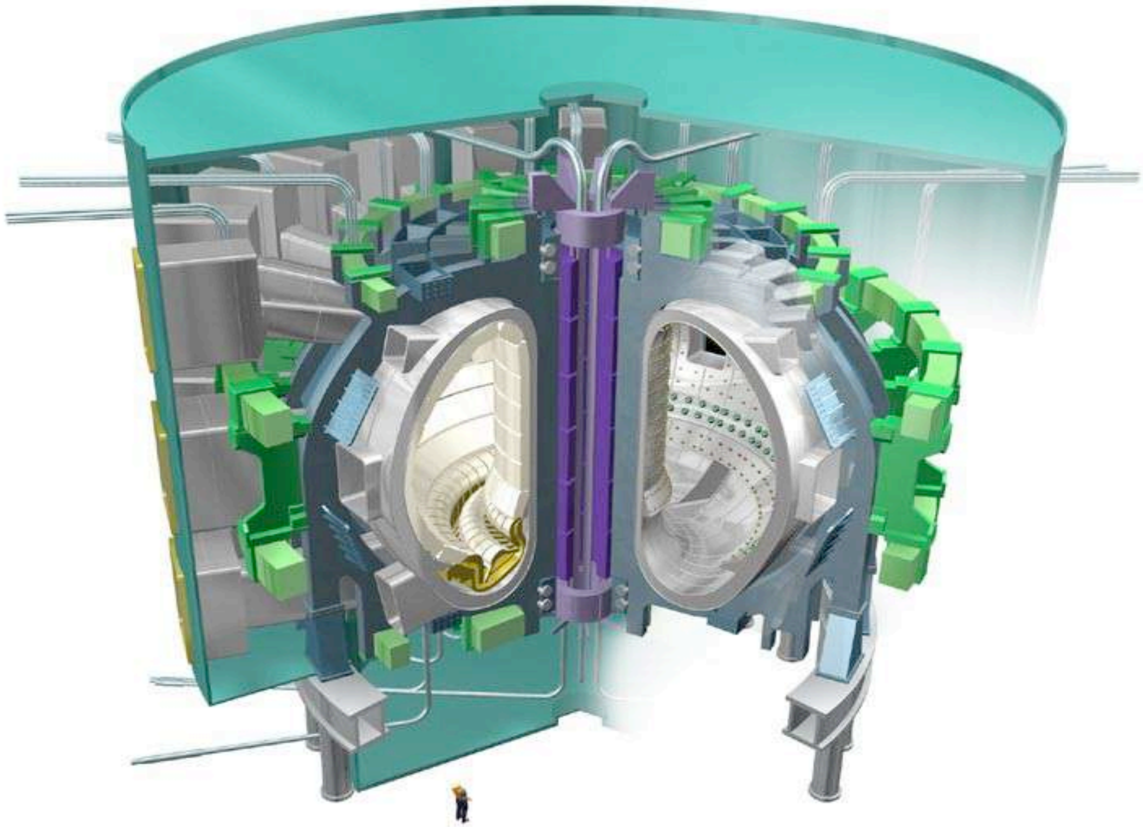
Design of ITER Plasma Facing Components



Mario Merola - ITER International Organization

Internal Components Division Head

Outline

Nuclear Fusion and the ITER Project

ITER Internal Components

ITER Divertor

ITER Blanket

Conclusions

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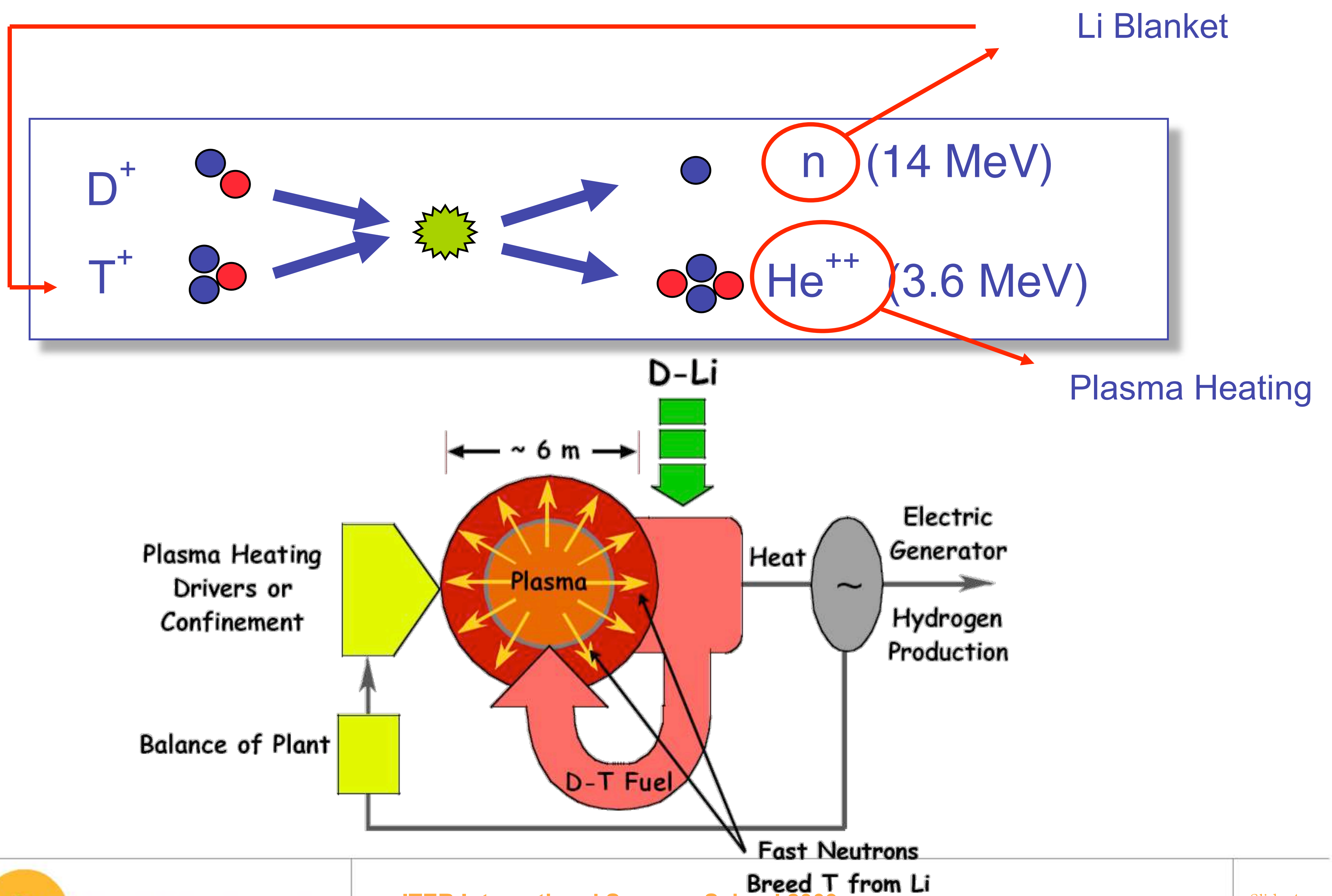
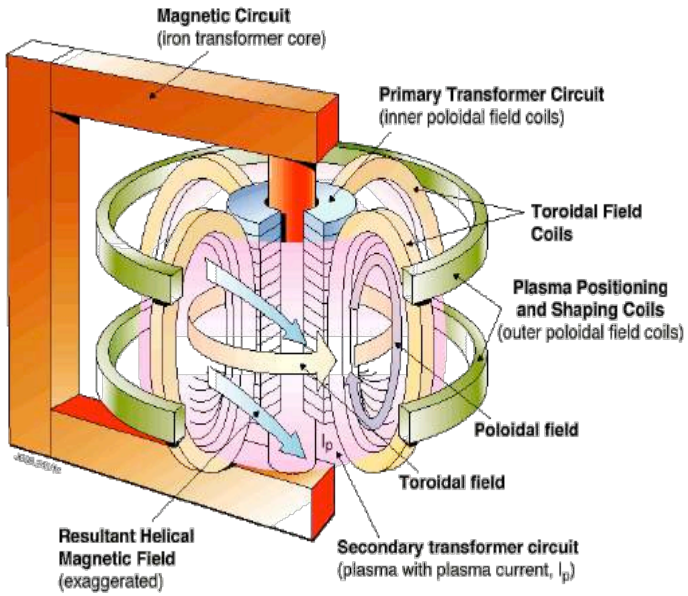
Elements of a D-T Fusion Energy System



Tokamak History

тороидальная камера с магнитными катушками
(*toroidal'naya kamera s magnitnymi katushkami*)
(Toroidal chamber with magnetic coils)



1956 - Experimental work starts in tokamak systems by a group of Soviet scientists led by Lev Artsimovich based on the work of Tamm , Sakharov and Lavryentev

1958 - World-wide declassification of magnetically confined fusion research at Geneva on Peaceful Uses of Atomic Energy

1960s : Tokamak established as leading contender for a thermonuclear system – First to achieve 1 keV temperature

1970s : Oil crisis propels major investment in fusion research facilities worldwide

1980s : Third generation of large tokamaks experiments come into operation : EU-JET ; US-TFTR ; URSS-T10 (all aimed at DT tests) and Japan- JT-60 (DD only)

1985 : ITER proposed at super power summit

The Way to Fusion Power – The ITER Story

The idea for ITER originated from the Geneva Superpower Summit on November 21,1985, when the Russian Premier Mikhail Gorbachev and the US-President Ronald Reagan proposed that an international Project be set up to develop fusion energy “as an essentially inexhaustible source of energy for the benefit of mankind”.

"All the News That's Fit to Print"

The New York Times

Late Edition

VOL. CXXXV . . No. 46,601 — NEW YORK, FRIDAY, NOVEMBER 22, 1985 — 30 CENTS

Text of the Joint U.S.-Soviet Statement: ‘Greater Understanding Achieved’

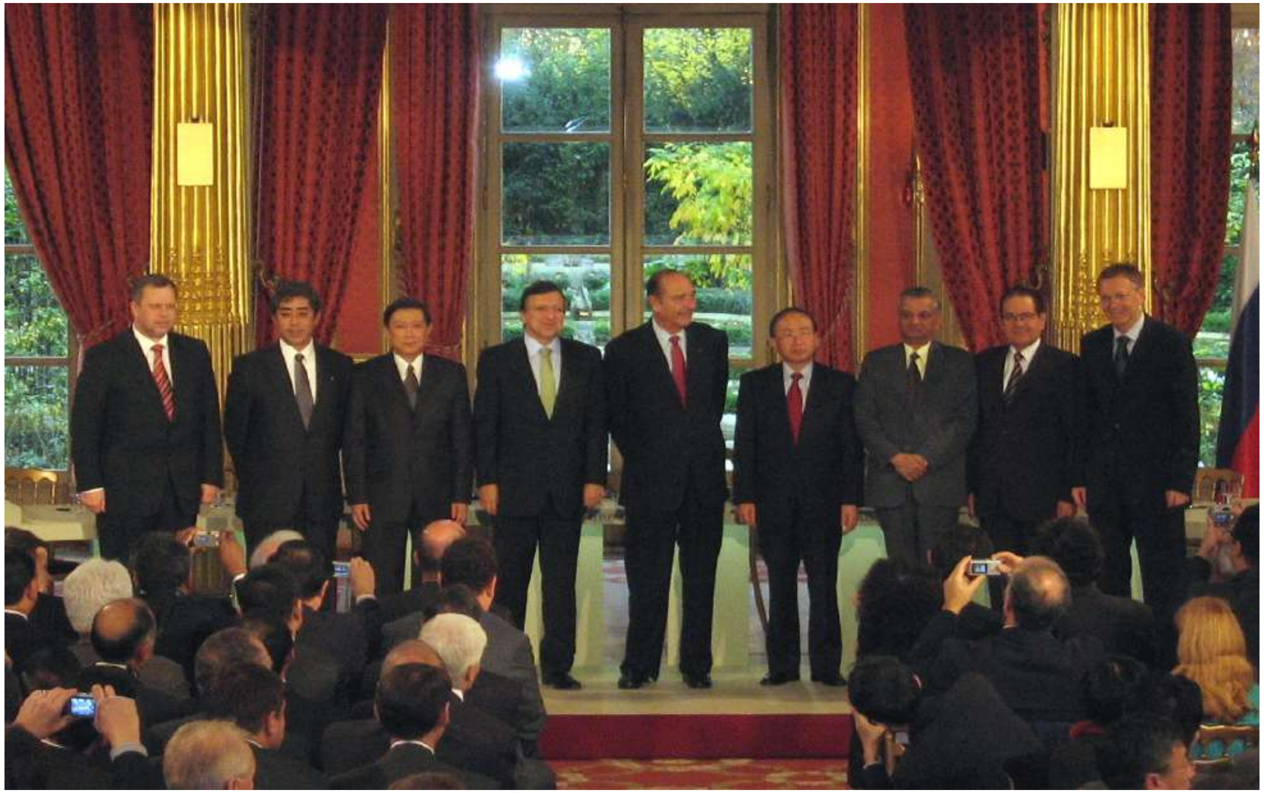
Fusion Research

The two leaders emphasized the potential importance of the work aimed at utilizing controlled thermonuclear fusion for peaceful purposes and, in this connection, advocated the widest practicable development of international cooperation in obtaining this source of energy, which is essentialy inexhaustible, for the benefit for all mankind.

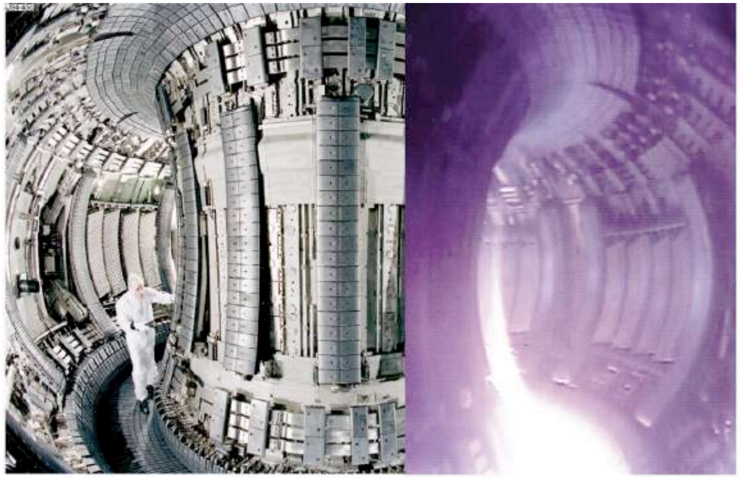
ITER Agreement

The agreement was signed on the
21st November 2006
at the Elysée Palace in Paris

International Organization started.



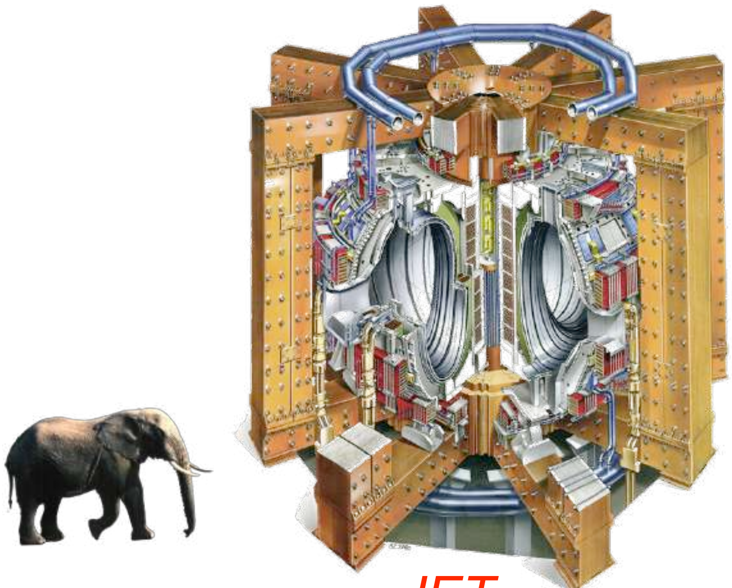
Tokamak's



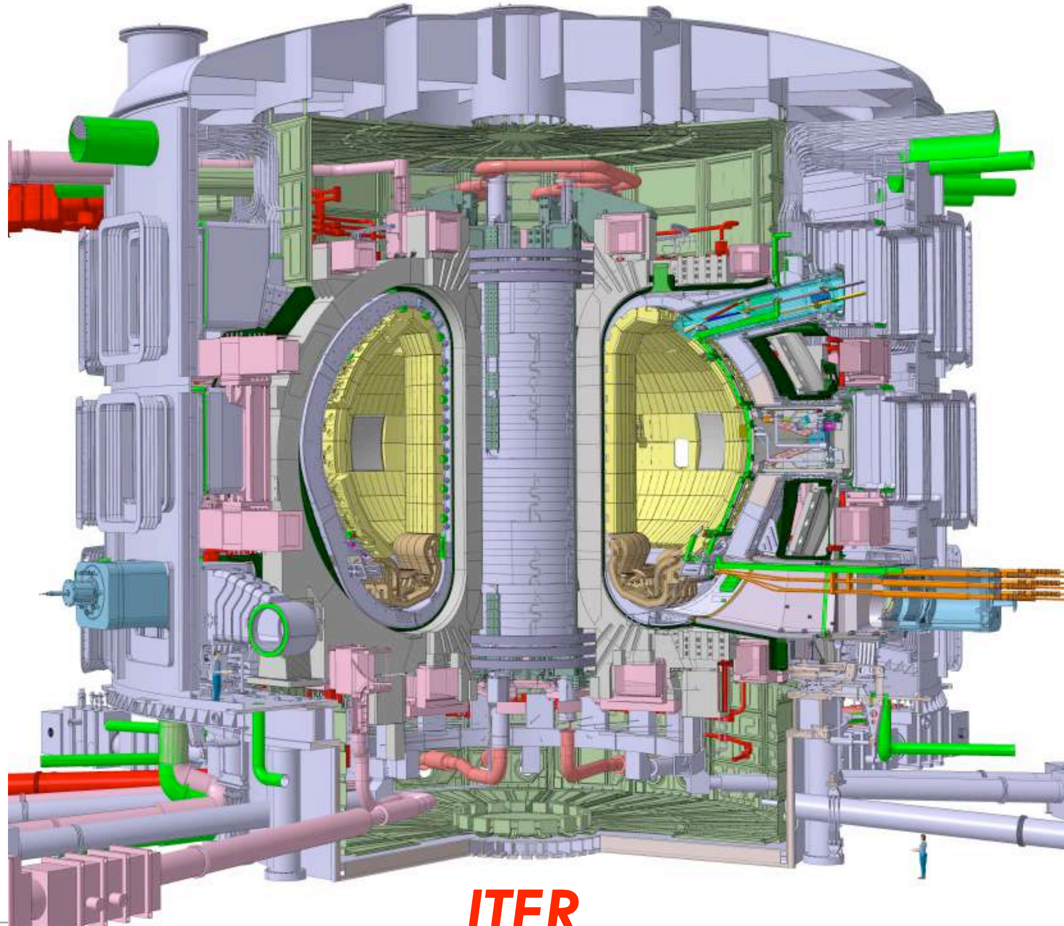
JET – Internals & Plasma

ITER will allow us to produce plasmas with
temperatures of 100 - 200 million °C
(10 times the temperature of the sun's core)

$\Rightarrow$ **500 Megawatts of fusion power**



JET



ITER

ITER – The way to fusion power

- ITER ("the way") is the essential next step in the development of fusion

- The world's biggest fusion energy research project, and one of the most challenging and innovative scientific projects in the world today.

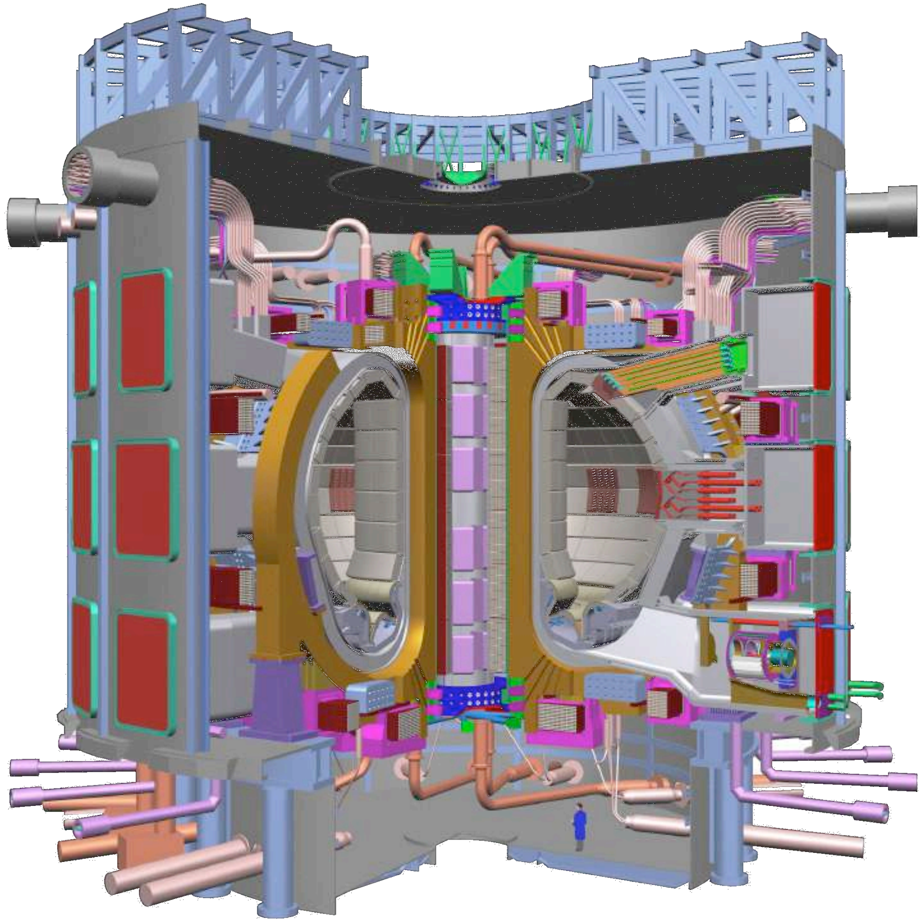
- Its objective:

  -to demonstrate the scientific and technological feasibility of fusion power

  -demonstrate extended burn of DT plasmas, with steady state as the ultimate goal.

  -integrate and test all essential fusion power reactor technologies and components.

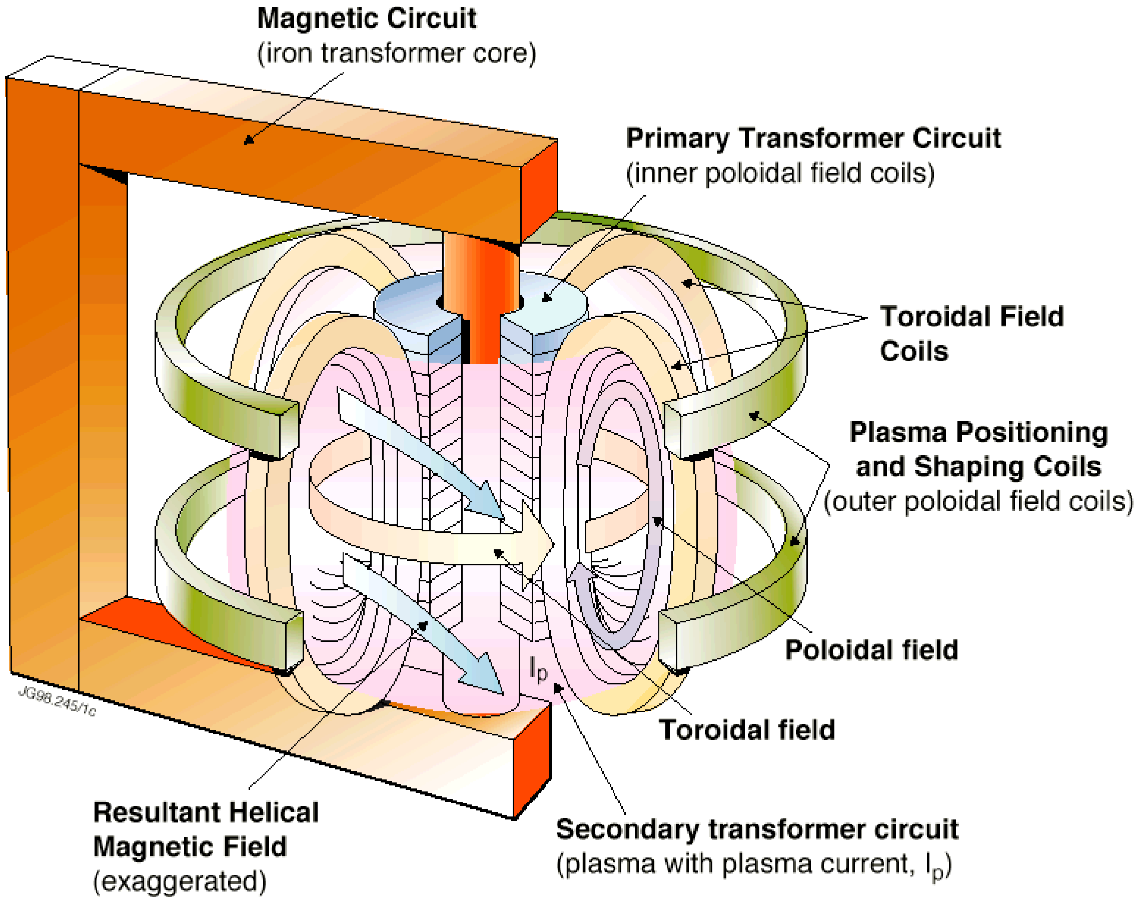
  -demonstrate safety and environmental acceptability of fusion.



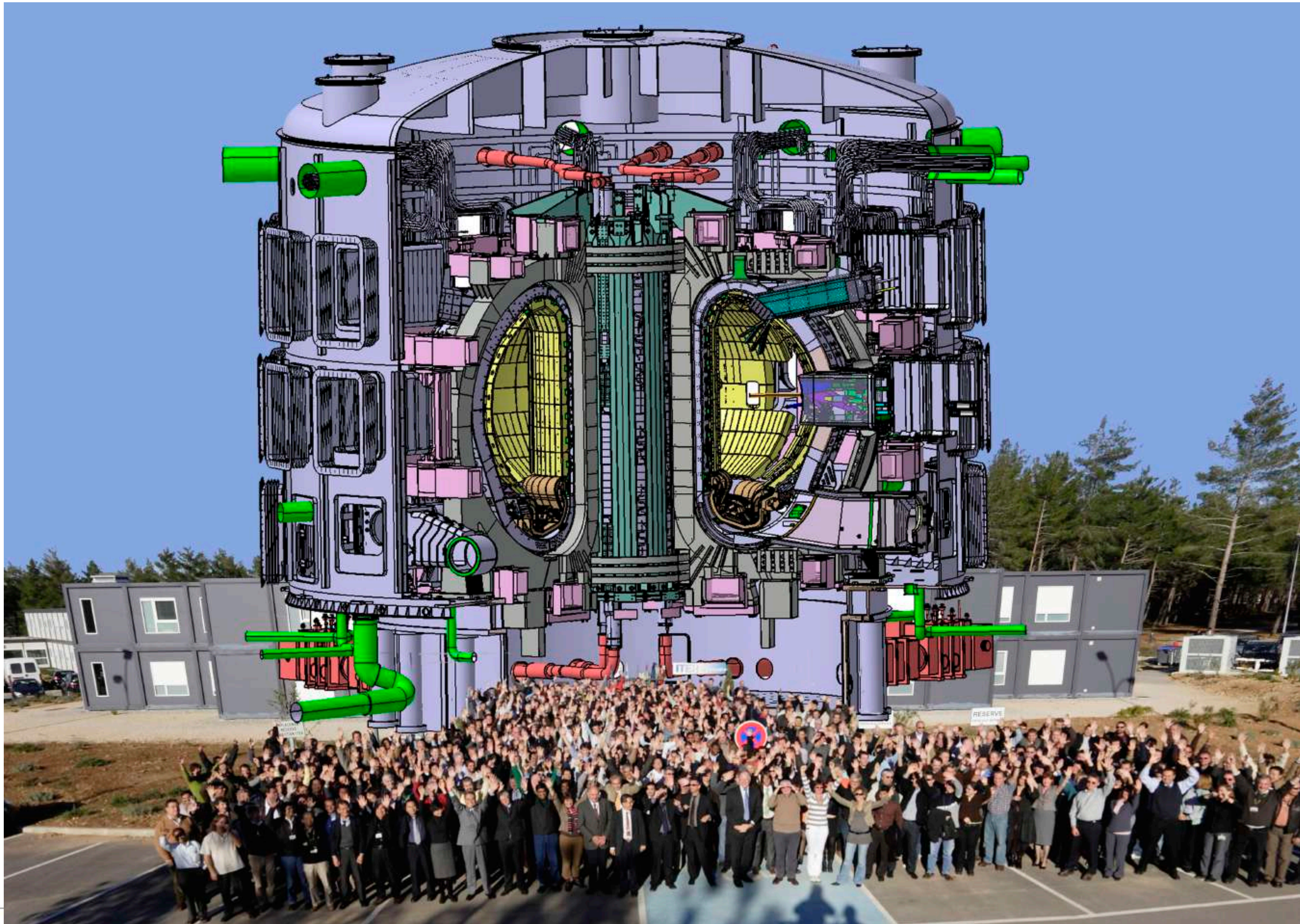
Magnetic Confinement in a Tokamak

The Tokamak:

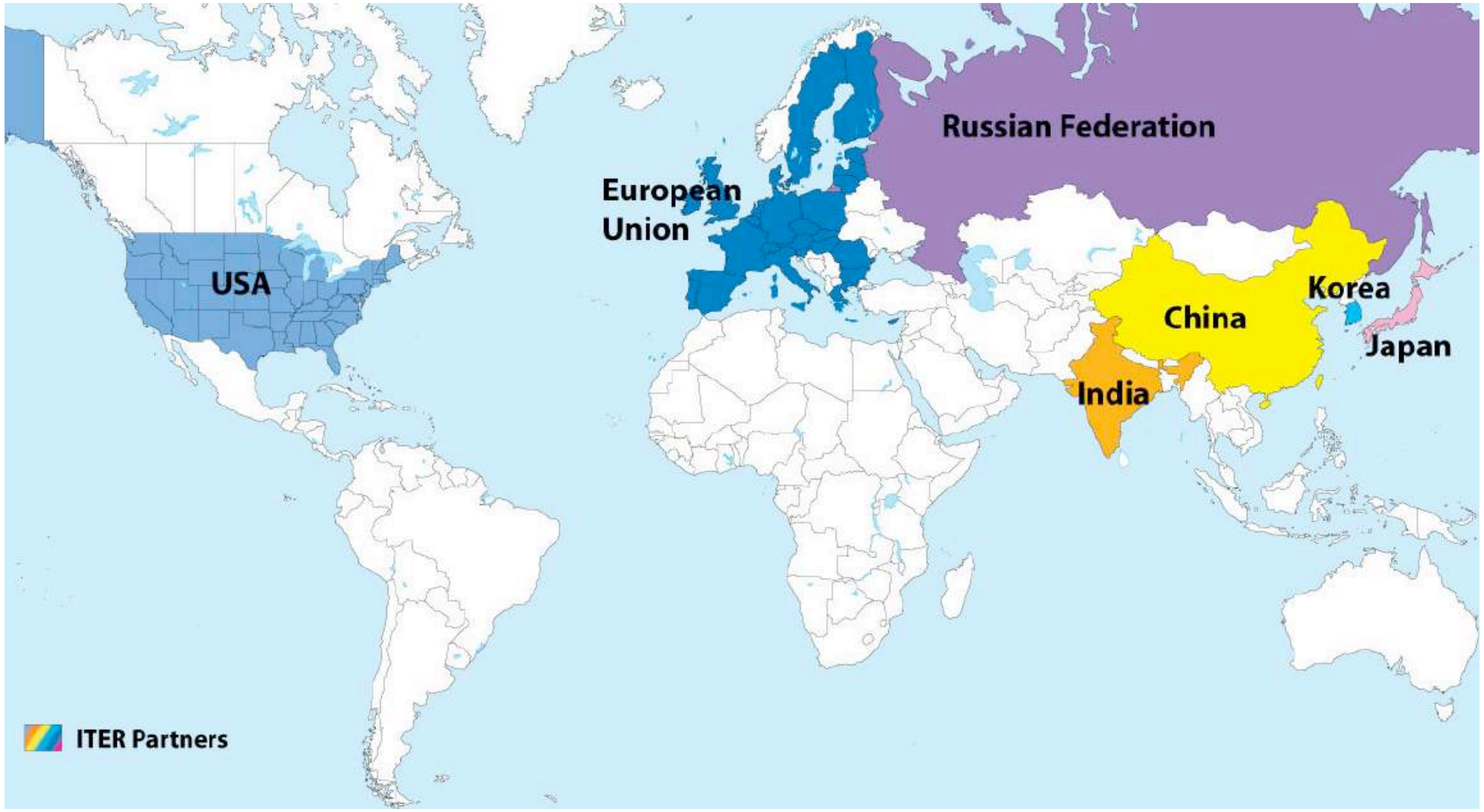
- toroidal magnetic field is produced by external magnetic field coils
- plasma current produces poloidal magnetic field
- result is a set of nested helical surfaces
  ⇒ plasma confinement



Tokamak – 29 m high x 28 m dia. & ~23000 t



ITER – Key facts

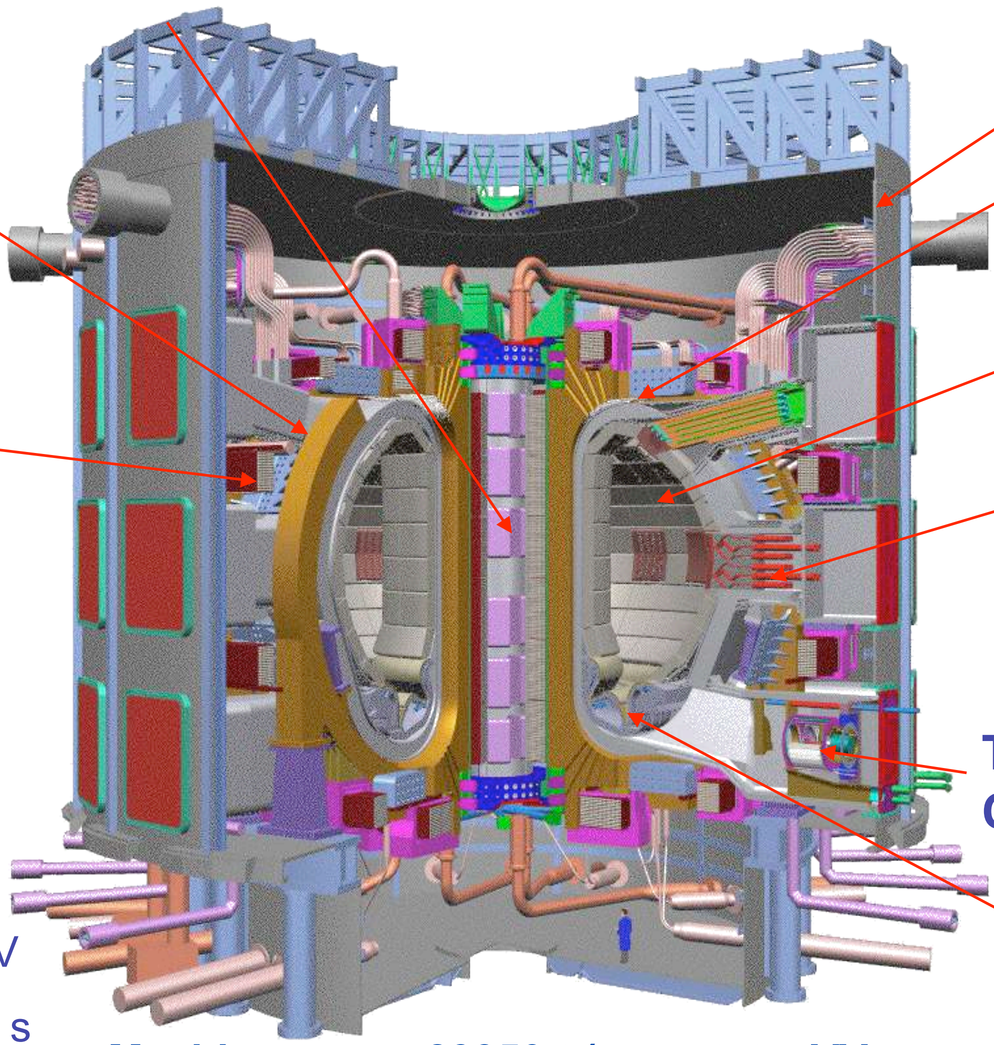


- Designed to produce 500 MW of fusion power for an extended period of time
- 10 years construction, 20 years operation
- Cost: 5 billion Euros for construction, and 5 billion for operation and decommissioning

ITER Site

ITER Headquarters :
Saint Paul-lès-Durance,
Provence-Alpes-Côte d'Azur,
France.
Site next to **Cadarache** , nuclear research center of the *Commissariat à l'Énergie Atomique* (CEA).

The Core of ITER



Major plasma radius 6.2 m

Plasma Volume: 840 m^3

Plasma Current: 15 MA

Typical Density: 10^{20} m^{-3}

Typical Temperature: 10 keV

Energy confinement time:3.4 s

Fusion Power: 500 MW

Machine mass: 23350 t (cryostat + VV + magnets)
- shielding, divertor and manifolds: 7945 t + 1060 port plugs
- magnet systems: 10150 t; cryostat: 820 t

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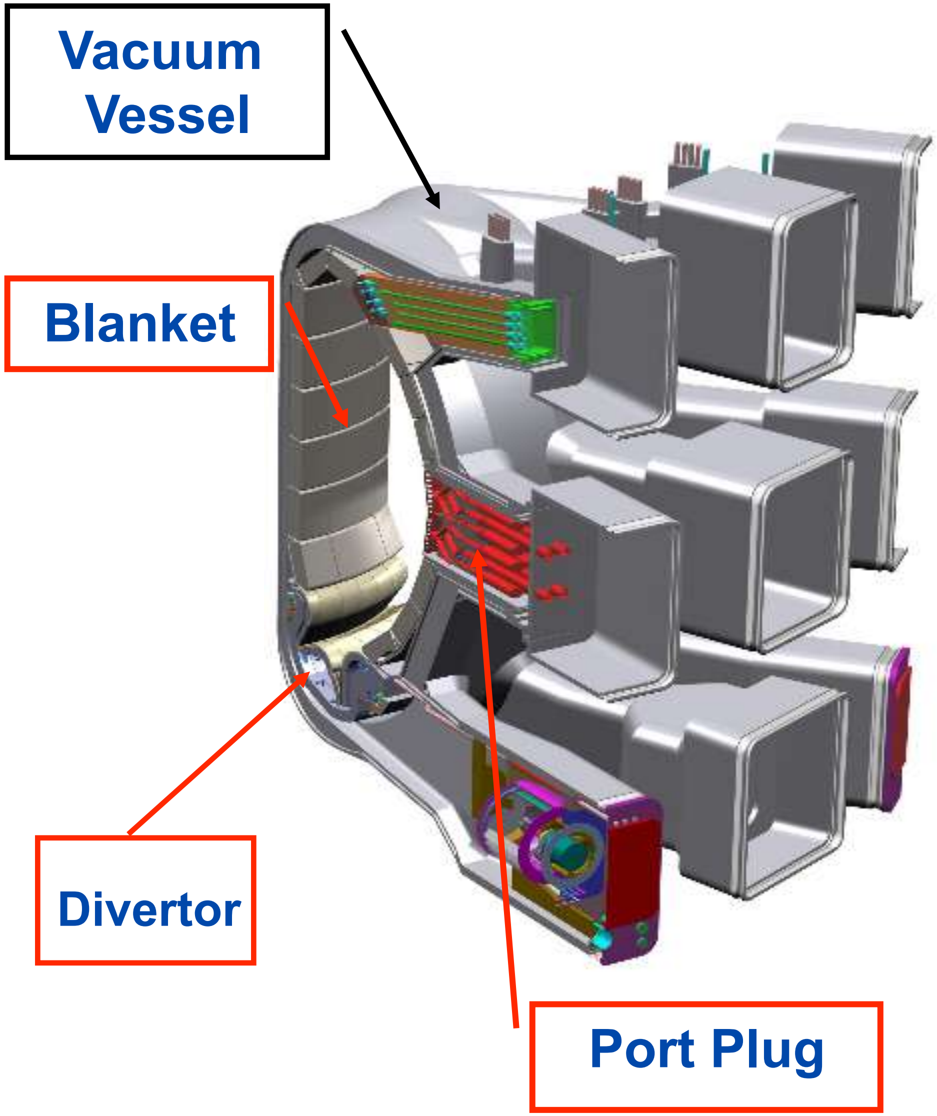
Vessel/In-Vessel ITER Components



•The vacuum vessel is lined by modular removable components: : **blanket modules**, **divertor cassettes** and **port plugs** (heating antennae, diagnostics and test blanket modules) All these removable components are mechanically attached to the VV.

•The functions of the in-vessel components are

-Minimize the impurity content of the plasma

-Absorb the radiated and conducted heat from the plasma and absorb the neutronic heating

-Shield the superconducting coils

-Withstand the electromagnetic induced forces during plasma instabilities

-Contribute to the plasma passive stabilization

Design inputs for PFCs

- **Surface heat flux** due to the radiative and particle flux from the plasma. This is of particular concern for the next generation of fusion machines where, due to the high number of operating cycles, a thermal fatigue problem is anticipated. Particularly harmful are the off-normal heat loads, which are associated to plasma instabilities (such as a plasma disruption or vertical displacement). Up to some tens of MJ/m^2 can be deposited onto the PFCs in a fraction of a second resulting in melting and evaporation of the plasma facing material. About 10% of the discharges are anticipated to end with plasma instability in the next generation of fusion machines, whereas this figure should decrease to less than 1% in a commercial reactor.

- **Neutron flux from the plasma.** The neutron flux is referred to as "wall loading" and measured in MW/m^2. This is the power density transported by the neutrons produced by the fusion reaction. The wall loading multiplied by the total plasma burn time gives the neutron fluence, which is measured in $MW\text{-}year/m^2$. The two main effects of the neutron flux are the volumetric heat deposition and the neutron damage.

Design inputs for PFCs

- The **volumetric heat deposition** has a typical maximum value of a few W/cm^3 in the FW structures and then decreases radially in an exponential way. It has mainly an impact on the design of the supporting structures, which thus need to be actively cooled.

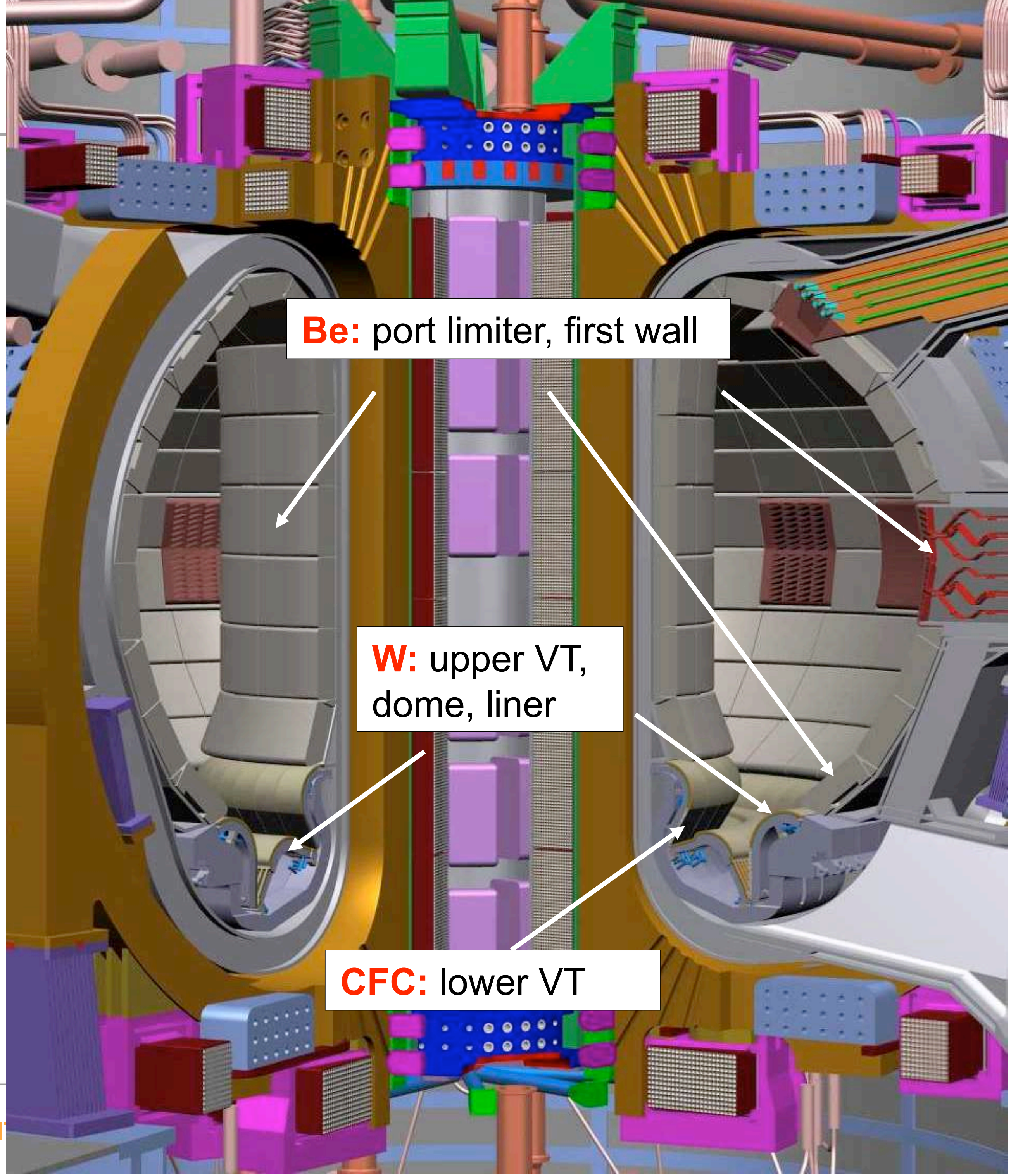
- The **neutron damage** will be the main lifetime limiting phenomenon in a commercial reactor. It is measured in “displacements per atom” (dpa) that is the number of times an atom is displaced from its position in the lattice due to the action of an impinging particle. The dpa is proportional to the neutron fluence. As an example 1 MW-year/m^2 causes about 3 and 10 dpa in beryllium and copper or steel, respectively. The dpa value is a measure of the neutron damage. Typical effects of this damage are embrittlement and swelling.

Design inputs for PFCs

- **Electromagnetic loads.** During a plasma instabilities eddy currents are induced in the PFCs. These currents interact with the toroidal magnetic field thus resulting in extremely high forces applied to the PFCs. These forces can generate mechanical stresses up to a few hundreds of MPa with a consequent strong impact in the design of the supporting structures.

- **Surface erosion.** The particle flux impinging onto the PFCs causes surface erosion due to physical sputtering (and also chemical sputtering in the case of carbon). One effect of this phenomenon is that the thickness of the plasma facing material is progressively reduced. Furthermore the eroded particles can migrate into the plasma thus increasing the radiative energy loss by *bremsstrahlung* and diluting the deuterium and tritium concentration. Another consequence is that some eroded particle (like carbon or beryllium oxide) may trap tritium atoms when they redeposit onto the surface of the PFCs (the so-called "co-deposition"). This results in an increase of the tritium inventory in the plasma chamber with the associated safety concerns.

Armour Materials



Beryllium

- Low atomic number
- Oxygen gettering capability
- Absence of chemical sputtering
- High thermal conductivity

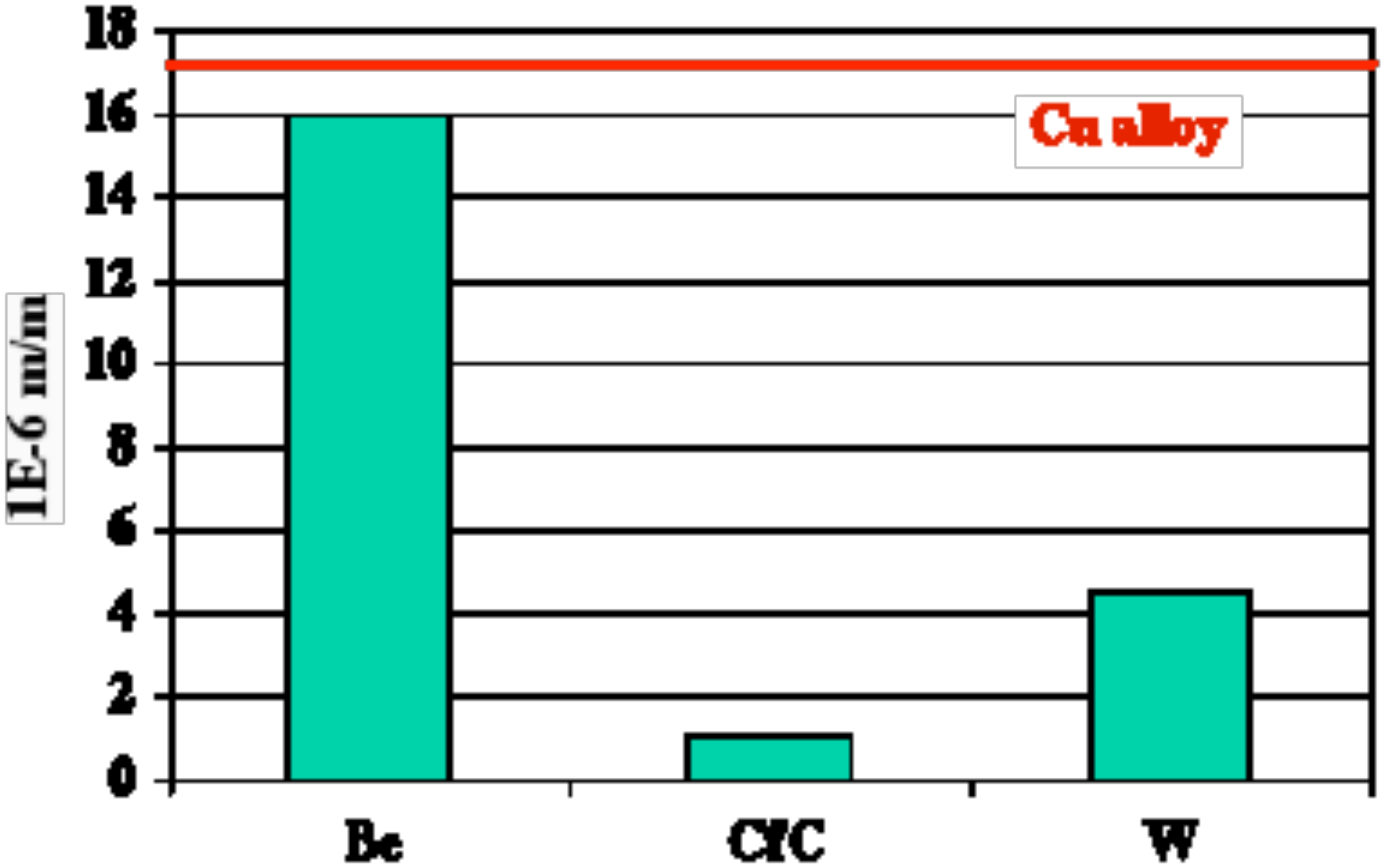
CFC

- Longest lifetime
- Absence of melting
- Excellent thermal shock resistance
- Very high thermal conductivity
- Low atomic number

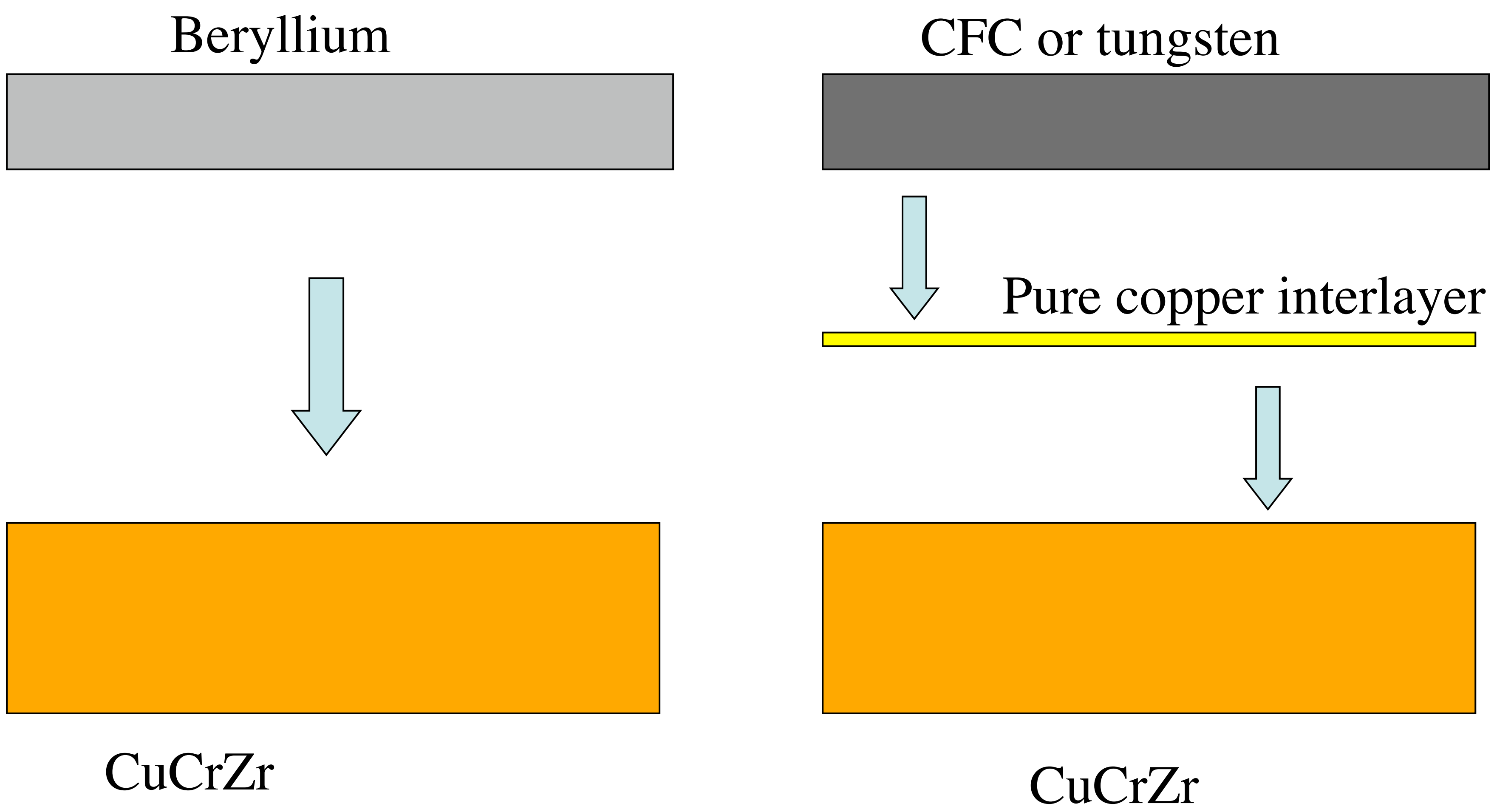
Tungsten

- Lowest sputtering
- Highest melting point
- High thermal conductivity
- No concerns over tritium inventory
- Reference grade: sintered and rolled pure tungsten

Thermal expansion at 300 °C



Armour to heat sink joints



Beryllium to copper alloy joint

Formation of brittle intermetallics (e.g. BeCu or Be_2Cu) above ~600 °C

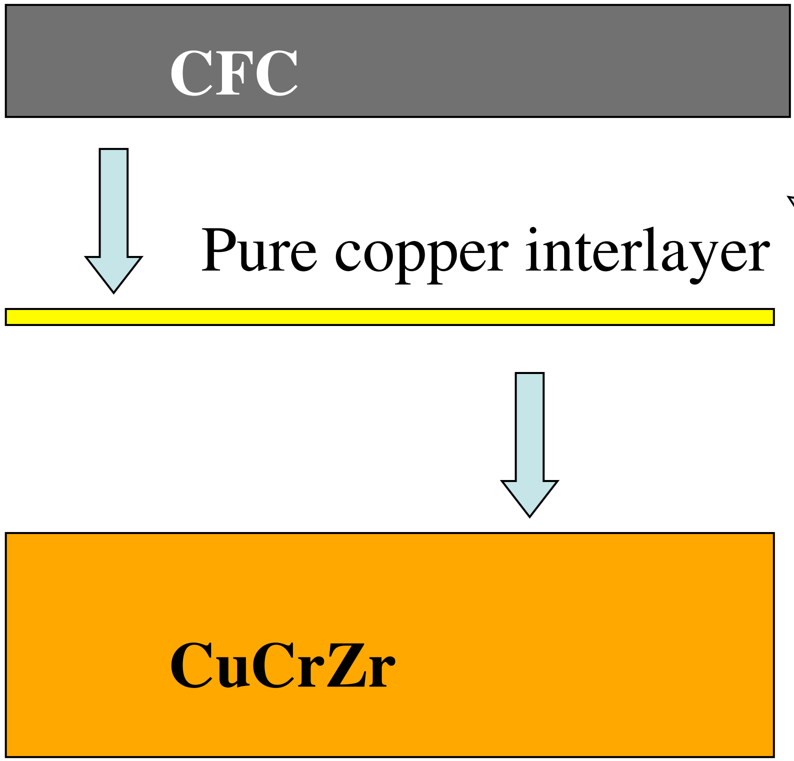
Use of diffusion barriers as Ti (Al)

HIP'ing at 500-850 °C

Proper selection of technology

Fast brazing
Cu electroplating of Be + HIP
Be plasma spray

Armour to heat sink joints

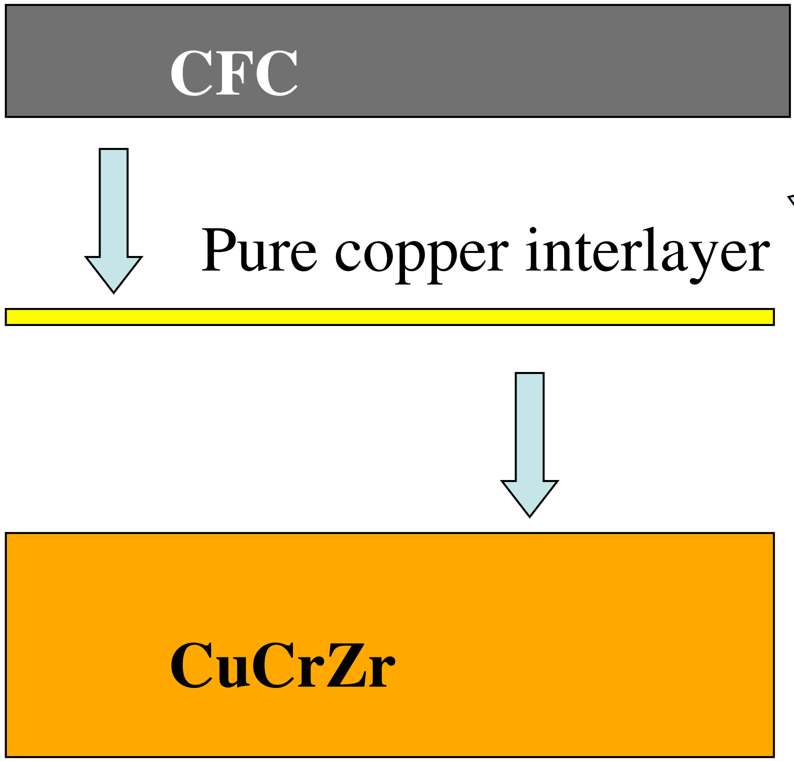


Copper does not wet carbon

Wetting agents may lead to the formation of brittle intermetallics or compounds with a low melting point

Large thermal expansion mismatch

Armour to heat sink joints

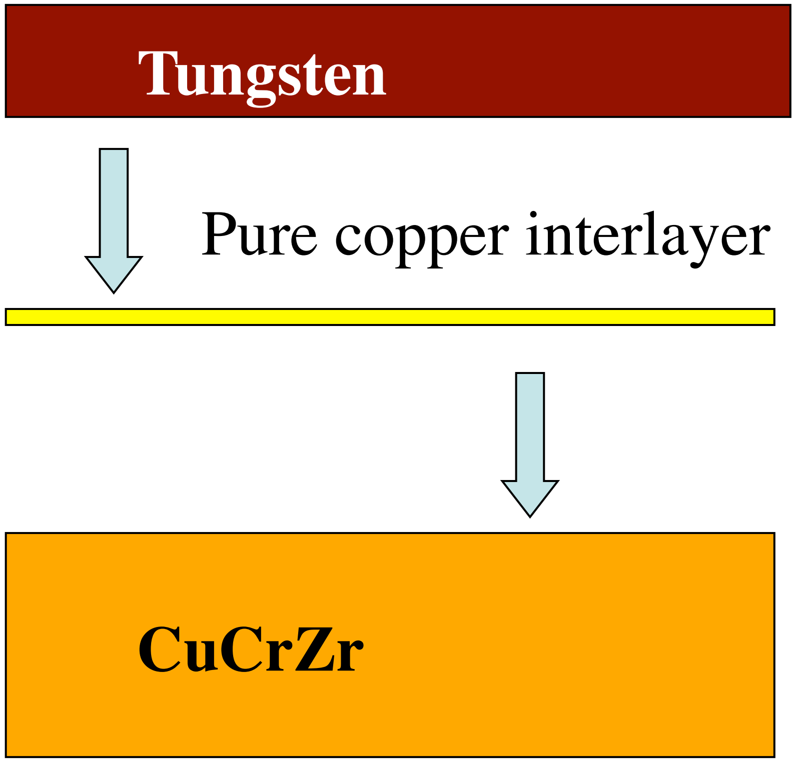


AMC

Pre-brazed casting

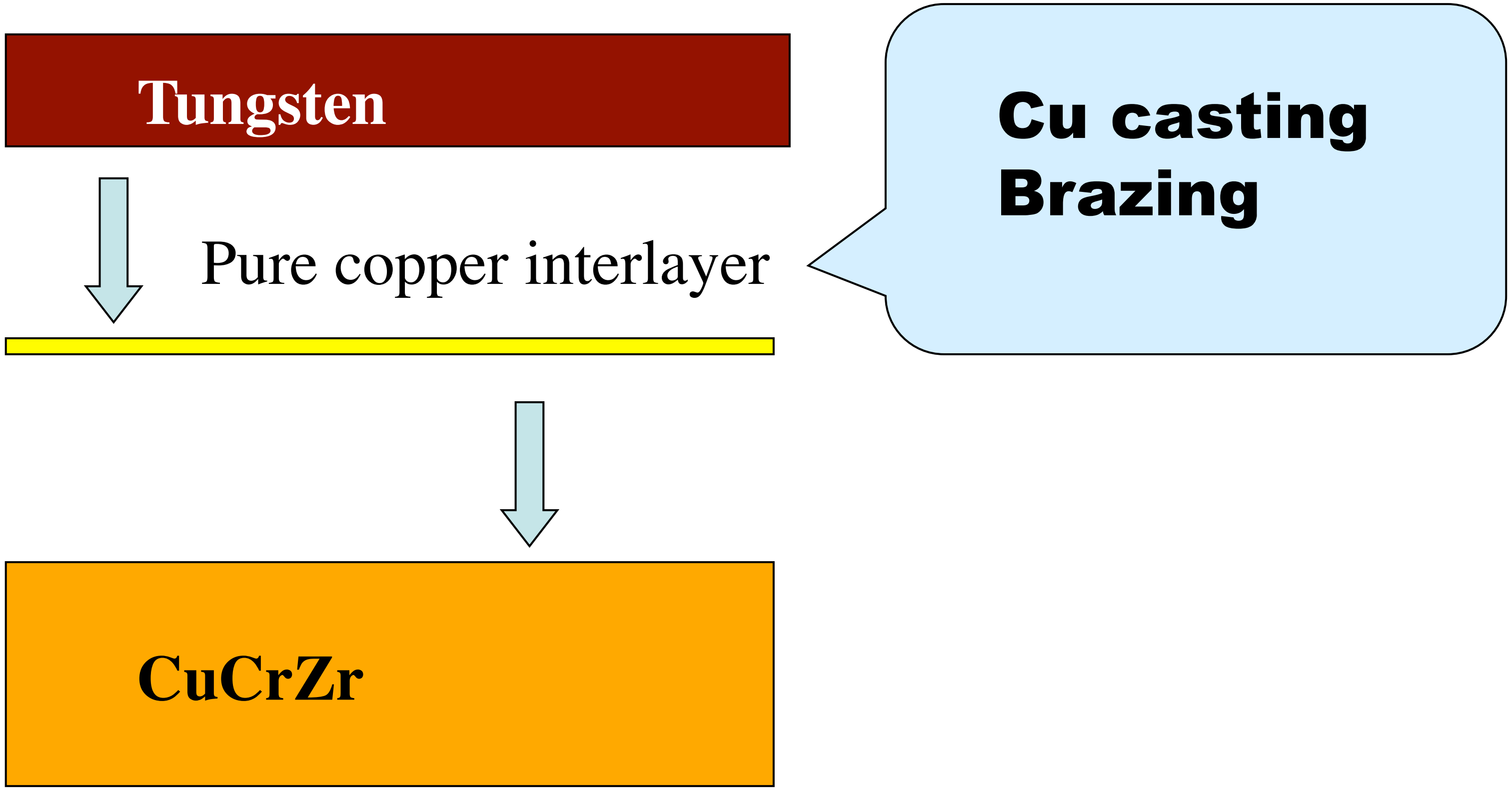
Brazing

Armour to heat sink joints

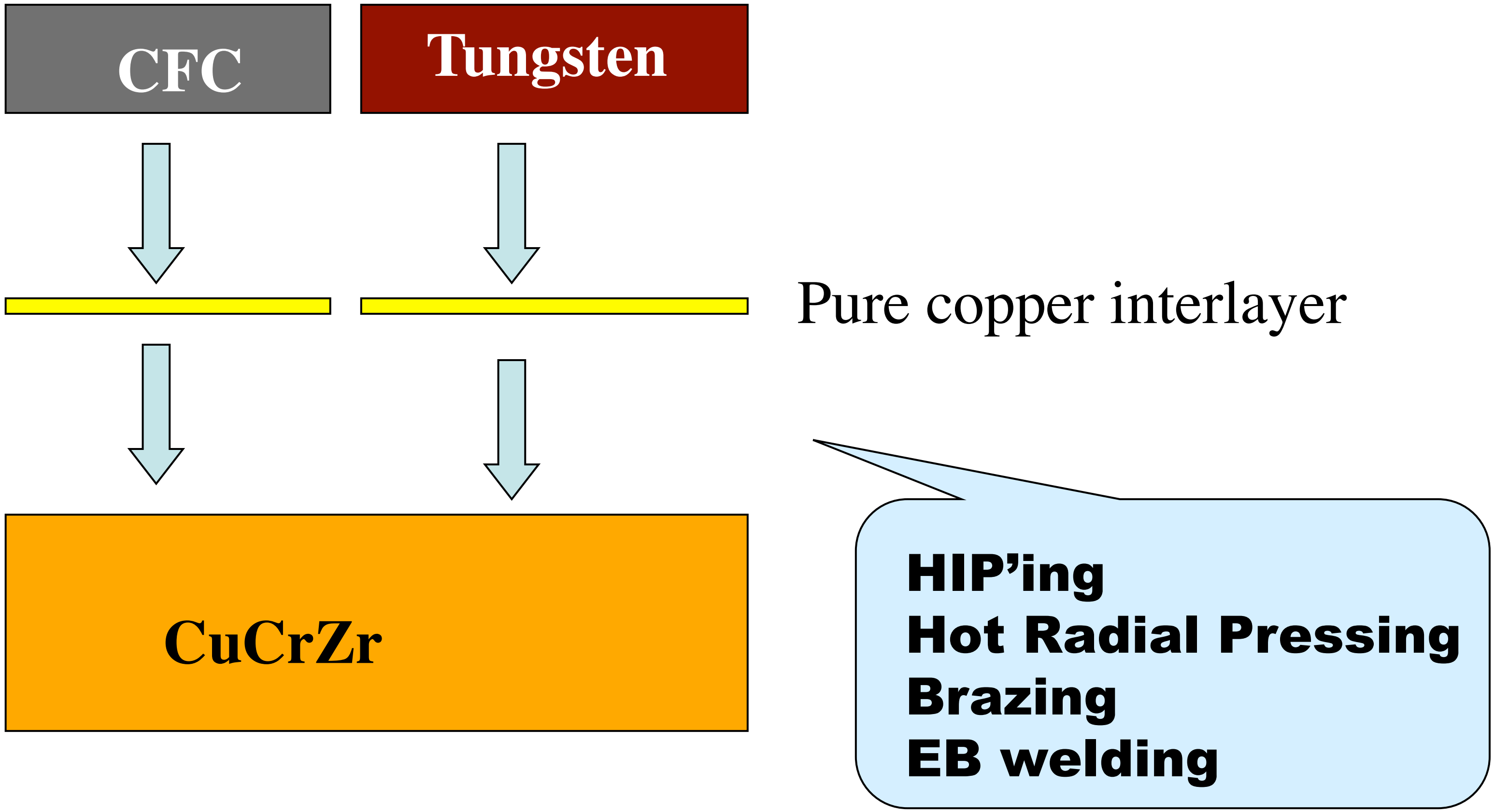


Casting Cu onto W requires adequate experience to ensure a good wetting of W and to prevent the formation of bubble in the cast Cu

Armour to heat sink joints



Armour to heat sink joints



CuCrZr: a “difficult” material

It reaches an optimum in strength after a thermo-mechanical treatment involving:

1) first a solution annealing at high temperature (>980 C) to dissolve the alloying elements (Cr, Zr)

2) then a water quench to keep the alloying elements in supersaturated solid solution at room temperature

3) finally an ageing treatment at intermediate temperatures (475 C, 3 hrs) to decompose the supersaturated solid solution into a fine distribution of precipitates.

CuCrZr: a "difficult" material

The manufacturing route shall be carefully defined.

Thermal excursions above the ageing temperature can overage the alloy with a significant decrease of strength.

Overageing affects also the thermal conductivity, which could be very much reduced by the dissolution of precipitates.

Identification of the allowable manufacturing brazing or high temperature HIP'ing cycle

Starting condition	SA	SA	SA	SA
Cool. Rate 970 --> 770 C	0.06 C/s	1.5 C/s	WQ	WQ
Cool. Rate 770 --> 370 C	0.03 C/s	0.8 C/s	=	=
HV after 475 C x 3 hrs	55	119	135	70 (no ageing)
Therm. Cond. (W/mK)	~320	~330	~320	~170

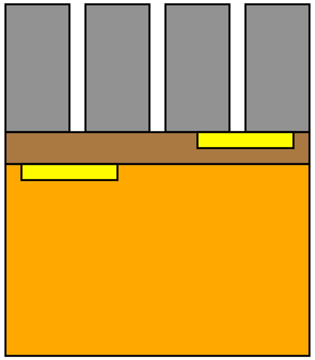
Values at RT

Identification of the allowable manufacturing brazing or high temperature HIP'ing cycle

- Any "fast cooling" after brazing with a rate > 1 C/s enables an acceptable recovery of the mechanical strength
- Thermal conductivity is very forgiving with respect to the manufacturing cycle

Non-destructive testing: Be/Cu alloy, W/Cu and Cu/Cu alloy joint

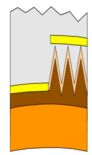
- Ultrasonic examination is the best technique.
- Defects of 2 mm can be detected reliably
- Inspection better performed from the rear side, prior to machining the cooling channels, when:
  - CFC armour
  - Be or W armour and fine castellation (< 10x10mm)
- Main issue: differences in the attenuation of the ultrasonic waves (up to 16 dB)



Non-destructive testing: CFC/Cu joint

<u>Ultrasonic examinations</u>

- Ultrasounds can hardly propagate inside the CFC material, therefore the CFC/Cu joint can only be inspected from the Cu side.
- The acoustic impedance of CFC and Cu is significantly different
- AMC: laser structured surface



CfC

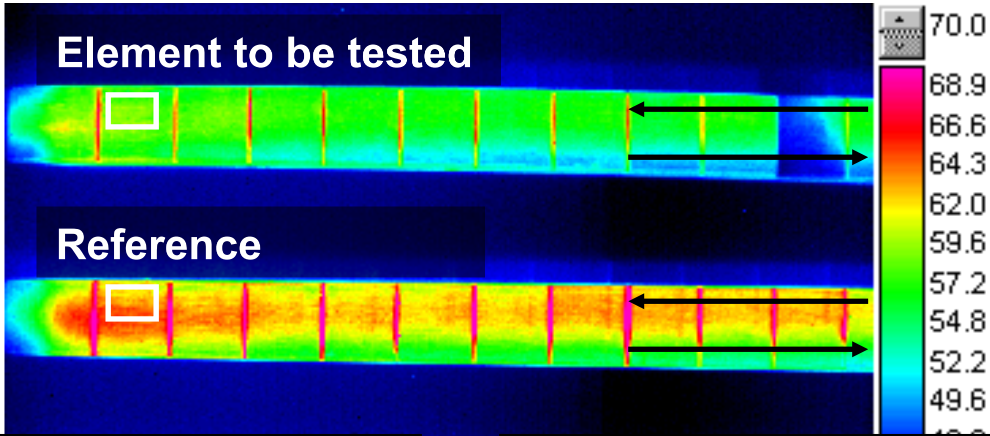
- Thermographic examination

Cu

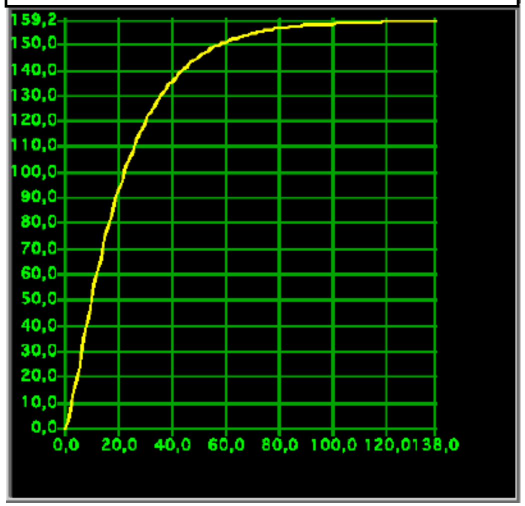
CuCrZr

Flow direction

Transient Thermography Inspection

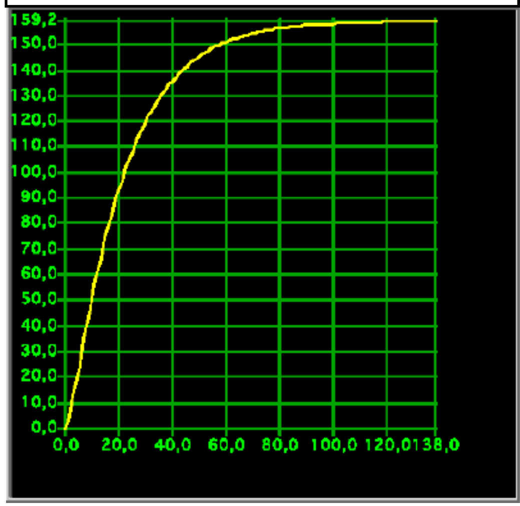


$T_{min,Ref}$ (t) temperature evolution of the reference tile



-

T_{min} (t) temperature evolution of the tile to be tested



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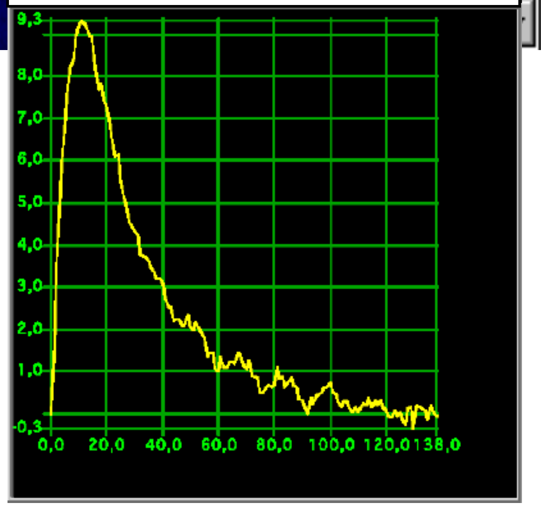
$\Delta T_{min,Ref}$ (t) maximum temperature difference



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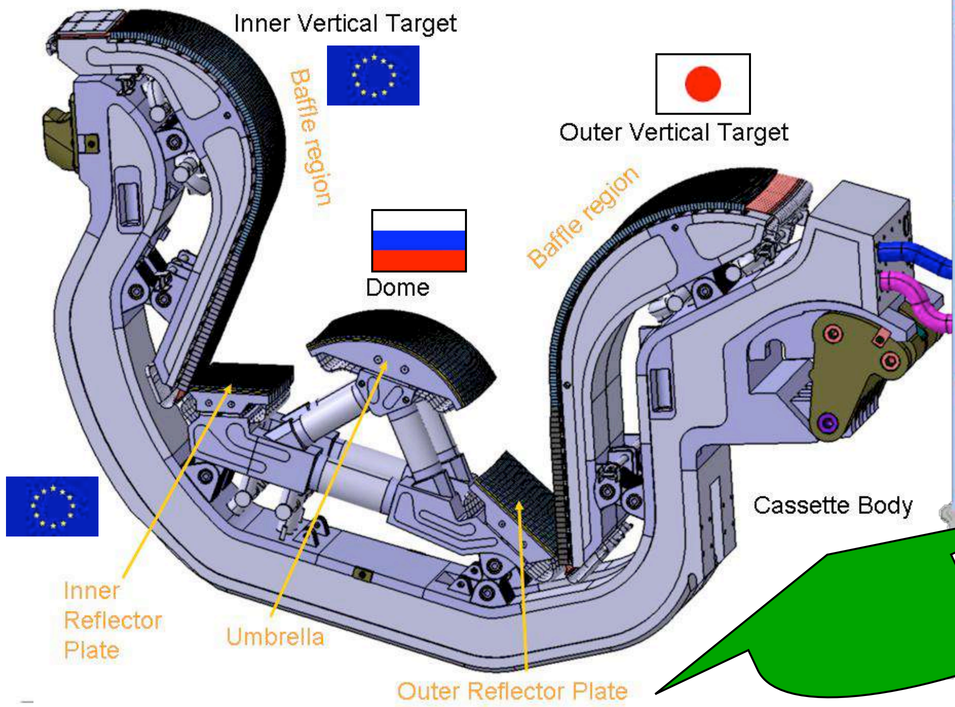
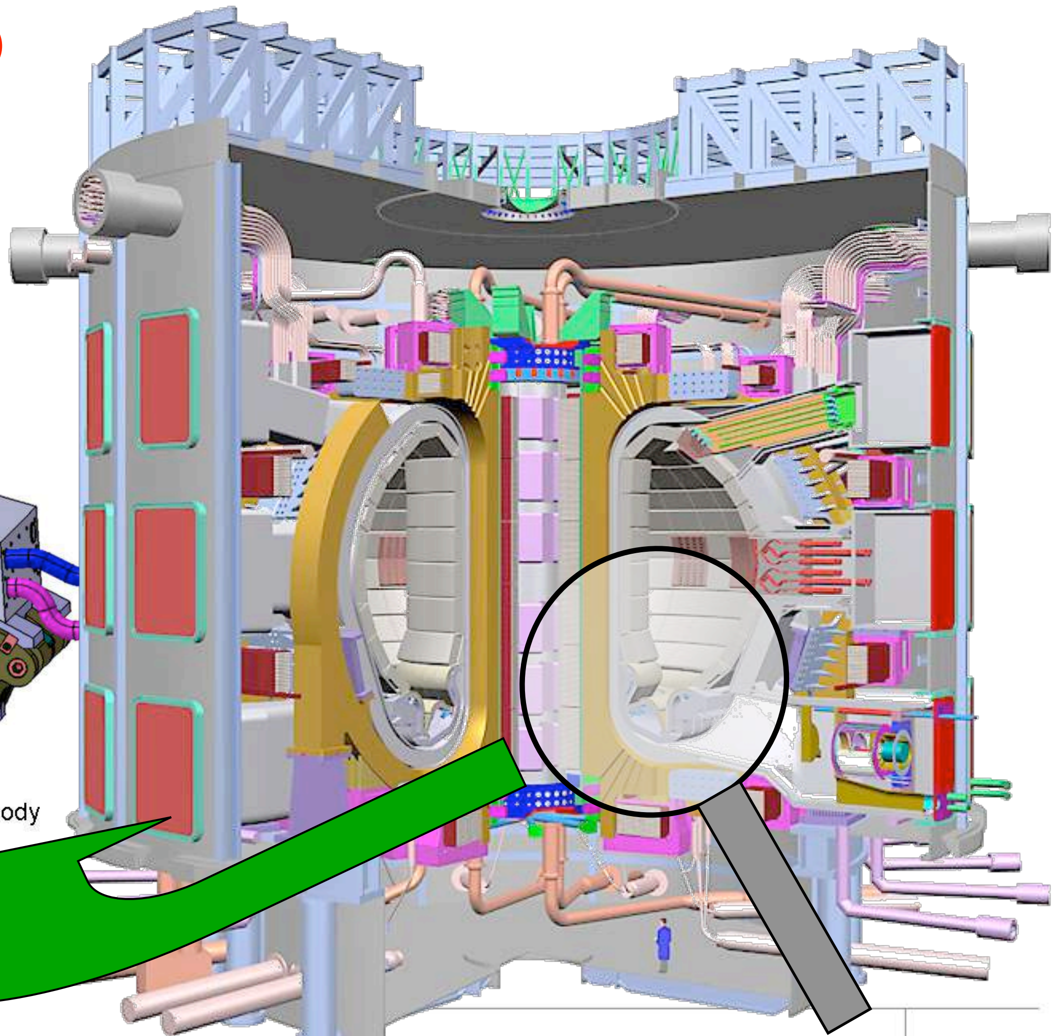
Divertor

Divertor system main functions :

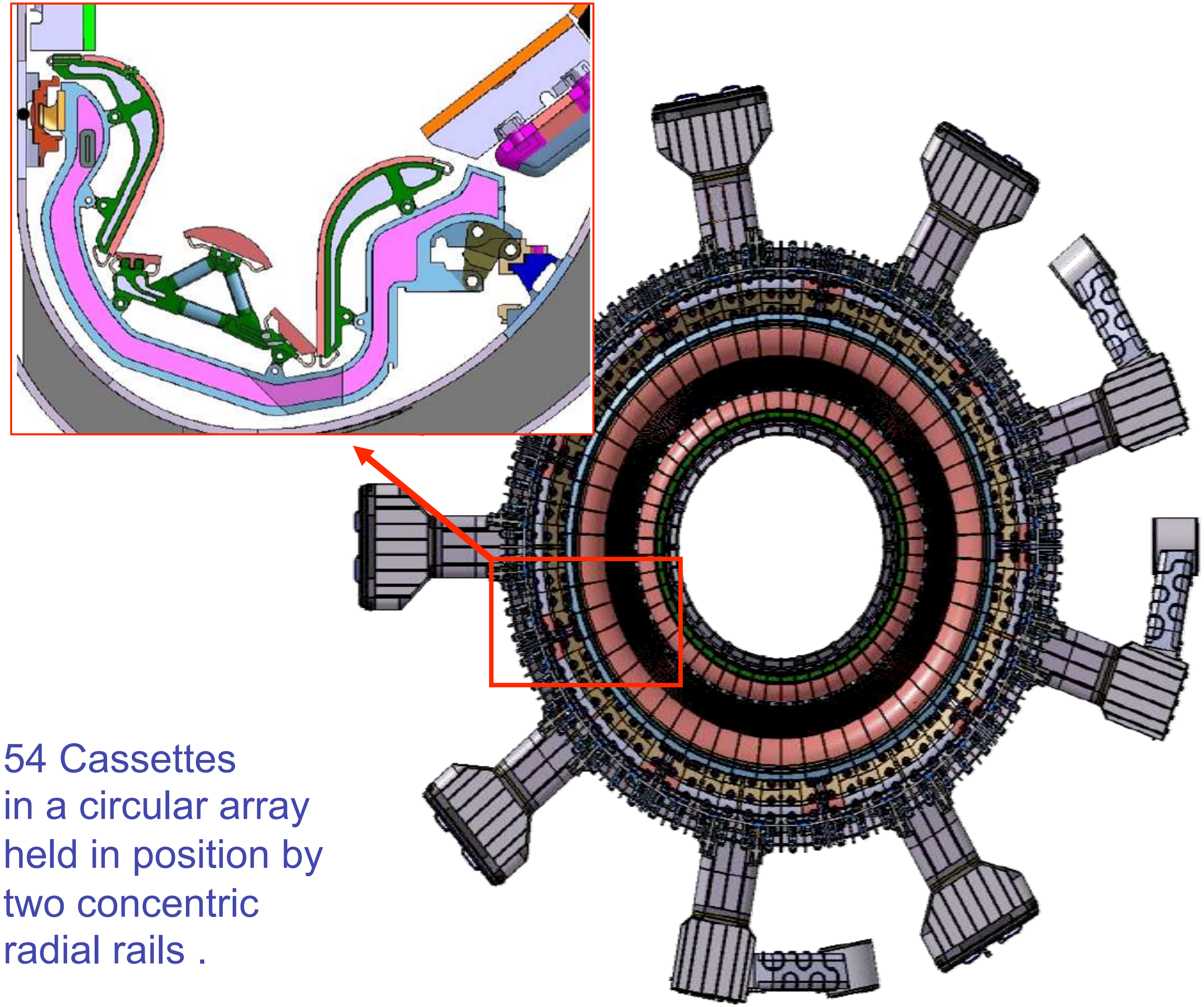
- Exhaust the major part of the plasma thermal power (including alpha power)

- Minimize the helium and impurities content in the plasma



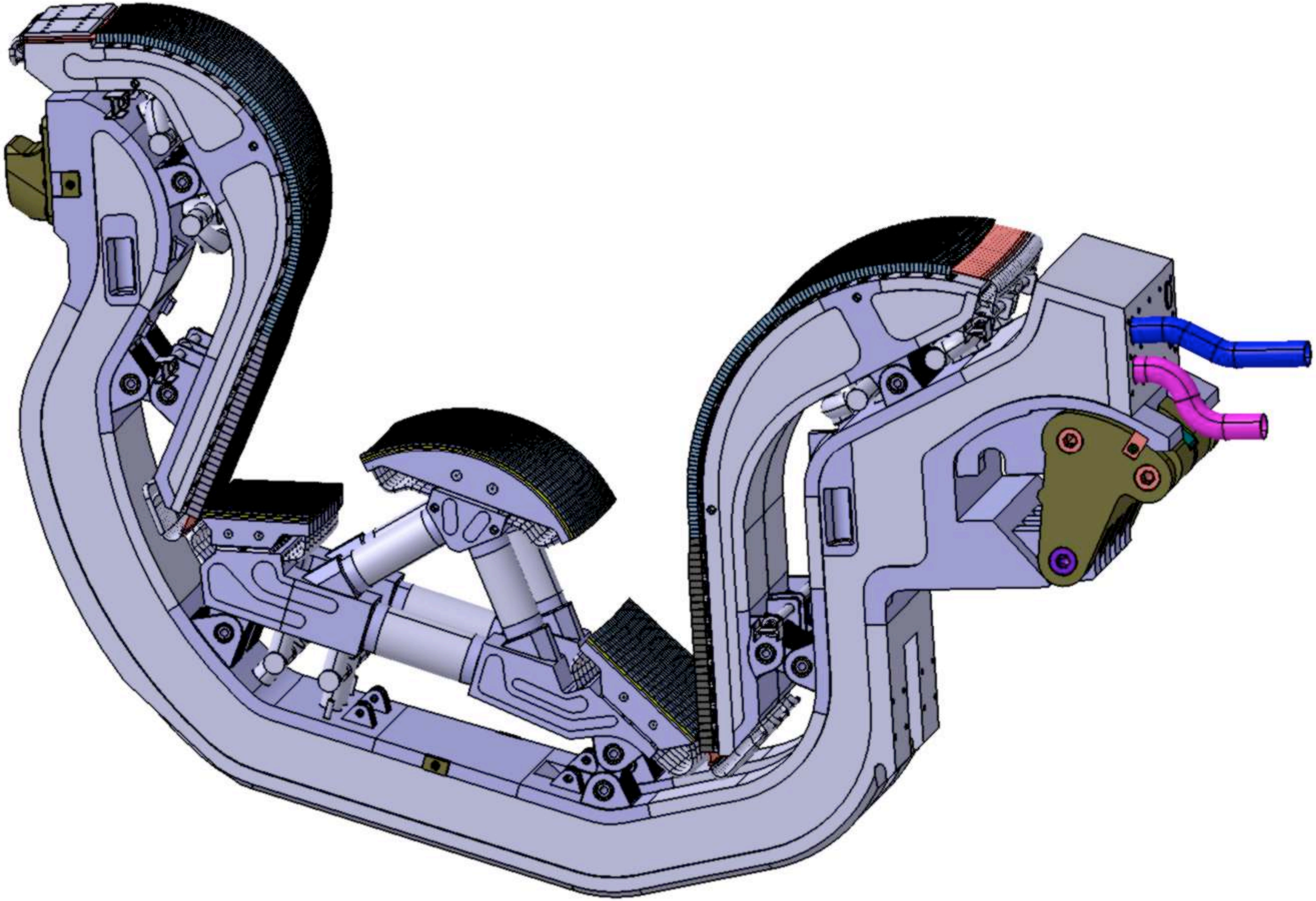


Divertor Cassette Layout



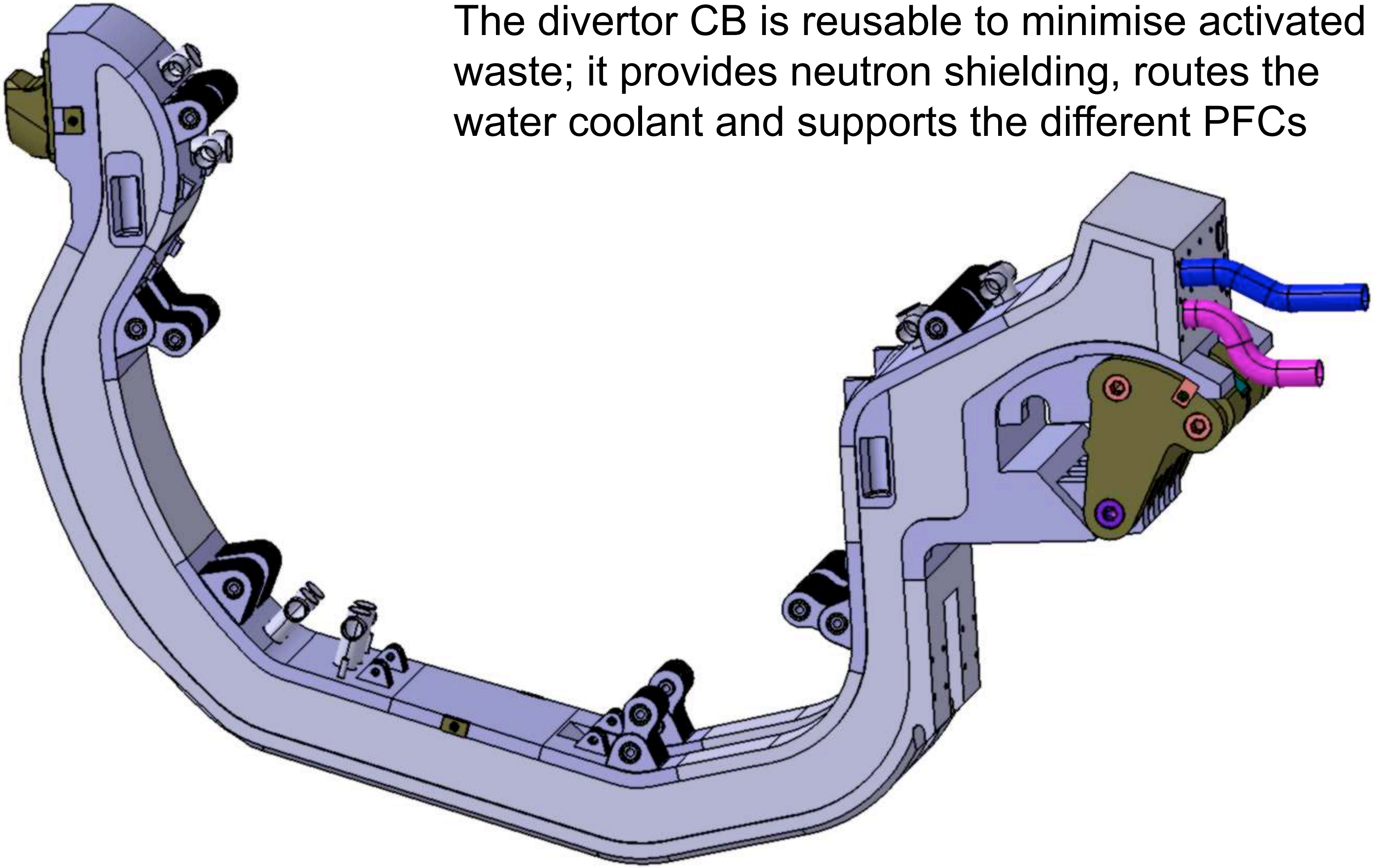
54 Cassettes
in a circular array
held in position by
two concentric
radial rails .

Divertor System

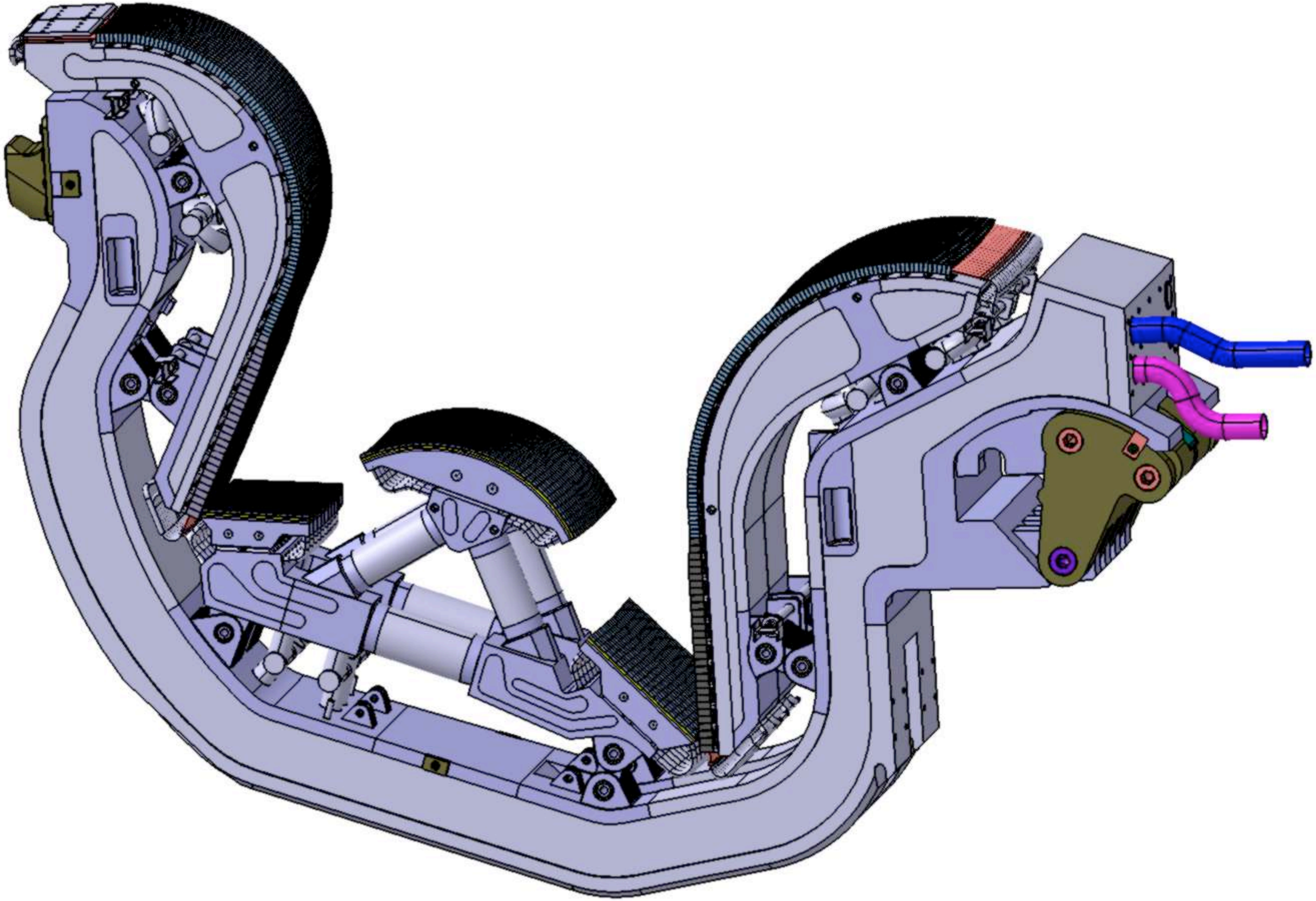


Cassette body

The divertor CB is reusable to minimise activated waste; it provides neutron shielding, routes the water coolant and supports the different PFCs

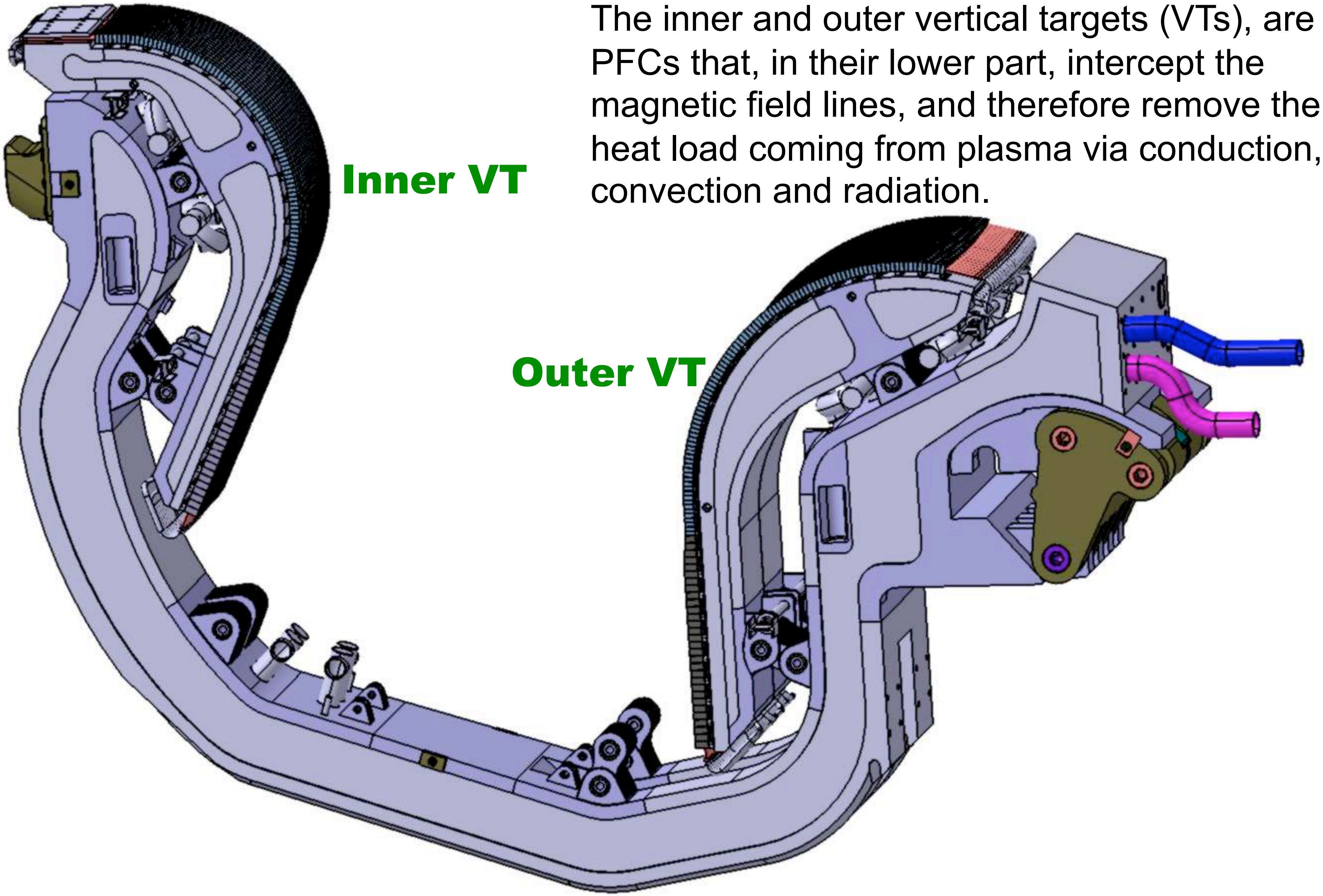


Divertor System

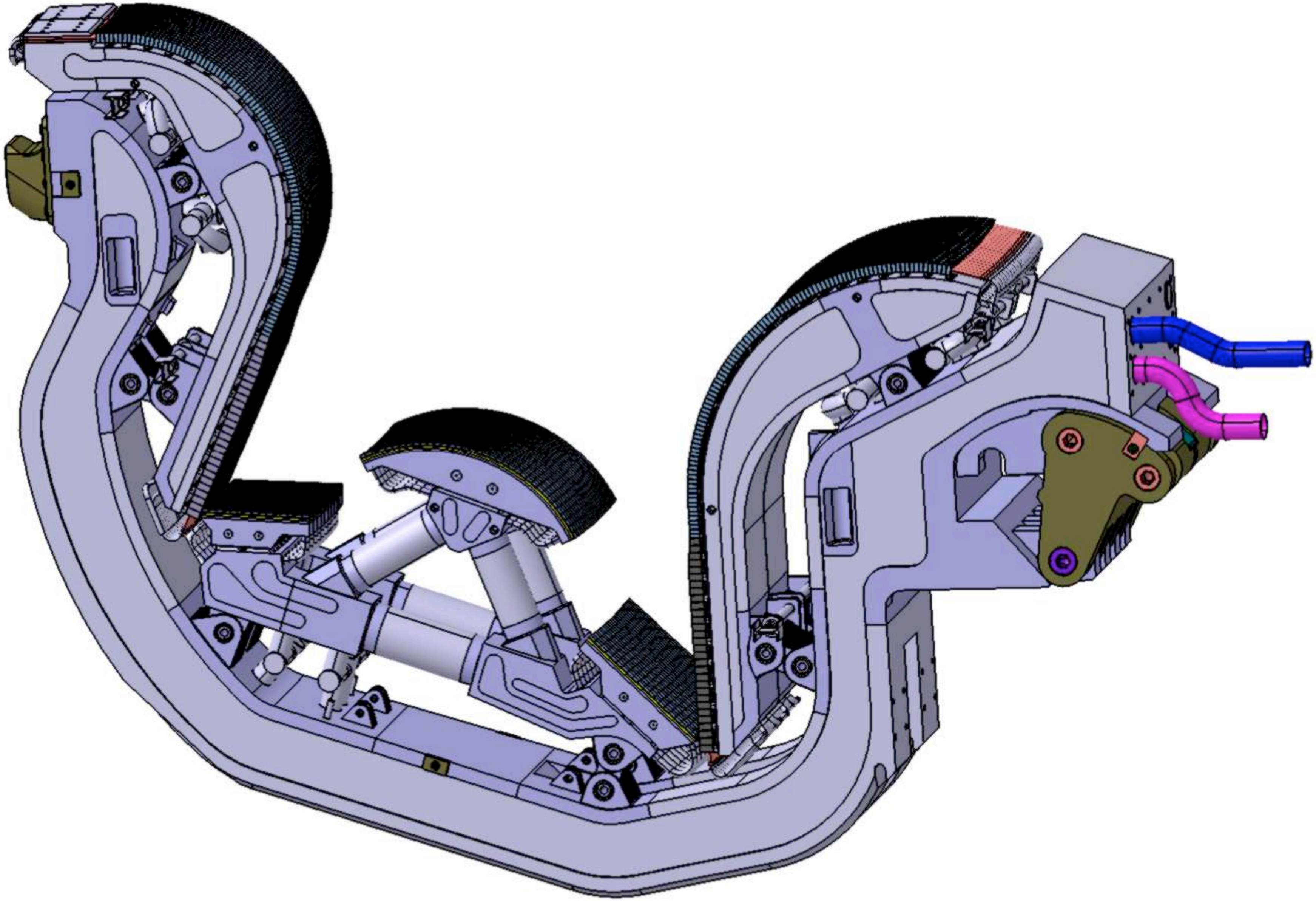


Vertical Targets

The inner and outer vertical targets (VTs), are the PFCs that, in their lower part, intercept the magnetic field lines, and therefore remove the heat load coming from plasma via conduction, convection and radiation.



Divertor System

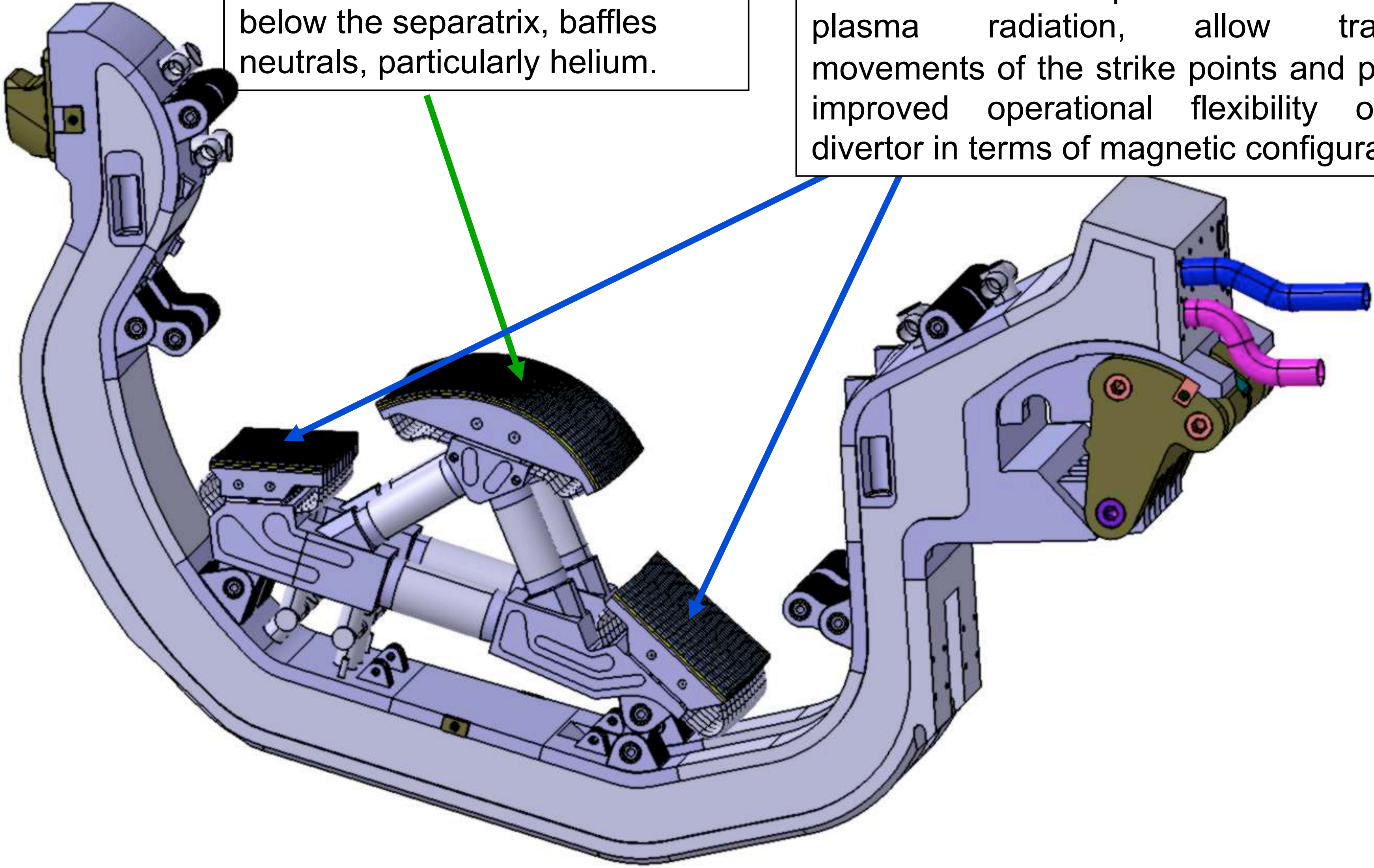


Divertor System

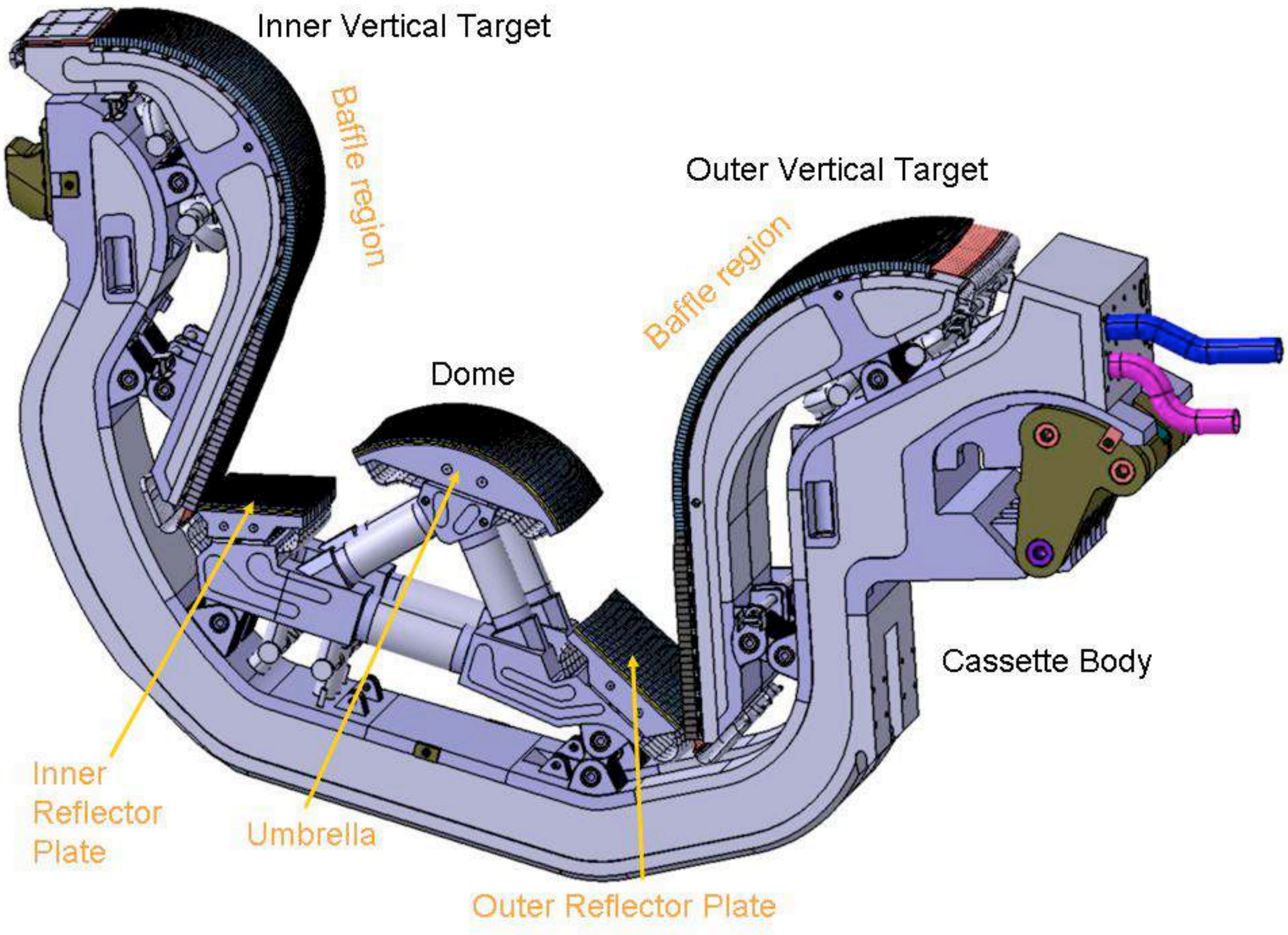
Dome

The "Umbrella", which is located below the separatrix, baffles neutrals, particularly helium.

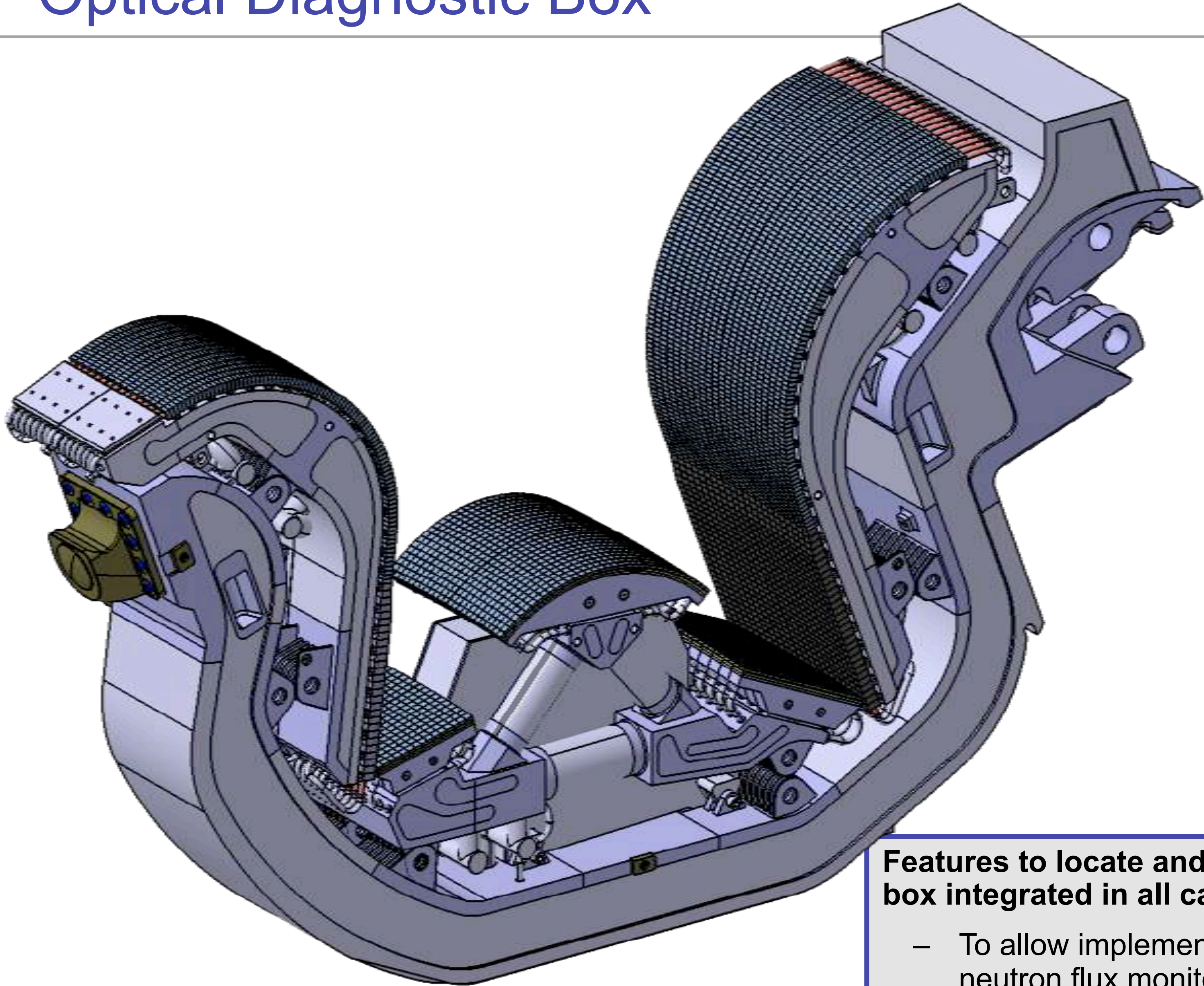
The inner and outer neutral "Particle Reflector Plates" protect the CB from plasma radiation, allow transient movements of the strike points and provide improved operational flexibility of the divertor in terms of magnetic configuration.



Summary of Terminology



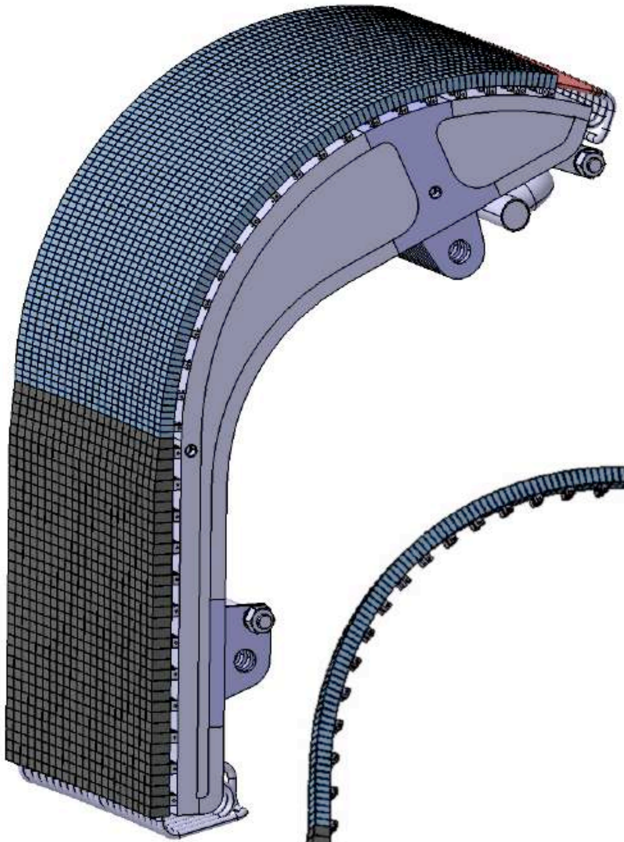
Optical Diagnostic Box

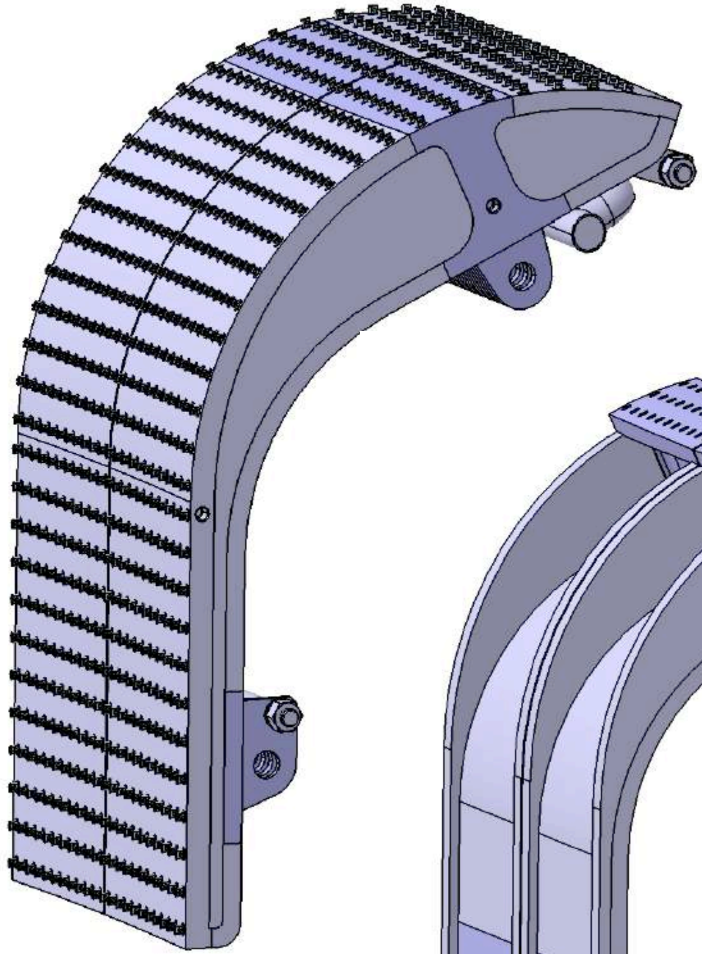


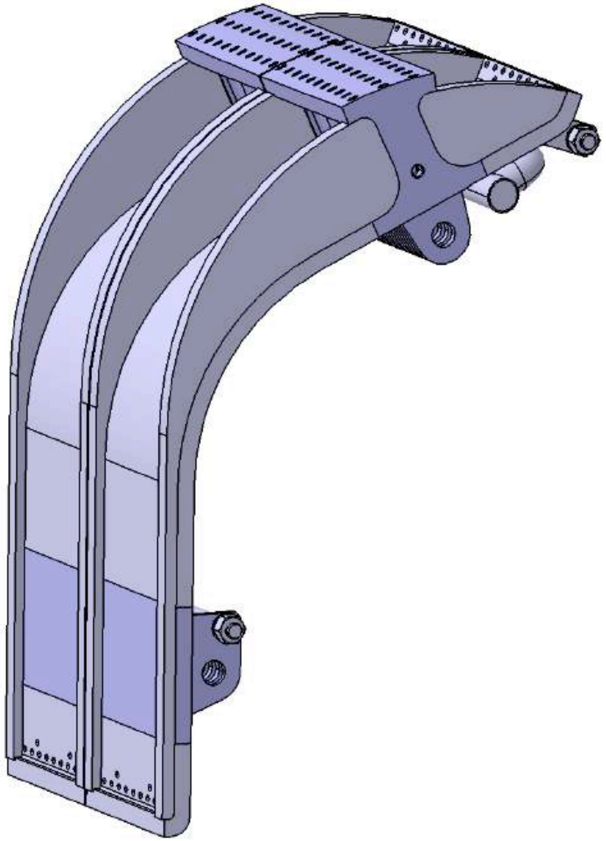
Features to locate and cool diagnostic box integrated in all cassettes

- To allow implementation of mirror box , neutron flux monitor , pick-up coils, bolometers
- Pipes plugged when no diagnostic box is required

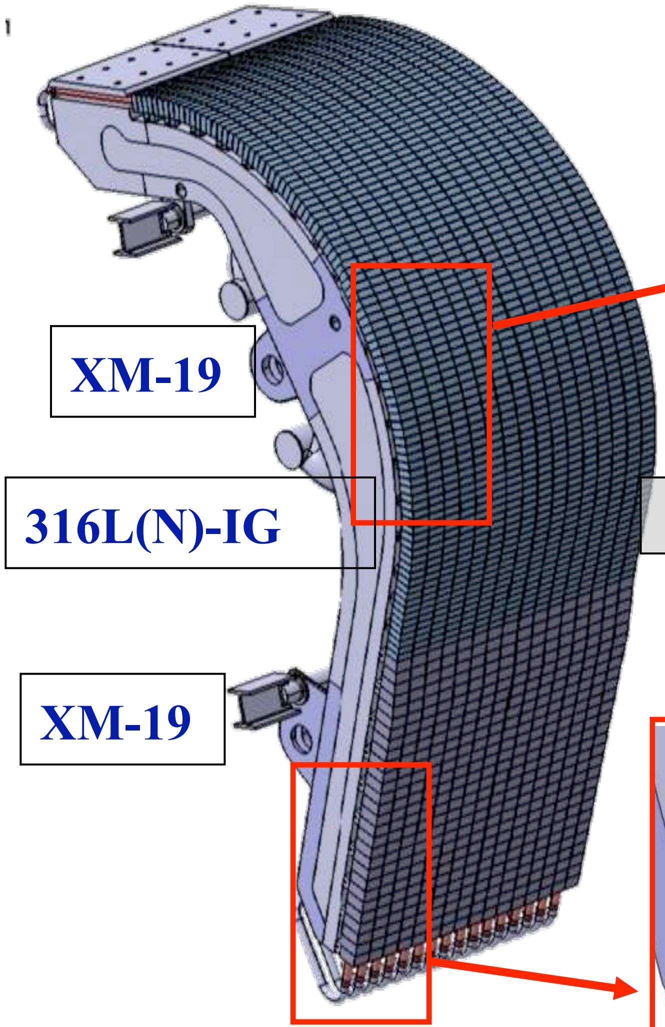
Plasma-Facing Components Vertical Targets





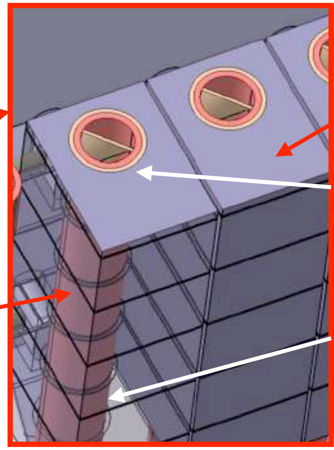


Plasma-Facing Components **Vertical Targets**



W monoblock

$5\ MW/m^2$



W monoblock

Copper Interlayer

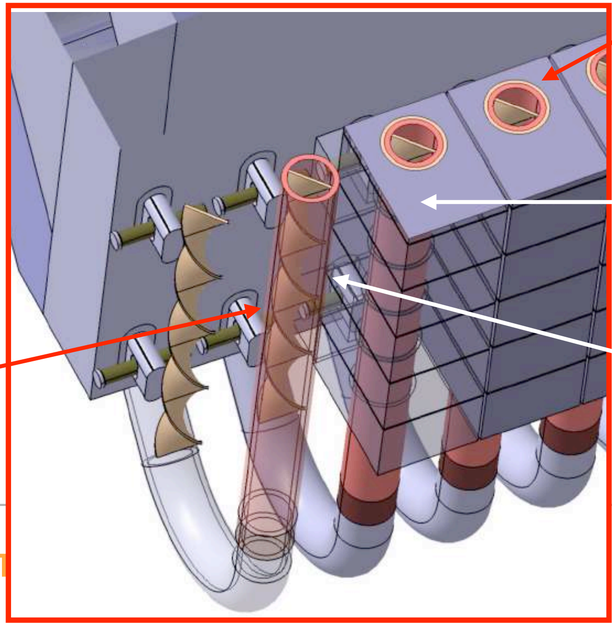
Smooth Tube

CuCrZr Heat Sink

CFC monoblock

$10\ MW/m^2$

$20\ MW/m^2$ 10 sec



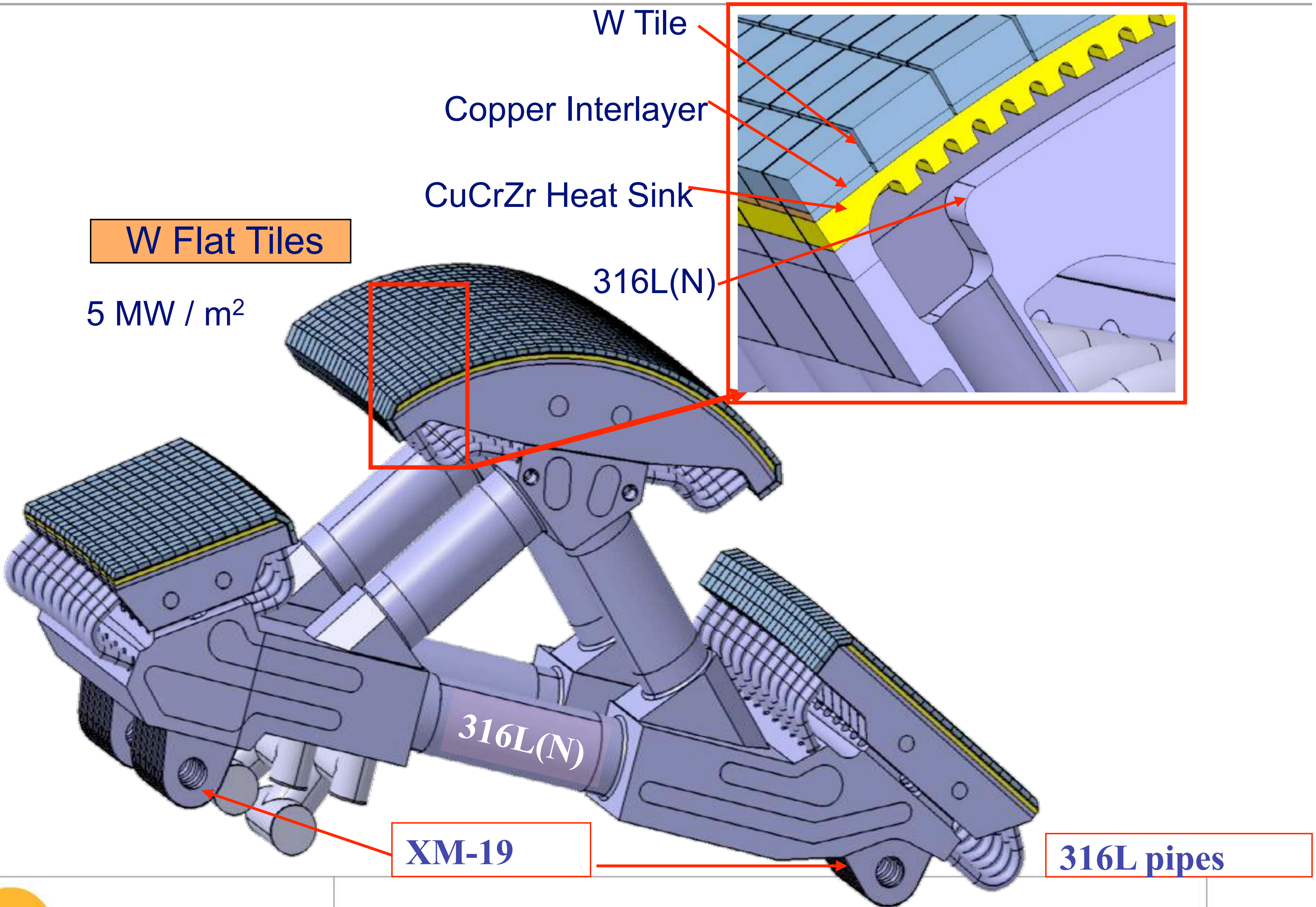
CFC monoblock
For first divertor set

Copper Interlayer

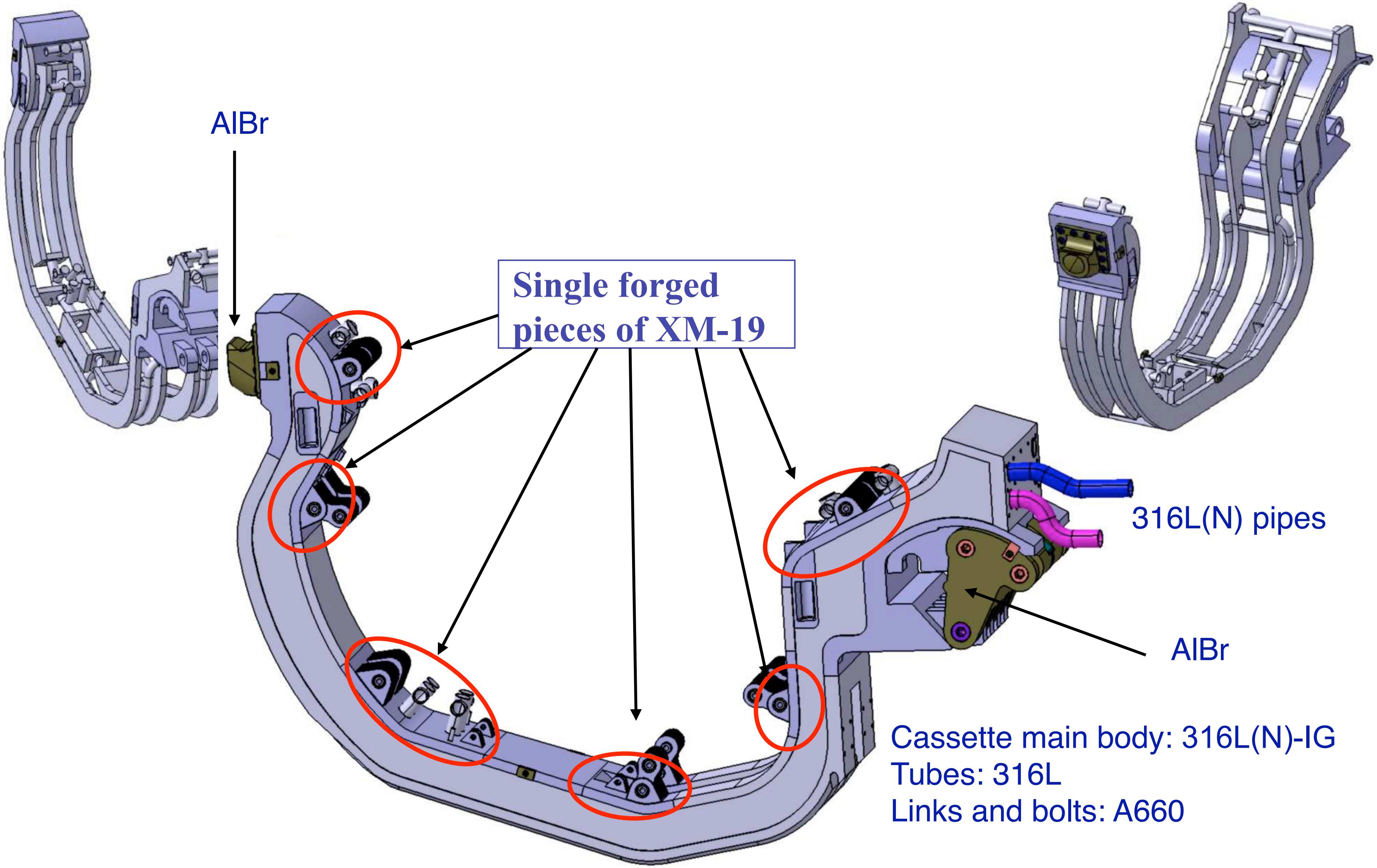
CuCrZr Heat Sink

Twisted tape: To increase the margins against the Critical Heat Flux

Plasma-Facing Components **Dome**



Cassette Body

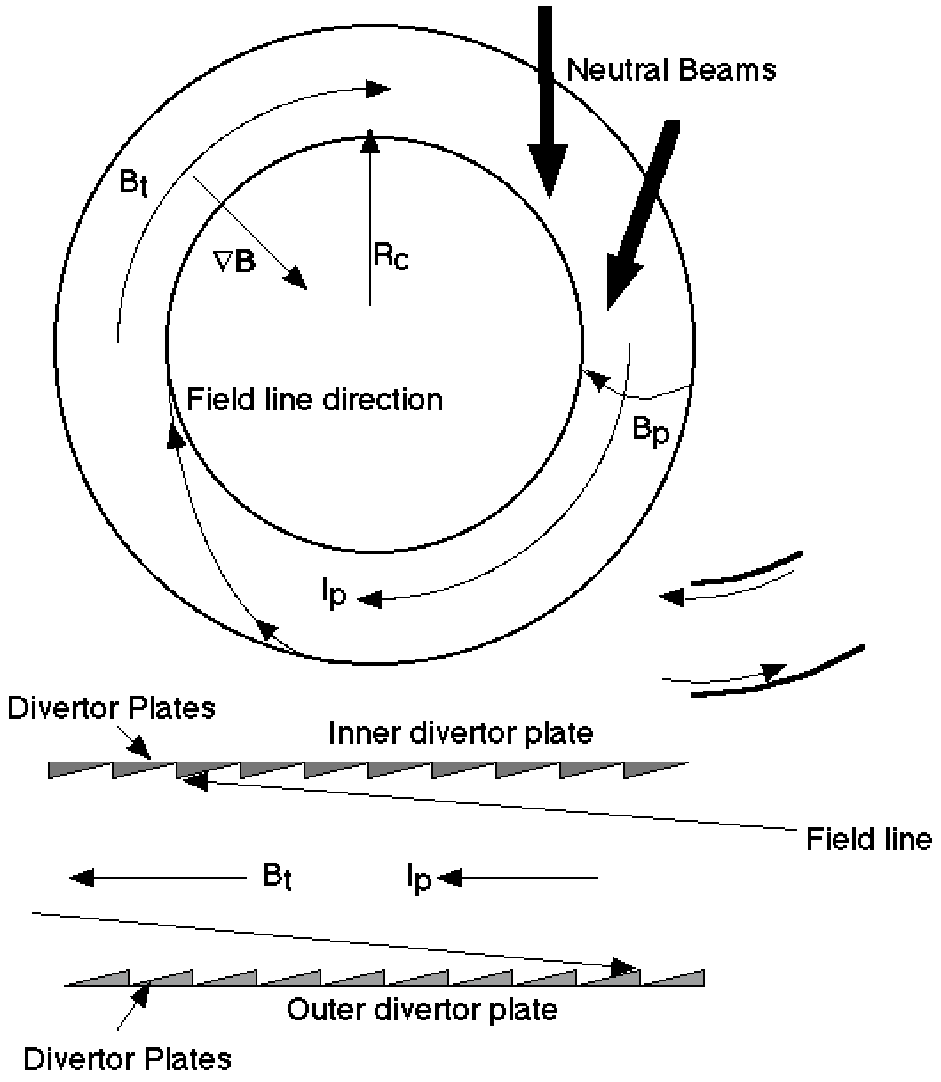


Alignment of PFCs

The PFCs shall be angled to avoid exposing the leading edges of the armour to the Scrape-Off Layer (SOL), otherwise the near normal incidence of the SOL on these edges would cause large amounts of carbon to be evaporated (or tungsten melted) with the inherent risk of poisoning of the plasma and/ or inducing a critical heat flux event in the water coolant.

A nominal step in the toroidal direction between adjacent targets of 3 mm is taken as a requirement

Orientation of Divertor, B_t, I_p and Neutral Beams

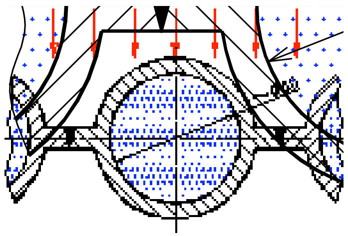
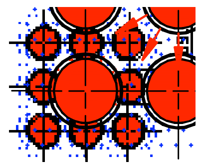
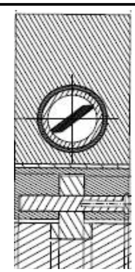


View fromTop

Power Handling **Design Values**

- The PFCs of the first divertor set are designed to withstand 3000 equivalent pulses of 400 s duration at nominal parameters, including 300 slow transients
- During normal operational conditions:
  - vertical target has a design surface heat flux up to 10 MW/m^2 (strike point region) and 5 MW/m^2 (baffle region)
- Under slow transient thermal loading conditions:
  - lower divertor vertical target geometry has a design surface heat flux up to 20 MW/m^2 for sub-pulses of less than 10 s
- The dome shall sustain design heat fluxes of up to 5 MW/m^2
- The umbrella and the particle reflector plates shall sustain local heat flux up to 10 MW/m^2, which can be transiently swept across the surface (about 2 s) as the plasma is returned to its correct position

Comparisons

HIGH HEAT FLUX COMPONENTS	FOSSILE FIRED BOILER WALL (ABB)	FISSION REACTOR (PWR) CORE	ITER DIVERTOR
DESIGN			 12/15 mm ID/OD
HEAT FLUX - average MW/m^2 - maximum MW/m^2	0.2 0.3	0.7 1.5	3 – 5 10 – 20
Max heat load MJ/m^2 Lifetime years Nr. of full load cycles Neutron damage dpa Materials	- 25 8000 - Ferritic-Martens. steel	- 4 10 10 Zircaloy - 4	10 ~ 5-8 3000 - 16000 0.2 CuCrZr & CFC/W
Coolant - pressure MPa - temperature °C - velocity m/s - leak rate g/s	Water-Steam 28 280-600 3 <50	Water 15 285-325 5 <50(SG)	Water 4 100 – 150 9 – 11 <10^{-7}

Coolant Flow Path

Divertor coolant design parameters

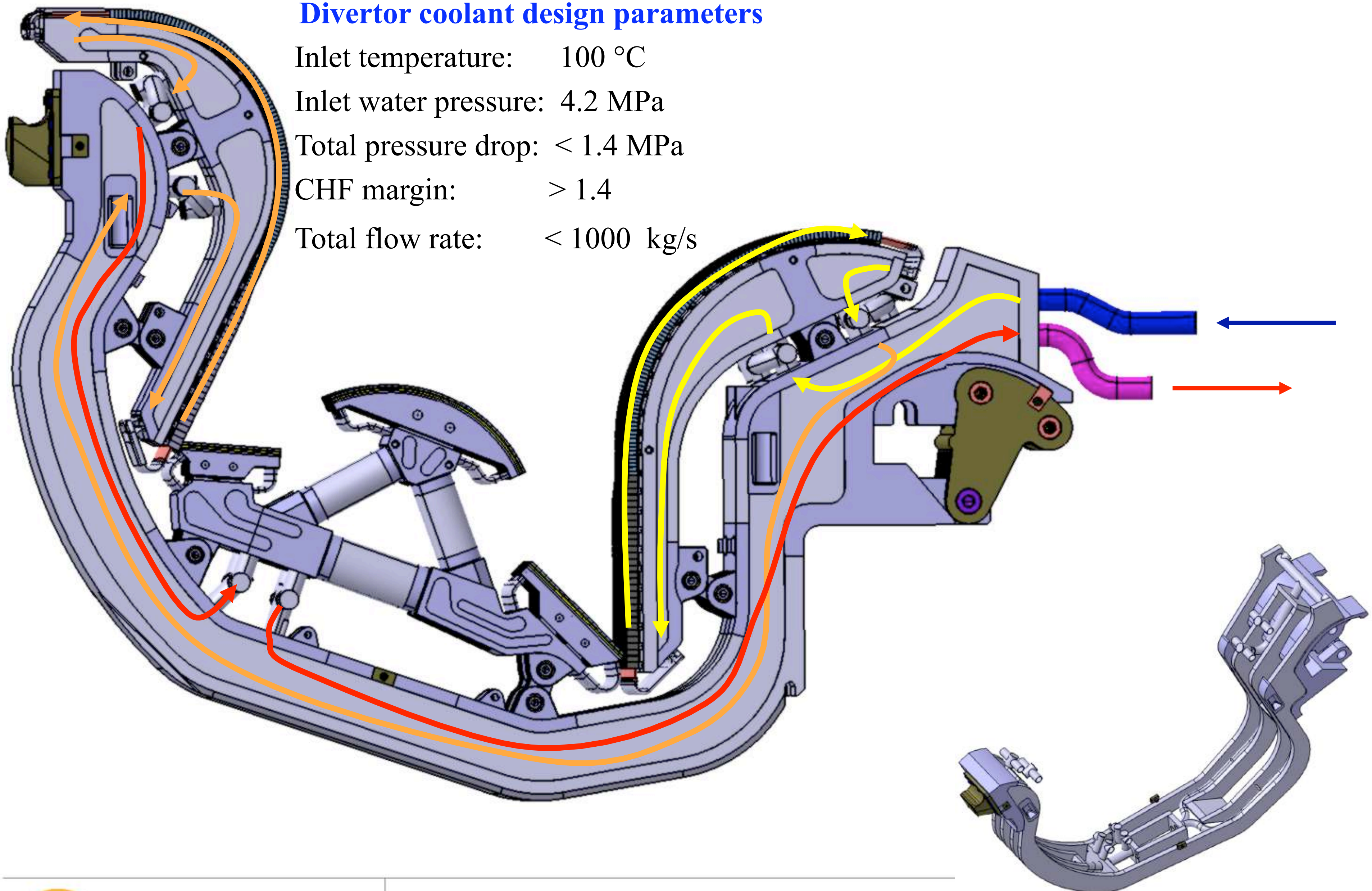
Inlet temperature: 100 °C

Inlet water pressure: 4.2 MPa

Total pressure drop: < 1.4 MPa

CHF margin: > 1.4

Total flow rate: < 1000 kg/s



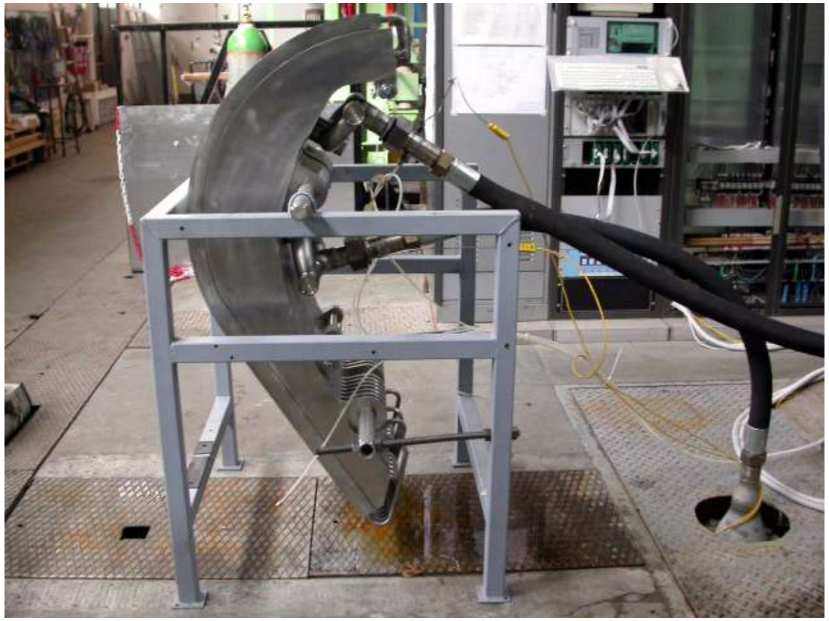
Thermo-Hydraulics Experiments

Pressure drop vs.flow rate have been measured on Outer and Inner Vertical target and Dome (ENEA Brasimone)

CEF 1 design parameters

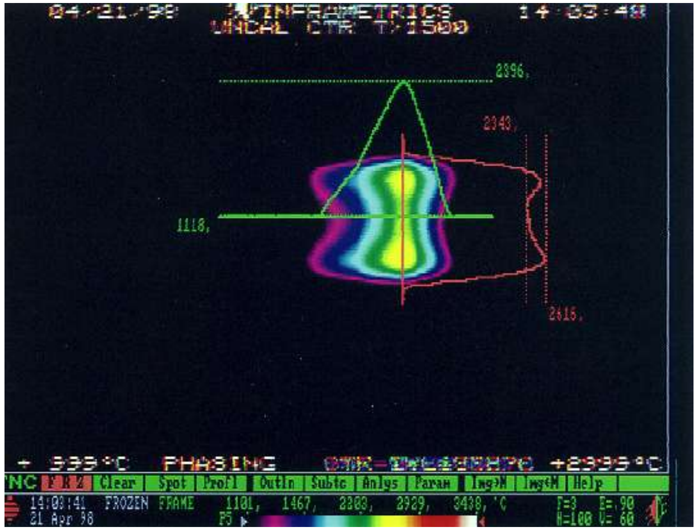
Tank design pressure (MPa)	0.5
Loop design temperature (°C)	140
Pump max. flowrate (kg/s)	2 x 70
Pump max head (MPa)	2 x 1.2
Electrical heater power (kW)	2 x 60

Hydraulic testing of DOME



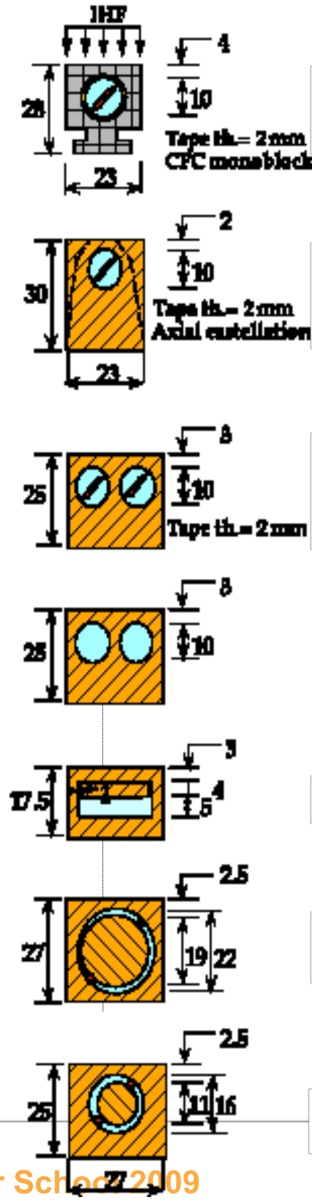
Hydraulic testing of IVT

Critical Heat Flux



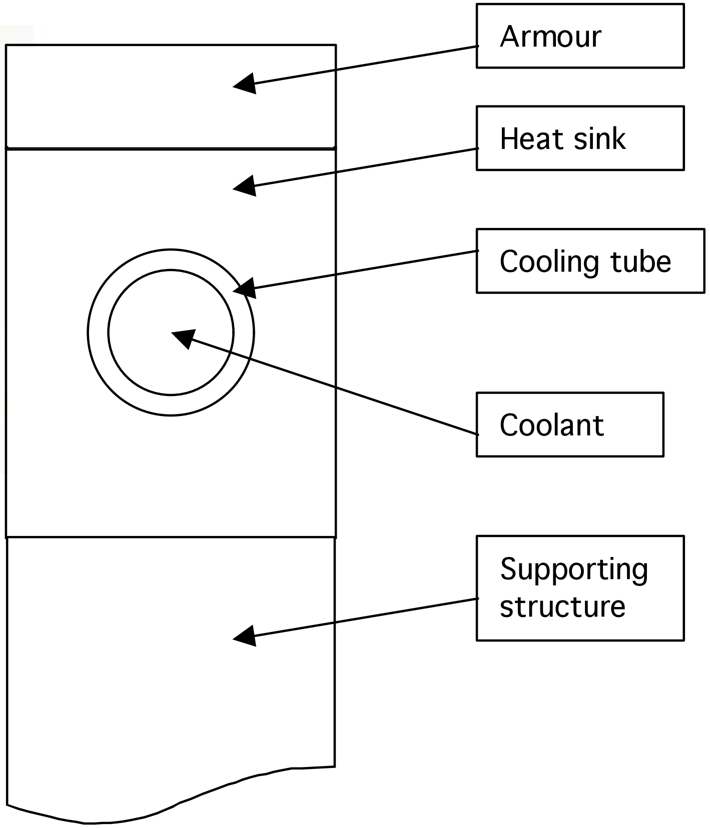
THERMO-HYDRAULIC TEST RESULTS BY CEA CADARACHE 1996-98

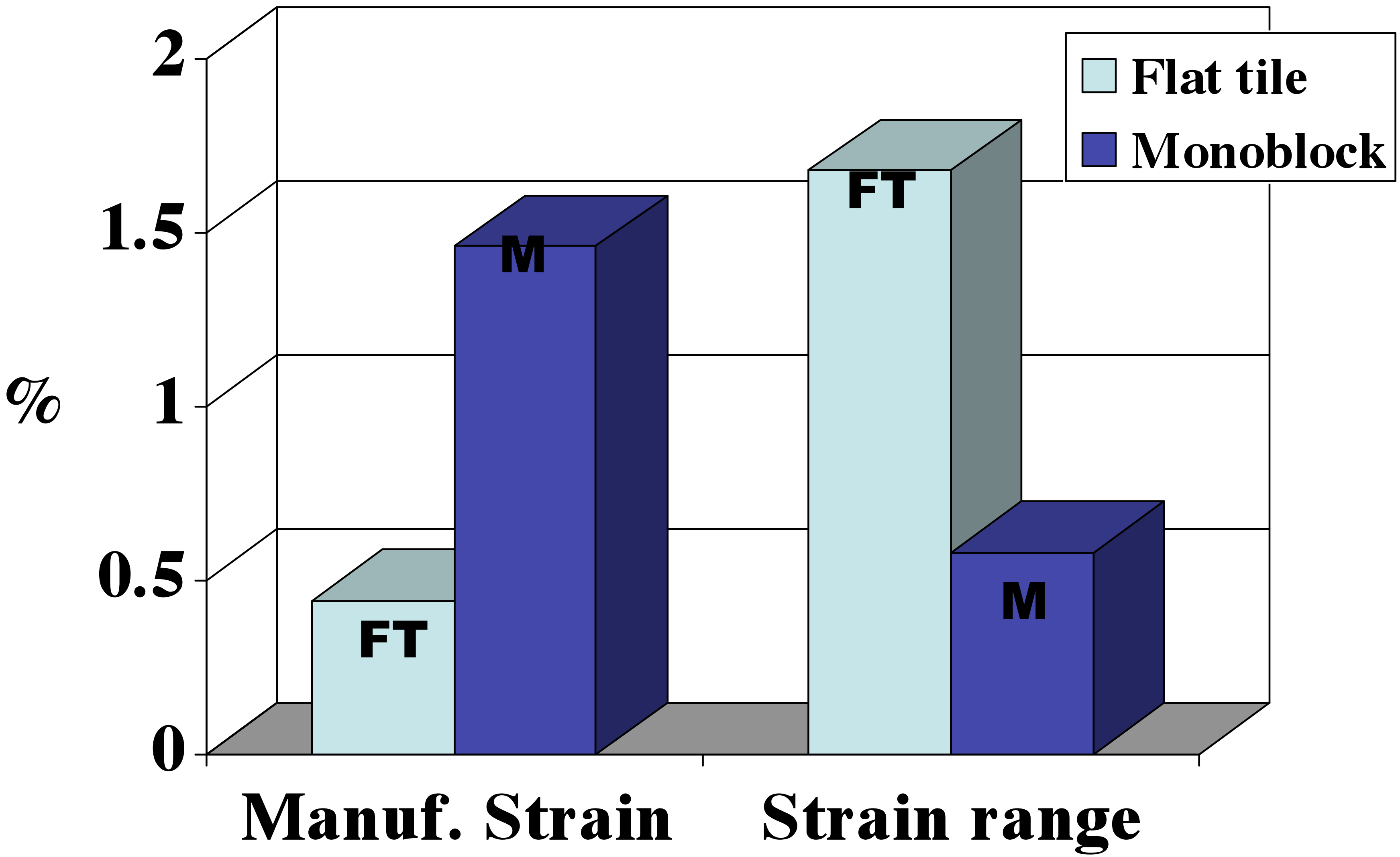
- Mock-ups: CuCrZr (1996), DS-Cu (1997), CFC monoblock (1998)
- Heated length: 100 mm uniform, 200 mm peaked heat flux profile
- Interpolated results for 3.5 MPa, 100 C subcooling, 12 m/s (ITER conditions)



Year	Type	Incident Critical Heat Flux (MW/m²): Uniform	Peaked	Pressure Drop (MPa/m)
1998	Swirl tube: Twist ratio 2	~ 22	~ 30	-
1997	Swirl tube: Twist ratio 2	~ 28	~ 34	~ 0.61
	Twist ratio 4	~ 21	~ 24	~ 0.41
1996	Swirl tube: Twist ratio 2:	~ 27	~ 45	~ 0.75
	Twist ratio 4:	~ 24	~ 35	~ 0.44
1996	Smooth tube:	~ 17	-	~ 0.20
1996	Hypervapotron:	~ 35	~ 43	~ 0.56
1996	Annular flow: Twist ratio: 2	~ 23	-	~ 1.2
1996	Annular flow: Twist ratio: 2	~ 27	-	~ 1.0

Terminology, Flat Tile and Monoblock



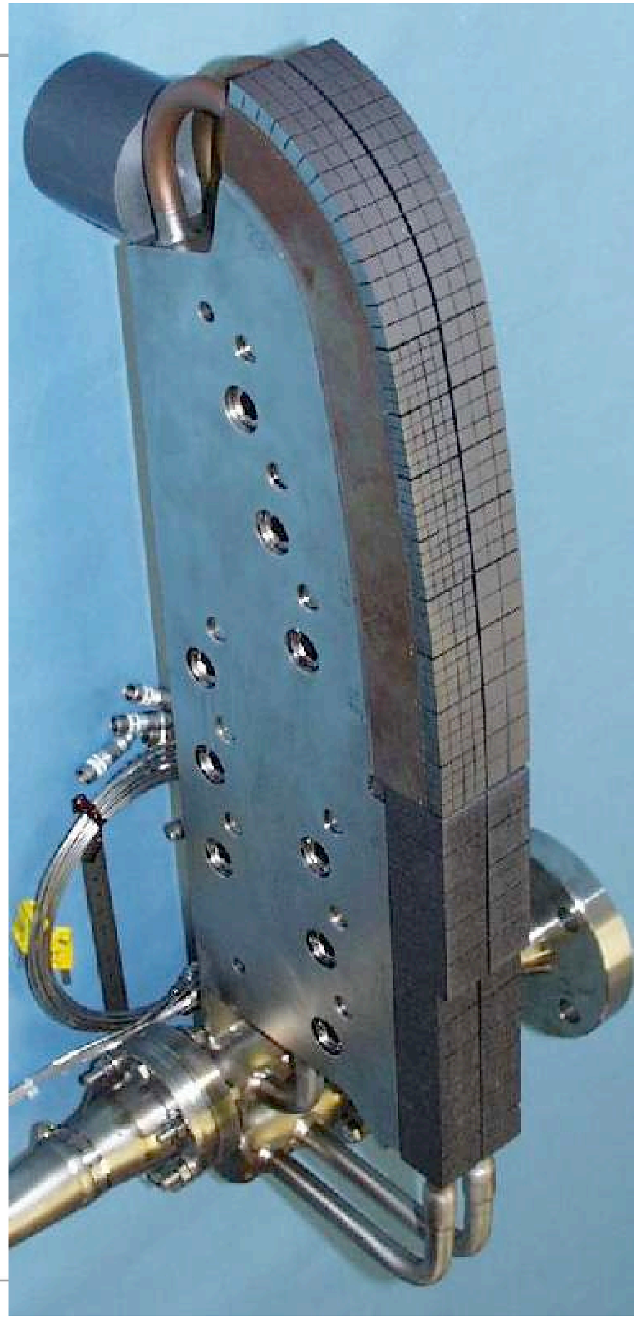


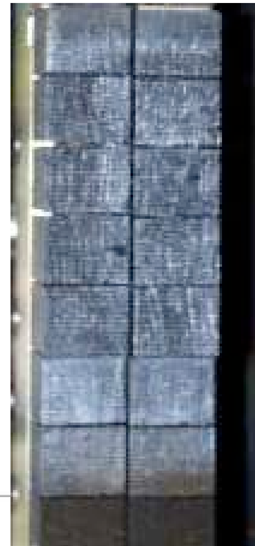
Flat tile
Monoblock
2
1.5
1
0.5
0
%
FT
M
FT
M
Manuf. Strain
Strain range

Vertical Target
Medium-Scale Prototype

Test results

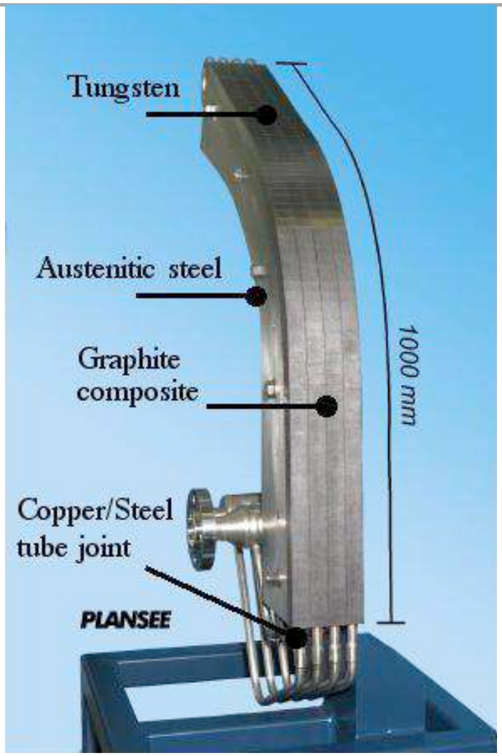
- **W macrobrush:**
  15 MW/m² x 1000 cycles
- **CFC monoblock**
  20 MW/m² x 2000 cycles
- **CHF test > 30 MW/m²**

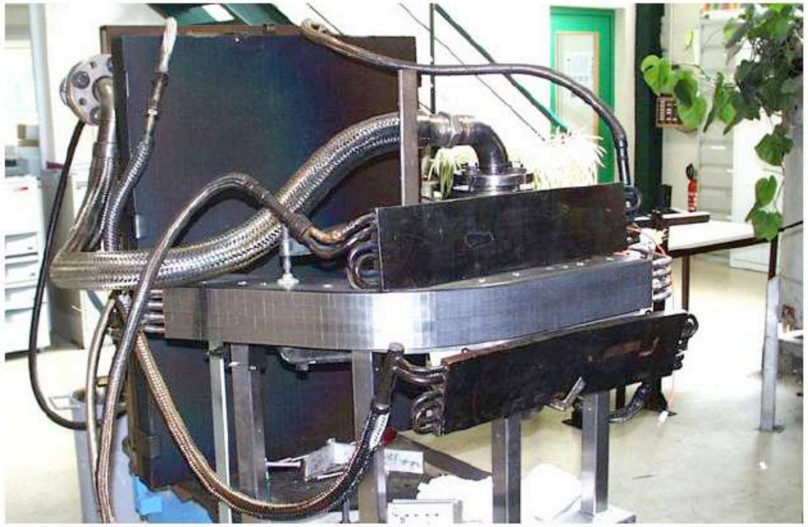




PLANSEE

Vertical Target
Full-Scale Prototype



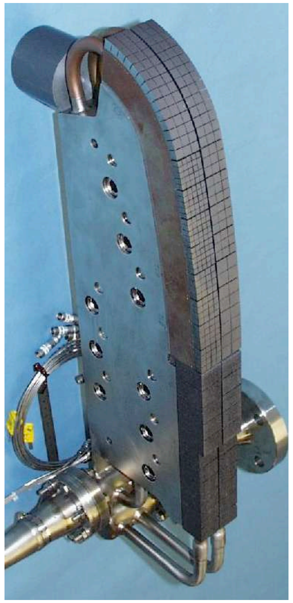


- **W monoblocks:**
  10 MW/m² x 1000 cycles

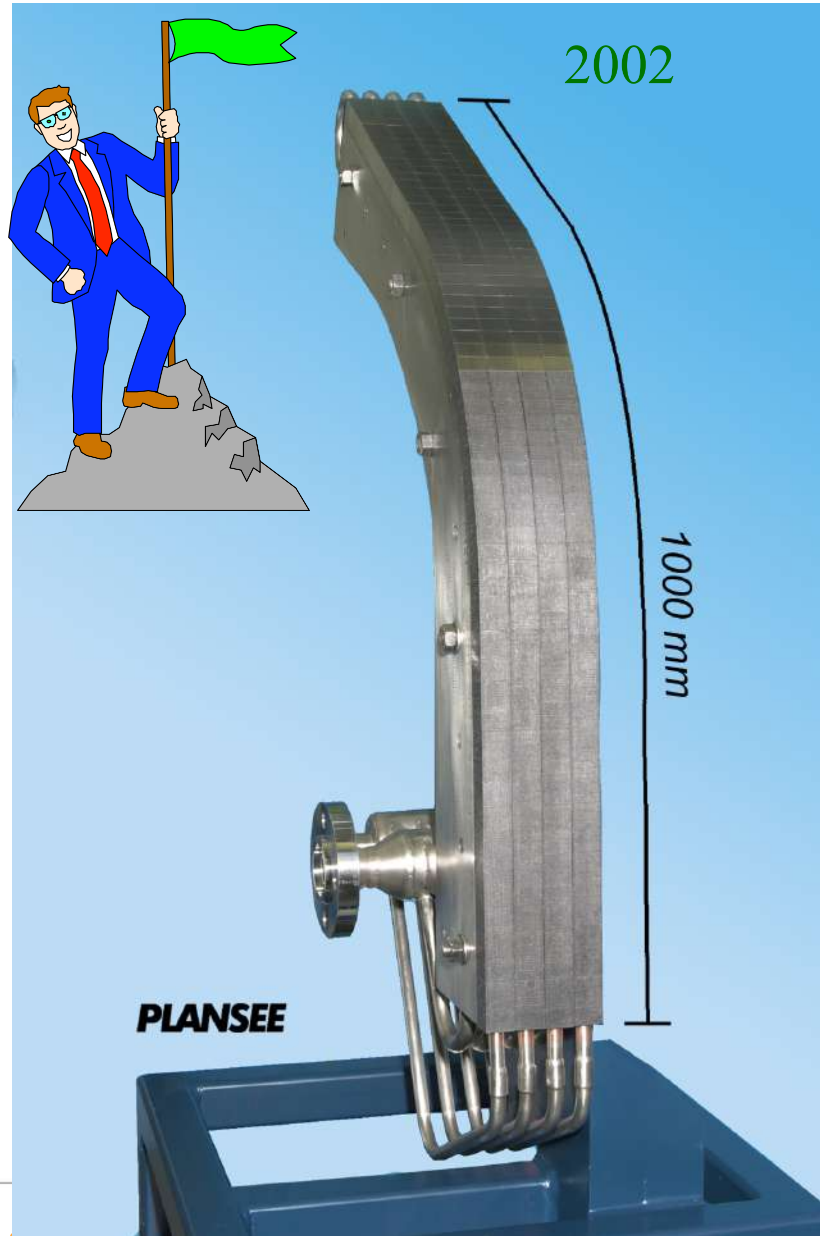
- **CFC monoblock**
  10 MW/m² x 1000 cycles
  20 MW/m² x 1000 cycles
  23 MW/m² x 1000 cycles

AREVA

1998



1995



Vertical Target component with W armour

Tested in FE200 facility (50°C-12 m/s – 3.3MPa)

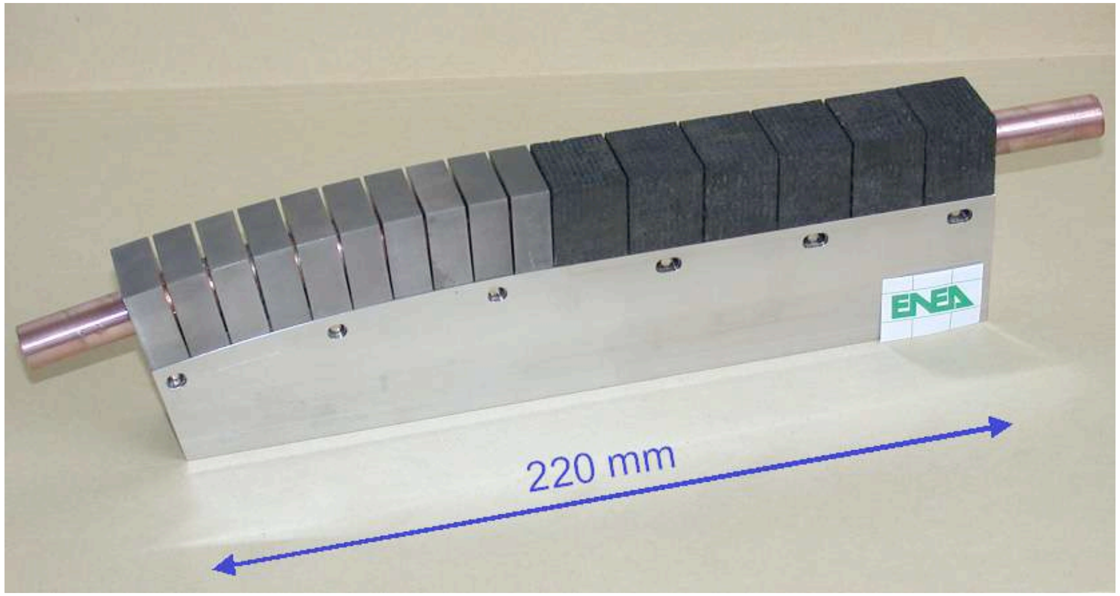
5 MW/m^2 x 100 cycles

10 MW/m^2 x 1000 cycles

20 MW/m^2 x 1000 cycles

Vertical Target Medium Scale Prototype by Hot Radial Pressing

The testing of medium-scale vertical target prototype manufactured by HRP-Hot Radial Pressing at ENEA Frascati in the FE200 facility (CEA-Areva)



3000 cycles at 10 MW/m^2

2000 cycles at 20 MW/m^2 on CFC and 15 MW/m^2 on W)

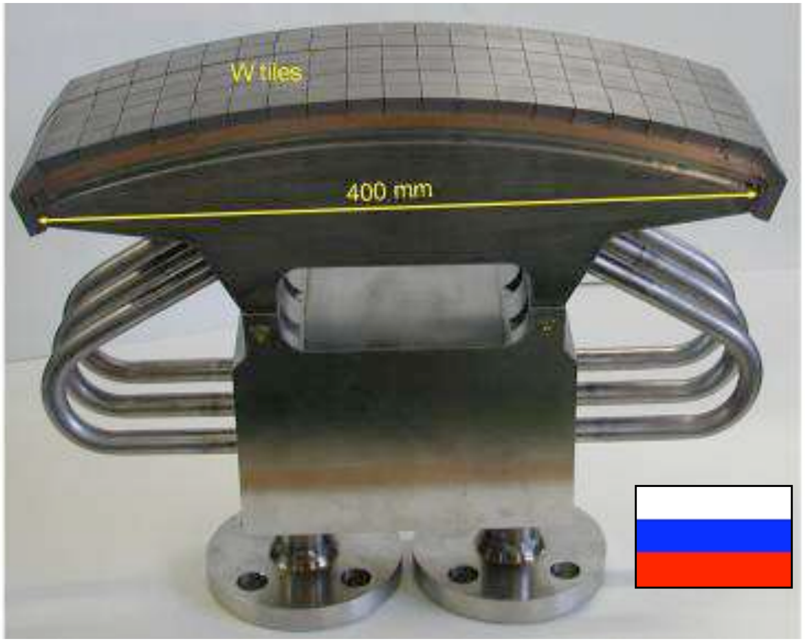
Experimental critical heat flux of 35 MW/m^2 on the CFC part – (10m/s, Tin=100 Tout=127 p=3.3MPa)

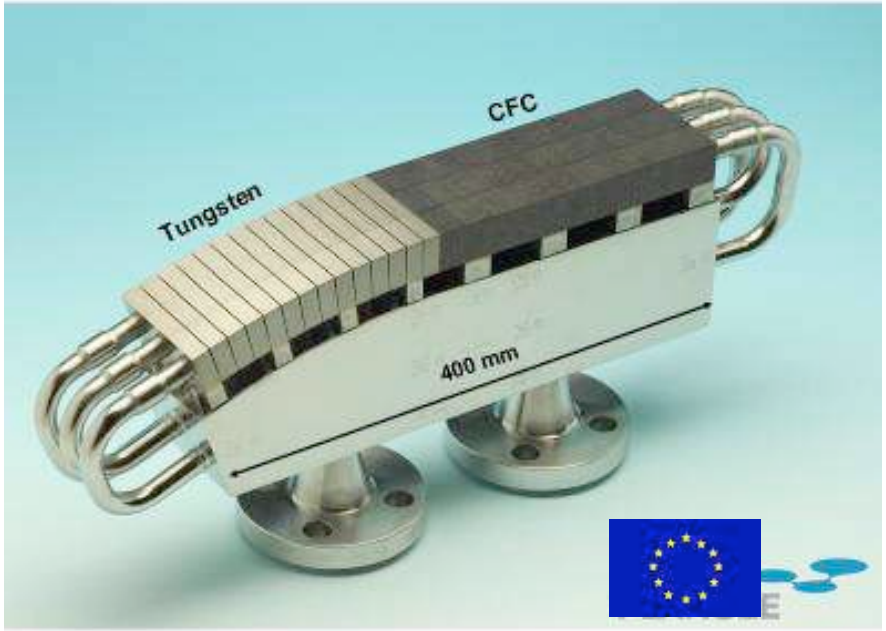
600 mm

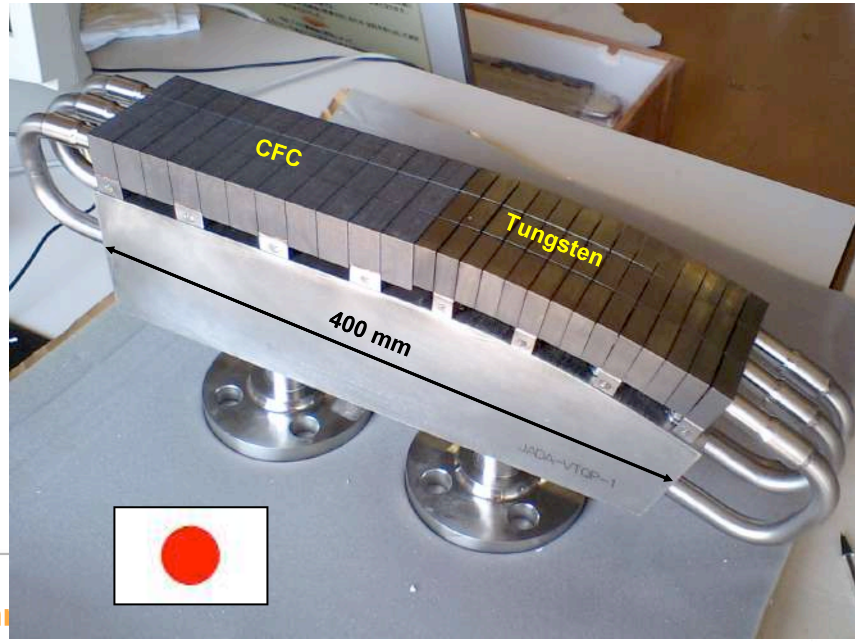
- W-Cu by casting
- Cu-CuCrZr by CuInSnNi (STEMET 1108) brazing
- 18.5 MW/m^2 x 1000 cycles

Divertor Qualification Prototypes

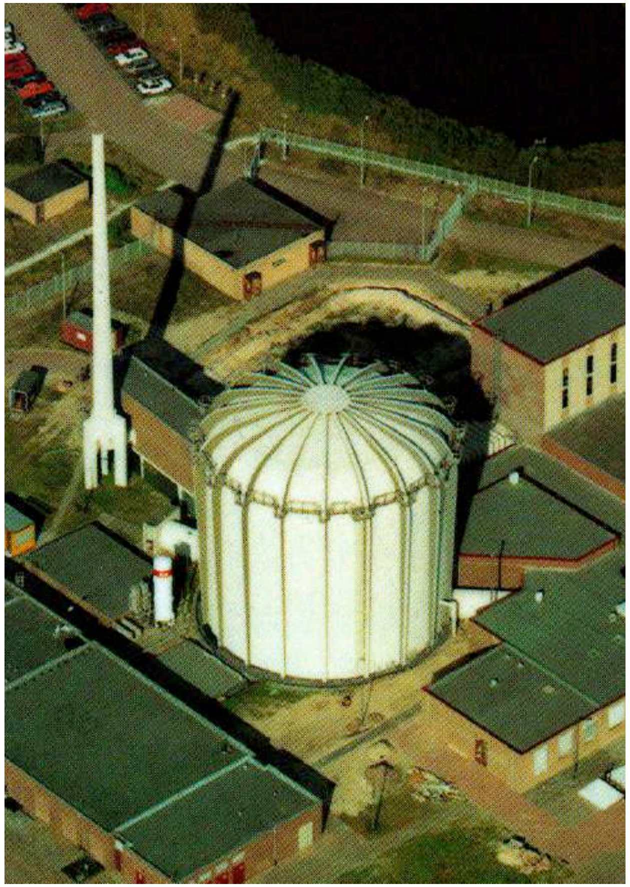
A qualification is "…*needed for the critical procurement packages shared by multi-Parties*…", including the divertor







Neutron-Irradiation Experiments PARIDE 1 - 4



High Flux Reactor
Petten, Netherlands

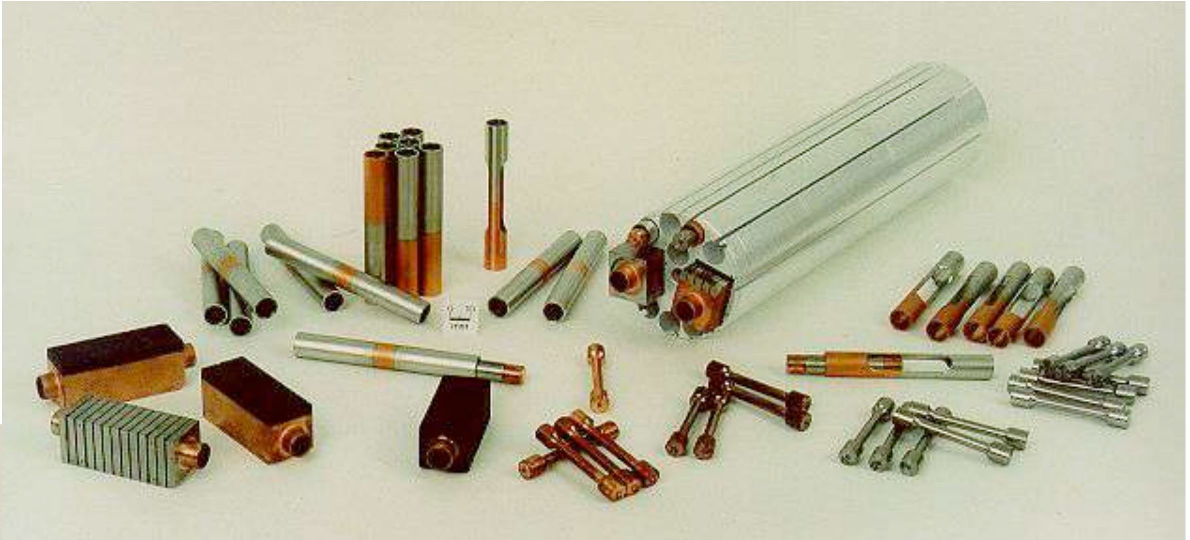
PARIDE 1:
- temperature: 350°C
- target fluence: 0.5 dpa

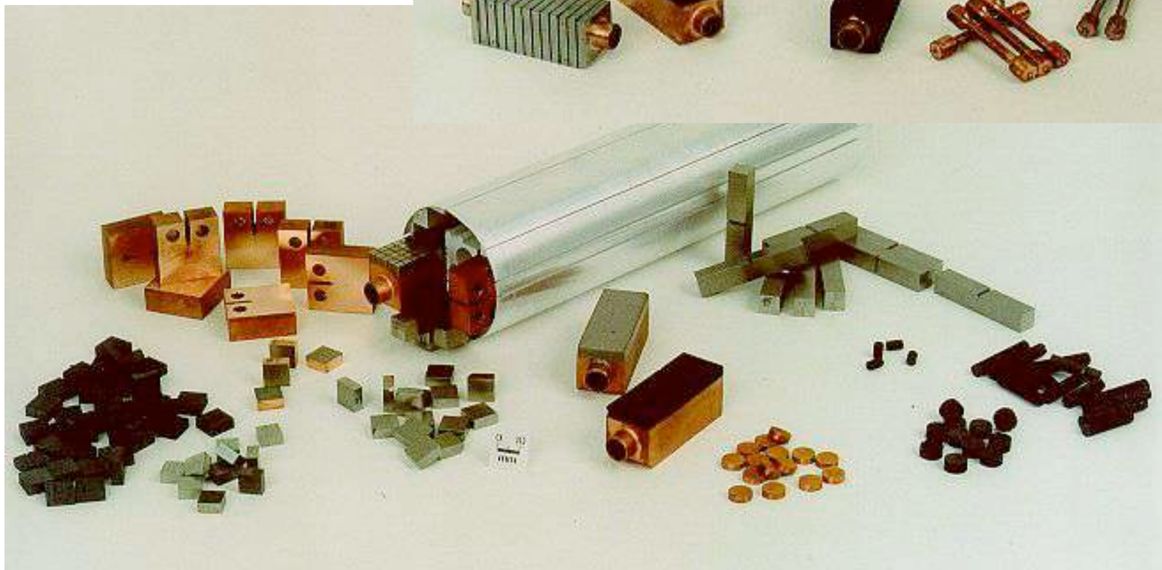
PARIDE 2:
- temperature: 700°C
- target fluence: 0.5 dpa

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

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

Irradiation





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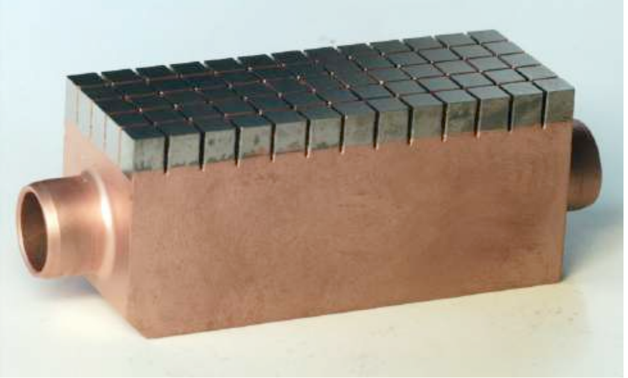
in der Helmholzgemeinschaft
EURATOM-Association

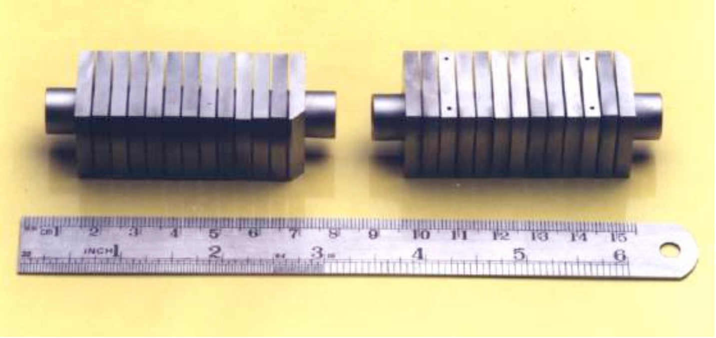
Irradiation

Testing of Tungsten Mock-Ups



Unirradiated
- **1000 cycles x 14 MW/m² – no failure**

200°C, 0.1 and 0.5 dpa in tungsten
- **Failure limit: 10 MW/m²**



Unirradiated
- **1000 cycles x 20 MW/m² – no failure**

200°C, 0.1 and 0.5 dpa in tungsten
- **Successfully tested up to 18 MW/m²**

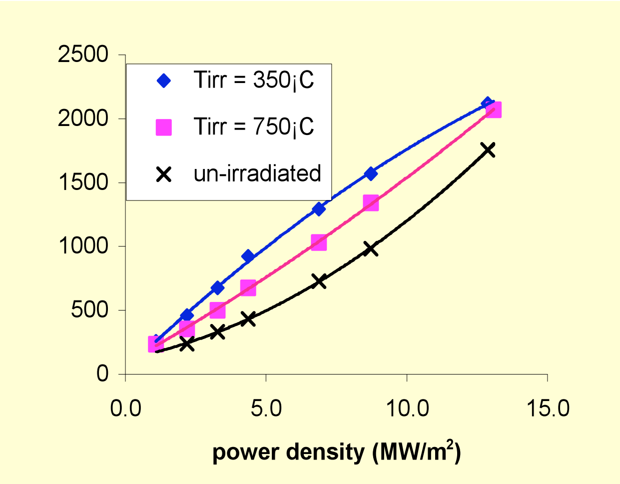
Testing of CFC Mock-Ups

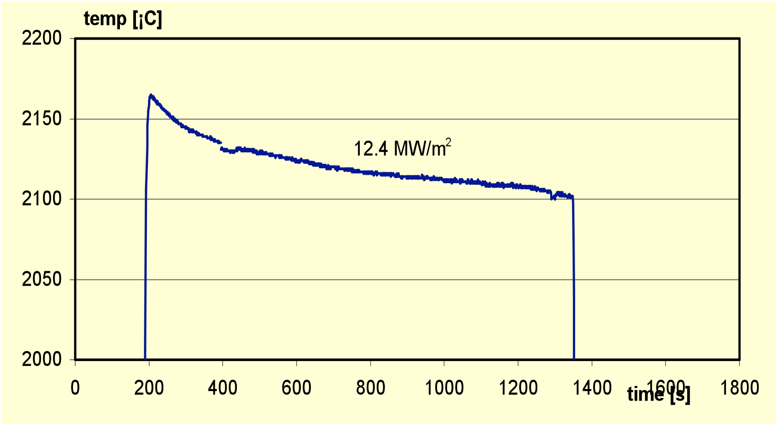
Unirradiated

- 1000 cycles x 19 MW/m^2 – no failure
- 700 cycles x 23 MW/m2 – no failure (erosion)

Irradiated at 200°C, 0.2 dpa in CFC

- 1000 cycles x 10 MW/m2 – no failure
- 1000 cycles x 12 MW/m2 – no failure





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Irradiation

- High heat flux testing of irradiated CFC mock-ups is limited by the surface temperature, which, however, decreases during testing due to thermal annealing
- The irradiated pure Cu interlayer leads to a reduction of the high heat flux performances in a flat tile geometry
- The irradiated pure Cu interlayer does not appear to reduce the high heat flux performances in a monoblock geometry → **Monoblock geometry appears mandatory for the vertical targets in the DT phase**

Outline

Nuclear Fusion and the ITER Project

ITER Internal Components

ITER Divertor

ITER Blanket

Conclusions

Outline

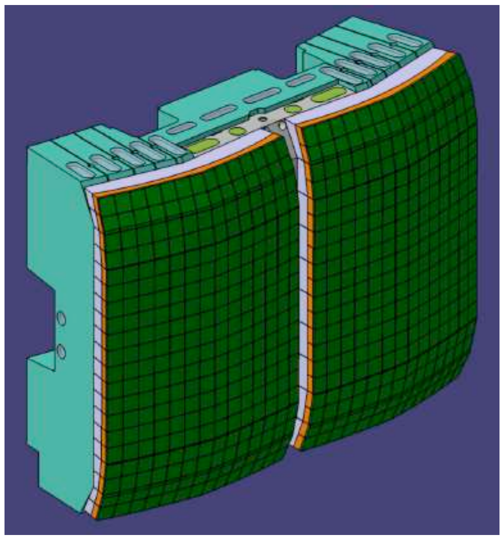
- Nuclear Fusion and the ITER Project
- ITER Internal Components
- ITER Divertor
- **ITER Blanket**
- Conclusions

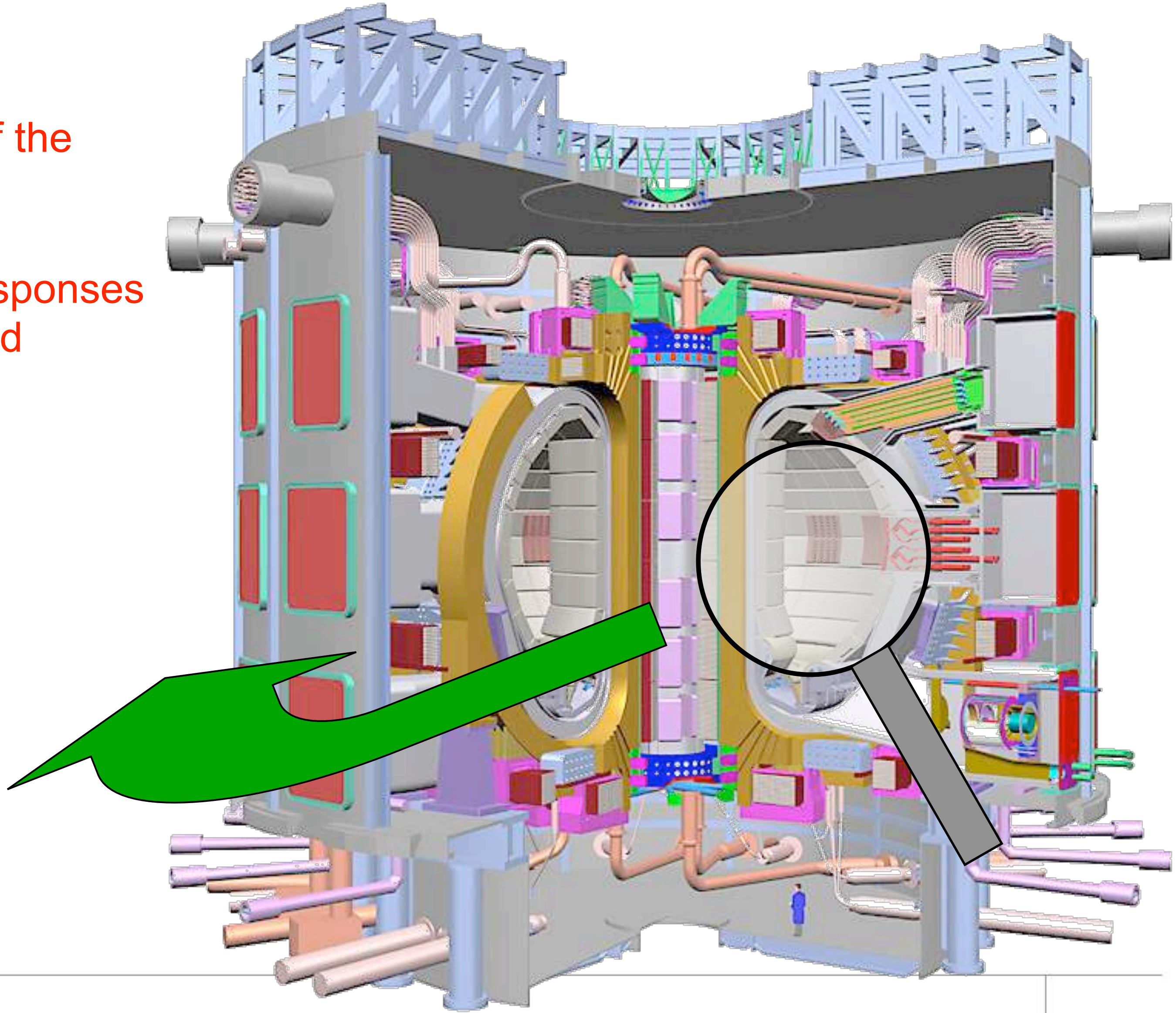
Blanket System

Blanket system main functions :

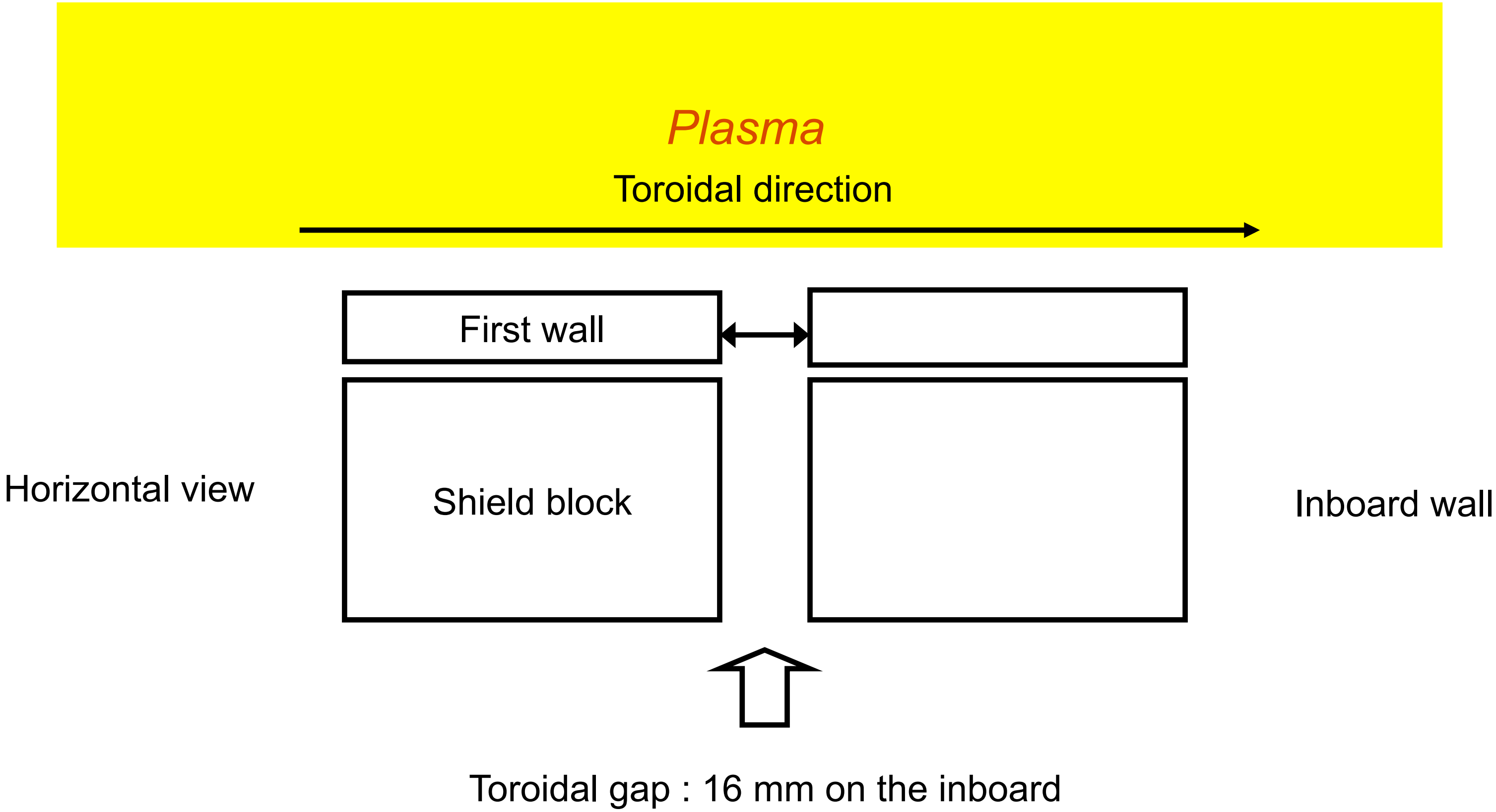
• Exhaust the majority of the plasma power

• Reduce the nuclear responses in the vacuum vessel and superconducting coils

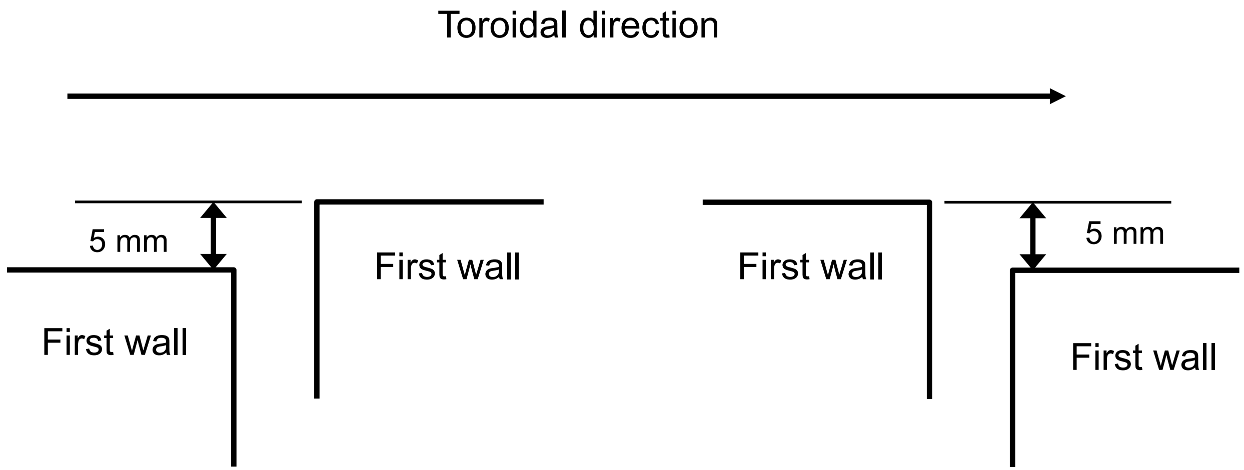




Why shaping is needed ?



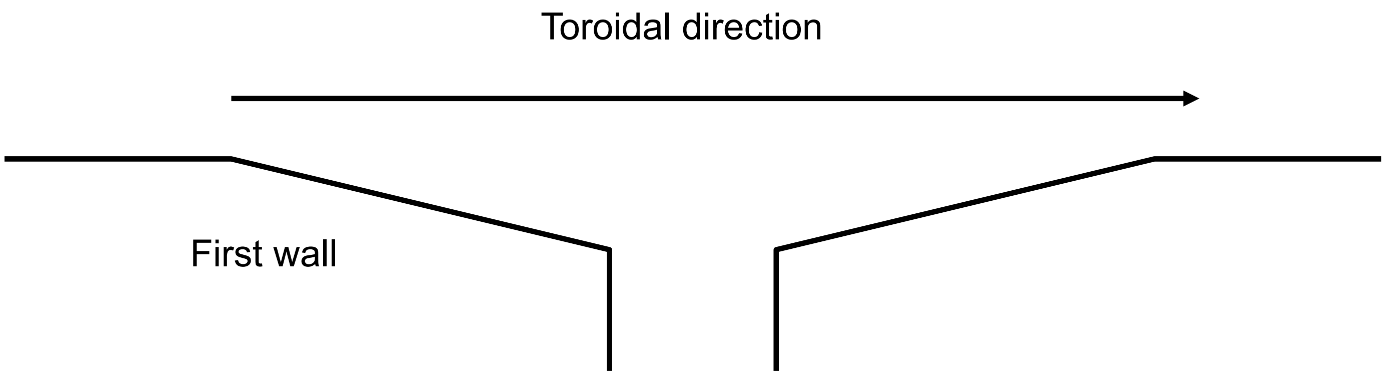
But the two sides have to be considered



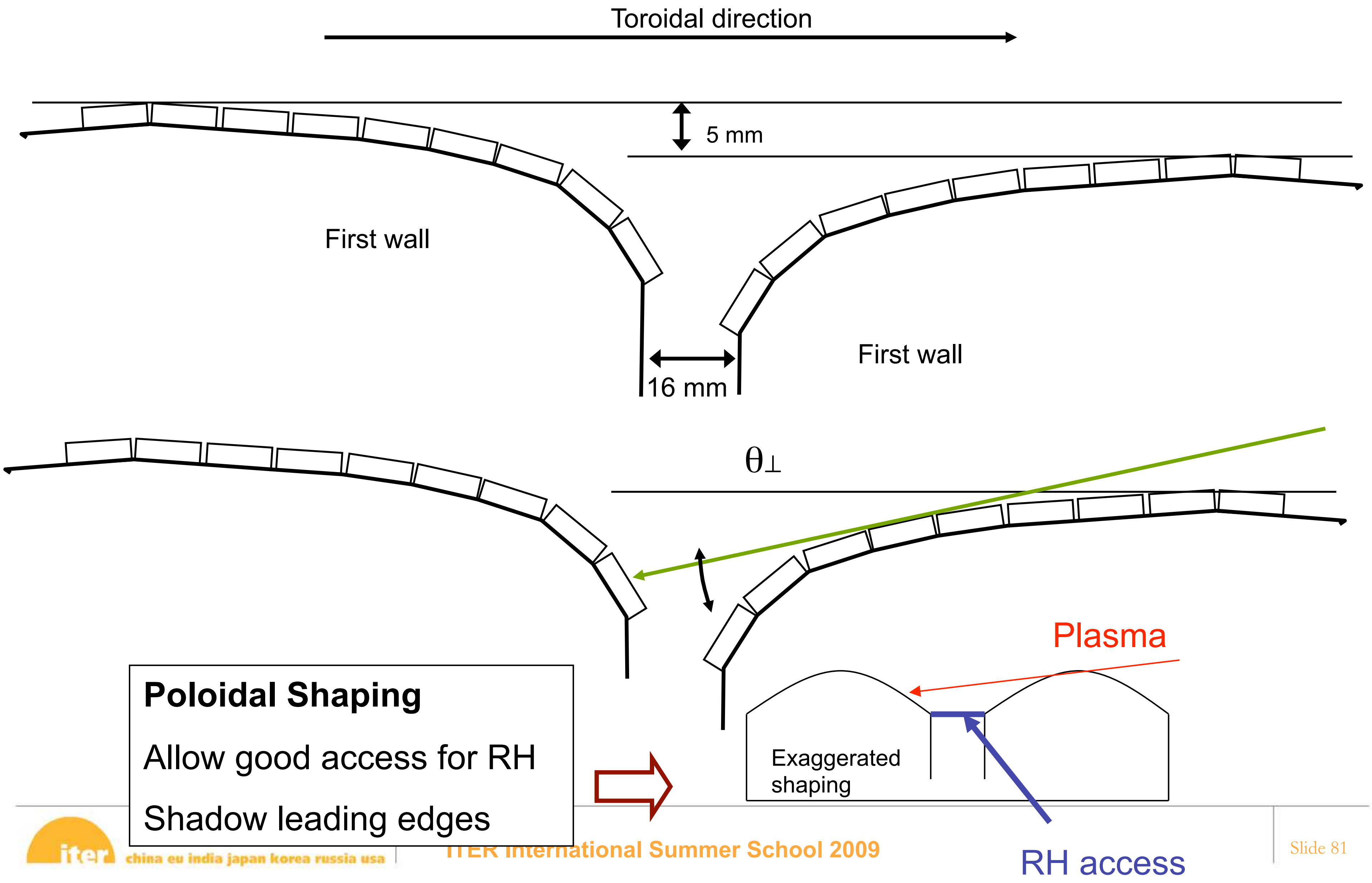
The two situations are equally <u>probable</u>

⇒

So chamfering on both sides is necessary



Definition of a key parameter : θ⊥

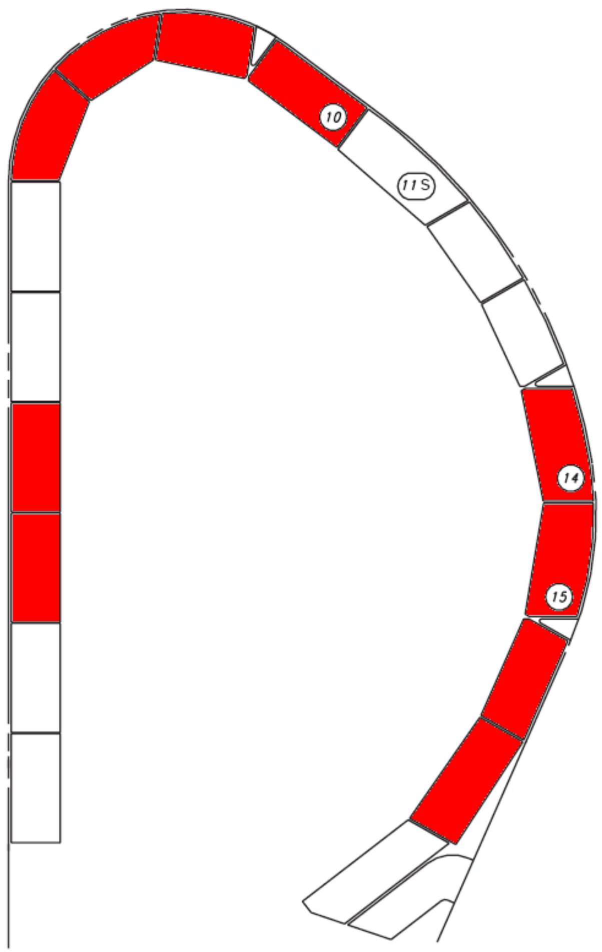


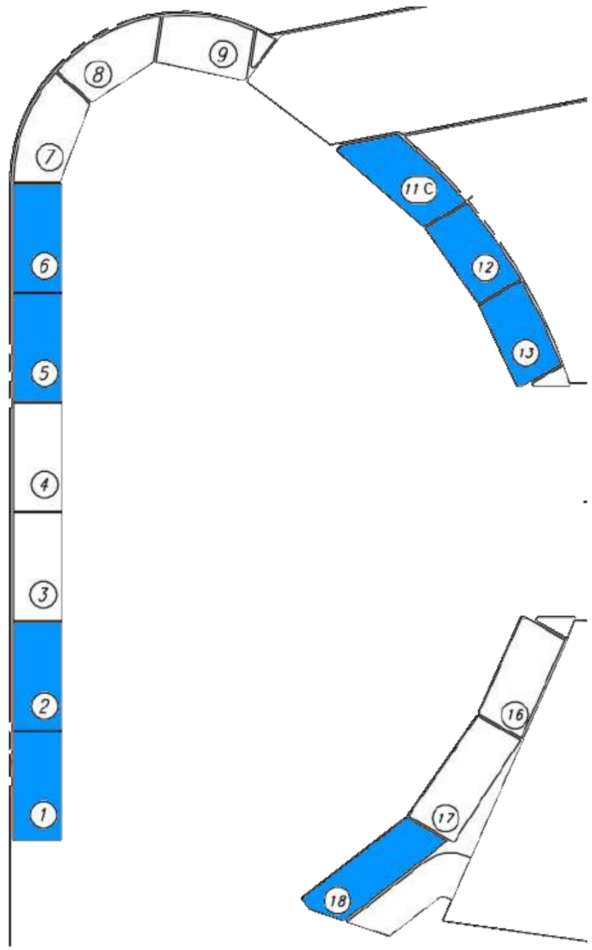
Poloidal Shaping

Allow good access for RH

Shadow leading edges

First Wall Panels: Design heat Flux



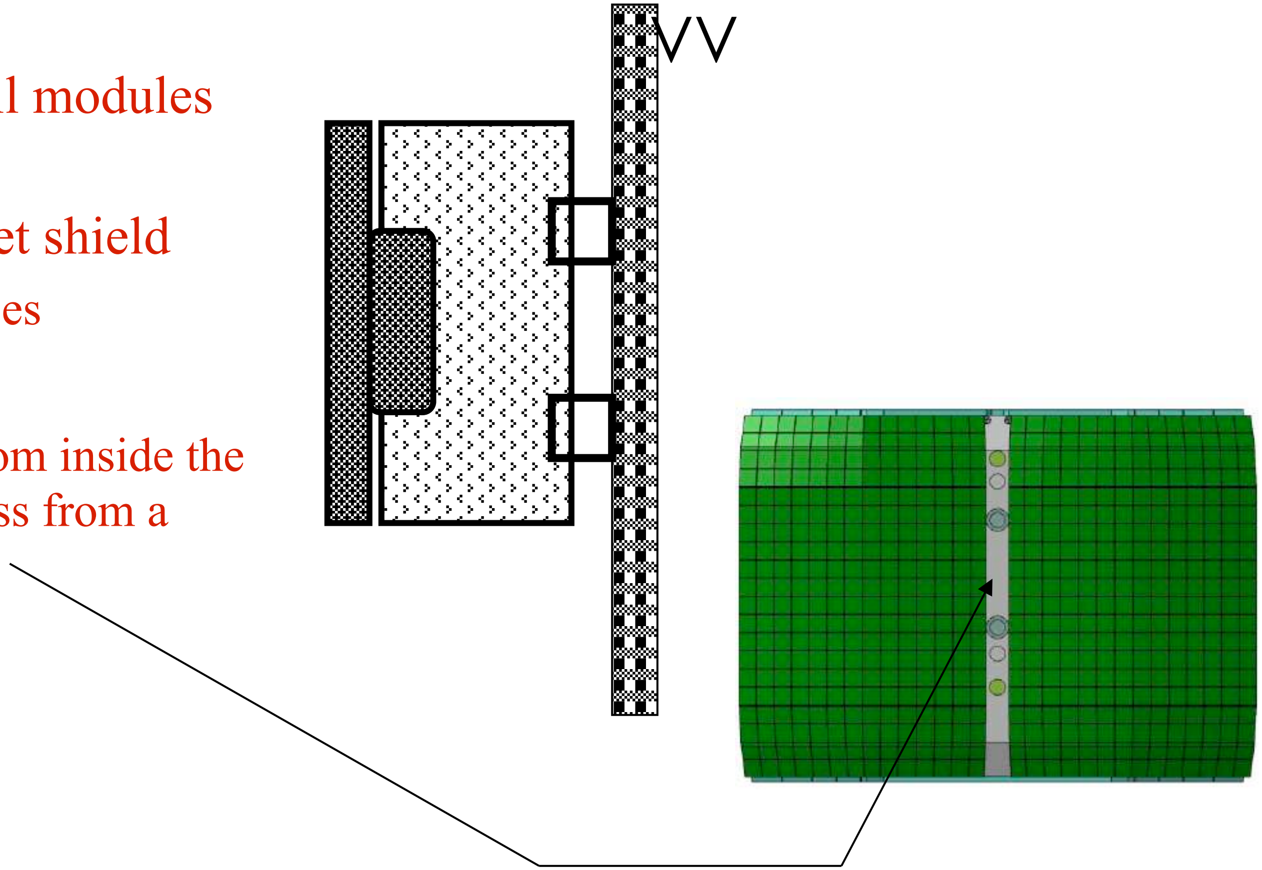


5 MW/m²

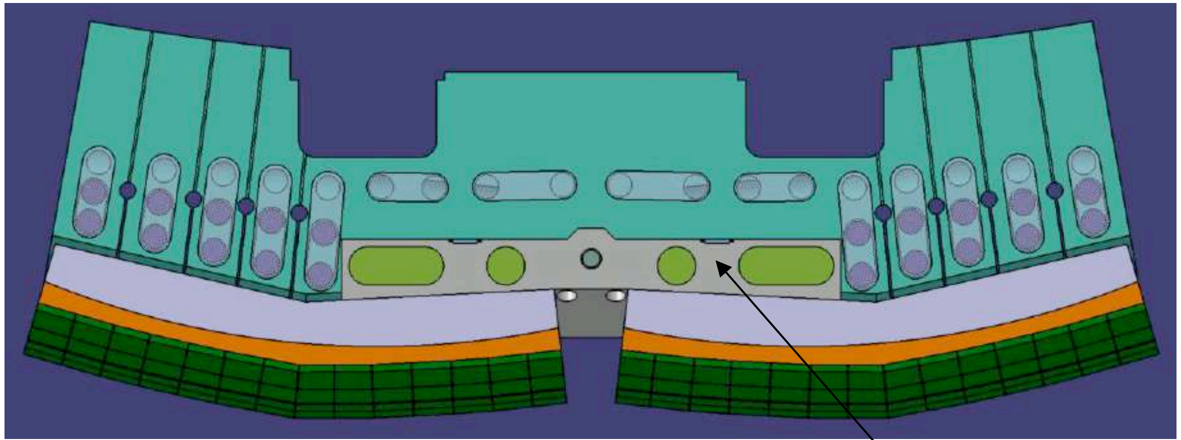
1 MW/m²

Conceptual design

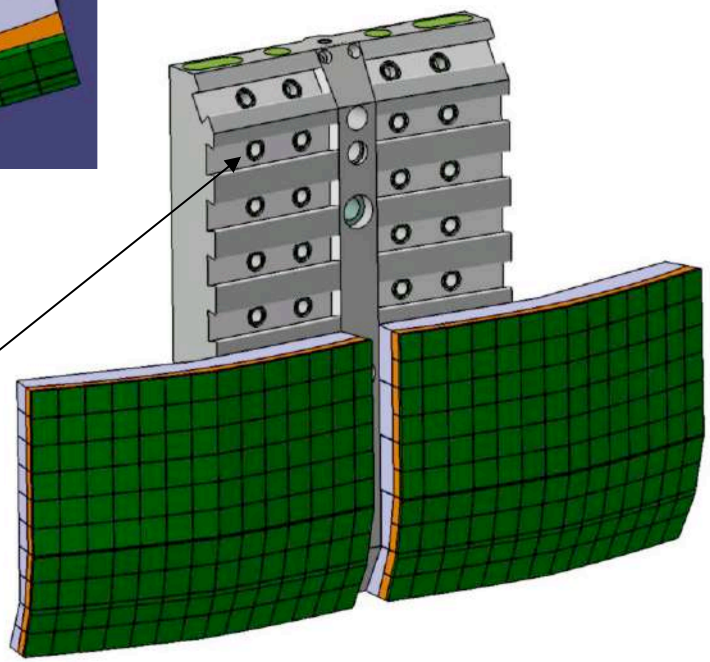
- Identical concept for all modules (inboard & outboard)
- Semi permanent blanket shield
  - Identical VV interfaces
- Separable first wall
  - Remote hand able from inside the vacuum vessel-Access from a central slot



First wall construction

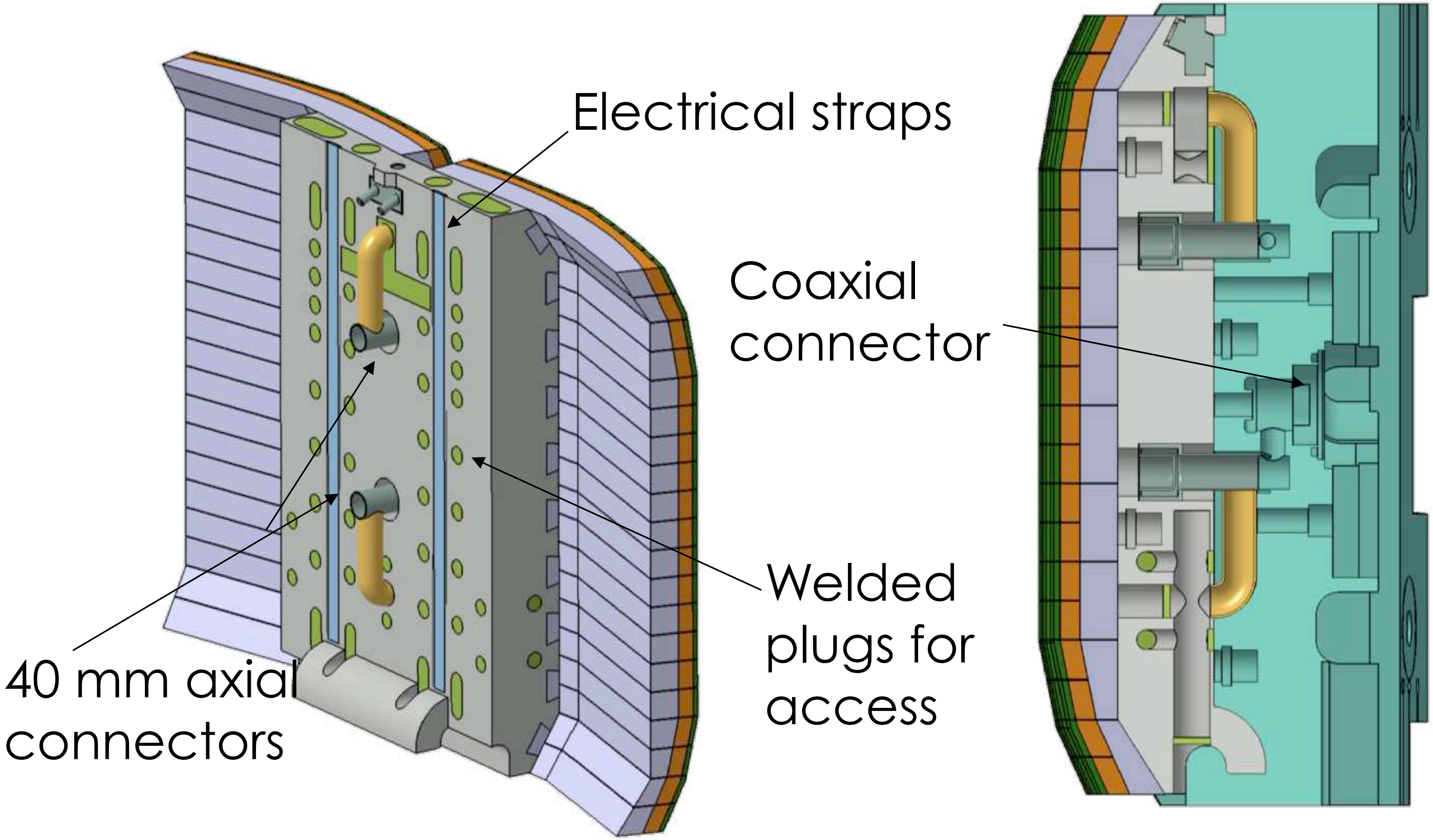


Increased beam width & reduced thickness



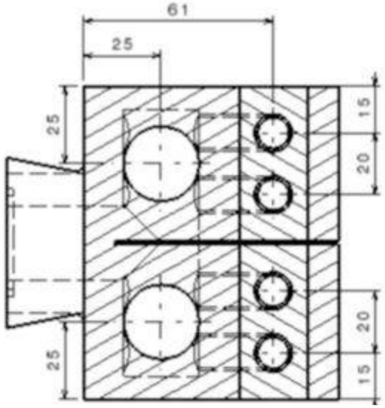
Dove tail Mechanical assembly

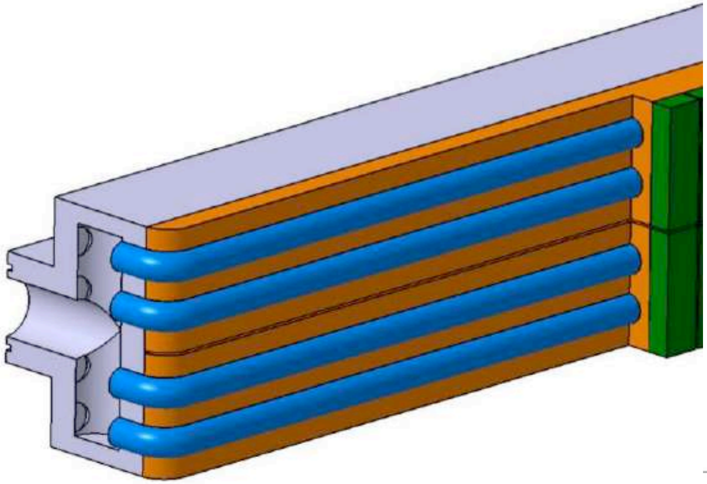
FW beam construction



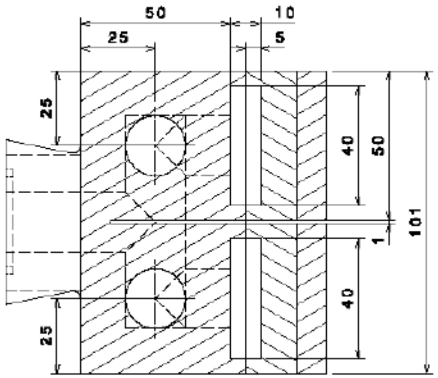
Individual finger construction

Low heat flux 1.0 MW/m²

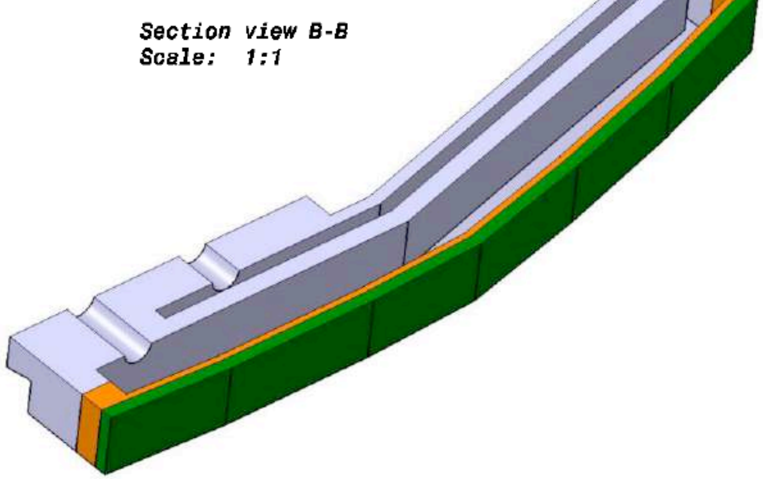




High heat flux 5MW/m²

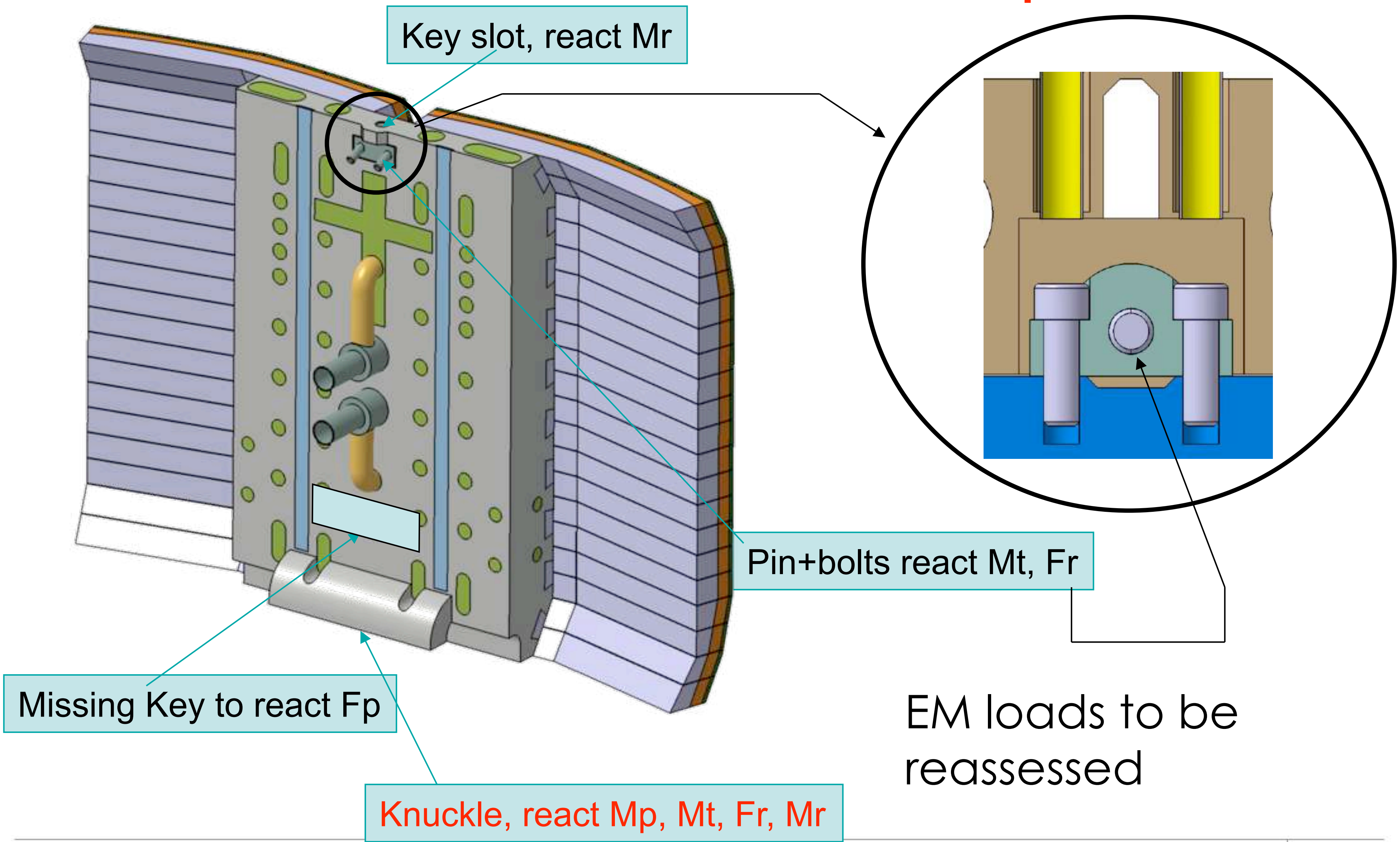


Section view B-B
Scale: 1:1



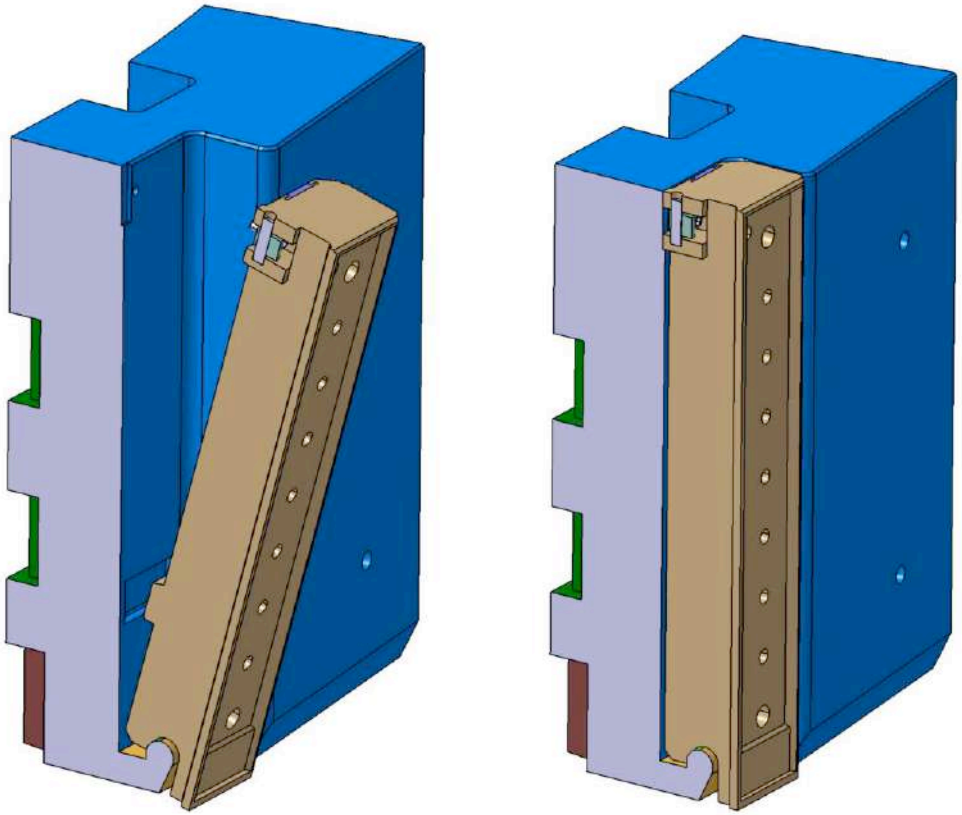
First wall attachment

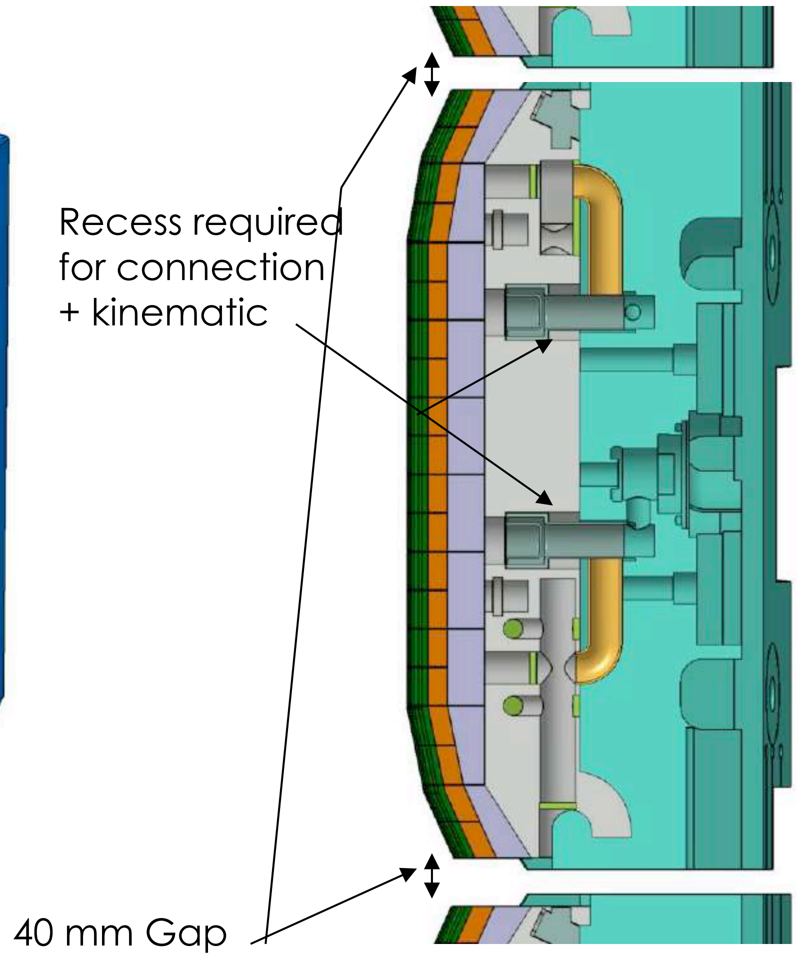
Knuckle concept



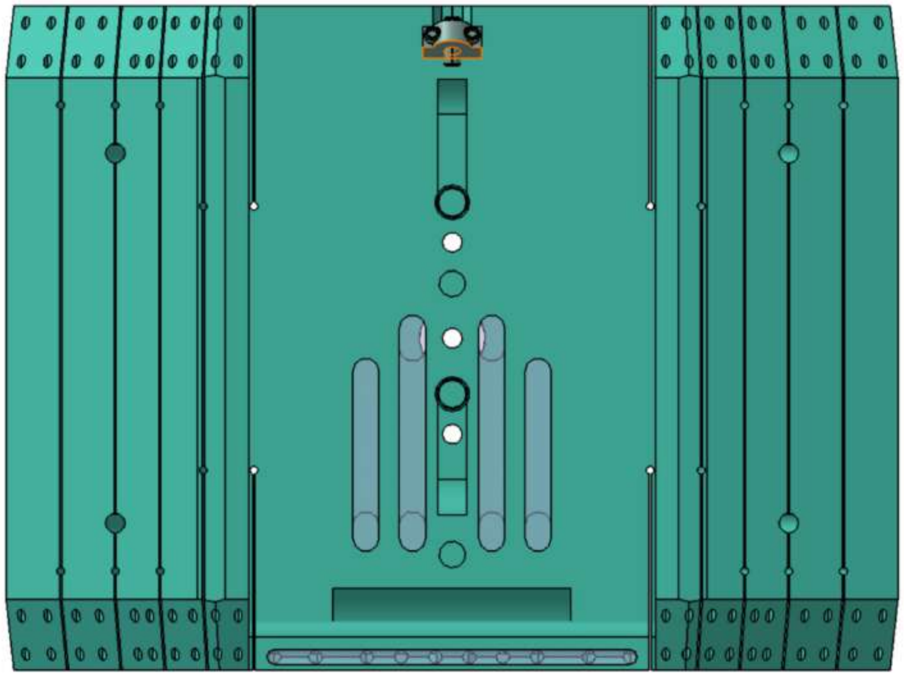
EM loads to be reassessed

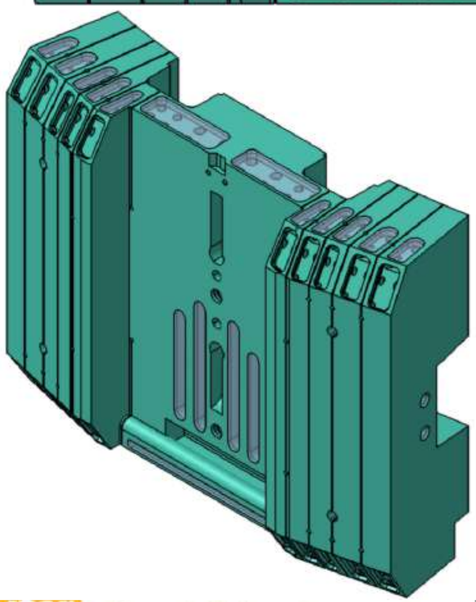
Knuckle kinematic

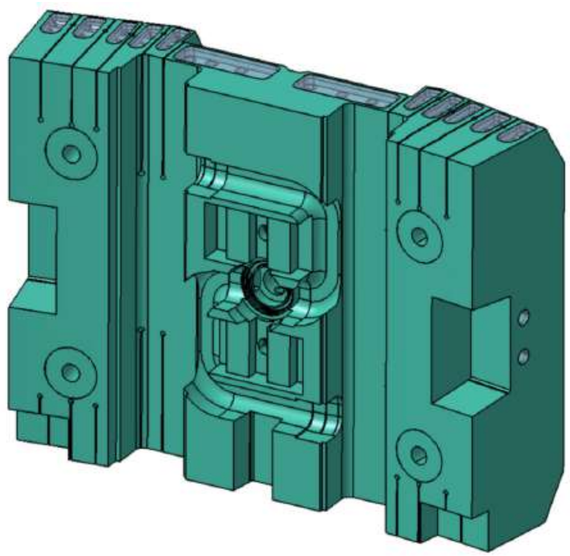




Shield Block Design







2 deep slits

D12 mm cooling holes

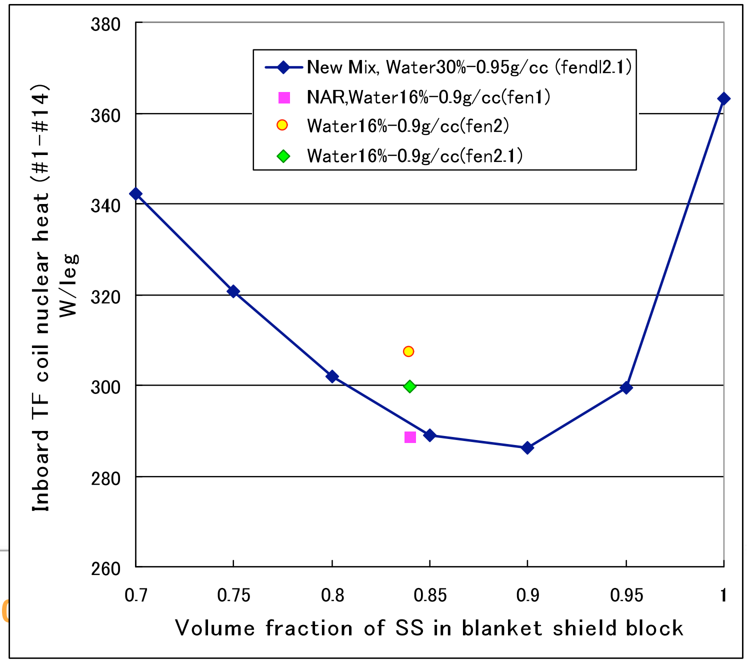
1m for one side drilling

Minimum wall thickness:

6mm (at access hole)

Covers: 10mm in thickness

Welding length: ~15m



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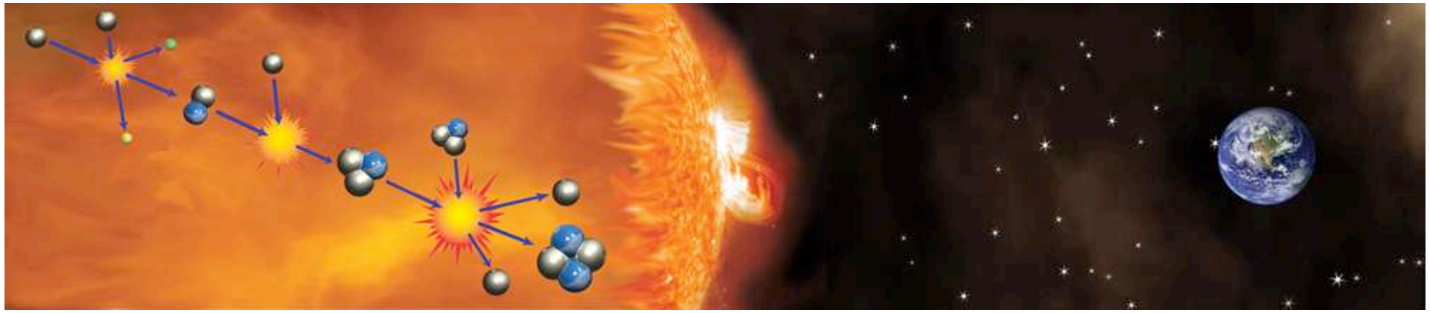
Conclusions

- The "fitness for purpose" of the armour to heat sink joining technologies was demonstrated on the basis of a "design by experiment" approach
- The available technologies are able to meet or even exceed the ITER design requirements
- The manufacturing of plasma facing components requires the mastering of a number of technologies in addition to the armour to heat sink joints
- An extensive R&D effort has been carried out world-wide to develop suitable engineering solutions for the PFCs
- The ITER Divertor design and R&D has reached a stage of maturity to allow the start of procurement

- A First Wall shape is being developed which both shadows leading edges, and provides for a generous RH access aperture
- High heat flux technology is required in some regions, but removes the need for start-up limiters

ITER The Way to the Future…



Tore Supra

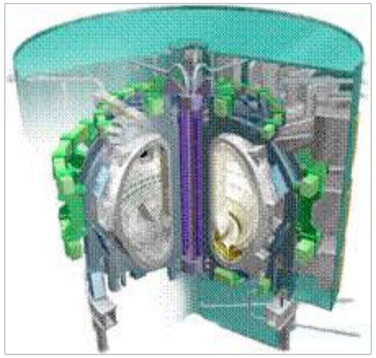
$25\ m^3$

$\sim 0\ MW_{th}$

JET

$80\ m^3$

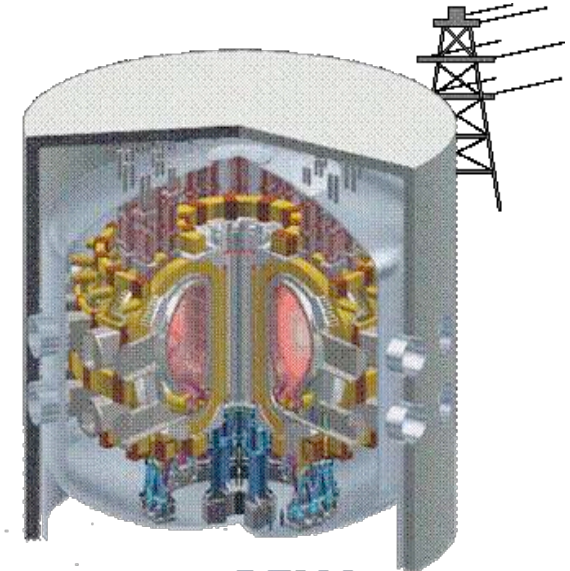
$\sim 16\ MW_{th}$



ITER

$800\ m^3$

$\sim 500\ MW_{th}$



DEMO

$\sim 1000 - 3500\ m^3$

$\sim 2000 - 4000\ MW_{th}$

- Dominant self heating ---------------->