Tritium in Chamber Materials, Trapping and Release
(plasma chamber in DEMO, irradiation damage and tritium trapping)

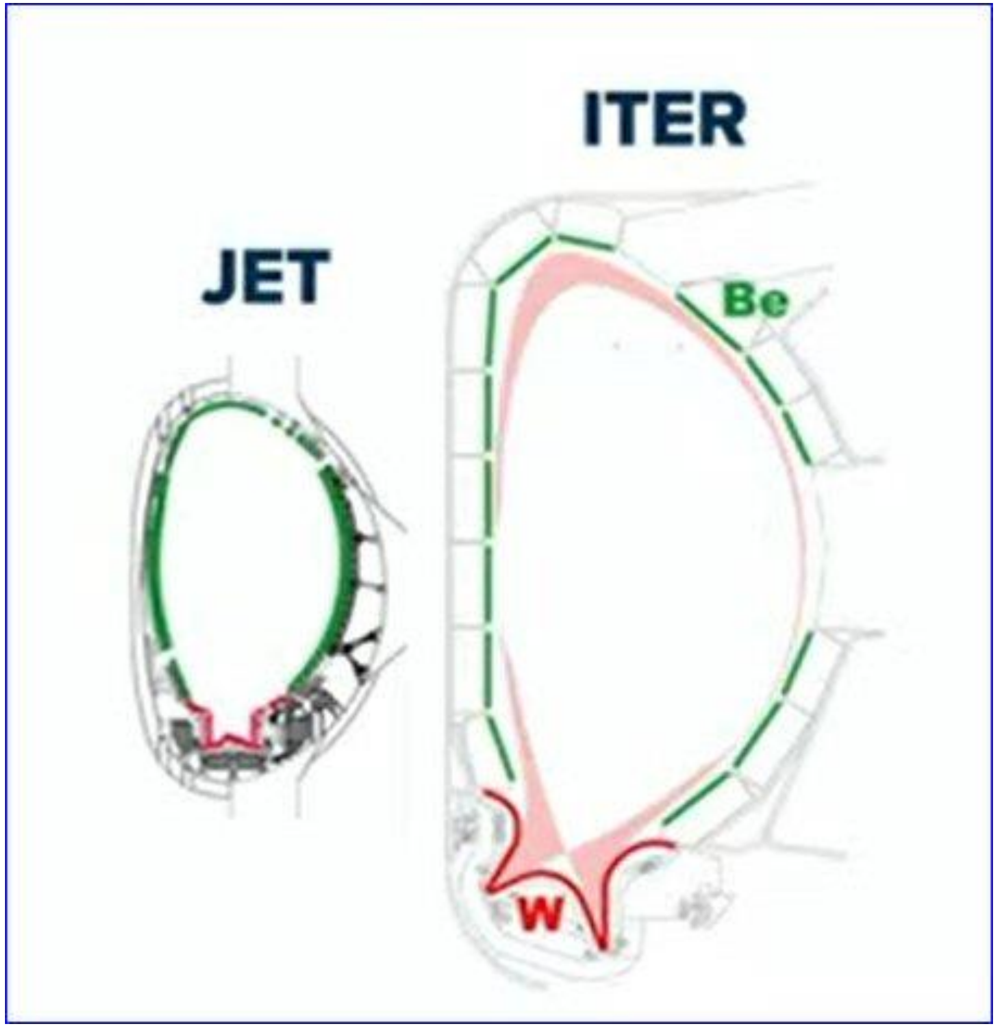
Part A: D and T retention in JET-ITER Like Wall (Japan-EU Broader Approach Collaboration)

Part B: Neutron irradiation effects on fuel retention in W

Reviews of recent publications

Yuji Hatano (University of Toyama, Japan)

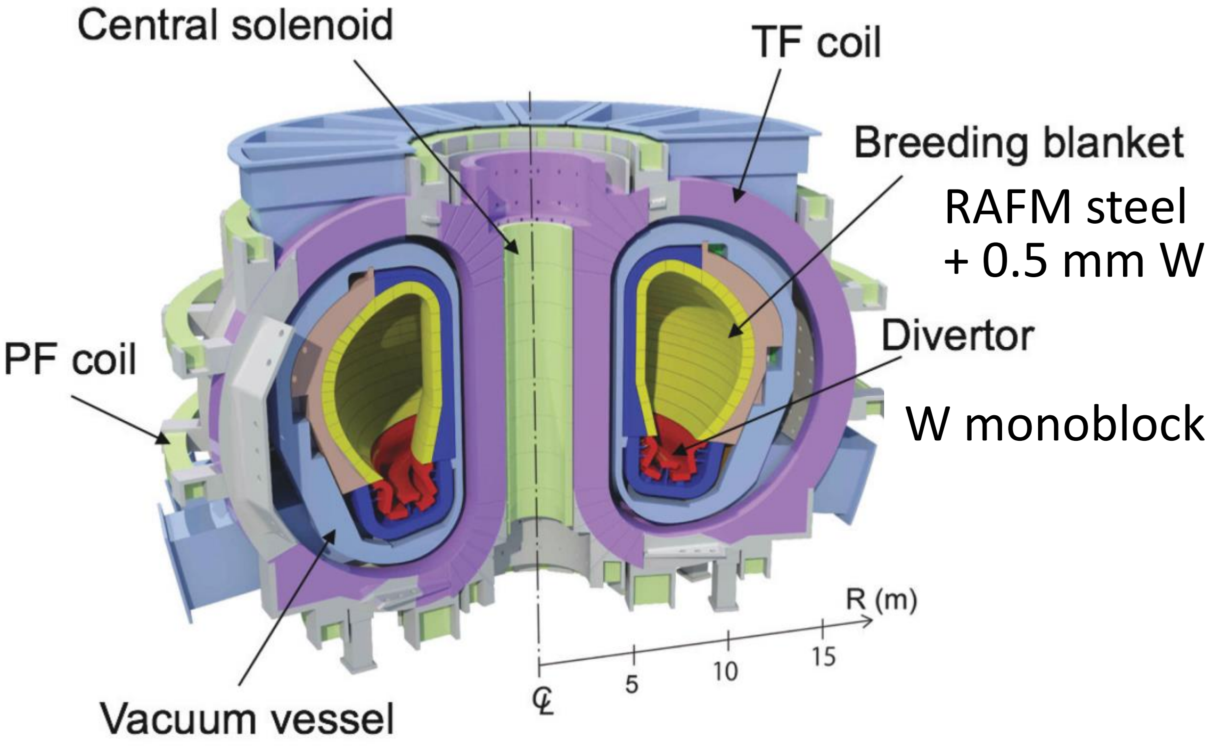
Neutron irradiation was performed in High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) under Japan-US Collaboration and Belgium Reactor 2 (BR2) at SCK•CEN under SCK•CEN-Tohoku University Collaboration program. We thank to colleagues in ORNL and SCK•CEN for cooperation.



Pulse operation
Co-deposition with Be and other impurities

https://news.newenergytimes.net/2022/02/09/u-k-jet-fusion-reactor-after-25-years-produces-more-five-second-reactions/

DEMO

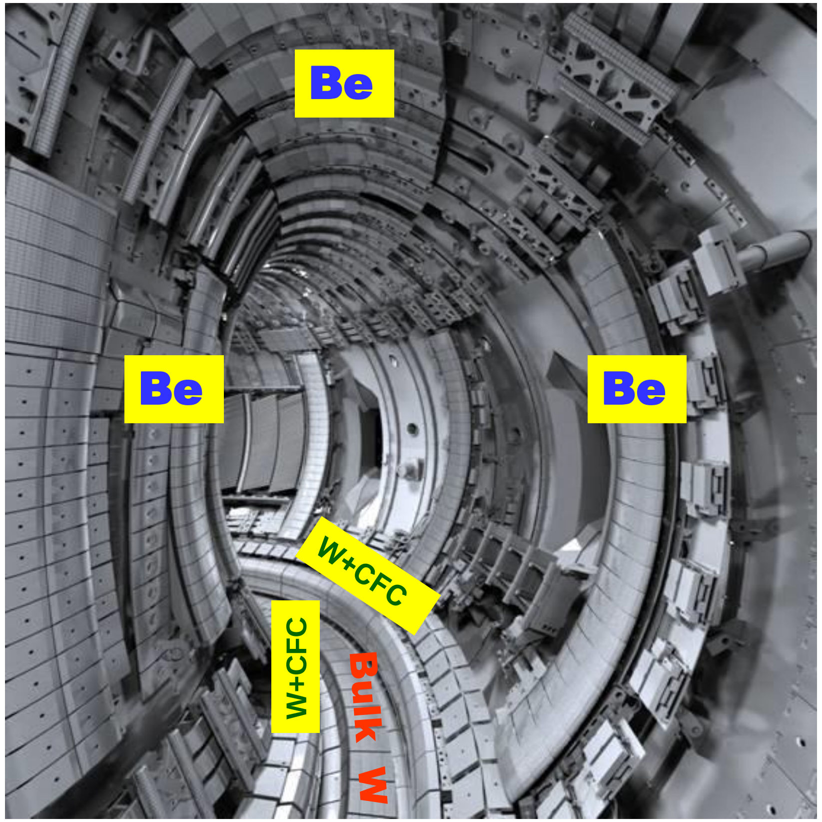


Japanese DEMO

K. Tobita et al., J. Phys. Conf. Ser. 1293(2019)012078

(Quasi-) Steady-state operation
Co-deposition
+ Penetration into W (and RAFM steel) by diffusion

Part A: D and T retention in JET-ITER Like Wall



JET vacuum vessel with ITER like wall

JET performed three DD campaigns with ILW. Tiles were extracted and analyzed after each campaign.

	Plasma time (h)	Limiter plasma (h)	Divertor plasma (h)	Input energy (GJ)
ILW1 / 2011-12	20.38	7.76	12.62	145
ILW2 / 2013-14	19.8	6.04	13.78	201
ILW3 / 2015-16	23.33	4.86	18.47	245

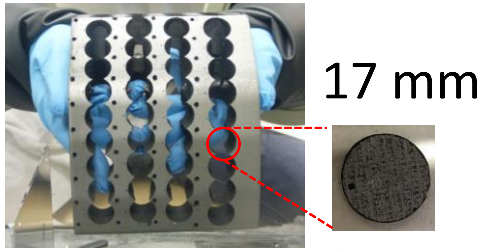
ILW2 ended with H-fuelled campaign (300 pulses) and ILW3 was finished with N_2 seeded H mode for 15 min.

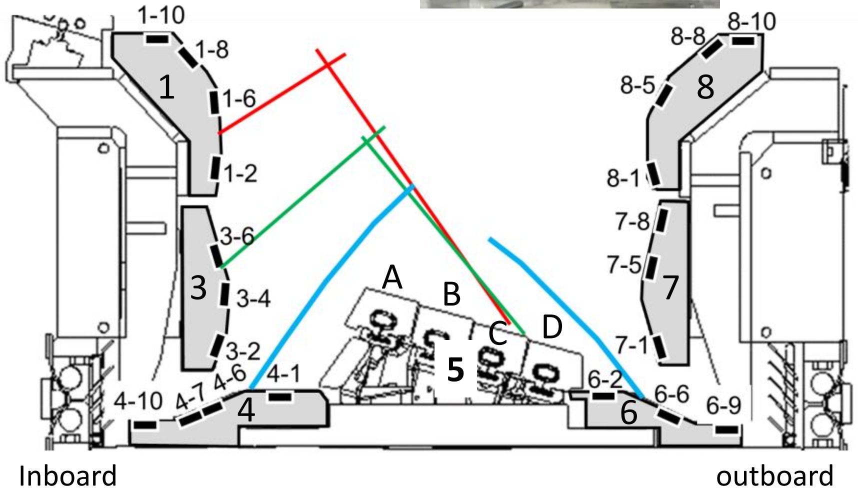
$$
\begin{aligned}
{}^{2}_{1}\mathrm{D} + {}^{2}_{1}\mathrm{D} &\rightarrow {}^{3}_{1}\mathbf{T}(\mathbf{1.01\ MeV}) + {}^{1}_{1}\mathbf{H}(\mathbf{3.03\ MeV}) \quad \mathbf{50\%} \\
&\rightarrow {}^{3}_{2}\mathbf{He}(\mathbf{0.82\ MeV}) + {}^{1}_{0}\mathbf{n}(\mathbf{2.45\ MeV}) \quad \mathbf{50\%}
\end{aligned}
$$

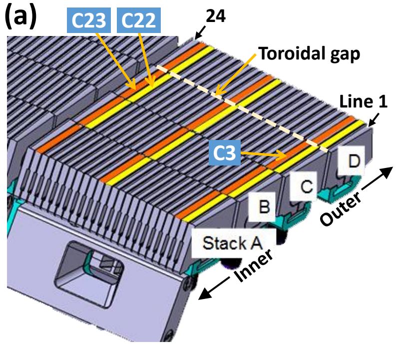
DT experiment (DTE2) was performed recently but tiles has not been extracted.

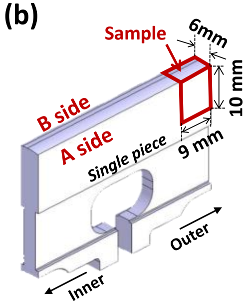
Sample preparation from JET ILW tiles

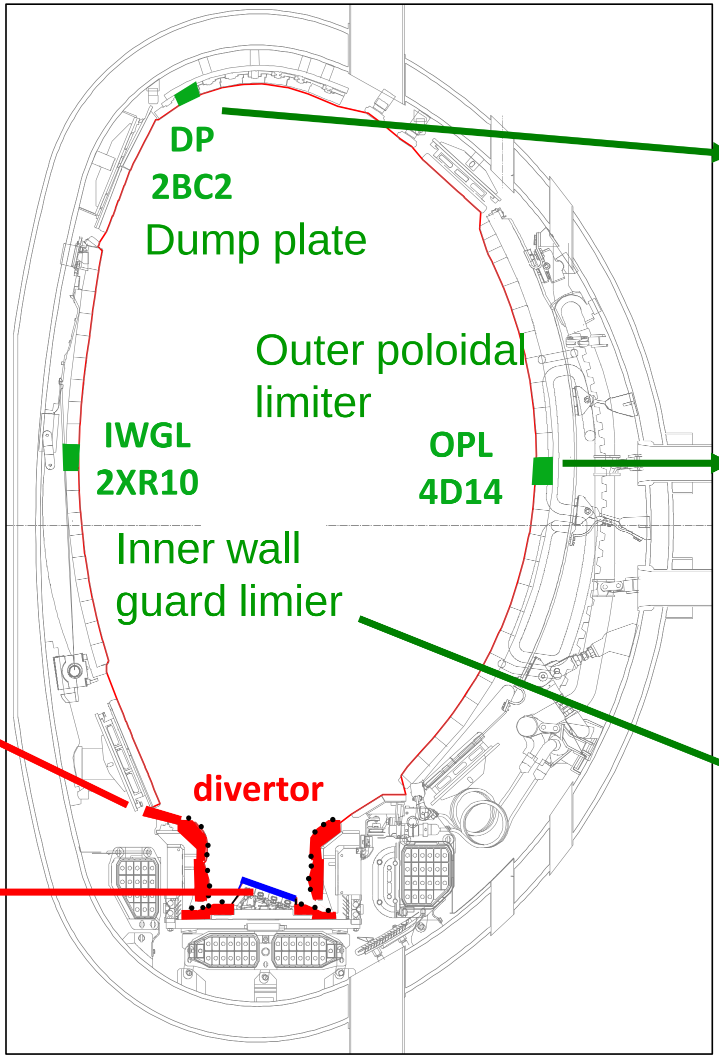
W-coated CFC tile after sample preparation

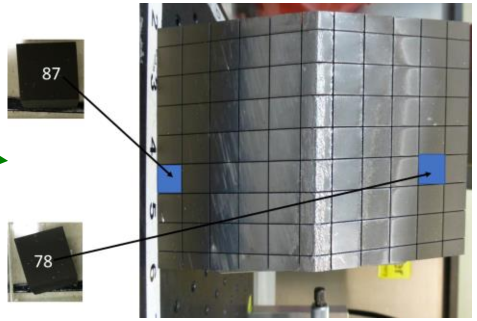


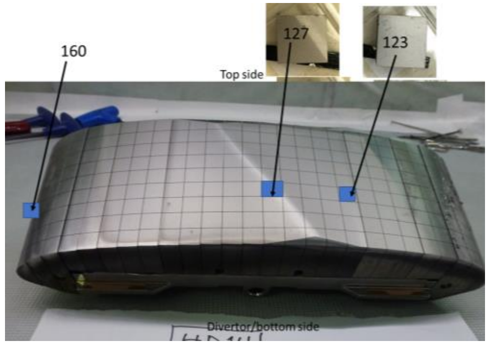


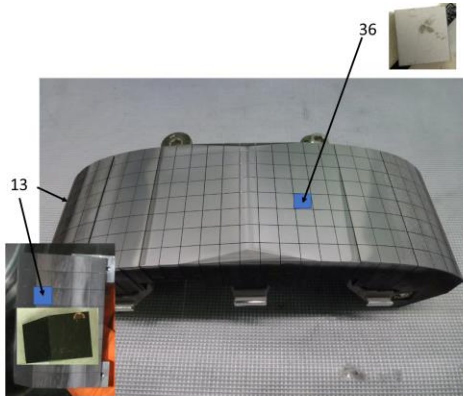






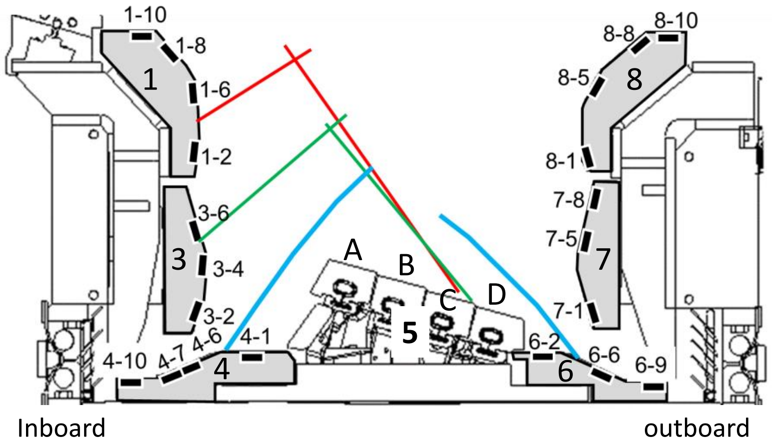






Part of samples were shipped and analyzed in Japan under Broader Approach Activities

D and T distributions in divertor region



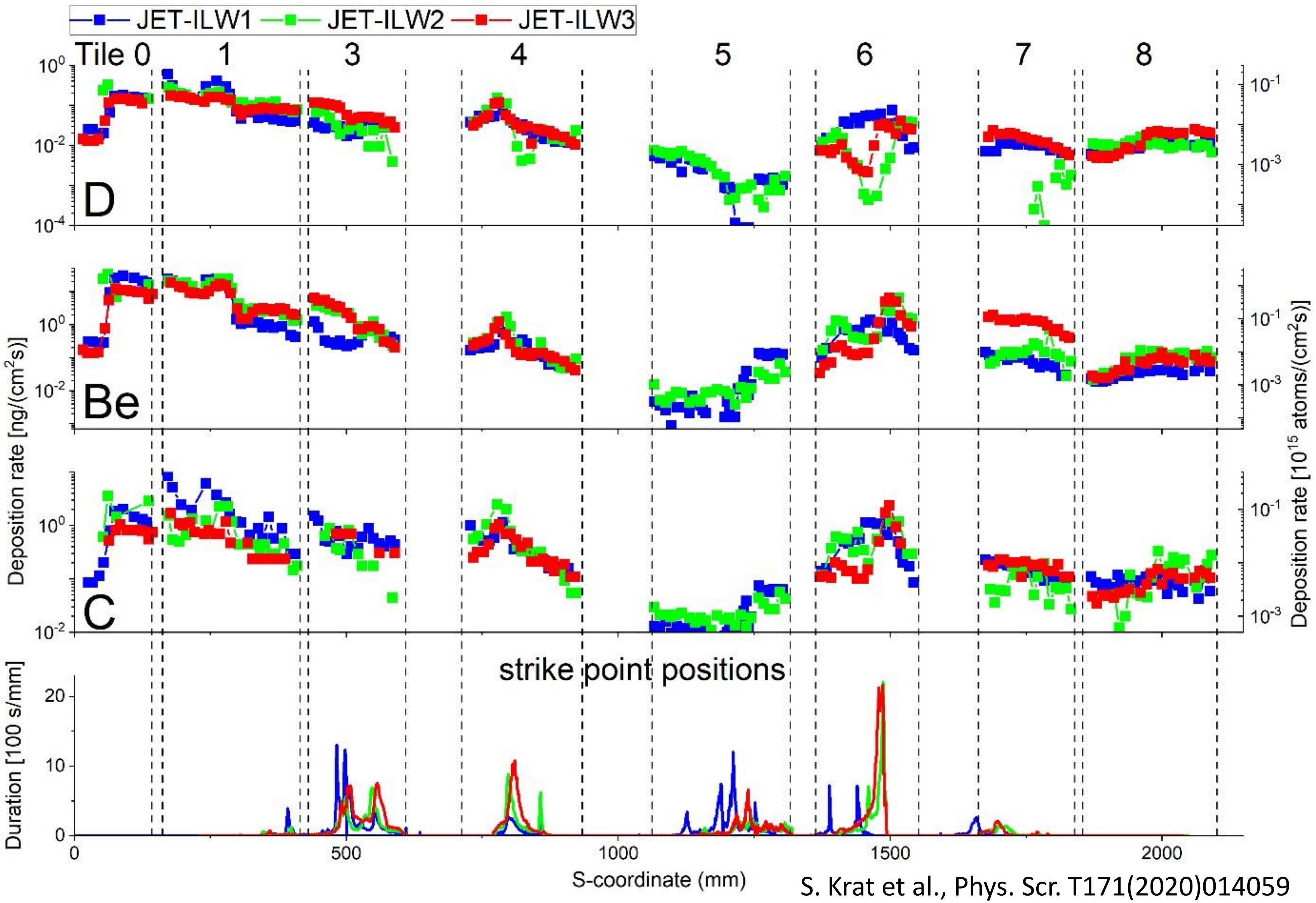
T concentration (Bq/cm^3) in deposited layers was sensitively dependent on C content; higher T content at high C areas.

S. E. Lee et al., Nucl. Mater. Energy 26(2021)100930

D penetration into porous W coating occurred if Be coverage is low.

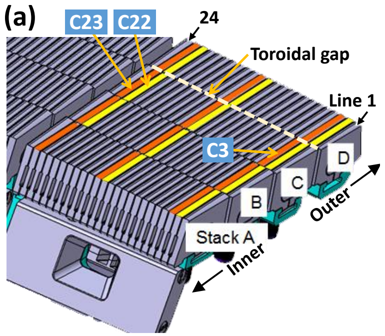
C. Ruset et al., Nucl. Mater. Energy 30(2022)101151

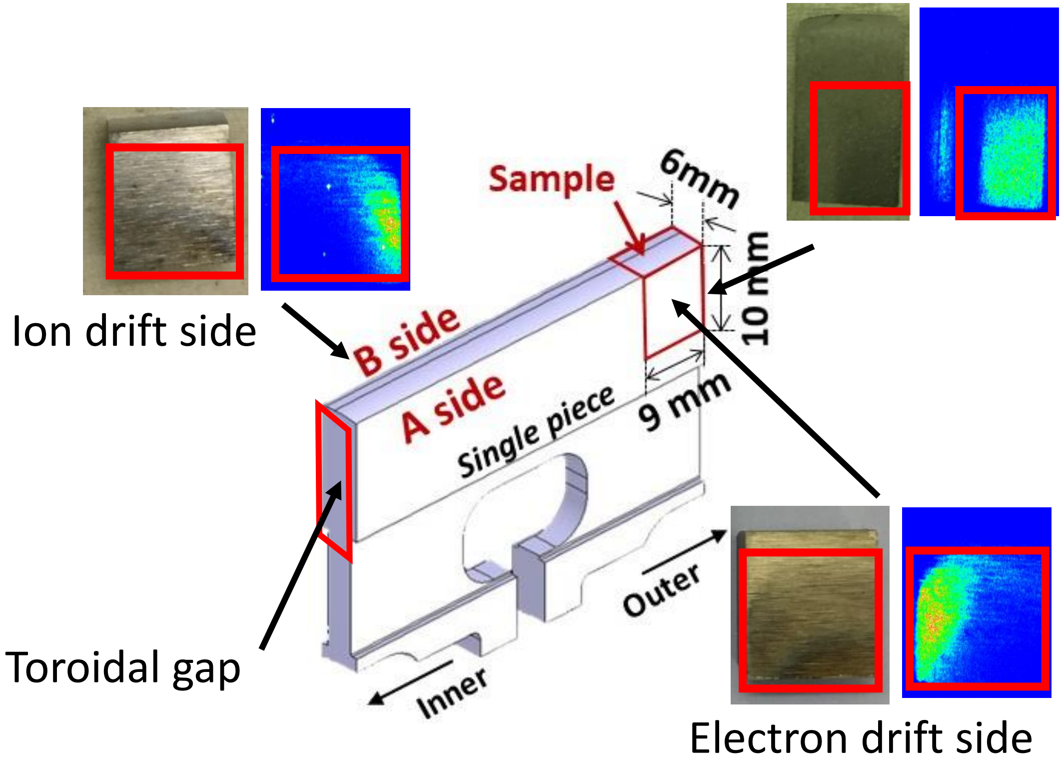
Correlation b/w position along divertor surface, deposition rates of D, Be, C and strike point distribution



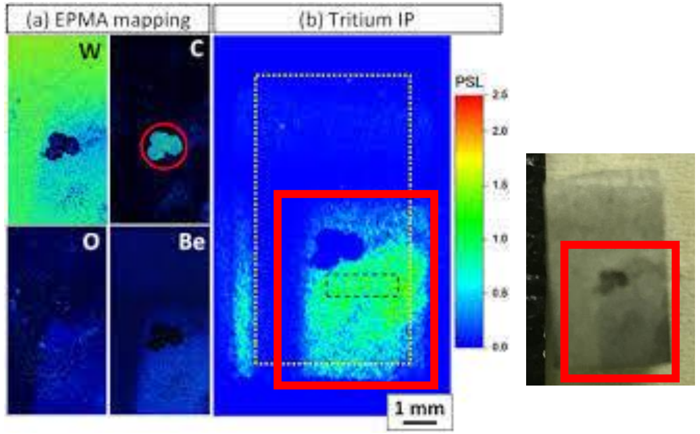
S. Krat et al., Phys. Scr. T171(2020)014059

Gap surfaces of bulk W lamellae





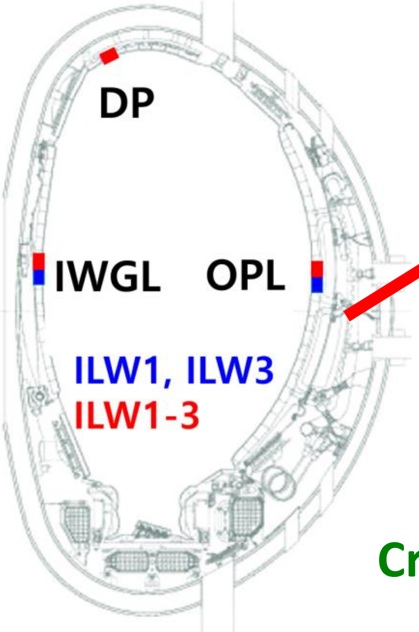
A schematic image of bulk W tile and IP images
of bulk W divertor tiles poloidal and toroidal gap

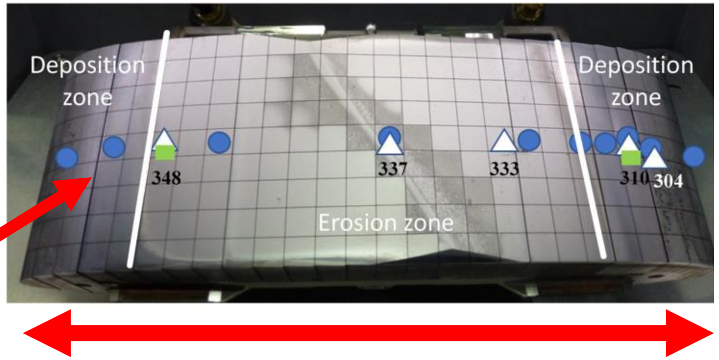


(a) EPMA mapping of W, C, O and Be and (b) tritium IP image
Deposition layer: 300-600 nm
T retention: ~40 kBq/cm^2

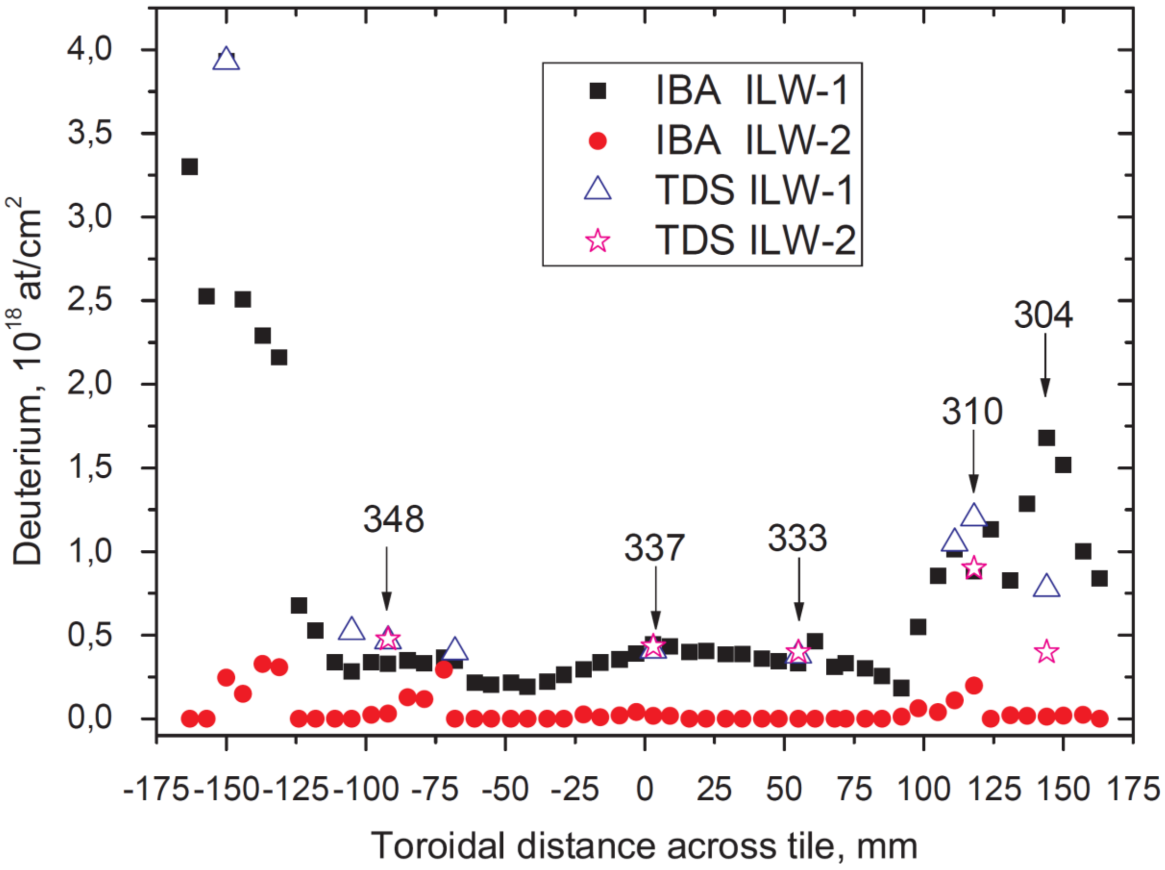
M. Tokitani et al., PS T171(2020)014010
S. E. Lee et al., Nucl. Mater. Energy 26(2021)100930

D retention on plasma-facing surfaces of Be limiters





Cross-section of JET-ILW



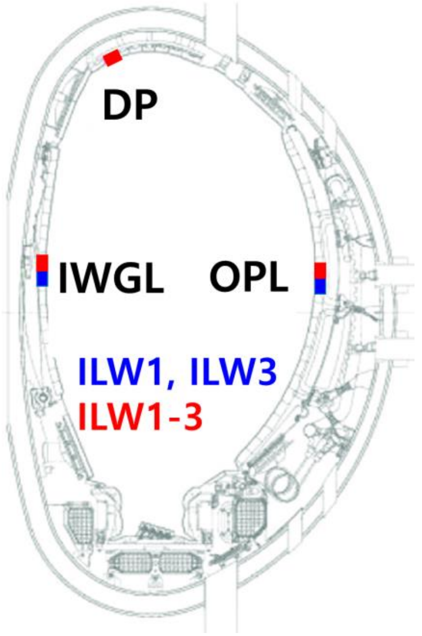
D and W metal distributions on OPL

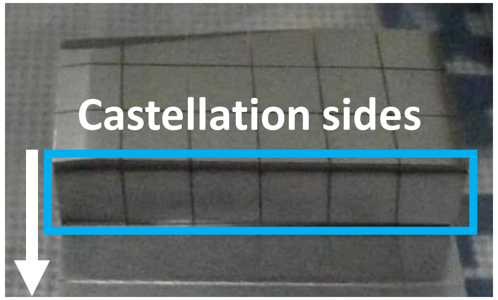
Higher T retention in deposition-dominant zone.

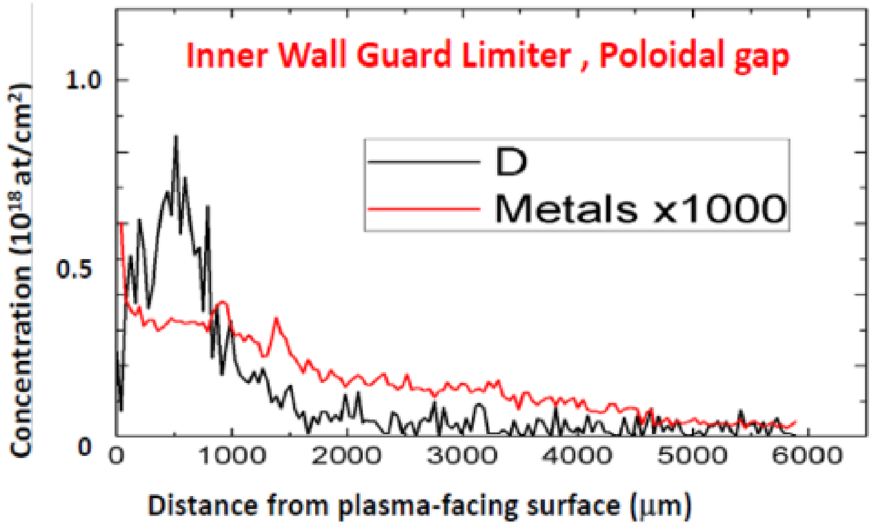
IBA showed lower D retention after ILW2 ended with H discharge. However, TDS showed comparable retention.

A. Baron-Wiechec et al., Fusion Eng. Design 133(2018)135

D and T retention in gaps of Be limiters





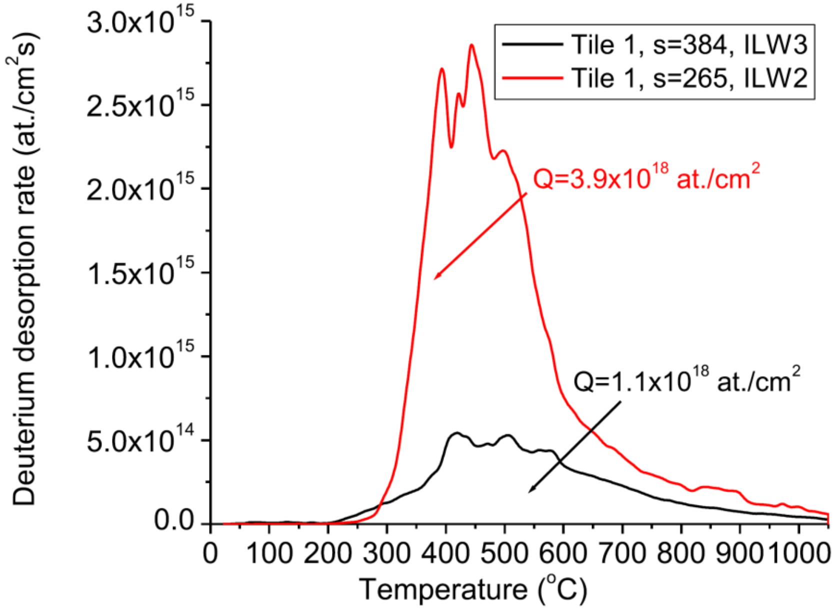


M. Rubel et al., Nucl. Fusion 57(2017)066027

D and T deposition with Be, C, O, metal impurities.

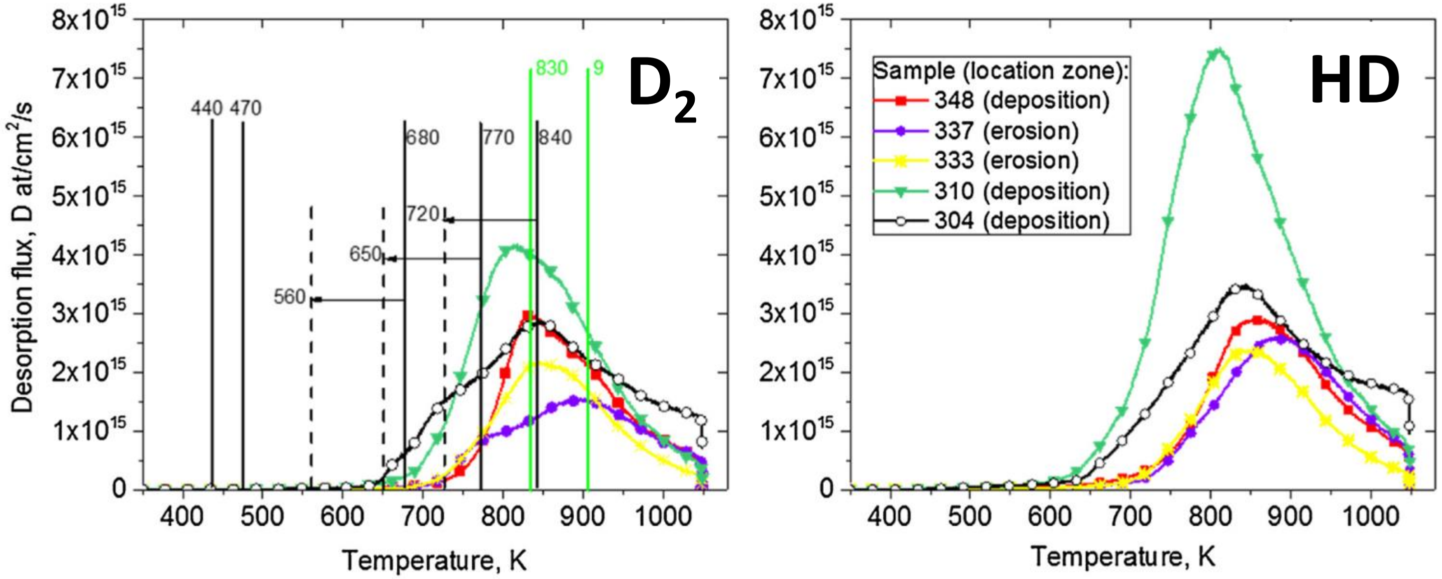
- Similar profiles for D and T
- Similar profiles regardless of location and orientation in the tokamak.

Thermal desorption of D



D release from W-coated CFC tile 1 used in ILW2 and ILW3 with Be deposition

C. Ruset et al., Nucl. Mater. Energy 30(2022)101151



D release from outer poloidal limiter used in ILW2

A. Baron-Wiechec et al., Fusion Eng. Design 133(2018)135

Melting point of Be: 1560 K

Summary of Part A

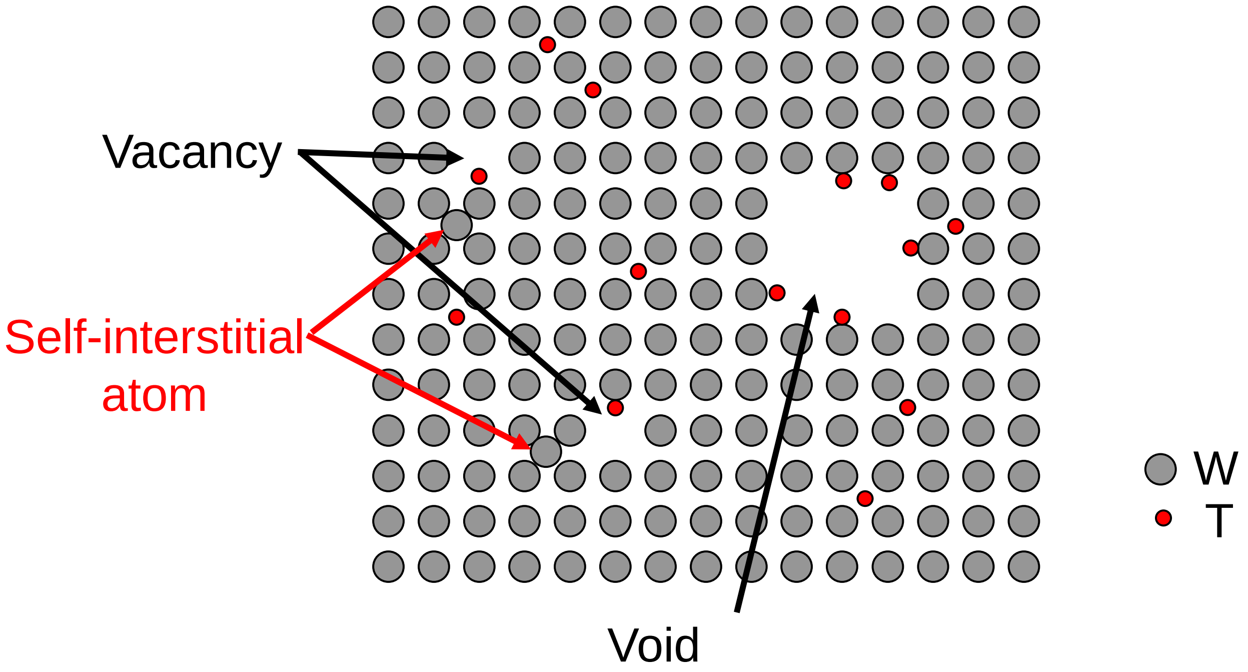
- Co-deposition with Be, C, O and other metallic impurities was the main mechanism of fuel retention in JET-ILW.
- The concentration of fuels in deposition layers increased with increasing C content.
- Co-deposition occurred in the gaps of castellated structure of Be tiles and those between W tiles.
- Thermal desorption peaked at 400–600 ℃ and continued up to the maximum temperature (~850 ℃) of the measurements.

Part B: Neutron irradiation effects on hydrogen retention in W

Plasma-facing material (W) will be exposed to D, T, He ions and 14 MeV neutrons.

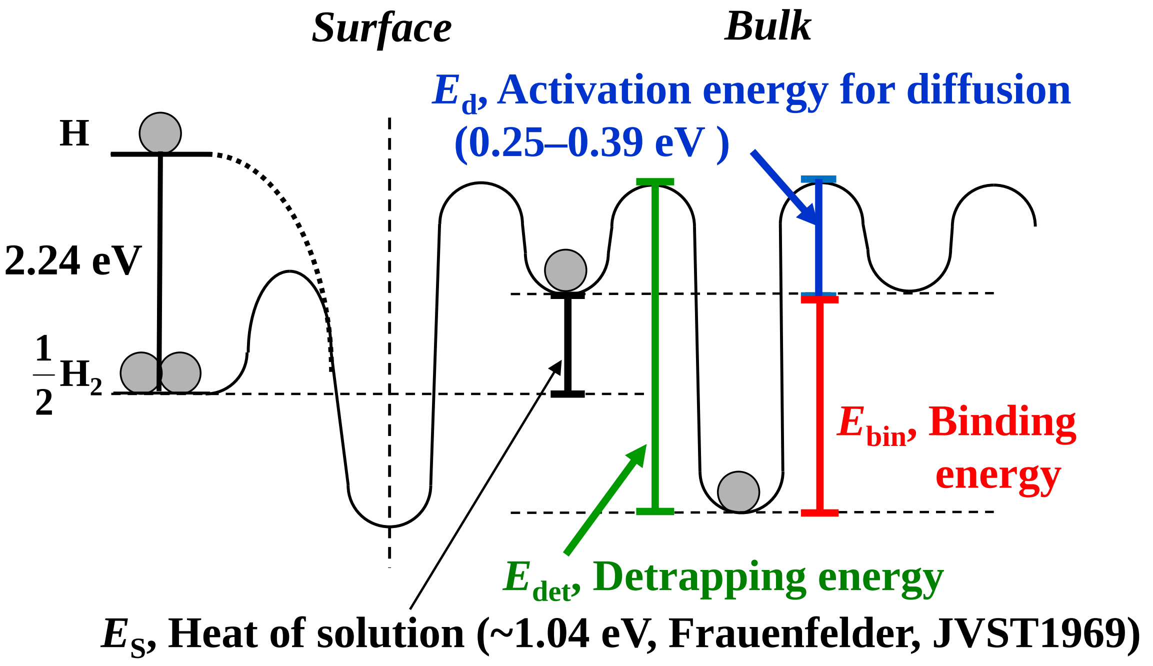
Vacancy type defects formed by neutron irradiation trap hydrogen isotopes strongly and increase fuel retention and tritium (T) inventory up to [D+T}/W = 0.001-0.01.

M. Shimada et al., Phys. Scr. 2011(2011)014051
Y. Hatano et al., Nucl. Fusion 53(2013)073006

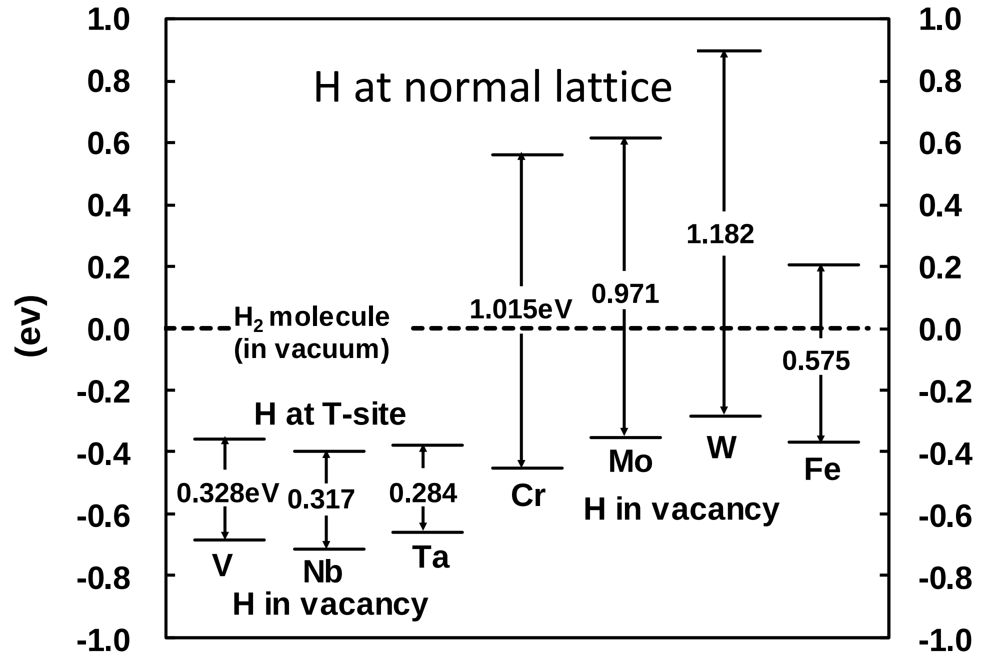


What is problem?

Hydrogen solubility in W is very small due to large positive heat of solution. The large positive heat of solution results in strong trapping at vacancy-type defects.



Potential diagram of H-W system



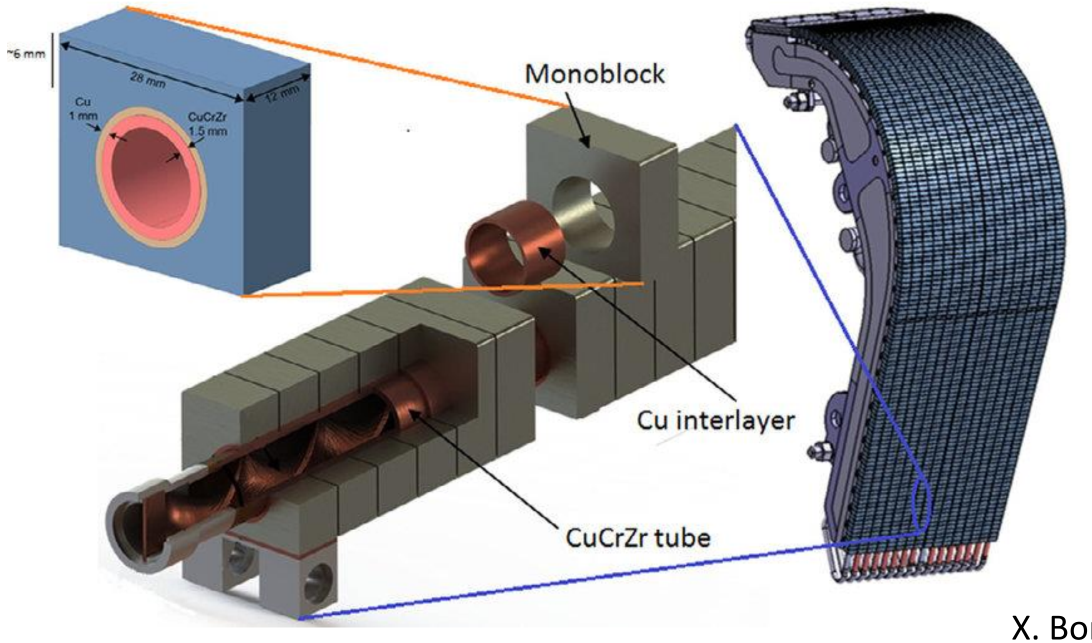
Energy levels of H in interstitial sites and vacancies in various metals

K. Ohsawa et al., Phys. Rev. B, 85(2012) 094102

In this presentation

B-1 Fuel penetration into neutron-irradiated W

B-2 Fuel release from neutron irradiated W by bakeout process at 300 °C



X. Bonnin et al., Nucl. Mater. Energy, 12(2017) 1100-1105

B-1 Fuel penetration into neutron-irradiated W

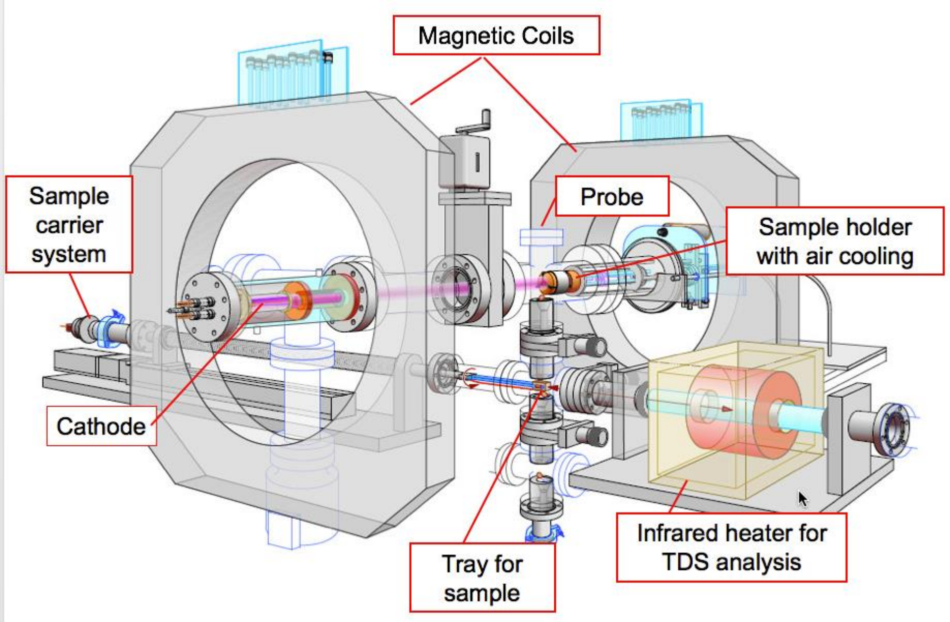
Samples: Disks of pure W (99.99 mass%, powder metallurgy, A. L. M. T. Co.)
ϕ 6 mm × t 0.5 mm, final heat treatment at 1173 K for 1 h (stress relieve annealing)

Irradiation: Belgium Reactor 2 (BR2) at SCK·CEN
MICADO-7-CALLISTO (2013−2014) 563 K, 1.1×10^{24} n m^{-2} (> 1 MeV), 0.06 dpa
MICADO-8-CALLISTO (2015−2016) 563 K, 3.0×10^{23} n m^{-2} (> 1 MeV), 0.016 dpa
dpa: displacement per atom

The irradiated samples were shipped to Tohoku U. and electropolished in 1 M NaOH solution to remove oxide films formed during irradiation.

Neutron irradiation was also performed at HFIR, ORNL.

Plasma exposure



Compact Divertor Plasma Simulator (CDPS)

N. Ohno et al., Plasma Fusion Res. 12 (2017) 1405040.

- ✓ A linear plasma device in radiation controlled area in Institute for Materials Research, Tohoku U.
- ✓ Good temperature control (±5 °C).
- ✓ Temperature rise before plasma ignition by electron bombardment.
- ✓ Thermal desorption measurement without air exposure.
- ✓ TEM sample (ϕ 3 mm, t 0.1 mm) is acceptable.

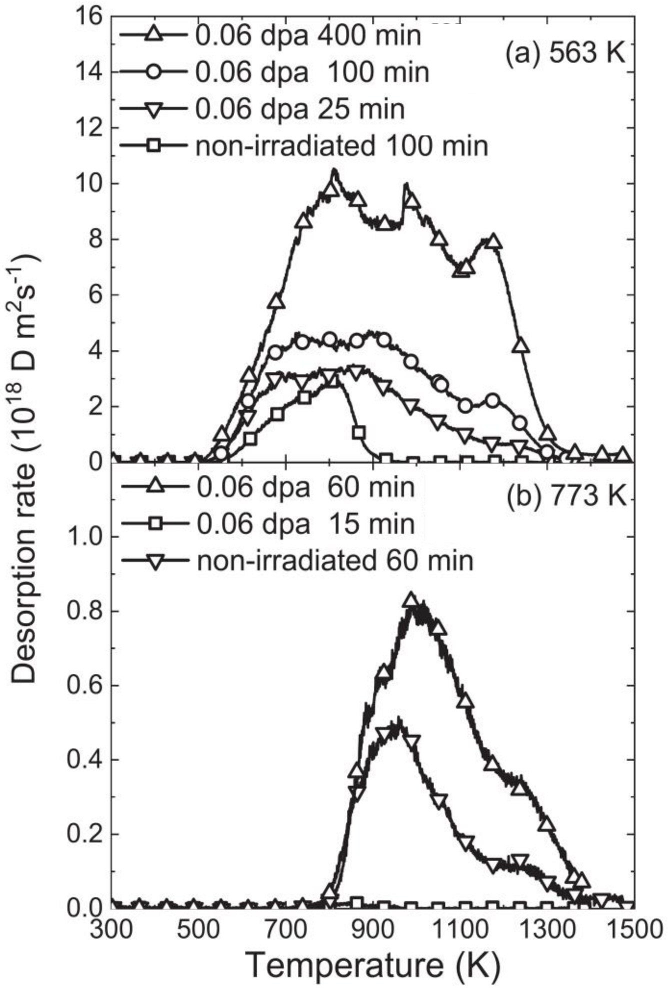
Plasma exposure at 563 K (irradiation temp.)

- Energy: 105 eV
- Flux: 5.4×10^{21} m^{-2} s^{-1}
- Duration of exposure: 25, 100 and 400 min.

Plasma exposure at 773 K

- Energy: 105 eV
- Flux: 9.6×10^{21} m^{-2} s^{-1}
- Duration of exposure: 15 and 60 min.

D penetration ··· Thermal desorption spectrometry (TDS)



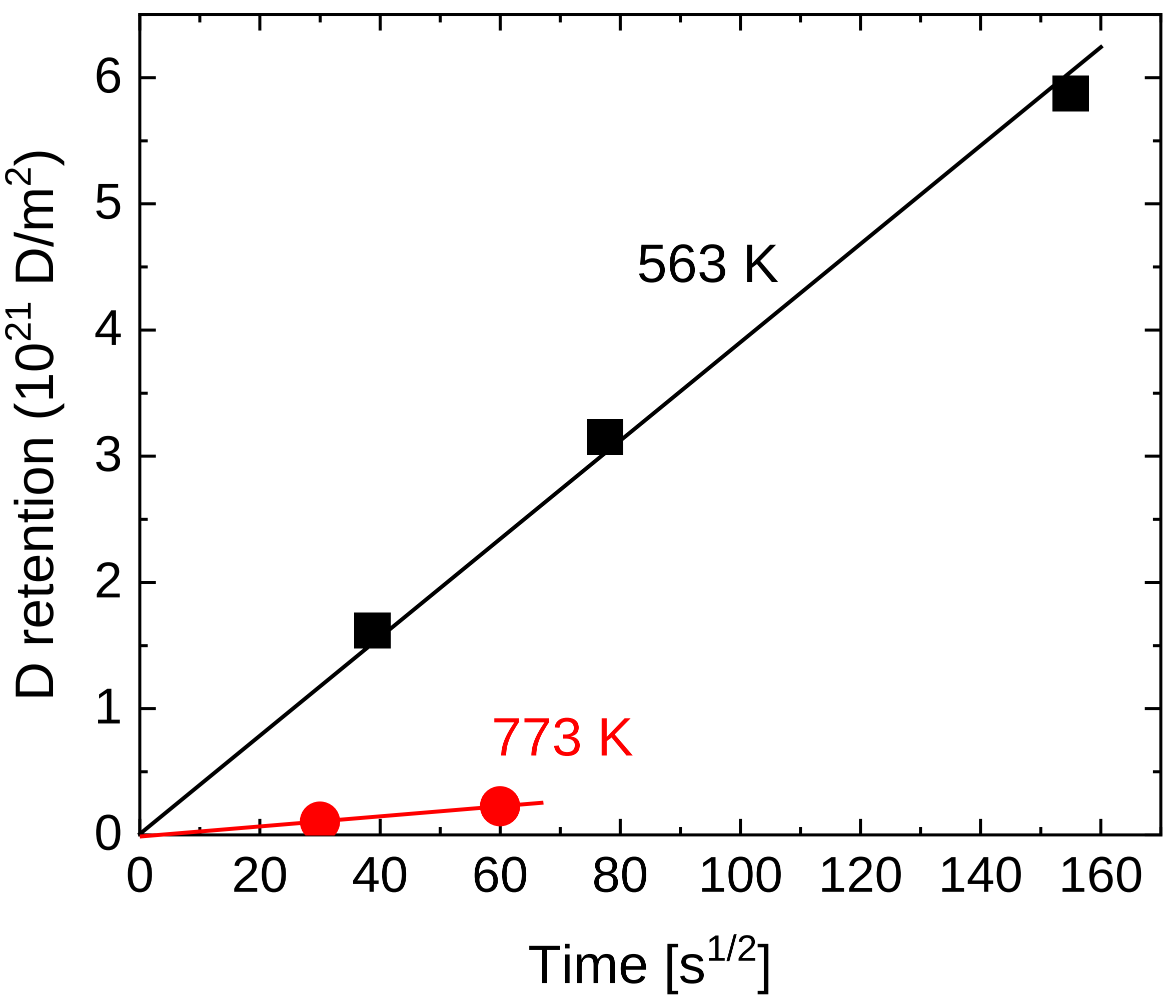
TDS spectra from neutron-irradiated and non-irradiated W samples

- ✓ MICADO-7-CALLISTO (2013−2014) 0.06 dpa
- ✓ Plasma exposure conditions
  - ・Energy: 105 eV
  - ・Flux: 5.4×10^{21} m^{-2} s^{-1} for 563 K
    9.6×10^{21} m^{-2} s^{-1} for 773 K

- ➢ Significant increase in D retention after neutron irradiation.
- ➢ High temperature shoulders appeared after neutron irradiation.

M. Yajima et al., Phys. Scr. 96(2021)124042

D penetration ··· Thermal desorption spectrometry (TDS)



Change in D retention in neutron-irradiated W with square root of plasma exposure time.

✓ MICADO-7-CALLISTO (2013−2014) 0.06 dpa

✓ Plasma exposure conditions

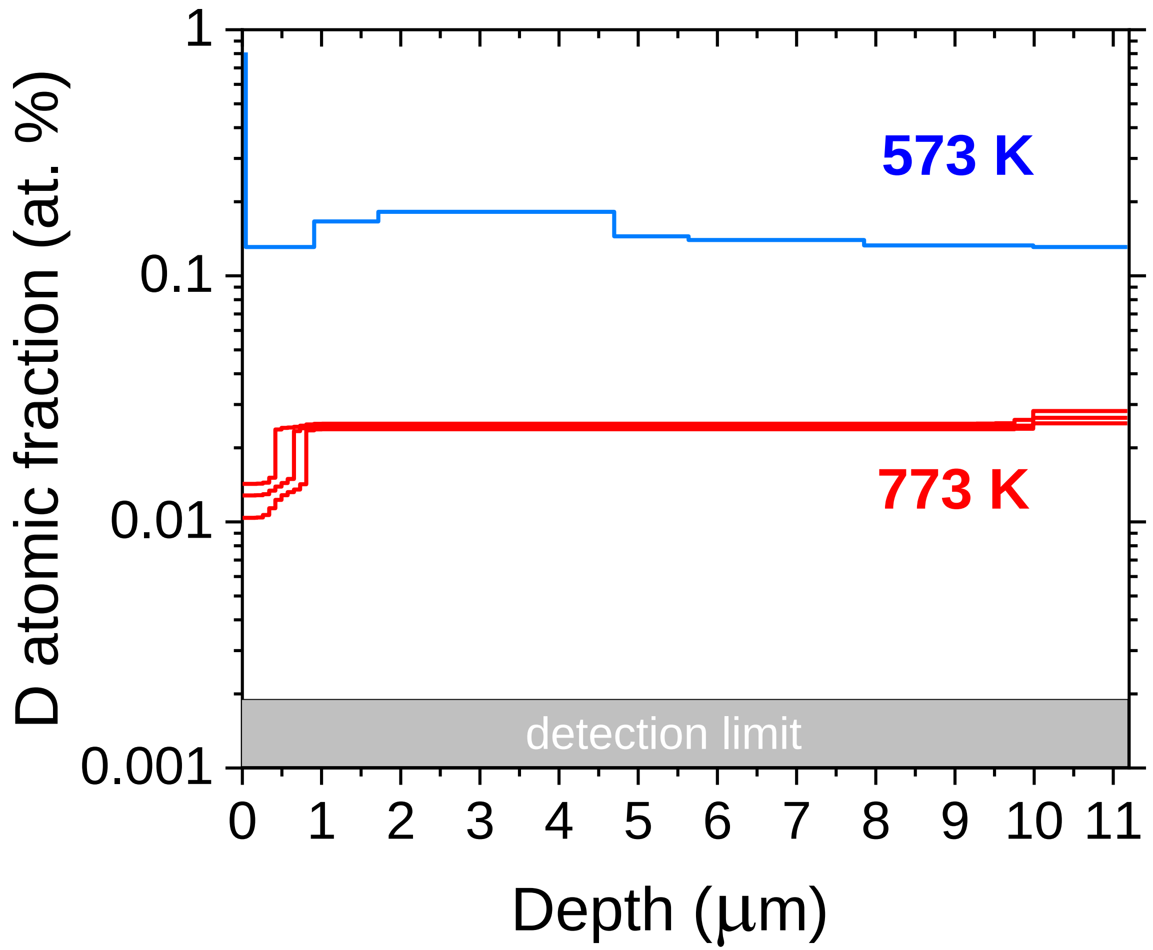
- Energy: 105 eV
- Flux: 5.4×10^{21} m^{-2} s^{-1} for 563 K
  9.6×10^{21} m^{-2} s^{-1} for 773 K

➢ D retention at 773 K << D retention at 563 K

➢ D retention was proportional to square root of plasma exposure time (= D fluence)

M. Yajima et al., Phys. Scr. 96(2021)124042

D penetration ··· Nuclear reaction analysis (NRA)



Depth profiles of D in neutron-irradiated W after plasma exposure at 573 K and 773 K.

- ✓ MICADO-8-CALLISTO (2015–2016) 0.016 dpa
- ✓ Plasma exposure conditions
  - ・Energy: 105 eV
  - ・Flux and exposure time
    - 573 K 3.61×10^{21} m^{-2} s^{-1} and 3150 s
    - 773 K 7.98×10^{21} m^{-2} s^{-1} and 1250 s
  - ・Fluence: 1.0-1.1 $\times 10^{25}$ m^{-2} s^{-1}

D diffused into the bulk beyond the detection depth of NRA.

D fraction: 0.15 at.% at 573 K & 0.025 at.% at 773 K

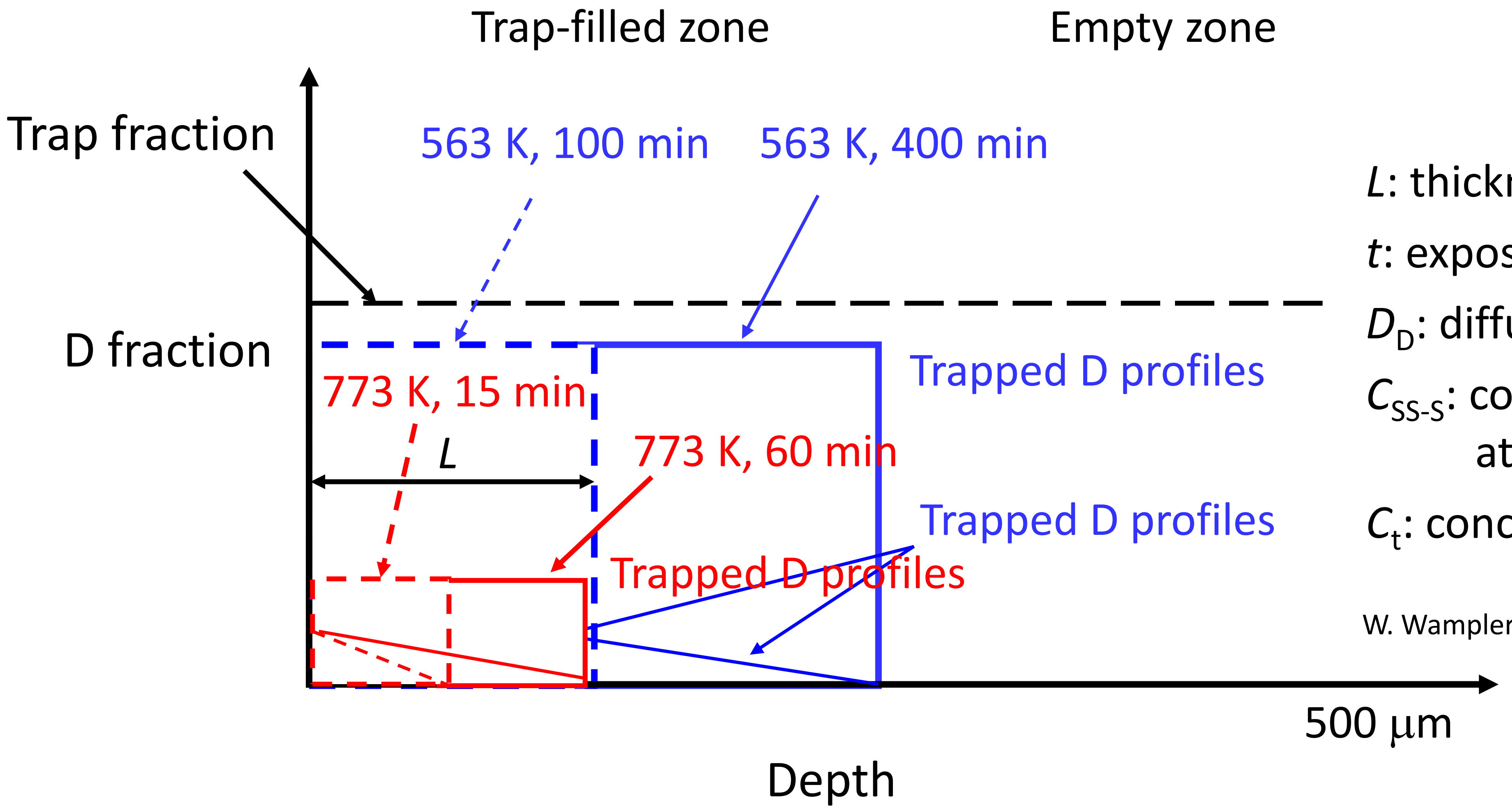
Diffusion length:

66 μm at 563 K, 400 min

14 μm at 773 K, 60 min

M. Yajima et al., Phys. Scr. 96(2021)124042

D penetration ··· Model



$L^2 = t\ (2D_D C_{SS\text{-}S}/C_t)$

L: thickness of trap-filled zone

t: exposure time

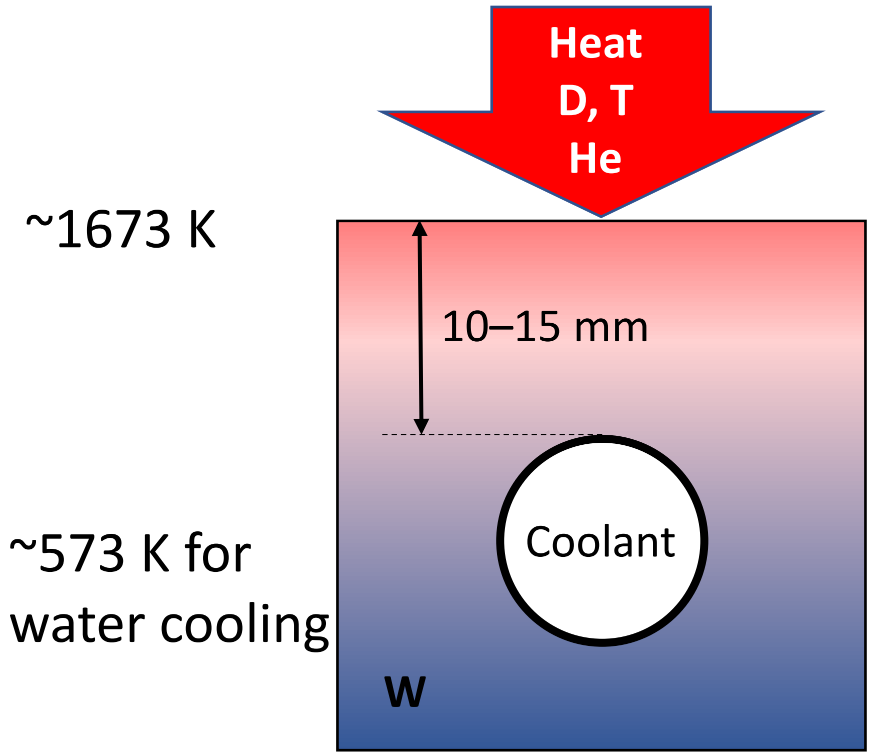
D_D: diffusivity of mobile D

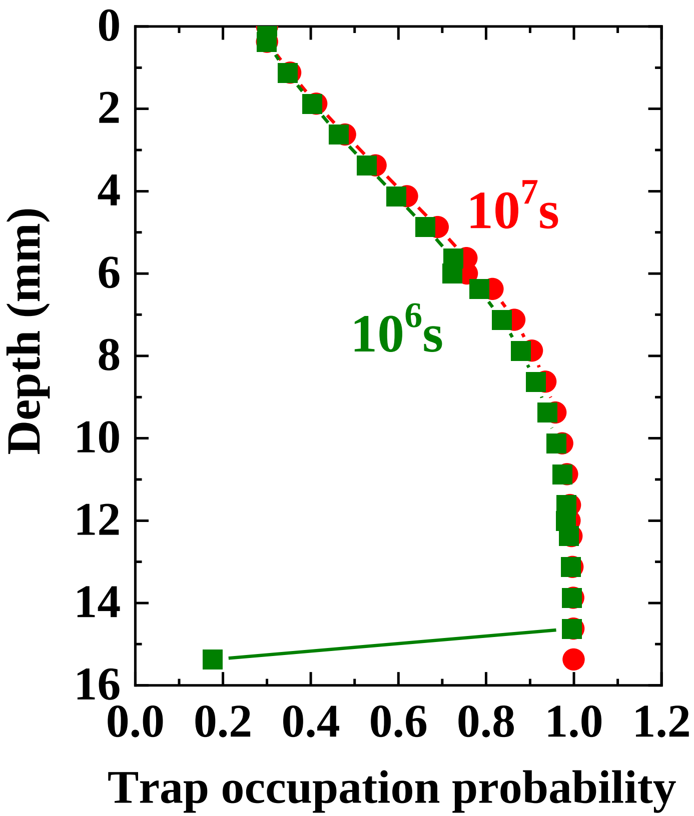
$C_{SS\text{-}S}$: concentration of mobile D at subsurface

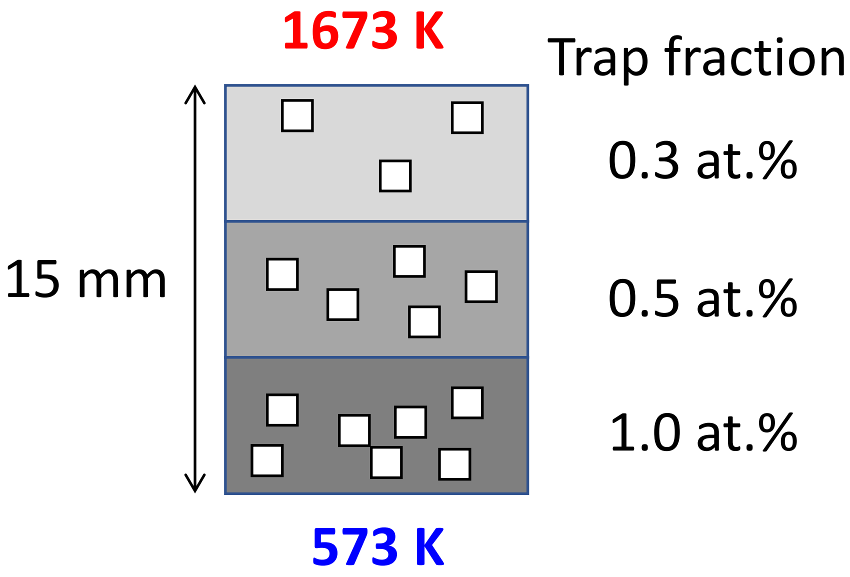
C_t: concentration of trap

W. Wampler, R. Doerner, Nucl. Fusion 49(2009)115023

D penetration ··· under temperature gradient







If W monoblocks are used as divertor structure, there are "cold" regions around cooling pipes. These "cold" regions dominate fuel inventory.

D penetration … Effects of He seeding in plasma

HFIR Rabbit
0.3 dpa

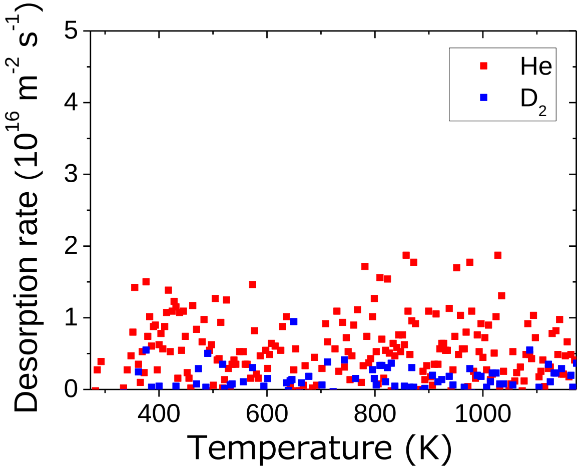
Neutron irradiation temperature

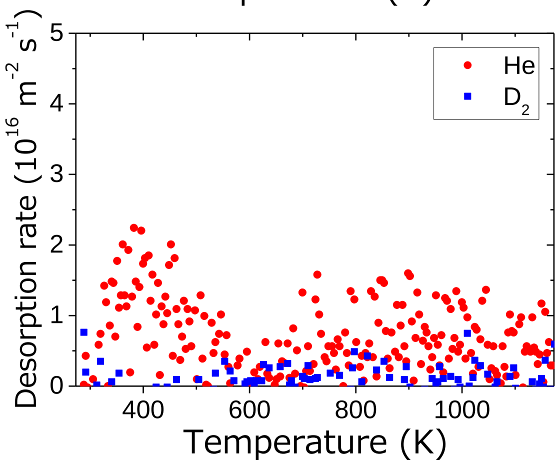
800 °C (1073K)

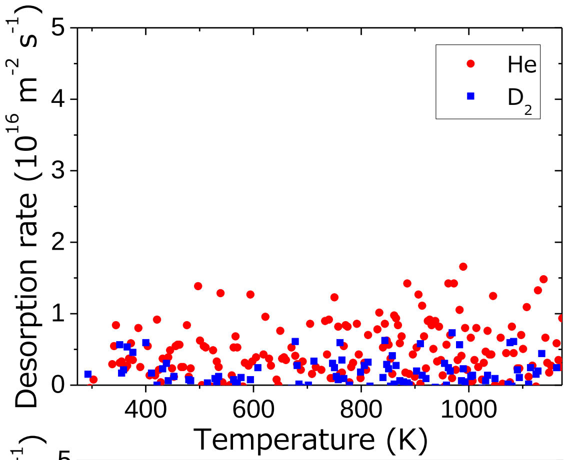
1100 °C (1373K)

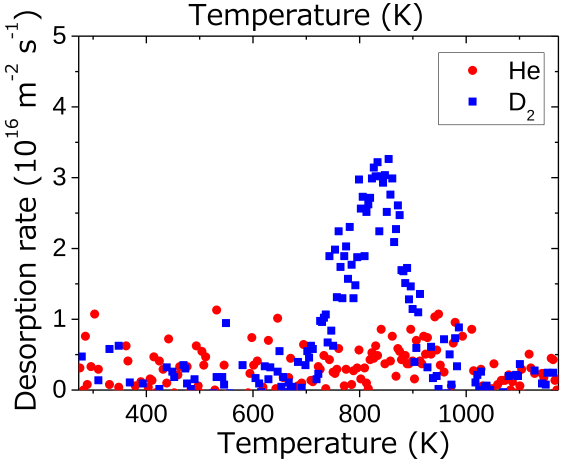
High He flow rate (D:He = 15: 10 [sccm])

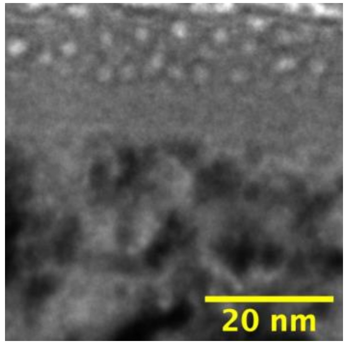
Low He flow rate (D:He = 24:1 [sccm])

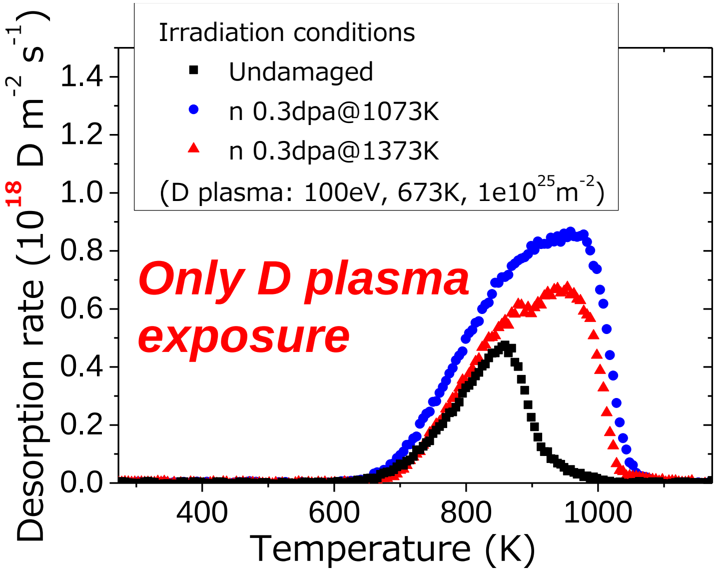












HFIR (ORNL) + TPE (INL)

Y. Nobuta, M. Shimada et al., Fusion Sci. Technol. 77(2021)76-79

D penetration ··· Alloying effects

Addition of Re and Cr to W enhanced vacancy annihilation and reduce fuel retention dramatically.

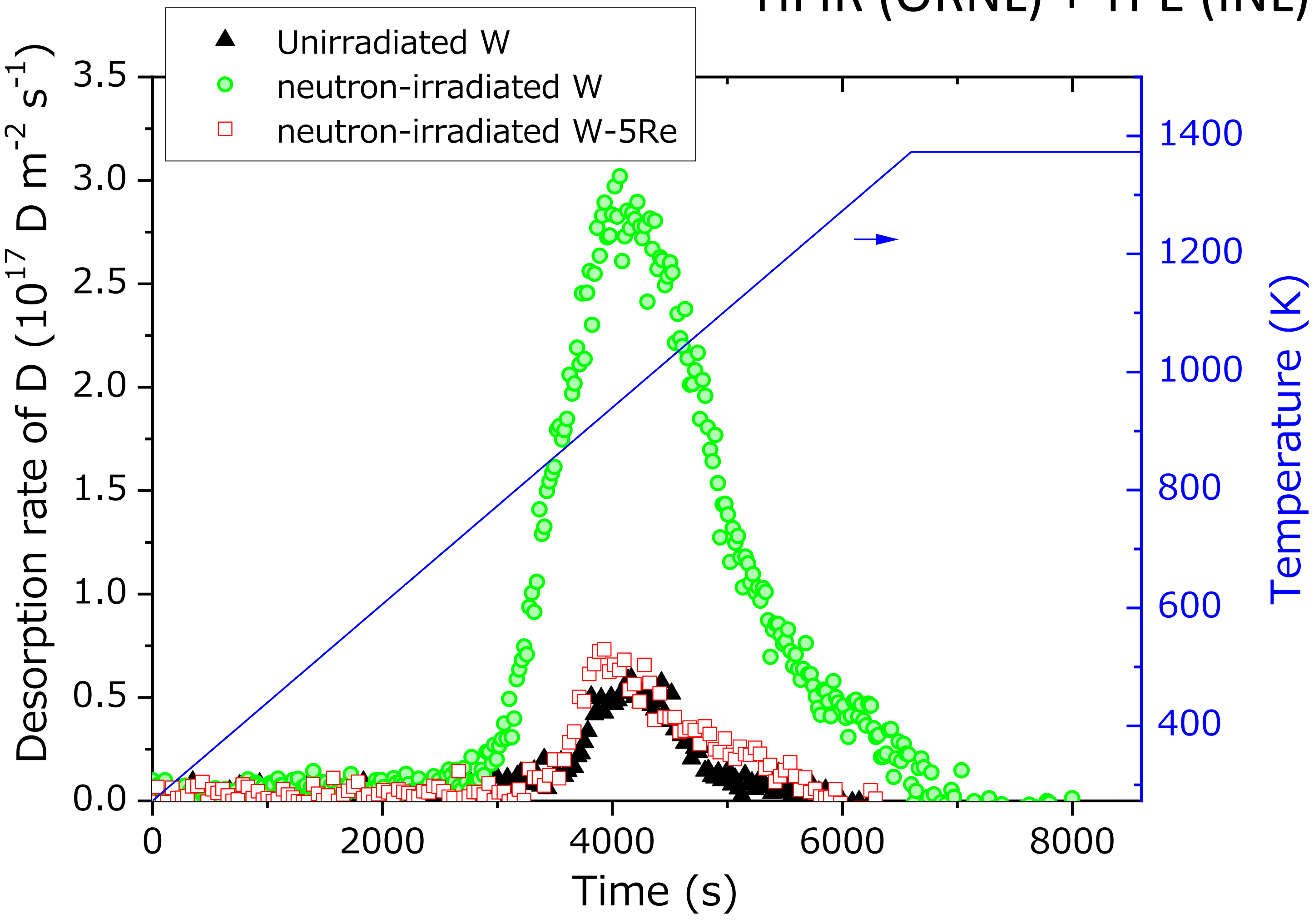
However, the effects are significant for irradiation at ≥ 773 K.

Y. Hatano et al., Nucl. Mater. Energy 9(2016)93-97
J. Wang et al., J. Nucl. Mater, 545(2021)152749
J. Wang et al., J. Nucl. Mater, 559(2022)153449

TDS spectra from non-irradiated and neutron-irradiated W and W-5Re alloy

Irradiation conditions: HFIR, ORNL
864 K and 0.35 dpa for pure W
792 K and 0.26 dpa for W-5Re alloy

HFIR (ORNL) + TPE (INL)



Y. Nobuta et al., J. Nucl. Mater, 77(2021)153774

B-2 Fuel release by bakeout process at 573 K

Background

In DEMO reactor, there will be about 1 month gap between termination of operation and start of maintenance work to wait decay of short-lived radioisotopes (cooling).

Can we remove T from PFCs during this period of time by heating PFCs with decay heat?

Maximum temperature may be ~573 K for water-cooled system.

Experimental

(1) Pairs of tungsten samples were irradiated to 0.016 displacement per atom (dpa) in BR2 rector at 573 , 673 and 773 K, and then exposed to D plasma (Flux 10^{21}–10^{22} D $m^{-2}s^{-1}$, Fluence 10^{25} D m^{-2}).

(2) D retention in the first sample of each pair was measured by TDS immediately after plasma exposure.

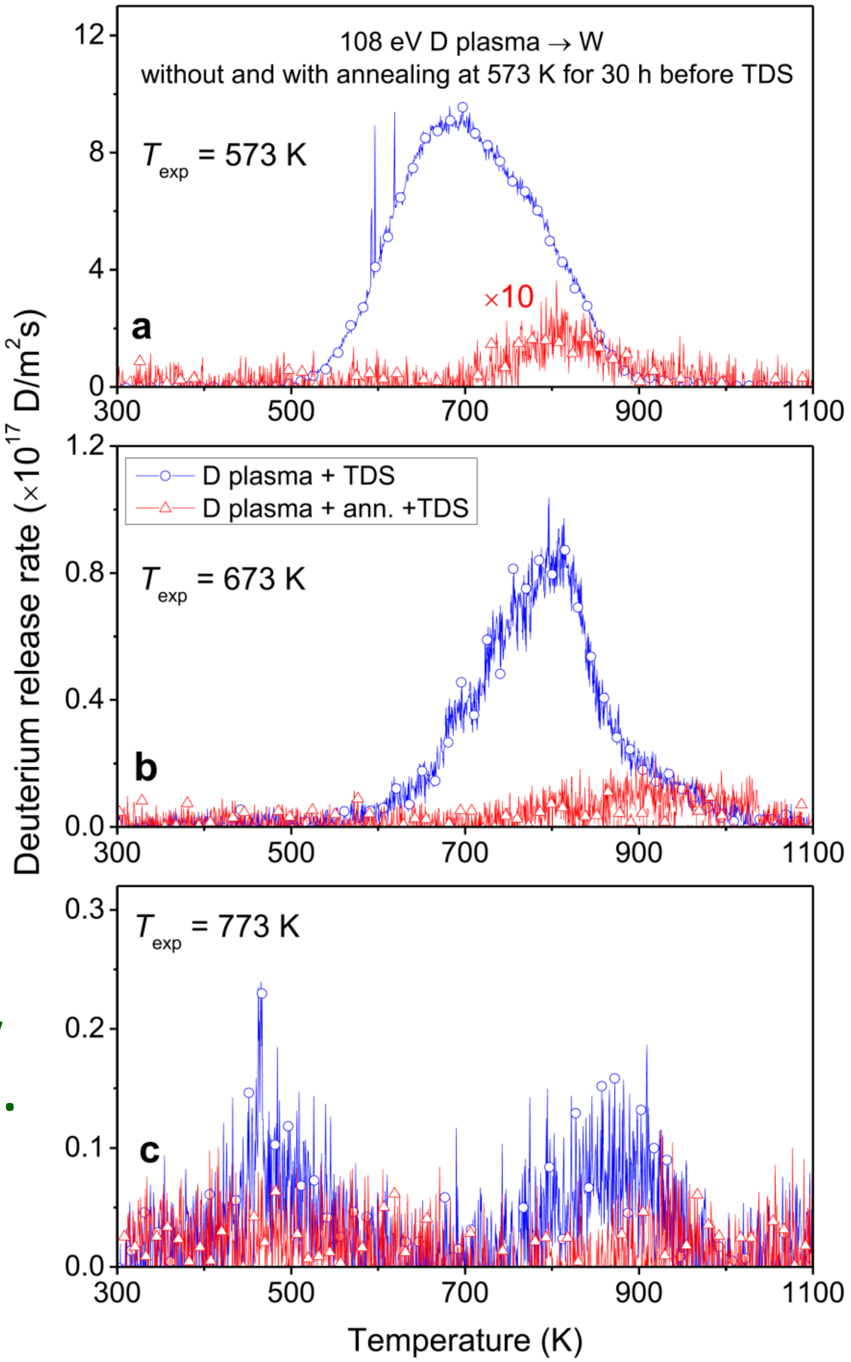
(3) The second sample was subjected to TDS measurement after heating in vacuum at 573 K for 30 h.

(4) The amount of D removed during heating at 573 K for 30 h was evaluated from the difference in D retention of the two samples.

(5) Non-irradiated samples were also examined.

Non-irradiated samples

- D retention decreased with increasing plasma exposure temperature.
- Almost all D was removed by heating at 573 K for 30 h.



TDS spectrum of D from non-irradiated immediately after plasma exposure and after heating at 573 K for 30 h.

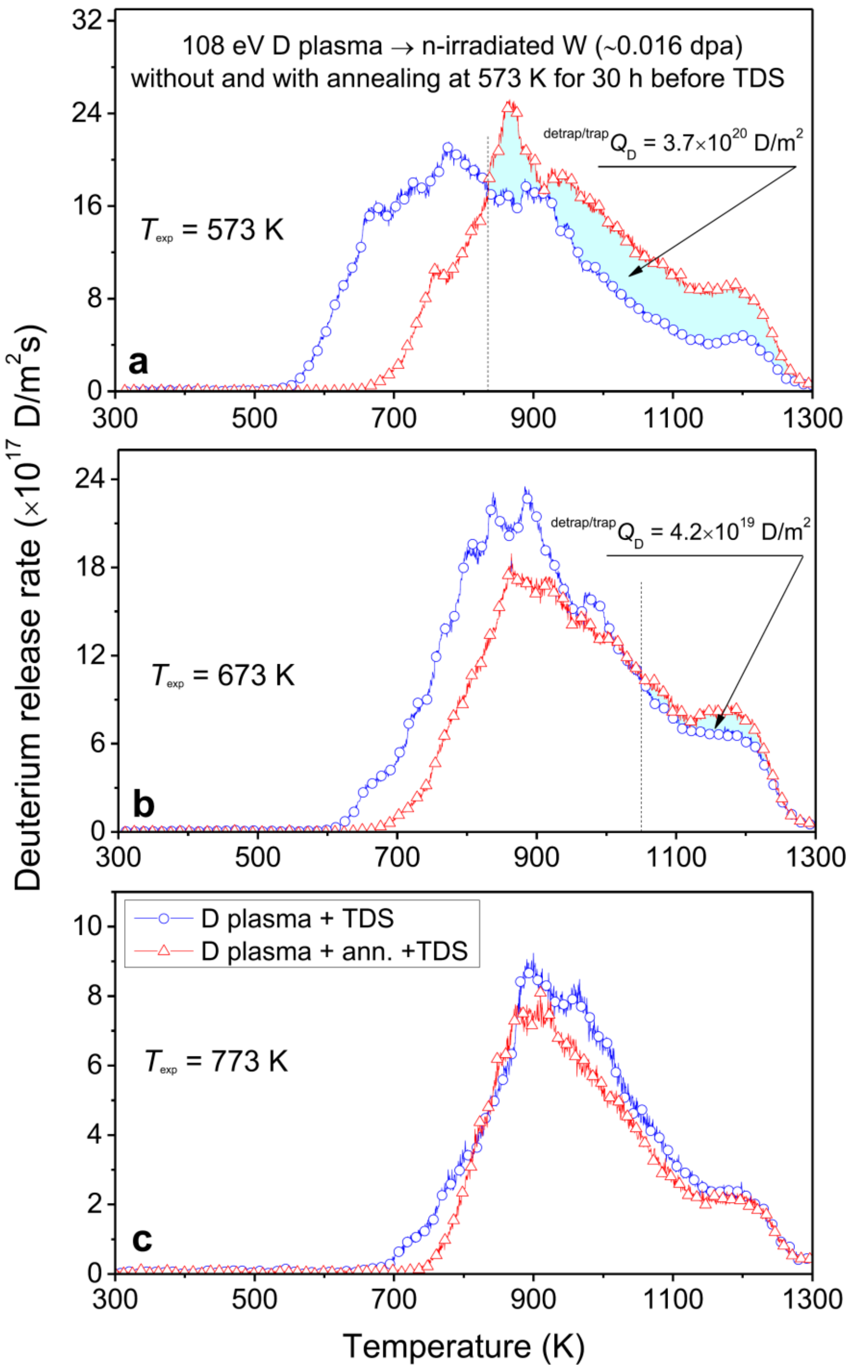
Blue: immediately after plasma exposure

Red: after heating at 573 K for 30 h

V. Kh. Alimov et al., Nucl. Fusion 60(2020)096025

Neutron-irradiated samples

- 10-20% D release by heating at 573 K for 30 h.
- Increase in D release at high temperature region after heating at 573 K for 30 h.



TDS spectrum of D from non-irradiated immediately after plasma exposure and after heating at 573 K for 30 h.

Blue: immediately after plasma exposure

Red: after heating at 573 K for 30 h

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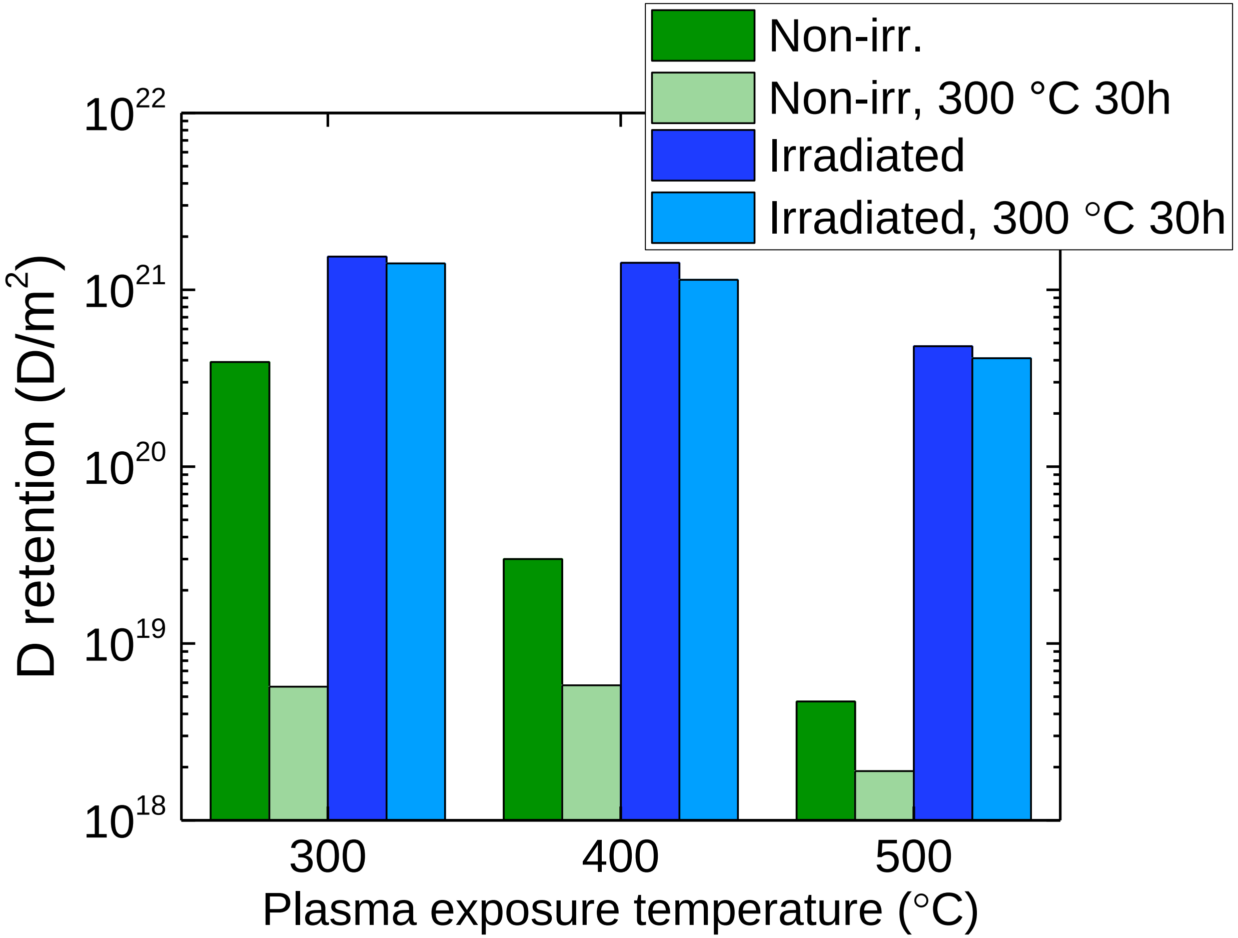
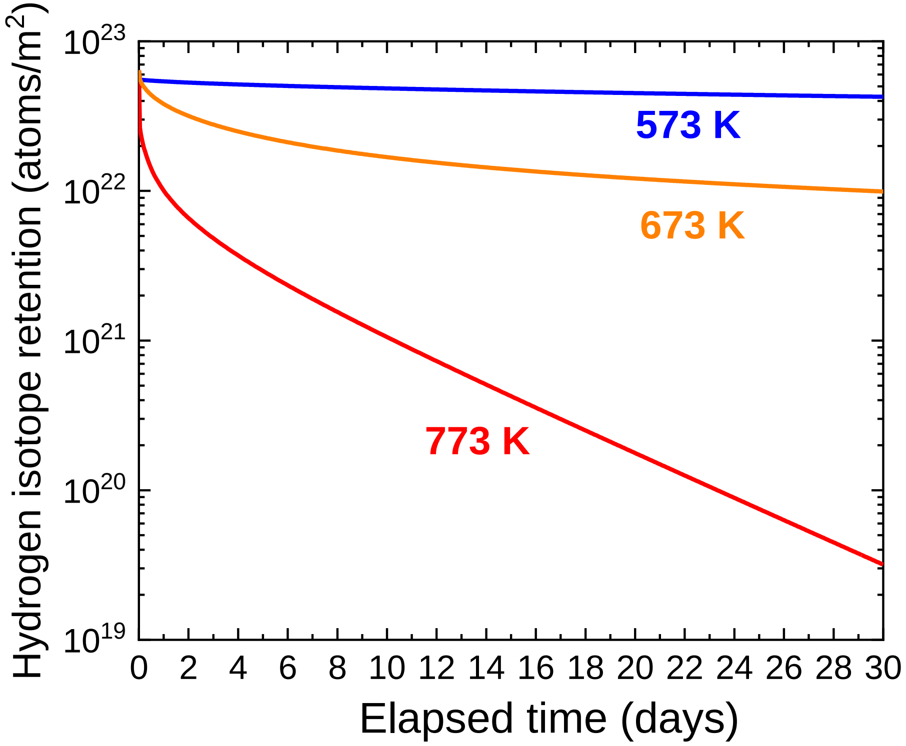
Comparison between irradiated and non-irradiated samples



D retention in non-irradiated and neutron-irradiated W with or w/o heating in vacuum at 573 K for 30 h.

W thickness: 0.5 mm
Uniform hydrogen isotope distribution before heating.
Trap density: 1.36 eV 0.16 at.%, 1.63 eV 0.044 at.%
Full occupation of traps by D before heating.



Fuel removal from neutron irradiated W by heating for 30 days.

Summary of Part B

- ✓ Neutron irradiation at 563 K resulted in formation of vacancies and their clusters (>V_{10}) and increase in D retention due to trapping effects.
- ✓ D retention was proportional to the square root of D fluence/plasma exposure time.
- ✓ If W monoblock is used, cold region around cooling pipe can dominate T inventory after long-term operation.
- ✓ He in plasma can prevent fuel penetration into neutron-irradiated W.
- ✓ Re and Cr alloying may be effective but need to be confirmed by neutron-irradiation experiment at 773 K and lower temperatures..
- ✓ D release at 573 K (bakeout temp.) was significantly hindered after neutron irradiation.
- ✓ TMAP simulation showed heat treatment at ≥773 K is effective for fuel removal from neutron-irradiated W (thickness 0.5 mm, first wall).
- ➢ Removal of He implanted layer by sputtering of seeding elements (N, Ne, etc.)?