Experience with Tritium retention and removal in JET-DTE2

D. Douai
CEA, IRFM, F-13108 Saint-Paul-lez-Durance, France
& Fusion Science Department, EUROfusion, D-85748 Garching, Germany

with special thanks to D. Matveev, T. Wauters, I. Jepu, S. Brezinsek, and JET Contributors

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the European Union's Horizon 2020 research and innovation programme under grant agreement number 633053.
The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Content

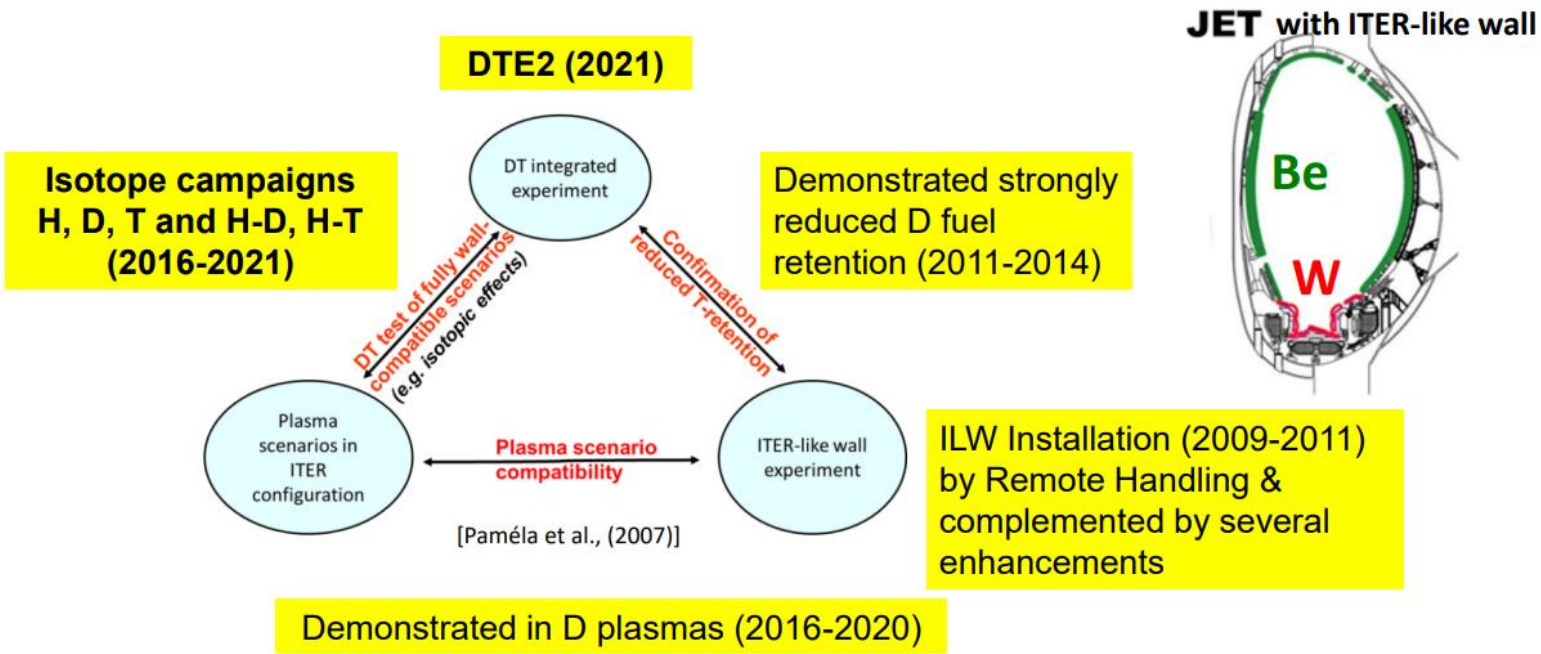
1. **JET operation ILW/DTE2:**
   - Tritium gas injection and T-NBI in DTE2
   - JET's Active Gas Handling System
   - Operational Budgets and Pattern for the DTE2 and Clean-Up Campaigns
     - Tritium, H isotopes and 14 MeV neutron budget for DTE2 / Clean-up
   - JET diagnostics for isotopic content in plasma
2. **Retention**
   - Retention & cleaning in DTE1
   - D fuel Retention in ILW
   - Gas balance in T
3. **Clean-up**
   - T removal strategy after DTE2 and TT
     - Baking, ICWC/GDC, RF-heated plasma and NBI heated plasmas
     - Raised Inner Strike Point Configuration
     - Quantification of T release
   - Modelling
4. **Summary – Lessons for ITER**

Content

1. **JET operation ILW/DTE2:**
   - Tritium gas injection and T-NBI in DTE2
   - JET's Active Gas Handling System
   - Operational Budgets and Pattern for the DTE2 and Clean-Up Campaigns
     - Tritium, H isotopes and 14 MeV neutron budget for DTE2 / Clean-up
   - JET diagnostics for isotopic content in plasma
2. **Retention**
   - Retention & cleaning in DTE1
   - D fuel Retention in ILW
   - Gas balance in T
3. **Clean-up**
   - T removal strategy after DTE2 and TT
     - Baking, ICWC/GDC, RF-heated plasma and NBI heated plasmas
     - Raised Inner Strike Point Configuration
     - Quantification of T release
   - Modelling
4. **Summary – Lessons for ITER**

JET's unique T capability with ILW

~10 years of preparations in view of DTE2

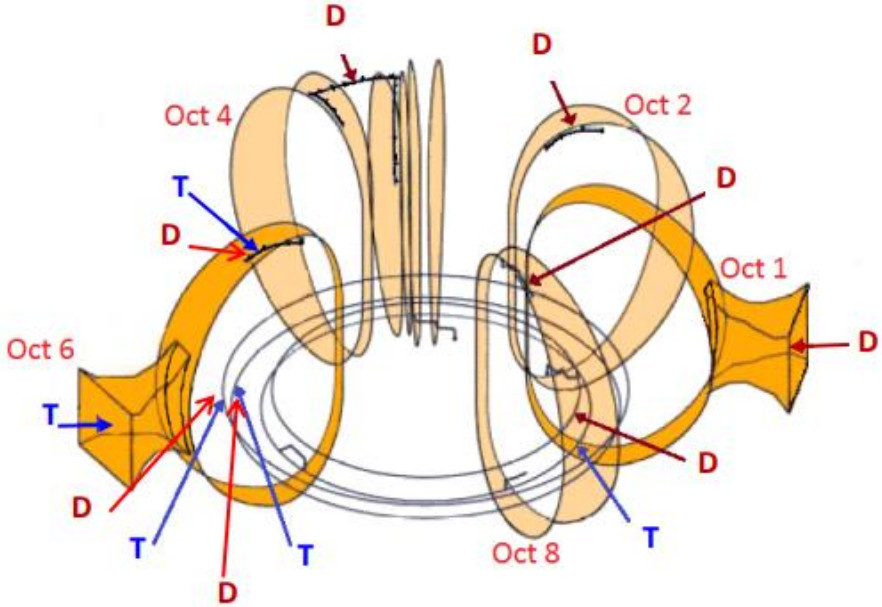


- Extensive refurbishment and upgrade of JET's Active Gas Handling Systems
- Significant expansion of diagnostic capability
- Safety case preparation…

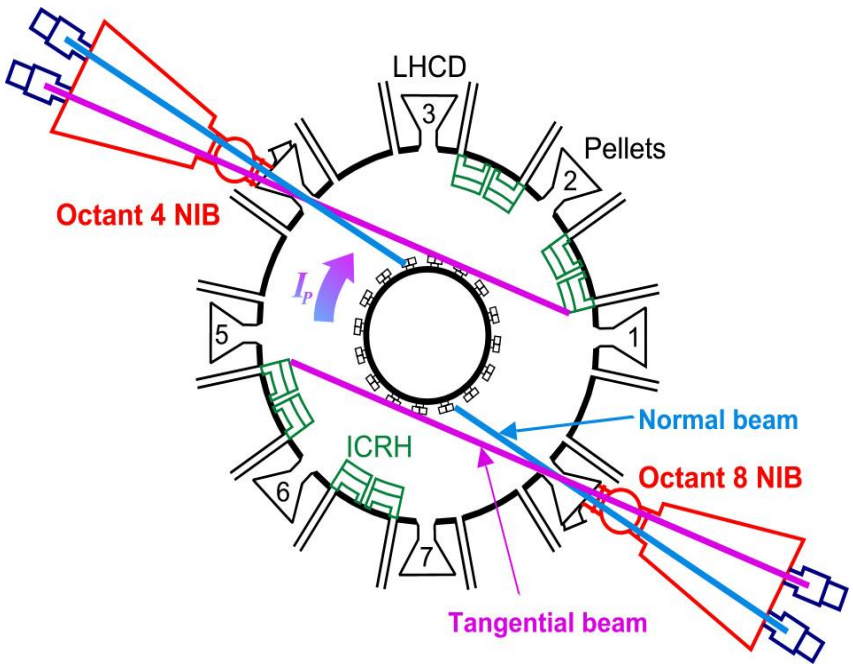
Tritium gas injection and T-NBI in DTE2

T gas injection: 5 Tritium Injection Modules (TIMs), in different areas of the vessel (only 1 module in DTE1)

T-NBI: T fed into both Neutral Beam Injection Boxes (only 1 NBI box in DTE1)



top view of JET

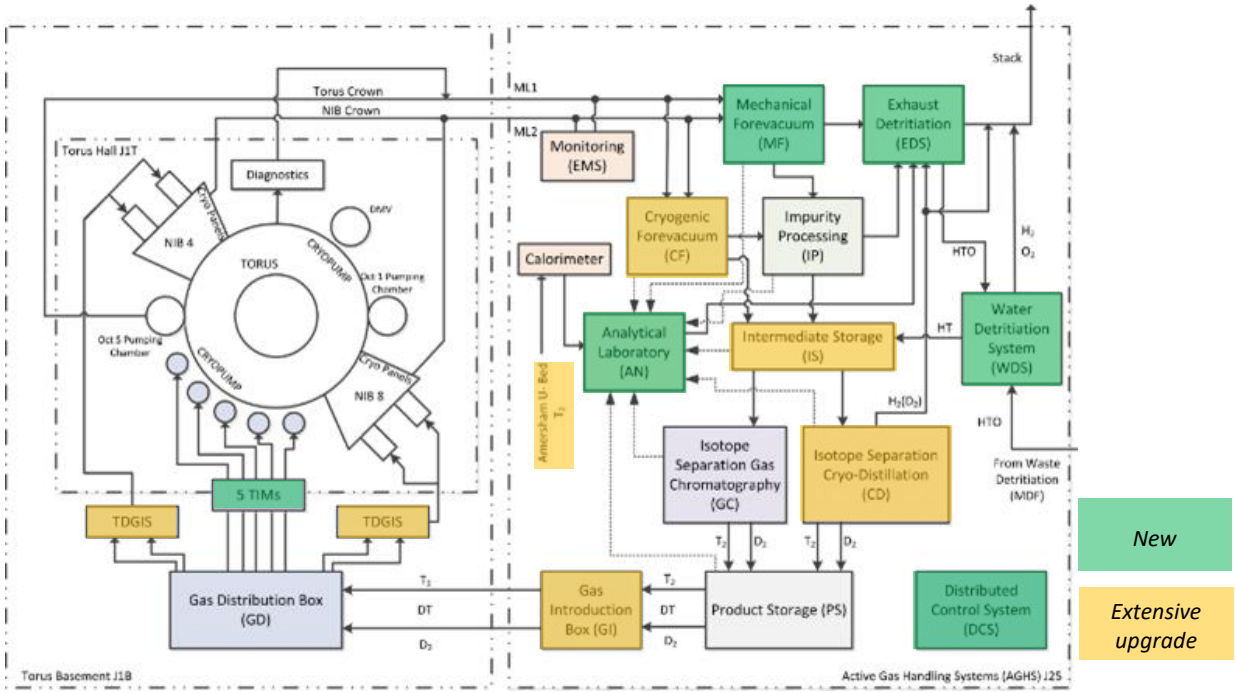


Capability for 100% high power Tritium experiments
69g Tritium on site for T and DTE2 (21g in DTE1)

Upgrade of JET's Active Gas Handling System

AGHS stores, supplies and recycles T going to and from JET

- Daily gas feed to torus & Neutral Beams
- Daily gas exhaust recovery and overnight storage
- Weekly reprocessing and accountancy every 3 weeks



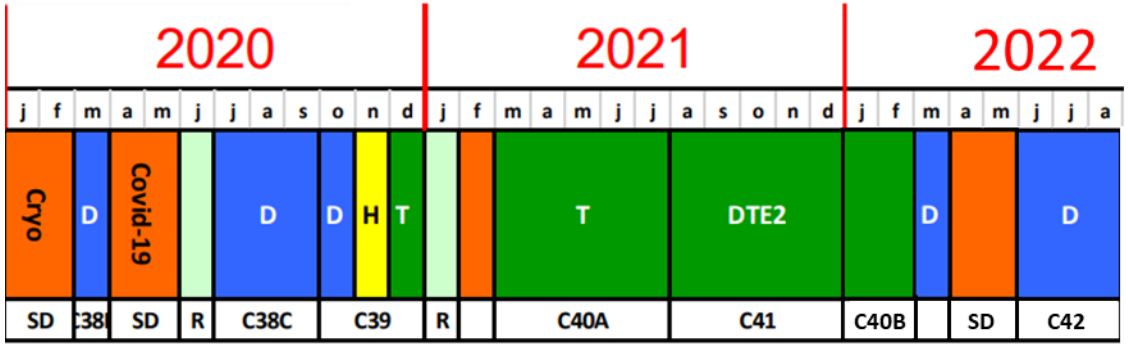
- Daily limit of **11g T** on the inventory allowed on torus & NBI cryo panels (JET Safety Case)
- Isotope separation process (Gas Chromatography) can process **up to 90 bar.L of all isotopes of hydrogen a day**
- **daily (overnight) regeneration of all cryopanels**

1Kg Tritium used for T + DTE2: 240 g (TIMs) and 763 g (T-NBI)

(DTE1 : ~ 100g T)

TT and DTE2 Schedule & Budget

- <u>Operational Pattern:</u>
  - Reprocessing in parallel with operation not possible in TT and DT
    - decision was taken not to connect the foreseen second Product Storage manifold, in order not to delay Tritium ops
  - ➢ 4 operational days per week
  - ➢ five-week cycle with three weeks of operation following by one week of tritium reprocessing and one week of tritium accounting In order to keep up with tritium reprocessing and accounting

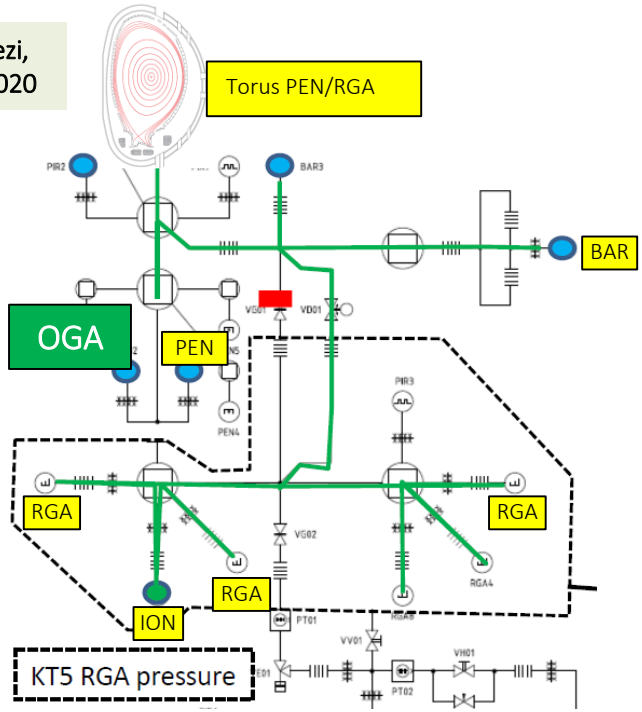


- <u>14 MeV Neutron budget:</u>
  - 1.3×10^{21} neutron budget for DTE2 (campaign C41)
  - 5×10^{19} for each of the TT (C39/40) and Clean-up (C42) campaigns

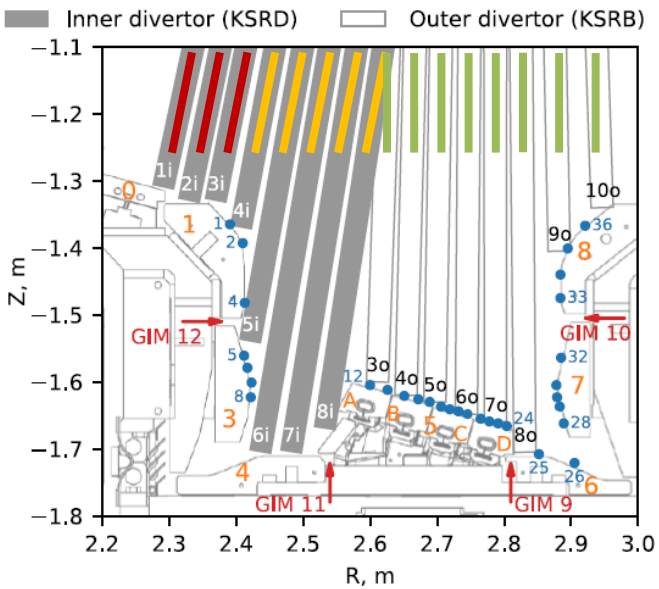
JET diagnostics for isotopic content in plasma

1. Sub-Divertor neutral gas diagnostics

U. Kruezi, JINST 2020

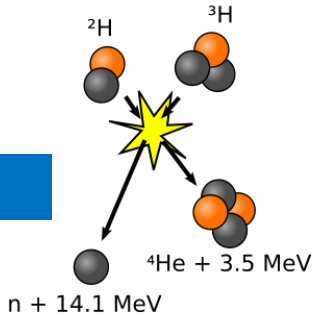


2. Divertor optical emission spectroscopy (OES)



V. Neverov, NF 2019

3. Neutron diagnostics



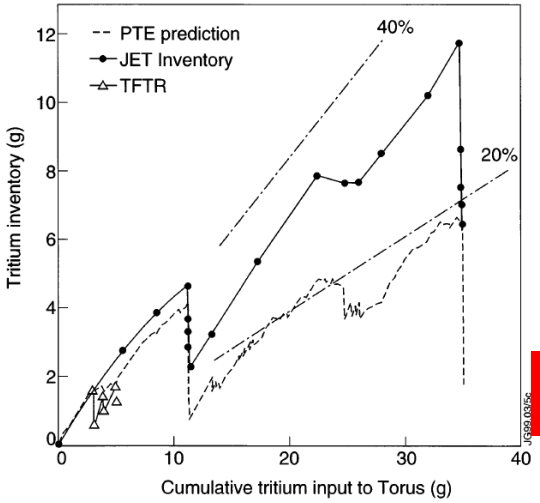
Expected detection limits for T minority

OGA:	T/D ~ 1-2%	(1)
OES:	T/D ~ 1-2%	(2)
14 MeV n:	T/D < 0.1%	(3)

Content

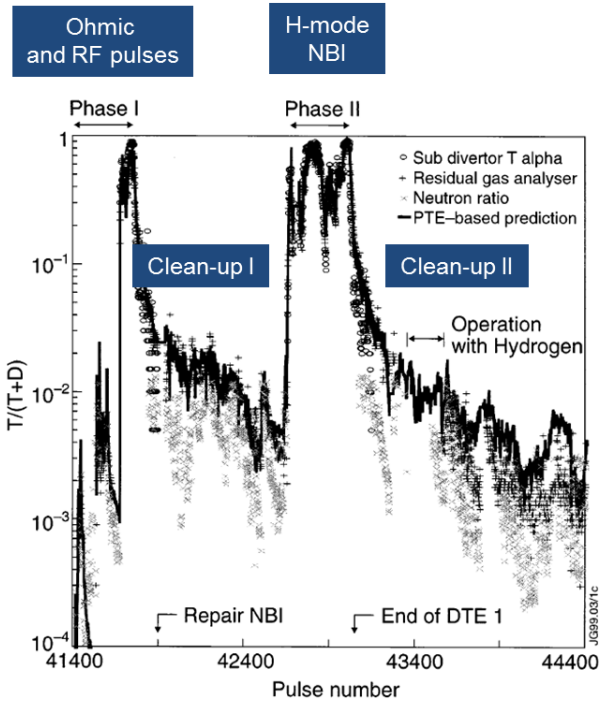
1. **JET operation ILW/DTE2:**
   - Tritium gas injection and T-NBI in DTE2
   - JET's Active Gas Handling System
   - Operational Budgets and Pattern for the DTE2 and Clean-Up Campaigns
     - Tritium, H isotopes and 14 MeV neutron budget for DTE2 / Clean-up
   - JET diagnostics for isotopic content in plasma
2. **Retention**
   - Retention & cleaning in DTE1
   - D fuel Retention in ILW
   - Gas balance in T
3. **Clean-up**
   - T removal strategy after DTE2 and TT
     - Baking, ICWC/GDC, RF-heated plasma and NBI heated plasmas
     - Raised Inner Strike Point Configuration
     - Quantification of T release
   - Modelling
4. **Summary – Lessons for ITER**

DTE1 results (retention / cleaning)



P. Andrew et al. FED 47 1999

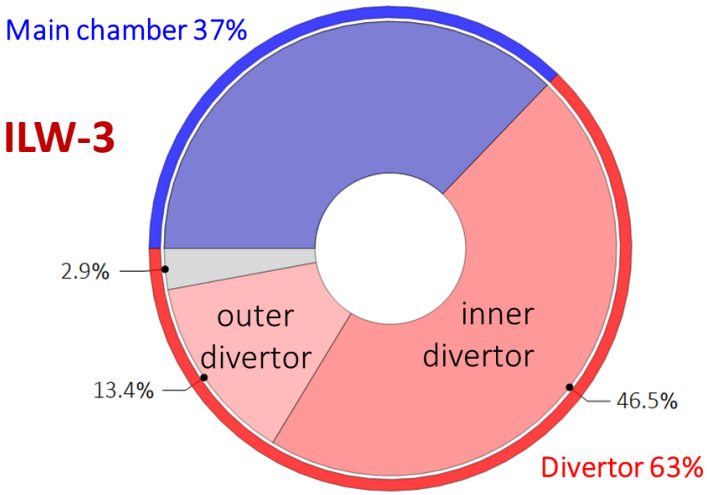
- JET DTE1 operation :
  - **35 g of tritium** introduced into the torus, mainly by gas puffing
  - torus T-inventory accumulated at a rate of about 40%



- Isotopic exchange with D or H plasmas (ohmic, RF and NBI heated, GDC, ECRH) :
  - plasma tritium fraction reduced below 1% in a few days of plasma ops in D or H + GDC, ECRH
  - inventory decreased to limit of about 6g (17%) of total input at the end of DTE1
- Ventilation of the torus (Post DTE1, in N_2, then air) → further 1/3 reduction
- ~3 g believed to remain retained in C flakes at the inner divertor

Fuel retention in JET-ILW (post-mortem analysis)

- Post-mortem analysis of JET components after 2015-2016 “ILW3” D operations completed
- IBA techniques, TDS, SIMS



- Fuel retention in the divertor is dominated by co-deposition with Beryllium (over implantation in Be/W)
- Inner divertor: ~47% of retained D in ILW-3

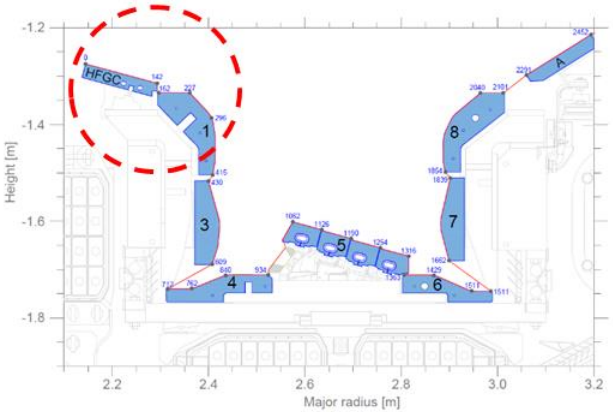
[Widdowson PFMC 2021]

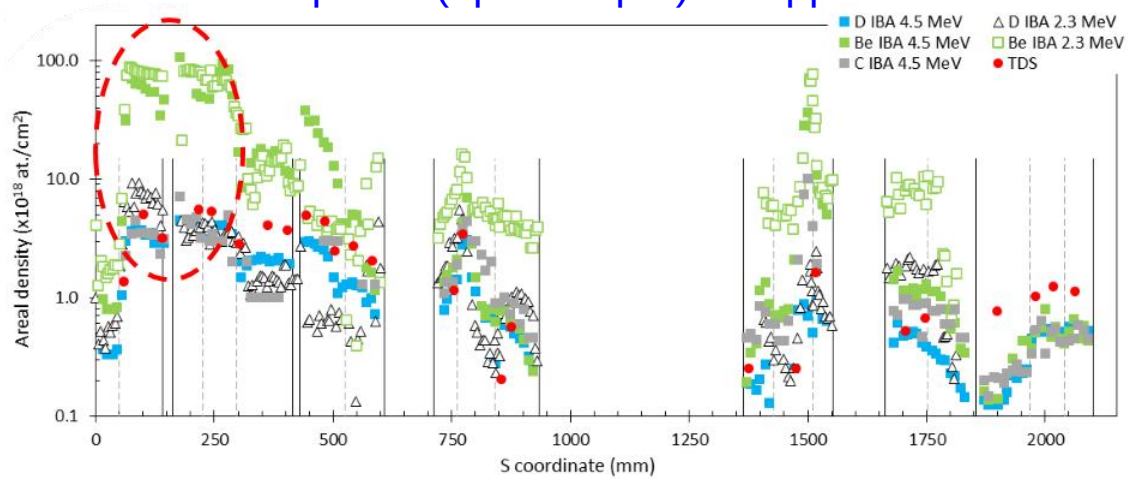
- 0.24% of puffed D retained
  - ILW-1: 3.9 x 10^{23} - ILW-3: 4.2x 10^{23}
  - Post mortem: 4 x 10^{18} D/s global retention rate

    Gas balance: 0.2 − 1.5 x 10^{20} D/s

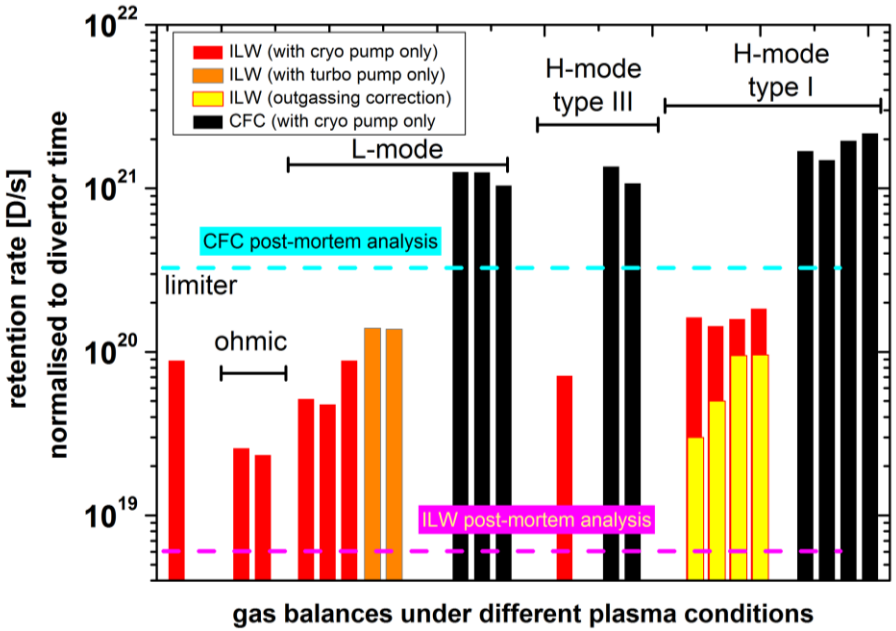
[Brezinsek NF2013]

- Thickest deposits (up to 40 µm) on upper inner divertor

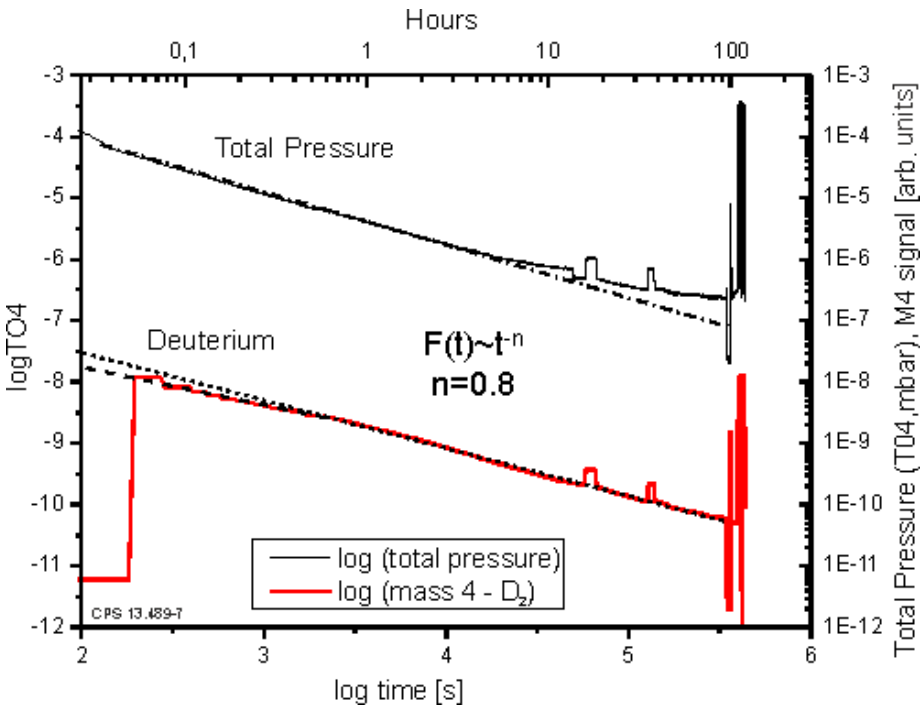




Gas balances in D



S. Brezinsek NF 2013

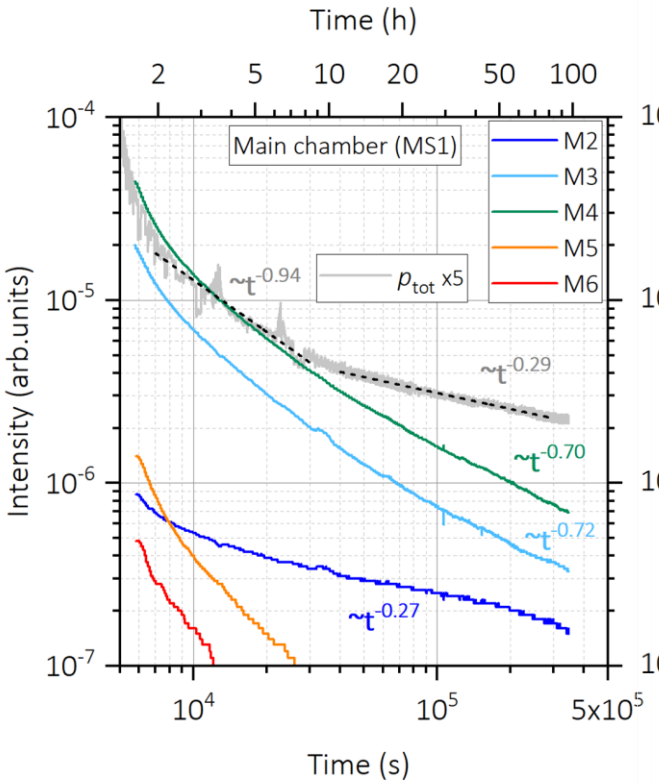


- JET-ILW fuel retention determined by 2/3 co-deposition in Be layers and 1/3 implantation in Be and W
- Large factor between global gas balance and ex-situ fuel retention analysis = > outgassing important

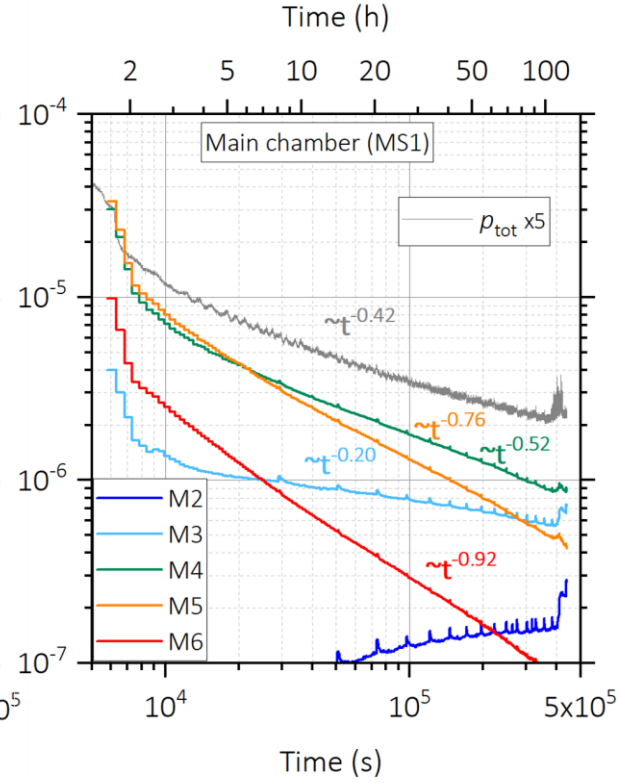
Short-term retention / long-term outgassing

Time evolution of RGA mass signals after

D pulses (#82449, 29.02.2012)



DT pulses (#99530, 02.10.2021)

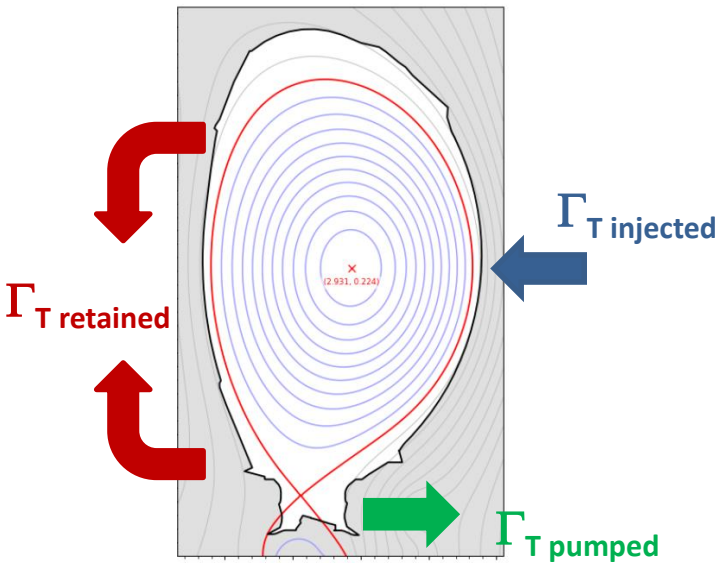


Values of α in ~$t^{-\alpha}$ for different masses

RGA signal	03/2012 (D)	10/2021 (DT)	12/2021 (DT)	Main species
M2	0.27	-	-	H_2^+, D^+
M3	0.72	0.20	0.29	HD^+, T^+, H_3^+
M4	0.70	**0.52**	**0.51**	D_2^+, HT^+, H_2D^+
M5	-	**0.76**	**0.77**	DT^+, D_2H^+, H_2T^+
M6	-	**0.92**	**0.92**	T_2^+, D_3^+

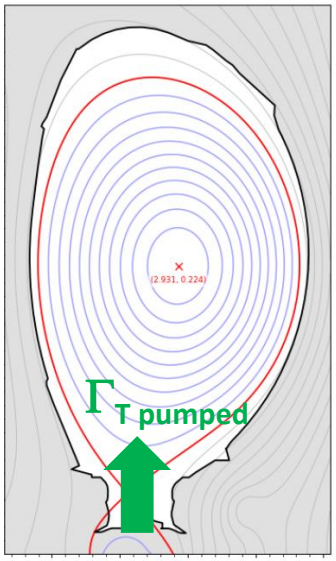
→ Faster decay of RGA signal at M/Z=6 amu (T_2) than at M/Z=4 amu (D_2)
→ Different decay times in D and DT still unclear
→ must be considered in global gas balances

Gas balance in T



- **Repeat identical discharges**
  - L-mode, 1.5MW ICRF
  - Limiter/Divertor time: 55/115s

Gas balance in T

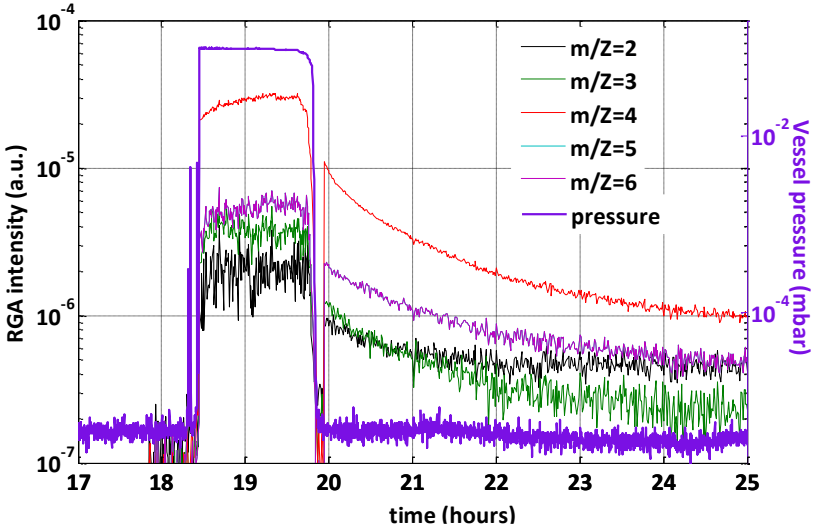


- **Repeat identical discharges**
  - L-mode, 1.5MW ICRF
  - Limiter/Divertor time: 55/115s

- **Global gas balance:** Pressure-Volume-Temperature (**PVT**) and Residual Gas Analysis (**RGA**) of gas released after Divertor cryopump regeneration

  → Includes inter-shot outgassing

  → Long-term retention



→ T Pumped: Δp x IR x Vol_{Torus}~ T Injected

with T/[H+D+T]~95%

→ Close match but high uncertainties!

→ To be repeated with better calibration for effective VV temperature

Content

1. **JET operation ILW/DTE2:**
   - Tritium gas injection and T-NBI in DTE2
   - JET's Active Gas Handling System
   - Operational Budgets and Pattern for the DTE2 and Clean-Up Campaigns
     - Tritium, H isotopes and 14 MeV neutron budget for DTE2 / Clean-up
   - JET diagnostics for isotopic content in plasma
2. **Retention**
   - Retention & cleaning in DTE1
   - D fuel Retention in ILW
   - Gas balance in T
3. **Clean-up**
   - T removal strategy after DTE2 and TT
     - Baking, ICWC/GDC, RF-heated plasma and NBI heated plasmas
     - Raised Inner Strike Point Configuration
     - Quantification of T release
   - Modelling
4. **Summary – Lessons for ITER**

Strategy for T removal after DT/TT ops

14 MeV neutron budget in D campaign following DTE2/TT : $5x10^{19}$

- assumes 1% residual T in DD plasmas → **T/(H+D+T) < 1% necessary**

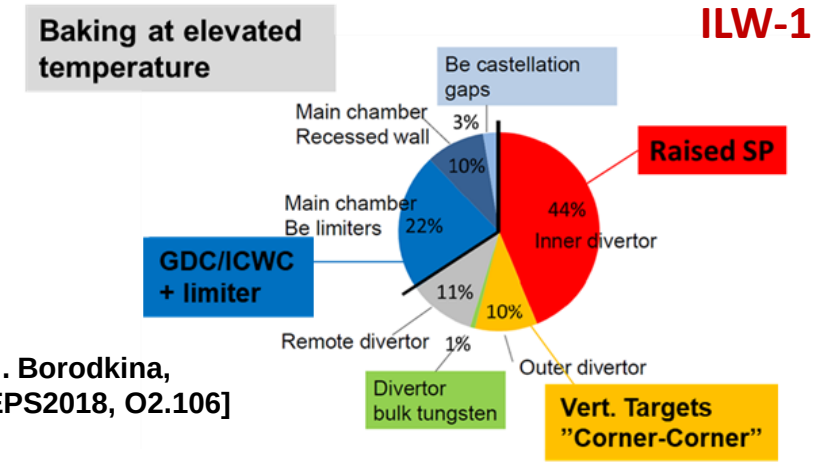
Strategy tested in H to remove D prior to TT

- Identical assumption (<1%D) and budget in TT campaign prior to DTE2

….- Oct /2020	Oct-Nov 2020	Nov-Dec 2020	Jan-July 2021	Aug-Dec 2021	Jan-Feb 2022	March 2022	July 2022 ….
D plasmas	D cleaning	H plasmas	T plasmas (T reprocessing+ no ops.)	DT plasmas (T reprocessing+ no ops.)	T plasmas (T reprocessing+ no ops.)	T cleaning	D plasmas

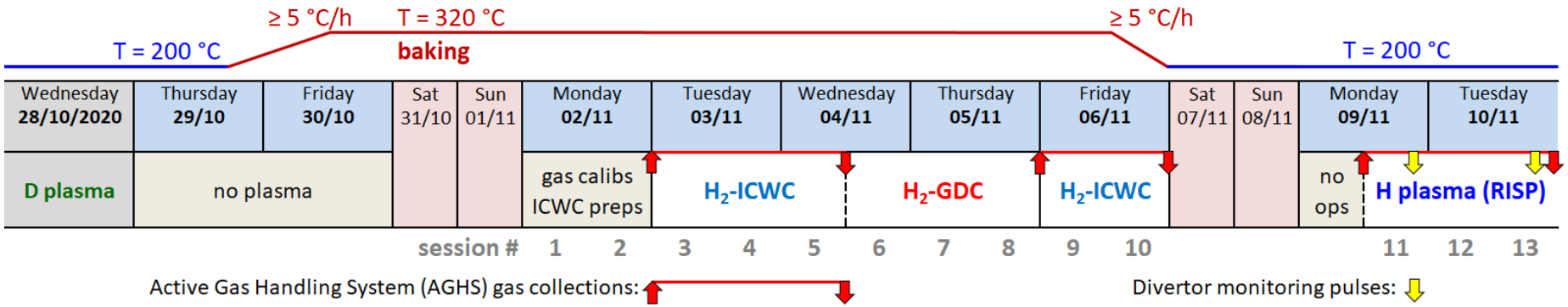
D/(H+D)<1% T/(H+D+T)<1%

Combination of techniques targeting different areas:



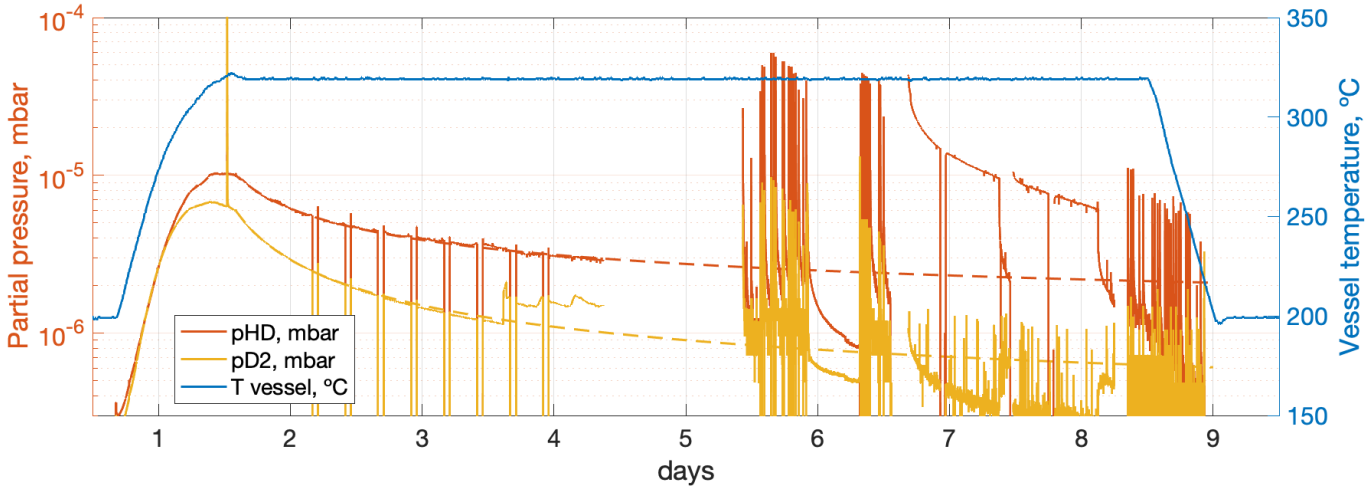
[I. Borodkina, EPS2018, O2.106]

D removal prior to TT

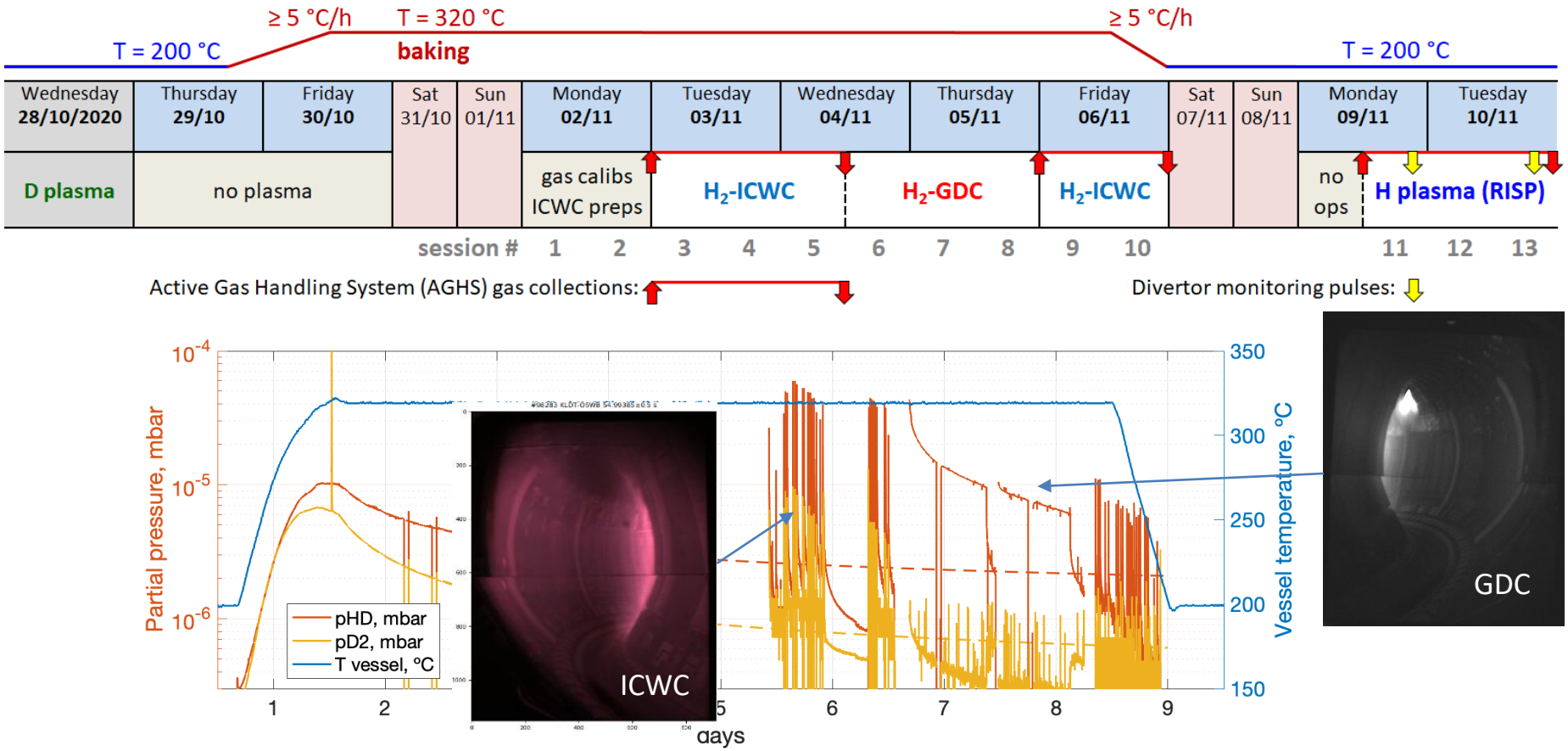


Active Gas Handling System (AGHS) gas collections: ↑ ↓

Divertor monitoring pulses: ⇩



D removal prior to TT

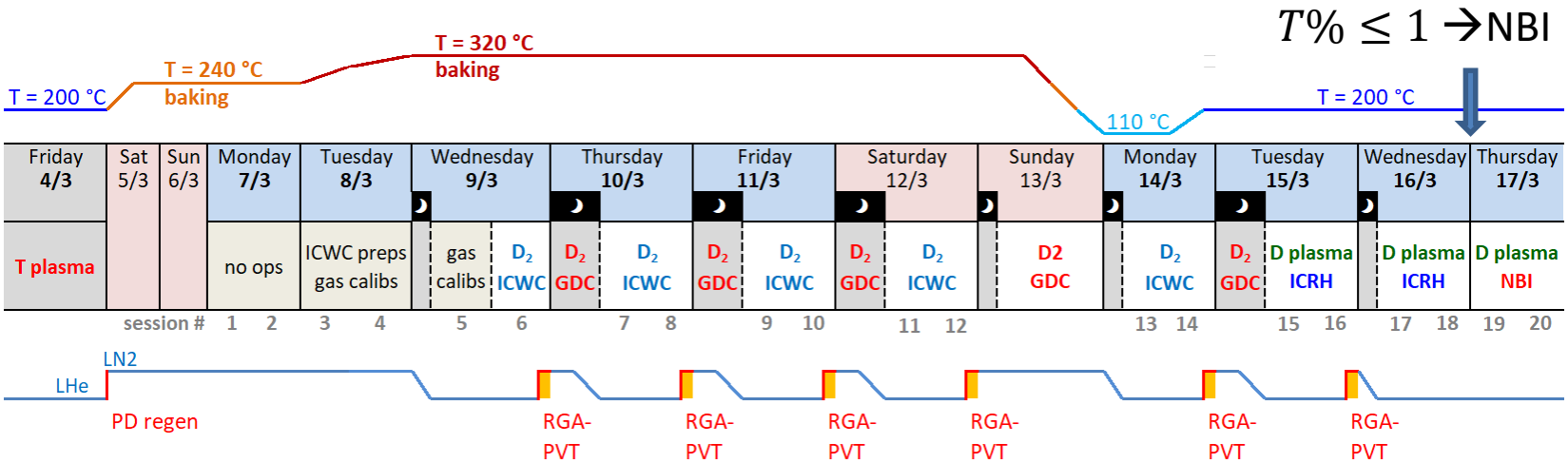


in total : 5.4 x 10^{23} D atoms removed (from gas chromatography, RGA and OGA)

- ~ 55% by 320°C baking
- ~ 40% by ICWC and GDC
- ~ 5% in following diverted plasmas

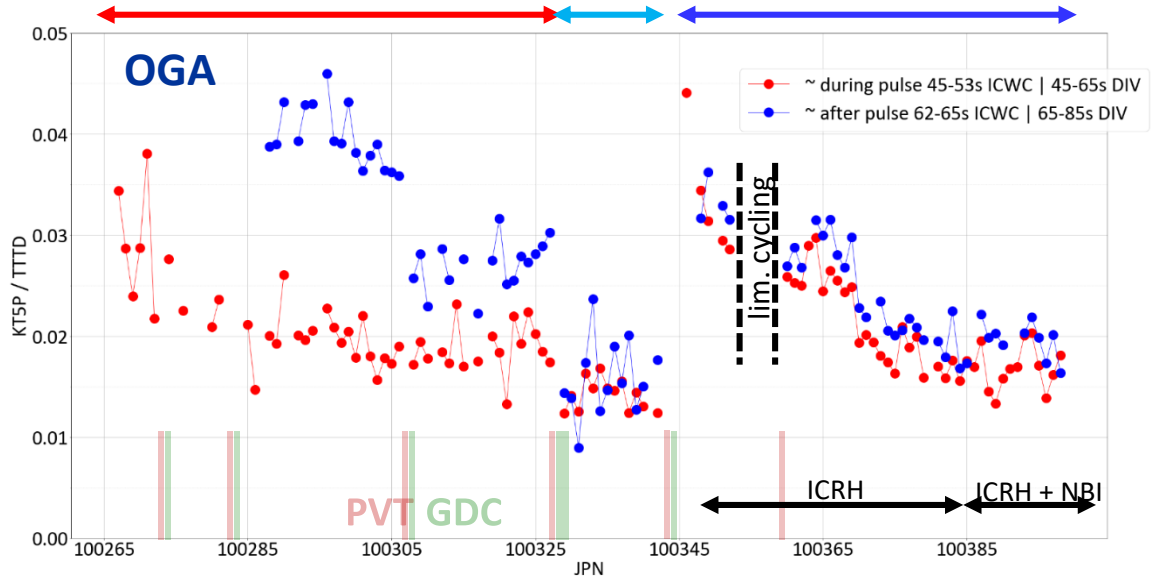
GDC & ICWC: ~×2 gain vs baking alone

T removal strategy after DT/TT operation



- low T fraction (~4-5%) after baking
- discontinuity of T/(H+D+T) around GDCs

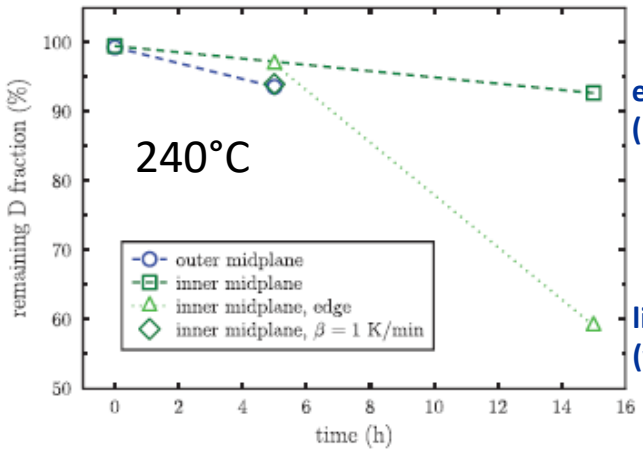
- Higher T content in plasmas than after ICWC/GDC (flux)
- Further decrease down to 1%T (0.1% estimated from neutron spectroscopy)



T/(H+D+T) from subdivertor optical penning gauge

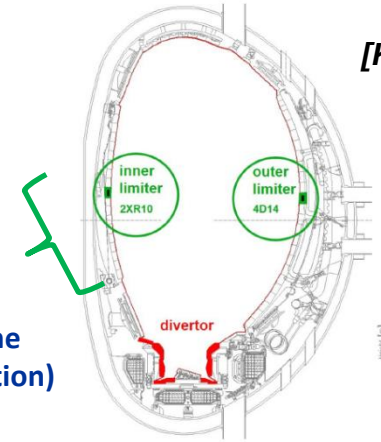
Thermo-desorption of ILW samples

- **bulk Be limiters and W-coated CFC (with Be codeposited layers)**

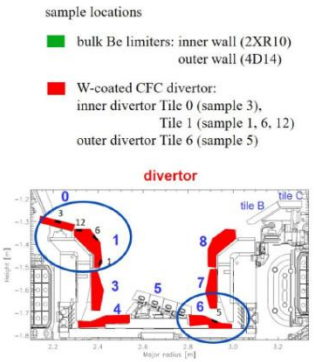


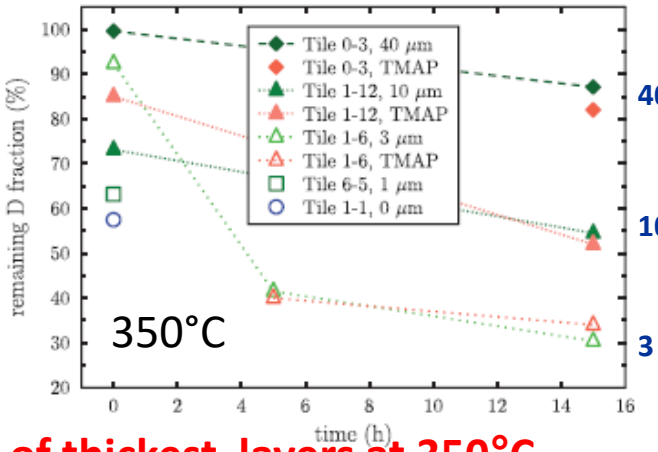
erosion zone (no deposition)

limiter edge (with deposists)



[K. Heinola et al., NF 2017]





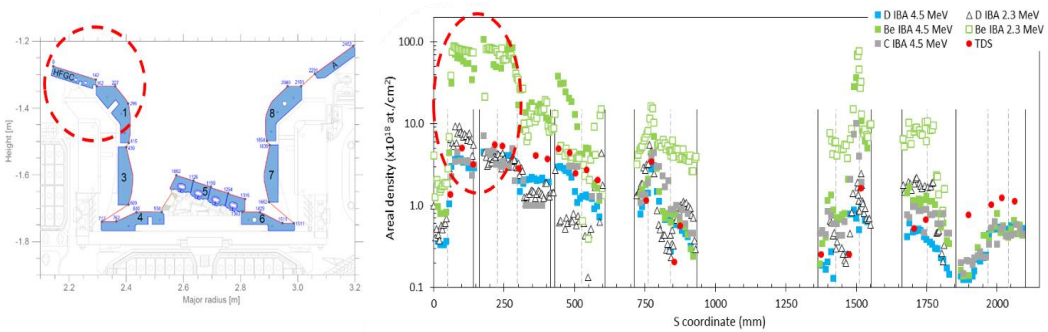
40 µm

10 µm

3 µm

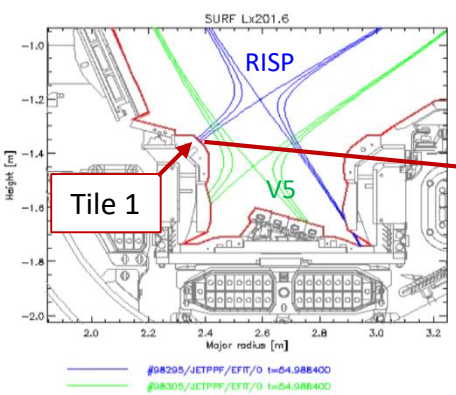
- **Co deposited layers:**
  - **release maxima at 328, 418 and 552°C**
  - **up to 90% of trapped fuel remaining after baking of thickest layers at 350°C**
- **90% of "implanted" fuel (eroded area) remaining after baking at 240°C**
  - **release from codeposits on limiter edge easier**
- **Fuel is recovered when T_{surf} raised up to 800°C**

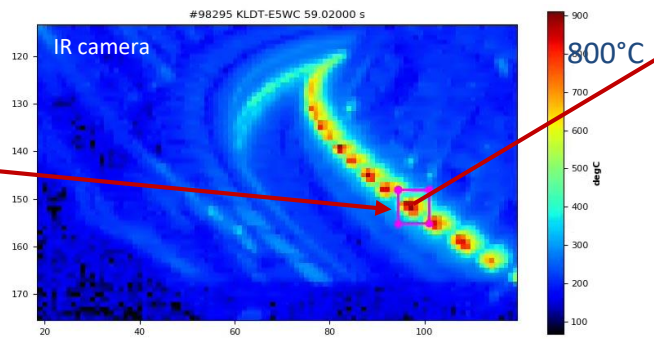
Accessing divertor deposits in JET

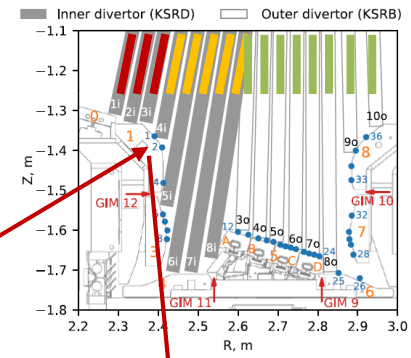


- Thickest deposits (up to 40 µm) on upper inner divertor

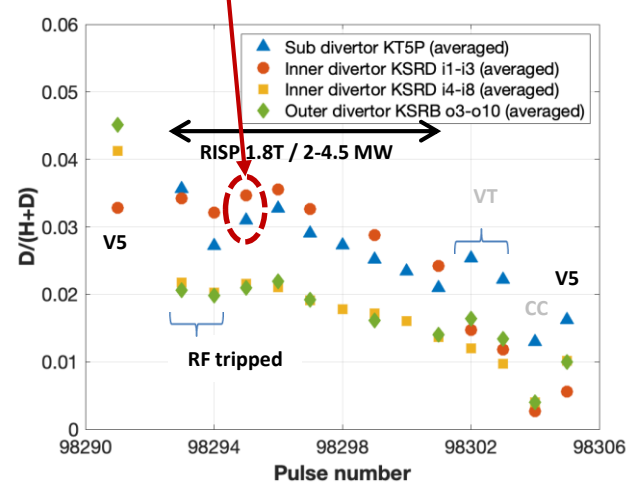
- Diverted plasma with raised inner strike point (RISP)







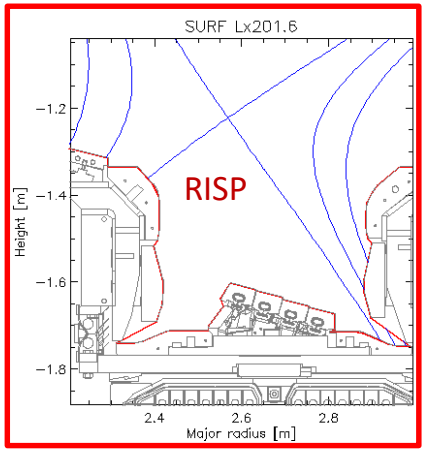
Divertor optical emission spectroscopy

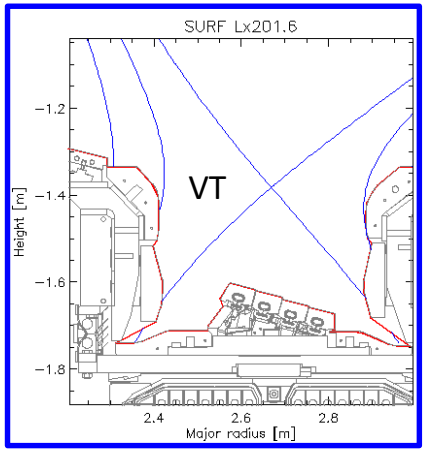


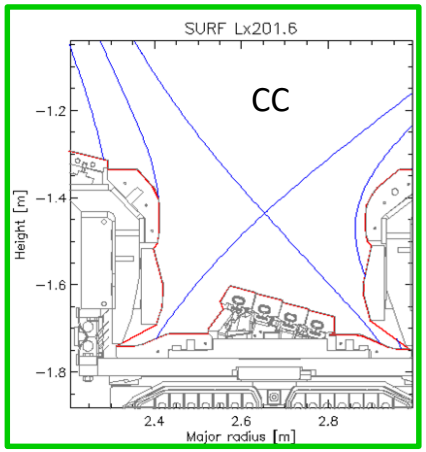
Divertor spectroscopy shows consistently higher isotopic ratio at tile 1

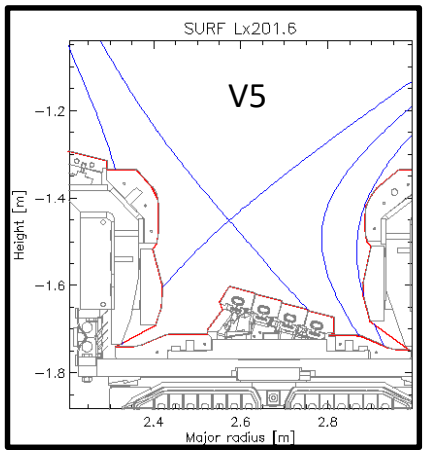
Diverted plasma configurations

Alternating different configurations for optimal cleaning







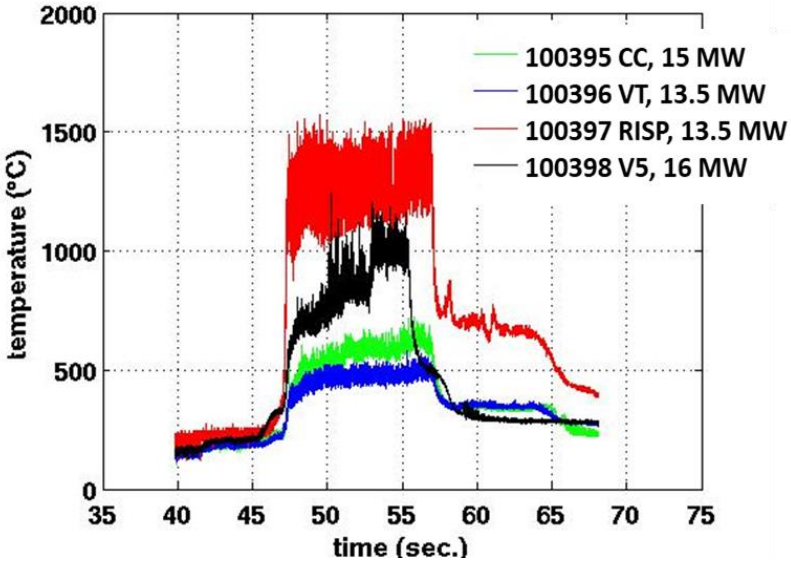


Correlation with neutron rate observed:

→ higher T_{surf} on Tile 1

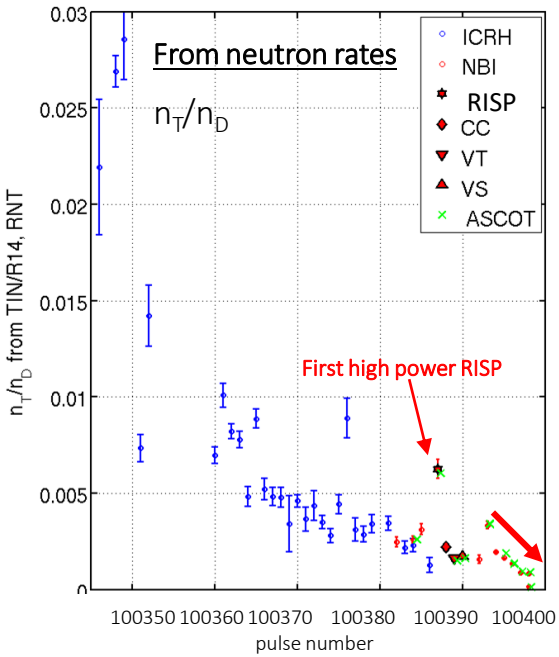
→ higher T release

→ higher 14 MeV neutron rate



highest T_{surf} on tile 1 (over 1200°C) with raised inner strike point configuration (RISP)

Isotopic content in diverted plasma

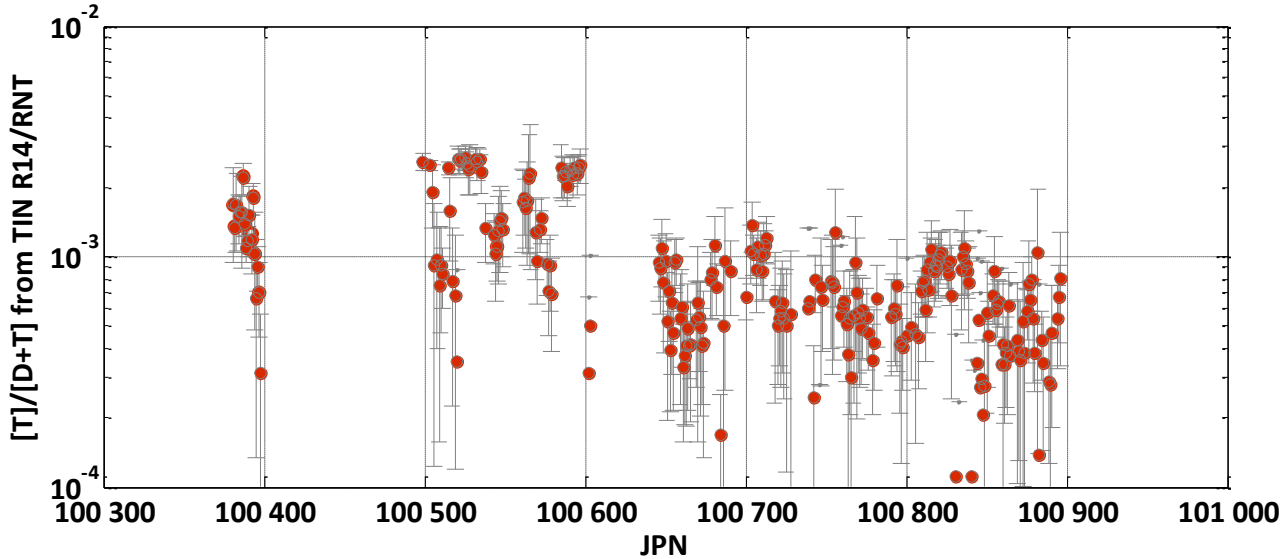


inferred from DD and TT neutron ratio

T/[H+D+T] well below 1% target at the end of clean-up sequence

Further decrease of the isotopic ratio after initial T clean-up sequence

averaged over NBI time
P_{TOT}>4MW, min. 1 MW NBI blips for neutrons

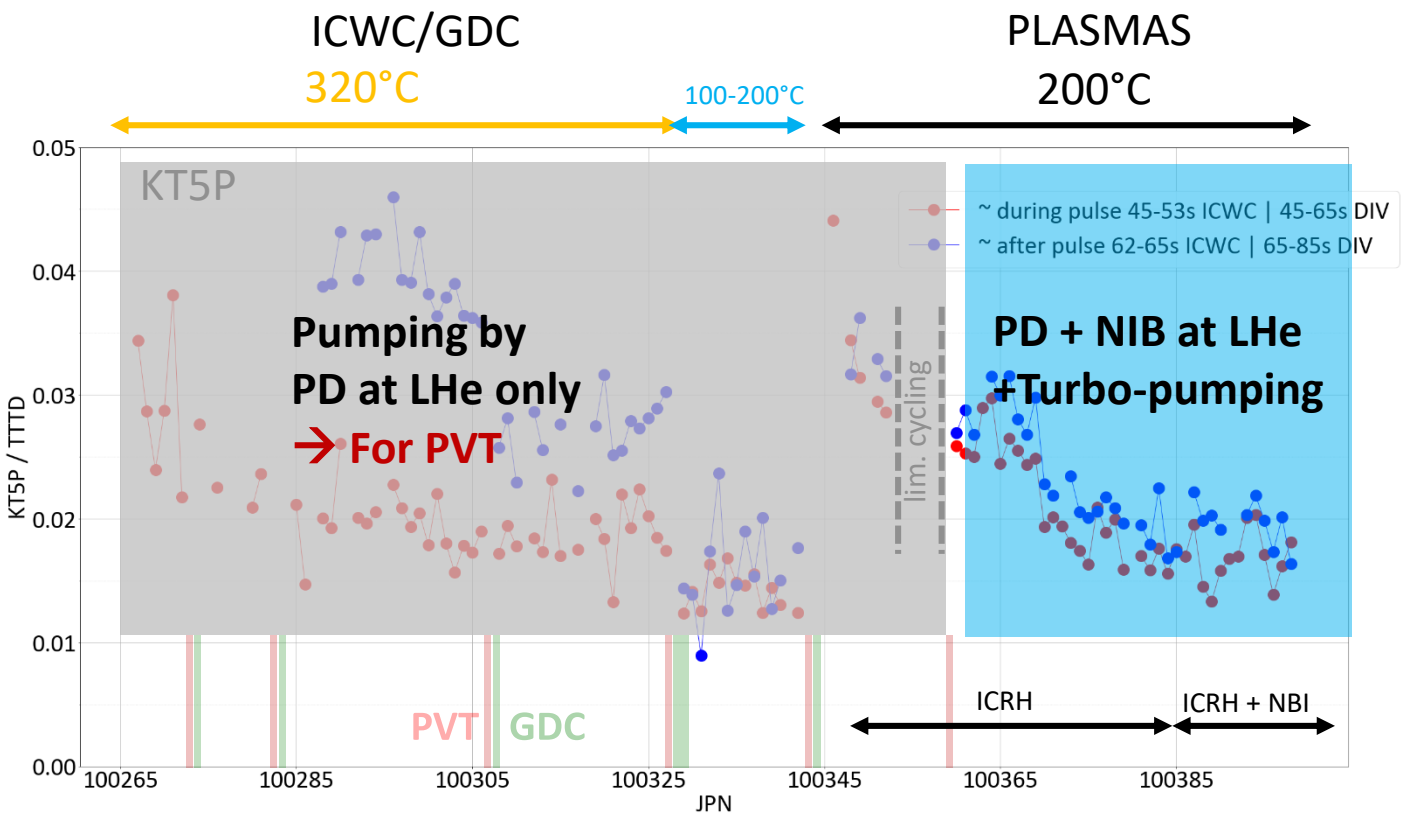


Quantification of T release by PVT

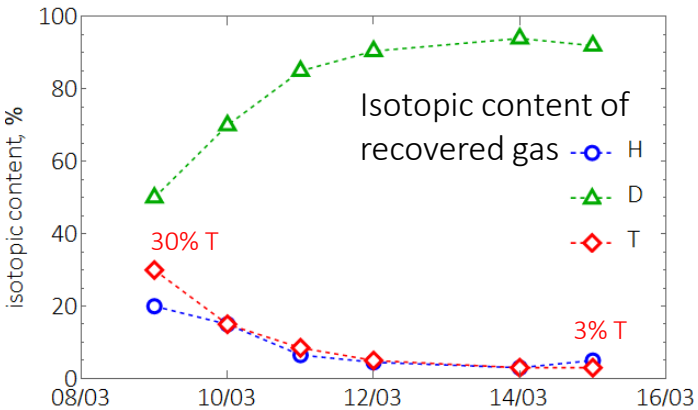
- **alternative technique to gas collections with AGHS**

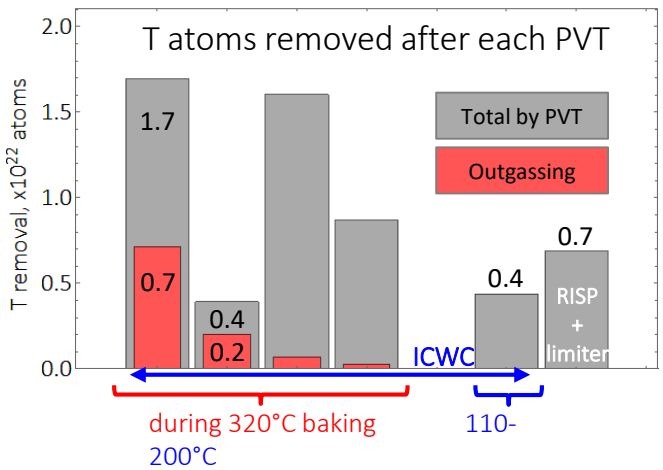
→ Pressure-Volume-Temperature (PVT) and Residual Gas Analysis (RGA) of the gas released after Divertor cryopump regeneration

→ Developed and qualified in C40 for gas balance/T-removal experiments



Quantification of T release by PVT





Systematically more gas collected than injected at 320°C

→ Contribution of outgassing at 320 °C

- 40-50% of collected gas during first 2 PVT days
  → almost no ICWC due to technical issues
- < 5% of collected gas later
  → full D_2-ICWC days & decrease of outgassing with time

Total T removal from RGA-PVT:	**5.7×10^{22} at.**	(~×2 vs baking alone)
Attributed to outgassing at 320 °C:	$(1.0 \pm 0.2) \times 10^{22}$ at.	18%
By ICWC alone at 320 °C:	$(3.6 \pm 0.2) \times 10^{22}$ at.	63%
By ICWC alone at 110-200 °C:	0.4×10^{22} at.	7%
By first 4x RISP + 4x limiter plasmas:	0.7×10^{22} at.	12%

Total by outgassing (baking alone): $(2.7 \pm 0.6) \times 10^{22}$ at.

Summary – Clean-up

- T removal by different methods

Changeover	Total	ICWC+GDC / Baking extr.*
D → H (2020)	5×10^{23}	~1.8
T → D (2022)	2×10^{23}	~3.3

For baking, GDC, ICWC and plasmas

- Faster reduction of ICWC plasma isotopic content for T than for D removal
- T accounting and long term retention is work in progress
- Implications for ITER

→ Baking seems to be less effective for T than for D :

- either there is less "mobilizable" T (e.g. stronger binding) or it is released faster

→ ICWC and GDC promote extra removal for both cases: D and T removal

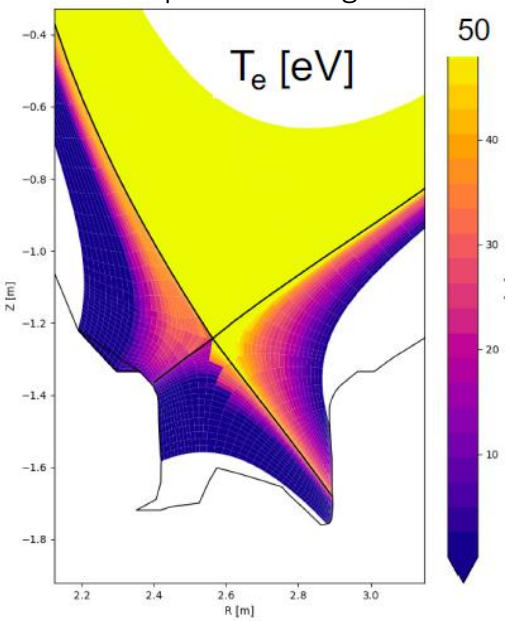
→ Available experimental and modelling data suggest efficient fuel removal from co-deposited layers in the inner divertor by RISP plasma → the role of re-deposition still to be assessed

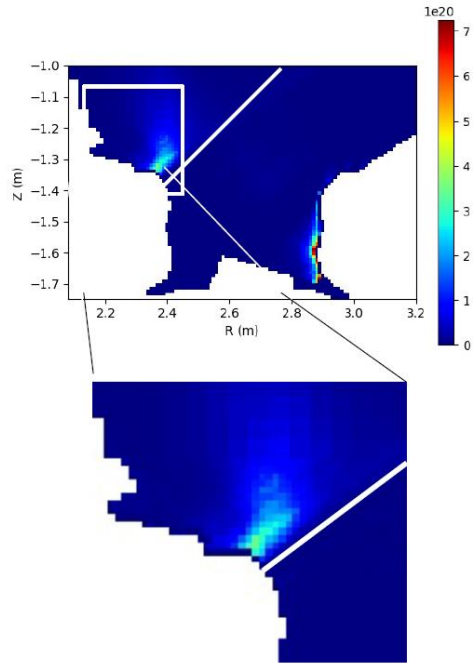
Modelling of erosion

Erosion of co-deposited layers in raised strike point plasma configuration

Be erosion rate (by D^+) at inner strike point for $n_e = 1.5 \times 10^{20}$ m^{-3} and T_e = 5-10 eV → ~10-100 nm/s !
→ 20 µm layer can be completely removed in 20-200 10s-long pulses (if no re-deposition takes place)
→ complete fuel release by erosion possible

OSM-EIRENE for plasma background





Tomographic reconstruction of Dα emission for validation of the model using synthetic diagnostics

Courtesy M. Groth

→ OSM-EIREINE PBG used in ERO2.0 to predict Be net erosion/re-deposition

Content

1. **JET operation ILW/DTE2:**
   - Tritium gas injection and T-NBI in DTE2
   - JET's Active Gas Handling System
   - Operational Budgets and Pattern for the DTE2 and Clean-Up Campaigns
     - Tritium, H isotopes and 14 MeV neutron budget for DTE2 / Clean-up
   - JET diagnostics for isotopic content in plasma
2. **Retention**
   - Retention & cleaning in DTE1
   - D fuel Retention in ILW
   - Gas balance in T
3. **Clean-up**
   - T removal strategy after DTE2 and TT
     - Baking, ICWC/GDC, RF-heated plasma and NBI heated plasmas
     - Raised Inner Strike Point Configuration
     - Quantification of T release
   - Modelling
4. **Summary – Lessons for ITER**

Summary

- JET has exploited its **unique capability for using T and DT** as main fuel in an ITER-like wall environment
- **Key information** obtained in a broad range of physics areas for improving predictions for ITER
  - See e.g. [Mailloux, Maggi, Garcia, EPS 22]…

- **T accounting and long term retention** is work still in progress
  - No major change on fuel retention wrt. to D ops seen so far
  - Need however additional data to confirm
  - T accounting still on-going (Product Storage still not fully regenerated !)

- **T removal** from Vacuum Vessel by combination of available techniques, targeting different areas
  - faster reduction of plasma isotopic content for T vs D removal
    - extra T removal by low Te GDC/ICWC plasmas
  - Access to thick T-rich Be layers with RISP in top of Inner divertor evidenced

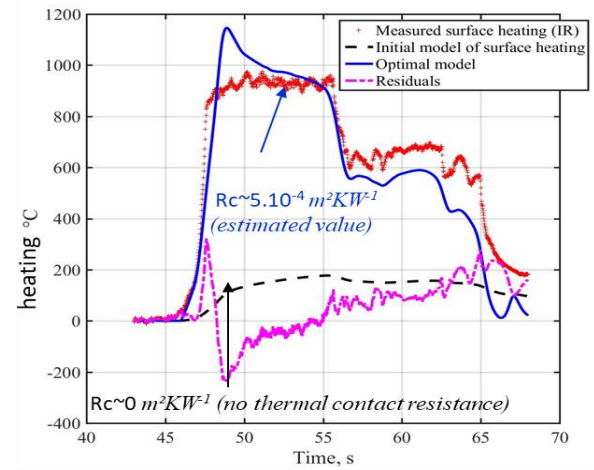
Raised strike point scenario in ITER

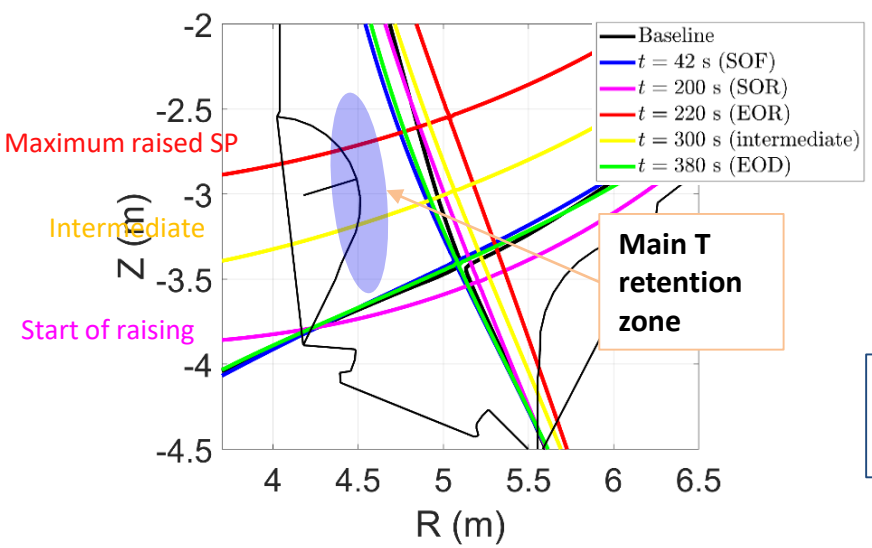
- Heating of divertor deposits in ITER assessed using "RISP-like configuration"
- Using thermal thermal resistance R_c of JET divertor deposits assessed in RISP plasmas

- 10 MA deuterium L-mode (intermediate case, 20 MW EC heated)
- Monoblock temperature estimated via SOLPS-ITER heat load
  - Layer temperature estimated using JET R_c estimate

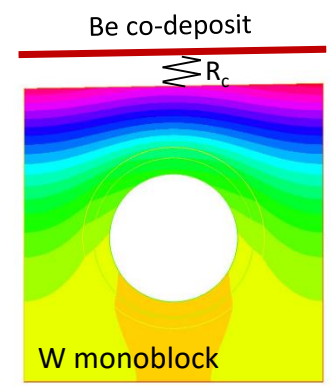
$$\rightarrow T_{Be}^{pk} = 750 - 1050\ ^{\circ}\mathrm{C}$$

R_c estimation from IR surface temperature and estimated heat flux from TC





[T. Wauters, PSI-25]



Heating of divertor co-deposits >800°C using raised strike points for fuel outgassing is feasible in ITER*