Magnetic confinement fusion: technological challenges

G.A. Spagnuolo

Acknowledge contributions: G. Federici, C. Bachmann, C. Luongo, M. Siccinio, M.L. Richiusa, F. Hernandez, J.A. Cordero, D. Marzullo

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Purpose/Outline

Purpose:
(1) provide a brief overview of tokamak fusion reactor
(2) describe the main considerations that affect the design of DEMO or any fusion nuclear reactor

- ✓ **The context**
- ✓ **Wall protection and plasma-facing components**
- ✓ **Vacuum Vessel and Magnet system**
- ✓ **Remote Maintenance of In-Vessel components**
- ✓ **Confinement/penetration/radiation protection**
- ✓ **Integration**
- ✓ **Conclusion**

In Europe, DEMO is considered to be the nearest-term reactor design to follow ITER capable of producing electricity, operating with a closed fuel-cycle and to be a facilitating machine towards a commercial reactor.

The context

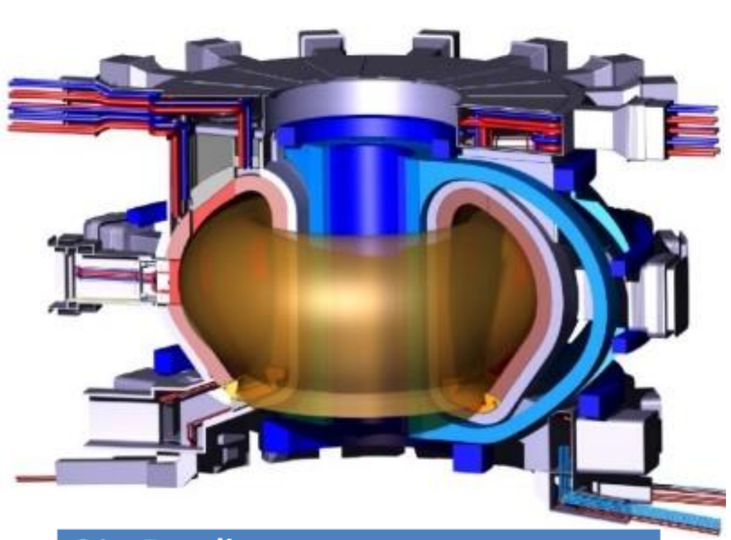
EU-DEMO: Goal

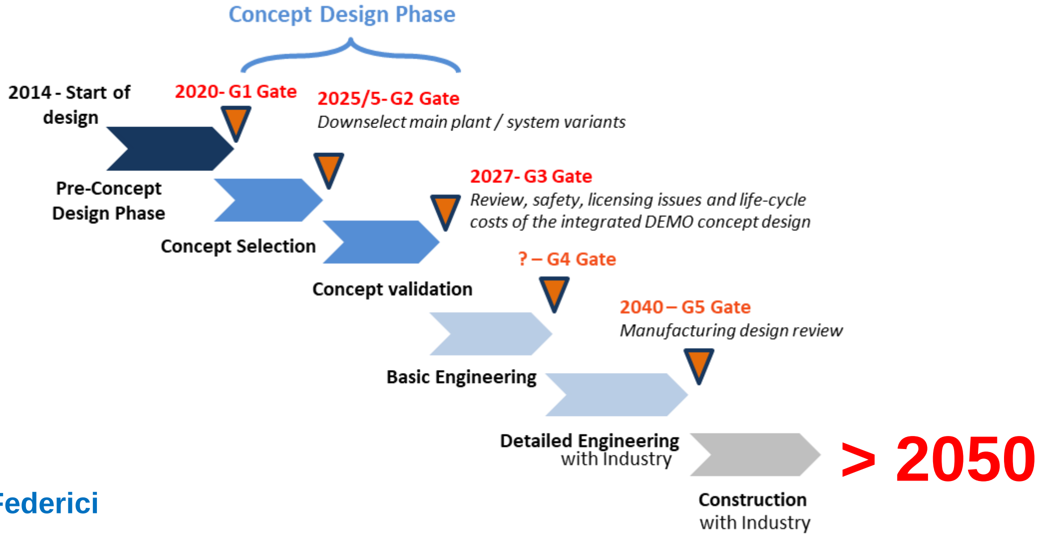
- The aim is to demonstrate all the functions necessary for the operation of a reactor.
  - Production of *net* electricity for a few hundred MW (currently 500).
  - Pulse duration > 1 hr.
  - Ability to produce all the tritium required for reactions in situ

- Technology qualification under reactor-relevant conditions (high neutron fluence: 50-70 dpa (ITER 1-3 dpa))

- Demonstrate high degree of repeatability of discharges and reliability of technology choices.

- **DEMO is not a commercial reactor:** the electricity produced does not have to be economically competitive



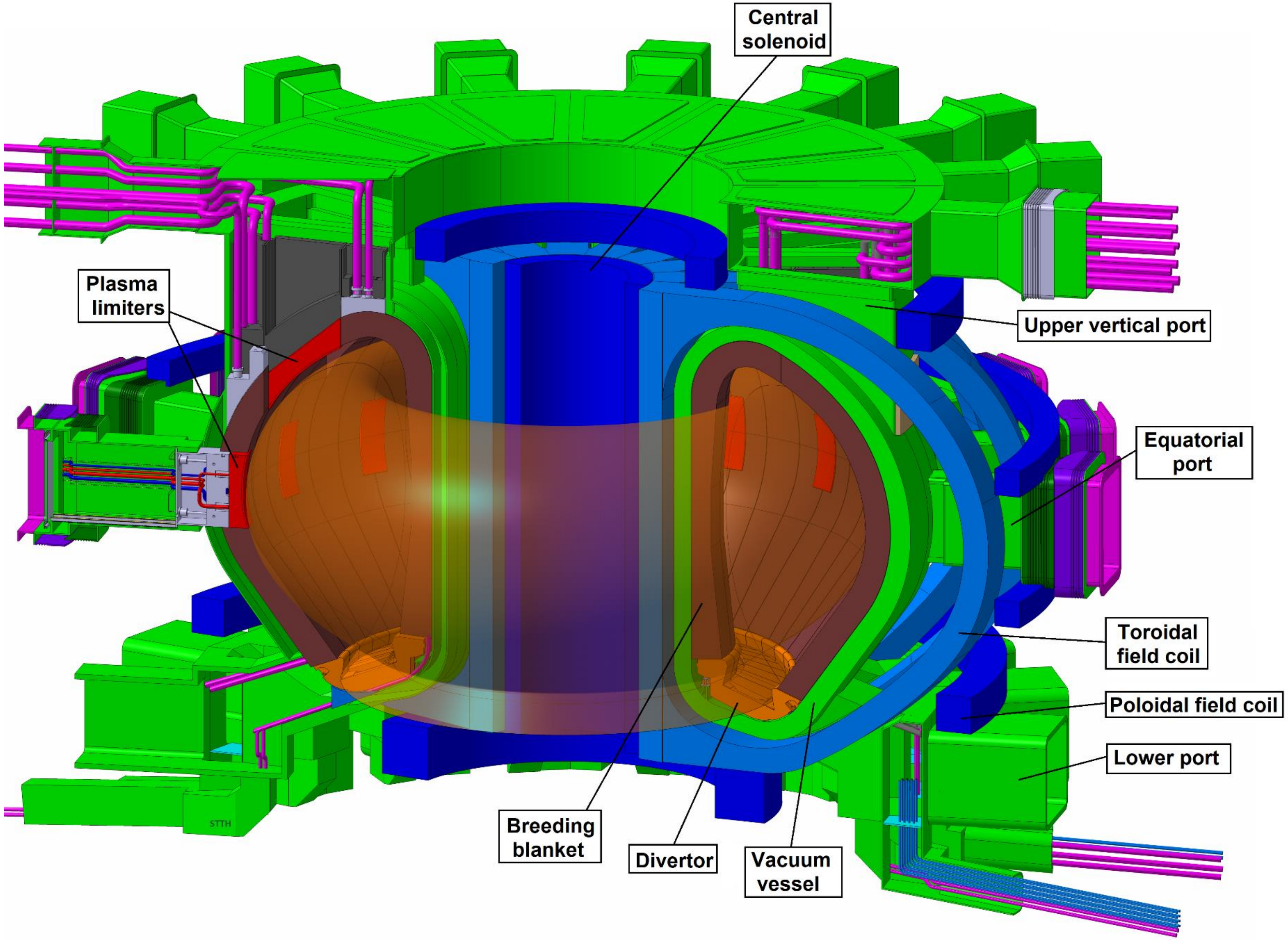


Courtesy of G. Federici

G1 - Baseline	
R [m], A	9.00, 3.1
B_0 [T]	5.9
q_{95}	3.89
κ_{95} , δ_{95}	1.65, 0.33
I_p [MA]	17.75
P_{fus} [MW]	2000
P_{el} [MW]	500
H_{98}	0.98
β_N [% mT/MA], $Fus.Gain\ Q$	2.5, >40
$P_{sep}B/q_{95}AR$ [MW T /m]	9.2
Pulse length [sec]	7200
NWL (MW/m^2)	~1
n-fluence: 20 dpa 1st BB/ 50 dpa 2nd BB	70 dpa

Nuclear performance

EU-DEMO layout



Wall protection and plasma-facing components

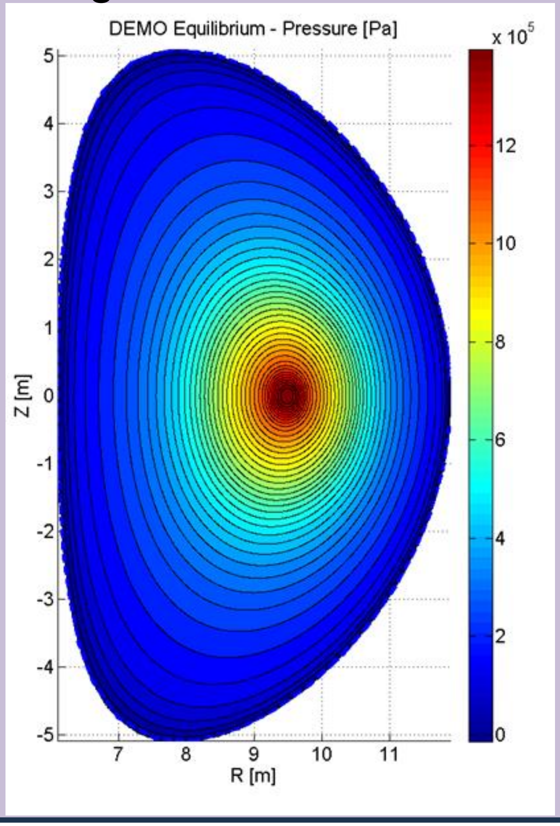
Plasma

Fusion plasma physics deals, broadly speaking, with two high-level topics:

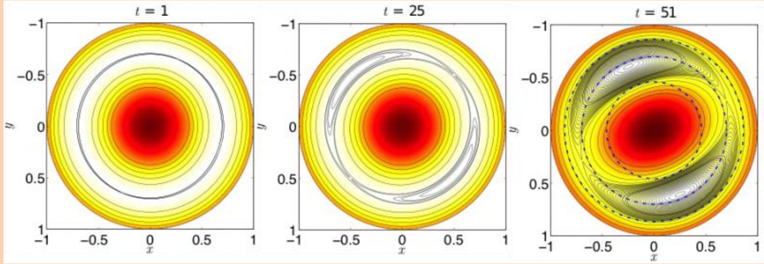
Courtesy of M. Siccinio

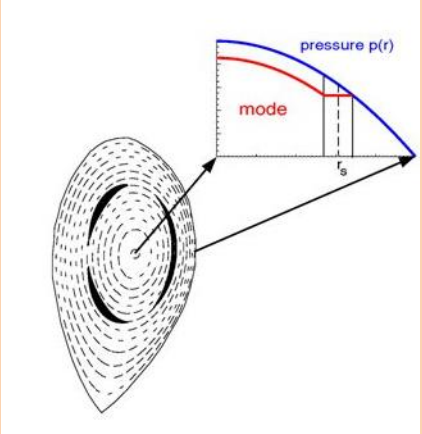
1. Maximise the performance in terms of fusion power production (plasma confinement)

The plasma must be **hot and dense in the centre**, but cold and less dense at the edge.

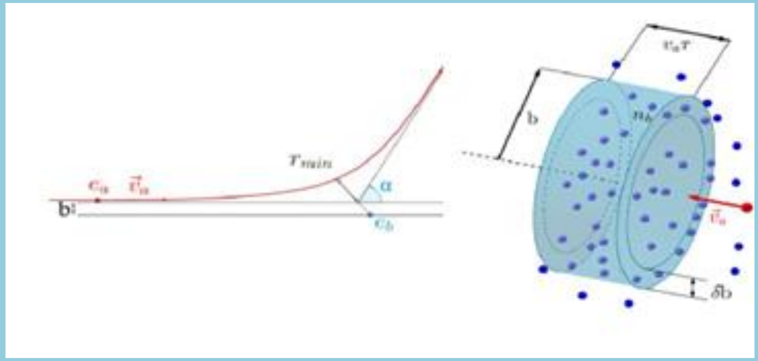


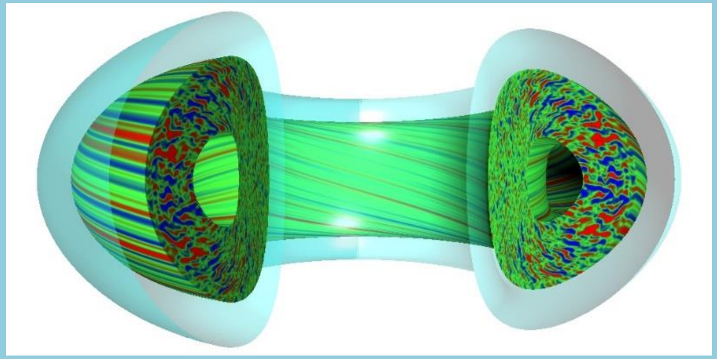
Avoid **plasma instabilities** which can deteriorate the machine confinement – like e.g. Sawteeth or NTMs





Understanding the physics of **magnetic confinement and plasma transport** (collisional and turbulent)





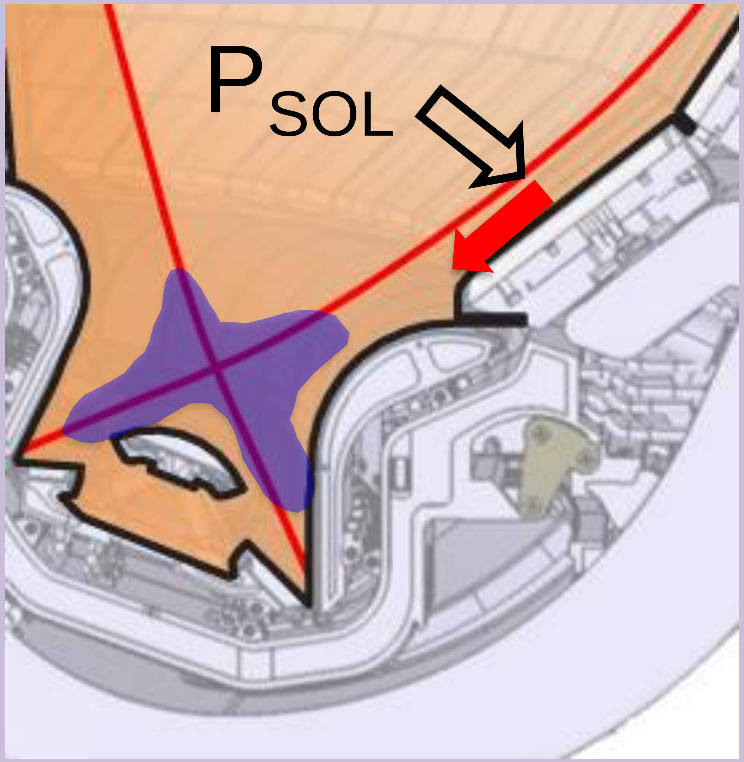
Plasma

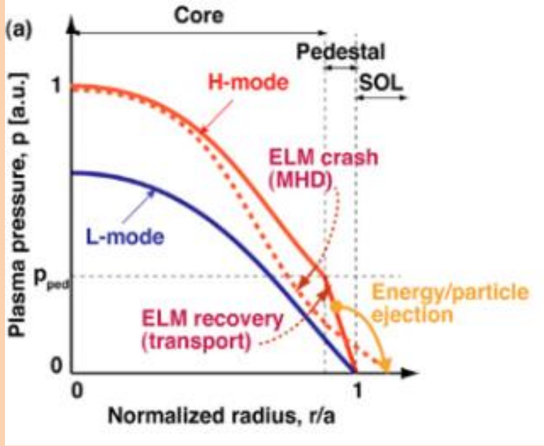
Fusion plasma physics deals, broadly speaking, with two high-level topics:

Courtesy of M. Siccinio

2. Ensuring the integrity of PFC during normal operation and during off-normal transients

The divertor is where the exhaust plasma is collected. In a reactor, heat is concentrated in a small area, with very high fluxes.

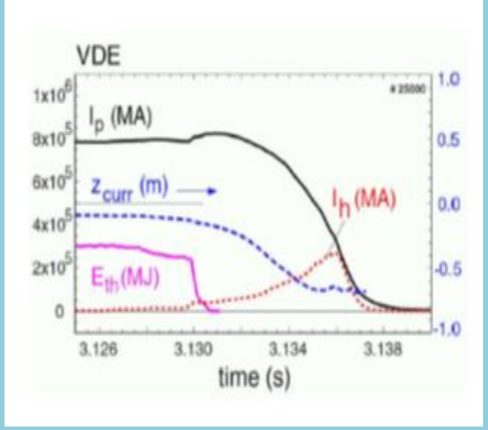


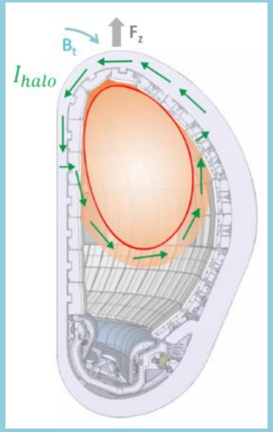


ELMs (Edge Localised Modes) are periodic relaxation (~1 Hz) of the plasma periphery, and lead to short but intense heat loads during plasma discharges

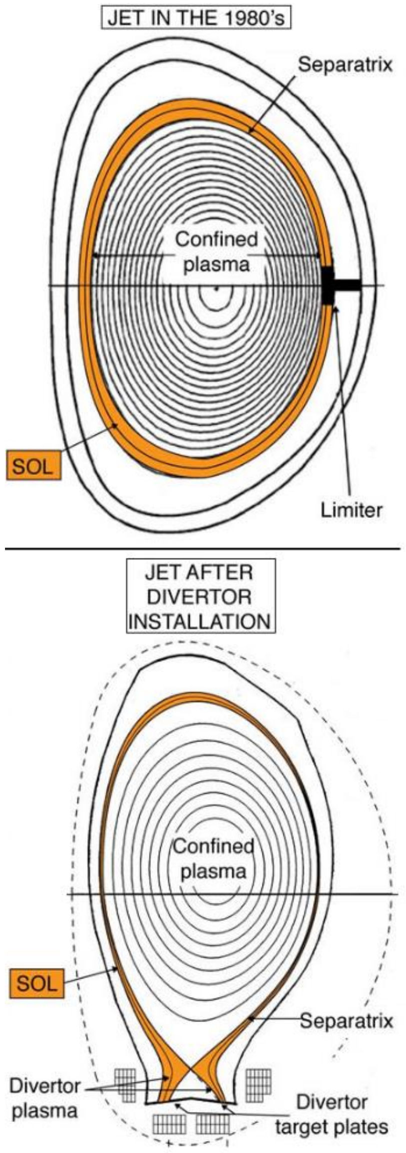
Disruptions are off-normal losses of control, with subsequent impact on the first wall.

They lead to huge heat load and forces in the structures.



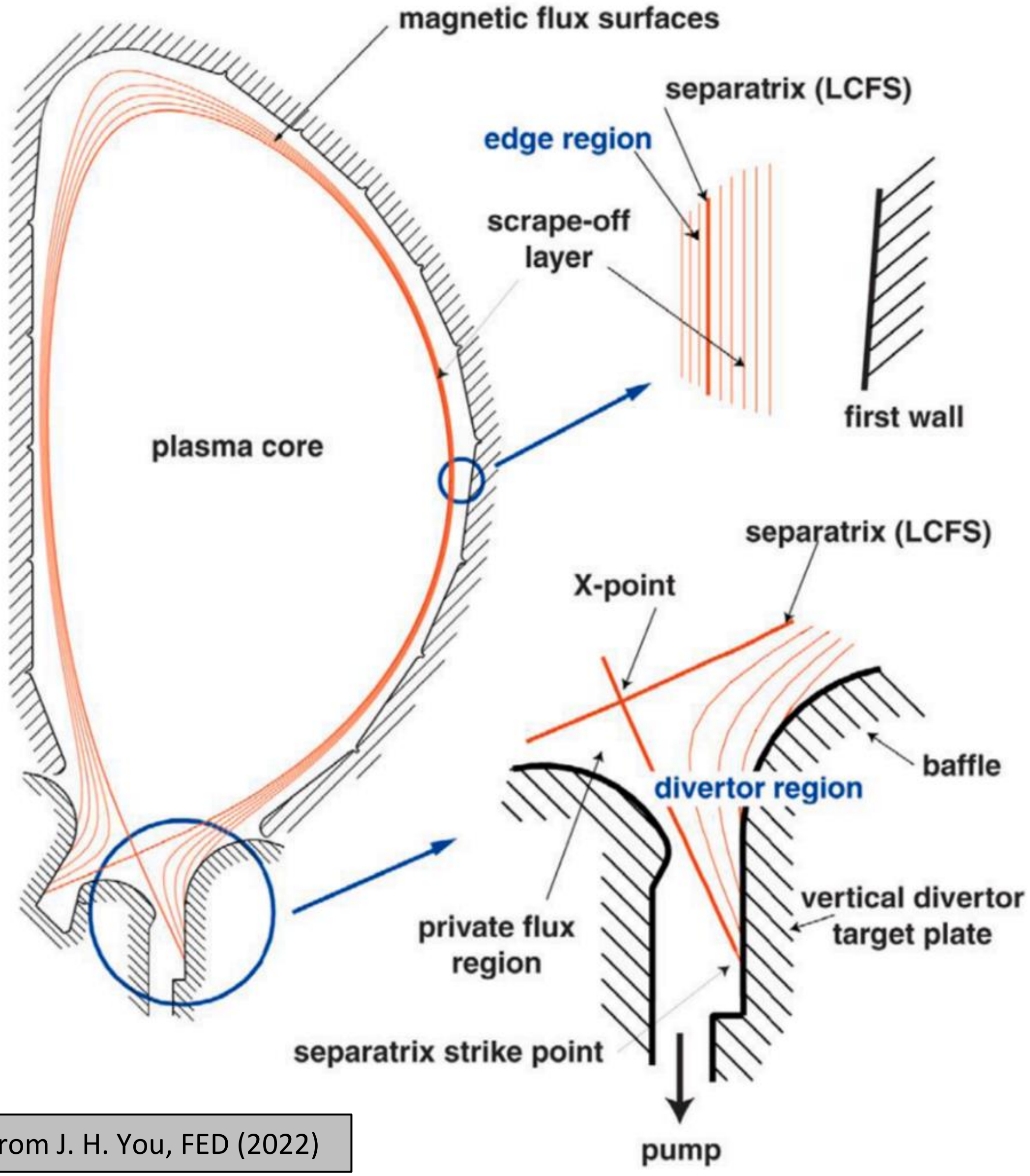


Power exhaust



Courtesy of C. Bachmann

- Charged particles escape from the plasma confinement (mostly on the outboard side) and enter the so-call scrape-off layer: SOL
- They travel along the poloidal magnetic flux lines until they impact on the part of the wall intersecting the SOL generating heat and causing erosion → the wall is coated with either Beryllium (low atomic number, does not severely impact on the plasma performance) or Tungsten (high resistance to erosion) or Carbon (not applicable in DEMO due to high T retention).
- The SOL is a thin layer of few centimetres → impact on the wall is concentrated in small areas.
- Divertor heat loads are a size driver for the machine!

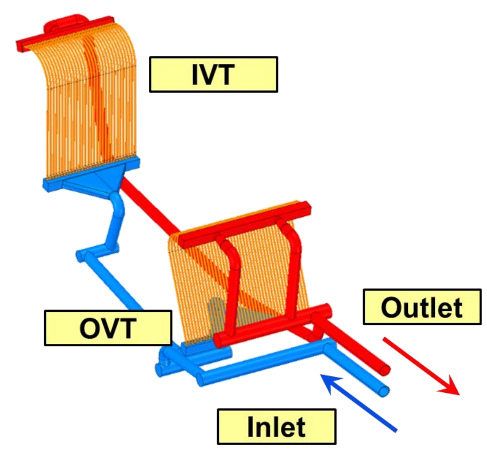


from J. H. You, FED (2022)

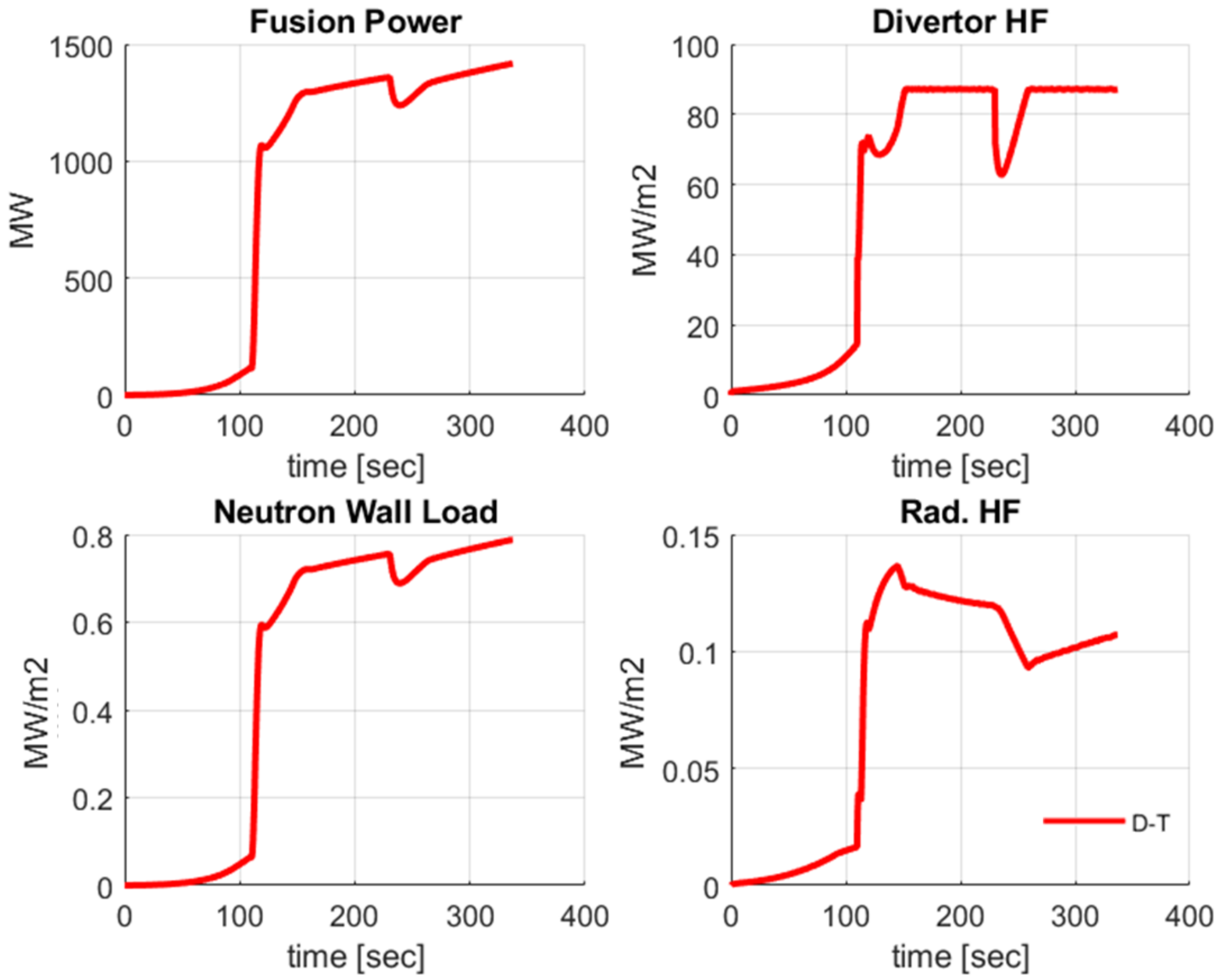
Divertor challenges: vertical target operation

The fusion power spike during RU at L-H transition is extremely nasty from many points of view.

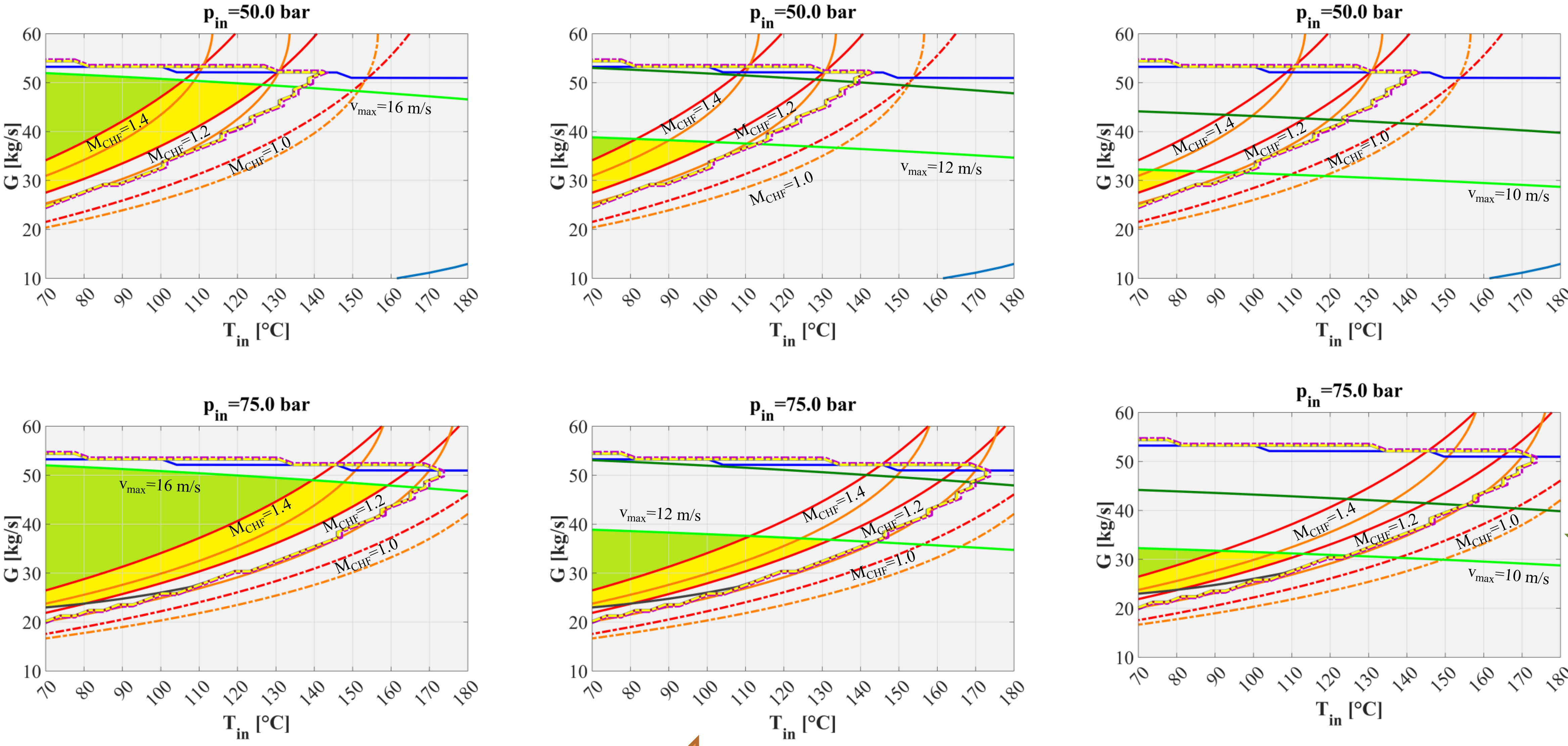
- ✓ Divertor,
- ✓ Vertical control,
- ✓ BB
- ✓ BOP
- ✓ …



Courtesy of M. Siccinio and D. Marzullo



Divertor challenges: vertical target operation



Coolant Inlet Pressure Increase

Max Velocity Increase

Courtesy of D. Marzullo

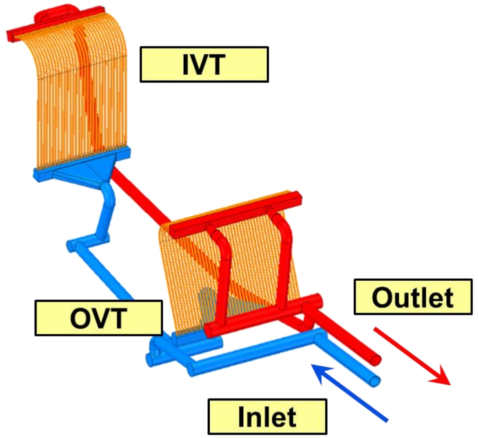
20 MW/m²

Divertor challenges: vertical target operation

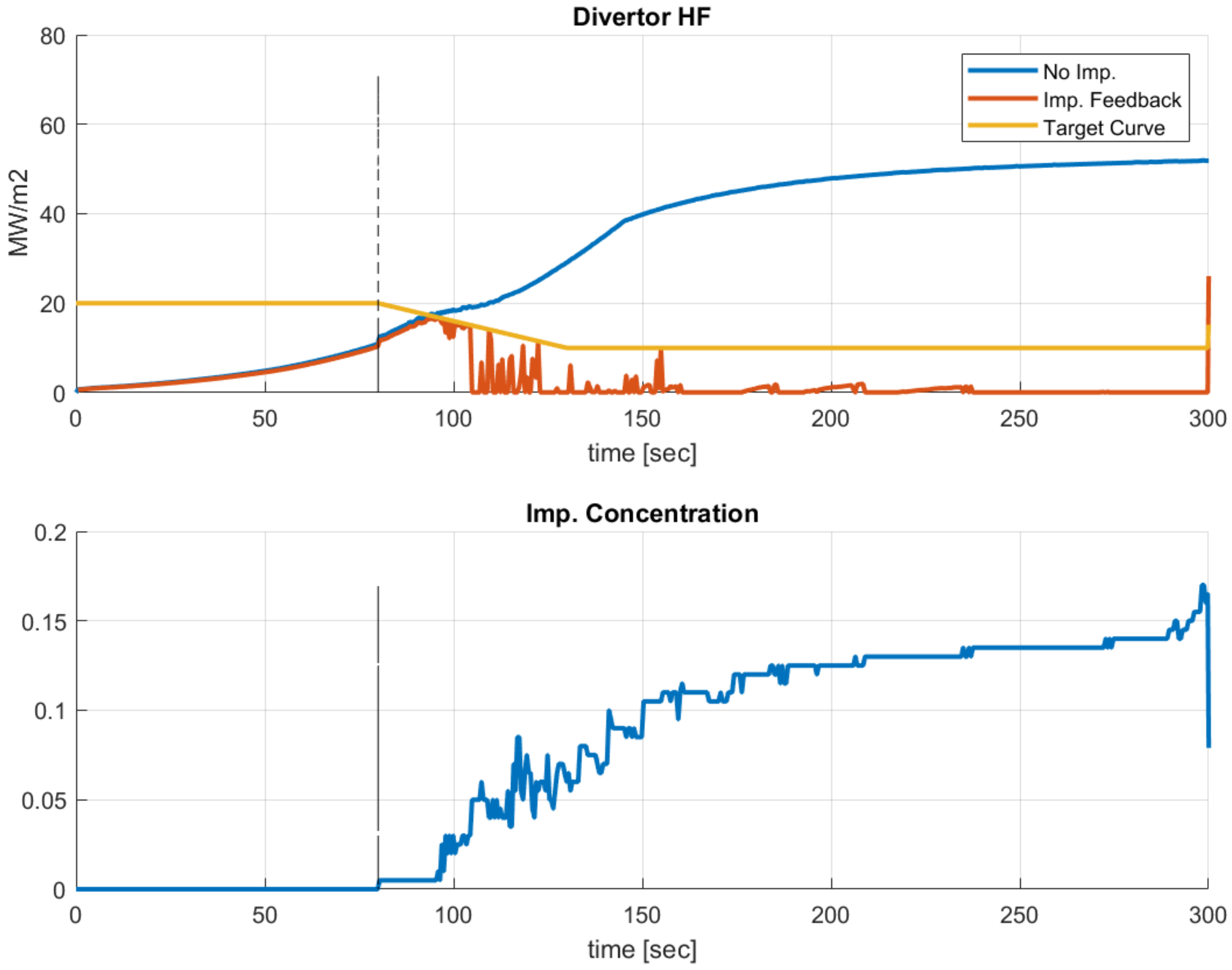
The fusion power spike during RU at L-H transition is extremely nasty from many points of view.

- ✓ Divertor,
- ✓ Vertical control,
- ✓ BOP
- ✓ …

Idea: dilute the plasma with He and reduce the spike. Then progressively pump He away.



Courtesy of M. Siccinio and D. Marzullo



Preliminary case – still to be optimised

Disruptions

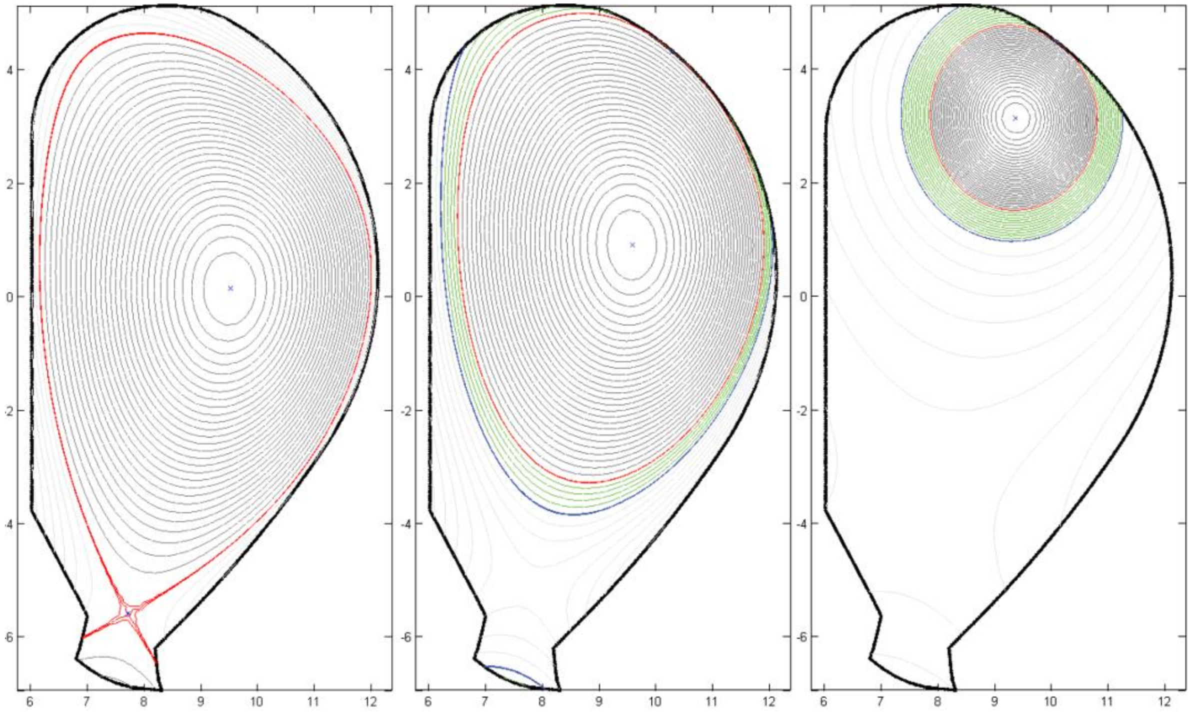
Plasma disruptions occur for various reasons, e.g. impurity influx, loss of confinement, error in the active control system.
Consequences:

- Plasma current (up to 15 MA in ITER) drops to zero within tens of ms → **large electromagnetic forces**
- Plasma will impact on the wall causing **huge heat impact loads** in the affected areas (GJ/m²) → evaporation and melting of the armour surface layers and potential damage of plasma facing components.
- In stellarators these problems are much reduced (no plasma current).

Induced currents in ITER blanket:

Courtesy of C. Bachmann

Vertical displacement event in DEMO:



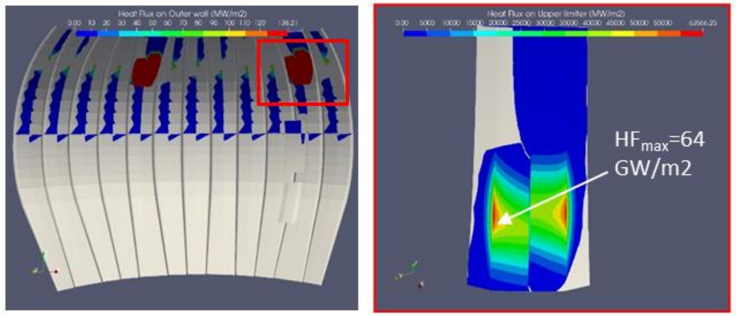
Heat loads on limiters

HF_{max} during Normal Operation

Components	HF_{max} [MW/m²]
FW	0.4
OML	0.5
OLL	0.1
IML	0.0
UL	0.8

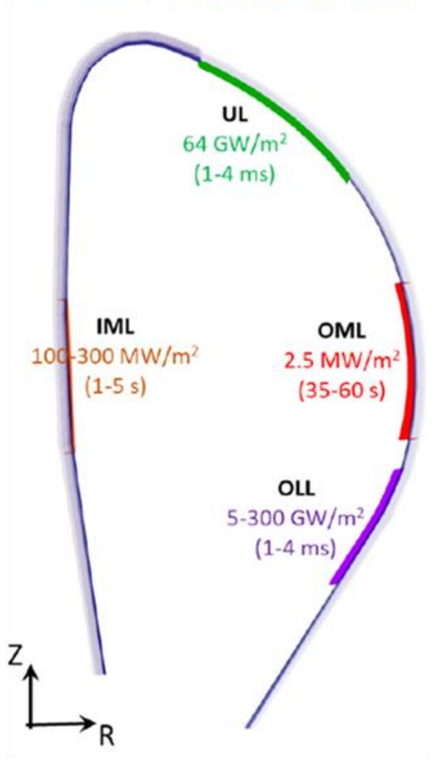
PFW Heat Loads

Mainly VDE heat load-driven design.

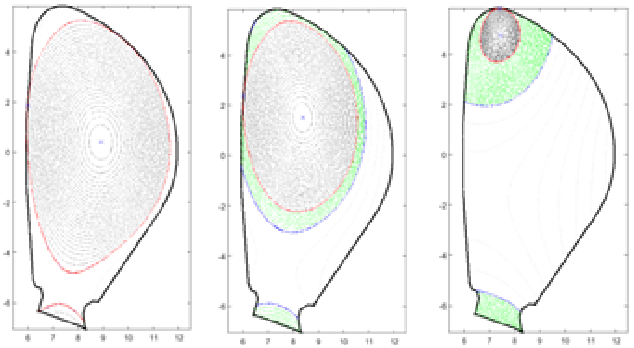


UL HF_{max} during a UVDE Thermal Quench.

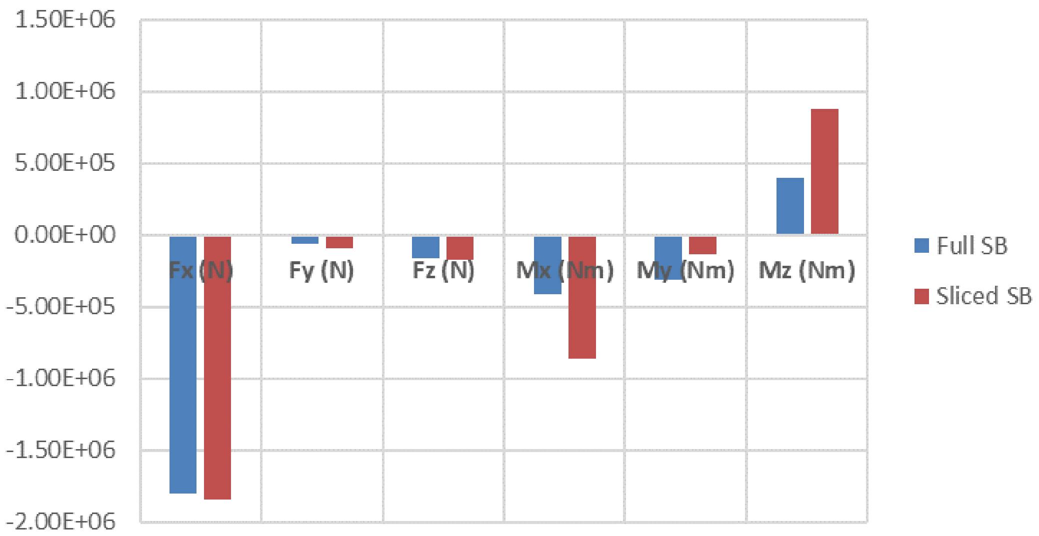
HF_{max} during UVDE, RU, DVDE, H-L transition



CarMa0NL simulation of plasma evolution during UVDE



Peak value on upper limiter @ TQ (1.053s)



F. Maviglia et al., Integrated design strategy for EU-DEMO first wall protection from plasma transients, Fusion Engineering and Design, 2022 https://doi.org/10.1016/j.fusengdes.2022.113067

Courtesy of M. L. Richiusa

Breeding blanket

<u>Why do we need tritium self-sufficiency?</u>

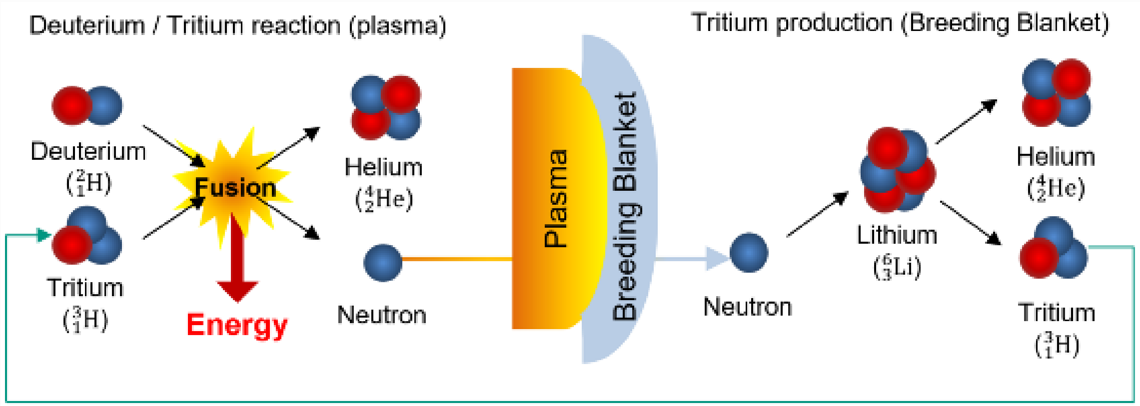
<u>Consumption:</u> P_{fus} ~2000 MW, 30% availability (DEMO)→ 33 kg T/year

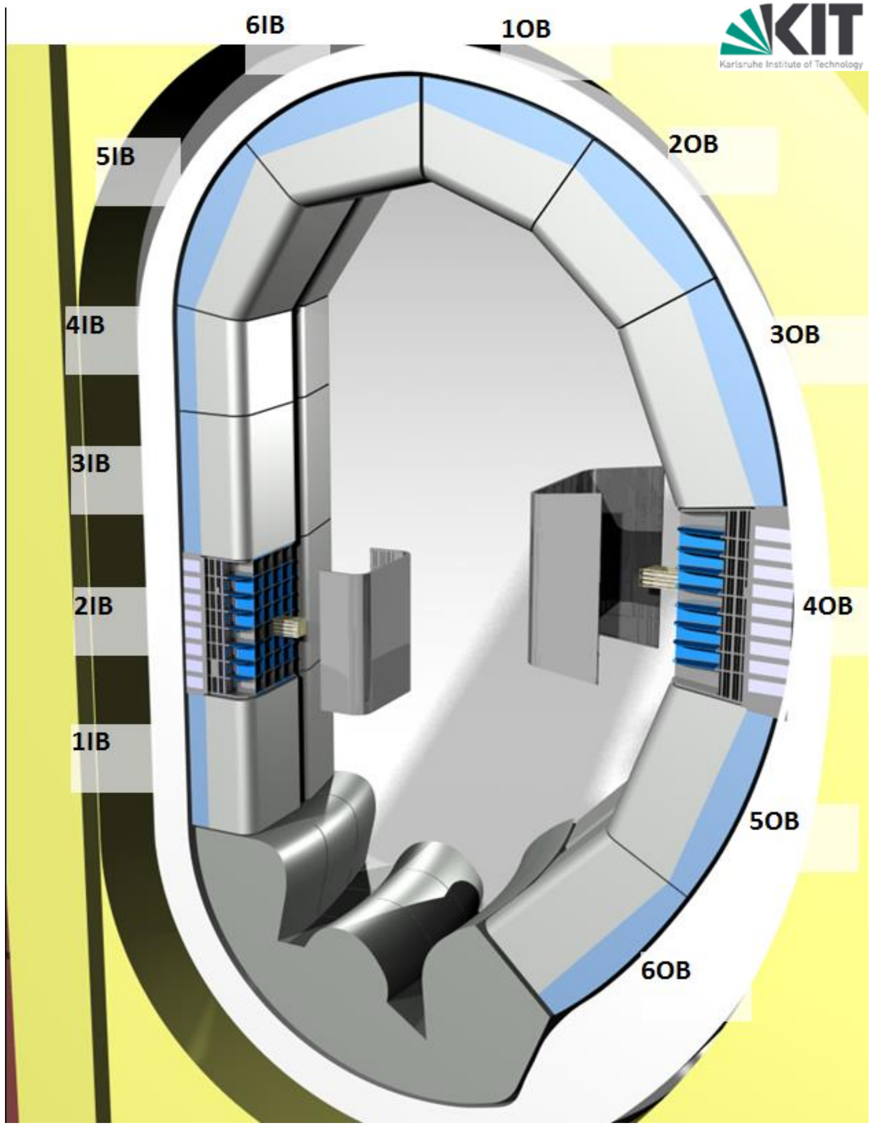
<u>Cost of Tritium:</u> ~25 M€/kg

<u>Breeding blanket overview :</u>

- Banana-shaped boxed made of Eurofer → BB segments
- Box is filled with breeder (^{6}Li) and neutron multiplier materials (Be, Pb): n + ^{6}Li → He + T + MeV
- Active cooling loop to remove the heat
- Active tritium extraction loop to remove T

→ ~85% of the plasma must be covered by the breeding blanket.



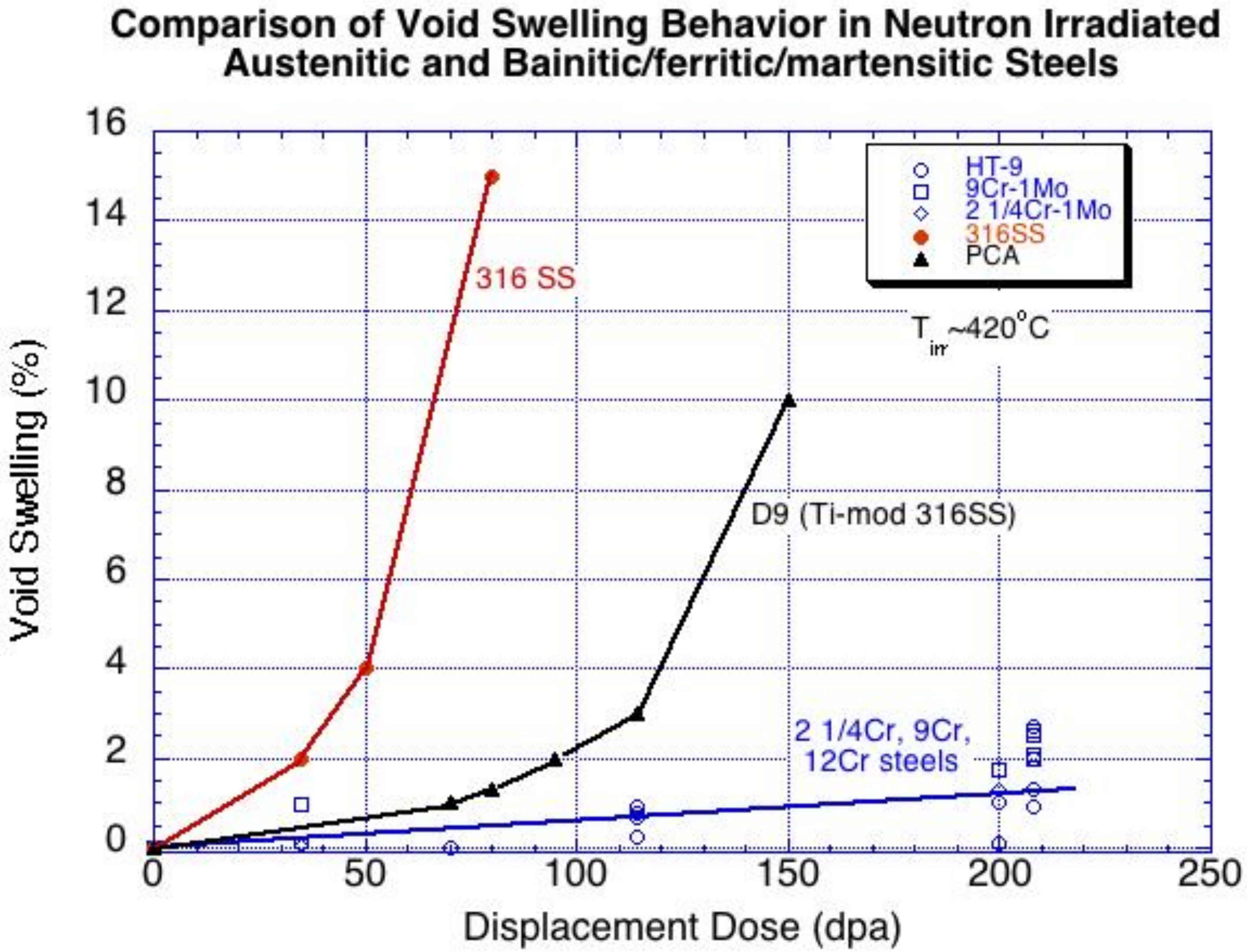


Courtesy of C. Bachmann

Irradiation resistant material

Lifetime neutron fluence:

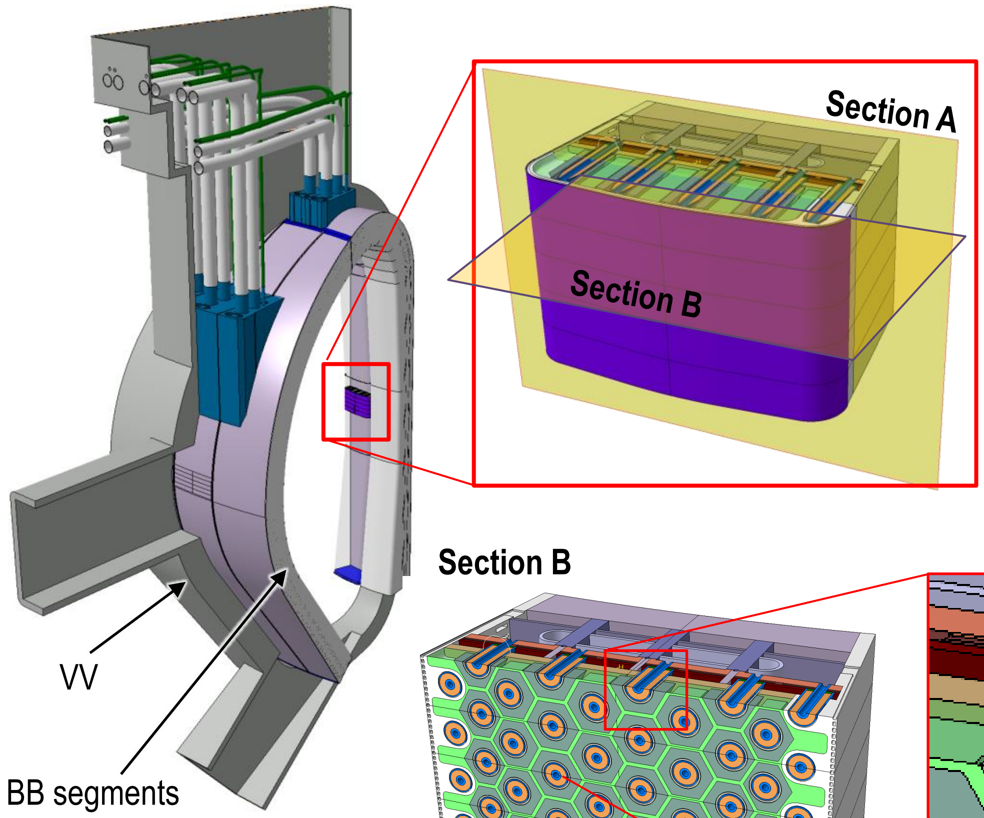
- ITER: ~1-3 dpa,
- DEMO: ~70 dpa,
- Fusion power plant: ~150-250 dpa

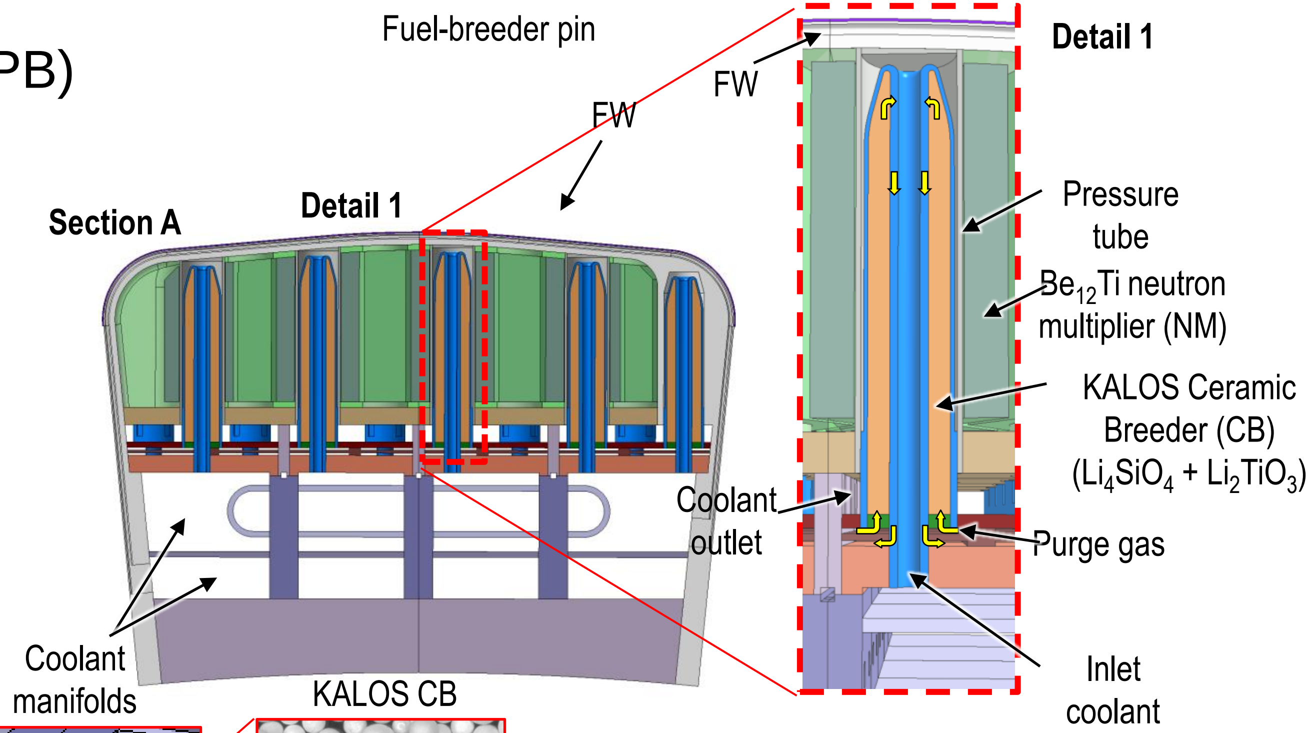


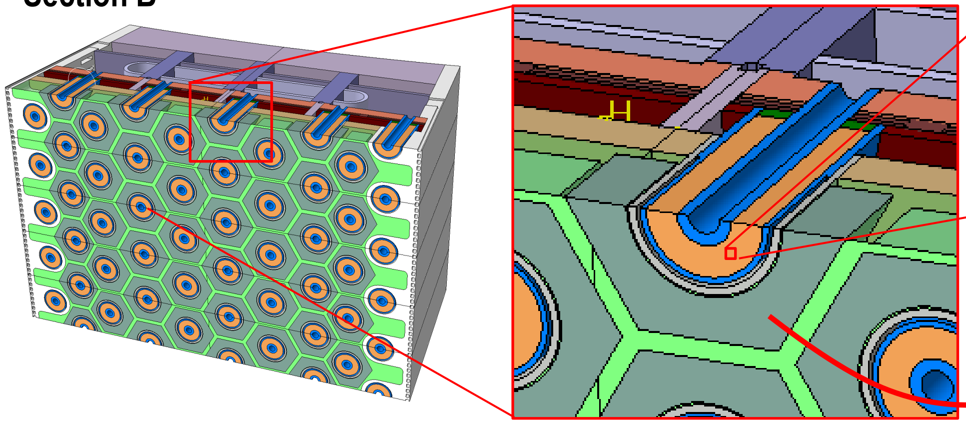
Courtesy of C. Bachmann

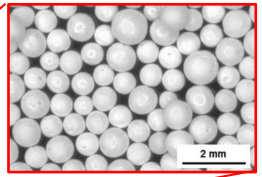
Current Design of HCPB BB

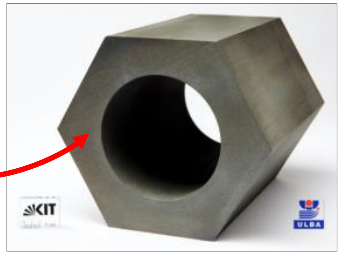
- Helium Cooled Pebble Bed Concept (HCPB)











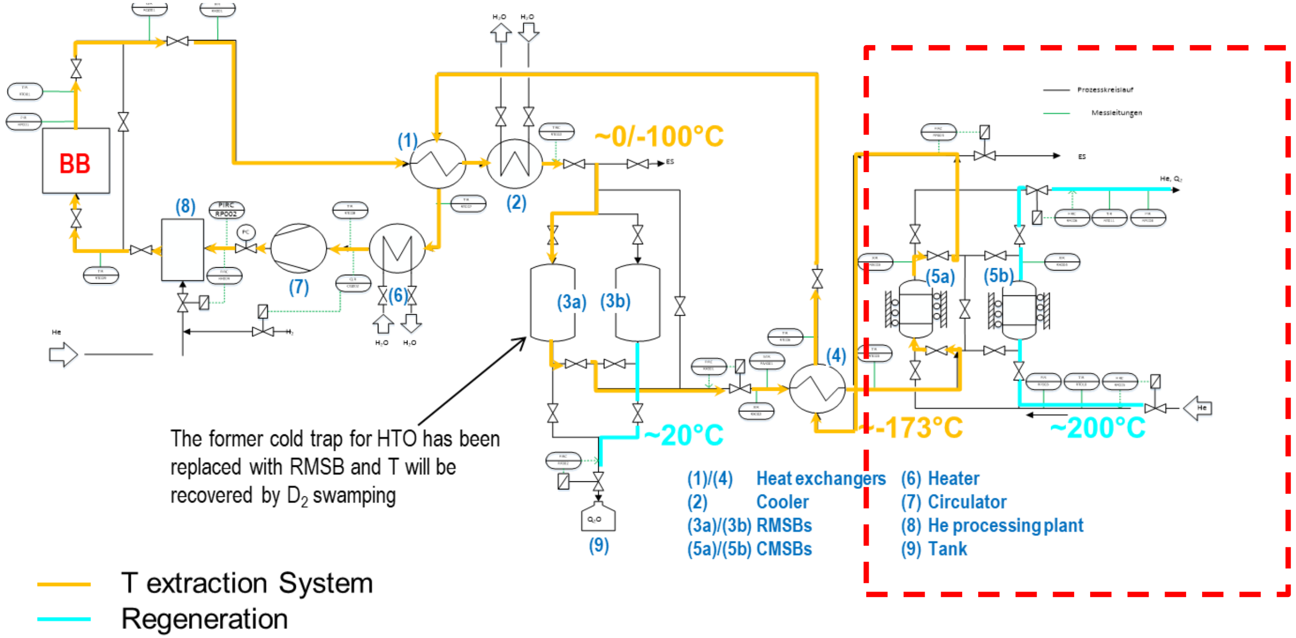
- He 80 bar, T_{in} = 300°C, T_{out} = 520°C
- ***T*** breeding: KALOS Ceramic Breeder
- ***n*** multiplier: $Be_{12}Ti$ blocks
- ***T*** extraction in BB through purge gas
- ***T*** removal from purge gas in TERS

Courtesy of F. Hernandez

F. Hernández, et al., Advancements in the Helium-Cooled Pebble Bed Breeding Blanket for the EU DEMO: Holistic Design Approach and Lessons Learned, Fusion Science and Technology, 75:5 (2018) 352-364.

Current Design of HCPB TER

- Non Evaporable Getter disks of ZrCo/ZAO alloy have been selected as second option.
  - ZrCo is a well-known material for tritium storage but it is also proposed as getter to extract small amounts of tritiated hydrogen isotopes from inert gas streams, such as the purge gas of the solid breeding blanket.
  - ZAO material is a new getter used also in industrial applications.
- Preliminary performance characterization with H_2 and D_2 has been carried out.

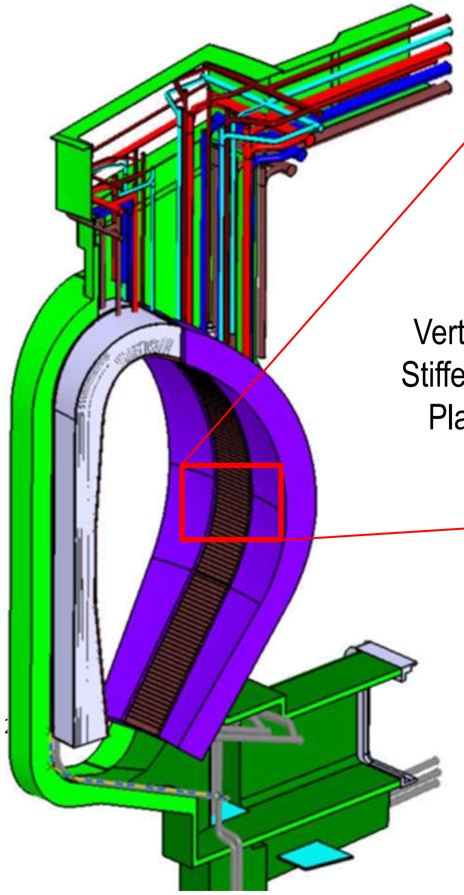


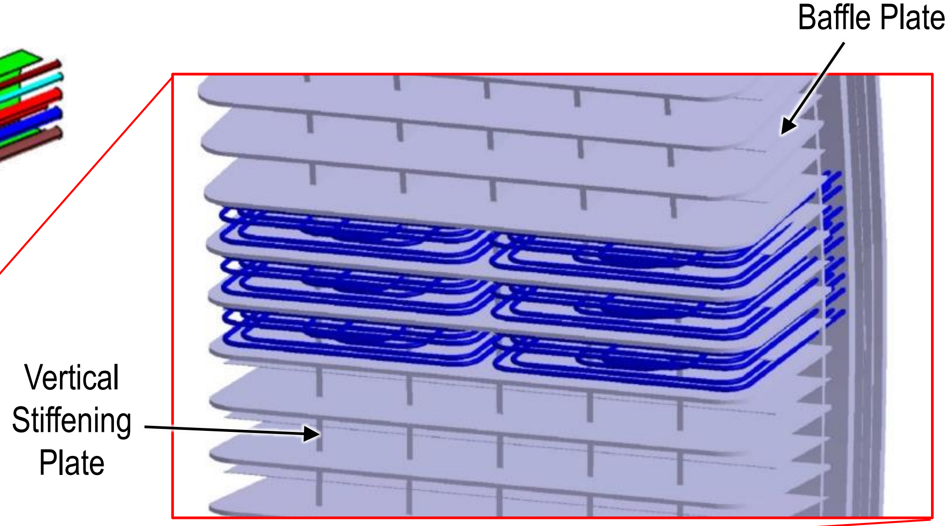
Laufende Nr.	1	2	3a	3b	4	5a	5b	6a	6b	7	8	9	10a	10b
Technische Daten	-	-	V=	V=	-	V=	V=	-	-	-	-	-	-	-
Betriebsdruck in bar	1,08	1,08	1,08	1,08	1,08	1,08	1,08	1,08	-	1,08	1,08	1,08	-	
Betriebstemperatur in °C	450	35	0 - (-100)	20	73	-173	250	-173	-		400 - 450	400 - 450	-	250
Durchfluss in m³h	10000	10000	10000	10000	10000	10000	1600	10000	-	10000	10000	10000	-	1600

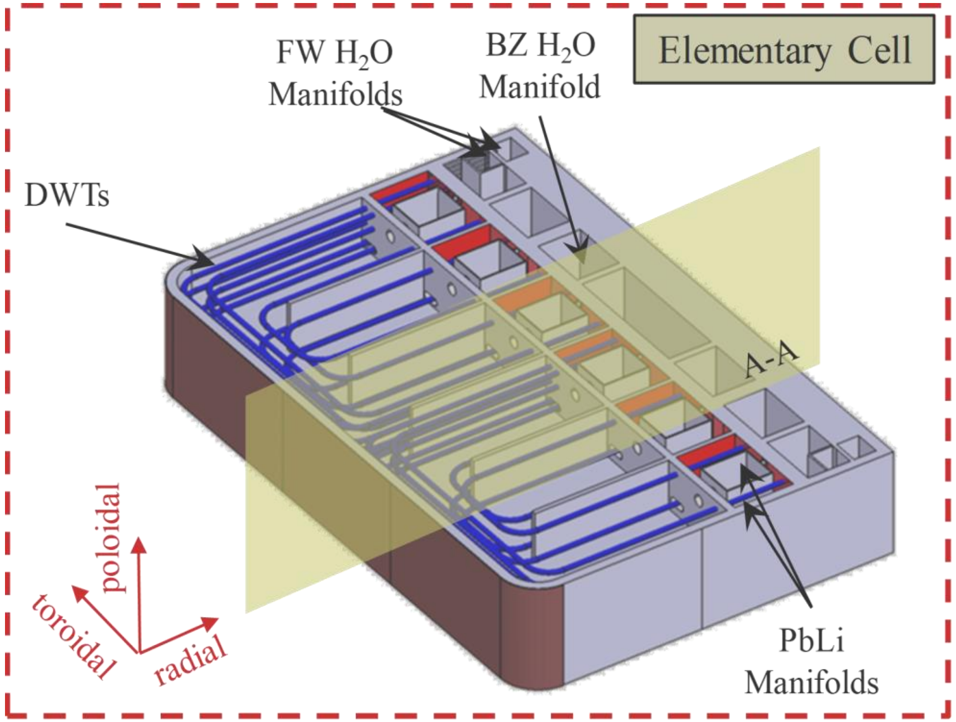
Current Design of WCLL BB

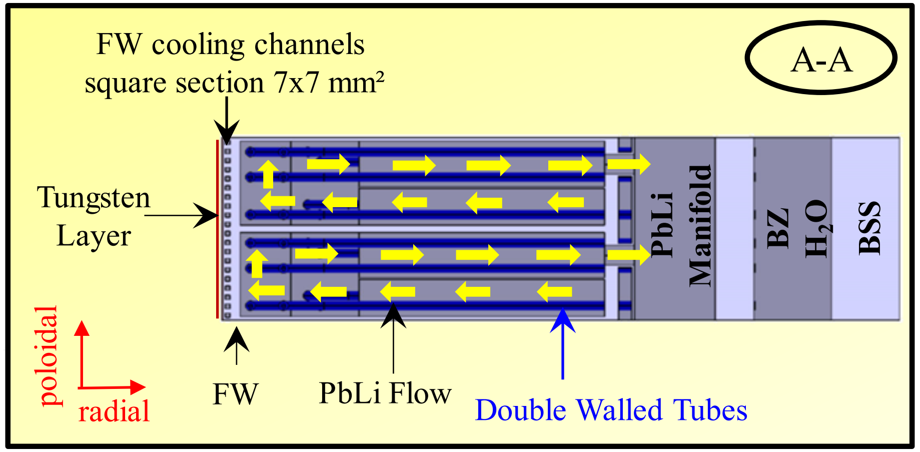
- Water Cooled Lead Lithium Concept (WCLL)

A. Del Nevo et al., Recent progress in developing a feasible and integrated conceptual design of the WCLL BB in EUROfusion project, Fusion Engineering and Design, 146 (2019) 1805-1809









- Water PWR: 155 bar, T_{in} = 295°C, T_{out} = 325°C
- ***T*** breeding and ***n*** multiplier: Pb-Li eutectic (liquid metal)
- ***T*** extraction and removal in TERS

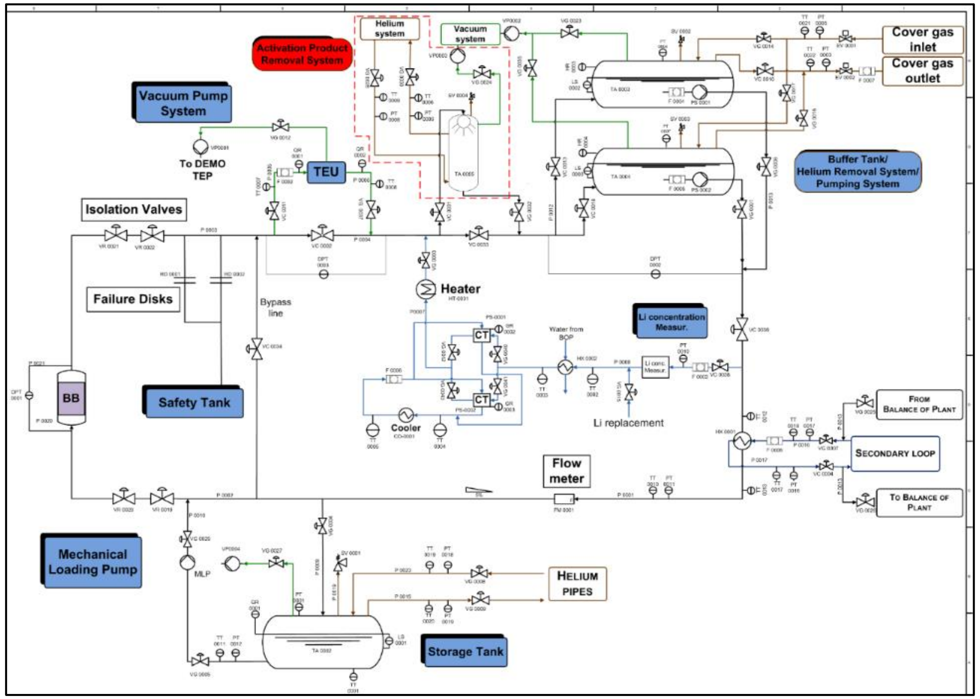
Italian National Agency for New Technologies, Energy and Sustainable Economic Development

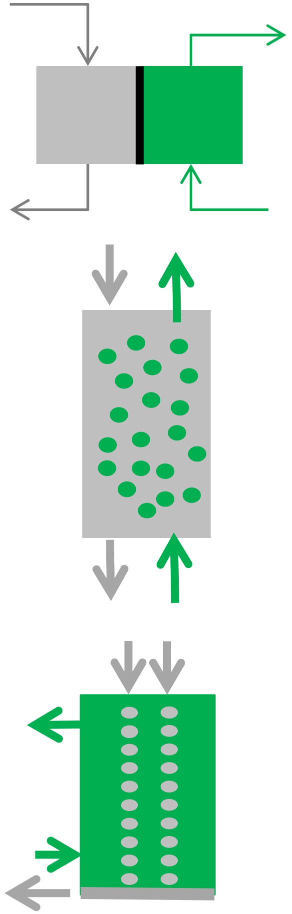
Courtesy of F. Hernandez

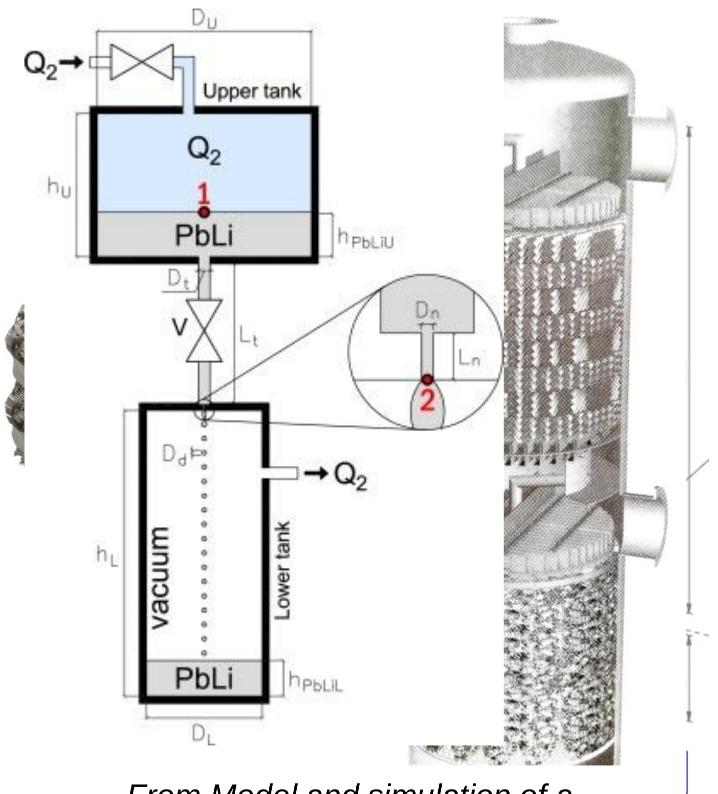
Current Design of WCLL TER

TER system is constituted by 6 PbLi Loops:

- 4 loops for the Outboard (OB) segments, one loop is connected to 4 OB sectors. Mass Flow/loop: 218.56 ±30% kg/s
- 2 loops for the Inboard (IB) segments, where one loop is connected to 8 IB sectors. Mass Flow/loop: 81.61 ±30% kg/s







From Model and simulation of a vacuum sieve tray for T extraction from liquid PbLi breeding blankets

- **PAV (permeation against vacuum)**: Necessitate large permeation surface, high T diffusivity in membrane materials (e.g. V, Nb)
- **Gas bubling**: Gas bubble flow in counter-current with respect to the PbLi flow. T concentrates in the gas bubbles → **GLC technology used in TBM**
- **Vacuum Sieves**: PbLi droplets fall in a vacuum chamber and T is extracted.

BB manufacturing strategies

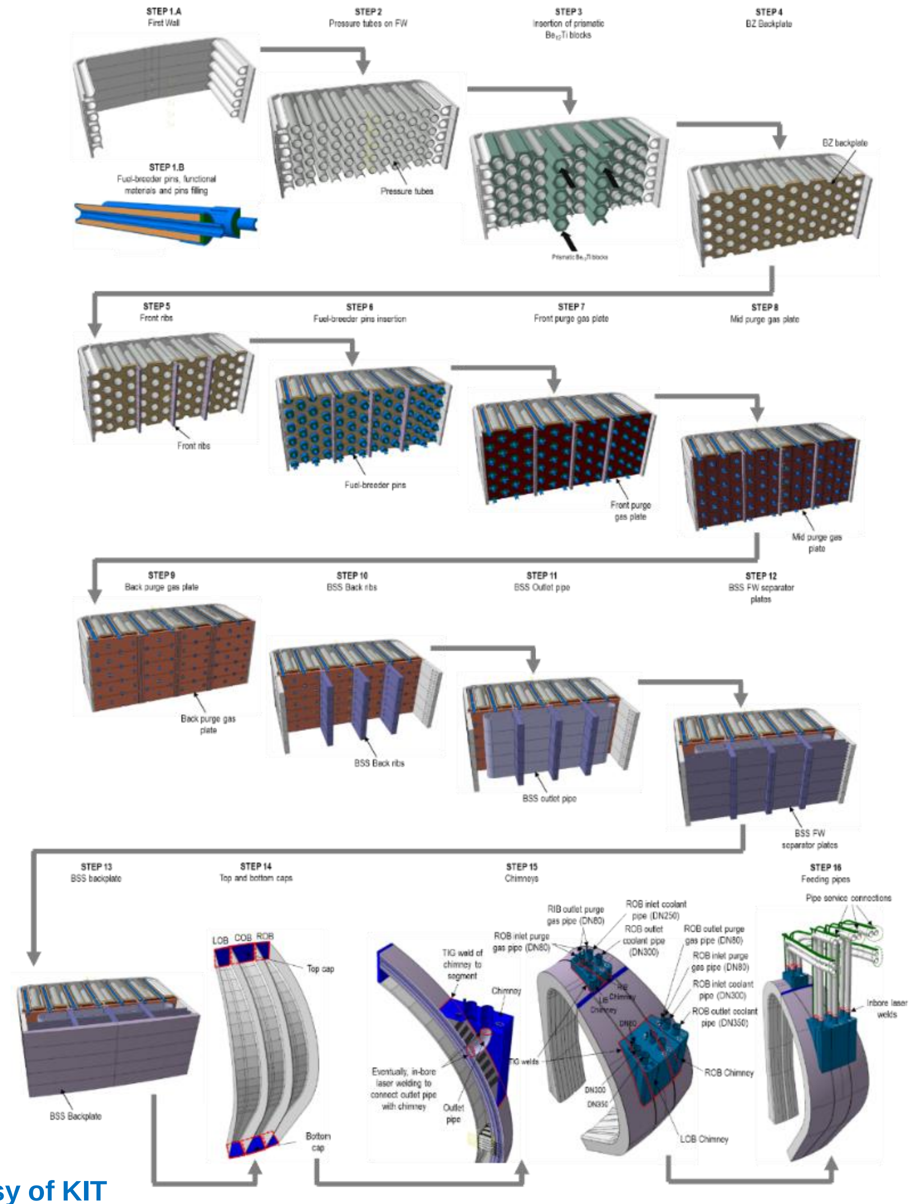
- HCPB assembly mainly formed by:
  - Pressure tubes and fuel pins
  - Breeder Zone backplate
  - Purge gas backplates
  - BSS outer pipe
  - BSS plates
  - Functional materials
  - First Wall

Industry „standard“ subcomponents

Industry „non-standard“ subcomponent

- First Wall:
  - Conventional routes:
    - Near-term: spark-erosion + bending (TBM)
    - Near/mid-term: fail-safe HIP
  - Advanced routes:
    - Mid-term: cold spray + HIP
    - Long-term: Selective Laser Melting

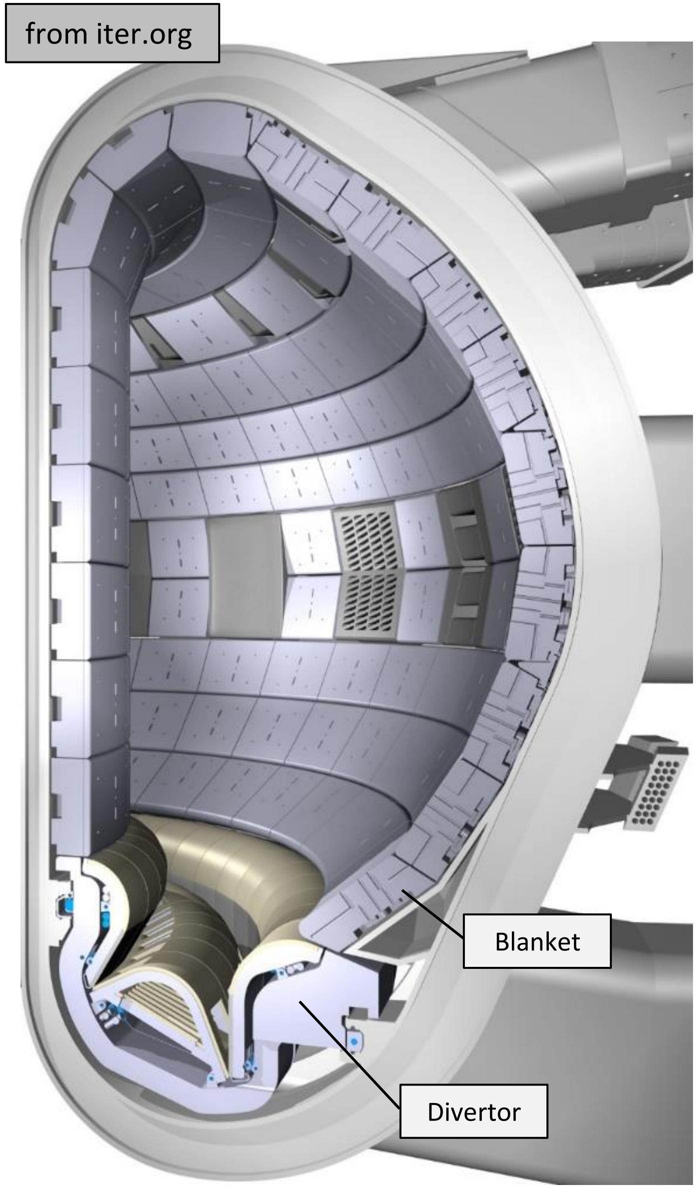
The BB reliability is one of the main challenge to be solved for a fusion power plant.



Courtesy of KIT

Vacuum Vessel and Magnet system

Vacuum vessel

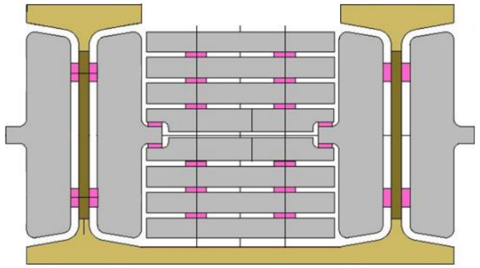


Videos of VV sector fabrication:

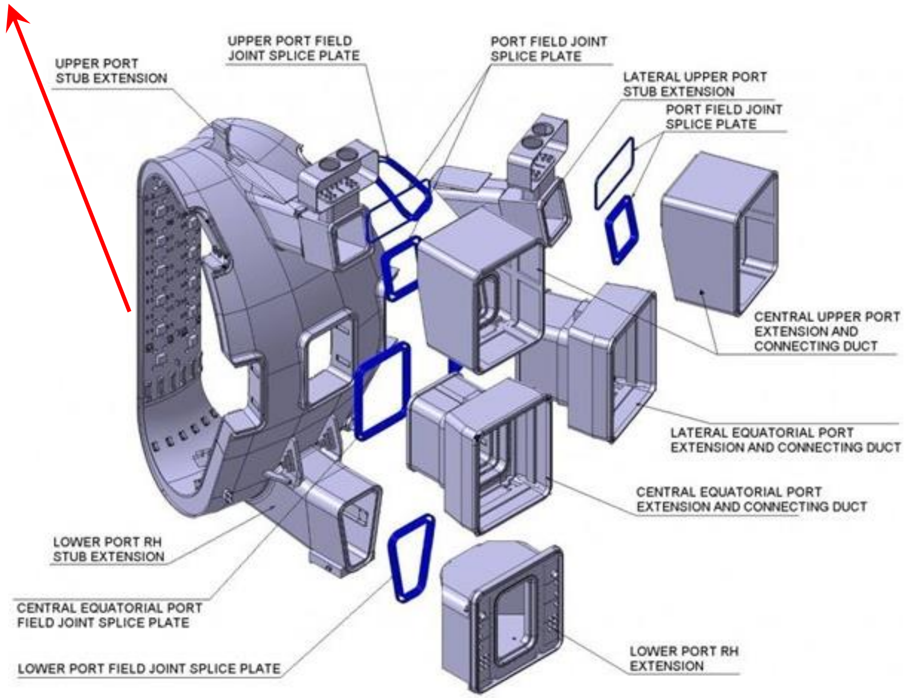
Hyundai Heavy Industries (Apr. 2020): https://www.youtube.com/watch?v=0qkqQFsNFy0&feature=youtu.be

Walter Tosto (Mar. 2021): https://www.youtube.com/watch?v=bATZcpXX6-0

- Main vessel of fusion reactor which surrounds the plasma.
- Double-wall fully welded pressure vessel (stainless steel).
- Actively-cooled with pressurized water.
- Port structures with removable port plugs and port closure plates.
- Nuclear component, highest level of qualification
- First confinement (practically excludes failure).
- Supports the in-vessel components.



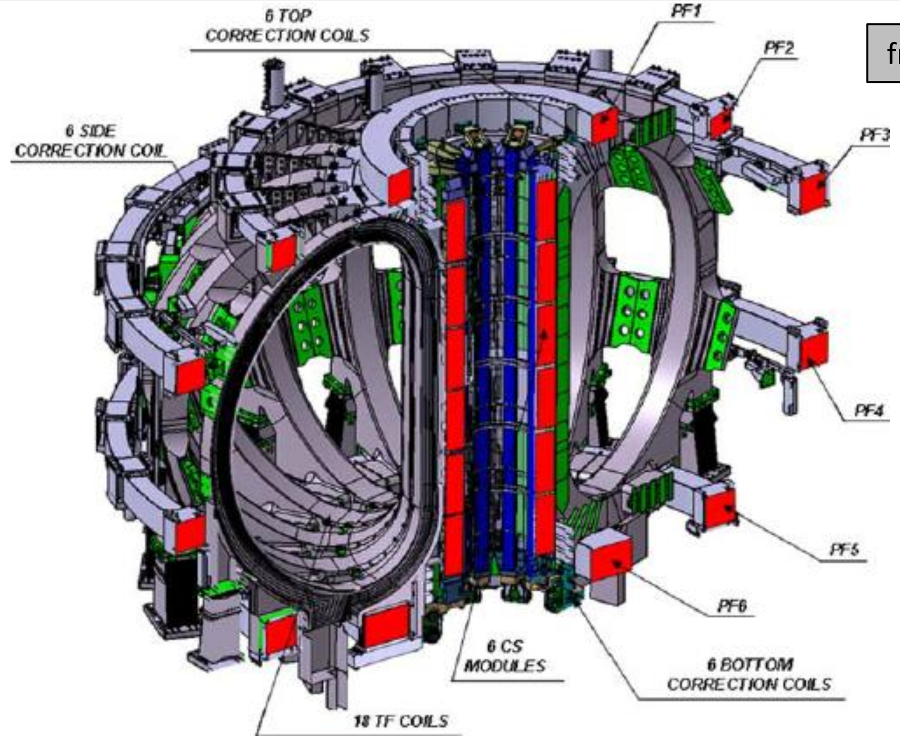
Courtesy of C. Bachmann and J.A. Cordero



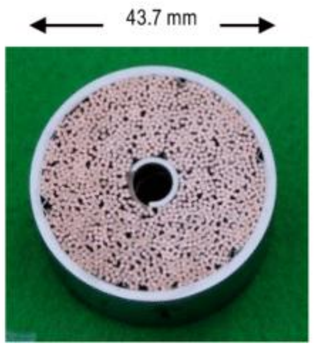
Magnet system

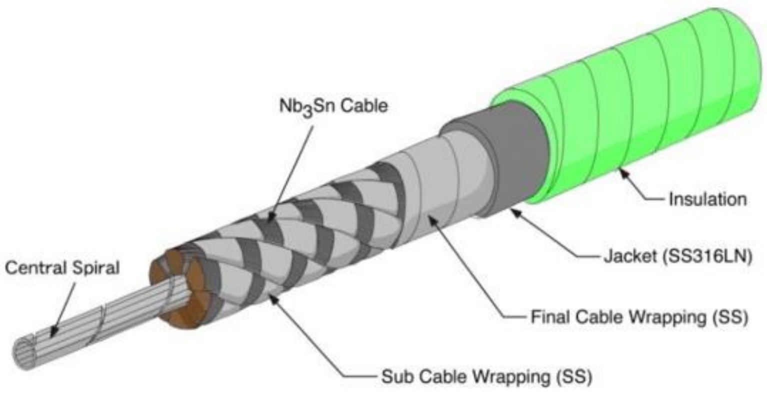
Three types of magnetic coils:
1. Toroidal field (TF) coils to confine the charged particles (together with the plasma current)
2. Poloidal field (PF) coils to shape and control the plasma
3. Central solenoid to inductively drive the plasma current

- Superconducting cables (zero electrical resistivity → long pulses):
    - Toroidal field (TF) coils, central solenoid (CS): Nb_3Sn (operated in magnetic field up to ~13T)
    - Poloidal field (PF) coils: mainly NbTi (operated in magnetic field < ~9T)
- The superconducting strands are cabled with copper wires to a cable that is fitted into a steel jacket integrating also a cooling channel for the liquid helium. A single cable is wound into a massive coil casing made of steel.
- **In-plane loads**: The TF coil casings withstand the large internal electromagnetic pressure due to the coil current (common for all coils)
- **Out-of-plane loads**: The TF coils are mounted together via bolted connections and shear keys to react the out-of-plane forces due to the interaction of the TF coil current with the poloidal field.



from iter.org



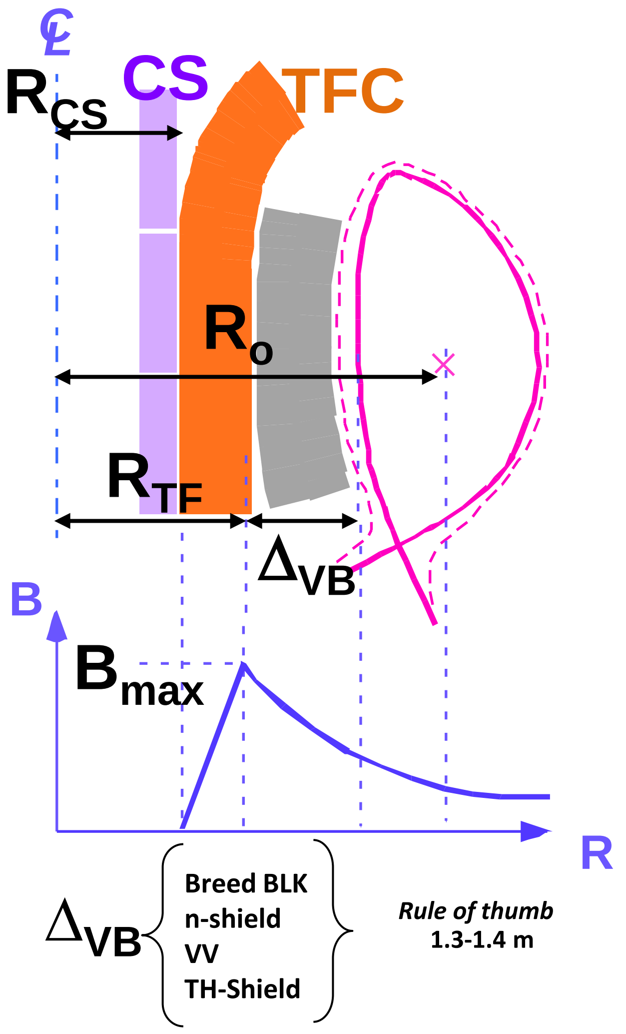


Courtesy of C. Bachmann

from Takayashi, Y., 23rd IAEA conference. 2010

TF coil impact on the size of the machine

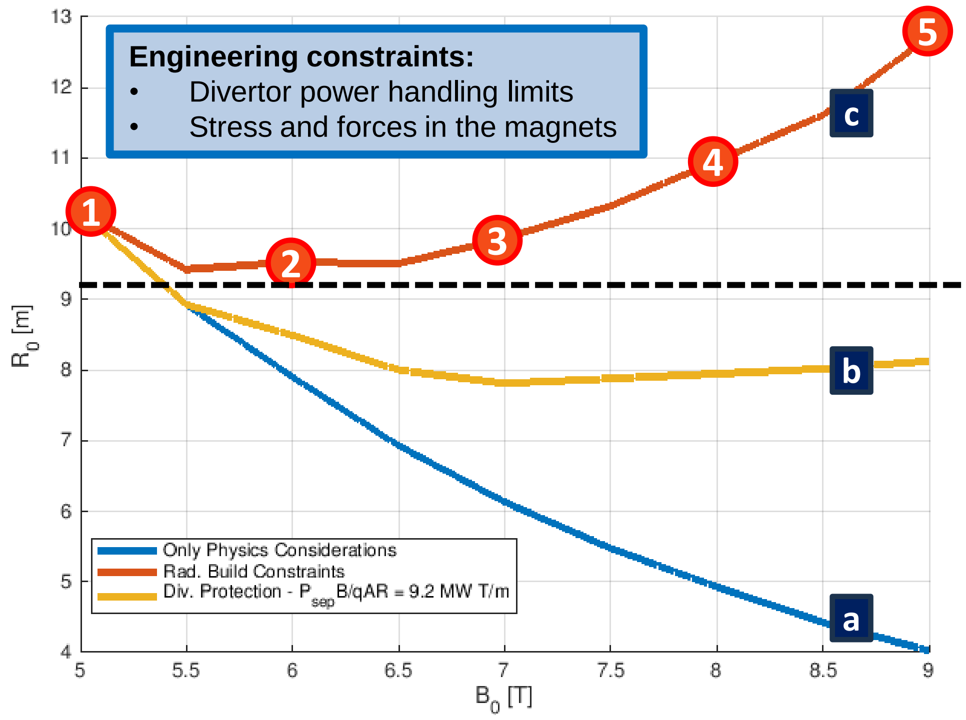
TF Inner Leg Space Allocation



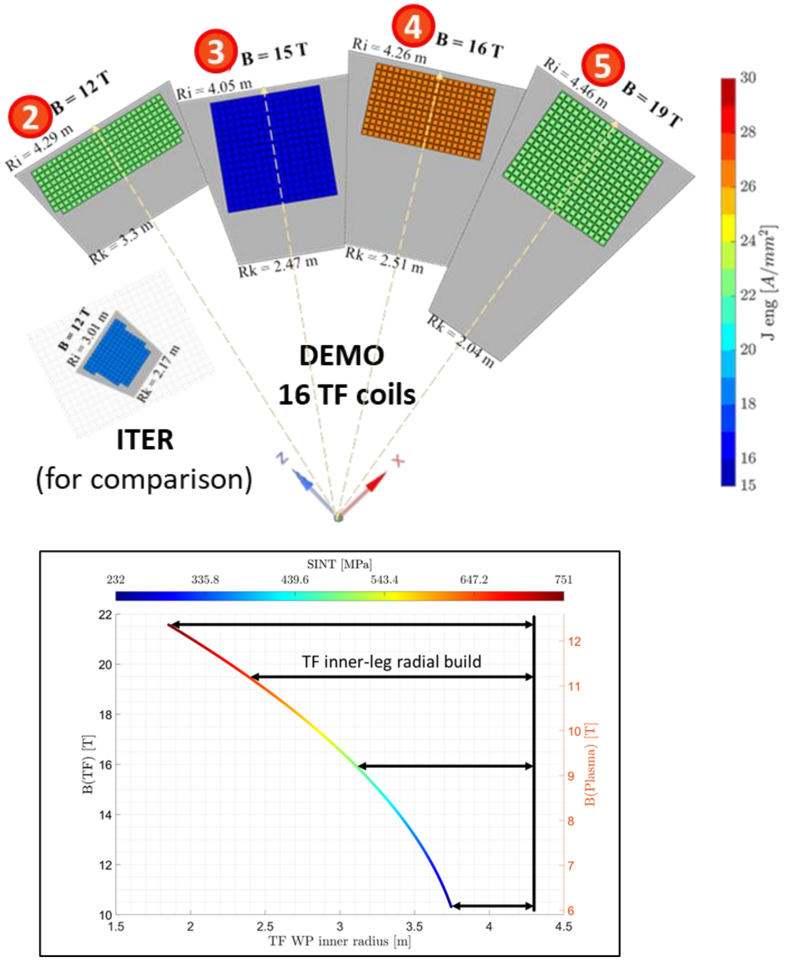
Rule of thumb
1.3-1.4 m

Courtesy of G. Federici

P_{fus}=2000 MW, pulse = 2 hrs, A=3.1



M. Siccinio, C. Bachmann. L. Giannini



Lorenzo Giannini (ENEA)

→**for higher B-field**, large structures are required to resist radial/vertical forces TF inner leg. Alternative mechanical concepts are very challenging or do not bring significant improvements

Magnet winding pack – Limits to current density

- Achievable current density in a magnet is usually determined by engineering constraints and not just the limits of the superconducting material

- Although the relationship between the critical current density and peak field is important in determining the ultimate operating limit; the current density in the coil pack also depends on quench protection and structural considerations

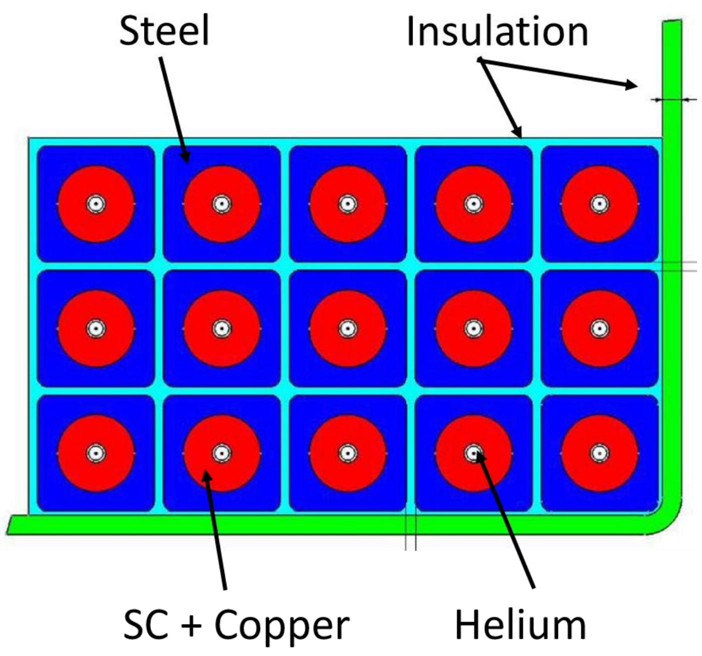
- Developing "high current density" superconducting materials is relevant, but the real challenges are: quality (which for HTS relates to ultimate performance), improving manufacturability and mechanical properties, radiation resistance, and lower cost.

$A_{tot} = A_{sc} + A_{Cu} + A_{He} + A_{ss} + A_{ins}$

A_{sc} – Superconductor
A_{Cu} – Copper (stabilizer)
A_{He} – Helium (coolant)
A_{ss} – Stainless steel (structure)
A_{ins} – Insulation

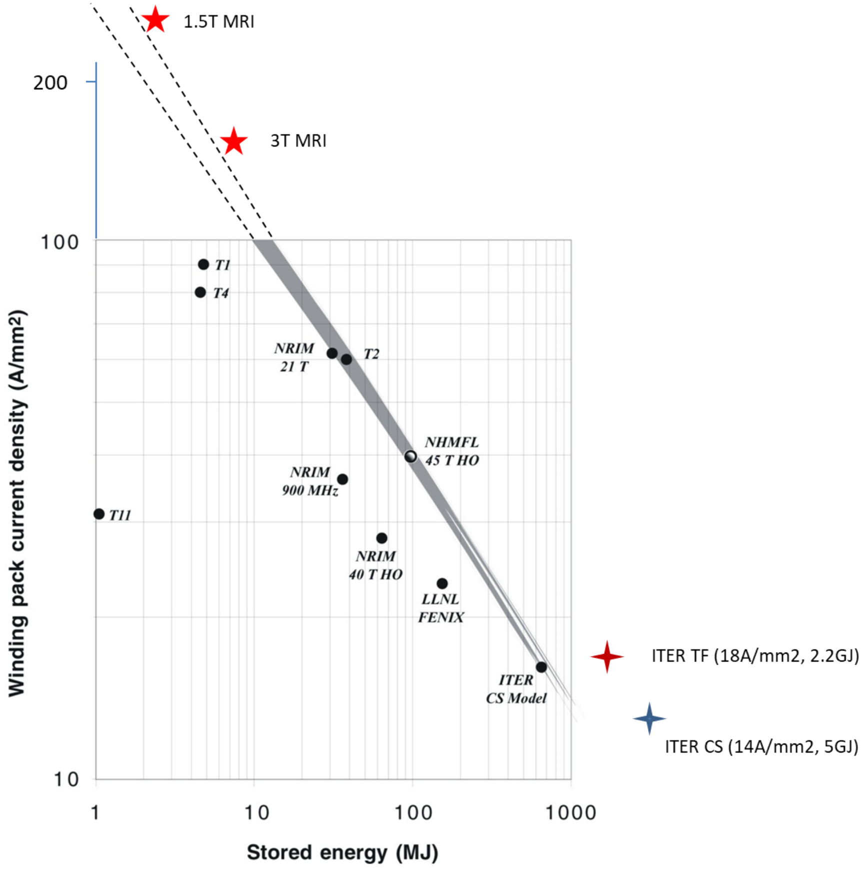
In simplistic terms:
A_{sc} ↑ with B_{max} ↑
A_{Cu} ↑ with E ↑ (stored energy)
A_{ss} ↑ with IxB ↑ (or B^2, proportional to E)

Therefore, we expect $J_{eng} = I/A_{tot}$ to relate more to E than B as the magnets become larger and A_{sc} to be less relevant



Courtesy of C. Luongo

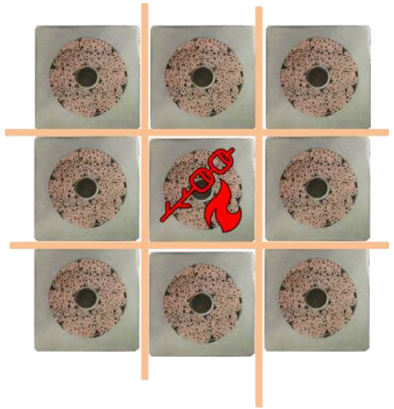
Magnet winding pack – Limits to current density

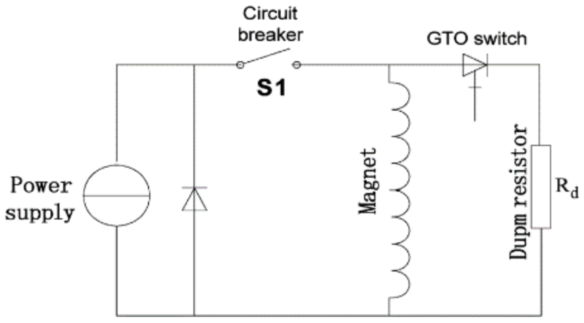


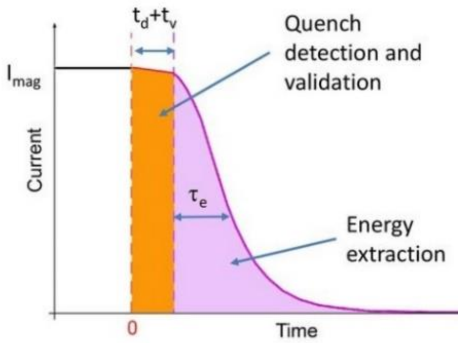
And that is exactly what we observe in practice, the achievable engineering current density (J_{eng}) is strongly correlated (limited) by the magnet stored energy; and nothing else.

Courtesy of C. Luongo

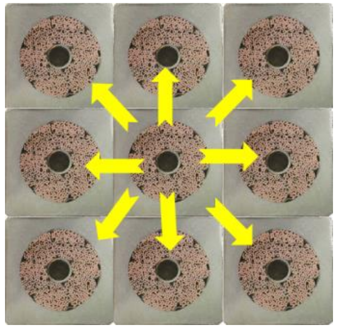
Quench protection







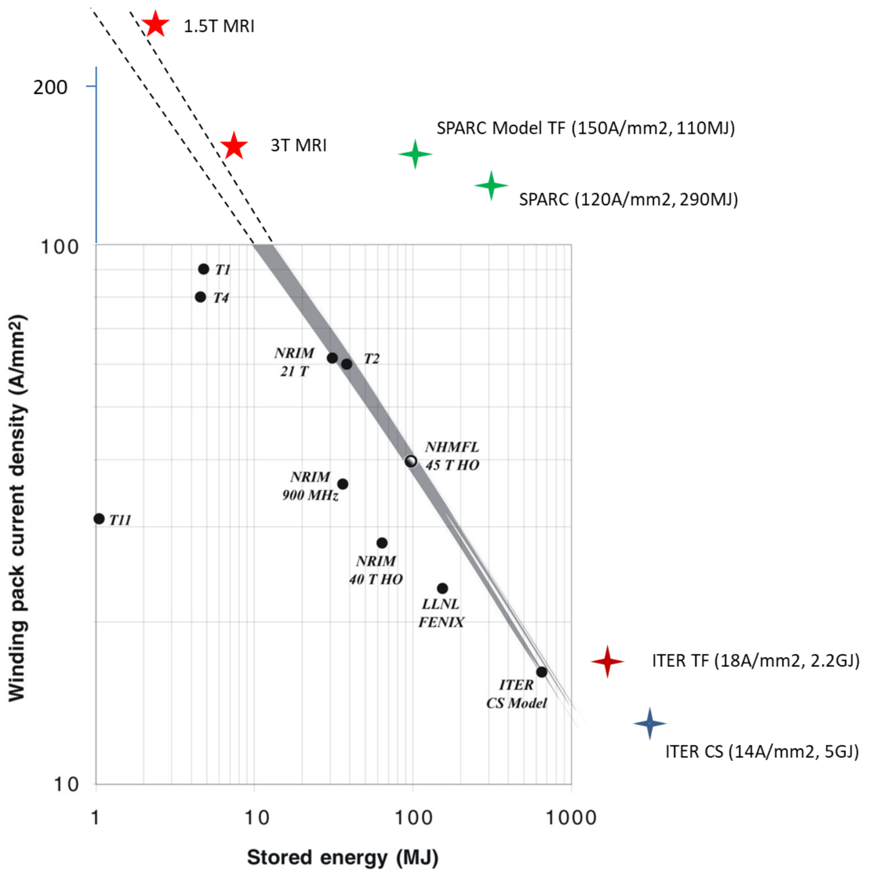
In an insulated coil the (traditional) quench protection assumes detection and external energy dump to bring down the current as fast as possible → quenched conductor has to survive ALONE



In a non-insulated (NI) coil quench protection assumes current will redistribute across the coil pack (quenched conductor SHARES its current with neighbouring turns to survive)

Principle not yet demonstrated in coils of relevant size for fusion

Magnet winding pack – Limits to current density



But here is why there is so much interest in NI coils. If they can be made to work, the limit on J_{eng} due to quench protection constraints could be broken and we can enter a new regime to attain higher coil current densities.

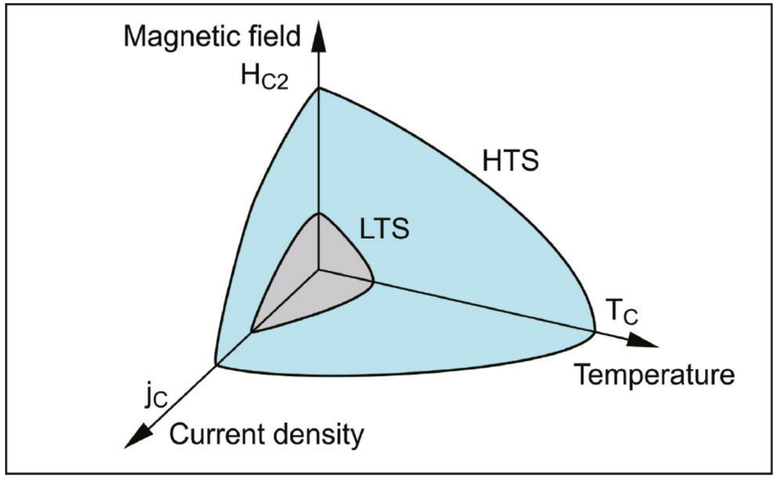
Courtesy of C. Luongo

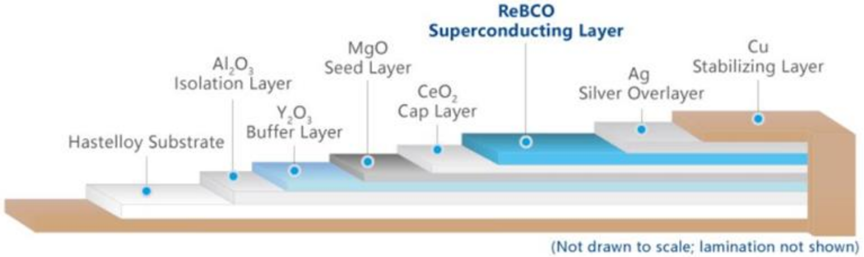
High Temperature Superconductors (HTS)

- First discovered in 1986, HTS offer the possibility of operating in superconducting mode at higher temperature and magnetic field than conventional (LTS) materials.

- Following decades of development HTS materials are now made via industrial production and are slowly gaining market penetration on a variety of applications (research, medical, energy, etc.). Fusion is potentially a major market pull for HTS allowing for further drops in cost.

- HTS would allow for more compact fusion tokamaks, also simpler and more reliable magnets. However, it may not have a significant impact on overall cost because of other technical limits on tokamak compactness, and the HTS cost, which today is significantly higher than LTS



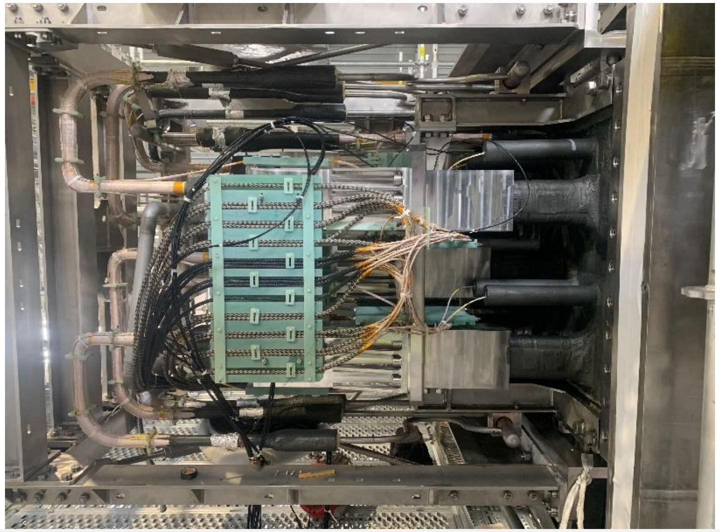


Courtesy of C. Luongo

High Temperature Superconductors (HTS)

- Operation at higher temperature is not necessarily a big advantage either:
  - Magnet cooling costs are negligible in the overall plant
  - Removal of nuclear heating loads is more difficult with a low-density gas than with a liquid

- The higher operating temperature should be used instead to:
  - Operate with higher temperature margin (reducing probability of quench) – Design non-isothermal magnets
  - With increased margin, allow for the design of greatly simplified coil packs, indirectly cooled by conduction (segregating the cooling and electrical functions of the conductor)



ITER TF coil terminals
Leads, wires, and pipes with electrical breaks, all penetrating the ground plane

This is what we would like to greatly simplify or eliminate
Both ITER CS module and JT-60 failures were initiated by Paschen discharge at the lead region

Courtesy of C. Luongo

Remote Maintenance of In-Vessel components

Replacement of in-vessel components (IVCs)

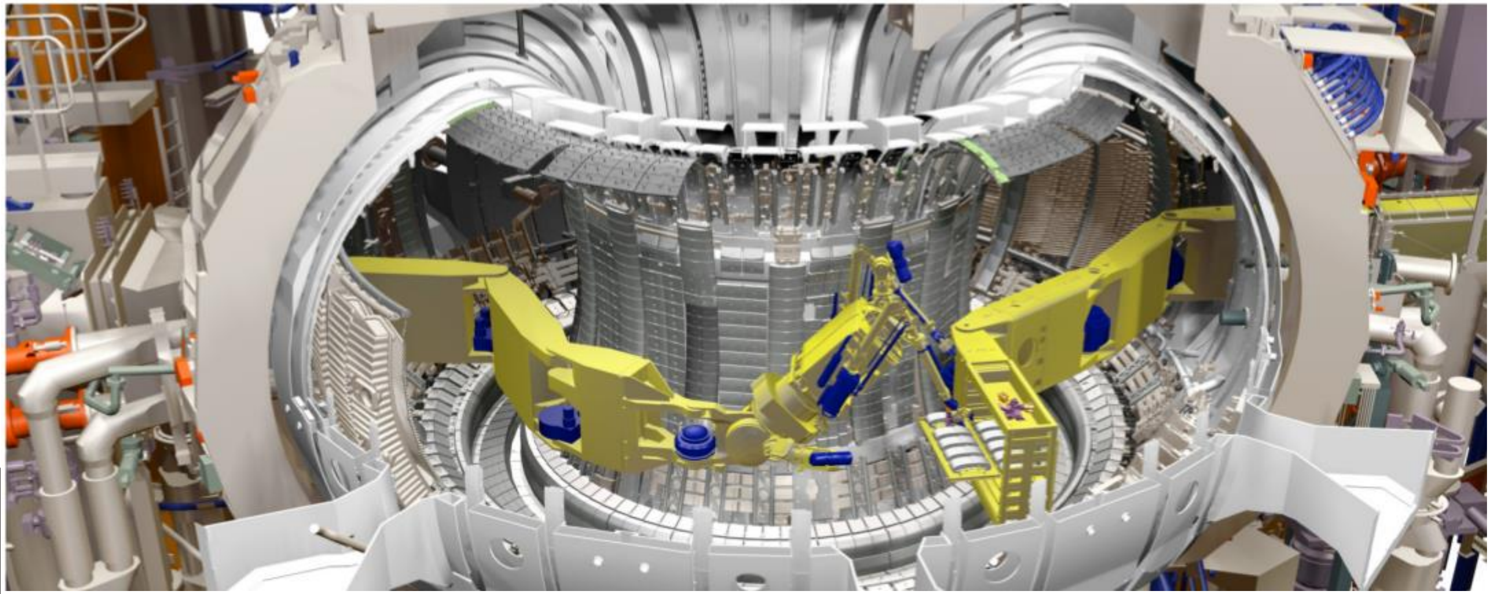
Main constraint: in-vessel gamma radiation level → robotic tools required

High neutron wall load highly activates the PFCs:

- ~2000 Gy/h: DEMO (8 weeks after shutdown)
- ~300 Gy/h: ITER (during maintenance)
- 530 Gy/h: Fukushima containment vessel (2017)

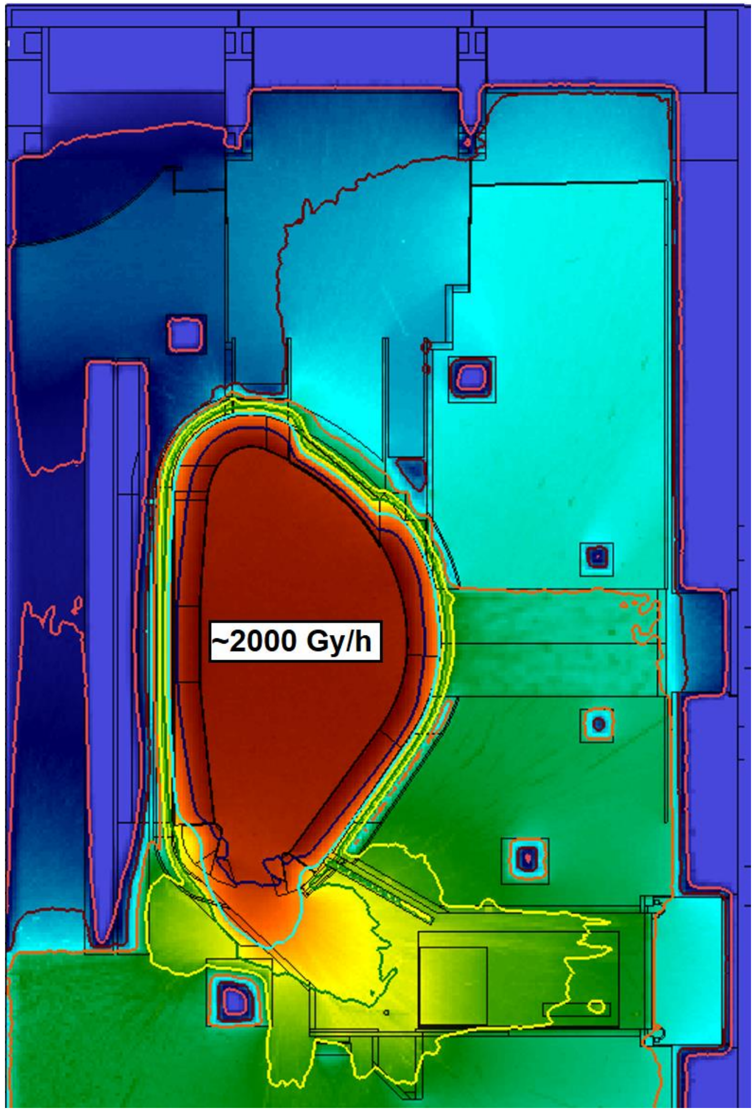
Inside ports, behind IVCs 2-4 orders of magnitude lower dose rate

Note: Absorption rate in silicon, 900 Gy/h = 1000 Sv/h



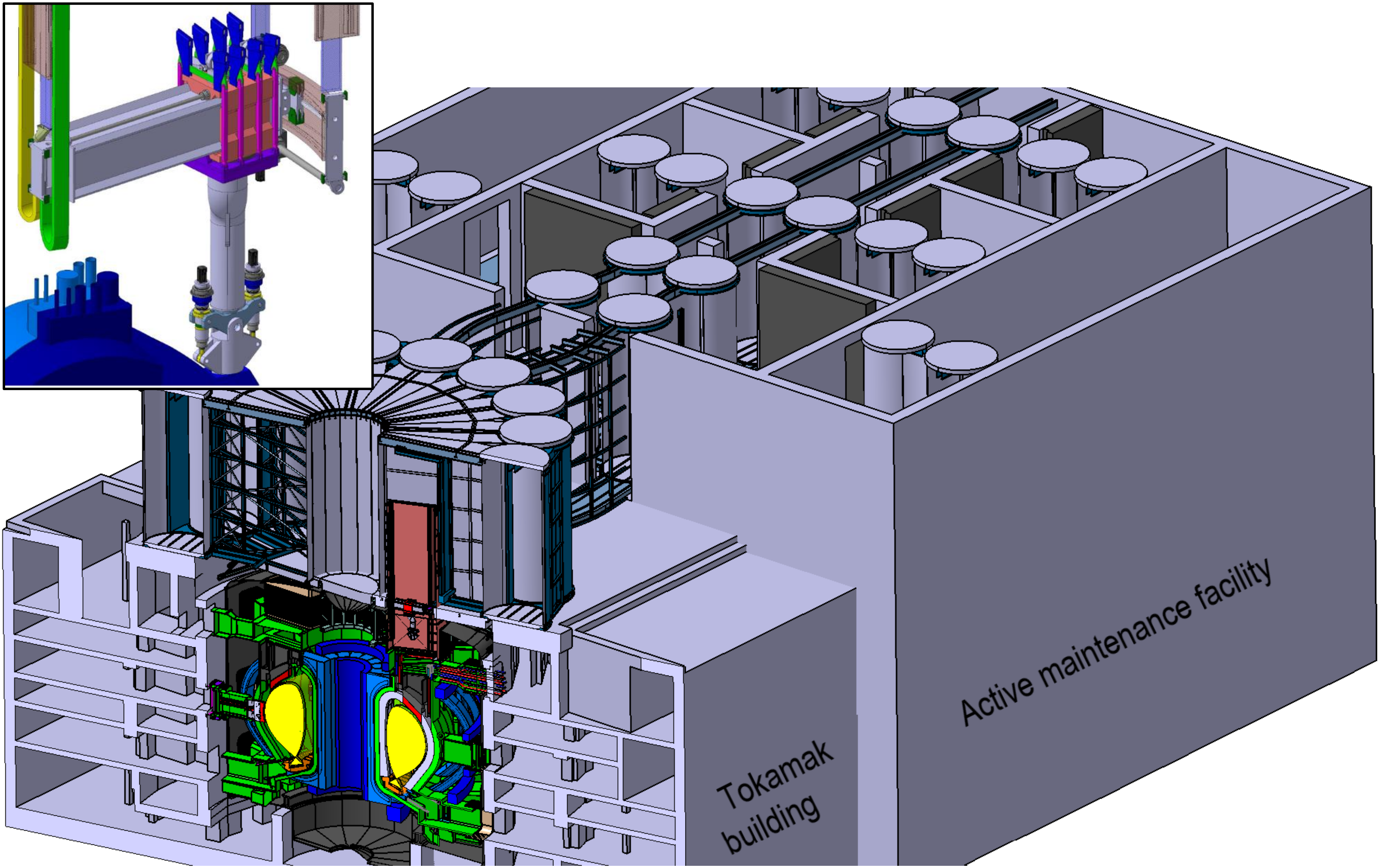
JET Boom

Sv/h in DEMO 8 weeks after shutdown:



Courtesy of C. Bachmann

Replacement of in-vessel components (IVCs)



- All DEMO in-vessel components require replacement
- Large blanket segments allow faster replacement and do not require the remote-controlled tools to enter into the high radiation zone in front of the plasma-facing components.
- Vertical removal to allow using a crane-like system to carry the large weight.
- Remote maintenance infrastructure larger than tokamak building.

Courtesy of C. Bachmann

Replacement of in-vessel components (IVCs)

Confinement/penetration/radiation protection

Dealing with water activation

In DEMO the principle of fission plants is adopted: primary coolant circuits inside secondary shielding walls.

<u>Secondary shielding requirements:</u>

- In fission plants secondary shielding ~0.8-1.2 m.
- In DEMO N-16 specific activity is ~300 times higher.
- Increase of concrete thickness required to keep same dose level (+40-50 cm)

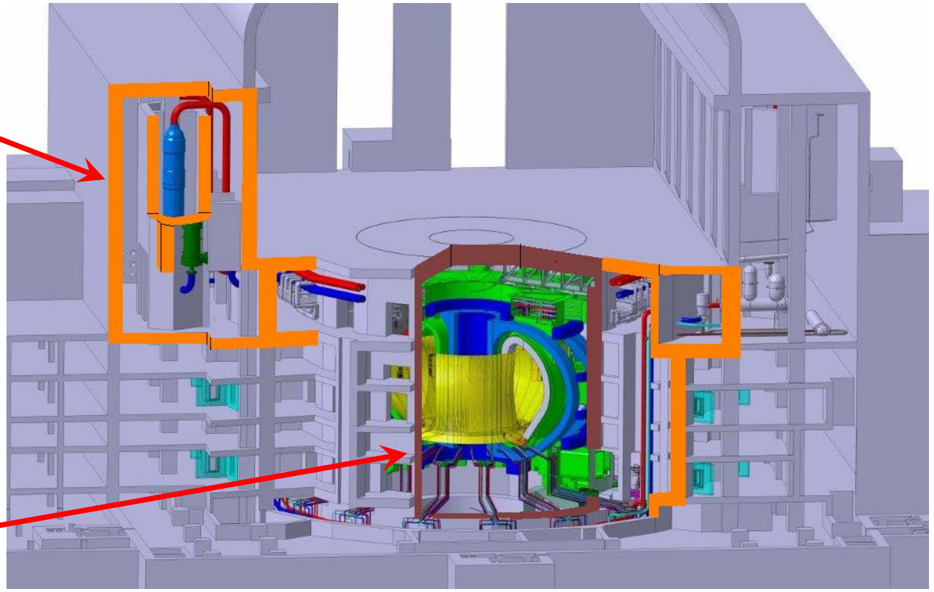
<u>DEMO concept based on lessons learned from ITER:</u>

- 1.25 m walls implemented around the pipes and steam generators (ITER: as low as 0.5-0.8 m)
- Water pipes inside cryostat routed in well-shielded structures.
- Activated water segregated from sensitive equipment
- During maintenance of primary system equipment, dose to workers is dominated by deposits of activated corrosion products → maintenance concepts as in fission plant are implemented.

C. Gliss et al., FED 168 (2021)

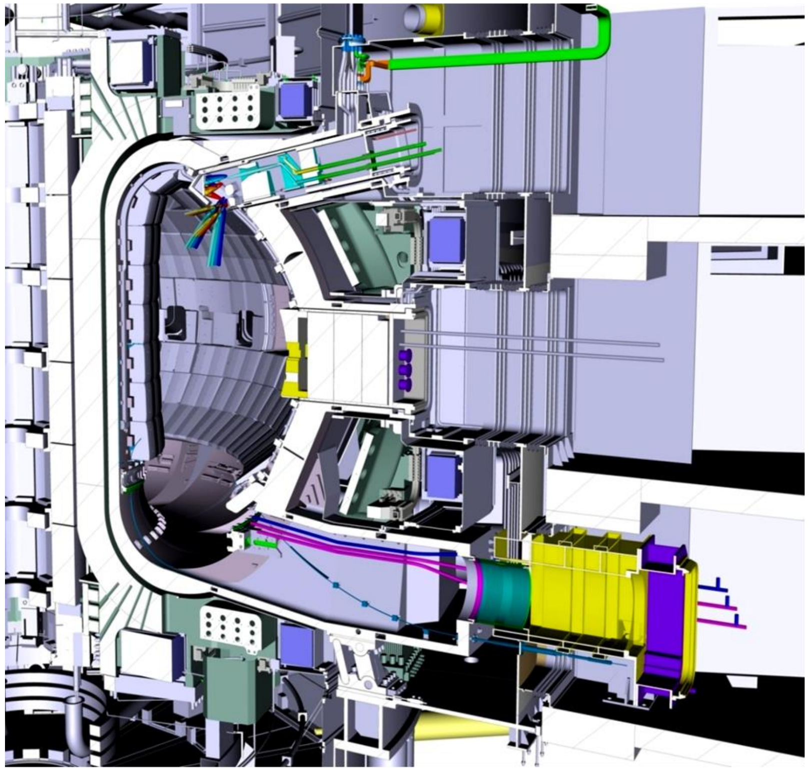
Secondary shielding

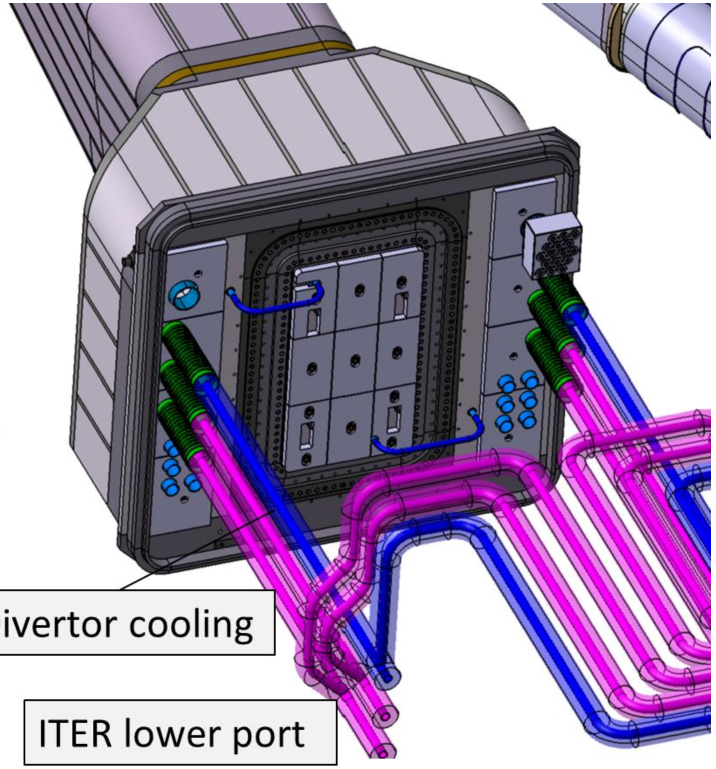
Bioshield

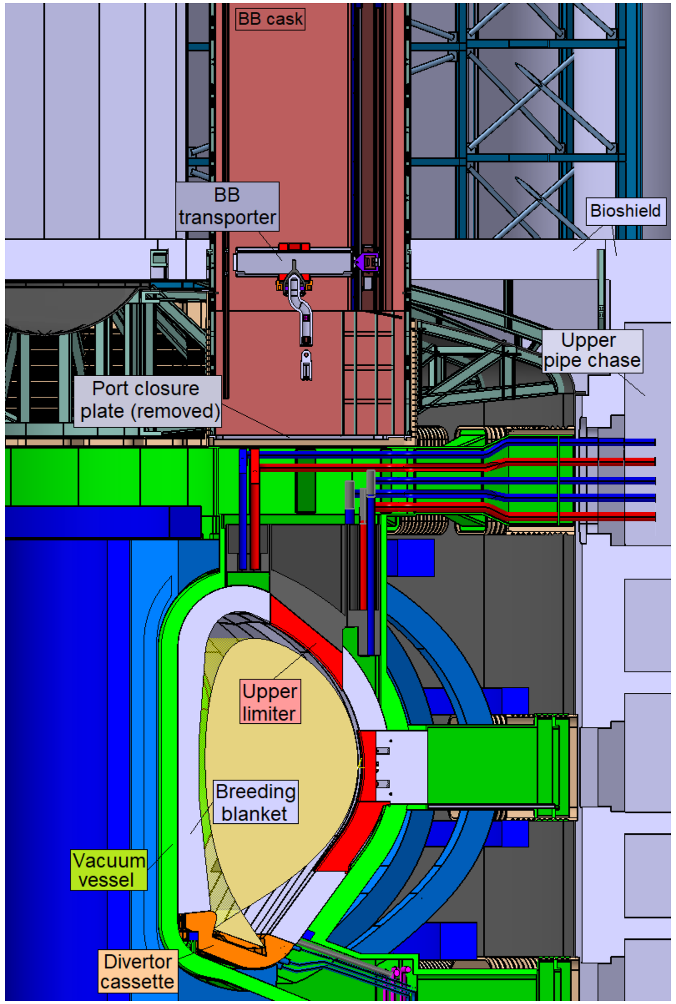


Courtesy of G. Federici

Penetrations of first confinement barrier







Courtesy of C. Bachmann

Radiation Protection Requirements

- The design should make provisions early in the design process to facilitate maintenance, fuelling, operation, etc., in view of the radiation protection

Radiological zones are defined for DEMO.

- In particular, it should provide for
  - appropriate accessibility,
  - adequate shielding, and
  - minimal handling of items performing safety functions

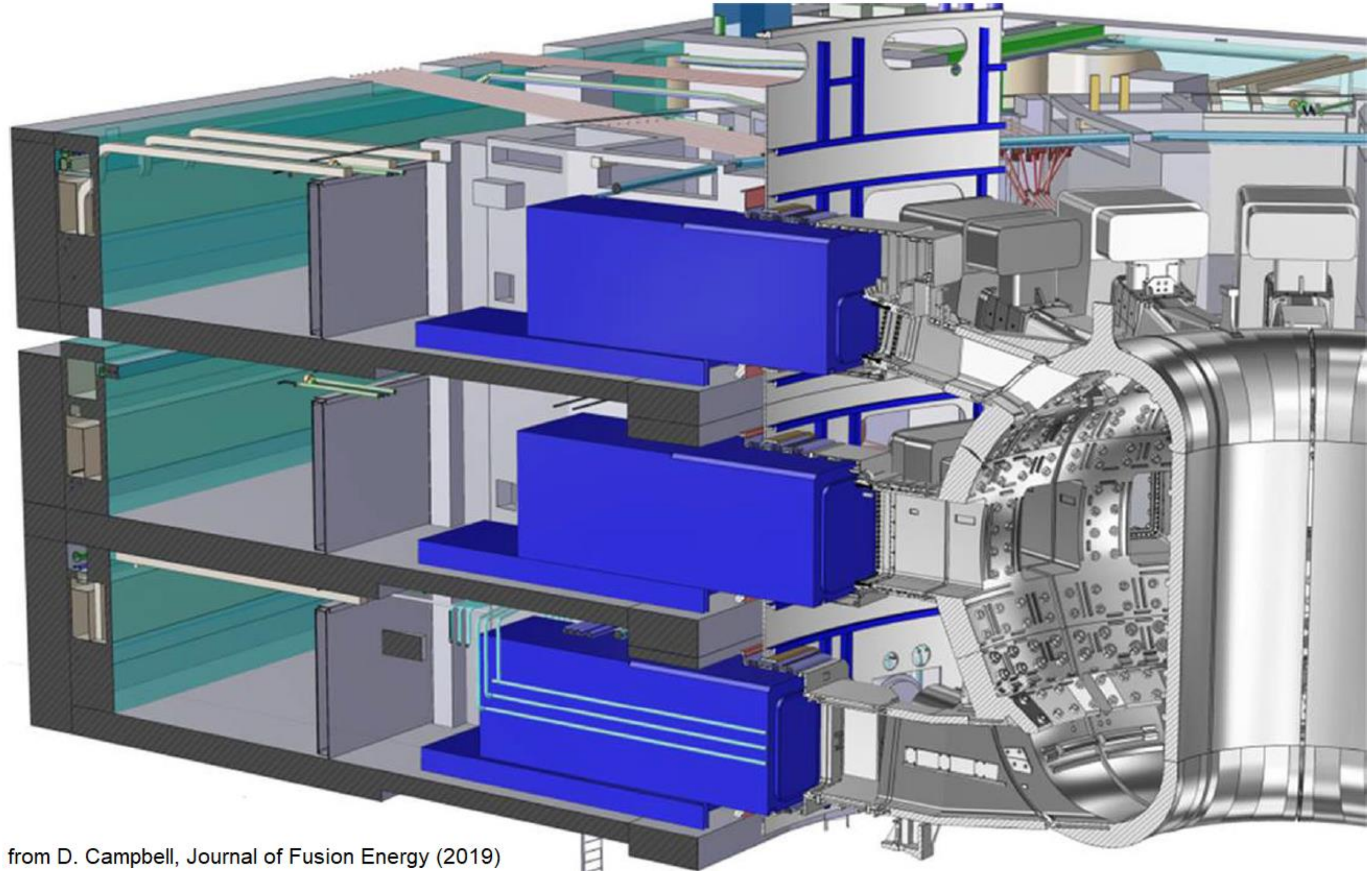
  in areas where there is the potential for contamination or exposure to radiation or hazardous materials, in order to facilitate maintenance and repair and to keep worker exposures ALARA.

<table>
<tr><th colspan="2">Zone type</th><th>Zone identification</th><th>Maximum total effective dose (external plus internal)</th><th>Maximum external dose to hands, forearms, ankles and feet</th></tr>
<tr><td colspan="2">Unregulated</td><td>White</td><td>80 µSv/month</td><td></td></tr>
<tr><td colspan="2">Supervised</td><td>Blue</td><td>7.5 µSv/hr</td><td>200 µSv/hr</td></tr>
<tr><td rowspan="4">Controlled</td><td>Limited</td><td>Green</td><td>25 µSv/hr</td><td>650 µSv/hr</td></tr>
<tr><td>Specially regulated</td><td>Yellow</td><td>2 mSv/hr</td><td>50 mSv/hr</td></tr>
<tr><td rowspan="2">Forbidden without specific authorization</td><td>Orange</td><td>100 mSv/hr</td><td>2.5 Sv/hr</td></tr>
<tr><td>Red</td><td>above 100 mSv/hr</td><td>above 2.5 Sv//hr</td></tr>
</table>

Courtesy of G. Federici

Contamination control during maintenance

The high expected level of contamination with tritium and dust inside the VV has led in ITER to the choice of sealed cask for the in-vessel maintenance. This is a major driver of plant complexity.



from D. Campbell, Journal of Fusion Energy (2019)

- During in-vessel maintenance the vacuum vessel (VV) is vented.
- The ITER VV contains activated dust and tritium. The spread of contamination into man-accessible areas must be prevented.

  → So-called casks are docked to the VV before the port closure plate is opened.

  → The air in the room hosting the cask is circulated to the detritiation system.

Courtesy of G. Federici

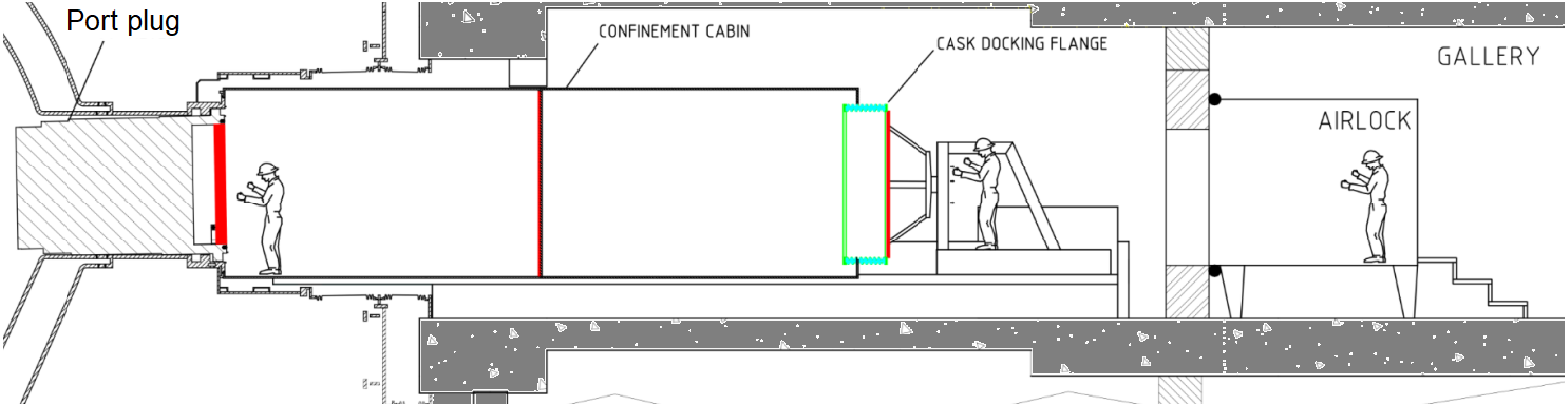
Protection of personnel

There is a need to minimise operator exposure during maintenance

Courtesy of G. Federici

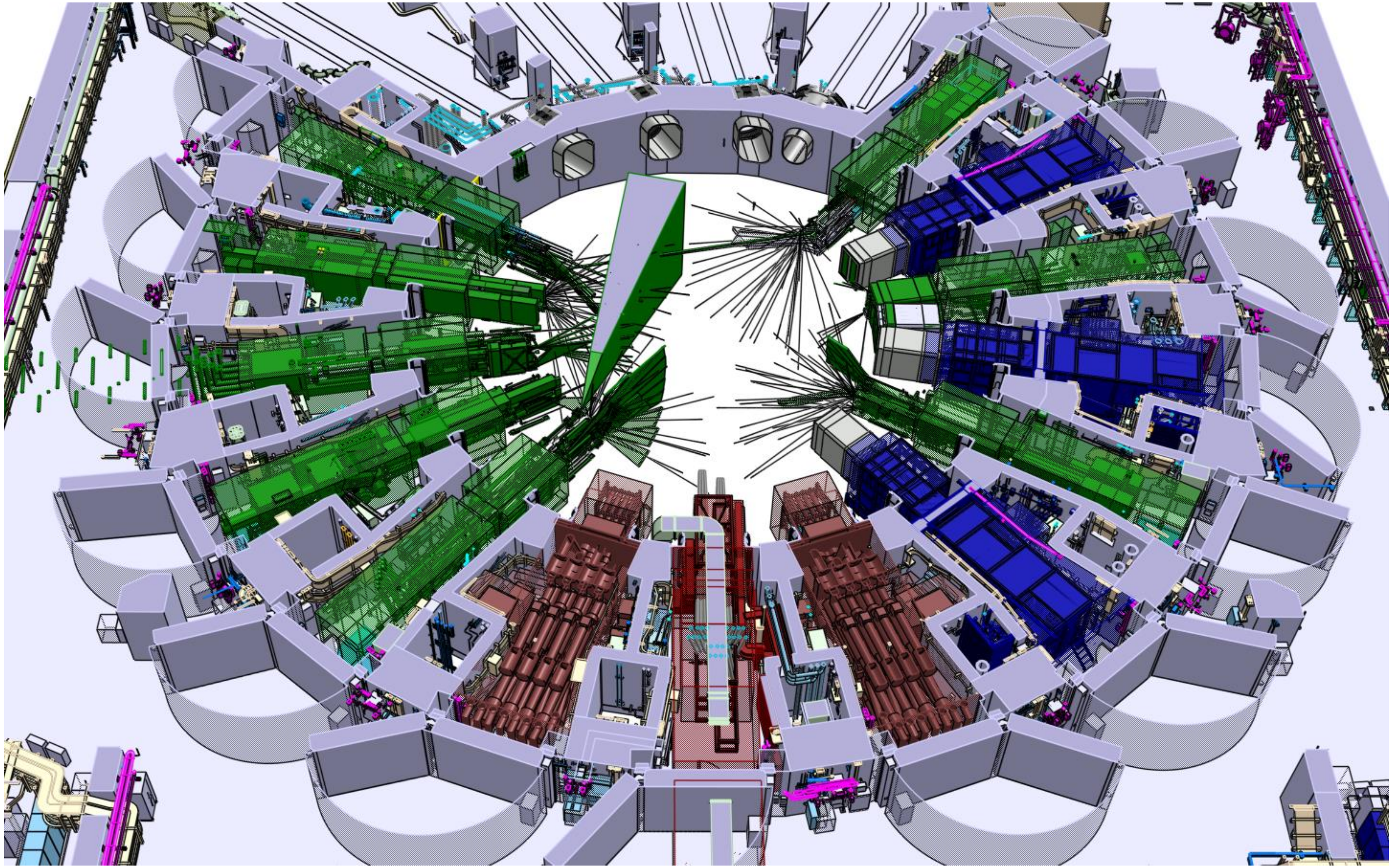
- During preparation of in-vessel maintenance and after cask operations access of personnel is required. Workers need to be protected from gamma radiation and contamination → use of shielding and confinement cabins.
- Other building areas must be protected from contamination → use of sealed volumes and airlocks.

Installation of decontaminated cask docking flange in ITER:

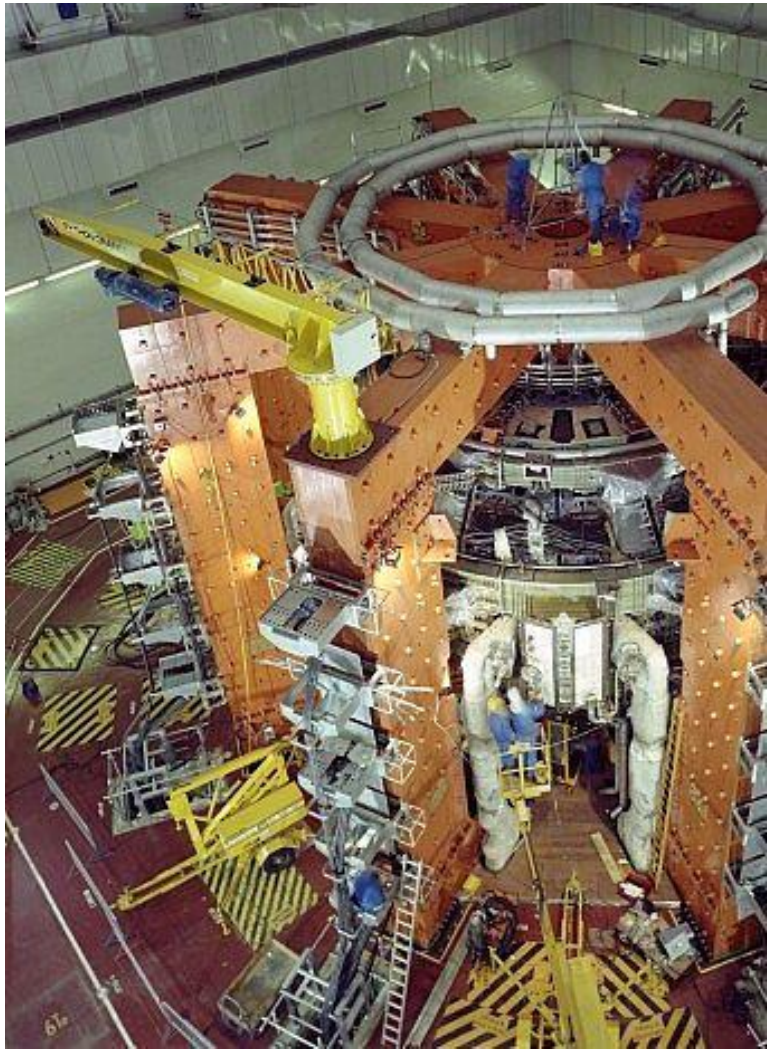


Building design for tokamak access with port cells

ITER port cells



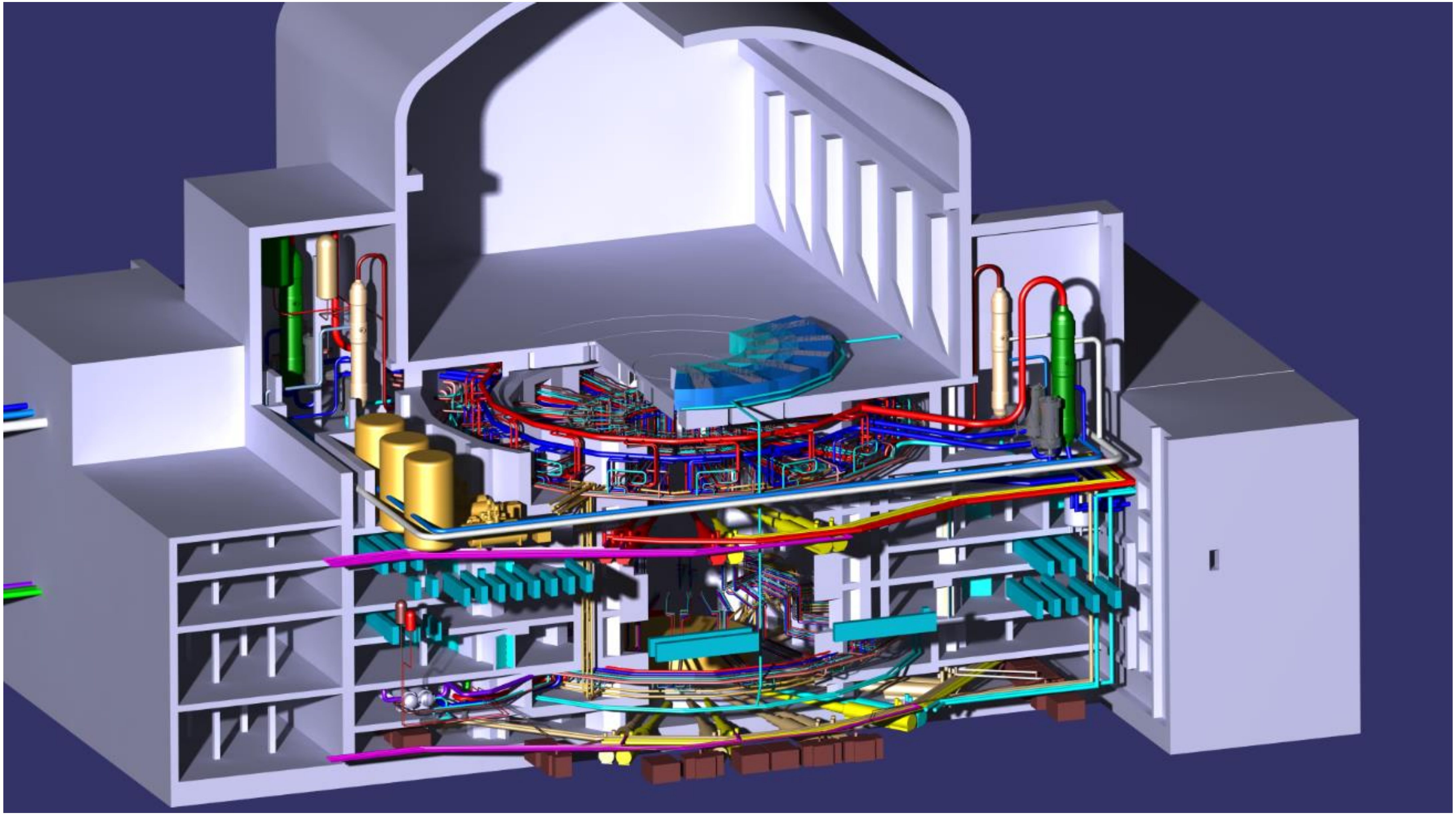
JET hall



Courtesy of C. Bachmann

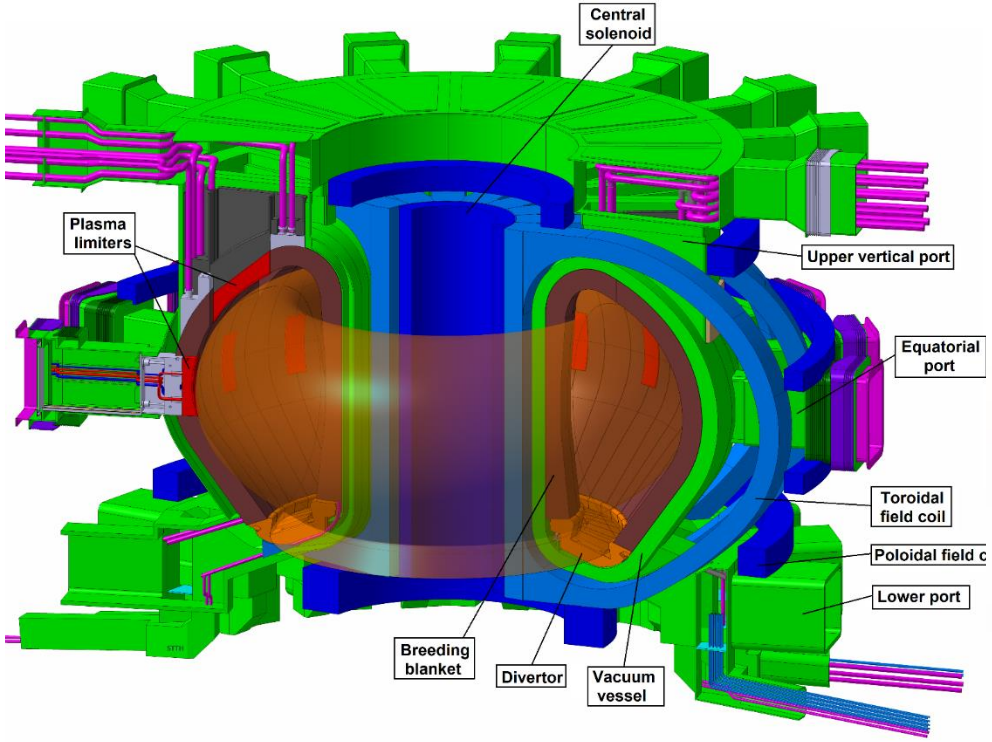
Integration

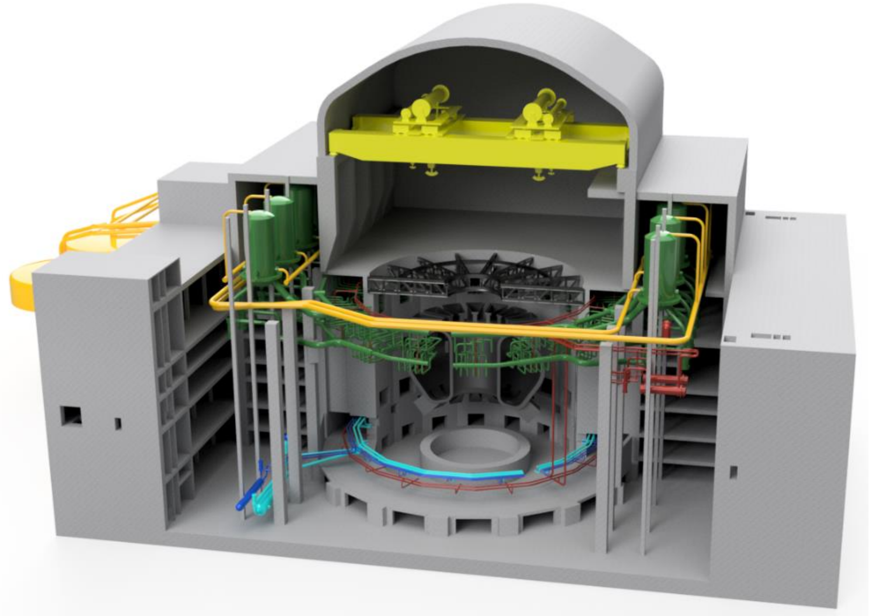
Reactor Integration



Conclusions

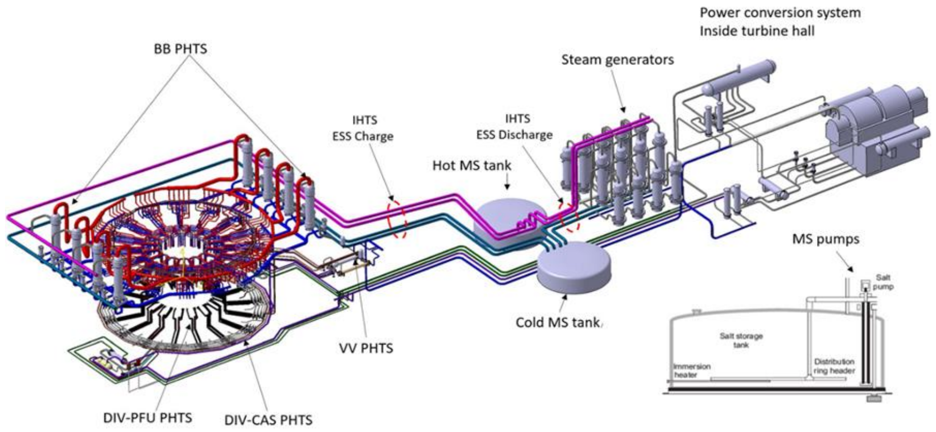
First results DEMO pre-conceptual design phase






Fusion Engineering and Design

A special issue on the DEMO Pre-Concept Design Phase activities has been completed for the Fusion Engineering & Design (FED) scientific journal. All articles can be accessed via this link: https://www.sciencedirect.com/journal/fusion-engineering-and-design/special-issue/10RRZQ6LW4H.



Key Takeaway Messages

- **Nuclear design considerations are essential for the design of nuclear fusion facilities**.

- **The nuclear shielding of critical systems in a nuclear fusion facility affects the utilization of critical space in the vacuum chamber, the design of the in-vessel components, and the port layout.**

- **Greater attention to integration and layout**: The design must be carefully integrated to ensure that all of the critical systems are compatible with each other.

- **Robust confinement systems**: The confinement systems must be robust enough to withstand the high levels of heat and radiation that will be generated by the fusion reaction.

- **The development or adaptation of codes and standards for fusion**: The design must be based on sound engineering principles and must meet all applicable safety standards.

- **The need to develop new materials and technologies that can withstand the harsh conditions of a nuclear fusion environment is essential to the progress of fusion research.**


FAIRNESS
Transparency
Collaboration
Loyalty
OPENNESS
Open doors
Open hearts
Open minds
Open ears
COMMITMENT
Ownership
Critical thinking
Determination
Respect
DIVERSITY
Cooperation
Equal opportunities
Inclusion