Tritium Retention and Removal in Tokamaks

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with credit to many colleagues in PSI community
2nd International ITER Summer School
July 22-25th 2008, Fukuoka, Japan

Outline

- Why are tritium and dust important ?
- TFTR & JET tritium experience
  - H retention in other tokamaks
- Tritium removal
- Projections for ITER

PPPL PRINCETON PLASMA PHYSICS LABORATORY

Tritium economy

- Most critical issue for any power source is fuel supply
- ITER expected to consume 15-18 kg-T.
- T is expensive ($31,000/g) and in short supply.
- Disposal of tritiated waste even more expensive (≈ x100?)

ISSUES BECOME MORE STRINGENT FOR DT REACTOR

- Extra tritium needed to fuel expanding number of DT reactors
- T is bred by $^{7}Li + n \Rightarrow {}^{4}He + T + n - 2.5$ MeV

  and $^{6}Li + n \Rightarrow T + {}^{4}He + 4.8$ MeV … looks like x2
- Actual T breeding ratio ~ 1.1 in tokamak geometry.

RETENTION LIMIT - 1 GW electric DT reactor burns ~ 350 g-T / day

- Max He in plasma core ~ 15%; (de)enrichment of He in divertor exhaust ~ 0.2
  Fuelling efficiency ≈ 10% - 80% (gas puff, mol. beam, pellets)
- T burn efficiency = (max He%) x (de-enrichment) x (fuel effic.) ≈ 1% .
- T fueling ~ 1,000 kg / mo, compared to ~ 1 kg safety limit.
- T Retention must be ≤ 0.1% for 1 mo continuous operation. *[D Whyte, C Kessel]*

Tritium safety

T inventory limit is derived from no public evacuation criterion (< 50 mSv dose)

GSSR analysis*

- Conservative weather, building wake, 1 km to site boundary

  => 90 g T tolerable ground level release.

  ground level release = T release x building confinement factor

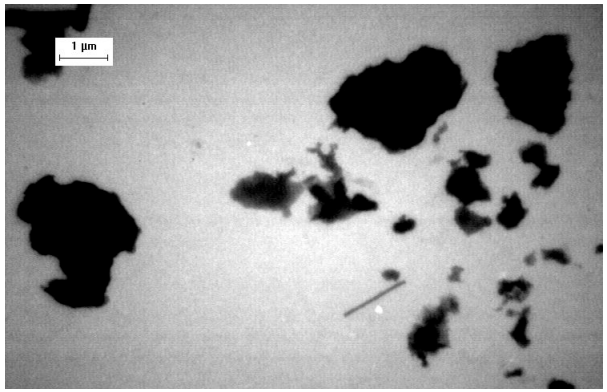
- Worst credible accident:
  - Vacuum vessel bypass event and
  - 8 hour blackout (8 h) and
  - In-vessel loss of coolant
- For 1 kg T inventory only 15 g tritium released to environment
  - good safety margin !

*Analysis now updated for Caderache site in Preliminary Safety Report.

- Tritium can be released in dust as well as T_2 and DTO gas
- W dust can also be activated

Tritiated dust more hazardous than HTO

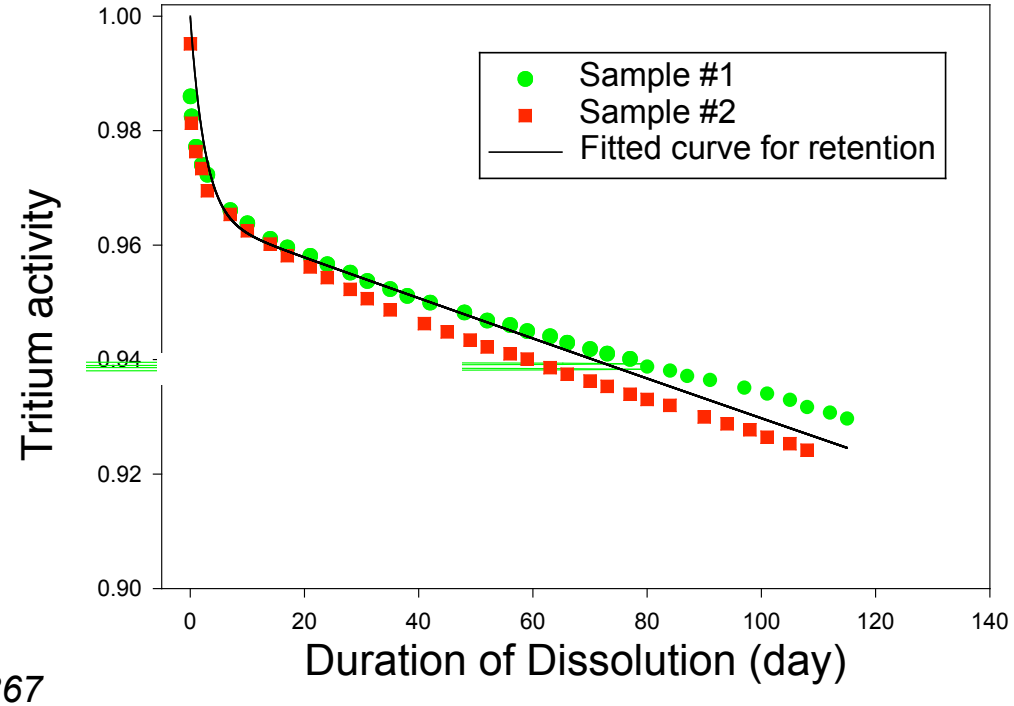
- Tritiated dust obtained from TFTR
- Size analysis showed it is respirable
  - CMD = 1.23 μm, GSD = 1. 72 μm
- *In-vitro* dissolution rate measured in simulated lung fluid.



Result:

- Only 8% of carbon tritide was dissolved after 110 days.
- Low solubility means tritium will remain for long time increasing radiation dose to lung.

- Data needed on a:BeT dust to determine allowable exposure !



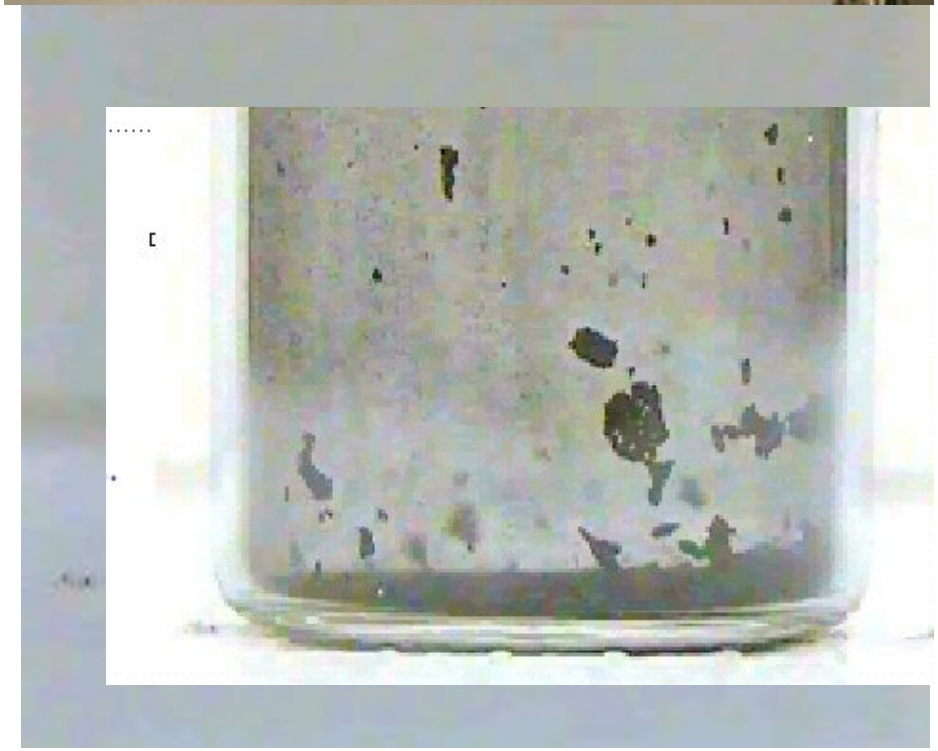
Cheng et al., Fus. Technol., 41 (2002) 867

Tritiated dust levitation by beta induced static charge

- Radioactive decay of tritium via beta emission leaves a positive charge on a dust particle.
- Tritiated particles could be uniquely more mobile than other dust.
- Movie of tritiated dust from TFTR-->

Good News:

- D/C in TFTR dust only 0.007
  T/C in TFTR dust only 0.0003
  (TFTR D/T fueling ratio 3%)
  Low D/C indicates high temperature
  H isotope outgassing in dust generation.
- cf. JET flakes D/C = 0.75
  higher value similar to codeposits.



Fus. Sci. Technol., 45 (2004) 11

Tritium and dust limits

To avoid evacuation of population in case of environmental release:

Safety assessment values ⇨ 1 kg in-vessel tritium limit

⇨ 1,000 kg in-vessel dust limit

Administrative in-vessel limits:

reduced by T inventory in cryo-pumps	-120 g
reduced by T accounting uncertainties	-180 g
Tritium administrative limit	⇨ 700 g
reduced by dust accounting uncertainties	-330 kg
Dust administrative limit	⇨ 670 kg

To avoid vessel overpressure accident:

Limit H-isotope production ≤ 2.5 kg if hot dust reacts with steam

e.g. $Be + H_2O \rightarrow BeO + H_2$

or potential pure dust or H/dust explosion.

Hot dust limit with carbon ⇨ 6 kg C, 6 kg Be, 6 kg W

Hot dust limit without carbon ⇨ 11 kg Be, 230 kg W

Difficult since limit is low and measurements uncertain

- Why are tritium and dust important ?
- TFTR & JET tritium experience
  - H retention in other tokamaks
- Tritium removal
- Projections for ITER

Tritium retention

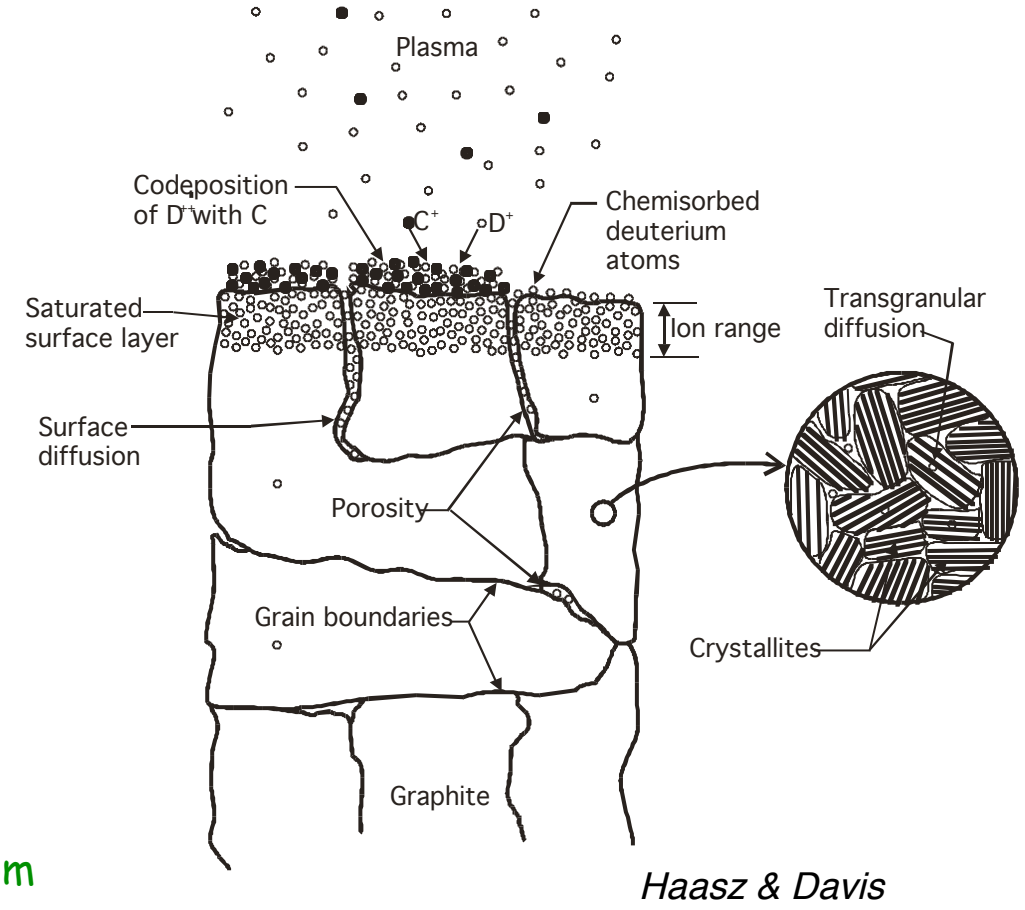
Basic mechanisms for retention

1. Short-term adsorption followed by outgassing (not a long-term problem).
2. Long-term deep implantation, diffusion, migration, trapping.
3. Long-term codeposition of tritium with plasma eroded materials e.g. C, Be.

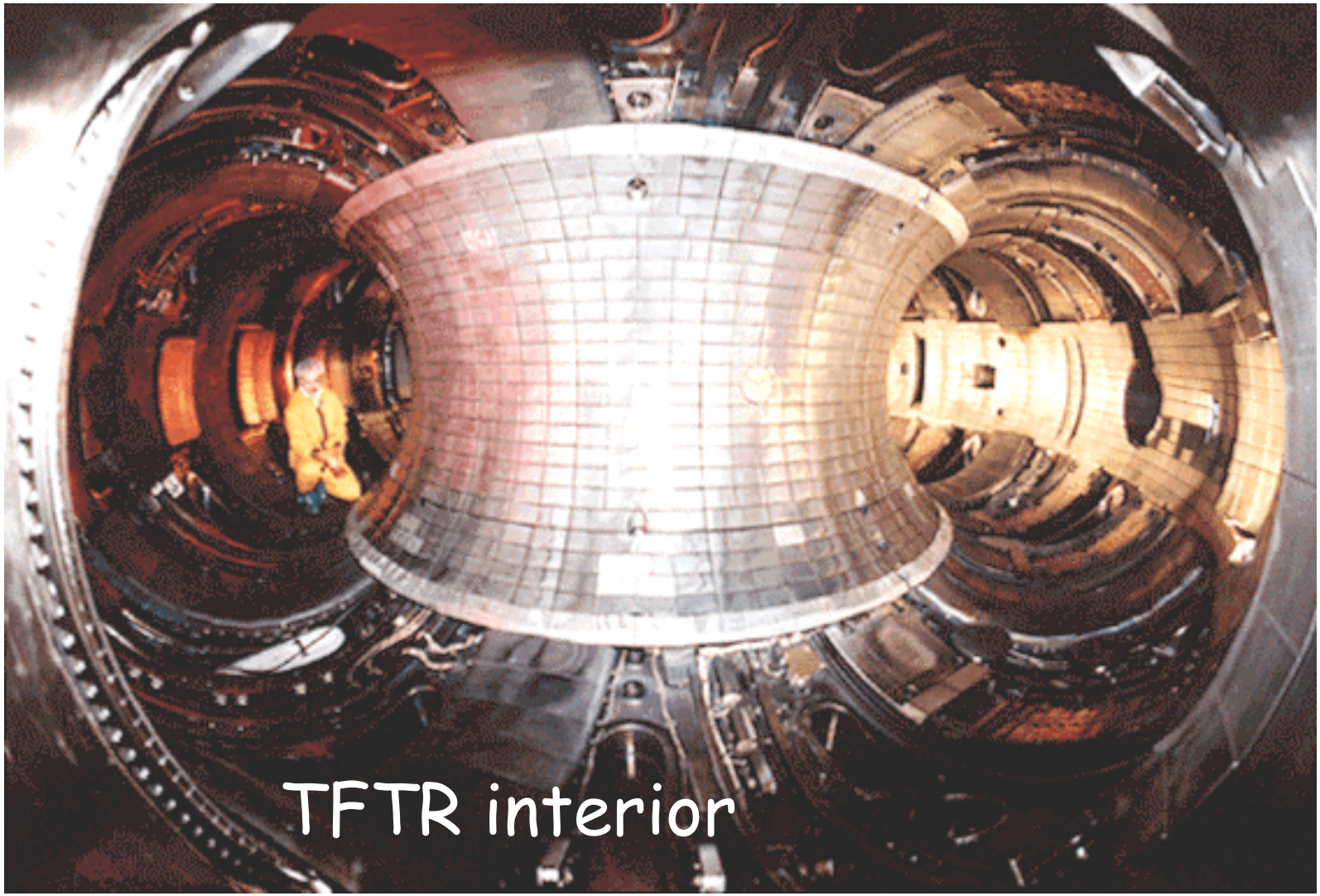
Two complementary methods to measure retention (R).

1. Gas balance, or fueling - exhaust (typically R≈ 10%-20%)
2. Analysis of components removed from vessel (typically R≈ 3% - 50%).

Retention in graphite



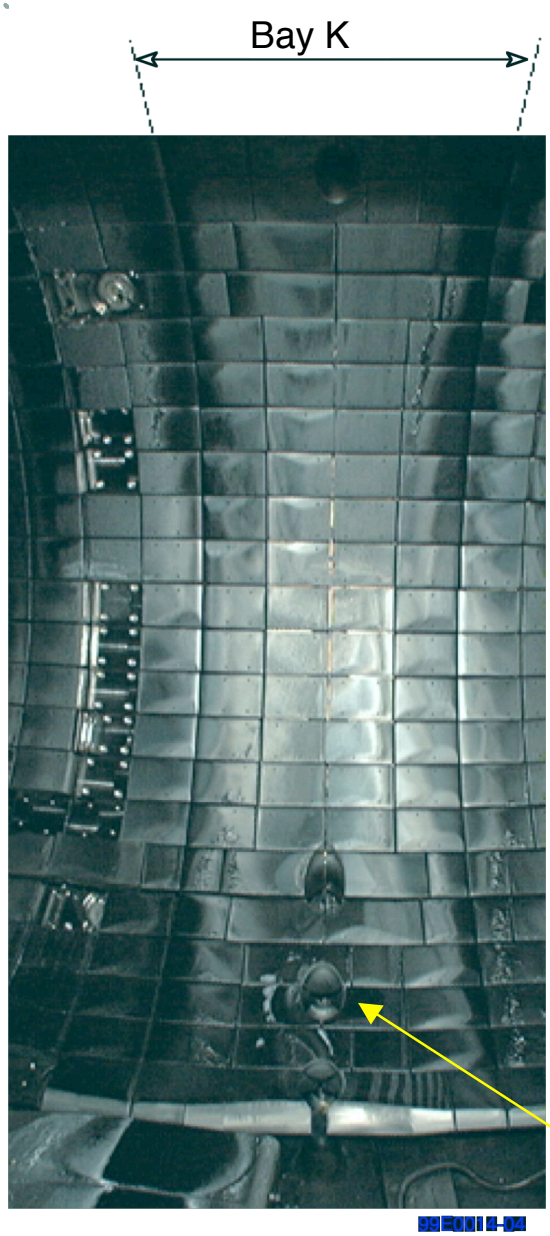
Haasz & Davis

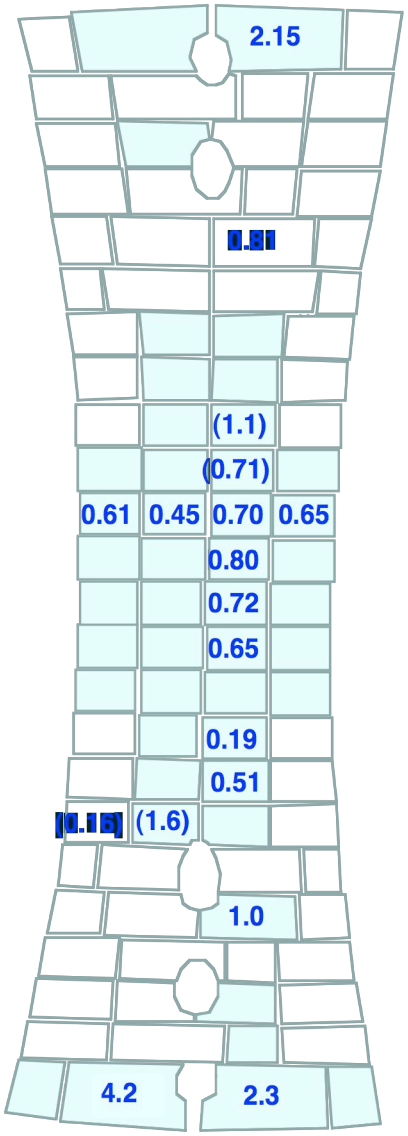


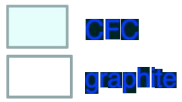
- DT operations 1993 - 1997
- Limiter machine - no divertor.
- Walls are deposition areas (not erosion)
- Walls heated only by plasma (limiter hotspots reached ≈ 800 C).

	TFTR SOL (TRANSP/DEGAS)	JET divertor (EDGE2D)
Ne	0.1 e19 – 1 e19 m-3	≈ 10 e19 m-3
Te	200 - 600eV	<30 eV

Tritium on TFTR bumper limiter.







After plasma operations tritium in TFTR was located on inner limiter (0.2 g), and outer wall (0.36 g).

Highest concentrations were at top and bottom of limiter.

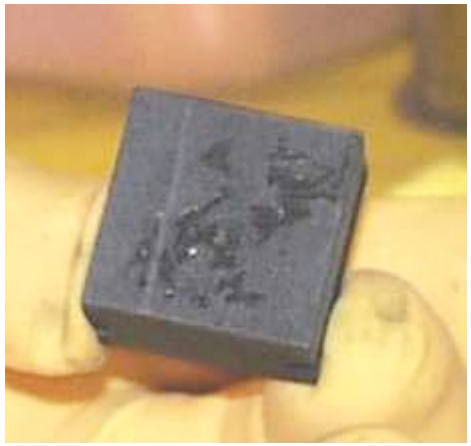
1 g-T = 9615 Ci

Numbers represent T (Ci) released by bakeout in air 500 C for 1 hour.

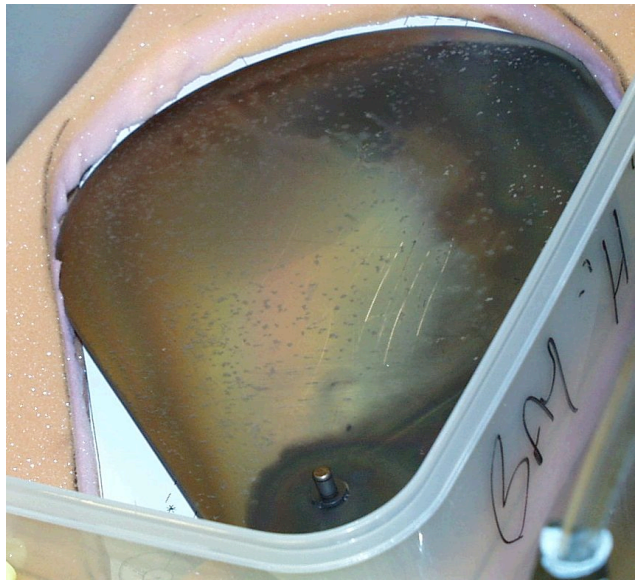
Diagnostic ports

C. H. Skinner et al., J. Nucl. Mater., 290-293 (2001) 486

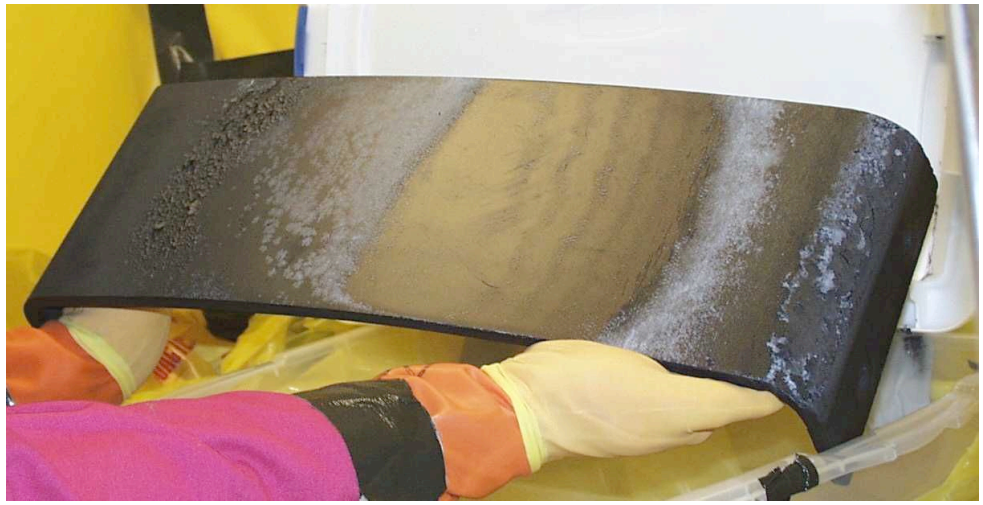
Samples from outboard side of vessel



Bay H midplane graphite coupon: 24 Ci/m^2
Bay N bottom graphite coupon: 65 Ci/m^2
Bay P midplane graphite coupon: 16 Ci/m^2



Bay H shutter (stainless steel) 9 Ci/m2

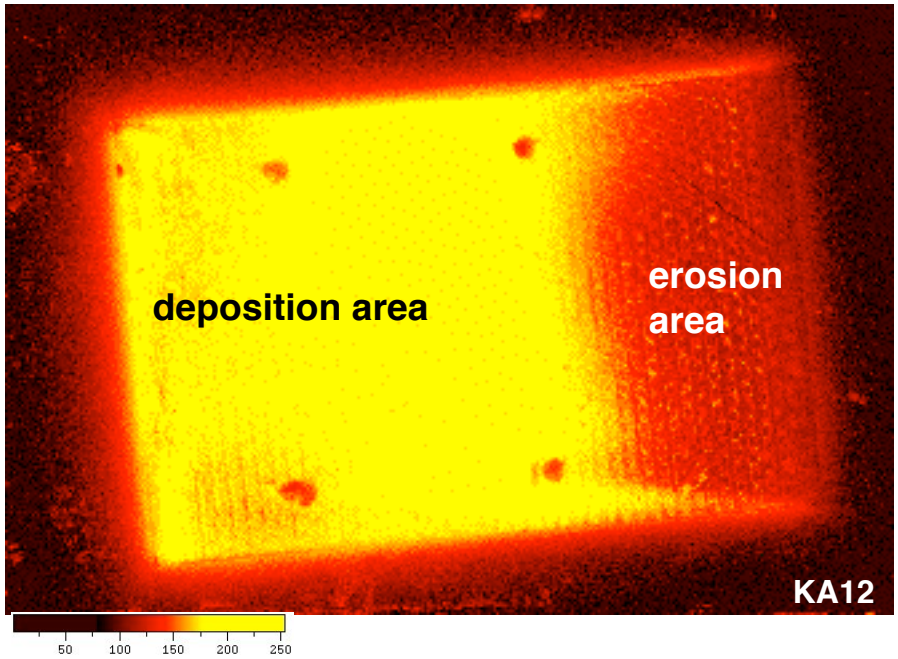


Bay O/N poloidal limiter tile: 31 Ci/m2

1 g-T = 9615 Ci

Images of tritium on TFTR tiles

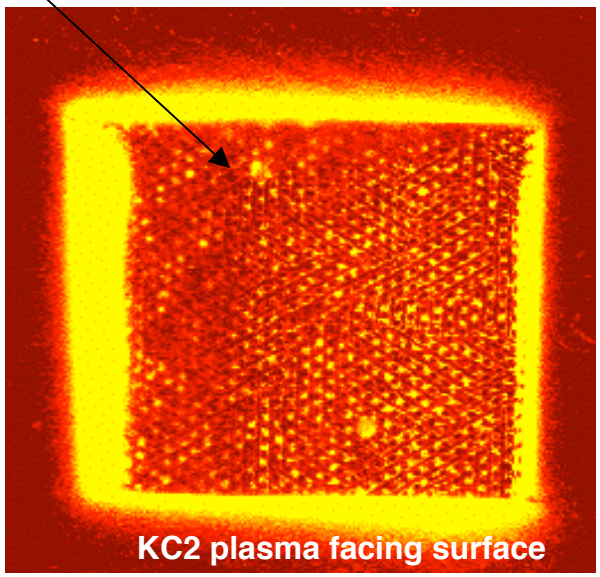
Tritium on TFTR CFC tile measured by imaging plate technique (false color).



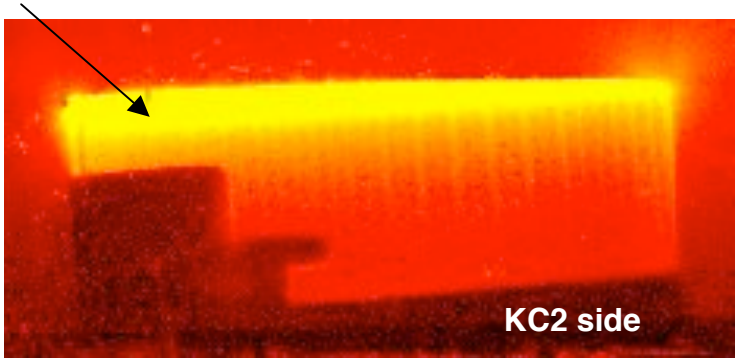
T Tanabe and K Sugiyama,
Fus. Sci. & Tech. 48 (2005) 577

15% D retained in TFTR tile gaps

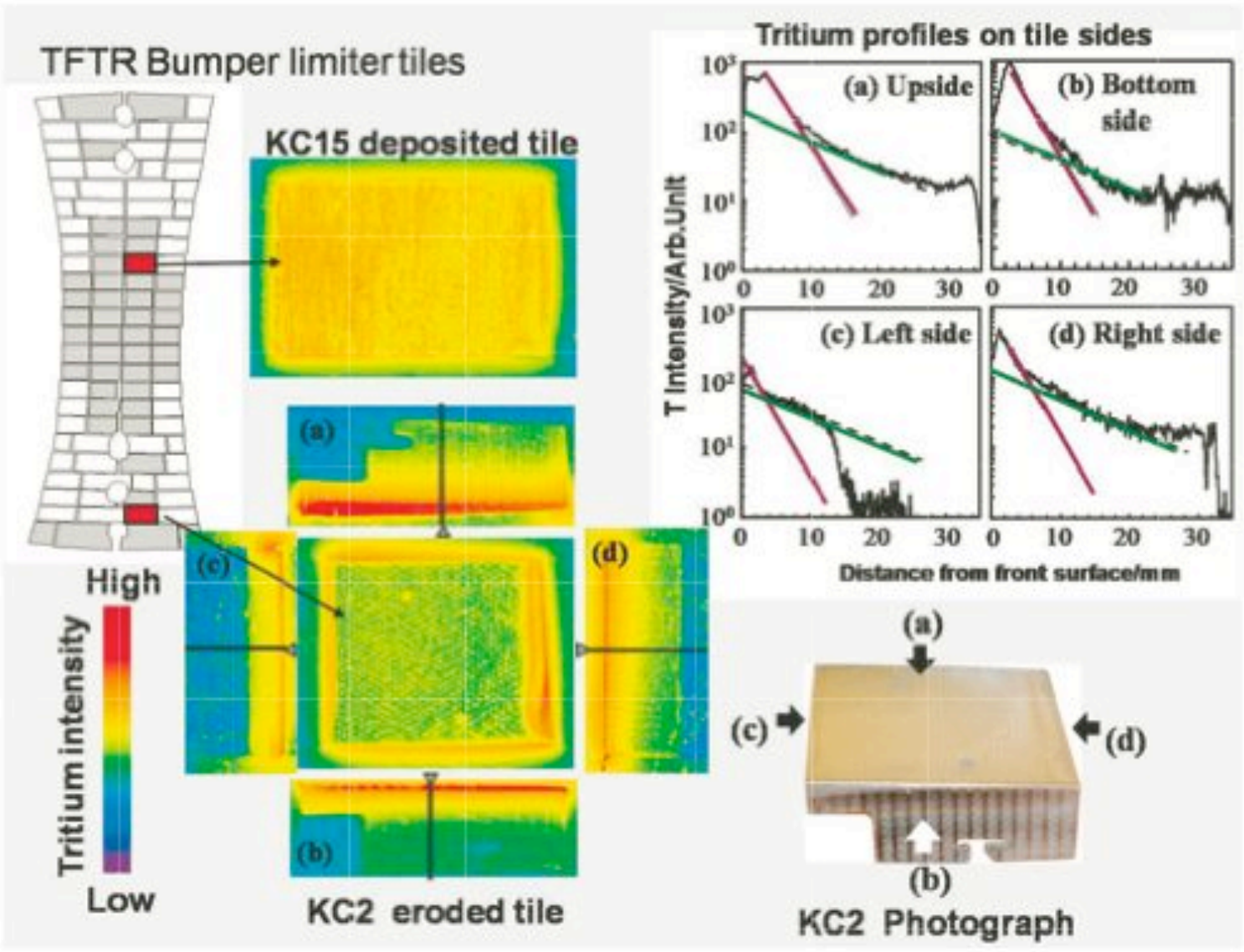
Tile from erosion region: tritium deposition in matrix between carbon fibers



Tritium deposition on tile sides



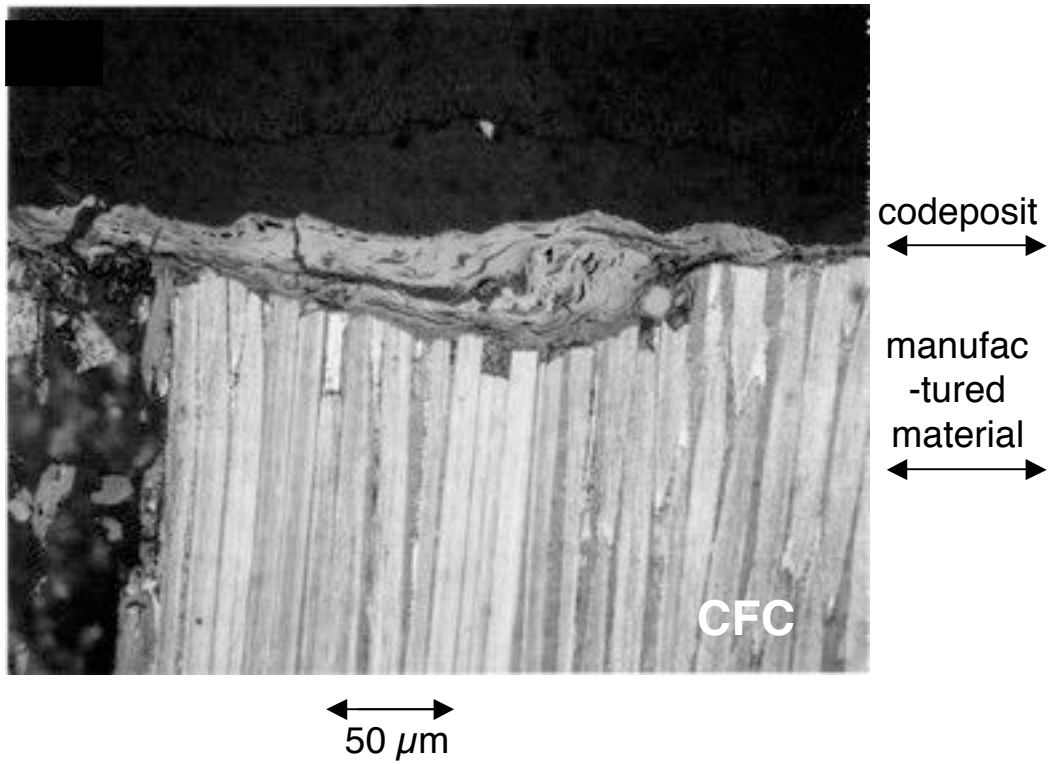
Images of tritium on TFTR tiles (2)



Penetration of T into gaps depends on magnetic field and population of high and low sticking probability hydrocarbons.

Important for tritium removal

TFTR codeposits containing tritium



TFTR tile samples impregnated with epoxy, polished and viewed in a metallurgical microscope.

Remarkably convoluted structure with distinct strata and voids that reflect the discharge history.

C. H. Skinner et al., Phys. Scripta T 103, 34-37 (2003)

R. Reiswig, S. Willms LANL

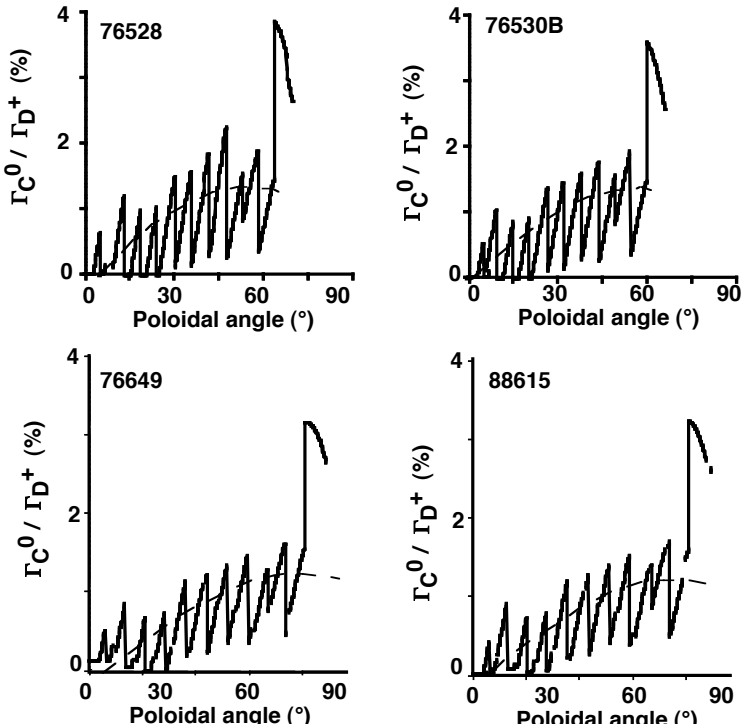
Modeling of C production and Tritium retention in TFTR

(John Hogan)

BBQ code describes:3D space, 3D velocity test particle Monte Carlo code for emitted C impurities from physical, chemical sputtering and radiation-enhanced sublimation (RES)

Parallel, perpendicular diffusion, electrostatic fields, friction with SOL flow, atomic/molecular physics *(includes Erhardt-Langer database for CD_4 breakup)*

Combines detailed TFTR Bumper Limiter geometry (CAD) with impurity SOL transport and redeposition



Local effective sputtering yield distribution on bumper limiter (emitted impurity flux / incident D+ flux for 4 representative discharges.

Extrapolate carbon erosion from selected representative discharges H-isotope/C ratio in co-deposits approximately 0.2 (NRA) – estimate retention….

C. H. Skinner et al., J. Nucl. Mater., 290-293 (2001) 486

Modeling can account for order of magnitude of retention

Location of TFTR Tritium inventory:

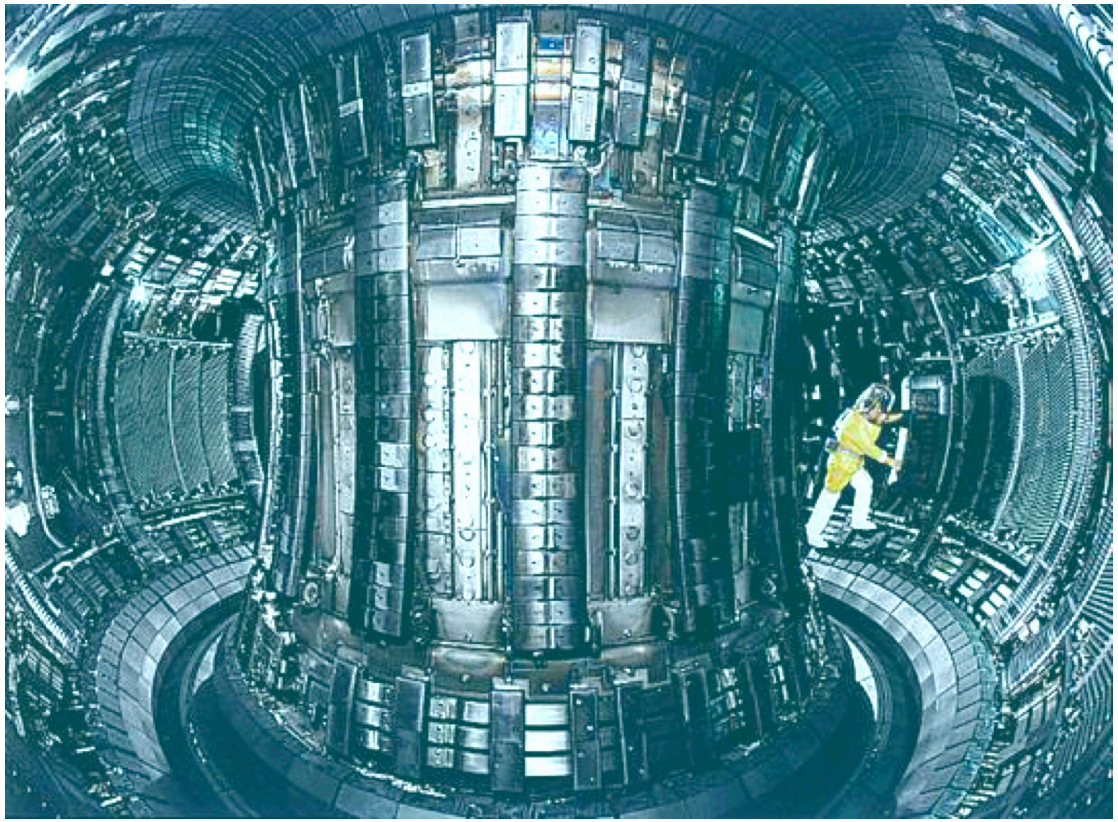
Location:	Area (m^2)	Average Ci/m^2 from bakeout + 10%	Inventory (Ci)	(g)
Bumper limiter	22	87	1,900	0.2
Outboard	110	32	3,500	0.36
Total			**5,400**	**0.56**
cf. fueling - exhaust			***6,200***	***0.64***

- *Average 51% of tritium retained during plasma operations*
  1/3 tritium on bumper limiter, 2/3 on outboard wall
- *Remarkably good agreement between extrapolation from tile analysis and gas balance (fueling less exhaust) and measurements at both PPPL and Savannah River.*
- *Also good agreement with modeling predictions.*

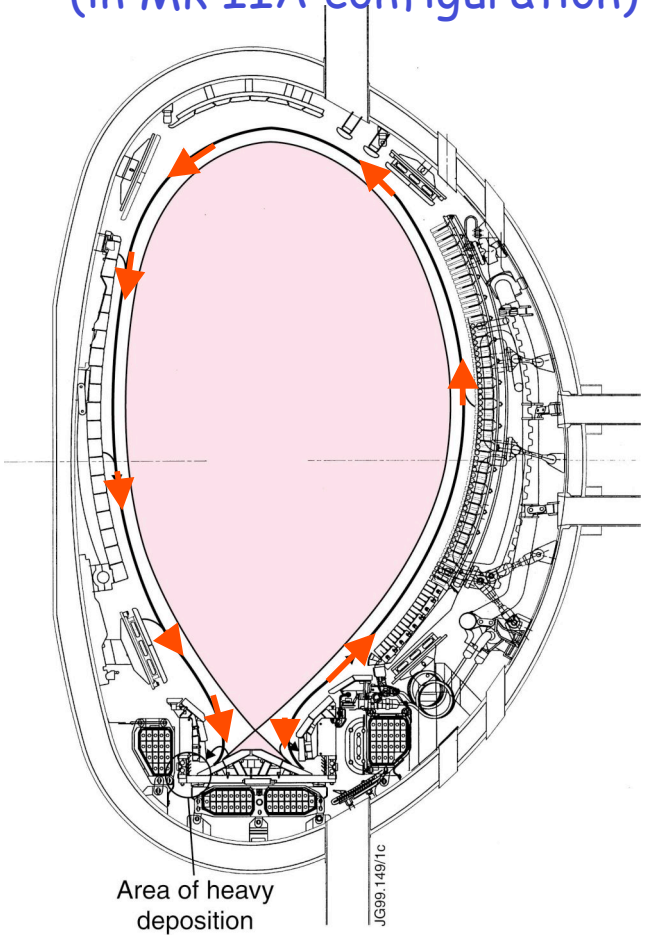
1 g-T = 9615 Ci

C. H. Skinner et al., J. Nucl. Mater., 290-293 (2001) 486

JET interior



Transport of impurities
(in Mk IIA configuration)

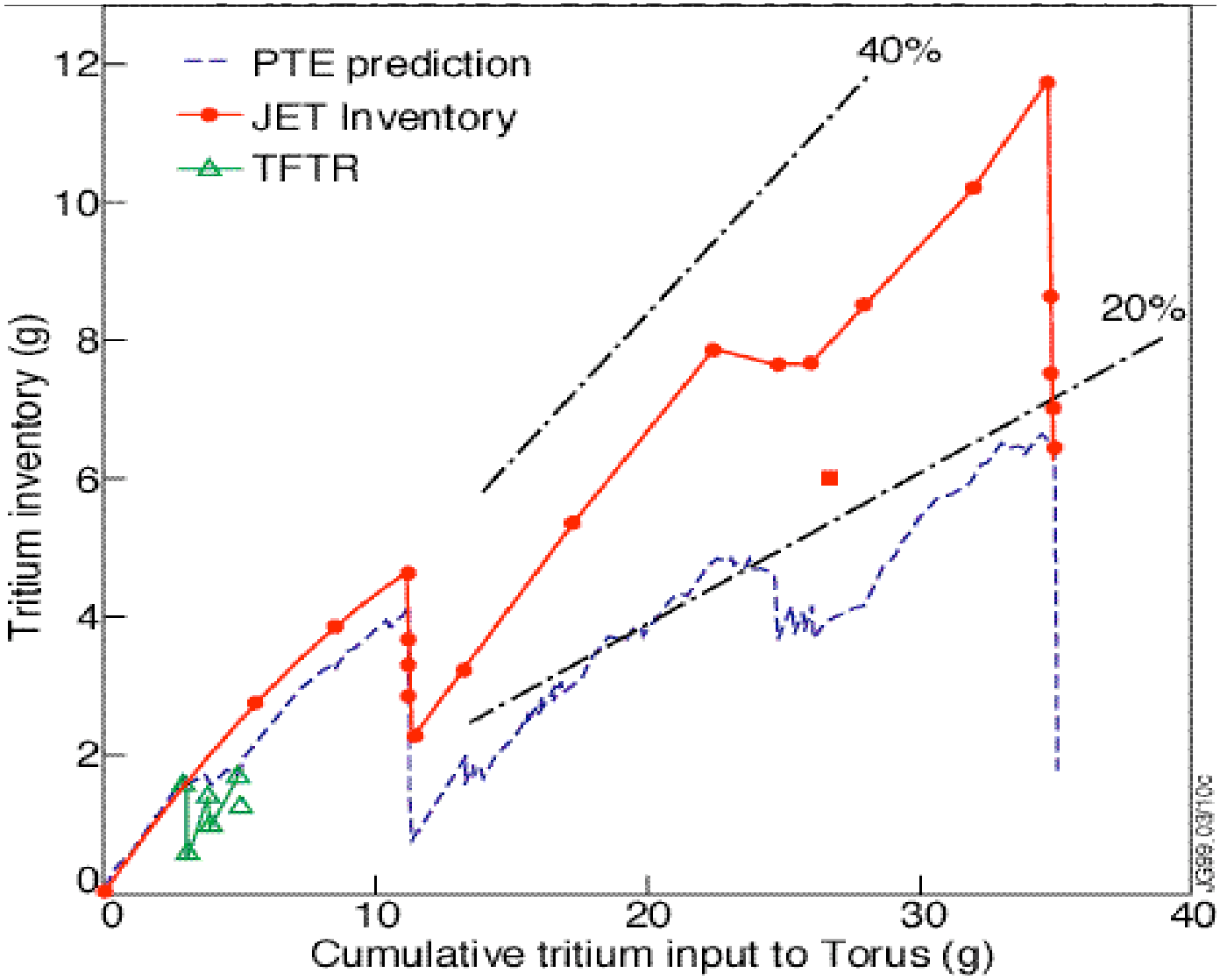


- JET DTE1 experiments 1997, (PTE 1991)
- JET has divertor.
- Walls are erosion areas
- Walls are heated 150-320 C.

	TFTR edge plasma	*JET divertor plasma*
Ne	*0.1 e19 – 1 e19 m-3*	*≈ 10 e19 m-3*
Te	*200 - 600eV*	*<30 eV*

Prompt retention rate higher than expected

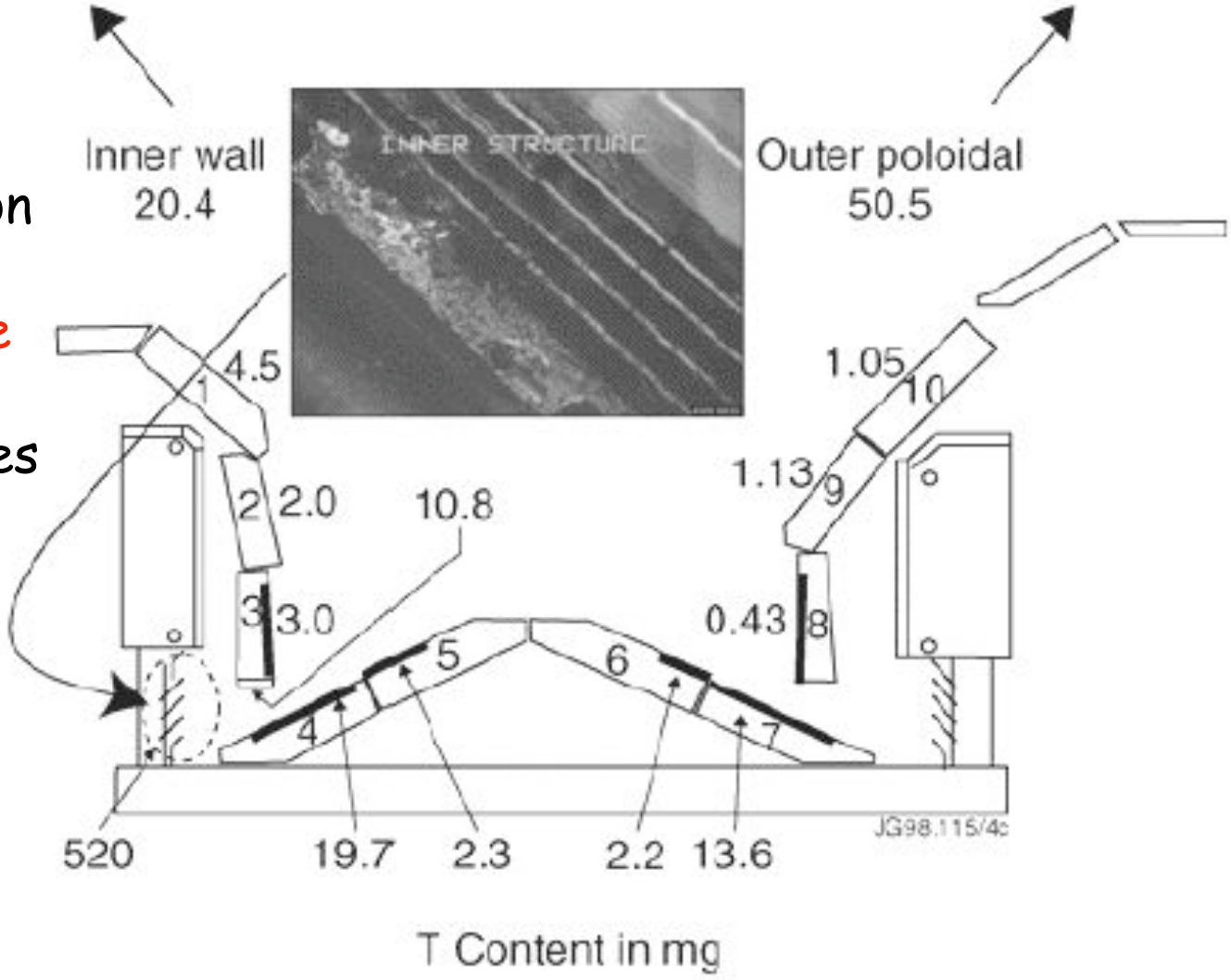
Gas balance measurements for DTE1



P. Andrew et al., Fus. Eng. Des., 47 (1999) 233

Tritium in JET tiles

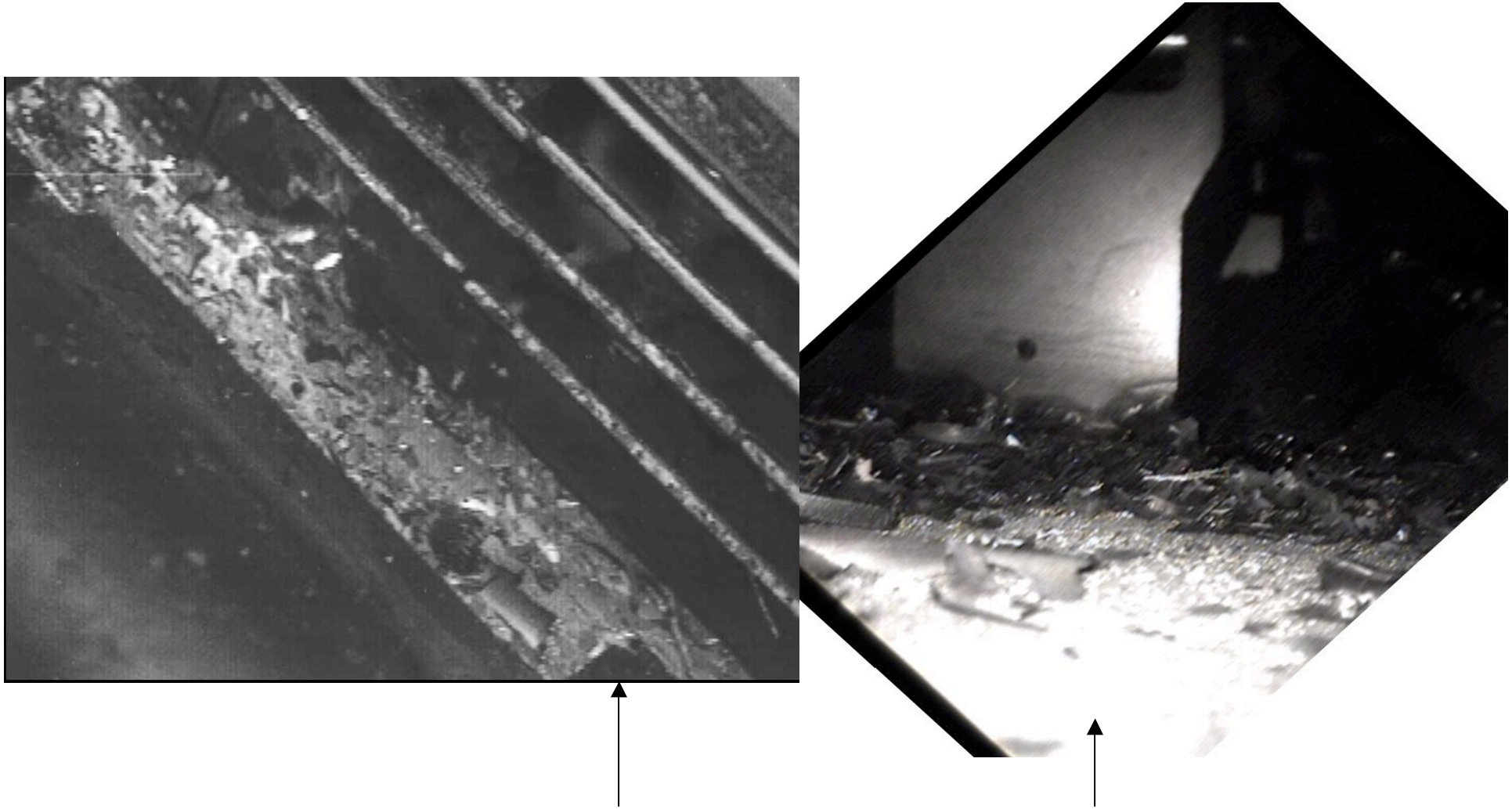
The majority of carbon deposition and tritium retention is on remote areas louvres and shadowed parts of tiles



1 g-T = 9615 Ci

P. Coad et al., J. Nucl. Mater., 313-316 (2003) 419.

Flakes at inner louvres of Mk IIA divertor



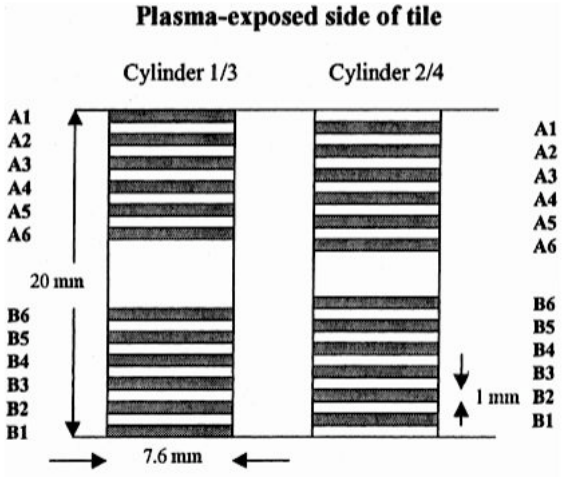
Tritium on the inner divertor louvres (0.5g) and sub divertor region (3.4 g).

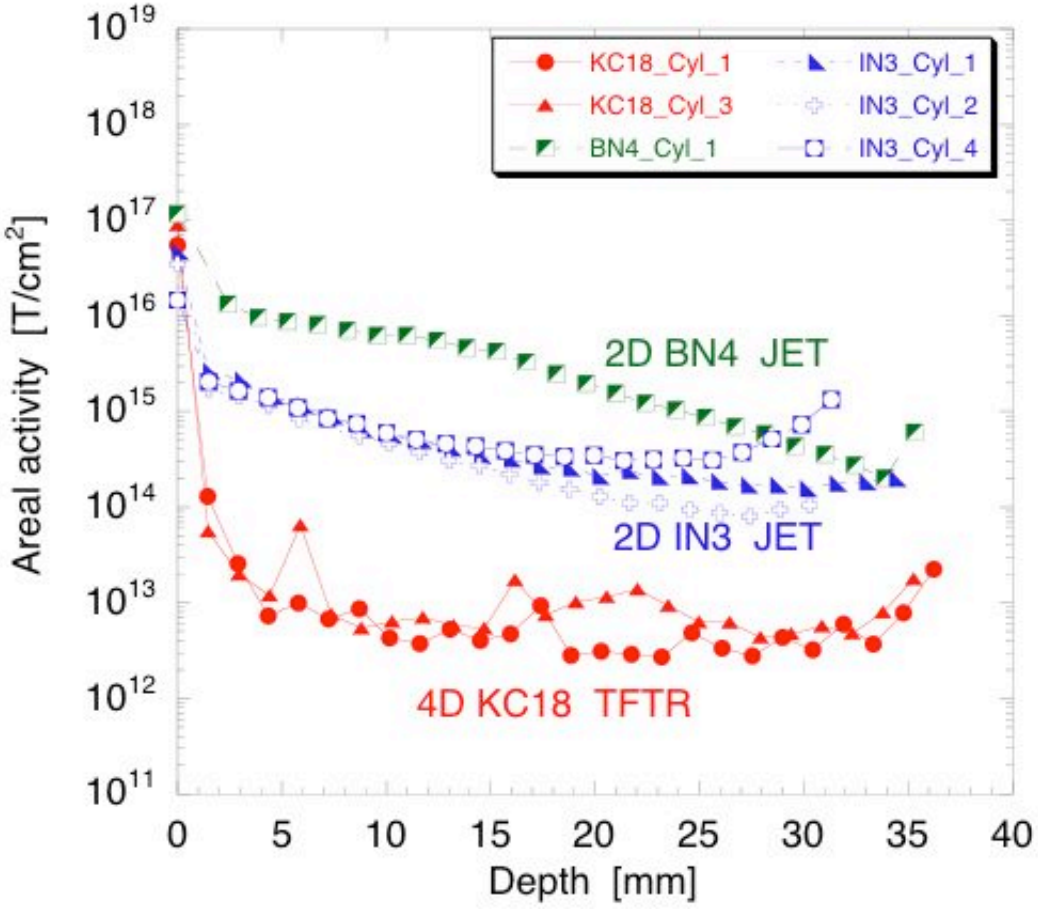
c.f. tiles (<0.1 g)

P. Coad, UKAEA/JET

Tritium can diffuse into carbon tiles

- Core samples of tiles are sliced into 1 mm discs
- These are incinerated to release all tritium.
- Tritium is measured by liquid scintillation counting.
- Results show 61% of retained tritium had diffused deep into bulk of JET 2D CFC tiles.
- This is a concern since removal from bulk is practically impossible.

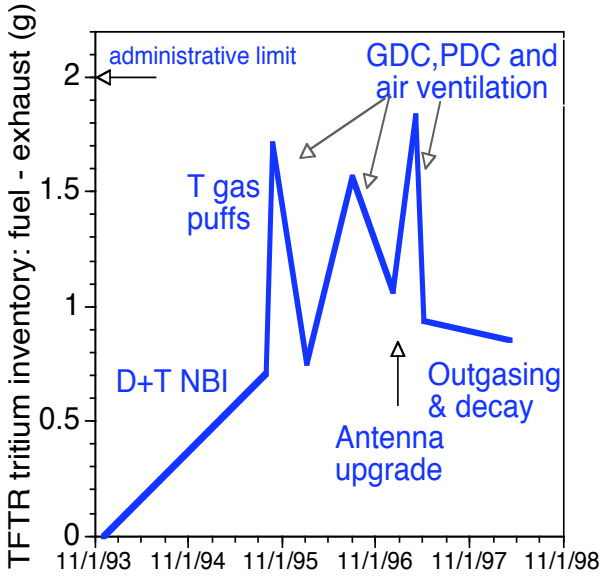


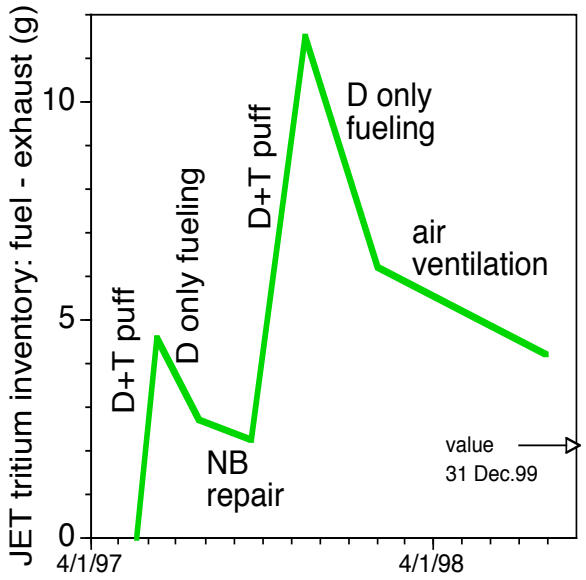


N. Bekris et al., J. Nucl. Mater., 313-316, 501, (2003)

Tritium retention high in TFTR and JET

In TFTR 5 g of tritium were injected into circular plasmas over a 3.5 year period, mostly by neutral beam injection.





In JET 35 g of tritium were injected mostly by gas puffing over a 6 month campaign.

Global Retention:	TFTR:	JET:
Total tritium injected, NBI	3.1 g	0.6 g
gas puff	2.1 g	34.4 g
Total tritium retained during DT operations	2.6 g	11.5 g
Initial % retention during T puff fueling (wall saturation + isotope exchange)	≈ 90%	≈40%
Longer term % retention including D only fueling (mostly co-deposition)	51%	17%
Tritium remaining in torus	0.85 g (4/98)	4.2 g (7/98)
Long term retention	16% (4/98)	12% (7/98) 6% (12/99)

Tritium retention and removal rate in TFTR and JET unacceptable for ITER

Temperature effect: JT-60U experience

Deuterium on tiles very low
D/C ~ 0.01 - 0.05
due to 300 °C wall temperature

Remote D/C ~ 0.75
same as JET 50°C

Retention ~ 1.3 e20 D/s (R~ 8%).

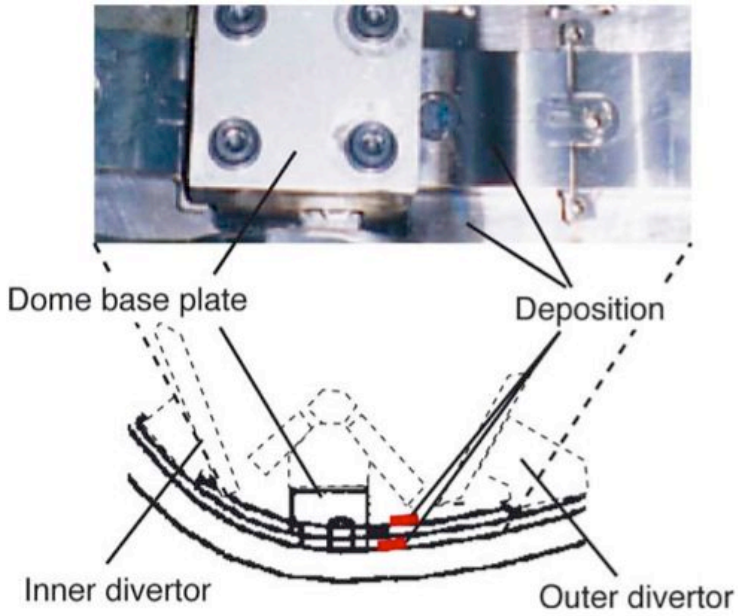
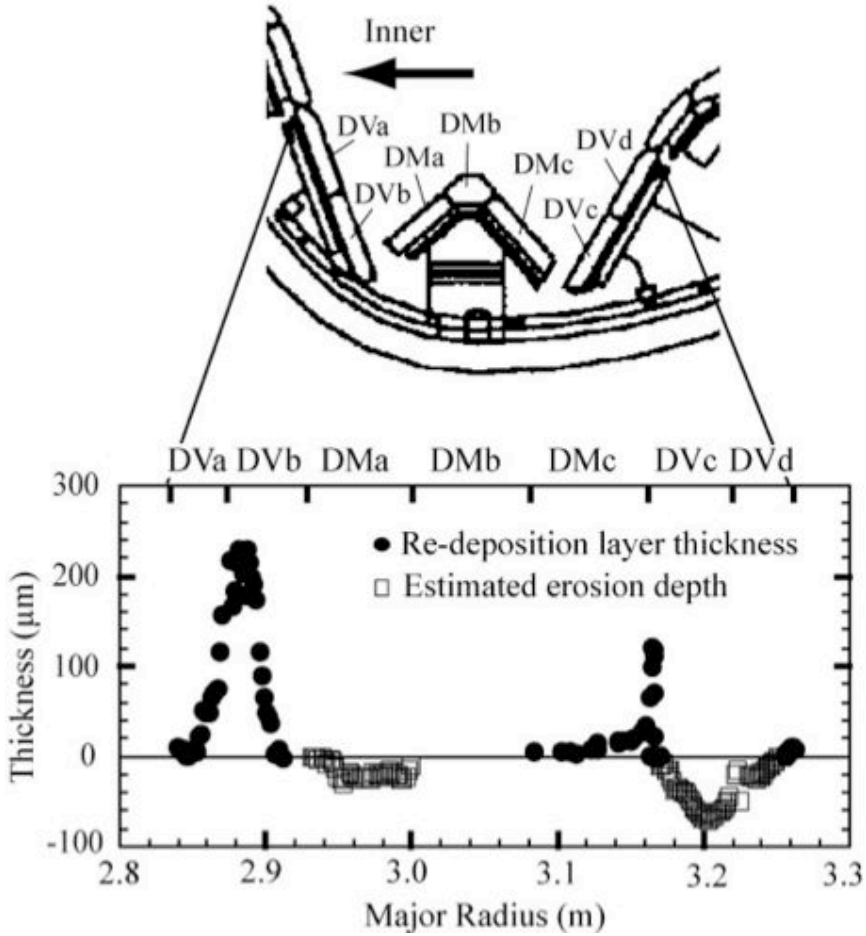


Fig. 8. Deposition underneath the divertor components.

Carbon tiles, 6 years - 8h20m of NBI



Erosion/deposition pattern of the JT-60U W-shaped divertor (1997-2002)

K. Masaki et al., Nucl. Fus., 47, (2007) 1577.
Th. Loarer PSI-18

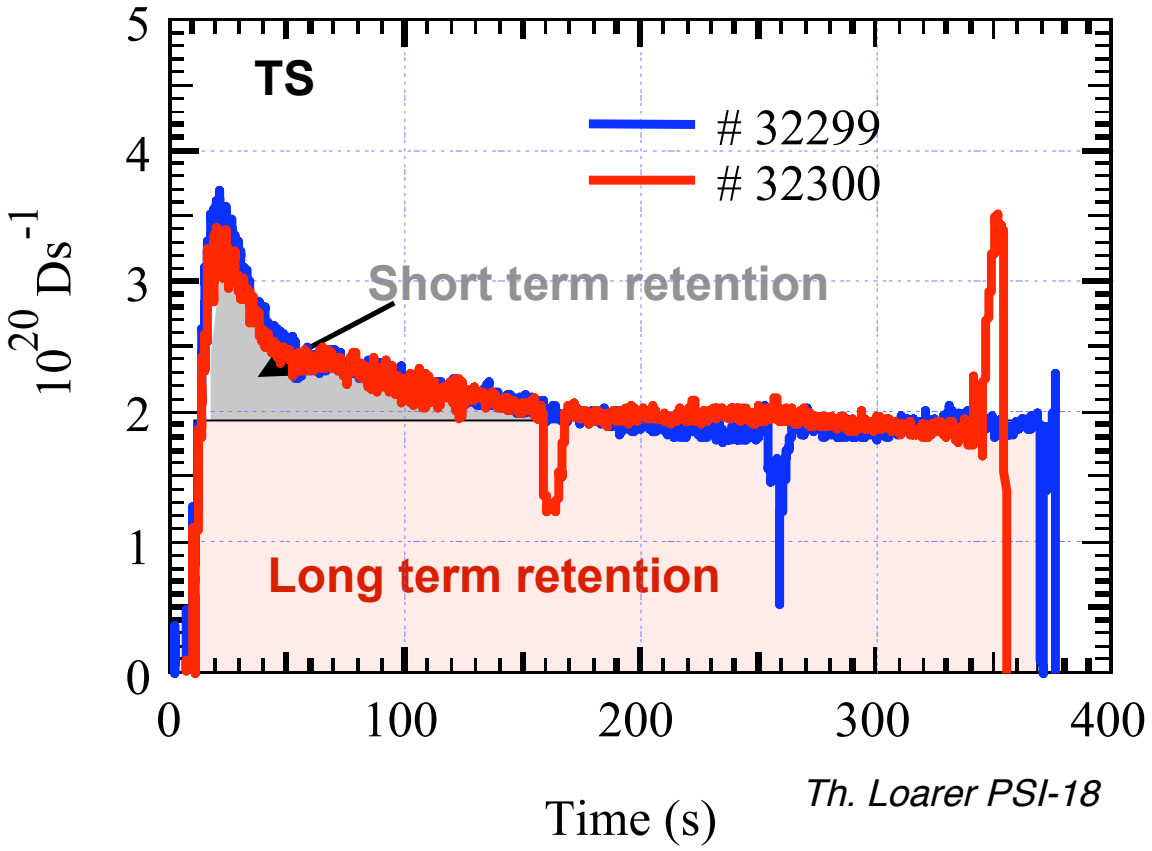
Long pulse effects: Tore Supra experience

Short term retention depends on plasma scenario and wall conditioning.
Limited to fast reservoir and recovered between pulses.

Long term retention:
60% decrease from L-mode to type III H-mode
60% increase from L-mode to type 1 H-mode.
=> correlated to carbon erosion

Steady state retention and wall inventory proportional to plasma duration (up to 5 h)
- no saturation.

Long pulse, actively cooled circular tokamak with carbon tiles



Th. Loarer PSI-18

Carbon plasma facing components
-> continuous increase of tritium inventory with plasma duration via codeposition

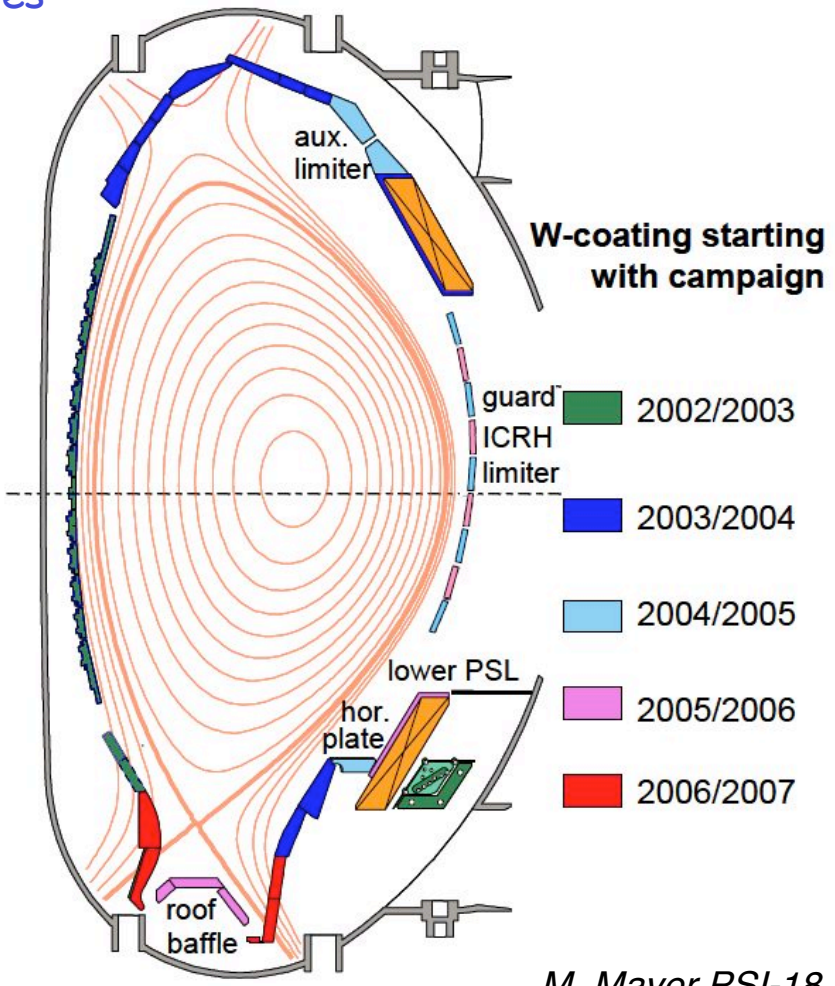
Encouraging results with metals at Asdex-U

Step by step replacement of C tiles by W coated tiles

- 3 – 4 μm W-PVD on most tiles
- 200 μm plasma-sprayed W at outer strike point

Results very encouraging:

- Decrease of trapped D in divertor by factor 5 – 10 from C-dominated to all-W machine. Retention ≤ 1%
- D-inventory in C-dominated machine determined by co-deposition in inner divertor + remote areas.
- D-inventory in all-W machine determined by deep diffusion into W at outer strike point.

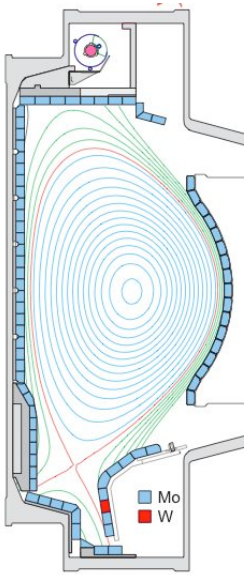


M. Mayer PSI-18

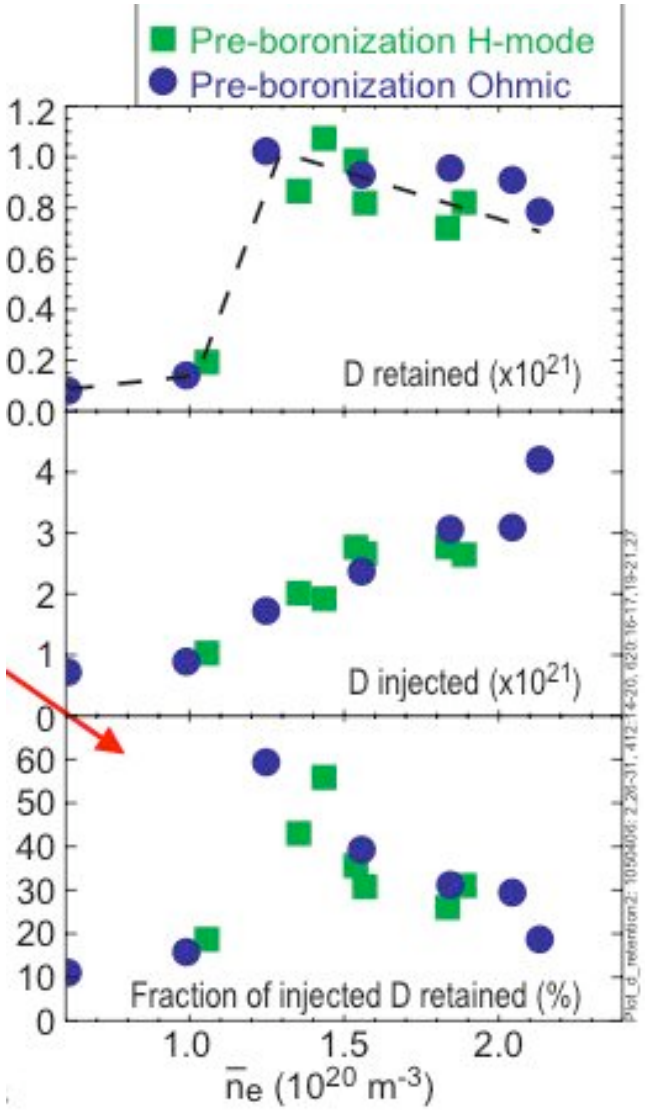
Control of the impurity transport in the plasma centre (ECRH) and in the H-mode edge barrier (ELM frequency) allows to achieve H-mode discharges with H-factor=1.2 and W concentration below 2E-5.

Surprising results from C-mod with Mo, W

- Short-term retention surprisingly similar to carbon PFC tokamaks single-discharge retention in magnitude (10-50%).
- Long-term retention: Much lower than for a single discharge.
  - Disruptions are clearly lowering retention
- Impacting ions appear to be changing the material's retention properties - creating, or expanding, traps for H retention
- Not clear how either short- or long-term retention scale to other conditions ITER, or reactor.



Single discharge retention
(pumping valves closed)



B. Lipschultz PSI-18

JET ITER-like wall will get experience with Be/W tiles as envisaged for ITER DT experiments

- Why are tritium and dust important ?
- TFTR & JET tritium experience
  - H retention in other tokamaks
- Tritium removal
- Projections for ITER

ITER scale up in duty cycle and tritium usage is larger step than change in core plasma parameters

Parameters:	TFTR experience	JET experience	ITER projections
Tritium in-vessel inventory limit	2 g	20 g site inventory	700 g limit
Typical pulse duration	≤ 8 s	30 s	400 s
Tritium retention rate (JET/TFTR including D only pulses)	51%	17%	1.4 – 5 % w/ carbon target plate
Cumulative DT discharge duration before inventory limit first approached.	708 pulses ≈ 33 min	500 pulses ≈ 250 min	250 -1,000 pulses 1,600 - 6,600 min
Period before inventory limit approached.	22 months	≈ 3 months	few weeks
Time devoted to tritium removal etc…	1.5 months	3 months	weekend (?)
Fraction of tritium removed	50%	50% (prior to venting)	should be close to 100%
Tritium removal rate	~ 1 g /month	2 g / month	≈ 10 g / h

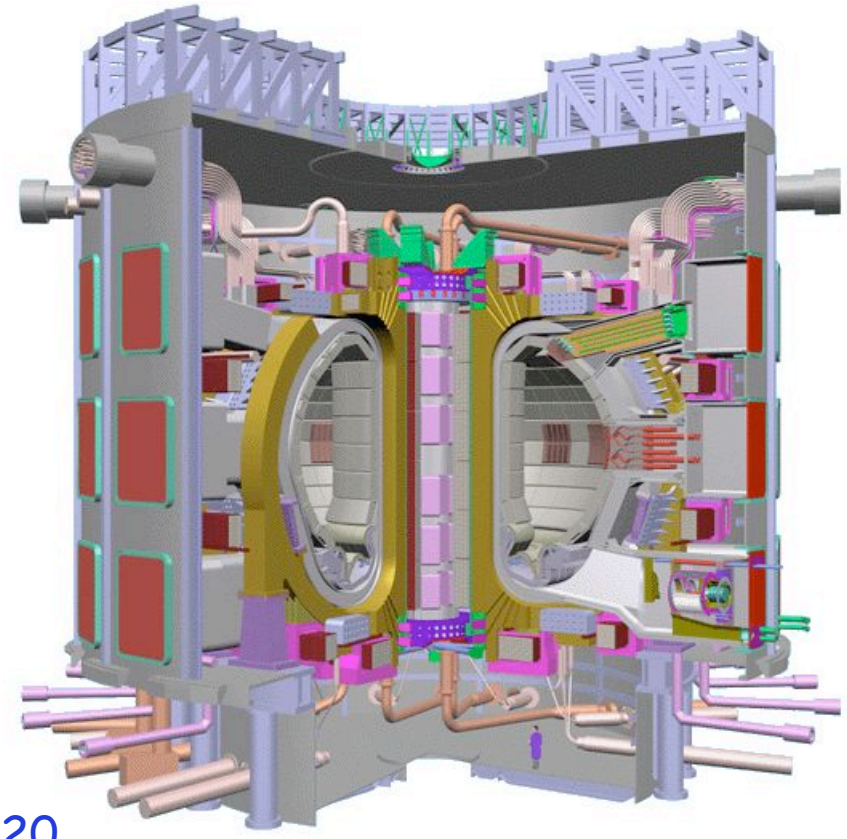
Bottom line:

- **Need to demonstrate in tokamaks rapid and efficient method to remove tritium at > ~ 1,000 scale up from TFTR & JET.**

Tritium removal options

Potential Options

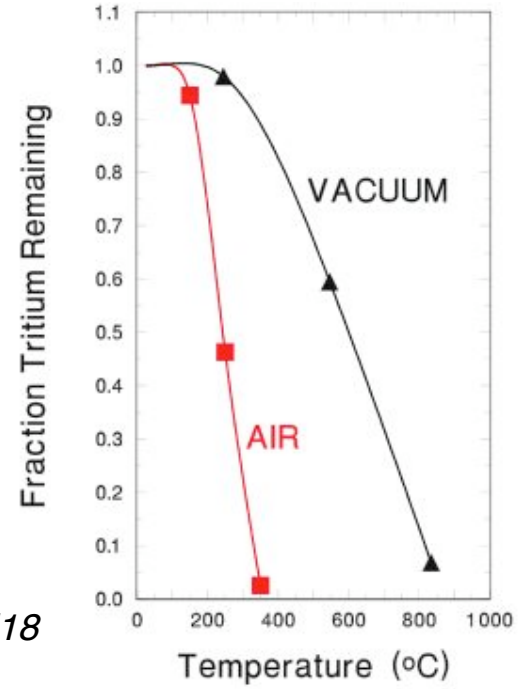
1) Remove whole codeposit by:
   - oxidation (maybe aided by RF)
   - ablation with pulsed energy (laser or flashlamp).
2) Release T by breaking C:T chemical bond:
   - Isotope exchange
   - Heating to high temperatures e.g. by laser



- Constraints:
  - 6.1 Tessla field at inner divertor
  - 10,000 Gy/hr gamma field from activation, 3 h after shutdown, after 20 years DT ops.
  - Access difficult, especially to hidden areas

Tritium removal by oxidation:

- Oxygen can remove carbon codeposits by oxidation to DTO, CO_2, CO.
- Removal rate depends on film structure - codeposits removed ~ 100x faster than manufactured tile
- D removal rate independent of codeposit thickness and Be content.
- Some experience on TFTR, JET, TEXTOR



Davis PSI18

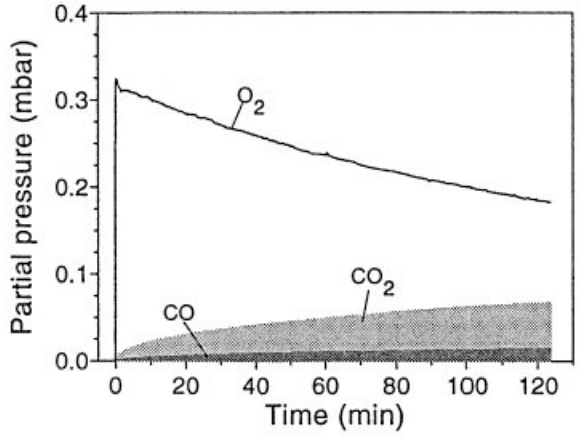
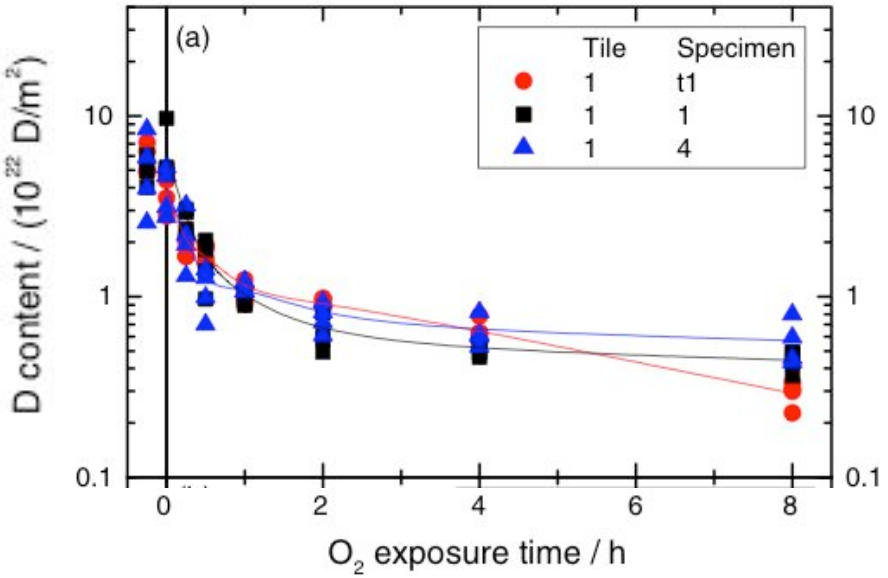


Fig. 2. Temporal behaviour of the partial pressures of O_2, CO and CO_2 after a ventilation of TEXTOR with $^{16}O_2$ to an initial pressure of 0.32 mbar. All external pumps are closed. Plasma facing wall temperatures range from about 520 to 650 K. (For more details see inside text.)

V. Philipps et al., J. Nucl. Mater., 266-269 (1999) 386.



Tsui et al., Nucl. Fusion 48 (2008) 035008
Also review by Davis in Physics Scripta T91, 33 (2001).

Tritium removal by oxidation (2):

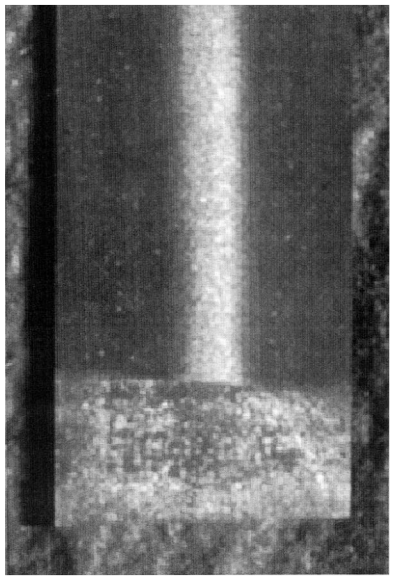
MERITS:

- Lab experience, some tokamak experience.
- access to all areas in vessel.
- Simple to implement, no in-vessel hardware.

LIMITATIONS:

- Temperature required, ~ 350 °C, not compatible with water cooling
  - additional baking system needed -$$$.
- No experience with repeated oxidation /deposition cycles.
- Potential for collateral damage to in-vessel components.
- DTO produced is 10^4 x more hazardous than T_2 and needs substantial investment in tritium plant to process.
- Not appropriate for BeT codeposits.
- Quantitative demo on tokamaks needed (planned for DIII-D).

Removal by ablation using excimer lasers or flashlamps

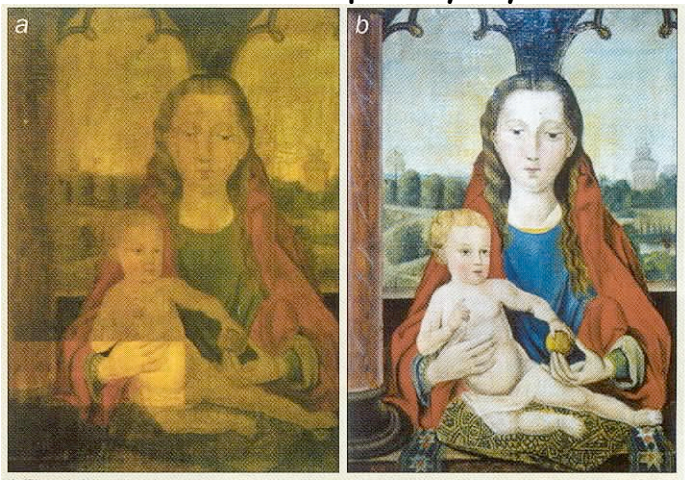


Flashlamp ablation:

CFC tile coated with a 28 μm aC:H film (darker regions). The lower region was masked during film deposition to act as a control. Deposition was removed in-vacuo using 10 pulses from the flashlamp.

G. F. Counsell & C. H. Wu , Physica Scripta T91 (2001) 70.

Art restoration partly by laser



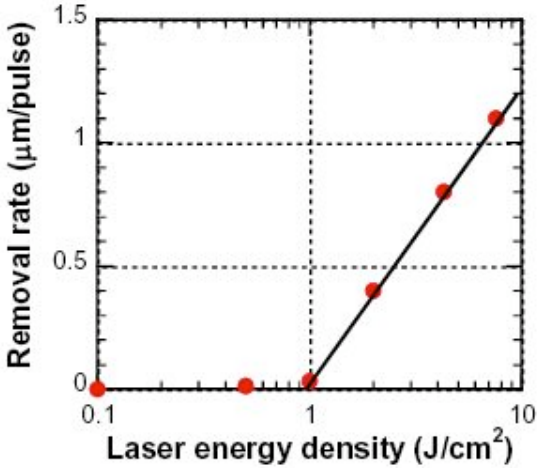
A Flemish painting cleaned using an excimer laser. (*a*) The original state of the painting. The yellowing is due to the aging of the varnish. The small area surrounding the Madonna's right hand has been laser treated to remove the top insoluble layer of polymerized varnish. (*b*) The painting after it was treated with the laser and the deeper layers of varnish were subsequently removed using traditional techniques. Photos courtesy of V Zafiropulos, Foundation for Research and Technology Hellas and M Doulgeridis, Conservation Department of the National Gallery of Athens.

K Hinsch & G Gülker Physics World Nov 2001 p.37

Excimer laser ablation:

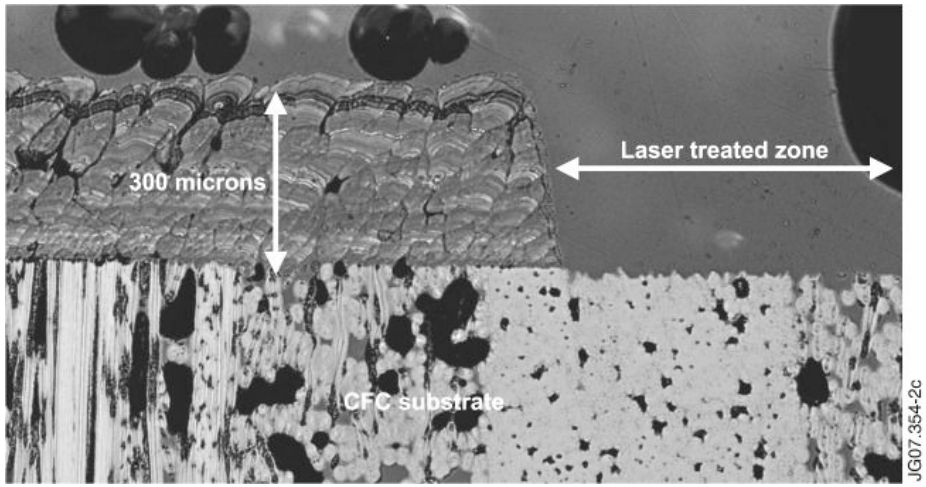
ArF laser removes JT60 codeposits

Shu et al., JNM 313 (2003) 585



Jet codeposit removed by 3 passes of Nd-YAG laser 100 μ spot, 120 ns 2 KHz.

A Widdowson et al., JFST 2008



Tritium removal by ablation (2)

MERITS:

- Lab & industrial experience,
- Whole codeposit removed

ISSUES:

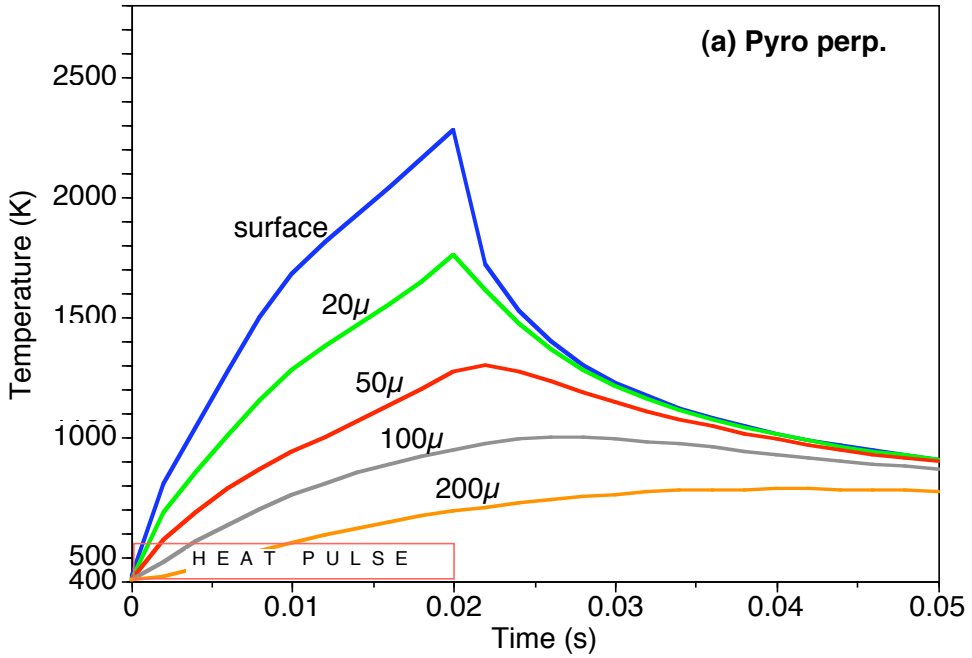
- In-vessel hardware needs to be developed.
- Access to hidden areas, tile gaps…?
- Fate of ablated products is major issue
  - Potential for dust & debris to fall in inaccessible areas (as in JET subdivertor).
  - Reactive radicals may be produced that would redeposit in-vessel
- Compatibility of in-vessel hardware with 6.1 T and 10,000 Gy/h field ?
- No in-tokamak demonstrations at required speed and efficiency

Detritiation by laser surface heating

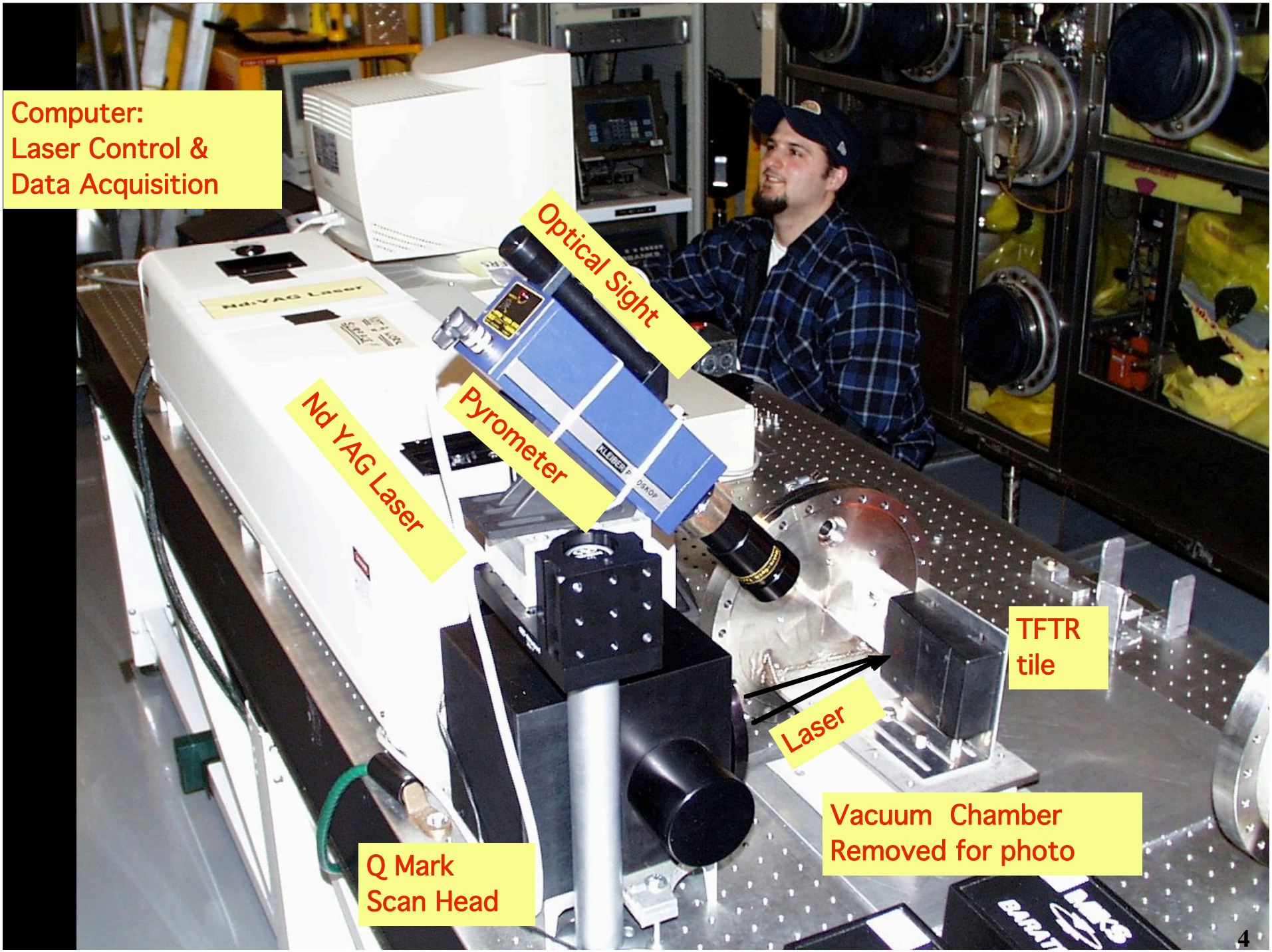
- Heating is proven method to release tritium but heating ITER vacuum vessel to required temperatures (~350 C) is incompatible with water cooling.
- But
  - most tritium is codeposited on the surface
  - only surface needs to be heated.
  - Modeling showed lasers could provide the required heating
  - Technique has been validated in extensive lab experiments on JET and TFTR tile samples

Modeling results:

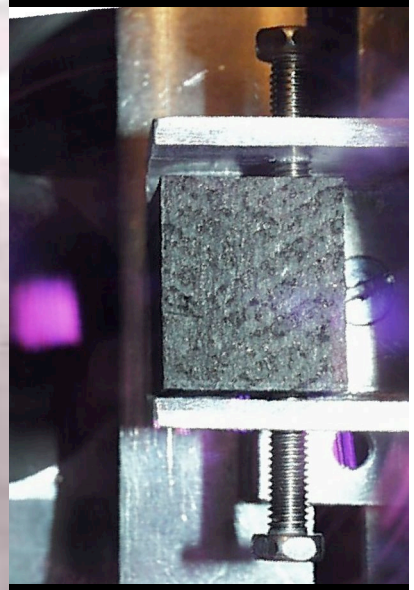
Temperature vs. time at different depths into pyrolitic perp. under 3,000 w/cm2 for 20 ms.



3000 w/cm² flux for ≈ 20 ms heats a 50 micron co-deposited layer to 1,000-2,000 K, appropriate for tritium release


Computer:
Laser Control &
Data Acquisition
Optical Sight
Pyrometer
Nd YAG Laser
TFTR
tile
Laser
Vacuum Chamber
Removed for photo
Q Mark
Scan Head

Nd laser in action:



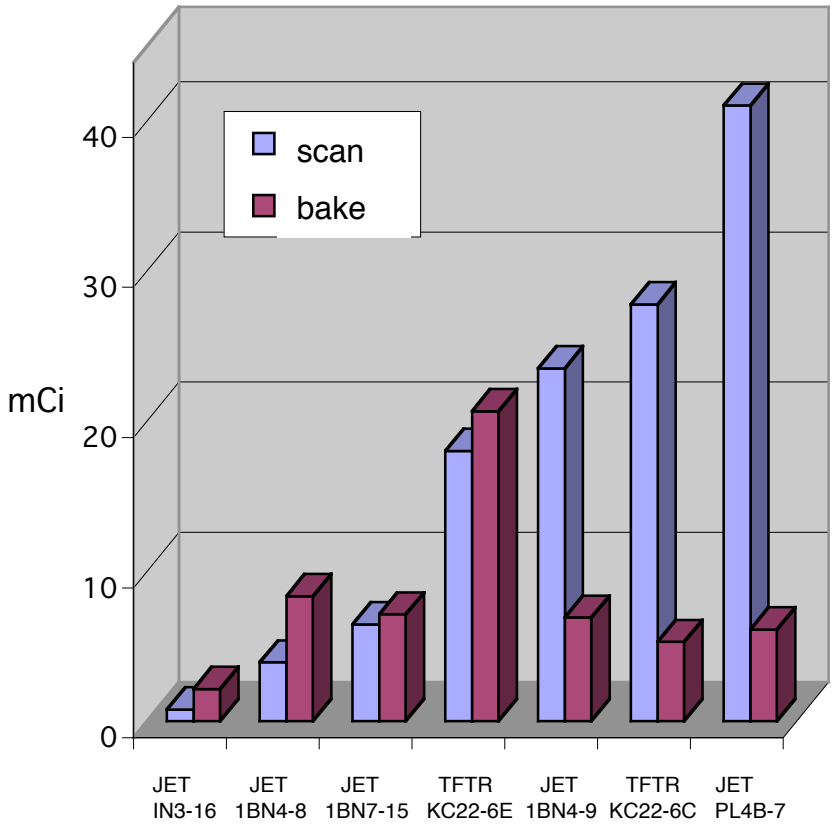
7/8” cube cut from TFTR tritiated tile inside chamber.

(KC17 2E)

Nd laser power only 6 w to avoid camera damage (300 w available)
TFTR DT tile cube KC17 2E in air at 200 mm/s.

How much tritium is released ?

- Scanning Nd laser heats codeposit surface to ≈2,000 C and thermally desorb tritium.
- Release fraction up to 87%
- Detritiation efficiency highest in regions of heavy deposition.
- remaining tritium measured by laser 'baking'.



Conclude: major part of co-deposited tritium can be released by scanning laser.

Detritiation by laser surface heating

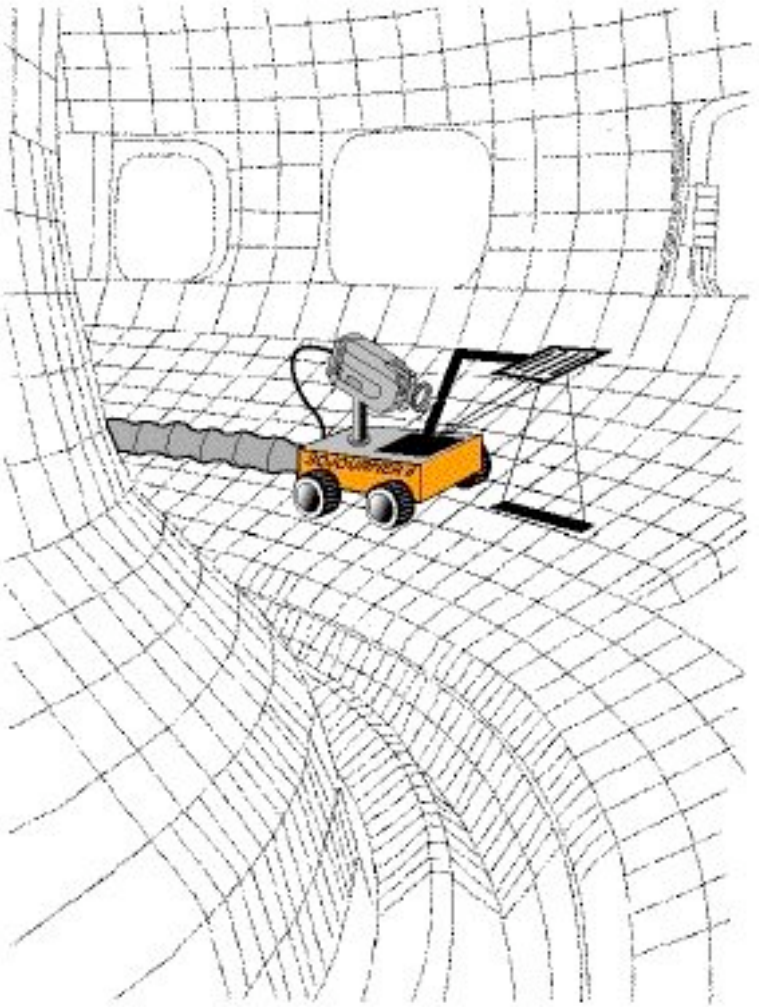
Application to ITER ?

- Fast cleanup - 6 kW laser can deliver energy to heat 50 m^2 surface in 3 hours in next-step device.
- Convenient fiber optic coupling.
- no HTO to process (HTO is 10,000x more hazardous than T_2 and expensive to reprocess).

Remaining issues:

- Development of miniaturized scan head for hard-to-access areas.
- Tokamak demonstration with remote handling and with plasma 'conditioned' codeposits.
  (funding needed)

Concept of potential in-vessel hardware



J. Nucl. Mater 313-316 (2003) 496.

Other methods:

Technique	Merits	Limitations
Glow discharge cleaning	Tokamak experience	Incompatible with 6 T field
ICRH	Tore Supra experience 4e22 C/m^2/h -> 1 μm/h	no access to shadowed areas collateral sputter damage
ICRH or ECRH + oxygen	Atomic O formed @ SNL ECRH 3.6 μm/h removal at 620K in Garching lab.	Time to recondition walls ? collateral damage ? HTO processing ? Access to hidden areas ? (contribution of neutrals)
N_2 scavenger gas	Inhibits codeposition	R&D needed.
Cathodic arc cleaning		Damage to underlying tile ?
CO_2 pellets		Damage to underlying tile
UV light		Ineffective
Ozone		Dissociates at 250 C.
Flame detritiation	effective	Only suitable ex-vessel

T removal rate required for ITER not yet demonstrated in tokamaks

ITER 2007 Design Review

ITER Design Review WG8 Interim Report 3 July 2007:

"The management of the in-vessel tritium and dust inventory.

- *The group understood that the issues of tritium and dust have been long standing, and that there is no obvious solution. However, it finds that the complete lack of a strategy, and the absence of any concrete provision in the design for potential measurement or cleaning techniques to be a serious risk to the achievement of the project goals.*"

Cambell email Sept. 27th, 2007

- "…the (ITER) project has taken a decision to follow a strategy in which we will plan to install an initial CFC divertor to get the plasma operations underway, but we will change to a tungsten divertor for the DT phase. …
- The essential reason for the proposed strategy is that there is no solution on the horizon to the problem of tritium removal from a device with Carbon PFCs and we see no evidence that the parties are preparing a focussed R&D programme which would develop a solution on the appropriate timescale.
- There is, on the other hand, a well-defined R&D programme in the EU, through AUG and JET, to learn how to operate the relevant plasma scenarios with tungsten PFCs.
- So while there remain issues which need to be resolved in the use of tungsten, there is an R&D programme in place to deal with them, whereas this is not the case for tritium removal in a carbon 'environment'."

- New R&D focus on remaining issues for T retention:
  - beryllium codeposits and retention in neutron damaged tungsten.

- Why are tritium and dust important ?
- TFTR & JET tritium experience
  - H retention in other tokamaks
- Tritium removal
- Projections for ITER

Dust generation

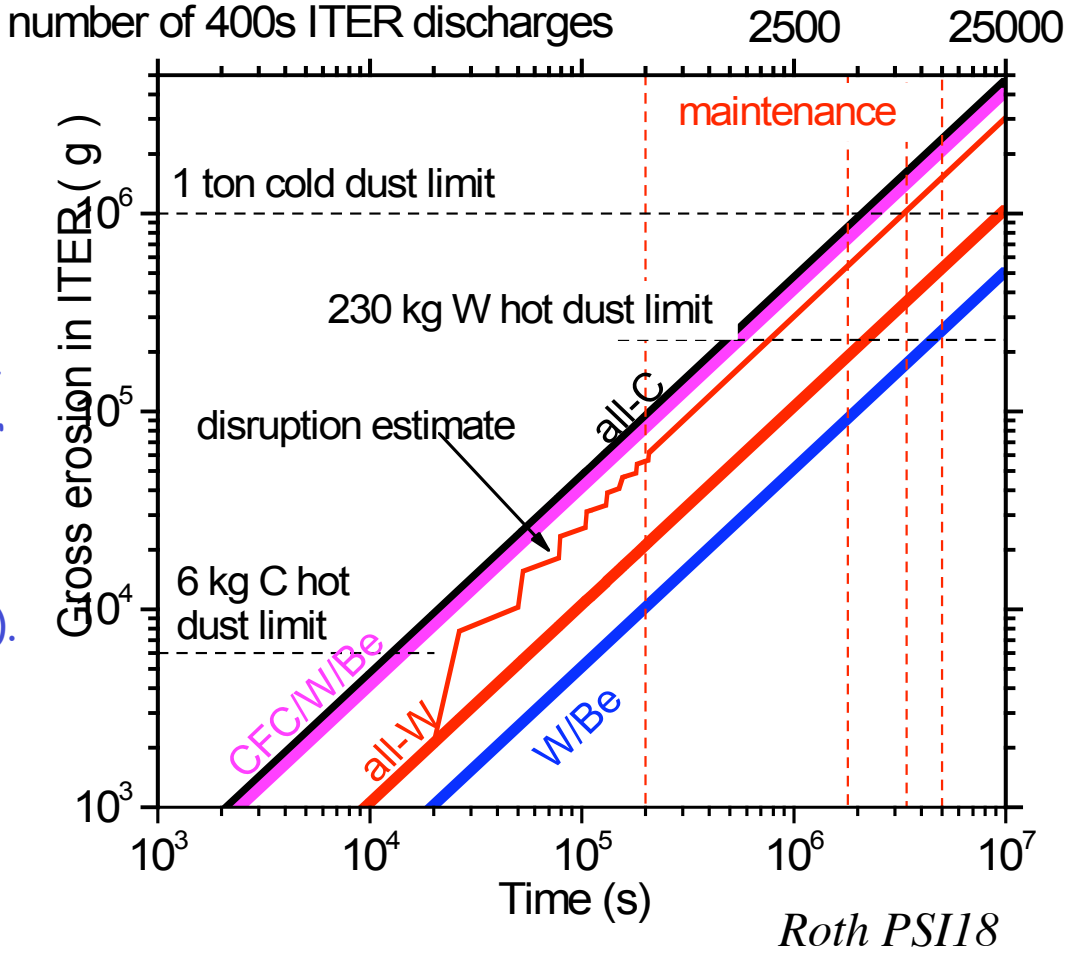
Dust generated by erosion, deposition, layer disintegration.
Dust inventory limit reached on same time scale as divertor erosion lifetime.

Present ITER strategy:

Remove dust when divertor is replaced (every ~ 4 years).

Monitor dust during shutdowns by:

1. Divertor erosion monitor (assume 100% conversion from erosion to dust for safety assessment.
   (10% in Tore Supra and JT-60U).
2. Local dust monitor (to be developed).
3. Local tritium measurement (tbd).
4. Global dust reactivity measurement.
5. Benchmarking in H-phase.



Roth PSI18

Dust detection and dust removal techniques needed !

P. Andrew

ITER retention depends on material choice

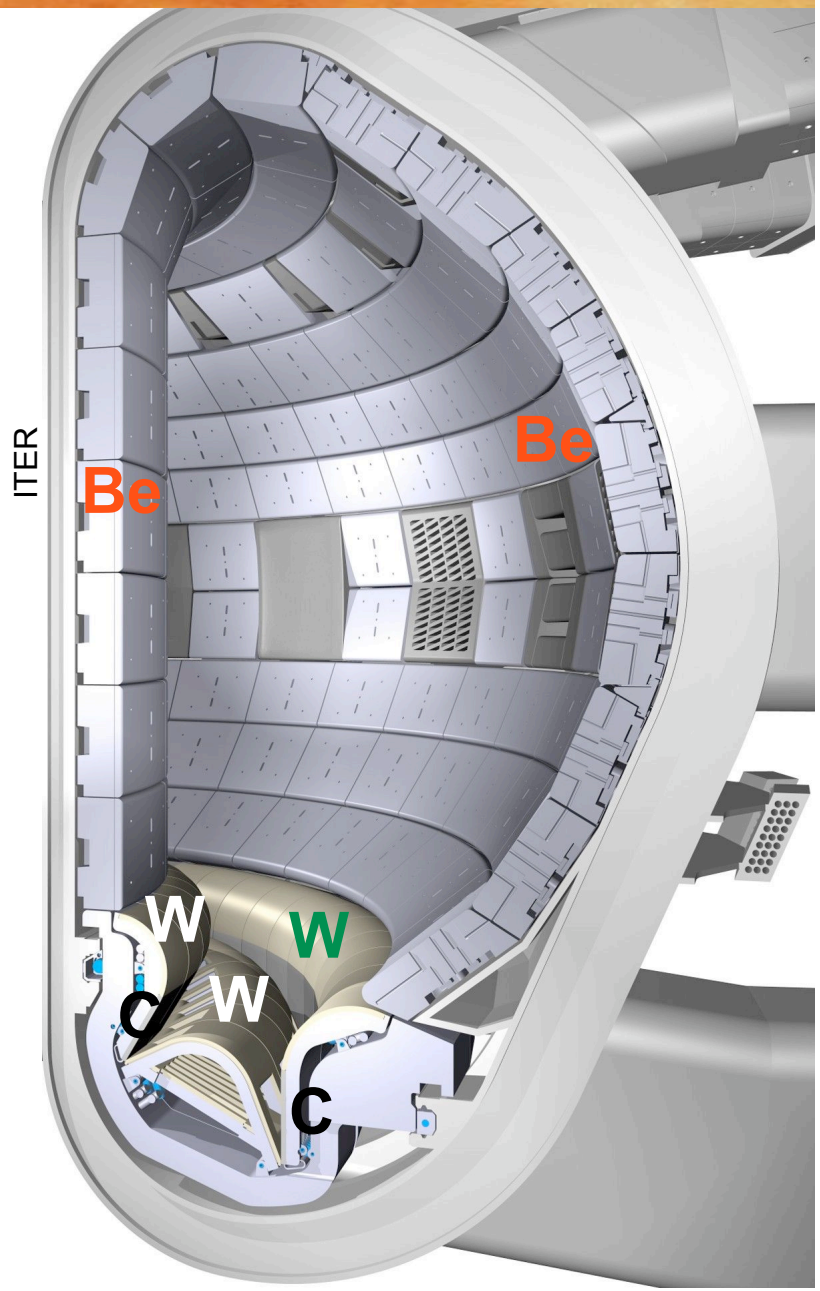
Present ITER strategy:

Initial hydrogen/deuterium phase:

- Beryllium wall, 700 m^2
  (low Z = low radiation losses, oxygen getter, but low melt temperature)
- Tungsten baffle and dome, 100 m^2
  (high melt temp, low erosion, low T retention, but high rad. losses)
- Carbon divertor target 50 m^2
  (does not melt, good radiator for plasma detachment, but T retention is major issue)

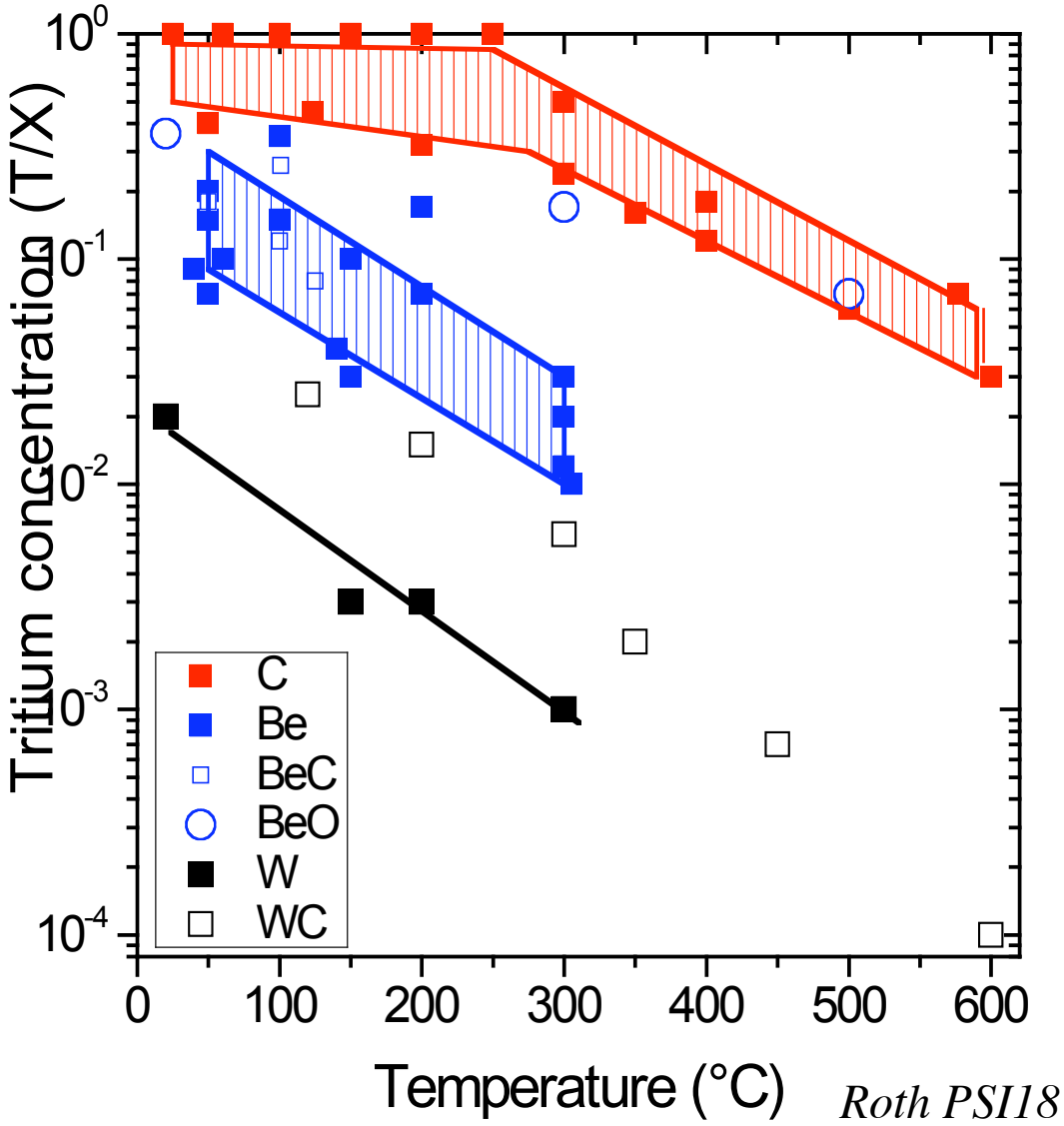
Before DT operation

- Change to full tungsten divertor.
- Timing depends on experience with H retention and dust
- All-W as future DEMO relevant choice



Erosion > co-deposition > tritium inventory

- Retention in ITER was evaluated by calculations of erosion, impurity transport including re-erosion, codeposition for various combinations of plasma facing materials
- Tritium concentration depends on final deposition conditions.

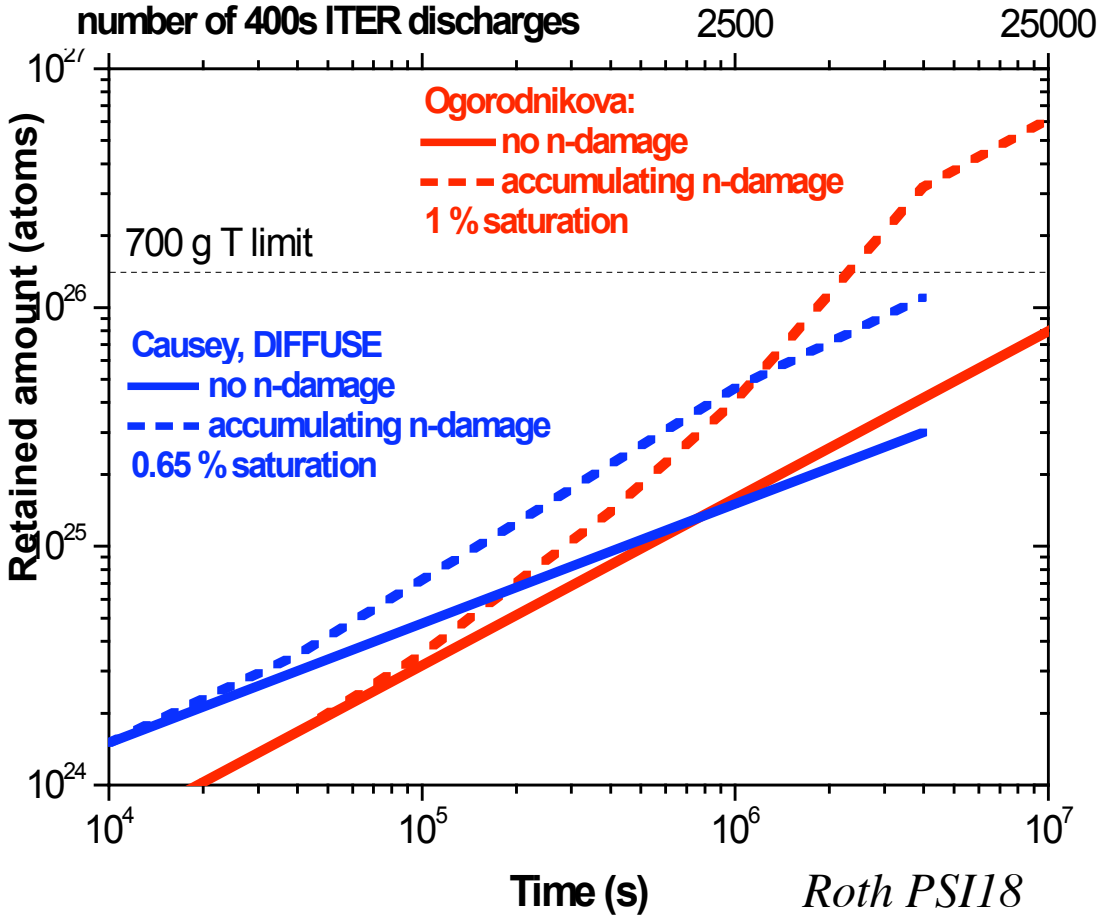


Roth PSI18

Implantation: D in W divertor tiles

- Code calculations (Ogorodnikova) based on experiments.
- Neutron irradiation assumes saturation at 1% additional trap sites.
- DIFFUSE code (Causey) predicts square root fluence dependence with and without neutron induced traps.

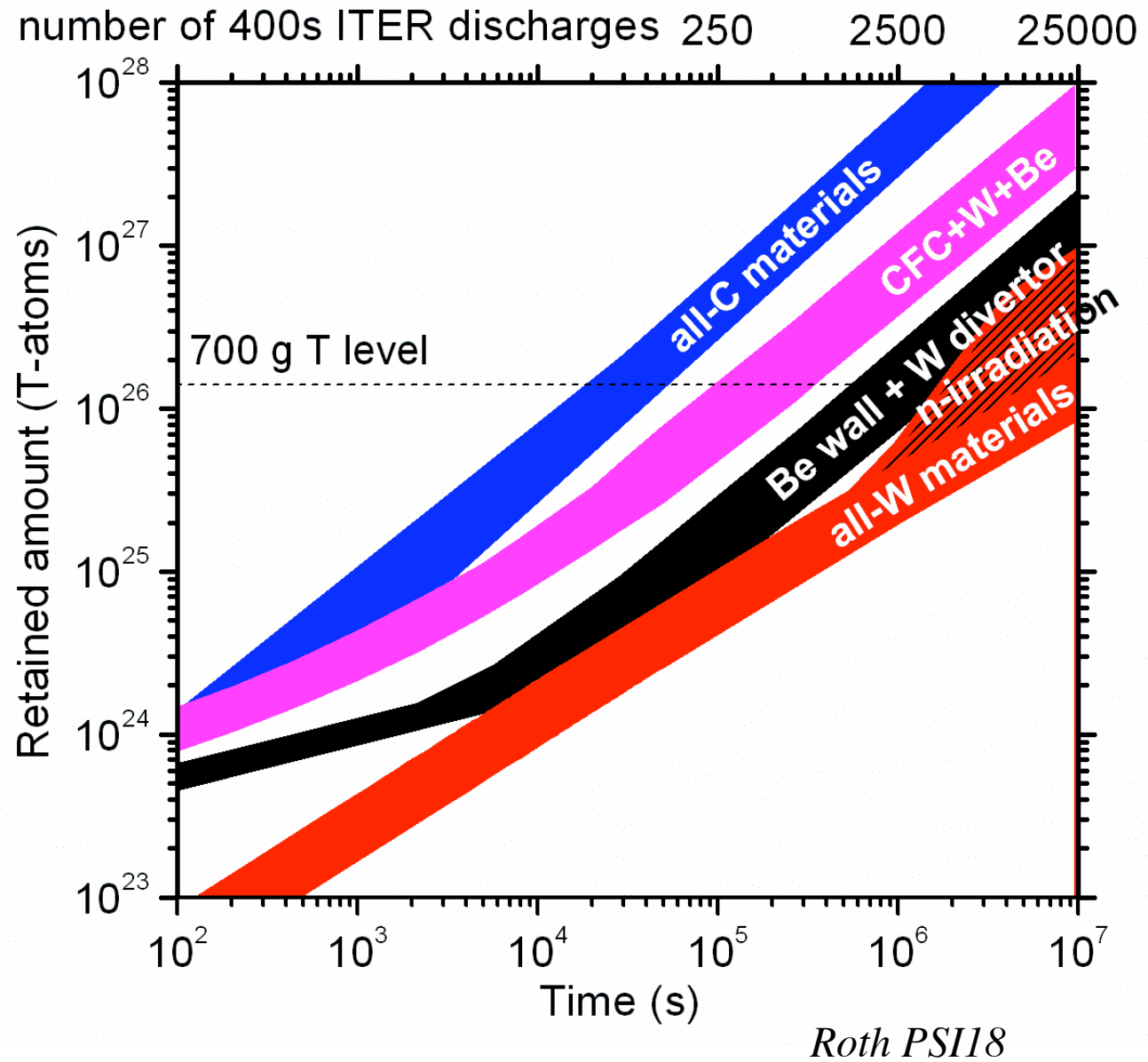
- Good agreement for case without neutron damage.
- Main uncertainties are in estimate of neutron displacements per atom (dpa) and hydrogen trapping.



Implantation + codeposition

Recent EU assessment of tritium inventory in ITER for various PFC material options
(to appear in PPFC)

Similar, independent plot by ITPA SOL/Div group (to appear in 2008 IAEA proceedings).



Summary:

- Managing tritium inventory is a challenge for ITER and future DT reactors.
- Strongly related to constraints on PFC erosion lifetime and dust generation, - essentially an astrophysical/terrestial interface issue.
- ITER strategy is to begin with Be wall, W divertor baffle and dome and CFC divertor target plate.
  - then switch to all W divertor to avoid serious loss of machine availability and risk of tritium removal.

High priority R&D:

1. Understand influence of neutron damage on tritium inventory in tungsten.
2. Extend experience with all metal machines - C-mod, Asdex-U and JET ITER-like wall experiment.
3. Control plasma wall interaction - mitigate ELMs and disruptions.
4. Demonstrate hydrogen isotope removal at relevant rate and efficiency on tokamaks.

Time to develop new energy sources is short….

J.Comiso

- The Arctic perennial ice cover has been decreasing at 9 to 10% per decade.
- Polar bears may be extinct by end of 21st century.
- Many Carribean reefs have seen a 80 % decline in coral reef cover partly due to global warming

References in Literature:

1. **"Recent advances on hydrogen retention in ITER's plasma-facing materials: Be, C, W."**
   C.H. Skinner, A.A. Haasz,V. Kh. Alimov, et al.,
   to be published in Fusion Science & Technology, November 2008
   http://www.pppl.gov/pub_report//2008/PPPL-4308.pdf
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   G. Federici, C. H. Skinner, J. N. Brooks, et al.,
   Nucl. Fus., 41, 1967 (2001).
3. **"Tritium Inventory in the Materials of the ITER Plasma Facing Components"**
   G. Federici and C. H. Skinner, in Nuclear Fusion Research - *Understanding Plasma-Surface Interactions*, Eds. D. Reiter and R. E. H. Clark, Vol. 78, *Springer Series in Chemical Physics,* ISBN 3-540-23038-6 Springer Verlag, Heidelberg, pp. 287-317 (2005).
4. **"Hydrogen Isotope Retention and Recycling in Fusion Reactor Plasma-facing Components,"**
   R. A. Causey, *J. Nucl. Mater.*, 300, 91 (2002).
5. **"Hydrogen Retention in and Release from Carbon Materials,"**
   A. Haasz and J. W. Davis, in Nuclear Fusion Research, editors R. E. H. Clark and D. H. Reiter, Springer-Verlag series in Chemical Physics, 2005, pp 225-248.
6. **"Tritium Removal from Carbon Plasma Facing Components,"**
   C. H. Skinner, J. P. Coad, and G. Federici, *Phys. Scr.,* T111, 92 (2004).
7. **"Gas balance and fuel retention in fusion devices"**
   T. Loarer et al., Nucl. Fus. 47 (2007) 1112.
   Also PSI-18 proceedings, J. Nucl. Mater., 2009
8. Copy of these slides available from cskinner@pppl.gov