DEN

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The PPCS In-Vessel Component Concepts
(focused on Breeding Blankets)

Presented by

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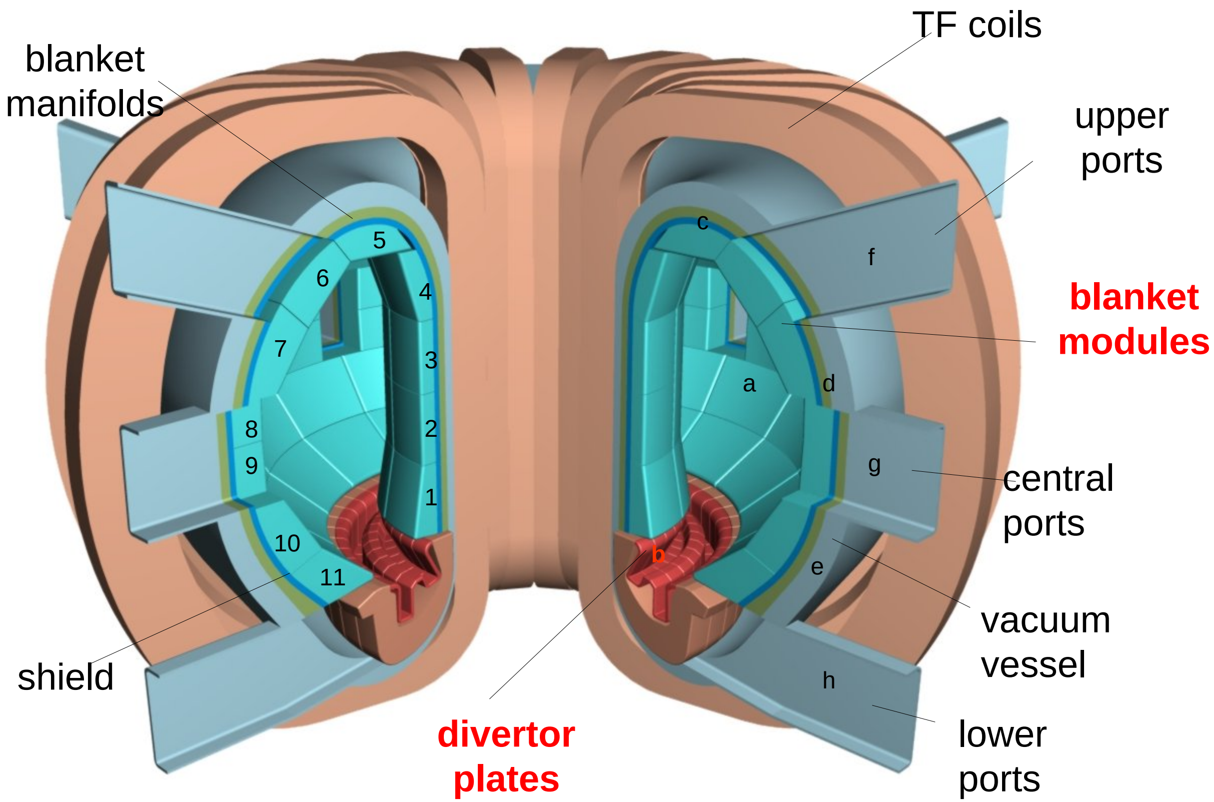
Plan of the Presentation

1. **Introduction on In-Vessel Components**
2. **Breeding Blankets**
    1. Functions and Strategy
    2. Present Choices for EU PPCS and DEMO
    3. Design & Analyses Details of the EU PPCS Blankets
    4. Blanket Systems and Integration Issues (examples)
    5. R&D Issues (medium term, long term)
3. **Divertors** *(short overview)*
    1. Functions and Strategy
    2. Choices for EU PPCS
4. **Conclusions**

In-Vessel Components Location & Conditions

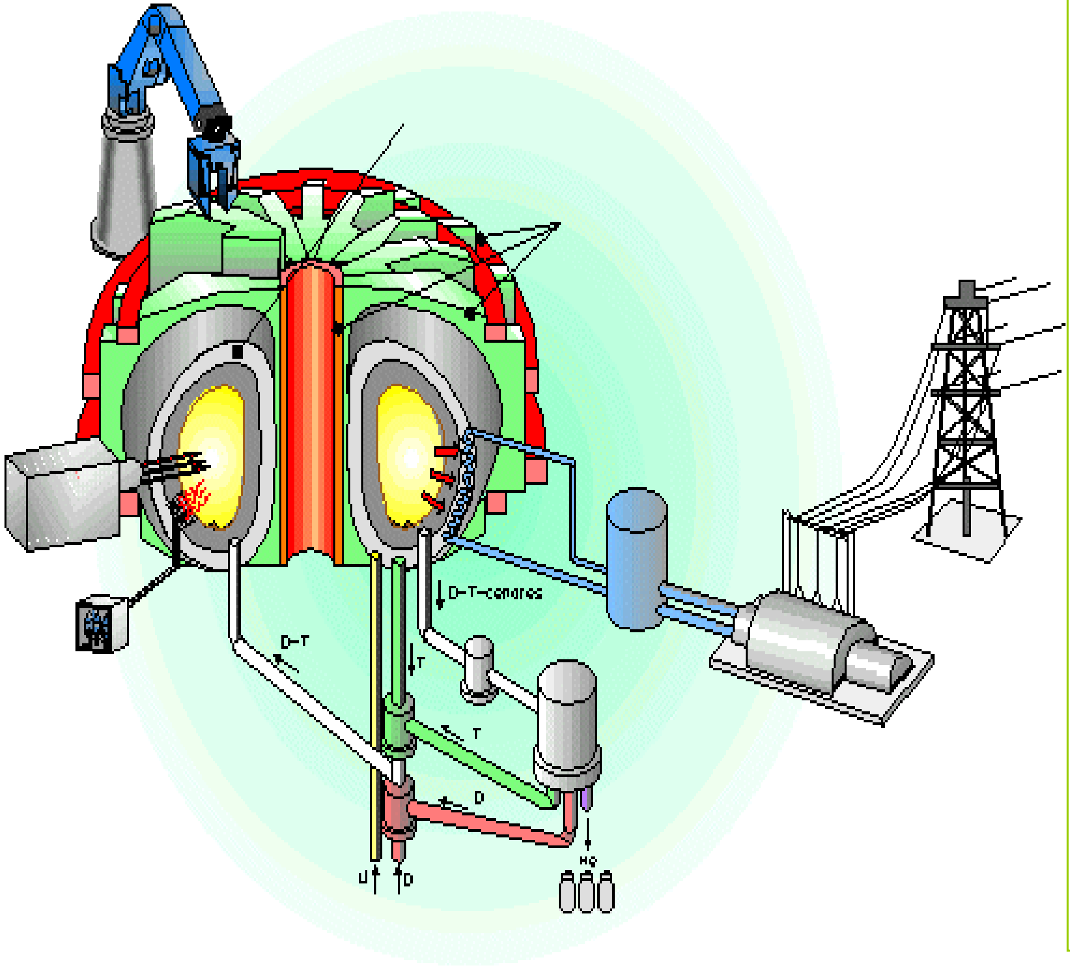


The most severe working conditions and requirements

- High surface heat flux 0.5 (FW) ➔ 5 MW/m^2 (div.)
- High neutron wall loading (FW) ~2.5 MW/m^2, ~150 dpa(Fe)
- Operation under void (plasma) ➔ low coolant leakages
- High magnetic field (~7 Tesla) (high MHD effects)

Moreover : Remote access in high radiation field (maintenance, inspection, repairs, diagnistics,…)

Reactor breeding blankets



Three crucial functions for a Power Plant

- **Convert** the neutron energy (80% of the fusion energy) in heat and collect it by mean of an high grade coolant to reach high conversion efficiency (>30%)
  ➔ in-pile heat exchanger
- Produce and recover all **Tritium** required as fuel for D-T reactors
  ➔ Tritium breeding self-sufficiency
- Contribute to neutron and gamma **shield** for the superconductive coils
  ➔ resistance to neutron damages

Tritium breeding self-sufficiency : a necessity

- Main Tritium production reaction : $^{6}Li + n \Rightarrow T + {}^{4}He + 4{,}8$ MeV
- Typical T-production rate in dedicated fission reactor : 1-2 kg T / GW_{th} / y
- Typical T burn rate in a fusion reactor : ~50 kg T / GW_{th} / y *(~150 gT/GW_{th}/day)*
- *N*eed to produce all T and collect it on-line (because of initial inventory and safety)

➔ ***Tritium Breeding Ratio, TBR = T_{prod}/T_{burn} > 1*** *(at least)*

Tritium breeding self-sufficiency : a severe constraint

$D + T \rightarrow {}^{4}He + n$

60-80%

$^{4}He + T \leftarrow {}^{6}Li + n$

➔**20 to 40 % of neutrons are not available for T-production**

Neutron Losses

✓ n-leakage through ports, divertor region, gaps
➔ 10 – 20 %

✓ n-parasitic absorptions on other materials (structures)
➔ 10 – 15 %

✓ blanket n-leakage (<5%)

Neutronics requirements

❑ ***minimize n-losses***
➔ small gaps and openings, minimization of structures and coolant fractions

❑ ***add a n-multiplier***
Pb or Be (n,2n) or Li^{7} (n, n'T)

Breeding blankets materials

Main available breeders
(Li-based compounds)

- ✓ Liquid Lithium (7.5% ^{6}Li)
- ✓ Eutectic Pb-17Li (T_m 235°C)
- ✓ Molten Salts :
  FLiBe, FLiNaBe
- ✓ Li-Ceramics :
  Li_4SiO_4, Li_2TiO_3, LiO_2

Neutron multipliers

- ✓ Be (n, 2n)
- ✓ Pb (n, 2n)
- ✓ ^{7}Li (n, n'T)

Main Structural Materials

- ✓ Ferritic/Martensitic Steels
- ✓ Vanadium Alloys
- ✓ Composites SiC/SiC

Main Coolants for Nuclear
(relevant T for good efficiency))

- ✓ Pressurized Water (PWR)
- ✓ Helium
- ✓ Liquid Metals : Li (Na), Pb-17Li

A breeding blanket has to be designed using a combination of these materials. Best concepts are an optimum compromise between all constraints which has to take into account:

☞ temperature, pressure, loads levels ☞ compatibility between materials under the defined working conditions ☞ safety ☞ low activation/low radwaste characteristics ☞ performances (e.g., TBR) ☞ technology availability and manufacturing processes as a function of timescale ☞ reliability & others…..

Breeding Blankets Development Timescale

Reactors	*JET, TFTR, JT60, Tore Supra Asdex, etc.*	*ITER* *(1) international*	*DEMO* *(1 or more ?)*	1st Commercial Power Plant *(1 or more ?)*	*10th of the kind* ➔ *PPCS*
Main	**Plasma Physics** *(control,stability, impurities, shutdown proc., heating, etc.)*	**Plasma Ignition Long D-T Pulses Systems Integration**	**Tritium Breeding Self-sufficiency High Safety Standards**	**Electricity Production at Competitive Cost** **High**	**Electricity Production at Competitive Cost** **Possible Advanced**
Goals	**Magnetic Field Advanced Systems** *(e.g., super cond. coils)*	**Test of DEMO Blanket Mock-ups** *(Test Module)*	**Electricity Production** *(high efficiency, but low reliability system)*	**Reliability** **Low Level Waste**	**Reliable Technology** **Low Level Waste**
Operation Years	**~1980 - 2010**	**~2014 - 2034**	**~2030 - 2050**	**From ~2050**	**> 2080**

Breeding Blankets Concepts

	Power Plant Model				
Basic Parameters	**A**	**B**	**AB**	**C**	**D**
Major Radius (m)	9.8	8.7	9.1	7.5	6.1
Aspect Ratio	3.0	3.0	3.0	3.0	3.0
Plasma Current (MA)	33.5	28.1	29.2	20.1	14.1
Toroidal Field on axis (t)	7.3	6.9	6.8	6.4	5.6
TF on TF Coil Conductor (T)	12.9	13.1	13.6	13.6	13.4
Elongation (95% and separatrix)	1.7/1.9	1.7/1.9	1.7/1.9	1.9/2.1	1.9/2.1
Triangularity (95% and separatrix)	0.27/0.4	0.27/0.4	0.27/0.4	0.47/0.7	0.47/0.7
Q	21	15	16	30	35
Engineering Parameters					
Fixed Electrical Output (GW)	1.5	1.5	1.5	1.5	1.5
Fusion Power (GW)	5.5	3.4	4.0	3.4	2.5
P_{add} (MW)	265	234	234	112	71
Avg. Neutron Wall Load FW (MW/m^2)	2.3	1.8	1.8	2.2	2.4
Surface Heat Flux on FW (MW/m^2)	0.5	0.5	0.5	0.5	0.5
Max. Divertor Heat Load (MW/m^2)	15	10	10	10	5
Selected Breeding Blanket Type	**WCLL**	**HCPB**	**HCLL**	**DCLL**	**SCLL**

PPCS Breeding Blankets Concepts

Short Term Concepts (for models A, B, AB)

✓ **A** - Water-Cooled Lithium-Lead (WCLL)
coolant T_{in}/T_{out} 285/325°C, 15.5 MPa

✓ **B** - Helium-Cooled Pebble-Bed (HCPB)
ceramic/Be blanket,
coolant T_{in}/T_{out} 300/500°C, 8 MPa

✓ **AB** - Helium-Cooled Lithium-Lead (HCLL)
coolant T_{in}/T_{out} 300/500°C, 8 MPa

➔ *To be tested in ITER (B & AB)*
➔ *Can be used for DEMO (after selection)*
➔ *Use of martensitic/ferritic steel structures (low activation EUROFER)*

➔ ***R&D still required, results expected on time for DEMO time schedule***

To demonstrate feasibility of Fusion for DEMO

To demonstrate performance objectives for Fusion

Medium Term Concept (model C)

✓ C - Dual-Coolant Lithium-Lead (DCLL)
& He ➔ He T_{in}/T_{out} 300/480°C, 8 MPa
LiPb T_{in}/T_{out} 480/700°C

➔ *Need SiC/SiC insulators + high Temp. HX*
➔ *Can be used in DEMO at later stage ?*
➔ *Use of martensitic/ferritic steel structures (low activation EUROFER)*

➔ ***Significant R&D required, relatively long time is required***

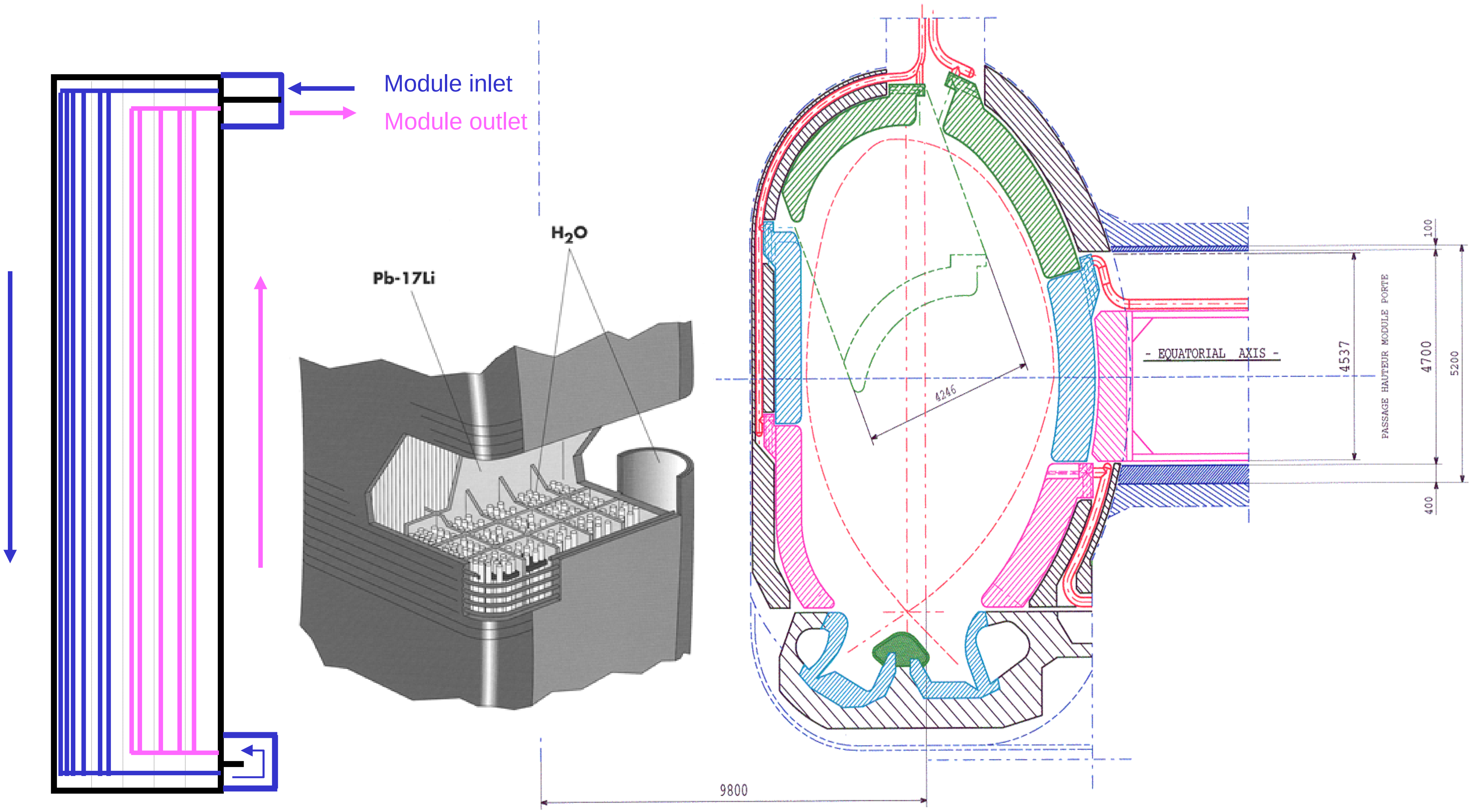
Long Term Concept (model D)

✓ D – Self-Cooled Lithium-Lead (SCLL)
LiPb T_{in}/T_{out} 700/1100°C

➔ *Use of SiC/SiC structures*
➔ *Need very large & lengthy R&D (50 y?)*
➔ *Very good efficiency & safety standard*

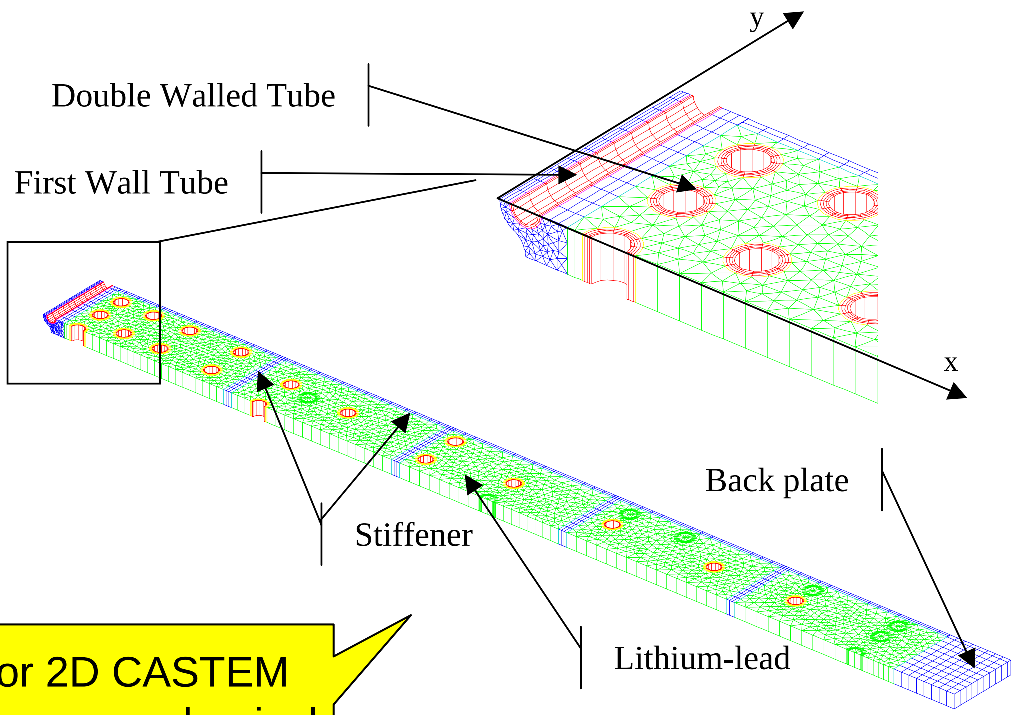
➔ ***Attractive but high development risk***

Blanket Concept A : Water-Cooled LiPb



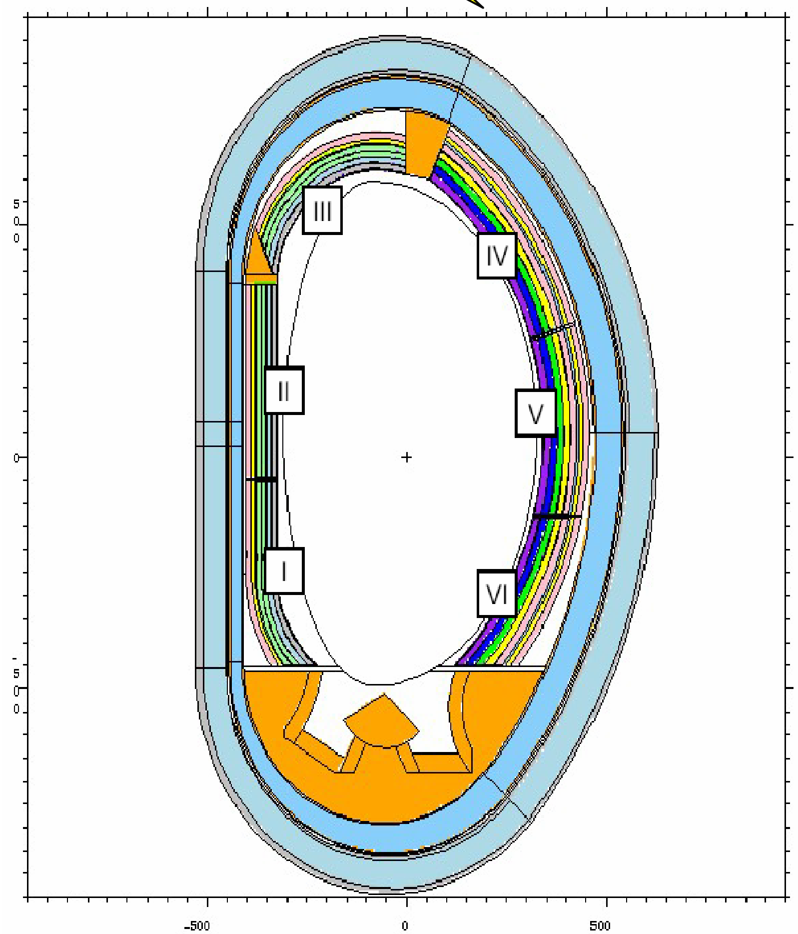
Blanket Concept A : Water-Cooled LiPb

Used Calculation Models



For 2D CASTEM Thermo-mechanical Analyses

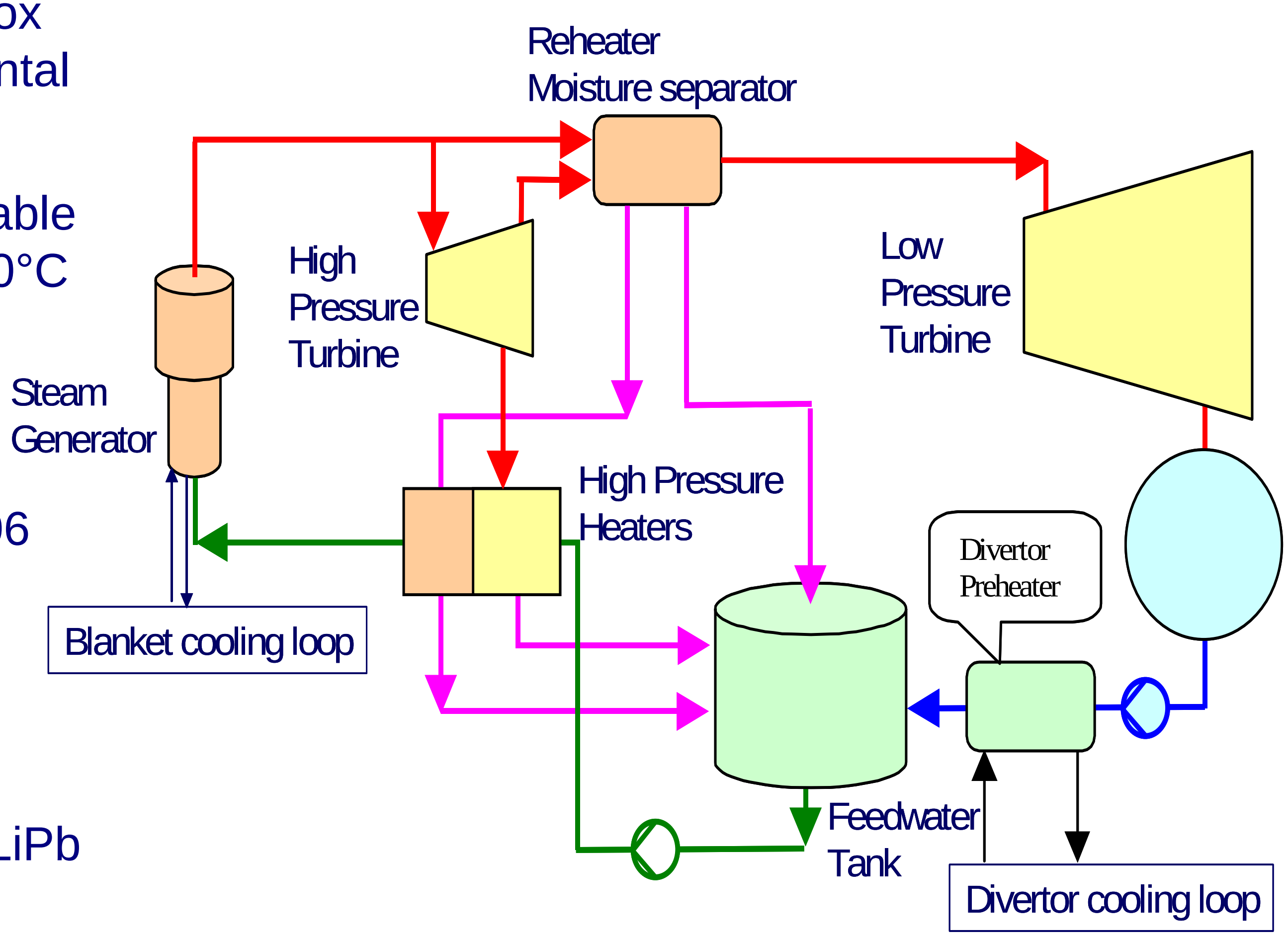
For 3D MCNP Montecarlo Neutronics Analyses



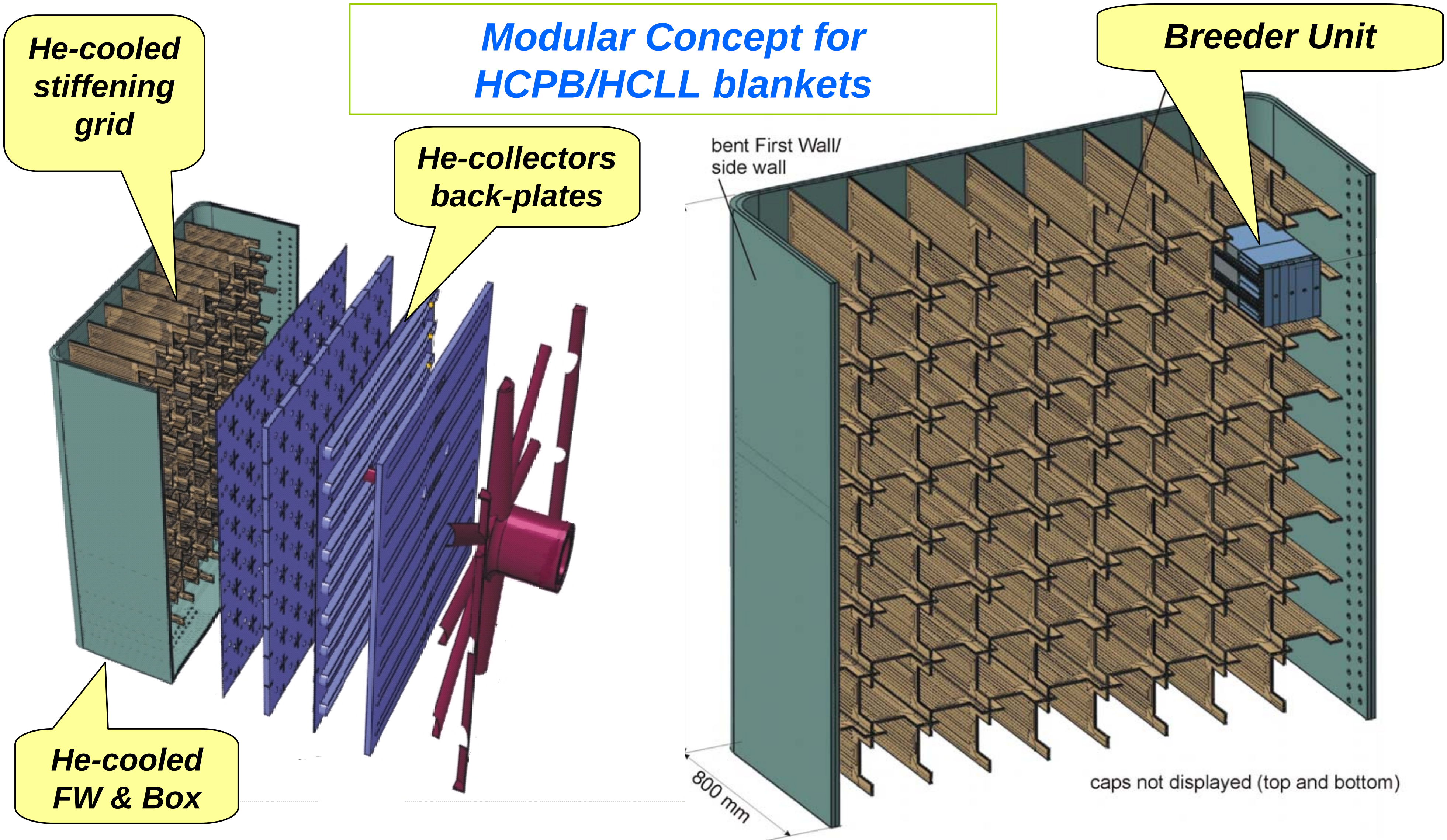
Main Analyses Results

- **Mechanics:** the modules box resists to 15.5 MPa in accidental conditions (RCC-MR)
- **Thermo-mechanic:** acceptable T (< 550°C in FW steel, < 480°C on Pb-17Li interface), acceptable stresses and deformation (RCC-MR)
- **Neutronics:** 3-D TBR = 1.06 at 90 at% ^{6}Li enrichment, 30 dpa/y in FW steel, 850 appm H/y, 300 appm He/y
- **Tritium:** maximize confinement and extraction (LiPb at ~2 mm/s to avoid MHD pressure drops)

Blanket Coolant External Circuits

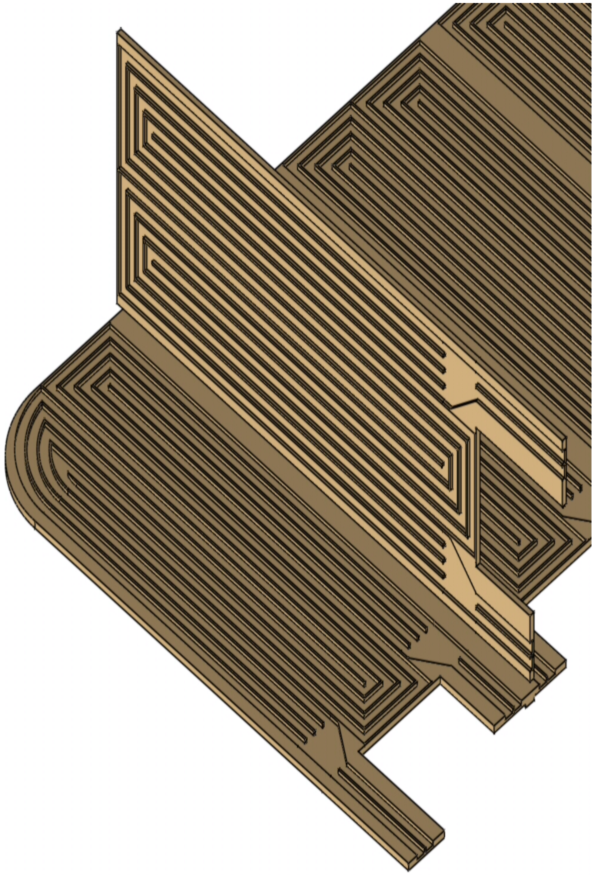


Blanket Concept B : Helium-Cooled Ceramic/Be

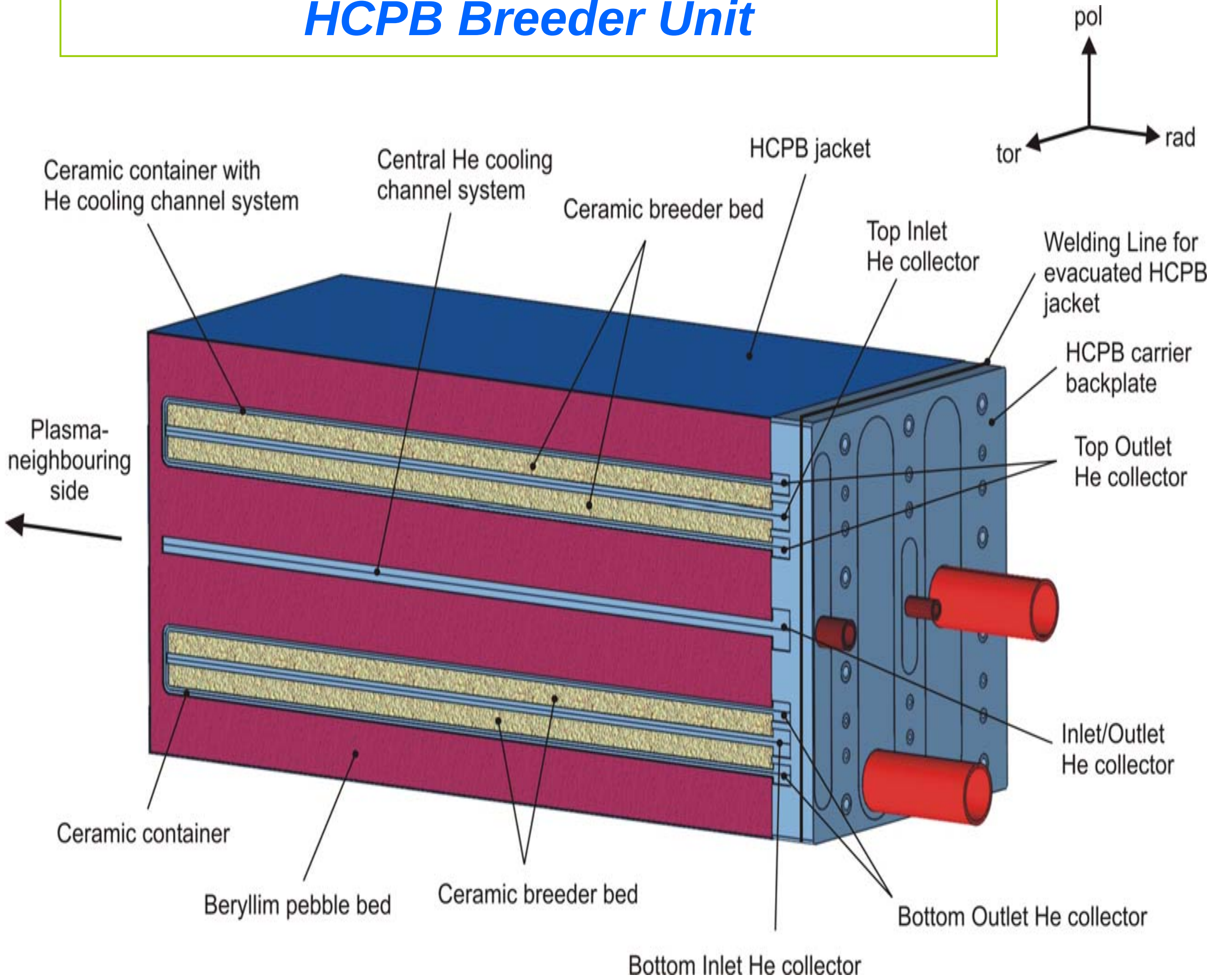


Blanket Concept B : Helium-Cooled Ceramic/Be

Stiffening Grid
(details)

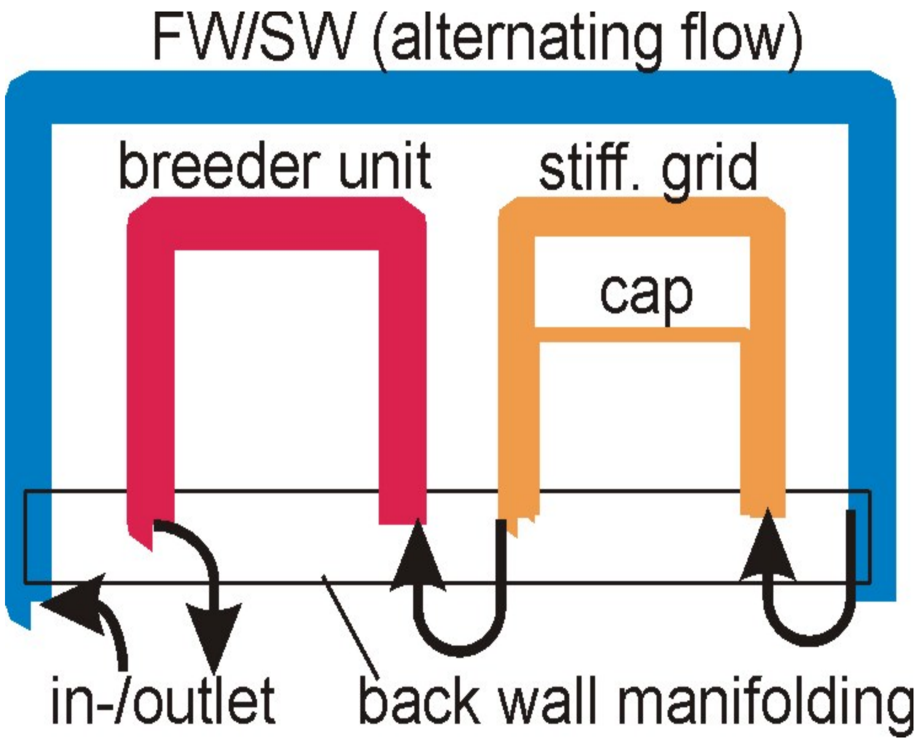


HCPB Breeder Unit

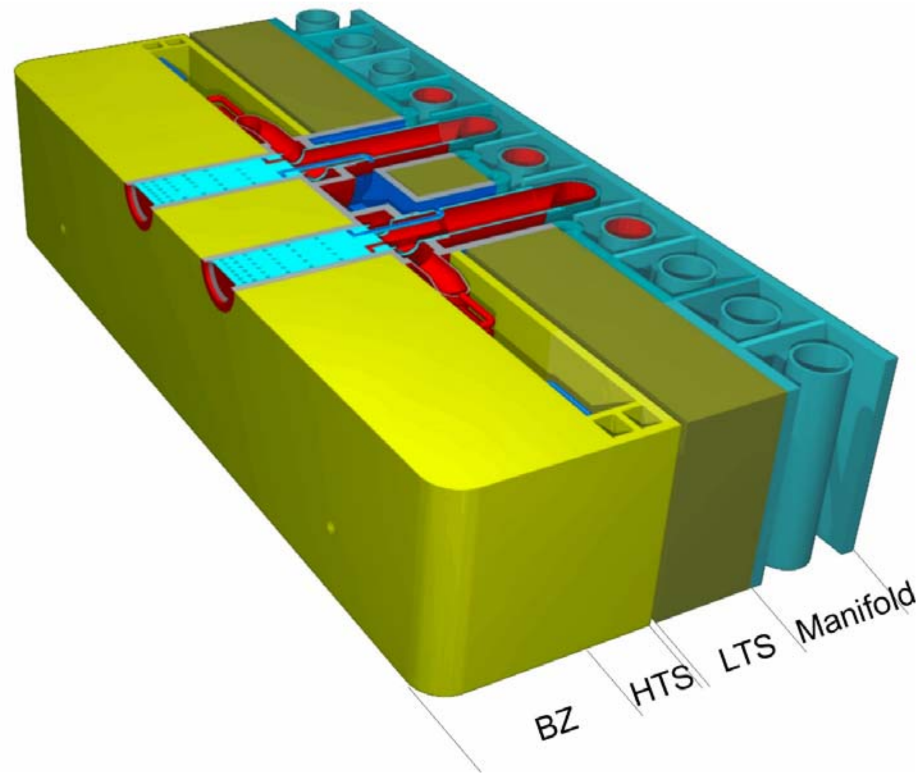


Blanket Concept B : Helium-Cooled Ceramic/Be

Scheme of the He-flow path in each module

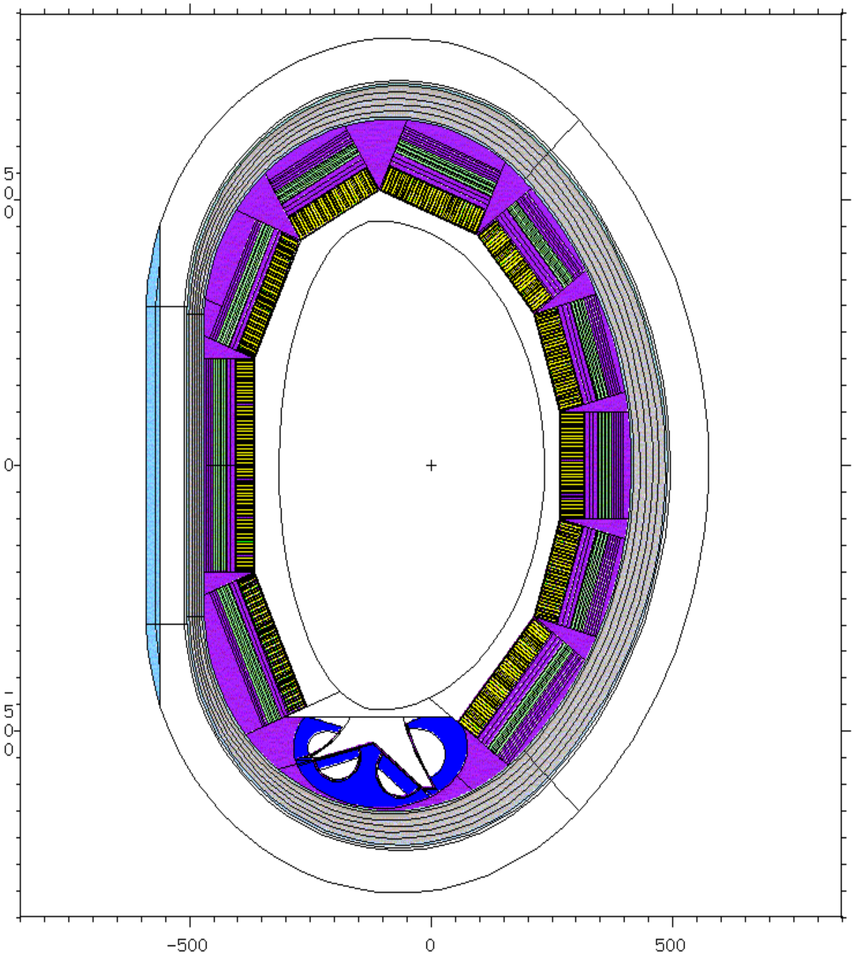


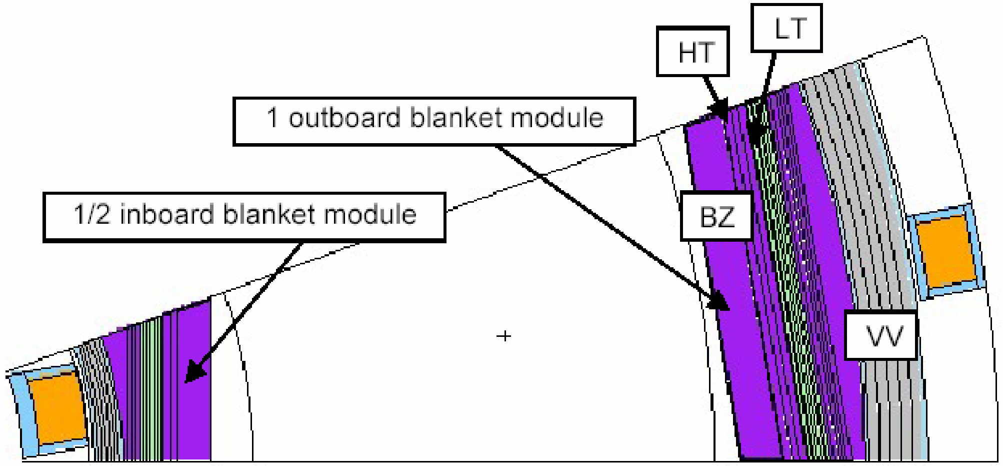
Scheme of a possible module attachment to the back shield



Blanket Concept B : Helium-Cooled Ceramic/Be

3D Montecarlo Neutronics Analyses (MCNP)

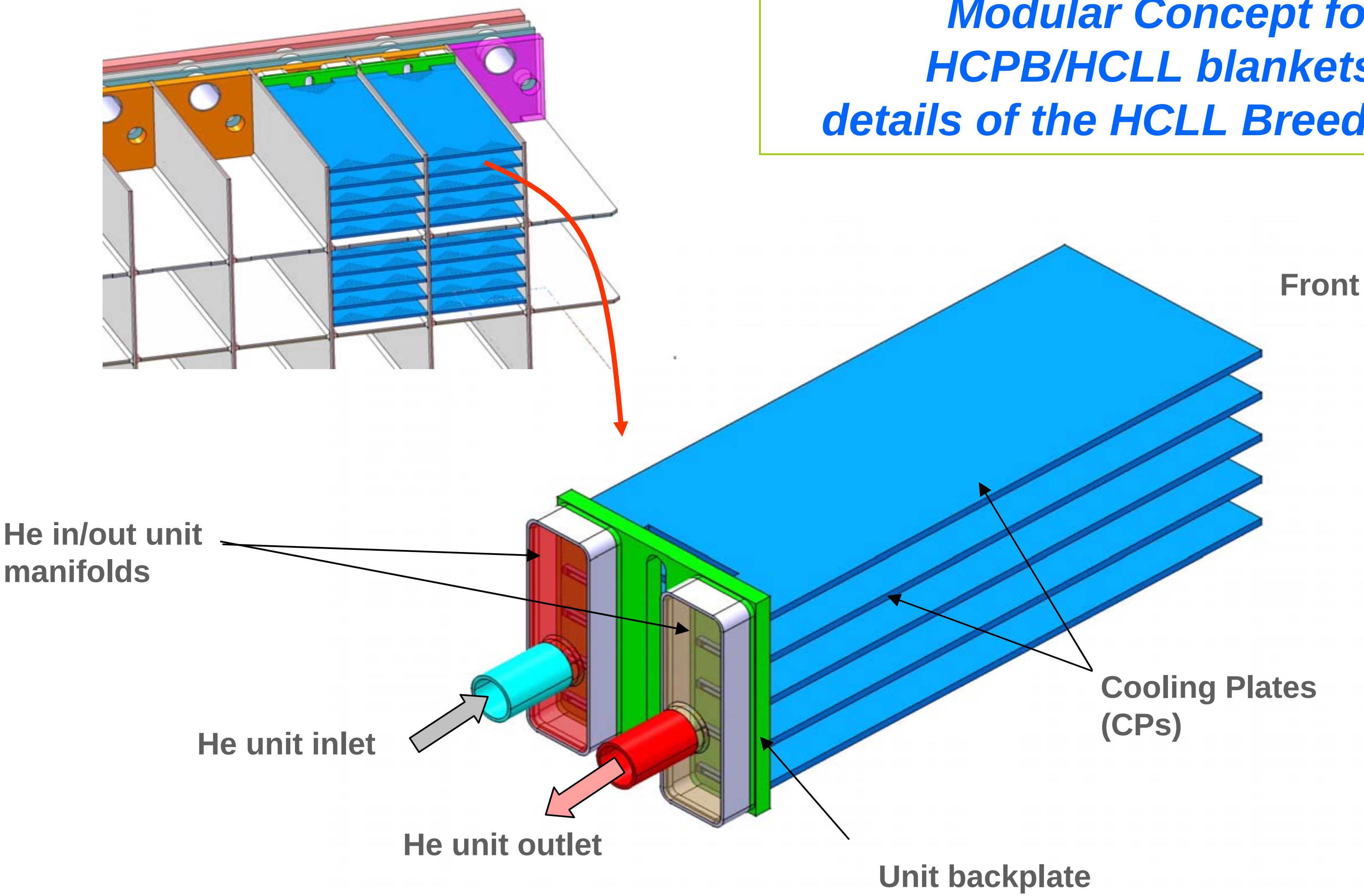




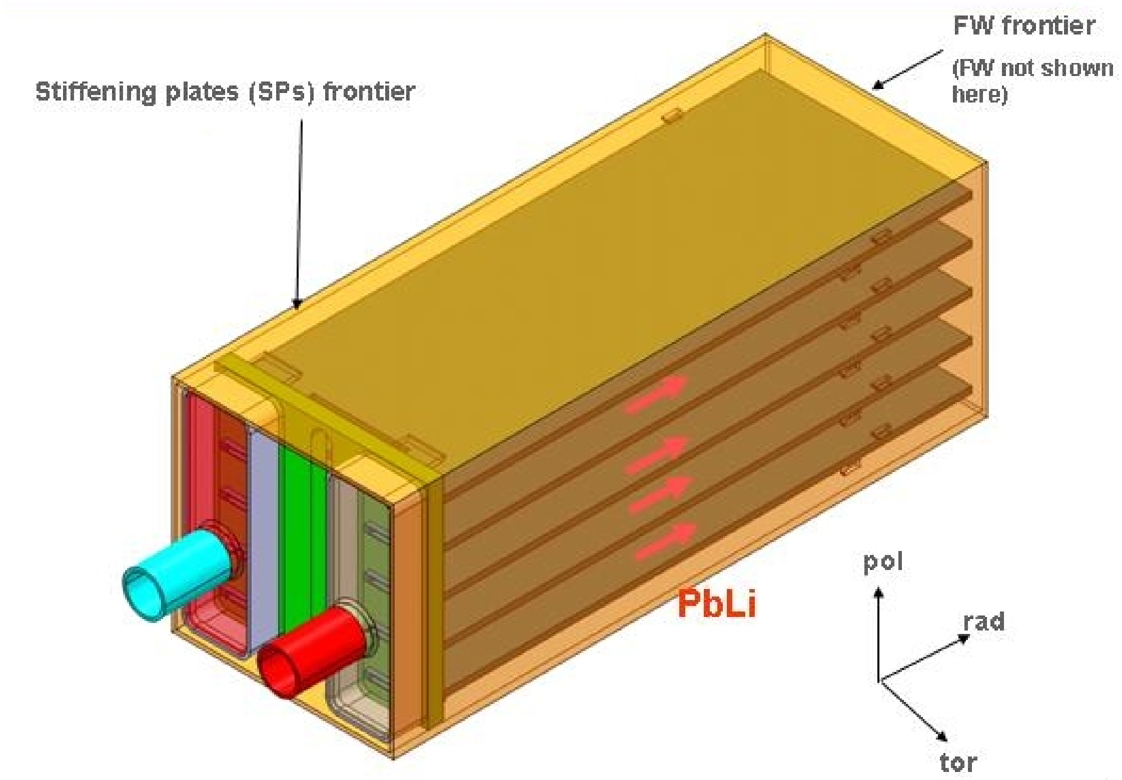
Breeder : Li_4SiO_4 (^6Li 30%)
Blanket thickness (inb/outb): 41/51 cm
Tritium Breeding Ratio: 1.12
Neutron multiplication: 1.78
Deposited Nuclear Power
- *whole blanket: 3000 MW*
- *Vacuum Vessel & Shield: 310 MW*

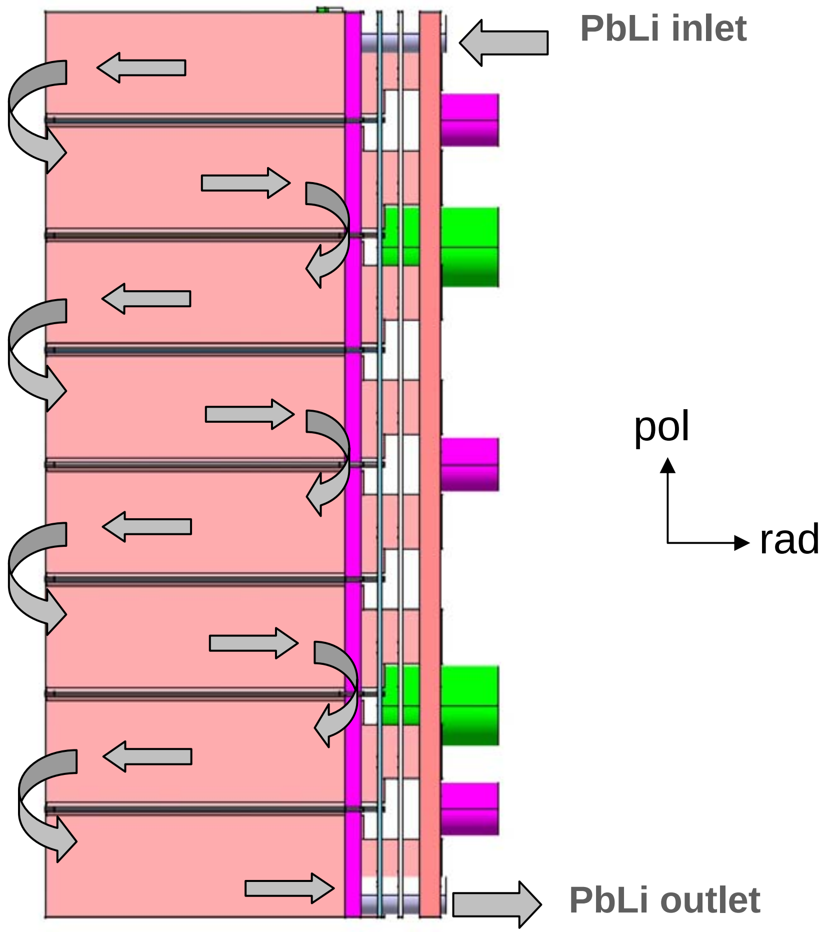
Energy multiplication: 1.38

Modular Concept for HCPB/HCLL blankets : details of the HCLL Breeder Unit



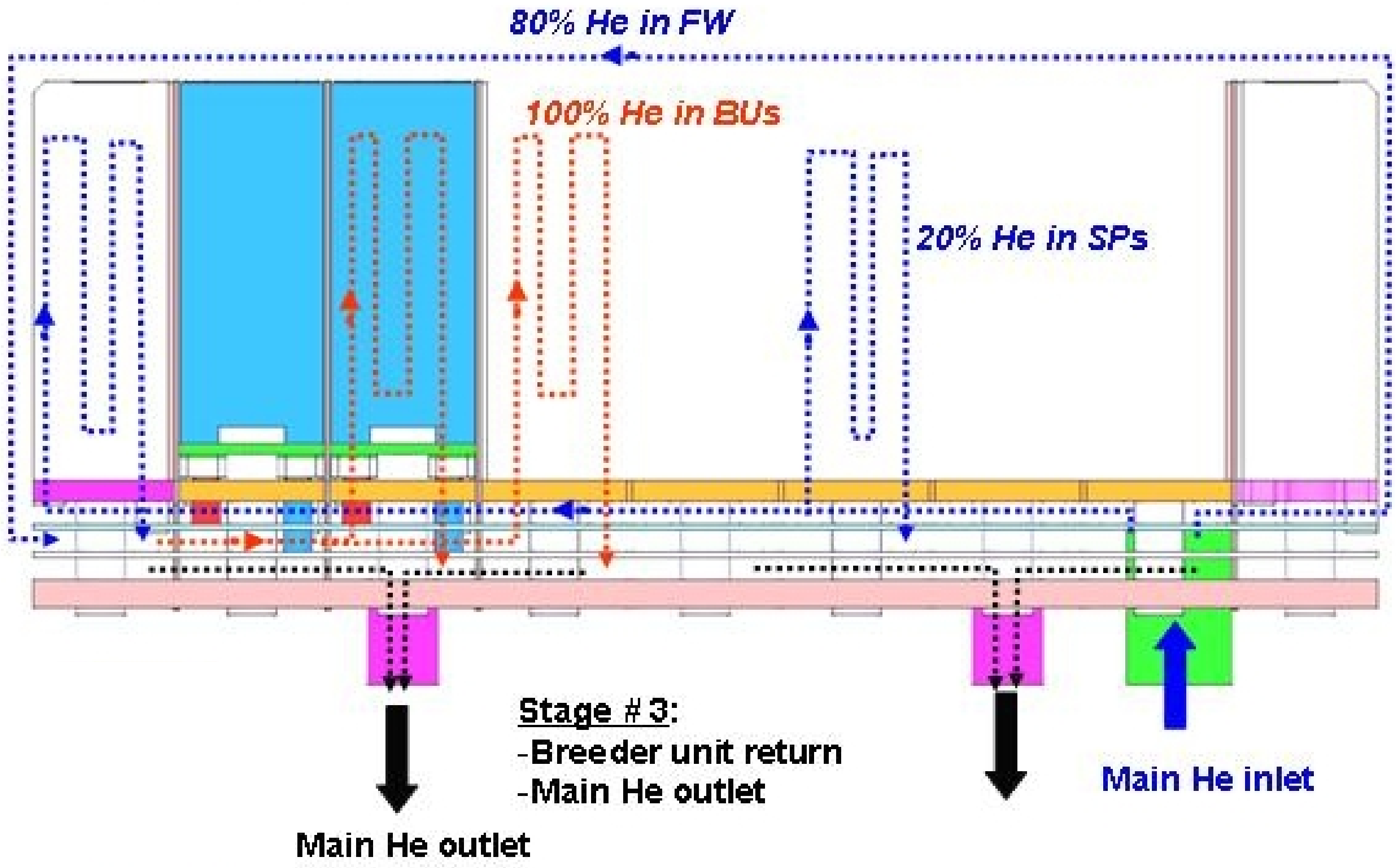
HCLL Breeder Unit & Module: LiPb flow path



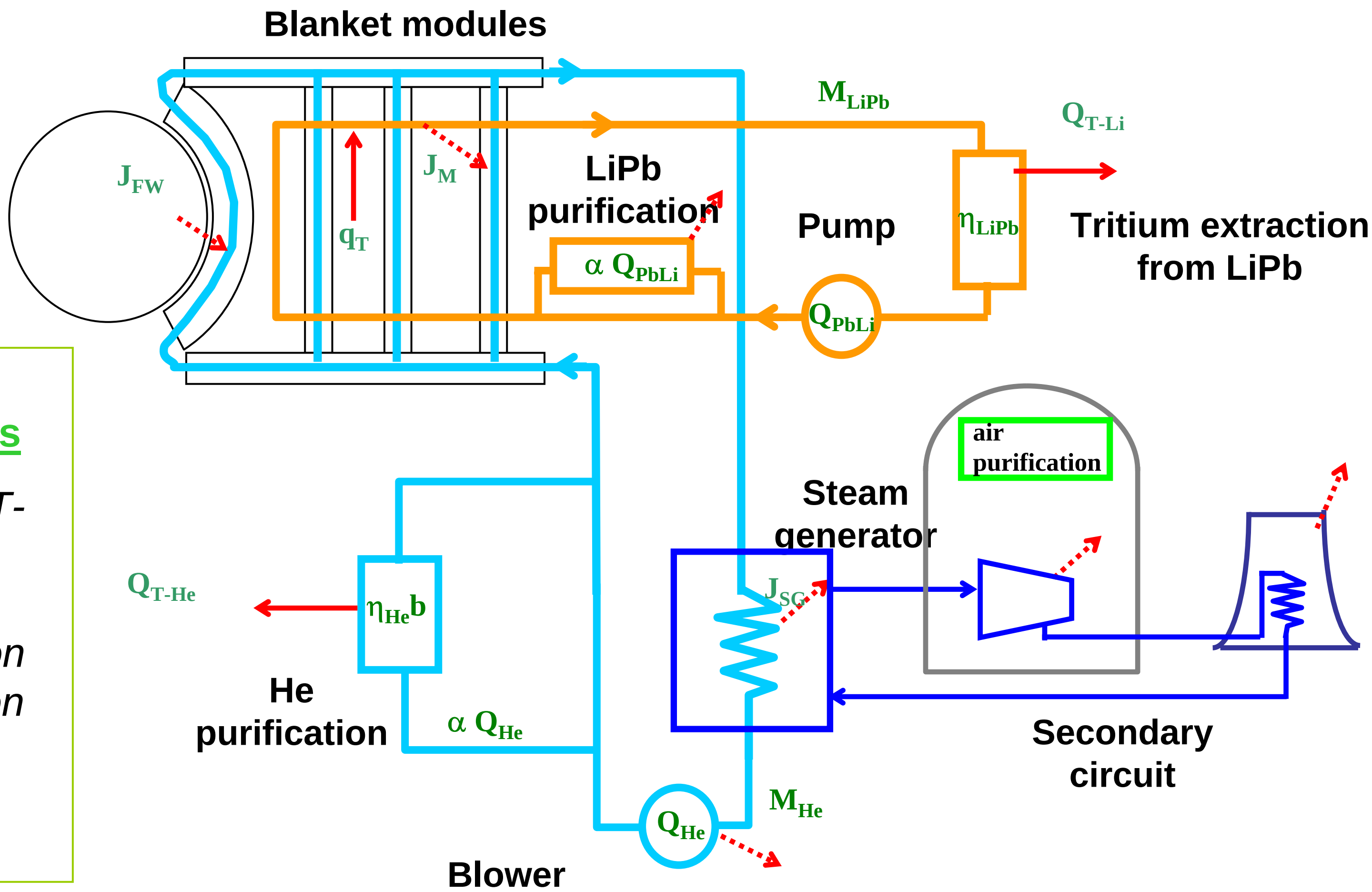


HCLL Module: Helium flow path

(Module top view)



Scheme for Tritium control in blanket modules and associated circuits: List of T-sources, possible T-permeation flux and leakages)



Some Questions

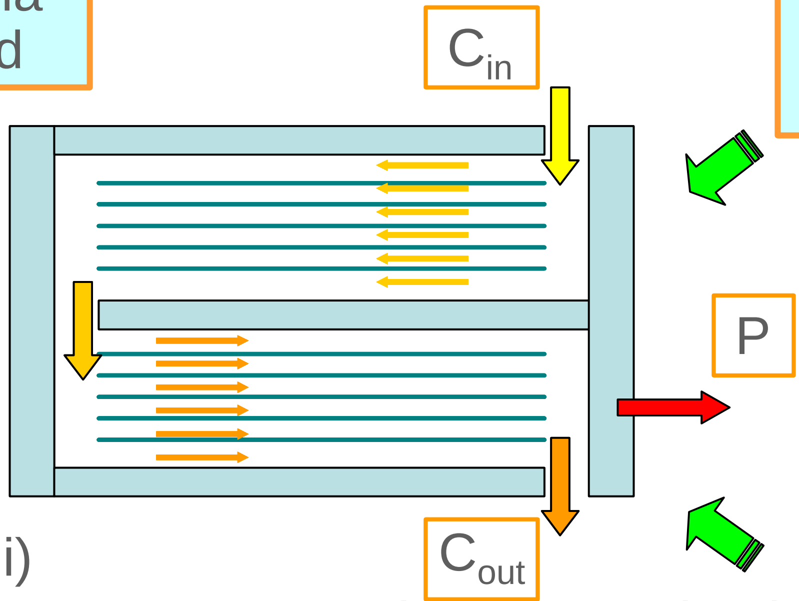
- ➔ *Need of T-permeation barriers?*
- ➔ *Extraction & purification efficiency?*
- ➔ *He-chemistry?*

Blanket Concept AB : He-Cooled Lithium-Lead

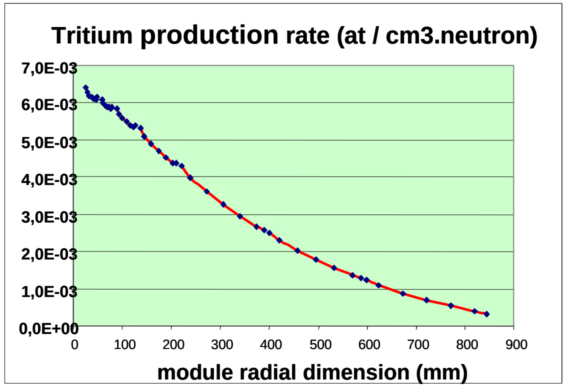
LiPb flow and T convection via cast3m_fluid

T diffusion and permeation via cas3m_fluid (heat-like)

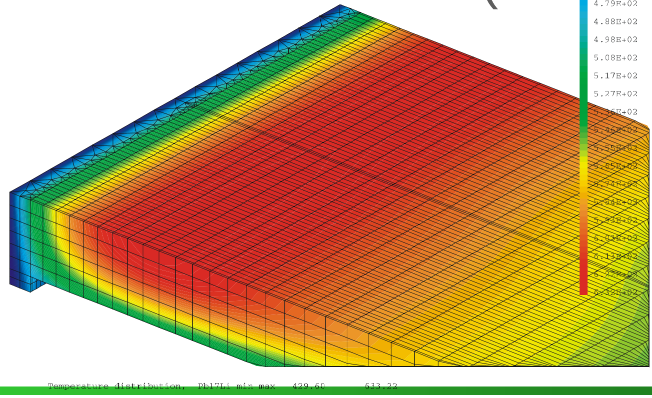
Computation procedure to be used for the estimation of T-permeation in a blanket module



Neutron code (Tripoli)



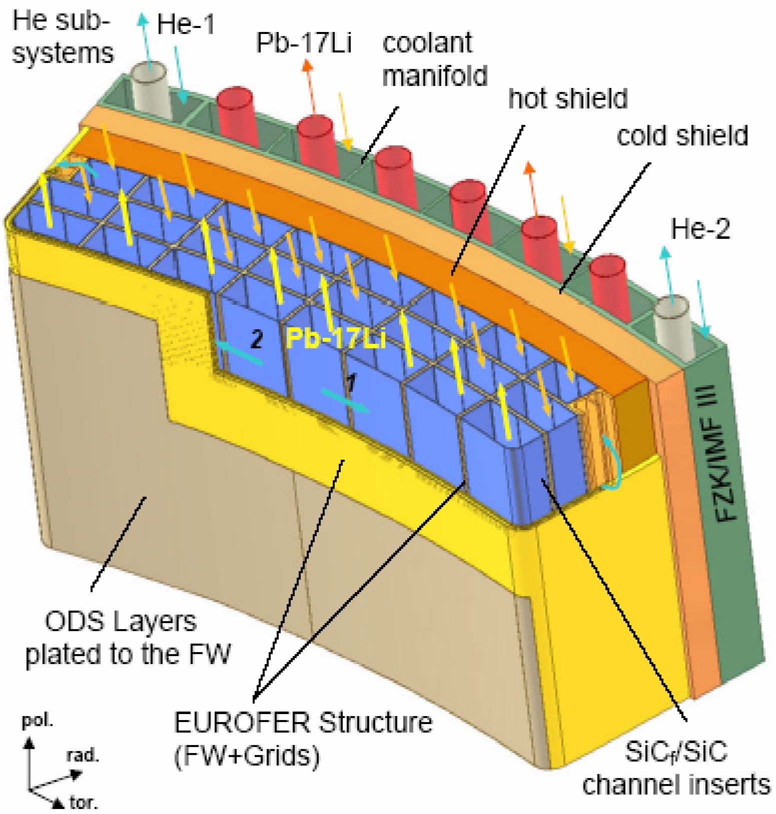
Thermal-mechanical code (cast3m)



Design Point for HCPB and HCLL blankets (selected for ITER EU-TBMs)

		HCLL	HCPB
TBR [breeder zone thickness]		1.15 [55 cm]	1.14 [46 cm]
Coolant (He): T_{min} - T_{max}	°C	300-500	300-500
Coolant passes Temperature:	°C	- FW: 300– 410 - BU: 410 – 500	- FW: 300 – 363 - Grid: 363 – 400 - BU: 400 – 500
Coolant (He): pressure	MPa	8	8
Breeder [^{6}Li enrichment]		$PbLi_{eu}$ [90%]	Li_4SiO_4 [40%] or Li_2TiO_3 [~70%]
Multiplier		-	Be
EUROFER: T_{min} - T_{max}	°C	300-550	300-550
Breeder: T_{min} - T_{max}	°C	400-660	400-920
Multiplier: T_{min} - T_{max}	°C	-	400-650

Blanket Design Scheme and Main Features

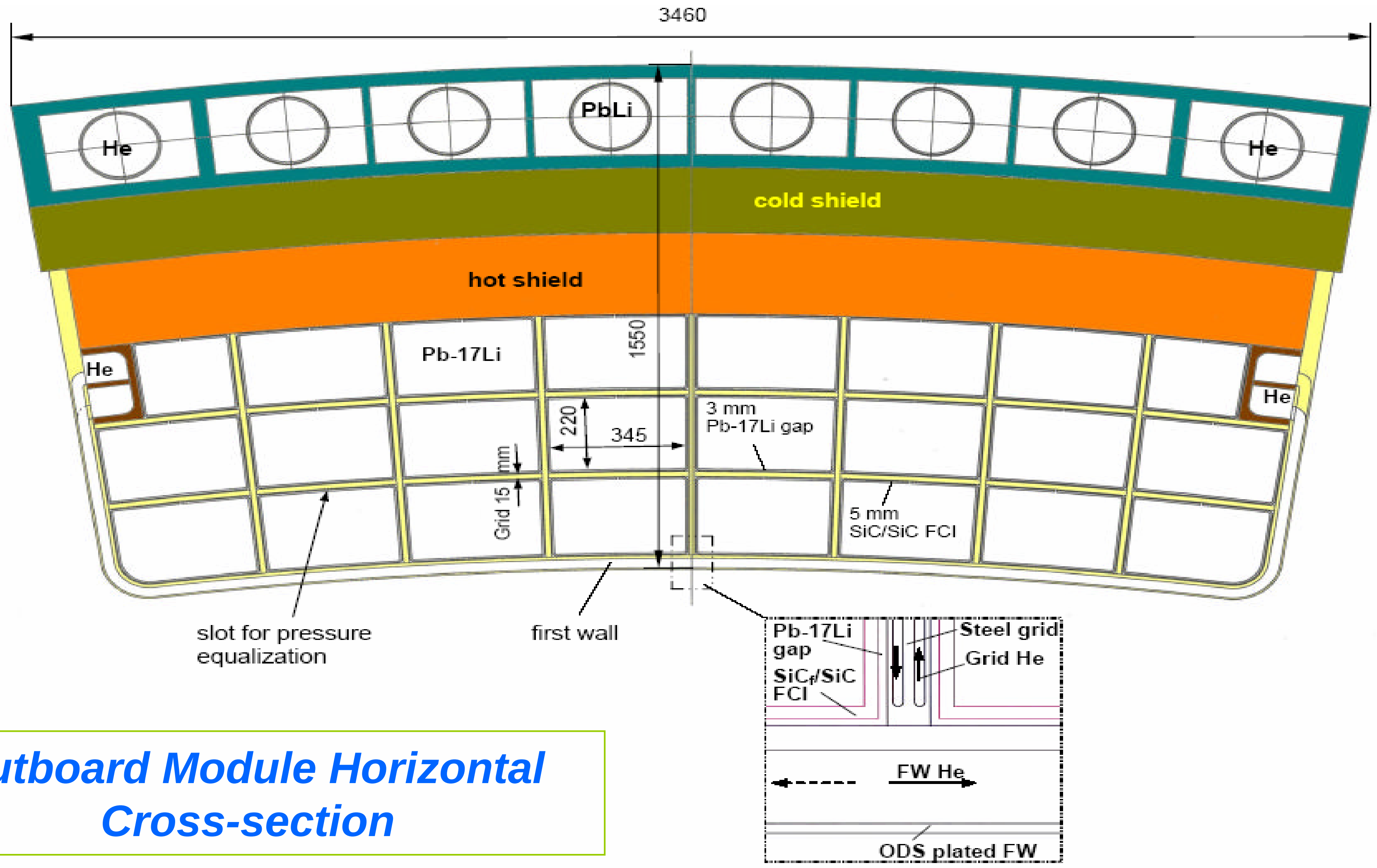


Main features:

- helium-cooled RAFM steel structures (EUROFER)
- ODS plated FW to use the high-temperature strength of ODS
- self-cooled breeding zone with Pb17Li as breeder and coolant
- SiC_f/SiC flow channel inserts as electrical (MHD) and thermal insulators leading to high exit temperature and high thermal efficiency

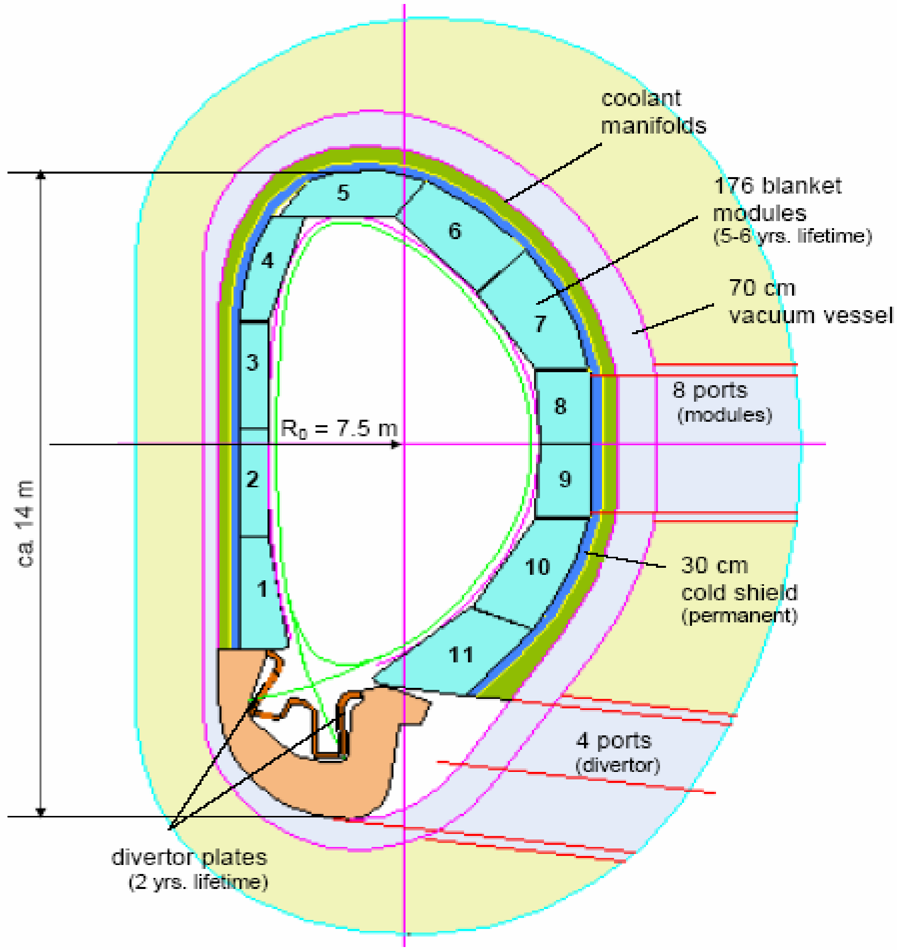
Dual Coolants	T_{Inlet} (°C)	T_{Outlet} (°C)	ΔT (K)
Helium (8 MPa)			
Overall blanket	300	480	180
FW	300	450	150
Grids	450	480	30
Pb-17Li	480	700	220

Blanket Concept C : Dual-Coolant (He & LiPb)



Outboard Module Horizontal Cross-section

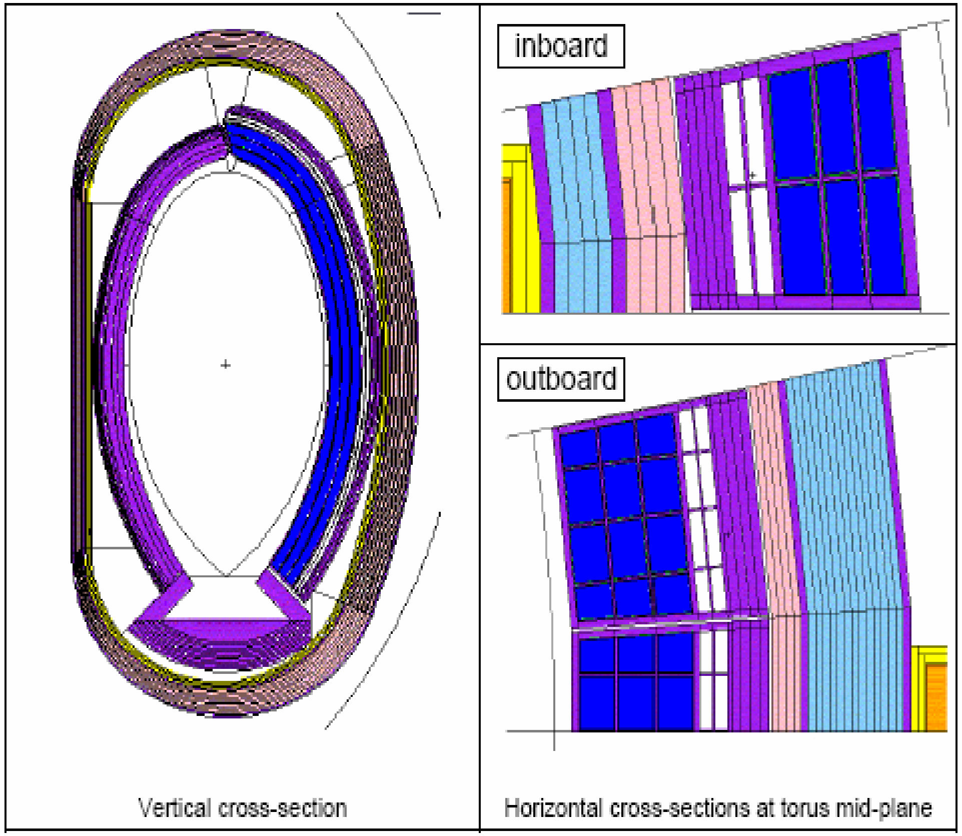
Poloidal Segmentation and Radial Built



Inboard Thickness [cm]	Cumulative	Outboard Thickness [cm]	Cumulative	Material	Component
4.4	4.4	4.4	4.4	Eurofer (0.45)	first wall
0.5	4.9	0.5	4.9	SiC_f/SiC	FCI
11.5	16.4	24.1	29	Pb-17Li	breeder/coolant
0.5	16.9	0.5	29.5	SiC_f/SiC	FCI
1.5	18.4	1.5	31	Eurofer	structure
0.5	18.9	0.5	31.5	SiC_f/SiC	FCI
11.3	30.2	23.5	55	Pb-17Li	breeder/coolant
0.5	30.7	0.5	55.5	SiC_f/SiC	FCI
1.5	32.2	1.5	57	Eurofer	structure
0.5	32.7	0.5	57.5	SiC_f/SiC	FCI
14.3	47	23.5	81	Pb-17Li	breeder/coolant
0.5	47.5	0.5	81.5	SiC_f/SiC	FCI
3	50.5	4	85.5	Eurofer	structure
6	56.5	9	94.5	He	He in/out
1.5	58	1.5	96	Eurofer	structure
6	64	9	105	He	He in/out
3	67	3	108	Eurofer	structure
13	80	25	133	Eurofer	HT shield
30	110	30	163	0.6 Eurofer/0.4 water	LT shield
35	145	75	238	steel/borated water	vacuum vessel

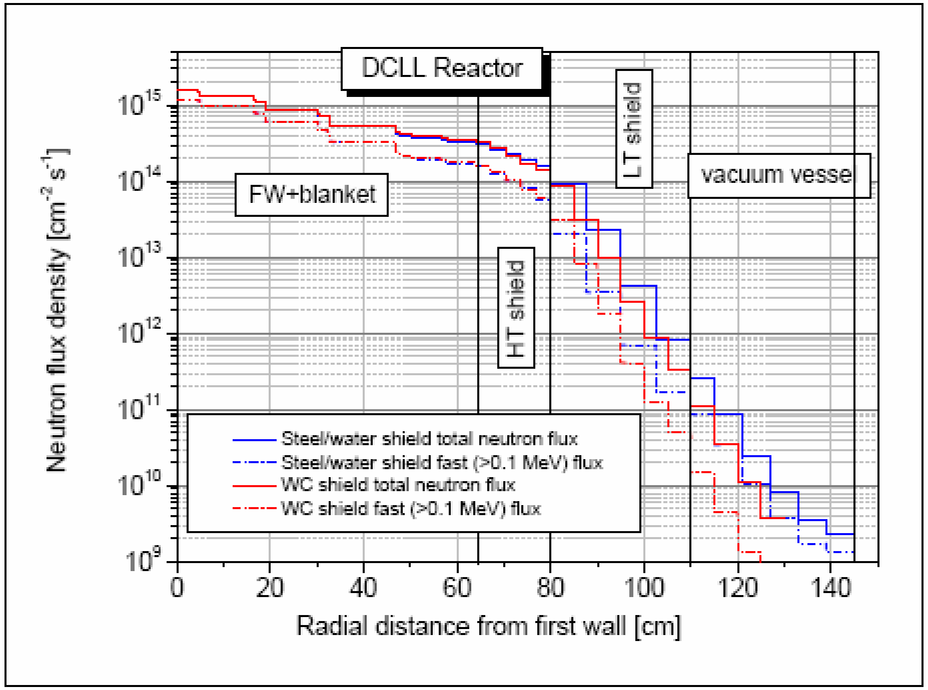
Blanket Concept C : Dual-Coolant (He & LiPb)

Used Neutronic 3D Model and Main Results



TBR = 1.15

(90% ^{6}Li, inb/outb breeder zone thk: 50.5/85.5 cm)



Neutron Flux Profile in Inboard mid-plane

Blanket Concept D : SiC/SiC - LiPb Self-Cooled

Design Rational *(maximization of efficiency & safety)*

- Structural Material: SiC/SiC (low afterheat, high temperature)
- Blanket working principle: co-axial Pb-17Li flow to maximize outlet tempertaure
- Pb-17Li external circuits location: horizontal deployment, to minimize pressure
- Blanket remote maintenance procedure: segment geometry, maintained through vertical ports after Pb-17Li draining
- Potential reduction of waste quantities: separation of outboard blanket in two zones in order to minimize radioactive waste (depending on lifetime evaluations)
- Magnet system: possibility of high T superconductors (77 K)
- Plasma heating system: acknowledge of need but not evaluated
- Advanced conversion systems: potential for H-production

Blanket Concept D : SiC/SiC - LiPb Self-Cooled

Assumed SiC/SiC Properties

Key SiC_f/SiC Properties and Parameters *	Assumed values [1] in the design analyses	Typical measured value
Density	~ 3000 kg/m^3	~ 2500 kg/m^3
Porosity	~ 5%	~ 10%
Young's Modulus	200-300 GPa	~ 200 GPa
Poisson's ratio	0.16-0.18	0.18
Thermal Expansion Coefficient	~ 4 x 10^{-6}/°C	4 x 10^{-6}/°C
Specific heat	190 J/kg-K	190 J/kg-K
Thermal Conductivity in Plane (1000°C)	~ 20 W/m-K (EOL)	~ 15 W/m-K (BOL)
Thermal Conductivity through Thickness (1000°C)	~ 20 W/m-K (EOL)	~ 7.5 W/m-K (BOL)
Electrical Conductivity	~ 500 /Ωm (under irradiation, EOL value)	~ 500 /Ωm (before irradiation)
Tensile Strength Trans-laminar Shear Strength Inter-laminar Shear Strength	300 MPa - -	300 MPa 200 MPa 44 MPa
Maximum allowable tensile Stress	Not used*	Unknown*
Max. Allowable Temperature (Irradiation Swelling basis)	~ 1000 °C	~ 1000 °C
Maximum Allowable Interface Temperature with breeder	~ 1000°C (flowing)	~ 800°C (static)
Min. Allowable Temperature (Thermal conductivity basis)	~ 600 °C	~ 600 °C
Cost	= $400/kg	~ 10 times larger

Blanket Concept D : SiC/SiC - LiPb Self-Cooled

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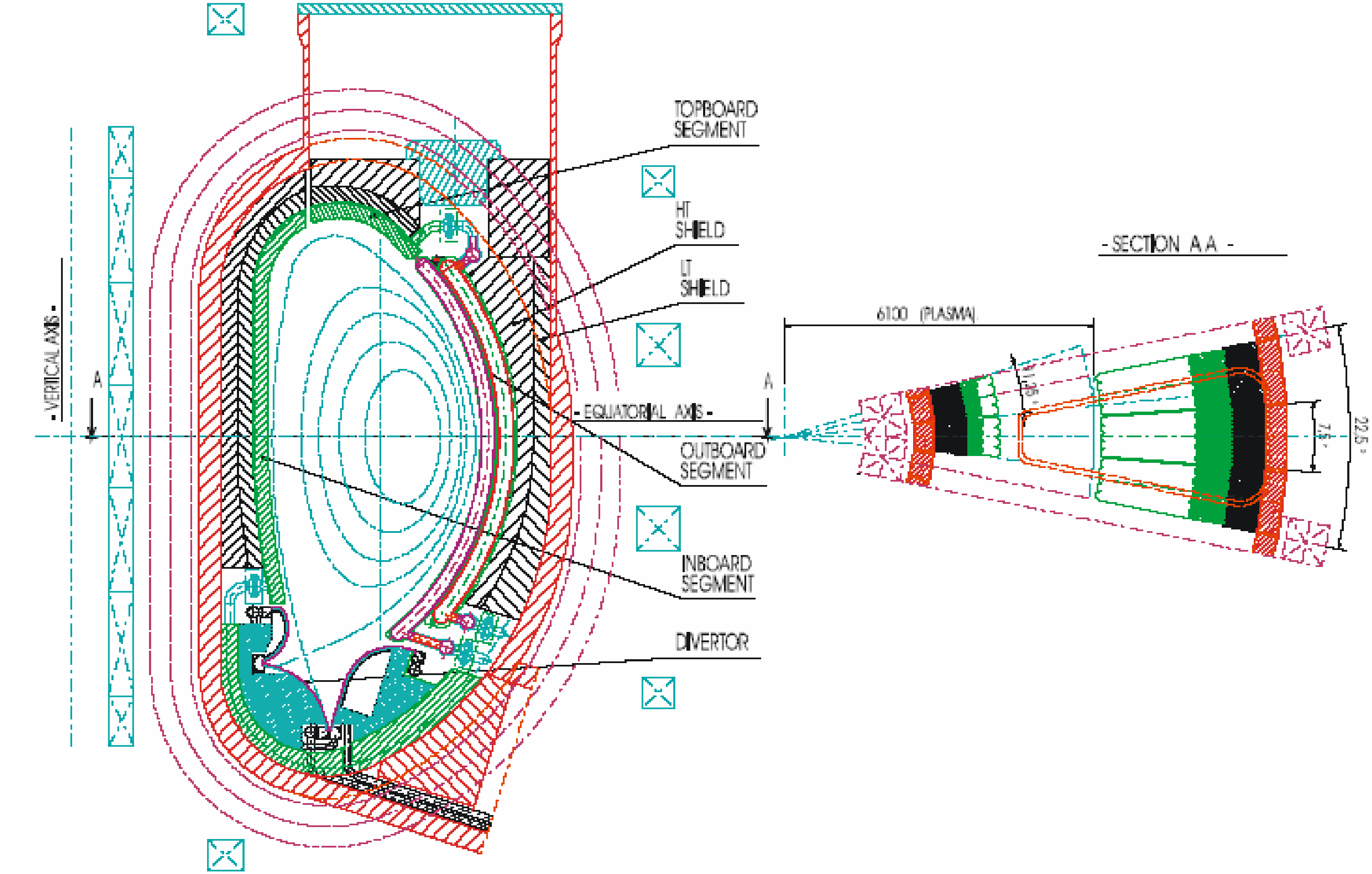
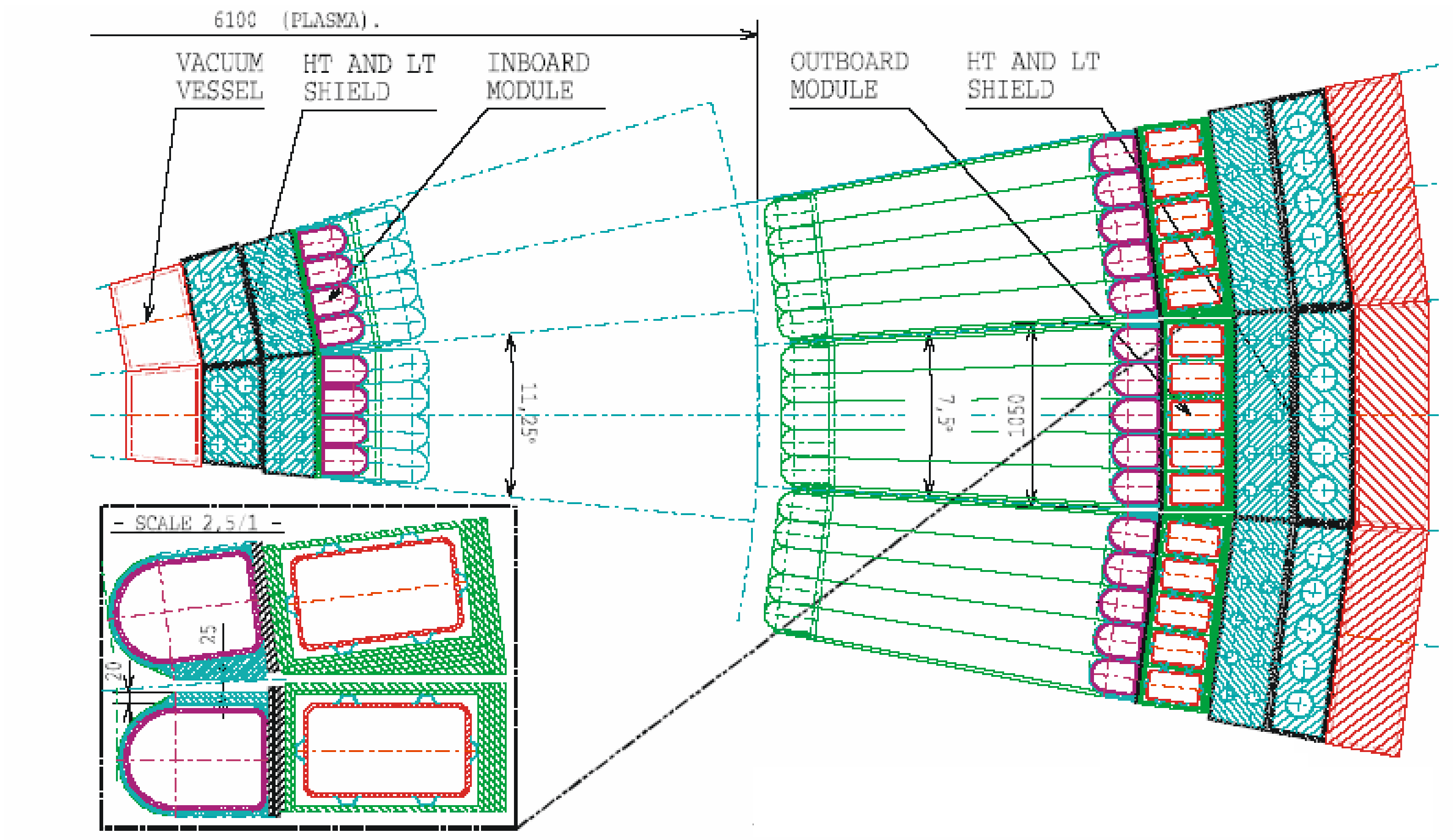
Design Scheme –Banana shaped Modules (poloidal Cross-Section)



Design Scheme –Banana shaped Modules (Horizontal Cross-Section)



Blanket Concept D : SiC/SiC - LiPb Self-Cooled

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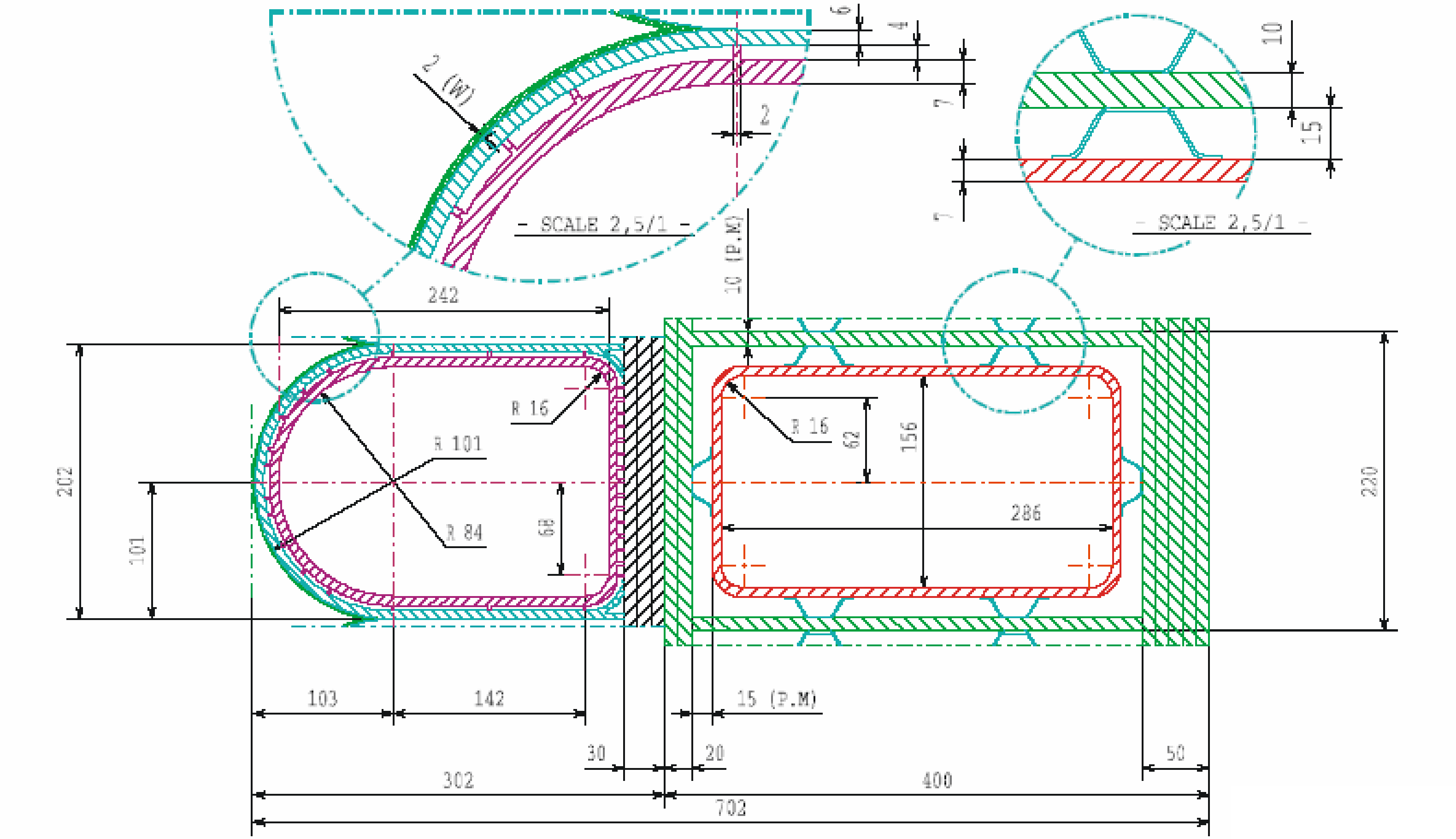
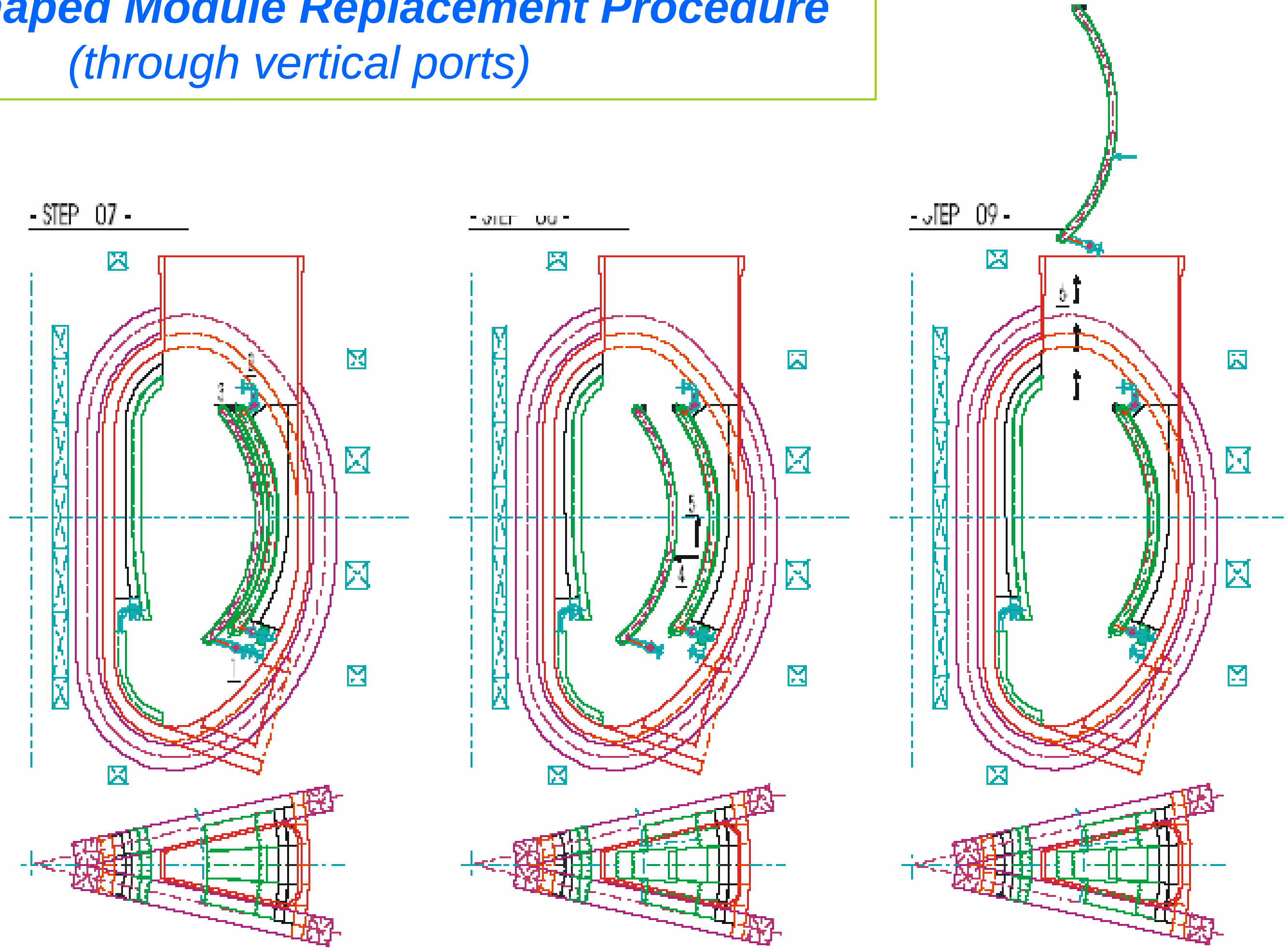
Design Scheme –Banana shaped Modules *(Outboard Module cross-section)*



Banana shaped Module Replacement Procedure

(through vertical ports)



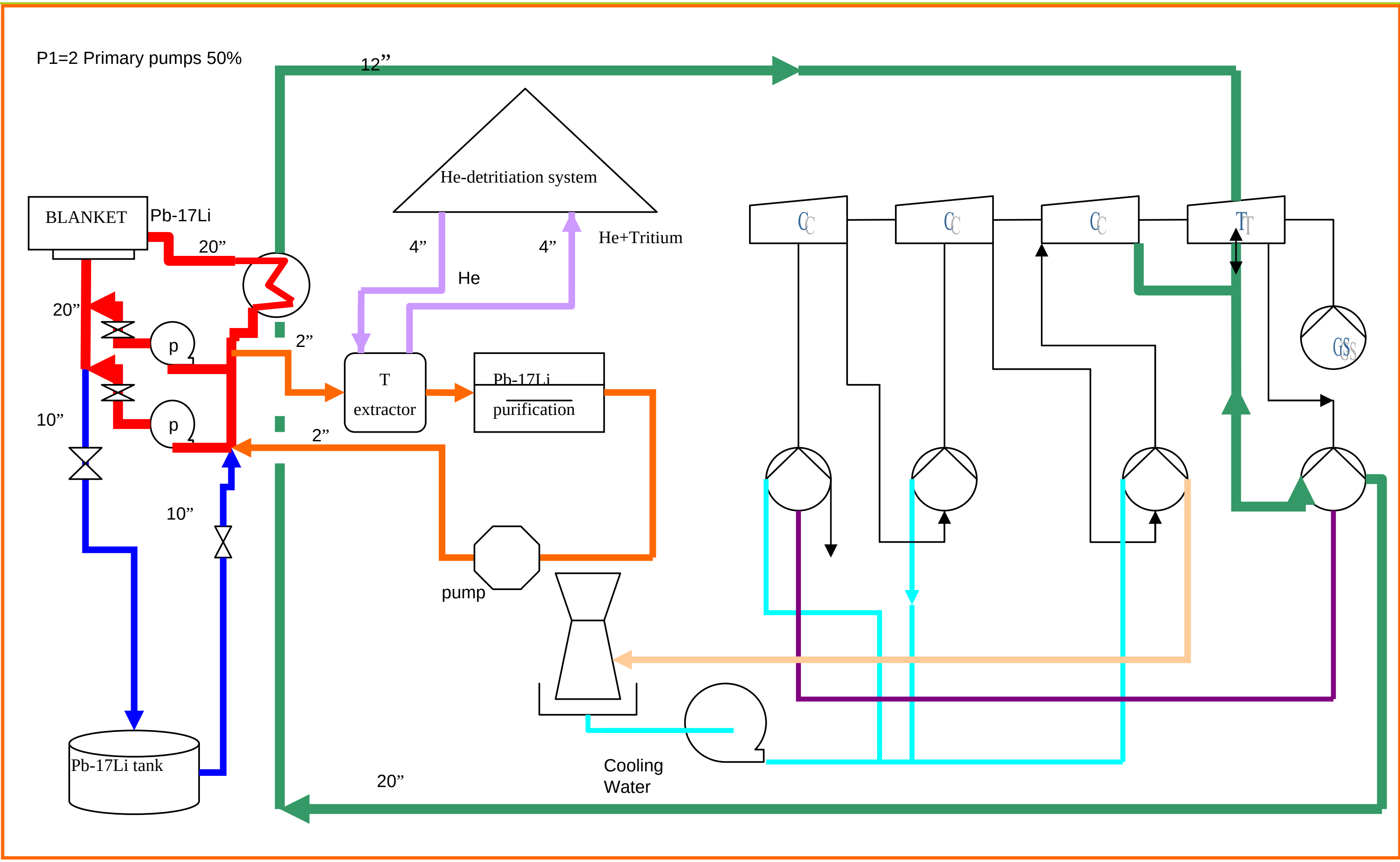
Blanket Design Point

- segment : SiC_f/SiC box acting as Pb-17Li container, 2 mm W protection
- use of 3D (industrial) SiC_f/SiC 3D , extensive use of brasing
- Pb-17Li: $T_{inlet/outlet}$ 700°C/1100°C *(div: 600/990°C)*, max. speed ~4.5 m/s
- dimensions outboard (front) segments: 5 modules h=8m, w=0.3m, thk=0.3m
- High T Shield: Pb-17Li-cooled WC, SiC_f/SiC structures (energy recovered)
- Low T Shield: He-cooled WC with B-steel structures (as for VV)

Main Analyses Results

- *Neutronics : TBR = 1.12 (0.98 blankets, 0.125 divertor, 0.015 HT shield)*
- *Thermal results :* T_{max} *~1100°C (if* λ *=20 W/mK, thk. FW 6 mm)*
- *Acceptable stresses (see criteria details) for Hydr. pressure = 0.8 MPa*
- *Acceptable MHD pressure drops*
- *1 independent cooling circuit for divertor, 4 for blanket,* ε *= 61 % (*P_{el}/P_{fus}*)*
- *T-extraction performed on the cold leg (Pb-17Li renewal ~1100 times/day)*

Scheme of the LiPB External Circuits–Banana shaped Modules



PPCS Breeding Blankets Concepts

Main Blanket R&D Issues

Short Term Concepts (for models A, B, AB)

- ✓ Further development of EUROFER (irrad.)
- ✓ Further development of structures manufacturing process (HIP, welds, etc…)
- ✓ Supporting systems, collectors and piping routes, remote mounting/dismantling
- ✓ Interaction LiPb/water & DWT (A)
- ✓ Tritium Management and Control (A, AB, B at lower extend)
- ✓ Permeation Coatings (A and AB) (irrad.)
- ✓ LiPb compatibility with EUROFER (A, AB)
- ✓ Ceramic and Be behavior under irradiation, T-inventory in Be (B)
- ✓ Pebbles beds behavior (B)

➔ *Promising on-going R&D*

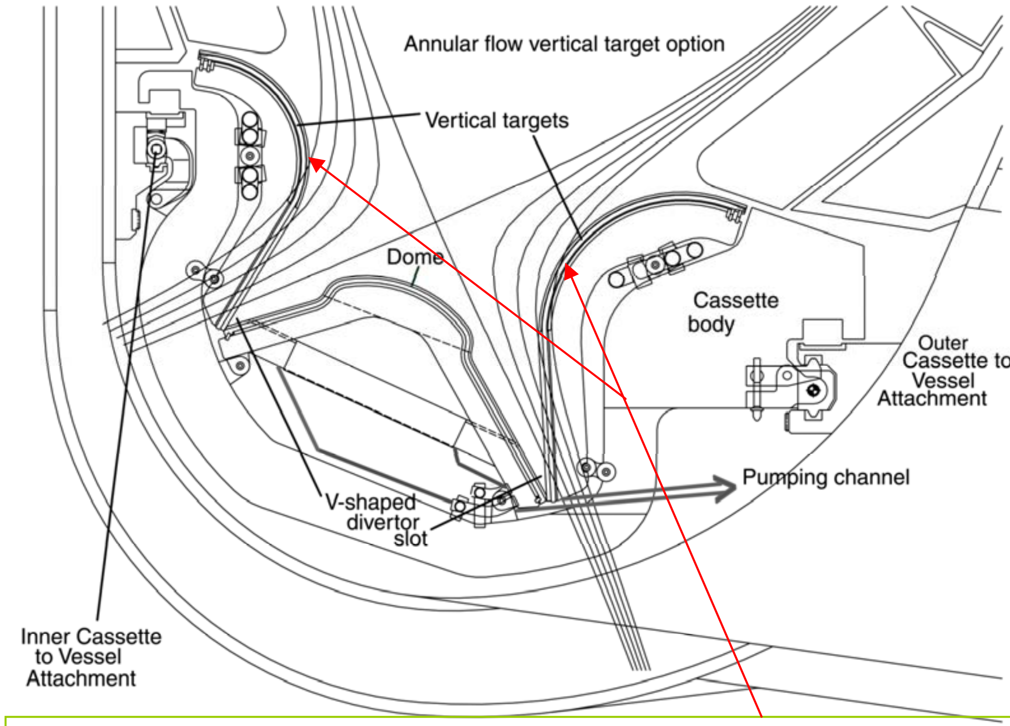
Medium Term Concept (model C)

- ✓ SiC/SiC related issues: FCI design (out of the main body), irrradiation, compatibility LiPb
- ✓ MHD: modeling of 3D inertial flow in expansion
- ✓ ODS: fabrication of ODS plated FW, irradiation
- ✓ Heat Exchanger Technology for High T LiPb
- ✓ T-recovery and purification (high LiPb flowrate)
- ✓ Integration aspects as for Short Term concepts

Long Term Concept (model D)

- ✓ SiC/SiC related issues requiring structural functions : irradiation (thermal conductivity, burn-up, lifetime), manufacturing (joints, pipe connections), reliability, modeling, compatibility with very high v &T LiPb
- ✓ MHD: modeling of 3D inertial flow in expansion
- ✓ T-recovery and purification (high LiPb flowrate)
- ✓ Heat Exchanger Technology and other components for High T LiPb
- ✓ Integration aspects as for Short Term concepts

PPCS Proposed Divertor Concepts

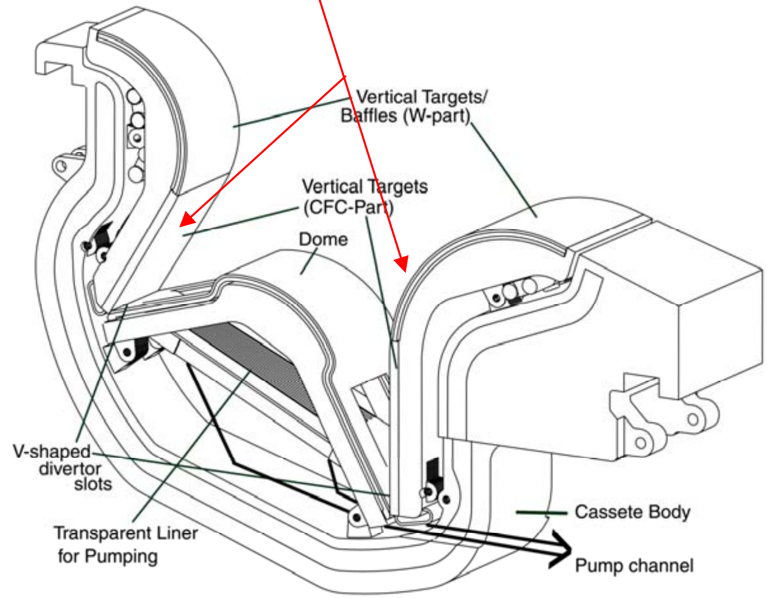


Main Functions *(ref. others speakers)*

- ✓ To control Plasma Boundary Conditions
- ✓ To divert magnetic lines to extract ashes and other impurities from the plasma

Vertical Targets are submitted to:

- ✓ very high heat fluxes (up to 15 MW/m^2)
- ✓ interaction with plasma ions/neutrals



➔ *Vertical Targets Designs*

✓ studied only for ITER, performances TBD
✓ *Additional requirements in a Power Plant*: account for neutron irradiation (~10 dpa/y), use of high T-coolant for acceptable power conversion efficiency, lifetime, …

➔***W-alloy appears to be the only acceptable armor material*** *(interaction with plasma)*

PPCS Divertor : Water-cooled concepts

Low-temperature concept

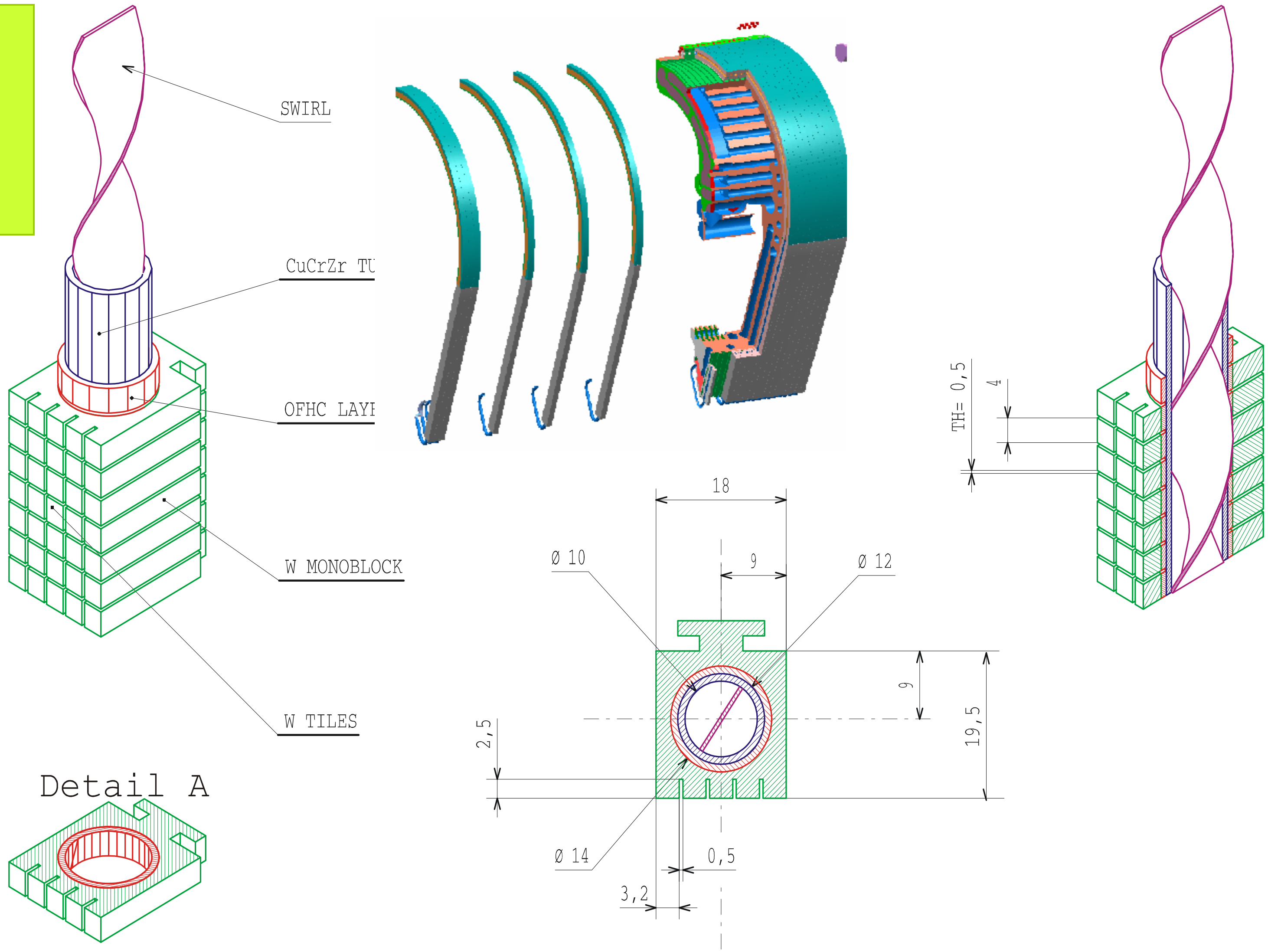
T_{inlet} = 150°C
Pressure : 4 MPa

Main features

- ✓ derived from ITER divertor studies
- ✓ W-alloy monoblocks
- ✓ CuCrZr tubes
- ✓ OFHC compliance layer
- ✓ CuCrZr Temp. limited to 400°C

Performances

Heat flux up to 15 MW/m²



PPCS Divertor : Water-cooled concepts

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High-temperature concept

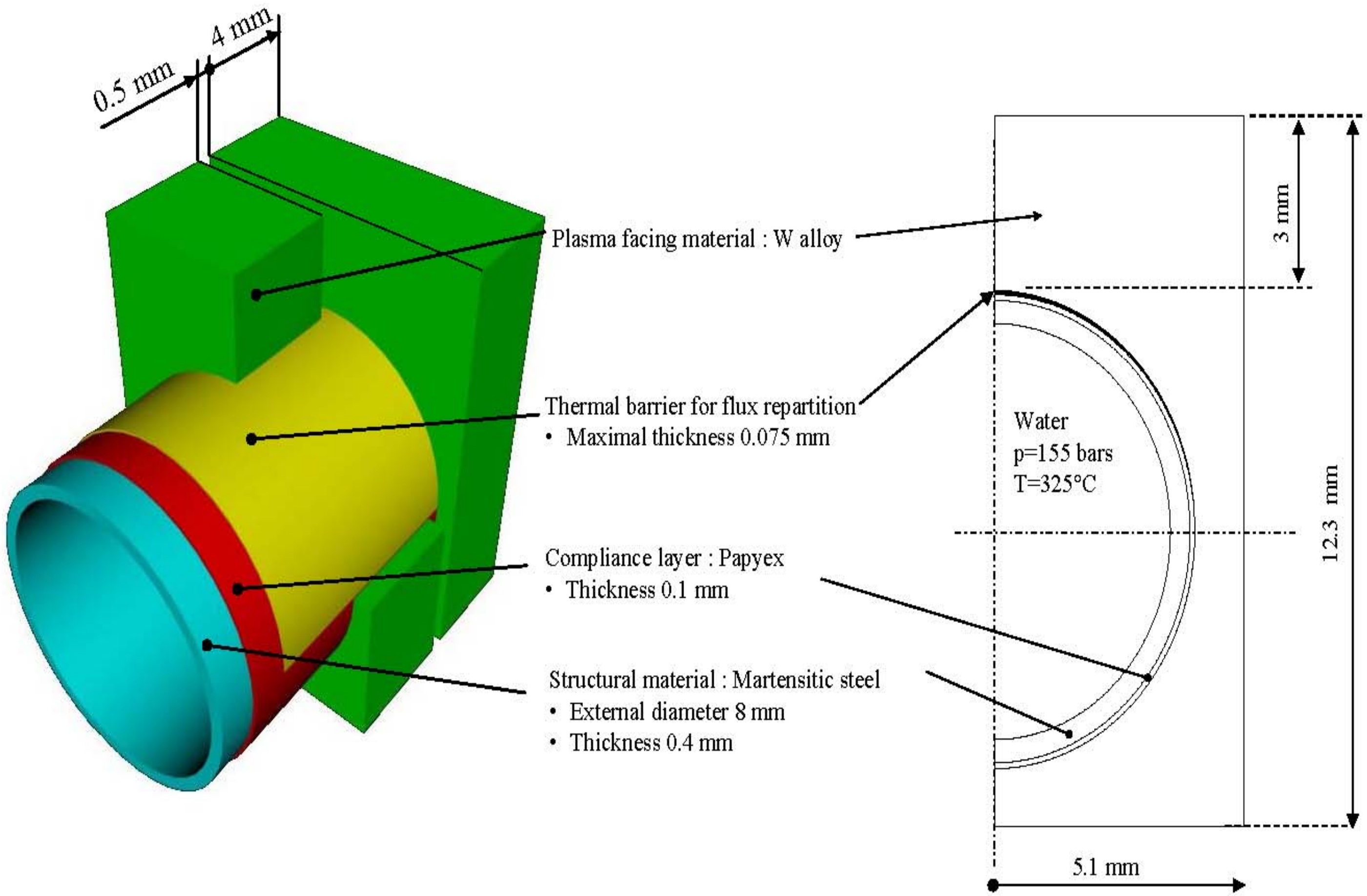
T_{outlet} = 325°C
Pressure : 15.5 MPa

Main features

- ✓ derived from low-T concept
- ✓ W-alloy monoblocks
- ✓ EUROFER tubes
- ✓ Papyex compliance layer
- ✓ Thermal barrier in the front part

Performances

Heat flux up to 15 MW/m²



PPCS Divertor : Helium-cooled concepts

Target-Plate Modular concept

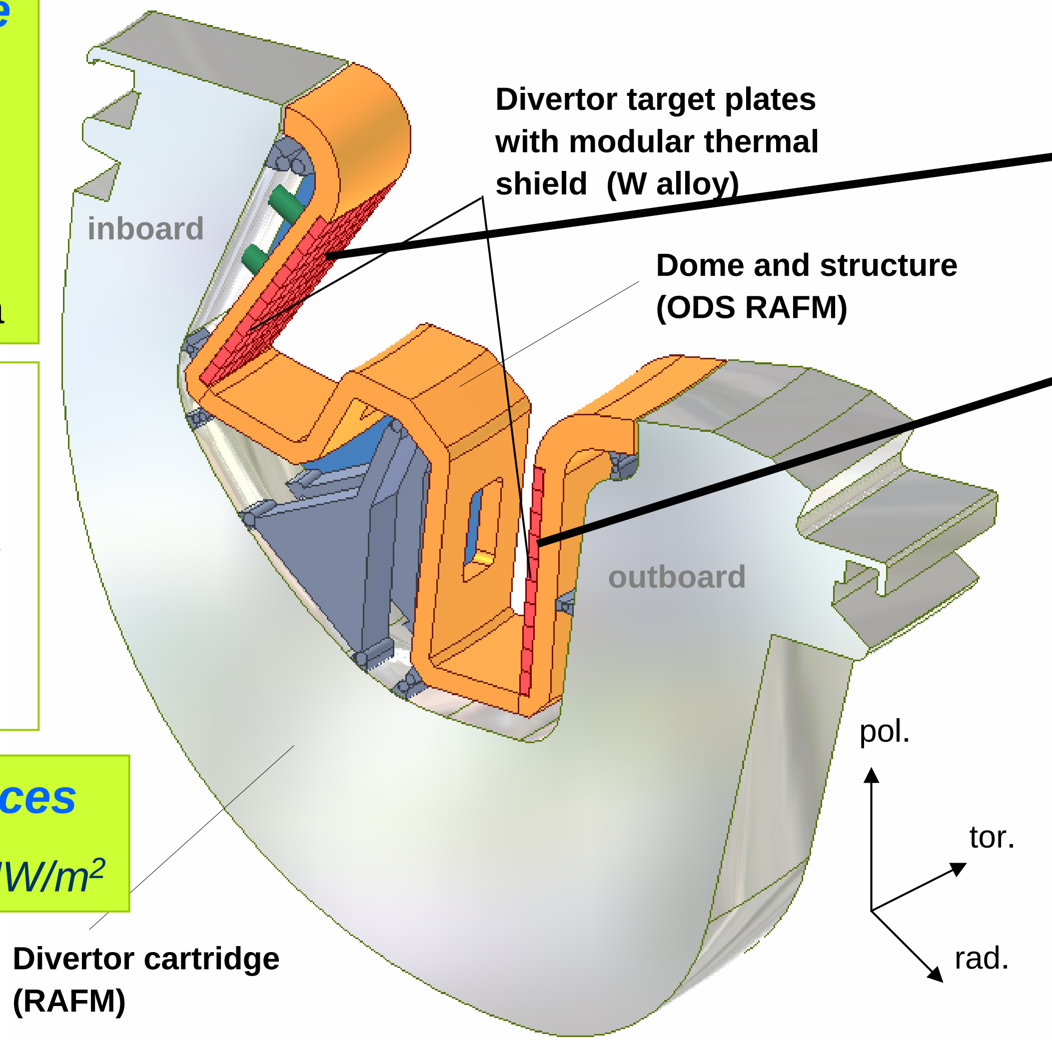
T_{inlet} = 600°C
T_{outlet} ~ 680°C
He P : 10 MPa

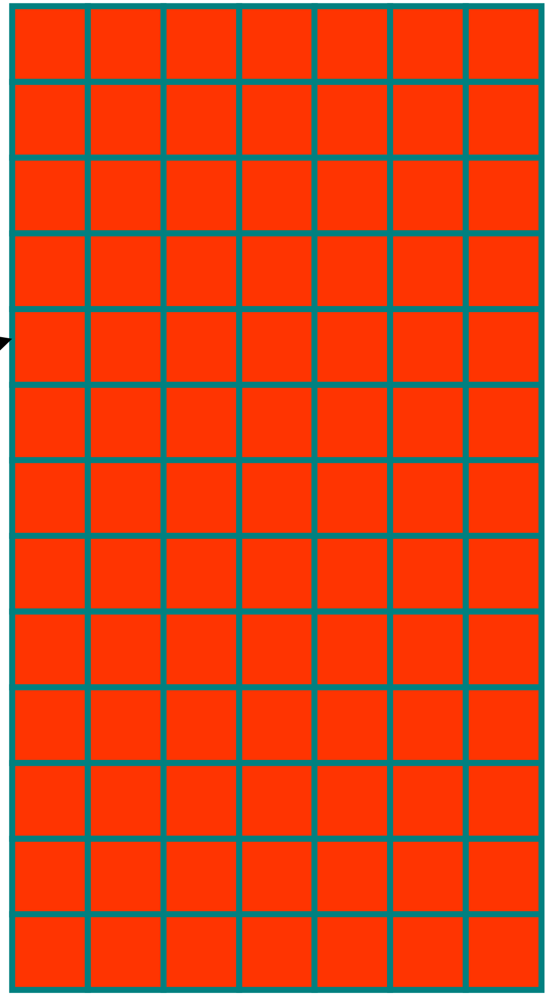
Main features

- ✓ W-alloy tiles
- ✓ W-alloy & ODS-steel structures

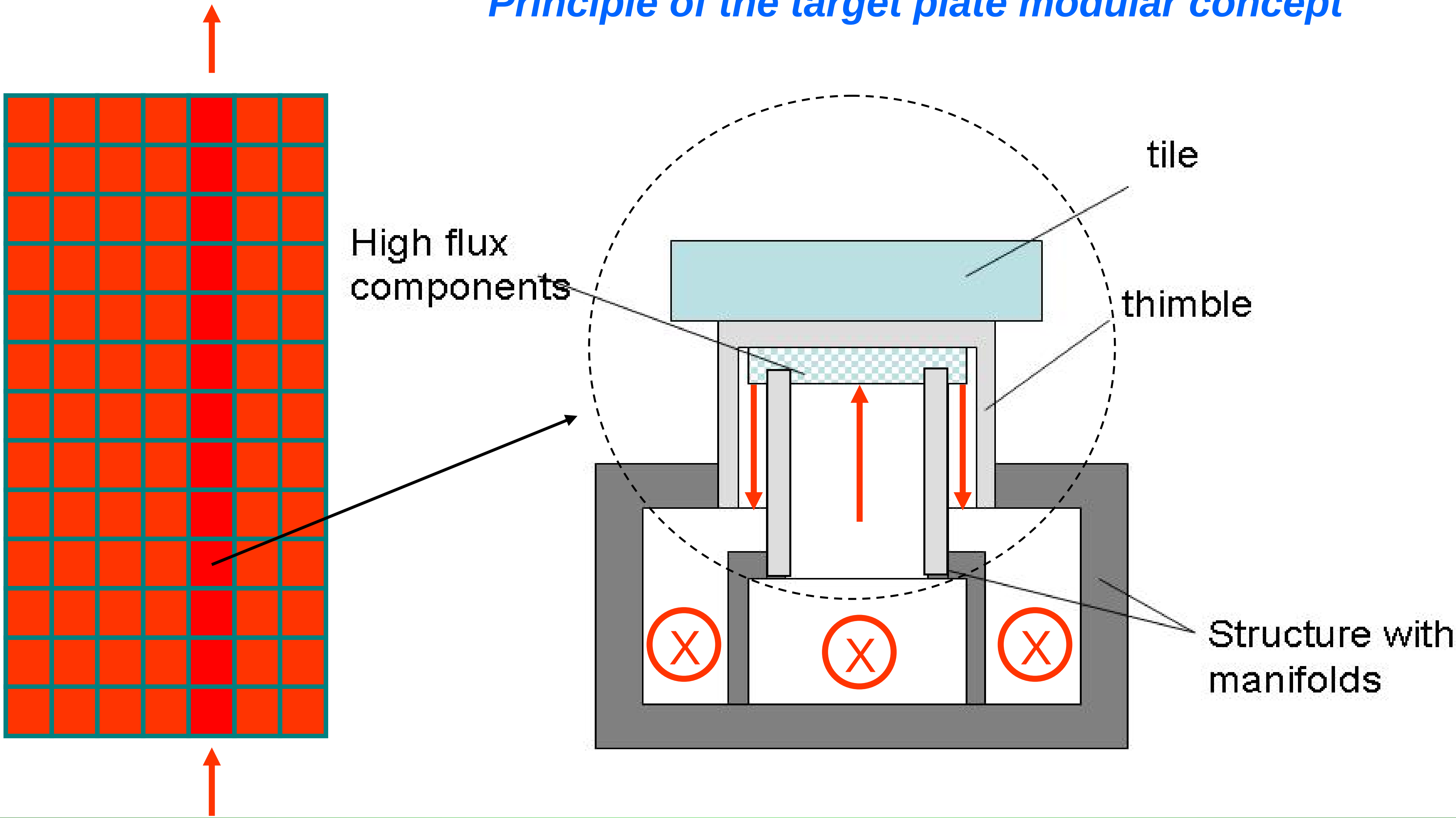
Performances

Heat flux 10 MW/m²

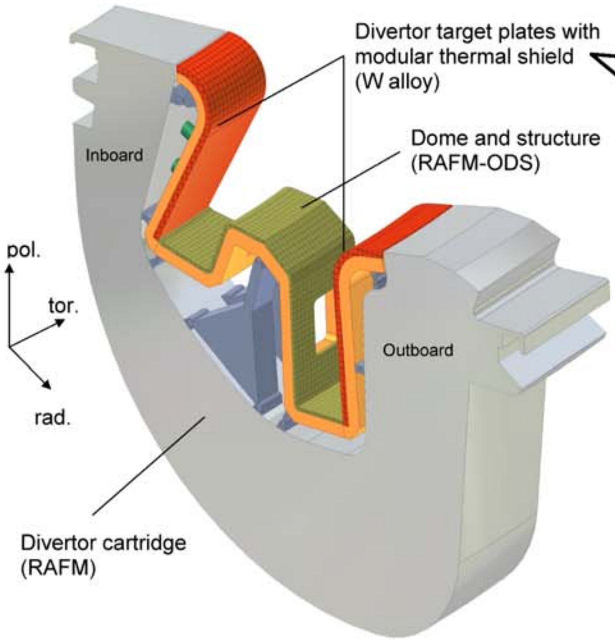


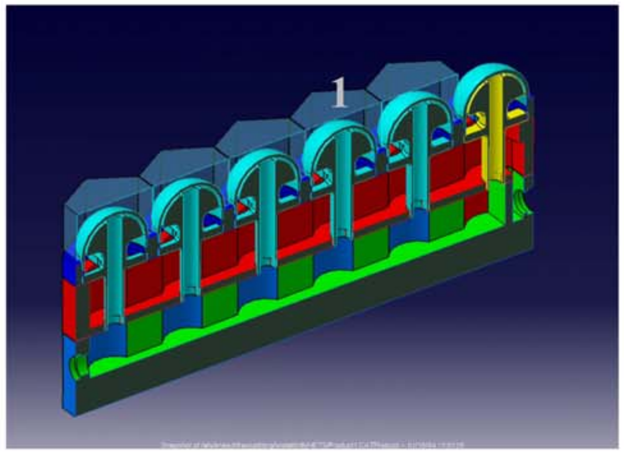


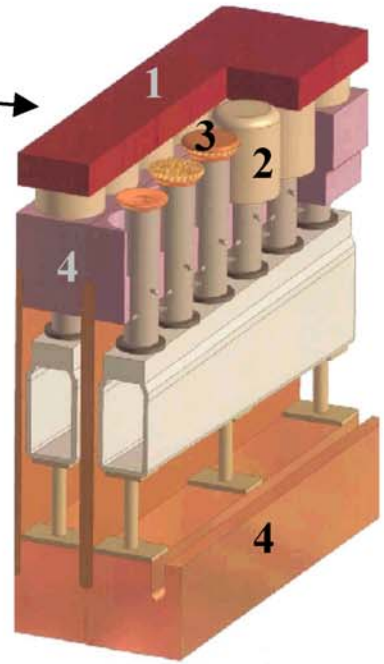
Principle of the target plate modular concept



Conceptual designs the vertical target and He-flow path







Various design principles:

HETS (ENEA)

Jet impingement

HEMP/HEMS (FZK)

***Flow promoter
(pin, slot arrays)***

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PPCS Divertor : Helium-cooled concepts

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Materials & Temp. Windows

Component	Material	Min Temp.	Max Temp.
Tiles	W	tbd (600°C) DBTT	2500°C Melting temp. 3410°C
High heat flux structure (high P He-containment)	W-alloy	600°C DBTT	1300°C re-crystallisation
Structure and manifolds	W-alloy ODS, ...	600°C DBTT 400°C DBTT	1300°C re-crystallisation 700-750°C strenght limits

Design Point

	Finger width [mm]	He T in/out [°C]	He pressure [MPa]	Mass flow [g/s]	Av. Heat Transfer Coeff. [kW/m^2 K]	Pressure drops [MPa]
HETS	**35**	**600-669**	**10**	**30**	**~ 30**	**0.063**
HEMP/ HEMS	**16**	**600-679**	**10**	**6.0**	**~ 30**	**0.11**

PPCS Divertor : LiPb-cooled (SiC/SiC) concept

Very high-T concept

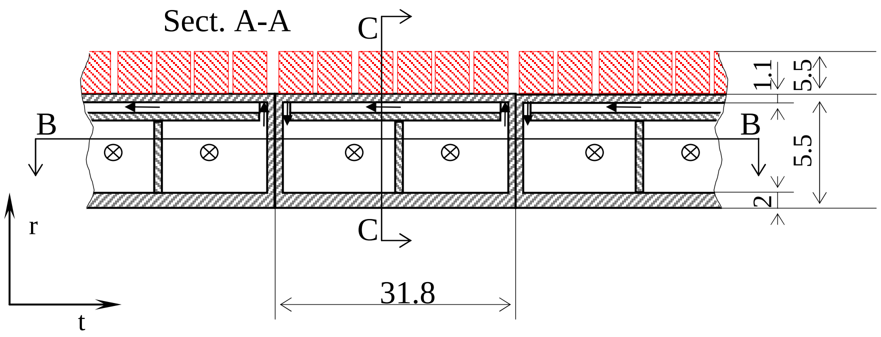
T_{outlet} = 1000°C
Low pressure (MHD)

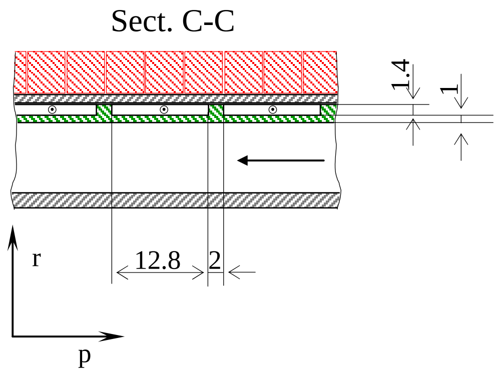
Main features

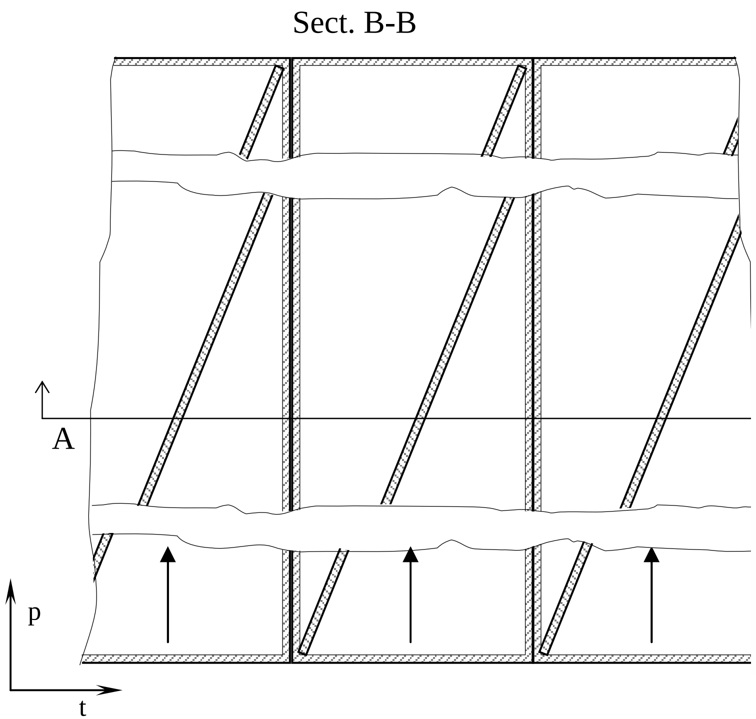
- ✓ derived from ARIES-ST studies
- ✓ W-alloy tiles
- ✓ high velocity LiPb in the toroidal direction (to limit MHD)
- ✓ SiC/SiC thin structures (R&D)

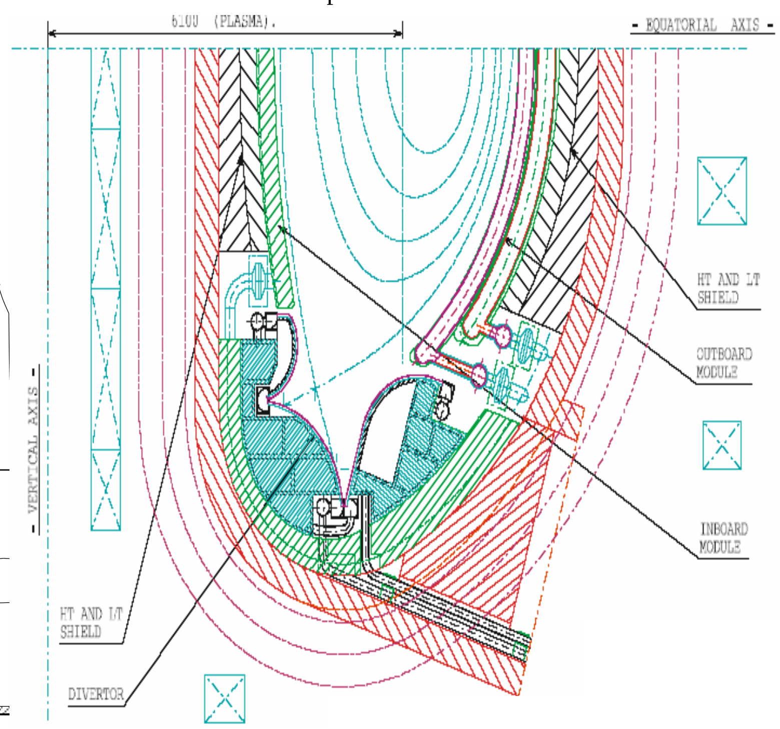
Performances

Heat flux up to ~5 MW/m²









PPCS Proposed Divertor Concepts - Summary

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Surface heat-flux in a reactor: TBD
15 MW/m² ➔ 10 ➔ 5 (long term) ??

He-cooled Concepts

- ✓ Achieve heat flux of 10 MW/m² (or more)
- ✓ Allow high T He-coolant (high efficiency)
- ✓ Avoid use of water with He-cooled blankets
- ✓ Development of W-alloy and ODS-steel manufacturing processes TBD
- ✓ Behavior of W-alloy under irradiation TBD
- ✓ Exp. demonstration of P-drops and heat transfer
- ✓ Joint W-alloys/steel TBD

➔ Significant R&D on Materials

Water-cooled Concepts

1 - Low-Temp. (as ITER, 150°C)

- ✓ Known manufacturing (as ITER)
- ✓ Achieve heat flux of 15 MW/m²
- ✓ Lifetime of Cu-alloy under irrad. TBD
- ✓ Deposited Power not used for Power conversion ➔ low reactor efficiency

➔ Low R&D but low performances

2 - High-Temp. (as WCLL, 325°C)

- ✓ Achieve heat flux of 15 MW/m²
- ✓ Lifetime of Papyex & th. Barrier, especially under irradiation TBD
- ✓ Manufacturing of interface TBD

➔ Significant R&D for the steel/W joint

LiPb-cooled Concepts

- ✓ Interesting concept if used jointly to the SCLL blankets, same R&D long term issues (SiC/SiC structures), reliability, modeling, compatibility with very high v & T LiPb TBD
- ✓ MHD: modeling of 3D inertial flow in expansion TBD
- ✓ Low achieved heat flux : 5 MW/m², large progress expected in divertor physics TBD

➔ Very large R&D (including physics)

Overall Conclusions

- ***Breeding Blankets*** **: a major new technological challenge for Fusion towards DEMO**

☞ Five blanket lines considered in the EU PPCS studies after a selection among the different possible options & materials combinations

☞ The blanket concepts corresponding to three of these lines (Water-cooled LiPb, He-cooled Ceramic/Be pebble beds, He-cooled LiPb) requires technologies which can be developed for DEMO (short term). In particular, several mock-ups of the two He-cooled concepts will be tested in ITER.

☞ The short term concepts show an acceptable efficiency (30-40%). Major design uncertainties are linked with reactor integration issues (module attachment, pipes routes, connection/disconnection, blanket coverage)

☞ Medium and long-term concepts (Dual-Coolant and LiPb Self-cooled SiC/SiC structures) show large attractiveness by higher development risk ➔ Large R&D required

- ***Divertor : a major technological improvement from ITER towards DEMO***

☞ Design and R&D performed for ITER. Additional requirements for DEMO and a Power Plant have to be taken into account. Possible solutions have been evaluated with EU (PPCS). Results will be used as a basis for launching appropriate R&D.