Studies of heat and particle loadings in long pulse plasmas and influence on PWI in JT-60U

Nobuyuki Asakura,

T. Nakano, H. Kubo, K. Masaki

Japan Atomic Energy Agency,

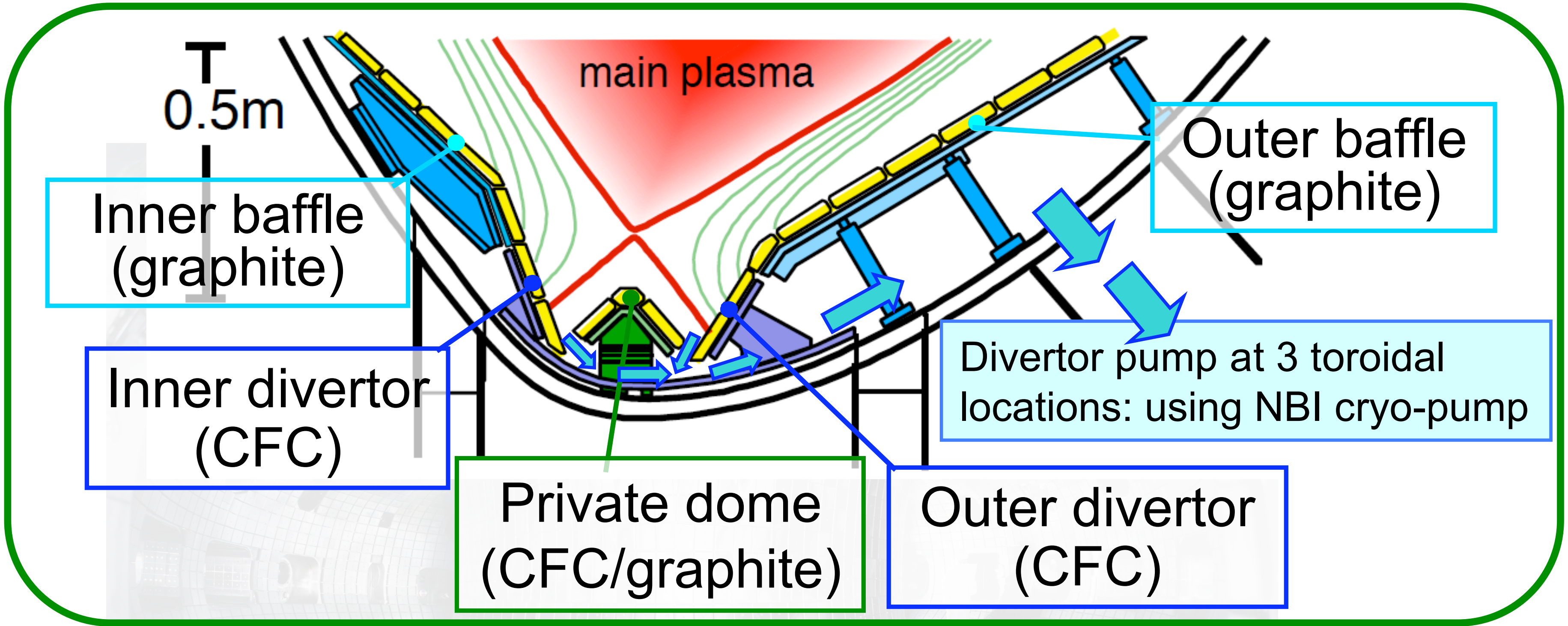
1st NIFS-CRC International symposium and
1st Korea-Japan workshop on "Edge Plasma and Surface Component Interactions in Steady State Magnetic Fusion Devices"

NIFS, 2007 May 21-22

Contents

1. Introduction:
    - Plasma Facing Components in JT-60U
    - Heat and particle control in long pulse
    - Carbon deposition and tritium retention
2. Wall-pumping and particle control in long pulse discharges
3. Carbon deposition and D/H retention at PFC
4. Summary

JT-60U W-shaped divertor with 3 cryo-pumps



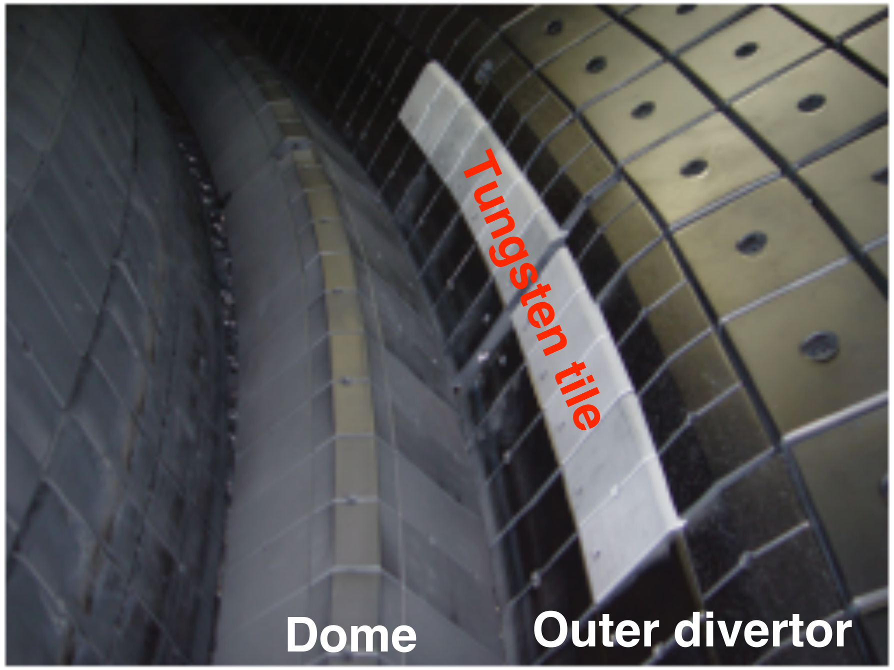
Divertor plays an important role in protecting first wall against large heat and particle fluxes.

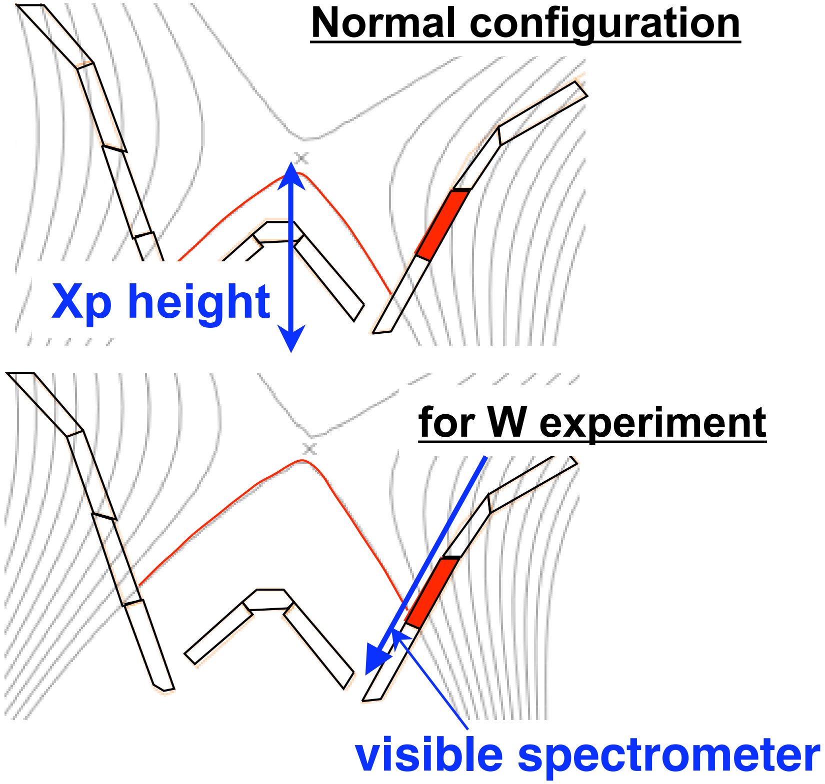
Particle exhaust of neutrals and He ash (low energy α-particle) is most important issue for steady-state operation.

W erosion and transport have been investigated (2003-2006)

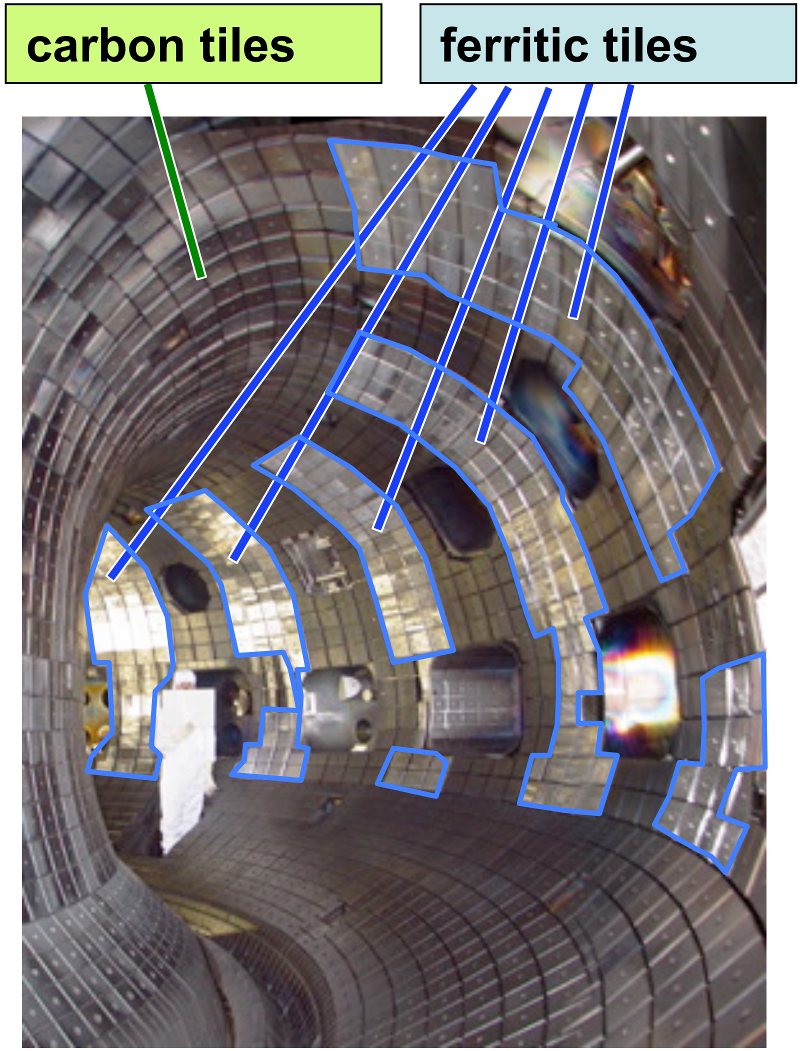
13 W-CFC tiles were installed at 1 toroidal section of outer divertor.
W-CFC tile: VPS coating on CFC: 50μm with Re multi-layer (3 layers)

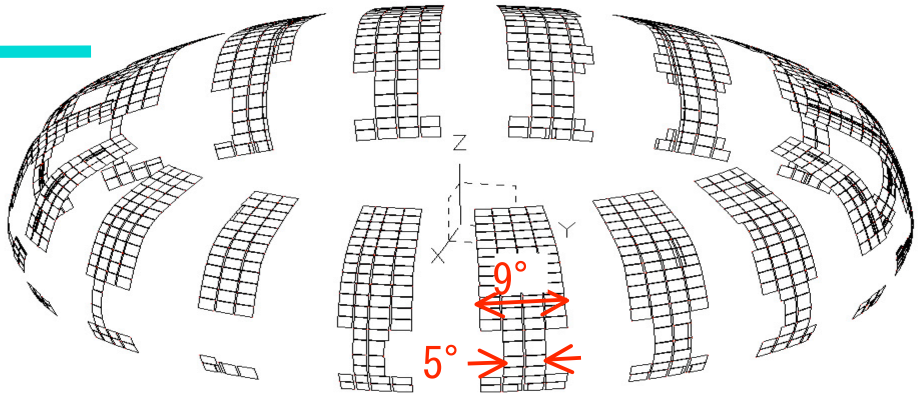
summarized by Ueda, et al. 17th PSI (2006).



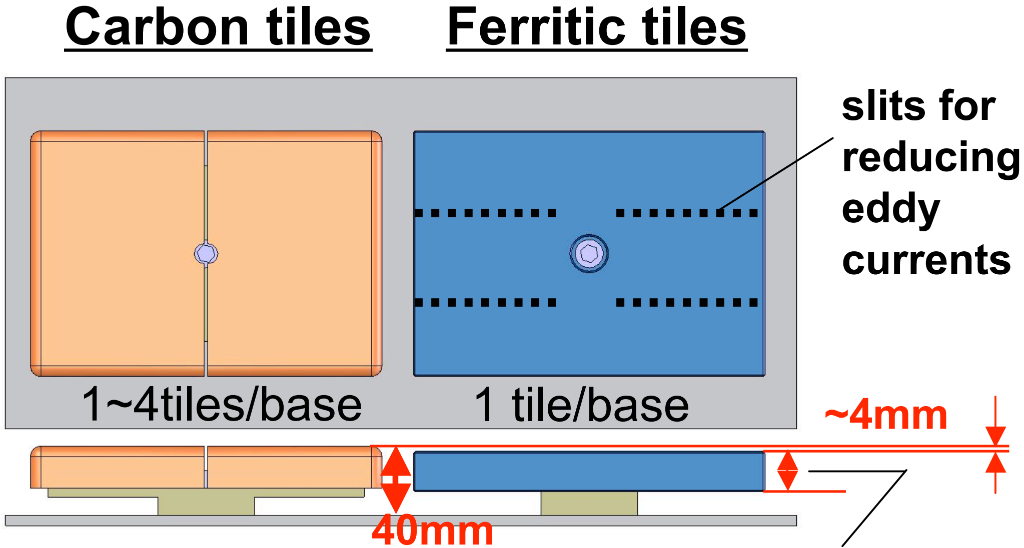


Ferritic tiles were installed to reduce toroidal field ripple and fast ion loss (2005-6)

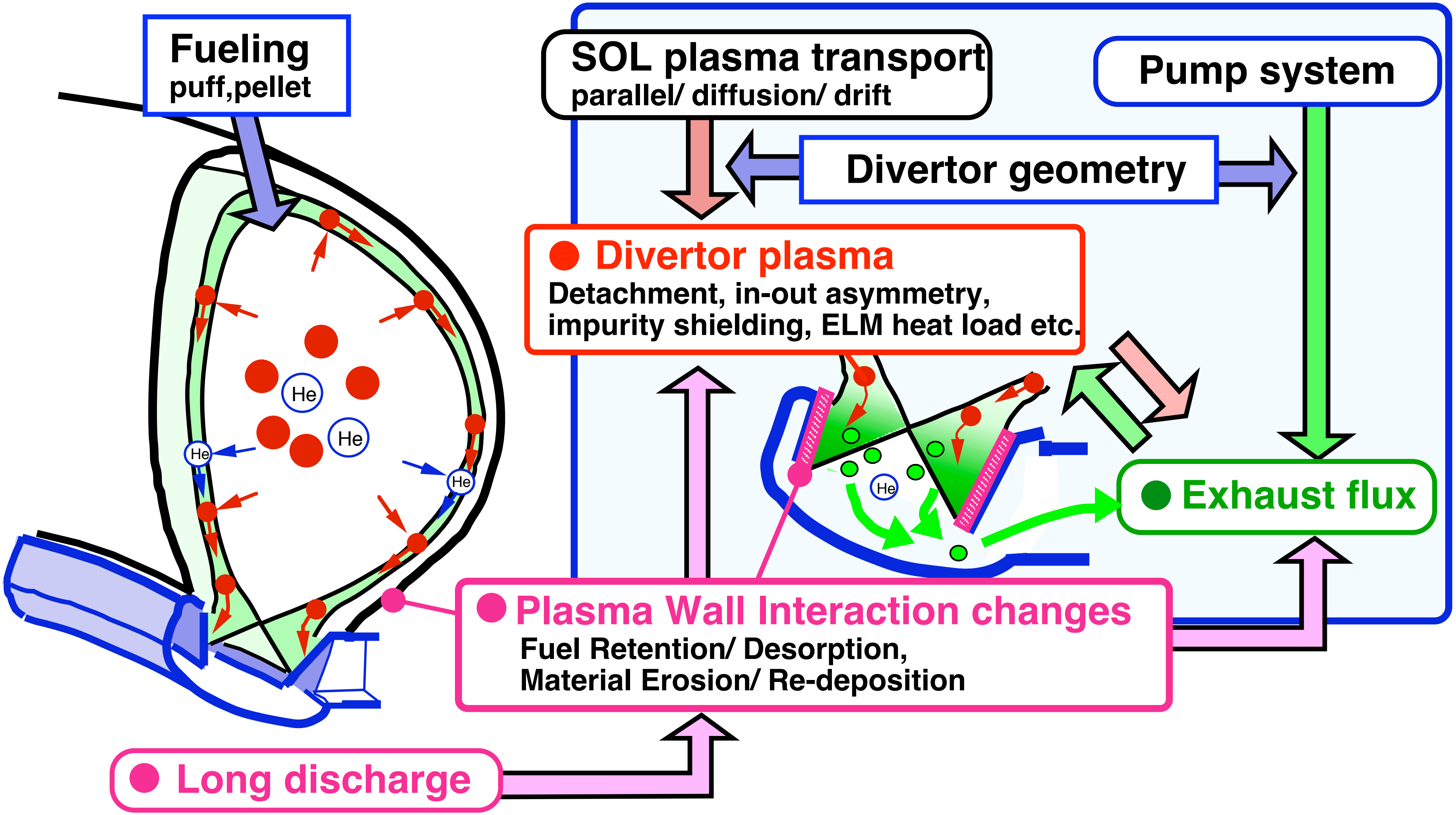




Material: 8%Cr 2%W steel
Magnetization: ~1.7 T at 573K



Heat and particle control in divertor for long discharge



Fueling
puff,pellet

SOL plasma transport
parallel/ diffusion/ drift

Pump system

Divertor geometry

● **Divertor plasma**
Detachment, in-out asymmetry,
impurity shielding, ELM heat load etc.

● **Exhaust flux**

● **Plasma Wall Interaction changes**
Fuel Retention/ Desorption,
Material Erosion/ Re-deposition

● **Long discharge**

Wall pumping/ Fuel retention ⇨ Saturation of wall/ divertor ⇨ Neutral/ Impurity shielding
Detachment/ MARFE control
Core confinement

In many tokamaks, boundary condition is determined by CFC/Graphite-PFC and Inertial/limited-active cooled PFC

	AUG	JET	JT60U	LHD	HT7	TRIAM	TS	ITER
Configuration	div	div	div	Island div	lim	lim	lim	Limiter / divertor
PFC material	C / W	C / Be	C	C	C	Mo	C	C / W / Be
Particle flux > 10^{22} D/m^2/s	yes	yes	yes	no (He)	no	no	yes	10^{23} D-T /m^2/s
1 < Te < 10 eV (partial detach)	yes	yes	yes	yes	no	yes	no	1-10 eV on targets
Pulse > 1 min	no	~1 min	~1 min	yes	yes	yes	yes	400s
Power flux 5~10 MW/m^2	yes	yes	yes	no	no	no	yes	10 MW/m^2 (+ transient)
Forced cooling	no	no	no	no	no	no	yes	yes

- **ITER : long pulse** will be maintained also in **limiter** (30 s)

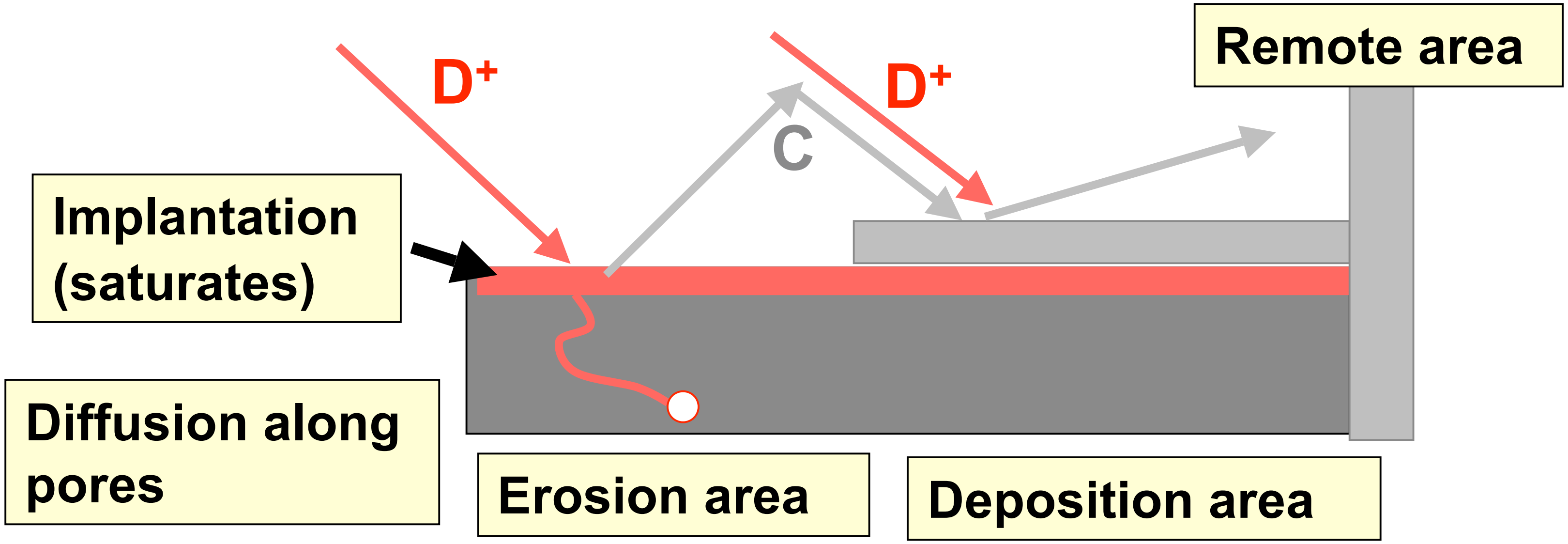
[G. Federici et al., JNM 313-316 (2003)]

- **No relevant material mix** : data shown here for **carbon**
- **No active cooling** (except TS) : data shown here for **evolving T^{surf}**

Tsitrone, et al. 2006 17th PSI, R-2

Fuel retention is closely related to PFC physics and chemical characteristics.

- Tritium is retained by co-deposition with Carbon, both on plasma facing sides and on remote areas
- Understanding of Tritium Co-deposition is understanding of where and how Carbon is eroded and how Carbon migrates globally and locally



Remote area

Implantation (saturates)

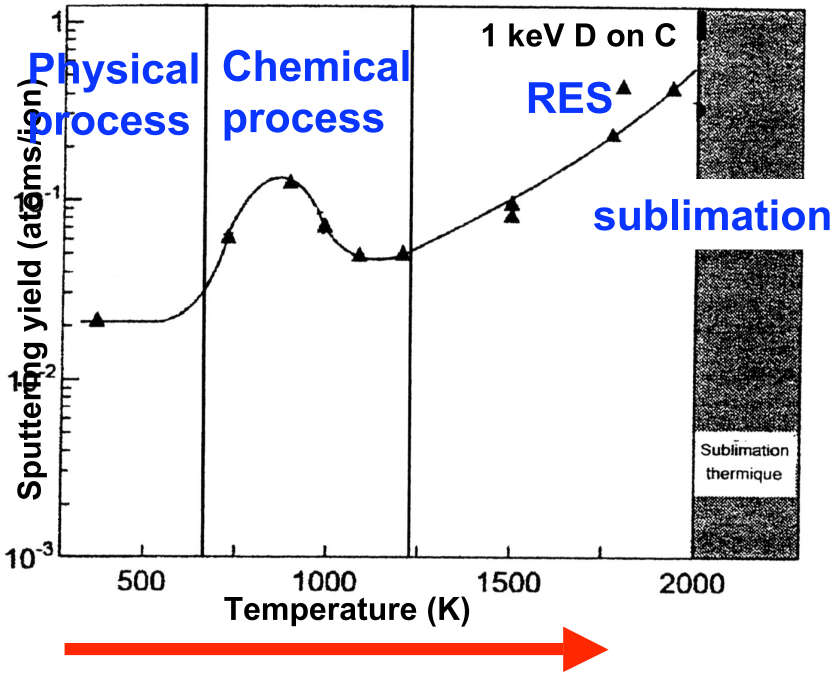
Diffusion along pores

Erosion area

Deposition area

Carbon characteristics change with temperature

Dominant erosion process of graphite (Physical/ Chemical/RES)

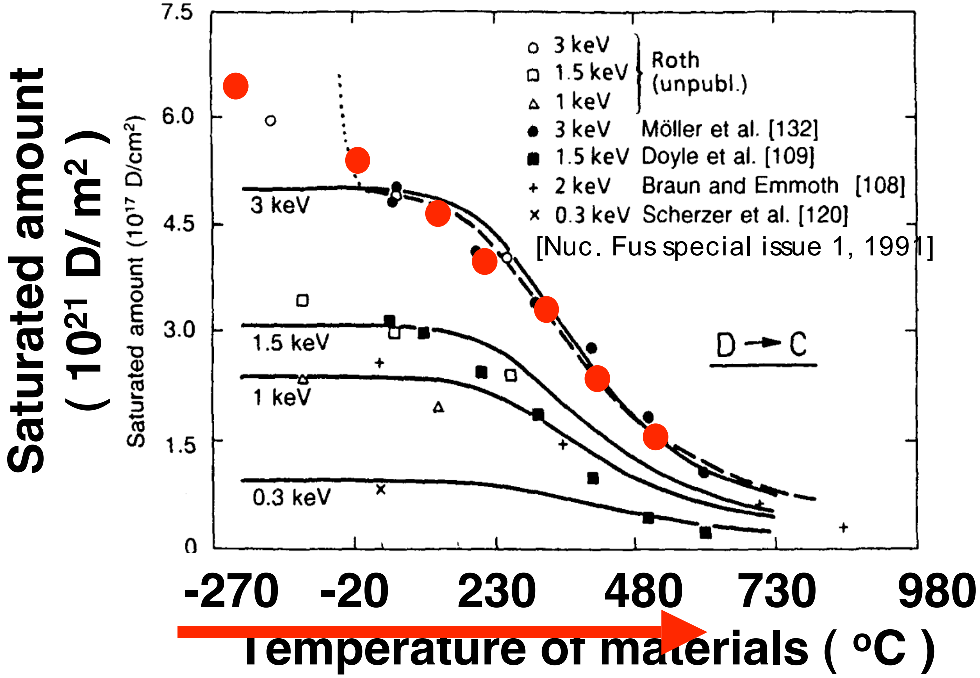


Increase in Carbon *erosion*

⇒ *Increase in Carbon deposition on lower temperature area,* and *Increase in H/D/T co-deposition*

D implantation to C materials

(W. Möller, J. Nucl. Mater. 1989)



Dueterium saturation decreases with increasing temperature.

⇒ Desorption of deuterium

2. Wall-pumping and particle control in long pulse discharges

Boundary plasma and PSI problems for ITER:

- Erosion by Type-I ELM will determine divertor life-time.
- Tritium retention in PFC (carbon) will determine operation period due to safety limit:

  Quantitative understanding of fuel retention mechanism has been progressed by particle balance experiments and PFC surface analysis.

This talk presents:

2-1) Particle balance and control using divertor pumping

2-2) Carbon erosion/ deposition and fuel retention

2-1) Particle balance and control using divertor pumping

Materials of the first wall

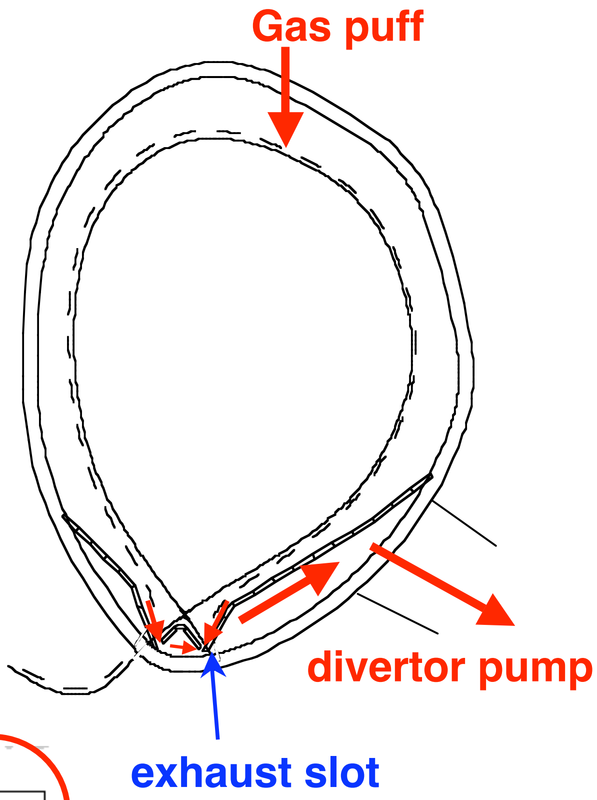
CFC: Divertor plates, top and outer dome

Graphite: main wall, baffle plates, inner dome

Ferritic steel: outboard main wall at TF coils (~10% of VV surface)

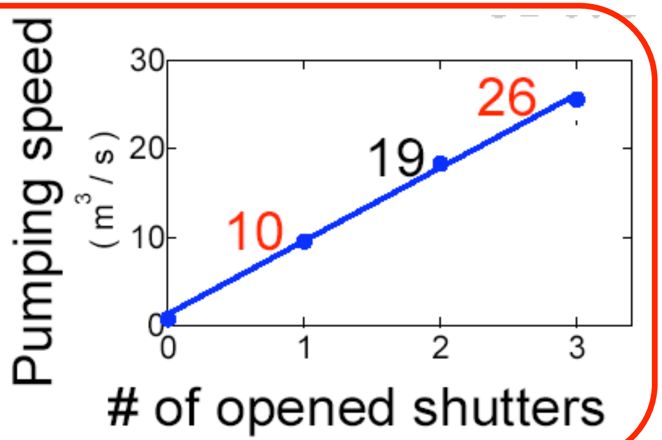
Baking temperature of the vacuum vessel:
150 or 300°C.

Pumping speed at exhaust slot :
0 - 26 m^3/s at 150°C,
0 - 47 m^3/s at 300°C.



$V_{plasma} \sim 74\ m^3$

Three pumps are available:
total pumping speed can be controlled during a discharge with opening/closing shutters.



"Globally saturated-wall condition" was observed after a few long-pulse discharges

Long-pulse (37s) high-density ELMy H-mode discharges have been repeated to investigate wall retention until wall saturation.

Wall pumping flux was evaluated from "Global particle balance":

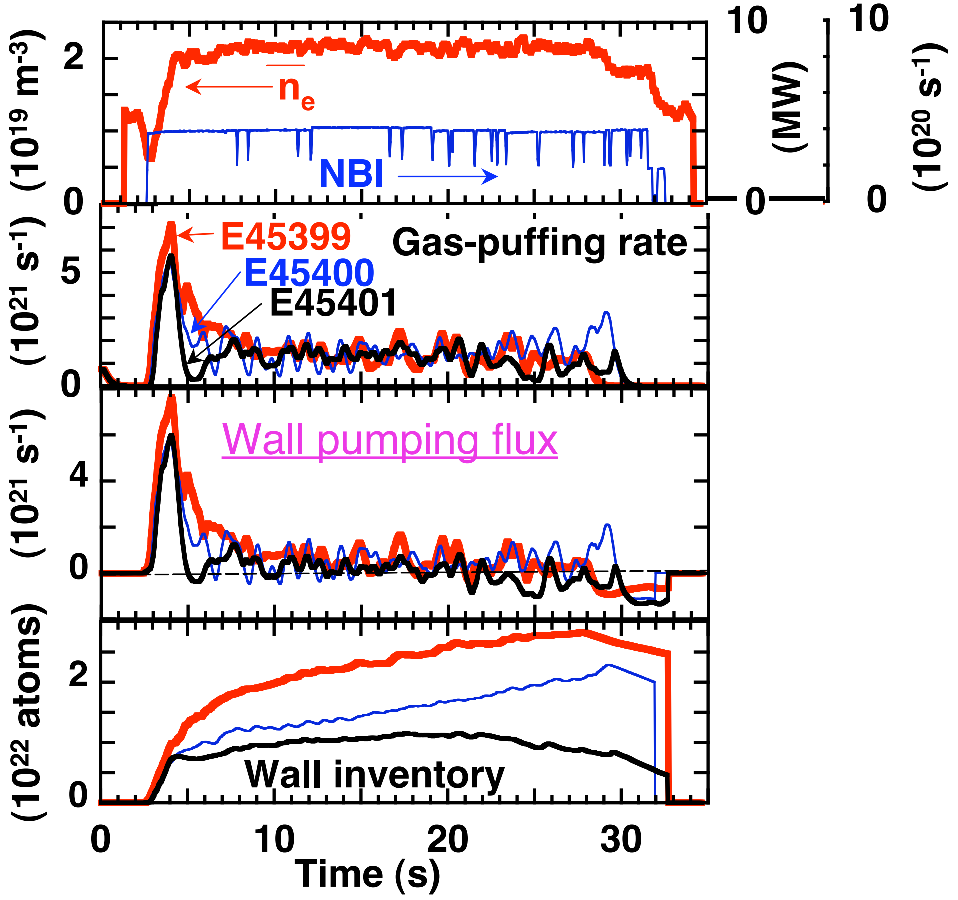
$$\frac{dN(t)}{dt} = \Gamma_{gas}(t) + \Gamma_{NBI}(t) - \Gamma_{div}(t) - \Gamma_{wall}(t)$$

n_e feedback controlled Γ_{gas} to maintain high n_e (> 0.5 n^{GW})

For early few discharges, wall-pumping rate was decreased during a discharge:

part of injected particles were accumulated at PFC surfaces.

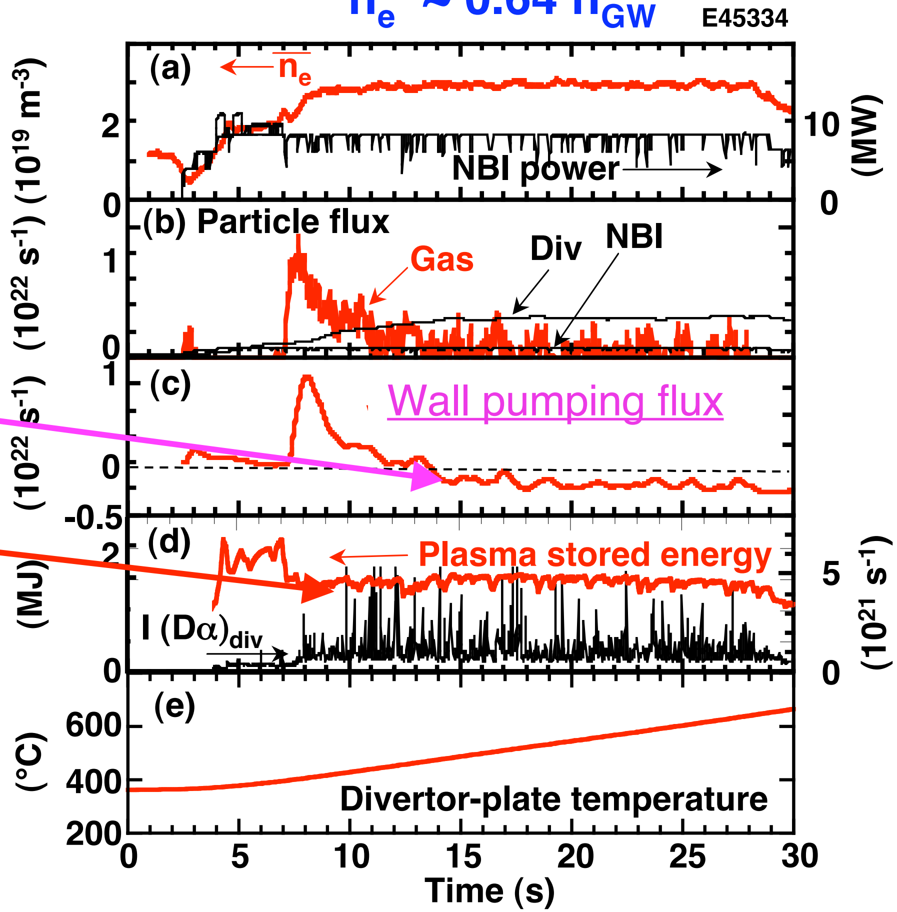
Nakano, et al. 2006 17th PSI, I-7
Kubo, et al. 2006 21st IAEA, EX-P4-11



I_p = 1.2 MA, B_t = 2.3 T, P_{NBI} = 4-8 MW for 30 s.

Under saturated wall condition, where outgas was observed, density was maintained by divertor pumping (~3x10^{21}D/s).

n_e ~ 0.64 n_{GW}



n_{GW}= 4.4x$10^{19}$$m^{-3}$

Wall-pumping flux was positive, then it decreased to be **negative** (**outgas: ~2x10^{21} D/s**) in *t* =14-32s.

Improved confinement with ELM activity was sustained.
(H_{89PL} ~ 1.3, $HH_{98(y,2)}$ ~ 0.71)

Maximum target temperature near the outer strike point increased from 380°C to 640°C (ΔT^{surf} ~260°C).

High-density (n_e/n^{GW}~0.85) plasma was maintained with divertor detachment by divertor pumping (~8x10^{21} D/s).

Wall-pumping flux was positive (3-10x10^{21}D/s) during discharge.

Increase in target temperature was small: 300°C-400°C (ΔT^{surf}~100°C), due to large radiation (X_p MARFE) and divertor detachment.

Evaluation of retention rate by co-deposition process: very small

Co-deposition rate ~0.26 x 10^{21} D/s

<< Wall-pumping flux ~ 3 x 10^{21} D/s

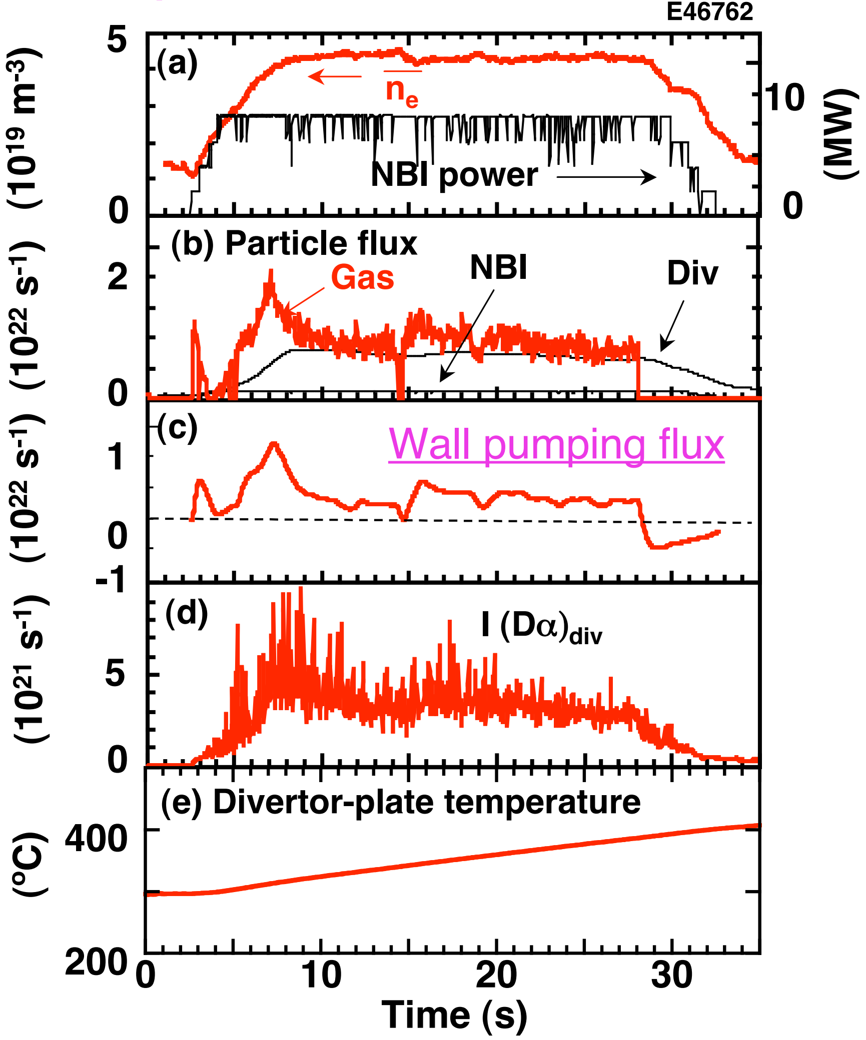
I (Dα) ~ 3 x 10^{21} phs/s

$R_{D\text{-ionization}} / R_{D\alpha\text{-emission}}$ ~ 18

Y_{carbon} ~ 0.1

D/C ~ 0.05 (measured in JT-60U)

n_e ~ 0.82 n_{GW}



Wall-pumping/outgas in "particle balance" was closely related to wall temperature, rather than injected flux.

$n_e \sim 0.65\ n_{GW}$ (#45953-45964) :

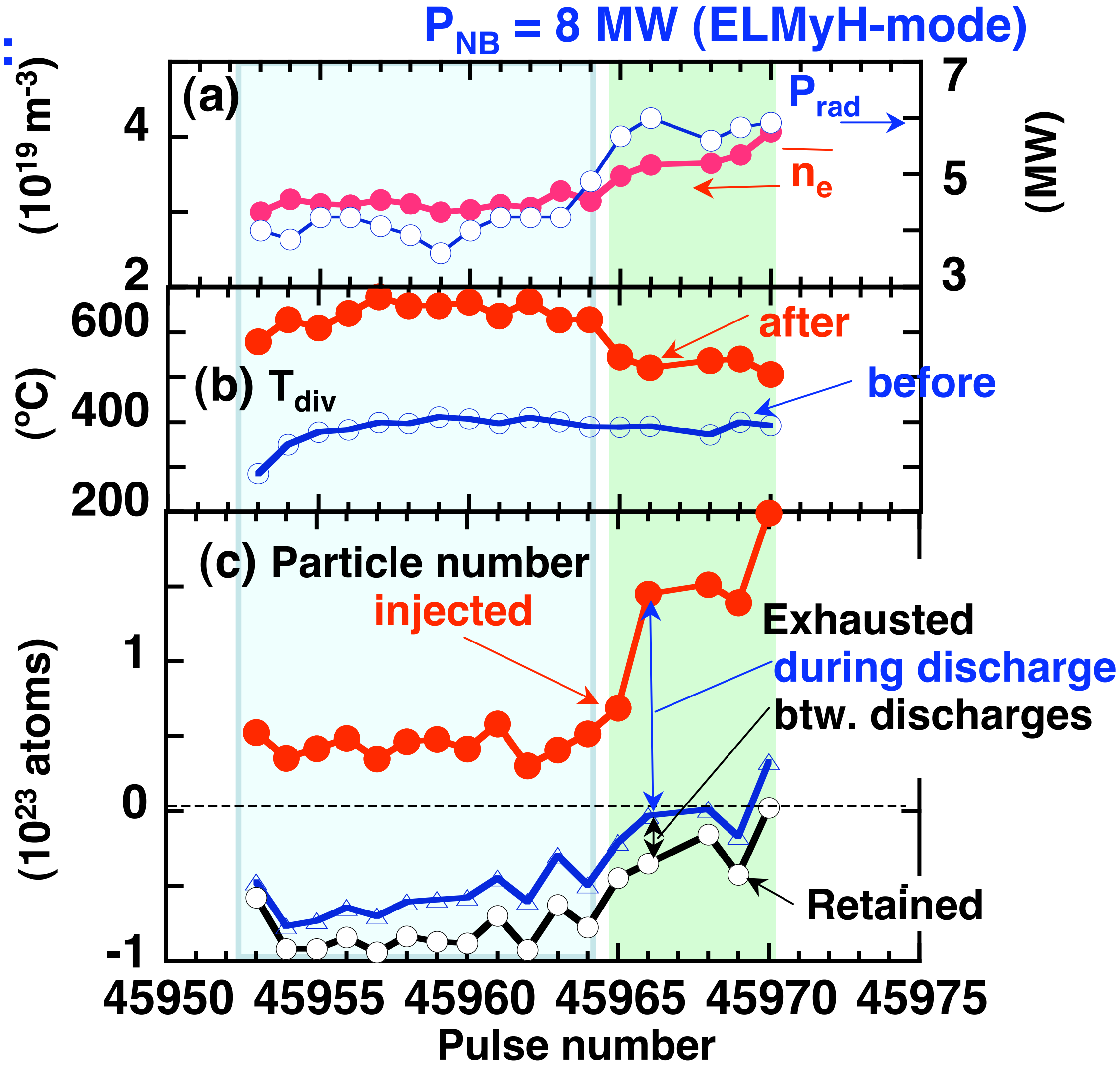
$\Delta T^{surf} \sim 250°C$

Particles were rather exhausted under "globally-saturated condition".

High density $n_e > 0.75\ n_{GW}$ (#45965-45970)

$\Delta T^{surf} < 130°C$

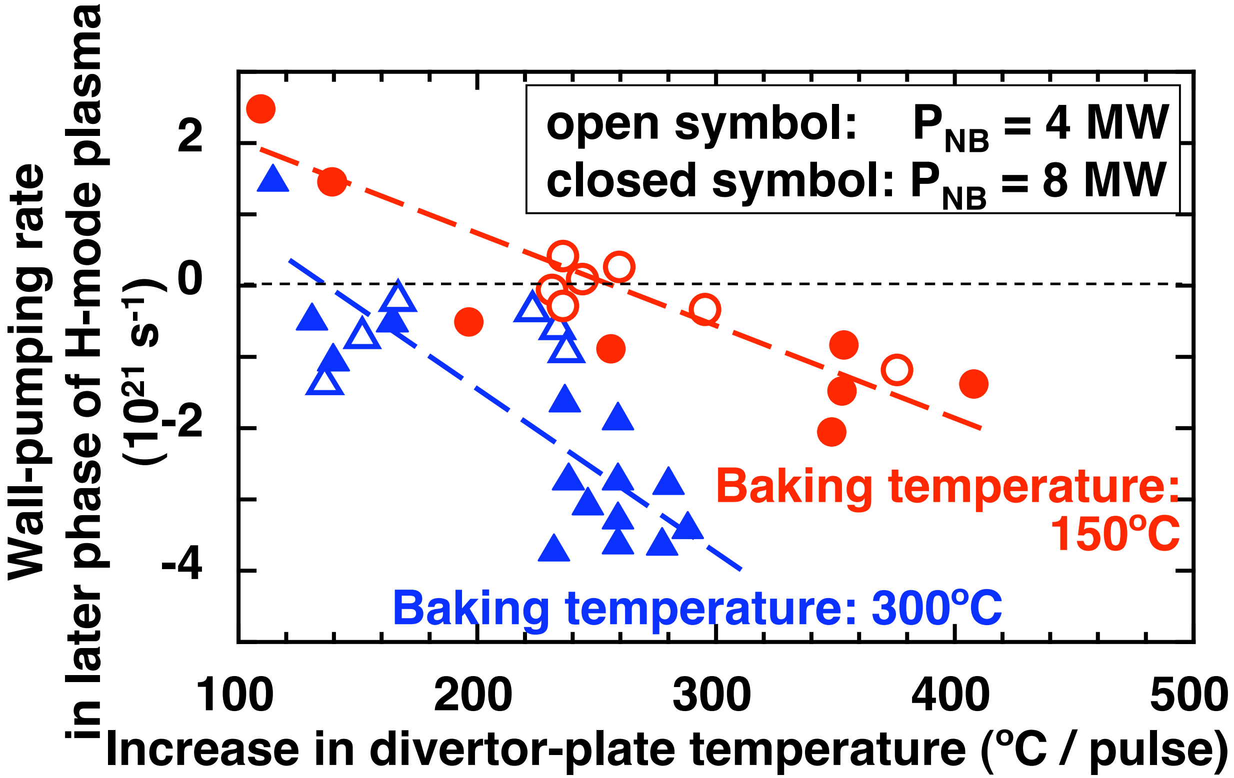
Small exhaust or retention was observed with handling large particle flux.



Outgas was increased with wall temperature

Similar characteristics of wall-pumping/ outgas depending on ΔT^{surf} were observed for baking temperatures of 150°C and 300°C.

Outgas was relatively smaller for low power P_{NB} (L-mode)



Note: Difference between the 150°C and 300°C cases is mostly attributed to difference in the outer strike point position.

Wall characteristics between two tokamaks are different for similar number of fuelling particles (10^{23}-10^{24}D):

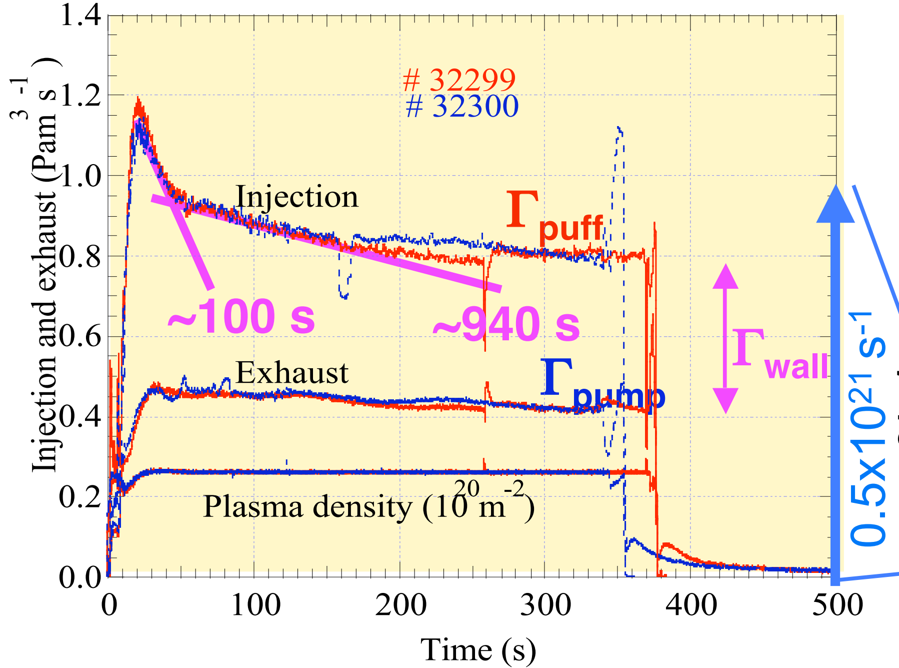
- **PFC temperature increased (JT-60U)/ mostly maintained(TS).**

Studies of C-deposition and bulk diffusion in PFC/pump geometry, base temperature, etc. have been in progress.

ToreSupra 6min. discharges: 1/2 of fuelling particles were accumulated in VV

JT-60U 37s discharges: saturated wall was observed after a few shots

($\Gamma_{wall} = \Gamma_{puff} - \Gamma_{pump} > 0$, $\Gamma_{wall}/\Gamma_{puff} \sim 0.5$)



378 s (6min 18s)

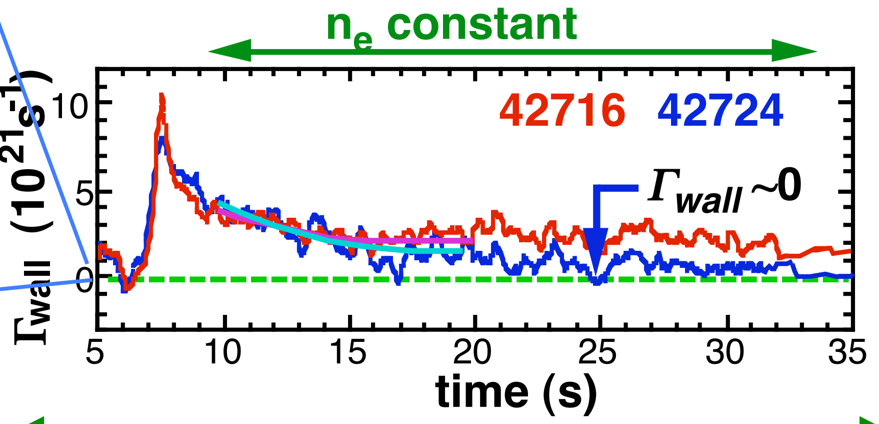
A. Grosman, et al. 16th PSI (2004)

($\Gamma_{wall} = \Gamma_{puff} - \Gamma_{pump} \le 0$)

Wall pumping flux (Γ_{wall}) decreased.

Desorption (outgas) was increased with divertor surface temperature

$$dN/dt = -\underline{[1-R(T_{surf})]\, N / \tau_p} + \Gamma_{inj} - \Gamma_{pump}$$

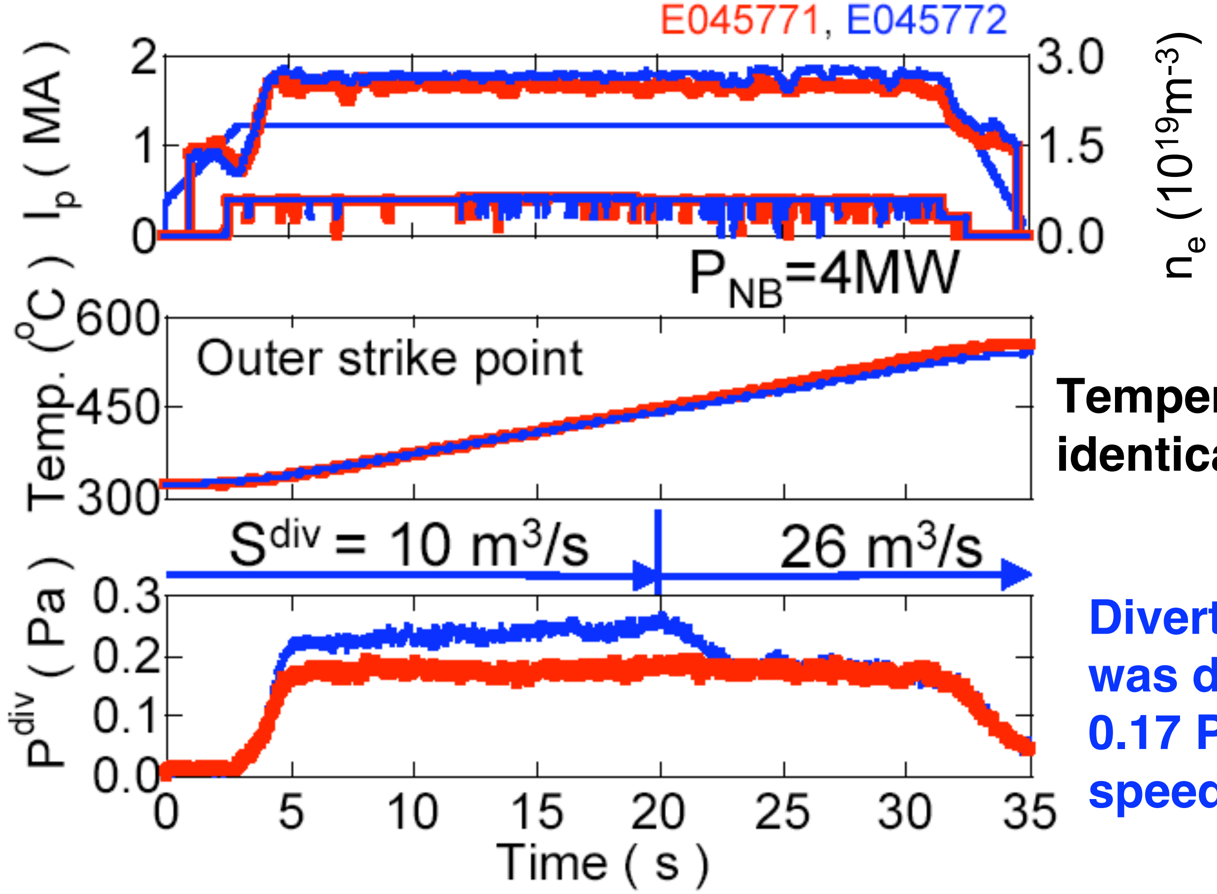


37 s

Response of wall-pumping during diveror pumping

Dynamic response of wall-pimping/ retention was investigated by changing pumping speed, in stead-state plasma (L-mode).

Shutters of two pumps opened at t = 20s (from 1 pump to 3 pumps): identical plasmas (n_e, P^{div}, recycling flux) were maintained.

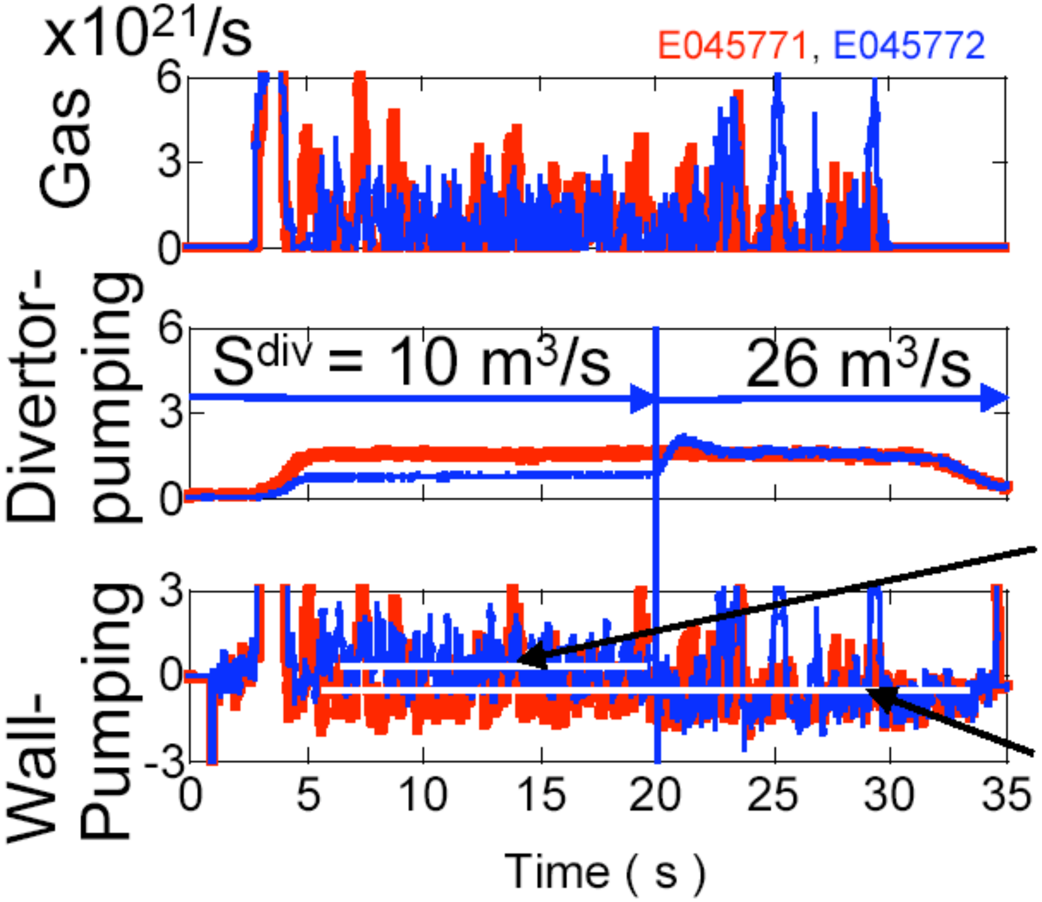


Temperature was identically increased.

Divertor neutral pressure was decreased from 0.25 to 0.17 Pa with pumping speed from 10 to 26m³/s.

Wall characteristics changed from *"pumping"* to *"outgas"* with increasing *divertor pumping flux*.

Γ_{wall} changed from 2×10^{20} (*wall pumping*) to -6×10^{20} D/s (*outgas*) in ~1s at t = 20s, just after Γ_{div} was increased from 0.8×10^{21} to 1.6×10^{21} D/s.



P_{NB} of 4MW
$\Rightarrow \Gamma_{NB} = 3.9\times10^{20}$/s

Γ_{div} increase: x 2

$\Gamma_{wall} = 2\times10^{20}$ /s
@S^{div}=10 m^3/s

$\Gamma_{wall} = -6\times10^{20}$ /s
@S^{div}=26 m^3/s

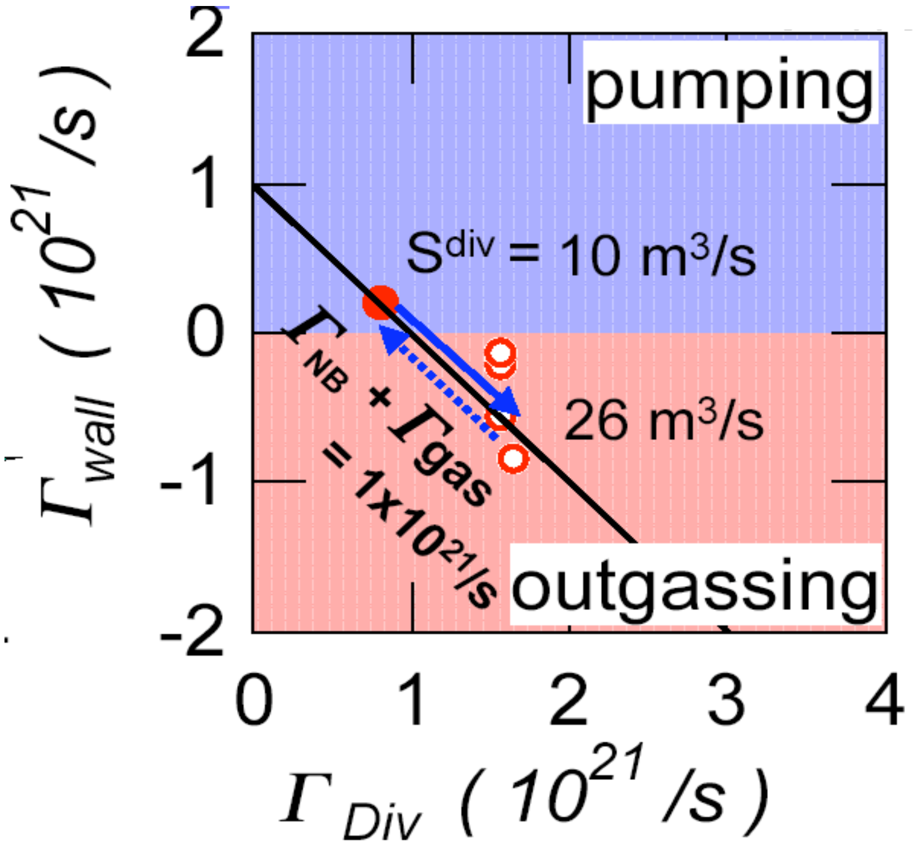
Similar characteristics of outgas/ wall-pumping from PFC with/ without divertor pimping were reported in DIII-D(without gas puffing) by R.Maingi, et al. Nucl. Fusion 36 (1996) 245

Retention mechanism/ condition (Γ_{wall}) changed with Γ_{div}.

Since both Γ_{NB} and Γ_{gas} are comparable before and after opening, wall-retention mechanism/condition (Γ_{wall}) may change with Γ_{div} or P^{div}.

Dynamic retention may decrease with decreasing plasma or neutral flux

suggesting large divertor pumping is for reduction of wall-retention?



$$\frac{d}{dt}N_p = \Gamma_{NB} + \Gamma_{gas} - \Gamma_{div} - \Gamma_{wall}$$

Const. N_p

$\Gamma_{NB}, \Gamma_{gas}$

$$\Gamma_{div} + \Gamma_{wall} = \Gamma_{NB} + \Gamma_{gas} = const.$$

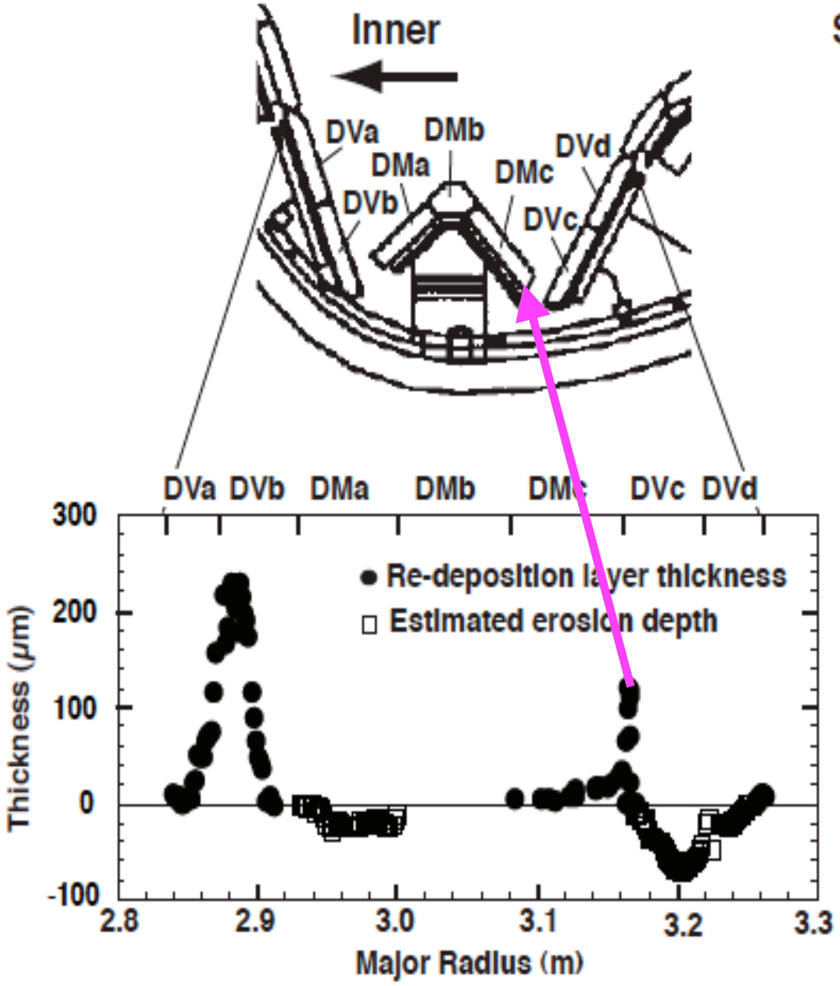
Dynamics of divertor/SOL plasma will be also investigated to model wall retention process.

2-2) Carbon erosion/ deposition and Fuel retention

Deposition layers (max. ~200μm) were seen at inner target, while erosion was dominant at outer divertor:

~40% of deposition may be originated from the first wall tile

Large local deposition was seen in shadow area at outer dome side



Eroded carbon goes to inner side.

Sample tiles : 1997~2002 **(W-shaped divertor)**

Integrated neutral beam injection time: 3 x 10^4s

Outer dome wing : 2 x 10^4s **(1999-2002)**

Inner divertor : 0.36kg (6 x 10^{20} C/s)

Inner dome wing : -0.09kg (-1.5 x 10^{20} C/s)

Outer dome wing : 0.18kg (4.5 x 10^{20} C/s)

Outer divertor : -0.25kg (-4.2 x 10^{20} C/s)

dust : 0.007kg (0.1 x 10^{20} C/s)

↓

density (deposition): 910kg/m^3

density (erosion): 1700kg/m^3

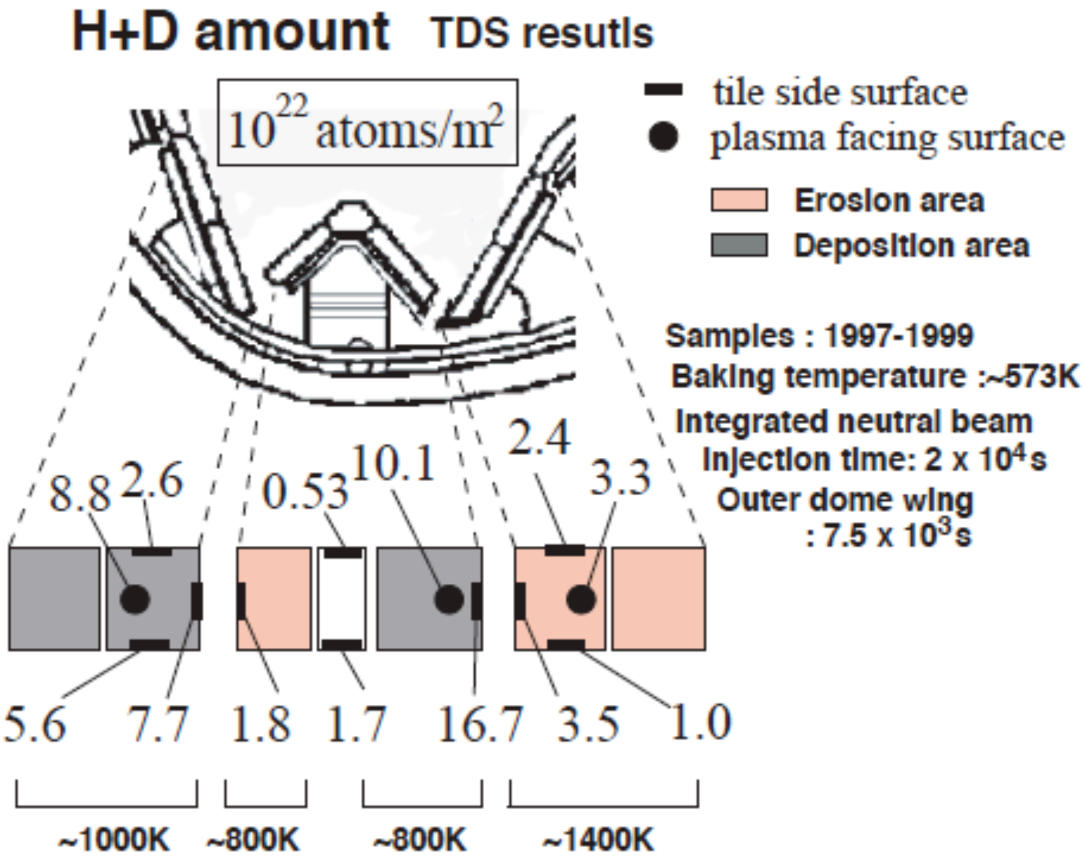
Total deposition : 0.55kg
(10.7 x 10^{20} C/s)

Total erosion : -0.34kg
(-5.7 x 10^{20} C/s)

0.21kg (5 x 10^{20} C/s: 40% of deposition of the divertor area) must be originated from the first wall (main chamber)

Large D+H retention was observed at outer dome (dep. layer): D and H tends to be remained at lower T^{surf} or ΔT^{surf} area.

Smaller D/H represents H-replacement just before ventilation: Large D retention at outer dome: *low temperature* and *small H plasma flux.*

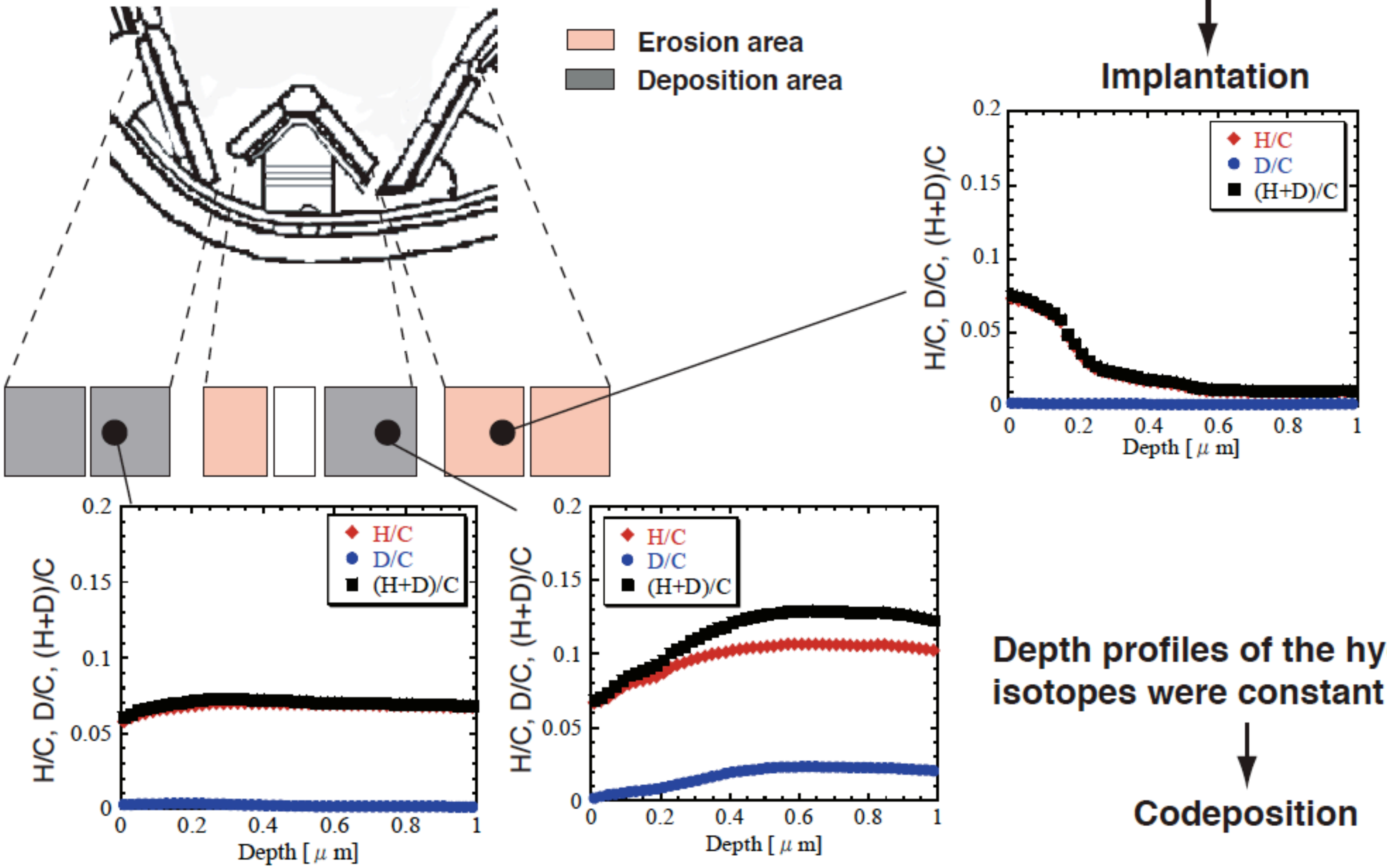


Max. surface temperature during NBI

Co-deposition was observed over C-deposition layers

Implantation was limited near eroded target surface (a few μm)
Co-deposition produced flat depth H+D/C profile on inner target & dome

H+D amount (ERD) **Samples : 1997~1998**



Hydrogen remained near surface region.

↓

Implantation

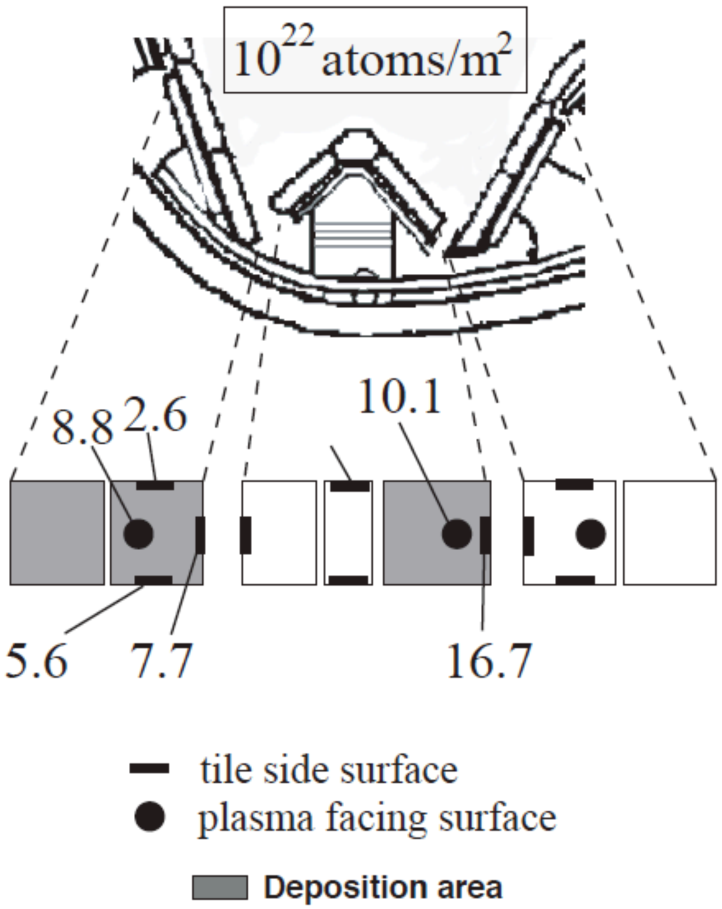
Depth profiles of the hydrogen isotopes were constant in depth.

↓

Codeposition

Averaged retention rate was evaluated in co-dep. layers:
estimated retention of ~2x10^{21} in 30s discharge are small

H+D amount Samples : 1997-1999



Inner divertor

Integrated neutral beam injection time: 2 x 10^4 s

Inner divertor surface : 8.8 x 10^{22} atoms/m^2

Area : 4.4m^2

H+D retention rate 2 x 10^{19} atoms/s

H+D retention rate : 1.2 x 10^{19} atoms/s

*Estimated from carbon deposition rate and (H+D)/C 6 x 10^{20} C/s, 0.02

Outer dome wing

Integrated neutral beam injection time: 7.5 x 10^3 s

Outer dome wing surface : 10 x 10^{22} atoms/m^2

Area : 2.6m^2

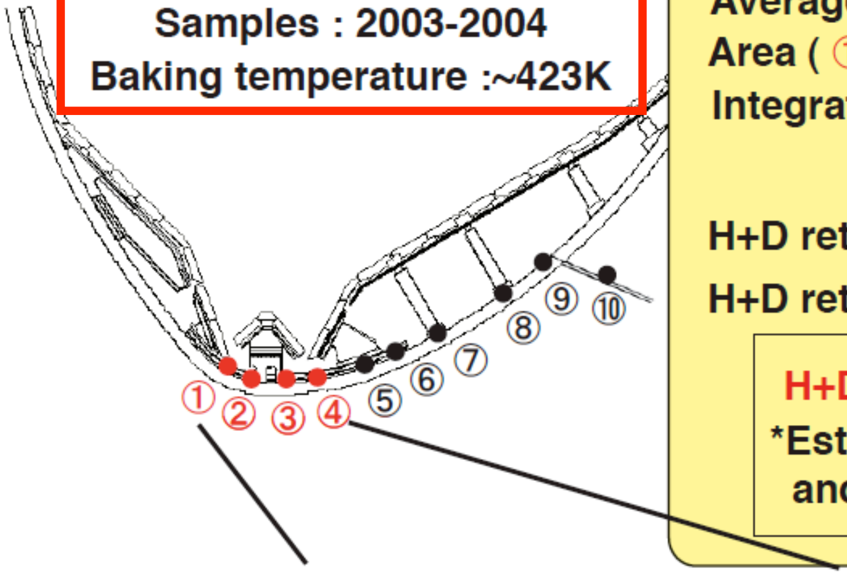
H+D retention rate 3.5 x 10^{19} atoms/s

H+D retention rate : 5.9 x 10^{19} atoms/s

*Estimated from carbon deposition rate and (H+D)/C 4.5 x 10^{20} C/s, 0.13

Large D/C was found under dome at base-T (150°C) region:

averaged retention rate (3-6x10^{19} D/s) was comparable to that on PFC.



Samples : 2003-2004
Baking temperature :~423K

Average H+D concentration : 8.56 x 10^{22} atoms/m^2
Area (① ② ③ ④) : 3.8 m^2
Integrated neutral beam injection time: 8 x 10^3 s

↓

H+D retention : 3 x 10^{23} atoms
H+D retention rate : 4 x 10^{19} atoms/s (Co-deposition)

H+D retention rate : 3~6 x 10^{19} atoms/s
*Estimated from carbon deposition rate and (H+D)/C 4~8 x 10^{19} C/s, 0.75

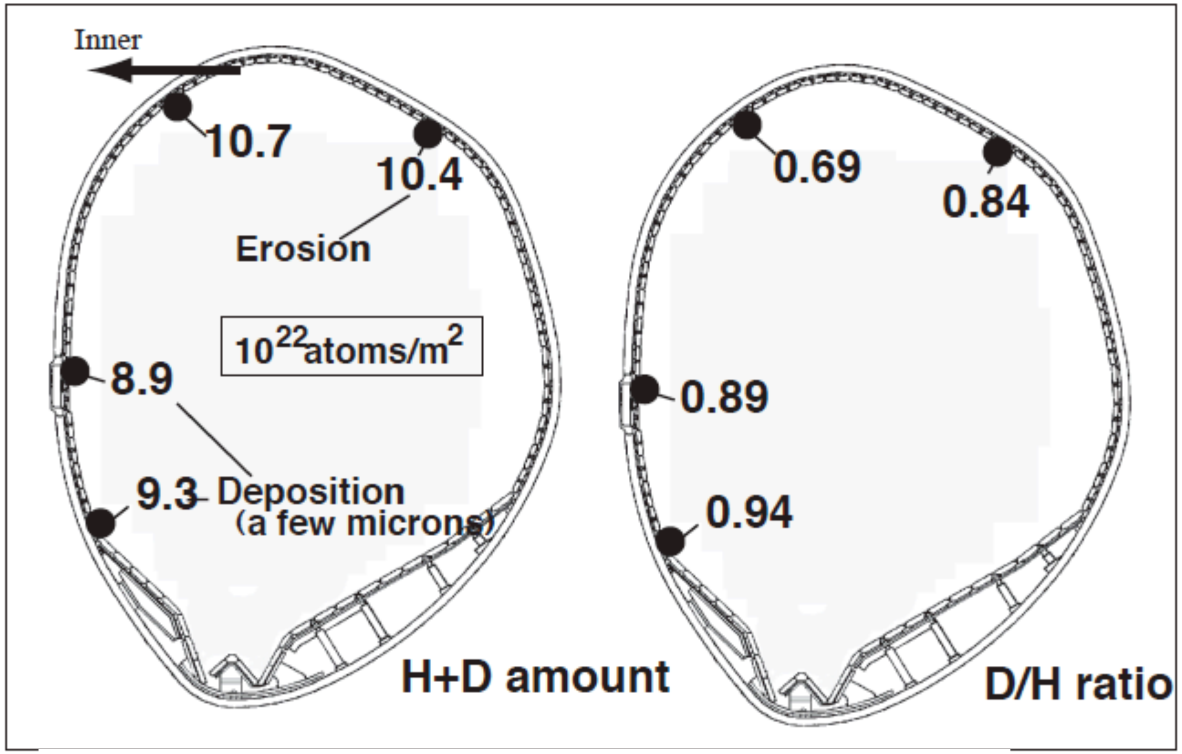
	①	②	③	④	⑤	⑥	⑦	⑧	⑨	⑩
H+D (x 10^{22} atoms /m^2)	3.22	14.1	12.6	4.32	Back.	—	Back.	—	—	Back.
D/H ratio	2.80	3.60	1.81	2.62						
Dep. Thickness (μm)	—	2.33	1.67	—						
C density (g/cm^3)		1.88	1.78							
(H+D)/C		0.64	0.85							

DD discharges: NB power ~9MW
NBI time : 7.4 x 10^3 s
HH discharges: NB power ~4MW
NBI time : 9.5 x 10^2 s

D+H was measured at first wall surface (base T region)

C: Erosion/ Deposition was often observed at Outer/ Inner wall surfaces
H+D: Retention was comparable to that at divertor.

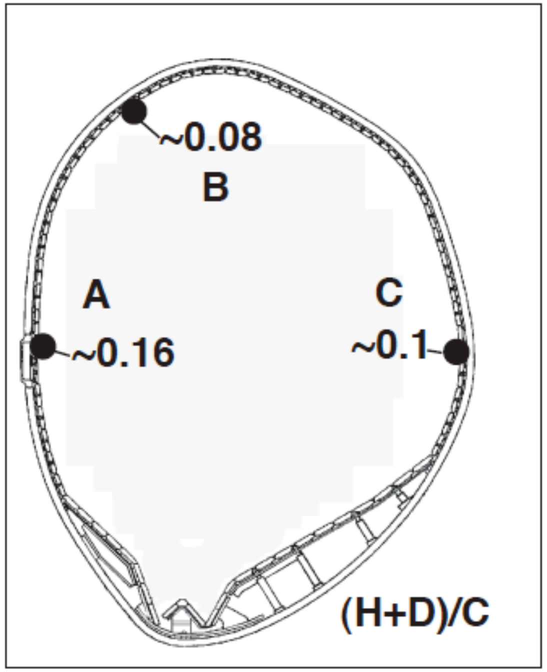
TDS results



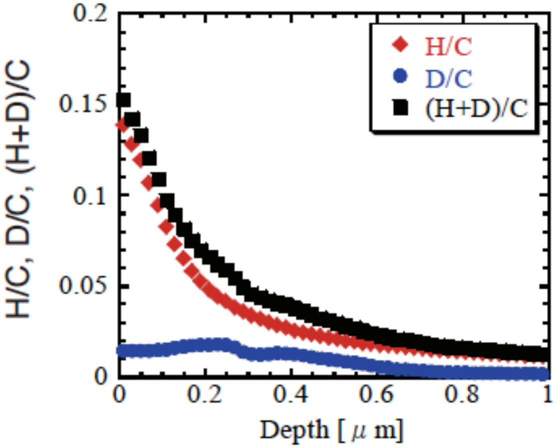
Frist wall area : 177m^2
(H+D) amount : ~10 x 10^{22}atoms/m^2
~1.8 x 10^{25} atoms

(H+D)/C profiles near wall surface showed implantation/ co-deposition: ratio was comparable to co-deposition at divertor.

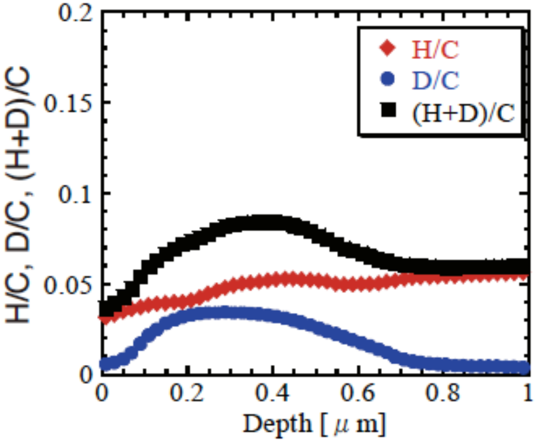
ERD results



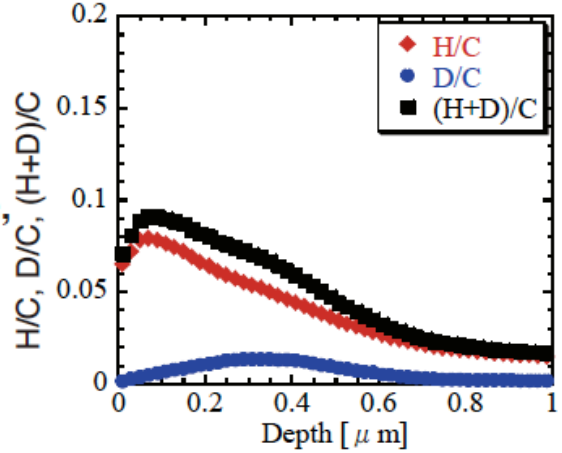
A



B



C



A and C

H remained in the near surface, and decreased with depth.

→ **Implantation**

B

H remained in the near surface, and increased with depth

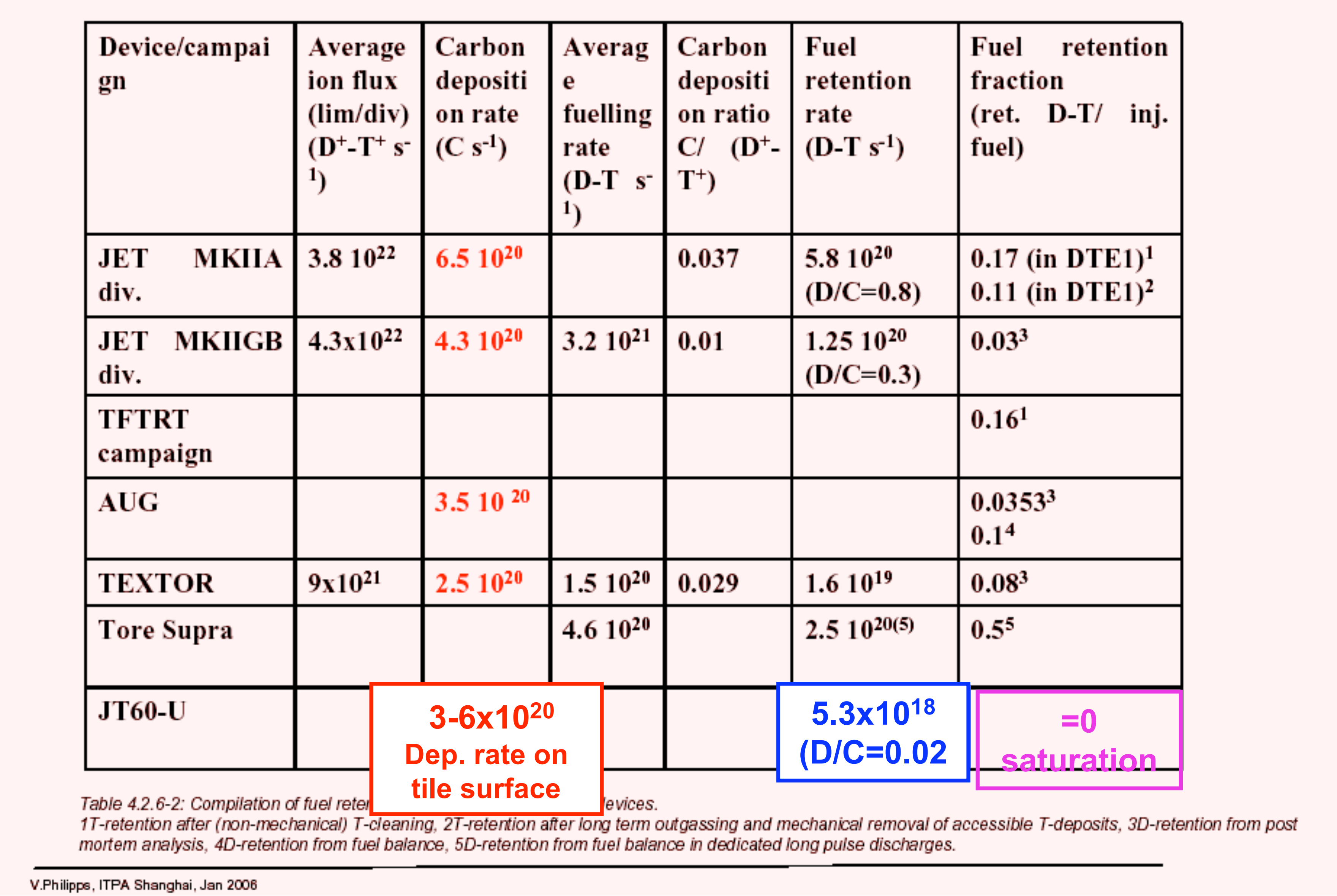
→ **Codeposition?**

D still remained deep inside influenced by boron layer or high energy ion implantation.

C-deposition rate in JT-60U was comparable

⇔ (H+D)/C was 1-2 orders of magnitude Lower:

Higher base temperature（300C until 2002, now 150C) ?



Device/campaign	Average ion flux (lim/div) (D^+-T^+ s^{-1})	Carbon deposition rate (C s^{-1})	Average fuelling rate (D-T s^{-1})	Carbon deposition ratio C/ (D^+-T^+)	Fuel retention rate (D-T s^{-1})	Fuel retention fraction (ret. D-T/ inj. fuel)
JET MKIIA div.	3.8 10^{22}	6.5 10^{20}		0.037	5.8 10^{20} (D/C=0.8)	0.17 (in DTE1)[1] 0.11 (in DTE1)[2]
JET MKIIGB div.	4.3x10^{22}	4.3 10^{20}	3.2 10^{21}	0.01	1.25 10^{20} (D/C=0.3)	0.03[3]
TFTRT campaign						0.16[1]
AUG		3.5 10 ^{20}				0.0353[3] 0.1[4]
TEXTOR	9x10^{21}	2.5 10^{20}	1.5 10^{20}	0.029	1.6 10^{19}	0.08[3]
Tore Supra			4.6 10^{20}		2.5 10^{20}(5)	0.5[5]
JT60-U		3-6x10^{20} Dep. rate on tile surface			5.3x10^{18} (D/C=0.02	=0 saturation

Table 4.2.6-2: Compilation of fuel reten evices.

1T-retention after (non-mechanical) T-cleaning, 2T-retention after long term outgassing and mechanical removal of accessible T-deposits, 3D-retention from post mortem analysis, 4D-retention from fuel balance, 5D-retention from fuel balance in dedicated long pulse discharges.

V.Philipps, ITPA Shanghai, Jan 2006

4. Summary

Particle control using divertor pumping and *Characteristics of Carbon PWI* has been investigated in long pulse discharges on JT-60U:

- Under "globally saturated wall" (outgasing) condition, plasma density (n_e/n^{GW}~0.65) was maintained by divertor pumping (~$3x10^{21}$ D/s).
- High-density (n_e/n^{GW}~0.85) plasma with detached divertor was maintained by divertor pumping (~$8x10^{21}$ D/s).
- Characteristics of wall-pumping/ outgas in "particle balance" (Γ_{wall}) was closely related to wall temperature and/or its change.
  At the same time, influences of C-deposition in PFC/pump geometry, base temperature, etc. have been investigated.
- Dynamic retention: Γ_{wall} can be change with divertor plasma flux (Γ_{div}) and/or neutral pressure (P^{div}).

Analysis of Carbon PFC surface:

- D+H co-deposition locations were inner diviverotr and outer dome edge: estimated retention of ~$2x10^{21}$ D in 30s discharge are small.
- Large D/C location was determined under dome in recent operation (150C).
- Erosion/deposition areas were seen in first wall: relatively large D/C.