The ITER Blanket System

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With contribution from Blanket IPT members

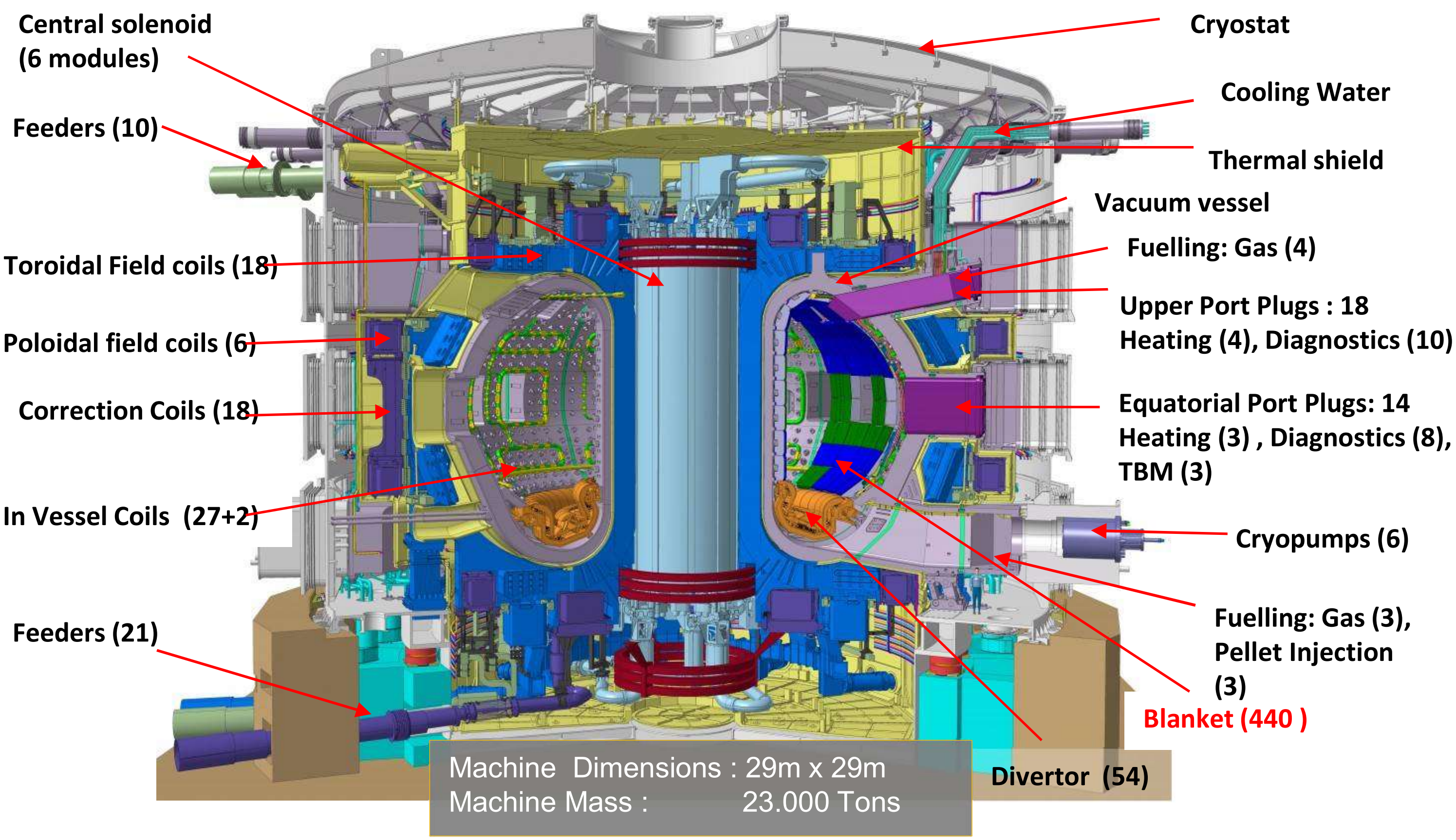
FuseNet PhD Event 2018

Amphitheatre, IO HQ, St Paul lez Durance

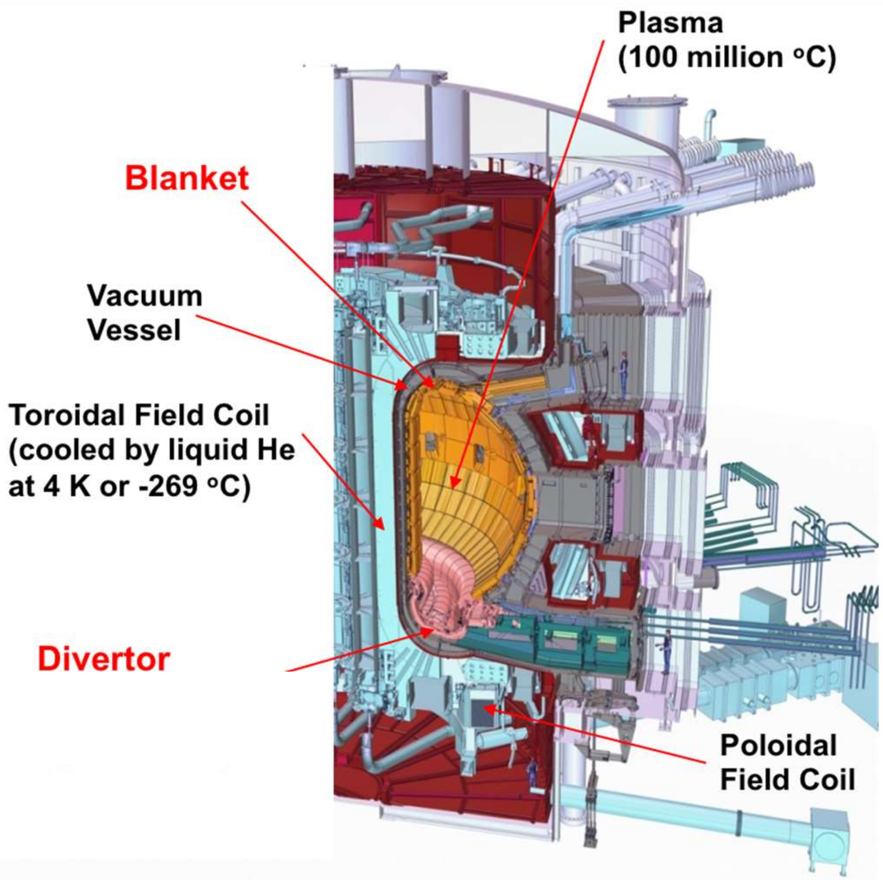
9 November, 2018

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

The ITER Tokamak



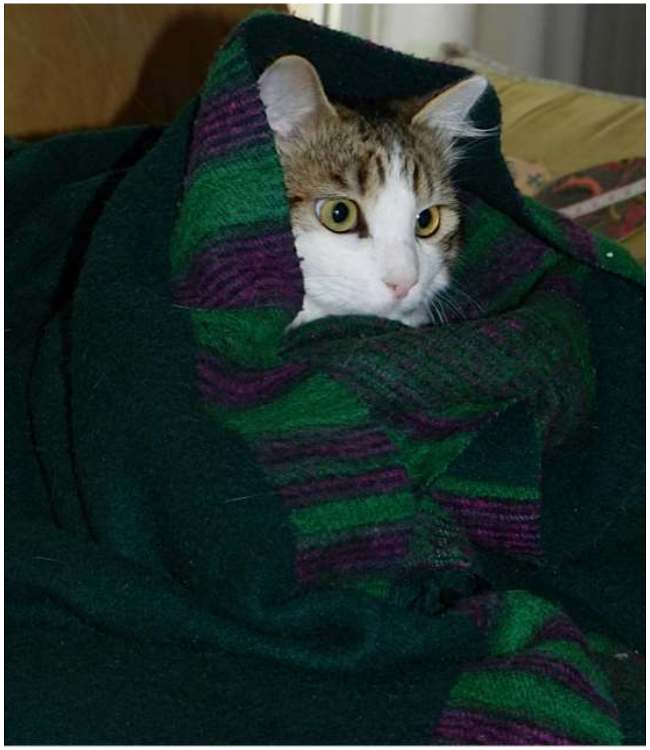
The Blanket is an Internal Component Directly Facing the Plasma



Outline

- What are the functions of a fusion Blanket
- Example of a fusion power plant Blanket
- Evolution of ITER Blanket design
- Current ITER Blanket design
- Challenges
- Off-normal events
- Accommodation of heat loads and EM loads
- First Wall shaping
- First Wall design
- Qualification – FW Panels and Shield Blocks
- Manifolds
- First Plasma Protection Components
- Summary → looking at the future

What Image Comes to Mind When You Think of a Blanket?



- Cover
- Protection
- Warmth
- Cozy feeling

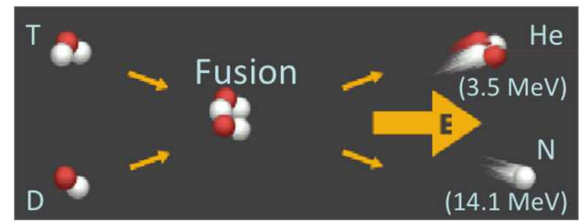
Also applicable to a Fusion Blanket …..except perhaps for the cozy feeling!

The Blanket System surrounds the plasma and blankets the outer components.

Main Functions of Blanket System

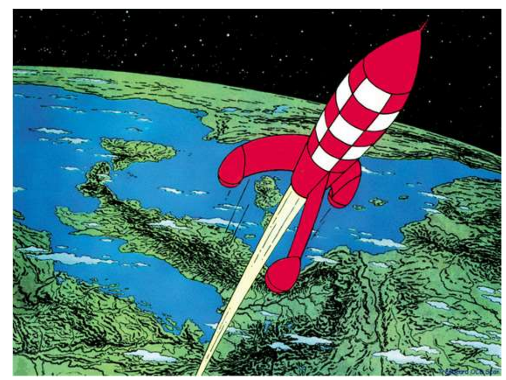
1. Major contributor to providing shielding to reduce heat and neutron loads in the vacuum vessel and ex-vessel components

- 80% of fusion energy carried by high energy neutrons (14.1 MeV).
- Neutrons cannot be controlled by magnetic or electric field.
- How to deal with these pesky neutrons which seem to have a mind of their own? → Blanket them!
- Energy from neutrons transferred to structure around the plasma, mostly to the Blanket by slowing down and stopping the neutrons before they reach the VV and Superconducting Coils.
- Unfortunately, in most cases, in so doing the material becomes activated, requiring the management of activated waste disposal at the end of the component life.



Fusion Reaction

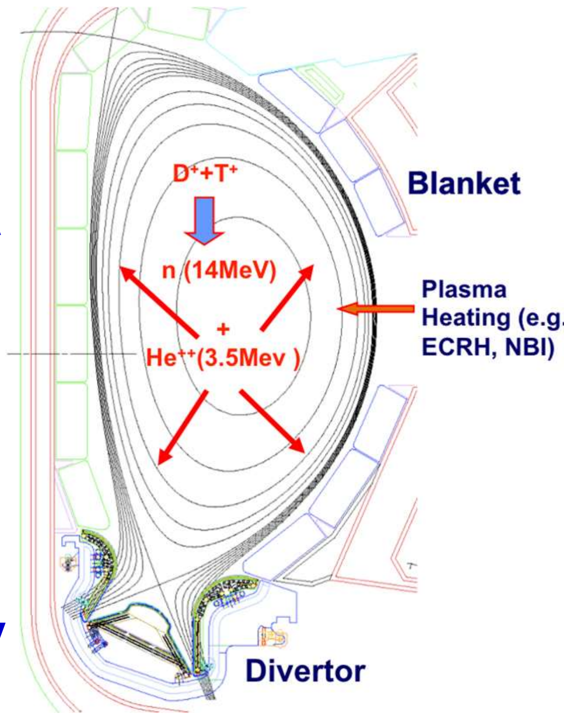
Without collisions, a 14.1 MeV neutron would reach the moon in 8 seconds.



Main Functions of Blanket System

2. Major contributor to absorbing radiative and particle heat fluxes from the plasma

- **20% of fusion energy carried by by alpha particle.**
- **Energy from alpha particle (and from plasma heating) deposited as radiative heat and particle fluxes on plasma facing components (Blanket First Wall and Divertor).**
- **The split between the energy deposited on the First Wall and that deposited on the Divertor depends on the plasma scenario and phase.**
- **The Blanket First Wall has to be designed to accommodate these plasma surface heat fluxes, typically of the order of 1 MW/m^2 for ITER.**



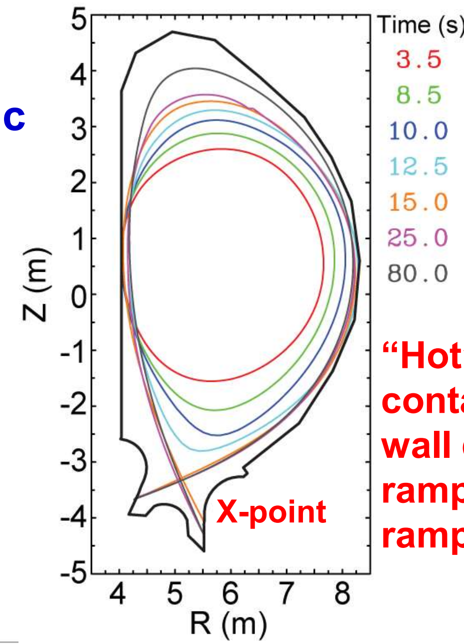
Main Functions of Blanket System

3. Provide a plasma-facing surface which is designed for a low influx of impurities to the plasma

4. Provide limiting surfaces that define the plasma boundary during startup and shutdown.

- To keep the plasma hot, need to minimize impurities and radiation losses.
- Radiation losses typically proportional to square of atomic number, Z^2.
- ITER Blanket First Wall armor chosen as a low-Z, plasma friendly material: beryllium (Z=4).

- Plasma starts in limiter configuration, typically on the inboard First Wall, then grows while the plasma current ramps up until the X-point is formed and the plasma switches then to divertor configuration.
- High resulting heat flux on FW key factor in FW shaping.

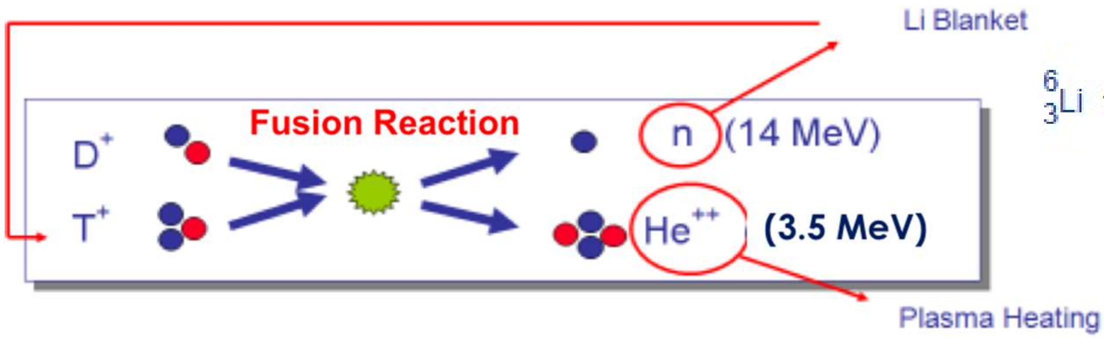


"Hot" plasma contacts the wall during ramp up and ramp down

Main Functions of Blanket System

5. Breed tritium to achieve tritium self-sufficiency

- **Tritium is not readily available and is also expensive to purchase.**
- **Tritium breeding is essential for a fusion power plant.**
- **Introduce lithium in the Blanket, which will react with the neutron formed by the fusion reaction to produce tritium.**



$$^{6}_{3}Li + n \rightarrow ^{4}_{2}He \ (2.05\ MeV) + ^{3}_{1}T \ (2.75\ MeV)$$

$$^{9}_{4}Be + n \rightarrow 2(^{4}_{2}He) + 2n$$

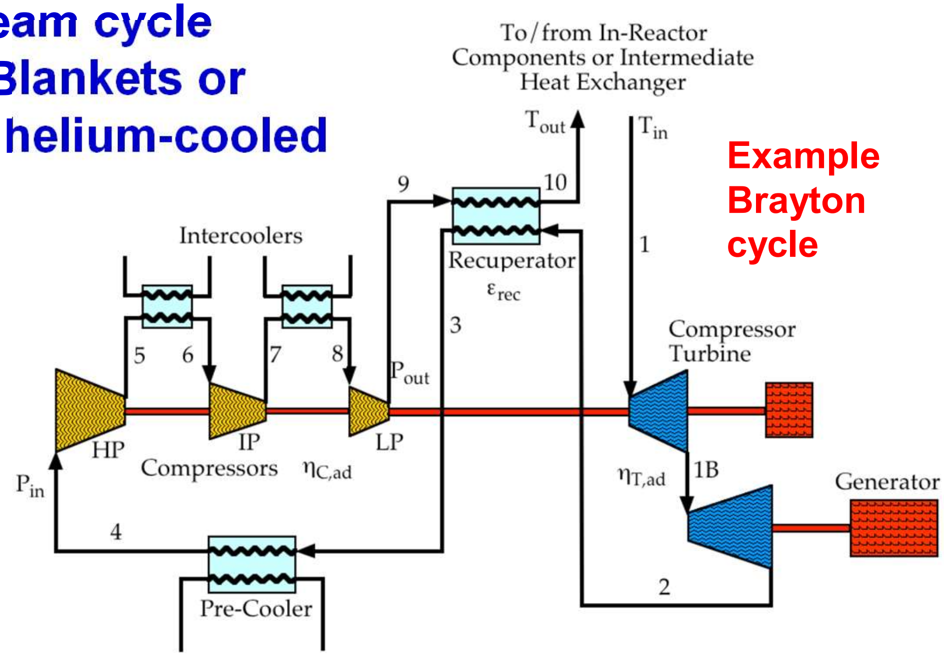
- **Not all neutrons will be captured by the lithium and the Blanket needs to have a neutron multiplier, such as Be or Pb to achieve tritium self-sufficiency.**
- **In so doing the neutron energy deposited in the Blanket is also enhanced – typical energy multiplication factor of 1.2 - 1.4.**

Main Functions of Blanket System
6. Remove energy deposition with high quality for effective power generation

- Higher temperature coolant and materials → Higher power cycle efficiency (Rankine for steam cycle usually associated with water-cooled Blankets or Brayton cycle usually associated with helium-cooled Blankets).

- Higher neutron fluences → Longer lifetime.

- Both of these affect the economics and attractiveness of a fusion power plant.



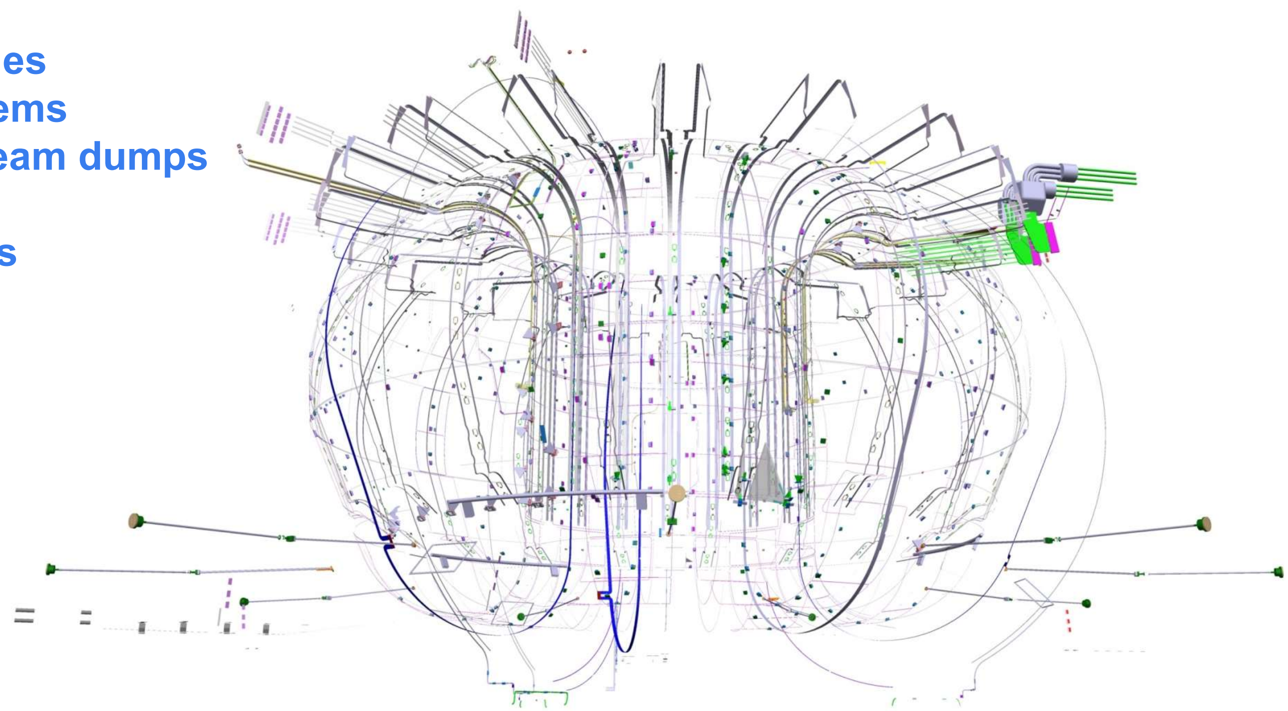
Example Brayton cycle

Main Functions of Blanket System

7. Provide passage for and accommodate interface requirements of the plasma diagnostics

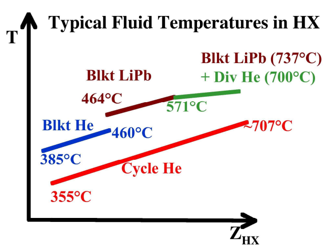
- **Many diagnostic systems interface with the Blanket including in the case of ITER:**
  - Neutron Activation Lines
  - Retro-reflectors
  - Dust and tritium samples
  - In-vessel viewing systems
  - Thomson scattering beam dumps
  - Neutron cameras
  - Micro-fission chambers
  - Wave guides
  - In-vessel looms
  - Bolometers
  - Magnetic loops
  - Rogowski coils
  - Magnetic sensors

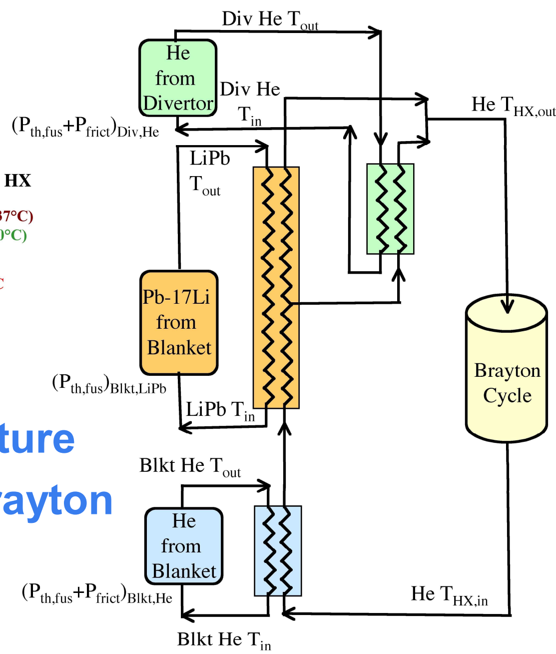
  etc….



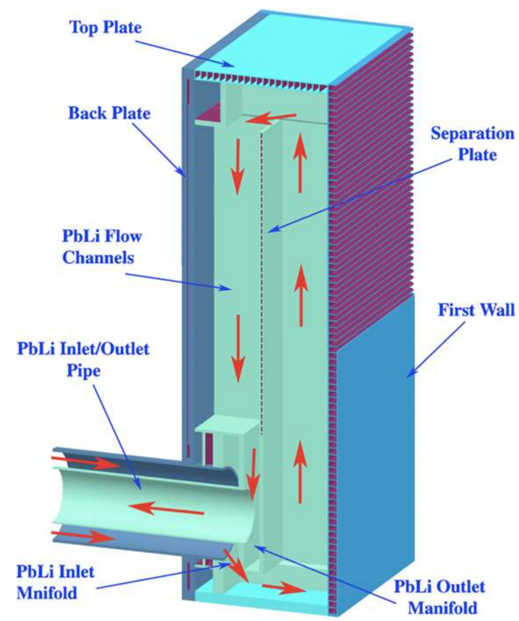
Example of an Attractive Power Plant Blanket Combining all Functions

- **Dual-cooled Pb-17Li/He Blanket Concept**
  - He coolant for more demanding First Wall cooling (no MHD uncertainties).
  - Self-cooled Pb-17Li (Li as breeder and Pb as neutron multiplier) for blanket region with more modest heat loads.





Dual Coolant Concept (ARIES-CS)



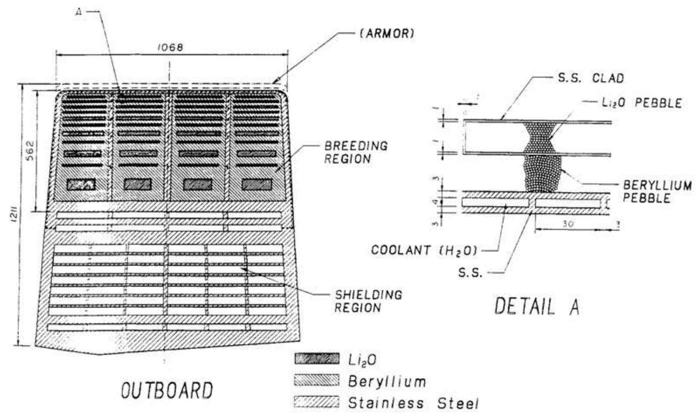
  - Higher temperature for attractive Brayton cycle efficiency (~43%).

- **However, a number of key issues still need to be addressed including MHD effects, electrical insulation integrity and material compatibility.**

ITER Blanket (CDA – EDA)

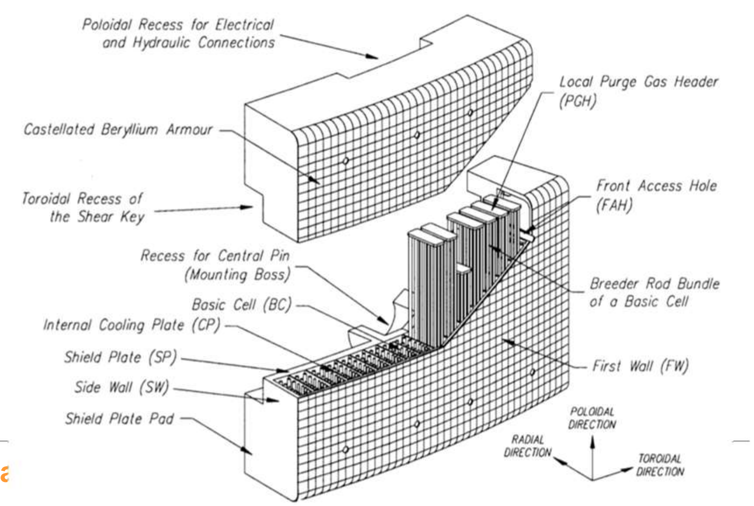
ITER Conceptual Design Activity Phase (late 1980s)

- Breeding blanket with lithium ceramic breeder or Pb-17Li and low temperature water (60-100°C) as coolant.



ITER Engineering Design Activity Phase (1990s)

- Basic Performance Phase (BPP) with non-breeding blanket.
- Enhanced Performance Phase (EPP) with non-breeding blanket replaced by breeding blanket (water-cooled lithium ceramic).



ITER Non-Breeding Blanket (BPP of EDA – late 1990s)

- **Water temperature 140-190°C.**
- **Integrated Shield Block and First Wall then.**
- **Can already see other major features of current ITER Blanket, such as:**
  - **Modular arrangement, Attachment scheme, Electrical straps, Branch pipe connections.**
- **Focus as of then clearly on Shielding Blanket to reduce complexity for the first ever Blanket to operate in a fusion reactor.**
  - **Non-breeding, Low temperature, Austenitic steel, Beryllium armor.**
- **Information about tritium breeding to be obtained from a dedicated TBM program.**

(Presentation by Jaap van der Laan)

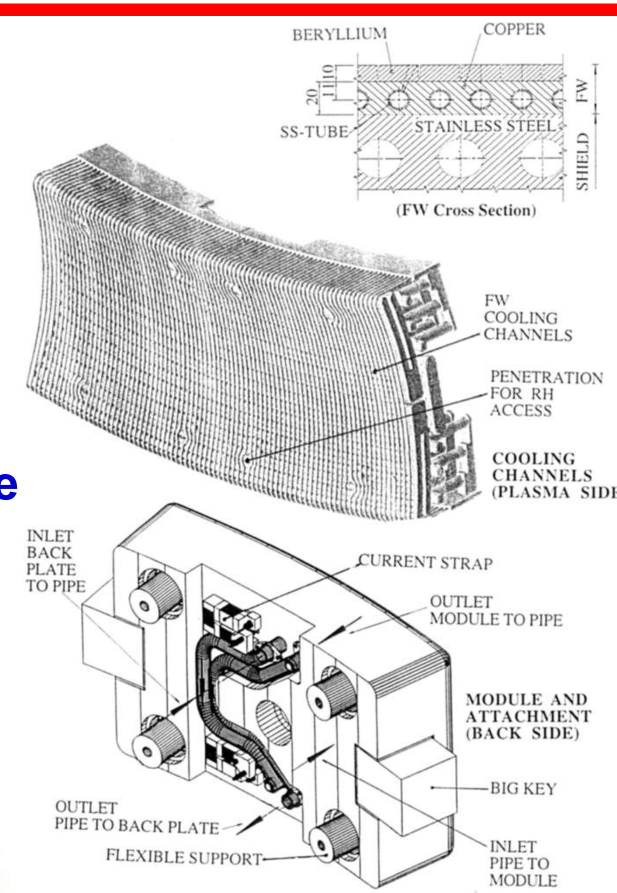
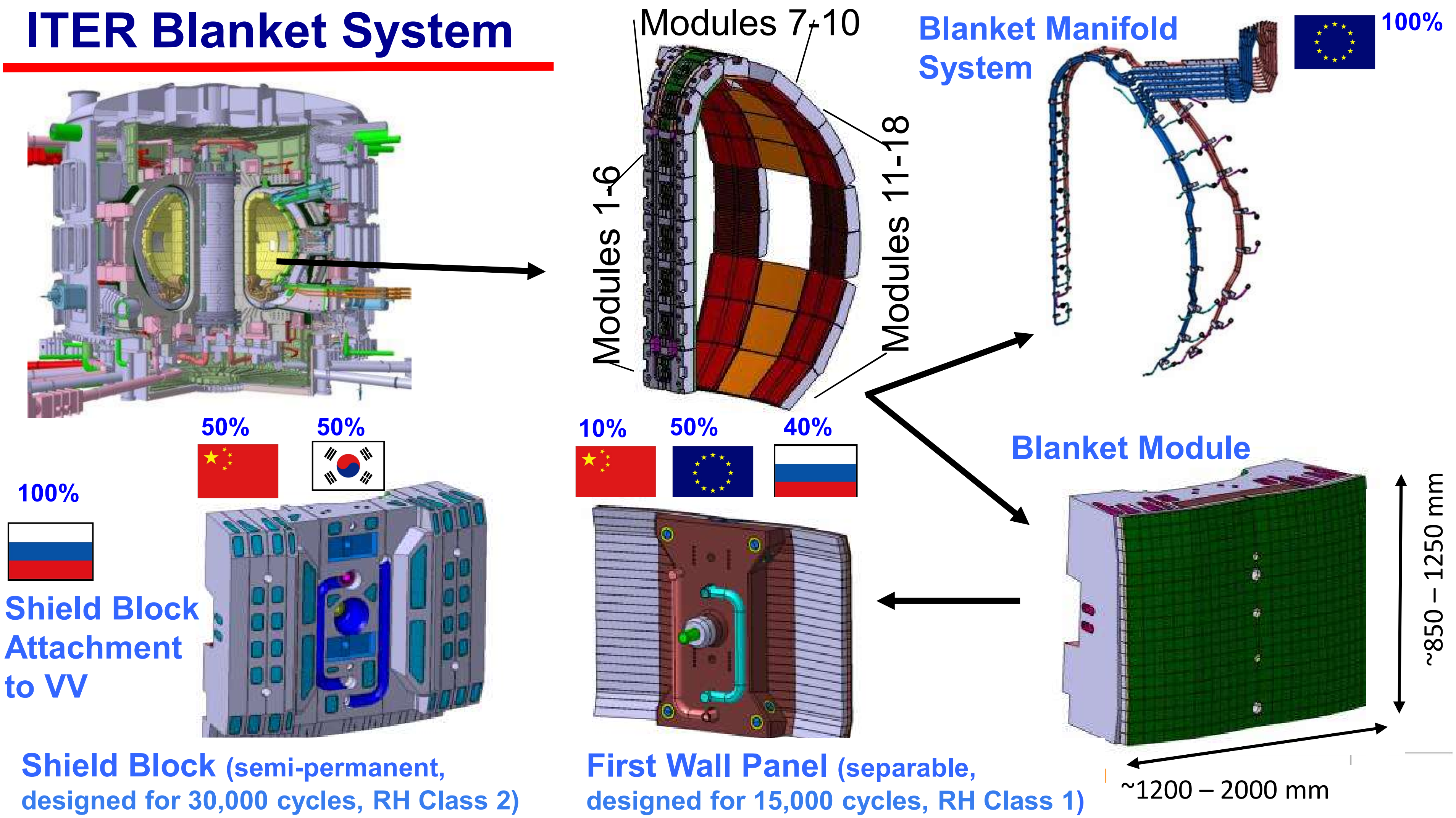


Fig. 2. FW/shield module.

ITER Blanket System



Blanket Design Challenge

- **First of a kind fusion blanket in first of a kind fusion experimental reactor.**

- **One of the most challenging components to be designed for ITER.**
  - **High heat fluxes from the plasma (steady state + off-normal).**
  - **Large EM loads from eddy and halo currents.**
  - **Interfaces with major systems and components.**
  - **Attached mostly against gravity.**
  - **Need to accommodate often conflicting requirements from interfacing systems (e.g. gaps and cut-outs vs neutron shielding).**

- **For historical reasons, interfacing components at substantially different levels of maturity → major interface issue.**
  - **Blanket right in the middle.**
  - **Ghostbuster syndrome…Who ya gonna call?**
  - **Major lesson for future reactor → integrated design with interfacing components should be developed over time at similar design level and with realistic tolerances.**

Design Must Take Into Account Interfaces with Many Other Systems, including:

- Physical Interfaces
  - Vacuum Vessel
  - Manifolds
  - In-Vessel Coils
  - Ports (Diagnostics, NBI, etc…)
  - Plasma Boundary
  - Diagnostics (many)
  - Fueling System
  - Plasma heating systems
- Functional Interfaces
  - Physics Scenarios (start-up/ramp down, flat top, off-normal events)
    - Heat and EM loads
  - Neutron Shielding (Presentation by Mike Loughlin)
    - Coils, Vacuum Vessel
  - Fusion Power Removal
    - Tokamak Cooling Water
    - Manifolds
- and others…

ITER Blanket System in Numbers

• Number of Blanket Modules:	440
• Max allowable mass per module:	4.5 tons
• Total Mass:	1530 tons
• First Wall Coverage:	~610 m^2
• Armor/Heat Sink/Structural material:	Be/CuCrZr/316L(N)-IG
• Max total thermal load:	678 MW (736 MW including ports)
• Cooling water inlet conditions:	4 MPa and 70°C

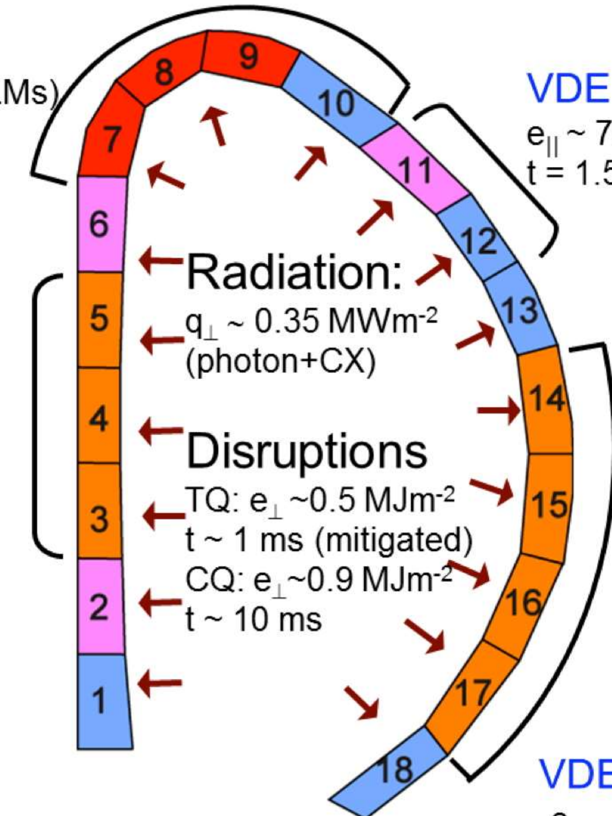
First Wall Subjected to Normal and Off-Normal Heat Fluxes from the Plasma

Steady state:
$q_{||} \sim 8\ MWm^{-2}$, $\lambda_{q||} > 4.0$ cm
$q_{||} \sim 24\ MWm^{-2}$, $\lambda_{q||} > 2.5$ cm (ELMs)

Disruptions:
$e_{||} \sim 45\text{-}120\ MJm^{-2}$, $\lambda_{q||} > 20$ cm
t = 3.0-6.0 ms

Start-up:
$q_{||} \sim 25\ MWm^{-2}$, $\lambda_{q||} \sim 5.0$ cm
Several seconds

Confinement transients:
$q_{||} \sim 250\ MWm^{-2}$
t ~2-3 secs



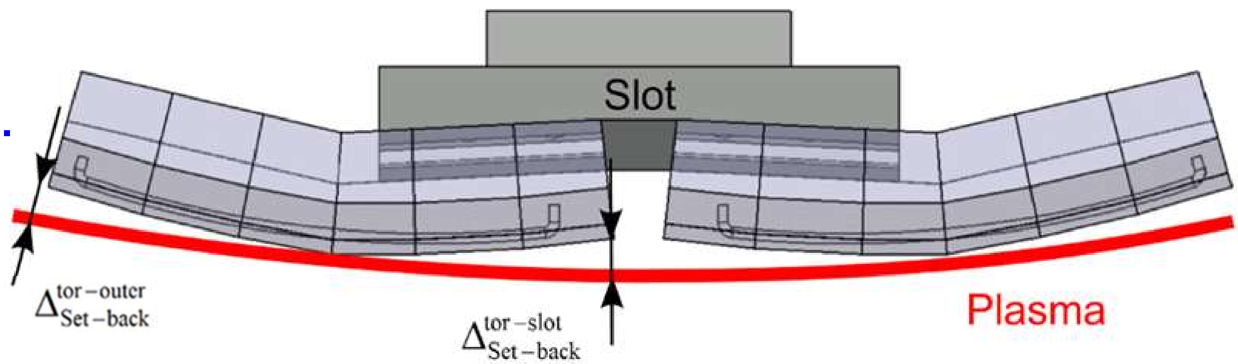
VDE (up):
$e_{||} \sim 70\text{-}270\ MJm^{-2}$, $\lambda_{e||} > 3.0$ cm
t = 1.5-3.0 ms

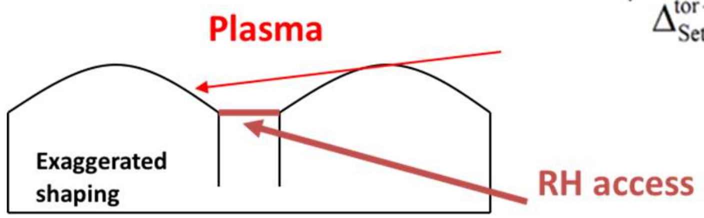
Start-up and rampdown:
$q_{||} \sim 40\ MWm^{-2}$, $\lambda_{q||} > 1.2$ cm
Several seconds

VDE (down):
$e_{||} \sim 90\text{-}300\ MJm^{-2}$, $\lambda_{q||} > 3.0$ cm

Shaping of First Wall Panel

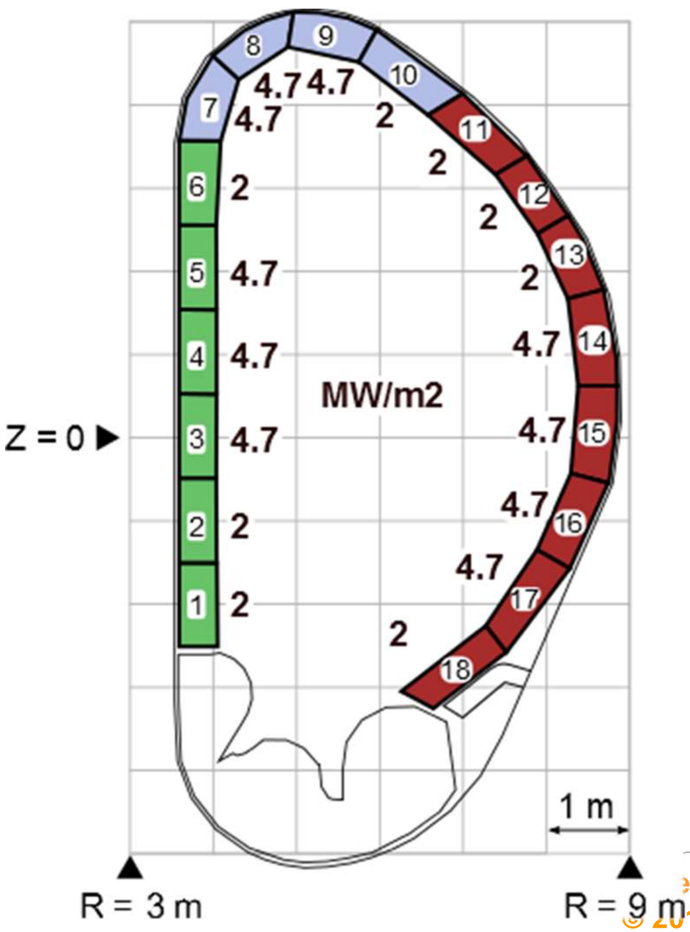
- Heat load associated with charged particles represents the main component of heat flux to the first wall.
- The heat flux is oriented along the field lines.
- Thus, the incident heat flux is strongly design-dependent (incidence angle of the field line on the component surface).
- Shaping of FW to shadow leading edges and penetrations and spread the incident heat flux on a larger area.
- Important to maintain FW alignment within tolerances.





- Allow good access for RH
- Shadow leading edges

Distribution of ITER FW Panels



2 Different First Wall Panel Configurations

- Normal Heat Flux (NHF) panels → 2 MW/m²
- Enhanced Heat flux (EHF) Panels → 4.7 MW/m²

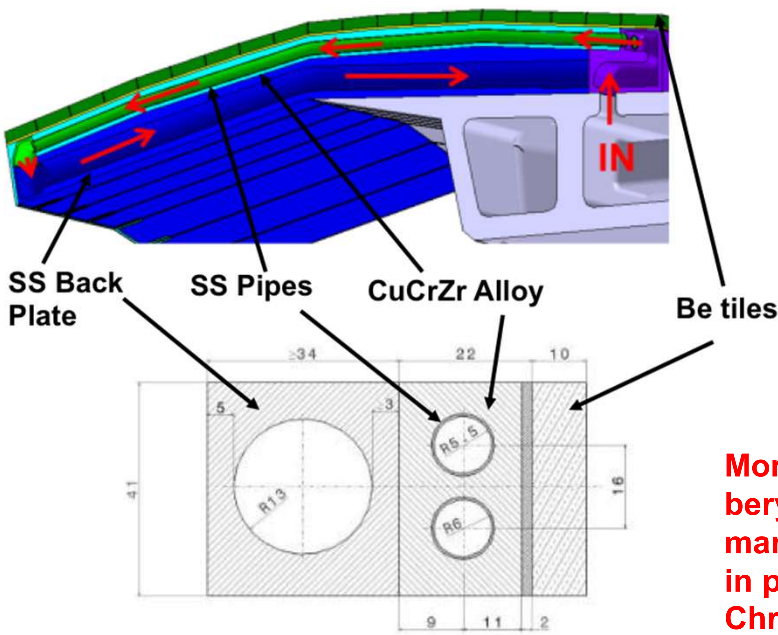
Summer sunny day
0.001 MW/m²

Space shuttle (re-entry)
0.5 MW/m²

First Wall Designs with Beryllium Tiles as Armor

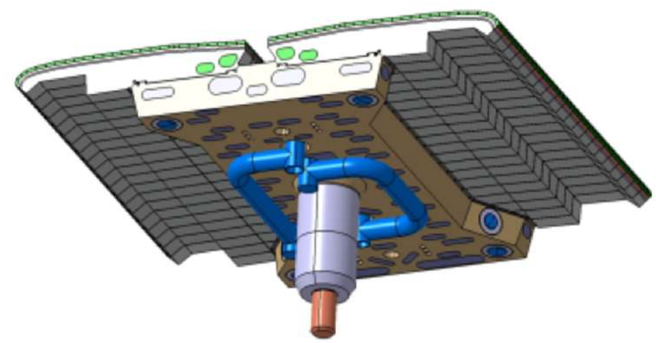
Normal Heat Flux (NHF) FW procured by EUDA:

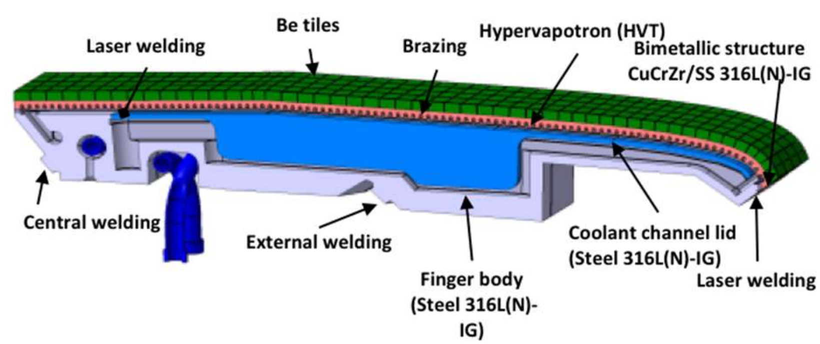
- $q'' = 2$ MW/m^2
- Steel Cooling Pipes
- HIPing cycles (SS/CuCrZr/Be)



Enhanced Heat Flux (EHF) FW procured by RFDA and CNDA:

- $q'' = 4.7$ MW/m^2
- Hypervapotron
- Explosion bonding or HIPing (SS/CuCrZr) + brazing (Be/CuCrZr)

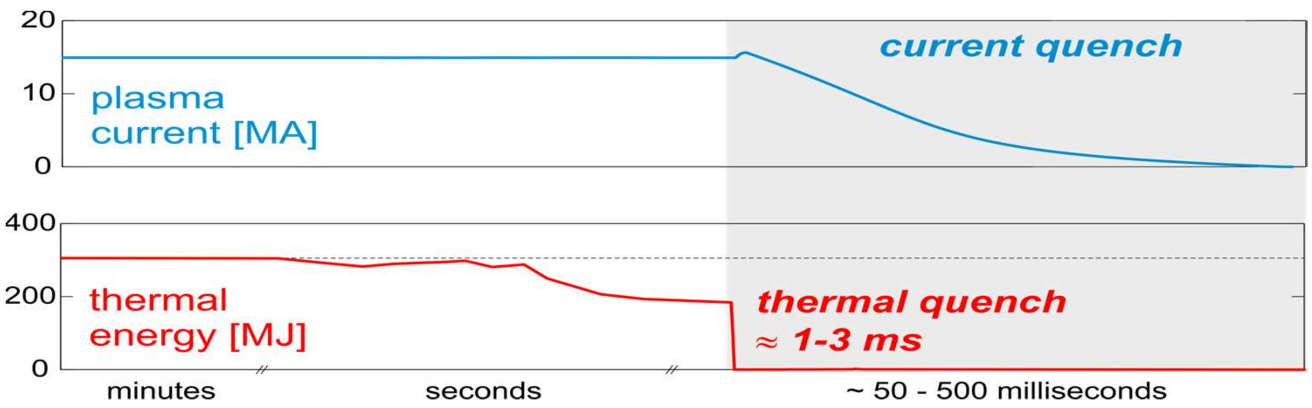




More detail on beryllium management at ITER in presentation by Chris Dorn.

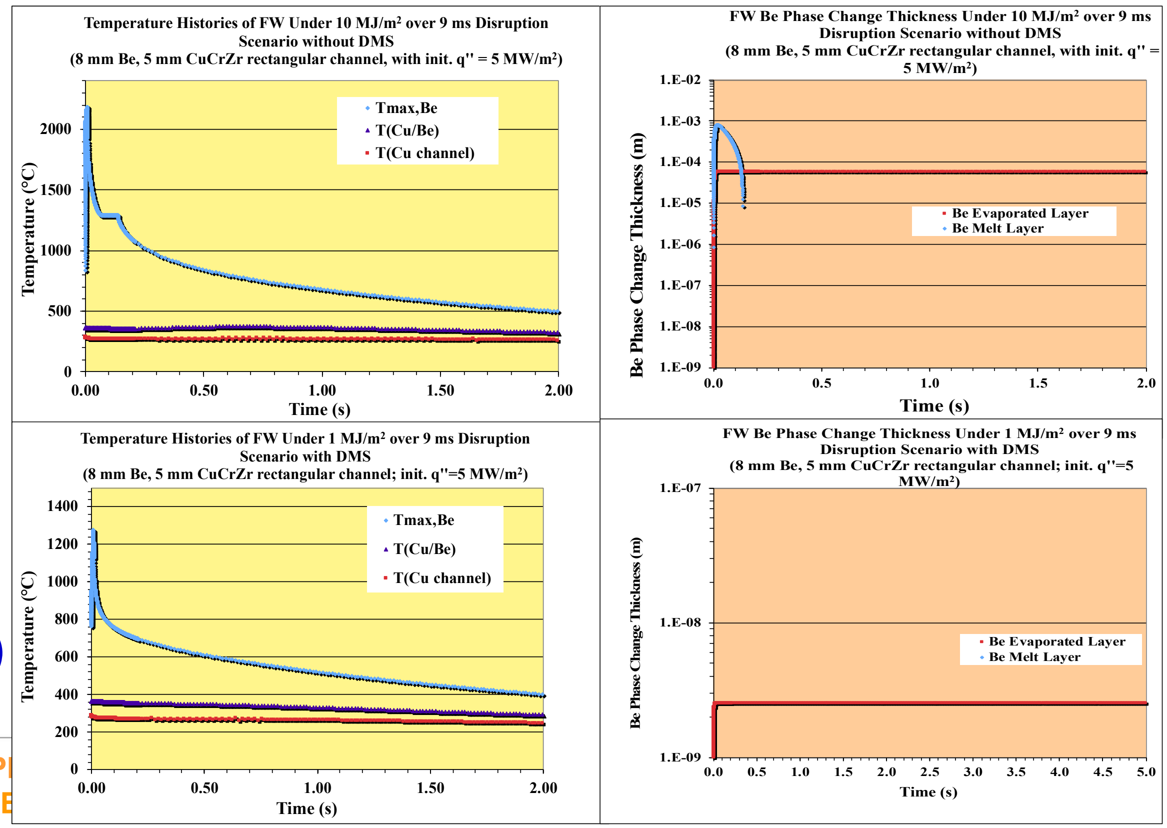
Off-Normal Events – Disruptions (from A. Loarte)

- Global plasma instability terminating tokamak plasma.
- Triggered by plant malfunction or other plasma instabilities.
- Plasma energy & current decrease very fast (loss of vertical position control) → huge power fluxes to PFCs + Forces ($j_{induced}$ x B) →V.V. & in-vessel comp.



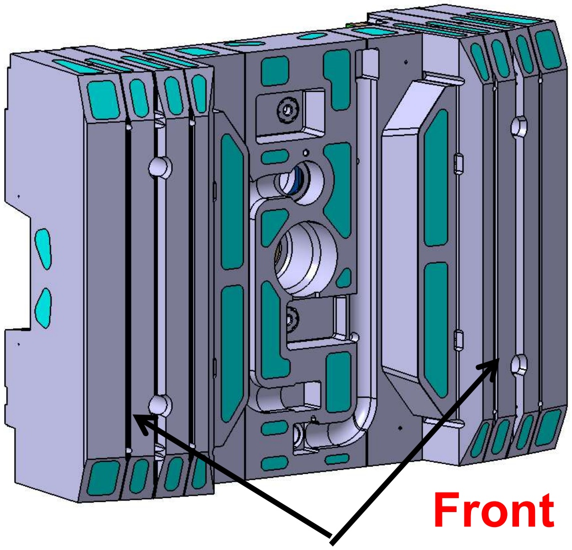
Off-Normal Energy Deposition

- **High energy deposition over a short time on the Be armor following off-normal events such as disruptions and VDEs.**
  - e.g. 10 MJ/m² over 9 ms.
  - Be melt layer + evaporation ~ 1 mm
  - Only a few such events could be accommodated.
  - Essential to have a Disruption Mitigation system (DMS) for an acceptable FW lifetime.

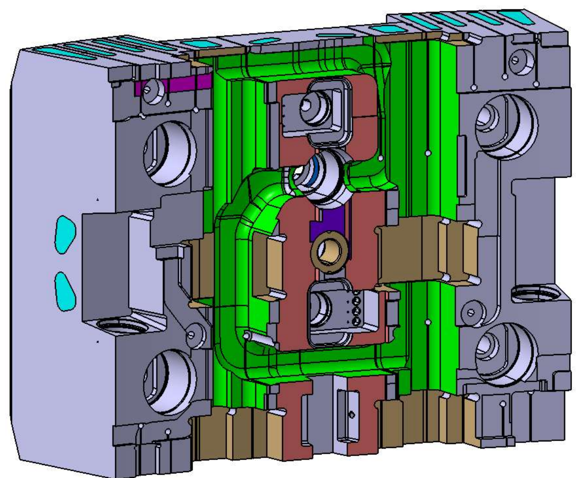


- **Strategy: set Be armor thickness based on steady state q'' with $T_{max,Be}$ <600°C (<800°C in some local zones)**
  - 8-10 mm

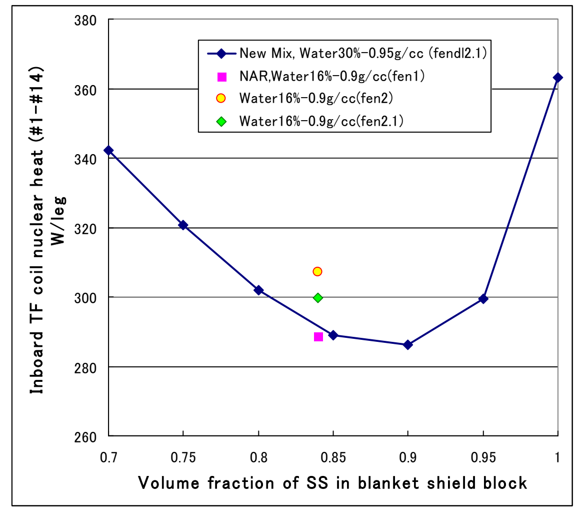
ITER Blanket Shield Block Design (e.g. SB04)



Front

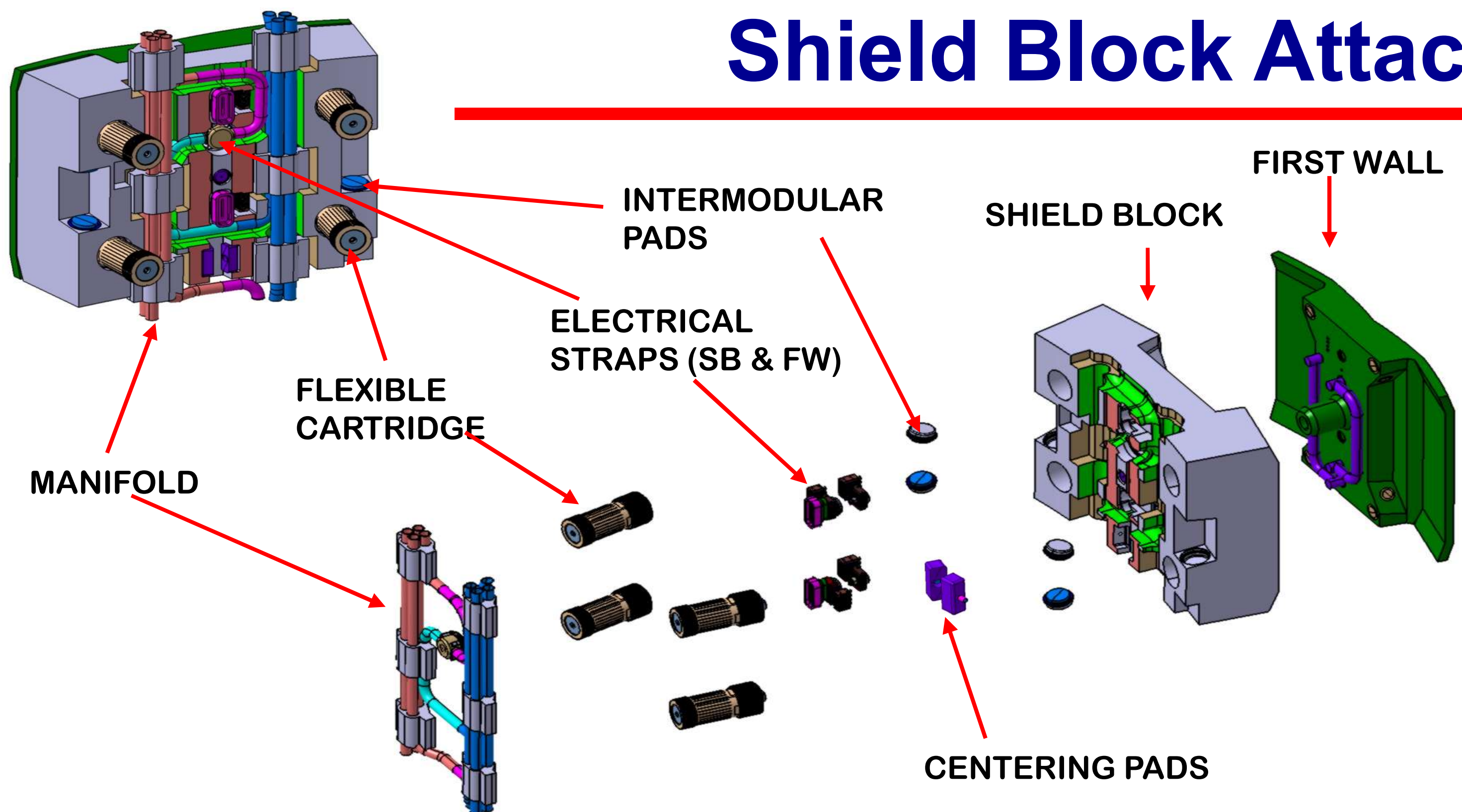


Back



- **Slits to reduce EM loads and minimize thermal expansion and bowing.**
- **Cooling holes are designed to provide required thermal-hydraulic performance while aiming for a Water/Stainless Steel ratio resulting in good nuclear shielding performance.**
- **Cut-outs at the back to accommodate many interfaces (including Manifold, Attachment, In-Vessel Coils).**

Shield Block Attachment

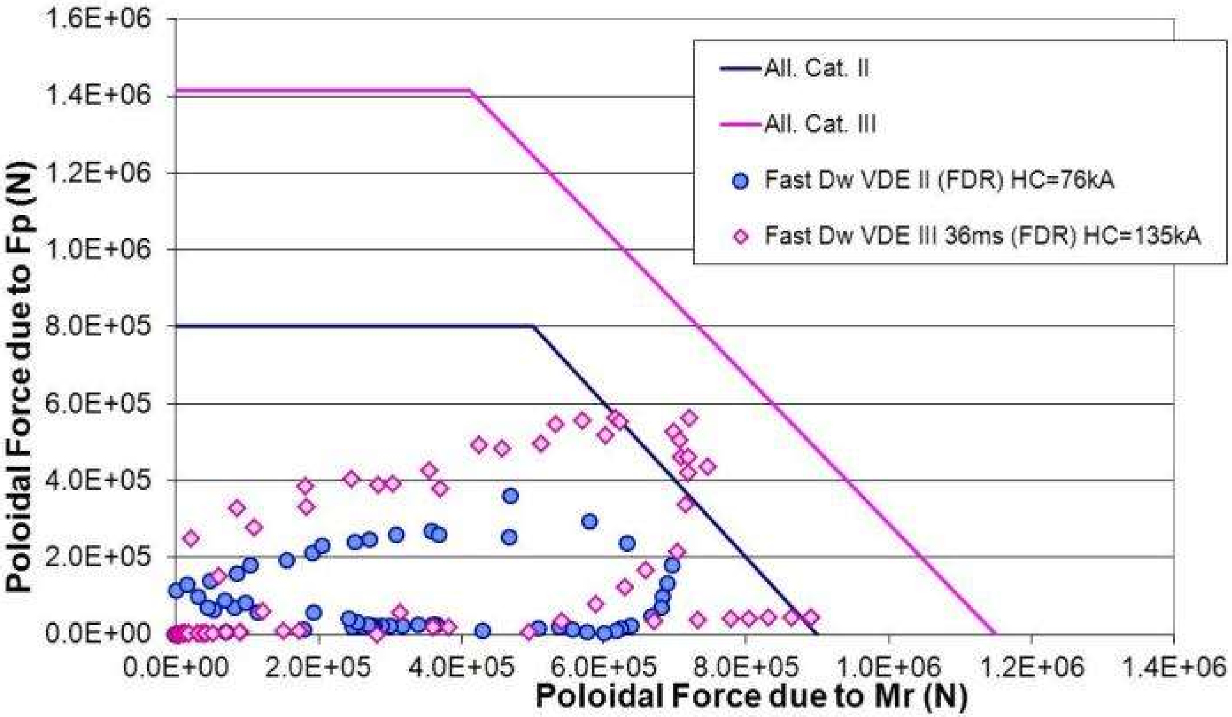


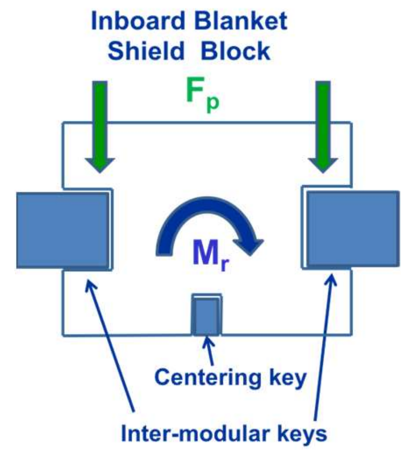
e.g. BM 4 Interfaces

- 4 flexible axial supports
- Keys to take moments and forces
- Electrical straps to conduct current to vacuum vessel

Shield Block and Attachment Designed to Respect Pre-Defined Load from Vacuum Vessel Load Specifications

- **Optimizing blanket design (radial thickness and slitting) to reduce EM loads based on analysis.**



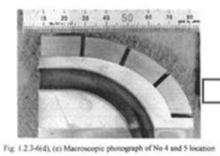


- **For example, results for BM 1 (one of the most loaded modules) show that the loads on the keys are within the VV LS for Cat. II and Cat. III cases.**

Qualification Program
e.g. Multi-Step development Program for NHF FW Panels

FW quality class 1 – includes qualification components

R&D



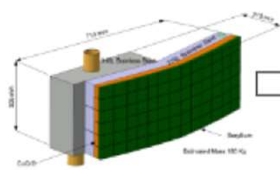
- Define bonding techniques
- Allowable of bonds (heat flux, tile size, acceptable designs)

Small scale mock-up

- Validation of bonding technology
- Small scale DA qualification
- Determination of tile size
- Qualification of NDE and Defect acceptance mock-ups
- Long cycle lifetime

2010 CDR

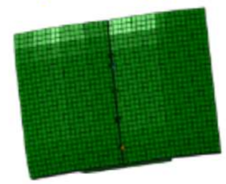
Semi prototype



- Includes all manufacturing features relevant to PFC final design
- Surface shaping
- PFC attachments to structure
- PFC hydraulic circuit + connection
- Statistical effect w/r number of tile

2011 PDR

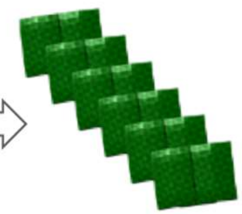
Full Scale prototype



- Tolerances
- Manufacturing documentation
- Improving efficiency of manuf. Route
- Full supporting structure
- 100% PFC test
- Pressure drop
- Handling full size comp.

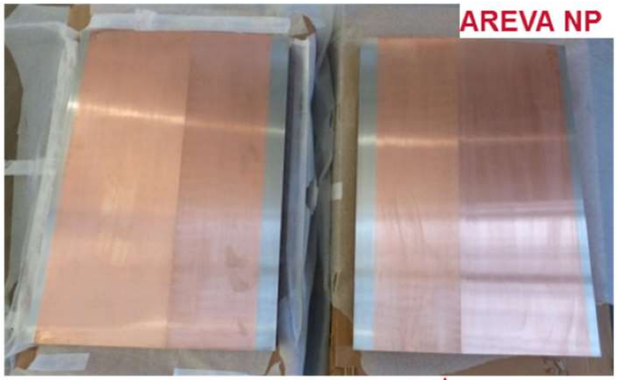
2013-2017 PA

Series



- Validation of manufacturing route
- Full qualification of final supplier
- Realism of schedule/cost

Qualification Program
Example Progress on NHF FW Full Scale Prototype in EUDA



Two wings of SS/CuCrZr FW structure before HIP joining cycle (1040°C for 2 hours)

Assembly of Be tiles before CuCrZr/Be HIP cycle (580°C for 2 hours)

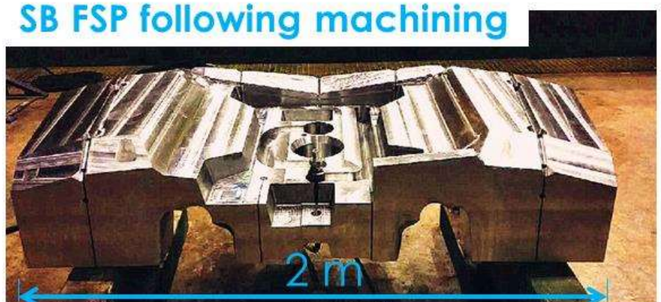
Example Progress on SB PA with CNDA

1.6.P1B.CN.01 FSP

The following processes have been successfully completed by mid-February 2018:

- Final machining
- Hydraulic pressure test
- Cold helium leak test
- Hot helium leak test following outgassing

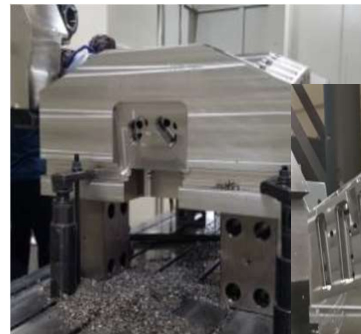
Blanket Shield Block Full Scale Prototype - CNDA



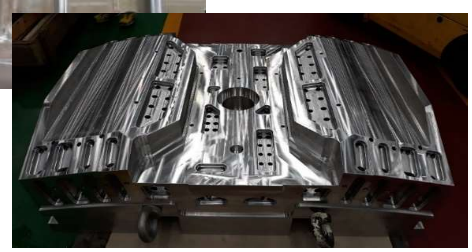
Example Progress on SB PA with KODA – Full Scale Prototype (SB06)

Visual inspection of cooling holes after deep drilling

Cooling header after machining



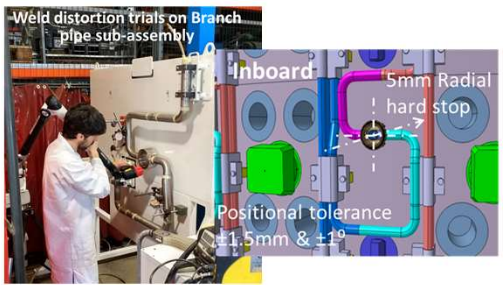
Cleaning before welding



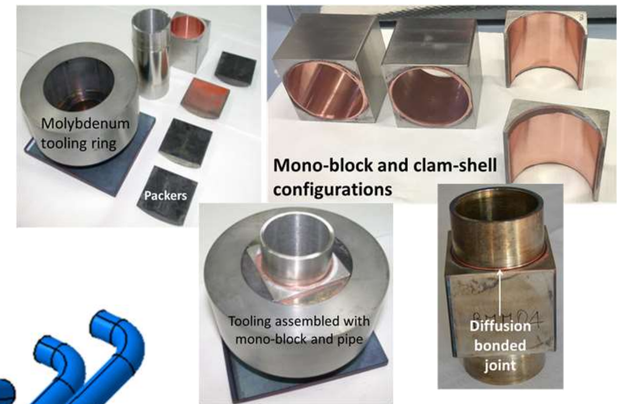
FSP after cleaning

FSP during welding

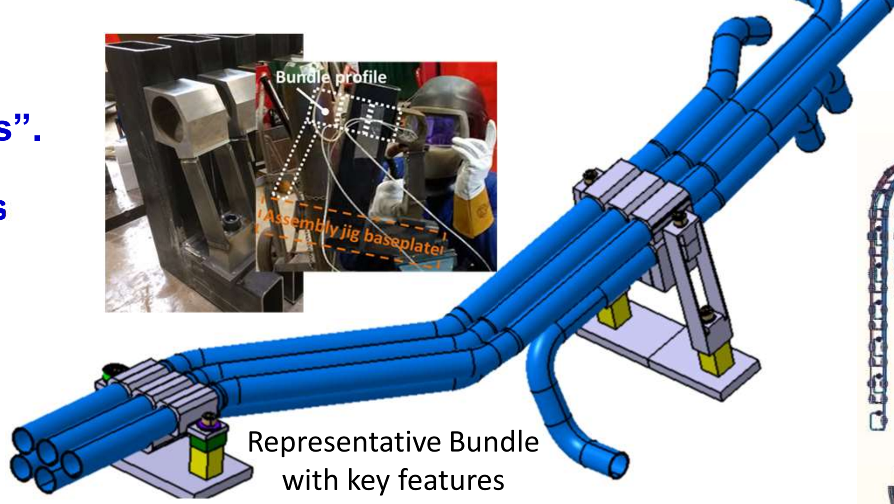
Blanket Manifold System

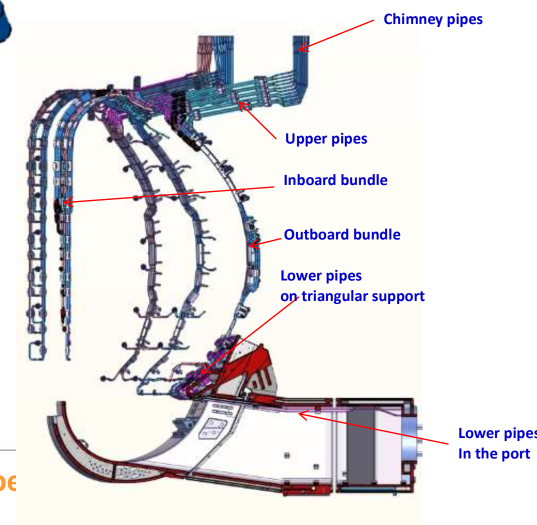


- Design being optimized – choice of diffusion bonded or bolted support attachment.
- Proof of principle and validation activities on-going.
- PA to be signed in 2019 with EUDA.



- Complex design because of thermal interface and EM load constraints.
- Not "just a bunch of pipes".
- e.g. must design supports to be thermally conductive to avoid too high heat load to VV and electrically insulating to maintain reasonable EM load levels.

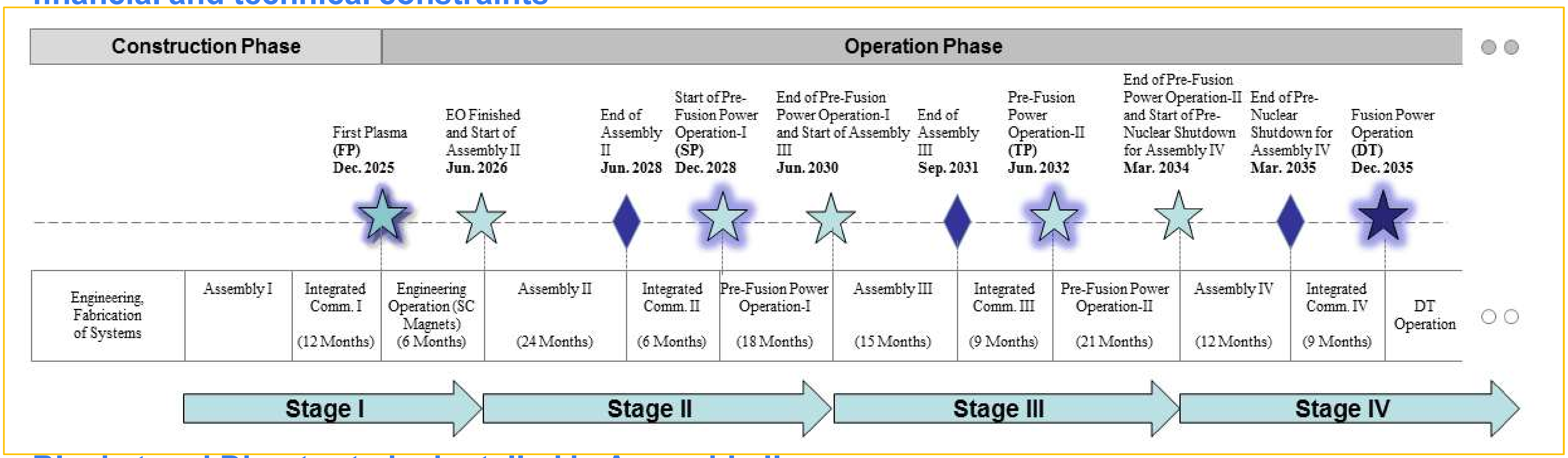




A Staged Approach to DT Plasma

Extensive interactions among IO and DAs to finalize revised baseline schedule proposal

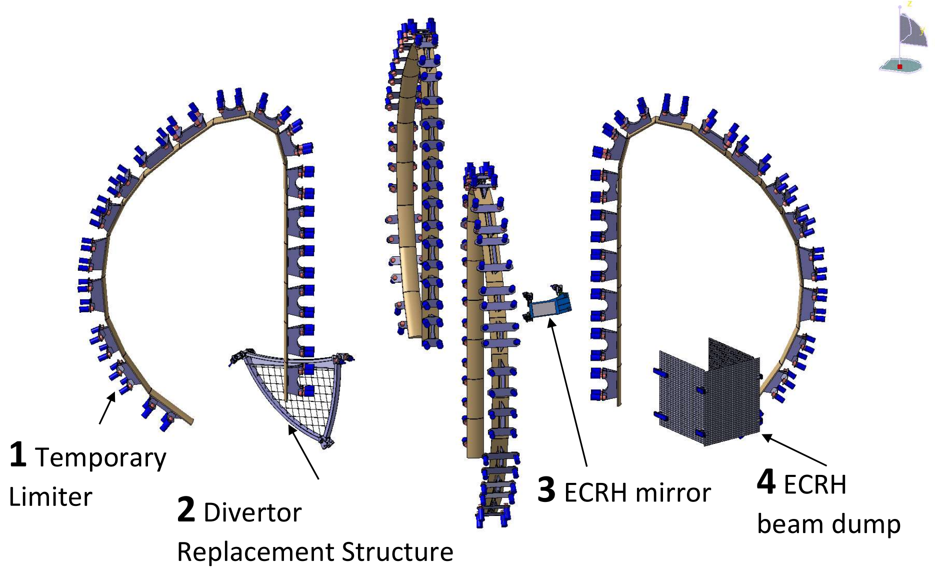
- Schedule and resource estimates through First Plasma (2025) consistent with Members' budget constraints.
- Proposed use of 4-stage approach through Deuterium-Tritium (2035) consistent with Members' financial and technical constraints

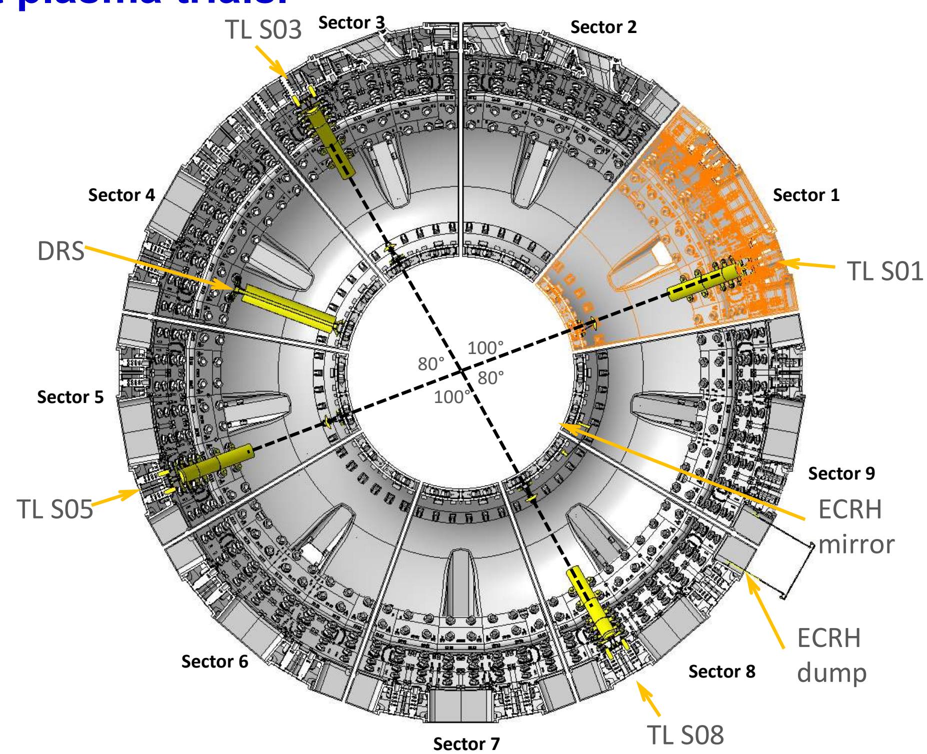


- Blanket and Divertor to be installed in Assembly II.
- First Plasma Protection Components installed in Assembly I in absence of Blanket and Divertor.

First Plasma Protection Components

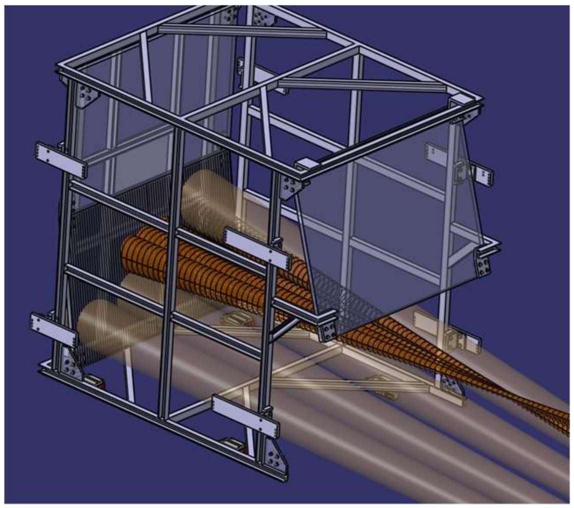
- **The ITER Baseline is based on a phased assembly for the ITER Tokamak. The blanket and divertor are not installed in the vessel for the first plasma. An in-vessel protection system (FPPC) is needed to protect components installed during the initial assembly phase from possible damage during the first plasma trials.**



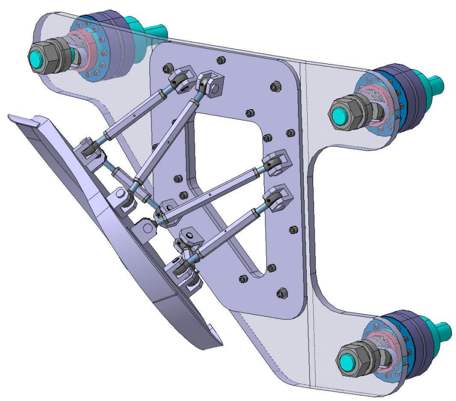


Details of First Plasma Protection Components

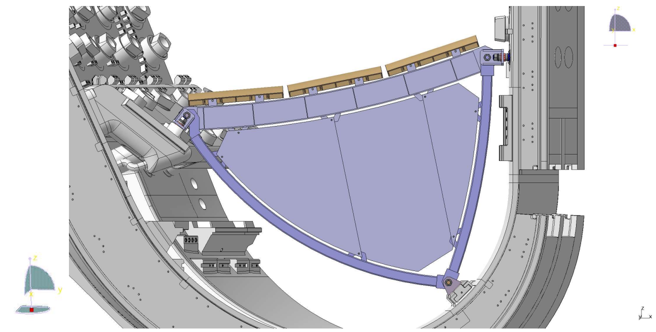
- Final design review is Planned in mid 2019.
- Components have to be manufactured and delivered on site for first assembly (First Plasma planned at the end of 2025)



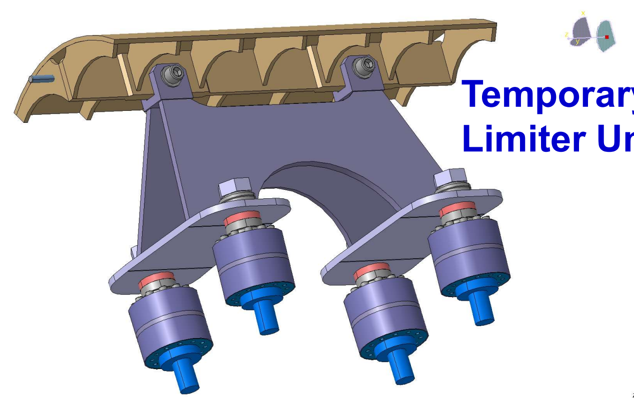
ECRH Beam Dump



ECRH Mirror



Divertor Replacement Structure



Temporary Limiter Unit

Summary

- **First of a kind Blanket to operate in first of a kind fusion reactor.**
- **Blanket with a lot of interfaces - very challenging and constraints.**
- **ITER will provide a range of key information in the development and operation of blankets for commercial reactors.**
  - **Important information from ITER Blanket on integration, interface management, tolerances, assembly, remote maintenance.**
  - **Test Blanket Modules (for DEMO and beyond) will be designed, fabricated, installed and tested to provide T breeding and higher temperature operation information.**
- **A full development program to DEMO would require testing of materials at high fluence (e.g. IFMIF) and possibly testing of blanket system at high temperature and fluence in a dedicated facility (CTF).**

Thanks to all Blanket Integrated Product Team Colleagues (IO and Domestic Agencies) for their Contributions

Participants in Blanket FDR (2013)