FIP/2-1Ra

Progress of Qualification Testing for Full-Scale Plasma-Facing Unit Prototype of Full Tungsten ITER Divertor in Japan

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FIP/2-1Rb

Progresses on WEST Platform Construction towards First Plasmas

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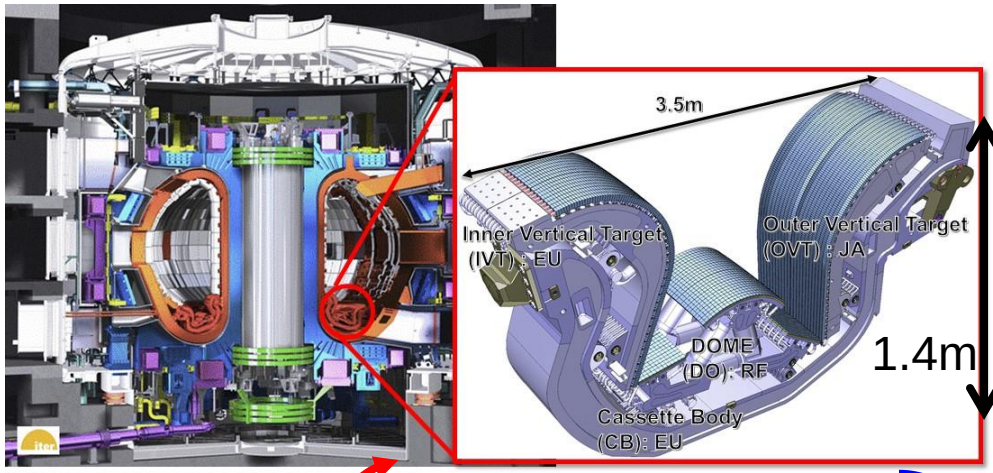
26th Fusion Energy Conference (FEC 2016)
Kyoto, Japan, 19 October 2016

Outline

FIP/2-1Ra (ITER divertor OVT)

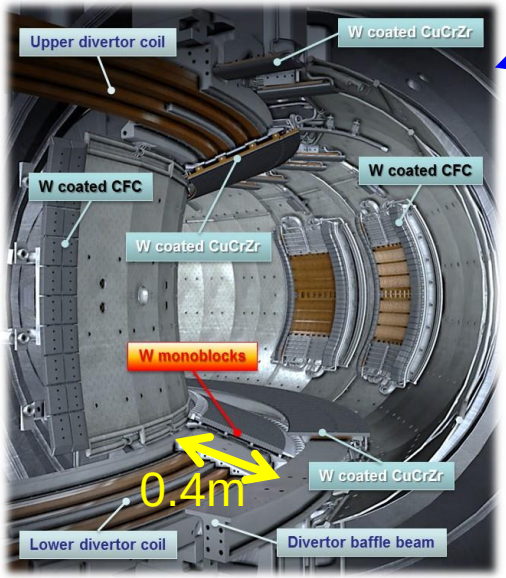
- Status of full-W ITER divertor outer vertical target
- Manufacturing full-W full-scale Plasma-Facing Unit (PFU) prototypes
- Profile tolerance of surface on PFUs
- Durability for HHF and deformation W monoblock
- Summary



Mutual-aid relationship

FIP/2-1Rb (WEST project)

- WEST Tokamak Features
- WEST Program: risk mitigation for ITER full W divertor
- WEST divertor PFU vs ITER divertor VT PFU
- WEST main component status
- Load assembly and experiment schedule
- Summary



Status of full-W ITER divertor Outer vertical target

Three challenges in Full-scale PFU prototype

(1) Soundness of all joint surfaces in PFU:
To be passed the ultrasonic testing (UT) and

(2) Strict profile tolerance of surface in target part:
To be stayed within ±0.25 mm.

(3) Durability for high heat flux (HHF) testing:
Requirement of the repeated heat load of 10 MW/m^2 × 5000 cycles and 20 MW/m^2 × 300 cycles.

Next Phase

ITER Divertor OVT
2nd Prototype and
Mass production

Completion
2014-2015

1st Full-scale
PFU prototype

Element test
for Material
2007-2011

W monoblcok Casted
Cu interlayer

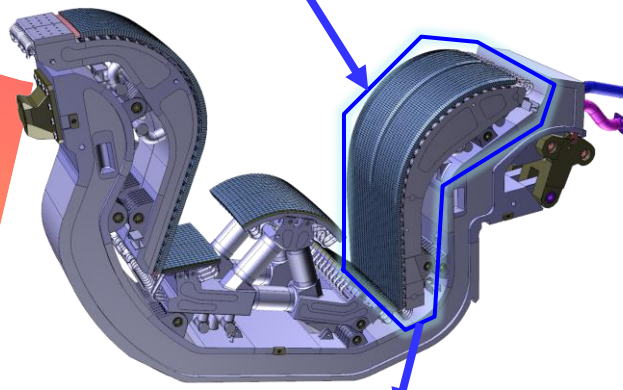
CuCrZr-pipe

Small-scale mock-up
2012-2013

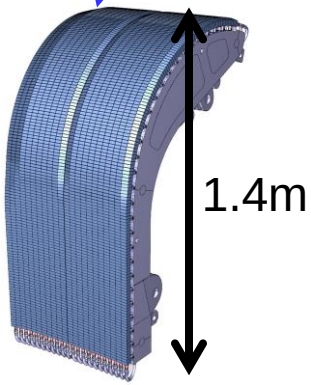
Establishment of
Brazing
30 W monoblocks

Crack-free after HHF testing

584 W monoblocks
Demonstration of
manufacturing feasibility



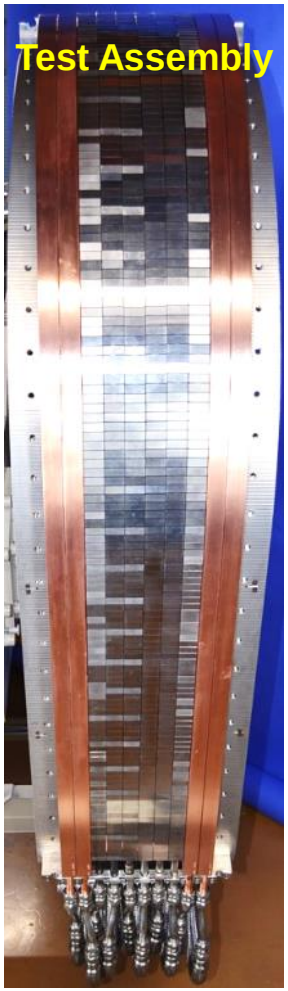
Profile Tolerance

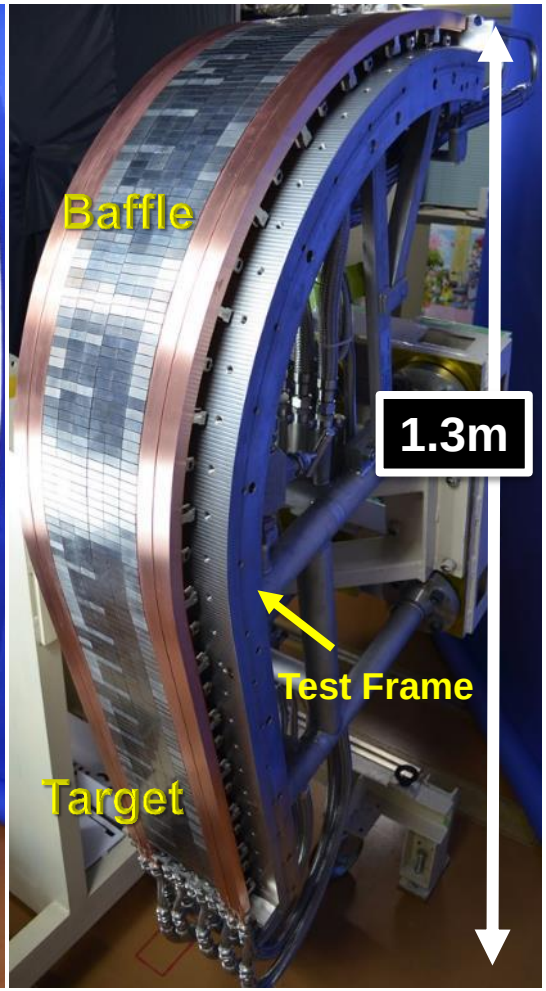


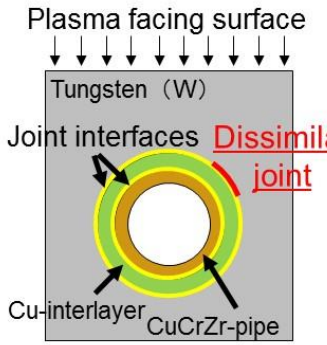
1.4m

200K W monoblocks
in 58 OVT

Manufacturing full-scale PFU prototypes





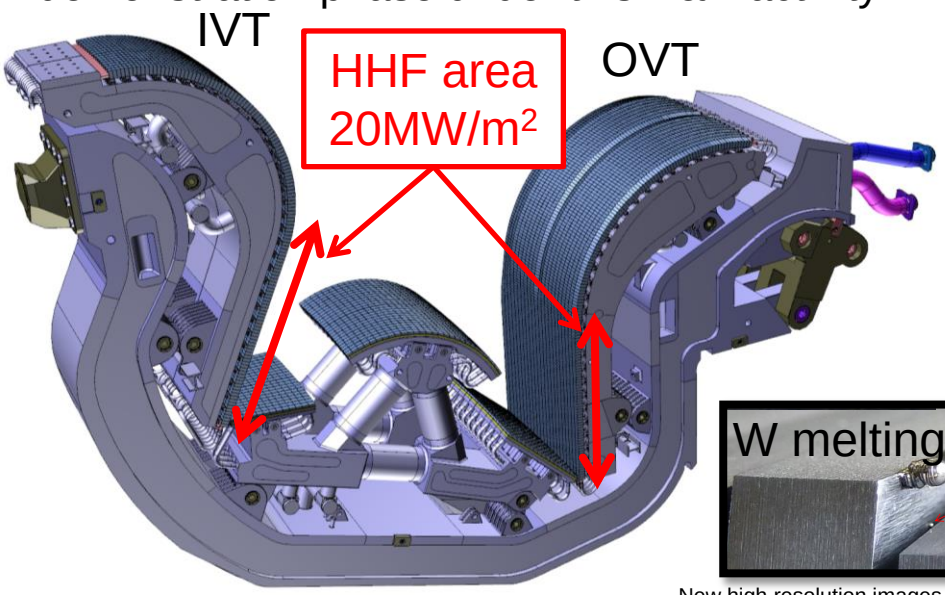


W monoblock for the target part of the prototypes was not chamfered in the toroidal direction for the subsequent high heat flux testing

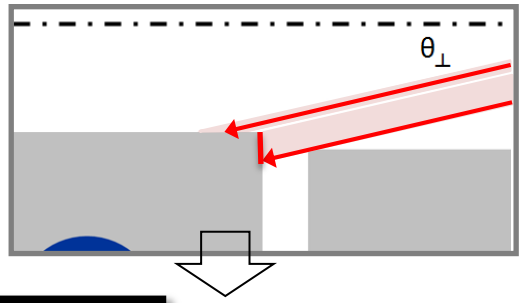
The results of UT for 4 PFU prototypes with W monoblocks and the casted Cu interlayers showed no disssimilar joint which degrades the heat removal capability in both joints of W/Cu and Cu/CuCrZr-IG among 584 W monoblocks (= 146 monoblocks/PFU × 4 PFUs).

Profile tolerance of surface on PFUs

Strict profile tolerance of the surface at the target part **is one of the challenges** in the full-scale demonstration phase under this R&D activity.



Problem : Step of W monoblock causes increasing heat flux at leading edges.

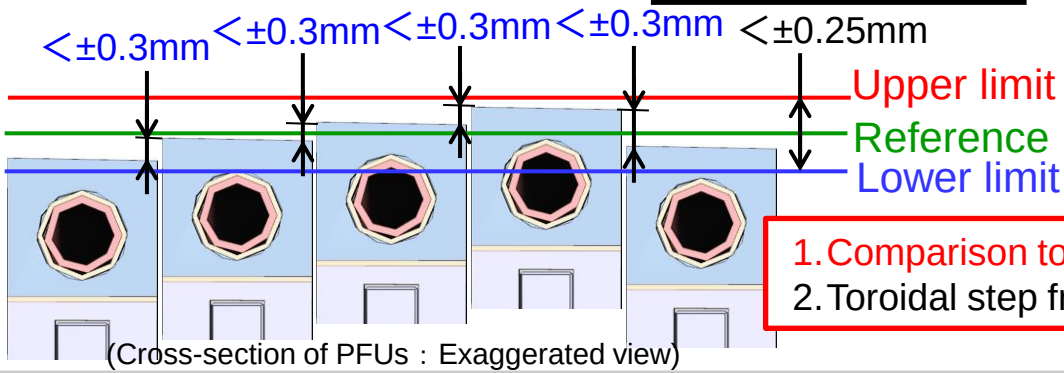


New high resolution images of W-melting in JET
[G. Matthews – PFMC 2015]

Two measures are taken for all W monoblocks to prevent the leading edges of the W monoblocks from over-heating.

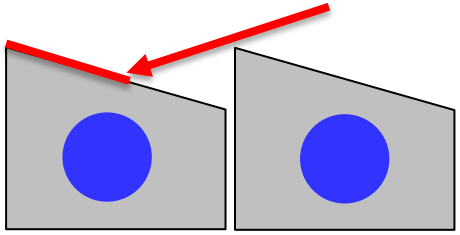
Measure 1 : Strict tolerance

Max. Toroidal Step **Profile tolerance**



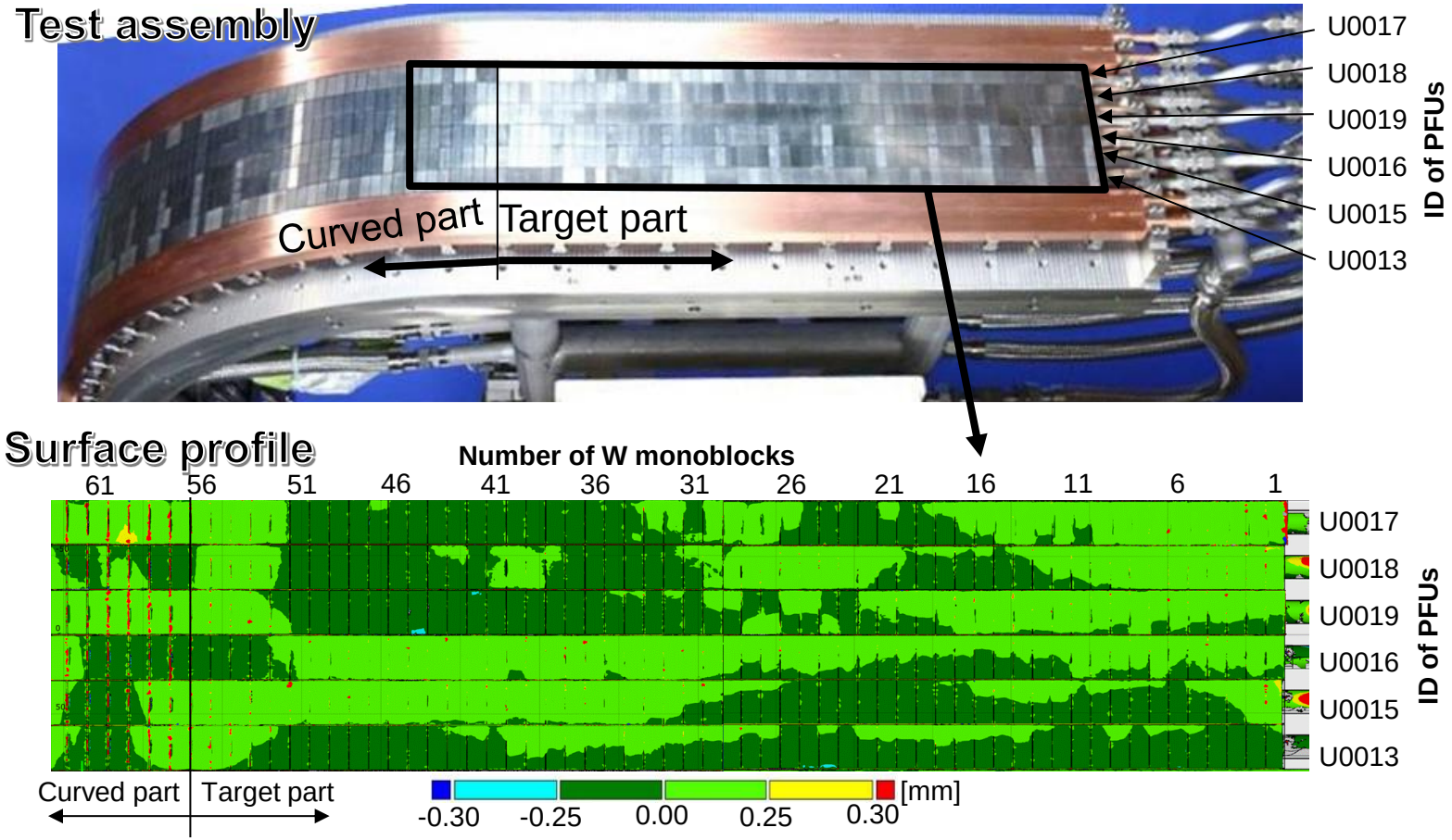
(Cross-section of PFUs : Exaggerated view)

Measure 2 : Chamfered W monoblock



1. Comparison to CAD model data : surface tolerance, ±0.25mm
2. Toroidal step from each neighbouring W monoblock, ±0.3 mm

3D measurement on surface profile of the 6 PFU prototypes



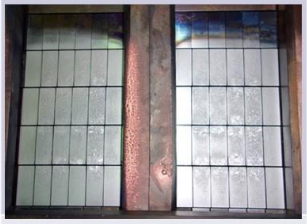
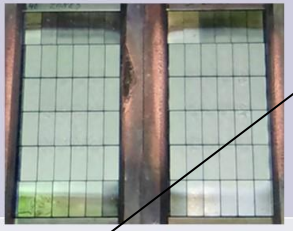
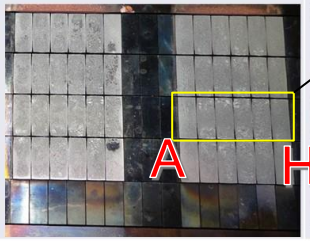
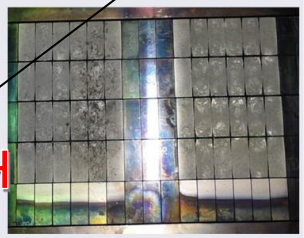
Optical measuring instrument (combination of ATOS Compact Scan 5M with TRITOP of Optical 3D Coordinate Measuring Machine)

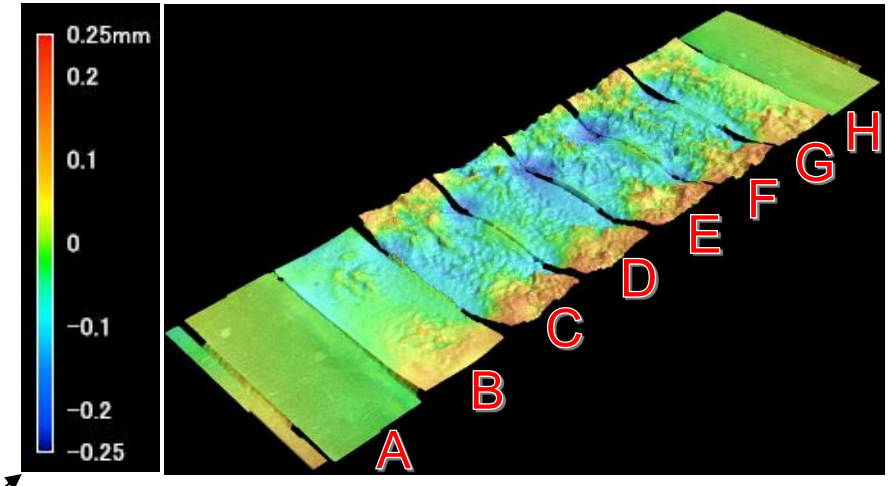
W monoblock surface in the target part was located within ± 0.25 mm from the CAD model data. JADA succeeded in demonstrating feasibility of the requirement of the surface profile with tight tolerance.

Durability for HHF and deformation W monoblock

Definition of zones heated in test assembly

Surface condition

Zone	2	3 and 4	5 and 6
After 300 cycles at 20MW/m^2			
After 1000 cycles At 20MW/m^2			



Deformation of W surface after HHF testing is appeared within ±0.25mm.

The HHF testing for four full-scale PFU prototypes was carried out at the Efremov Institute.
All of the tested 116 W monoblocks endured the repetitive heat load of 10 MW/m^2 × 5000 cycles and 20 MW/m^2 × 1000 cycles which is more than three times higher than the requirement of 300 cycles.
Any W monoblock did NOT show macroscopic cracks along the coolant tube axis after HHF testing.

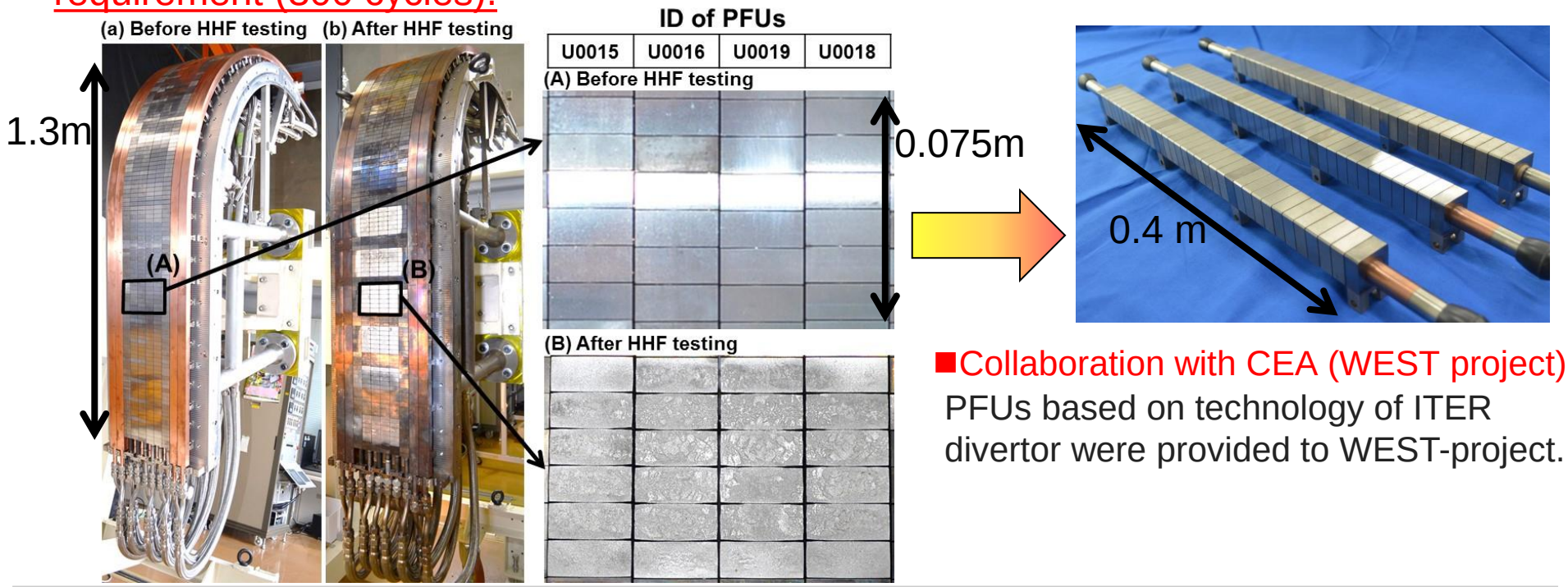
Summary of FIP/2-1Ra

Through full-scale prototyping, JADA demonstrated the manufacturing ability of the full-tungsten plasma facing unit.

■All joint surfaces in four PFUs with a casting Cu interlayer successfully passed ultrasonic testing.

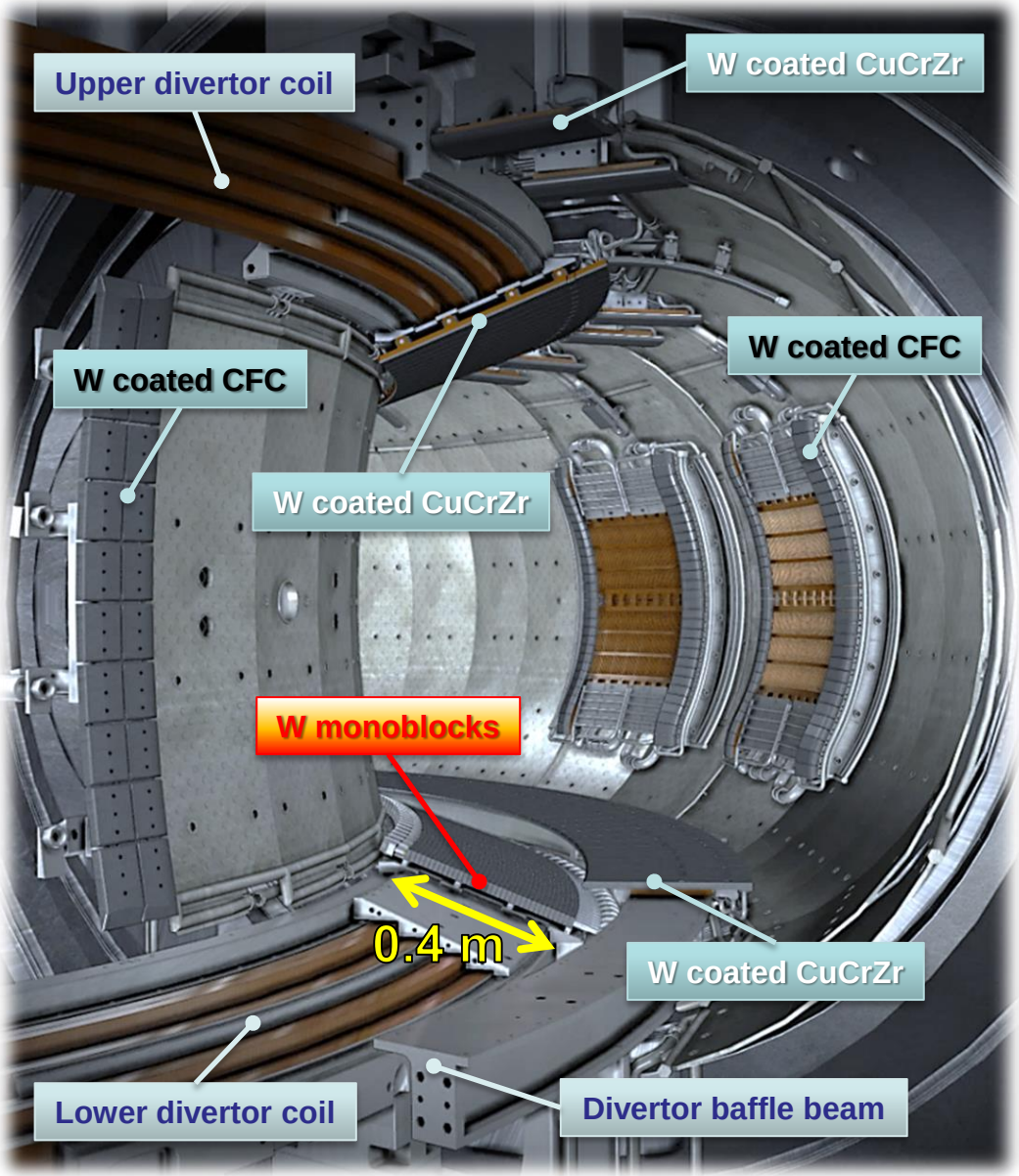
■Surface profile in the target part stayed within the required profile tolerance of ± 0.25 mm.

■Full-scale prototype withstood the repetitive heat load of 10 MW/m^2 × 5000 cycles and 20 MW/m^2 × 1000 cycles which is more than three times higher than the requirement (300 cycles).



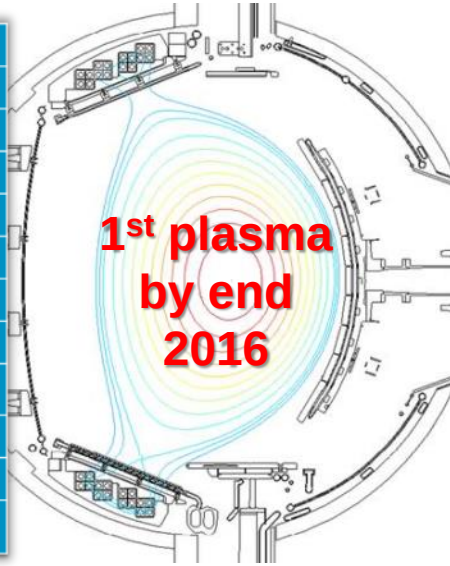
■Collaboration with CEA (WEST project)
PFUs based on technology of ITER divertor were provided to WEST-project.

WEST Tokamak Features



- Tungsten environment
- Actively cooled PFC → high heat exhaust capability
- Flexible divertor conf. (LSN, DN, USN) → heat load control
- Large aspect ratio and high shaping
- RF heating → no external momentum input
- Long pulse capability → 1000s

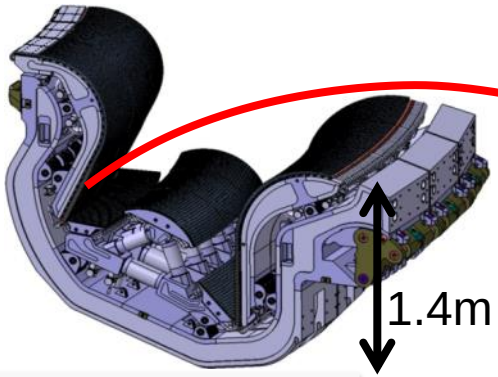
I_p (q_{95}~2.5)	1 MA
B_T	3.7 T
R	2.5 m
a	0.5 m
A	5-6
κ	1.3-1.8
δ	0.5-0.6
V_p	15 m^3
n_{GW} (1MA)	1.5 $10^{20} m^{-3}$
P_{ICRH}	9 MW
P_{LHCD}	7 MW
$T_{flattop}$ (0.8 MA)	1000 s



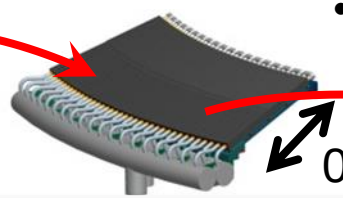
WEST Program: risk mitigation for ITER full W divertor

WEST in support to ITER divertor procurement and operation

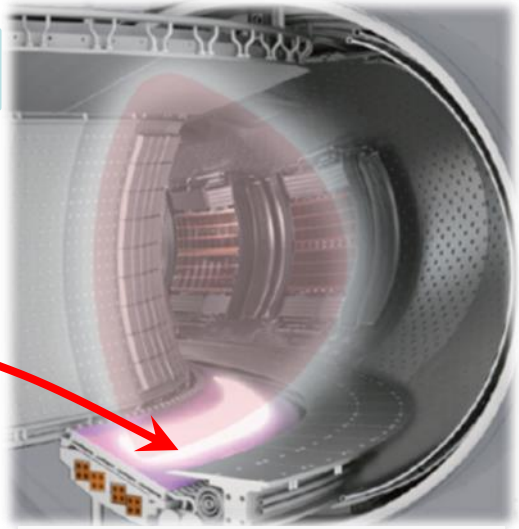
- PFC technology validation
- PFC lifetime



ITER divertor cassette



WEST divertor sector (30°)



Tokamak plasma exposure

Large scale collaboration between ITER partners

- ITER Organization and Domestic Agencies: F4E and JADA
- ITER partners: China, India, Korea, Japan, USA and Europe

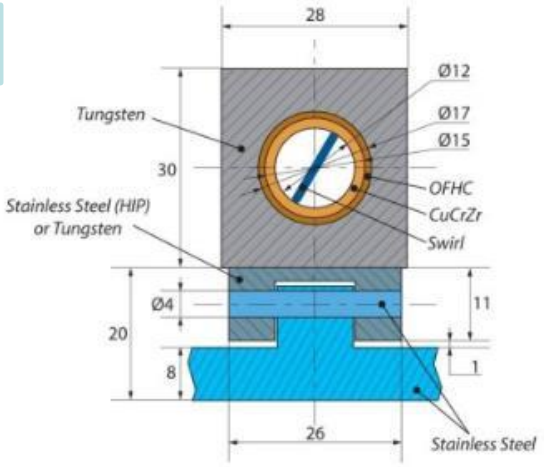
Scientific program build with the input of the international fusion community (cf. WEST Research Plan) and steered by a Governing Board (WEST partners)

- Phase 1 (2016-2018): prototypes testing* / short pulses / power handling issues
- Phase 2 (2019-2020's): large scale testing / long pulses / high particle fluences

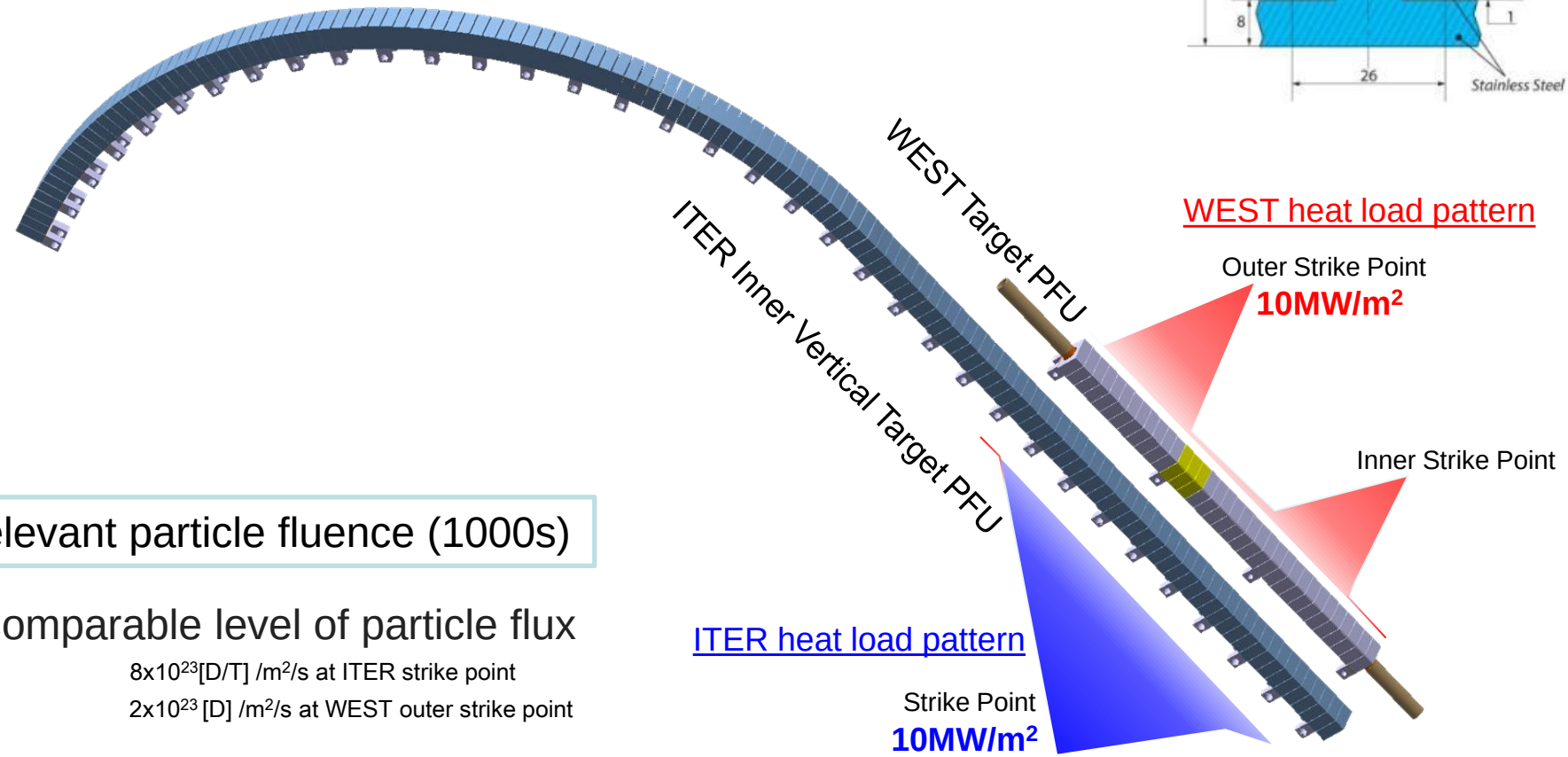
**complemented by inertial tungsten coated graphite elements*

WEST divertor PFU vs ITER divertor VT PFU

Similar geometry and technology (straight part of ITER VT)



Relevant steady-state heat load (10-20 MW/m^2)



Relevant particle fluence (1000s)

Comparable level of particle flux

8x10^{23}[D/T] /m^2/s at ITER strike point

2x10^{23} [D] /m^2/s at WEST outer strike point

WEST main component status

In-vessel divertor coils

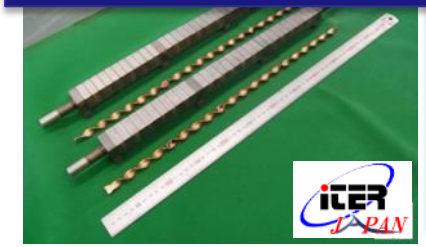
Installation completed

Divertor power supplies

Commissioning started

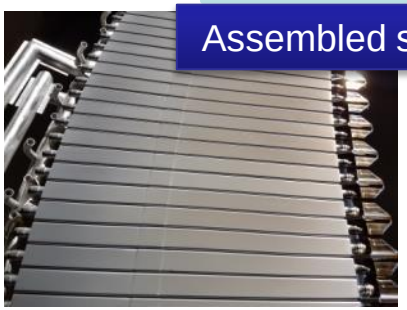
ITER-like target prototypes

Pre-characterization completed



Tungsten coated PFC

Assembled sectors installation ongoing

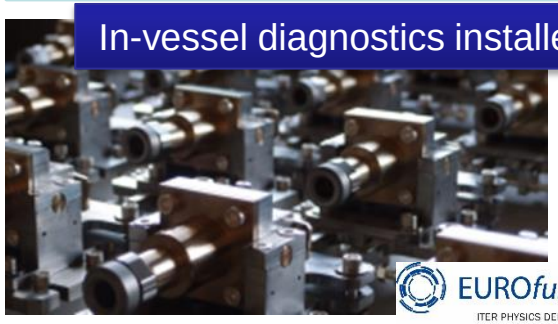


ELM-resilient ICRH antennas

1st antenna assembly ongoing

Key diagnostics

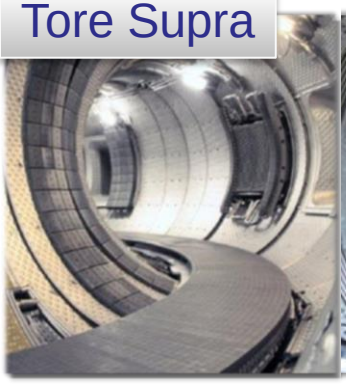
In-vessel diagnostics installed

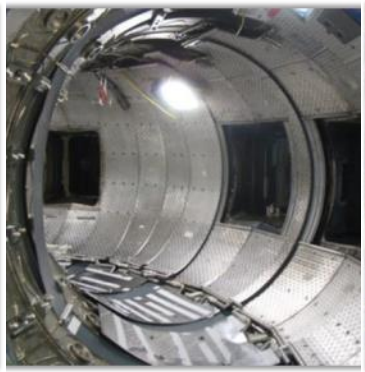


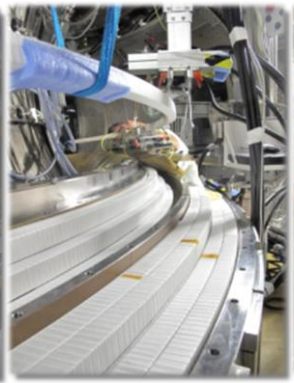
EUROfusion
ITER PHYSICS DEPARTMENT

Load assembly and experiment schedule

Tore Supra








WEST

2013 — 2014 — 2015 — 2016 → Sept. 2016

Major milestones achieved!

- Divertor coil SS thick casings assembled inside the vacuum vessel
- Divertor coil windings completed (>150 in-situ brazing joints)
- Vacuum vessel protection panels installed (outer and inner wall)

Last load assembly sequence ongoing with PFC sectors installation

Vessel evacuation and coil impregnation scheduled in Nov.

December 2016

1st Plasma

Experimental timeline for WEST phase 1

2017 C1 – Heat loads / L-H transition → C2 – PFC testing / H mode characterization → 2018 C3 – Preparation for long pulse → C4 – He campaign

Summary of FIP/2-1Rb

WEST to support ITER divertor strategy

- Diverted tokamak with W environment and long pulse capability (1000 s)
- WEST divertor target representative of ITER divertor target
- Risk mitigation for ITER tungsten divertor procurement and operation

WEST to start plasma operations soon

- Major components delivered and installed or being installed
- Last load assembly phase ongoing
- Vessel evacuation early November
- Plasma operations foreseen in December

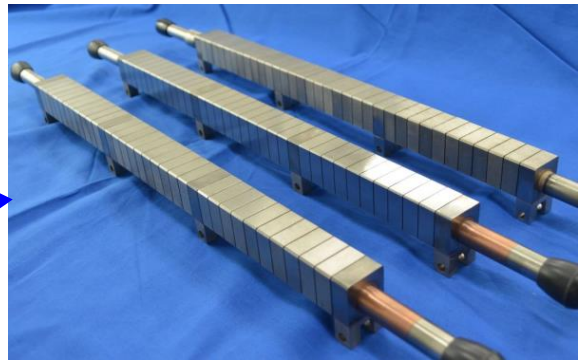
WEST first experimental campaigns

- Staged approach for WEST exploitation
- First phase with a mix of ITER-like and inertial Plasma Facing Units (PFU)
- ITER-like PFU prototypes in characterization phase
- First prototypes will be exposed during the first campaign to start in February

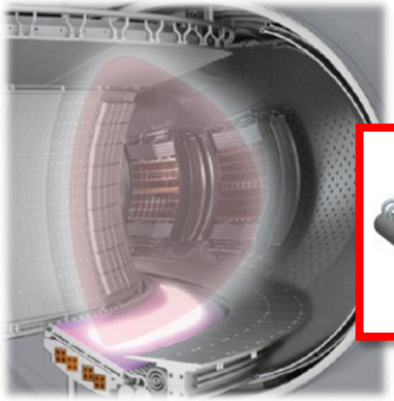
Summary of FIP/2-1

Full-W divertor is common challenge for the international community.
Mutual-aid relationship is on time for each schedule.

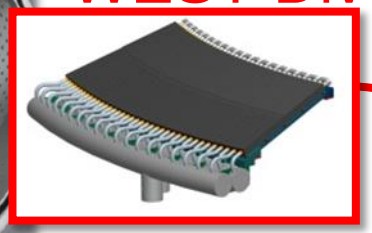
Experiment in Plasma



WEST-CEA



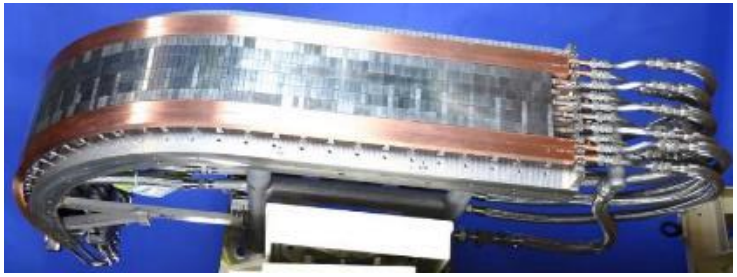
WEST-Divertor



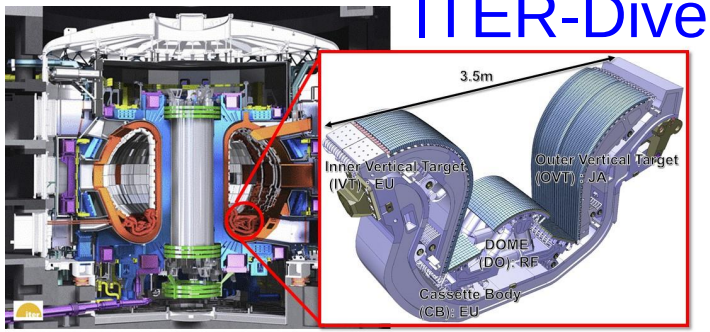
Supply of PFU

PFC technology validation & PFC life time

Manufacturing Technology



ITER, JADA



ITER-Divertor

Thank you