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

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Motivation for this talk

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

- **700m² Be** first wall and start-up limiter modules
- **100m² W** divertor dome and baffle region
- **50m² 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)**

ITER plasma facing materials mix
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

- 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

- C\textsuperscript{13}D\textsubscript{4} puffed into outer divertor
- Tile analysis (SIMS/IBA) to track C\textsuperscript{13}
- EDGE2D/NIMBUS used to model C\textsuperscript{13} trajectories in background plasma
Be transport will impact C erosion

- >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\textsuperscript{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
Clear need for T removal schemes

- Acceptable ITER operation ~2500 shots before maintenance period

⇒ 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

• 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 m² \rightarrow 215 m²
W baffle & dome (1.2M rods): 100 m² \rightarrow 460 m²
Be main wall (300,000 tiles): 680 m² \rightarrow 1290 m²

C\textsubscript{x}H\textsubscript{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)
Potential for significant T inventory

TZM castellated monoblocks exposed to plasma for 200s in TEXTOR

- D retention fall-off dependent on $\Gamma_D$ and $T_{\text{tile}}$
- $D_{\text{gap}} \sim 0.4\% - 4\%$ of $\Gamma_D$, between low and high $\Gamma_D$ at $T_{\text{tile}} \sim 200-260^\circ\text{C}$
- Factor 10 decrease in $D_{\text{gap}}$, $30 \rightarrow T_{\text{tile}} \rightarrow 200^\circ\text{C}$

- Extrapolation to ITER based on $\Gamma_D$ 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
  ⇒ 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 ∝ fluence^{0.5}
- Calc. suggest this may initially exceed co-deposition in Tore Supra
  ⇒ 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)

<table>
<thead>
<tr>
<th>Method</th>
<th>Time (h)</th>
<th>T release (g)</th>
<th>Efficiency (recovery/inventory)</th>
<th>Recovery (gT/h/150m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>D₂ tokamak discharges with S/P sweep</td>
<td>7</td>
<td>5.5</td>
<td>45%</td>
<td>2</td>
</tr>
<tr>
<td>Flushing with D₂ (1 – 10 Pa)</td>
<td>4</td>
<td>0.1</td>
<td>2%</td>
<td>0.06</td>
</tr>
<tr>
<td>D₂ GDC/ECRH</td>
<td>5</td>
<td>&lt;0.04</td>
<td>&lt;1%</td>
<td>&lt;0.02</td>
</tr>
<tr>
<td>Baking (135 °C) under vacuum</td>
<td>24</td>
<td>0.006</td>
<td>&lt;1%</td>
<td>&lt;0.001</td>
</tr>
<tr>
<td>Flushing with N₂ (350 Pa, 150°C)</td>
<td>8</td>
<td>~0.15</td>
<td>3%</td>
<td>0.05</td>
</tr>
<tr>
<td>Flushing with air (100kPa)</td>
<td>2000</td>
<td>1.85</td>
<td>30%</td>
<td>0.002</td>
</tr>
</tbody>
</table>

- 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

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

\[
aC:D/T + O \rightarrow CO_x + DTO:D_2O:T_2O
\]

- 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
**O₂ baking efficient, but at high T_{wall}**

- Molecular chemistry – O₂ penetrates all regions of deposition but ….
  
  - Low D removal efficiency below ~300°C (cf ITER wall bakeout temp 240°C)
  - High O₂ pressure needed for high removal efficiency
  - Co-deposit not fully removed – becomes flaky and peels off
    \[\Rightarrow\text{Inhibited O penetration and release of volatiles due to carbide formation with impurities? WC and BeC may form in ITER ...}\]
  - O₃/O₂ mix effective at <200°C and low pressure but damage to bulk CFC seems to be too high

<table>
<thead>
<tr>
<th>Treatment</th>
<th>D Content (10²⁰ m⁻²)</th>
<th>Removal efficiency (%)</th>
<th>Removal rate (gT/h/150 m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Original surface</td>
<td>121</td>
<td></td>
<td></td>
</tr>
<tr>
<td>300°C, air, 2h</td>
<td>35</td>
<td>70</td>
<td>1.6</td>
</tr>
<tr>
<td>300°C, air, 10h</td>
<td>2.4</td>
<td>98</td>
<td>0.45</td>
</tr>
<tr>
<td>550°C, air, 1h</td>
<td>7</td>
<td>94</td>
<td>4.3</td>
</tr>
<tr>
<td>1000°C, vacuum, 1h</td>
<td>6</td>
<td>95</td>
<td>4.3</td>
</tr>
<tr>
<td>357°C, 0.3mb O₂ (TEXTOR)</td>
<td>800 µm</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
O-Plasma – effective at room temp

- ECR plasma in 100% O₂
- Products: CO, CO₂, H₂, H₂O

Erosion rate, νₑ
- increases with Tₚsurf \Rightarrow \text{chemical reactions}
- and with bias volts \Rightarrow \text{collisions}

\Rightarrow 2 \text{ step process: surface damage by ion bombardment then chemical erosion}

He/O₂ mixture:
- νₑ limited by He ion flux at high %O₂
\Rightarrow νₑ \text{ saturates above few %O₂}
He/O GDC in the tokamak environment

<table>
<thead>
<tr>
<th></th>
<th>Asdex Upgrade</th>
<th>TEXTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>O₂/(O₂+He)</td>
<td>2% 6.4 x 10⁻³ mb 3 x 1.8 A 600 V</td>
<td>0% - 100% 0 - 5 x 10⁻³ mb 4 x 1.5 A 400 V 120 W</td>
</tr>
<tr>
<td>chamber pressure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>discharge current</td>
<td></td>
<td></td>
</tr>
<tr>
<td>discharge voltage</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RF assistance</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- CO and CO₂ dominant
- T₂O 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?

**Asdex Upgrade:**
49h, 25g removed, 7x10¹⁸ C-at/s
\[ \nu_E \sim 1.4 \times 10^{17} \text{ C-at m}^{-2}\text{s}^{-1} \]

**TEXTOR:**
3h, 5.2g removed, 2x10¹⁹ C-at/s
\[ \nu_E \sim 5.7 \times 10^{17} \text{ C-at m}^{-2}\text{s}^{-1} \]

i.e. 0.075 - 0.3g T/h over 150m²

- Tokamak and Lab studies less clear on removal from tile gaps
- O+/O may penetrate several mm into sufficiently wide gaps
  \[ \Rightarrow \text{Castellation and tile gap design may be important for ITER} \]
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 $\Gamma_i \approx 2.5 \times 10^{18}$ m$^{-2}$s$^{-1}$, 8mbar, 20% O$_2$ in He

- Similar $\nu_E$ to tokamak GDC
- $\nu_E$ 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
Not all injected $O_2$ is pumped out of the vessel during GDC

- Some $O$ retained in metal oxides
- $O$ retention higher at larger $V_{\text{bias}}$
  $\Rightarrow$ 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 $150^\circ$C, 10h He GDC & boronisation

Will recovery extrapolate to BeO?
Alternative chemistry may have a role

- N\textsubscript{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
    – mopping-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\textsubscript{2}/N\textsubscript{2} plasmas
      \( v_E \) up to 1\( \mu \)m/hour for lab deposits in ECR plasma
      less than O, but not optimised
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 ⇒ 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

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

- Photon flux from 500J, 140µs flash-lamp ⇒ 3.6MW
- Rep. rate 5Hz
- Focused using semi-elliptical cavity –
- Footprint ~30cm² @ 30mm ⇒ 375MWm⁻², 6J/cm²

JET 2004 trial showed engineering feasibility of flash-lamp technology
1st demonstration of T-removal

- Total T release \( \sim 9\mu g \).
- Decreasing efficiency with number of pulses
- 40% of T inventory & 70-90 \( \mu m \) co-deposit, removed (off gas & SEM)

\[ \Rightarrow 0.075g \text{T/h over } 150m^2 \]

- \( 7\mu m \) de-tritiation at surface of treated zone
  \[ \rightarrow \text{Consistent with FE calcs of bulk heating above 700K} \]
- Build-up of Ni at surface \( \rightarrow \) explanation for roll-over of tritium release/pulse? (similar results for Be on other treated tiles)
Big ‘toolbox’ needed for ITER

- 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

<table>
<thead>
<tr>
<th></th>
<th>No Action</th>
<th>‘Good housekeeping’</th>
<th>%T removal/mitigation</th>
<th>Possible technique</th>
</tr>
</thead>
<tbody>
<tr>
<td>During the shot</td>
<td>3g</td>
<td>3g ⇒ 1.8g</td>
<td>40%</td>
<td>N₂ Scavenging</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Optimisation of fuelling</td>
</tr>
<tr>
<td>End of shot &amp;/or inter-shot</td>
<td>3g</td>
<td>1.8g ⇒ 1.1g</td>
<td>40%</td>
<td>D-only phase (20%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Disruption cleaning</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D-only discharges</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D⁻μW-plasma</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D₂ flush</td>
</tr>
<tr>
<td>Overnight (10 hours)</td>
<td>30g</td>
<td>11g ⇒ 9g</td>
<td>20%</td>
<td>O₂/He or N₂ μW-plasma</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>and D-μW recovery</td>
</tr>
<tr>
<td>Weekends (2 days)</td>
<td>150g</td>
<td>45g ⇒ 30g</td>
<td>35%</td>
<td>O₂/He or N₂ μW-plasma</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>and D-μW recovery</td>
</tr>
<tr>
<td>Monthly (9 days)</td>
<td>450g</td>
<td>90g ⇒ 45g</td>
<td>50%</td>
<td>O₂/He or N₂ μW-plasma</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>O₂/He GDC (fields off?)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>and D-μW recovery</td>
</tr>
<tr>
<td>Annual (4 months)</td>
<td>3.6kg</td>
<td>350g ⇒ 35g</td>
<td>90%</td>
<td>Photonic-cleaning by flash-lamp or laser (RH entry)</td>
</tr>
</tbody>
</table>

- 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 $\alpha$-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