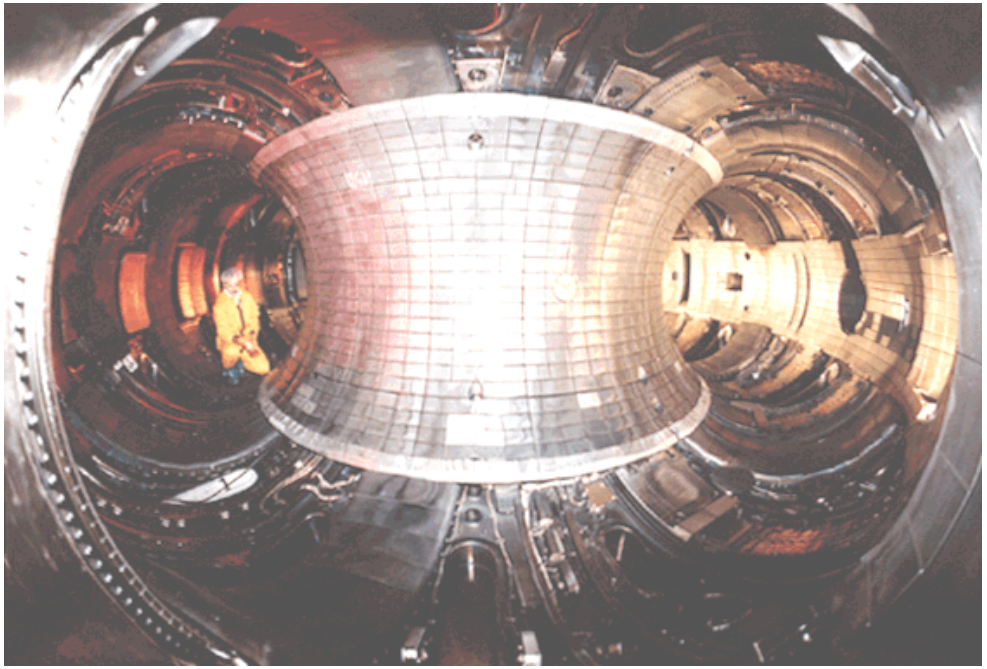
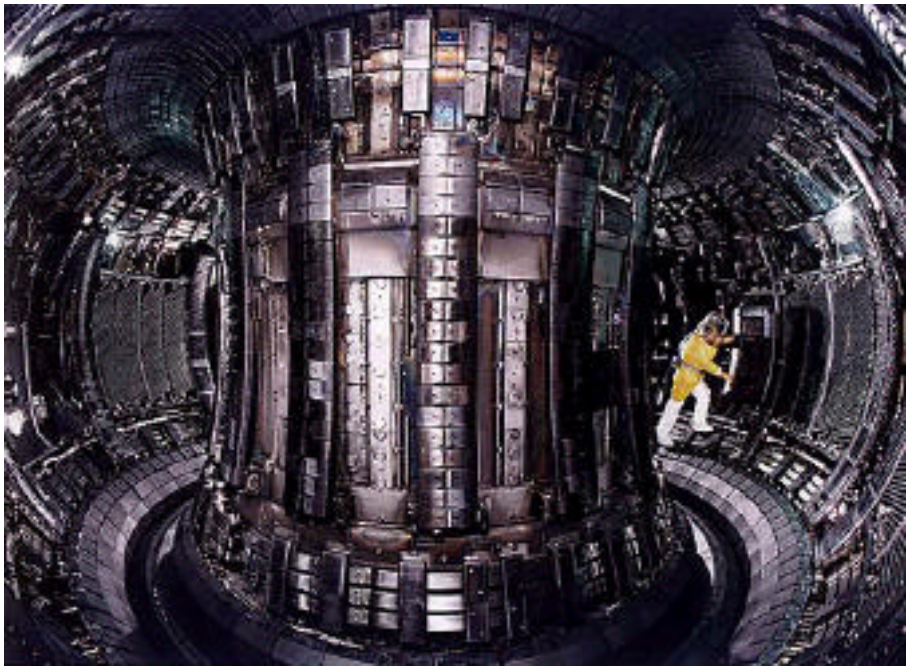
Tritium in Tokamaks - past, present & future

Charles H. Skinner
Princeton Plasma Physics Laboratory
with contributions from
Paul Coad, Gianfranco Federici, Charles Gentile, John Hogan, Yung-Sung Cheng, and many others

- **Synopsis**
  - DT experience of TFTR and JET
  - Retention in C-mod with Mo walls
  - Reactor issues
    - carbon PFC's
    - metal PFCs,
  - New results:
    - Dust & Flakes
    - Laser based tritium removal

Tritium experience of TFTR & JET

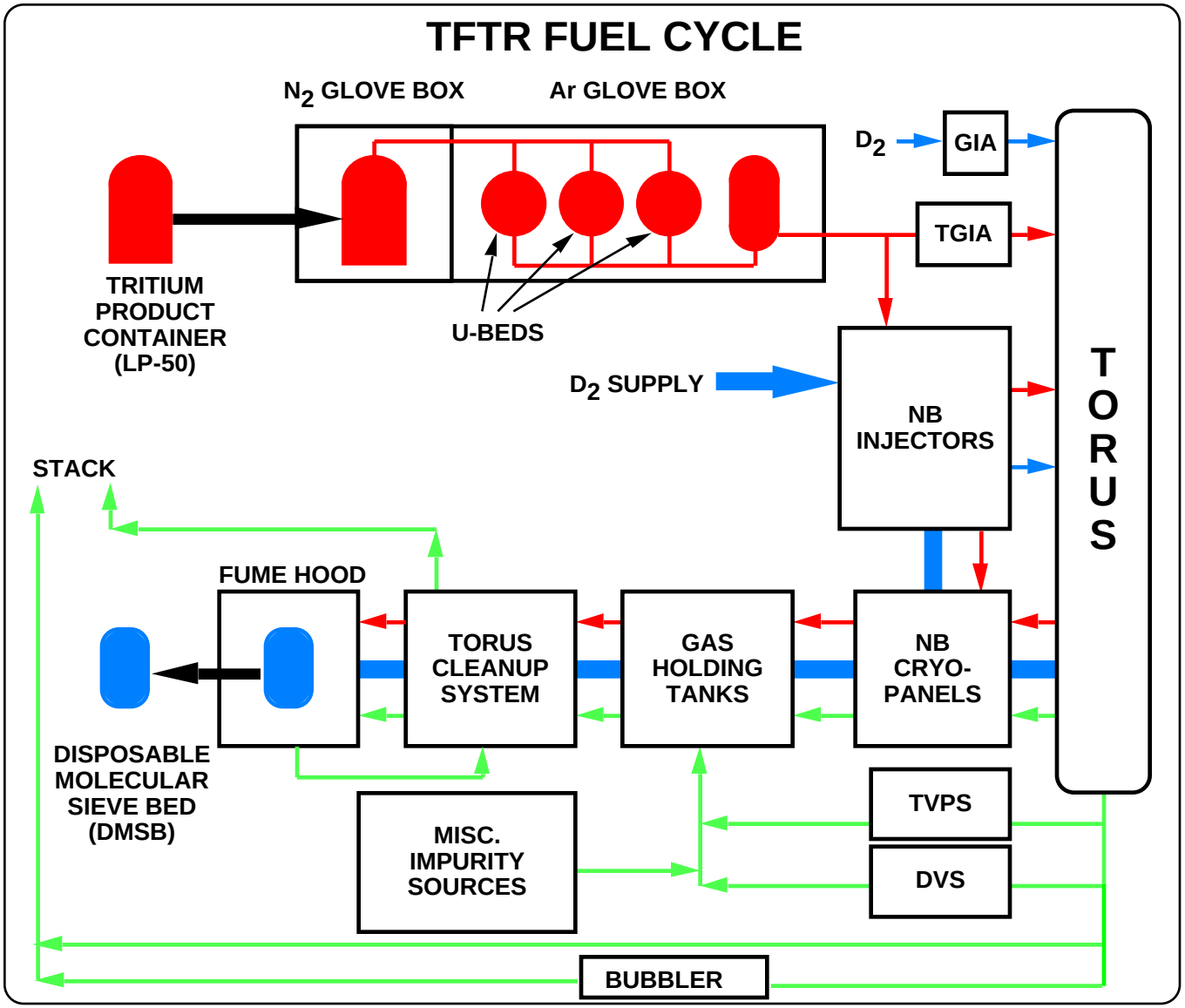




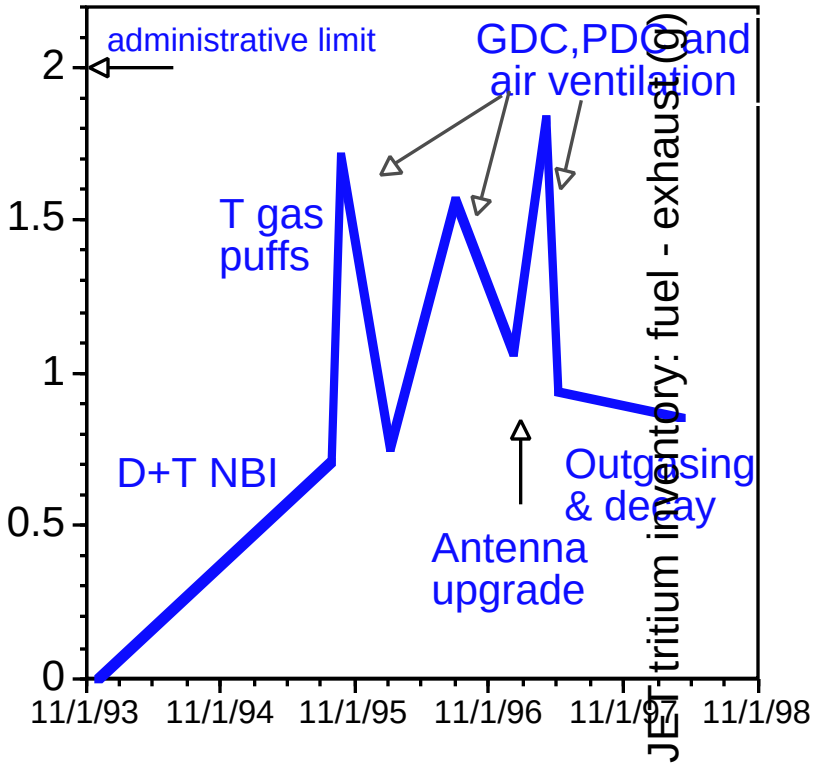
- TFTR
- 10 MW fusion power
- Limiter machine
- typical SOL parameters
- Ne ~ 0.1e19 - 1e19 m-3
- Te ~ 200 eV - 600 eV

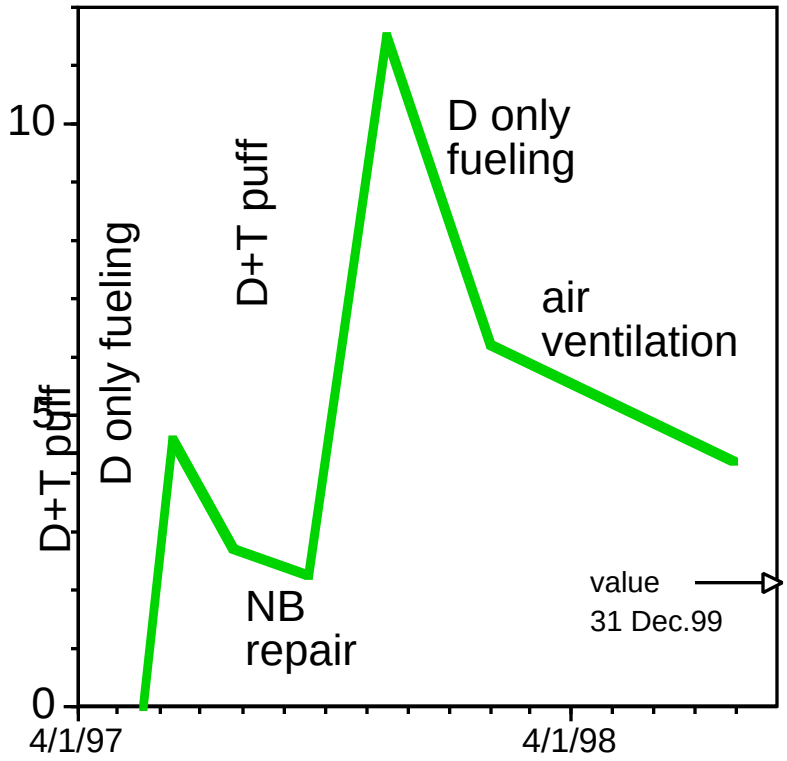
- JET
- 16 MW fusion power
- Divertor machine
- typical divertor parameters
- Ne ~ 10 e 19 m-3
- Te ~ < 30 eV

TFTR fuel cycle:



Chronology of tritium retention in TFTR & JET



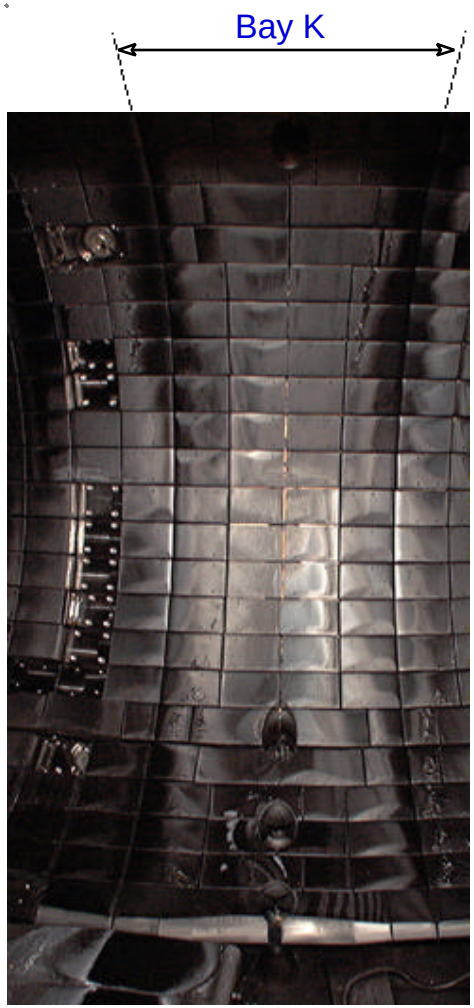


Tritium retention

	TFTR: 3 run periods over 3.5 y	**JET: DTE1, over 6 m**
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)

- ***Larger source of carbon (for co-deposition) in TFTR limiter***
  - ***TFTR limiter conditioned to low D/C before T gas puffing.***
- ***D pulsing removed T from JET dynamic inventory leaving ~1/2 in co-deposits***

Location of Tritium in TFTR



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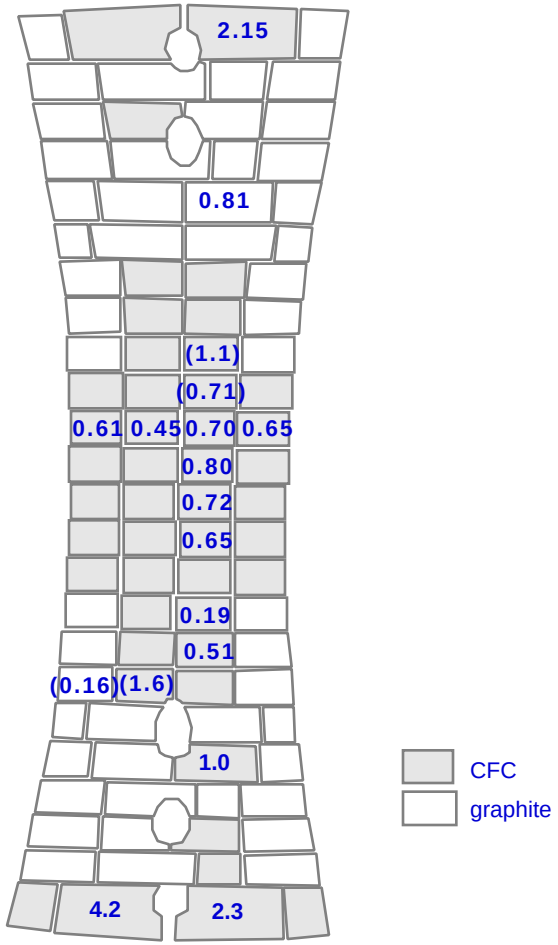
Co-deposition, flaking,
on bumper limiter at Bay K.



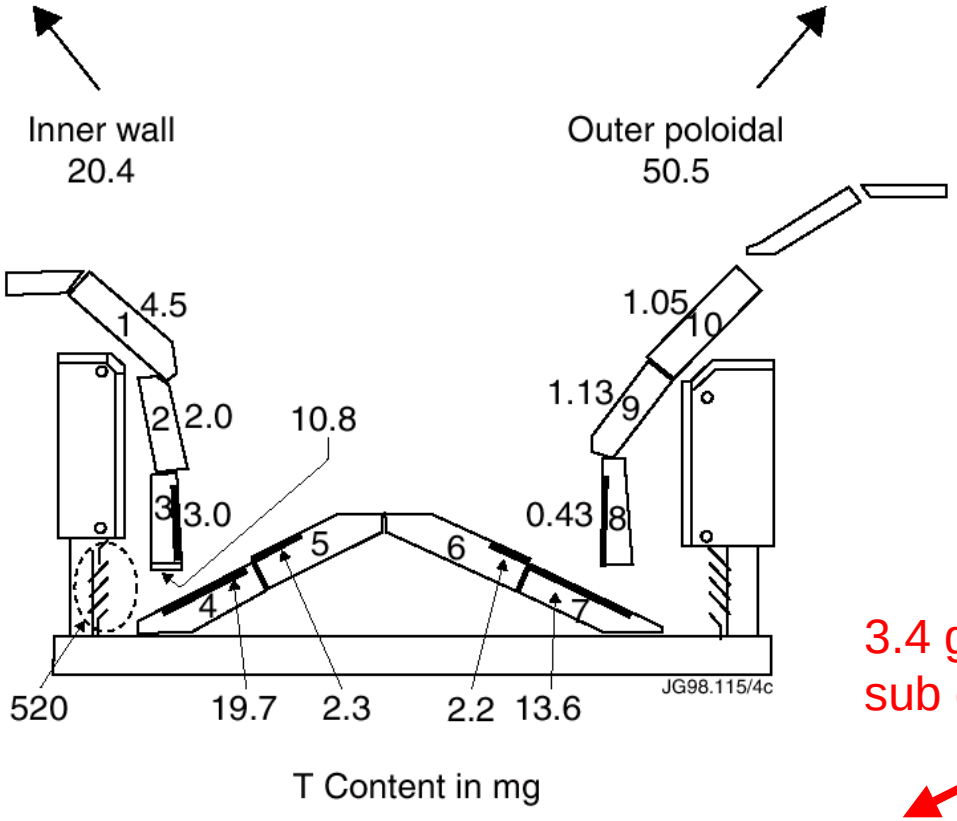
Tritium released (Ci) by 1 hour, 500C
bake of Bay K (L) tiles.

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***

- *1/3 tritium on bumper limiter, 2/3 on outboard wall*
- *Remarkably good agreement between extrapolation from bakeout measurements and difference inventory (fueling less exhaust) and measurements at both PPPL and Savannah River.*

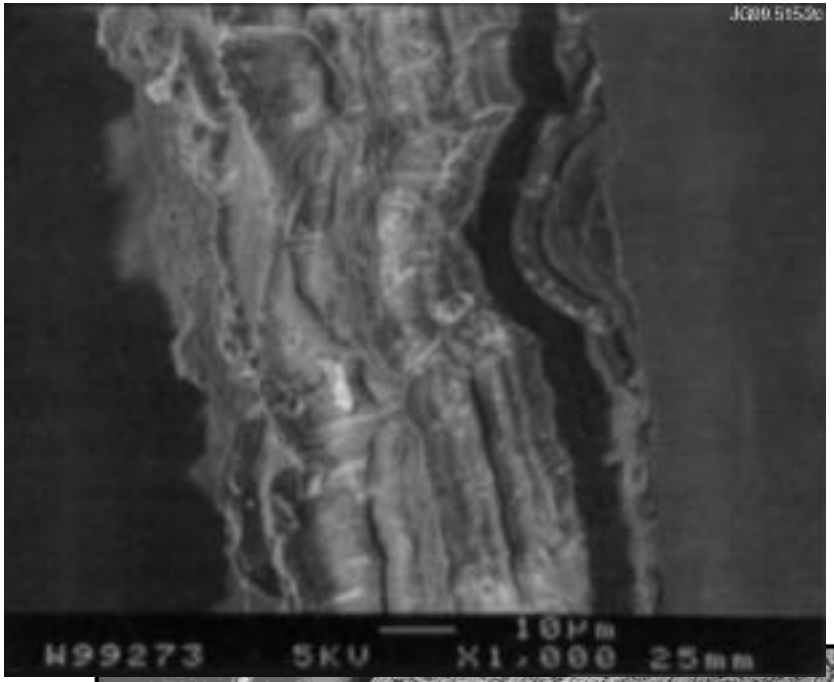
Tritium retention in JET higher than expected



3.4 g tritium in sub divertor ?

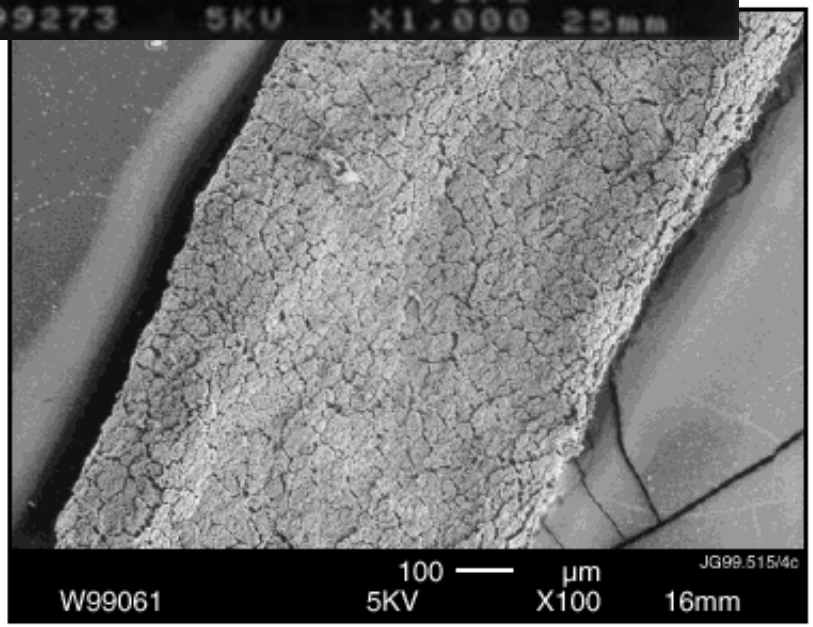
- Flakes seen on louvres in divertor.
- Tiles contain less tritium than expected.
- Tiles 3 & 4 showed unexpectedly high bulk tritium concentrations
- Remaining tritium believed to be in flakes in sub-divertor

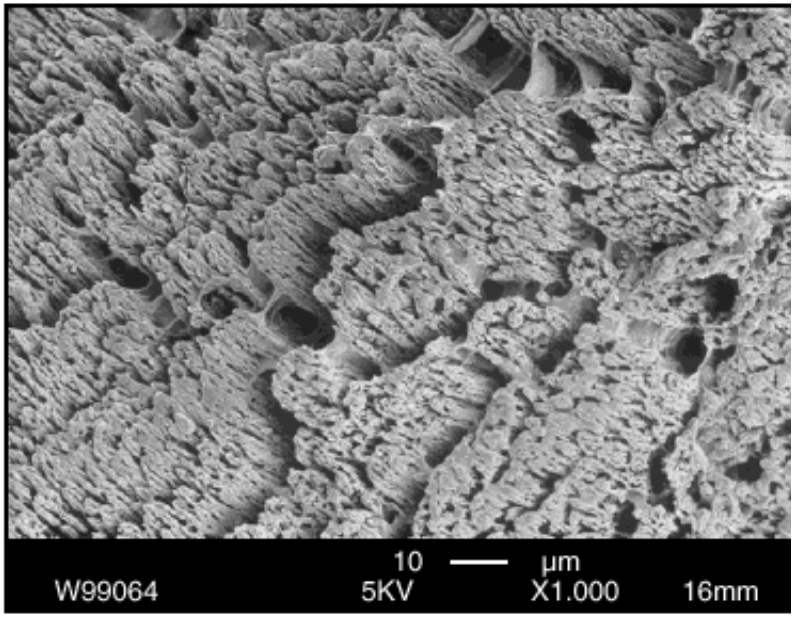
Flakes and heavy deposits in JET



Flakes from JET louvres
at inner divertor

Heavy deposits on JET inner
divertor tile 4 .





Summary from TFTR & JET:

- Tritium retention due to isotope exchange in dynamic wall inventory and co-deposition with eroded carbon.

- **TFTR had large source of eroded carbon from limiter for co-deposition.**
  - Tritium retention in line with deuterium experience
  - Modeling shows high erosion / deposition of C and Li at limiter leading edges.
  - T retention high on leading edges, in line with predictions**.**

- **JET Tritium retention higher than expected from preliminary tritium expt.**
  - JET used intensive T gas puffing which exchanged with D in wall.
  - Material eroded in main chamber flows into inner divertor. There, carbon is chemically sputtered and migrates to (cool) shadowed areas to form thick deposits with high D(plus T)/C

- Retention measurements, surface analysis and modeling give consistent picture.

But.....

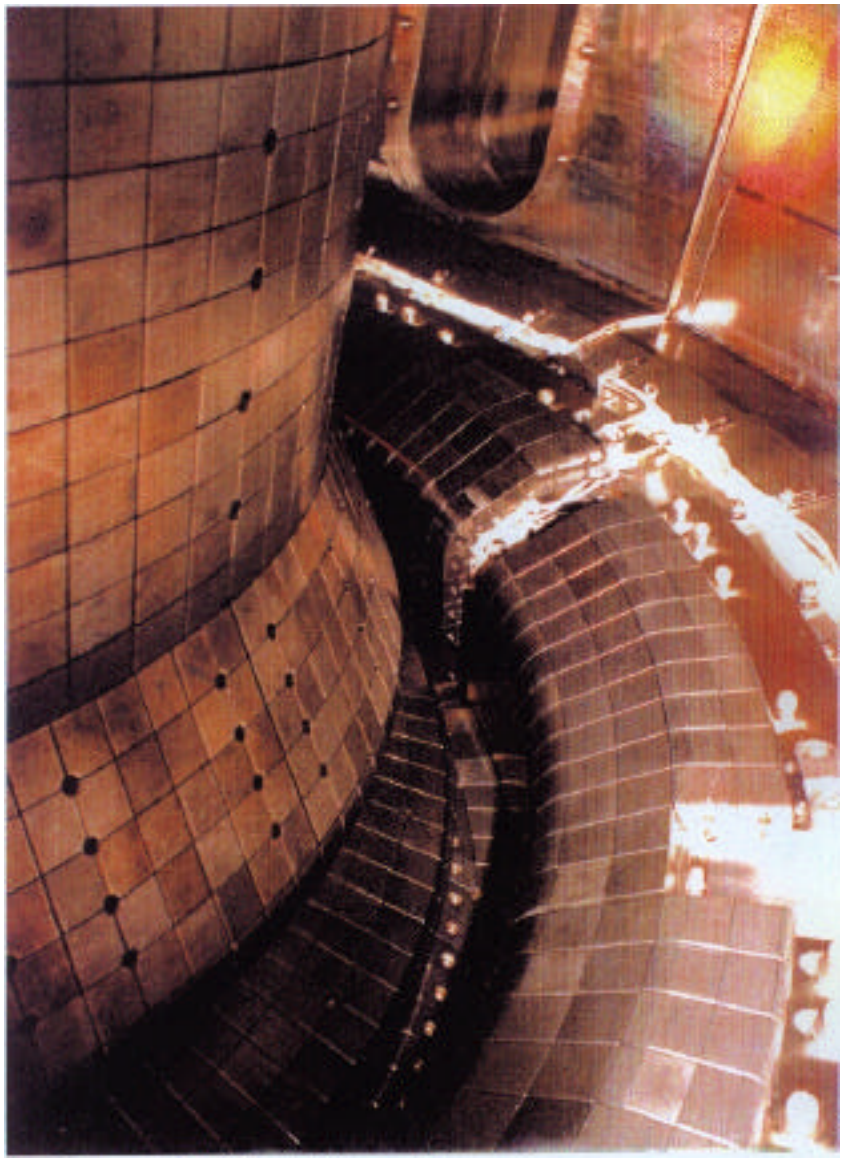
- Future DT power reactor needs retention fraction <~ 0.1% to be self sufficient in tritium.
- ***Carbon plasma facing components are unacceptable for a DT power reactor.***

Retention in metal walled tokamaks

- C-mod is lined with Mo tiles - there are no carbon PFC's
- Boronization used to reduce plasma impurities (carbon present at very low level)
- Fuel is deuterium, nuclear reactions in plasma generate low levels of tritium.
- Tile analysis by Wampler (SNL) showed most of D inventory implanted (not codeposited) on main chamber wall

- **Fraction of tritium produced that is retained is less than 0.002, 100x smaller than with carbon PFC's**

- **Use of metal PFC's in reactors depends on minimising impurity transport and melt layer loss during disruptions.**



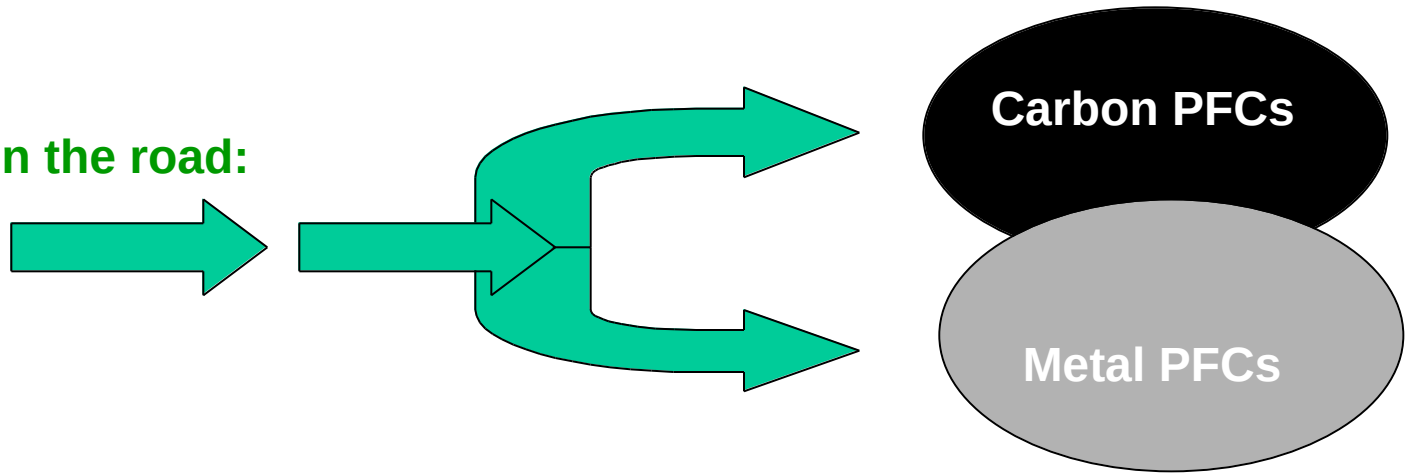
Alcator C-mod

Recent review:
"Plasma Material Interactions in Current Tokamaks and their implications for Next-Step fusion reactors."

- recommend download from http://www.pppl.gov/pub_report/
- PPPL-3531/IPP-9/128 Preprint: January 2001, UC-70
- *Highly relevant to aim of town meeting*

- **Chapter 6 devoted to Future R&D priorities:**

Fork in the road:



R&D issues for carbon PFCs

Some key points:

- Frequent replacements of PFCs needed due to ~10nm/s erosion
- Flux dependence of chemical erosion yield and sticking coefficients of radicals still an open question.
- SOL flows need to be better diagnosed and understood
- Behavior of mixed materials uncertain
- Disruption and ELM loads a major challenge issues include vapor shielding, brittle destruction....
- *Tritium retention unsustainable in power reactors*

R&D issues for metal PFC's

- Encouraging results from C-mod (Mo wall), and ASDEX (W-coated divertor plates and central column).
- W 'brush' materials tested up to 20 MW/m^2.

but....

- Control of transport and MHD in alpha heated plasmas critical (core high–Z impurity levels detrimental to plasma performance even at ~$10^{-5)}$.
- Disruption and ELM loads a major challenge
  - issues include melt layer loss, vapor shielding
  - Disruptions need to be very rare
  - High confinement without ELMs needed.
- Data on neutron effects on tungsten sparse due to activation.
- Public acceptance of handling/ disposal of activated tungsten

Common R&D issues

- Minimization, control and accountancy of tritium inventory a critical issue.
- Tritium needed in burning plasma is small fraction of total.
  - Fast regeneration of in-vessel cryopumps would help reduce inventory.
  - Efficient fueling reduced needed total tritium inventory and aids tritium self sufficiency.
- Ar and Ne injection planned to control divertor detachment but..
  - will become activated, making current tritium detectors unusable for exhaust stream - new detection technology needed.
- Advanced plasma scenarios with high edge temperatures will result in severe erosion.
- Wall conditioning e.g. boronization, over 1000 s pulses, an issue
- Behavior of mixed materials uncertain
- In-vessel dust diagnosis to demonstrate compliance with regulatory limits a major challenge
- Tritium removal/decontamination in areas that require hands-on maintenance also challenging.

Dust and Flakes

Flaking on TFTR limiter

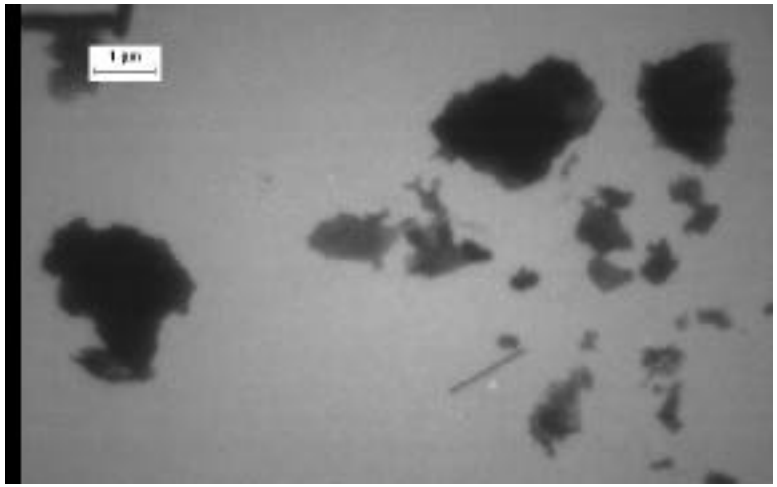
Particles on TFTR vessel floor

- All tokamaks generate dust.
- Flake/dust production will inevitably increase with the increase in duty cycle in next-step devices with graphite plasma facing components.
- Carbon tritide from tokamaks is toxic, radioactive and chemically reactive.
  - quantitive assessment is needed.
- Just diagnosing how much dust is in existing machines is a major challenge

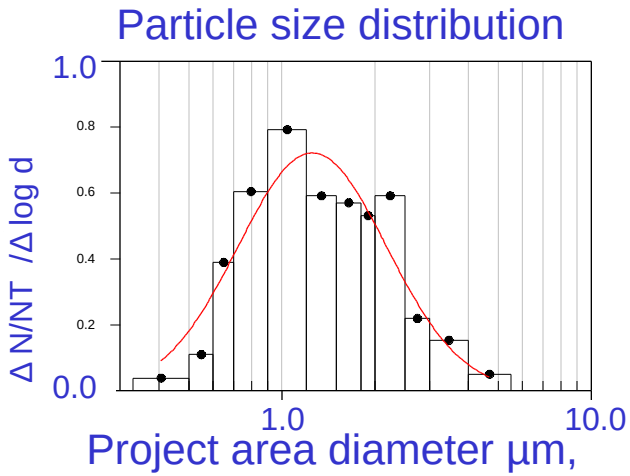
TFTR dust analysis

(collaboration with Y.S. Cheng, Lovlace Respiratory Research Institute)

- Dust is respirable and can stay suspended in air for a long time
- Count Mean Diameter (CMD) =1.25 µm
- Geometric Standard Deviation (GSD) =1.74 µm
- Currently NO standards for occupational exposure to tritiated graphite dust.
- *In vitro* dissolution rate in simulated lung fluid: > 90% of tritium remained in particles after 110 d (HTO eliminated from body in 10 d)
- ICRP modeling suggests occupational limit (DAC) is 4.4x lower than HTO - new techniques for real-time monitoring technology needed.
- *In vivo* biological studies recommended



Microscopic image of TFTR dust



Tritium removal by Nd laser

- **Motivation**
  - In TFTR several weeks were needed for tritium removal after only 10-15 min of cumulative DT plasmas
    - *Future reactors need T removal rate >> retention rate*
  - Heating is proven method to release tritium but heating vacuum vessel to required temperatures (~ 350 C) is expensive.
  - Present candidate process involves oxidation, requiring lengthy machine re-conditioning and expensive DTO processing
  - But
    - most tritium is codeposited on the surface
    - only surface needs to be heated.
  - Modelling indicates that exposure to ~multi-kw/cm^2 laser flux for ~ 10 ms heats a 50 micron surface layer up to 2,000 C enabling tritium release.

Heating releases tritium:

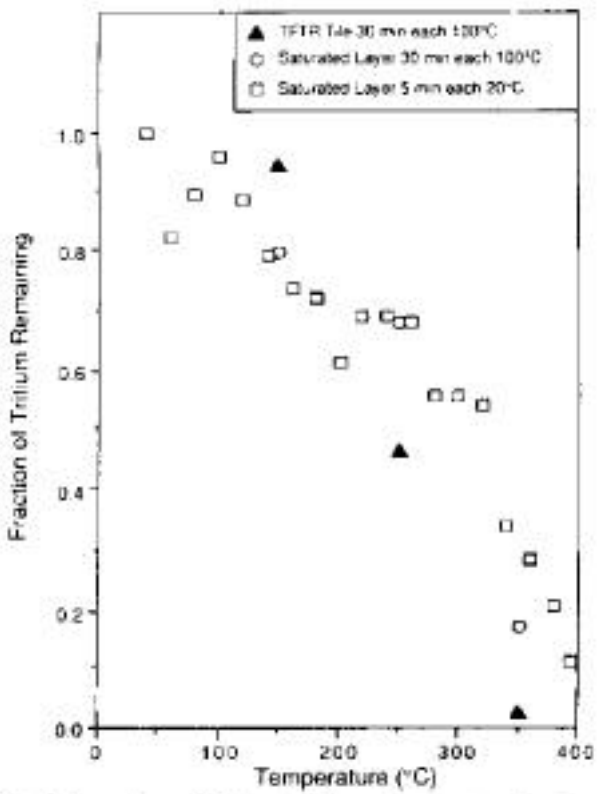


Fig. 2. Comparison of the in air hydrogen isotope retention in a codeposited film to that for a saturated layer.

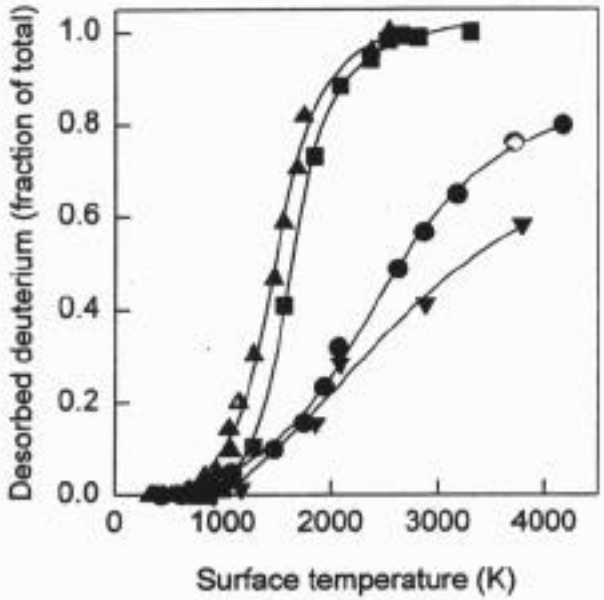


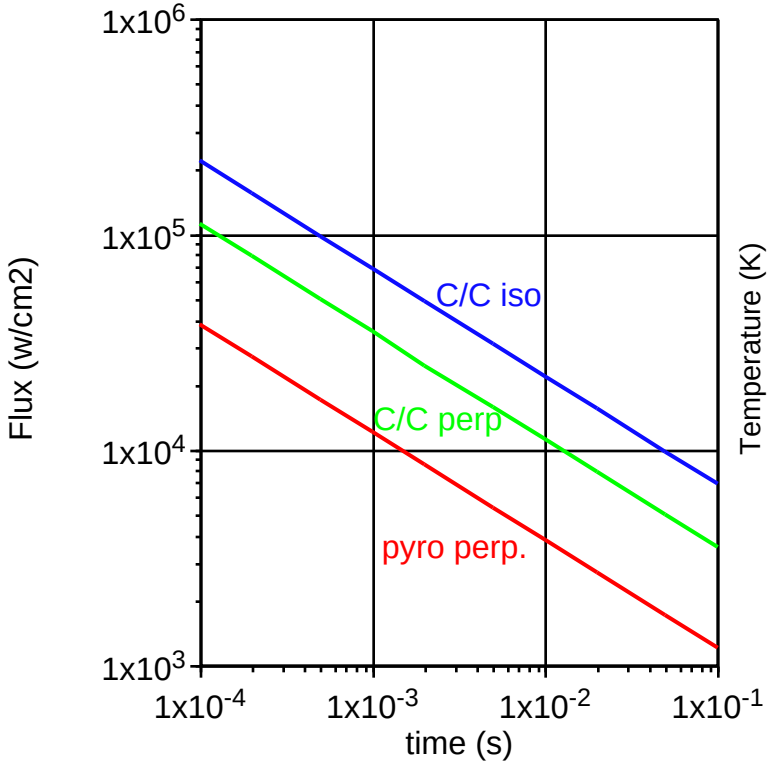
Fig. 7. Cumulative desorption curves for four types of carbon implanted with 1.5 keV deuterium as a function of the maximum surface temperature reached during the laser shot. The x-axis was calculated with the computer code DTRLAS up to the graphite melting point. ● isotropic graphite (4.1×10^{20} D/m^2), ■ vitreous carbon (8.2×10^{20} D/m^2), ▲ pyro a (2.1×10^{20} D/m^2) and ▼ pyro c (4.1×10^{20} D/m^2).

Heating of co-deposited TFTR tile and C implanted with D (Causey et al.)

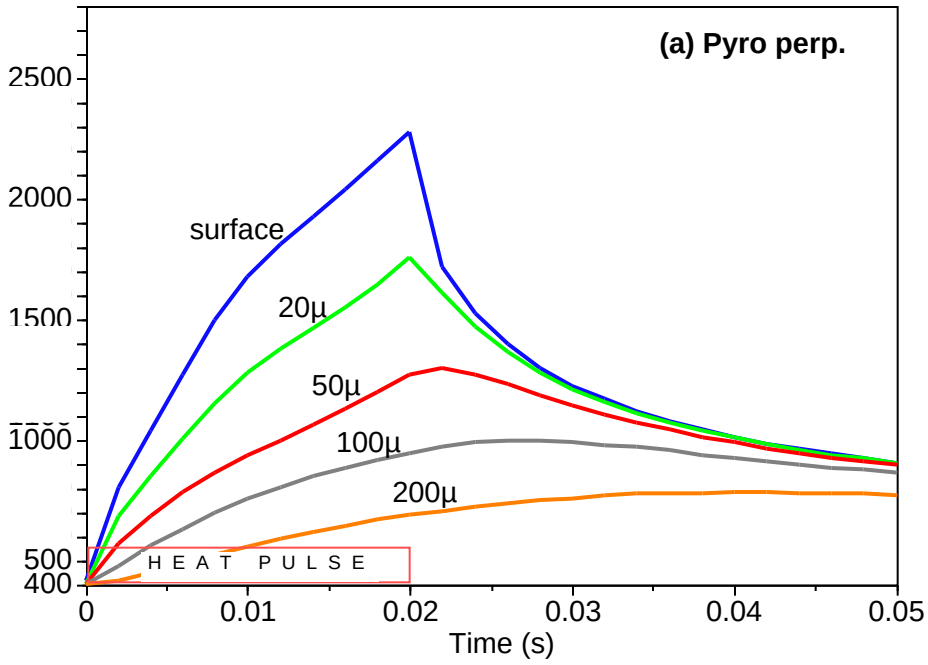
30 nsec laser pulse on 4 types of C, implanted with 1.5keV D (Keroack et al.)

Modeling of laser heat pulse:

Heat flux necessary to attain a 400 -> 2,000 K surface temperature rise for different graphites.



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

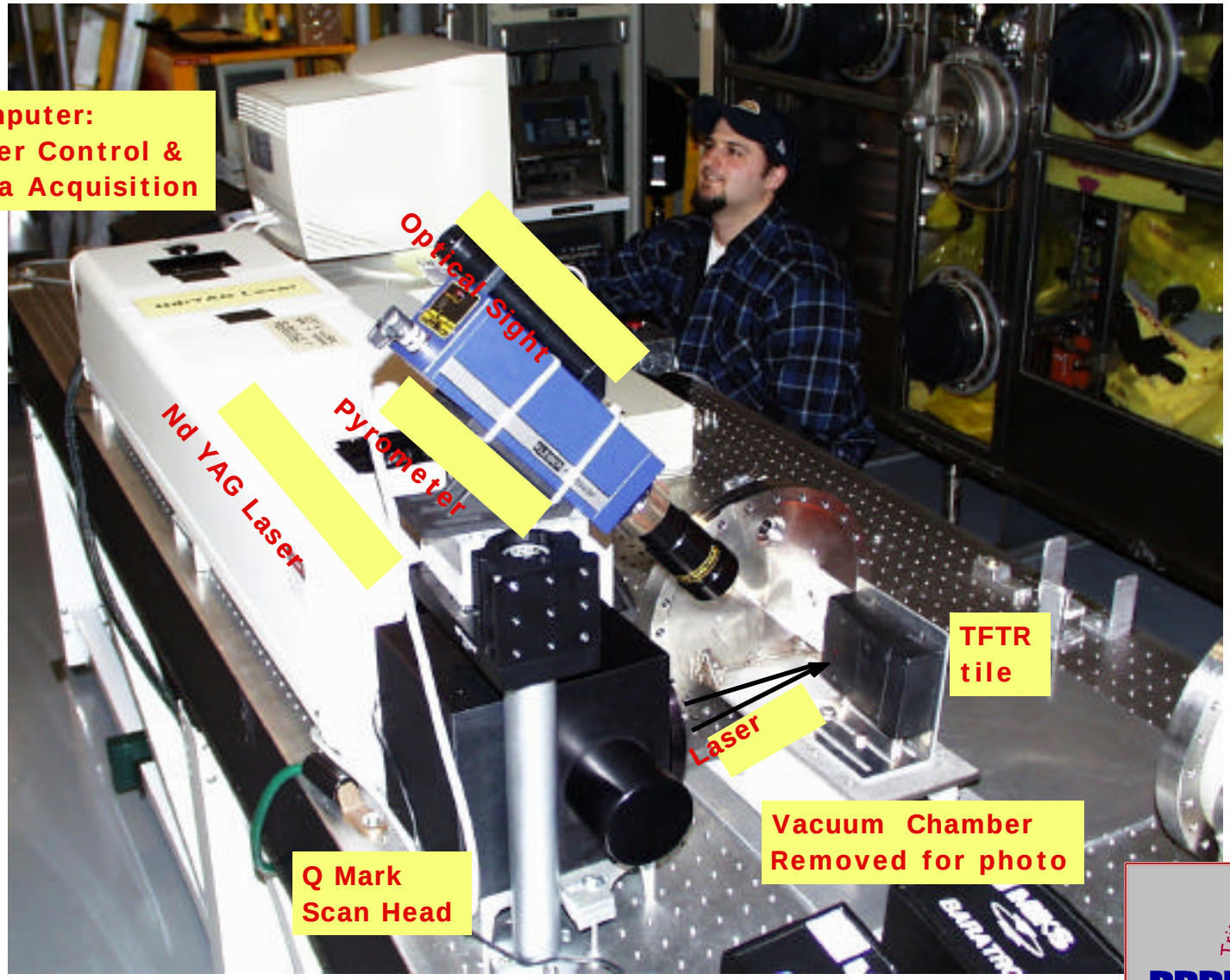


from numerical heat code HEAT1DS by M. Ulrickson

Note: wide differences between carbon materials

- no thermal coefficients available for co-deposited amorphous tritiated carbon.

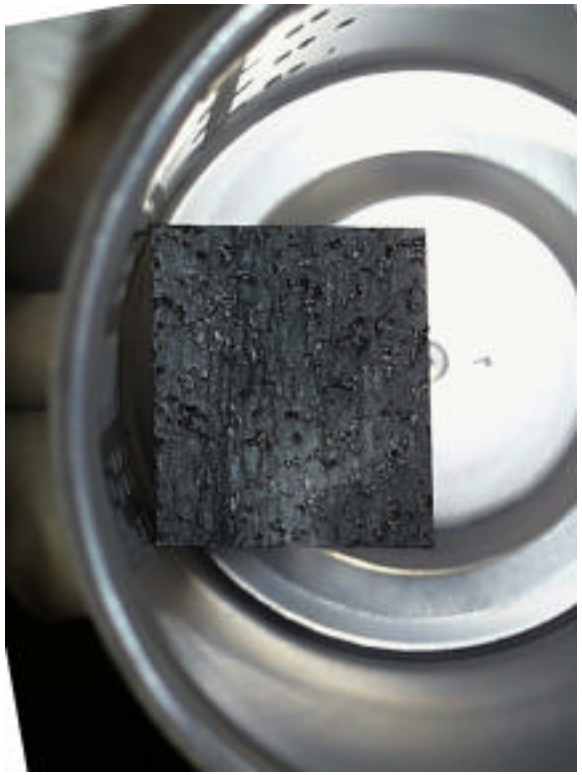
Experiments have begun:



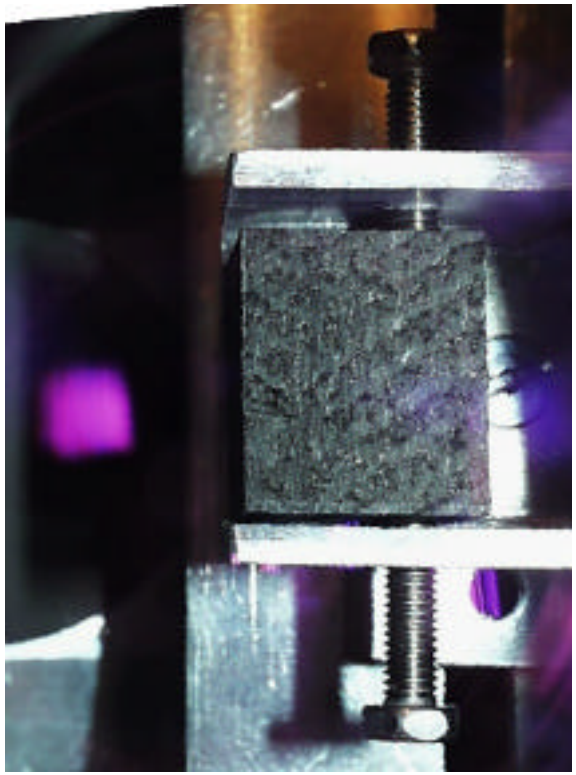
Tritium Technology

PPPL JAERI

1st results: TFTR DT tile before and after Nd laser exposure

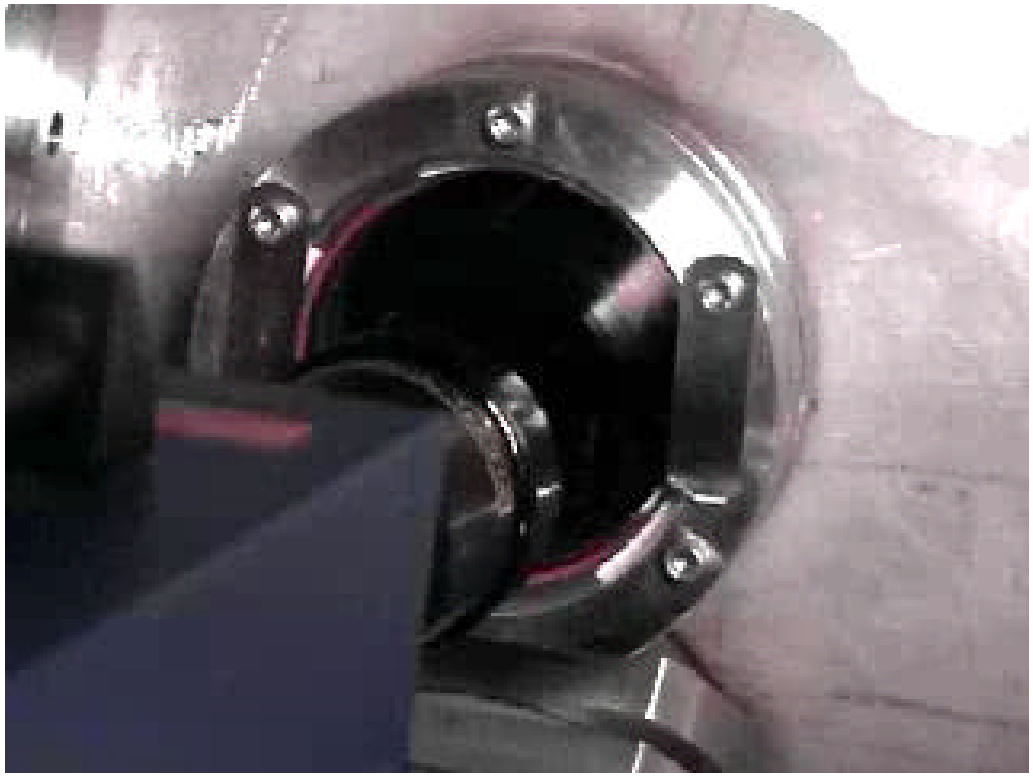


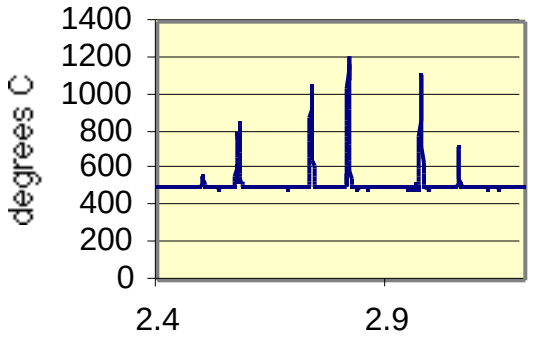
Cube cut from CFC tile KC17 2E before laser exposure
size: 7/8” a side

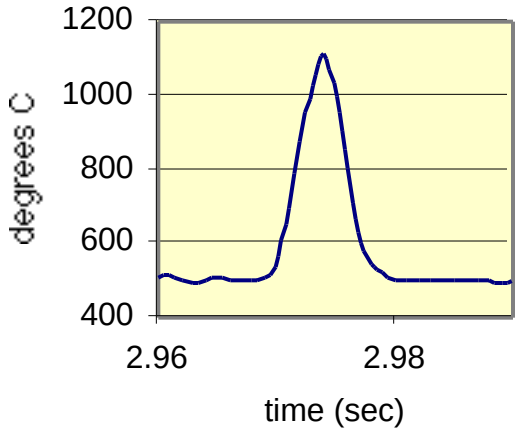


In chamber after exposure to Nd laser
2000 mm/s @ 40 W and
200 mm/s @ 6 W

Laser in action:







Nd laser power only 6 w (300 w available)

~0.5mm focal spot, 200mm/s scanning across TFTR DT tile cube in air.

Pyrometer views 0.7 mm area on DT codeposit on CFC tile on 2nd scan (temp.>2,300C on cube 3E)

(pyrometer range 500 C - 2300 C)

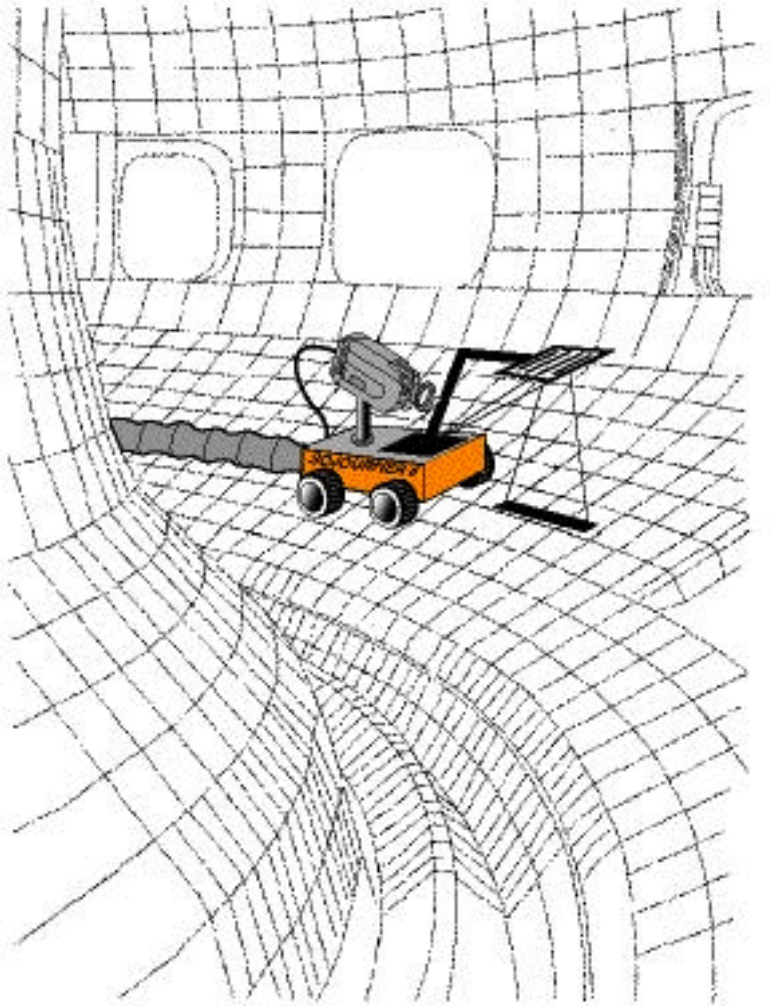
Preliminary data on tritium released in first expt.

- **Surface tritium (measured with open wall ion chamber) decreased**
  - from 51 µCi/cm2 before
  - to 14 µCi/cm2 after Nd laser exposure
- **2.6 mCi released by Nd laser**
- **10.7 mCi released on exposure on baking at 500 C for 1 hour in air.**
- **Non plasma facing surface on cube 3E heated to > 2300 C by Nd laser showed complete removal of surface tritium.**
  - Powerful decontamination technique.
- **Need to optimize scan rate, laser power, laser focal spot size, investigate desorption processes, surface effects....**

The Future ? - Voyager II

- **3e7 J required to heat top 100 microns of 50 m^2 divertor.**
  - corresponds to output of 1kW laser for only 8 hours !
- **Nd laser can be coupled via fiberoptic**

- **Potential for oxygen free tritium release in operating tokamak**
  - avoid deconditioning plasma facing surfaces
  - avoid HTO generation (HTO is 10,000x more hazardous than T_2 and very expensive to reprocess)



Further reading:

- "Studies of tritiated co-deposited layers in TFTR"
  J. Nucl. Mater. 266(1999) 941. PSI-14 J. Nucl. Mater in press
- "Tritium Retention and Cleanup in JET"
  Fus. Eng. & Des. 47 (1999) 233
- "Long Term Retention of Deuterium and Tritium in Alcator C-mod"
  Proceedings of 18th Symposium on Fusion Energy, (1999) p.267.
- "Tritium experience in large tokamaks: Application to ITER"
  Nuclear Fusion 39, 271, (1999)
- "Plasma-material Interactions in Current Tokamaks and their Implications for Next-step Fusion Reactors."
  PPPL-3531/IPP-9/128 Preprint: January 2001, UC-70
  download from http://www.pppl.gov/pub_report/