Status of the ITER TBM Program and overview of its technical objectives

Luciano Giancarli, Tritium Breeding Blankets Project
15 September 2023

Disclaimer
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization

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Acknowledgements to Co-Authors

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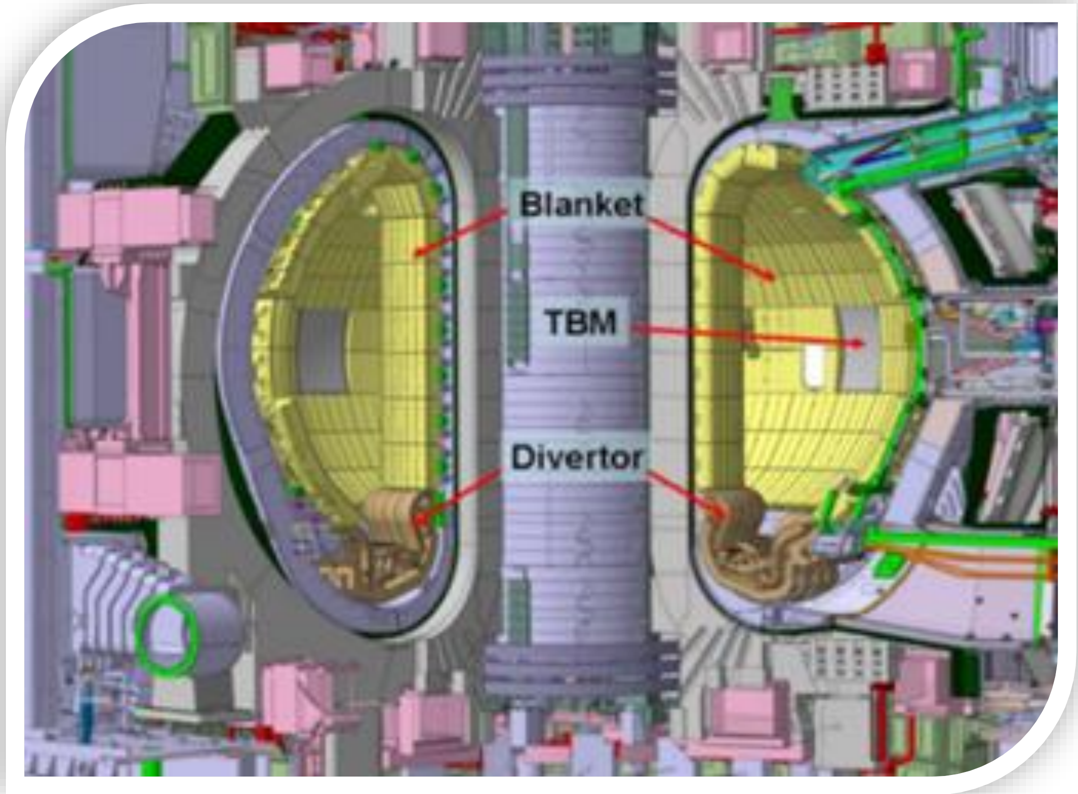
- **The presenter wishes also to thank the TBM Teams from CN, EU, KO, and JA for some of the pictures used in this presentation**



Outline of the presentation

A. Rational of the ITER TBM Program

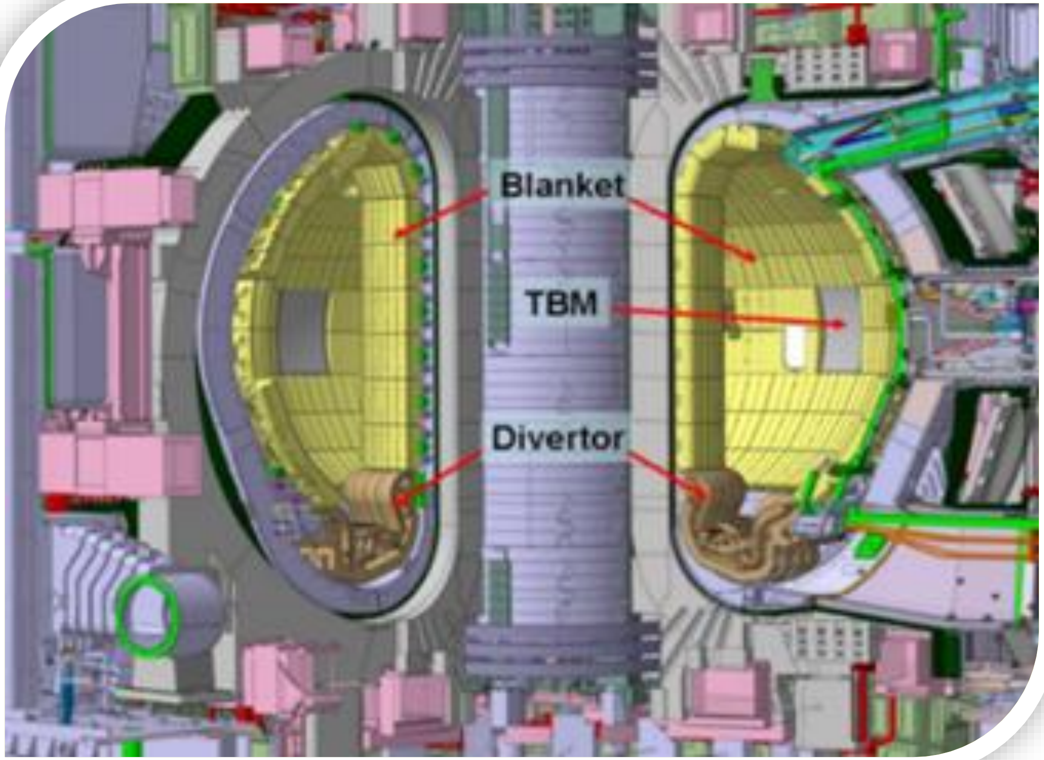
B. Main features of the Test Blanket Systems

C. Examples of on-going R&D on Test Blanket Systems and Components

D. ITER TBM Program Testing Plan and main overall objectives

E. Technical objectives and achievements of the TBM Program in support of the DEMO TBBs development

F. Final Considerations and Conclusions



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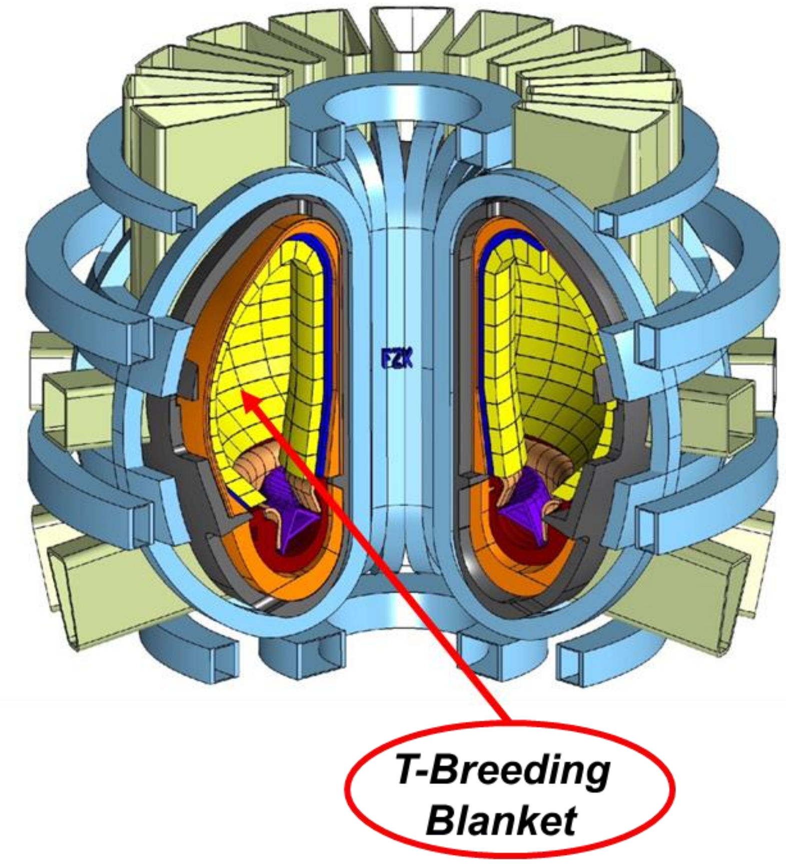
What is the ITER TBM Program ?

- A Fusion Power reactor needs to produce by itself all the Tritium that is needed as fuel for the D-T plasma (Tritium-breeding self-sufficiency) while ITER is using external Tritium source.

- To support this need, one of the ITER missions is the following (cf. Project Specifications):
  "***ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high grade heat and electricity production.***"



All the ITER Organization (IO) design and R&D activities related to this mission, both in IO Central Team and in the seven IO Domestic Agencies, form the so-called ITER TBM Program.

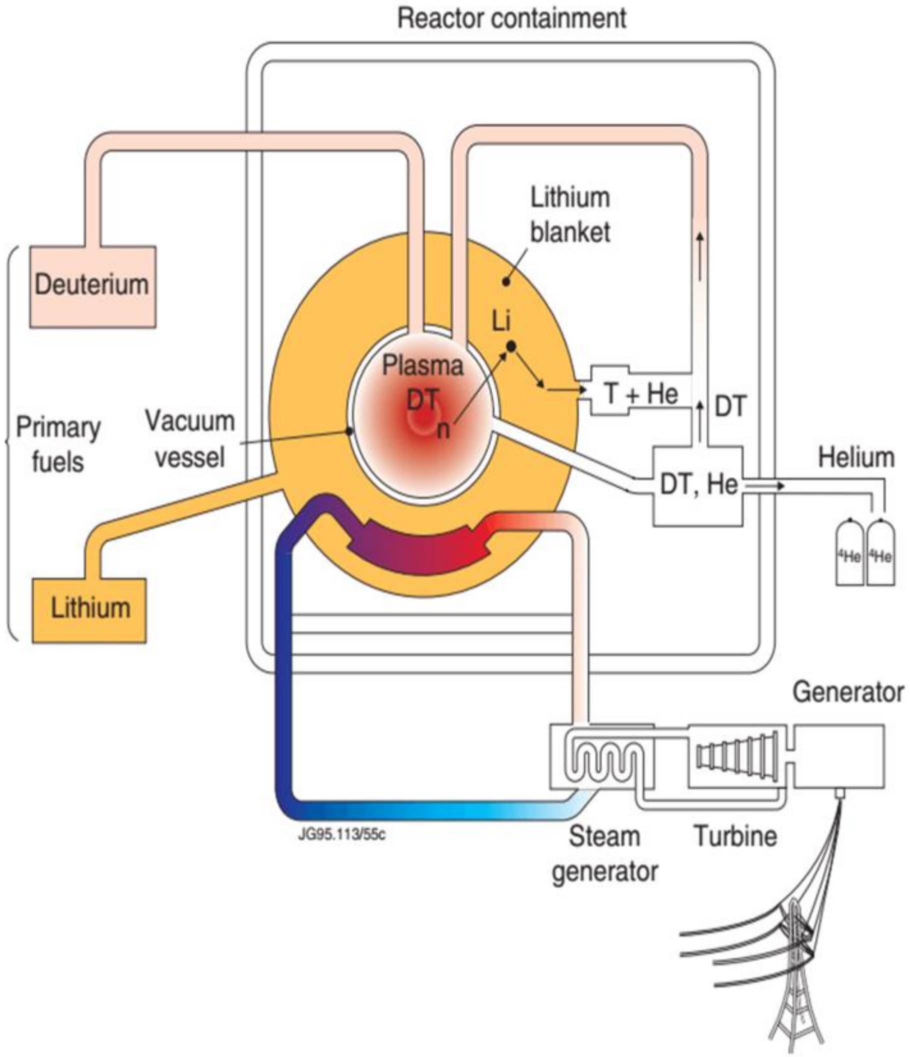
➢ The TBM Program is therefore a specific research activity run in ITER as part of the ITER Research Plan

Tritium Breeding Blankets (TBB) Functions in a Fusion Power Reactor

- **Convert** the neutron energy (80% of the fusion energy) and the gamma energy coming from the plasma in heat and collect it by mean of an **high-grade coolant** to reach high conversion efficiency (>30%)
  ➔ use of high pressure, high temperature coolant

- **Produce all Tritium** required as fuel for D-T reactors via nuclear reactions on Lithium isotopes present in the breeding blanket and **fully RECOVER** it by using, for instance, He purge-gas and proceed to a continuous re-injection in the plasma
  ➔ achieve Tritium breeding self-sufficiency

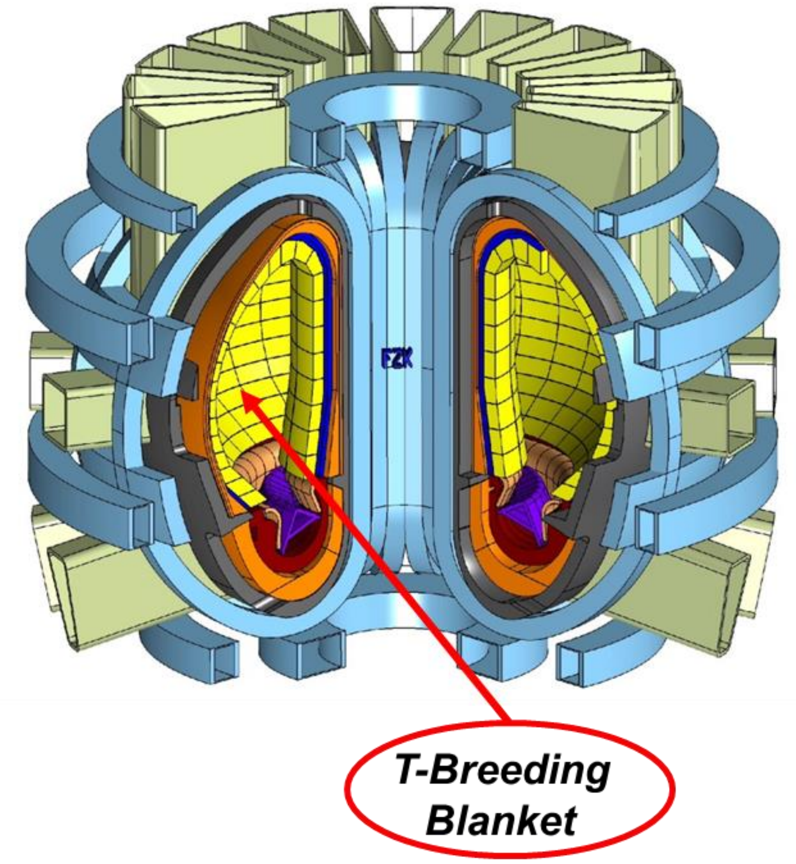
- Contribute to neutron and gamma **shielding** to protect the superconductive coils



Typical TBB Operating Conditions expected in a Fusion Power Plant

The breeding blanket, including the Fist Wall (FW), fully surrounds the plasma. It is therefore submitted to severe working conditions, for instance:

- ✓ High surface heat flux > 0.5 MW/m^2 on FW
- ✓ High neutron wall loading ~2.5 MW/m^2
- ✓ Operating for at least 5 years: ~150 dpa(Fe) in FW
- ✓ Operation in vacuum (plasma) ➔ low coolant leakages
- ✓ High magnetic field (~7 Tesla) ➔ high MHD effects



Typical “medium-term” materials and associated functions that would need to be present in a DEMO Tritium Breeding Blankets and potentially envisaged/needed for TBMs

- **Structural materials:** Reduced-Activation Ferritic/ Martensitic steels. These types of steels are under development by several worldwide laboratories to avoid rad-waste with lifetime longer than 100 years (that would be present using current nuclear steels). They are very important for the public acceptance of future of D-T fusion power.

- **Coolants:** Helium (~300/500°C, 8 MPa), pressurized water (~280/325ºC, 15.5 MPa, same as in PWR), Pb16Li (eutectic liquid metal, melting point 235°C).

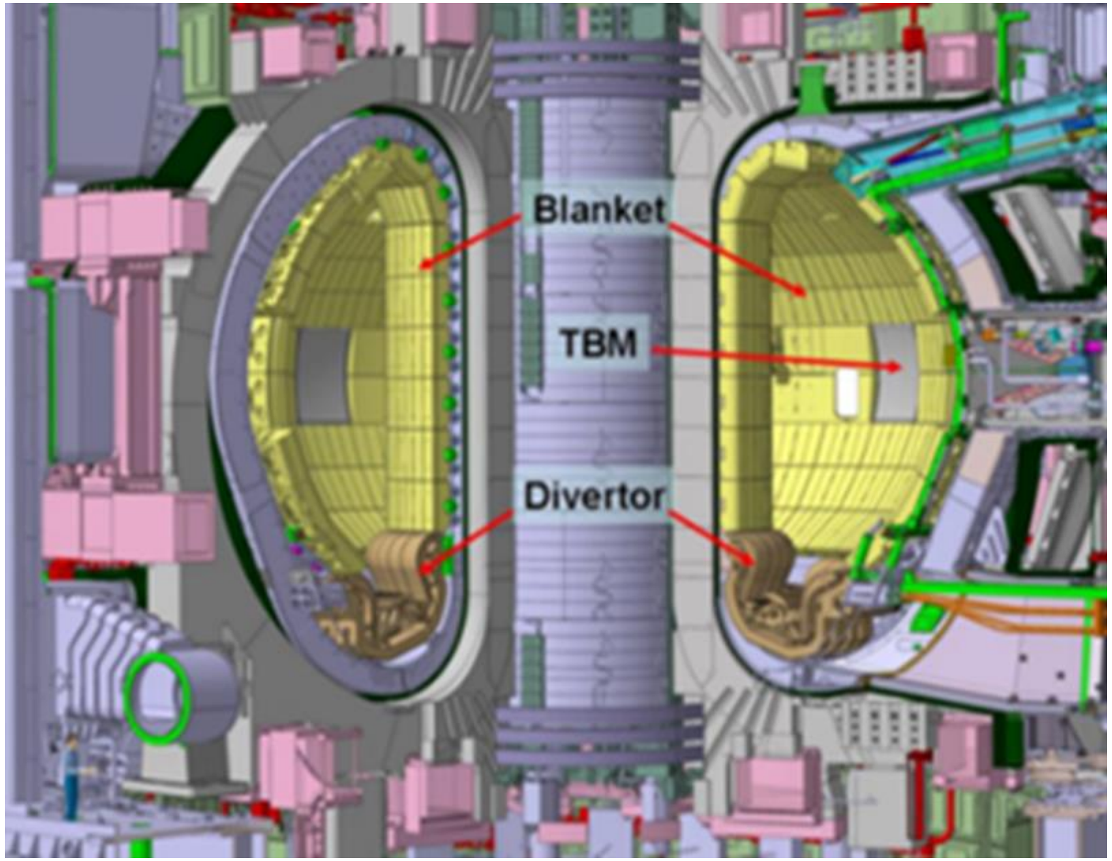
- **Tritium breeder materials:** Lithium-based ceramics (solid) such as Li_4SiO_4 and Li_2TiO_3, Pb16Li (Li as part of the liquid metal).

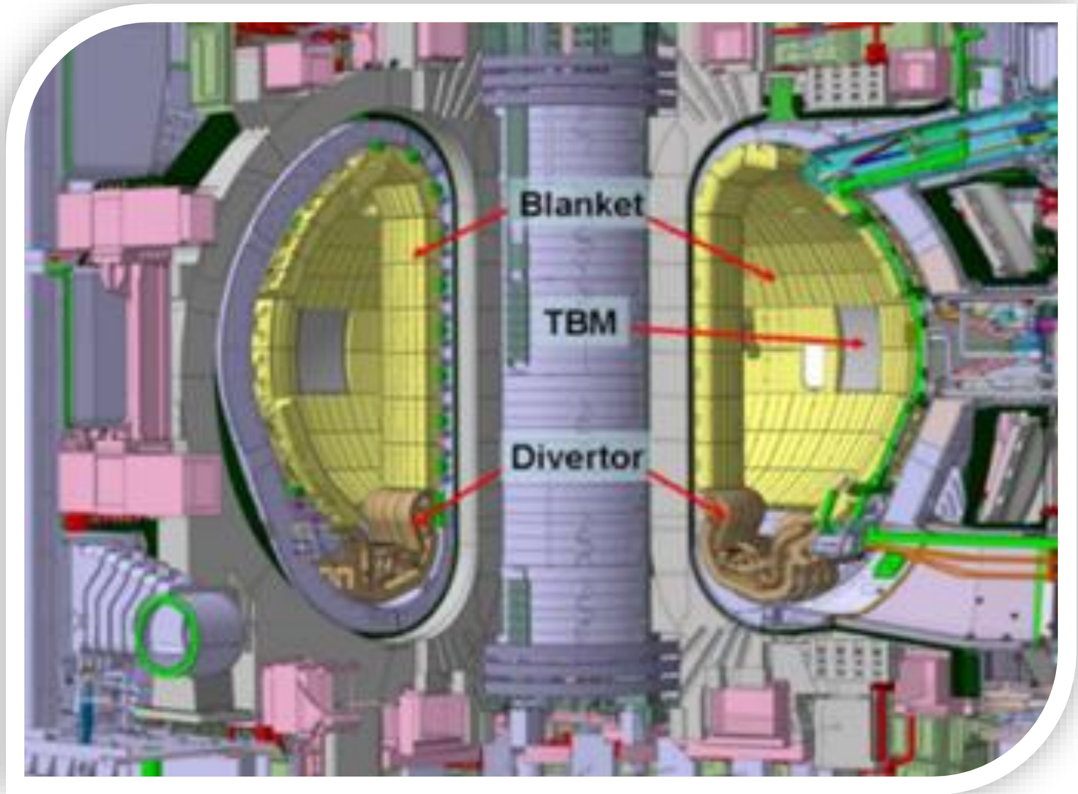
- **Neutron multipliers**: Beryllium (solid), Pb16Li (Pb as part of the liquid metal).

Testing of Tritium Breeding Blankets (TBB) in ITER

- ITER is a unique opportunity to test complete mock-ups of TBB (called Test Blanket Systems) in DEMO-relevant conditions in order to demonstrate that **Tritium Breeding self-sufficiency** can be achieved (involving the whole process of tritium production, extraction, and re-injection in the plasma).

- Therefore, the “ITER TBM Program” foresees to install and operate in ITER **several Test Blanket Systems** that include the Test Blanket Modules (TBMs) (i.e., the in-vessel part) and the associated ancillary systems.





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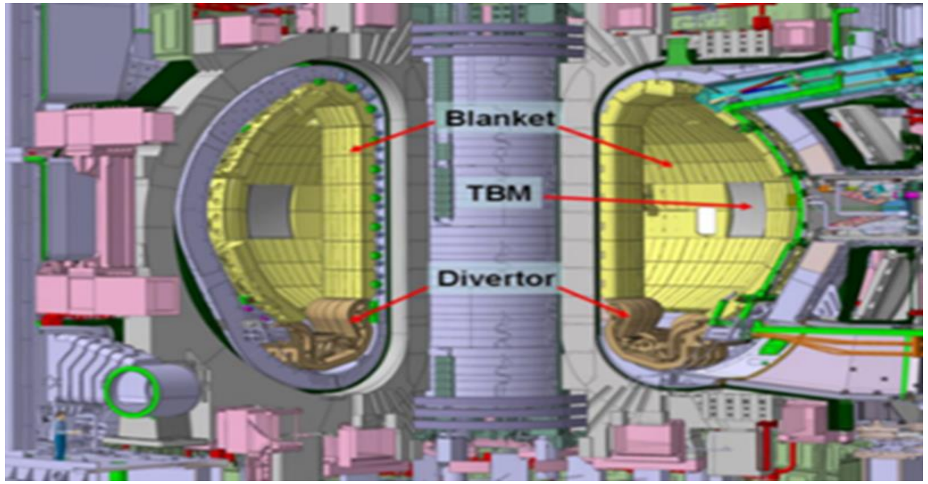
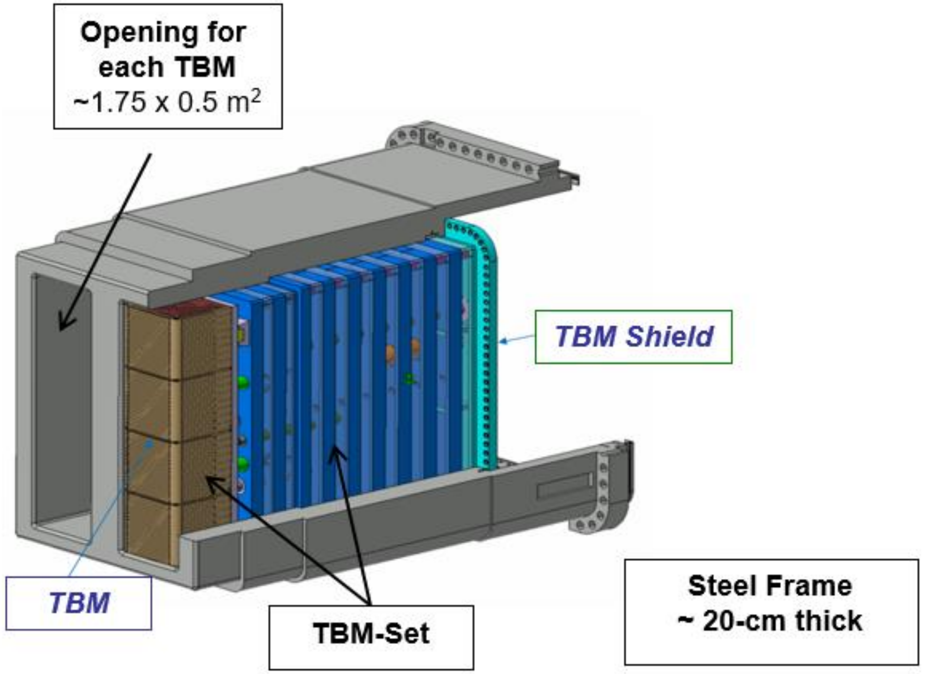
Boundary Conditions in ITER for TBMs test

- The "ITER TBM Program" foresees the installation of four Test Blanket Systems (TBSs) whose TBMs are located in the **2 ITER equatorial ports #16 and #18** (2 TBMs per port) inside the Vacuum Vessel.
- The TBMs are installed in a water-cooled steel frame (together with the associated shield).
- Each TBS is formed by a TBM and the associated ancillary systems (e.g., coolant, coolant purification, Tritium extraction, Tritium accountancy, Instrumentation & Control, maintenance tools and equipment) located in the Tokamak Complex. It is a complete mock of the whole DEMO Tritium Breeding Blanket Systems.
- It means that ITER is planning to simultaneously operate four independent Test Blanket Systems.





Port Allocation for the four Test Blanket Systems of the Initial Configuration (InCo)

Port Nº	First Concept	Second Concept
16 (F)	TBS-1: *Water-Cooled Lithium-Lead (EU)* ➔ *Water at 15.5 MPa, 280-325ºC*	TBS-2: *Helium-Cooled Ceramic Pebbles (joint KO & EU)* ➔ *Helium at 8.0 MPa, 300-500ºC*
18 (S)	TBS-3: *Water-Cooled Ceramic Breeder (JA)* ➔ *Water at 15.5 MPa, 280-325ºC*	TBS-4: *2nd Helium-Cooled Ceramic Breeder (CN)* ➔ *Helium at 8.0 MPa, 300-500ºC*

Note: Test Blanket Systems are designed & procured by the ITER Members that retain the TBSs ownership. IN, RF & US contributes to the TBM Program by performing supporting R&D.

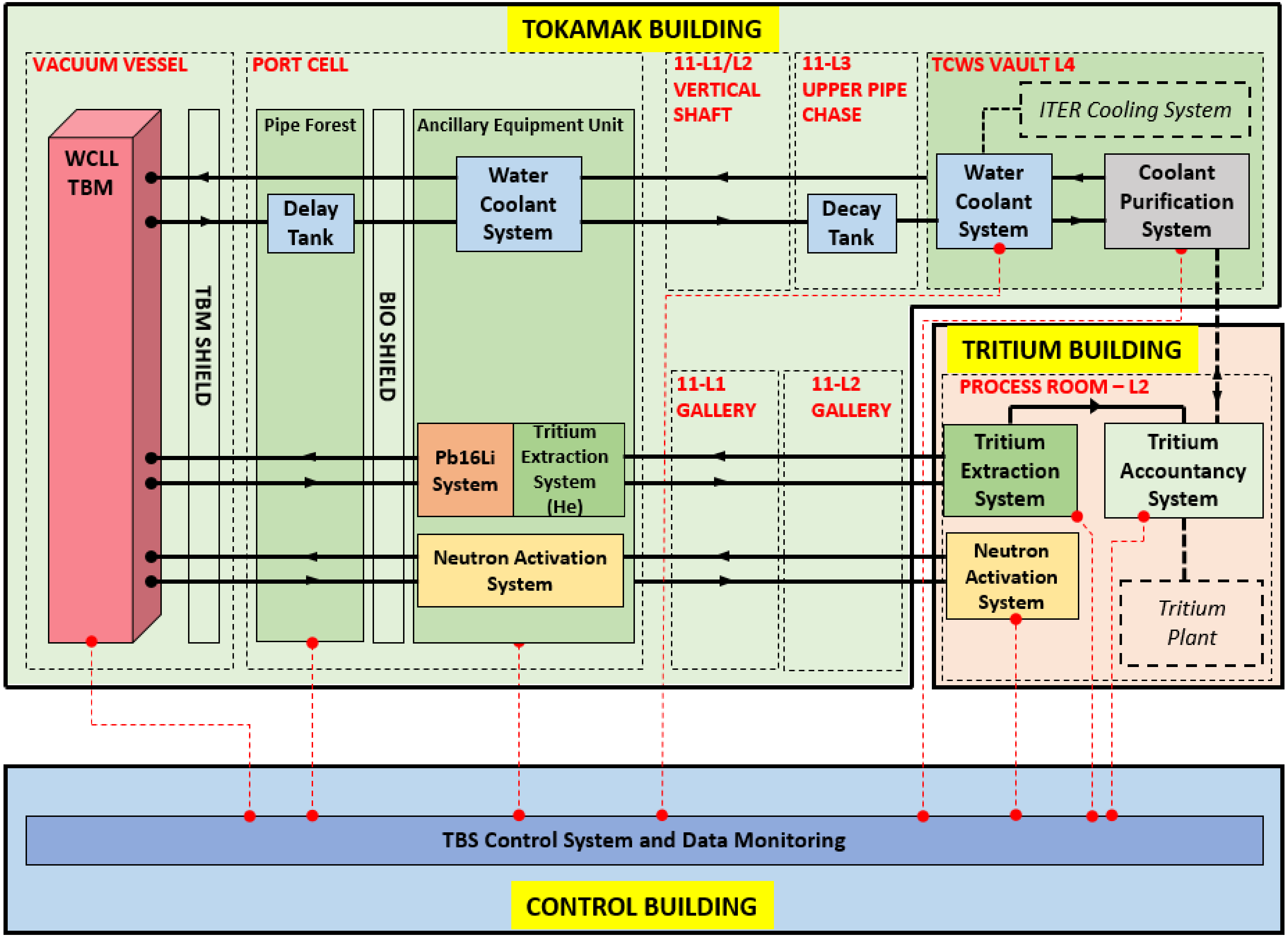
Duration of the TBS operations

The TBS forming the Initial Configuration are expected to be operated for several campaigns after the First Plasma.

After such testing campaigns, different TBSs (possibly designed and procured by different ITER Members) could be installed and operated for some further campaigns. They would form the “second TBS Configuration” to be discussed and agreed several years before their installation.

Main Characteristics for the 4 InCo TBMs and associated Ancillary Systems

<table>
<tr><th>Equatorial Port #16</th><th>Equatorial Port #18</th></tr>
<tr><td>TBS-1 – WCLL-TBM (proposed by EU)
▪ Eurofer Steel (structure), Pb16Li (multiplier/breeder).
▪ Coolant: H_2O at 15.5 MPa, 280 /325°C.
▪ T-removal gas (from Pb16Li): He + 0.1% H_2 at 0.4 MPa.
▪ Maximum Tritium production: 20-25 mg/day</td><td>TBS-2 – HCCP-TBM (joint KO/EU design)
▪ RAFM Steel (structure), Be pebbles (multiplier); Li_4SiO_4 or Li_2TiO_3 pebbles (breeder).
▪ Coolant: He at 8 MPa, 300/500°C.
▪ Purge gas: Helium + 0.1% H_2 at about 0.3 MPa.
▪ Maximum Tritium production: 20-25 mg/day</td></tr>
<tr><td>TBS-3 – WCCB-TBM (proposed by JA)
▪ F82H Steel (struct.), Be pebbles (mult.), Li_2TiO_3 pebbles (breeder).
▪ Coolant: H_2O at 15.5 MPa, 280 /325°C;
▪ Purge gas: He + 0.1% H_2 at 0.1 MPa.
▪ Maximum Tritium production: 20-25 mg/day</td><td>TBS-4 – HCCB-TBM (proposed by CN)
▪ RAFM Steel (structure), Be-pebbles (multiplier), Li_4SiO_4 pebbles (breeder).
▪ Coolant: He at 8 MPa, 300/500°C.
▪ Purge gas: He + 0.1% H_2 at about 0.3 MPa.
▪ Maximum Tritium production: 20-25 mg/day</td></tr>
</table>



Scheme of a water-cooled TBS: ➔ Example of the WCLL-TBS

With indication of the main Locations in the various rooms of the ITER Tokamak Complex

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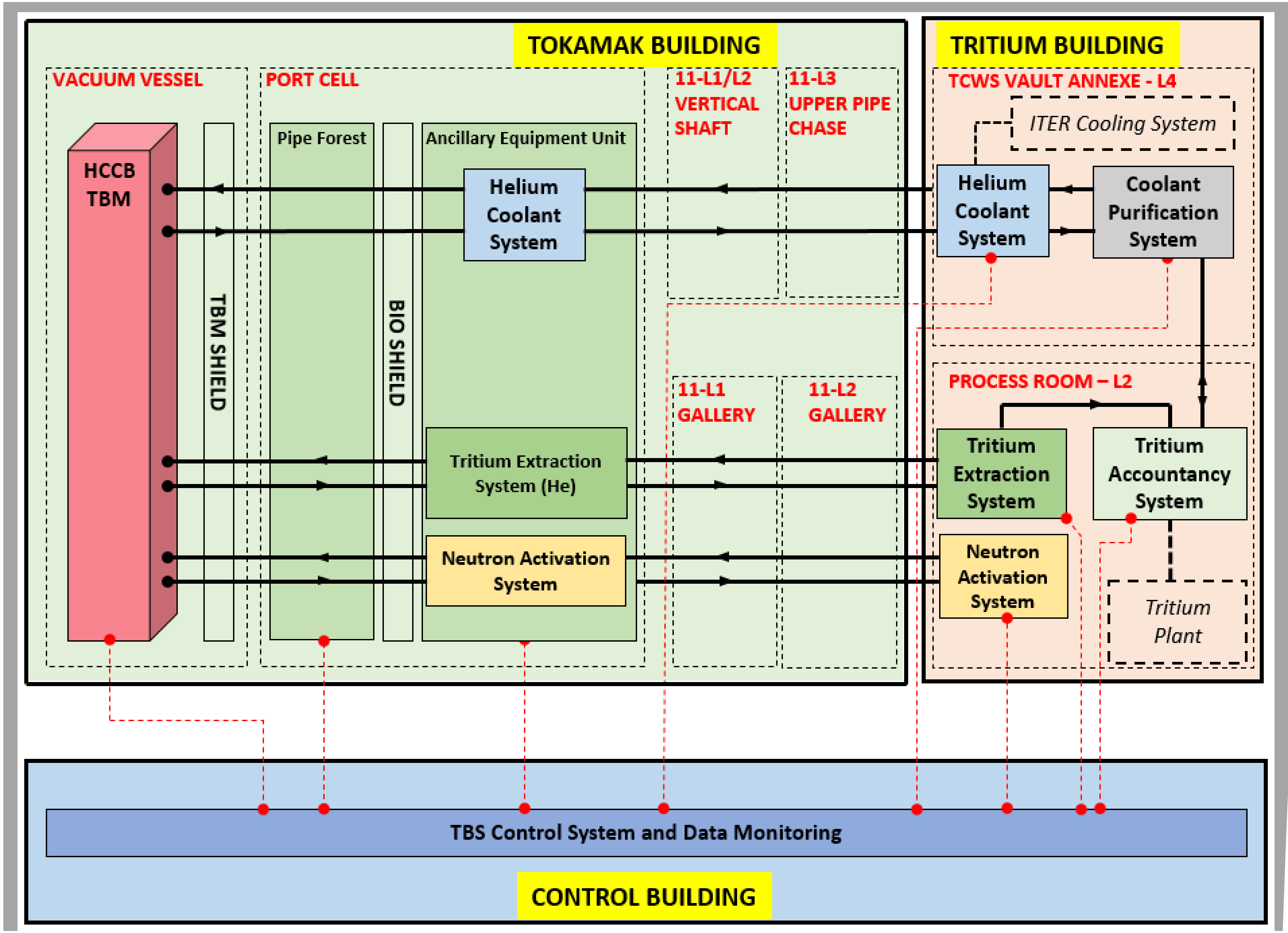
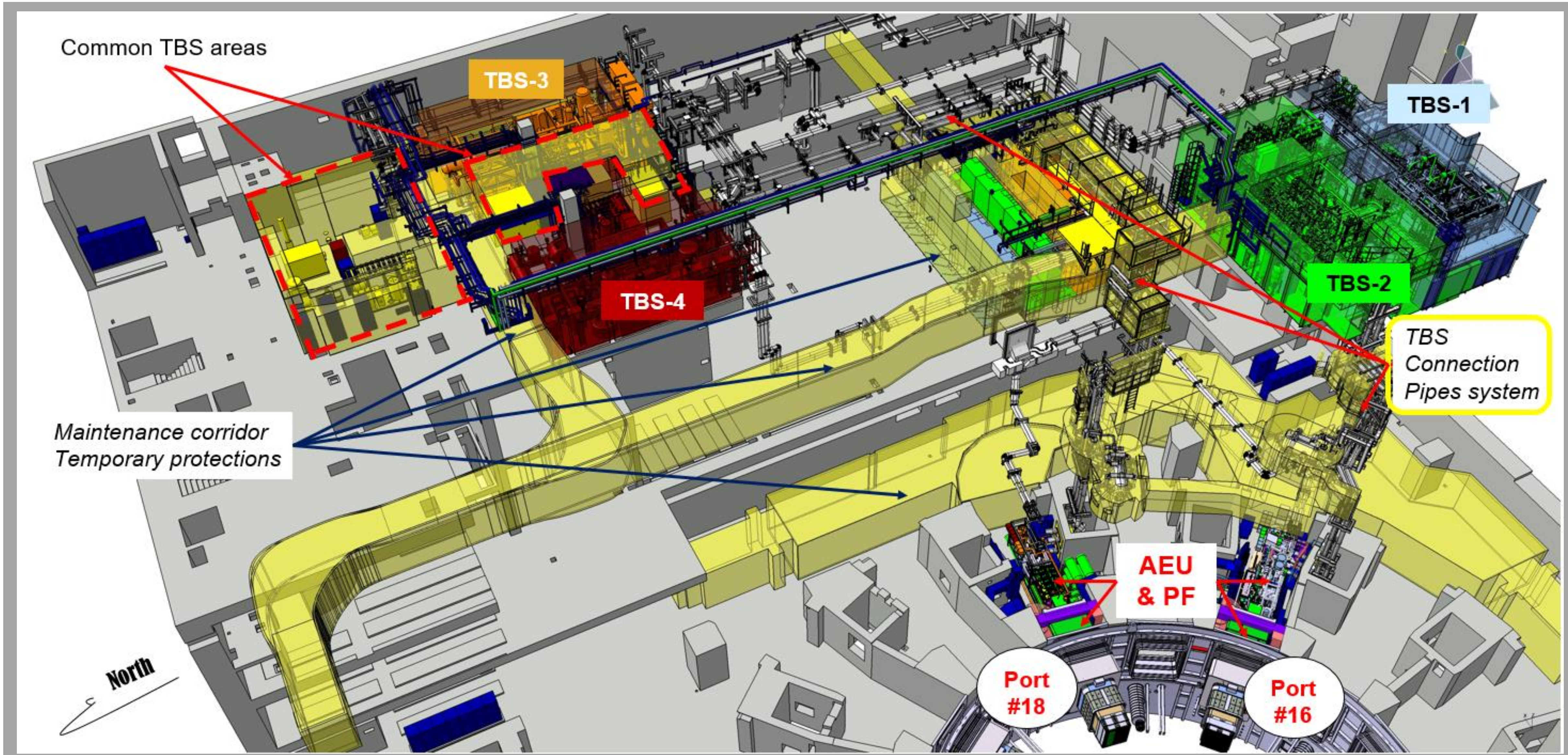
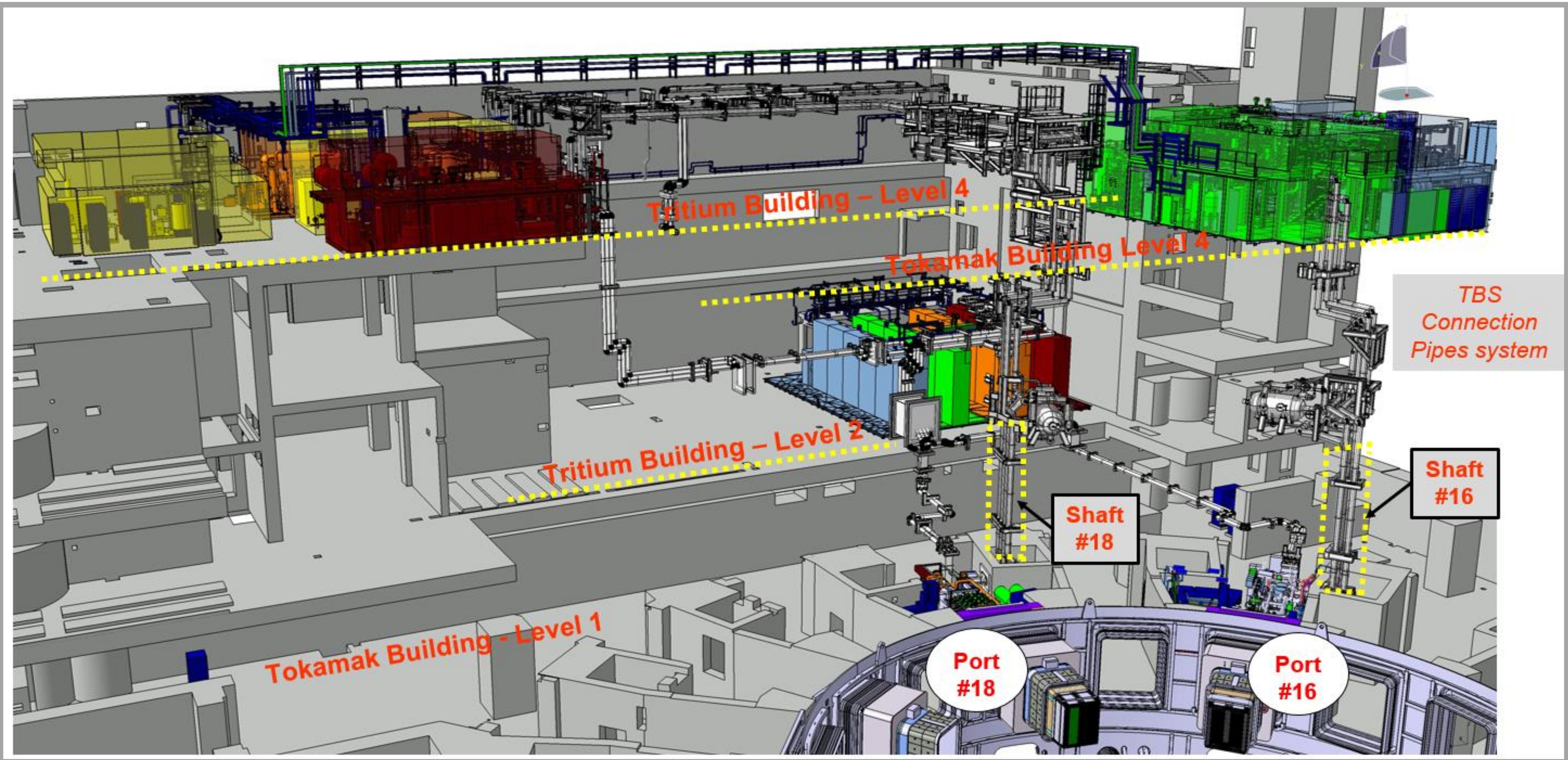


Scheme of a Helium-cooled TBS: ➔ Example of the HCCB-TBS

With indication of the main Locations in the various rooms of the ITER Tokamak Complex



Overall View and location of the 4 Test Blanket Systems within the Tokamak Complex



Vertical View and location of the 4 Test Blanket Systems within the Tokamak Complex

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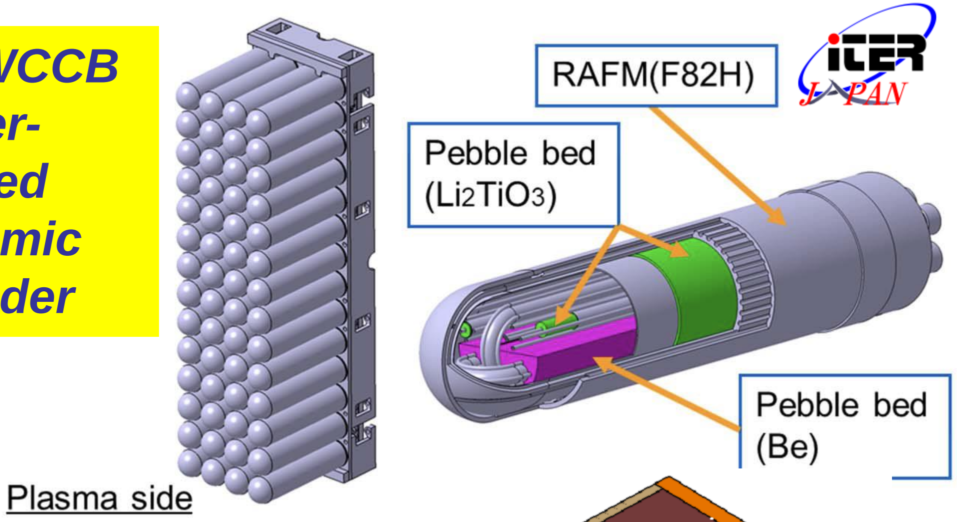
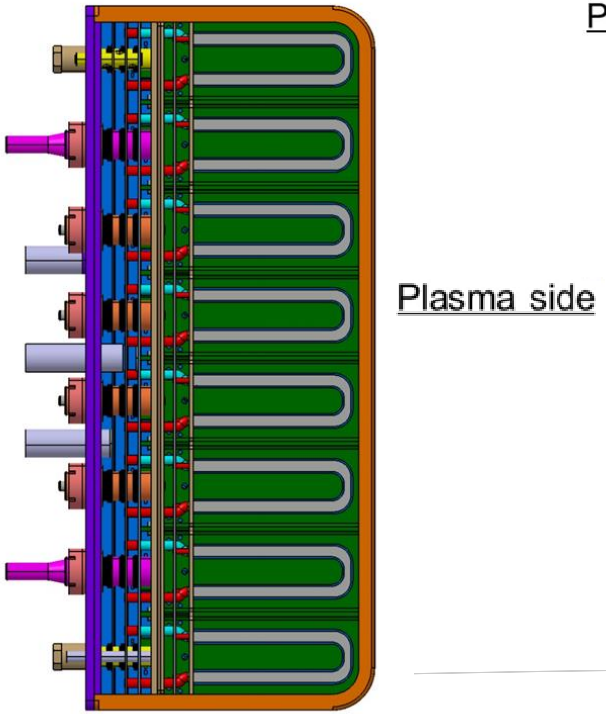
View of the internals of the four TBMs of the Initial Configuration

➔ The design (including materials) of each TBM is based on same structures and materials of the corresponding DEMO breeding blanket ➔ fully relevant mock-ups !!

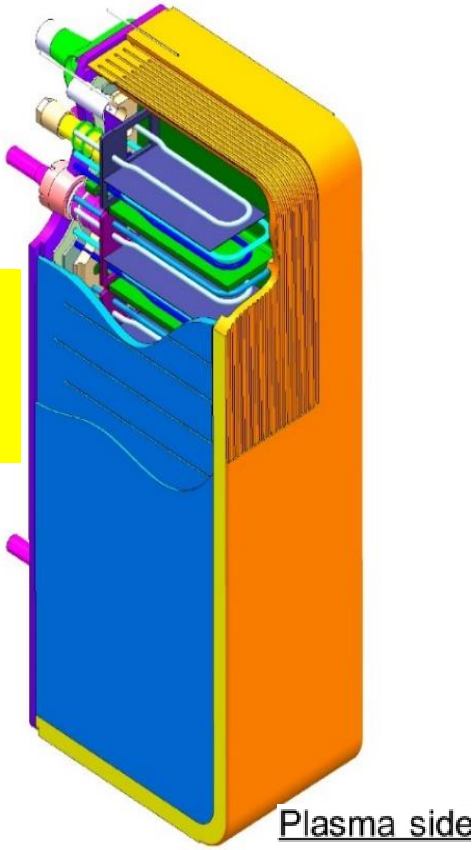
JA-WCCB Water-cooled Ceramic Breeder



KO/EU-HCCP Helium-cooled Ceramic Pebble

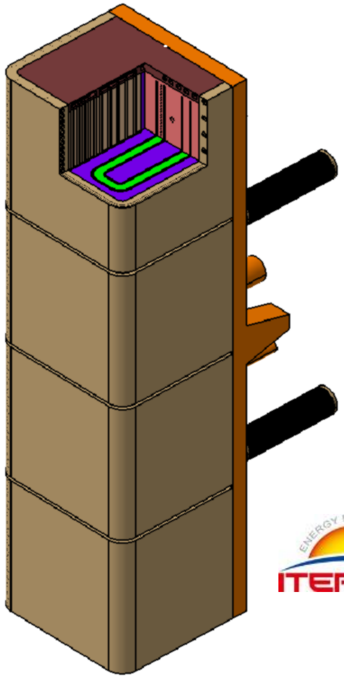


EU-WCLL Water-cooled Lithium-Lead



CN-HCCB Helium-cooled Ceramic Breeder

Plasma side



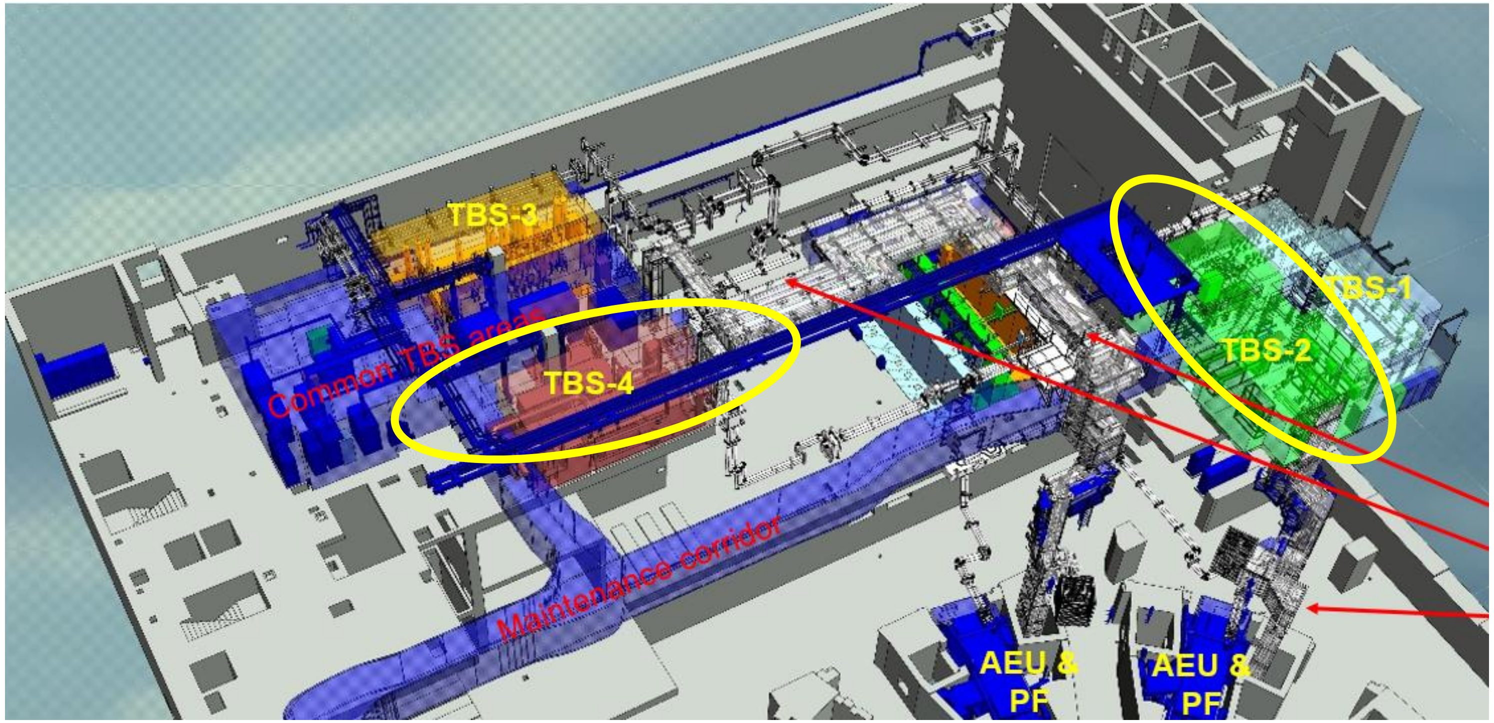
Main features of an Helium Cooling System (+CPS) for TBSs

For each of the two He-coolant system + CPS in ITER:
Footprint : ~80 m^2
Height ~7 m

Main characteristics of the Helium Coolant Systems:

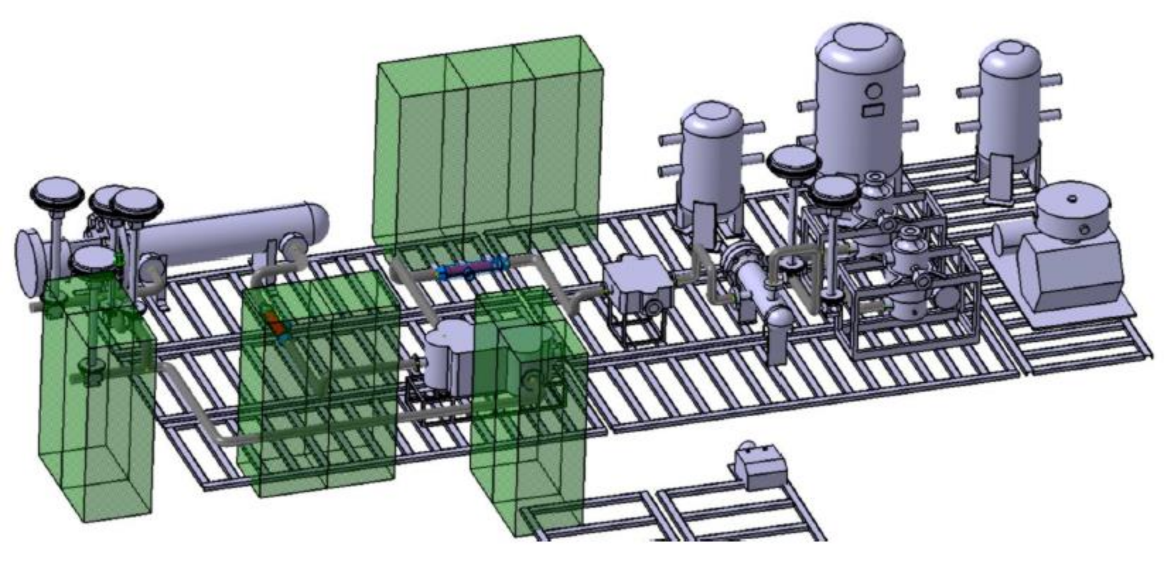
- Inlet/Outlet Temperature: 300/500 °C
- Operating Pressure: 8 MPa
- Flow-rate: 1.3 kg/s
- Total He inventory: ~40kg
- Temperature of the secondary water coolant: inlet/outlet 31/42 °C

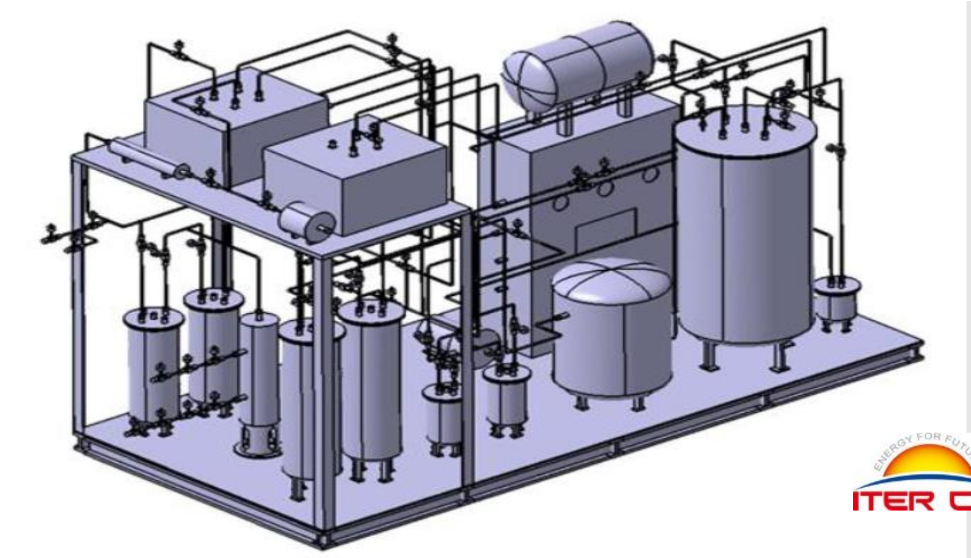
Need of an electrical heater of ~400 kW at 400 Vac to increase the He-temperature at inlet value.



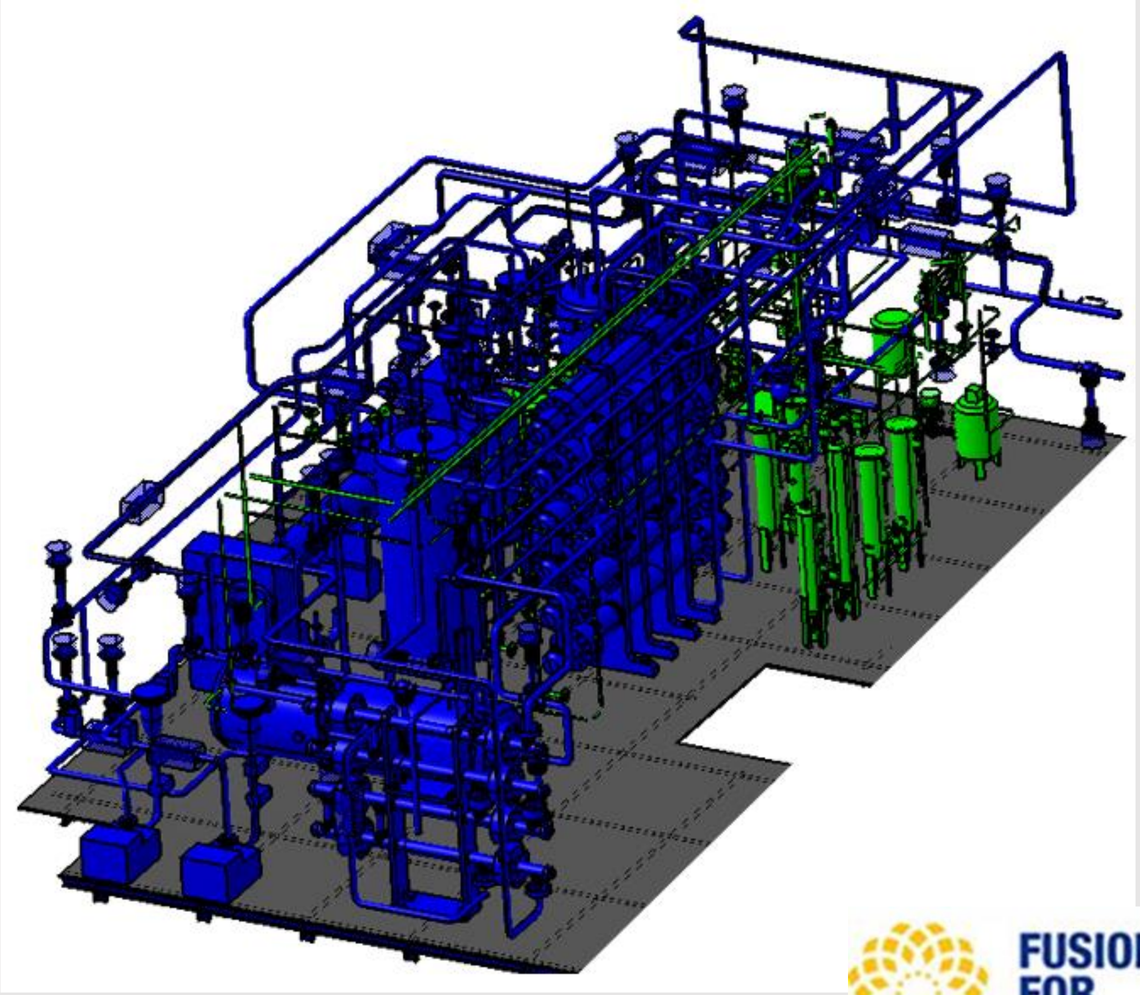
Typically, need of two circulators per system in order to improve reliability (redundancy).

- Cooled by chilled water (inlet ~6 °C), operating with He at low temperature (<100 °C)
- Power supply: ~300 kW at 400 Vac.

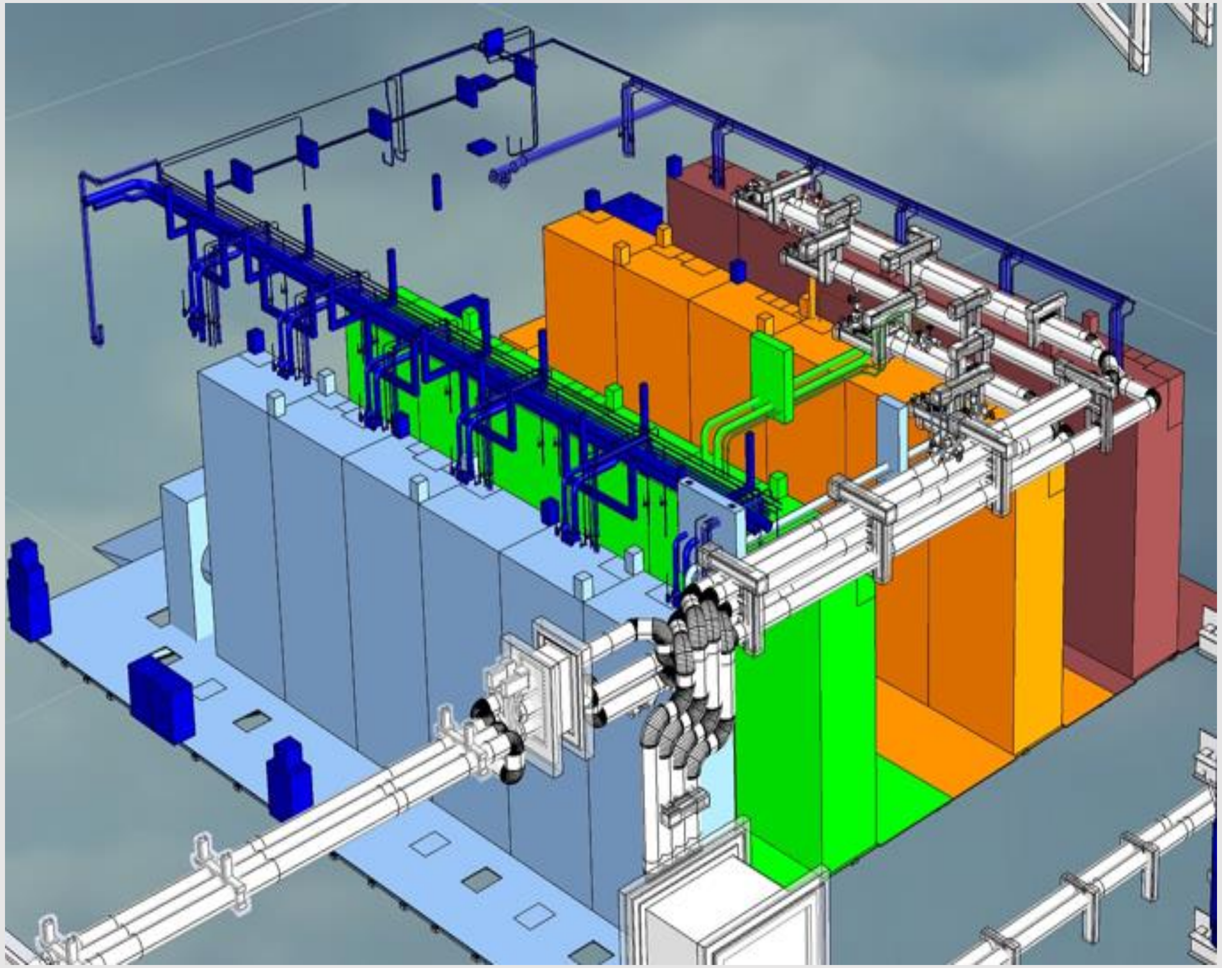




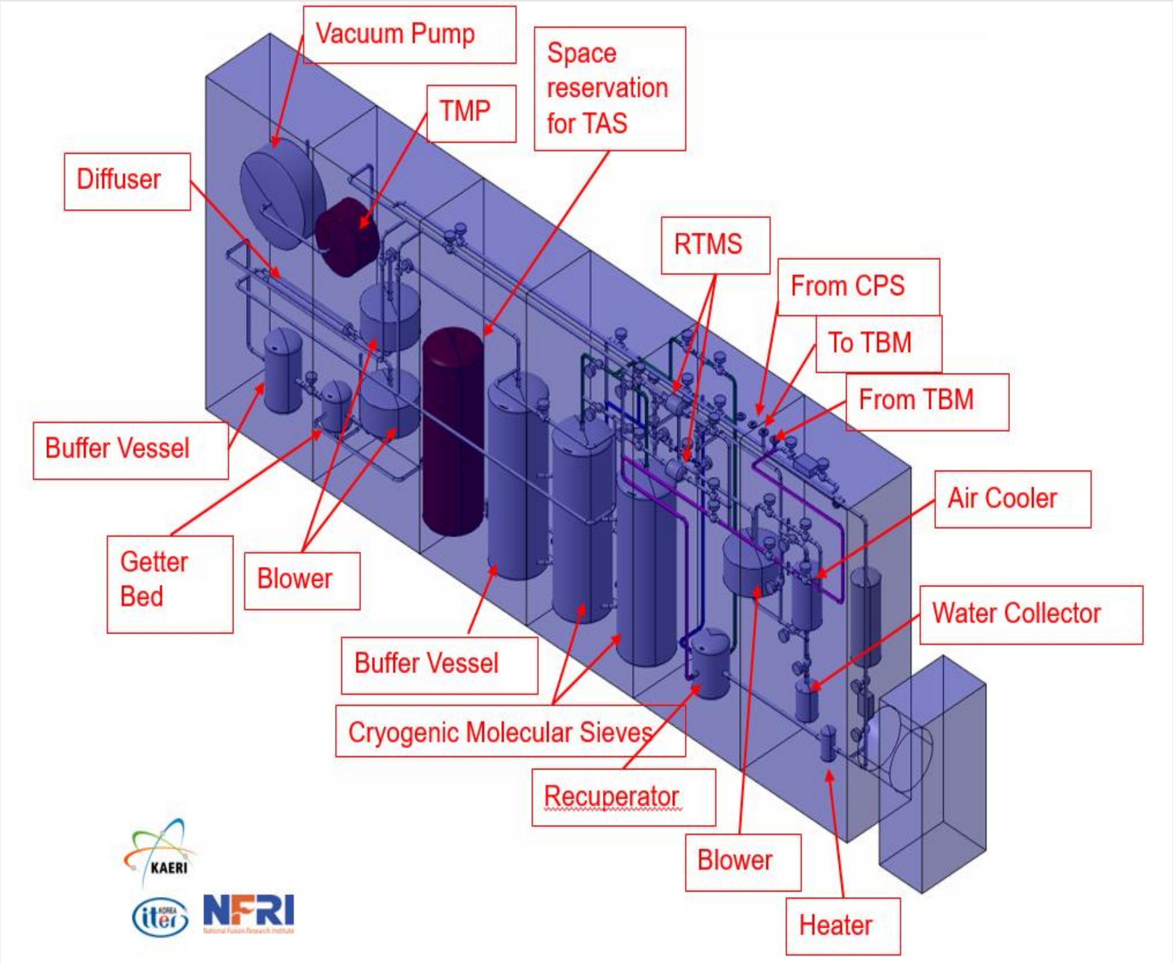
Layout & components of the HCCB HCS (above) and CPS (below)



Layout & components of the WCLL WCS



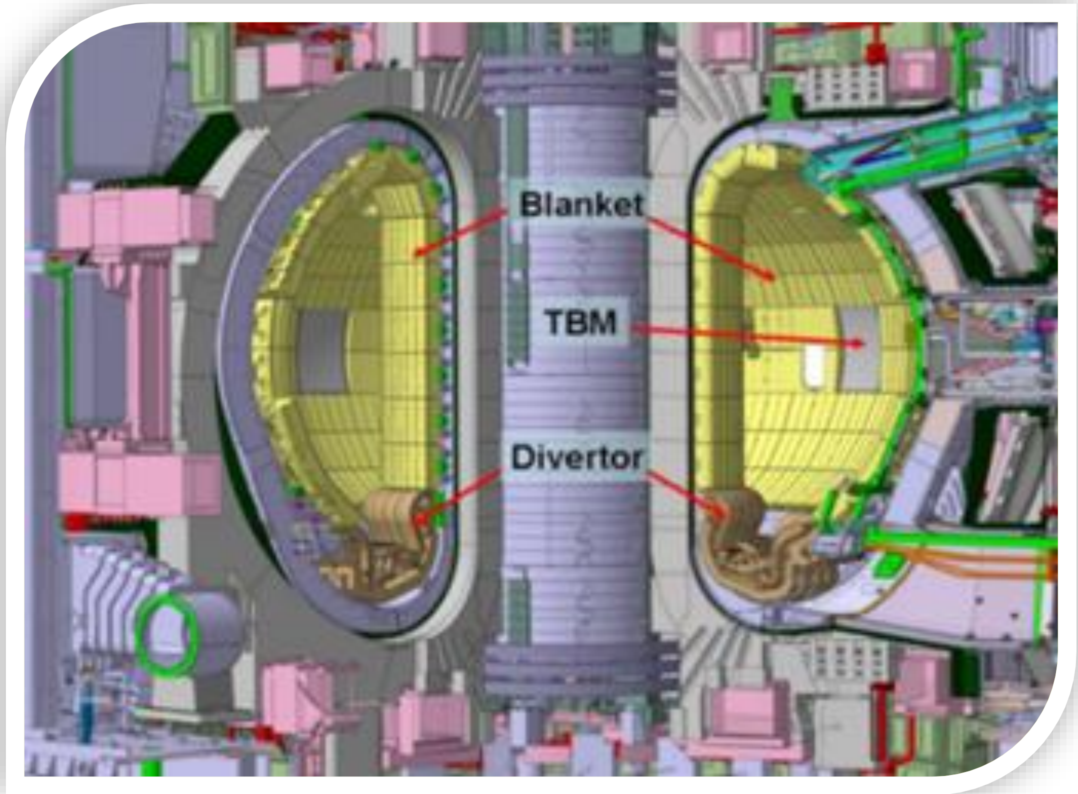
Volume space reservation for the 4 TBSs



Example of components integrated in the space (e.g., glove-box)

Details of room 14-L2-24: 4 Tritium Extraction Systems + 4 Tritium Accountancy Systems + 4 Neutron Activation Systems

Outline of the presentation



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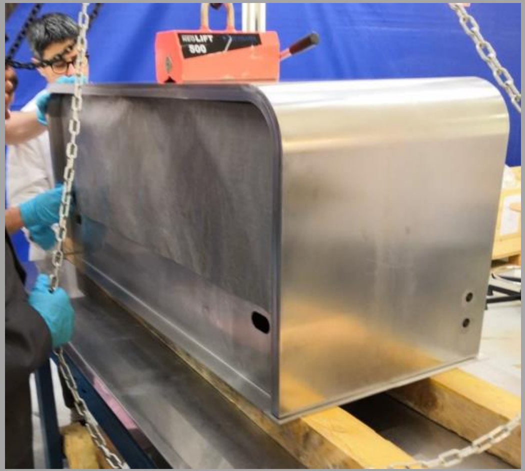
R&D on Reduced Activation Ferritic/ Martensitic Steel Eurofer-97 (EU)

Manufacturing technologies using EUROFR-97



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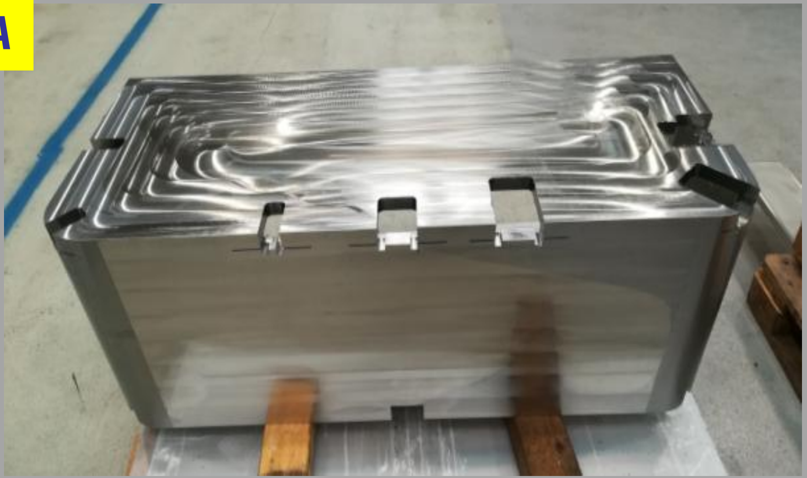

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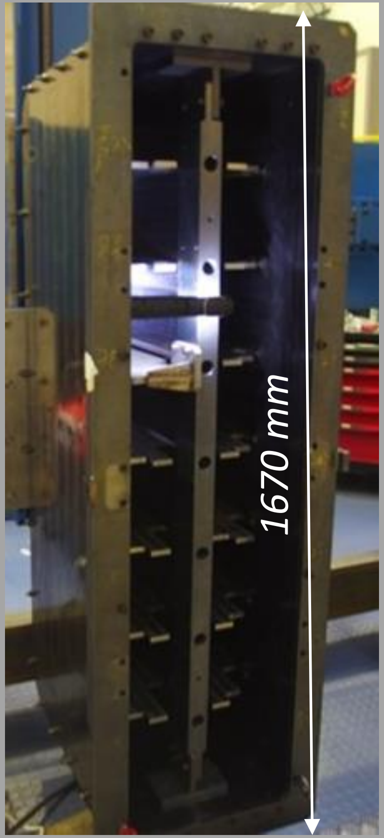

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A - Welding external walls by HIPing : 2-step HIP process, NDT and DT completed

B - Welding TBM internals by robot : TIG welding robot assembling TBM mock-ups with narrow space accessibility

Each IM develops RAFM steels:
- ***AARA** by Korea*
- ***CLF-1** and **CLAM** by China*
- ***F82H** by Japan*

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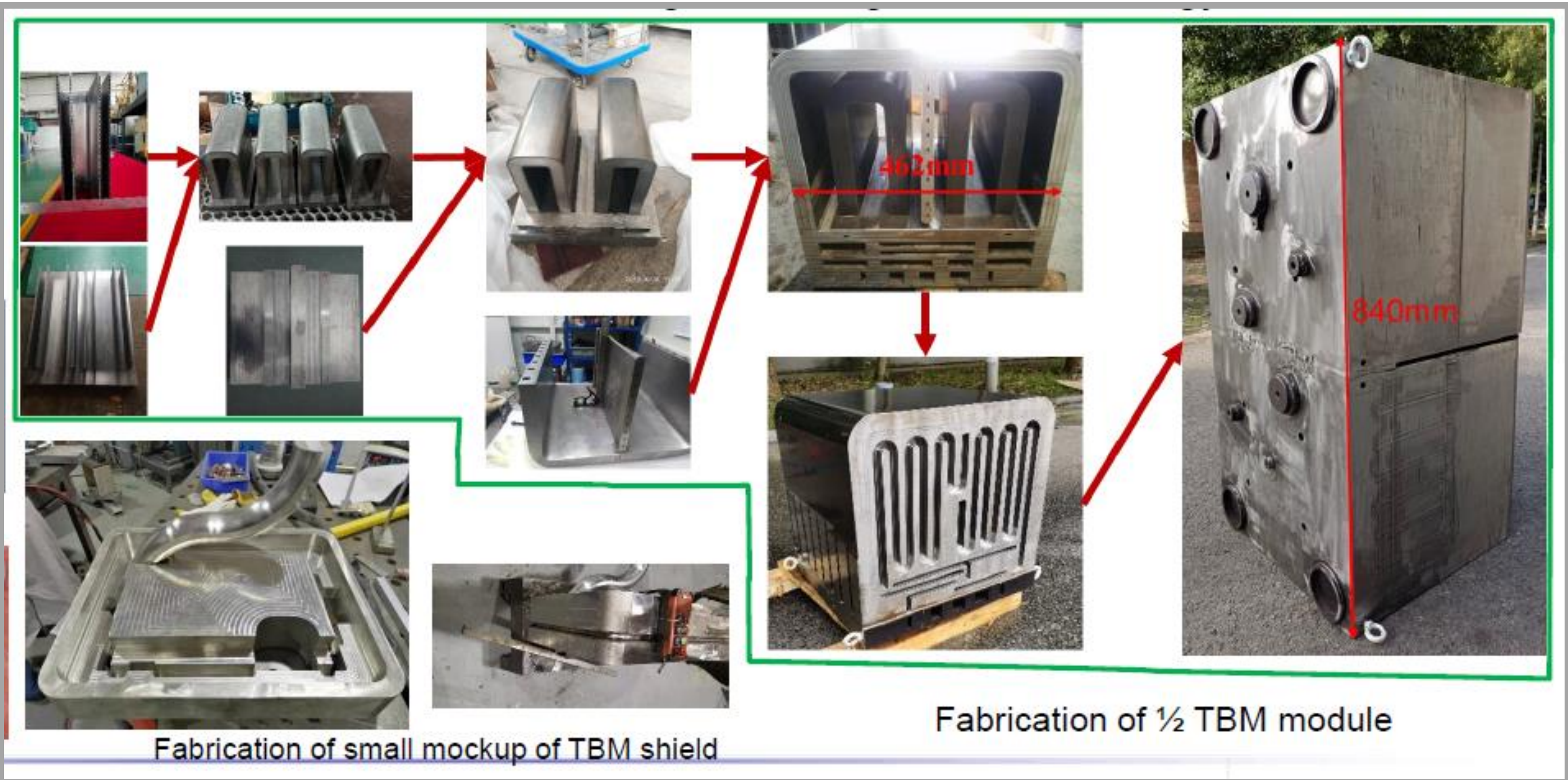
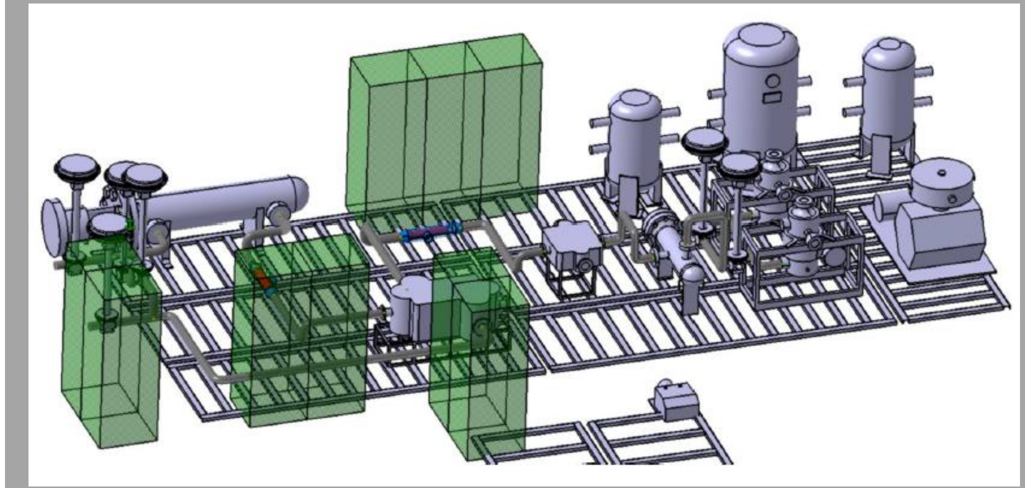
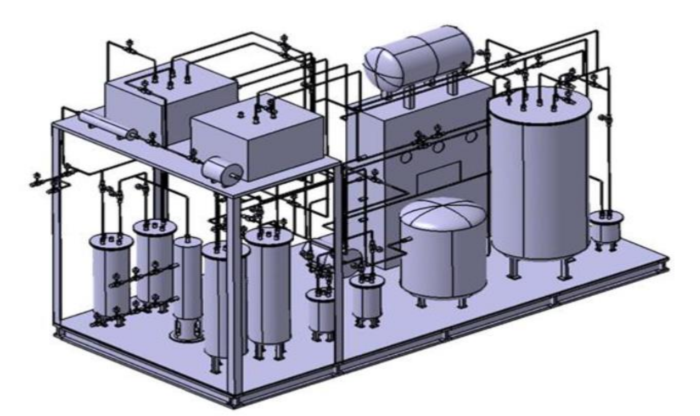


R&D on HCCB-TBM fabrication process and technology are on-going, developed and verified by making components mock -ups and semi–prototype of TBM box in CLF-1 (including NDT technology).





Example of the Helium-coolant system for HCCB-TBS (CN)

Main characteristics of the Helium Coolant System:

- Inlet/Outlet Temperature: 300/500 °C
- Operating Pressure: 8 MPa
- Flow-rate: 1.3 kg/s
- Total He inventory: ~40kg
- Temperature of the secondary water coolant: inlet/outlet 31/42 °C

Constructed He-coolant loop prototype: HeCEL-3 (2.5kg/s@8-12MPa & 550℃) with electromagnetic bearing circulator and Printed-Circuit Heat Exchanger (PCHE)

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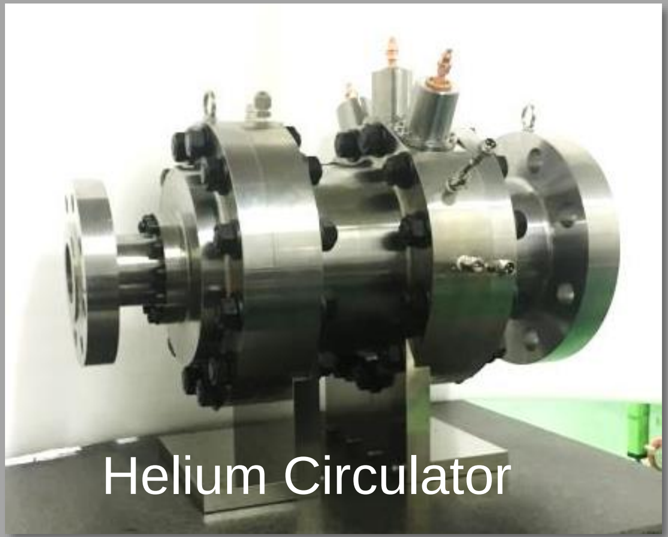
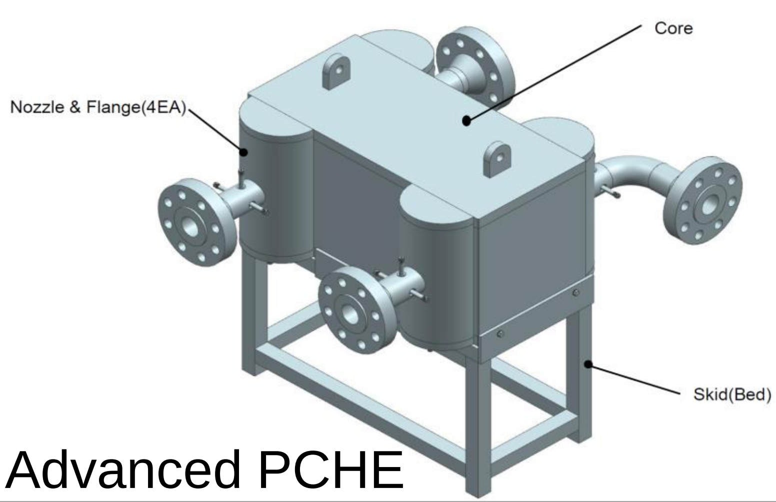


Helium Circulator



Advanced PCHE

Example of the Helium-coolant system for HCCP-TBS (KO)

Upgrade of HCS test facility named Helium Supply System (HeSS) (10 MPa, 550°C) for performance tests:

- Advanced PCHE (Printed Circuit type Heat Exchanger) = Compact HX with lower pressure drop (<50 kPa)
- Air bearing type circulator (ongoing R&D)

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TRIEX-II experimental facility at ENEA-Brasimone

Operated to determine the extraction efficiency of a gas liquid contactor for WCLL-TBS.

Experimental campaigns have been carried out using D_2 as dissolved gas in Pb16Li stripped by He+H_2 (0.1%vol.).

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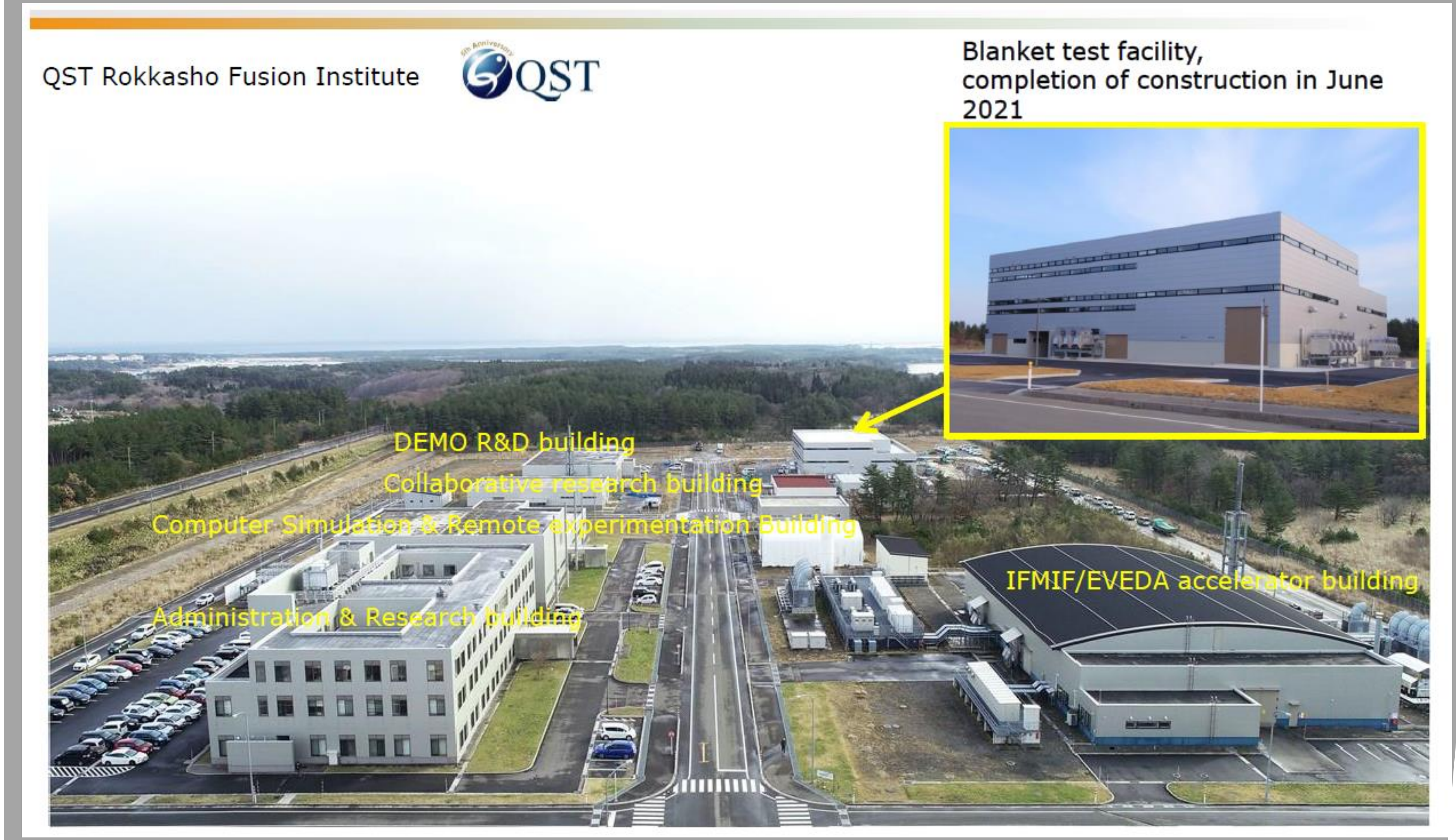

View of the Blanket Test Facility completed in Rokkasho site in July 2021

(first of the kind in the world)

Fully dedicated to WCCB-TBS and DEMO Breeding Blankets R&D

Planned/on-going tests are for instance:

- Heat load test on TBM FW
- In-Box LOCA test
- Reaction between water and Beryllium pebbles
- Cooling-water flow and accelerated corrosion test.

High heat flux test

Flow accelerated corrosion test

Coolant leakage propagation test

Be – Water reaction rate measurement

Test Facilities for physical mock-ups test related to the WCCB-TBS

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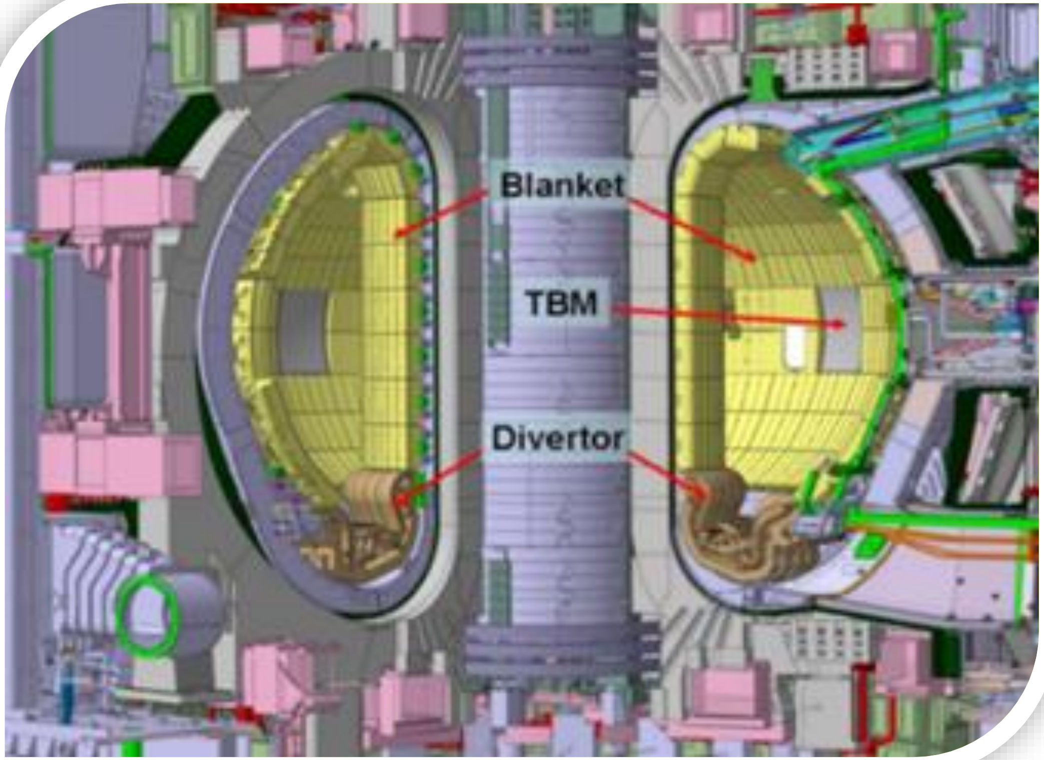

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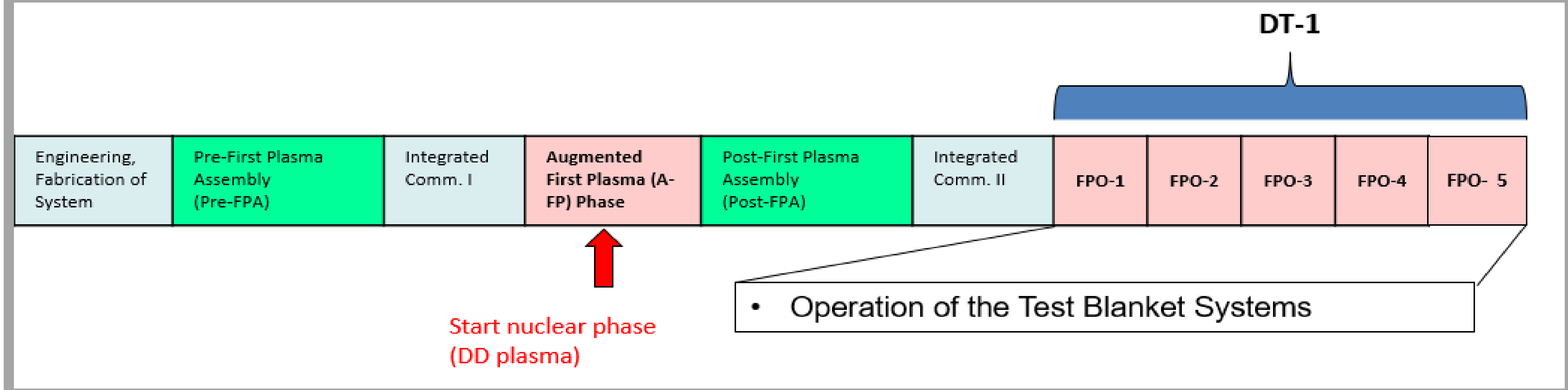
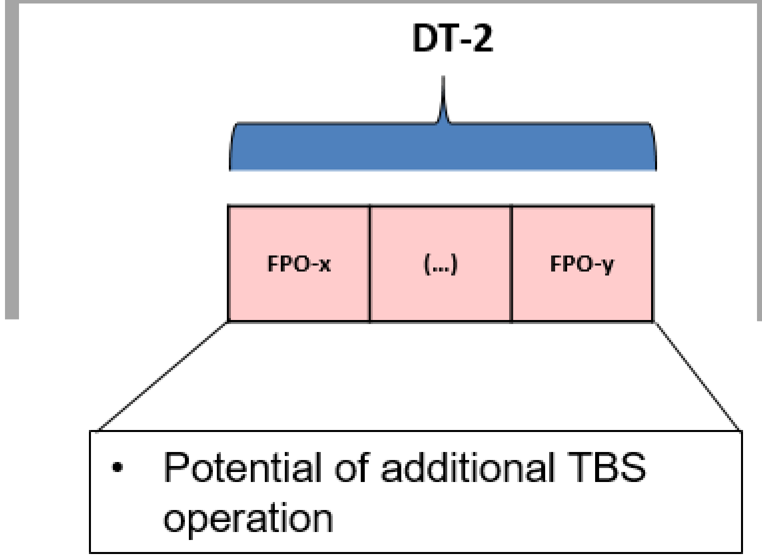
A new ITER Scenario based on a step-wise approach to safety demonstration is under development and will be finalized in 2024





Three Phases:

1. **AF-P : Augmented First Plasma Phase**
2. **DT-1 : nuclear phase formed by 5 campaigns lasting about 10 years (with neutron fluence reduced to about 1% of the project specification goal of 3×10^{27} neutrons)**
3. **DT-2 : completing the achievement of ITER objectives**

Duration of the TBS operations

The TBSs forming the Initial Configuration are expected to be operated for several campaigns starting at the beginning of DT-1 (i.e., FPO-1).

Phase	Duration (months)	Plasma species	Neutron fluence ($x10^{24}$)	Heat flux on TBM FW (MW/m^2)	DT Neutron flux on TBM FW (MW/m^2)
FPO-1	16-18	H, DD	<0.02	0.2 (peak)	Negligible
FPO-2	16	DD	<0.3	0.2 (peak)	Negligible
FPO-3	16	1/5 shots DD and 4/5 shots from (D+10%T) up to DT	<2.7	0.25 (peak)	~0.3 (peak)
FPO-4	16	(D+10%T) gradually up to DT	<9.0	0.3 (peak)	~0.7 (peak)
FPO-5	16	some (D+10%T) quickly up to DT	<18.0	0.3 (peak)	~0.7 (peak)
		Total DT-1	~30	0.3 (peak)	~0.7 (peak)
		Total DT-2	~3000	0.3 (peak)	~0.7 (peak)

Present estimation of the main characteristics of the expected DT-1 campaigns relevant for the TBM Program

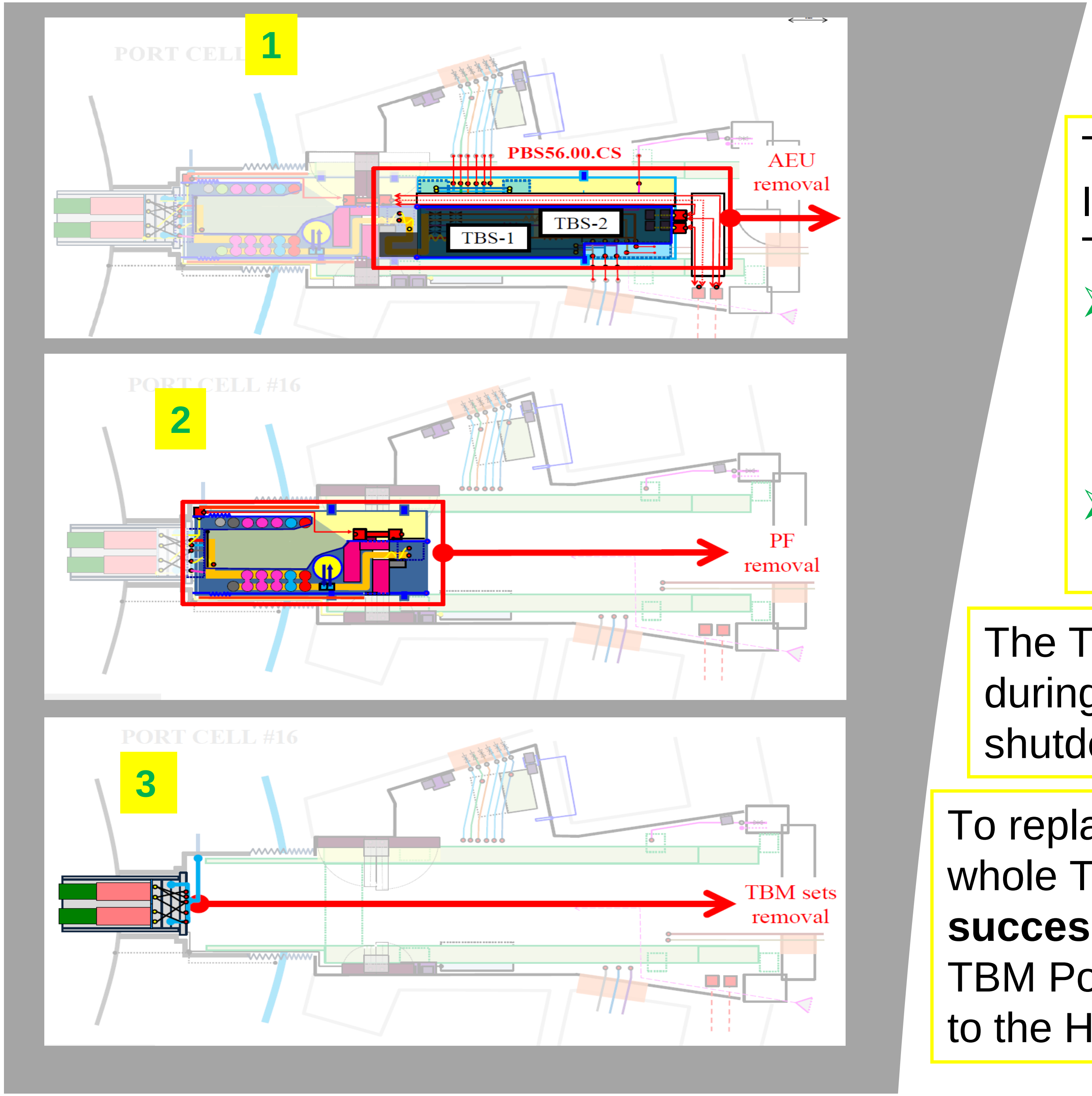
Peak values mean values occurring during at least one pulse (=design values)

Main Overall Objectives of the TBM Program

- ✓ **Operation** of a whole TBS including all sub-systems from the Control Room (non-nuclear). Therefore, simulation for the first time of the operation a DEMO breeding blanket systems under relevant tokamak environment. Demonstration of capability of withstanding disruption and capability of surface heat flux management in TBM FW.
- ✓ **Validation** of the nuclear response predictions with existing modelling codes and nuclear data during TBS operations ➔ Extrapolation to DEMO BB Tritium Breeding.
- ✓ **Experimental verification** of the TBMs thermo-mechanical behaviour at relevant temperature and volume heat sources using RAFM steels as in DEMO.
- ✓ **Demonstration** of heat removal with high temperature/high pressure coolant relevant for electricity production in DEMO BBs.
- ✓ **Demonstration** of the tritium management, including validation of Tritium extraction techniques and permeation reduction capability. **Validation** of modelling for extrapolation to DEMO.
- ✓ Tritium Breeding Blanket **performance** for an extended period of time in order to obtain initial reliability data.
- ✓ Post-Irradiation Examinations (**PIEs**) for TBM material/process data in the IMs premises (after transfer to the IMs premises from the IO Hot Cell Facility)

"The overall objective of the TBM Program is to demonstrate, in a relevant tokamak-fusion environment, the feasibility of the self-sufficiency of the Tritium production using relevant coolant temperature and pressure needed for the next tokamak generation (e.g., DEMO)"

Adopted Strategy for the TBM Program Testing Plan



The TBM testing plan has to be adapted to the ITER operation plan ➔ Need of testing several TBM versions per each TBS in order to:

- Consider the different ITER operating conditions in each campaign while simulating the DEMO conditions (i.e., engineering scaling),
- Install/implement different instrumentation in order to measure different parameters.

The TBMs will be replaced **after each campaign** during each Long Term Maintenance (**LTM**) shutdown *(approximately each 2 years).*

To replace a TBM implies to replace the whole TBM Port Plug and **to transfer successively** all the components in the TBM Port Cells (i.e., AEU, PF and TBM-PP) to the Hot Cell using the RH Transfer Cask.

Detailed technical testing objectives derived from the overall objectives

versus

Plasma-pulses parameters drivers

(first part)

Expected simulations/results from the TBS testing	**Driver plasma-pulse parameters for TBSs testing objectives**
Capability of TBMs to withstand disruptions (seen as accident)	Plasma current and plasma performance perturbations. *Integration of appropriate strain gauges and displacement sensors in the TBMs.*
Thermo-mechanical gradients in the specific TBM First Wall to check its structural load capacity	(*Thermal response time constant of 80-100 s*) Surface heat flux on the FW – Plasma pulse length. *Typical implemented measurements devices include thermocouples and strain gauges.*
Thermomechanical load capacity of the whole TBM-box	(*Time constant of some hundreds seconds*) Surface heat flux on the FW - neutron and gamma-ray fluxes (=volumetric heat) – Plasma pulse length. *Typical implemented measurements devices include thermocouples and strain gauges.*
Capability of circulating the Lithium-Lead (melting point: 235ºC) – For WCLL-TBS only.	Neutron flux (=volumetric heat) – Plasma pulse length - Magnetic field (MHD). *Typical devices are mass flowmeters and pressure drops detectors compatible with the Pb16Li.*

Detailed technical testing objectives derived from the overall objectives

versus

Plasma-pulses parameters drivers

(second part)

Expected simulations/results from the TBS testing	**Driver plasma-pulse parameters for TBSs testing objectives**
Capability to release the Tritium from the Li-ceramic pebbles requiring high temperatures, up to peak temperatures of 800ºC-900ºC to confirm pebble-bed performance	Neutron flux – Plasma pulse length – Coolant temperature and flow-rate. *Typical devices are thermocouples and devices measuring the tritium concentration in various TES locations.*
Demonstrate the Tritium extraction efficiency of the Tritium extraction systems ➔ need to reach quasi-equilibrium of the Tritium concentrations in TBS components	(*time constant of some thousands seconds*) – Neutron flux– plasma pulse length – pulse repetition time - number of pulses per day. *Specific devices need to be developed to measure tritium concentration in various TES locations.*
Estimate the Tritium permeation into the coolant, the Tritium coolant purification system efficiency ➔ need to reach quasi-equilibrium of the Tritium concentrations in all TBS components	*(time constant of some thousands seconds)* – Neutron flux – plasma pulse length – pulse repetition time - number of pulses per day. *Specific devices might need to be developed to measure tritium concentration compatible with high-temperature high-pressure environment.*
Behaviour of steel, Li-ceramics and Be-multiplier under irradiation	Limited neutron fluence seen by each TBM during each campaign (replaced during each LTM).

How to achieve the detailed technical objectives during the DT-1 campaigns (FPO-1, FPO-2, FPO-3, FPO-4 and FPO-5) within the limit of the DT-1 neutron fluence?

It is planned to have some ITER operation days (in particular during FPO-3, FPO-4 and FPO-5) fully dedicated to the TBM Program with the following main characteristics:

A **Fusion Power of at least half of the maximum** (i.e., about 250 MW). It gives sufficient heat loads on the TBM FW and a sufficiently high neutron wall loading in order to have a sufficient tritium production rate.

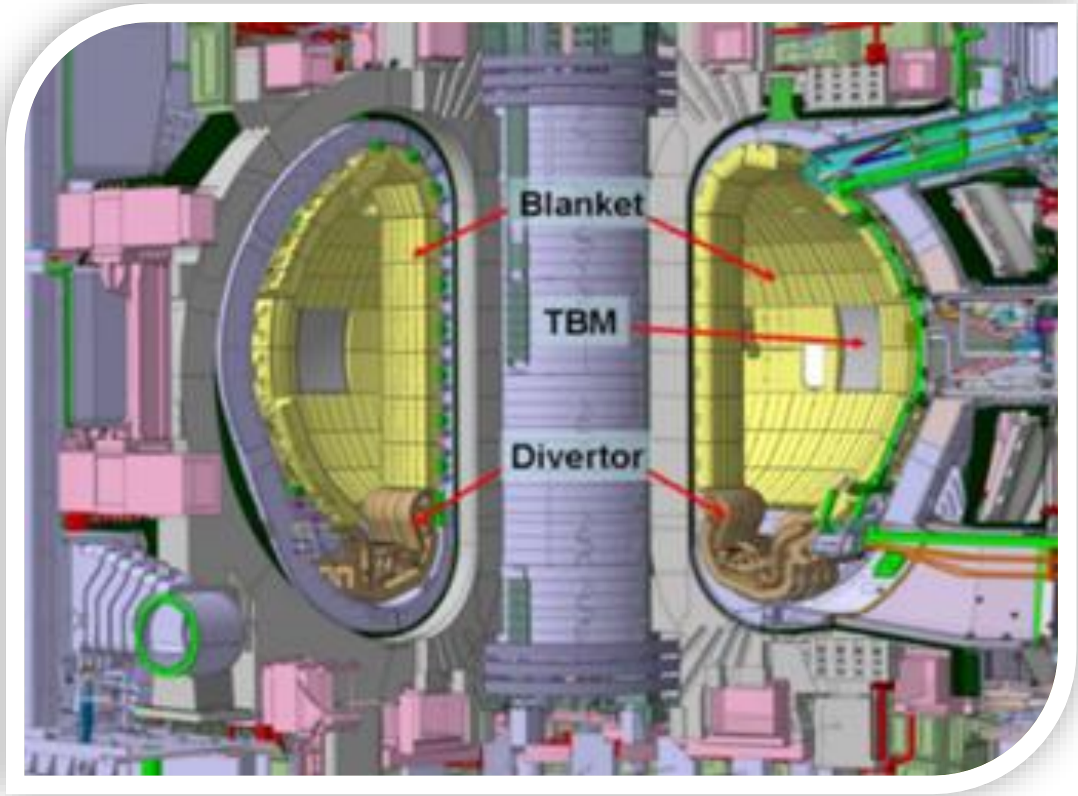
A pulse length with **a flat top of at least 300 seconds** in order to achieve the thermal quasi-steady state in TBMs.

A high repetition time, for instance having around 32 “back-to-back” pulse for 16 hours (a day with 2 shifts), in order to achieve a pseudo equilibrium among the various tritium-related characteristics (i.e., production, extraction, permeation and coolant tritium purification).

Reinforced R&D dedicated to TBS operations:

- Develop **adequate measurement systems** and detectors
- Measured results need to be interpreted **with adequate modelling** and simulation capability, in particular for tritium-related data.

Outline of the presentation



A. Rational of the ITER TBM Program

B. Main features of the Test Blanket Systems

C. Examples of on-going R&D on Test Blanket Systems and Components

D. ITER TBM Program Testing Plan and main overall objectives

E. Technical objectives and achievements of the TBM Program in support of the DEMO TBBs development

F. Final Considerations and Conclusions

TBM Program Achievements prior the TBS operations in ITER

"Much before the TBSs operations in ITER, the R&D performed during the TBM/TBS design and manufacturing phases give already a large amount of data and information in support of the design of a TBB for DEMO"

- The previous slides have addressed the objectives of the TBM Program during TBS operations starting in the FPO-1 campaigns. These objectives can be fulfilled if and only if the ITER TBSs use DEMO BB design parameters, DEMO-relevant BB materials and DEMO-relevant sub-systems components.

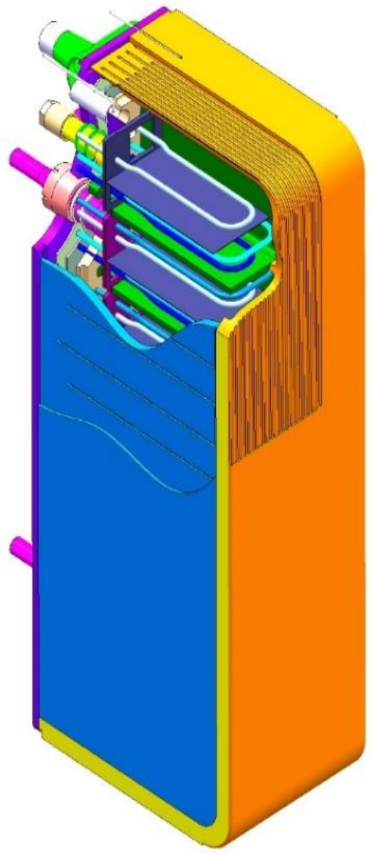
- It means that the results of most R&D performed for the TBSs design and manufacturing is relevant for the DEMO TBB. In some cases it is directly applicable.

- As examples, the following five aspects are briefly addressed:
  1. **TBM structural materials**
  2. **Tritium management**
  3. **TBS Safety aspects**
  4. **Measurements and control systems**
  5. **TBS components reliability/availability**

❶ TBMs structural materials (1/2)

Mastering RAFM steels manufacturing

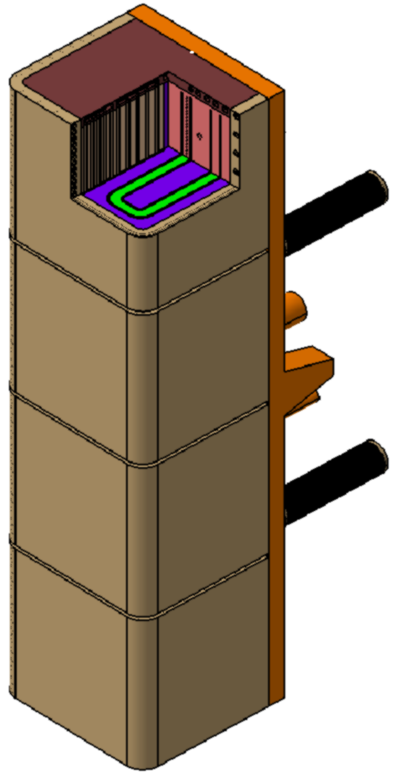
- ✓ **The TBMs are mock-ups** of the corresponding DEMO TBBs since they use the same structural materials: Reduced-Activation Ferritic/Martensic (RAFM) steels and relevant design architecture and manufacturing technologies.
- ✓ Therefore, **the RAFM steel manufacturing technologies** developed for TBMs have direct consequences on the choices to be made for TBBs. Each procuring ITER Member has developed a specific RAFM steel for its TBM and future TBB application: EUROFER-97 in EU, CLF-1 and CLAM in CN, F82H in JA and ARAA in KO. Their composition is relatively close to each other's.
- ✓ The manufacturing of RAFM steels products has already gained a **relevant industrial experience** (e.g., bars and plates up to 50mm-thick, few tens of tons), in particular in mastering the chemical composition (i.e., alloying elements and impurities control), the metallurgical state after heat treatment and the target tensile and impact properties.
- ✓ Typically, these products are currently delivered with an EN 10204 type 3.1 (EU), or even type 3.2 (CN) inspection **certificates**, which are required in EU for the manufacturing of (Nuclear) Pressure Equipment. It is a requirement of nuclear codes to get return on experience from several industrial batches.



➊ TBMs structural materials (2/2)

Toward the codification of RAFM steels design and construction

- ✓ The TBMs are classified as **(Nuclear) Pressure Equipment** and **Quality Class-1** in ITER. Therefore, their design and manufacturing shall fulfil the Essential Safety Requirements (ESR) of the European Pressure Equipment Directive and, when relevant, of the French Ministerial Order for Nuclear Pressure Equipment. The ITER operator has prescribed the use of RCC-MR(x) for the components providing confinement barrier of the plasma chamber, i.e., the rear back-plate of the TBM-Set.
- ✓ Procuring ITER Members have adopted **the RCC-MRx** as the reference code for the design and construction of the whole TBM-Set. The way to implement the ESR (without presumption of conformity with the regulation) is defined in the code. However, none of the RAFM steels is today part of the RCC-MRx.
- ✓ Most IMs decided not to integrate for the time being their RAFM steel in the RCC-MRx. On the other end, the EU has engaged since 2012 in integrating the EUROFER steel in the RCC-MRx. Recently CN has also started the integration but with a different approach making use of Chinese HJB standards. In all cases, **the overall effort is an extensive programme of mechanical characterization**, the definition and structure of design limits, the production of regulatory justification documents. It is also a way to capitalize the effort for future fusion application.



2 Tritium management

Tritium balance and modelling applicable to DEMO

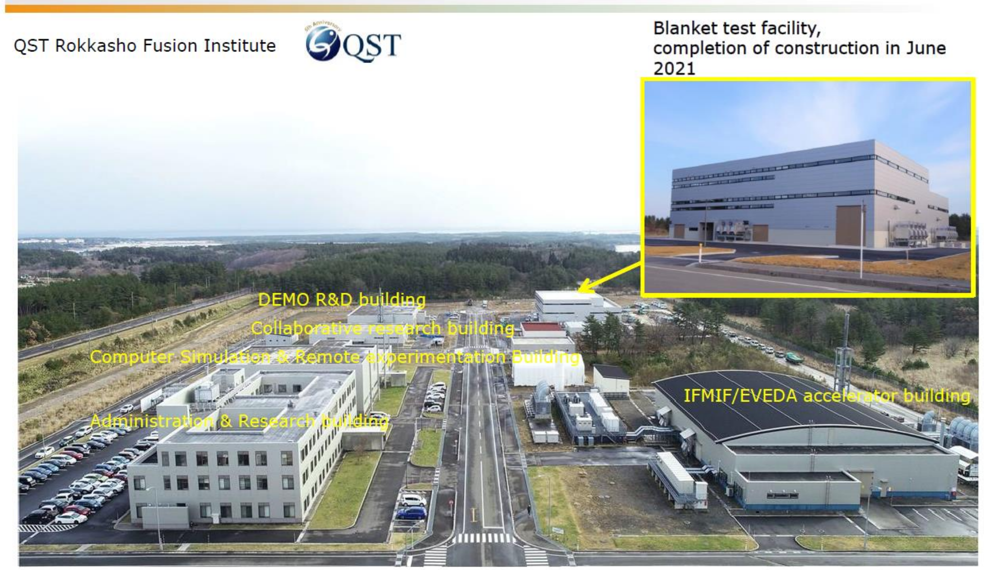
- Tritium managament is considered as one of the major issues for D-T fusion reactors since a long time.
- For this reason a significant modelling efforts is made by the IMs since already a few years. Examples are the development of EcoSimpro in EU, of modelling tool THETA-FR in KO, of simulation tool TriSim in China, of similar simulation tool in Japan, as well as COMSOL Multiphysics and TMAP (used by several IMs).
- The EU EcoSimpro code is an advanced simulation tools able to predict the tritium migration as a function of tritium partial pressure and temperature in the various location of the TBS sub-systems with the implementation of all the knows phenomena governing these aspects, such as surface phenomena, material diffusion, impact of material defects.

Tritium permation barriers (TPB)

- Based on present modeling capability, it appears that the TBSs He-coolant system (because of the high outlet temperature of 500°C) might have a too high Tritium permeation in the Tokamak Complex and might require permeation barriers having a **Permeation Reduction Factor (PRF) above 10**.
- No TPB are implemented within the TBMs, therefore, it has been decided **to implement TPB** (alumina, pack-cementation type) **in the internal surface of He-pipes** (planned to be manufactured in 2025/26) by using an industrial manufacturing process. This implementation will allow to assess the compliance of the TPB with the Pressure Equipement requirements. It could be needed also for He-cooled DEMO TBBs.

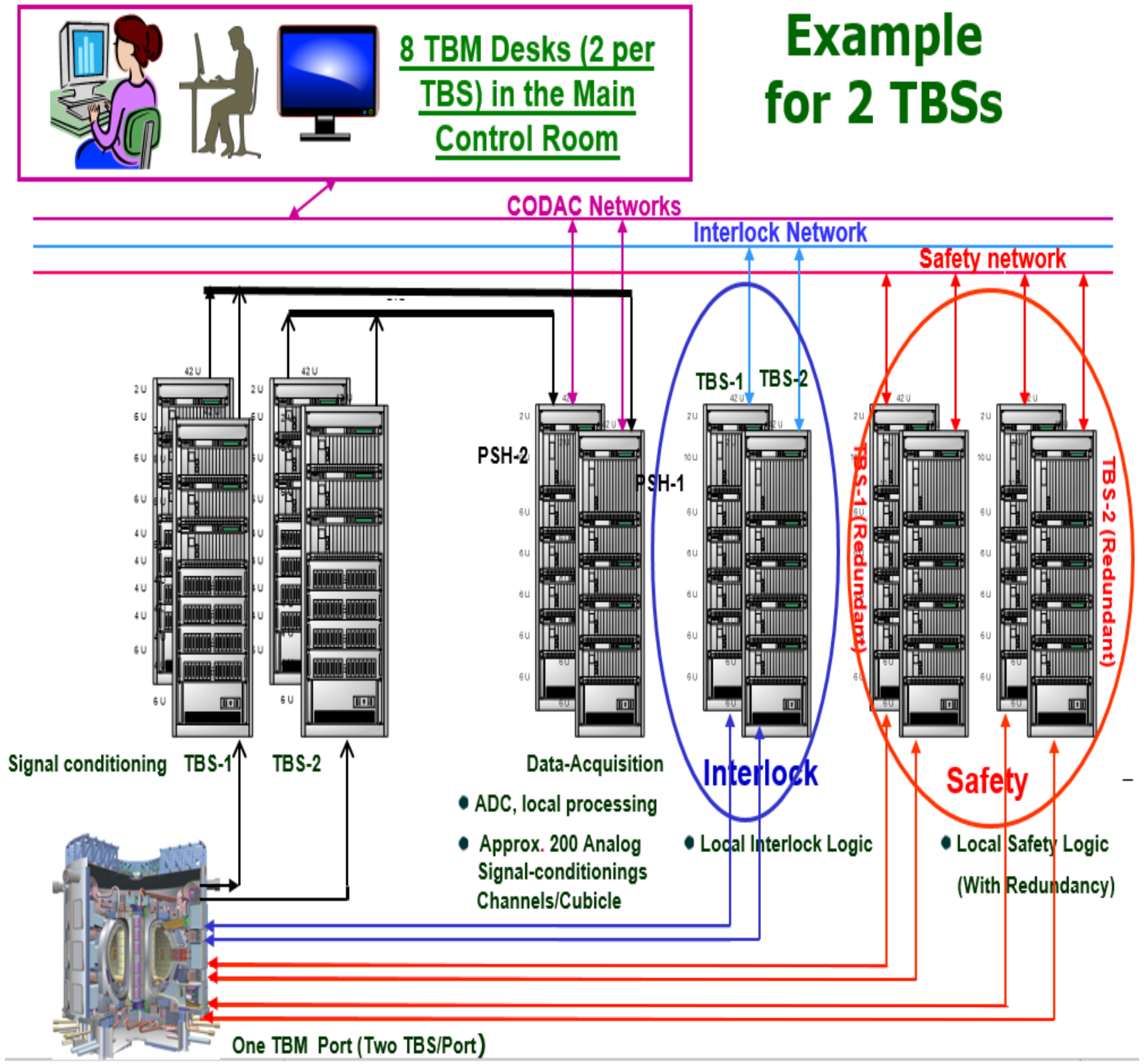
❸ TBS Safety Aspects

- A large amount of Safety analyses are on-going for each TBS and will have to be completed for achieving the approval of the TBS Final Designs.
- The list of possible accidents/incidents has to be exhaustive and the categorization of the various accidents has to be agreed by the IO as Nuclear Operator. All the Safety Functions have to be properly defined and the associated safety-related instrumentation (e.g., detectors and actuators) and components (e.g., safety valves) need to be well defined and qualified in the appropriate environmental conditions (such as radiation field, seismic loads, electromagnetics field, pressure and temperature fields). This list of accidents and the implementation of safety functions would give relevant guidelines for the safety engineering of DEMO TBB systems.
- Safety analyses validation experiments have been performed or being planned. Good examples of safety-related experimental work are the safety-related experiments planned in the previously–mentioned Blanket Test Facility constructed in Rokkasho such as the corrosion tests, the coolant leakage propagation tests and the beryllium-water interaction tests. The results of these validations experiment would be fully applicable to the TBB systems of DEMO.



4 Measurements and control systems

- A significant work is on-going within the ITER Members on the development of **appropriate detectors, instrumentation and measurement systems** able to measure the various TBSs operating parameters which are required to operate and to control the TBSs from the main Control Room.
- These detectors and instrumentations needs **to measure and transmit data** to be used by the Safety Systems (Central+Plant) to operate the defined safety functions, the Interlock (Investement Protection) Systems (Central+Plant) and the regular CODAC system. These systems need to be indipendent and the Safety system needs to be redundant (=2 systems).
- The instrumentation and cabling technologies developed and finally selected for the TBSs will be **directly applicable** to the DEMO TBB.
- The **whole network architecture** is specifically developped for the TBSs and can be used as the basis for the DEMO TBBs measurement systems.



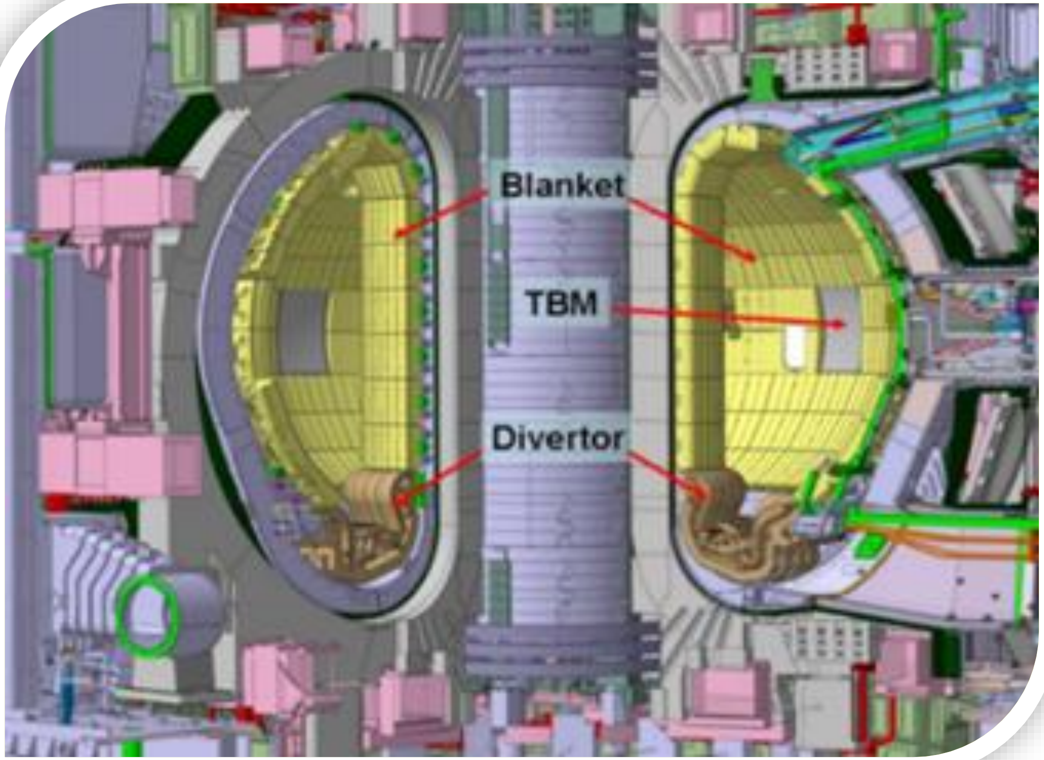
❺ TBS components reliability/availability

- A significant anayses work is on-going to address the **TBS components RAMI** (Reliability/ Availability/ Maintainability/ Inspectability). Only a limited number of relevant input data are available (i.e., failure rates), therefore large unceratinties are present.
- It is known that for the DEMO TBB system components, **RAMI is a critical issue**. It is also a critical issue for the TBS components.

$$\text{Availability} = \frac{\text{MTBF}}{\text{MTBF} + \text{MTTR}}$$

MTBF – Mean time between failures
MTTR – Mean time to repair

- **TBSs components failures** could need machine shutdown. Depending on the locations, the replacement of the failed components could need long time. It is the case for the components within the Vacuum Vessel but also in other areas with high radiation fields and high contamination, such as in the TBM Port Cells.
- Therefore, already for the TBSs, it is necessary **to improve the MTBF of several components** since for the OFF-THE-SHELF versions it could be too low. The on-going RAMI analyses will indicate the type and number of components for which a significant R&D have to be launched for achieving reasonnable values.
- Since many TBS components/technologies are expected to be used also for the DEMO TBB, these TBS developments **can directly profit to DEMO**.



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Final Considerations and Conclusions

- The TBM Program has the objective to install and simultaneously operate in ITER four Test Blanket Systems (=mock-ups of the corresponding DEMO TBB), starting from the beginning of the ITER DT-1 phase (i.e., the TBS Initial Configuration).
- *All the four TBSs have passed the Conceptual Design Phase, while the part concerning the TBS Connection Pipes (the captive part) is already in the final design and partially in the manufacturing phase.*
- The design of the various TBMs is progressively improving taking into account fabrication process and R&D results. Many R&D activities are on-going within the ITER Members laboratories, in particular on manufacturing technologies and on experimental loops.
- *The main objectives of the TBM Program have been discussed placed in the context of the recent proposed modification of the ITER Research Plan that implies a reduced neutron fluence during the DT-1 phase when the four TBSs of the Initial Configuration are planned to be operated.*
- The most important testing objectives of the TBM Program are achievable in DT-1 phase if several operational days of back-to-back pulses, fully dedicated to the TBM Program, are performed.
- *It was shown that, even during the TBS design and manufacturing phases, several results and experimental data concerning the TBSs are of significant benefit for the development of the breeding blankets systems of the next generation of the fusion power reactors (such as DEMO).*

Thank you!

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