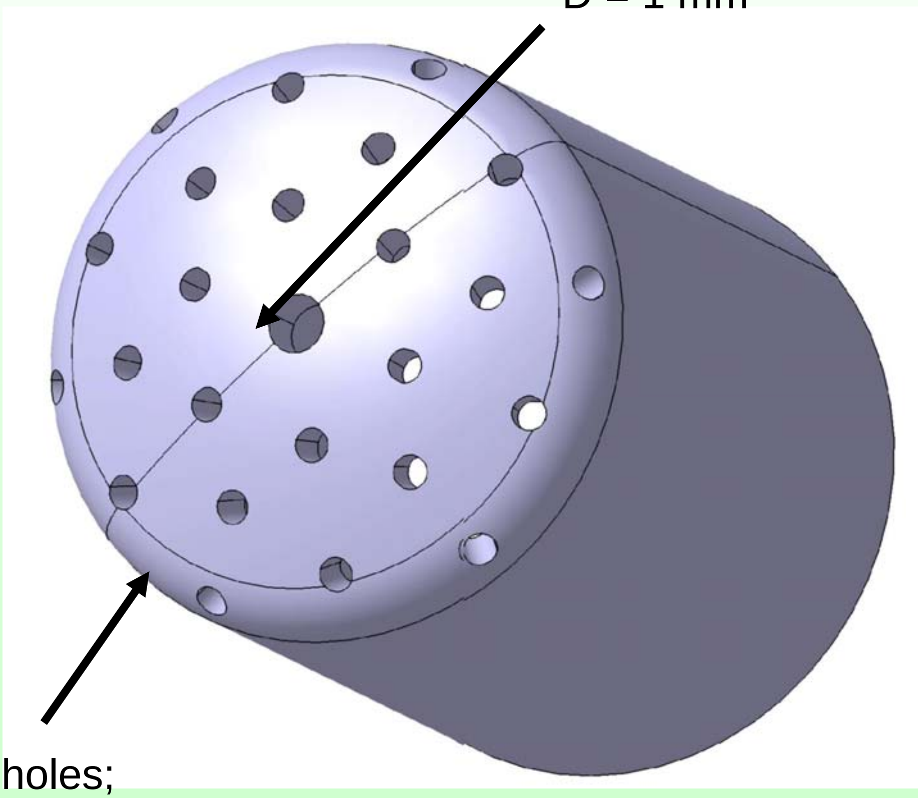
Armor (W)

- **Prevention of crack failures**
- **Prevention of bulge during operation**
- **Prevention of failures due to local stress peaks**
- **High Cooling Surface (Hex. Armor, Domed Caps)**

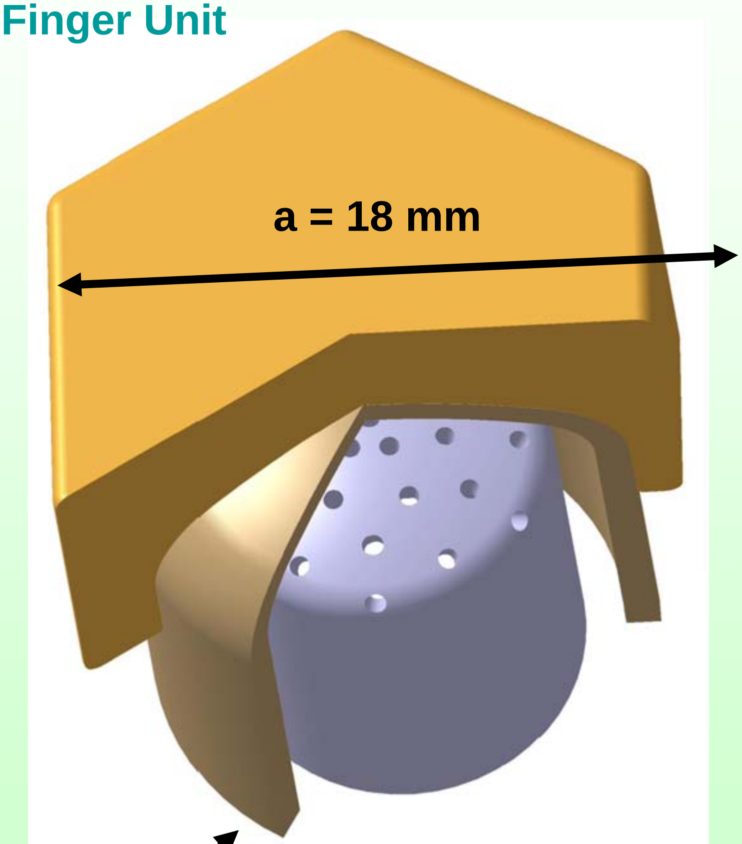
Jet Cooling

Layout example for Multiple-He-Jet-Cooled Divertor

Multiple Jet Cartridge



Finger Unit



HEMJ cartridges for simulation experiments

HEMJ 0.6-2

HEMJ 0.6-2 J1-b

HEMJ 0.6-2 J1-c

HEMJ 0.6-2 J1-d

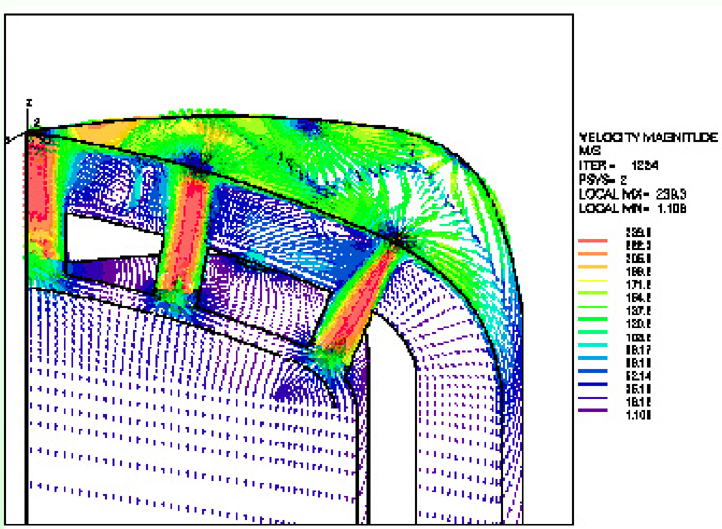
HEMJ 0.6-2 J1-e

Tube for mockup

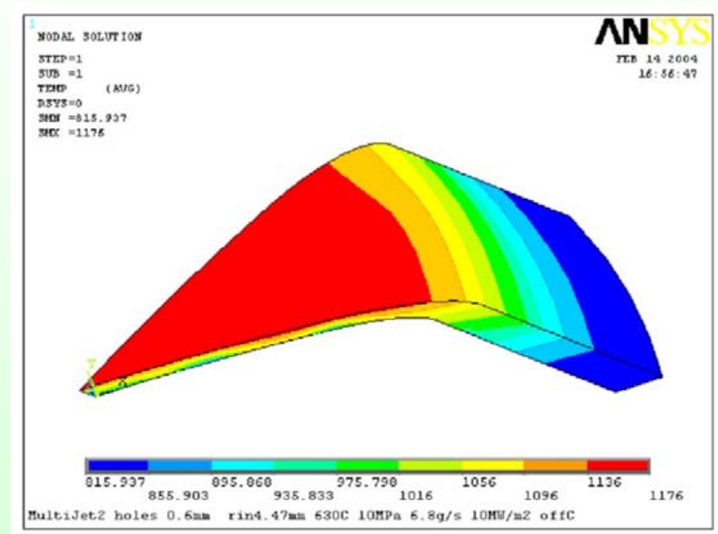
Jet Cooling

CFD-Results for layout example, (Efremov)

Flow Velocities



Temperatures in Cap



B.C.: **<u>Helium, 10 MPa, T_in 630 °C, heat load: 10 MW/m²:</u>**

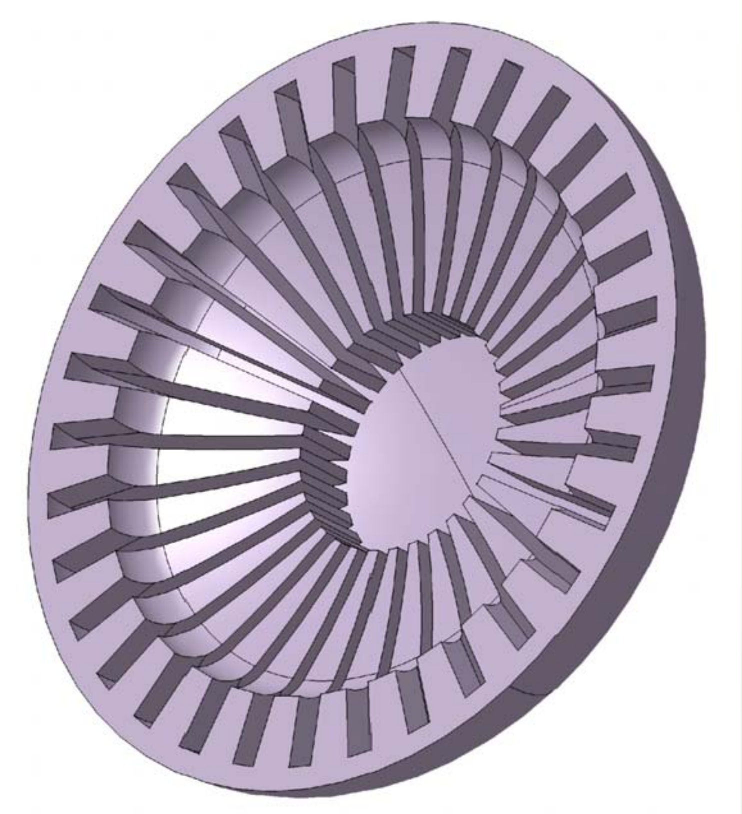
Results:

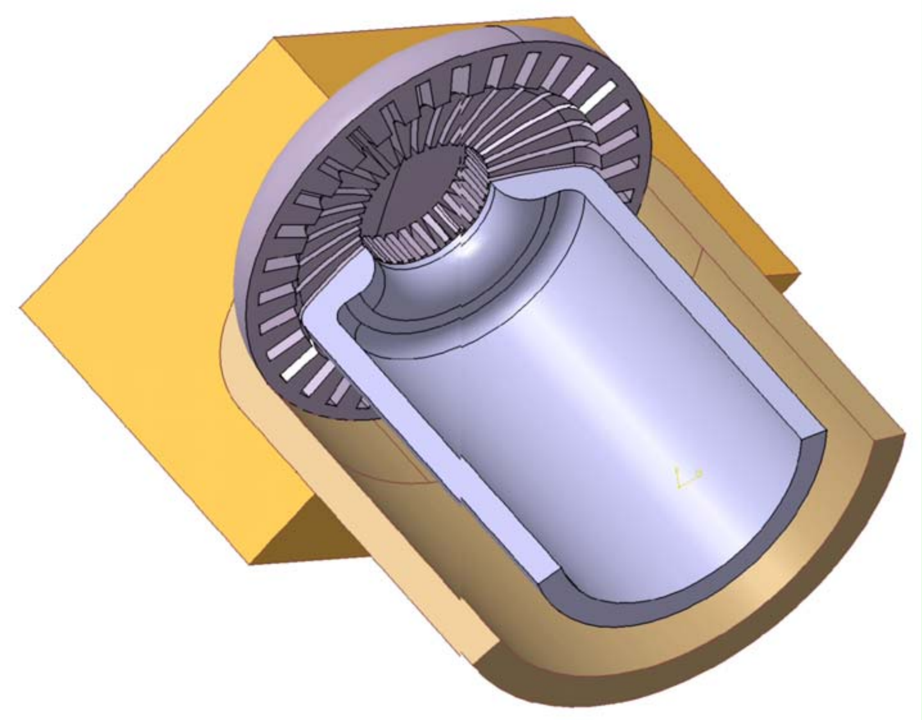
- **max. Temperature in W-Cap: 1176°C (< 1300°C)**
- **Pressure drop: 0.15 MPa**
- **Mass flow per finger: 6.8 g/s**

Flow Promoter

Optimization of slot concept (HEMS-C): Domed slot array

Domed slot array





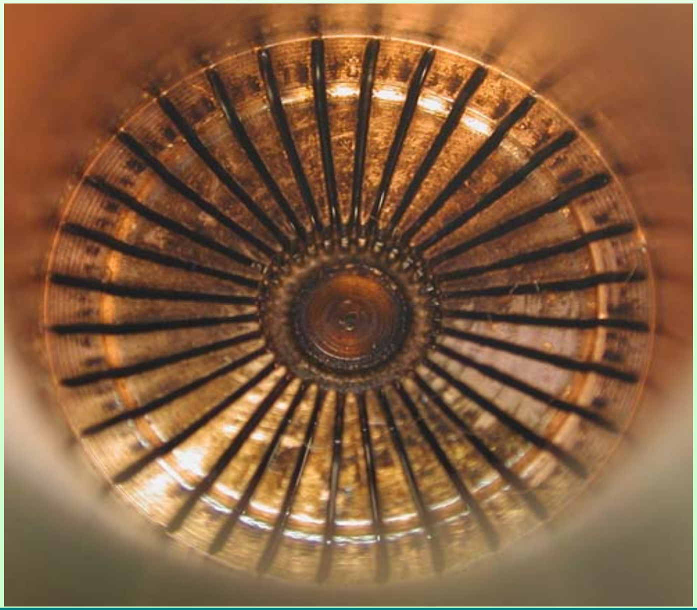
finger unit with
domed slot array

EDM manufacturing of CuCrZr mockups

- Roughness Ra~0.4 mkm
- EDM duration ~ 40 hours
- Slot width 0.2 mm

H=1.2 mm

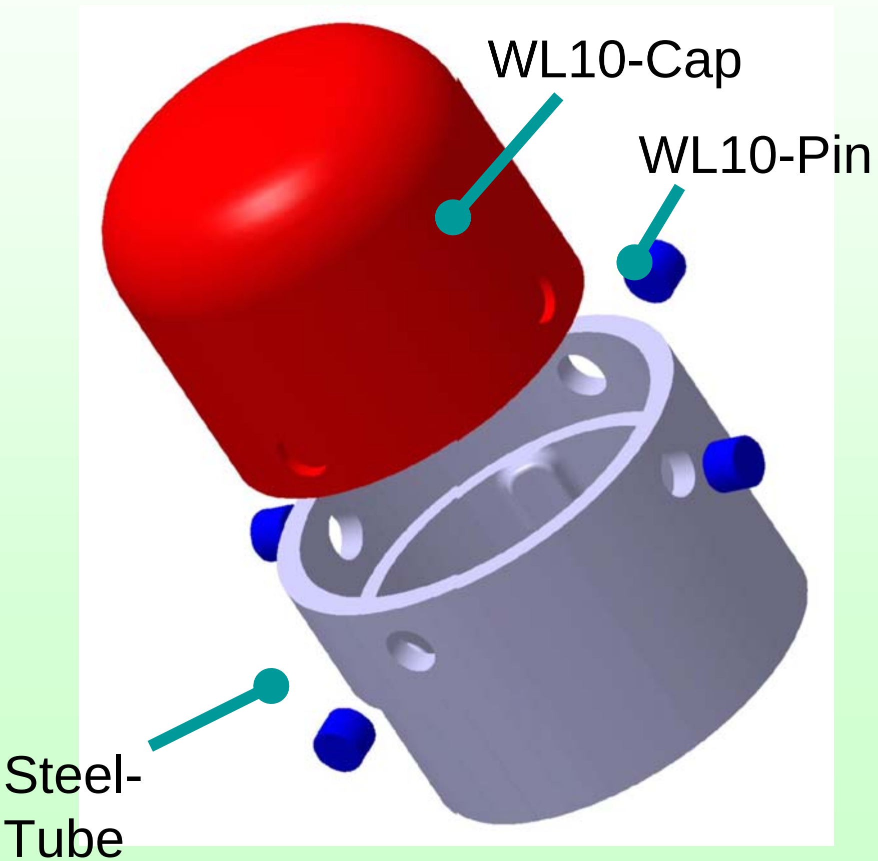
H=1.7 mm



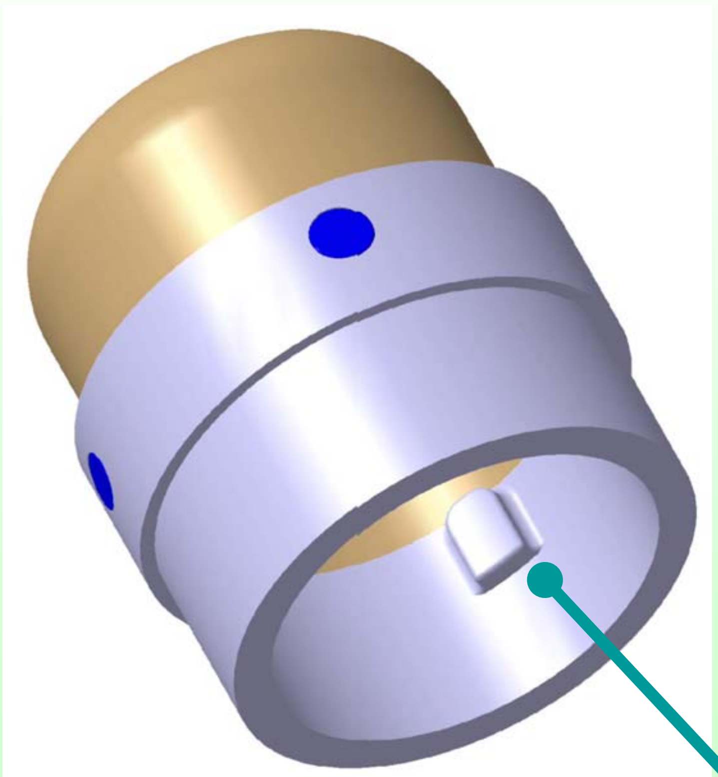
New Design

W-Cap – Steel Joint

W-Cap – Steel-Tube Joint

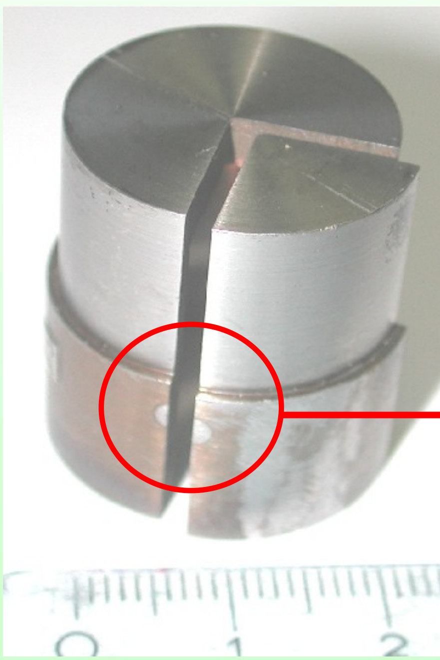


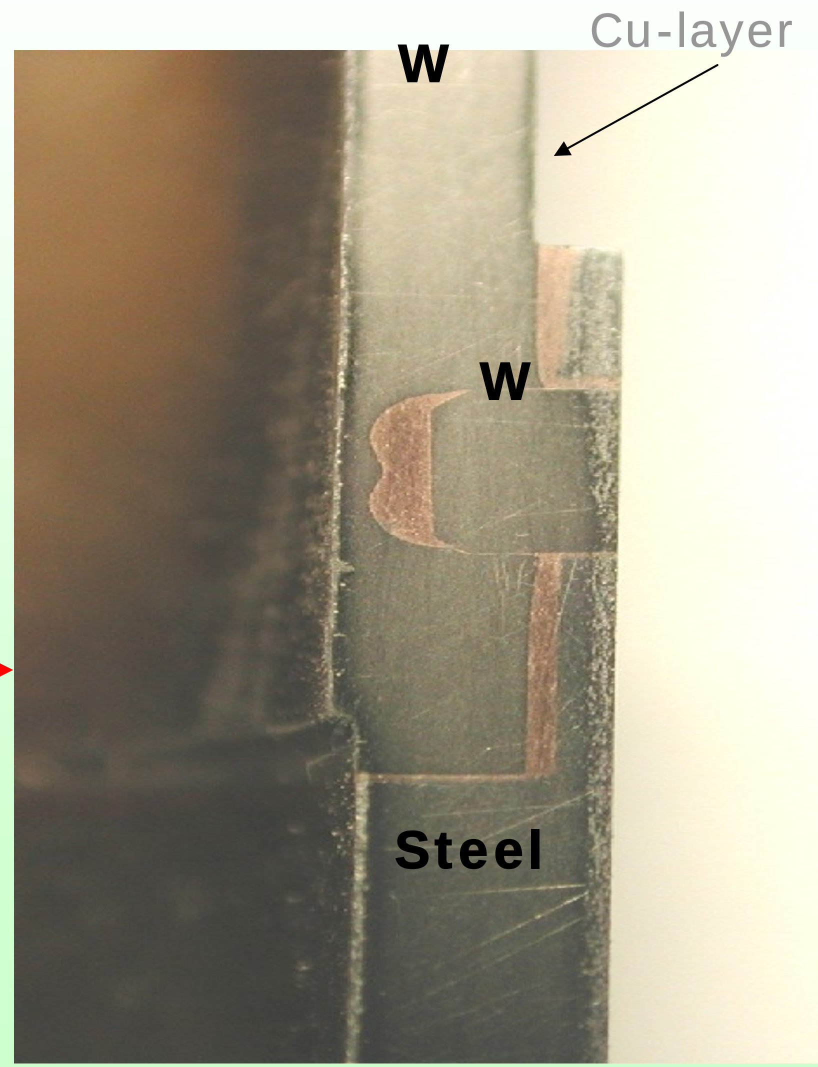
Cap/Tube with Spacers for Cartridge



Current results (pins option)

Cross-section of lock-area

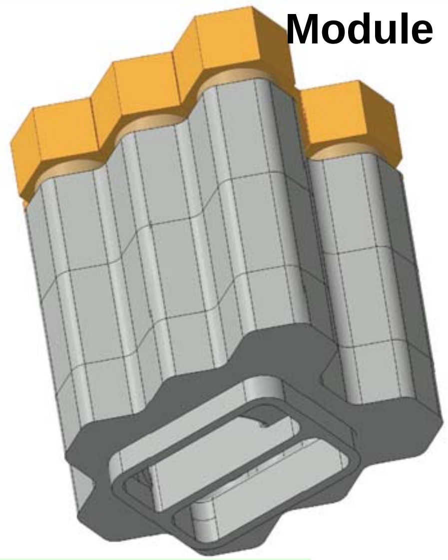




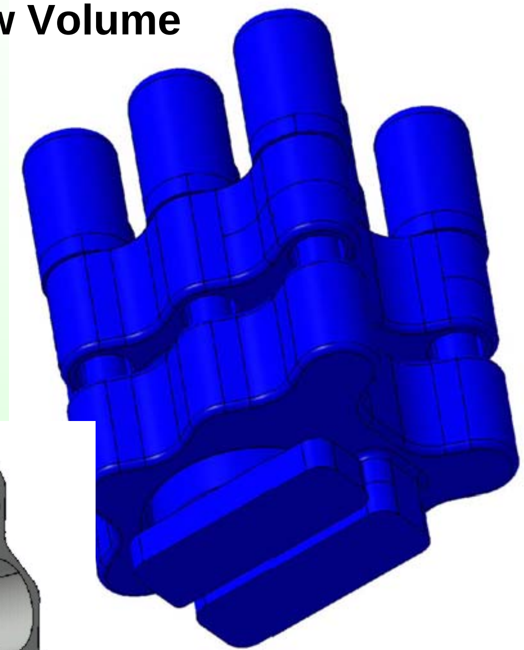
DESIGN

Advanced 9-Finger module

Module



Flow Volume



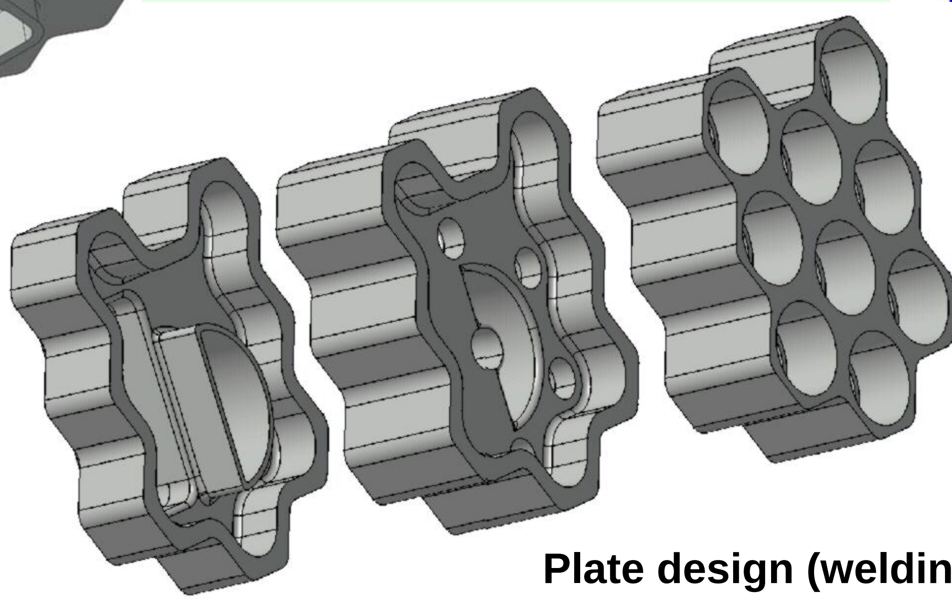


Plate design (welding / diffusion welding / HIP)

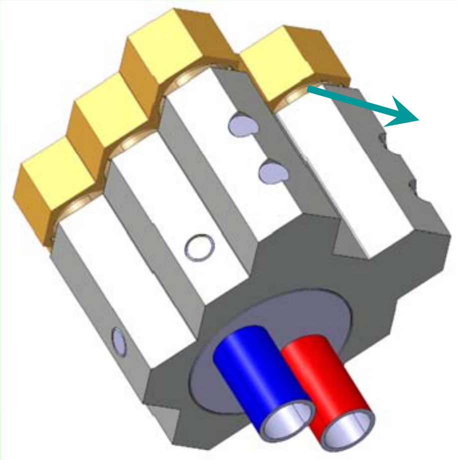
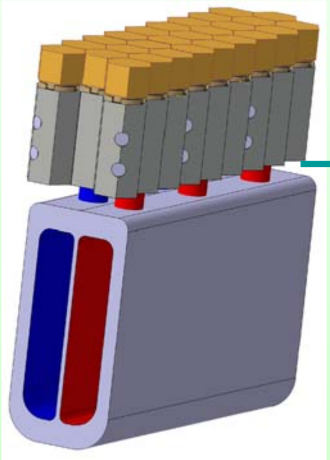
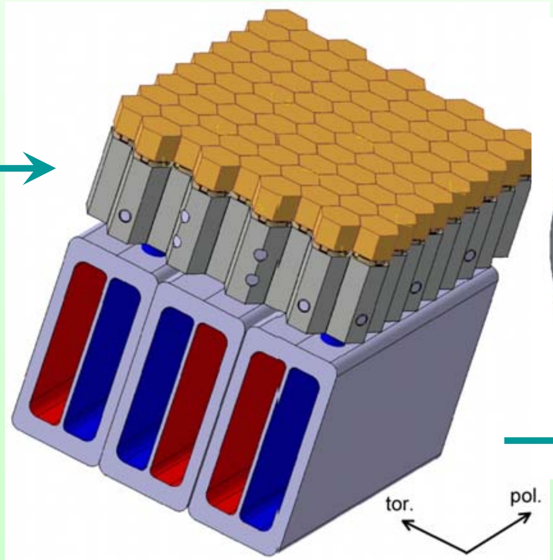
DESIGN

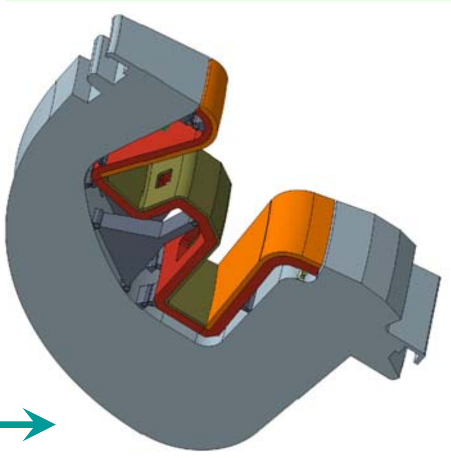
High reliability, low waste during fabrication

Construction Kit Principle:

Small Units advantageous for R&D progress
Small Units can be tested before assembling

9-Finger Unit **Stripe-Unit** **Target Plate** **Divertor Cassette**



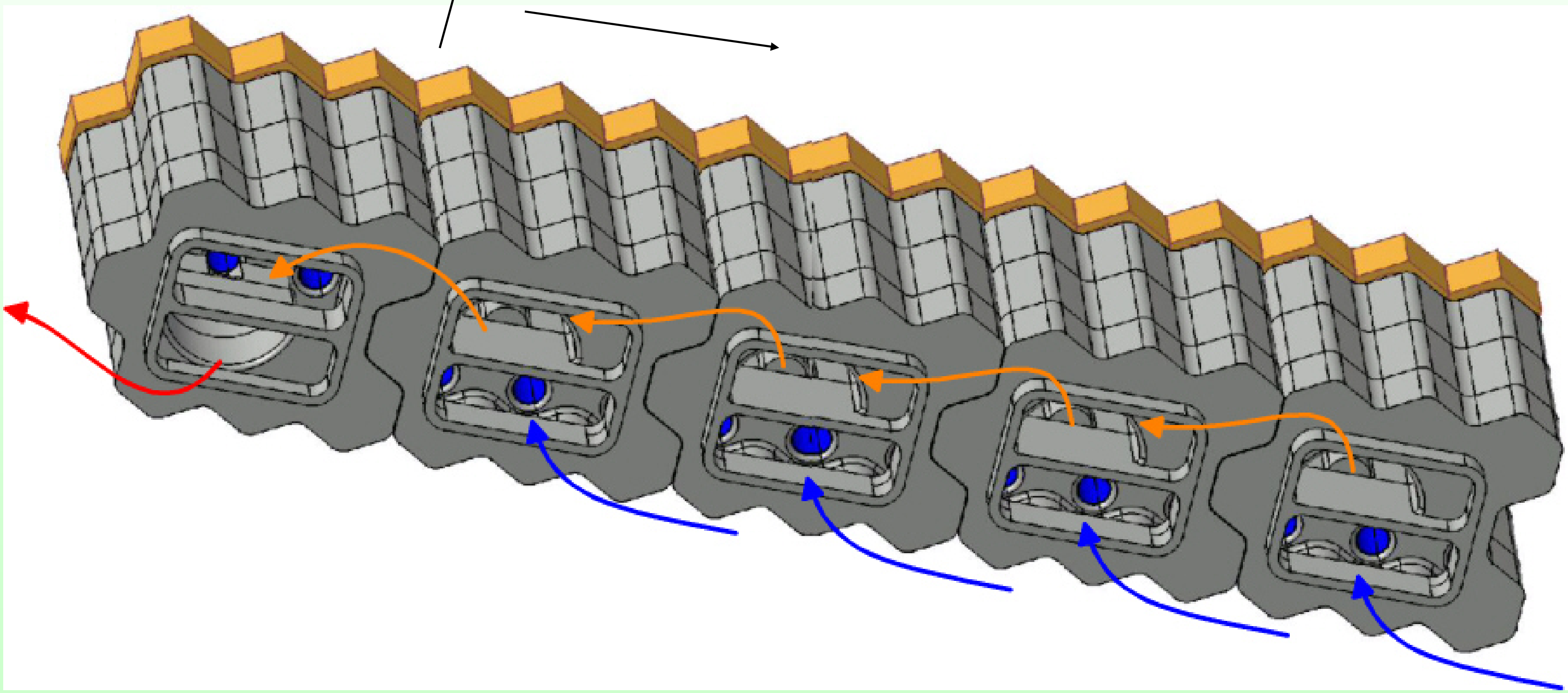
Test, Assembling → Test, Assembling → Test, Assembling → …

Assembling of 9-Finger-Modules (series connection)

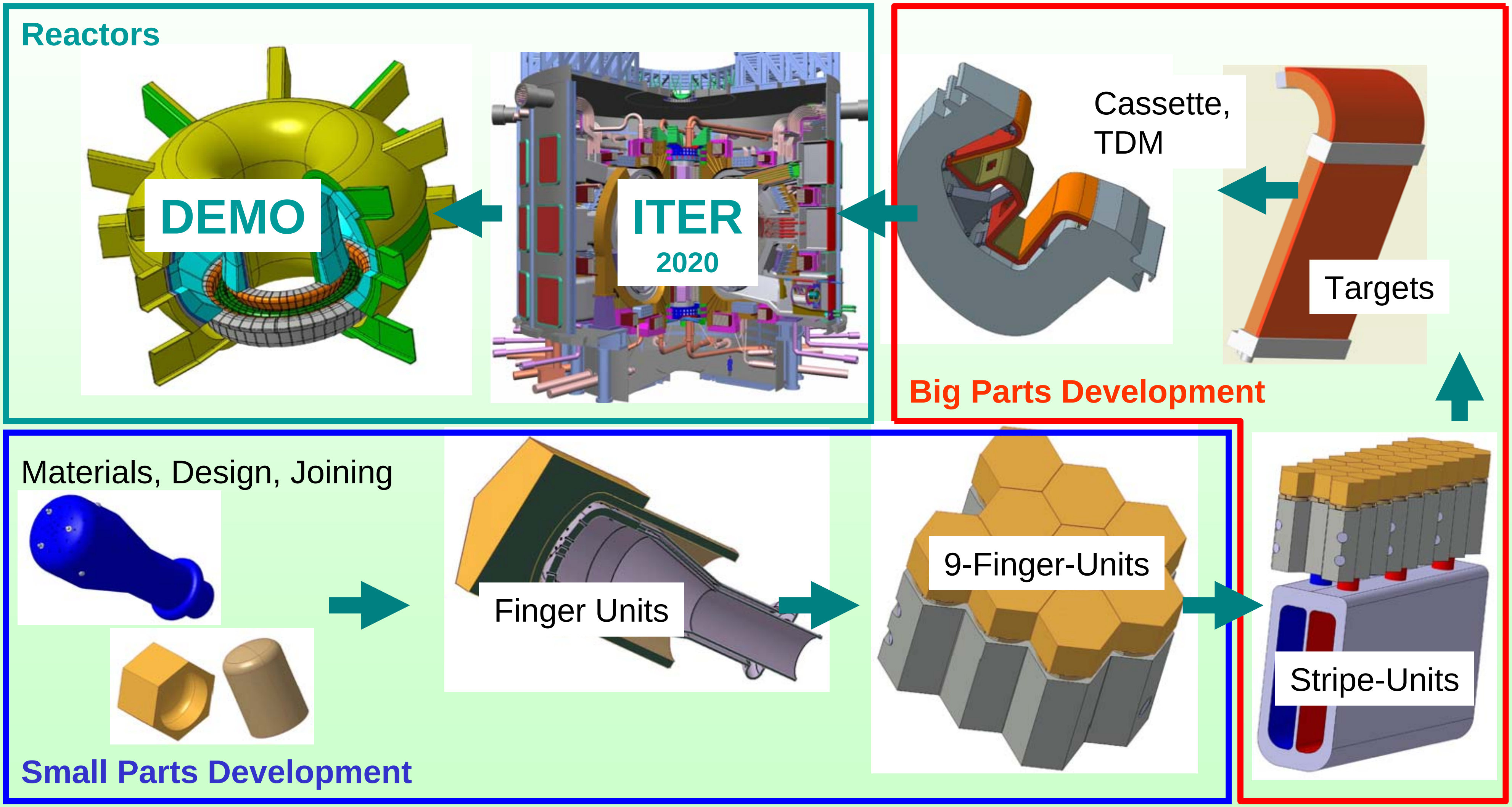
DESIGN

„left“ Modules

„right“ Modules



Outlook: He-cooled Divertor on the way to ITER & DEMO



Summary

- **Power Balance and Momentum Balance together with the need to avoid X-point Marfes limits the radiation losses in the SOL and the Divertor**

- **A few simple rules allow to design an Optimized Divertor**

- **Solutions with CuCrZr as heat sink and CFC or W as plasma facing material exist for a water cooled divertor up to 20 MWm^{-2}**

- **The integration of the divertor into a real machine produces additional boundary conditions**

- **A He cooled Divertor with up to 15 MWm^{-2} seems feasible**

Experiments

Scheme of Efremov GPF2