Overview of Fuel Inventory in JET with the ITER-Like Wall

EUROfusion

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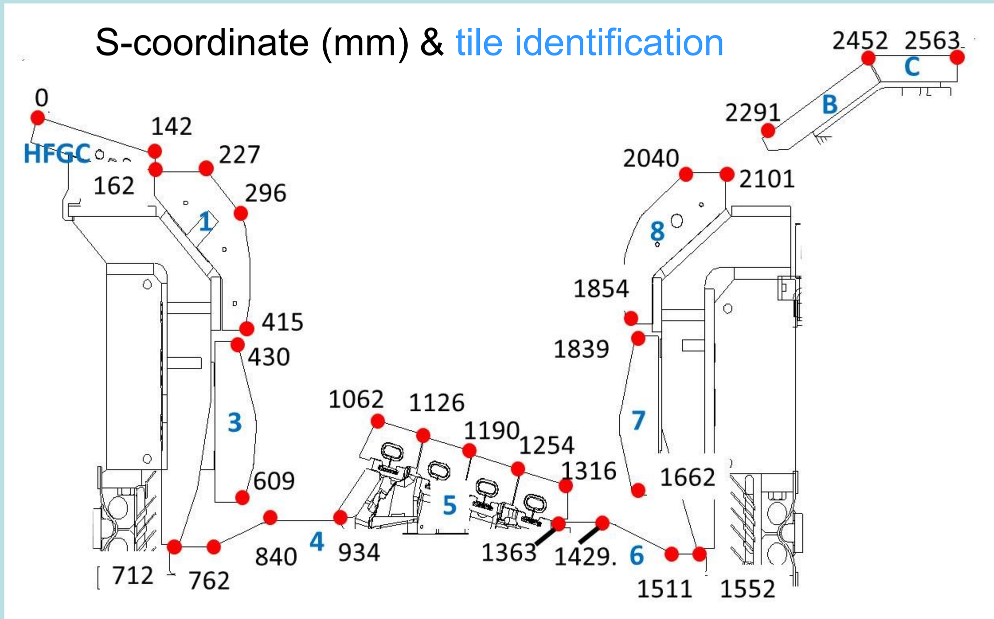
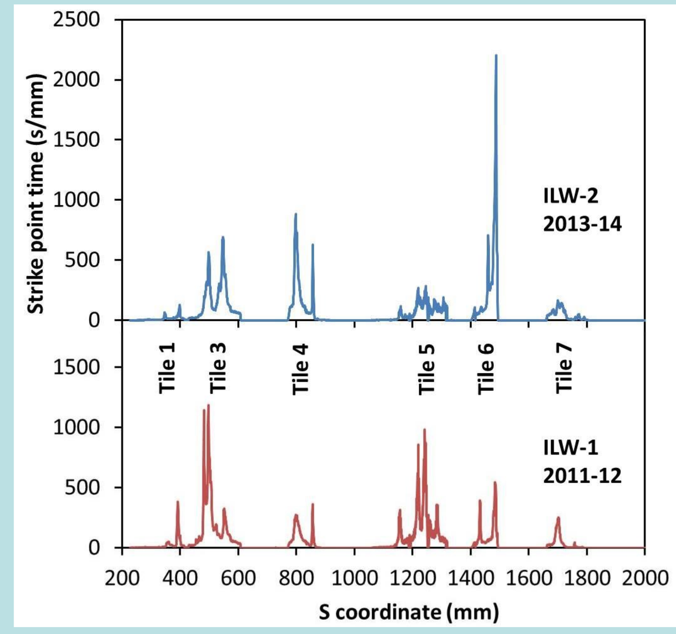
- An overview of fuel retention in the JET ITER-like wall configuration (JET-ILW) drawing on a range of analysis techniques is presented.
- Two experimental campaigns have now been completed with JET-ILW; 2011-2012 (ILW-1) and 2013-2014 (ILW-2).
- Post mortem analysis is completed on components removed from JET-ILW after each campaign; ion beam analysis (IBA), secondary mass spectrometry (SIMS), scanning electron microscopy & electron dispersive spectroscopy, surface profiling, thermal desorption spectroscopy, optical microscopy, photography.
- Results from the analysis of these components provide direct measurement of material erosion, re-deposition, fuel retention, surface and particulate morphology.
- Examples below illustrate key fuel retention and erosion/deposition results for JET-ILW.

JET CAMPAIGN INFORMATION

	ILW-1: 2011-2012	ILW-2: 2013-2014
Limiter phase	6 hours	6 hours
Divertor phase	13 hours	14 hours
Hydrogen campaign	None	10% pulses at end of campaign (~300 pulses)
Inner strike point (ISP)	Predominantly Tile 3	Tiles 3 & 4
Outer strike point (OSP)	Predominantly Tile 5	Predominantly Tile 6





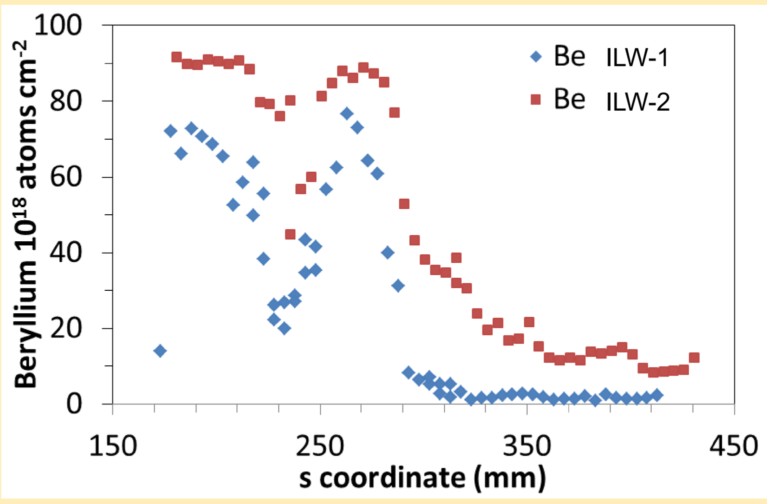
UPPER INNER DIVERTOR: HFGC & TILE 1

⇒ Global fuel retention dominated by co-deposition.

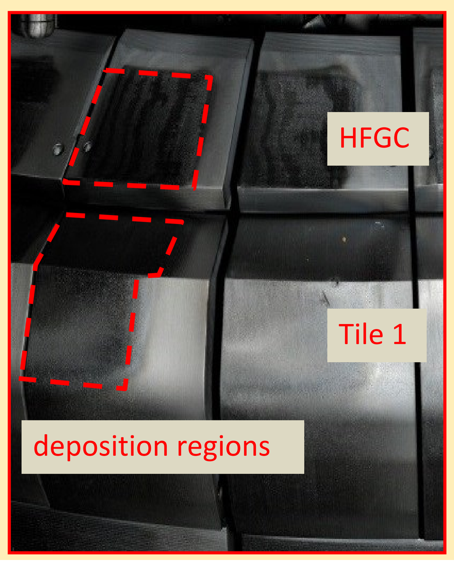
- HFGC and Tile 1 are regions of highest beryllium deposition and highest fuel retention by co-deposition.
- Following ILW-1 1/3 of global fuel retention was on HFGC and Tile 1 surfaces.
- HFGC and Tile 1 in Scrape Off Layer (SOL) due to inner strike point on Tile 3 & Tile 4 resulting in deposition.

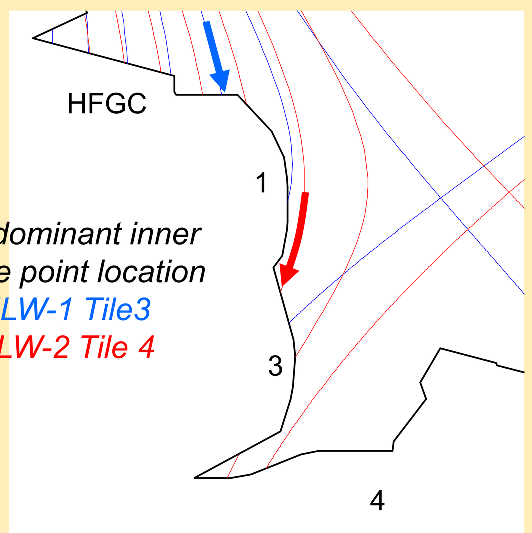
⇒Global fuel retention has decreased with JET-ILW

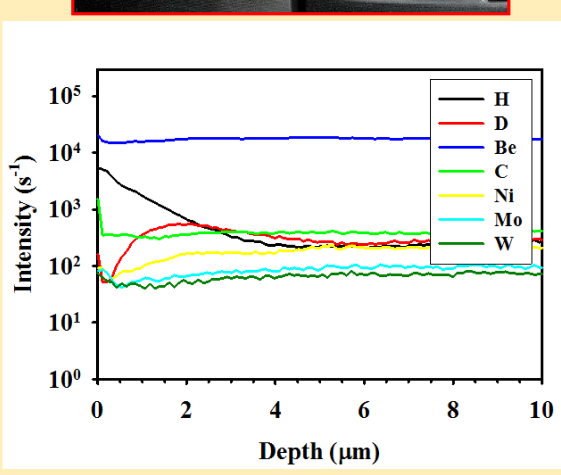
- Deposition in JET-ILW is at least an order of magnitude lower than for the JET carbon wall.



More deposition on Tile 1 following ILW-2 due to inner strike point location on Tile 4





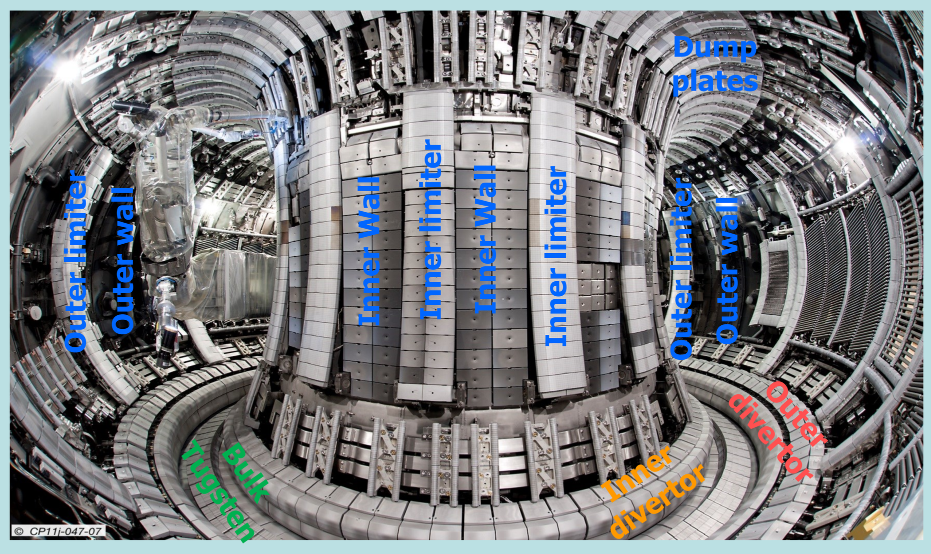


- **ILW-2 campaign ended with 300 H_2 pulses**
- **Evaluation of fuel retention complicated by H campaign**
- **H increased decreased D concentration at the surface**

Increased H concentration at surface of HFGC. Results from SIMS.

FUEL INVENTORY FOR JET-ILW DIVERTOR AND MAIN CHAMBER SURFACES FOLLOWING ILW-1

Divertor Tungsten	Inventory (10^{22} D atoms)	Main chamber Beryllium	Inventory (10^{22} D atoms)
Plasma facing surfaces			
Inner divertor*	17	Inner limiters*	1.4
Outer divertor*	3.9	Outer limiters*	5.2
Bulk tungsten†	0.3	Dump plate*	2.1
Recessed/remote surfaces and gaps			
Inner corner*	2.0	Inner wall*	2.8
Outer corner*	2.2	Outer wall*	0.9
		Castellation gaps‡	1.0



*Heinola et al., *Phys. Scripta* 2016 T167 014075, ‡ Rubel, IAEA-FEC 2016 proceedings, † results shown below

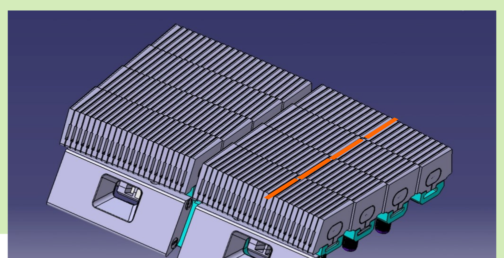
BULK TUNGSTEN†: TILE 5

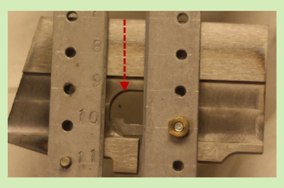
⇒ Bulk tungsten surface contributes an order of magnitude lower fuel retention.

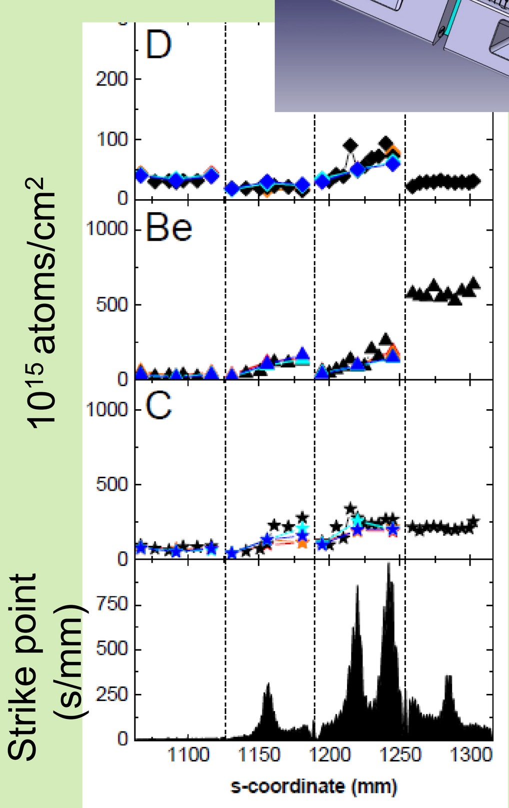
- Fuel retention governed by surface temperature, either due to outer strike point location or shadowing

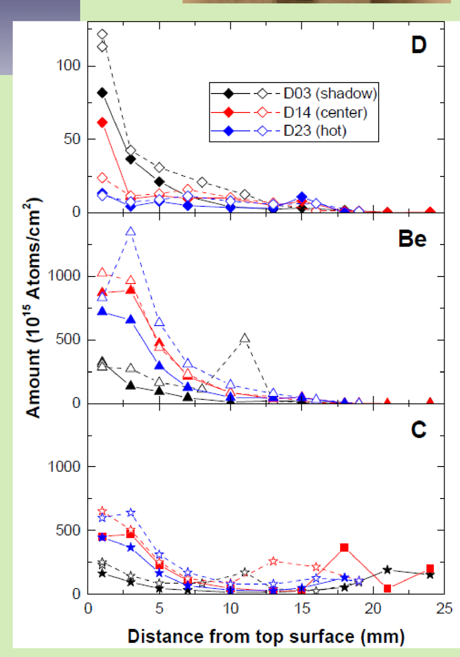
⇒Lamellae gaps contribute <1% to global fuel retention

ILW-1: Deposition on plasma facing surface of tungsten lamellae. Results from IBA.









ILW-1: Deposition in gaps between tungsten lamellae <1 mm. Results from IBA.

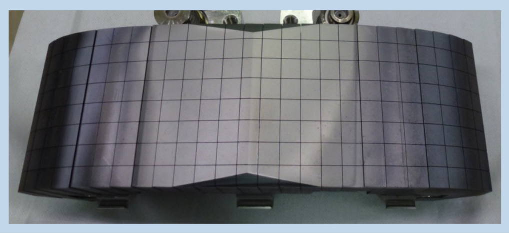
MAIN CHAMBER: INNER LIMITER TILE

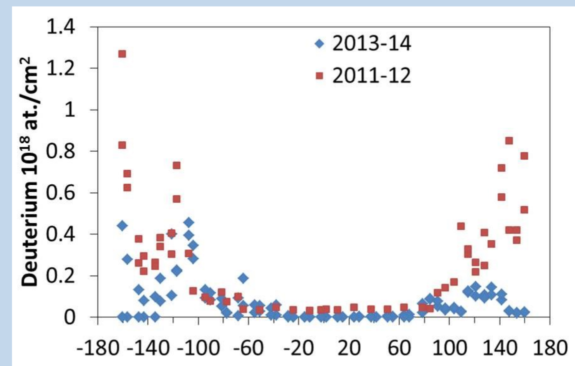
⇒ Fuel retention in plasma facing limiter tiles ~22% of global inventory

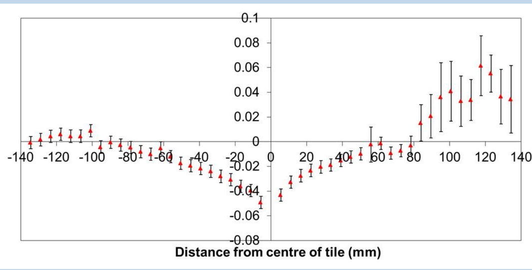
- Erosion at inner limiter during limiter phase
- Fuel retention dominated by local co-deposition at ends of limiter tiles

⇒ Limiter gaps contribute 2.5% to global fuel retention ‡

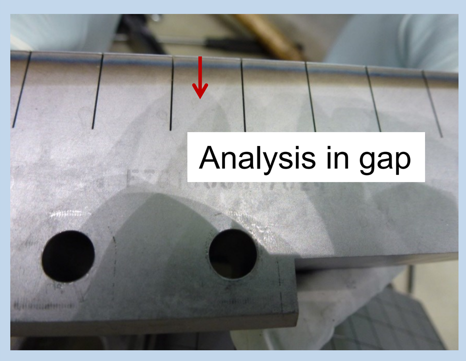
⇒Keep gaps narrow to reduce fuel retention

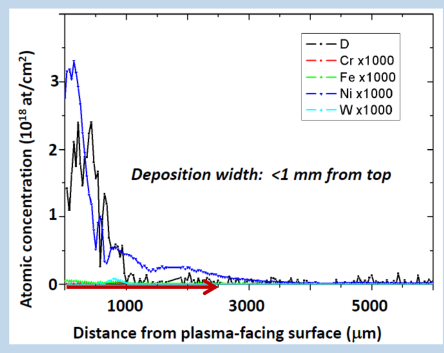






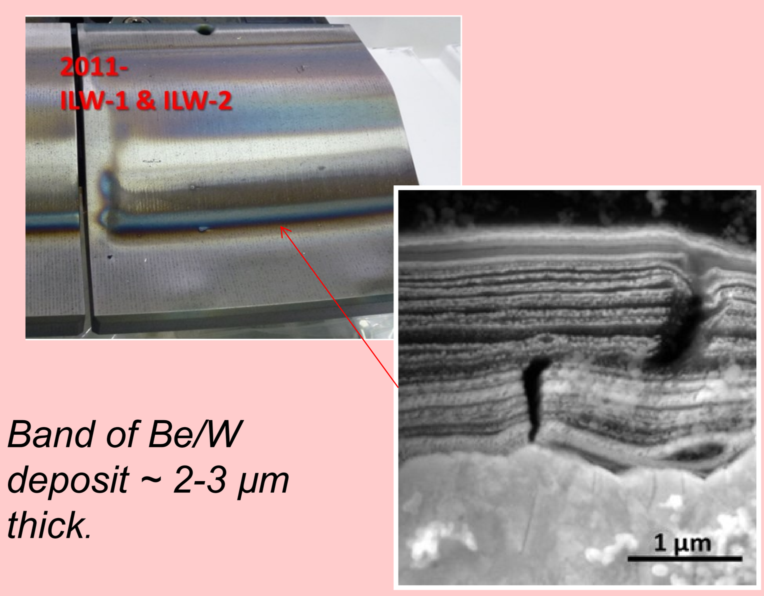
Fuel retention and erosion/deposition pattern for mid-plane inner limiter tile. Results from IBA and surface profiling.



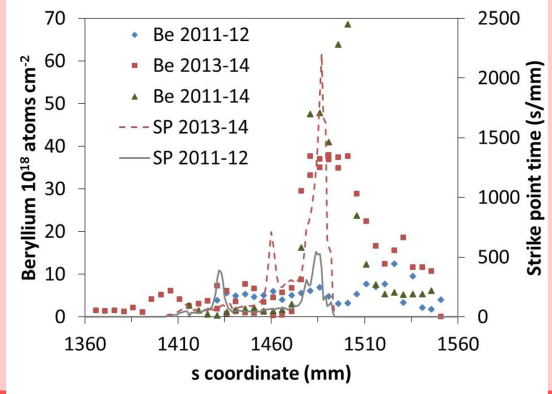


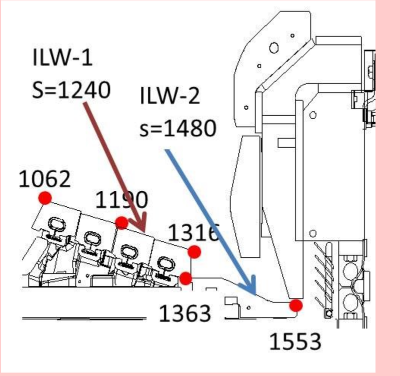
‡ see Rubel, IAEA-FEC 2016 proceedings

OUTER DIVERTOR CORNER: TILE 6



Band of Be/W deposit ~ 2-3 µm thick.





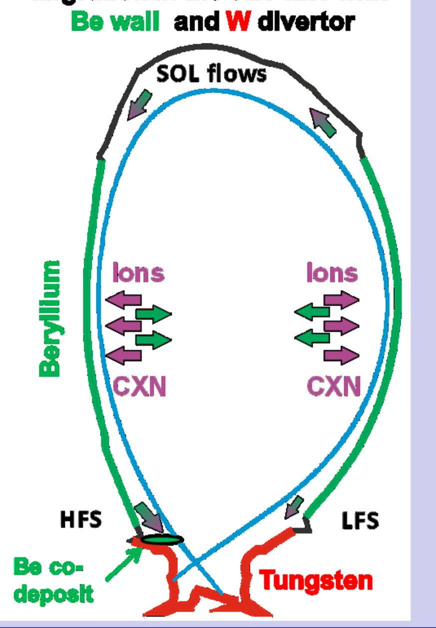
⇒ Deposition pattern governed by outer strike point location

- After ILW-2 prompt redeposition just beyond outer strike point location
- Be max. concentration increased compared with ILW-1.

CONCLUSIONS

Lower fuel retention with ITER plasma facing materials - beryllium and tungsten - compared with carbon

Advantages of beryllium and tungsten Plasma Facing Components over carbon PFCs
Lower erosion in main chamber resulting is lower migration of impurities to divertor
Transport of deposited material is divertor is by sputtering and redisposition ⇒Limited chemical sputtering of beryllium
Lower material migration to remote divertor surfaces



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