# Problems of Plasma Surface Interactions in Fusion Devices

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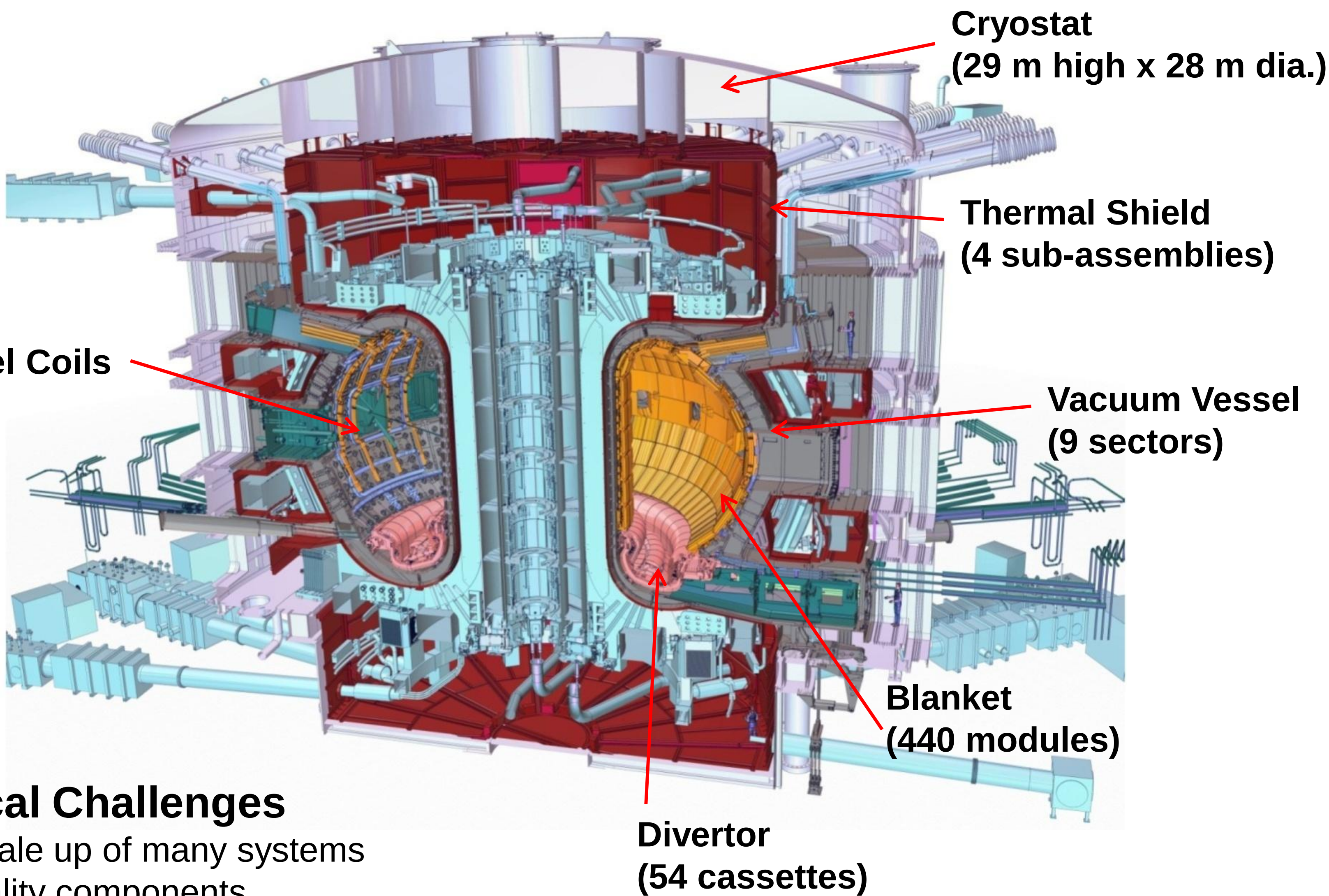
***National Nuclear Research University MEPhI***

*Plasma in astrophysics and in the laboratory. Ignitor challenge. Italian-Russian meeting*
*20-21June 2011*

# ITER Facility




## Technical Challenges

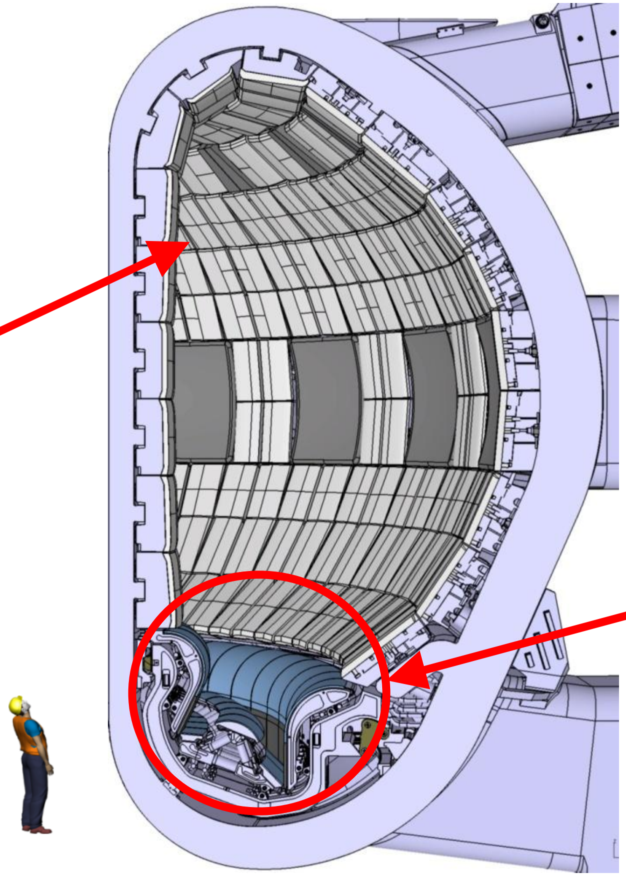
- Large scale up of many systems
- High quality components
- Tight tolerances
- Manufacturing around the world
- Highly integrated design

# Materials choice

ITER will be the first tokamak in which plasma-facing components will play a major role in machine performance and availability

→ **ITER must demonstrate that sustained fusion performance can be compatible with materials facing the plasma**

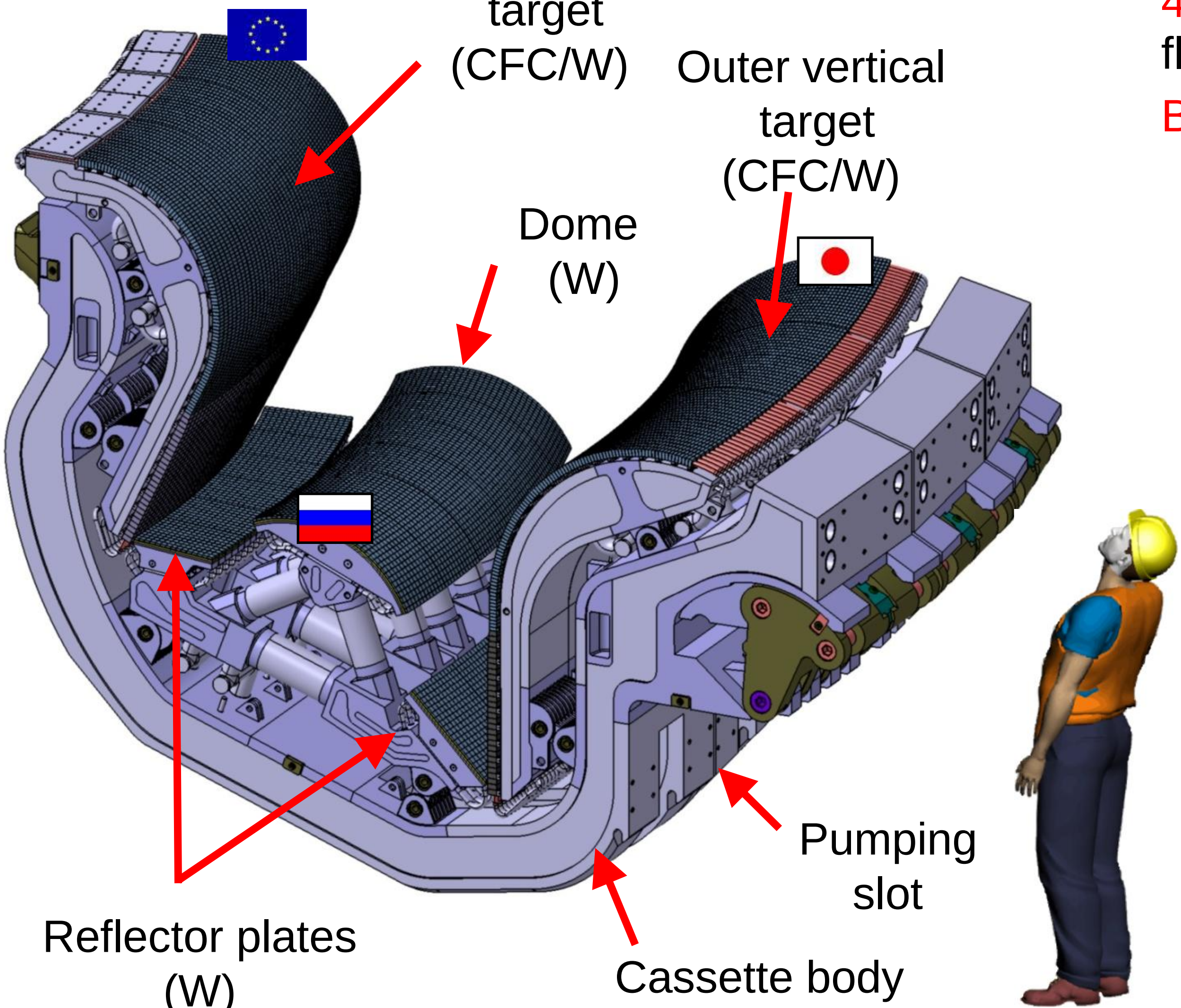
First wall
~700 $m^2$
Beryllium
replaceable → high Z (W) after first DT phase?



Divertor
~160 $m^2$
CFC/W,
non-active phase
All-W,
for nuclear phases

**PWI – is the key problem in ITER performance!**

# Divertor assembly




54 divertor assemblies (~8.7 tonnes each)

4320 actively cooled heat flux elements

Bakeable to 350°C

First divertor (non-active operation):

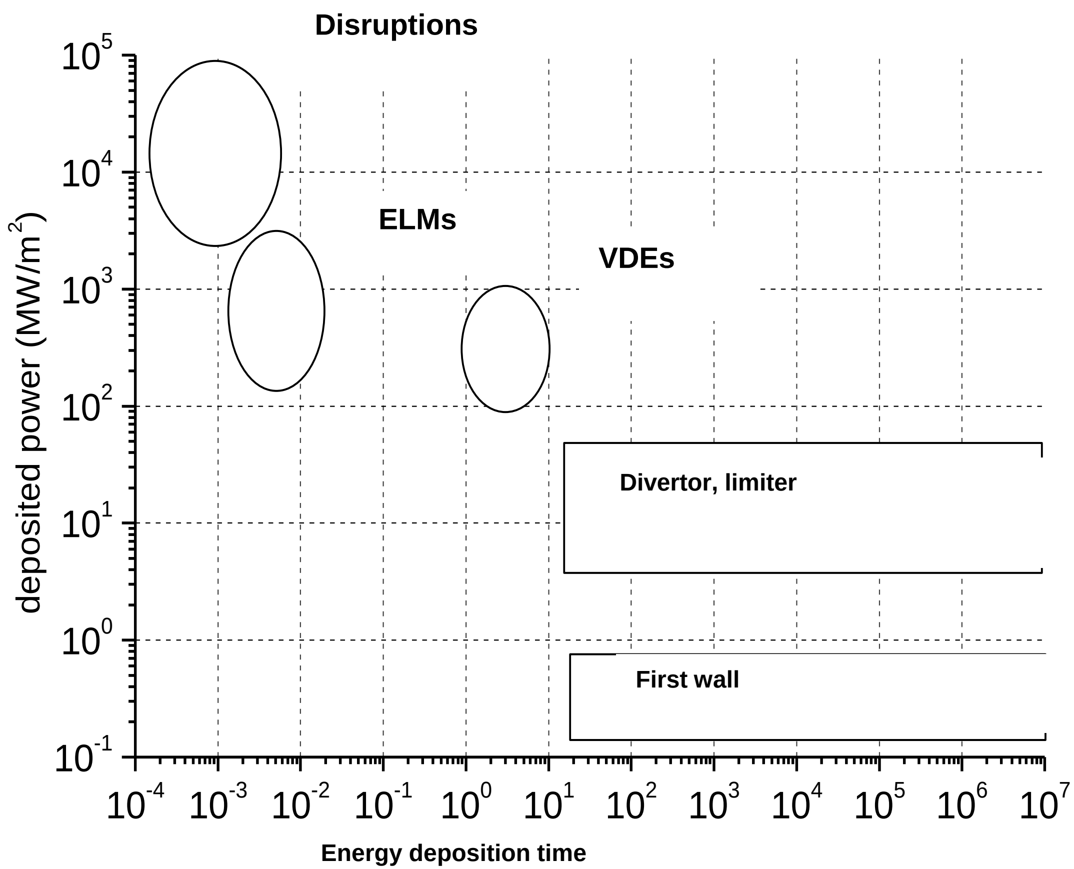
CFC at strike points
W on the baffles

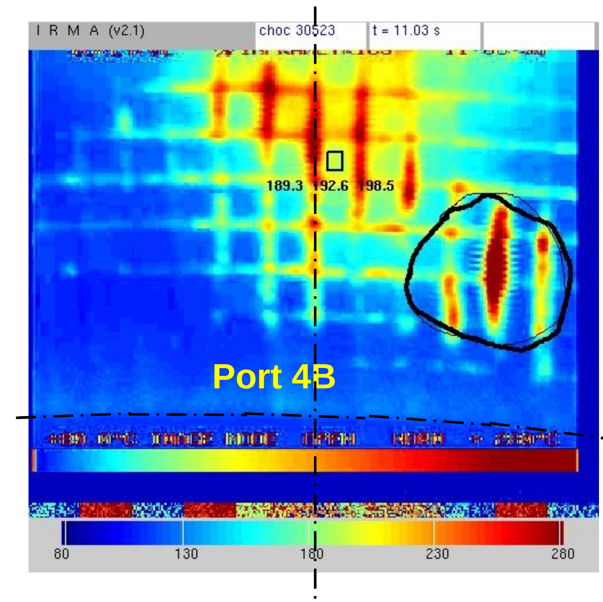
All-W for nuclear phase

**Technical Challenges**

- High steady state heat flux up to
10 MW/m2 (3000 cycles) &
20 MW/m2 (300 cycles)
- Material bonding techniques
- Remote handling requirements

# Heat loads in fusion devices

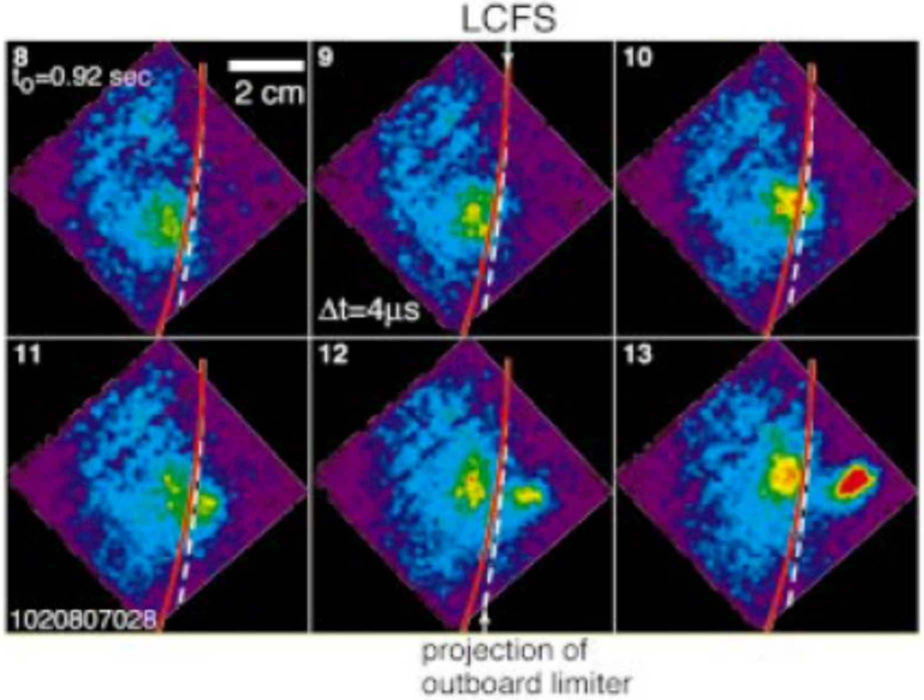



Thermography of Tore Supra plasma faced component

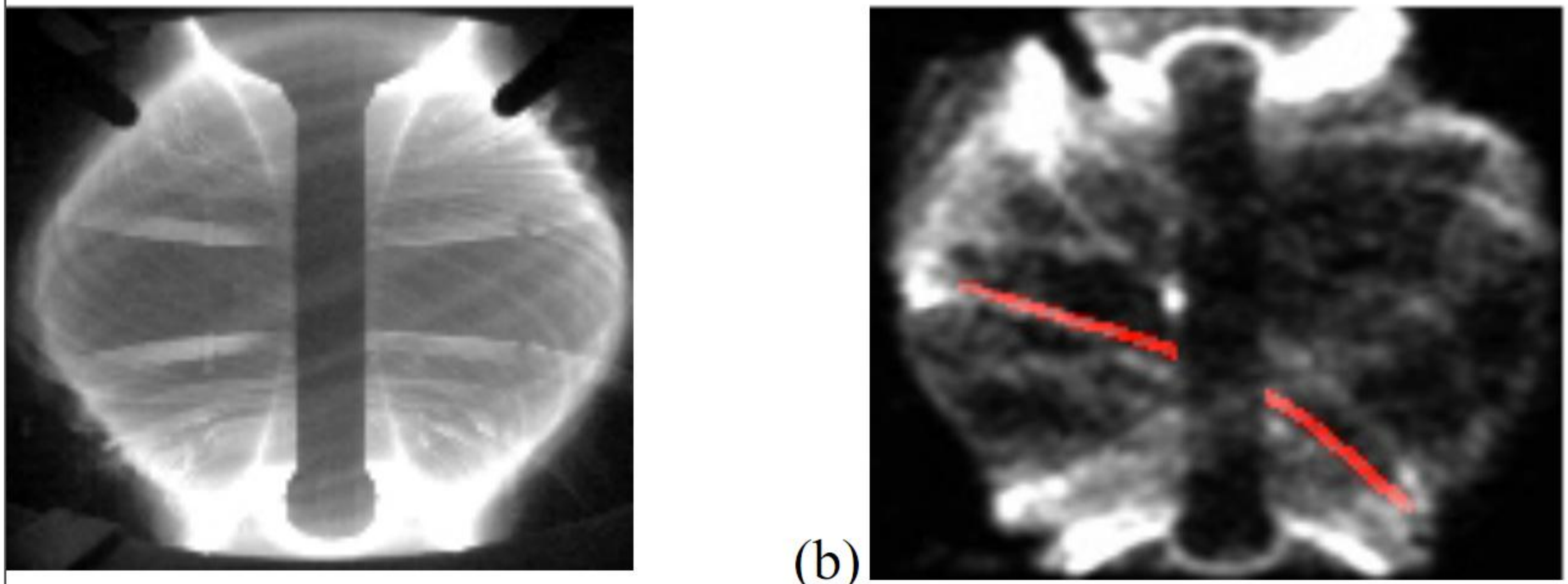
R.Behrish et al

# Turbulence at plasma edge




Blobs dynamics at the outer edge of Alcator C-Mod

# Edge localized modes at NSTX




Plasma blob (a) and ELM filaments in NSTX tokamak

# Transients - ELMs and steady state loads

**Start-up/ramp-down -** Both LFS & HFS start-up scenarios are available

**"Steady state" loads**

**Importance of magnetic equilibrium –** position of separatrix

**Far SOL plasma fluxes -** far SOL tails in $n_e$ and $T_e$ due to turbulence driven convective (filamentary) transport

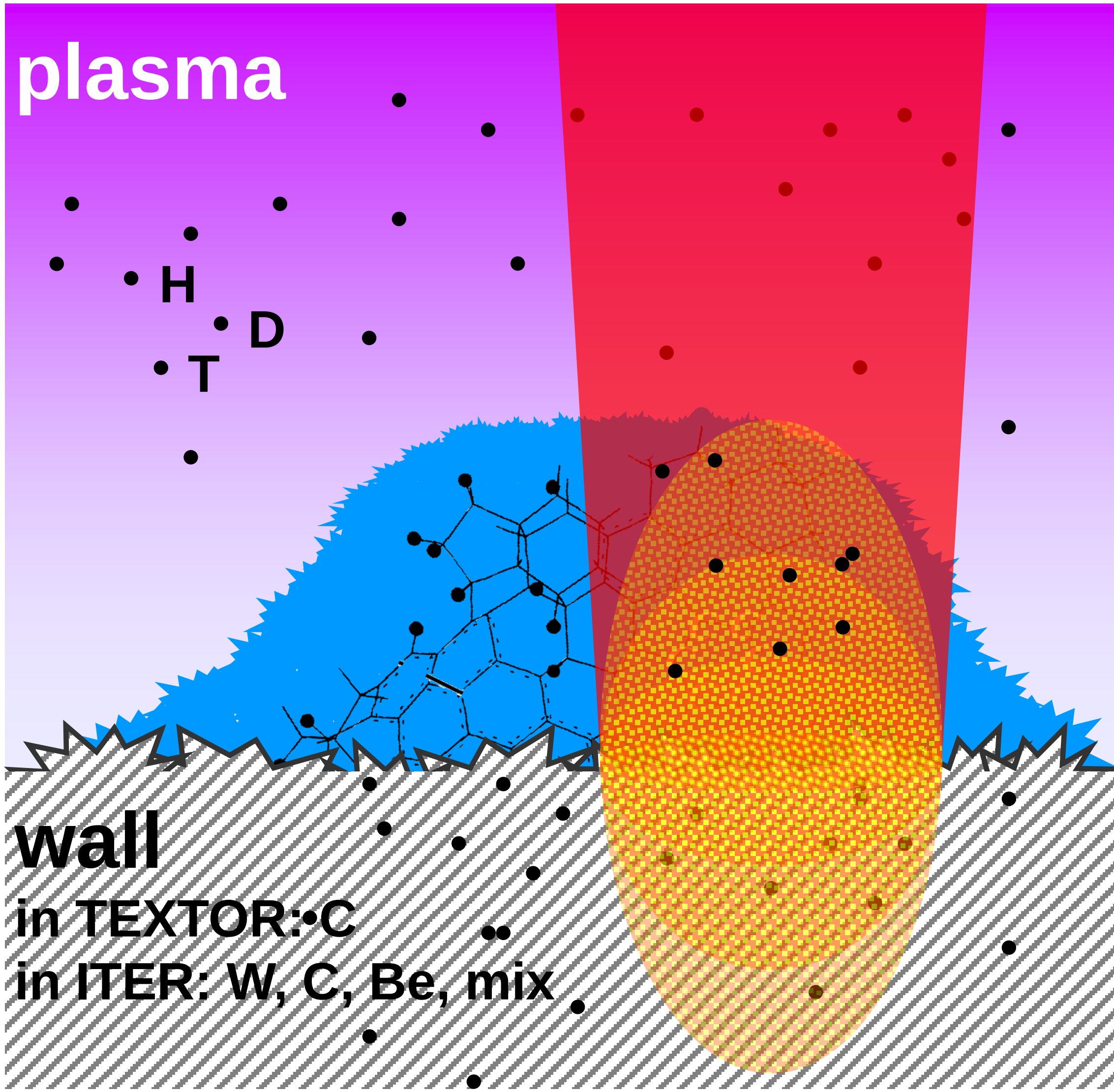
**ELM wall loads -** to convect power to far SOL regions through filamentary transport, → Max ELM-averaged loads at upper X-pt: 12-24 $MWm^2$, ($f_{ELM}$ = 20 – 40 Hz). To be added to the peak static load of 8 $MWm^{-2}$

W exposed to 100 pulses of 1.5 $MJm^{-2}$

QSPA-T


- Significant melting, bridging of castellation gaps, lifetime reduction
- Issues of operability on damaged targets, dust production

# Physics of Retention & LID: overview




**plasma-wall-interactions:**

**co-deposition of H in layers**
e.g. amorphous hydrocarbon layers (a-C:H layers)

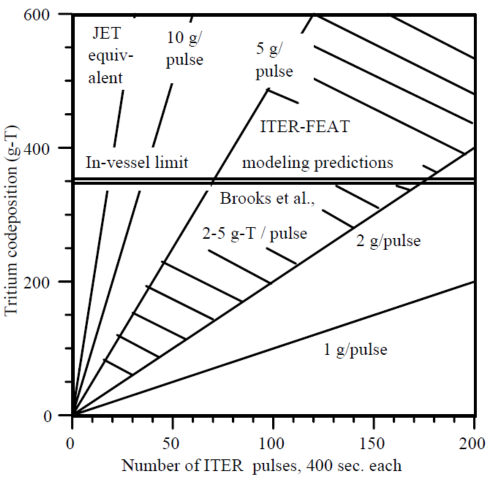
also: Be:H layers, W/C:H or Be/W/C:H etc. (mixed materials)

**implantation of H by plasma**

# Tritium retention

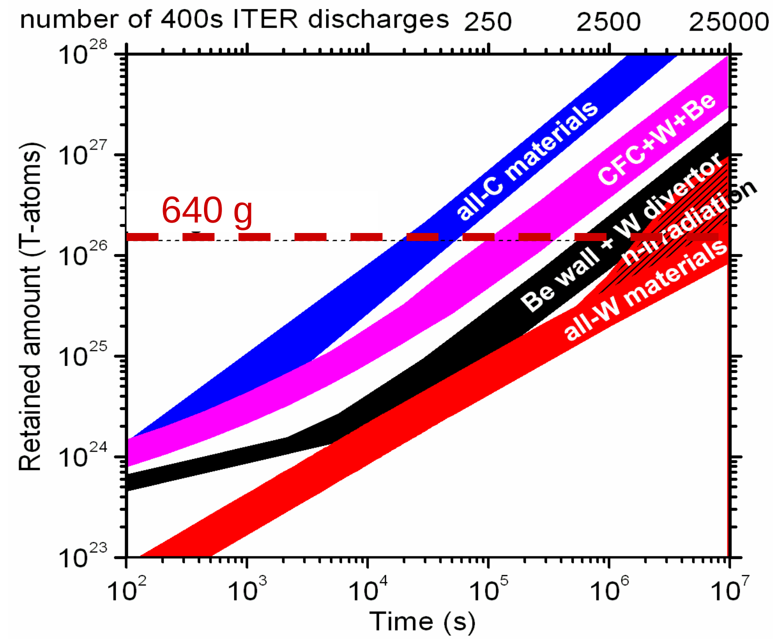
4kg of tritium will be held on ITER site → only about 20 kg of tritium anywhere in the world at any one time!

- ITER predicted tritium accumulation rate is that experienced in JET
- Model underestimates JET retention by factor x40.




*J. Brooks (ANL), A Kirschner (FZJ)*

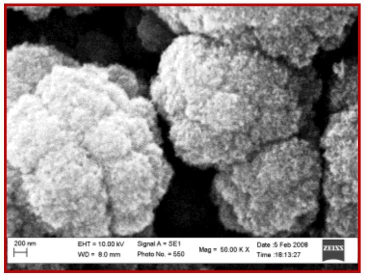
(2004)




- Main chamber erosion = divertor deposition model:
  - → 1500 – 3000 full burn shots before T-limit for Be/W
  - → 100 – 1000 only for C/Be
- But these approximate calculations assume no net redeposition in the main chamber

**→ The real situaiton could be worse .....**

**QSPA-T**
**1 MJ/m2**



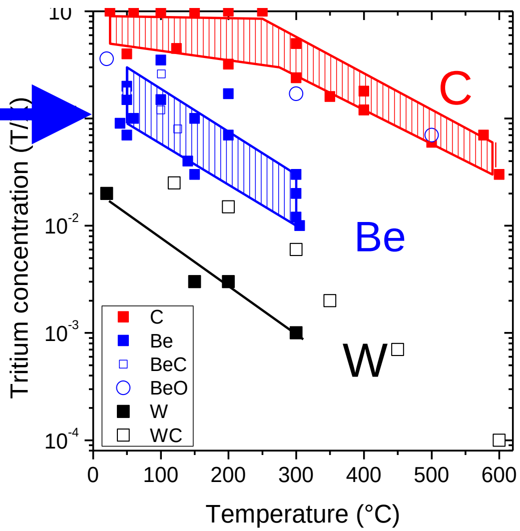
# Tritium retention - experiments

Co-deposition / bulk

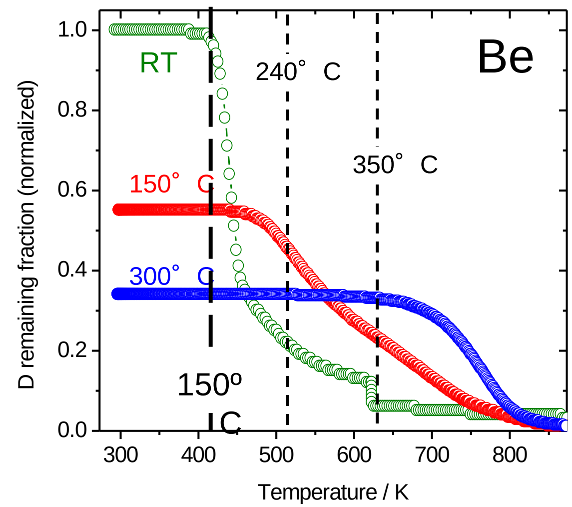
J. Roth, K. Sugiyama et al., 14th DivSOL ITPA, Korea, Oct. 2010

co-deposition with Be can be high

Retention in W (even irradiated) probably not a serious issue in ITER




R. P. Doerner et al., NF (2009) 035002

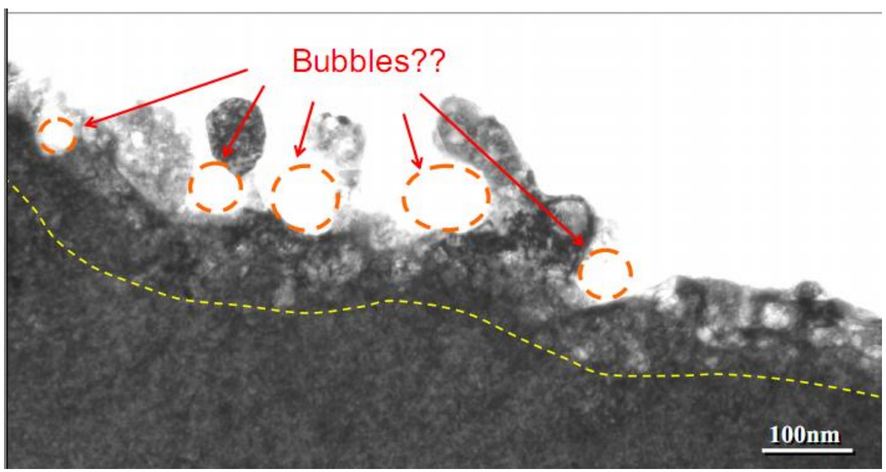


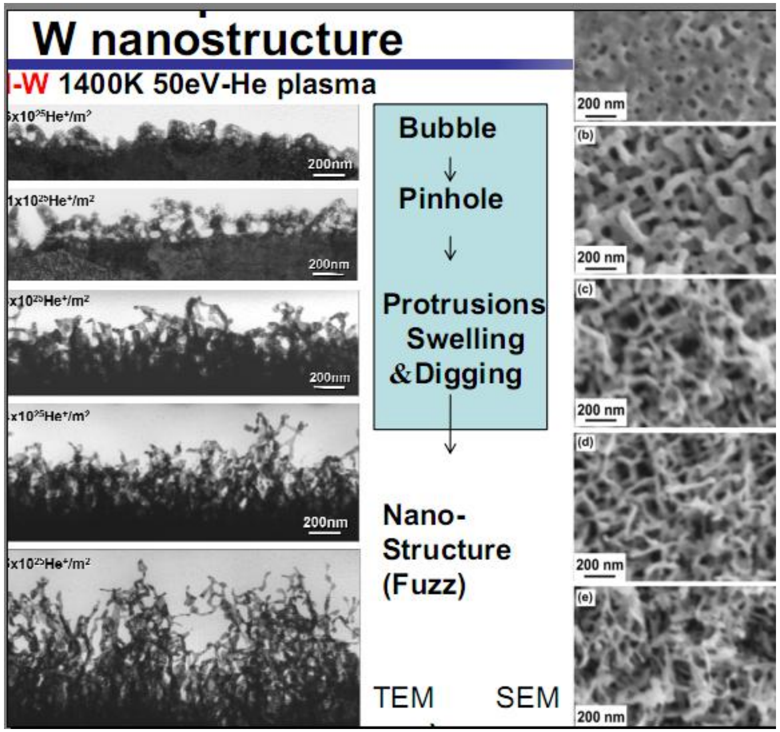

- Less retention at higher implantation temperature
- But need higher bake temperature to release

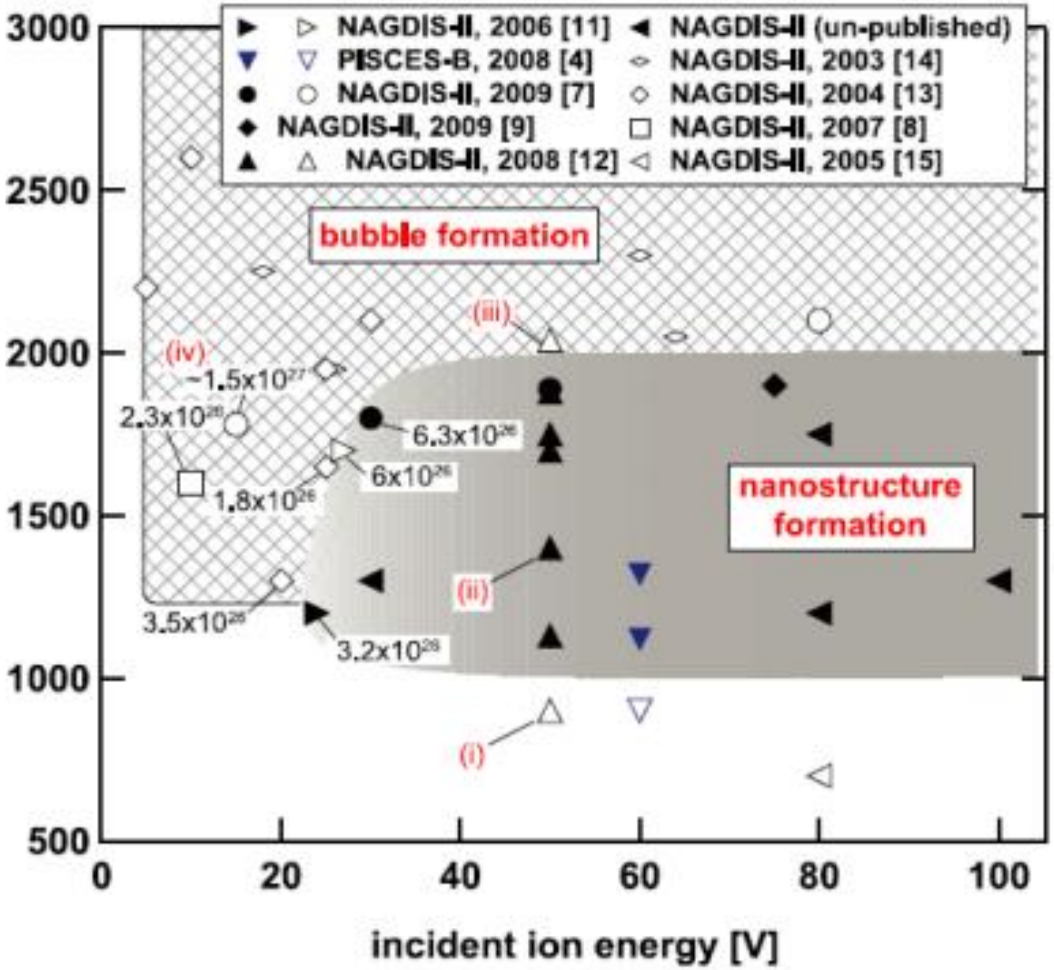
ITER main walls will operate between 150 - 200ºC during burning plasmas but only ~100ºC in non-active phases

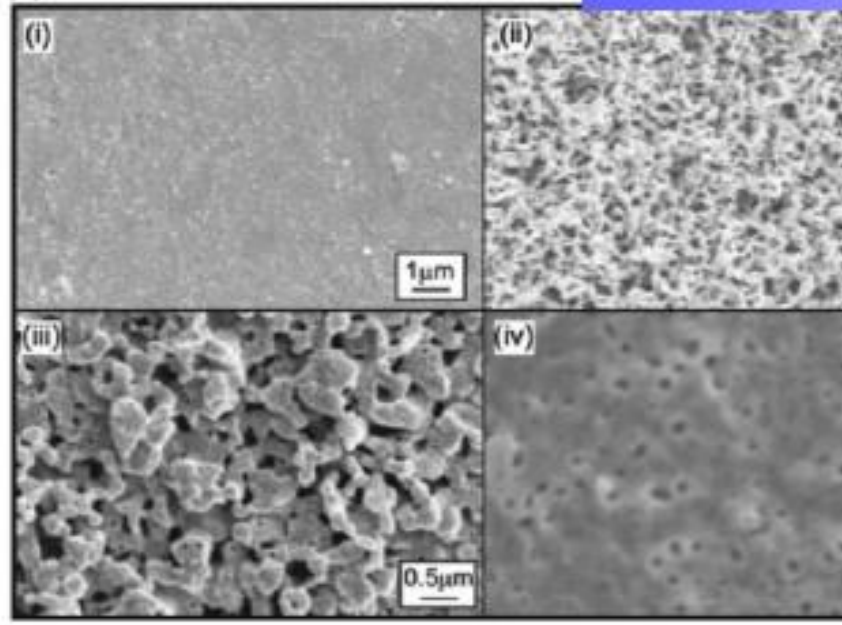
# W in tokamaks

**Fuzz formation at He irradiation of hot W surface**





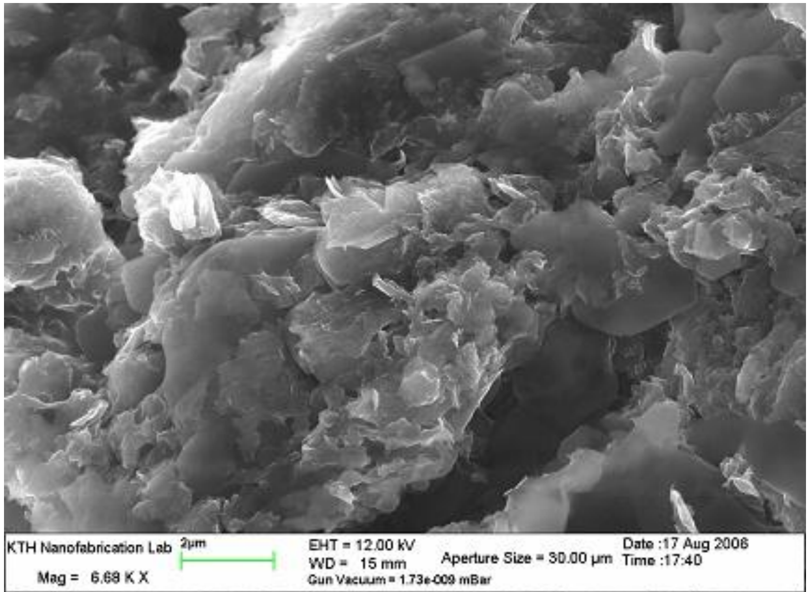



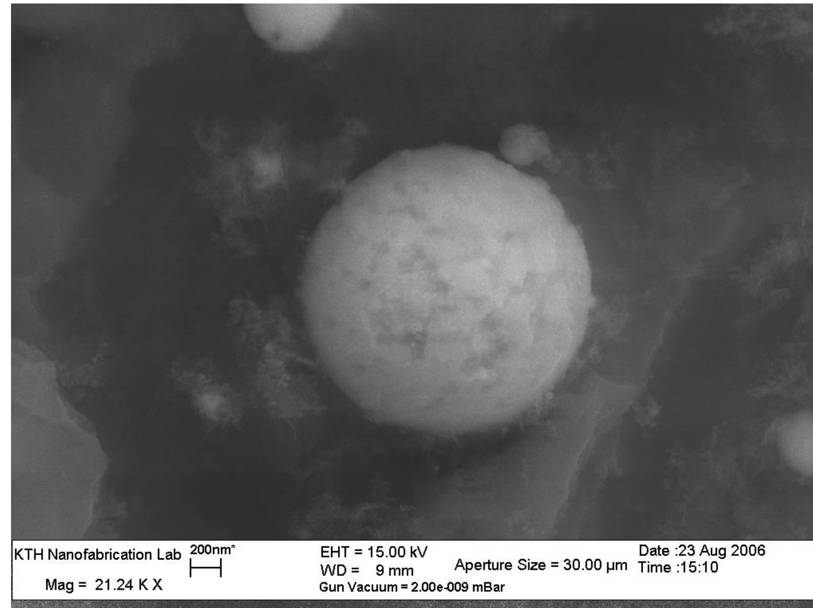

- Closed markers with nanostructure
- Open markers without nanostructure

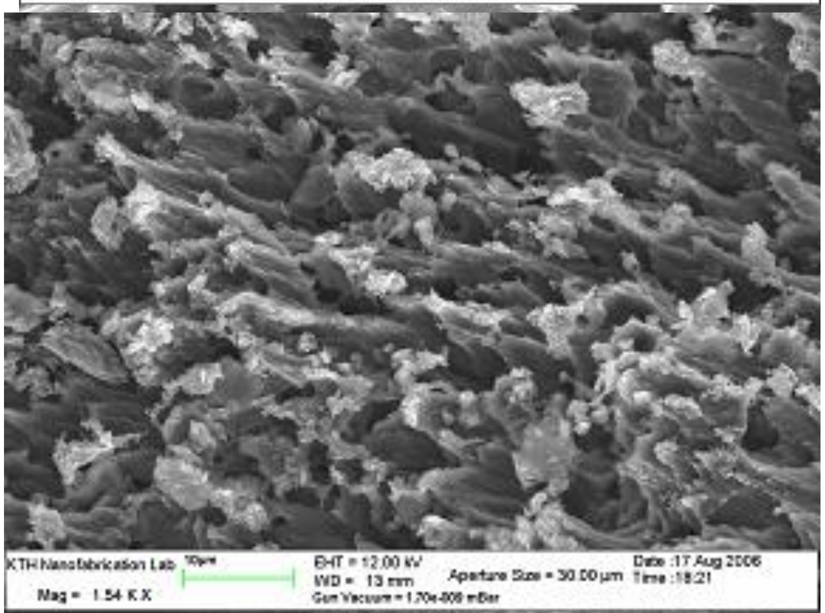
Surface Temp: **1000 K < T < 2000 K**
Ion Incident Energy**>20 eV**

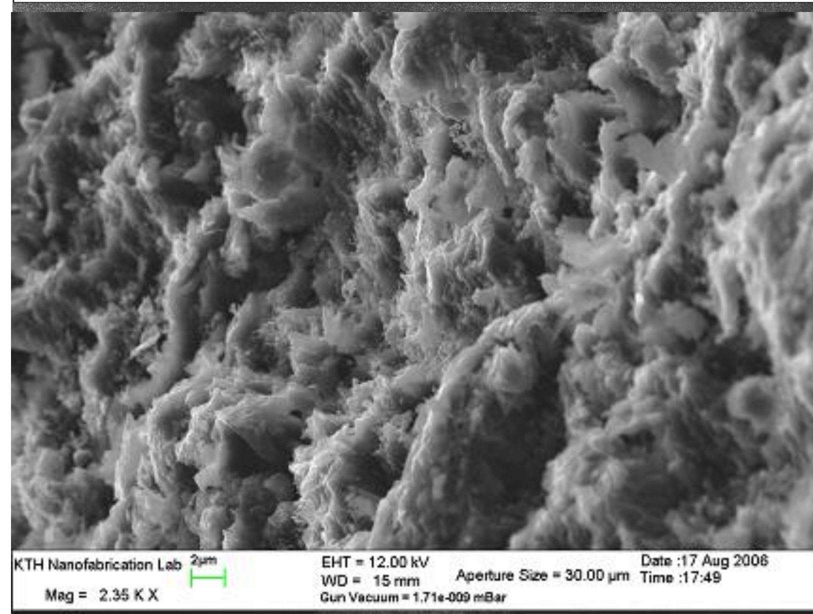
[4] M. Baldwin NF (2008).
[7] W. Sakaguchi JNM (2009)
[8] S. Kajita, NF (2007).
[9] S. Kajita, NF (2009).
[11] S. Kajita, J. Appl. Phys. (2006).
[12] W. Sakaguchi, Proc. 18th Int. Toki Conf.
[13] D. Nishijima, JNM (2004).
[14] D. Nishijima, JNM (2003).

# Surface transformation in tokamak T-10

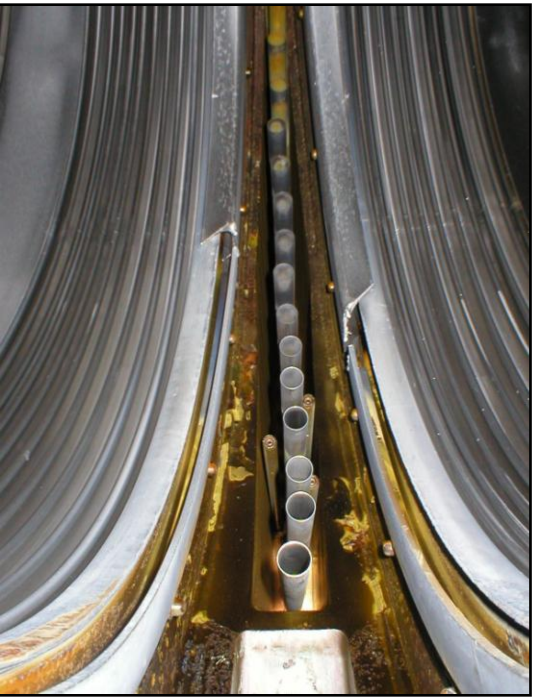


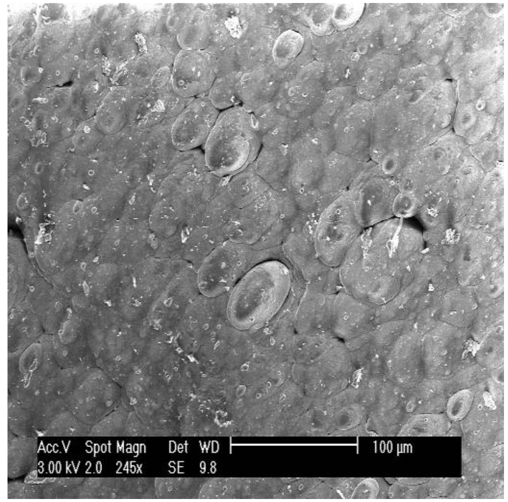



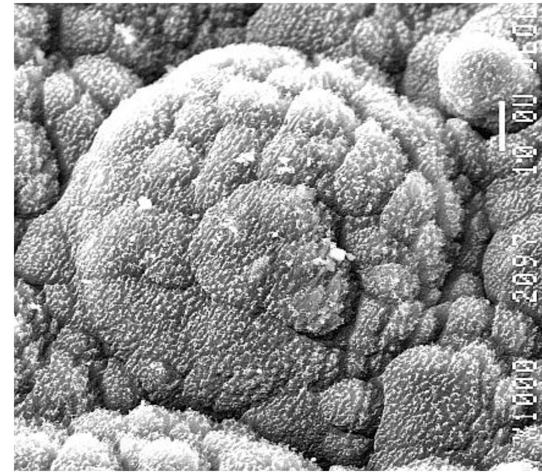



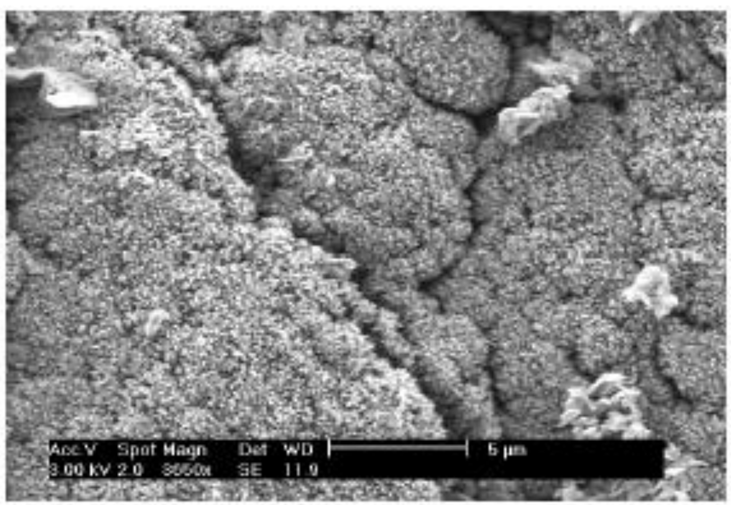



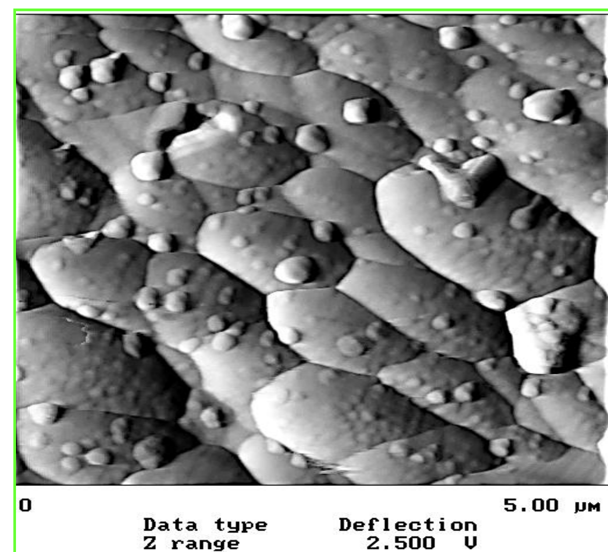
# Codeposits in tokamaks (T-10)








**В.П.Будаев**

14 конференция-семинар
«ВЗАИМОДЕЙСТВИЕ ПЛАЗМЫ С ПОВЕРХНОСТЬЮ»
МИФИ 4 февраля 2011г


T+: -36.552 ms
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Cam: Phantom v.7
AcqRes: 800 x 600
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Exp: 495 µs
EDR: 0 µs
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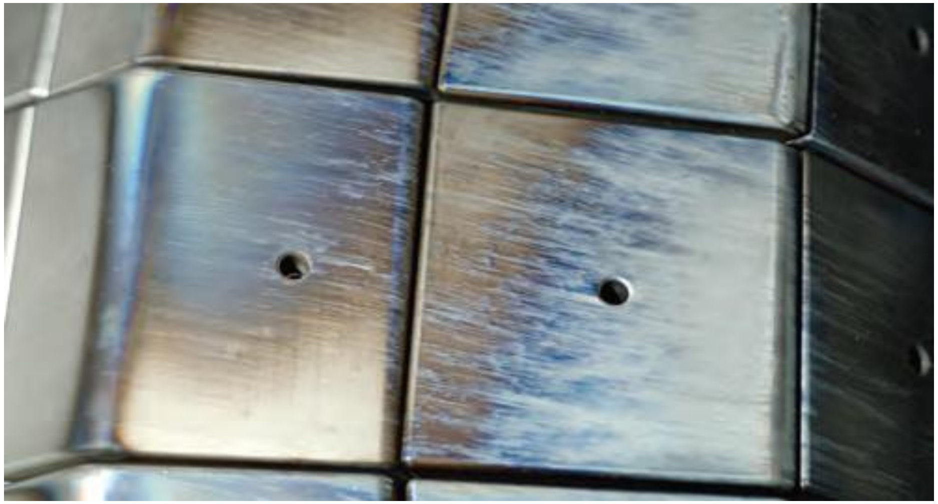
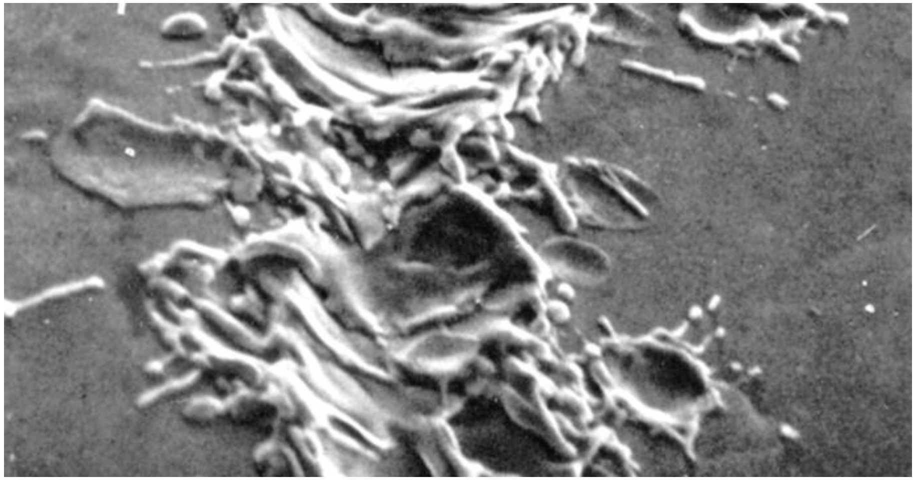
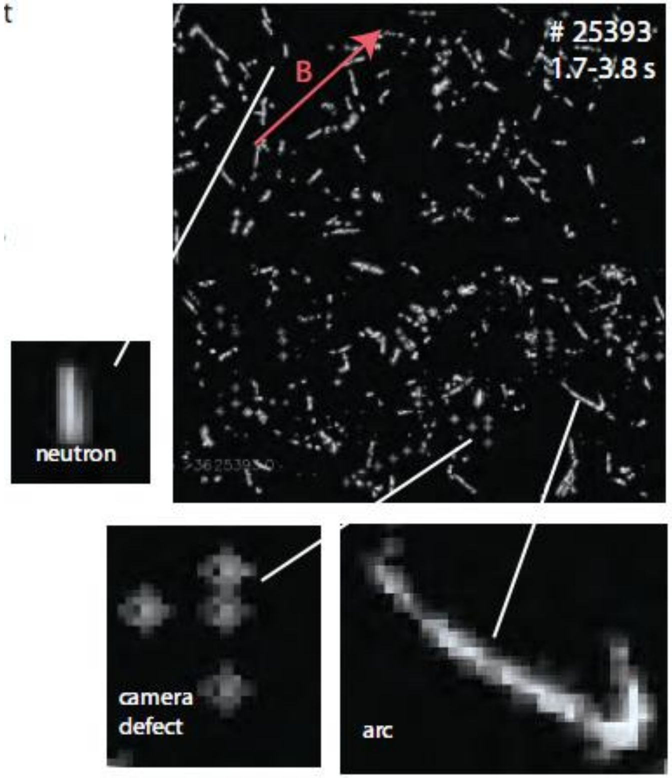
Range data:
Tore Supra #43493 - Runaway control experiments - C. Reux, F. Saint-Laurent

# Arching in tokamaks

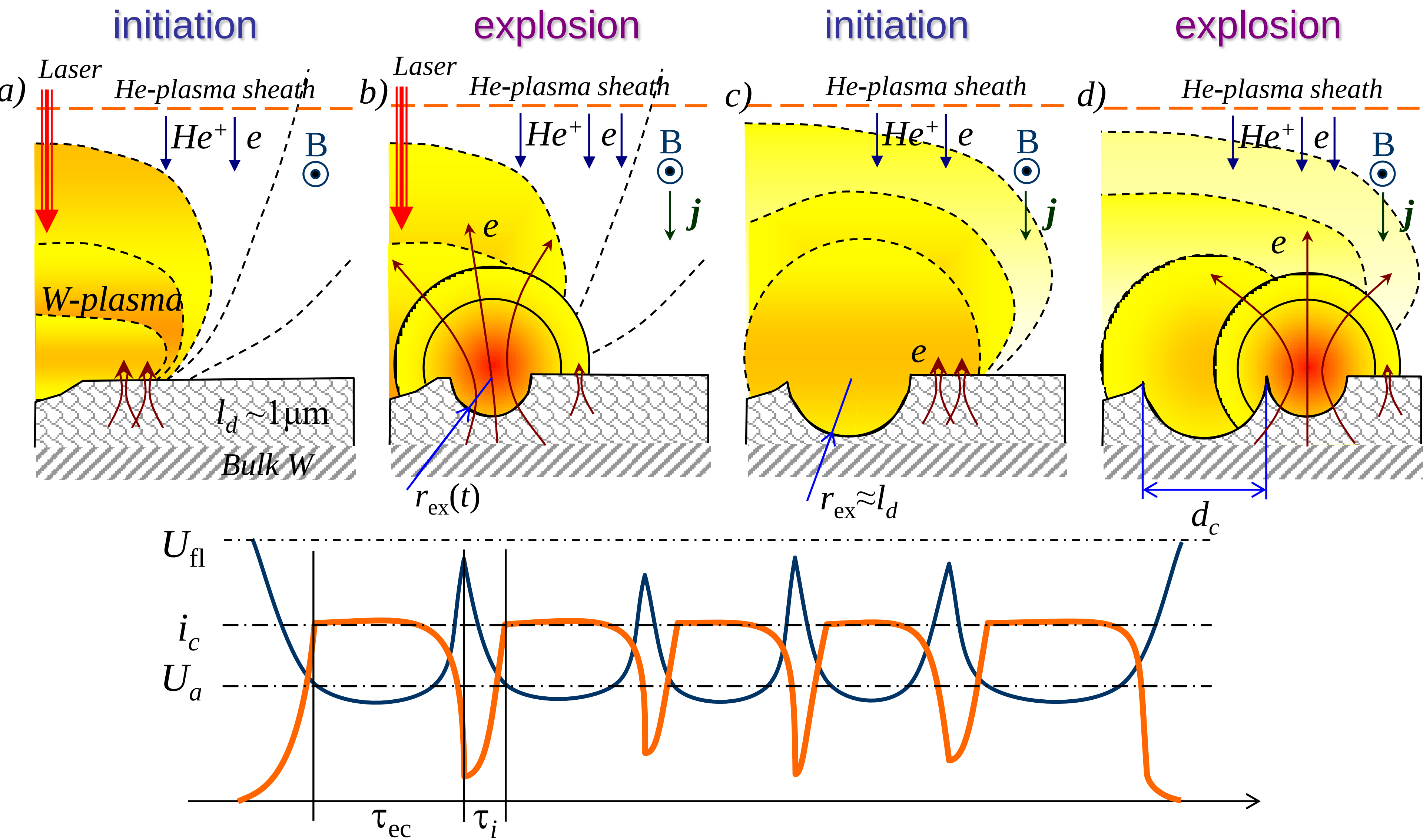






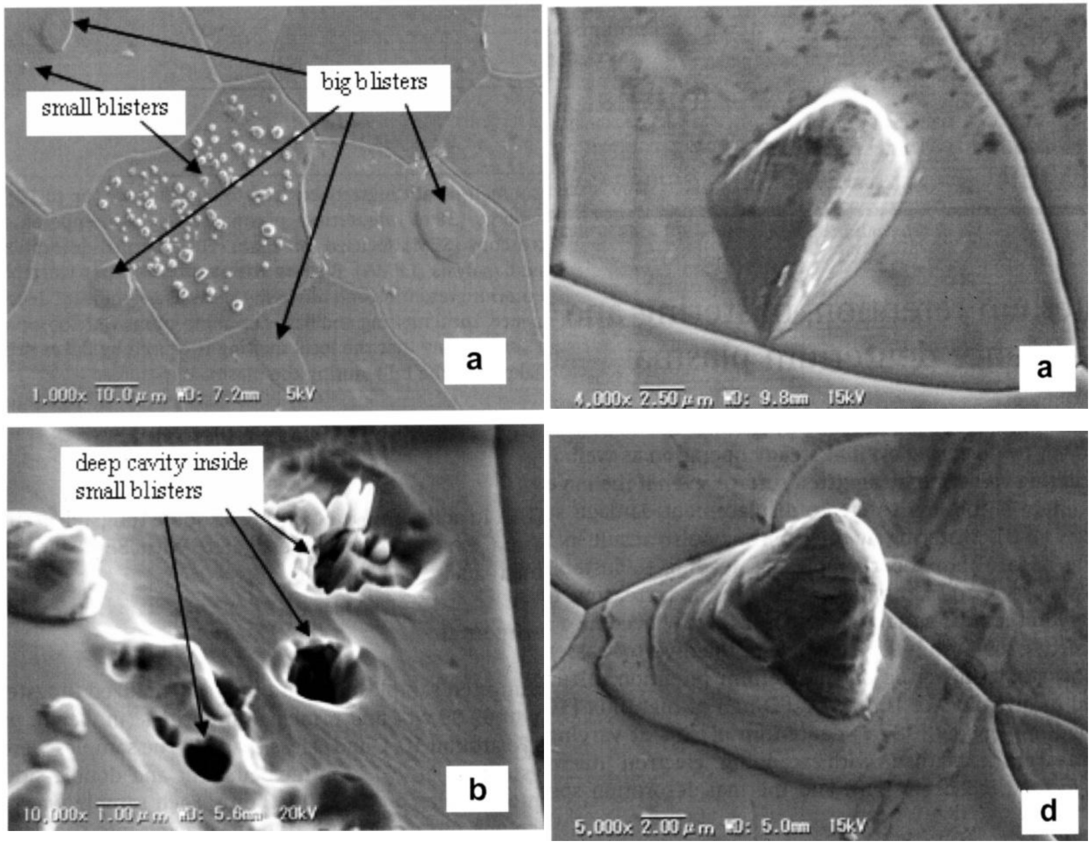
# ECTON MECHANISM OF UNIPOLAR ARC…

S A Barengolts G A Mesyats M M Tsventoukh 2010 *Nucl. Fusion* **50** 125004



# Tungsten . BLISTERING

## (cones and pyramids)




The temperature of blistering is specific for cooper irradiated with moderate fluxes. (T=480-520 K)

Fig.1. Blisters (cones and pyramids) on the W surface irradiated by high-flux ($10^{22}m^{-2}s$), high-fluence (up to $10^{27}m^{-2}$) and low-energy (38 eV) deuterium plasma at T=480-520 K [1] . a) Small blisters, b) cavities inside them, c) pyramid-like big blister, d) cone-like big blister,


[1] W.M.Snu, M Nakamichi, V.KH. Alimov, G.-N. Luo, K. Isobe, T. Yamanishi et al. J. of Nucl. Mater. 390-391(2009)1071.

# Tungsten : BLISTERING

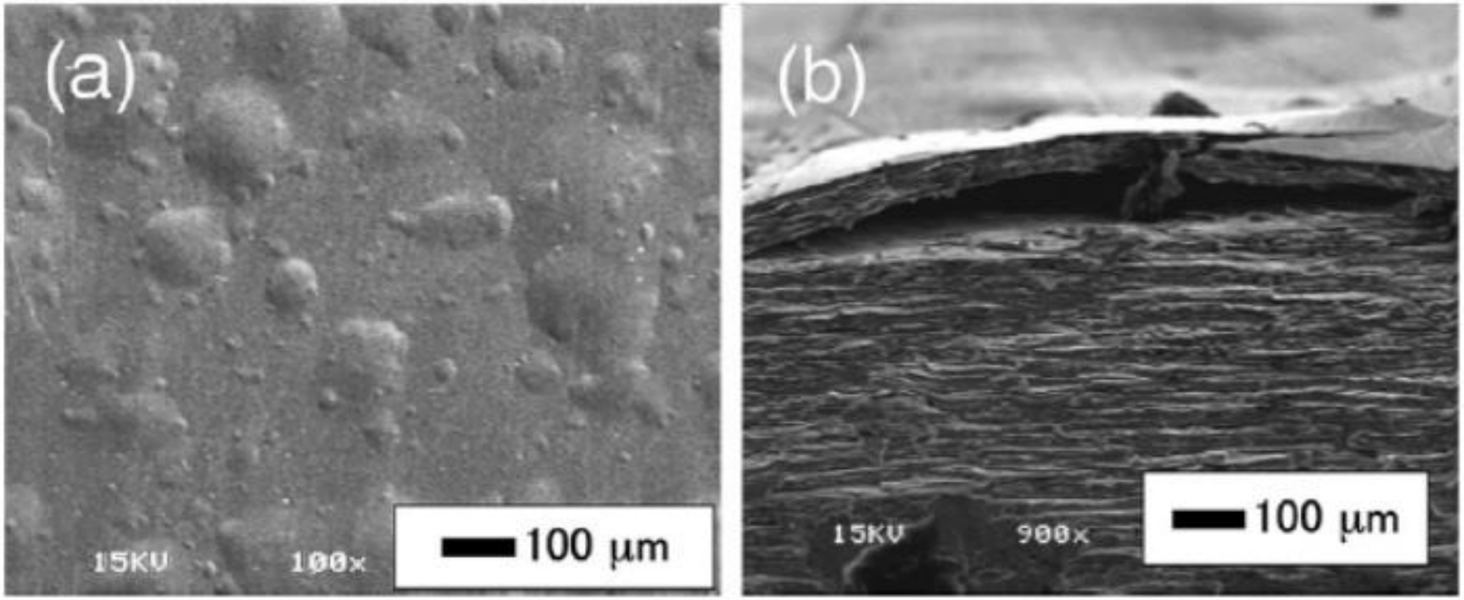
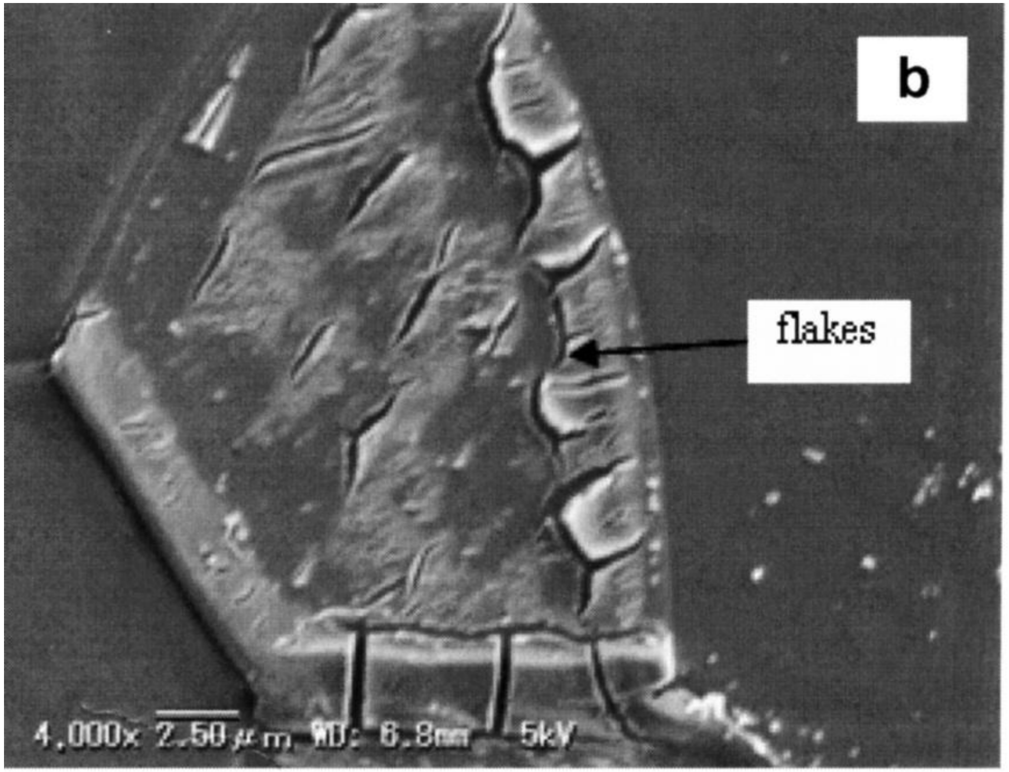



Рис. Blisters on the W surface irradiated in deuterium plasma
E =90 eV, Φ = 3,4 · $10^{25}$ $м^{-2}$, T= 550K [2]

The temperature of blister appearance is specific for cooper irradiated with moderate fluxes. (T=480-520 K)

2. Ohno 2007

# Tungsten : FLAKING

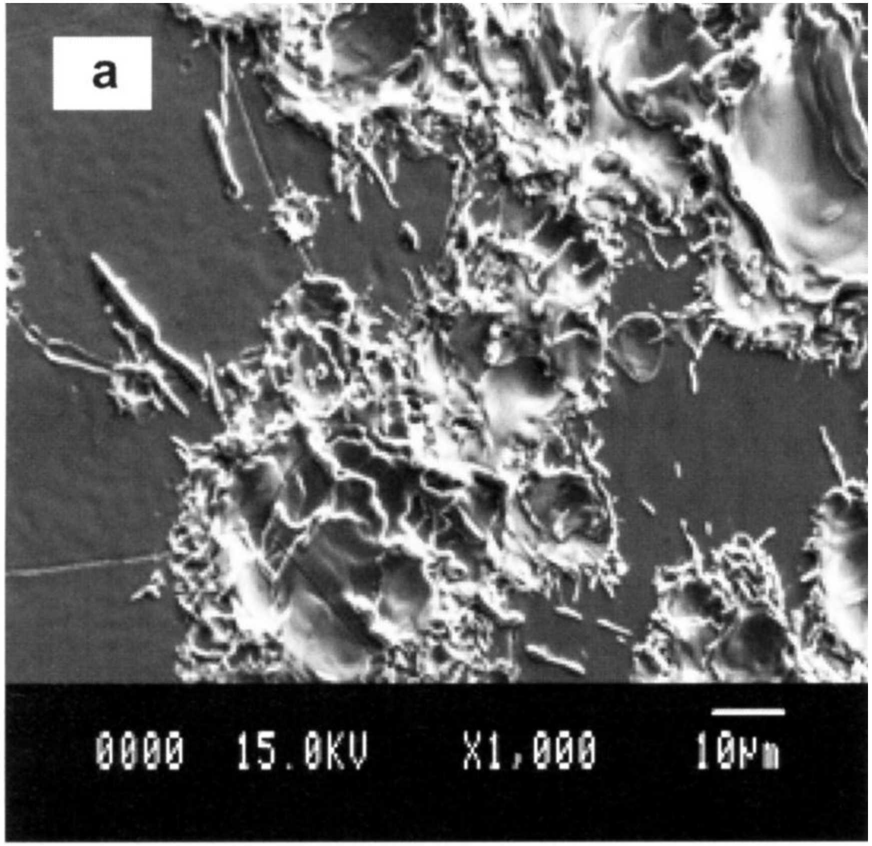



The temperature of flaking is specific for cooper irradiated with moderate fluxes (T=618 K)

Fig.1. Flakes on the W surface irradiated by high-flux ($10^{22} m^{-2}s$), high-fluence (up to $10^{27} m^{-2}$) and low-energy (38 eV) deuterium plasma at the temperature T= 618 K [1].

[1] W.M.Snu, M Nakamichi, V.KH. Alimov, G.-N. Luo, K. Isobe, T. Yamanishi et al.
J. of Nucl. Mater. 390-391(2009)1071.

# Tungsten : MELTING OF FLAKS



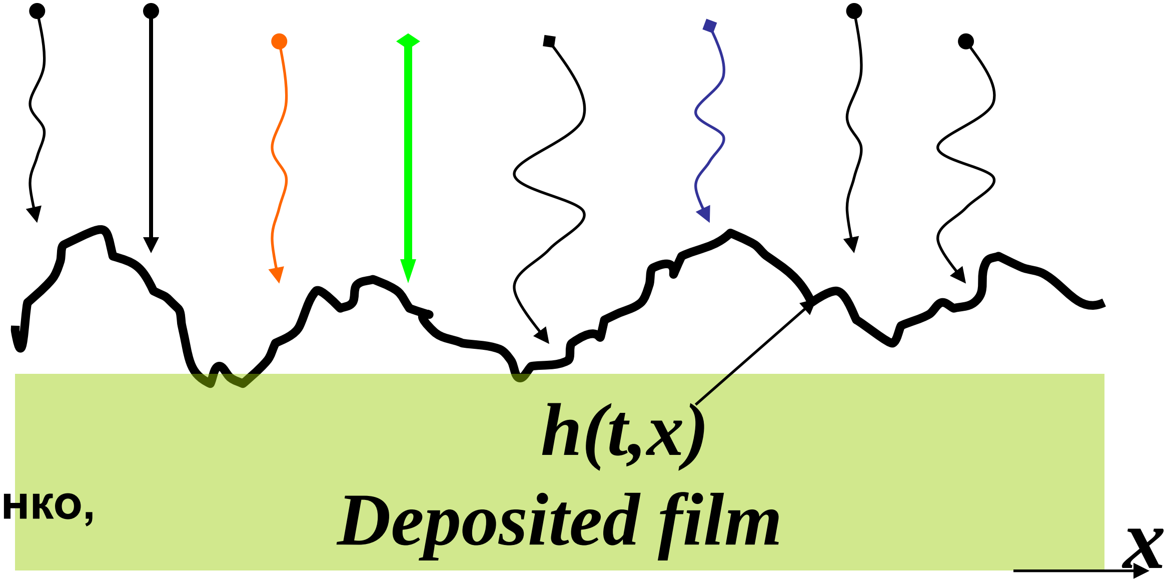

The temperature of flake melting
Is estimated to be (T=1300-1400 K).
It is the temperature range of cooper melting point

Local melting of W surface irradiated by high-flux ($10^{22}m^{-2}s$), high-fluence (up to $10^{27}m^{-2}$) and low-energy (38 eV) deuterium plasma.
Surface temperature is 618 K. Estimated temperature of flake melting is 1300-1400 K

[1] W.M.Snu, M Nakamichi, V.KH. Alimov, G.-N. Luo, K. Isobe, T. Yamanishi et al.
J. of Nucl. Mater. 390-391(2009)1071.

Surface morphology is very sensitive to impinging fluxes fluctuations = turbulence

Fractal structures can arise at a surface




Будаев, Химченко, ЖЭТФ 2007, ВАНТ 2008
Предсказание роста фрактальных структур из W И Be в ИТЭР

Deposition can be enhanced two orders of magnitude!

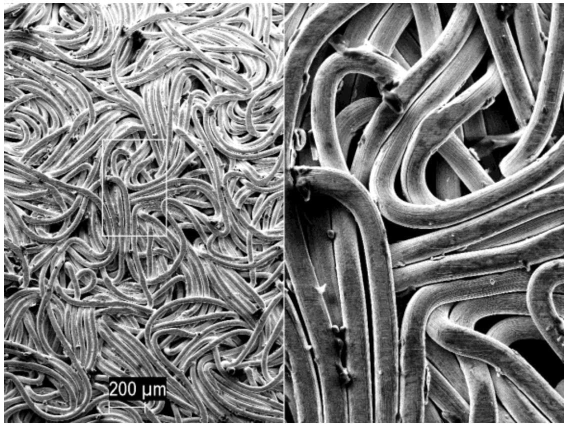
# PWI Ignitor features

In general close to Alcator C-Mod and FTU, but:

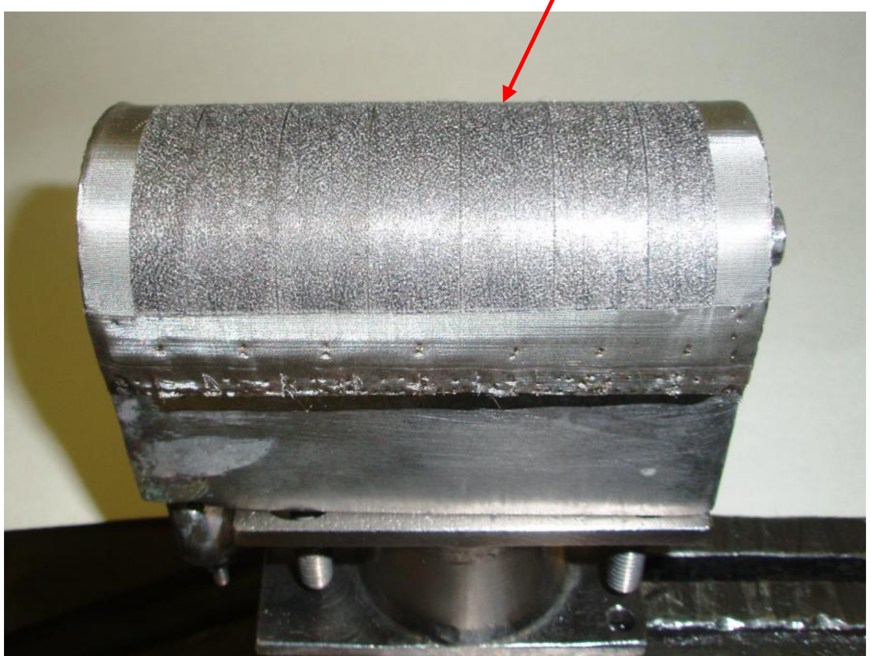
- Higher ion temperature
- Admixture of He ash
- Presence of tritium and other isotopes (H, $^3$He)
- Low $Z_{eff}$
- Neutron irradiation of first wall during the pulse and as a result long term secondary radioactivity
- Fusion reaction shut down
- High density regimes without divertor

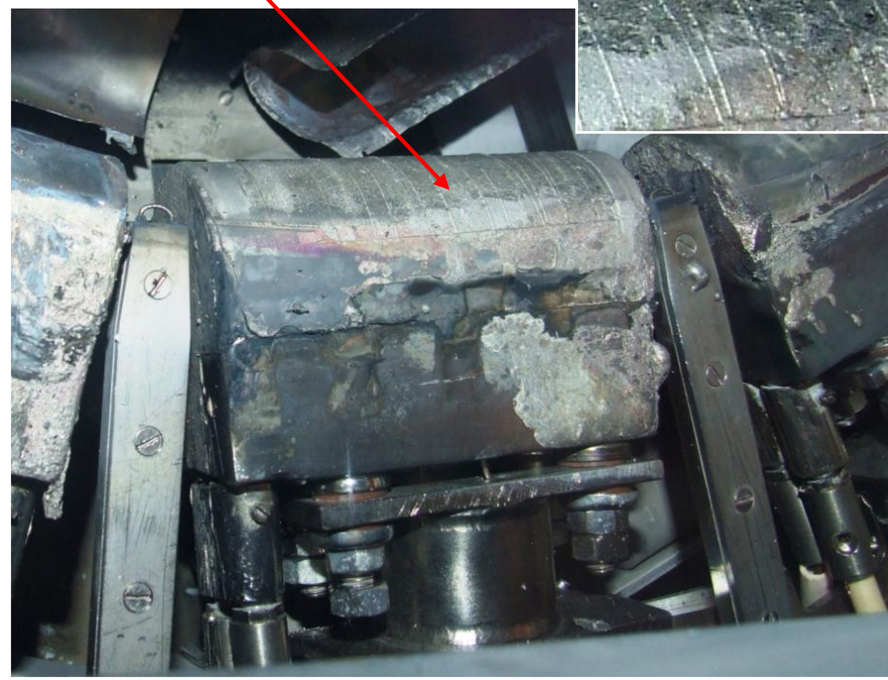
PWI control is necessary!

# On the base of W felt one element of FTU lithium limiter has been modified and tested




**Tungsten felt CPS**



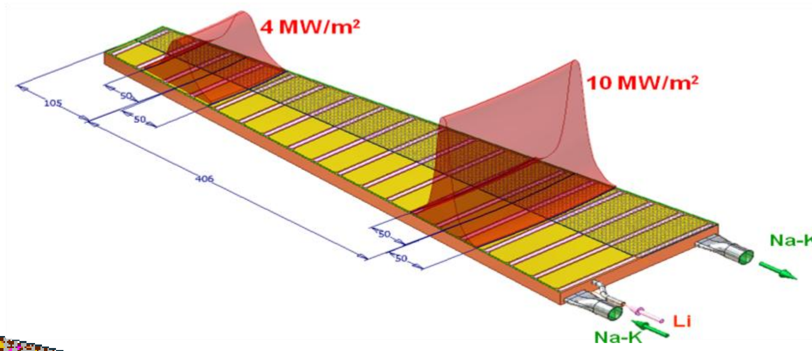


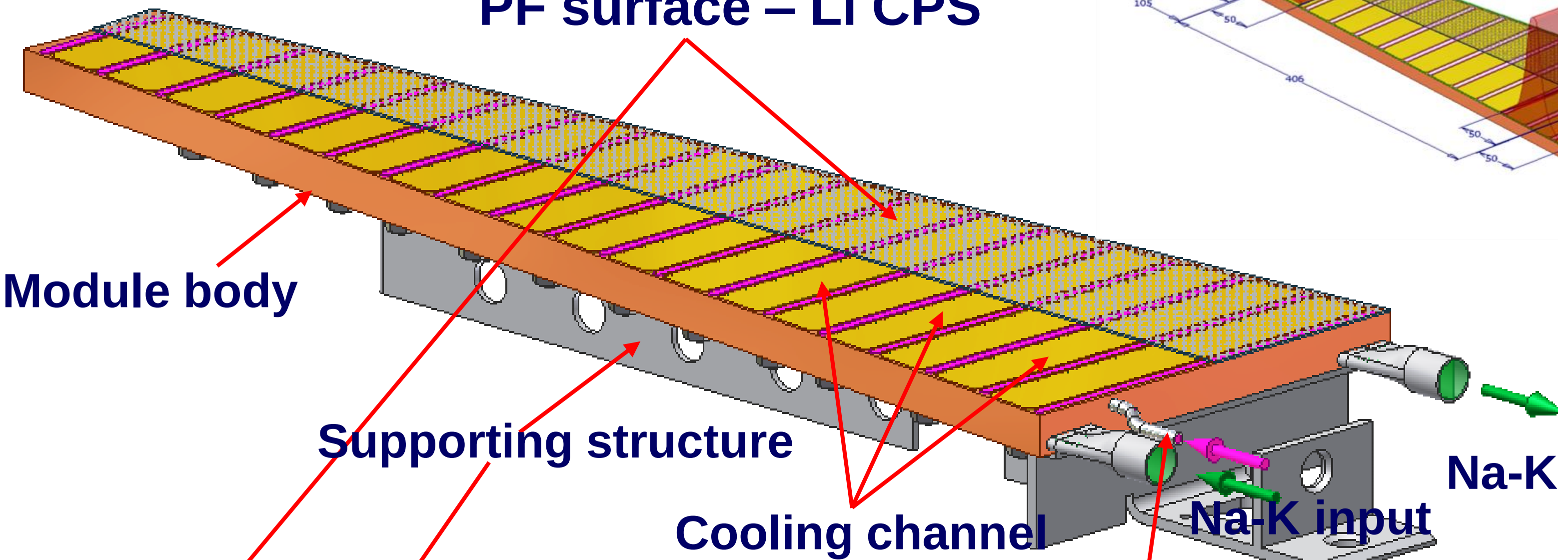
**No surface deformation and damage of CPS structure after FTU test**

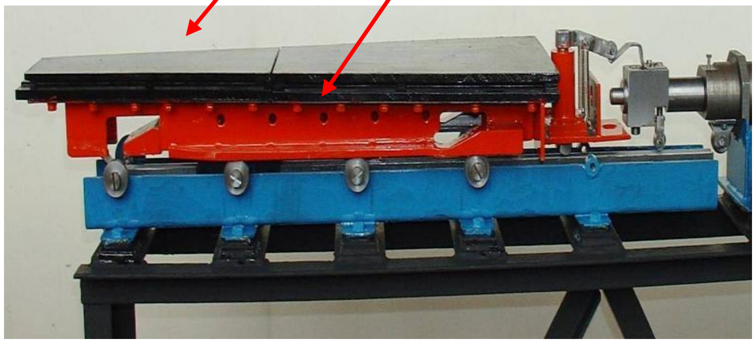
*By I.Lyublinsky et al* Red Star , Rosatom

# Module of lithium divertor will be placed on supporting structure of KTM

Joint project: I.Lyublinsky et al,FSUE «Red Star», S.Mirnov,TRINITI, I.Tajibaeva IAE Kazakhstan
G. Mazzitelli, ENEA RC Frascati, I. Agostini, ENEA RC Brasimone, Italy



**The module with renewed lithium surface will be able to operate under specific heat loads of from 2 to 10 MW/m2, in quasi-stationary mode duration up to ~5 sec.**

# Parameters of $B_4C$. SPUTTERING

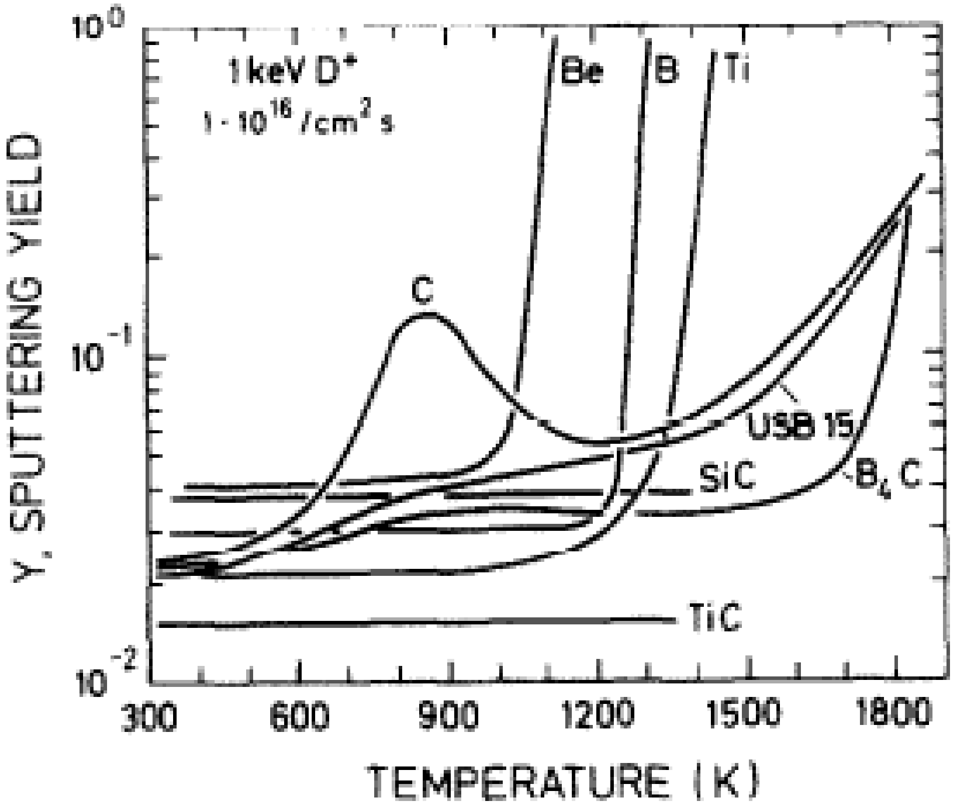



Fig.3. Temperature dependence of the erosion yield due to 1 keV D+ sputtering and evaporation for various materials [6].


[6] J.Roth, J.Nucl.Mater. 176&177 (1990) 132-141

# Parameters of $B_4C$. HYDROGEN RETENTION

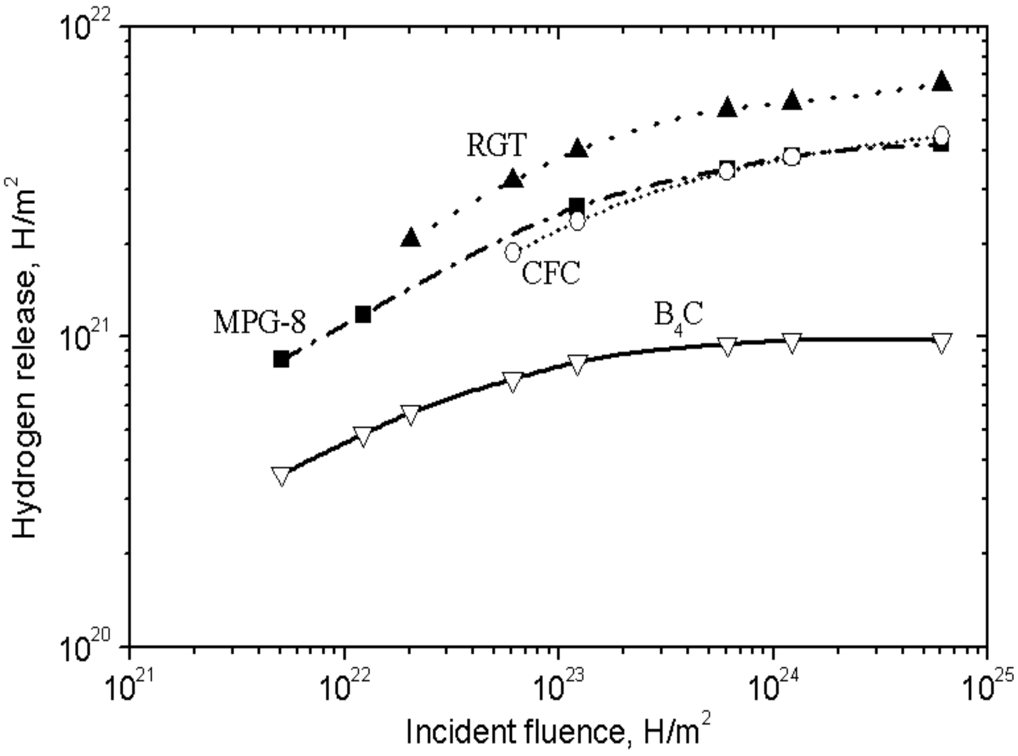



Fig.4. Fluence dependence of hydrogen retention for different materials ($H^+$, $E_i$=100 eV, j=5.610$^{19}$ m$^{-2}$) [7].


[7] L. Begrambekov, O. Buzhinsky, A. Gordeev, E. Miljaeva, P. Leikin, P. Shigin. Physica scripta, N108 (2004), p.72-75.

# Parameters of $B_4C$. In situ coating deposition

| | |
|---|---|
| **Deposition conditions** | **Plasma providing total dissociation of the molecules of initial substance** |
| Devices used for $B_4C$ deposition | Tokamak T-11M, PISCES-B ($T_e$~40 eV, $n_e$~2·10$^{17}$m$^{-3}$, electron flux~2·10$^{17}$m$^{-2}$c$^{-1}$) [9]. |
| Initial substance | Non-toxic, non-explosive, and non-hazardous carborane ($C_2B_{10}H_{12}$) |
| Deposition rate | ≈ 30 nm/s (1µm/min) in PISCES-B discharge |

[9] Buzhinskij O.I., Otroschenko V.G., Whyte D.G. et al. J. Nucl. Mater., 313—316 (2003) 214.

# Advantages of $B_4C$ coating

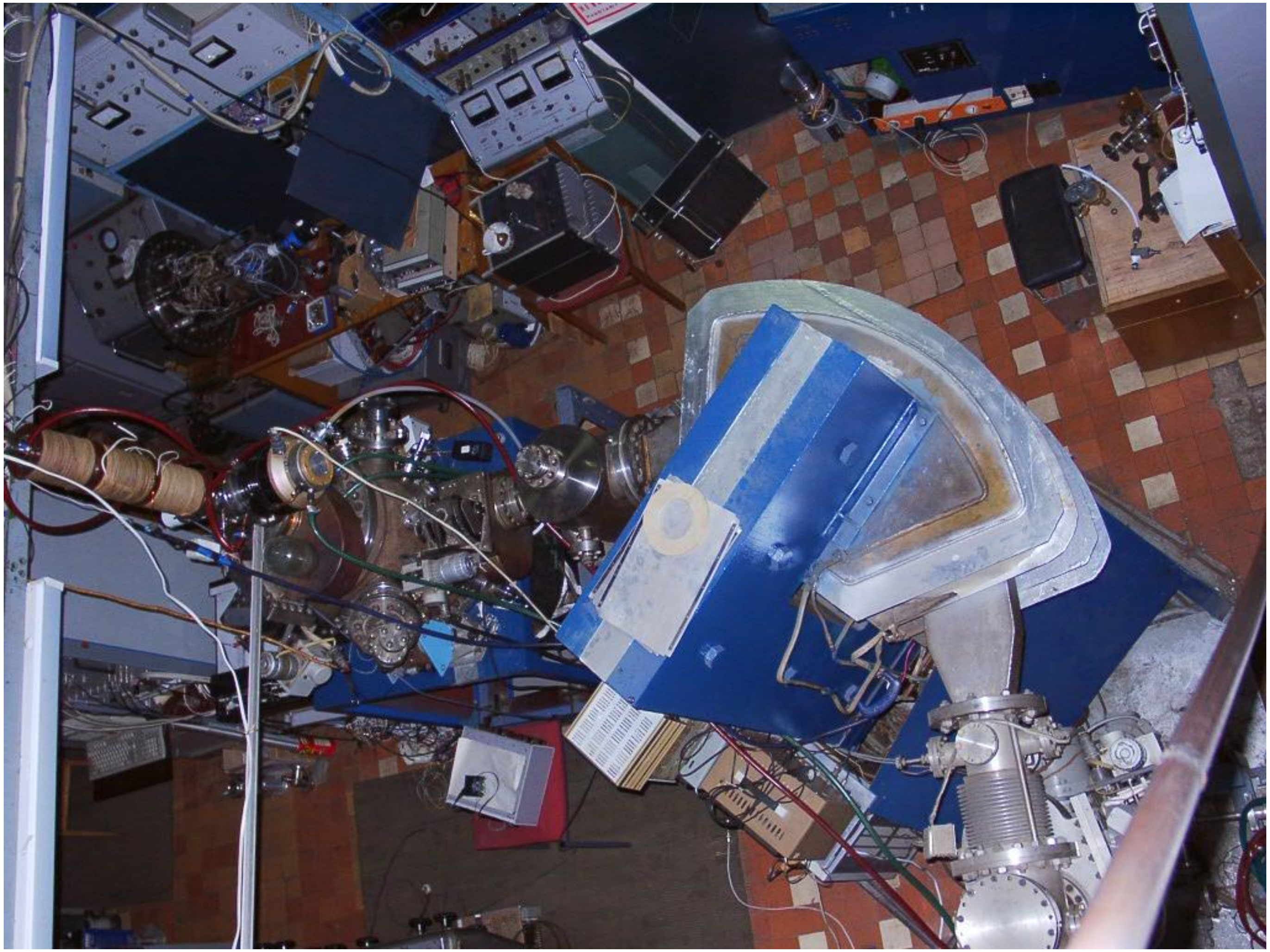
- $B_4C$ coating will provide low rate erosion of divertor tile surfaces.
- $B_4C$ coating will prevent erosion of the tiles as well as penetration of tritium into and trapping in the bulk of the tiles.
- $B_4C$ coating can be deposited and renewed during regular tokamak discharges
- $B_4C$ coating can withstand high thermal fluxes (13.0 $M \cdot W \cdot m^{-2}$)
- Erosion of $B_4C$ coating will lead to deposition of easily outgases and easily removed H/C/B films.
- Expanded investigation of deposition and behavior of $B_4C$ coating in tokamak conditions is needed

# PWI activity at PPhD of MEPhI

- Investigations on ions and plasma interactions with materials for controlled fusion devices started in our University more than 40 years ago due to proposal of Kurchatov Centre and Rosatom.
- For these years many original experimental facilities, devices, methods, codes were developed at MEPhI.
- 7 Dr Sci, ~30 Candidates and ~100 Masters of science in this field were trained in MEPhI and work now in Russia and other countries.

# Experimental facilities (1)

***Two- beam ion mass monochromator***



**$E_1$ =1-40 keV**

**$1<M/Z<100$,**

**$\Delta E/E$ =0.003**

**$P_{res}= 10^{-6}$ Pa**

**$E_2$ =0.05-5keV**

**$1<M/Z<40$,**

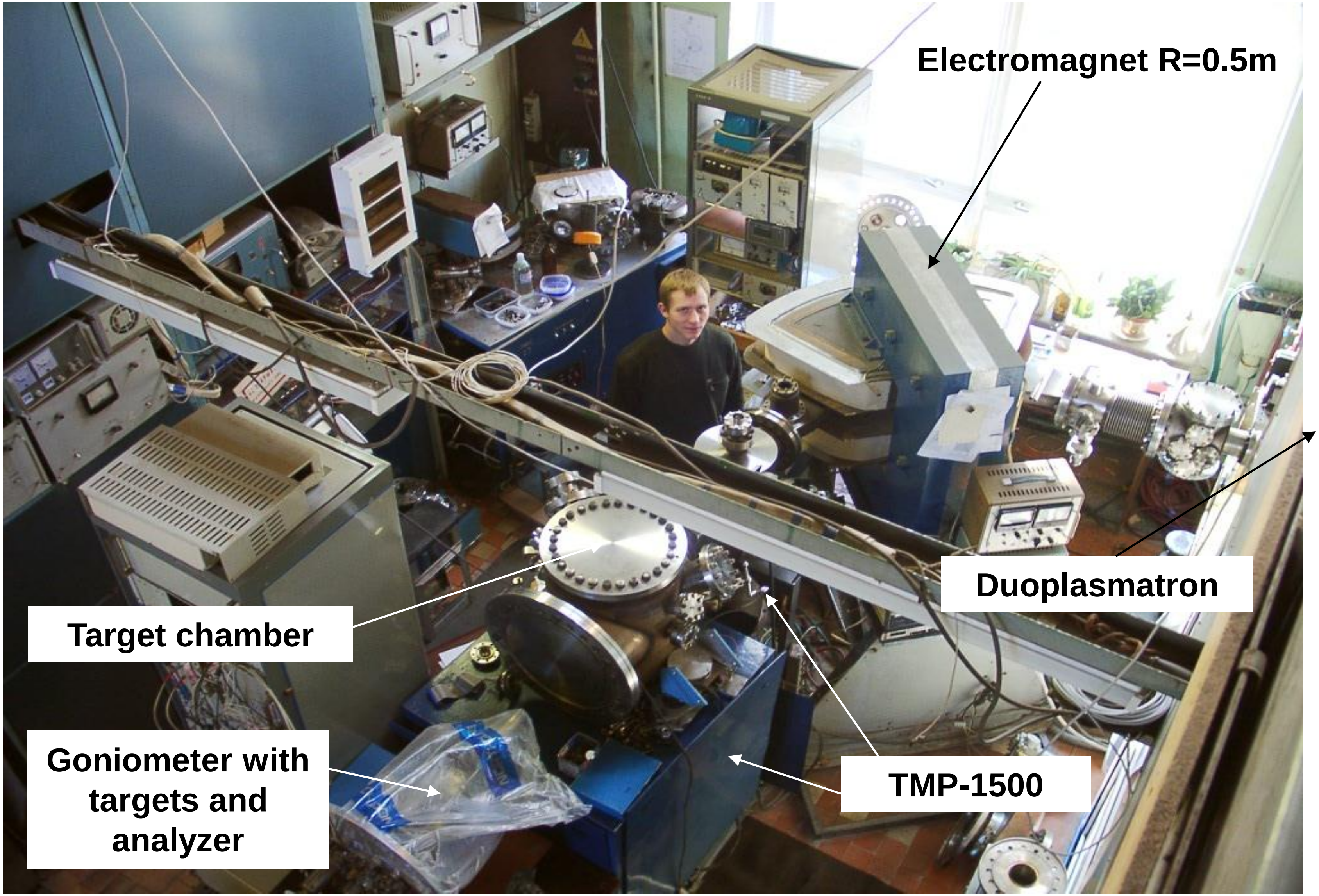
**$\Delta E/E$ =0.03**

**Measurements:**

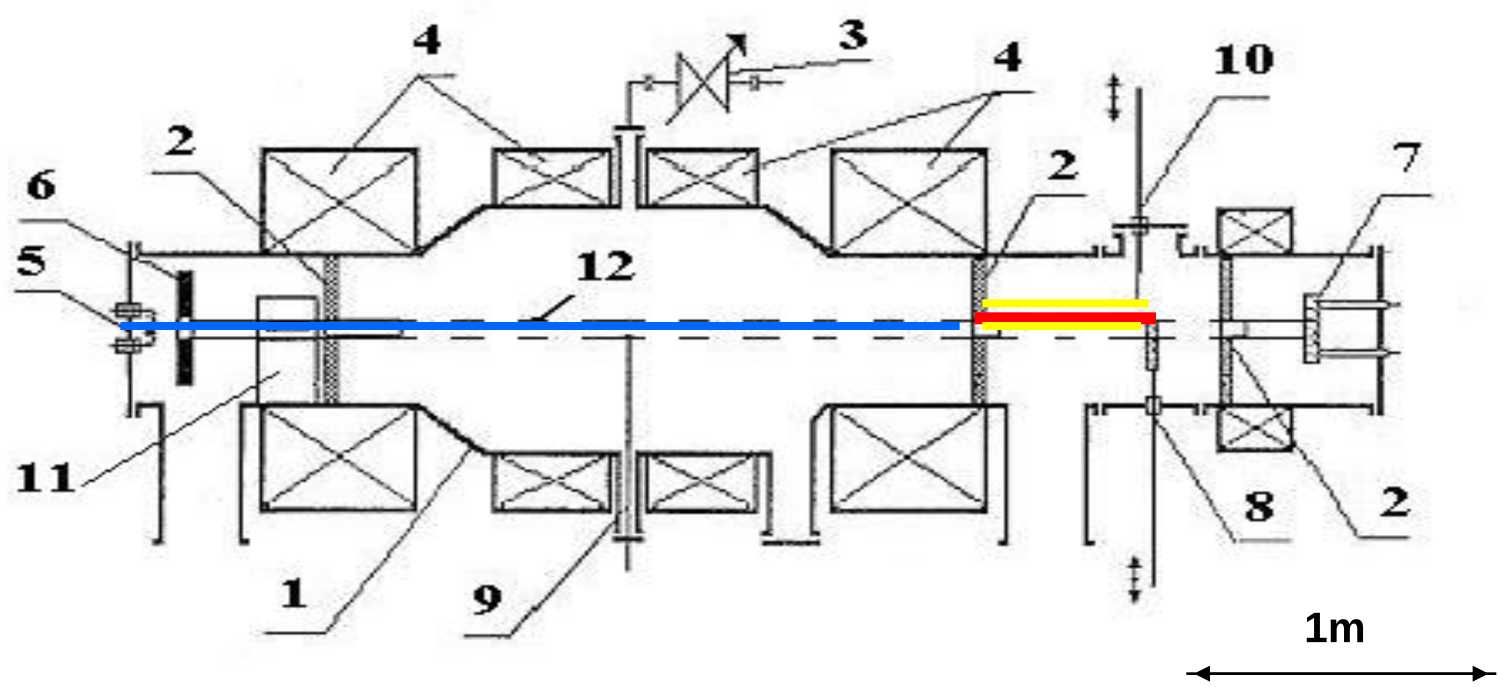
**energy, angular, charge resolved distributions of reflected and emitted particles.**

# The main ion channel

**of *two- beam ion mass monochromator***

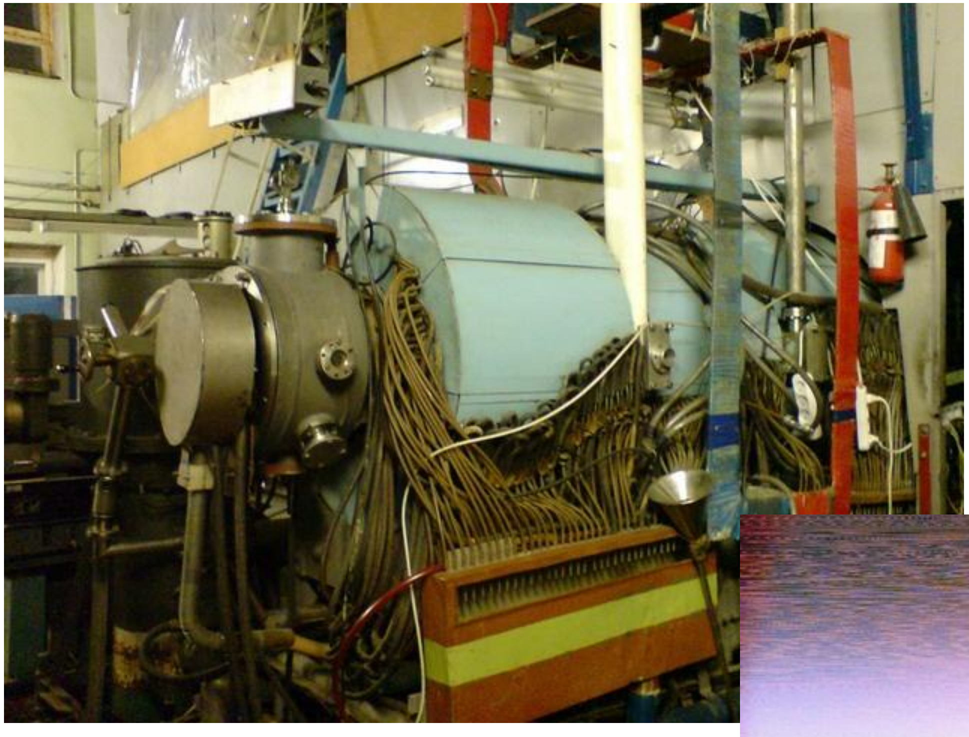


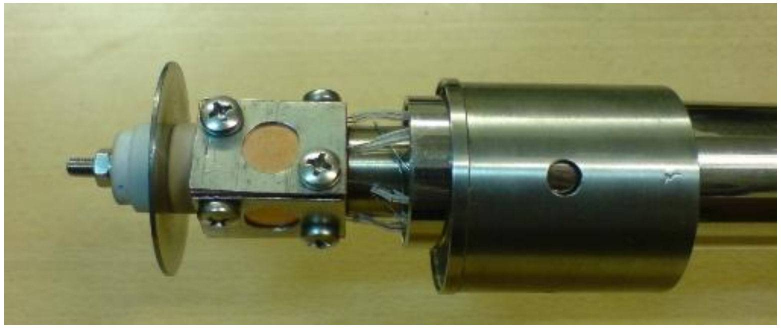
# Linear simulator with PB discharge (1)

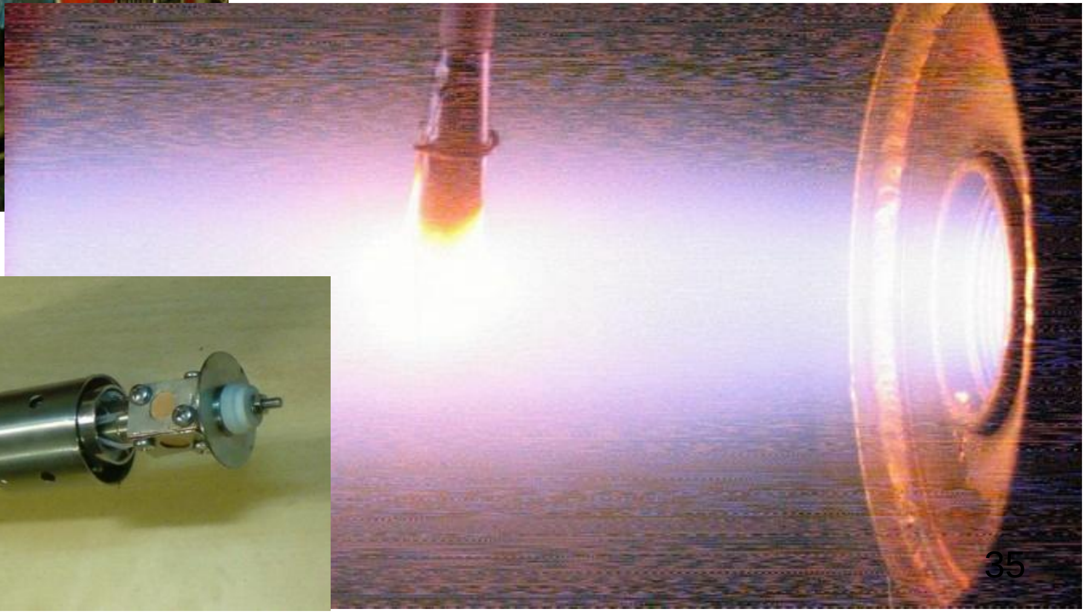



**1– vacuum chamber, 2– diaphragms, 3– gas feeding, 4– magnetic coils, 5– cathode, 6– anode, 7– collector, 8– loading lock №1, 9– Lengmuir probe, 10– loading lock №2,11– build- in mass analyzer, 12– plasma column.**

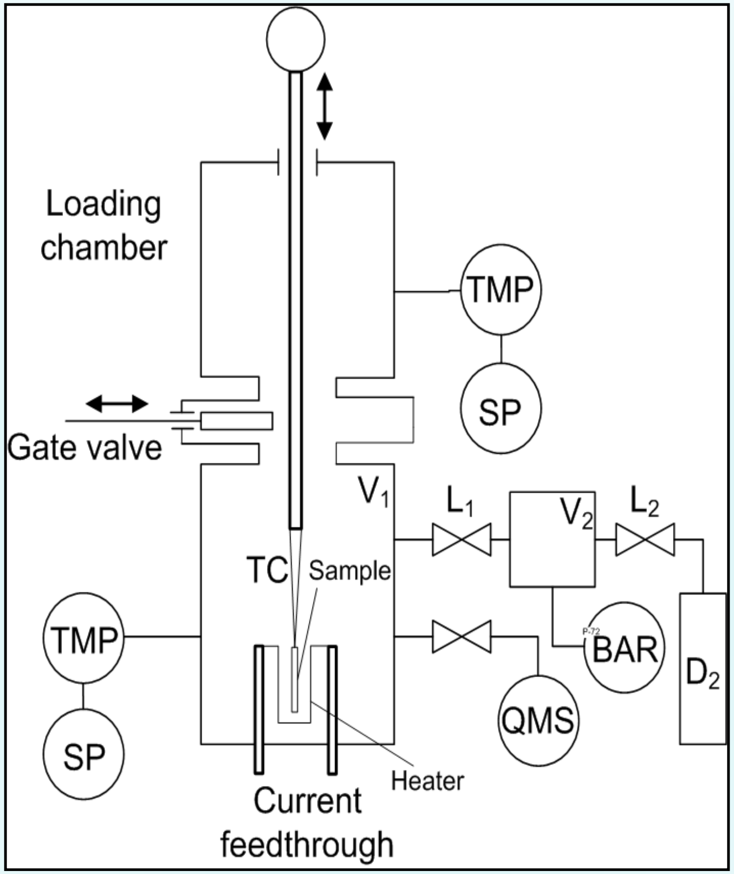
# NEW – high flux irradiation in PBD



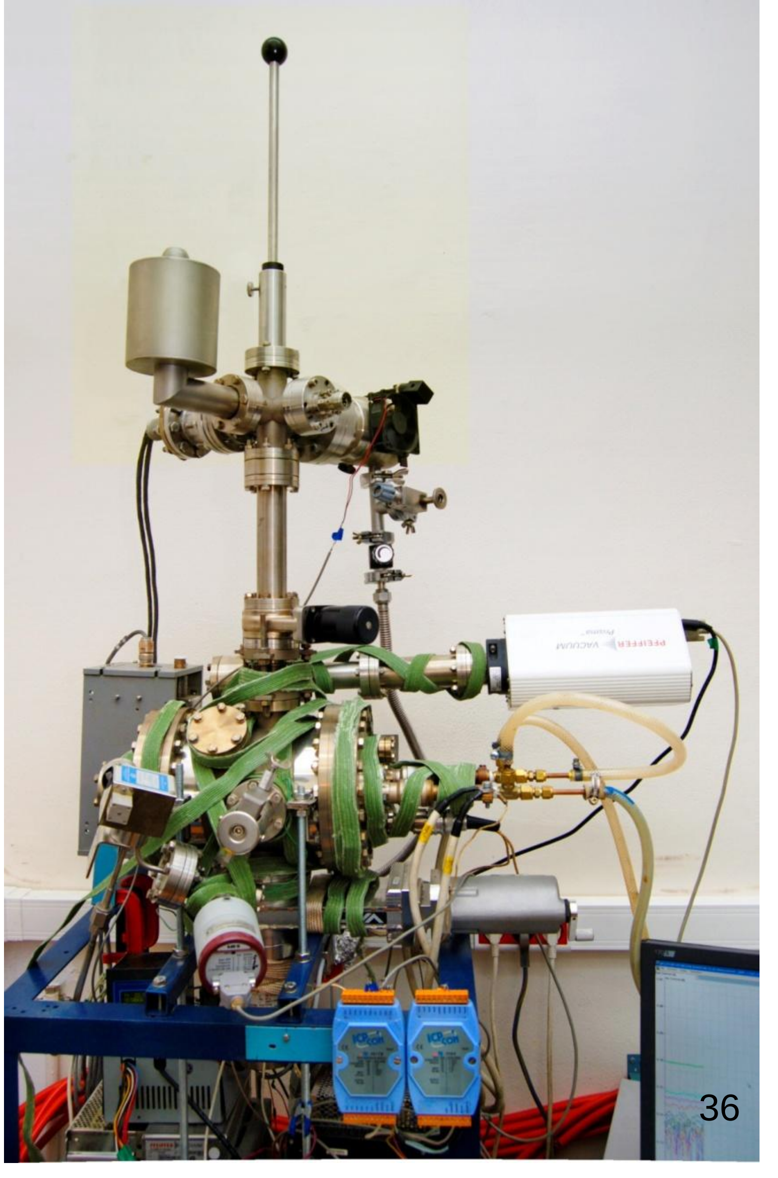




# New – UHV TDS stand




The residual gas pressure: $10^{-9}$ mbar
Linear heating ramp : 2 K/s
Maximum temperature: 1600-1750 K



# New MEPhI device for sample irradiation/implantation in plasma and for TDS analysis

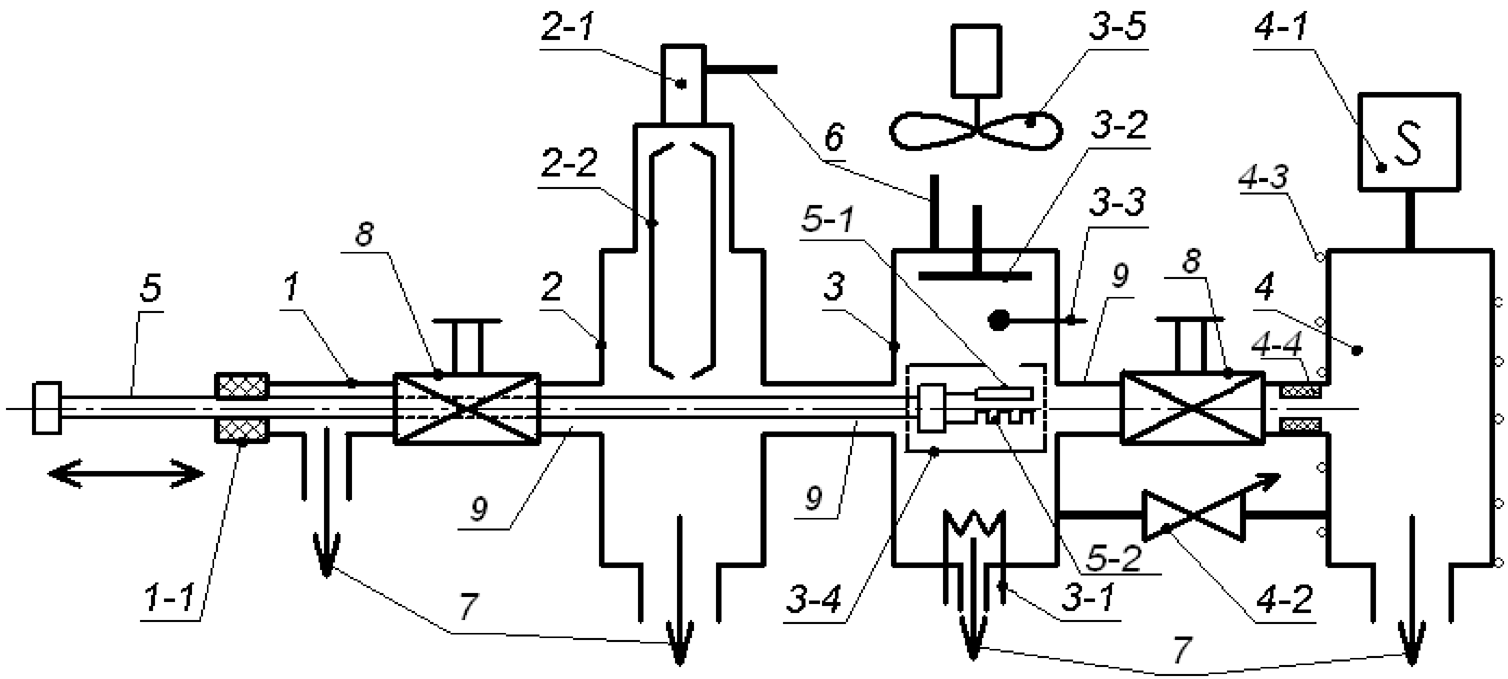



**Fig. 3. 1. The scheme of thermal desorptional stand TDS-2**

1. **Lock chamber.**
   1.1. Feed through of movable sample manipulator.
2. **Ion gun chamber**
   2.1. Ion gun
   2.2. Retarding and focusing system

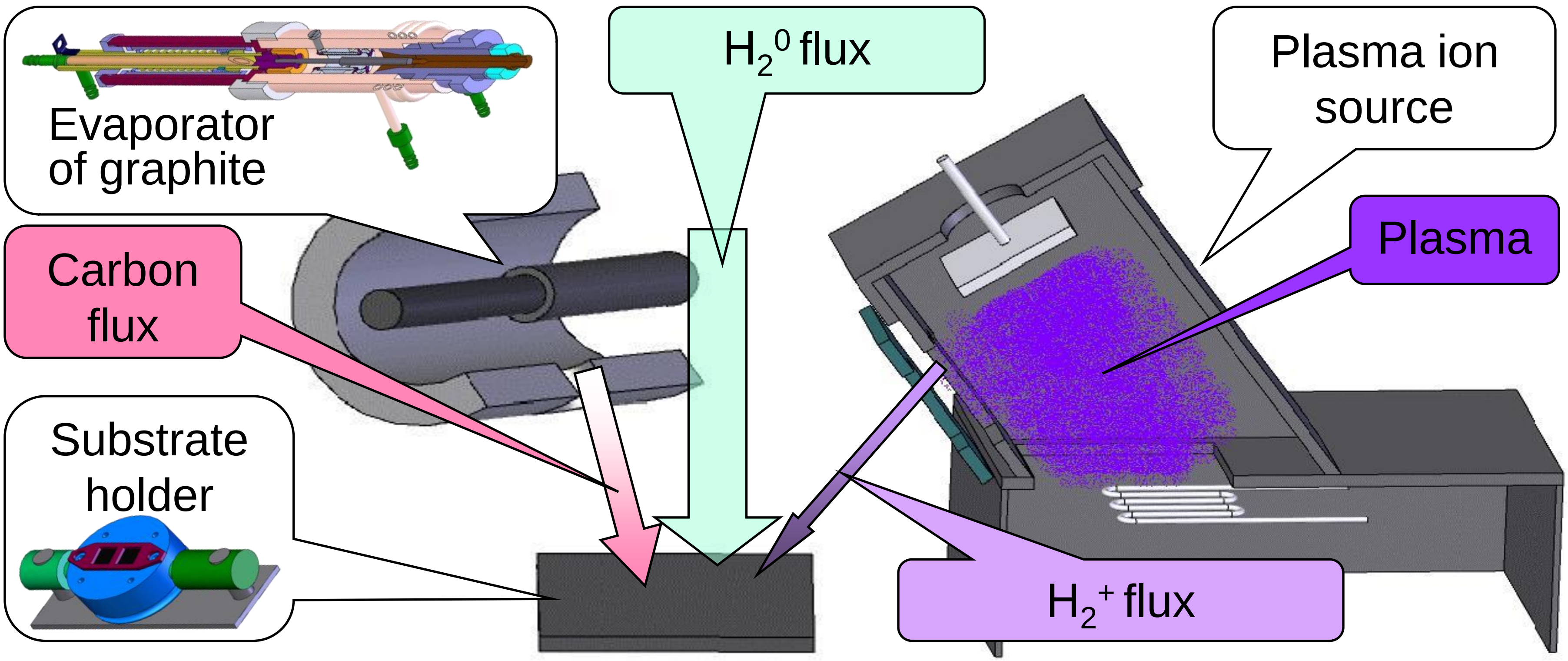
3. **Plasma chamber**
   3-1. Cathode
   3-2. Anode
   3-3. Probe
   3-4. Screen
   3-5. Air cooling system

4. **TDS chamber**
   4-1. Quadruple mass-spectrometer
   4-2. Plasma composition analysis line
   4-3. TDS chamber water cooling
   4-4. Sample manipulator sealing

5. **Sample manipulator**
   5-1. Sample
   5-2. Sample heater

6. Working gas inlets
7. Pumping
8. Gate valves
9. Pipe branches

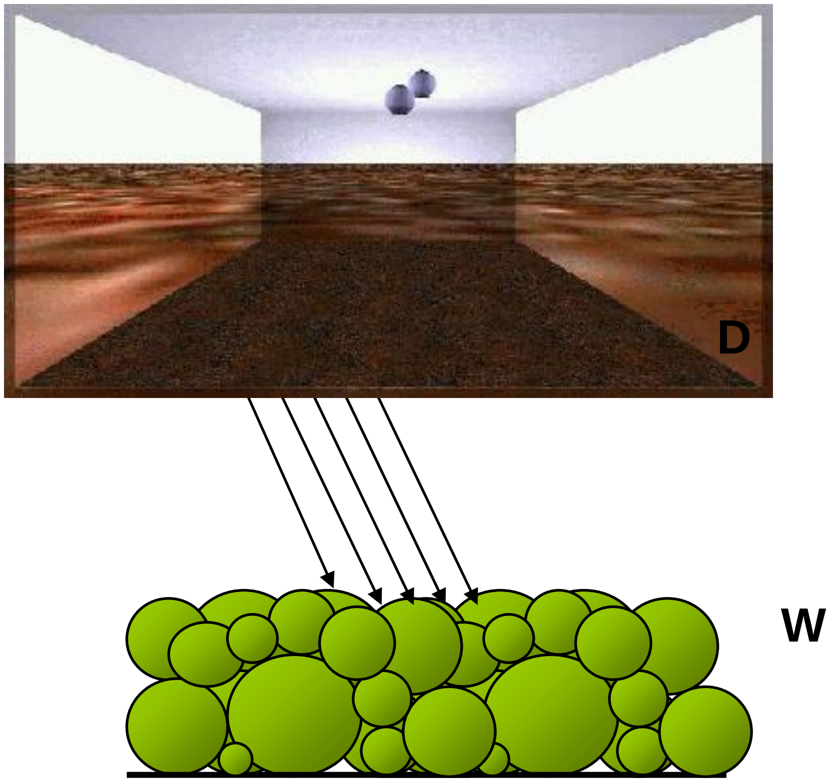
# Device for films deposition in hydrogen atmosphere and under plasma irradiation

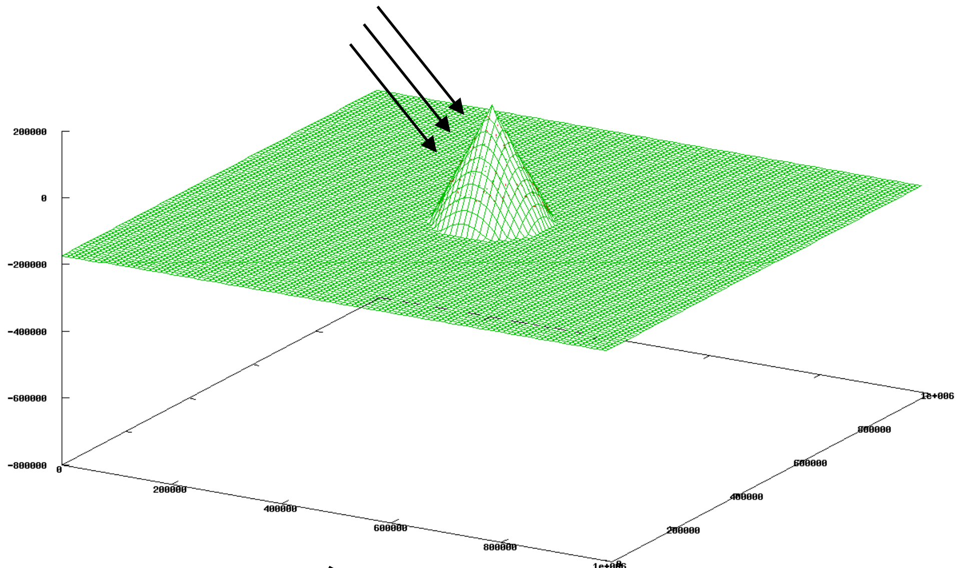



| | | |
|---|---|---|
| Deposition rate | | 0.5 – 0.01 nm/s |
| Pressure of the deposition atmosphere | $H_2$ | $1 \cdot 10^{-3} - 2.5$ Pa |
| | $H_2O$ | $5 \cdot 10^{-4} - 1 \cdot 10^{-2}$ Pa |
| Concurrent plasma irradiation | Ion energy | ≈ 10 – 1000 eV/atom |
| | Ion flux | $1 \cdot 10^{15} - 2 \cdot 10^{16}$ $cm^{-2} \cdot s^{-1}$ |
| Substrate temperature | | 300 – 1500 K |
| Thickness | | 20 – 2000 nm |

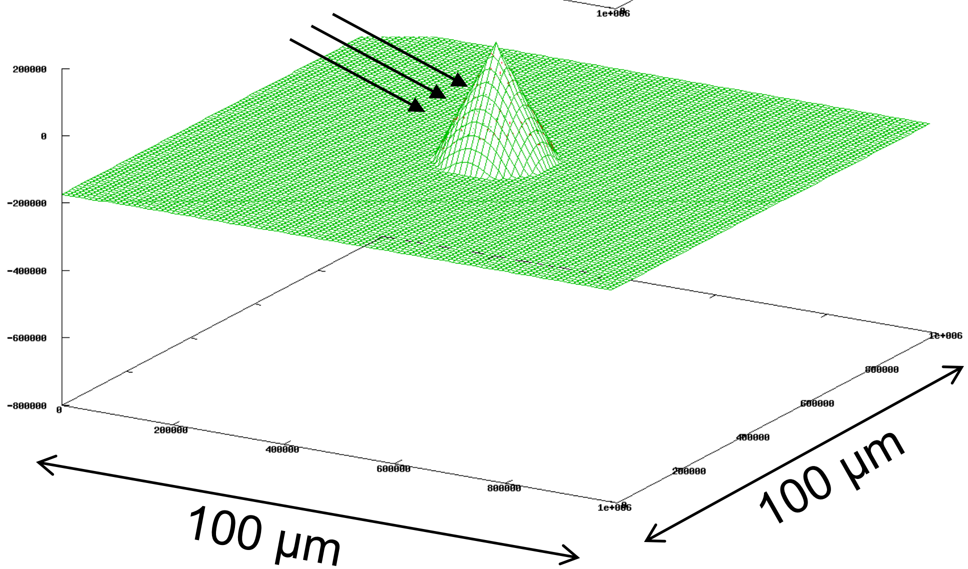
# Computer simulations of ion interaction with solids

## Simulation of plasma sprayed tungsten used as plasma faced material in German tokamak ASDEX






Sputtering of a cone with α=60° by Ar (1keV) ions at the incidence angle  30°




Sputtering of a cone with α=60° by Ar (1keV) ions at the incidence angle  630°
к нормали

# Conclusion

- PWI demonstrates very complicated and multifaceted physics
- PWI influences fusion device performance very much
- In spite of high plasma density and low temperature at the edge in Ignitor, PWI in this machine should be of keen interest and care.
- Relevant to Ignitor wall conditioning should be developed.

Thank you for your attention!