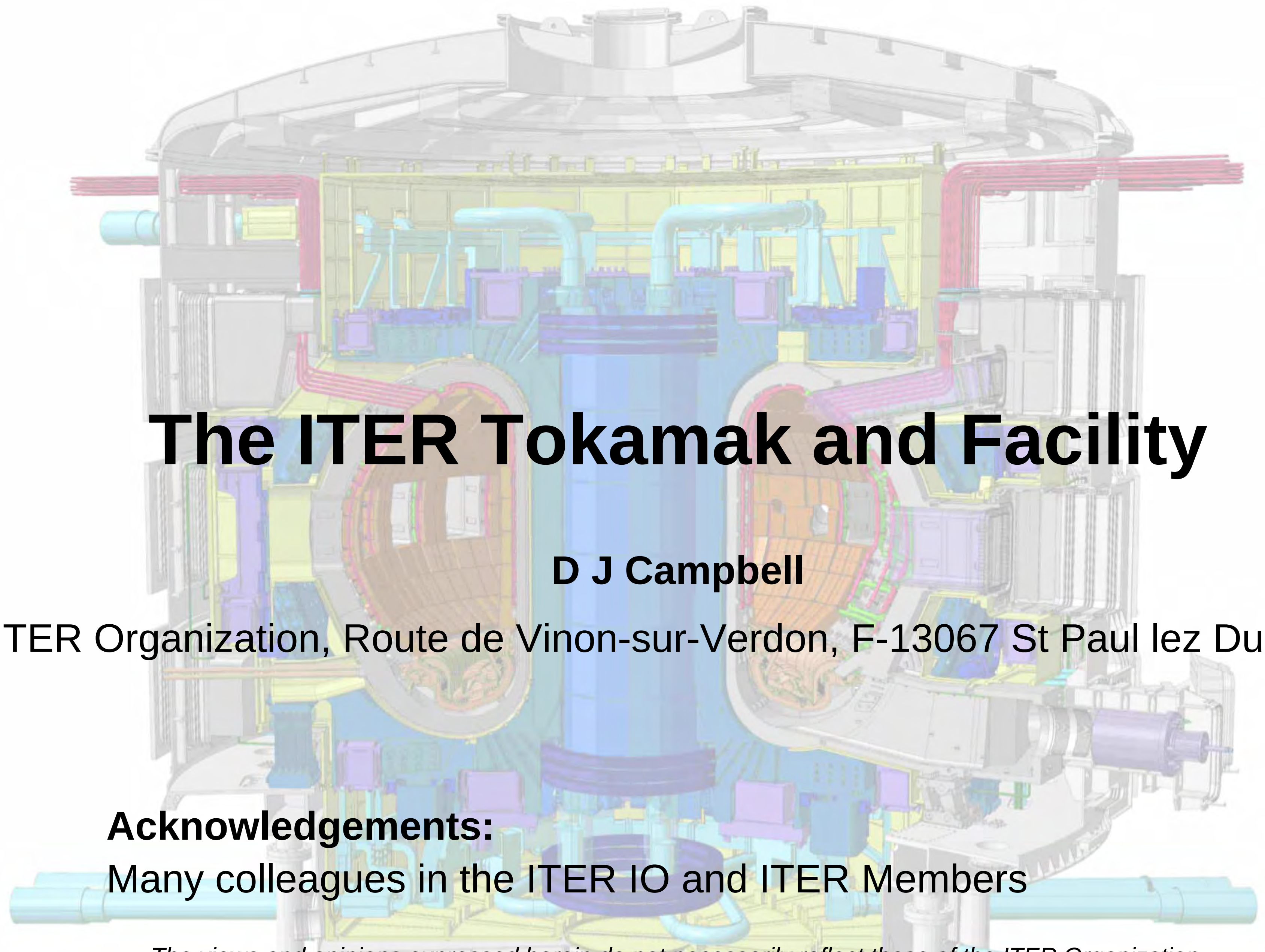


The ITER Tokamak and Facility

D J Campbell

ITER Organization, Route de Vinon-sur-Verdon, F-13067 St Paul lez Durance

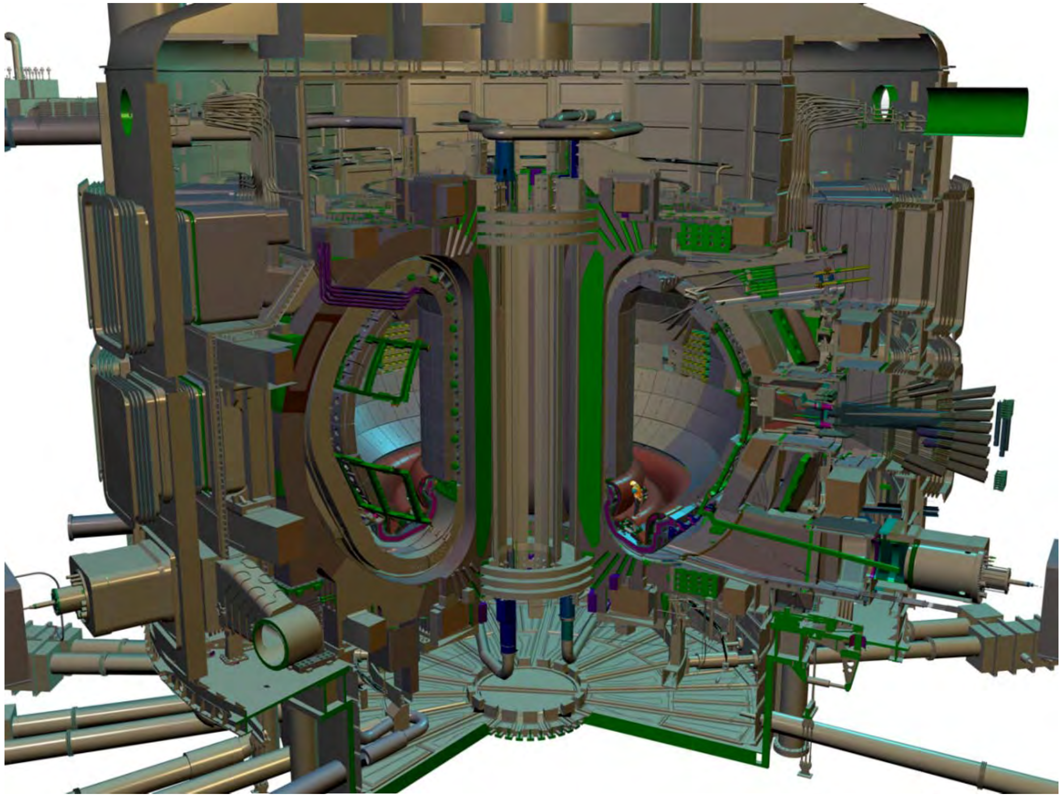
Acknowledgements:

Many colleagues in the ITER IO and ITER Members

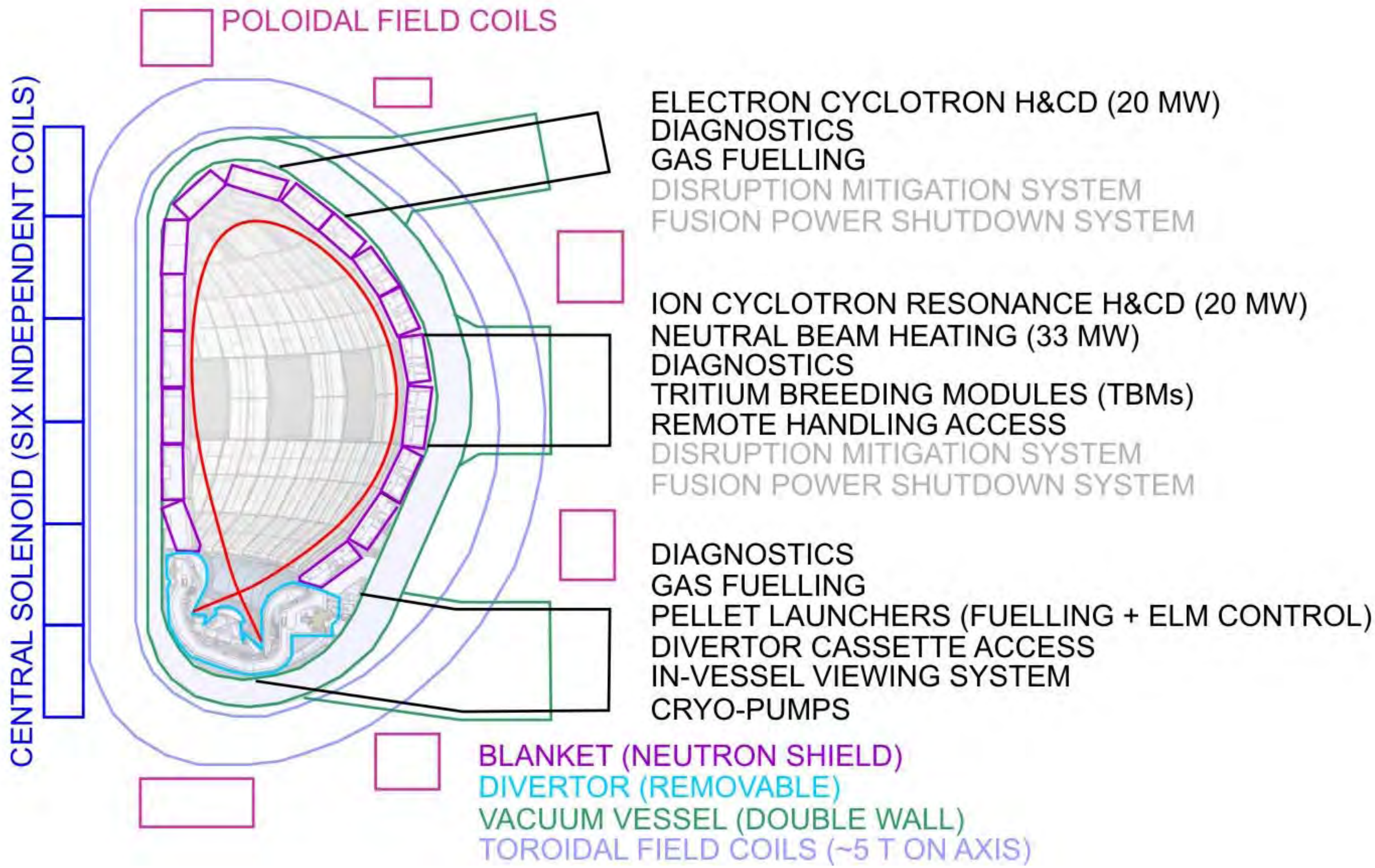
The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Synopsis

- **ITER core components**
- **ITER plant systems**
- **ITER ancillary systems**

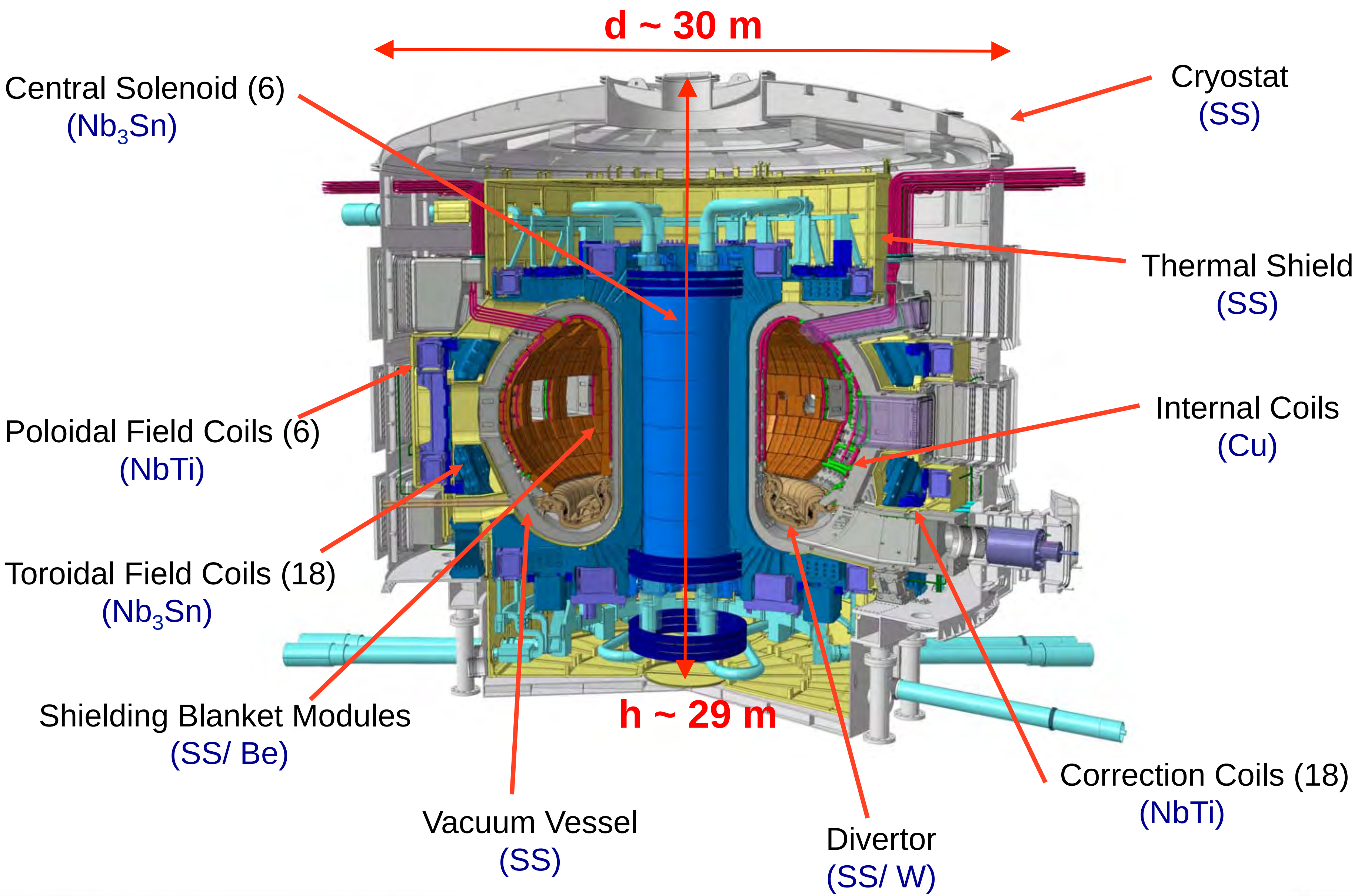


ITER – Overview of Major Systems

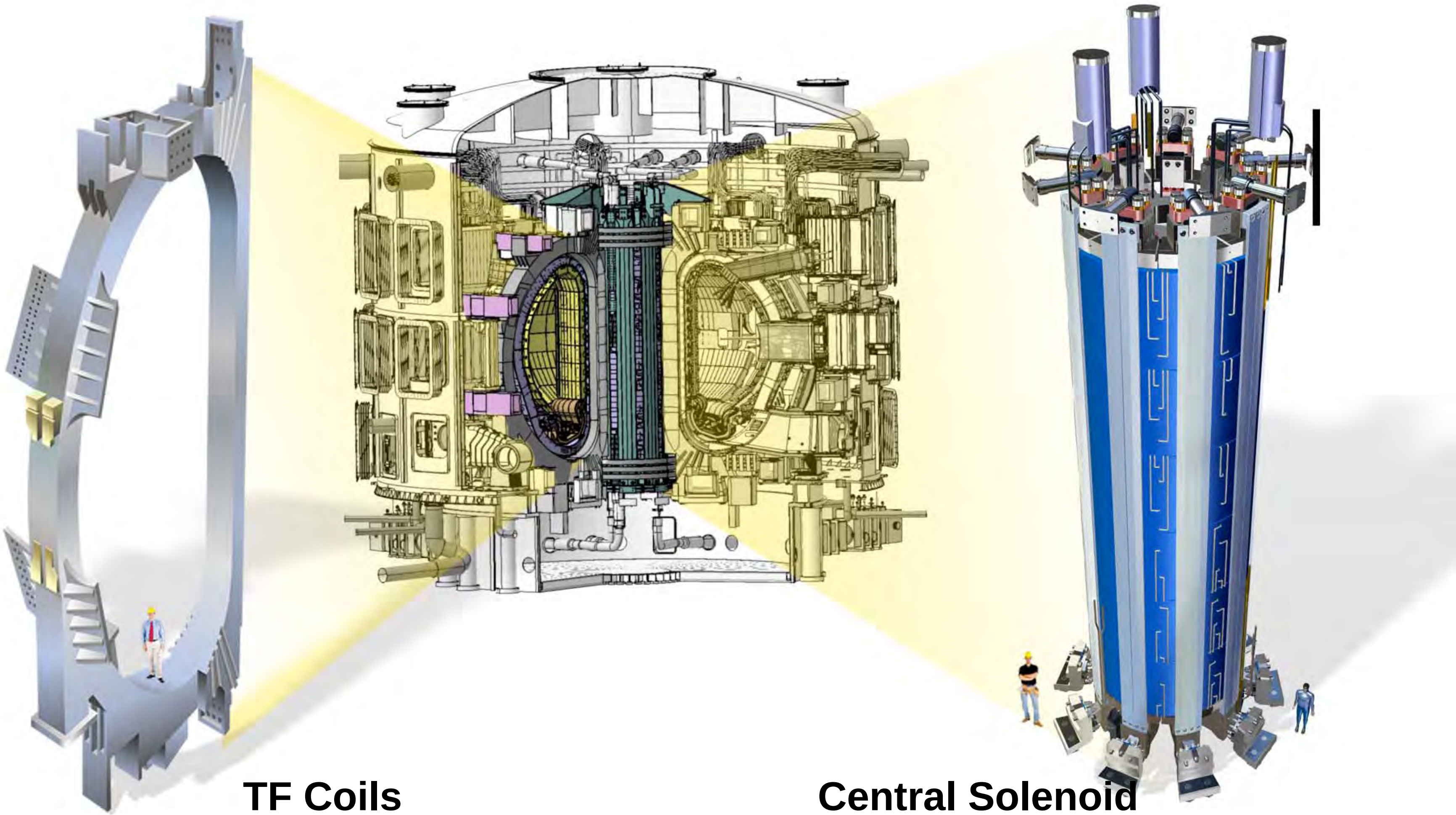


ITER Core Components

ITER – Tokamak Core Components



Magnets - Unprecedented Size and Performance



TF Coils
11.8 Tesla, 41 GJ
400 MN centering force

Central Solenoid
13 Tesla, 7 GJ
20 kV, 1.2 T/s

ITER Magnet Supply: 10 PAs

CS with US
(General Atomics)

Pre-compression rings: EU
(EADS CAS)

PF1: RF
(Efremov)

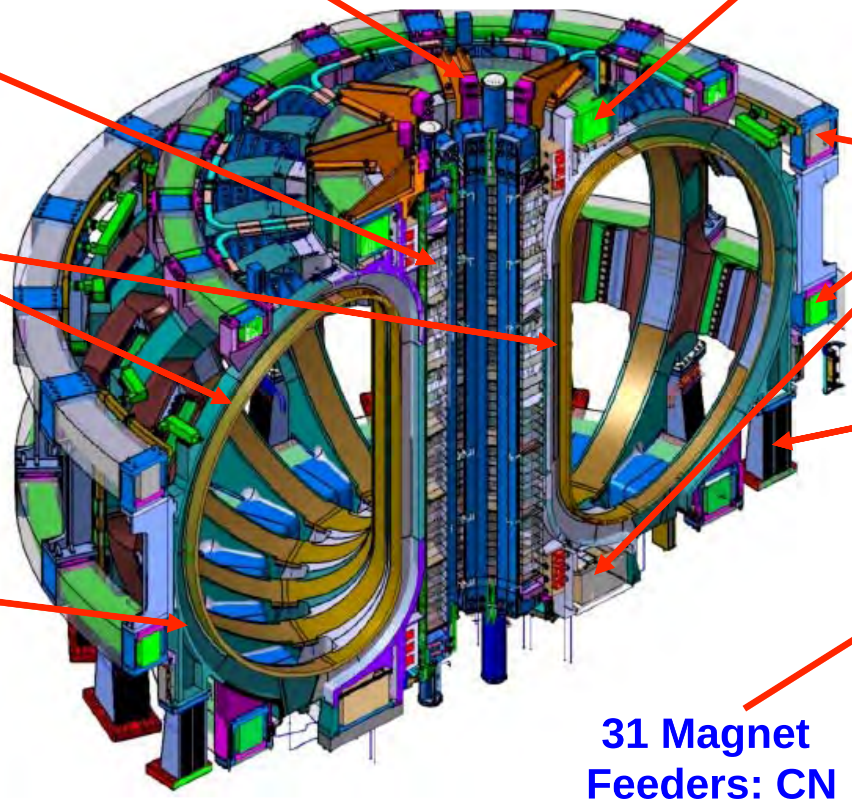
9+1 TF coils with EU
(Iberdrola/ASG/Elytt
Consort.)

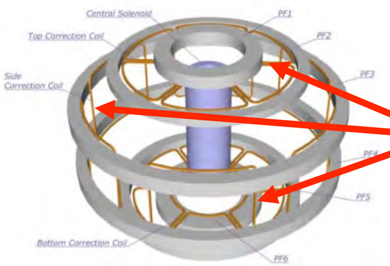
PF2-6: EU
(*not decided*)

9 TF coils with JA
(MHI)

Magnet Supports: CN
(SWIP)

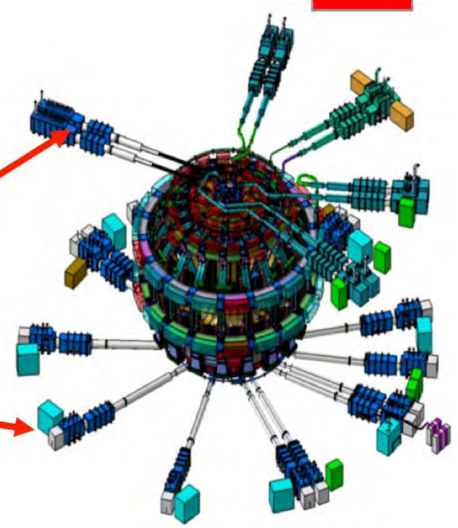
19 TF coil cases: JA
(MHI/Hyundai
Consor





9 pairs of CCs: CN
(ASIPP)

31 Magnet
Feeders: CN
(ASIPP)



ITER Conductor Supply: 11 PAs

Nb_3Sn

TF Conductor

(dummy)
CS Conductor

PF Conductor

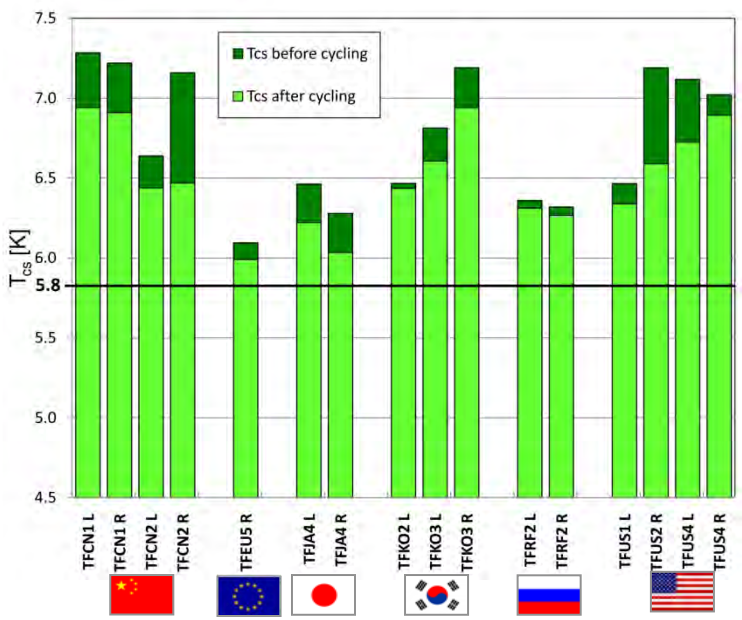
MB Conductor

Nb–Ti

CC Conductor

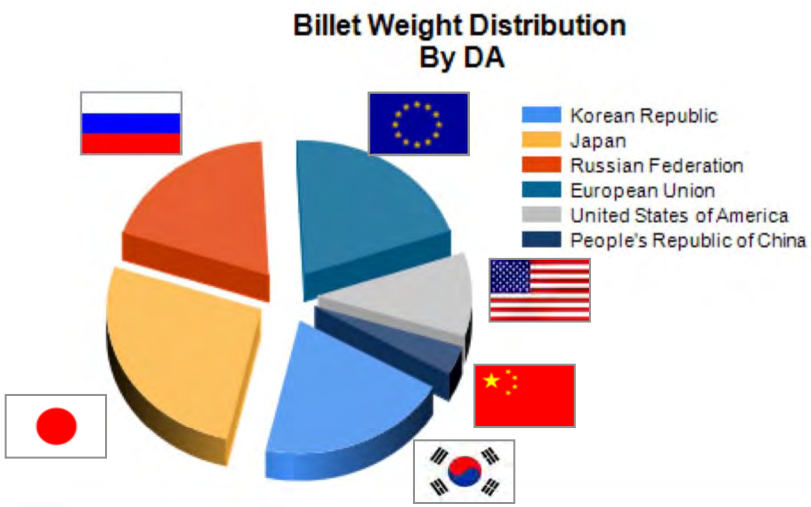
(dummy) CB Conductor

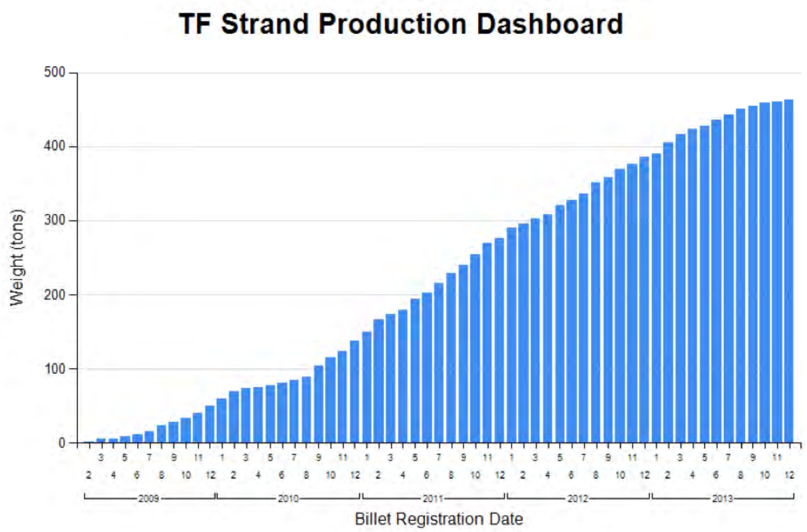
TF Conductor Production



TF Qualification Sample Summary

- **All TF conductor qualification samples were manufactured at CRPP and enabled supplier qualification in all 6 DAs involved in TF conductor production**
- **Currently, >95% (95,000 km) of required 450t of Nb_3Sn strand has been produced around the world**



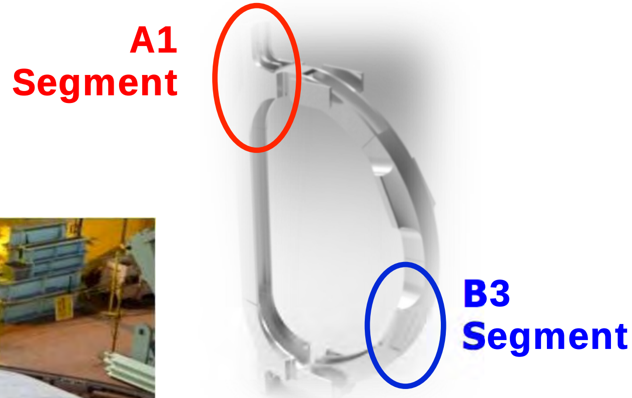


TF Strand Production Summary

Progress on Prototyping TF Structures

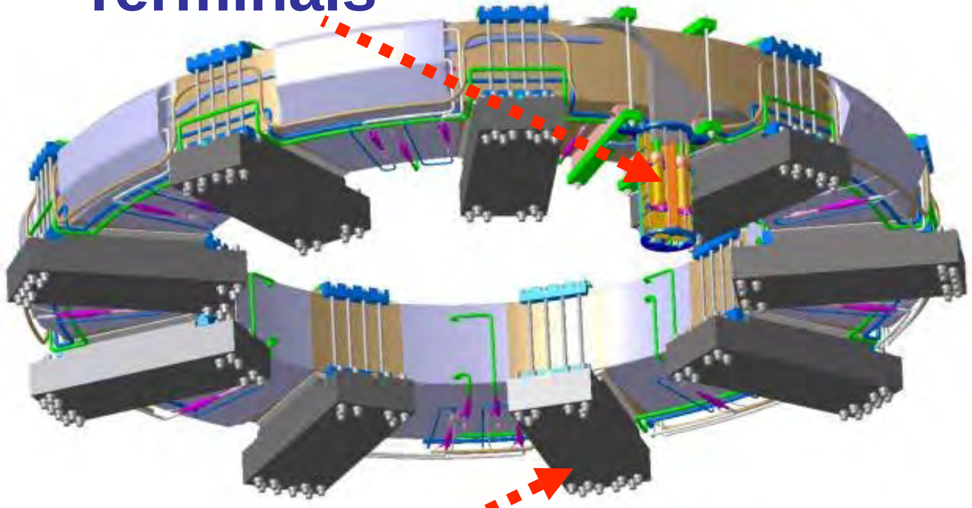
- **EU has manufactured 2 full size Radial Plate (RP) prototypes, while JA has manufactured 1 full size RP prototype**

Full-Size rDP RP Prototype in EU

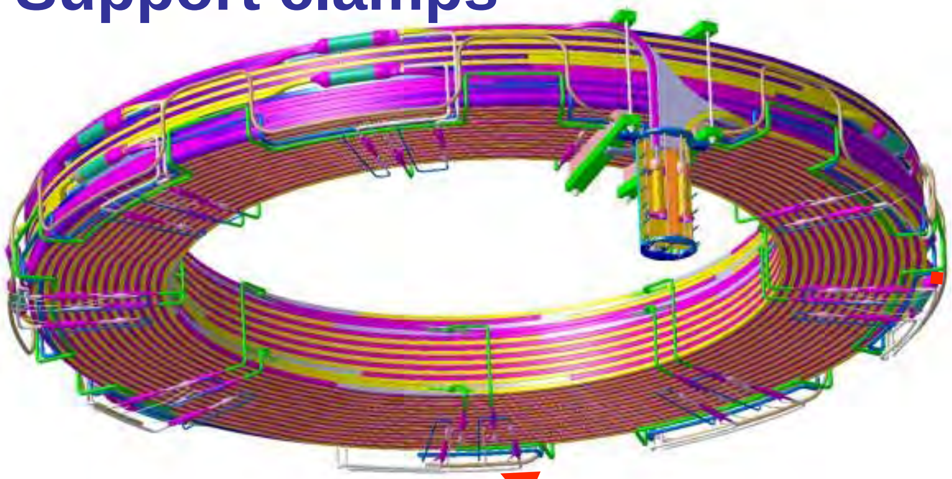


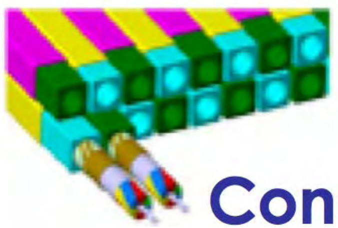
- **JA has manufactured full-scale mock ups of 2 TF coil structure segments**

Poloidal Field Coils

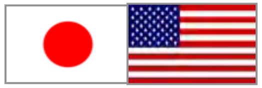


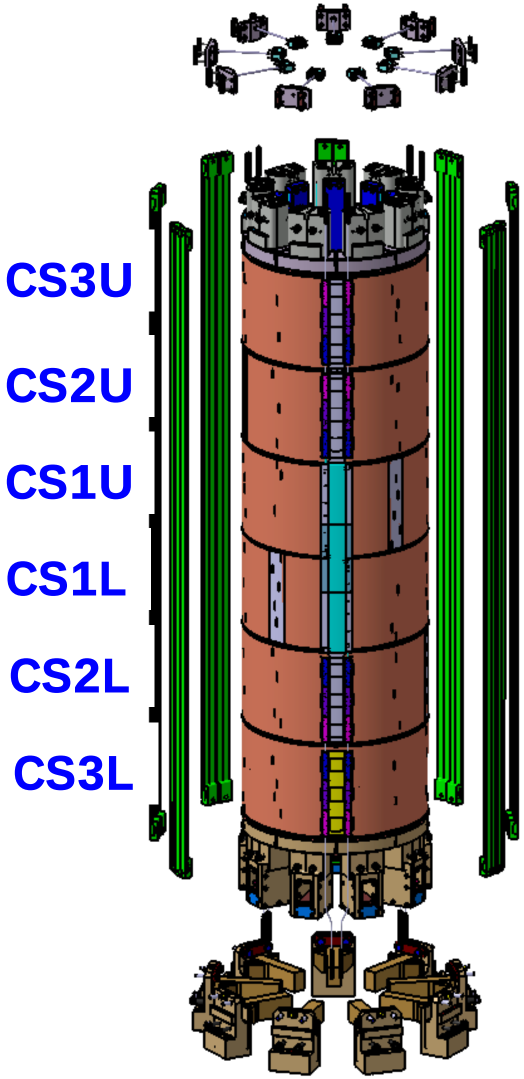
- So large that most must be manufactured on site
- PF3: 24.5 m dia. & 386 ton
- Building is 250 m long x 45 m wide and is the first building on site!





Central Solenoid R&D and Construction Advancing





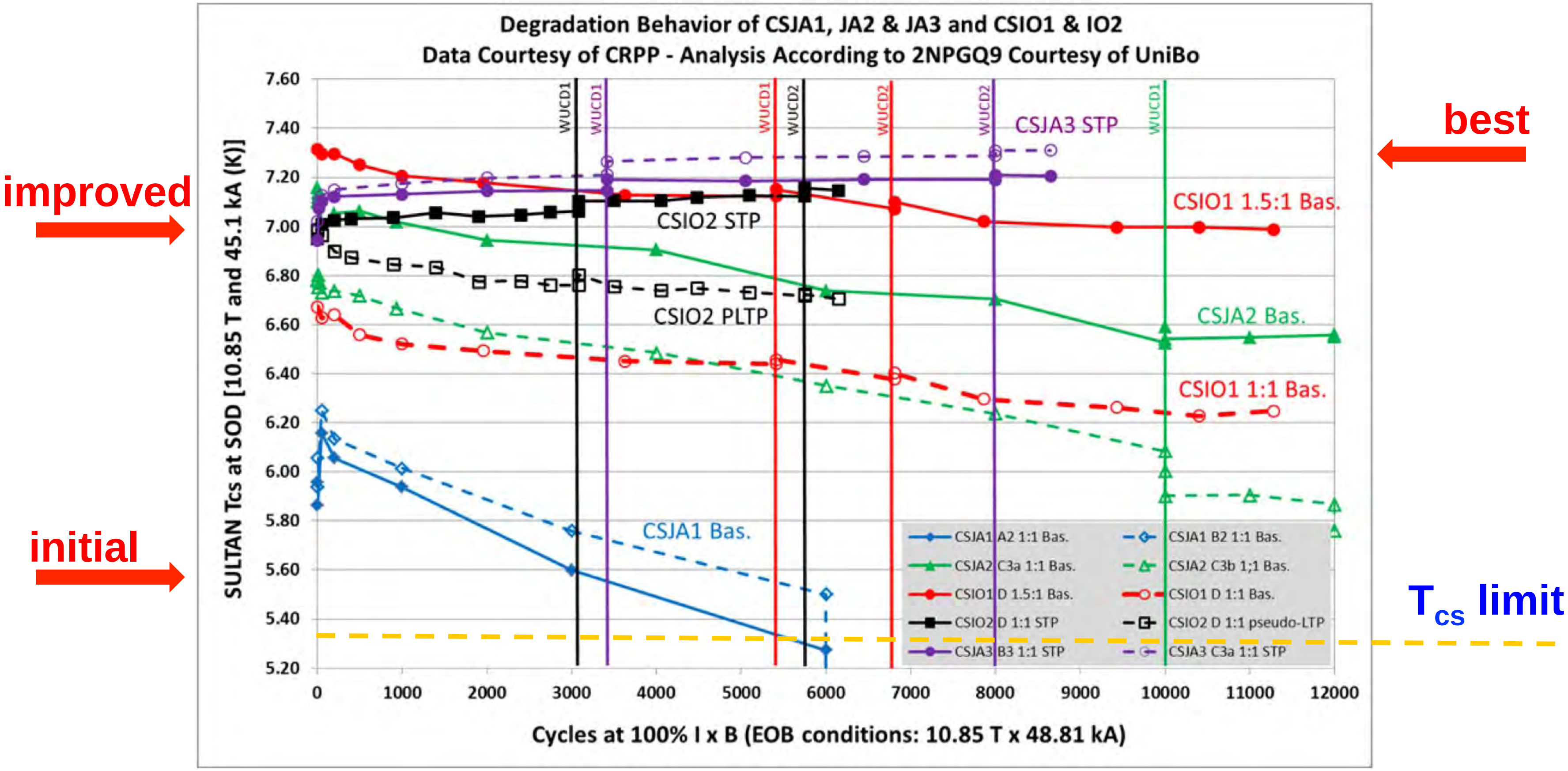
- **In comparison to the TF coils, which are operated in a steady state, the CS and PF coils must drive inductively 30,000 x 15 MA plasma pulses with a burn duration of ~400 s**
- **During their life time, the CS coil modules will have to sustain severe and repeated electromagnetic (EM) cycles to high current and field conditions**

⇒ ***Well beyond anything large Nb_3Sn coils have ever experienced***

Note: CS conductors are to be procured in kind by Japan; they are 100% funded by EU (Broader Approach) and will be delivered to the USA for coil manufacture

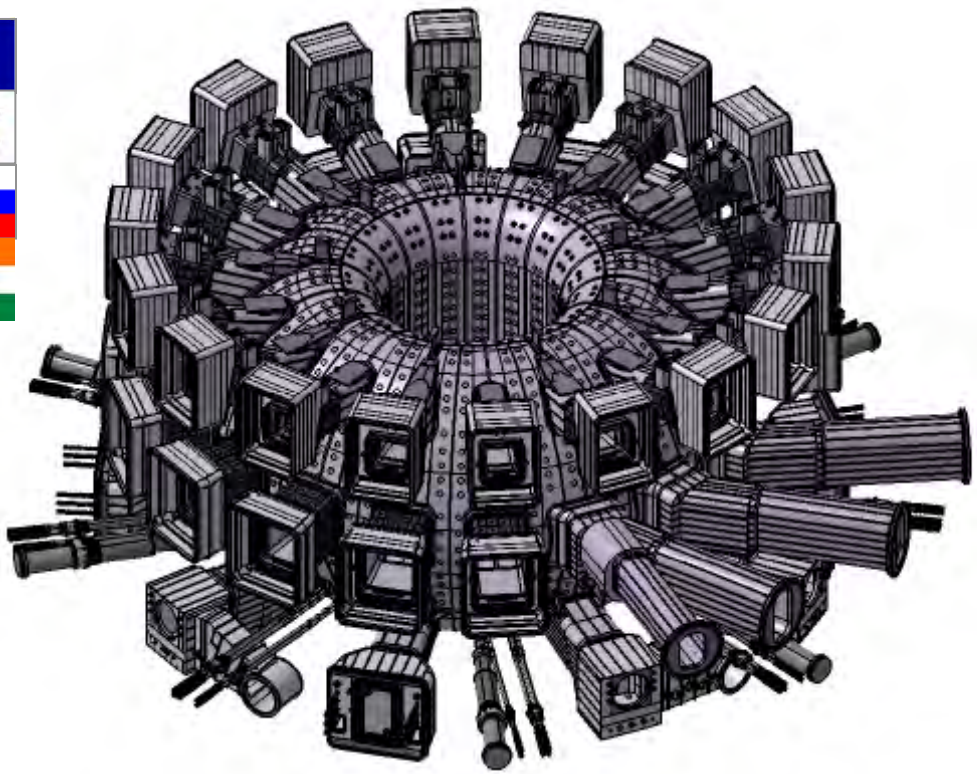
CS Conductor Meeting ITER Specifications

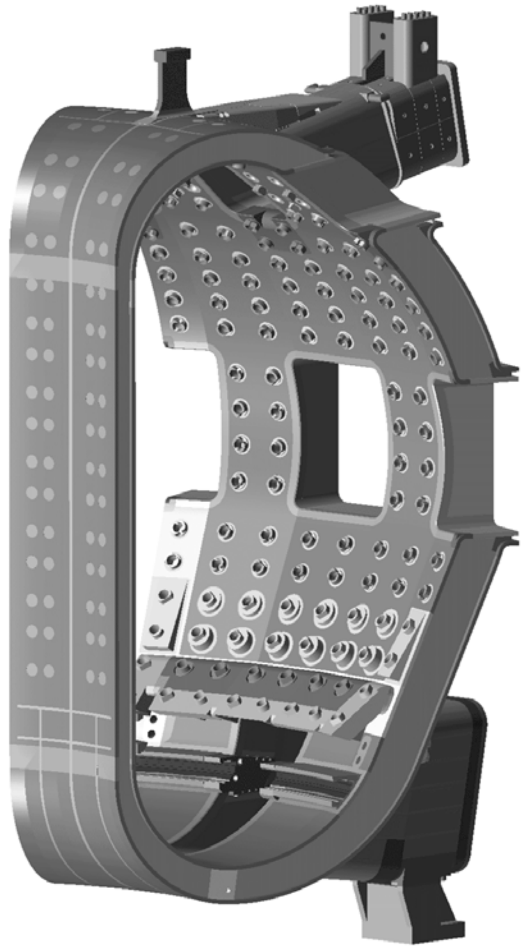
Successful Qualification Test



- Rapid development of improved Nb_3Sn conductor in response to initial problems identified in CS conductor qualification tests

Vacuum Vessel Manufacturing Contracts Awarded

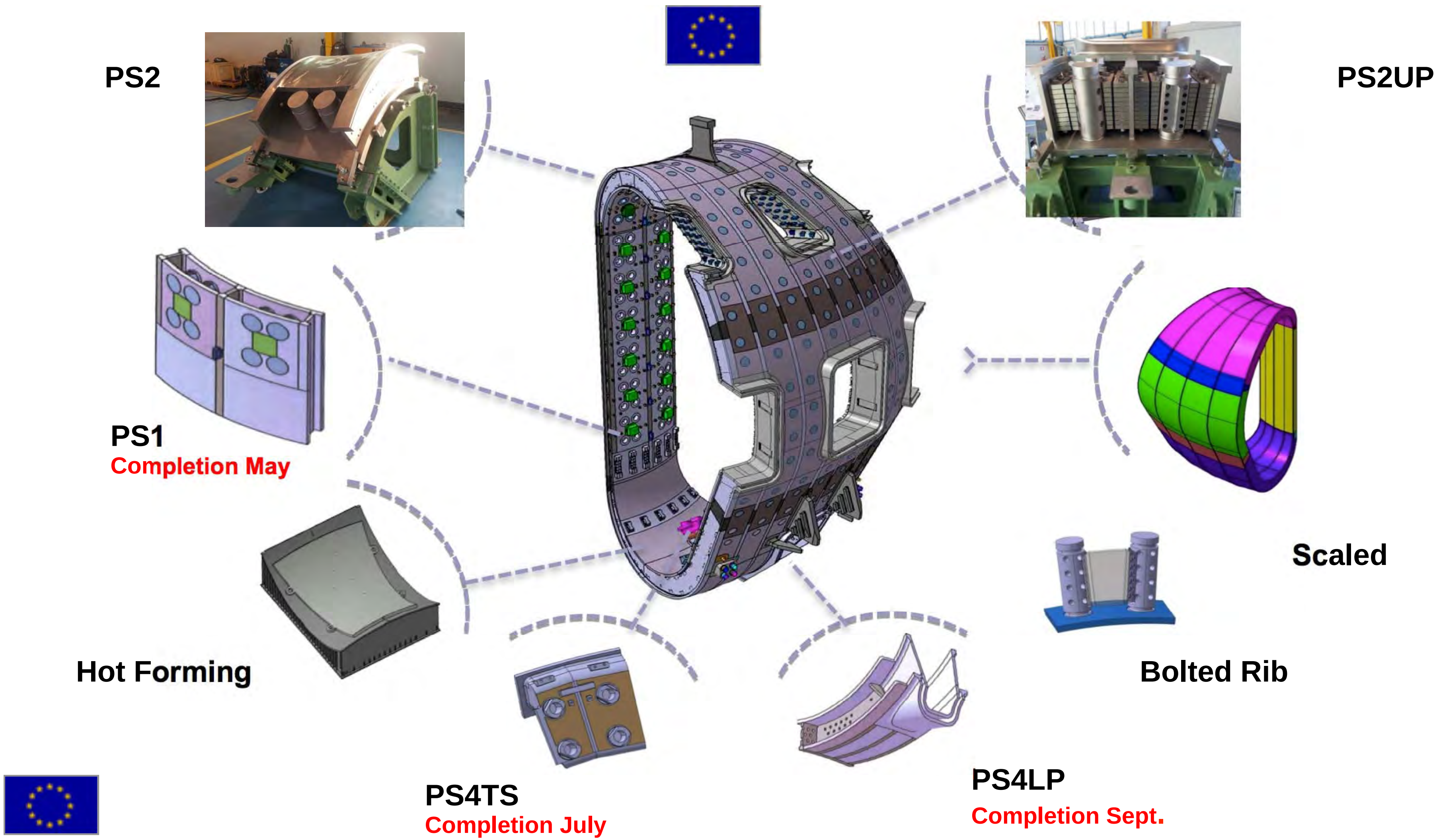




- **Vacuum Vessel is a double-walled stainless steel structure**
  - 19.4m outer diameter, 11.3m height, 5300 tonnes
  - provides primary tritium confinement barrier
- **VV sector, port and in-wall shielding PAs signed (EU, KO, RF, IN)**
  - industrial contracts awarded in each area by corresponding DA

ITER Vacuum Vessel - 7 Sectors (EUDA)

<u>Main mock-ups almost completed, to validate manufacturing route</u>



ITER Vacuum Vessel – 2 Sectors (KODA)

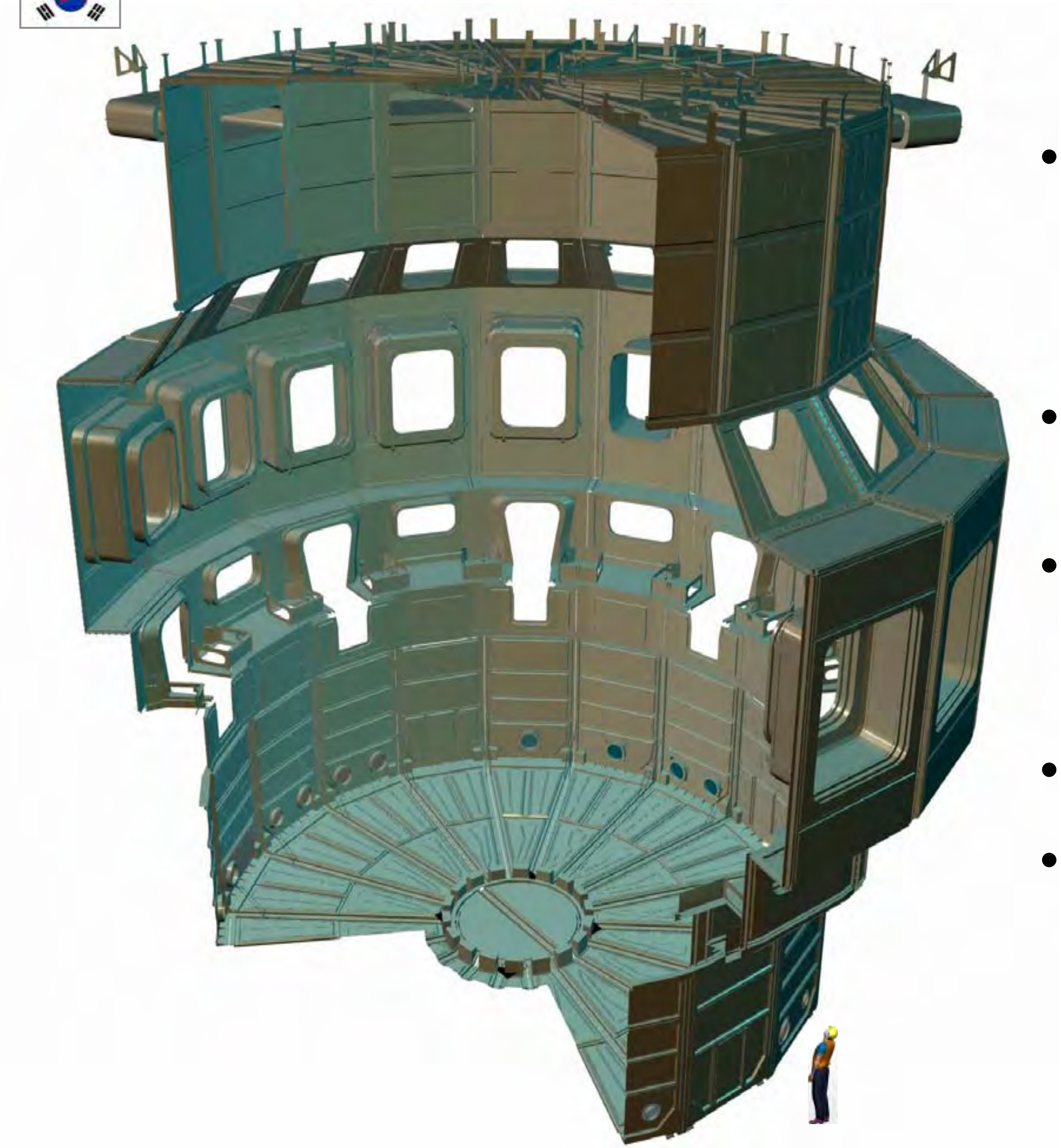
Fabrication of Upper Segment (PS2)

Welding of Centering Keys and Inter-Modular Keys

Port Stub

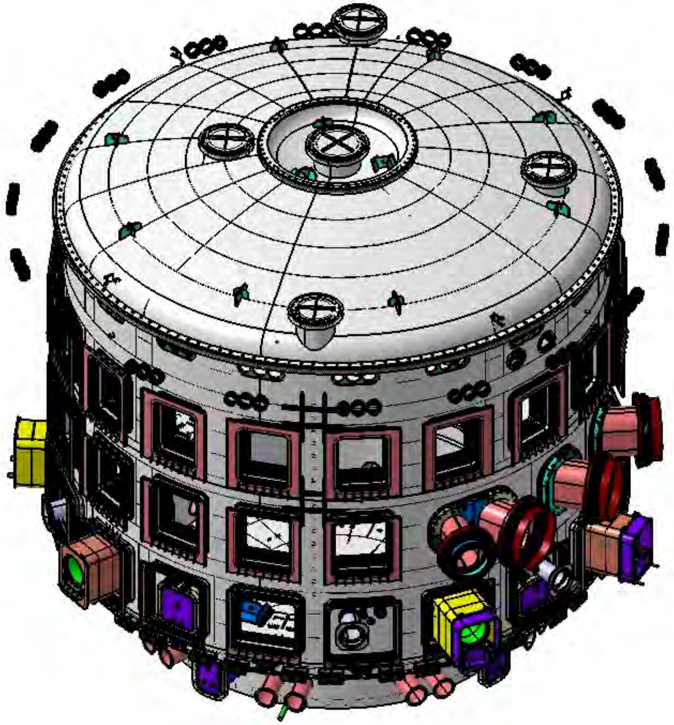
R&D for IVC Rail Support

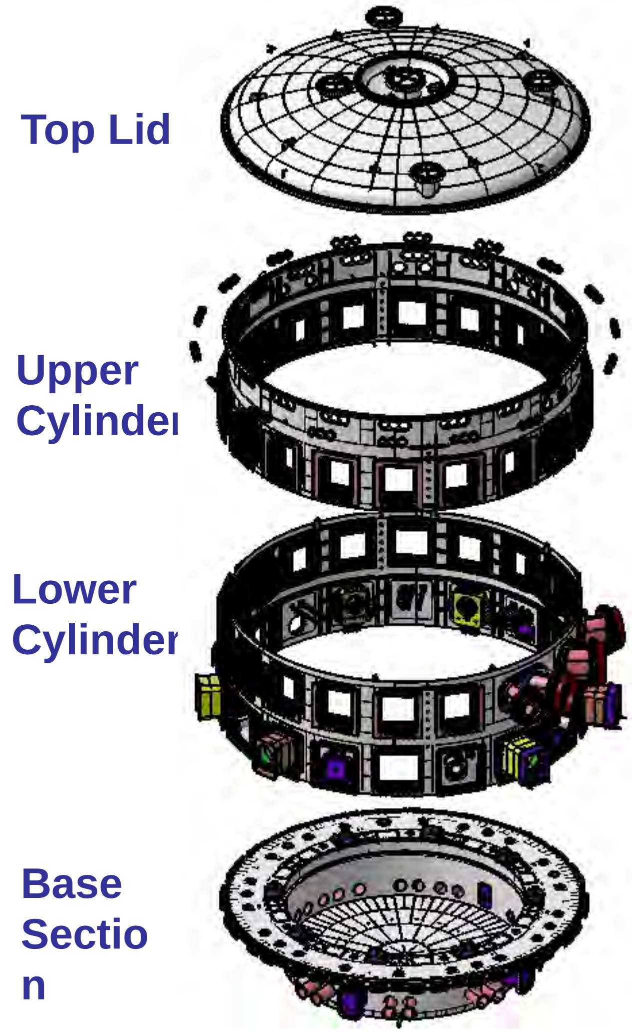
Main Inner Heat Shield



- Provides barrier for thermal loads from warm components to the superconducting coils (4.5K)
- Operates at 80 K (gaseous He in cooling pipes)
- Stainless steel panels are silver coated to reduce emissivity
- Mass: ~1000 t
- A smaller shield isolates the TF coils from the vacuum vessel

Cryostat



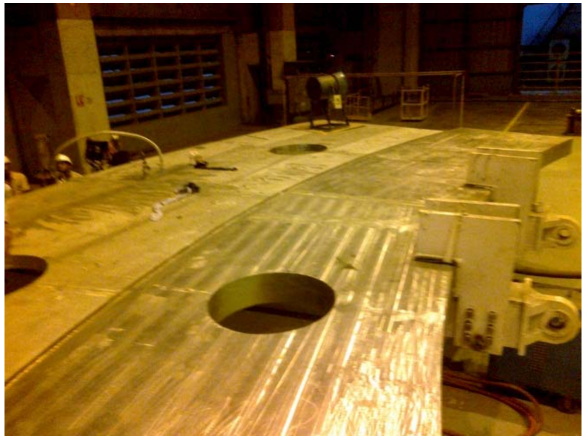


- 304L Stainless steel
  40 – 180 mm thick
- Diameter: 29.4 m/ Height: 29 m
- Weight ~3500 tonnes
- Base pressure < 10^{-4} mbar
- Transfers loads to tokamak complex floor

- **IN-DA signed PA September 2011**
- **Contract awarded in August 2012**

40° Prototype Cryostat Base Section

Cryostat pedestal ring- top plate 200 mm and skirt plate 105 mm thick welded together



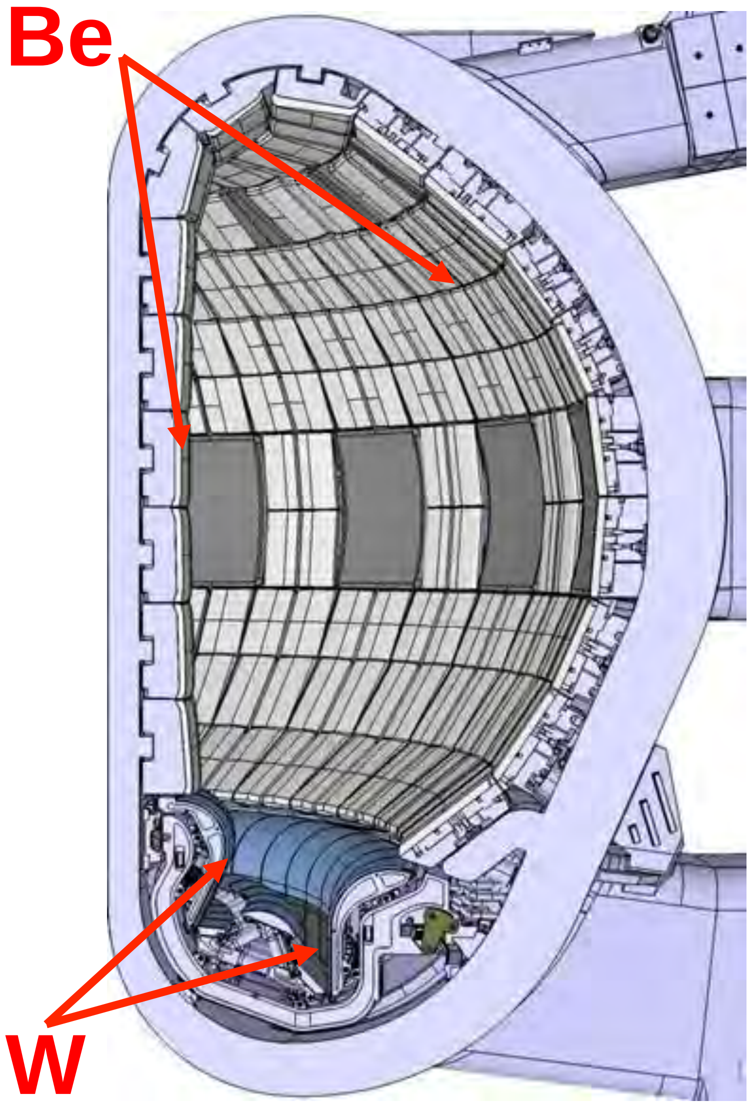
Cryostat pedestal ring- bottom plate (180 mm & side plates (120 & 80 mm thick being welded

ITER Plasma Facing Components

ITER will operate with all metal PFCs

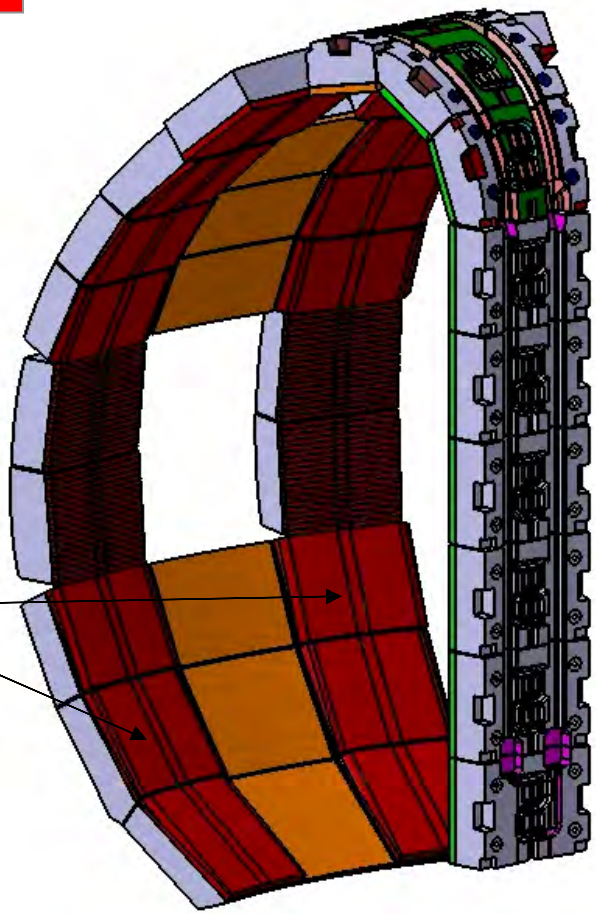
- **Be first wall (~700m²):**
  - low-Z limits plasma impurity contamination
  - low neutron activation
  - low melting point
  - erosion/ redeposition will dominate fuel retention
  - melting during disruptions/ VDEs
  - dust production

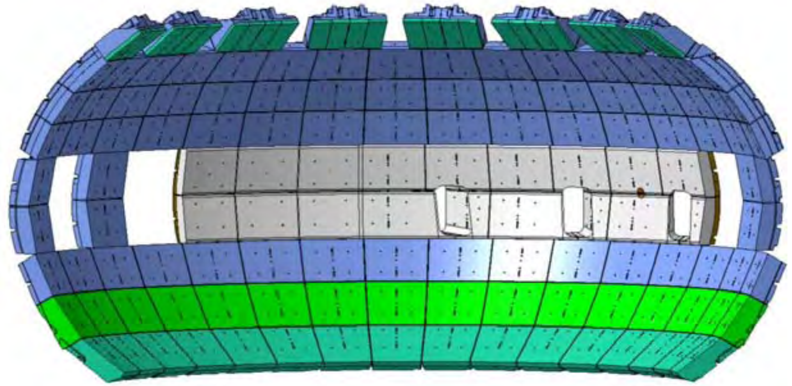
- **W divertor (~150m²):**
  - resistant to sputtering
  - limits fuel retention (but note Be)
  - melting at ELMs, disruptions, VDEs
  - W concentration in core must be held below ~ 2.5 × 10^{-5}



In-Vessel Components - Blanket/ First Wall

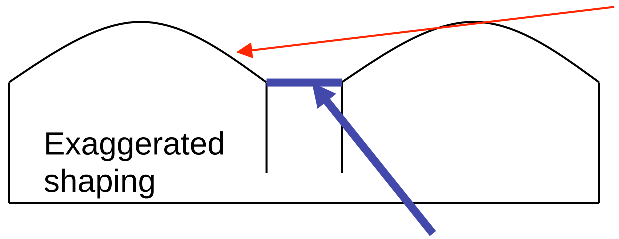
Poloidal strips of modules advanced to protect port region





Toroidal Asymmetry

Single Module Section



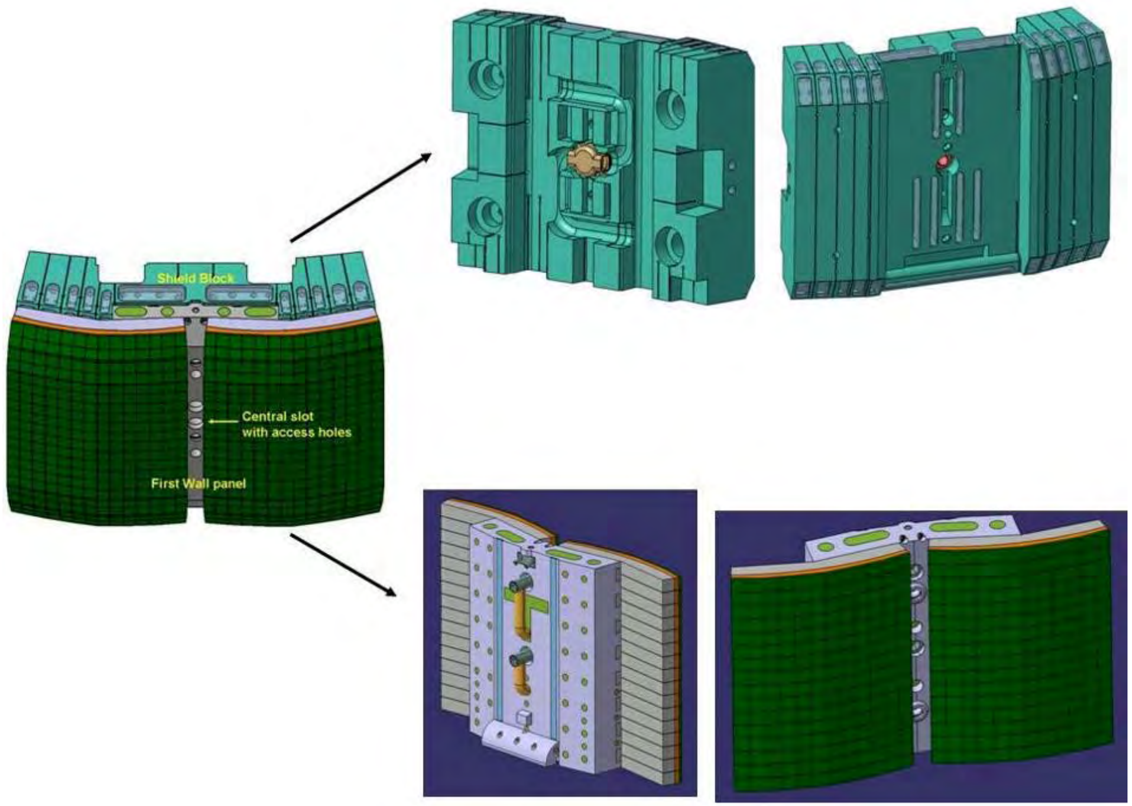
Blanket/ FW:

- Contributes to neutron shielding for superconducting coils
- Exhausts majority of plasma power
- Provides limiting surfaces for plasma start-up and shutdown

Blanket Shield Modules and First Wall Panels

Shield Module

- **Facts:**
  - 440 blanket modules
  - ~4 tons each
  - 18 poloidal rows
  - 18 or 36 toroidal rows
  - ~40 different modules
  - Mass: 1530 tons



- **Technical Challenges:**
  - Large electromagnetic loads
  - High heat flux ~ 5 MW/m^2
  - Material bonding techniques
  - Plasma-material interactions
  - Integration with in-vessel coils, diagnostics and blanket manifold.
  - Remote handling requirements

First Wall Panel

ITER Blanket – FW panel heat handling

Secondary divertor interactions

Start-up

Start-up

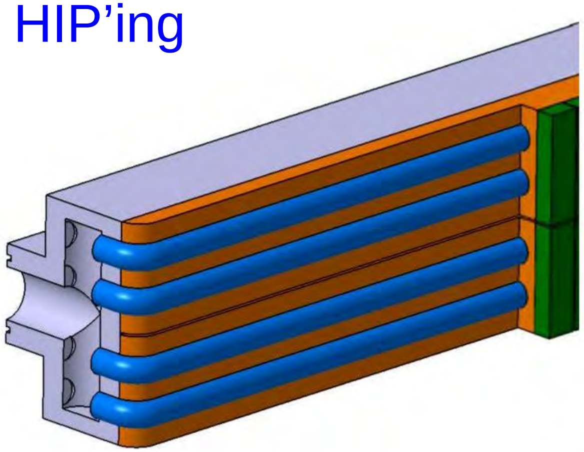
Ramp down

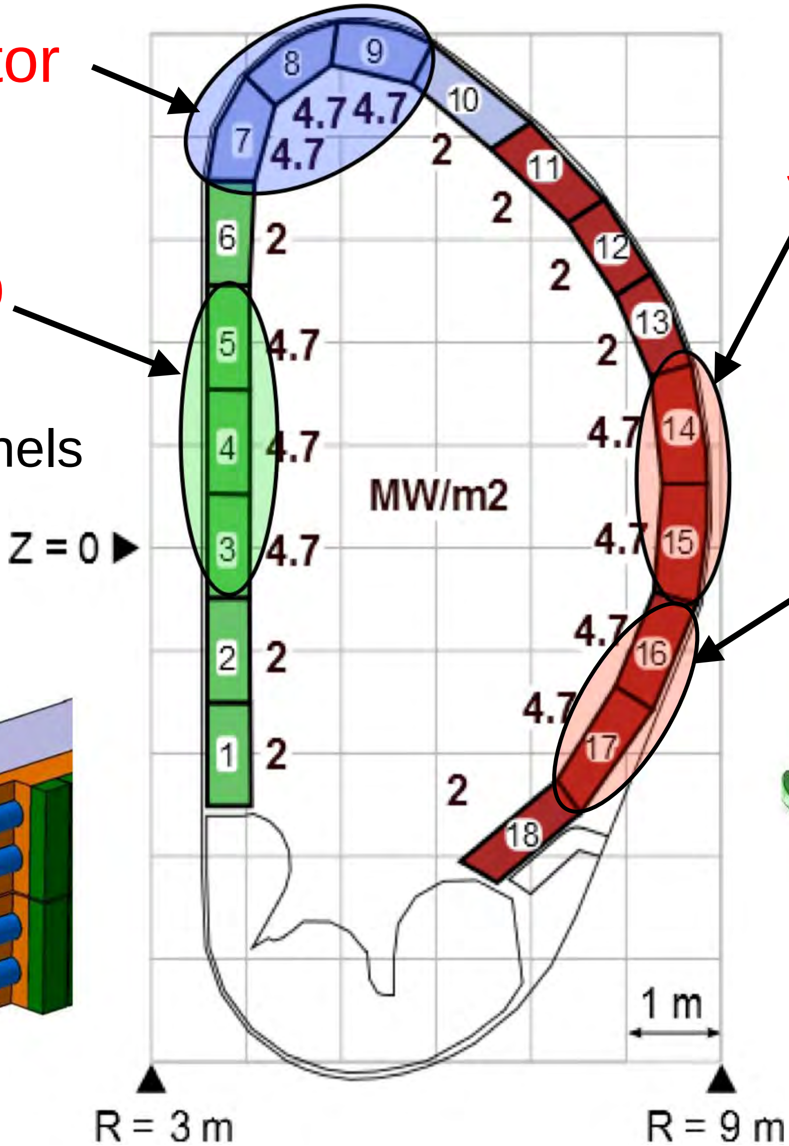
Normal Heat Flux panels (NHF)

$q_{\perp} \approx$ **1- 2 MWm²**

Steel cooling pipes

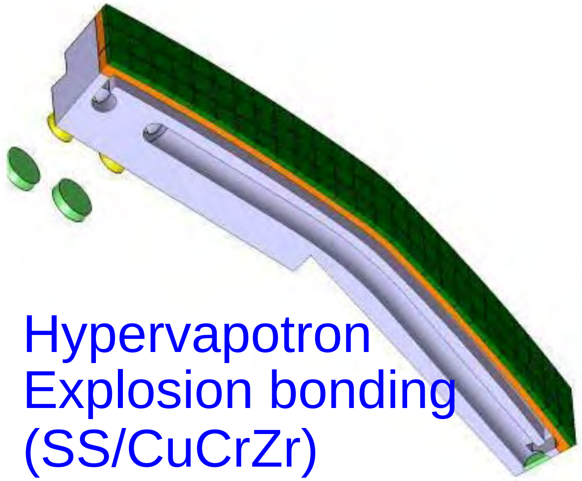
HIP'ing





Enhanced Heat Flux panels (EHF)

$q_{\perp}$ up to **4.7 MWm²**

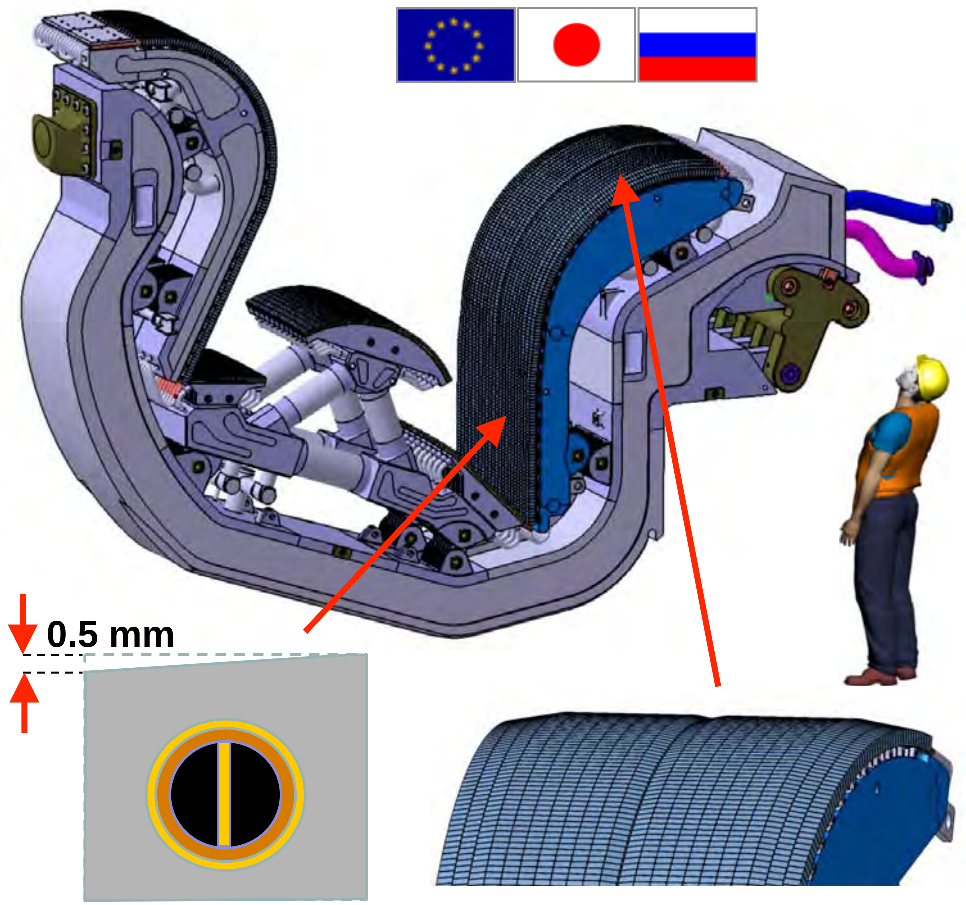


Hypervapotron

Explosion bonding (SS/CuCrZr)

Brazing (Be/CuCrZr)

All-Tungsten Divertor



- Particular attention required to shaping for leading edge protection ⇒ avoid worst cases of melting due to transients

- Significant contribution to cost containment for ITER Project
- Experience gained in operation with W-divertor in non-active phase, including development of ELM-mitigation techniques
- Low fuel retention and lower dust inventory

But:

- Will require more cautious approach in non-active phase
- Need to ensure effective disruption and ELM mitigation early in operational period
- Need to develop suitable operational scenarios, particularly for non-active phases of operation

Divertor Outer Vertical Target - Testing

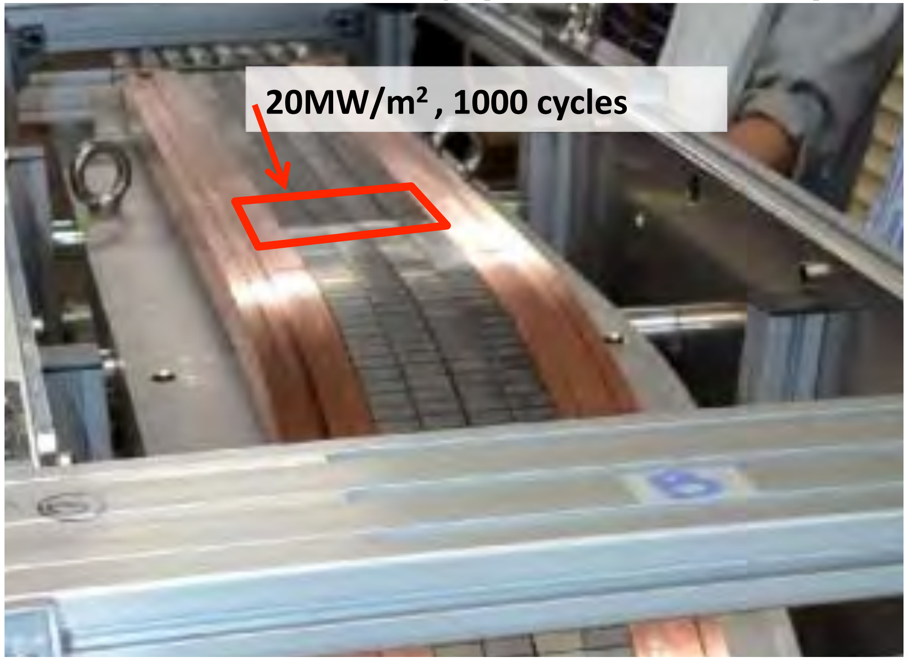
- **4 full-scale OVT PFUs manufactured and inspected by Kawasaki Heavy Industry (1st set of PFUs) were HHF tested in the IDTF in late 2012 (RFDA) . After HHF test, PFUs were sent back to JADA in early 2013 for final assembly into a full-scale prototype OVT.**

Main results:

- **Straight part W – tested at 10 MWm^{-2} 5000 cycle & 20 MWm^{-2} 1000 cycle**
- **Curved part W – tested at 5 MWm^{-2} 1000 cycle**

Steel Support Structure after rough machining

4 full-scale CFC/W OVT PFUs mounted on HHF test assembly (Before HHF test)

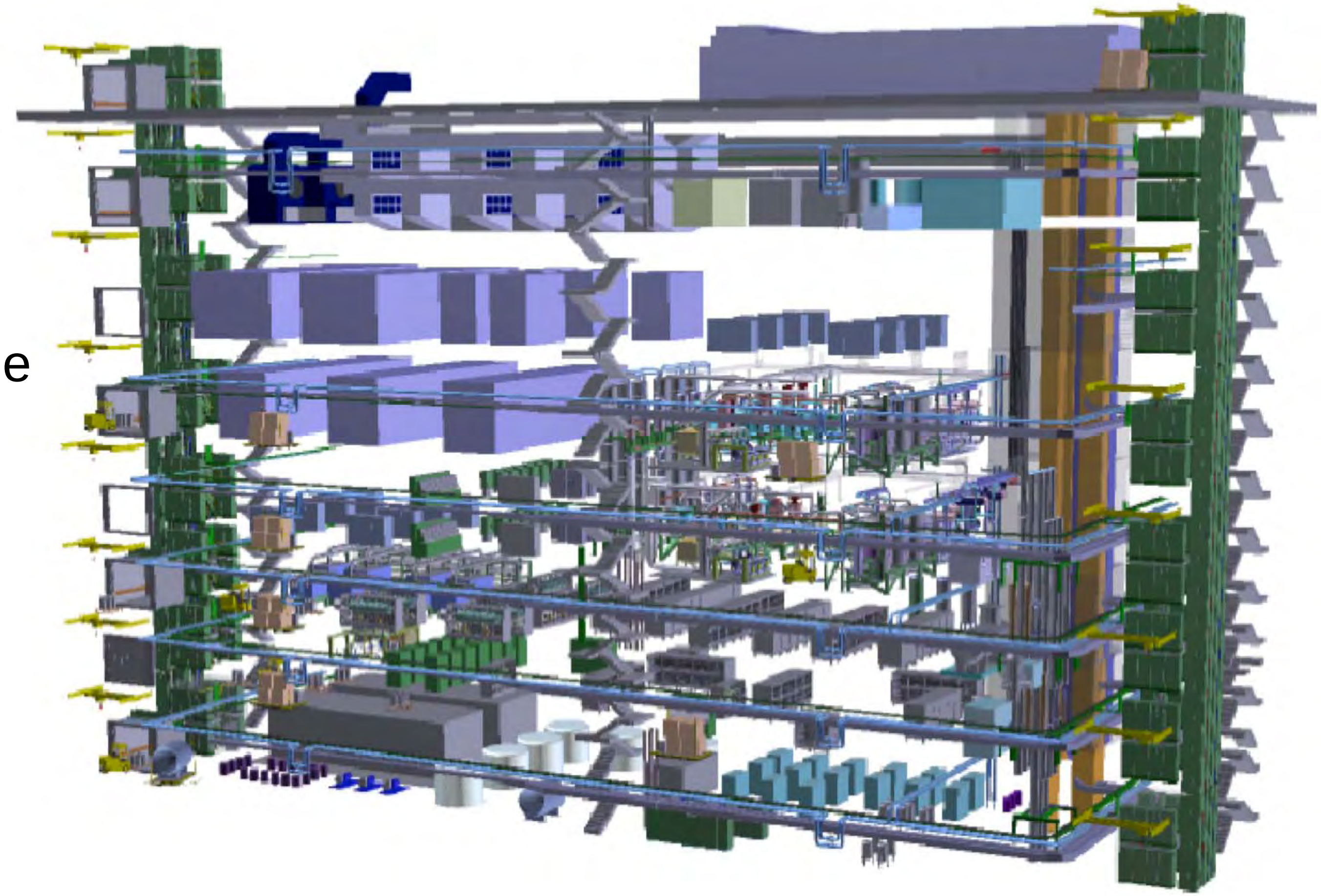


ITER Plant Systems

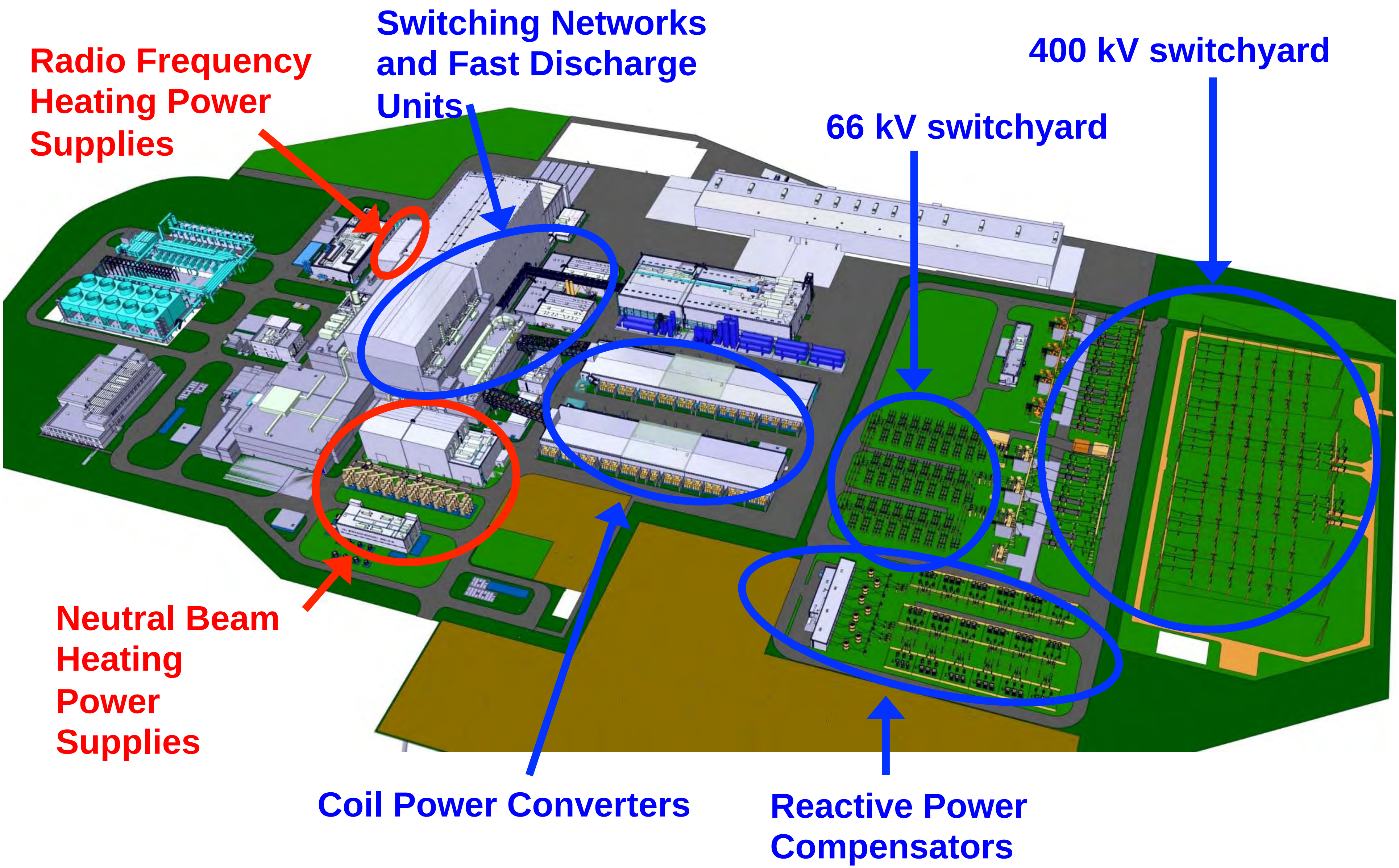
Tritium Plant

- **4 kg of tritium will be stored on-site**
- **~100 g tritium required for a standard Q=10 pulse, but only <1% is actually burned:**
  – tritium reprocessing required

- **7 Stories**
  – 2 below grade
- **L = 80 m**
- **W = 25 m**
- **H = 35 m**

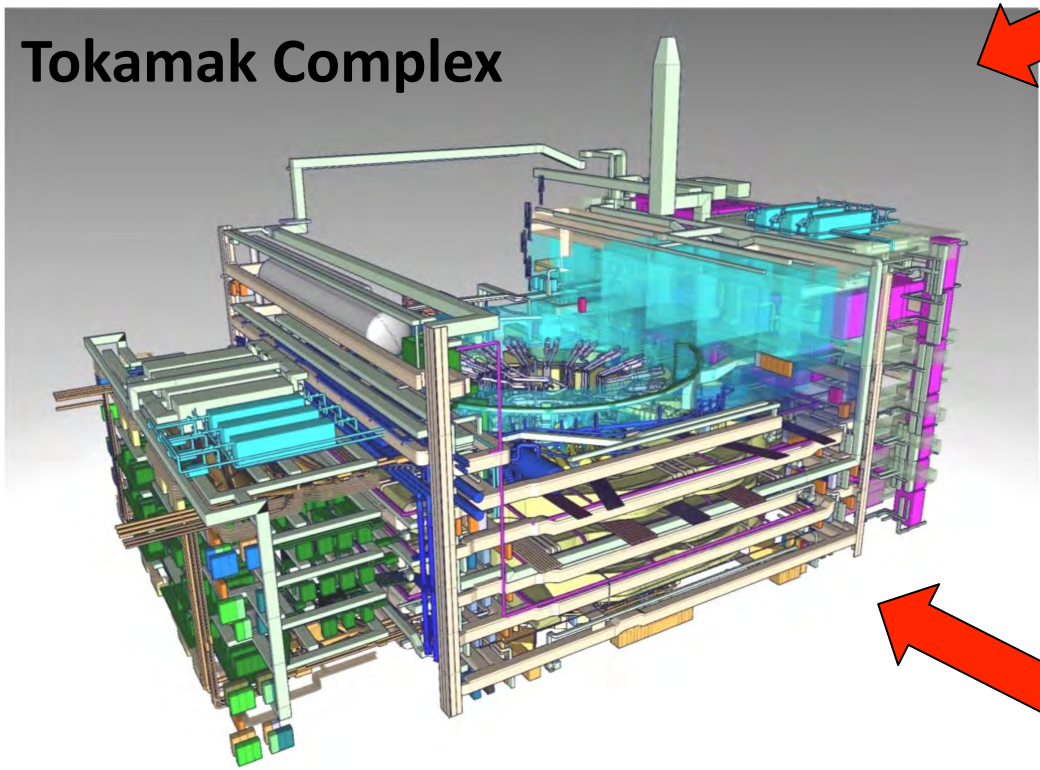


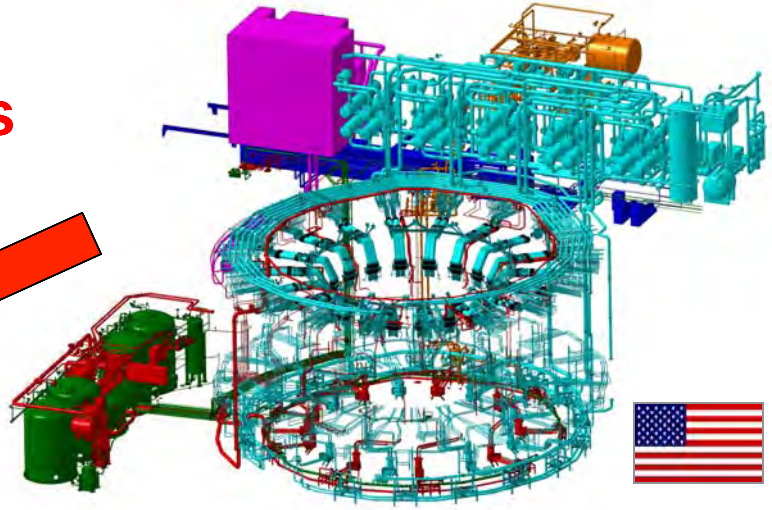
Main pulsed power supply components



Plant Systems' Configuration Models

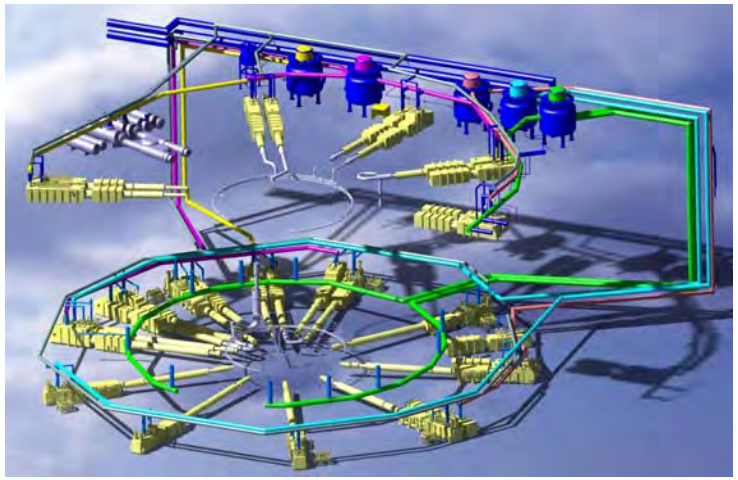
Major focus on integrating systems into the Tokamak Complex





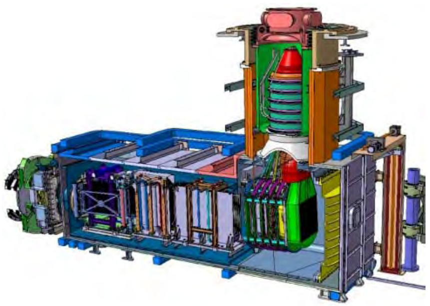
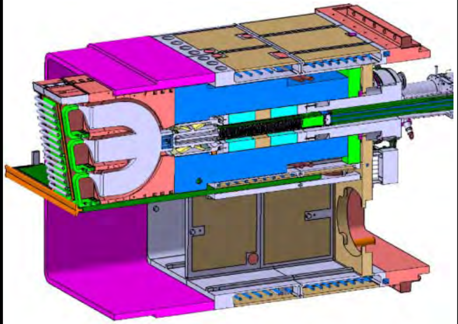
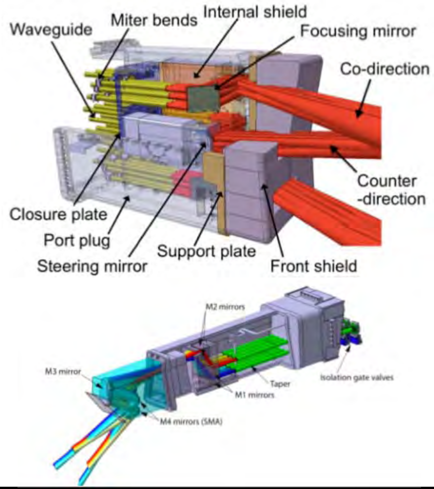
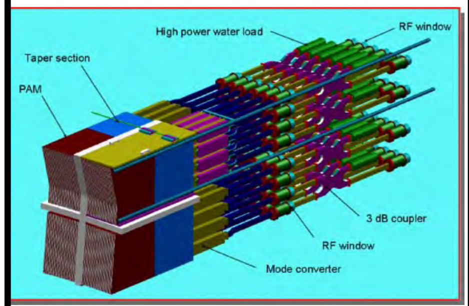
Tokamak Cooling Water System

Cryodistribution



ITER Ancillary Systems

ITER heating and current drive systems

<table>
<tr><th>NB</th><th>IC</th><th>EC</th><th>LH</th></tr>
<tr><td>Neutral Beam
- 1 MeV</td><td>Ion Cyclotron
40-55MHz</td><td>Electron Cyclotron
170GHz</td><td>Lower Hybrid
~5 GHz</td></tr>
<tr><td></td><td></td><td>
</td><td>
</td></tr>
<tr><td>33MW*
+16.5MW#</td><td>20MW*
+20MW#</td><td>20MW*
+20MW#</td><td>0MW*
+40MW#</td></tr>
<tr><td>Bulk current drive
limited modulation</td><td>Sawtooth control
modulation < 1 kHz</td><td>NTM/sawtooth control
modulation < 5 kHz</td><td>Off-axis bulk current
drive</td></tr>
</table>

***Baseline Power**

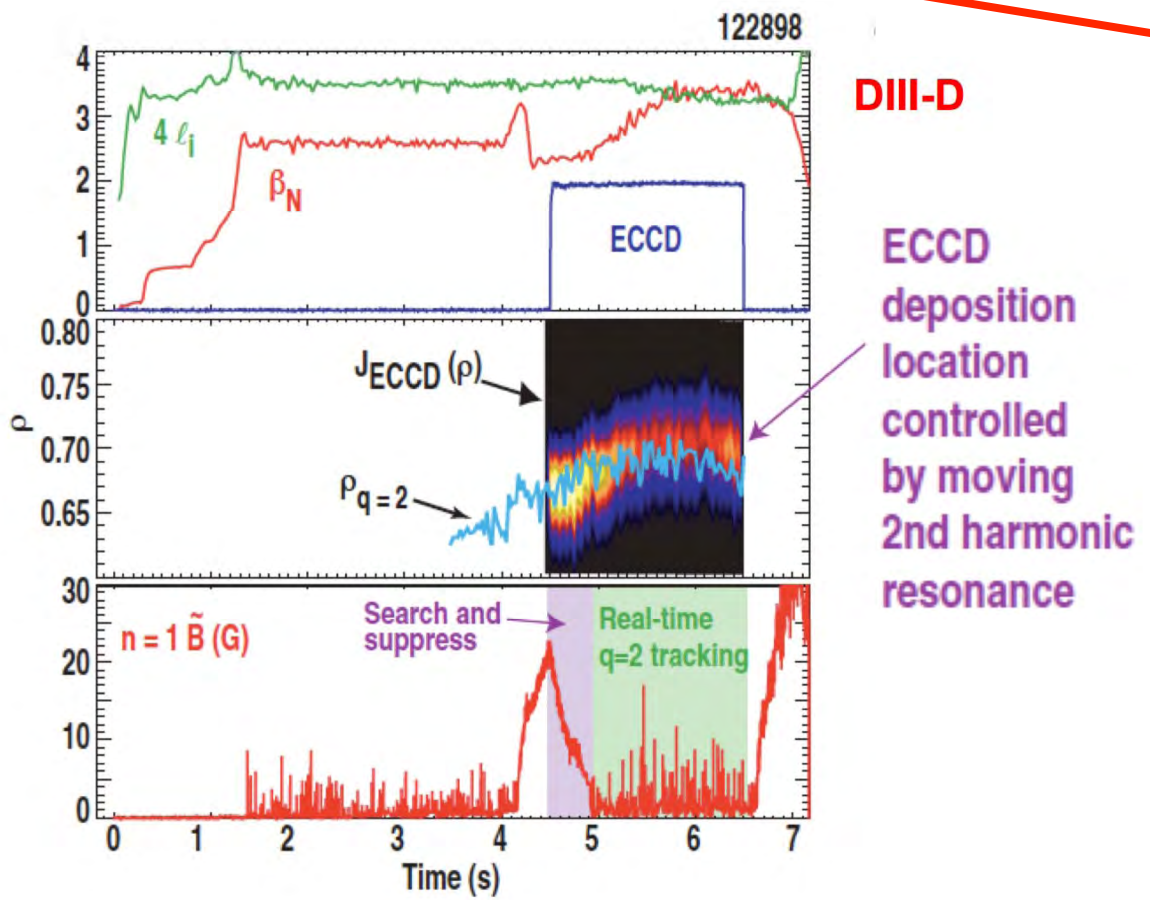
#Possible Upgrade

Why 4 Heating Systems?

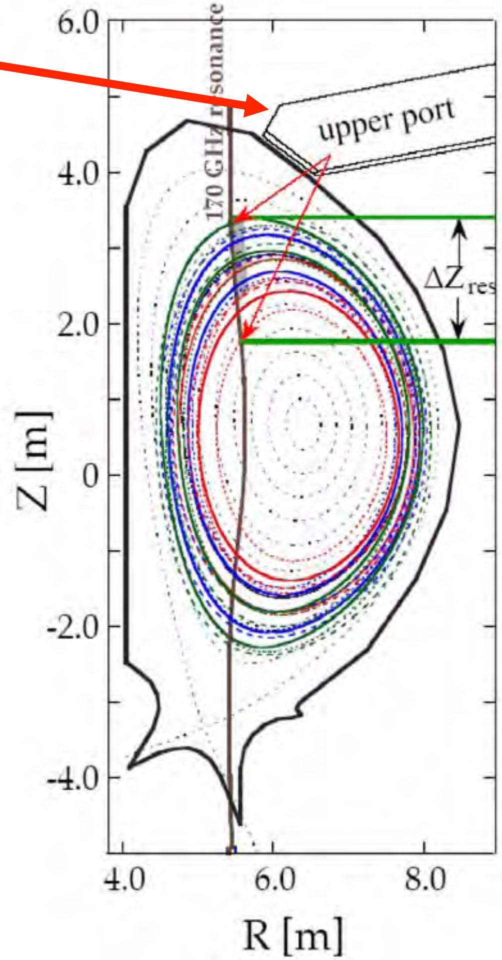
- **Technology:**
  - ICRF and LHCD fairly conventional
  - NBI and ECRH source technology challenging
- **Coupling to plasma:**
  - NBI and ECRH straightforward
  - ICRF and LHCD problematic: antenna design challenging due to difficulty in coupling wave through (evanescent) plasma edge
- **Radial localization:**
  - Resonance condition favours ECRH and ICRF radial localization
  - NBI and LHCD more global in effects
- **Current drive:**
  - NBI and LHCD most efficient
  - ECRH and ICRF used in more specialized applications where space localization important

Control of Neoclassical Tearing Modes

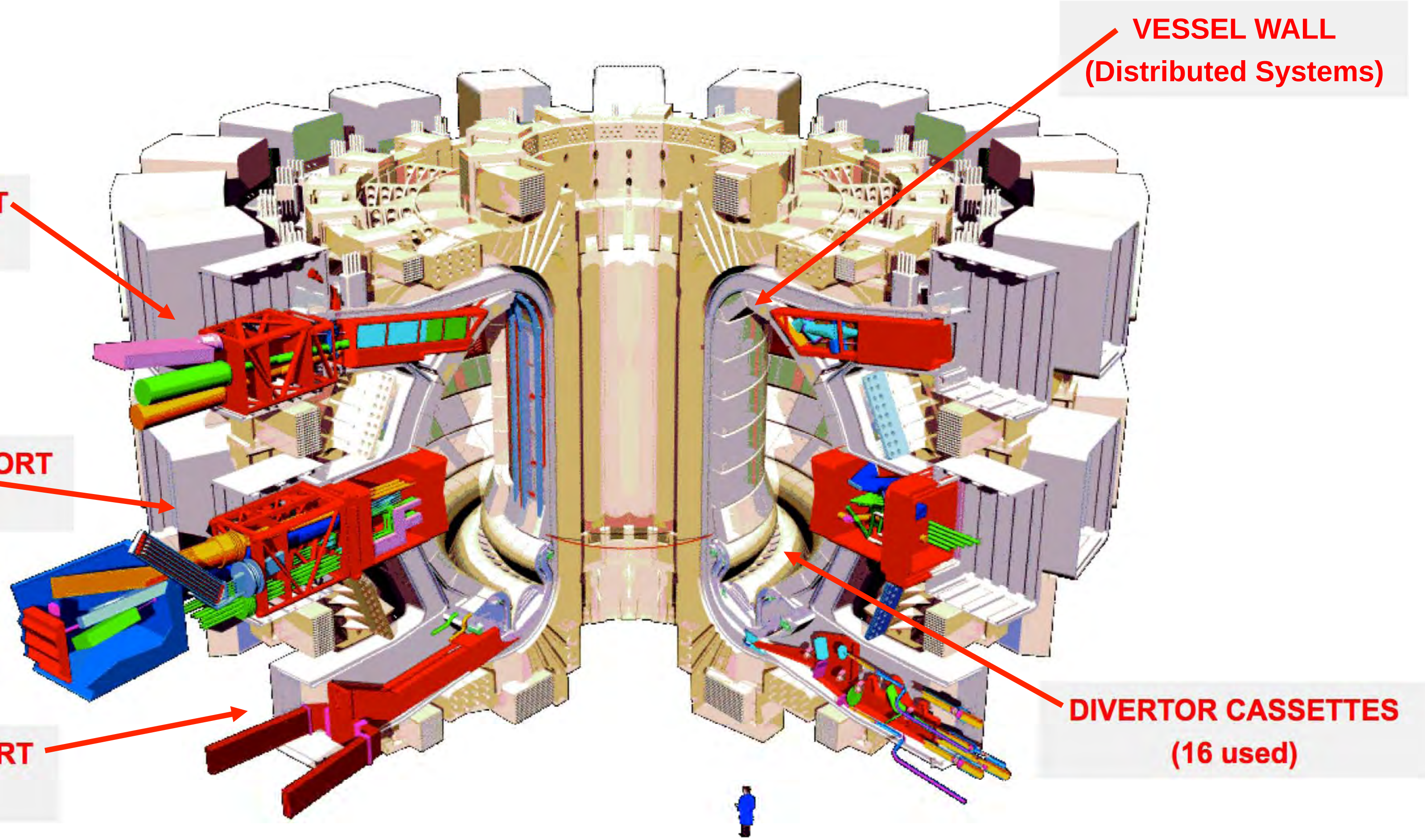
- **An MHD instability is detected (magnetically, SXR, ECE …):**
  - localized electron cyclotron current drive is used to suppress the instability
  - ITER has 4 steerable upper ECH&CD launchers launching 20 MW



M R Wade et al, FEC-2006, OV1-4

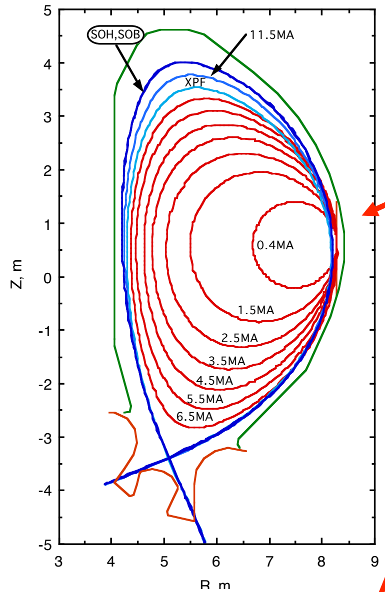


Analyzing the Plasma - ITER Diagnostics



- **About 40 large scale diagnostic systems are foreseen:**
  - Diagnostics required for protection, control and physics studies
  - Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  - Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ….)

Fusion Plasma Diagnostics

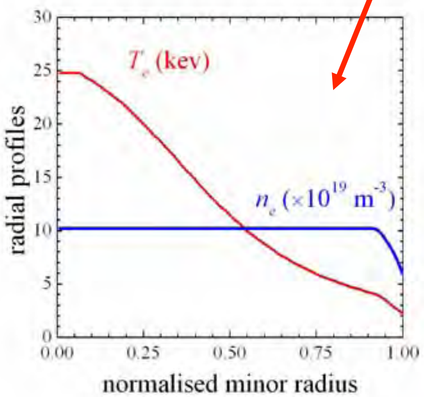


Plasma shape evolution (ITER)

Plasma-wall interaction (JET)

Plasma density and temperature (ITER)

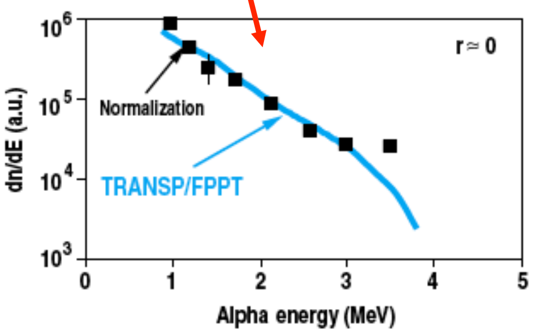
Plasma radiation (ASDEX-U)

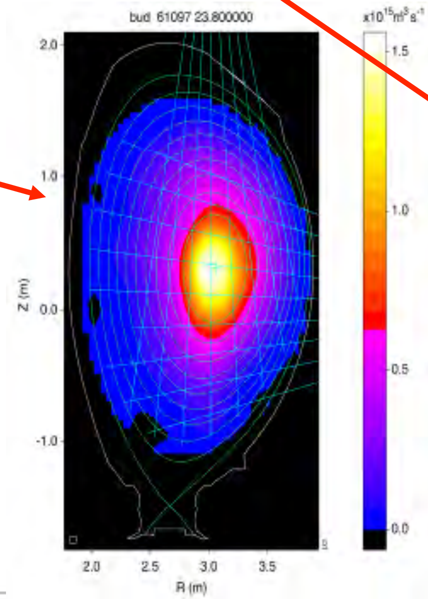


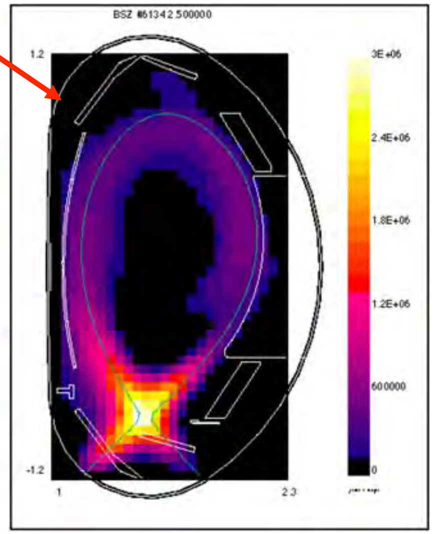
Fusion power:

14MeV neutron profile (JET)

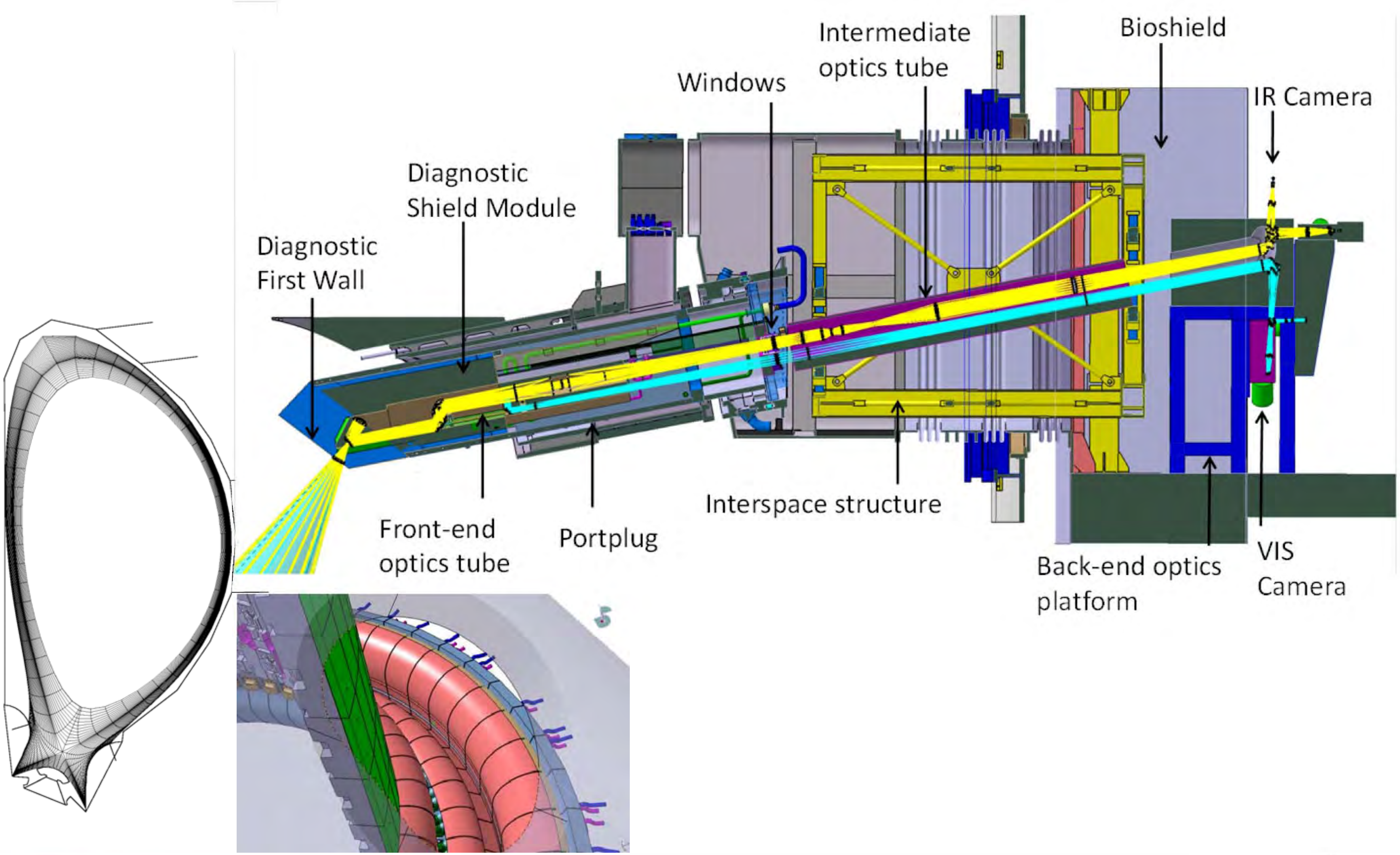
α-particle spectrum (TFTR)







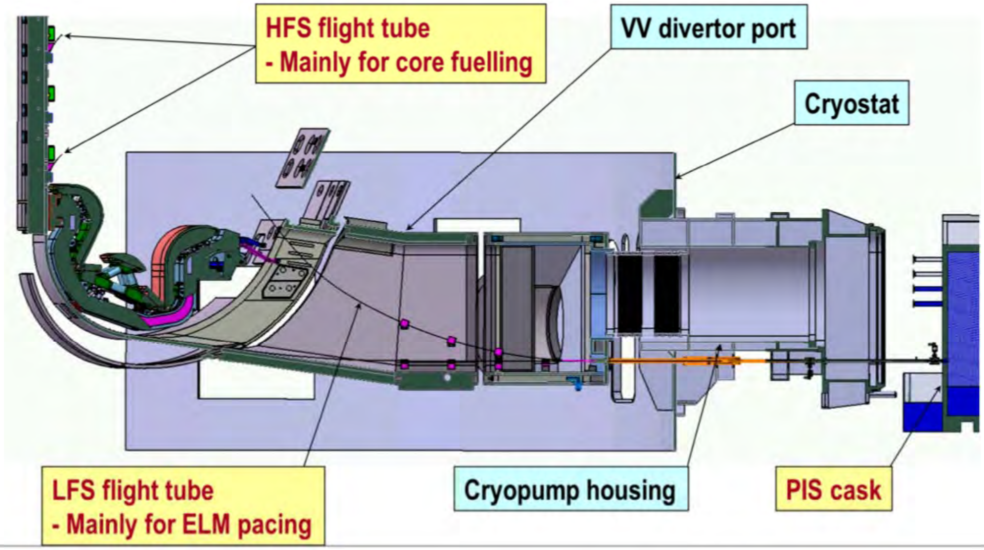
Diagnostic Integration: Upper Visible/IR Camera



Fuelling Systems

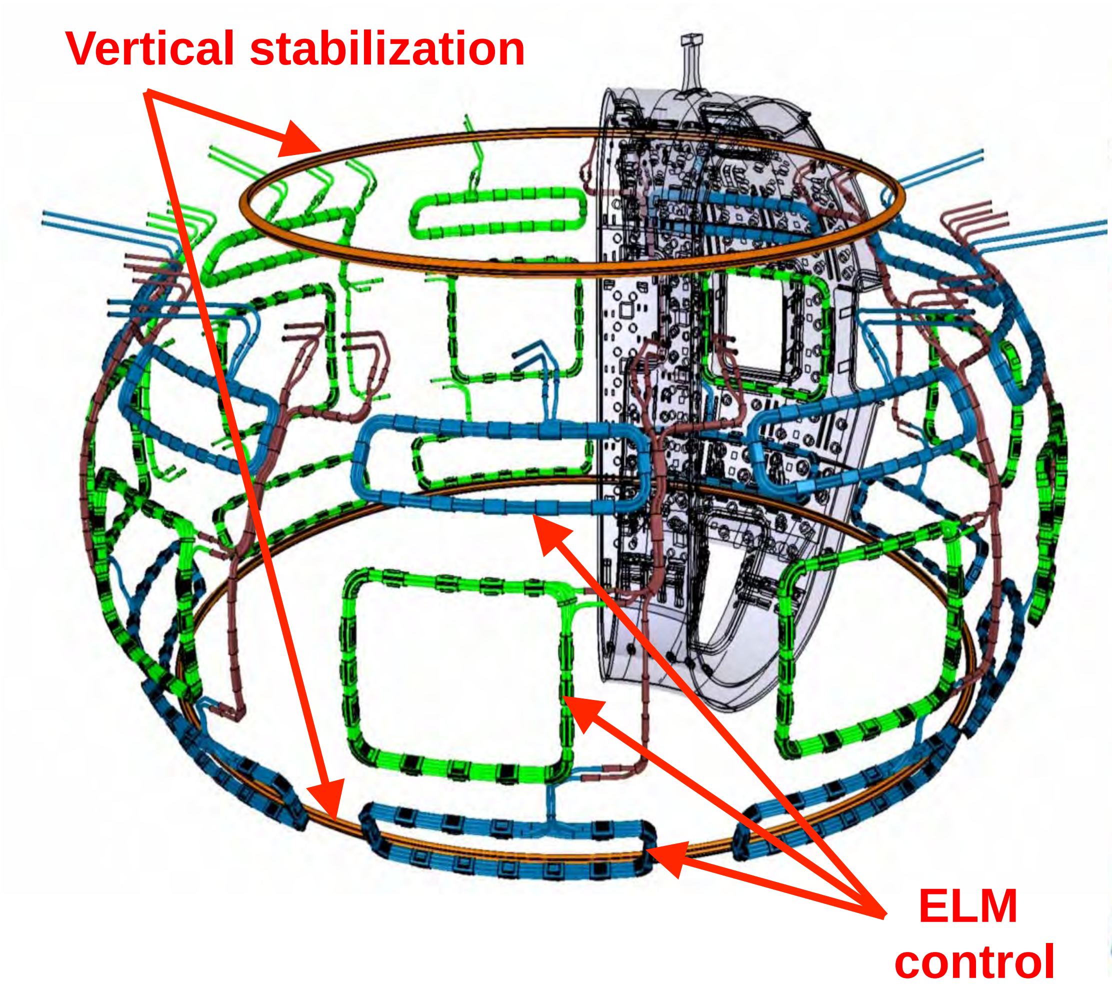
- **Plasma density is controlled by 3 active systems:**
  - Gas injection:
    - toroidally and poloidally distributed in torus
    - can inject hydrogenic and impurity gases
  - Pellet injection:
    - injects cryogenic pellets on hydrogenic isotopes at 10s of Hz and velocities ~ 1 km.s^{-1}
    - can inject from outboard and inboard side of torus
  - Divertor (cryo-)pumping:
    - in ITER, 6 cryo-pumps located in toroidally distributed ports
    - exhausts fuel and impurity gases and helium "ash" (activated charcoal)

ITER Pellet Injection layout showing outboard and inboard launch tubes

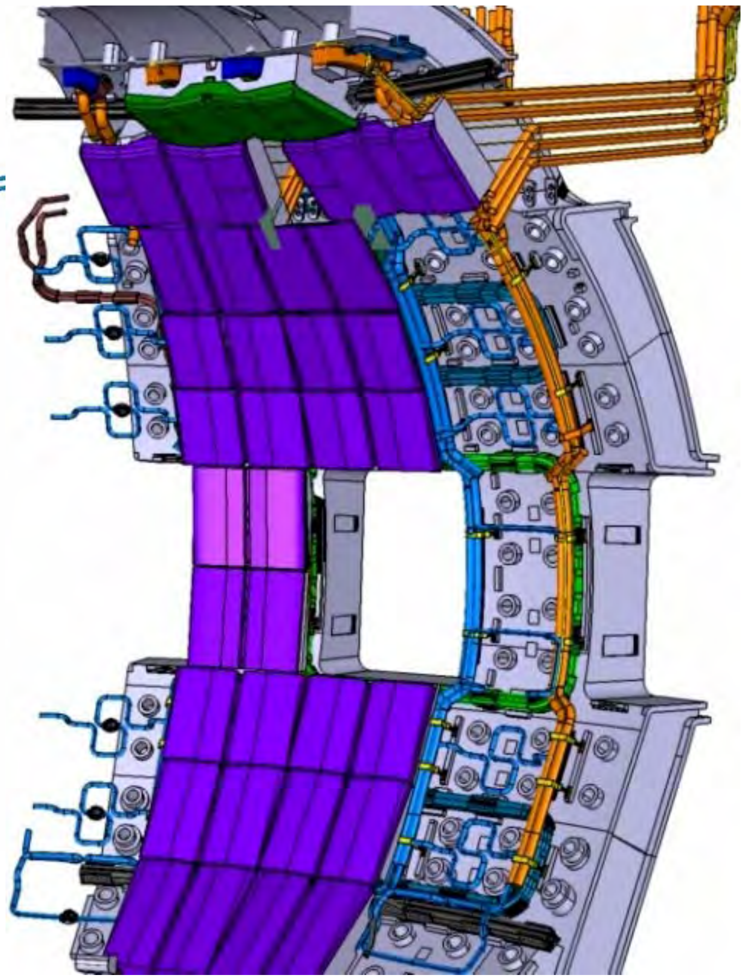


In-Vessel Coils: Vertical Stabilization/ ELM Control

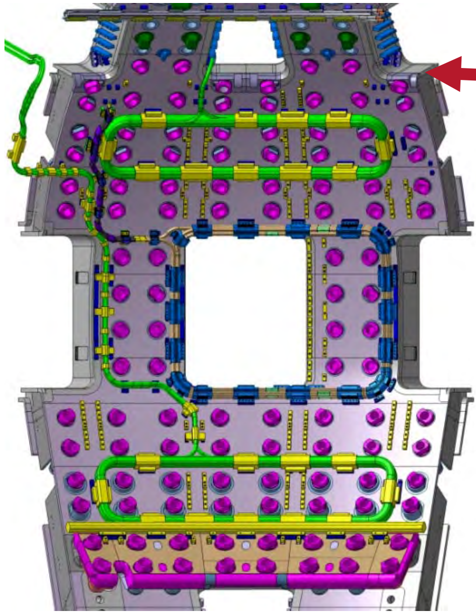
27 ELM control coils (n=4) – 3 per vacuum vessel sector (40°)

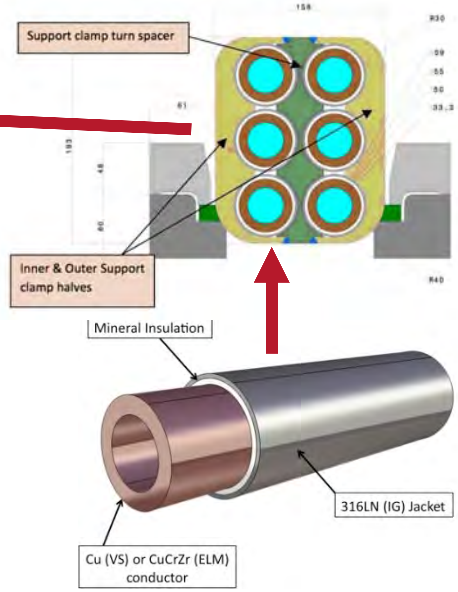


ELM Control Coils

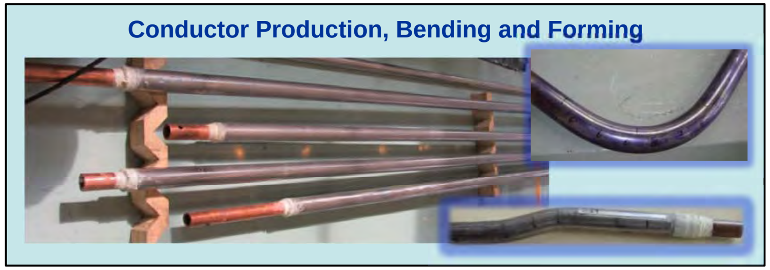


In-Vessel Coils





Conductor

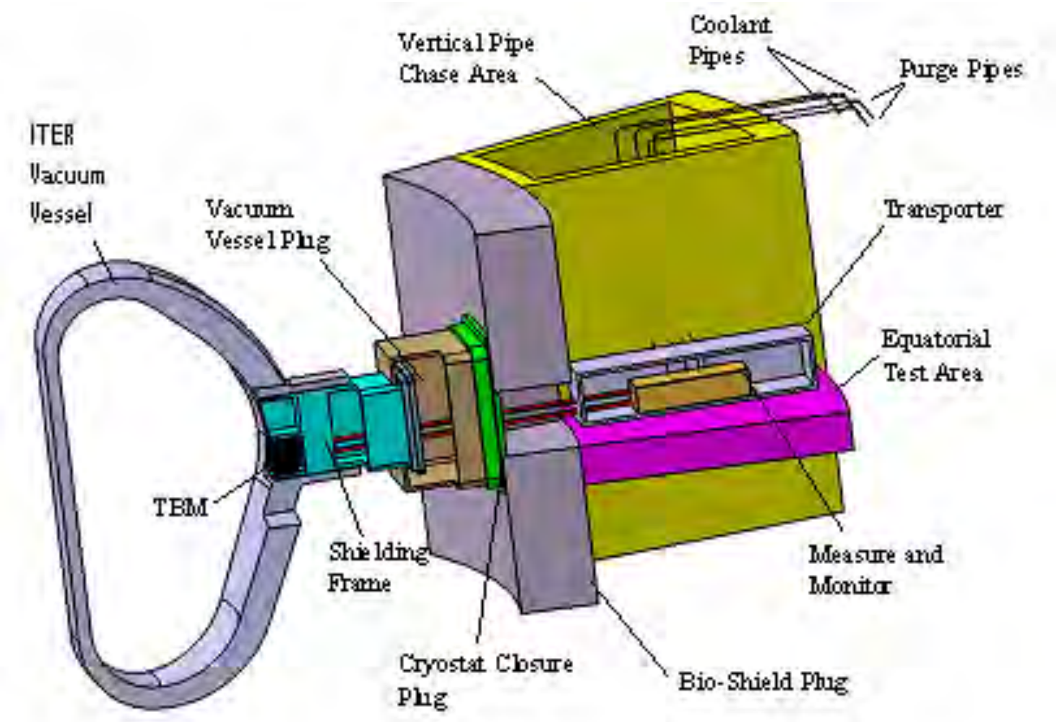


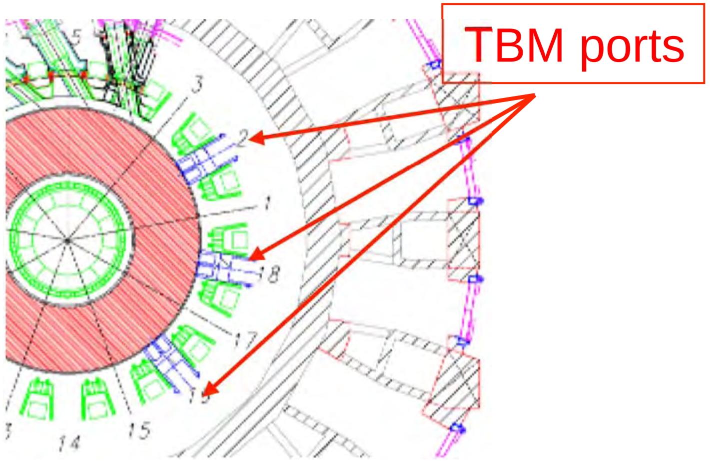
Technical Challenges:

- High currents in neutron environment (~60 kA @ 2.3 kV)
- Scale up of conductor (26 to 59 mm diameter)
- Remote handling
- Encouraging results from R&D programme:
  - prototype coils being tested
  - at present two alternative concepts being investigated

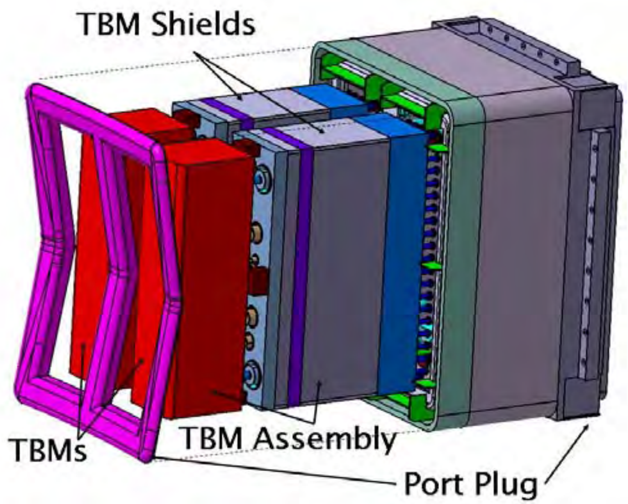
Test Blanket Modules - Tritium Breeding

$$n + {}^{6}Li \rightarrow T + {}^{4}He + 4.8\ MeV$$
$$n + {}^{7}Li \rightarrow T + {}^{4}He + n - 2.47\ MeV$$





TBM Port Plug (exploded view)



- **Three dedicated stations for testing up to six tritium breeding concepts**

Test Blanket System Testing in ITER

(TL = TBM Leader)

Port Number	First Concept	Second Concept
16	HCLL (TL: EU)	HCPB (TL: EU)
18	WCCB (TL: JA)	HCCR (TL: KO)
2	HCCB (TL: CN)	LLCB (TL: IN)

HCLL: Helium-cooled Lithium Lead; **HCPB:** He-cooled Pebble Beds
WCCB: Water-cooled Ceramic Breeder (+Be); **HCCR**: Helium-Cooled Ceramic Reflector
HCCB: He-cooled Ceramic Breeder (+Be); **LLCB**: Lithium-Lead Ceramic Breeder (He/LiPb)

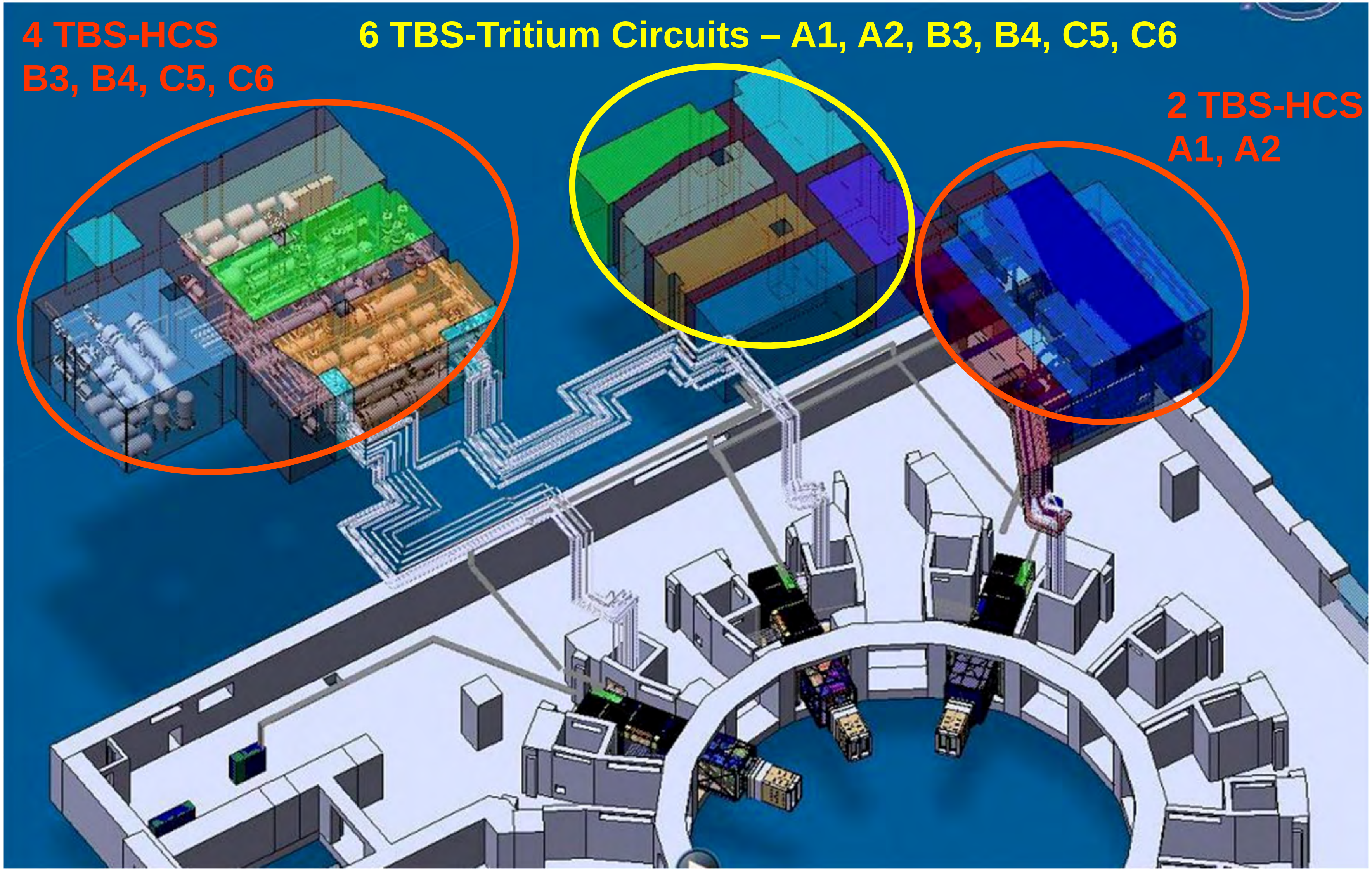
Demo-BB typically use a Reduced-Activation Ferritic/Martensitic steel ⇒ TBMs can correctly represent Demo-BBs only if they use the same structural material

These steels:

i. ☺ *Do not generate rad-waste with lifetime longer than 100 years*
   ⇒ important for future of D-T fusion power !!!

ii. ☹ *Are ferromagnetic ⇒ they perturb the magnetic field in the plasma.*
   To better understand this issue, experiments in DIII-D have been performed in 2009, 2011 and 2014

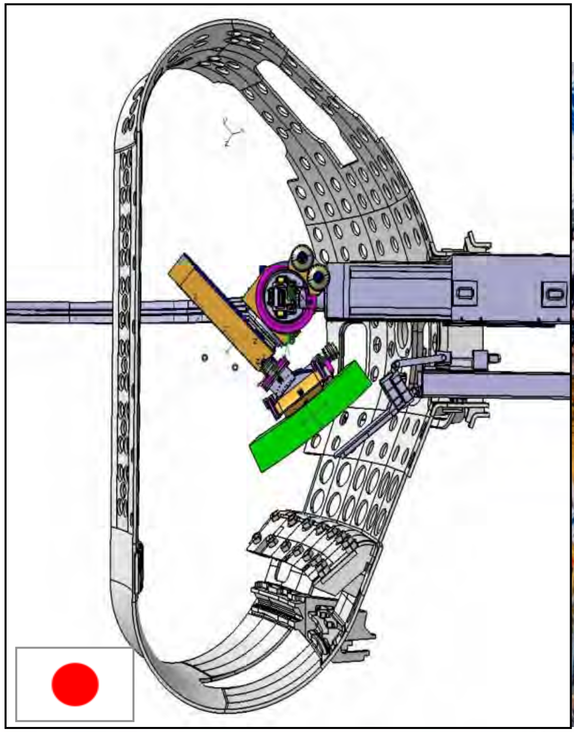
Overview of 6 TBM Systems

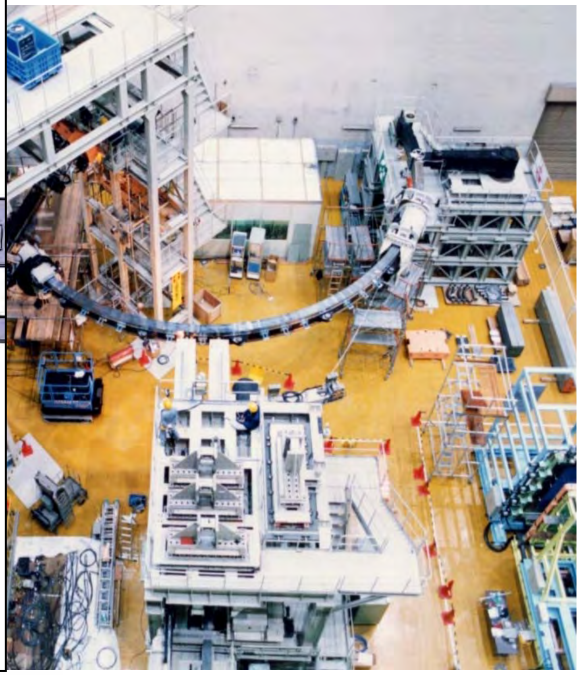


Remote Handling

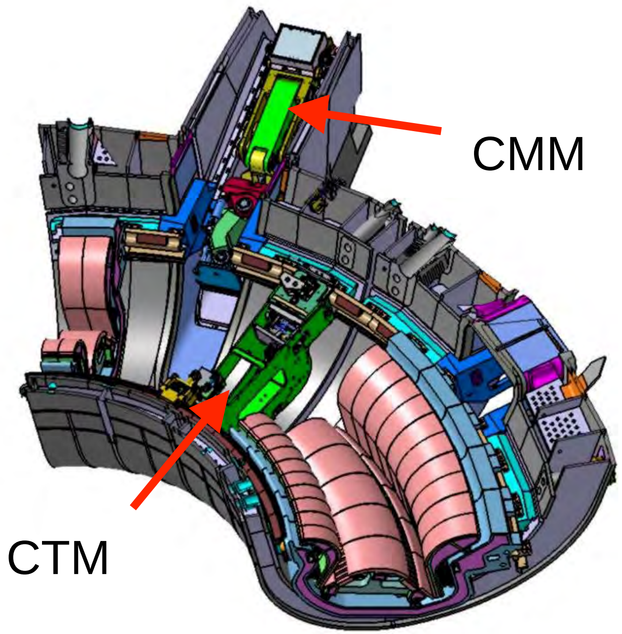
- **A major part of the ITER activity ⇒ extremely challenging to repair and replace complex and heavy components in a nuclear environment**
  - Dedicated, state-of-the art systems for both Blanket and Divertor
  - Divertor replacement 2-3 x in the machine lifetime (~6 months to exchange)
  - First wall panels replaced at least once

Blanket RH procured by JA





Divertor RH procured by EU



Conclusions

The ITER device integrated many advanced technologies and is driving major technology R&D programmes within the partners

- **Since the establishment of the ITER Baseline in July 2010, the ITER project has moved fully into the Construction Phase**
  - on-site construction of the Tokamak Complex is underway
  - Domestic Agencies have launched large scale manufacturing contracts for many major components
  - extensive prototyping is ongoing in preparation for series manufacture
- **Substantial progress in design and R&D for In-Vessel Components, Plasma Auxiliary Systems, Remote Handling etc**
  - many Procurement Arrangements have already been signed in these areas

Successful exploitation of ITER will not only realize the limitless possibilities of fusion energy, but open new areas of fusion plasma research and fusion technology, including tritium breeding