KYUSHU UNIVERSITY

先端エネルギー理工学専攻
Department of Advanced Energy Engineering Science

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

Tritium issues in plasma wall interactions

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Organizer

Grand in Aid for Scientific Research, MEXT, Priority area No.467

Tritium Science and Technology for Fusion

http://tritium.nifs.ac.jp/

Tritium

Radioactivity requires safety handling.

Limited resource requires effective breeding and recovery.

3rd International ITER Summer School, Provence Univ. July 22-26, 2009

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Part I Tritium issues in a fusion reactor

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A brief introduction of our research project started at 2007 in Japan (Tritium is so important)

DT fusion reactor (Ignition and continuous burning)

$$D + T = {}^{3}He\ (3.7MeV) + n\ (14MeV)$$

To establish reliable and safe tritium fuel cycles and safe tritium confinement to build economic and safety fusion reactor

Encouraging yang scientist and students

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Grant in Aid for Scientific Research

Tritium Science and Technology for Fusion Reactor

Organizer Professor Tetsuo Tanabe
Kyushu University Faculty of Engineering Science

Research purpose

The main aim of this project is to establish tritium safety in a D-T fusion reactor. Since huge amount of radioactive tritium must be introduced into the reactor as a fuel, we are facing to lots of safety concerns newly appeared to be solved.

Main efforts will be to establish tritium safety in (1) a fueling system keeping continuous D-T burning, (2) tritium exhausting, recovering and refining processes, (3) a tritium breeding system with a breeding rate over 1.05, and (4) tritium monitoring and accounting systems.

In addition, easy isotopic exchange reactions of tritium with hydrogen in water and hydro-carbons result in the contamination of the systems, which require decontamination techniques. The project also aims to provide new insights into basic tritium science and technology.

Limited resource requires safety T breeding system compatible with power production and T breeding

Reactor

- **Recycling of fugue amount of T**
- **Safety confinement to avoid possible contamination**
- **Difficulty of extrapolation of limited experience of T handling to fusion system**
- **Poor understanding of isotope effect**

Production of hazardous inorganic tritium

Contamination by permeation and leakage

Multi step contamination

Safety Confinement for regulation

Containment Physically & Chemically

ITER at France and a Test reactor in Japan require large numbers of tritium experts.

2. Characteristics of a DT reactor as an energy source

Already 50 years has passed after finding nuclear reactions give energy.

Fission reactors are already established as energy sources.

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

Significant amount of energy is required to overcome Coulomb potential.

The first priority has been plasma confinement to establish DT burning, and we will soon attain Q=10 in ITER.

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

Lots of engineering issues are remained to be solved.

Tritium safety and economy are critical issues.

Hydrogen related 5 fusion reactions

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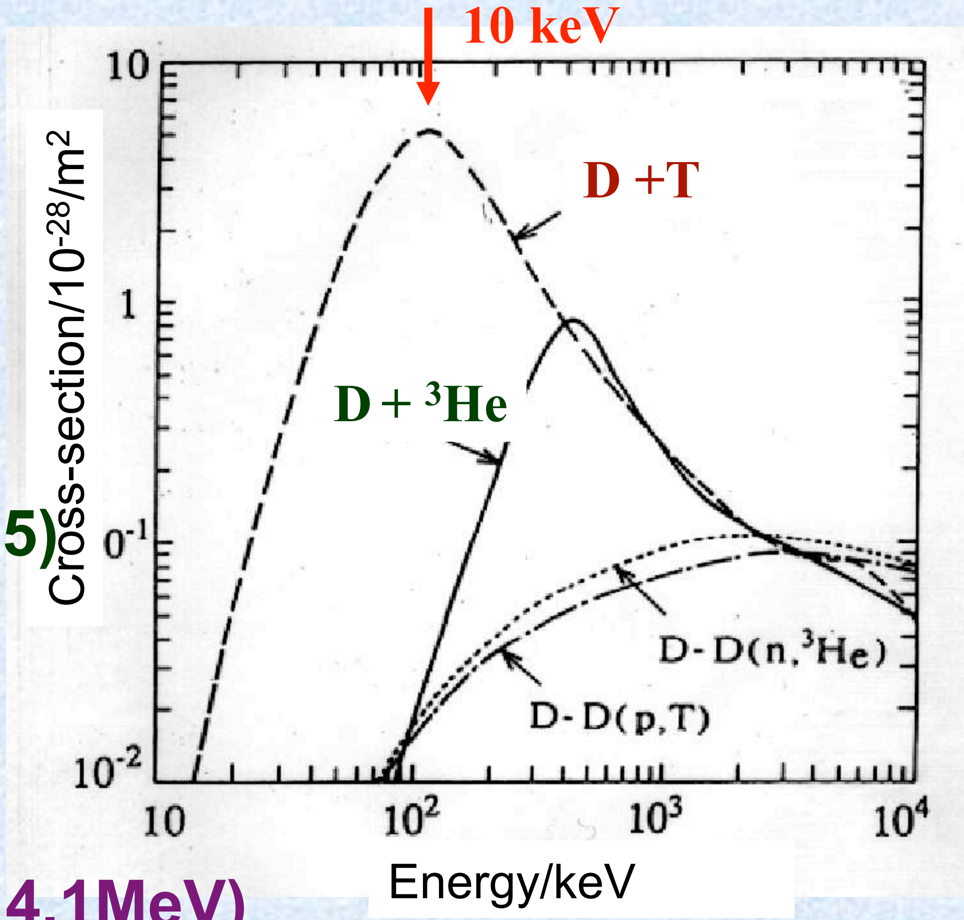
D +T → ^{4}He + n + 17.6MeV (1)

D + D → T + H + 3.98M (2)

D + D → ^{3}He + n + 3.25MeV (3)

T + T→ ^{4}He +2n + 11.3MeV (4)

D + ^{3}He → ^{4}He + H + 18.3 MeV (5)



Easiest reaction is DT reaction (1)

D +T → ^{4}He (3.5MeV) + n(14.1MeV)

plasma heating Energy and T breeding

The D ^{3}He reaction is very much attractive for *no neutron production*, though accompanying DD reactions do produce it.

Comparison of fission and fusion as energy sources

Fission reactor

Most of things;
energy conversion,
fuel breeding,
waste-confinement
in fuel pins of diameter of **~ 1cm**

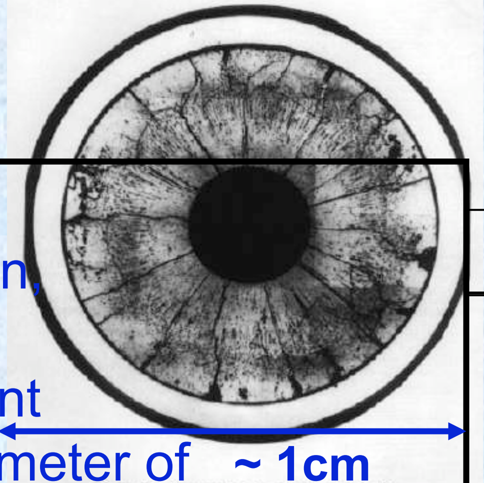


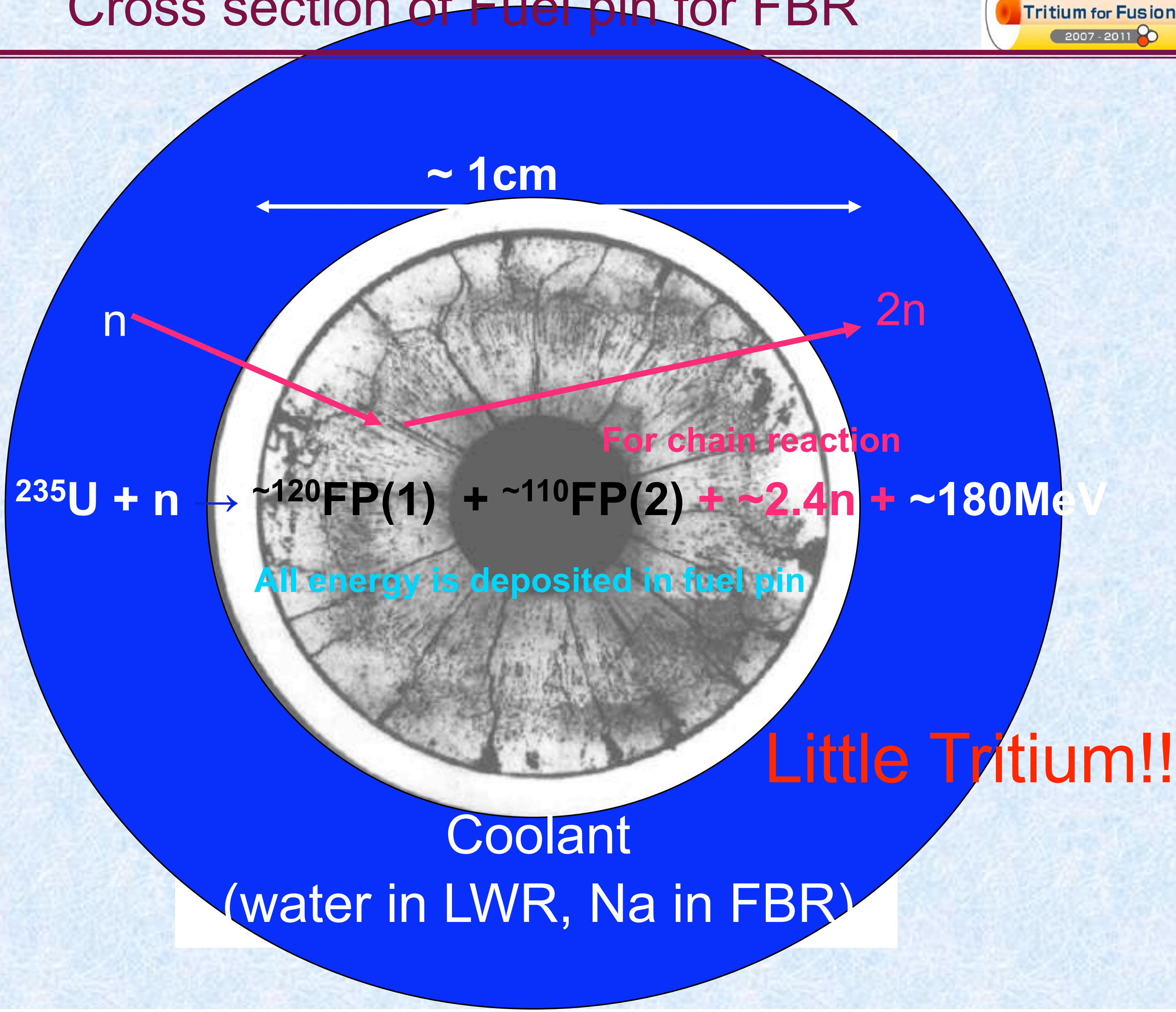
Fig. 3.9(a) Cross-section of an irradiated UO_2 fuel element, showing restructuring, ANL.

Fusion reactor

Fugue volume of tritium handling system with less energy density.

	Fission reactor	Fusion reactor
Energy Input	**Nearly zero**	**Huge energy required** **Poor fueling efficiency**
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, we need neutron multipliers (Be, Pb).** **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.**

Cross section of Fuel pin for FBR



T resource is very limited → need T breeding

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

plasma heating Energy and T breeding

– Deuterium can be extracted from natural water (SMOW (standard mean ocean water) contains 0.016% D)

– Tritium must be imported (limited) or bred internally from lithium

- 56 kg tritium is required per GW year (thermal) of fusion power
- About 100 g tritium is produced per year in a standard CANDU fission unit
- 20 to 25 kg tritium (mainly in Canada) will be available for operation of ITER
- Tritium must be bred by reactions in blanket systems

$${}^{6}Li + n \rightarrow T + {}^{4}He + 4.8MeV$$

$${}^{7}Li + n \rightarrow T + {}^{4}He + n - 2.5\ MeV$$

$${}^{9}Be + n \rightarrow 2n + 2\ {}^{4}He - 2.5\ MeV$$

$${}^{A}Pb + n \rightarrow 2n + {}^{A-1}Pb - 7\ MeV$$

- Overall breeding ratio is expected to be above ~1.1 (must)

Very hard to attain

3. General safety issues in a fusion reactor

Fusion Safety Issues (General) are
mostly owing to tritium and neutron activated materials
because

- **The Fusion Process Is Inherently Safe**
  - **No chain reaction**
  - **Reaction is thermally self-limiting**
  - **Limited to a few second burn without re-fueling**
  - **Power/energy densities in the reactor and plasma are low**
  - **Reaction products**
    - **Helium (totally inert)**
    - **Neutrons**
      - **Used to breed tritium**
      - **Absorbed in the surrounding material**

- **Most serious hazard involve the tritium fuel and activated dust from erosion of plasma facing components**

Fusion Safety Issues (General) Cont.

- **Hazard and Containment**
  - **Principle of defence-in-depth**
    - **Vacuum vessel**
    - **Cryostat**
    - **Building ventilation systems (sub-atmospheric condition)**
  - **Passive safety features (natural physics) are used as extensively as possible**
    - **In case of active cooling system failure, decay heat from activated materials is low enough that all in-vessel components can be cooled by natural convection**
    - **Reactor "melt-down" is physically impossible**

- **Environmental Impact**
  - **Currently, materials are not optimized for low-activation under neutron irradiation**
    - **Can be recycled for re-use after 50-100 years**
  - **In the future, material optimized for low-activation can be readily recycled for use in fusion power-plant reactors.**

Public Safety

Emission of Tritium must be As Low As Reasonably Achievable (ALARA)

– **Under normal operation:**

- **Total releases will cause doses below 1% of that of natural background radiation: ~ 2 mSv/year, or 200 mrem/year.**

–**Under the worst case: the most severe hypothetical event, or the holy-Moses-oh-my-God-we-are-all-done-for scenario:**

- Fusion reactor site boundary dose will be less than 50 mSv (5000 mrem).

•In comparison:

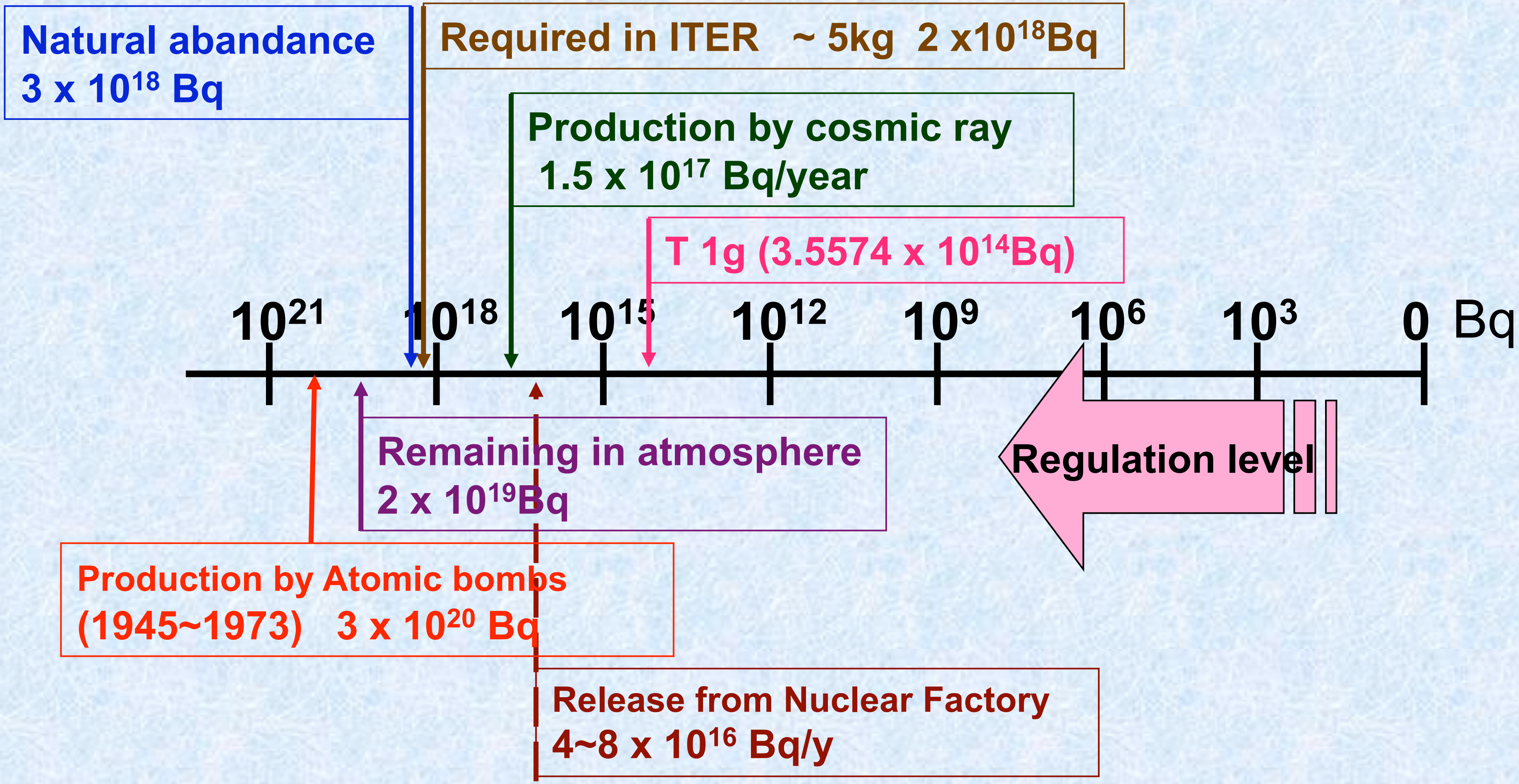
- 50 mSv/year is the US NRC dose limit for adults working with radioactive material.

- 100 mSv is considered "low-dose"; correlation with adverse biological effect (e.g. cancer) currently could not be established.

- Plant workers and fire fighters battling the fire at Chernobyl received 700~13400 mSv of radiation; 20% of them died from radiation effects.

Concerns are coming from Tritium and nuclear activation

Tritium Abundance
(limited resources and regulation for safety)



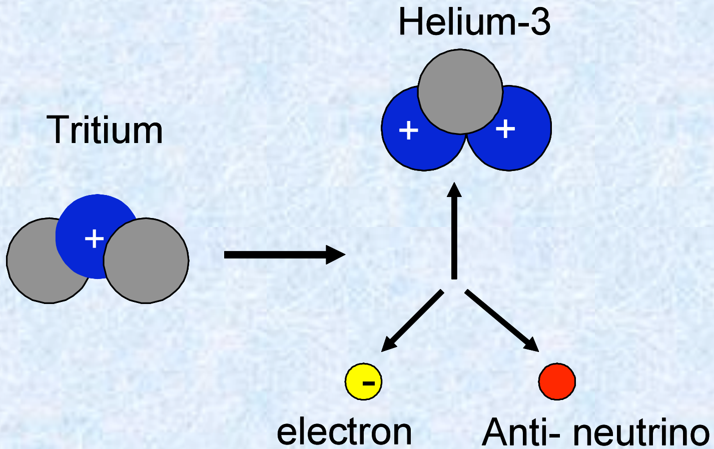
Quantitative analysis	Radiation heat; Gravimetric; Volumetric (PVT); Disintegration (dps)

No single method can cover whole range.
Poor resolution inhibits cross-cheek

4. Tritium as a fuel of a DT fusion reactor

Tritium ($^3T \rightarrow {}^3He + \beta$ electron)

Half life $t_{1/2}$ = 12.323 ∓ 0.004 years
(about ~5.5% is disappearing in a year)



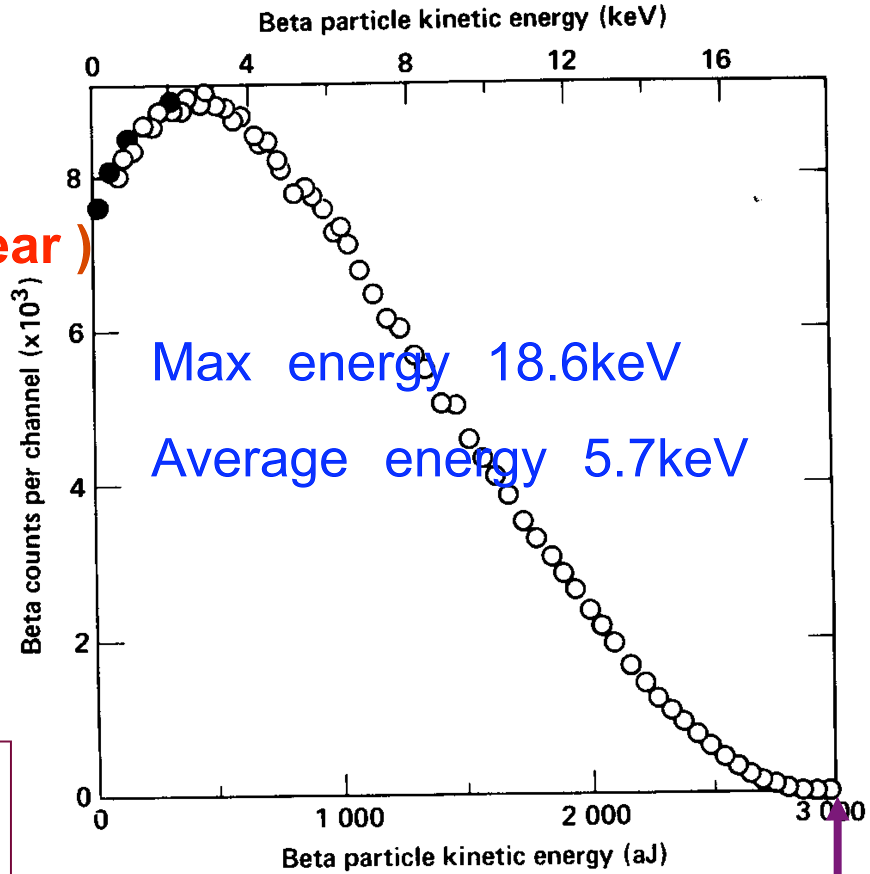


Fig. 16.2. A high-resolution spectrum of the tritium beta-particle kinetic energy (open circles). The closed circles are special low-energy points from another source. The spectrum was measured by J. J. Simpson, University of Guelph.

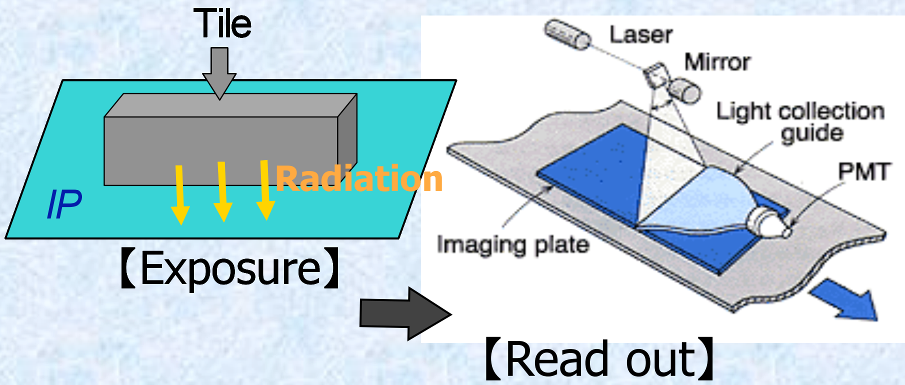
Precise measurements of edge energy would give neutrino mass

Maximum range of electron
Air 6mm
Metals <~1mm
Shielding of tritium radiation is not really a issue (Except direct exposure of organs)

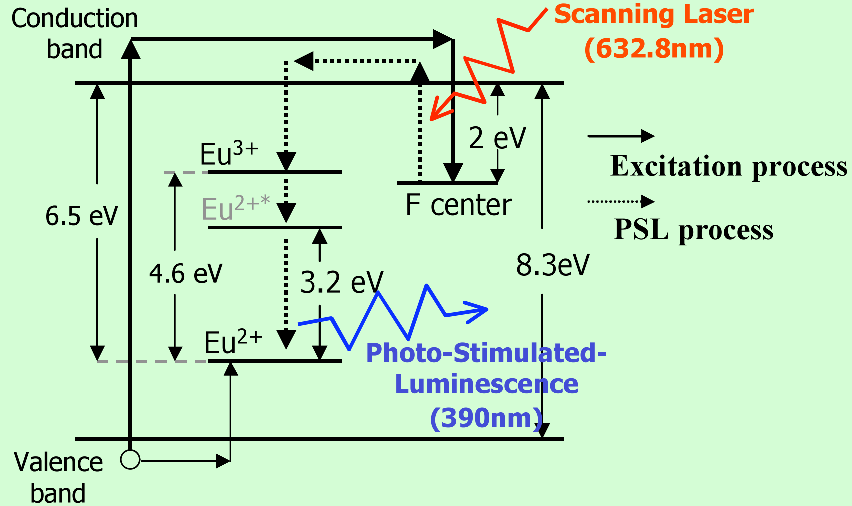
- Electrons emitted to neighboring molecules would enhance some chemical reactions.
- Effect of self irradiation would appear only at very high conc.
- **Decay heat : 324 mW/1g could enhances T release from solid**

Detection of T is rather easy (ex. Imaging Plate Technique)

to obtain the 2-D image or profiles of radioactivity image & profiles.

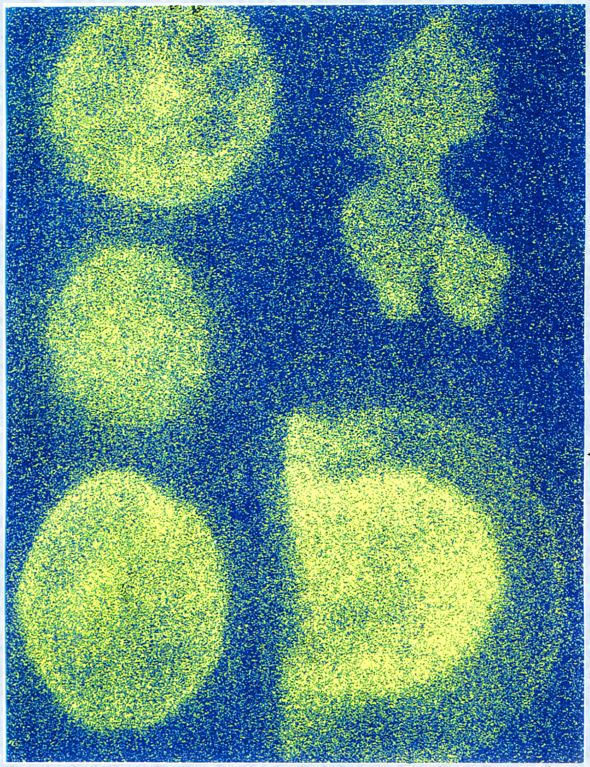


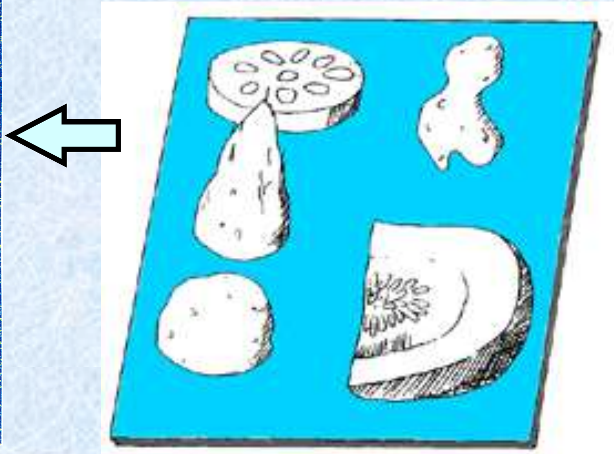
IP is a 2-D radiation detector with high sensitivity and resolution, utilizing a photostimulable phosphor ($BaFBr:Eu^{2+}$).



* Fig. Energy level diagram for the PSL mechanism in $BaFBr:Eu^{2+}$

Potassium (K) in foods





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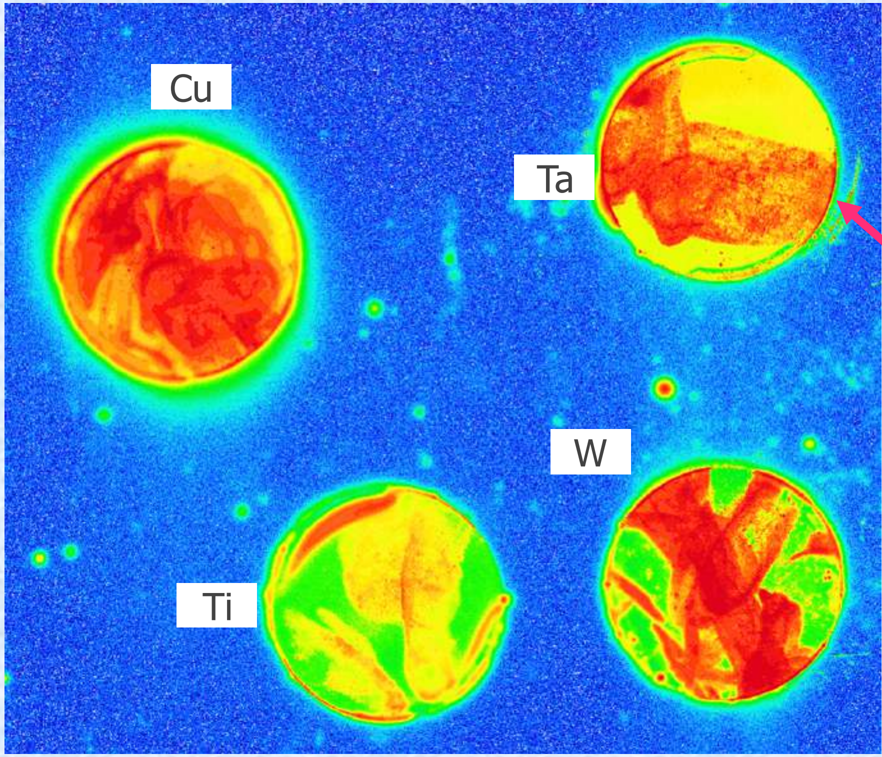
Cross (Multi-step) contamination

Ex. Contamination by gloves in safety box

Metal plates exposed to D plasma in TPL and handled in a T handling glove box



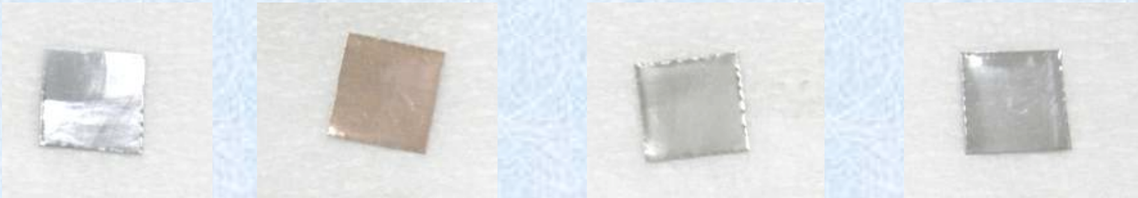
Traces of glove fingers

Possible contamination by permeation

T precipitated on surface behaves differently from bulk T
(when bulk concentration is very low)

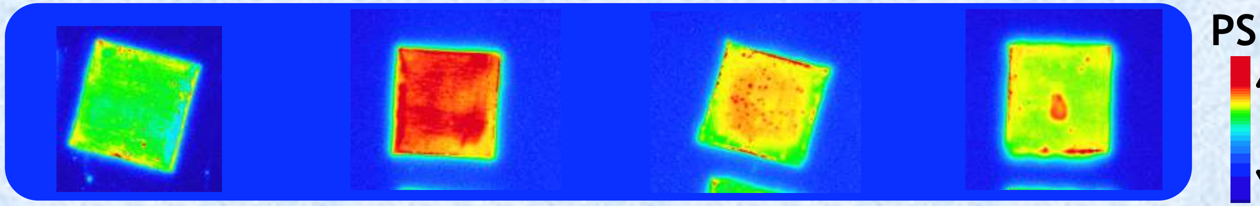
Fe Cu Ni Mo

Photo surface



10 mm

Tritium distribution



PSL intensity
High
Low

	Fe	Cu	Ni	Mo
Tritium activity	704	2380	1070	811
H solubility at 873K (H/M)	10^{-5}	10^{-5}	10^{-3}	10^{-7}

At high concentration, both behave similarly.

But it is quite hard to detect bulk T

Why T surface precipitation occurs?

Production of hazardous inorganic tritium

T can easily radio chemically replace the ubiquitous lighter hydrogen isotopes, above all the protium (H) / deuterium in water and hydrocarbons in air

$$HT + H_2O = HTO + H_2 - \Delta G$$

$$HT + CH_4 = CH_3T + H_2 - \Delta G$$

Also, any solid surfaces absorb water molecules resulting in

surface precipitation of T

$$M + HTO = MOT + \frac{1}{2}H_2 - \Delta G$$

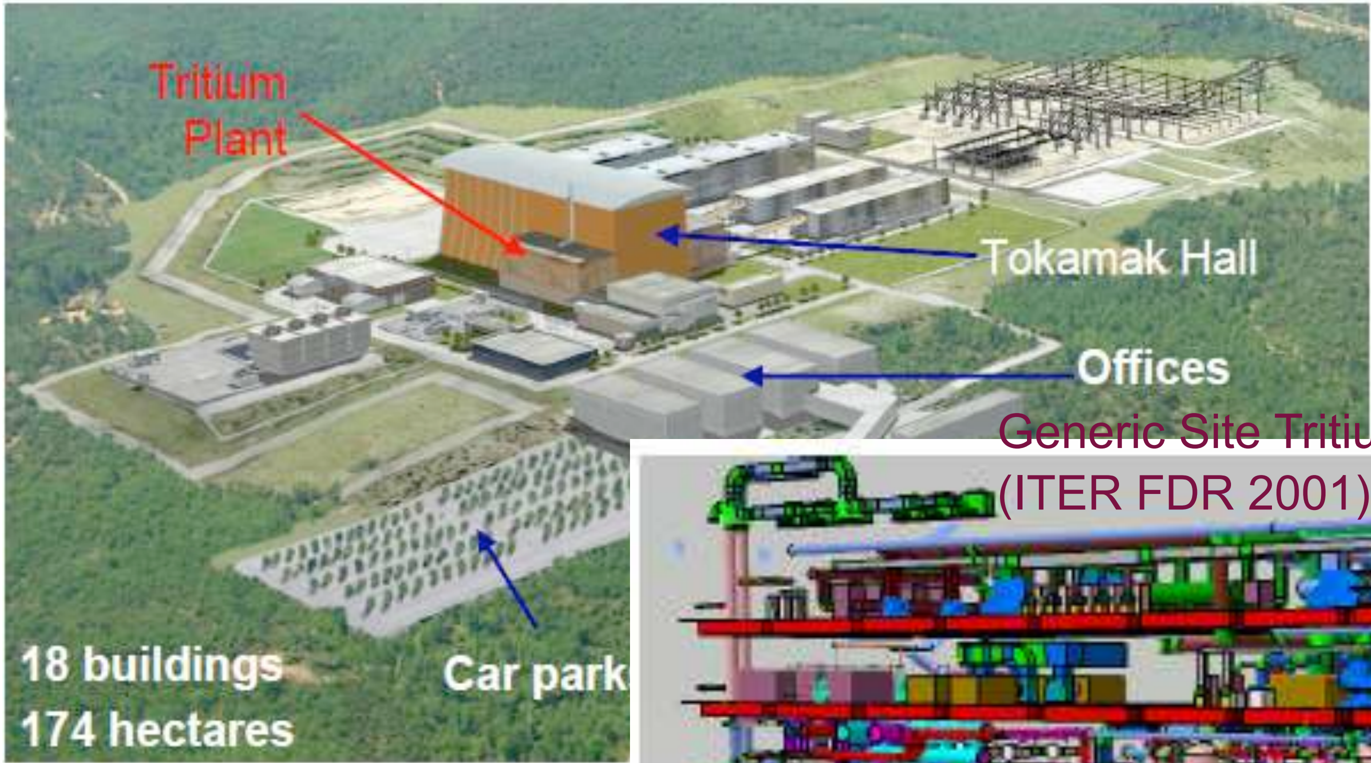
Exposure of skin is not so important owing to thin penetration of β-electron, while tritium in organs are dangerous

In case, T is going in your body, you should drink water to remove it. For that purpose, Beer is very good!

Tritium and ITER

- First fusion machine fully designed for equimolar DT operation
  - Tokamak vessel will be fuelled through gas puffing & Pellet Injection (PI)
  - Neutral Beam (NB) heating system will introduce deuterium
- Employing DT as fusion fuel has quite a number of consequences
  - It causes alpha heating of the plasma
  - The fusion reaction will eventually provide energy
  - Closed DT loop is required due to the small burn-up fraction
    - Primary tritium systems for processing of tritiated fluids
    - Auxiliary systems necessary for the safe handling of tritium
  - Multiple barriers vital for DT confinement
    - Atmosphere & Vent Detritiation are crucial elements in the concept

After all a rather complex chemical plant, i.e. the Tritium Plant of ITER is needed for deuterium-tritium fuel processing

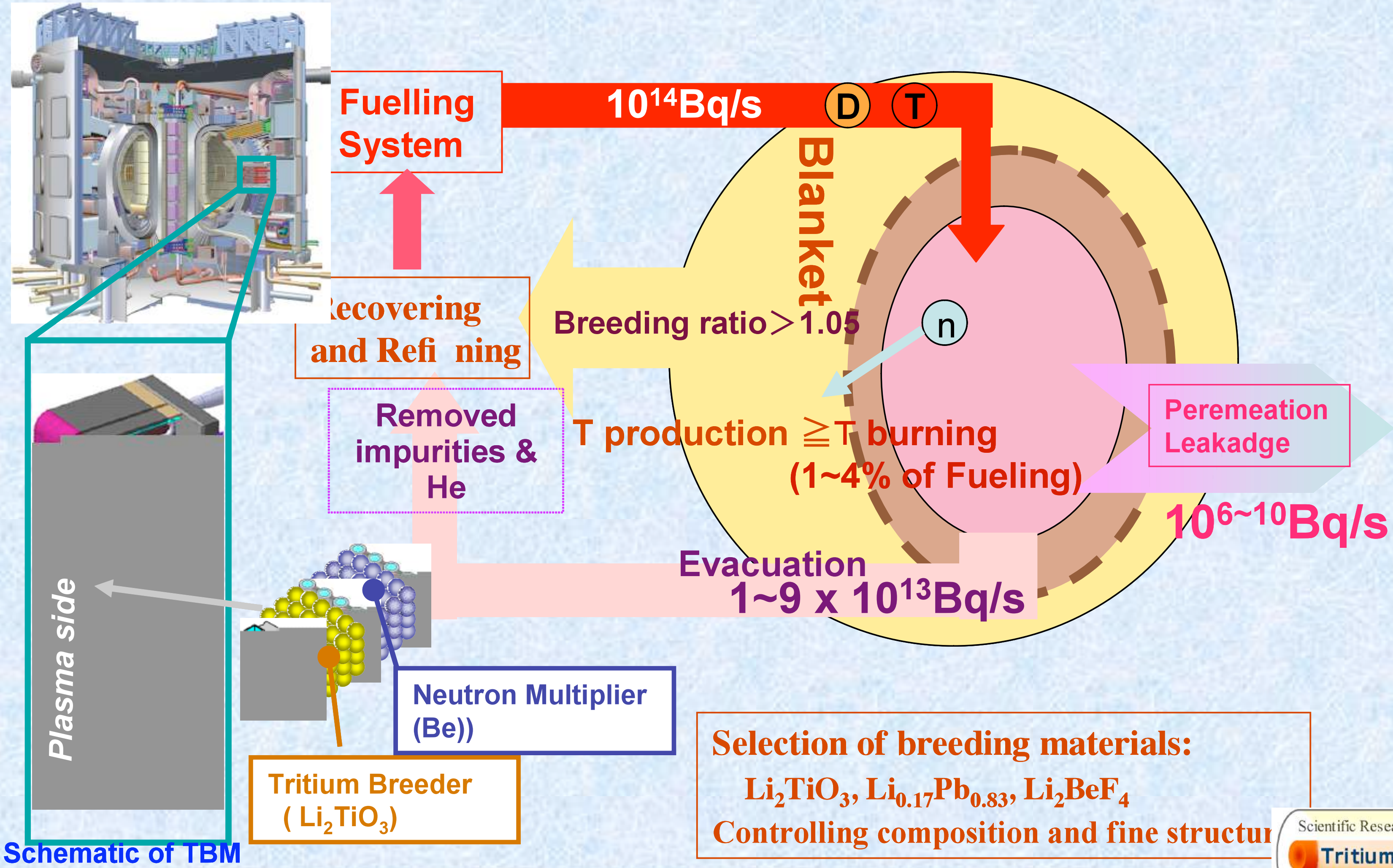


Generic Site Tritium Plant Building Layout (ITER FDR 2001)

- **Dimensions**
  - **Length: 79 m**
    Width: 20 m
    Height: 34 m
- **Space occupation**
  - **HVAC: 18%**
  - **Detritiation systems 16%**
  - **Tritium processing systems 30%**
  - **Non Tritium Plant systems 21%**
  - **Non process areas 15%**

Tritium issues relating fuel cycles and T breeding

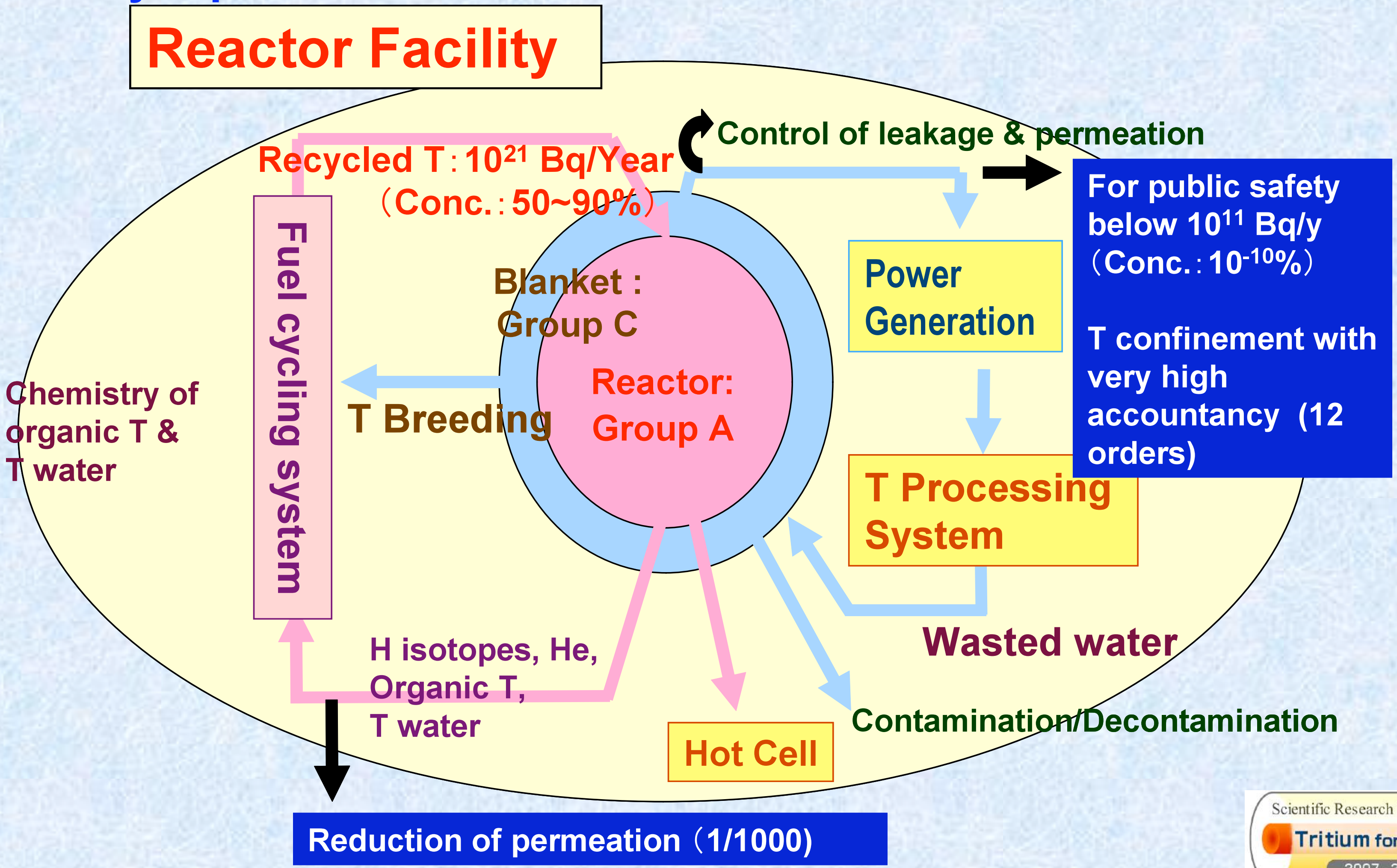
- Tritium breeding with enough margin and compatible with energy conversion
- Limited resource of Tritium (CANDU reactors are the main source)
- Tritium recovery in fuel cycles and breeding systems and its refinement



Selection of breeding materials:
Li_2TiO_3, $Li_{0.17}Pb_{0.83}$, Li_2BeF_4
Controlling composition and fine structur

Tritium relating issues in power generation and surroundings

- **Physical confinement and Safety confinement**
- **Detritiation and/or decontamination**
- **Safety reposition**



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Difficulties related to tritium summary I

- Difficulty of detection and quantitative analysis measurement with high accuracy.
- No way to measure tritium in bulk except combustion detection and calorimetry.
- T behavior in a DT reactor might not be simulated by that in DD plasma machine
- Large mass difference among all hydrogen isotopes
- Tritium breeding must be compatible with energy conversion (or economic)
- Tritium is chemically very active and react with most of impurities, in particular water and hydrocarbon molecules, in air to make more hazardous.
- Permeation and leakage are unavoidable

Summary of Part I (Tritium in Fusion)

Amount to be handled $10^{1} \sim 10^{17}$ Bq

monitoring 1kBq release

Temperature $10^{1} \sim 10^{9}$ K

Pellet(20K)、 Gas at RT(300K)、Plasma ($10^{5} \sim 10^{9}$K)

Characteristics of Tritium

Chemistry of excited state and non-equilibrium thermodynamics

Effect of β electron emission and/or radiation heat

Defect formation by electron excitation and He production

Adsorption, solution, diffusion and permeation in materials

Reacts with impurities to produce inorganic hazard

Difficulty in quantitative analysis (accountancy)

No way to measure tritium in bulk except combustion/calorimetry.

Counting of disintegration（ $1 \sim 10^{6}$Bq limited to T near surface）

Mass and pressure measurements

Radiation heat measurement（accompanying large error）

Summary part I (Tritium in Fusion) Cont.

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Tritium handling system, which uses mostly established techniques, can be build for ITER or even reactor.

However, handling of huge amount of tritium in ITER gives somewhat different problems. (Mostly relating tritium behavior in tokamak)

→ Huge inventory in tokamak and its accountancy

→ Controlled fuelling of DT

→ Possible permeation and leakage leading to cross-contamination

→ Contamination of remote handling system

Most of tritium problem is directly related to the safety of operators and/or professionals. But public safety does not seem to become significant problems.

Tritium breeding must be compatible with energy conversion (or economic)

It should be metioned that we are facing a world wide lack of experts in tritium science and technology.

Tritium in burning plasma

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Inefficient fuelling, Inefficient fuel cycling system

D and T are different in fuelling efficiency, escaping flux, pumping speed
D and T must be separately fuelled

Difficulty in controlling DT ratio 1 in plasma to attain efficient burning

D, T concentration

Quantitative evaluation of D and T in plasma center is not easy

Plasma opacity could disturb optical measurements like Thomson scattering

Feed back from neutron yield

Possible but quite dependent on confinement time which could be significantly different for D and T,

Influence of toroidal and poloidal inhomogeneity

Fugue in-vessel inventory

Significant isotope effect among H, D and T due large mass differences

Effect of different mass on velocity and flux among hydrogen isotopes gases

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Simple molecular kinetics tells that velocity for D and T at the same energy different. So as rotational and vibrational state are.

Maxwell-Boltzman's law gives

$$\bar{v} = \sqrt{\frac{8RT}{\pi m}}$$, hence $$\overline{v_H} / \overline{v_D} = \sqrt{2} \qquad \overline{v_H} / \overline{v_T} = \sqrt{3}$$

Molecular kinetics gives incident flux to wall surface under pressure *P*

$$J = nv = \frac{P}{(2\pi mkT)^{1/2}} \qquad J_H / J_D = \sqrt{2} \qquad J_H / J_T = \sqrt{3}$$

Isotope effects

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Mass ratio of H, D and T is 1:2:3

under the same pressure $v_H / v_D = \sqrt{2}$ and $v_D / v_T = \sqrt{3/2}$

$$\phi_H / \phi_D = \sqrt{2} \qquad \phi_D / \phi_T = \sqrt{3/2}$$

to give the same flux $p_H / p_D = 1/\sqrt{2}$ and $p_D / p_T = \sqrt{2/3}$

Relating to

Different confinement	Outgoing flux ratio would be SQR(2/3)
Impinging energy to wall surface	?
Reflection coefficient	May be SQR(3/2) but no data for T
Recycling flux ratio	Unknown retention time
Pumping speed ratio	For mechanical pumping SQR(3/2) Unknown for cryo-pump

Tritium retention (solubility, diffusivity and permeability, trapping effect)
Surface residence time

Part II Tritium issues in plasma wall interactions

How to extrapolate results on hydrogen retention
in present tokamaks to ITER and beyond

1. Tritium retention on plasma facing materials caused by DT experiments in TFTR and JET

2. Behavior of Tritium produced by DD reactions
   (could not be used to simulate behavior of T fuel)

3. Deuterium and hydrogen retention in JT-60
   for understanding of DT fuel

Estimation of in-vessel tritium inventory includes very large error and uncertainty

- Evaluation of **hydrogen retention** in present tokamaks is of high priority to establish a **database** and **a reference for ITER** (400 s…usually 10-20 s today).

- T retention constitutes an **outstanding** problem for ITER operation particularly for materials **choice** (low Z or high Z ?)

- A **retention rate of 10%** of the T injected in ITER would lead to the in-vessel T-limit (350/700g) in **~35/70 pulses**. (every ~ **35/70** shots require removing of in vessel T)

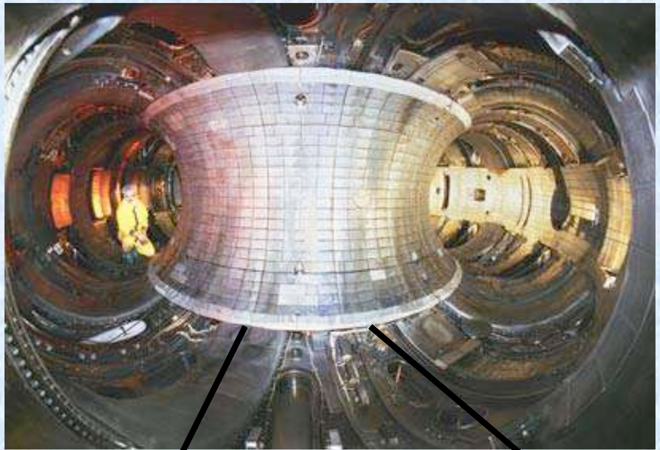
- Retention rates of this order **or higher** (~20%) are regularly found using **gas balance**.

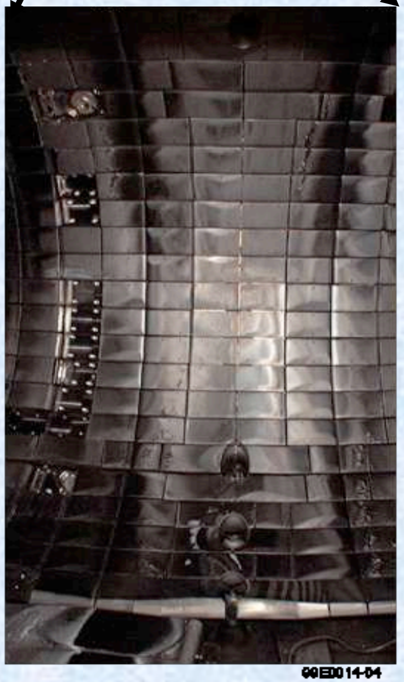
-Retention rate **often lower** (3-4%) are obtained using **post mortem analysis**

- **T breeding can not compensate such high inventory**

II-1. Tritium retention on plasma facing materials caused by DT experiments in TFTR and JET

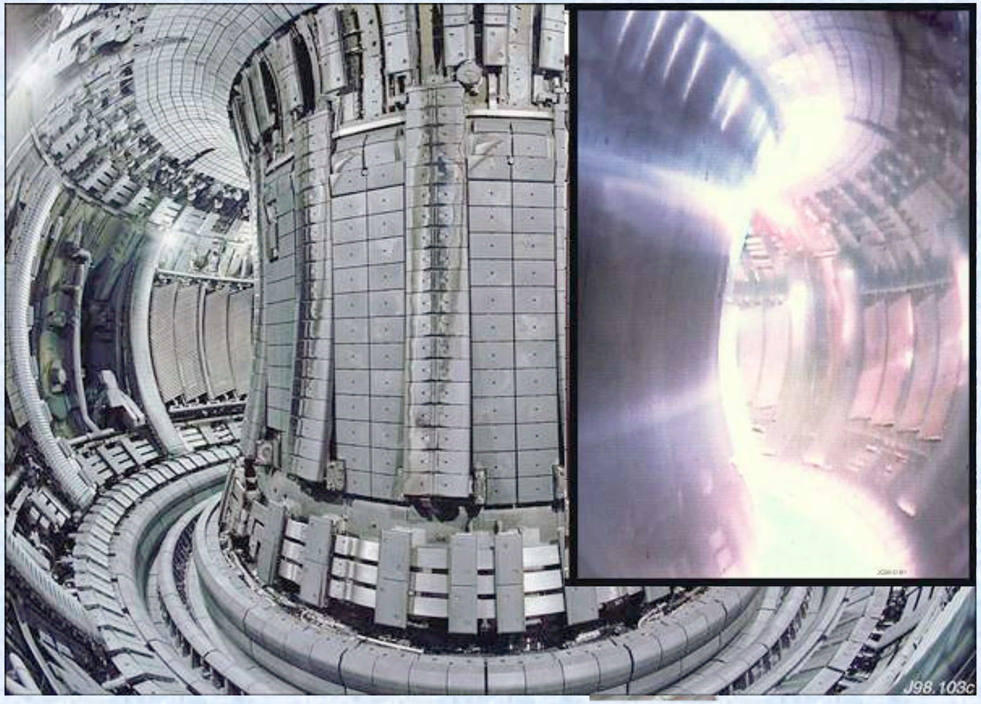
TFTR : a limiter tokamak



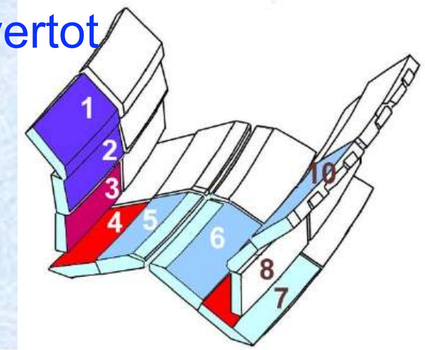


JET : a divertor tokamak

EUROPEAN FUSION DEVELOPMENT AGREEMENT

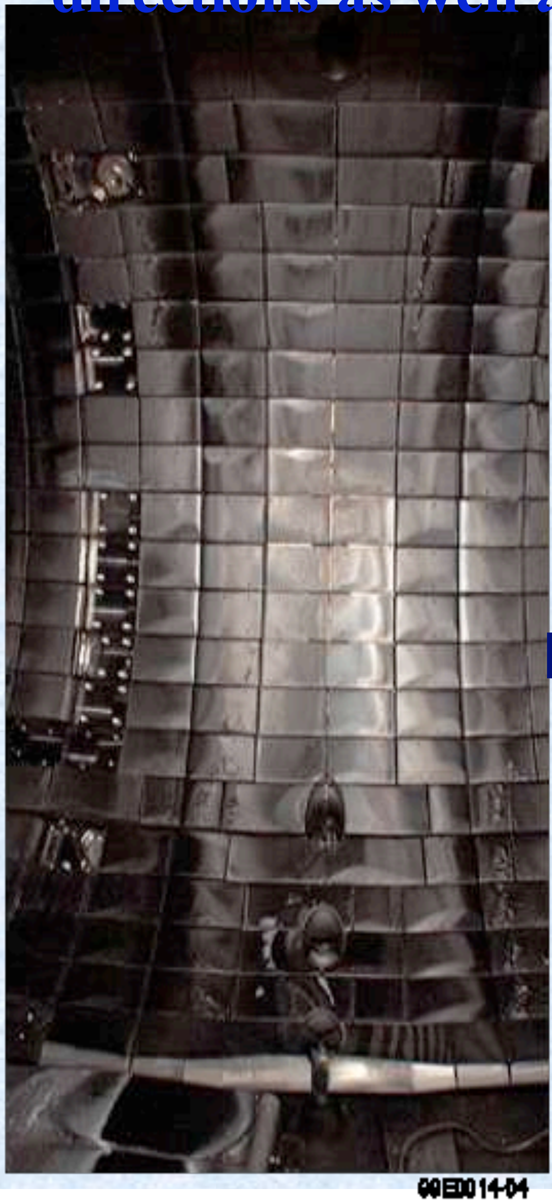


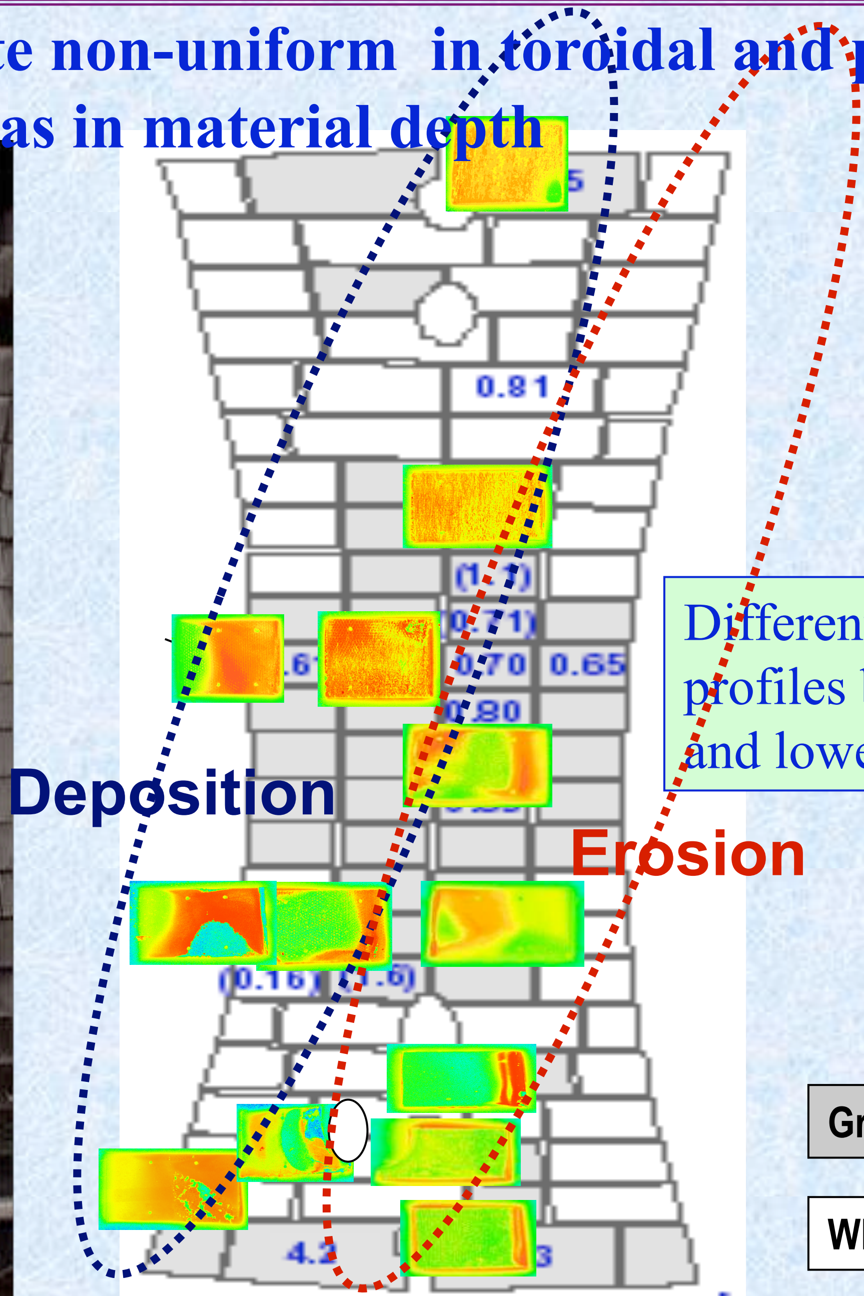
DTE Campaign using
MarkII-A divertot



Experience of DT discharges in TFTR

T retention is quite non-uniform in toroidal and poloidal directions as well as in material depth





Different deposition profiles between upper and lower area

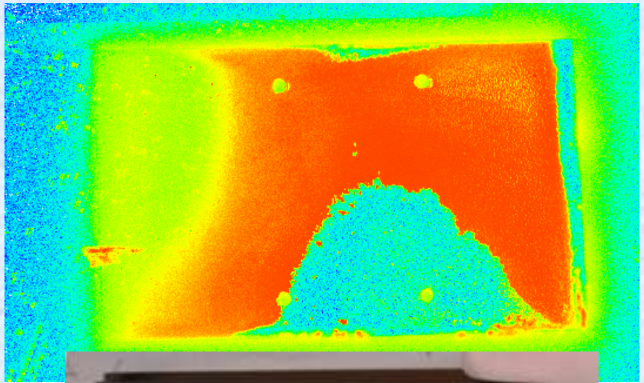
Grey : CFC

White : Graphite

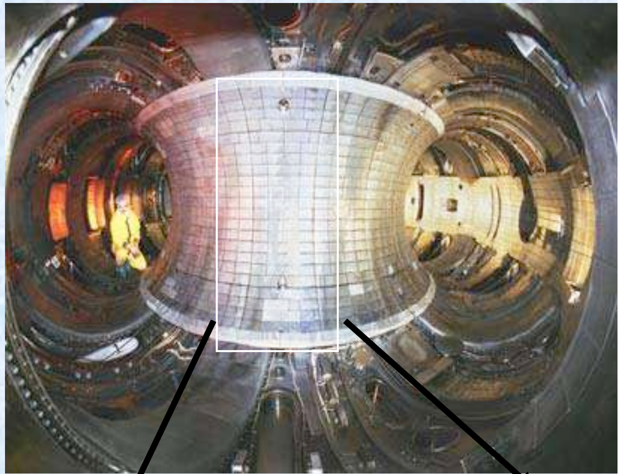
Experience of DT discharges in TFTR

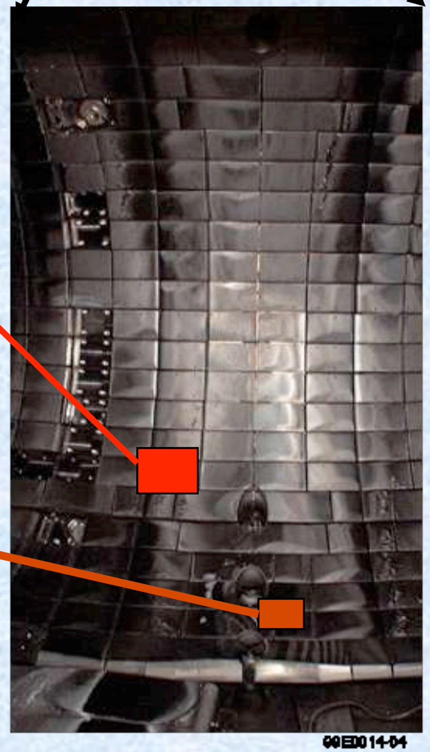
- T retention is mostly in redeposited layers
- Inhomogeneous retention
- Machine dependent distribution

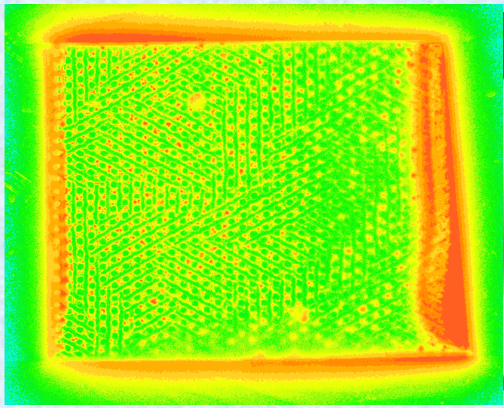
Concerns on T inventory in Carbon machiens

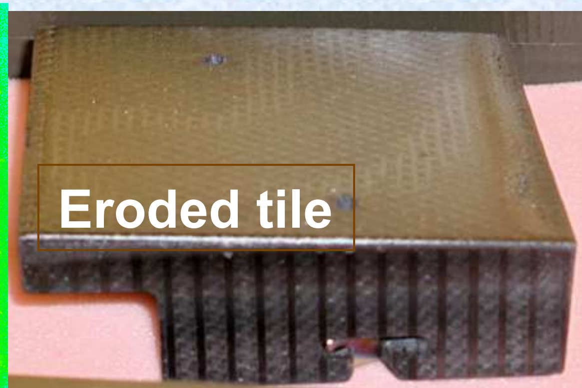


TFTR

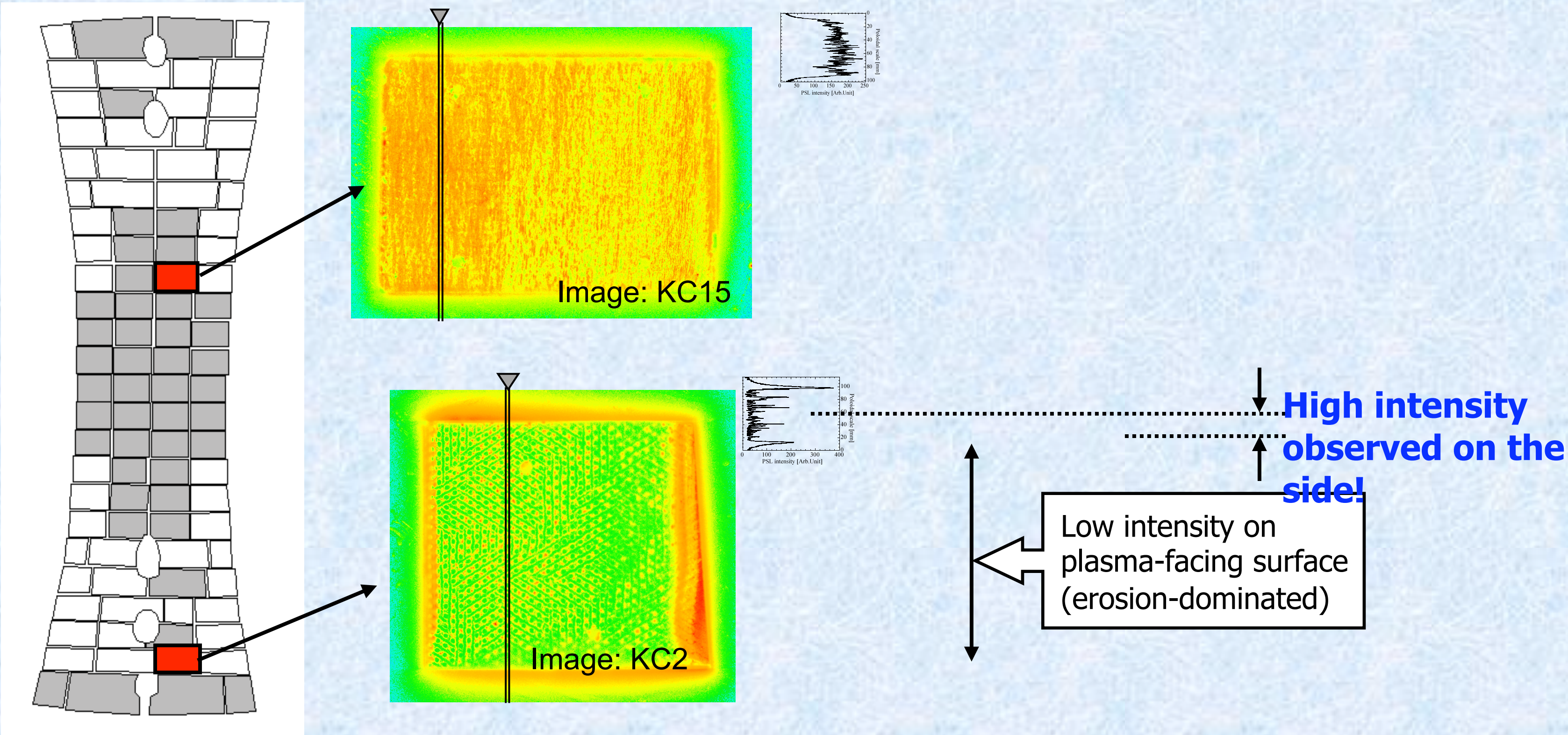








- TFTR bumper limiter
  - Tritium was mostly codeposited with carbon.
  - Heavier codeposition on the edge of the erosion dominated tiles [1].



- Main source of the codeposition on the side was prompt deposition of carbon which was sputtered on the plasma-facing surface!

[1] K.Sugiyama et al., Physica Scripta **T108** (2004) 68

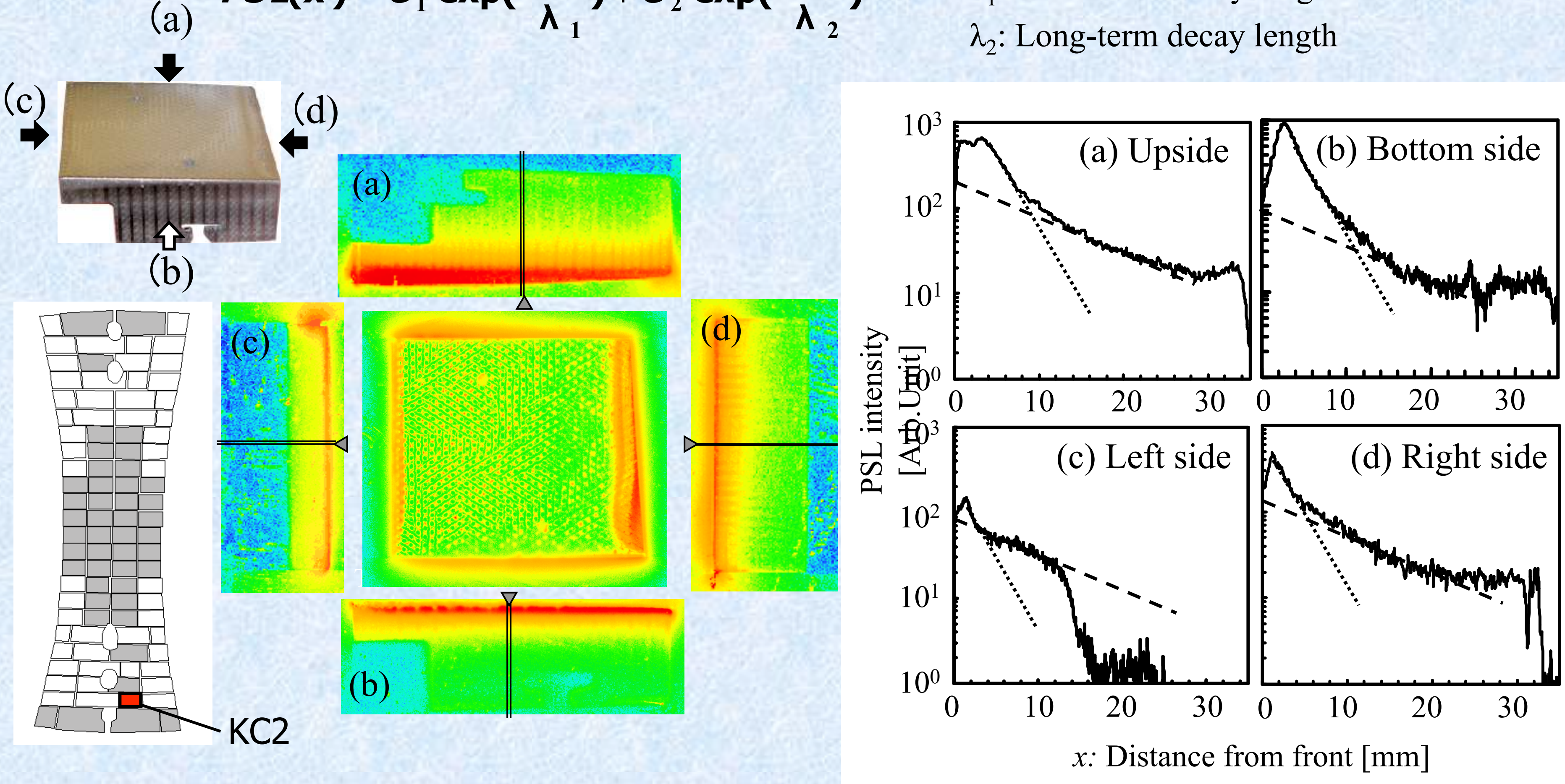
T retention profile on tile sides consists of two exponential decay components

$$PSL(x) = C_1 \exp\left(-\frac{x}{\lambda_1}\right) + C_2 \exp\left(-\frac{x}{\lambda_2}\right)$$

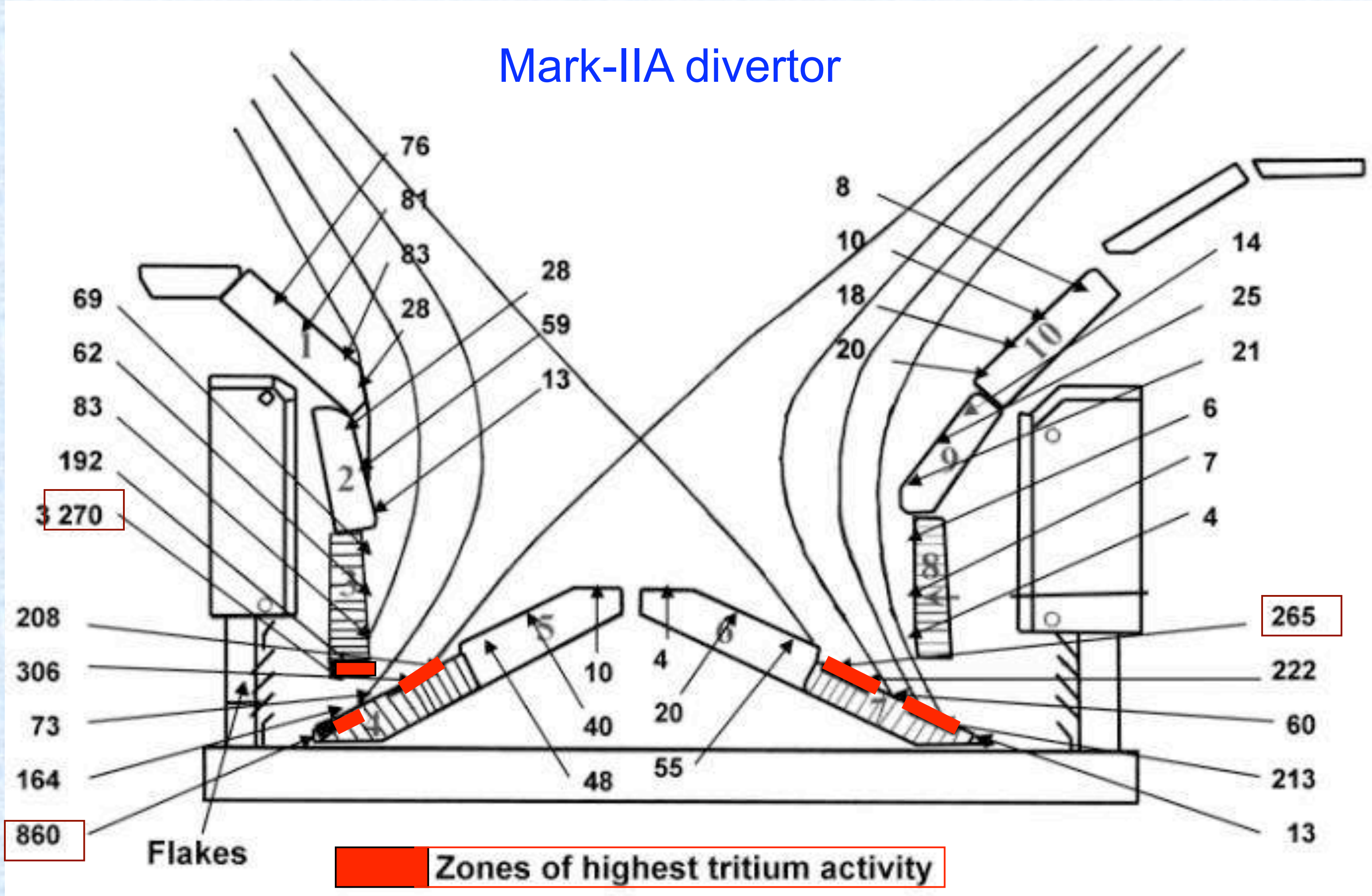
x: Distance from front

λ_1: Short-term decay length

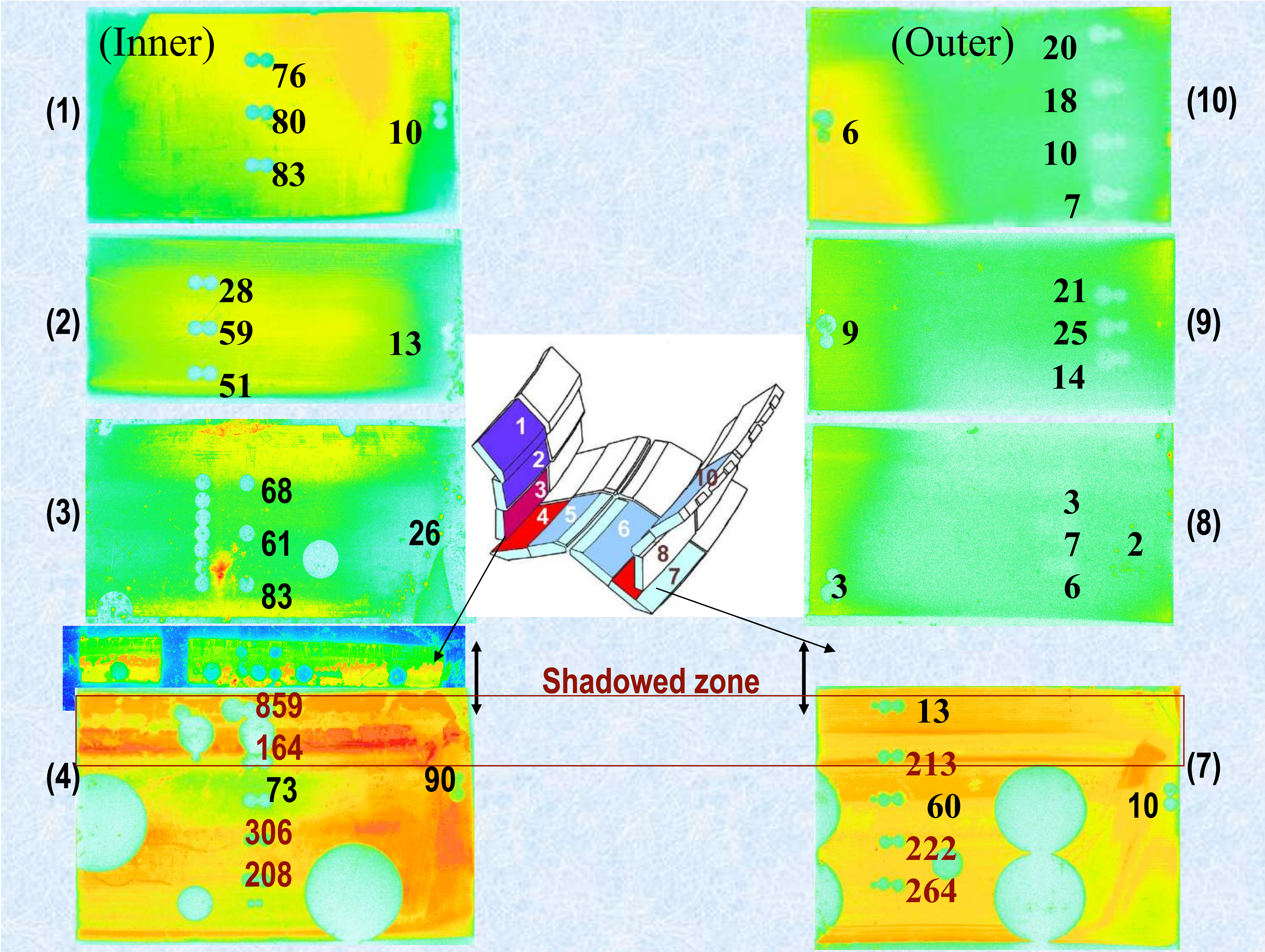
λ_2: Long-term decay length



T retention in JET tiles measured by combustion method



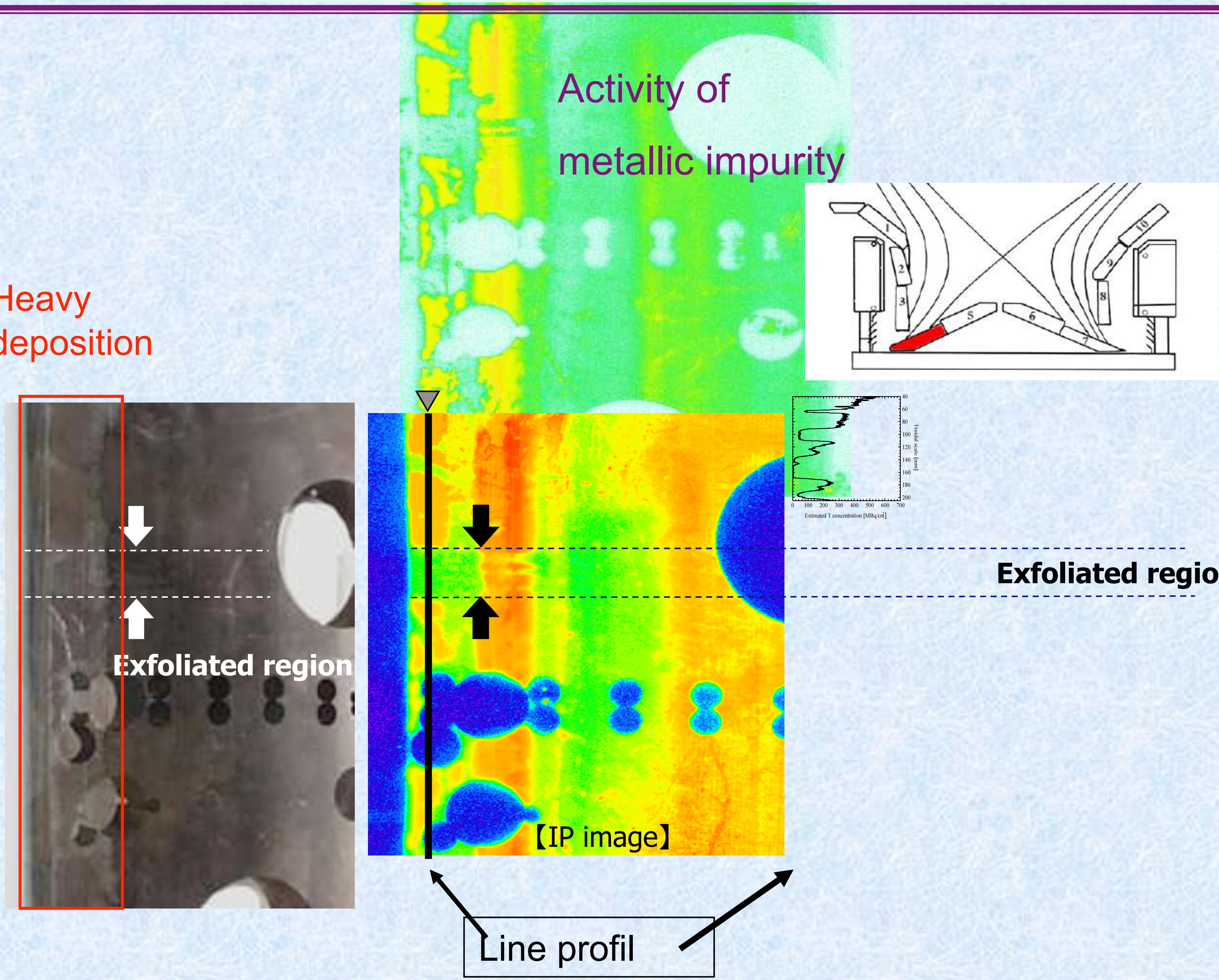
- Tritium codeposits with carbon and other impurities at low temperature region
- No detailed profile
- Necessity to develop removal technique



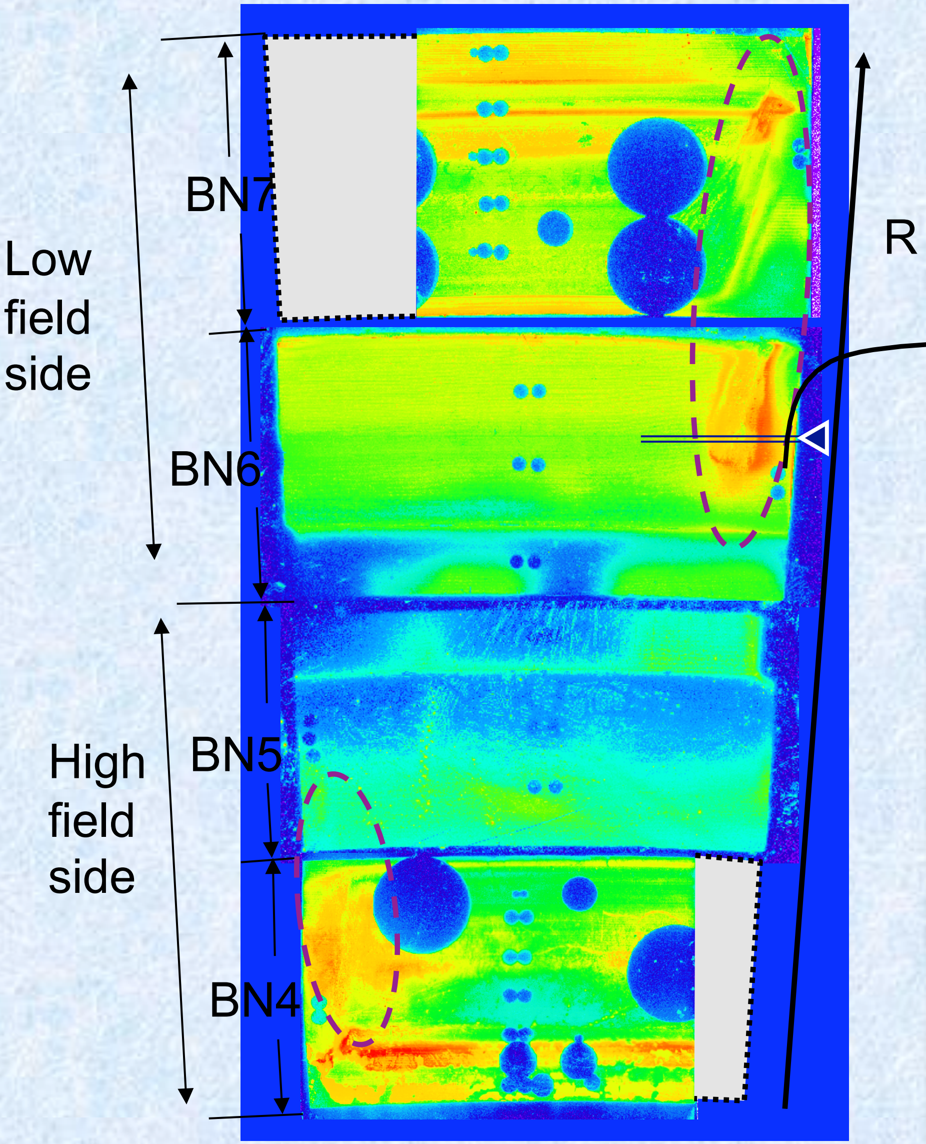
(Inner)
(Outer)
(1)
76
80
83
10
(2)
28
59
51
13
(3)
68
61
83
26
(4)
859
164
73
90
306
208
(10)
20
18
10
7
6
(9)
21
25
14
9
(8)
3
7
6
2
3
(7)
13
213
60
10
222
264
1
2
3
4
5
6
7
8
10
Shadowed zone

Most of Tritium is in deposited carbon

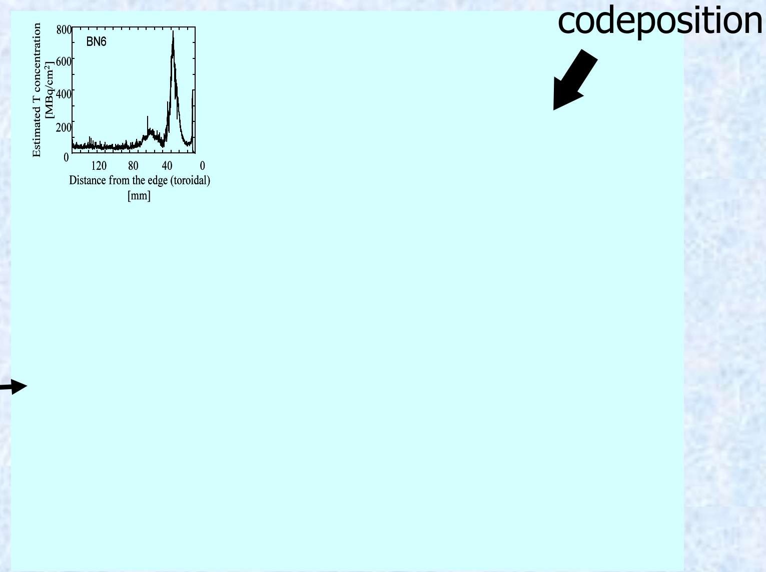
Heavy
deposition



Clear asymmetry owing to tile alignment

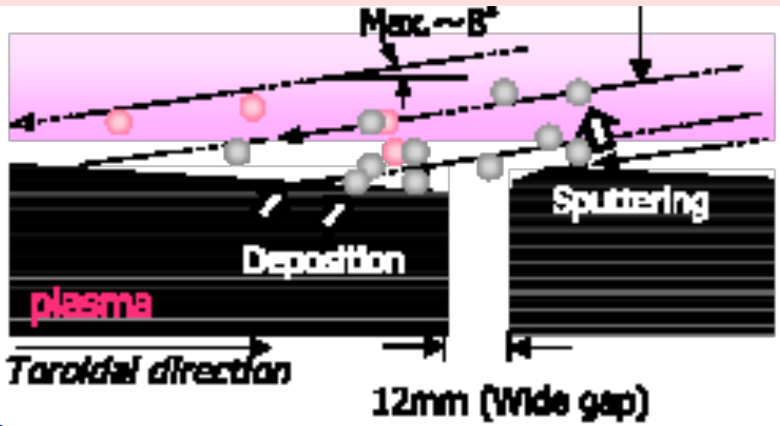


【Tritium image of the divertor floor tiles】

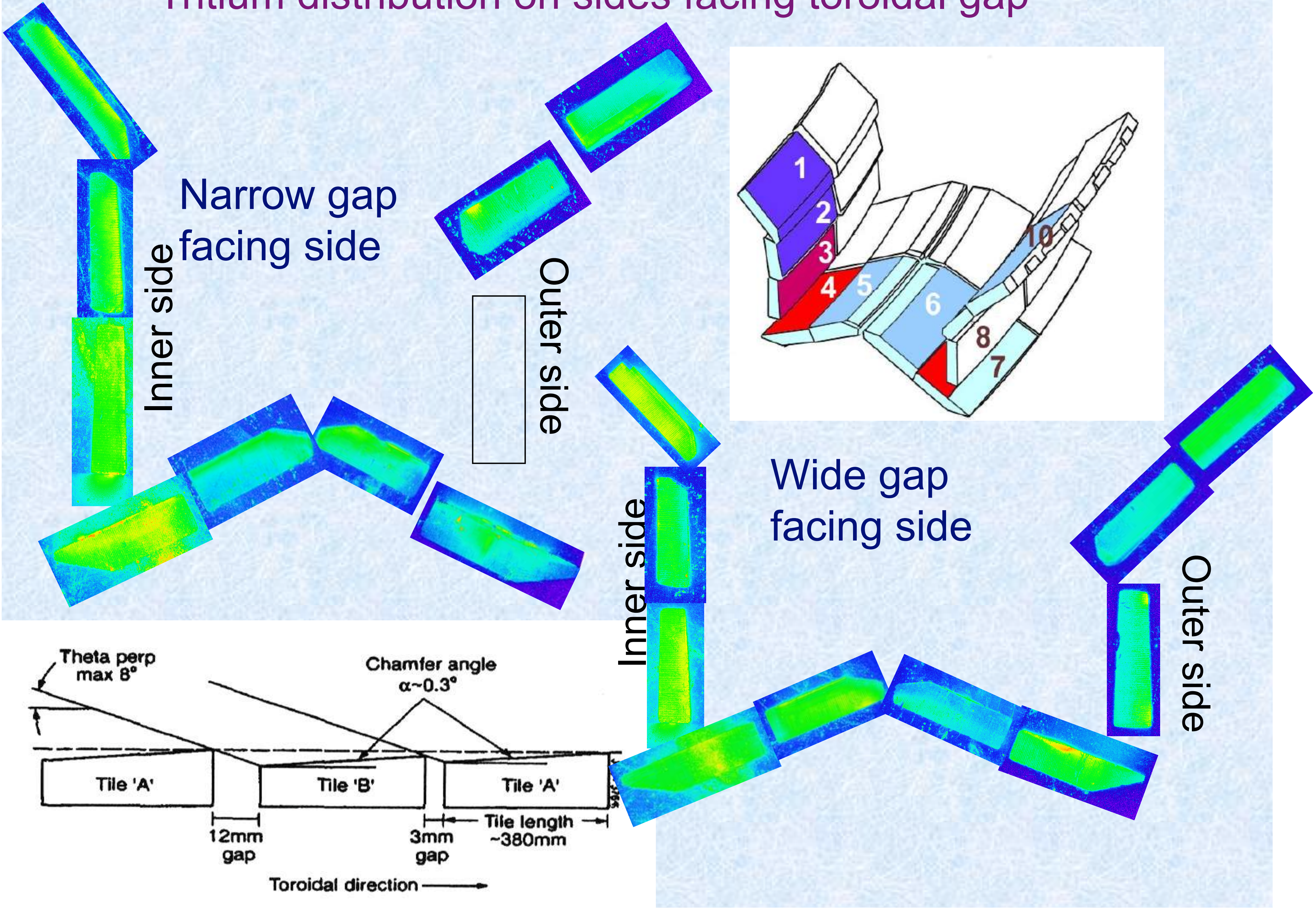


- Local codeposition

  ⇒ because of the codeposition of tritium and carbon sputtered at adjacent tile.

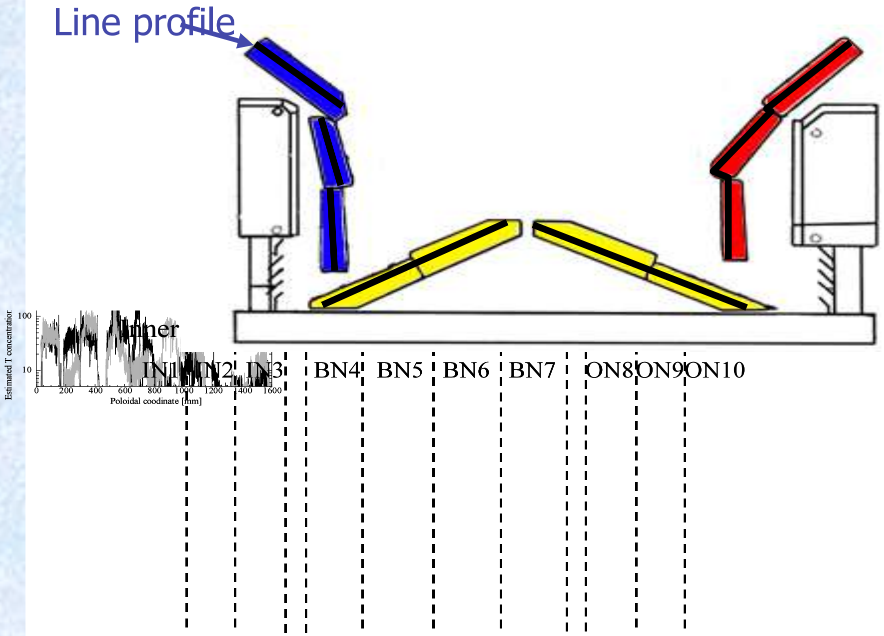


Tritium distribution on sides facing toroidal gap



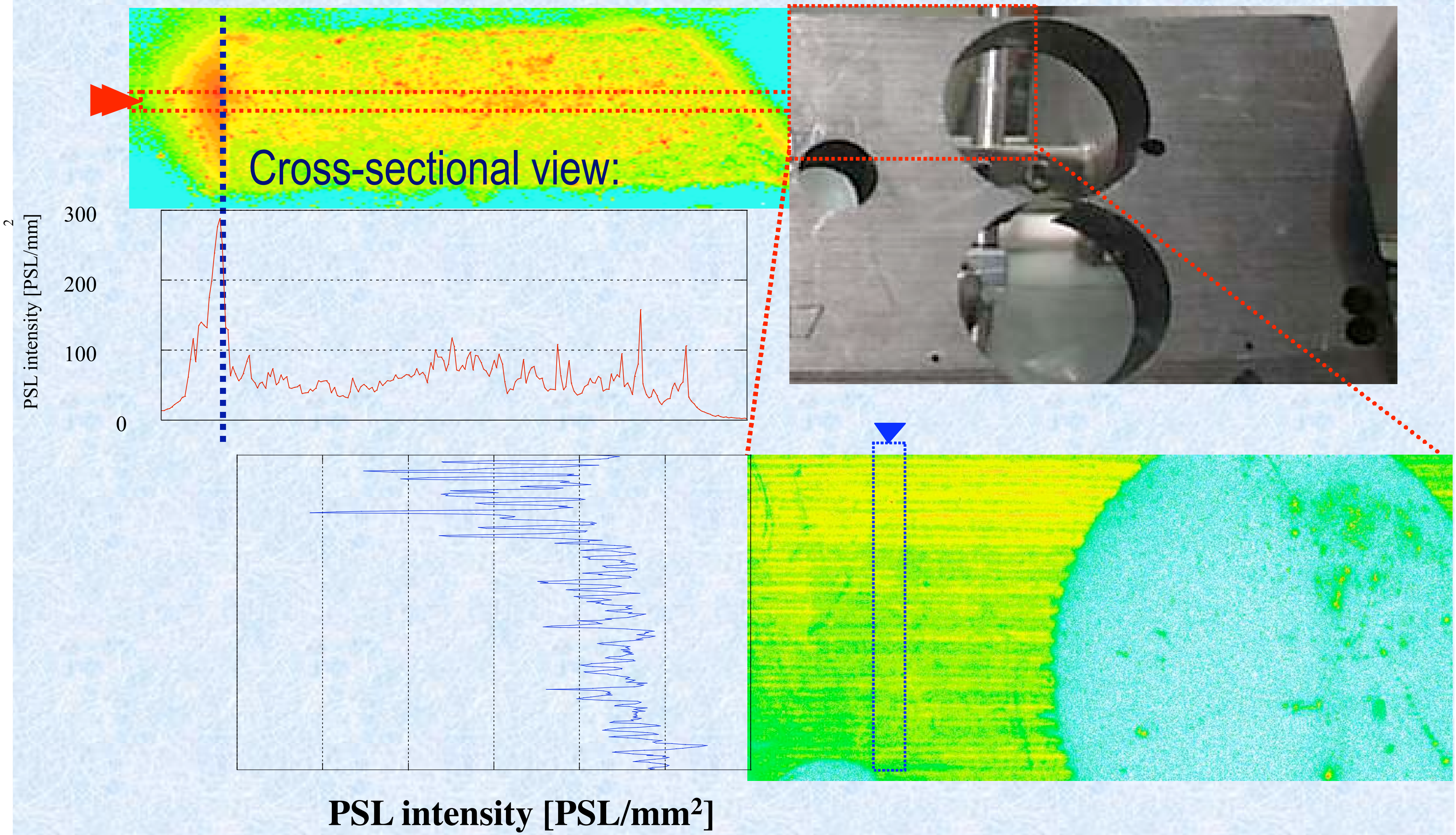
Deposition on the toroidal gaps does not seem problem

- Little codeposition on the toroidal facing sides
- Clear In-Out asymmetry, but little difference between the both sides



CFC is a porous material allowing deep T penetration and its matrix and filler shows quite different tritium retention

Front Surface **Rare Surface**



II-2. Behavior of Tritium produced by DD reactions

At the beginning we thought behavior of T produced by DD reactions should be similar to that of fueled T in tokamak.

We have found that was wrong.

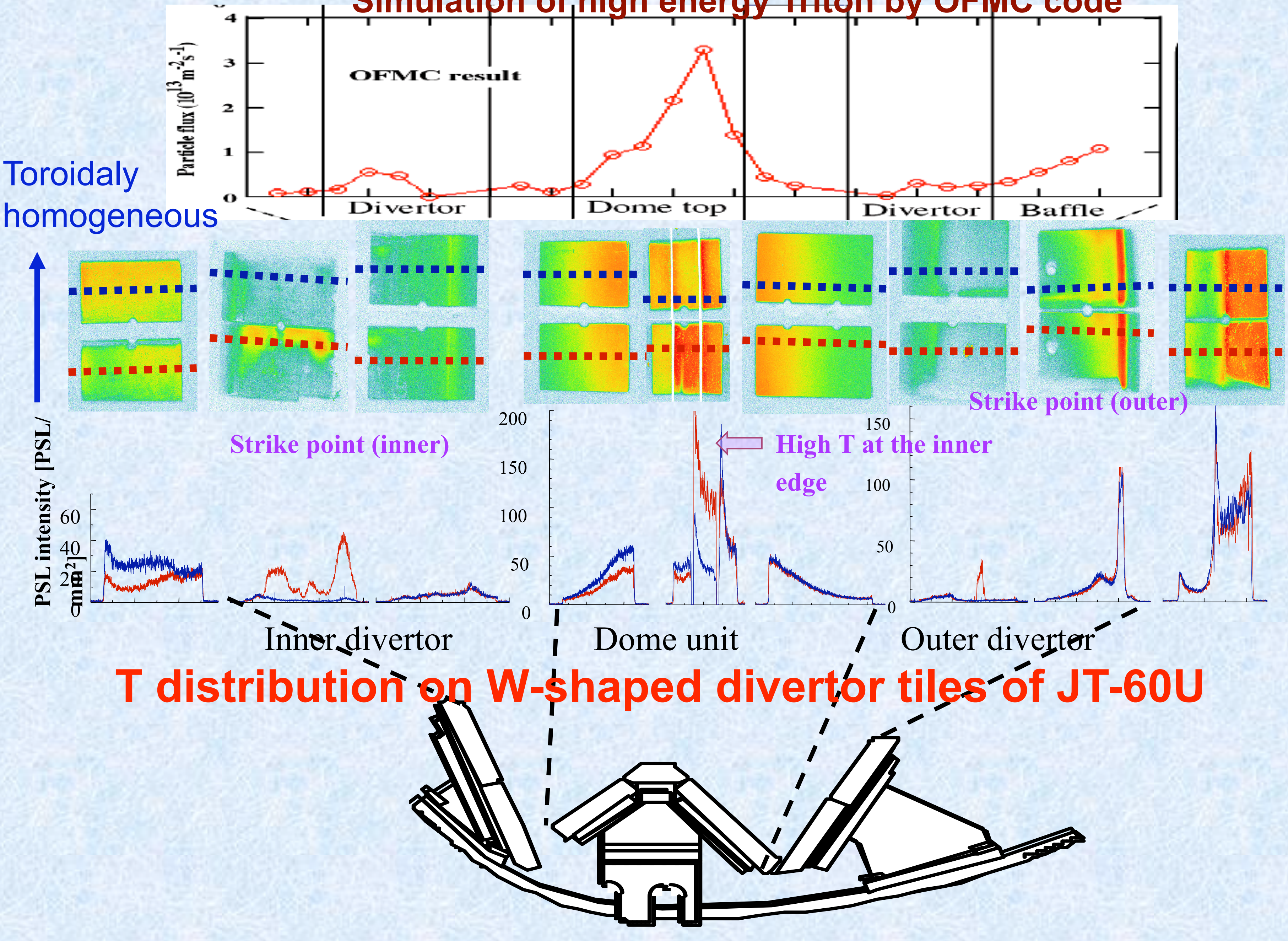
Tritium produced by DD reactions in JT-60U, ASDEX-U and TEXTOR) do not reflect behavior of fueled tritium.

Simultaneously we have found that

- Most of T produced by DD reactions (which initially have energy of 1MeV) do not fully loose their energy and are directly implanted into subsurface of the plasma facing materials (in present tokamaks).

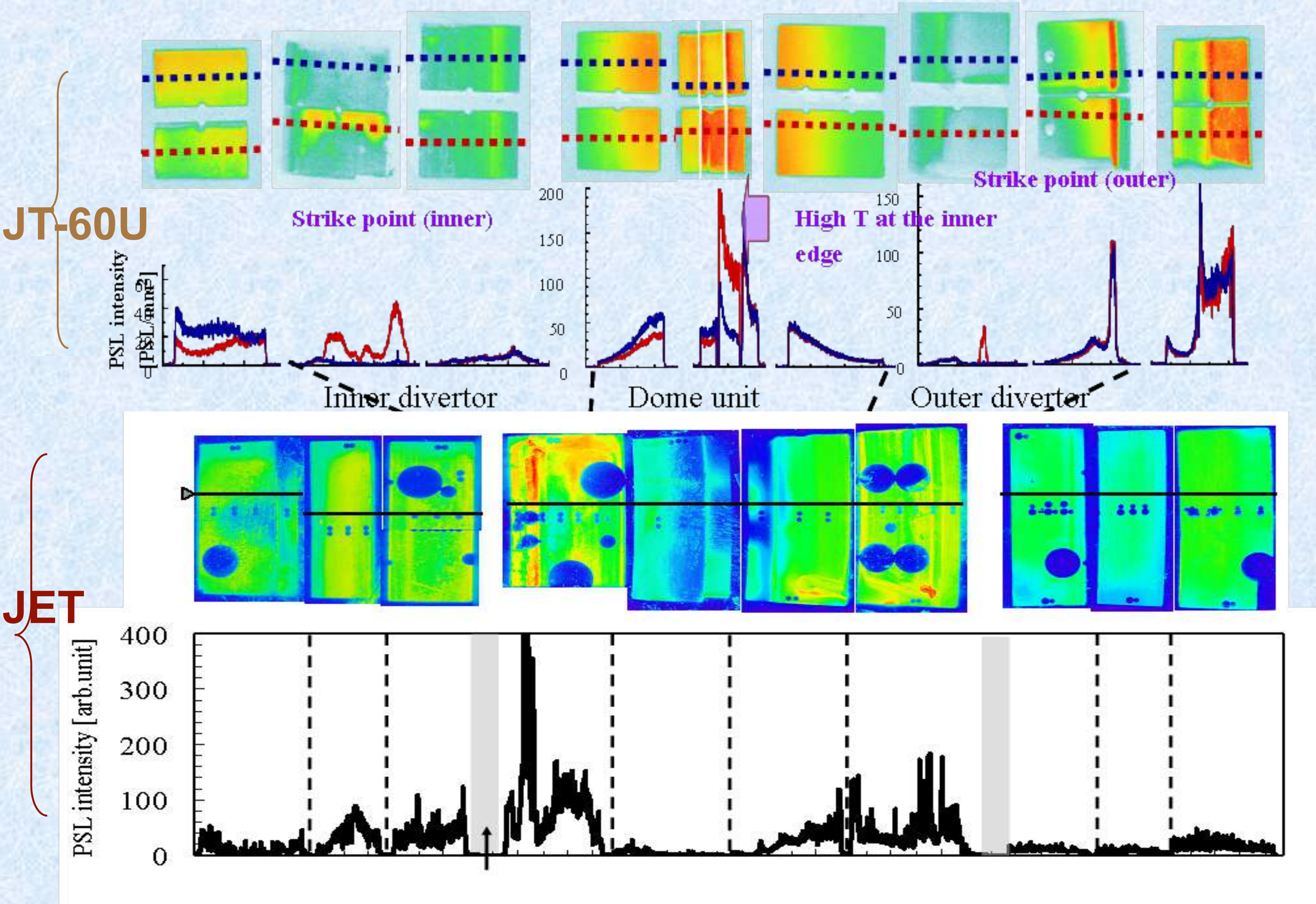
i.e.
we can study behavior of high energy particles escaping from plasma like NBI particles and He ash.

Simulation of high energy Triton by OFMC code



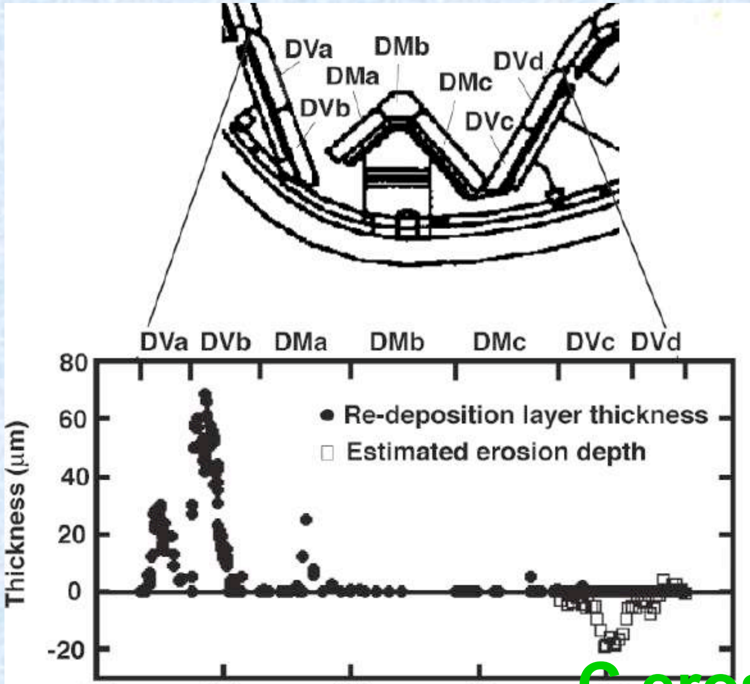
T distribution on W-shaped divertor tiles of JT-60U

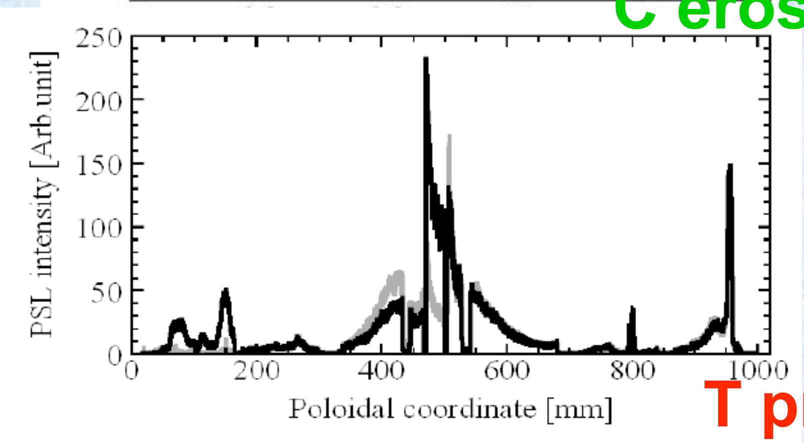
Completely different T profiles at divertor area between JT-60U and JET



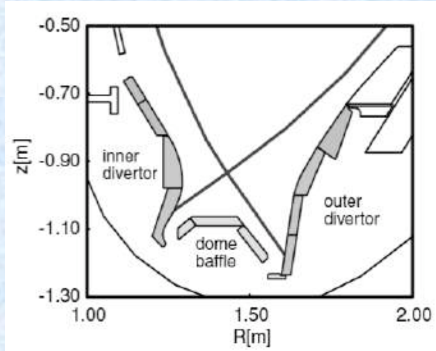
T profile is completely different from C deposition

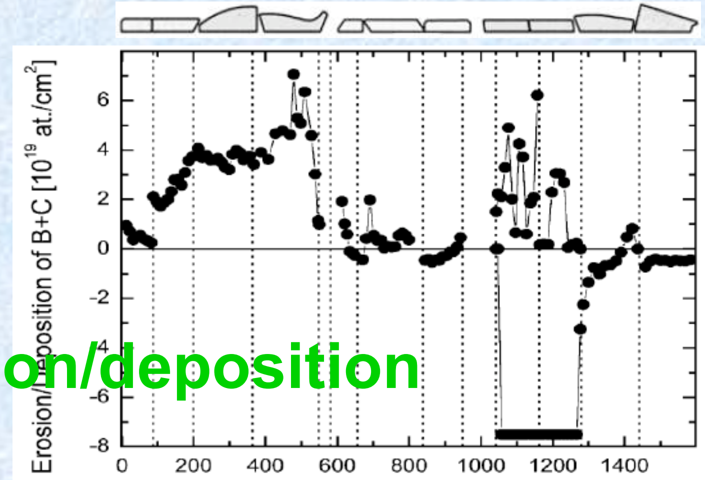
JT-60U



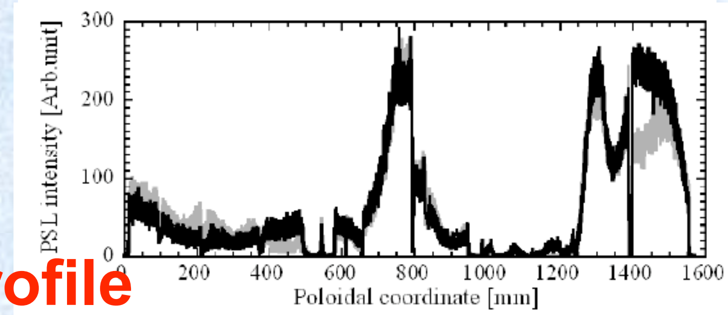


ASDEX-U



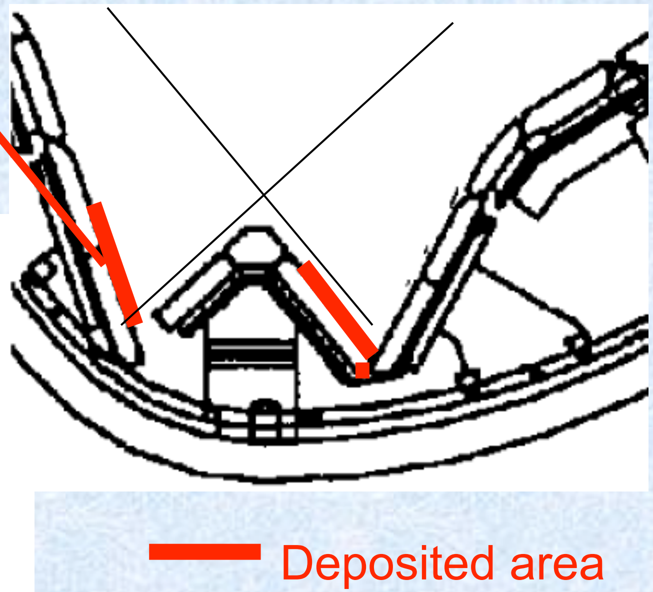


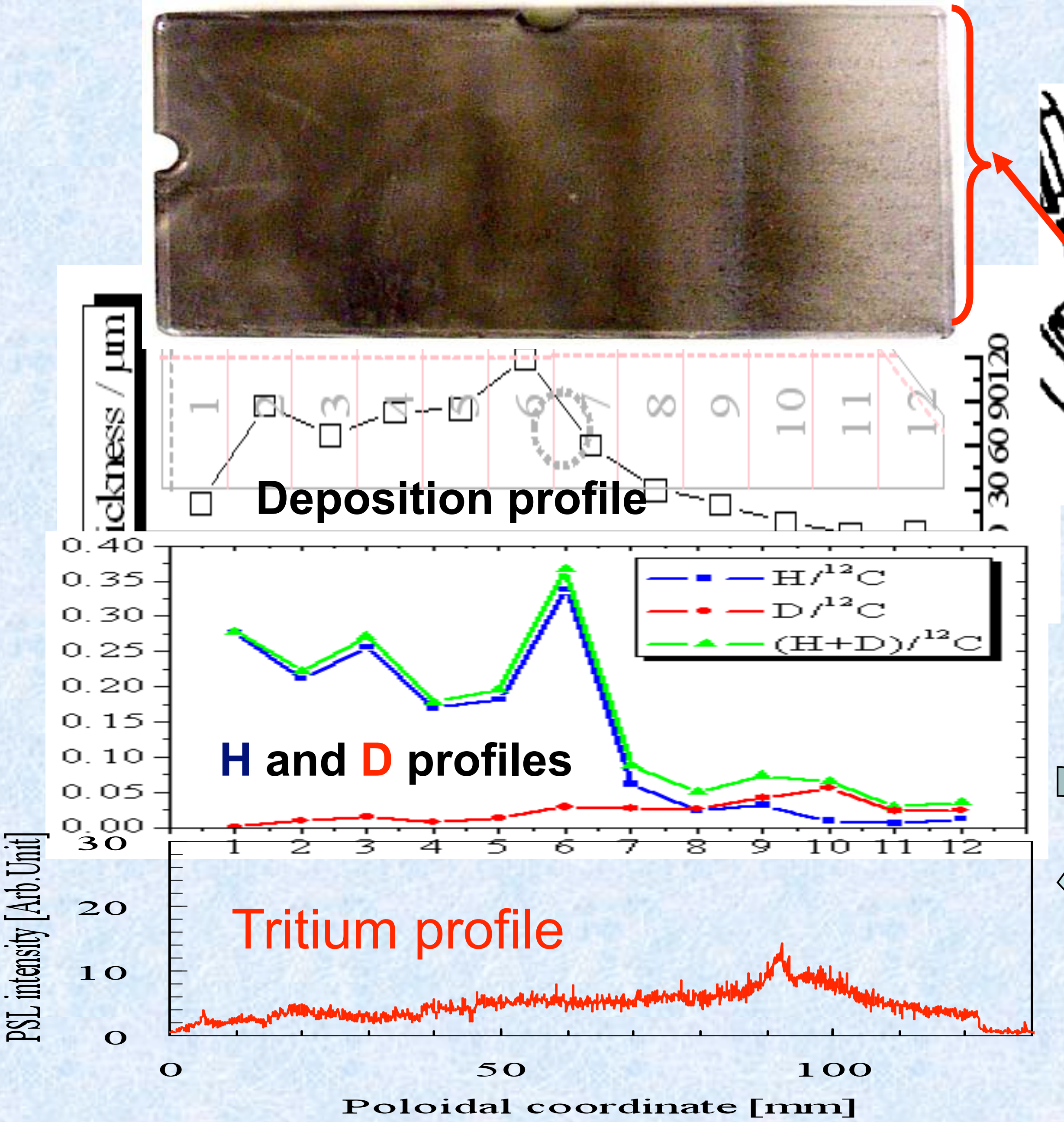
C erosion/deposition



T profile

Typical example for different behavior of T and H/D





Completely different

OFMC simulation

High energy Triton escaping from plasma

<Simulation results using OFMC code>

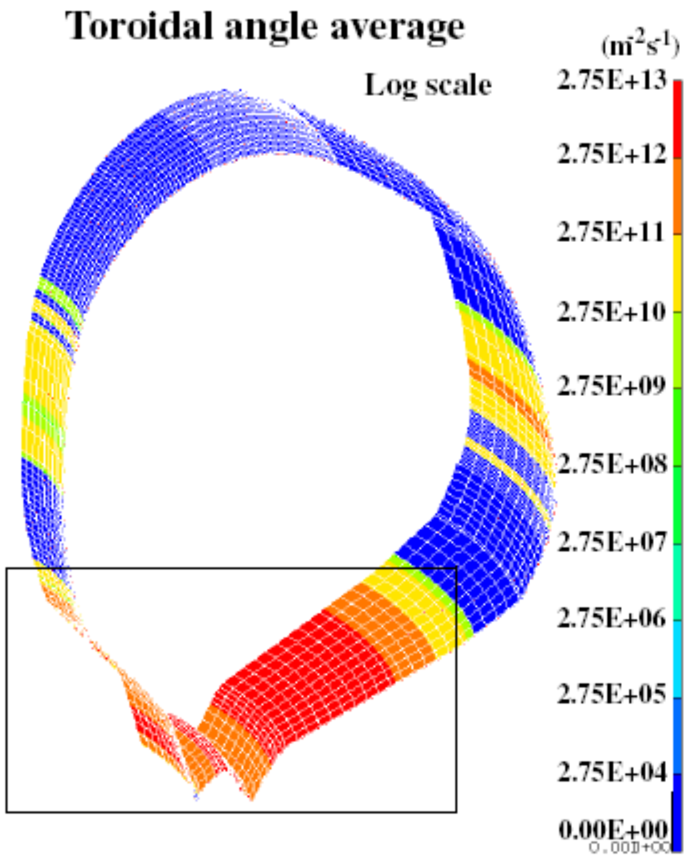
- **31% of produced tritons was lost and implanted to the wall.**
- **12% of produced tritons impinged on the divertor region**
- **Fluxes of the tritons were**

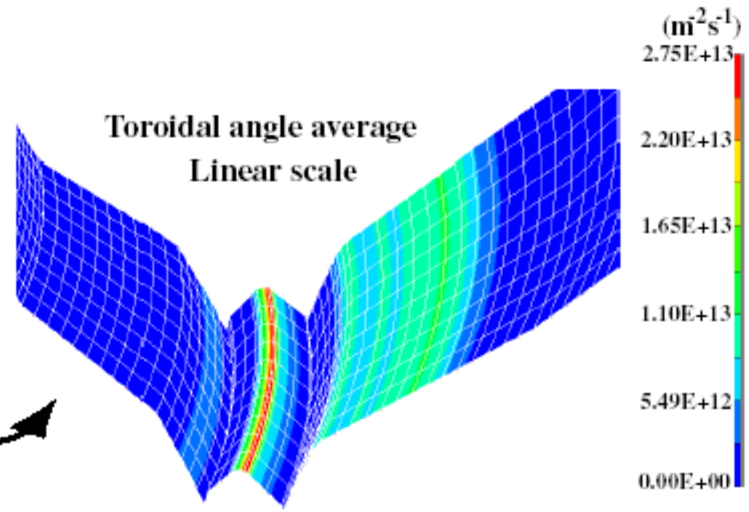
In a case of neutron production of ~10^{14} s^{-1}

~10^{11} $m^{-2}s^{-1}$: inner and outer midplane first wall

~10^{12} $m^{-2}s^{-1}$: inner divertor target and inner baffle plate

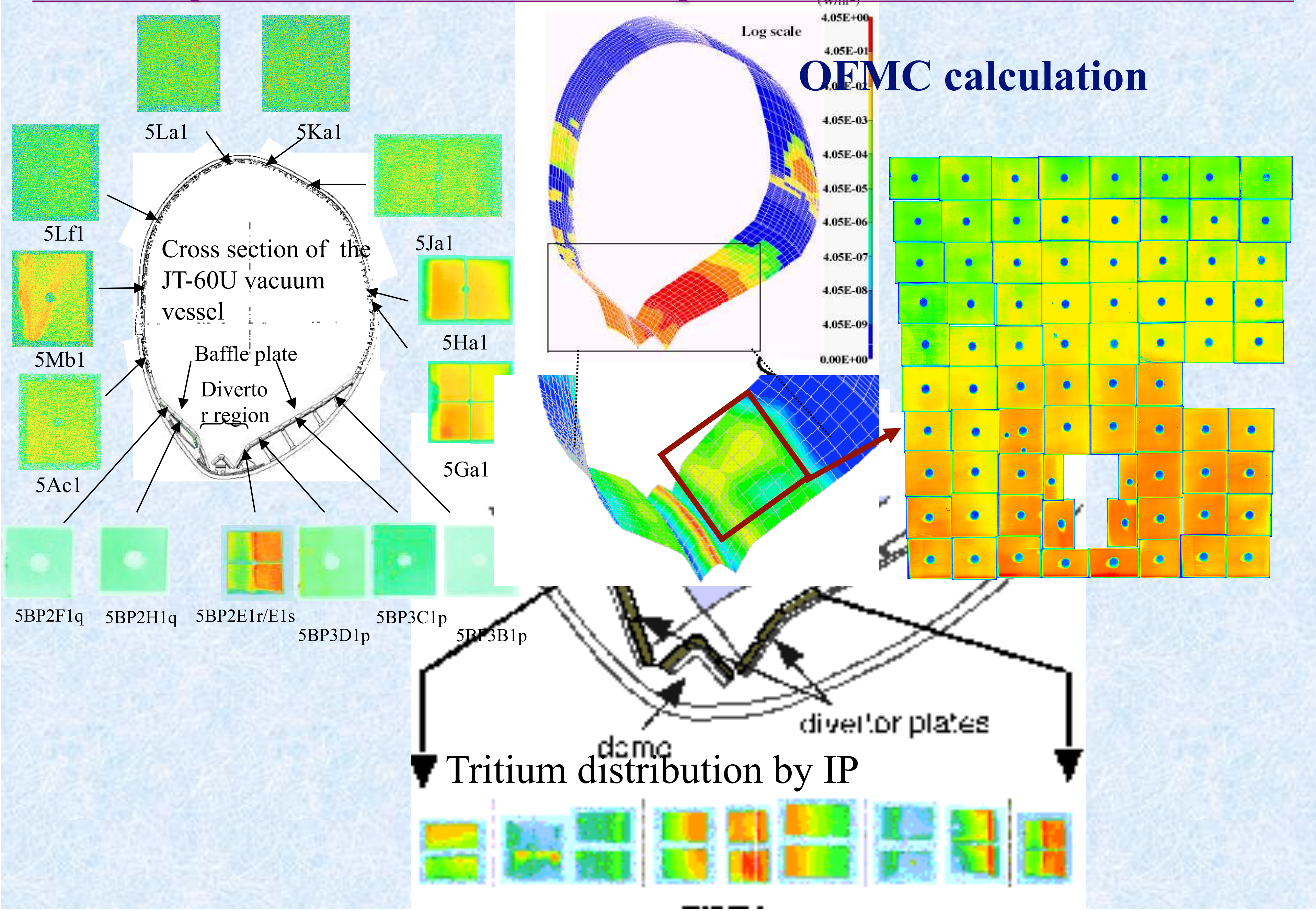
~10^{13} $m^{-2}s^{-1}$: dome top and the outer baffle plate





Masaki et al. 15th PSI

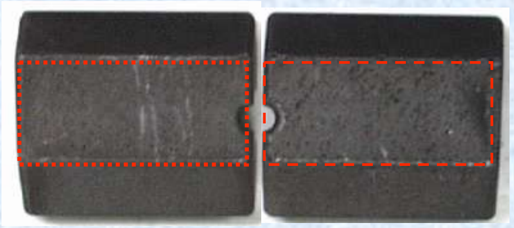
Comparison of observed T profiles and simulation



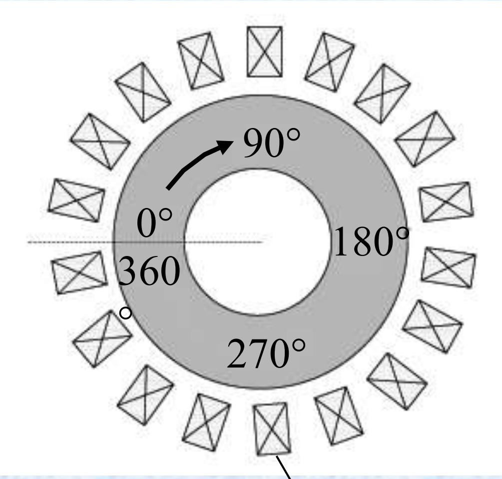
Tritium on dome top tile - full toroidal distribution-

 # 240 peaces of tiles

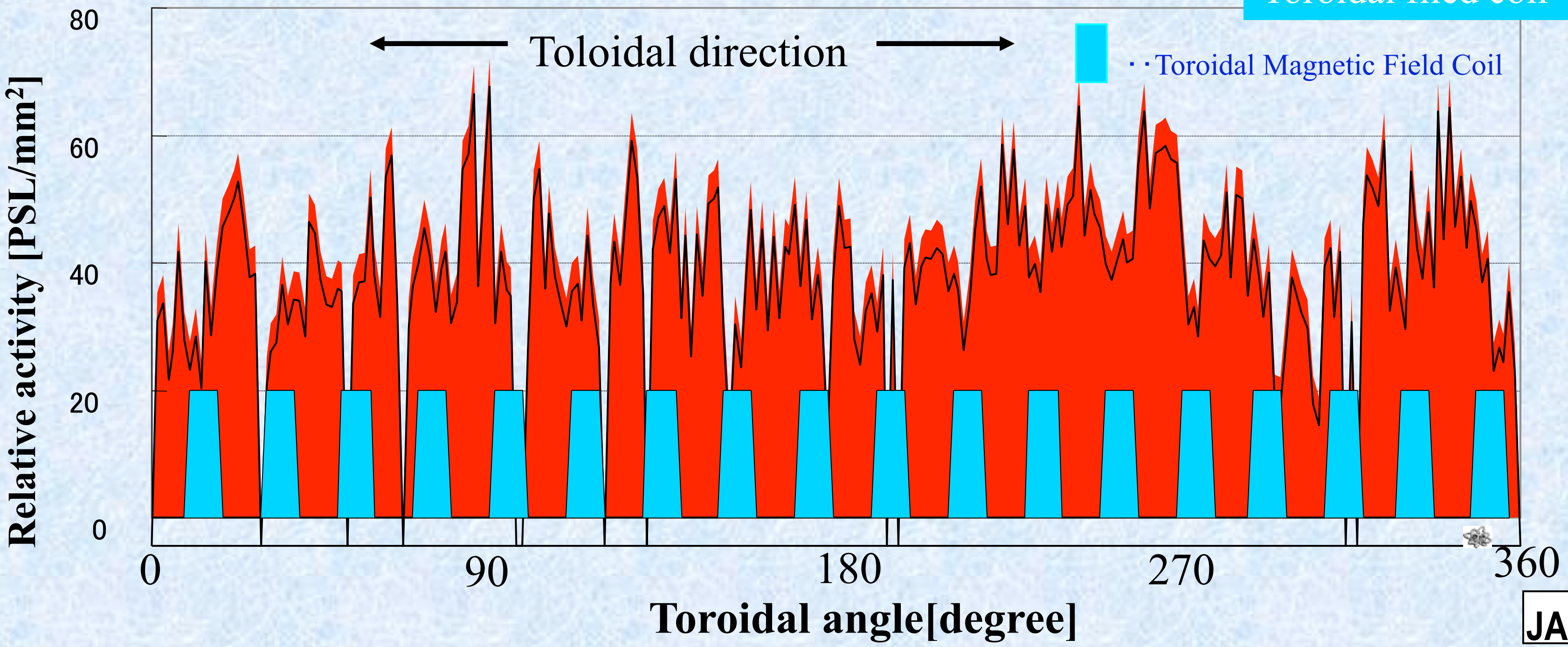
 # relation between tritium distribution and toroidal magnetic coils



Measuered area



Toroidal filed coil



JAERI

Tritium retention: comparison between observation and calculation

Observation

Long term tritium retention : Roughly 40% of produced tritium (18GBq)

<Tritium concentration>
Inner divertor: 2 kBq/cm^2
2 Dome top: 60 kBq/cm^2
Outer divertor: 250 Bq/cm^2

<Tritium retention>
Divertor region
10% of produced tritium

OFMC calculation

31% of tritons produced by nuclear reaction are lost from plasma

Dome : 6% of the produced tritons, ~0.7 MeV
First wall: 1%, ~1 MeV
Divertor: 3%, ~0.5 MeV
Inner baffle plate: 1%, ~1 MeV
Outer baffle plate: 20%, ~0.6 MeV

<Tritium retention>
Divertor region
12% of produced triton

Tritium behavior in current tokamaks (summary)

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

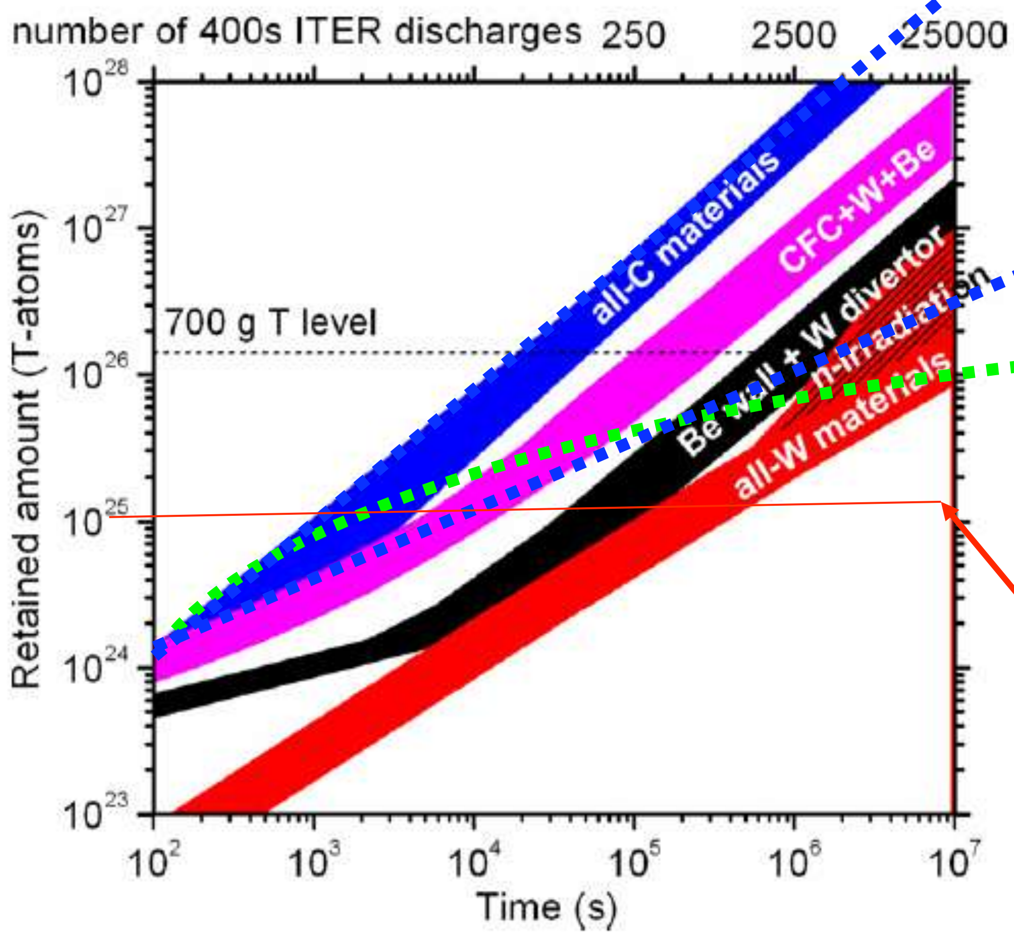
DT discharge experiments In JET and TFTR have shown that significant amount of tritium retained in redepoited carbon layers.

Behavior of T produced by DD reactions is completely different from that of fueled T.

For detailed understanding of T behavior and estimation of T inventory in ITER and a reactor, behaviors of D in various tokamaks have been extensively studied. And no we believe D behaves similar to T but is not so sure. (We don not how large isotopic effects are)

In anyway, large D retention in carbon redeposits make us to avoid carbon as PFM in a DT reactor.

Current estimation of T inventory in ITER is not saturated!



$\propto \phi$

No saturation in deposits at plasma shadowed area

$\propto \sqrt{\phi}$

Possible saturation ?

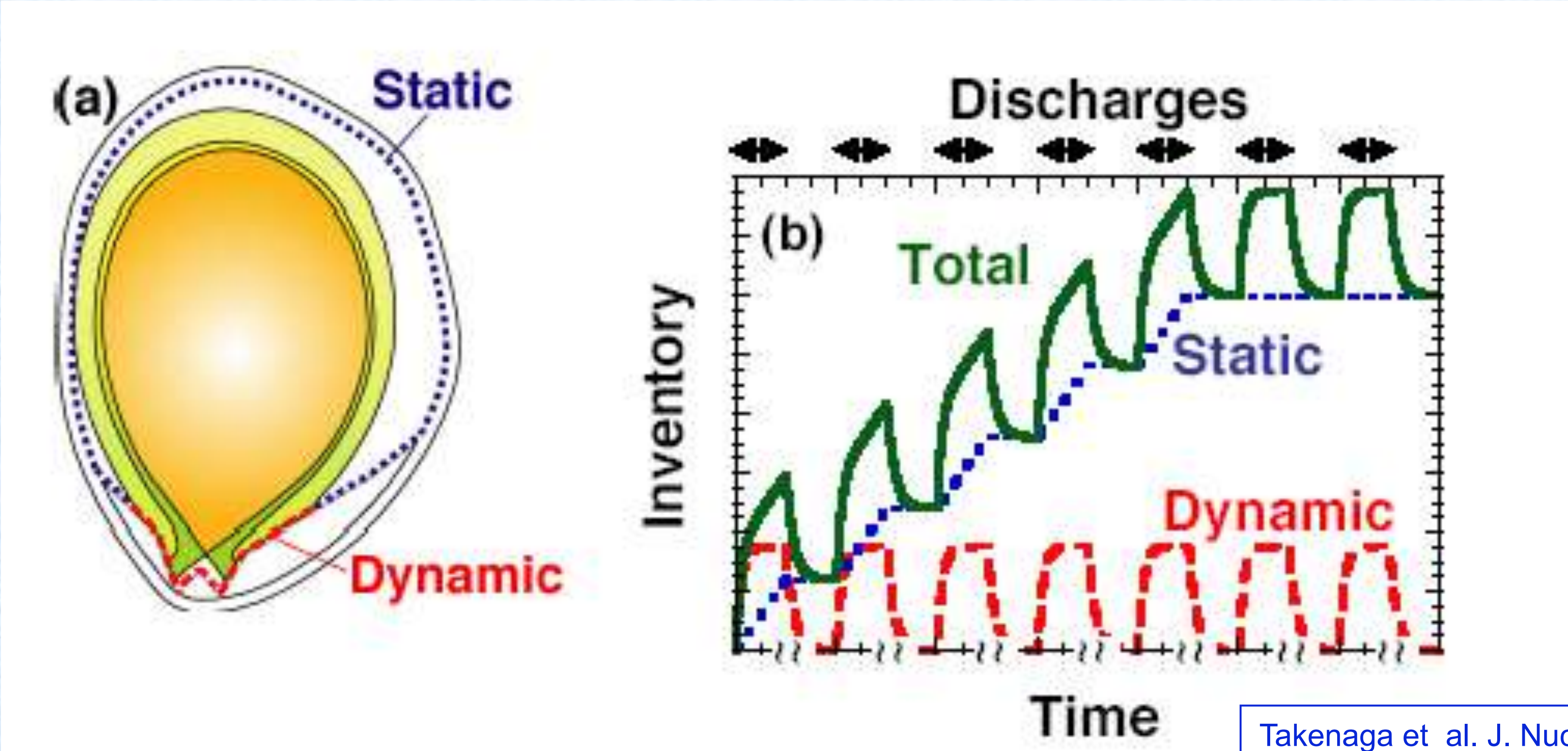
In the present tokamaks
Maximum retention appears around 10^{23} /m^2
X100m^2

Tritium inventory in ITER plasma-facing materials and tritium removal procedures
J. Roth, E. Tsitrone, T. Loarer, V. Philipps, S. Brezinsek, A. Loarte, G. F Counsell, R. P. Doerner,
Plasma Phys. Control. Fusion **50** (2008) 103001

It is critically important whether hydrogen retention saturates or not.

Static retention : incorporated in redeposited carbon layers at plasma shadowed area

Dynamic retention : retained in plasma facing surface area both eroded and deposited



Takenaga et al. J. Nuclear Fusion, 46 (2006) S39-S48

Which is large, Static or Dynamic?

In Tore-Supra; Static >> Dynamic and $\partial S/\partial t$ > $\partial D/\partial t$

In JT-60U ; Static > Dynamic but $\partial S/\partial t$ < $\partial D/\partial t$

Concerns of large tritium retention in carbon materials minimize the utilization of carbon materials in ITER,

- **Be ; First wall**
- **W ; Divertor dome and buffer plates**
- **C ; only for the divertor target.**

However,utilization of tungsten blocks below their DBTT could result in the total failure of the machine through cracking of cooling pipes, we should keep carbon materials as an alternative for armor tiles even for a reactor.

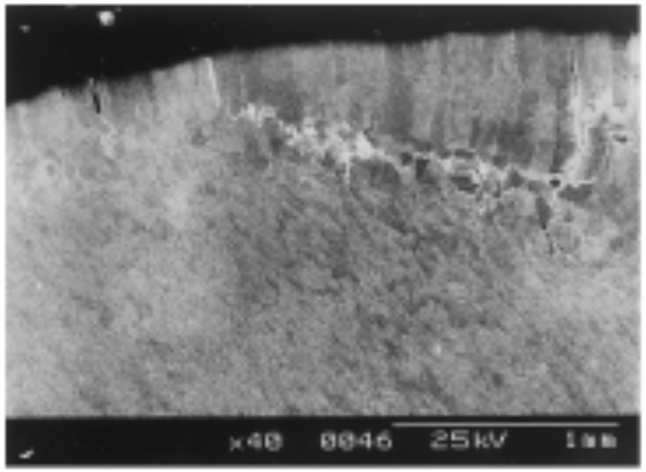
(Matrial selections will be discussed in Wednesday evening)

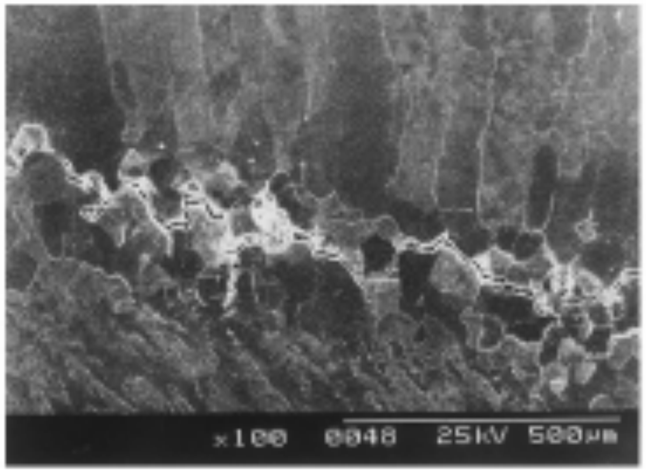
Bulk W and Mo must not be used below DBTT

W

Break-up by thermal shock

Mo





Recrystalization to columnar grains results in cracking
Congruent melting with substrate metals leads cracking as well as melting
(Be and W would give same result)

However,utilization of tungsten blocks below their DBTT could result in the total failure of the machine through cracking of cooling pipes, we should keep carbon materials as an alternative for armor tiles even for a reactor.

(Matrial selections will be discussed in Wednesday evening)

This motivate us to examine carbon erosion/deposition and H and D retention in plasma facing carbon materials of JT-60U in detail.

Remaining questions to quantify tritium inventory

Remaining questions to be solved for application of carbon as PFM in DT machines are,

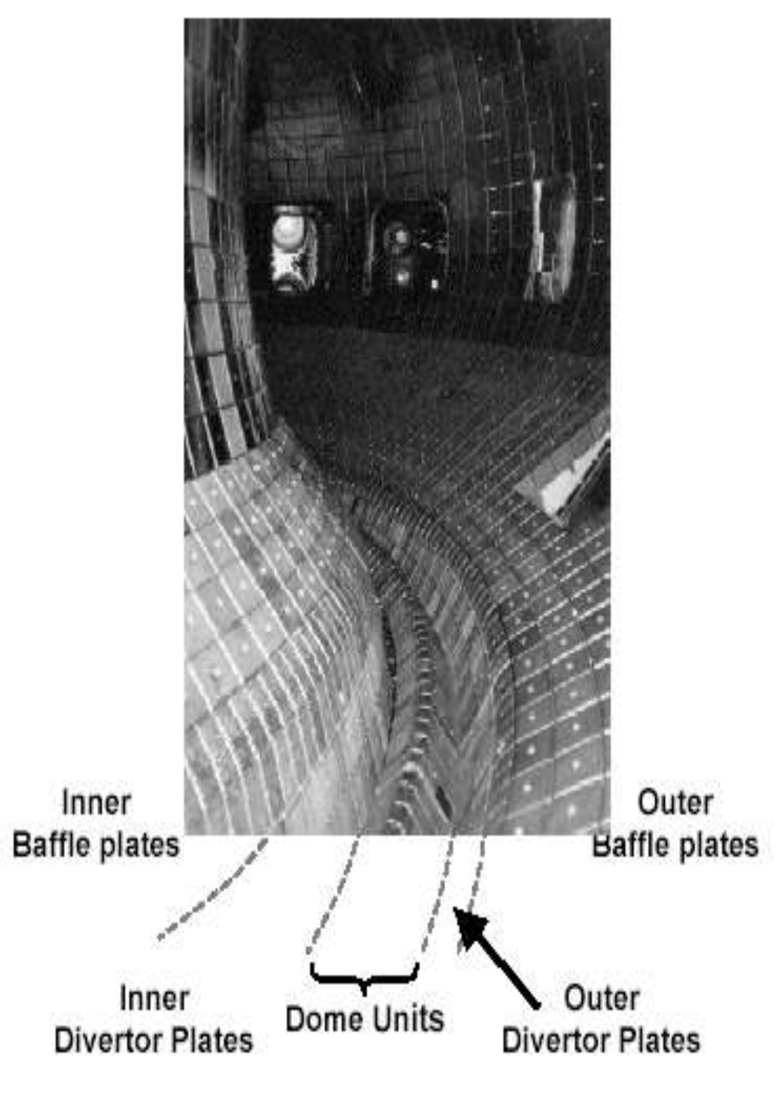
- Where and how much is carbon eroded and redeposited?
- Do erosion and redeposition saturate?

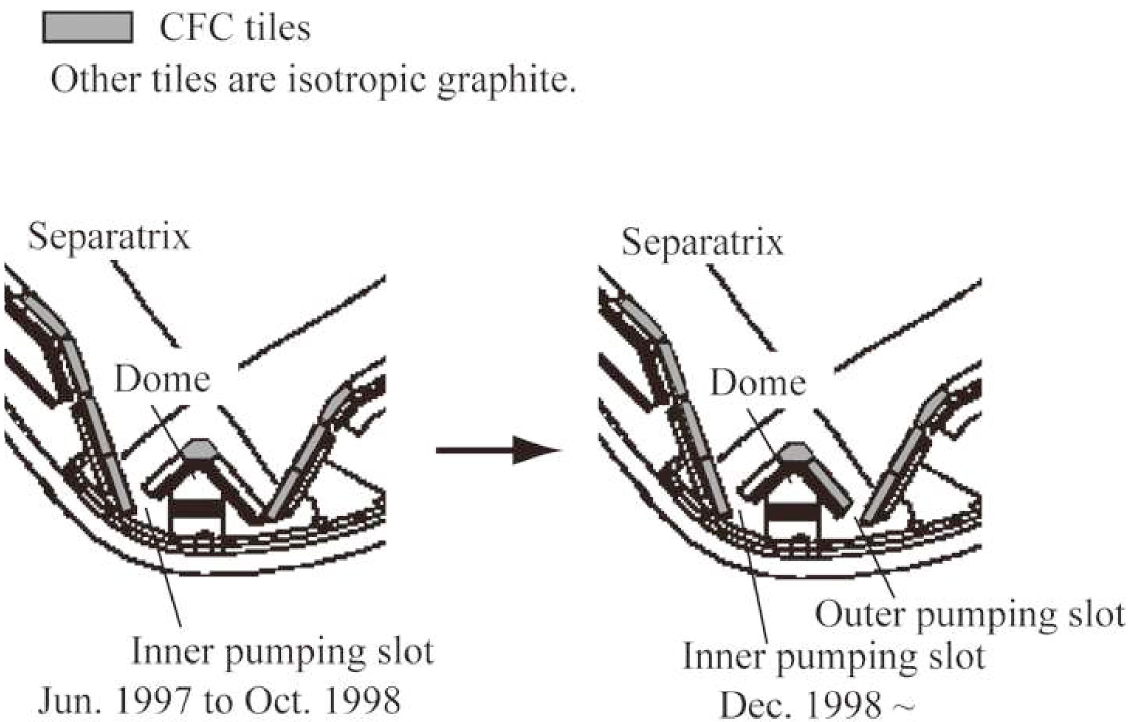
- Where the largest redeposition occurs, plasma facing surface, shadowed area or far remote area?

- Where is tritium (T) retained?
  How related to carbon deposition?
  How large is retention in eroded area and main chamber?
  Does T retention saturate?

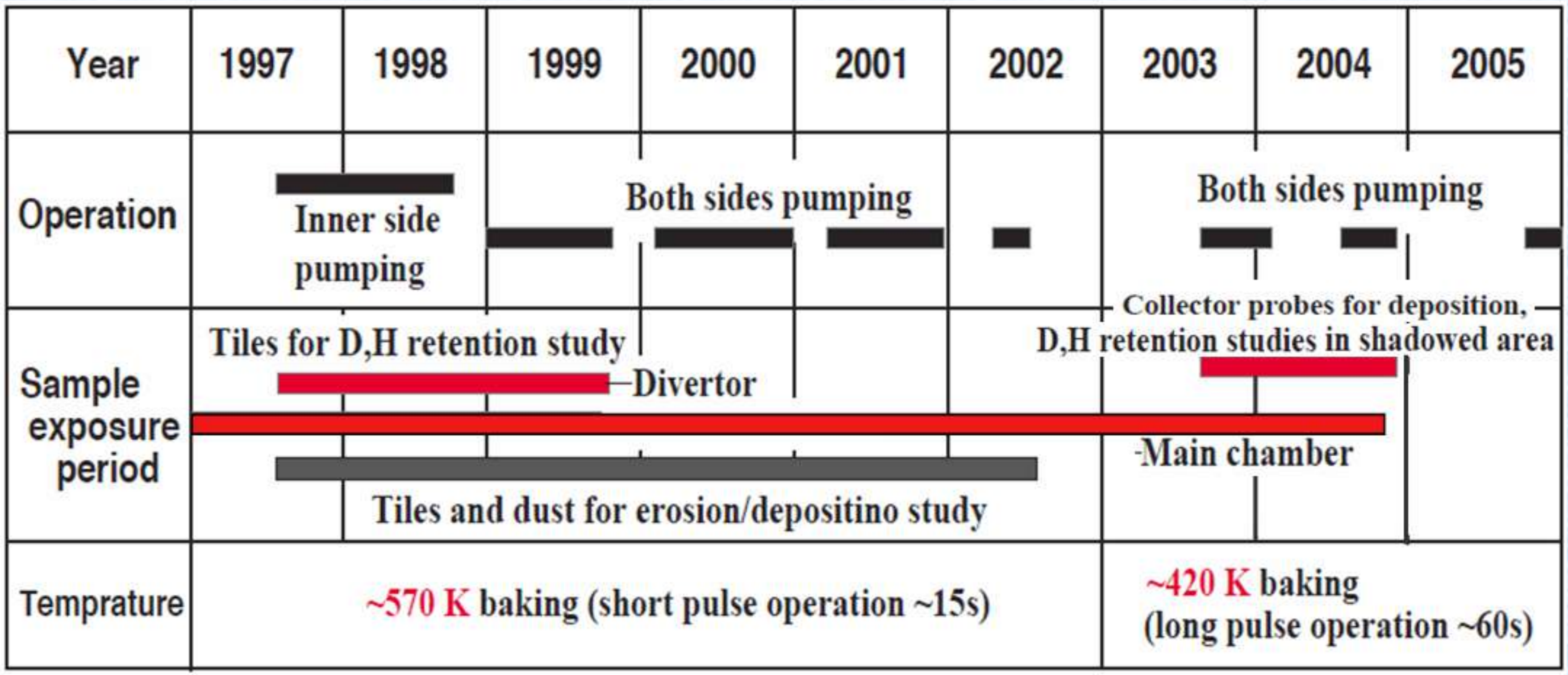
- How to recover or remove the retained T?

II-3. Carbon erosion/deposition and D and H behavior in JT-60 for understanding of DT fuel





History of plasma operation and of plasma exposure of analyzed tiles

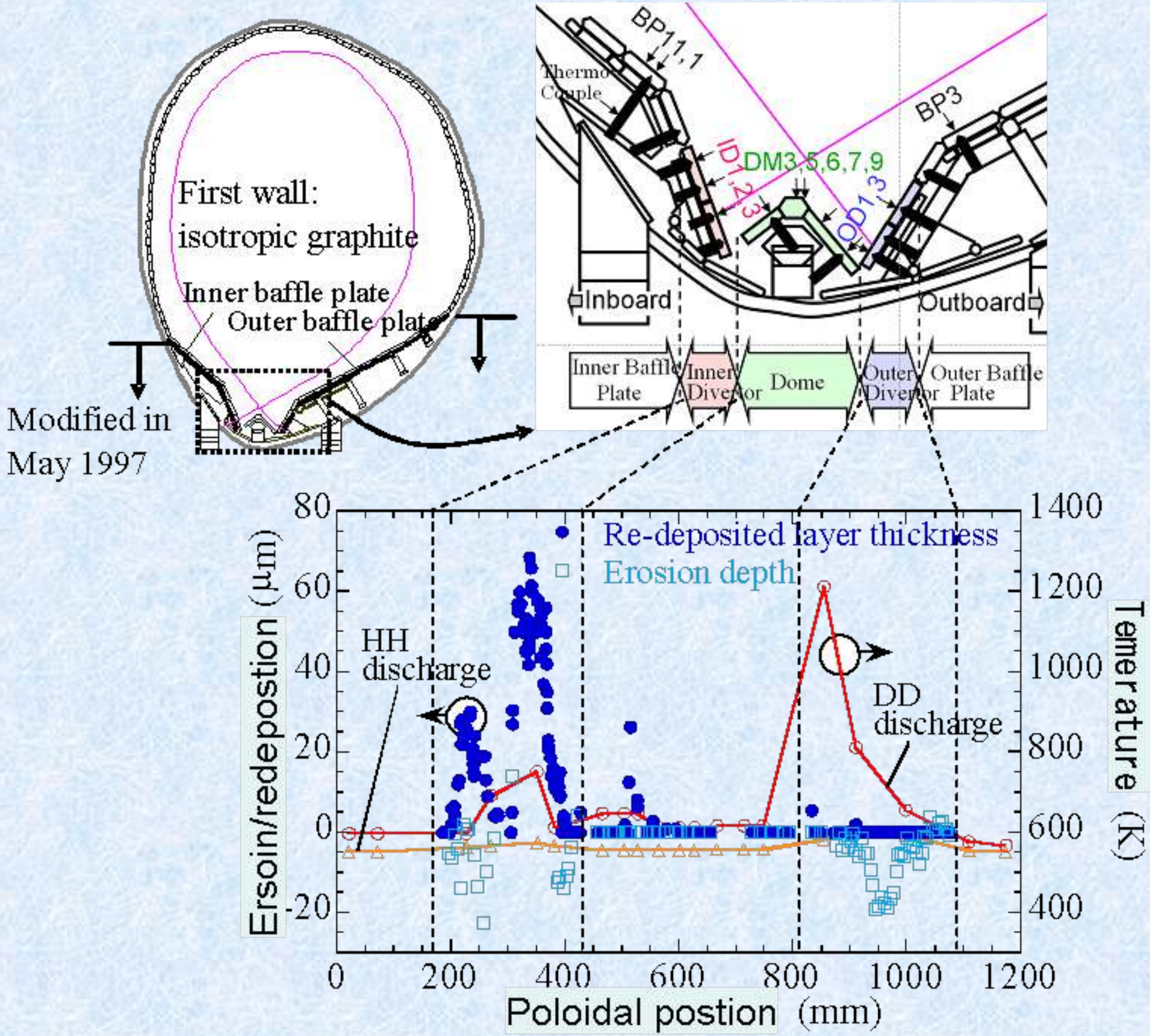


Normal plasma operation is done by D discharges with D NBI (DD discharges)

Usually each campaign terminated by HH discharges to remove T produced by DD reactions.

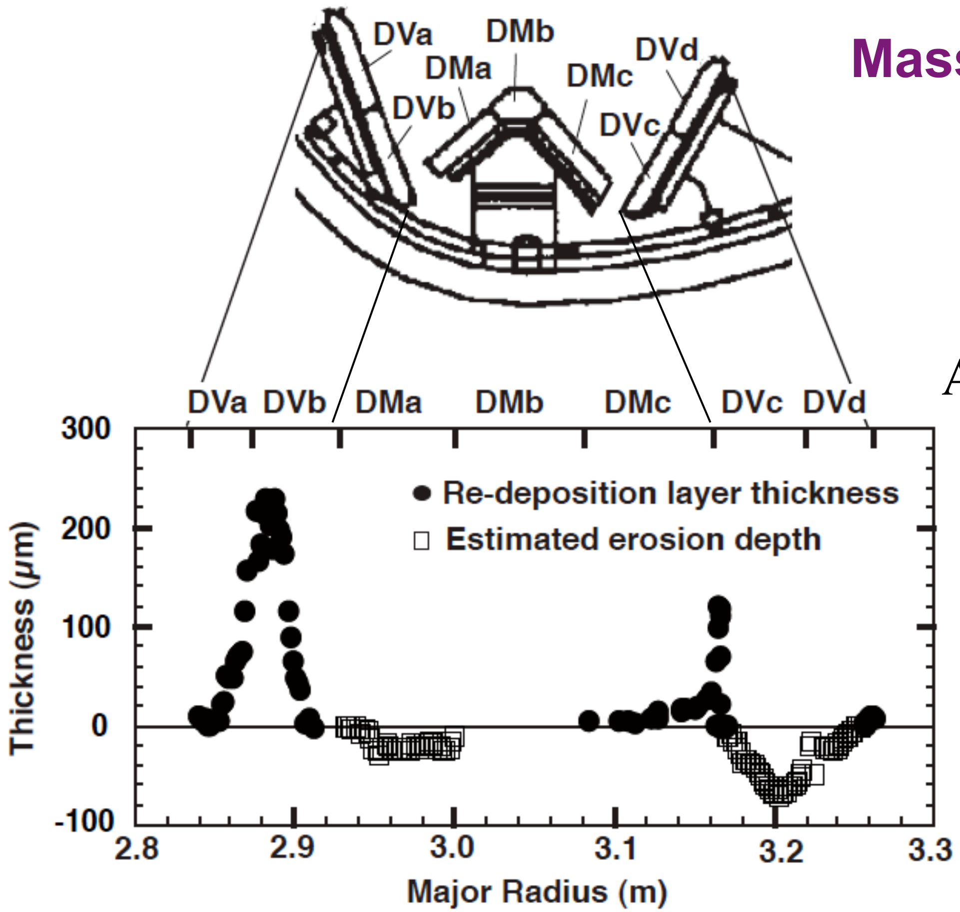
The temperature of plasma facing surfaces increased 50 -1000K owing to the plasma heating. Because of less heating power of H NBI, the temperature increments of plasma facing surface under HH discharges were significantly lower than that under the DD discharges.

Temperature rise owing to plasma heat load



Erosion/deposition profiles at divertor region

SEM observation and Micrometer measurements



Samples : 1997~2002

NB injection time :
3 x 10^4 s (outer dome wing:2 x 10^4s)

Mass balance of erosion and deposition

Materials density
Deposited layers : 0.91 g/cm^3,
Eroded region : 1.70g/cm^3)

Assuming toroidal symmetry,
Deposition : 0.55 kg (10.7 x 10^{20} C/s)
Erosion : -0.34 kg (-5.7 x 10^{20} C/s)

Missing : 0.21 kg (5 x 10^{20} C/s)

40% of the deposition on the divertor area must be originated from the main chamber wall.

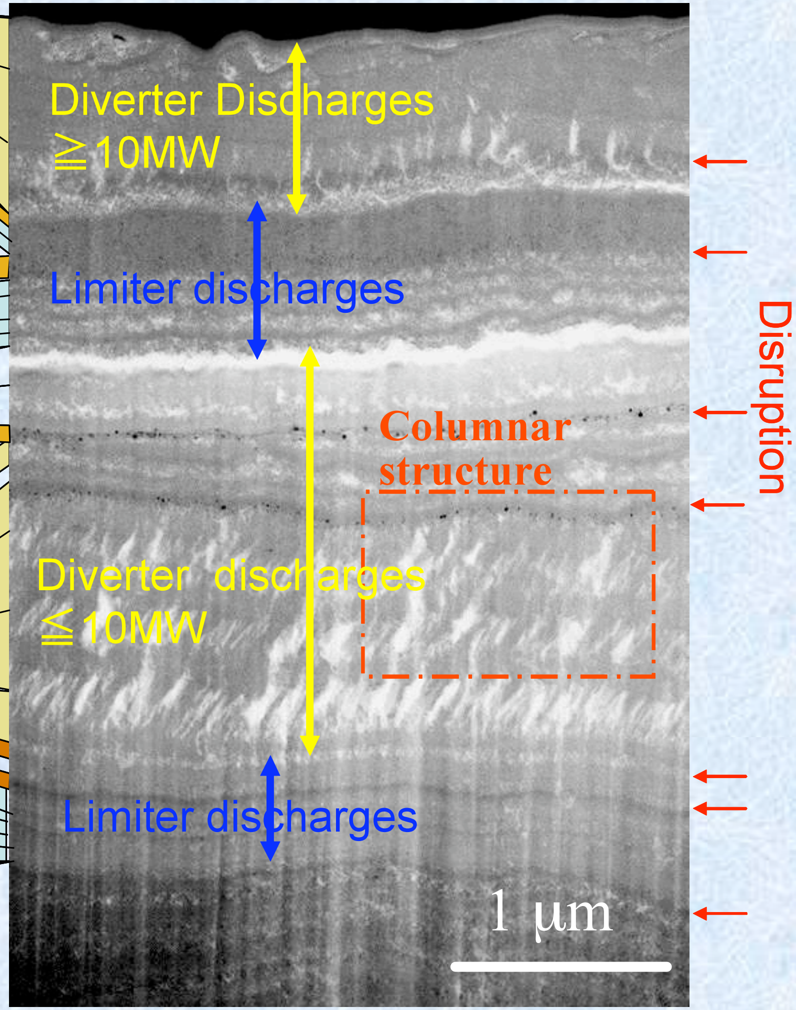
Y. Gotoh et al. J. Nucl. Mater. 357(2006) 138

Growth of redeposited layers in JT-60

Different growing mechanisms depending on discharge conditions, flux, temperature and so on

40 shots

		Δx(mm)	PNB(MW): L	PNB(MW): D	IP (kA)	t (s)	PNB*t(MJ)	Ris(mm)	Hard X-ray
8303	137.5	60		16	968	3.6	55		
8302	145.9	50		26	955	3.9	98		
8301	135.8	62		10	968	3.9	37		
8300									
8299			0.4		461				
8298			7.8		1499	1.7	13		60
8297			25		1461	3.9	99		
8296			24		1452	3.9	92		20
8295					458				
8294			1.9		1749	3.5	0		60
8293					468				
8292			1.9		1822	3.5	0		160
8291									
8290									
8289									
8288			10		1999	1.7	17		74
8287			8		1439		0		
8286			4		1455	3.3	13		
8285			2		1449	3.3	7		
8284			6		1464	3.3	19		
8283			4		1449	3.3	13		
8282									
8281	141.7	55		25	987	2.4	61		
8280	141.7	55		25	965	3.9	97		
8279			7.2		1034	0.5	4		
8278			16		946	1.9	30		
8277			16		936	1.9	30		
8276			16		955	1.9	31		
8275			16		949	1.9	31		
8274			25		949	0.5	12		
8273			0.3		471				
8272	141.7	55		25	977	3.9	98		
8271			3.8		1468	2.9	11		
8270									
8269	150.1	45		9	1493	2.5	22		61
8268	149.3	46		4	1474	2.5	9		
8267	154.4	40		11	1775	3.4	39		
8266									
8265	154.4	40		6	1775	3.6	20		
8264			20.3		2126	0.8	0		102
8263			24.5		2119	3.5	71		
8262			15.7		2189	1.1	8.8		10
8261			5.7		2138	3.3	17.8		
8260			0.2		487				
8259			3.6		2193	0.4	1		88
8258			25.2		949	2.5	63		
8257			25.0		942	2.5	63		
8256			23.2		971	2.5	58		
8255			22.8		971	2.5	57		
8254			25.2		2113	3.3	83		
8253			0.4		477				
8252			24.7		2221	0.8	0		78
8251	141.7	55		24.3	987	3.9	94		
8250	141.7	55		25.0	1002	3.9	96		
8249	141.7	55		23.1	980	4.7	108		8



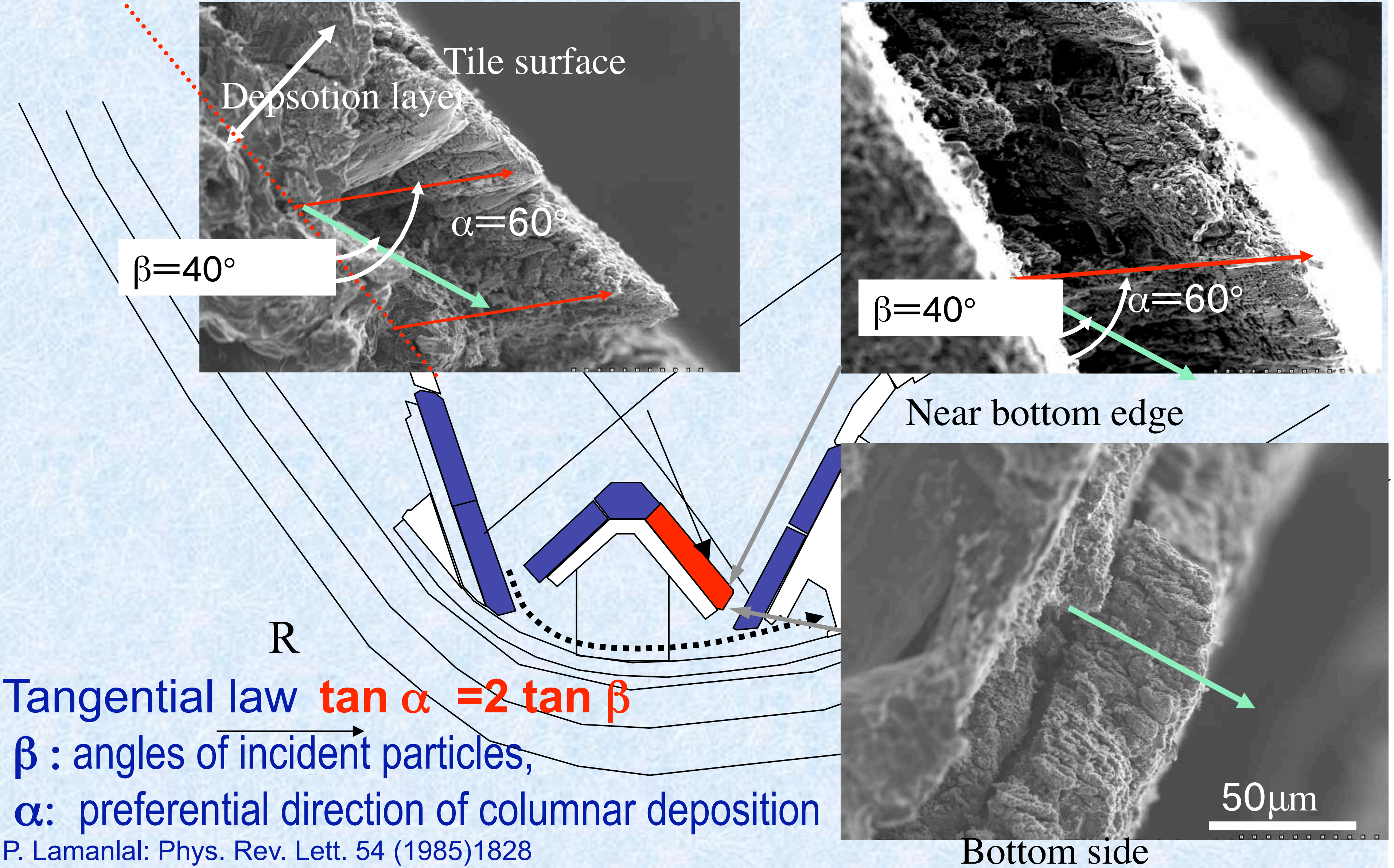
Thickness : Divertor discharges > limiter discharges

Lower power divertor discharges give columnar structure

Gotoh et al. J. Nucl. Mater., 329-333 (2004) 8404

Mechanism of Carbon Transport

Line of site deposition on outer dome wing



Tangential law **tan α =2 tan β**

β : angles of incident particles,

α: preferential direction of columnar deposition

P. Lamanlal: Phys. Rev. Lett. 54 (1985)1828

Mechanism of Carbon Transport

Deposition at inner target

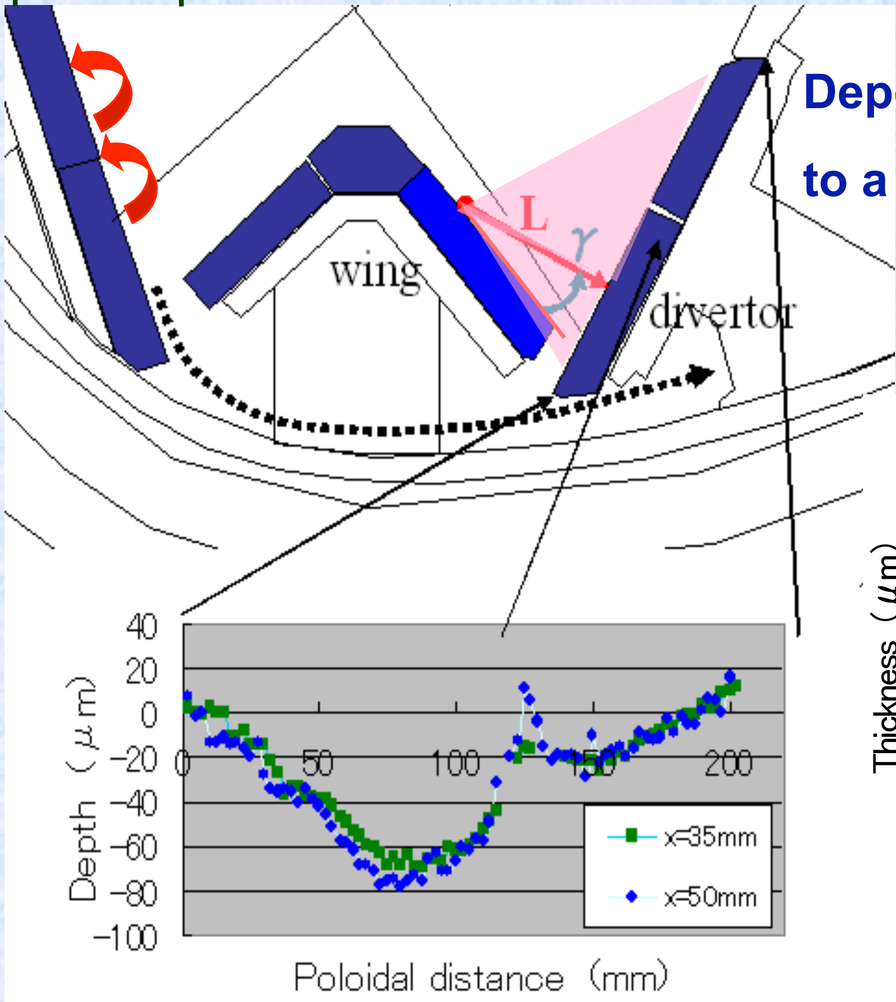
Repetition of erosion and prompt redeposition

Deposition at outer dome

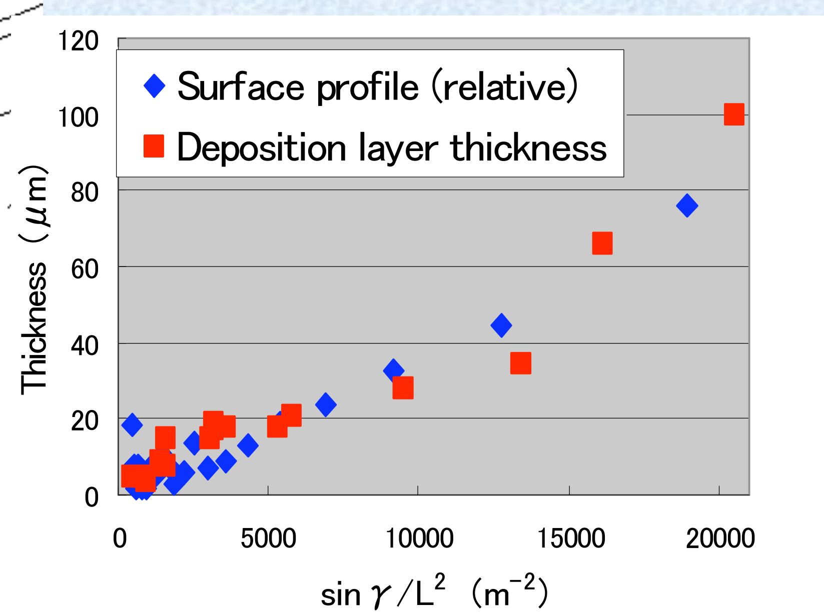
Direct transport of eroded carbon at outer divertor

Deposition thickness is proportional to a solid angle from outer divertor

$$d \propto \sin\gamma / L^2$$



Erosion profiles of outer divertor tiles

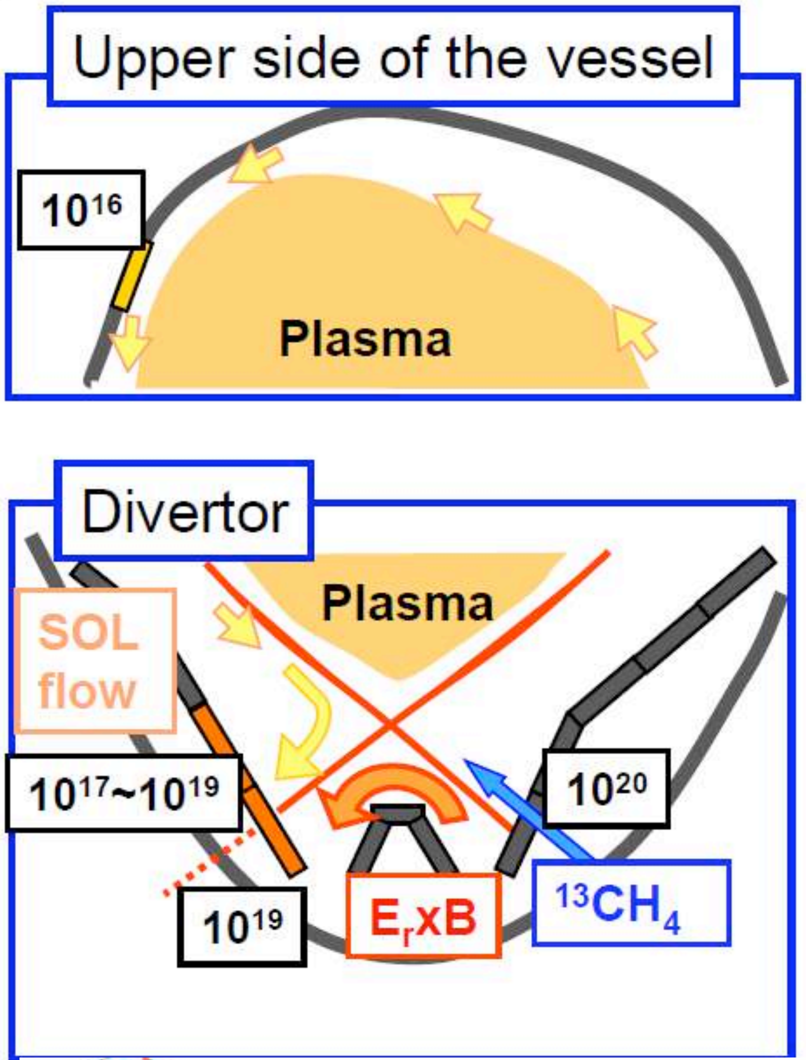


Y. Gotoh et al. J. Nucl. Mater. 357(2006) 138

Mechanism of Carbon Transport

JT-60U

Schematic of ^{13}C transport projected to poloidal cross section



$^{13}CH_4$ puffing at outer divertor

- ^{13}C deposition on surface first wall tiles was very small
- ^{13}C deposition peak is slightly shifted toward the pumping slot than the peak position of C deposition
- Direction of the drift flux in the private region was toward the inner divertor (Reciprocating mach probe measurements)

⇩

Carbon transport through private region

Deposition at remote area (Bottom of Divertor)

caused by line of sight transport from eroded area

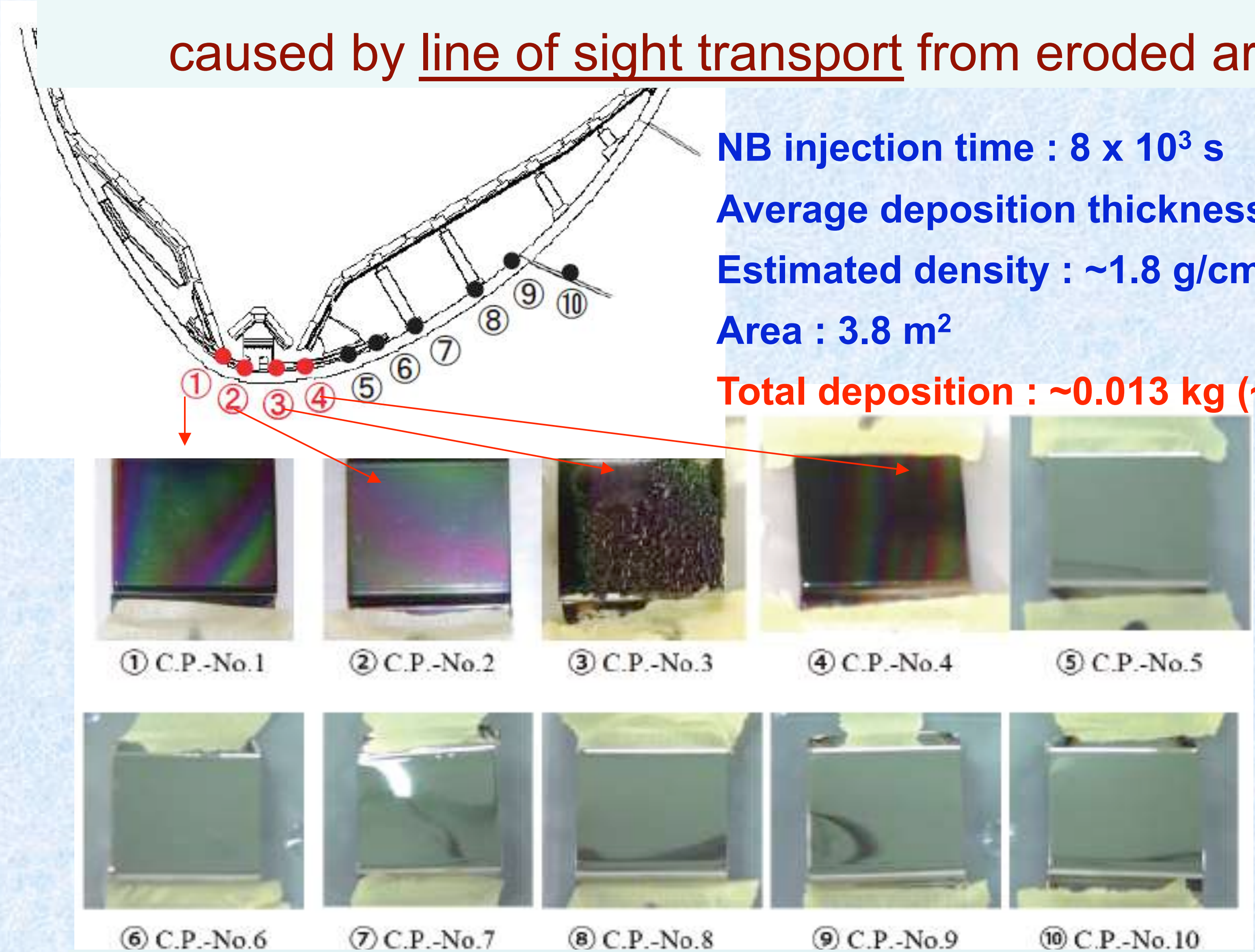
NB injection time : 8 x 10^3 s

Average deposition thickness : ~2μm

Estimated density : ~1.8 g/cm^3

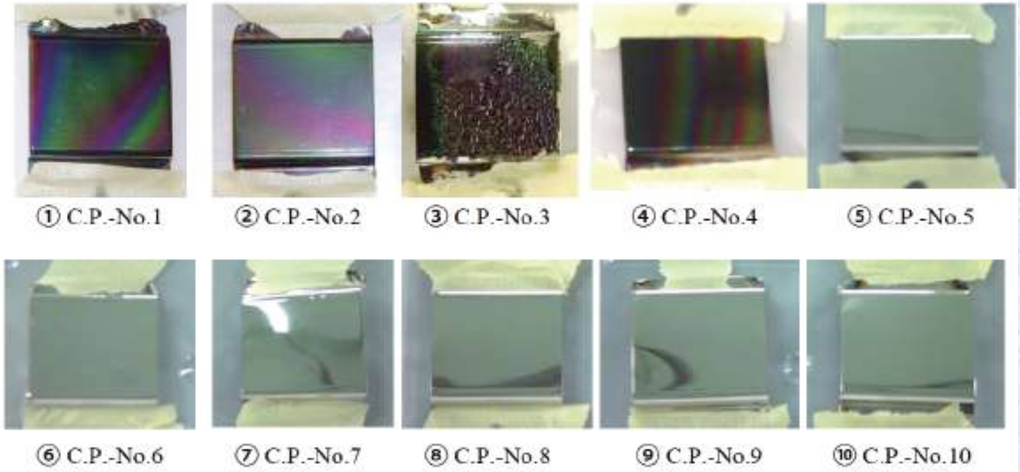
Area : 3.8 m^2

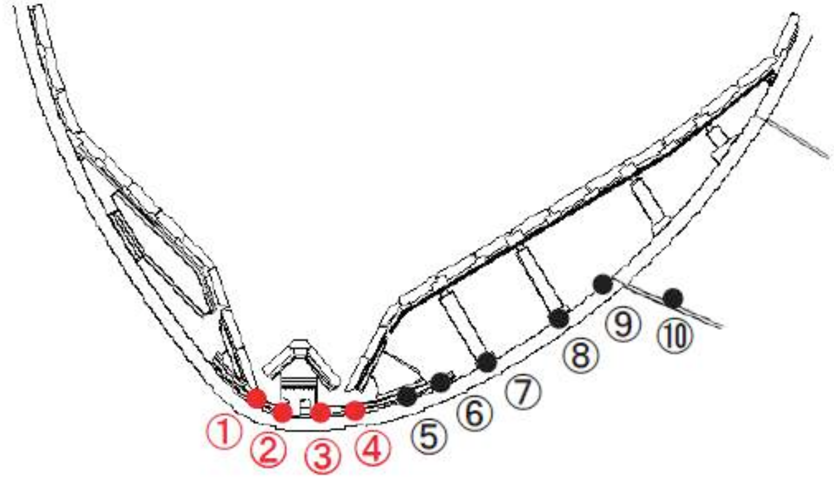
Total deposition : ~0.013 kg (~8 x 10^{19} C/s)



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

Carbon deposition pattern at remote area well corresponds Tritium profile

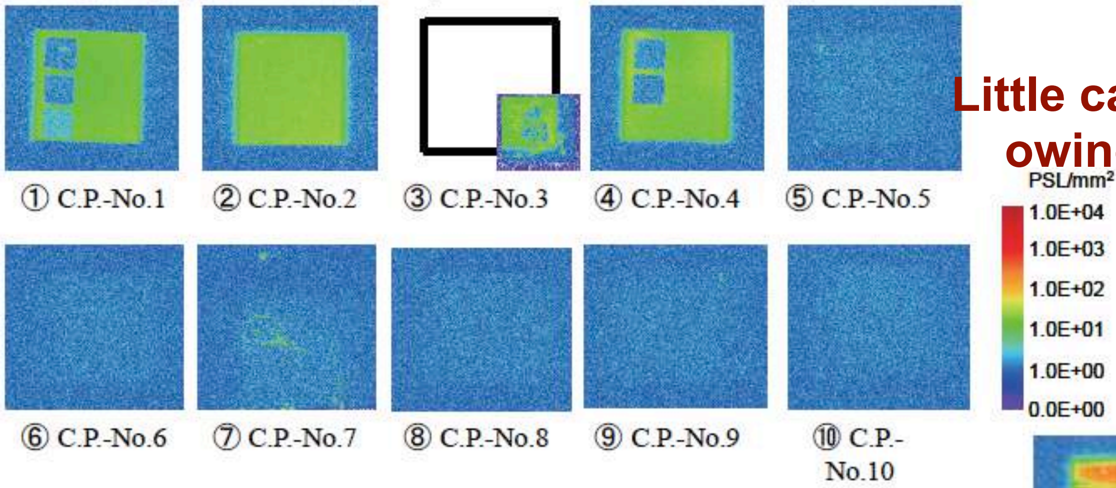


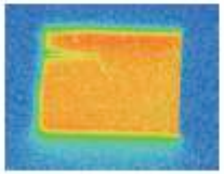


No Carbon deposition at far remote area evidenced by tritium retention

Little carbon exhaust owing to high temperature operation?

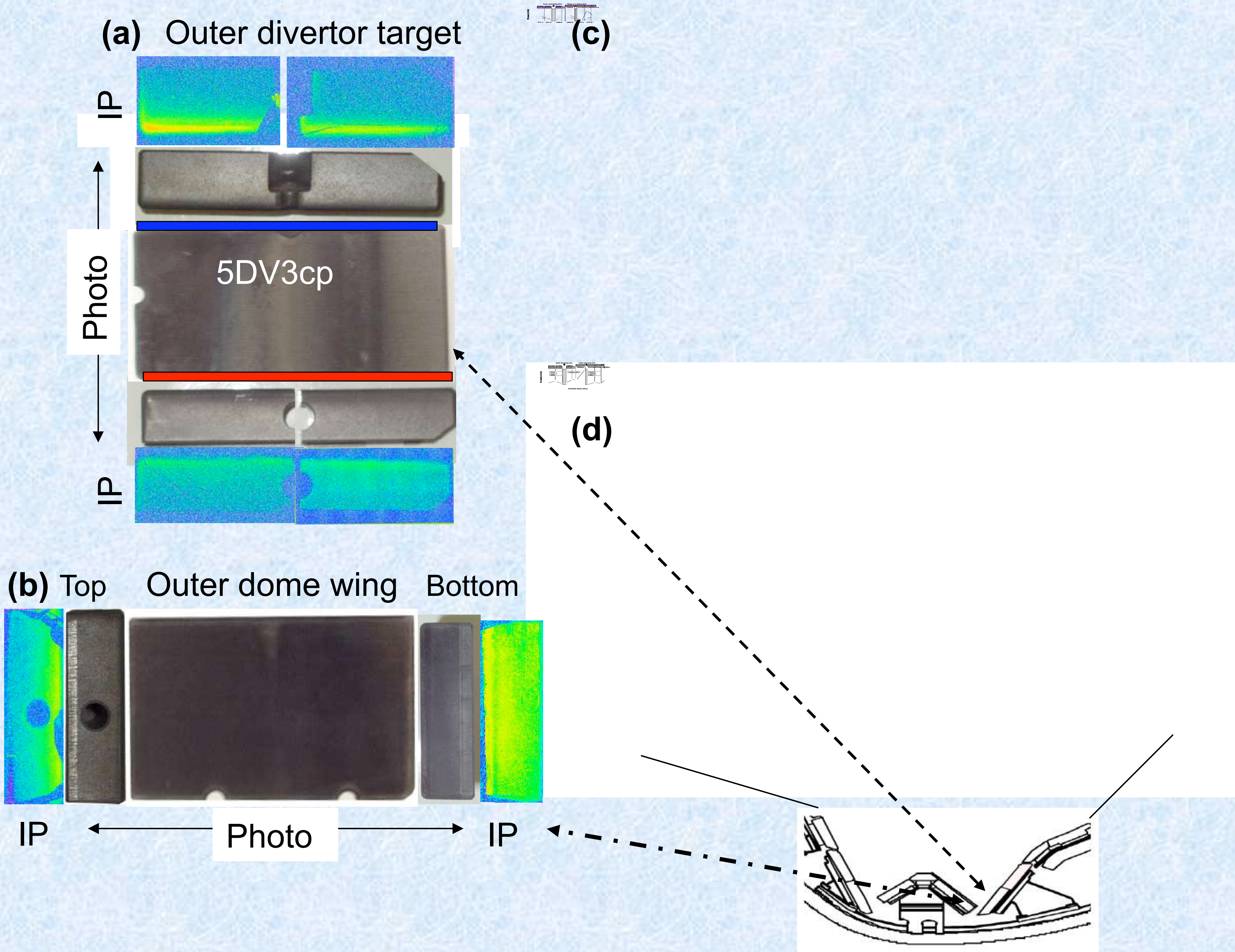
Imageing plate analysis (Tritium measurement)





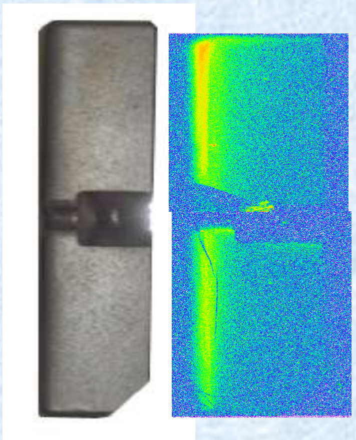
Reference:
Dome top tile
~ 60 kBq/cm^2

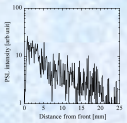
K. Masaki et al. Nucl. Fusion, 47(2007)1577


(a) Outer divertor target
IP
Photo
5DV3cp
IP
(c)
(d)
(b) Top
Outer dome wing
Bottom
IP
Photo
IP

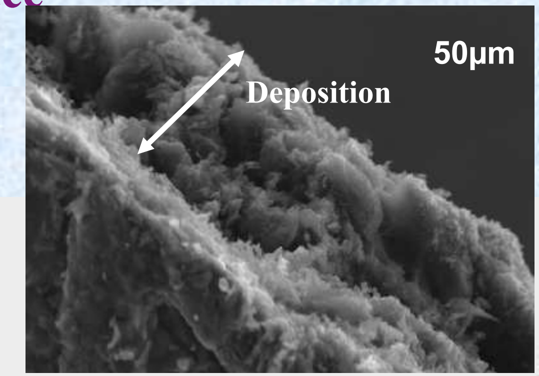
Carbon depsotion at tile gaps -Two different mechanisms -

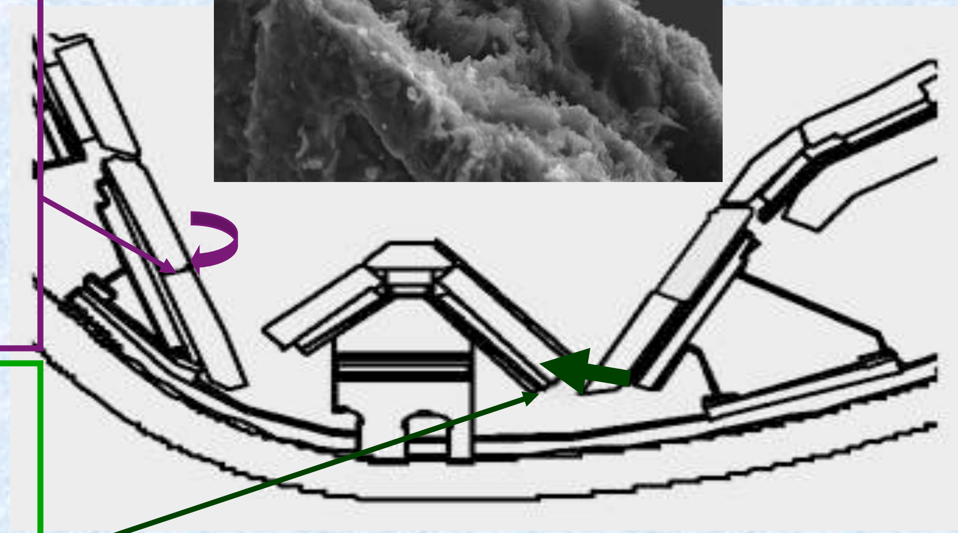
Prompt redeposition of eroded carbon at front surface



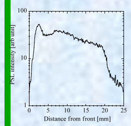


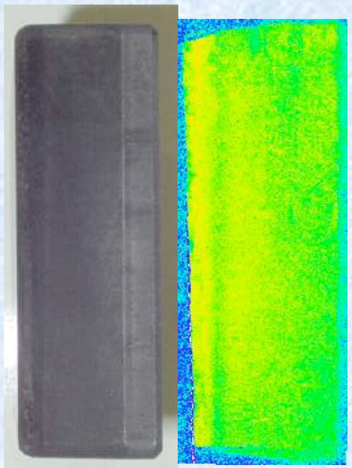
e-folding length: **7 mm**

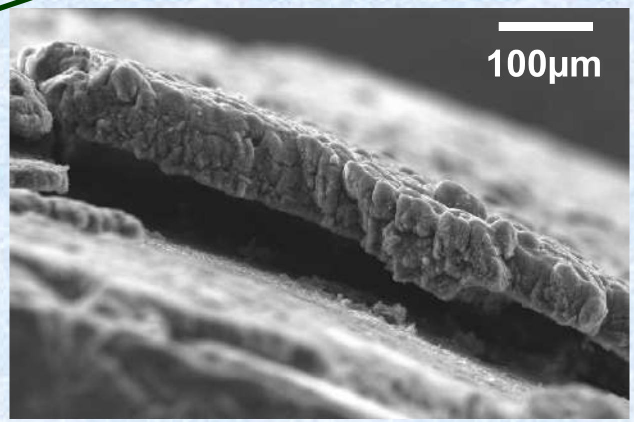




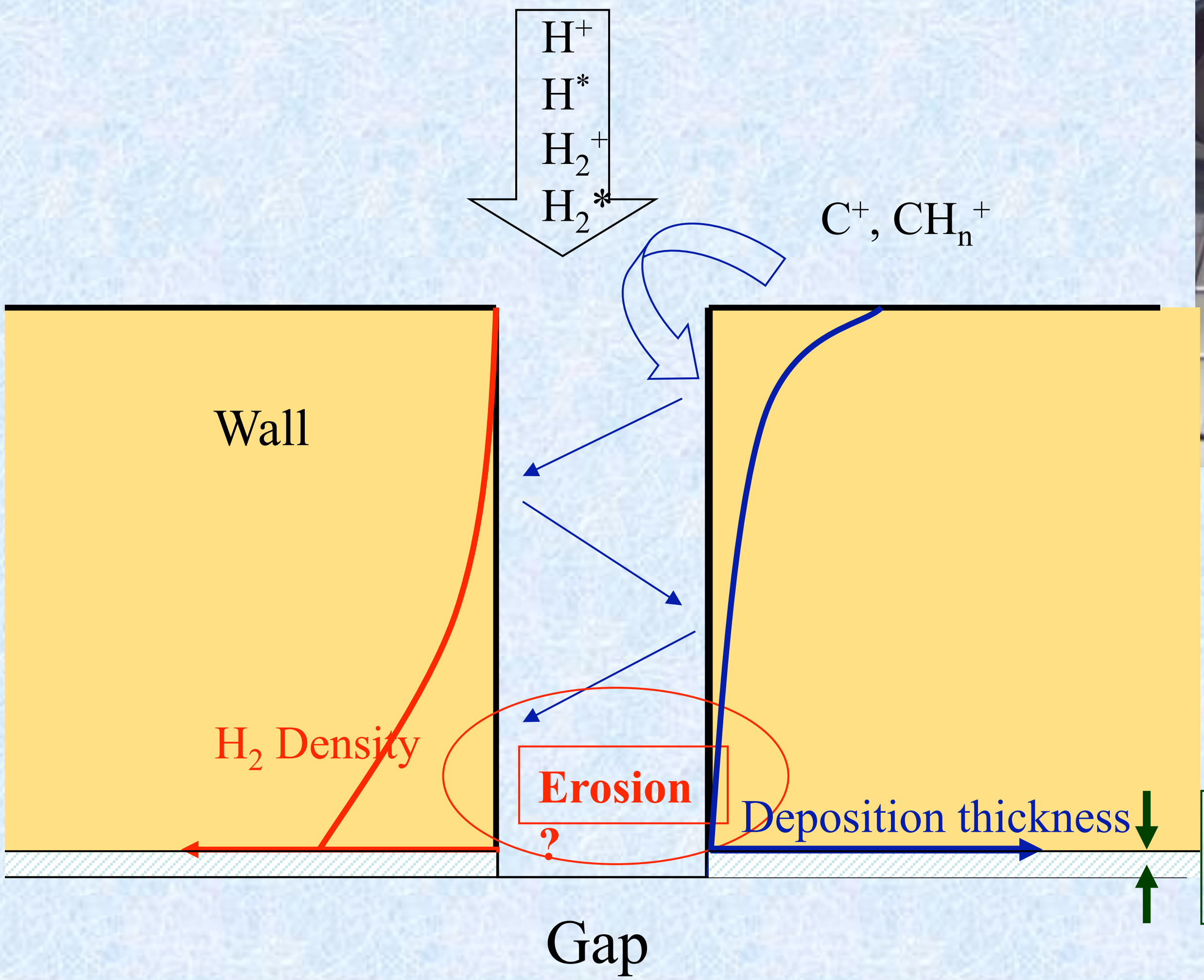
Line of sight deposition from eroded area

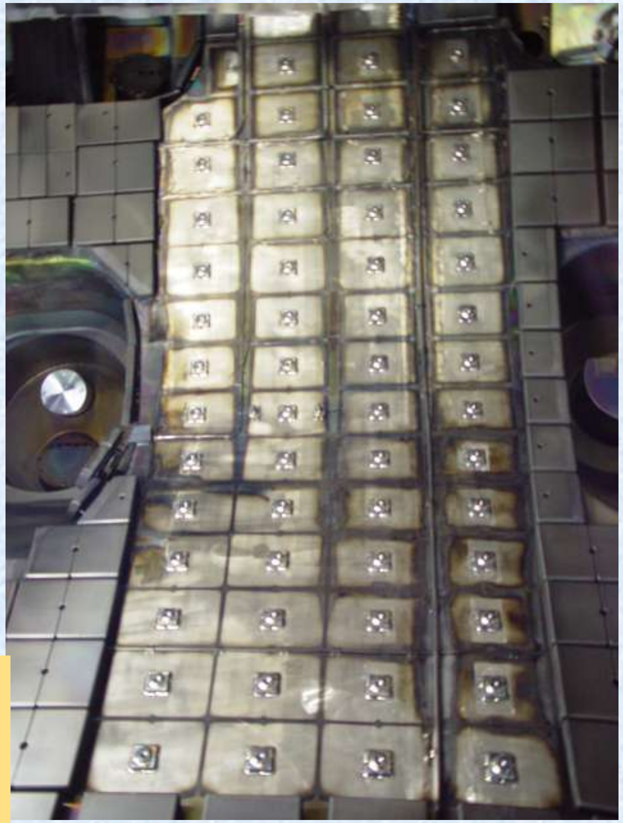






Deposition at the deeper area and/or bottom of closed gap seems small





Deposition at gap between tile and base plate

Summary on Carbon deposition observed in JT-60

1. Deposition at PFS

- Large at inner divertor and outer dome wing
- Different deposition mechanisms ;
  Lamellar type, Columnar Type, Amorphous Type (High H/C case)

2. Deposition in the gap

- The gaps at the first wall would be very small

3. Deposition at remote area

- Can not be avoided.

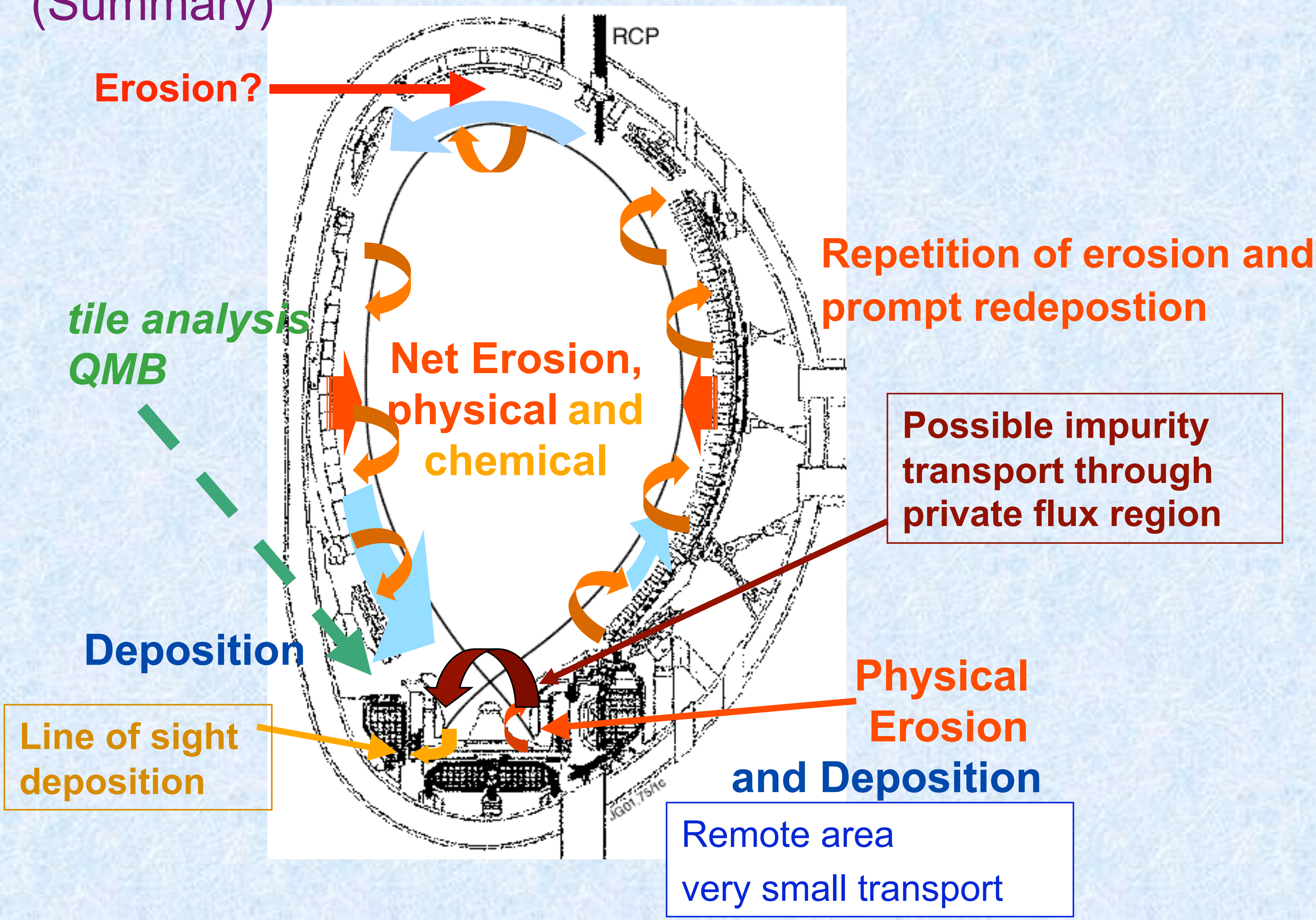
  Mostly appeared at the line of site from the eroded area.

  Could be reduced by appropriate divertot geometry

4. Far remote area

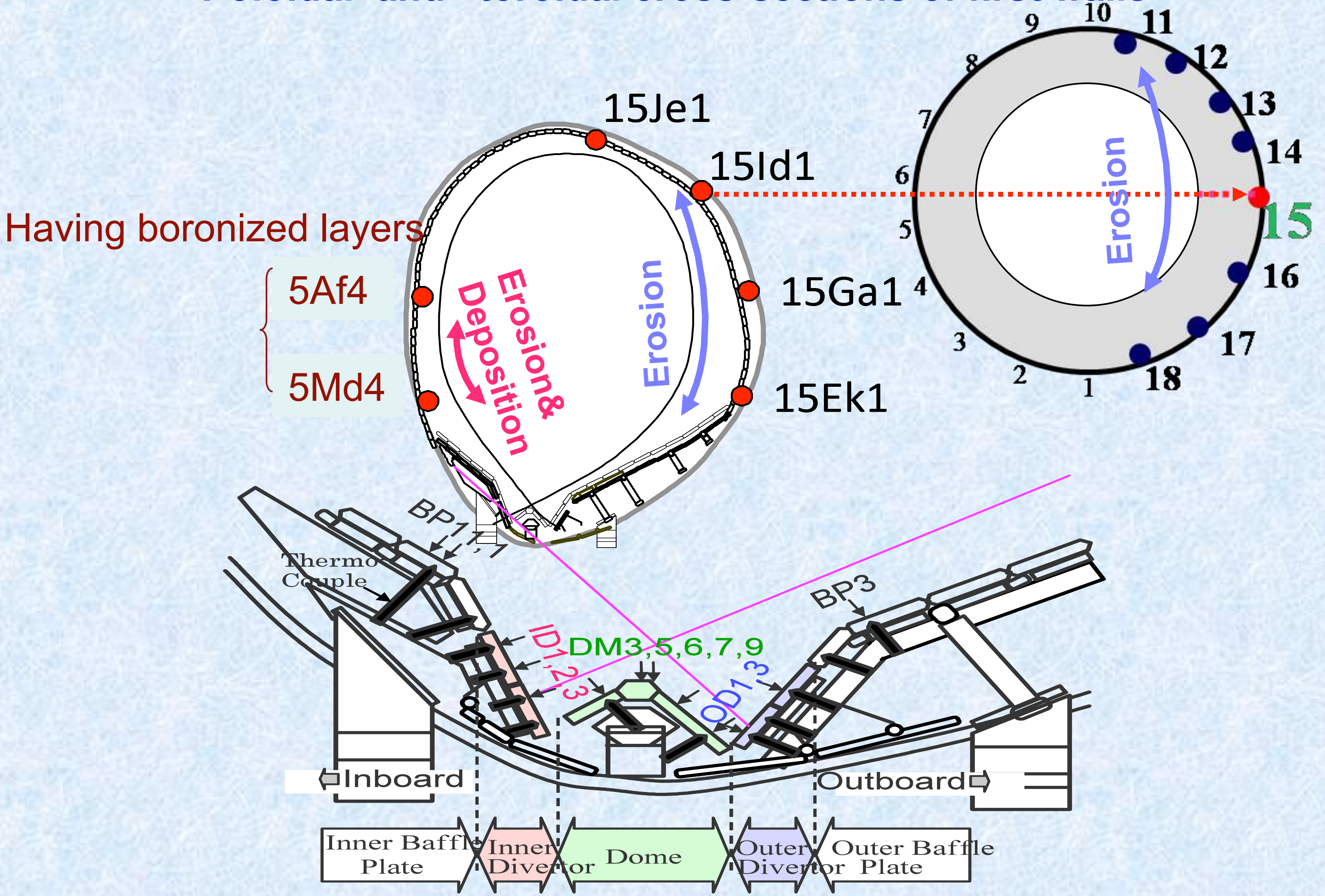
- Little deposition (Could be owing to high temperature operation)

Possible mechanisms of carbon erosion and transport (Summary)



Retention of D and H - Locations of analyzed tiles -

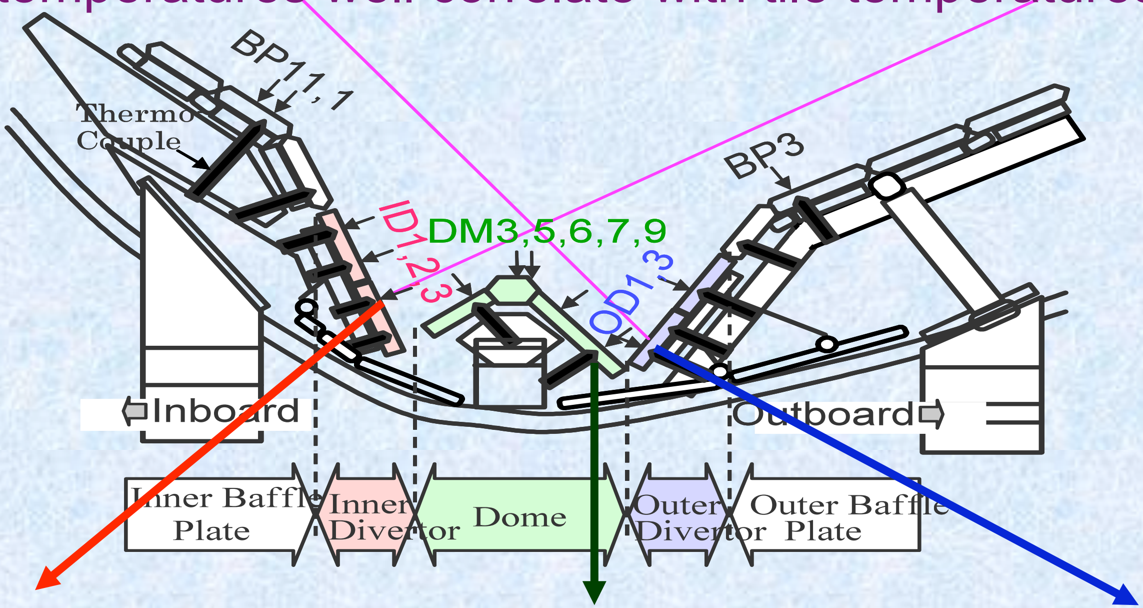
Poloidal and toroidal cross-sections of first walls



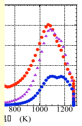
W shaped open divertor

Hydrogen retention determined by TDS

Peak temperatures well correlate with tile temperatures



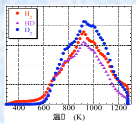
Redeposits on ID3



1000K

Temperature/K

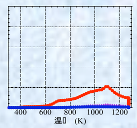
Redeposits on DM9



800K

Temperature/K

Eroded area OD1



1400K

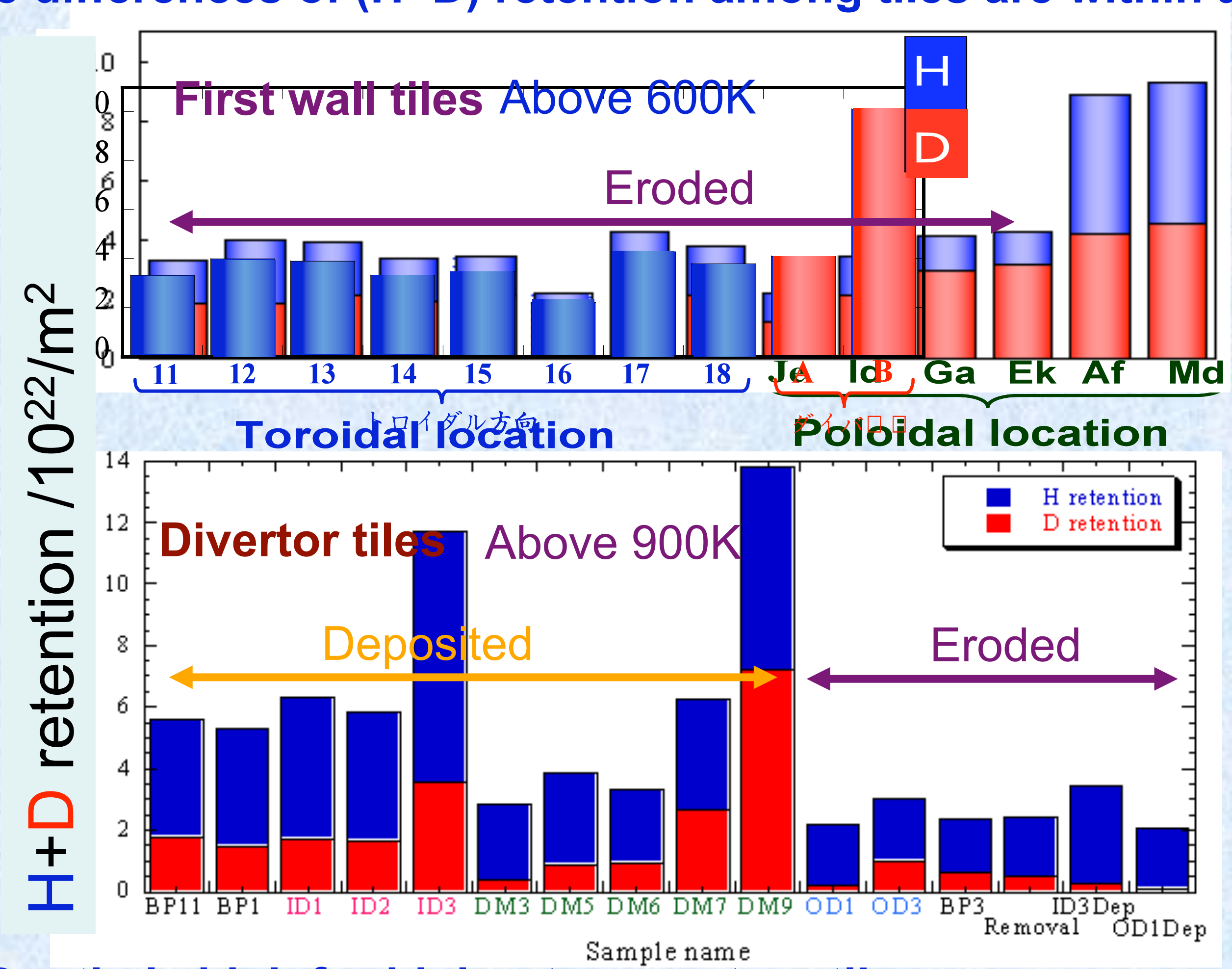
Temperature/K

Desorbed amount/ 10^{19} molecules·m^{-2}·s^{-1}

Comparison of H+D retention in near surface layers

Total retention ~ $10^{23}/m^2$

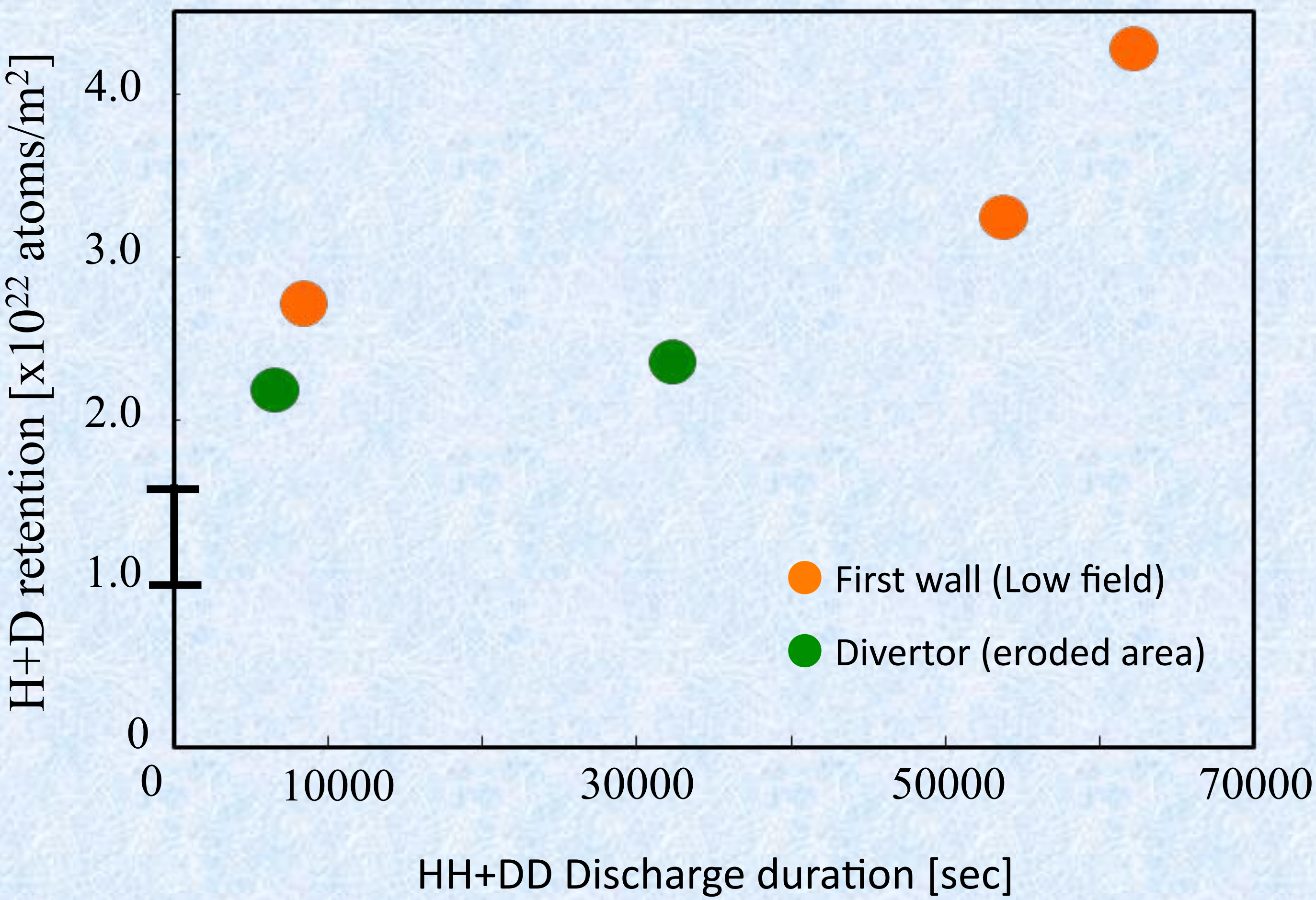
- The differences of (H+D) retention among tiles are within a factor of 10.



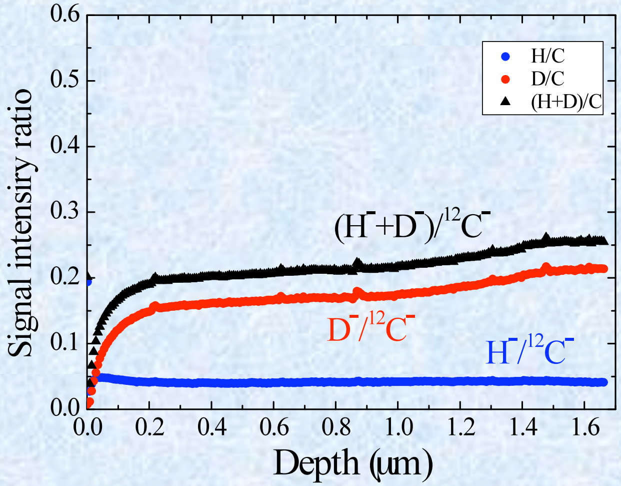
- H/D ratio is high for higher temperature tiles.

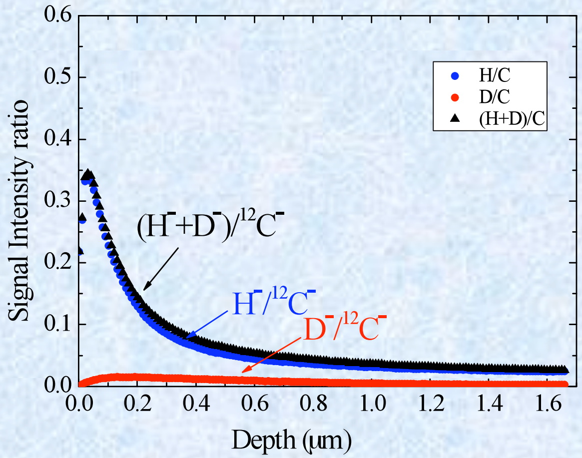
- Deuterium once retained in the wall during the DD shots was isotopically replaced by H under the HH discharges.

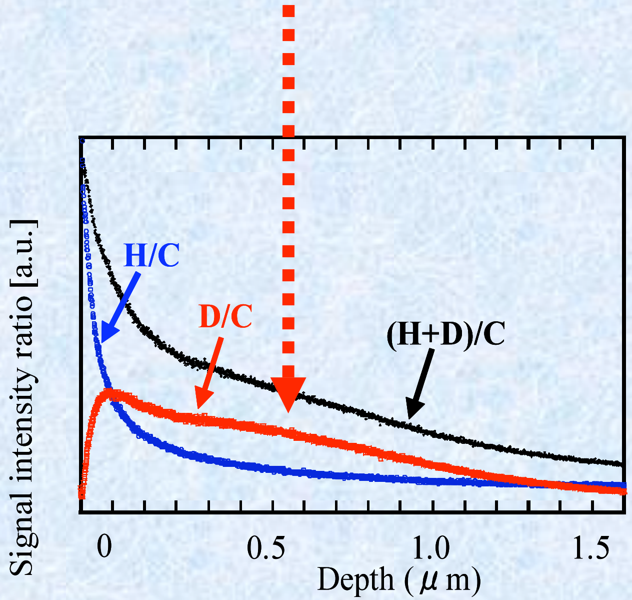
H+D retention in plasma facing surface layers is likely saturated

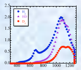


Depth profile by SIMS indicates deep D implantation

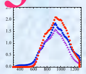




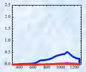




Redeposits on ID3



Redeposits on DM9



Eroded area OD1

1000K ↓

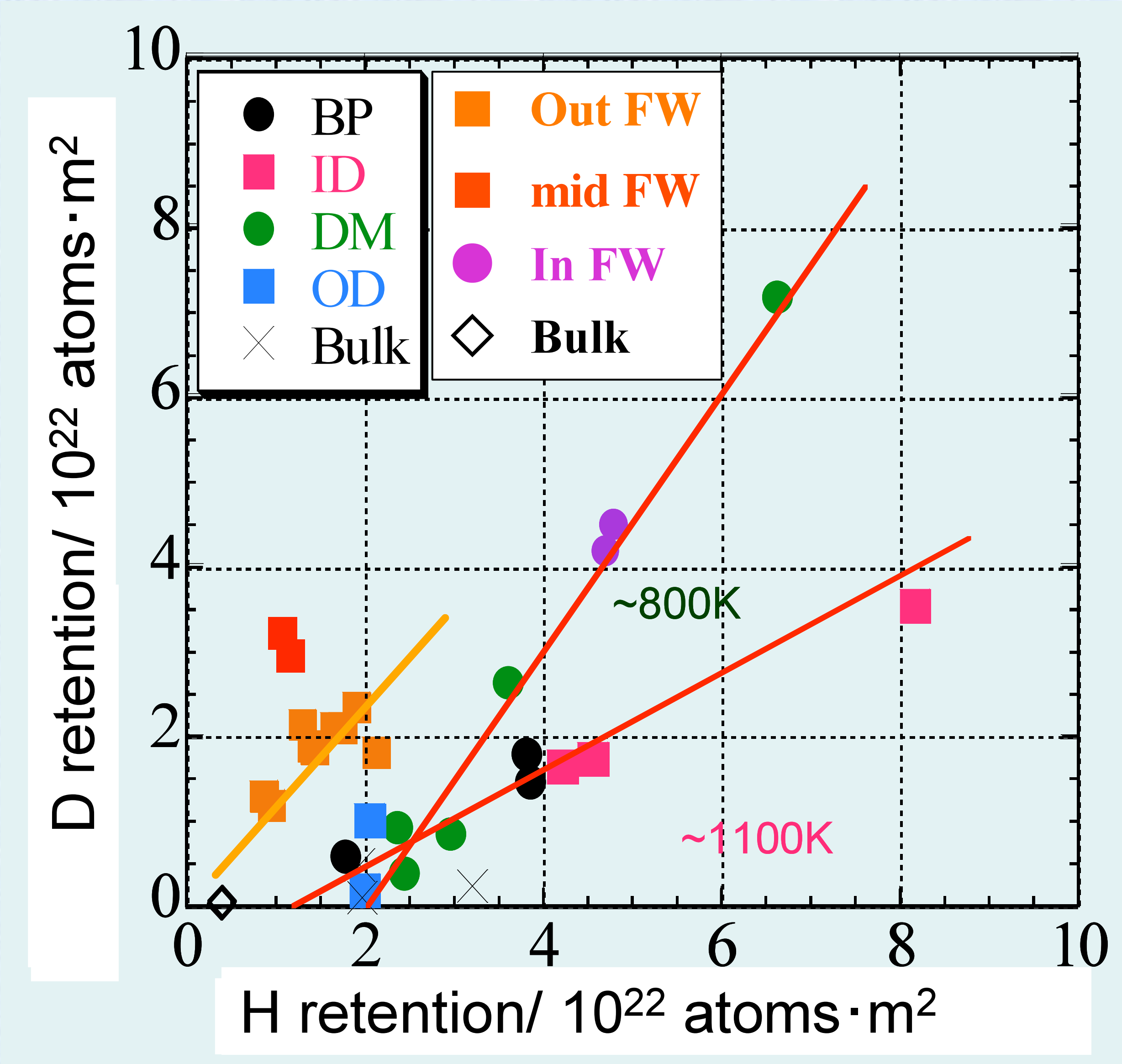
800K ↓

1400K ↓

H > D

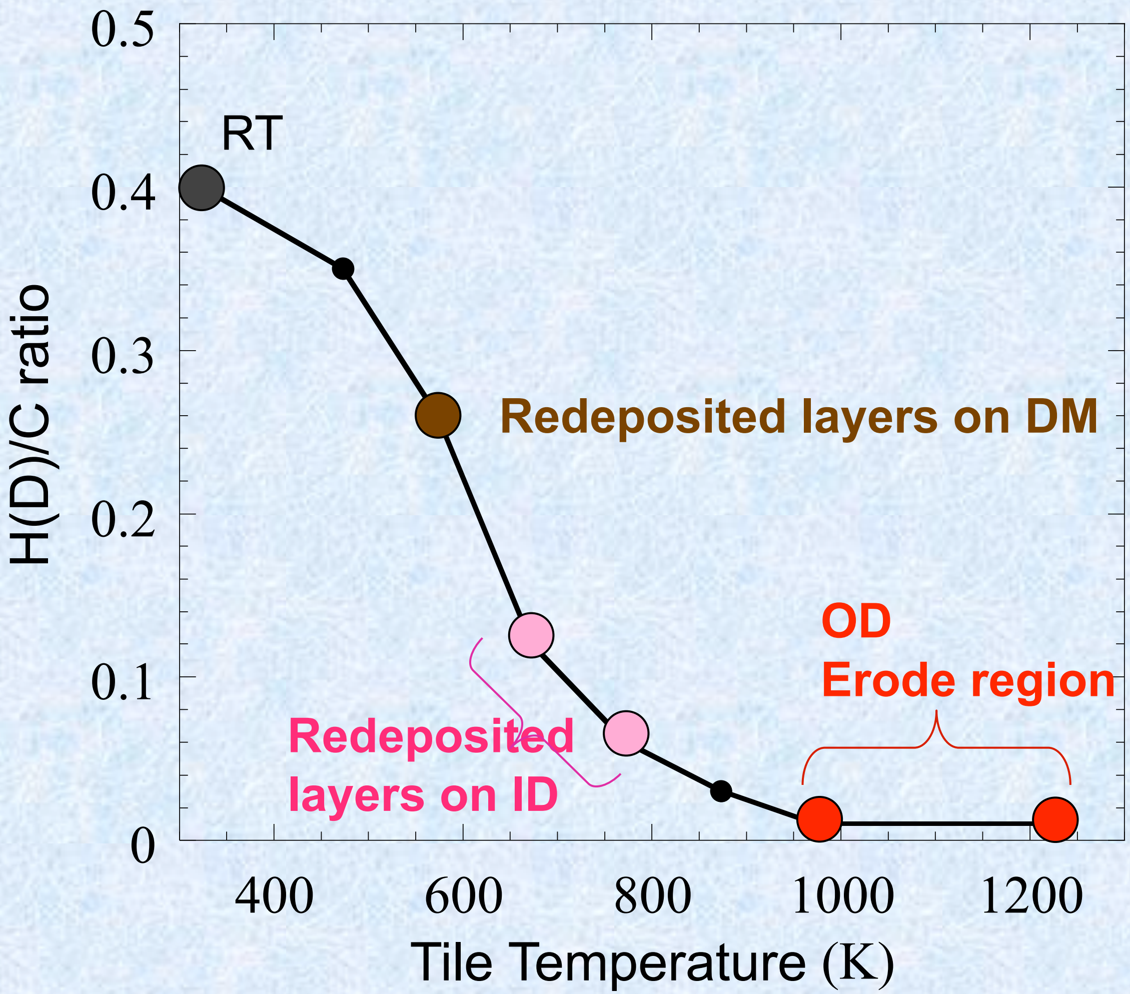
H ~ D

H > D



BP
ID
DM
OD
Bulk
Out FW
mid FW
In FW
Bulk
~800K
~1100K
D retention/ 10²² atoms·m²
H retention/ 10²² atoms·m²
0
2
4
6
8
10

Temperature dependence of H+D retention



Redeposited layers on DM (Dome)

Lowest Temp.

Inhomogeneous retention

ID (Inner divertor)…

Higher Temp (Thermally isolated from the substrate)

Homogeneous distribution

Eroded region

OD (Outer divertor)…

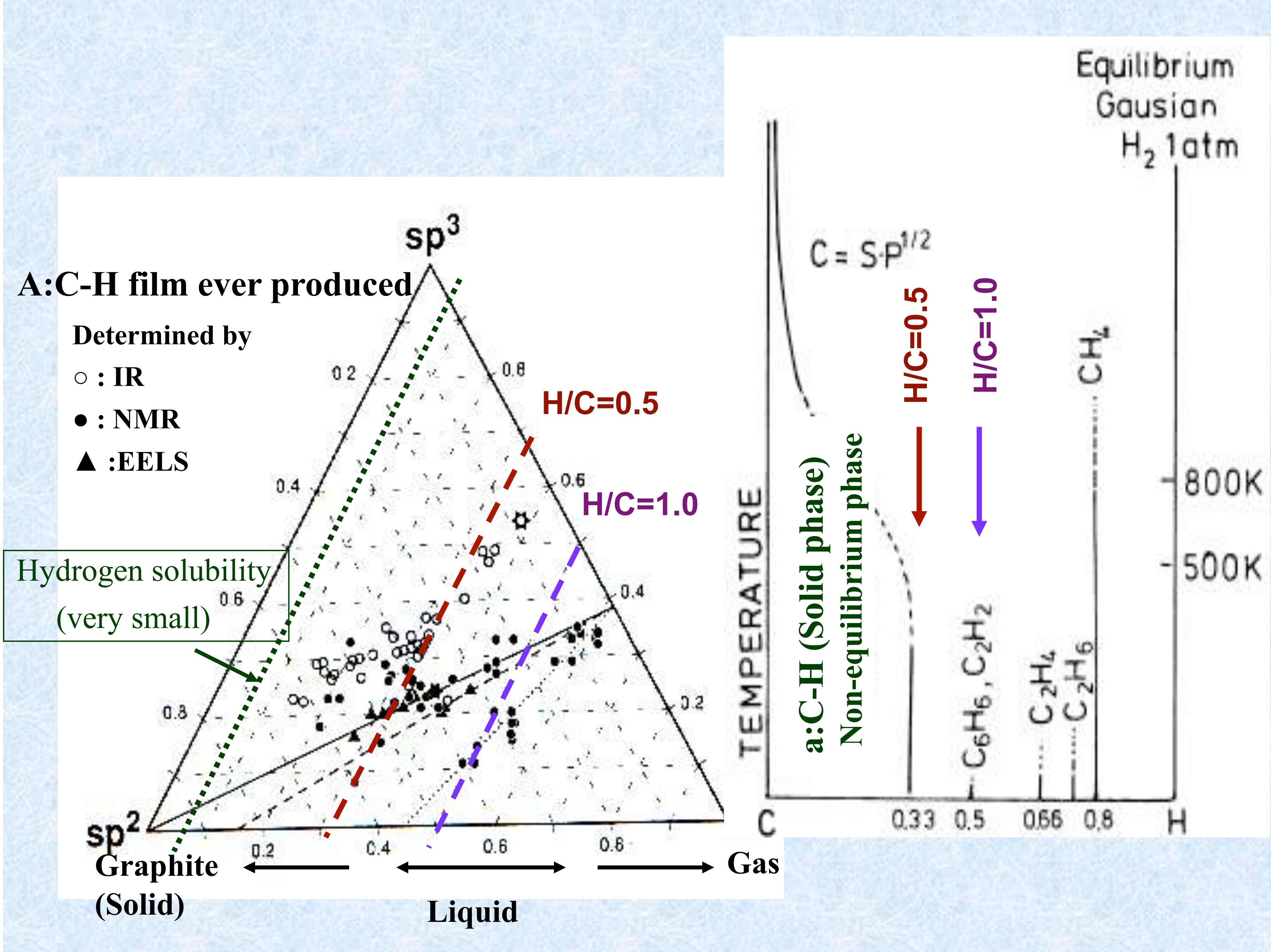
Highest Temp.

Lowest Conc.

Benefit of high temperature operation

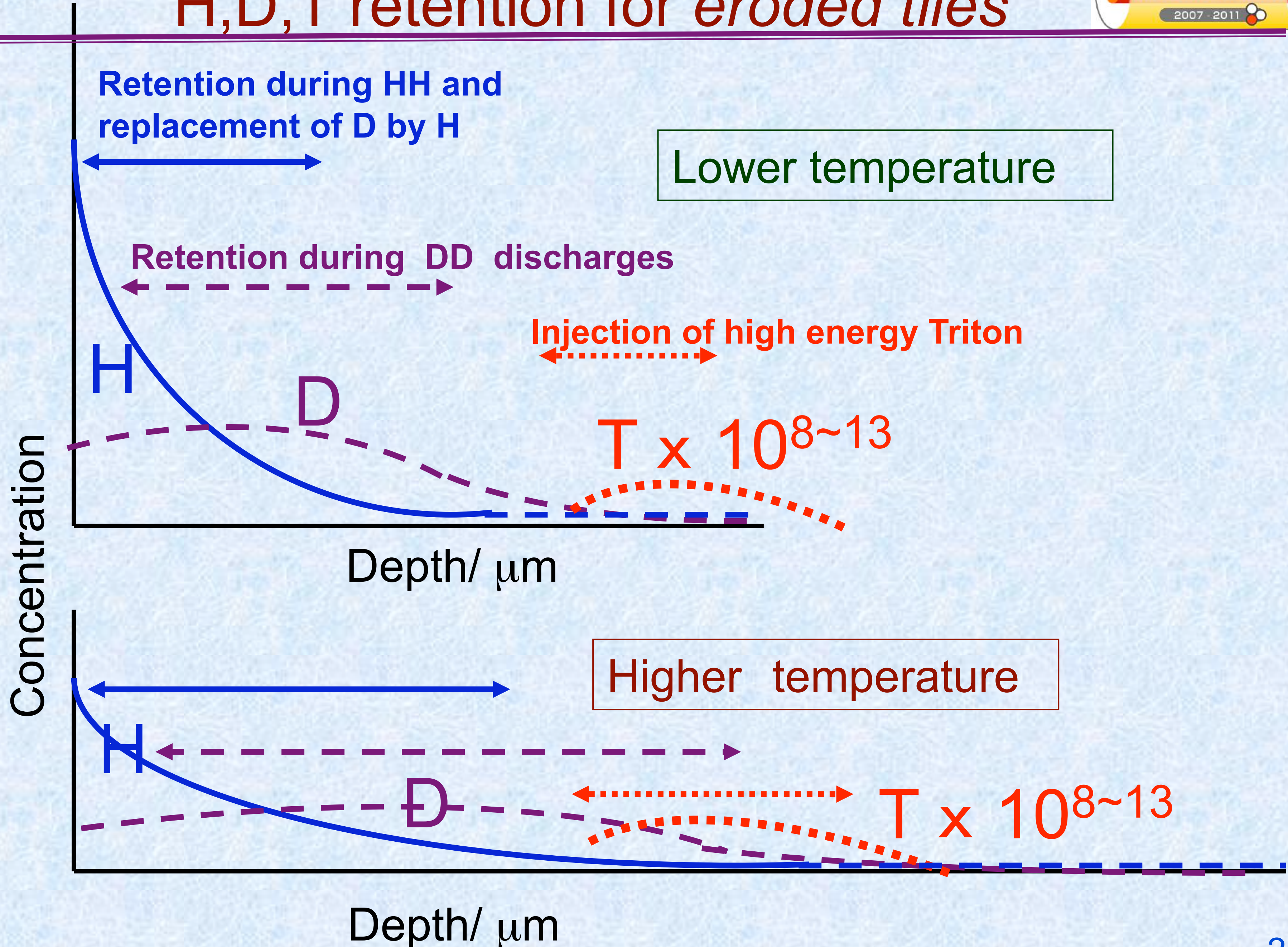
Surface hydrogen concentration is very likely saturated.

The saturated concentrations for the eroded area and redeposited layers were nearly the same and only depending on the surface temperature.



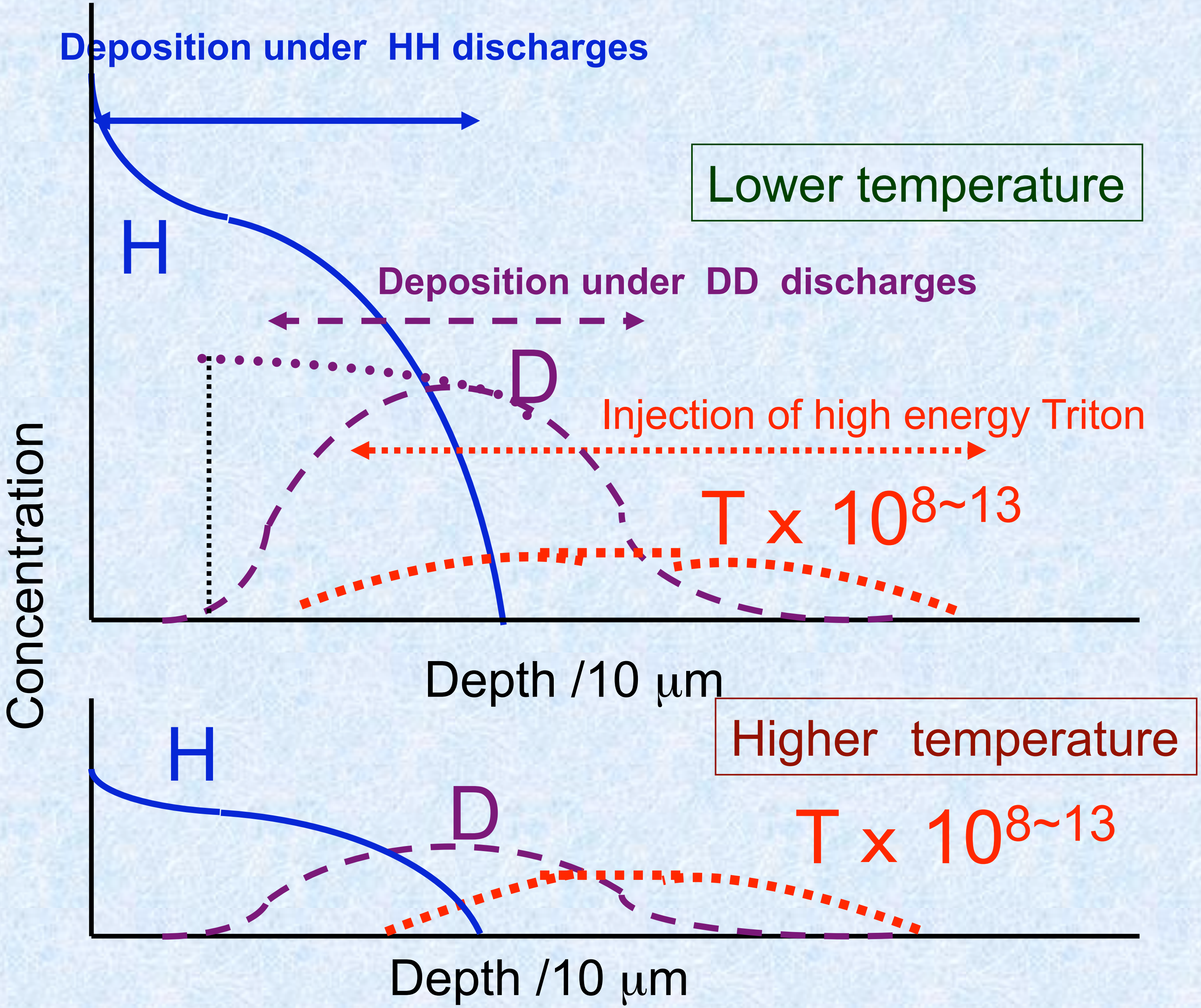
A:C-H film ever produced
Determined by
○ : IR
● : NMR
▲ :EELS
sp3
sp2
0.2
0.4
0.6
0.8
H/C=0.5
H/C=1.0
Hydrogen solubility
(very small)
Graphite
(Solid)
Liquid
Gas
Equilibrium
Gausian
H2 1atm
C = S·P1/2
TEMPERATURE
a:C-H (Solid phase)
Non-equilibrium phase
800K
500K
C6H6, C2H2
C2H4
C2H6
CH4
C
0.33
0.5
0.66
0.8
H

H,D,T retention for *eroded tiles*



H,D,T retention for *redeposited tiles*

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011



Summary of Part II (Erosion and H retention) Cont'd

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

1. Carbon erosion and deposition in JT-60U

- Deposition occurs mostly at the inner divertor probably owing to repetitive process of eorosion and prompt redeposition
- Deposition at outer dome wing and divertor shadowed area is caused by line of sights from the eroded area
- Deposition in tile gaps is not large, except open gaps connected to pumps

2. Retention and depth profiles of three hydrogen isotopes in JT-60U

- Hydrogen retention at the plasma facing area is very likely saturated and would not linearly increase with time.
- The isotopic ratios of retained hydrogen near surface layers are always equilibrated with incoming hydrogen fluxes (H/D/T).

3. Effects of high temperature deposition

- Possible saturation of T retention at plasma facing surface and less T retention
- Deposits at high temperature have less T and show strongly adhesion

4. Importance of Geometry

- Tile alignment, Gap width, Divertor geometry could reduce erosion.
- Plasma shaping could suppress erosion

Summary of Part II (Retention of H and D) Cont'd

Scientific Research in Priority Areas
Tritium for Fusion
2007 - 2011

- Saturation of H retention on the plasma facing surface would not allow linear increase of T retention with time.
- The isotopic ratios of retained hydrogen near surface layers are always equilibrated with incoming hydrogen fluxes (H/D/T).
- Depth attaining this equilibrium is quite thick owing to the porous nature of carbon materials and is increased by temperature rise.
- Hence tritium retention in plasma facing surfaces (both eroded and redeposited) would be significantly reduced by isotopic replacement by DD discharges subsequently made after DT discharges.
- All these results from JT-60 is promising to use carbon as PFM at high temperature (above 800K)

Summary of Part II (Deep implantation)

- Since JT-60U had rather large magnetic ripple loss, the loss or injection of high energy triton to the deep in the first wall was appreciable. Different from the plasma particle injection which would cause the near surface saturation of hydrogen, the deep implantation with less flux could pile up for long time.
- The deep implantation of energetic hydrogen could enhance hydrogen retention even for metallic first wall.

Deposited area	Location	Deposition rate x 10^{20} atoms/s	H+D retention rate x 10^{19} atoms/m^{2}s	(H+D)/C	D/H
	Inner divertor	~ 6	~ 1	~ 0.02	~ 0.4
	Outer dome wing	~ 4.5	~ 6	~ 0.13	~ 1.2
	Bottom of divertor (Base Temp. 420K)	~ 0.85	~ 6	~ 0.75	~ 3.6
	First wall (low field side)	~ 0.0015	~ 0.0024	~ 0.16	~1
Eroded area		Erosion rate x 10^{20} atoms/s	H+D total retention x 10^{22} atoms/m^2	(H+D)/C	D/H
	Inner dome wing	~ 1.5	~ 2	not evaluated	~ 0.07
	Outer divertor	~ 4.2	~ 3	~ 0.07	~ 0.31
	First wall (low field side)	not evaluated	~ 2-4	~0.0004	1 ~ 4