Lecture in 4th International school on Physics of Plasma
Surface interactions, July 14 , 2020, MEPHI, Moscow

Plasma materials interactions in a fusion reactor

T. Tanabe

Osaka City University and National Research Nuclear University “MEPHI”

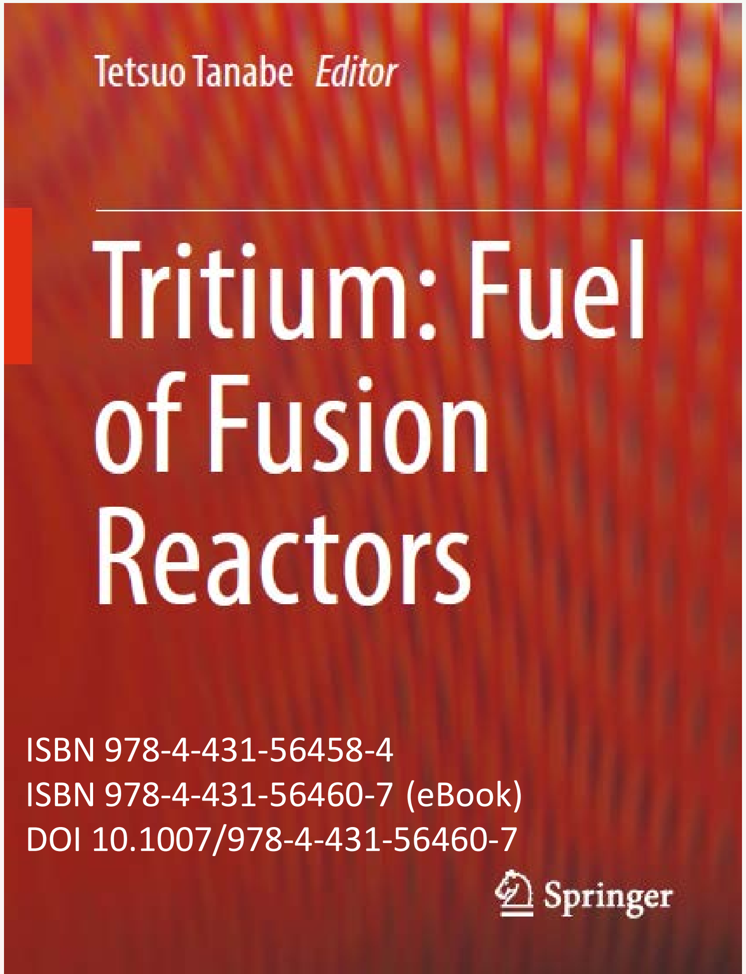
This talk focuses to explain “Why PMI concerns?”

“Modifications of plasma facing materials by huge power load
and their influence on plasma performance”

Important issues related to PMI not covered here are

- Conversion of neutron energy to electricity, and T breeding
- Tritium safety and economic efficiency

Please refer textbooks



Tetsuo Tanabe *Editor*

Tritium: Fuel of Fusion Reactors

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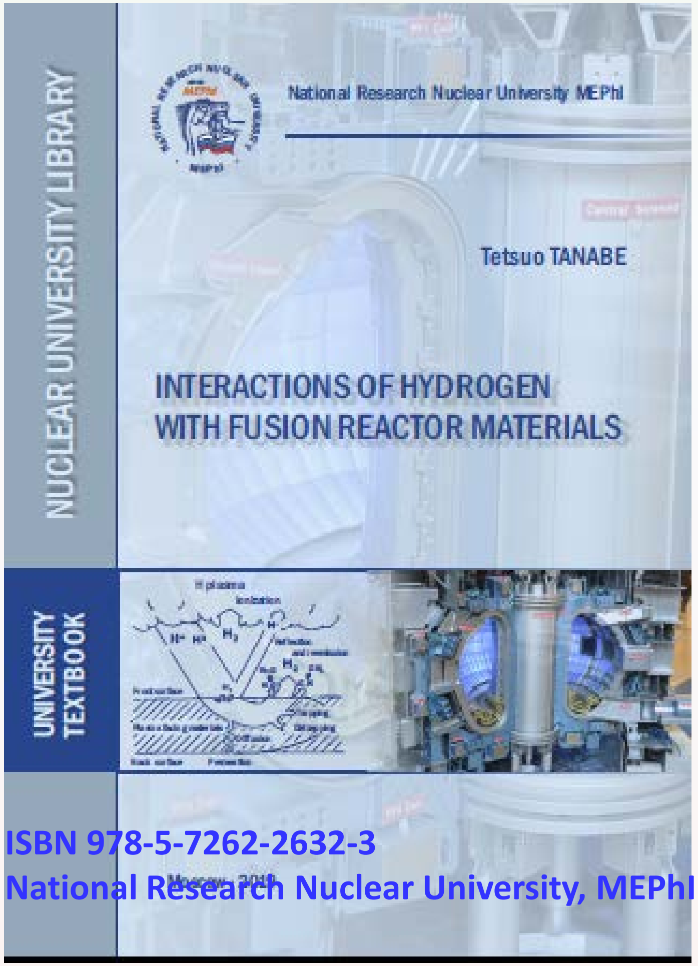

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Tetsuo TANABE

INTERACTIONS OF HYDROGEN
WITH FUSION REACTOR MATERIALS

UNIVERSITY TEXTBOOK

ISBN 978-5-7262-2632-3
National Research Nuclear University, MEPhI

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1. Characteristics of a DT reactor as an energy source

Nearly 60 years have passed after finding fusion reactions give energy.

Fission reactors are already established as energy sources.

$^{235}U + n \rightarrow$ $^{\sim 120}FP(1) + ^{\sim 110}FP(2) +$ ~ 2.4n + ~ 180MeV

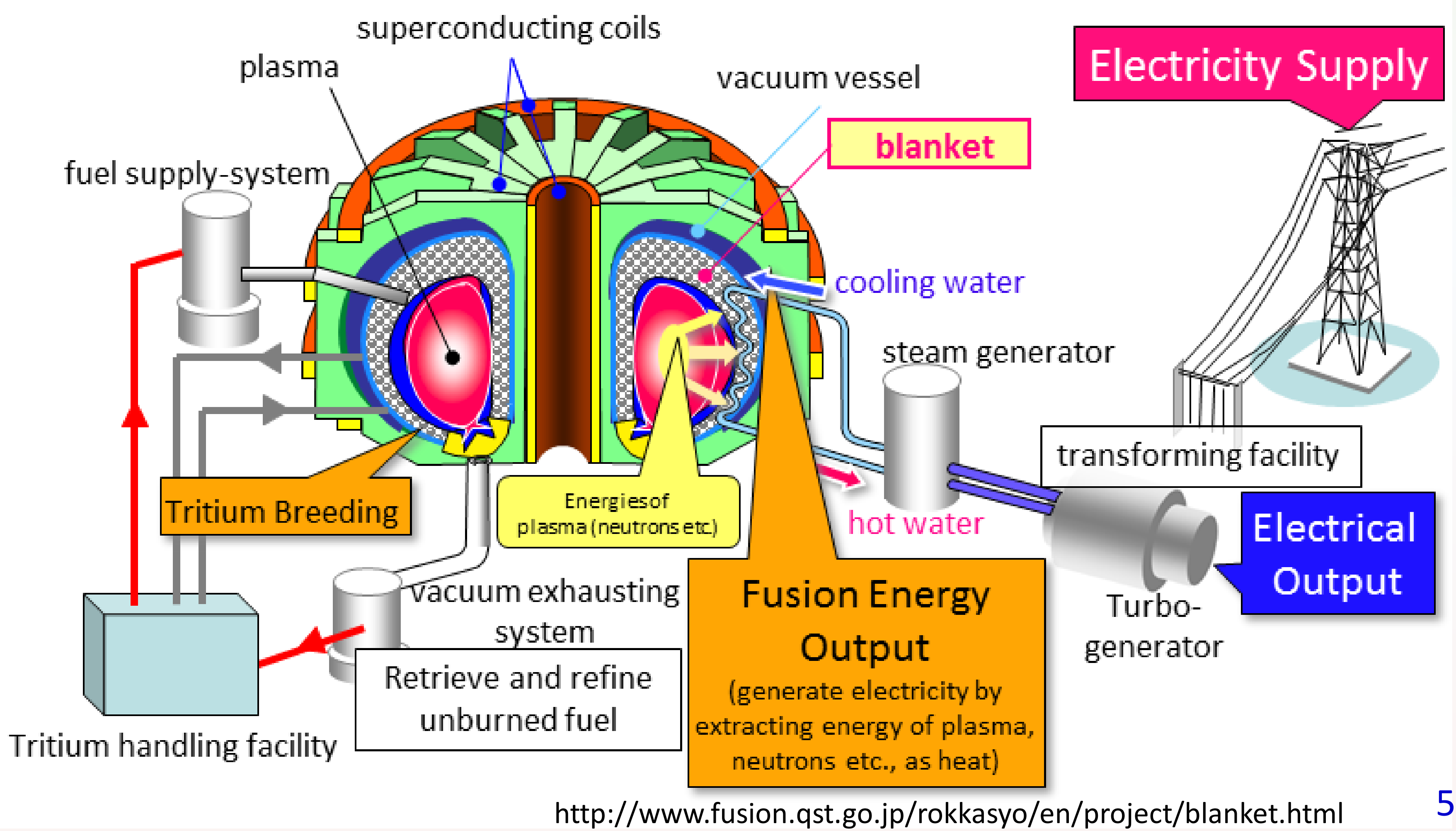
Concept of fusion reactor has been well established but not realized yet

$D + T \rightarrow$ ^{4}He (3.5MeV) + n (14.1MeV)

Plasma heating — **Power production and T breeding**

Fusion reactor system for generation of electricity

- Huge energy required to start burning
- Energy production in burning core plasma
- Energy conversion and T breeding in blanket using fusion neutrons
- Electricity generation out side of a reactor (out of scope of ITER)



http://www.fusion.qst.go.jp/rokkasyo/en/project/blanket.html

1. Characteristics of a DT reactor as an energy source

Nearly 60 years have passed after finding fusion reactions give energy.

Fission reactors are already established as energy sources.

$^{235}U + n \rightarrow\ ^{\sim 120}FP(1) + ^{\sim 110}FP(2) + \sim 2.4n + \sim 180MeV$

Concept of fusion reactor has been well established

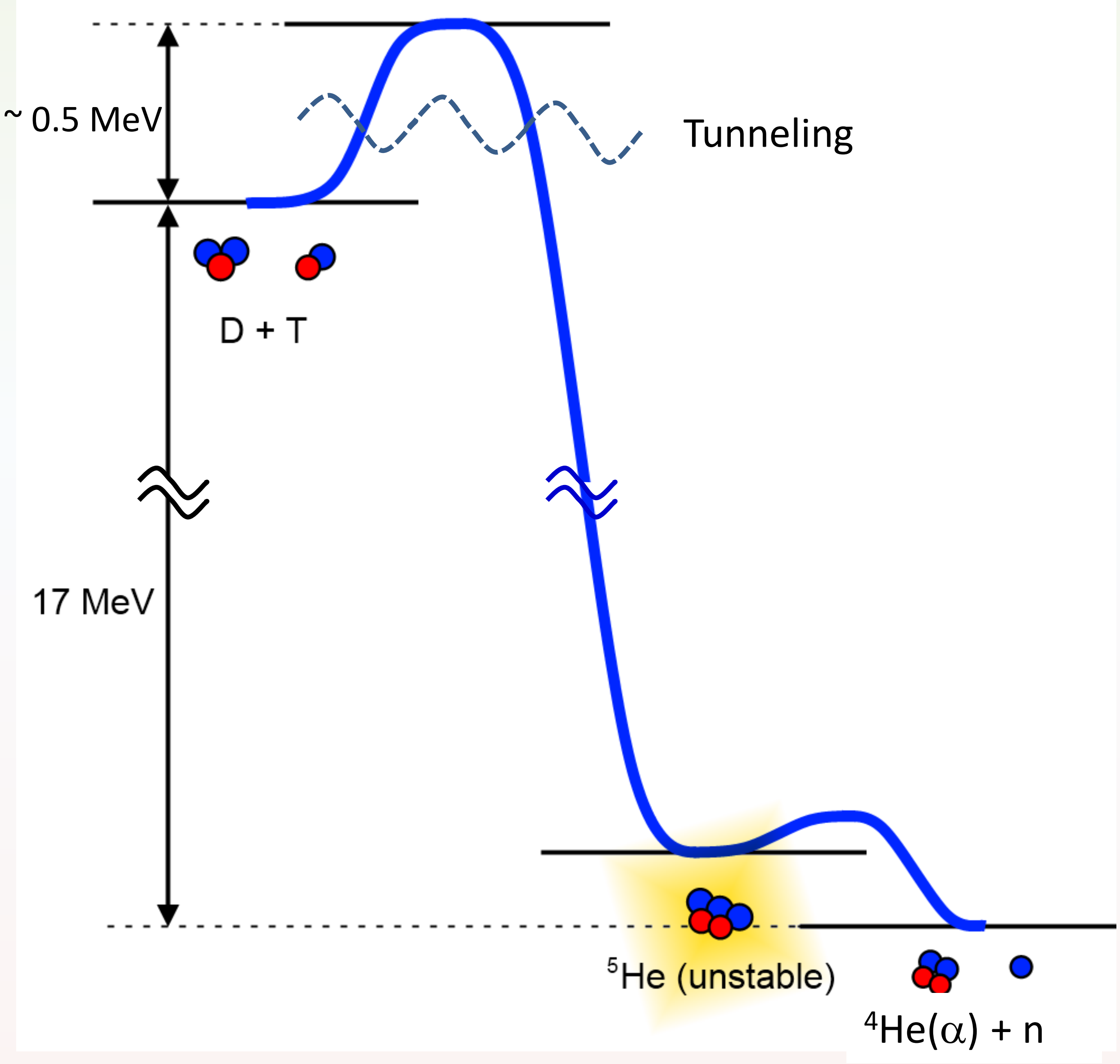
D + T → ^{4}He (3.5MeV) + n (14.1MeV)
Plasma heating   **Power production and T breeding**

Why much longer time has been required for fusion than fission?

Significant amount of energy is required to start and continue fusion reactions (to overcome Coulomb potential)

Potential energy diagram for D-T fusion reaction

Significant amount of energy is required to start and continue fusion reactions
(to overcome Coulomb potential)



1. Characteristics of a DT reactor as an energy source

$D + T \rightarrow {}^{4}He$ (3.5MeV) + n (14.1MeV)
Plasma heating Power production and T breeding

Significant amount of energy is required to start and continue fusion reactions (to overcome Coulomb potential)

Soon we will get energy gain, Q=10 in ITER.

But this is not enough for fusion to be an energy source!!.

Huge power load to plasma facing surface

Conversion of neutron energy to electricity, and T breeding

Tritium safety and economic efficiency

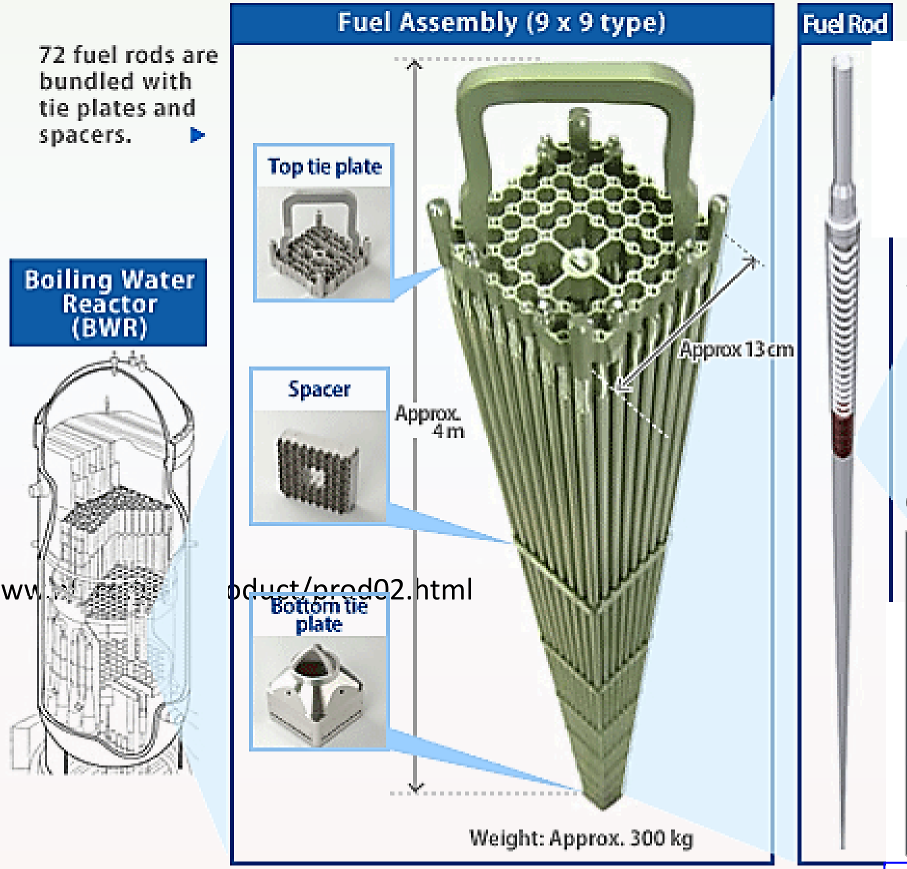
Fission reactor

In fuel rods, nuclear chain reactions occur

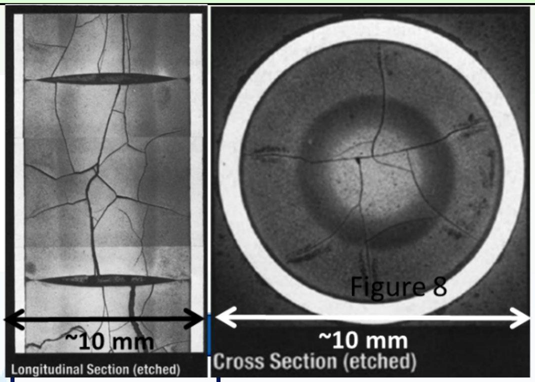
$$^{235}U + n \rightarrow {}^{\sim120}FP(1) + {}^{\sim110}FP(2) + \sim 2.4\, n + \sim 180 MeV$$

Most of released nuclear energy is carried by Fission Products and converted to heat.

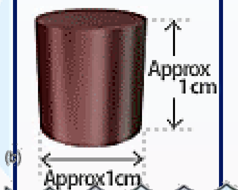
All radioactive FPs are also encapsulated.

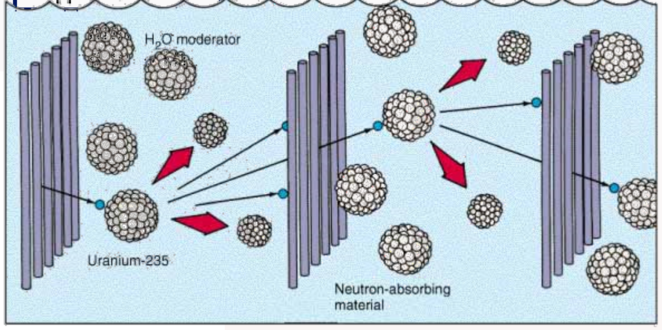


http://www.[illegible]oduct/prod02.html



H. Stehile, J. Nucl. Mater. **153** (1988) 3- 15





Energy required to make fission chain reactions is quite small compared to fusion

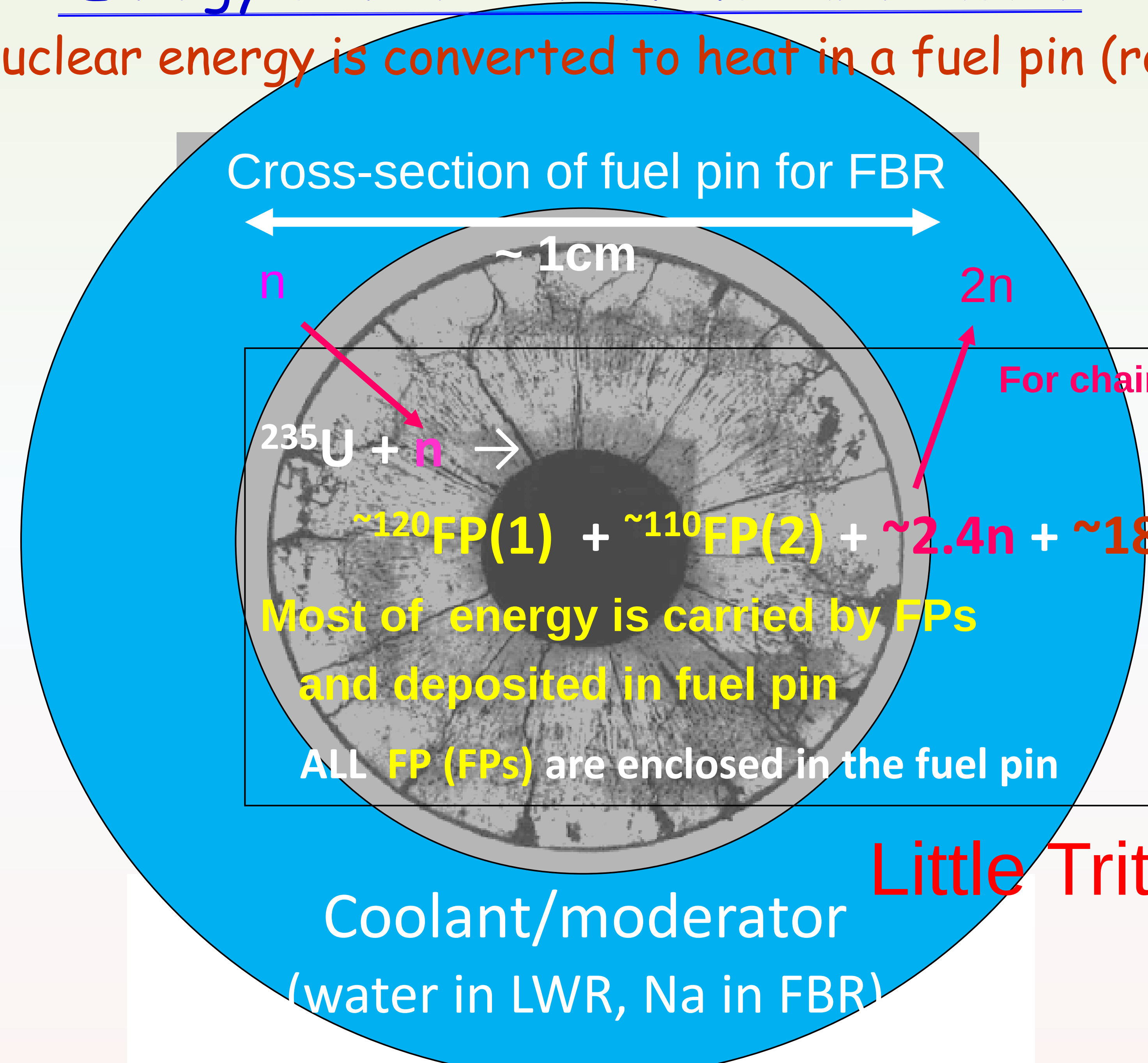
Easy burning control

Control rod (Neutron absorber) : Boron, Cadmium

Moderator : H_2O, D_2O,

Energy conversion in Nuclear reactor

Nuclear energy is converted to heat in a fuel pin (rod)



Cross-section of fuel pin for FBR

~ 1cm

n

2n

For chain reaction

$^{235}U + n \rightarrow$

$^{\sim 120}FP(1) + {}^{\sim 110}FP(2) + \sim 2.4n + \sim 180MeV$

Most of energy is carried by FPs and deposited in fuel pin

ALL FP (FPs) are enclosed in the fuel pin

Little Tritium!!

Coolant/moderator

(water in LWR, Na in FBR)

Comparison of fission and fusion as energy sources

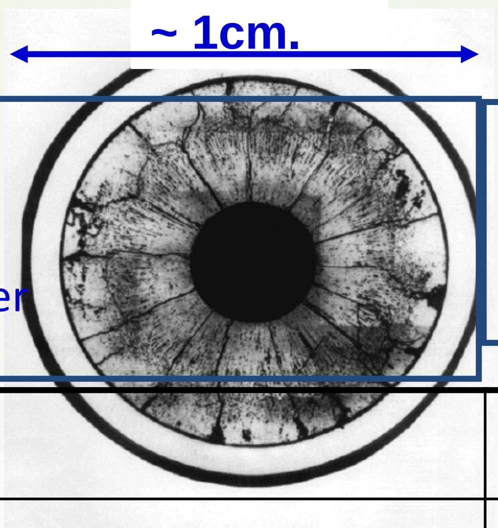


Fig. 3.9(a) Cross-section of an irradiated UO_2 fuel element, showing restructuring. ANL.

In a fission reactor, energy conversion, fuel breeding, waste-confinement in fuel pins of diameter of 1 cm

Fusion reactor is an open tritium handling system with a huge volume, requiring special care on T safety.

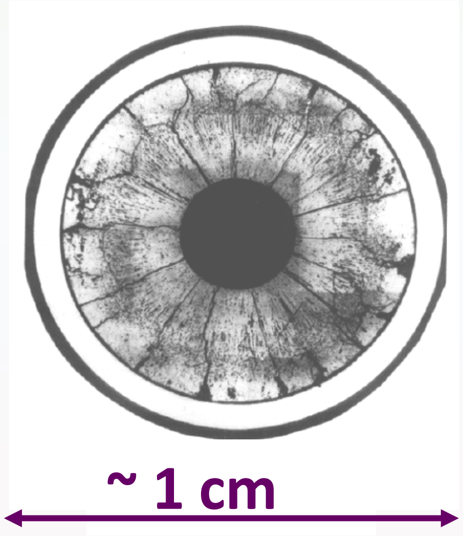
	Fission	Fusion
Energy Input	Nearly zero	Huge energy is required Poor fueling efficiency requires huge fuel throughput
Energy conversion	Energy carried by fission products (FP, heavy ions) (~170MeV) is deposited in fuel pins.	Energy carried by neutron (14MeV) must be converted in large volume of blanket system
Fuel breeding and recovery	One fission produces more than 2 neutrons, easy to keep chain reactions and to breed fuels. Fuel pins retain both FP and new fissile and spent fuels are reprocessed to remove/recover them.	To keep breeding ratio more than 1, neutron multipliers (Be, Pb) are required. Tritium breeding and energy conversion must be done simultaneously.
Nuclear Waste	Long life radioactive FPs must be handled with special care and will be reposed deeply under ground.	Waste is limited to activated structure materials, could be recycled.

Output energy density is nearly the same

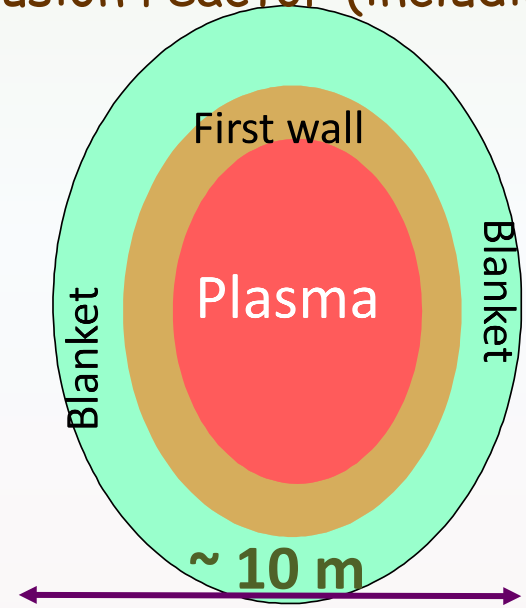
Because energy conversion to heat for electricity production requires similar total volume of a reactor

Comparison of energy density of reactors with thermal output power of a few GW)

fission reactor



fusion reactor (including blanket)

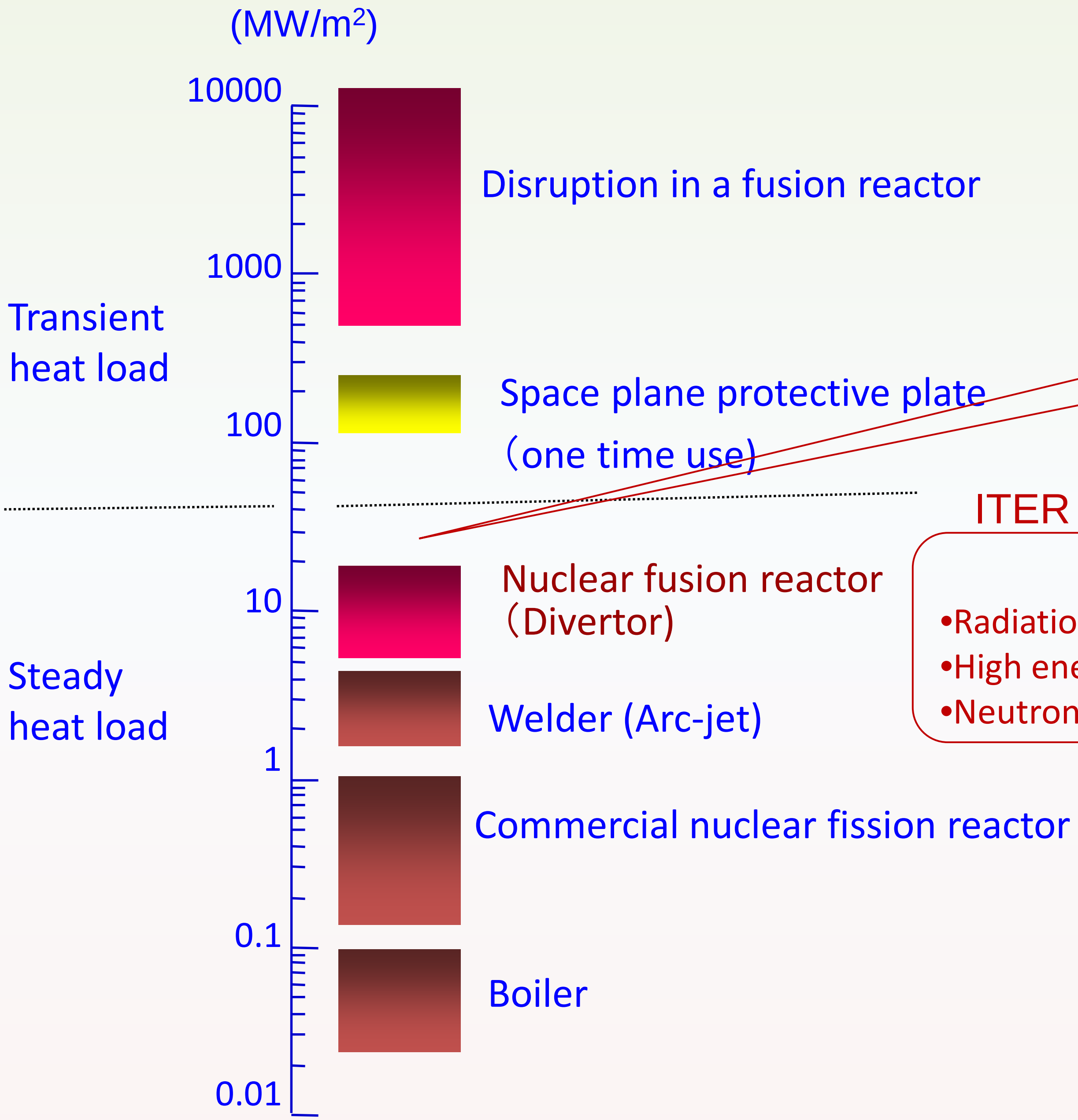


0.1 - 1 W/cm^3 for a fuel pin = **0.1- 1 MW/m^3** for first wall and blanket region

(More than 10 times higher In burning plasma)

In contrast , power loads to system walls are quite different . NEXT

2. Comparison of Surface heat load in energy systems



Transient heat load

Disruption in a fusion reactor

Space plane protective plate
（one time use)

Steady heat load

Nuclear fusion reactor
（Divertor)

Welder (Arc-jet)

Commercial nuclear fission reactor

Boiler

Melting/evaporation appears

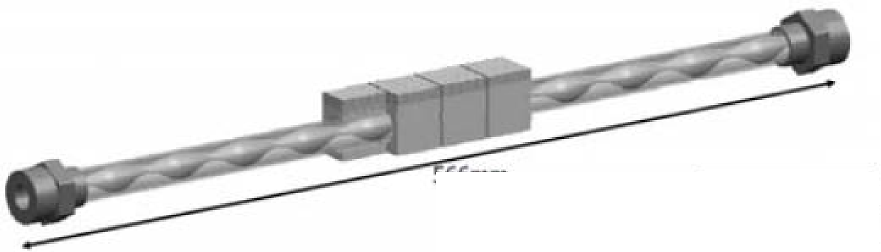
ITER divertor structure

Wall load:

- Radiation (heat) ⎫ ~ 15-20 MW/m^2
- High energy particles ⎭
- Neutrons ~2 MW/m^2

How large power can materials tolerate?

Effort to avoid melting damages

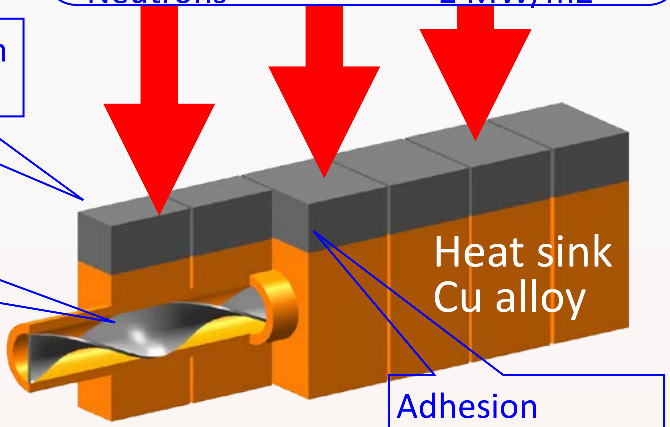


- Hot pressing condition
  - Temp.= 900 °C
  - Compression= 500 kgf
- Braze condition
  - Temp. = 850 °C
  - Braze mater.= TiCuAg

ITER divertor structure

Wall load:

- •Radiation (heat) } ~ 15-20 MW/m^2
- •High energy particles
- •Neutrons ~2 MW/m2

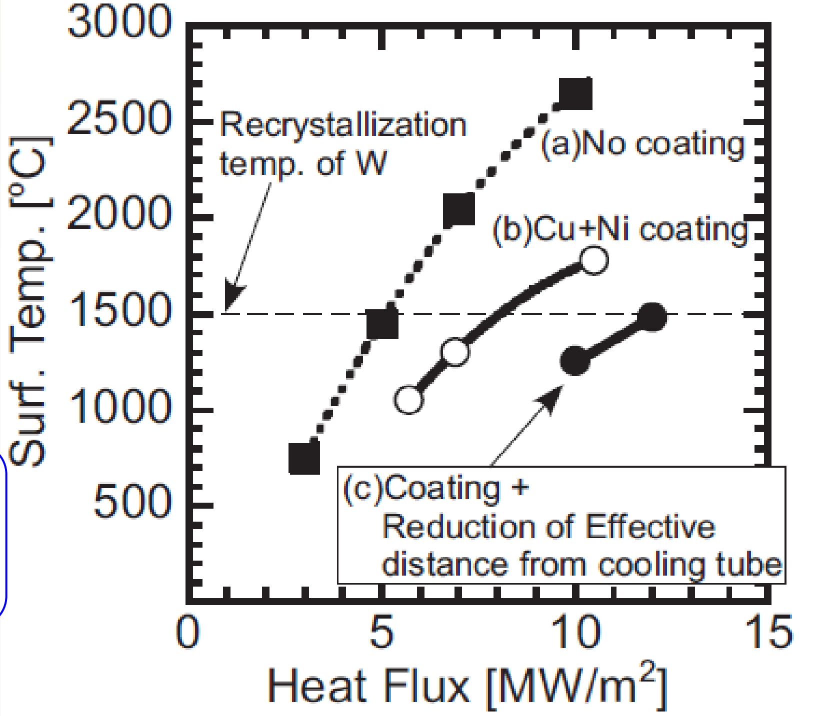


Armor tile with high TC and low erosion

Swirl tube for higher heat removal

Adhesion Brazing/Welding

Maximum tolerable heat load at steady state would be around 15MW/m^2



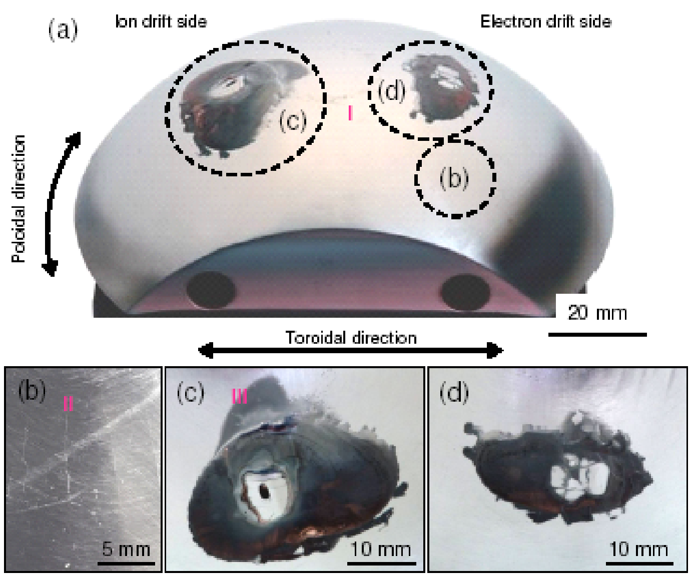
Comparison of thermal response of W-hot pressing mock-ups with different joining conditions in high heat flux experiment

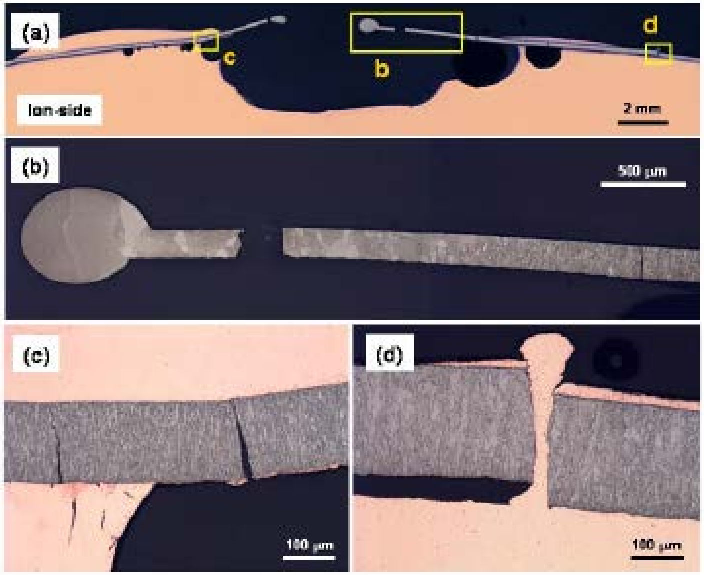
K. Ezato IAEA Conf.

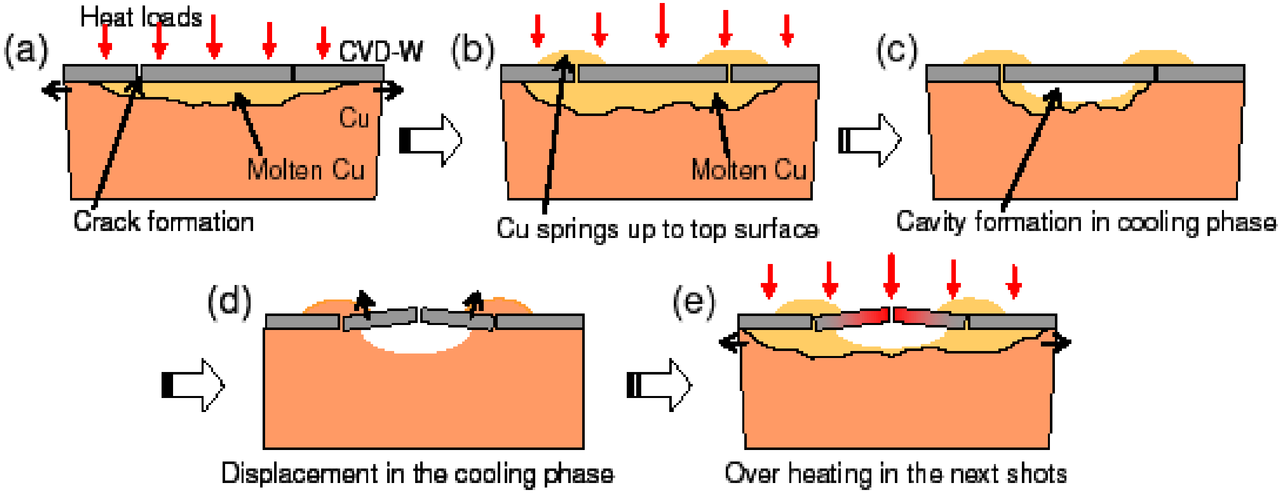
Bad experience on CVD – W on Cu !

Cu melted first, then W layer lost thermal contact and melted

TEXTOR: CVD-W on Cu Limiter







Energy conversion and power load to plasma facing surface

Simple estimation for a spherical reactor with fusion power of 3 GW

Radius (r)	Surface area $4\pi r^2$	Power flux	Volume $(4\pi r^3/3)$	Energy density
3 m	113 m^2	26.5 MW/m^2	113 m^3	26.5 MW/m^3
5 m	314 m^2	9.5 MW/m^2	523 m^3	5.7 MW/m^3

- 2/3 of output power is carried by 14 MeV neutrons and is converted to heat in blanket having large volume
- 1/3 is by particles and radiation (= 3-10 MW/m^2) and deposited to near surface layers (very dangerous for PFM)

Exercise :

Estimation of particle flux if power is loaded by fuel particles with energy of 100 eV

$1\ MW/m^2 = 1M\ J/m^2s = 10^6 \times 6.2 \times 10^{18}\ eV/m^2s = 6.2 \times 10^{24}\ eV/m^2s$

To give the power load of 1 MW/m^2 with 100eV ions

$\phi(\mathrm{ion}100eV) \ = \ = 6.2 \times 10^{24}\ eV/m^2s\ /100\ eV = \underline{6.2\mathrm{x}10^{22}/m^2s}$

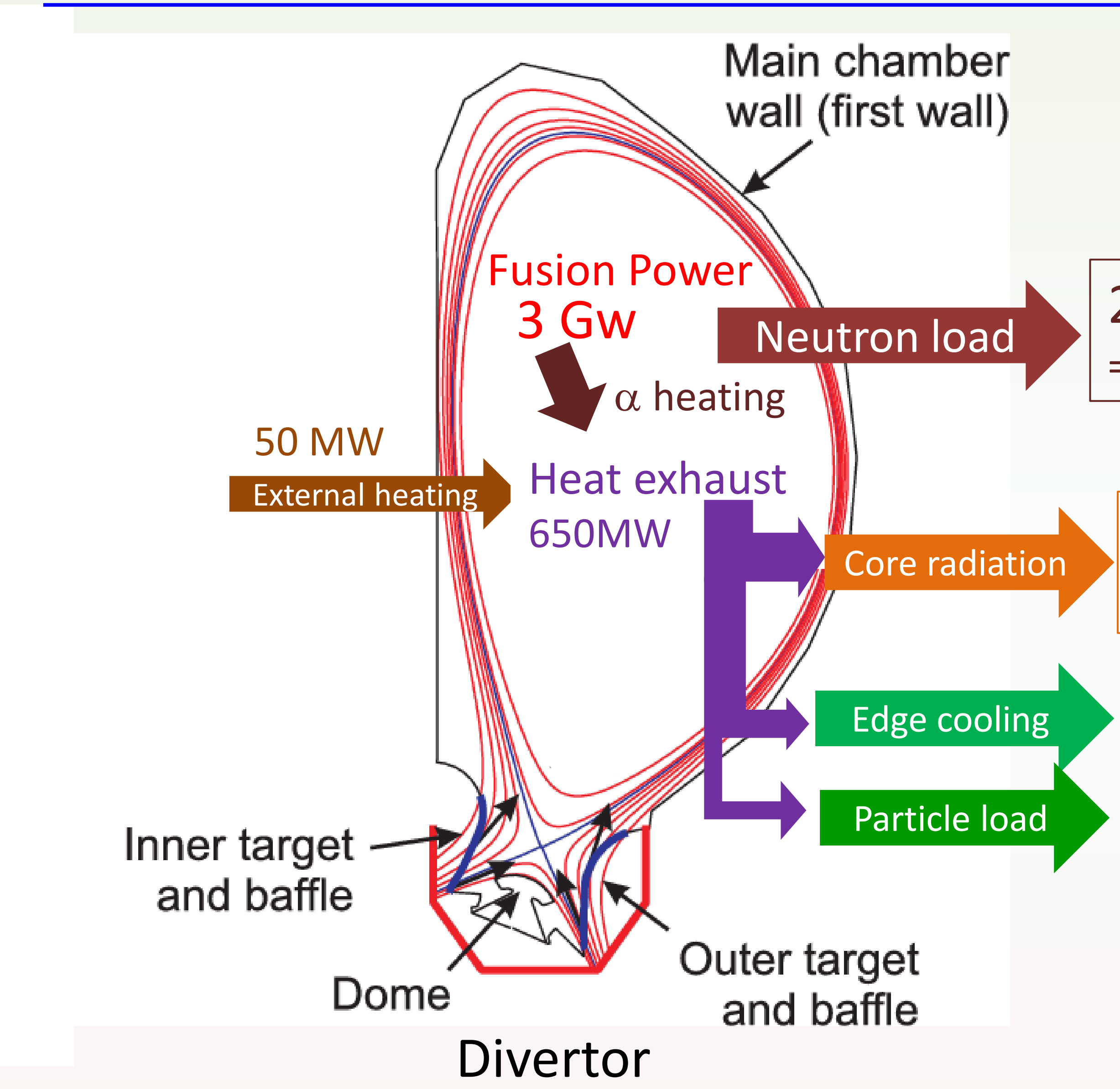
x 100 in divertor area

Conf. Areal atomic density of solid surface = 10^{19} atoms/m^2

According to simple molecular kinetics

$\phi = p/(2\pi mkT)^{1/2} \rightarrow \phi\ H_2\ (1atm, RT) = 10^{28}/m^2s$

Details of steady power load in a 3 GW_{th} reactor

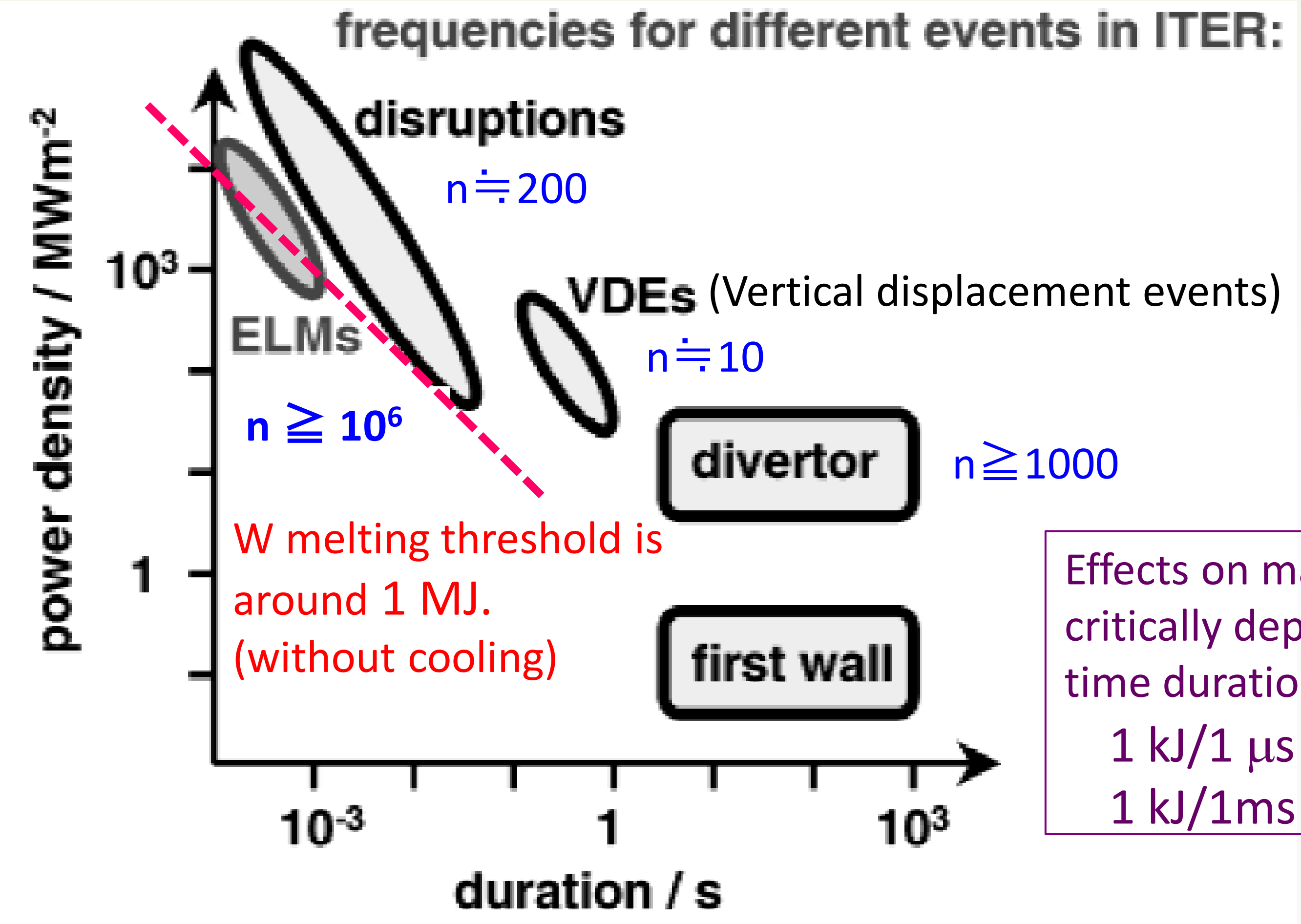


2.4 GW
=> 2.4 MW/m^2(av.)

500MW
=> 0.5 MW/m^2 (av.)

150MW
=> 1 - 2 MW/m^2
(First wall)
=> 10 - 20 MW/m^2
(Divetor plate)

Concerns of transient/off-normal power loads



n≒200

(Vertical displacement events)

n≒10

$n \geqq 10^6$

n≧1000

W melting threshold is around 1 MJ. (without cooling)

Effects on materials critically depend on time duration.

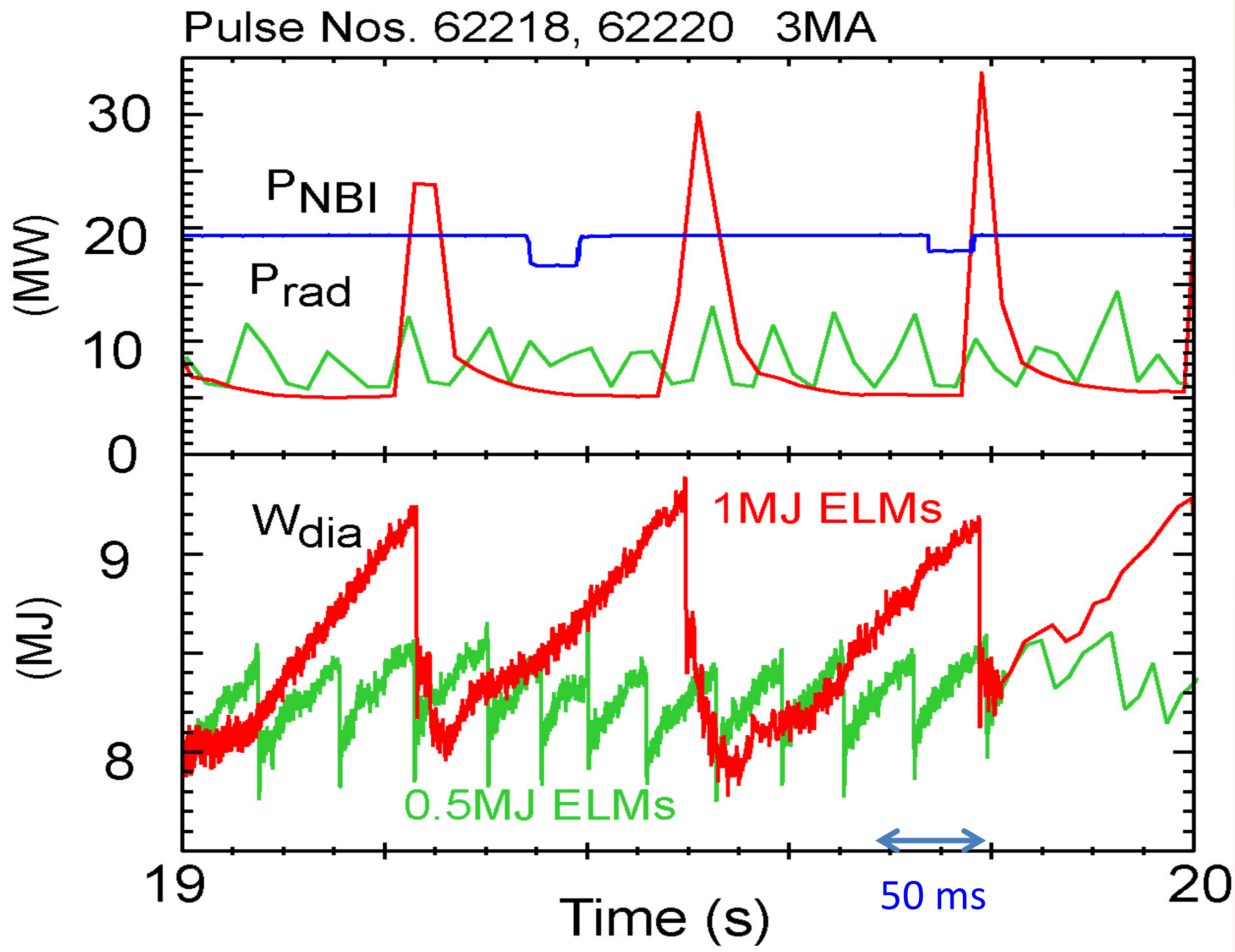
1 kJ/1 μs = 1 GW,

1 kJ/1ms = 1 MW

Plasma-induced thermal loads on PFCs in ITER;
n is the expected frequency for these events

J. Linke, Fusion Sci. Technol. 46(2004)142-151

Transient power load by ELMs in JET over 1 MJ (nearly W melting threshold)



J. Paméla et al. 17th PSI

Observed temperature rise of divertor by ELM in JET

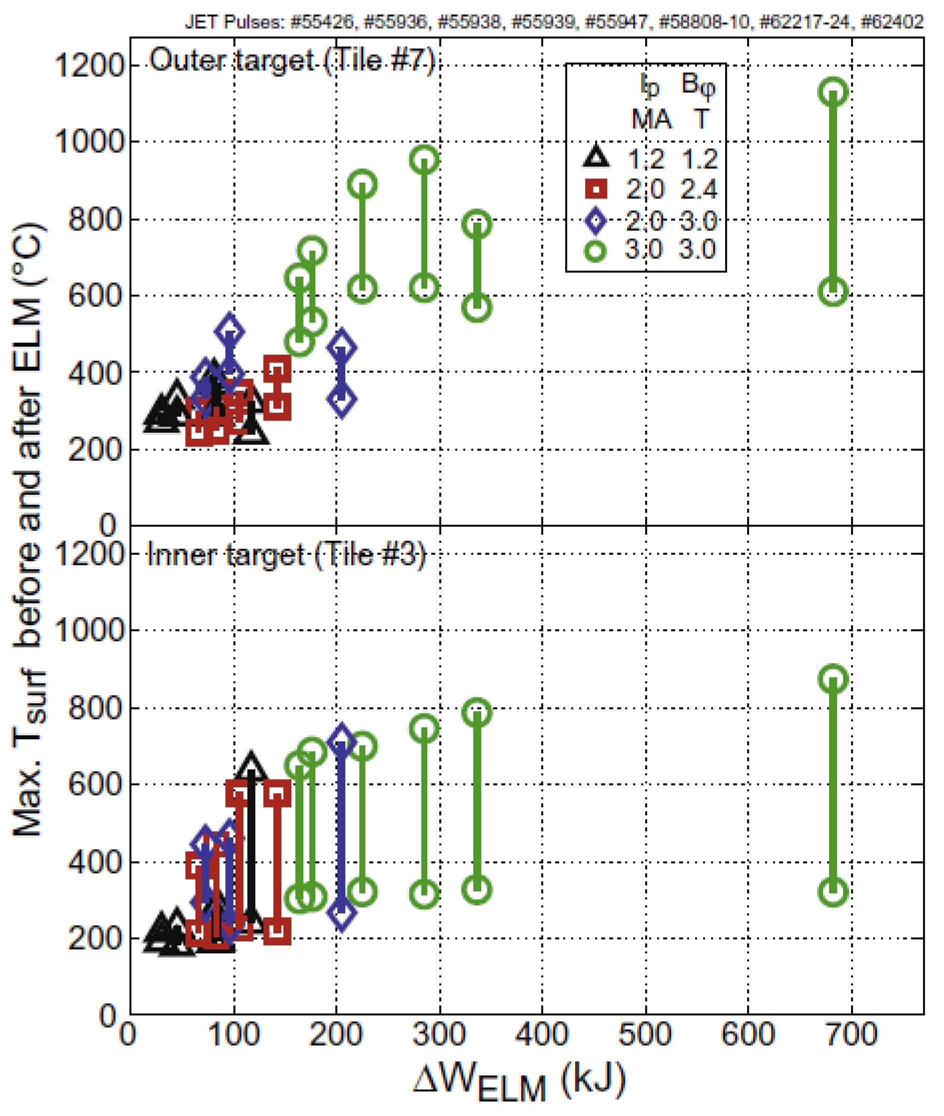
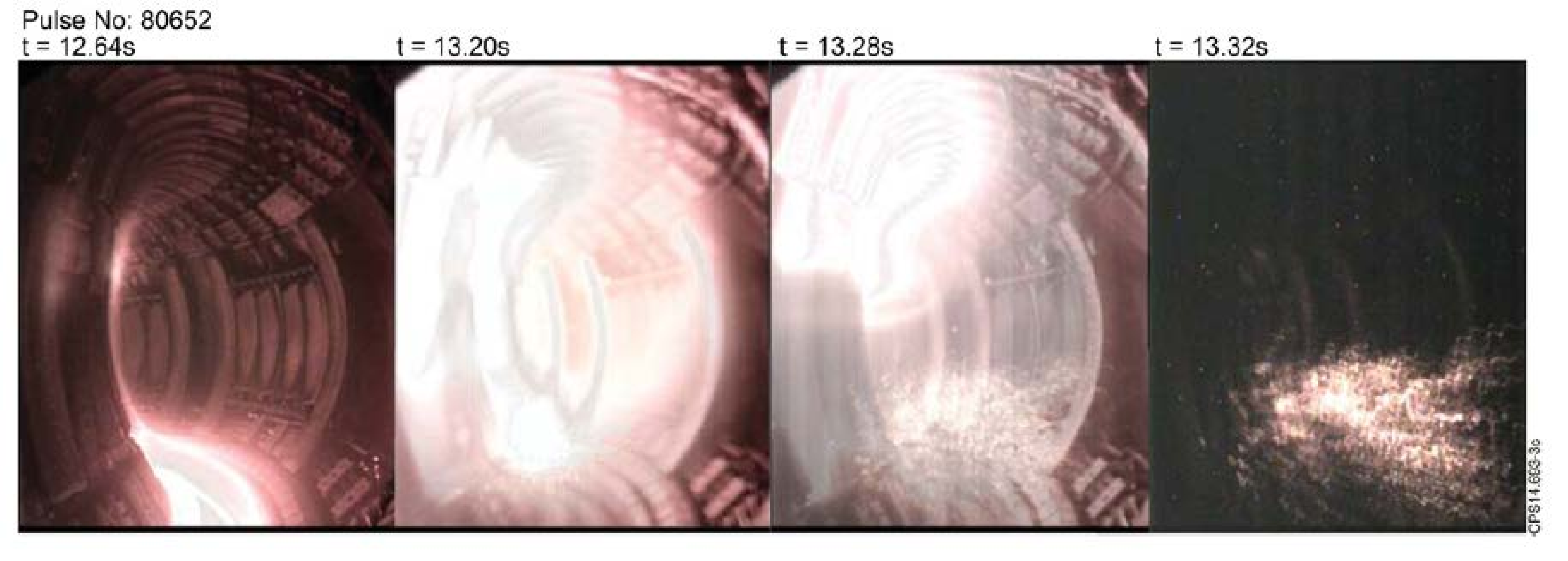


Fig. 5. Peak divertor T_{surf} before and during ELMs. Results are obtained from coherent averaging of ELM groups near the end of the H–mode phase.

R.A. Pitts et al. / Journal of Nuclear Materials 390–391 (2009) 755–759

Disruption made debris (droplets of melt layers, exfoliation of deposited layers)

Production of debris is important safety issue



Images from JET's COHU colour CCD camera (KL1) from octant 4 for JET pulse 80652. The plasma is terminated by a disruption with *td* = 13.14 s and *tm* = 13.18 s. Induced vessel forces are recorded as 0.43 MN. A cloud of debris is seen at *t* = 13.20 s and in subsequent frames. Note that the images in frames after *t* = *td* show distortion due to camera shaking.

J C Flanagan , *Plasma Phys. Control. Fusion* 57(2015) 014037

Summary of wall load (radiation and particles)

- Power load by transient or off normal events, like giant ELMS and disruption should be kept below threshold of destruction of any plasma facing components.

- Radiation (Impurities and Bremsstrahlung)

  Central radiation of high Z impurities should be avoided (high Z).
  Heat exhaust is required with edge cooling by radiation of seeded impurities.
  Melt damage should be avoided.

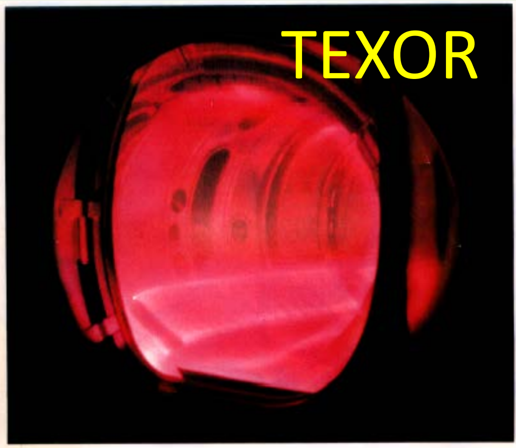
- Particles (electrons, ions and neutrals of fuels and impurities)

  Effects not fully understood are ,
  - Extremely high flux (over $10^{24}/m^2 \cdot s$, cf. $10^{19}/m^2$ of areal atomic density)
    Flux dependence of sputter erosion
    Immediate surface saturation with fuels making density control hard
    Increased of T inventory with high surface concentration and deep penetration with large diffusion flux (in metals)
  - Simultaneous injection of electrons with ions
    Changes surface chemical nature with high density electron excitation in surface
    (Ex. an insulator could become a conductor)
  - Effects of seeded impurities (Increase of sputter erosion)

- Neutrons (to be used as heat by volumetric energy conversion in blanket)
  - Power load to PFM is not large
  - Degrading thermo-mechanical properties of fusion rector materials by neutron damages
    (Loss of ductility, reduction of thermal conductivity, heat shock resistance, fatigue, etc.)
  - Possible increase of T inventory by trapping at defects (loss of ductility)

3. PMI in large tokamaks (TFTR, JET, JT-60U, etc.)

Response of PFS to high heat and particle load

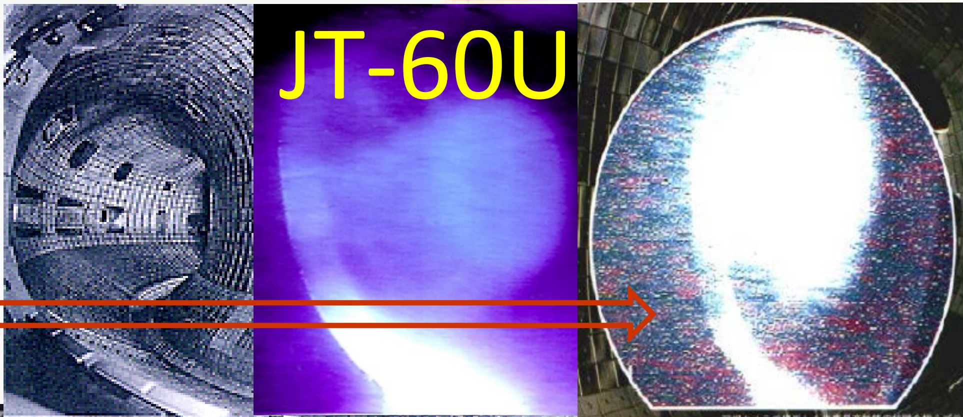


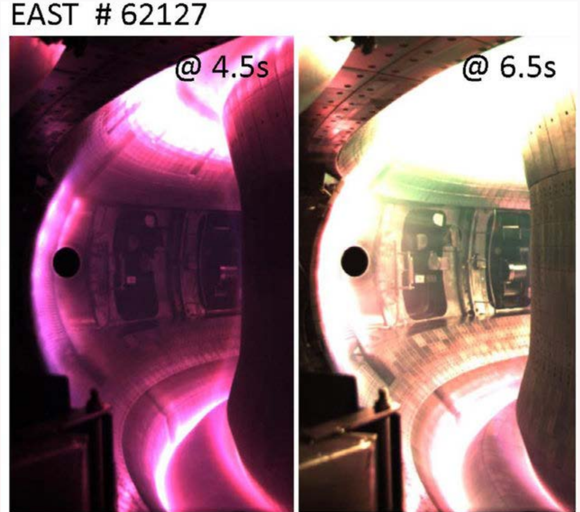
Radiation from high T plasma (center) is not visible, while radiations from limiters and divertor plates are appreciable.

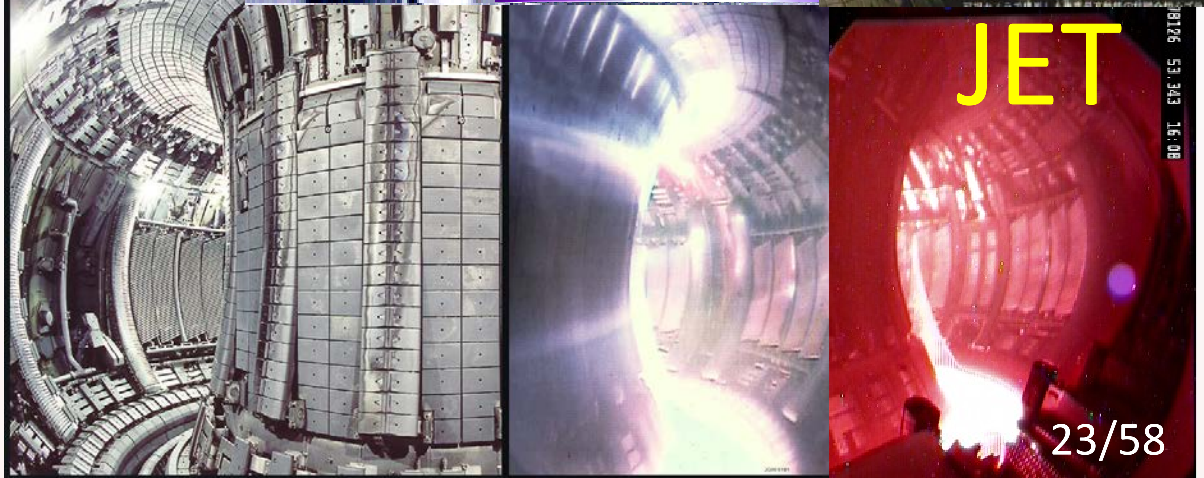
Different colors in edge plasmas owing to different impurities

Plasma temperature of a few to 10 keV
  Significantly high heat load
  Erosion/deposition became appreciable
  Difficulty of density control owing to
    Fuel saturation in wall
    or release from wall caused by Temp. up

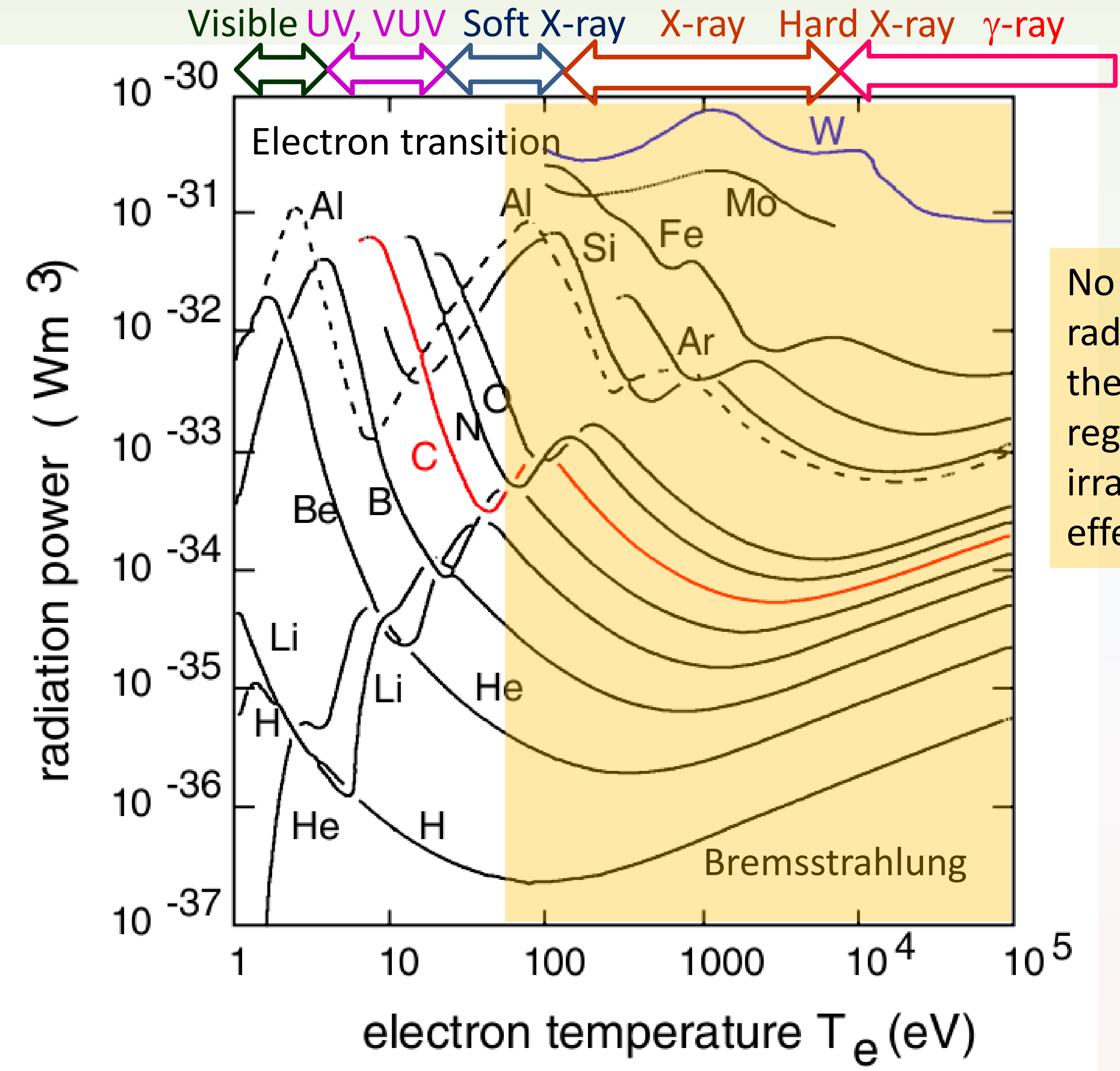
Neutron production by DD-reaction in JT-60U gives electric noise on a CC camera







Radiation power based on corona model

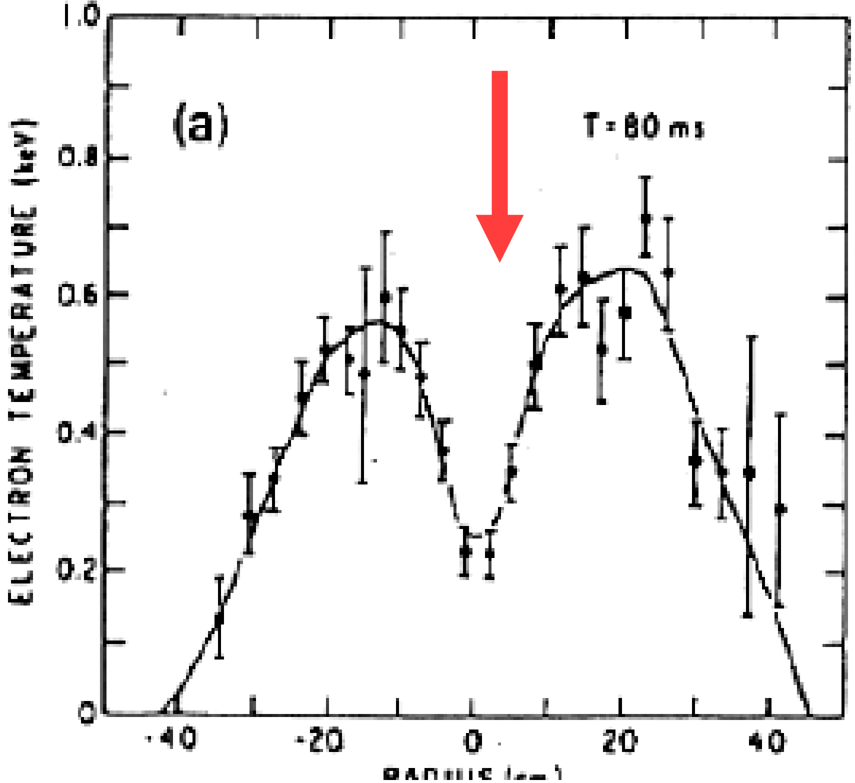


No high power radiation sources in these wave length region to examine irradiation effects

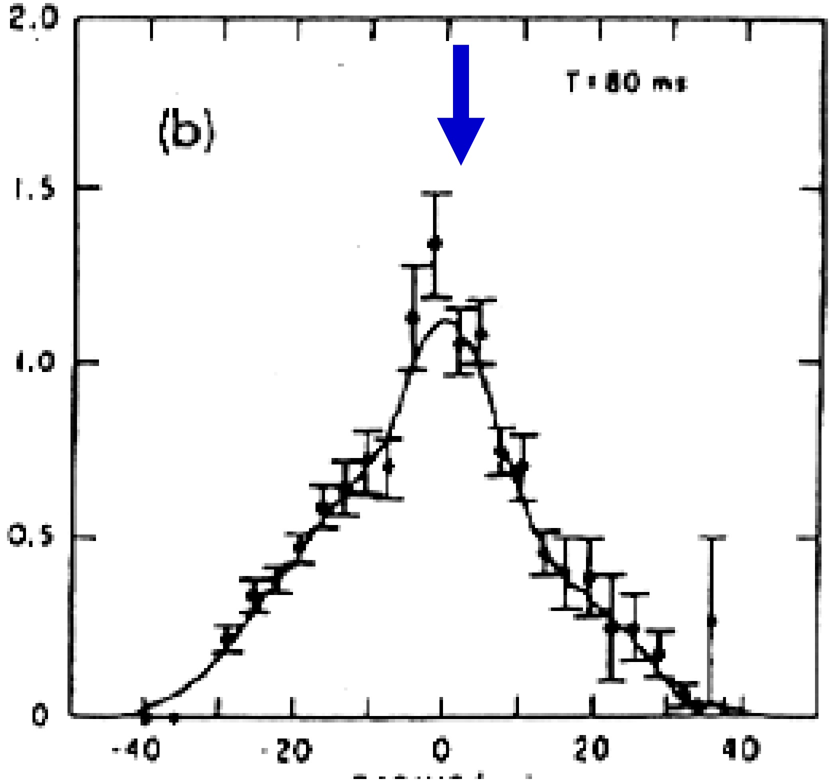
R. Schneider, X. Bonnin, K. Borrass, et al.. Plasma edge physics with b2-eirene., Contributions to Plasma Physics, 46(1-2):3191, 2006.

W accumulation was observed in ORMAK tokamak plasma center at 1979

Resulting central radiation and hollow temperature profile



Hollow temperature profile due to W accumulation



Recovery of peaked temperature profile by controlled gas puffing

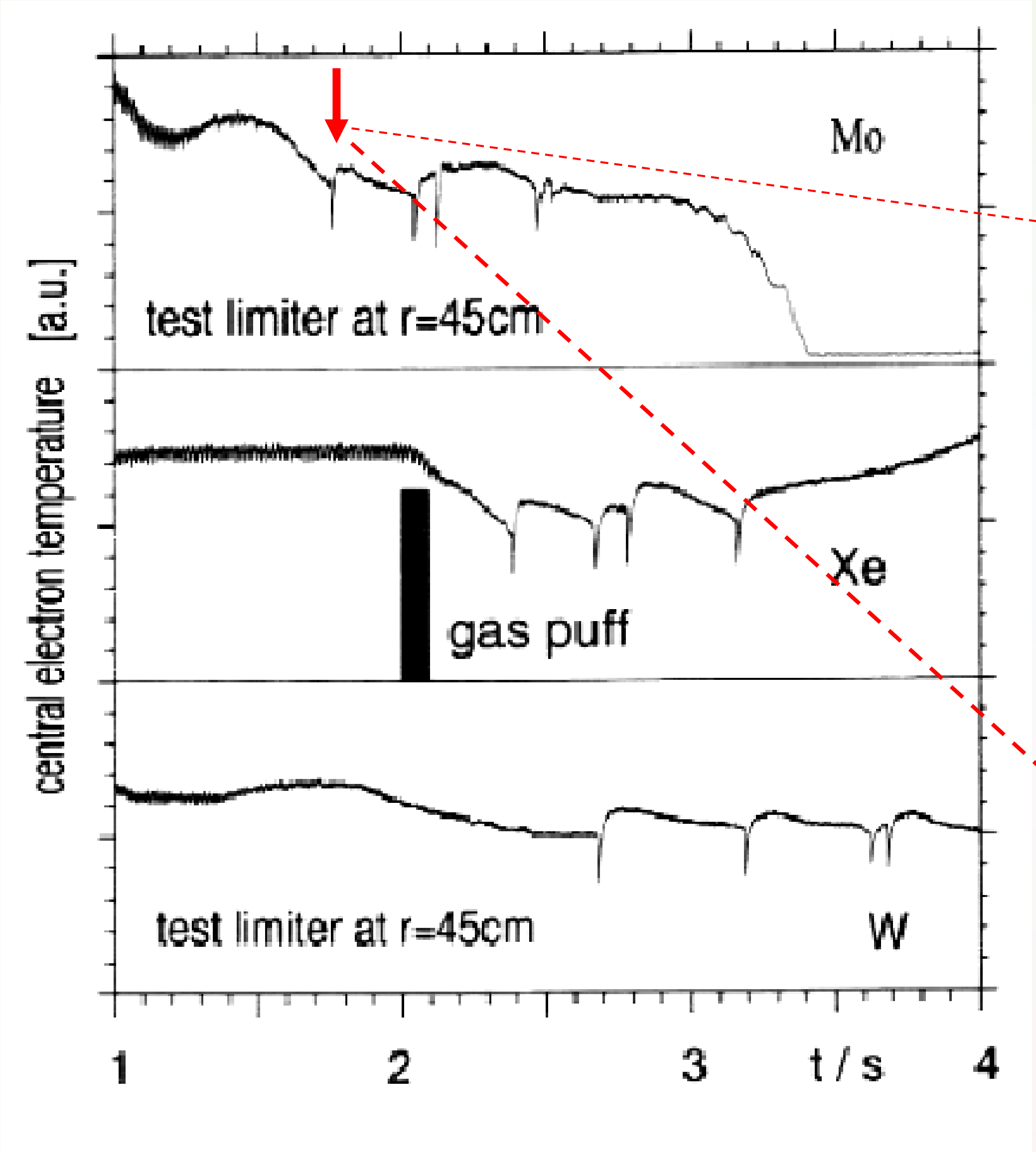
Hawryluk et al. Nucl. Fusion 19(1979)1307.

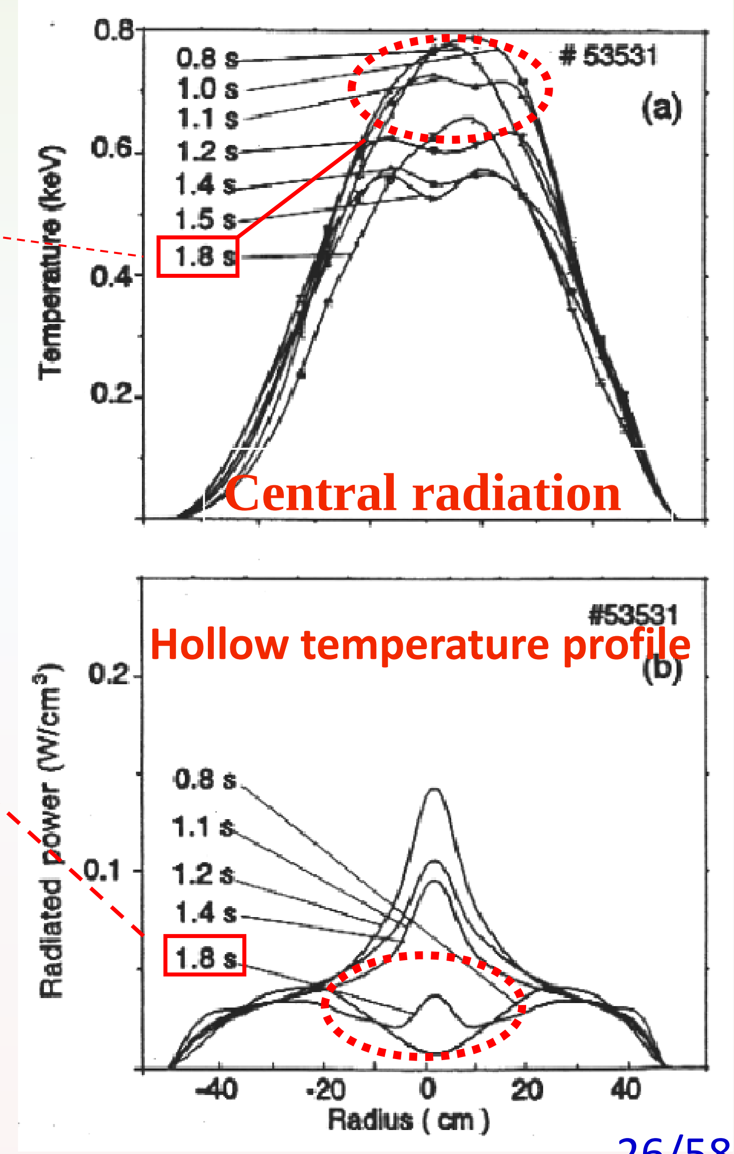
After this finding , W had been avoided to use as plasma facing wall,

until high Z limiter experiments started in TEXTOR under IEA cooperation at 1993.

High Z impurity accumulation results in repetitive minor disruptions.

(High Z gas gives similar results)





Central radiation

Hollow temperature profile

4. Material modification/degradation by high power load

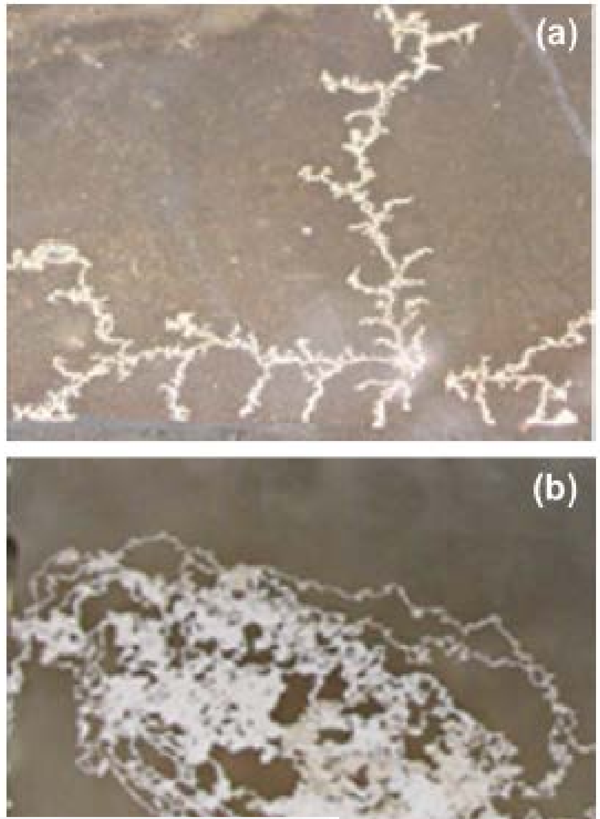
Consequence of huge power load by radiation and particles

- Material modification (mostly degradation) by high power load
  - Significant in W wall.
    - Limitation of lifetime of plasma facing wall
      - Erosion caused by
        - Melting/evaporation, sputtering, surface cracking/exfoliation
      - Degradation of material properties caused by
        - Heat shock, thermal (cyclic) heat load and neutron load
  - For C neutron irradiation effects concerns.

- Erosion, transport and deposition
  - Significant in Carbon wall
    - Limitation of lifetime by chemical erosion
    - Significant T retention in deposited layers in particular plasma shadowed areas

Examples of “unmagnetized” arc tracks in DIII-D: on a bottom surface of a mid-plane port (a), and on a surface of a metallic mirror recessed in another midplane port (b).

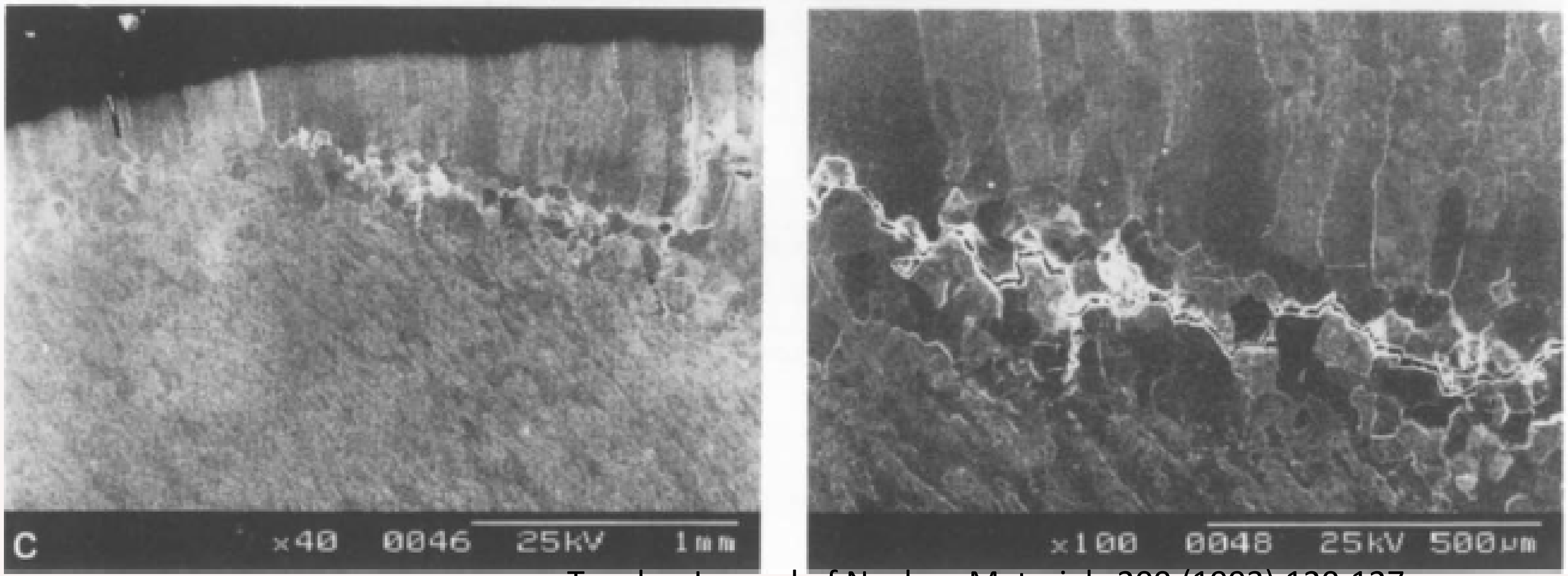
Rudakov, Journal of Nuclear Materials 438 (2013) S805–S808



After once melting or cyclic heat load

Easy crack propagation along boundary between recrystallized or once melted zone and matrix.

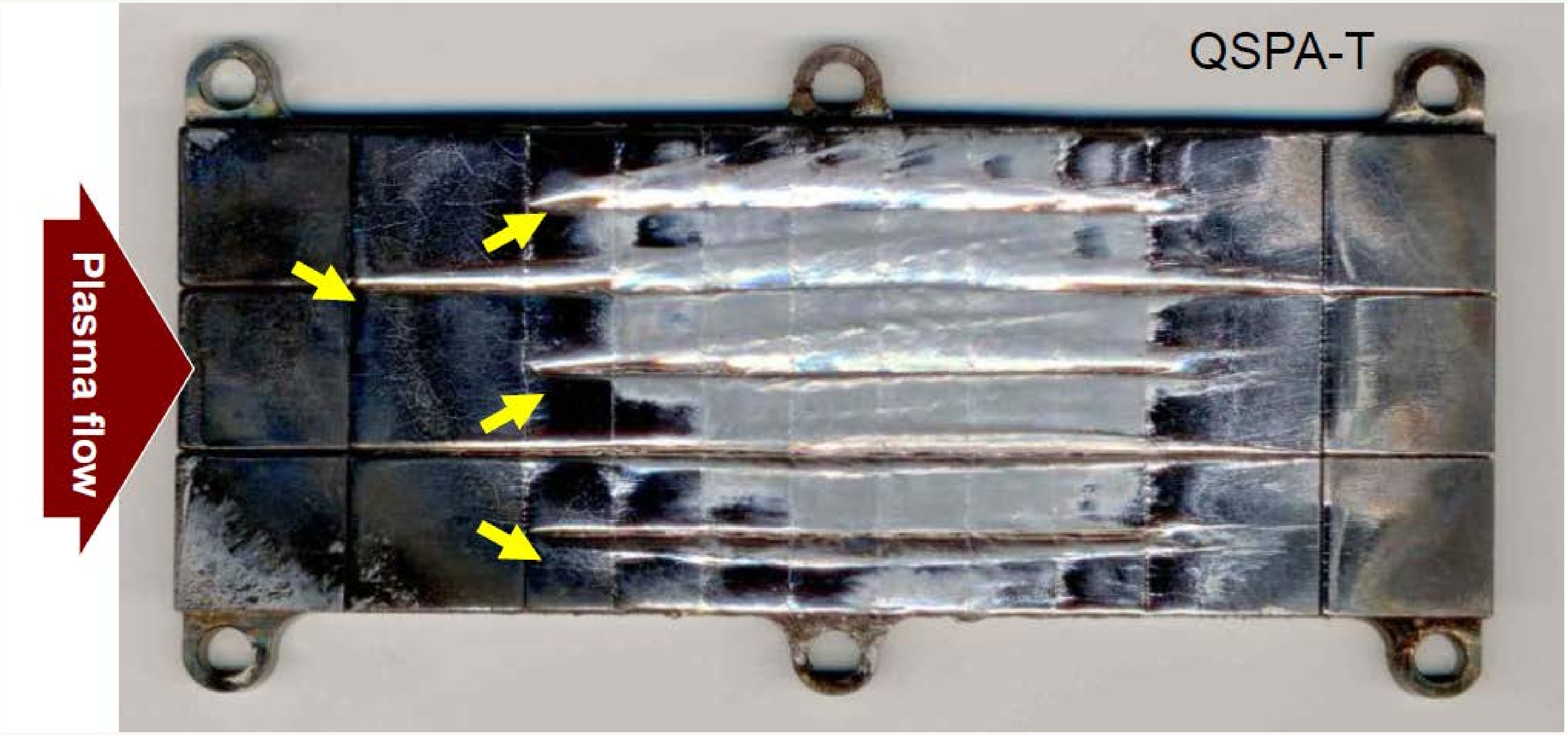
Accumulated impurities (C, O, S, probably He and H) at recrystallized boundary loosen the bonding between grains



Tanabe, Journal of Nuclear Materials 200 (1993) 120-127

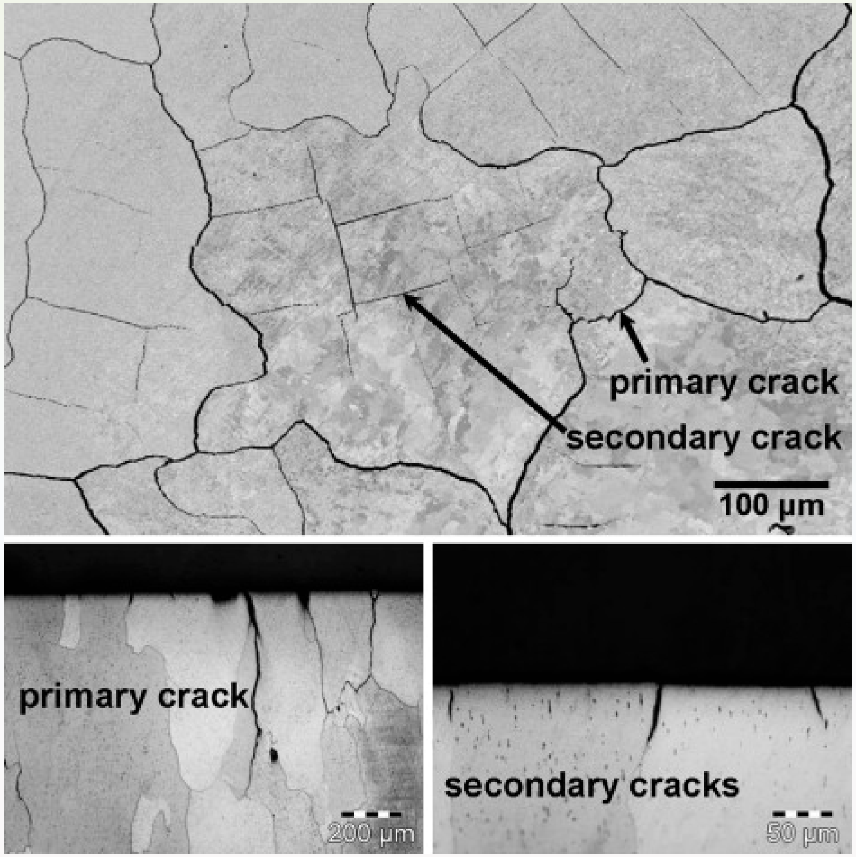
FLM simulation using quasi-stationary plasma accelerator (QSPA)

W exposed to 100 pulses of 1.5 MJm-2

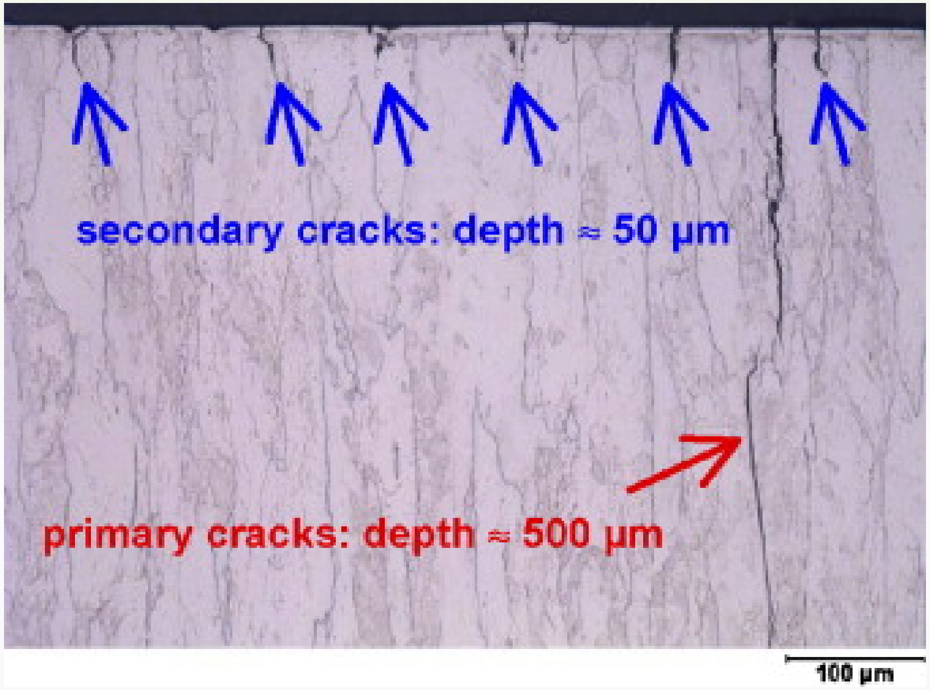


Effect of ELMs on ITER divertor armour materials”, A. Zhitlukhin et al., J. Nucl. Mater. **363-365** (2007) 301

Cracks perpendicular to the surface along grain-boundaries in matrix



SEM-image (top) of the tungsten grade M1 at the boundary between loaded and not loaded surface ($P = 0.88$ GW m^{-2}, single shot)—primary and secondary cracks; cross-section: LM-images of primary and secondary cracks (bottom).



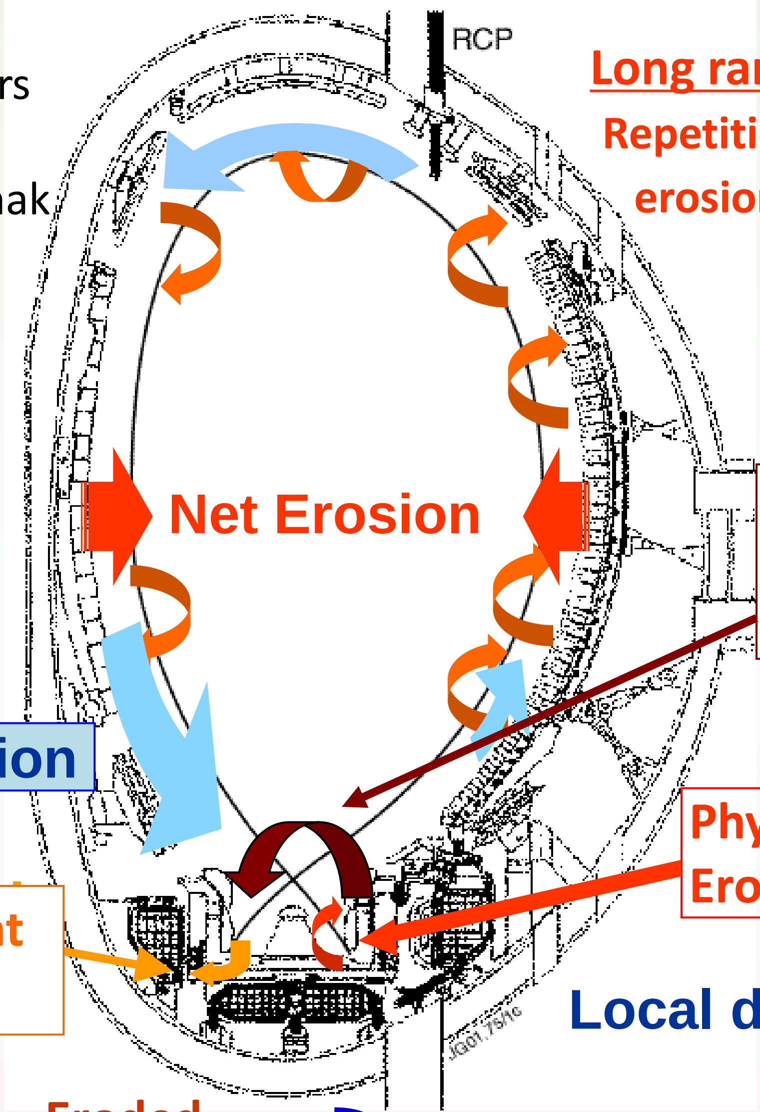
The cross section of the tungsten sample after 100 shots in QSPA facility with heat load of 0.9 MJ/m^2 and 0.5 ms time duration. Meandrous pale vertical lines are the boundaries separating elongated tungsten grains, perpendicular to the sample surface. Molten tungsten layer of 3–5 µm thickness is seen at the irradiated surface. Bold dark lines are the cracks.

G. Pintsuk, Fusion Eng. & Design 82 (2007) 1720–1729

S.E. Pestchanyi, Fusion Engineering and Design 82 (2007) 1657–1663

5. Erosion and deposition; Materials migration

Erosion and deposition occurs different location resulting materials transport in Tokamak



Long range transport

Repetition of erosion & prompt redeposition

Net Erosion

Possible impurity transport through private flux region

Self shaping or surface smoothing occurs

Deposition

Physical & Chemical Erosion

Line of sight deposition

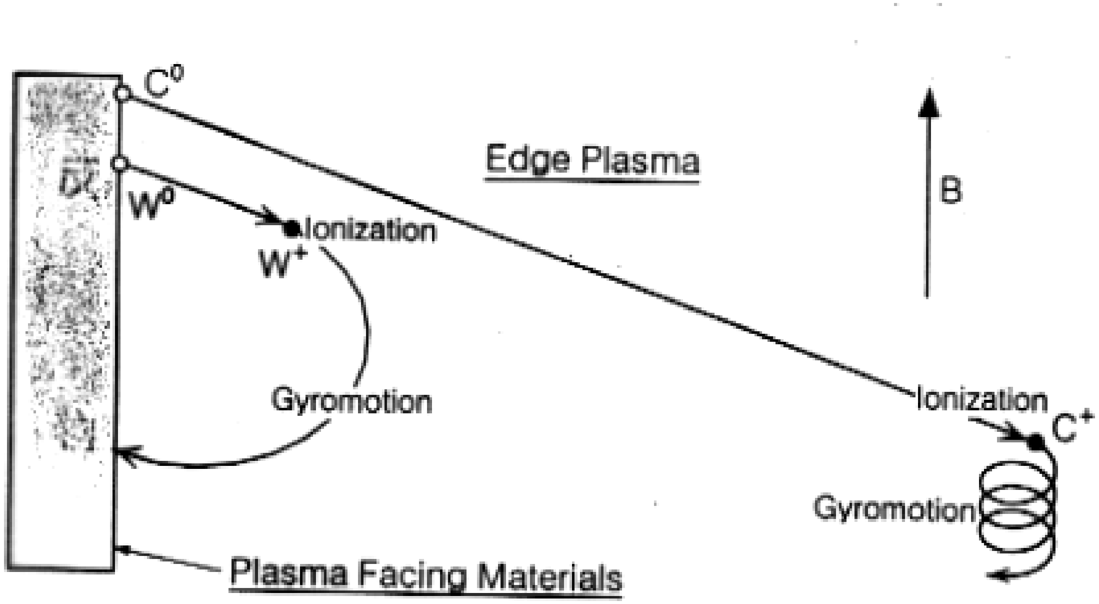
Local deposition

First wall	**Eroded**
Outer divertor	**Eroded**
Inner divertor	**Deposited**

Mass balance is missing!

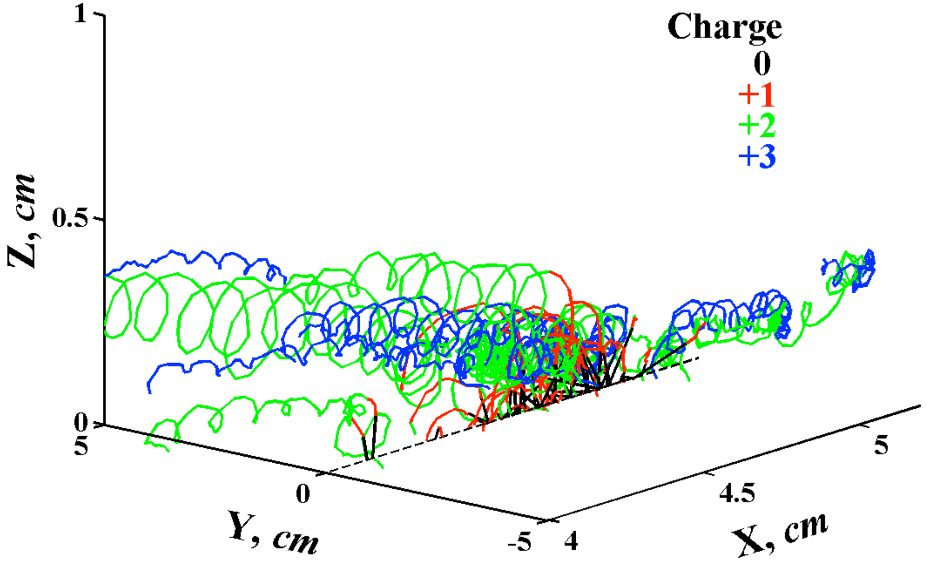
Prompt redeposition of sputtered particles suppresses net erosion

- No direct long range transport of eroded particles -



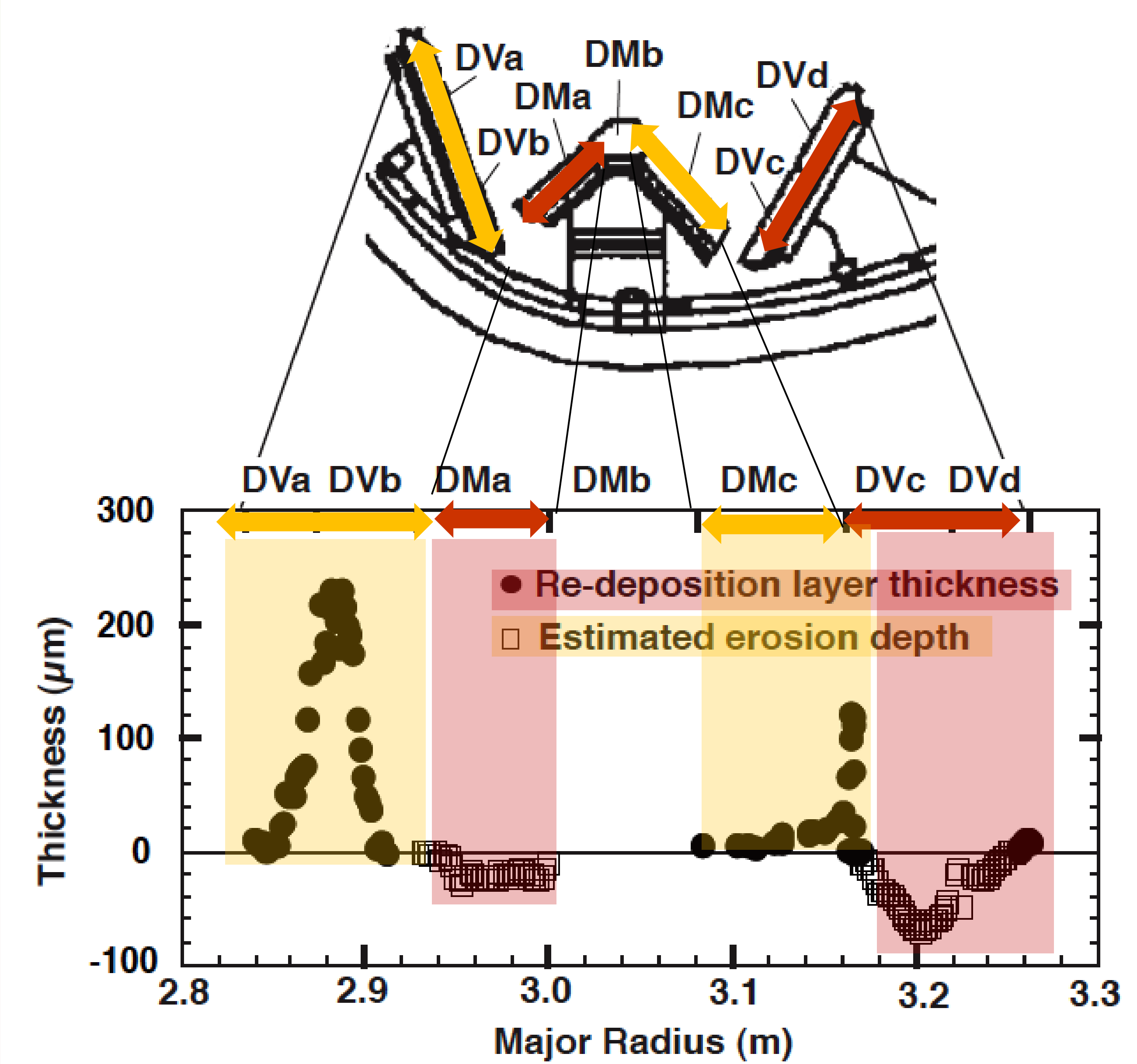
Heavier ions are easier to return to the surface due to larger Larmor radius

Trajectories of 50 typical sputtered Mo particles launched near middle of ALCATOR C-MOD outer vertical divertor computed by WBC code computed



Broks. J. Nucl. Mater. 415 (2011) S112-S116

Erosion and deposition profiles on W shaped divertor in JT-60U



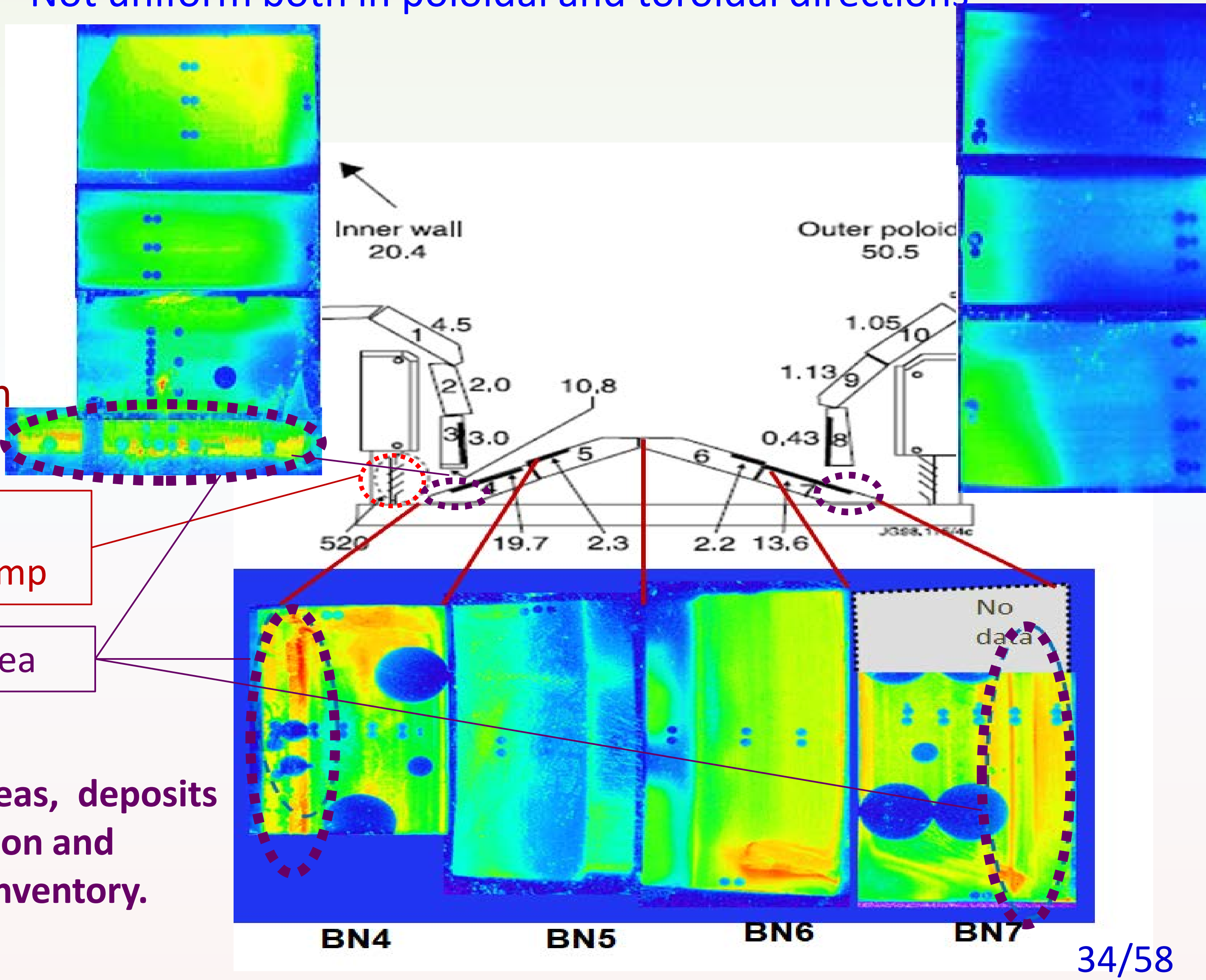
Samples: 1997~1999 (outer dome wing :1999)

Y.Gotoh et al. J. Nucl. Mater, **357**(2006)138-146

Eroded materials redeposit or codeposits with fuels on JET Mark-IIA divertor

T distributions well correlate to C deposition profiles.

Not uniform both in poloidal and toroidal directions



Significant deposition appeared at

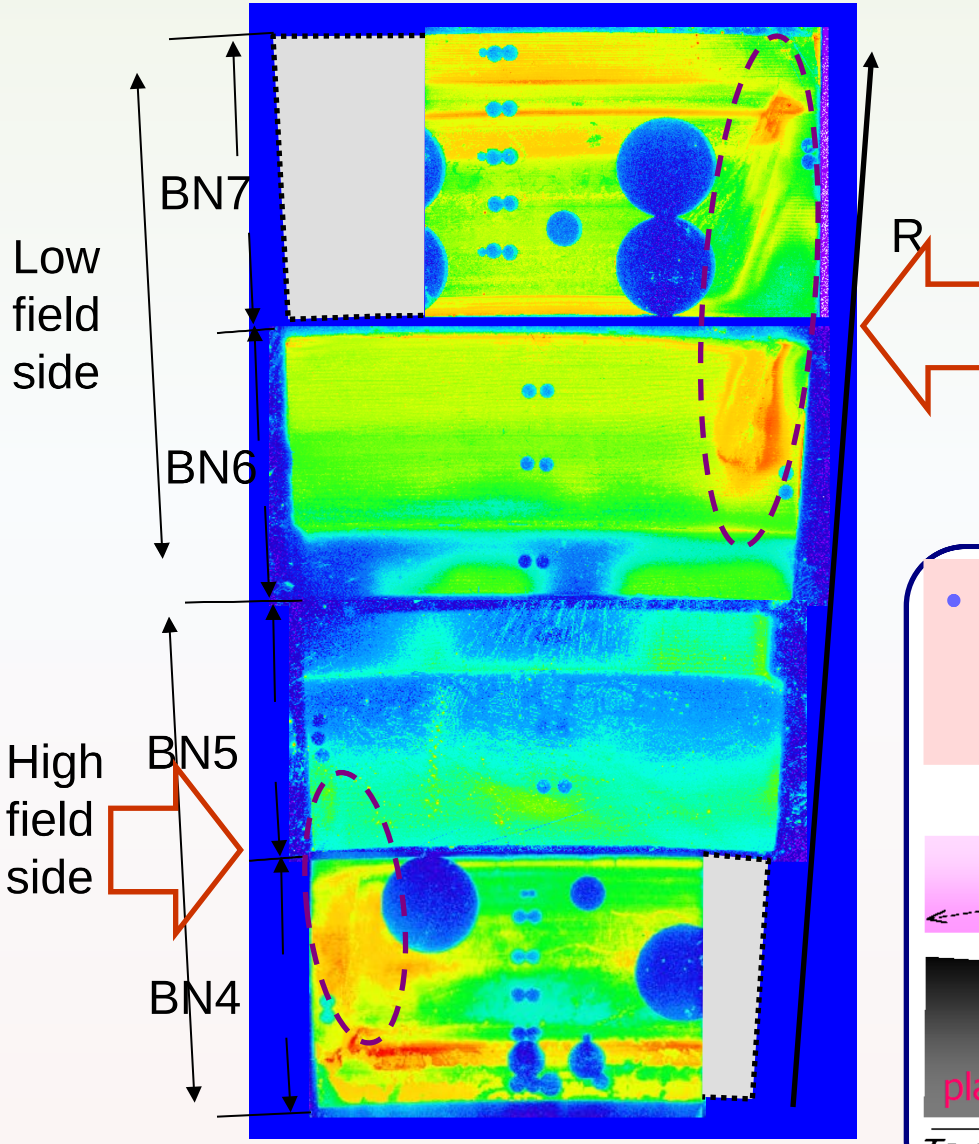
louvers
in front of cryo-pump

plasma shadow area

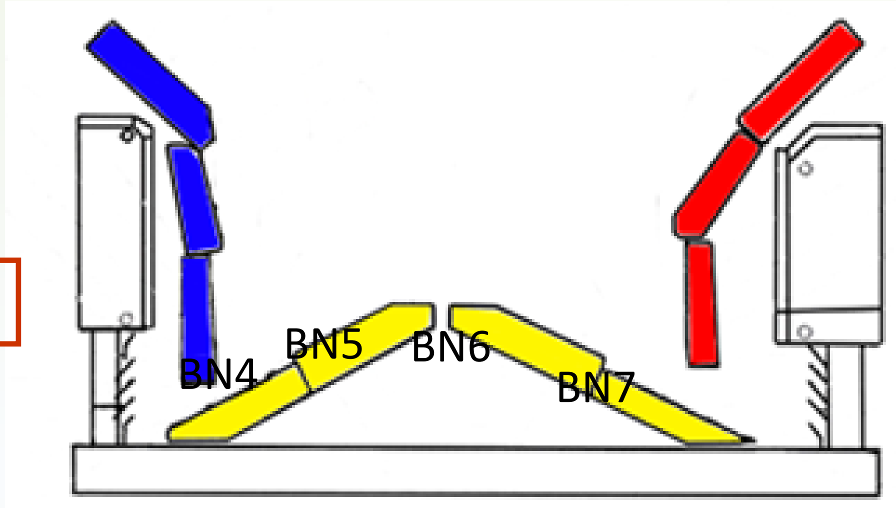
In plasma shadowed areas, deposits pile up without re-erosion and contribute largely in T inventory.

Non-uniform deposition in both toroidal and poloidal directions

Tile alignment is quite important, and self shaping would work to reduce erosion

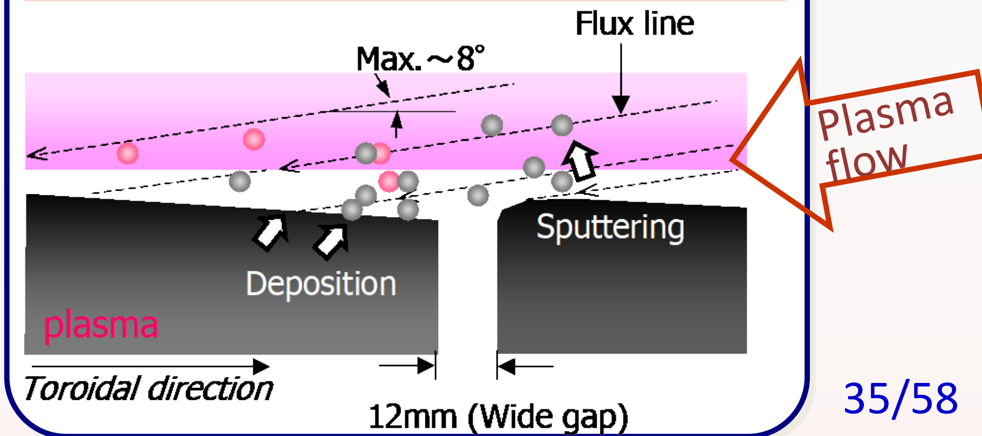


【Tritium image of the divertor floor tiles】



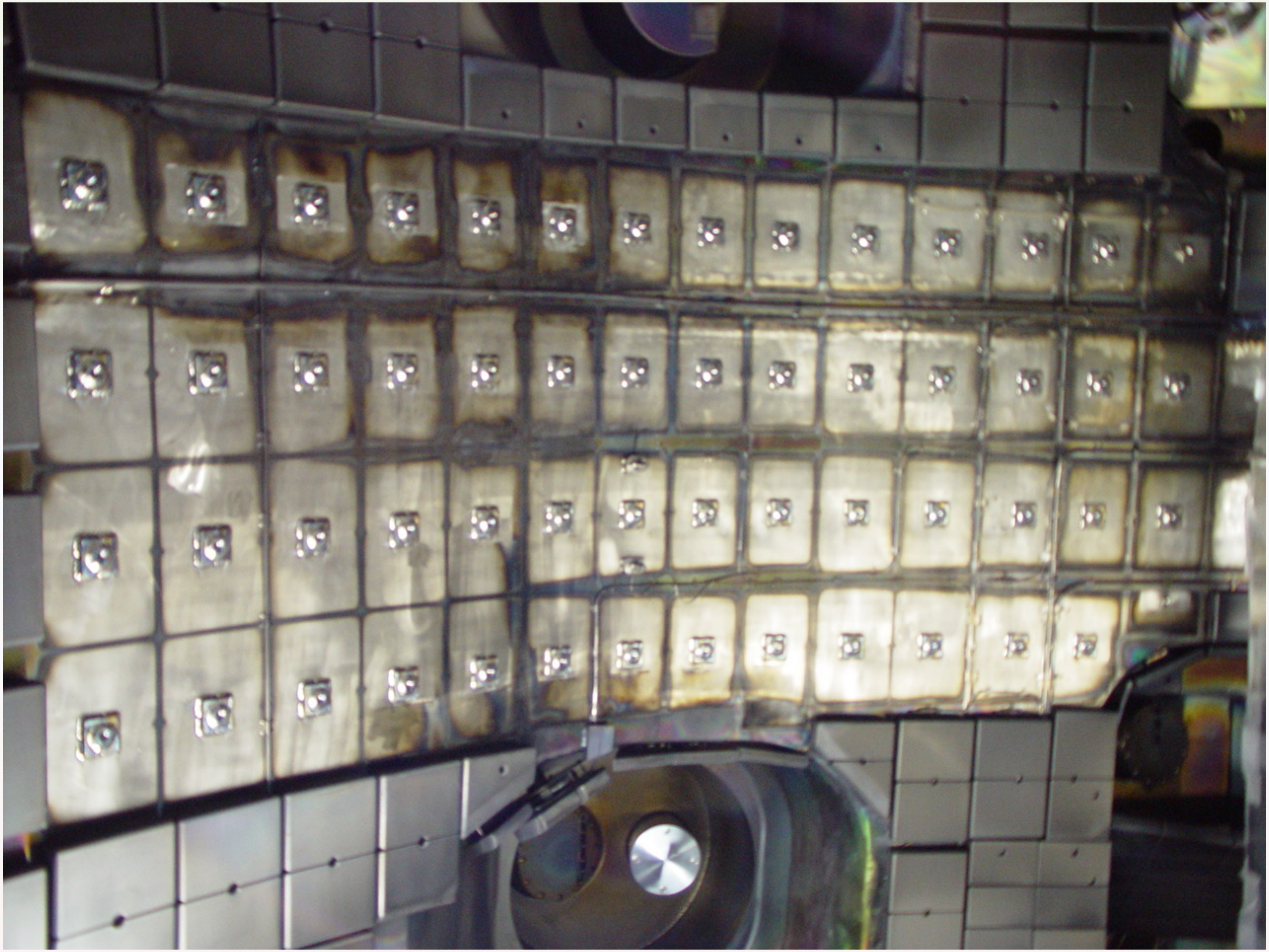
- Local erosion/deposition

  caused by different height depending on location of tiles



Deposition/Erosion in gaps and behind of tiles

observed in JT-60U outboard side after removal of carbon tiles



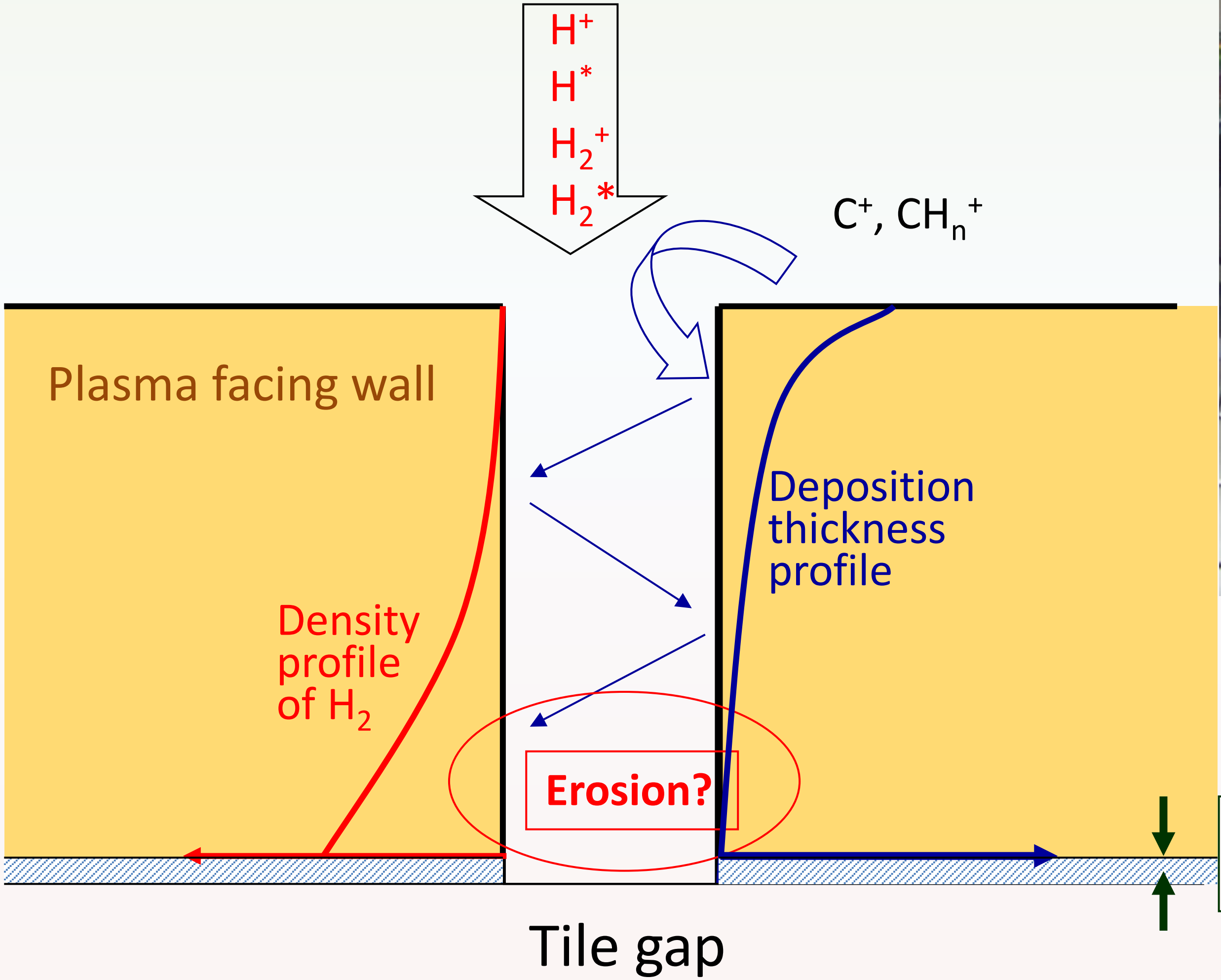
JT-60U outboard side after removal of carbon tiles

Appearance of deposition or erosion depends on the location of gaps tiles)

Deposition at the bottom of deeper and/or closed gaps seems small

Prompt redeposition is working!

Neutralized hydrogen at the bottom would re-erode C deposits



H^+
H^*
H_2^+
H_2^*

C^+, CH_n^+

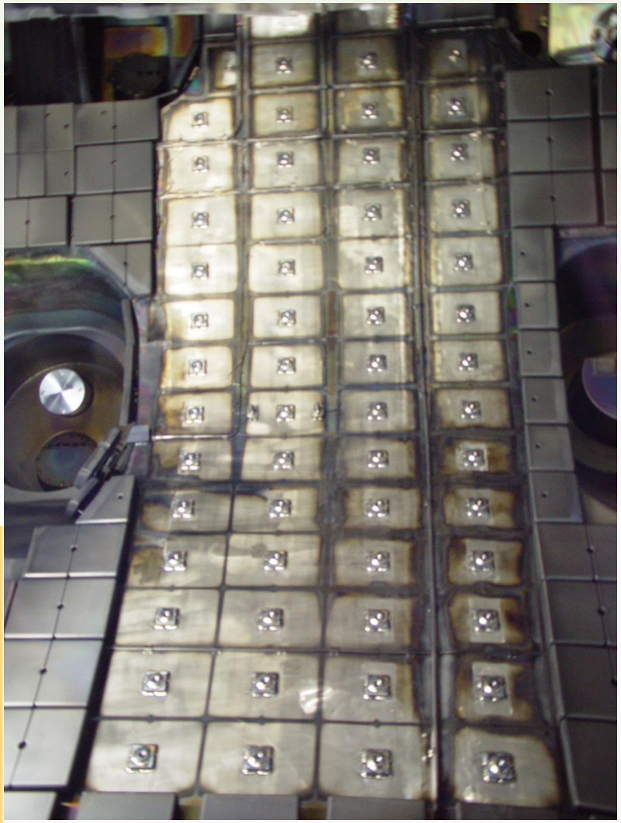
Plasma facing wall

Density profile of H_2

Deposition thickness profile

Erosion?

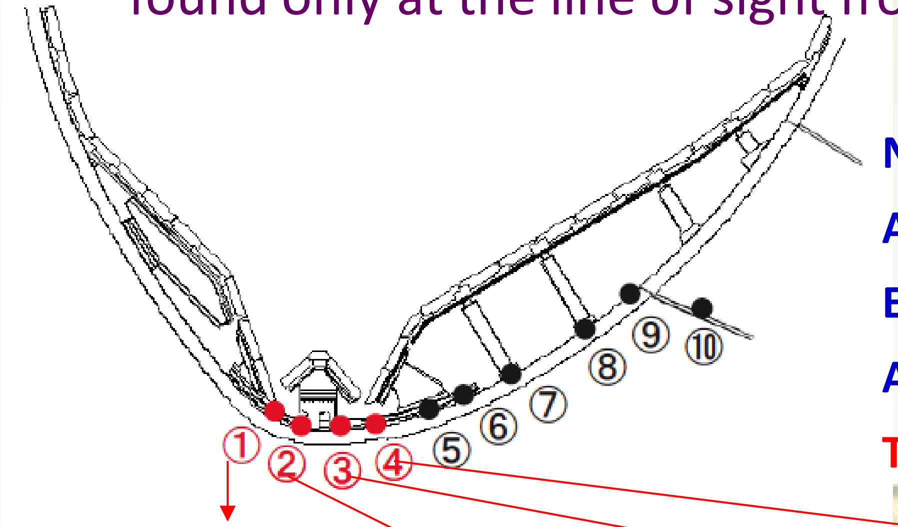
Tile gap



Deposition behind tile
gap between tile and base plate

Deposition at remote area (Bottom of Divertor) in JT-60U

found only at the line of sight from plasma (no long range transport)



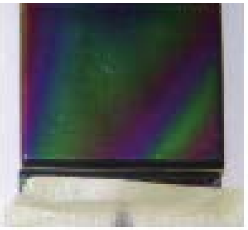
NB injection time : 8 x 10^3 s

Average deposition thickness : ~2μm

Estimated density : ~1.8 g/cm^3

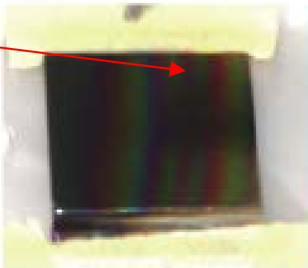
Area : 3.8 m^2

Total deposition : ~0.013 kg (~8 x 10^{19} C/s)


① C.P.-No.1


② C.P.-No.2


③ C.P.-No.3


④ C.P.-No.4

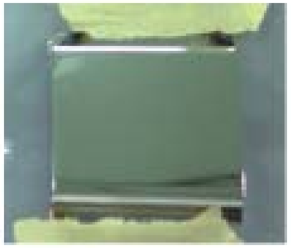

⑤ C.P.-No.5


⑥ C.P.-No.6


⑦ C.P.-No.7

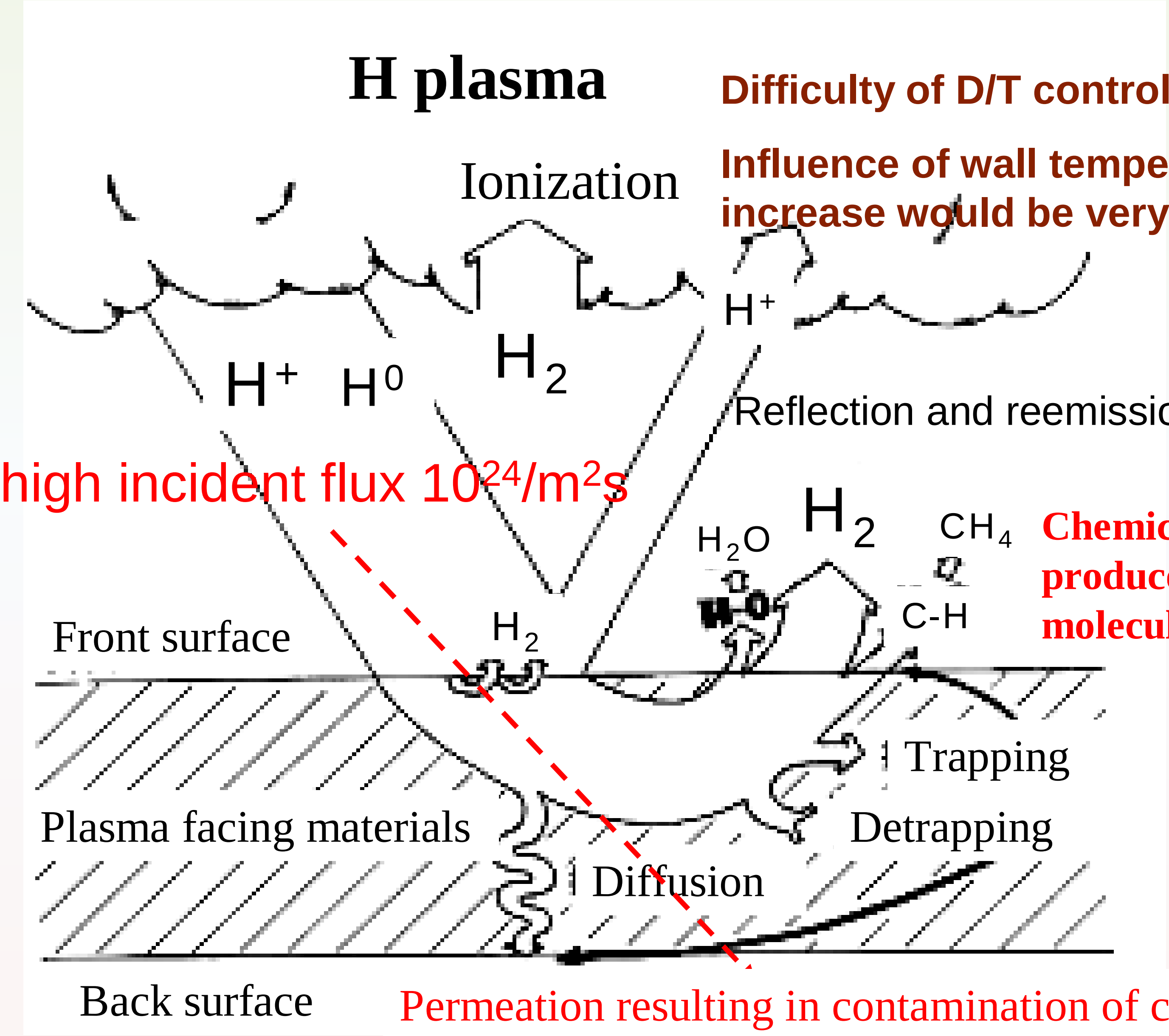

⑧ C.P.-No.8


⑨ C.P.-No.9


⑩ C.P.-No.10

Owing lower temperature (420K) operation (H+D)/C in redeposits is very high, 0.6 ~0.8, which makes their structure amorphous like.

6. Hydrogen recycling and fuel retention



Difficulty of D/T control in plasma

Influence of wall temperature increase would be very large

Very high incident flux $10^{24}/m^2s$

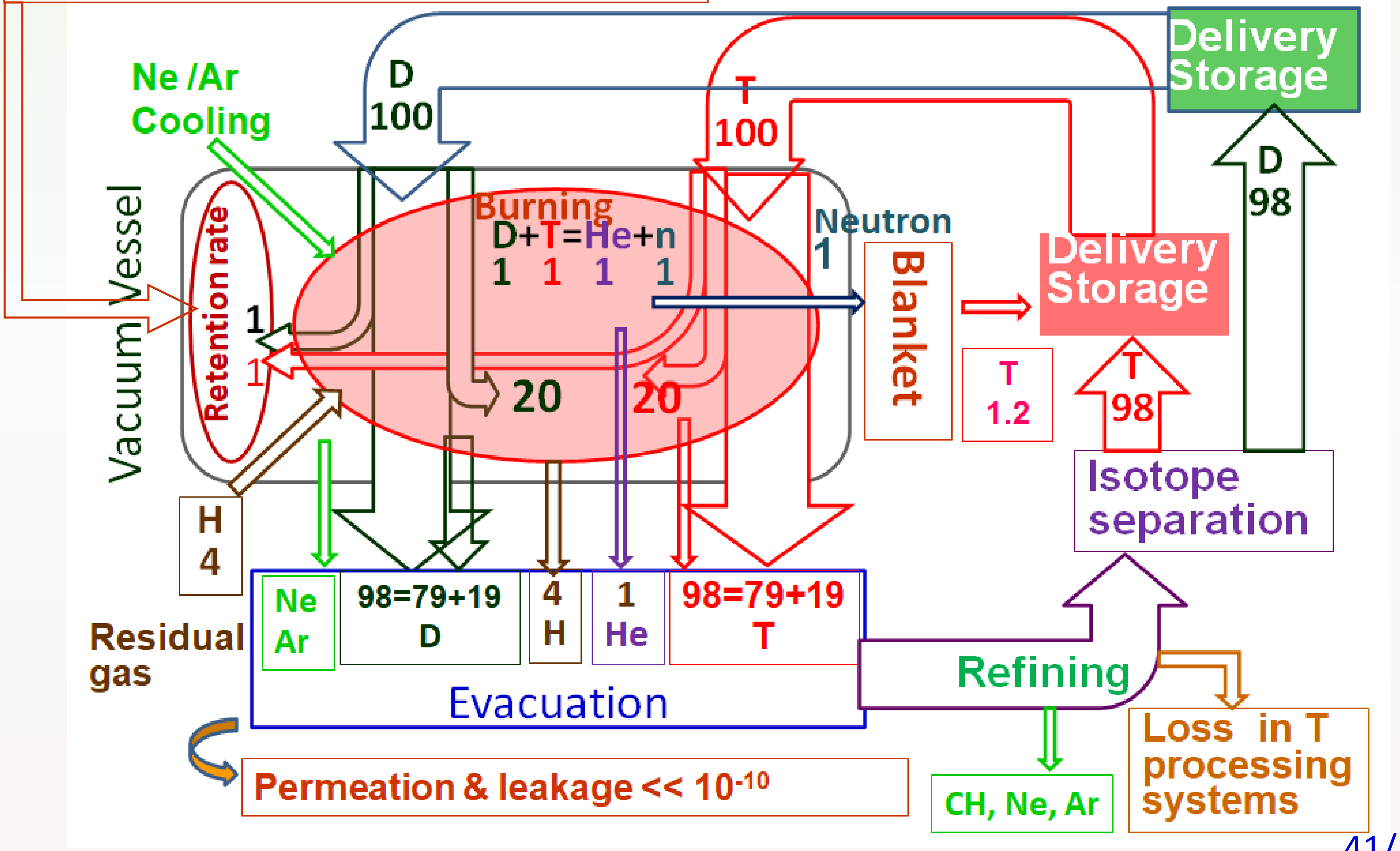
Chemical reactions produces hazardous molecules and dusts

Permeation resulting in contamination of cooling water

Recycling of fuels and their balance

Fuel retention in PFW is the one of the key to establish a fusion reactor as energy source

Assumption
Fueling efficiency: 20% of throughput
Burning efficiency: 5 % of plasma
Breeding rate: 1.2



To keep fuel self-sufficiency

Assumption

Fueling efficiency: 20% of throughput
Burning efficiency: 5 % of plasma
Breeding rate: 1.2

1.2 T production
for every 100 T throughput

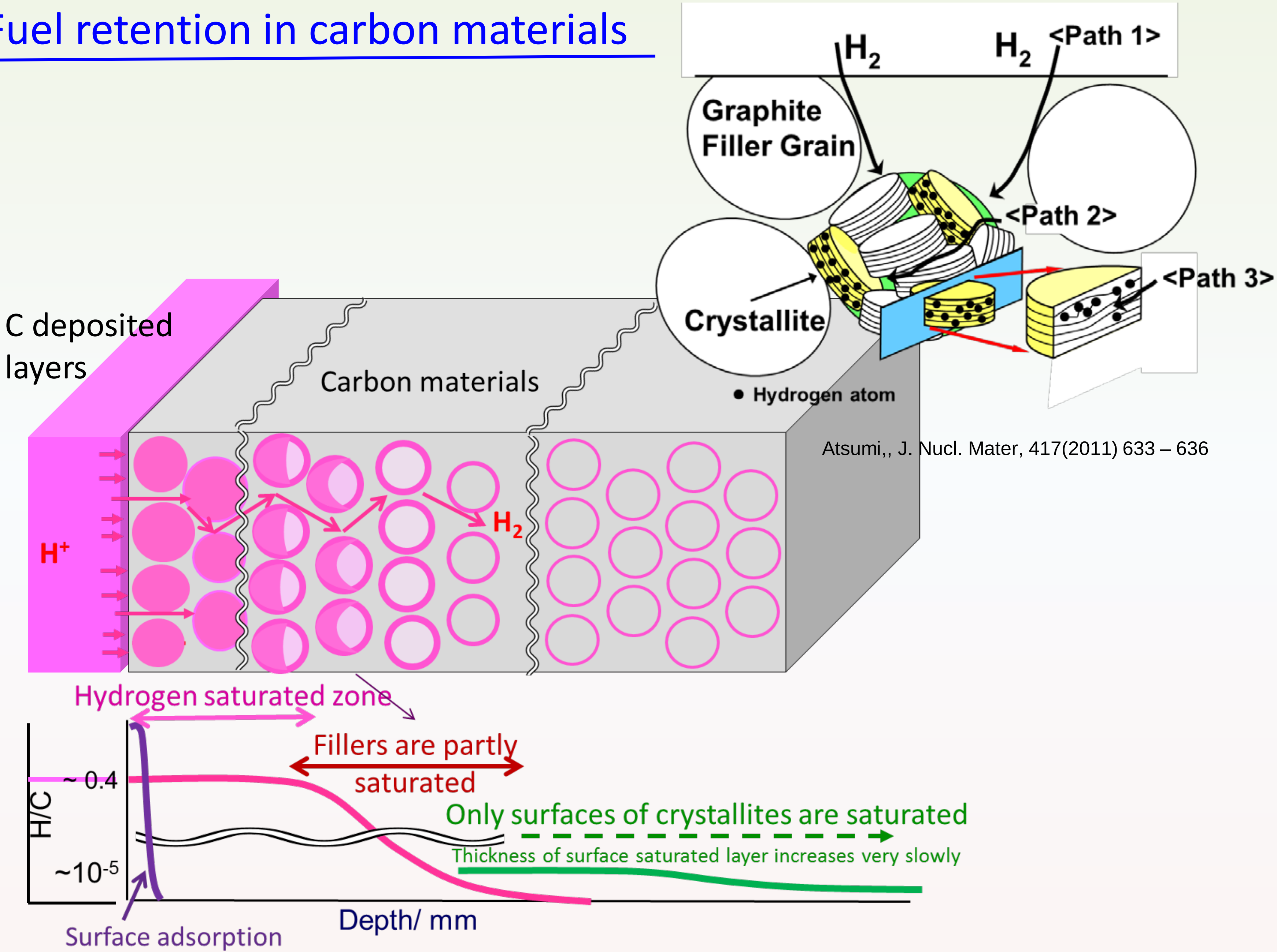
Wall retention rate should be
less than 0.2% and kept as small as possible
To compensate additional T losses of
5%/y by decay during the storage
~ 0.1% in T recovering process

The retention rate of 0.1% is still too large due to the site limit of T inventory in ITER

Exercise: Estimation of the amount of T retained in the wall
Suppose $\phi = 10^{21}/\mathrm{m}^2\cdot\mathrm{s}$, retention rate 0.1 %
Particle retention rate becomes $10^{18}/\mathrm{m}^2\cdot\mathrm{s} \simeq 10^{-5}\ \mathrm{gT}/\mathrm{m}^2\cdot\mathrm{s}$
Multiplying surface area of ~ $10^2\ \mathrm{m}^2$
The total retention rate becomes 1 mg/s

Cf. for ITER discharge of 400 s ; Retention rate is 0.4g/shot
to become site-limit of 1 k g only 2500 shots are required

Fuel retention in carbon materials



Atsumi,, J. Nucl. Mater, 417(2011) 633 – 636

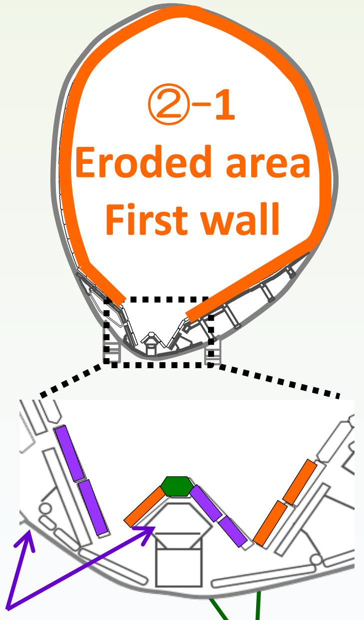
Fuel retention is caused by deposited C layers in a full carbon wall tokamaks

Device/ Campaign	Carbon deposition rate (C/s)	Fuel retention rate (D-T/s)	Fuel retention fraction (retained D-T/ injected fuel)	Remarks
JETMK-IIA Div.	6.5 x 10^{20}	5.8 x 10^{20} (D/C=0.8)	0.17 (in DTE1) 0.11 (in DTE2)	- T retention after non-mechanical cleaning - T retention after long term outgassing and mechanical removal of accessible T deposit
JETMK-IIB Div.	4.3 x 10^{20}	1.25 x 10^{20} (D/C=0.3)	0.03	- D retention from post mortem analysis
TFTR Campaign			0.16	
AUG	3.5 x 10^{20}		0.035 0.1	- D retention from post mortem analysis - D retention from fuel balance
TEXTOR	2.5 x 10^{20}	1.6 x 10^{19}	0.08	- D retention from post mortem analysis
Tore Supra		2.5 x 10^{20}	0.5	- D retention from fuel balance in dedicated long pulse discharges
JT-60U	3~6 x 10^{20} (only plasma facing area)	5.3 x 10^{18} (D/C=0.02)	0.0 (for saturated wall)	- Wall saturation appeared at 573K opeartion

Only high temperature operation at 573 K in JT-60U realizes wall saturation

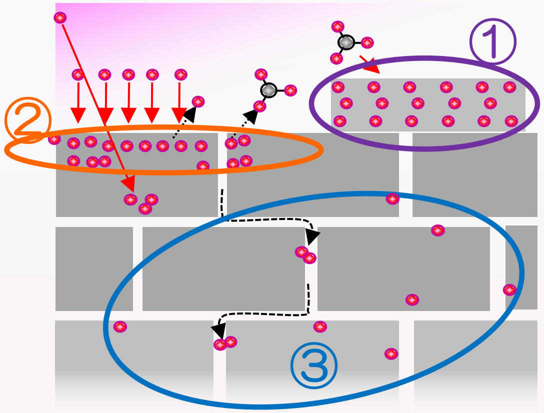
Such high T retention in full carbon wall machines is
one of the main reasons for excluding carbon as PFM in

Observed fuel retention in JT-60U operated at 573K

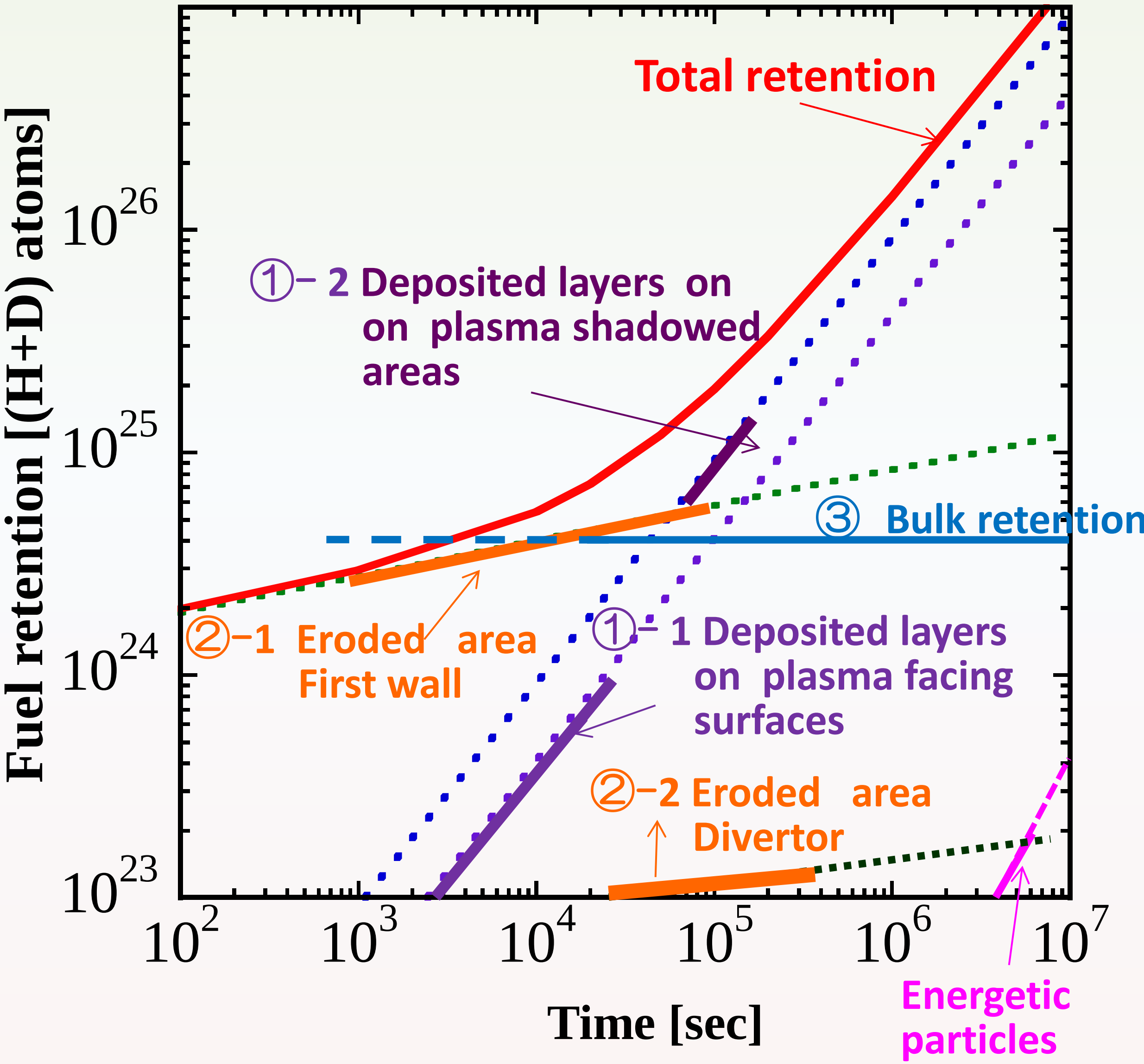


① Deposited layers

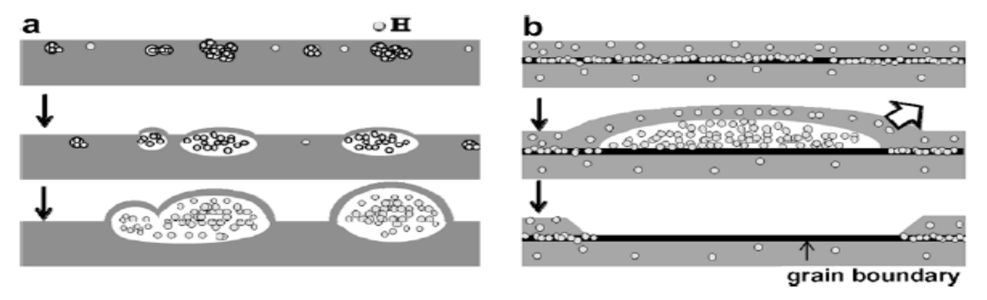
②−2 Eroded area



Carbon tile



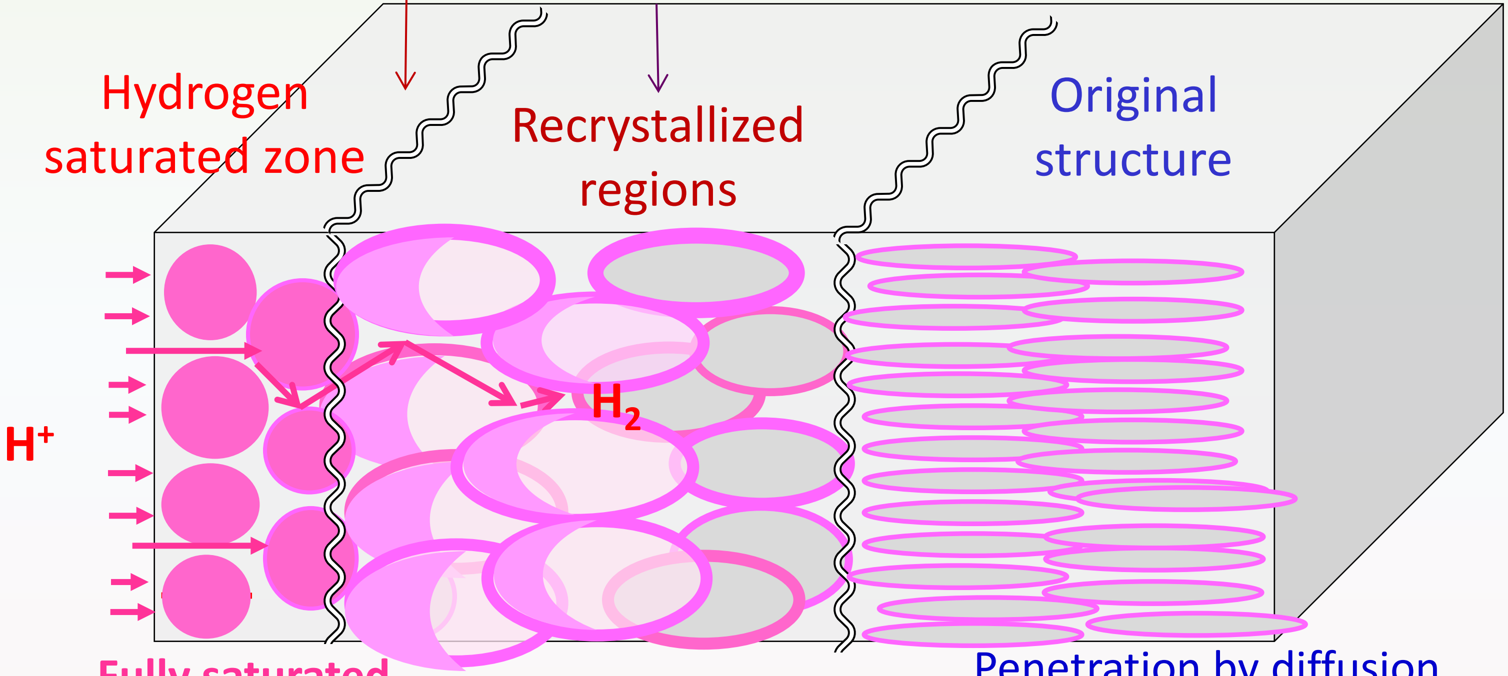
Hydrogen ret PMW



Blistering

by accumulation of bubbles

by exfoliation of grain boundaries



Hydrogen saturated zone

Recrystallized regions

Original structure

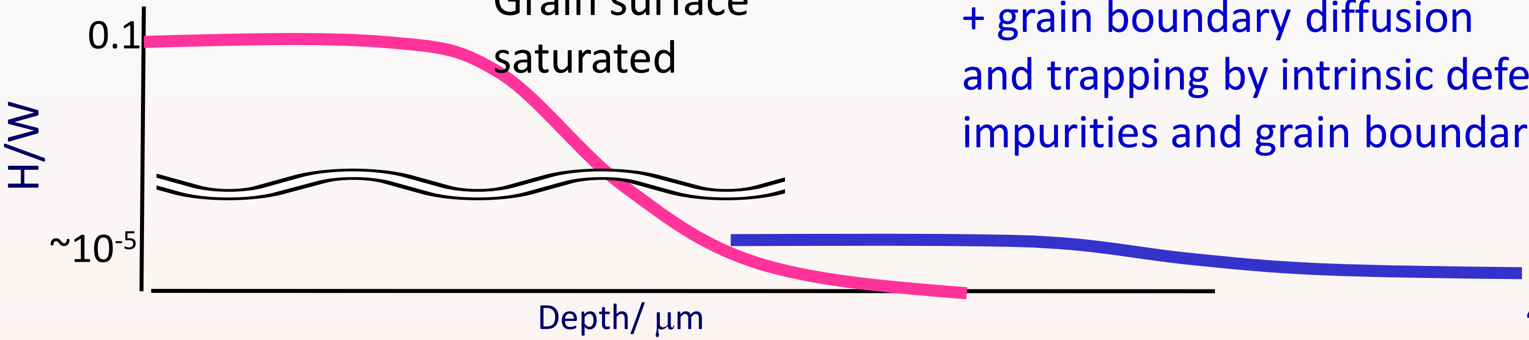
H^+

H_2

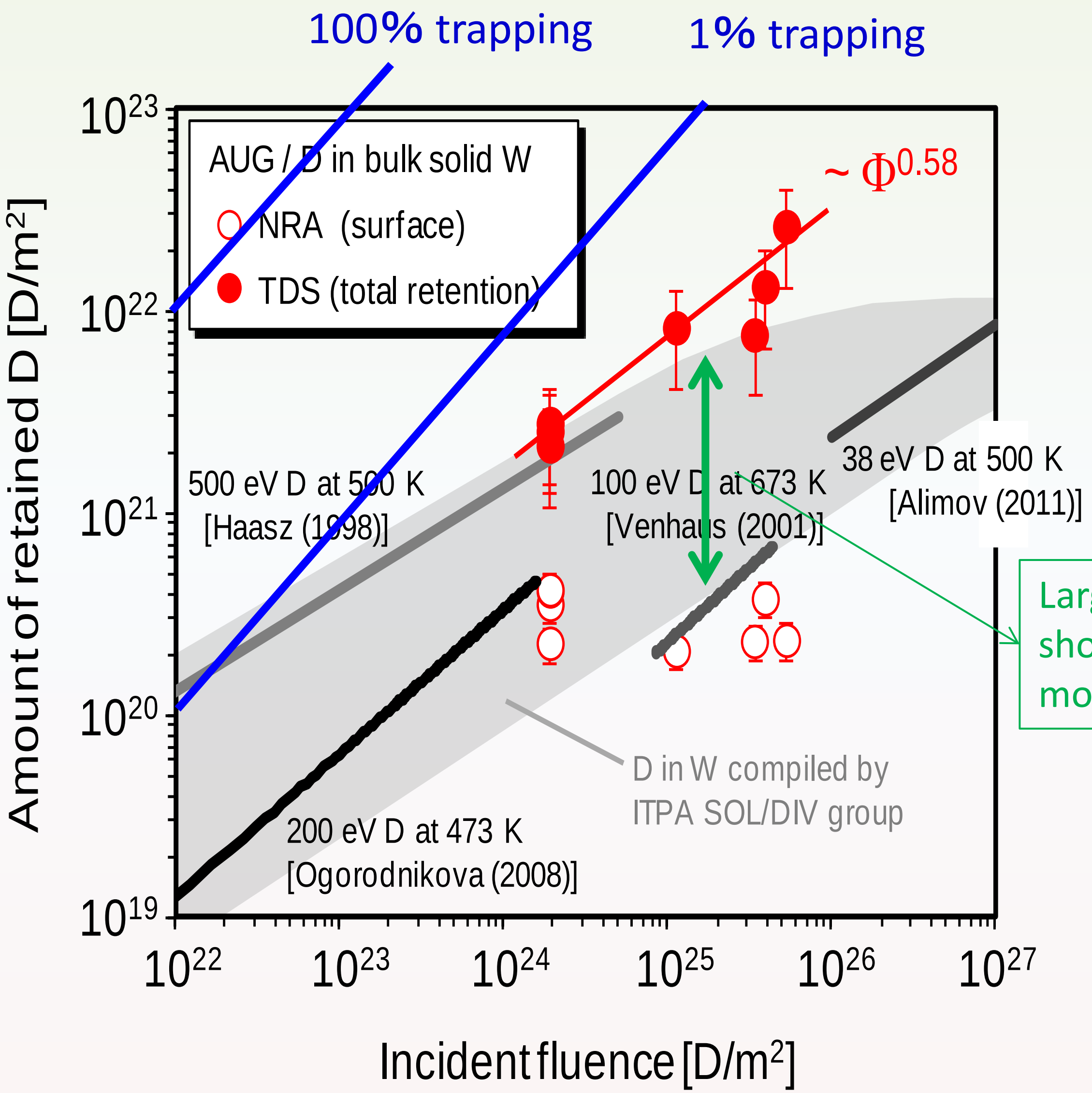
Fully saturated

Grain surface saturated

Penetration by diffusion
+ grain boundary diffusion
and trapping by intrinsic defects,
impurities and grain boundaries



D retention in bulk W



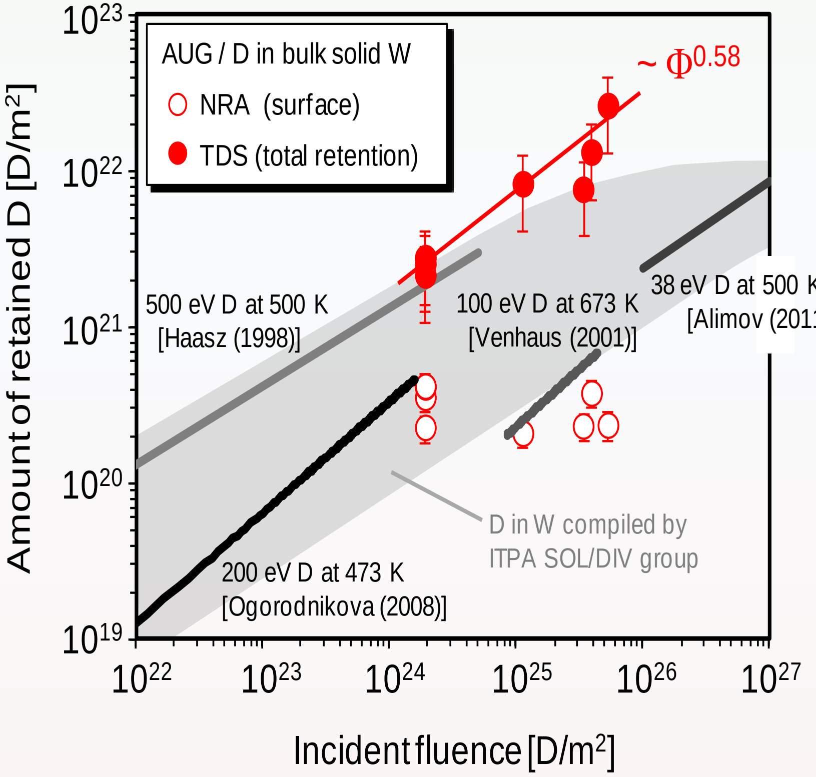
Large differences between NRA and TDS show Importance of H in bulk or depth of more than mm

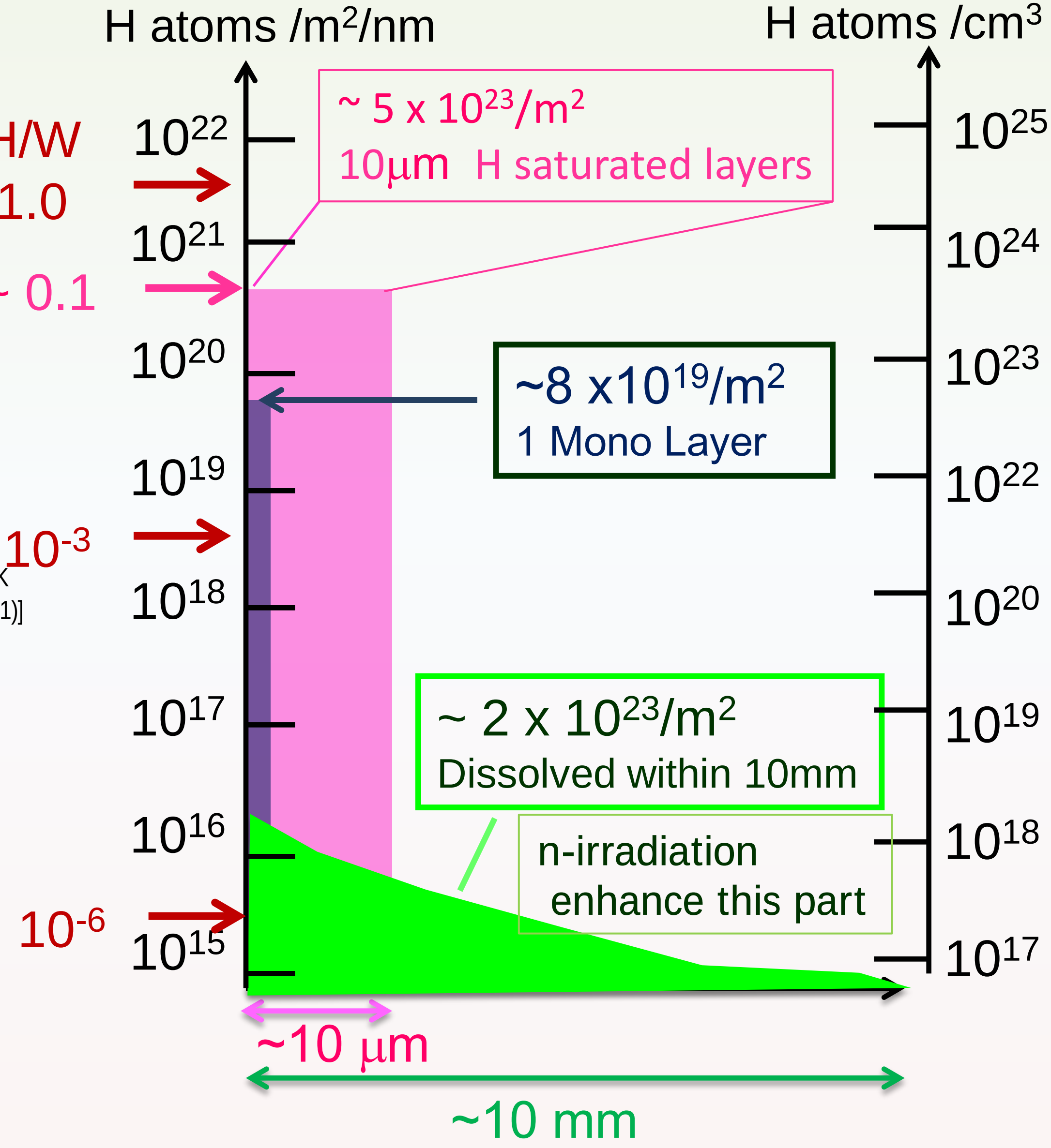
K. Sugiyama et al. Phys. Scripta T145 (2011) 014033

How H retained in W (Tanabe's picture)

ITER Limit 700g ~ 2 x 10^{26}
~ 4 x 10^{23}/m^2 (500m^2)

~ 5 x 10^{22}/m^2
Observed Max





~ 5 x 10^{23}/m^2
10μm H saturated layers

~8 x10^{19}/m^2
1 Mono Layer

~ 2 x 10^{23}/m^2
Dissolved within 10mm

n-irradiation
enhance this part

~10 μm

~10 mm

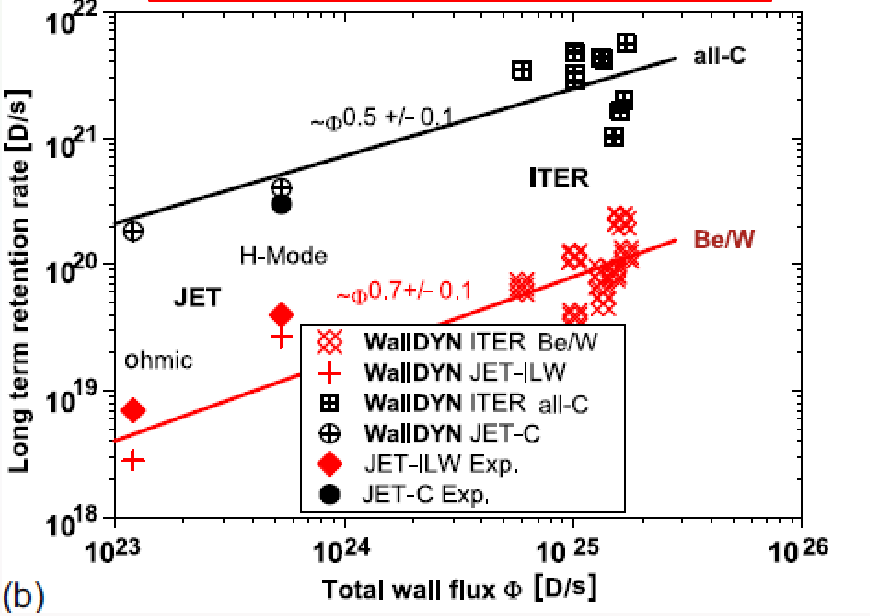
K. Sugiyama et al. Phys. Scripta T145 (2011) 014033

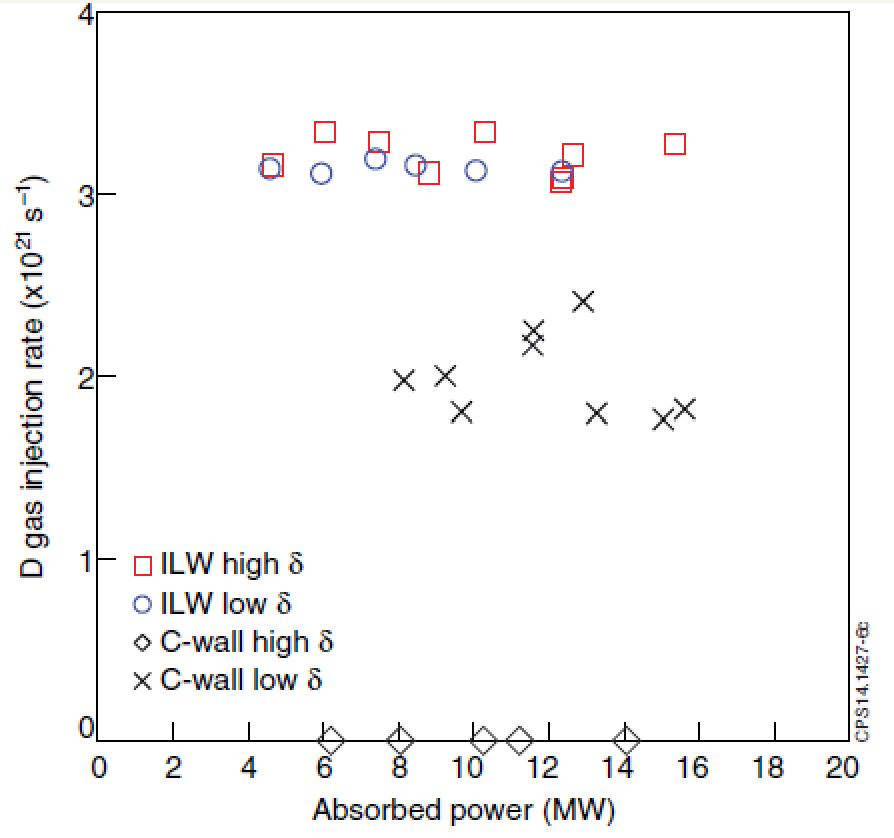
JET-ILW shows significant reduction of fuel retention rate, while requires more throughput!

Typical for a metallic wall; larger throughput with larger recovery after discharge

Total D retention rate

JET-C 2001-2004	6g	3.7%
JET-C 2007-2009	~50g	2.1%
JET-ILW 2010-2012	≦1.5g	≦0.3%





Global gas balance in JETILW benchmarks

and Long-term retention rate predictions for ITER made by WallDYN [57]

Brezinsek, Journal of Nuclear Materials 463 (2015) 11–21

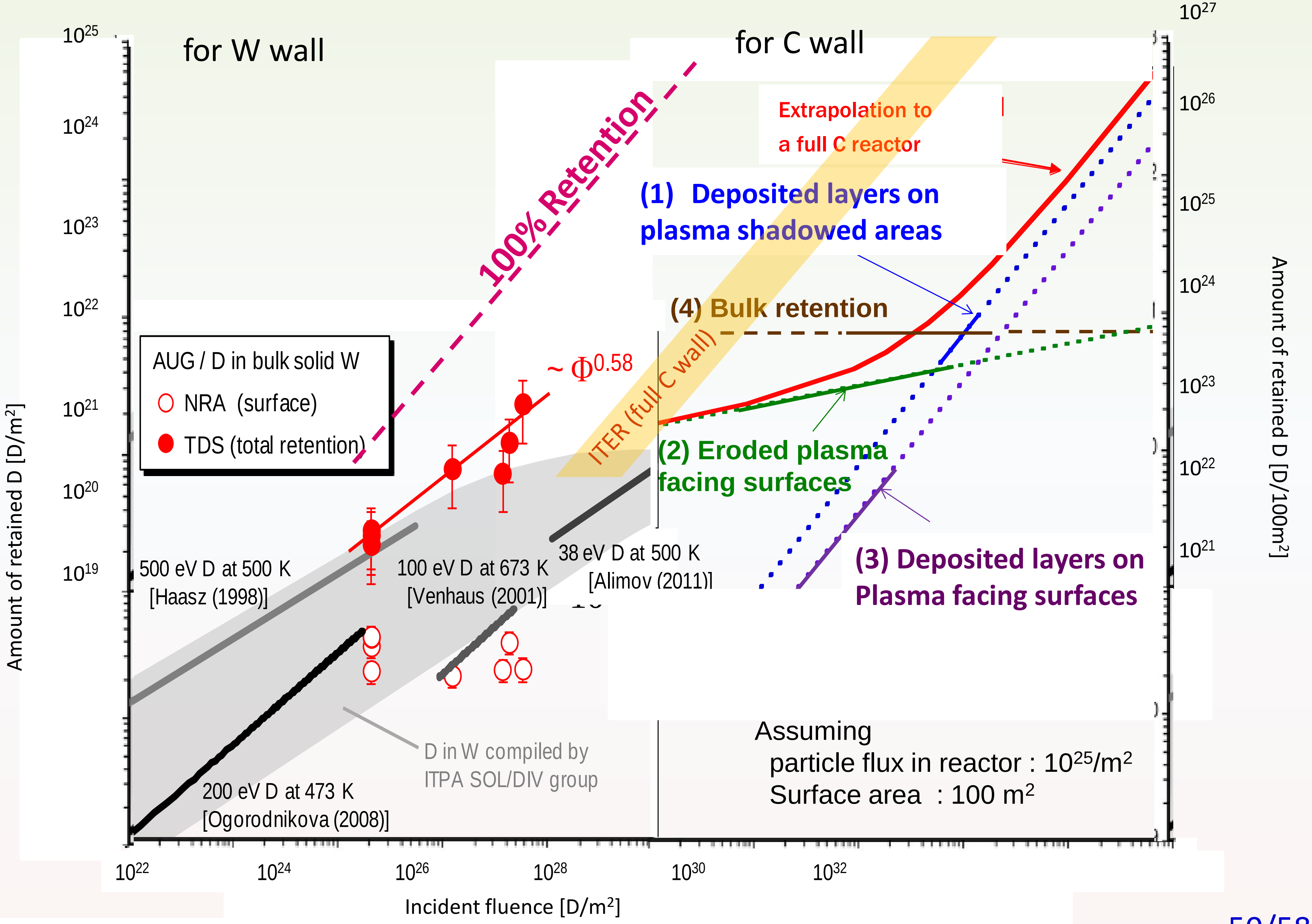
Could be more optimistic, because H retention in Be deposits is large in JET ITER-Like wall

Deuterium gas injection rate

during the main heating phase as a function of absorbed power for the four power scan experiments.

Challis, Nucl. Fusion 55 (2015) 053031 (18pp)

Comparison of estimated fuel retention in an ITER like reactor



K. Sugiyama et al. Phys. Scripta T145 (2011) 014033

M. Yoshida et al. / J. Nucl. Mater. 438 (2013) S1261–S1265

W seems to be a very good PFM

- If the central accumulation of high Z impurities can be avoided by suitable transport control.
- “Prompt redeposition” of high Z atoms is very promising for the reduction of erosion.
- Long range transport and deposition at remote area are small
- Heat deposition is likely reduced for higher Z materials owing to its higher reflection coefficient, which must be confirmed in the future.
- Utilization of higher temperature above DBTT is promising, while embrittlement by grain growth is concerned.

Concerns from PMI aspect

- Plasma operations are carefully done in JET with ITER like wall
- High Z accumulation in plasma center gives high radiation loss could cause disruption
- Dust production due to blistering (He effect)
- No data on hydrogen effect (possible H accumulation at grain boundary)

Concerns from materials aspect

- Local melting (melt layer motion and Electro-magnetic field
- Behavior of re-solidified layers (very brittle) (effect of cyclic heat load and ELM)
- Loss of ductility due to neutron irradiation due to damages, nuclear transmutation, and Helium (decay product of T)
- High activation by neutron irradiation
- Hard to be used as heat sink (or direct cooling) requiring brazing
- Hard to machine and heavy
- High chemical activity to hot water（Accident, WO_3 is volatile ）

7. Criteria for selection of PFM materials of a reactor

1. Impurity release vs. plasma contamination

2. Surface damage

3.T retention

4. Effects of neutron irradiation

5.Maintenance

Low Z and lower erosion are better. However some impurities are required for edge cooling. (Impurity seeding)

W radiates lots if they are localized in the edge

Methods to avoid the accumulation of high impurities are required. (Impurity source and transport are separated)

Material loss by melting

Loss of ductiluty by resolidifcation and recrystalliztion

CFC (on high heat flux areas)

Self –shaping reduces peak heat load, while redeposits at plasma shadowed area are concerns

eithre C or W

Either Low Z (Li, B, an C) or High Z
(Be is not likely used owing to its low MP)

These would not be primary criteria

Criteria for selection of PFM materials of a reactor

T retention

Lower T retention
Most of T is retained in deposits

↓

W
(less long range transort)
Question on easy T recovery owing to bulk retention

↓

CFC at high Temp. use
(lower erosion and small bulk retention)

Effects of neutron irradiation

Lower activation

↓

Carbon
Reduction of thermal conduct. of Carbon is concerned

↓

W
Useage as armor tiles allowing high Temp. operation mitigates neutron effects
(Dimensional change of CFC by n irradiation concerns)

↓

Maintenance

Brazing of armor tiles is difficult to replace.

Is mechanical fixing for easy replacement gives enough heat removal?

C layers allow in-situ repairing

↓

CFC

↑

Critically important for a reactor are
- Low retention for T safety
- Easy recovery for T self-sufficiency

Either Low Z (Li, B, an C) or High Z
(Be is not likely used owing to its low MP)

Concerns on C to use as PFM and their mitigation

Concerns	Remarks	Mitigation/Prospects	
High erosion	Erosion will not continue at same place	Divertor sawing In-situ repairing by CVD, PVD or plasma assisted processes	
	Fine alignment of tiles is required	Plasma shaping will also work	
T retention in tiles	Formation of T saturated layers is limited near surface	Saturated concentration (T/C) significantly decreases with Temp. Isotopic exchange works	
T retention in redeposited layers	Location of redeposited layers is predictable	Tile gap: Tile replacement will reduce	
		Divertor: Installation of cooling plate to take-up redeposition	
Dust formation	Mainly caused by exfoliation of redeposited layers	Chemical instability of C-H bond at higher Temp. reduces formation of redeposited layers	
Neutron damage	Loss of thermal conductivity	High T operation decelerates damage formation	Tile replacement
	Dimensional change		
	Increase of T trapping		Isotopic exchange

Possible use of Carbon as PFM
allowing temperature rise with mechanical fixing to heat sink material

Exercise: Estimation of temperatures of surface (T_1) and boundary (T_2)

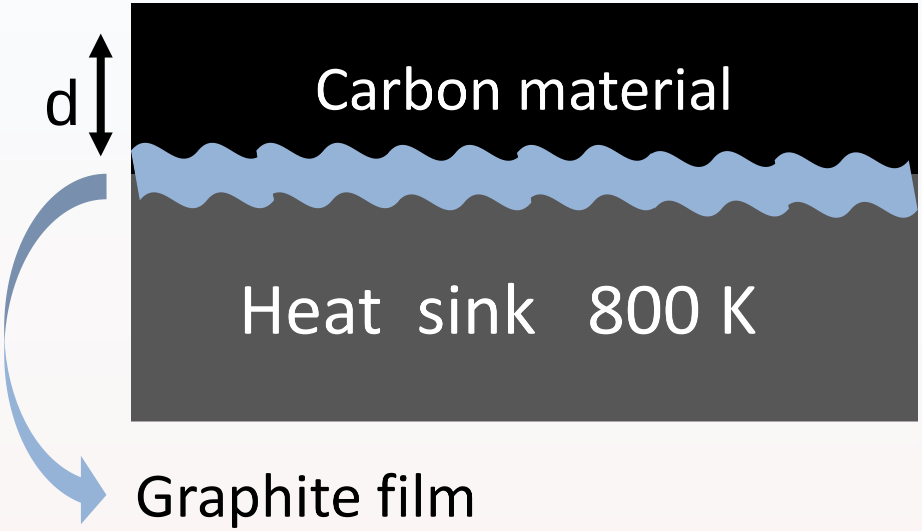
$$Q = \lambda(T_1 - T_2)/d$$ for steady state heat conduction

$$T_1 = \frac{Qd}{\lambda} + T_2$$

$\lambda = 50 W/mK \quad d = 5 \times 10^{-3}\ m$ for carbon armor

$Q = 10\ MW = 10^7 W$ Power input

$$T_1 = 1 \times 10^3 + T_2$$



T_1 = 2800 K

T_2 = 1800K

T_2 = 800K

Graphite film

Thermal contact coefficient 10^4 W/m^{2}K

10^4 W/m^{2}K x ΔT = 10^7 W/m^2 , ΔT =10^3

8. Summary : Consequences of huge heat and particle loads

Erosion and deposition & influence on plasma

Erosion
- Sputtering & Chemical Sputtering
- Radiation enhanced sublimation
- Radiation induced evaporation

Transport or eroded materials(impurities)
- in main and scrape off plasmas
- Plasma contamination

Deposition
- Prompt redeposition
- Line of sight deposition
- Deposition at shadowed area
- Deposition far remote area
  (long range transport)

Deposited materials
- Mixed material (Often non-crystallized)
  used to be referred as Tokamakium
- Incorporation of H (Large T inventory)

Material modification/degradation

Surface melting and evaporation
- Vapor shielding?

Melt layer motion

Materials damage
- Embrittlement
  - once melted layer
  - recrystallization of heat influenced zone
- Degradation of thermophysical properties by neutron damage
- Nuclear transformation
- Increase of fuel trapping

Fuel recycling and T inventory

- Wall saturation
- Density control
- T Fuel self sufficiency, Safety

Influences of these modifications on burning plasma would be significant .
However, power load in present machines is far less than that expected in a reactor .
Hence data extrapolation should be carfully done and ITER should be used as a test bed.

Thank you very much for your attention