Advanced Course on Fusion Engineering & Technology – 02/12/2024

# Cooling and Balance of plant and radiation protection

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Acknowledgments: DEMO Central Team

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# Outline

Part 1 – Design of Tokamaks for power generation:

1. Tokamak basic design
2. Plasma –facing components
3. Radiation shielding and contamination control

Part 2 – DEMO design and challenges:

1. Sizing DEMO
2. Irradiation resistant materials
3. First wall protection
4. Tritium breeding blanket

Part 3 – Fusion Power Plant Energy Conversion Systems

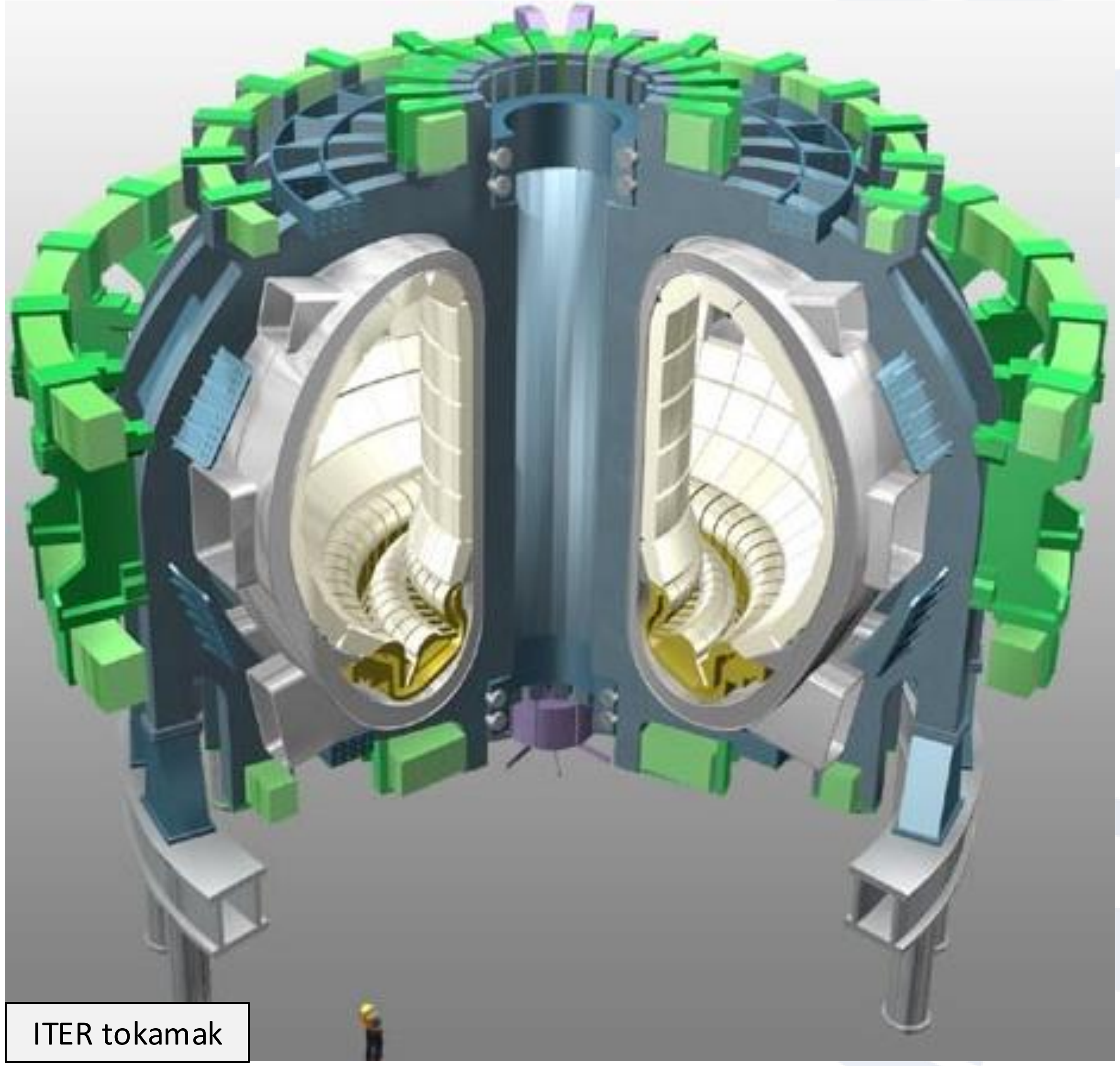
1. Fusion Energy Conversion Cycle and DEMO Balance of Plant Solutions
2. The design of DEMO Tokamak Coolants System
3. Pulsed operations in the Tokamak Coolant Systems
4. Water chemistry in fusion reactors

Part 4 - Radiation Protection approach for the coolant systems

# ITER tokamak – basic design

# Tokamak has three main systems

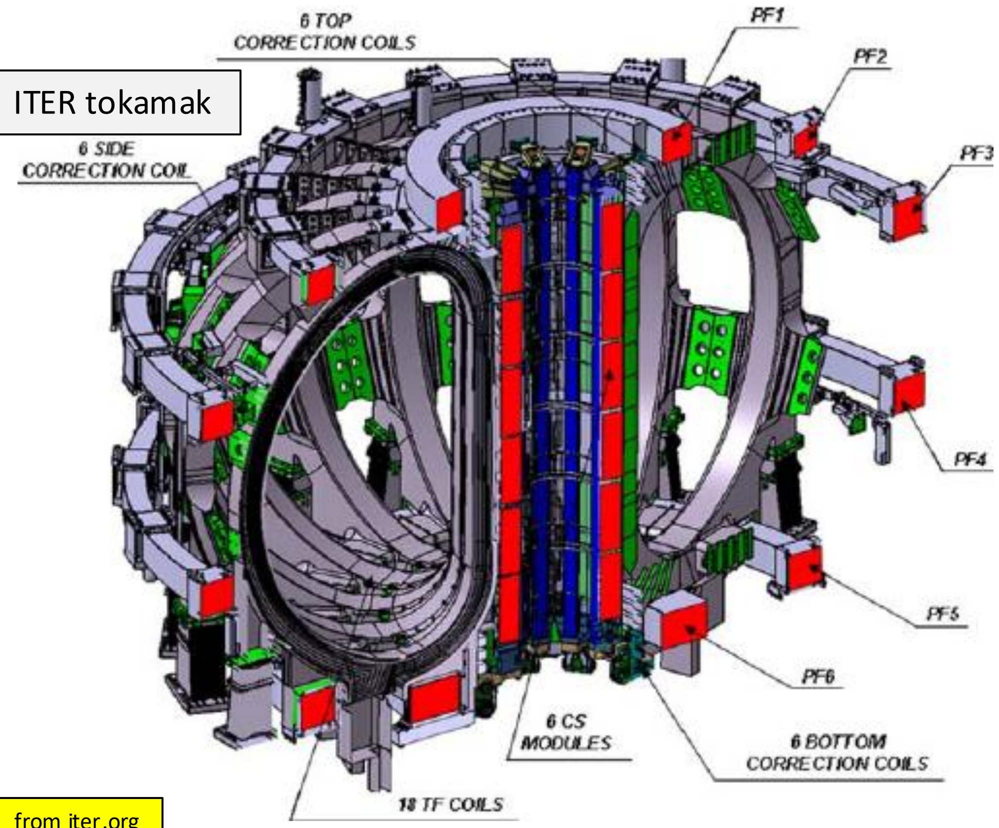
1. Magnets
2. Vacuum Vessel, supports the wall components
3. Cryostat (in case magnets are cryogenic) → next slide



ITER tokamak

# Magnet system

ITER tokamak

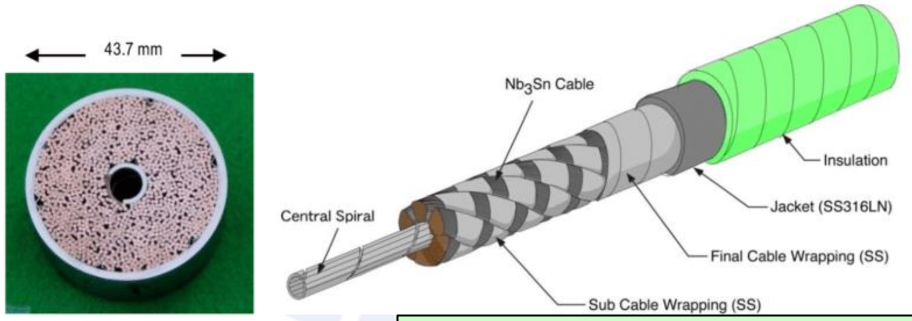



from iter.org

- 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, which is designed to withstand forces generated interaction of magnetic field and currents.

<u>Three types of magnetic coils:</u>

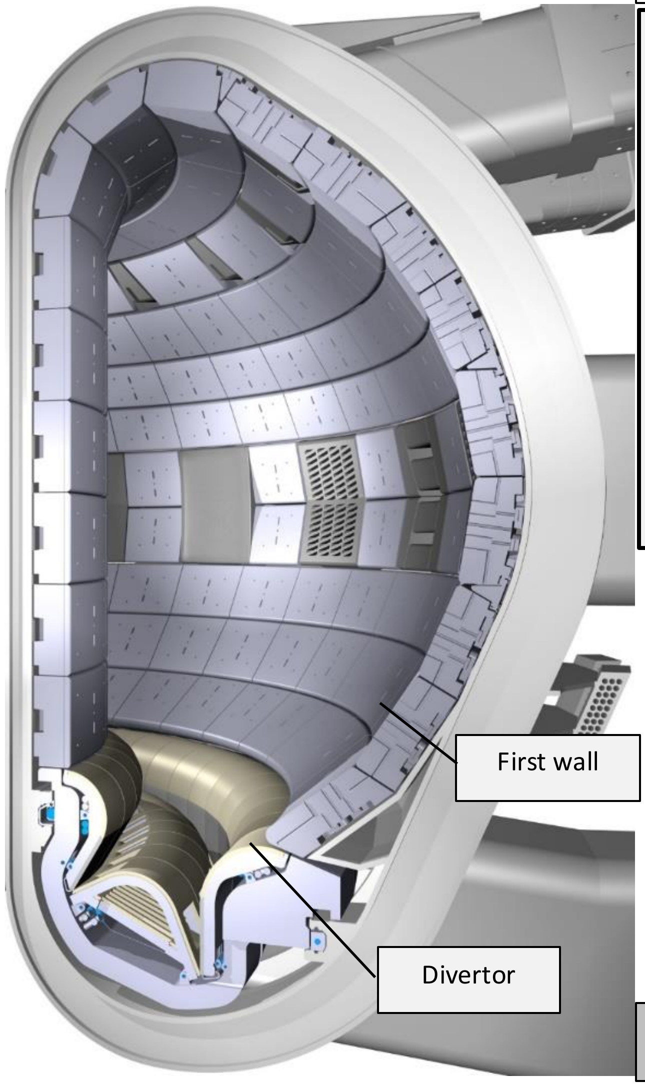
1. Toroidal field (TF) coils to confine the charged particles (together with the plasma current)
2. Poloidal field (PF) coils to shape and control the plasma
3. Central solenoid to inductively drive the plasma current



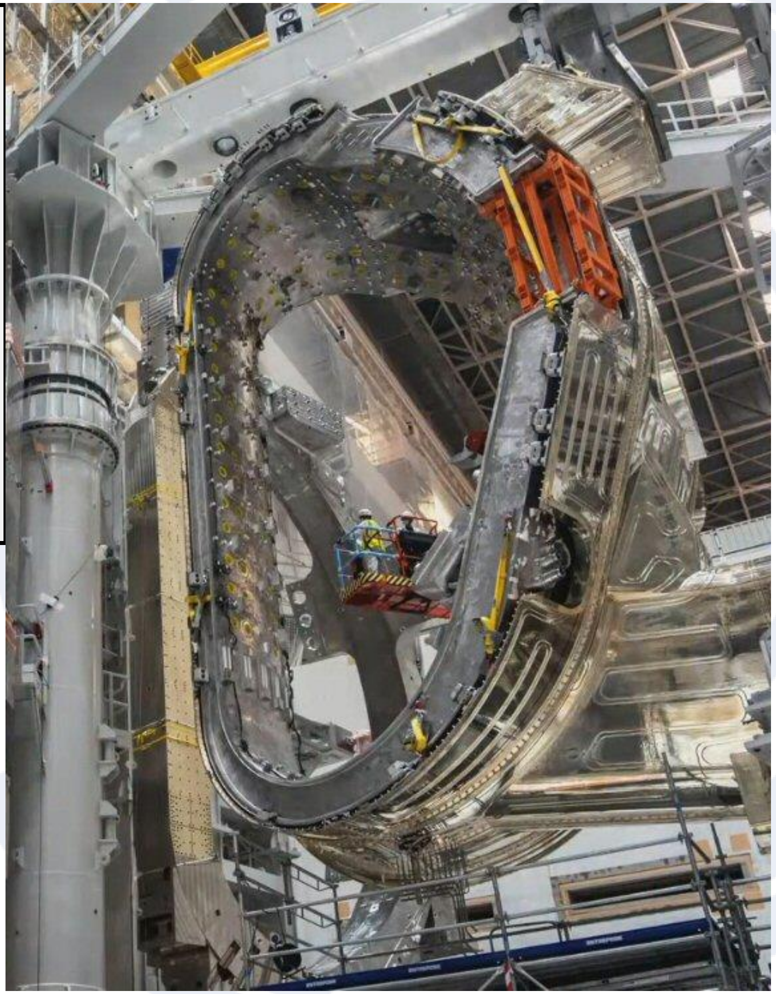

from Takayashi, Y., 23rd IAEA conference. 2010

# 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

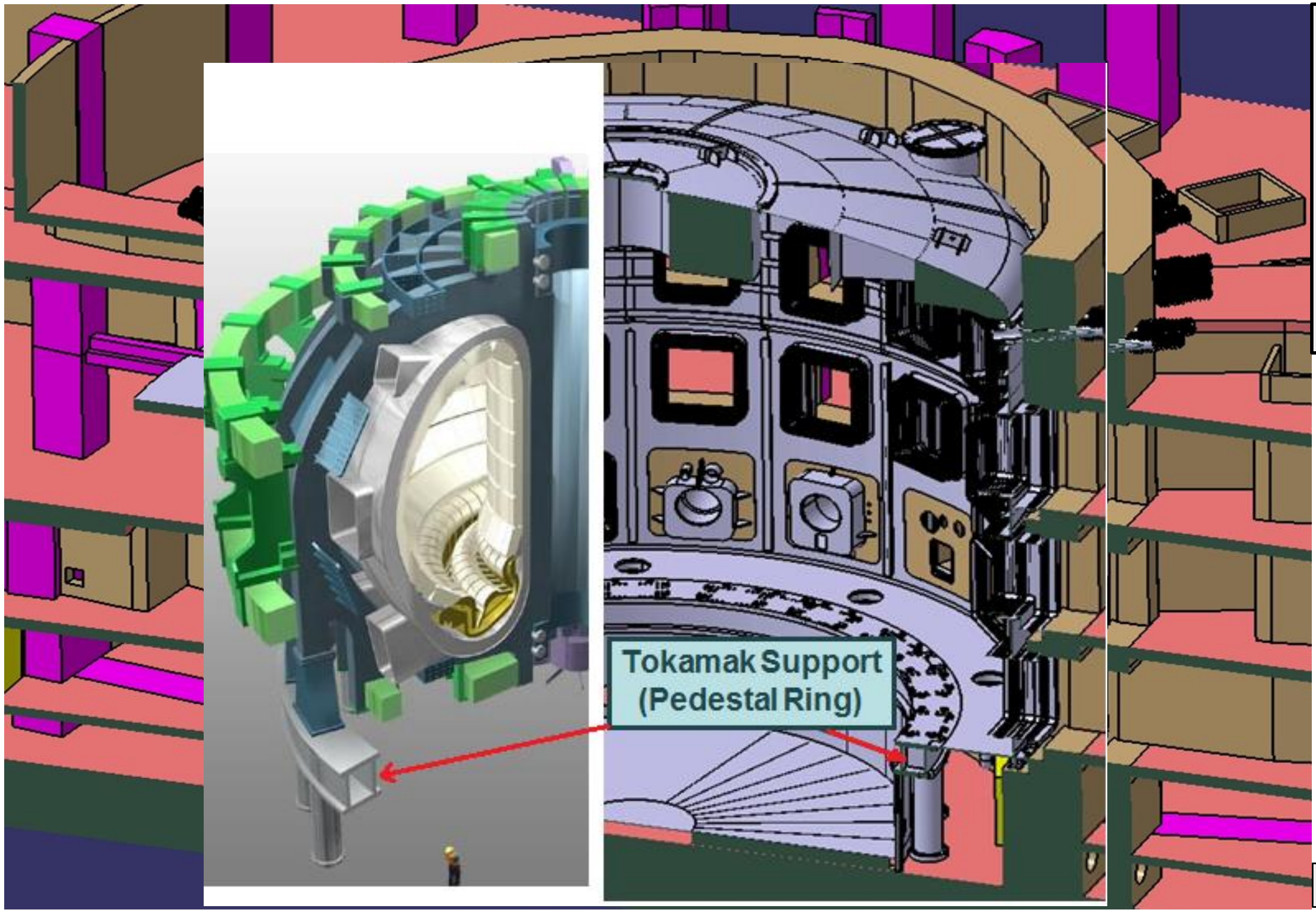



- Double-wall fully welded pressure vessel (stainless steel).
- Actively-cooled with pressurized water.
- Port structures with removable port plugs and port closure plates.
- Nuclear component, highest level of qualification practically excludes failure.
- Supports the in-vessel components



from iter.org

# Cryostat in the building

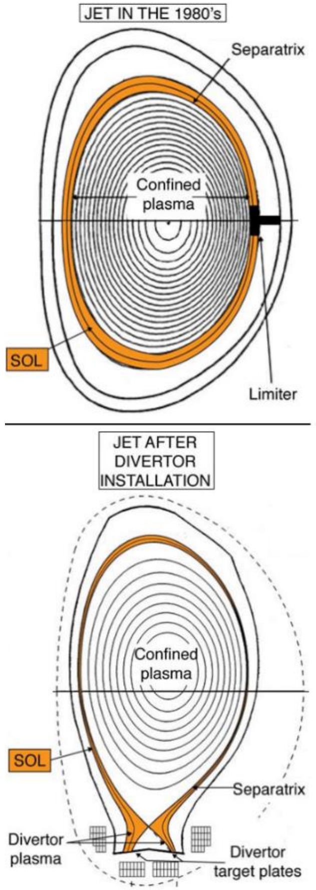



ITER cryostat

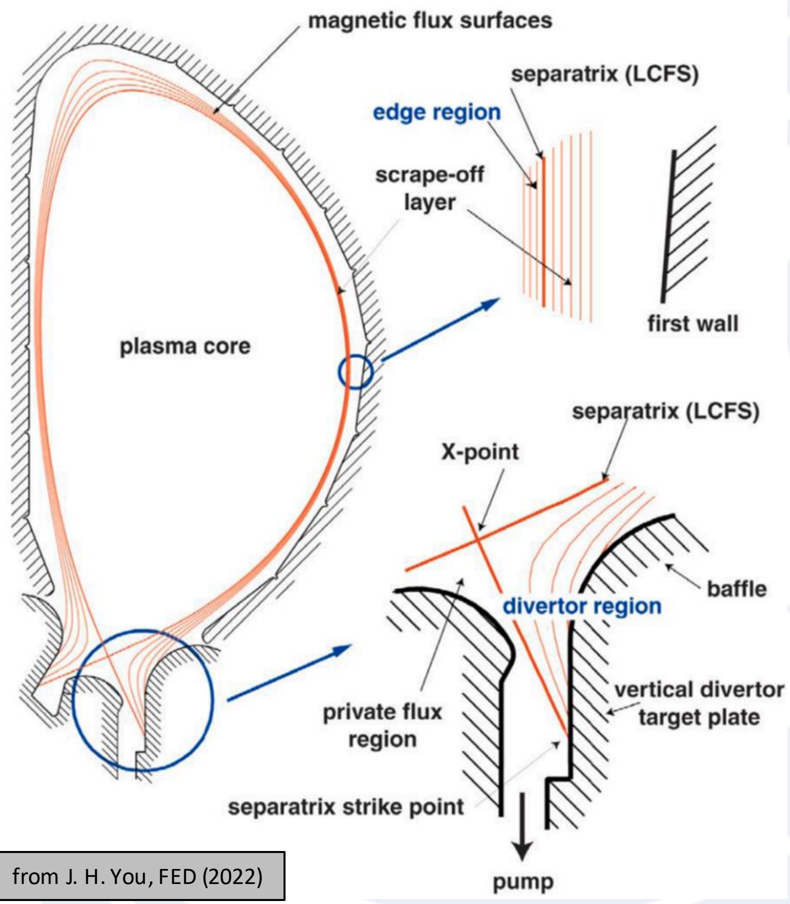
- Conventional vacuum chamber (except its size) to thermally insulate the magnet coils
- Supports the tokamak.
- Inside the bioshield.
- Final fabrication on site.

# Wall protection and plasma-facing components

# Power exhaust



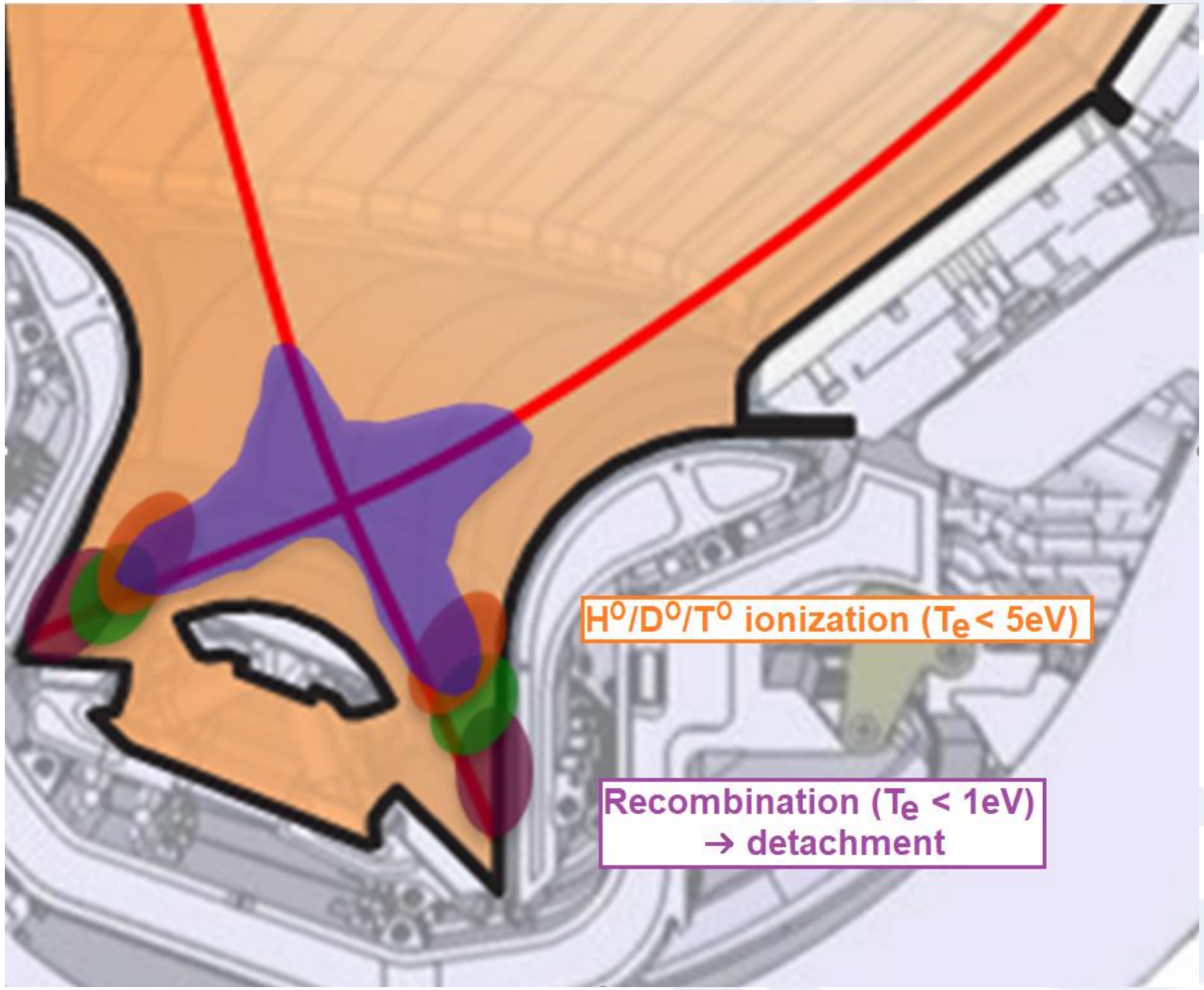

- Charged particles escape from the plasma confinement (mostly on the outboard side) and enter the so-call scrape-off layer: SOL
- They travel along the poloidal magnetic flux lines until they impact on the part of the wall intersecting the SOL generating heat and causing erosion → the wall is coated 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.
- Toroidally continuous divertor target plates at shallow angles to the separatrix cause flux expansion.




from J. H. You, FED (2022)

# Divertor detachment

- In medium-power machines neutrals reflected from the target plates generate a cloud above the divertor, which the incoming particles cannot penetrate and hence radiate their power → detached divertor.
- In high-power machines like ITER the high-energy incoming particles suppress the cloud → injection of neutral gases. (too much causes impurities in the plasma and can trigger a plasma disruption).
- Divertor heat loads are a size driver for the machine!



# Disruptions

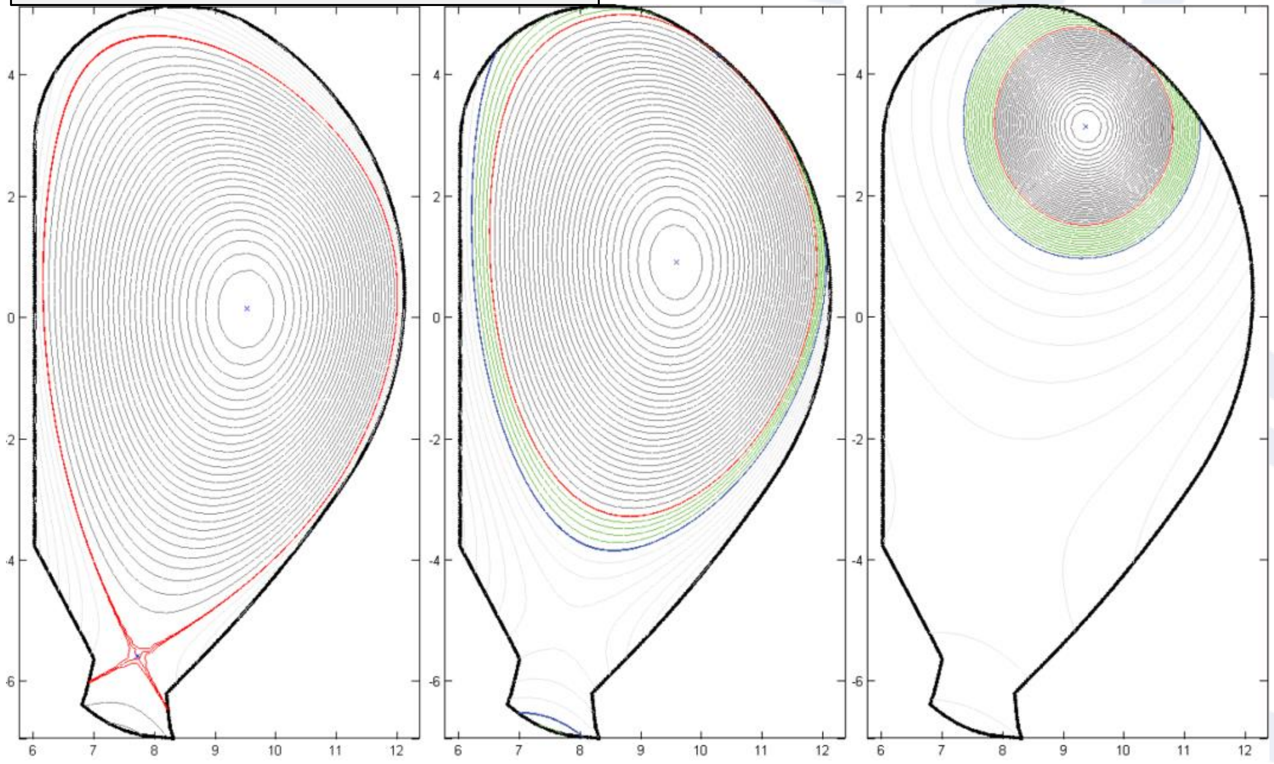
Plasma disruptions occur for various reasons, e.g. impurity influx, loss of confinement, error in the active control system.

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).

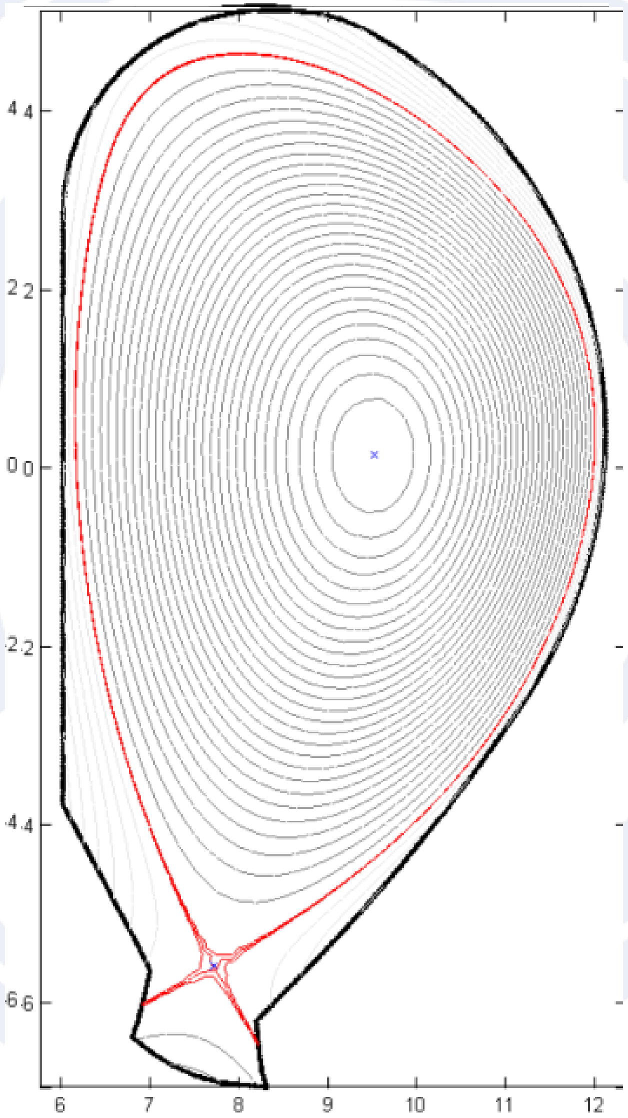
Induced currents in ITER blanket:


Vertical displacement event in DEMO:



# Protection of the first wall


- <u>During flat top</u> the plasma does not contact the wall except the divertor target plates.
- The ITER FW panels are foreseen to touch the plasma <u>during plasma transients</u>, i.e., ramp-up/down and disruptions.
- <u>During flat top</u>, the edges of those panels in the area of the null point (lower and upper) are also subject to particle impact.
- The ITER FW panels are made of CuCrZr and have high heat flux capability. They must be well aligned (±5-10mm) to prevent particle impact concentrated on leading edges.



# Technology of plasma-facing components (PFCs)

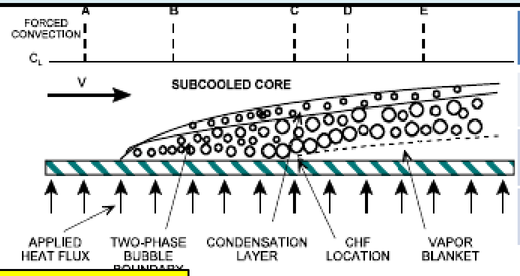

High heat flux on the plasma facing wall (in ITER ~0.1-0.2 MW/m² due to radiation):
→ The entire wall must be covered with PFCs.

Long plasma pulses:
→ steady-state heat flux, active cooling required for all PFCs.

Charged particles locally increase the heat flux significantly:
- Divertor strike area: 10-20 MW/m²
- First wall: 2-5 MW/m² (ITER)

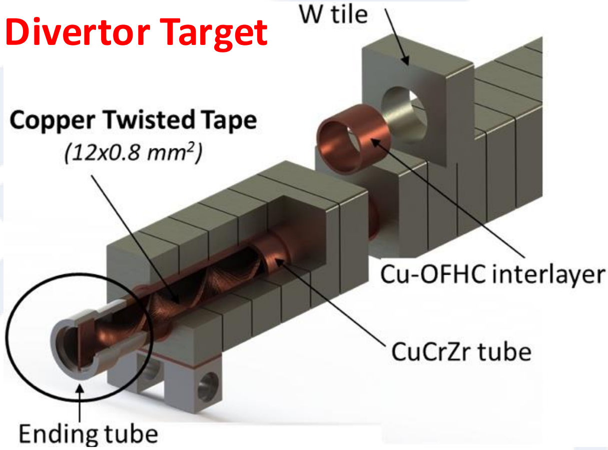
→Large temperature gradients, high surface temperatures

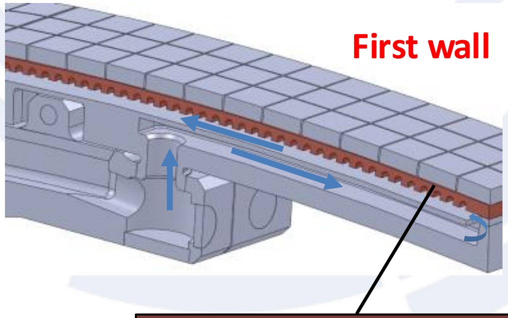
ITER design choices:
- Water is used as coolant making use of phase change.
- Copper alloys are used as heat sink structure.




from iter.org

| | Thermal conductivity, k |
|---|---|
| CuCrZr | ~330 W/mK |
| Steel | ~20-40 W/mK |
| Tungsten (not a structural material) | ~130 W/mK |

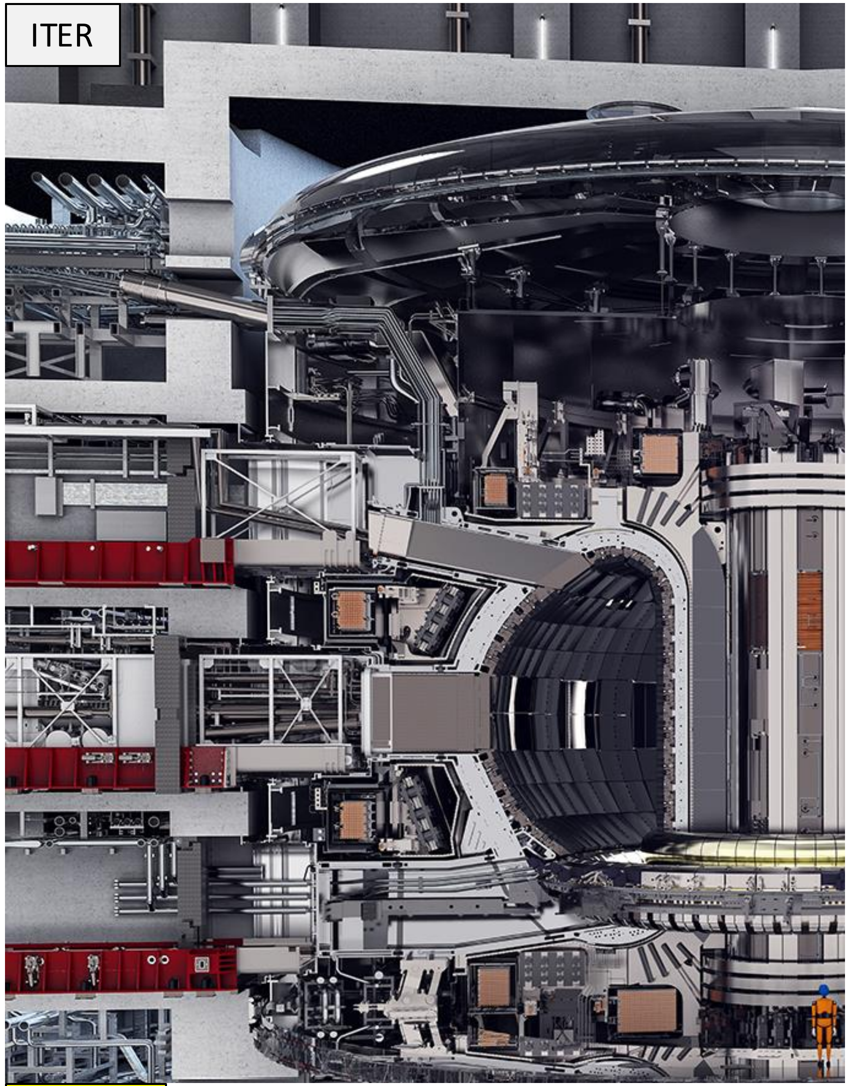



CuCrZr hypervapotron structure for turbulence enhancement

# Radiation shielding

# Radiation shielding




from iter.org

A D-T fusion plasma generates an enormous number of neutrons, with consequences:

→**Heating** (affected: in-vessel components, cryogenic magnets)

→**Material damage** (affected: structural materials of in-vessel components, functional materials even outside the bioshield)

→**Material activation** (affected: maintenance access, sustainability of fusion energy)

**Interaction of neutrons with matter generates secondary gammas that must be shielded as well**

<u>Three main types of neutron and shielding structures:</u>

1. In-vessel components (IVCs) and vessel (VV), ~1m of steel/water mixture
2. Port plugs
3. Bioshield (to allow access of personnel outside the bioshield, see next slide)

# Bioshield

ITER bioshield


from iter.org

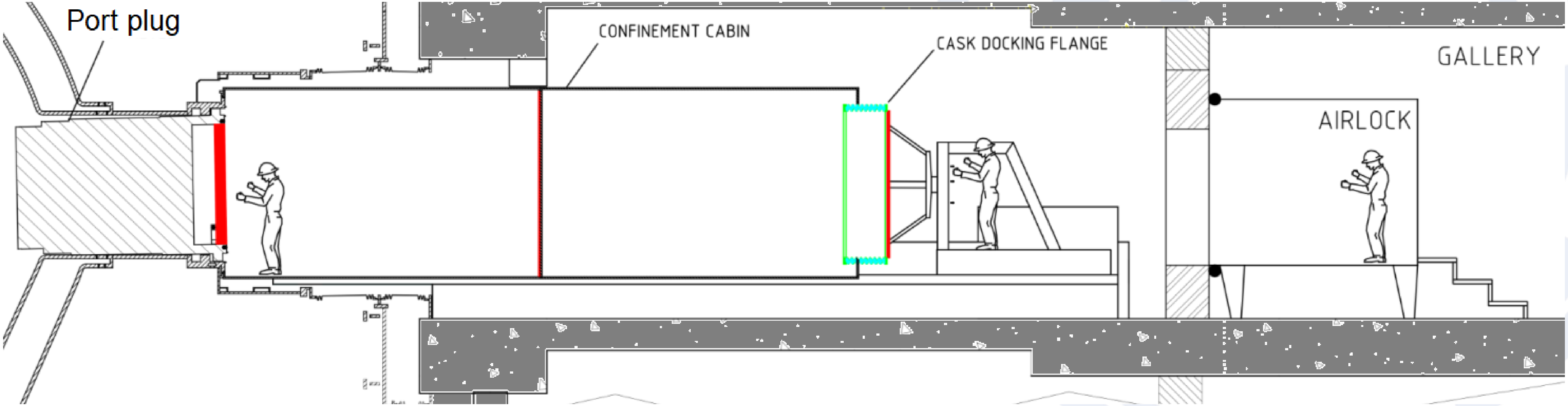
Numerous large openings in the bioshield required for access to the various tokamak systems impairs its radiation shielding function.

→ Bioshield plugs (with penetrations).

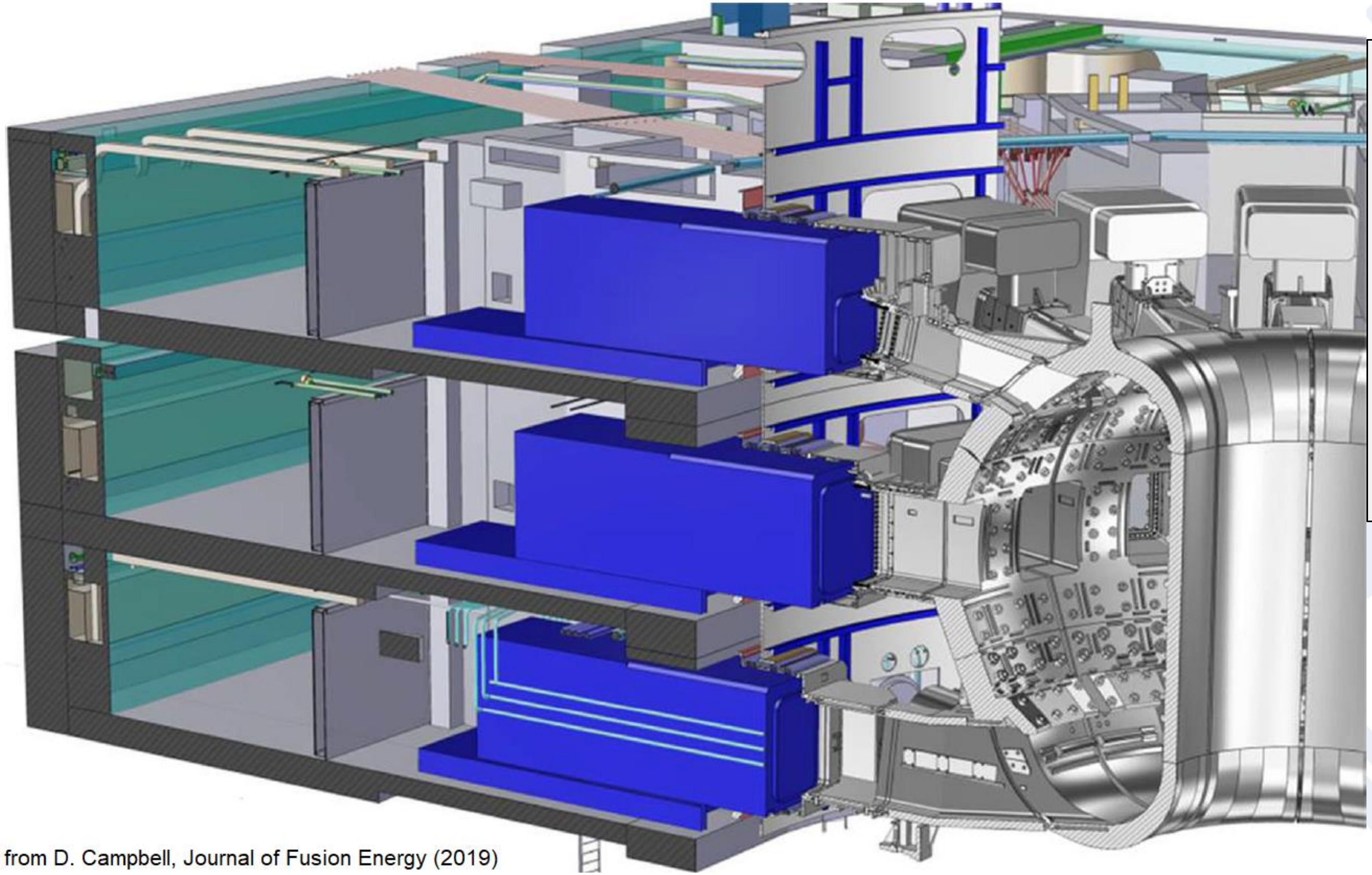
# Protection of personnel

- When workers access the VV they must 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:



# Contamination control during in-vessel maintenance



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

- During 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.

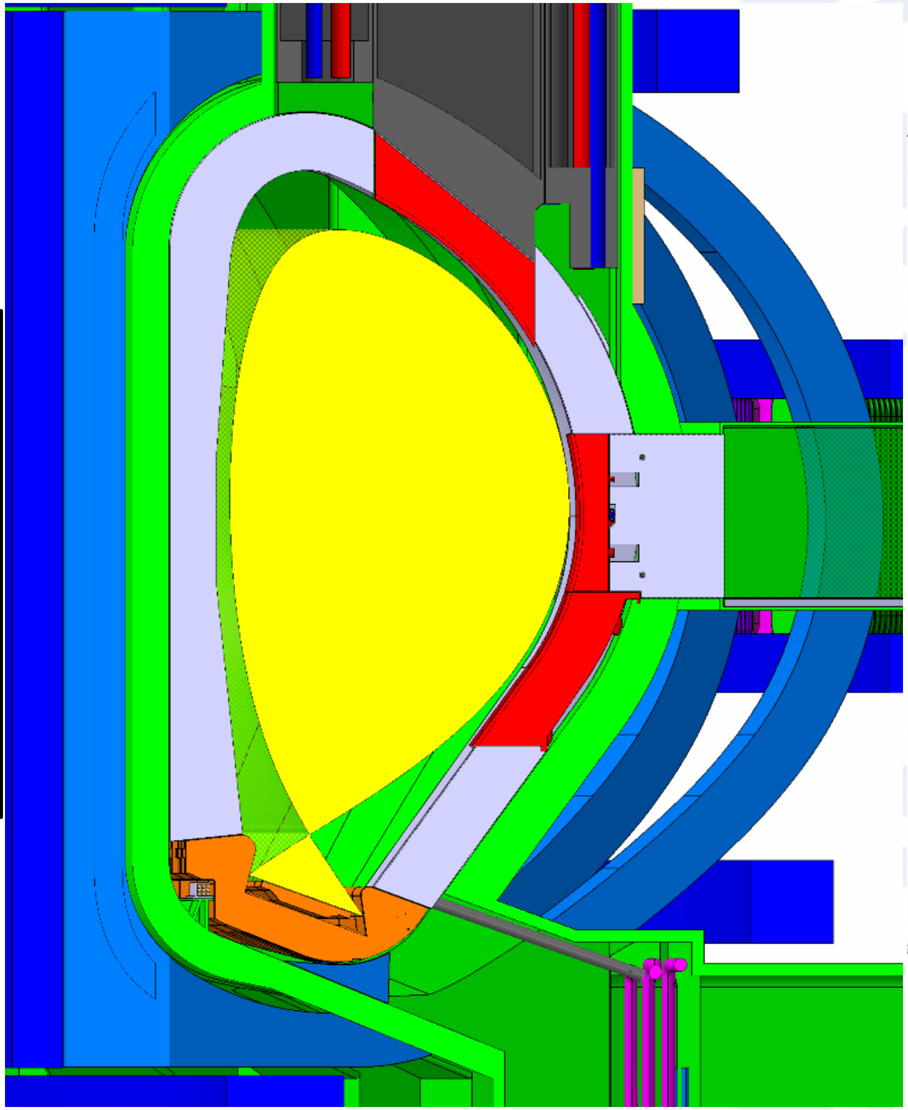
→ So-called casks are docked to the VV before the port closure plate is opened.

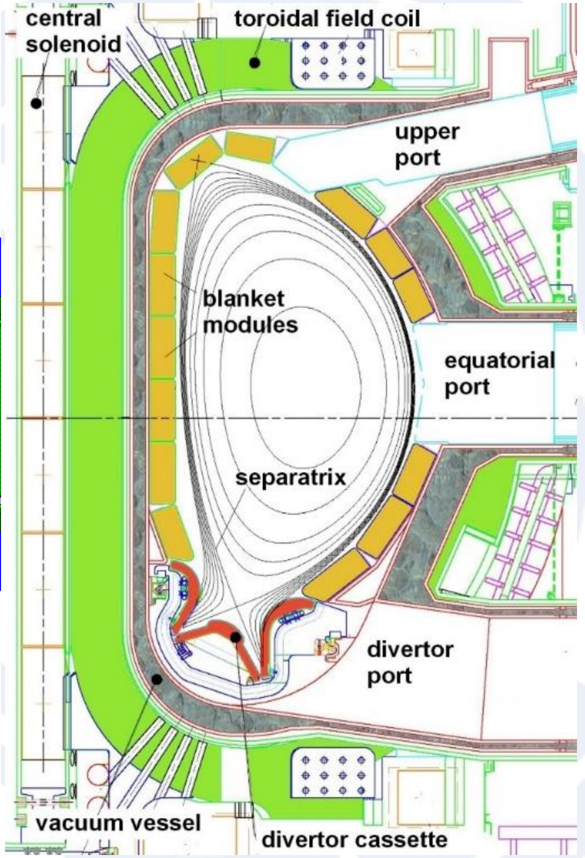
→ The air in the room hosting the cask is circulated to the detritiation system.

# From ITER to DEMO

# Differences between DEMO and ITER

a) Demonstration power plant: 300-500 MW net electric → higher fusion power
b) High neutron fluence (7 instead of 0.5 full power years) → irradiation-resistant materials required
c) Pulse length (2hrs rather than 400s)
d) Tritium breeding blanket
e) Heat conversion into electricity



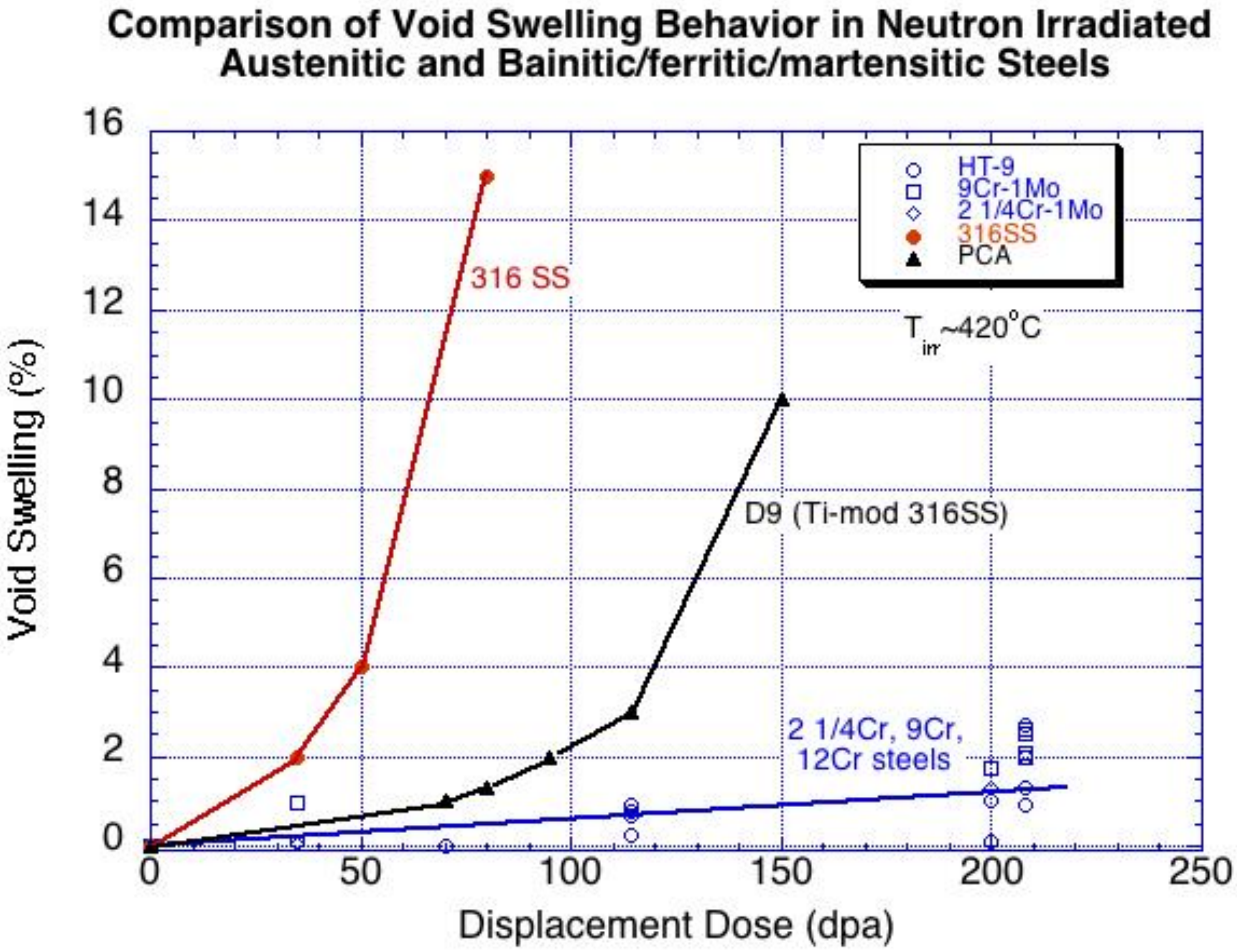


# Irradiation resistant material for in-vessel component

<u>Lifetime neutron fluence:</u>

- ITER: ~1-5 dpa,
- DEMO: ~70 dpa,
- Fusion power plant: ~500-1000 dpa

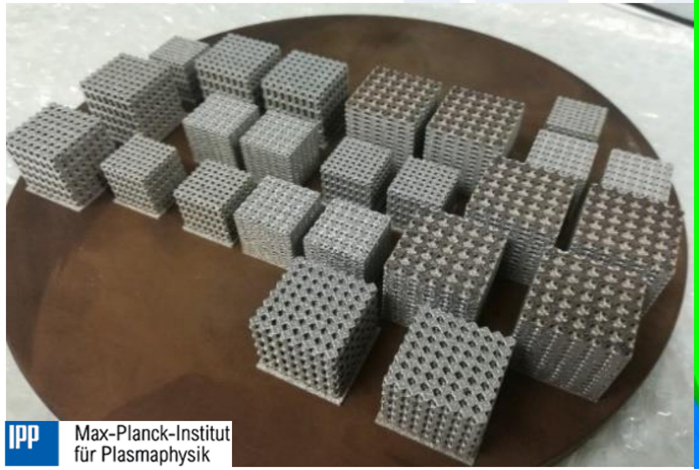
1) **Breeding blanket** made of ***Eurofer***, must be replaceable
2) **Cu-based PFCs** have moderate lifetime (< 10-20 dpa, i.e. 1-2 fpy)
   → Quick replacement
   → Only in **divertor** and plasma **limiters**

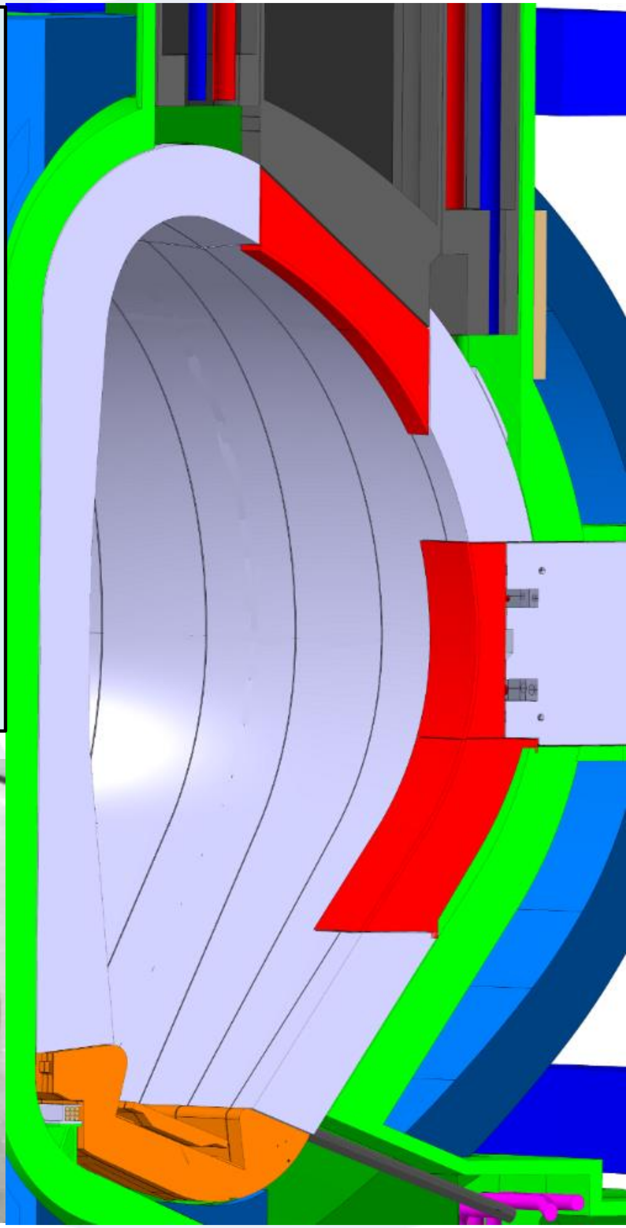


# Protection of the first wall


- The ITER first wall touches the plasma during plasma transients, i.e. ramp-up/down and disruptions.
- CuCrZr with high heat flux capability (up to 5 MW/m²).

- The DEMO breeding blanket first wall is made of steel (Eurofer) with moderate heat flux capability (up to ~1 MW/m²).
- It is protected from charged particles by protruding limiters [6].
- Limiters require PFCs of a new type: Extreme heat loads on the surface shall not be conducted to the cooling structure. At the same time moderate steady-state heat loads must be tolerated.



# Breeding blanket

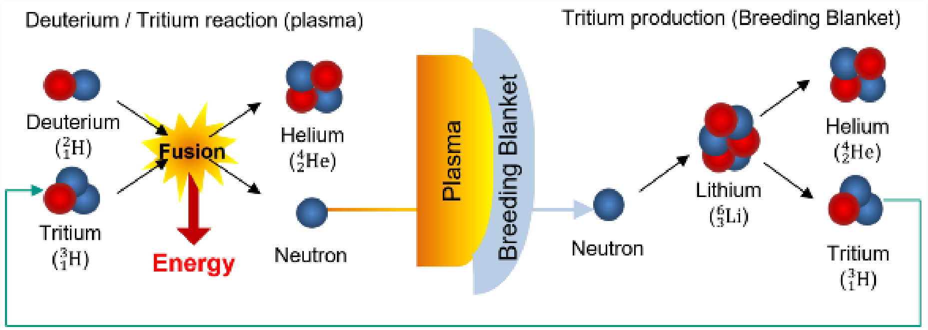
Why do we need tritium self-sufficiency?

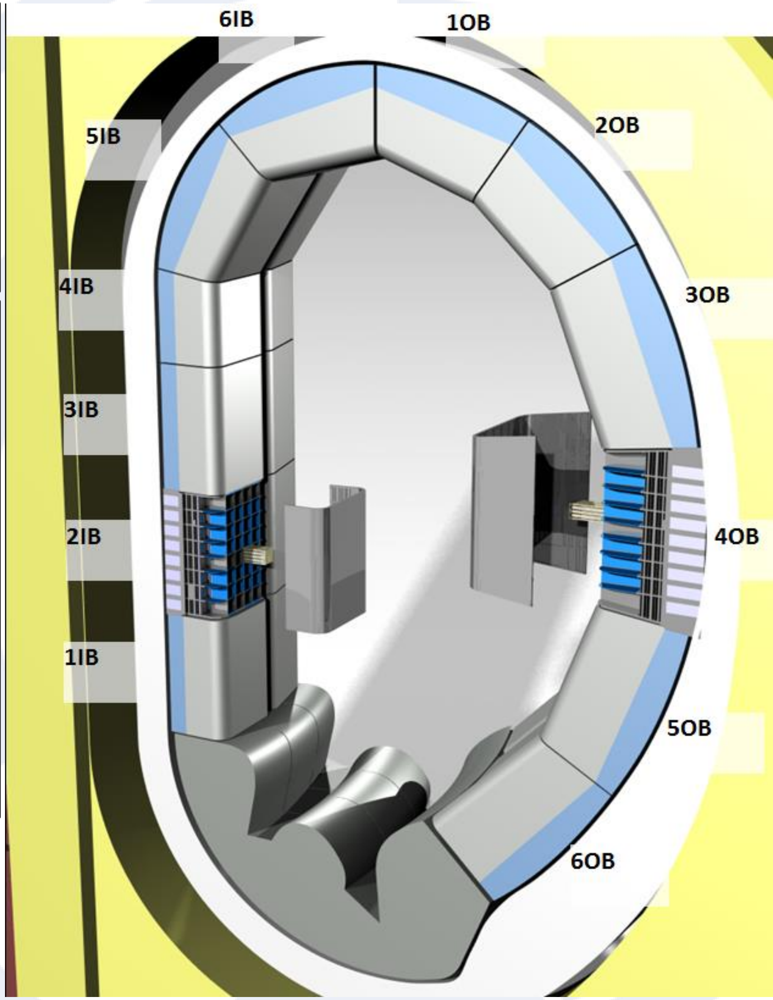
Consumption: $P_{fus}$ ~2000 MW, 30% availability (DEMO)→ **33 kg T/year → 112 kg T/FPY**

Cost of Tritium: **≈30 M€/kg**

Breeding blanket overview :

- Banana-shaped boxed made of Eurofer → BB segments
- Box is filled with breeder, Lithium (preferably enriched in $^6$Li) and neutron multiplier materials (Be, Pb): n + Li → He + T
- 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.

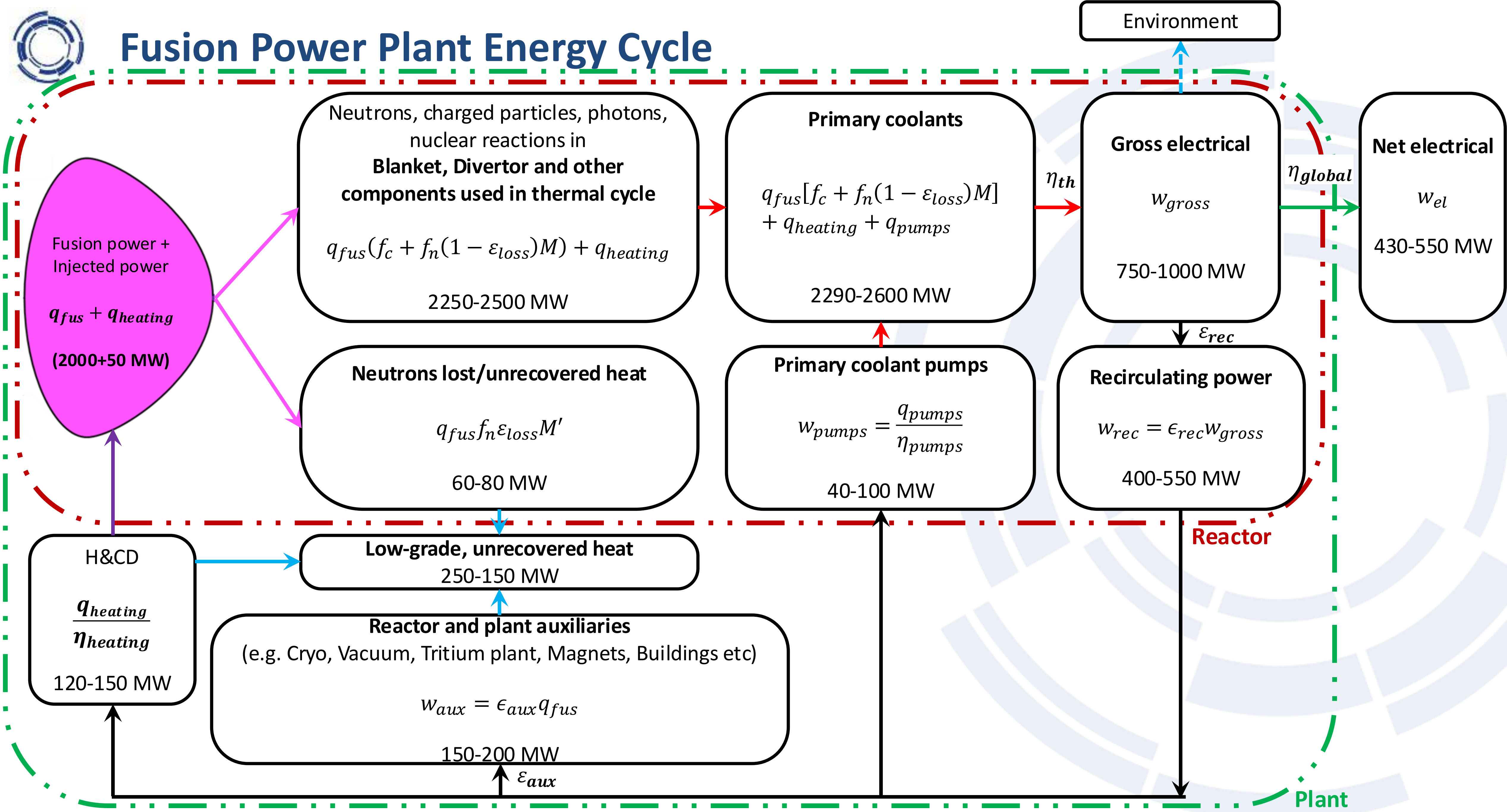



Courtesy: KIT

# Fusion Energy Conversion Cycle and DEMO Balance of Plant Solutions

# Fusion Power Plant Energy Cycle



# Fusion Power Plant Energy Cycle

Calculate the numbers (exam)

**The total heat added to the thermal cycle is**

$$q_{in} = q_{heating} + q_{fus}[f_c + f_n M(1 - \varepsilon_{loss})] + q_{pumps}$$

**The gross electric output at the generator is**

$$w_{gross} = \eta_{th}\{q_{heating} + q_{fus}[f_c + f_n M(1 - \varepsilon_{loss})] + q_{pumps}\}$$

**The net electric output is**

$$w_{net} = w_{gross} - w_{rec} = \eta_{th}[q_{heating} + q_{fus}[f_c + f_n M(1 - \varepsilon_{loss})] + q_{pumps})] - \frac{q_{pumps}}{\eta_{pump}} - \frac{q_{heating}}{\eta_{heating}} - w_{aux}$$

**The global efficiency is**

$$\eta_{global} = \frac{w_{net}}{q_{fus}(f_c + f_n M)}$$

**Remembering that**

$$Q = \frac{q_{fus}}{q_{heating}}$$

**Assuming that pumping power and auxiliary plant power are proportional to the carried power and the fusion power, respectively.**

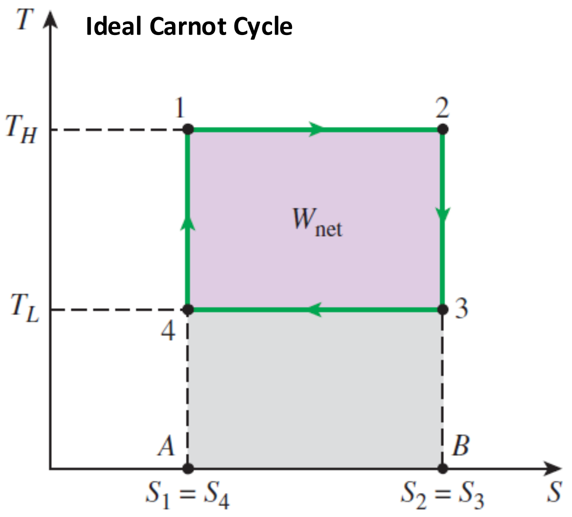
$$q_{pumps} = \epsilon_{pump}\{q_{heating} + q_{fus}[f_c + f_n M(1 - \varepsilon_{loss})]\} \qquad w_{aux} = \epsilon_{aux} q_{fus}$$

**The global efficiency can be easily rearranged as follow**

$$\eta_{global} = \overbrace{\eta_{th}(1 + \epsilon_{pump})\left(1 + \frac{1}{Q(f_c + f_n M)}\right)}^{\text{Gain due to the total heat injected into the thermal cycle}} - \overbrace{\frac{\eta_{th}\epsilon_{loss} f_n M}{(f_c + f_n M)}(1 + \epsilon_{pump})}^{\text{Fusion heat lost/unrecovered}} - \overbrace{\frac{\epsilon_{pump}}{\eta_{pump}}\left(1 + \frac{1}{Q(f_c + f_n M)} - \frac{\epsilon_{loss} f_n M}{(f_c + f_n M)}\right)}^{\text{Recirculating pumping power}} - \overbrace{\frac{1}{\eta_{heating} Q(f_c + f_n M)}}^{\text{Recirculating additional heating power}} - \overbrace{\frac{\epsilon_{aux}}{(f_c + f_n M)}}^{\text{Other auxiliaries}}$$

# Steam Rankine Cycle: old but gold

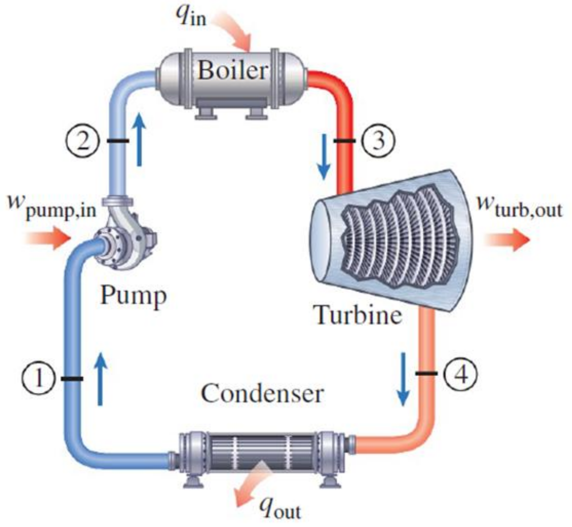
Adapted from *V. Dostal, "A supercritical Carbon Dioxide Cycle for Next Generation Nuclear Reactors", Ph.D. thesis, MIT, 2004*

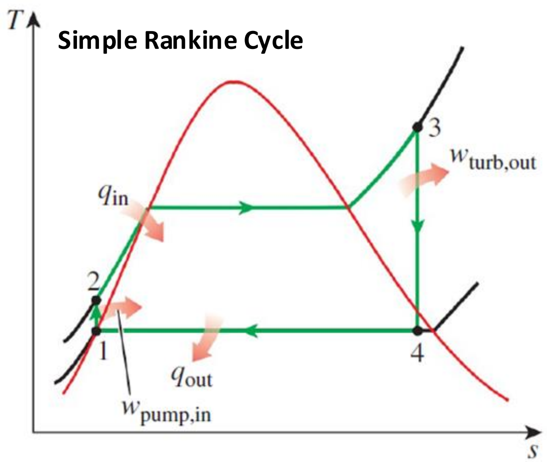



$$\eta_{CARNOT} = 1 - \frac{T_L}{T_H}$$

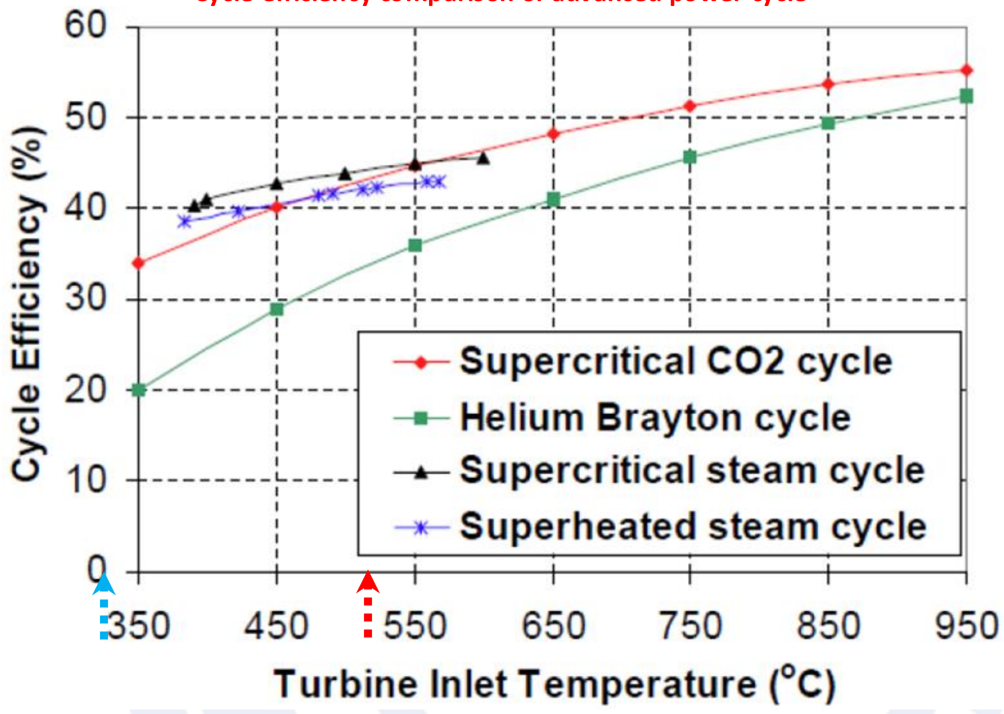
$$\eta_{Cycle} = \frac{w_{out} - w_{in}}{q_{in}} = 1 - \frac{q_{out}}{q_{in}} = 1 - \frac{\overline{T_L}}{\overline{T_H}}$$

Where $\overline{T} = \frac{\int T dS}{\int dS}$




Cycle efficiency comparison of advanced power cycle



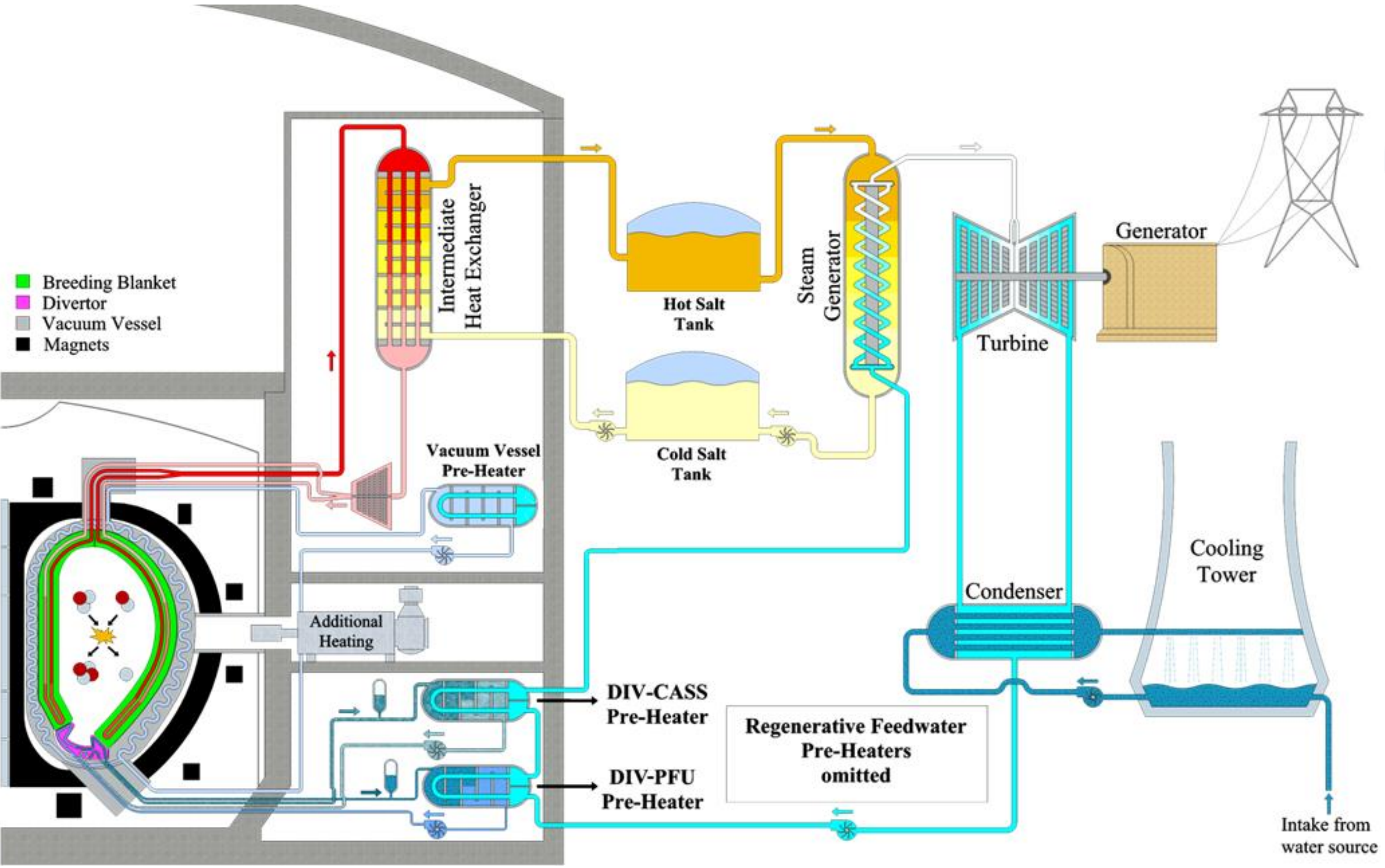

**Max temperature of coolant at blanket outlet:**

- **Helium-Cooled: 520 °C** → Limit mainly imposed by EUROFER mechanical properties
- **Water-Cooled: 328 °C** → Limit imposed by water inherent properties

**For practical max. temperature, steam cycles are still the most favorable power cycles to be used.**

# Simplified concept scheme for HCPB BoP

- **BB PHTS** coupled with **IHTS** that stores energy
- **Divertor+Limiters** PHTSs directly connected to the **PCS**
- **IHTS** delivers thermal power to the **PCS** continuously with the aim of almost constant electric power output




**BB PHTS**
He at 8 MPa,
$T_{in}/T_{out}$ = 300/520 °C

**IHTS**
HITEC at ~0.8 MPa,
$T_{in}/T_{out}$ = 270/465 °C

**DIV+LIM PFCs PHTS**
$H_2O$ at 5.0 MPa,
$T_{in}/T_{out}$ = 130/136 °C

**DIV+LIM shield PHTS**
$H_2O$ at 3.5 MPa,
$T_{in}/T_{out}$ = 180/210 °C

**PCS**
Dual Rankine Steam Cycle
~12.5 MPa, $T_{max}$ = ~445 °C

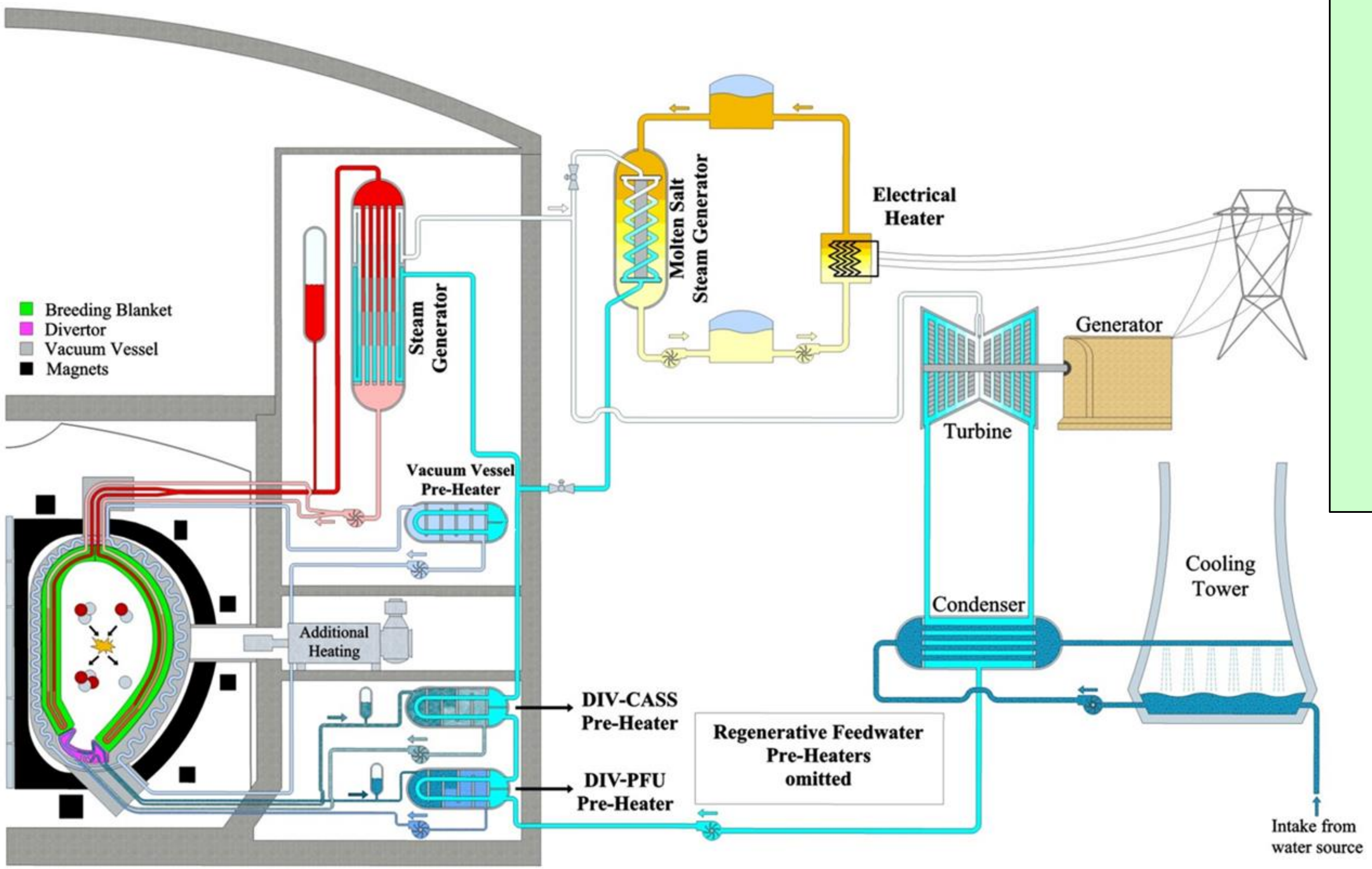
**$\eta_{th}$= ~ 36-37% (BoP load only)**

# Simplified concept scheme for WCLL-BoP

- **All PHTSs** directly connected to PCS
- **Adoption of two tanks**
- **Heating system based on electrical heaters**

**During DWELL:**

- OTSGs are idle
- Small storage → $Q_{DWELL}$ =10%$Q_{pulse}$




**PULSE**

**BB PHTS**
$H_2O$ at 15.5 MPa,
$T_{in}/T_{out}$ = 295/328 °C

**DIV+LIM PFCs PHTS**
$H_2O$ at 5.0 MPa,
$T_{in}/T_{out}$ = 130/136 °C

**DIV+LIM shield PHTS**
$H_2O$ at 3.5 MPa,
$T_{in}/T_{out}$ = 180/210 °C

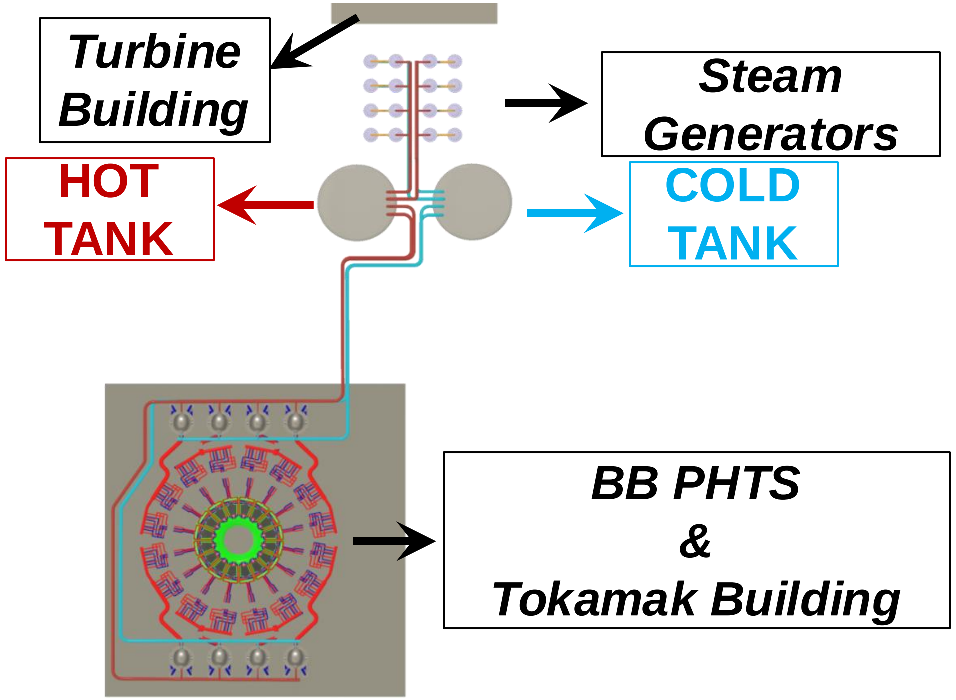
**DWELL**

**IHTS+ESS**
HITEC at ~0.1 MPa,
$T_{in}/T_{out}$ = 282/330 °C

**PCS**
Rankine Steam Cycle
64.1 MPa, $T_{max}$ = ~299 °C

**$\eta_{th}$= ~ 30-31% (BoP load only)**

# Intermediate Heat Transport System and Energy Storage system




| IHTS+ESS main data for HCPB BoP | |
|---|---|
| Coolant [-] | HITEC |
| Thermal Storage Capacity [MWh] | 426 |
| $T_{cold}/T_{hot}$ tanks [°C] | 270/465 |
| Operating tank pressure [bar] | ~1 |
| Number of loops [-] | 2 |
| Tanks nominal volume [m³] | 2 x 3000 |
| Tanks Size (ΦxH) | 24 x 8 |
| Nominal salt inventory per tank [ton] | 5600 |

**SOLANA (Arizona) Concentrated Solar Plant:**

- 280 $MW_e$
- 4470 MWh
- 12 tanks (6 hot/cold)
- 135000 tons of salt
- Tank size 37 x 10

# Steam turbine challenges at low and very low load operations

Since the ‘70s, Lagun et al. highlighted the critical issues of low-pressure turbines operating at very low flow rates. During continuous low load operations, the steam turbine Last Stage Moving Blades acts like a compressor experiencing high temperatures.

The increase in temperature and the prolonged running time at low load could lead to a series of mechanical issues if not properly evaluated and monitored:

- undesirable restriction of time for the turbine to run with low load
- possible uneven heating of the exhaust casings leading to serious distortion and adverse effects on turbine alignment
- blade tip rubbing
- exhaust sprays continuously activated leading to trailing edge erosion
- lifetime reduction of last stage blades

In NPPs and others “baseload” plants, operations at low loads (<15% rated power) occur not as often as it would be required in DEMO
→ about 11 start-ups/shutdowns per day are foreseen!

Turbine must be designed to guarantee reliable operations avoiding that low load operation jeopardizes the lifetime and plant availability



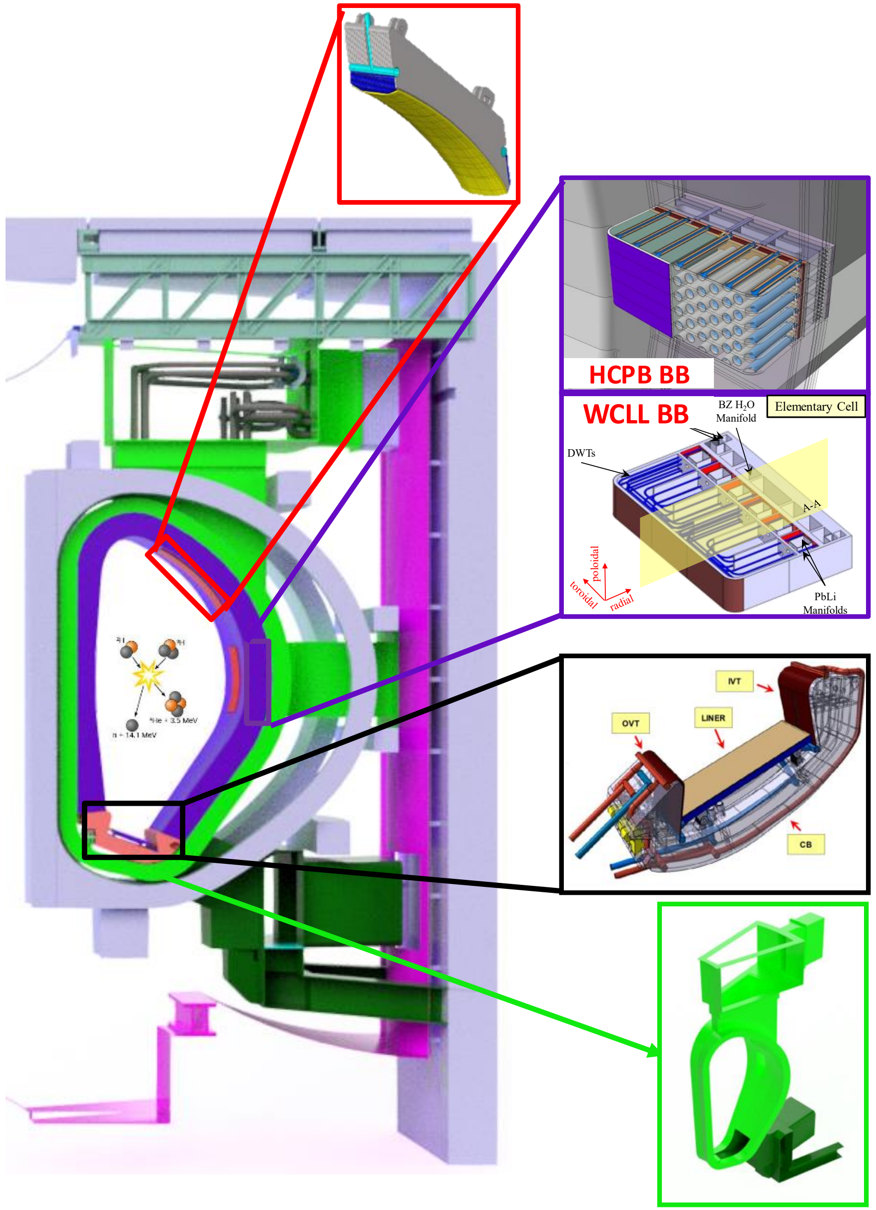
Typical dimensions of Turbine of 700-1000 MW

- Last Stage Blade Length - **from 99 to 144 cm**
- Power train width - **up to 17 m**
- Power train length (w/Generator) - **up to 56 m**

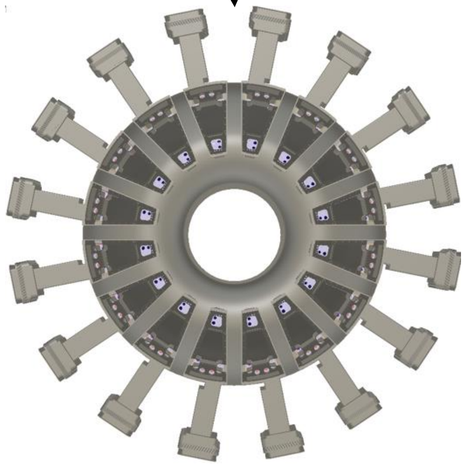
# The design of DEMO Tokamak Coolants System

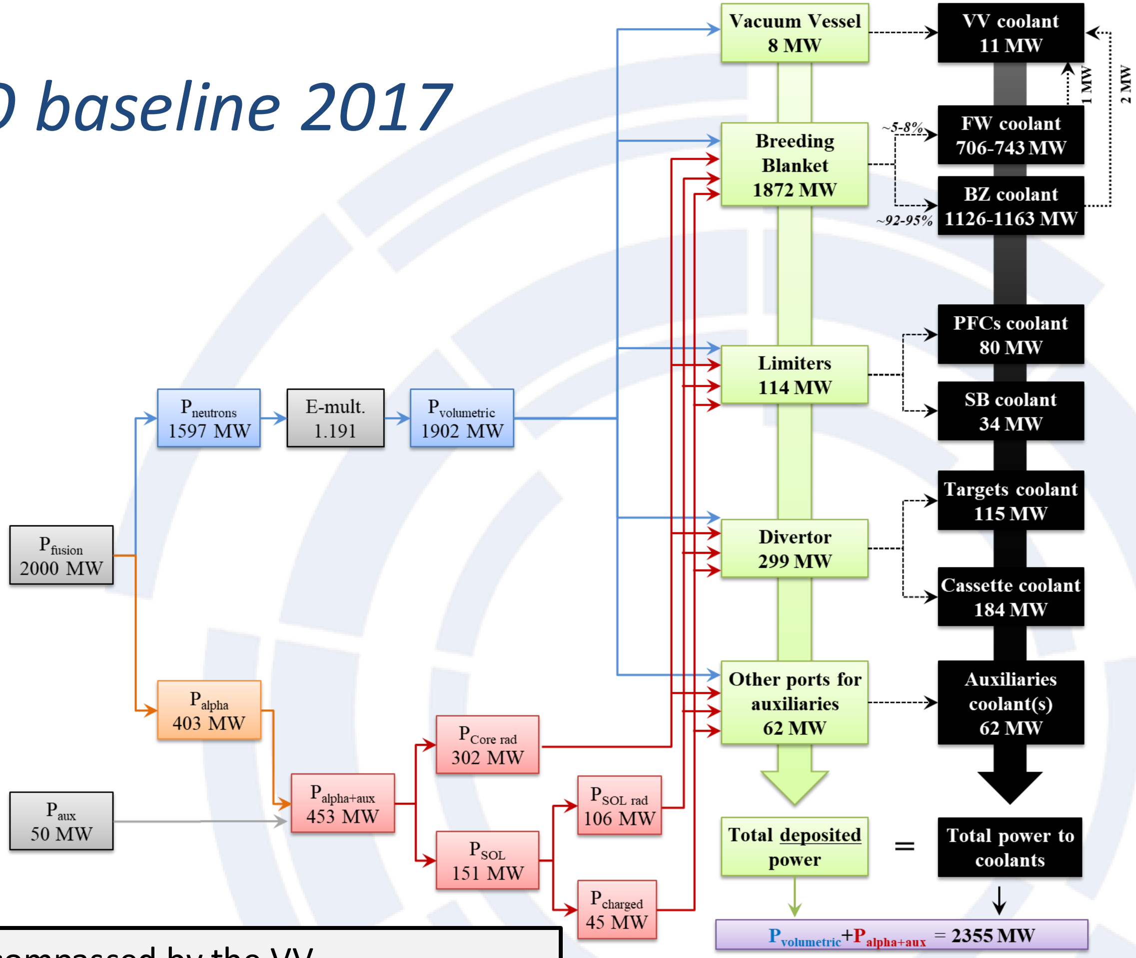
# EU-DEMO In-VV and VV components

## *Thermal power breakdown as for DEMO baseline 2017*




**Tokamak top view with the 16 sectors**



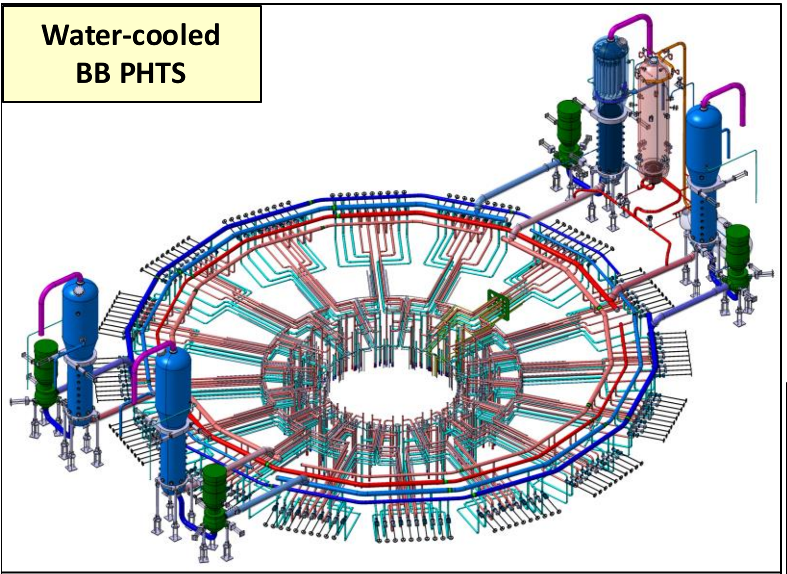



Several In-Vessel components encompassed by the VV.

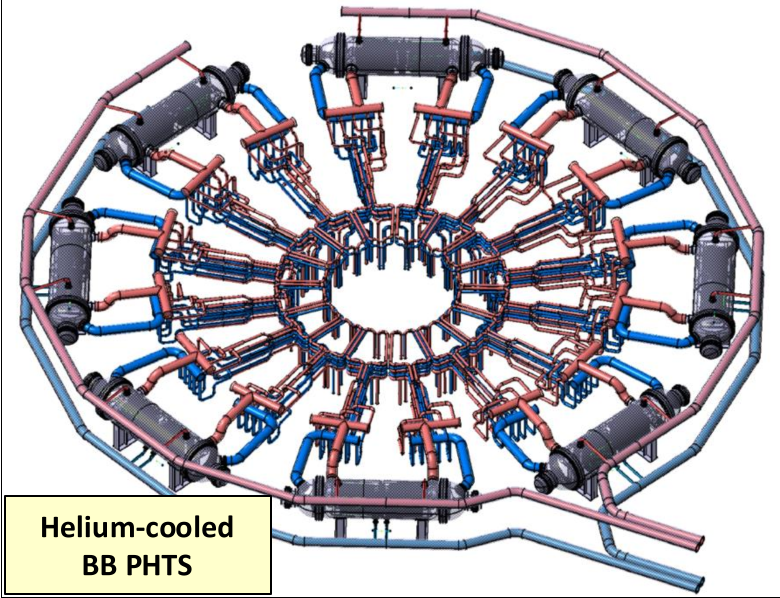
- **Breeding Blanket**: 80 segments (48 outboard BB + 32 inboard BB)
- **Divertor**: 48 cassettes (3 per tokamak sector)
- **Limiters**: 20 discrete limiters are currently considered.
- **Vacuum Vessel**: 16 sectors
- **Other auxiliaries** (additional heating antennas, port plugs etc.)

Due to the different requirements of the clients
a single coolant circuit cannot be adopted to feed the various components.

# Tokamak Coolants System status

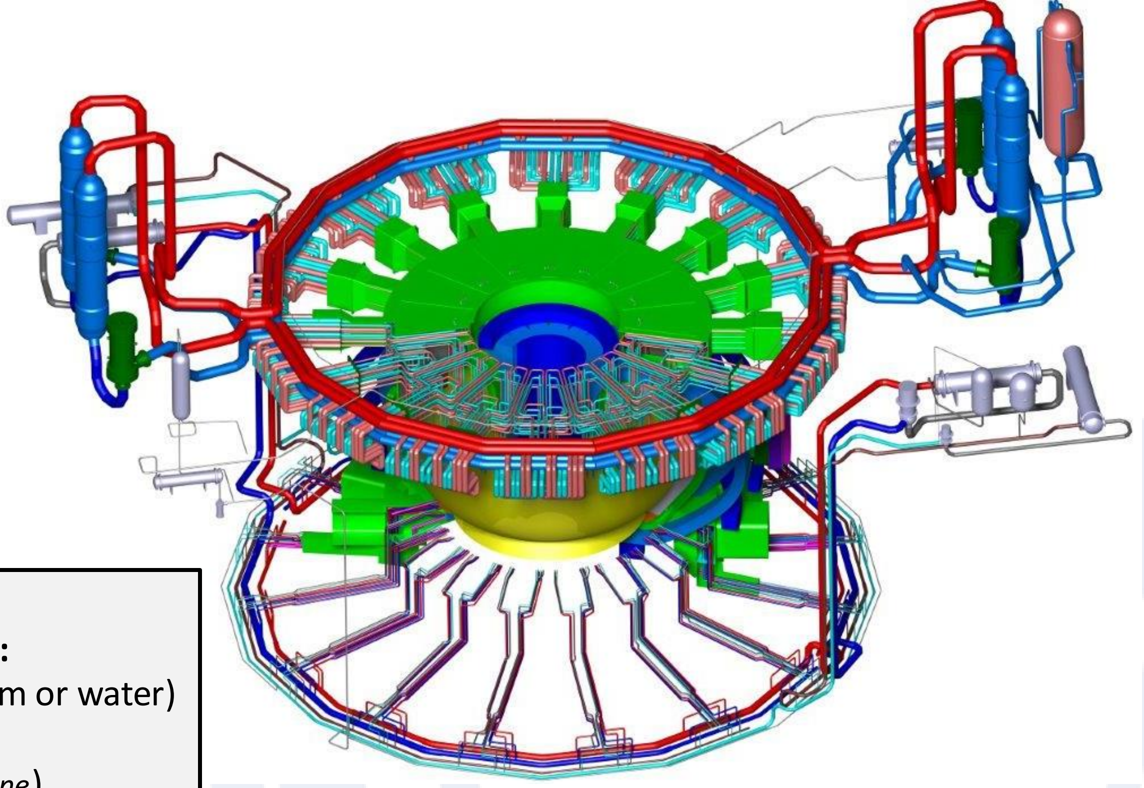


Water-cooled BB PHTS



Helium-cooled BB PHTS

**Tokamak Coolants system**:
**4** main **Primary Heat Transport Systems:**
- **1** for **Breeding Blanket** (either helium or water)
- **2** for **Divertor + Limiters**
  (*1 high heat flux comp. + 1 for shielding zone*)
- **1** for **Vacuum Vessel**

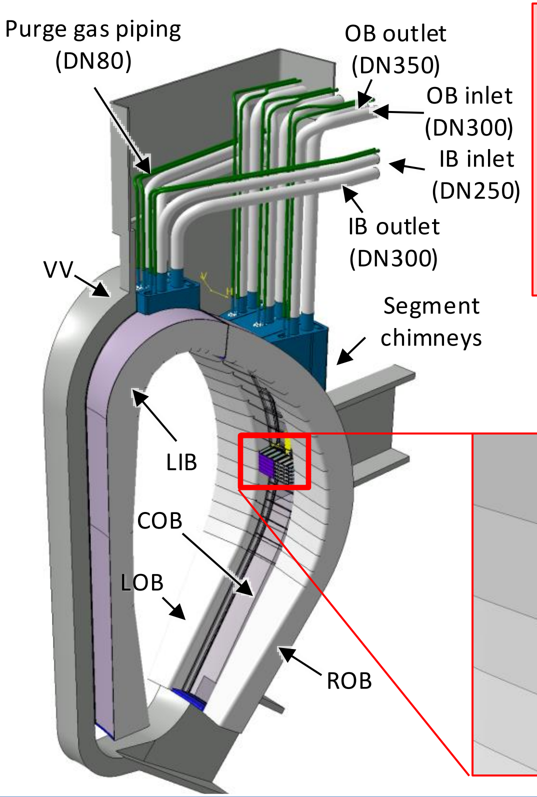


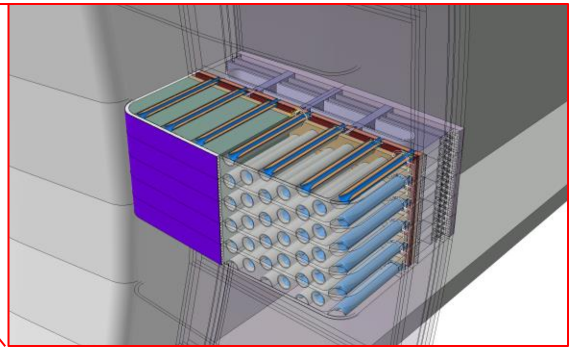
Integration of Tokamak Coolants System for water-cooled Blanket option

| Quantity | Breeding blanket PHTS | | Shielding Components PHTS | High Heat Flux PHTS | Vacuum Vessel PHTS |
|---|---|---|---|---|---|
| | WCLL | HCPB | | | |
| Thermal-power [MW] | 1900 | 2100 | 230 | 240 | 48 (HCPB)/11 (WCLL) |
| Primary volume [$m^3$] | 630 | 1900 | 140 | 280 | 580 (≈500 VV only) |
| Primary temperatures °C | 295-328 | 300-520 | 180-210 (295-328)* | 130-136 | ≈50 |
| Primary pressure [MPa] | 15.5 | 8.0 | 3.5 (15.5)* | 5.0 | 1.3 |
| Mass flow rate [kg/s] | 9754 | 1840 | 1718 | 9394 | TBD |

**Values in () for high temperature option*

# Helium-Cooled Pebble Bed Breeding Blanket PHTS



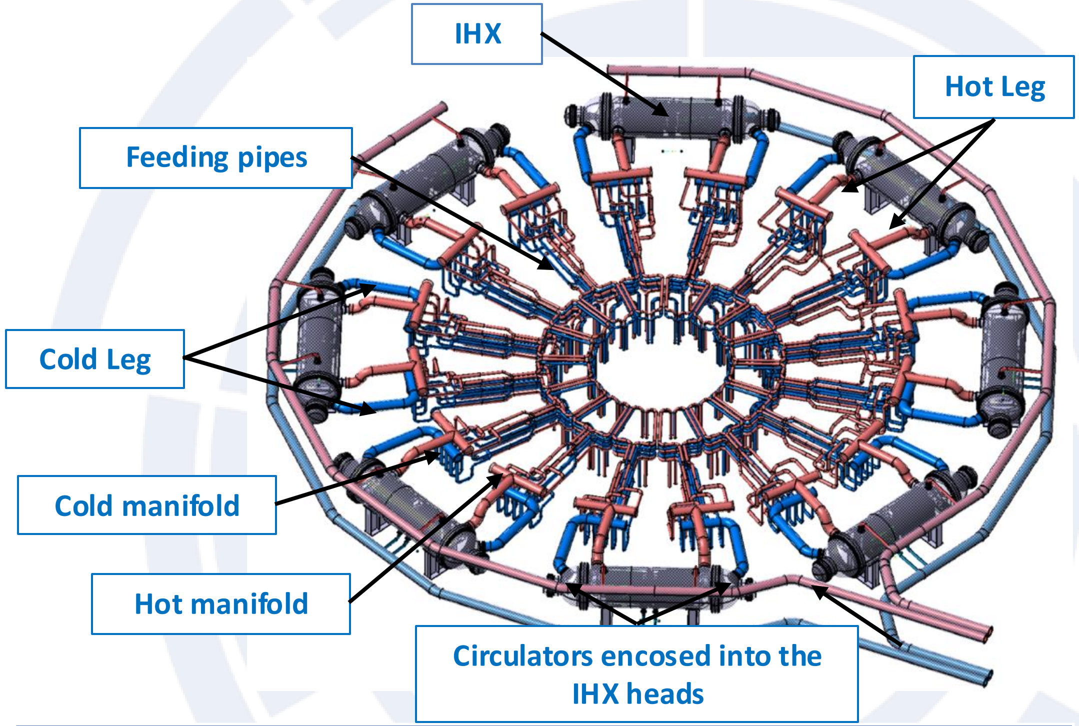



T/H inputs per loop:

- Thermal power: **254 MW**
- $T_{in/out}$: **300/520 °C**
- $\Delta P_{In\text{-}Vessel\ circuits}$: **0.8 bar**

**AISI 316L(N) steel pipes:**

- **DN between 900 and 250**
- **Thickness up to 50 mm**



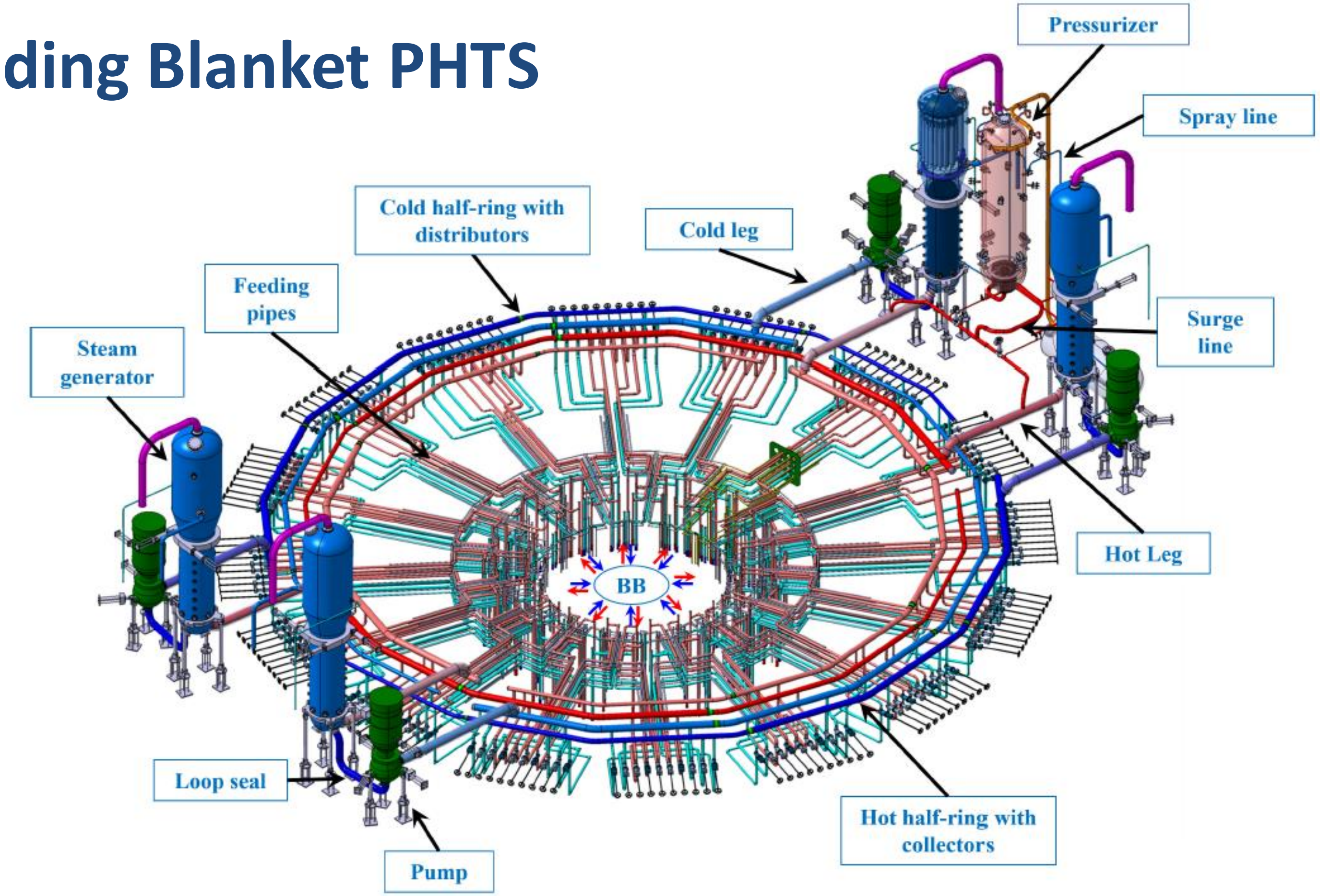

| BB PHTS main components per loop | |
|---|---|
| Hot/Cold feeding pipes | 10/10 |
| Hot/Cold headers | 2/2 |
| Hot/Cold leg | 2/2 |
| Circulators | 2 |
| Heat exchanger | 1 |

| HCPB - BB PHTS main data | |
|---|---|
| Thermal power [MW] | 2029.1 |
| Circulator power [MW] | 92÷91 |
| Total helium volume [m³] | 1882 |
| Total piping length (In+Ex-VV) [m] | 4850 |
| Cooling loops [-] | 8 |

# Water Cooled Lithium Led Breeding Blanket PHTS

**AISI 316L(N) steel pipes:**
- DN between 550 and 100
- Thickness up to 52.37 mm

T/H inputs per loop:
- Thermal power: ~ **955 MW**
- $T_{in/out}$: **295/328 (285/325) °C**




| WCLL BB PHTS main components per loop | |
|---|---|
| Hot/Cold feeding pipes | 80/80 |
| Hot/Cold ring | 2/2 |
| Hot/Cold leg | 2/2 |
| Pump | 2 |
| Heat exchanger | 2 |
| Pressurizer | 1 |

| WCLL BB PHTS main data | |
|---|---|
| Thermal power [MW] | 1884.0 |
| Pumping power [MW] | 26 |
| Total water volume [$m^3$] | 580 |
| Total piping length (In+Ex-VV) [m] | ≈9000 |
| Cooling loops [-] | 2 |

# Pulsed operations in the Tokamak Coolant Systems

# Facing pulsed load and other operational transients

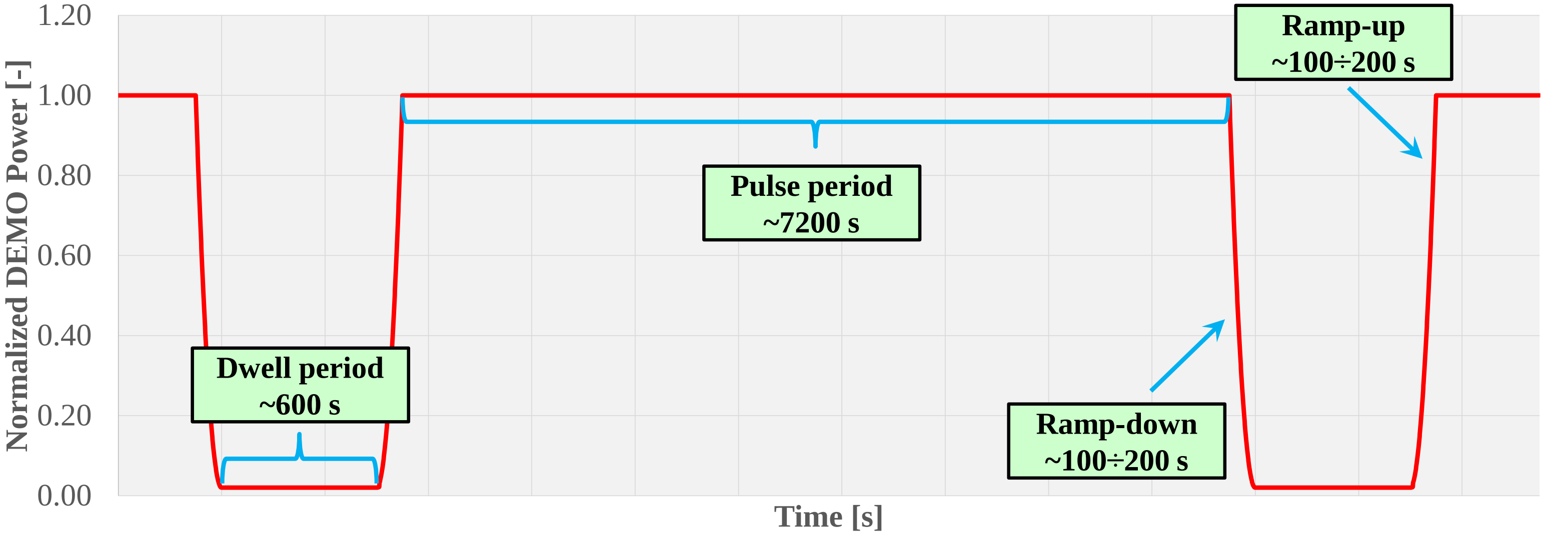
The service life of any coolant system pressure component depends upon a number of factors: material irradiation, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

All components of primary cooling system must be designed to withstand the effects of cyclic loads due to coolant system temperature and pressure changes.
Thermal and loading cycles are introduced by a number of events:

- **Normal conditions**: heatup and cooldown cycles, power changes, normal cyclic variations, baking, … → 100s to $10^6$s cycles
- **Upset conditions:** reactor trip, turbine trip, loss of flow, … → 10s to 100s of cycles
- **Emergency and Faulted conditions**: → small to large LOCAs, feedwater and steam line breaks, pump locked motor… → a few to 10s cycles
- **Test conditions** → primary/secondary side hydrostatic tests, primary/secondary sides leak tests, tube leak tests … → 10s to 100s cycles

The EU-DEMO foresees about **11 pulses per day** with a with **100%-0% reactor load swing up to 1%/s**

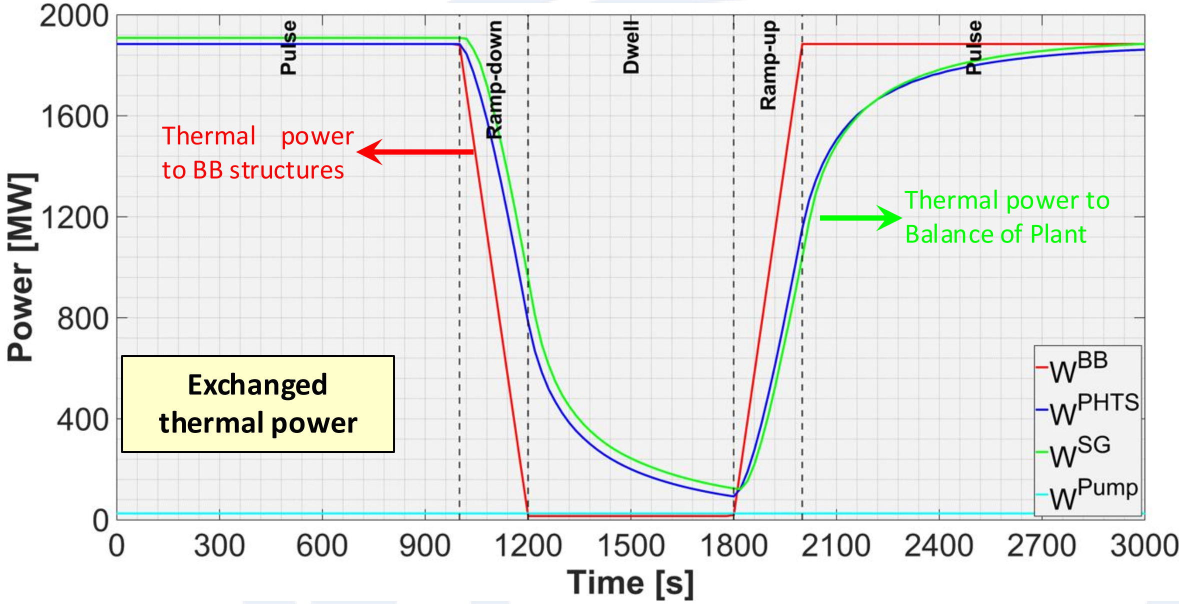
Fission reactors generally consider 1 load variation per day in the range 15% to 100% of full power.

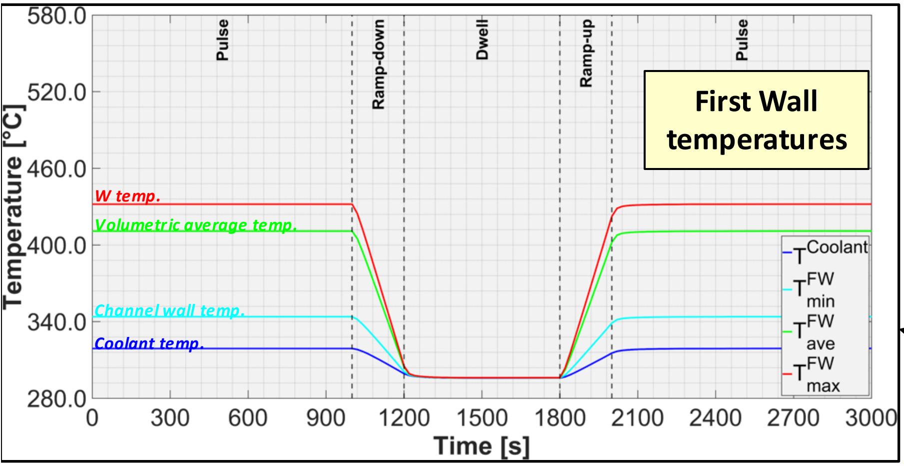


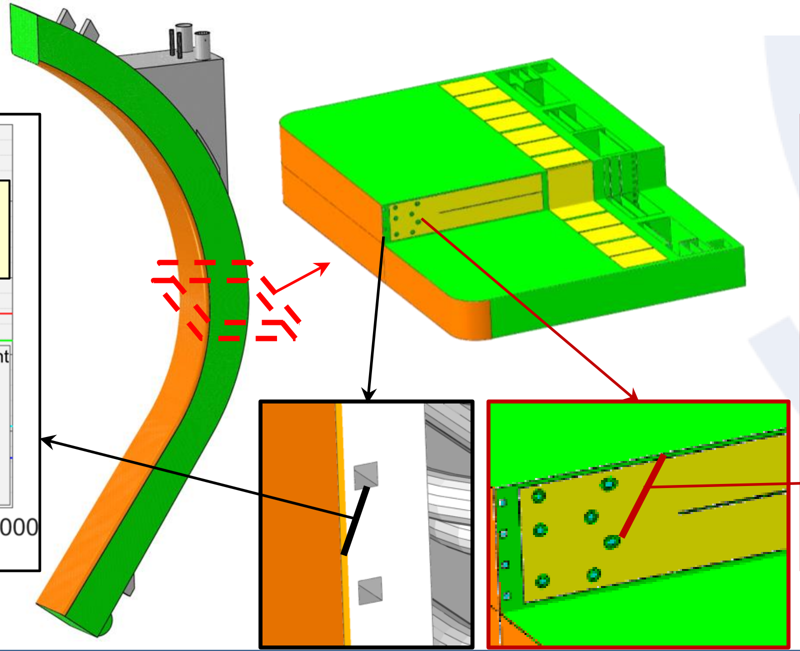
# Ramp-up/down thermal behaviour of In-VV components

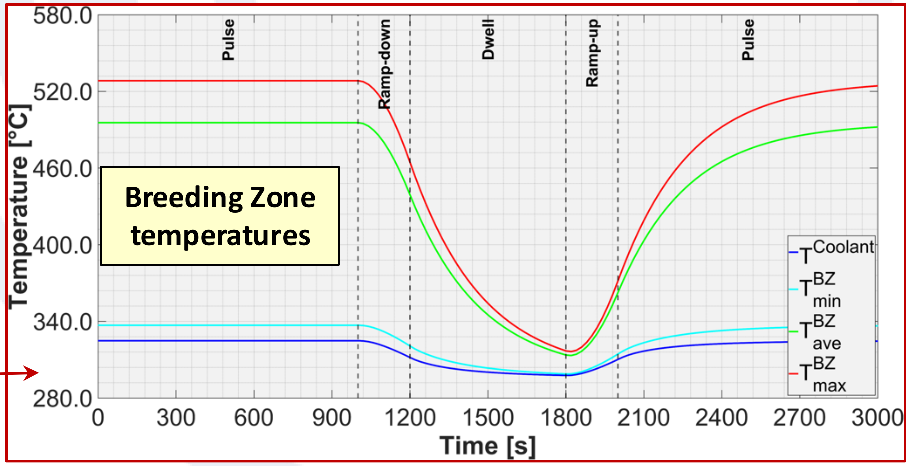
## *Example of WCLL Breeding Blanket cooling circuit*

- 200s ramps assuming linear power decrease
- Thermal-power delivered to the balance of plant at the end of 10 min Dwell still **7% of nominal power**.
- First wall thermal inertia nearly negligible, whereas breeding zone far from reaching steady thermal conditions in between two flat-tops. Overall, the **BB thermal inertia is significant**.
- BB temperature histories require detailed transient assessments.



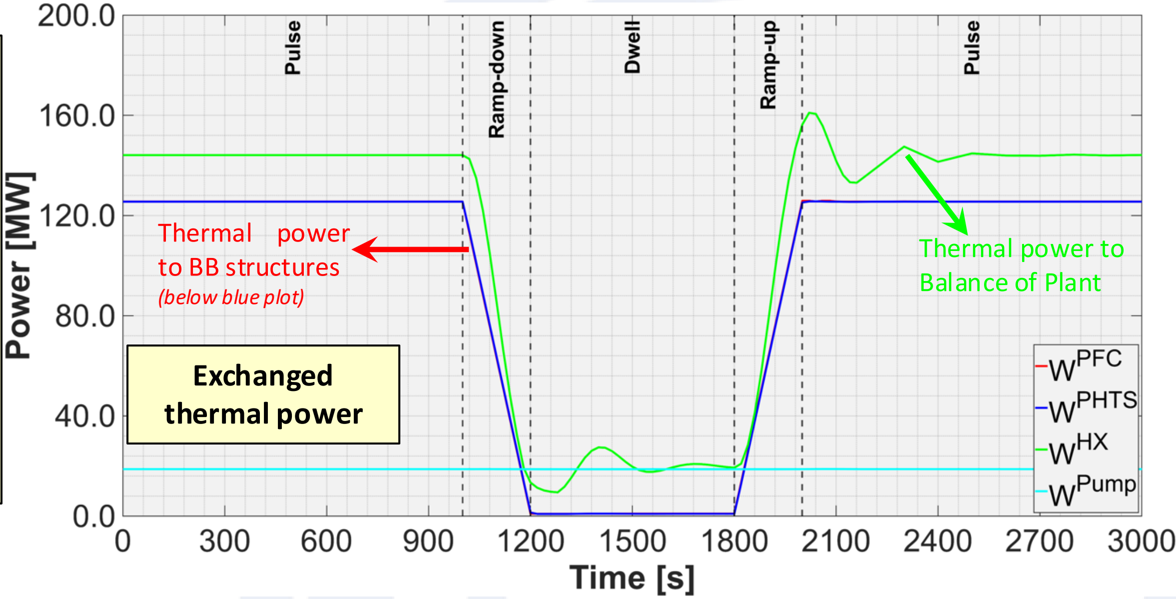


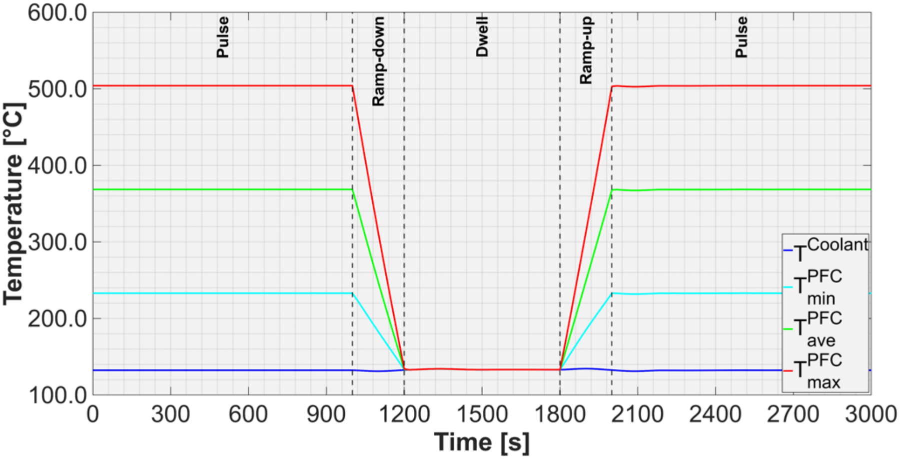

***All temperatures are indicative***

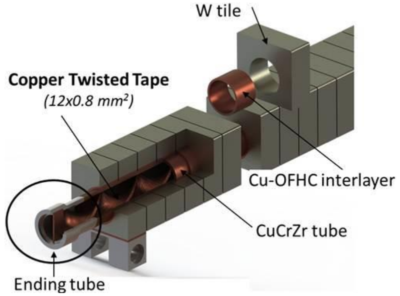
# Ramp-up/down thermal behaviour of In-VV components

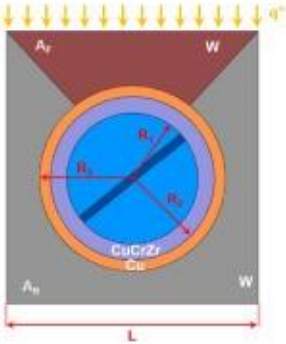
*Example of Divertor Plasma Facing Components cooling circuit*

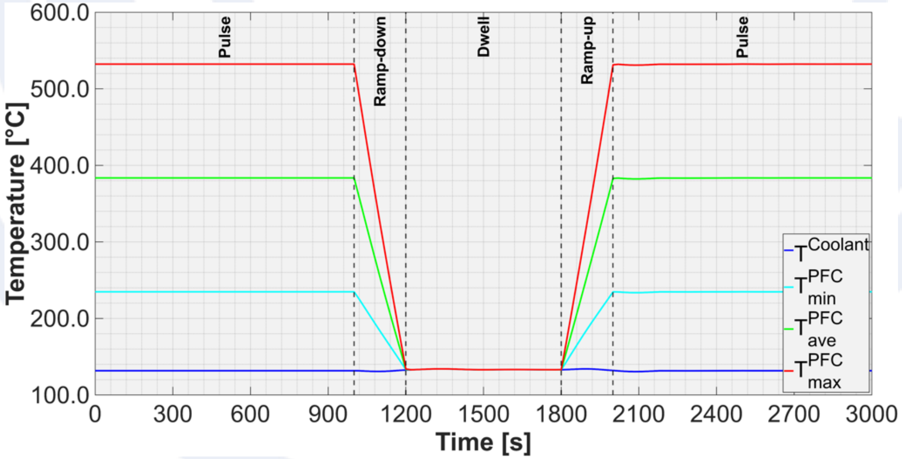
- Divertor load conditions: fully-detached divertor with heat flux <u><10 MW/m²</u>
- 200s ramps assuming linear power decrease
- Thermal-power delivered to the balance of plant at the end of 10 min Dwell “only” **decay heat + pumps**.
- Monoblocks thermal inertia negligible. As expected, divertor PFCs **thermal inertia is basically negligible**.



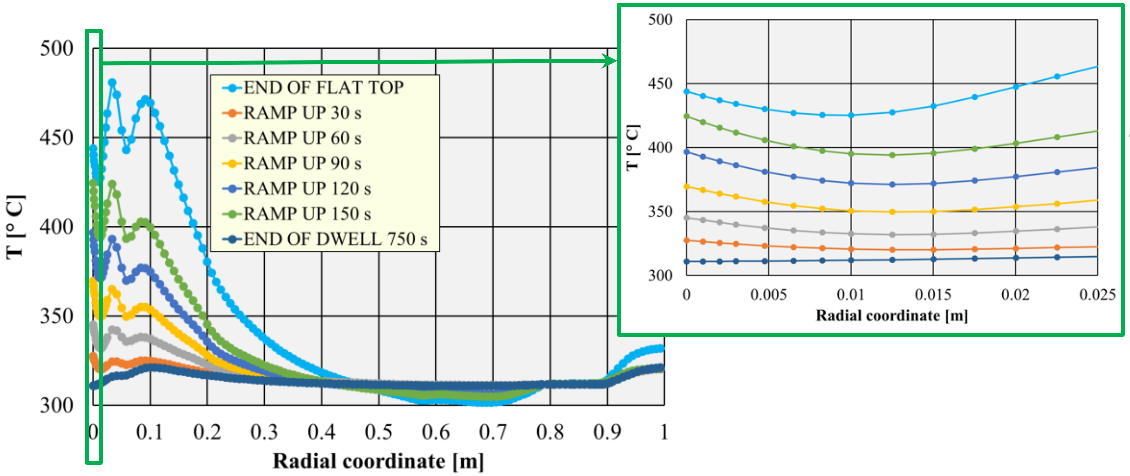



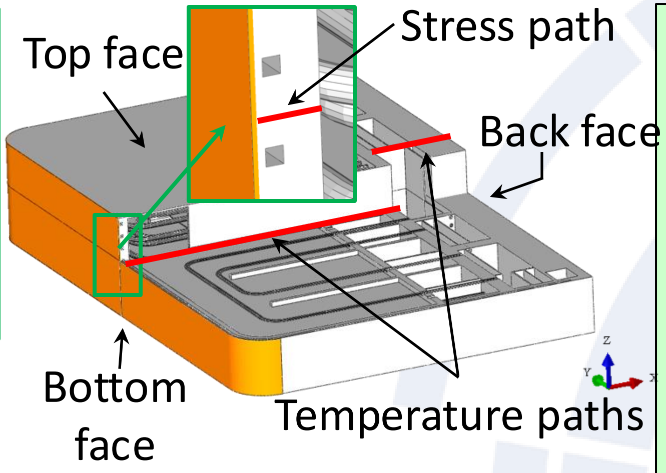



# Ramp-up/down thermal behaviour of In-VV components

*Example of stress field in breeding blanket first wall during ramp-up*

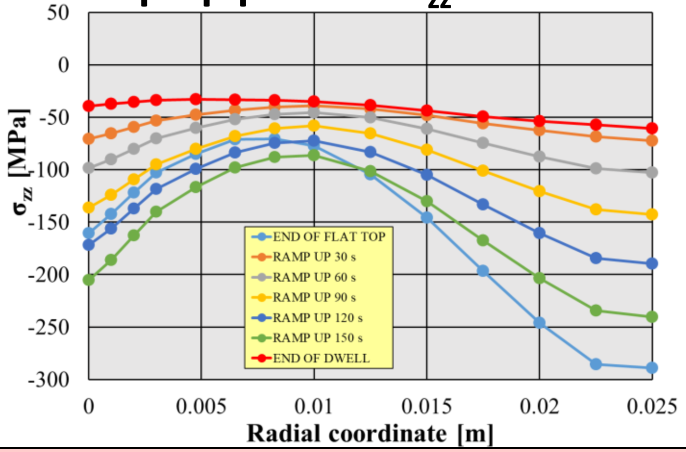


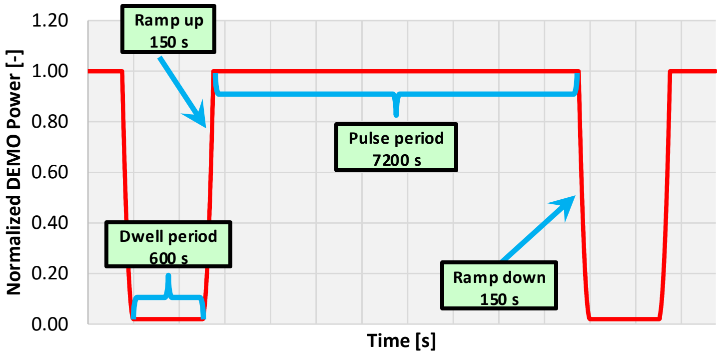

**Loads and boundary conditions**

- Time-dependent heat flux and nuclear heating (as per the DEMO duty cycle).
- Constant coolant mass flow rate.
- Coolant design pressure (17.8 MPa).
- Generalised plane strain/Z symmetry conditions on the model top/bottom face.
- Radial- toroidal contraints on the model back face.

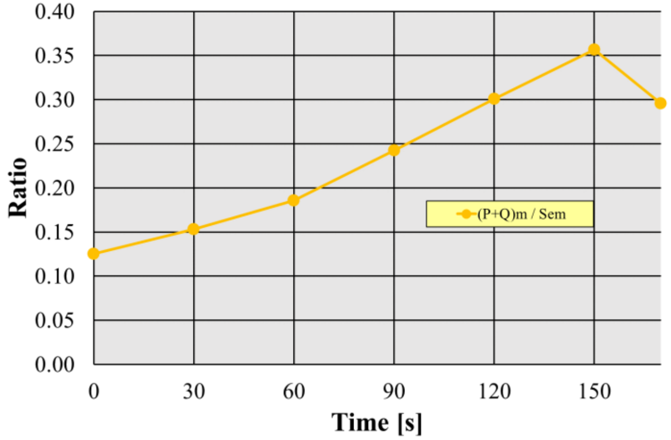
## Ramp up phase – $\sigma_{zz}$ stress radial profile




## DEMO duty cycle




## Immediate plastic flow localisation criterion




- In the back region, temperature decreases during ramp up.
- Within the first wall, some stress components achieve values during the ramp up higher than the end of flat top.

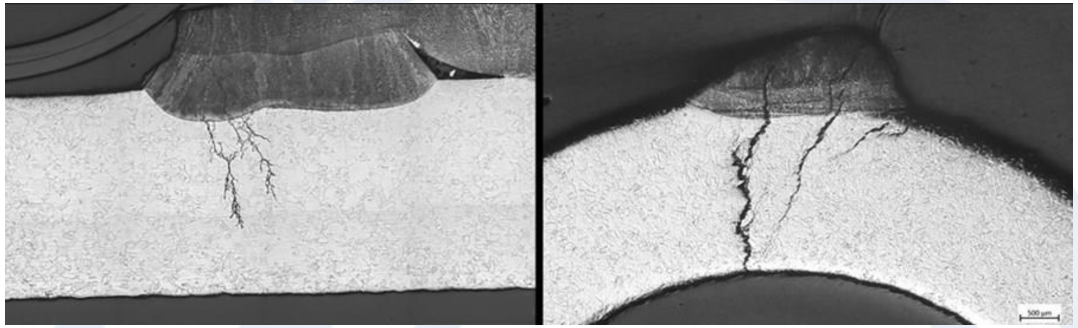
- Generally, most severe conditions can be reached during ramp up than at the end of flat top. Transient analysis is necessary.

# Water chemistry in fusion reactors

# Corrosion and its control in nuclear environment

*General pictures, status and perception*

- Corrosion impacts performances as well as safety in nuclear industry
- In general, corrosion issues costs lots of money
- Corrosion degradation in the nuclear industry is super expensive → e.g. ITER
- Corrosion was seldom or superficially considered in early NPPs designs
  - Bad assumptions
  - Poor corrosion testing techniques
- Significant amount of LWR lost availability and costs can be avoided



*Stress corrosion cracking in ITER thermal shield tubes due to chlorine residues.* [*https://www.iter.org/newsline/-/3818*](https://www.iter.org/newsline/-/3818)

- Corrosion is not an attractive idea → anyone likes decay, aging and disease
- Decisions on initial design are not usually made by engineers who understand corrosion → corrosion is usually one of the last consideration in design
- Corrosion is too complex and "magical"

# Corrosion and its control in nuclear environment

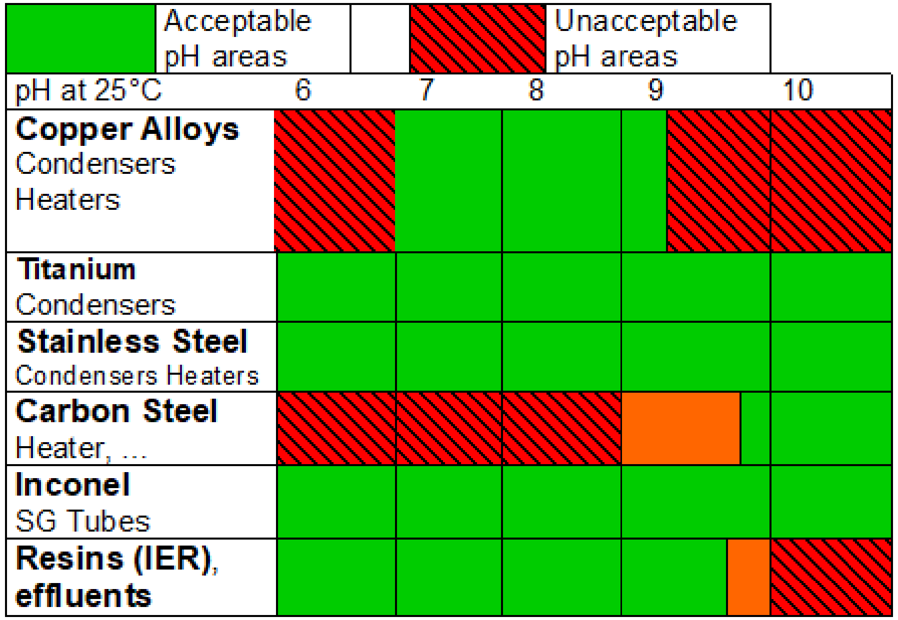
*Main materials used in DEMO PHTSs*

In-VV components

- **EUROFER-97 →** blanket, divertor cassette body and limiters shield blocks.
- **Copper based alloy** – High heat flux components → Divertor and Limiters PFCs
- **Austenitic SS 316L(IG) →** Vacuum Vessel

Ev-VV components

- **Low alloy steels →** base materials for pressure vessels
- **Austenitic SS 304/316 (or variants)** → piping and vessels cladding
- **Nickel based alloy →** steam generator tubes and vessels cladding



# Corrosion and its control in nuclear environment

*Several forms of corrosion–Chemical conditions for materials' protection*

Macro-localized corrosion

- General or uniform corrosion
- Galvanic corrosion
- De-alloying corrosion
- Velocity phenomena->erosion-corrosion, cavitation, impingement, fretting and FAC
- Crevice corrosion

Micro-localized corrosion

- Pitting corrosion
- Intergranular corrosion
- Corrosion fatigue
- Stress corrosion cracking

**Two main objectives for the water chemistry control:**

1) protect the pressure boundary materials' integrity
2) control of dose rates by minimising production and transport of corrosion products

**EUROFER-97** → Low $O_2$, Alkaline pH, low ionic impurities

**Copper based alloy** → Very limited $O_2$, neutral or alkaline pH, high water purity

**Stainless steels/nickel alloys** → Limited $O_2$, alkaline pH, Low $Cl^-$

# Corrosion and its control in fusion environment

## *pH level for circuits equipped with Eurofer97*

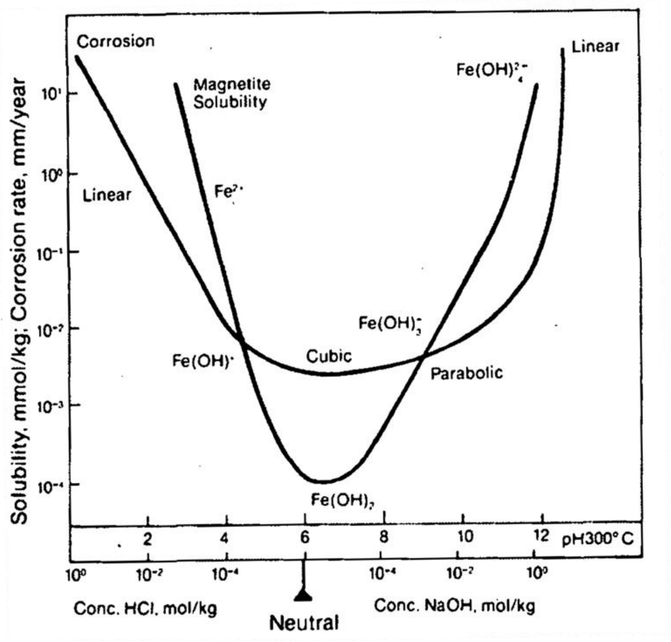
Corrosions in steels show a broad minimum in slightly alkaline conditions.

- $Fe_3O_4$ solubility mirrors the minimum.

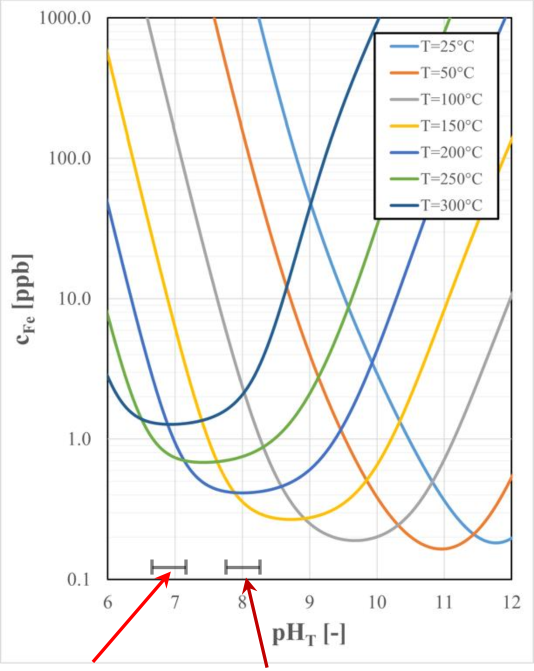
Slightly alkaline condition, with $pH_{25}$ between 10 to 12 would be beneficial to limit general corrosion

- In practice, $pH_{25}$ kept <u>below 11</u> to minimize the possibility of generating caustic environments

| T [°C] | $pH_T$ Min $Fe_3O_4$ solubility | $pH_{25}$ min $Fe_3O_4$ solubility | Li ppm |
|---|---|---|---|
| 25 | 11.81 | 11.81 | 63.78 |
| 50 | 10.97 | 11.69 | 47.14 |
| 100 | 9.71 | 11.44 | 25.75 |
| 150 | 8.69 | 11.07 | 10.39 |
| 180 | 8.32 | 10.94 | 7.68 |
| 200 | 7.94 | 10.69 | 4.20 |
| 250 | 7.43 | 10.30 | 1.69 |
| 300 | 6.95 | 9.65 | 0.37 |
| 310 | 6.79 | 9.40 | 0.20 |

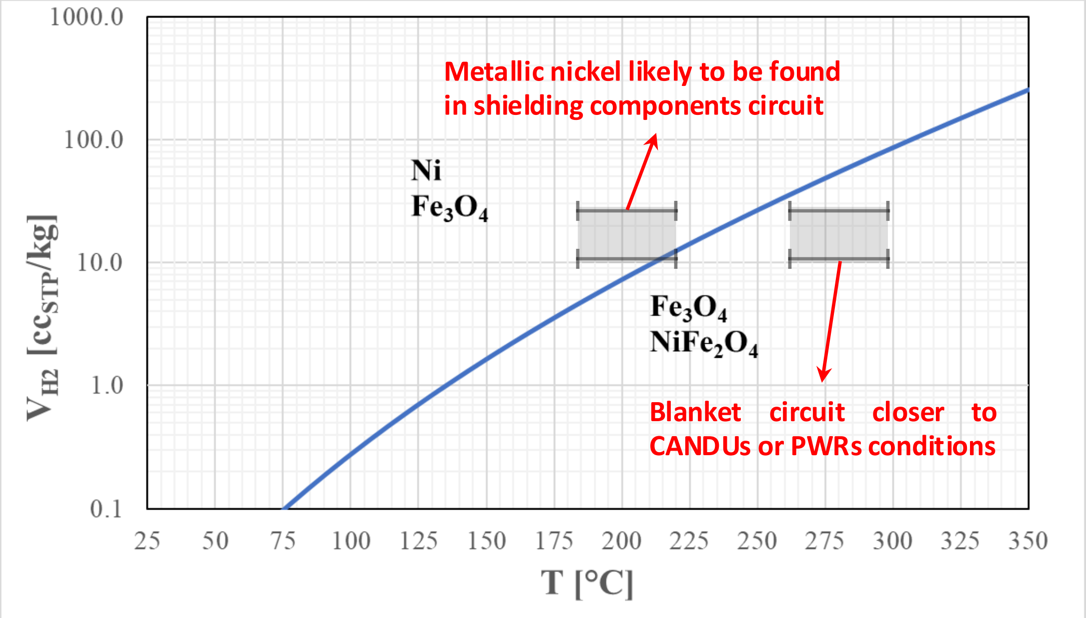



*After G.M.W. Mann (1977).*




Corrosion of Eurofer97 higher than SS and Inconel:

- Magnetite anticipated to be dominant as Fe/Ni> 2

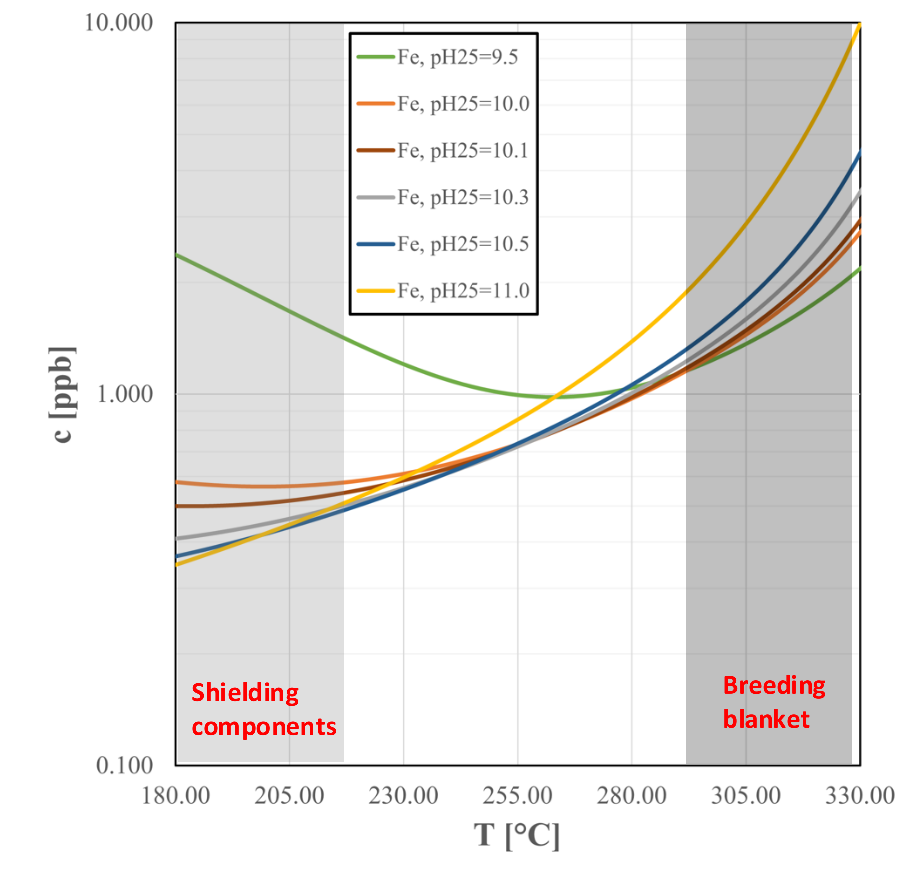


# Corrosion and its control in fusion environment

## *pH level for circuits equipped with Eurofer97*

Positive solubility gradients within in-flux areas are expected to minimize precipitation in the small channels and tubes

➢ less radioactivation, but also cleaner heat transfer surfaces (influence on heat transfer, pressure drops, …).




- For breeding blanket circuit ($T_{ave}$≈310°C) is recommended the following operating band: **9.5< $pH_{25}$ <10.5** (0.26 ppm< Li < 2.7 ppm)

- For shielding components ($T_{ave}$≈195°C), slight shift of the minimum $pH_{25}$ is suggested to minimize deposition onto in-flux surfaces → **10.1 < $pH_{25}$ <10.5** (1.1 ppm< Li < 2.7 ppm)

Assessments on maximum lithium concentration to be made → 2.7 ppm are currently assumed.

Simplistically:

$$\frac{dW_{in-flux}}{dt} = -R - k_{dis/pre}\boxed{[C_w(T) - C_b(T)]} - k_{er}W_{in-flux} + k_{dep}C_{ins}$$

For positive solubility gradient, scale tends to dissolve from hotter wall to cooler water and no precipitation occurs

# Choice of the alkalizing agent

*Is tritium a problem or an asset?*

LiOH, KOH or NaOH are common alkalizing agent used to increase water pH

**NaOH**: ruled out due to higher risk of caustic SCC

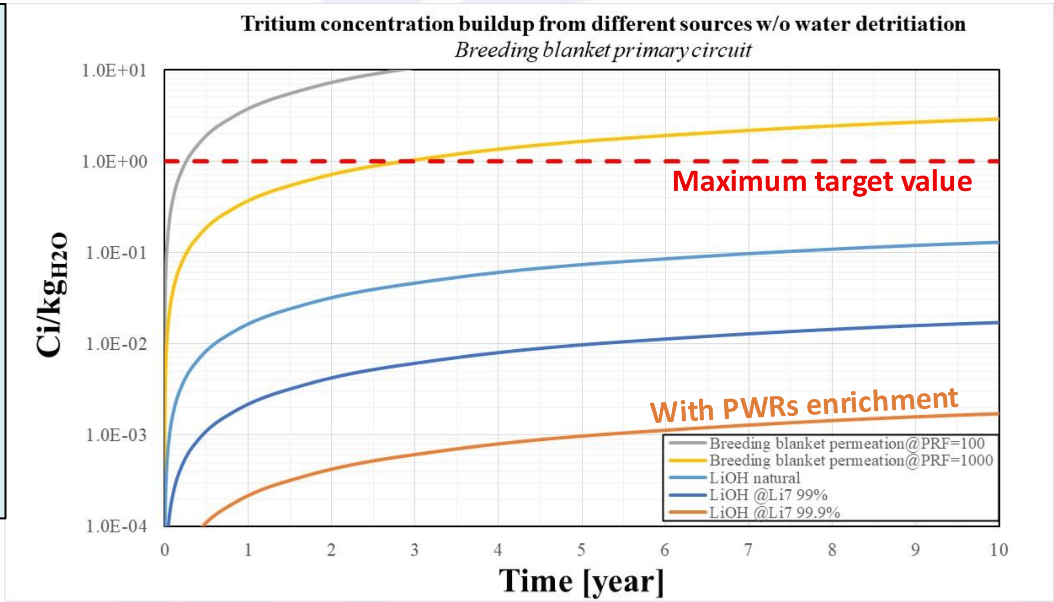
- *$^{23}Na(n,\gamma)\,^{24}Na \rightarrow T_{1/2} \approx 15$ h; 1.37 + 2.7 MeV gammas*

**KOH**: WWER experience, EPRI studies

- *$^{39}K(n,\gamma)\,^{40}K \rightarrow T_{1/2} \approx 1.25\cdot10^{9}$ y; 1.46 MeV gamma*
- *$^{41}K(n,\gamma)\,^{42}K \rightarrow T_{1/2} \approx 12.3$ h; 1.5 MeV gamma*

**LiOH**: Western PWRs experience

- *$^{6}Li(n,He)^{3}H$ and $^{7}Li(n,n\alpha)\,^{3}H \rightarrow T_{1/2} \approx 12.7$ y*




**Main data in a nutshell**:

- > 320 g/day of tritium being produced in breeding blanket for 2 GW fusion power
- With coatings reducing by a factor 100 the tritium permeation from breeding zone, migration into coolant can be as high as 450 mg/day for WCLL BB → **water detritiation system needed for primary coolant**
- Production from natural LiOH in coolant would be max. 2 mg/day → **negligible**
- Tons of depleted $^{7}$Li available from $^{6}$Li enrichment, thus completely different supply chain respect to western PWRs → **Cost effective solution**

**LiOH** is currently judged as the most elegant solution for fusion power plants

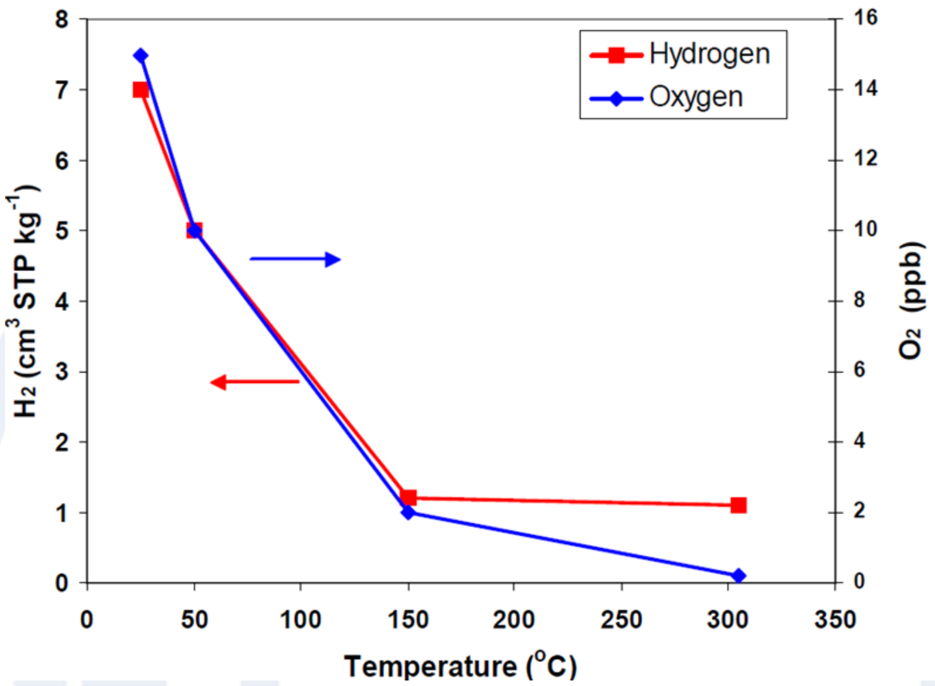
# Radiolysis of water
## *Can we keep sufficient reducing environment?*

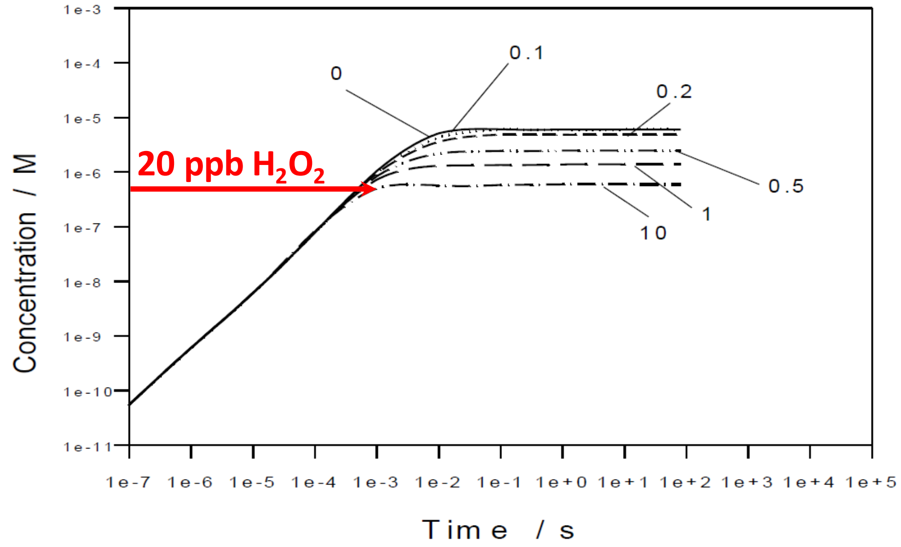
Simplistically, ionizing radiation decomposes water to form the species $H^+$ , $OH^-$, H, OH, $H_2$, $O_2$, $H_2O_2$, $e^-_{aq}$ and $HO_2^-$.
The stable products are $H_2$, $H_2O_2$ and $O_2$.

Reactions are reversible and hydrogen addition can suppress the $H_2O_2$ and $O_2$ formation

Recombination increases with temperature
- Pressurized Water Reactor conditions are beneficial

Oxidizing conditions could be difficult to suppress in "cold" circuits
- in high heat flux components electrochemical corrosion potential could remain high.




**Hydrogen is strongly recommended to decrease the oxidant formation.**
**Preliminary selected range 10-25 cc (STP)/kg**

*D.D. Macdonald, Corros. Mater. Degrad., (2022).*

# Radiation Protection approach for the coolant systems

# Radiation protection

## *Calculation of the source term in the coolant system*

| Main water activity [kBq g$^{-1}$] | DEMO – *2000 MW fusion power* | | | NPP |
|---|---|---|---|---|
| | BB PHTS | Shielding Component PHTS | High heat flux components PHTS | AP600 *(1933 MW$_{th}$)* |
| $^{16}$N | $4.38\cdot10^{6}$ | $4.46\cdot10^{6}$ | $8.40\cdot10^{5}$ | $6.03\cdot10^{3}$ |
| $^{17}$N | $6.06\cdot10^{2}$ | $3.78\cdot10^{2}$ | $1.17\cdot10^{2}$ | 1.77 |
| $^{3}$H | $\leq7.40\cdot10^{4}$ | $<3.70\cdot10^{4}$ | $<7.40\cdot10^{4}$ | $1.3\cdot10^{2}$ * |

*In CANDU 6 the $^{3}$H concentration in HTS is typically limited to $\leq9.25\cdot10^{4}$ kBq g$^{-1}$

| Long-term CRUD deposit specific activity [kBq $g_{crud}^{-1}$] | DEMO | NPP |
|---|---|---|
| | BB PHTS | AP600 *(1933 MW$_{th}$)* |
| $^{58}$Co | $7.89\cdot10^{4}$ | $4.44\cdot10^{5}$ |
| $^{60}$Co | $1.61\cdot10^{5}$ | $2.22\cdot10^{5}$ |
| $^{59}$Fe | $2.22\cdot10^{4}$ | $1.85\cdot10^{4}$ |
| $^{54}$Mn | $1.31\cdot10^{6}$ | $5.18\cdot10^{4}$ |

Fusion power plant have inherent safety features compared to fission power plant, however some specific hazards must be considered:

- $^{16}$N specific activity ~**700 times higher than NPPs**
- **Tritium** contamination in coolant **can be higher than CANDU reactors**
- Activated corrosion products as in NPPs

**To cope with these challenges, several actions are taken**:

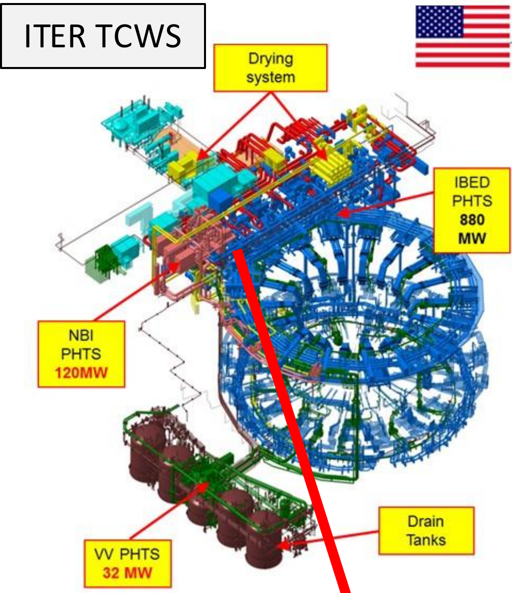
- Tight control of water chemistry (pH, $H_2$, low halides content, etc.) to minimize corrosion issue and radiation field
- **Purification half-life up to 60 min** to effectively remove corrosion products and ensure high water purity
- Dedicated **active detritiation system** for coolant based on water distillation technology

Radioactive contamination of water drives the integration in the building and systems design approach

# Radiation protection – Shielding approach

*Lesson learnt from ITER (1/2)*

In fusion power plants the principle of fission plants must be adopted: primary coolant circuits inside secondary shielding walls.




Secondary shielding requirements:

- Dose outside the tokamak building walls ≤2.5 µSv/h
- In fission plants secondary shielding ~0.8-1.2 m.
- In DEMO $^{16}N$ specific activity can be more than 700 times higher than NPPs.
- Increase of concrete thickness needed to keep same dose level (+30-40 cm)

DEMO concept based on lessons learned from ITER:

- Up to 1.5 m thick walls implemented around the pipes and heat exchangers (ITER: as low as 0.5-0.8 m)
- Water pipes routed through intra-bioshield walls.
- Activated components segregated from other equipment
- During maintenance of primary system equipment, dose to workers is dominated by deposits of activated corrosion products → frequently maintained/operated components are further shielded as for fission plant rationale.

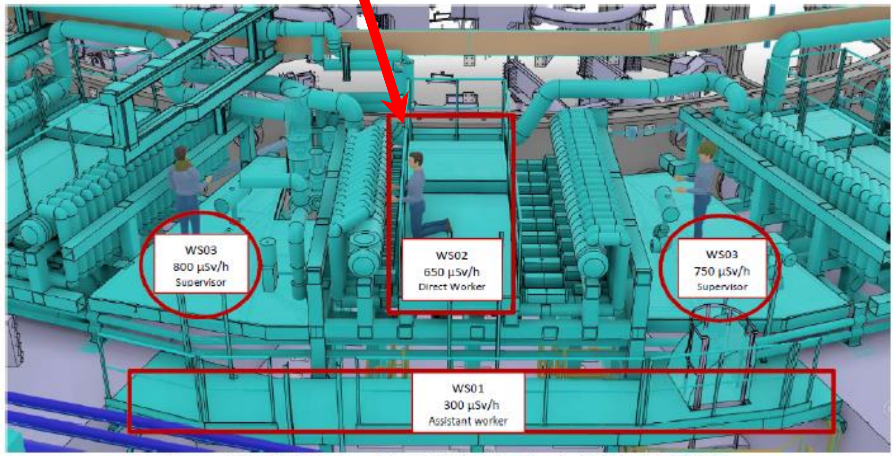
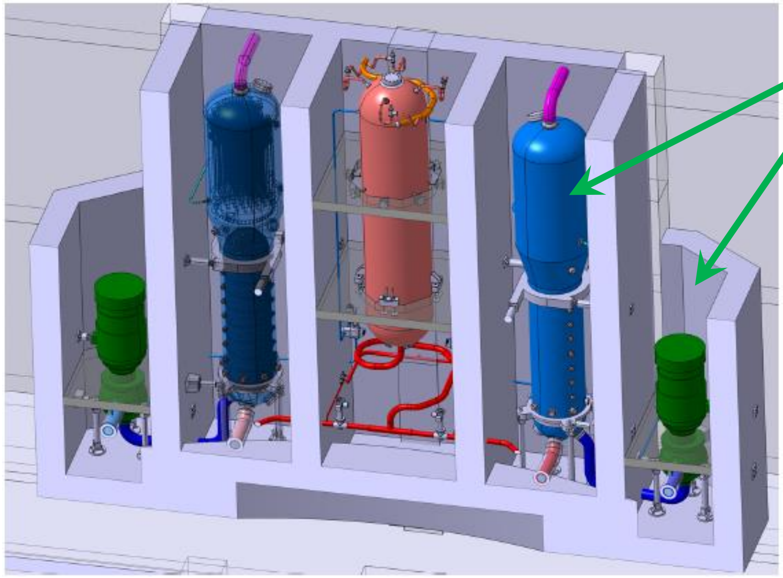
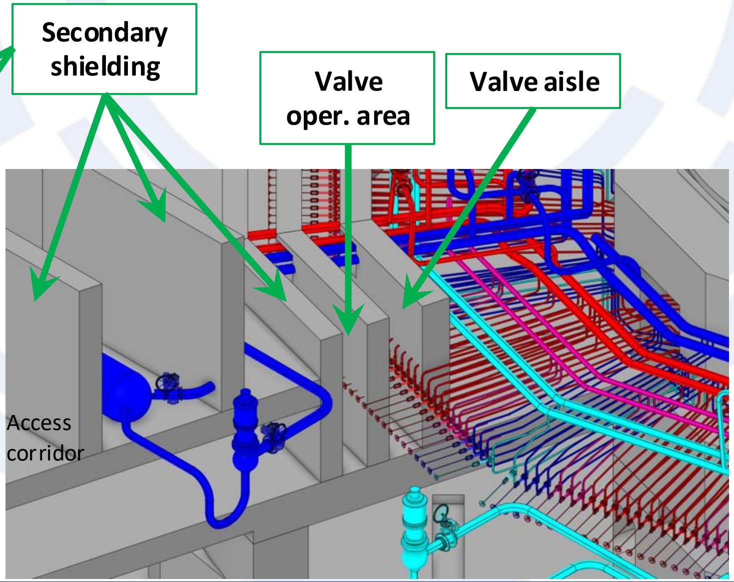



*Figure 10-1. WorkStations 01, 02 & 03 at the Jungle GYM – (11-L3-03S).*





# Radiation protection – Shielding approach

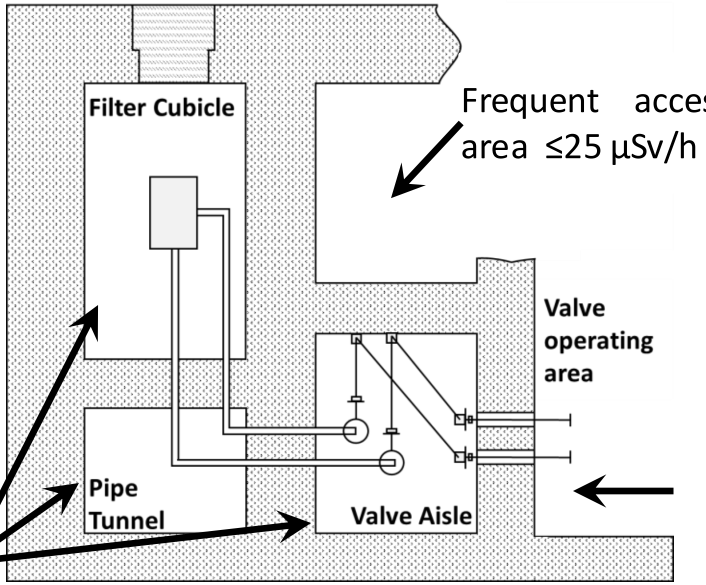
*Lesson learnt from ITER(2/2)*

**Auxiliary Building to house PHTS support systems and other equipment**

- Coolant purification systems
- Component cooling water system
- HVAC
- Coolant storage tanks

## Radiation protection drives layout

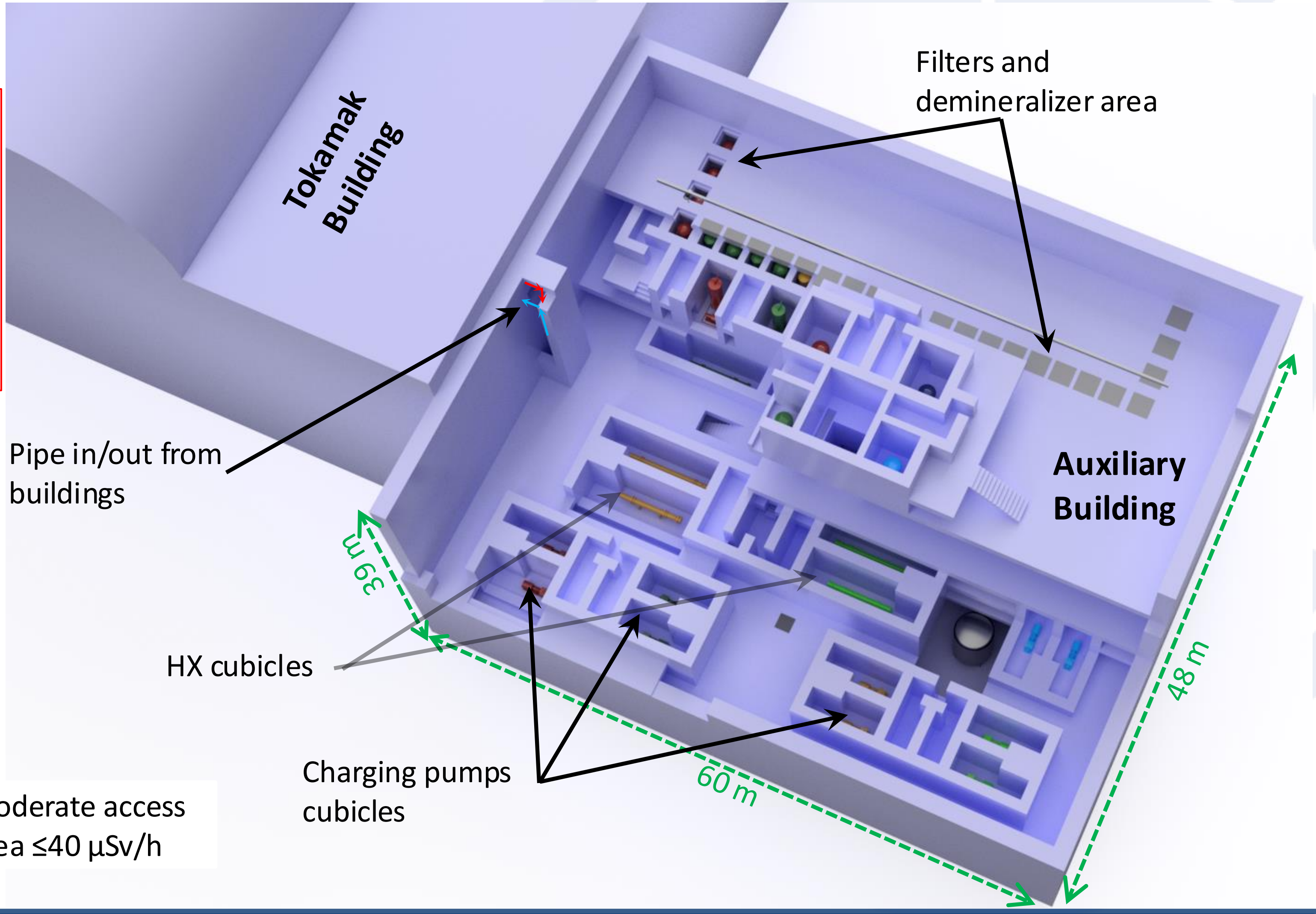
- Separation of radioactive and nonradioactive pipe tunnels
- Valve operated from well shielded areas
- Frequently access areas (40hr/week) like corridors are shielded so that radiation level is ≤25 µSv/h




Frequent access area ≤25 µSv/h

High radiation area ≥1 mSv/h

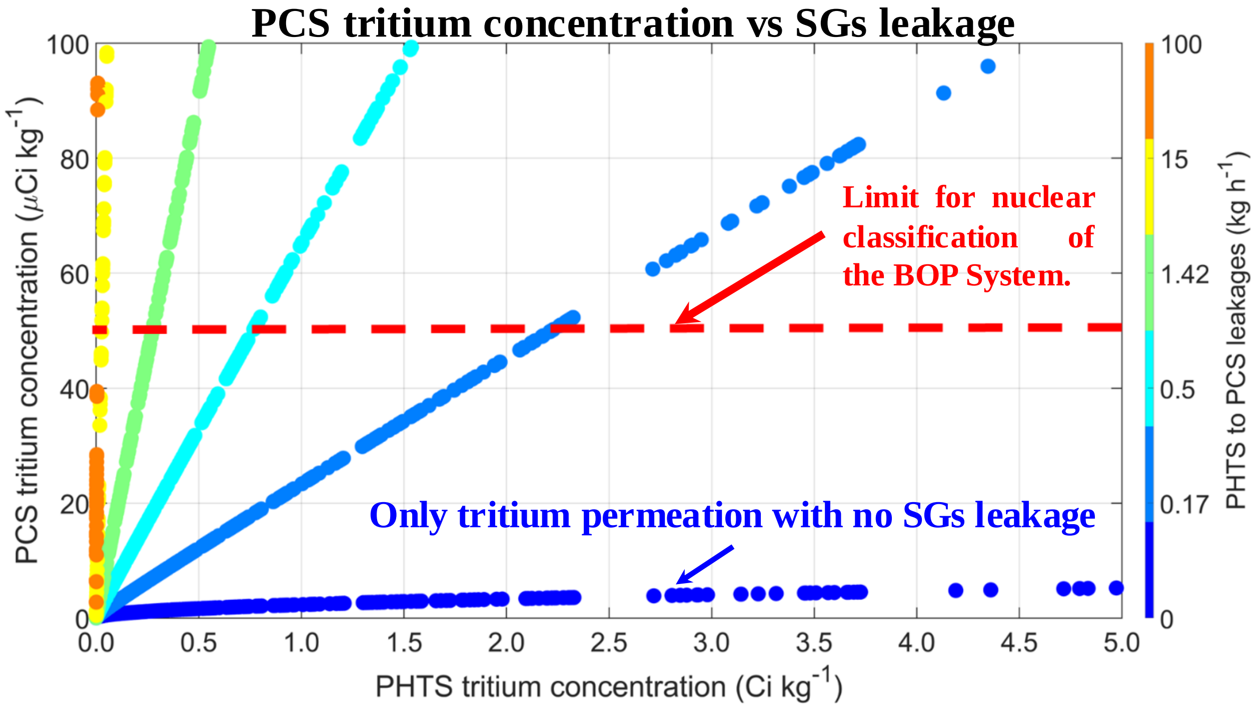
Moderate access area ≤40 µSv/h



# Radiation protection approach
## *Limit for tritium release into secondary*

| Main water activity [kBq g$^{-1}$] | DEMO – *2000 MW fusion power* | | | NPP |
|---|---|---|---|---|
| | BB PHTS | Divertor CB & Limiter SB PHTS | PFCs PHTS | AP600 (*1933 MW$_{th}$*) |
| $^{3}$H | ≤7.40·10$^{4}$ | <3.70·10$^{4}$ | <7.40·10$^{4}$ | 1.3·10$^{2}$ * |

*In CANDU 6 the $^{3}$H concentration in HTS is typically limited to ≤9.25·10$^{4}$ kBq g$^{-1}$




Secondary system is mostly located in a non-nuclear building and it is usually designed as **non-nuclear** system.

As for the ESPN Classification (2005), a pressurized equipment is classified as nuclear when, in case of failure, it leads to release of activity above **370 MBq**.

The release of activity shall be evaluated as follows:

- For a container, the product of its volume times the activity concentration of the fluid contained, calculated as the sum of the activity concentration due to all elements present except T, N-13, O-15 and 19, F-20, 21 and 22, Ne-19 and 23, multiplied by a coefficient 1.

- The activity concentration due to tritium, nitrogen 13, oxygen 15 and 19, fluorine 20, 21 and 22, neon 19 and 23, multiplied by a coefficient **1/1000**

**Thank you for your attention**

The activity $A(t) = -\lambda N(t)$ of a sample of radioactive nuclides $N(t)$ provides the number of nuclides that decay in the interval $dt$. Where $\lambda$ [s$^{-1}$] is the decay constant.

It follows that the decrease in number of radioactive nuclides of the sample in the same interval $dt$ can be written as $\frac{dN}{dt} = -\lambda N$

Integrating the equation we find that the number of nuclides in the sample decays exponentially from the initial value $N_0$ . Multiply both side by $\lambda$ we obtain the activity of the sample at the time $t$

$$N(t) = N_0 e^{-\lambda t} \xrightarrow{\lambda} A(t) = A_0 e^{-\lambda t}$$

The activity is measured in Bequerel [Bq] in SI, which measures the disintegration per second of a given radioactive isotope. Another unit which can be often encountered is the Curie [Ci]. 1 Ci= 3.7E+10 Bq= 37 GBq.

The *unshielded* radiation flux from a gamma (neutron) point source of activity A, at a distance R is

$$\phi_0 = \frac{A}{4\pi R^2} \left[\frac{photons}{cm^2 s}\right]$$

When a shield is placed between the source and the probing point, the *uncollided* flux decreases according to the well known exponential law

$$\phi = \phi_0 e^{-\mu x} = \frac{A}{4\pi R^2} e^{-\mu x}$$

Where μ [cm$^{-1}$] is the attenuation coefficient, depending on material and energy of incident flux and
$x$ [cm] is the shield thickness

If interactions of photon with material of the shielding and lack of "good" geometry are present, buildup factor should be considered

$$\phi = B\phi_0 e^{-\mu x} = B\frac{A}{4\pi R^2} e^{-\mu x}$$

$B$ depending on material, shield thickness and energy of incident flux