Tritium inventory control in ITER

Charles Skinner with key contributions from
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Gianfranco Federici (ITER JCT).

- *Motivation*
- *How much time does ITER have to remove tritium ?*
- *T removal from ITER*
  - *-oxidation*
  - *ablation by flashlamp or lasers*
  - *other techniques*
  - *thermal desorption by disruption or laser heating*
- *Conclusions / Recommendations*

Tritium inventory control

Major milestone in US:
National Research Council Report "Burning Plasma - Bringing a Star to Earth"
Sept 26th, 2003
P. 38
... "high confidence in readiness to proceed with burning plasma step"

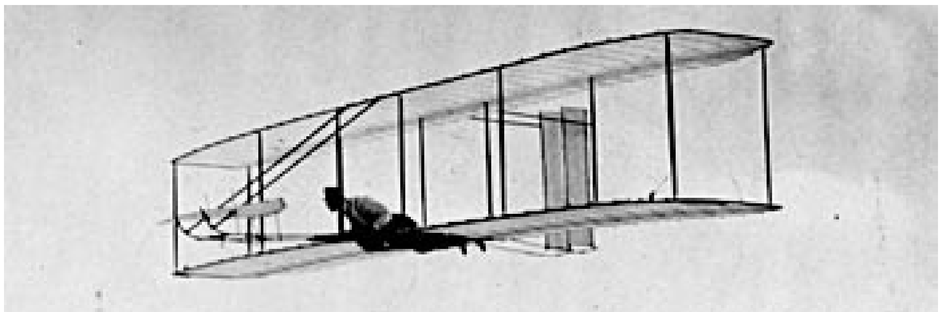
P. 55:
"In ITER [codeposition] could result in a limit of 10-100 shots before the tritium in the chamber reaches the maximum permitted"

Worrisome issue:
Once at the tritium limit there won't be any more burning plasmas until the tritium is removed.

Where will the tritium be located ?
and exactly how will it be removed ?

Control, a century ago:

Wright Brothers' 1902 glider, the world's first aircraft with fully controllable yaw, pitch and roll (albeit unpowered).



"If you are looking for perfect safety, you will do well to sit on a fence and watch the birds; but if you really wish to learn, you must mount a machine and become acquainted with its tricks by actual trial."

Wilbur Wright, on learning to ride a flying machine

http://www.nasm.si.edu/wrightbrothers/

Langley Aerodrome A

The first test flight of the Aerodrome A was on October 7, 1903. Immediately after launching, the Aerodrome plunged into the river at a forty-five-degree angle... The second crash of the Aerodrome A ended the aeronautical work of Samuel Langley. His request to the Board of Ordnance and Fortification for further funding was refused and he suffered much public ridicule.

Langley's simple approach was merely to scale up the unpiloted Aerodromes of 1896 to human-carrying proportions. This would prove to be a grave error, as the aerodynamics, structural design, and control system of the smaller aircraft were not adaptable to a full-sized version. The control system was minimal and was also poorly conceived.

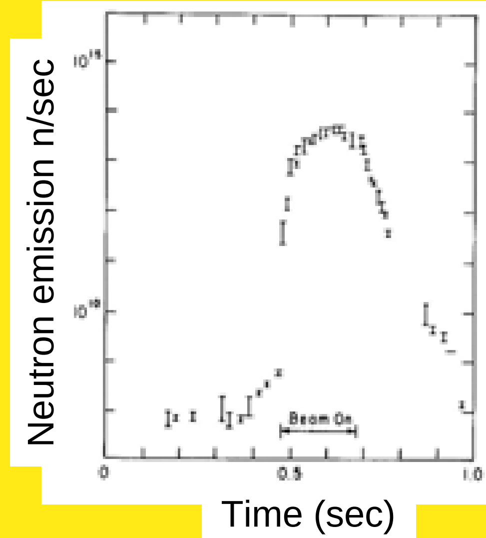
http://www.nasm.edu/nasm/aero/aircraft/langleyA.htm

Motivation

- Decades of R&D have established a strong physics and technology base for ITER
  - BUT one major development task remains.
- Tritium removal at unprecedented speed and efficiency will be necessary for ITER with carbon PFCs to support a credible physics program.
- This situation is in striking contrast other technology development e.g. superconducting magnets, remote handling, and surprising in view of public sensitivities to radioactivity (e.g. closure of High flux beam fission reactor at Brookhaven).
- Alternative of tungsten PFC also carry significant risks of plasma contamination, melt layer loss, and time lost to divertor replacement.
- ITER could initially be a hugely expensive plasma wall interaction (PMI) experiment
  (≈ $ 100,000 / hour for unplanned outages).
- Only if PMI solutions are found, will a burning plasma program be possible.

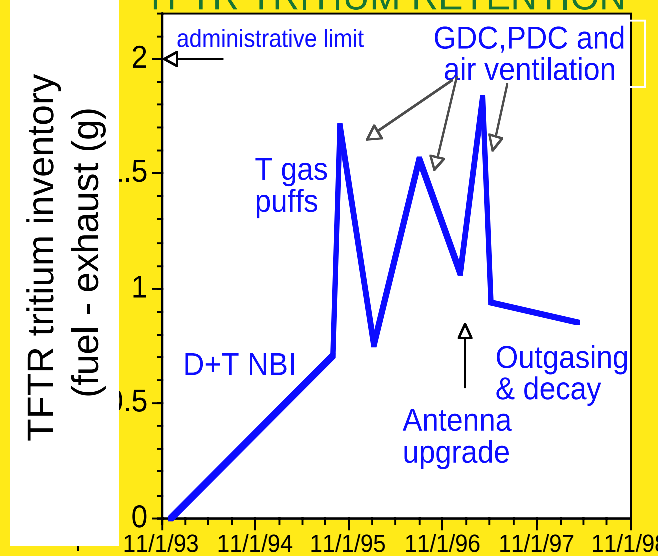
History:

1978: Changing from tungsten to carbon limiter enabled PLT to access low collisionality Ti ≈ 5.5 keV, plasmas (Eubank et al., 1978 IAEA)

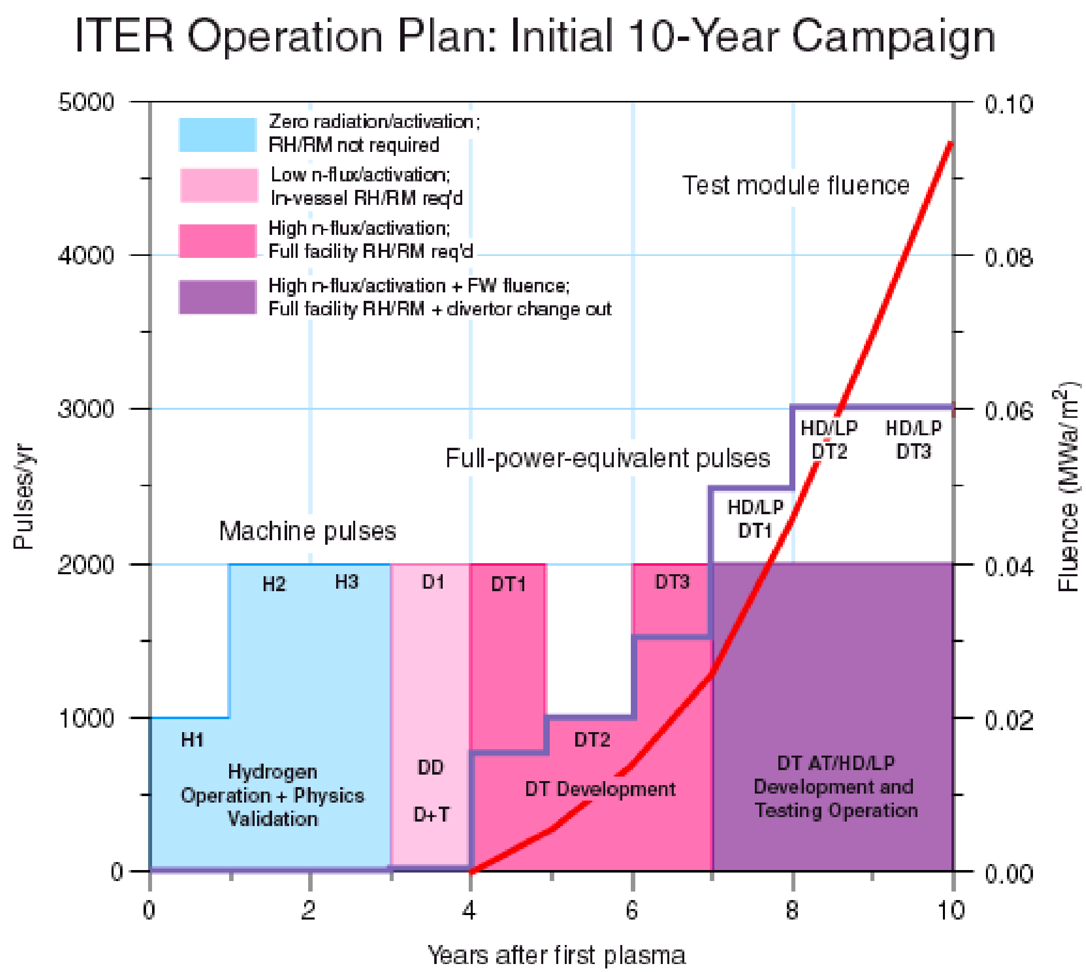


But eroded carbon is codeposited with tritium

TFTR TRITIUM RETENTION



ITER duty cycle is biggest scale-up from current tokamaks



(1 MWa/m2
= 3.15 x 107 MJ/m2
= 1.39 x 1025 n/m2)

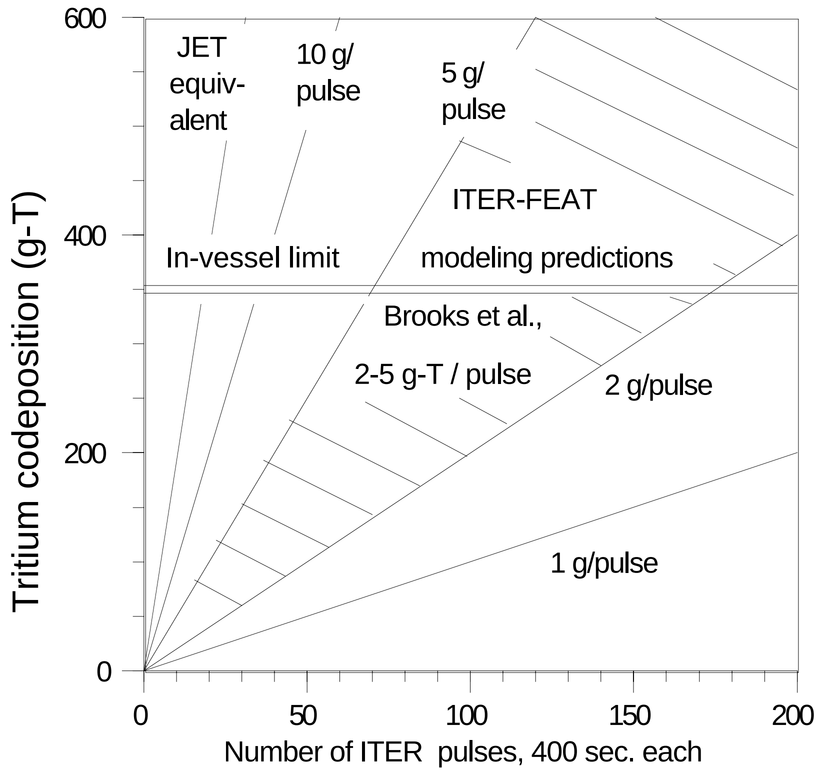
- 2000 pulses / year means 2 shifts (14 hr/day) 5 days / week,

  3 weeks / month, 8 months/ year with 70% availability.
- Tritium accumulates much faster, with much less time available for removal than TFTR or JET

ITER retention could be 50 - 125 g / day in 50 µm codeposit

Note:

- ITER predicted tritium accumulation rate is 10x less than that experienced in JET
- But model underestimates JET retention by factor x40.
- location of tritium unclear (flakes, bulk of CFC tiles ? ....)



Carbon PFCs limited to divertor strike point

J Brooks, A Kirschner, D. G. Whyte, D. N. Ruzic, D. A. Alman

- Modeling of detached plasmas a challenge. Predictions uncertain due to
  - uncertain chemical erosion yield of redeposited material,
  - effects of mixed materials,
  - lack of code validation in detached plasma.

Scale up in duty cycle and tritium usage is larger step than change in plasma parameters

Parameters:	TFTR experience	JET experience	ITER projections
Tritium in-vessel inventory limit	2 g	20 g site inventory	350 g
Typical pulse duration	≤ 8 s	30 s	400 s
Tritium retention rate (JET/TFTR inc. D only pulses)	51%	17%	≈ 3%
Cumulative DT discharge duration before inventory limit first approached.	708 pulses ≈ 33 min	500 pulses ≈ 250 min	≈70–170 pulses 466 – 1133 min
Period before inventory limit approached.	22 months	≈ 3 months	≈ 1 week (± uncertainties)
Time devoted to tritium removal etc…	1.5 months	3 months	est. ≈ 5 h overnight
Fraction of tritium removed	50%	50% (prior to venting)	close to 100%
Tritium removal rate	~ 1 g /month	2 g / month	Up to 25 g / h or 10 µm codeposit / h

Bottom line:

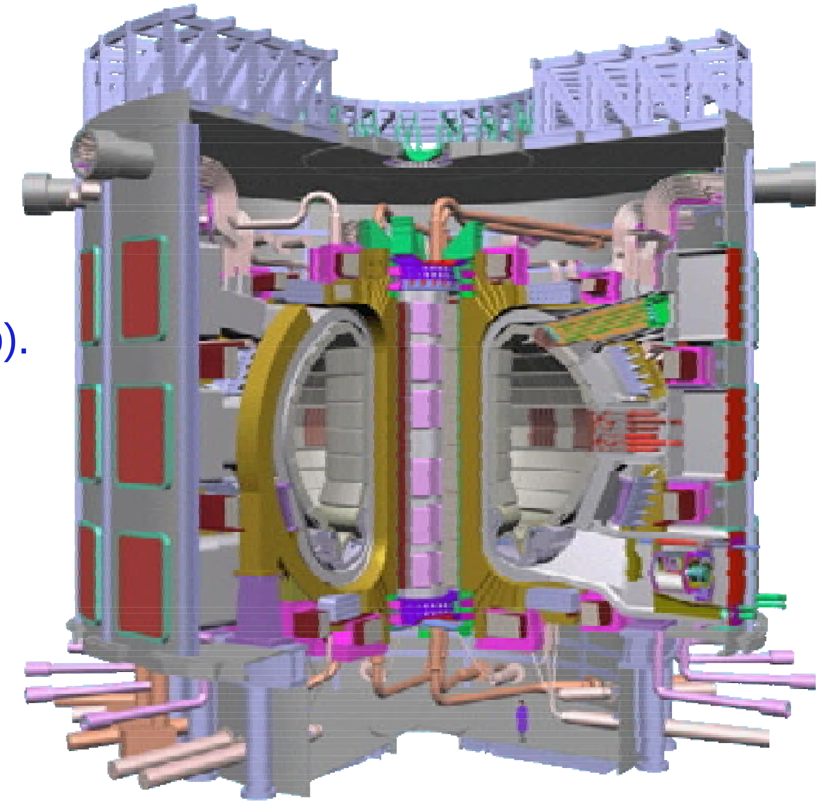
•**Need to demonstrate method that can efficiently remove up to 125 g of tritium from 50 micron codeposit overnight.** (Removal rate scale up from TFTR & JET ~ x10^4)

•**Access for tritium removal should be integral part of divertor design.**

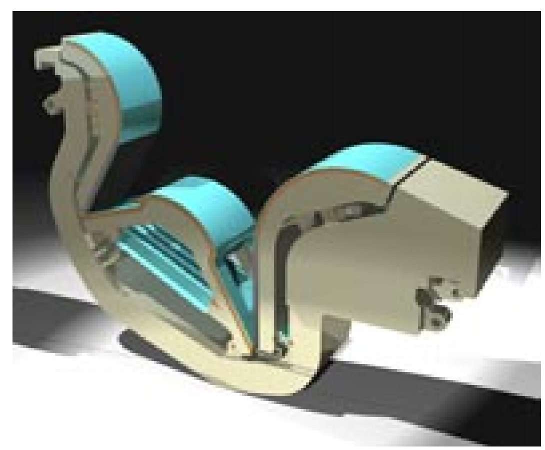
Tritium removal: potential options & constraints:

Potential Options

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



- 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 codeposits by oxidation to H_20, CO_2, CO.
- removal rate depends on film structure - codeposits removed ~ 100x faster than manufactured tile
- ‘soft’ films removed at lower temperatures
- removal rate up to 50 μm/h measured by Haasz et al. for TFTR codeposit in lab tests.
- Some experience on TFTR, JET, TEXTOR

  see Wang et al.,Maruyama et al., Alberici et al.,

  see review by Davis in Physics Scripta T91, 33 (2001).

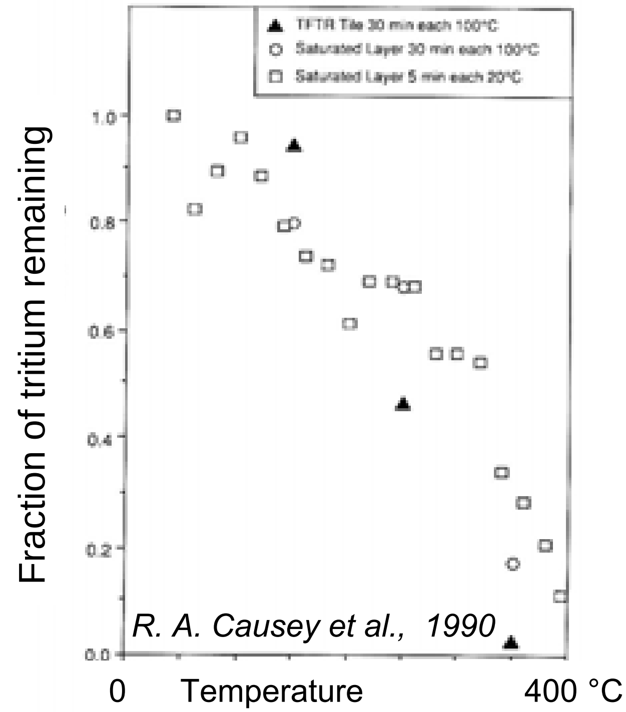
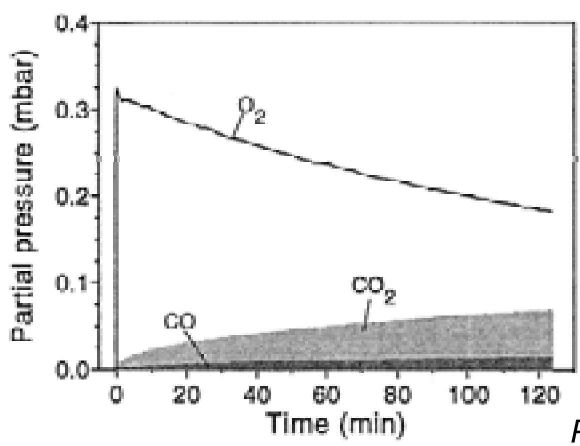
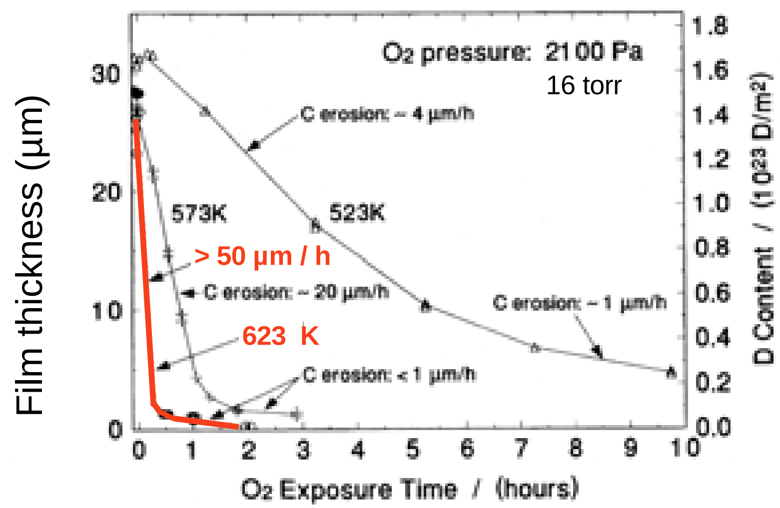


Fig. 2. Comparison of the in air hydrogen isotope retention in a codeposited film to that for a saturated layer.



Philipps et al

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.)



Haasz & Davis 1998

Tritium removal by oxidation - overview:

MERITS:

- Lab experience, limited tokamak experience
- Access to all areas in vessel
- Simple to implement, no in-vessel hardware

LIMITATIONS:

- Temperature required for fast removal higher than the 240 C attainable with pressurized water cooling.
- Potential for collateral damage to in-vessel components.
- Appears impossible to re-condition plasma facing surfaces in time available.
- Is Be wall then BeO? Will Bel continue to getter oxygen ?
- Tungsten or boron impurities found to inhibit oxidation of codeposit (Davis & Haasz)
- DTO exhaust is more hazardous than T_2 and needs substantial investment in tritium plant to process

To be credible for ITER, demonstrations in current tokamak of fast and nearly complete removal of codeposited H-isotopes at 240 C without collateral damage are needed.

Why tokamak tests are essential:

"I guess I am also missing why you can't just process one of our tiles in a side lab experiment -- not sure why the tokamak part is so important.
How would it be more convincing doing it in DIII-D? "
- Steve Allen.

1. CONDITIONING: The surface of tiles used in ex-situ detritiation experiments is not exactly the same as the 'conditioned' surface of tiles in operating tokamaks.
   XPS analysis of removed TFTR tiles showed an extensive zone of oxidised carbon (O content 20-50%). Some codeposits detached (flaked off) from substrate.
   - To measure the efficacy of a T removal technique on plasma-conditioned tiles you need to do it in a tokamak.

2. REABSORPTION: Tritium may be released from tiles as 'sticky' hydrocarbon radicals that are redeposited before being pumped out of the vessel. The tritium removal rate of HeO GDC in TFTR was 20 times less than reported in laboratory measurements
   - To demonstrate that redeposition is not an issue, tokamak experiments are essential.

3. RE-CONDITIONING: A key constraint is how long it takes to restore good plasma performance after tritium removal. At present there is no specific allowance in the ITER operational schedule for either tritium removal or recovery of good wall conditions.
   - The time needed to restore good plasma performance can only be measured in a tokamak.

4. CREDIBILITY: How can oxidation be a credible tritium removal technique for ITER if current tokamak operators are afraid to prove its efficacy because of fears of collateral damage ?

Tritium removal by ablation using excimer lasers or flashlamps

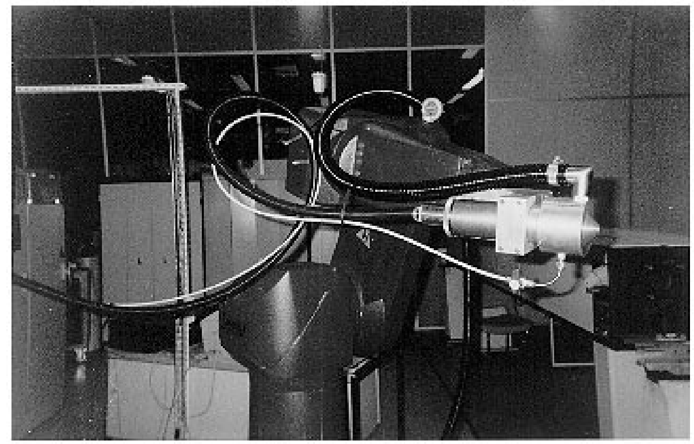
Art restoration by laser

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

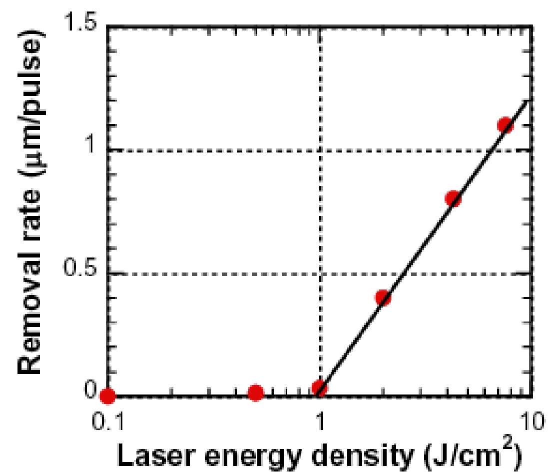
Automated XeCl laser unit developed for radioactive metallic oxide decontamination. 2-6 m^2/h, fiber ≤ 5 m.

Sentis et al., Quantum Electronics 30 495 (2000)



Excimer laser ablation:

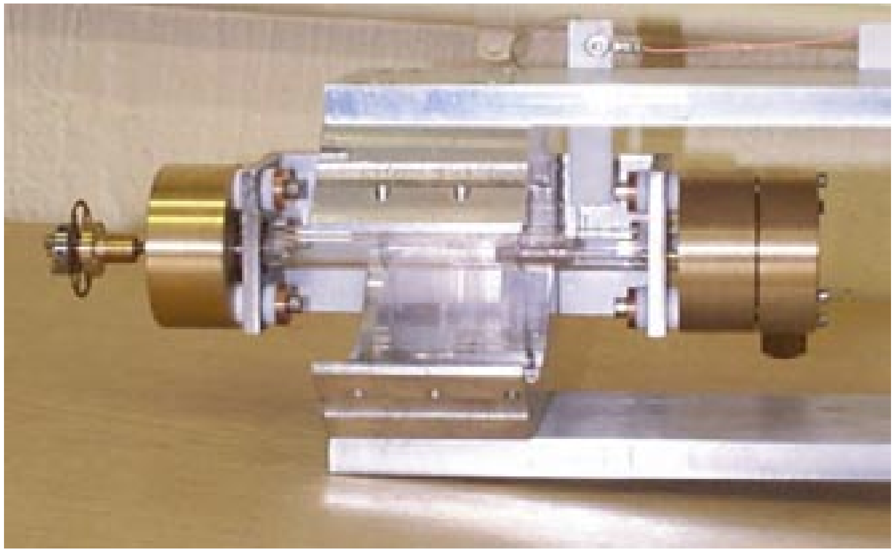
ArF laser removes JT60 codeposits
Shu et al., JNM 313 (2003) 585

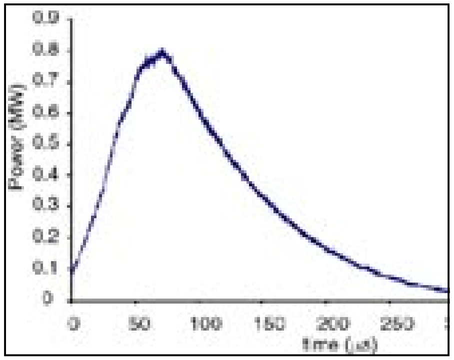


Flash-lamp detritiation

Glenn Counsell

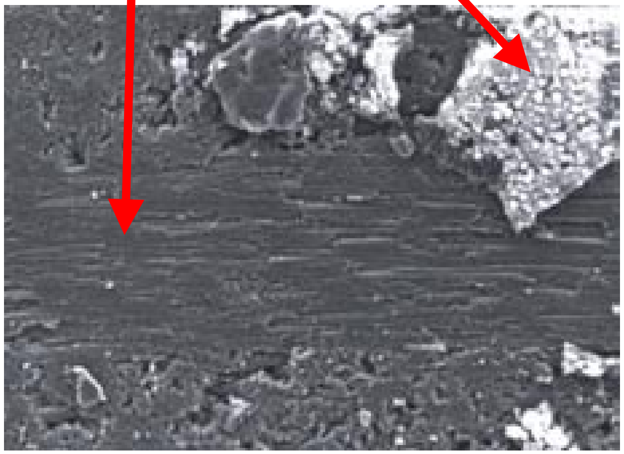
- High power flash-lamps are being studied as means of detritiating and/or removing co-deposited films in ITER during short maintenance periods (in vaccuo and with coils magnetised)
- ELM-like power densities possible ($1 GW/m^2$) in $10\ cm^2$ area.
- Surface temperature of typical co-deposit raised by >1000 K in one pulse without substrate damage.
- 50 µm film removed in 5 pulses
- Cleaning rates > $3\ m^2$/hour demonstrated with 4 Hz prototype.
- Co-deposit removal produces significant amounts of H_2, CH_4 and higher hydrocarbons but also dust
- Balance of gaseous/solid debris still to be determined/optimised





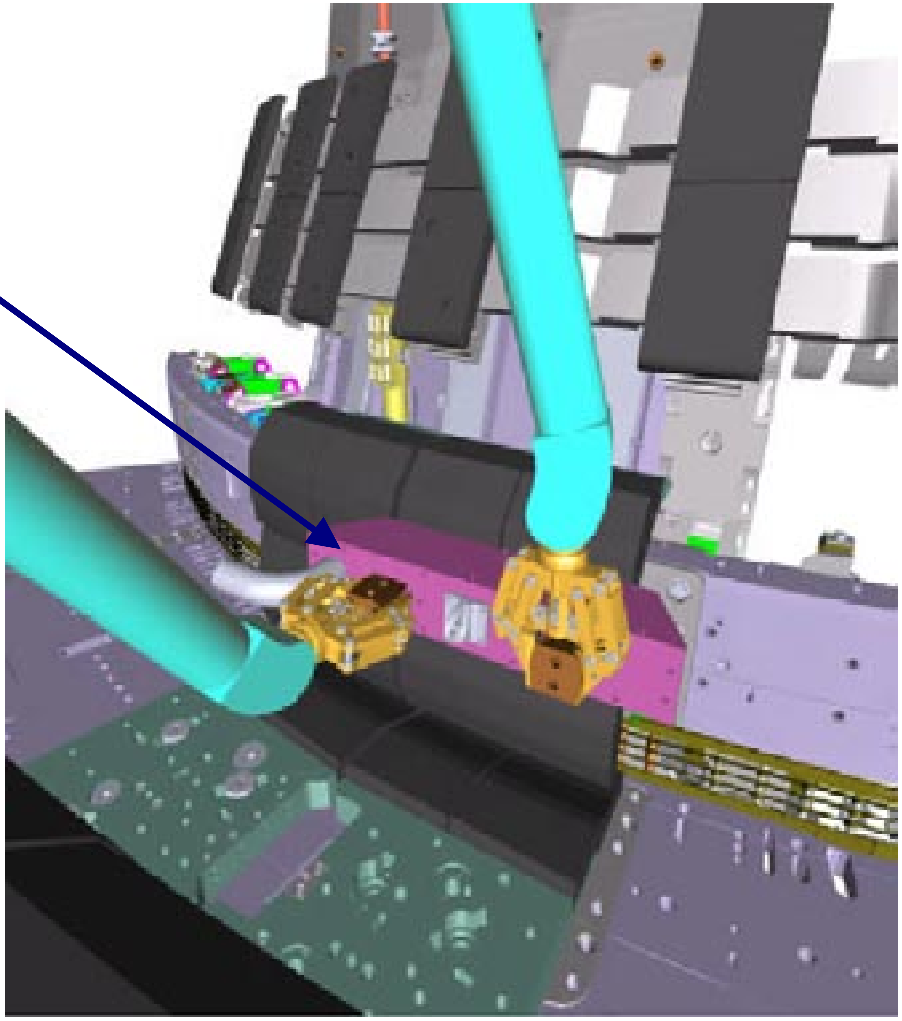
Cleaned substrate

Co-deposit



Cleaning trials planned on JET

- New flash-lamp system developed for JET trials
- 500 J, 5 Hz flash-lamp and power supply (*cf* 100 J, 4 Hz prototype)
- Flash-lamp and optics housed in MASCOT robotic arm head
- Cleaning trials (at atmosphere) planned for heavily co-deposited inner divertor region
- Flash-lamp head supplied with power and cooling water via umbilical
- Attached via vacuum pump and filter to JET tritum handling system



- Trials planned at reduced energy (<100 J) operation to simulate laser detritiation.
- Energy insufficient to remove co-deposit but sufficient to outgas retained tritium

Tritium removal by ablation - overview

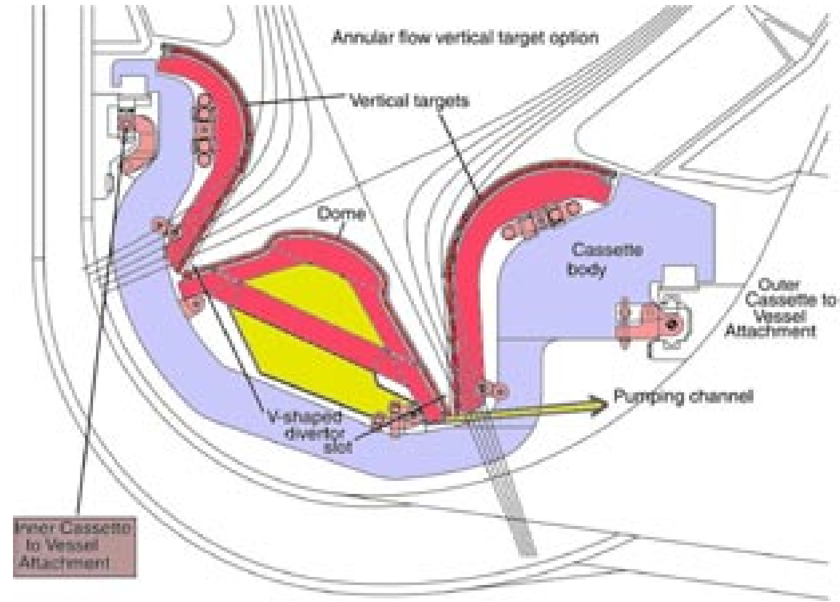
MERITS:

- some lab & industrial experience,
- whole codeposit removed

ISSUES:

- Fate of ablated products ?
  - potential for debris to fall into inaccessible areas reactive
  - radicals could be produced that would redeposit in-vessel
- For excimer lasers: is fiber optic transmission sufficient over required distance ?
- Is removal rate sufficient ? (≈100 g T / 5 h needed)
- Can hidden areas be accessed ? - >>
- Is hardware compatible with 6.1 T ?
- Is hardware compatible with 10,000 Gy/h field ?

Tokamak experience needed to validate technique



ITER divertor

Tungsten armor

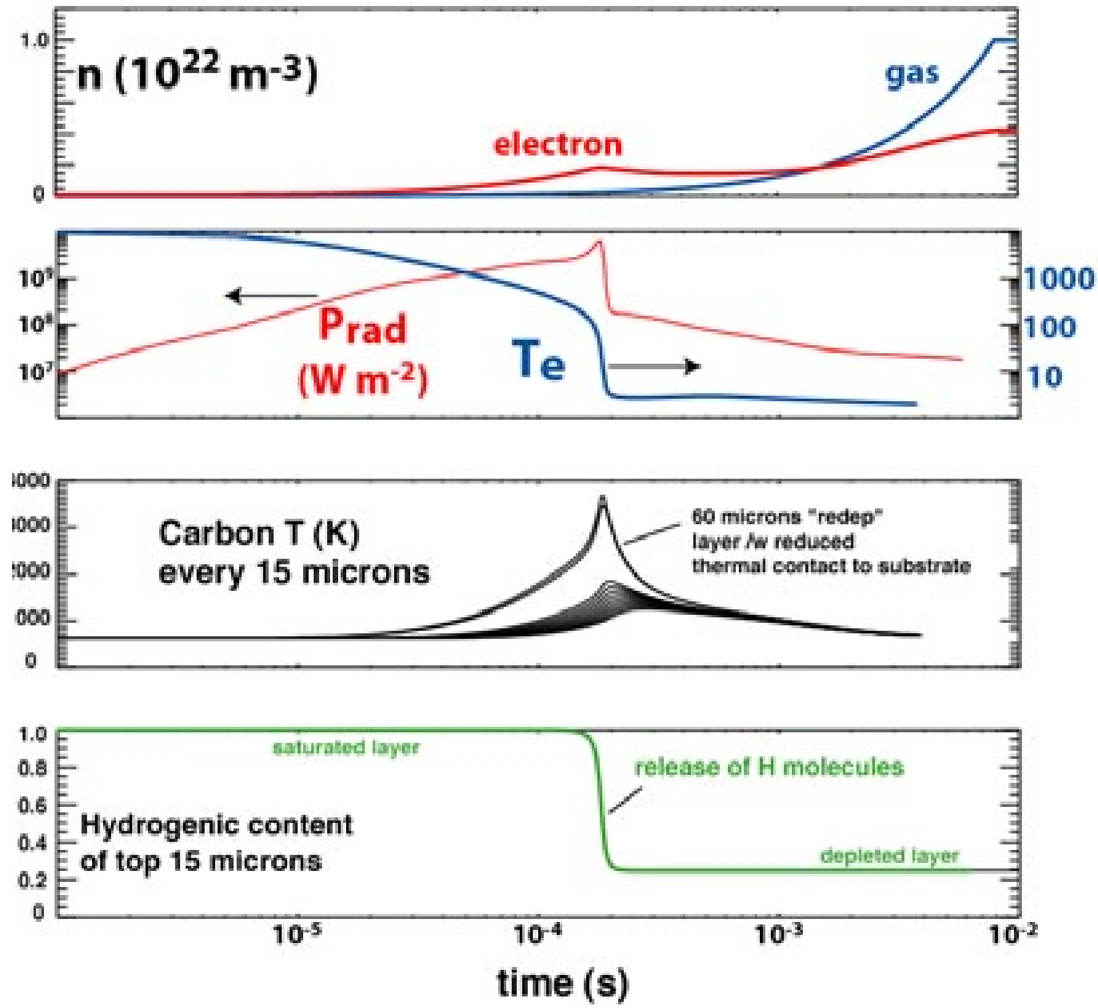
Tritium removal by radiative heating proposed:

Dennis Whyte, as proposed at St. Petersburg ITPA.

- Either: routine gas-jet termination during plasma current rampdown.
- Or: dedicated, short duration low-I_p discharges

- How it works:
  - Large stored energy (~100's MJ) release in < ms via neon radiation
  - All plasma-viewing surfaces are irradiated and heated simultaneously.
  - H/D/T desorbed from surface layers after rapid heating
  - Low ionization fraction and low-energy sheath in post thermal quench plasma do not implant H/D/T back into surface (demonstrated w/ Ne and Ar)
  - H/D/T and injected gas, with total pressure < mbar are pumped by vacuum system (cryopumps or turbopumps) on longer timescale after the termination.

Example: neon termination of ITER



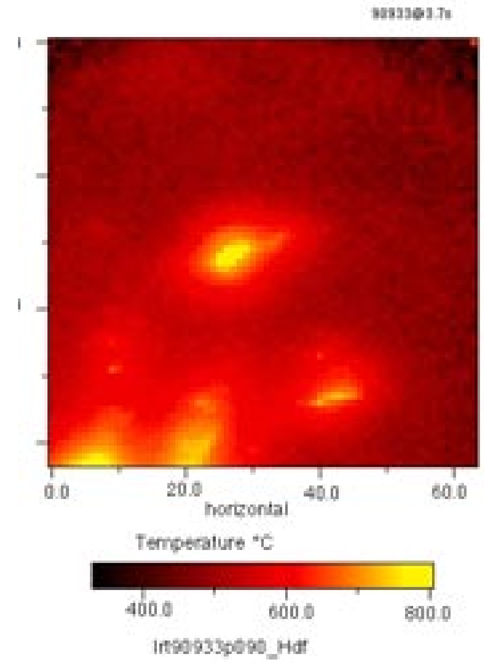
Dedicated gas-jet terminations have several advantages

- Uses only existing features of ITER
  - No vacuum break necessary.
  - No cycling of B_t necessary.
  - Normal pumping system and T processing used.

- Opens possibility of shot-to-shot T inventory control in plasma current ramp down, particularly if predominant codep location is a plasma-viewing surface
  - Technically good idea: the thicker the codep layer, the more difficult it is to remove via heating.
  - Politically good idea: pro-active operational ability to attempt to stay far away from T safety limit.

- Issues and R&D
  - Variability in thermal properties of films. ->
  - Minimization of side-effects (divertor over-heating, substrate damage, diagnostics)
  - Design and implementation of test on present devices (difficult due to lower energy density).
  - Tritium on -hidden surfaces not addressed.

TFTR Limiter Temperature @ 28 MW NBI



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	Tokamak 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
Laser heating….	See next slides….	

T removal rate required for ITER not yet demonstrated in tokamaks

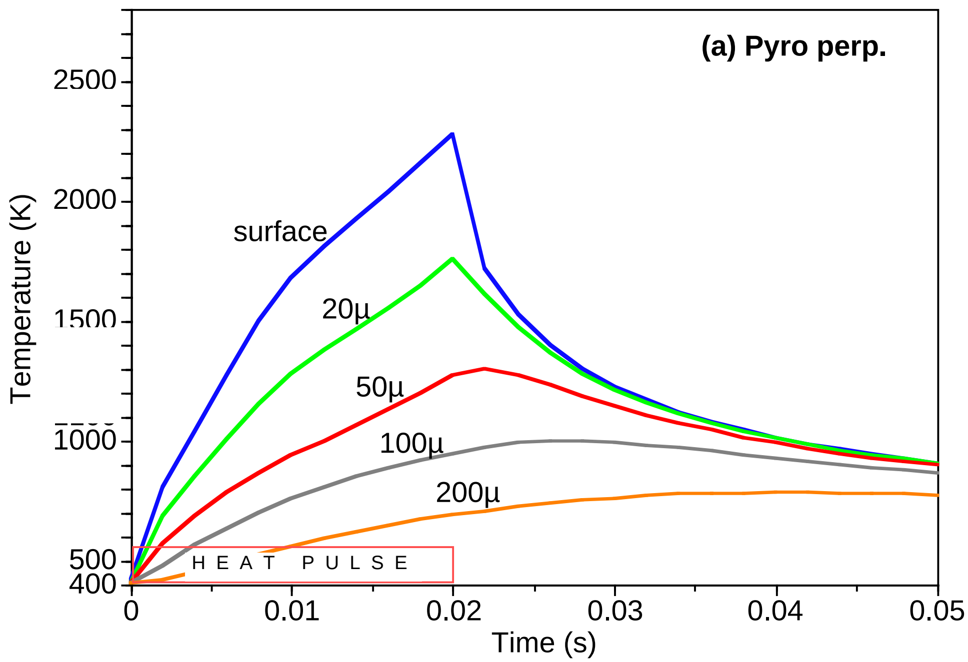
Detritiation by laser surface heating

- Heating is proven method to release tritium but heating ITER vacuum vessel to required temperatures (~350 C) is impractical.

- 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:

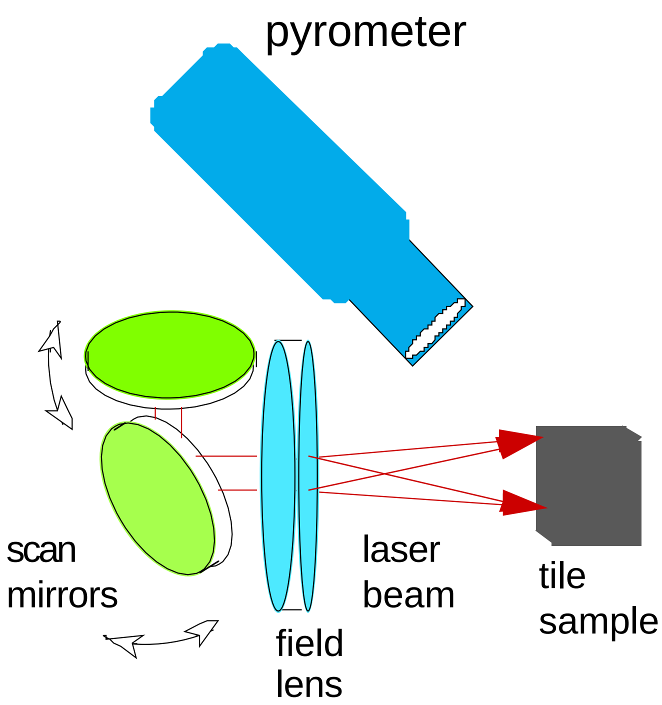
Temperature vs. time at different depths into pyrolitic perp. under 3,000 w/cm2 for 20 ms.



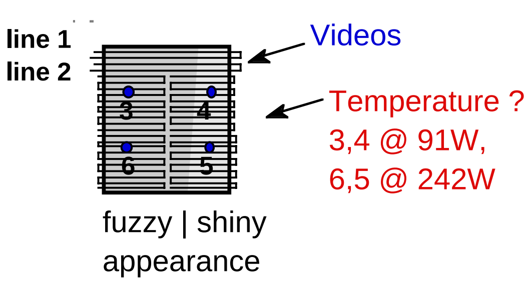
3000 w/cm^2 flux for ≈ 20 ms heats a 50 micron co-deposited layer to 1,000-2,000 K, appropriate for tritium release

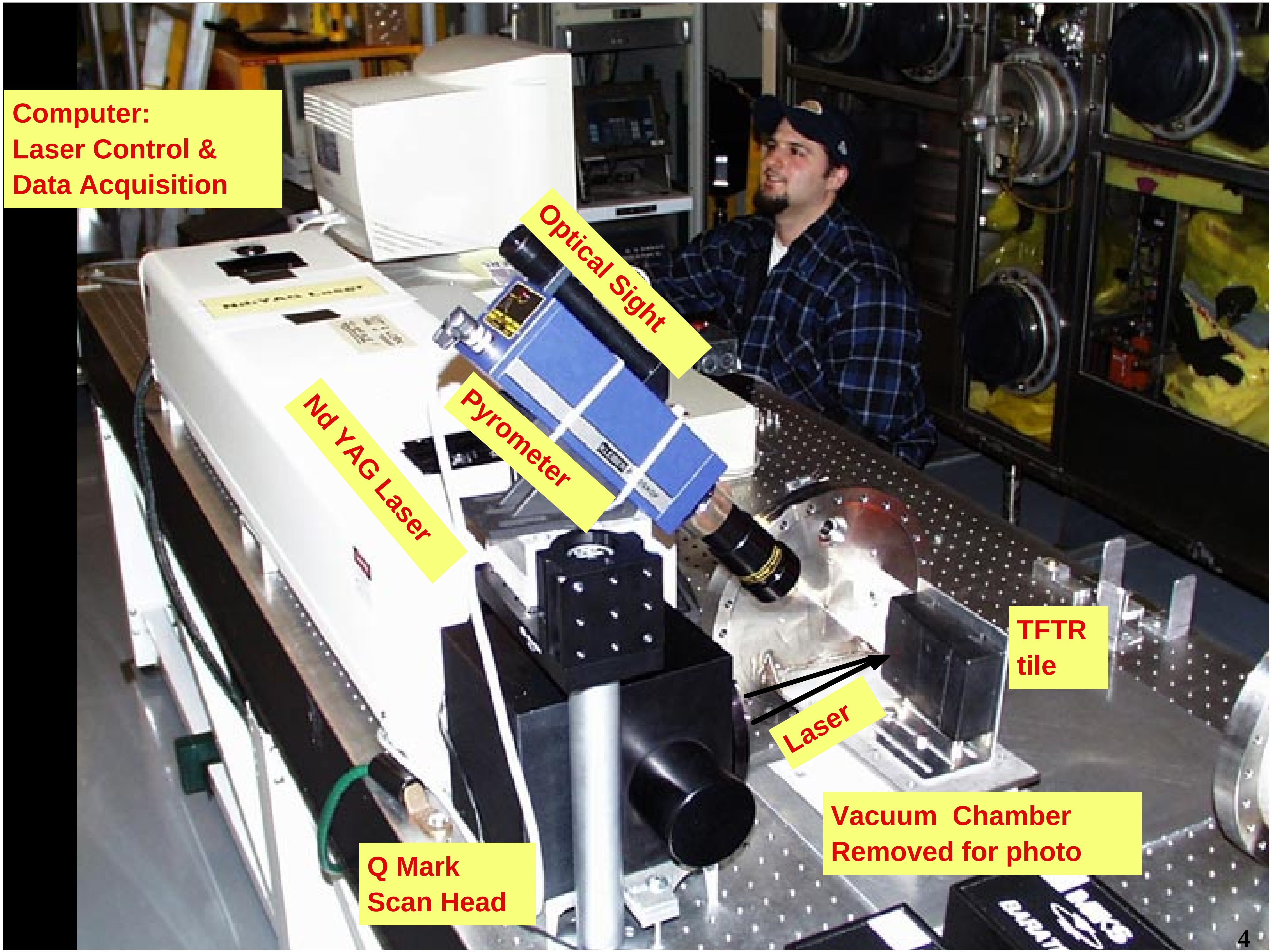
Experimental setup in PPPL tritium area:

- *Nd:Yag laser, continuous wave, 300 watt.*
- *Computer programable laser scanning unit*
- *Samples cut from TFTR and JET tiles exposed to DT plasmas*
- *Irradiated w/laser in Ar or air atmosphere.*
- *Vary raster pattern, laser power, laser focus, scan speed,*
- *Temperature measured by fast (0.3ms), high spatial resolution(0.7mm) pyrometer*
- *Microscope images taken before, during and after laser irradiation*
- *Tritium measured by ion chambers & Differential Sampler.*

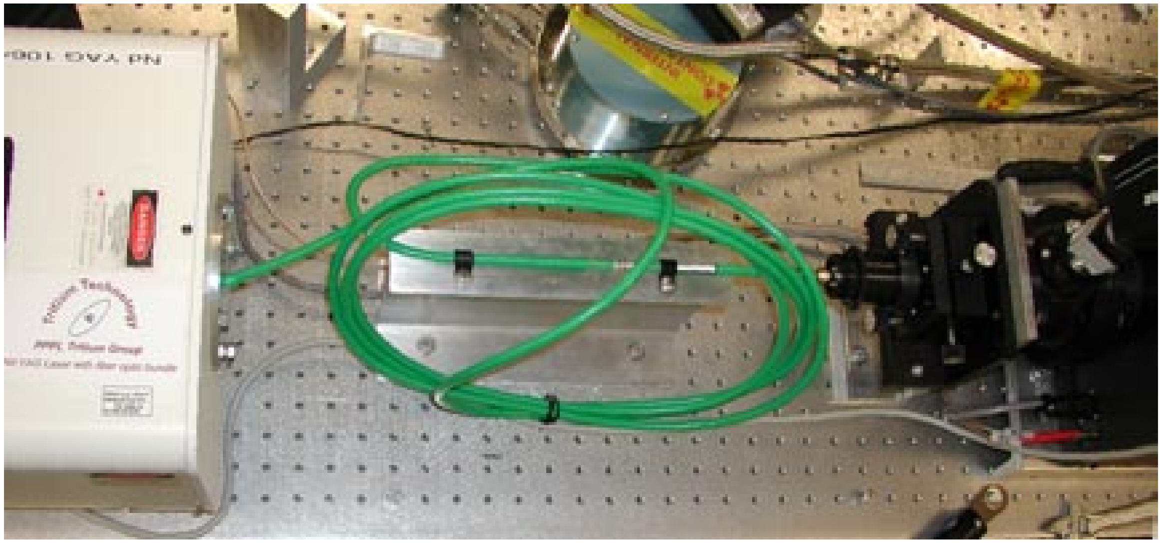


6-zone raster pattern.
line spacing 0.5 mm
•= pyro. view 0.7 mm

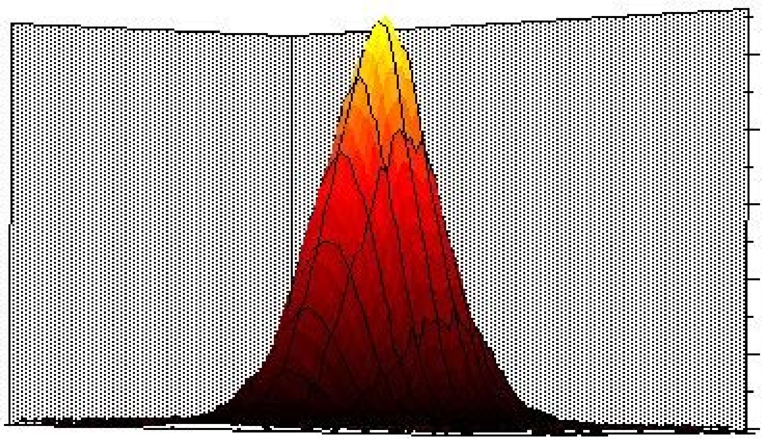



Computer:
Laser Control &
Data Acquisition
Optical Sight
Nd YAG Laser
Pyrometer
TFTR
tile
Laser
Vacuum Chamber
Removed for photo
Q Mark
Scan Head

Fiber Optic Coupling installed between laser and scanner

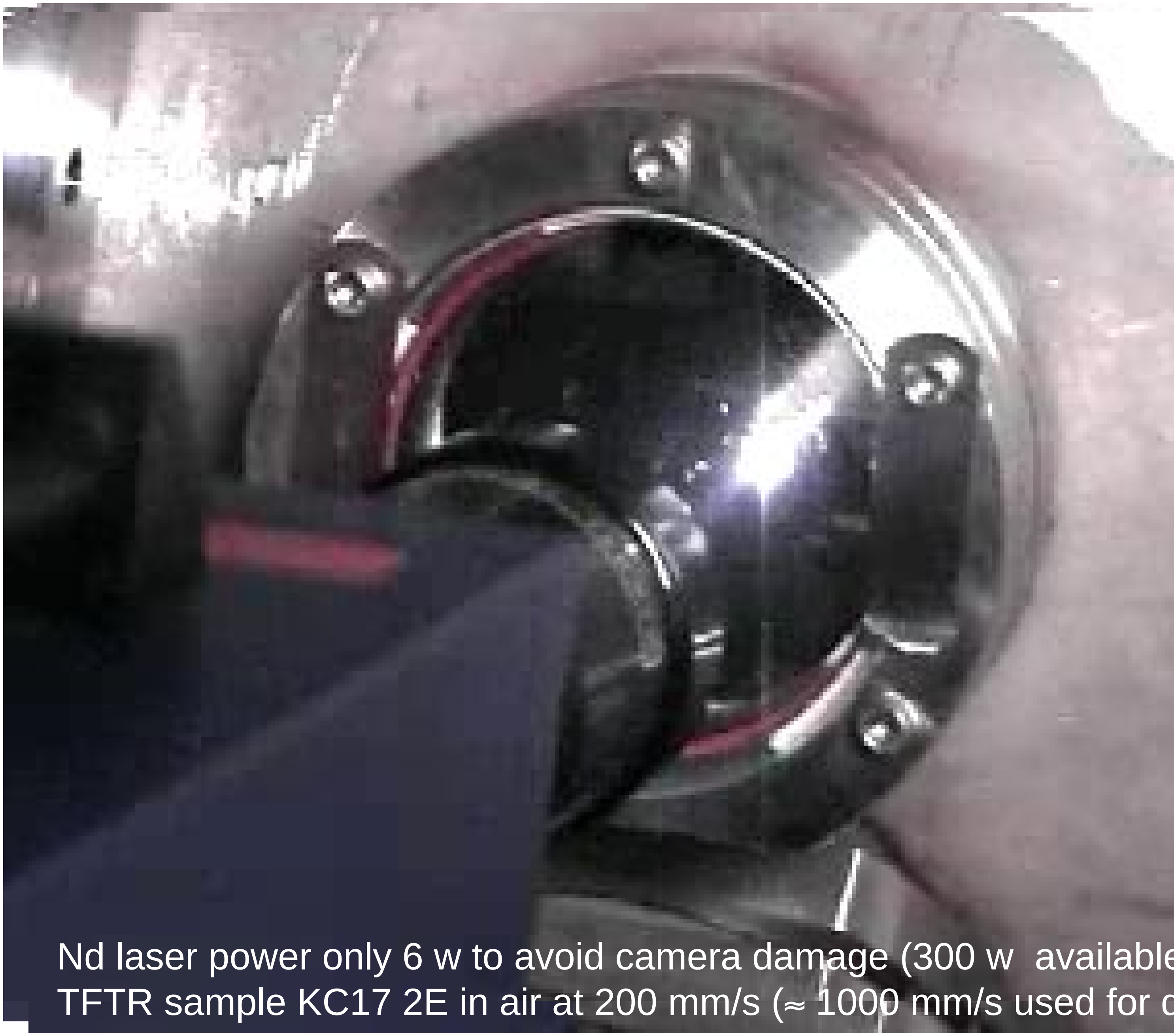


- *For future tokamak applications, laser beam can be transmitted to in-vessel scanner by fiber optic.*
- *5 m fiber optic installed (50 m available)*
- *Fiber diameter 600 microns, armor jacketed, transmission > 90%*



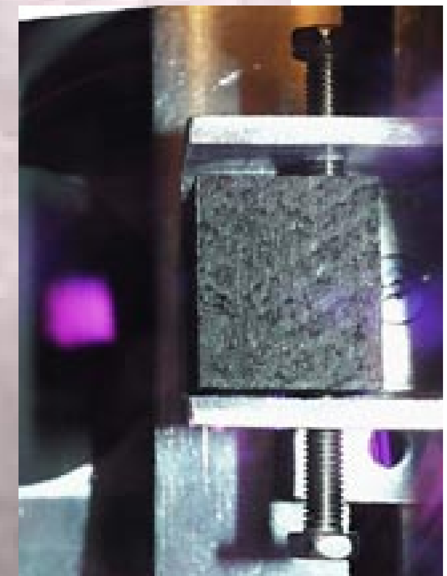
Focal spot intensity profile
FWHM 1.6 mm, 128 W/mm^2

Nd laser in action:



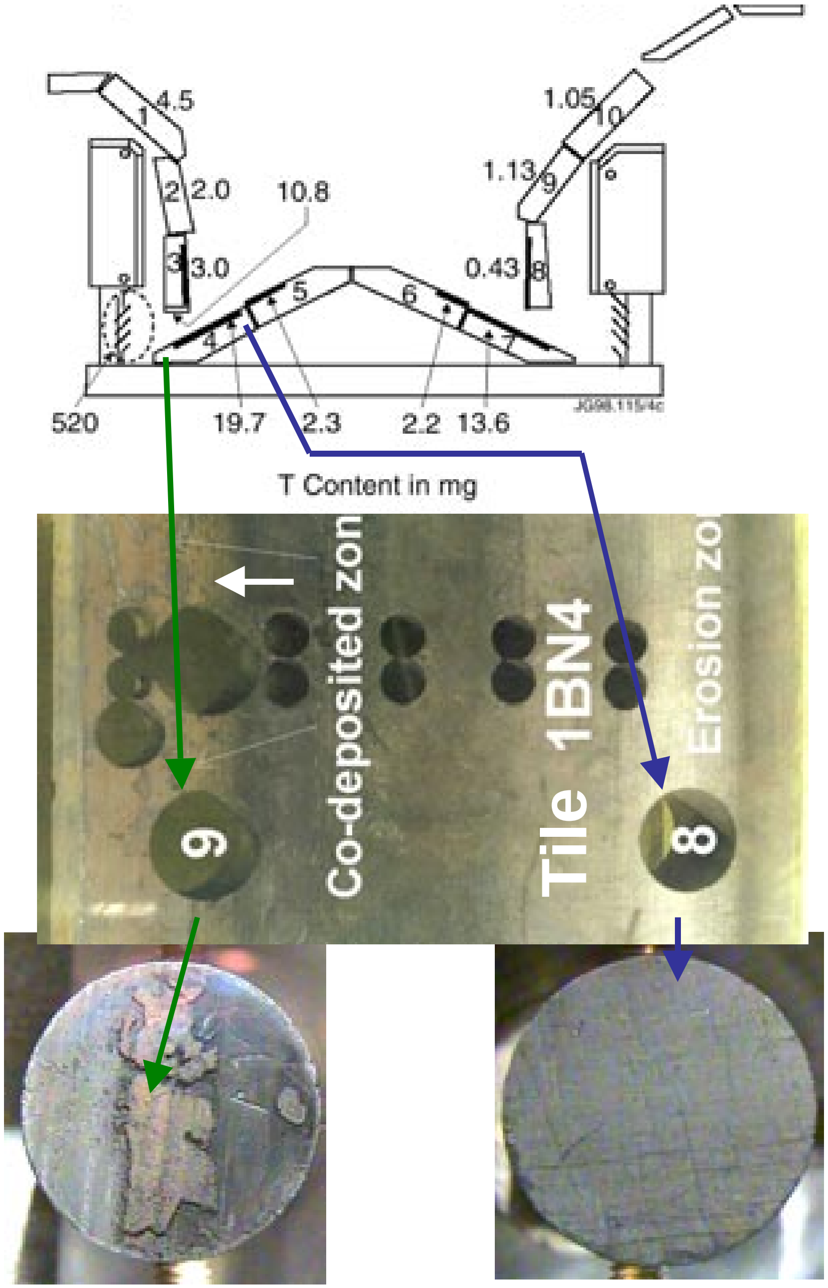
7/8" cube cut from TFTR tritiated tile inside chamber.

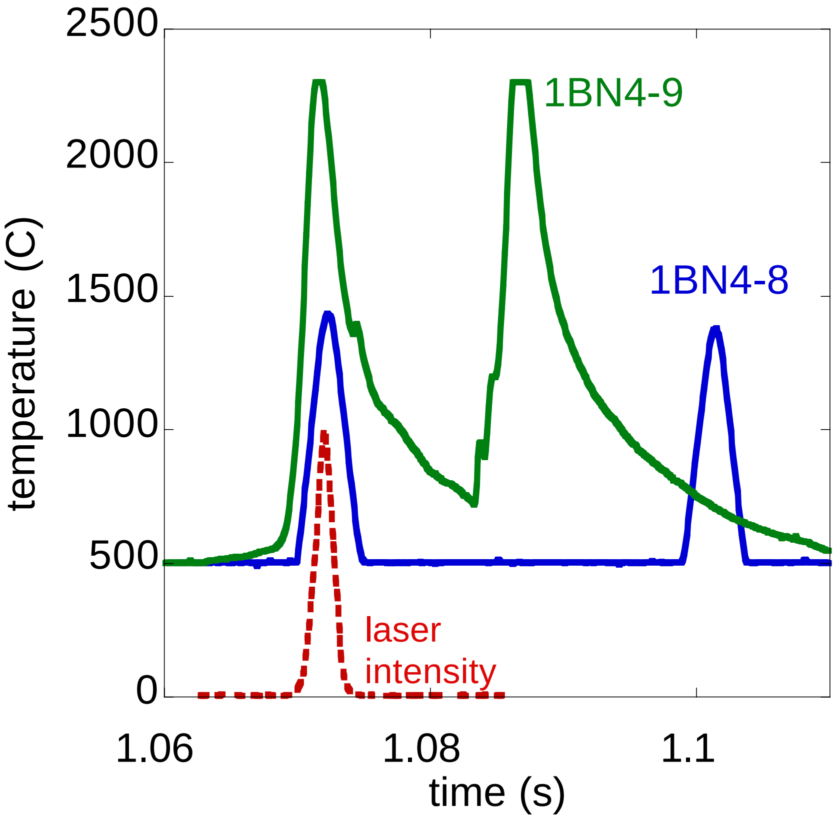
(KC17 2E)



Nd laser power only 6 w to avoid camera damage (300 w available)
TFTR sample KC17 2E in air at 200 mm/s (≈ 1000 mm/s used for detritiation).

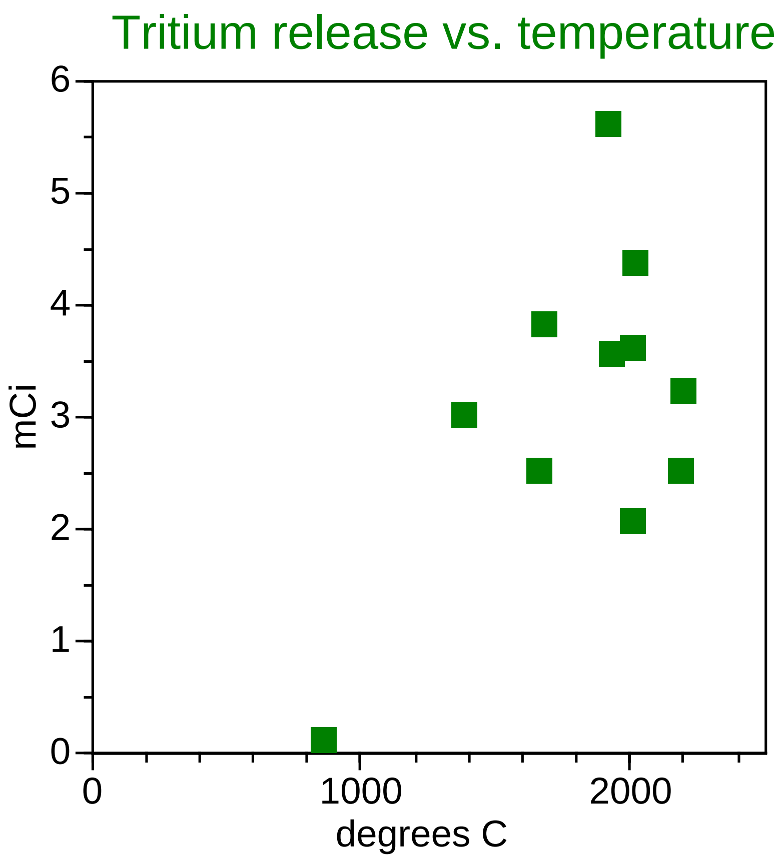
Temperature rise much higher on deposition areas

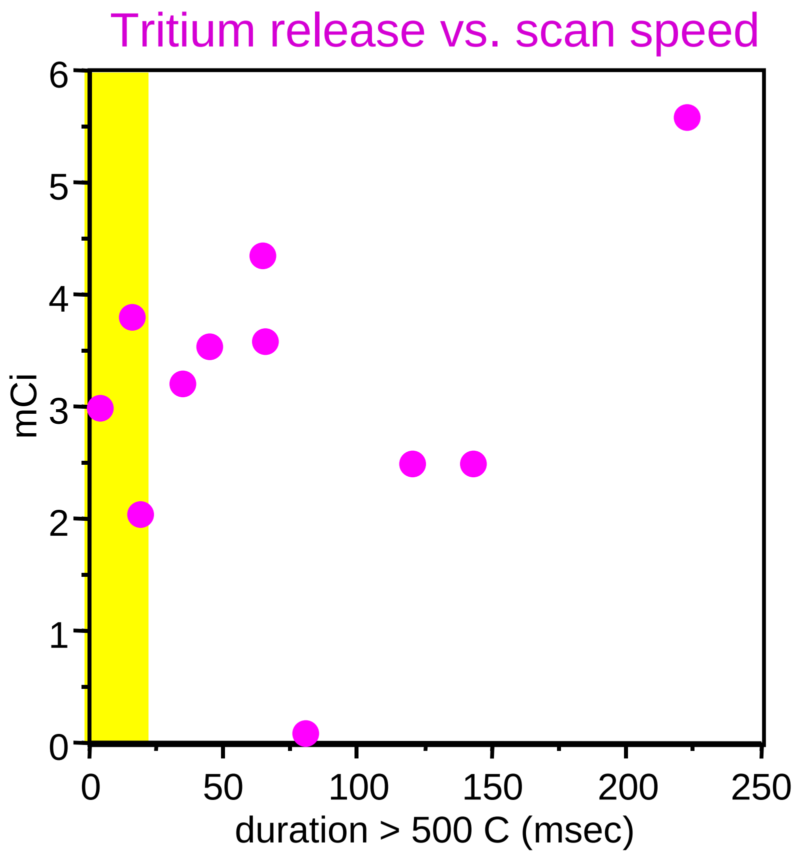




Thermal response to two successive laser pulses (both ≈80 W/mm^2 and 1 m/s) of neighboring erosion and deposition areas on JET divertor tile 4.
As TFTR, the deposition area has much higher temperature excursion + 'tail'.

TFTR tritium release:

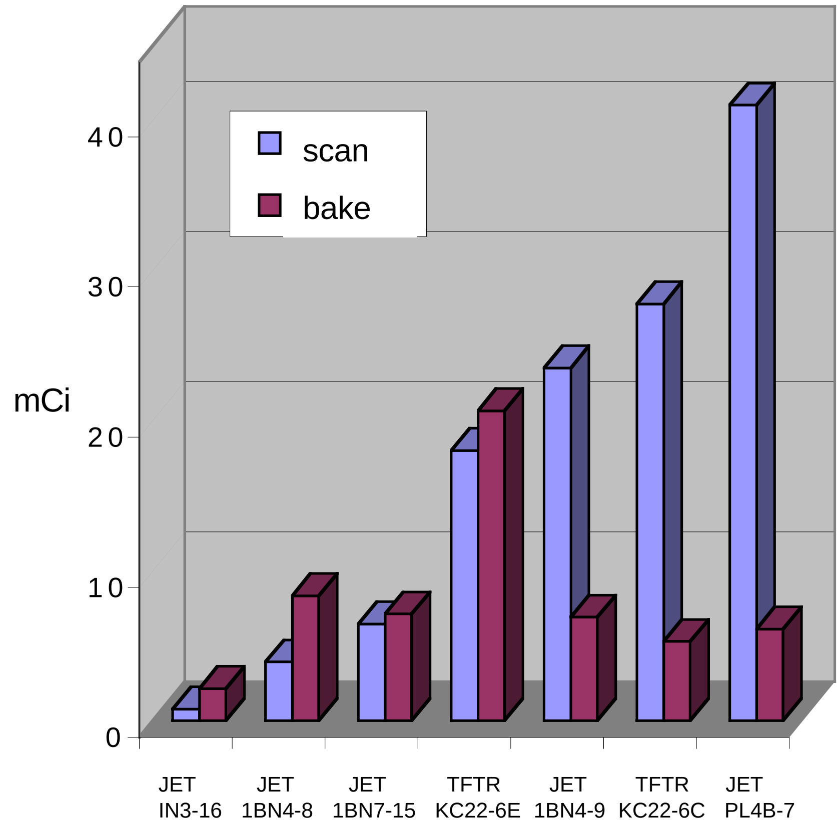




- TFTR surface tritium density varied over factor 3 from sample to sample
- ~ 10 ms heating to ≈ 2000 C gives good tritium release with minimal change in surface (yellow area)

How much tritium is released ?

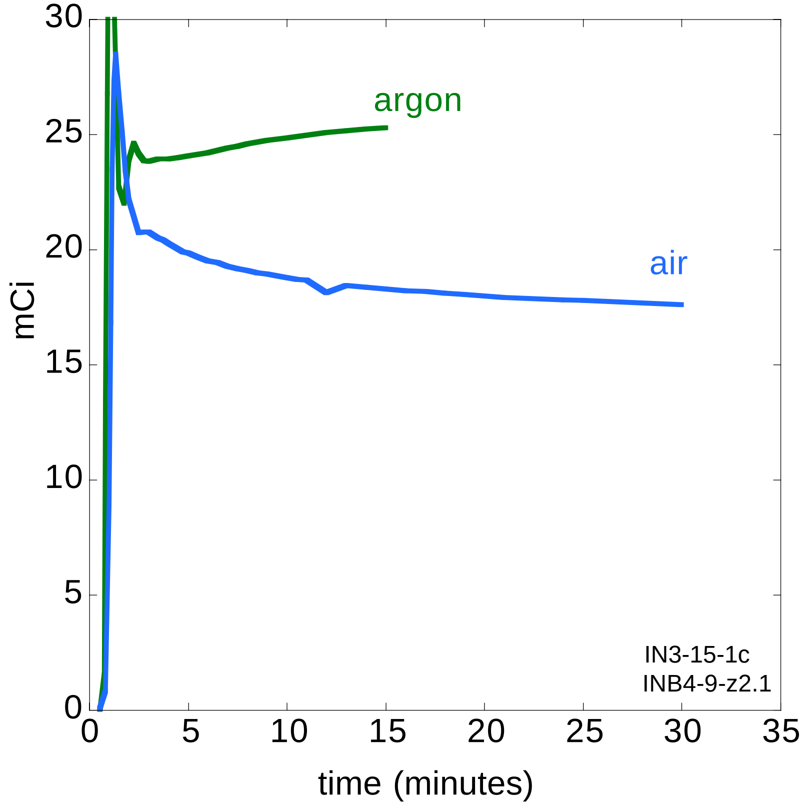
- Release fraction up to 87%
- Scan conditions not all optimized, but detritiation efficiency highest in regions of heavy deposition.
- remaining tritium measured by laser 'baking'.



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

Where does the tritum go ?

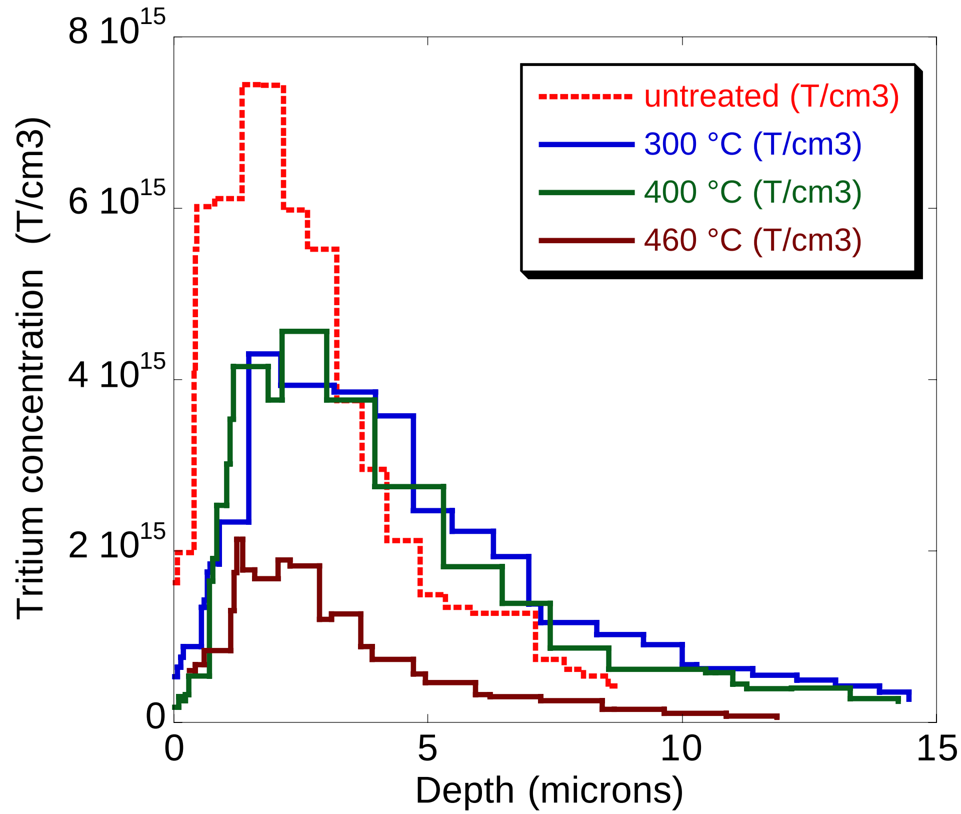
Tritium in closed loop after laser scan



Tritium stays gaseous and can be pumped

no reabsorption in inert atmosphere

Tritium depth profile vs. temperature of JET tile samples (accelerator mass spectrometry).
[Penzhorn et al.,Fus. Eng. & Des. 56-57 (2001) 105.]
(without laser scan)

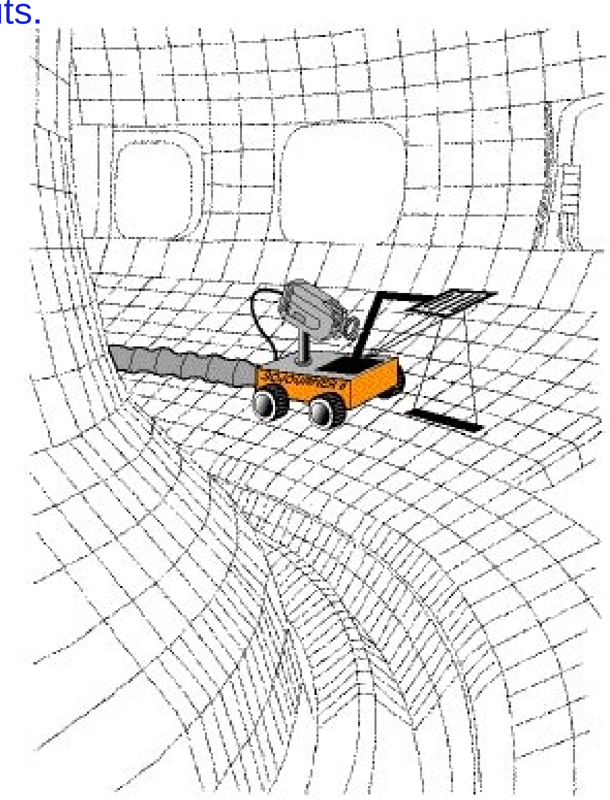


Location of tritium peak unchanged on heating in moist air to 460 C

Status of laser detritiation

- Laboratory measurements show scanning Nd laser can heats codeposit surface to ≈2000 C and thermally desorb tritium [J. Nucl. Mater 313-316 (2003) 496.].
- Up to 87% of tritium has been removed from TFTR and JET codeposits.
- Application to next-step device looks promising.
  - fast cleanup in a next-step machine ($50\ m^2$ in 3 hours with industrial 6 kW laser).
  - convenient fiber optic coupling to in-vessel scanner.
  - 1 micron wavelength of laser minimises gamma induced fiber damage
  - no oxygen to decondition PFC's
  - no HTO to process

artists concept
of potential
in-vessel hardware



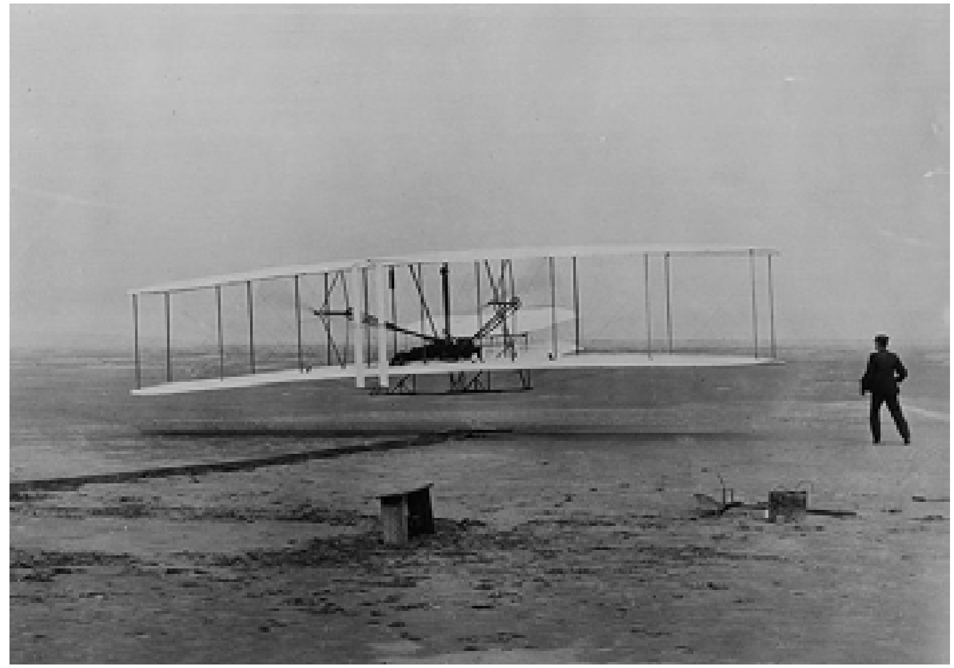
Remaining issues:

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

Demonstration proposed for JET attracted widespread support but no US funding.
How to make the case for near term funding increase for unique ITER needs ?

Concluding Remarks:

- Tritium issues will be heavily scrutinized by regulatory authorities in licensing process.
- Scale-up of removal rate required is ≈ 10^4, higher than any other parameter.
- Understanding tritium migration will not be sufficient. Without tokamak demonstration of tritium removal at relevant rate ITER will not be allowed to use carbon PFCs for DT.
- Lack of major effort 15 years after codeposition discovery suggests a ‘cultural issue’ - tritium cleanup is ‘housekeeping’ and not a concern of real physicists ?
- ITER PFC procurement contracts may be set in as little as 5 years.
- For decades of work on carbon to be relevant to next-step tokamak there is an urgent need for tokamak demonstration of tritium removal by a method that is extrapolable to ITER.



First controlled, powered flight - Wright Brothers 1903

Question:

Will ITER be primarily a hugely expensive plasma wall interaction experiment ?

Or will we attract talent and resources to overcome the outstanding issues and ‘take flight’ to a new era of fusion energy....