IAEA Technical Meeting on Plasma Physics and Technology Aspects of the Tritium Fuel Cycle for Fusion Energy

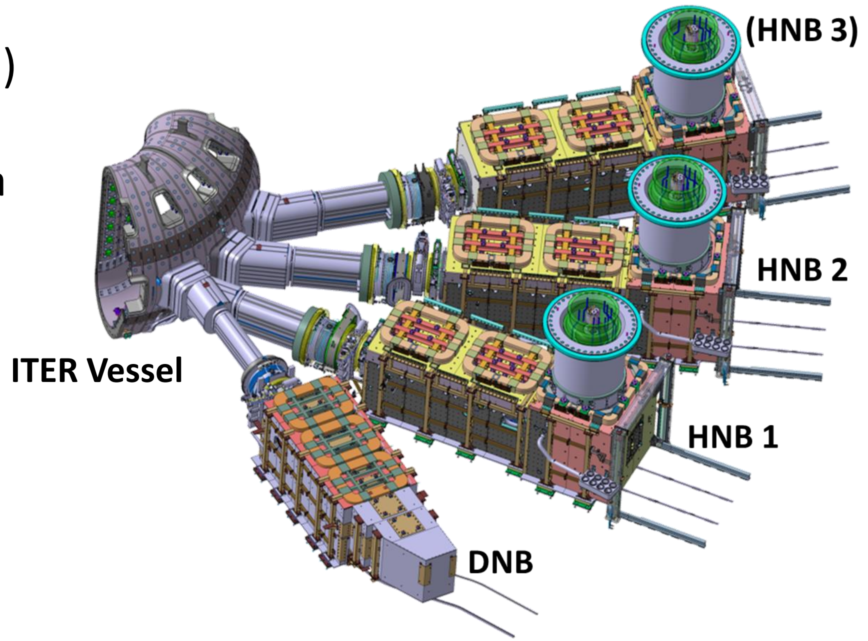
Neutral beams and the requirements they place on the fuel cycle

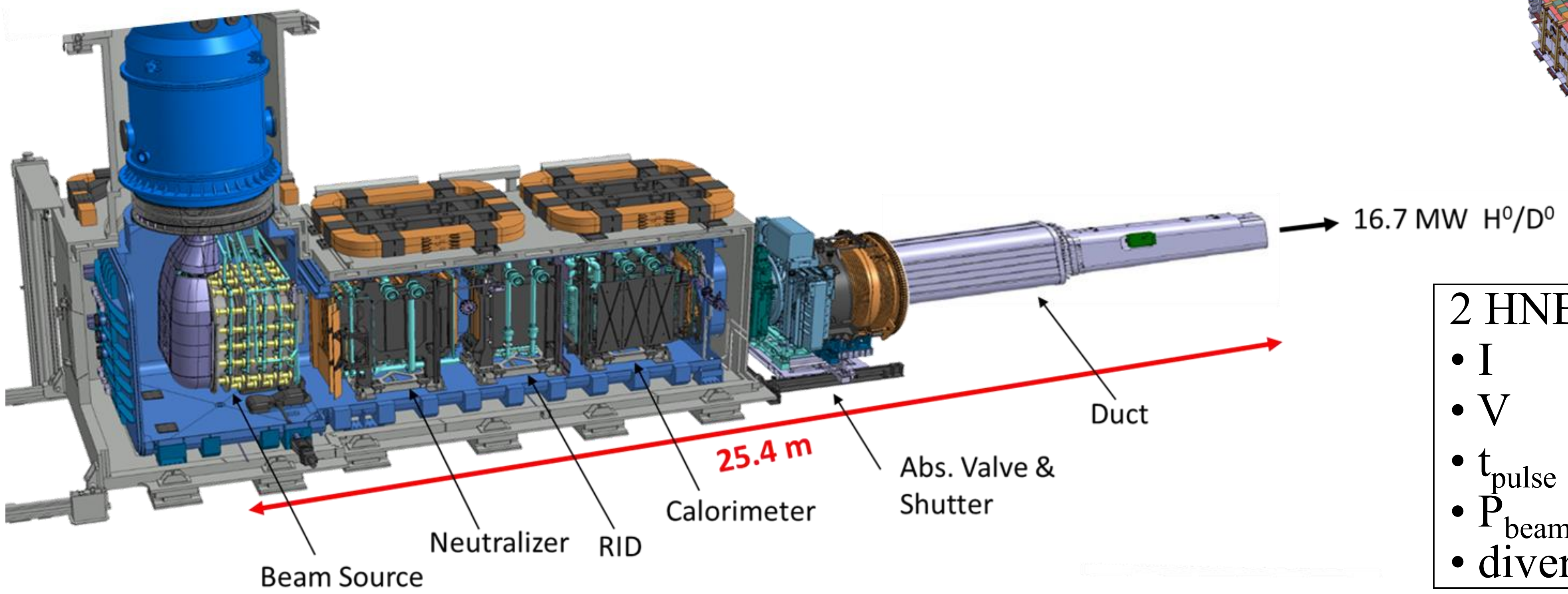
P. Veltri, NB Section
ITER Organization

Special Thanks to:
U. Fantz, B. Heinemann (IPP), D. Boilson, S. Willms, (IO), E. Sartori (RFX), R. Hemsworth, for fruitful discussions.
A. Litvinov, K. Roux (IO) for CAD support

Introduction

- Neutral are the beams main plasma heating method for present fusion device (They also provide current drive)
- Atom beam production based on the conversion of ions in a gas cell ("Neutralizer")
- For large fusion device like ITER, 1 MeV necessary to access the core of the plasma
- Requirements for ITER NBI extremely demanding

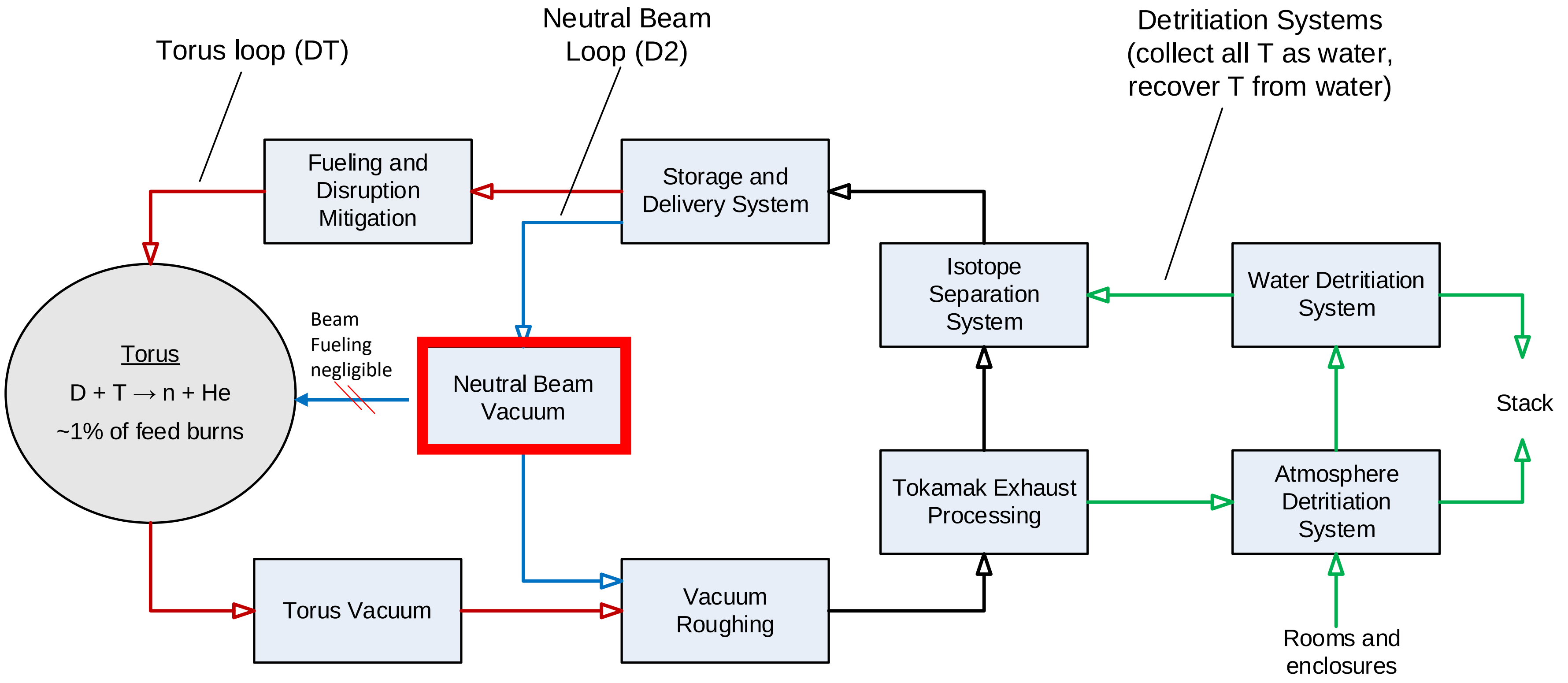




2 HNBs (+1): H_2 / D_2
- I = 46 / 40 A
- V = 0.87 / 1 MV
- t_{pulse} = 1000 / 3600 s
- P_{beam} = 16.5 MW
- divergence ≤ 7mrad

Introduction

- Fuel Cycle consists of vacuum, tritium processing and fueling technologies
- Deuterium-Tritium is circulated through the reactor
- **Deuterium is circulated through heating beams**
- Tritium is recovered from water and gases



Fluxes of Gas in the beamline

- D_2 gas in injected in the ion Source to ignite a RF plasma
- D_2 gas in injected in the Neutralizer to convert the ions into neutrals (gas cell)
- D_2 / T_2 gas recycling from tokamak was estimated to be up to 4e20 atoms/second/m2, corresponding to a throughput of 0.88 Pa m3/s (50% T , 50% D) at duct entrance
- Main Concern: the Purity of Gas in the Source, depending on all the above
- Present Requirement: 200 ppm (0.02%) T_2 in D_2

2.2.2 Gas quality and quantity requirements

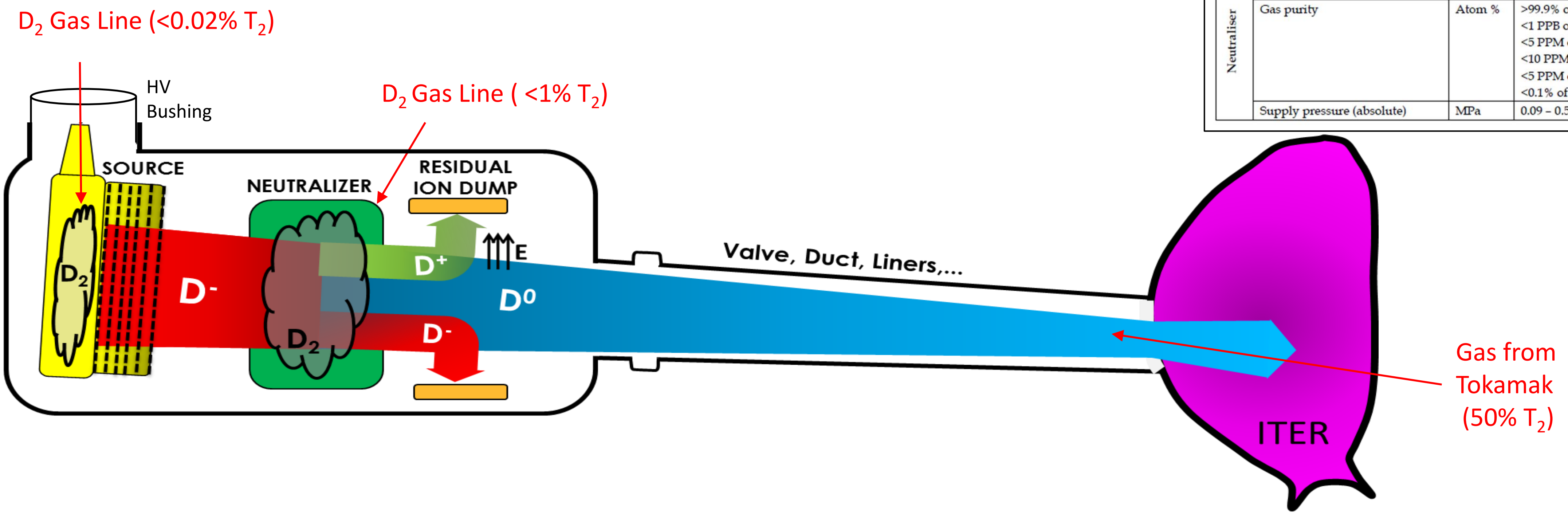
[5301s610-R] The gas requirements that shall be considered for one NB Heating and CD System injector for the two phases of operation.

[5301s611] **Table 2.2: Gas requirements for one NB Heating and CD System injector for the two phases of operation** (1)

[5301s718]

(1) Refer to interface sheet for exact data. Gas purity requirement to be confirmed by NBTF operation.

	Parameter	Unit	H/He Phase	DD/DT Phase
Ion Source	Gas type		H_2	D_2
	Gas flow	Pam^3/s	5.1	3.6
	Gas purity	Atom %	>99.999% of H_2 <1 PPB of ^3H <1 PPM of O_2 <10 PPM of N_2 <1 PPM of H_2O <0.001% of other gases	>99.7% of D_2 <200 PPM of ^3H <1 PPM of O_2 <10 PPM of N_2 <1 PPM of H_2O <0.001% of other gases
	Supply pressure (absolute)	MPa	>0.5	>0.5
Neutraliser				
	Gas flow	Pam^3/s	43	19
	Gas purity	Atom %	>99.9% of H_2 <1 PPB of ^3H <5 PPM of O_2 <10 PPM of N_2 <5 PPM of H_2O <0.1% of other gases	>99% of D_2 <1% of ^3H <5 PPM of O_2 <10 PPM of N_2 <5 PPM of H_2O <0.1% of other gases
	Supply pressure (absolute)	MPa	0.09 – 0.5	0.09 – 0.5



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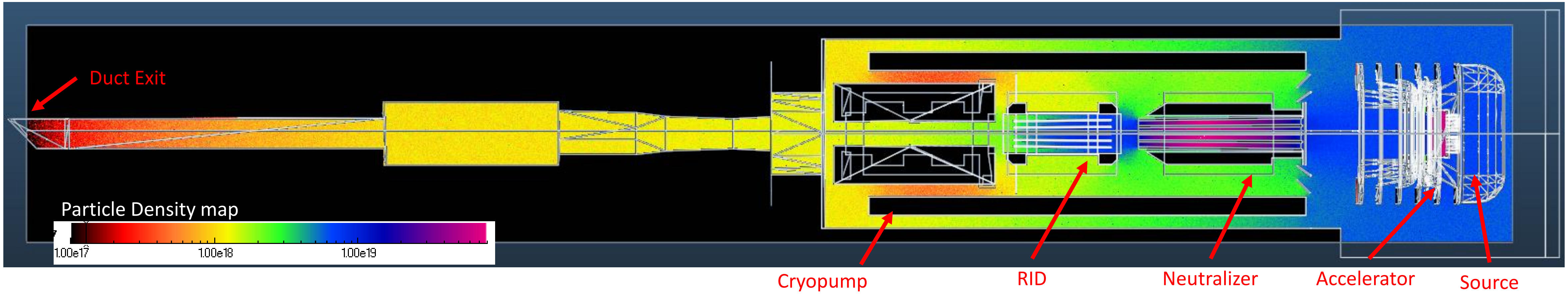
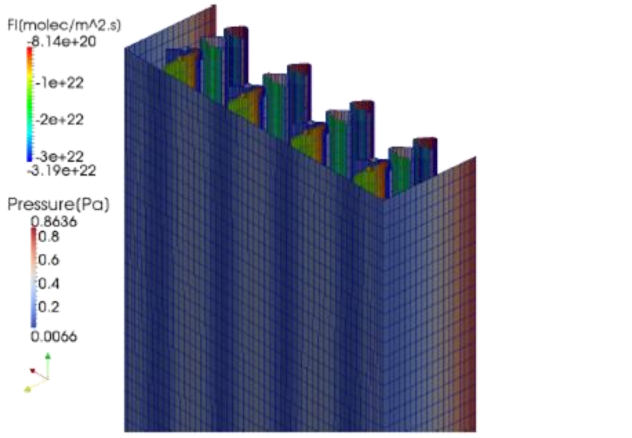
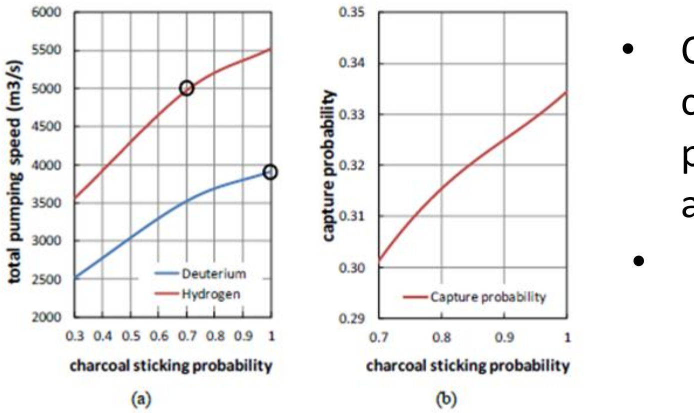
- Main Concern: the Purity of Gas in the Source, depending on all the above
- Can be Calculated by Montecarlo Codes (molecular regime)

	D_2	T_2
Pump Capture Coefficient	0.3	0.25
Qsource (Pa·m3 / s)	3.6	0.00072
Qneutralizer (Pa·m3 / s)	19	0.19
QTokamak	0.44	0.44





Horizontal section - rear view of the simulated cryomodule, colours by flux of impinging particles at pumping surfaces and pressure at solid surfaces.



(a) Pumping speed and (b) capture probability of the cryopanels calculated with Avocado model as a function of the charcoal sticking probability

- Cryopumps too complex: one can use a sub model to obtain the capture probability [1] starting from the geometry and the sticking probability of charcoal, known from experiments[2]→ about 0.3, and scales with mass
- In addition to cryopumps the duct exit toward ITER VV is also pumping, with **sticking=1**

[1] E. Sartori et al. RFX-MITICA-TN 147 rev2 (2014)
[2] M. Dremel, C. Day, S. Hanke, X. Luo, Cryopump design development for the ITER Neutral Beam Injectors, Fusion Engineering and Design 84 (2009) 689–693

Fuel

D_2 / T_2 Distribution Along the Beamline

- T_2 From Tokamak Negligible
- T_2 From Source filling line reflect the set purity (200 ppm)
- T_2 From Neutralizer contribute to 600 ppm in the ion source
- Source-Neutralizer Fluxes could be slightly different
- T_2 from Neutralizer 0.05-0.1% > 0.02%

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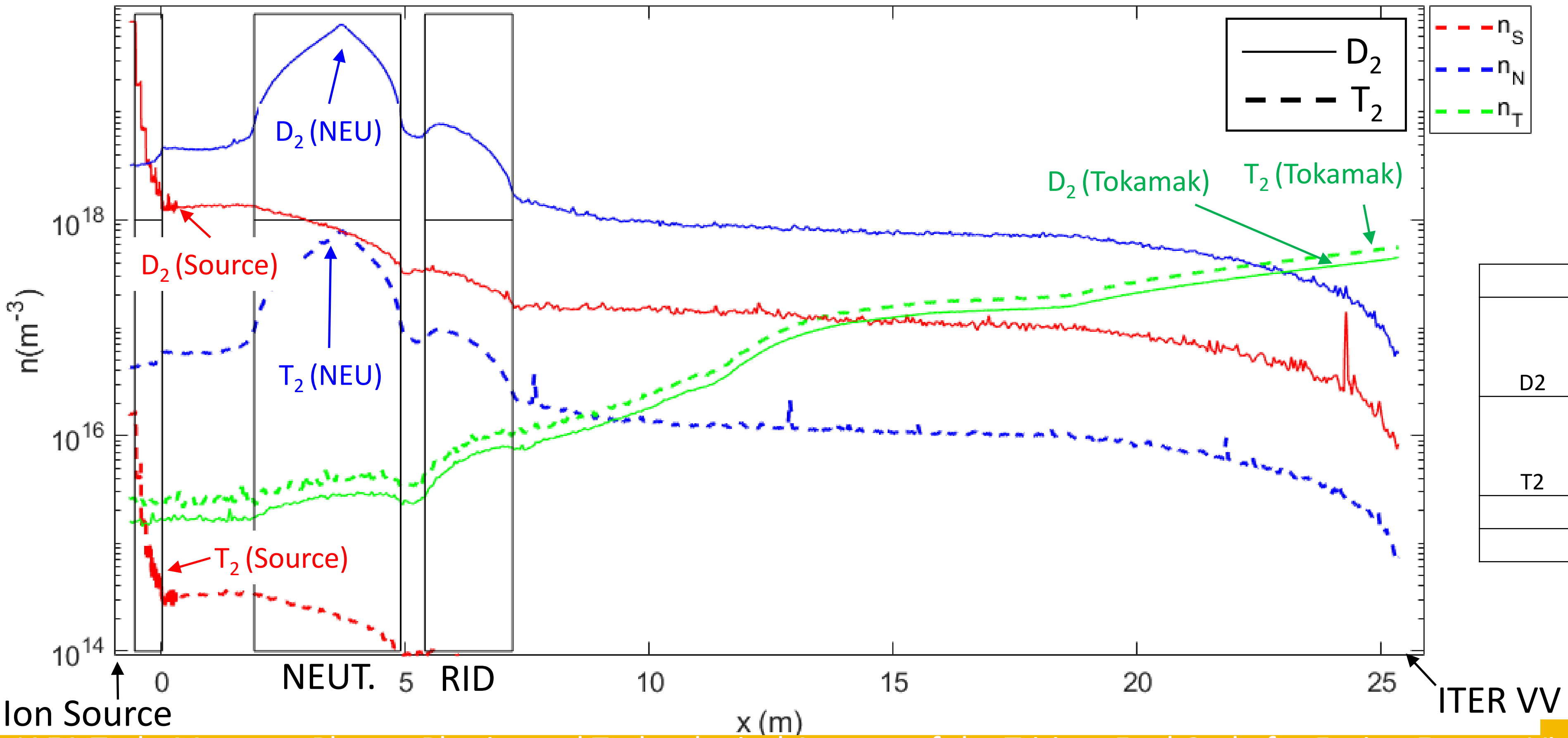
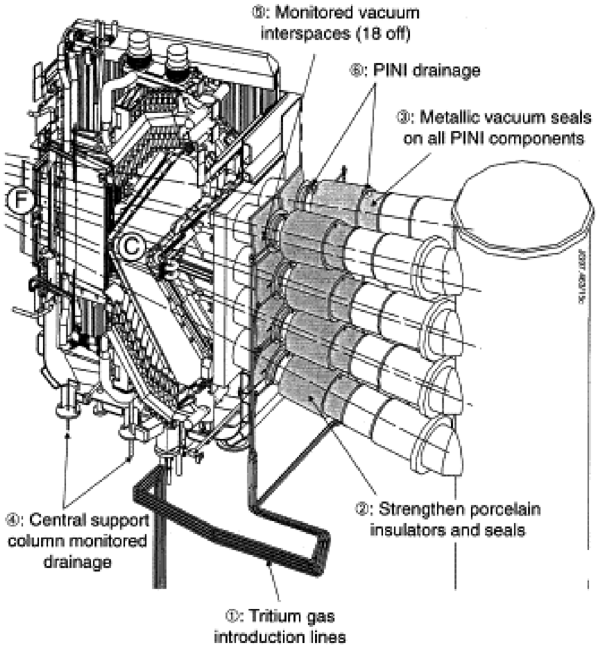


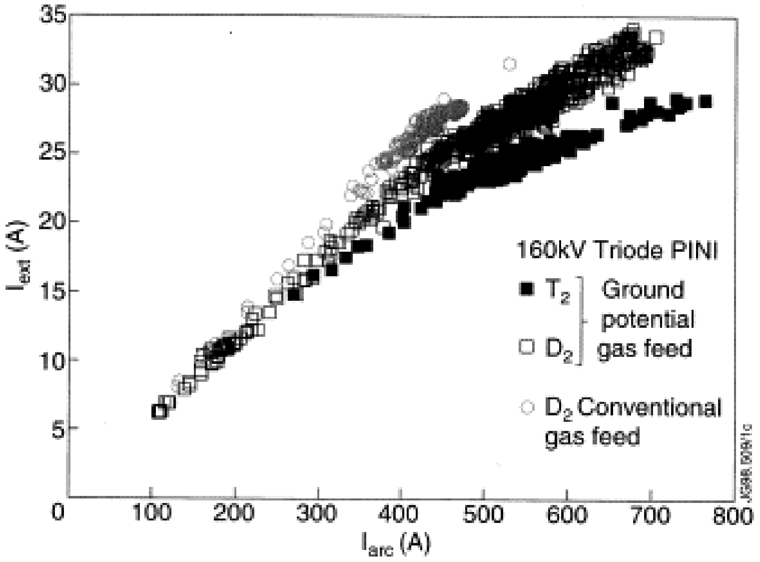
		Density (1/m3)	%
D2	From Source	6.90E+19	95.09%
	From Neutr.	3.50E+18	4.82%
	From Tokamak	1.50E+15	0.00%
T2	From Source	1.50E+16	0.02%
	From Neutr.	**4.30E+16**	**0.06%**
	From Tokamak	2.60E+15	0.00%
	ToT	7.26E+19	

Is this an issue?

Tritium traces in Positive ion sources

In principle the use of T instead of D/H affect the source performances (arc efficiency,…), but not at low fraction





T.T.C. Jones et al. Fus. Eng.and Des. 47 (1999) 205–231

Prior to D-T experiment at JET, a campaign with T traces in D2 (1% doping) were carried out at JET positive ions injectors of D_2 [1]

Result:

“PINI operation with such a gas mixture did not require any specific commissioning, since at such a low tritium concentration the PINI operating characteristics are indistinguishable from those of pure deuterium.”

T. Jones et al. JET report: JET–P(99)08

BUT

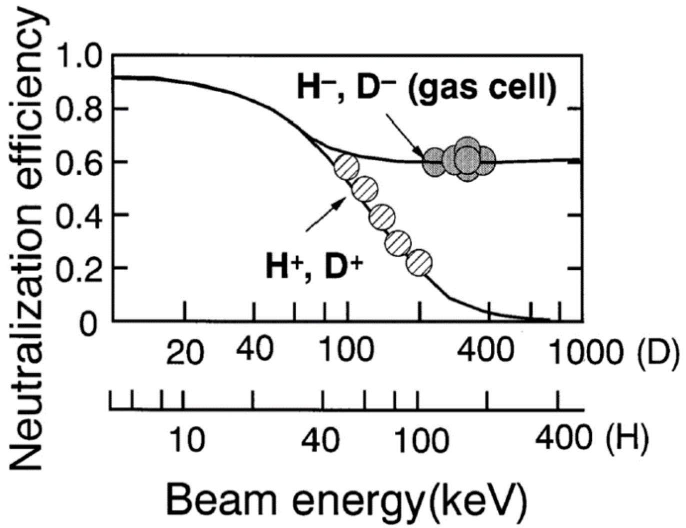
Of course the physics involved in positive ions sources is much simpler than for H- sources….

Negative ion sources: Why?

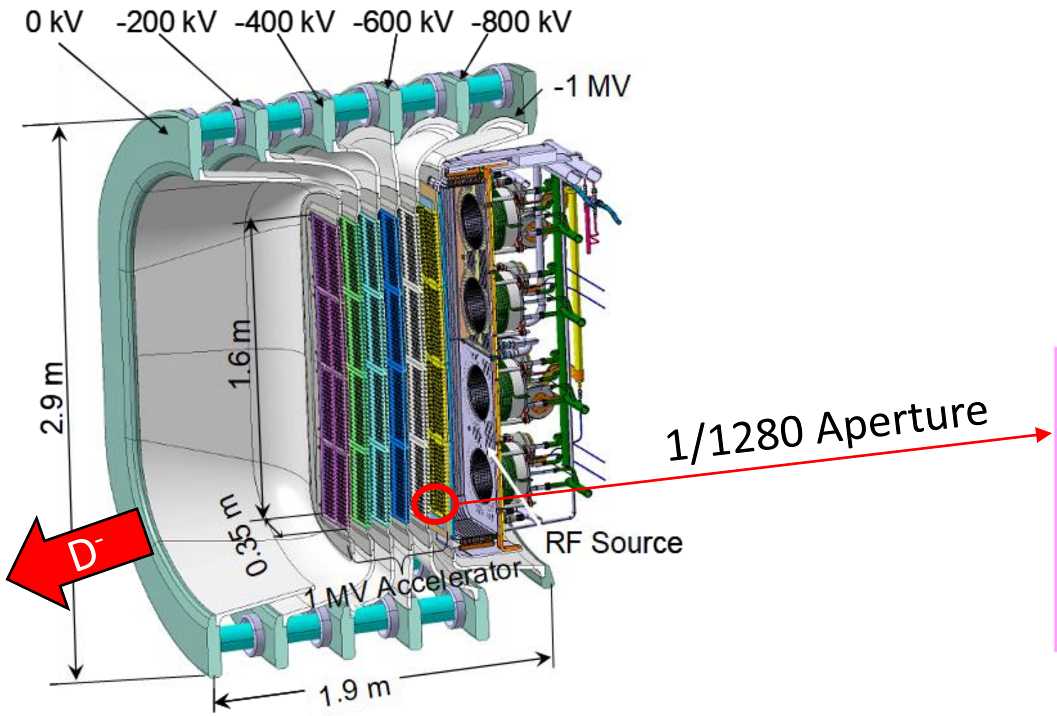
- Large Fusion devices requires the beam to travel long distances into the plasma
- In the case of ITER, efficient beam absorption requires D^0 Energy in the 1 MeV range
- Negative ions are mandatory!

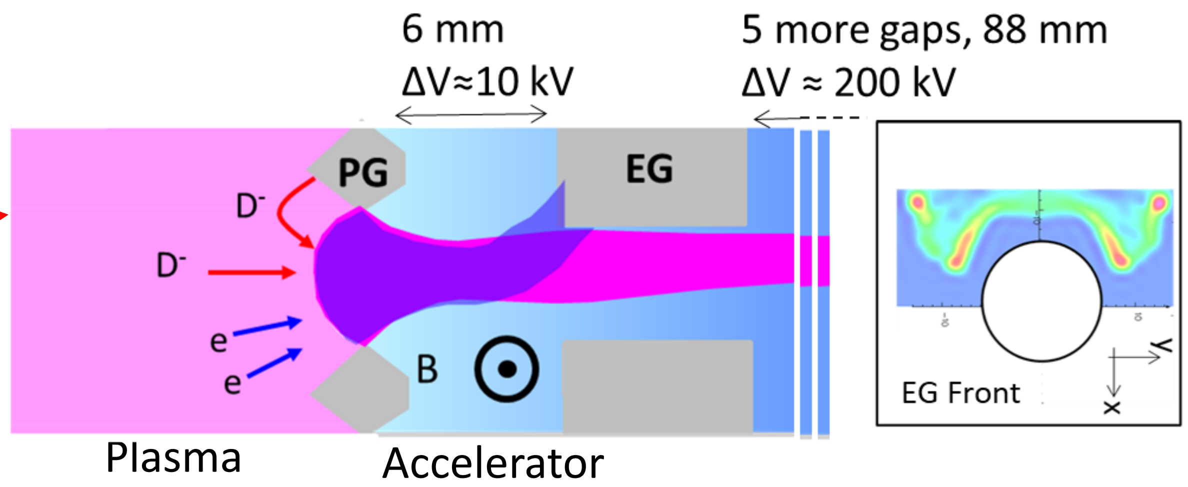
Main Consequences:

- H-/D- generation much more challenging than H+/D+
- H-/D- are fragile, destroyed by hot e- in the plasma
- Their extraction from the source is accompanied by Co-extracted electrons (see later)



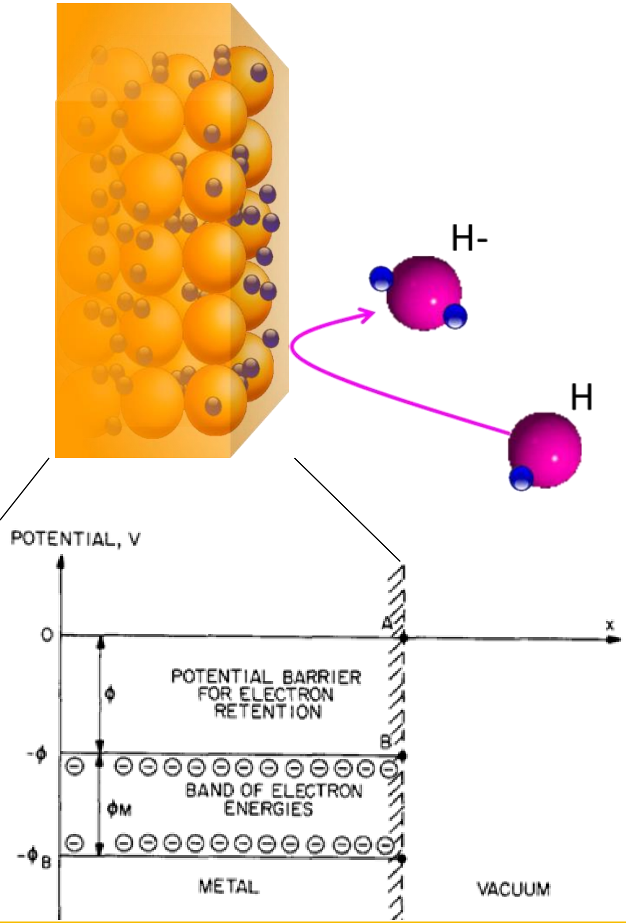
Experimental Measurement at JT-60U
[M. Kuriyama, Fus. Eng. Design, 39-40:115, 1998]



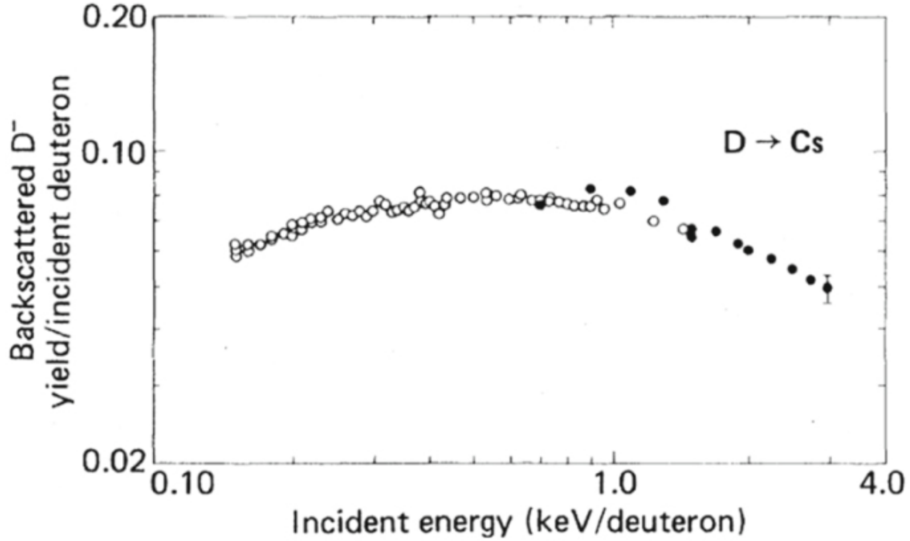


Negative ion Generation

- Negative ions are (mainly) created by surface conversion of atoms / positive ions on plasma facing materials
- Need electrons! Electrons in a solid are confined by the ions charge and cannot escape
  - When the electron affinity E_a of the atom is $E_a > \phi$ (work function) the probability that an electron is captured from the surface and a negative ion is formed is enhanced:



$$\beta^- = \frac{2}{\pi} e^{-\frac{\pi(\phi_W - E_A)}{2 a v_z}}$$

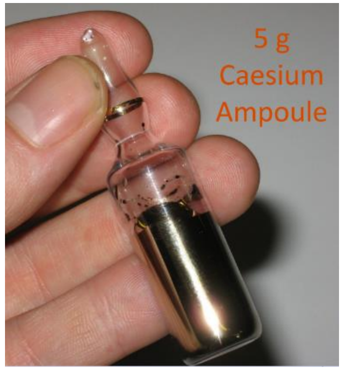


M. Bacal and M. Wada, Plasma Sources Sci. Technol. 29 (2020) 033001

- E_a of H- / D- is around 0.75 eV
- Coverage of alkali metals on source wall lowers the work function
- Cs has the lowest work function (2.2 eV) and is the best electron donor

Negative ion sources: Cs dynamic

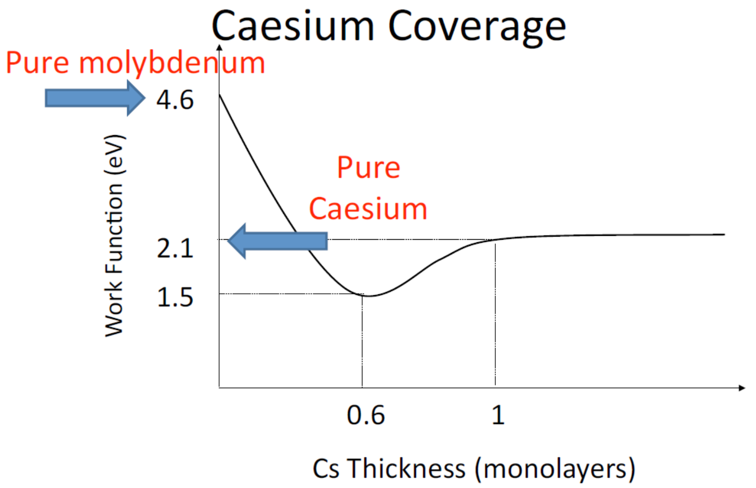
- Small quantity of Caesium vapor injected in the source by temperature controlled Cs ovens
- Caesium is transported and re-distributed by the plasma
- The work function of surfaces (PG electrode, in Moly-coated Cu) is reduced (≈pure Cs)
- Negative ion production dramatically enhanced→ “ion-ion plasma”
- <u>Co-extracted electrons reduced</u>

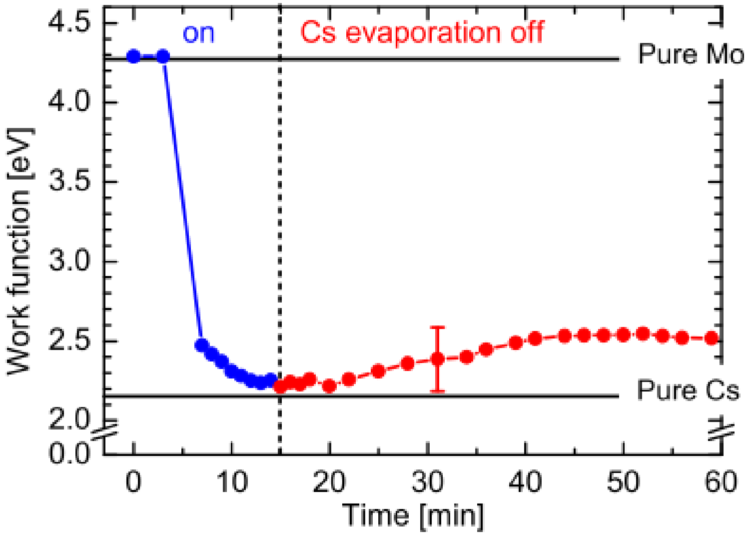


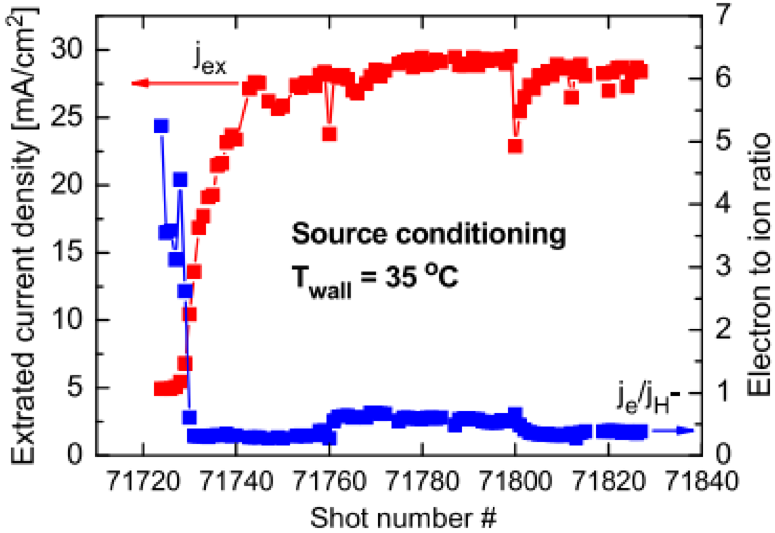
D. Faircloth, ISIS, CERN Acc. School on Ion Sources, 2012

Practical Usage:

- Source Conditioning needed →several plasma pulses to re-distribute Cs
- Plasma grid temperature @ 100°- 250° “, source body temperature at > 35° to avoid trapping of Cs on the walls
- Cs is highly reactive! Requires an impurity-free environment (O_2, …)







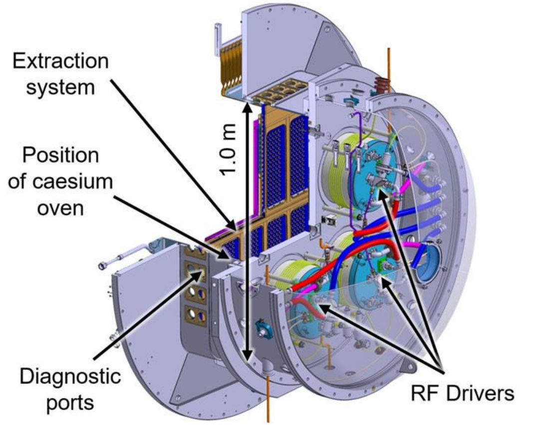
U. Fantz et al. / Chemical Physics 398 (2012) 7–16

Negative ion sources: Isotope Effect

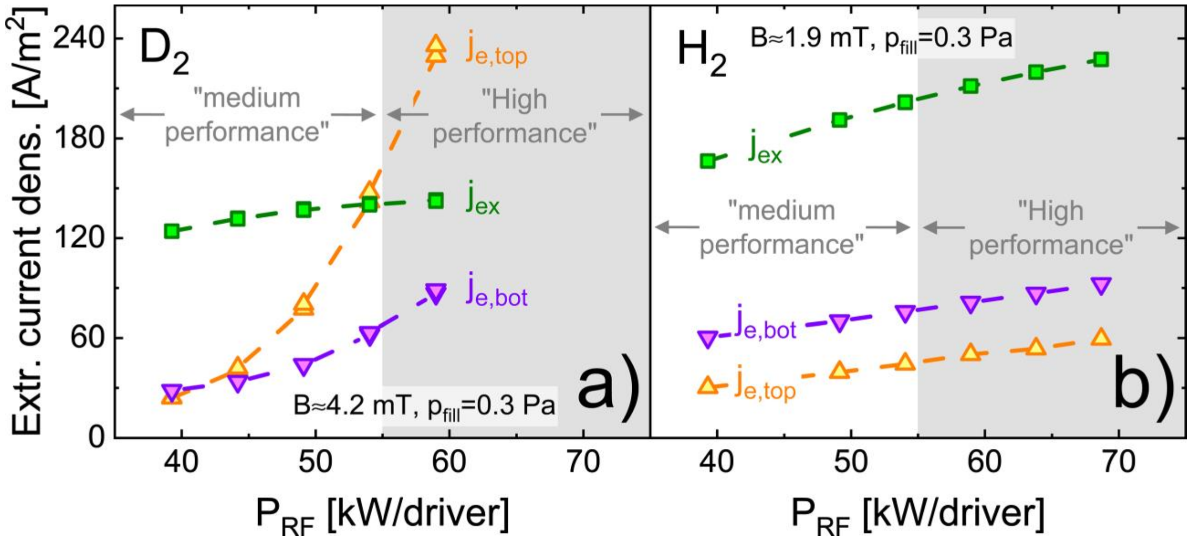
Operation of the sources in D_2 plasma much more challenging than operation in H_2 plasma:

- Co-extracted electron current increases
  - At high power (ITER source to be operated at 80-100 kW/driver)
  - in time (issue for long pulses)
- Saturation of extracted current (j_{EX})
- D_2 Operation accompanied by higher Cs densities in the plasma

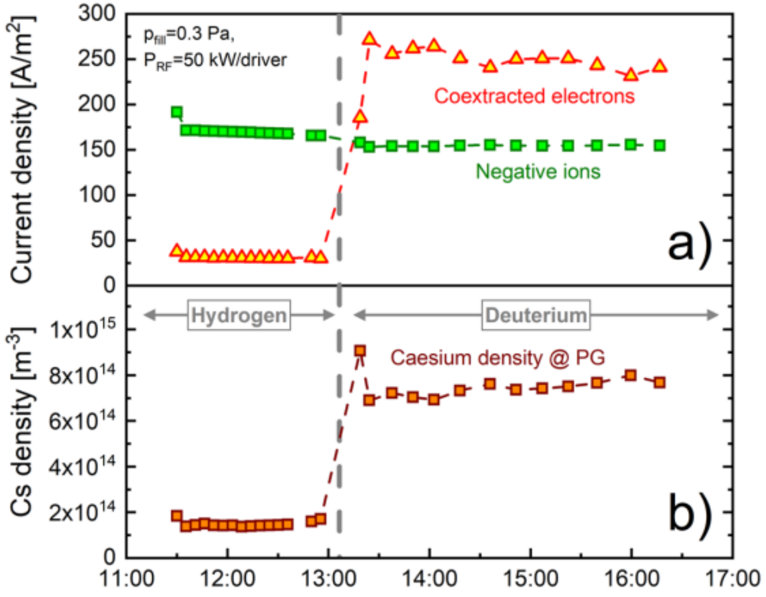
→Root cause seems to be linked with degradation of the Cs layer (enhanced Sputtering?)



Experimental DATA from ELISE ion source @ IPP Garching



*D. Wünderlich et al 2021 Nucl. Fusion **61** 096023*



D. Wünderlich et al. Rev. Sci. Instrum. 90, 113304 (2019)

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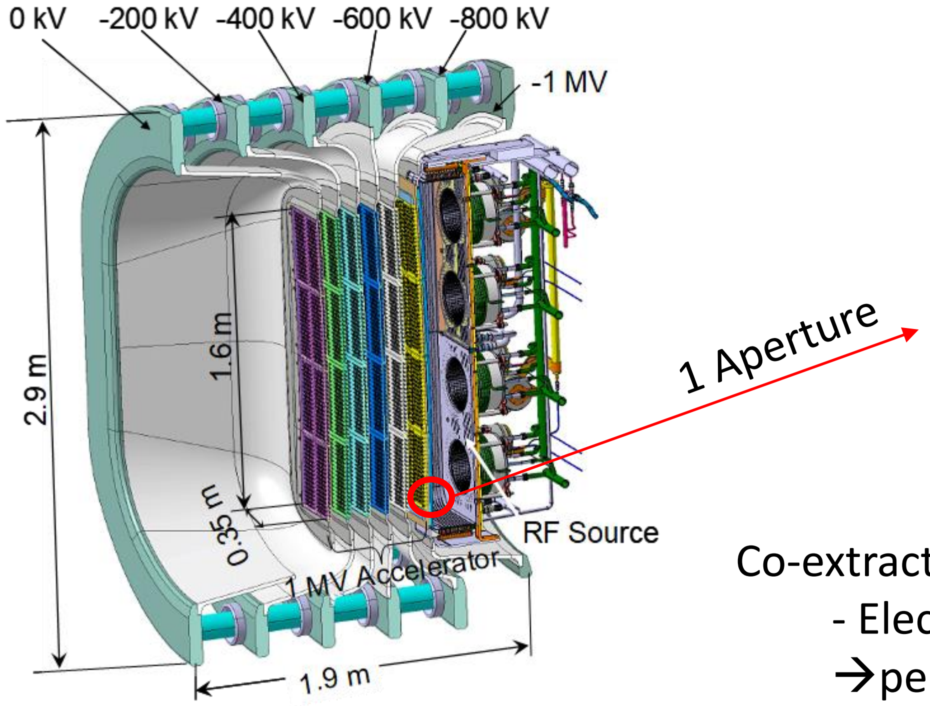
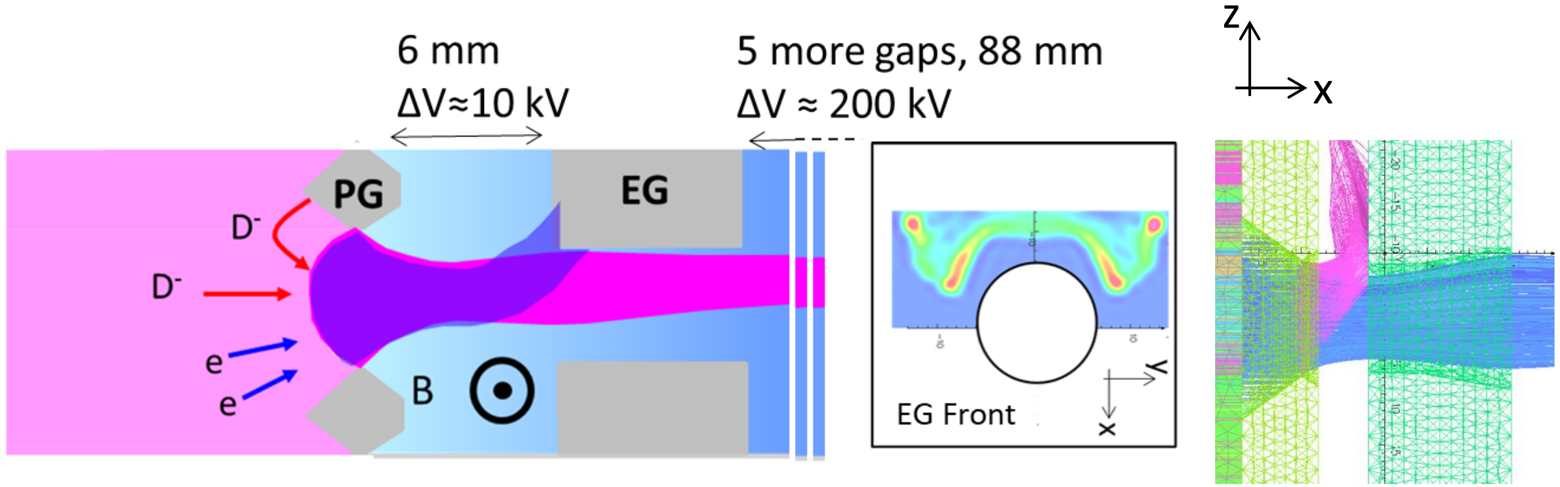




Co-extracted electrons

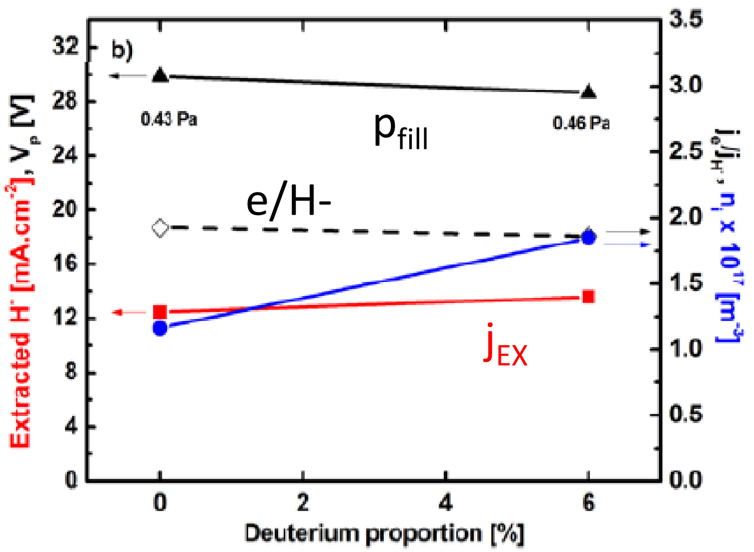
- Electrons need to be dumped ASAP, before they gain too much energy.

→permanent magnets in 1st electrode, called Extraction grid (EG): Load ≈10 MW/m2

Negative ion sources: Isotope Effect

iter

H_2 doping tests at BATMAN test bed in IPP [1]: H_2 plasma with D_2 doping (6%, 15%, 28%)



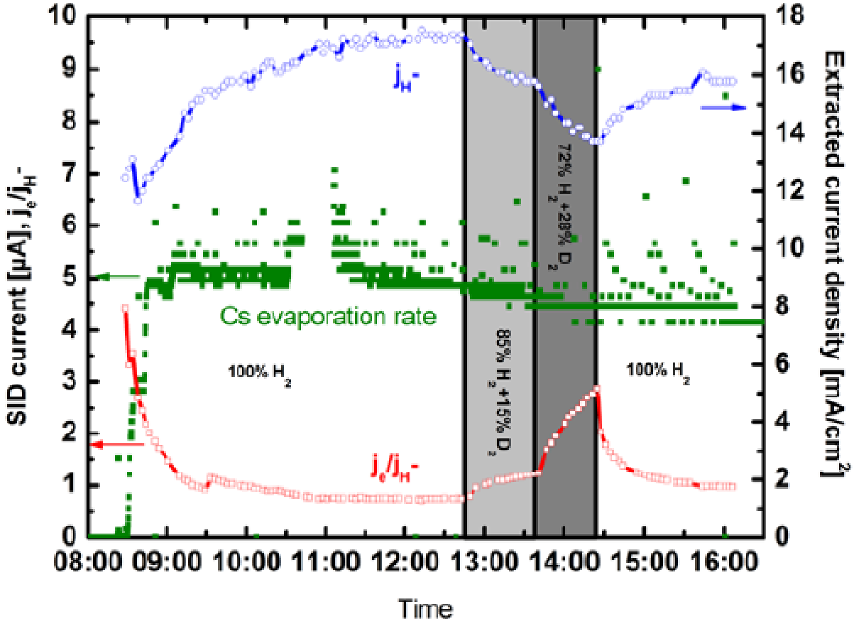


Figure 7: Variation of the extracted H- and the je/jH- as a function of the deuterium proportion during one day. The measured Cs evaporation rate is also plotted.

[1] IPP-FinalReportPRIMA-WP13v1

CAVEATS:

- The study was done for D_2 doping in H_2 (effect of T_2 in D_2 might be different)
- It is at relatively high pressure (test done at 0.4 and 0.7 Pa)
- These results were obtained when the Cs content in the oven was low,

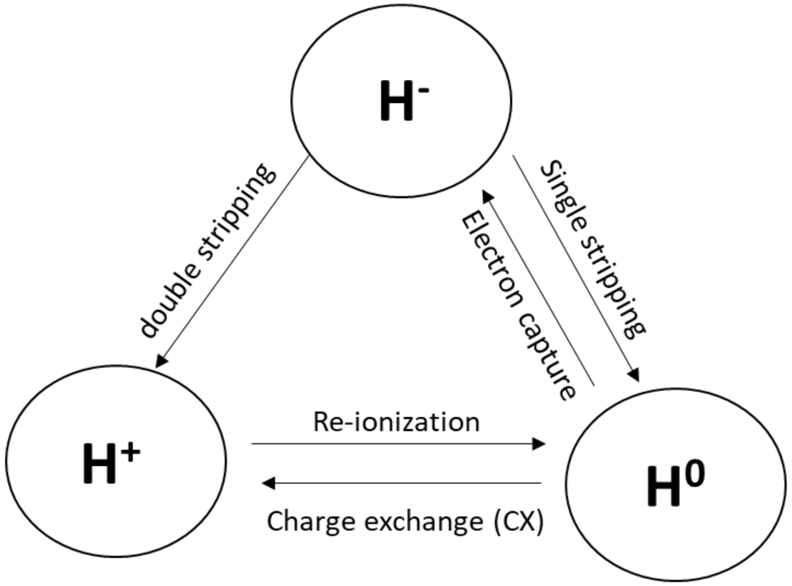
RESULTS

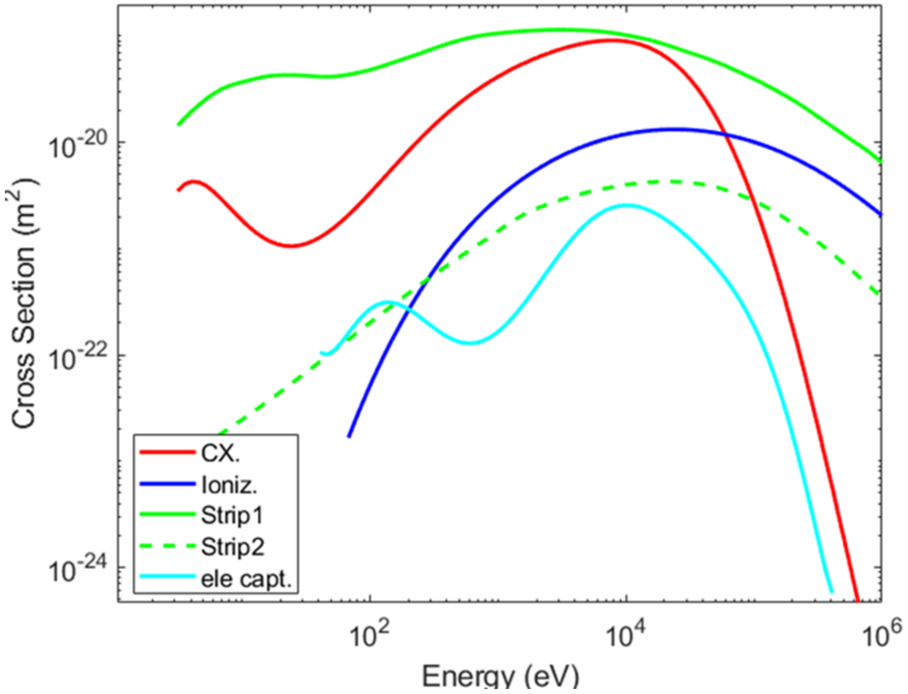
- At 6% experiments were done by adding 6% D_2 to baseline pressure of H_2.→ Effects of doping might be hidden by those related to overall pressure increase (for example: electron current decrease)
- Test with constant pressure were only done of higher D fraction → "A *proportion of 15% of deuterium shows a slight reduction of the extracted negative ion current and a moderate increase of the co-extracted electrons.*"

0.02% Seems really too stringent → Relaxation by a factor 10-20 is acceptable (from now on I will assume 0.2%)

Power Loading Considerations: Beam fraction evolutions

- Assuming equal ionization/dissociation/negative ion yield as per D_2 , a T^- beam having 0.2% of D- current (60 A) is extracted/accelerated
- Beam interact with the background gas in the beamline: losses
- With the gas profile and the cross sections we can evaluate the beam fractions at any point along the beamline





- cross section from ORNL Redbook (Barnet 1990)
- Electron capture process can be neglected (very minor effect) → H- can only be destroyed

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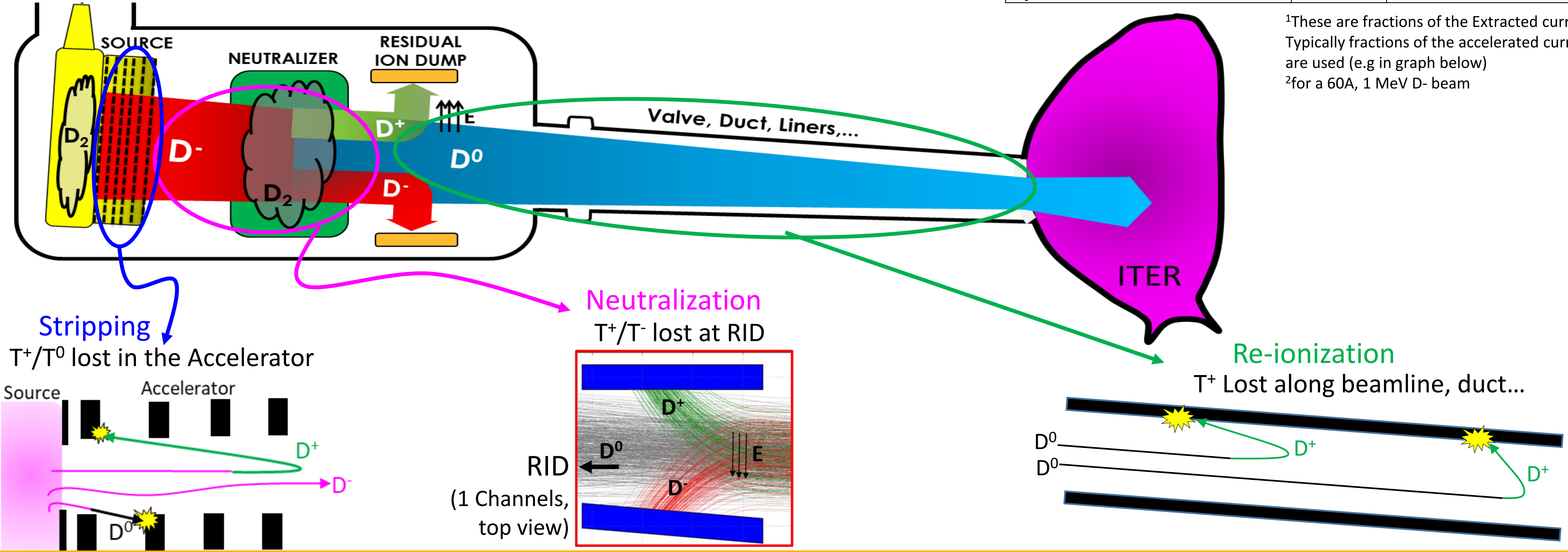
	Fraction[1] (%)	Lost Current/Power[2] (mA/kW)
Extracted Beam Current T-		120
Lost in The Accelerator	34%	40.8
Lost at RID (A)	29%	35.2
Lost In the Duct, …	9%	11.2
Injected in ITER	27%	32.8

[1]These are fractions of the Extracted current. Typically fractions of the accelerated current are used (e.g in graph below)
[2]for a 60A, 1 MeV D- beam



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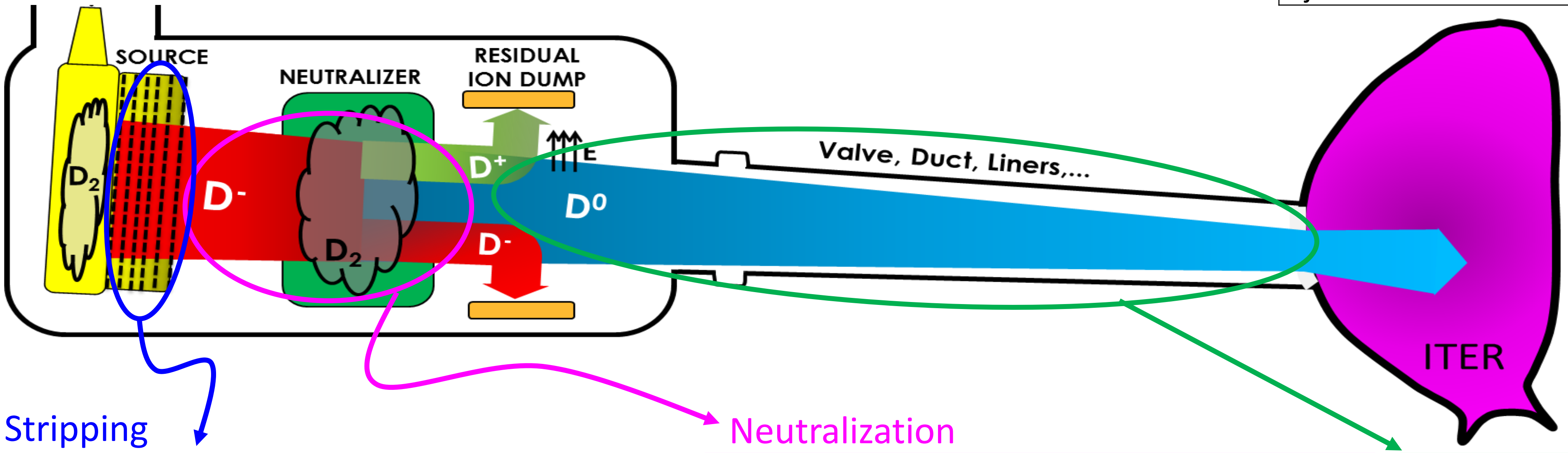
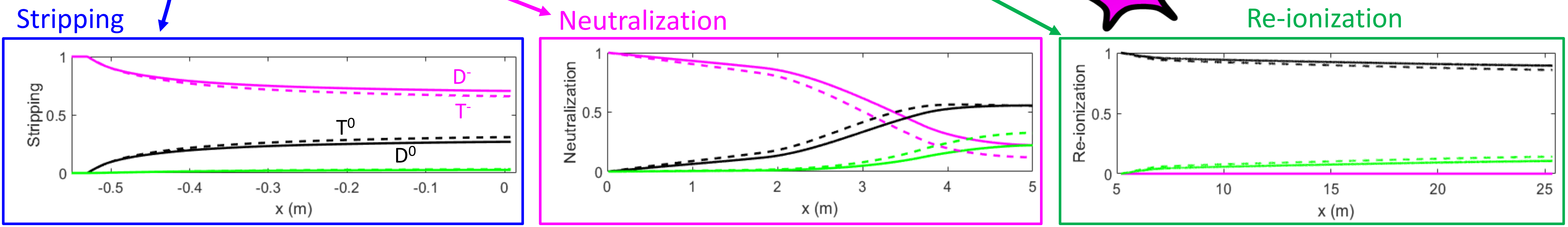
[2]for a 60A, 1 MeV D- beam
[3]This figure shall be intended for the lost current only. The associated power is much less, as a large part of the losses occur at low voltage





Power Loading Considerations: Power on the Duct

Beam interact with the background gas in the beamline: losses

Stripping→ T^+/T^0 lost in Source/Accelerator

Neutralization→ T^+/T^- lost at RID

Re-Ionization→ T^+ Lost along beamline, duct…

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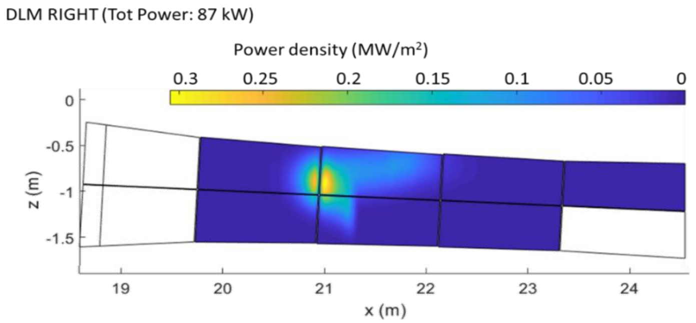
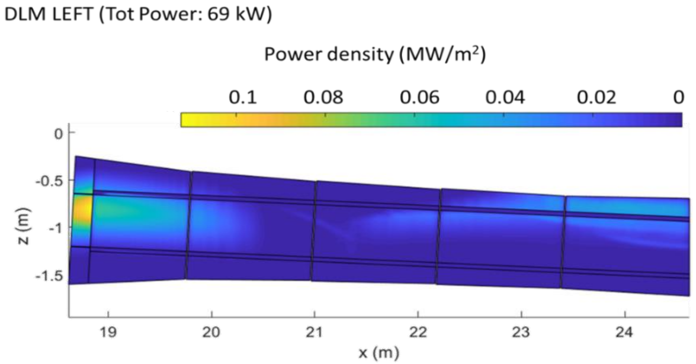
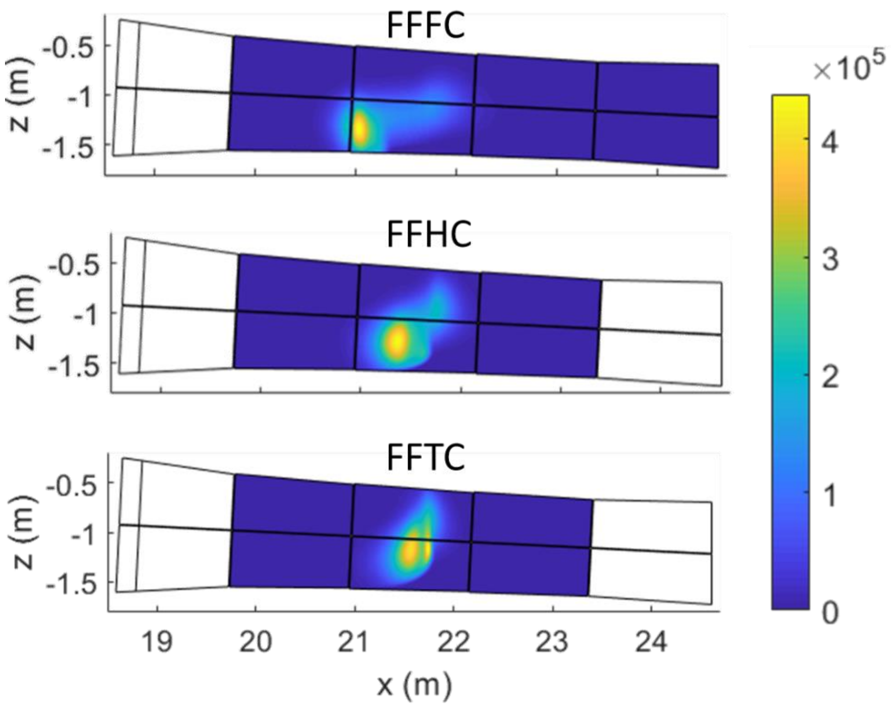
Power Loading on NB Duct (Worst Case)





Power Loading on NB Duct for three B field Scenarios

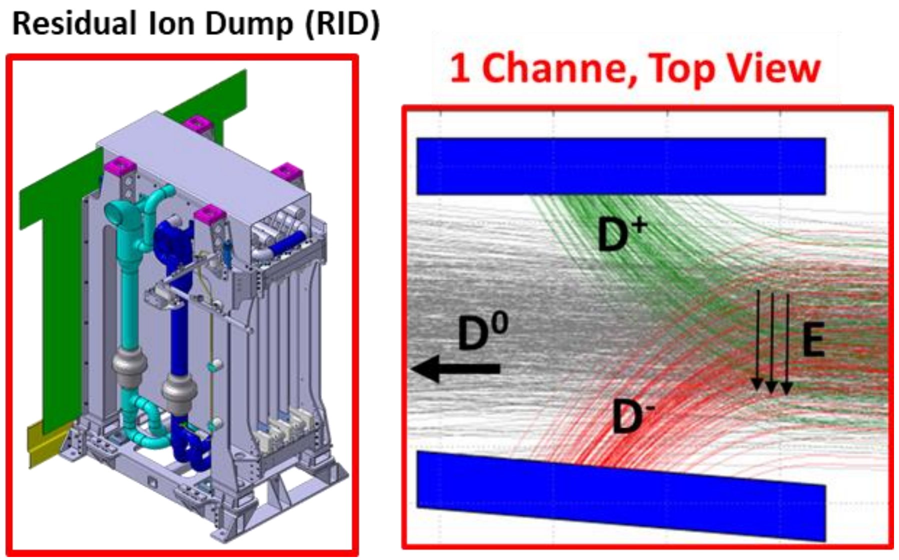


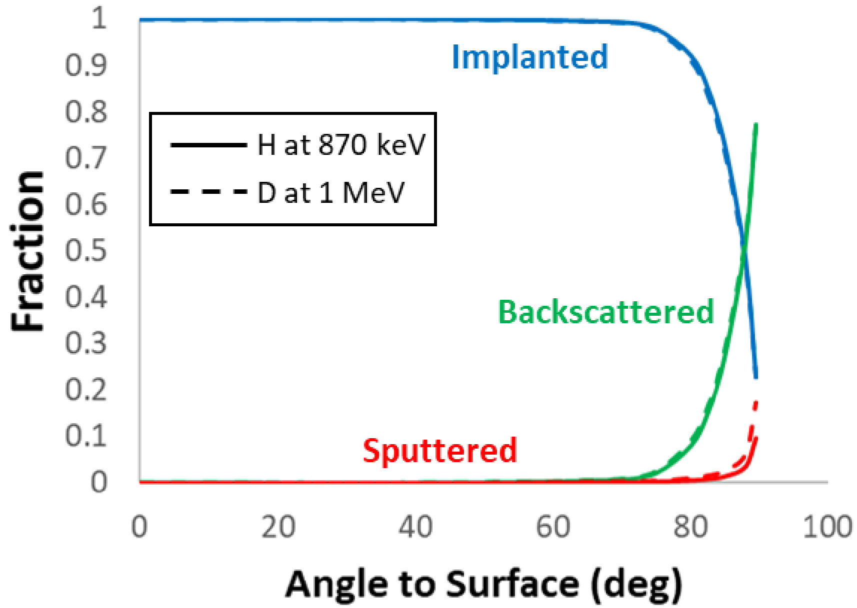
- Duct liner is already cooled almost everywhere to cope with re-ionized power at different B fields
- Power loading on Front end Components to be checked

Tritium Implantation and Retention: RID

- Part of the accelerated T beam impact with surfaces with non negligible probability to be implanted.
- T^+/T^- fluxes on RID, T^+/T_2^+ T_3^+ back-streaming towards the source
- Let's take the RID case. TRIM calculation [1] shows that the implantation fraction for D at 1 MeV on copper is >90%. Should be similar for T.
- Fraction for T+/T- on RID are similar, therefore an ionic tritium flux of about 2.5e17 s^{-1} particles are therefore implanted in the RID. →This would correspond to **40 g** over ITER lifetime.
- Similar Amount on Calorimeter, but for much shorter time (only conditioning).

	Fraction (%)	Particle Flux (1/s)	tot in ITER Lifetime (g)
Lost in The Accelerator	34%	2.5E+17	49.0
Lost at RID (A)	29%	2.2E+17	42.2
Lost In the Duct	9%	7.0E+16	13.4
Injected in ITER	27%	2.0E+17	39.4

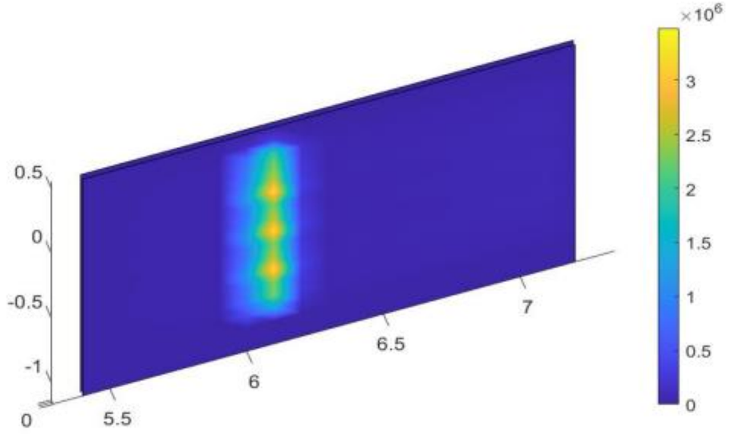




[1] C. Hopf, TRIM calculation of D and H beam on copper (ITER IDM ref.: 4HDWTC)

Tritium Implantation and Retention: The case of the RID

- Part of the accelerated T beam impact with surfaces with non negligible probability to be implanted.
- T^+/T^- fluxes on RID, T^+/T_2^+ T_3^+ back-streaming towards the source
- Let's take the RID case. TRIM calculation [1] shows that the implantation fraction for D at 1 MeV on copper is >90%. Should be similar for T.
- Fraction for T+/T- on RID are similar, therefore an ionic tritium flux of about 2.5e17 s^{-1} particles are therefore implanted in the RID. →This would correspond to **40 g** over ITER lifetime.
- Similar Amount on Calorimeter, but for much shorter time (only conditioning).



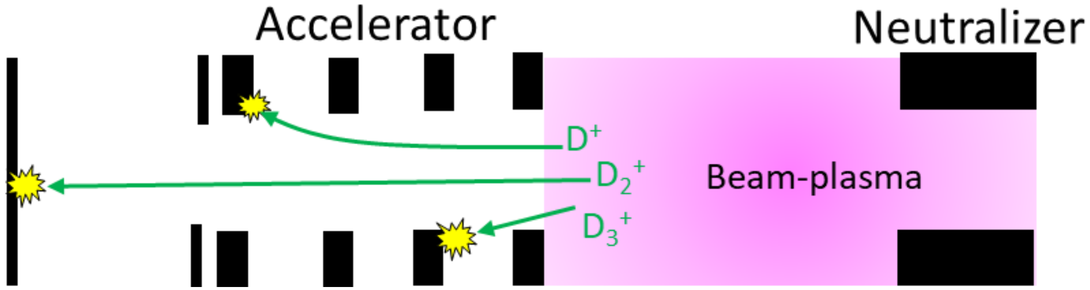
Power Load on 1 Panel of RID

BUT

- a saturation in the material should be reached earlier→ T ion/atoms recombine in molecules and are released from metal
- Studied for T saturation at JET suggest 20% saturation in atomic density [2] on Copper

- Rough calculation with the figures above and assuming the impact surface on RID panels, and implantation depth of 1 micron → Saturation at 0.5-1 g of accumulated tritium

- Another Example: Back streaming ions can also be implanted in the ion source but flux is smaller: 1.2e16 s^{-1}
- Positive T flux from beam-gas interaction in the accelerator about 1e16 s^{-1}. Large part back-streams
- Positive T flux from beam-plasma in the Neutralizer can be calculated from Bohm velocity from plasma density and Te (1e14, 3 eV→ 2e15 s^{-1}

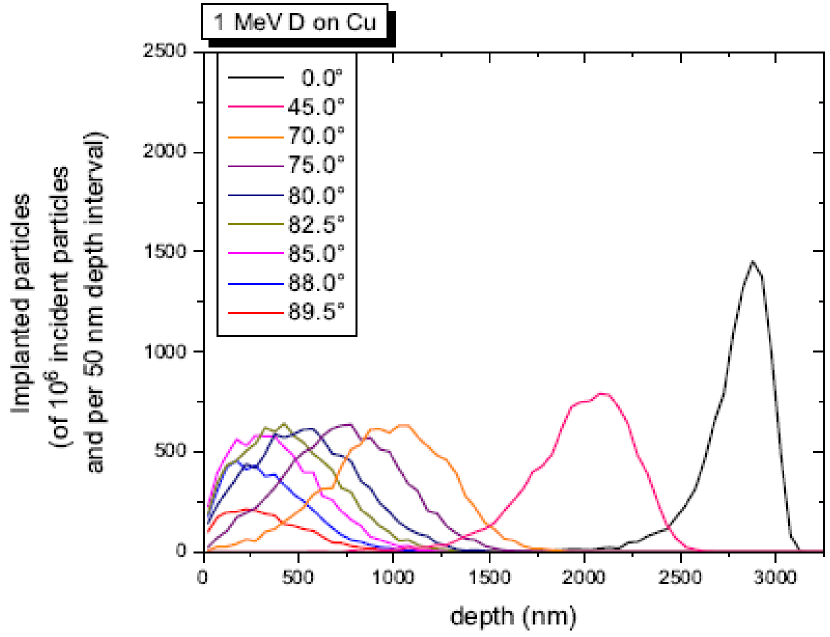


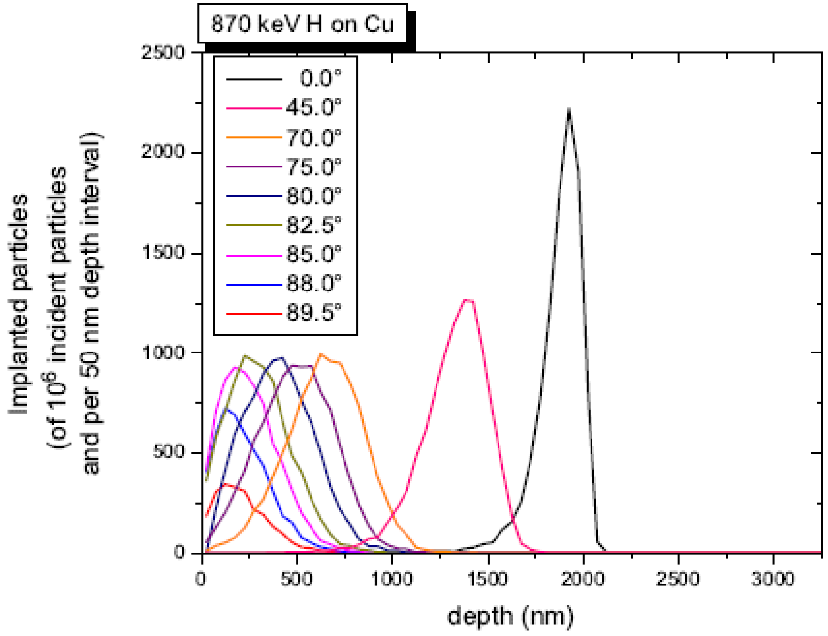
[1] C. Hopf, TRIM calculation of D and H beam on copper (ITER IDM ref.: 4HDWTC)

[2] T.T.C. Jones et al. / Fusion Engineering and Design 47 (1999) 205–231

Tritium Implantation and Retention: “Clean-up”

- Experimental results from JET “PTE” experiment showed that the use of D and H beams after the T-beam campaign effectively “clean-up” surfaces from T. For example in [1] it is concluded that *“The implanted tritium can be removed by operating the injector using a different isotope.”*





- Penetration depth is different, then cleanup is not 100%,
- The position of impact at RID is the same for D/T, then clean-up is effective→ retained quantity is decreased
- In the source back-streaming ions fall at slightly different location (mv/B) on the source back plate→ cleanup less effective
- Also along duct and FECs the positions of implantation differ but these components will be anyway activated

[1] H-D Falter et al., Implantation and desorption of tritium and tritium recovery from the JET neutral beam injectors, J. Nucl. Mater. 196 1992

Summary

- Present Requirement for Gas purity in the ion Source NBIs (<200 PPM of T_2) of are probably too stringent

- The number was set on the basis of the concern on:
  - Degradation of source performances (co-extracted electrons)
  - Location of Power loading of re-ionized T+ atoms

- Experiment at IPP show that fraction at about 0.2-0.5% are probably acceptable.

- T ion implantation would happen in the in-vessel component of the NBIs (Ion Source, RID, Neutralizer) increasing the nuclear dose there. Qualitative consideration suggest that this would not be a serious issue, but more detailed calculation are advisable

- None of the point above justifies the present requirement on gas purity

- A relaxation by an order of magnitude is quite possible (up to 0.2%), that is still challenging for the Isotope Separation System!

Thank you!

Implantation of Tritium and clean-up

“Beam-target emission could be successfully described by a ‘local mixing model’ taking into account the stopping function of the incident particles.

- *This model assumes that the local concentration of hydrogen isotopes cannot exceed a given saturation level (which is dependent upon temperature).*
- *When the saturation level is reached locally, one hydrogen atom is released for every incoming atom which stops at that particular location; the probability that the released atom is of a given isotope is assumed to reflect the local isotopic mix.*
- *The displaced atom is assumed to diffuse rapidly to the surface without being trapped in any adjacent non-saturated region; at the surface it is assumed eventually to re-combine to form a molecule and to leave the material.*
- *The local saturation density of all the hydrogen isotopes implanted in the Cu material was taken to be 20%*

[…] It is therefore concluded that up to 20% of the initial tritium content of the calorimeter panels is retained at the end of the clean-up phase. In order to scavenge this residual tritium content, it would be necessary to employ deuterium beams with at least 20% HIGHER ENERGY”

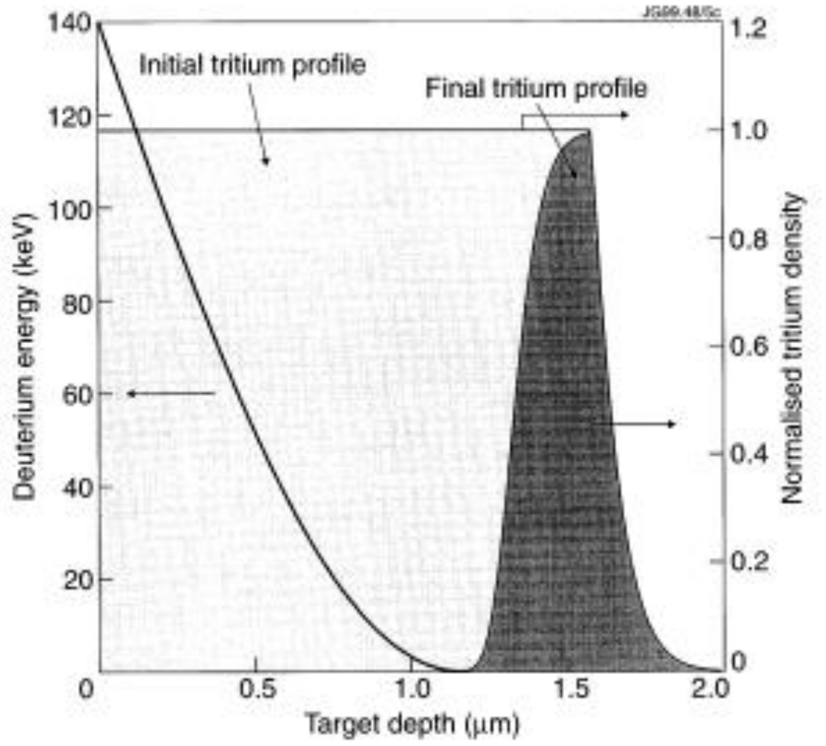


Fig. 19. Tritium implantation profile at the start and end of the cleanup phase of operation, and the average range of implantation of deuterium ions as a function of incident energy.

T.T.C. Jones et al. / Fusion Engineering and Design 47 (1999) 205–231