STUDIES OF IN-VESSEL COMPONENT INTEGRATION FOR A HELIUM-COOLED DEMO FUSION REACTOR

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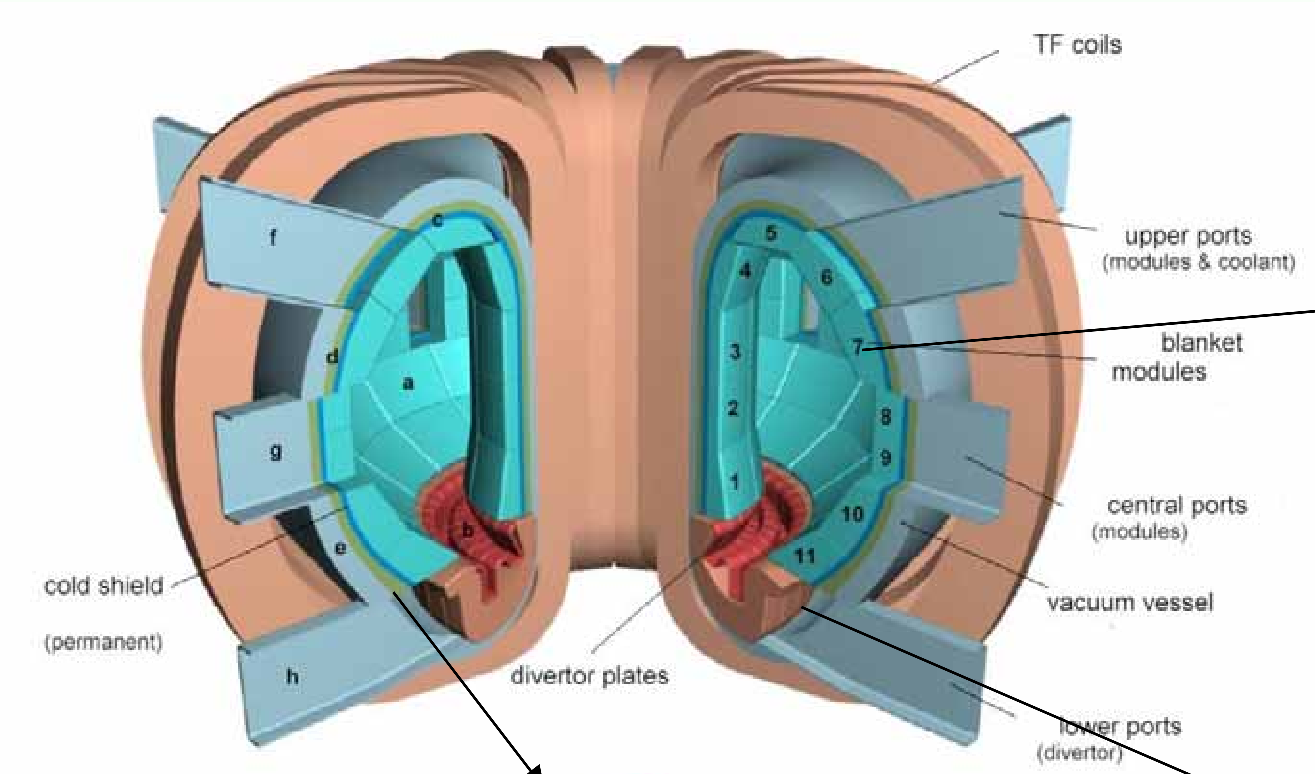
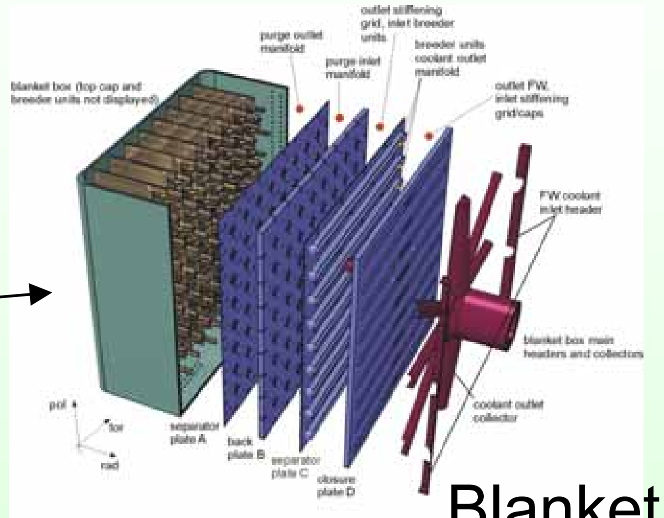
Outline

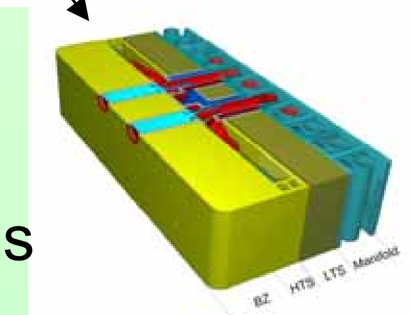
- Introduction
- Blanket and divertors for DEMO and the first generation of FPP
- Reactor Integration
- Conclusions

Integration of in vessel components in the fusion power plant

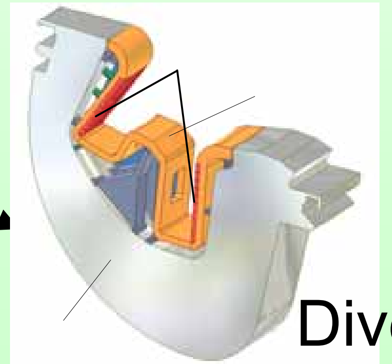




Blanket



Shield
Manifolds

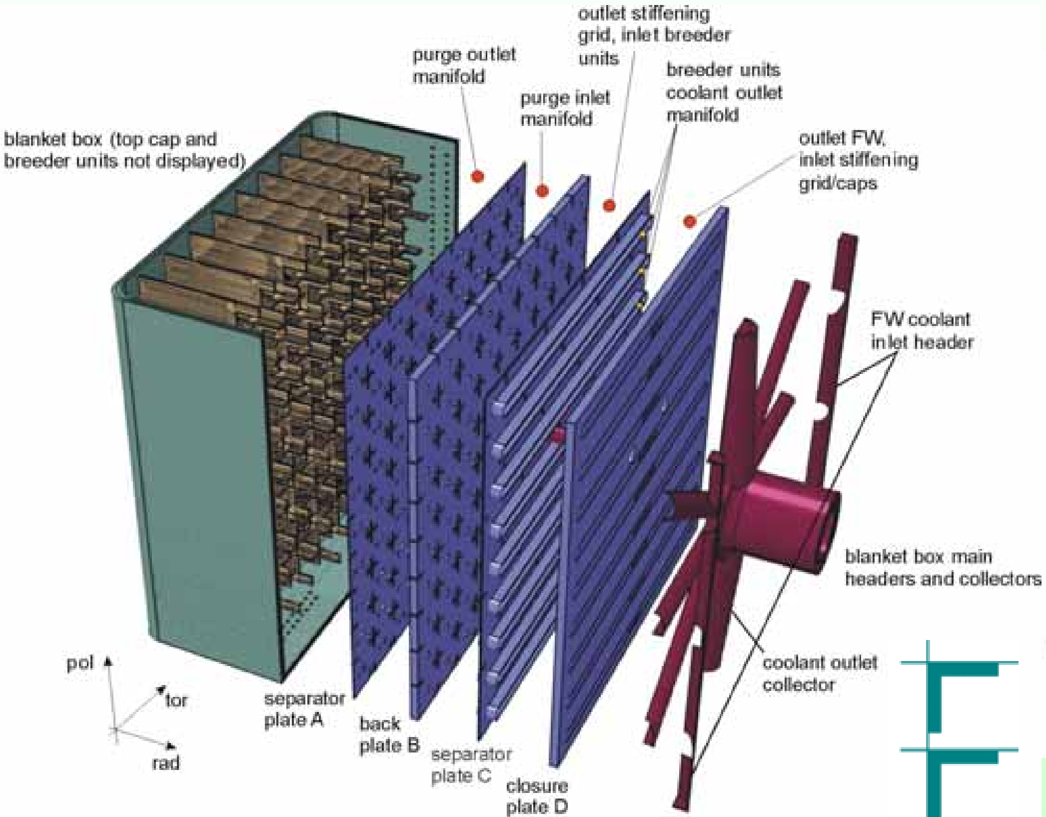


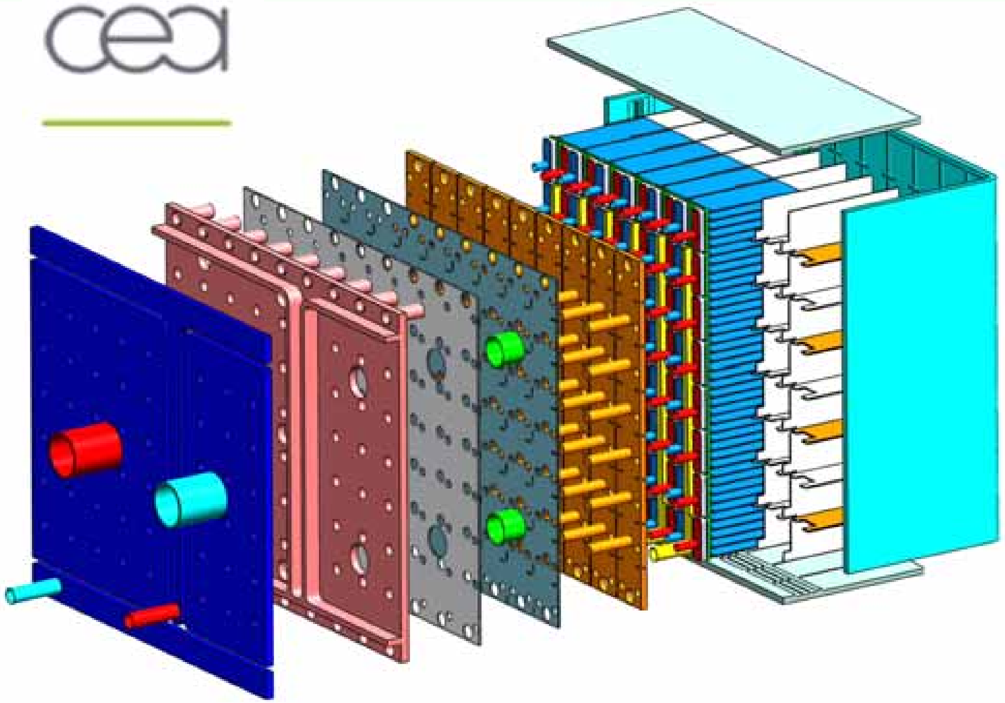
Divertor

Power Plant Concepts

	Model B	Model AB	Model C
Blanket Type	HCPB (Solid Breeder)	HCLL (Stagnant liquid)	DCLL (Dual Coolant)
Structural material	EUROFER	EUROFER	EUROFER (ODS in FW)
BrederMaterial	$Li4SiO4 – Li2TiO3$	Pb/Li_{eut}	Pb/Li_{eut}
Multiplier	Beryllium	"	"
Coolant	Helium	Helium	Helium (40 %) Pb/Li_{eut} (60 %)
Divertor type	He-cooled	He-coooled	He-cooled
Coolant	Helium	Helium	Helium
Structural material	W-alloy / ODS steel	W-alloy / ODS steel	W-alloy / ODS steel

Design of the HCPB and HCLL Blanket for DEMO

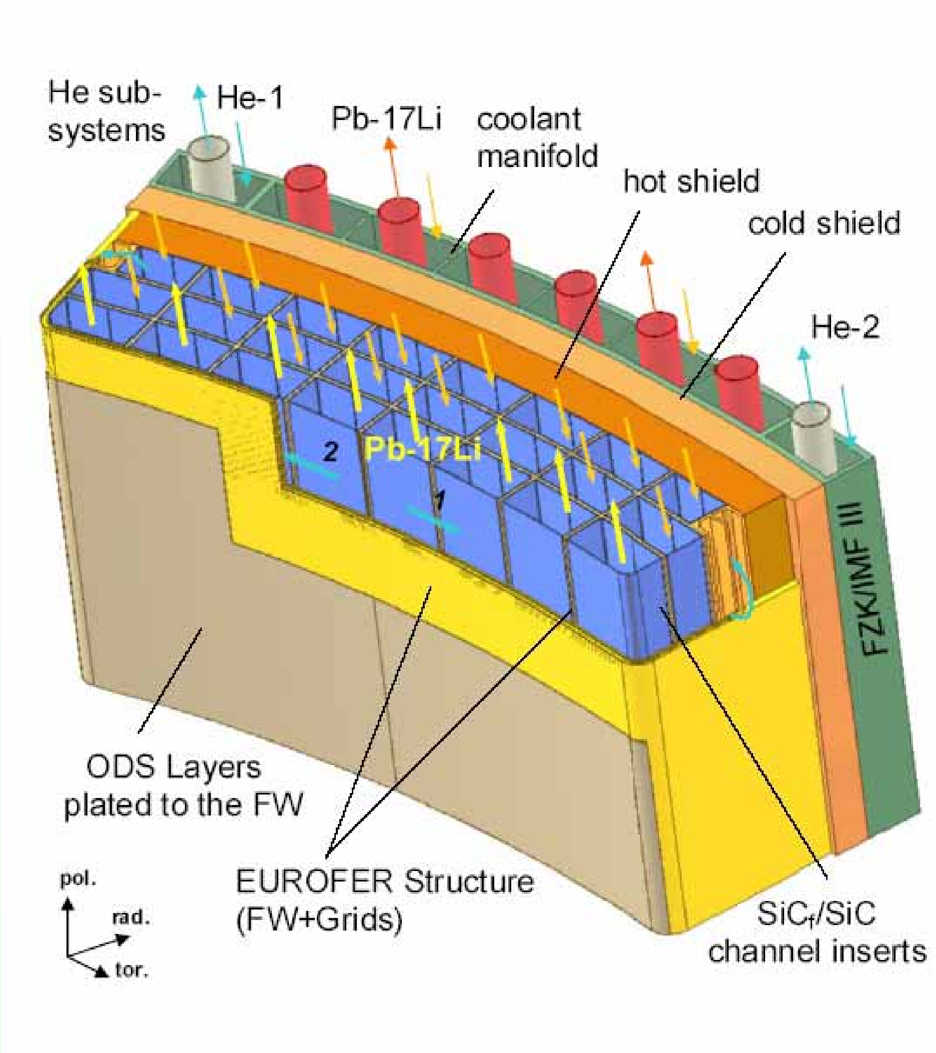


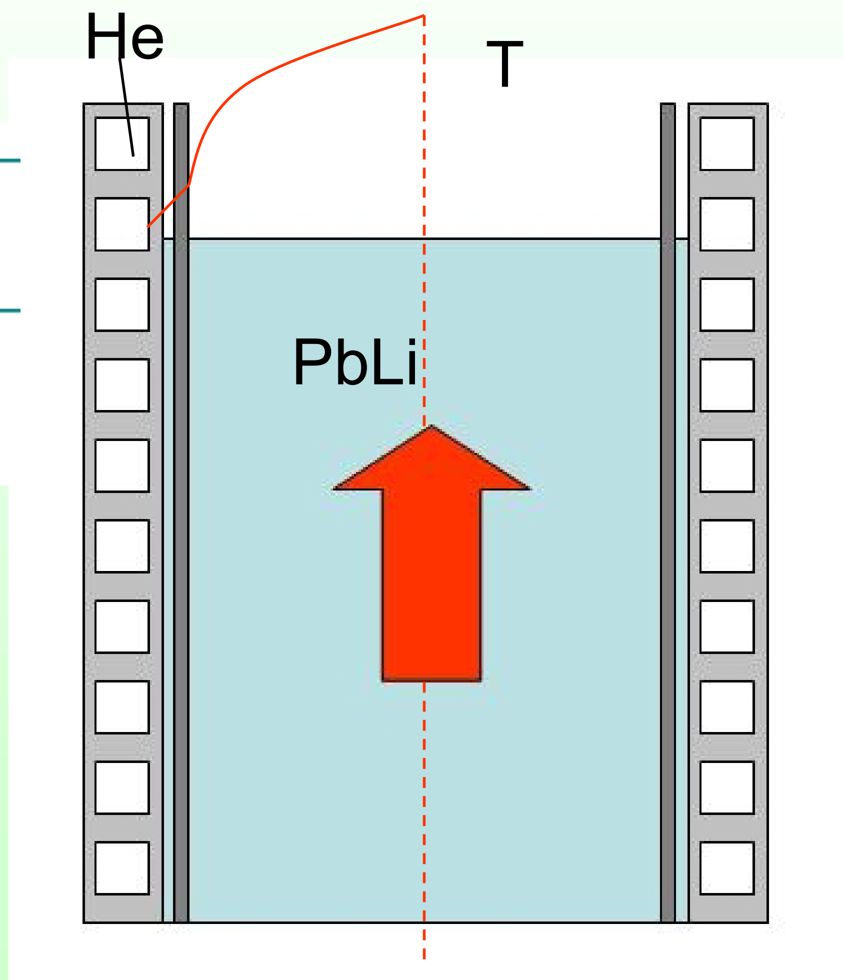


HCPB: Helium Cooled Pebble Bed (FZK)

HCLL: Helium Cooled Lithium Lead (CEA)

FZK Dual Coolant

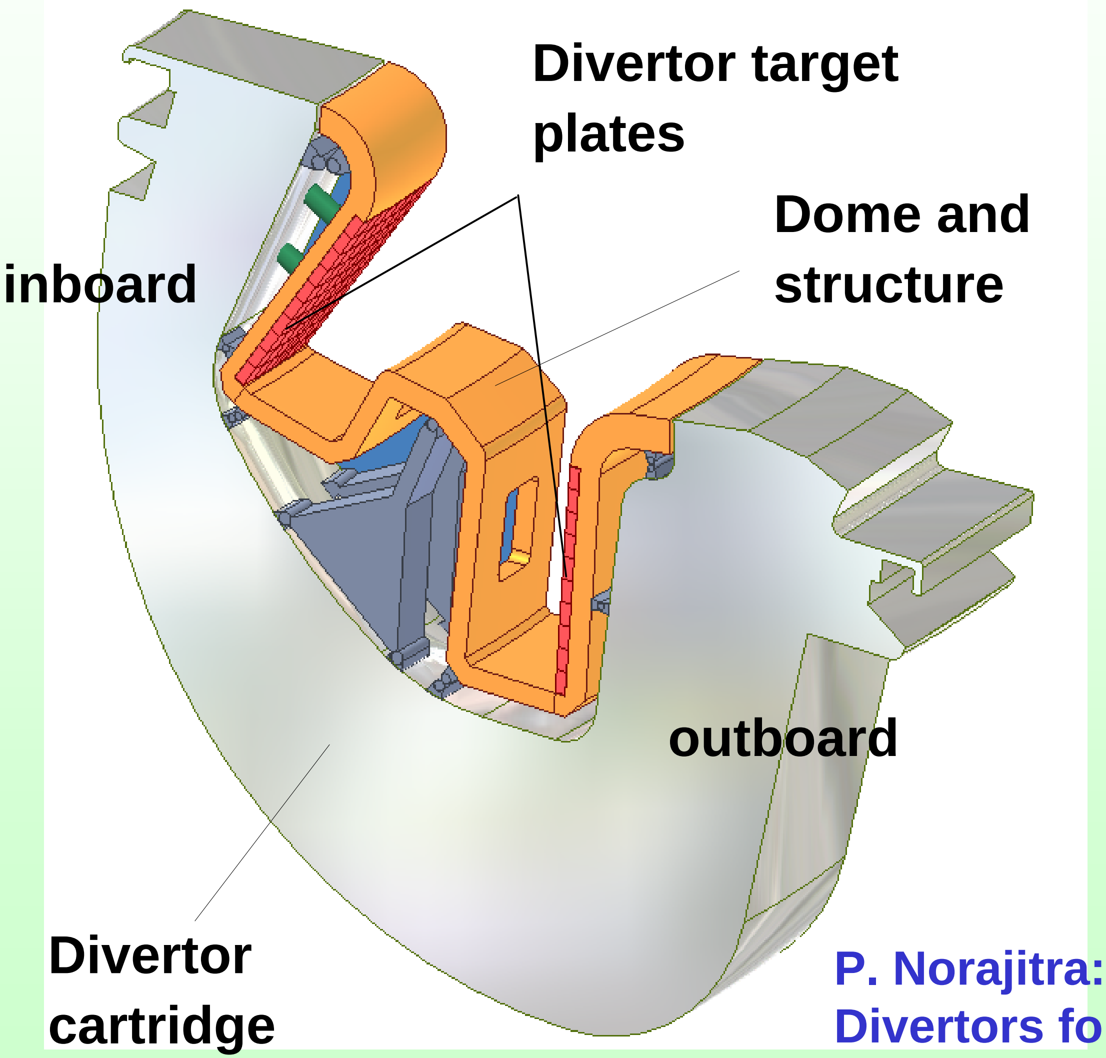




Blanket Performances and coolant requirements

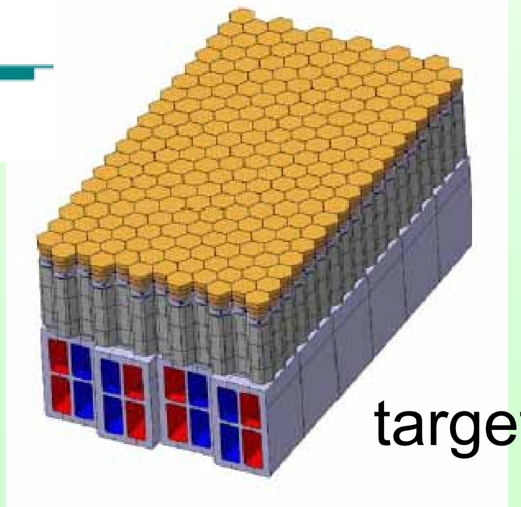
	HCPB	HCLL	DCLL
Electricity power	1.3 GW	1.5 GW	1.5 GW
Blanket Thermal Power	4.3 GW	4.5 GW	3.5 GW
Coolant temperatures	He: 300-500°C	He: 300-500°C	He: 300-480°C PbLi: 480-700°C
Dimensions (major rad.)	8.6 m	9.6 m	7.5 m
Coolant mass flow	He: 4.9 t/s	He: 5.1 t/s	He: 1.5 t/s PbLi: 46 t/s
Coolant flow area (hot leg): (~ 75 m/s for He) (~ 1 m/s for PbLi)	He: 13.1 m^2 (5.6 %)	He: 13.6 m^2 (4.7 %)	He: 4.6 m^2 (2.3 %) PbLi: 4.2 m^2 (2.6 %) tot: 8.3 m^2 (4.9 %)

He cooled Divertor



Thermal Load:
- heat peak: 10 MW/m2
- average heat: 5 MW/m2

He Coolant:
- temperatures: 543-700°C
- pressure: 10 MPa



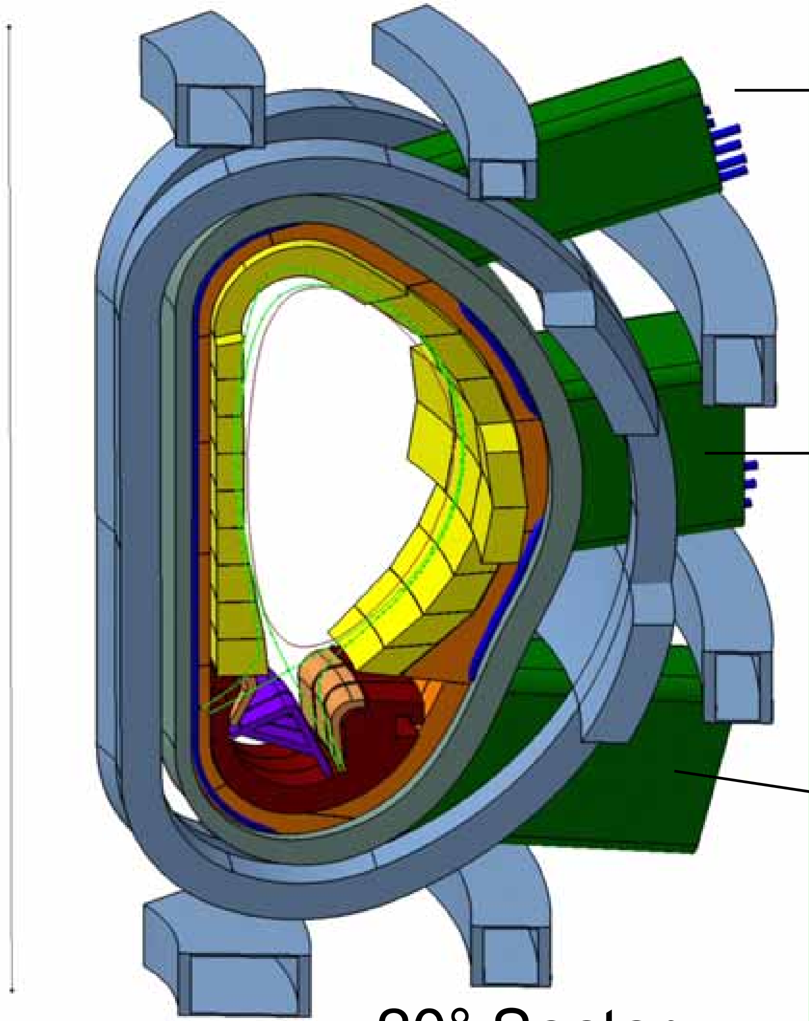
target plate

P. Norajitra: "Development of Helium-Cooled Divertors for Fusion Power Plants", this conference.

Scheduled component replacement

- The lifetime of the blanket is determined by the neutron damage in the structural material of the FW (EUROFER from 75 to 150 dpa, that means ~3 -> 6 FPY at 2.4 MW/m2 neutron wall load).
- Divertors will be limited by erosion of the target plates, with an envisaged lifetime of ~2 FPY
- Part of shield and manifolds can be designed as permanent o semi-permanent components
- For unscheduled maintenances these permanent components should be designed for RH.
- The replacement strategy should assure a power plant plant availability >70%.

"Transporter" concept for a Fusion Power Plant



20° Sector

Upper Port: 4 ports for RH of the 54 "Blanket Cassettes"

Equatorial port: 4 ports for the RH of the equatorial IB and OB Blanket modules

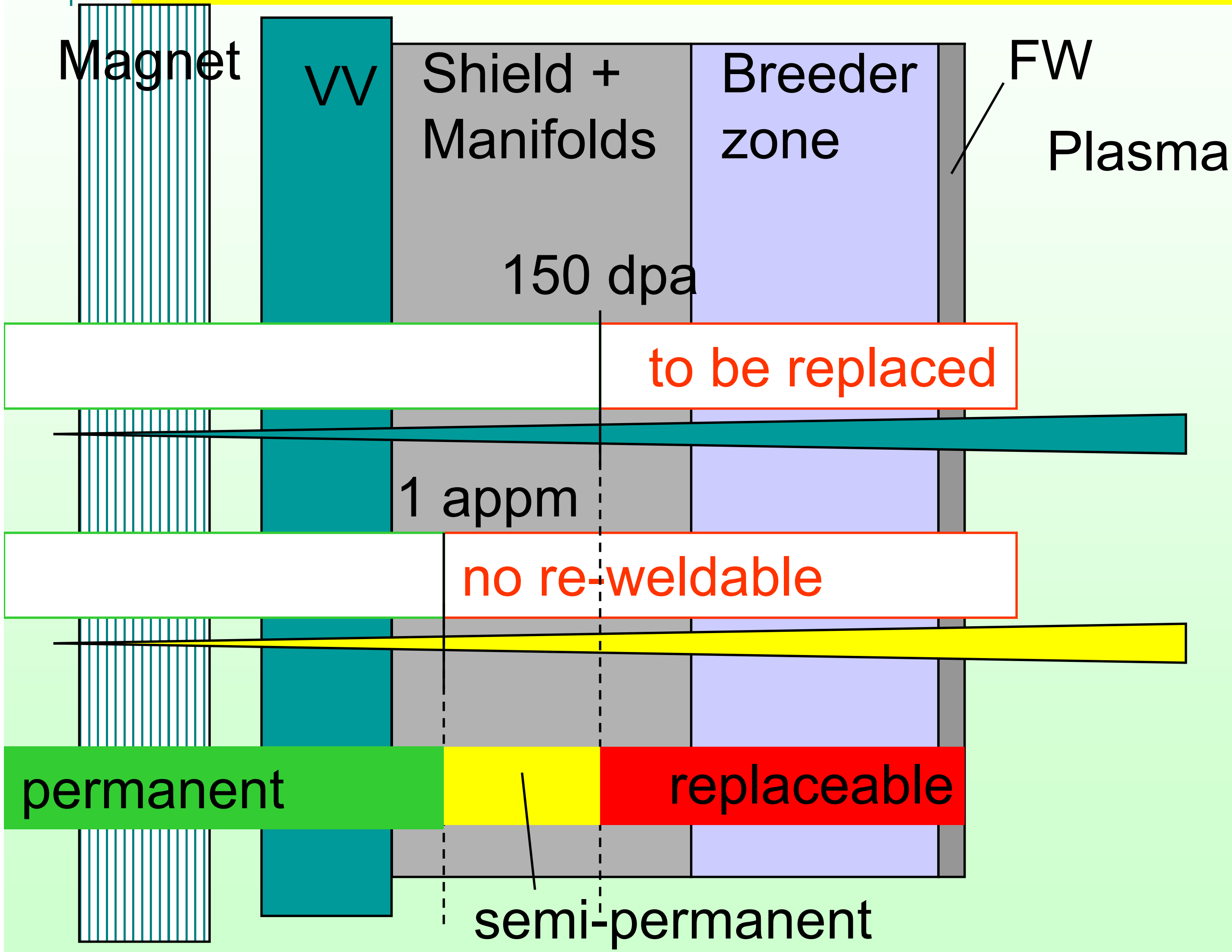
Lower port: 4 ports for the 54 "Divertor Cassettes"

8.5 m plasma major radius, 1500 MW_{el}

From ITER to Reactor

	ITER	Fusion Power Plant
Dimensions:	**Major plasma radius 6.2 m**	**7.5 – 9 m (for 1000-1500 MW_{el})**
Power densities:	**0.78 MW/m2 as neutron wall load** **0.25-0.5 MW/m2 as surface heating**	**2.5 MW/m2** **0.50 MW/m2**
Fluences	**max 0.5 MWa/m2 at the FW**	**~100 MWa/m2 (for 40 FPY at FW)**
Pulse length:	**400s (1000-3000 in advanced scenarios) and long dwell: ~1200s**	**10000 s and short dwell** **(or steady state)**
Blanket	**No tritium production (but..)** **Low coolant temperatures (no electricity production)** **Water cooling**	**Tritium production and extraction** **Higher temperatures for electricity production** **He cooling** **High shielding capability**
Divertor	**“Cold divertor”**	**Divertor integrated in the power generation system (divertor heat ~17% of the reactor thermal power).**
Availability:	**10%**	**>70-75%**

Radial build-up of the fusion reactor core (1/2)



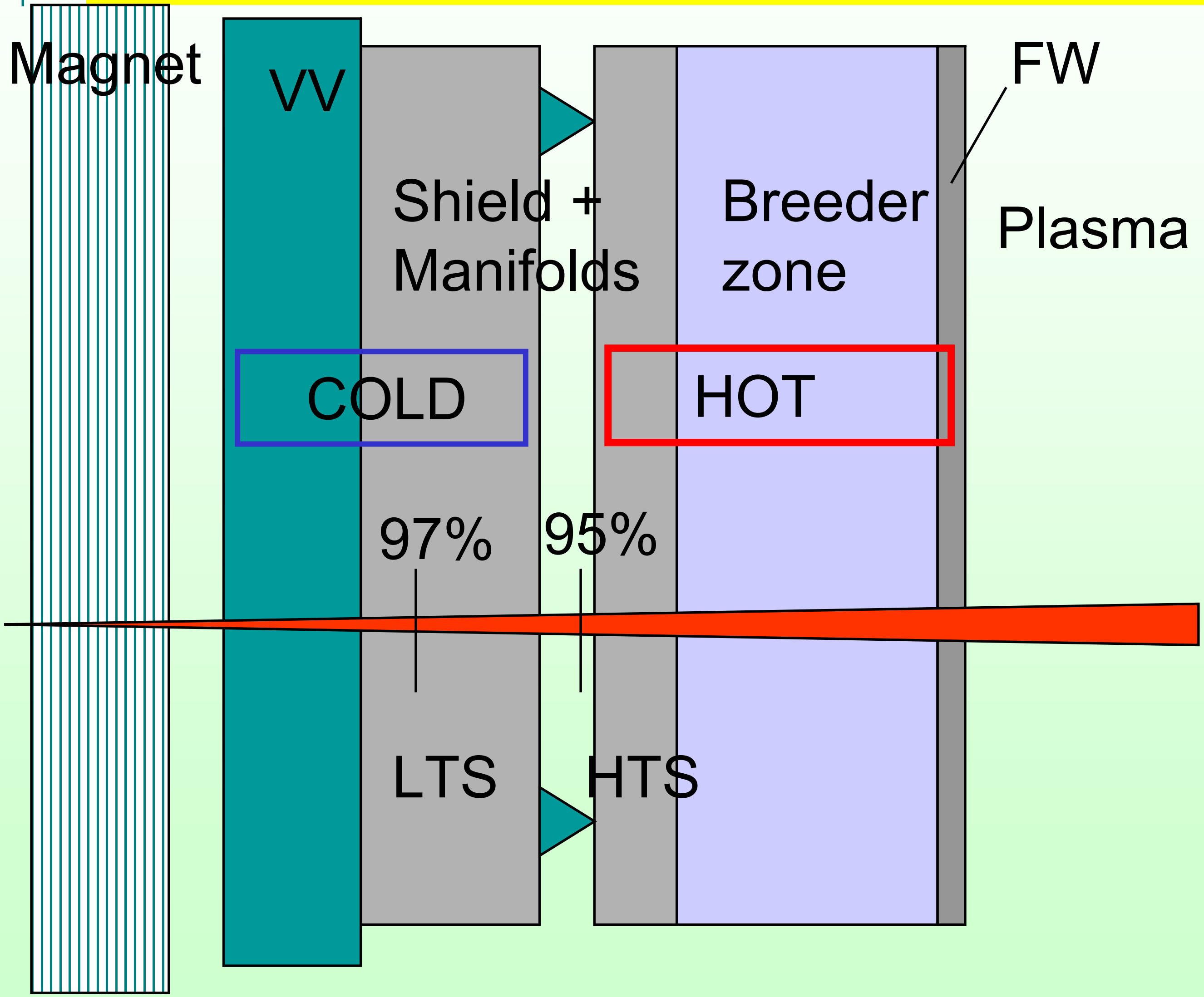
General criteria:

- Magnet protection
- VV re-weldable

damages in steel:
(dpa cumulated
in the lifetime)

Helium production:
(appm He)

Radial build-up of the fusion reactor core (2/2)

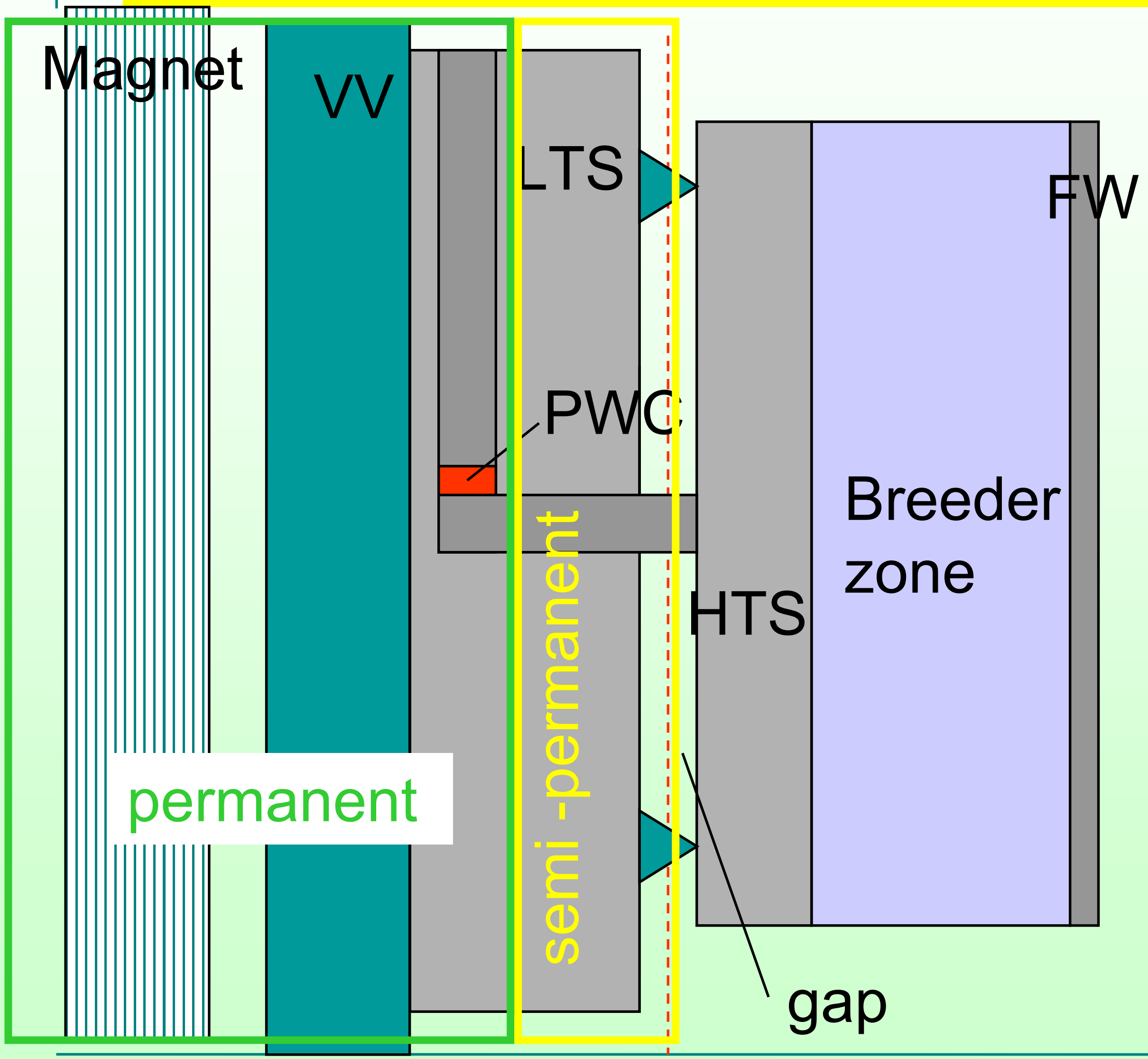


Cold: 150°C

Hot: 300-550°C

energy deposition
% of therm. power

Manifold as permanent components for the module concept

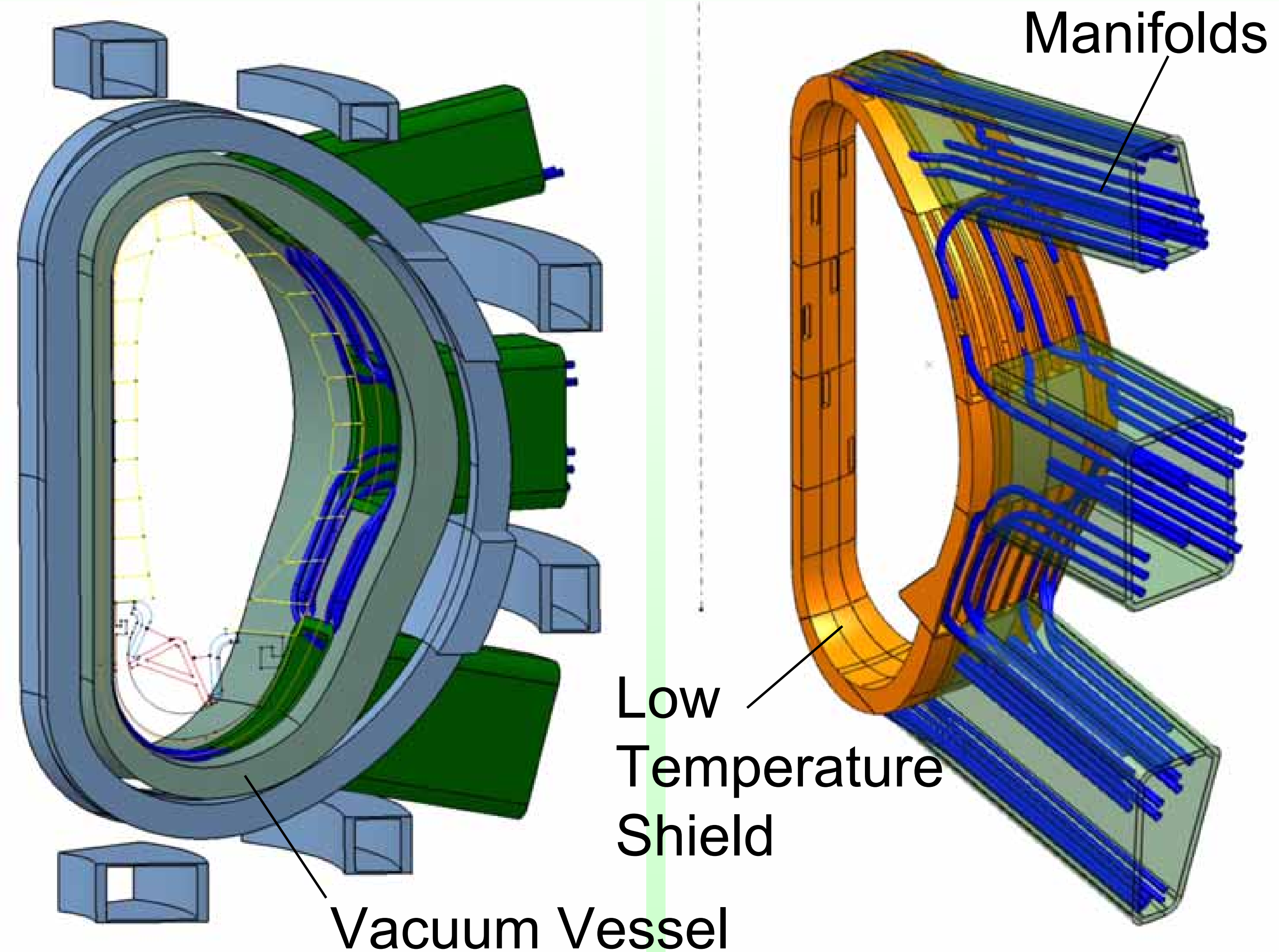


HTS: high temperature shield

LTS: low temperature shield

PWC: pipe weld connection

Vacuum Vessel and Low temperature shield



Vacuum Vessel:

-T ≈ 150°C

-Water cooled

-permanent component

Low Temperature shield:

-T ≈ 150°C

-Water cooled

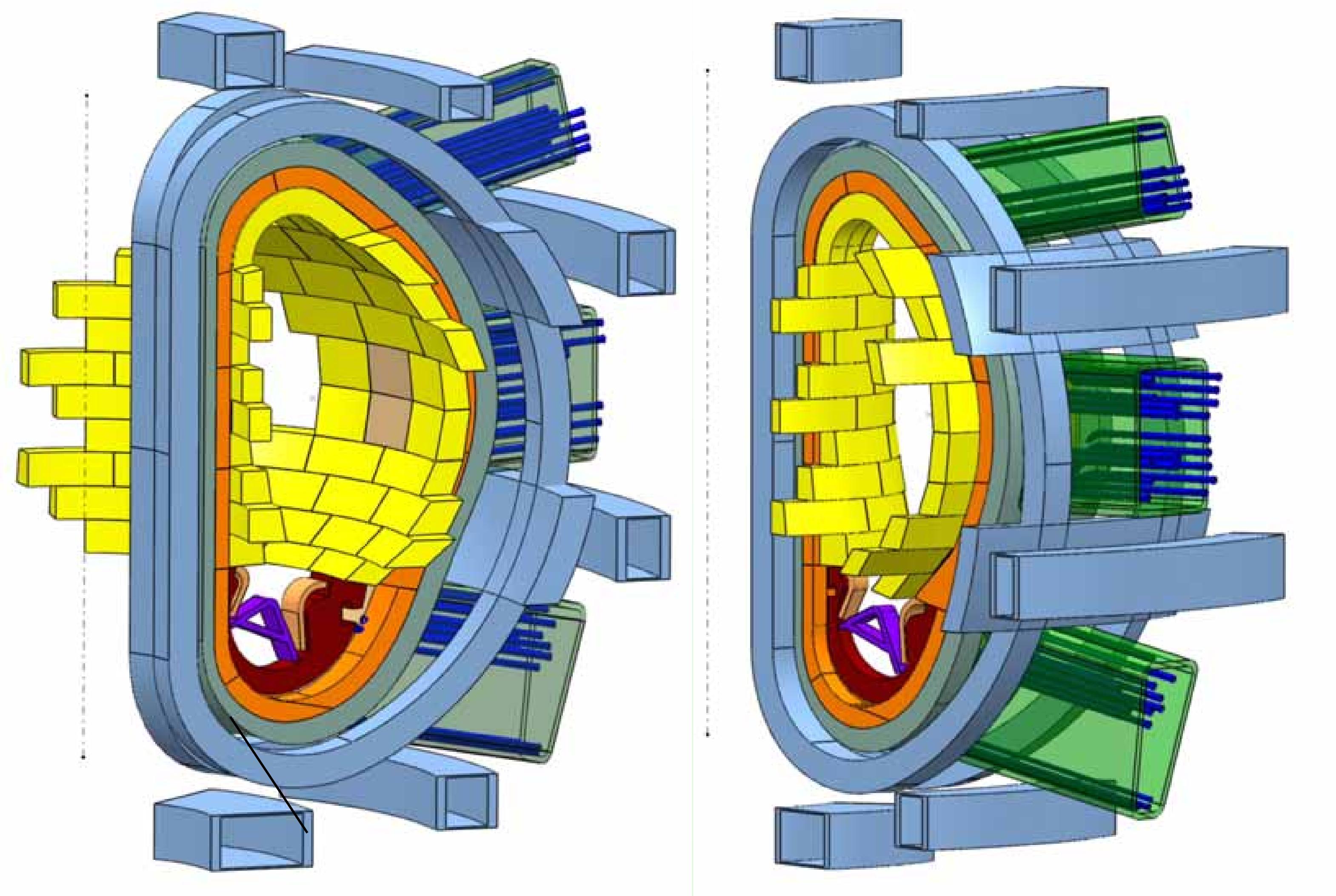
-semi-permanent component

Manifolds:

-T ≈ 300-500°C

- permanent components

Blanket and Divertor



Blanket:

- T ≈ 300-550°C
- Helium Cooled
- replaceable

Divertors:

-T ≈ 550-650°C

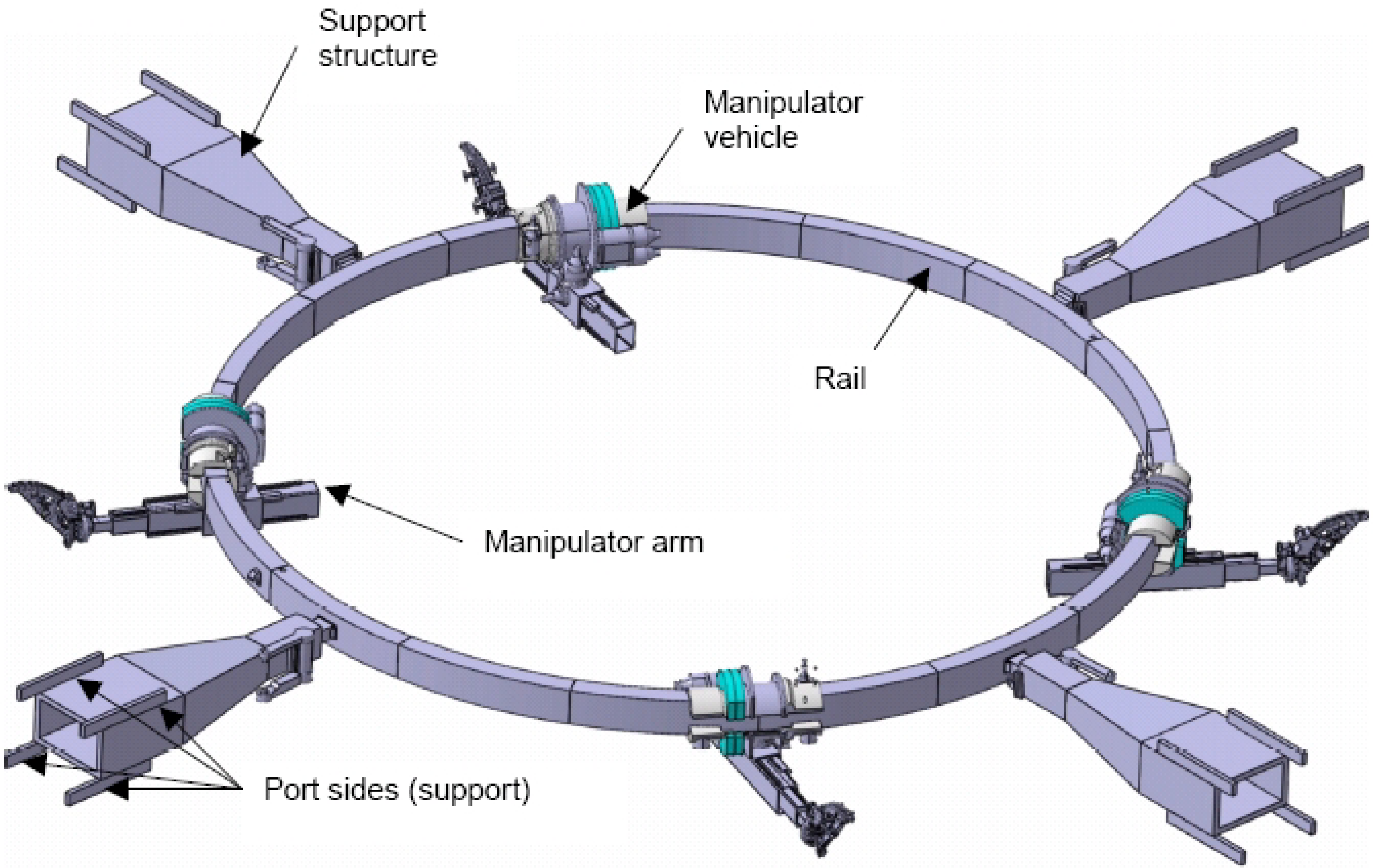
-Helium Cooled

-replaceable

Blanket segmentation

System:	Overall	Units per 20°-segment	Pipes per 20°-segment	ID pipes	RH method
Blanket Cassettes	52 cassettes	3	6	200 mm	Cassette
Inboard Modules	108 modules	6	12	150 mm	In- vessel mach.
Outboard:					
- Upper Mod.	72 modules	4	8	200 mm	In- vessel mach.
- Middle Mod.	36 modules	2	4	170 mm	In- vessel mach.
- Lower Mod.	96 modules	5-6	10-12	200 mm	In- vessel mach.
- Port Plugs	36 modules	2 (1 plug)	4	200 mm	Plug
Divertor Cassettes	52 cassettes	3	6	200 mm	Cassette
Note: 10 t modules	400 blanket 52 divertor				

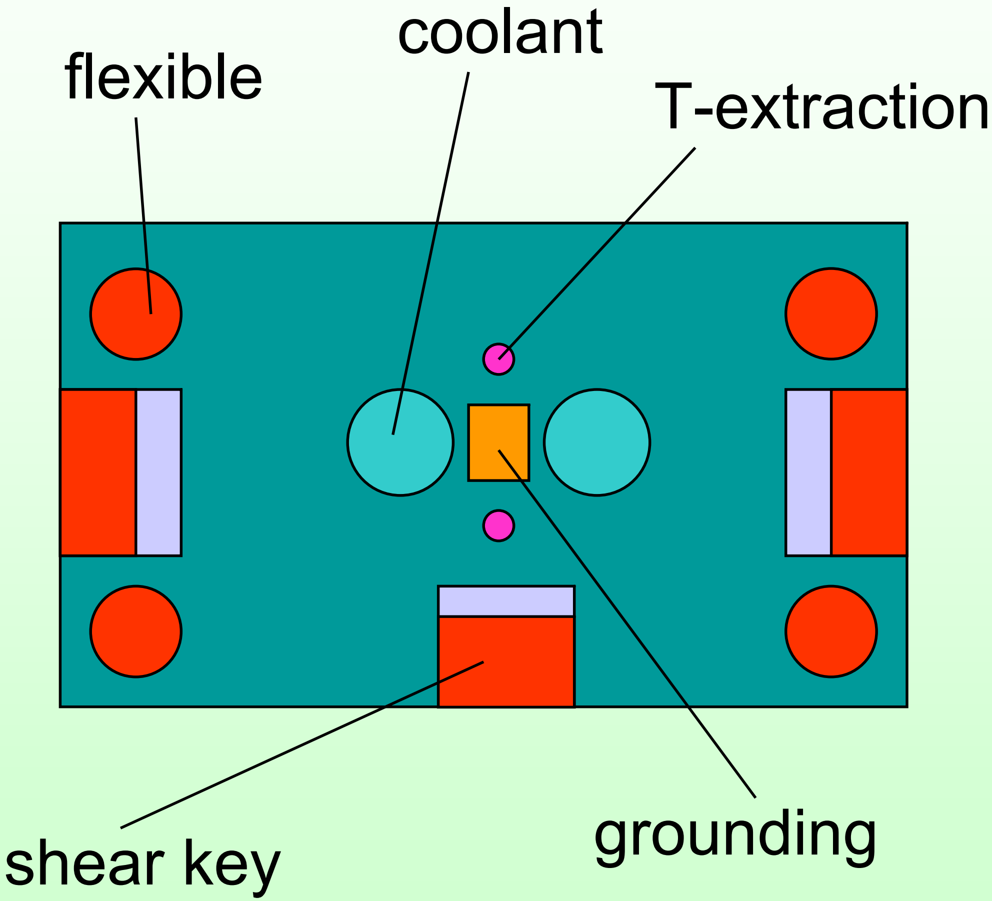
In-vessel machine for FPP (10 t)



- 10 t per module
- 4 independent manipulators

D. Nagy: "In-vessel Remote Handling Machine for Blanket Replacement in the Demo fusion reactor", this conference.

Typical interface of a module (10 t)



1) Mechanical attachment (red)

-3 shear keys, 4 flexibles

2) Coolant pipes (light blue):

-2 x 150-200 mm pipes

[DCLL: 4x100 mm pipes]

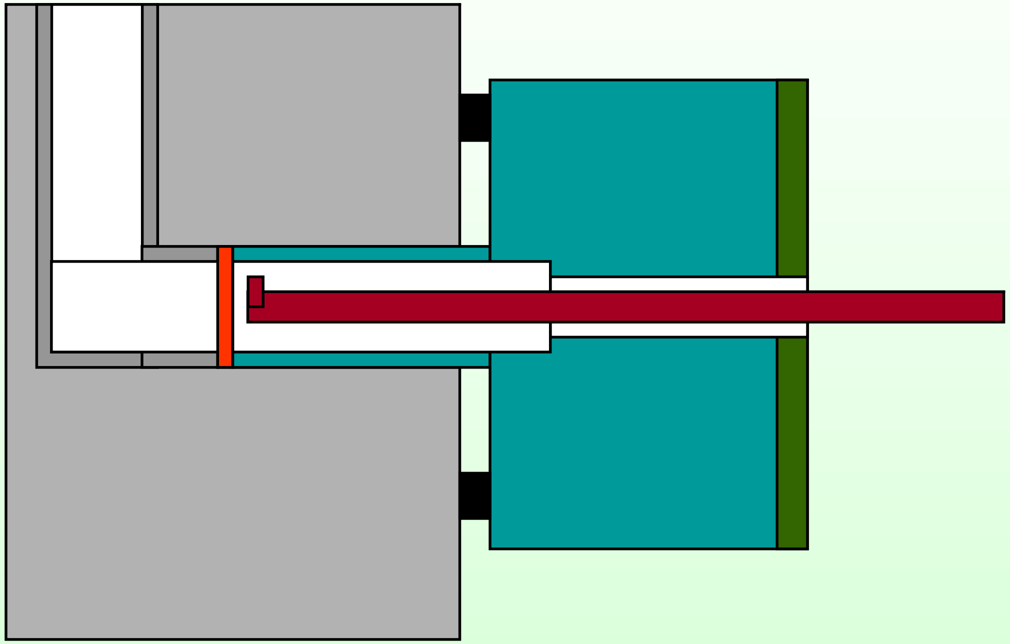
3) Tritium recovery (violet):

-2 x 70 mm pipes

4) Electrical grounding (orange):

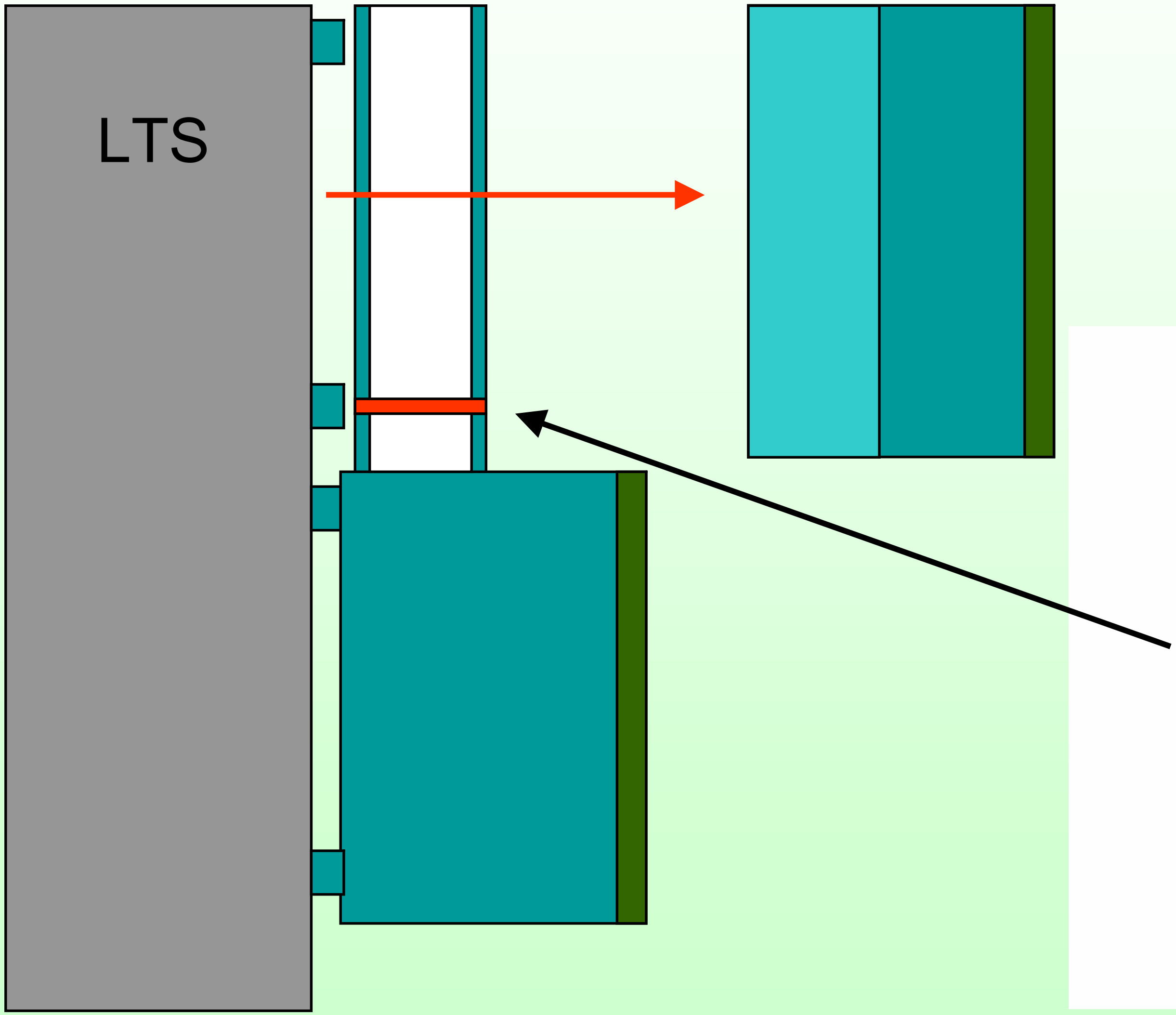
- connecting straps (tbd)

Cutting Rewelding of hydraulic connections with frontal access



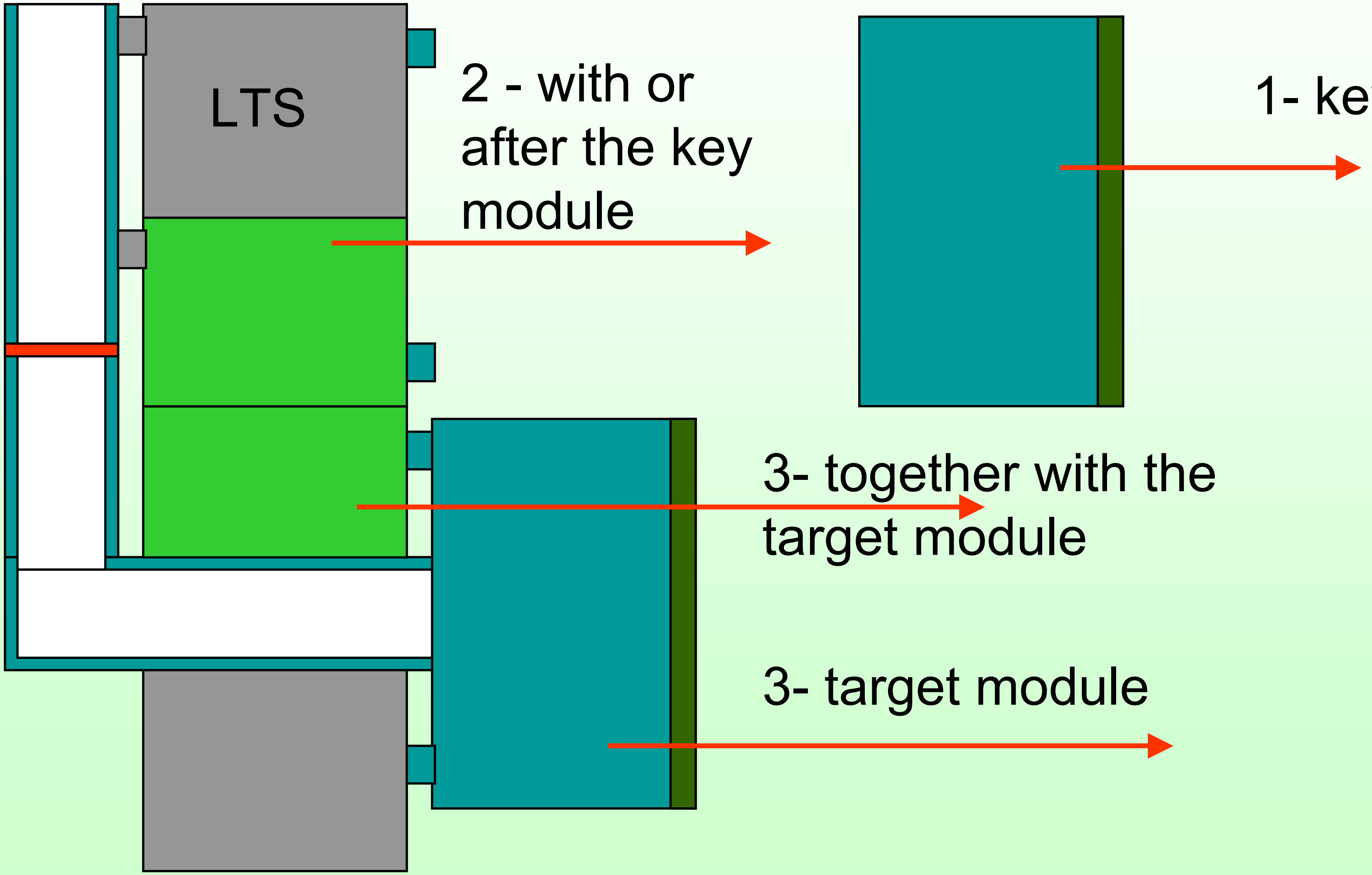
- The pipes have a diameter of 150-200 mm. It is questionable an access of only 3 cm-diameter.
- The cutting/welding zone is too deep in the low temperature shield and the re-weldeability criteria is not demonstrated.
- Helium cannot shield neutron like water !

Orbital tools

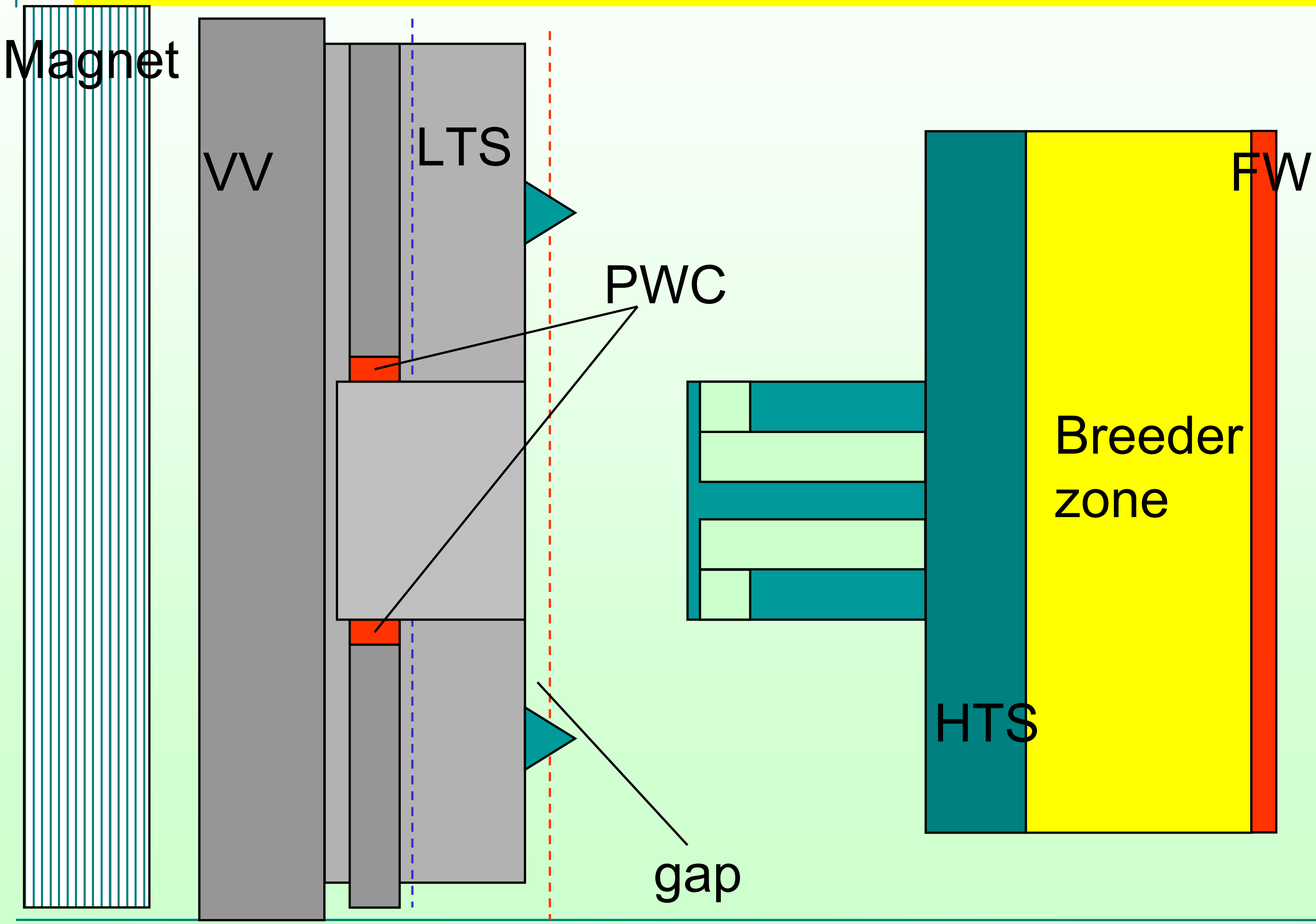



Model 95-1500 Low Profile

Orbital tools for permanent pipes



Use of in-bore tools in the DEMO Transporter concept



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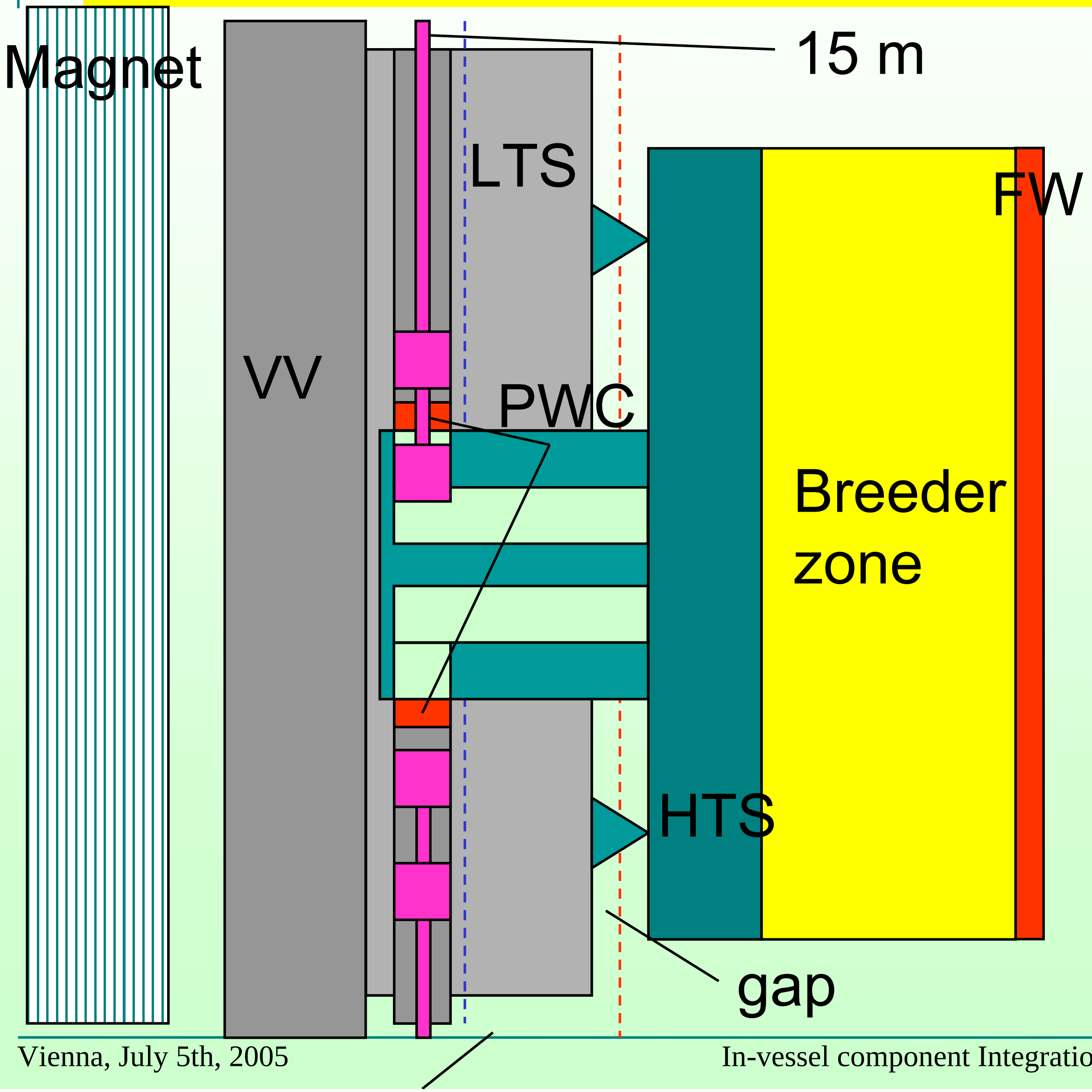
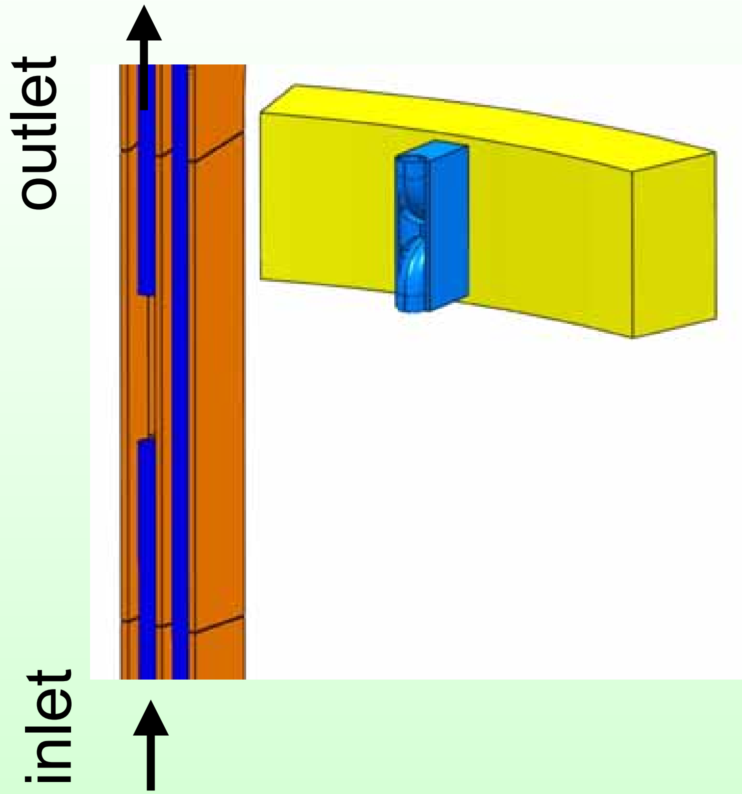


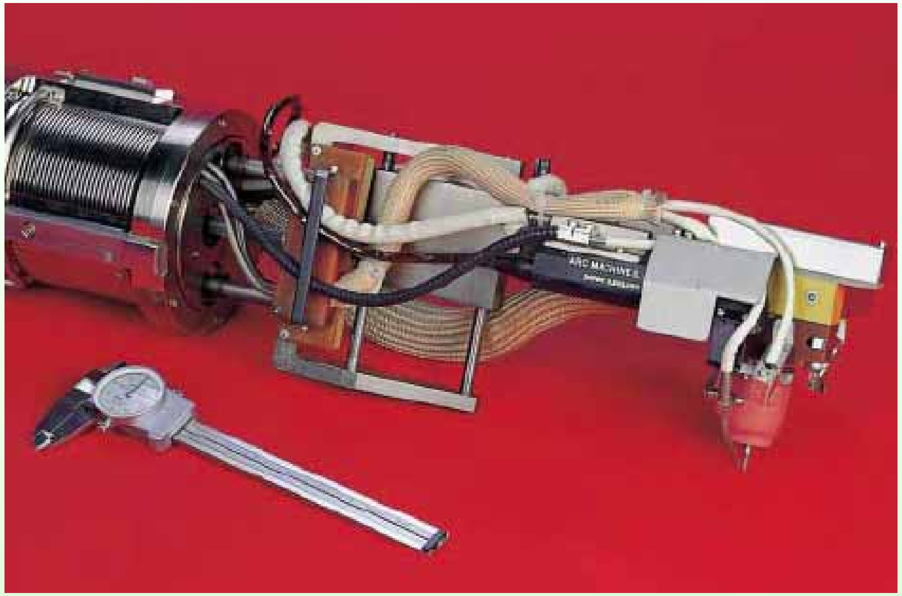


LTS: low temperature shield

HTS: high temperature shield

PWC: pipe weld connection

In bore tools (J. Rey)



1) Realisation of path reference:

- Optical detection
- Detection by 1-D tire roll motion
- Detection with 3-D gyroscopic system
- Detection with 3-D Laser scanner system

2) Cutting:

- Because of internal dust only LASER torch system
- With Inertgas radial dust outblow
- Internal vacuum system for dust minimisation

3) Re-welding:

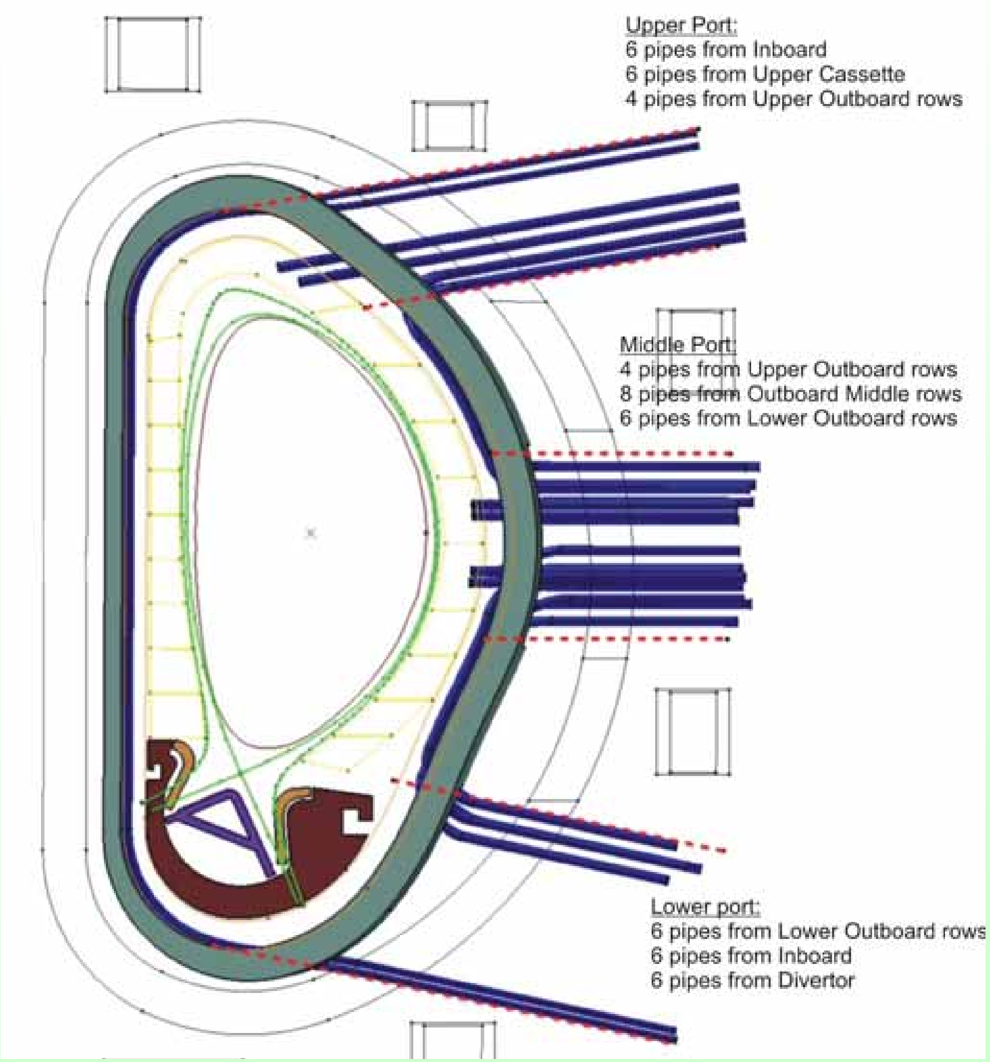
- Axial, radial and missangle tollerances $\Rightarrow$ TIG and Hybrid Laser with additional wire
- wall thickness 10-15mm
- Miniaturisation.

4) Inspection systems

- Visual with micro camera
- Eddy current (surface cracks)
- Thermal mapping
- US
- He leakage tests

J. Rey: "In-bore tools for the Blanket Replacement in the Demo fusion reactor", this conference

DEMO transporter: pipe lay-out



Upper Port:

- 6 pipes (out) for Inboard Blanket
- 6 pipes (in/out) for Blanket cassette
- 4 pipes (out) for Outboard

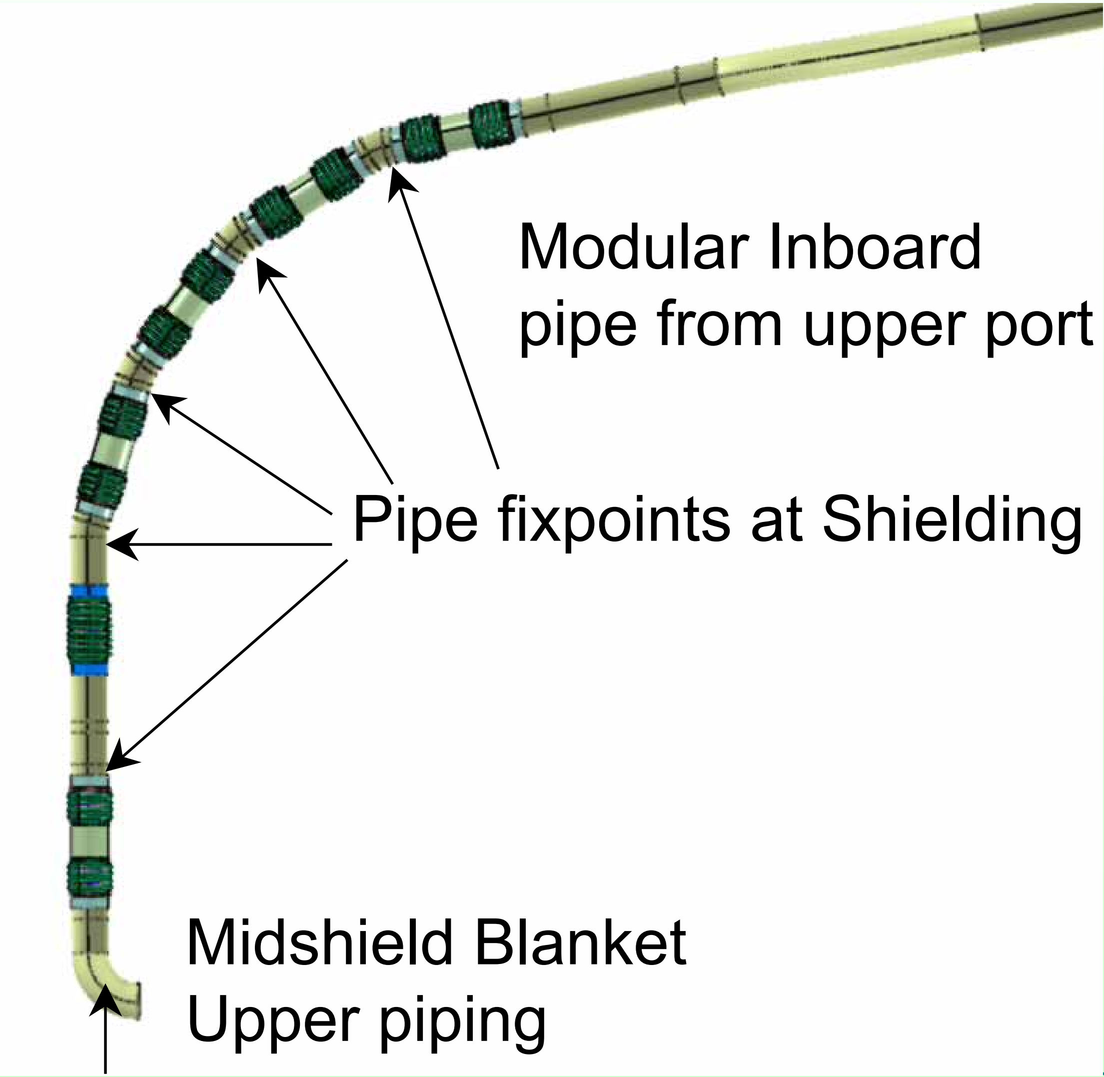
Middle Port:

- 14 pipes(in/out) for Outboard Blanket
- 4 pipes (in/out) for plug modules

Lower Port:

- 4-6 pipes (in) for Outboard Blanket
- 6 pipes (in) for Inboard Blanket
- 6 pipes (in/out) for Divertor Cassette

Pipes thermal compensation (pipe fixed at the LTS)



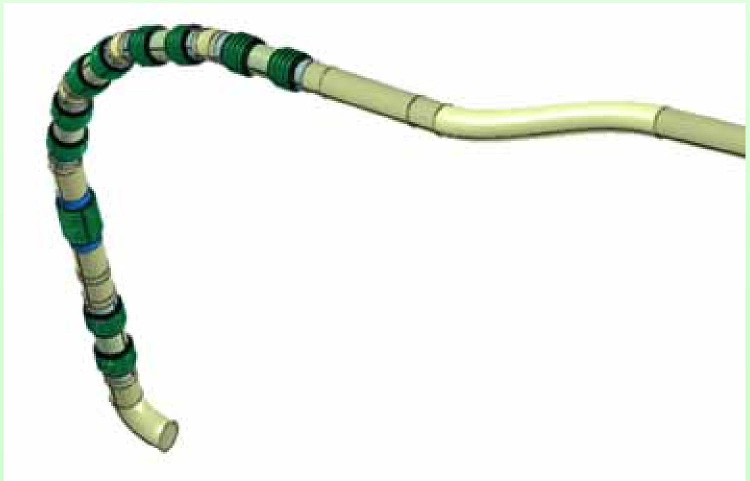
Requirements:

ID = 150-200 mm

ΔT = 150-350 K

Δp = 8 MPa

cycle = 100000



Conclusion 1

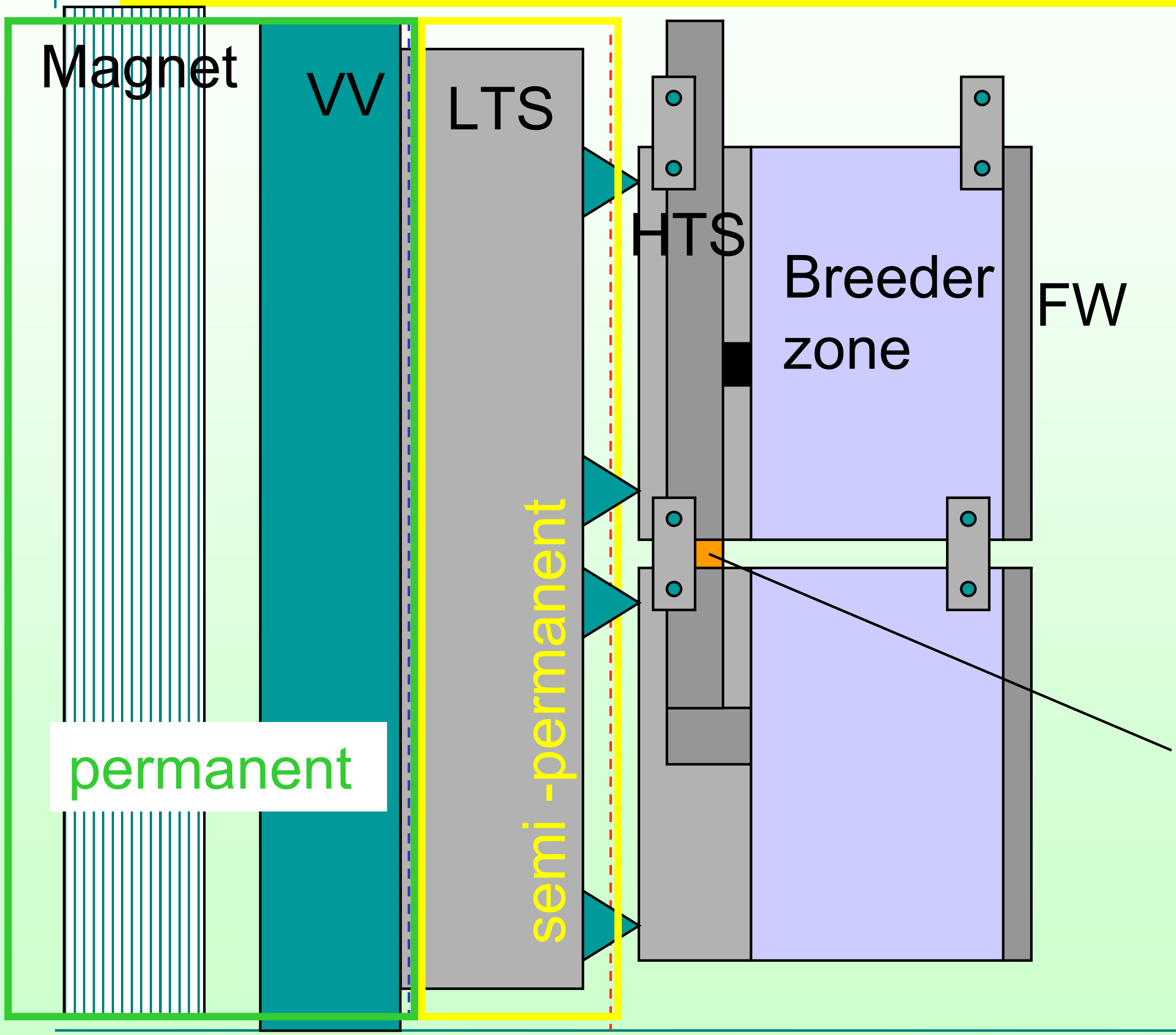
The adaptation of the transporter concept of ITER for a FG-FPP has been investigated:

- **radial build-up**
- **blanket and divertor segmentation**
- **cutting/welding procedures for He pipes**
- **thermal compensation**
- **piping lay-out**
- **mechanical attachment**
- **in-vessel RH machine**

Further analyses will be concentrated on:

- **critical technologies (in-bore tools and pipe thermal compensation)**
- **availability**
- **safety performances**

Alternatives: MMS with manifold as replaceable components

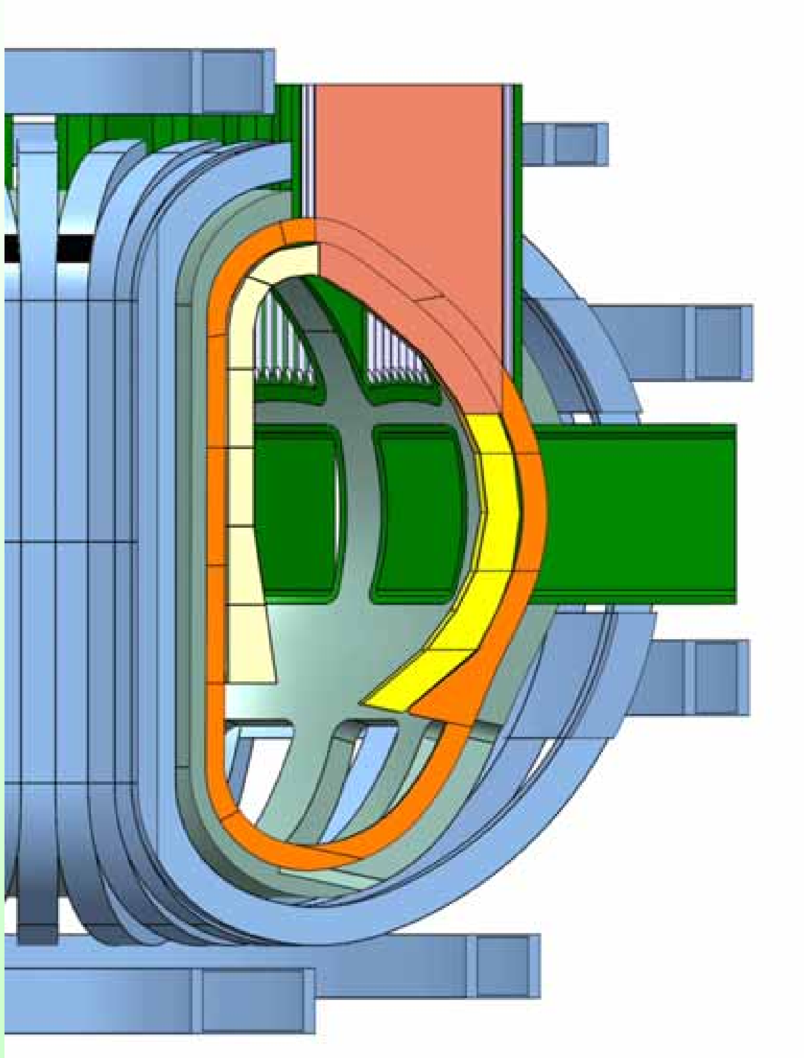


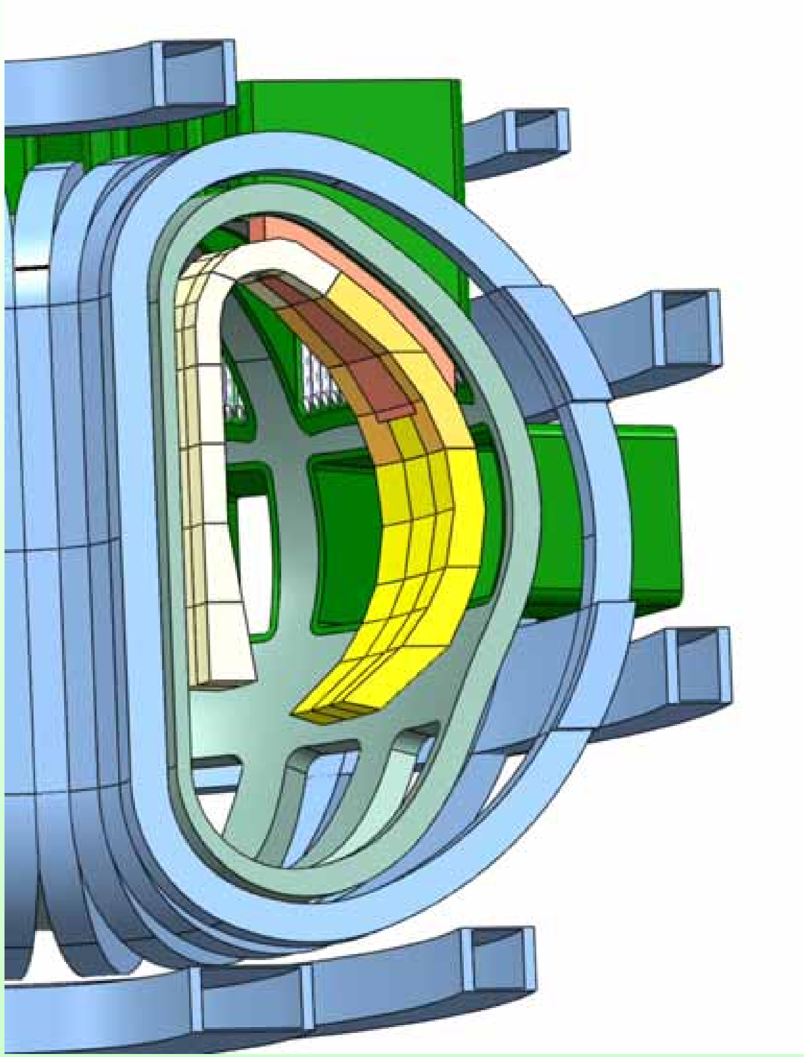
HTS: high temperature shield

LTS: low temperature shield

TC: thermal compensator

DEMO Multi Module Segments





Conclusions

- The engineering of the fusion core is one of the most challenging enterprise for the design of the first generation of fusion reactor.
- It is the key for reaching the high availability, that is essential for the economical competitiveness of the FPP in respect of other energy sources
- ITER is the first example of this integration, but it is not a power plant. The definition of specific requirements for the FPP is still under investigation.
- In FZK a study started last years to investigate the integration of the Helium cooled in vessel components. A model of ITER derivation has been analysed and the issued defined.
- Alternative configurations have been proposed and the most promising selected for future investigations.