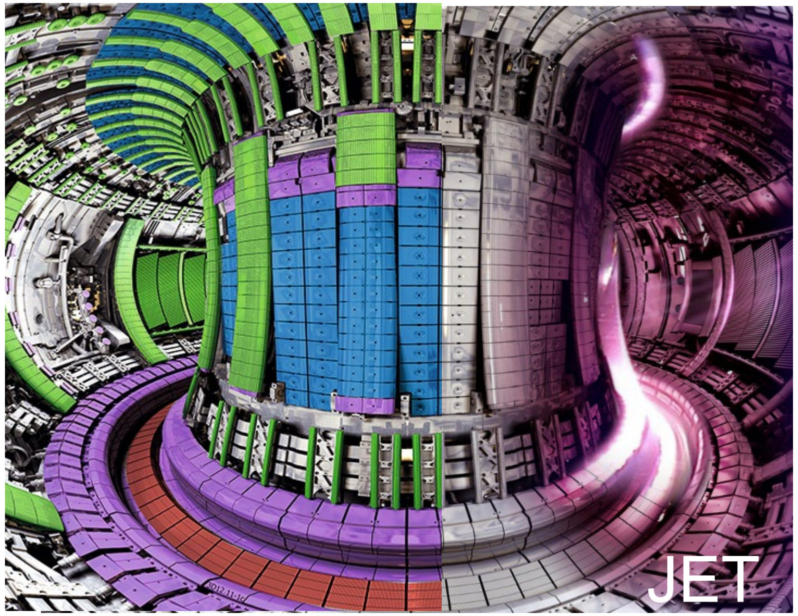
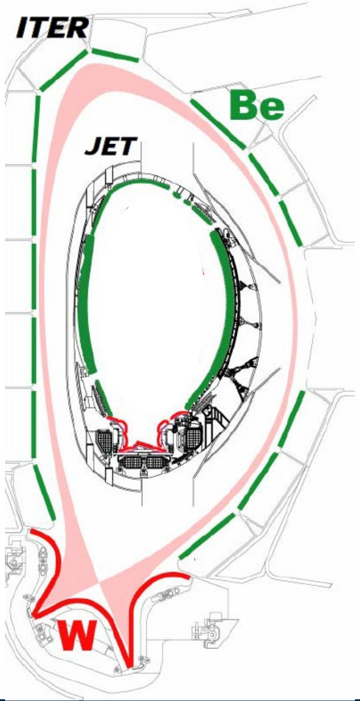
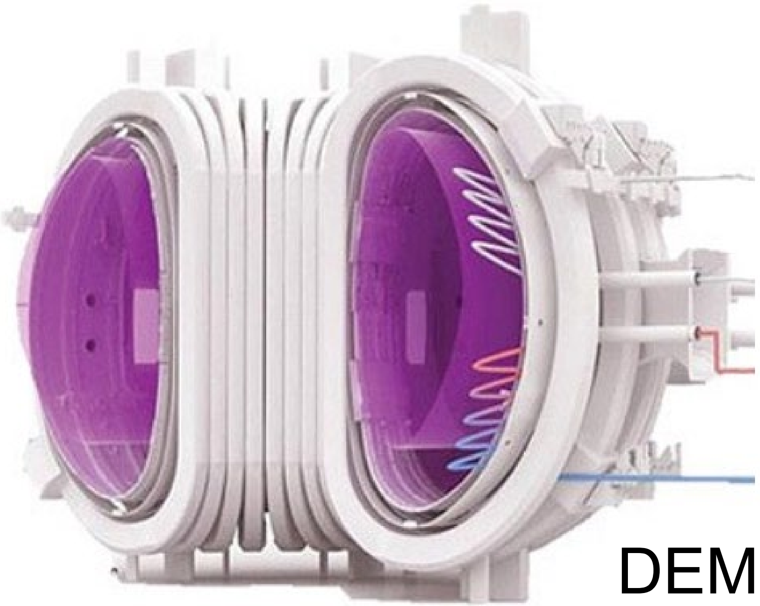






Plasma-material interaction in the main chamber of fusion reactors: the role of high-Z and low-Z wall materials on erosion, dust, and fuel retention

17.11.2021 I Sebastijan Brezinsek and EUROfusion WP PFC, PWIE and JET contributors | IEK-4 Plasmaphysik | Forschungszentrum Jülich

EUROfusion

Member of the Helmholtz Association

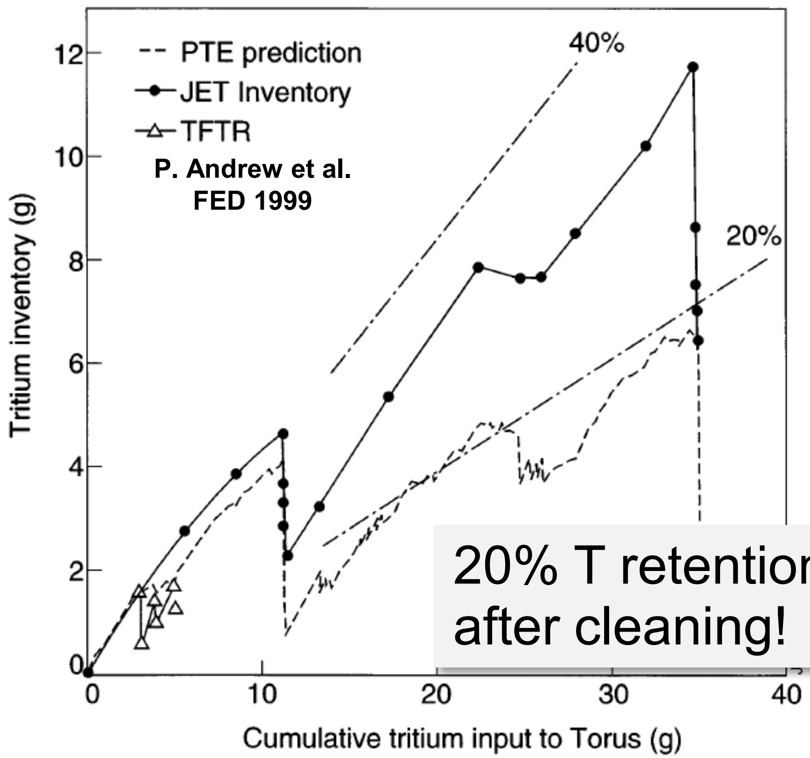
IAEA -Technical Meeting on Plasma Physics and Technology Aspects of the Tritium Fuel Cycle for Fusion Energy
11.10.-13.10. Wien

JÜLICH Forschungszentrum

Why not keep graphite plasma-facing components?

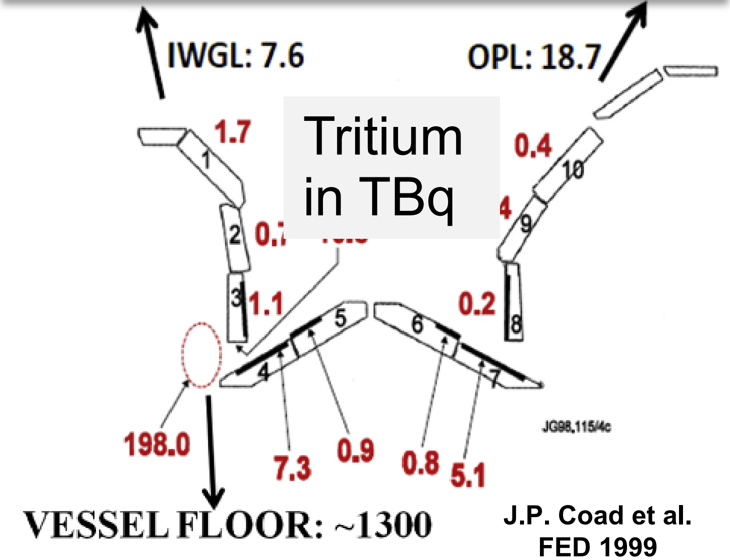
Experience from JET and TFTR operation with graphite walls

High tritium retention

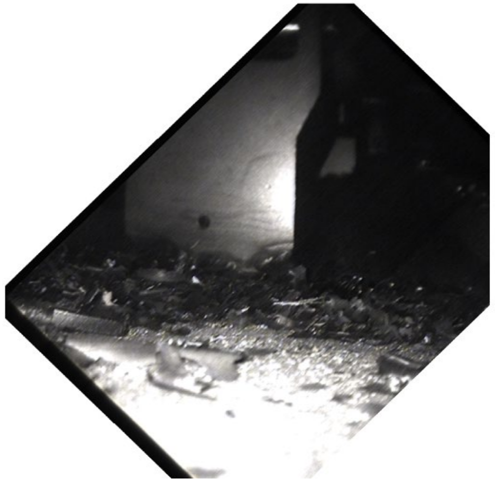


20% T retention after cleaning!

High T co-deposition with C
=> dust formation



Tritium
in TBq



Results of new JET DT, TT campaigns with low retention in presentation of Douai and Garcia

Challenging conditions for long-pulse steady-state operation (fusion reactor)
Main cause: chemical erosion of graphite (~1 nm/s => 10 cm/fpy)

Outline

Plasma-Surface Interaction

- *Low-Z and high-Z plasma-facing materials (PFM)*
- *Plasma-wall interaction processes (PWI)*
- *Processes leading to hydrogenic isotope (HI) retention*

Operation in JET and conclusions for ITER

- HI retention in JET with Be/W material mix
- Be material migration in JET with Be/W material mix
- ERO2.0 and WallDYN simulations of material migration and retention
- Outlook to HI retention and migration in ITER

Next step: nuclear fusion reactor at high duty cycle

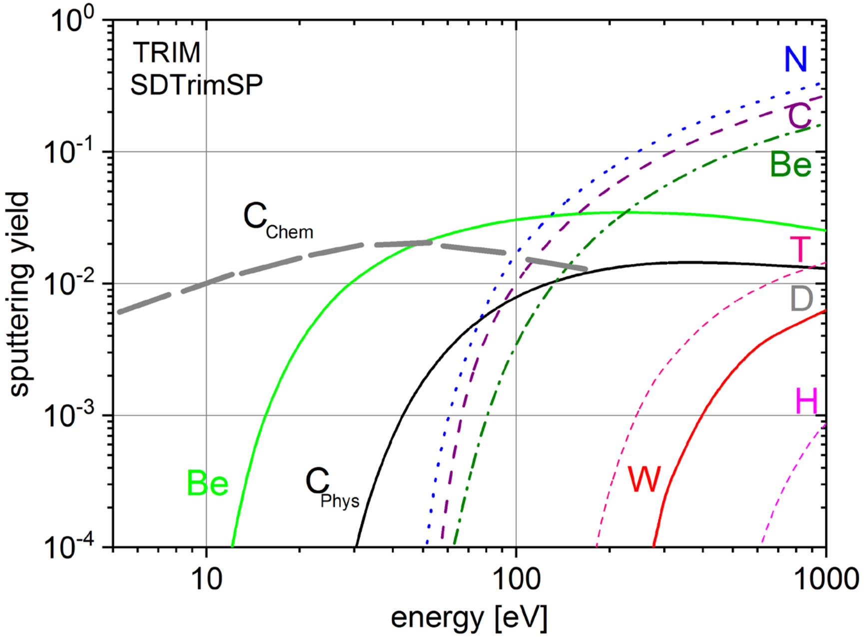
- A hypothetical full-W ITER
- Outlook to an European DEMO with full-W first wall

Conclusion

Selection of plasma-facing materials: low vs. high Z

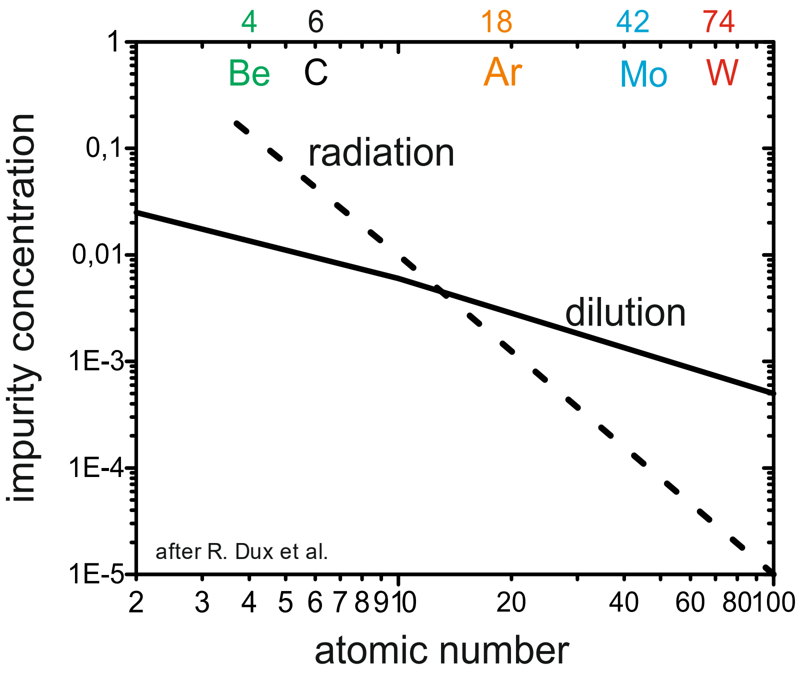
Steady-state plasmas operation for a DT plasmas in a reactor class device

Lifetime & safety



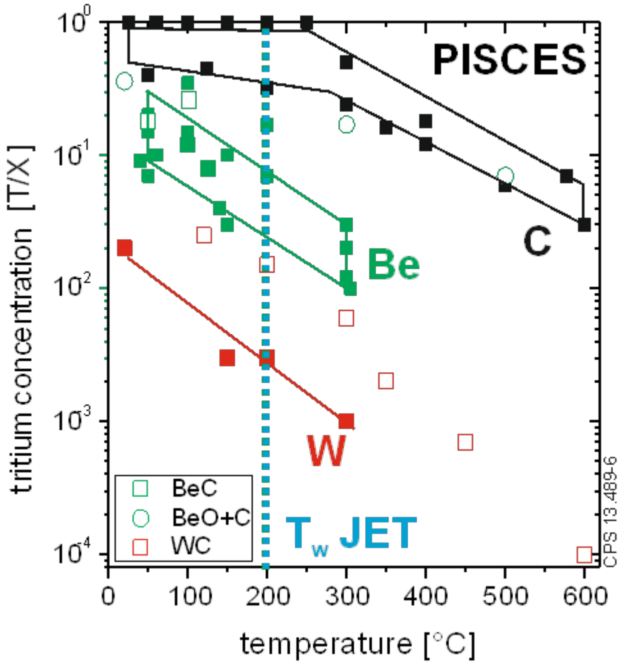
Low-Z: strong wall erosion;
for C due to chemical erosion
High-Z: low erosion;
for W mainly induced by impurities

Core plasma compatibility



Low-Z: limited by dilution
High-Z: limited by radiation

Tritium cycle & safety

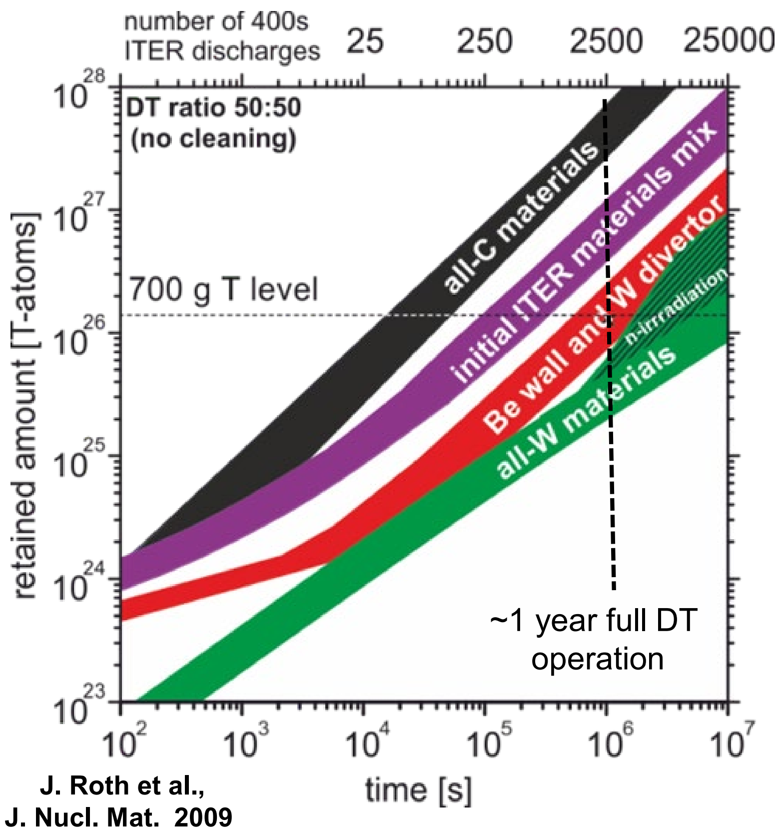


Main fuel retention process
for C and Be: co-deposition
W: implantation only
All: neutron damage effects

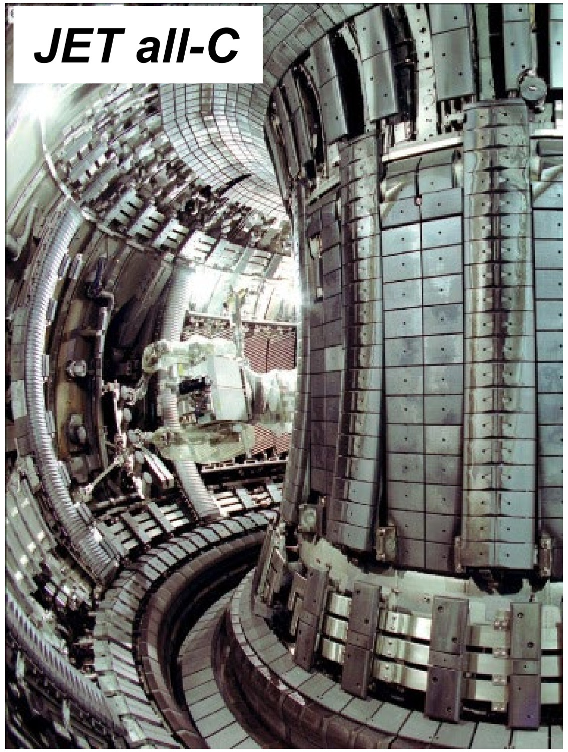
Single-step exchange of materials: JET from C to Be/W

Long-term fuel retention as main driver to change wall material

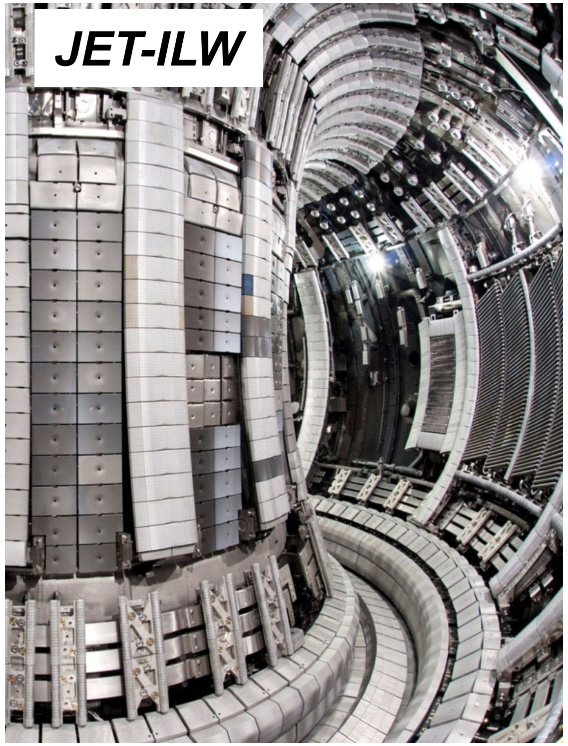
Fuel retention predictions for ITER by the EU Task Force PWI in 2009



J. Roth et al., J. Nucl. Mat. 2009



G. F. Matthews et al., Phys. Scr. 2011



S. Brezinsek, J. Nucl. Mater. 2015

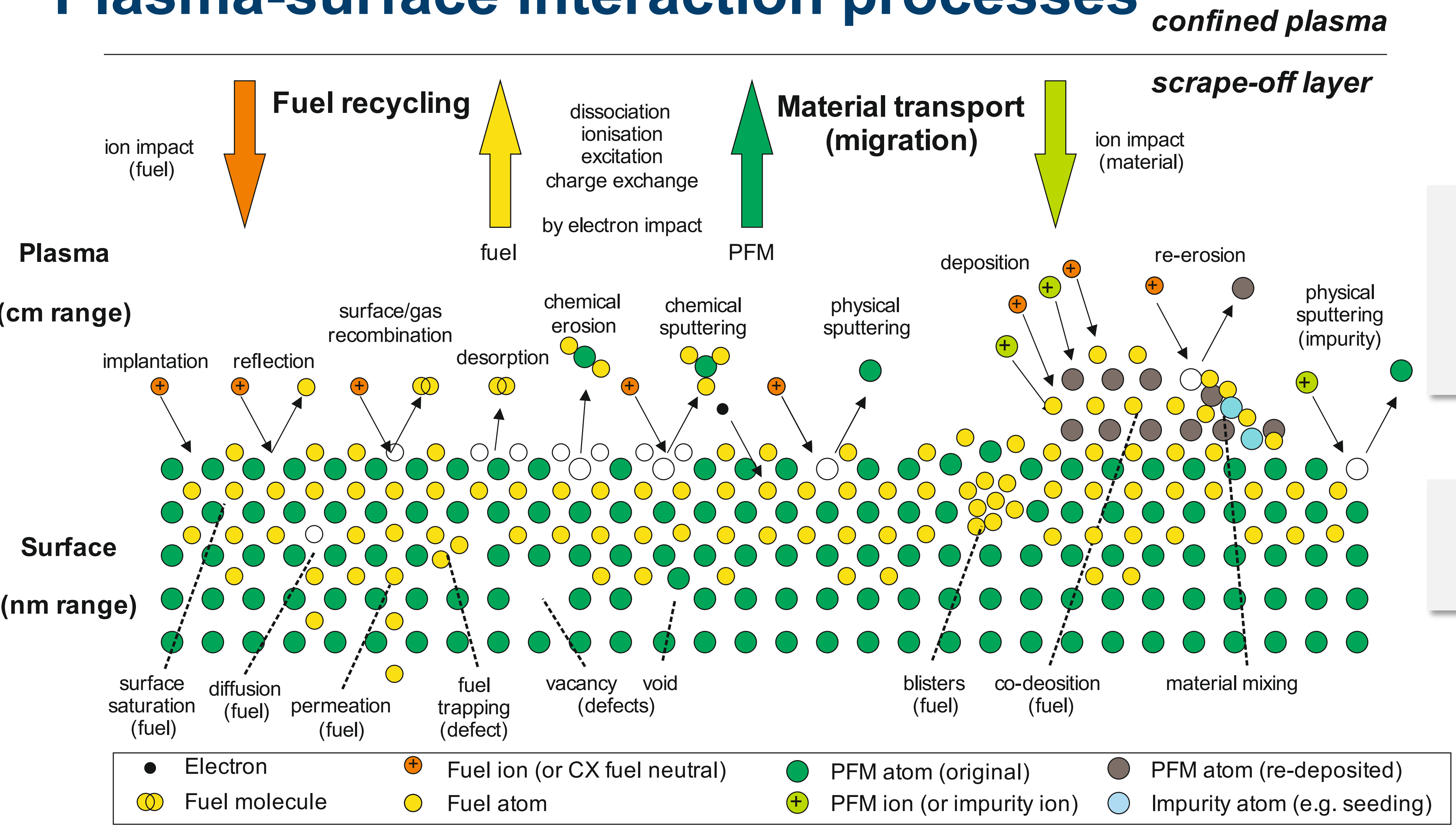
Safety:
- fuel retention
- dust

Lifetime:
- erosion
- melting

Operation:
- performance
- power exhaust

Predictions:
- modeling
- theory

Plasma-surface interaction processes



Divertor und SOL plasma conditions:

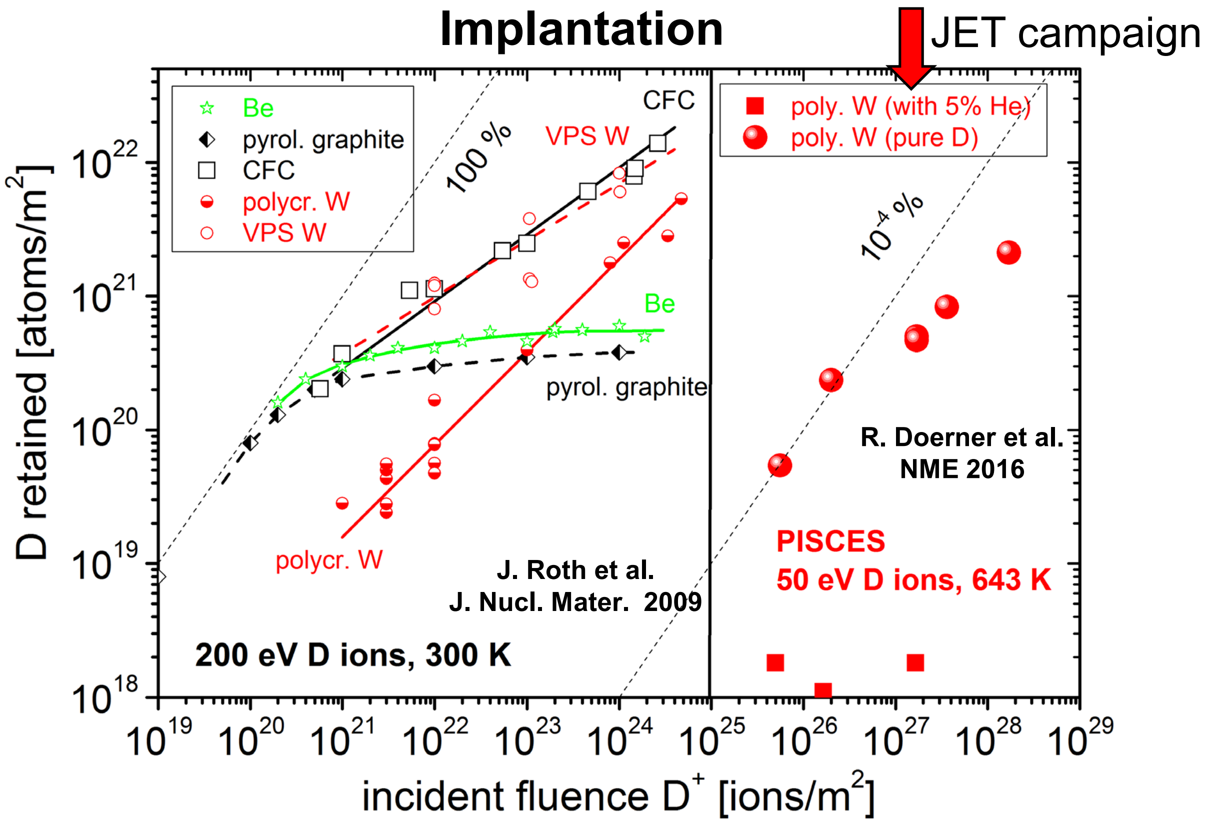
T_e=1-100 eV
$n_e=5x10^{17}-1x10^{21}m^{-3}$

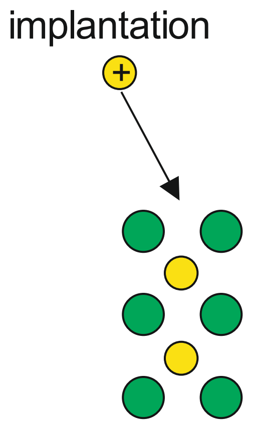
Impact energies:
E_{in}=1-500 eV HI ions
E_{in}=1-2000 eV CX HI

HI = Hydrogen Isotopes

Fuel retention mechanisms: implantation

Safety and fuel cycle: long-term fuel retention





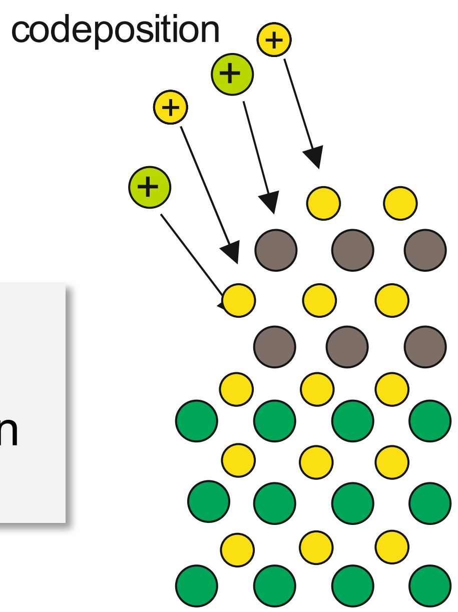
Components:

- Permeation occurs
- Temperature critical

- Material samples naturally outgassed after extraction from exposure chamber
- Retention measured ex-situ by thermal desorption spectrometry

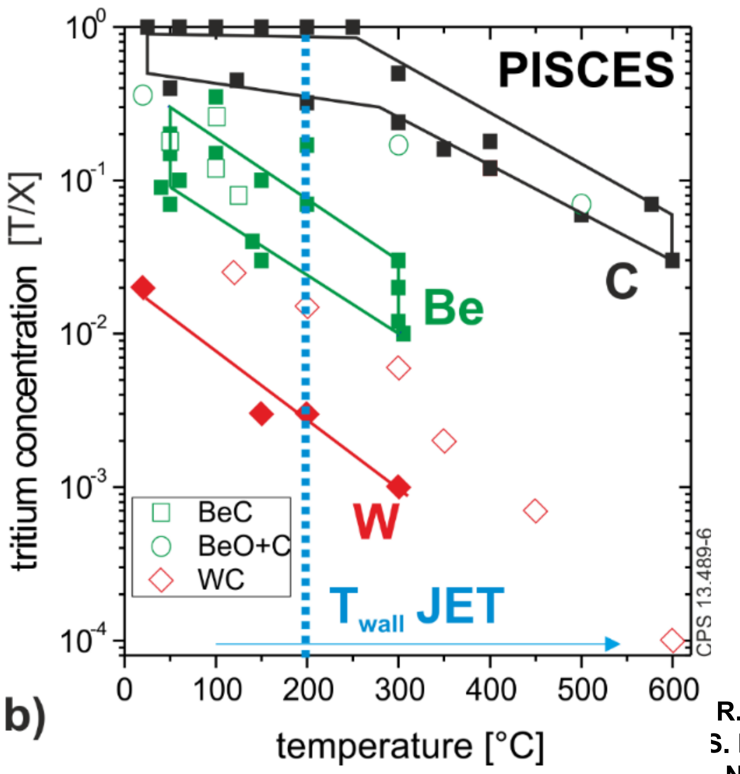
Fuel retention mechanisms: co-deposition

Safety and fuel cycle: long-term fuel retention



Components:
- Continuous layer growth
- Instability => dust formation
- Temperature critical

Co-deposition



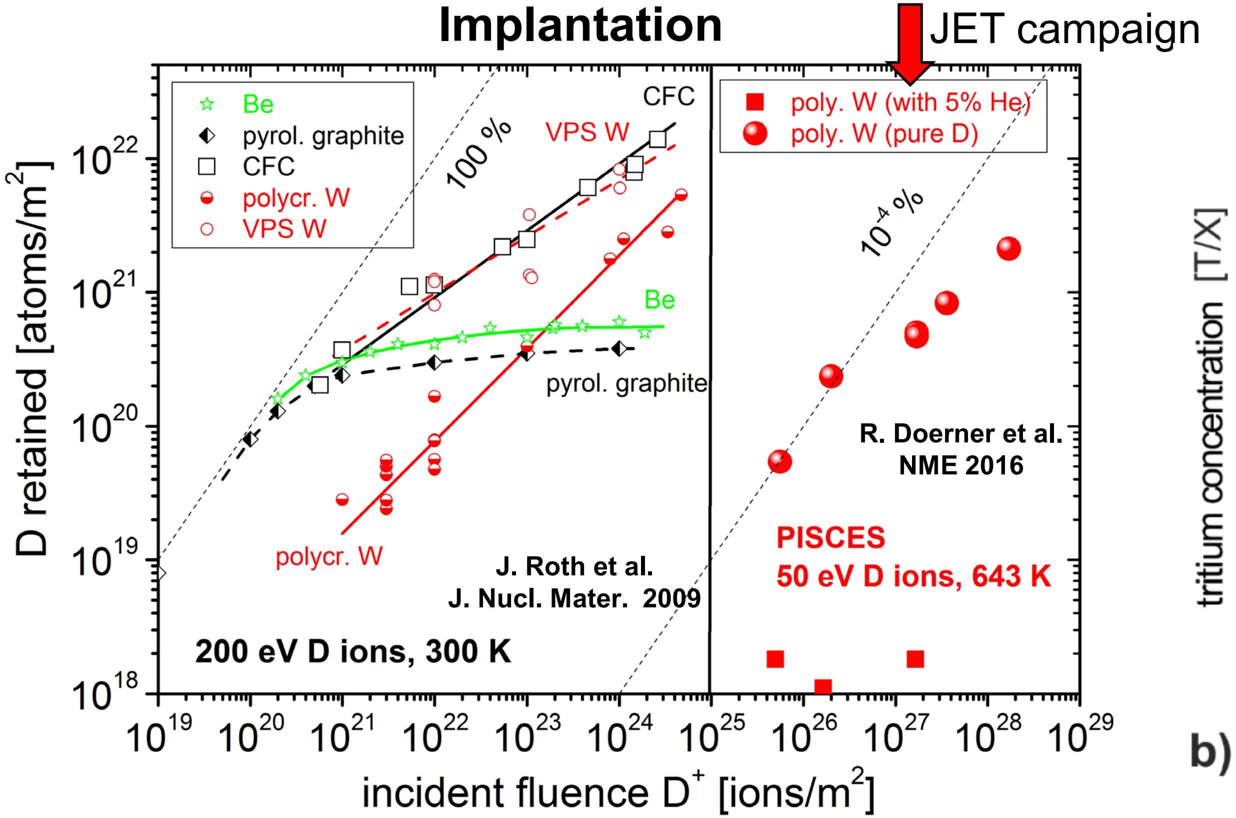
R. Doerner et al., S. Brezinsek et al. Nucl. Fus. 2013

- Material samples naturally outgassed after extraction from exposure chamber
- Retention measured ex-situ by thermal desorption spectrometry

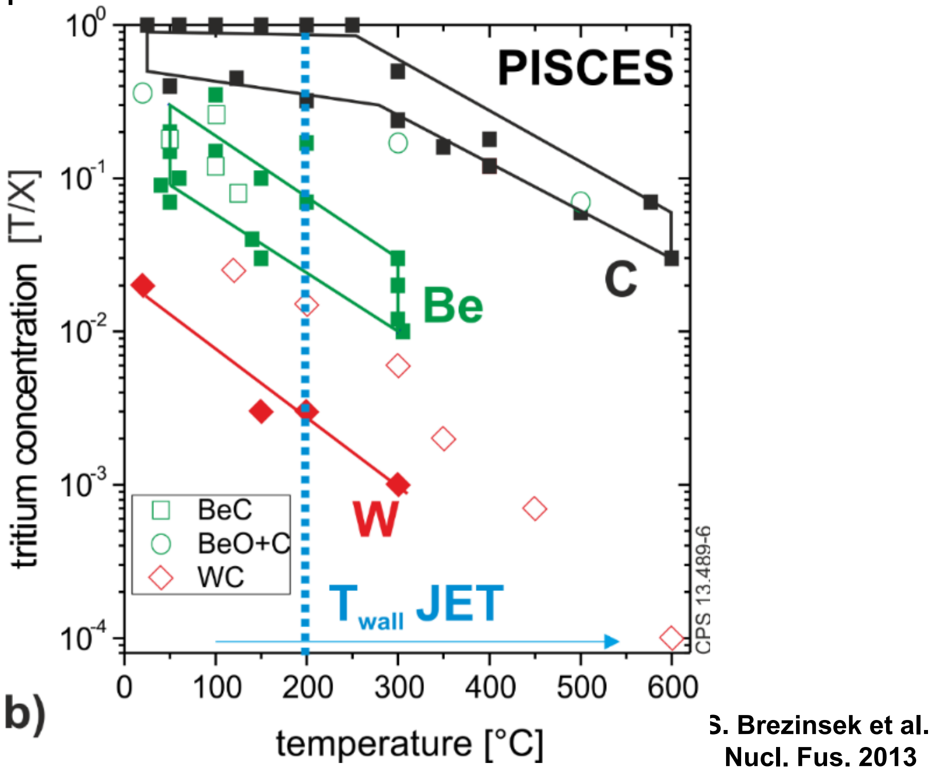
Fuel retention mechanisms

Safety and fuel cycle: long-term fuel retention

Implantation



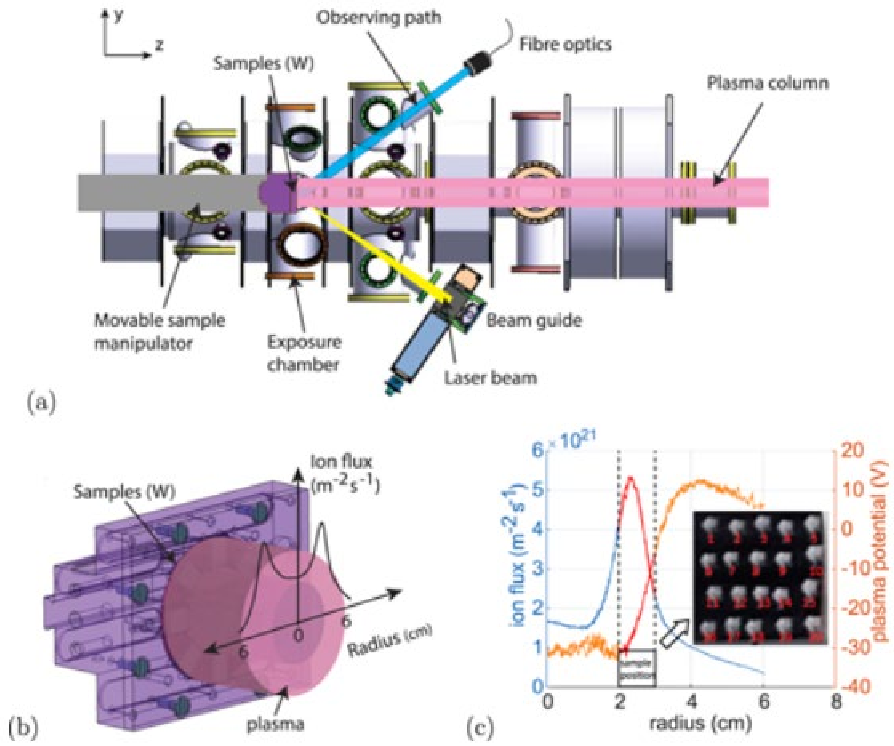
Co-deposition

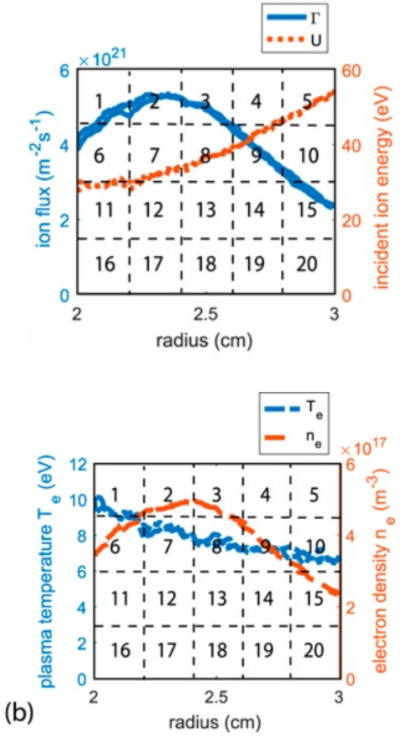


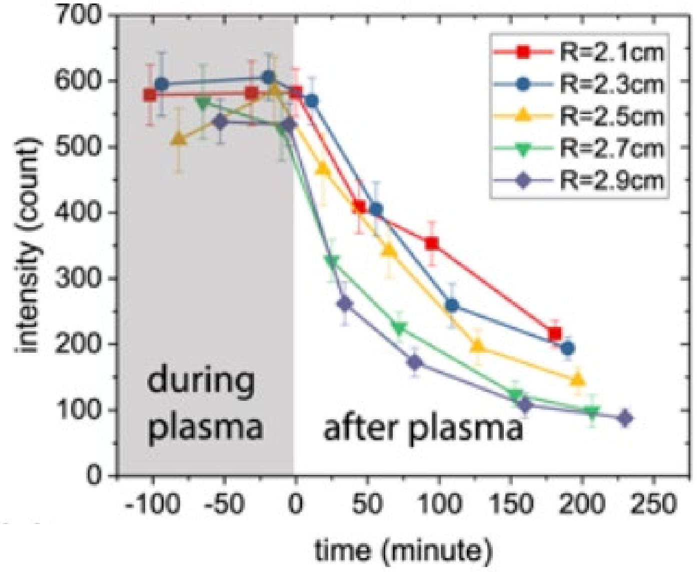
- Co-deposition strong with Be and C, but two orders of magnitude lower with W
- Fuel retention by implantation is orders of magnitude lower than by co-deposition
- Large difference in implantation between bulk W and W-coating

Short-term fuel retention

(Over-)saturation of surfaces: example W







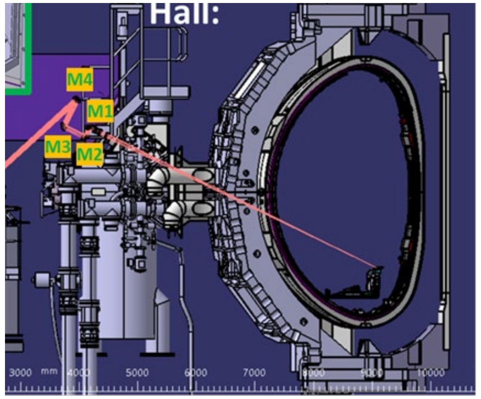
X. Jiang et al.
Nucl. Fus. 2021

- LIBS-probing of D content in W during and post plasma exposure in PSI-2
- During plasma operation reflects recycling flux at W surface
- Post plasma operation reflects near surface retention as function of T_e, n_e, ion flux, and T_{surf}
- Good agreement in D retention between TDS and LIBS after days of outgassing

Fuel retention measurement techniques

LID-QMS or LIBS

- Local measurement
- Inter- or intra-shot
- Short- or long-term retention

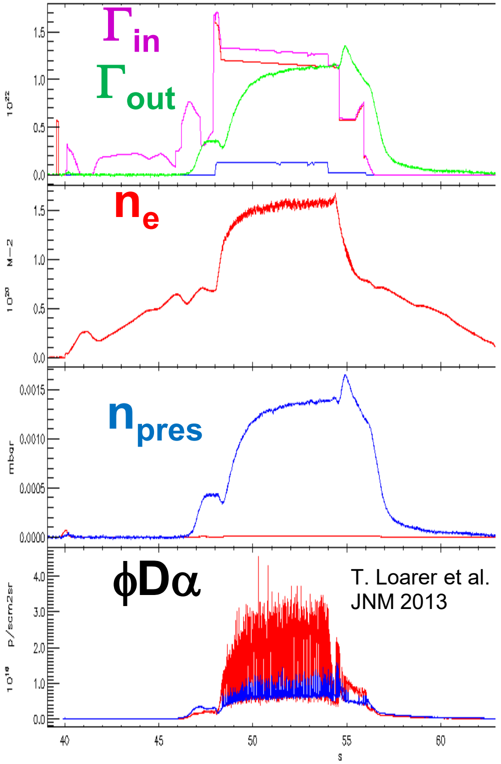


M. Zlobinski et al.

Local D content (to control or to validate modelling)

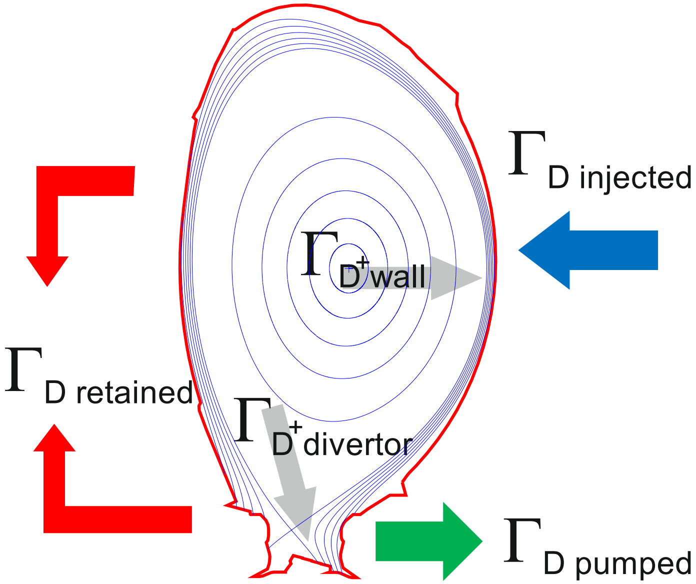
Intershot gas balance

- Single discharge only
- Transient effect
- Short-term retention



Global gas balance

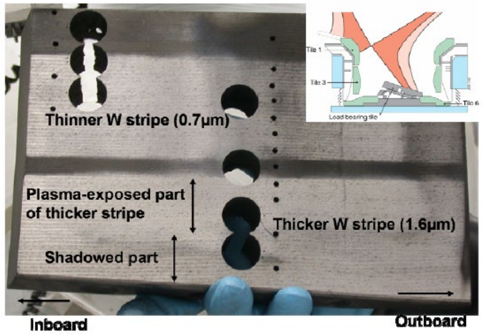
- Multiple identical discharges
- Includes inter-shot outgassing
- Long-term retention (1/2 day)



Upper limit of D remaining in the vessel

Post mortem analysis

- All campaign discharges
- Includes outgassing in campaign / intervention
- Long-term retention (1year)



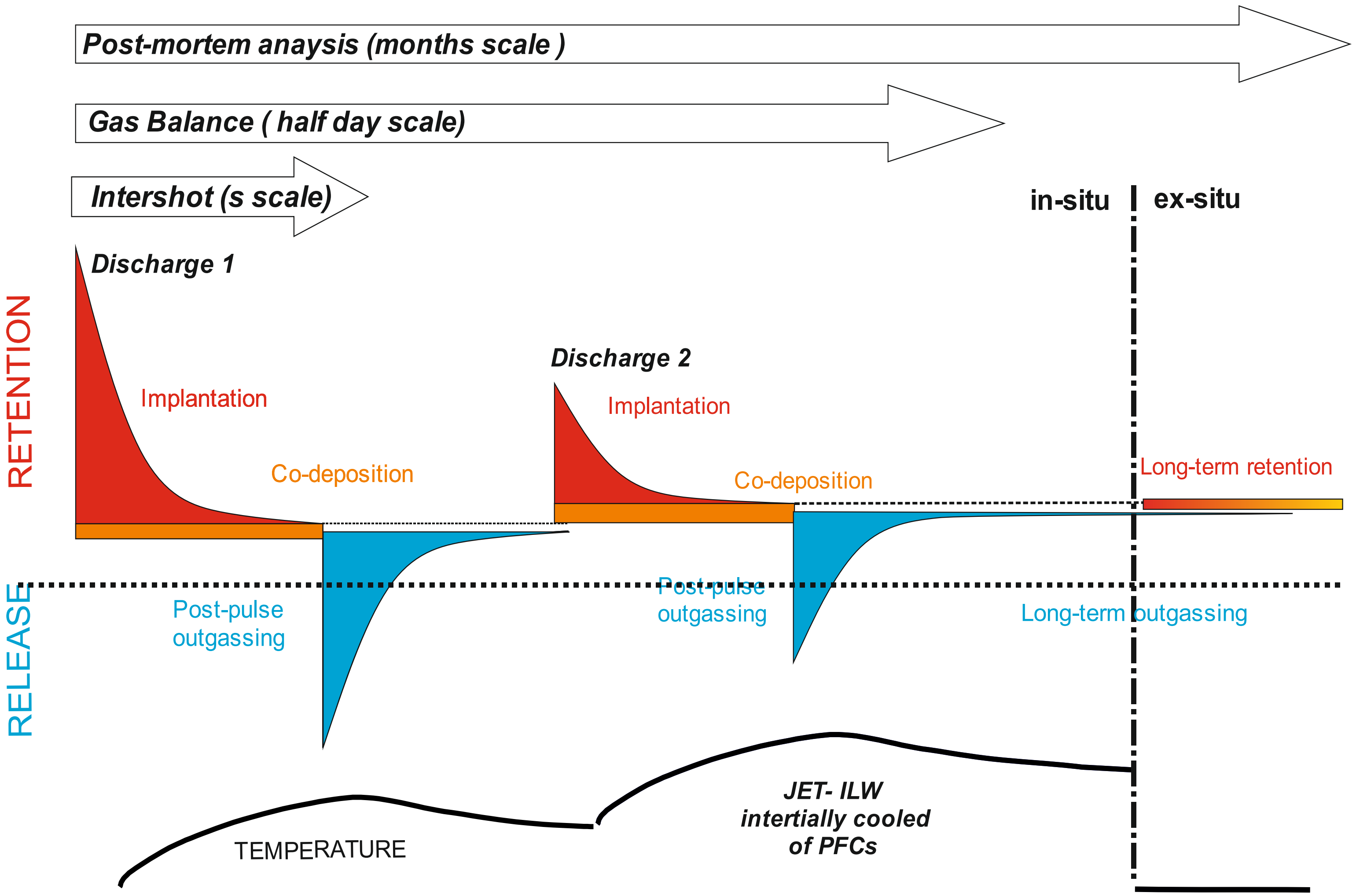
P. Coad et al.
Phys. Scripta 2012

Permanent-stored D In the vessel

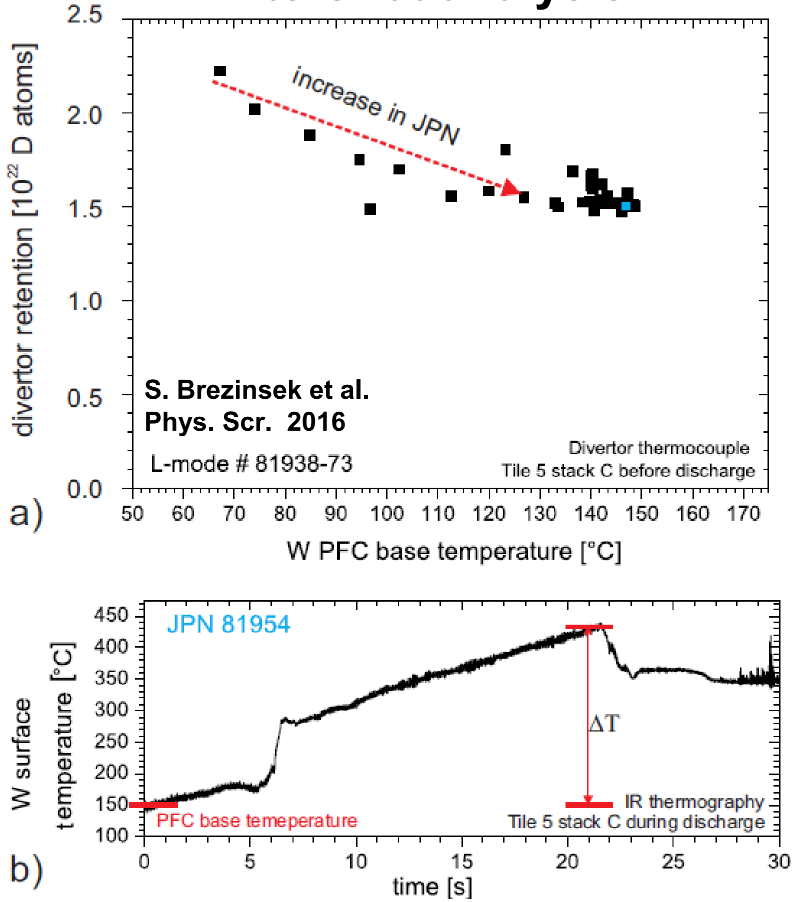
In T plasmas: T accounting with whole pipe-works and processing chain (not only vessel)!

Dynamic cycle: retention vs. outgassing in JET

Short-term retention, long-term retention and outgassing



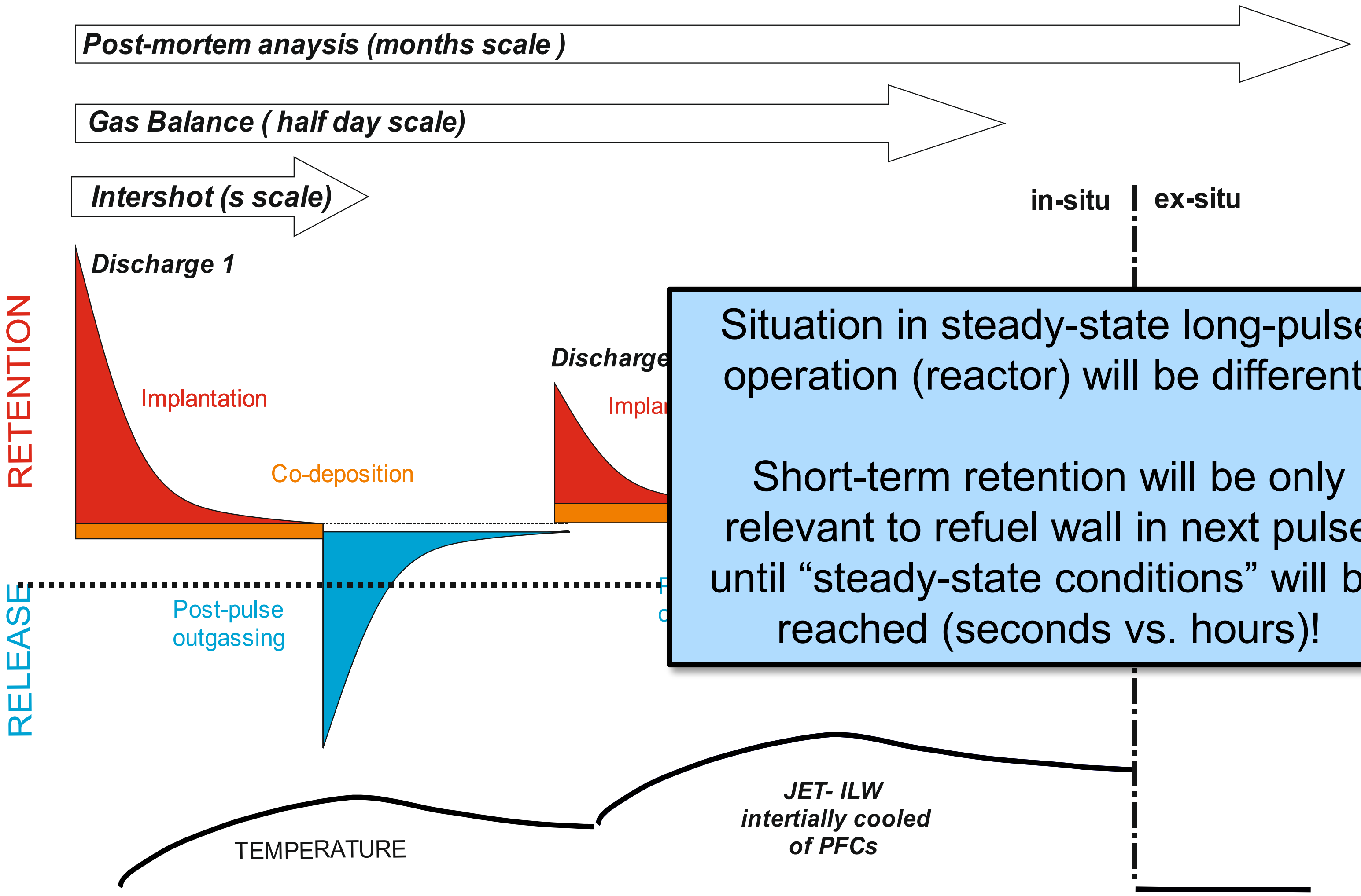
intershot analysis



- Required fuel of plasma depends on PFC temperature conditions
- Inertially cooled components in JET

Dynamic cycle: retention vs. outgassing in JET

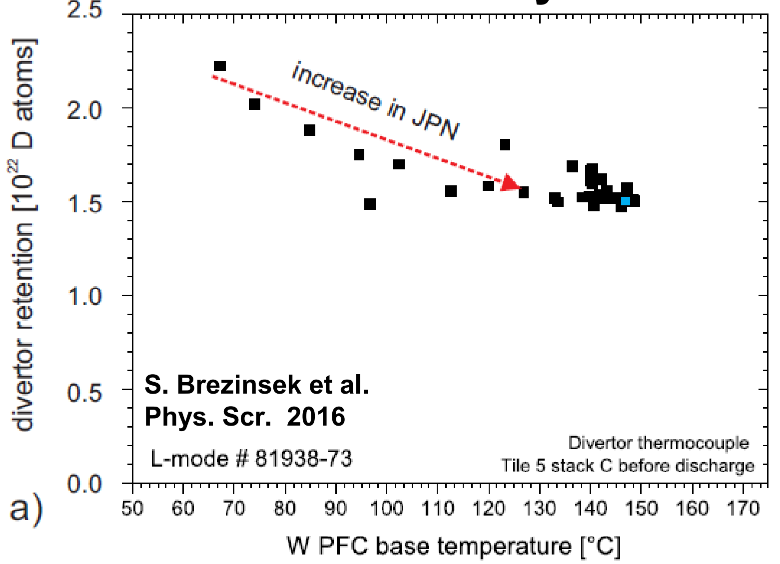
Short-term retention, long-term retention and outgassing

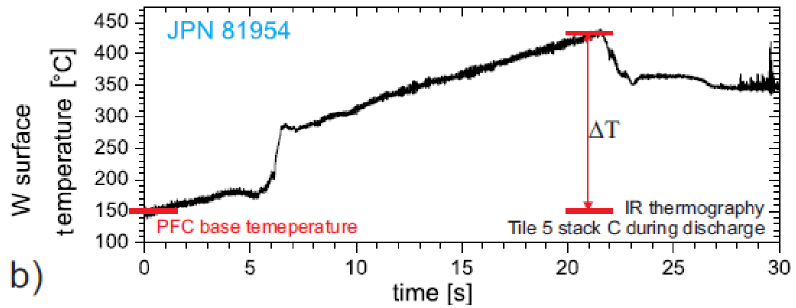


Situation in steady-state long-pulse operation (reactor) will be different!

Short-term retention will be only relevant to refuel wall in next pulse until "steady-state conditions" will be reached (seconds vs. hours)!

intershot analysis





- Required fuel of plasma depends on PFC temperature conditions
- Inertially cooled components in JET

Outline

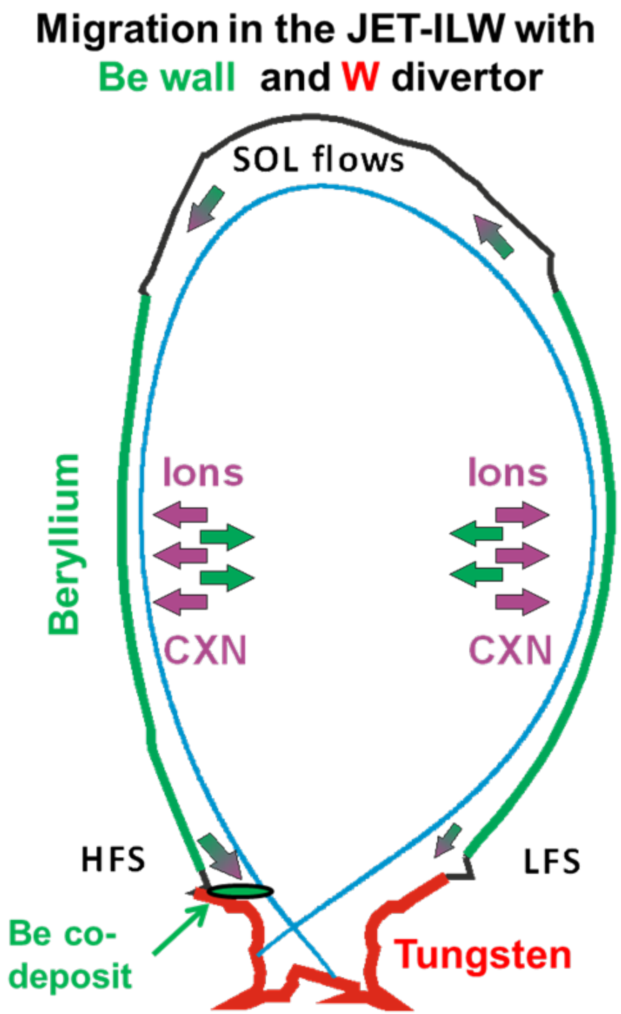
Plasma-Surface Interaction
- Low-Z and high-Z plasma-facing materials (PFM)
- Plasma-wall interaction processes (PWI)
- Processes leading to hydrogenic isotope (HI) retention

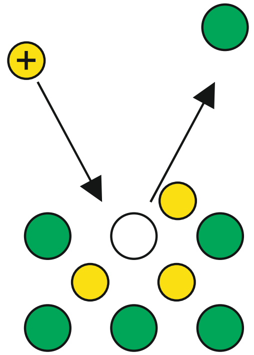
Operation in JET and conclusions for ITER
- *HI retention in JET with Be/W material mix*
- *Be material migration in JET with Be/W material mix*
- *ERO2.0 and WallDYN simulations of material migration and retention*
- *Outlook to HI retention and migration in ITER*

Next step: nuclear fusion reactor at high duty cycle
- A hypothetical full-W ITER
- Outlook to an European DEMO with full-W first wall

Conclusion

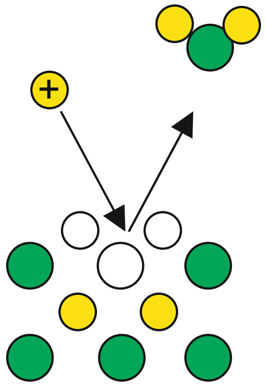
Beryllium erosion processes in JET

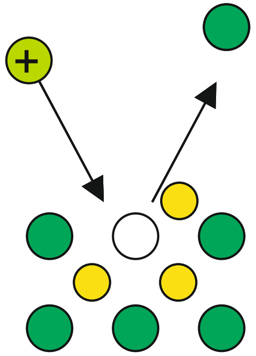




Physical sputtering of beryllium by D^+ (or impurity ions)

Chemically assisted physical sputtering of beryllium by D^+ (or impurity ions)



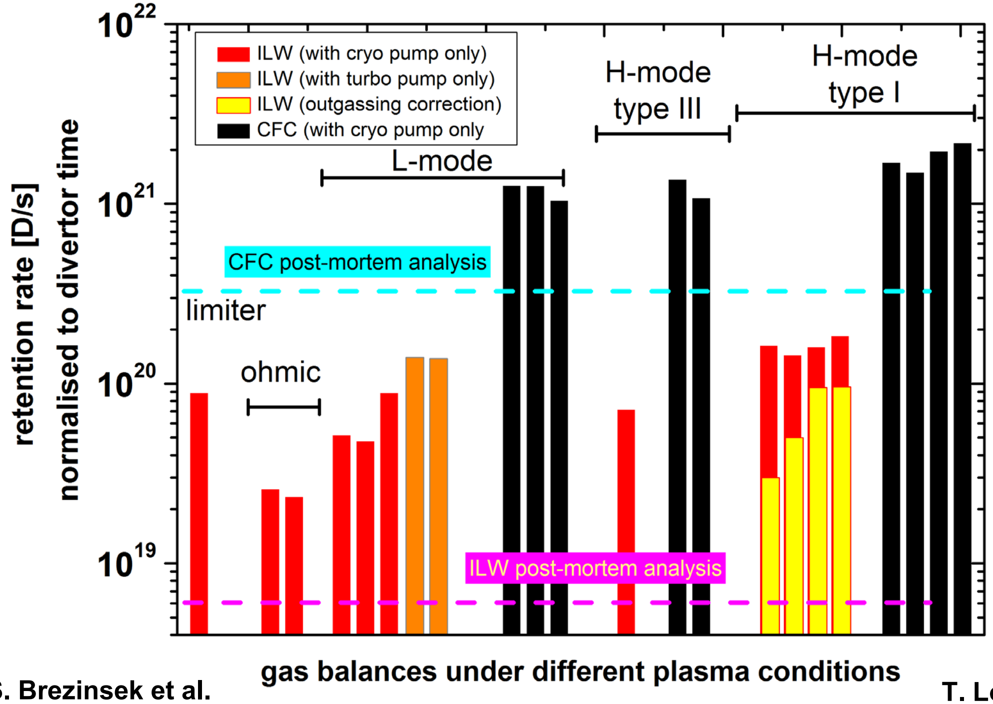


Self-sputtering of beryllium by Be ions

No chemical erosion like in C as energy threshold exist!

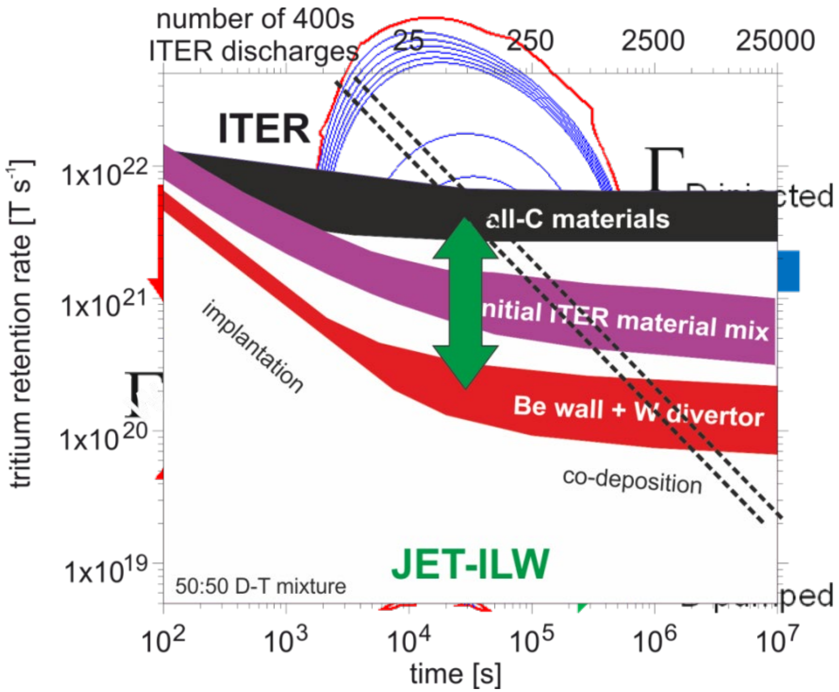
Long-term fuel retention in JET and extrapolation to ITER

JET demonstrates low deuterium retention in the Be/W material mix



S. Brezinsek et al. Nucl. Fus. 2013

T. Loarer et al. J. Nucl. Mater. 2009



S. Brezinsek Nucl. Fusion 2013

- In operando gas balance demonstrates a factor 15-20 reduction with strong post-plasma wall outgassing
- Retention drop as expected: retention by co-deposition (2/3) dominates over implantation (1/3)
- Reasons for the reduction from JET-C to JET-ILW:
  - Be primary source and Be transport to divertor in JET-ILW smaller than C in JET-C
  - Lower fuel content in pure Be co-deposits in comparison with C co-deposits

Fuel content in divertor – post plasma outgassing

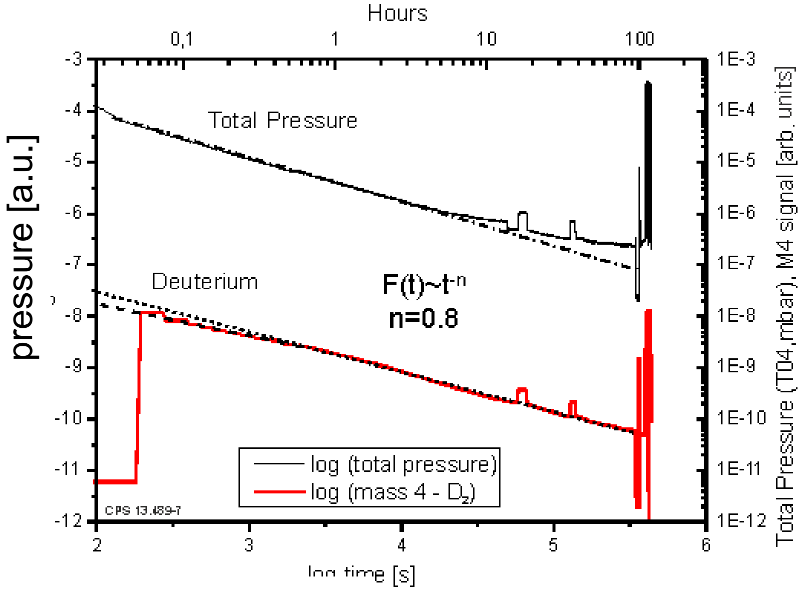
Short-term retention, long-term retention and outgassing

Co-deposition fully determined by D with Be, C, O

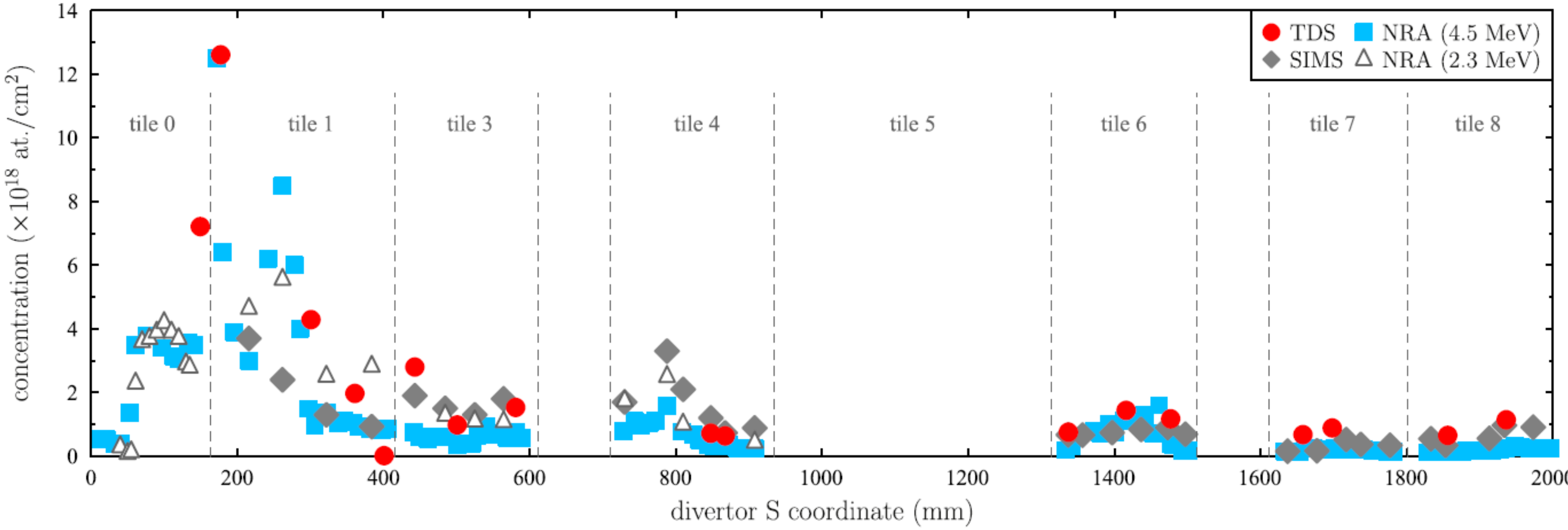
Implantation in Be PFCs and clean W surfaces

Implantation in W-coated CFC: ~1x10^{22} D/m^2
Implantation in bulk W:~5x10^{20} D/m^2

K. Heinola et al. Nucl. Fus. 2015



V. Philipps et al. PFMC 2013



- 50% of D are on top of tile 0 and 1 (co-deposited with Be)
- 15% of D implanted and co-deposited with impurities tile 3-4 and 6-8
- 35 % of D in main chamber due to implantation and limiter operation

Fuel content in divertor – post plasma outgassing

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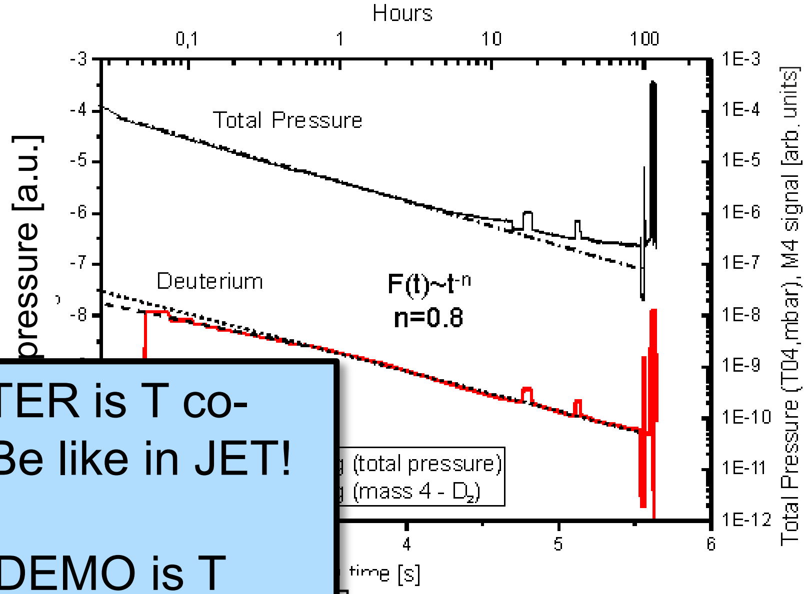
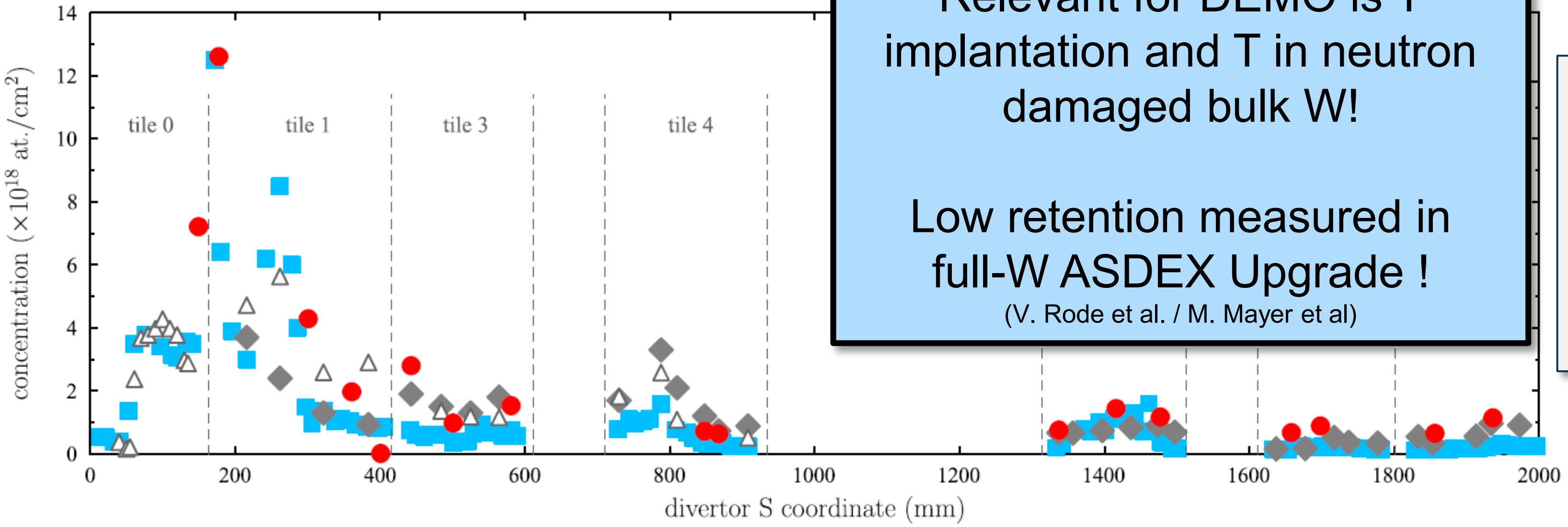


V. Philipps et al.
PFMC 2013

Relevant for ITER is T co-deposition with Be like in JET!

Relevant for DEMO is T implantation and T in neutron damaged bulk W!

Low retention measured in full-W ASDEX Upgrade !
(V. Rode et al. / M. Mayer et al)



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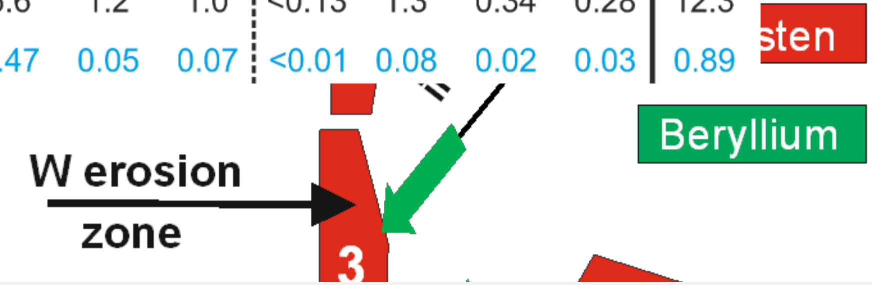
- 15% of D implanted and co-deposited with impurities tile 3-4 and 6-8
- 35 % of D in main chamber due to implantation and limiter operation

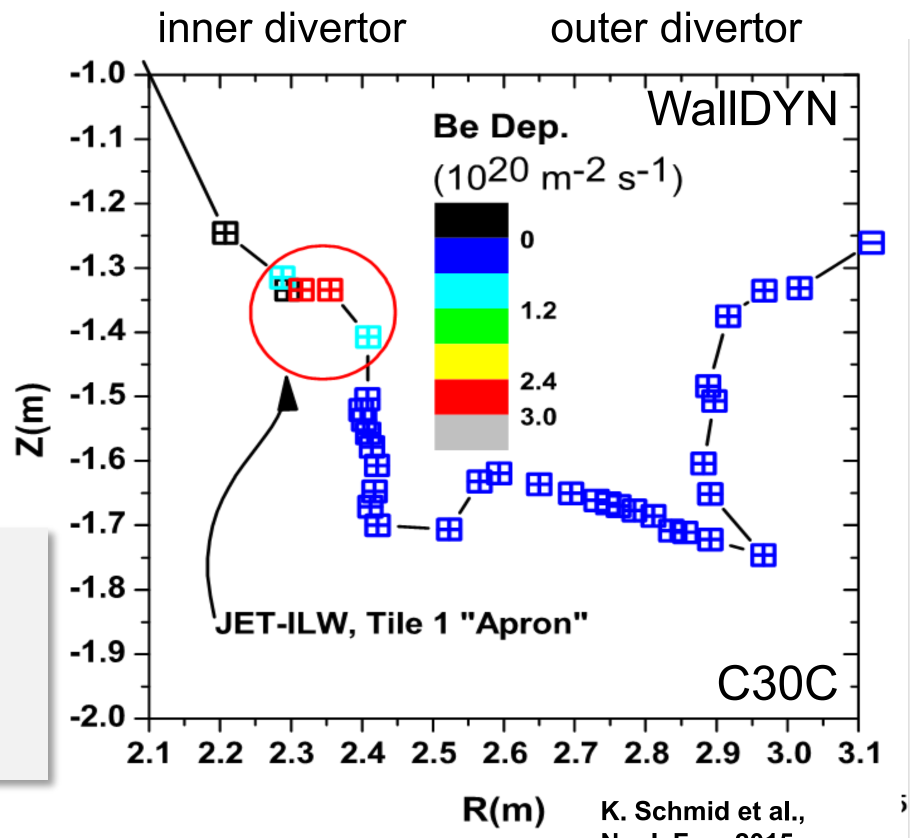
Long-term D retention and Be deposition in JET

Be migration within the divertor: low transport to remote areas in V5

	inner divertor				outer divertor				
	0	1	3	4	5*	6	7	8	SUM
Be [g]	21.7	27.1	0.91	0.45	<0.08	1.5	0.18	0.10	52.0
C [g]	1.4	6.6	1.2	1.0	<0.13	1.3	0.34	0.28	12.3
D [g]	0.16	0.47	0.05	0.07	<0.01	0.08	0.02	0.03	0.89

M. Mayer et al., Phys. Scr. 2016





K. Schmid et al., Nucl. Fus. 2015

- Strong Be deposition on top of inner W divertor (NRA, RBS)
- No re-erosion at lowest impact energies with Be and reduced stepwise transport (contrast to chemical erosion of graphite)
- De co-deposition with Be measured ex-situ (TDS, NRA)

A. Kirschner et al., J. Nucl. Mater. 2015

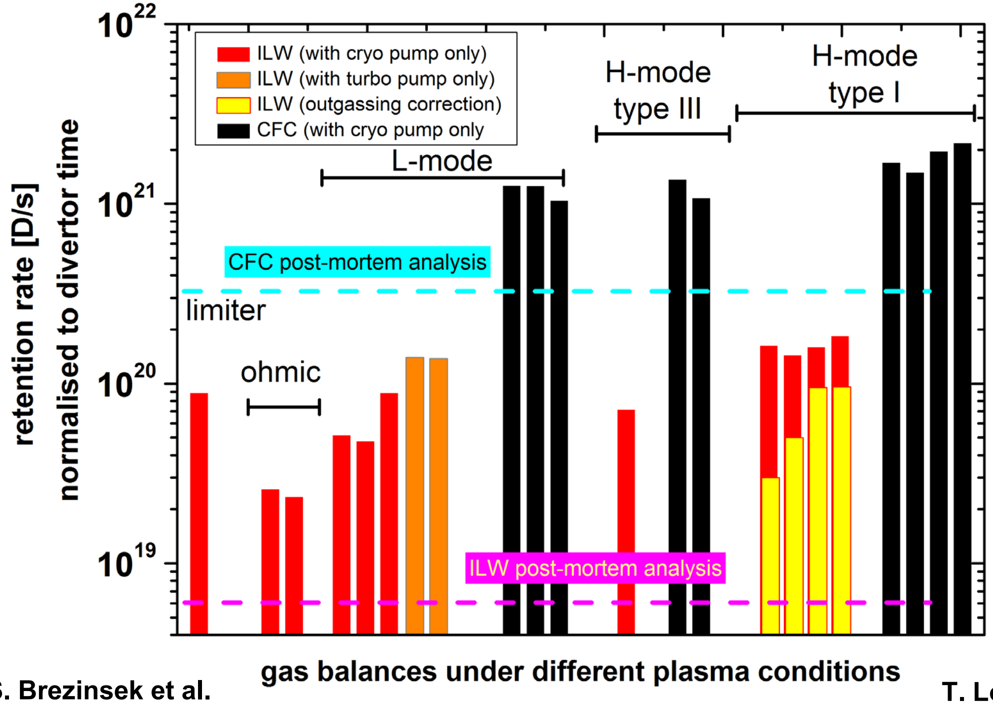
G. Sergienko et al., NME 2017

K. Heinola et al., Phys. Scr. 2016

A. Widdowson et al., NME 2019

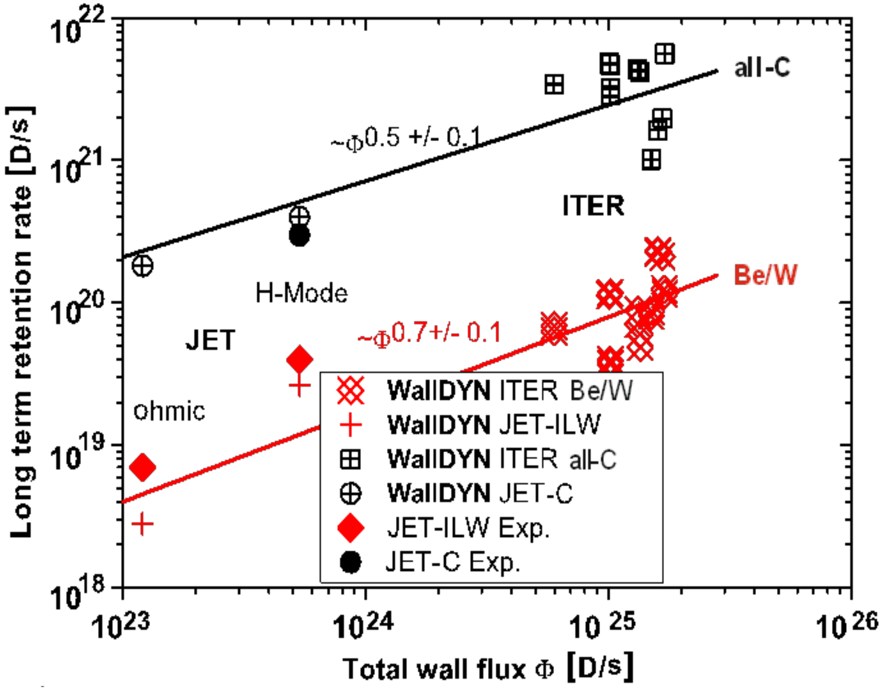
Long-term fuel retention in JET and extrapolation to ITER

JET demonstrates low deuterium retention in the Be/W material



S. Brezinsek et al. Nucl. Fus. 2013

T. Loarer et al. J. Nucl. Mater. 2009



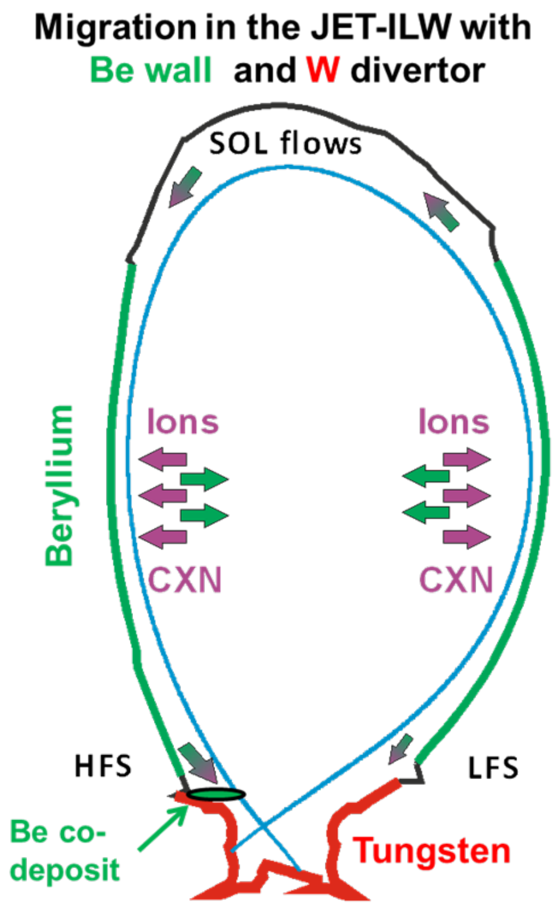
K. Schmid et al.. J. Nucl. Mater. 2015

- In operando gas balance demonstrates a factor 15-20 reduction with strong post-plasma wall outgassing
- Deposition pattern and absolute value of deuterium retention reproduced with WallDYN simulations
- Initial 2D WallDYN simulations predict ITER T inventory limit within 3000 – 20 000 discharges without cleaning

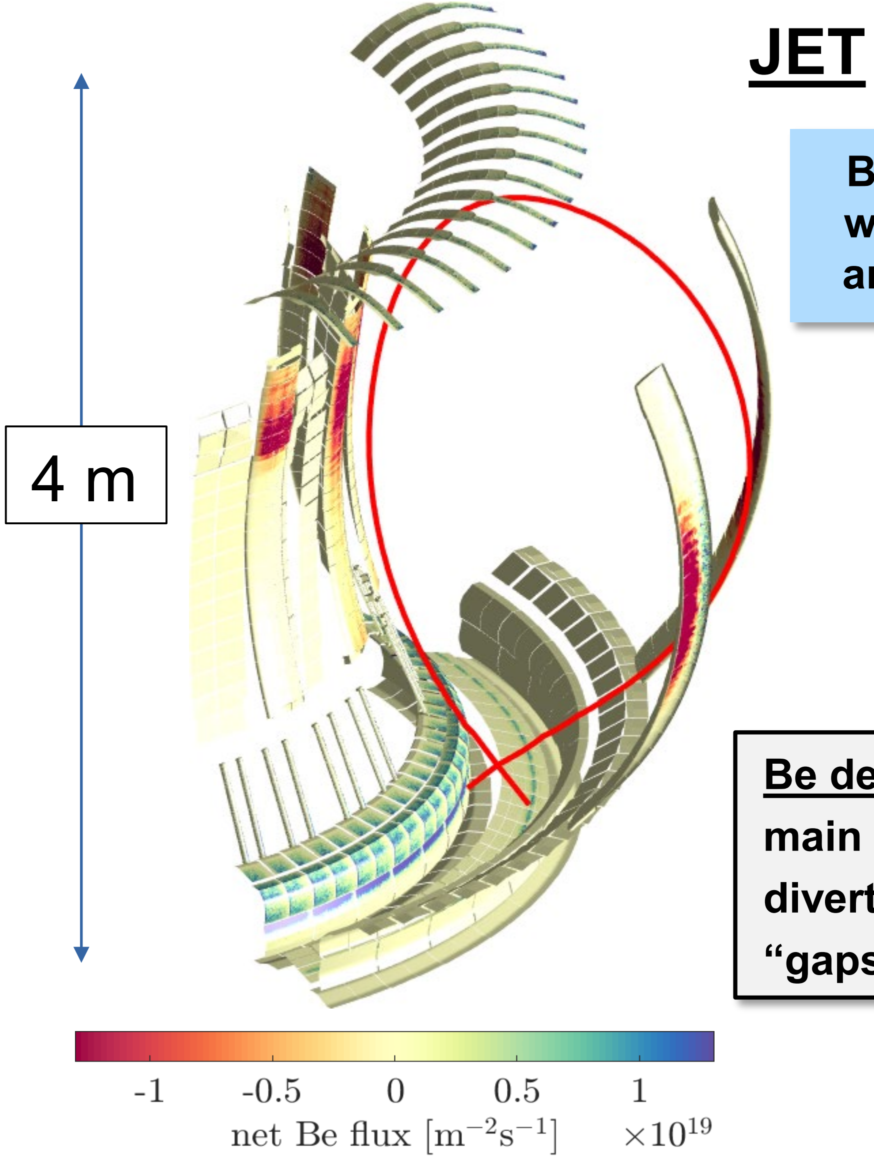
Material migration in JET deuterium plasmas

Common between JET-C and JET-ILW:
- sputtering at recessed wall by low energetic ions (E_{in}<10eV) and by charge-exchange neutrals
- transport of material due to scrape-off layer flows preferred into the inner divertor
- outer divertor: erosion zone at strike line

Difference between JET-C and JET-ILW:
- absence of chemical erosion in the Be case
- factor 5 lower primary source with Be
- factor 7-10 lower migration into divertor with Be
- factor 100 less low-Z dust (in remote areas) with Be



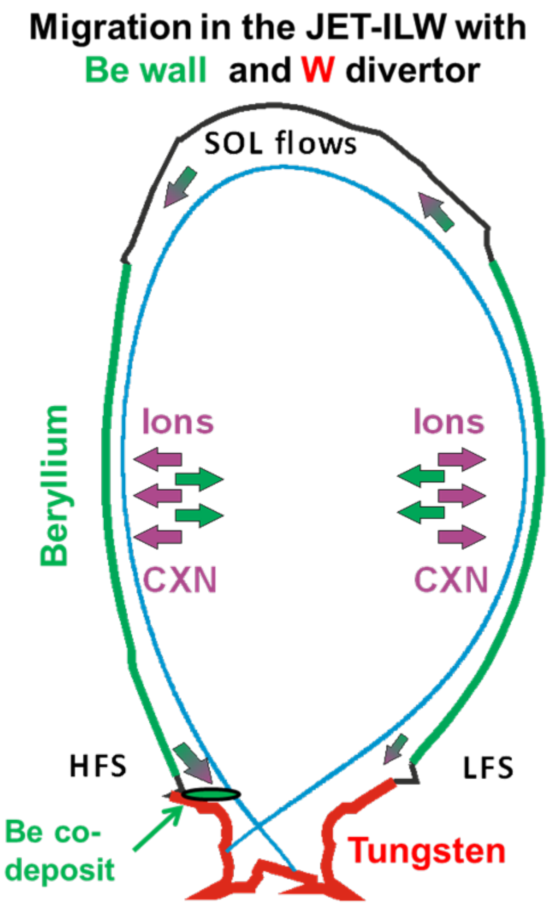
Interpretation of Be migration in JET-ILW with ERO2.0 (3D)



Balance of first wall Be erosion and deposition!

Be deposition:
main chamber: 18%
divertor: 58%
"gaps": 24%

J. Romazanov et al. NME2021

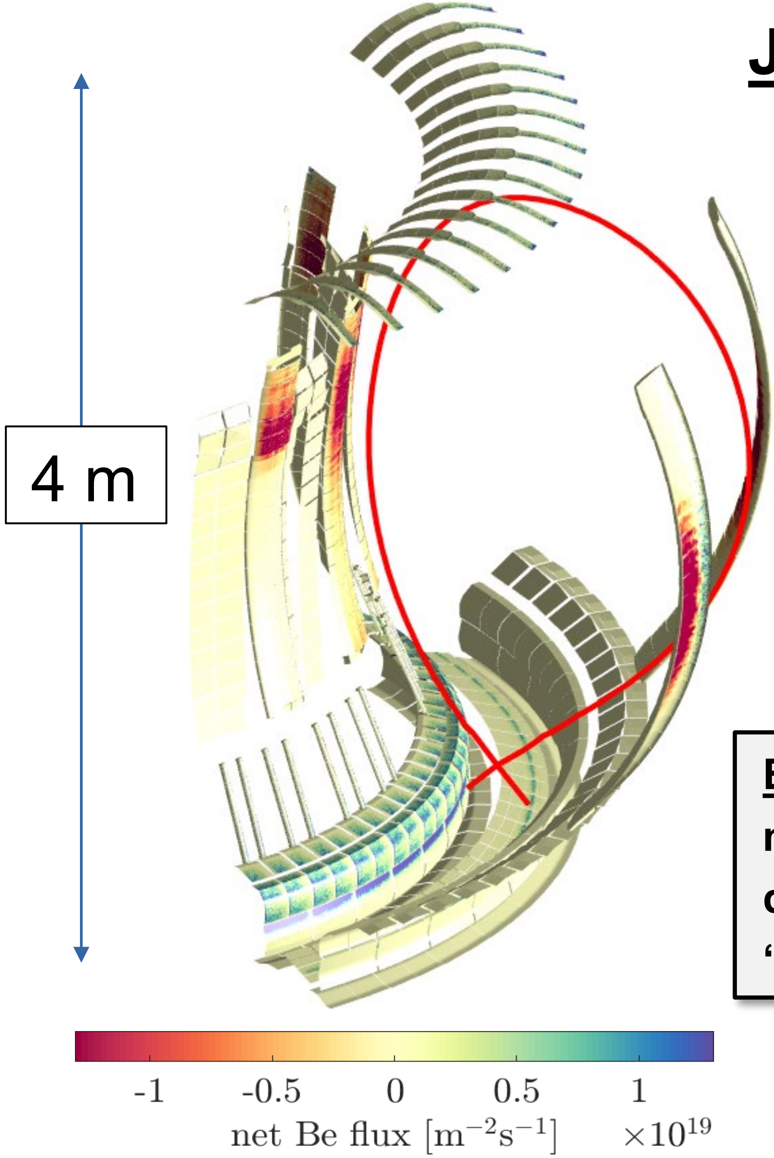


- ERO2.0 reproduces global erosion/deposition pattern
- Provides in 3D highest deposition/retention location
- Residual HI ion and CX fluxes important

Comparison of JET and ITER with ERO

Balance of first wall Be erosion and deposition!

JET



Dominant Be deposition on inner divertor apron

Be deposition:

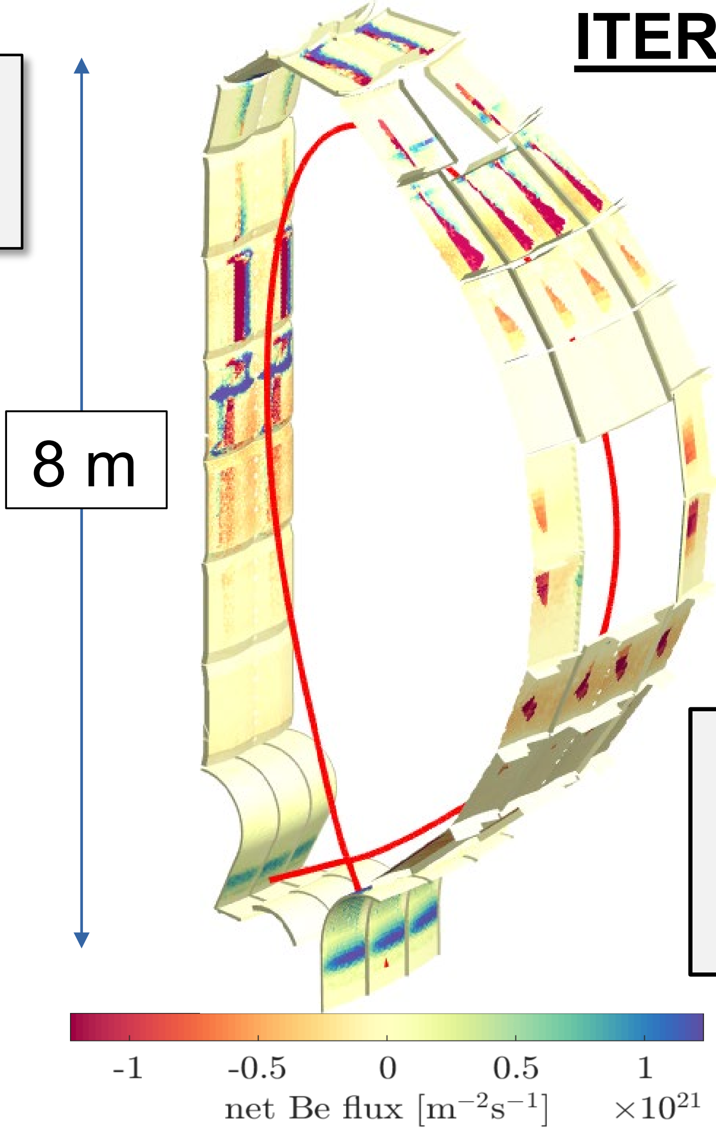
main chamber: 18%

divertor: 58%

“gaps”: 24%

ITER

H mode with zero SOL flow



Dominant Be re-deposition at main chamber wall

Be deposition:

main chamber: 90%

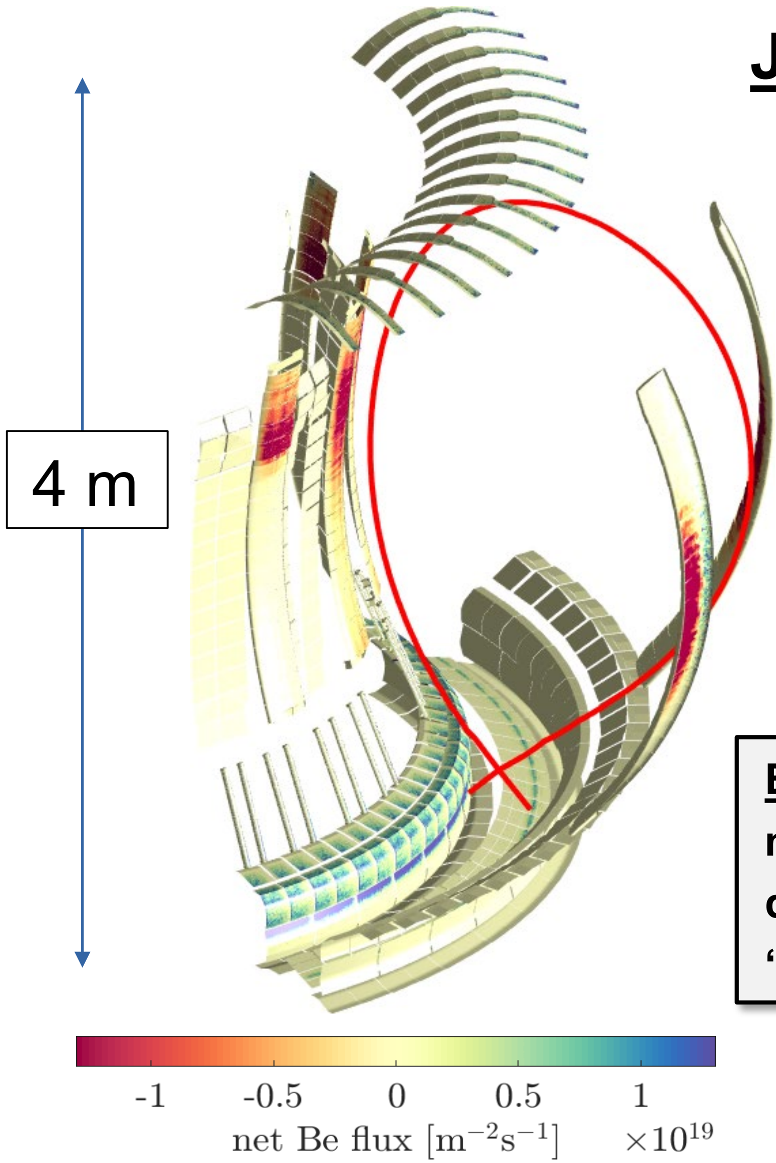
divertor: 10%

“gaps”: <1%

J. Romazanov et al.
NME2021, NF2022

Comparison of JET and ITER with ERO

JET



Balance of first wall Be erosion and deposition!

Domin
depos
diverto

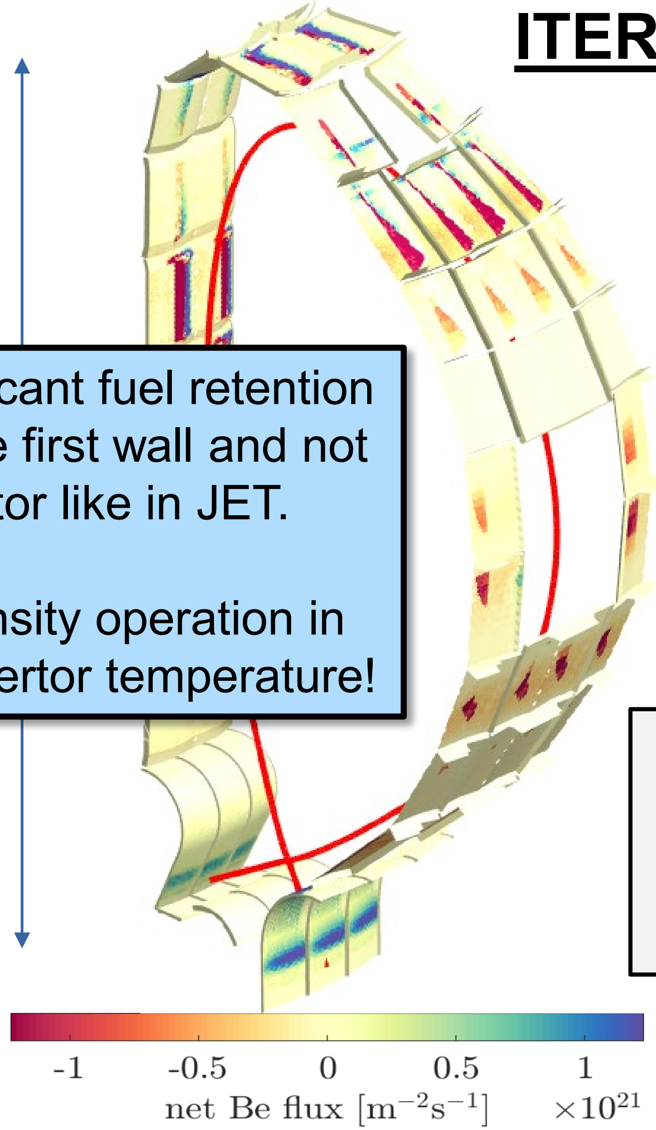
Note: in ITER significant fuel retention will also occur at the first wall and not only in the divertor like in JET.

Reason is high density operation in SOL and high W divertor temperature!

Be deposition:
main chamber: 18%
divertor: 58%
“gaps”: 24%

ITER

H mode with zero SOL flow



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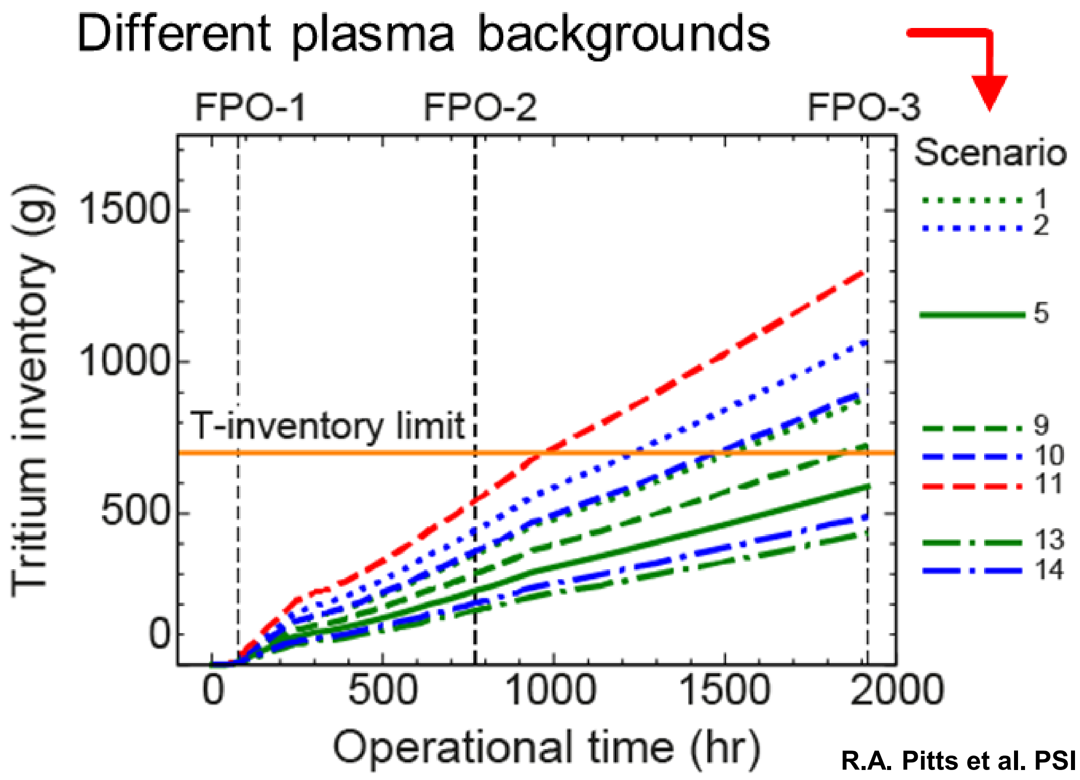
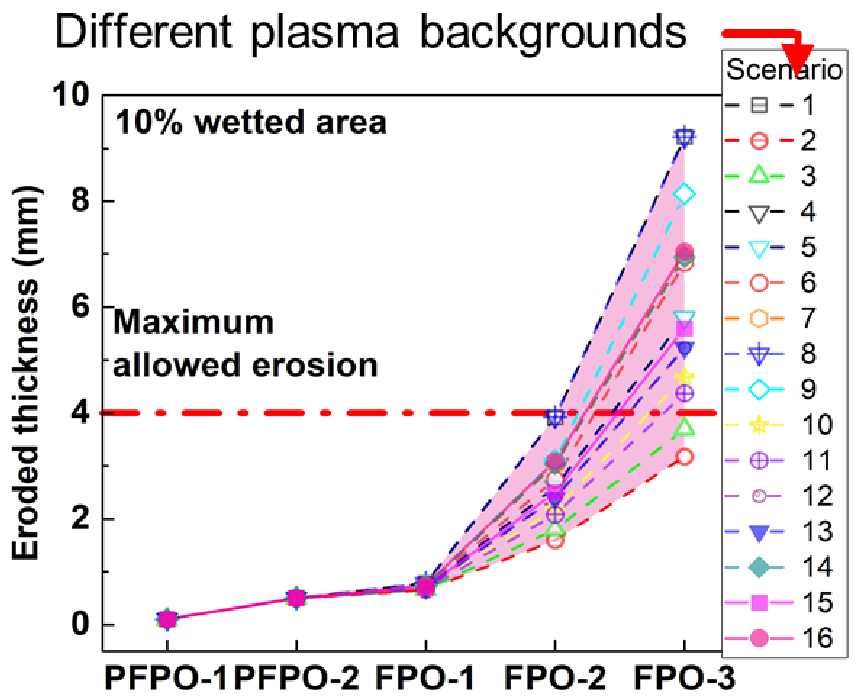
J. Romazanov et al.
NME2021, NF2022

ITER retention & erosion predictions: ERO2.0 & WallDYN

Set of potential ITER scenarios with different SOLPS plasma background, flows and magnetic shapes



R.A. Pitts et al. PSI 2022



G. De Temmerman, et al., in preparation for Nuclear Fusion

- Promising results simulating life-cycle of ITER assuming consecutive identical scenarios
  - peak Be erosion thickness just critical for some locations and scenarios in FPO-2
  - first tritium cleaning activities will be required likely late in FPO-2

=> Details in ITER presentation Loarte

ITER retention & erosion predictions: ERO2.0 & WallDYN

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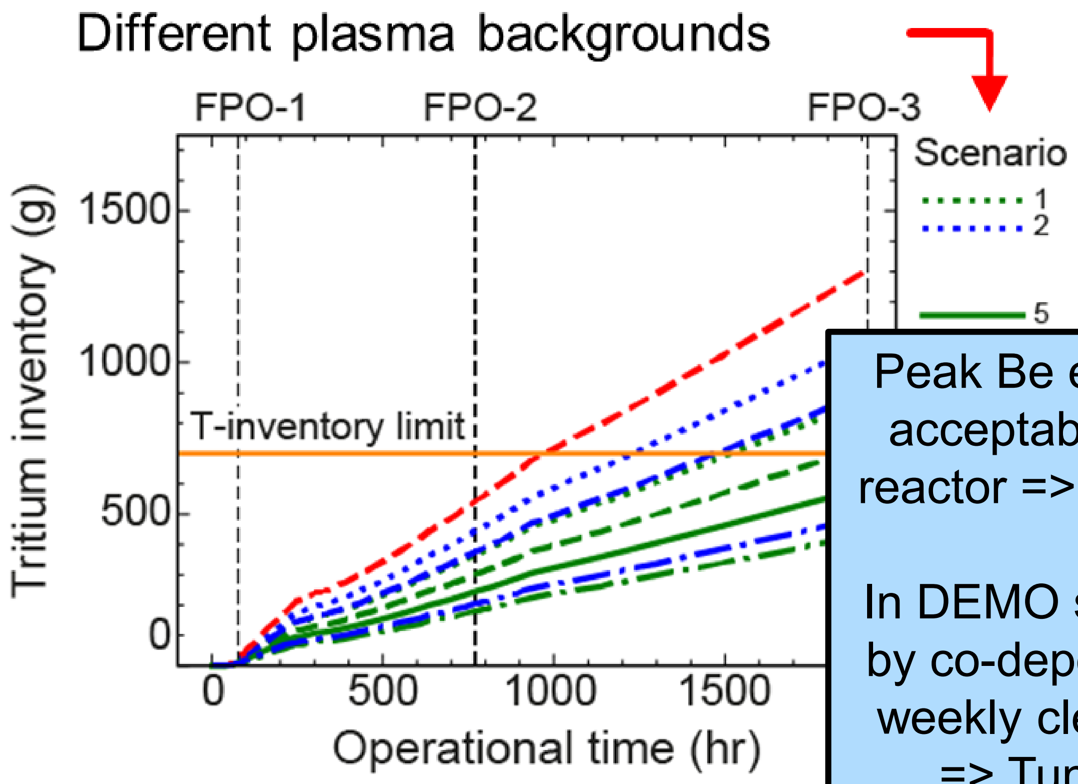
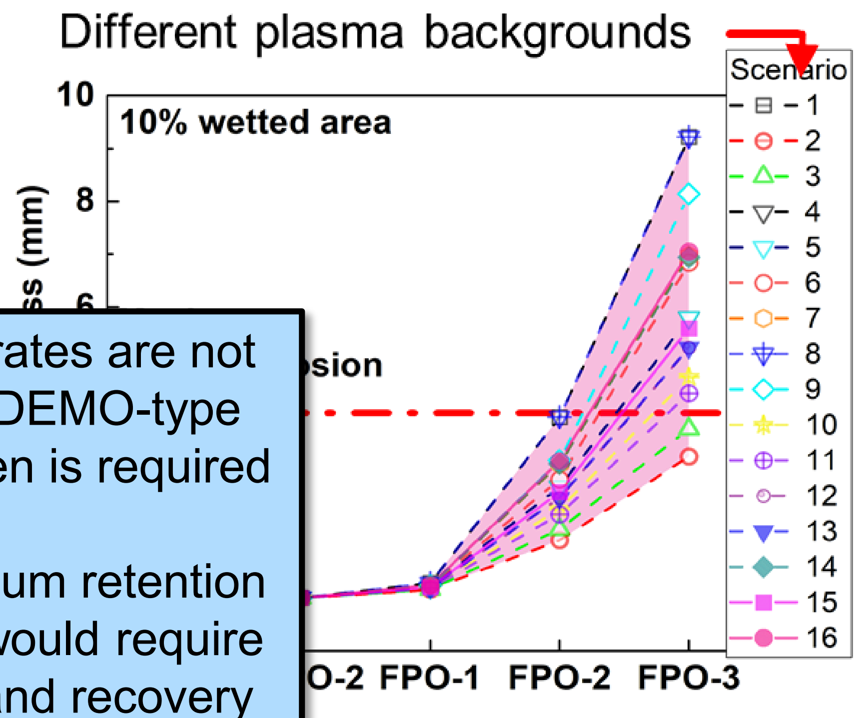


Peak Be erosion rates are not acceptable for a DEMO-type reactor => Tungsten is required

In DEMO such tritium retention by co-deposition would require weekly cleaning and recovery => Tungsten is required



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Outline

Plasma-Surface Interaction

- Low-Z and high-Z plasma-facing materials (PFM)
- Plasma-wall interaction processes (PWI)
- Processes leading to hydrogenic isotope (HI) retention

Operation in JET and conclusions for ITER

- HI retention in JET with Be/W material mix
- Be material migration in JET with Be/W material mix
- ERO2.0 and WallDYN simulations of material migration and retention
- Outlook to HI retention and migration in ITER

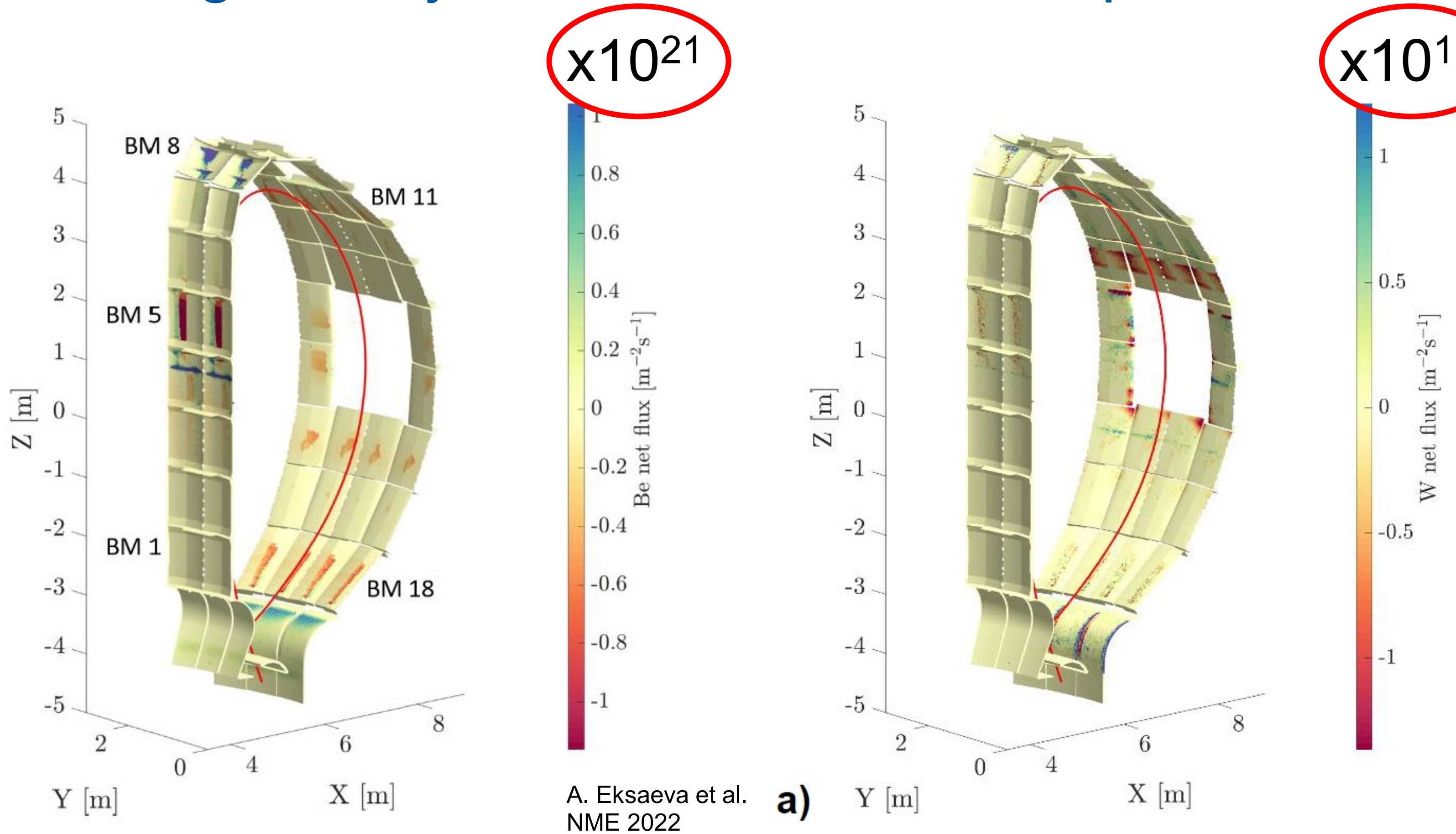
Next step: nuclear fusion reactor at high duty cycle

- *A hypothetical full-W ITER*
- *Outlook to an European DEMO with full-W first wall*

Conclusion

What would be different in a hypothetical full-W ITER?

Exchange Be by W in ERO2.0 and keep the semi-detached Q=10 scenario*



A. Eksaeva et al.
NME 2022

Would results in "full-W ITER":
- Low first wall W erosion
- Lowest co-deposition with W
- Residual T retention by implantation in the wall
- Increase of T retention with neutron damage in W (<1dpa)

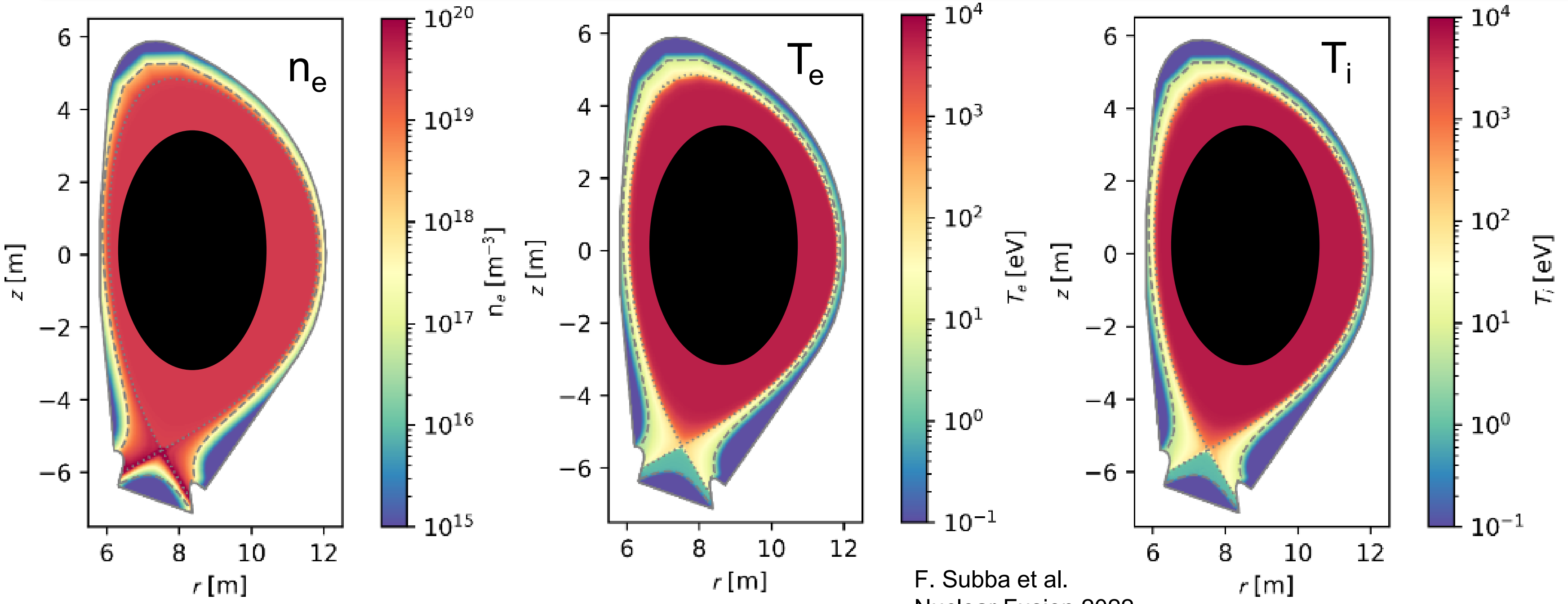
*ELMs are mitigated

- Net erosion of the first wall is drastically reduced by about 4 orders of magnitude for the peak erosion
- Residual sputtering caused mainly by impurity ions, 0.5% Ne ions (used to seed), and energetic CX HI

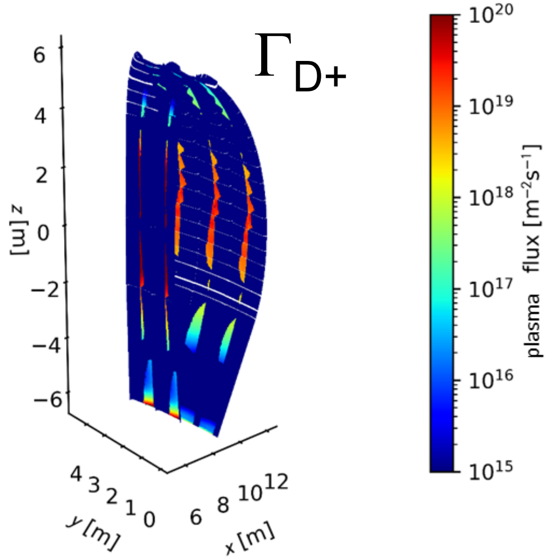
What can be expected in for an European DEMO?

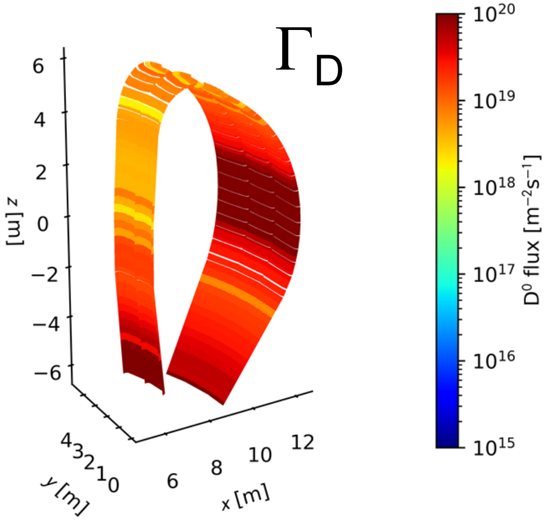
Following the ITER prediction route in a multi-year project

- SOLPS-ITER plasma background with Ar seeding in lower single null configuration
- ELM-free H-mode is assumed with additional core radiator (Kr or Xe)



F. Subba et al.
Nuclear Fusion 2022

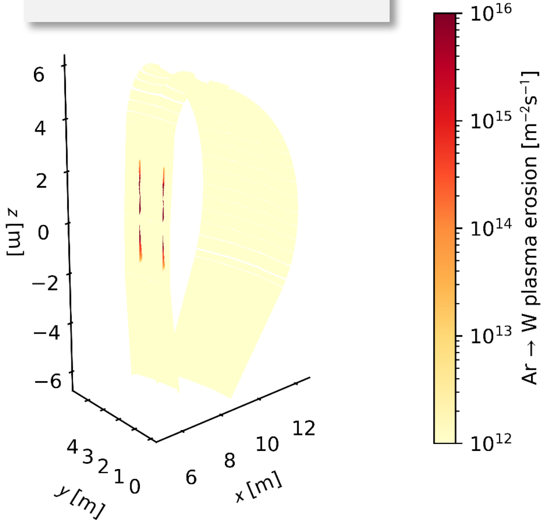




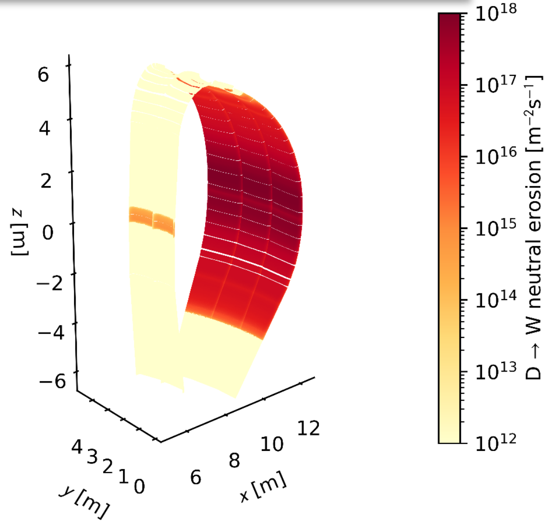
- Extension of the grid to the first wall assuming a decay length (ITER studies)
- Critical: CX HI flux, energy and angle distribution to the first wall (EIRENE)

W sputtering at the DEMO first wall (ERO2.0 / preliminary)

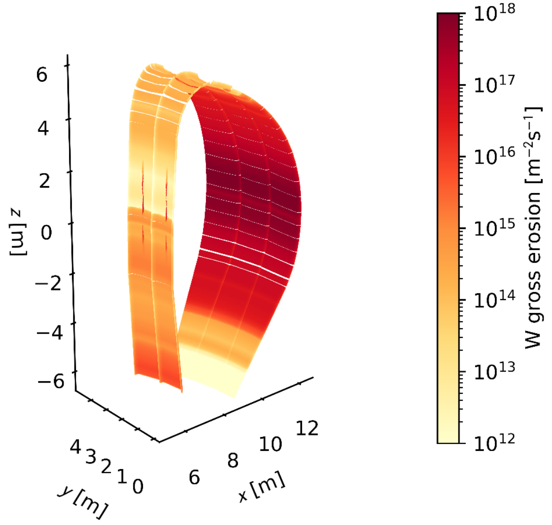
Ar ions => W



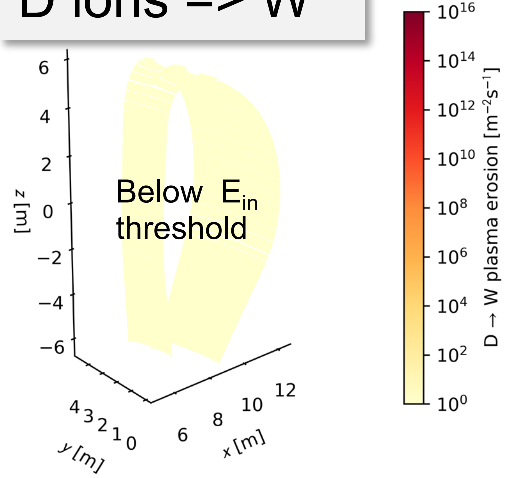
CX D atoms => W



W gross erosion rate (log. scale)

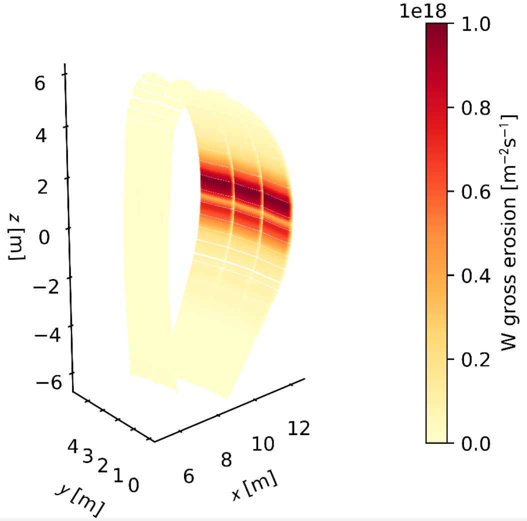


D ions => W



Main causes of W gross erosion:
- High energetic CX HI (keV)
- Residual Ar ions

Next steps:
- W net erosion estimation
- HI retention estimation

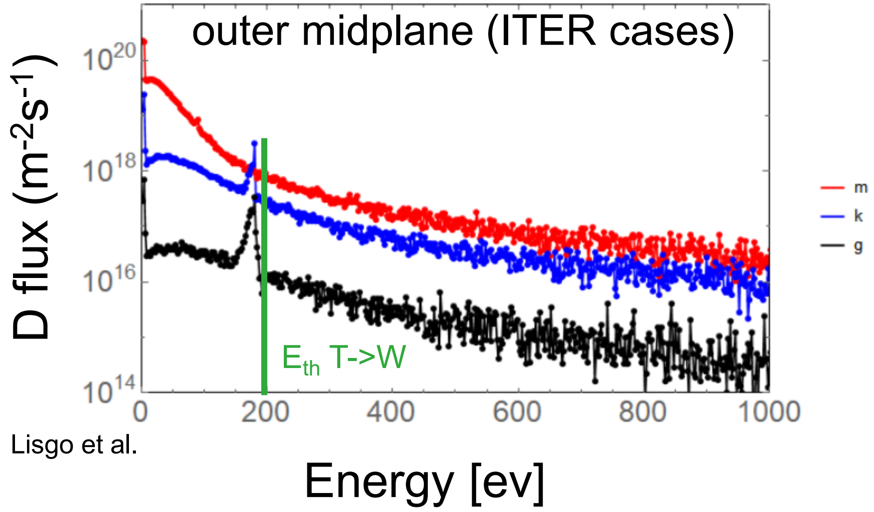


TSVV-7
Ch. Baumann et al.

W gross erosion rate (lin. scale)

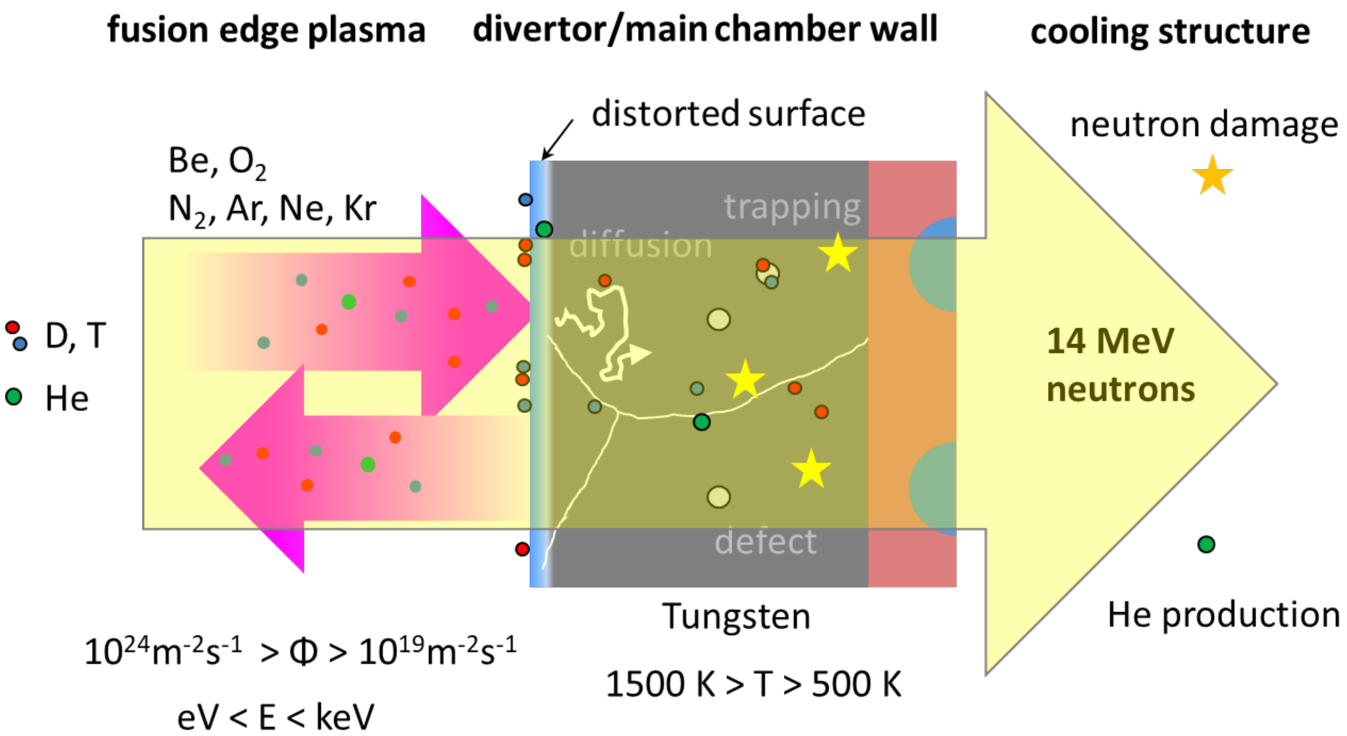
Critical: energetic CX HI and synergistic effects in PFCs

First wall erosion prone to CX HI



- High energetic tail characterisation
- Extension to recessed areas
- Structural material (steel)

Extension of T diffusion, trapping, permeation, release studies to PFCs



- Inclusion of neutron (-like) damages in W
- Synergistic effects with He
- Fuel release capabilities and schemes
- Permeation barriers

=> Impact of neutron damage in Schwarz-Selinger presentation

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Conclusion

Conclusion

JET/ITER class device (without significant neutron fluence)

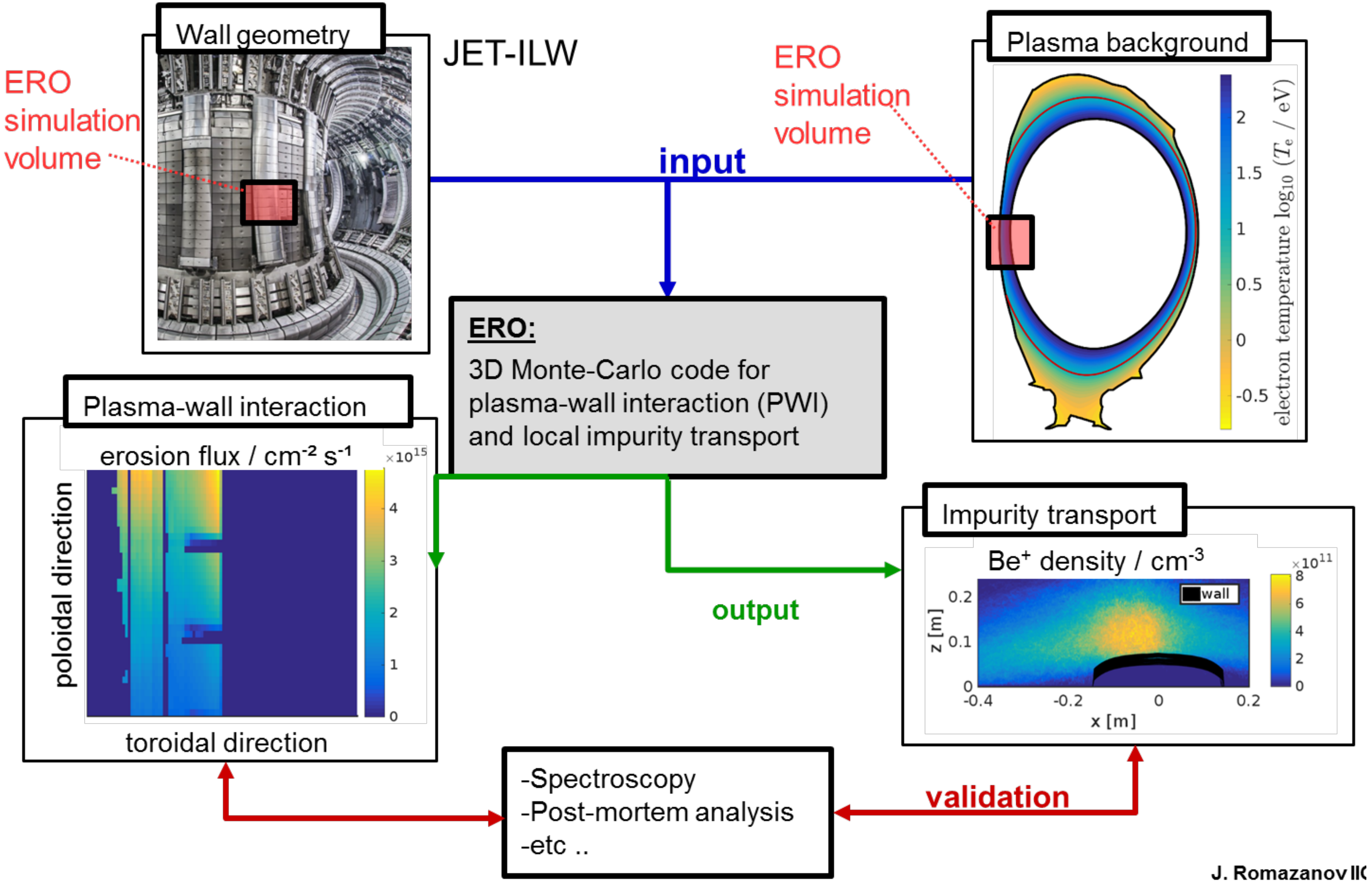
	positive	negative
graphite:	low central radiation radiation in boundary overload tolerance	high erosion and dust tritium co-deposition destruction by neutrons neutron-enhanced T retention
berrllium:	**oxygen getter**	
tungsten:	low erosion no tritium co-deposition resistant to neutrons	high central radiation helium-tungsten interaction critical with overload (melting) neutron-enhanced T retention

Effective plasma control with ELM and disruption mitigation is with both solid material solutions mandatory in a reactor to reach high duty cycle!

DEMO/reactor class (with significant neutron fluence)

Spare

ERO code description (3D simulation)



ERO1 = small volume
ERO2 = full vessel

WallDYN code description (2D simulation)

- **Main features**
  - **Non iterative** merge of **global impurity transport** (DIVIMP) with **surface models** (Sputtering, Chemical erosion, Sublimation, Seeding...)
  - Includes **re-erosion** of deposited material
  - Maintains a strict **global material balance**
- **Main calculation results**
  - Time evolution of impurity fluxes to / from the wall

    → Feedback to plasma imp. conc. → **Virtual diagnostics**
  - Surface composition evolution (areal densities)

    → Layer deposition

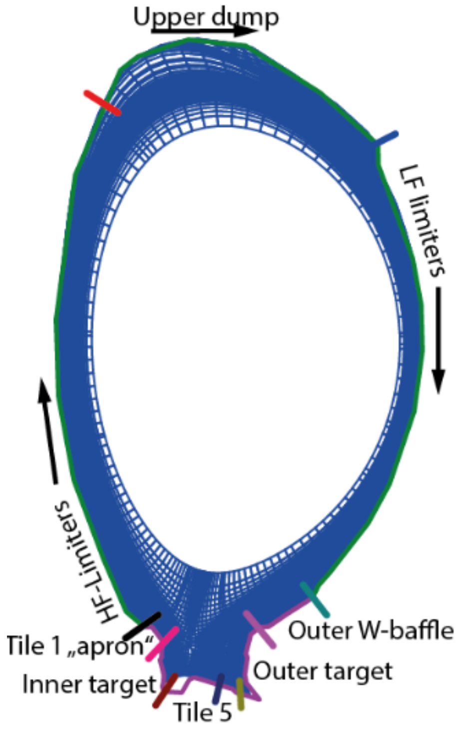
+ ***D/X ratios[1,2]*** → **Fuel retention via co-depostion**

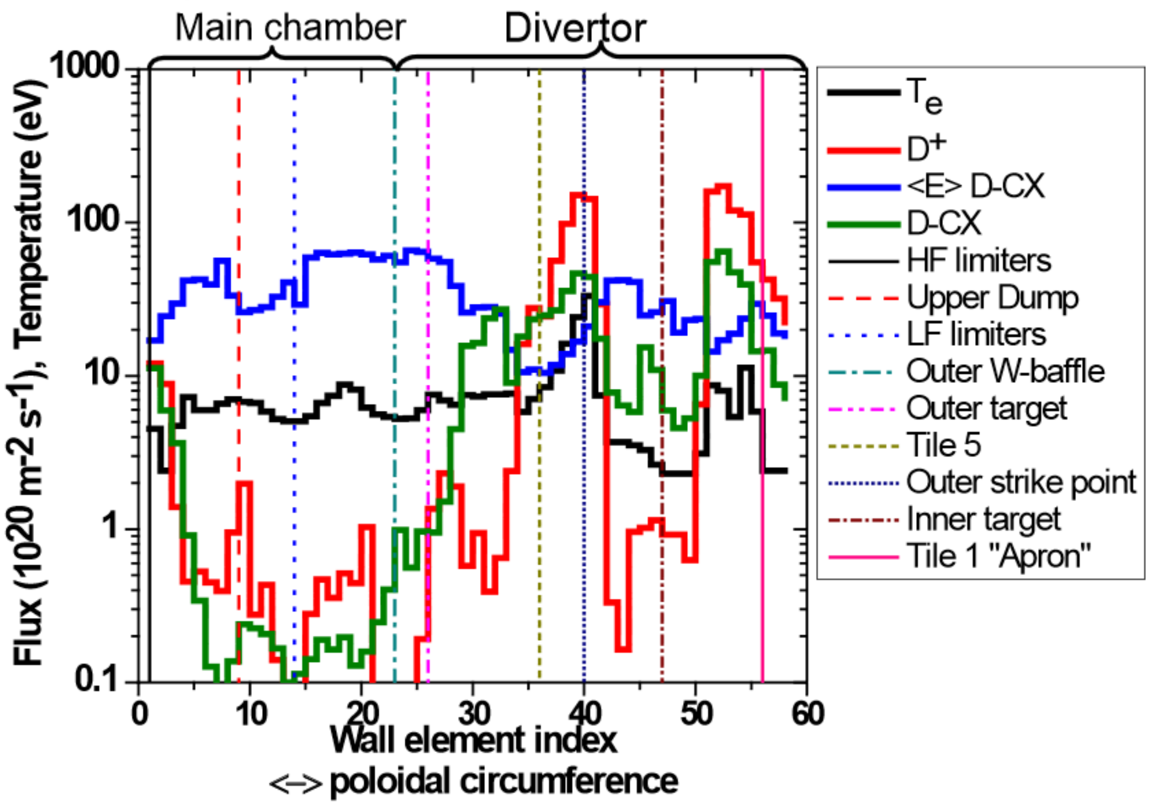
[1] G. De Temmerman et Al Nucl. Fusion 48 (2008) 075008 (7pp)
[2] R. Doerner Nucl. et Al Nucl. Fusion 49 (2009) 042002 (3pp)

K. Schmid et al.

WallDYN code description (2D simulation)

❖ Plasma paramters at wall: Example ohmic shot 80295

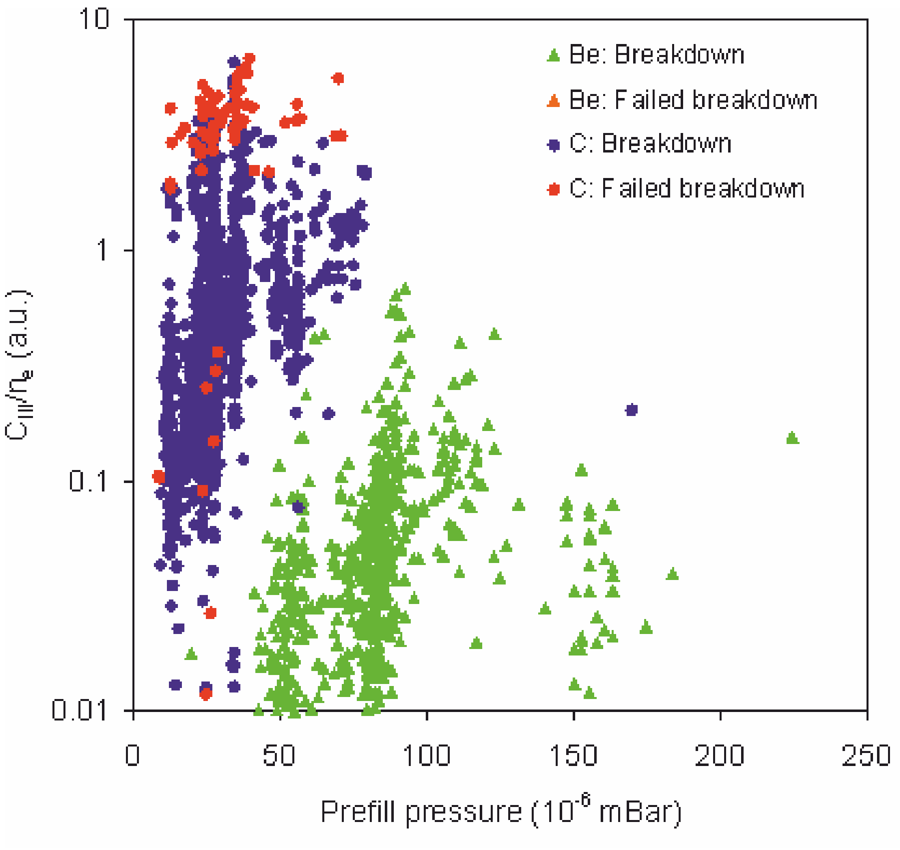


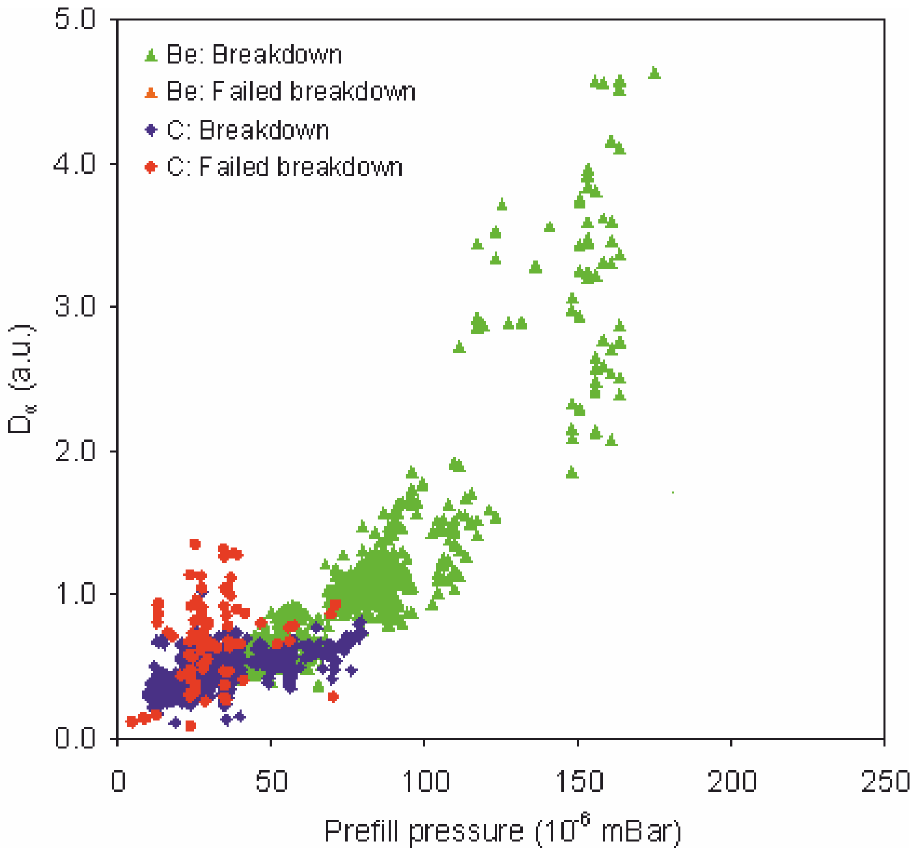


- Pure D plasma
- High Te at outer strike point of 30eV
- Still W sputtering by D-CX only

What about impurities?
(Except Be and W)

Breakdown with JET-ILW: low wall reservoir



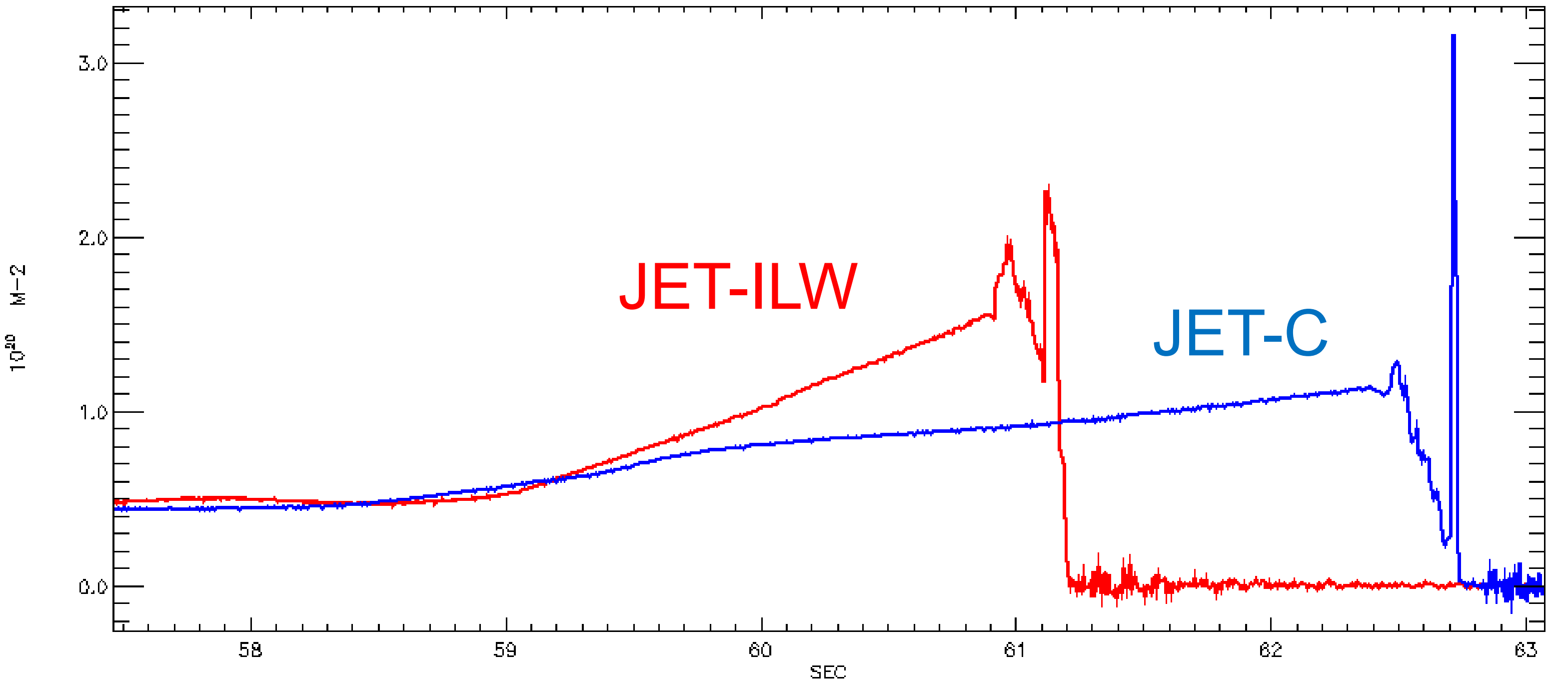


P. De Vries
NF 2013

- No issues with non-sustained breakdowns due to wall saturation like in JET-C
- Carbon radiation much reduced during breakdown with respect to JET-C
- Strong outgassing of fuel between discharges ensures better D control with Be walls

Denisty Limit in JET

L-mode density limit in high triangularity configuration (HT3L)



Higher density limit in pulses with Be/W wall in comparison with CFC walls,
but RF-plasmas with increased W content inhibit increase with power!
In CFC increase of density limit with input power observed.