Is carbon a realistic choice for ITER's divertor ?

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ORNL Workshop on Plasma-Surface Interactions for Fusion
Oak Ridge, Tennessee, March 21–23, 2005.

ITER plasma facing materials:

Brief History:

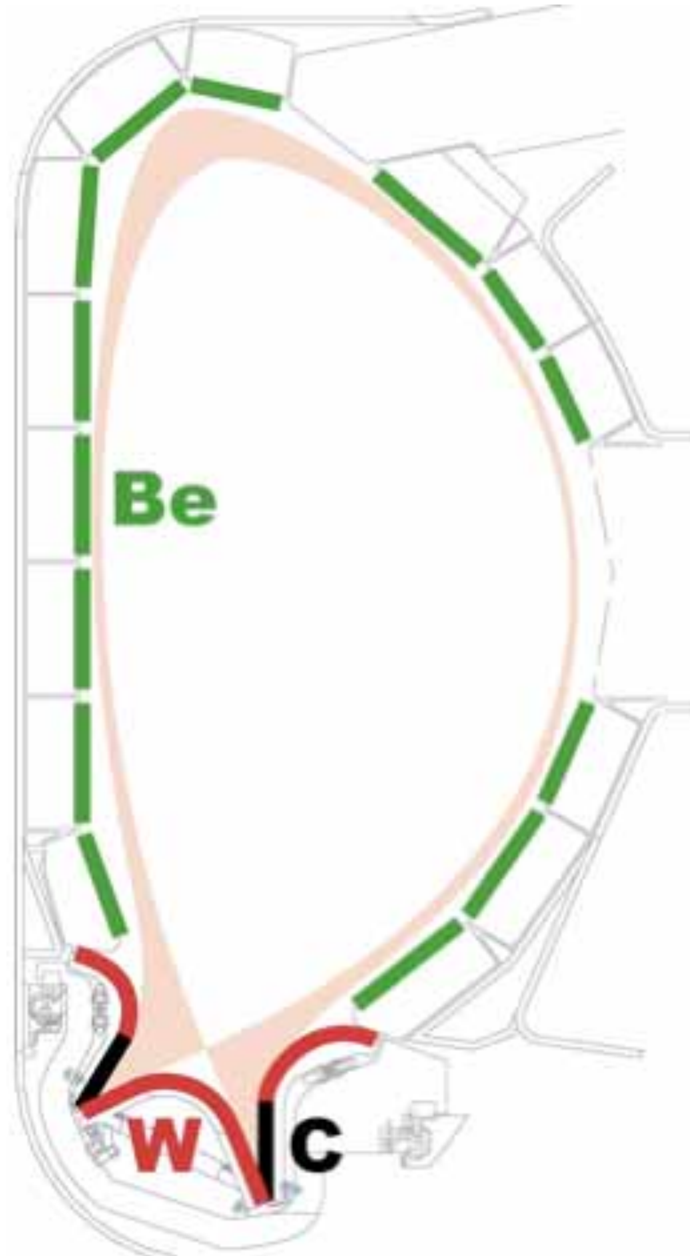
- 1978 PLT switch from W to C limiters enables first thermonuclear temperatures.
- 1988: Codeposition discovered on JET & TFTR
  - recognized as problem for T inventory.
- Be tested in ISX-B, then on JET wall + divertor to mitigate codeposition and getter oxygen.
- 1990 JET Be divertor melted - back to carbon for divertor
- Early 1990's: Be chosen for ITER wall, W for ITER dome & baffle and minimal C for divertor strike points to minimise codeposition and erosion.

Since then:

- Heavy T retention on TFTR/JET
- Cross field transport, ELMs
- Be/W alloys (PISCES) …

Present ITER PFC strategy:

- Use CFC in divertor for H/D operation,
- "Assess" H-isotope retention and melt layer loss for W.
- Decide on W or C divertor for DT operations



International Experimental Thermonuclear Reactor is inevitably an experiment.

PFCs could be biggest technical risk:

Risk	Potential consequence	R&D to mitigate risk
Divertor damage by ELMs, disruptions or VDEs.	Premature divertor failure, long down time to replace it	ELM research worldwide, disruption mitigation on DIII-D, W brush on C-mod, Dis. Simulat.
Damage on Be limiters, upper strike point	Longer down time for ITER wall replacement, success questionable.	Be/W planned on JET 2009. Modeling still immature
W influx to core plasma	Failure to achieve Q = 10	Asdex-U investigates central heating, full W wall by 2007; Be/W planned on JET 2009.
Unforseen mixed materials effects	PFC failure, long down time.	$Be_{22}W$, $Be_{12}W$ alloys on PISCES; Be/W planned on JET 2009.
Neutron damaged material behaves poorly ?	????	Materials Test Facility envisioned.
Radioactive/toxic/explosive dust particles accumulate	Plasma ops. suspended till dust removed (how?)	???
Tritium removal unsuccessful	ITER limited to ~ 100 pulses	???

Risk Management Perspective:

- Fusion Energy Research can be seen as a risk management project.
  Ongoing R&D can:
  - better quantify risks,
  - discover unforseen risks, and
  - validate on current tokamaks innovative solutions that minimise risks to ITER's burning plasma program.

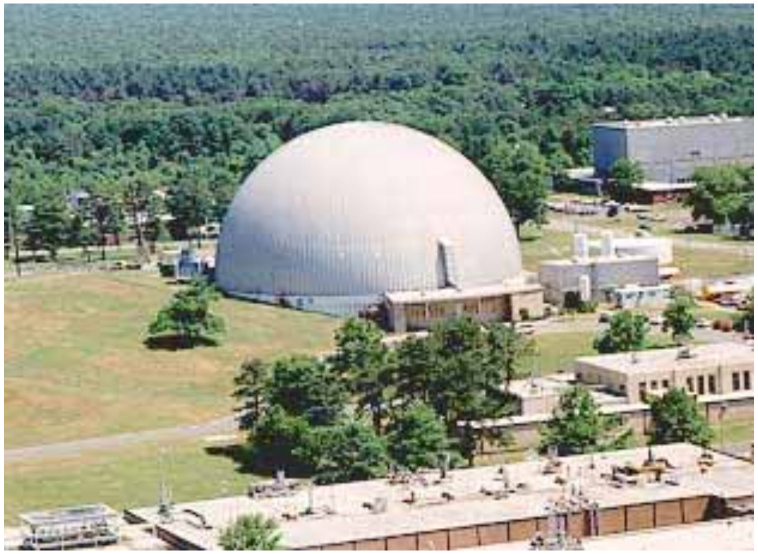
- Risk management issues common in space exploration, investment, insurance, and new product development e.g. drugs, software….

- Worthwhile looking at experience and lessons learned in other fields when considering workshop goal "to suggest most important next steps…".

e.g. "Programatic Risk Analysis for Critical Engineering Systems under Tight Resource Constraints", R. L. Dillon, et al., Operations Research 51 (2003) 354.


Cassinni-Huygens mission to Titan

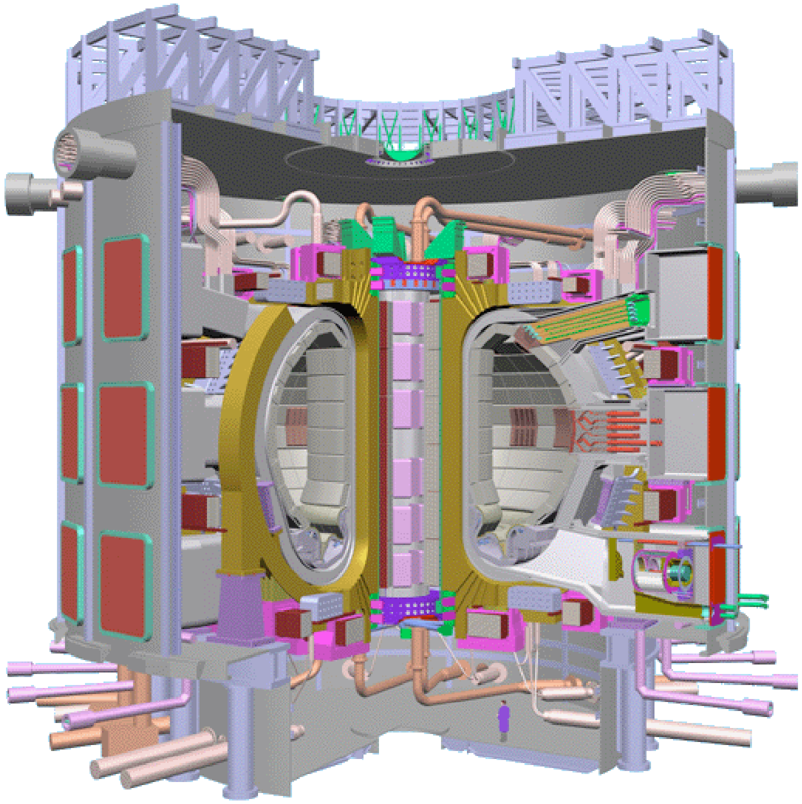
http://www.nasa.gov



Brookhaven High Flux Beam Reactor once the premier source for neutron science, is undergoing decommissioning

Risk Assessment:

1. What is the potential impact of the problem?
2. How well is the underlying physics understood ?
3. What technology is needed to resolve the problem ?
4. Are we on a path to develop the technology required ?
5. What R&D strategies are available to mitigate the risk ?



Risk Assessment (1):

What is the potential impact of the tritium removal problem?

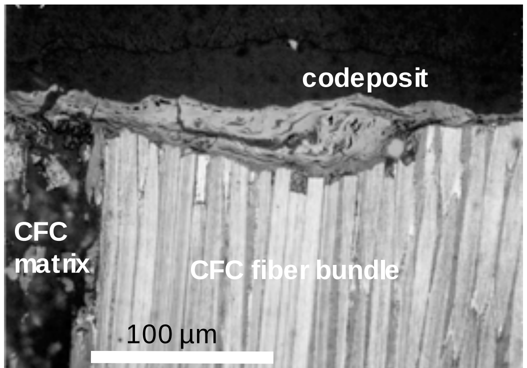
- Tritium inventory is a major safety factor and will be heavily scrutinized by regulatory authorities in licensing process.
- Public very sensitive to tritium issues.
- Cost of unforseen delays ≈ $1 million / day.
- At stake is not just the success of ITER, but the public credibility of fusion energy if ITER spends too long as PWI experiment.

Brookhaven High Flux Beam Reactor Area now cleared of experimental equipment

Tritium retention

TFTR - 51%

Microstructure of codeposit



(M Paffet)

Tritium on limiter (Ci)

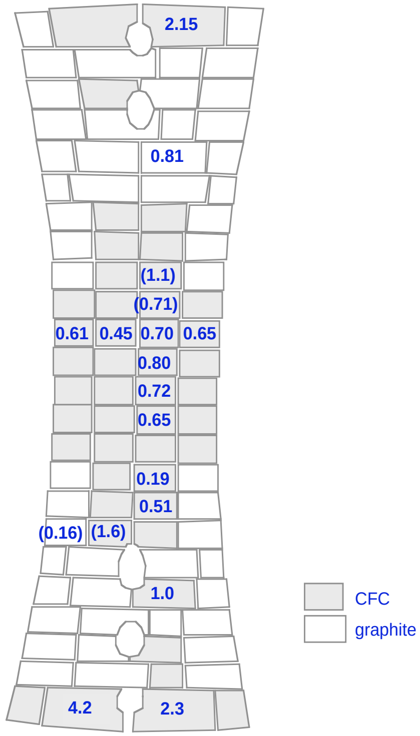
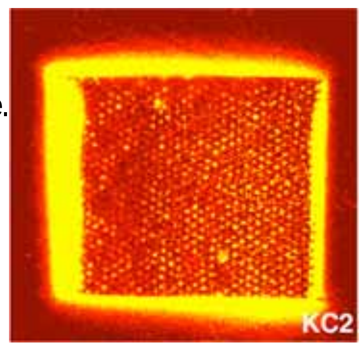


Image of tritium on tile. Note most T is on sides. (Overal 15% of D in tile gaps)



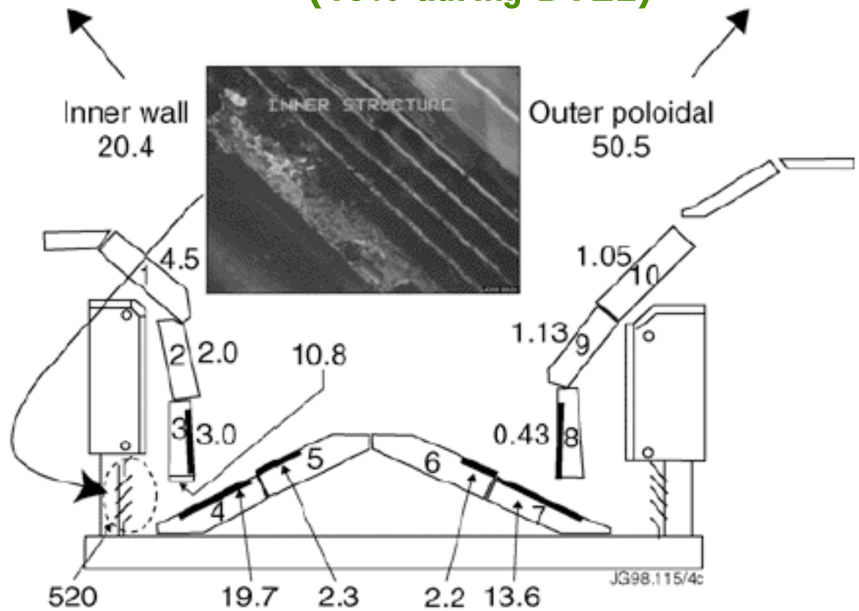
(T.Tanabe)

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.

JET - 17%

(40% during DTE1)



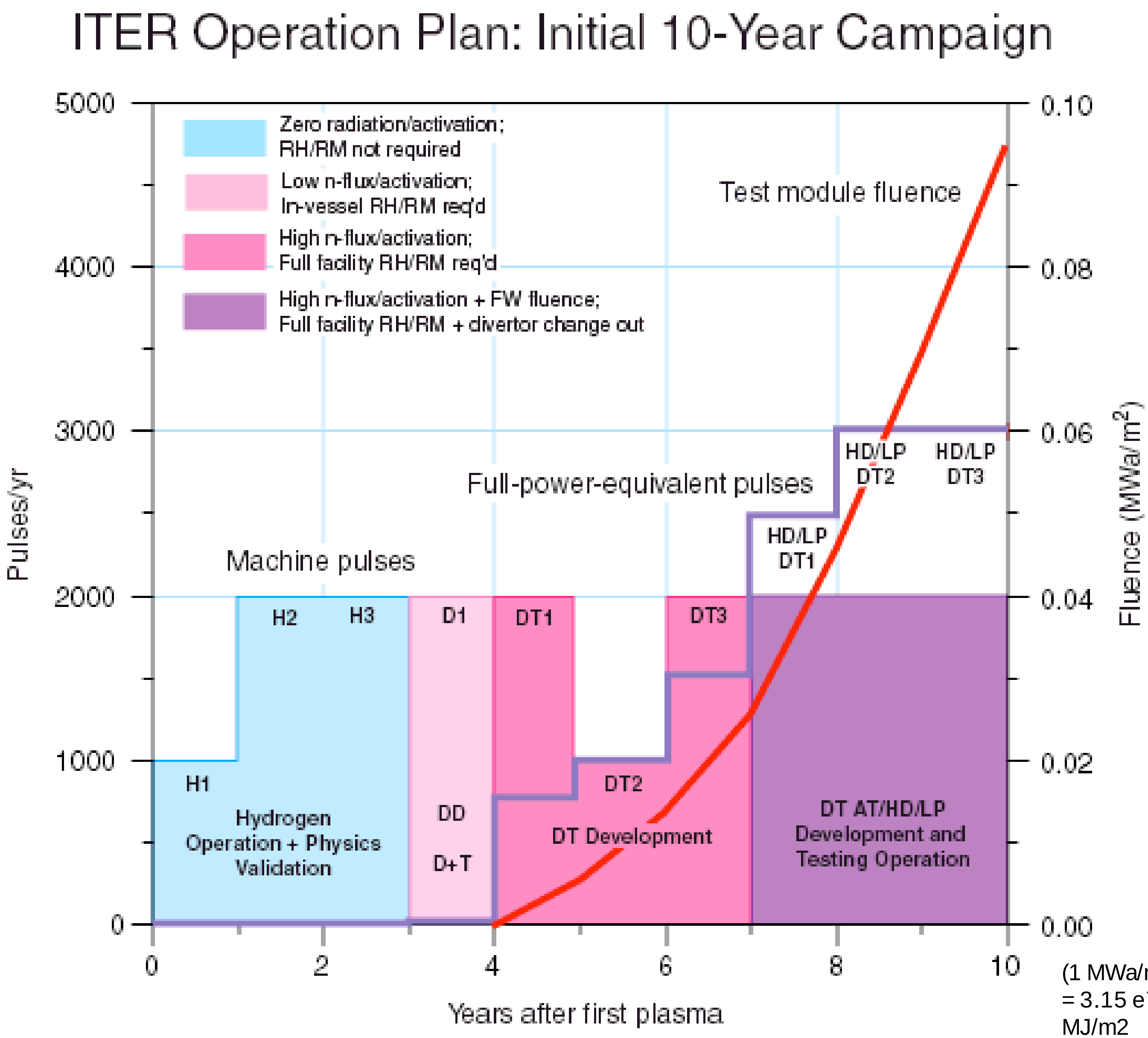
T Content in mg

P. Coad, UKAEA/JET

Most remaining tritium in sub divertor region (3.4 g).
Tritium was also found on the inner divertor louvres (0.5g) and tiles (<0.1 g).

Some tritium found in bulk of CFC tiles (about 10% of surface tritium, R.-D. Penzhorn). (tritium in bulk can only be removed by replacing divertor.)

ITER duty cycle is biggest change from current tokamaks



(1 MWa/m2
= 3.15 e7
MJ/m2
= 1.39 e25

- Pulses 400 s duration, ~20 pulses / day
- Tritium can accumulate very quickly, with little time available for removal

ITER scale up:

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%	4 - 10%
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	≈ 10 h (overnight)
Fraction of tritium removed	50%	50% (prior to venting)	> 90% necessary
Tritium removal rate	~ 1 g /month	2 g / month	Up to 100 g / d or 50 µm codeposit / d

x10^4 scale-up required T removal rate - higher than any other ITER parameter **+** Lower R&D effort than any other area **=** High risk

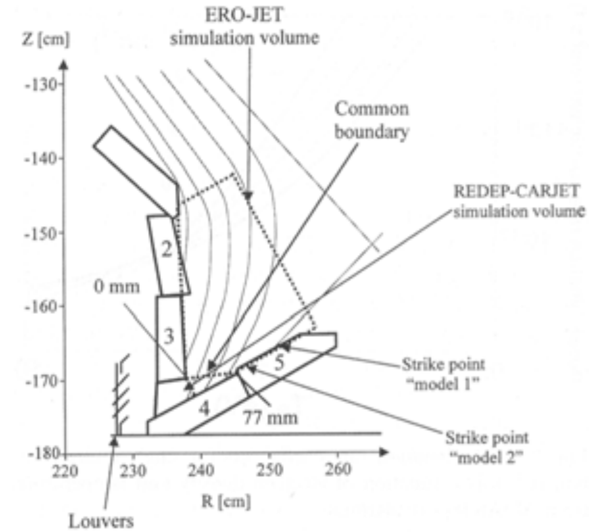
Risk Assessment (2):

How well is the underlying physics understood?

- Modelling underestimates JET retention x40
- Model cannot reproduce detached plasmas on DIII-D (but has been successfully benchmarked in attached plasmas (Whyte)).
- Major uncertainty is in chemical erosion yield
- Retention could be lower if:
  - Be layer impedes chemical sputtering (Doerner)
  - Chemical sputtering flux dependent (Roth)
- Retention could be higher if:
  - Wall is deposition area (Kukushkin)
  - Significant C erosion by ELMS
- Additional uncertainties from mixed materials

"considerable uncertainties" remain

- use conservative 2-5 g / pulse predictions

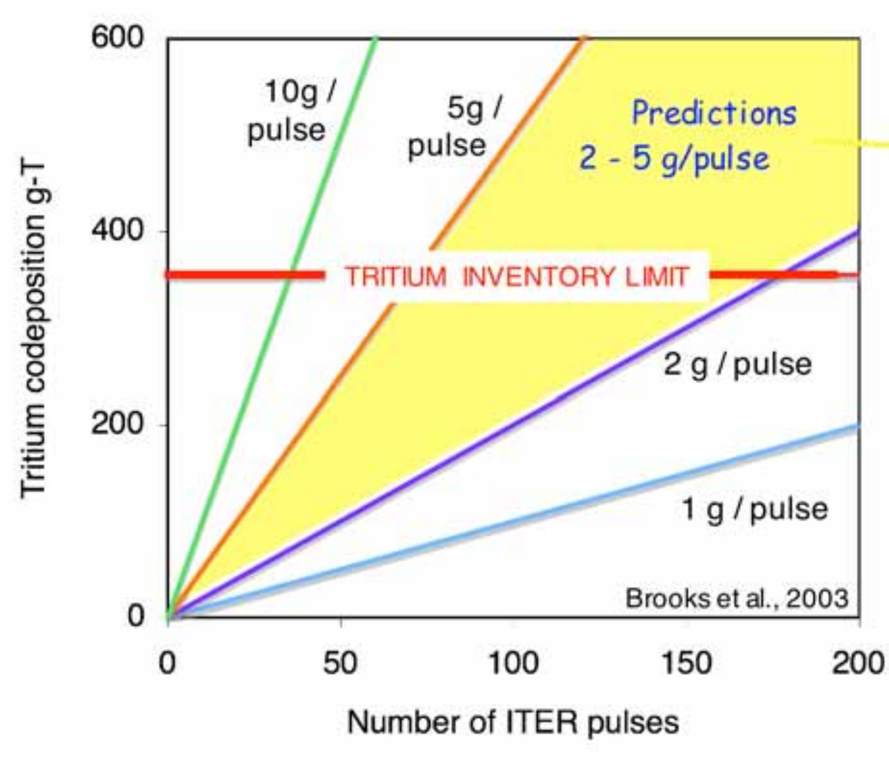


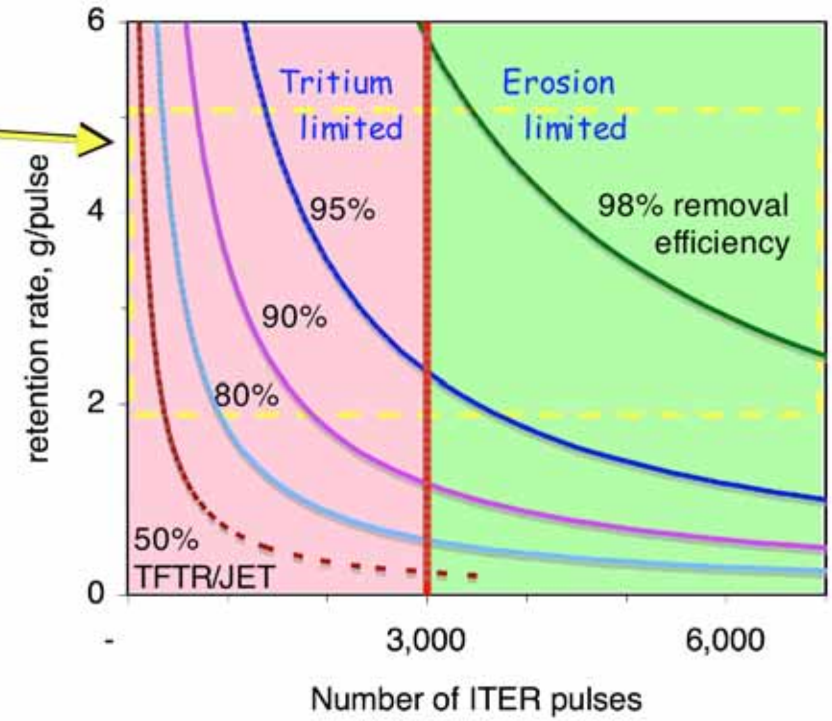
JET MkIIA inner divertor geometry and calculation setup.

Brooks et al., J, Nucl. Mater 313-316 (2003) 424

- Coupled REDEP and ERO-JET impurity transport calculations for sputtered wall/divertor carbon.
- MolDyn molecular dymanics calculations of carbon/hdrocargon particle reflection at hydrogen-saturated carbon surfaces.
- ADAS full collisional radiative carbon ion recombination rate coefficients.

ITER retention could be 40 - 100 g / day in 50 μm codeposit





Residual tritium remaining after active removal can also stop operations at 350 g T limit.
Divertor exchange may be only way to remove it - IF it is on divertor.

Appropriate removal goal to enable plasma operational schedule as planned:

= capability to remove 98% of tritium and restore wall conditions in overnight shift

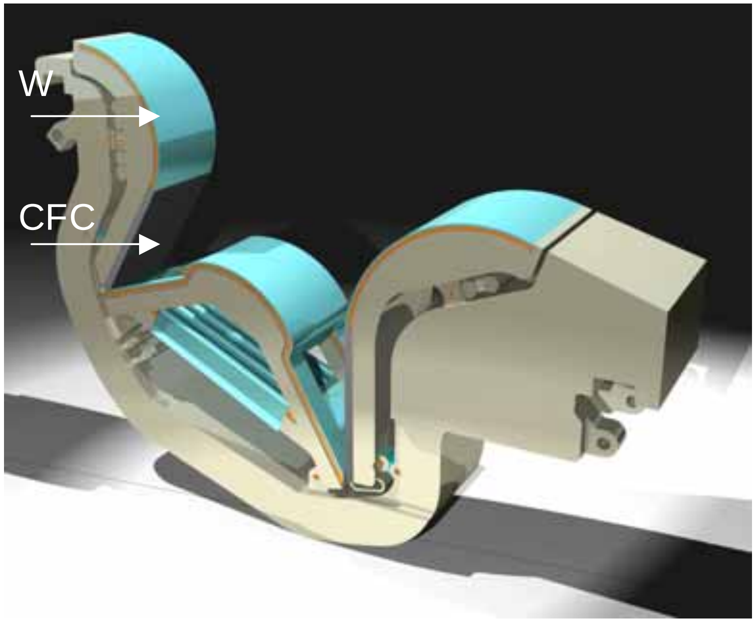
Risk Assessment (3):

What technology is needed to resolve the problem?

Tritium removal - 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
   - or ...

ITER divertor cassette



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

Castellated structures for W and CFC.
140m^2 of gaps, 1μm layer > 35g-T !!!

–location of tritium uncertain (under divertor dome, in flakes, bulk of CFC tiles....)

Tritium removal by oxidation:

MERITS:

- access to all surfaces in vessel (including hidden areas)
- Simple to implement, no in-vessel hardware
- Lab experience, some tokamak experience (albeit at low or unmeasurable removal rates)

CURRENT R&D:

- Laboratory tests by Haasz Group for later application to DIII-D (constrained by funding).
- DIII-D proposal (Stangeby) in conjunction with ^{13}C tracking
- TEXTOR: HeO RF and HeO GDC
- China HT-7: baking , glow discharge, oxygen ICR
- FZ- Juelich / CIEMAT / IPP Garching: ozone, alternatives, GDC….

RISKS:

- Collateral damage to in-vessel components.
- Delays - Fast removal requires T> 240 C, incompatible with pressurized water cooling.
- Delays in re-conditioning of plasma facing surfaces, Be gettering of O in plasma questionable.
- Tungsten or boron impurities found to inhibit oxidation - more delays.
- Redeposition inside tokamak before recovery ?
- "..processing something like one liter of water containing about one hundred gram tritium on a more or less daily basis is a showstopper for fusion." Manfred Glugla
- NO ITER REQUIREMENT FOR PROCESSING EXHAUST FROM OXIDATIVE DETRITIATION !

Tritium removal by ablation

MERITS:

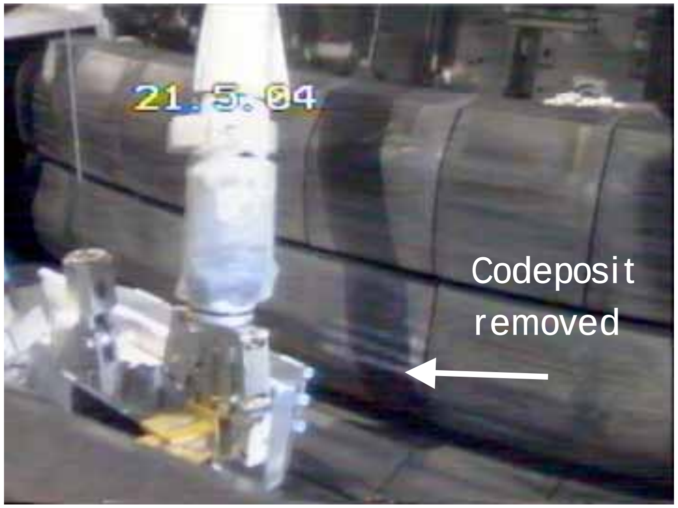
- some tokamak, lab & industrial experience,
- whole codeposit removed

CURRENT R&D

- EU FT task force- assessment of JET tiles ablated by flashlamp.
- CEA- pulsed Nd laser system being developed for JET.

RISKS:

- Access difficult to hidden areas, under divertor dome, tile gaps… ?
- Debris falling in inaccessible areas ?
- Reactive radicals produced that would redeposit in-vessel ?
- Lack of excimer laser fiber optic transmission over required distance.
- Flashlamp incompatible with 6.1 T ?
- Compatibility with 10,000 Gy/h field ?
- Will removal rate and efficiency be sufficient ?



Laser ablation demo in JET
(Gibson et al. PSI-16)

Tritium removal by laser heating:

MERITS:

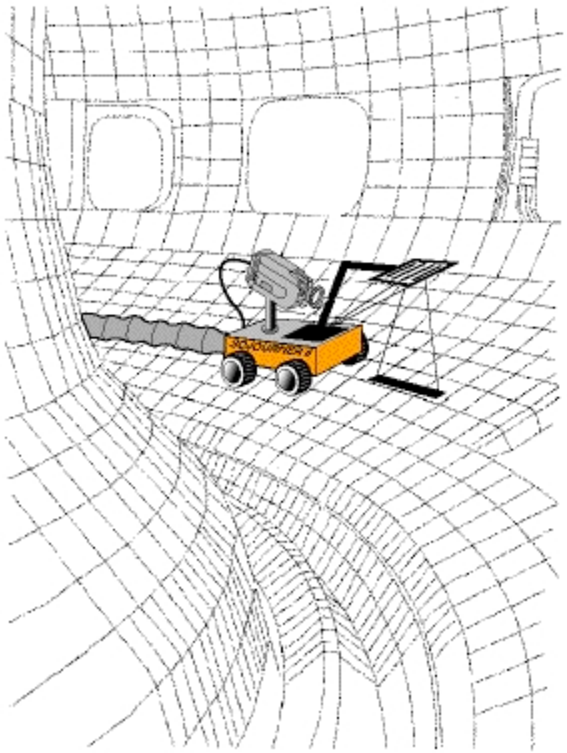
- PPPL Lab experience very encouraging, up to 87% T removed
- T_2 gas recovered - avoids DTO processing
- Fast - 3 kW laser could clean 50 m^2 in 3 hours
- Avoids deconditioning plasma facing surfaces
- Convenient fiber optic coupling at 1 μm wavelength
- Compatible with 6.1 T
- Compatible with 10,000 Gy/h field
  - Extrapolable to ITER.

CURRENT R&D

- None - no funding

RISKS:

- Need to demonstrate detritiation of tile gaps and other hidden areas.
- Need to increase efficiency to > 90%
- No tokamak experience (planned but not funded)
- Stability of thick codeposits remaining ?



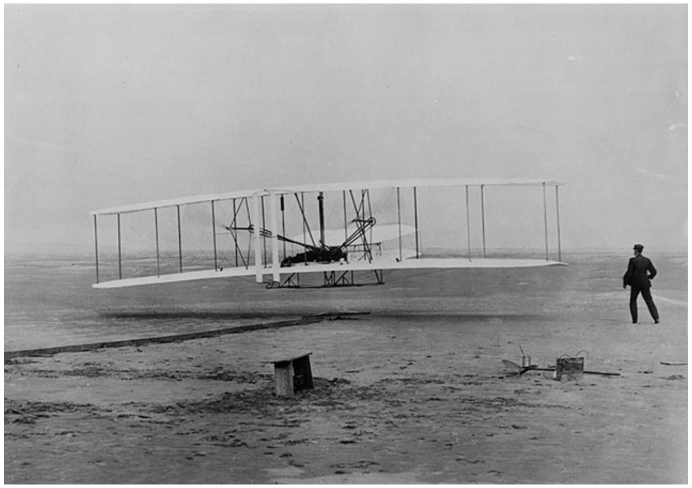
Sojuner II

Conceptual design for laser detritiation of ITER inspired by successful Sojuner I mission to Mars

Other methods:

Technique	Merits	Current R&D	Limitations/RIsks
Glow discharge cleaning	Tokamak experience	HT-7, FZ-Juelich, IPP-Garching	Incompatible with 6 T field
ICRH	Tore Supra experience 4e22 C/m^2/h -> 1 µm/h	HT-7	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)
Flash heating from controlled disruptions	Easy to implement	DIII-D	No access to hidden areas. Damage of PFCs ?
N_2 scavenger gas	Inhibits codeposition	Moderate decrease seen on JET	R&D needed on gas phase chemistry.
Cathodic arc cleaning			Damage to underlying tile?
CO_2 pellets			Damage to underlying tile
UV light			Ineffective

Are tokamak tests really necessary ?



"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

1. 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.

2. WALL CONDITIONS: 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 a tokamak.

3. RE-CONDITIONING: At present there is no 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 think it too risky because of potential collateral damage ?

Risk Assessment (4):

Are we on a path to develop the technology required ?

Realistic near-term milestone:

50 μm scale of codeposit in current tokamaks = that expected after 1 day of ITER ops.

Can we remove >90% of D from current tokamak <u>and</u> run high performance plasmas next day ?

PRESENT STATUS:

1. Development path from laboratory tests to ITER not specified.
2. Tokamak tests are rare and way short of removal rate required.
3. Implications for wall conditioning not explored
4. Implications for tokamak exhaust processing system not explored
5. Funding low or non-exisistent
6. Risks unacknowledged - whose problem is it - physicists or engineers ?
7. Compare 14 talks on ELMs at PSI-16 to just 2 on tritium removal !

CONCLUSION: Rate of progress over last 17 years does not extrapolate to success

Risk Assessment (5):

What R&D strategies are available to mitigate the risk ?

1. R&D to address risks of CFC divertor in DT phase
2. R&D to address risks of metals in DT phase

   (is funding sufficient for both ?)

3. R&D to address risks of mixed materials
4. R&D to address risks of dust

1. CFC divertor in ITER DT phase:

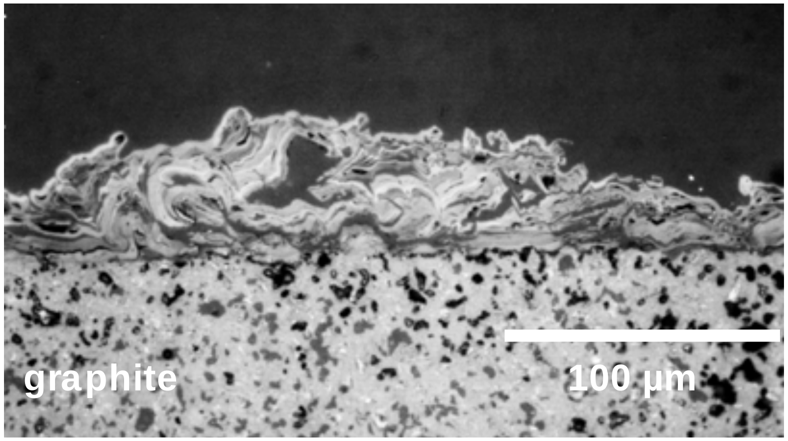
MERITS:

- Carbon more robust for machine commissioning.
- No melt layer loss.
- Q=10 supported by ITER physics base developed on carbon machines

RISKS:

- Present path does not extrapolate to ITER relevant tritium removal technology
- DT operations are stopped for safety reasons when T inventory reaches limit
- Removal methods are inadequate - long delays
- Public reaction terminates support for project.

Cross Section of TFTR co-deposit.



R&D to address risks:

- Given 17-year history above, ONLY meaningful step is intensive development of promising H-isotope removal techniques with goal of 1-day >90% D removal <u>in current tokamak</u>(s) with high performance plasmas next day and funding profile for completion within few years
- PLUS commitment to make changes in ITER design (pumps, divertor dome, tile gaps .... to make T removal feasible on ITER).
- PLUS massive R&D program on processing DTO, AND on deposition diagnostics.
- Without this, MD research on chemical sputtering of carbon for example, will be exciting science, but have NO value for ITER-DT since carbon will not be used.

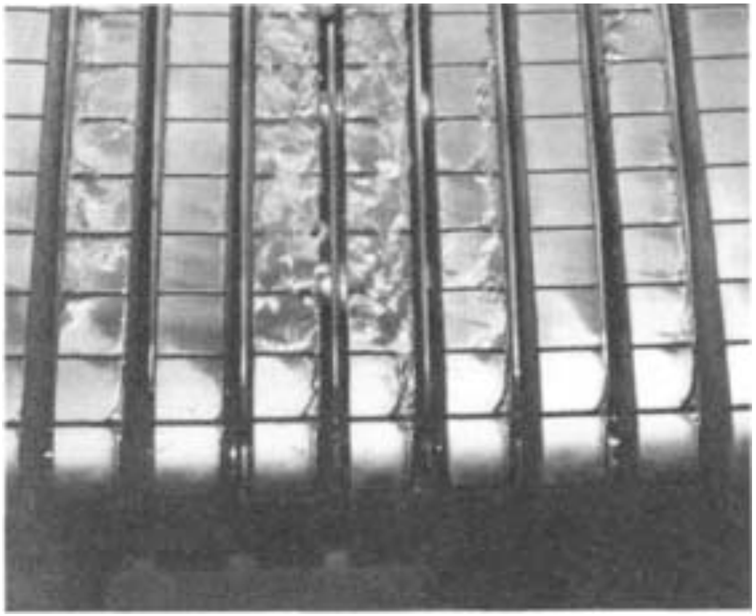
2. Metals in DT phase:

MERITS:

- Tritium retention not major issue.
- Carbon is not relevant to power fusion reactor anyway.

RISKS:

- Plasma ops different to ITER physics base
- Restricted operational space may not allow Q=10
- Detachment physics different with extrinsic impurities.
- Enhanced divertor erosion with extrinsic impurities.
- Melt layer loss prematurely terminates divertor lifetime.



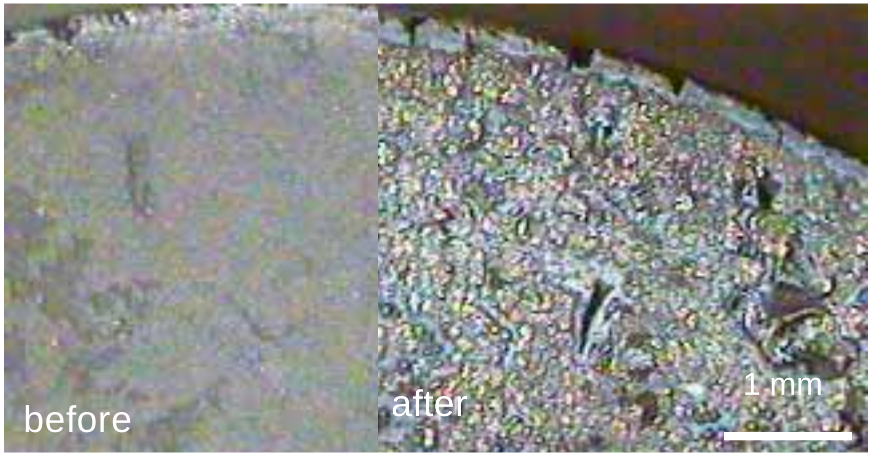
Be limiter damage observed on JET (Loarte PSI-16)

R&D to address risks:

- On going tokamak experience: Mo in C-mod, W in Asdex, Be/W on JET.
- Continue research solid state properties of tungsten
- Ongoing ELM & Disruption mitigation R&D
- Develop ITER performance scaling from all-metal machines
- Transport of tungsten melt layer loss after ELM/disruption melting.
- Benchmark disruption simulation codes (e.g. HEIGHTS) against experimental disruption simulators.
- Adapt diagnostics (e.g. CHERS) to carbon-free situation.
- Expand atomic physics base (emission lines, collision cross sections...) of highly ionized tungsten.
- Develop advanced high heat flux components

3.R&D to address risks of mixed materials:

- A long pulse tokamak makes its own plasma facing surface.
  Realistic R&D intrinsically difficult in laboratory.
- Be/W PFCs planned for JET in 2009
- Be/W results from PISCES (Doerner this workshop)



Be/C layer on JET tile formed 'beads' at 2100C before & after exposure to laser heat flux.

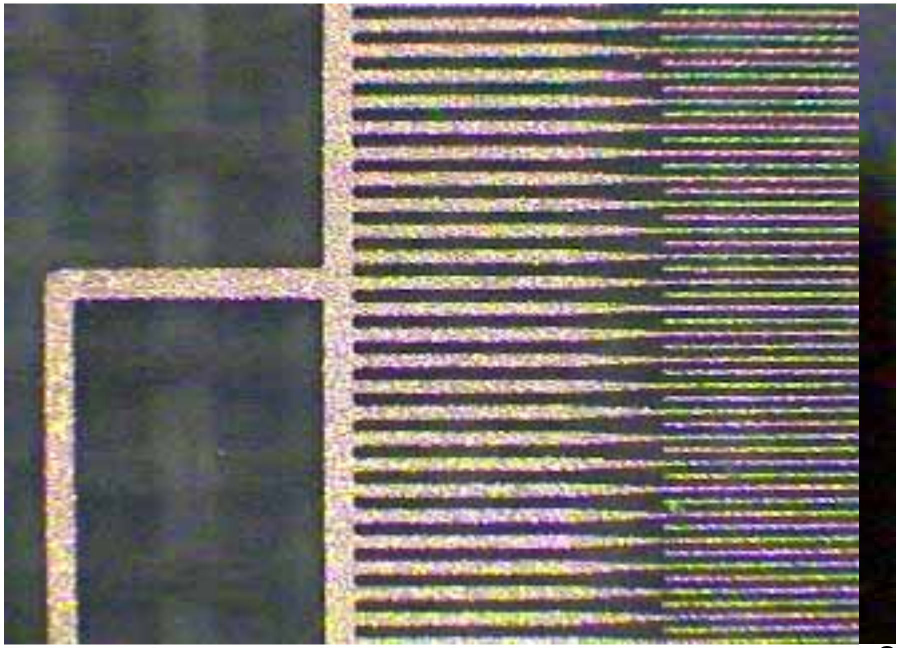
4. R&D to address risks of dust:

Novel electrostatic surface particulate detector:

A fine grid of interlocking traces spaced ≥25 μm is biased with 30-50 v DC.

Impinging dust produces a short circuit and current pulse that vaporises the dust and provides a signal.

(but funding almost non-existent)



Workshop goals: "The aim of the Workshop is to exchange opinions on the present depth of knowledge of surface properties for the main fusion-related materials…

Opinion:

- ITER has uncomfortable choice of:
  1. sticking with carbon (maybe not 100% impossible, but appears unrealistic for DT)
  2. switching to tungsten (but ITER physics base is mostly from carbon machines)
     - both options have serious risks
  3. Sitting on fence (current strategy) BUT
  - H and DD experience will not help much as:
    - Retention in hydrogen phase will be obscured by H_2O in tiles.
    - Deposition diagnostics to measure codeposits in ITER are NOT part of requirements.
    - R&D funding dissipated in directions that will inevitably be abandoned.
  - Carbon may be more robust for machine commissioning but a switch to W in DT phase entails serious delays to develop new plasma scenarios + potential complications with mixed materials from residual carbon.
  - Maybe biggest risk that 6 ITER parties will concentrate on their contractual obligations to produce major items of equipment and reduce R&D funding (as in FY2006 budget proposed by US administration).

Possible new directions for US:

Risk Management:

- Diversity of options always reduces risk only IF R&D budget sufficient to cover all options.
- Some solutions are closely coupled, probability of success depends on weakest link. Plan must be consistent, if solution needs both X and Y, no good researching only X
- Incentive needed for someone to take ownership of thorny interdiciplinary “not-my-problem” issues such as diagnosing retention in H/D phase, tritium removal, dust removal, or processing DTO exhaust.

- Quantify melt layer loss from tungsten in ITER (add JXB to plasma gun).
- Benchmark melt layer codes (HEIGHTS) against disruption simulators, extrapolate to ITER.
- Study Mo central accumulation and mitigation by central heating in C-mod (do not boronize)
- Study carbon-free detachment on C-mod with a view to its relevance to a W ITER.
- Study carbon-free CHERS diagnostic with a view to its relevance to a W ITER.
- Modeling SOL transport of Be and W
- Divertor erosion lifetime including neon sputtering (as necessary for detachment)
- US/EU collaboration with JET on Be wall initiative
- Initiate risk management modelling for prioritizing R&D for ITER.

Further info:

- PSI review: Nucl. Fusion 41 (2001)1967; T removal: Physica Scripta T111, 92-97, 2004.
- G. Federici and C. H. Skinner, "Tritium Inventory in the materials of the ITER plasma-facing components" 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).

Time is short….

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

[redacted]

Tritium removal by oxidation:

- Oxygen can remove codeposits by oxidation to H_2O, 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 one TFTR codeposit (also earlier talk on *Pressure Dependence of Oxidative Removal of Tokamak Codeposits*).
- Some experience on TFTR, JET, TEXTOR
  Reviewed by Davis in Physics Scripta T91, 33 (2001).

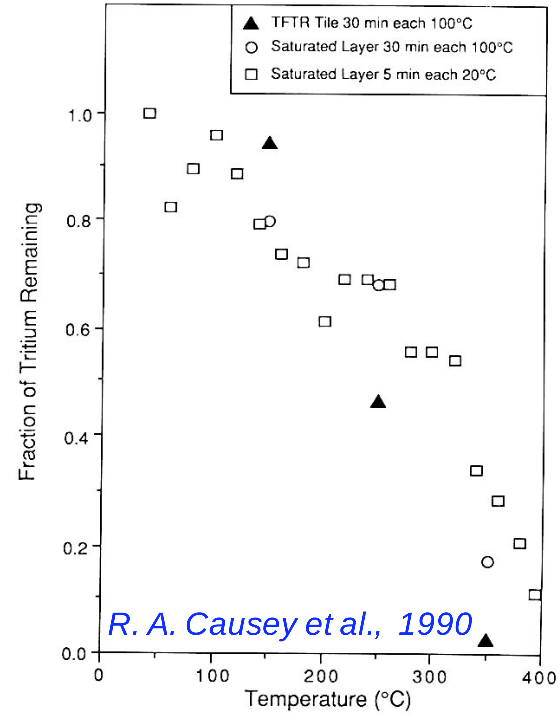
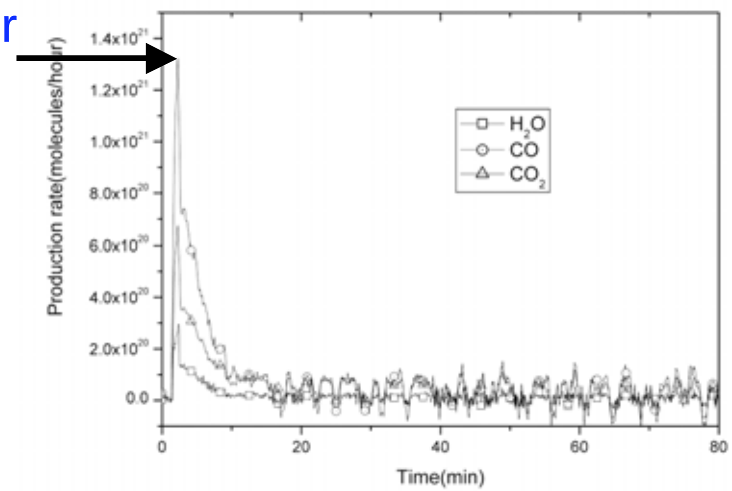
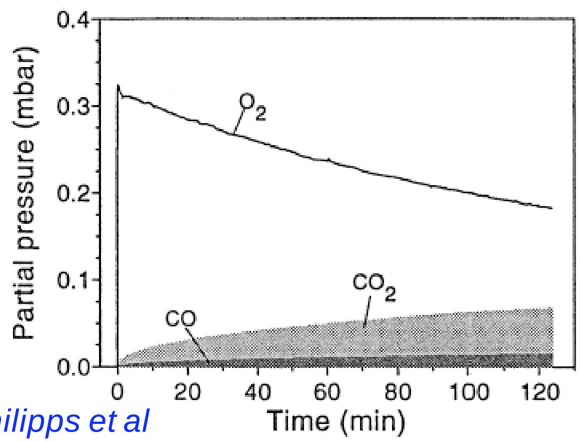


Fig. 2. Comparison of the in air hydrogen isotope retention in a codeposited film to that for a saturated layer.

Hu et al., HT-7 first results from O-bake 177 C, 0.24 torr

4 mg/hr





D release difficult to measure

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.)

Tritium removal by ablation using excimer lasers or flashlamps

Laser ablation demo in JET (Gibson et al. PSI-16)

Codeposit removed

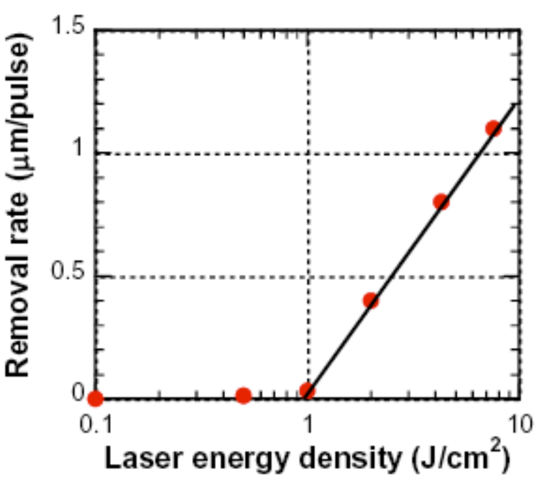
Art restoration by laser

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

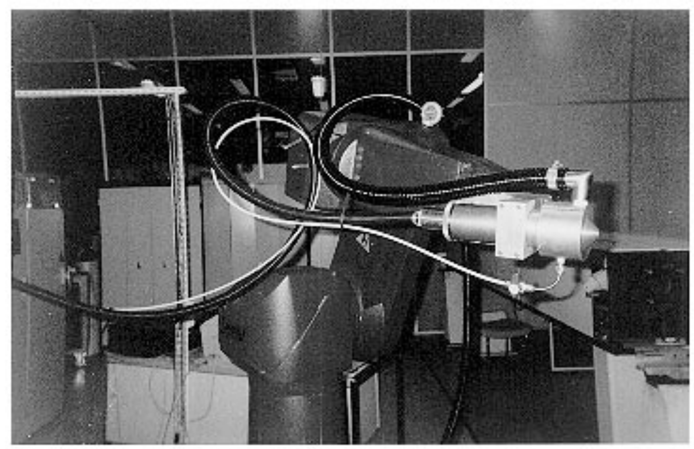
Excimer laser ablation:

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



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

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



Tritium removal by laser heating

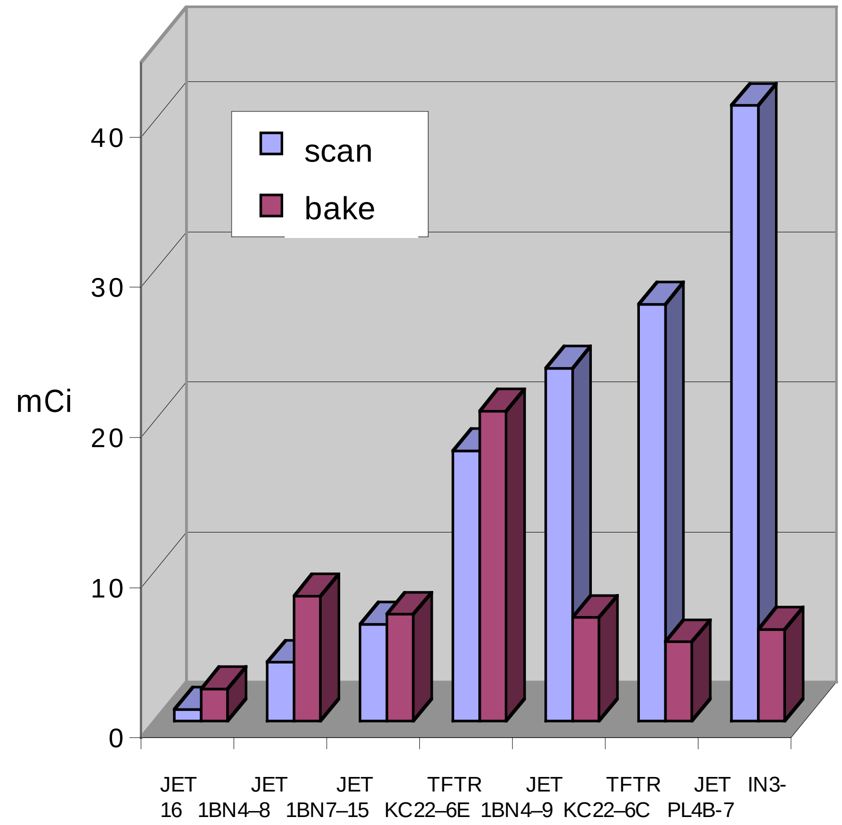
- Heating is proven method to release tritium but heating ITER vacuum vessel to required temperatures (~350 C) is impractical.

But

- – most T is codeposited on the surface
- – only surface needs to be heated.
- – Continuous wave lasers can provide the required heating without ablation.
- – Technique has been validated in extensive lab experiments on JET and TFTR tile samples

- Release fraction achieved up to 87%
- Detritiation efficiency highest in regions of heavy deposition.
- Extrapolates well to ITER

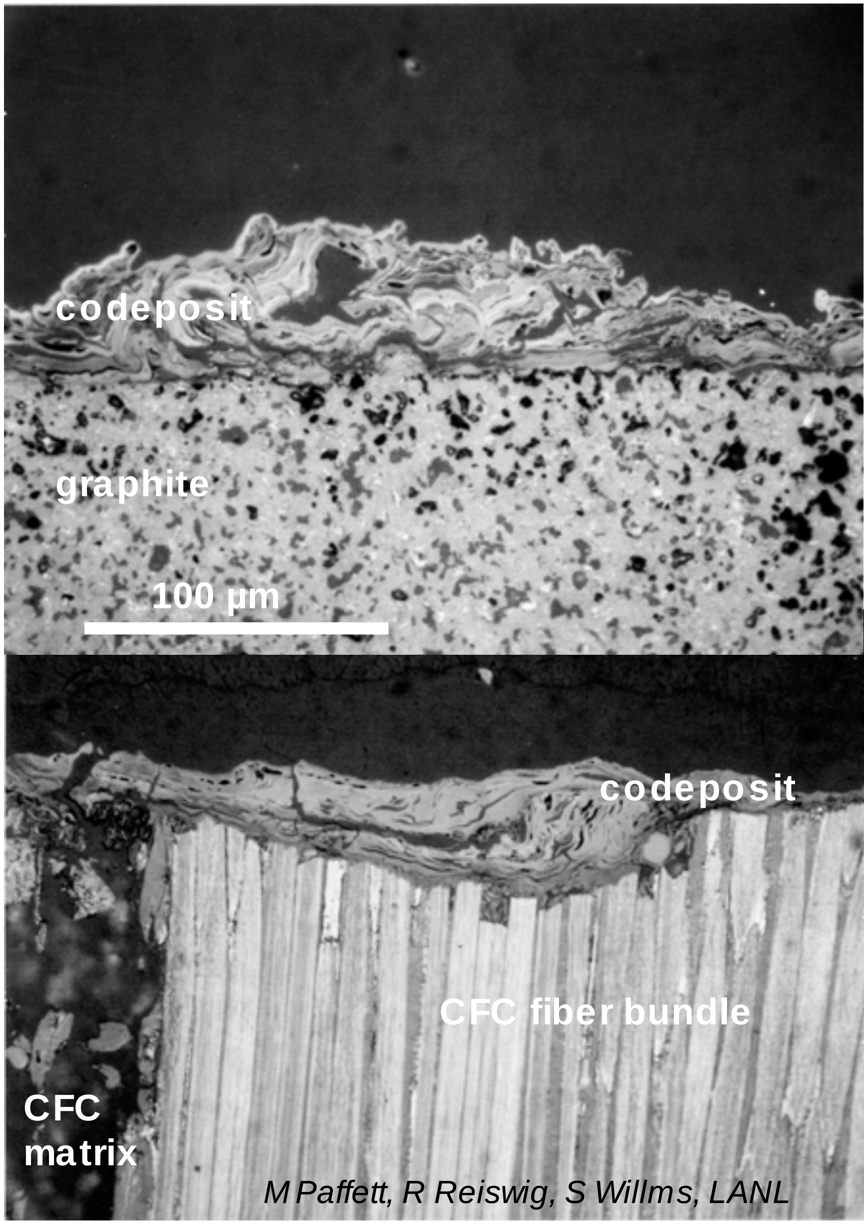
PPPL results 2001



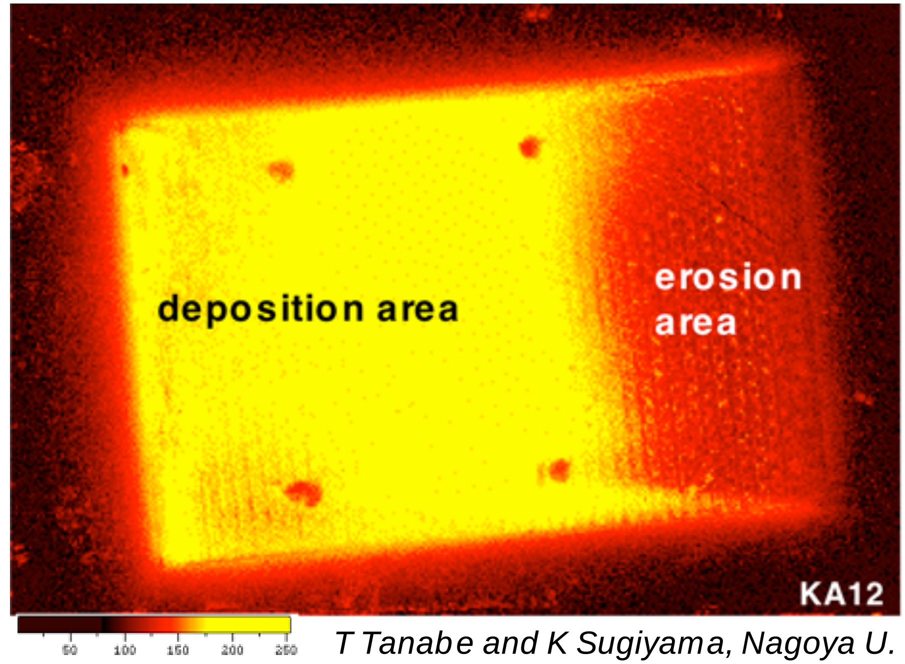
Major part of co-deposited tritium released by scanning laser.

Tritium location

Cross Section of TFTR co-deposit.

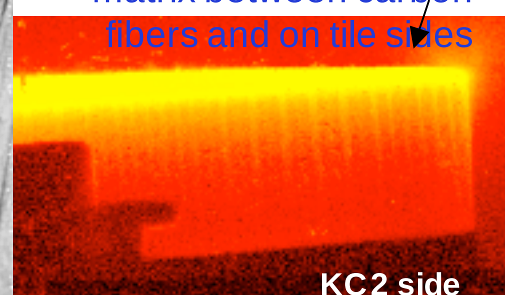


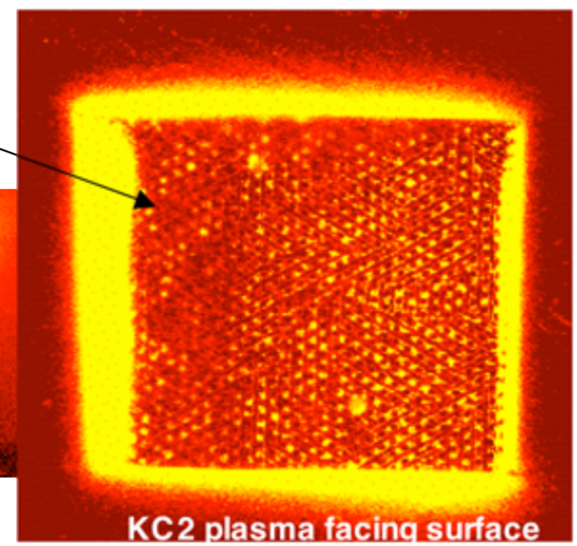
Imaging plate: tritium and ^{60}Co on TFTR CFC tile



T Tanabe and K Sugiyama, Nagoya U.

Tile from erosion region.
tritium deposition in matrix between carbon fibers and on tile sides

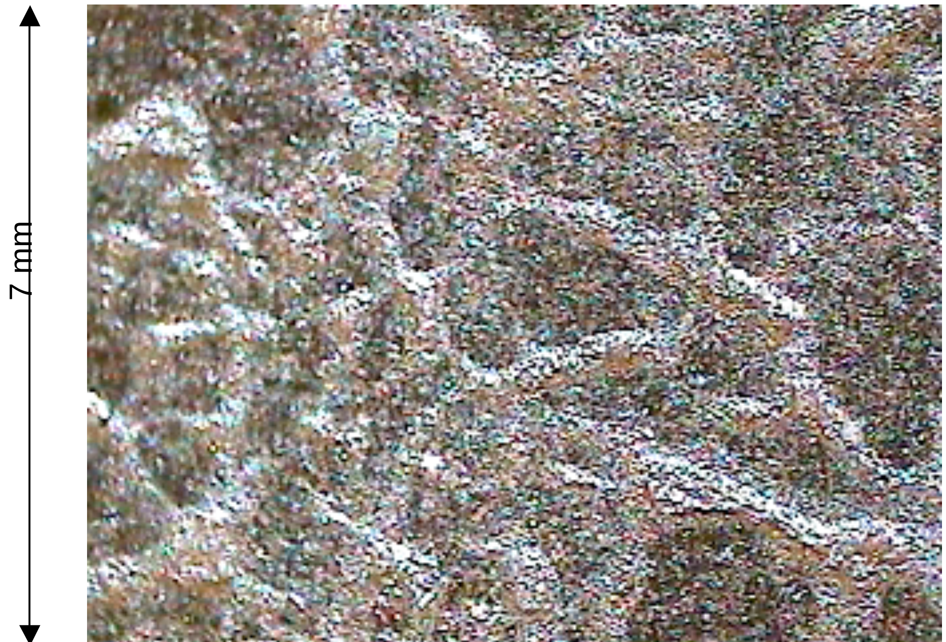




15% D retained in gaps

Thermal response of ion damaged tungsten

W sample implanted with 1e21 D @ 200 eV (courtesy of J. Roth).



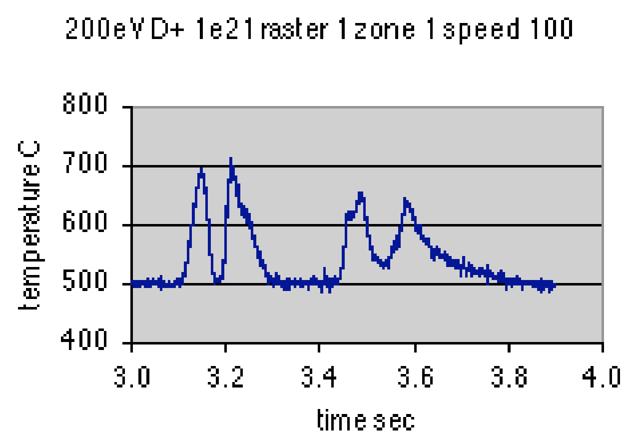
200evD+1e21x1x4_e.BMP

low power x4
optical microscope

Use scanning Nd laser to transiently heat (laser has smooth focal spot)

During laser scan

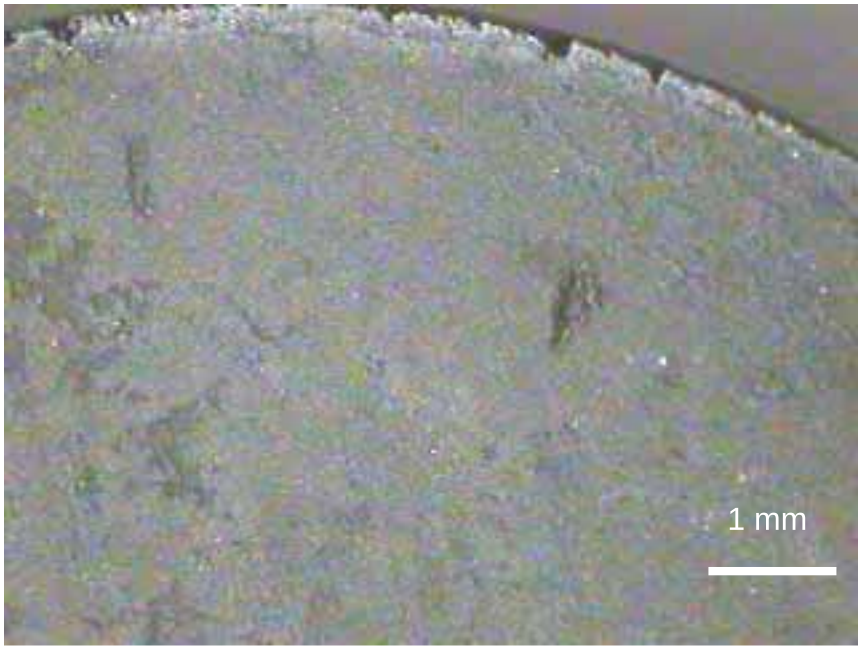
W_03still.jpeg



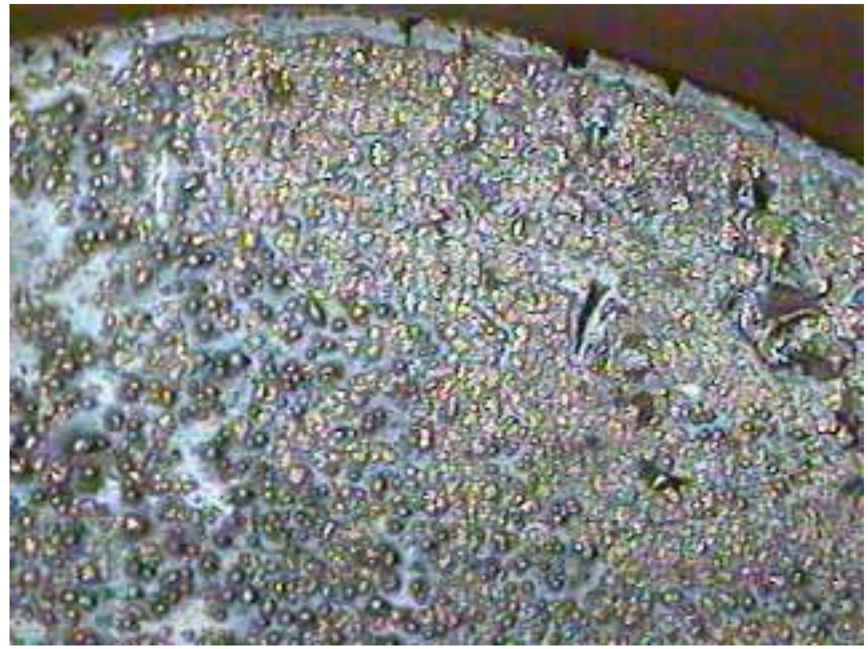
Ion implantation features change thermal response.
- PFC surfaces manufactured by tokamak and may not have same properties as factory-manufactured material.

Be/C layer appears to form 'beads' in response to heat flux.

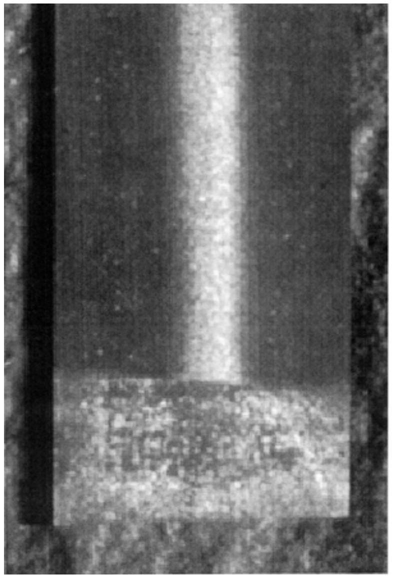
JET tile IN3–16 (vertical tile, inner divertor) before Nd laser scan



After laser scans



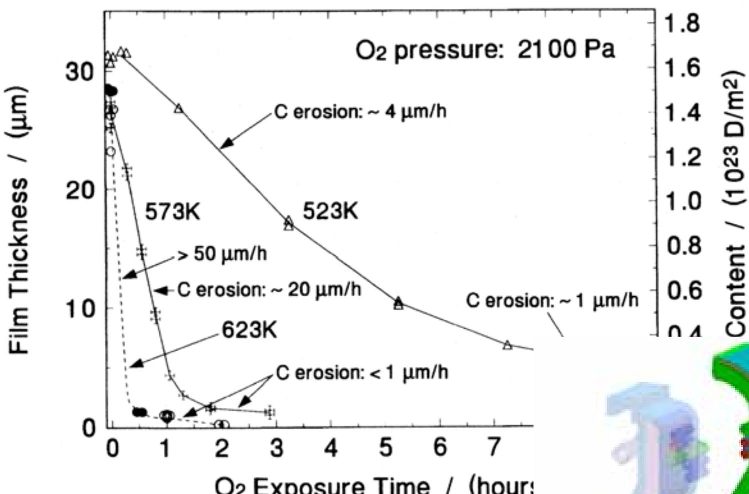
'Globules' of Be formed after laser heating.
21 temperature excursions above 1000 C, peak temperature 2,100 C.
Subsequent scans at same laser power and speed resulted in lower temperatures (1,601 C then 1,314 C) as layer became more thermally conducting similar to the manufactured material.
Other codeposits (without Be) did not show this major temperature decrease.

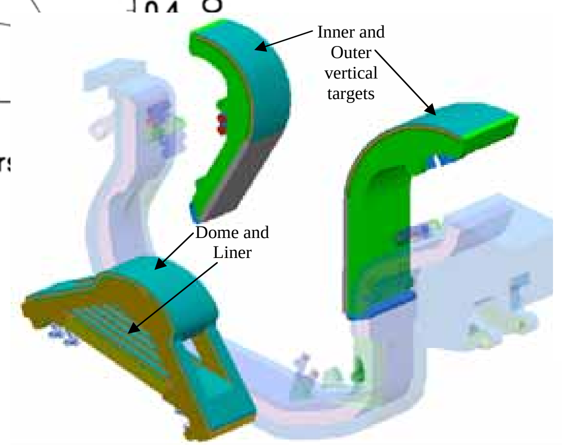


Flashlamp ablation:

CFC tile coated in 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-vacuuo using 10 pulses from the flashlamp.

G. F. Counsell & C. H. Wu ,8th Carbon Workshop, Physica Scripta T91 (2001) 70.





Haasz & Davis 1998