Politecnico di Torino
corso magistrale: 2015-16

Blanket Technology for the Fusion Reactor

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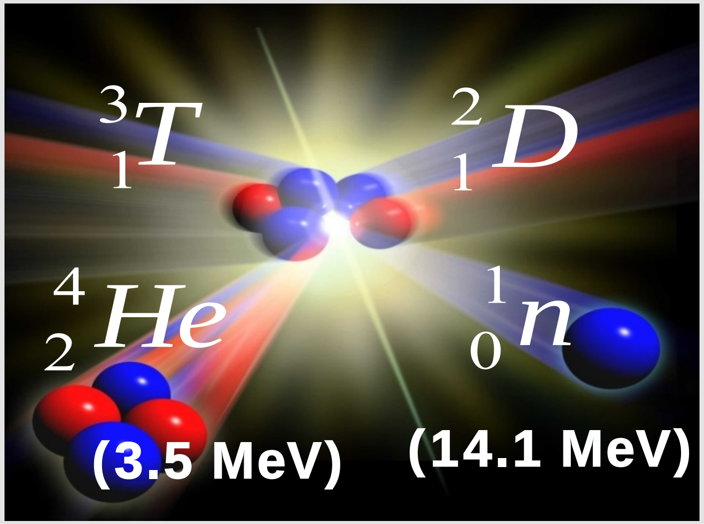
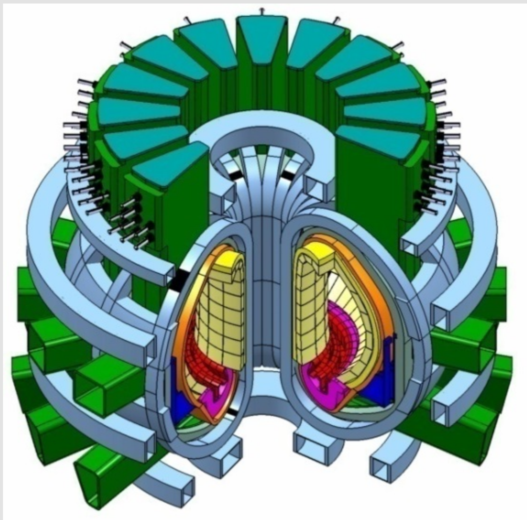
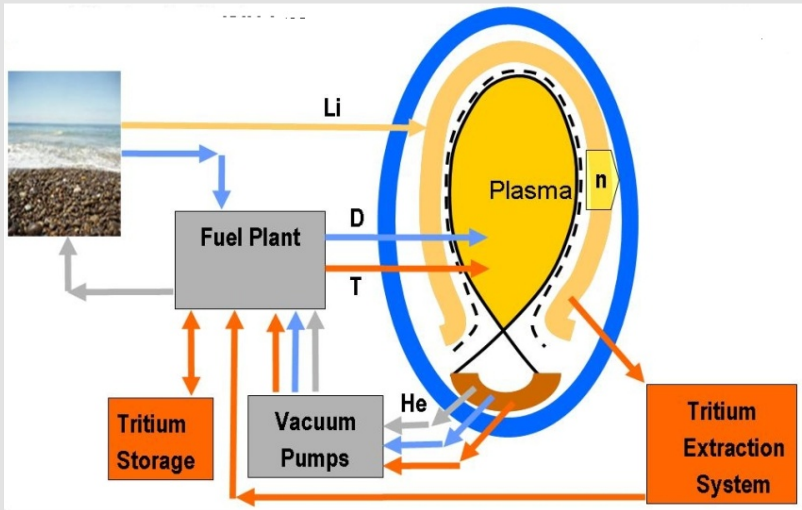




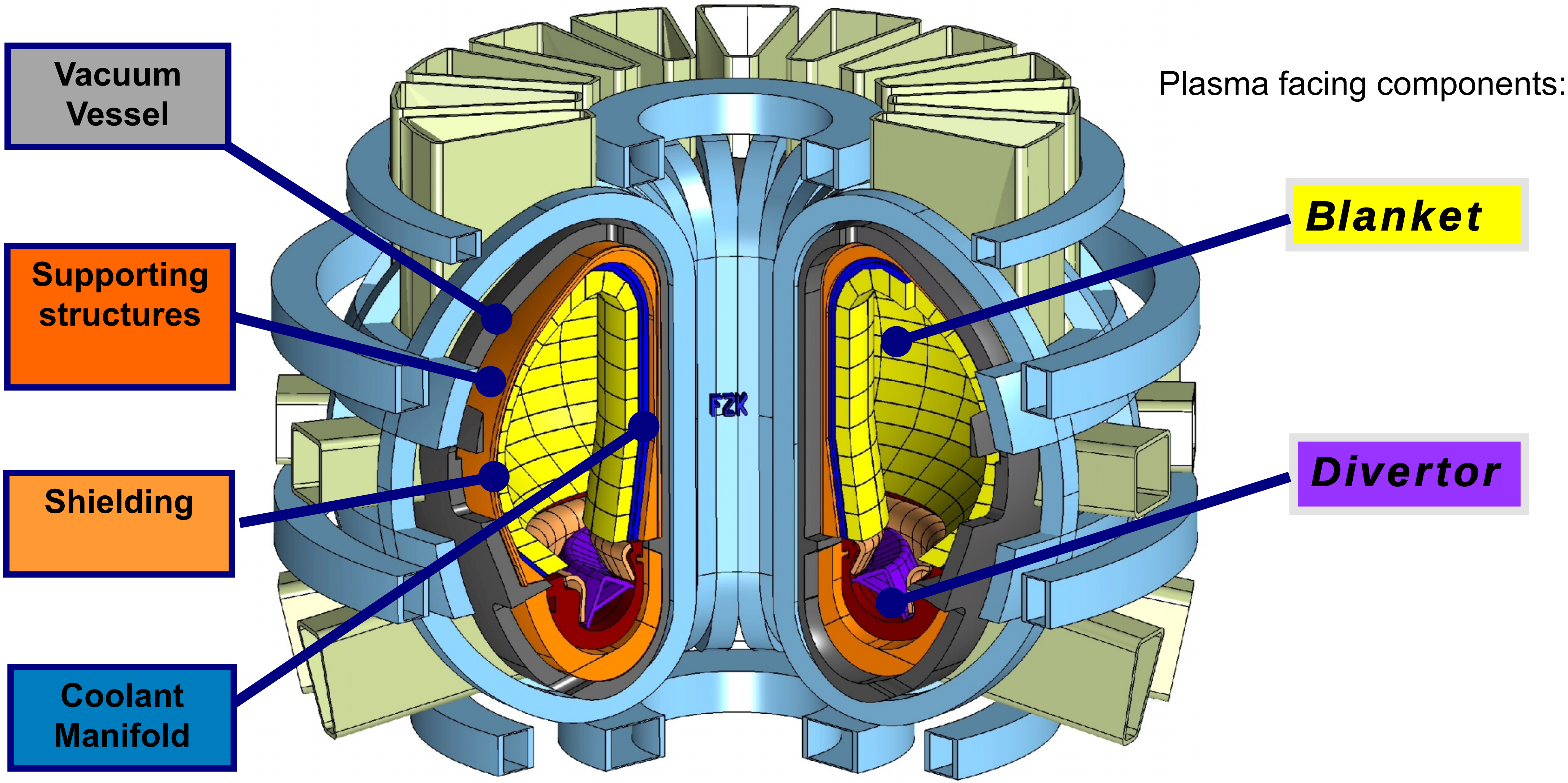


Outline

- Introduction
- Blanket fundamental functions (power extraction, T production, shielding)
- Reactor integration (e.g. maintenance systems).

- Blanket concepts (according to breeder, structural material, coolant, mechanisms of heat and T removal)
- DEMO and Power Plant blanket solutions: near term and advanced concepts.
- The Test Blanket programme in ITER

Fusion Reactor: Thermonuclear Core

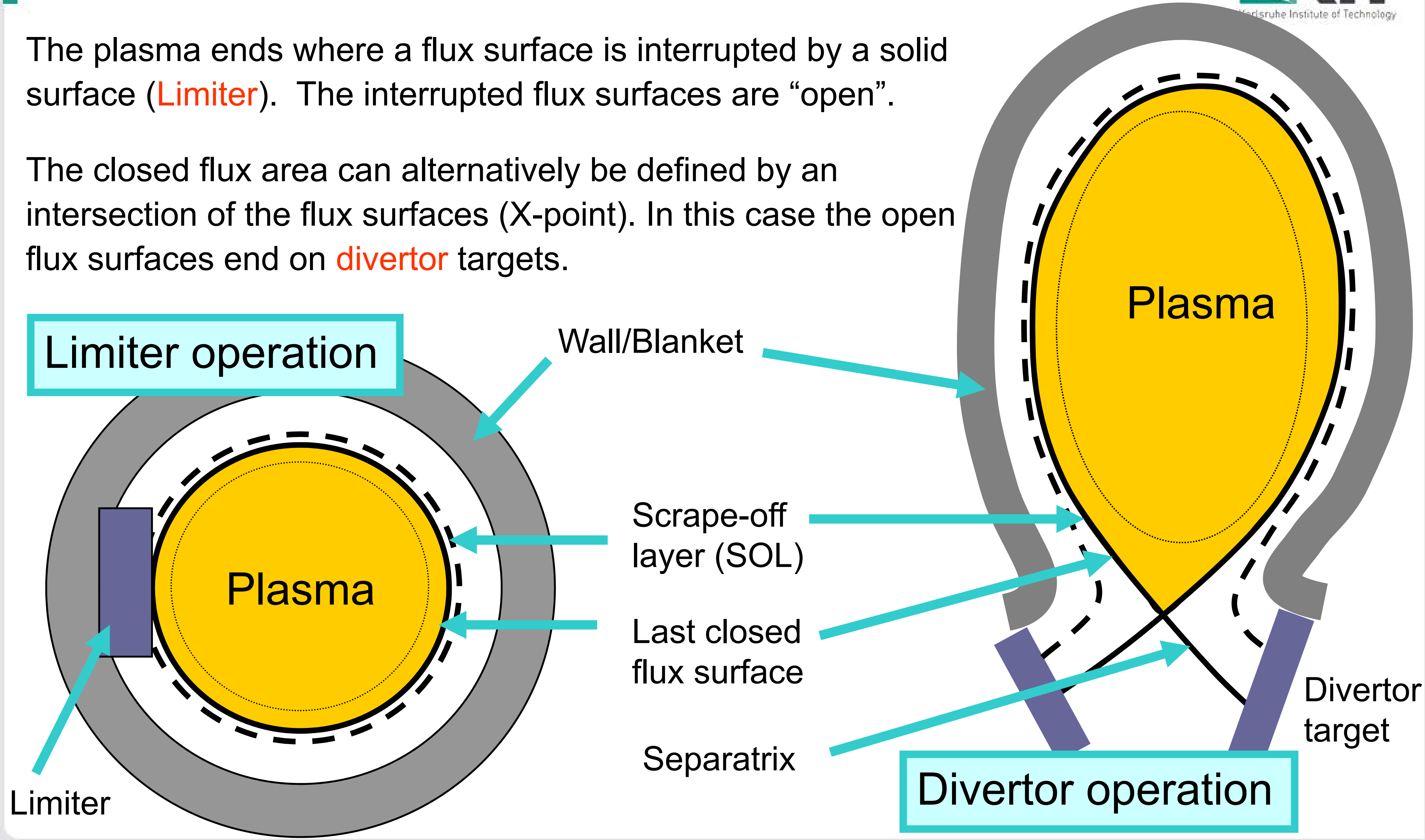


KIT Design for a DEMO (Demonstration Fusion Reactor Plant)

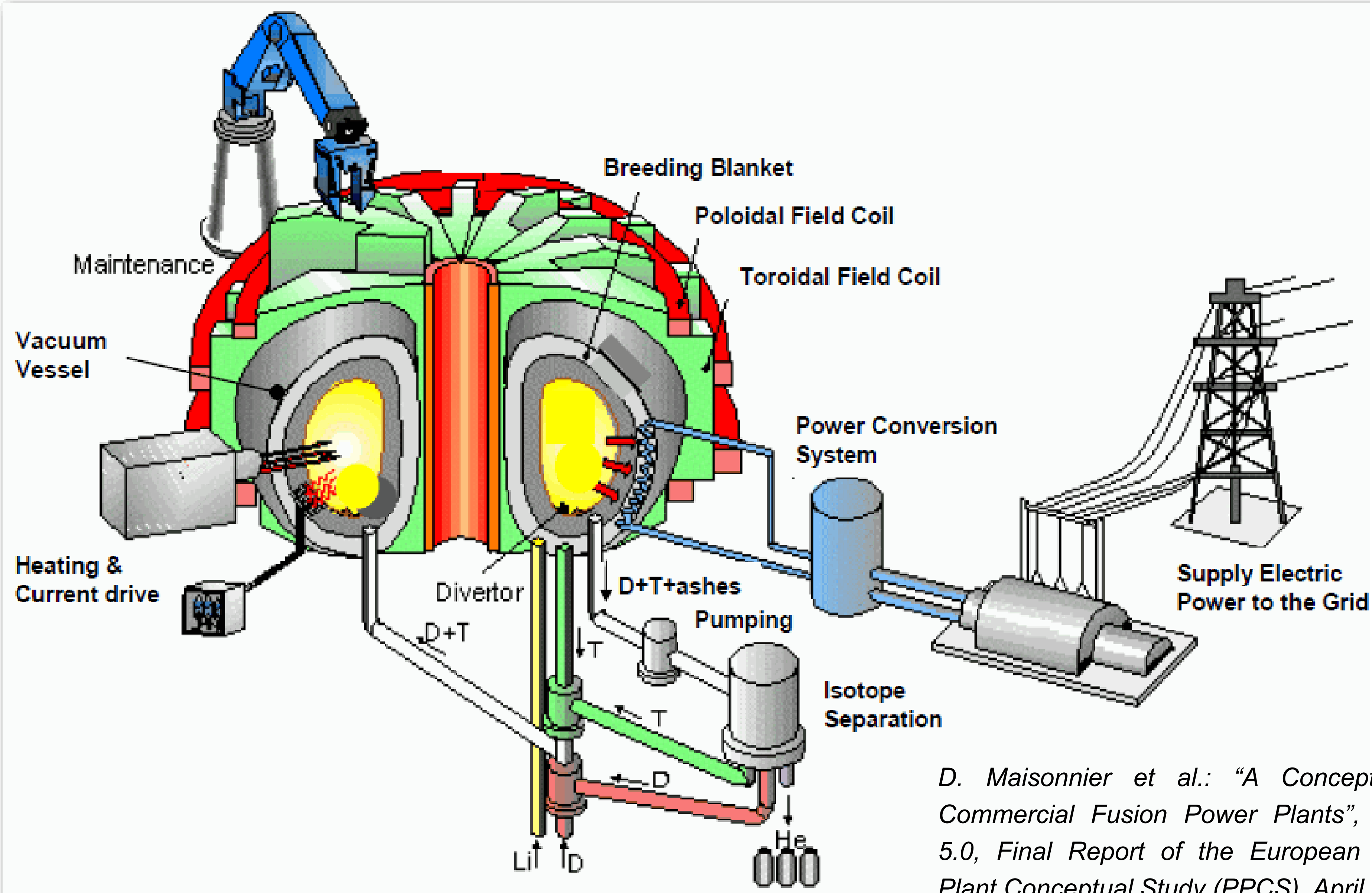
Plasma magnetic confinement: plasma limit

The plasma ends where a flux surface is interrupted by a solid surface (Limiter). The interrupted flux surfaces are "open".

The closed flux area can alternatively be defined by an intersection of the flux surfaces (X-point). In this case the open flux surfaces end on divertor targets.



Fusion Reactor: external systems



D. Maisonnier et al.: "A Conceptual Study of Commercial Fusion Power Plants", EFDA-RP-RE-5.0, Final Report of the European Fusion Power Plant Conceptual Study (PPCS), April 13th, 2005.

Breeding Blanket Functions and Requirements

Fundamental functions of the breeding blanket:

1. **Cooling:** conversion of fusion energy into heat and removal of the heat for electricity production
2. **Breeding:** Tritium production and extraction
3. **Shielding:** Contribute to the shielding of the Vacuum Vessel and Toroidal Field Coils

The design has to be featured in order to achieve:

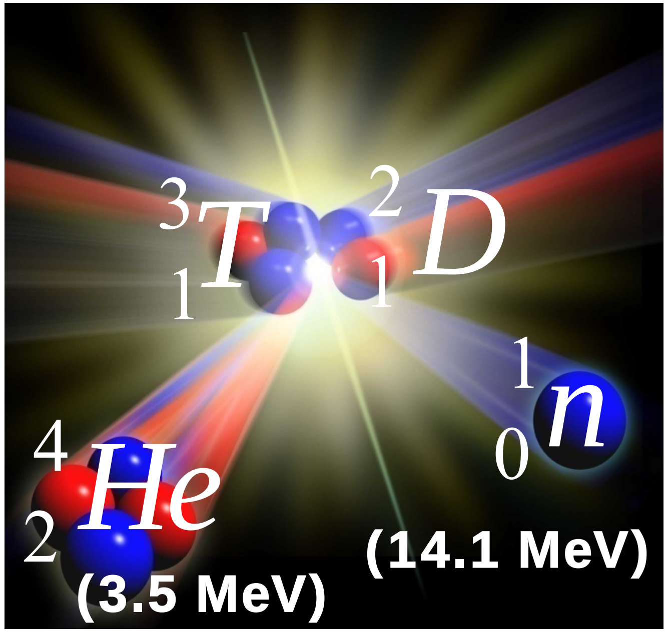
1. Low maintenance time (=> high availability)
2. Sufficiently long lifetime (=> high availability)
3. High safety level (e.g. accidents, operations, etc.) and low environmental impact (including waste)
4. Reasonable electricity costs (e.g. small dimensions, high efficiency, etc.)

Deuterium-Tritium Fusion

$^{2}D + {}^{3}T \rightarrow {}^{4}He + n + 17.6\ MeV$

⊕ ⊕ ⊕⊕ ○
○ ○○ ○○

⊕ Proton
○ Neutron



- Kinetic energy (½ m v^2) of α-particles (^{4}He) and neutrons = **17.6 MeV**
- α-particles and neutrons shall have the **same impulse (m * v)** → neutron must be 4 times faster than Helium

➡ kinetic energy distributed by inv. mass ratio m_α/m_n= 4/1=(14.1/3.5),

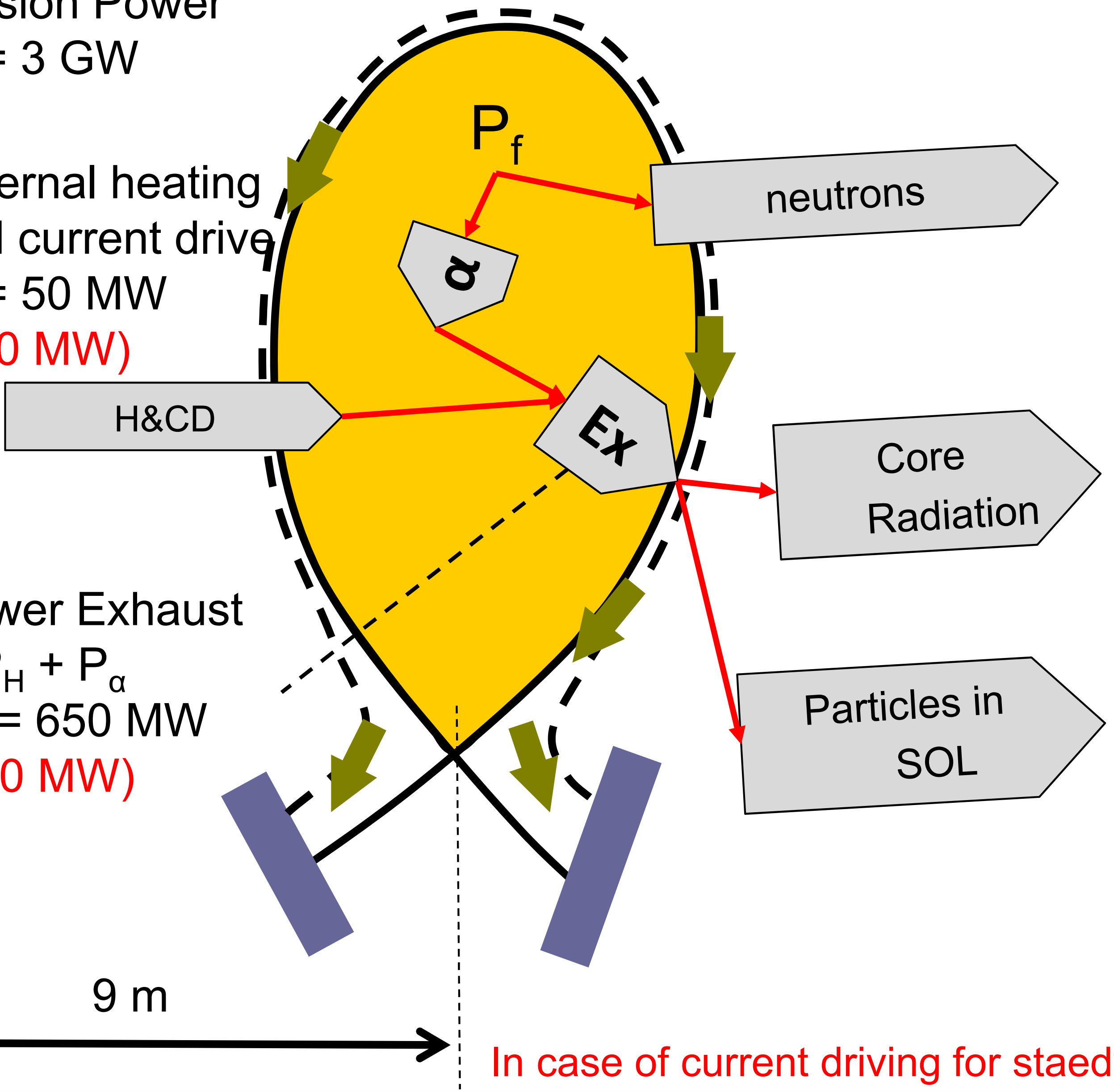
➡ **80% of energy in kinetic neutron energy**

Fusion Power Balance (example)

Fusion Power
P_f= 3 GW

External heating
and current drive
P_H= 50 MW
(200 MW)

Power Exhaust
= $P_H + P_\alpha$
P_{ex}= 650 MW
(800 MW)



In case of current driving for staedy state

PFC Loading:

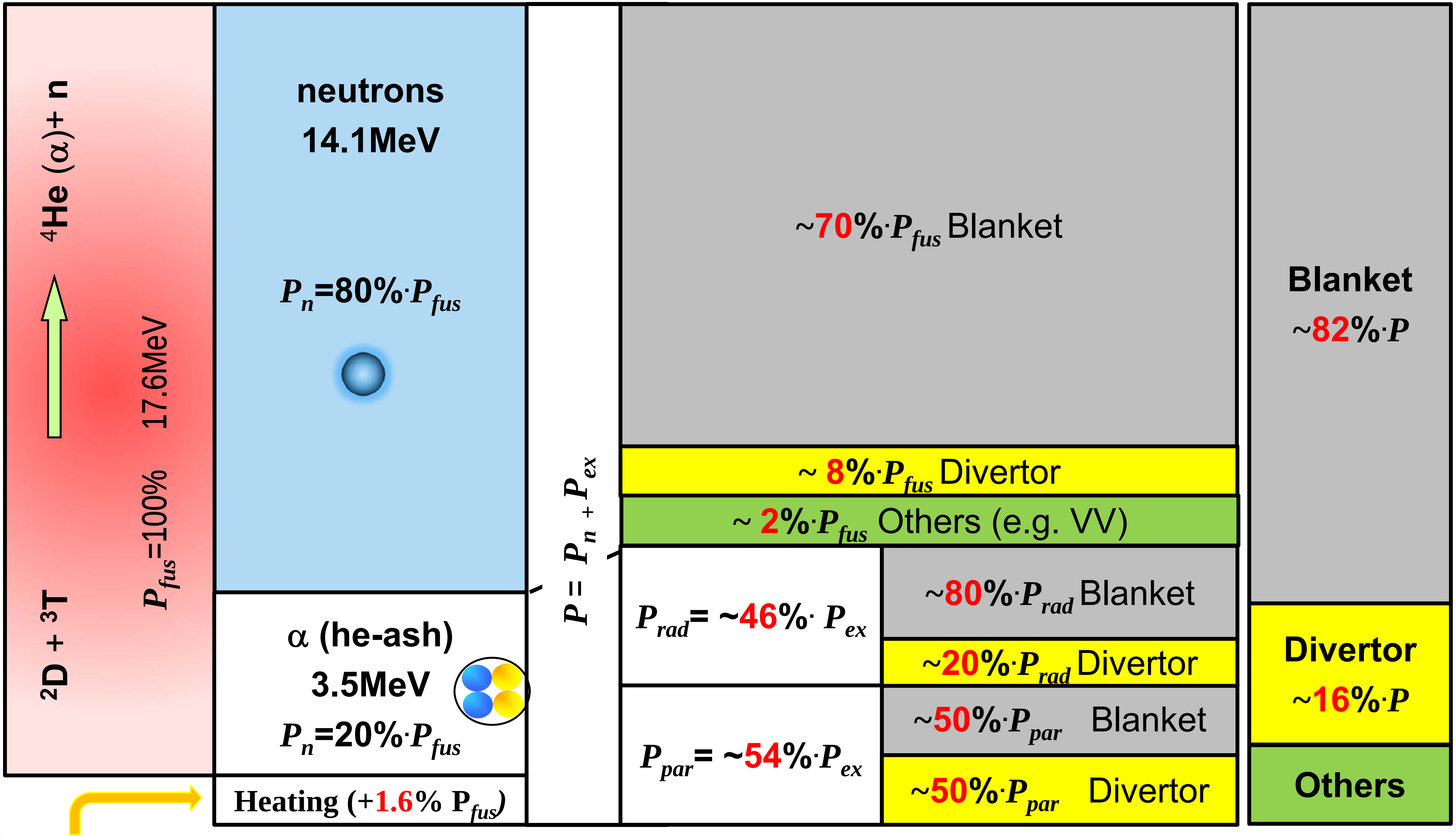
P_n= 2.4 GW
=> 2.4 MW/m² (av.)
=> 2.9 MW/m² (max.)

P_R= 300 MW
=> 0.3 MW/m² (av.)

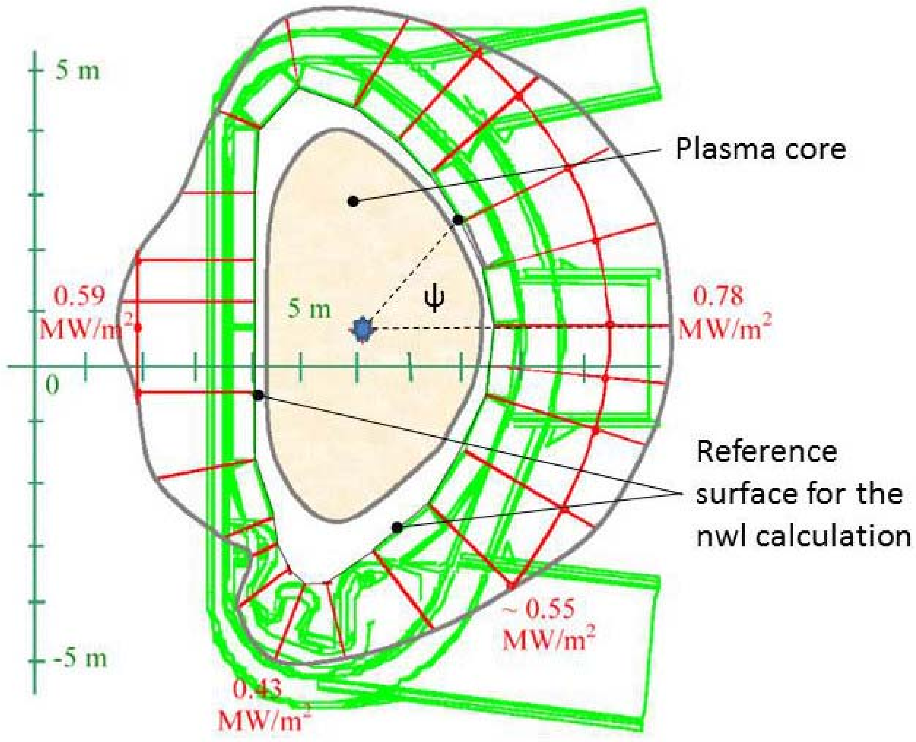
P_P= 350 MW
=> FW local spots (tbd)
=> 10-20 MW/m² on divertor plates

For a Blanket FW area of ~1000 m²

Thermal loads on plasma facing components

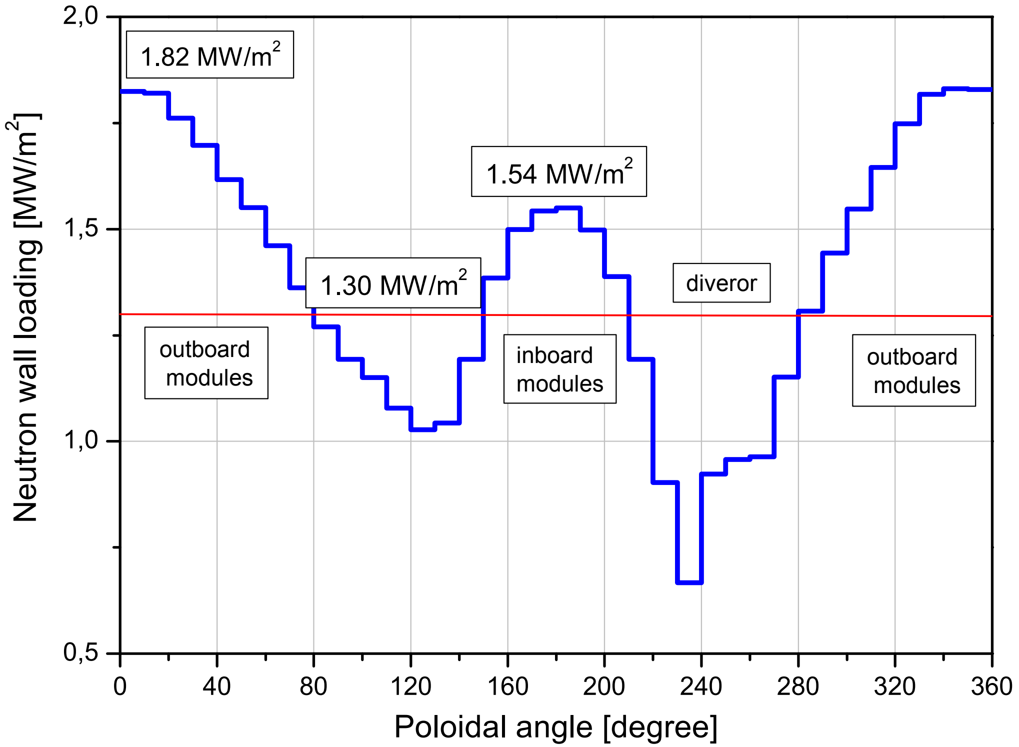


Neutron source and neutron wall load



- The plasma acts as a volumetric 3D-neutron source, practically axisymmetric and picked at the equatorial plane. Peak factors (actual value/average value) of 1.2-1.4 are usual for ITER and DEMO.

- **Left figure:** ITER neutron wall load (nwl) at 500MW fusion power

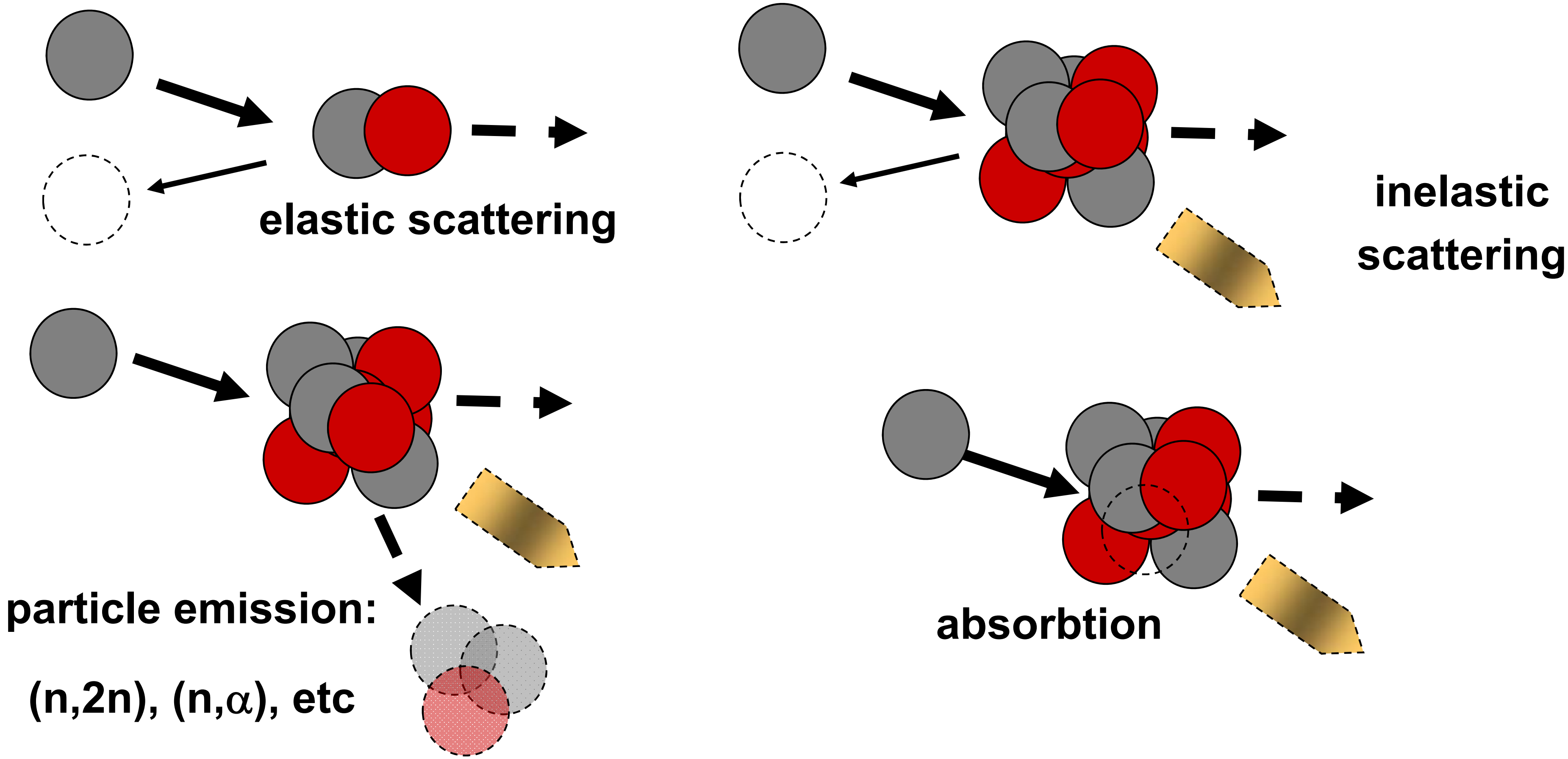


- **Right figure:** DEMO (~2.5 GW_f) neutron wall load as function of the poloidal angle coordinate ψ in the vertical-radial plane z-r.

- This quantity (nwl) is calculated with respect to a reference surface that follows the FW contour.

Neutron interation with matter

Several interactions possible



Crucial parameter: nuclear cross-section σ (measured in barn=10^{-24}cm^2)

σ dependent on incident neutron Energy (E) and angle (φ)

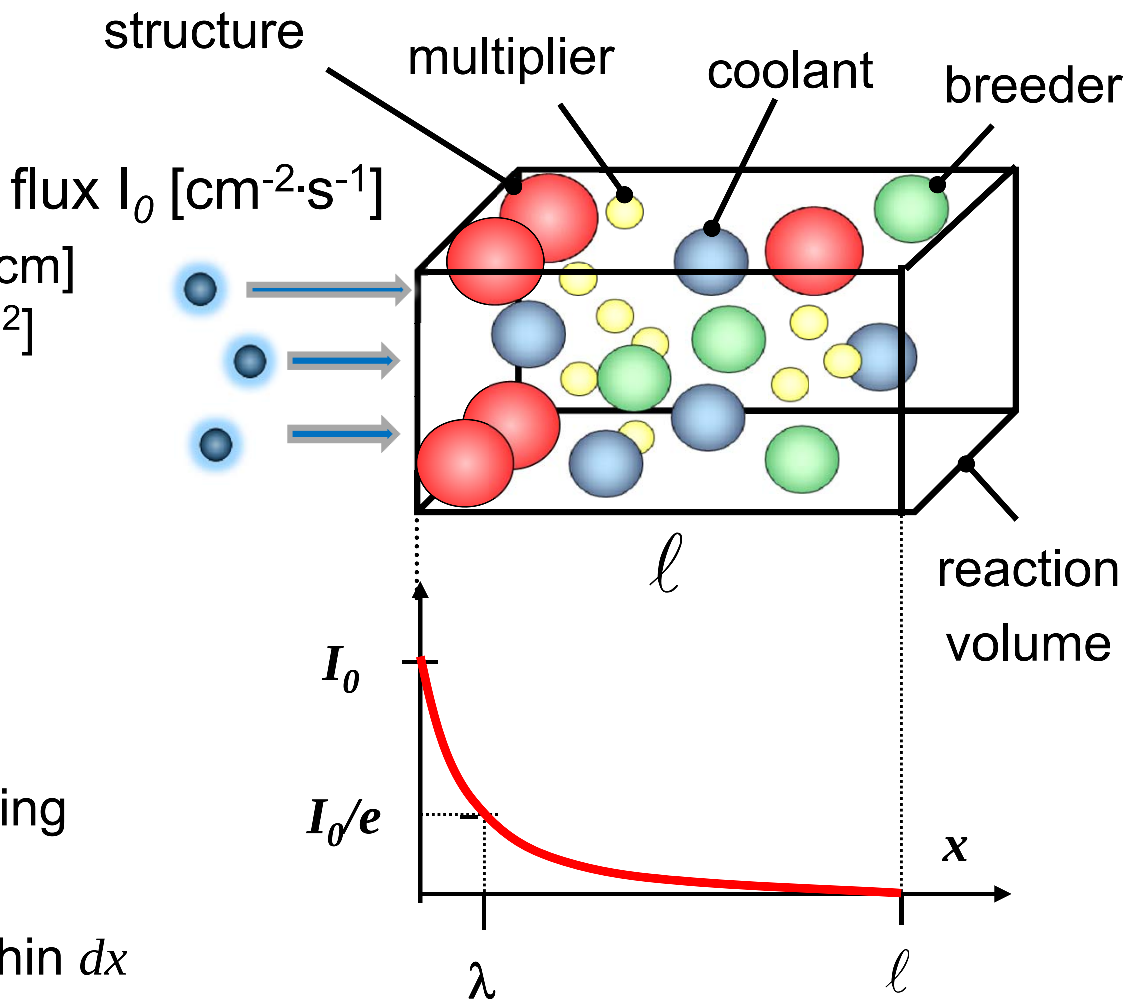
Volumetric heating by P_n

Interaction of different nuclei with matter scoping all reactions

- elastic and inelastic scattering,
- absorption,
- particle emission
- and associated γ-ray emission)

Essential $\boxed{\Sigma = N \cdot \sigma}$

Σ=macroscopic nuclear cross-section [1/cm]
σ=microscopic nuclear cross-section [cm^2]
N=number nuclei [1/cm^3]
(standard unit for σ is barn=10^{-24} cm^2)



Lambert-Beer-attenuation law

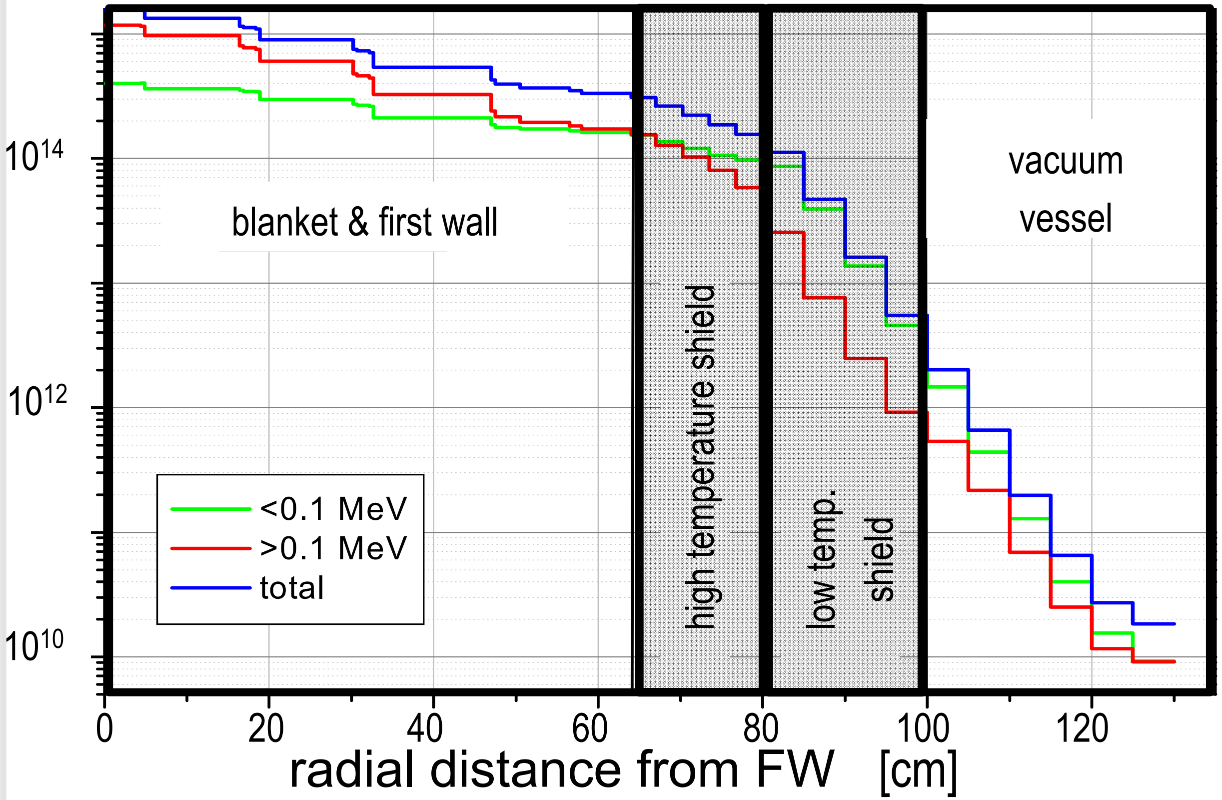
$$\boxed{I_x = I_0 \cdot e^{-\Sigma x}}$$

- $\Sigma^{-1}=\lambda$: mean free path (average travelling distance before collision)
- $\Sigma \cdot dx$: collision probability of neutron within dx

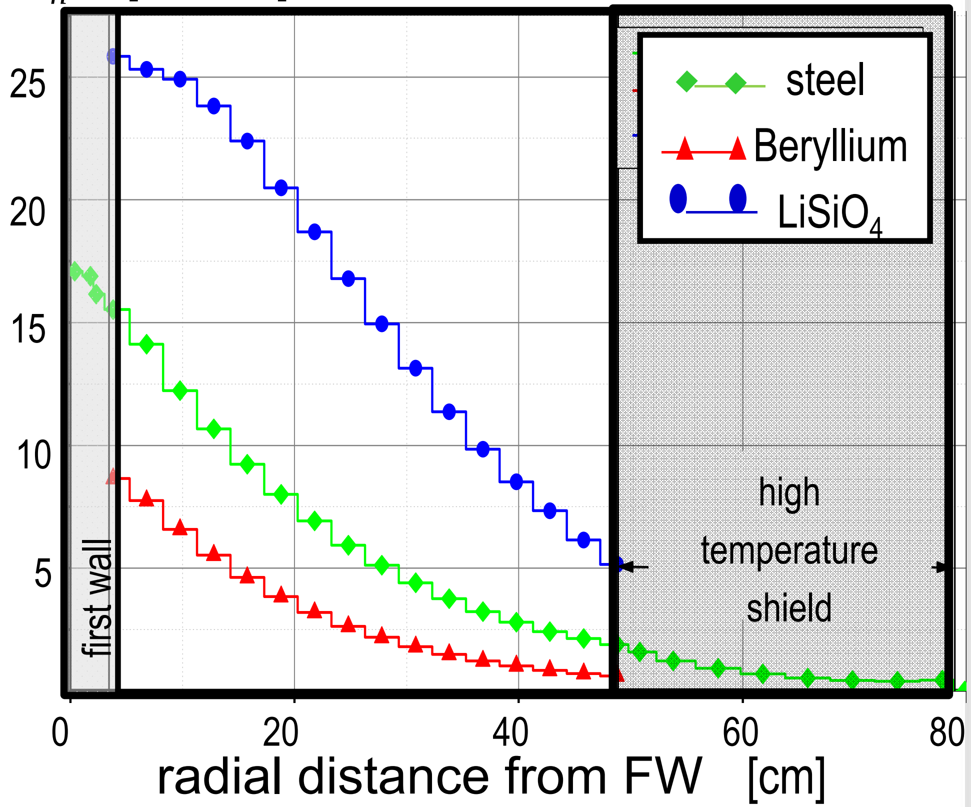
Volumetric heating

Given Fusion reaction P_{fus} ➡ Φ_n ➡ volumetric heat release

Φ_n [1/(cm²·s)]



P_n [MW/m³]



Volumetric power density: Magnitude Example: for 3 MW/m² neutron wall load

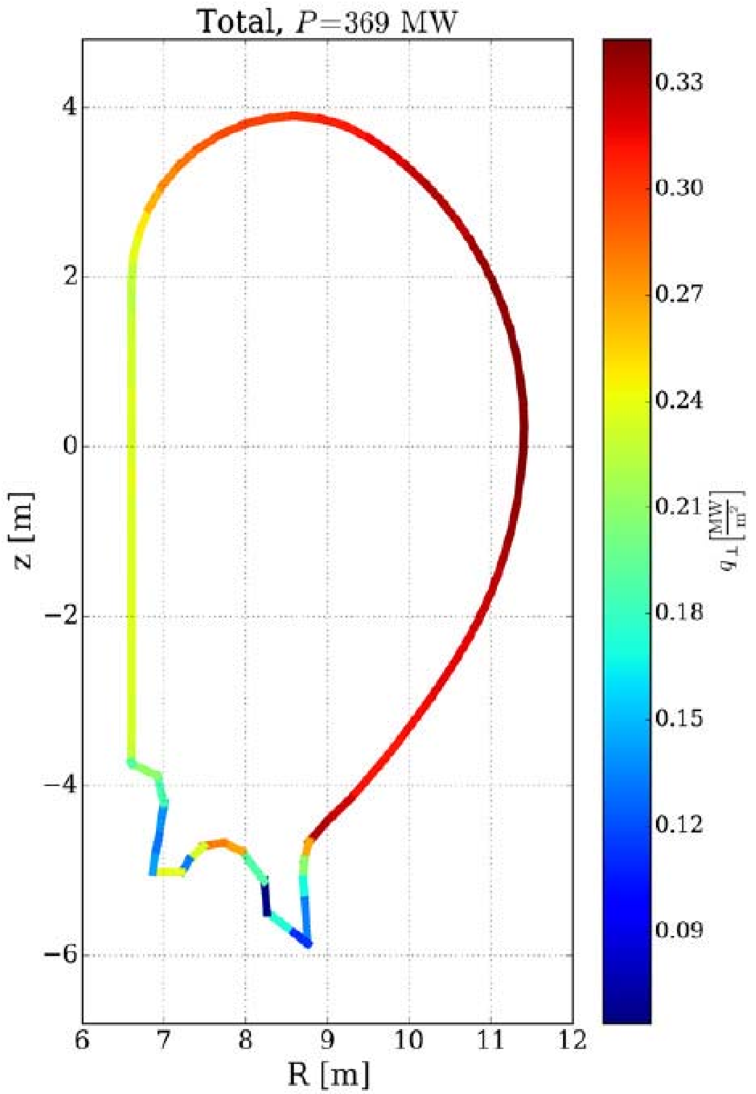
- In the FW, first 5 mm ➡ $q'''_{n,max} \approx 25$ MW/m³ that corresponds to a $q''_{s,max} \approx 0.125$ MW/m²
- Average in a 1m blanket ➡ $q'''_{n,max} \approx 2.9$ MW/m³

In comparison in a LWR fuel rod ➡ $q'''_{n,max} \approx 100$ MW/m³

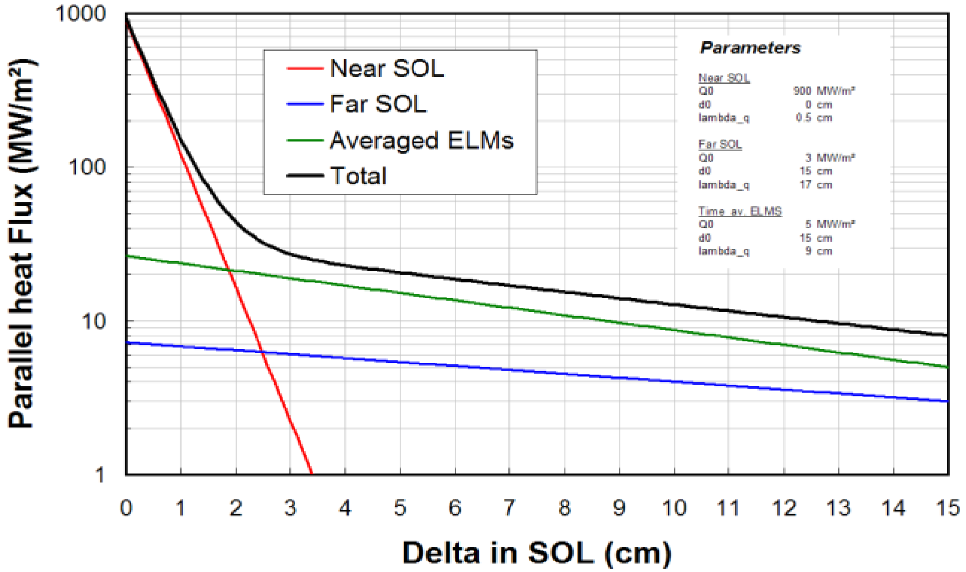
Surface Heating: core radiation

- Heat flux evenly distributed along outer wall
- Peak heat flux below material limit 0.35 MW/m²
- Inner wall receives reduced heat load 0.24 MW/m²
- Divertor is well shielded from main chamber radiation.

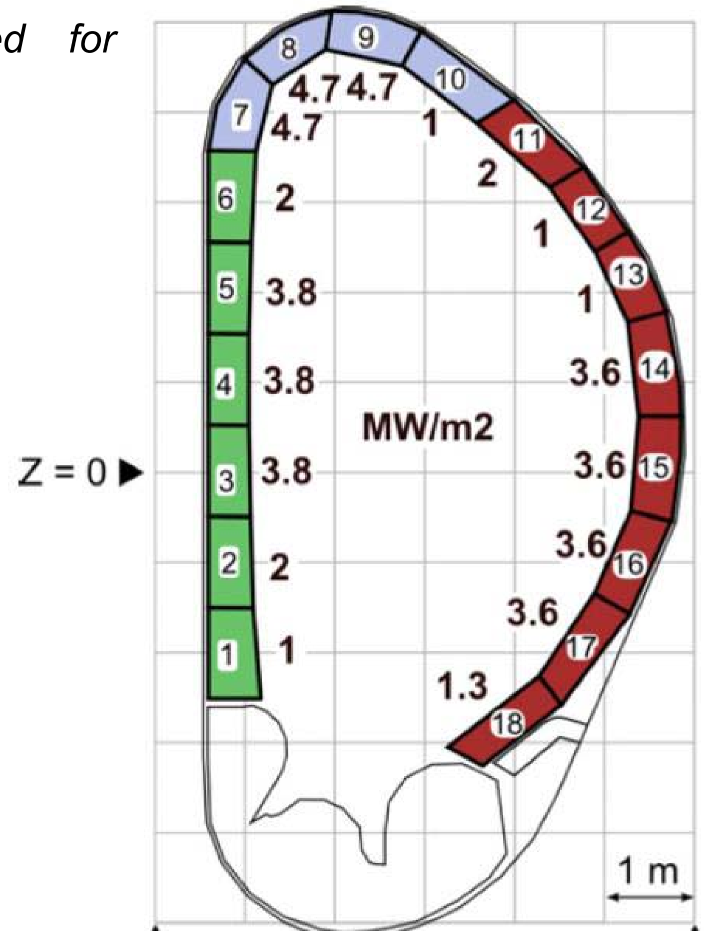
Distribution of the radiative flow according to Sieglind, Radiative Wall Load, DEMO Physics Forum, 17 November 2015.



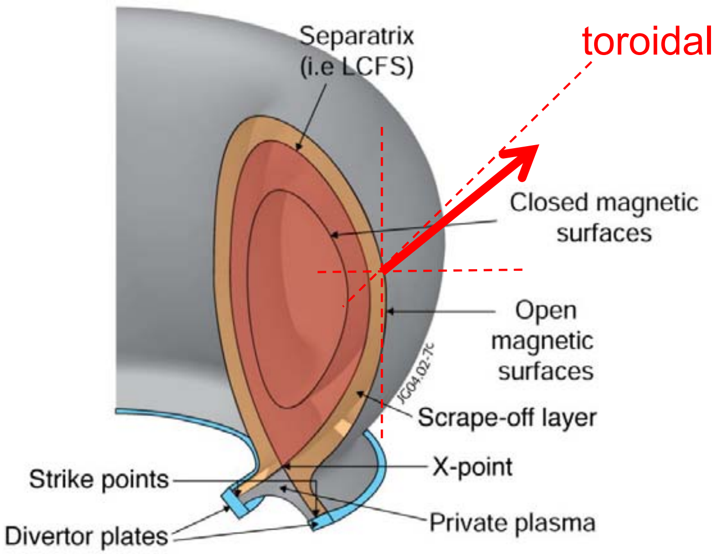
Surface heating: Plasma flux

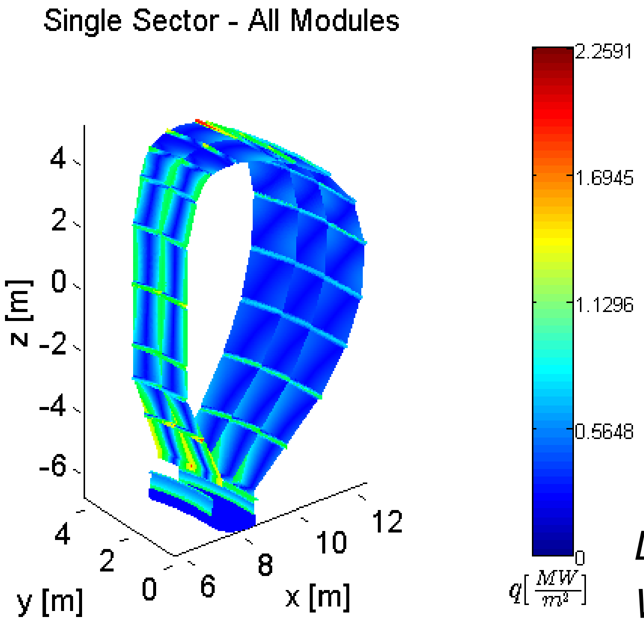


Plasma heat flux ("parallel" component) calculated for ITER (ITER: Mitteau)



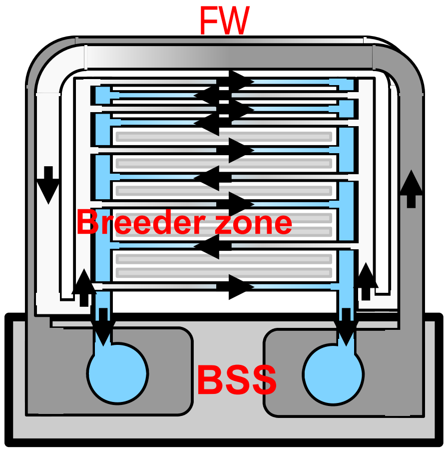
ITER: Mitteau (JNM 2011)



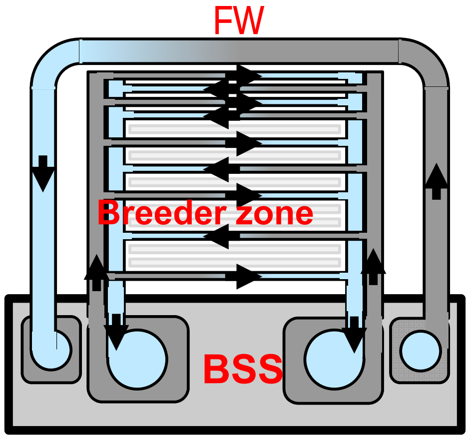


L. Barbato, 3rd DEMO Physics Forum, "First Wall Loads – First Estimates"

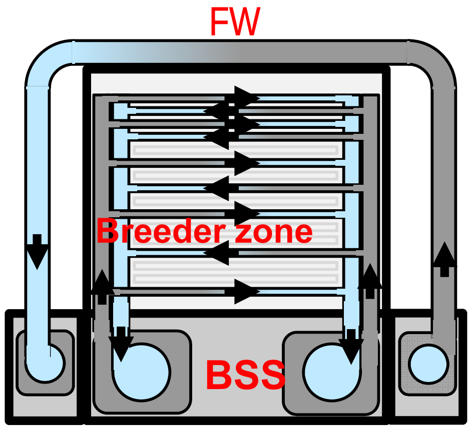
Primary heat transfer system (PHTS)



i) Integrated FW concept



ii) Hydraulic decoupled concept



iii) Fully decoupled concept

Three architecture options have been proposed:

i) a fully thermo-hydraulically and mechanically integrated option (current reference design),

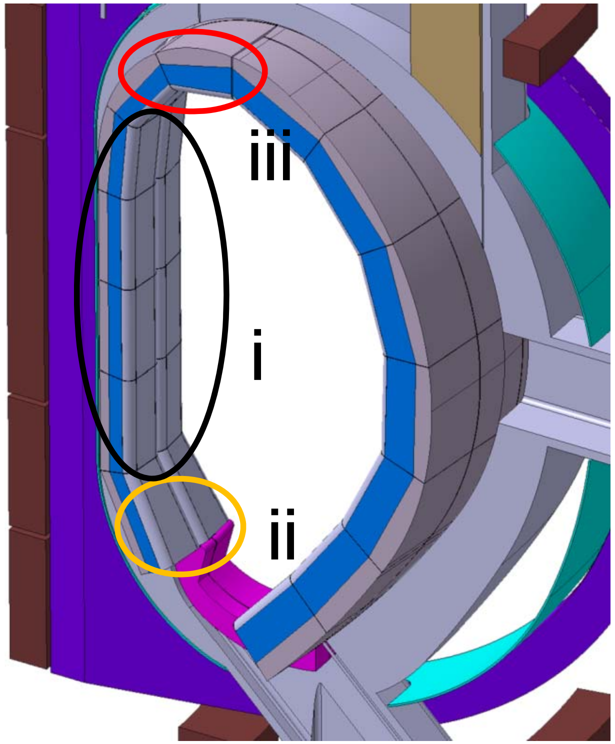
ii) a thermo-hydraulically decoupled option;

iii) a fully (also mechanically) decoupled option.

The adoption of these different configurations will depend on the local loads acting on the blanket surface.

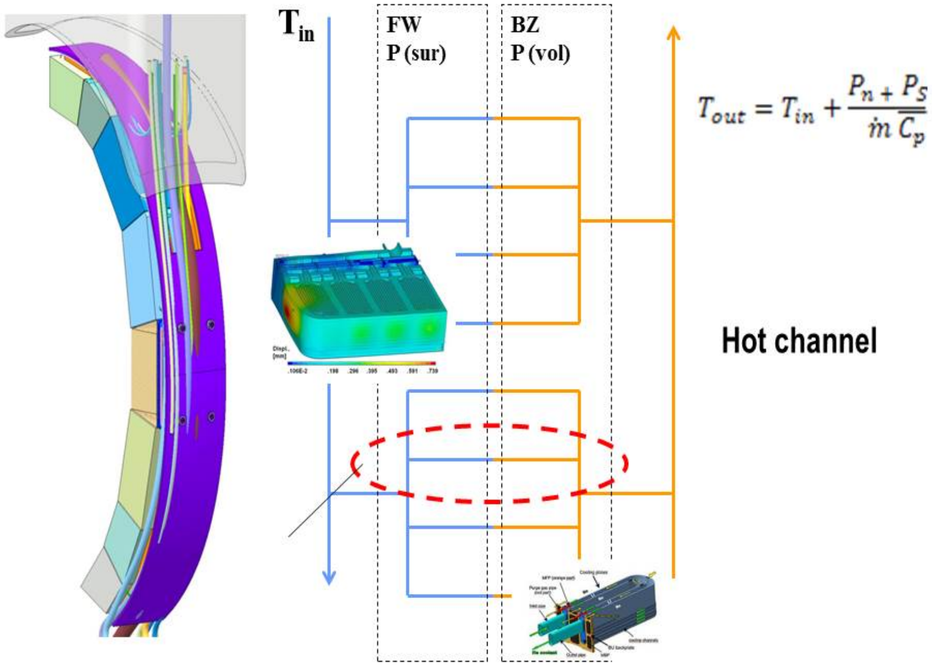
Different architecture of the FW probably to be used for different part of the segment poloidal plasma surface.

Limiters as FW protection are also under investigations.

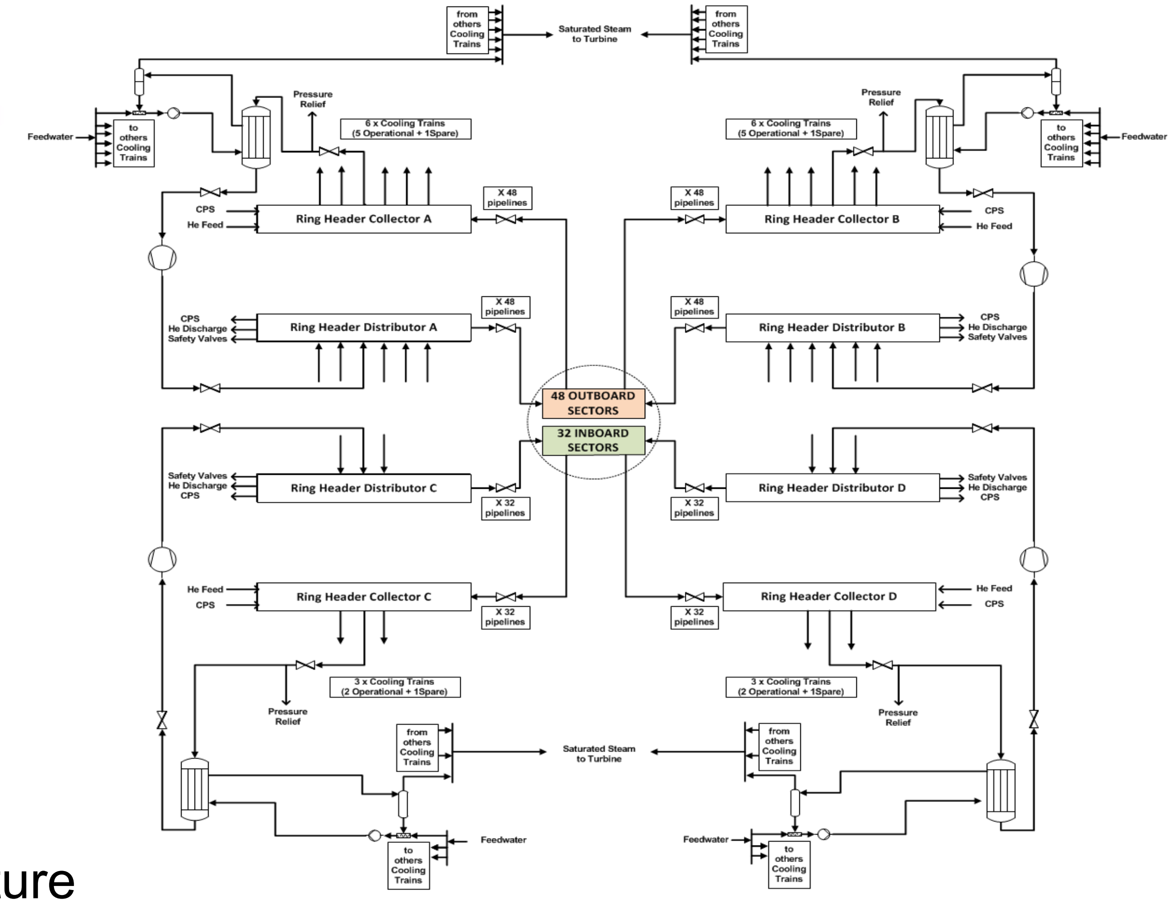


<u>T.R. Barrett</u> *et al., "Progress in the Engineering Design and Assessment of the European DEMO First Wall and Divertor Plasma Facing Components", ISFNT-12, Jeju, Korea (2015).*

Primary heat transfer system (PHTS)



PHTS lay-out in the blanket



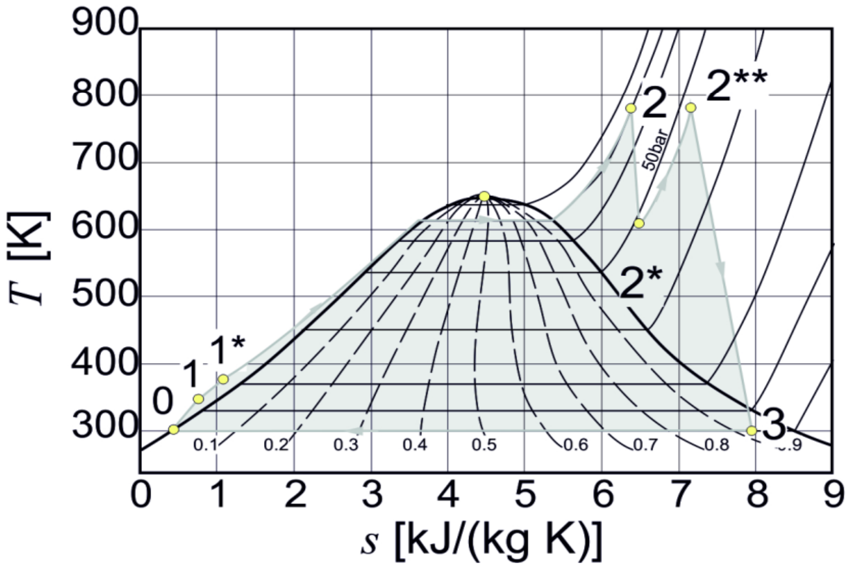
PHTS for Helium Cooled Blankets (Siemens KWU, 1992)

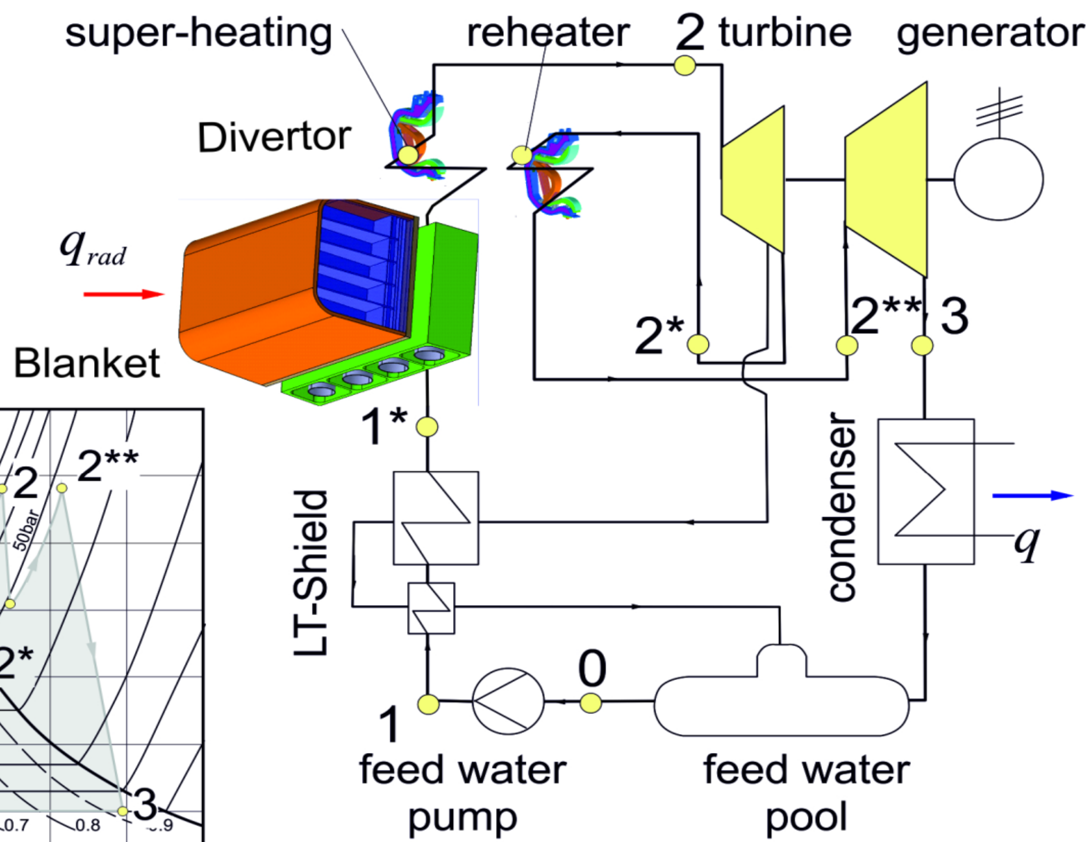
Requirement: sufficient coolant temperature for high efficiency h_{th} Power Conversion Systems Types.

- Joule-Brayton (Gas turbine cycle)
- Clausius-Rankine (steam turbine)

Clausius- Rankine Process

P_{fus} DEMO	4050 MW
$P_{el,n}$	1500 MW
Plant efficiency, $P_{el,net} / P_{fus}$	37 %
P_{pumps}	400 MW
η_{th}	42 %



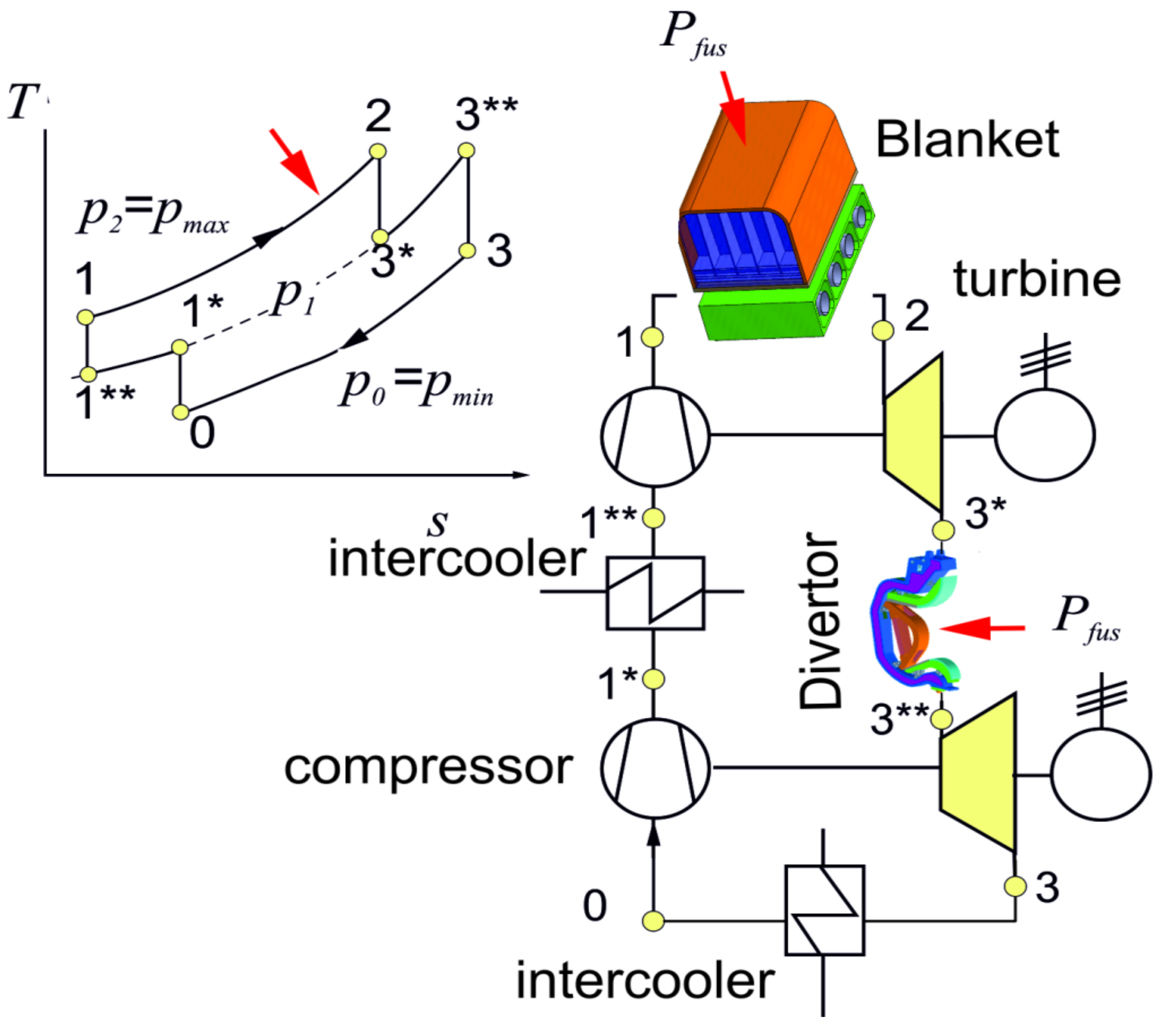


- lower mean average temperature
- but multi-stage required
- operation threshold higher than advanced PWR

Joule-Brayton -Process

- use of inert He
- demand for high η_{th} temperatures >700°C
  - ➡ high material challenges
  - ➡ high pumping power ≈8-10%· P_{fus}

- For efficiencies η_{th} >40% staggered heating required to maintain in acceptable stress and performance limits of components
  - Stage 1 300-480°C
  - Stage 2 480-620°C
  - Stage 3 480-700°C
  - Stage 4 700-800°C



Tritium Consumption

For 2700 MW Fusion (i.e. ~1000 MW Electrical) Power :

Note: 1eV = 1.602 10^{-19} As*V = 1.602 10^{-19} Joule

1. Energy per fused tritium atom (17.6 MeV in Joule):
$17.6 \cdot 10^{6} * 1.602 \cdot 10^{-19} = 2.82 \cdot 10^{-12}$ Joule;

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2. Fusion frequency = P/E = $2700 \cdot 10^{6}$ J/s / $2.82 \cdot 10^{-12}$ J = $9.57\ 10^{20}$ 1/s;

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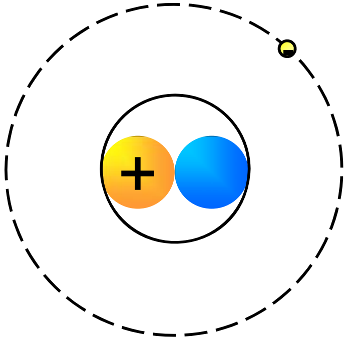
3. Tritium mass flow = 3 * mass of proton (neutron) * frequency =
$3 * 1.67 \cdot 10^{-27}$ kg * $9.57 \cdot 10^{20}$ * 24 * 60 * 60 * 1/day = **0.41 kg/day**

Tritium T (^3H):	**~ 0.41 kg/day**	**~ 150 kg/fpy**
Deuterium D (^2H) :	**~ 0.27 kg/day**	**~ 100 kg/fpy**

fpy **= full power year**

To compare:1000MW coal plant @ η=40% requires 2.16Mt coal and produce 5Mt CO_2

D: Deuterium



$$^{2}_{1}H$$

Property	D_2O (Heavy water)	H_2O (Light water)
Freezing point (°C)	3.82	0.0
Boiling point (°C)	101.4	100.0
Density at STP(g/mL)	1.1056	0.9982
Temp. of maximum density (°C)	11.6	4.0
Viscosity (at 20°C, mPa·s)	1.25	1.005
Surface tension (at 25°C, µJ)	7.193	7.197
Heat of fusion (cal/mol)	1,515	1,436
Heat of vaporisation (cal/mol)	10,864	10,515

Can be found: in ocean water at a H/D ratio of ~6500 (~150 ppm)

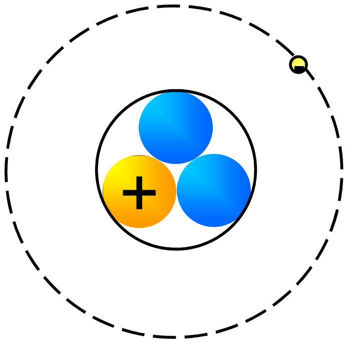
Used: in nuclear energy (e.g. D_2O in CANDU reactors)

Production methods (D_2O): e.g. Girdler-Sulfid-Proces (isotopical exchange) + vacuum distillation

Extimated earth availability: $5*10^{16}$ kg (in oceans)

Sufficient for several billion years !!

T: Tritium



$$^{3}_{1}H$$

Radioctive with a half life of 12.33 y

$T \rightarrow {}^{3}He + e^{-} + 18.6\ keV$

Can be found in nature:

in neglegible amount as product of cosmical rays

$$^{14}N + n \rightarrow {}^{12}C + {}^{3}T$$

Production as waste of nuclear D_2O reactors (e.g. CANDU):

$$D + n \rightarrow T \qquad \sim 2\ kg/year$$

No available production to support fusion reactors as external sources.

ITER will consume less than 20 kg in its whole life.

T production in reactor

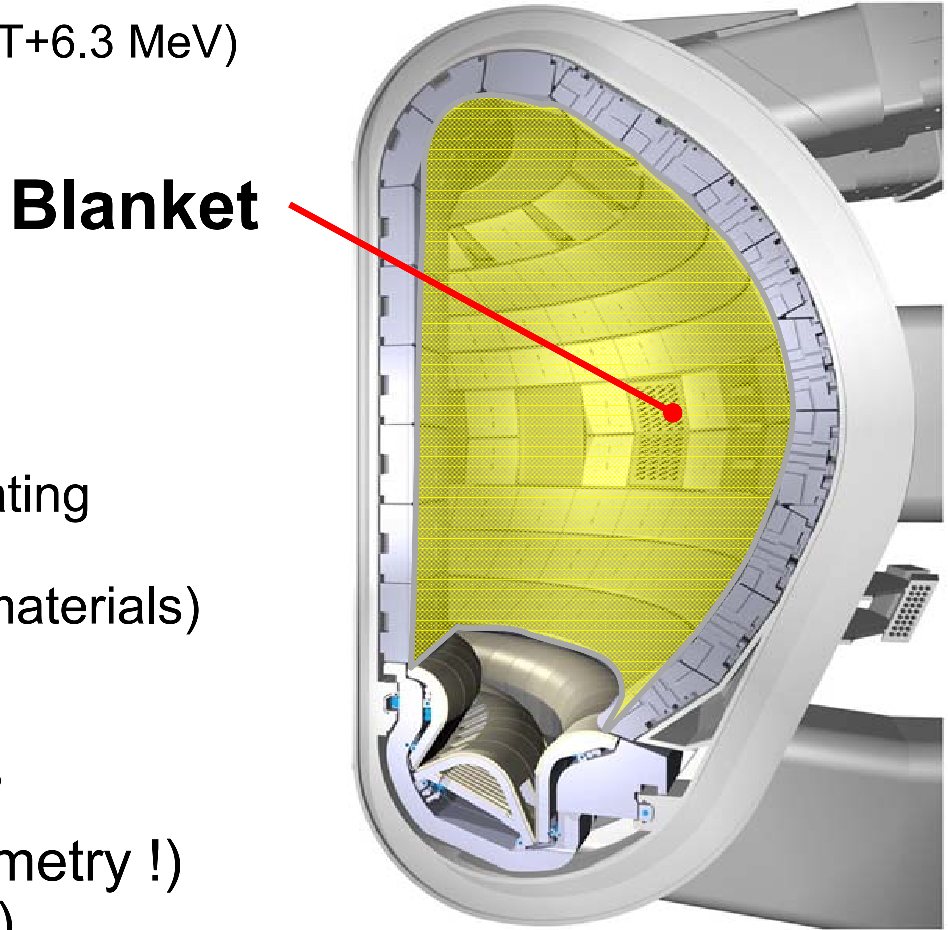
Use of T breeder:

$^{6}Li + n = T + ^{4}He +$ 4.8 MeV or $^{6}Li(n,a)\ T\ + 4.8\ MeV$

$^{7}Li + n = T + ^{4}He + n$ - 2.5 MeV or $^{7}Li(n,n'a)T\ - 2.5\ MeV$

(other potential reactions $^{3}He(n,p)T + 0.8$ MeV, $^{2}H(n,\gamma)T+6.3$ MeV)

- **Which configuration ?**
  - Breeder arrangement in ***Blanket*** around plasma chamber so that neutron absorbed in breeder
  - Reactor constraints
    - Plasma chamber (80% for Blanket -coolant, structure material- 20% divertor & plasma heating devices)
    - Parasitic neutron absorptions (non breeding materials)
    - Neutron leakage (ports, diagnostics)
    - ⇨ **Need for neutron multiplication**
- **How to prove tritium breeding capability?**
  - neutronic calculation (⇨ method/data/geometry !)
  - calculation validation against experiment(s)

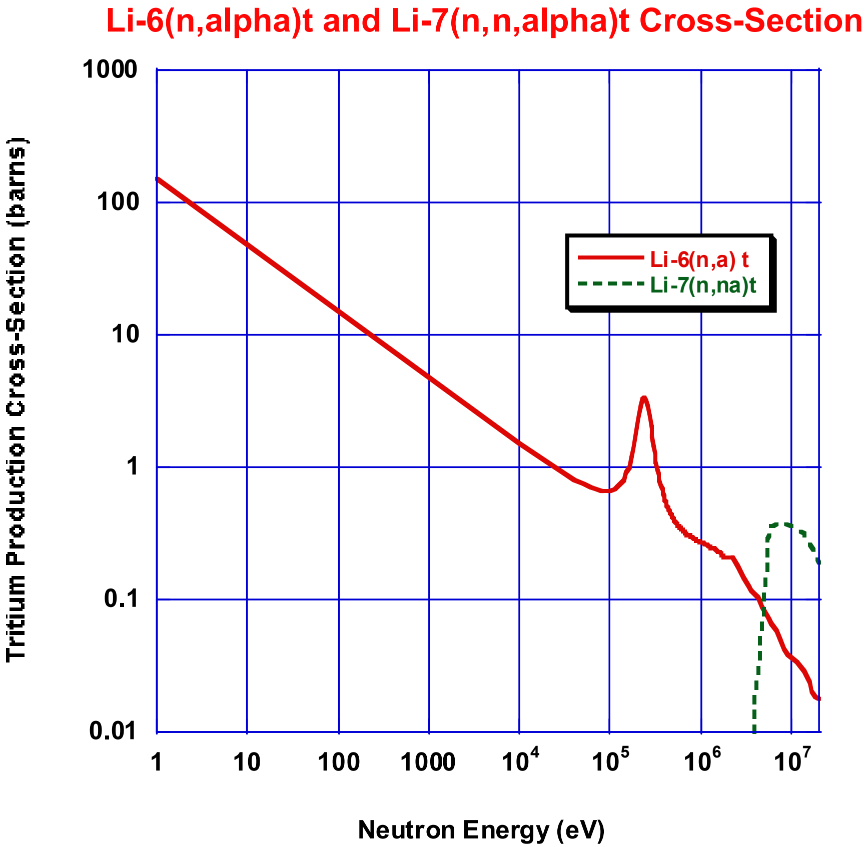


Tritium Production in reactor: Li reactions

Production with Li:

$^{6}Li + n = T + {}^{4}He + 4.8$ MeV.

$^{7}Li + n = T + {}^{4}He + n - 2.466$ MeV



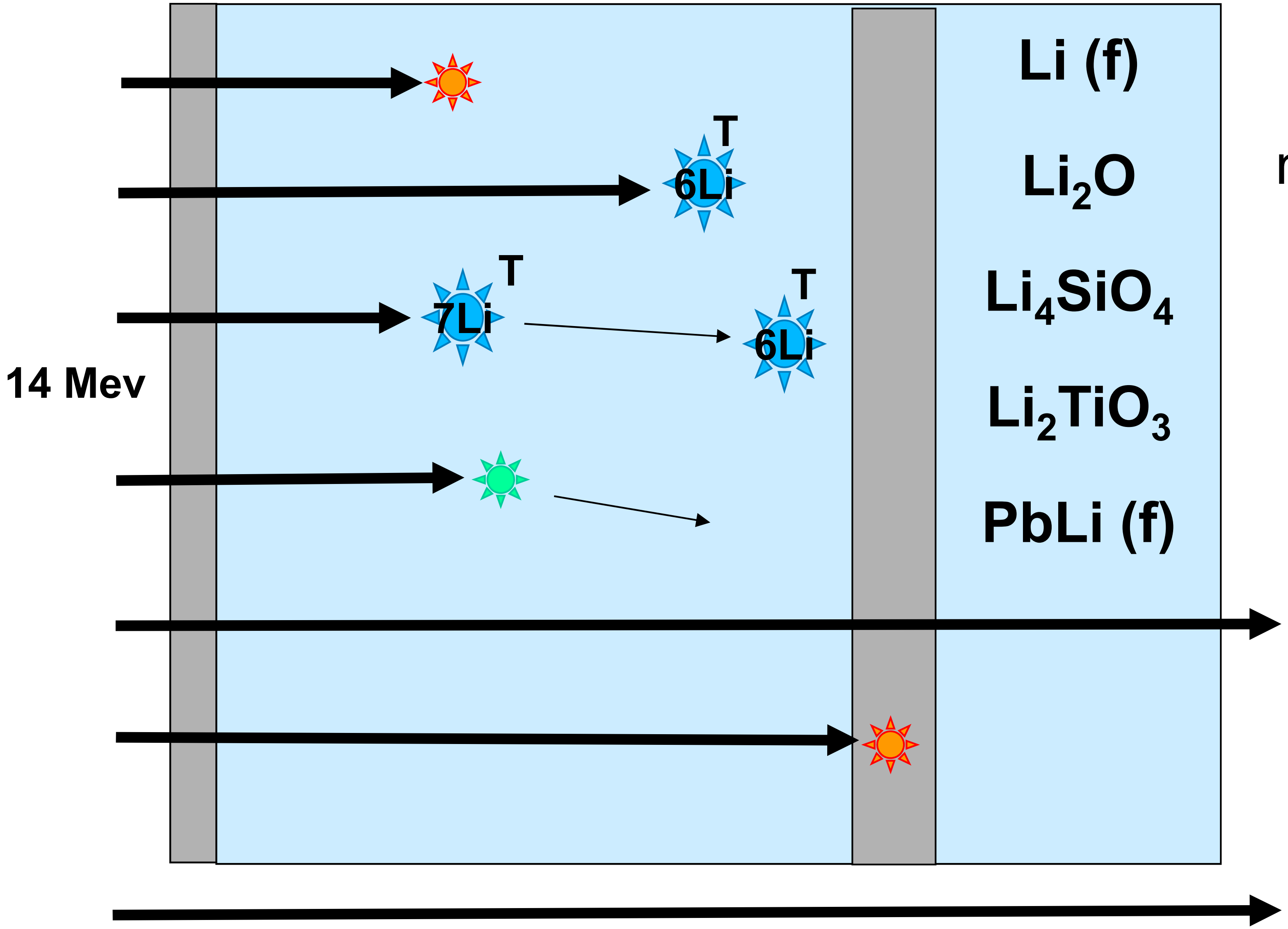
The ^{6}Li reaction has a very high cross section especially in the low-energy region.

The ^{7}Li reaction works with the high energy neutrons. It produce an additional n that is available for a successive reaction.

Natural mixture: ^{7}Li 92.5 % at., only 7.5 % ^{6}Li

About 10^{11} kg Lithium in landmass: sufficient for 30'000 years.
About 10^{14} kg Lithium in oceans: sufficient for 30 million years !!

Production of Tritium in Blanket



$n_T = n_f * X_b * X_T * m$

shall be $n_T / n_f > 1$

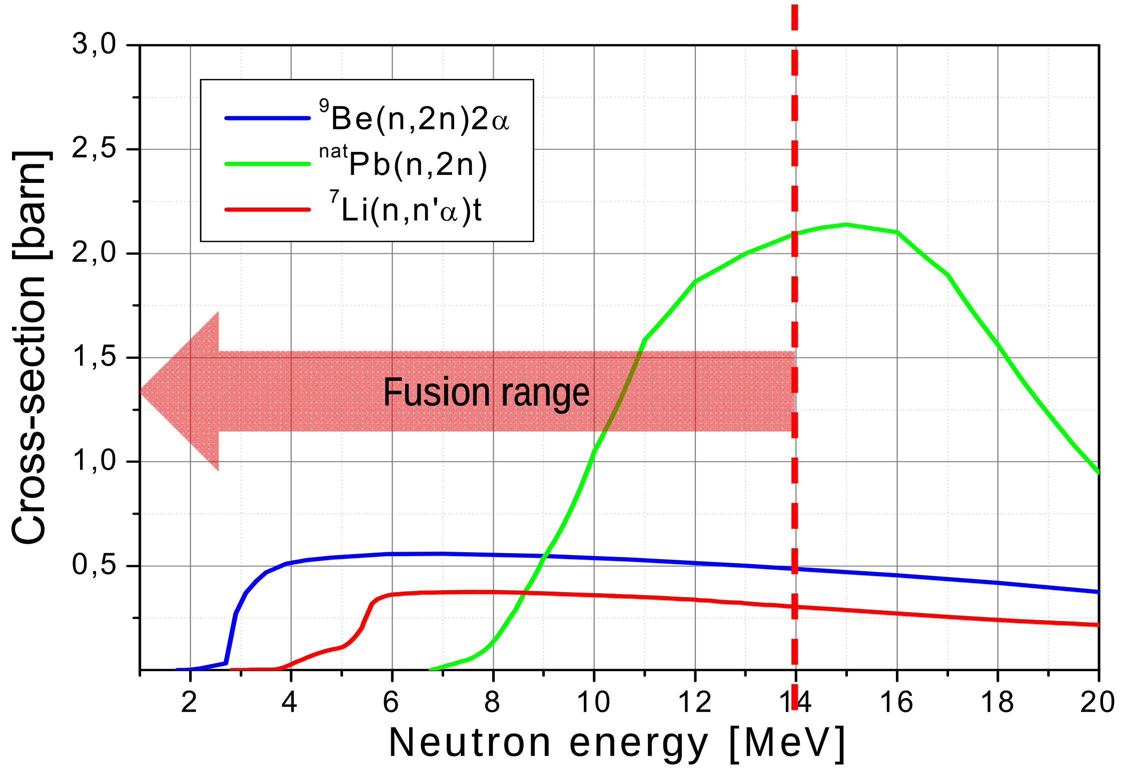
Neutron Multiplication

Required (n,2n) reactions with high σ in Energy range up to 14MeV

Li(nat):
- Sufficient, but only with very low n-absorber materials
- Strong reaction with water and air
- Getter for T (difficult recovery)

Beryllium (Be):
- low E threshold for (n,2n)
- good moderator (shielding)
- small world resources
- exothermal reaction with water (H production) beyond 600°C
- Be dust are toxic

Lead (Pb):
- high availability, low cost
- can be used as coolant
- corrosion with material (e.g. steels)
- weight
- activation through Po formation
- Melting point ~235°C
- (moderate) reaction with water



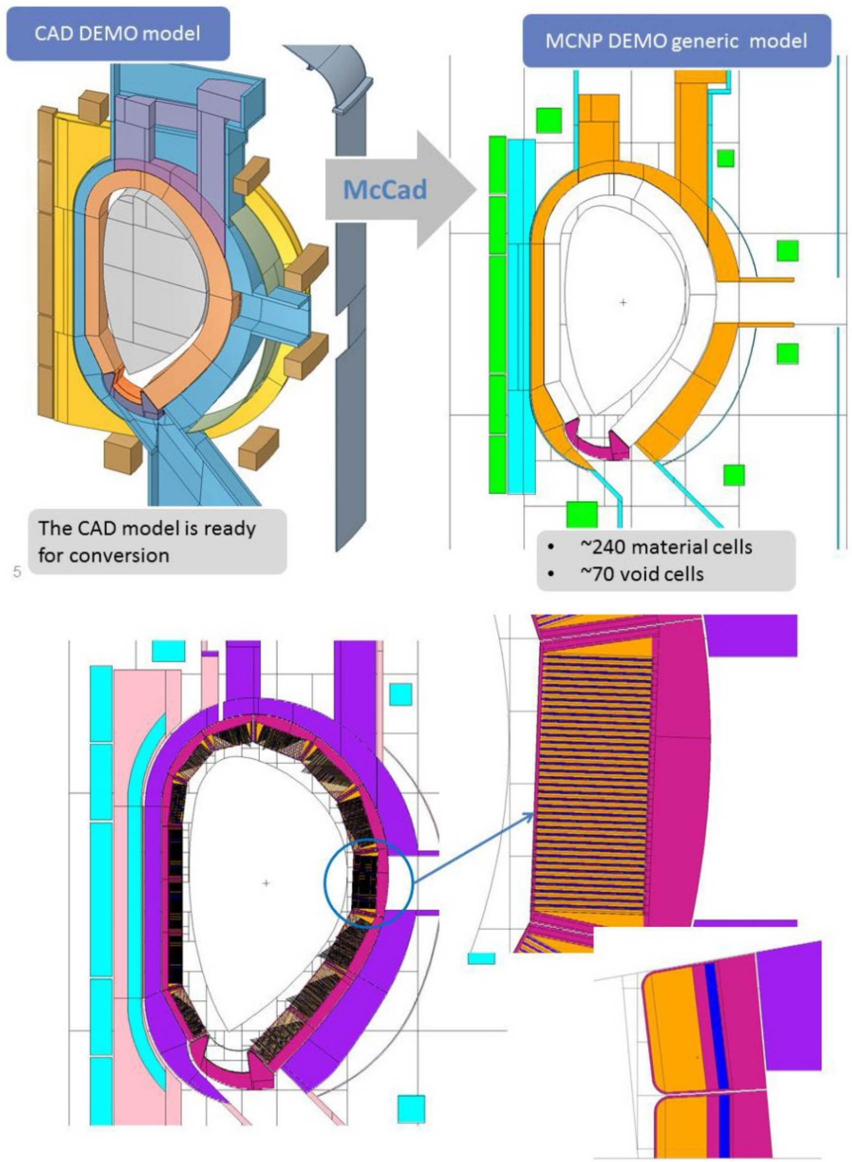
Possible strategies:
- Li(nat) with V as structural material.
- **Better** => Adding a more effective multiplier (Be or Pb) and increasing the ^{6}Li enrichment (40%-90%) to use efficiently neutrons at lower energies.

Tritium Breeding Ratio

$$TBR = \frac{\text{number of tritons produced per second in blanket}}{\text{number of fusion neutrons produced per second in plasma}} > 1$$

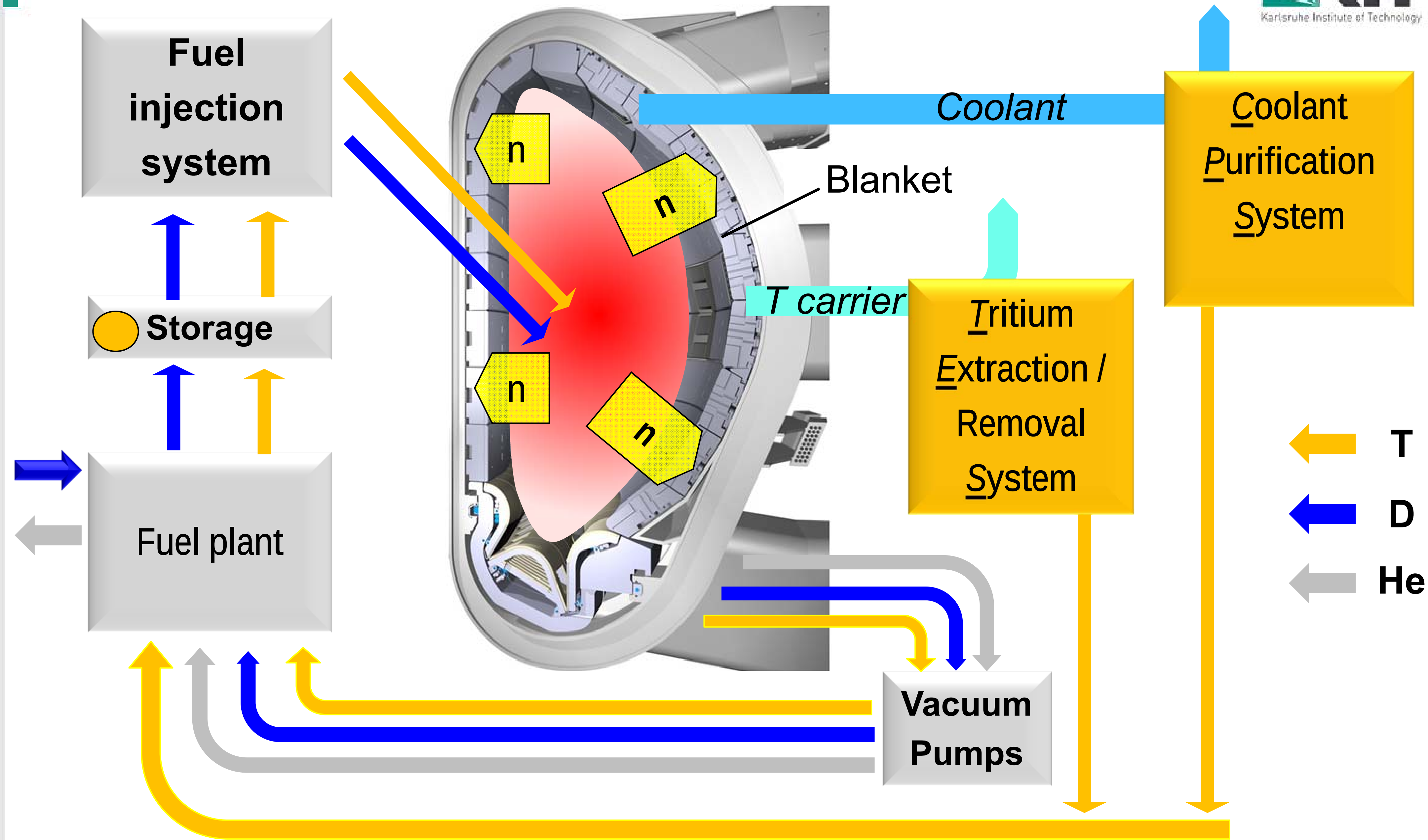
How large the TBR should be actually be during the lifetime of the FPP is depending on the “dynamics” of the entire fuel cycle for the D-T plant. It account for:

- possible losses of tritium in the different parts of the process and tritium trapping in the materials
- Reduction of breeding in the time with the Li burn-up
- duplication time (time necessary to produce the tritium amount necessary to start a new FPP)
- storage capabilities
- natural decay of tritium (half-life of 12.3 y)



Pavel Pereslavtsev, KIT

Fuel Cycle: inner and outer cycle



Blanket - Operational functions- „Structures“

Additional effects

➡ **<u>Some effects reduce the available structural material quality</u>**

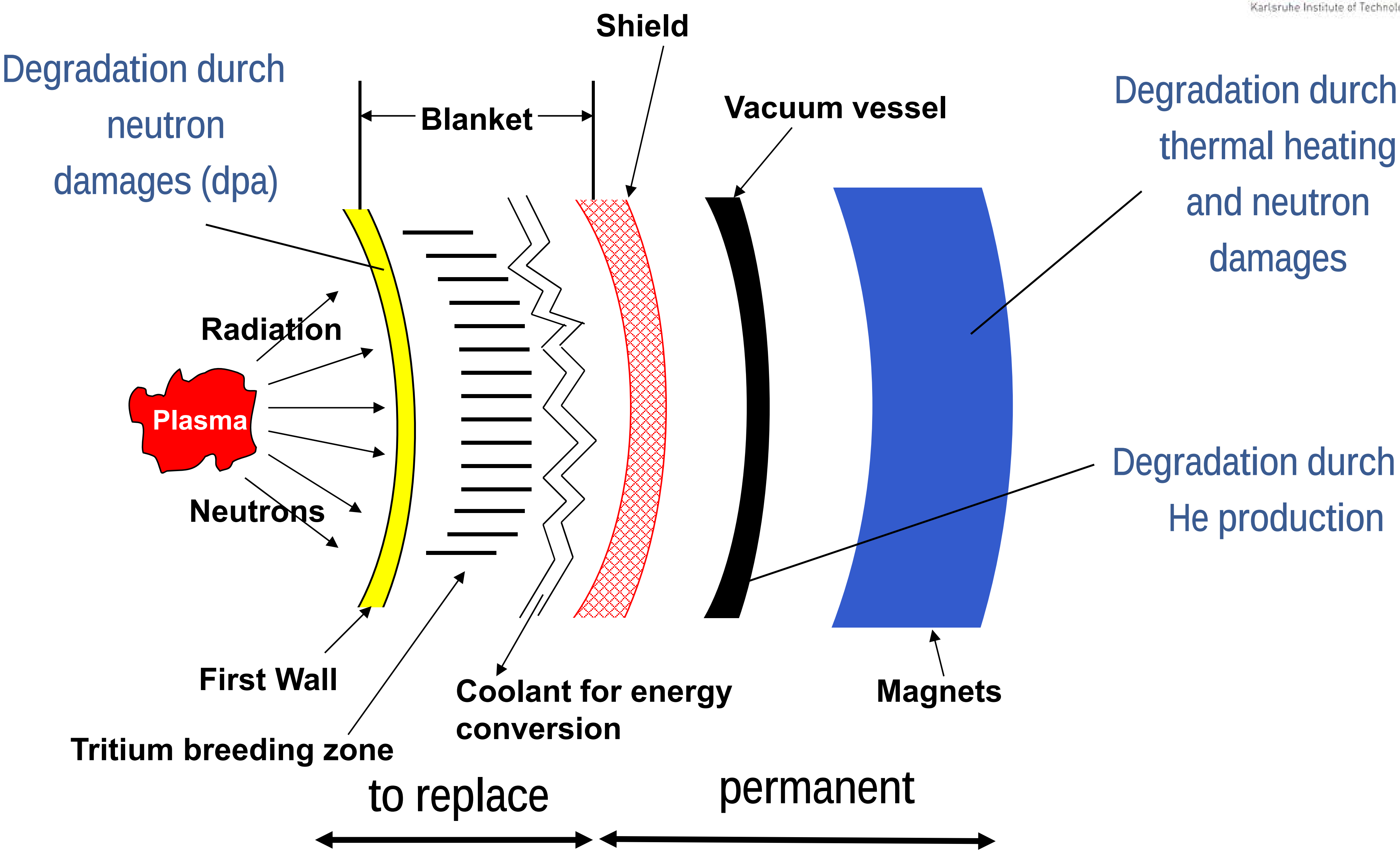
- **Cyclic loads**
- **Thermal creep**
- **Displacement of lattice atoms through collisions (radiation damage) in dpa**
- **Generation of gases (He, H_2) through transmutation reactions (e.g. (n,α)) in appm/year**

First wall of blankets at 3 MW/m^2 neutron wall load:

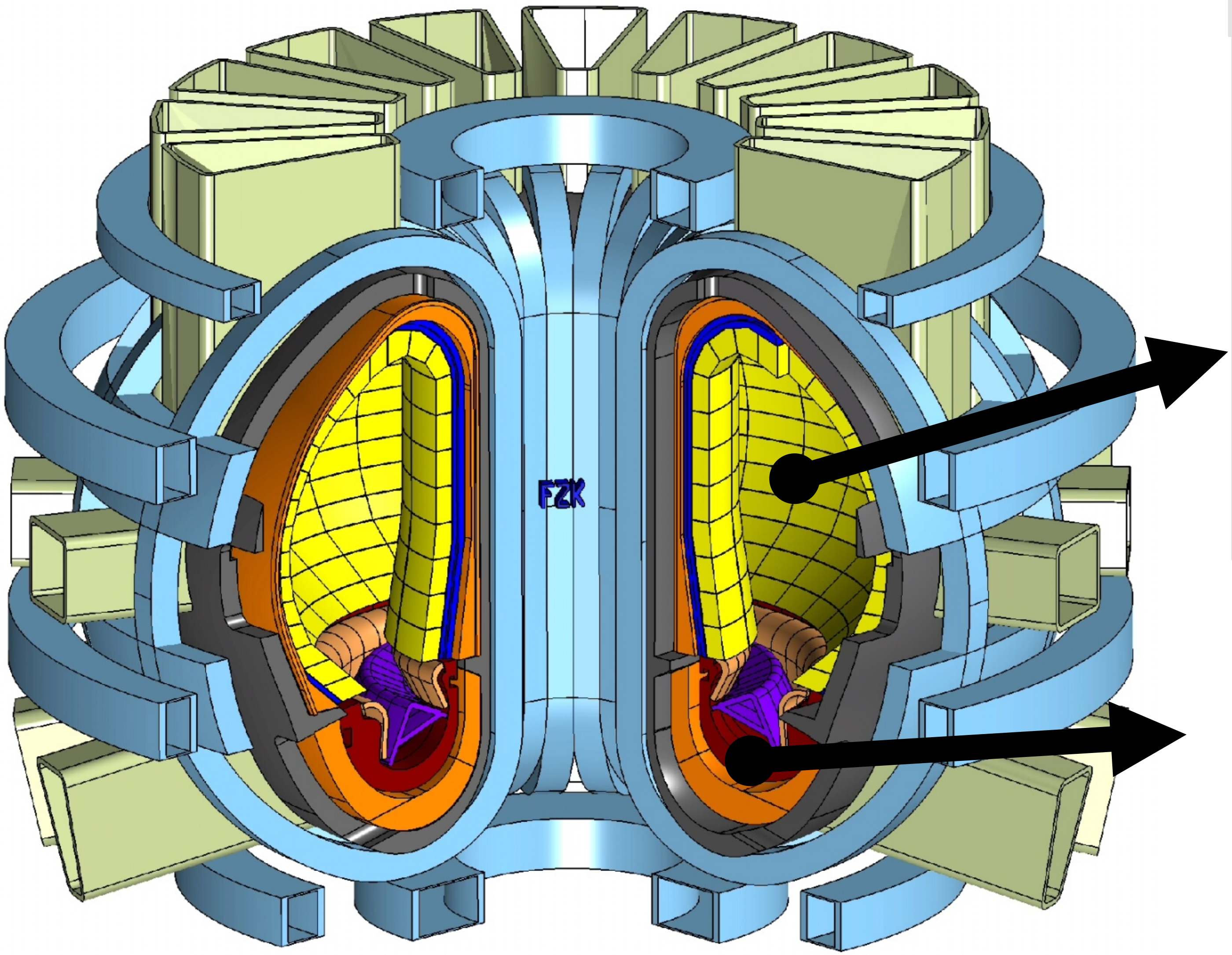
36 dpa/year and **440 appm/year**

σ_m is function of time, temperature and irradiation history

Component damages: life time



Thermonuclear Core replacement



Blanket

-T production

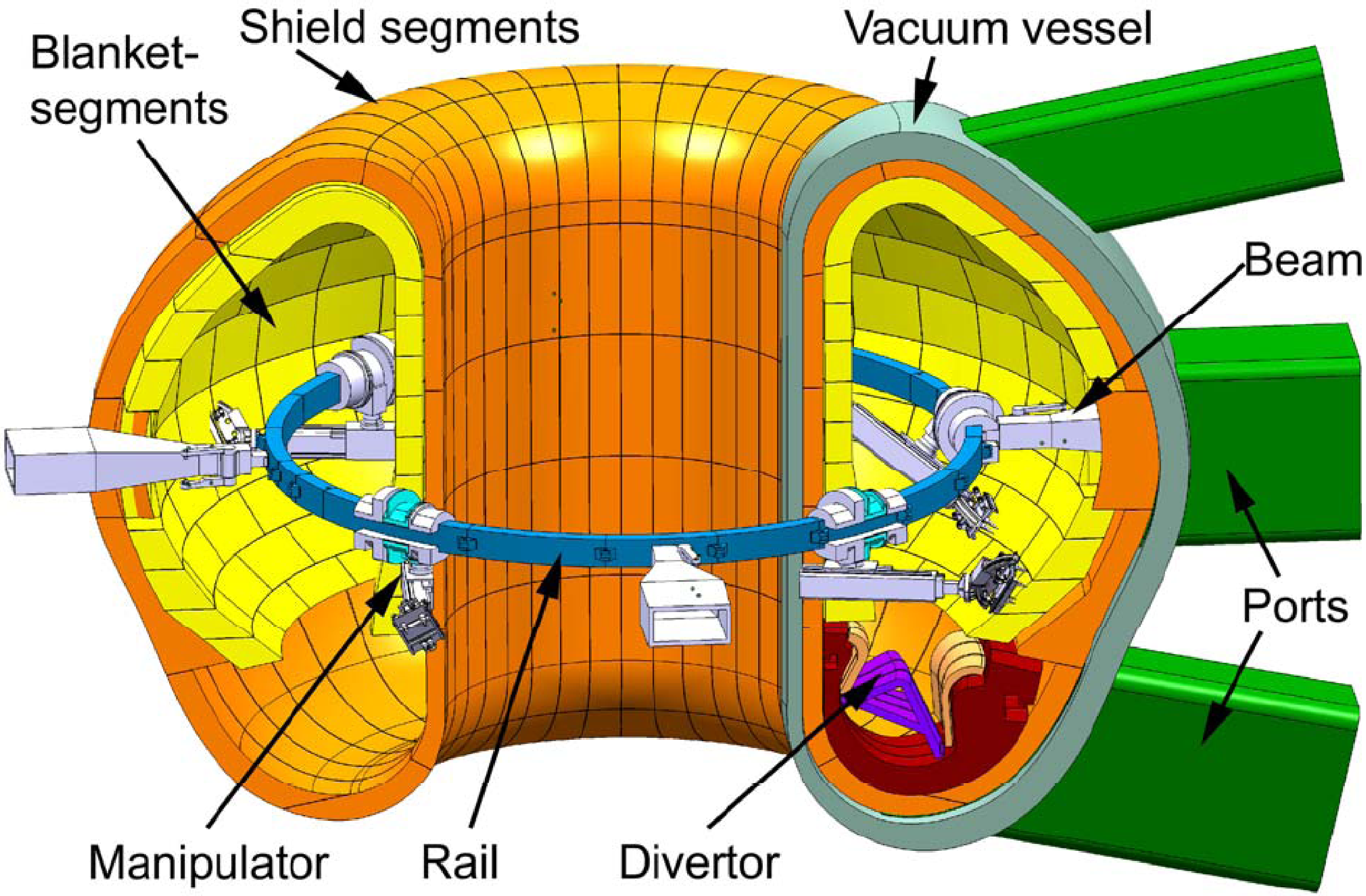
- high temperature coolant for electricity production

- Remote replacement 3 - 5 years (80...150 dpa)

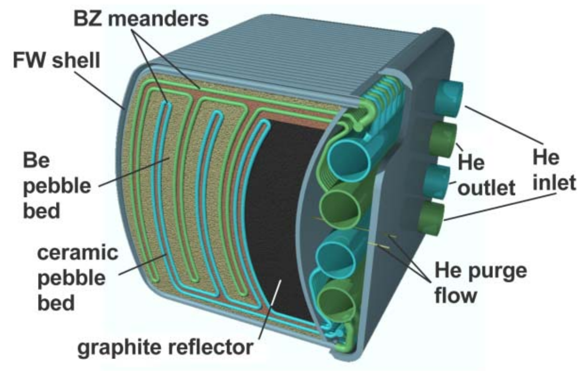
Divertor

- Exhaust alfa and impurities from plasma
- Remote replacement 1.5 - 2.5 years

Maintenance system: ITER Configuration



Blanket Module

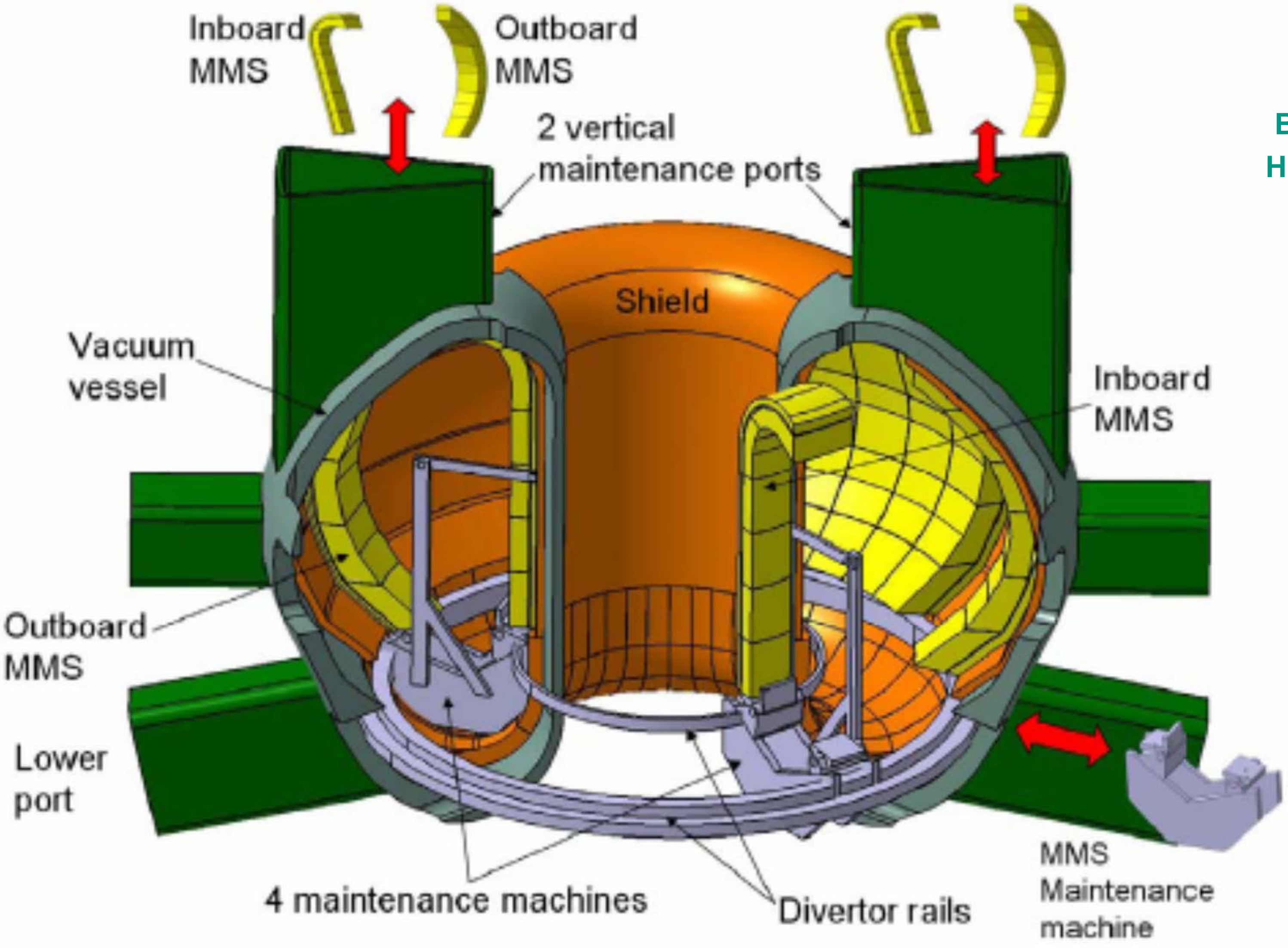


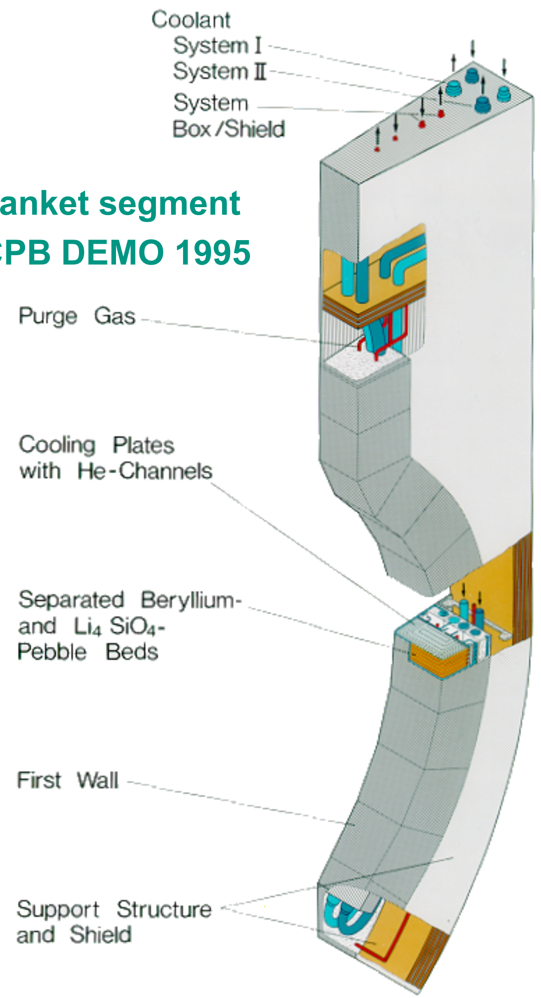
E.g. 18 modules per sector

x 16 sectors

=> 288 modules

Maintenance System: Vertical port

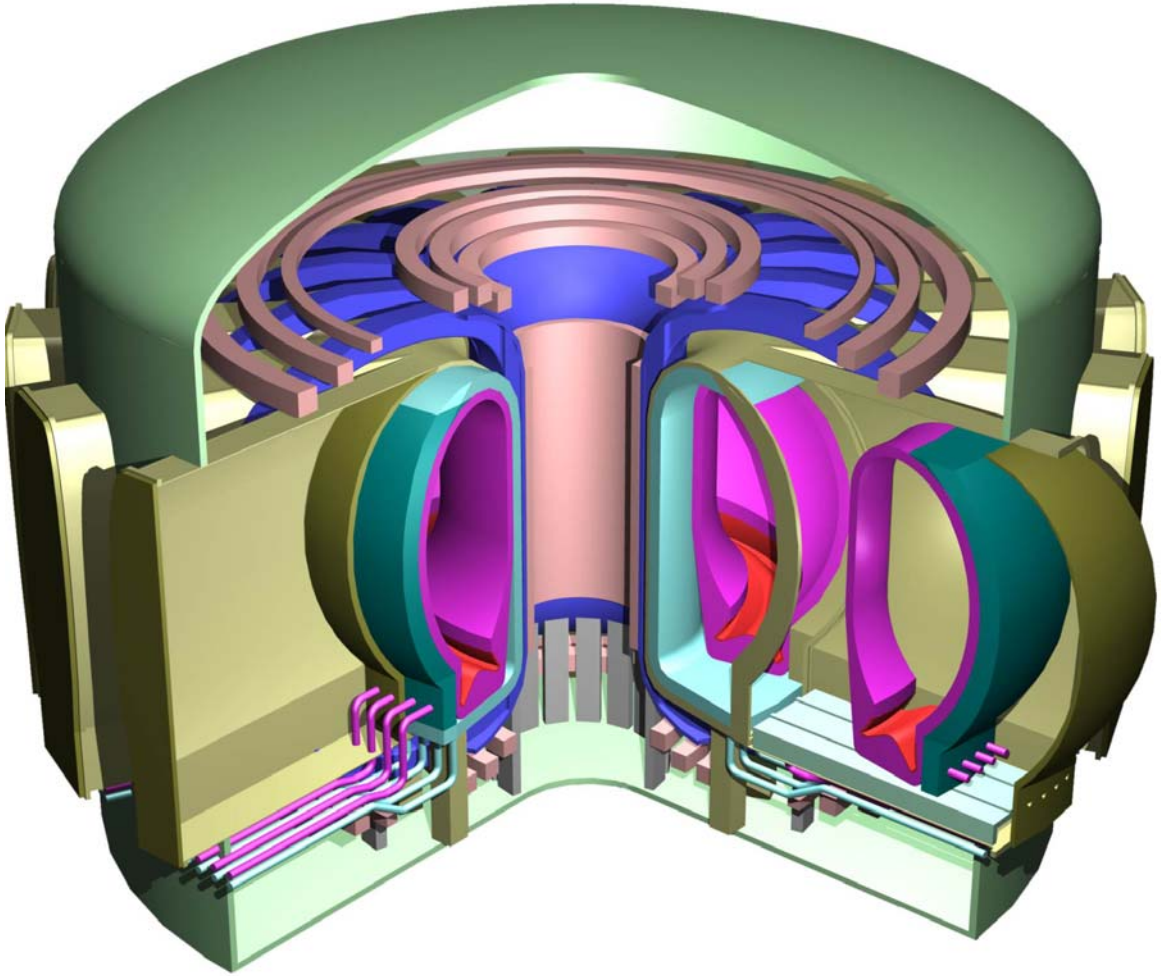




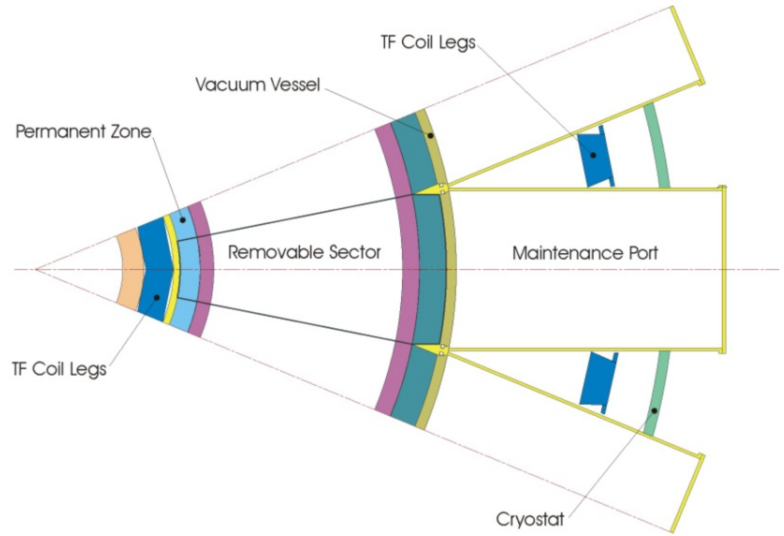
Blanket segment
HCPB DEMO 1995

5 segments per sector
x 16 sectors
=> 80 segments

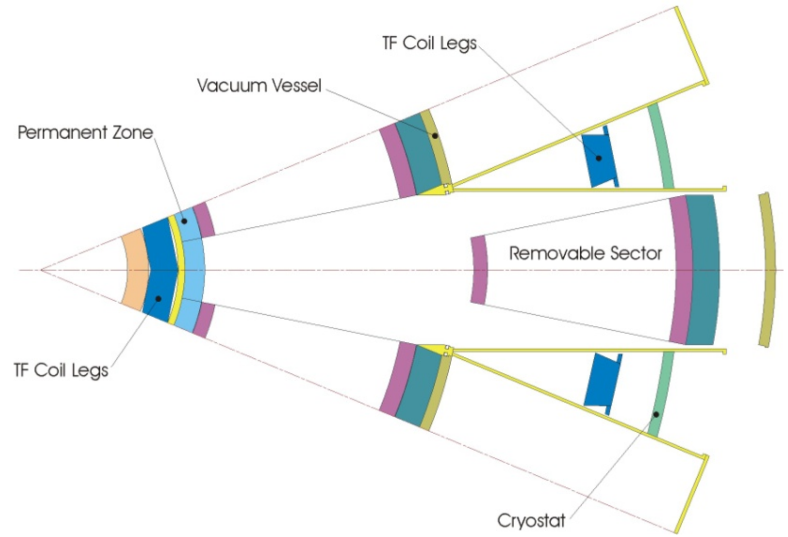
Maintenance System: Large sector concept



Aries Studies, US

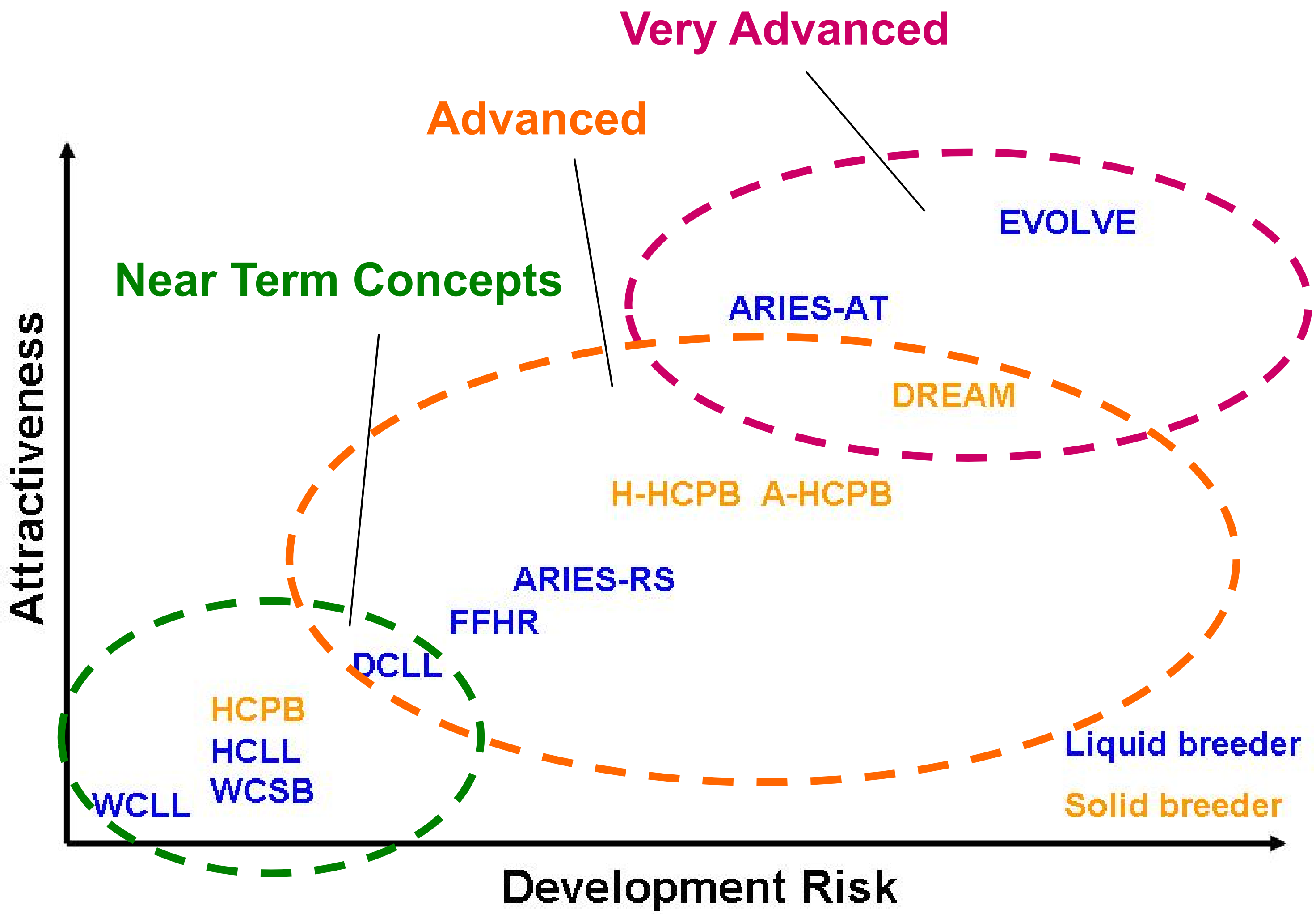


=> 16 sectors



Maintenance concept analogy

Possible Blanket Concepts



Not exhaustive

Classification according to:

– Maturity level (near term->Very Advanced)

– Structural material (e,g. steel, SiC_f/SiC or V-alloy)

– Breeder / multiplier (solid and liquid breeder)

– Coolant (water, gas, liquid metal, molten salt)

– Heat and T extraction (e.g. Self Cooled, Dual Coolant)

Different Blanket Concepts in the “near term” class

Concept class	Structural Material	Breeder/Multiplier	Coolant	T-extraction	Breeding Blanket Concept for DEMO and ITER testing
HCSB	RAFM steel	Ceramic Breeder / Beryllium	Helium	Low pressure He purging	**EU HCPB DEMO + ITER test** China HCSB DEMO + ITER Test Korea HCSB DEMO + ITER Test
WCSB	RAFM steel	Ceramic Breeder / Beryllium	Water	Low pressure He purging	JA HCSB DEMO + ITER test China WCSB DEMO Korea WCSB DEMO
WCLL	RAFM steel	Lithium Lead (Pb15.8Li)	Helium	LL slow circulation	**EU WCLL DEMO**
HCLL	RAFM steel	Lithium Lead (Pb15.8Li)	Water	LL slow circulation	**EU HCLL DEMO + ITER test**
DCLL	RAFM steel	Lithium Lead (Pb15.8Li)	Helium PbLi	LL fast circulation	**EU DCLL-LT DEMO** US DCLL-HT DEMO

Legenda:

HC: Helium cooled
WC: Water Cooled
RAFM: Reduced Activation Ferritic-Martensitic
SB: Solid Breeder
LL: Lithium Lead
DC: Dual Coolant
LT: Low Temperature
HT: High temperature

Structural material: which? and which requirements?

EUROFER: is a ferritic martensitic steel: C12%, Cr9%. Very low content of Ni, Mo, Nb. Substituted by V, W and Ta. Well know material, licensing process under irradiation ongoing.

SiC_f/SiC: Silicon carbide composites are attractive as structural materials in fusion environments because of their low activation, high operating temperature and strength.

V-alloy: it exhibit favorable neutronic properties which include lower parasitic neutron absorption leading to better tritium breeding performance (e.g Li-V blankets).

Requirements:

Withstand vs. radiation damages: e.g. EUROFER target 70 up to 150 dpa.

Temperature window: EUROFER >300-550°C, SiC_f/SiC ~600-1100°C, V-4Cr-4Ti ~400-650°C

Compatibility with breeder/multiplier and coolant (e.g corrosion): EUROFER is compatible (up to 550°C with Solid Breeder and Be); corrosion with PbLi is an issue starting from 450°C. Water corrosion.

Thermal properties: conductivity, thermal expansion, allowable stress, etc.: good for EUROFER and V-alloy. Worst for SiC_f/SiC due to low thermal conductivity.

Reduced activities characteristics: e.g EUROFER can be recycled after ~100 years storage.

Code and Standards: e.g. EUROFER is under an EU programme with the aim to qualify it for ITER up to 3 dpa (in 2018) and up to 20-70 dpa for DEMO (~2030-50).

Coolants used in the Breeding Blanket

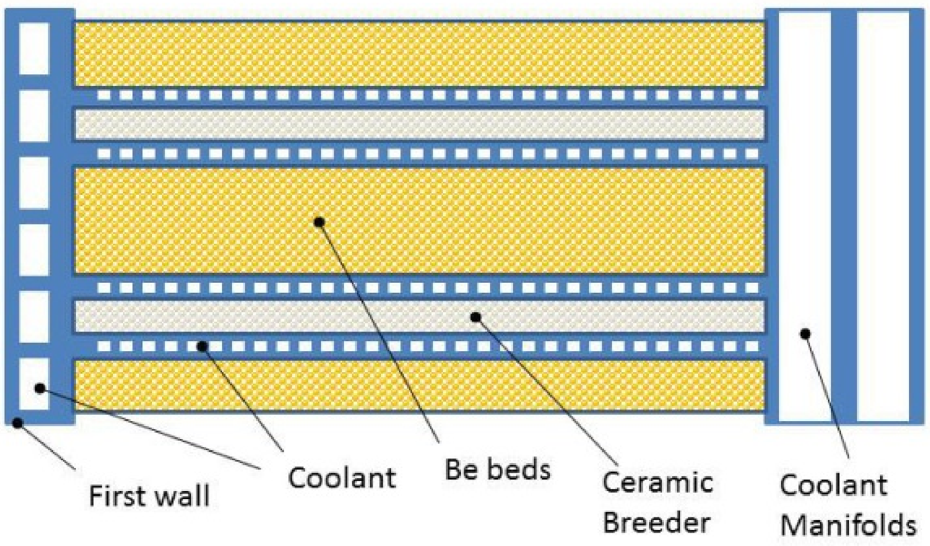
Water: exceptional cooling capability. High density that allow small flow section. Low ΔT in Blanket. PWR range (275-315°C @15.5 MPa): temperature at the lower region for use with F/FM steels. Corrosion. Issue with T contamination. Chemical compatibility.

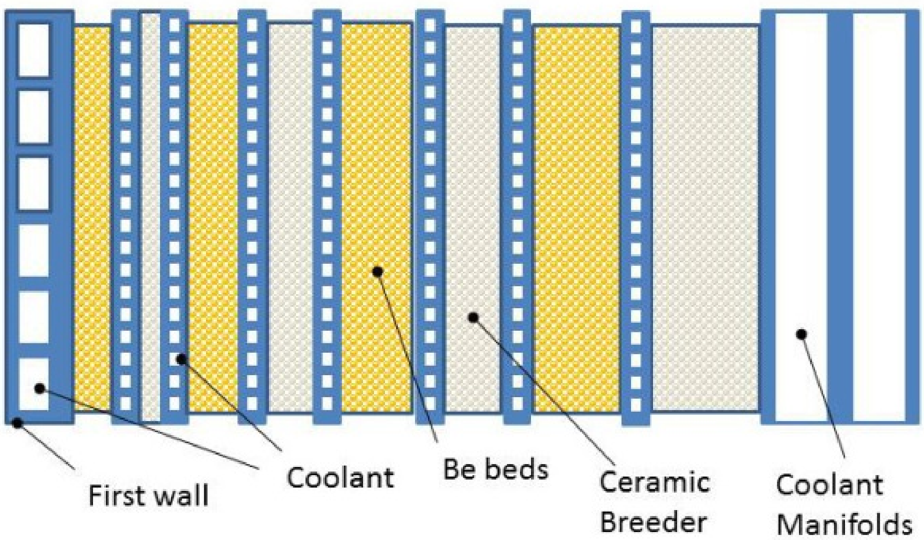
Helium: exceptional compatibility with the material used in blanket and other part of the reactor. Possibility to cope with all the temperature windows of the materials. Lower heat removal features and higher pumping power. Large tubes with low shielding features (issue for the reactor integration of pipes and manifolds).

Liquid Metal (PbLi and Li): high heat removal capability but strongly conditioned by MHD effects (suitable only if insulation barriers with conducting structures are available). Low pressure. Used as breeder, they must accomplish the double functions of heat removal and T transport. Corrosion and chemical compatibility issues.

Molten salt (FLiBe): Low pressure. Must accomplish the double functions of heat removal and T transport. High corrosion issues. Low thermal conductivity. Difficult chemistry.

Solid Breeder Blanket concepts





Main features of breeding zone organization:

- the solid material fills the blanket box that is formed by structures actively cooled for heat extraction;
- Breeder and Be usually in form of a pebble bed;
- Breeder cooled by these external plate or tube;
- T extracted from the breeder by means of an independent low pressure gas flow in order to extract the T produced (double chambered components)
- The two chambers are divided by thin walls for a very large surface. This results, however, in the permeation of tritium from the breeding beds to the coolant which is a critical issue.
- Beryllium is required in a large amount (>3:1 volume ratio in comparison to the breeder)
- Bed arranged in thin layers of few cm (horizontal or vertical configuration).

Solid Breeder materials

Solid breeders proposed in pebble form to reduce further fragmentation and control thermal conductivity of the beds. Possible materials: Li-oxide (Li_2O) and ternary ceramics, like Li-orthosilicate (Li_4SiO_4), metatitanate (Li_2TiO_3), metazirconate (Li_2ZrO_3) or allumitate ($LiAlO_3$).

Ideal requirements:

- quick and easy tritium release → reduction of T inventory,
- high Li density → improved neutronic performance
- high mechanical resistance → reduce fragmentations,
- high temperature windows
- compatibility with other materials (including water in the air moisture for out-of-reactor handling),
- simple fabrication and recycling routes.

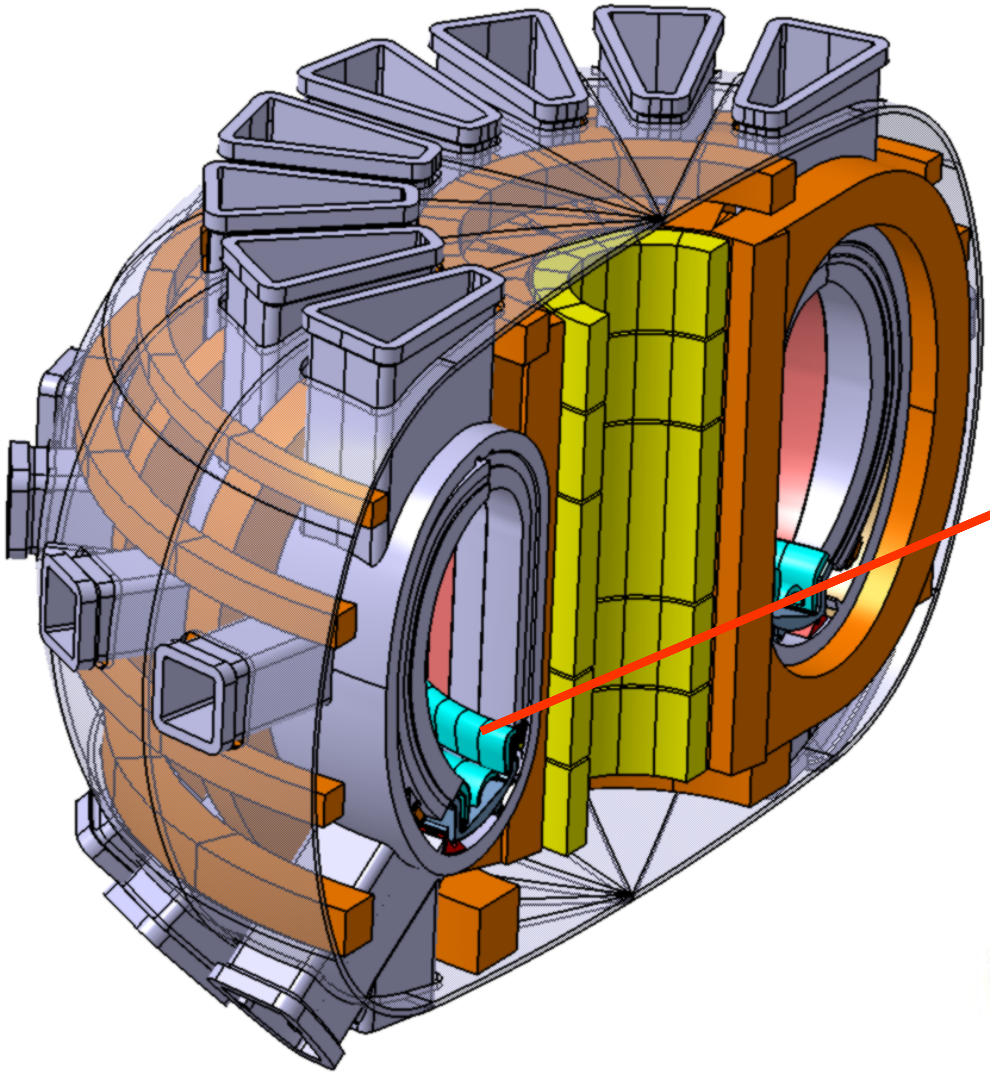
Li_4SiO_4	**medium lithium content,** fair mechanical properties, hygroscopic, fair tritium residence time, higher thermal expansion
Li_2TiO_3	**low lithium content,** good mechanical properties, not hygroscopic, fair tritium residence time, lower thermal expansion
LiO_2	**highest lithium content,** good thermal conductivity; large thermal expansion, poor mechanical behavior, precipitate formation (LiOH) → loss of Li

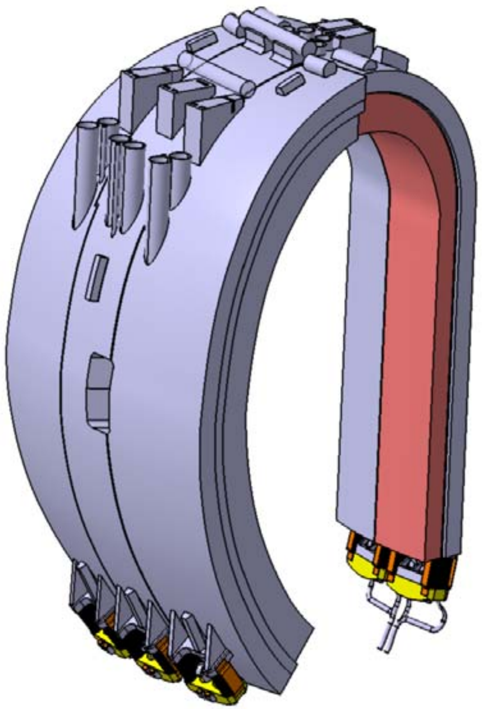
Li-ceramic breeder list with main properties

Items \ Materials	Li_2O	Li_2TiO_3	Li_2ZrO_3	Li_4SiO_4	γ-$LiAlO_2$
Li Density (g/cm^3)	0.94	0.43	0.38	0.51	0.27
Thermal Conductivity (500°C)•(W/m/°C)	4.7	2.4	0.75	2.4	2.4
Thermal Expansion (500°C)•(DL/L_0%)	1.25	0.8	0.50	1.15	0.54
Reaction of Water	very	less	less	little	little
Residence Time (440°C)•(h)	0.03	(-)	0.01	3.0	50
Swelling* (DV/V_0%)	7.0	(-)	<0.7	1.7	<0.5
Transmutation Nuclides	^{16}O(n,p):7s	^{46}Ti(n,p):84d ^{47}Ti(n,p):3.4d ^{48}Ti(n,p):1.8d	^{90}Zr(n,p):64h ^{91}Zr(n,p):57d ^{94}Zr(n,2n):10^6y ^{96}Zr(n,2n):64d	^{28}Si(n,2n):4s ^{29}Si(n,p):6m ^{30}Si(n,α):9m	^{27}Al(n,2n):6s ^{27}Al(n,p):9.5m ^{27}Al(n,α):15h
Melting Point (°C)	1430	1550	1615	1250	1610

HCPB: EU DEMO reactor integration

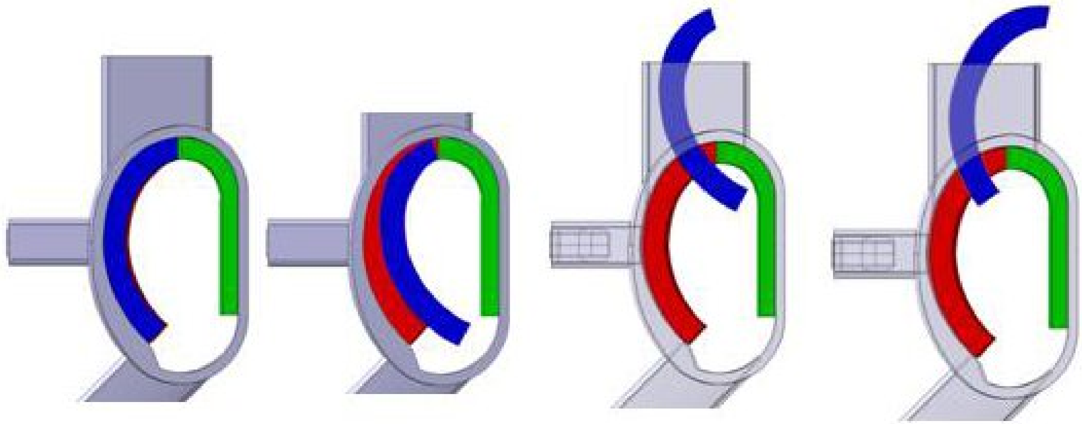
Karlsruhe Institute of Technology





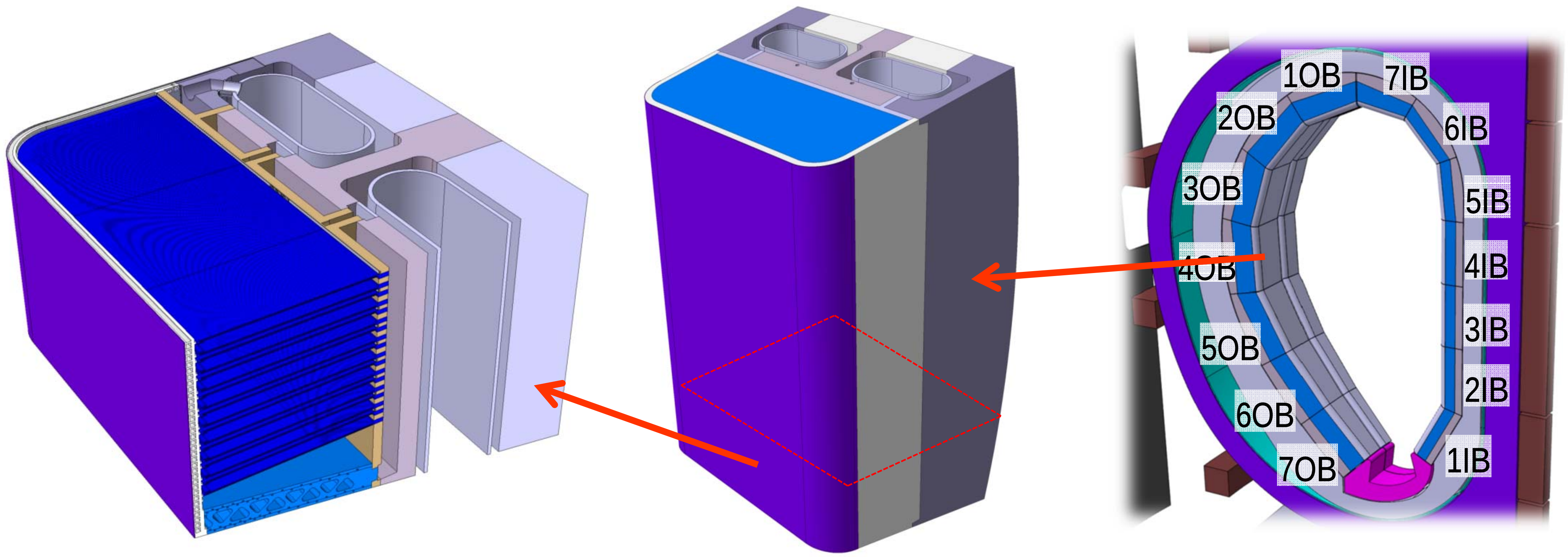
5 blanket segments (3 OB and 2 IB) for each 22.5°- sector (16 TF coils)

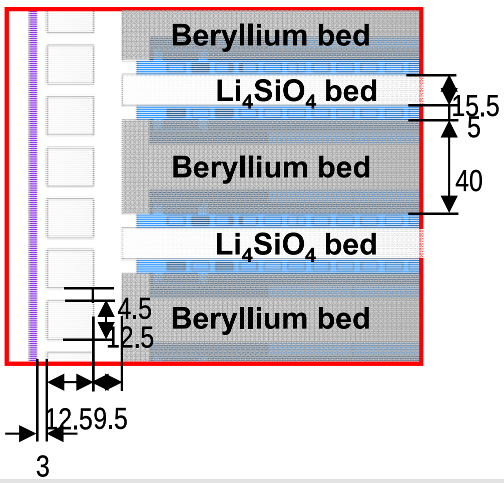
Daniel Iglesias et al., Blanket Segment Remote Maintenance, EFDA-WP13-DAS07-T05 (CCFE)



Example of kinematics for OB segment extraction

HCPB Blanket: EU PPPT-DEMO (KIT, 2015)

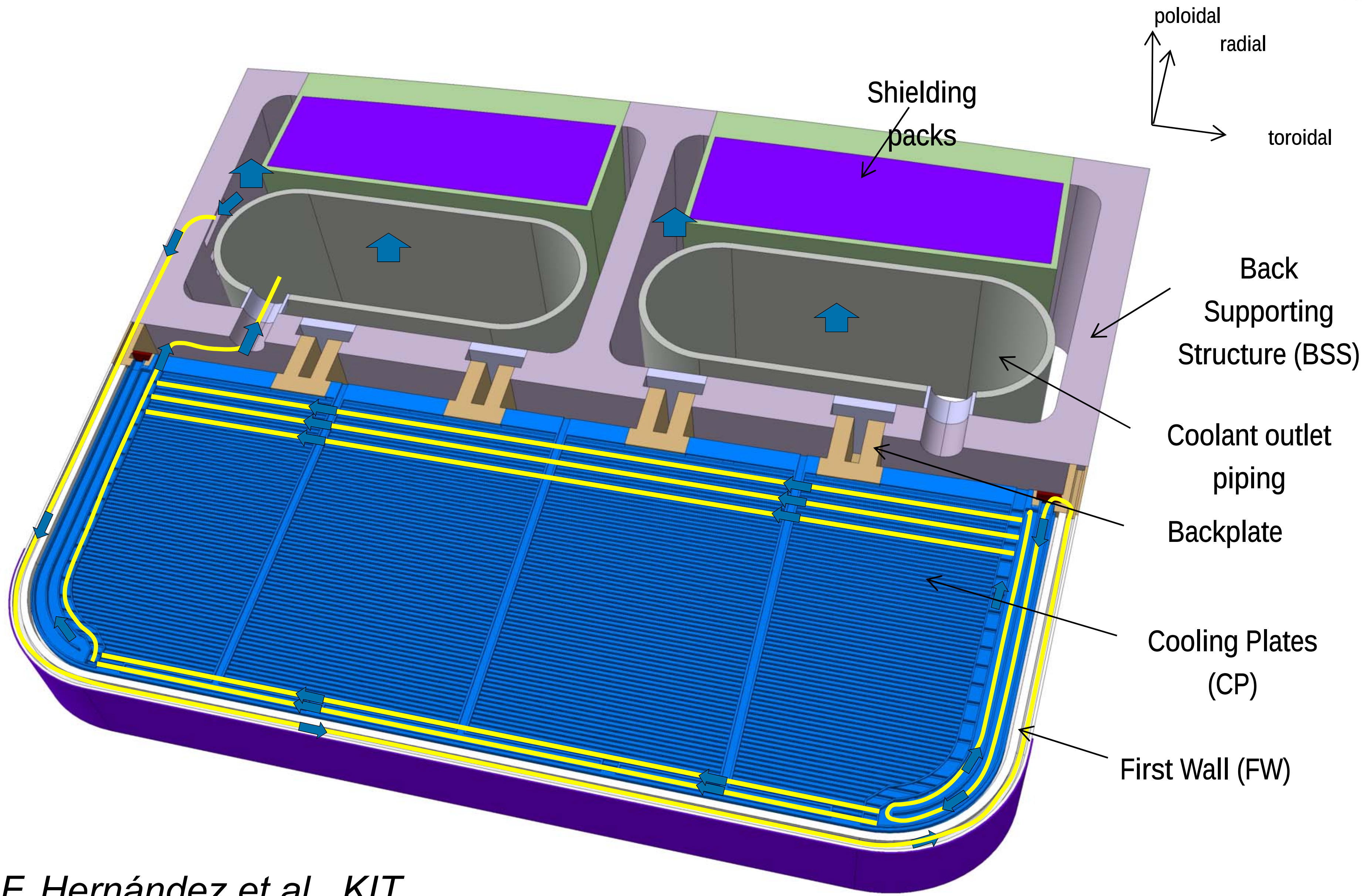




Box dimensions: 1.8m (pol) x 1.4m (tor) x 1.6m (rad)

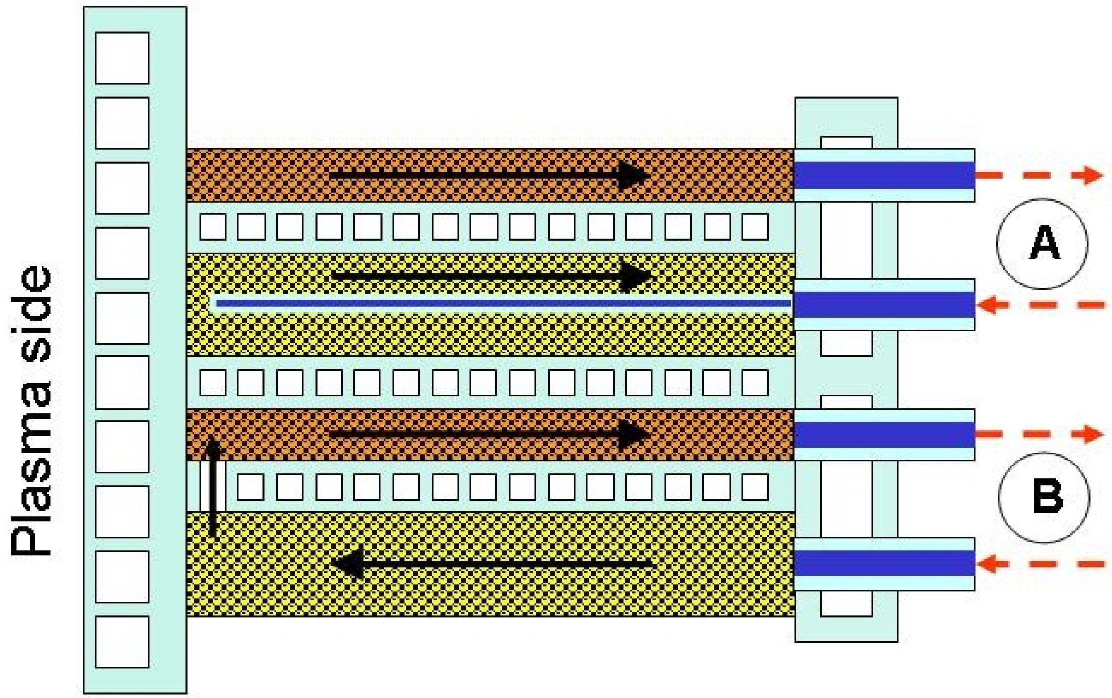
Operational parameters	Values
FW heat flux (peak)	0.5 MW/m^2
Neutron wall load (peak)	1.7 MW/m^2
He Coolant pressure	8 MPa
He Coolant temp. (inlet/outlet)	300 / 500°C
Power Generation System	Water-steam Cycle (Rankine)

HCPB Blanket: Coolant flow scheme



F. Hernández et al., KIT

HCPB: Tritium extraction concept



To possible variants: in B the radial direction in the Be beds is reversed. The direction in CB is important for T control.

Purge flow:

Tritium extraction in CB and Be.

Independent loop.

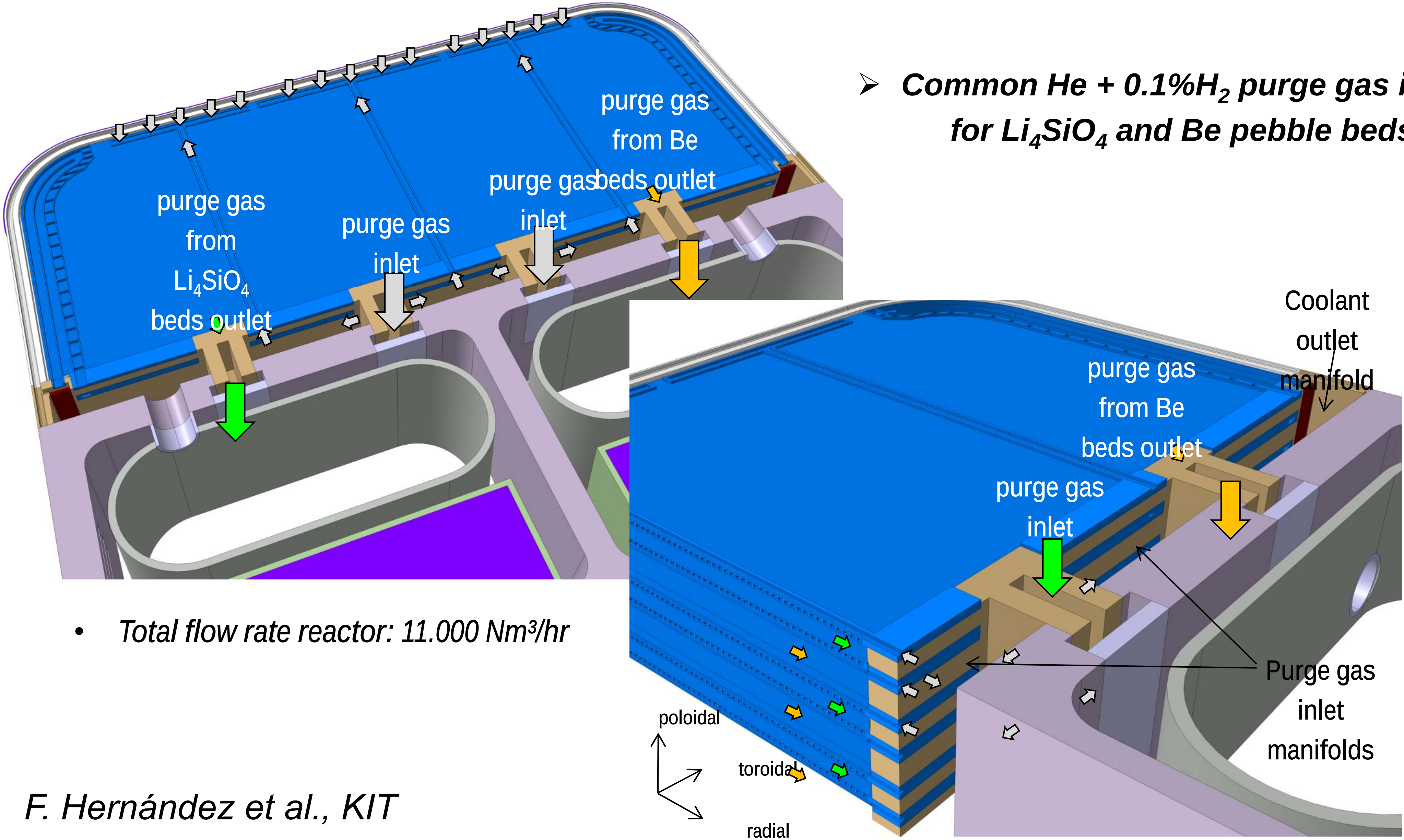
Pressure ~0.1-0.2 MPa

Chemical composition (H_2 addition, ref 0.1% to enhance T extraction from CB)

Mass flow: to optimise in order to reduce H circulation and minimise T partial pressure (~10 cm/sec).

CB=Ceramic Breeder

HCPB Blanket: Purge flow scheme

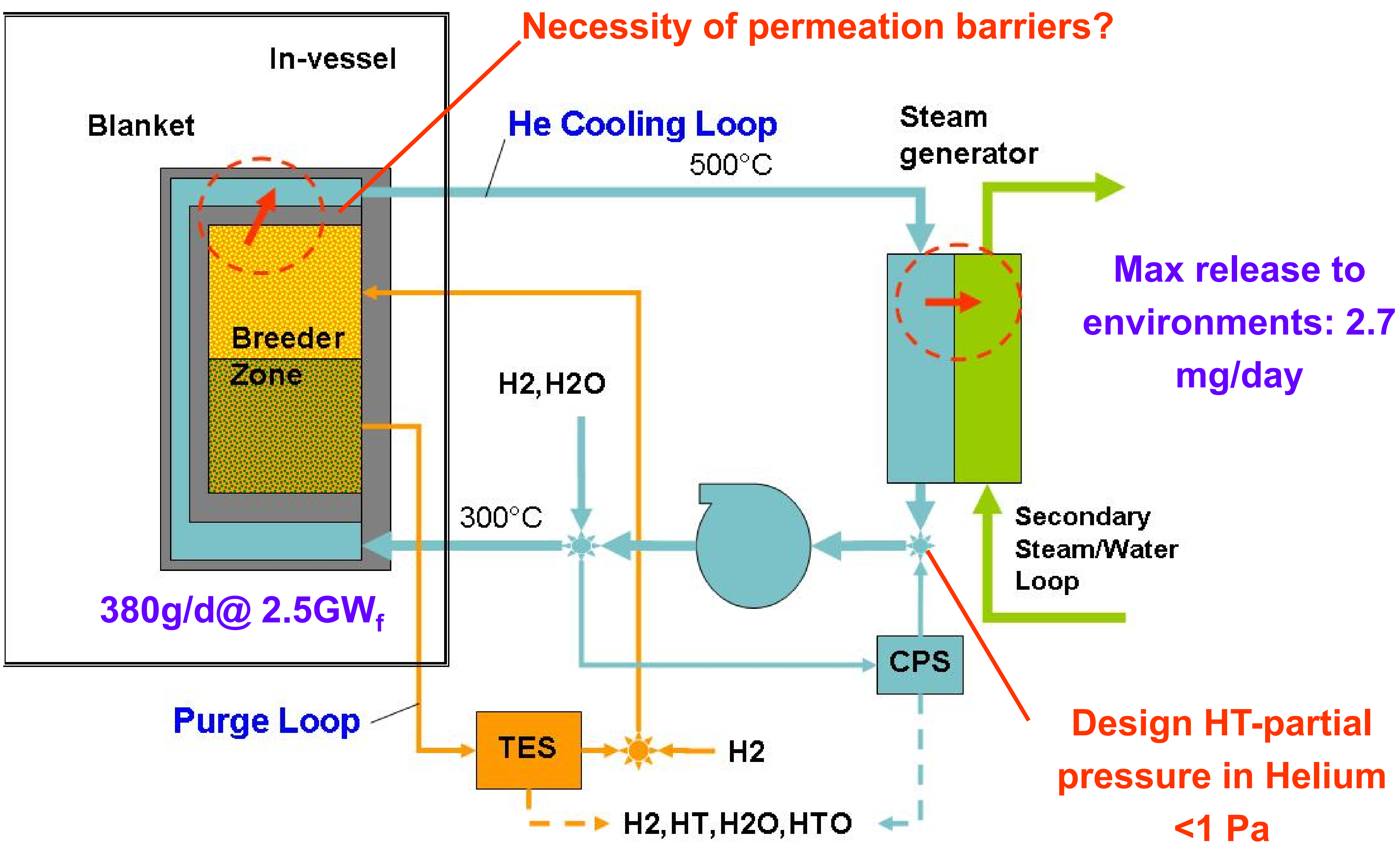


➢ ***Common He + 0.1%H$_2$ purge gas inlet for Li$_4$SiO$_4$ and Be pebble beds***

- *Total flow rate reactor: 11.000 Nm³/hr*

F. Hernández et al., KIT

HCPB Blanket Concept: Tritium Control

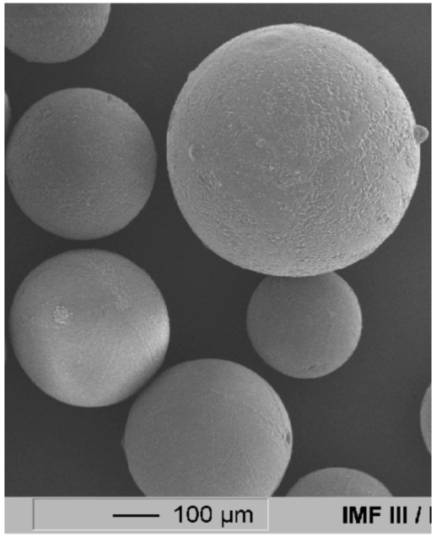


Li-ceramic pebbles in the HCPB

- Two ceramic breeder (CB) materials have been extensively investigated in the last years, namely the Lithium Orthosilicate (Li_4SiO_4) and Metatitanate (Li_2TiO_3).

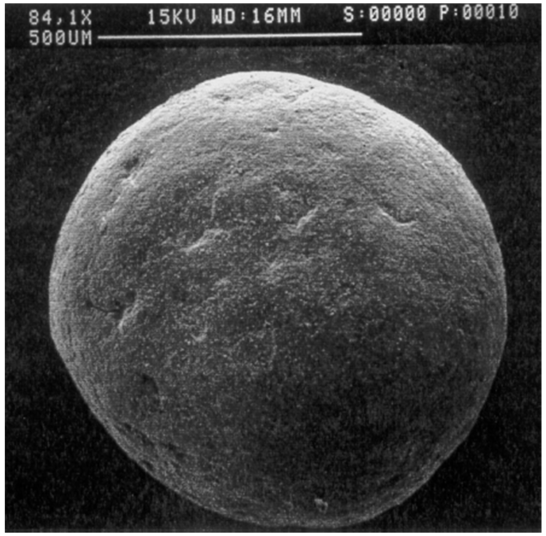
- In these years, pebbles made with both the materials have reached a semi-industrial production stage;
  - the Orthosilicate pebbles (KIT-Schott) are produced by melting and spray process
  - the Metatitanate pebbles (CEA-CTI) by extrusion-spheronisation-sintering process.
  - KIT-KALOS: 80 mol% lithium orthosilicate + X mol% lithium metatitanate (X=20-35) [R. Knitter and al, Fabrication of modified lithium orthosilicate pebbles by addition of titania, Journal of Nuclear Materials, 442 (2013), S433–S436]

- The properties of the non irradiated pebbles are satisfactory for both the ceramics in term of geometry, chemical composition, mechanical properties, and microstructure.



Orthosilicate

d = 0.2 - 0.6 mm



Metatitanate

d ≅ 1.0 mm

See also: R. Knitter, P. Chaudhuri, Y.J. Feng, T. Hoshino, I.-K. Yu, Recent developments of solid breeder fabrication, Journal of Nuclear Materials 442 (2013) S420–S424

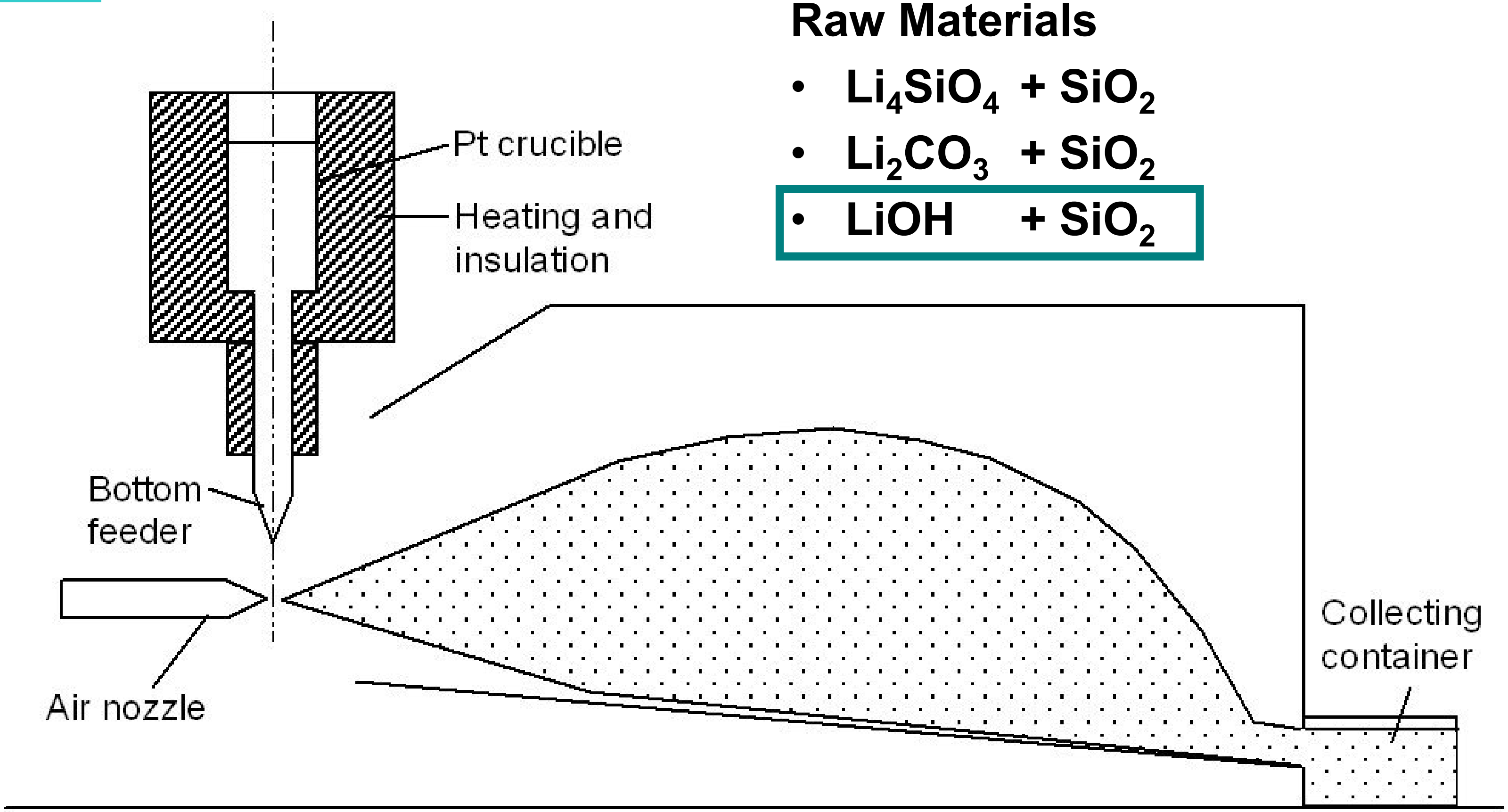
Li-orthosilicate fabrication

Melt-Spraying
Facility at
Schott Glas, Mainz

➔Aim Li_4SiO_4 + 2.5 wt% SiO_2

Raw Materials

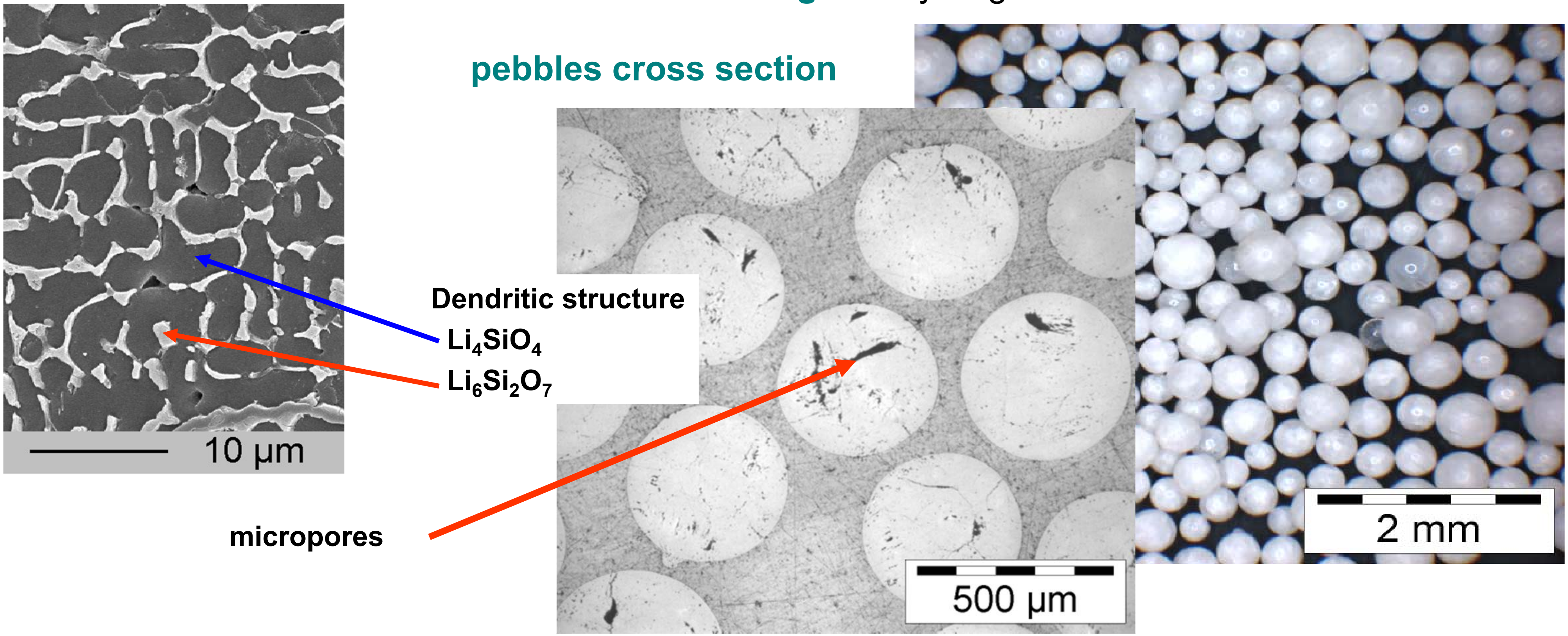
- Li_4SiO_4 + SiO_2
- Li_2CO_3 + SiO_2
- LiOH + SiO_2



Li-orthosilicate fabrication

Melt-Spraying Facility at Schott Glas, Mainz

Production	2 x 1.5 kg per day
Yield of screened pebbles	50 % (250 – 630 µm)
	200 – 300 kg per year
Drawbacks	Batch processing
	Limited control of fabrication parameters
➔	Variation of batch properties
Advantage	Recycling of material



T-residence time and inventory

- Pellets or pebbles are swept by He flowing during the reactor operation. Tritium in gaseous forms are carried by the He purge; adding H2 (as isotope swamping) to He the removal rate is improved
- Tritium residence time is the key parameter to calculate the tritium inventory in the ceramic materials. In the recent years, several correlations have been derived from results of in-pile tritium release experiments, either fitting of data from pebble bed experiments or extrapolating from existing data of materials in other forms. For the two ceramic materials, the following expressions of the tritium residence time τ [h] as function of the temperature T [K] have been used in the design calculations for HCPB blanket concepts:
- $\tau = 1.280\ 10^{-4} \exp(9720/T)$ for Li_4SiO_4
- $\tau = 1.995\ 10^{-5} \exp(10315/T)$ for Li_2TiO_3
- The tritium inventory I [g] in the breeder material at asymptotic conditions can be calculated as follow:

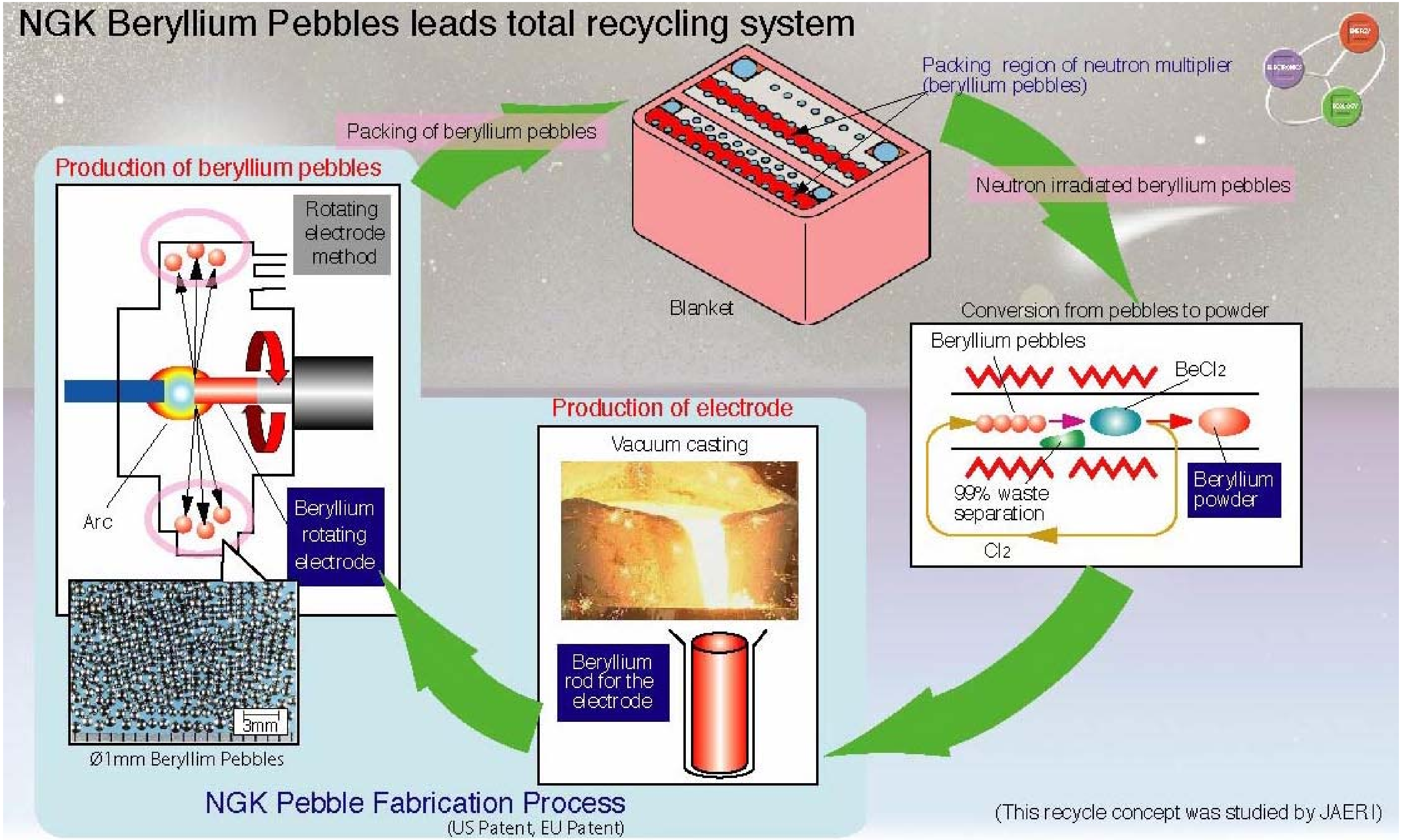
$$I = \int_{\Omega} \dot{m}(\vec{r})\ \tau(T(\vec{r}))dV$$

with

m(r) local tritium production [g s^{-1} m^{-3}]

τ (T) tritium residence time [s] as function of the breeder temperature T [K]

Fabrication cycle of neutron multiplier



Be pebbles: Chemical Composition

Be	99.5
BeO	0.23
Al	< 0.04
Fe	0.09
Mg	0.03
Si	>0.02
U	<0.01
Co	<0.001
Sc	< 0.005
Mn	0.01
Others	~ 0.065

Activation requirements:

- In reactor the amount of U should be lower as possible (<0.005 wt%) to reduce both fission products and long-lived actinide (e.g. 239Pu, 241Am).
- major contributors to the γ-dose rate are 56Mn, 24Na and 60Co (from 55Mn, 24Mg and 59Co respectively) at the short time

Mechanical requirements:

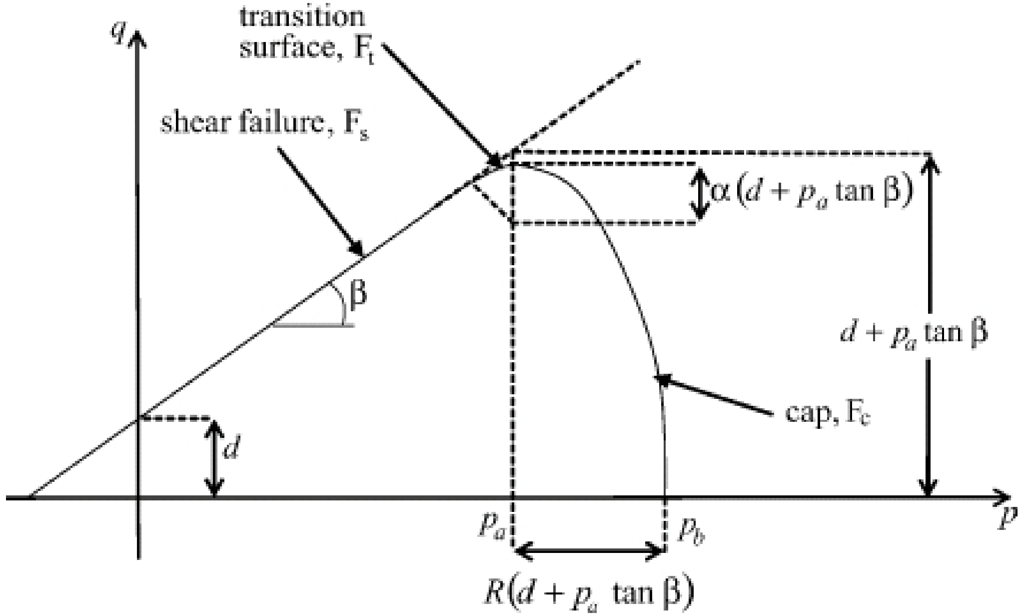
- To avoid low melting inter-metallic phases, it is required:

Al(wt %) / Fe(wt %) < 0.5 (atomic ratio Al/Fe < 1) Mg(wt%)/ Si(wt%) < 1.7 (atomic ratio Mg/Si < 2)

Be Irradiation behavior: T retention and Swelling

- The major design issue is connected with the tritium inventory in Be at high irradiation. Lacunae in the database and in the modelling give large uncertainties in the design calculation of the EOL Tritium inventory in Be in Demo/FPP conditions.
- The helium produced mainly by the reaction $^{9}Be(n,2n)2\ ^{4}He$ is the dominant cause of beryllium swelling and embrittlement, which are the major lifetime limiting factors for the material. The extimated swelling in DEMO condition is about 8%.
- An irradiation campaign to obtain data on Be at 3000 appm of Helium in 2 irradiation years and 6000 appm Helium in 4 irradiation years with temperatures in the range 500-700° C has been done in Petten within the framework of the European HIDOBE task. With these data, the modelling should be improved and, complementarily, an empirical extrapolation to DEMO conditions (18000 appm) could be attempted. Status: PIE phase ongoing under F4E contracts.

Thermomechanics of pebble beds



A non-linear elastic model and a modified Drucker-Prager-Cap theory

Temperature dependent parameters:

- thermal stresses
- Safety margin for the materials
- Tritium permeation
- Tritium release/inventory
- Swelling of materials

Thermal conductivity as function of the strain:

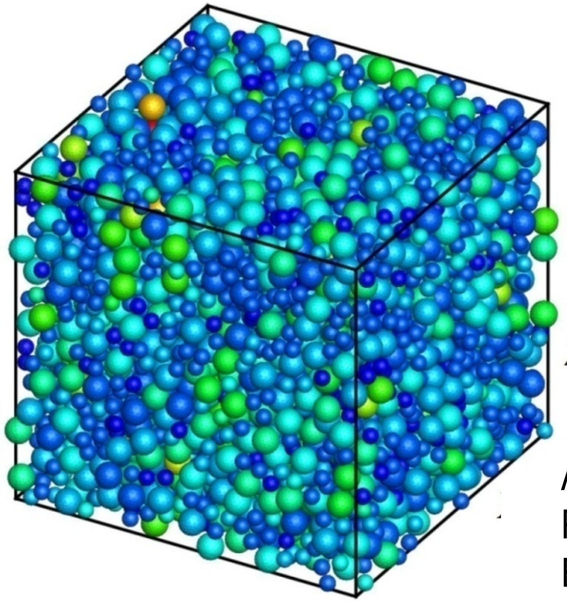
$$k\ (\mathrm{W/mK}) = 1.81 + 0.0012\ T(\circ\mathrm{C}) - 5 \times 10^{-7}\ T(\circ\mathrm{C})^2 + (9.03 - 1.386 \times 10^{-3}\ T(\circ\mathrm{C}) - 7.6 \times 10^{-6}\ T(\circ\mathrm{C})^2 + 2.1 \times 10^{-9}\ T(\circ\mathrm{C})^3)\ \varepsilon(\%)$$

Phenomena in pebble beds:

- Build-up of stress during start-up
- Creep at high temperature
- Gap formation
- De-mixing in binary beds

Discrete element Method (DEM)

Discrete Element Method based on the idea that discrete particles could be displaced independently from one another and interact with each other through contact forces.

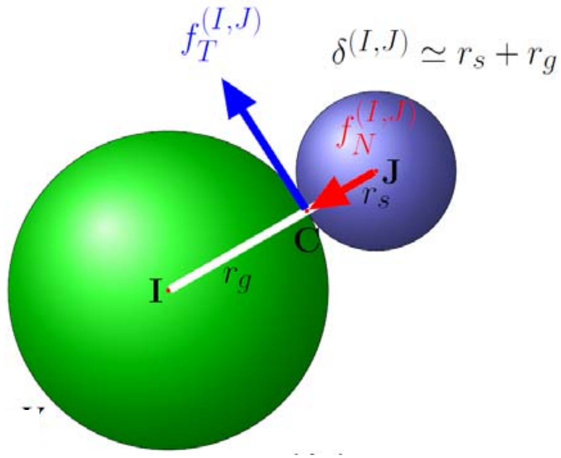


Pebble Assembly

Idealizations:
A unit cell of interest from a large assembly
Periodic boundary conditions

Assumptions:
Pebble shape: spherical
Elastic contact only

Pebble-Pebble Interactions



Macroscopic stress

$$\overline{\sigma}_{ij} = \frac{1}{V}\left(\sum_{I<J} \delta^{(I,J)} f_N^{(I,J)} n_i n_j + \sum_{I<J} \delta^{(I,J)} f_T^{(I,J)} n_i t_j\right)$$

Average normal force and hydrostatic pressure

$$f_{\mathrm{ave}} = 2\frac{\sum_{I<J} f_N^{(I,J)}}{n_c N}, \quad p = \frac{\overline{\sigma}_{ii}}{3} = \frac{1}{3V}\left(\sum_{I<J} \delta^{(I,J)} f_N^{(I,J)}\right)$$

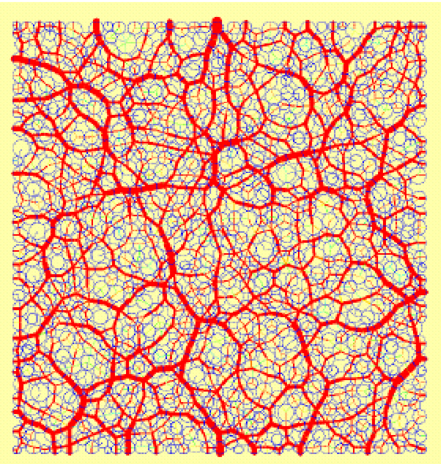
Y. Gan et al, JMPS 2010

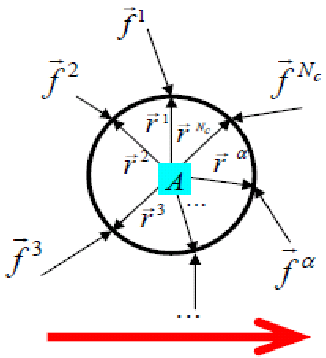
Discrete element Method (DEM)

Visualization of DEM simulation results

Deformation, as well as the internal force distribution of the pebble bed assembly can be obtained numerically

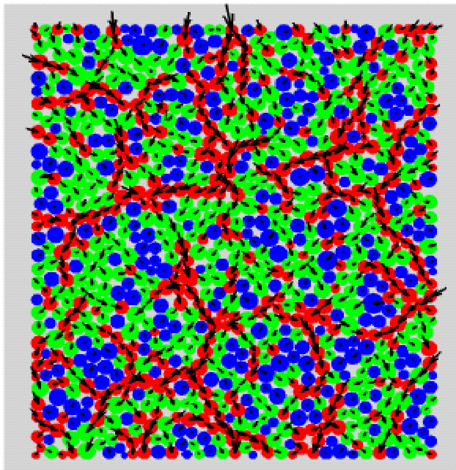
- **Force Map**

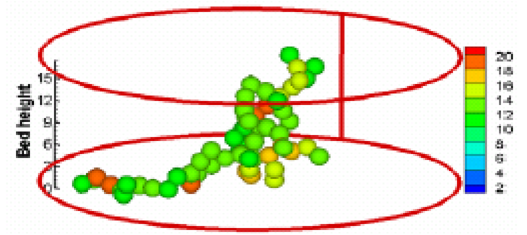


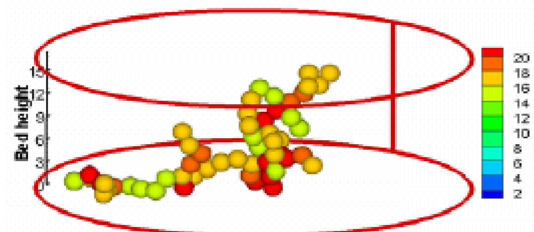


$$\overline{\sigma}_{ij} = \frac{1}{V}\sum_{\alpha=1}^{N_C} f_i^{\alpha} r_j^{\alpha}$$

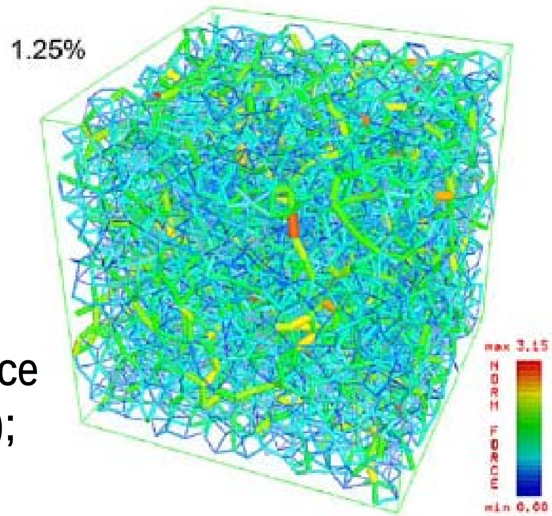
- **Stress Map**

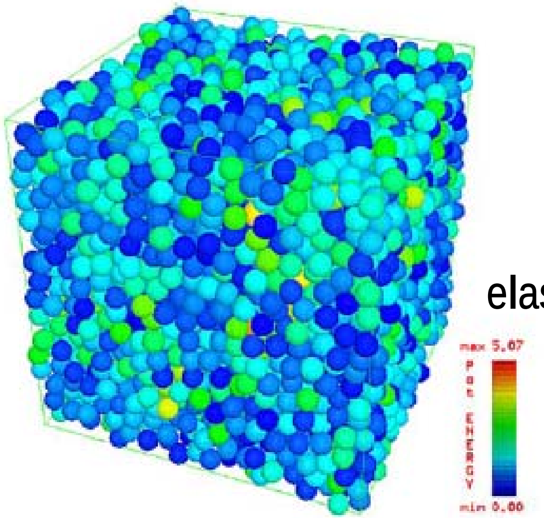






(J. An et al. 2006)





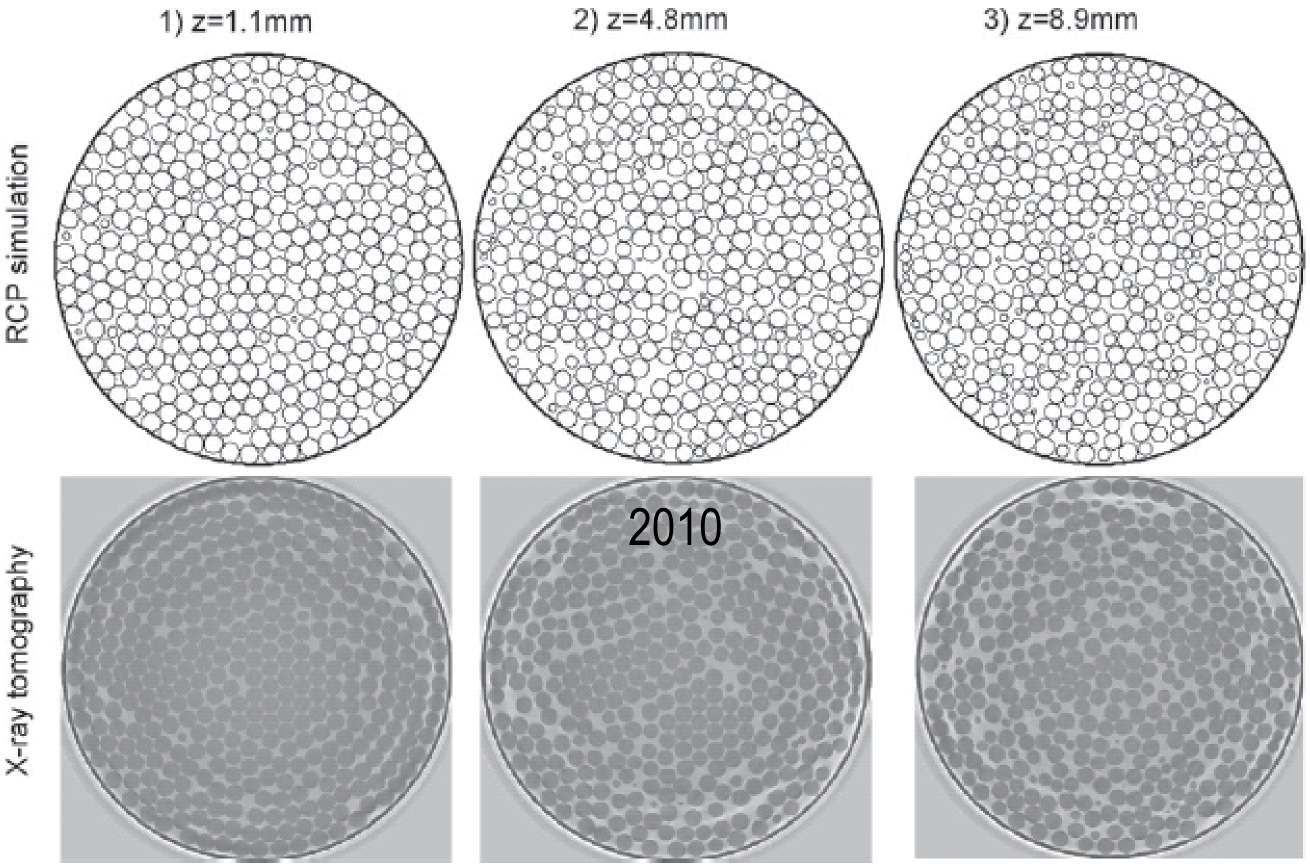
inter-particle force chains (unit: N);

elastic potential energy (unit: 10^6J) .

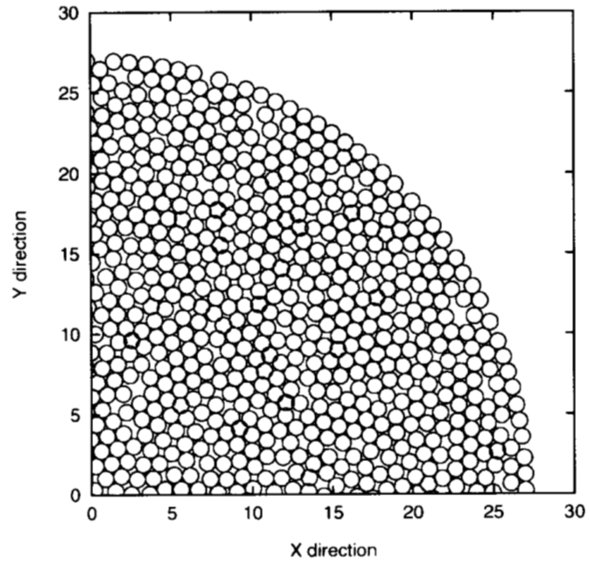
(Y. Gan et al. 2010)

DEM and X-ray Tomography

Computer simulated packing structures in pebble beds show the DEM approach is promising in studying Pebble bed thermomechanics

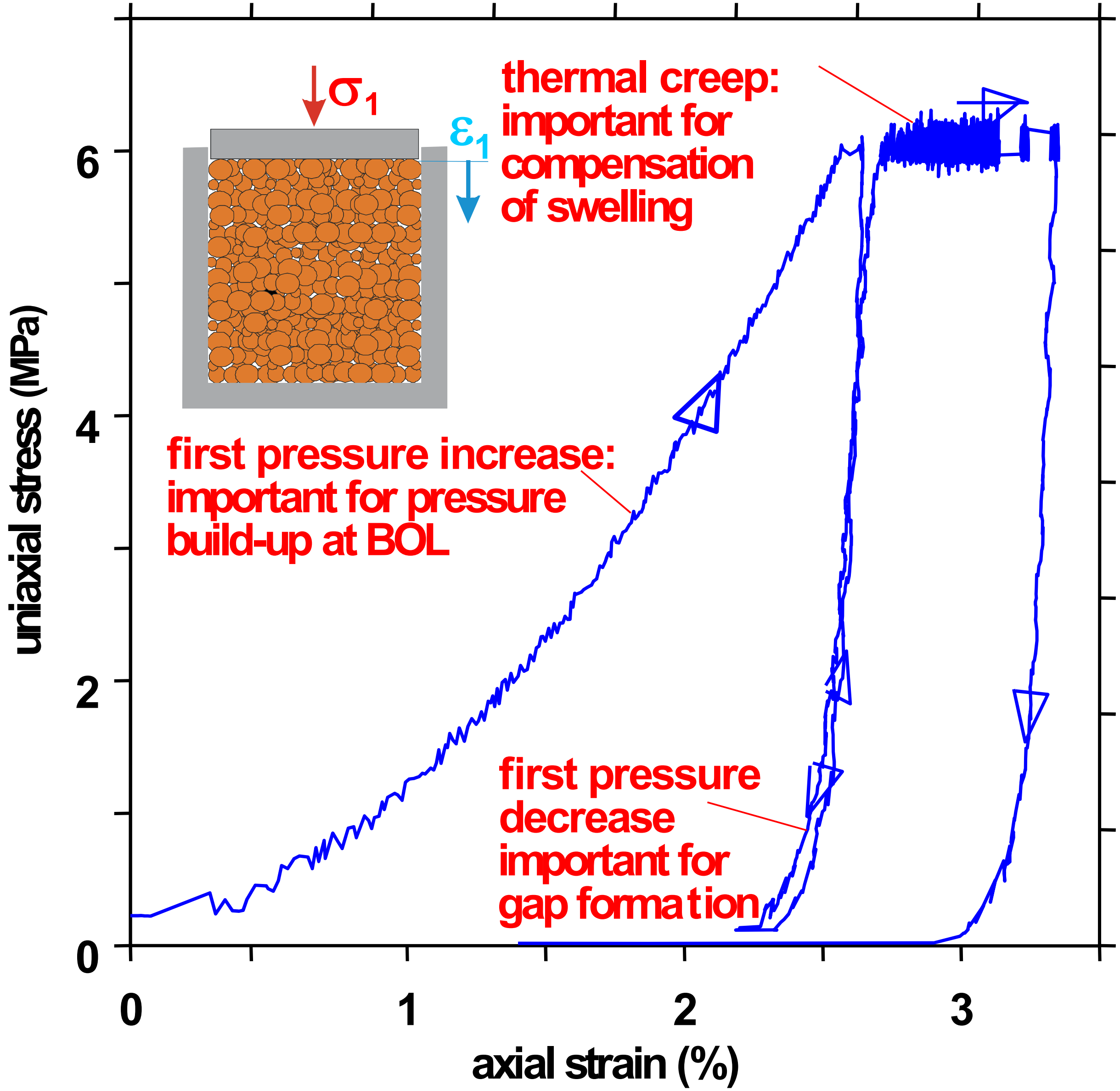


Comparison with the experiment: The first row is the plot from simulation, and the second row is from the X-ray tomography experiment. Container dia = 48.9 mm, height~ 50 mm. Sphere particle: 2.3 or 5 mm *(Y. Gan et al., 2010)*



Experimentally observed orthorhombic packing obtained numerically (particle dia: 1 mm, total: 26,010 particles)
(J. An et al., 2004)

Characteristic UCT result at elevated temperature



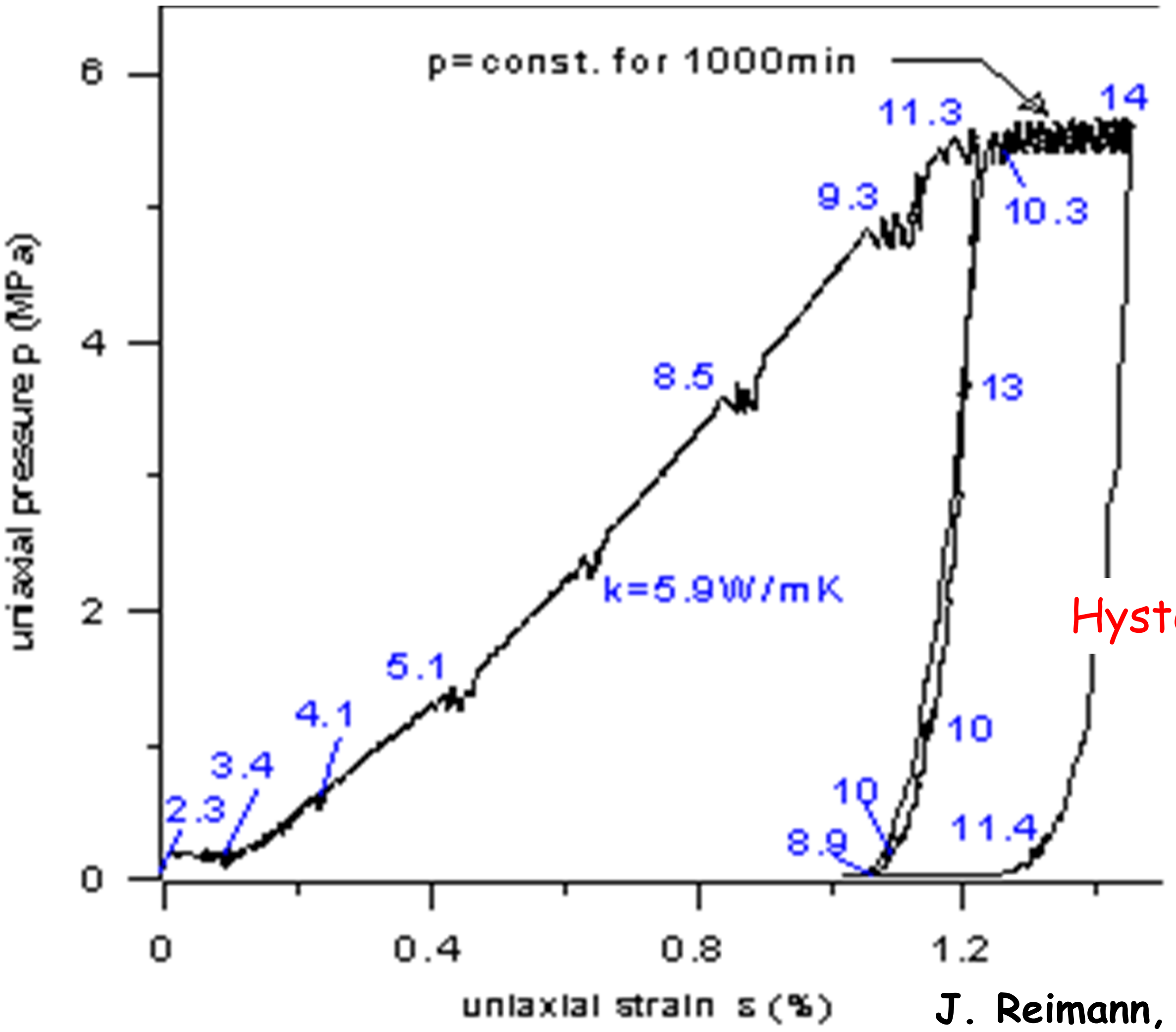
Uni-axial-compression tests (UCT):

- **First pressure increase:** Mechanisms: irreversible displacements of particles and elastic/plastic particle deformation. This curve determines the stress build-up during the first blanket operation.
- **First pressure decrease**, second pressure increase: Mechanisms: elastic and plastic deformations.These curves and the curve"1st pi" determine the formation of gaps during blanket operation.
- **Thermal creep:** important for stress relaxation during the first days of blanket operation and for compensation of irradiation induced swelling.

J. Reimann, KIT

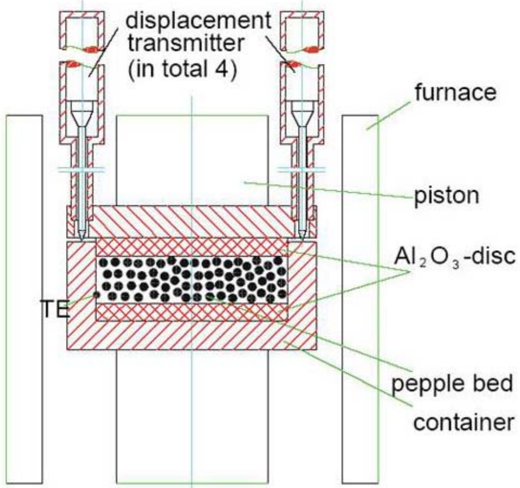
Effective Thermal conductivity in Be pebble beds

Effective thermal conductivity of 1 mm Be pebble bed (475 °C) depends on the compressive strain



J. Reimann, KIT

Hysteresis effect



HECOP

point contact

compression

area contact

(increase in k)

stress or swelling

Blanket manufacturing

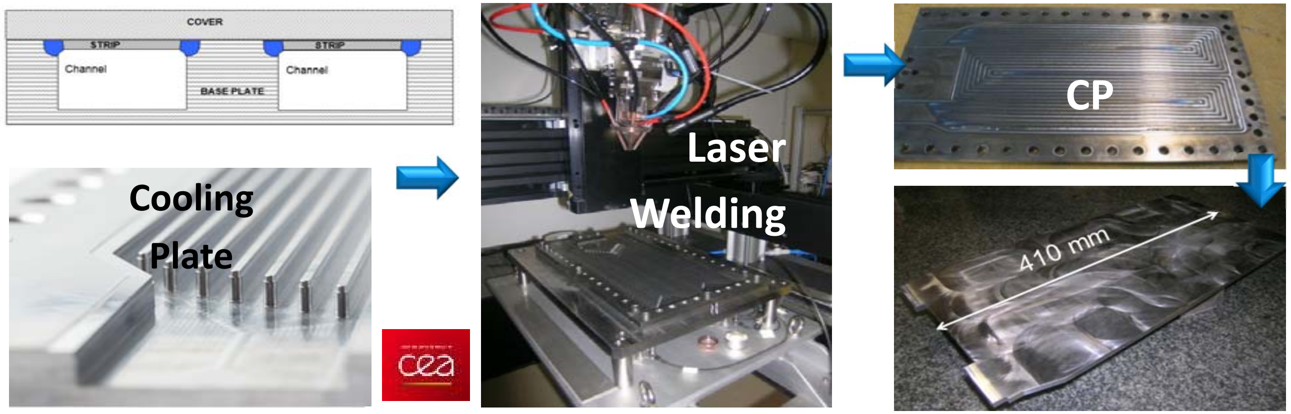
In EURO*fusion* programme:

- Assessment of Blanket Segment manufacturing (FW, Breeder zone, small/large boxes, manifold systems, attachments to VV).
- Technology development with production of fabrication mock-ups (e.g. Double Wall Tubes for WCLL, W coating in FW)
- Test of mock-ups in helium and water facilities.

A fail-safe and cost effective fabrication route for blanket First Walls; Commin, L.; Rieth, M.; Dafferner, B.; Zimmermann, H.; Bolich, D.; Baumgärtner, S.; Ziegler, R.; Dichiser, S.; Fabry, T.; Fischer, S.; Hildebrand, W.; Palussek, O.; Ritz, H.; Sponda, A.; Journal of Nuclear Materials, Volume 442, Issue 1, p. 538-541

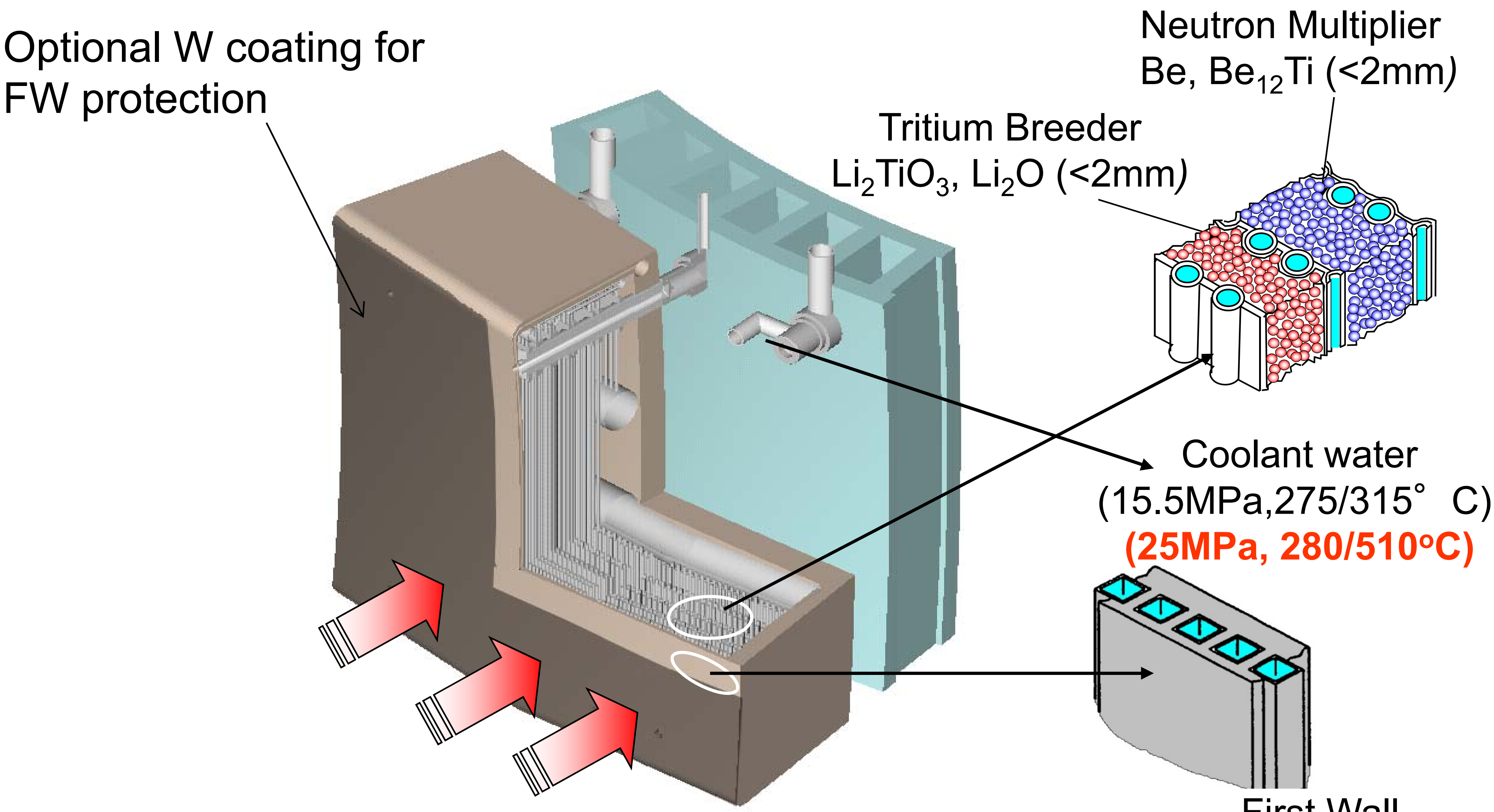
In TBM Programme:

- Qualification of key technologies (like plates with channels and welding assembly) suitable for integrated FW, breeder Zone and small boxes.



Laser welding channels recostruction + final HIP

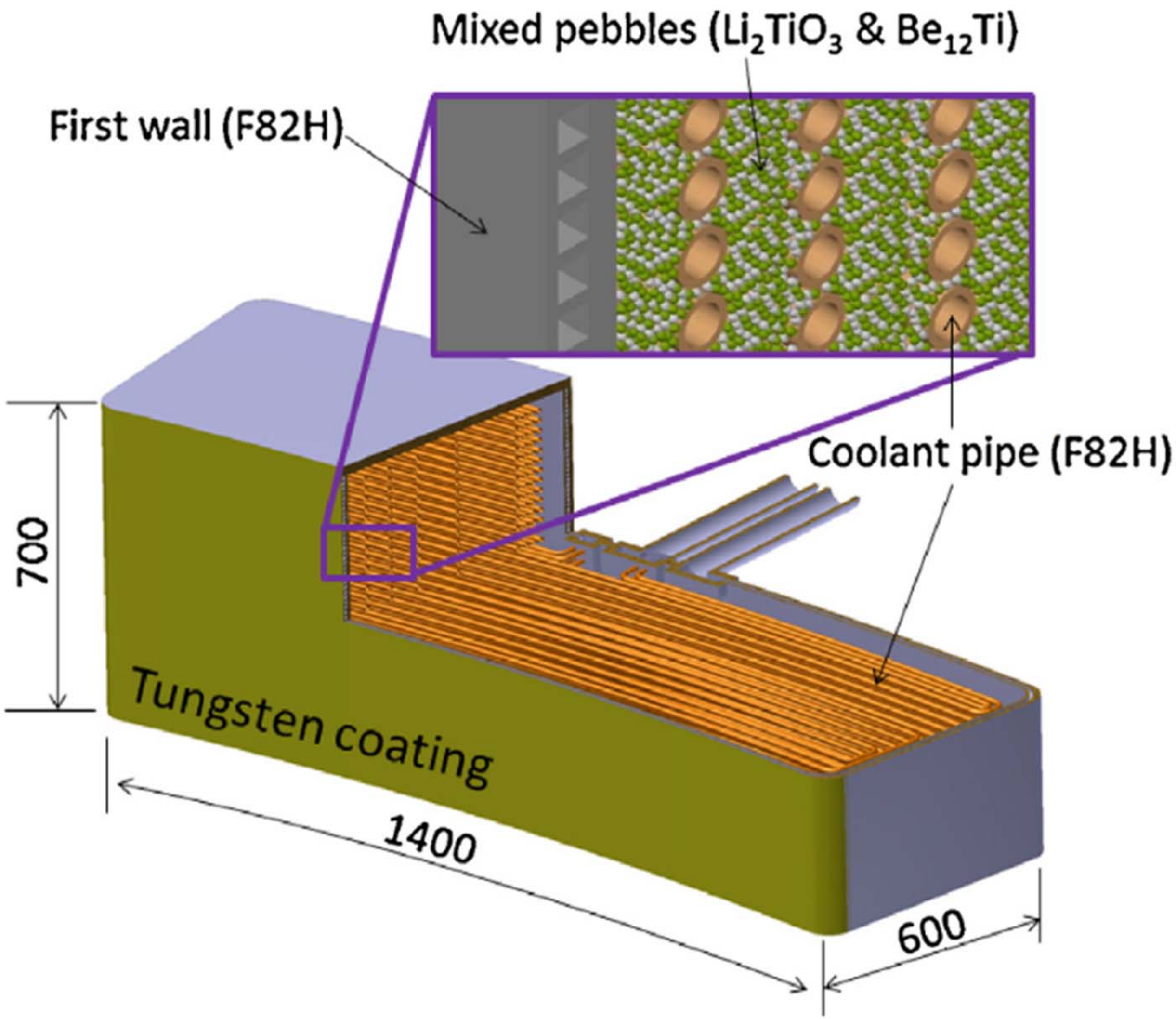
JA Water-Cooled Solid Breeder Blanket



Surface Heat Flux: 1 MW/m^2

Neutron Wall Load: 5 MW/m^2(1.5×10^{15}n/cm^{2}s)

JA WCSB Blanket (JA DEMO 2015)



Mixed beds:

- **Use of $Be_{12}Ti$ as neutron multiplier**
- **Use of Li_2TiO_3 as breeder.**

✓ **More safety margin for Be-steam reaction**
✓ **Increase of mechanical strength at high temperatures**
✓ **Chemical compatibility between breeder and multiplier**
✓ **Better T release from Be**

❖ **Reduction of neutronics performances.**

Y. Someya and al., "Design study of blanket structure based on a water-cooled solid breeder for DEMO", Fusion Eng. Des. 98-99 (2015) 1872–1875.

Liquid Breeder Blanket Concept

The breeder is used in liquid form. Potentially the breeder can be used as coolant.

Type of Liquid Breeder:

a) **Liquid Metal**: Li, $PbLi_{eu}$(15.8 at%).

High conductivity, low Pr number, melting point: ~235°C for $PbLi_{eu}$

Dominant issues: MHD, chemical reactivity for Li (corrosion and water reaction), tritium permeation issues for LiPb concepts, T gettering for Li, corrosion with steel at >450°C.

b) **Molten Salt:** Flibe $(LiF)_n \cdot (BeF_2)$, Flinabe (LiF-BeF_2-NaF)

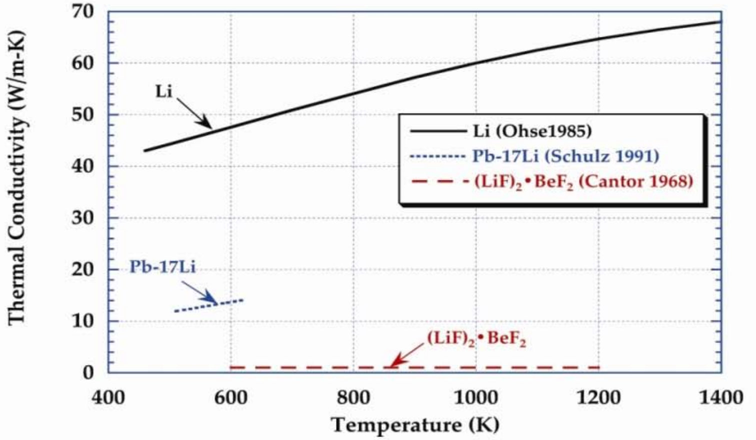
Low conductivity, high Pr number, melting point Li_2BeF_4 : ~459°C,
not flammable and does not react with air or water

Dominant Issues: Higher melting point, chemistry, tritium control, corrosion.

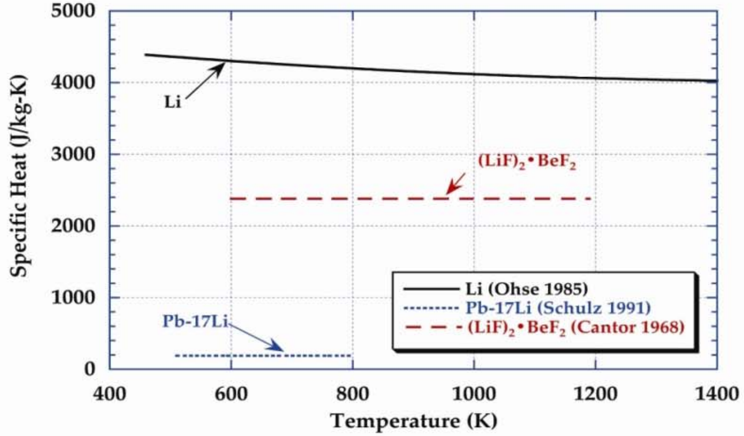
Liquid Breeder Blanket Concept

Summary of some physical property data for candidate molten salts

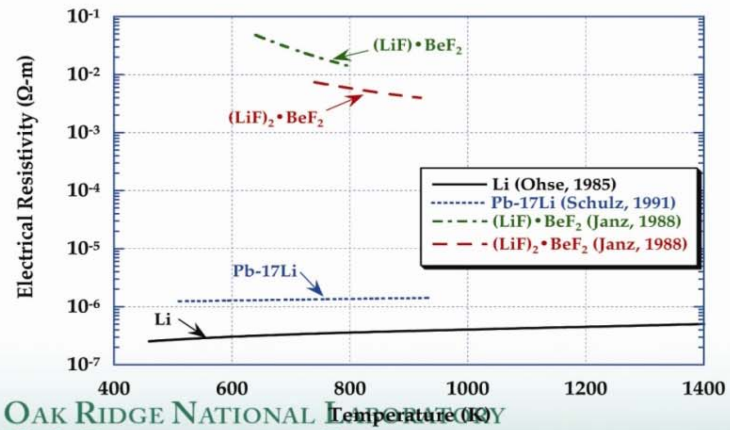
COMPARISON OF THE THERMAL CONDUCTIVITY OF LIQUID COOLANTS



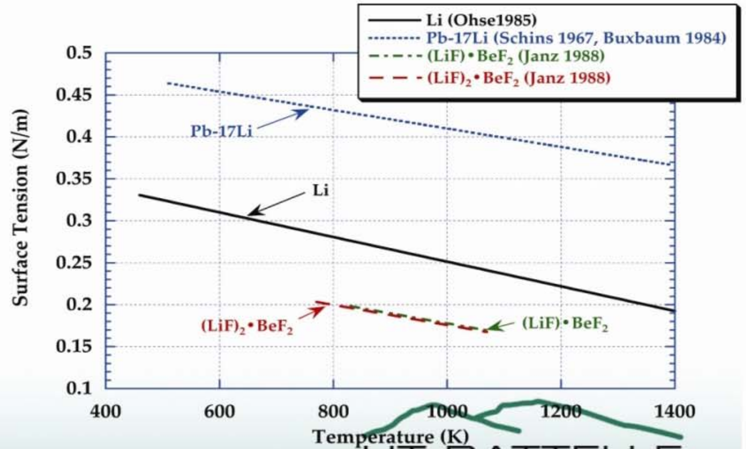
COMPARISON OF THE SPECIFIC HEAT OF LIQUID COOLANTS



COMPARISON OF THE ELECTRICAL RESISTIVITY OF LIQUID COOLANTS



COMPARISON OF THE SURFACE TENSION OF LIQUID COOLANTS

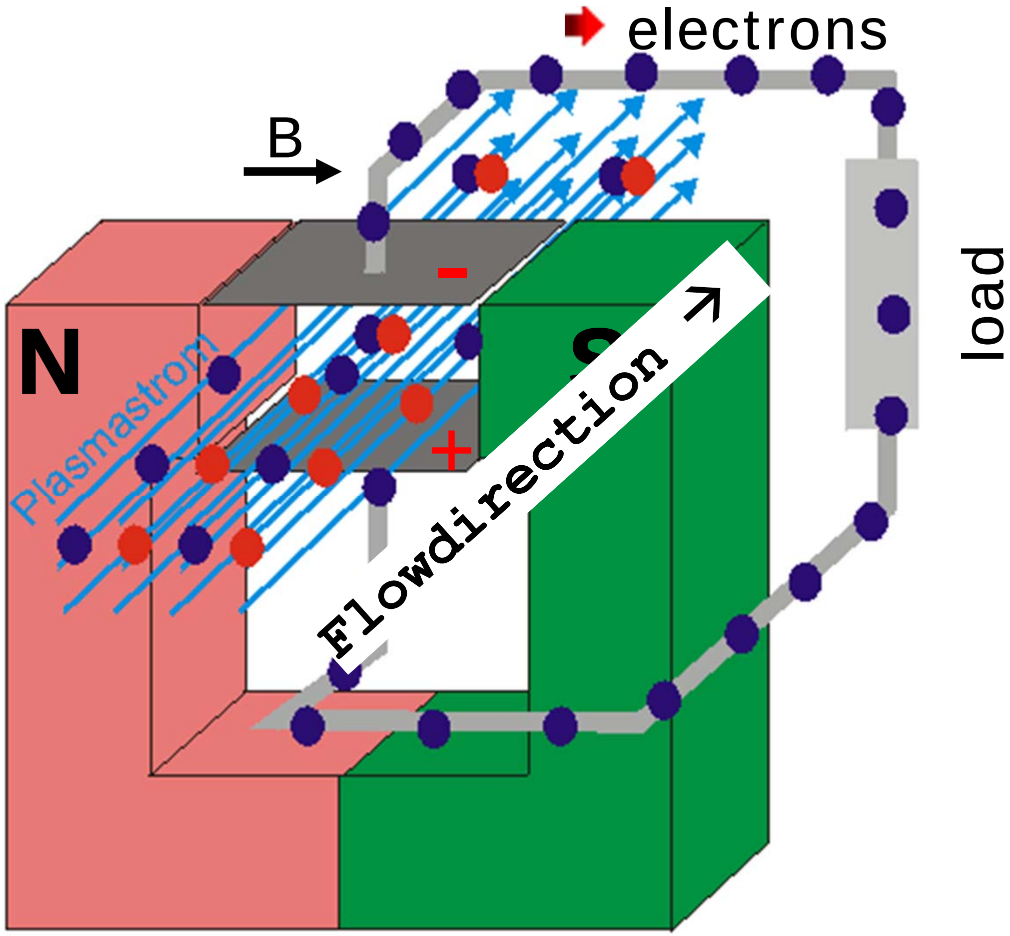


OAK RIDGE NATIONAL LABORATORY
U. S. DEPARTMENT OF ENERGY

UT-BATTELLE

Liquid metal flow in a magnetic field [1/3]

MHD=Magnetohydrodynamics

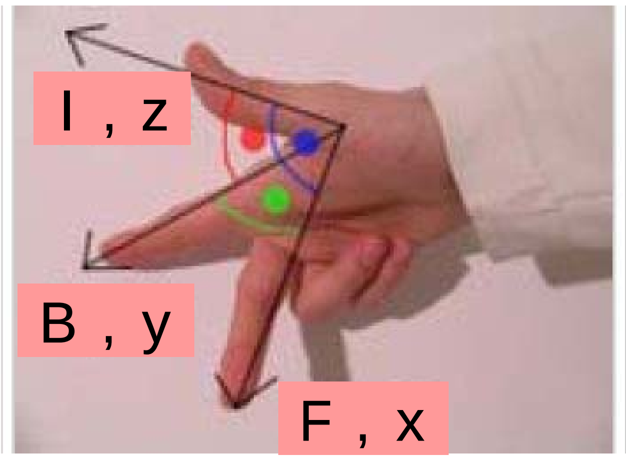


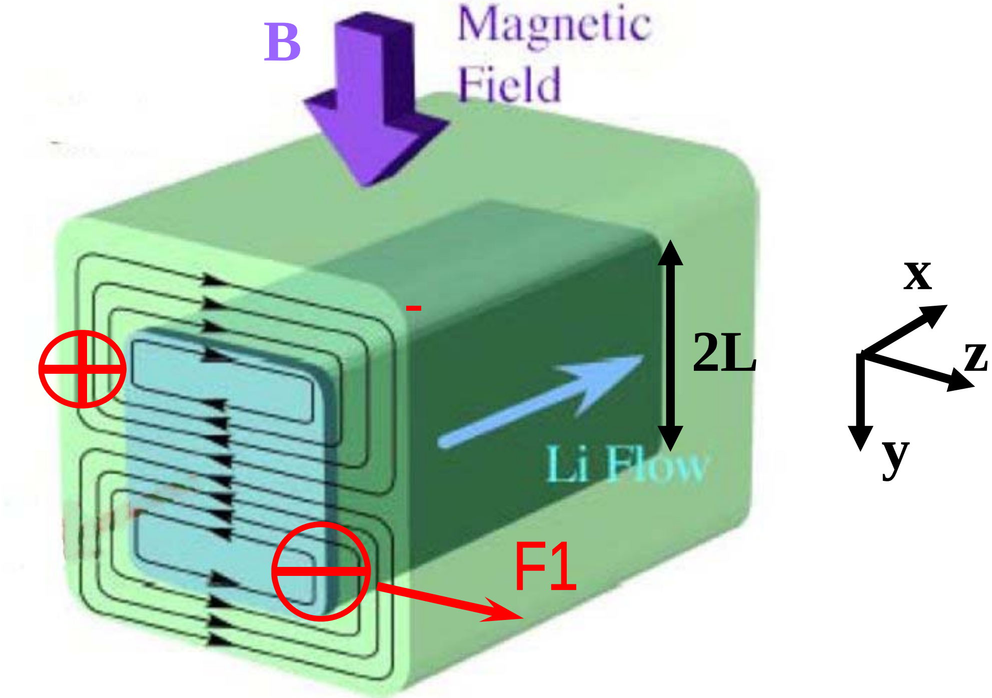
- charges are separated in the magnetic field, effect can be used for MHD generators

Liquid metal flow in a magnetic field [2/3]

- Liquid metal flows orthogonal to the magnetic field B
- = Charges are moved in magnetic field → Force F1 on electrons

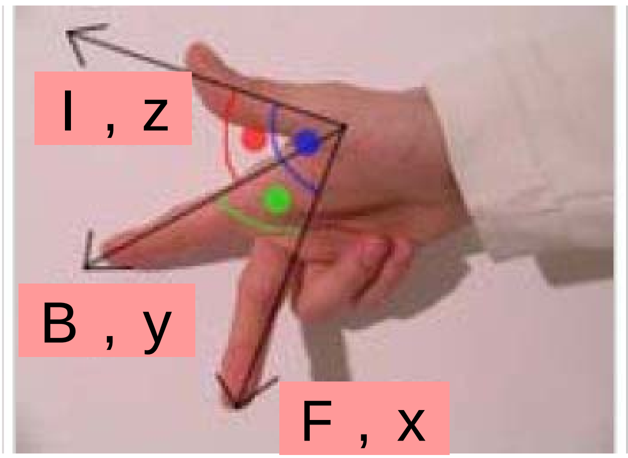
→ separation of charges results in voltage in z-direction
→ eddy currents close either in the fluid (insulated walls)
or over the walls (conducting walls)

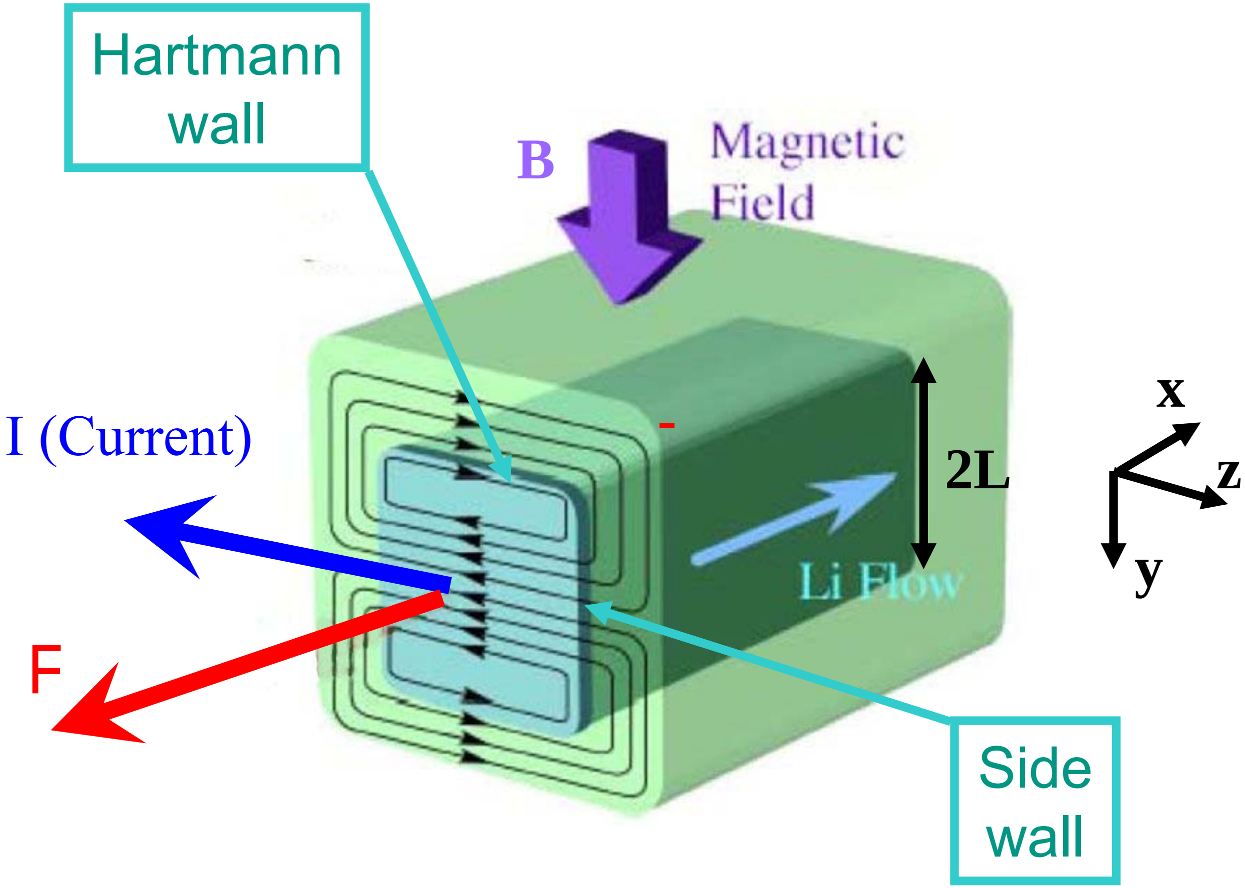




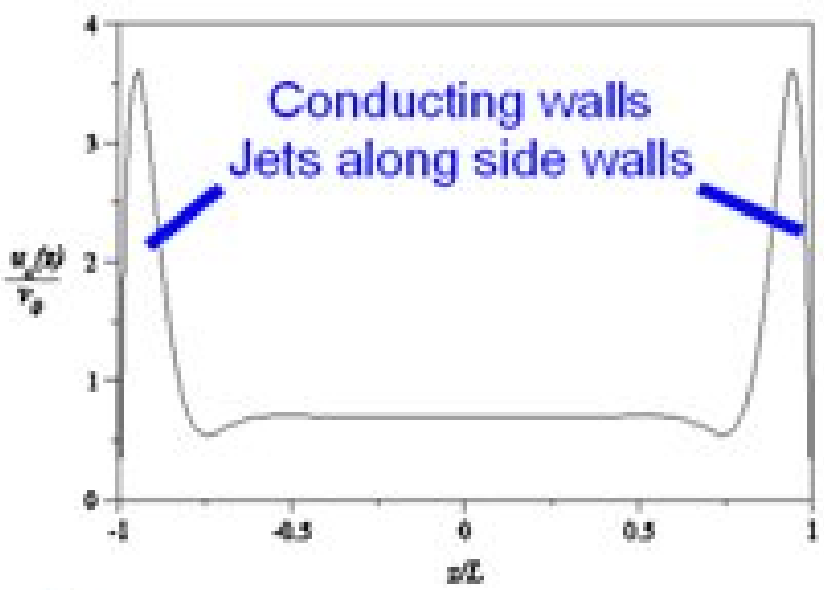
Liquid metal flow in a magnetic field [3/3]

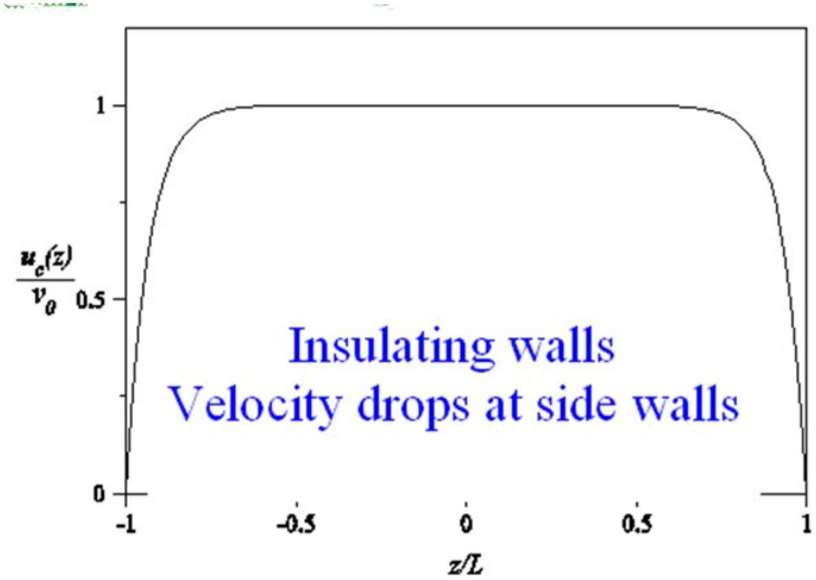
- In the fluid bulk the current flows orthogonal to the magnetic field B
- Resultant forces are opposite to the Liquid metal flow

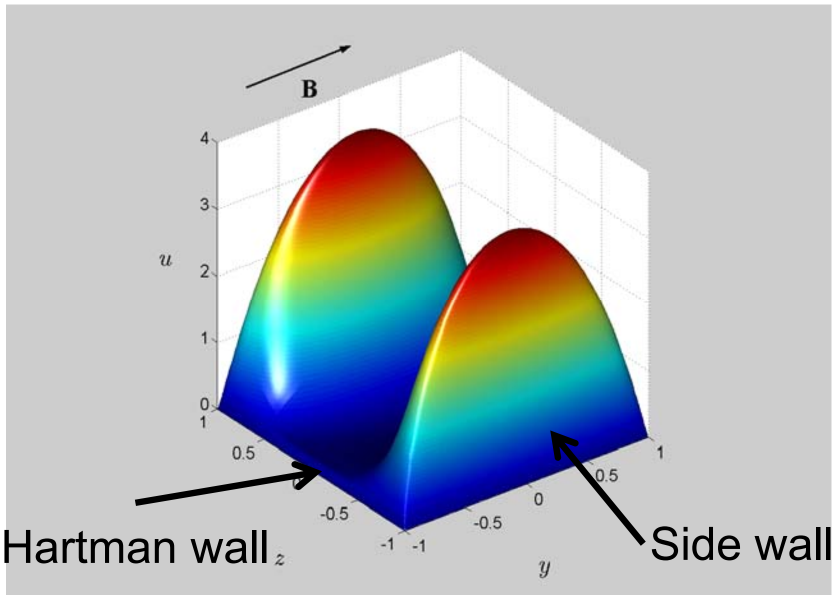


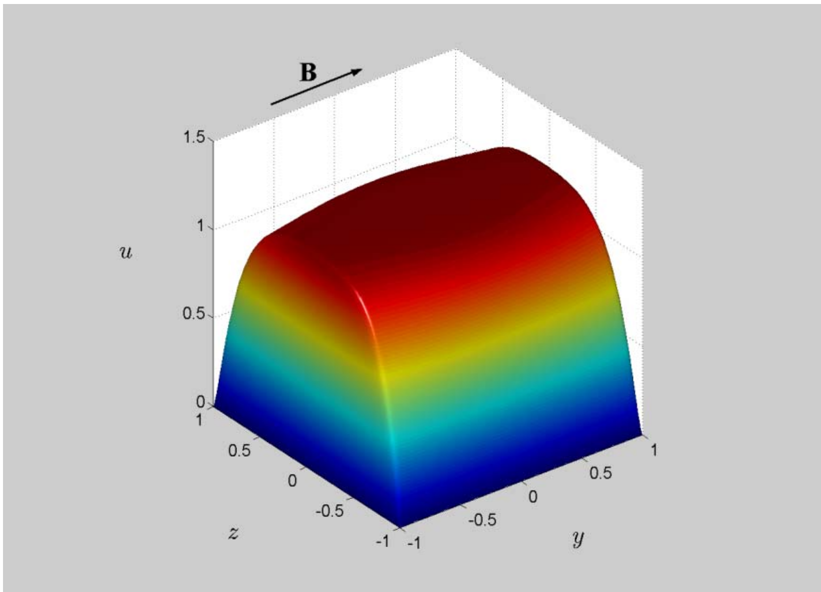


Flow with conducting and insulating walls

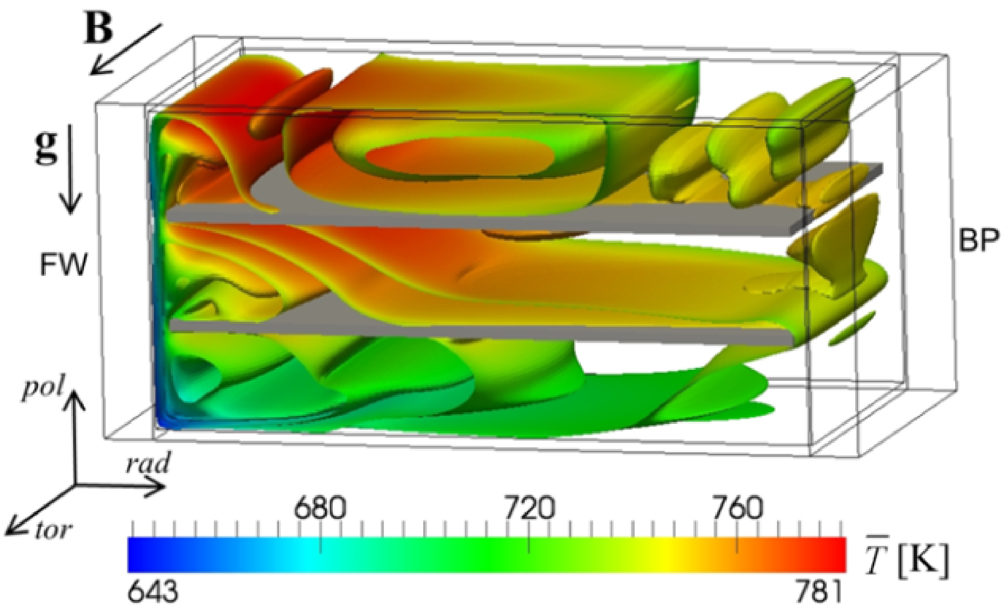


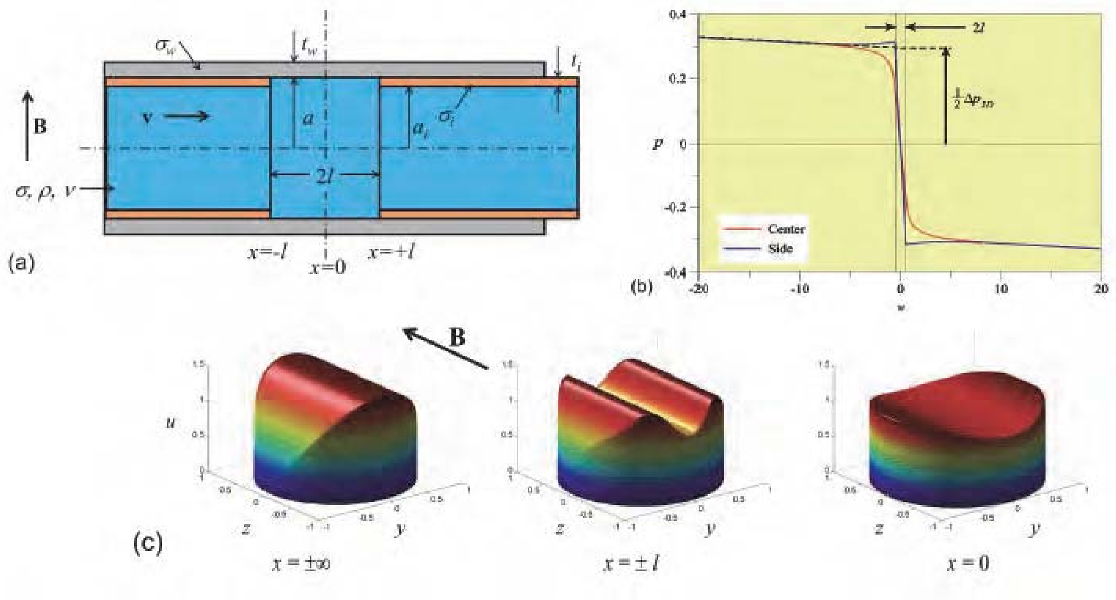






MHD effects in liquid blankets



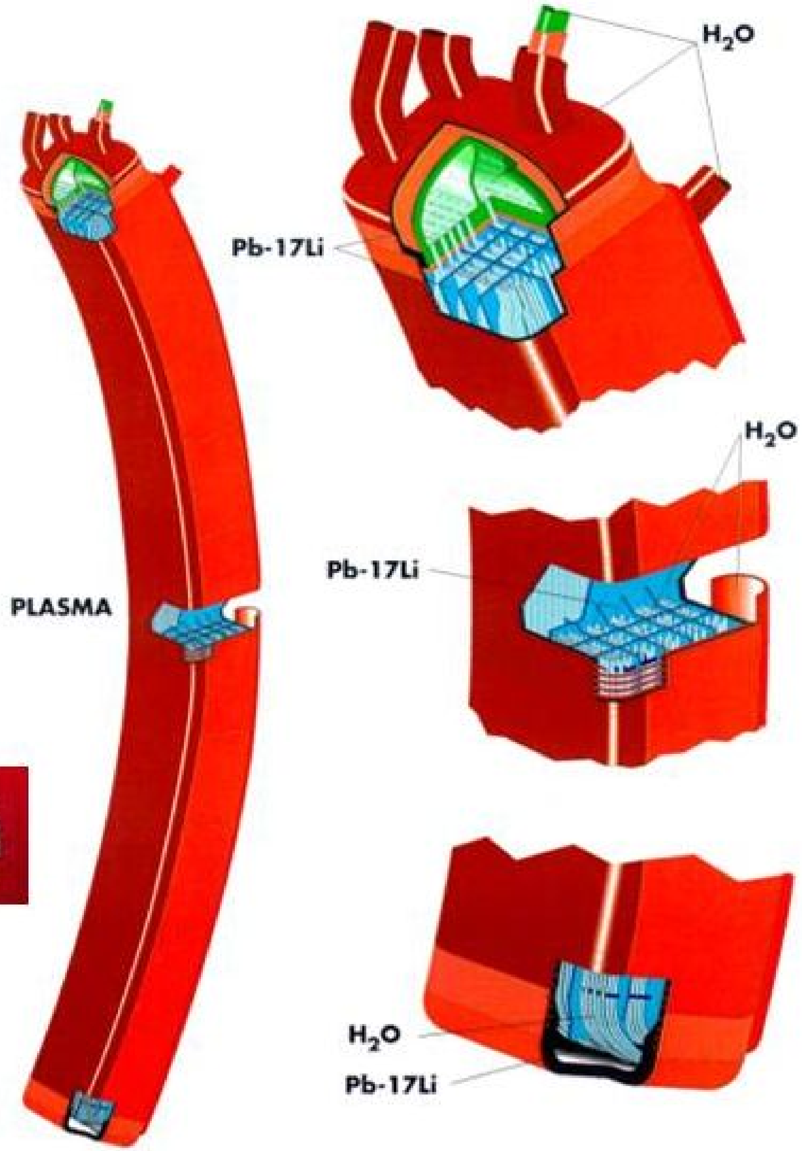


Iso-surfaces of time averaged electric potential, colored by averaged temperature, for flow at Ha = 2000 and Gr = 1.2 108 in a cavity. Relevant for HCLL blankets.

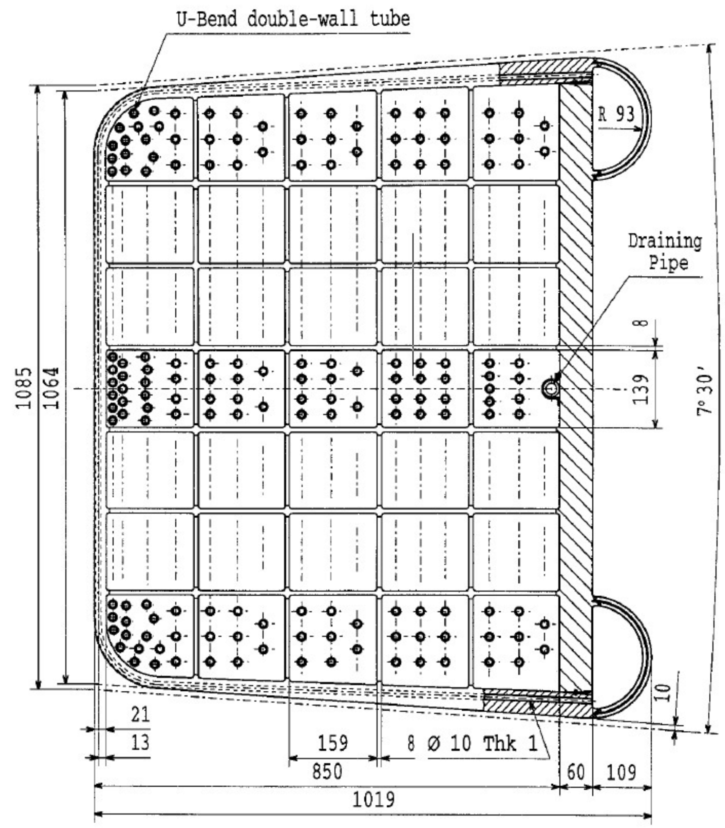
C. Mistrangelo, L. Bühler and G. Aiello [Mist14]

Impact of gaps in electrical insulation at the junction of different Flow Channel Insert (FCI) pieces. Gap between two FCIs (a) geometry; (b) dimensionless pressure distribution along the pipe axis y=0 and along the side wall y=±1; (c) typical velocity profiles in fully developed regions (x±∞), at the beginning and end of the gap (x=±l) and in the middle of the gap (x=0). In DCLL.

WCLL Blanket Concept (CEA, 2000)

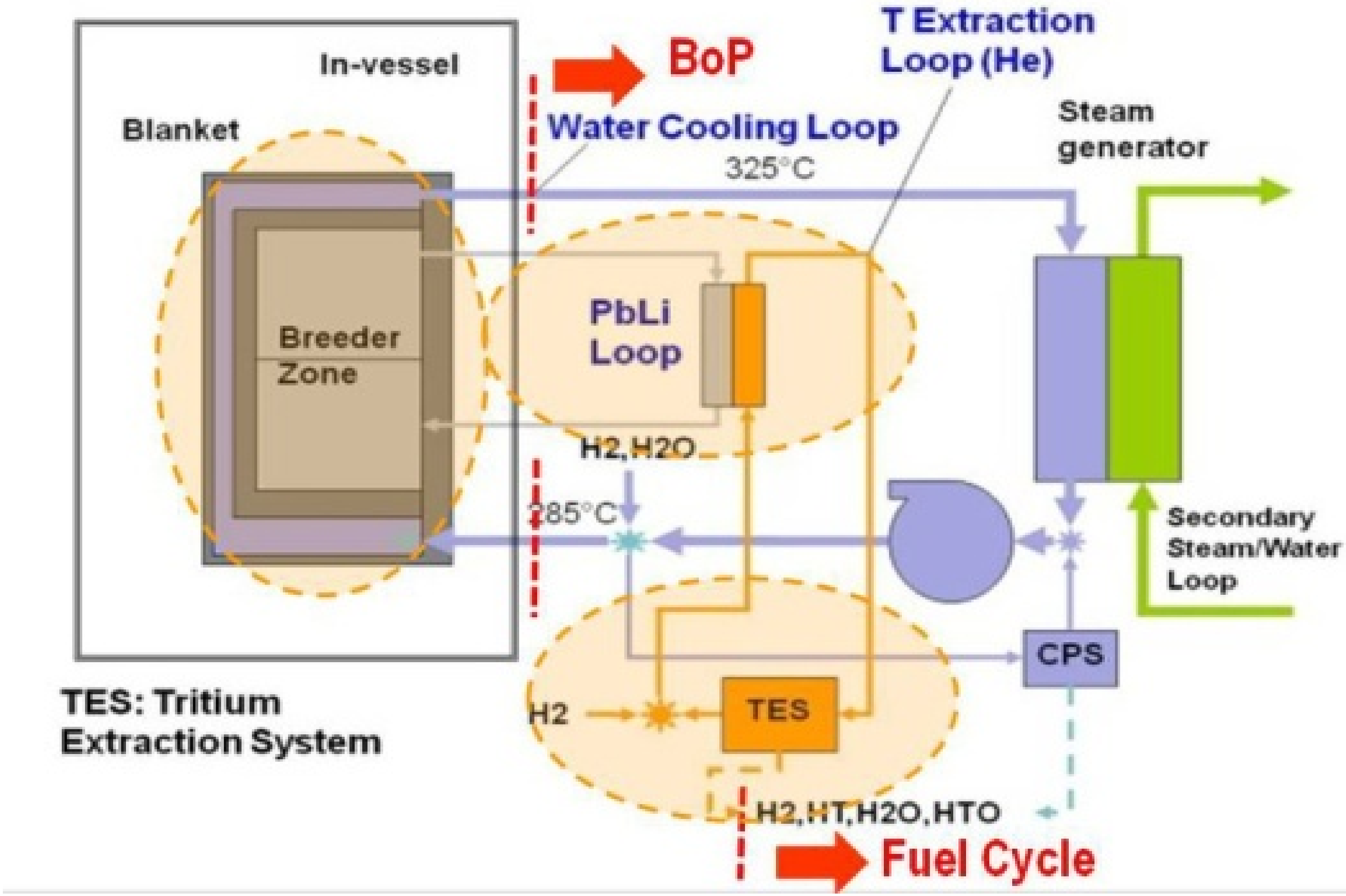


- PbLi as breeder. Quasi stagnant (10 inventories changed per day), only as T carrier.
- Water as coolant (285-325°C @ 15.5 MPa)

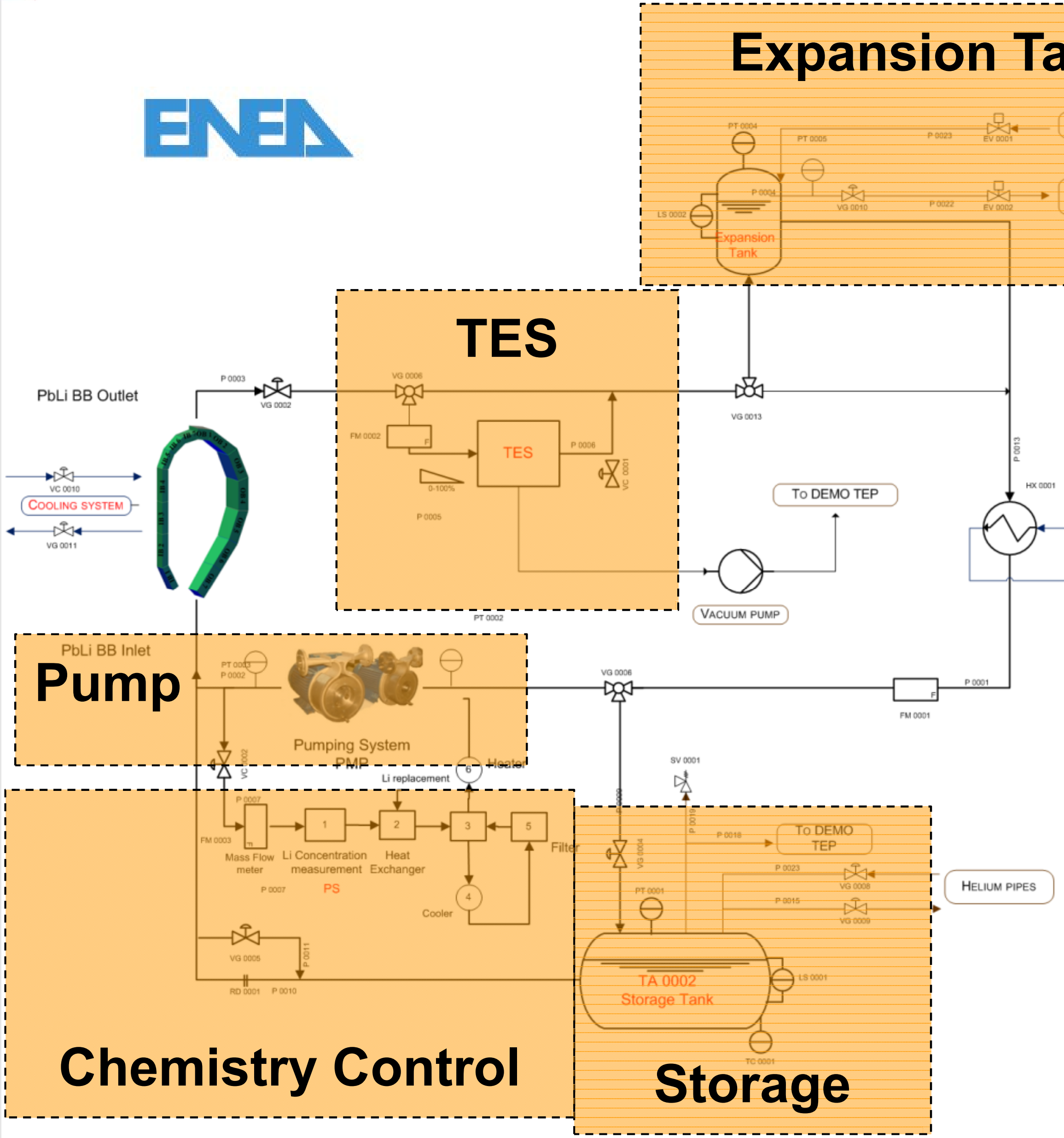


P. Sardain at al., "Power plant conceptual study - WCLL concept", Fusion Engineering and Design 69 (2003) 769-774

WCLL Blanket Concept (CEA, 2000)



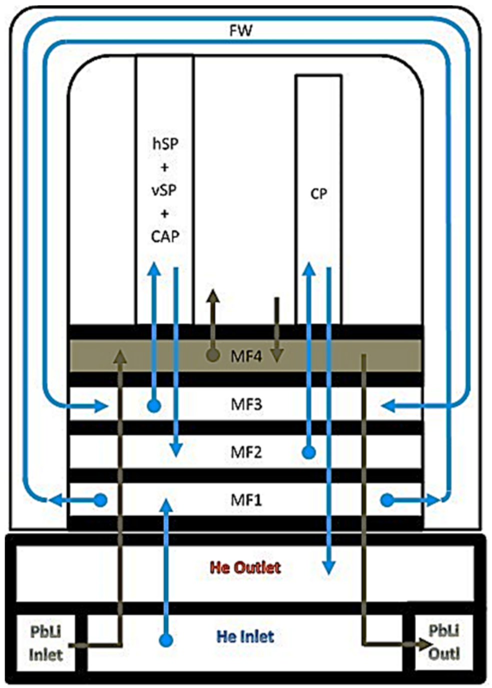
WCLL: PbLi loops



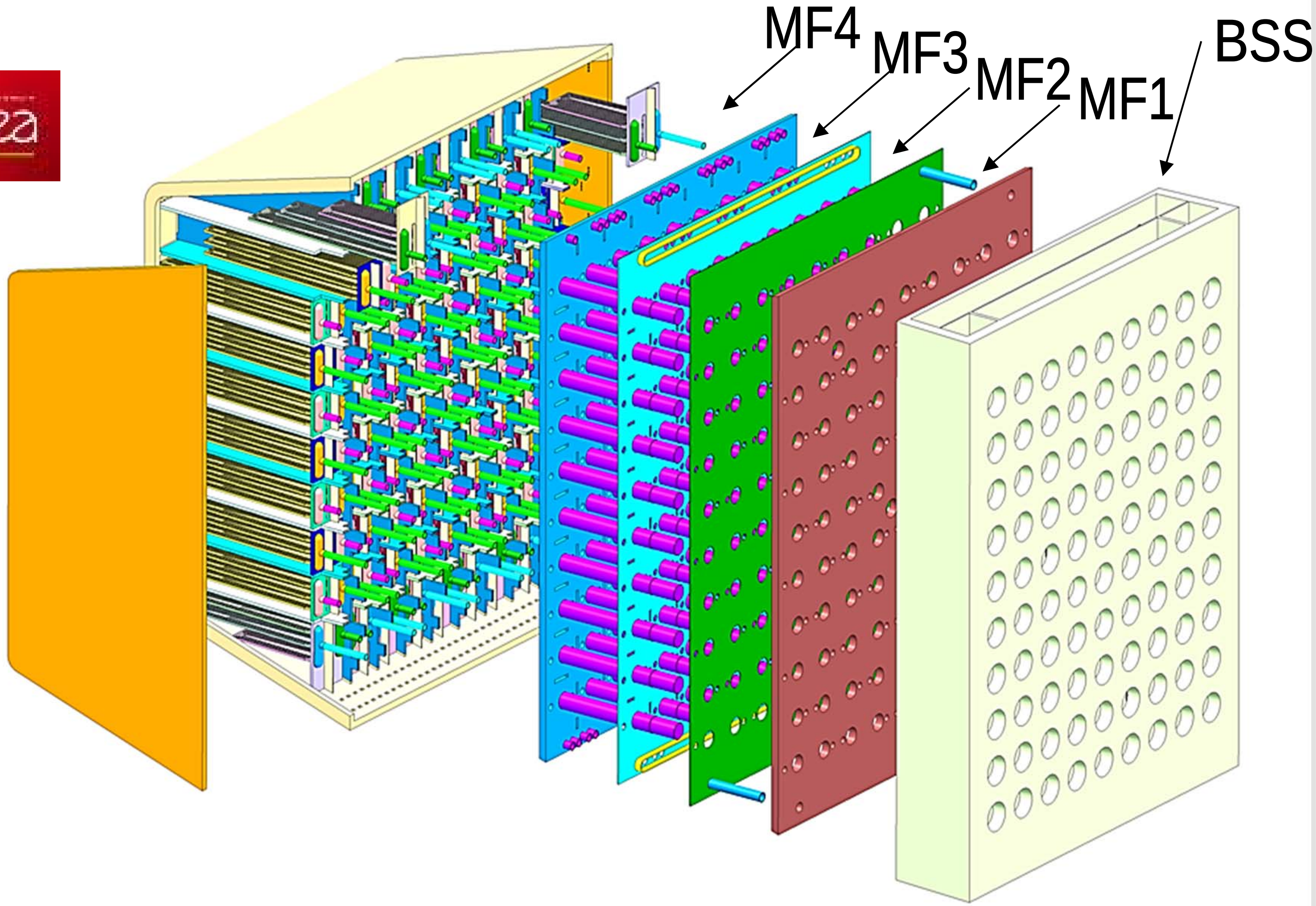
- Eight Pumping System => 70kg/s per loop
  - ✓ 4 loops <=Inboard Module (4 Segments)
  - ✓ 4 loops <= Outboard Module (4 Segments)

PPPT Demo Lay-out	WCLL		
PbLi volume [m3]	5,21E+02		
PbLi weight [kg]	5,12E+06		
Recirculation/day	8	10	15
PbLi mass flow [kg/s]	448	560	804
PbLi mass flow segment [kg/s]	28	35	53
ΔT PbLi(out-in) max	211		
Pth [MW]	15	22	33

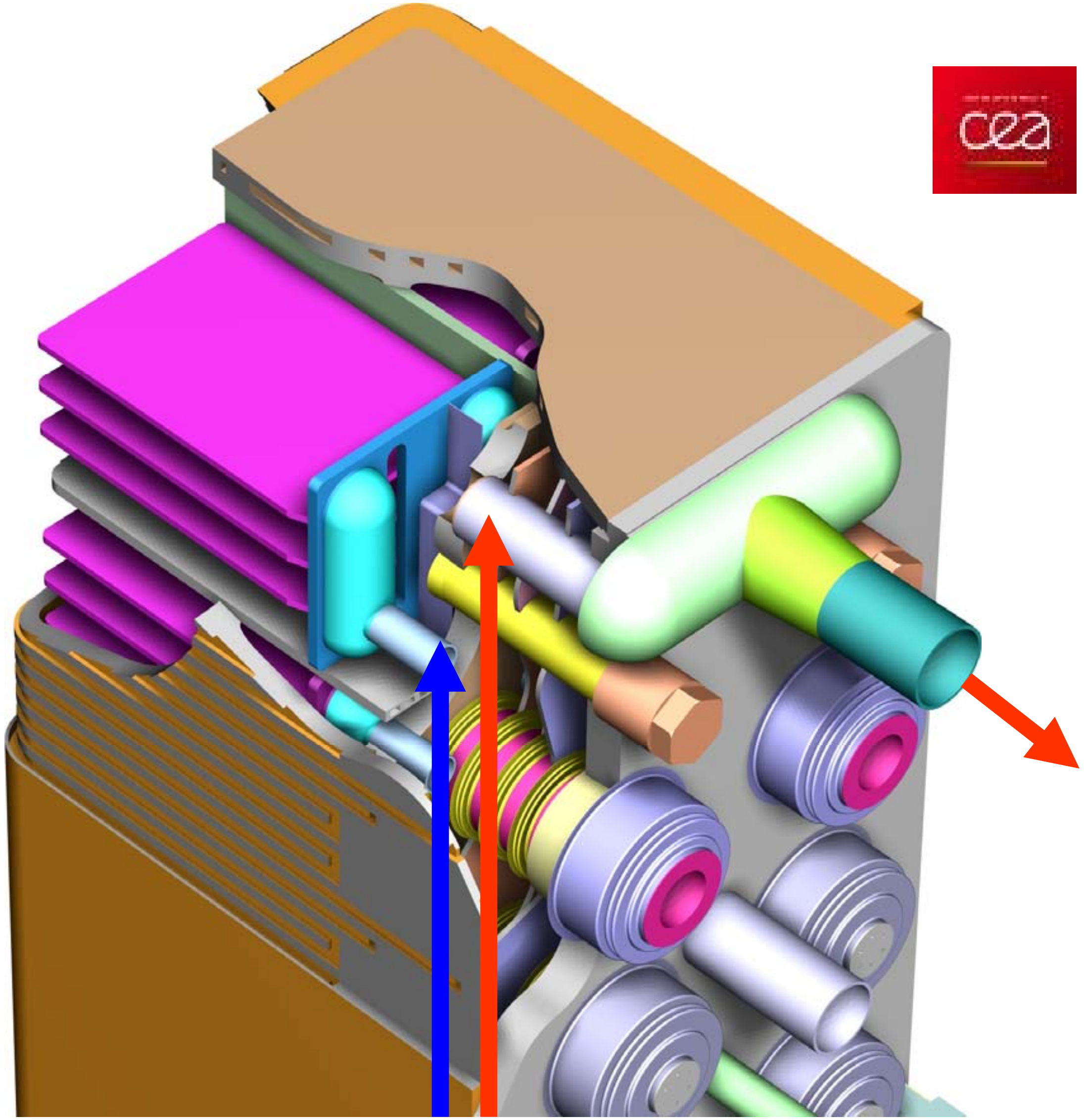
HCLL Blanket Concept (CEA, 2015)

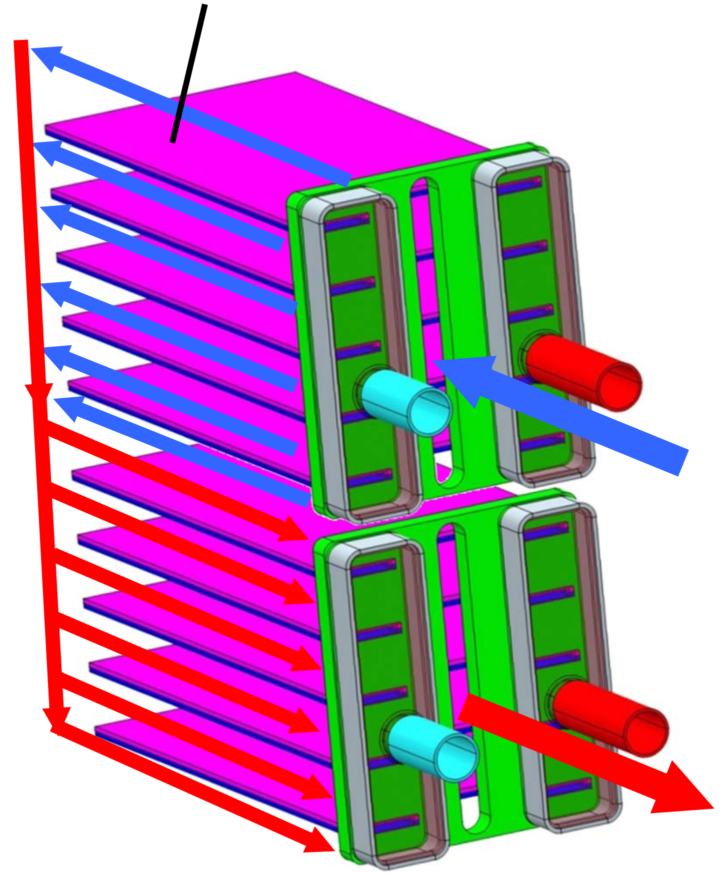


HCLL Flow Scheme



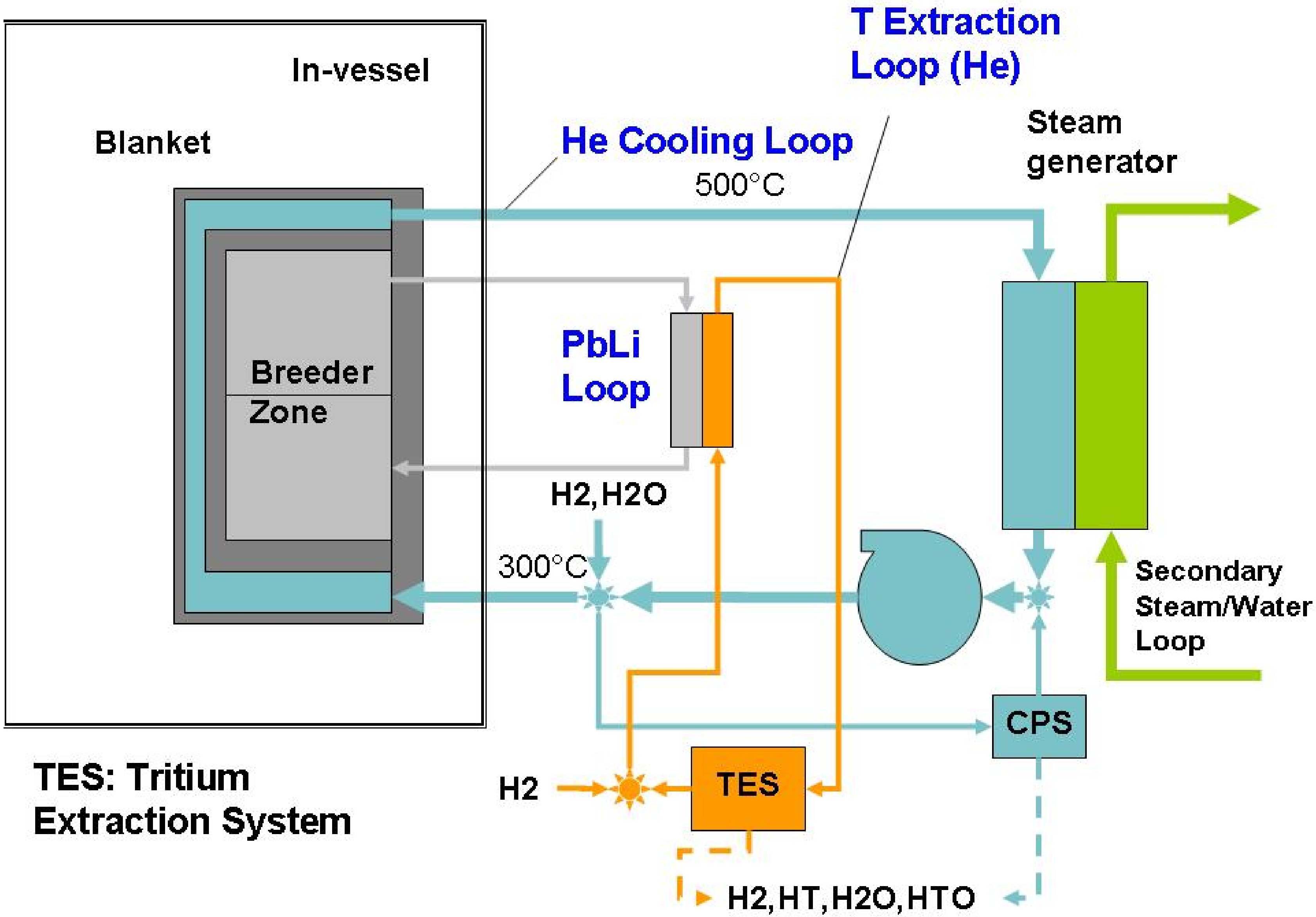
HCLL: PbLi Distribution





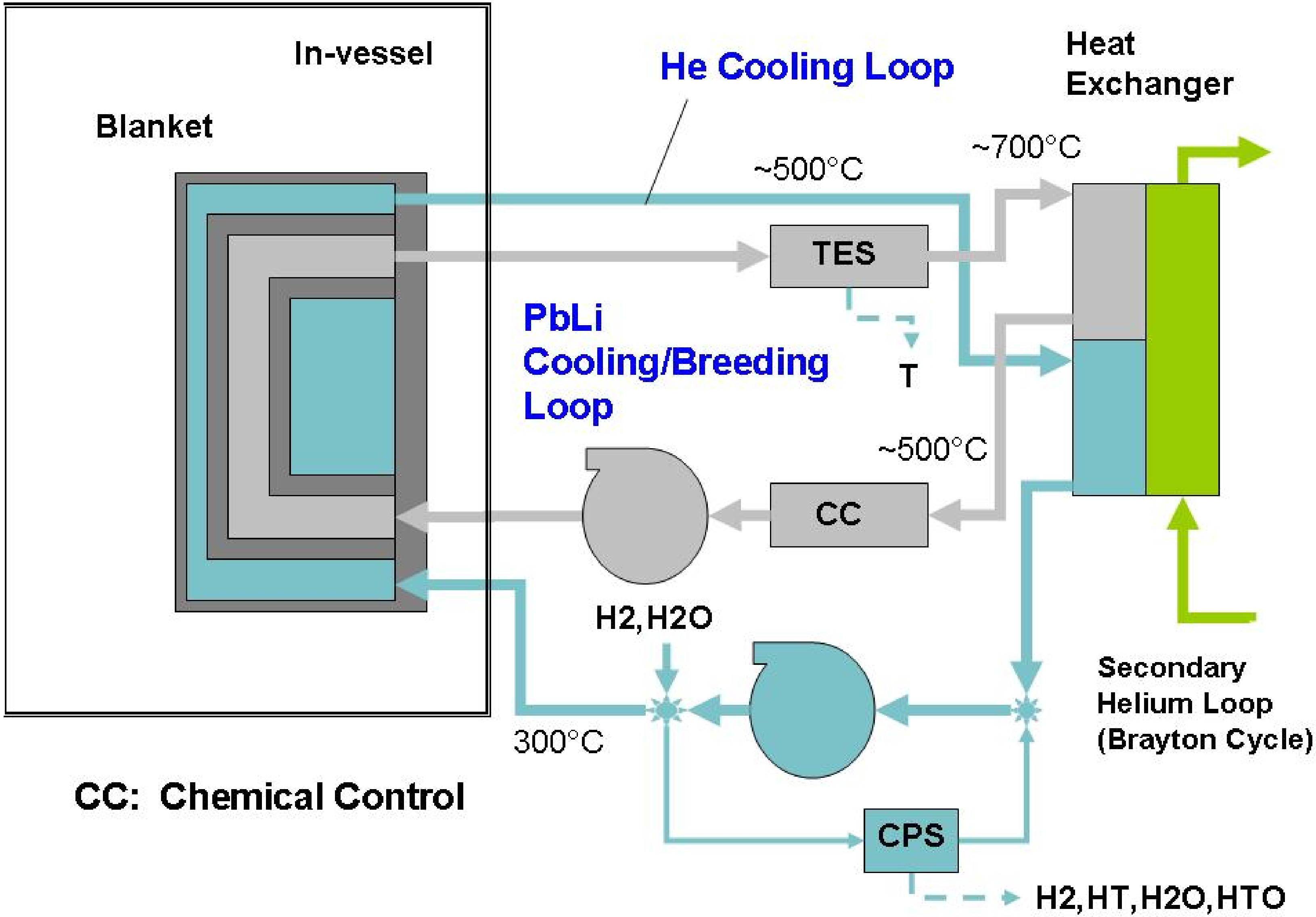
G. Rampal, HCLL TBM 2008

HCLL Blanket Concept (CEA, 2000)



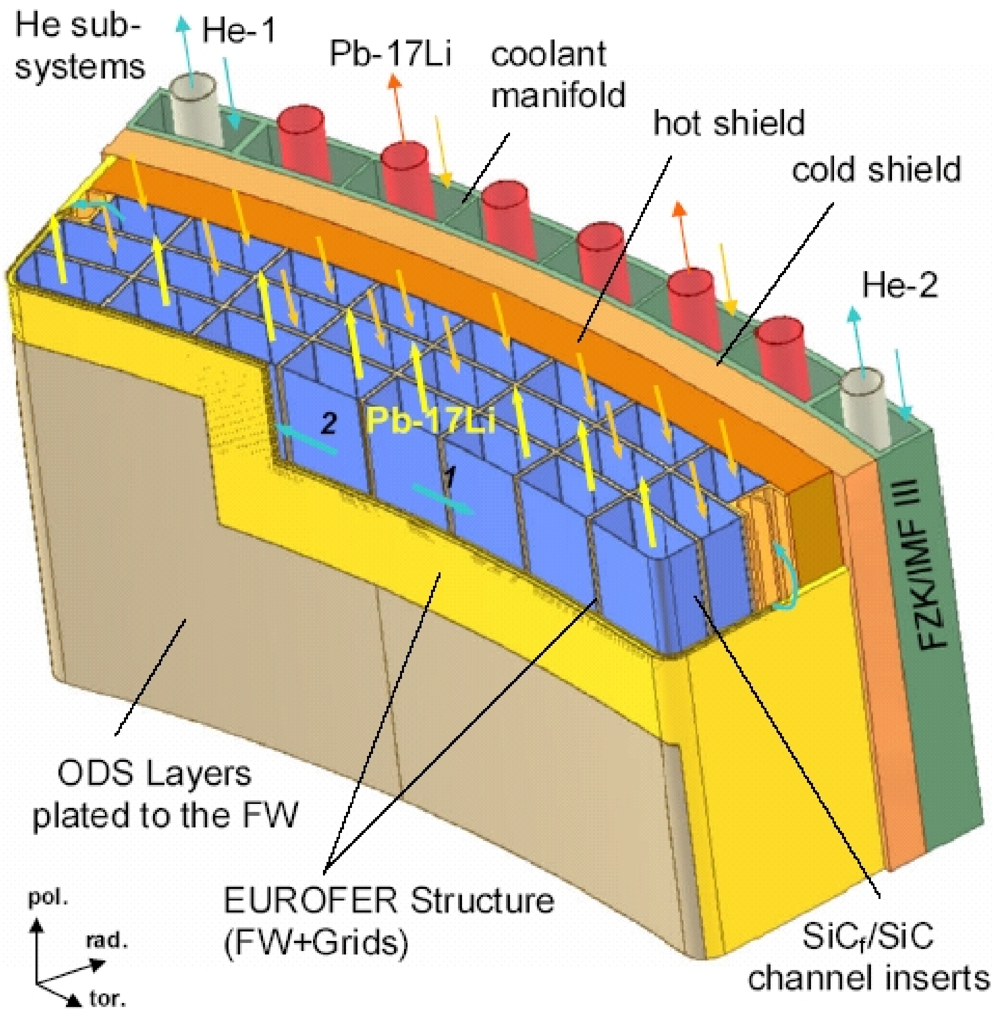
TES: Tritium
Extraction System

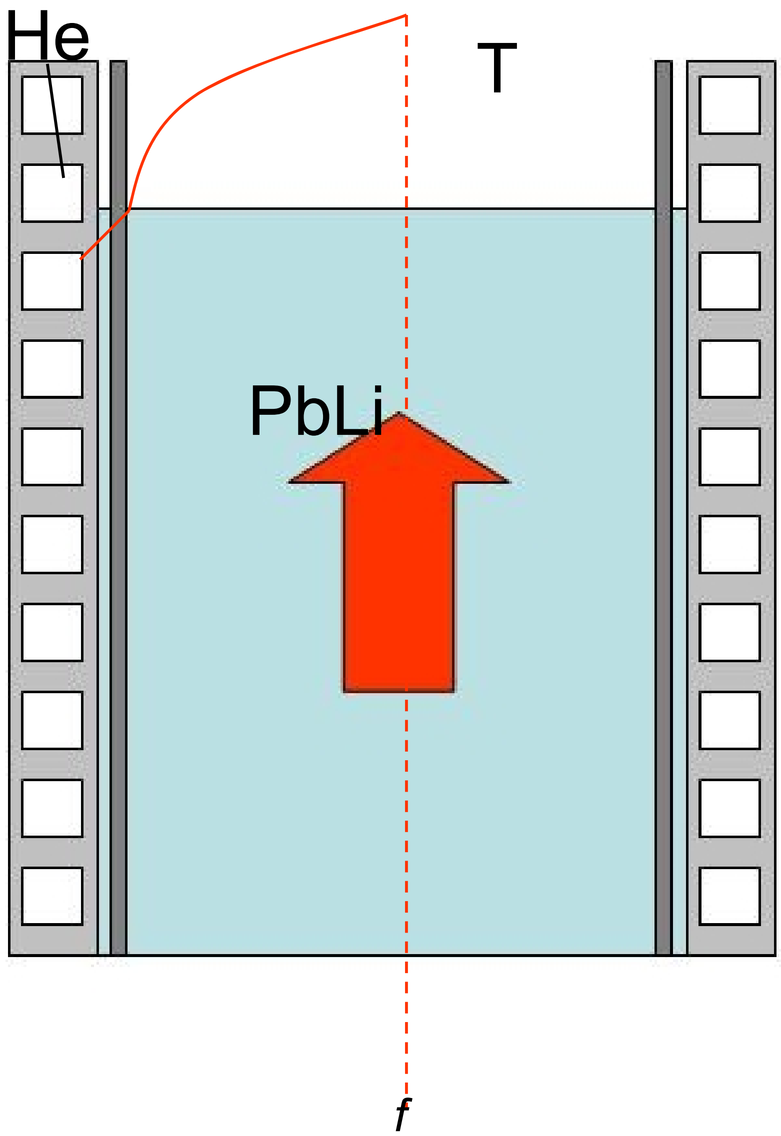
DCLL-HT Blanket (US & KIT, 2000)



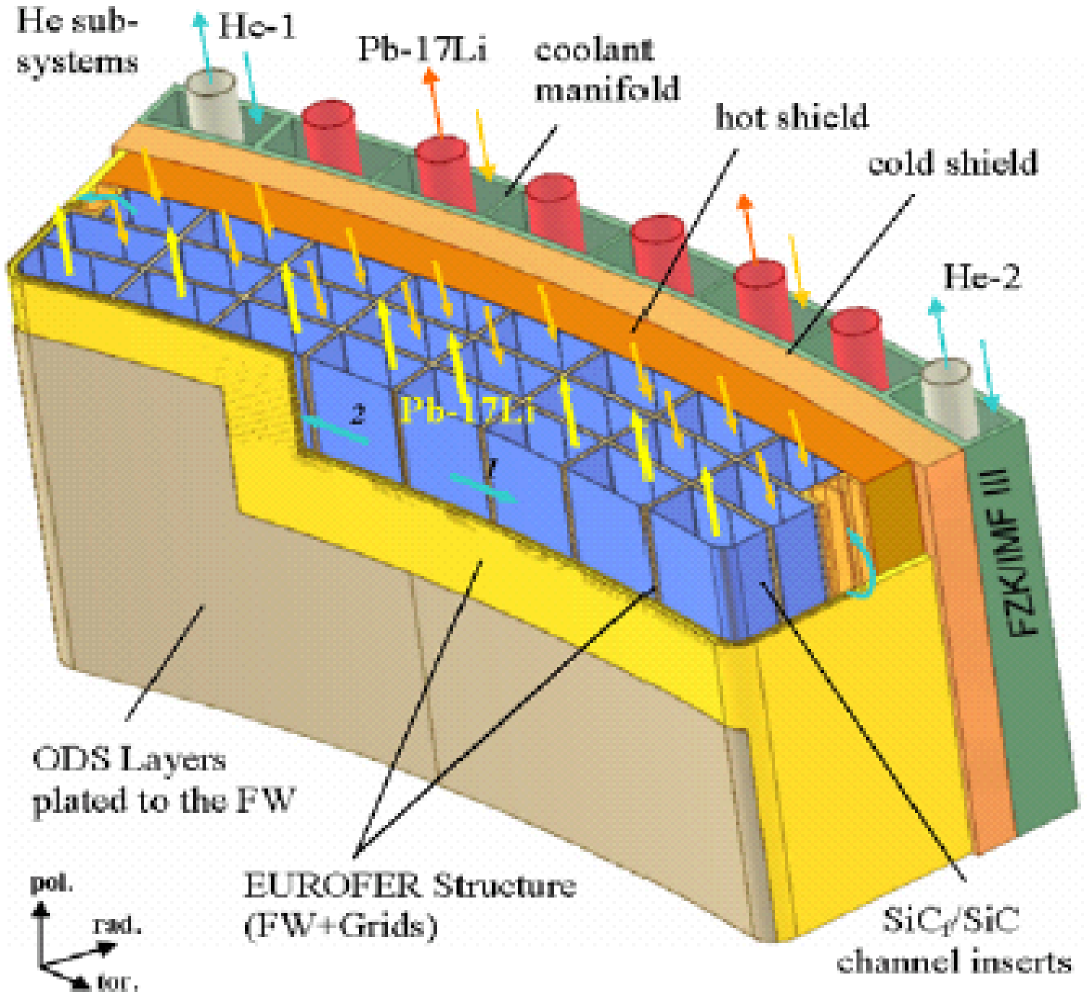
CC: Chemical Control

FZK DCLL Blanket (EU-PPCS Model C, 2002)





FZK DCLL Blanket (EU-PPCS Model C, 2002)

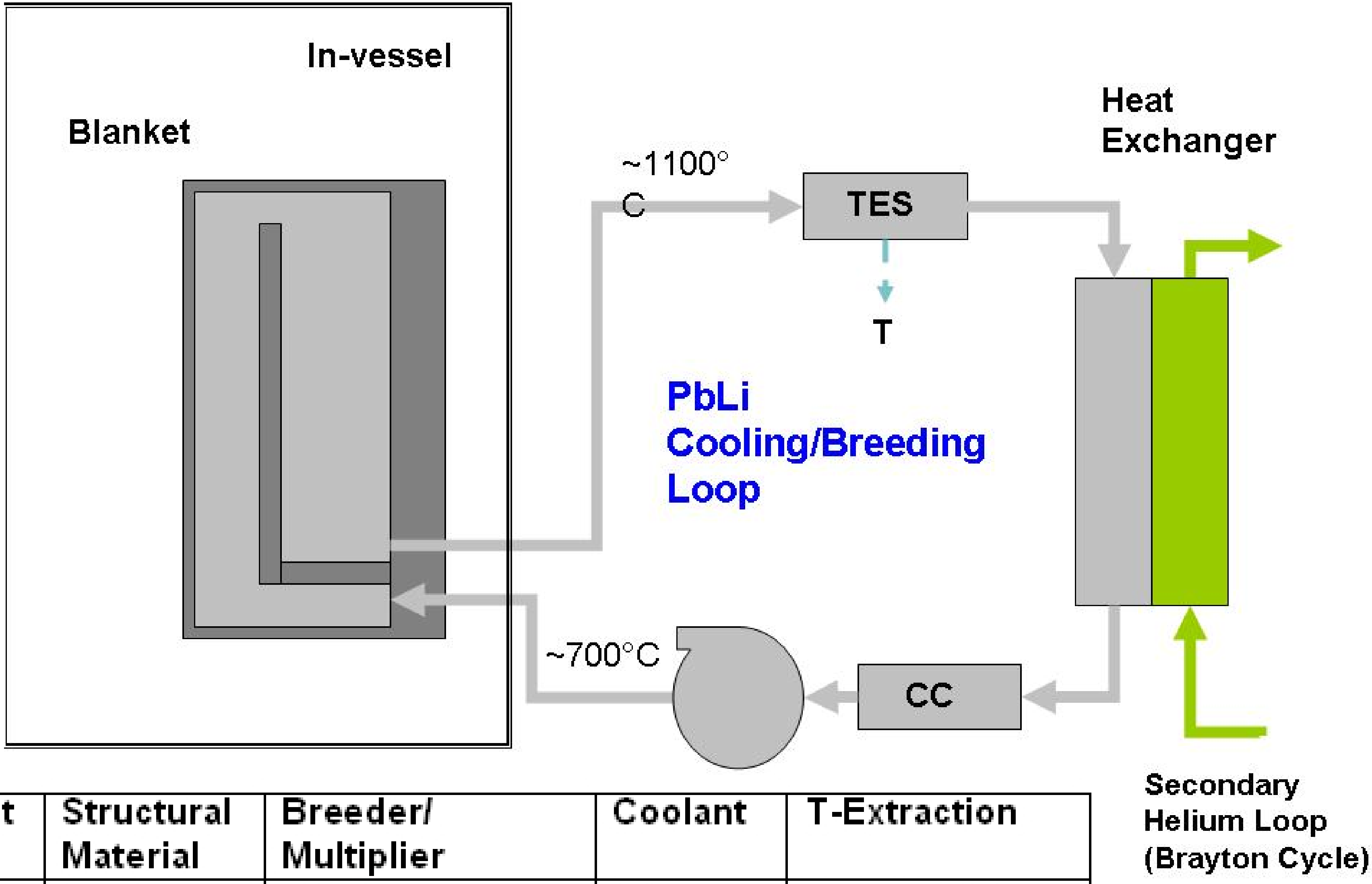


Main features:

- 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

US ARIES-AT, Blanket system

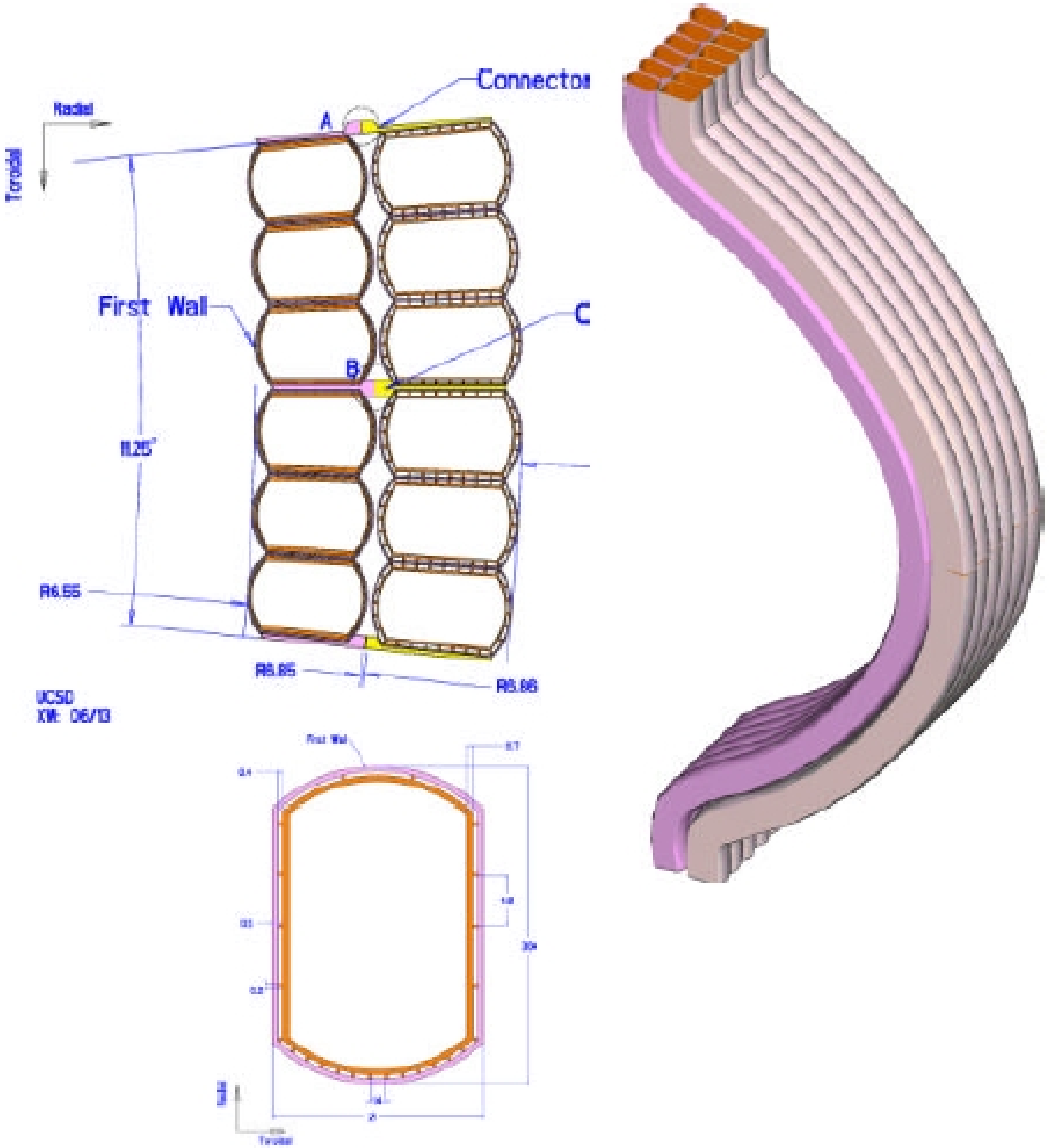


Concept	Structural Material	Breeder/ Multiplier	Coolant	T-Extraction
SCLL	Sic/Sic$_f$	PbLi	PbLi	Coolant PbLi

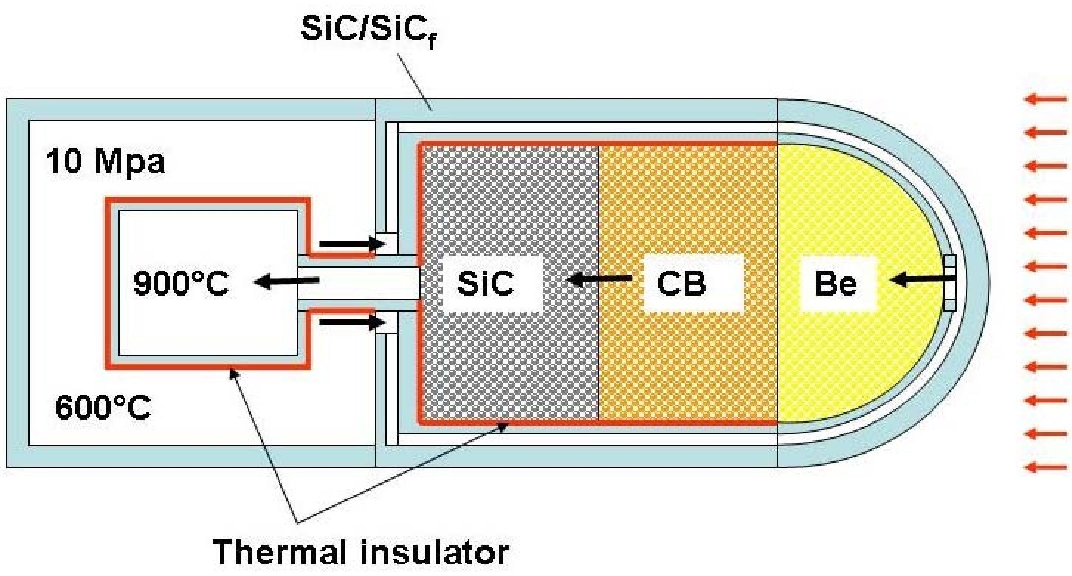
US ARIES-AT (2000)

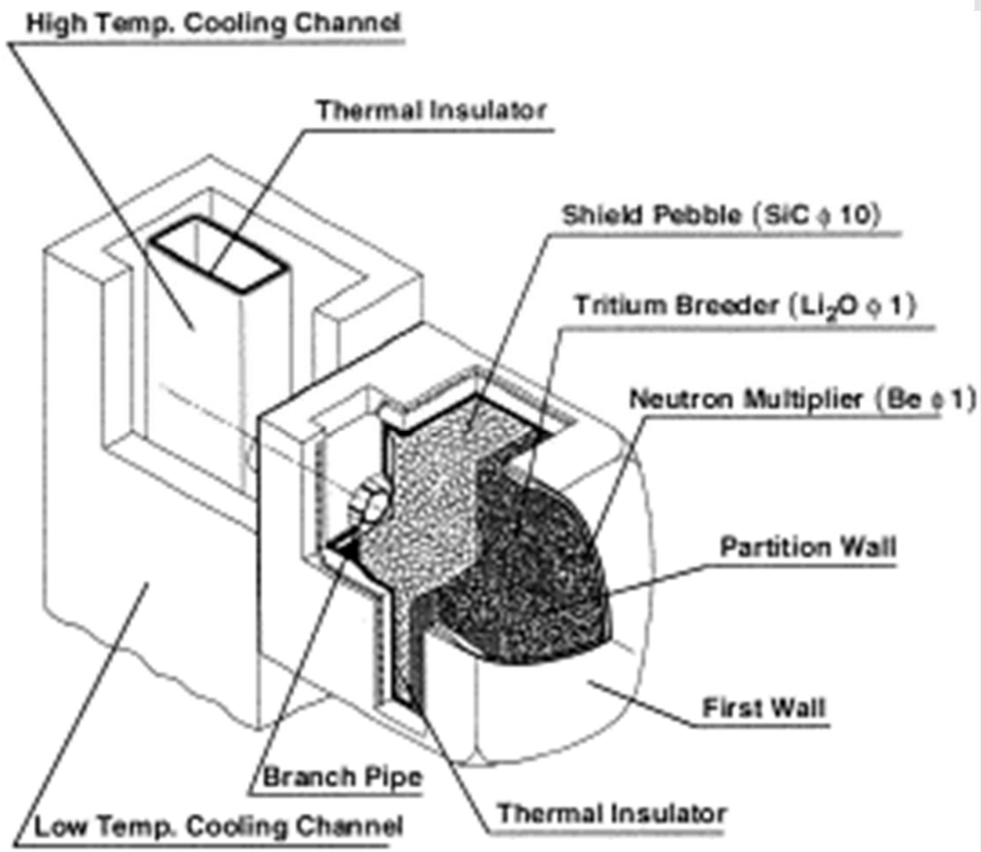
- Simple, low pressure design with SiC structure and LiPb coolant and breeder.
- Innovative design leads to high LiPb outlet temperature (~1100°C) while keeping SiC structure temperature below 1000°C leading to a high thermal efficiency of ~ 60%.
- Simple manufacturing technique.
- Very low afterheat.
- Class C waste by a wide margin.
- LiPb-cooled SiC composite divertor is capable of 5 MW/m^2 of heat load.

Outboard blanket & first wall



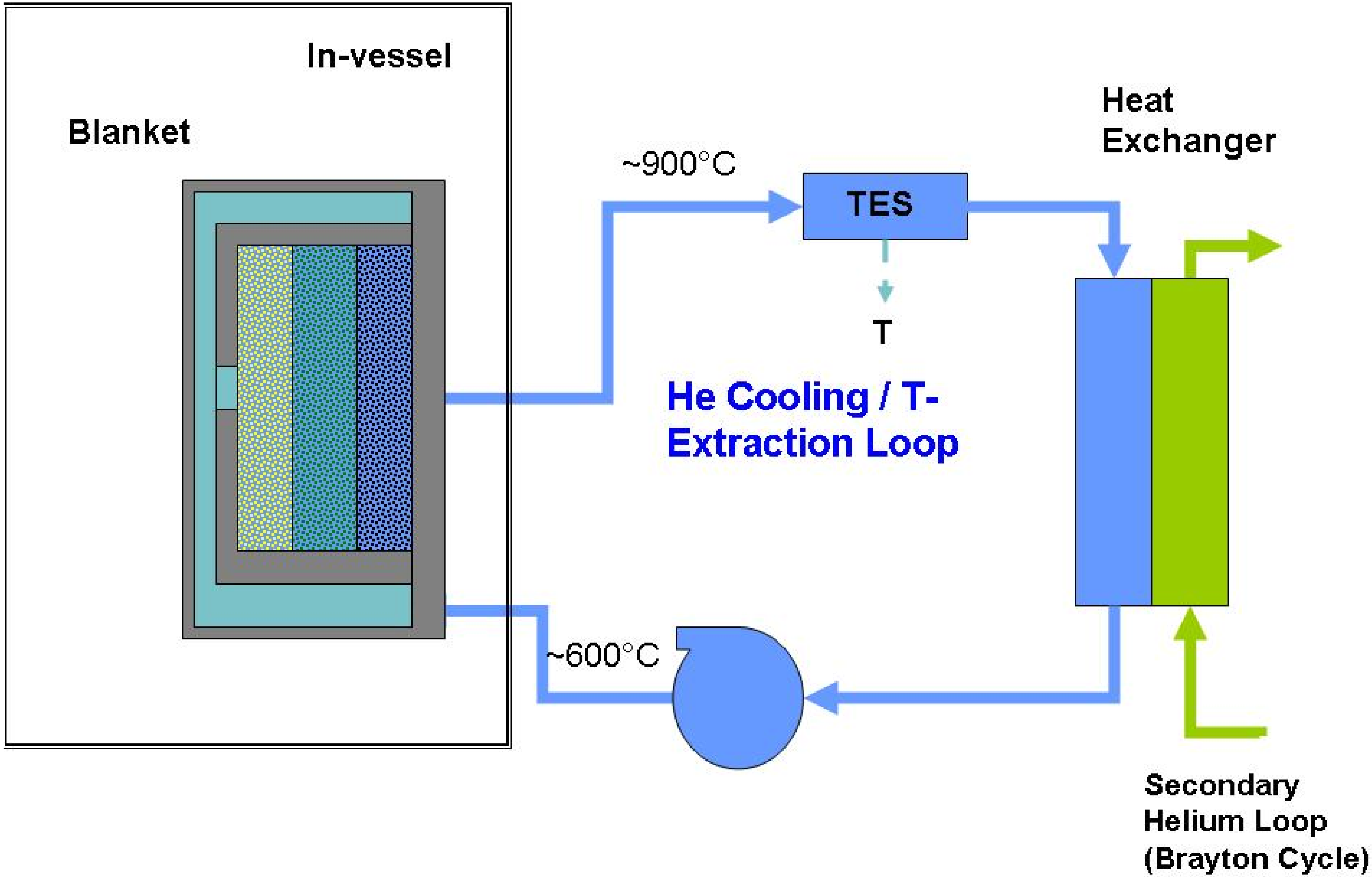
(Very) Advanced Solid Breeder Concepts: JA Dream



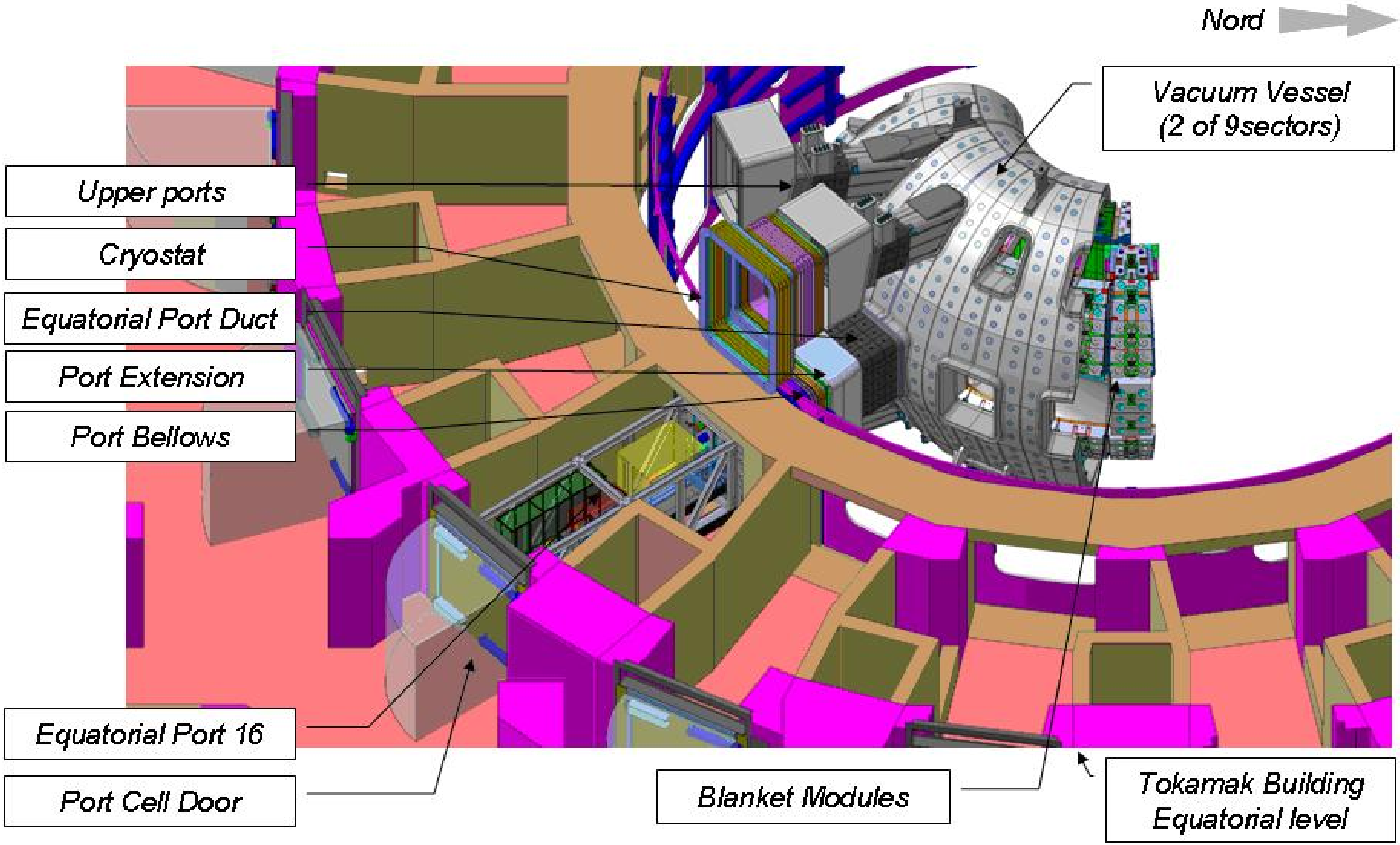


Concept	Structural Material	Breeder/ Multiplier	Coolant	T-Extraction
HT-HCSB	Sic/Sic$_f$	CB / Be-alloy	He	Coolant He

(Very) Advanced Solid Breeder Concepts: JA Dream



EU Test Blanket Systems in ITER



The two EU TBM Systems are located in the equatorial port #16 in a common Port Plug.

Strategy and Requirements of the TBM Programme

Limitation in the TBM test in comparison to a DEMO component conditions:

Major loads	**in the ITER TBM [Gian12]**	**in EU DEMO (2015) [Fede16]**	**in EU FPP (PPCS-2004) [Mais05]**
Neutron wall load $[MWm^2]$	0.78	1.3-1.8	>2.2
Radiative surface heating $[MW/m^2]$	0.30	0.50-0.70	0.50-0.70
Conductive surface heating $[MW/m^2]$	None*	Tbd (up to 5) **	Negligible***
Pulse length [h]	0.11	2.5	Stationary or long pulses (>8)
Cumulative fluence at the FW [dpa]	3	70	>800

* the ITER TBM FW is recessed of several cm respect to the Shielding Blanket FW and so protected from the plasma flux at the edge

** It will be dependent from the final FW design and protection strategy (e.g. limiters)

*** It was assumed almost negligible at 15 cm FW distance from the separatrix.

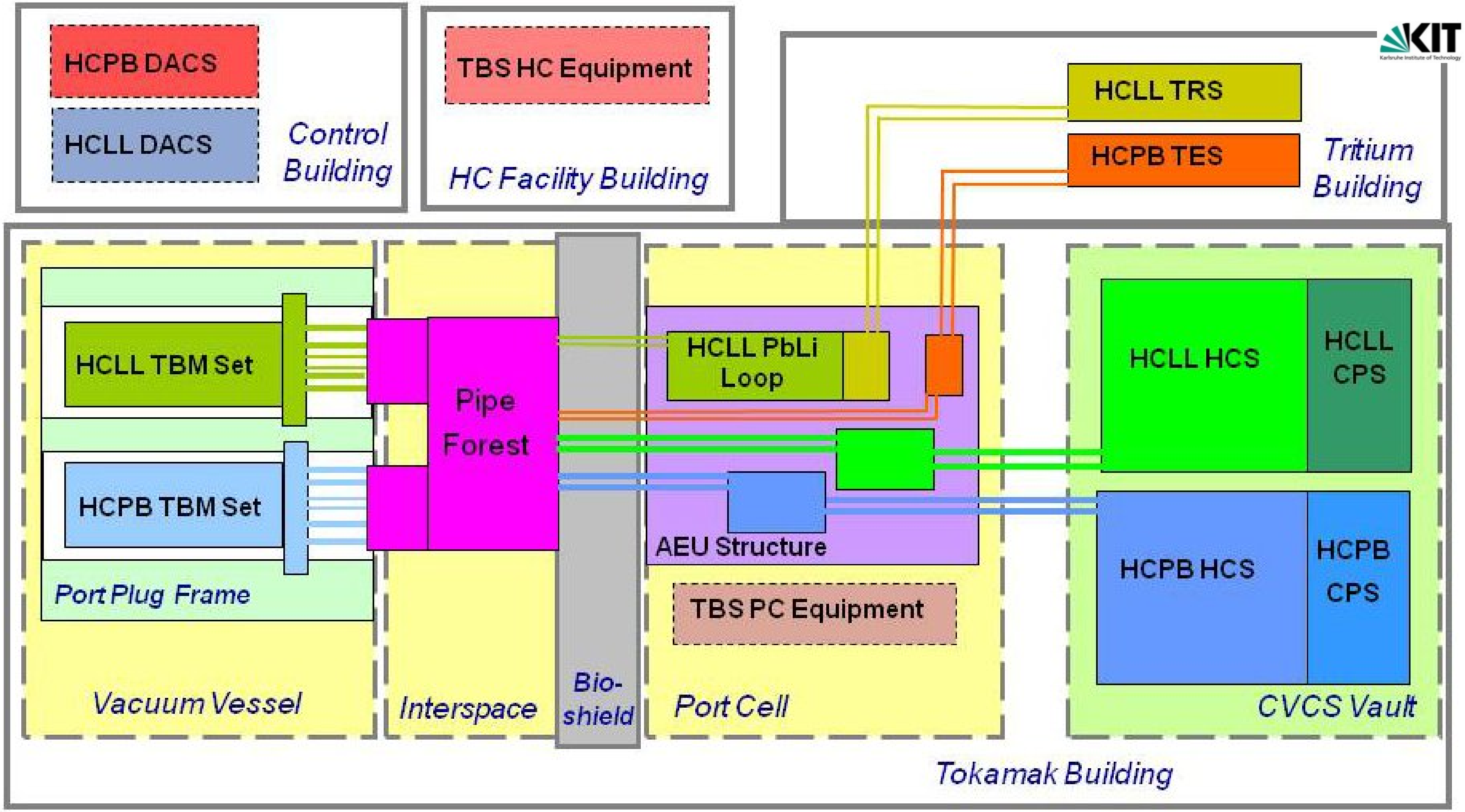
Strategy of testing developed in several studies (e.g. [Bocc02], [Poit07]); the main objectives can be summarised:

1) Development of tools (experimental and computational) to capitalise the experimental results in ITER and extrapolate this to DEMO.
2) Development of manufacturing technology with EUROFER
3) Create a precedent in the safety and licensing of blanket components
4) Integral test of processes, materials and technologies (at least for BOL conditions).

EU Test Blanket Systems

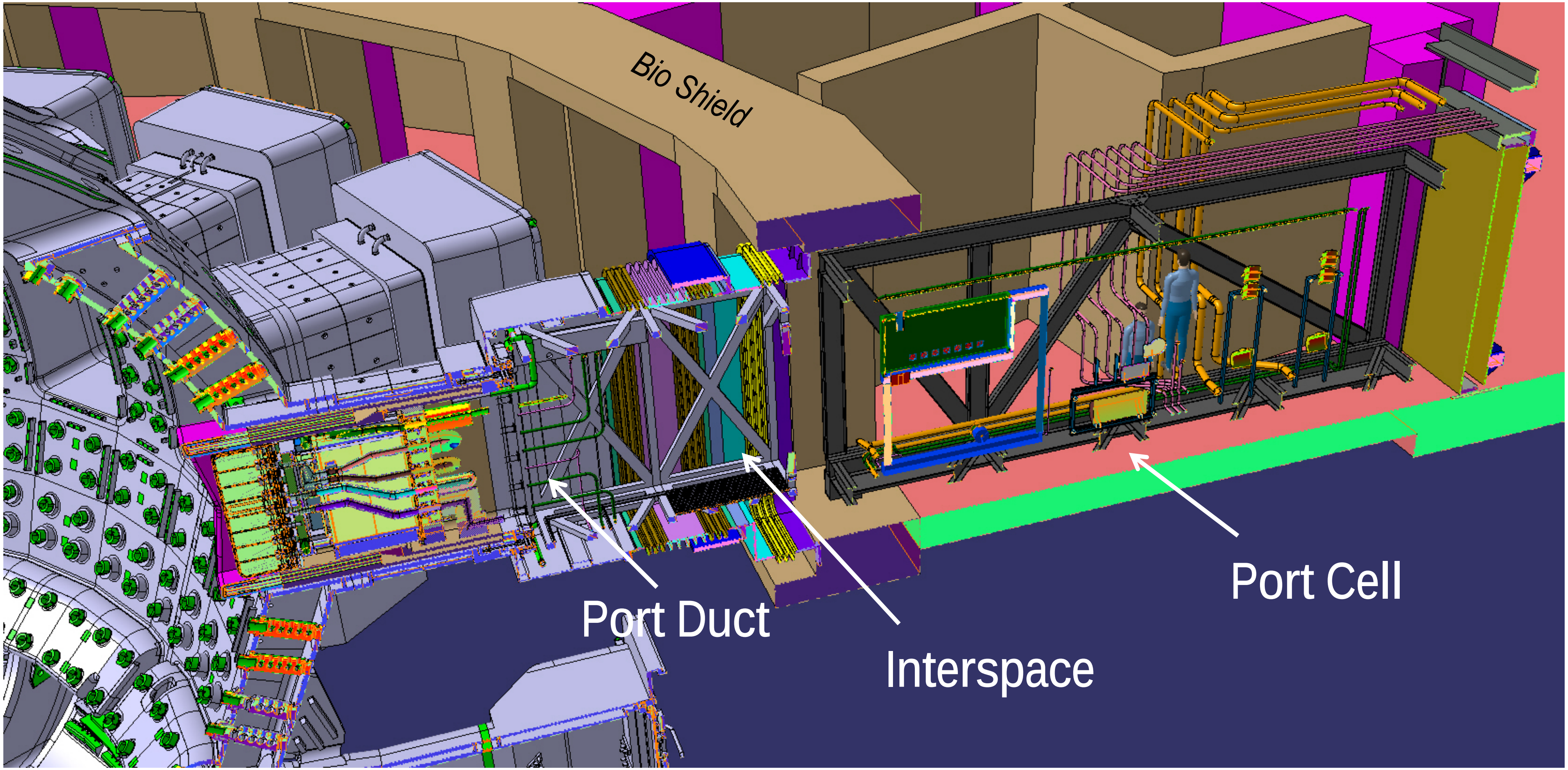
Breeder blanket concepts: HCPB and HCLL.

Presentation based on: **[Bocc11].** Recent development from 2012 are not included.

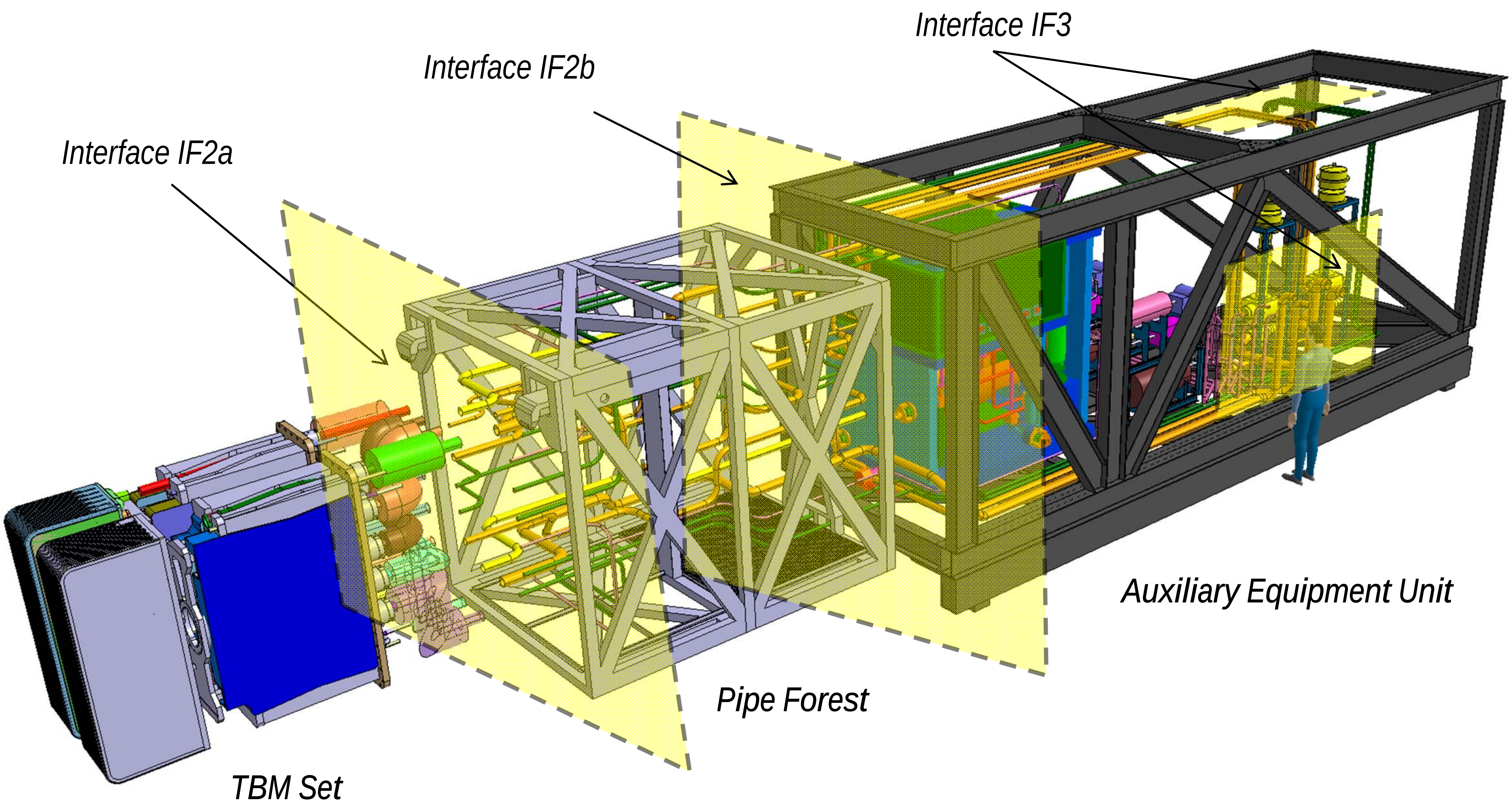


Integration in the ITER VV and Port Cell

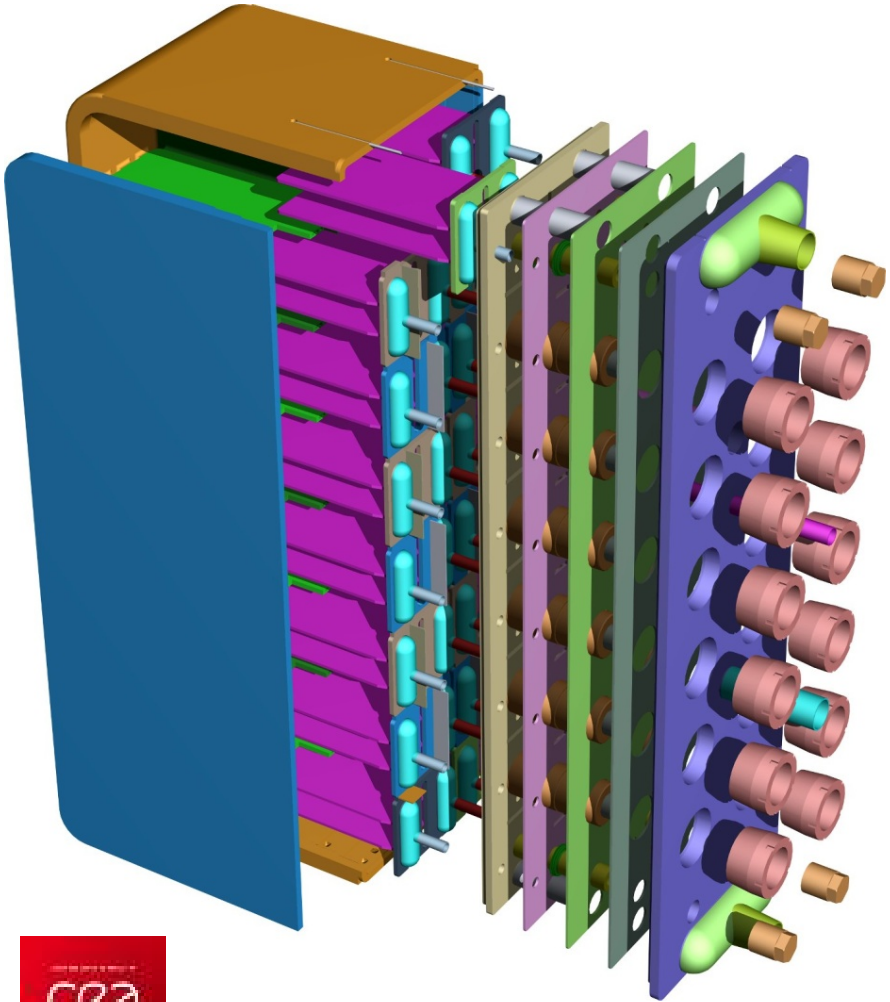
KIT
Karlsruhe Institute of Technology



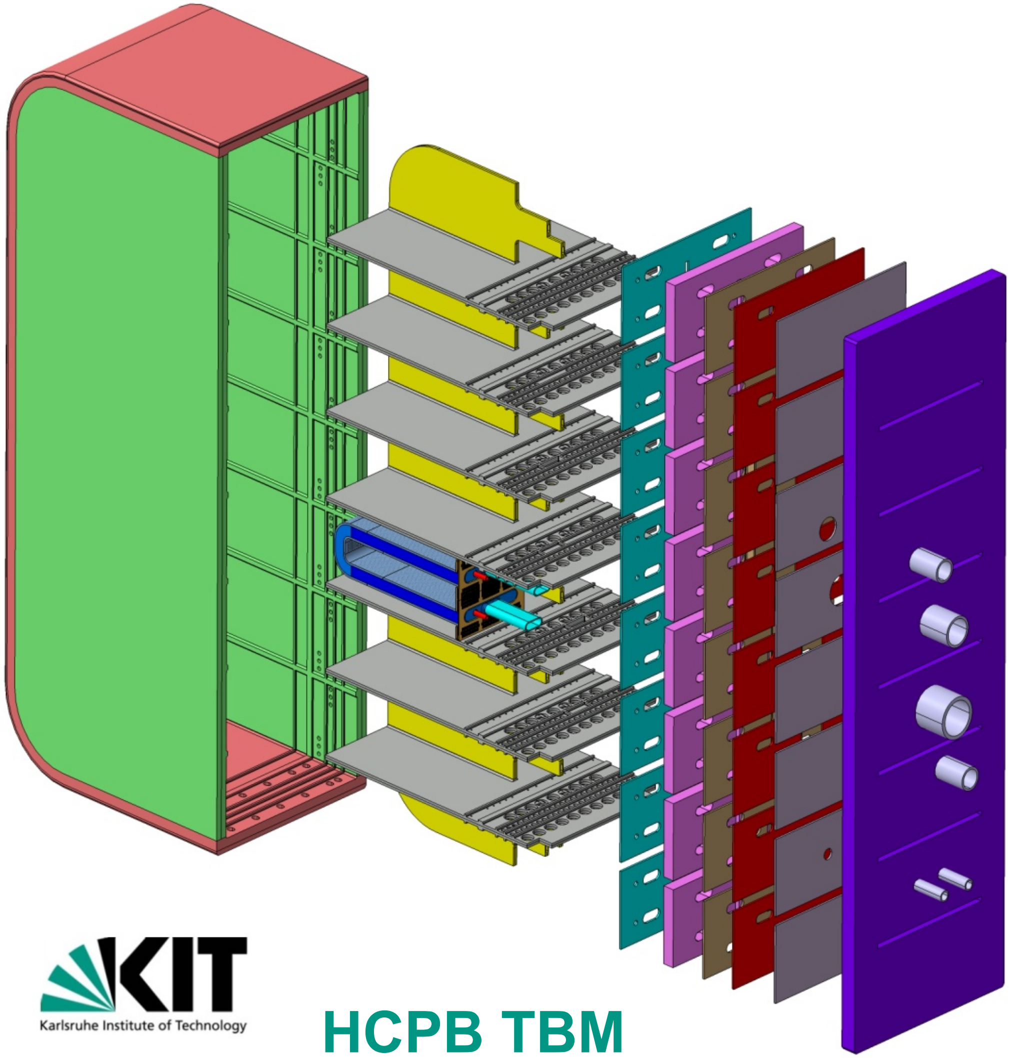
The TBS equipment in VV-Interspace-Port Cell



EU Test Blanket Modules

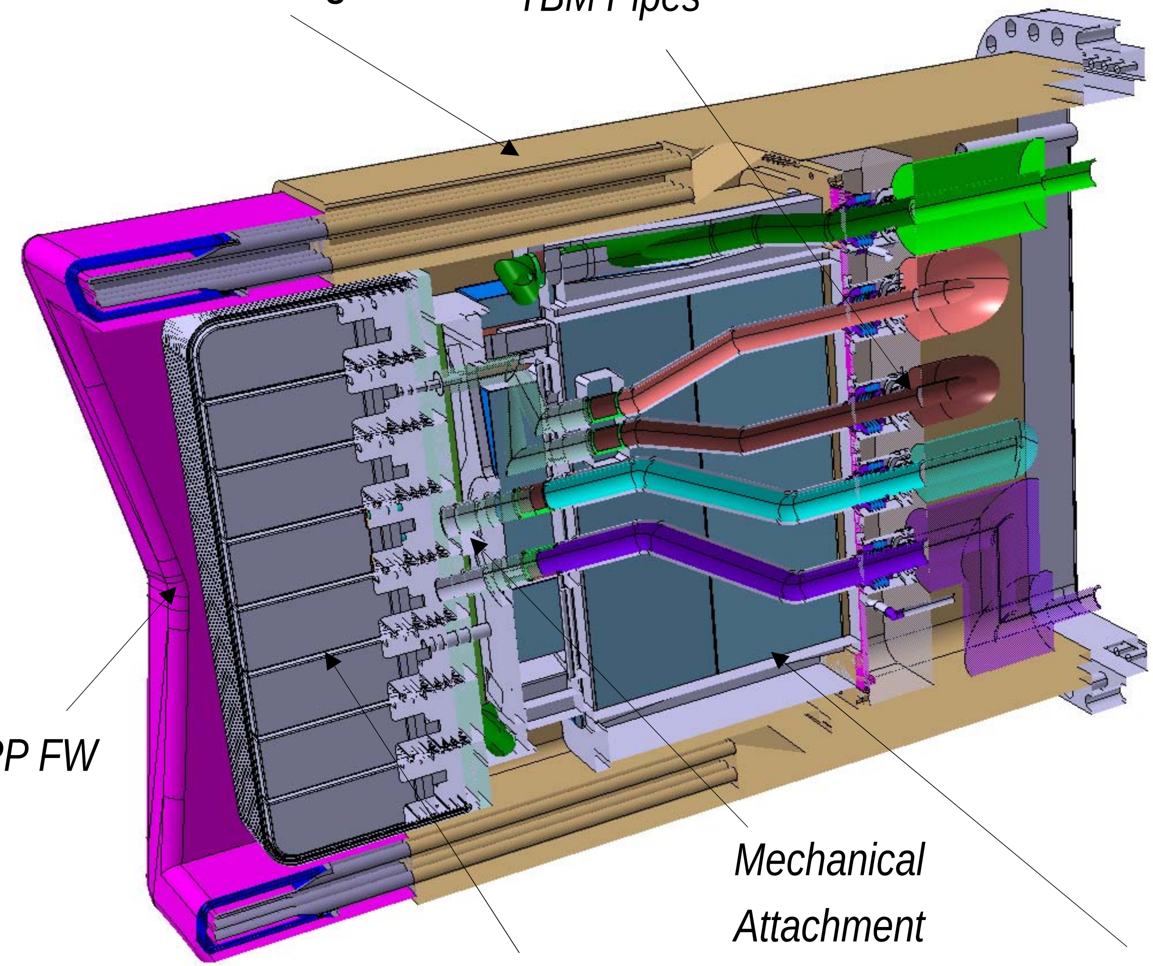


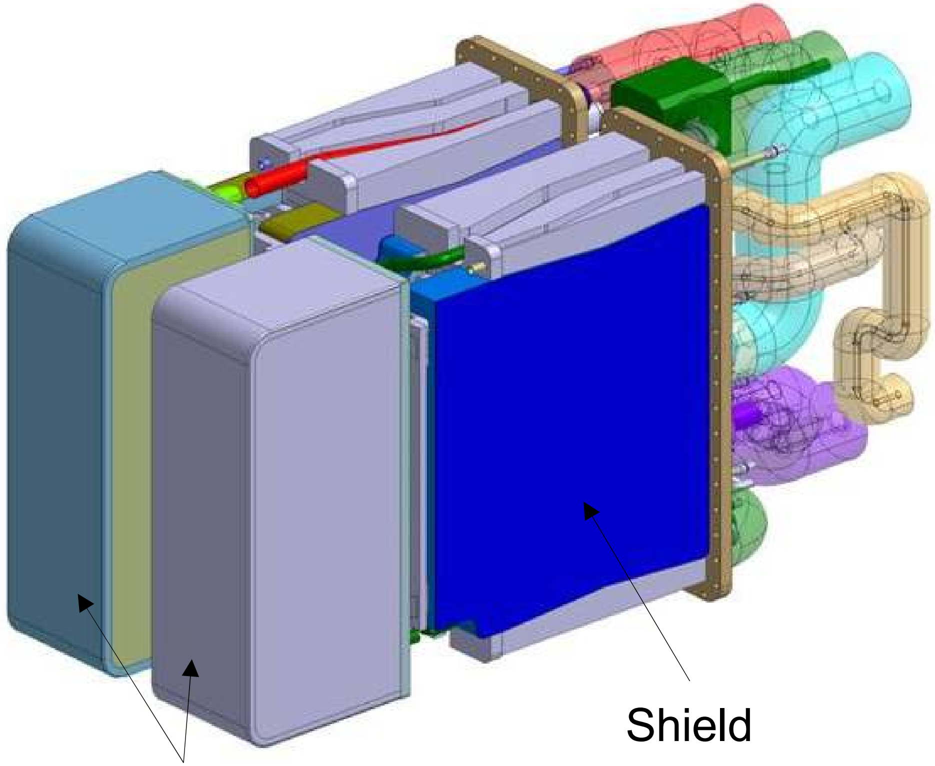
HCLL TBM



HCPB TBM

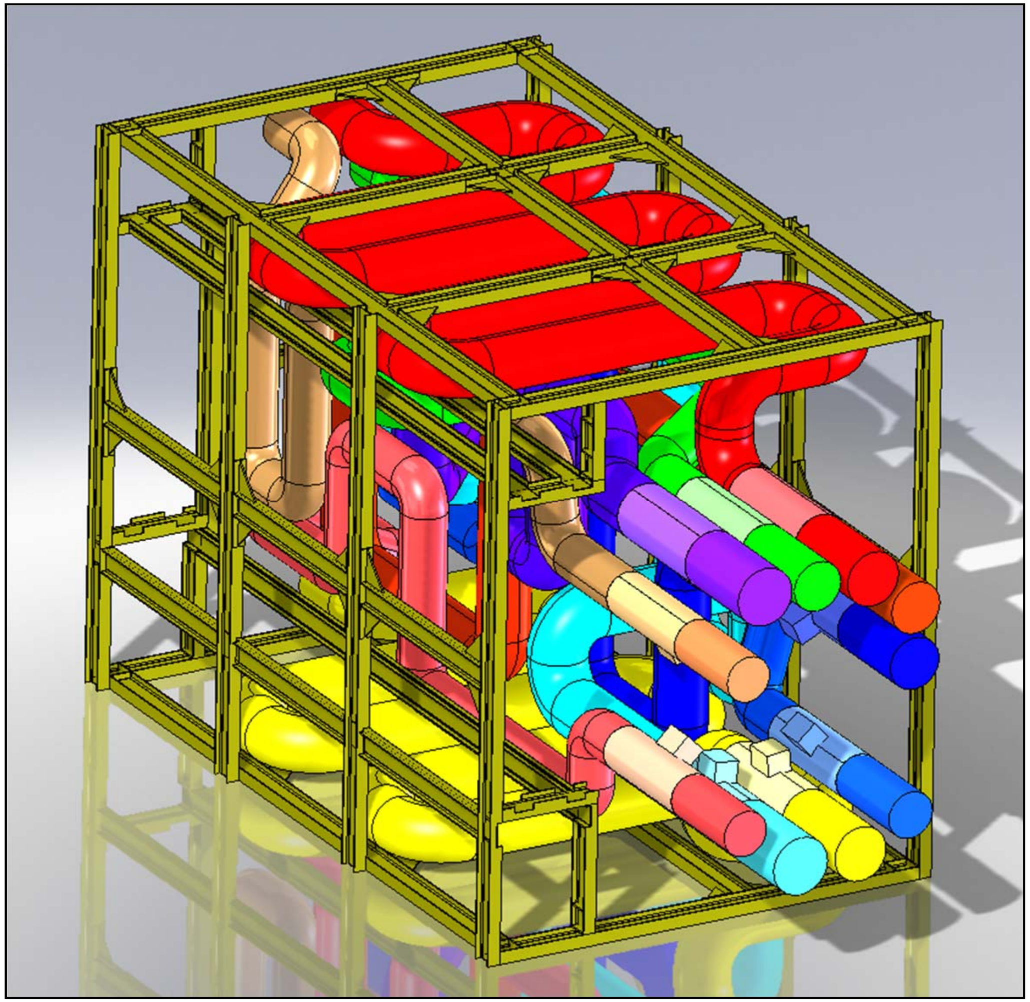
TBM integration in the Port Plug Frame

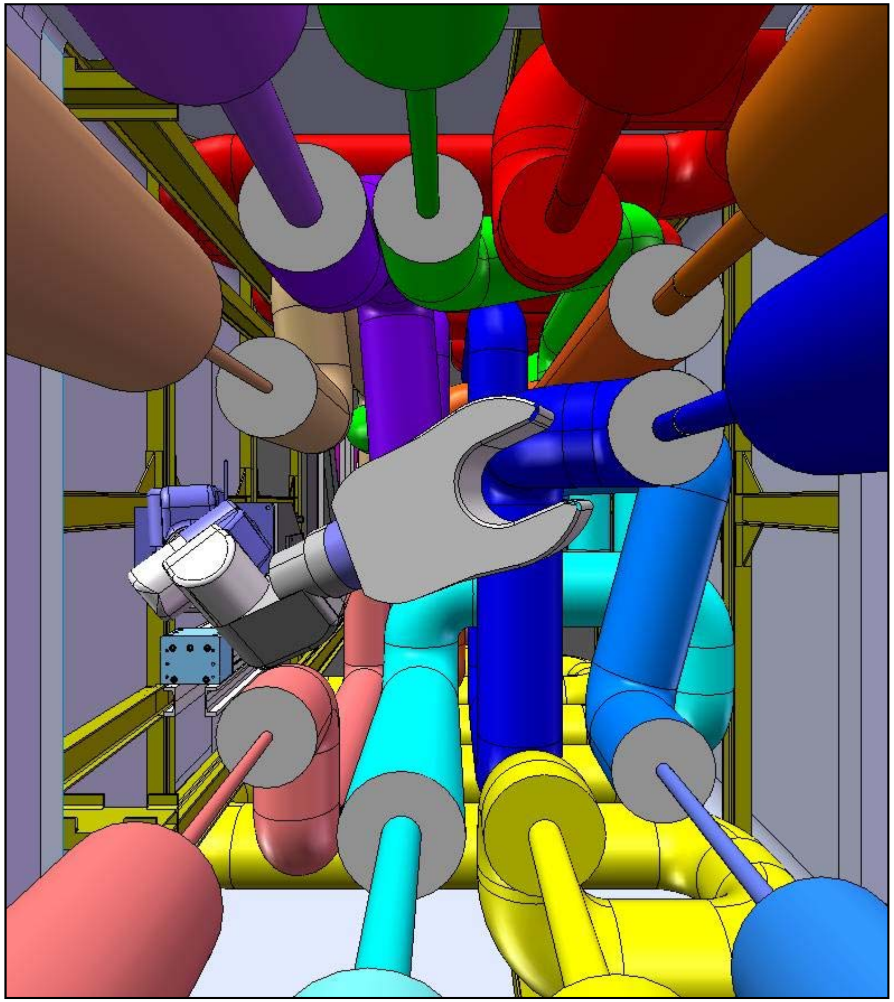




HCLL and HCPB TBM-Set

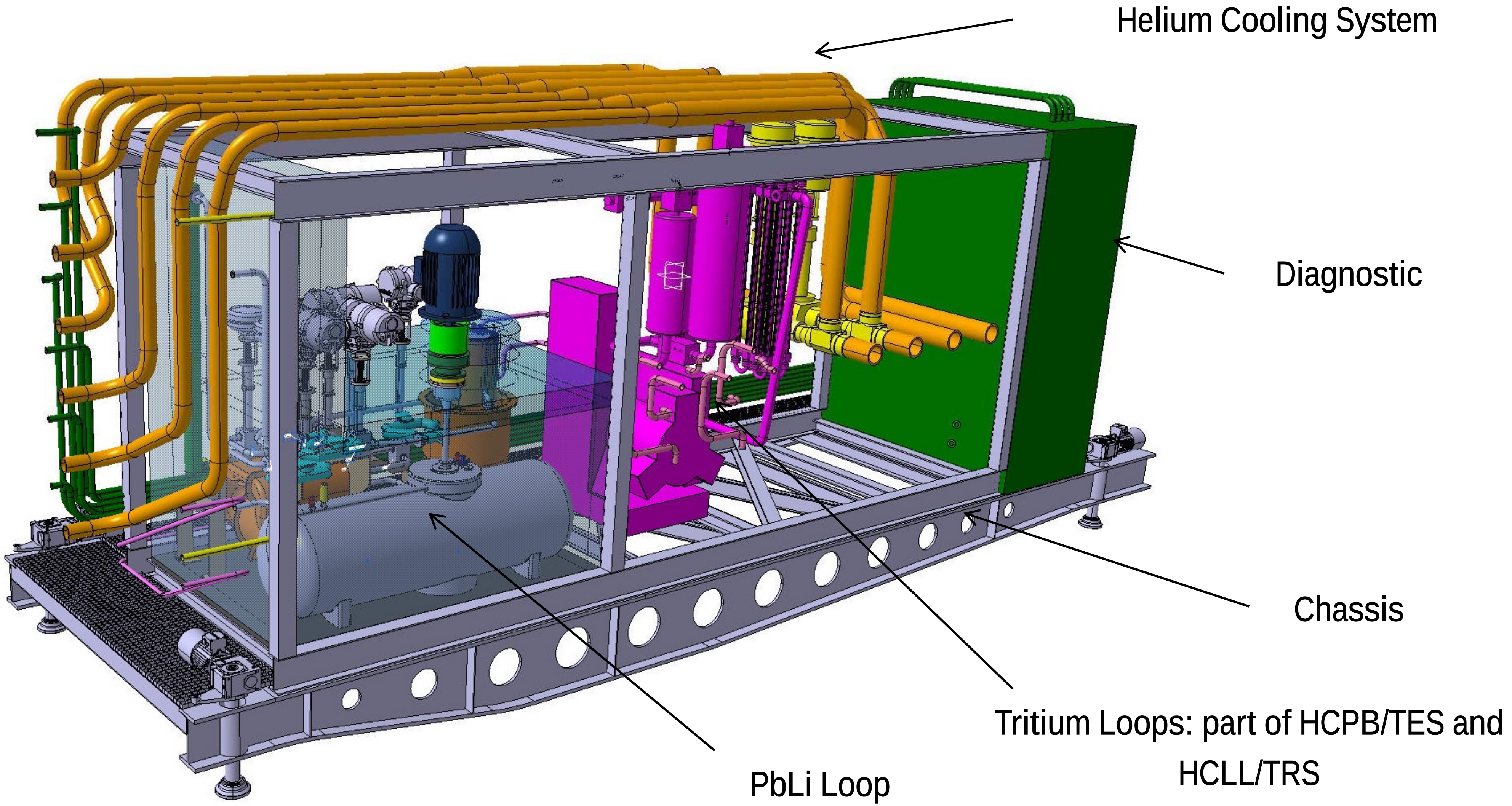
Pipe Forest and RH at the interface IF2a





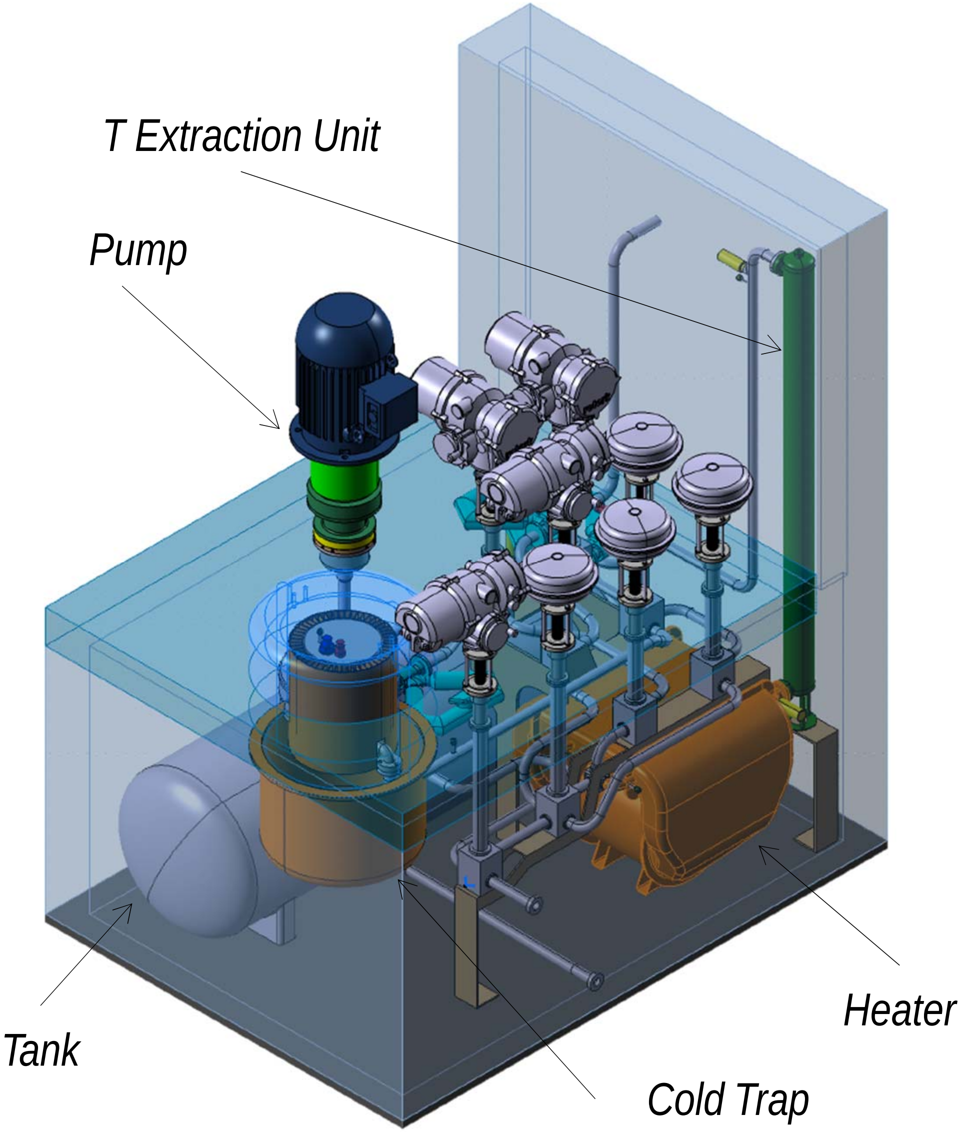
D. Keller (CEA)

AEU Structure and TBS Subsystems



T. Ilkei (RMKI)

The PbLi Loop

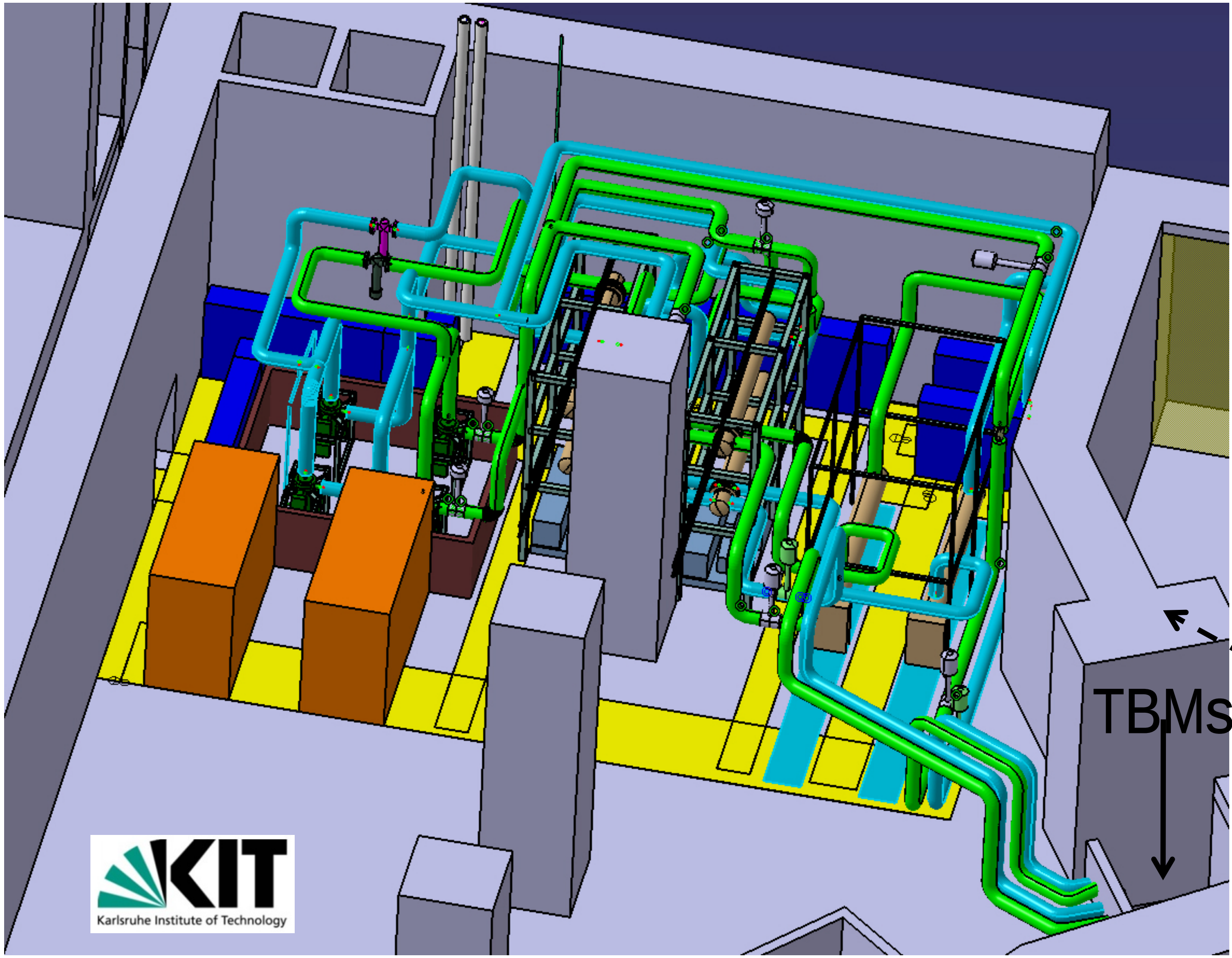


Operational Parameters:

PbLi mass flow range:	0.1-1kg/s
PbLi volume:	0.5m^3

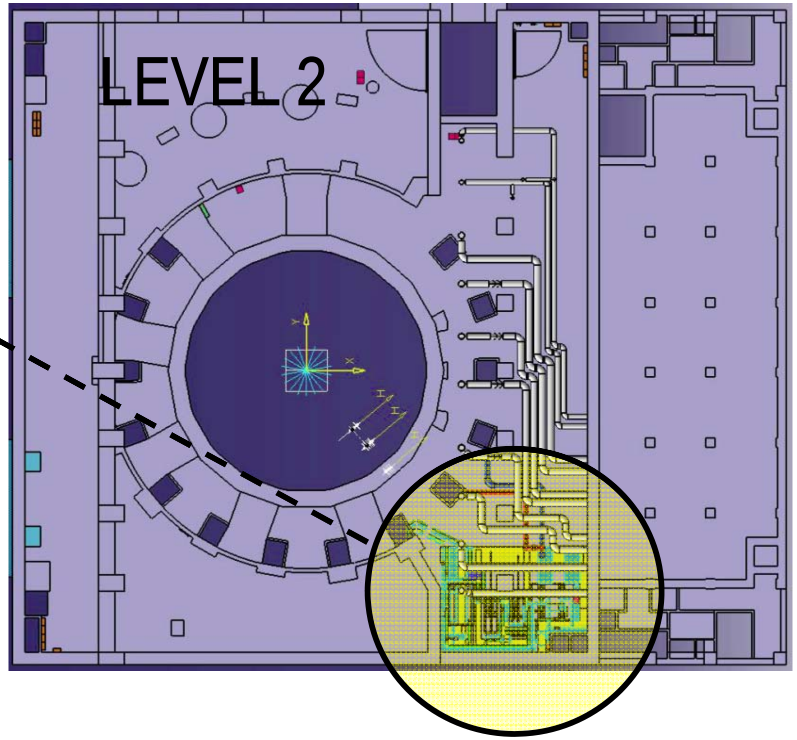
L. Kosek (NRI)

The Helium Cooling Systems: integration in the CVCS



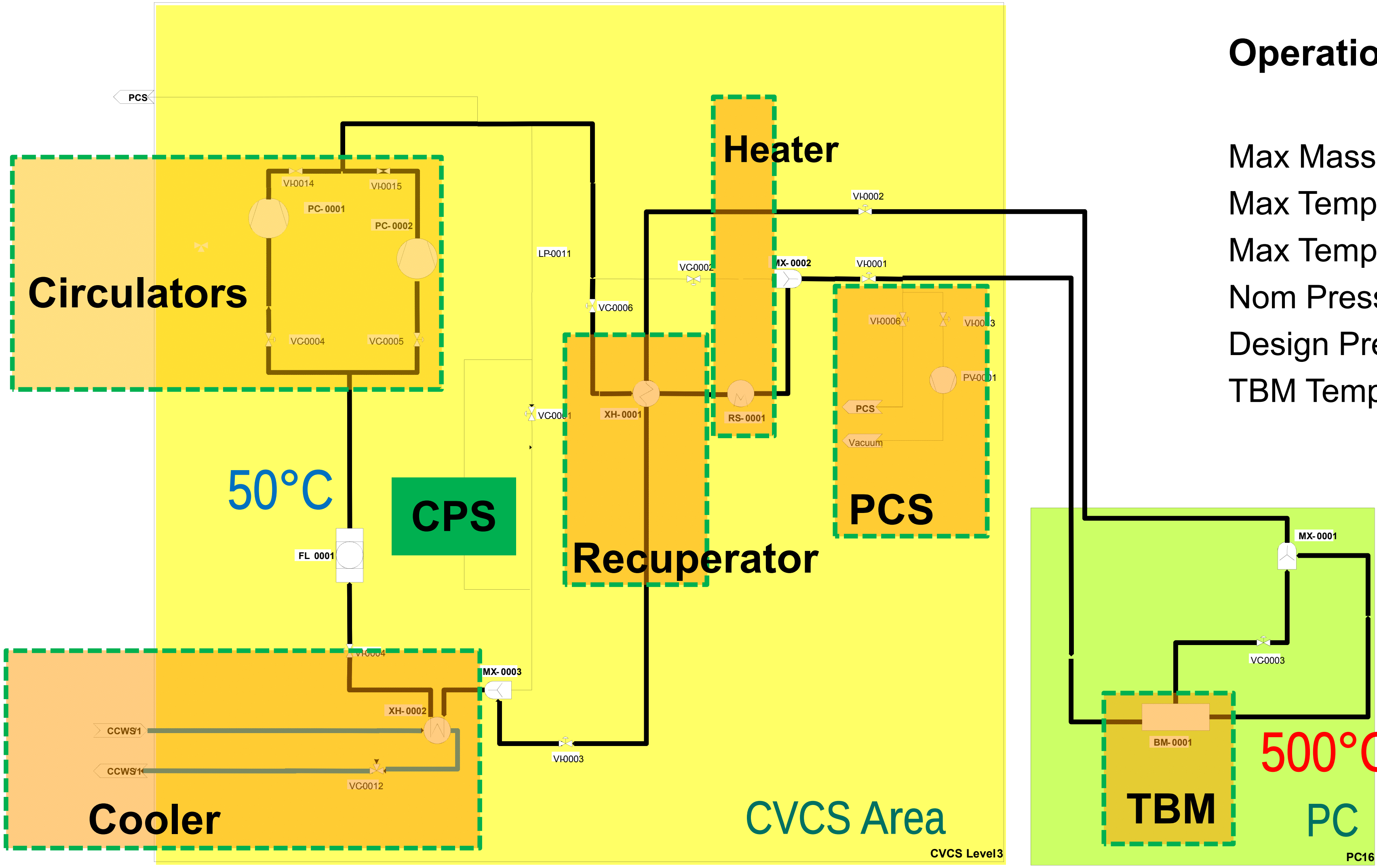
Both HCLL and HCPB HCS are arranged in the CVCS area (in a reserved space of about 222 m^2).

Each HCS should cool 0.8 m^2 of the ITER FW, removing about 0.6 MW heat during the pulses.



B-E. Ghidersa, KIT

The Helium Cooling Systems: flow diagram



Operational Parameters:

Max Mass flow:	1.4 kg/s
Max Temperature:	500°C
Max Temp. at circ. inl.:	50°C
Nom Pressure:	8 MPa
Design Pressure:	10 MPa
TBM Temp. inlet:	300°C

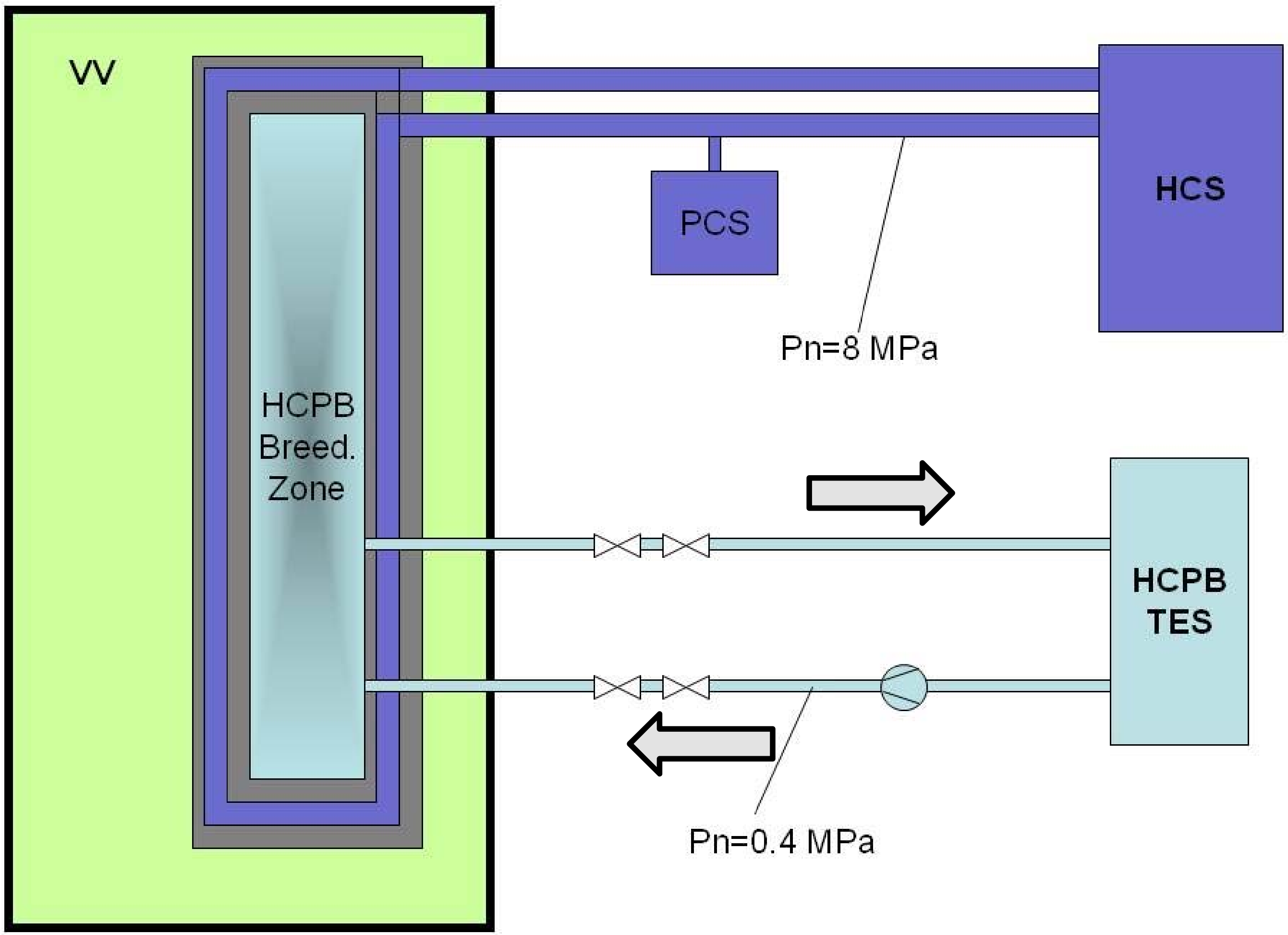
PCS: Pressure Control System

CPS: Coolant Purification System

PC: Port Cell

B-E. Ghidersa, KIT

The Tritium Extraction in the HCPB concept



<u>Ceramic breeder</u> in form of a pebble bed.

- Pebbles with D<1mm
- Materials: Li_4SiO_4 or Li_2TiO_3
- Single bed (PF 64%)

<u>Be Multiplier</u>:

- Pebble beds
- Pebble D<2mm
- Single beds (PF 64%)

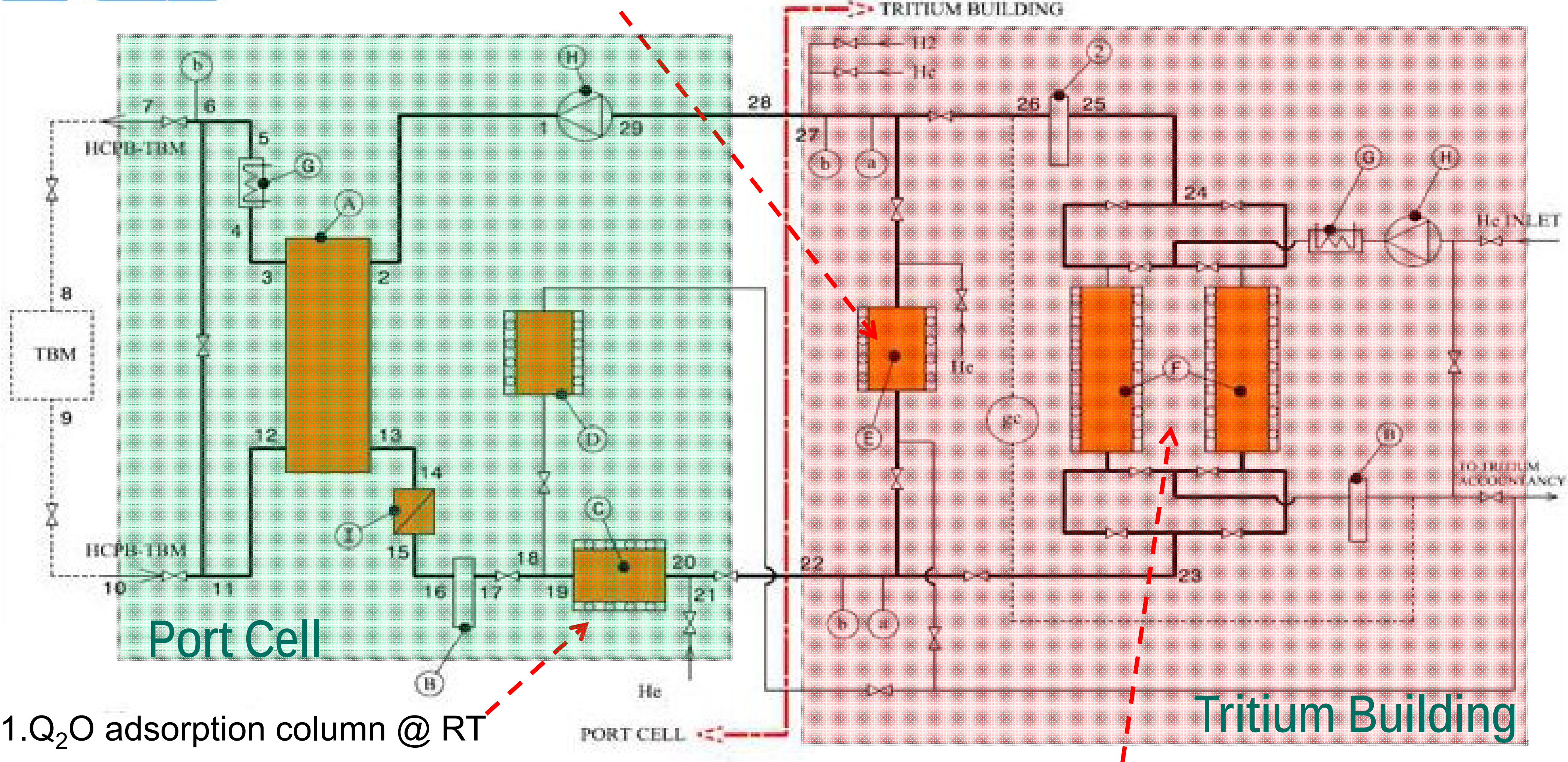
<u>Purge gas composition:</u>

Inlet: He +H_2 addition (0.1%)

Outlet: H_2, H_2O, HT and HTO

The Tritium Extraction System (HCPB specific)

2. For LD-phase: Q_2 getter beds based ZrCo @ RT

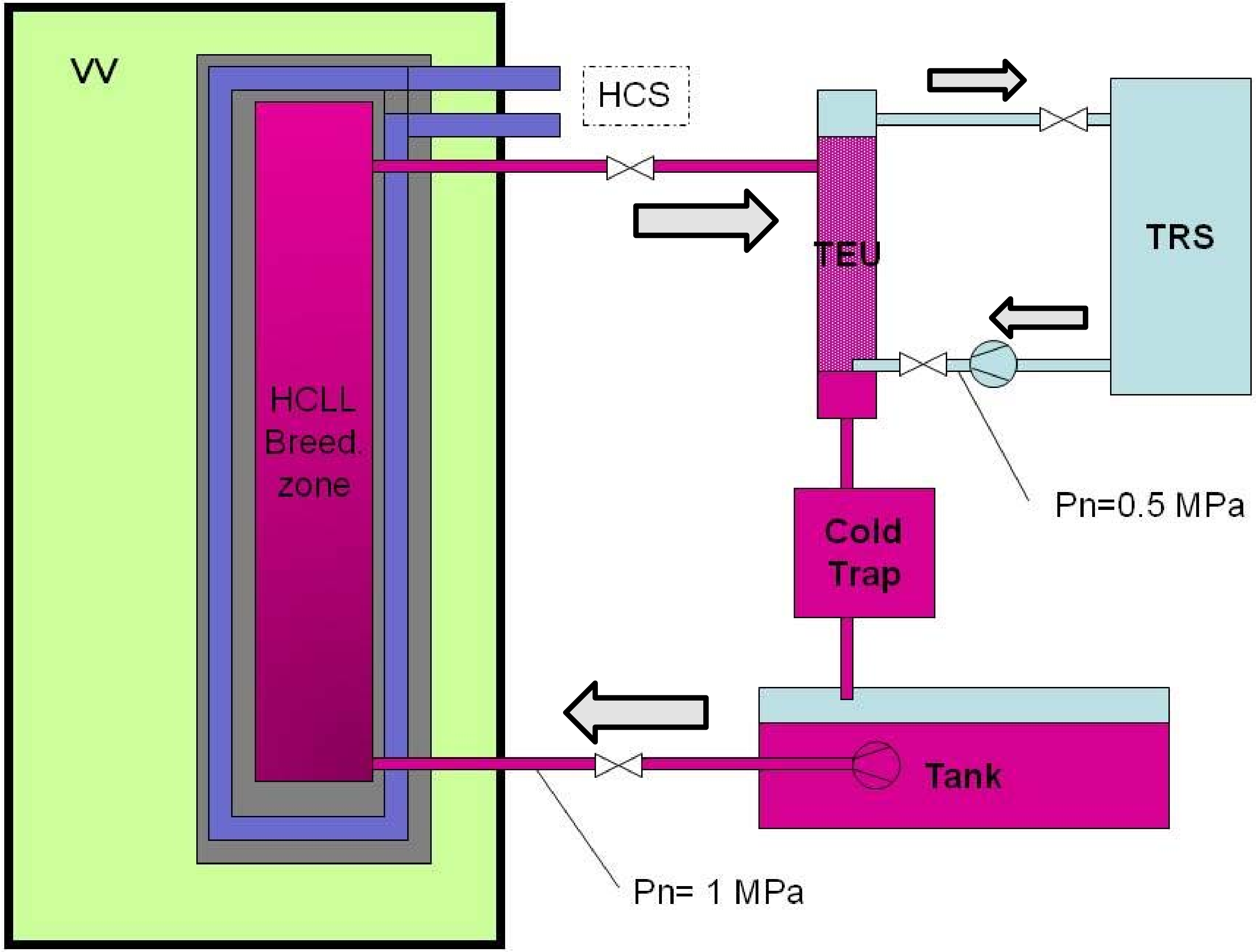


1. Q_2O adsorption column @ RT

3. For HD-phase: 2 Q_2 getter beds based ZrCo @ RT

A. Ciampichetti (ENEA)

The Tritium Extraction in the HCLL concept



TRP Gas composition:

Mass flow: 1.3 Nm_3/h

Inlet: He + 1% of H_2

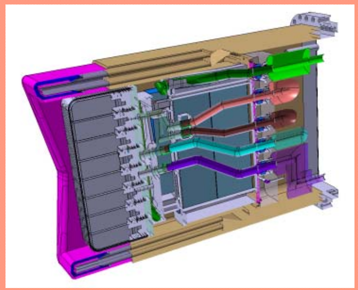
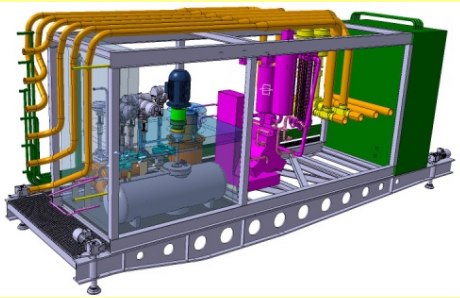
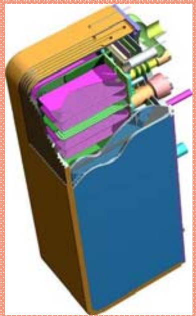
Outlet: H_2, HT

TEU: Tritium Extraction Unit

TRS: Tritium Removal System

VV: Vacuum Vessel

Operation in Hot Cell Facility for the TBS

1) Refurbishment of the TBM Port Plug		Dedicate TBM Plug Workstation in red zone: the TBM Set is separated from the Port Plug Frame.
2) Refubishment of the AEU		Dedicate AEU Workstation in yellow zone: maintenance of the AEU Subsystems and pipe preparation.
3) Preparation of TBM transport for PIE		Dedicate TBM Workstation in red zone: the TBM is separated from the shield and in case speciments are prepared

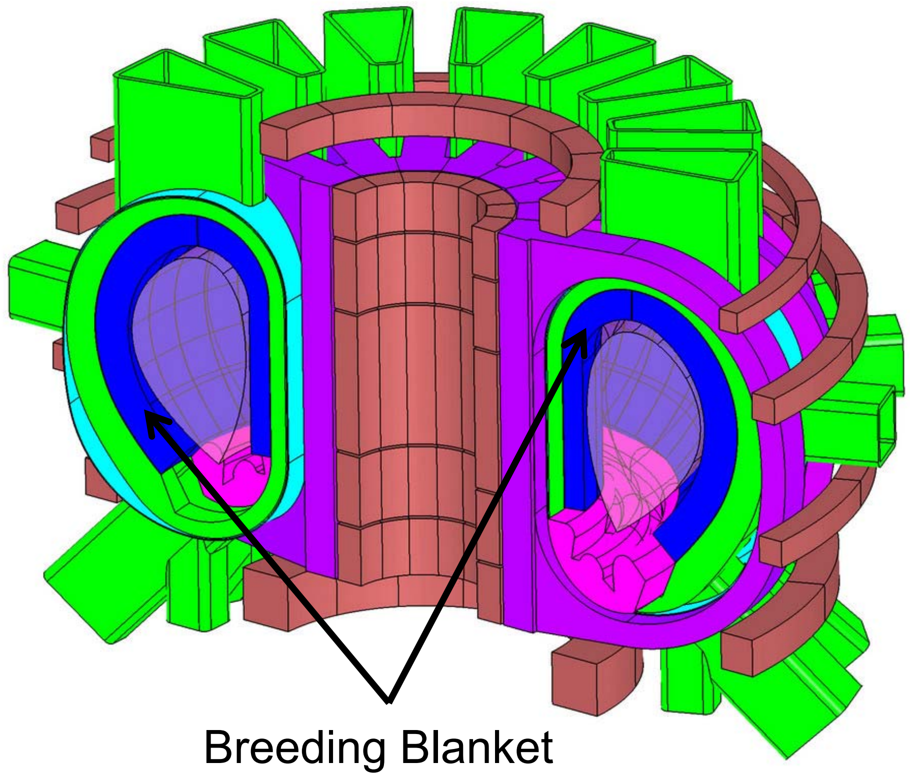
T. Ilkei (RMKI)

The PPPT DEMO

The EU fusion roadmap Horizon 2020 foresees a DEMO to built around 2050. The programme is implemented by the EUROfusion Consortium.

Major requirements of the DEMO plant are:

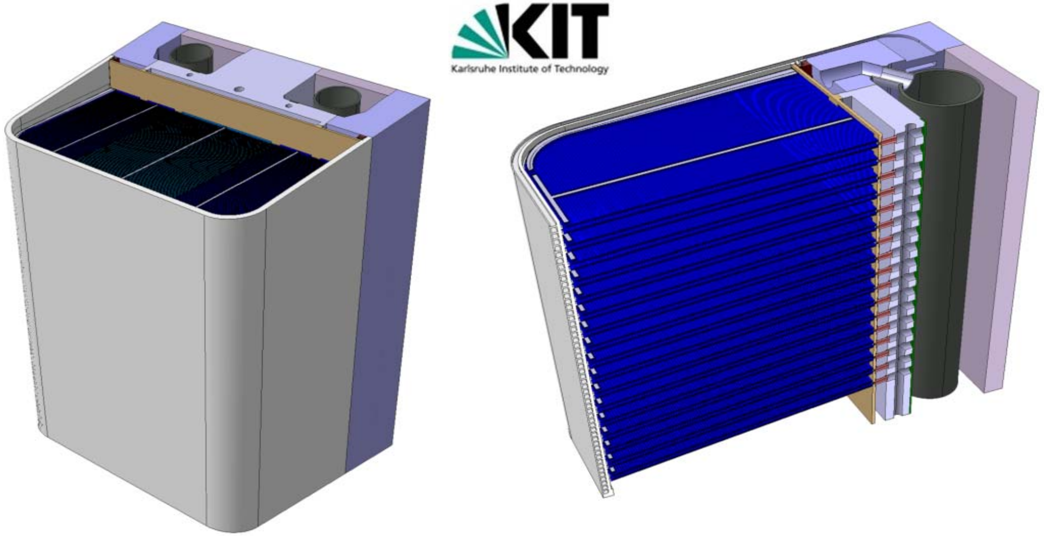
- To breed all the T necessary for the thermonuclear reaction;
- To deliver several hundreds of MW of electrical power in net



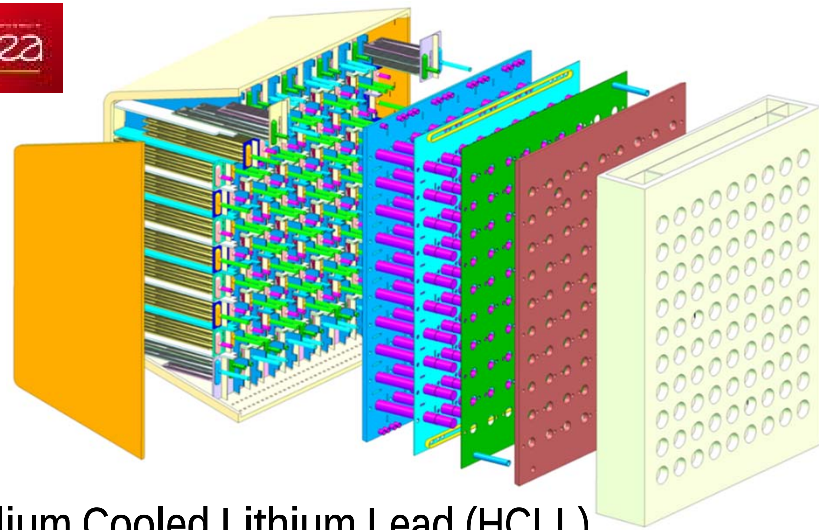
H. Hurzlmeier, EFDA_D_2M9JM7 v2.0 (2014).

The BB Project is part of the EUROfusion consortium and started in May 2014. It is responsible for the development of blanket concepts suitable for the EU DEMO concept as described in the EU Roadmap. The study take profit from a large tradition of EU (and outside EU) blanket studies: e.g DEMONET (1994-1999) and PPCS (2000-2004) and from the EU TBM Programme in ITER. See **[Fede16]** and **[Bocc16]**.

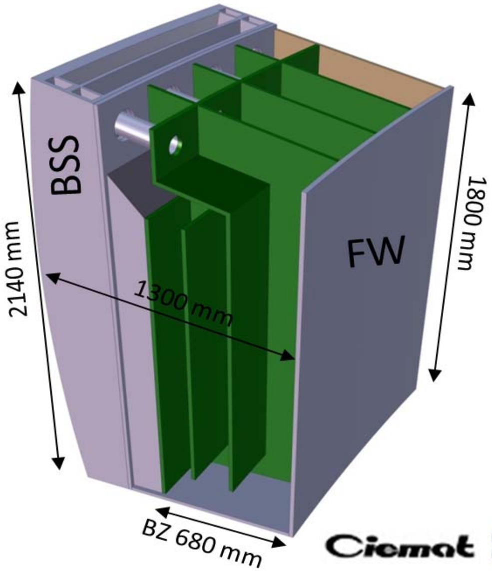
Blanket concepts considered in PPP&T



Helium Cooled Pebble Bed (HCPB)

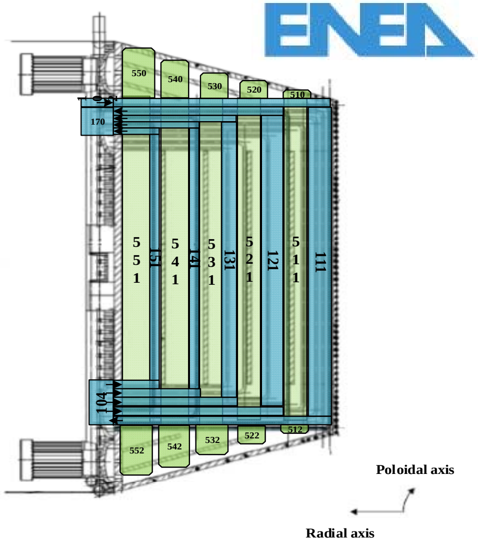


Helium Cooled Lithium Lead (HCLL)



Concept	Breeder/ Multiplier	Coolant	T-Extraction	Other
HCPB	Ceramic Breeder / Beryllium	Helium	He low pressure purging	(Permeation control)
HCLL	PbLi	Helium	PbLi slow recirculation	(Corrosion-permeation barrier)
WCLL	PbLi	Water	PbLi slow recirculation	(Corrosion-permeation barrier)
DCLL	PbLi	Helium PbLi	PbLi fast recirculation	FCI of Allumina or SiC/SiC$_f$

Dual Coolant Lithium Lead (DCLL)



Water Coolant Lithium Lead (WCLL)

Milestones and deliverables for the WPBB

Karlsruhe Institute of Technology

Deliverables for each BB concept:

- Design Description Document (Blanket Segment + Aux)
- CAD Models, PFDs, etc.
- Experimental R&D test reports
- Manufacturing Feasibility Assessment Report
- Risk Analysis Report

Deliverables for the Project:

- Project Management Plan (Primavera)
- Blanket system SRD and ICDs
- Project Risk Assesment

2014 ----|------ 2015 -----|---- 2016----|------ 2017------|------ 2018-----|-------- 2019 --------|------ 2020---

MP1: Establishment of project Requirements, blanket configuration and blanket Development Plan. Preliminary DDD and CAD released.

MP2: System Requirement Review

MP8: Definition of Blanket selection criteria

MP3: Selected technology R&D and testing to enable down-selection most promising design option for each breeding blanket concept. Consolidated DDD and CAD released.

MP4: Manufacturing feasibility demonstration of key technologies, including performance tests, structural integrity reports, etc.

MP5: Finalise breeding blanket concept design. Conceptual design DDD and CAD released.

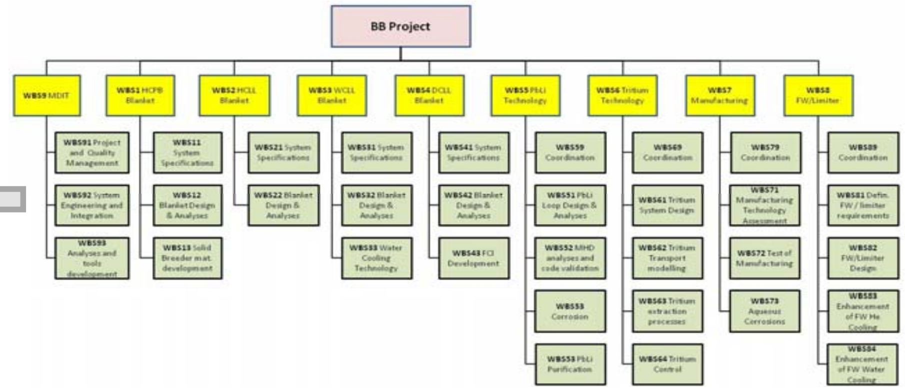
MP6: Documentation for the Plant Review meeting issued

MP7: End of the Plant Review Meeting

Breeding Blanket Project: WBS and PMP

WBS in the Breeding Blanket Project:

- **WP1: HCPB Blanket Design including Ceramic Breeder and Beryllium development and characterisation.**
- **WP2: HCLL Blanket Design**
- **WP3: WCLL Blanket Design including Water cooling technology**
- **WP4: DCLL Blanket Design including Flow Channel Insert development**
- **WP5: PbLi technology for HCLL, WCLL and DCLL.**
- **WP6: Tritium Technology**
- **WP7: Manufacturing Technology**
- **WP8: FW and Limiter technology**
- **WP9: System Engineering, System Modelling, Neutronic and EM analysis development.**

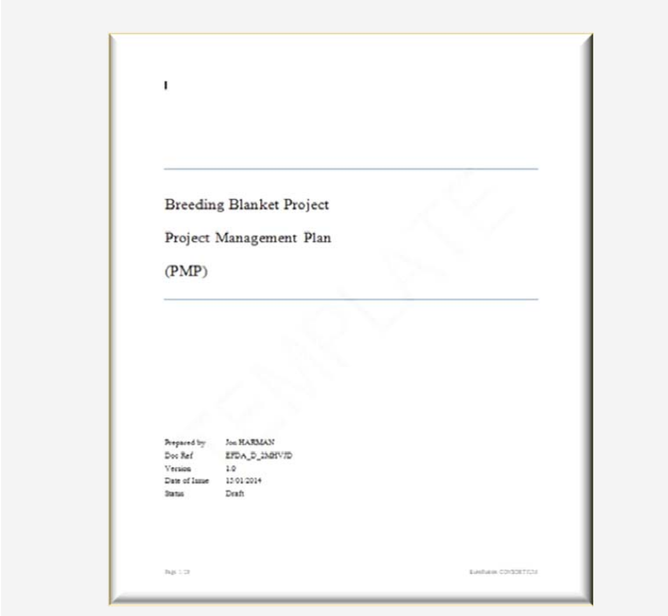


Resources:

- **In 2014-2018 about 260 ppy and 19 M€.**

EU Research Unit involved:

- **CCFE, CEA, CIEMAT, ENEA, IPP.CR, KIT, Wigner RCP.**

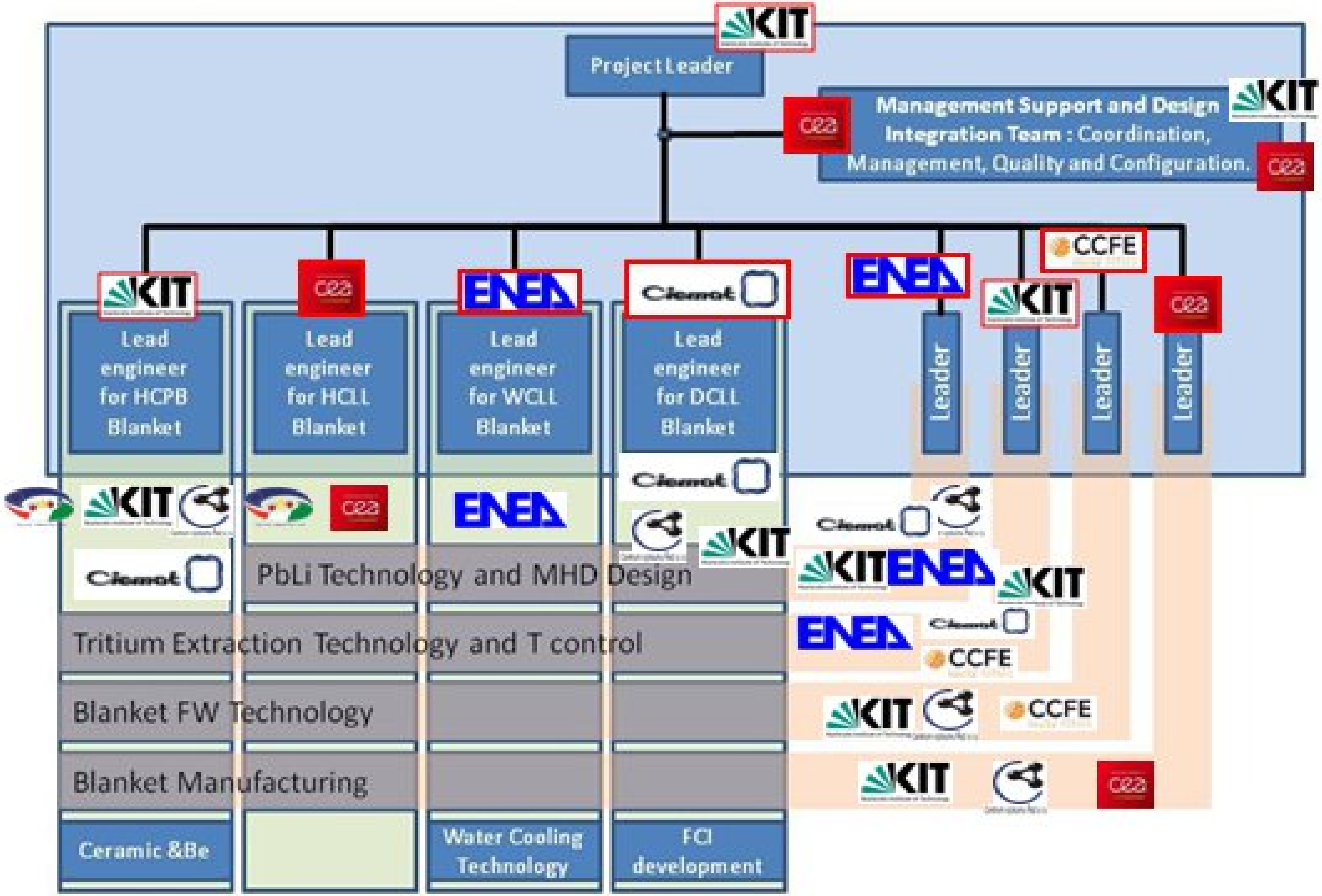

Breeding Blanket Project

Project Management Plan

(PMP)

Project Management Plan: https://idm.euro-fusion.org/?uid=2MDAMJ&version=v4.0&action=get_document

WPBB Organisation Chart



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