Tritium Retention Issues in the ITER-FEAT Tokamak

presented by
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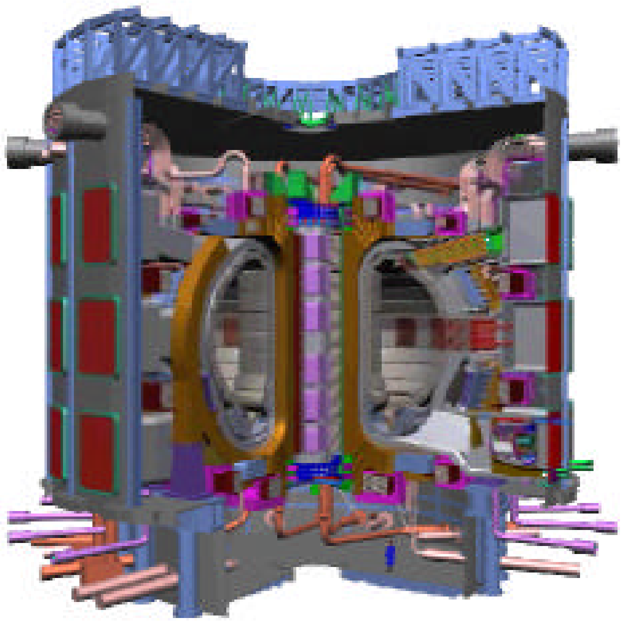
Outline:



A. Design, materials and operating conditions of PFCs in ITER-FEAT
B. Erosion and tritium retention in the ITER tokamak
  - Mechanisms
  - Database
  - Model predictions
C. Further R&D needs
D. Conclusions

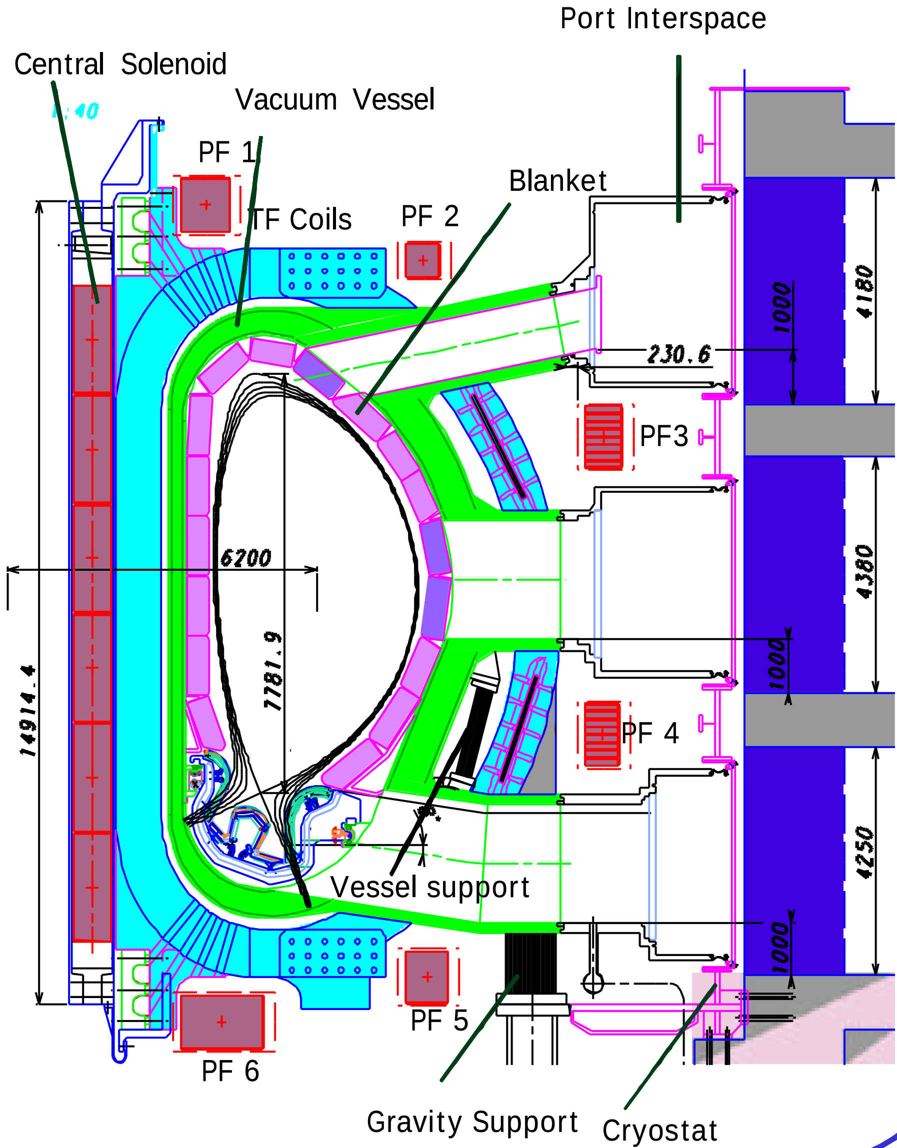
Tritium Town Meeting
Hilton Garden Inn, Livermore CA
6-7 March 2001

ITER-FEAT Machine Parameters

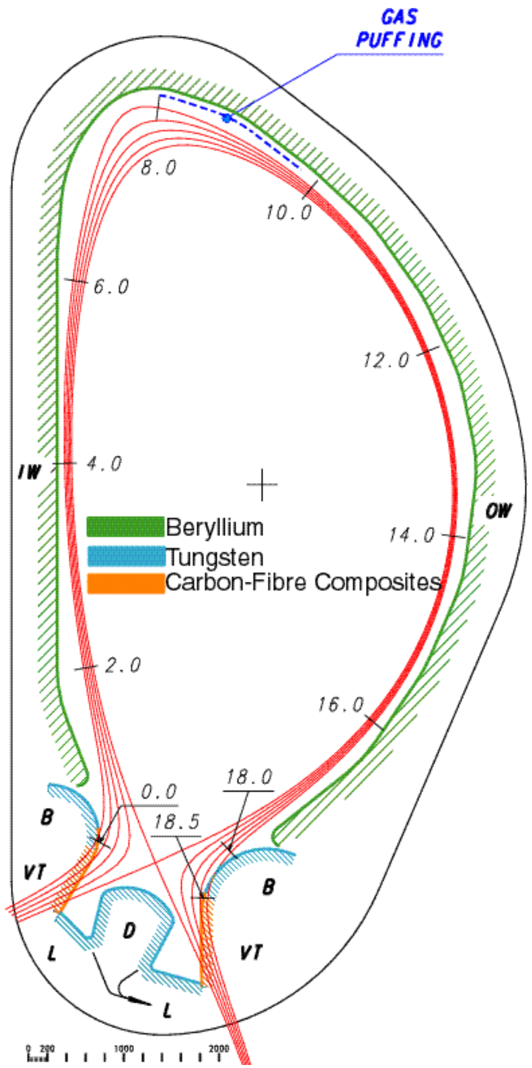
Parameters of ITER-FEAT

Parameter	Nominal	Max
Major radius, R	6.2 m	<=
Minor radius, a	2.0 m	<=
Plasma current, I_p	15.0 MA	17.4
Additional heating & CD power	73 MW	100
Fusion power	500 MW	700
Toroidal field at major radius, Bo	5.3 T	<=
Elongation at 95% flux, κ_{95}, κ_X	1.7, 1.85	<=
Triangularity at 95% flux, δ_{95}, δ_X	0.33/0.49	<=
Plasma volume	837 m^3	<=
Plasma surface	678 m^3	<=
MHD safety factor at 95% flux, q_{95}	3	2.6
Average neutron wall load	0.57 MW /m^2	0.80

Inductive pulse length ~ 400 sec



A. Design and Materials of PFCs in ITER-FEAT



•The material for the first-wall is **Beryllium**. The strike-point regions of the divertor are of **CFC** and the baffle regions of the target, the dome and the transparent liner in the private flux region below the dome are all in **tungsten**.

	PFCs	Armour	Area (m^2)	P. Fluxes ($DT/m^2 \cdot s$)	Heat F. (MW/m^2)
FW	First-Wall	Be/10 mm	680	10^{19}-10^{20}	<0.2
	Start-up limiter	Be/8 mm	10	1.10^{22}	10 st.up
Divertor	Baffle	W/10 mm	50	10^{20}-10^{22}	3
	Dome	W/10 mm	30	10^{20}-10^{22}	3
	Vertical target	CFC/20	55	<10^{24}	10
	Liner	W/10 mm	60	<1.10^{23}	<1

A. Choice of Plasma-Facing Materials

Justifications for selection

- **Be: FW (~700 m^2), SU limiters (~10 m^2):**
  - good oxygen gettering;
  - low Z (low plasma contamination, low radiation losses);
  - absence of chemical sputtering;
  - low tritium inventory;
  - potential for plasma-spray repair.
- **W: Baffle (60 m^2), dome (35 m^2), liner (50 m^2)**
  - low physical sputtering yield and high threshold energy for physical sputtering.
- **CFC:** target near-strike points (**~50 m^2)**
  - withstand disruptions without melting.

Issues

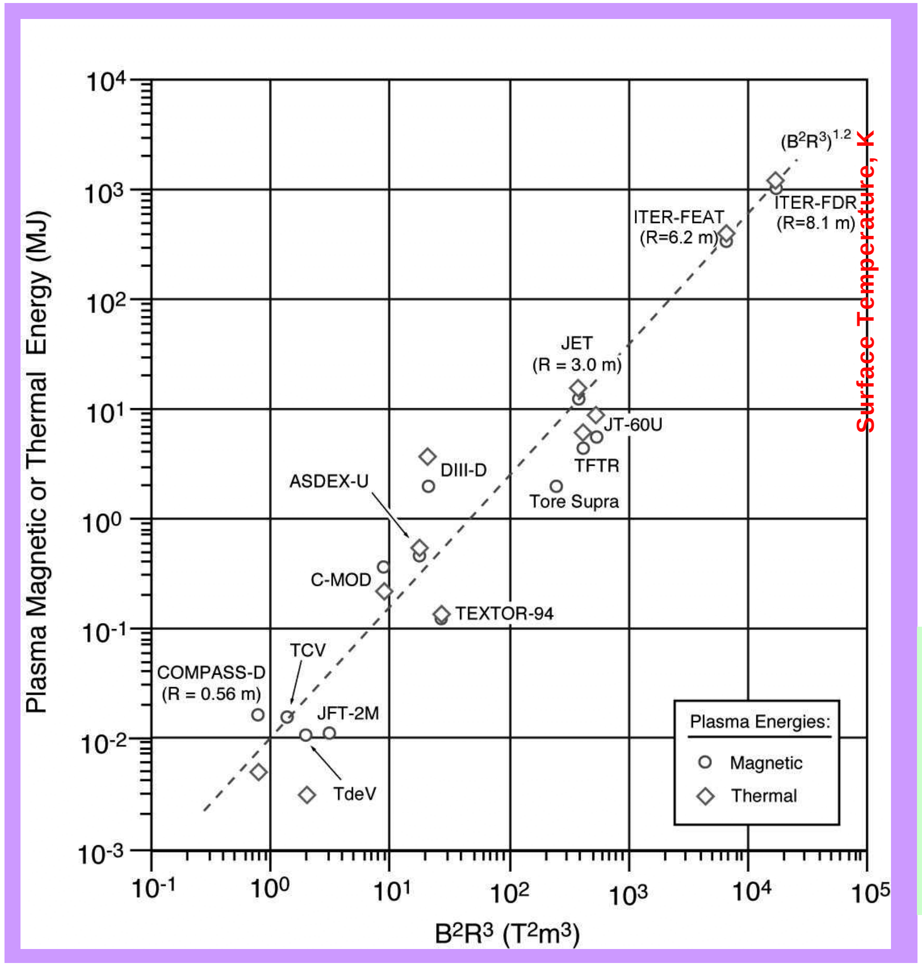
- **Be**
  - Be dust production (10-20 kg limit) -> hydrogen production -> explosion.
- **W**
  - W dust production (100 - 300kg);
  - disruption erosion - melt-layer loss;
  - plasma compatibility.
- **CFC**
  - T-co-deposition (350 g limit);
  - no adequate cleaning method available;
  - C dust production (~100kg limit).

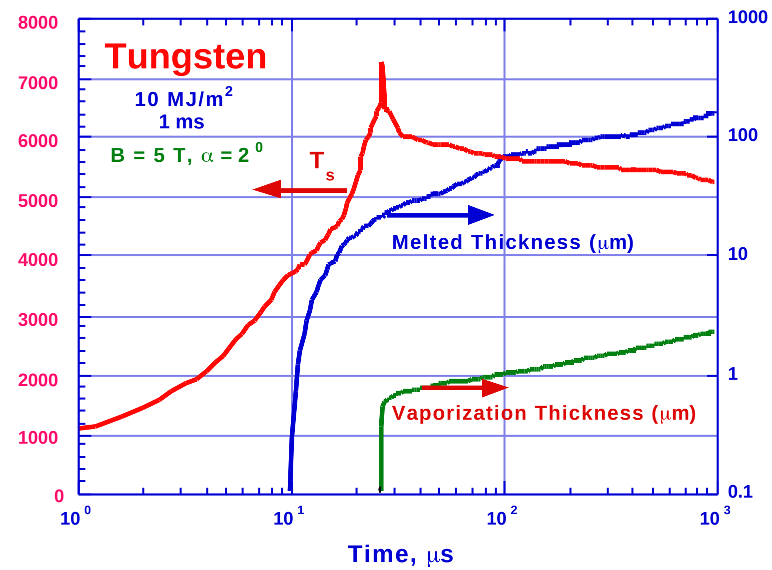
=> Mix-materials effects are expected to play an important role.

- The current scenario in ITER is initially to install C as armour on the targets, and to maintain the option to switch to more reactor relevant all-W armoured targets prior to D-T operation.
- The decision to make this change will depend on the progress made in controlling the plasma, in particular, on the frequency and severity of disruptions, and on the other hand, the success achieved in mitigating the effects of T co-deposition.

A. Disruptions and its Consequences in ITER

The use of alternatives to carbon poses problems for the extreme pulsed heat loads of disruptions and ELMs because they melt rather than subliming.





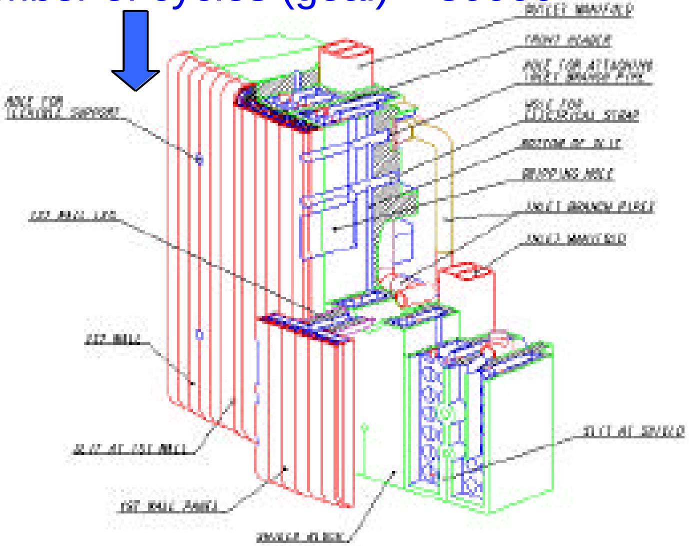
- The melt-layer thickness of W is 10-100 times higher than vaporisation losses.
- Therefore, the dynamic response of a ML layer during the course of disruptions is most important.

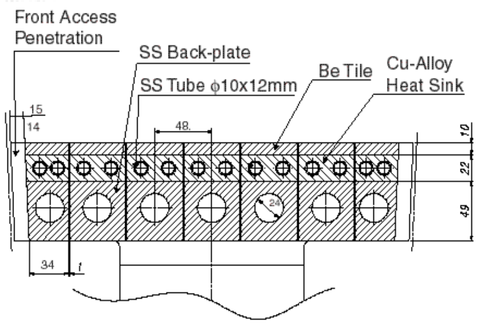
A. Beryllium is the armour for the ITER First-Wall and Limiter

Requirements: First Wall

Max heat flux – 0.5 MW/m^2

Number of cycles (goal) – 30000

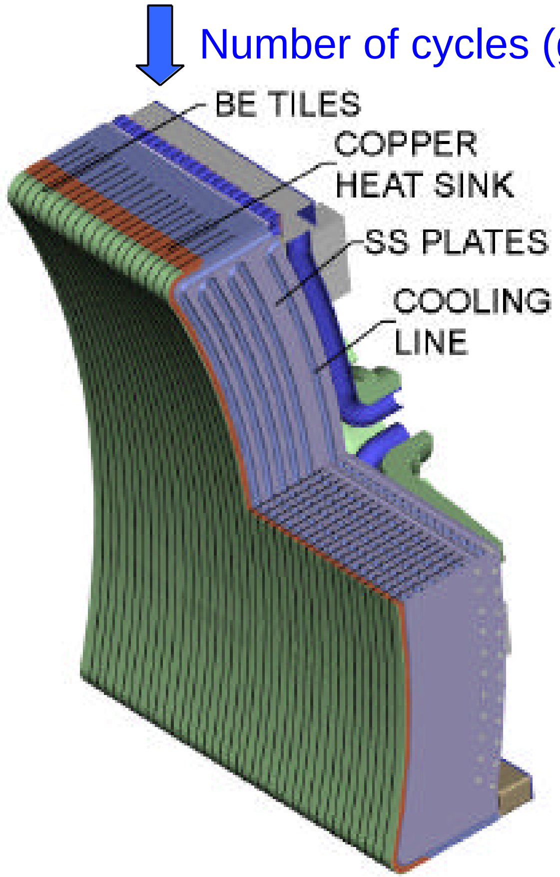


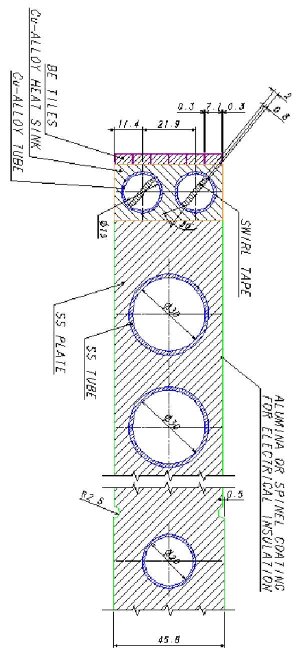


Limiter

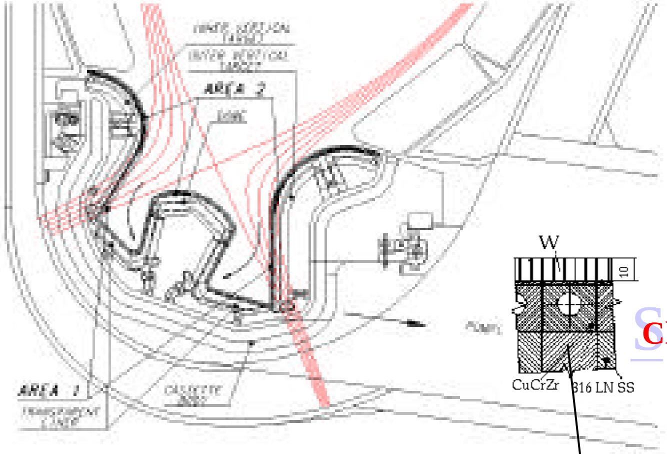
Max heat flux – 8 MW/m^2

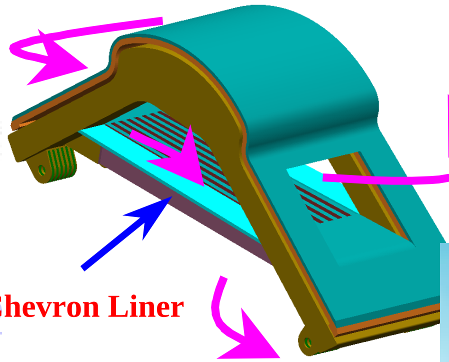
Number of cycles (goal) – 10000

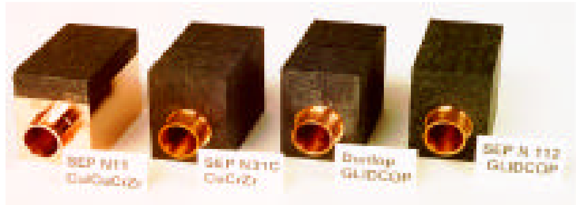


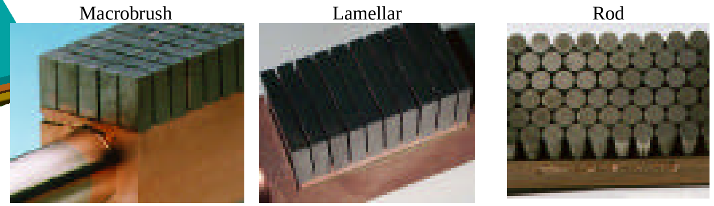


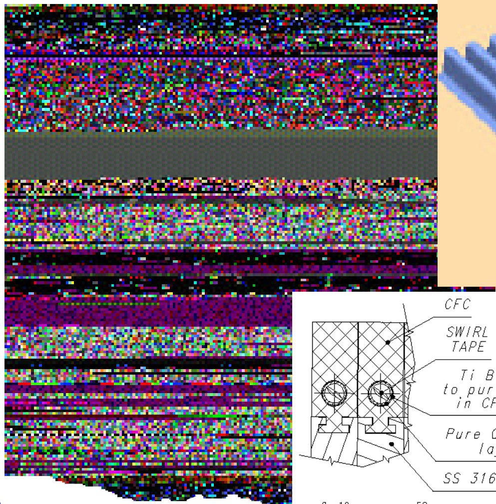
A. The ITER Divertor

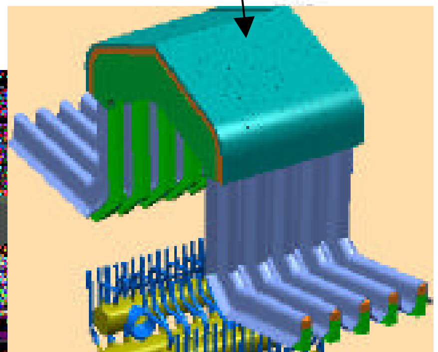


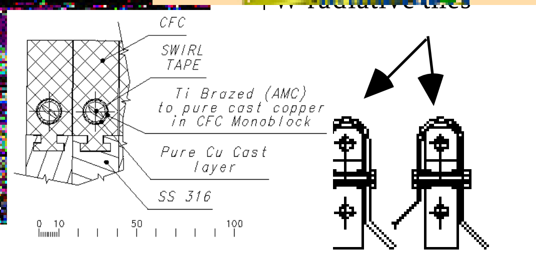


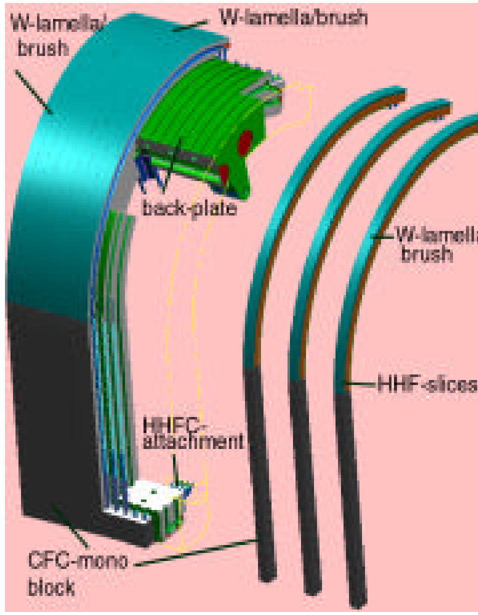


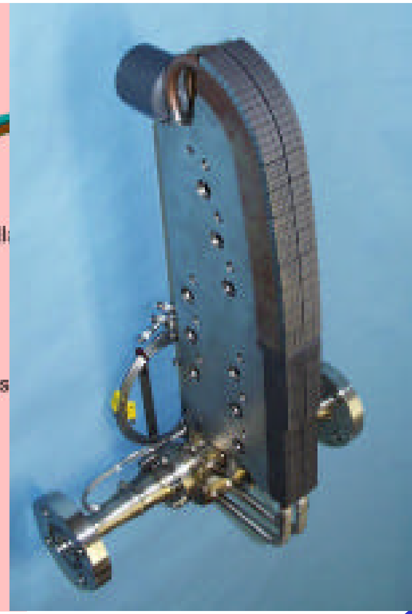










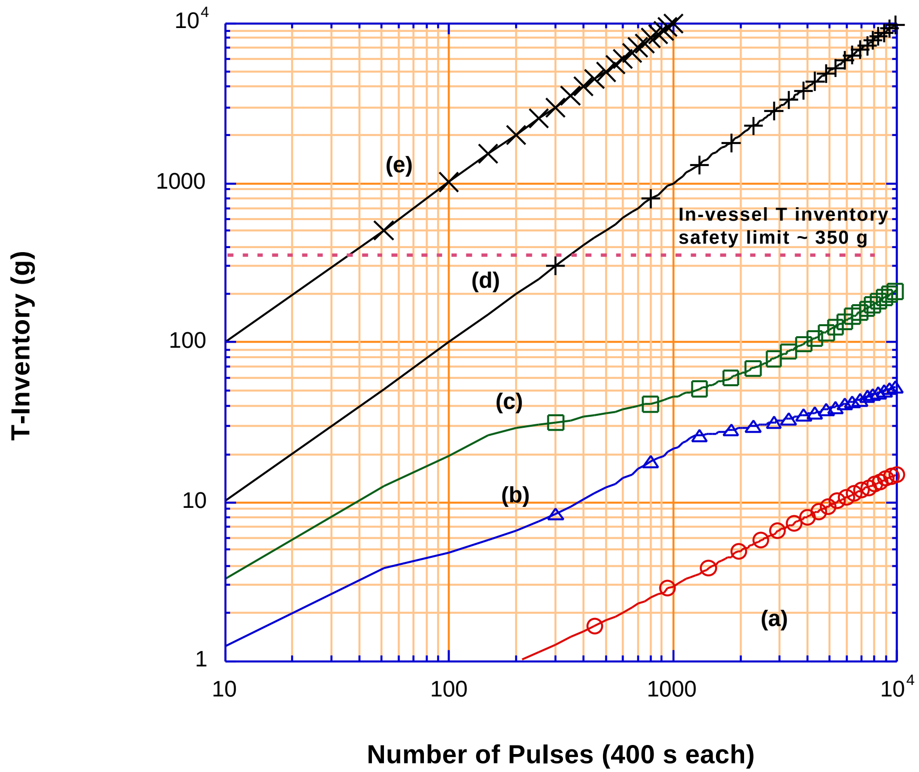


B. Tritium Retention Mechanisms in ITER

- **Co-deposition** (mainly in the divertor)
  - of carbon (mainly eroded near the strike-points) and T (D) in regions, which are shaded from ion flux, e.g., gaps, ducts, etc.;
  - possibly with beryllium eroded from the wall and transported on the regions in-light-of-sight of the plasma.

- **Implantation**
  - In the Be wall (relatively small contribution).

- **N-induced transmutation reactions**
  - In the Be wall (small contribution).



Implantation

(a) without n-effect;

(b) with 0.1% traps;

(c) with 1% traps;

Co-deposition:

(d) 1 g/pulse

(e) 10 g/pulse

B. Surface Damage in Beryllium

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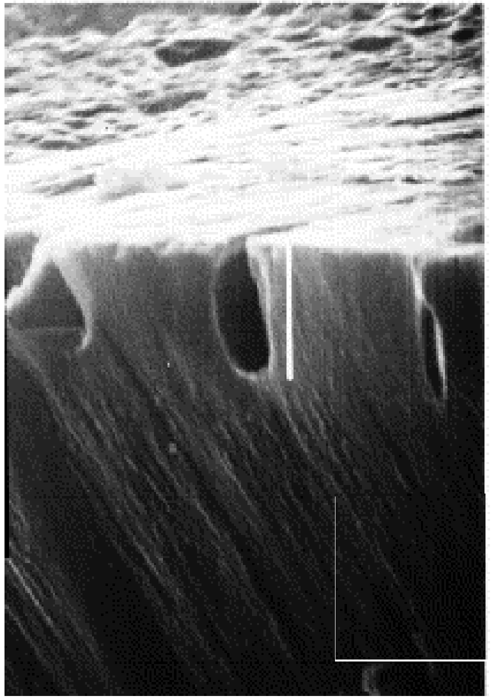
is seen experimentally

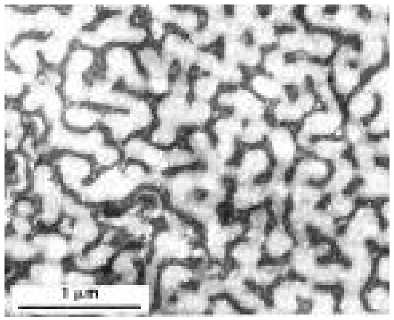


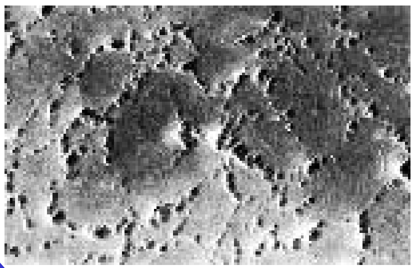
INEEL Be at 470 C
Ion flux 6×10^{19} D/m^{2}s
Ion fluence 2.3×10^{24} D/m^2

Implantation depth 16 nm

Damaged to 1 μm

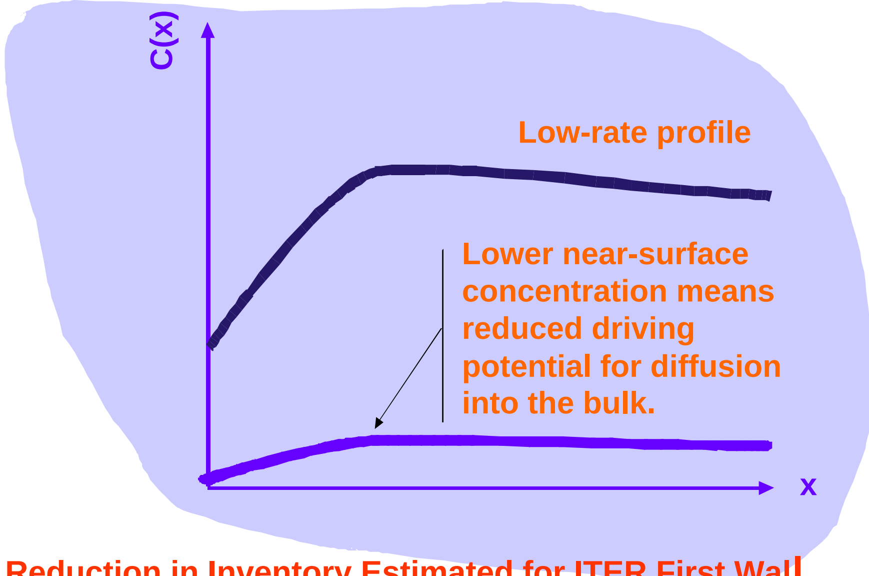
Subsurface labyrinths generated by 10-keV D in Russian Be at 425°C, ion fluence 2×10^{21} D/m^2



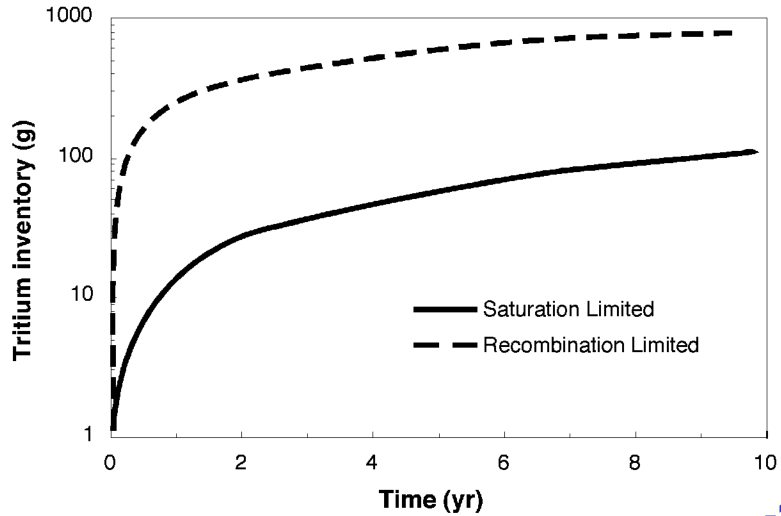


Russian Be at 435°C
ion fluence 1.2×10^{24} D/m^2
showed similar damage

Implantation, Diffusion, and Re-emission

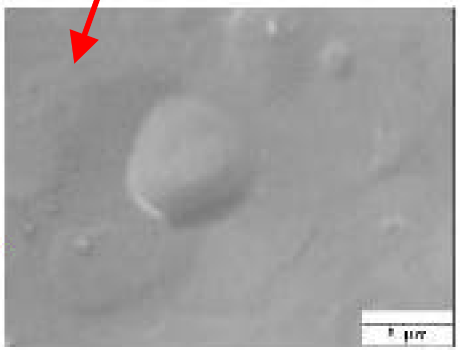


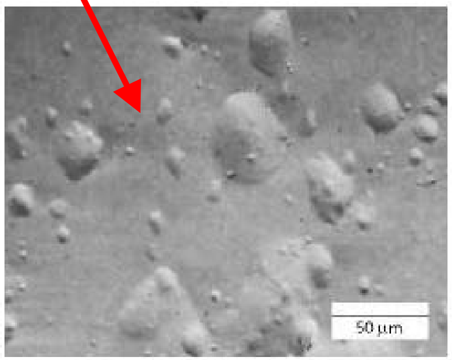
Reduction in Inventory Estimated for ITER First Wall



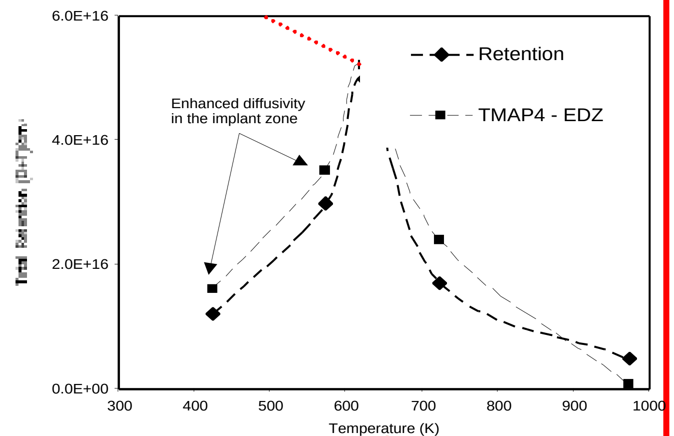
B. Behavior of W Exposed to High Fluxes of Low Energy H-Isotopes

- Some mechanism is causing an enhancement of the re-emission of atoms from the surface.
  - Blistering on the surface is observed, pressure in the bubbles caused blisters to rupture releasing trapped gas.
  - Number of blisters and their size increases with increasing D ion fluence.
    - 7×10^{24} D + /m^2 (@100 eV), no blisters observed
    - 7×10^{25} D + /m^2 (@100 eV)
    - 3×10^{26} D + /m^2 (@100 eV)
    - No blister formation observed after low-energy (100 eV) helium plasma exposure of W samples





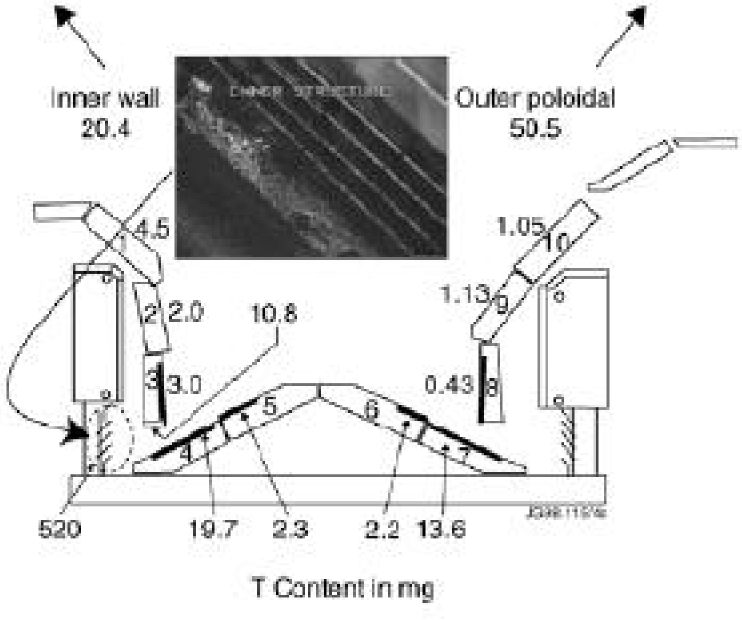
- Venhaus and Causey proposed a new model for low temperature data (423 and 573 K), which consists of adjusting diffusivity in implant zone to enhance release.



Courtesy of Causey/ Venhaus (SNL)

=> Calculated tritium inventory in the ITER divertor is less than ~10 g after ~3000-4000 pulses.

B. Co-deposition Occurs in Areas Shaded from the Plasma



Courtesy of J.P. Coad (JET)

Main chamber

JET

- Main chamber wall: erosion by CX
  M. Mayer ***et al.***, J. Nucl. Mater 241-243 (1997) 469
- Guard limiters: Erosion at plasma face
  Deposition at side faces

Divertor

- Deposition at inner divertor leg
- Deposition at louvres:
  Hydrocarbon radicals?
- Sub divertor area? Deposits in ASDEX-U

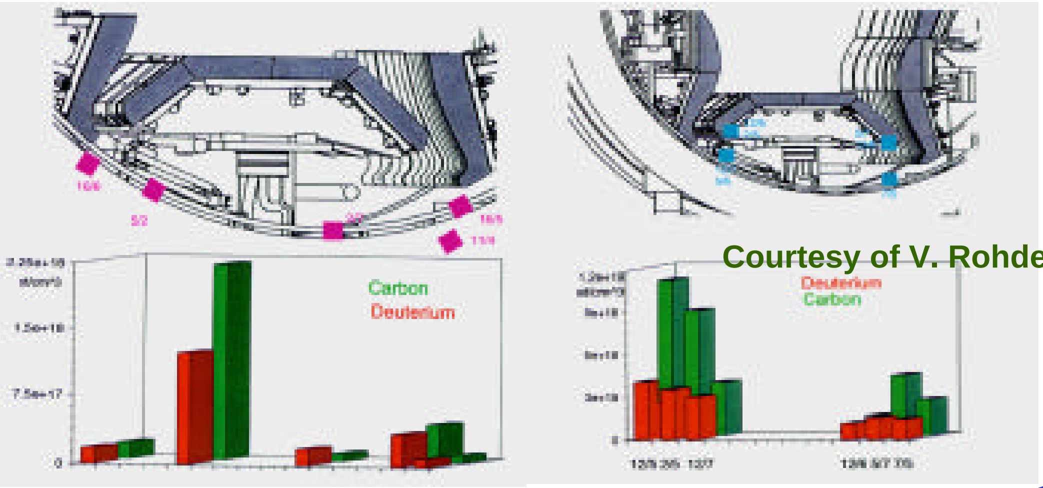
ASDEX-Upgrade

Inner/outer divertor: 3:1

Two kind of layers:

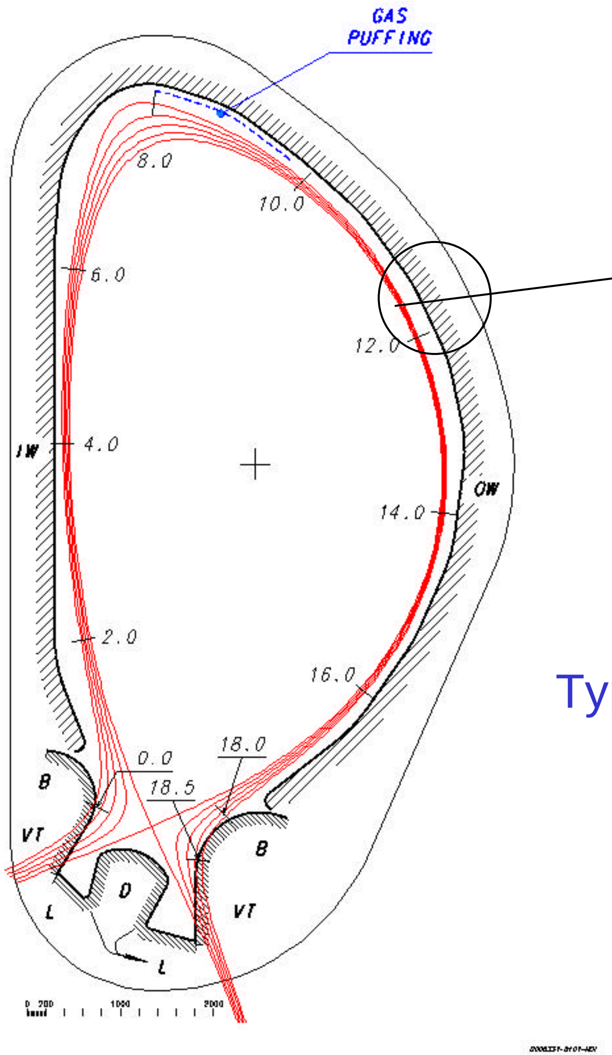
1) brownish; 2) transparentv

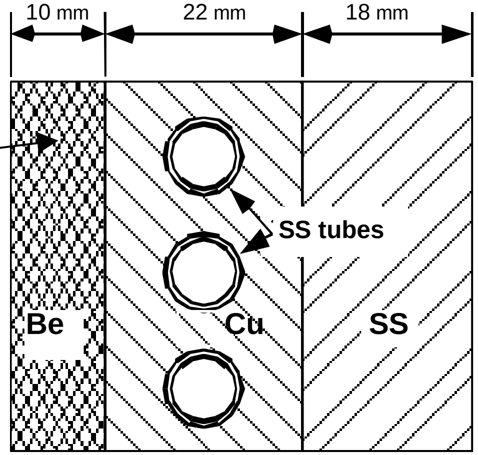
Mechanisms of formation still unclear:
ion bombardment, photo-ionisation, etc.



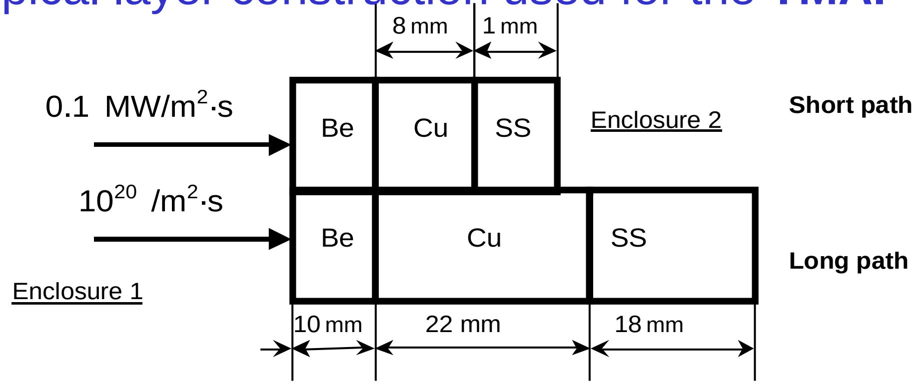
Courtesy of V. Rohde (IPP)

B. Tritium Tritium Implantation Analyses

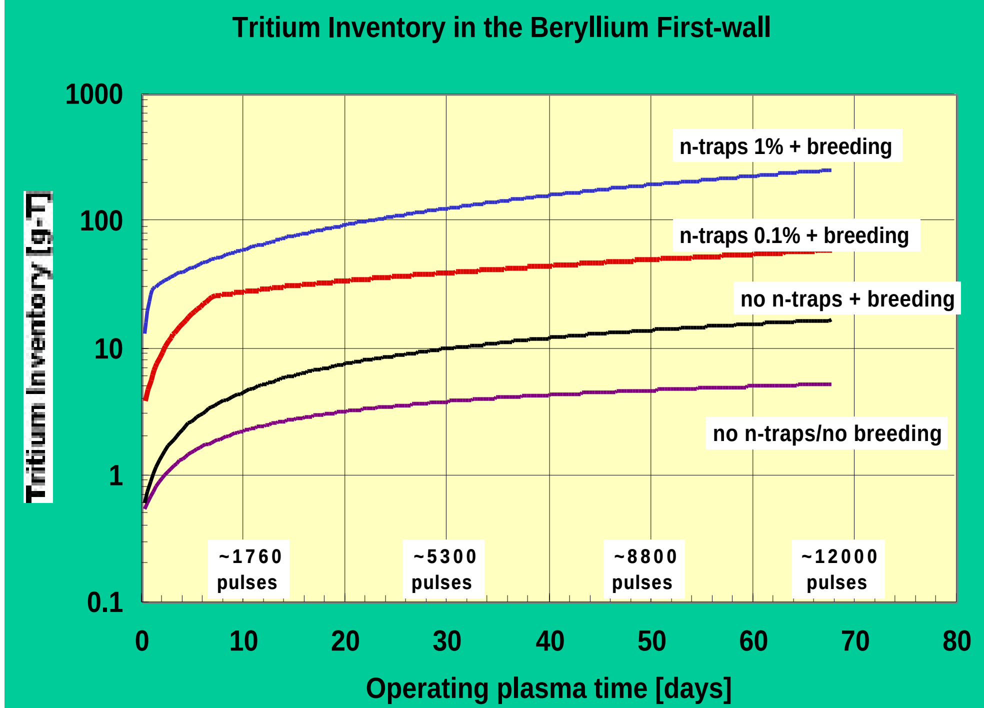


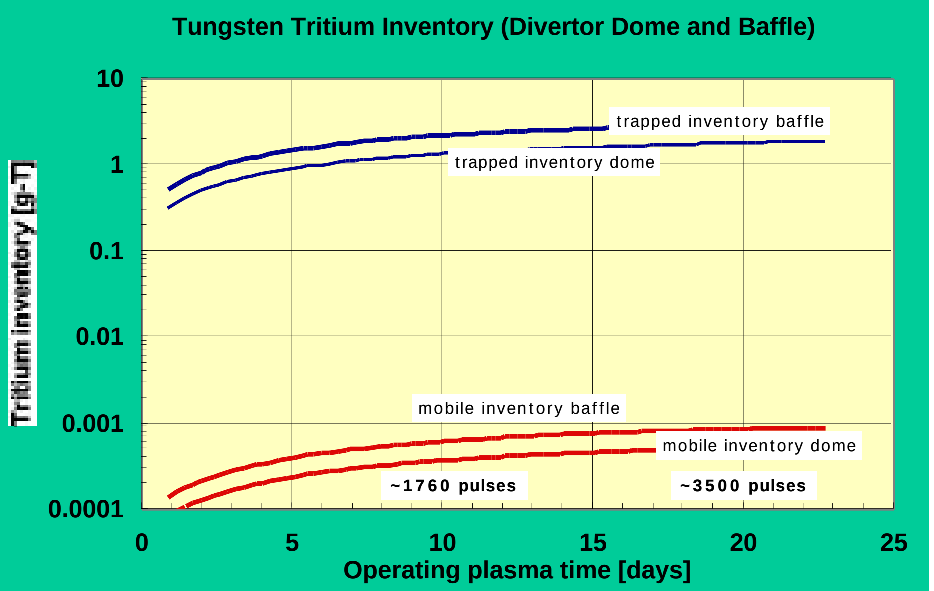


Typical layer construction used for the **TMAP4 model**.



B. Summary of Results





20 grams without including n-induced traps

including effects of n-induced traps and breeding the total inventory after 12000 pulses is between 90-250 g.

No perm. breakthrough

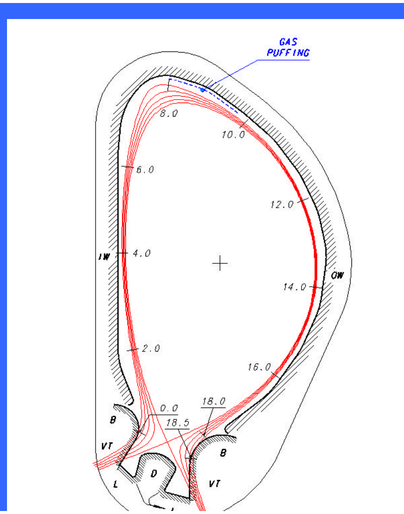
Tritium inventory is negligible < 10 g after ~3000-4000 pulses.

No permeation breakthrough.

B. Erosion of the ITER Be First-Wall

Assumptions:

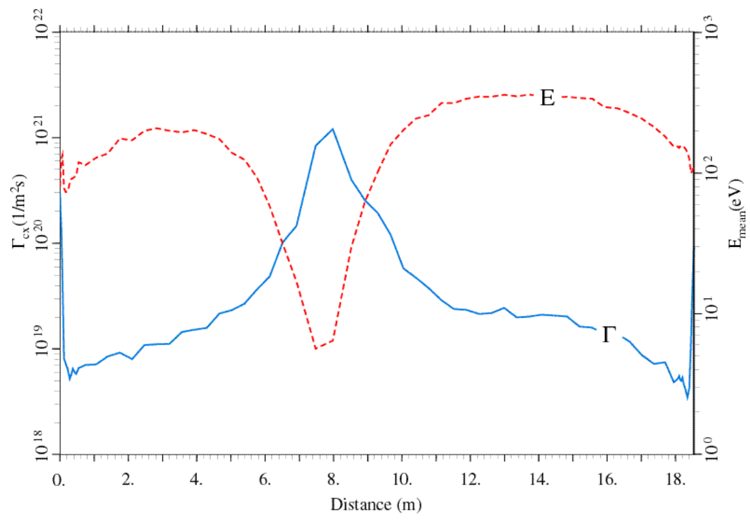
- CX fluxes and spectra calculated with EIRENE;
- ion fluxes from B2-EIRENE; values at the grid edge scaled to the wall using the e-folding length of 3 cm;
- ion species: D, T, He^{+}, He^{2+}, C^{+}- C^{6+};
- shifted Maxwellian distribution function;
- physical sputtering only;
- no re-deposition of sputtered material;
- sputtering yields Y(E) form Bohdansky formula taking 2 as the angle enhancement factor.



$$\Gamma_{CX} = \int \Gamma(E)dE$$

$$E_{mean} = \frac{\int \Gamma(E)E\,dE}{\int \Gamma(E)dE}$$

$$\Gamma_{SP} = \int \Gamma(E)Y(E)dE$$

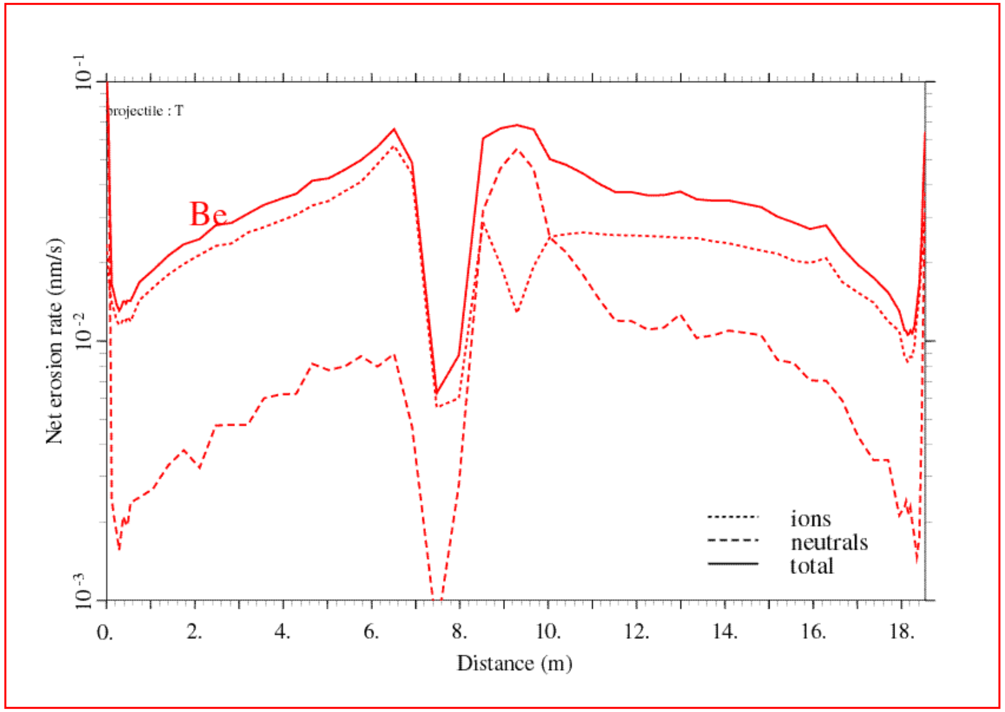


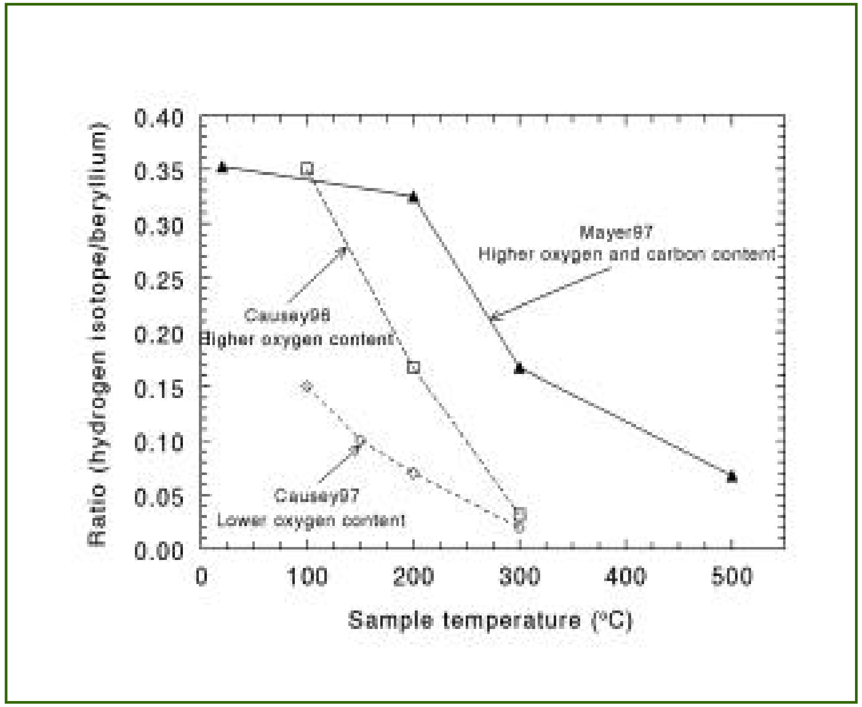
Results (see details next slides)

Peak erosion rate is estimated to be ~ 0.1 nm/s (0.4 µm/pulse).

Worst-case co-deposition from this Be transported to the cold surfaces of divertor is <0.5 g/pulse.

B. Erosion of ITER Be First-Wall

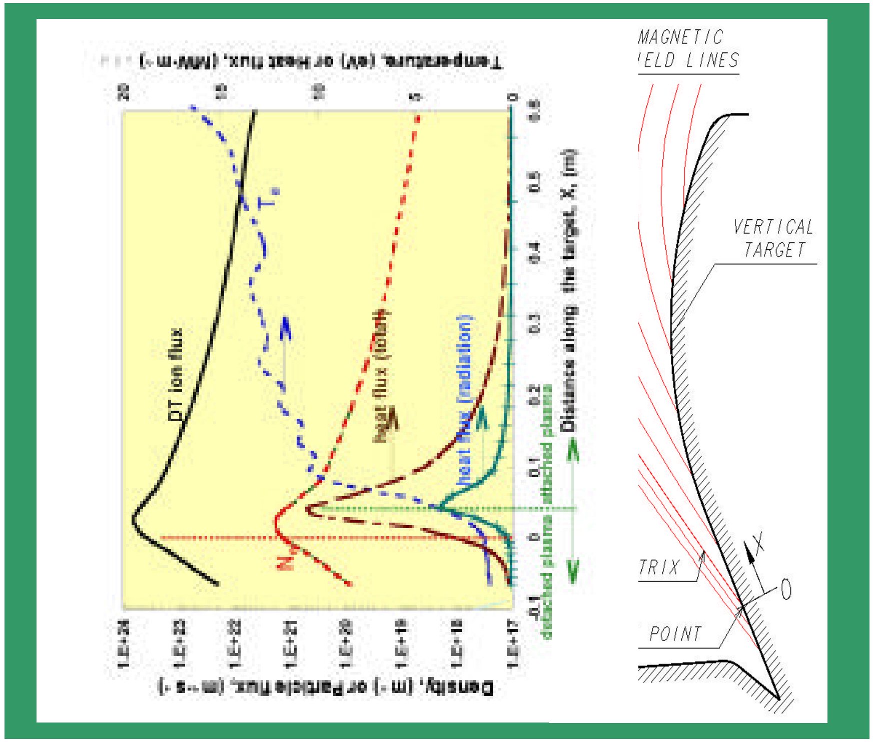




	$\Gamma(s^{-1})$	$\delta(Be\cdot s^{-1})$	δ_{Be} $(g\cdot s^{-1})$ /(g/pulse)	δ_T (g-T/pulse) T@200˚T T/Be=0.1	δ_T (g-T/pulse) T@300˚C T/Be=0.02
$CX_{(D+T)}$	$6.3\cdot10^{22}$	$9.1\cdot10^{20}$	0.014/5.4	0.2	0.04
$Ions_{(D+T)}$	$5.3\cdot10^{22}$	$1.7\cdot10^{21}$	0.025/10	0.3	0.07
$Ions_{(He+C)}$	$1\cdot10^{21}$	$1\cdot10^{20}$	0.002/0.8	0.02	0.004
Total		**$2.7\cdot10^{21}$**	**0.04/16.2**	**~0.5**	**~0.1**

B. Erosion/co-deposition in the ITER Divertor

- The reference ITER divertor plasma is "semi-detached", with moderate plasma temperature (T_e ~ 30 eV) away from the strike point, and detachment (T_e ~ 1-5 eV) near the strike point.



Main assumptions:

- Total chemical erosion yields (C atoms/ion) based on the 'mean values' resulting from the Garching and Toronto procedures

D^+	E(eV)	500 K	600 K	700 K	800 K
	5	0.008	0.010	0.012	0.007
	10	0.012	0.016	0.018	0.014
	15	0.014	0.019	0.022	0.018

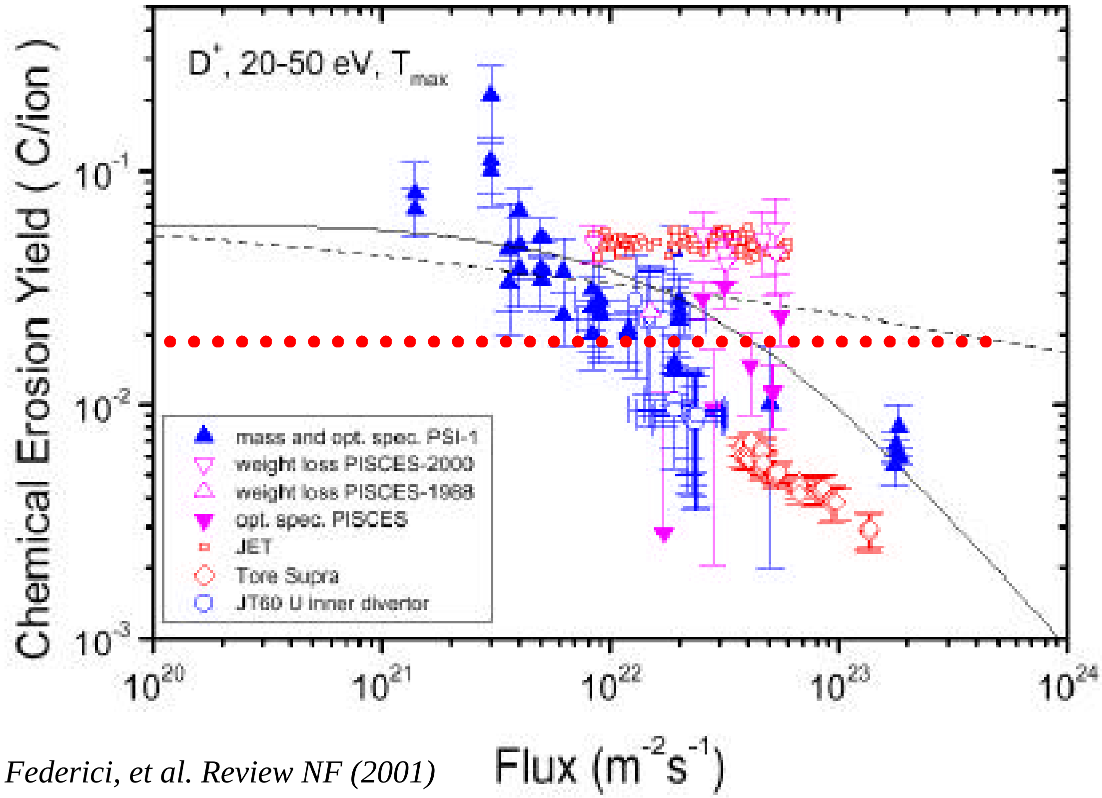
T^+	E(eV)	500 K	600 K	700 K	800 K
	5	0.009	0.014	0.014	0.009
	10	0.015	0.019	0.021	0.016
	15	0.017	0.022	0.025	0.020

- Use of the full low-energy chemical sputtered hydrocarbon spectrum including methane and higher hydrocarbon emission and rate coefficient package and;
- Use new C/hydrocarbon reflection model, predicting much less reflection (higher sticking) than the previous model.

B. Chemical Erosion of Carbon

- Further work is needed to clearly establish the extent to which parameters other than flux (e.g., energy, redeposition, photon efficiency and viewing geometry in spectroscopic measurements, etc.) affect the observed erosion rates.

- The results of recent investigations to study chemical erosion yields at high fluxes (i.e., 10^{22} - 10^{23} m^{-2} s^{-1}) in PISCES and in-situ tokamak measurements at JET show less pronounced or almost no flux dependence, in contrast to previous studies.



G. Federici, et al. Review NF (2001)

B. Summary of Co-deposition Analysis

Main conclusions:

- The calculations for ITER-FEAT, yield a T co-deposition rate of 2-5 g-T/pulse.
- Because of uncertainties in the modelling and the underlying database, the present results provide general trends rather than precise predictions.

		ITER-98	ITER-FEAT
Peak first-wall erosion	[nm/s]	**~0.1**	**~ 0.1**
Peak div. gross erosion rate	[nm/s]	**158**	**65**
Peak div. net erosion rate	[nm/s]	**2.86-16**	**6.4**
Divertor erosion lifetime†	[No. Shots]	**~1250**	**>7000**
Tritium codeposition rate from divertor	[g-T/pulse]	**2*-14****	**0.4*-5****
Tritium codeposition rate from wall eroded beryllium	[g-T/pulse]	**0.1-0.6**	**0.1-0.4**
Operation prior to recovery of codeposited T	[No. Shots]	**70 - 500**	**70-900**

† **20 mm carbon, sputtering only (no ELMs and disruptions)**
* **physical sputtering only**
** **with chemical sputtering.** ***Courtesy of Brooks (ANL)***

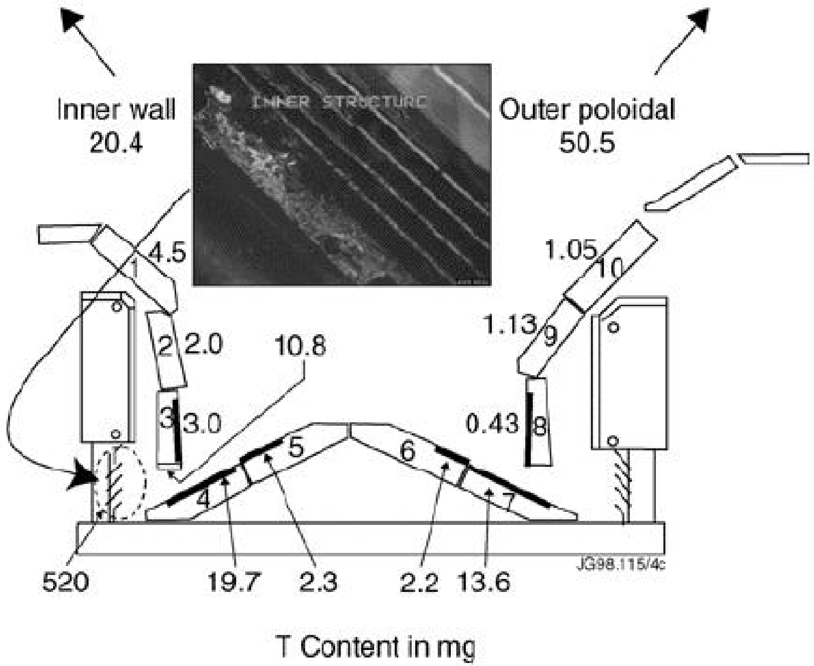
ITER FDR (1000 s pulse/ 1000 g max. allowed T- codeposited inventory)
ITER FEAT (400 s pulse/ 350 g max. allowed T- codeposited inventory)

Large uncertainties:

- Models used to study detached-plasma C erosion have not yet been fully validated for these conditions.
- Still unresolved poor agreement between code predictions with JET erosion/ co-deposition results (see next slide).

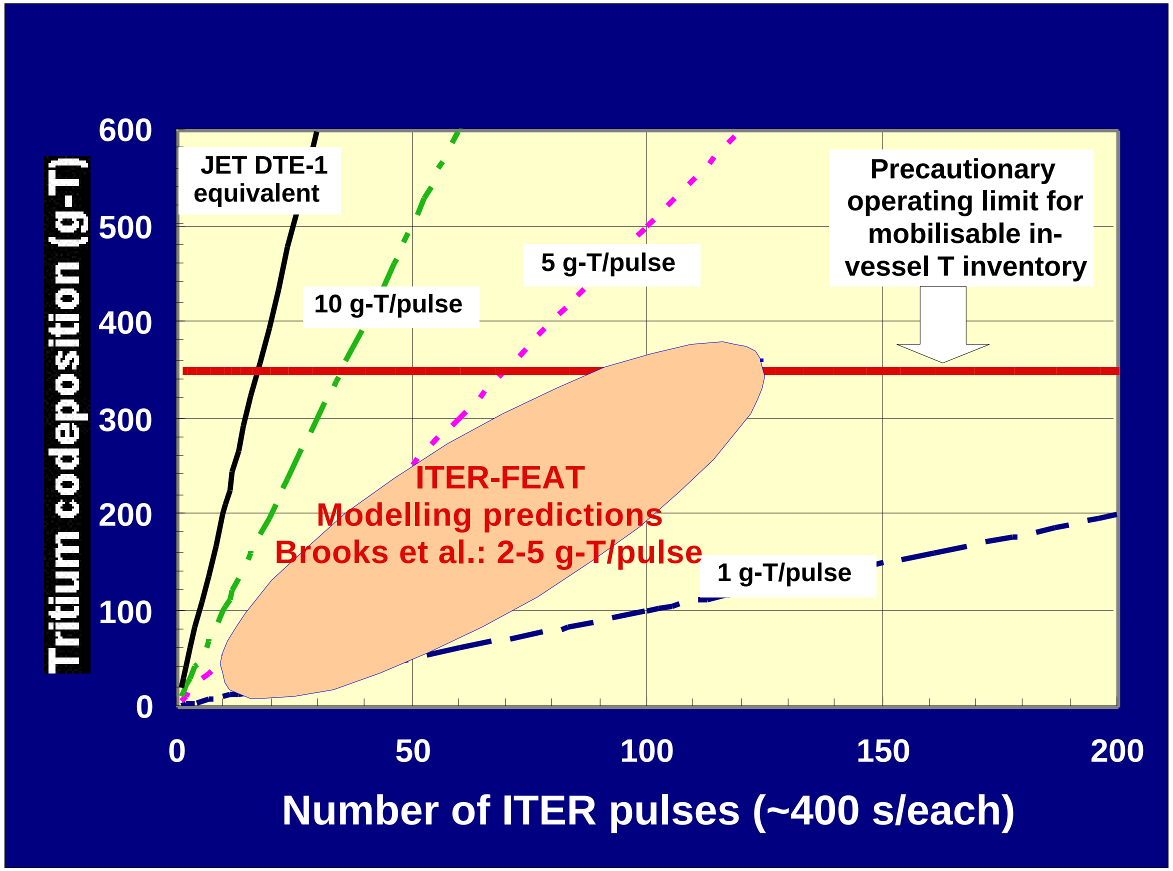
B. Modelling Erosion/Co-deposition at JET

- The JET erosion/deposition asymmetry and the absolute magnitude of the carbon deposition is not correctly modelled by DIVIMP or REDEP/WBC using ``conventional'' values of low-energy chemical sputter yield (~ 1% C/D), sticking probabilities and other parameters.
- Agreement between observed deposition and DIVIMP modelling was achieved by including three additional mechanisms, which alter deposition patterns in JET by more than an order of magnitude.
- These mechanisms are:
  - (i) drift in the SOL as has been observed in JET;
  - (ii) additional interaction with the main chamber wall, and
  - (iii) enhanced erosion of the re-deposited carbon films at the inner target.



B. In-vessel Tritium Control: Requirements

- Maintaining C in ITER strongly impacts the control of the in-vessel T inventory.
- Removal of the co-deposited layers is required.
- Frequency of clean-up depends on the co-deposition rate (**modelling/design**) and in-vessel tritium hold-up limits (**safety).**
- A precautionary operating limit of **350 g-T** for the mobilisable in-vessel tritium (1000g in 1998 ITER design) is now set in ITER-FEAT, based on safety considerations.

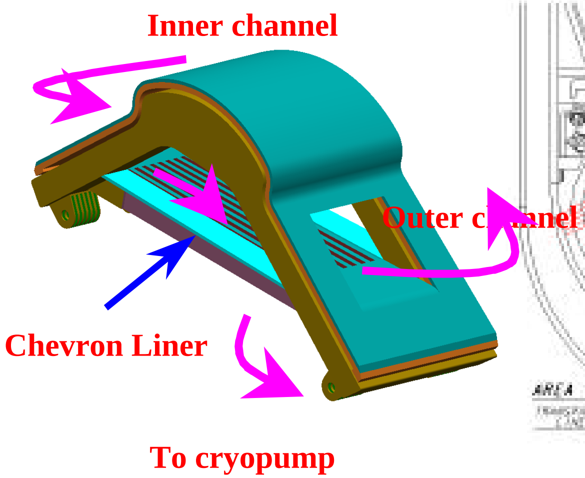


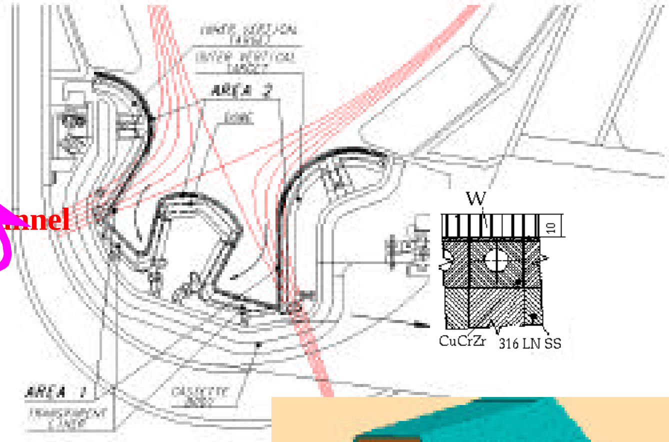
B. In-vessel Tritium Removal: Options Available

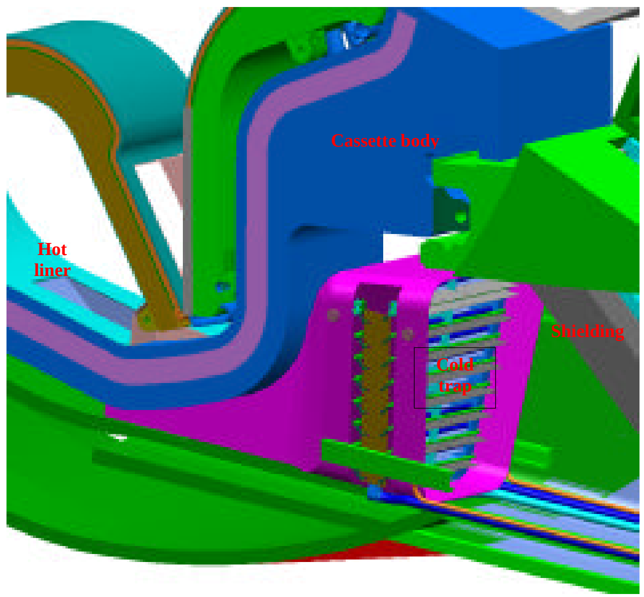
- There is currently no good way to remove T from co-deposited C--based films.
- The only methods proven effective for removing T involve **(1)** oxidation of the co-deposited layers (e.g., thermo-oxidative erosion > 250˚C, or O plasma discharges) or **(2)** physical removal.
- For carbon co-deposited films, erosion rates strongly depend on the microstructure of layers.
- Although 240˚C may remove soft films, due to the variability of film properties, a baking capability at temperatures (> 300˚C) would be desirable, in particular to reduce ratcheting effects, i.e. accumulation of residuals (cleaning efficiency >95 %).
- Drawbacks of using oxygen, especially at elevated temperatures, are collateral effects on other reactor vessel components, and recovery time for normal plasma operation.

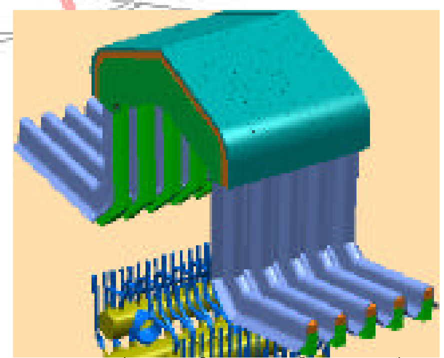
B. In-vessel Tritium Control: Design Optimisation

- Besides developing effective techniques of removal, design solutions for the divertor PFCs are being explored which could minimise co-deposition, either by keeping the surfaces hot or by using cold traps.
- Implementation of these measures in the design is, however, very difficult.

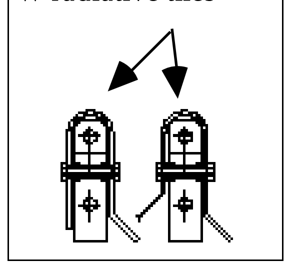








W radiative tiles



C. Key R&D Needs

- There are still large uncertainties in quantifying in-vessel T inventory of ITER-FEAT.
  - Carbon chemical erosion yields at high fluxes;
  - Very poor validation of E/R models for plasma detached conditions;
  - Poor knowledge of underlying mechanisms in the plasma edge and on the walls (e.g., drifts, transport and deposition of low sticking radicals).
- There is a need to better quantify carbon sources in todays' machines, e.g., to match experience of JET, and to make reliable extrapolation to ITER.
  - We need an experiment at JET with a Be first-wall with C and W in the divertor, which would closely mirror the ITER case and would help answer some of the relevant questions and address synergistic effects that cannot be determined in laboratory simulation experiments and one-material lined tokamaks.
- Experiments must be continued in ASDEX-Upgrade (with a W-clad inner wall) and in C-Mod (with an all-Mo) to fully demonstrate operation compatibility and the merits of high-Z materials.
- Implications of mixed-materials effects could be large and require R&D:
  - e.g., T co-deposition in Be/O mixed-layers, reduced erosion on a Be-covered C-target, higher temperatures required for T-removal.

C. Key R&D Needs (cont'd)

- For tungsten, while the damaged surface structure, appears to be beneficial by allowing enhanced reemission from the exposed surface, the resulting blister cracking may lead to material being injected into the plasma and need further investigation.
- Investigate transport and deposition of hydrocarbon radicals (e.g., sticking probability as a function of temperature) on surfaces hidden from the plasma.
- Besides developing effective techniques of tritium removal, design solutions for the divertor PFCs should be validated, which could minimise co-deposition, either by keeping the surfaces hot or by using cold traps. Implementation of these measures in the design is, however, very difficult.
- Important goals for the use of W on the targets are the minimisation of thermal quench frequency, the reduction of the thermal quench energy deposition through dissipative methods, and high performance regime with smaller ELMs.

D. Conclusions

- Co-deposition remains the major tritium repository for the ITER-FEAT design, even if the use of carbon is minimised to the divertor strike plates.
  - Modelling shows that the T-co-deposition rates for ITER are still high: **(1) 2-5 g-T/pulse,** and the administrative in-vessel T-inventory limit could be reached after **70-400** pulses.
  - Retention of tritium by other mechanisms such as implantation/diffusion with trapping and breeding, is expected to be only few hundred grams at the EOL.
- These estimates are subject to several uncertainties due to the lack of relevant data and code validation. More modelling work is needed to resolve poor agreement between codes and tokamak results.
- As long as C is used in ITER, efficient techniques are needed to remove in-situ the co-deposited tritium.
- Besides developing effective techniques of removal, design solutions are being explored which could minimise co-deposition, e.g., hot-liner, cold-traps, use of W.
- Important goals for the use of W on the targets are the minimisation of thermal quench frequency, the reduction of the thermal quench energy deposition through dissipative methods, and high performance regime with smaller ELMs.

PMI Review

- Plasma-material Interactions in Current Tokamaks and their Implications for Next-Step Fusion reactors
  - Authors: G. Federici, C.H. Skinner, J.N. Brooks, J.P. Coad, C. Grisolia, A.A. Haasz, A. Hassanein, V. Philipps, C.S. Pitcher, J. Roth, W.R. Wampler, D.G. Whyte.

- Submitted to Nuclear Fusion (Jan. 2001)
- It is available also as a joint IPP/PPPL report: PPPL-3531/IPP-9/128
- It is also available on the web (in pdf 8.8 MB);
  - http://www.pppl.gov/pub_report/2001/PPPL-3531-abs.html