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Tritium retention in next step devices and the requirements for mitigation and removal techniques

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Motivation for this talk

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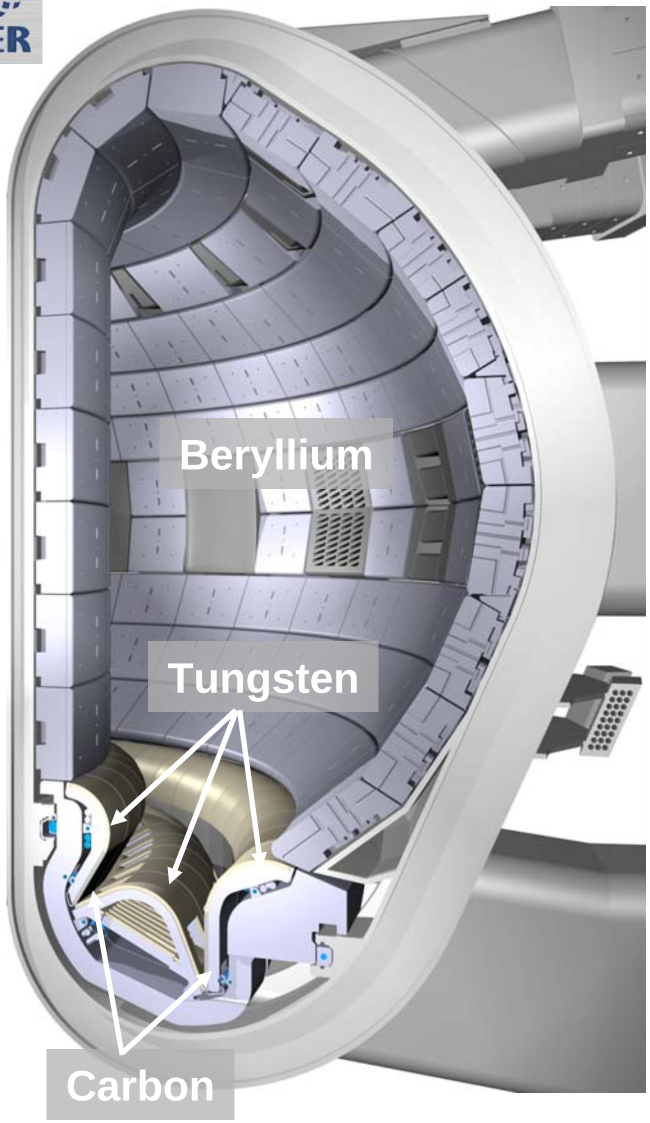

ITER plasma facing materials mix

- In the DT phase, ~50 g T injected/400 s pulse
- Mobilisable tritium inventory limit (safety) - 350g

- **700m^2 Be** first wall and start-up limiter modules
- **100m^2 W** divertor dome and baffle region
- **50m^2 Carbon Fibre Composite** (CFC) for the divertor strike point tiles

- **Carbon plasma facing components known to cause trapping of hydrogenic atoms for ~18 years (and tritium for ~15)**

Outline and aims

- Challenge of operating with CFC and tritium mix
- Growing body of experimental data on tritium retention with carbon and improvements in understanding of the underlying physics
- Current status of research into tritium removal schemes; efficiency and applicability
- Integration of tritium removal into ITER operations

D/T retention linked to C transport

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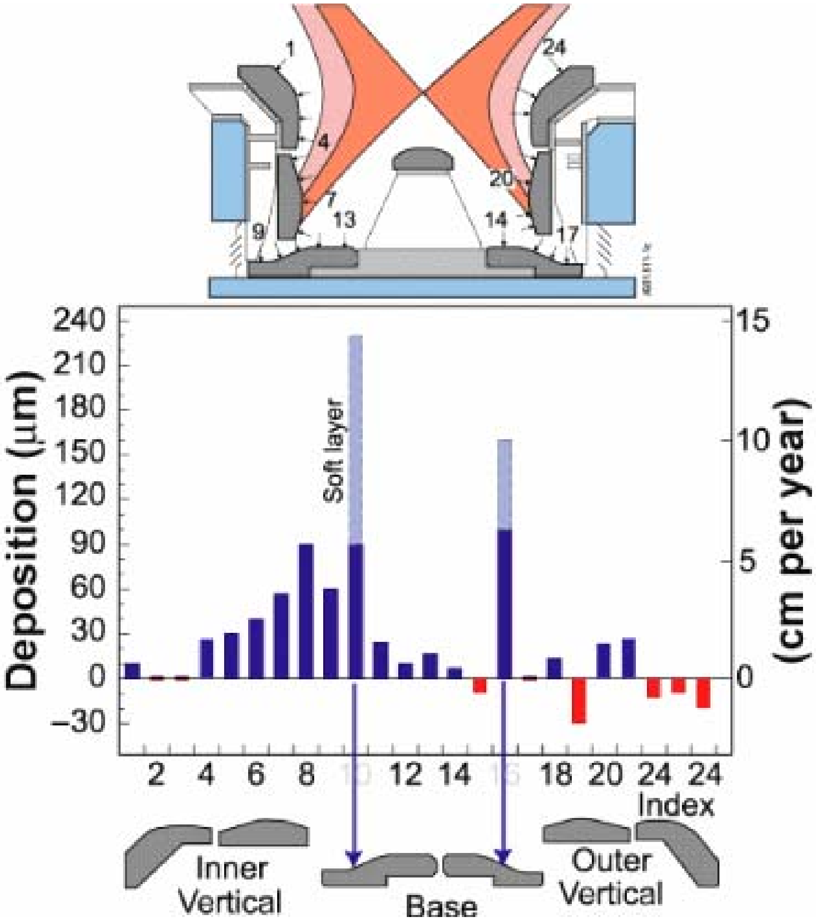
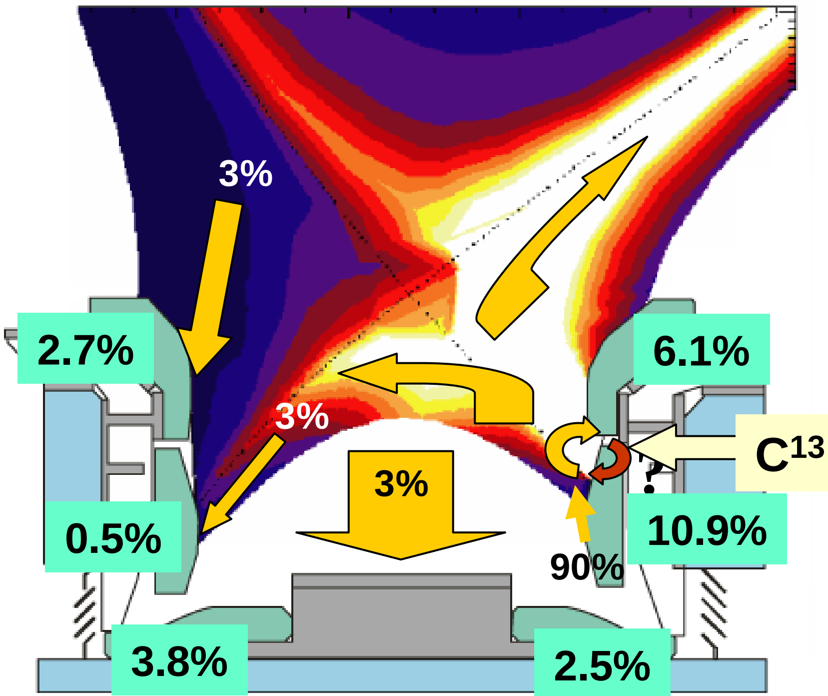


- $C^{13}D_4$ puffed into outer divertor
- Tile analysis (SIMS/IBA) to track C^{13}
- EDGE2D/NIMBUS used to model C^{13} trajectories in background plasma



- **D/T trapped in aC:H codeposited layers ⇒**
- Understanding C erosion and redeposition mechanisms is key

- Reasonable agreement with redeposition at inner divertor – **EXB drifts, SOL flows, ELMs all play a role**

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Be transport will impact C erosion

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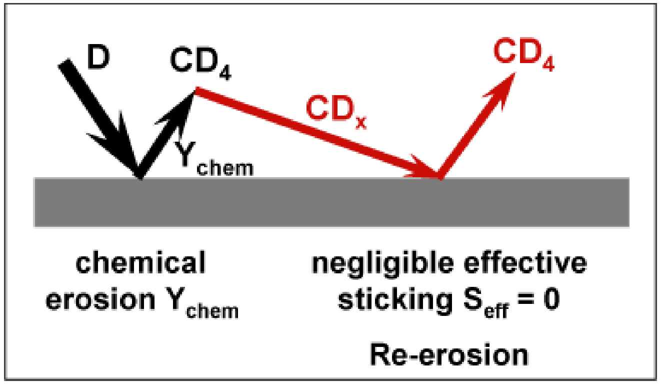
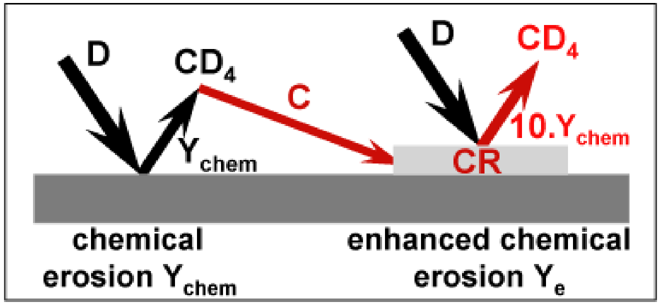
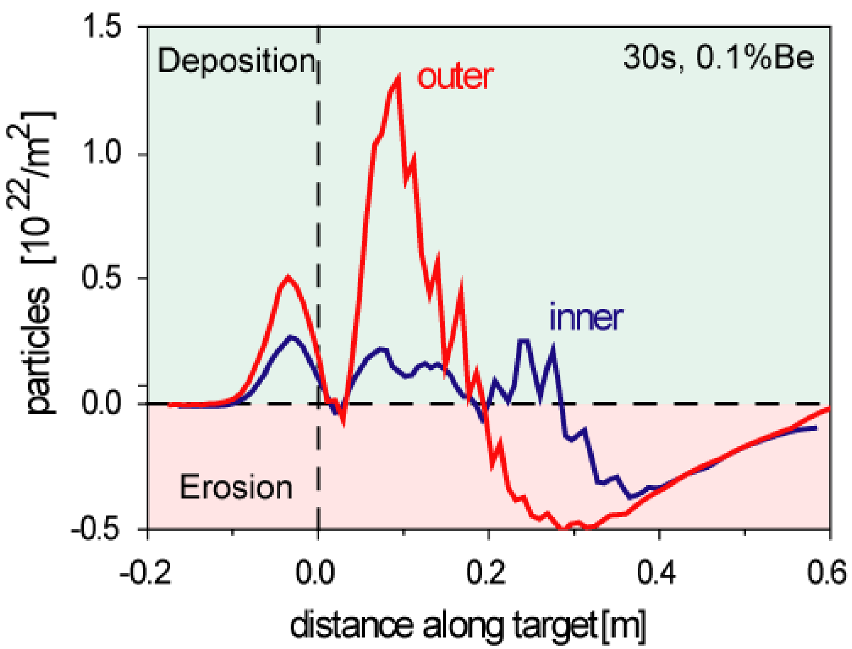
- >80% of wall area in ITER is beryllium
- Eroded Be will transport to divertor (as ions)

⇒ **modify erosion and co-deposition**





- ❑ **Preliminary** modelling using local erosion & deposition model ERO (still many open questions)
- ❑ Model assumptions validated against TEXTOR C^{13} injection experiments



- ❑ Be concentration in plasma plays key role
- ❑ Balance of inc. Be coverage on target (dec. C erosion) and inc. C erosion due to Be flux
- ❑ For a range of Be conc., T/C and T/Be ratios

⇒ **0.5g – 6.4gT/400s shot**

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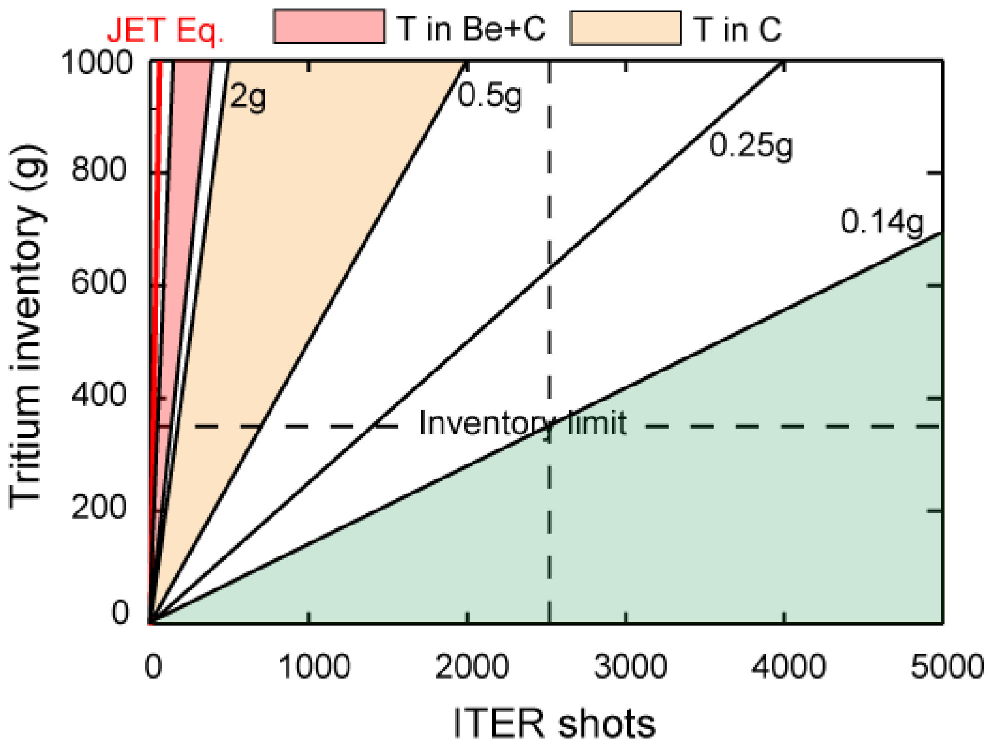
Clear need for T removal schemes



- Acceptable ITER operation ~2500 shots before maintenance period

$\Rightarrow$ Long term T retention/shot must be < 0.14g/400s shot

- ❑ Strategies for T removal essential if CFC targets in DT phase
- ❑ Removal efficiency must be 80% - 98%

aC:H co-deposits form in tile gaps

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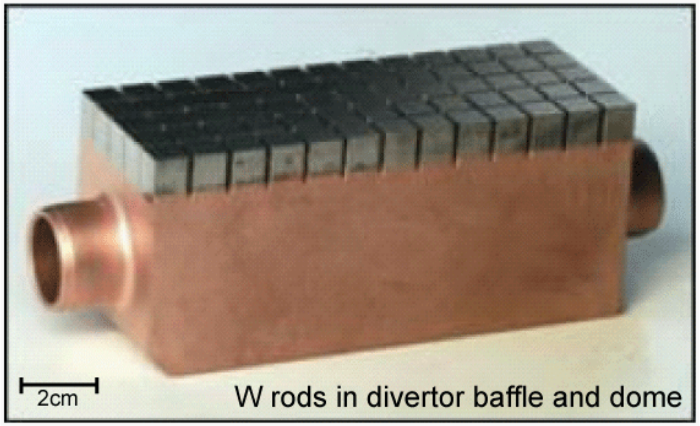
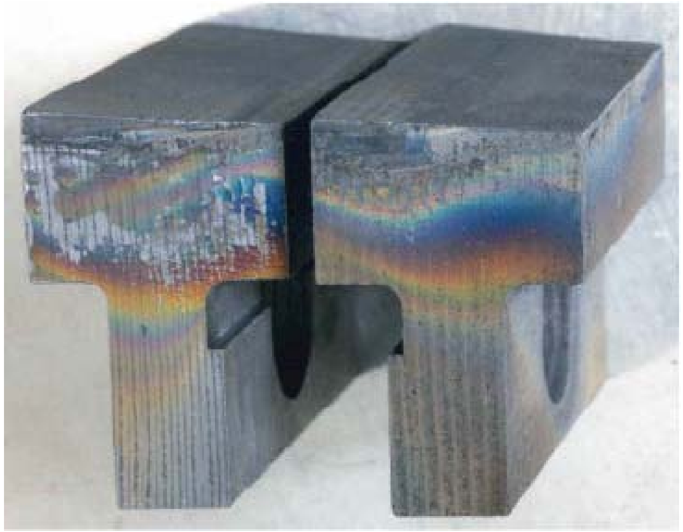
- All ITER plasma facing components will be castellated
- >2,000,000 Gaps in ITER (typ. 0.5-1mm x 10mm)
- Increases plasma exposed areas by factor 2 - 5

CFC target (90,000 monoblocks):	50 m^2	$\rightarrow$ 215 m^2
W baffle & dome (1.2M rods):	100 m^2	$\rightarrow$ 460 m^2
Be main wall (300,000 tiles):	680 m^2	$\rightarrow$ 1290 m^2

ITER mock-up



❑ C_xH_y molecules and radicals form a:C:H co-deposits deep in gaps – how much and how deep is on-going research



CFC tile segments from JET Mk1 divertor, 6mm gaps

Retention in gaps twice that on plasma-facing surfaces (protected from re-erosion)

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Potential for significant T inventory

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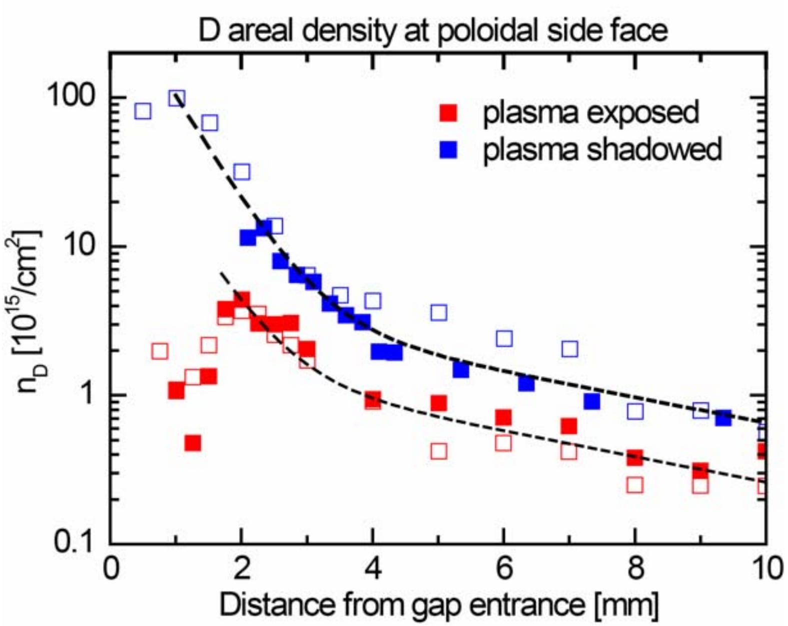


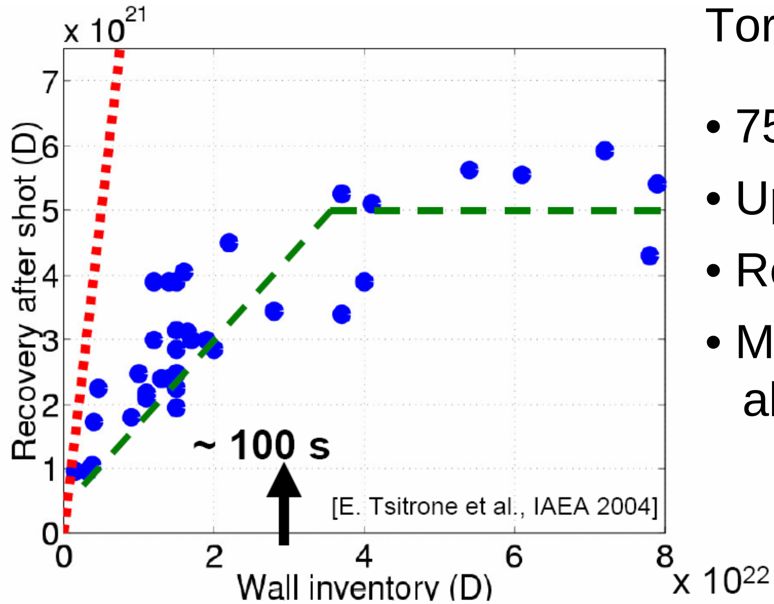
TZM castellated monoblocks exposed to plasma for 200s in TEXTOR

- ❑ D retention fall-off dependent on Γ_D and T_{tile}
- ❑ D_{gap} ~ 0.4% - 4% of Γ_D, between low and high Γ_D at T_{tile} ~200- 260°C
- ❑ Factor 10 decrease in D_{gap}, 30 → T_{tile} → 200°C

- Extrapolation to ITER based on Γ_D from B2-EIRENE modelling (Kukushkin, 2005):
  $\Rightarrow$ **0.5 – 5gT/400s shot**

- Maybe other factors, however:
  - ❑ strong function of gap width
  - ❑ carbon source (local or remote)
  - ❑ period of exposure

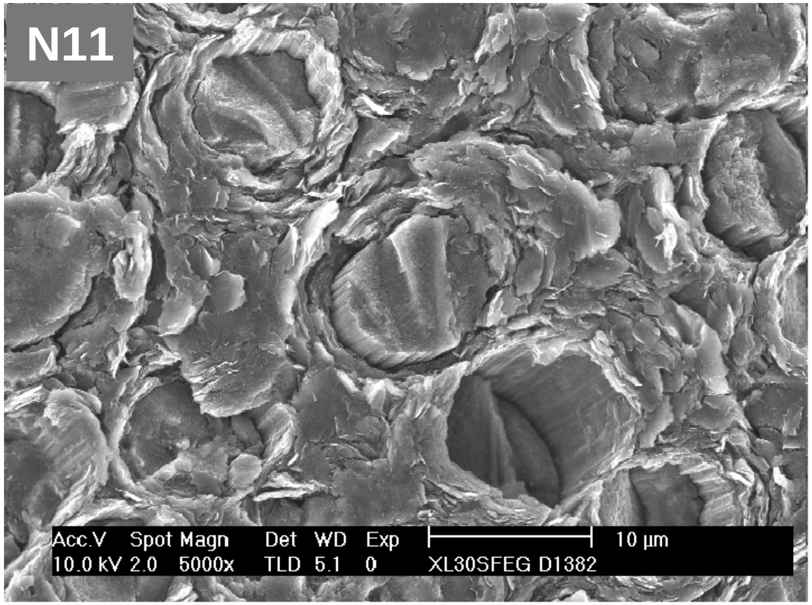
D/T retention in CFC bulk



ToreSupra:

- 75-85% D retention in short shots (<30s)
- Up to 100% D retention in long shots (>100s)
- Retention in short shots easily recovered by He glow
- Measurements of C erosion suggest co-deposition alone may not explain retention

$\Rightarrow$ more than 1 mechanism?



- ❑ Retention in bulk CFC being considered for high fluence conditions
- ❑ Lab studies indicate D retention to several μm in bulk
- ❑ D inventory $\propto$ fluence$^{0.5}$
- ❑ Calc. suggest this may initially exceed co-deposition in Tore Supra

$\Rightarrow$ could affect choice of CFC for ITER

Conventional T-recovery schemes

- Tritium operation in JET required tritium recovery before manned vessel entry
- Traditional conditioning schemes (but able to evaluate effectiveness with tritium)

	Time (h)	**T release (g)**	**Efficiency (recovery/inventory)**	**Recovery ($gT/h/150m^2$)**
D_2 tokamak discharges with S/P sweep	7	5.5	45%	**2**
Flushing with D_2 (1 – 10 Pa)	4	0.1	2%	**0.06**
D_2 GDC/ECRH	5	<0.04	<1%	**<0.02**
Baking (135 °C) under vacuum	24	0.006	<1%	**<0.001**
Flushing with N_2 (350 Pa, 150°C)	8	~0.15	3%	**0.05**
Flushing with air (100kPa)	2000	1.85	30%	**0.002**

- ❑ Efficiencies much less than 80 – 98% that is required for ITER

  ⇒ **Need to develop new T removal schemes**

- ❑ Must address all sources of retention and be compatible with ITER operation

T-removal through oxidation

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- Tritium trapped in aC:D/T co-deposits $\Rightarrow$
- Oxidation an obvious candidate for detritiation through the reaction :

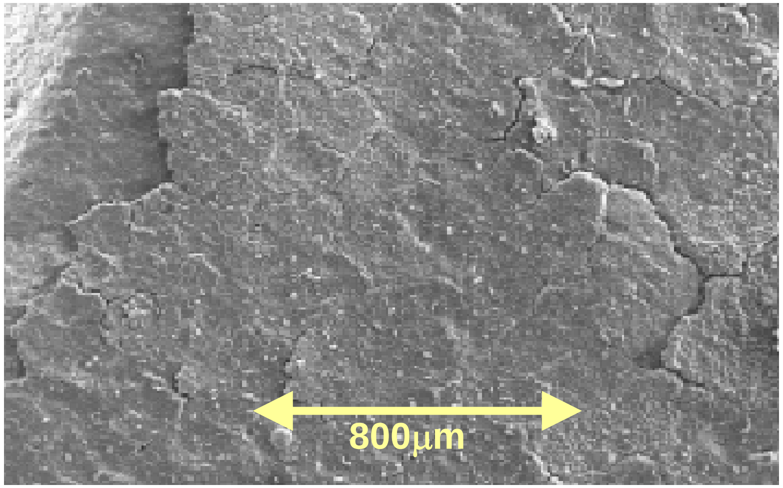
$$\text{aC:D/T} + \text{O} \rightarrow \text{CO}_x + \text{DTO:D}_2\text{O:T}_2\text{O}$$

- In-situ – no need for vessel entry
- Volatile products pumped from vessel
- Several schemes under investigation:

  - ❑ Baking in O_2
  - ❑ ECR or ICR μ-wave plasma in O_2 or He/O_2 mix
  - ❑ DC Glow discharge cleaning in He/O_2 mix

- Studies on-going in both laboratory and tokamak environments and both laboratory produced and tokamak co-deposited films

O_2 baking efficient, but at high T_{wall}



Treatment	D Content ($10^{20}m^{-2}$)	Removal efficiency (%)	Removal rate ($gT/h/150m^2$)
Original surface	121		
300°C, air, 2h	35	70	**1.6**
300°C, air, 10h	2.4	98	**0.45**
550°C, air, 1h	7	94	**4.3**
1000°C, vacuum, 1h	6	95	**4.3**
357°C, 0.3mb O_2 (TEXTOR)			**0.03**

- ❑ Molecular chemistry – O_2 penetrates all regions of deposition but ….

- Low D removal efficiency below ~300°C (*cf* ITER wall bakeout temp 240°C)
- High O_2 pressure needed for high removal efficiency
- Co-deposit not fully removed – becomes flaky and peels off

  ⇒ Inhibited O penetration and release of volatiles due to carbide formation with impurities? WC and BeC may form in ITER …
- O_3/O_2 mix effective at <200°C and low pressure but damage to bulk CFC seems to be too high

O-Plasma – effective at room temp

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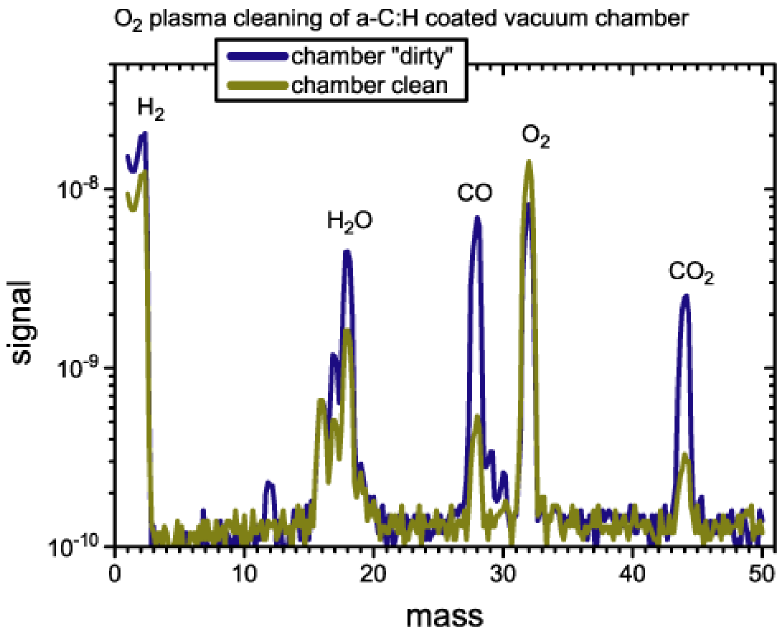
- ECR plasma in 100% O_2
- Products: CO, CO_2, H_2, H_2O



He/O_2 mixture:

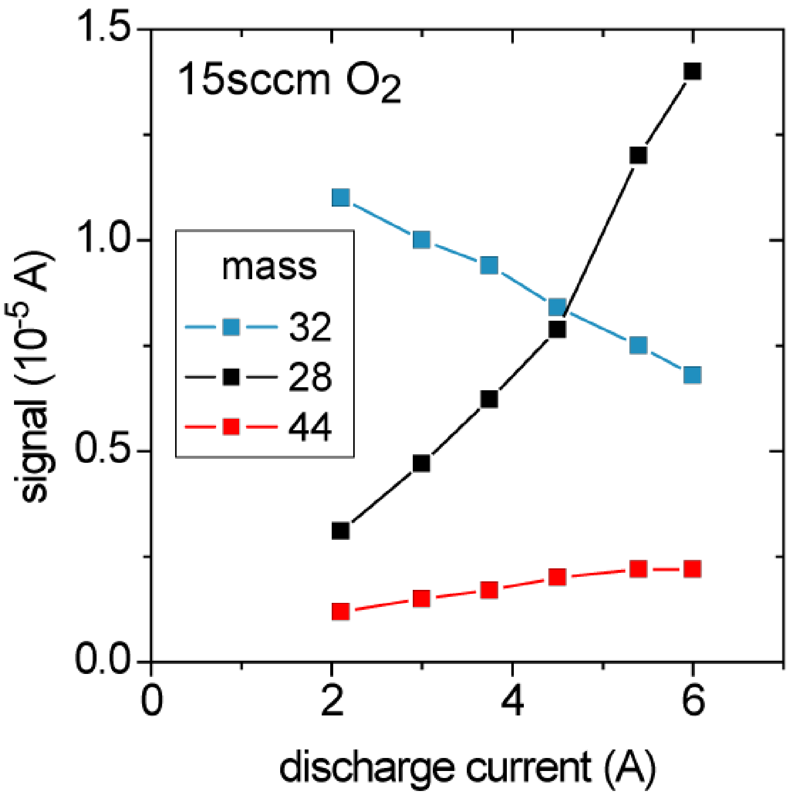
- ν_E limited by He ion flux at high %O_2

$\Rightarrow$ **ν_E saturates above few %O_2**

Erosion rate, ν_E

- increases with T_{surf} $\Rightarrow$ **chemical reactions**
- and with bias volts $\Rightarrow$ **collisions**

$\Rightarrow$ 2 step process: surface damage by ion bombardment then chemical erosion



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He/O GDC in the tokamak environment

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	Asdex Upgrade	TEXTOR
$O_2/(O_2+He)$	2%	0% – 100%
chamber pressure	6.4×10^{-3} mb	$0 – 5 \times 10^{-3}$ mb
discharge current	3 x 1.8 A	4 x 1.5 A
discharge voltage	600 V	400 V
RF assistance		120 W

Asdex Upgrade:

49h, 25g removed, 7×10^{18} C-at/s

$\Rightarrow \nu_E \sim 1.4\times10^{17}$ C-at $m^{-2}s^{-1}$

TEXTOR:

3h, 5.2g removed, 2×10^{19} C-at/s

$\Rightarrow \nu_E \sim 5.7\times10^{17}$ C-at $m^{-2}s^{-1}$

i.e. **0.075 - 0.3g T/h over 150m²**

- CO and CO_2 dominant
- T_2O 30 times higher than He GDC
- Production saturates at low O %

❑ **No removal from shadowed areas**:
a:C:H coated samples behind first wall, deep in divertor untouched

❑ **or boronised regions**:
B-coated sample coupons and boron coated co-deposited tiles unaffected

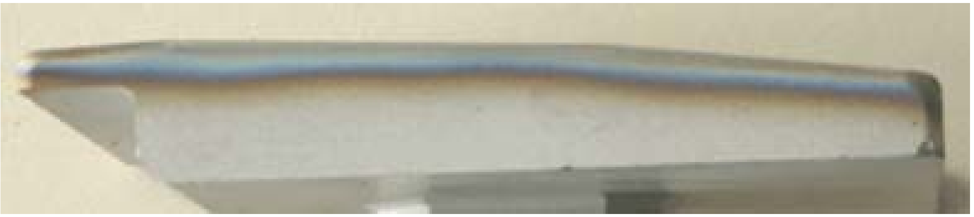
- Impact of WC, BeC in ITER?

- Tokamak and Lab studies less clear on removal from tile gaps
- O+/O may penetrate several mm into sufficiently wide gaps

  ⇒ Castellation and tile gap design may be important for ITER

Impurities in aC:H reduce efficiency

- Tokamak produced aC:H co-deposits on W substrate efficiently cleaned in He/O discharge



Fully removed with 6.25 hours lab GDC
Γ_i~2.5x10^{18} $m^{-2}s^{-1}$, 8mbar, 20% O_2 in He

- ❑ Similar ν_E to tokamak GDC
- ❑ ν_E for tokamak co-deposits up to factor 10 less than for laboratory produced
- ❑ 80% co-deposit eroded during first 20% of plasma exposure

$\Rightarrow$ **effect of impurities in co-deposit building up at surface?**

- W, Be will mix with aC:H in ITER

Collateral damage and recovery OK

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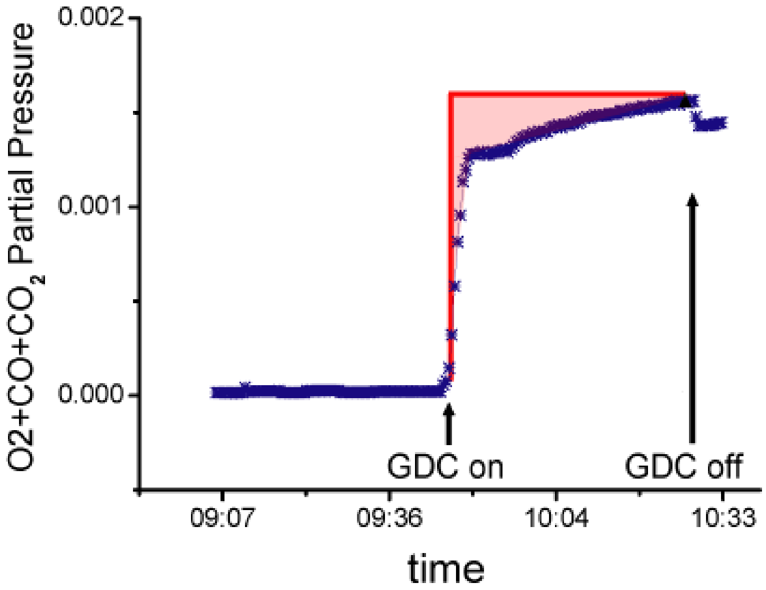
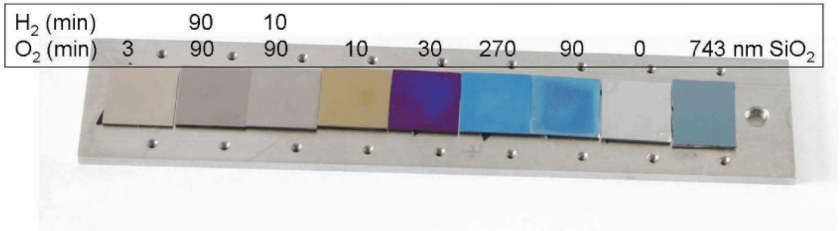
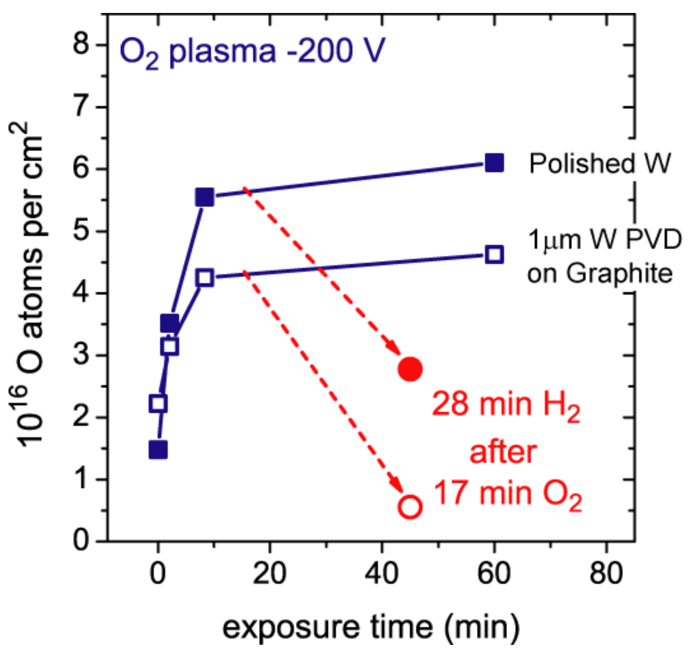
Not all injected O_2 is pumped out of the vessel during GDC



- Some O retained in metal oxides



- O retention higher at larger V_{bias}
  $\Rightarrow$ opportunity for optimisation
- H_2 discharge effective at removing oxides



- ❑ TEXTOR Recovery: 66h H_2 GDC, 0.5h He GDC & boronisation
- ❑ Asdex Upgrade Recovery: 72h baking at 150°C, 10h He GDC & boronisation)

Will recovery extrapolate to BeO?

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Alternative chemistry may have a role

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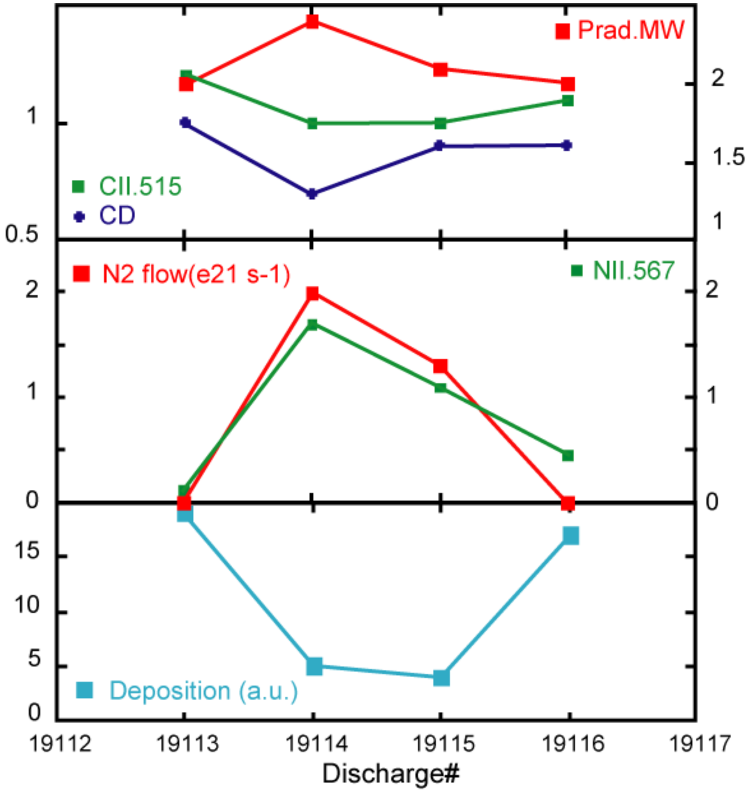
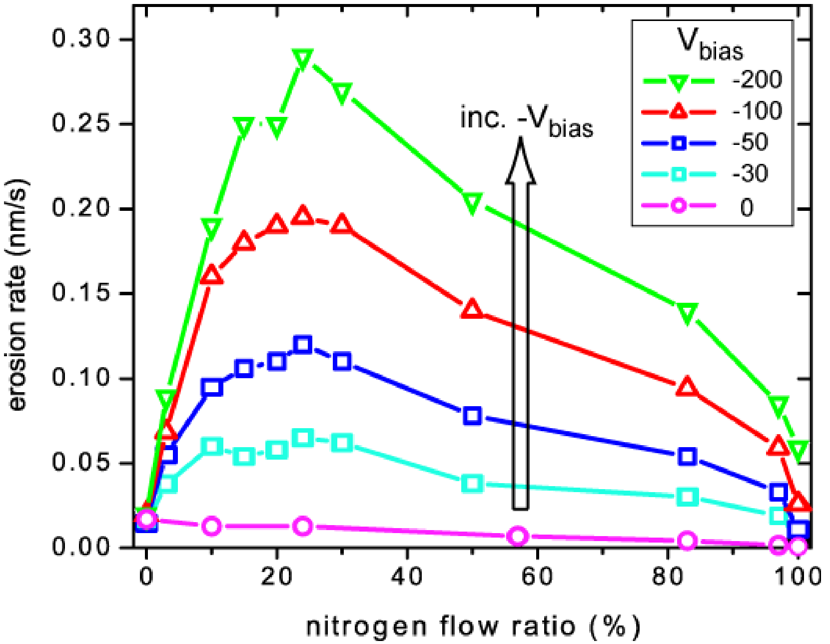


- ❑ N_2 injection into Asdex Upgrade sub-divertor
- ❑ Factor 5 reduction in aC:H net co-deposition rate
- ❑ No significant N retention
- ❑ Effect not seen with Ar (laboaratory studies)

- 'Scavenging' proposed as one mechanism
  – moping-up of reactive radical pre-cursors
- But also alternative explanations -



- Synergistic interaction of H and N at surface – peaks at ~ 75%:25%
- Erosion rates high in H_2/N_2 plasmas

  ν_E up to 1μm/hour for lab deposits

  in ECR plasma

  less than O, but not optimised

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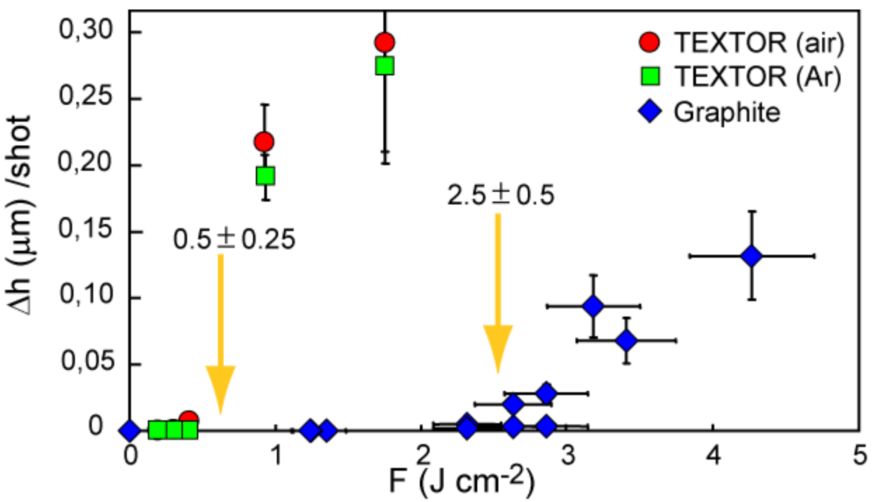
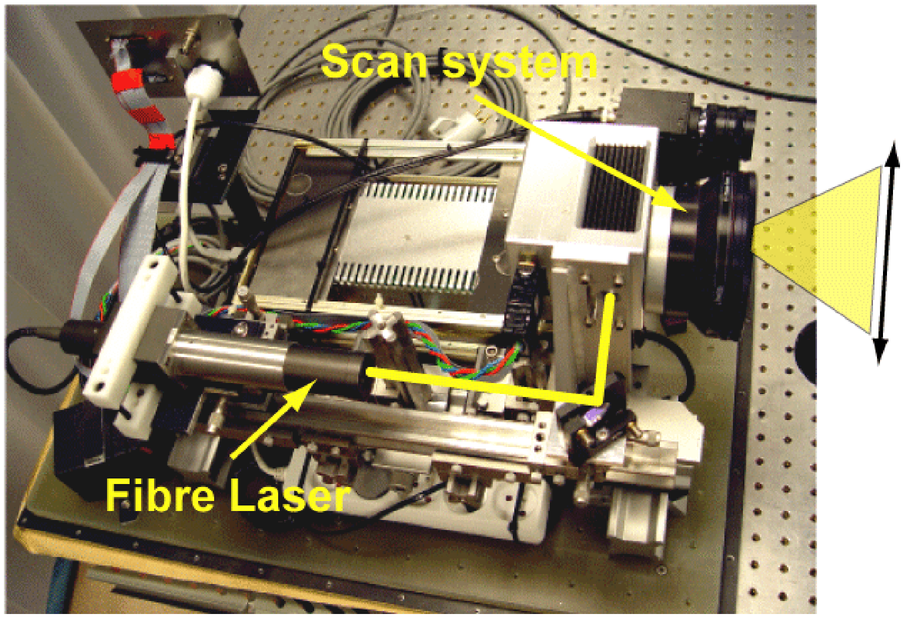
T-removal through ‘photonic cleaning’

- aC:H co-deposits have poor thermal conductivity compared to substrates (CFC, Be, W)
- Surface heat flux leads to rapid temperature rise in co-deposit $\Rightarrow$ ablation or chemical ‘bond-breaking’
- Two ‘photonic cleaning’ schemes under investigation:
    - ❑ LASER
    - ❑ Flash-lamp
- Requires vessel access, but can operate in high magnetic fields and in vacuuo, inert gas or atmospheric conditions
- Studies on-going in both laboratory and tokamak environments and both laboratory produced and tokamak co-deposited films

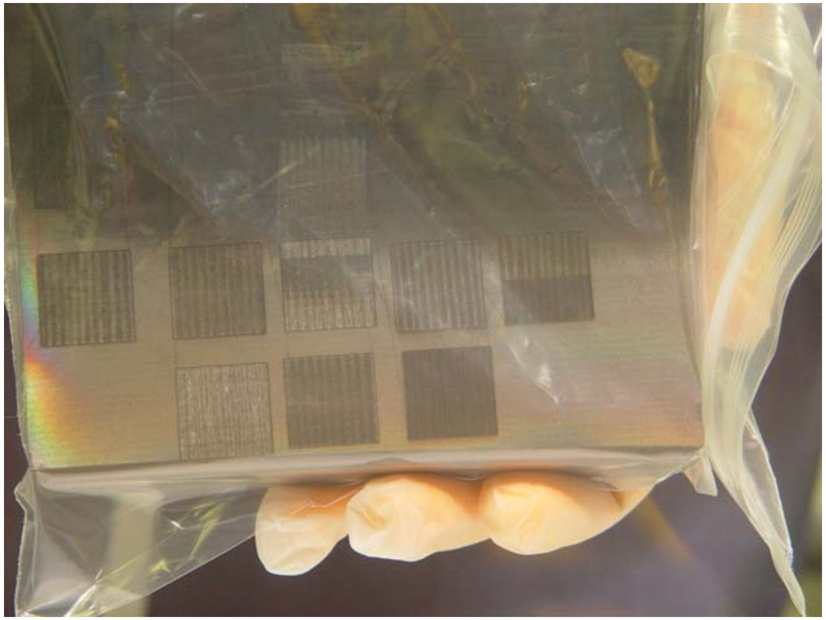
Laser cleaning of TEXTOR tile



- Energy density threshold for removal
- Threshold factor 5 lower for co-deposit compared to graphite – selective removal
- No difference between active and inert gas environment



Galvo-scanning fibre laser developed for JET



- Trials conducted in JET BeHF
- Co-deposit easily removed but only 10% T released ⇒ micro-particulate?

Flash-lamp cleaning of tritiated aC:H

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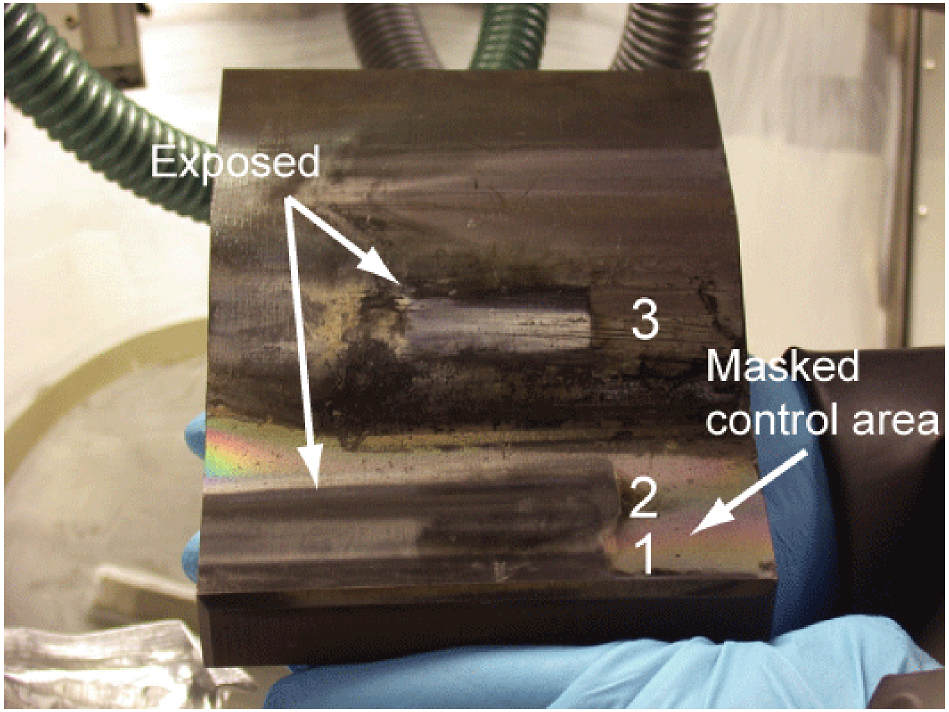
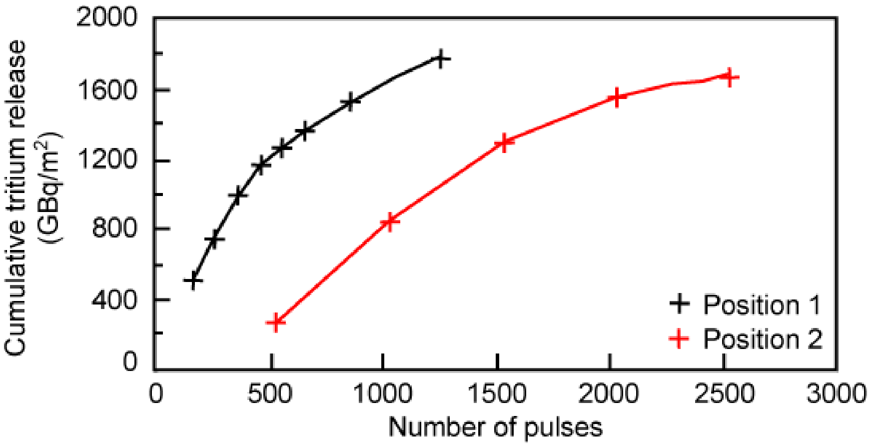
- Photon flux from 500J, 140μs flash-lamp
  $\Rightarrow$3.6MW
- Rep. rate 5Hz
- Focused using semi-elliptical cavity –
- Footprint ~30cm^2 @ 30mm
  $\Rightarrow$375MWm^{-2}, 6J/cm^2

- Trials now conducted using flash-lamp in JET berylium handling facility
- Aim to clean thick, tritiated co-deposit from inner divertor CFC tile

JET 2004 trial showed engineering feasibility of flash-lamp technology

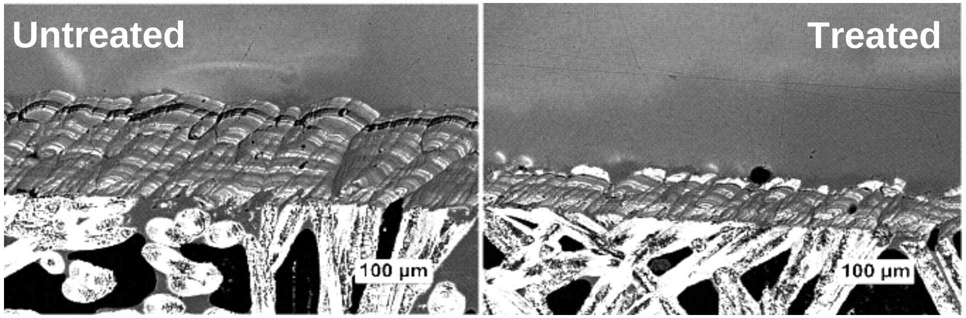


1st demonstration of T-removal



- Total T release ~9μg.
- Decreasing efficiency with number of pulses
- 40% of T inventory & 70-90 μm co-deposit, removed (off gas & SEM)

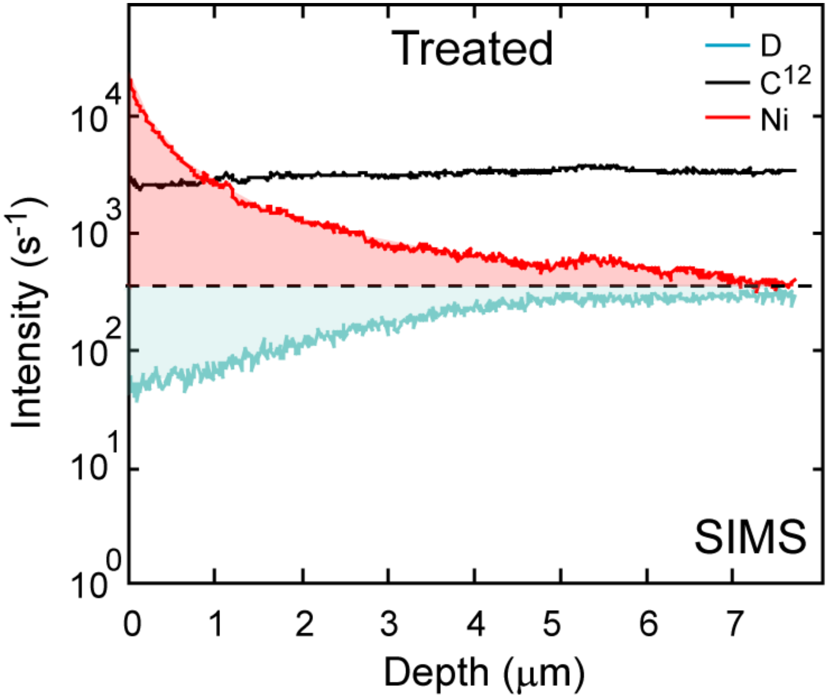
⇒ **0.075g T/h over 150m²**



- ❑ 7μm de-tritiation at surface of treated zone

  → Consistent with FE calcs of bulk heating above 700K

- ❑ Build-up of Ni at surface → explanation for roll-over of tritium release/pulse? (similar results for Be on other treated tiles)



Big ‘toolbox’ needed for ITER

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- No single T-removal scheme likely to be sufficient – let’s not close any doors
- Integration of different schemes on different timescales will probably be required – the ‘good housekeeping’ approach

	No Action	‘Good housekeeping’	%T removal/ mitigation	Possible technique
During the shot	3g	3g $\Rightarrow$ 1.8g	40%	N_2 Scavenging Optimisation of fuelling
End of shot &/or inter-shot	3g	1.8g $\Rightarrow$ 1.1g	40%	D-only phase (20%) Disruption cleaning D-only discharges D μW-plasma
Overnight (10 hours)	30g	11g $\Rightarrow$ 9g	20%	D μW-plasma D_2 flush
Weekends (2 days)	150g	45g $\Rightarrow$ 30g	35%	O_2/He or N_2 μW-plasma and D-μW recovery
Monthly (9 days)	450g	90g $\Rightarrow$ 45g	50%	O_2/He or N_2 μW-plasma O_2/He GDC (fields off?) and D-μW recovery
Annual (4 months)	3.6kg	350g $\Rightarrow$ 35g	90%	Photonic-cleaning by flash-lamp or laser (RH entry)

- ❑ **Example** of T-removal integrated into ITER operating schedule
- ❑ Extrapolated from predicted/measured T-removal rates allowing for future optimisation

Concluding remarks

- Challenge of long term tritium retention with carbon known for at least 18 years but efforts to diagnose, model and resolve only expanding in last few years
- Considerable way to go before models providing reliable estimates for tritium retention with carbon in ITER are available
- Several T-removal schemes now being investigated but all have drawbacks – no easy solutions. Much more effort needed to provide ITER with reliable technology
- Even if DT phase of ITER does not include CFC, co-deposit removal required to ensure carbon not present in vessel
- T-retention does not vanish in an all-metal ITER – trapping with intrinsic BeO or in α-damaged W may not be trivial (e.g. 0.2g/400s shot)
- T-removal schemes for an all metal ITER (and future devices) may be necessary – and may be more difficult – but little or no effort yet