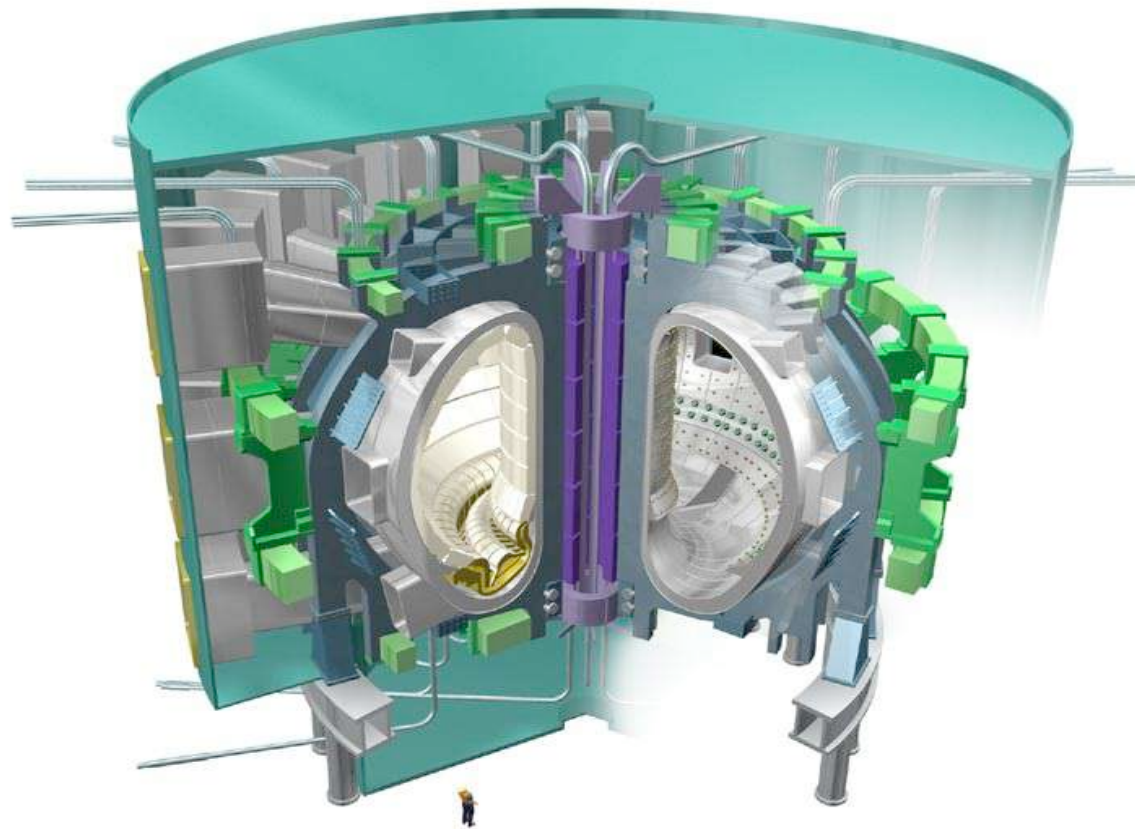


Design of ITER Plasma Facing Components



Mario Merola - ITER International Organization
Internal Components Division Head

Outline

Nuclear Fusion and the ITER Project

ITER Internal Components

ITER Divertor

ITER Blanket

Conclusions

Nuclear Fusion and the ITER Project

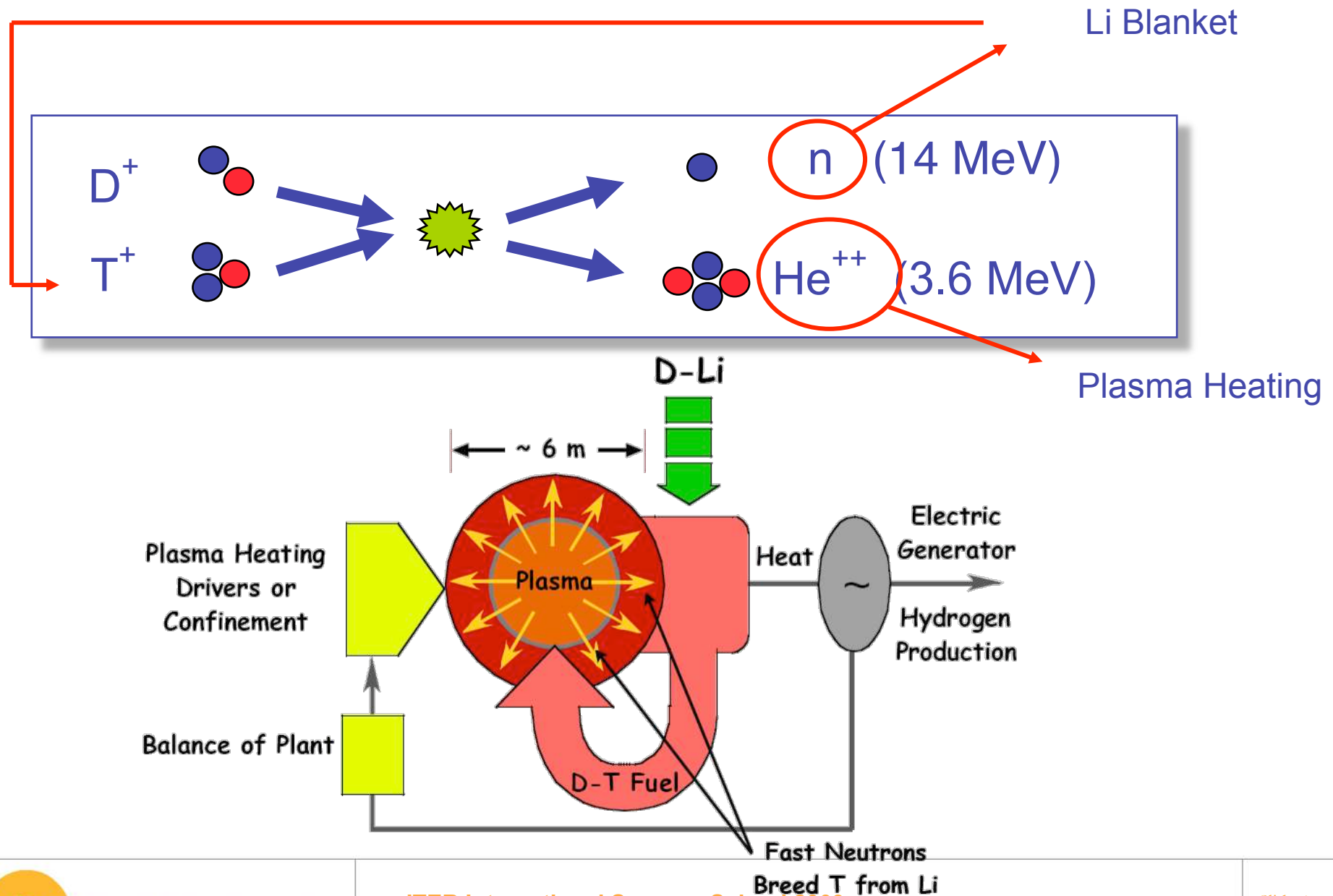
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Conclusions

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

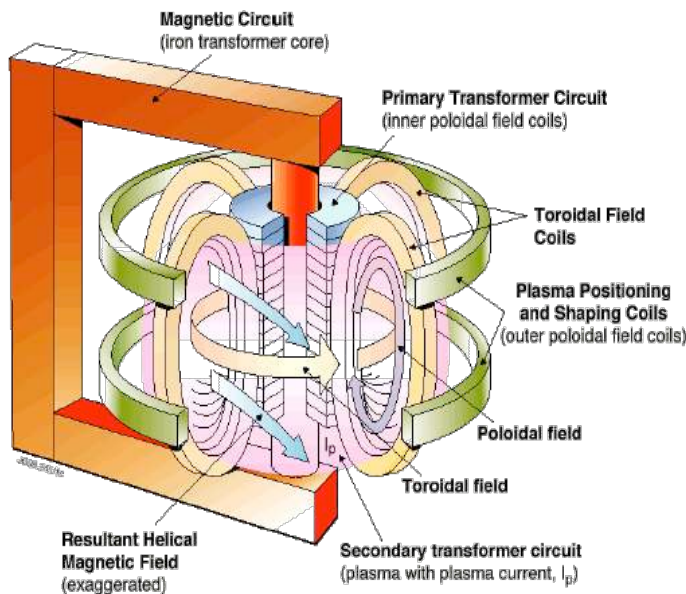
1958 - World-wide declassification of magnetically confined fusion research at Geneva on Peaceful Uses of Atomic Energy

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 tokamak 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”.



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

Special to The New York Times
GENEVA, Nov. 21 — Following is the text of the joint Soviet-American statement at the end of the summit meeting today, as made public by the White House:

By mutual agreement, the President of the United States, Ronald Reagan, and the General Secretary of the Central Committee of the Communist Party of the Soviet Union, Mikhail S. Gorbachev, met in Geneva Nov. 18-21. Attending the meeting on the U.S. side were Secretary of State George P. Shultz; chief of staff, Donald T. Regan; Assistant to the President, Robert C. McFarlane; Ambassador to the U.S.S.R., Paul H. Nitze; special advisor and the Secretary of State, Rosamund L. Ridgway; and the President's Special Representative for Arms Control, Paul H. Nitze. Attending on the Soviet side were the Foreign Minister, Gennadiy A. Yevseyev; the Deputy Foreign Minister, Andrei I. Kostikov; the Deputy Prime Minister, Viktor I. Stetsko; and the Deputy Prime Minister, Viktor I. Stetsko. These comprehensive talks covered the basic Soviet relations and national situation. Frank and useful discussions took place on a number of issues. While acknowledging the differences in their views on some issues, the two sides agreed to improve U.S.-Soviet relations and to work for the benefit of mankind.

In this connection the two sides have confirmed the importance of an ongoing dialogue, reflecting their strong desire to seek common ground on existing problems. They agreed to meet again in the nearest future. The General Secretary accepted an invitation by the President of the United States to visit the United States of America, and the President of the United States accepted an invitation by the General Secretary of the Central Committee of the C.P.S.U. to visit the Soviet Union. Arrangements for the timing of the visits will be agreed upon through diplomatic channels. In their meetings, agreement was reached on a number of specific

ple of 50 percent reductions in the nuclear arms of the U.S. and the U.S.S.R., appropriately applied, as well as the idea of an interim I.N.F. agreement. During the negotiation of these agreements, effective measures for verification of compliance with obligations assumed will be agreed upon.

Risk Reduction Centers
The sides agreed to study the question at the expert level of centers to reduce nuclear risk taking into account the issues and developments in the Geneva negotiations. They took satisfaction in such recent steps in this direction as the modernization of

firming that they are in favor of a general and complete prohibition of chemical weapons and the destruction of existing stockpiles of such weapons. They agreed to accelerate efforts to conclude an effective and verifiable international convention on this matter.

The two sides agreed to intensify bilateral discussions on the level of experts on all aspects of such a chemical weapons ban, including the question of verification. They agreed to initiate a dialogue on preventing the proliferation of chemical weapons.

Mutual Basic Force Reduction
They agreed on the importance of resolving humanitarian cases in the spirit of cooperation. They believe that there should be greater understanding among our peoples and that to this end they will encourage greater travel and people-to-people contact.

Northern Pacific Air Safety
The two leaders also noted with satisfaction that, in cooperation with the Government of Japan, the United States and the Soviet Union have agreed to a set of measures to promote safety on air routes in the North Pacific and have worked out steps to implement them.

Civil Aviation Consulates
They acknowledged that delegations from the United States and the Soviet Union have begun negotiations aimed at resumption of air services. The two leaders expressed their desire to reach a mutually beneficial agreement at an early date. In this regard, an agreement was reached on the simultaneous opening of consulates general in New York and Kiev.

Environmental Protection
Both sides agreed to contribute to the preservation of the environment

ministries and departments in such fields as agriculture, housing and protection of the environment have been useful. Recognizing that exchanges of views on regional issues on the expert level have proven useful, they agreed to continue such exchanges on a regular basis.

The sides intend to expand the programs of bilateral cultural, educational and scientific-technical exchanges, and also to develop trade and economic ties. The President of the United States and the General Secretary of the Central Committee of the C.P.S.U. attended the signing of the Agreement on Contacts and Exchanges in Scientific, Educational and Cultural Fields.

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Environmental Protection
Both sides agreed to contribute to the preservation of the environment

— a global task — through joint research and practical measures. In accordance with the existing U.S.-Soviet agreement in this area, consultations will be held next year in Moscow and Washington on specific programs of cooperation.

Exchange Initiatives
The two leaders agreed on the utility of broadening exchanges and contacts including some of their new forums in a number of scientific, educational, medical and sports fields (inter alia, cooperation in the development of educational exchanges and software for elementary and secondary school instruction; measures to promote Russian language studies in the United States and English language studies in the U.S.S.R.; the annual exchange of professors to conduct special courses in history, culture and economics at the relevant departments of Soviet and American institutions of higher education; mutual allocation of scholarships for the best students in the natural sciences, technology, social sciences and humanities for the period of an academic year; holding regular meetings in various sports and increased television coverage of sports events). The two sides agreed to resume cooperation in combating cancer diseases.

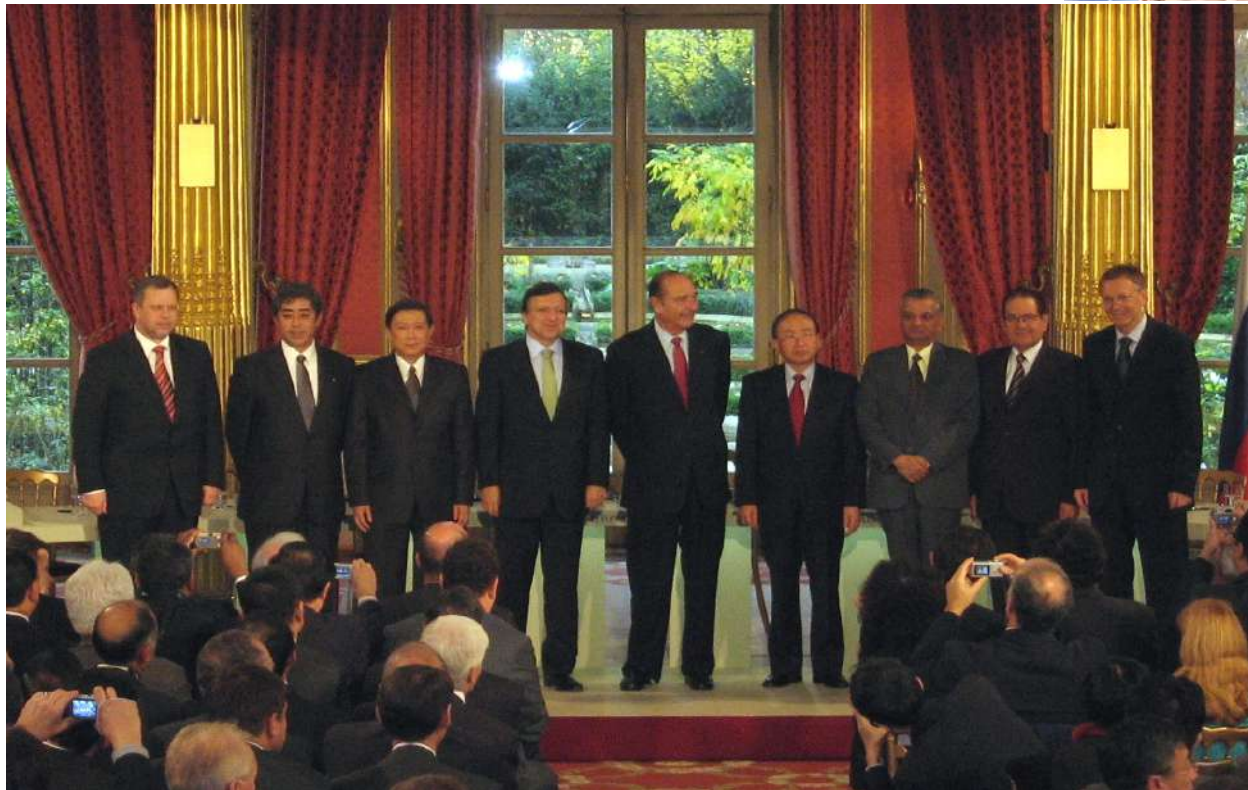
The relevant agencies in each of the countries are being instructed to develop specific programs for these exchanges. The resulting programs will be presented to the leaders at their next meeting.

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 essentially inexhaustible, for the benefit of all mankind.

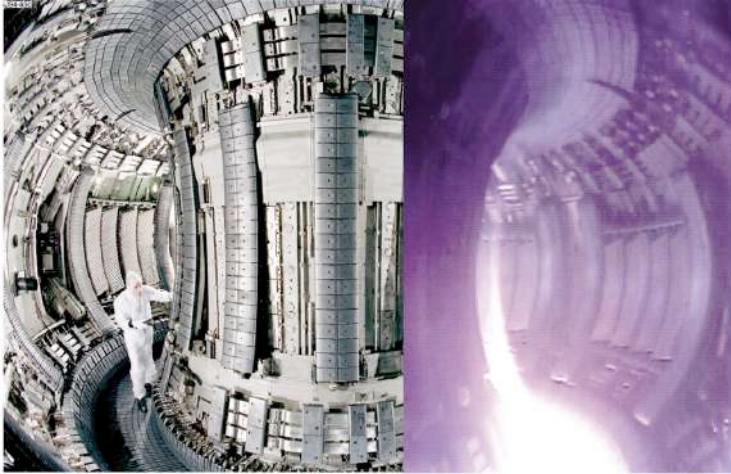
ITER Agreement

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

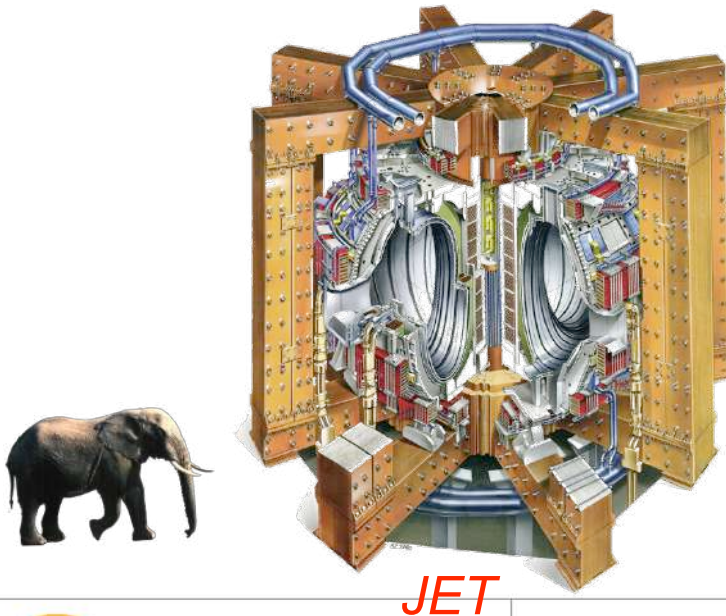
International Organization started.



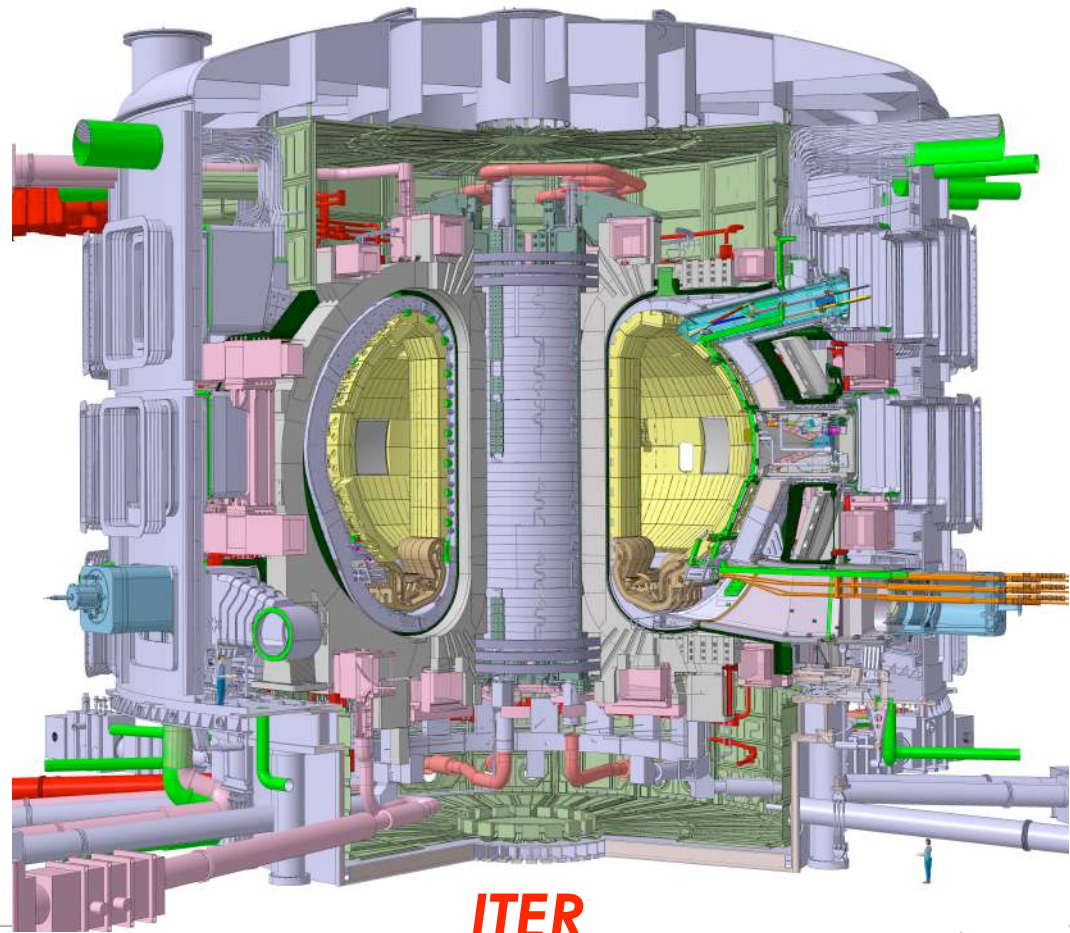
Tokamak's



JET – Internals & Plasma



JET

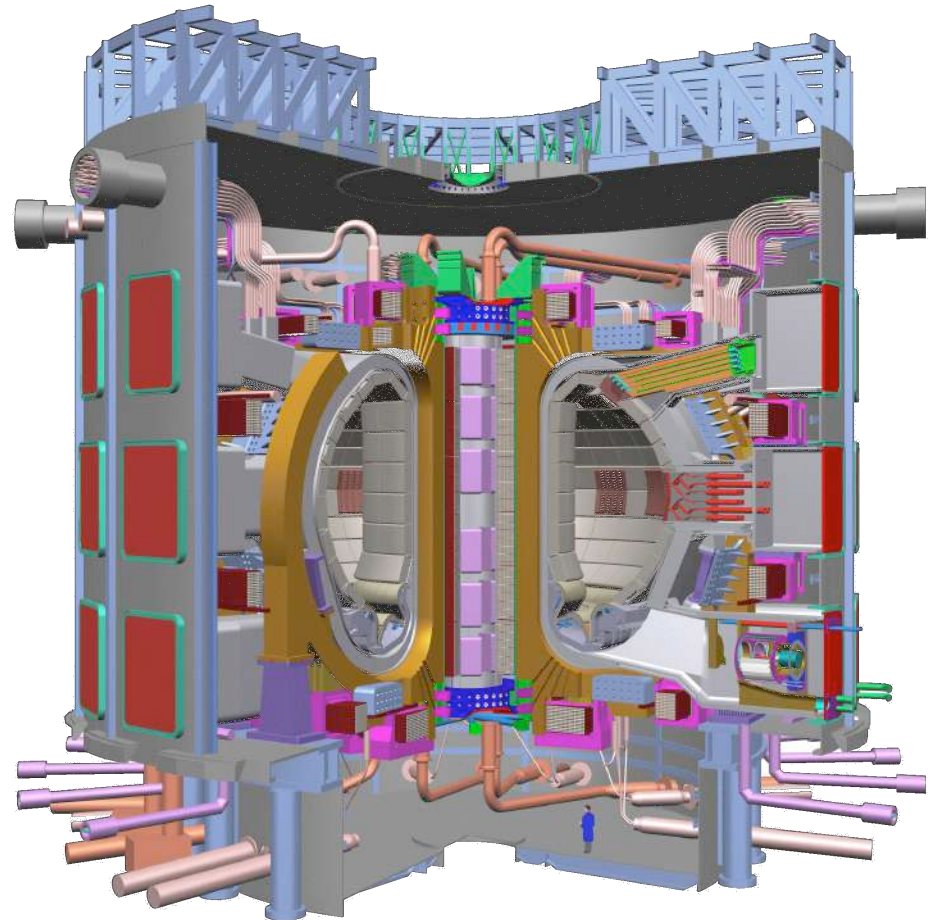


ITER

ITER will allow us to produce plasmas with
temperatures of 100 - 200 million °C
(10 times the temperature of the sun's core)
⇒ 500 Megawatts of fusion power

ITER – The way to fusion power

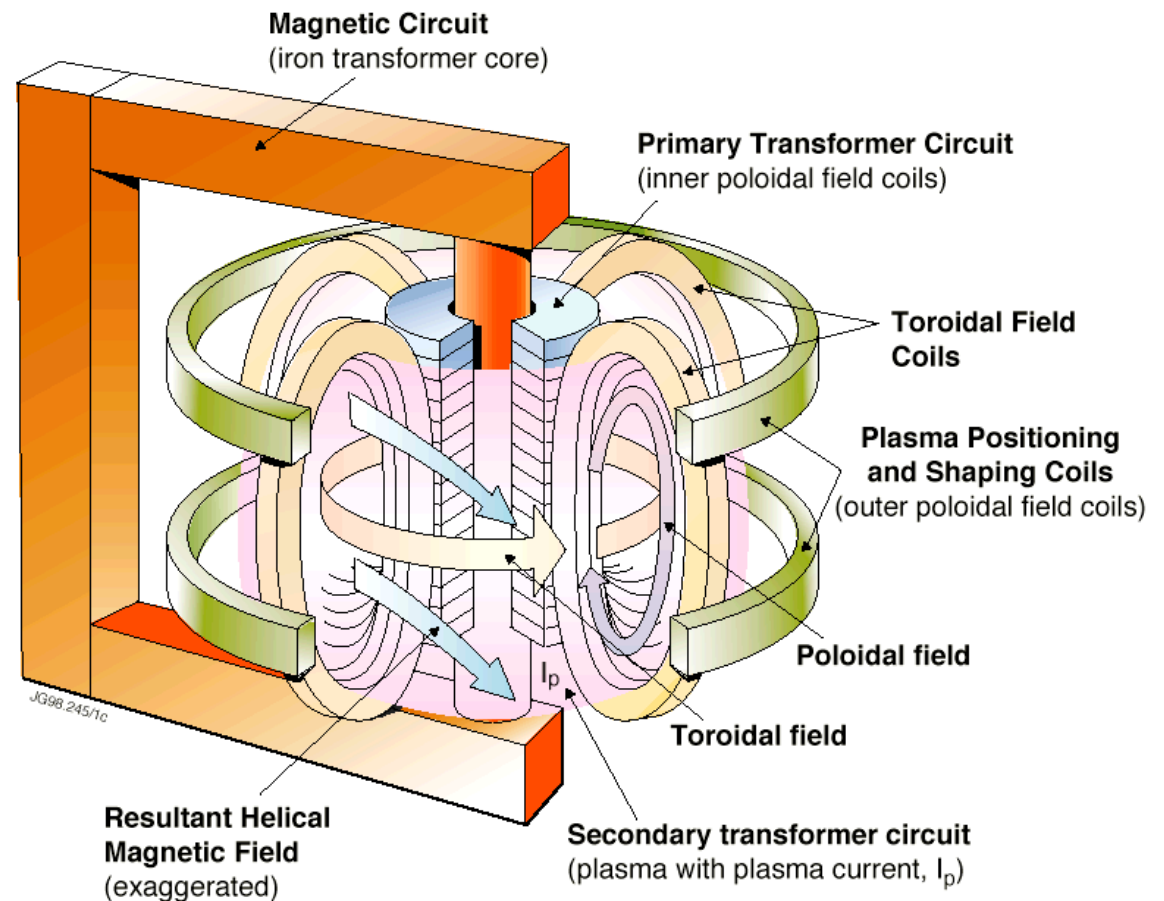
- ITER (“the way”) is the essential next step in the development of fusion
- The world’s biggest fusion energy research project, and one of the most challenging and innovative scientific projects in the world today.
- Its objective:
 - to demonstrate the scientific and technological feasibility of fusion 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.



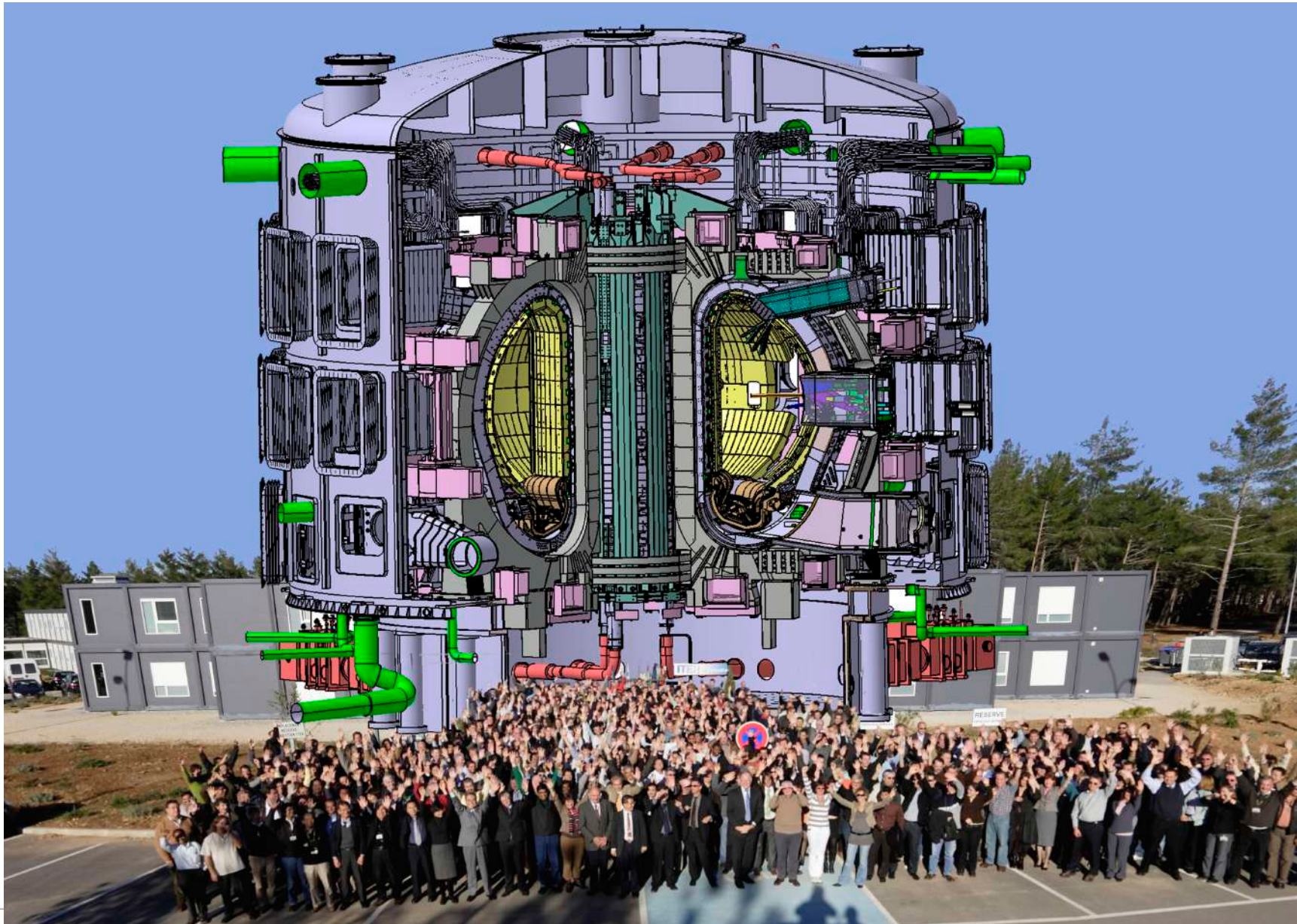
Magnetic Confinement in a Tokamak

The Tokamak:

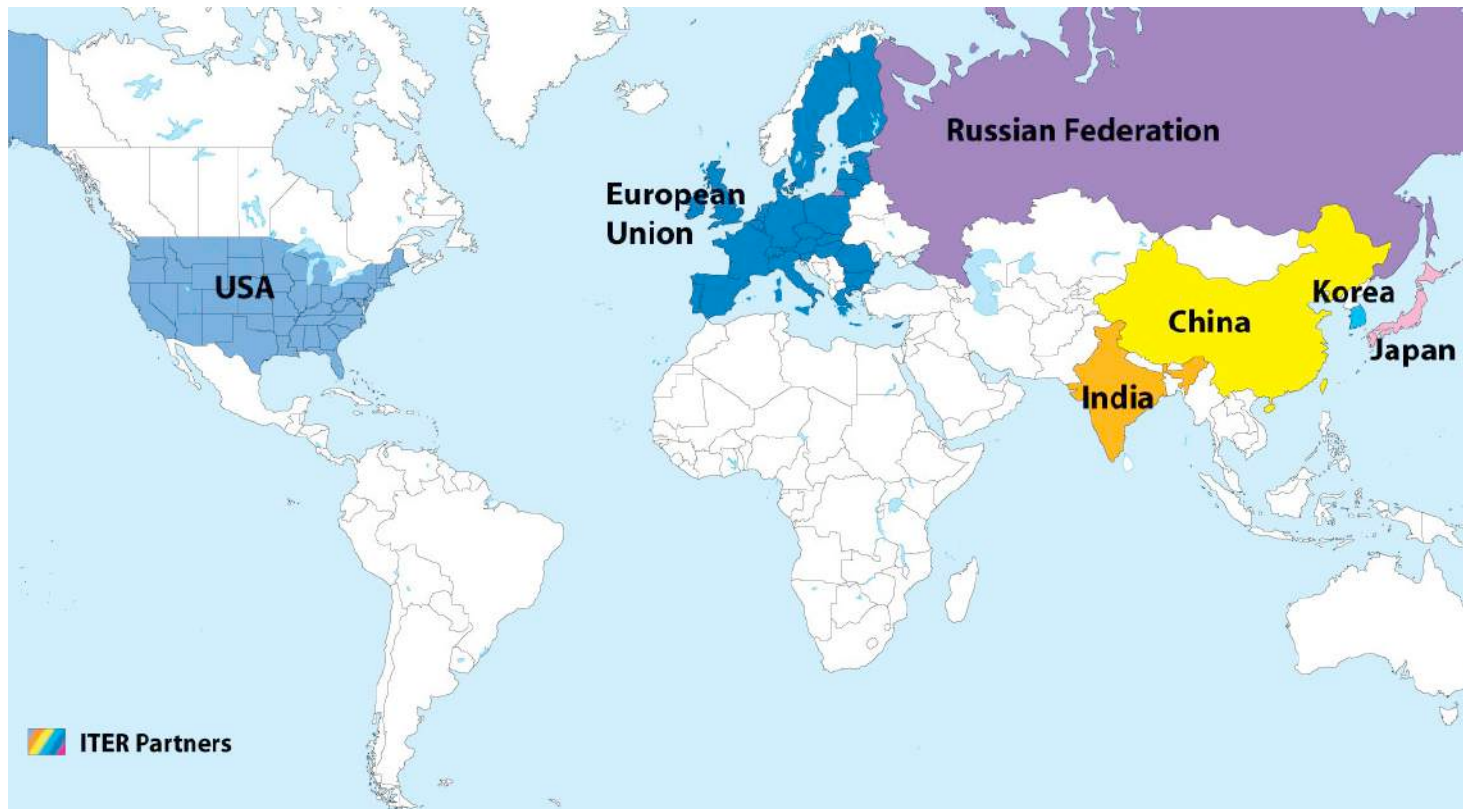
- **toroidal** magnetic field is produced by external magnetic field coils
- plasma current produces **poloidal** magnetic field
- result is a set of nested **helical surfaces**
⇒ **plasma confinement**



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 :

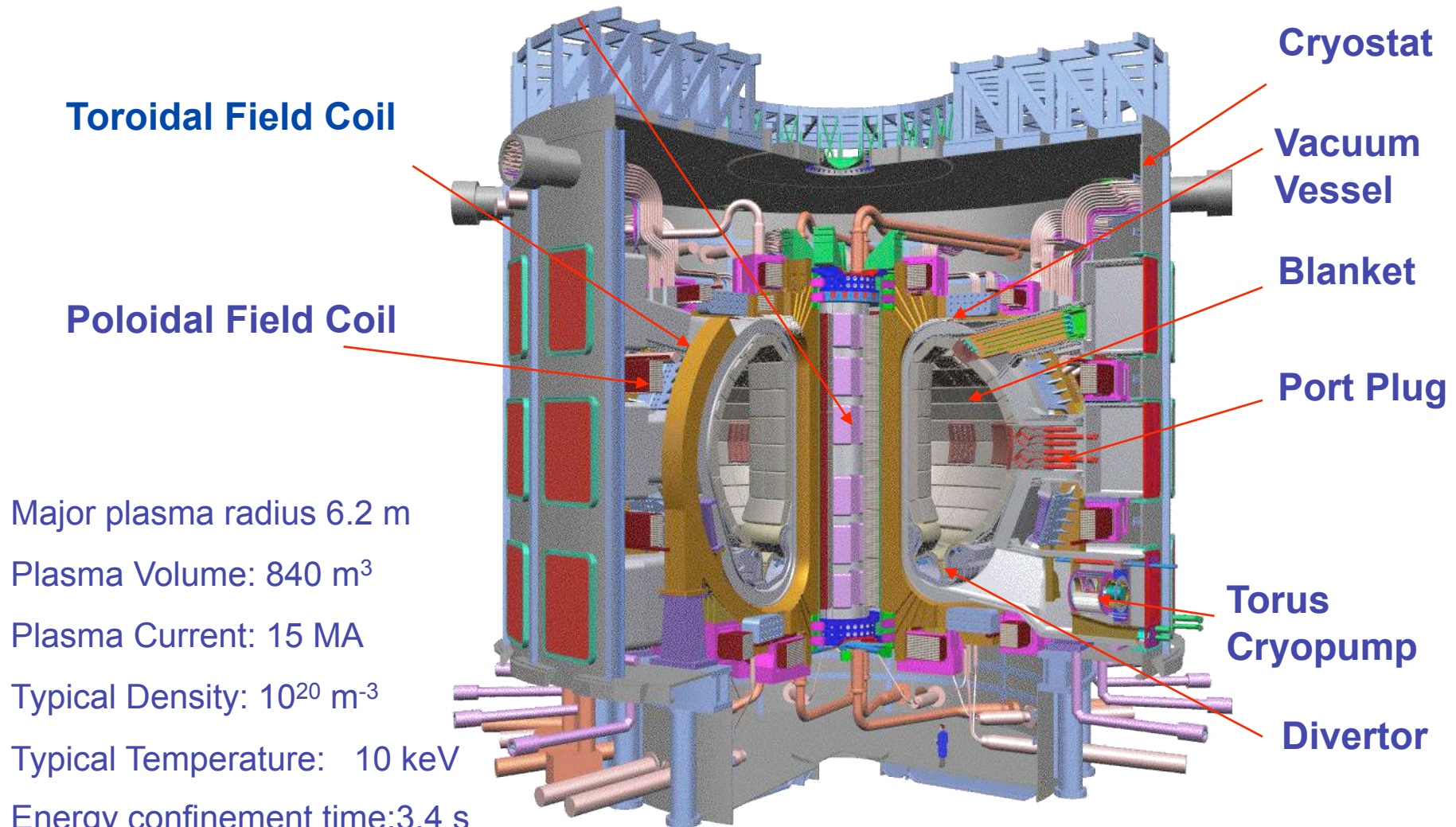
Saint Paul-lès-Durance,
Provence-Alpes-Côte d'Azur,
France.

Site next to **Cadarache** , nuclear research center
of the *Commissariat à l'Énergie Atomique* (CEA).



The Core of ITER

Central Solenoid



Major plasma radius 6.2 m

Plasma Volume: 840 m³

Plasma Current: 15 MA

Typical Density: 10²⁰ 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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ITER Divertor

ITER Blanket

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Nuclear Fusion and the ITER Project

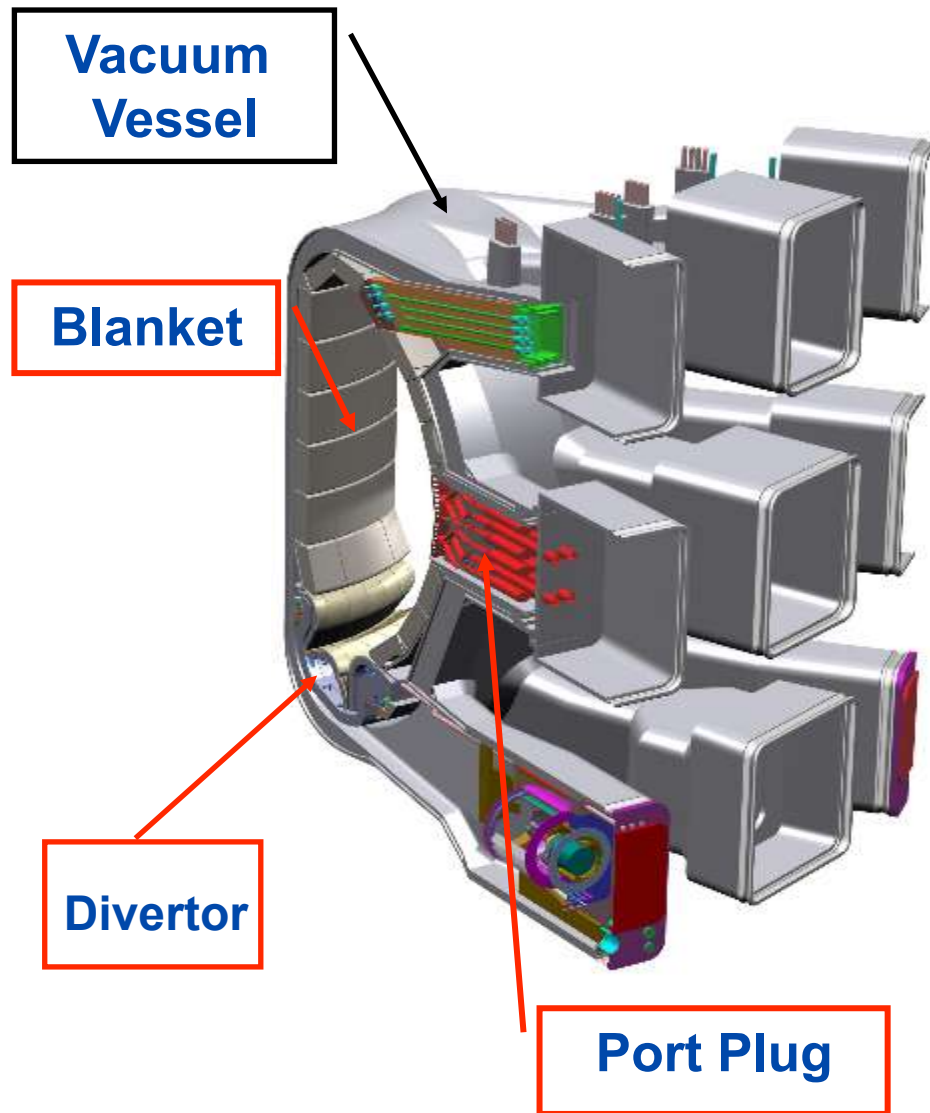
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Vessel/In-Vessel ITER Components



- 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-year/m^2 . The two main effects of the neutron flux are the volumetric heat deposition and the neutron damage.

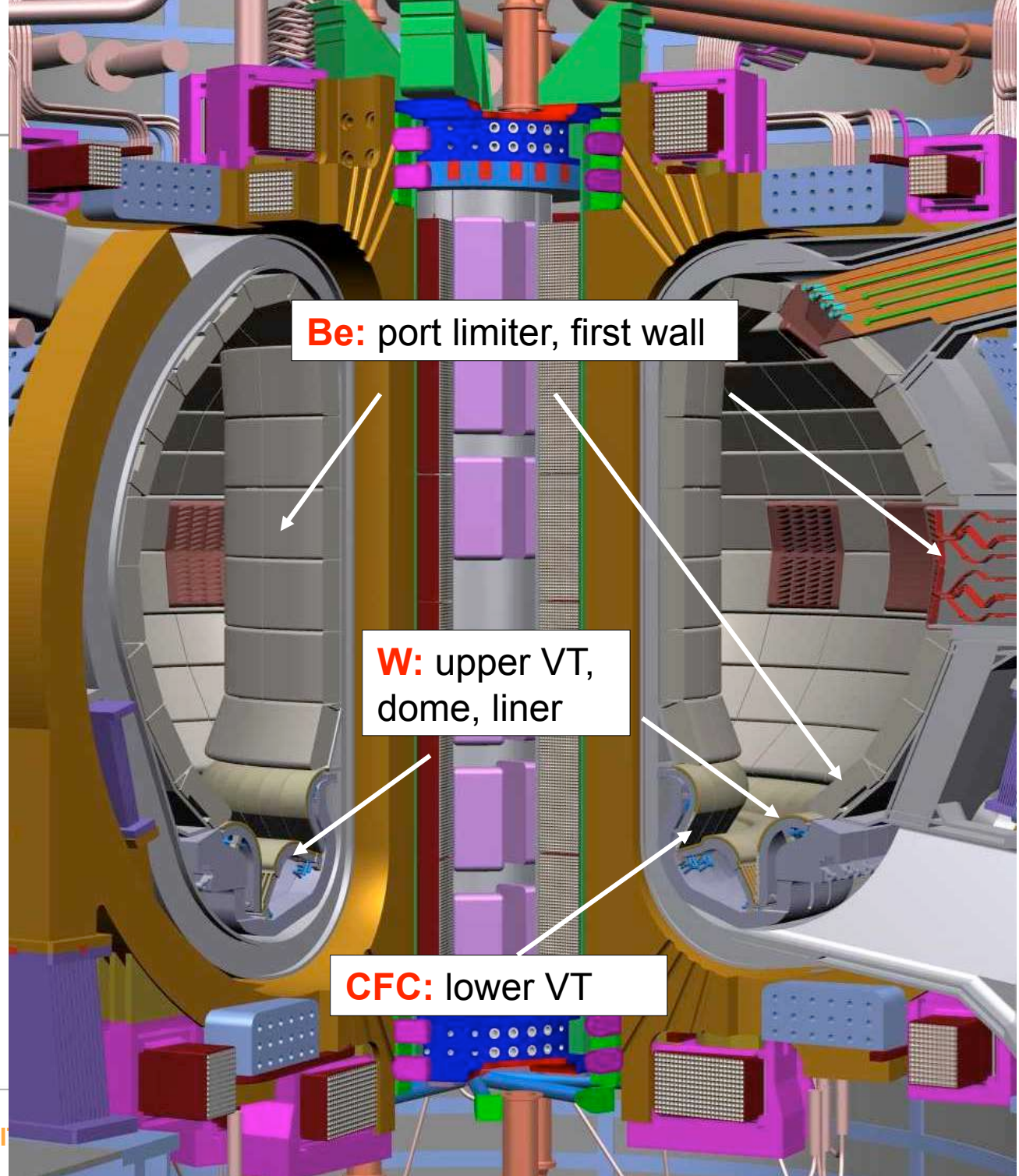
Design inputs for PFCs

- 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

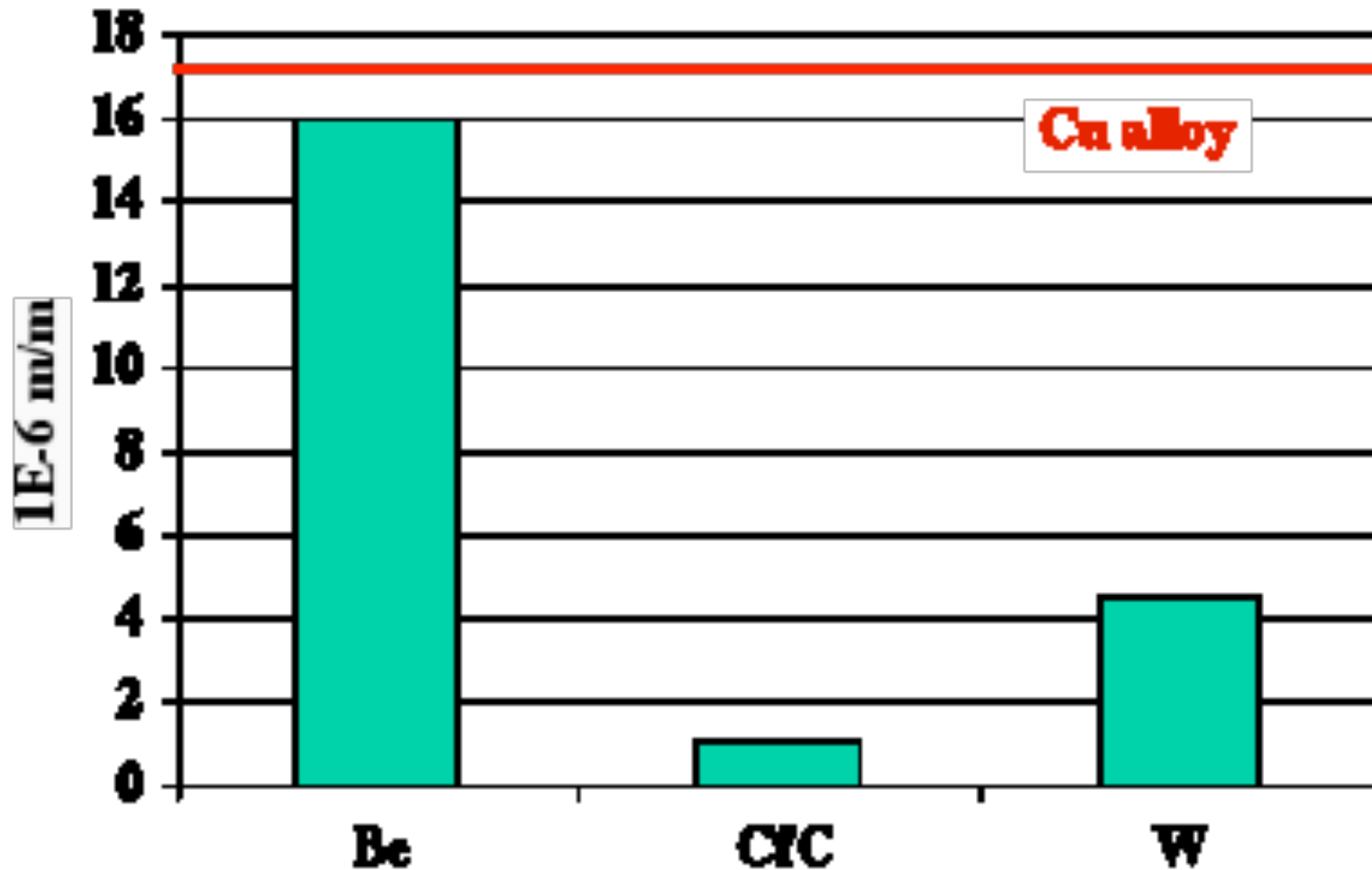
CFC

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

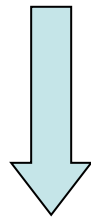
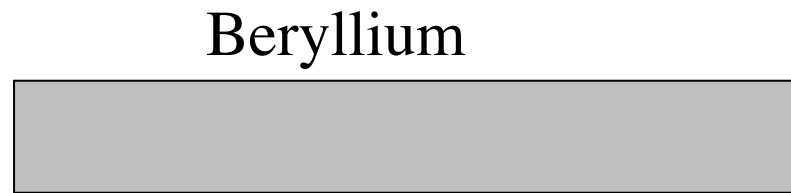
Tungsten

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

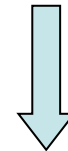
Thermal expansion at 300 °C



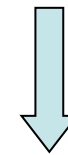
Armour to heat sink joints



CuCrZr



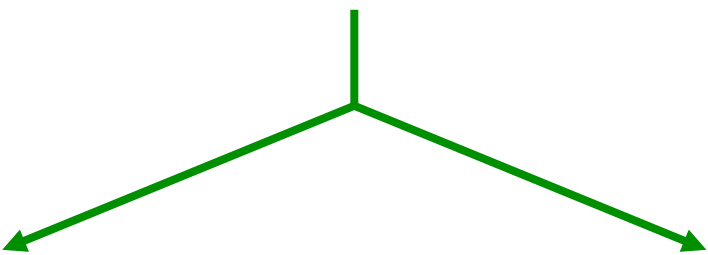
Pure copper interlayer



CuCrZr

Beryllium to copper alloy joint

Formation of brittle intermetallics
(e.g. BeCu or Be₂Cu) above ~600 °C



Use of diffusion barriers
as Ti (Al)

HIP'ing at 500-850 °C

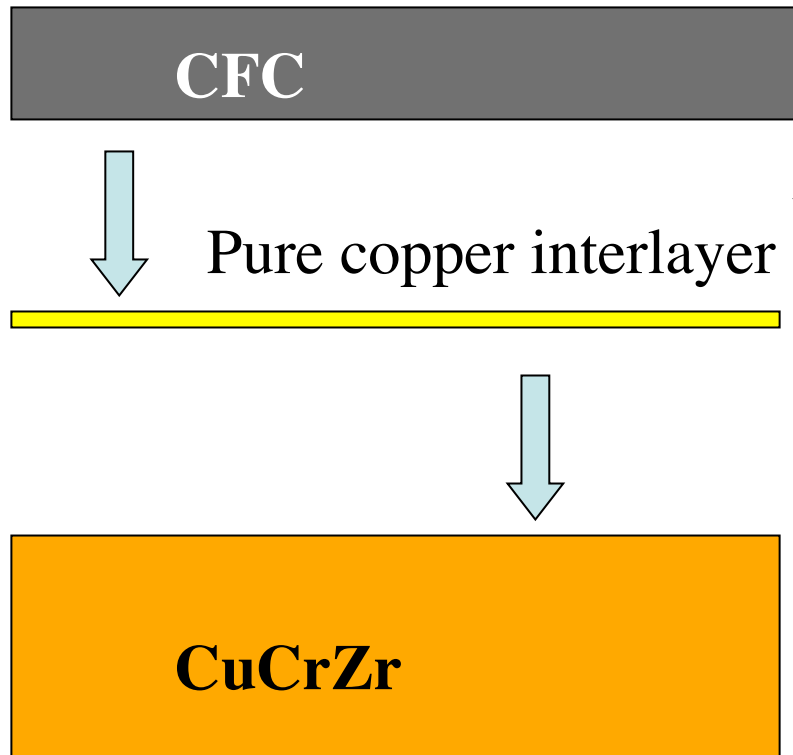
Proper selection of technology

Fast brazing

Cu electroplating of Be + HIP

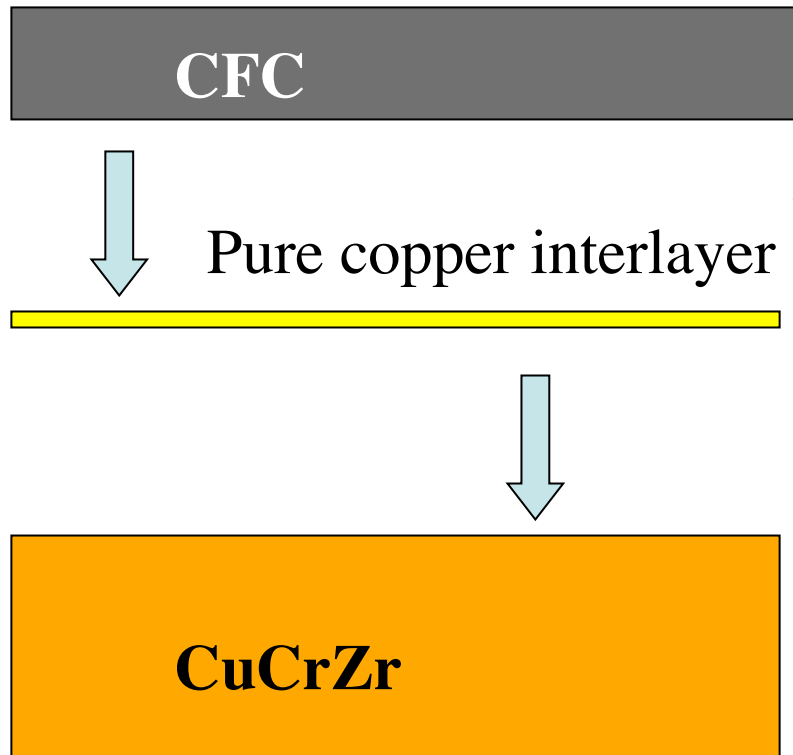
Be plasma spray

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

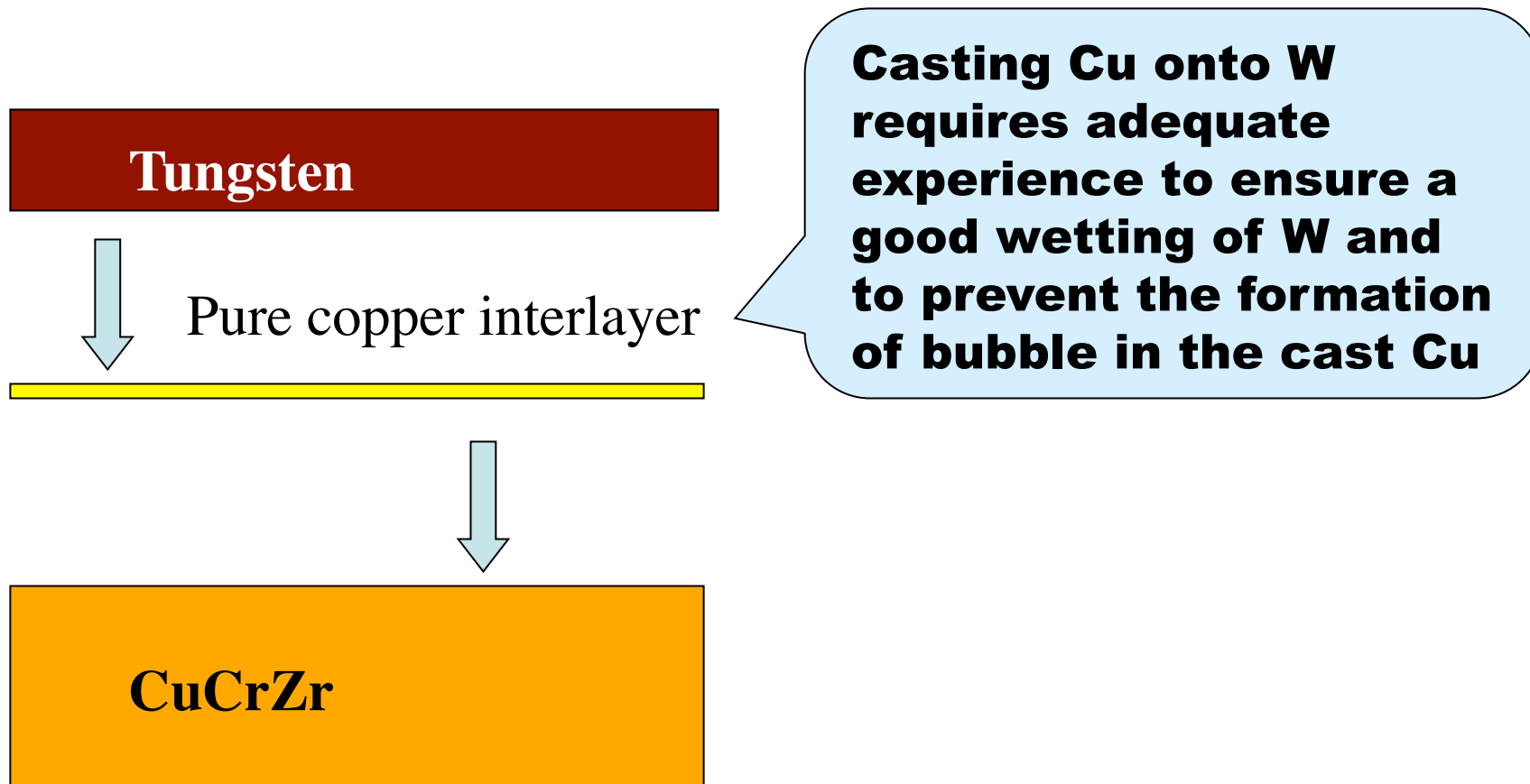


AMC

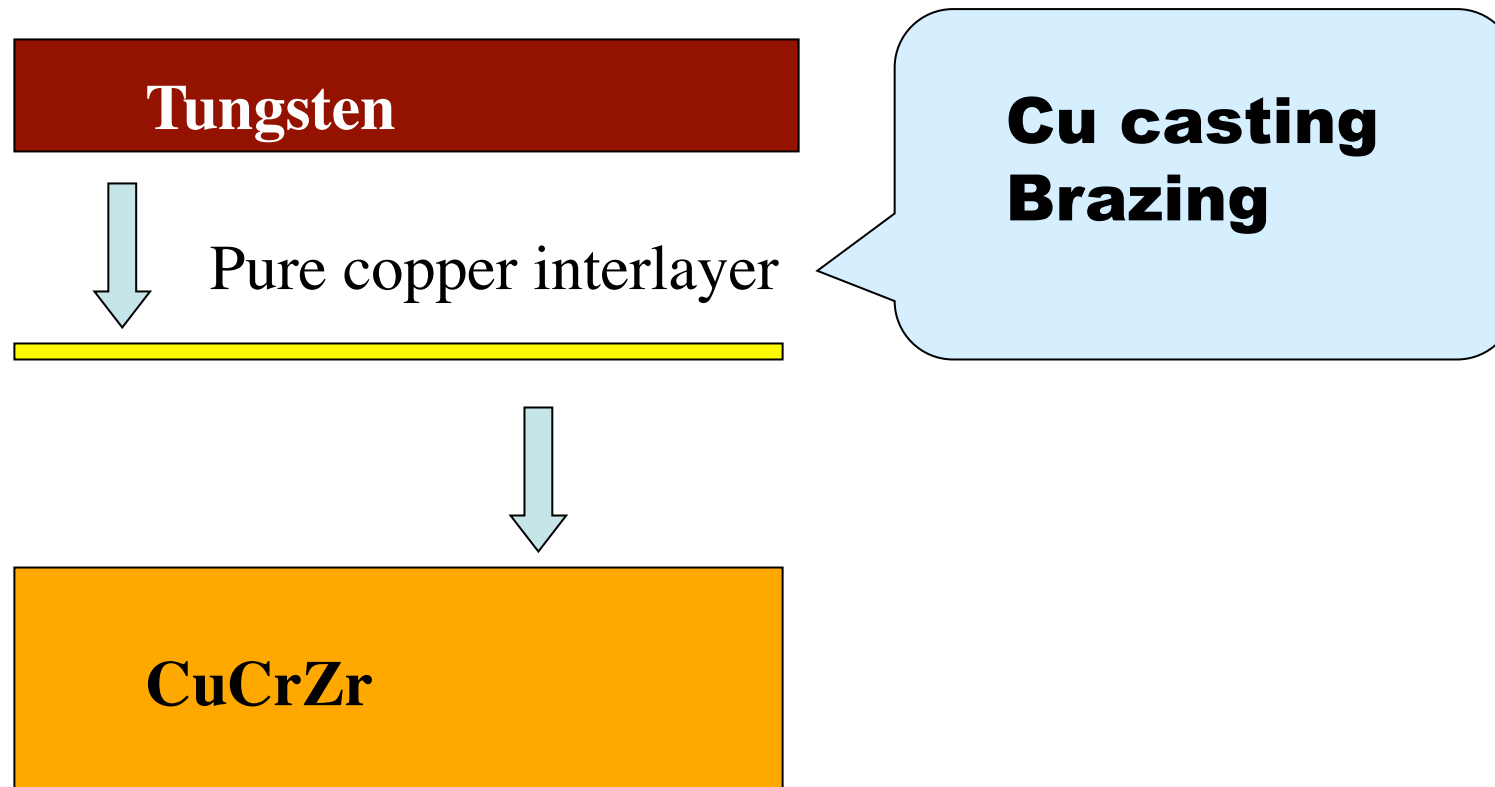


**Pre-brazed casting
Brazing**

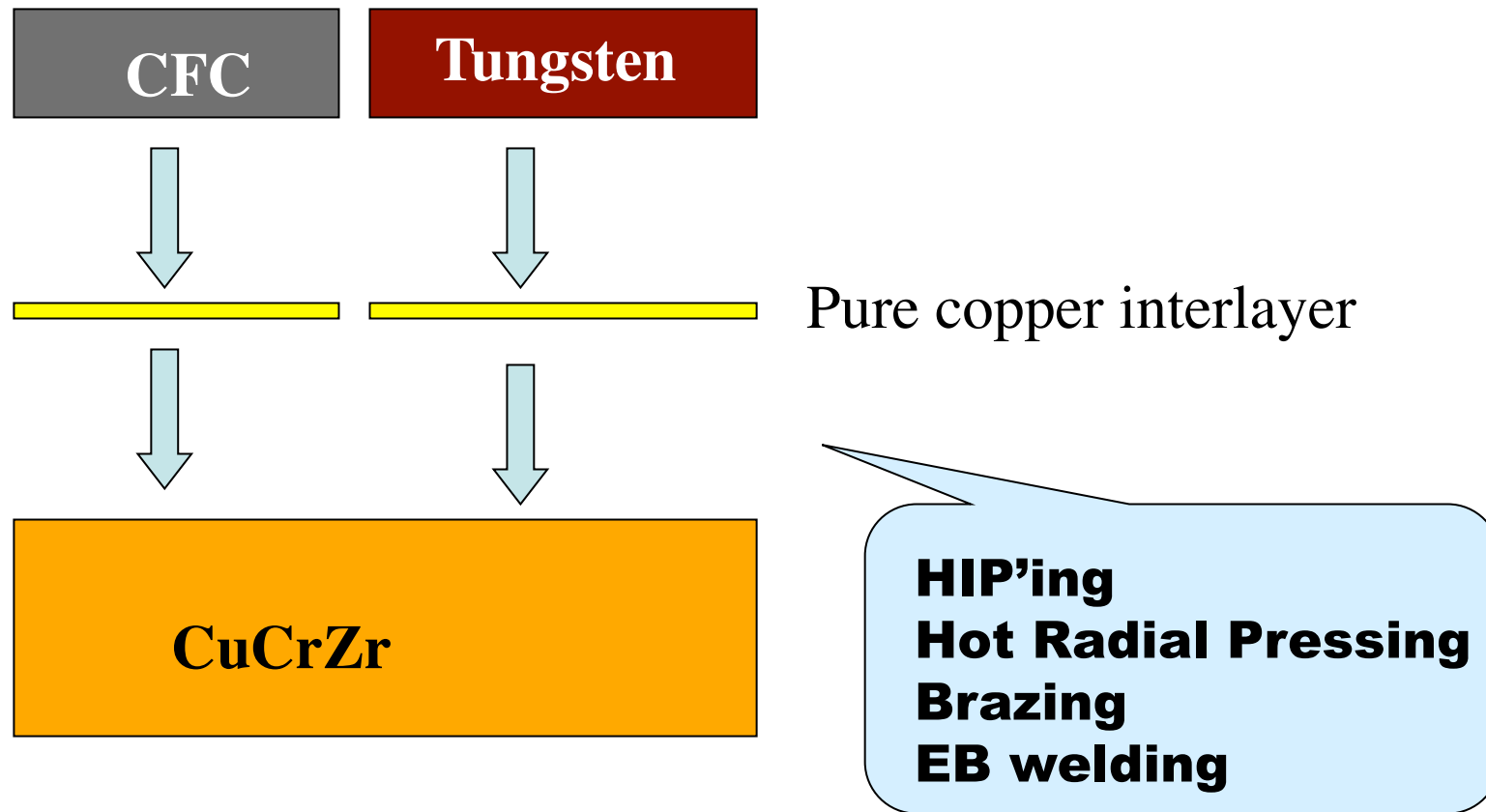
Armour to heat sink joints



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

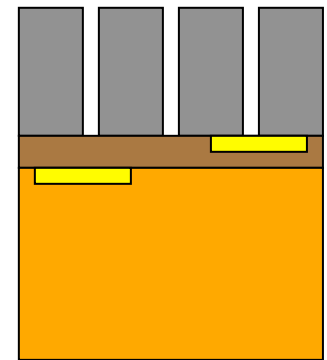
Values at RT

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

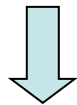
- Ultrasonic examination is the best technique.
- Defects of 2 mm can be detected reliably
- Inspection better performed from the rear side, prior to machining the cooling channels, when:
 - CFC armour
 - Be or W armour and fine castellation ($< 10 \times 10 \text{ mm}$)
- Main issue: differences in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



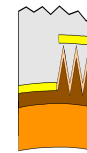
Non-destructive testing: CFC/Cu joint

Ultrasonic examinations

- 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



- Thermographic examination

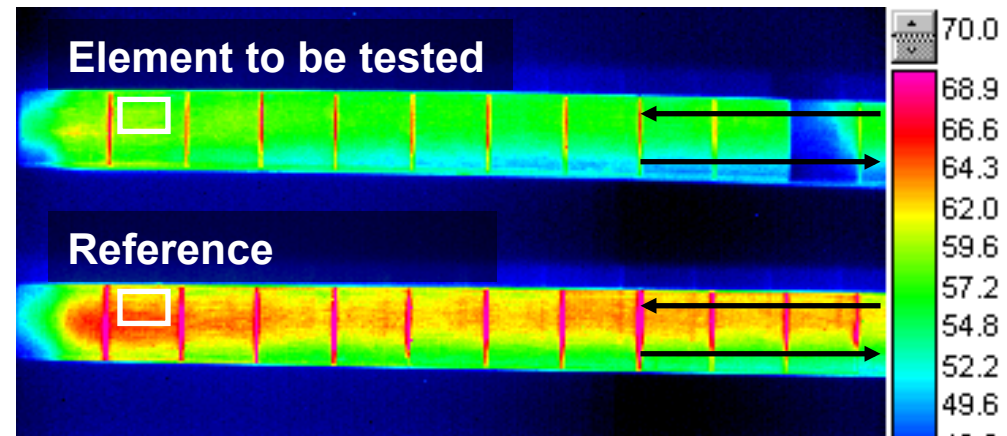


CfC

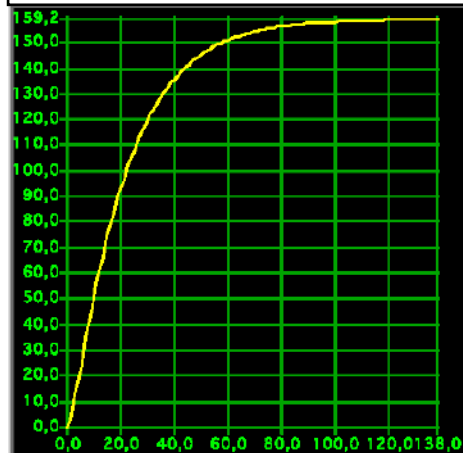
Cu

CuCrZr

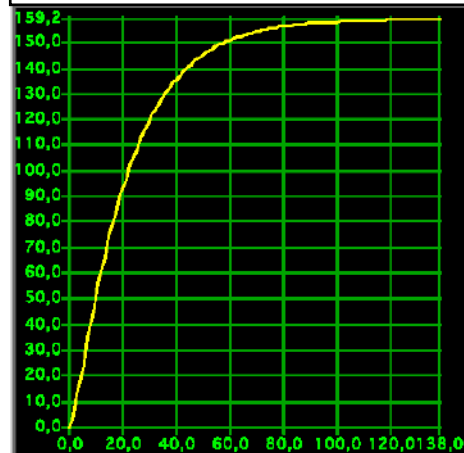
Transient Thermography Inspection



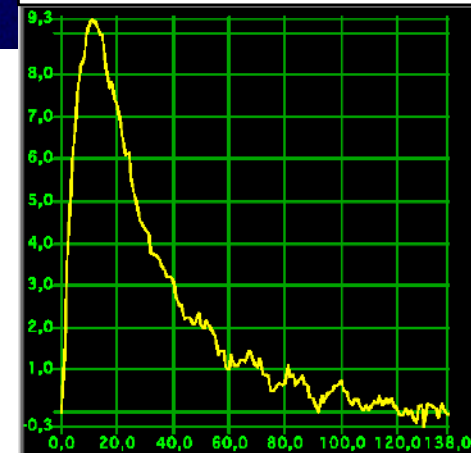
$T_{\min, \text{Ref}}(t)$
temperature evolution of
the reference tile



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



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



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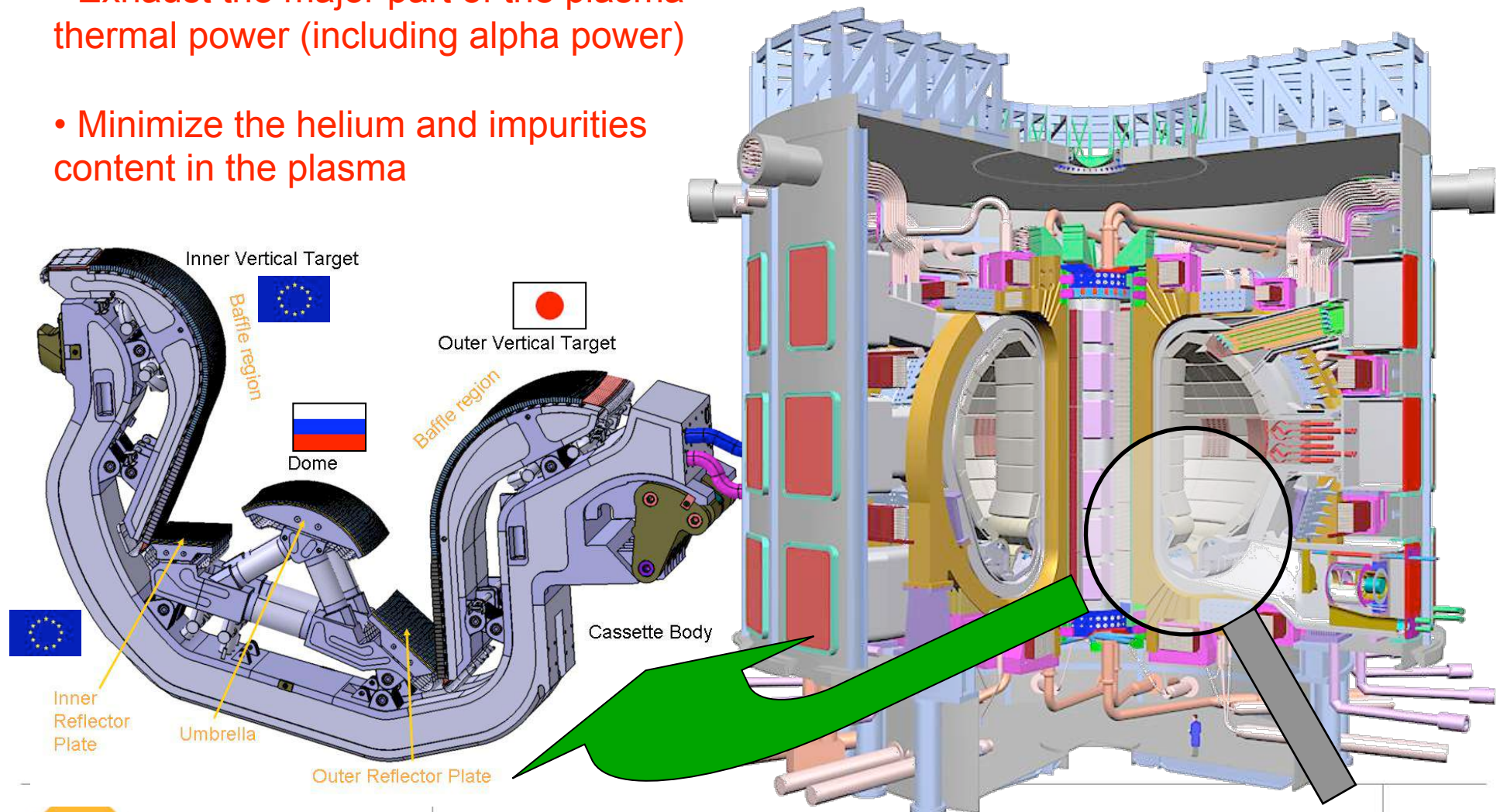
ITER Blanket

Conclusions

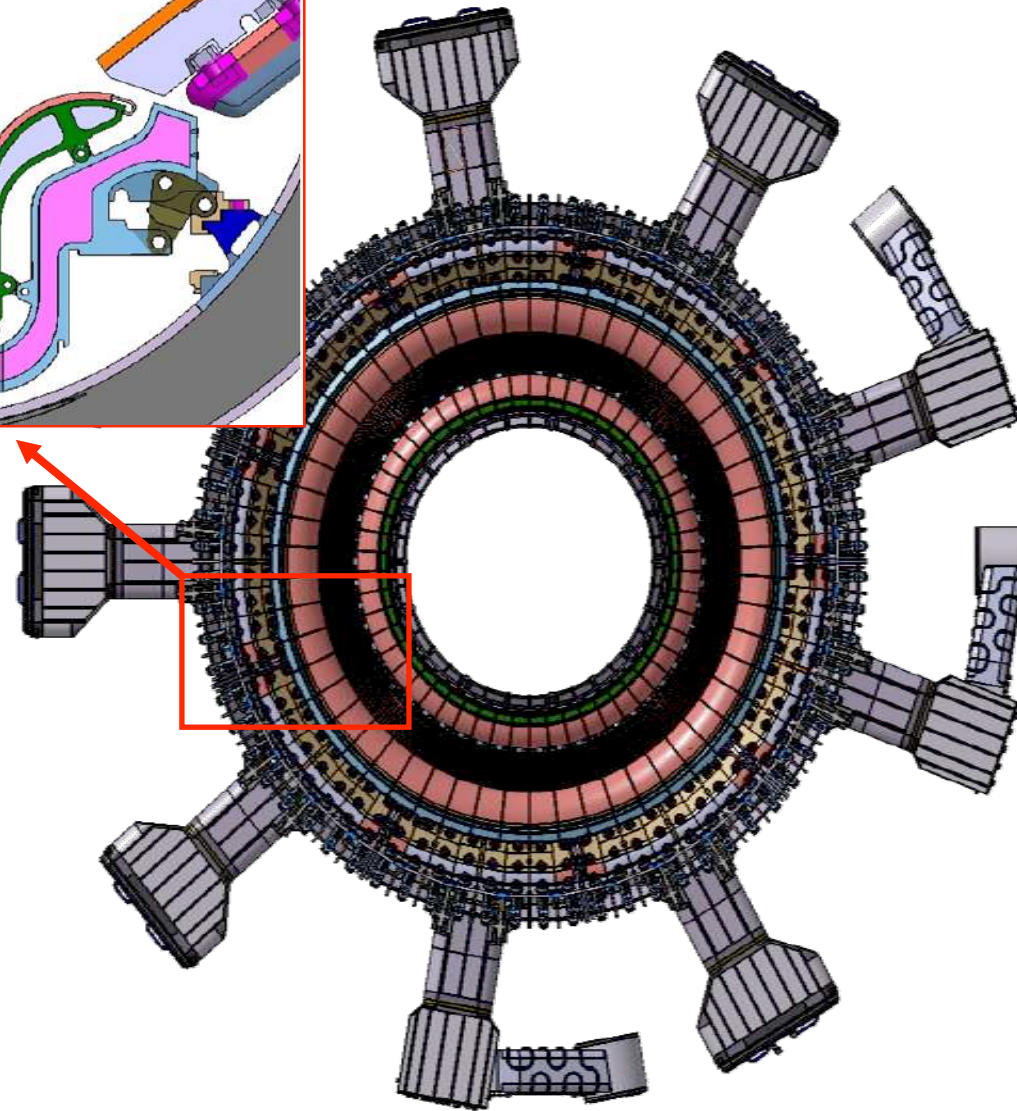
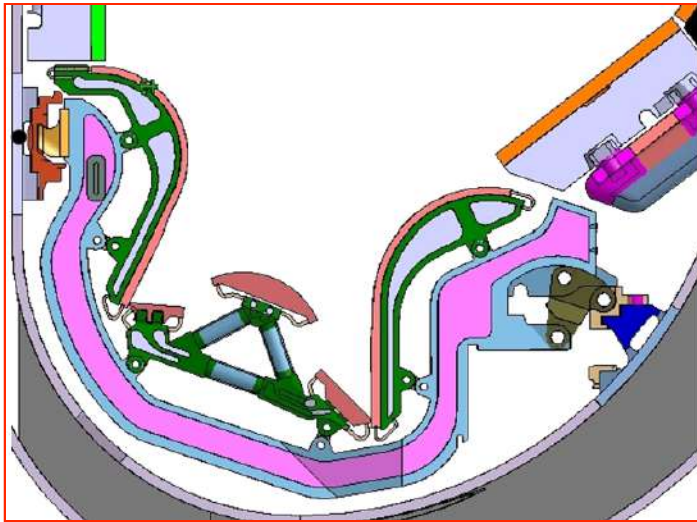
Divertor

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

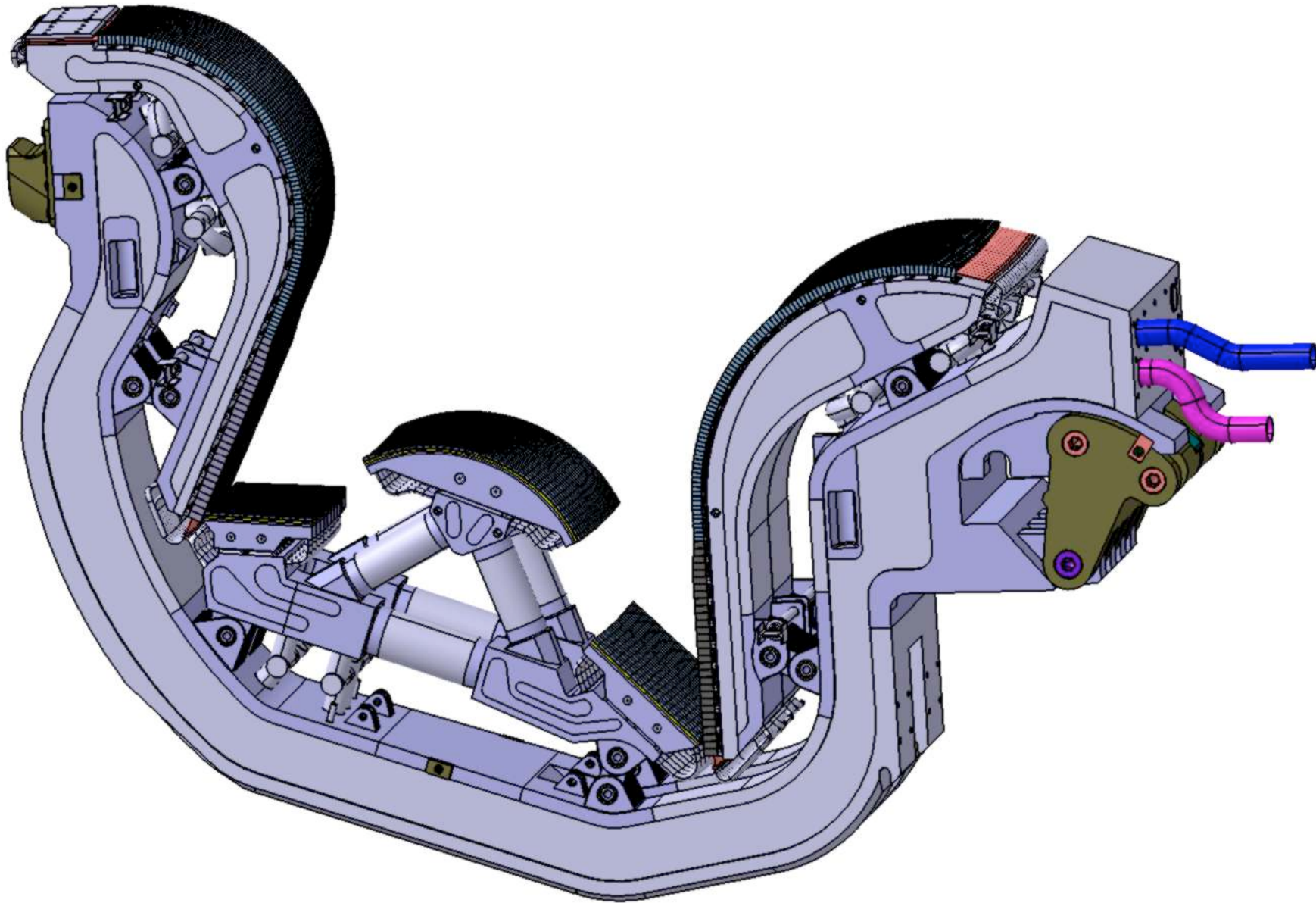


Divertor Cassette Layout

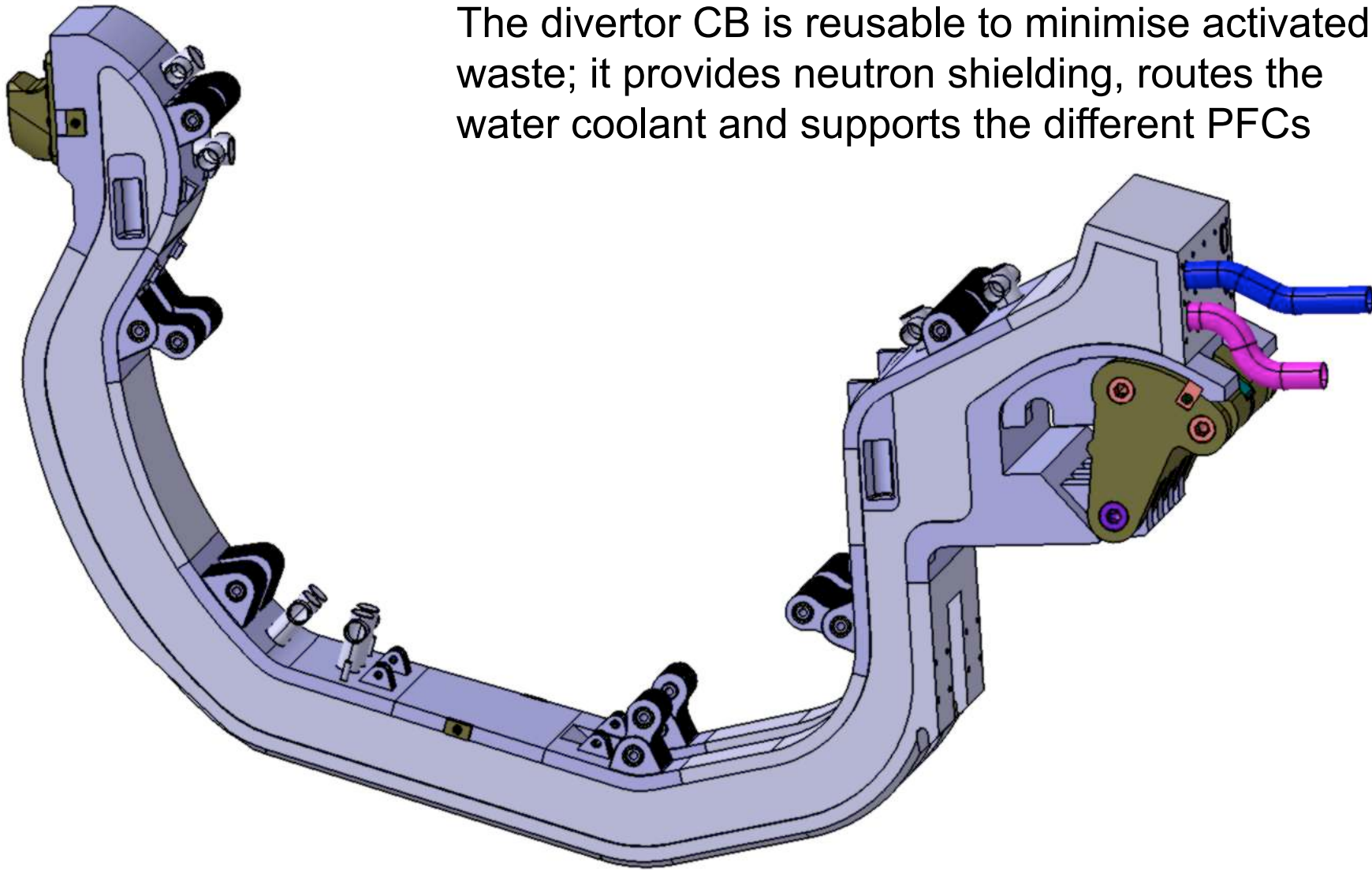


54 Cassettes
in a circular array
held in position by
two concentric
radial rails .

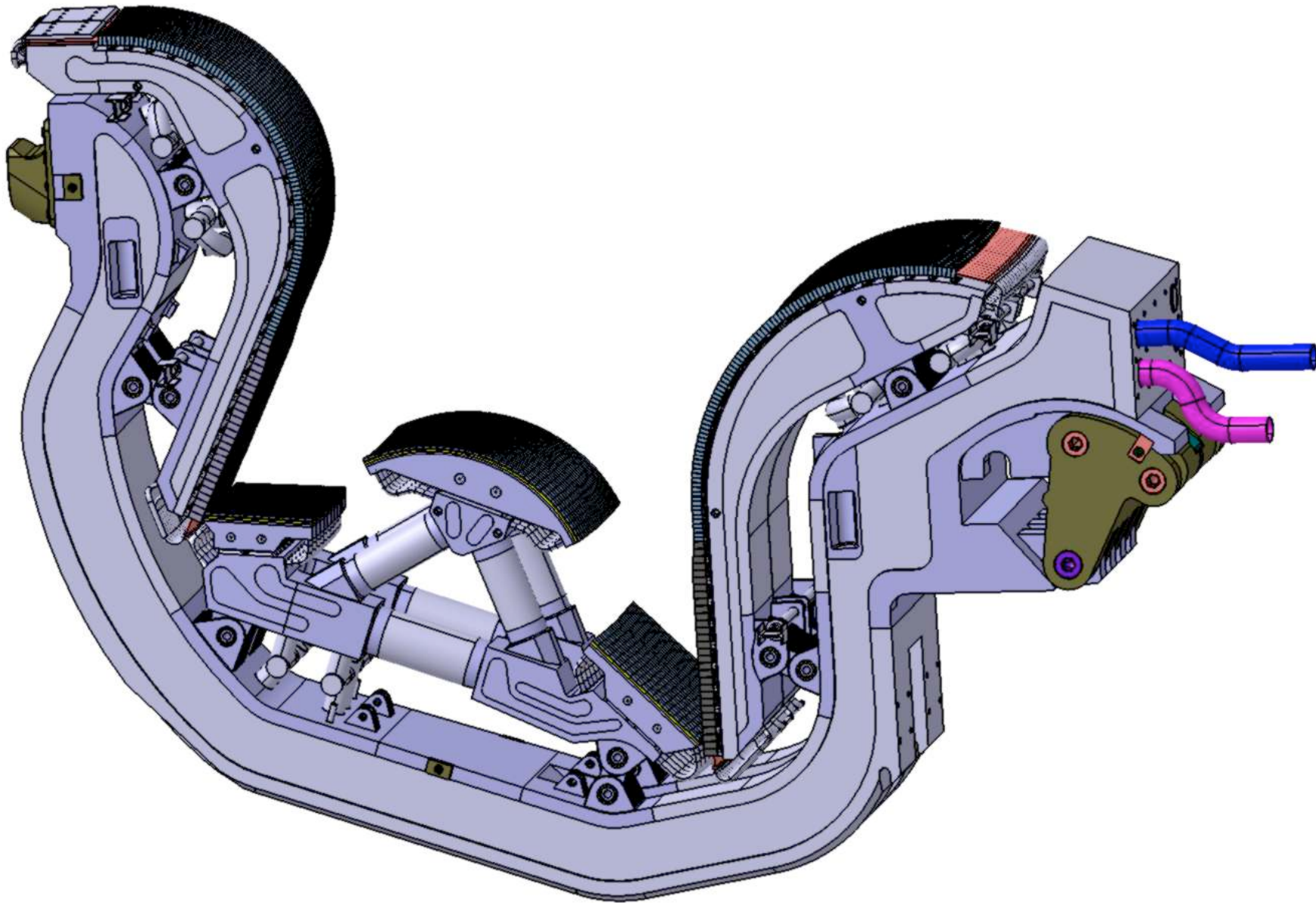
Divertor System

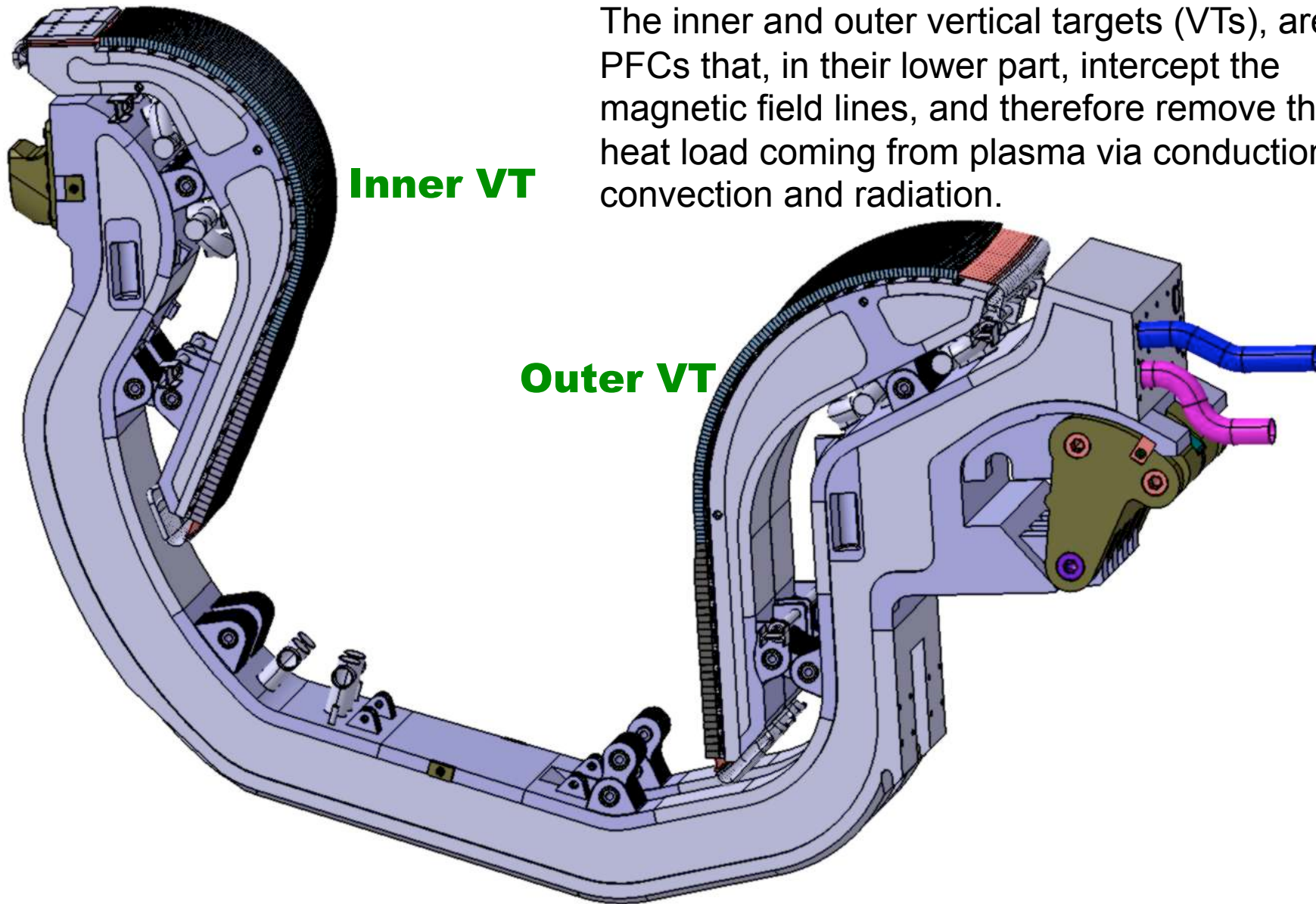


The divertor CB is reusable to minimise activated waste; it provides neutron shielding, routes the water coolant and supports the different PFCs



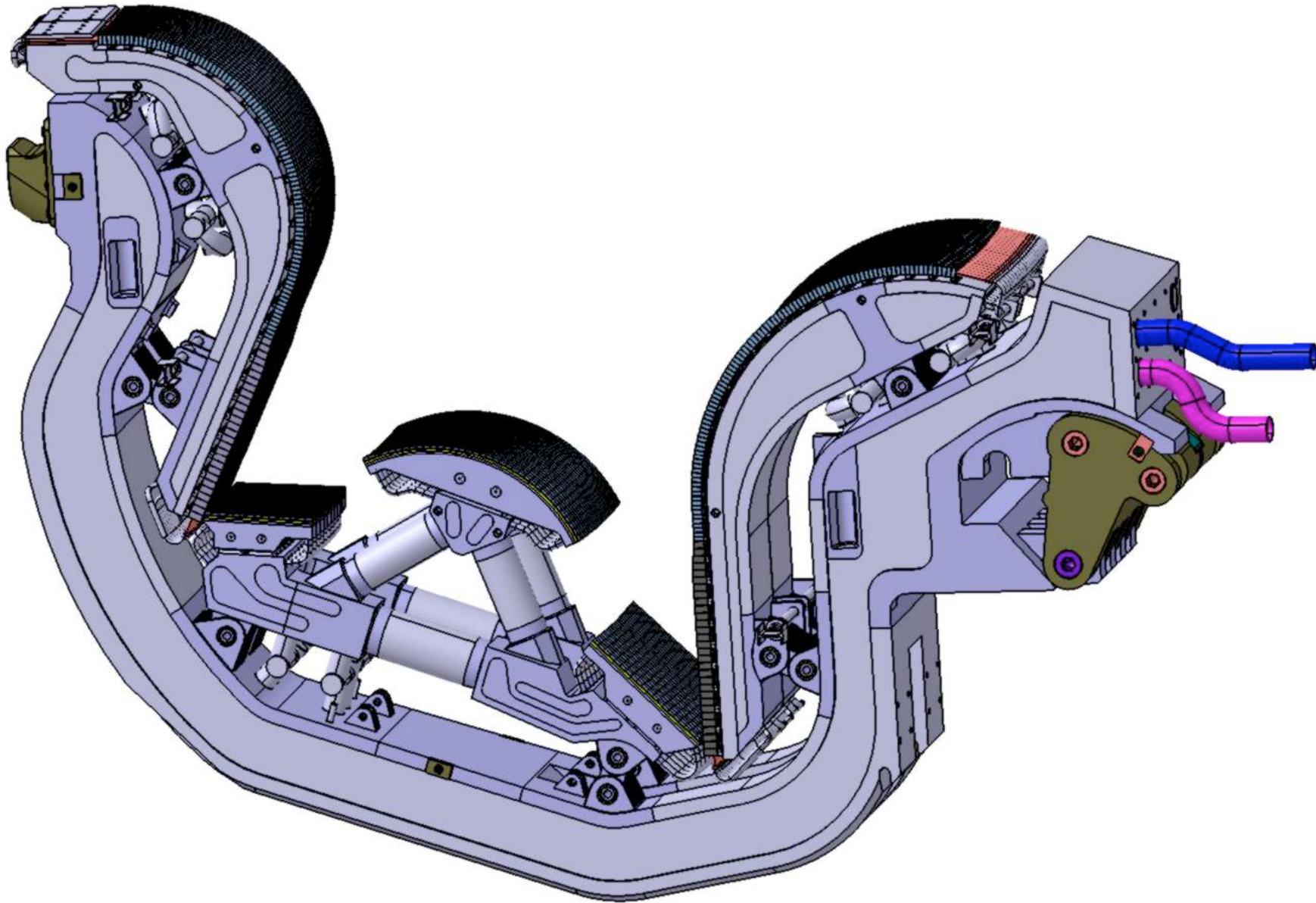
Divertor System





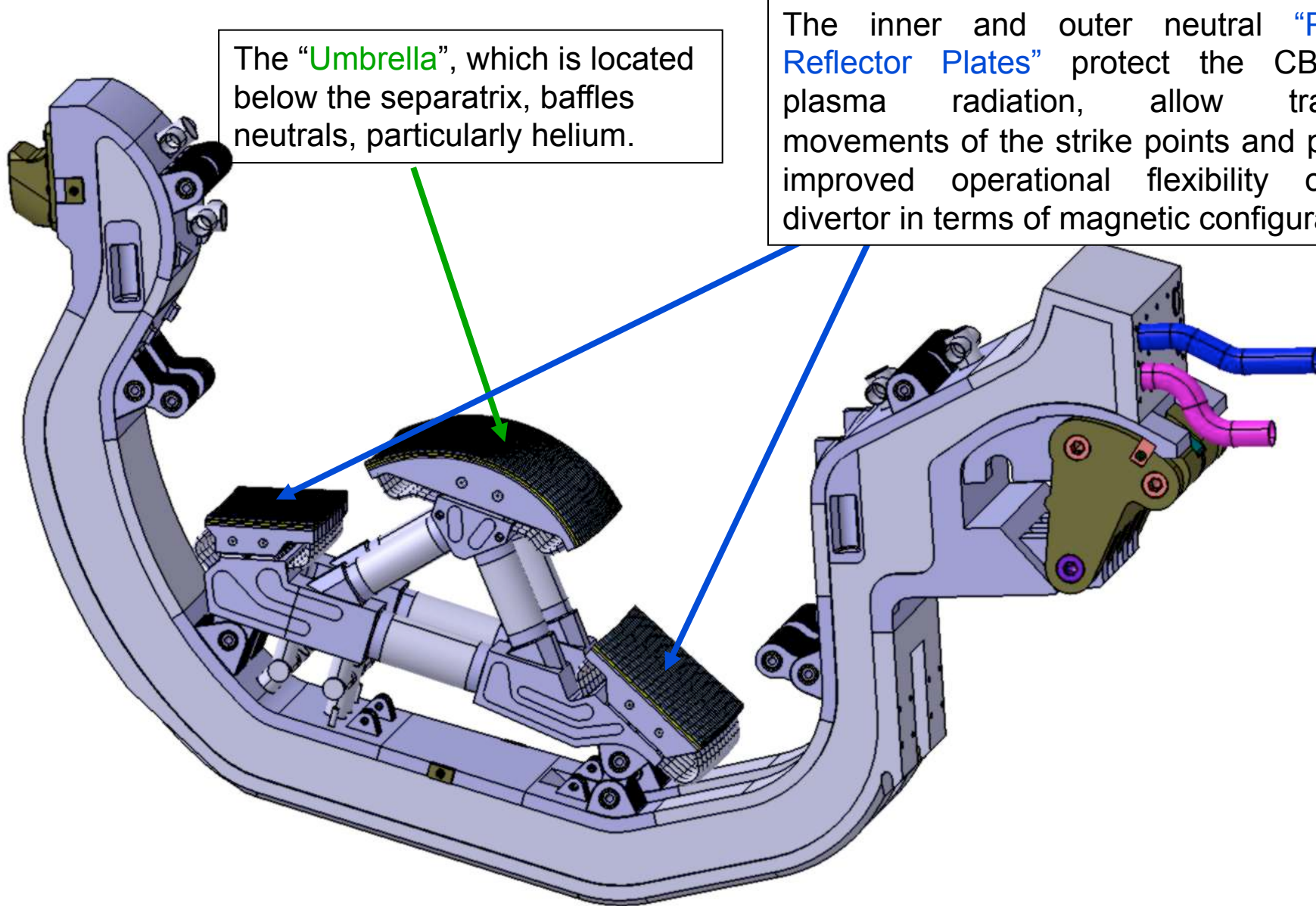
The inner and outer vertical targets (VTs), are the PFCs that, in their lower part, intercept the magnetic field lines, and therefore remove the heat load coming from plasma via conduction, convection and radiation.

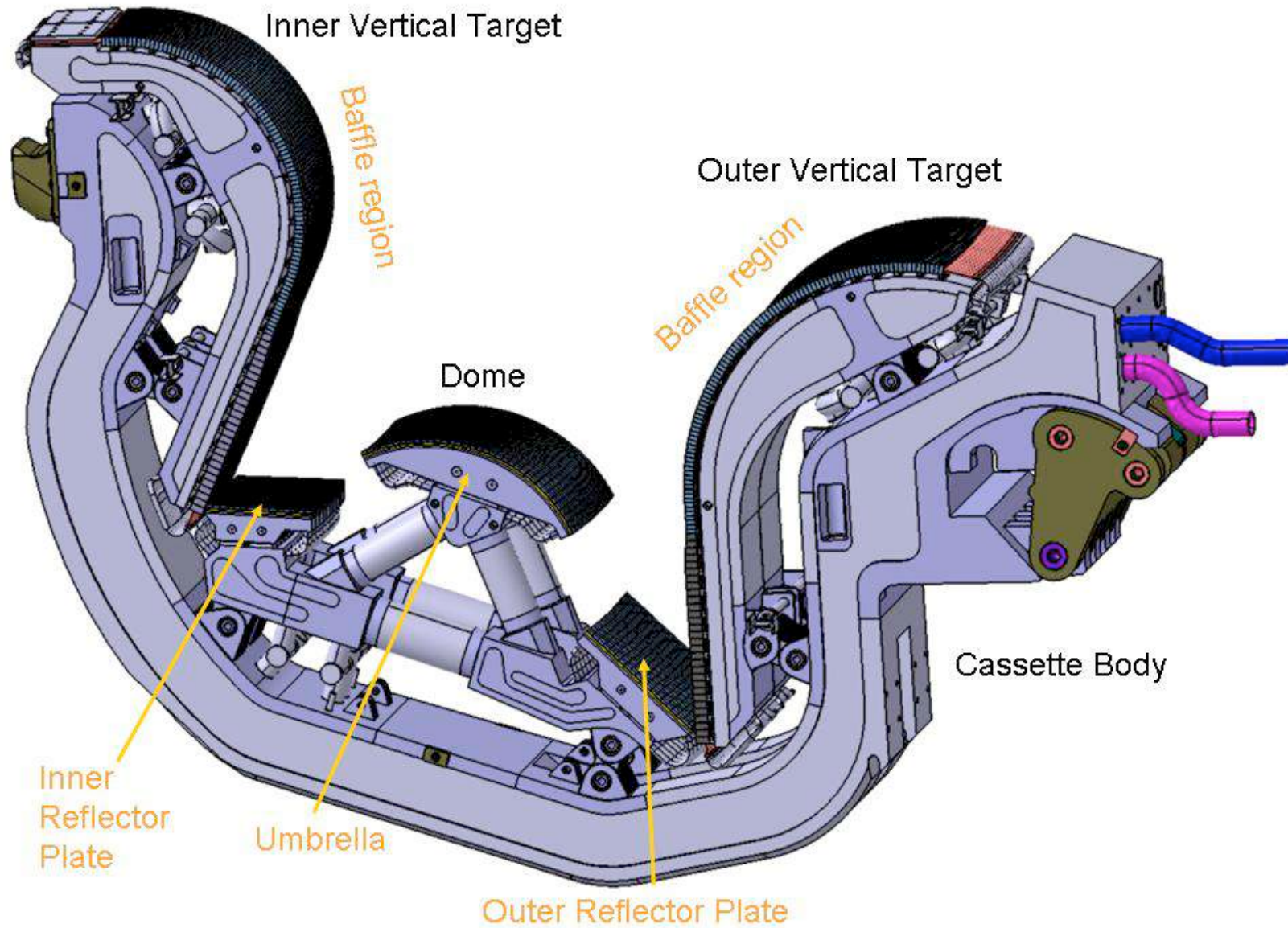
Divertor System



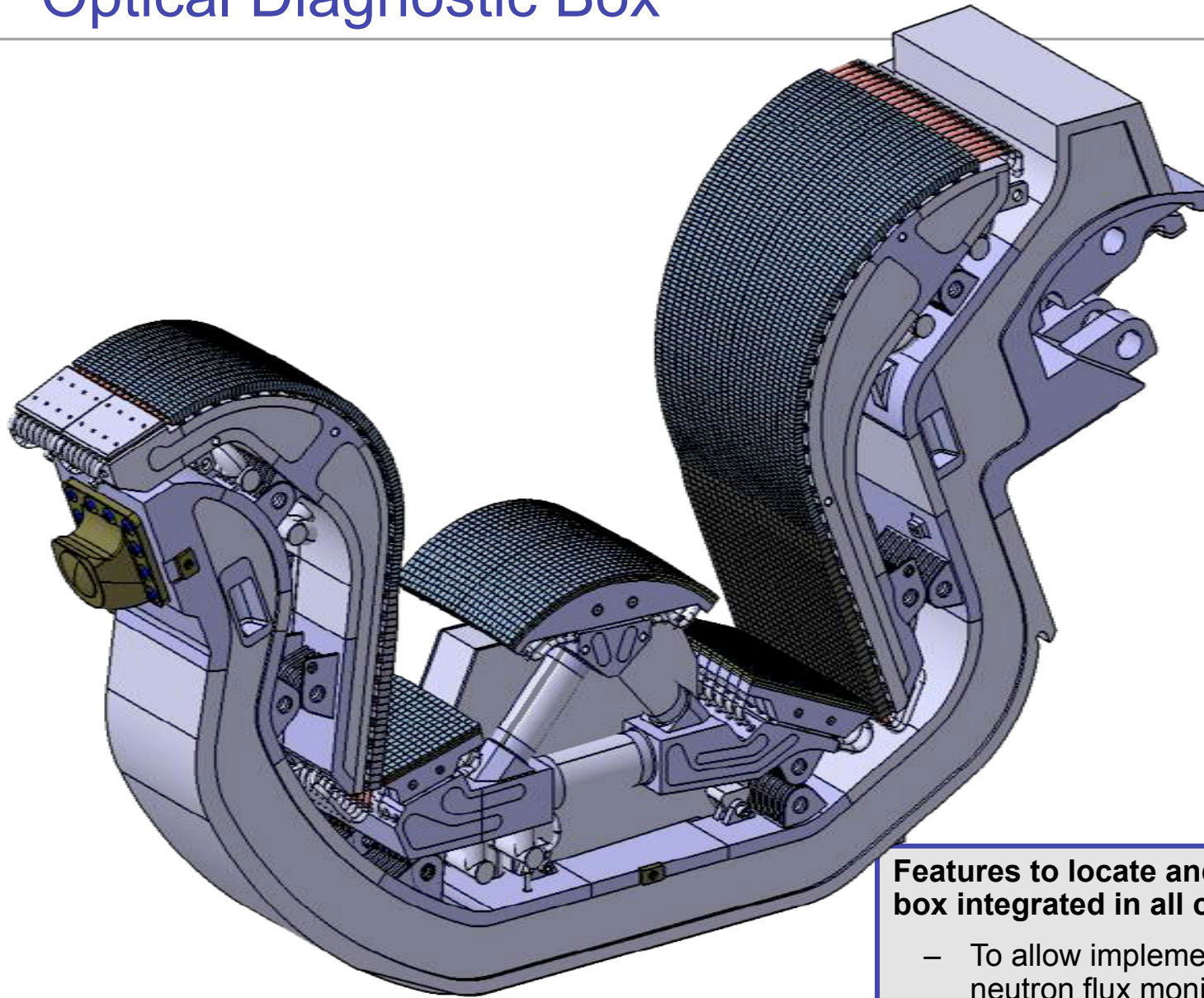
Divertor System

Dome



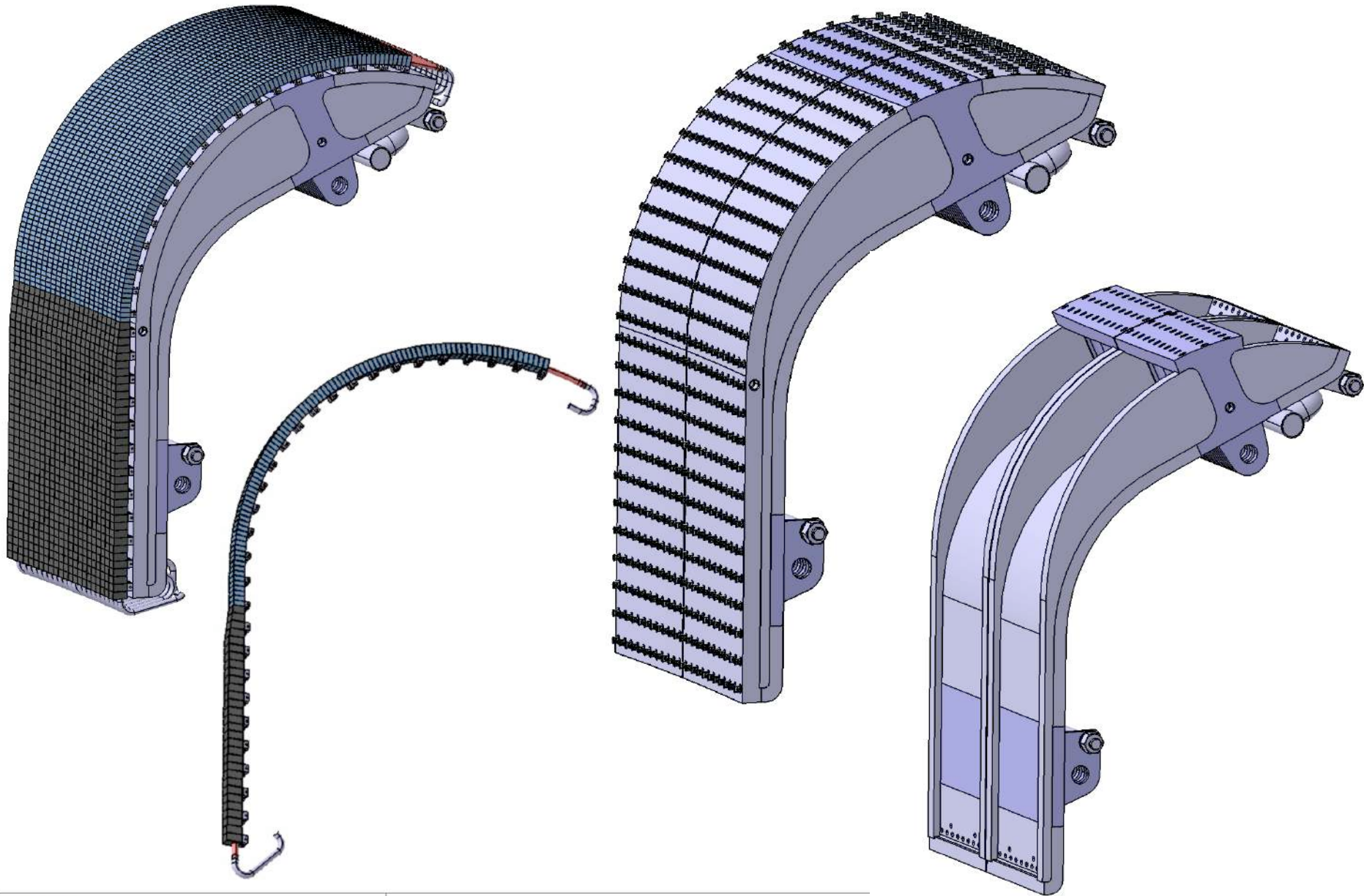


Optical Diagnostic Box



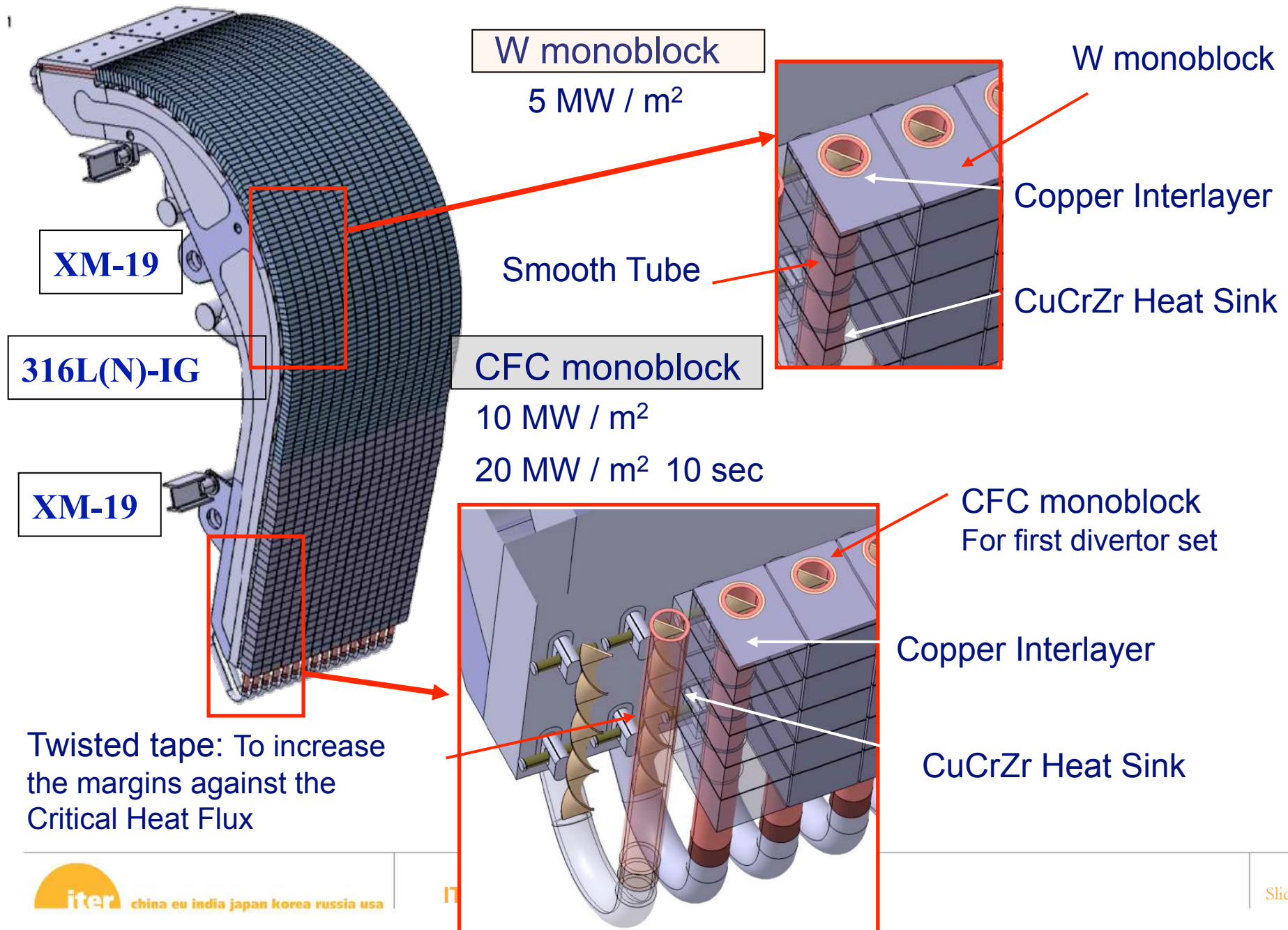
Features to locate and cool diagnostic box integrated in all cassettes

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



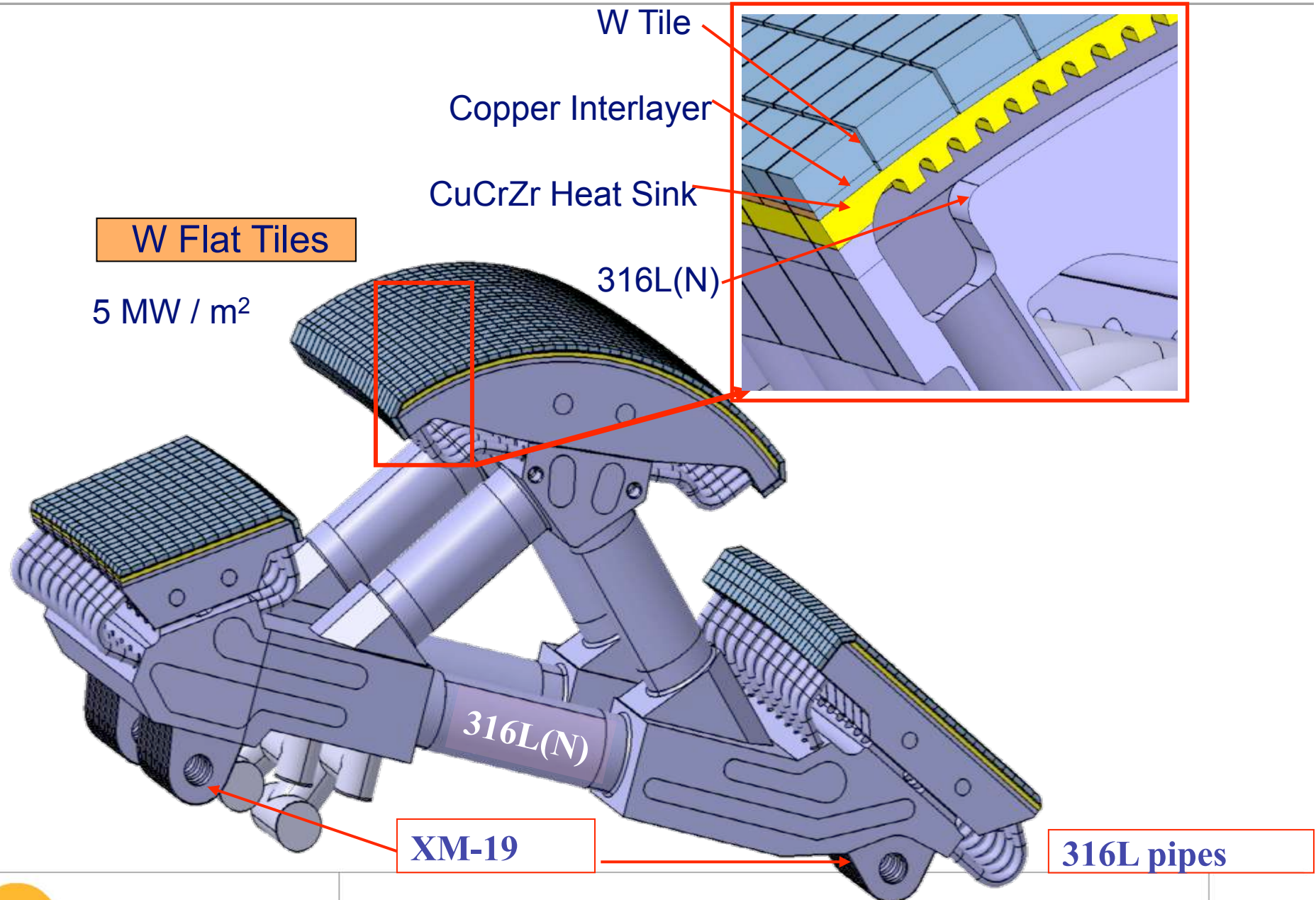
Plasma-Facing Components

Vertical Targets

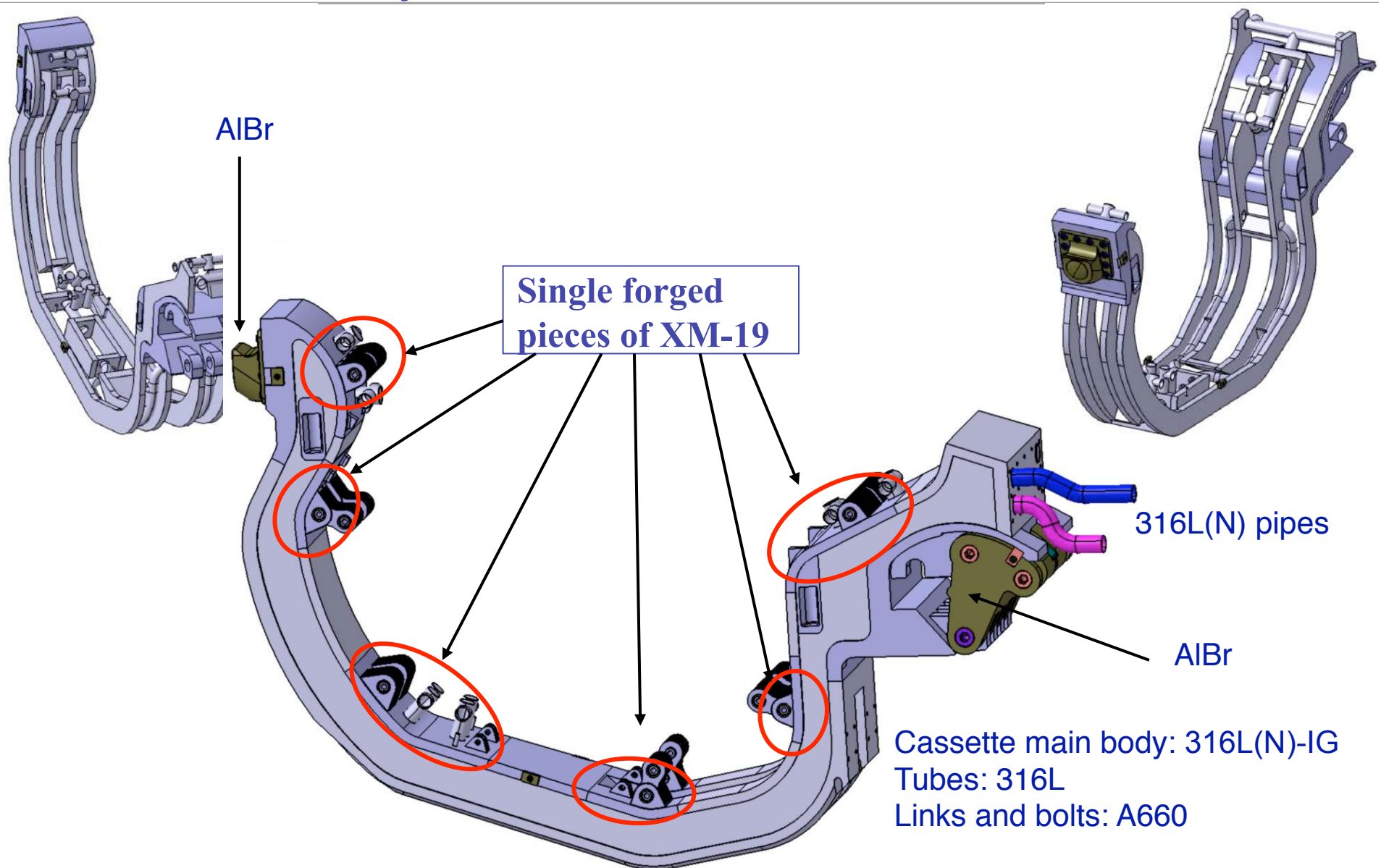


Plasma-Facing Components

Dome



Cassette Body



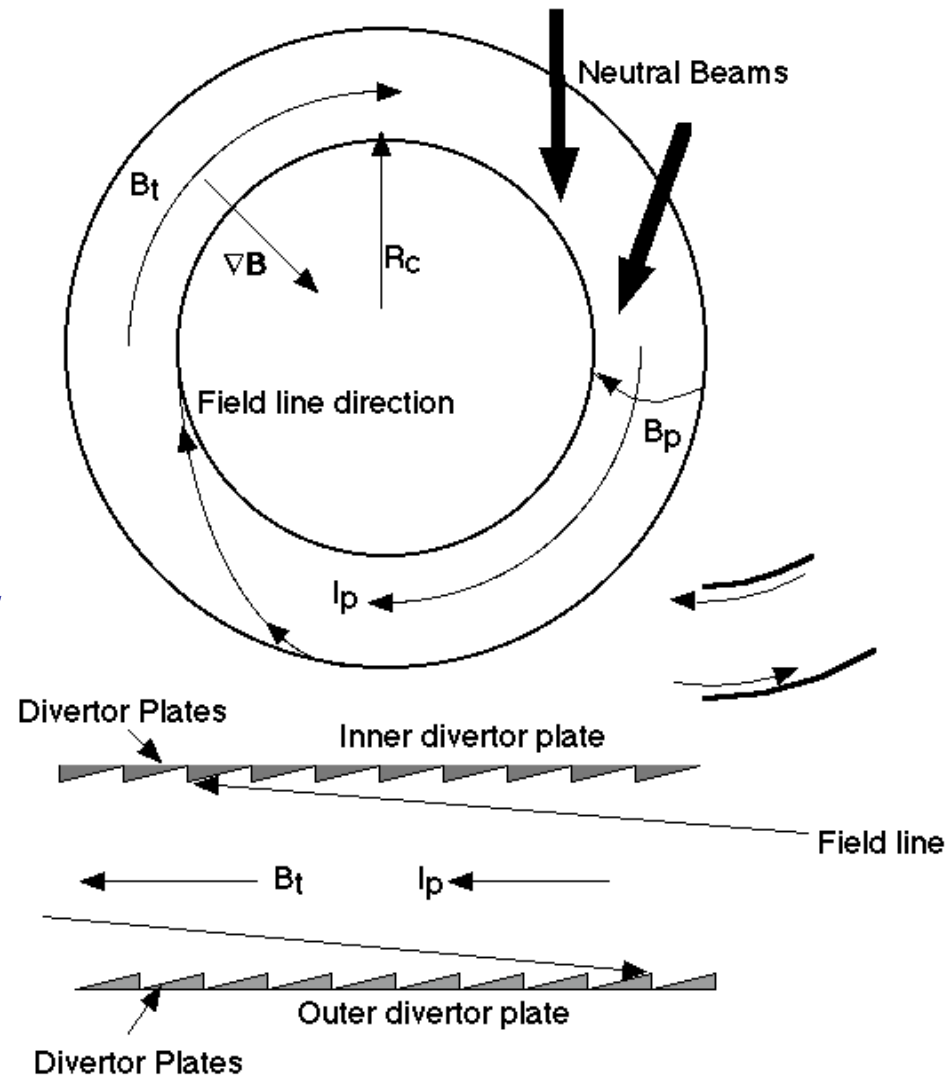
Divertor System

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

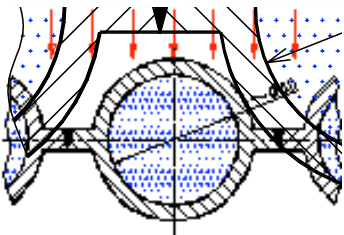
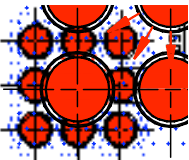
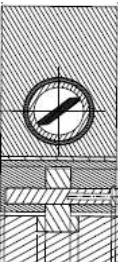
Alignment of PFCs

Orientation of Divertor, B_t , I_p and Neutral Beams



View from Top

- 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/m² (strike point region) and 5 MW/m² (baffle region)
- Under slow transient thermal loading conditions:
 - lower divertor vertical target geometry has a **design** surface heat flux up to 20 MW/m² for sub-pulses of less than 10 s
- The dome shall sustain **design** heat fluxes of up to 5 MW/m²
- The umbrella and the particle reflector plates shall sustain local heat flux up to 10 MW/m², which can be transiently swept across the surface (about 2 s) as the plasma is returned to its correct position

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
<u>Max heat load MJ/m²</u> <u>Lifetime years</u> <u>Nr. of full load cycles</u> <u>Neutron damage dpa</u> <u>Materials</u>	- 25 8000 - Ferritic-Martens. steel	- 4 10 10 Zircaloy - 4	10 ~ 5-8 3000 - 16000 0.2 CuCrZr & CFC/W
<u>Coolant</u> - 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

Divertor coolant design parameters

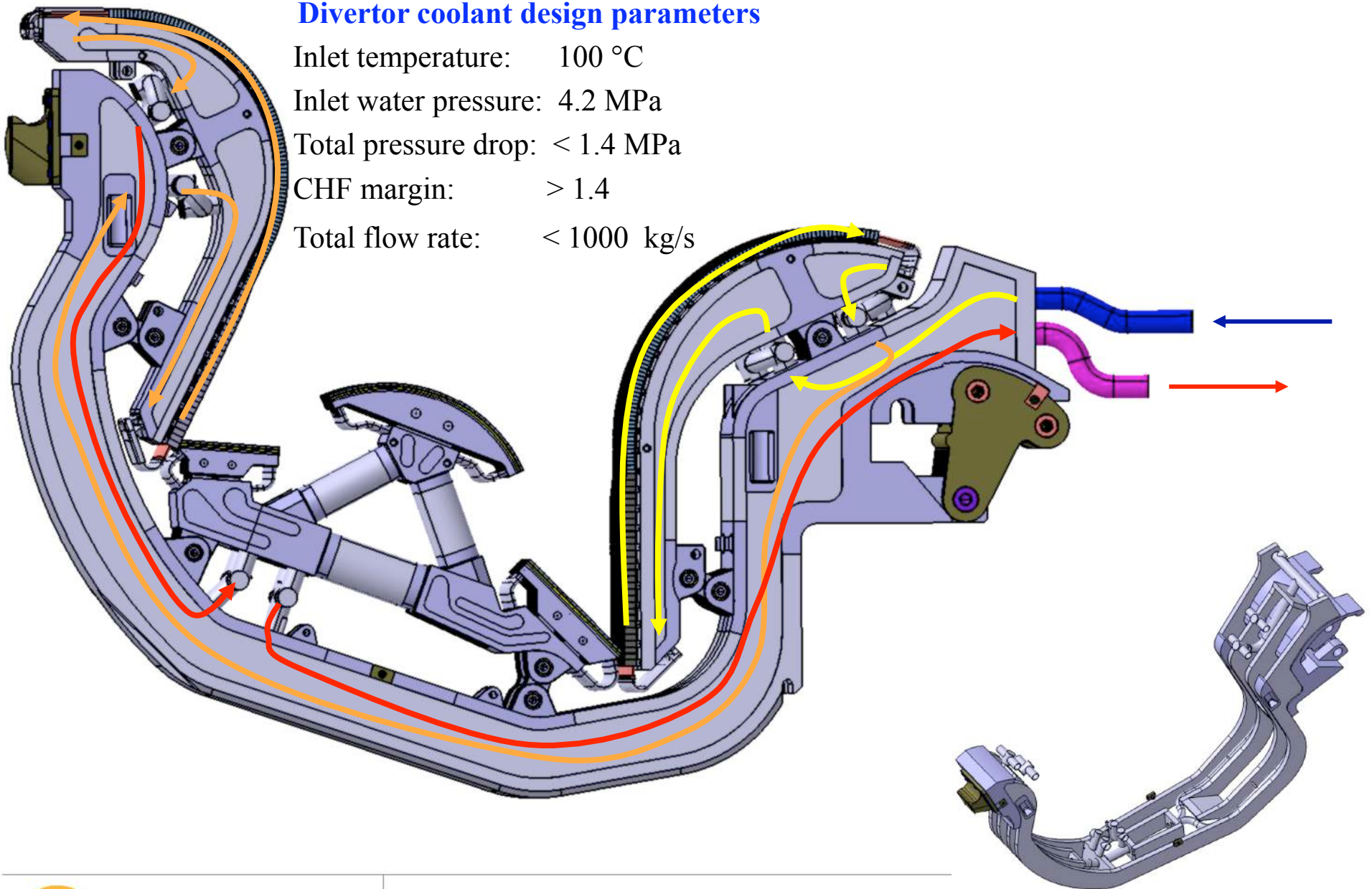
Inlet temperature: 100 °C

Inlet water pressure: 4.2 MPa

Total pressure drop: < 1.4 MPa

CHF margin: > 1.4

Total flow rate: < 1000 kg/s



Thermo-Hydraulics

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

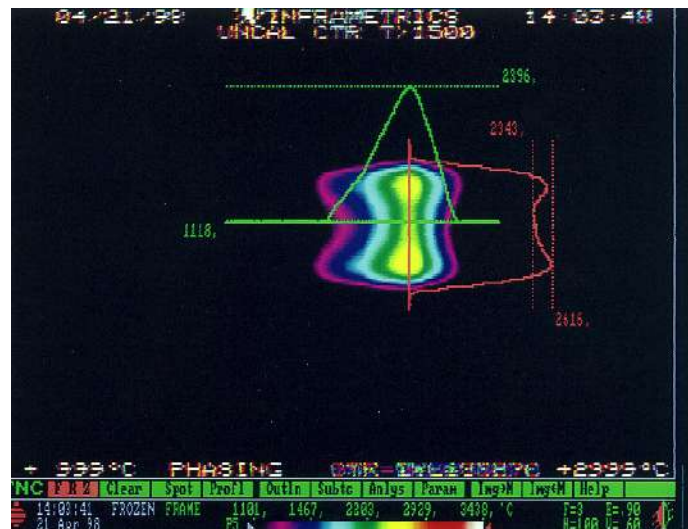
Experiments



Hydraulic testing of DOME

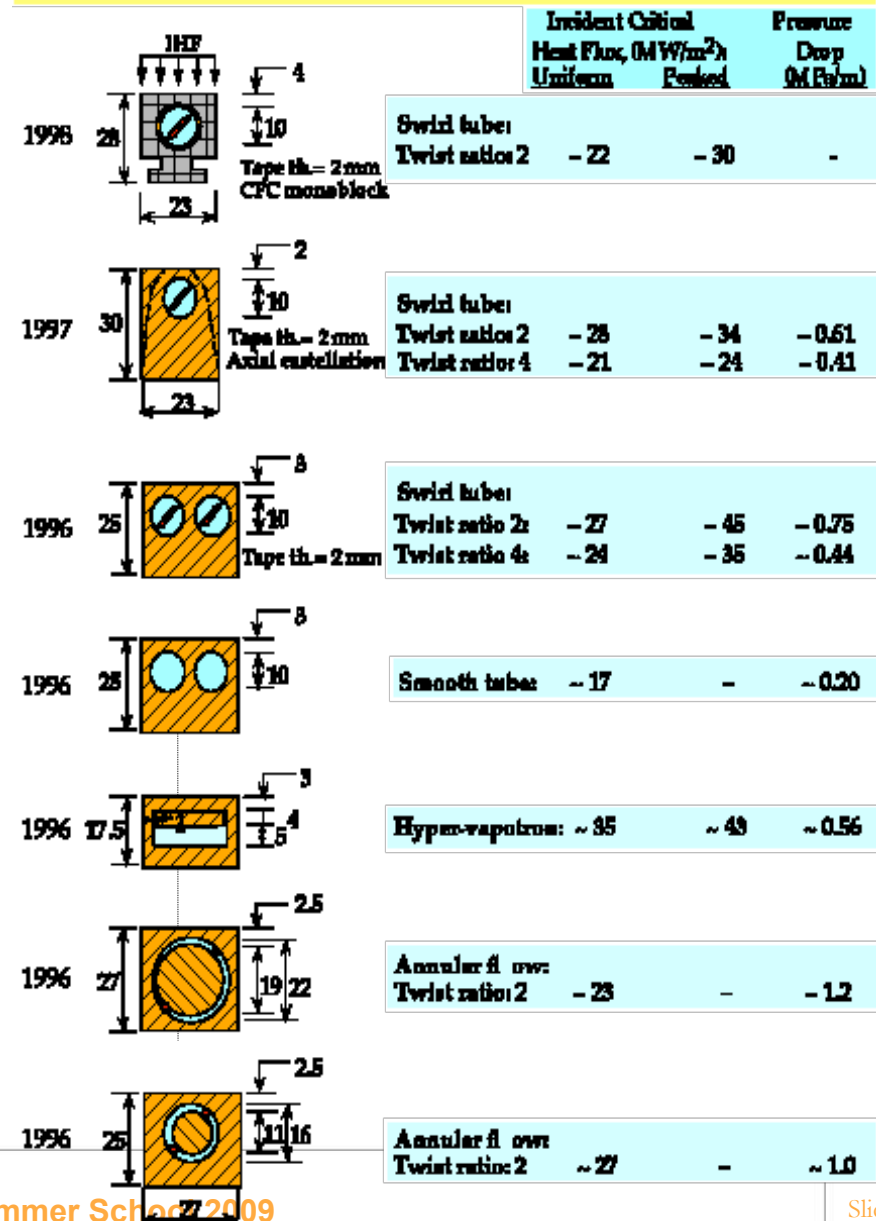


Critical Heat Flux

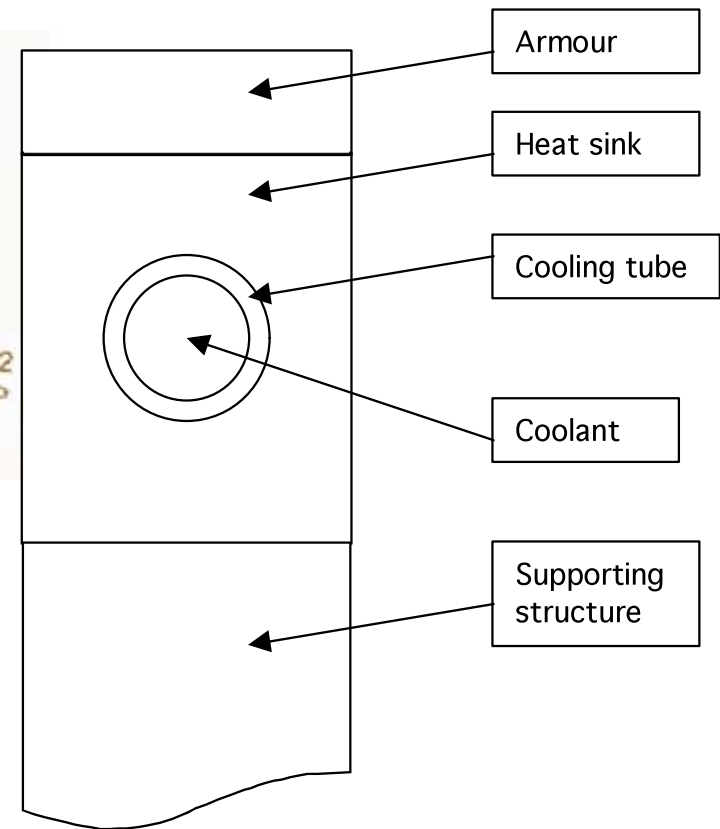


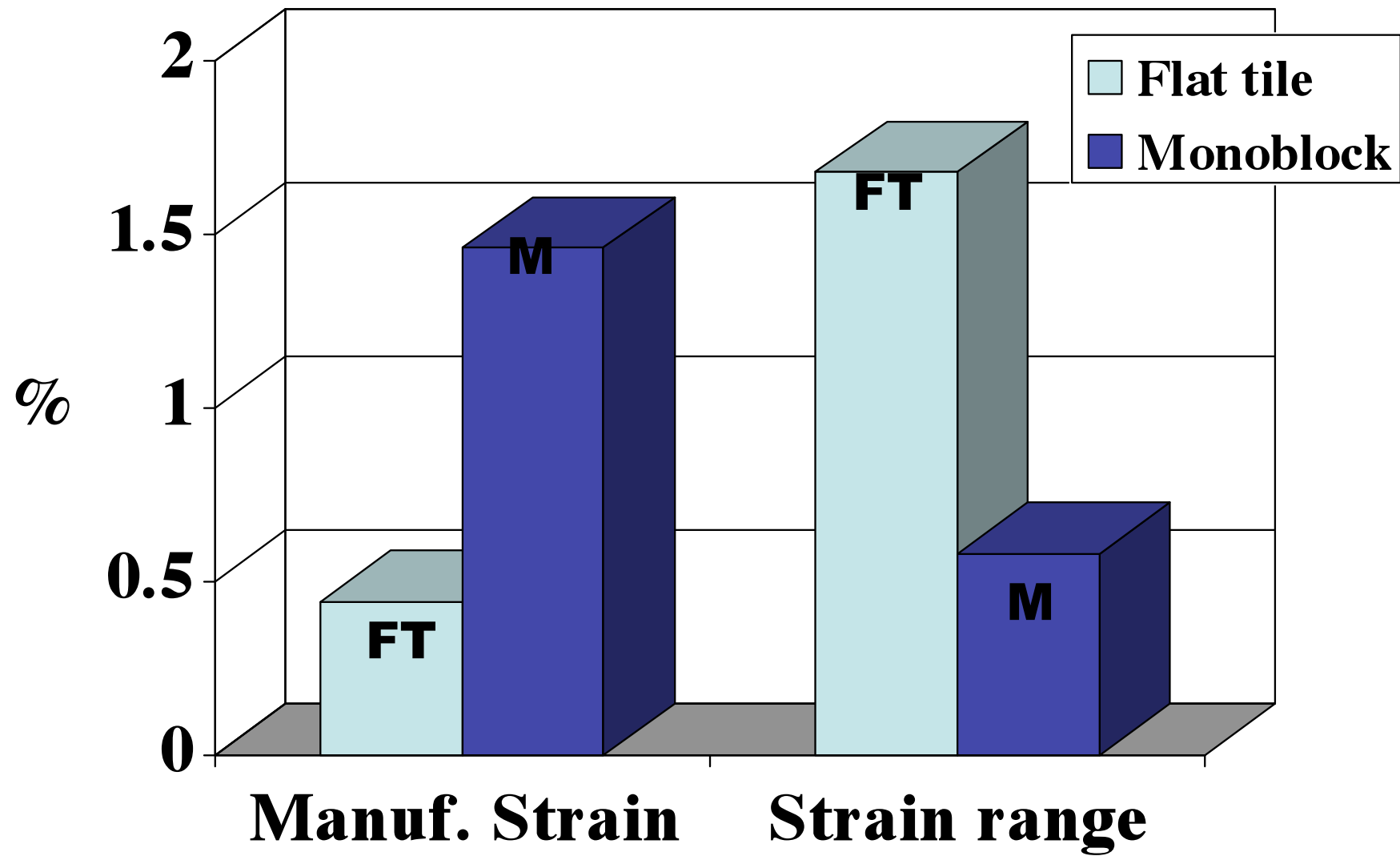
THERMO-HYDRAULIC TEST RESULTS BY CEA CADARACHE 1996-98

- Mock-ups: Cu-CuZn (1996), DS-Cu (1997), CPC 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)



Terminology, Flat Tile and Monoblock



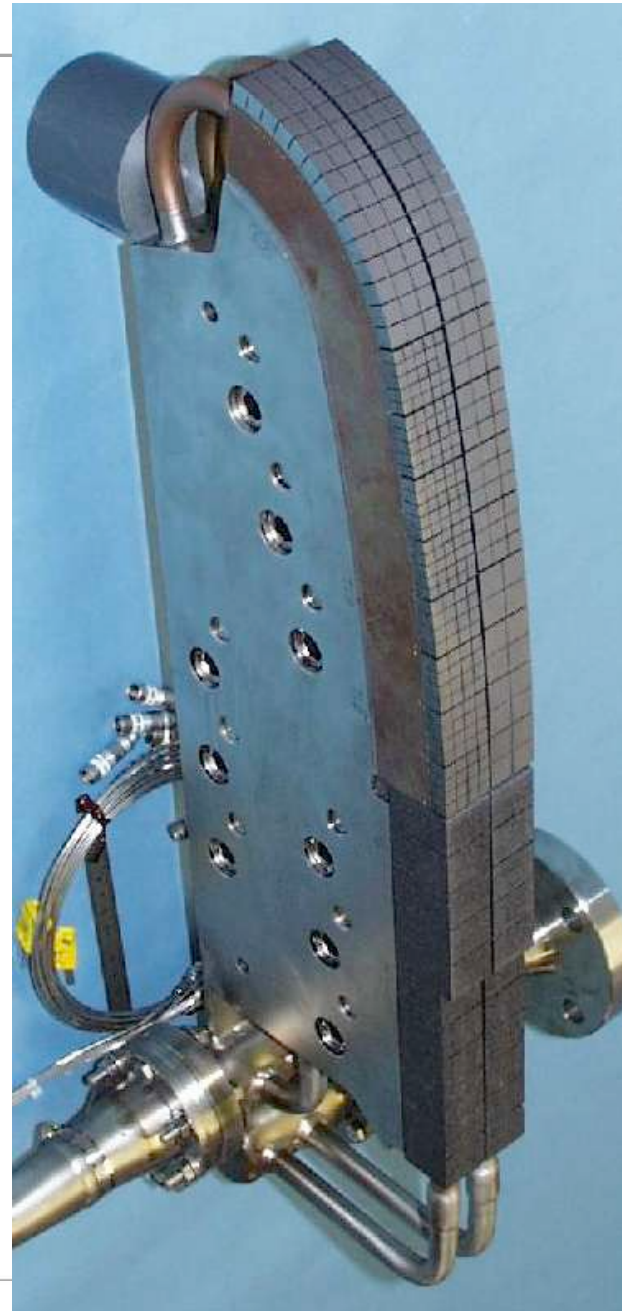


Vertical Target Medium-Scale Prototype

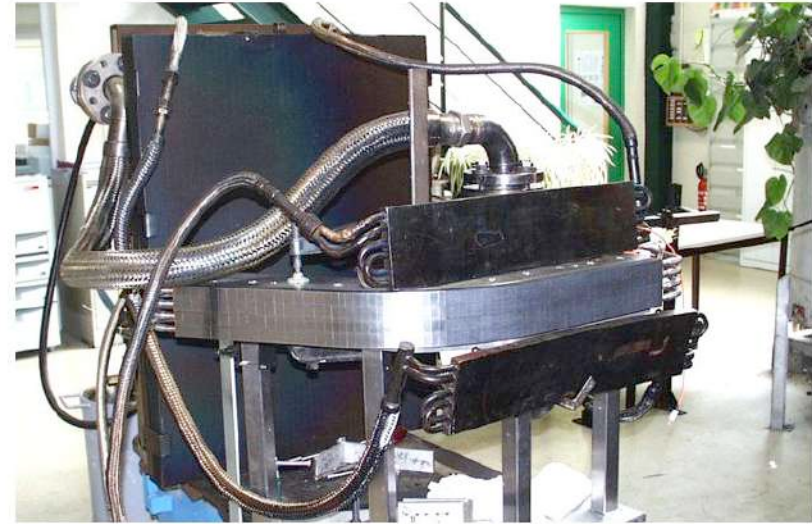
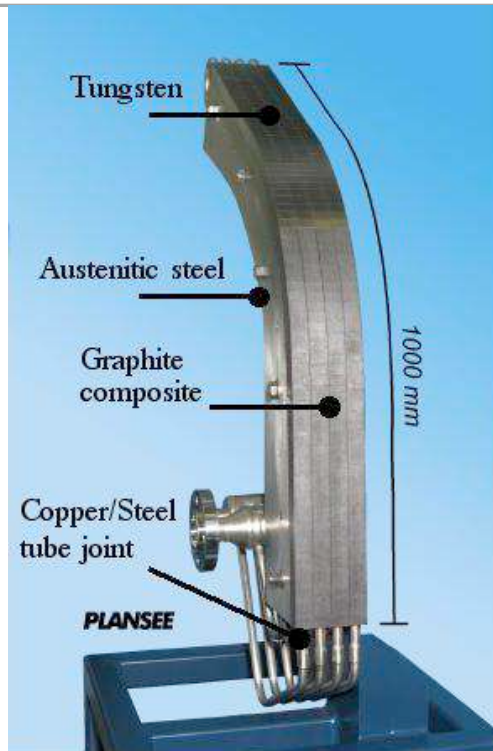
Test results

- **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

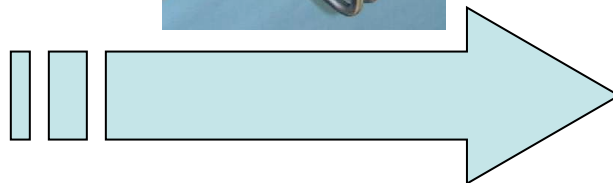


- **W monoblocks:**
10 MW/m² x 1000 cycles
- **CFC monoblock**
10 MW/m² x 1000 cycles
20 MW/m² x 1000 cycles
23 MW/m² x 1000 cycles

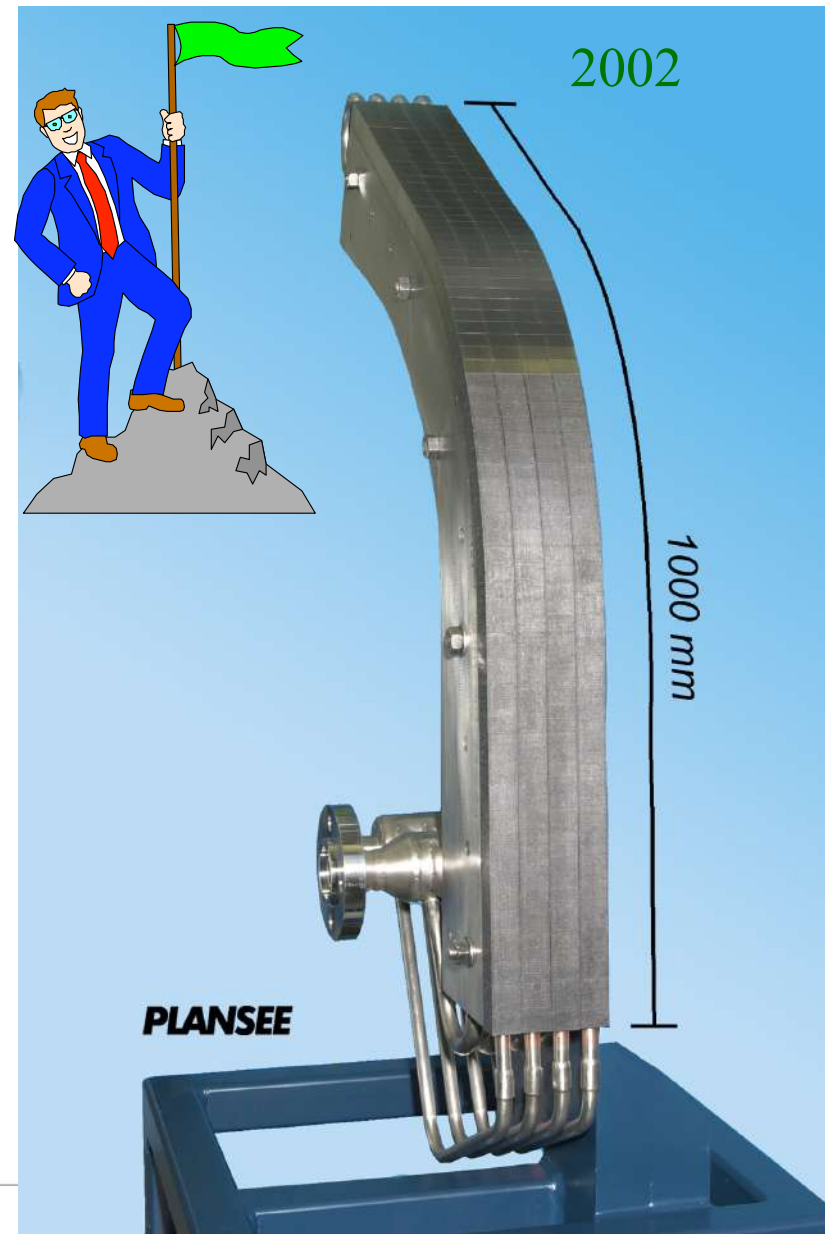
1995



1998



2002



Vertical Target component with W armour



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

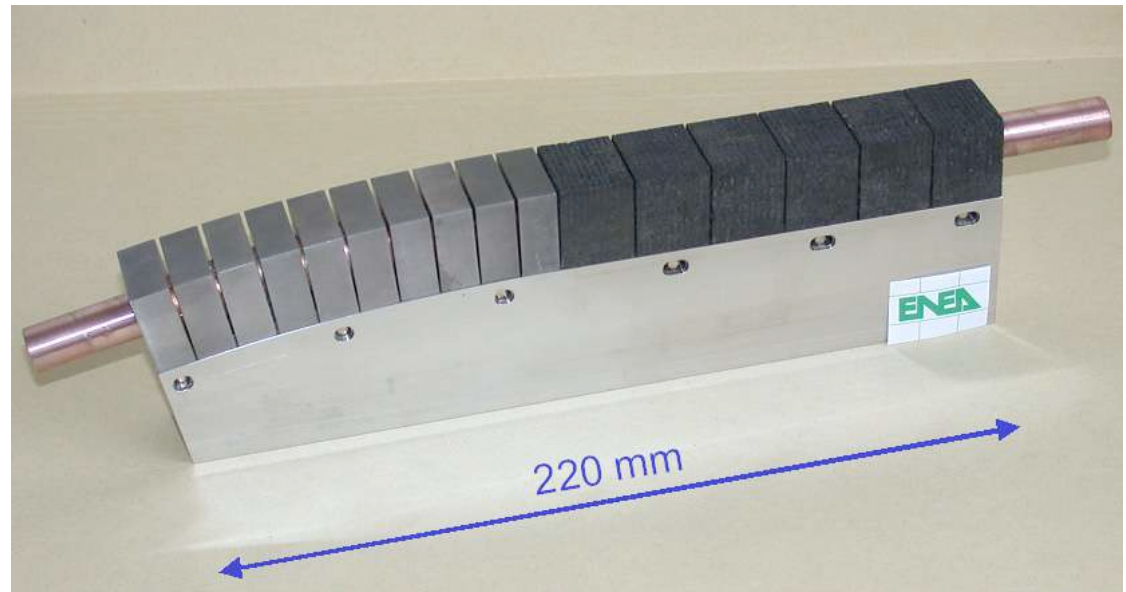
5 MW/m² x 100 cycles

10 MW/m² x 1000 cycles

20 MW/m² 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/m²

2000 cycles at 20 MW/m² on CFC and 15 MW/m² on W)

Experimental critical heat flux of 35 MW/m² on the CFC part – (10m/s, $T_{in}=100$ $T_{out}=127$ $p=3.3$ MPa)



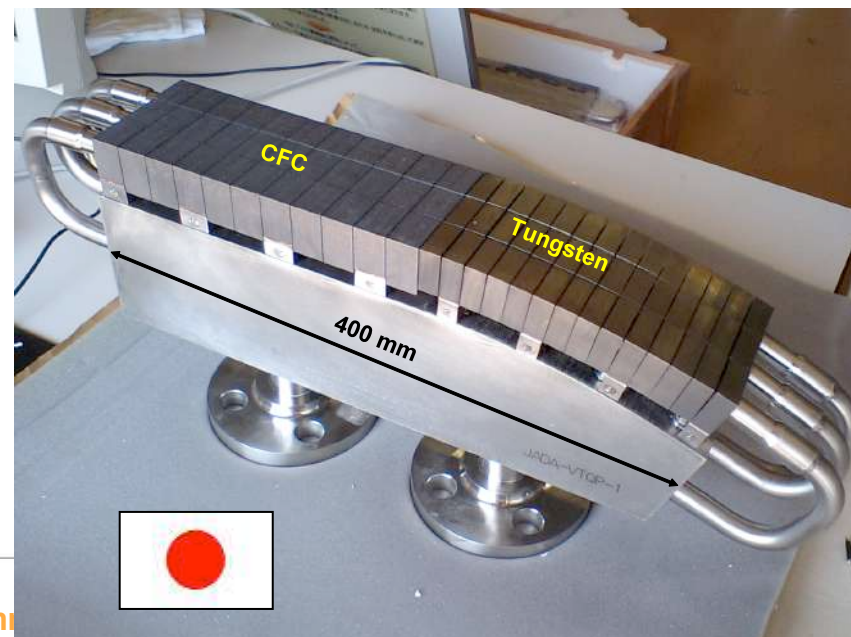
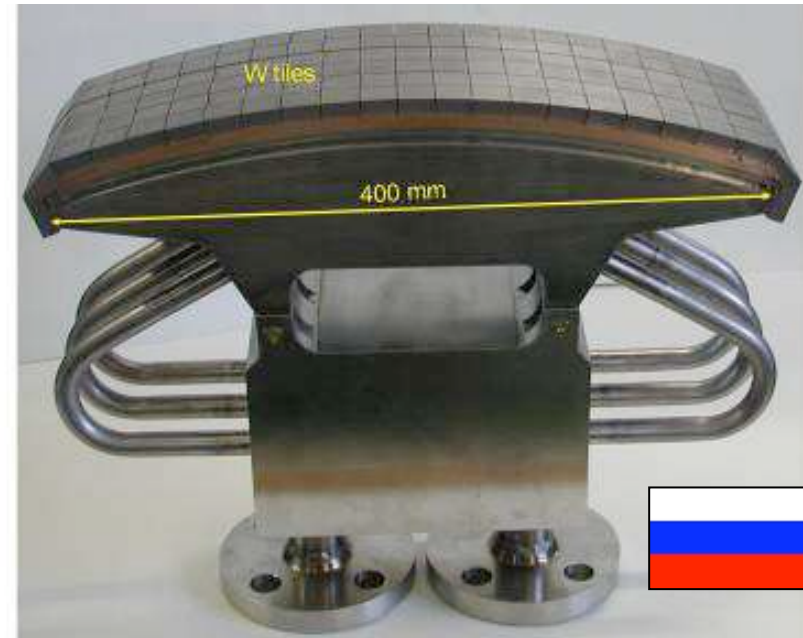
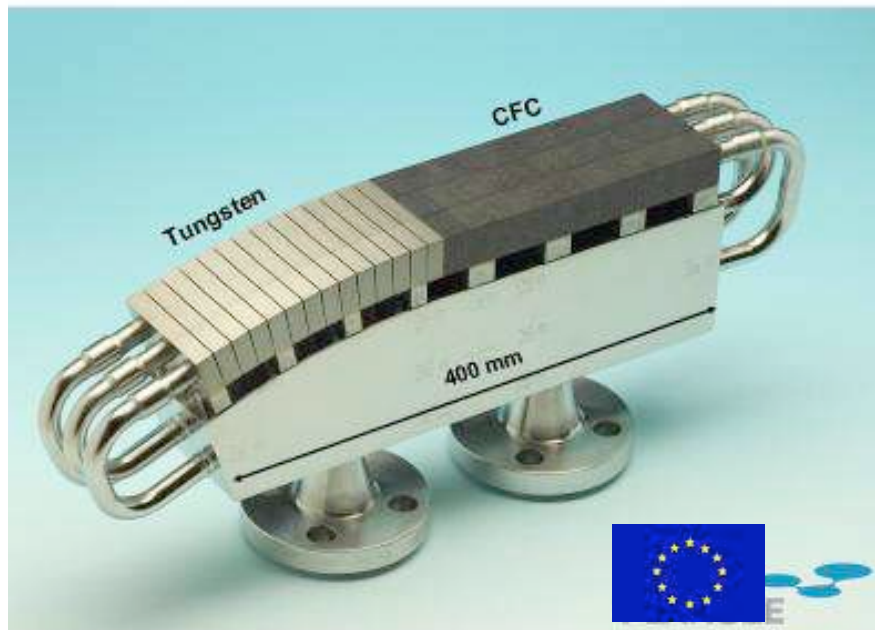
600 mm

- W-Cu by casting
- Cu-CuCrZr by CuInSnNi (STEMET 1108) brazing
- $18.5 \text{ MW/m}^2 \times 1000 \text{ cycles}$

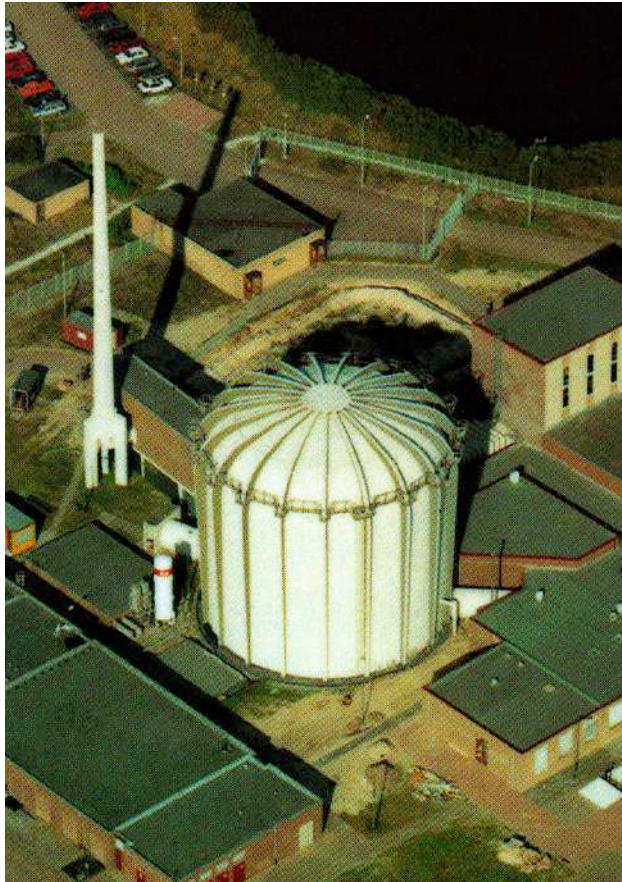


Divertor Qualification Prototypes

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:

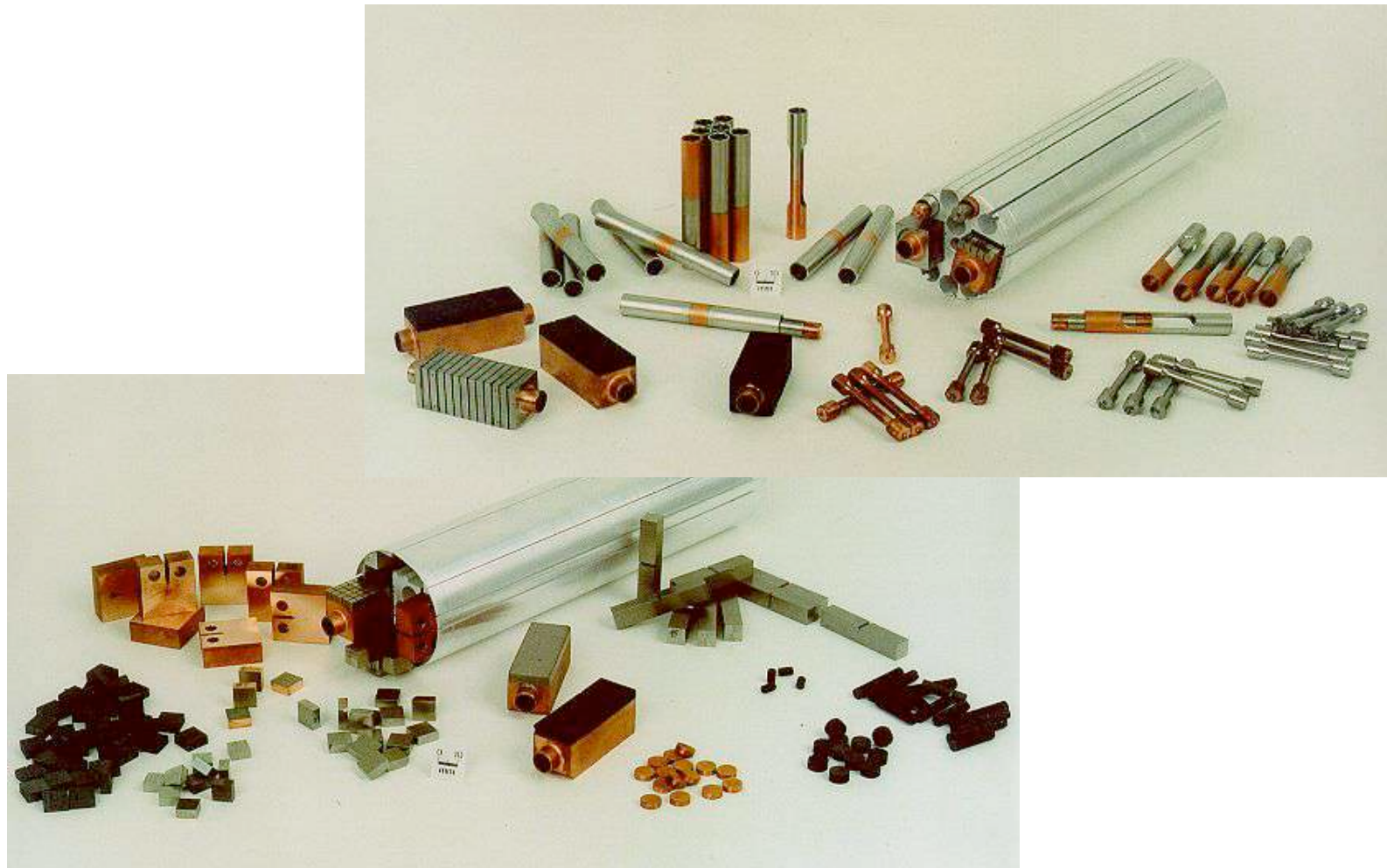
- temperature: 700°C
- target fluence: 0.5 dpa

PARIDE 3:

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

PARIDE 4:

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



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in der Helmholtzgemeinschaft
EURATOM-Association



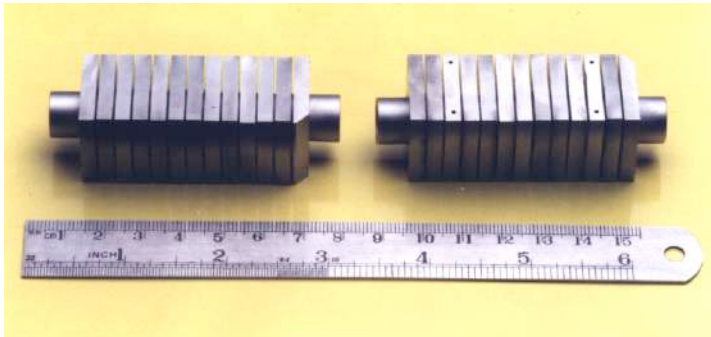
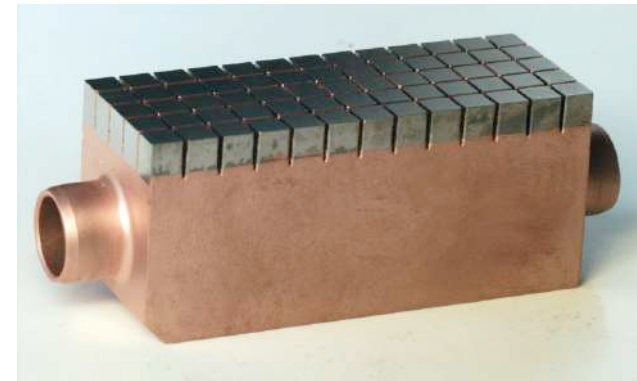
Testing of Tungsten Mock-Ups

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**

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in der Helmholtzgemeinschaft

EURATOM-Association



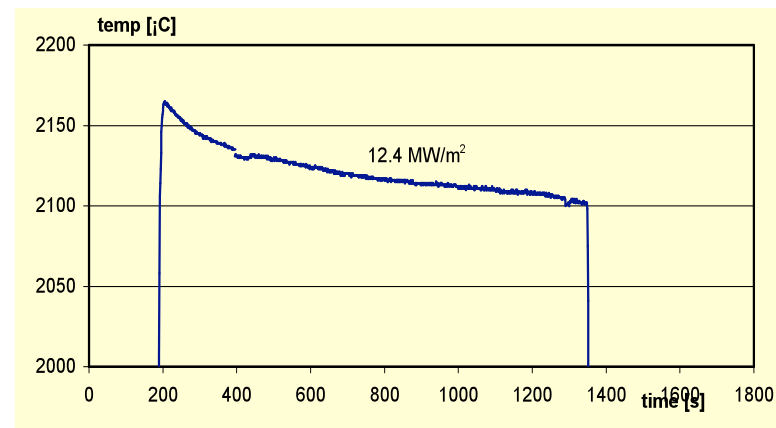
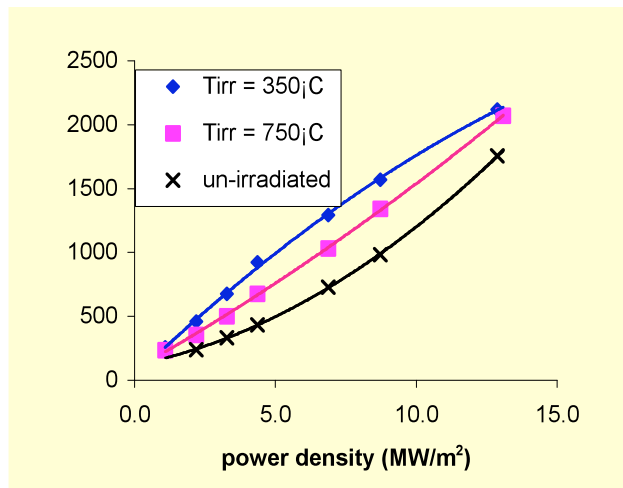
Testing of CFC Mock-Ups

Unirradiated

- 1000 cycles x **19 MW/m²** – 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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EURATOM-Association



- 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

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

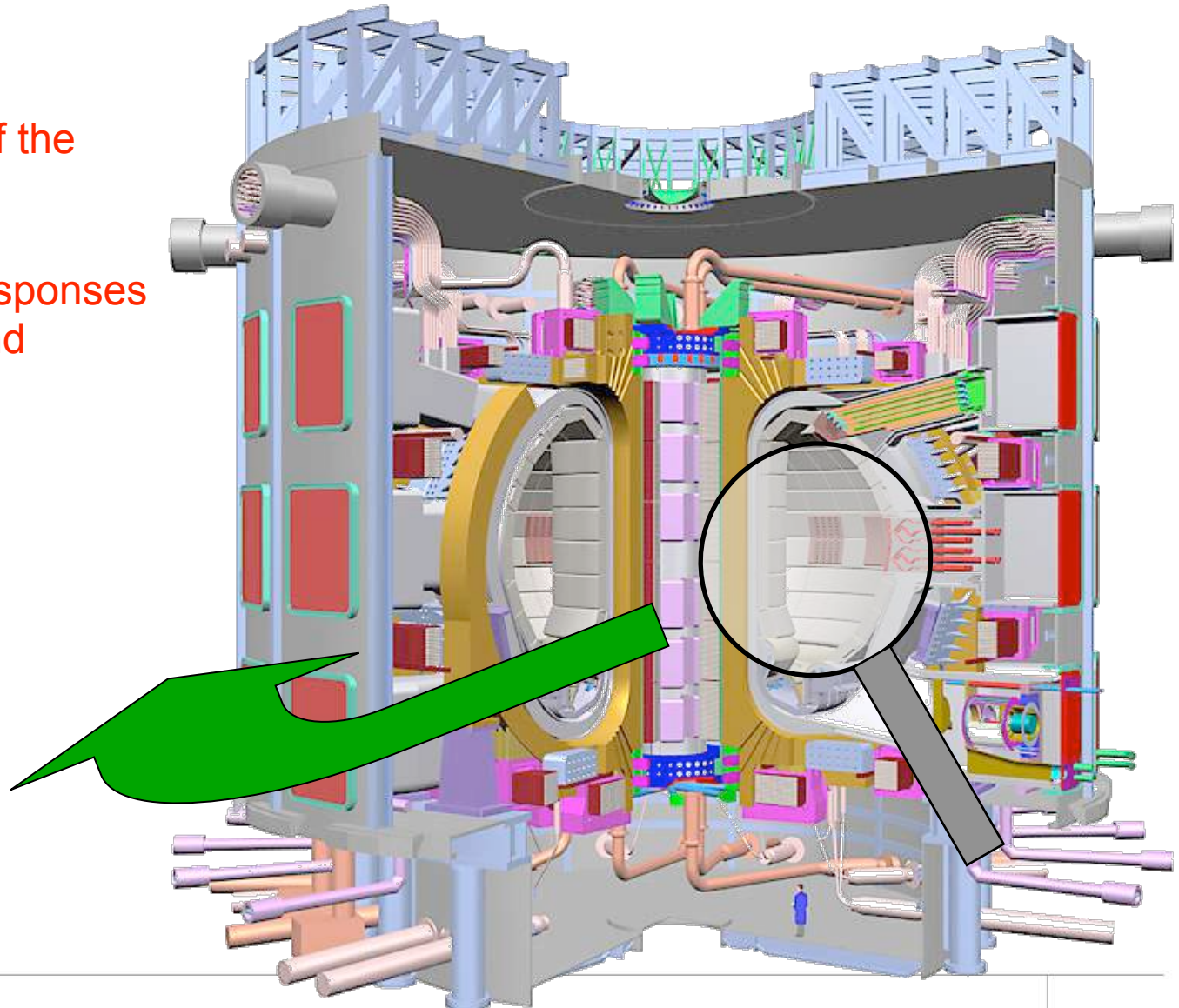
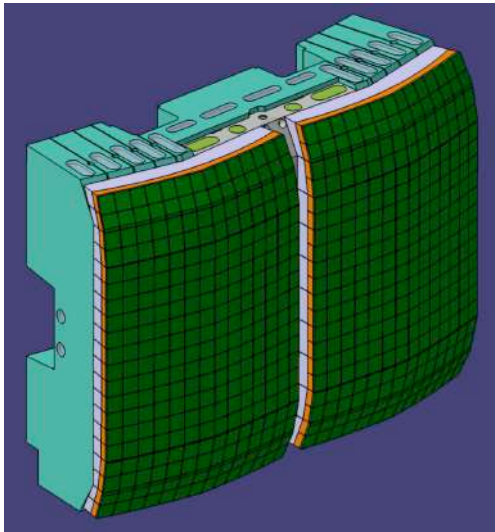
ITER Blanket

Conclusions

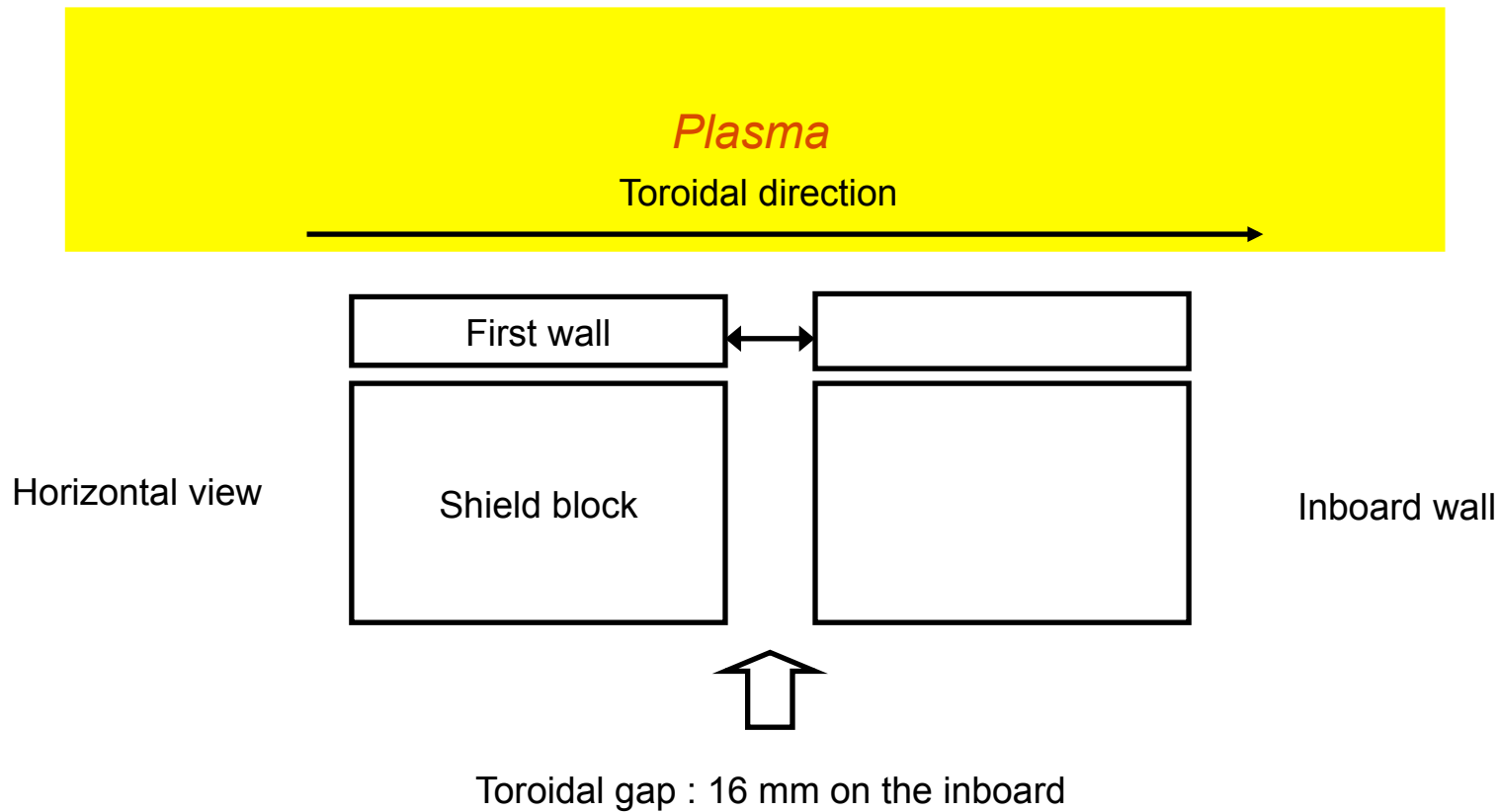
Blanket System

Blanket system main functions :

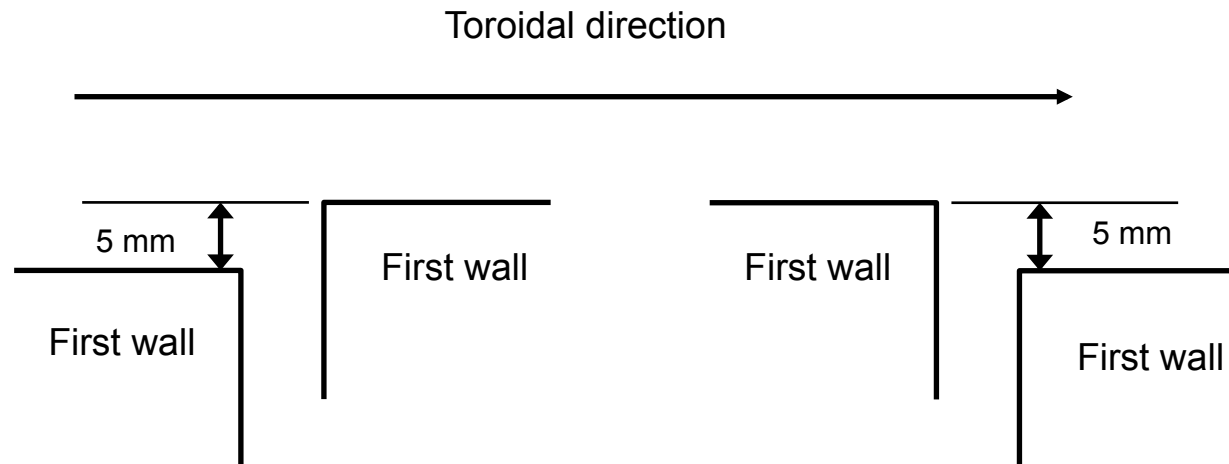
- Exhaust the majority of the plasma power
- Reduce the nuclear responses in the vacuum vessel and superconducting coils



Why shaping is needed ?



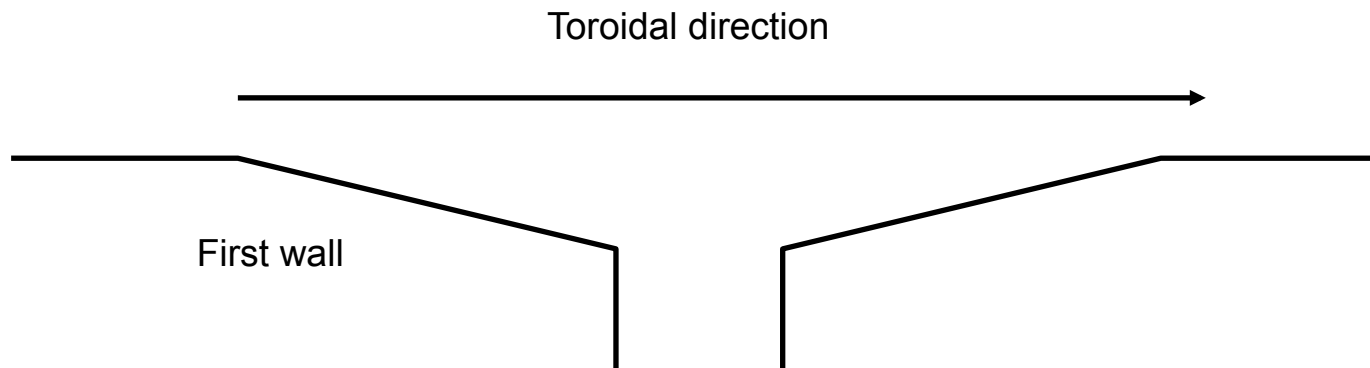
But the two sides have to be considered



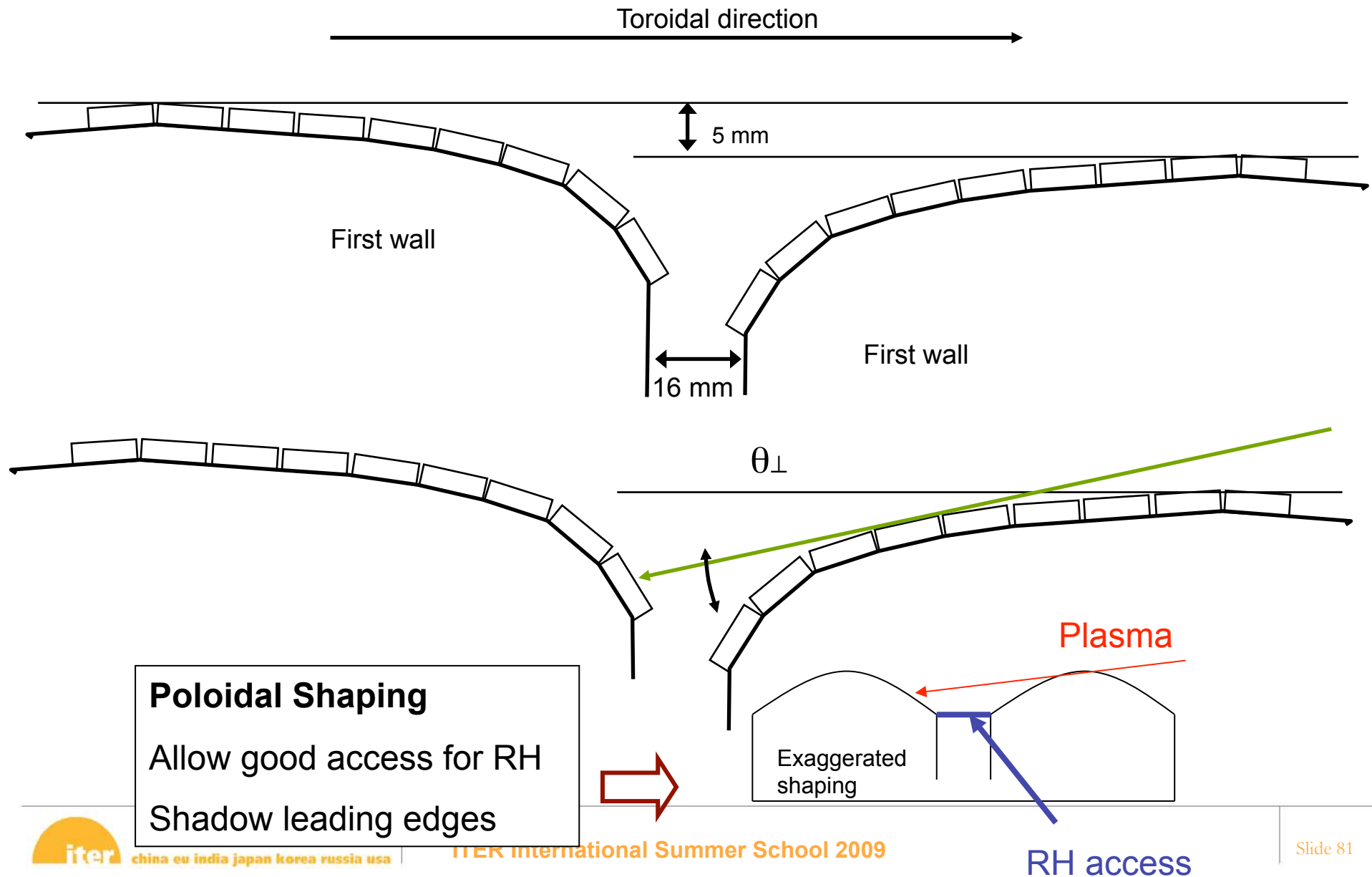
The two situations are equally probable

⇒

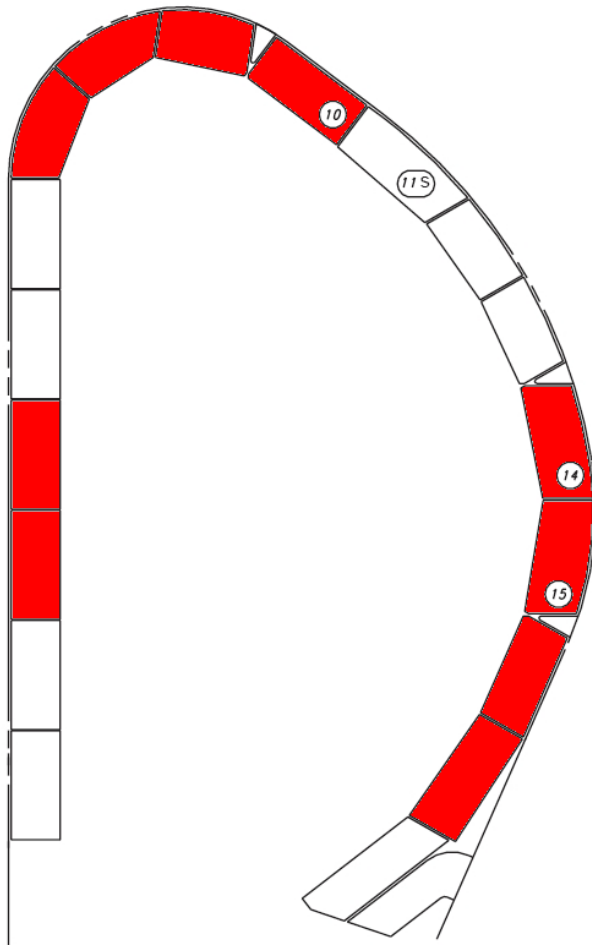
So chamfering on both sides is necessary



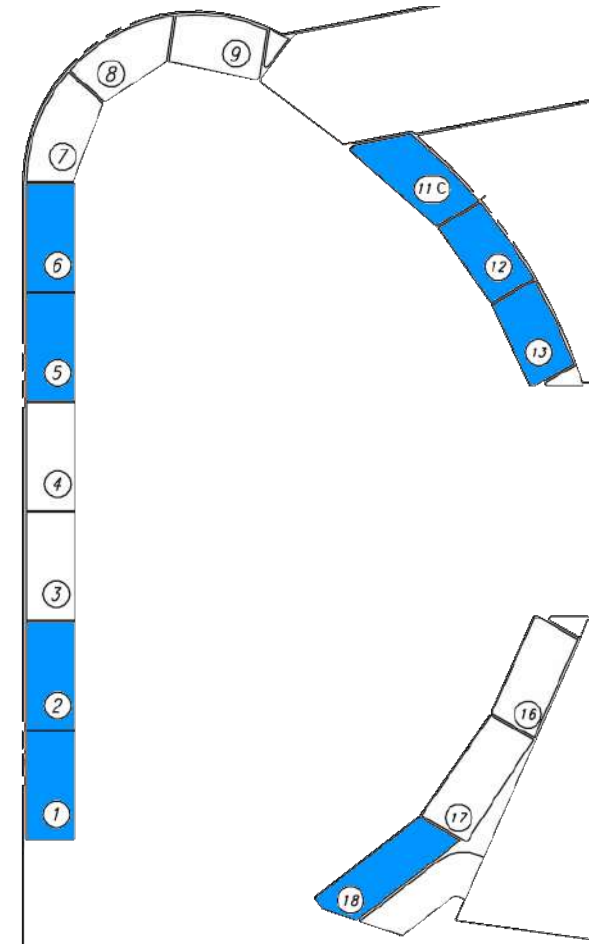
Definition of a key parameter : $\theta_{\perp}$



First Wall Panels: Design heat Flux



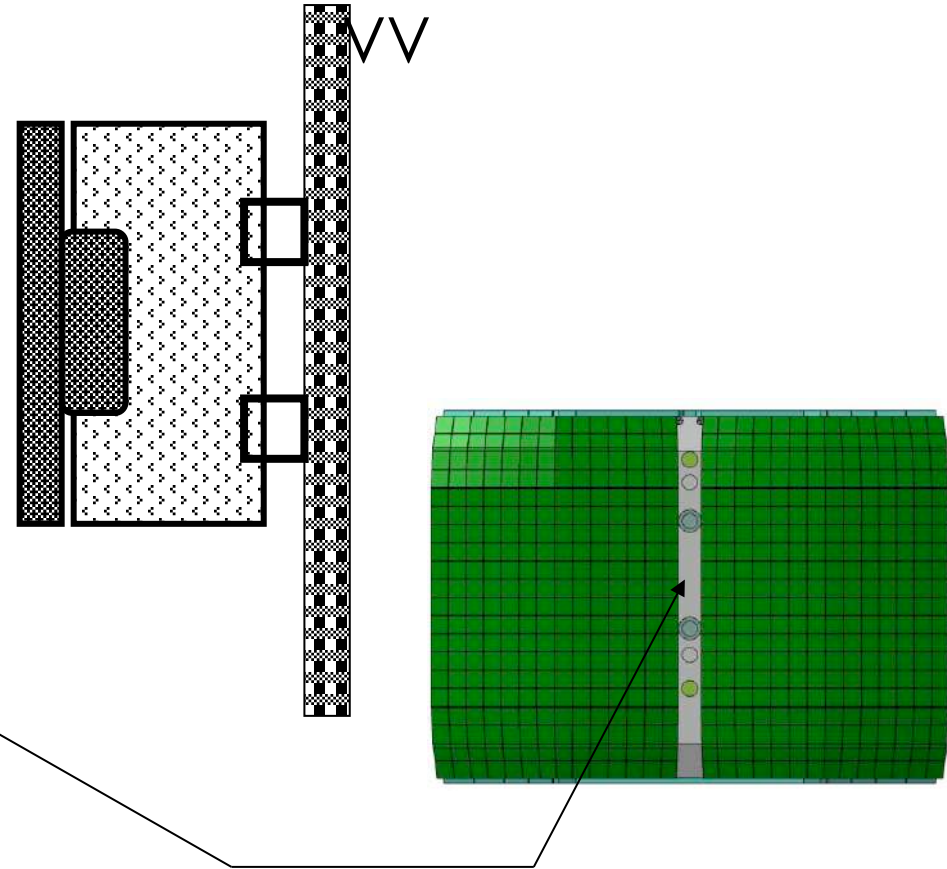
5 MW/m²



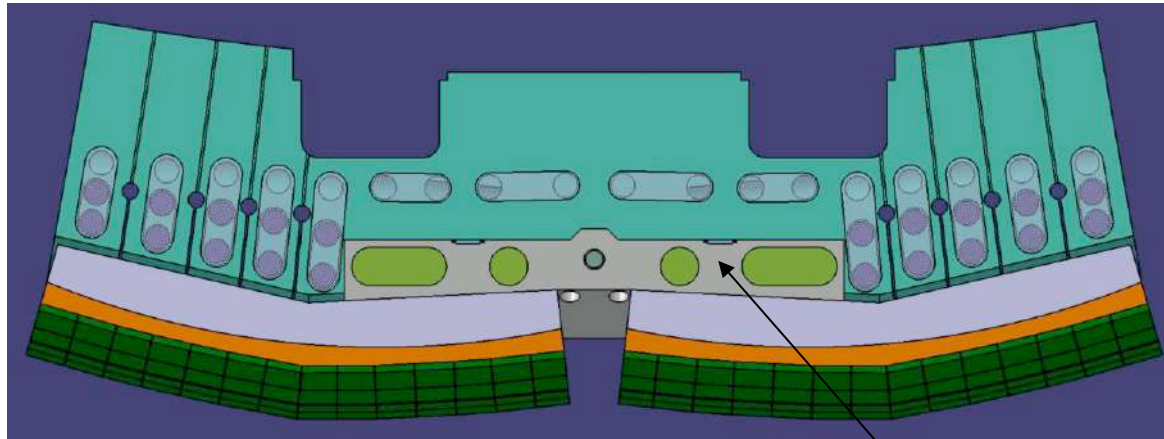
1 MW/m²

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

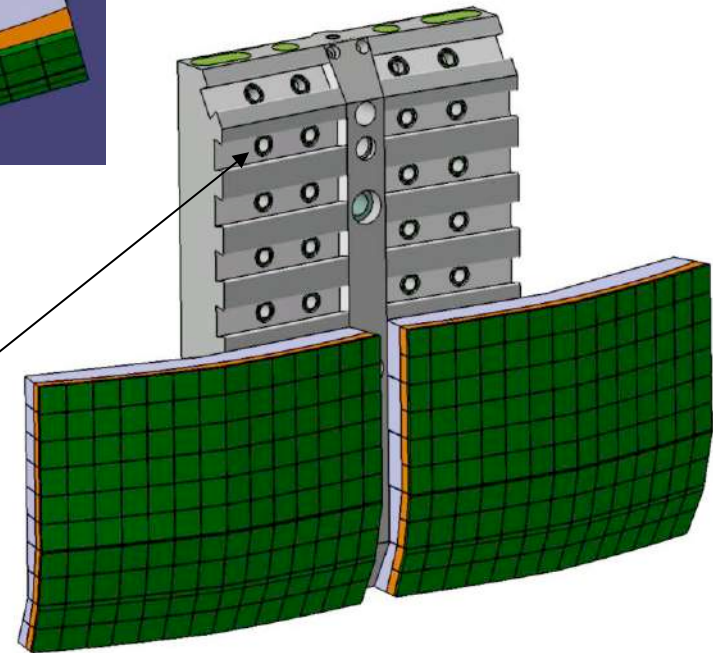


First wall construction

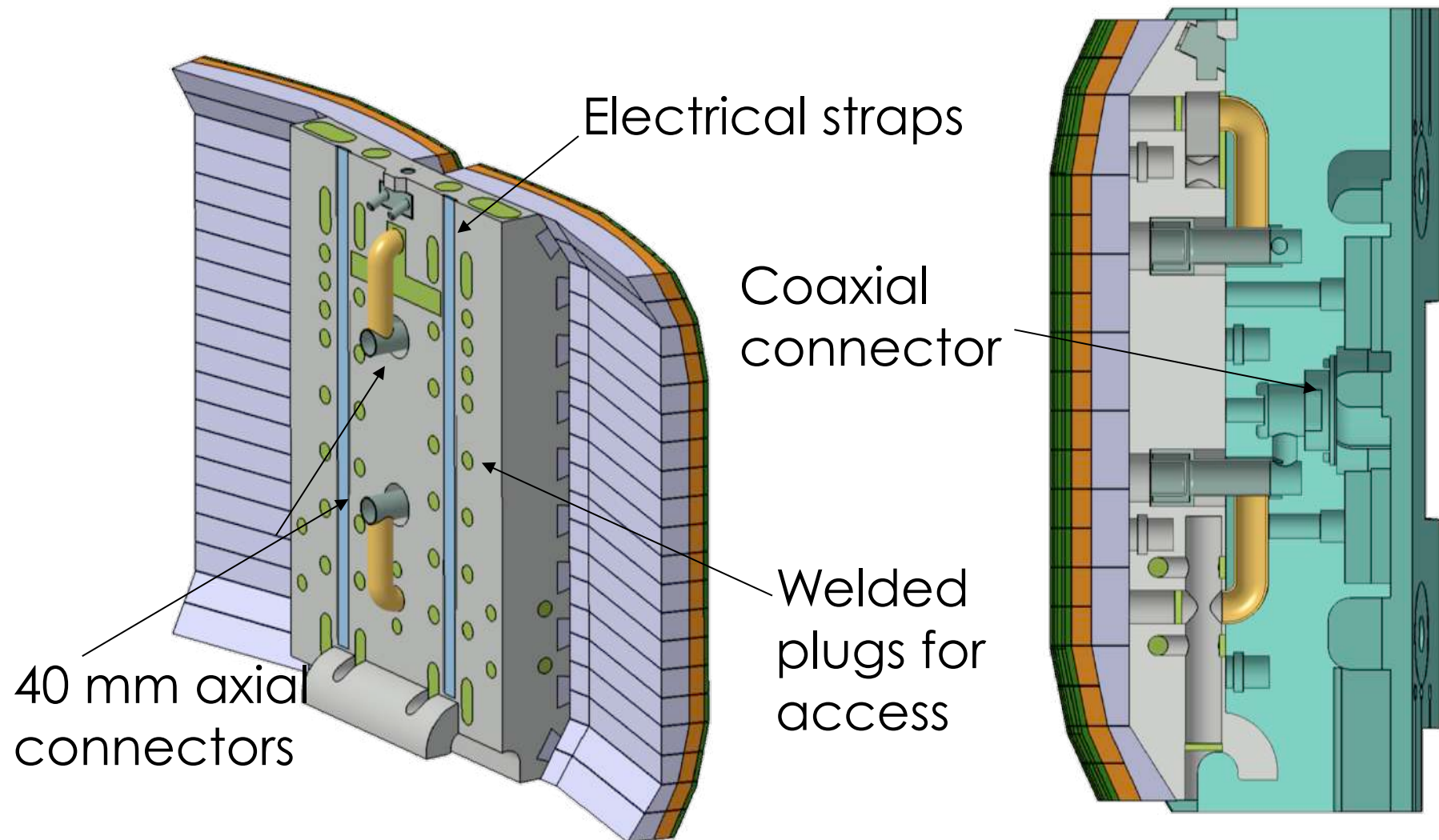


Increased beam width &
reduced thickness

Dove tail Mechanical
assembly

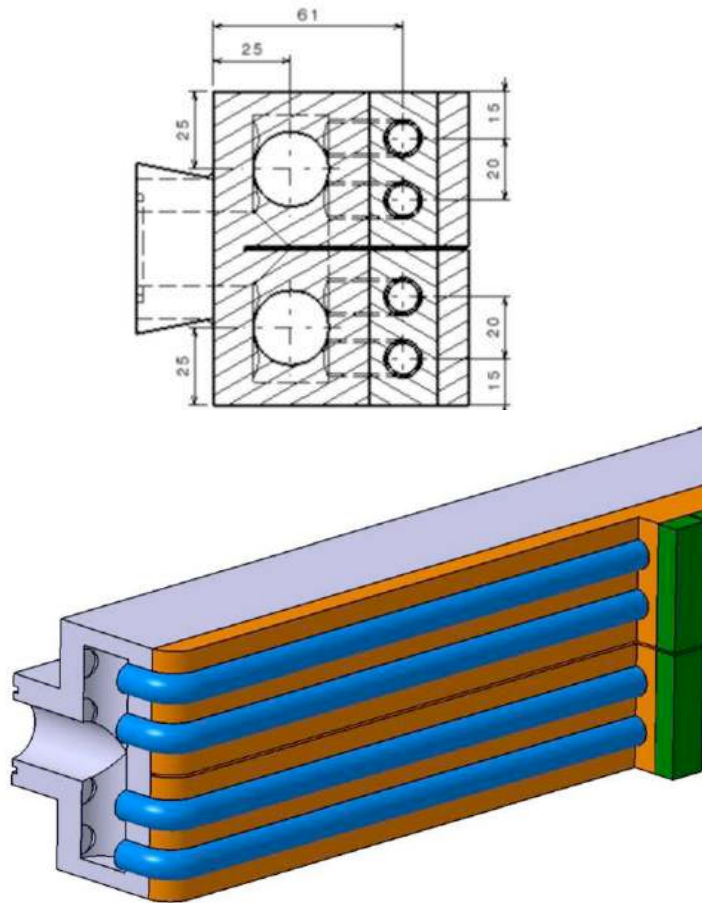


FW beam construction

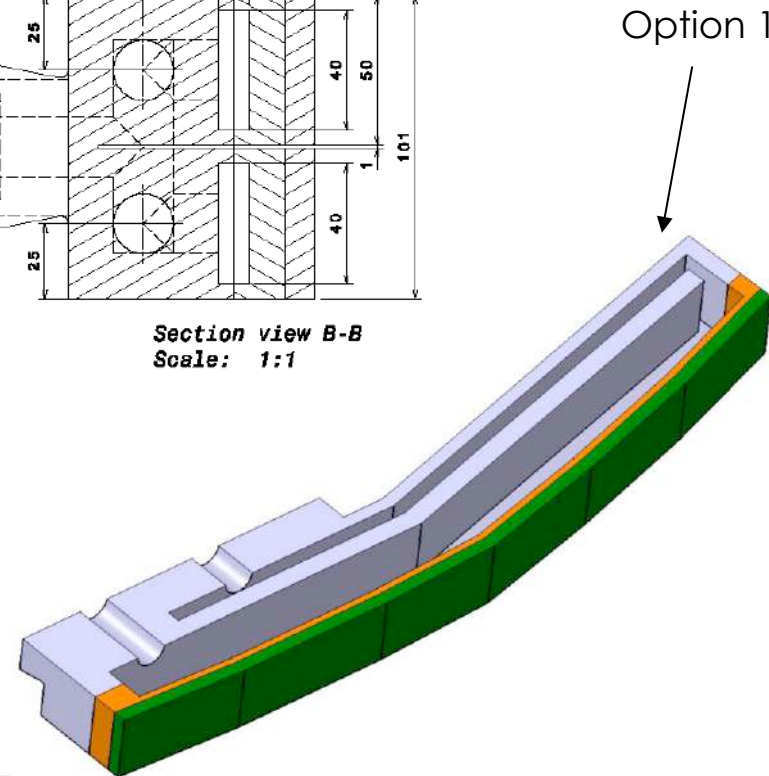
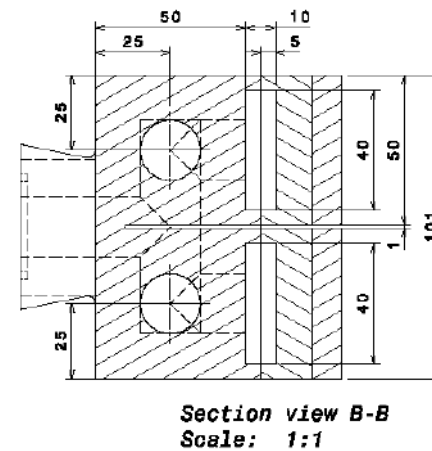


Individual finger construction

Low heat flux 1.0 MW/m^2

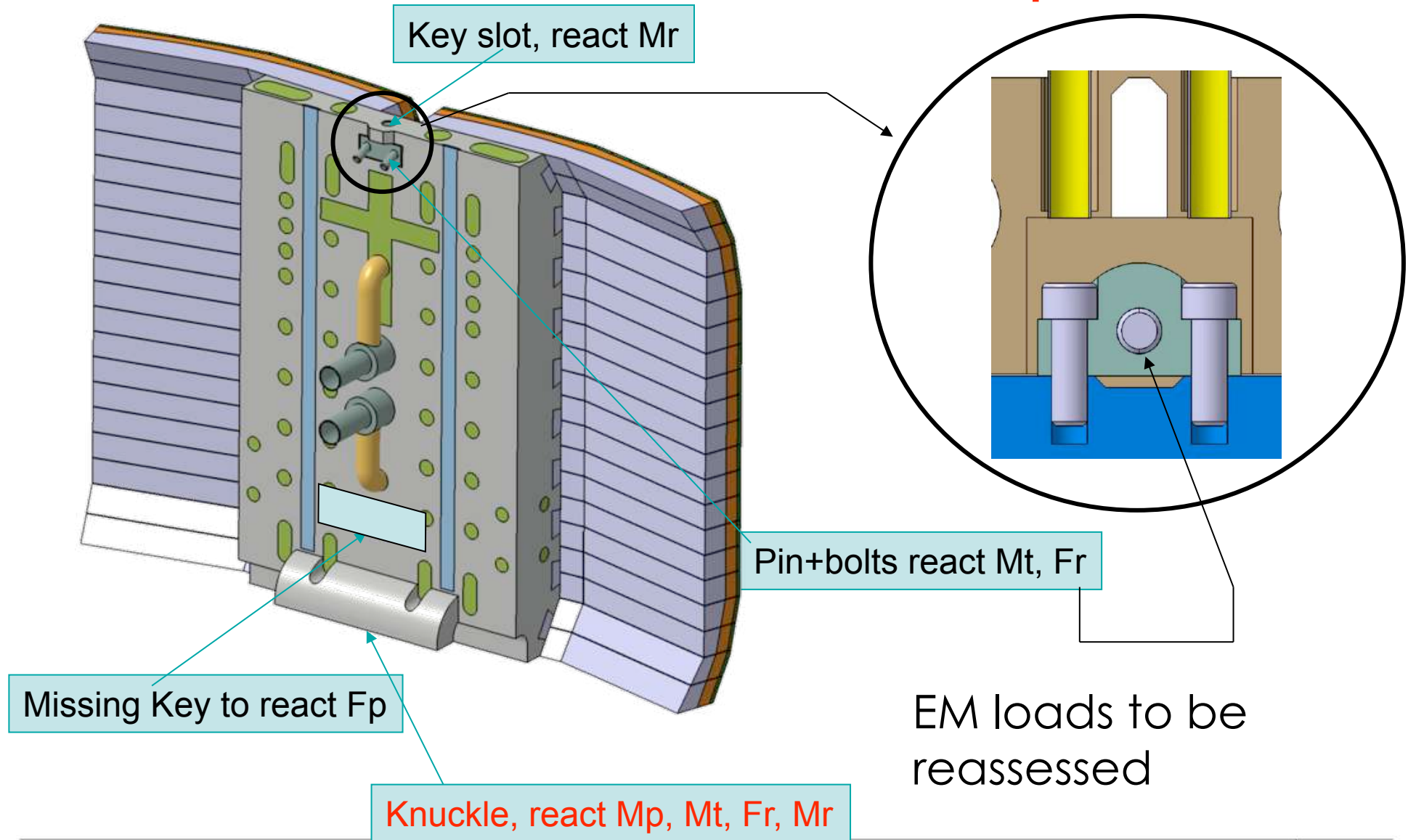


High heat flux 5 MW/m^2

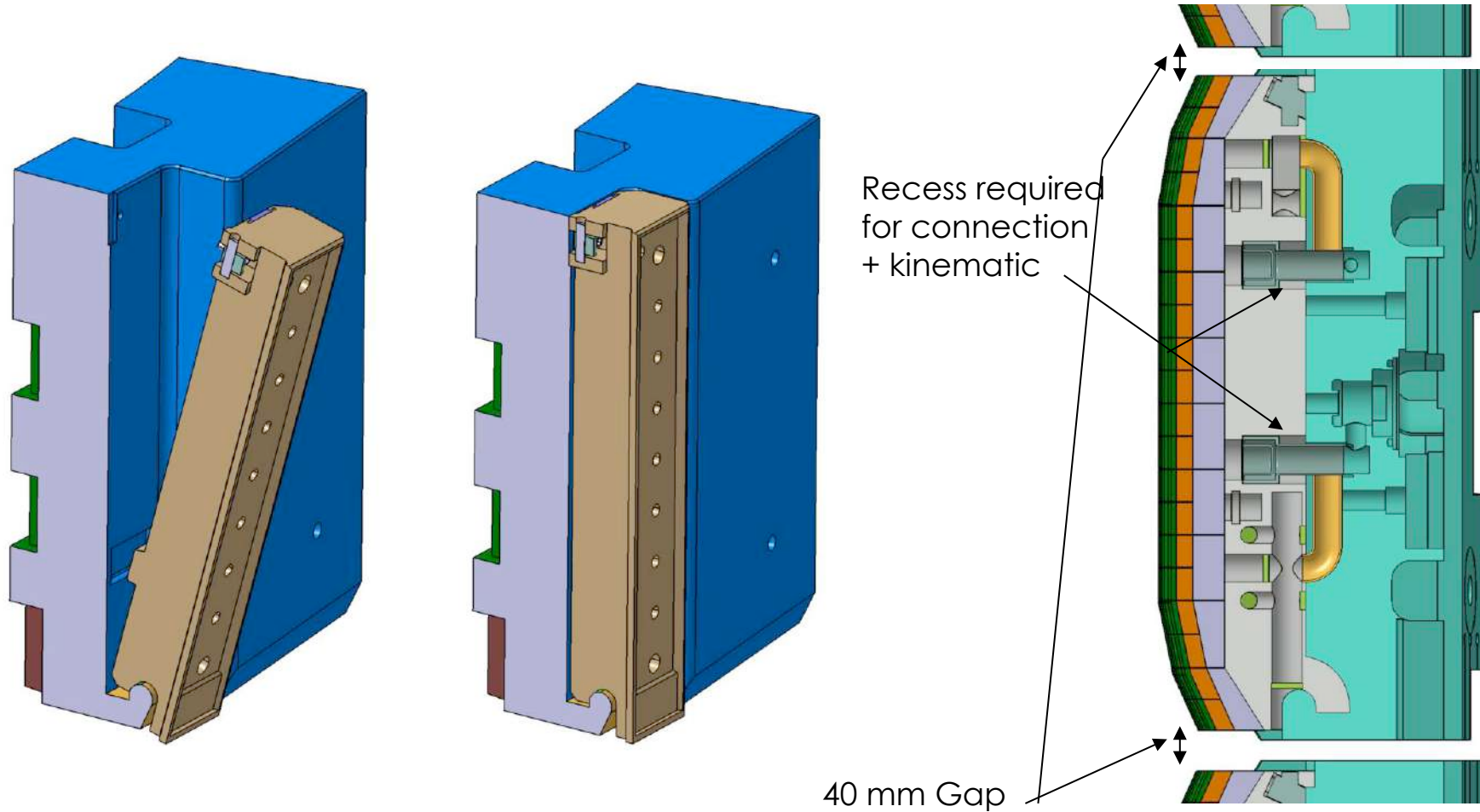


First wall attachment

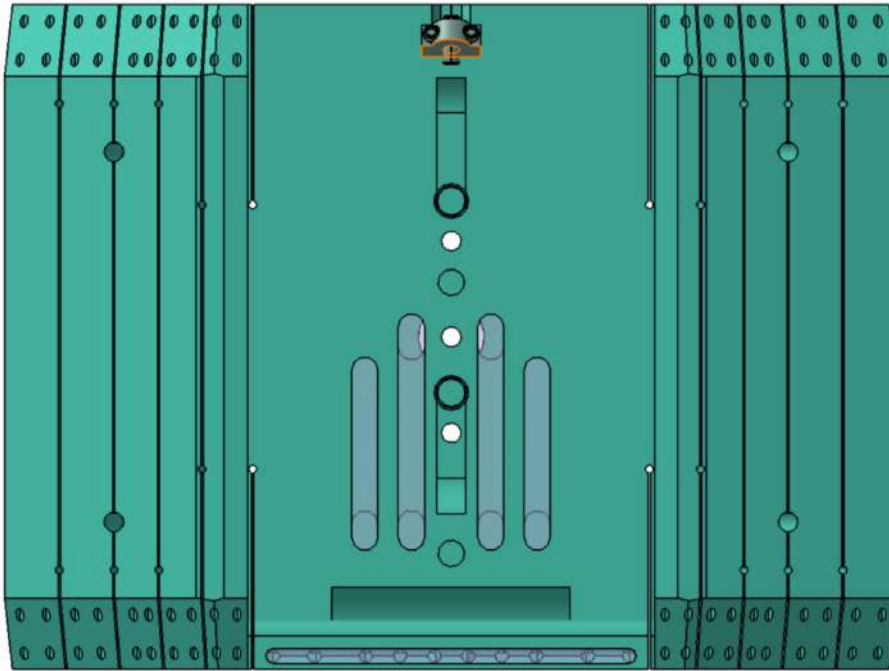
Knuckle concept



Knuckle kinematic



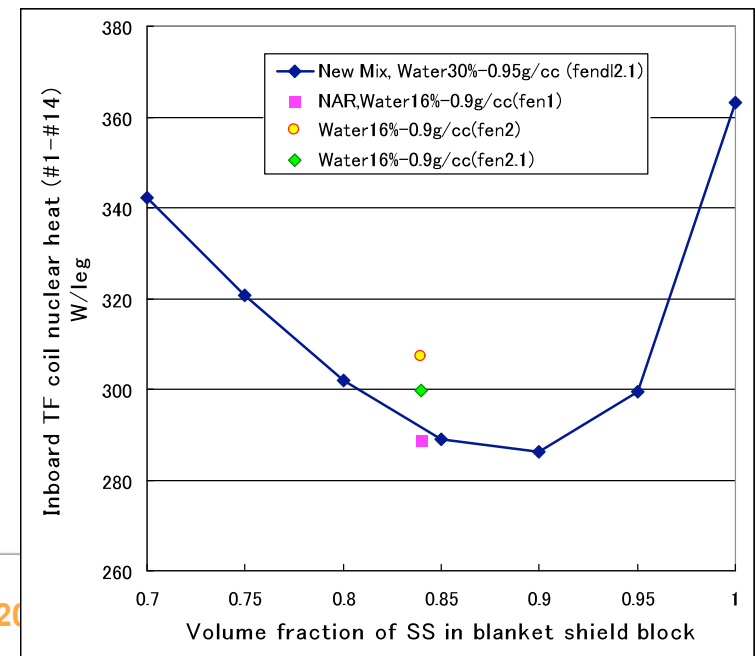
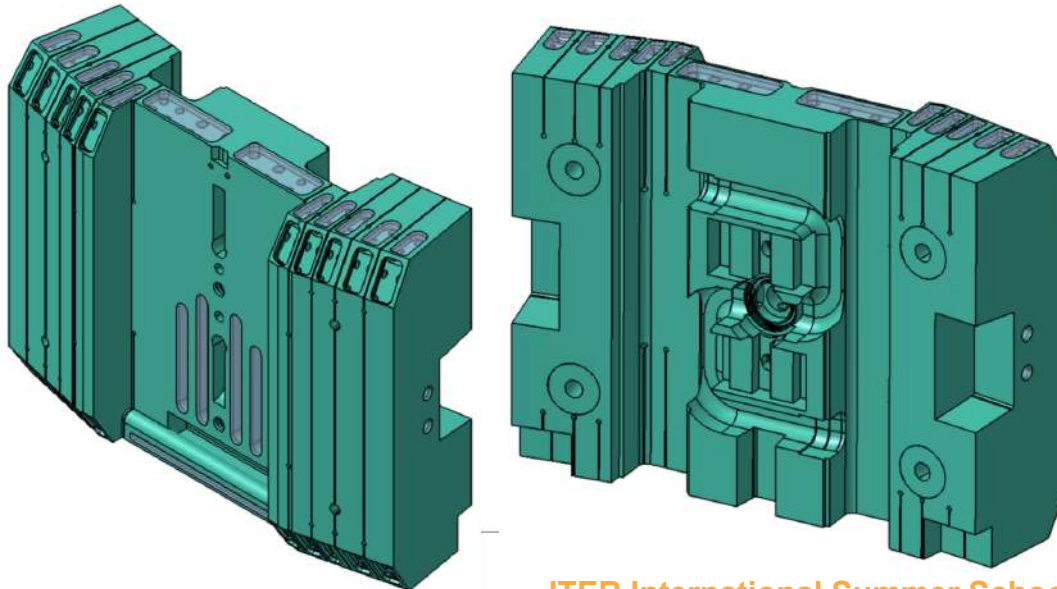
Shield Block Design



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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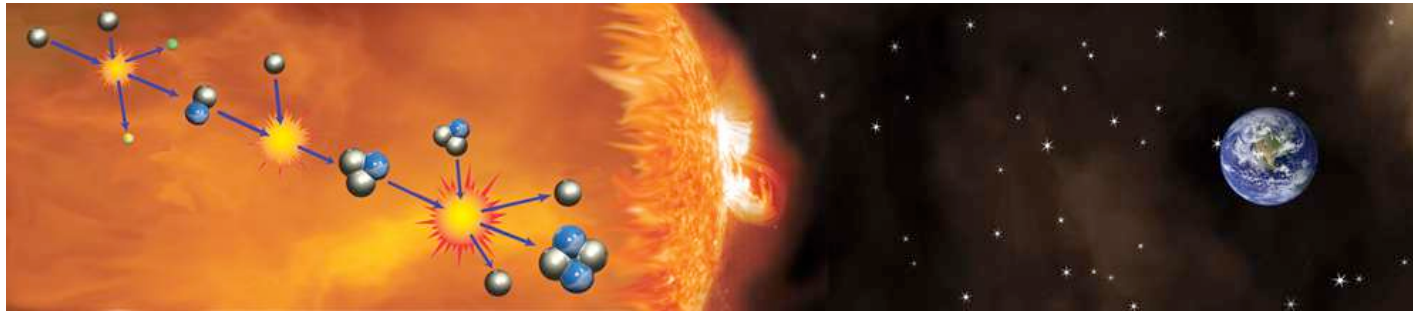
ITER Blanket

Conclusions

Conclusions

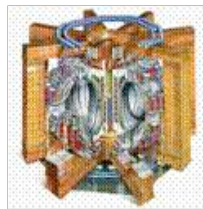
- The “fitness for purpose” of the armour to heat sink joining technologies was demonstrated on the basis of a “design by experiment” approach
- The available technologies are able to meet or even exceed the ITER design requirements
- The manufacturing of plasma facing components requires the mastering 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...



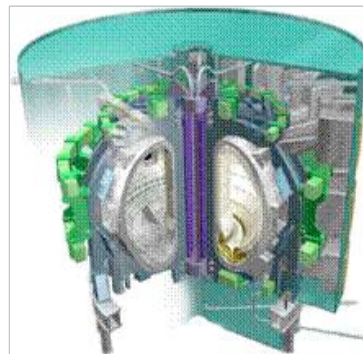
Tore Supra

25 m³
~ 0 MW_{th}



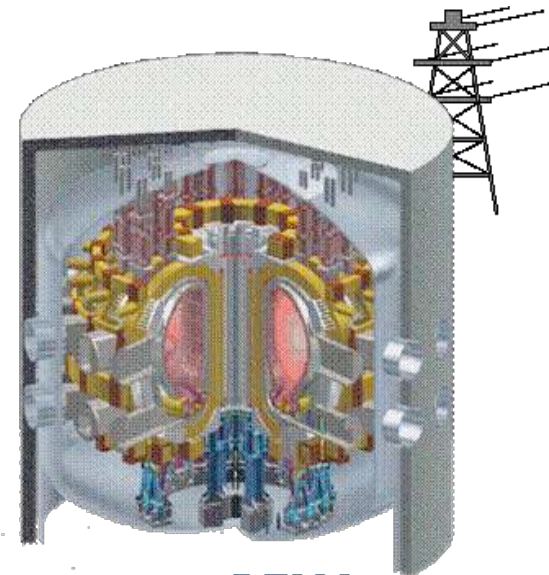
JET

80 m³
~16 MW_{th}



ITER

800 m³
~ 500 MW_{th}



DEMO

~ 1000 - 3500 m³
~ 2000 - 4000 MW_{th}

- Dominant self heating ----->