This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

Liquid Breeder Validation Module geometrical configuration for permeation control optimization

Fourth DONES Users Workshop.
CIEMAT. Madrid (13th-15th October)

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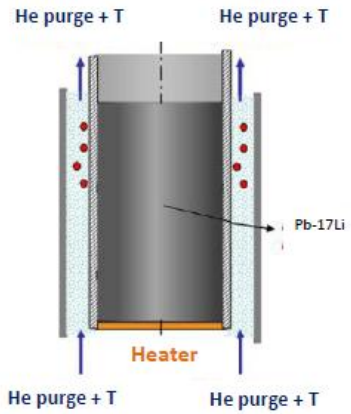
This work has been carried out in the framework of the E4XTREM project,ref. PLEC2024-011136, funded by MICIU/AEI/10.13039/501100011033/FEDER,UE.

Objective

We want to be users of IFMIF DONES to test Tritium Breeding related materials

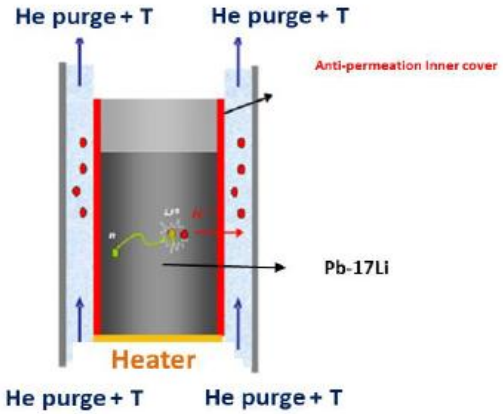
Experiment 1:

- Tritium production monitoring (online)
- Lithium-Lead alloy Transmutation (post-irradiation)



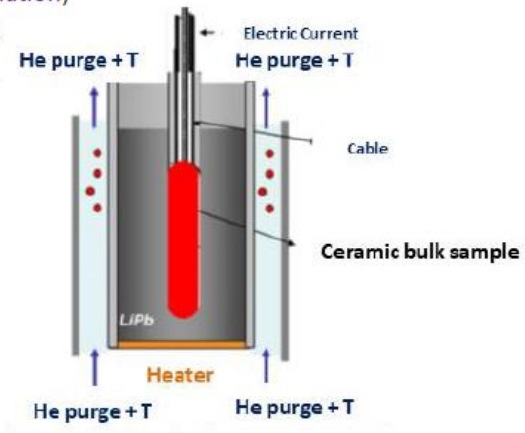
Experiment 2:

- **Coating materials** assessment
  - Antipermeation degradation - Tritium permeation monitoring (online)
  - Damage and corrosion characterisation (post-irradiation)

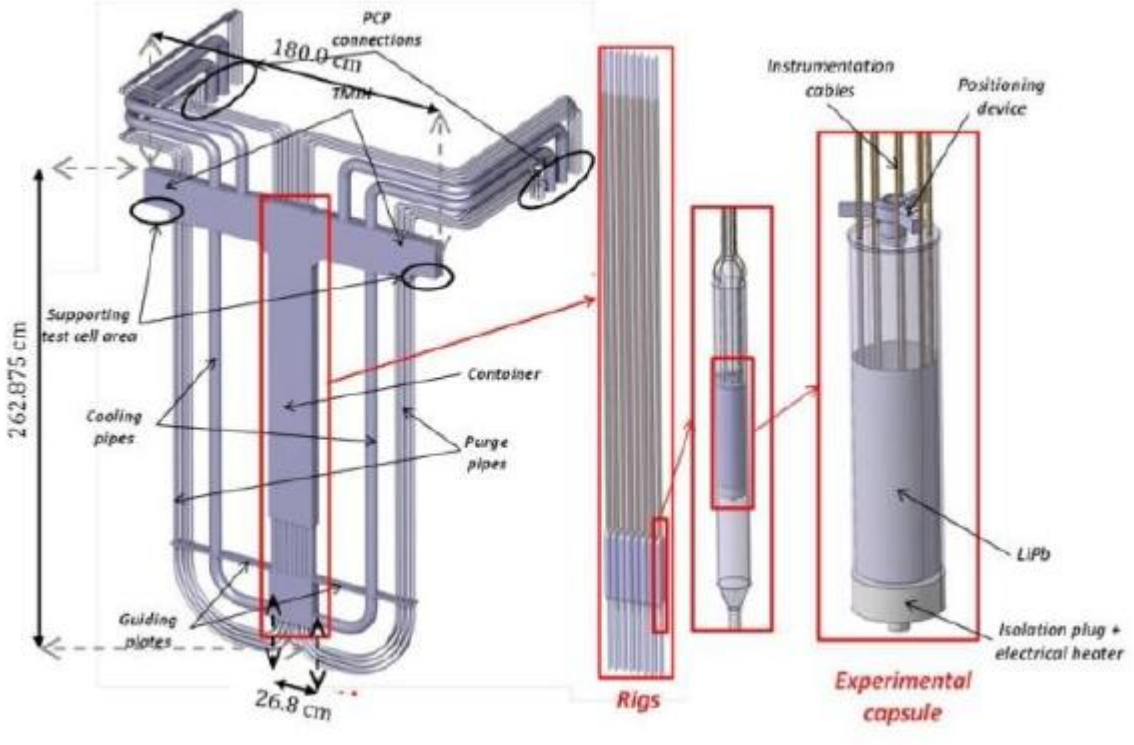


Experiment 3:

- **Bulk materials** assessment
  - Electrical resistivity measurement (online) Important analysis for MHD effects reduction material selection for DCLL concepts.
  - Antipermeation (post-irradiation)
  - Corrosion(post-irradiation)

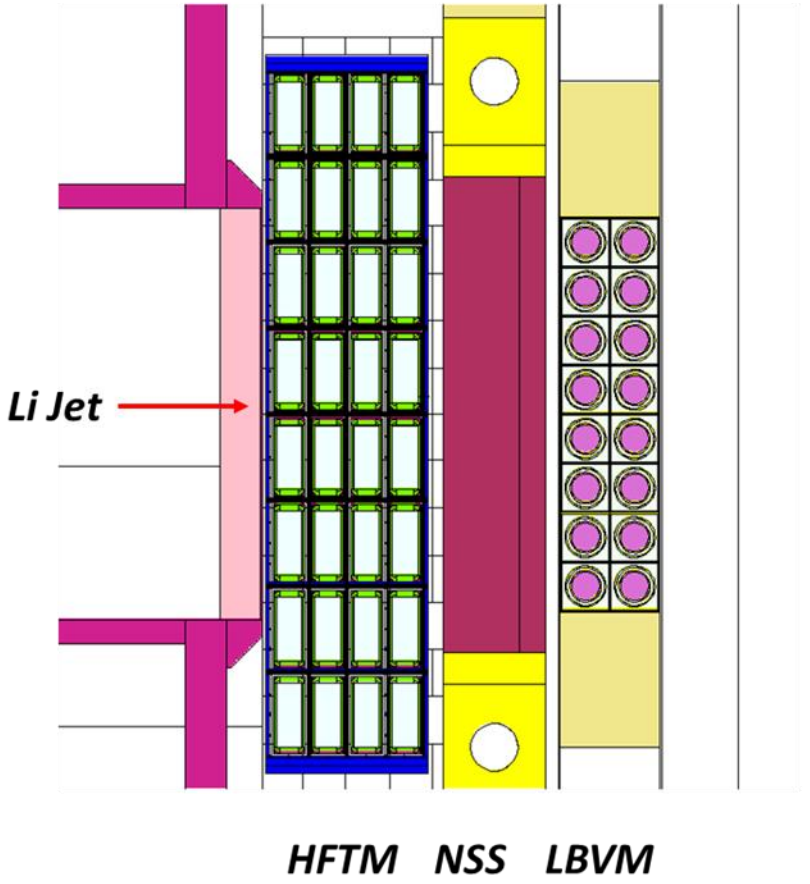


Original design

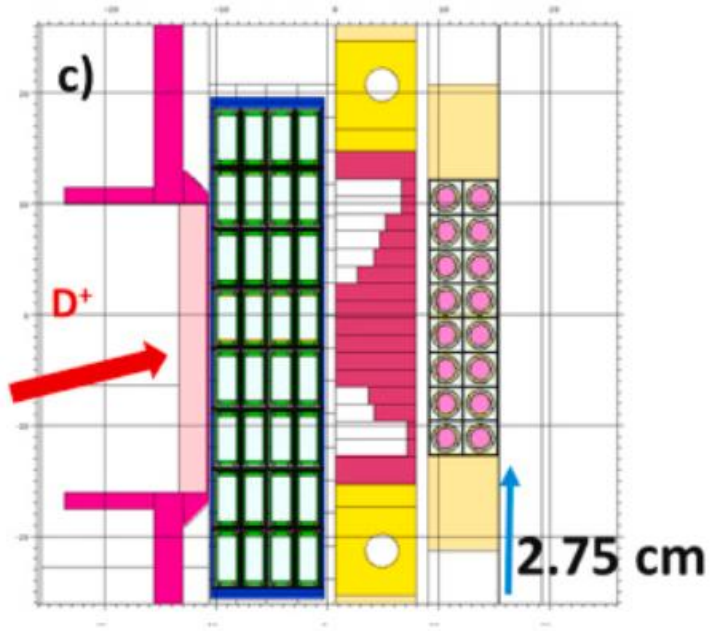


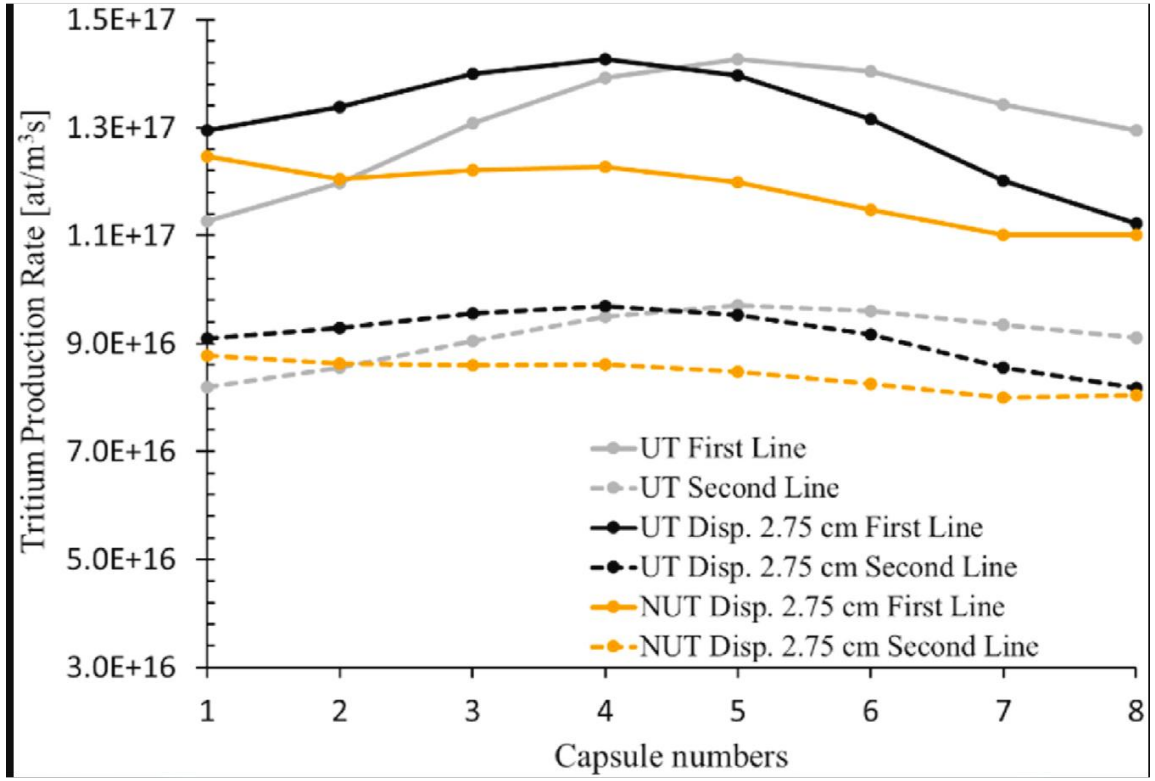
PhD Thesis N. Casal (2016)

IFMIF-DONES Test Cell 2024(horizontal cross section)



Update of the NSS





F. Mota et al. Non Uniform Thickness Neutron Spectrum Shifter to fit radiation BB effects on material in IFMIF-DONES . https://doi.org/10.1016/j.net.2025.103983

Requirements for more detailed design

Internal radius



R = 15 mm for internal coating

Capsule wall



300–550 °C

BZ temperatures
Max 450 W

Purge He gap



As small as possible

Need to fit many capsules

Exp. tube



< 100 °C

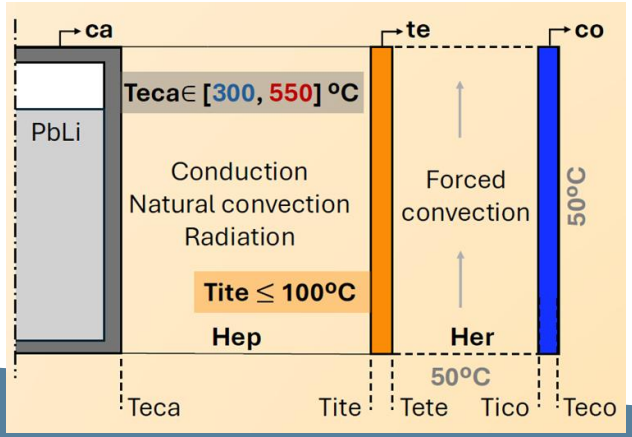
Low T to minimize T permeation

Cooling He

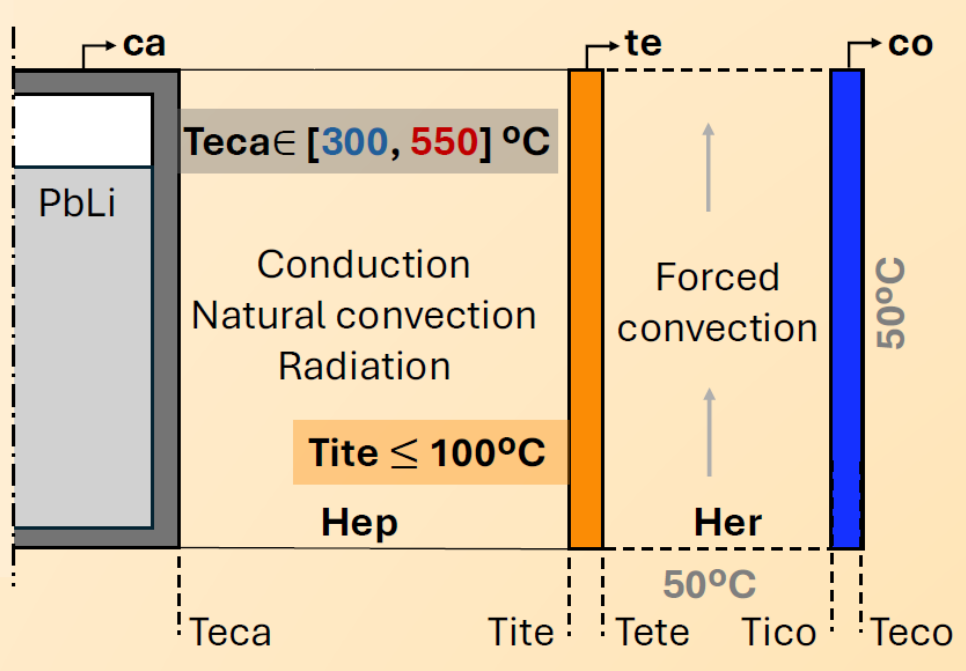


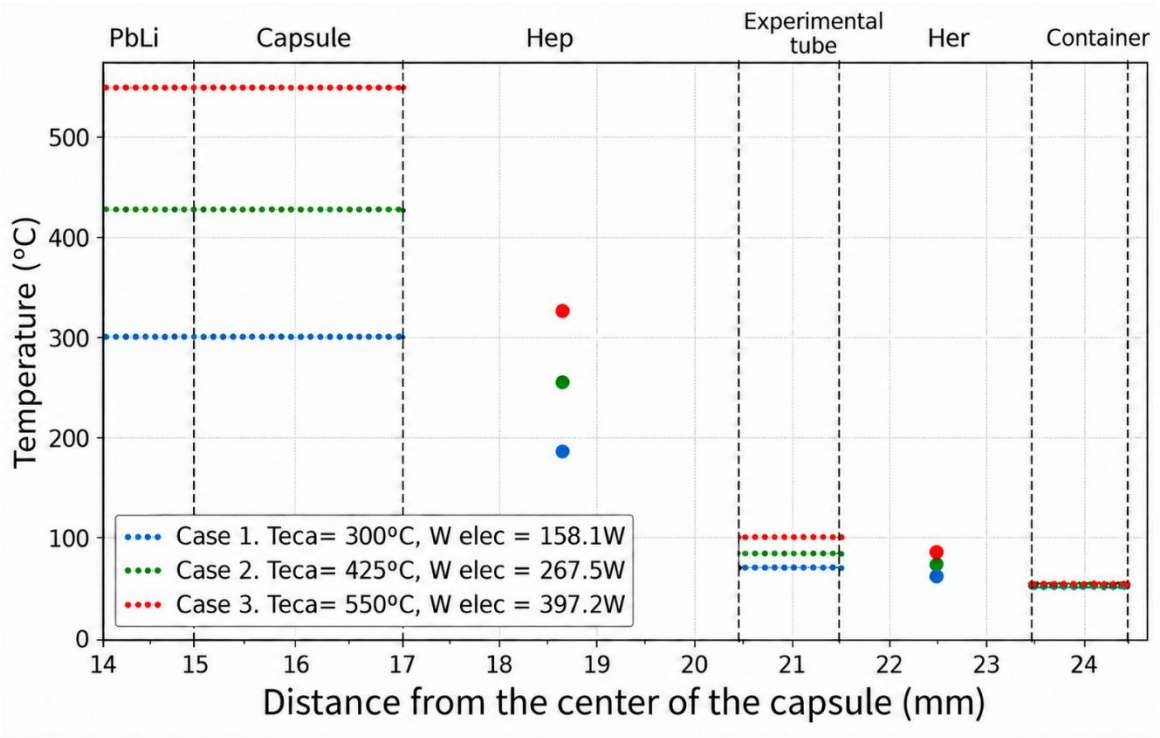
As small as possible

But enough for TCs + manufacturing tolerances



Thermal calculations

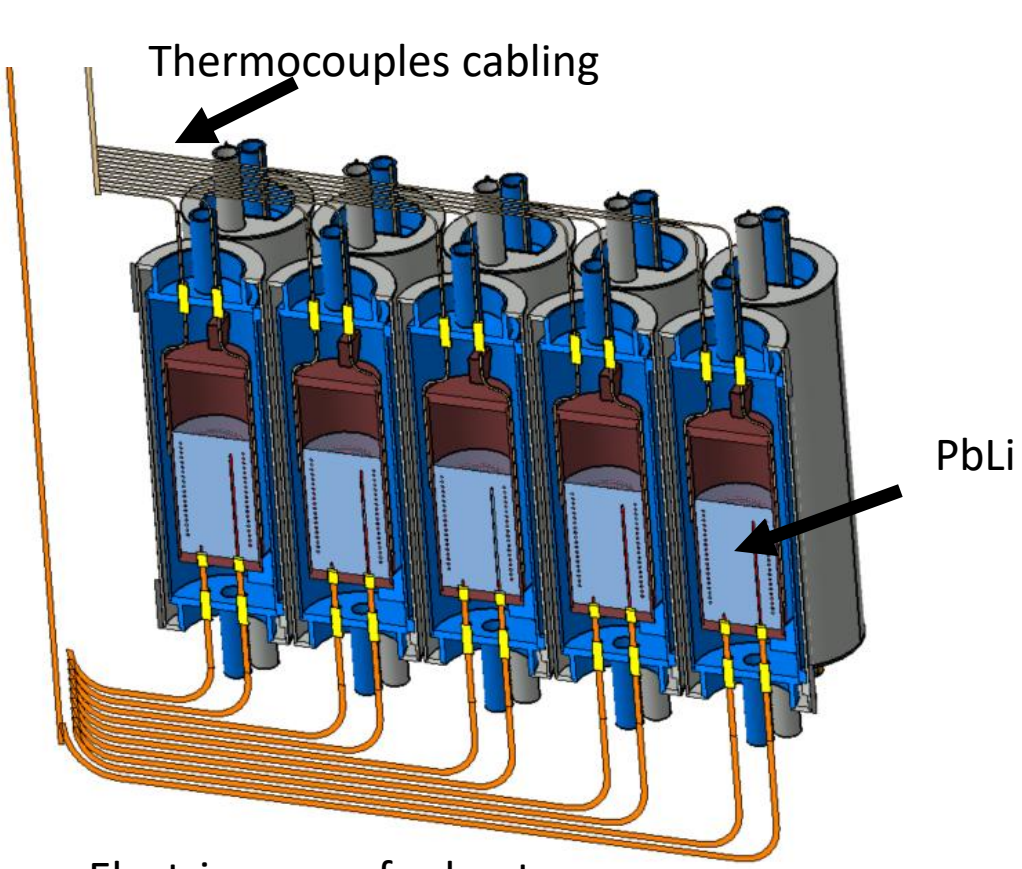


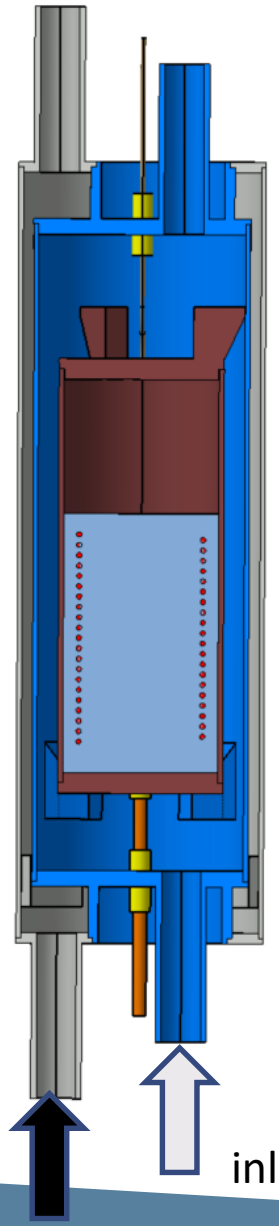


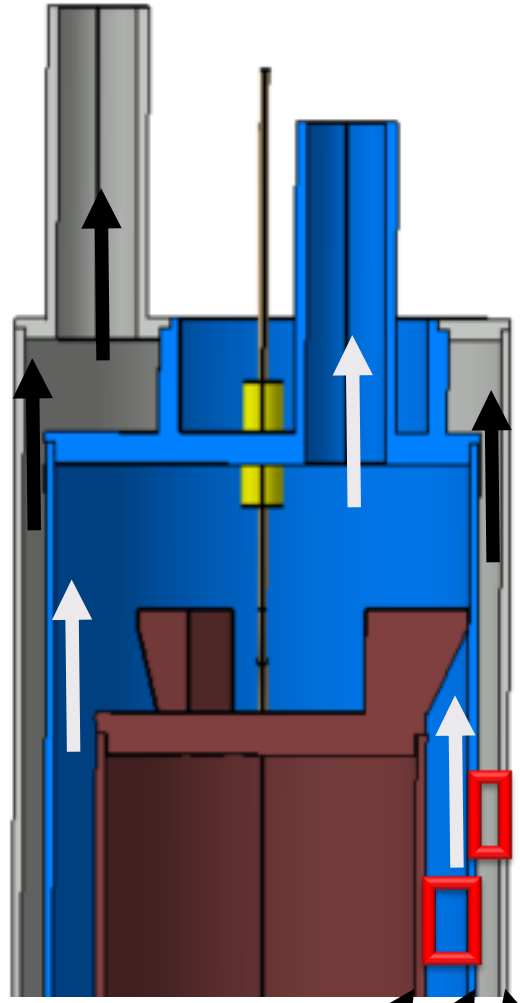
An iterative calculation provides :

- minimum purge gap: 3,4 mm
- required electric power to reach 550ºC on the capsule: 397 W

Updated design of the capsule

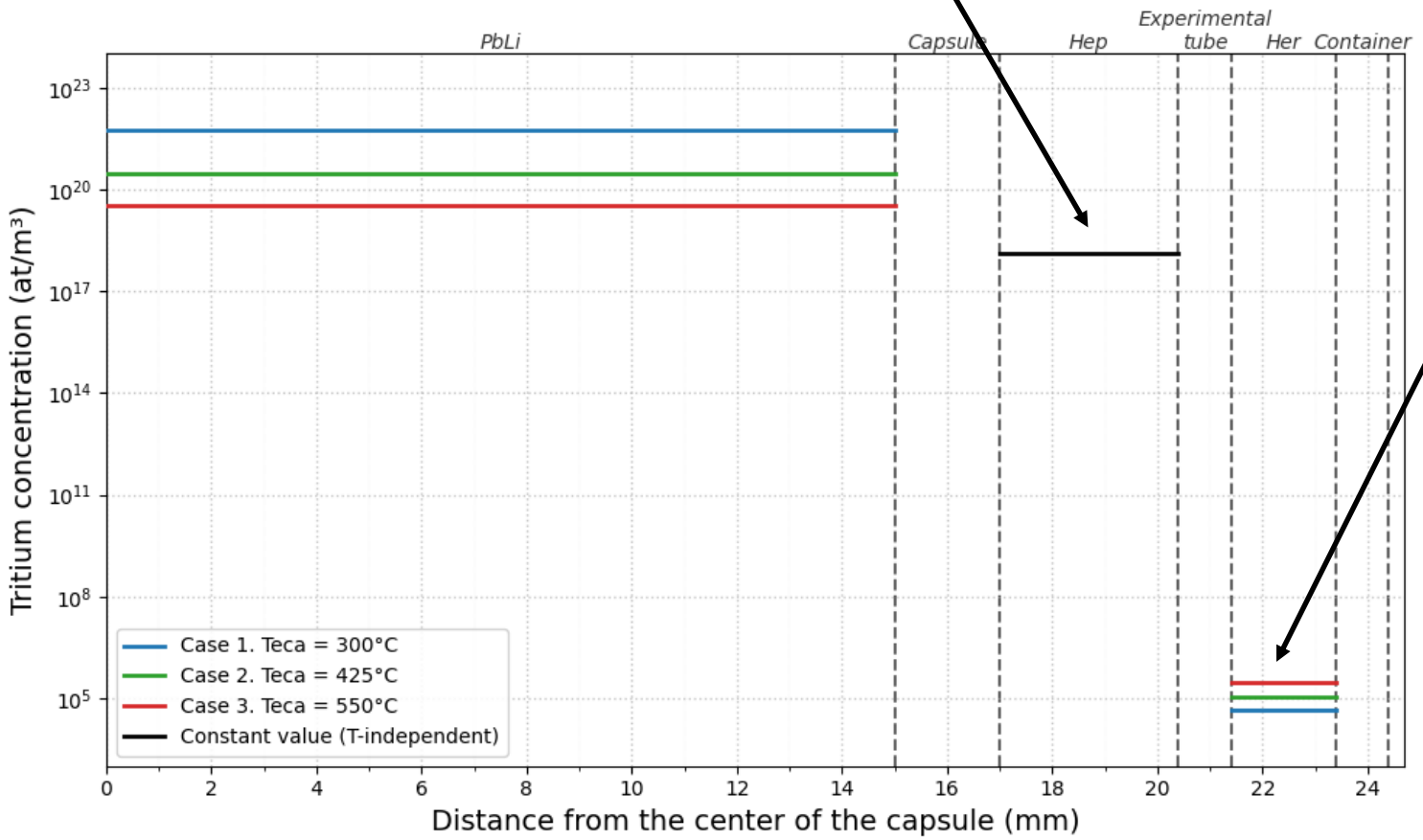




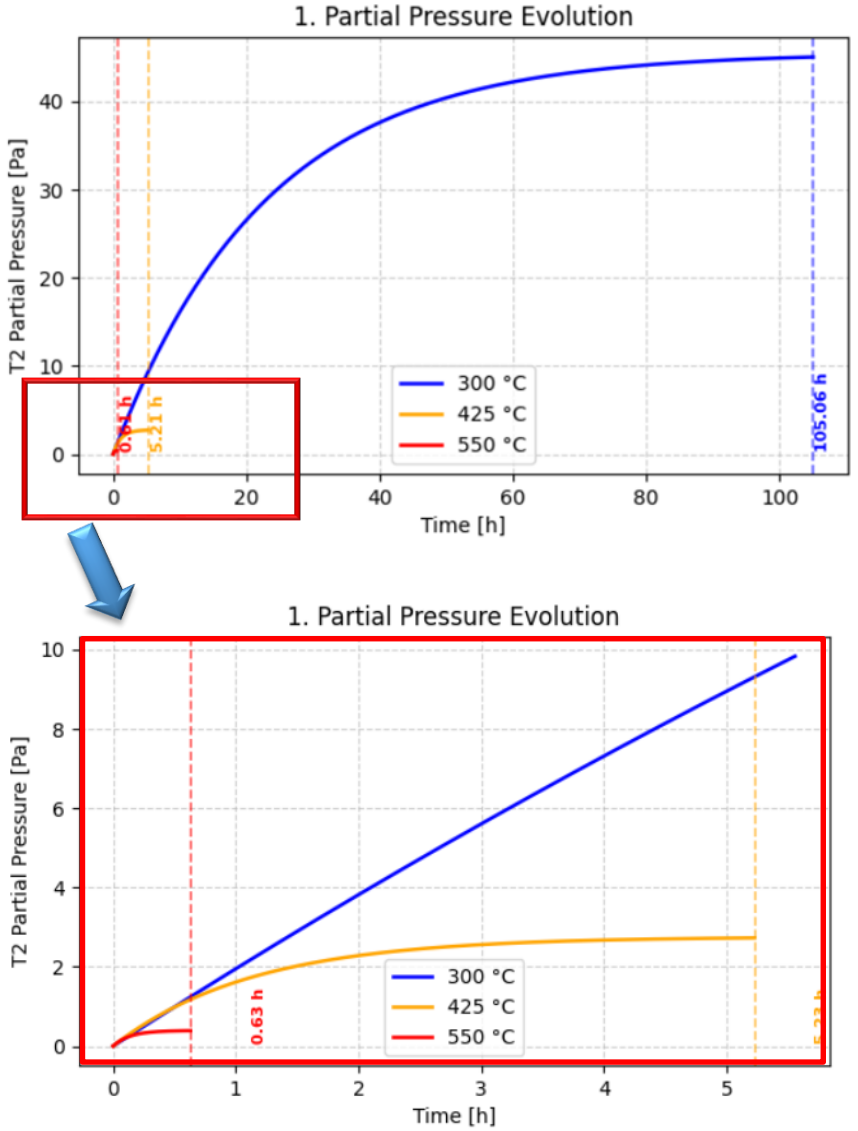


Estimates of Tritium concentration

Case	C_{PbLi} $[at/m^3]$	$J_{s,ca}$ $[at/s]$	C_{Hep} $[at/m^3]$	$J_{s,te}$ $[at/s]$	C_{Her} $[at/m^3]$
300ºC	1	$5.75 \cdot 10^{21}$	$4.45 \cdot 10^{12}$	$1.32 \cdot 10^{18}$	$1.90 \cdot 10^{3}$	$4.69 \cdot 10^{4}$
425ºC	2	$2.88 \cdot 10^{20}$			$4.58 \cdot 10^{3}$	$1.13 \cdot 10^{5}$
550ºC	3	$3.46 \cdot 10^{19}$			$1.15 \cdot 10^{4}$	$2.83 \cdot 10^{5}$



Time to reach stationary state




```
----- 300 °C (300 °C / 573.3 K) -----
  Iterations completed : 11819
  Elapsed time         : 378208.0 s (6303.47 min / 105.06 h)
  Final relative error : 1.000 %
  Final pressure       : 4.505e+01 Pa
  Final permeation(Js) : 4.409e+12 at/s

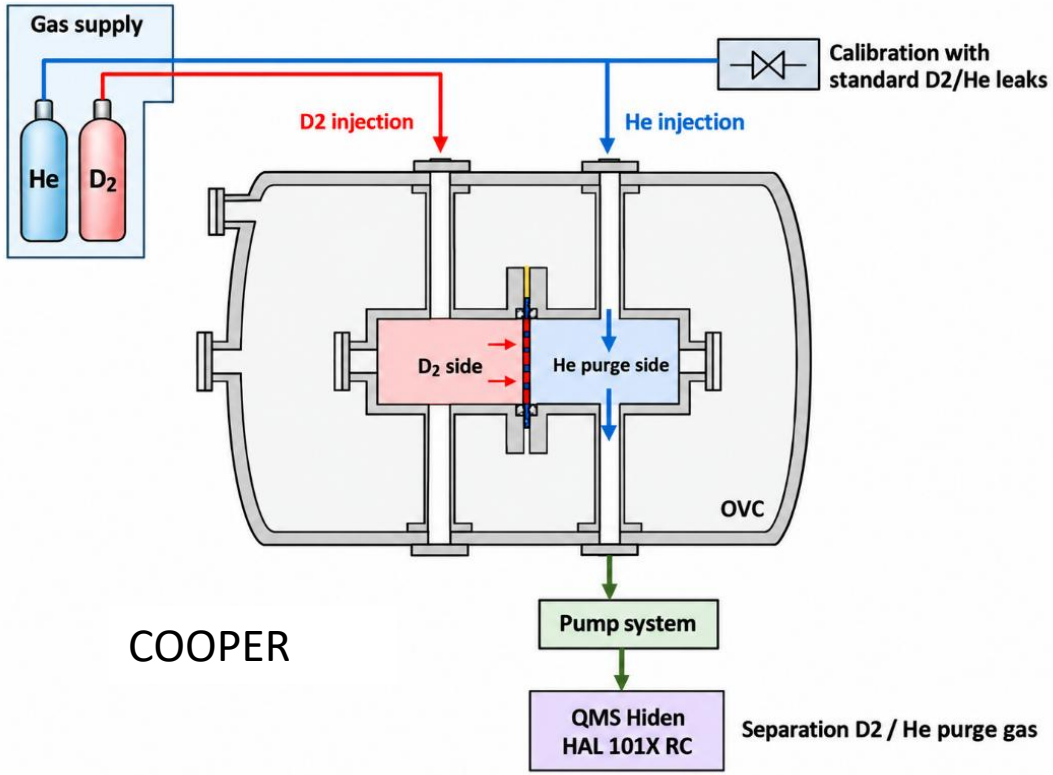
----- 425 °C (425 °C / 698.0 K) -----
  Iterations completed : 586
  Elapsed time         : 18752.0 s (312.53 min / 5.21 h)
  Final relative error : 0.998 %
  Final pressure       : 2.729e+00 Pa
  Final permeation(Js) : 4.409e+12 at/s

----- 550 °C (549 °C / 822.3 K) -----
  Iterations completed : 69
  Elapsed time         : 2208.0 s (36.80 min / 0.61 h)
  Final relative error : 0.981 %
  Final pressure       : 3.885e-01 Pa
  Final permeation(Js) : 4.409e+12 at/s
```

4.4 days

Capsule prototypes at CIEMAT

Ongoing prototypes and experiments



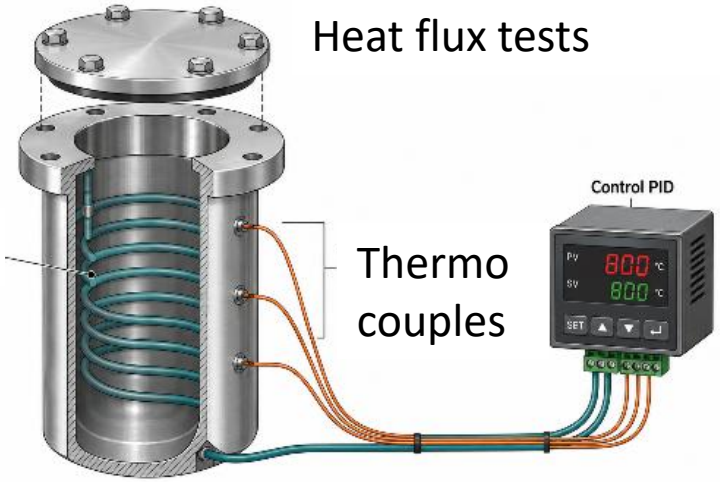
COOPER

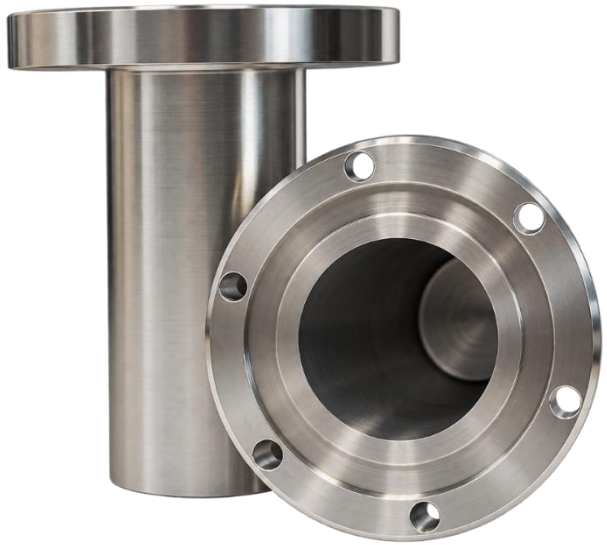
S.J. Hendricks et al. Design of a multi-isotopic hydrogen co- and counter-permeation experiment for HCPB related tritium mitigation studies. https://doi.org/10.1016/j.fusengdes.2021.112991

Heat flux tests

Heating wire

Thermo couples





Future work

* Advance in **detailed engineering design** and experimental procedure
* Analyse and define **Tritium measurement** equipment
* **Life cycle** definition (manufacturing, installation, irradiation, removal, decomissioning,…)
* **Safety** aspects analysis

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Thank you for yor attention
Questions or comments?

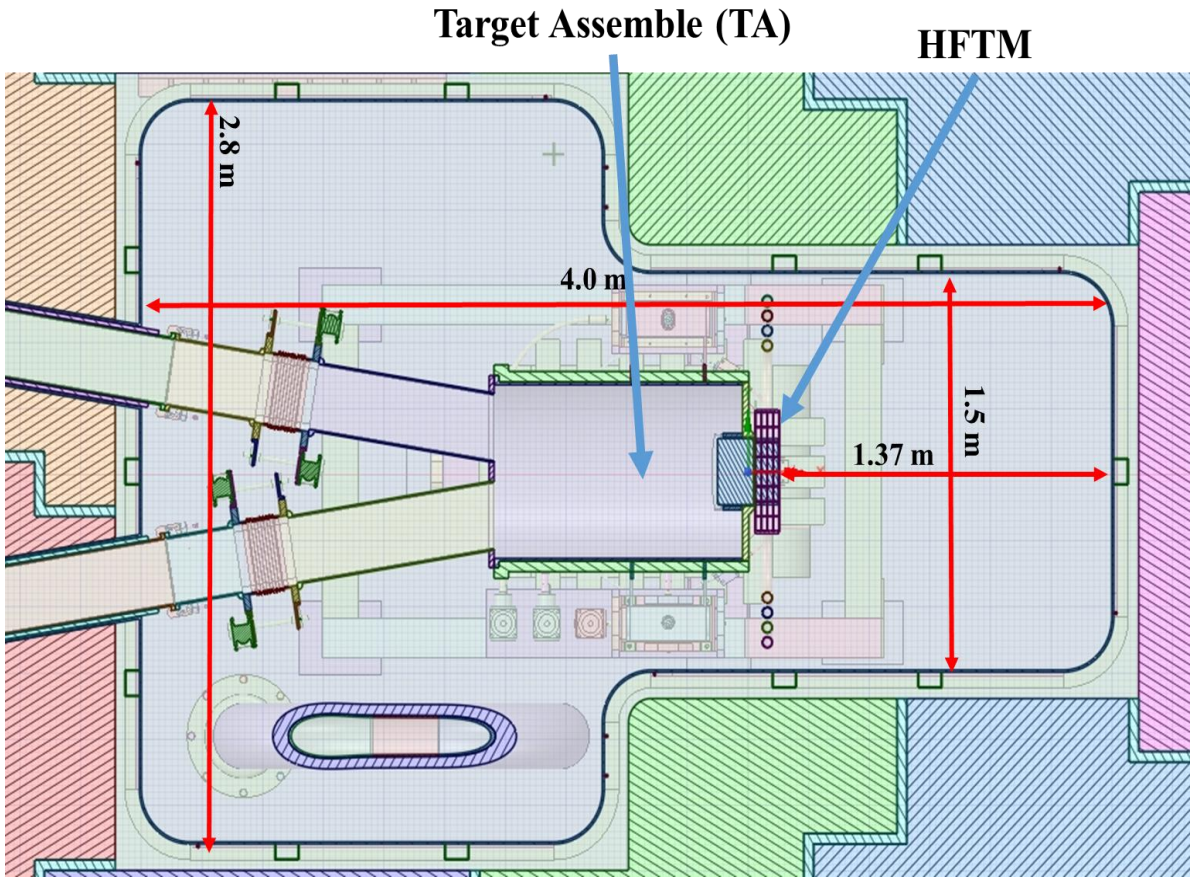
fernando.arranz@ciemat.es

This work has been carried out in the framework of the E4XTREM project,ref. PLEC2024-011136, funded by MICIU/AEI/10.13039/501100011033/FEDER,UE.

Additional slides

IFMIF-DONES test area

- Available irradiation **volume** in DONES is **large** and represents an important benefit → margin to propose new experiments.



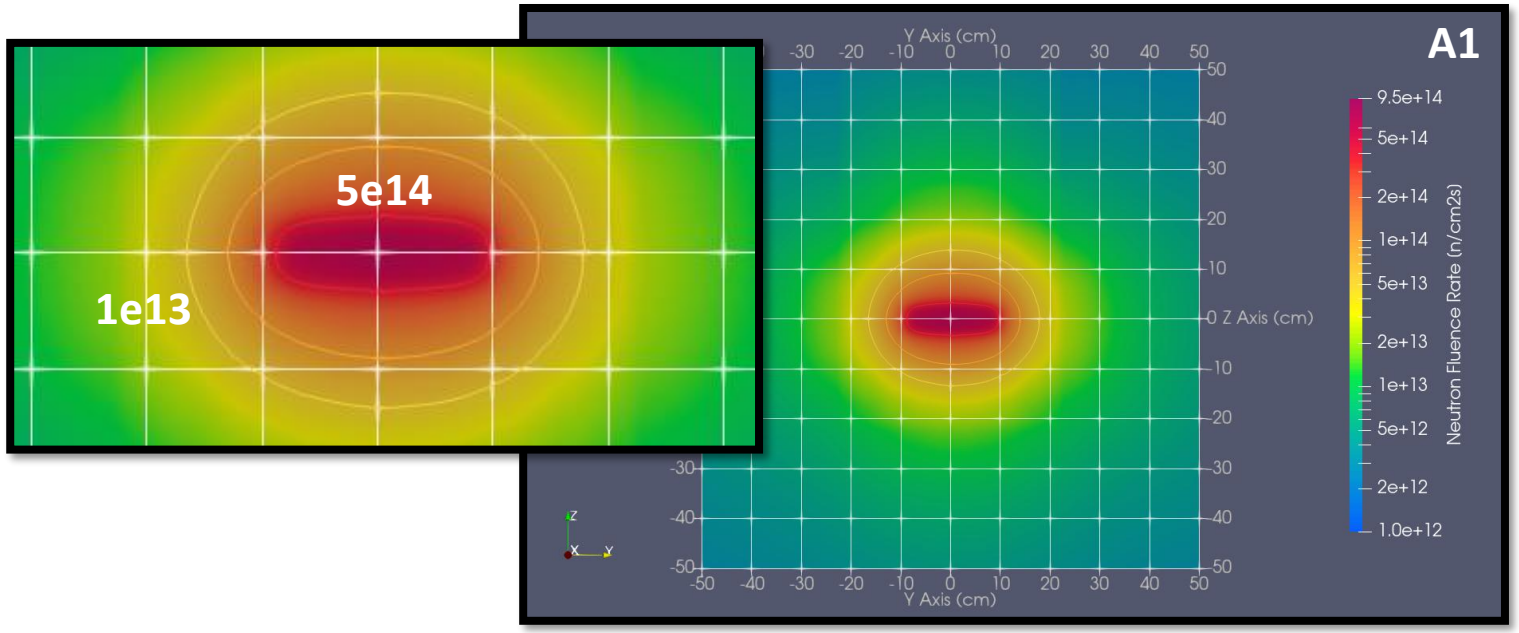
x-axis: 1.37 m in the direction of the beam (excluding the HFTM)
y-axis: 4 m in the vertical direction
z-axis: 1.5 m in the horizontal direction

Effective irradiation volume

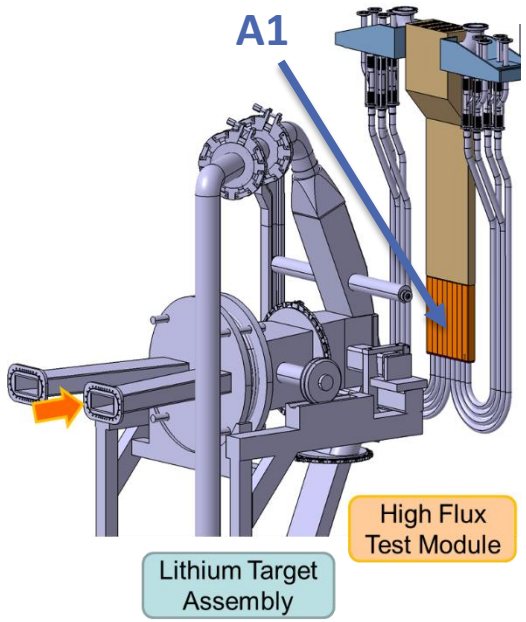
- The volume where the experiments will be relevant (in some way) for the BB qualification
- As in any other neutron source, depending on the used materials and components, the irradiation field could be modified
- **Specific calculations are needed**

IFMIF-DONES test area

- Most of people think on this kind of neutron distribution, but this is the footprint just **immediately behind the Target Assembly** (or the front of the HFTM).

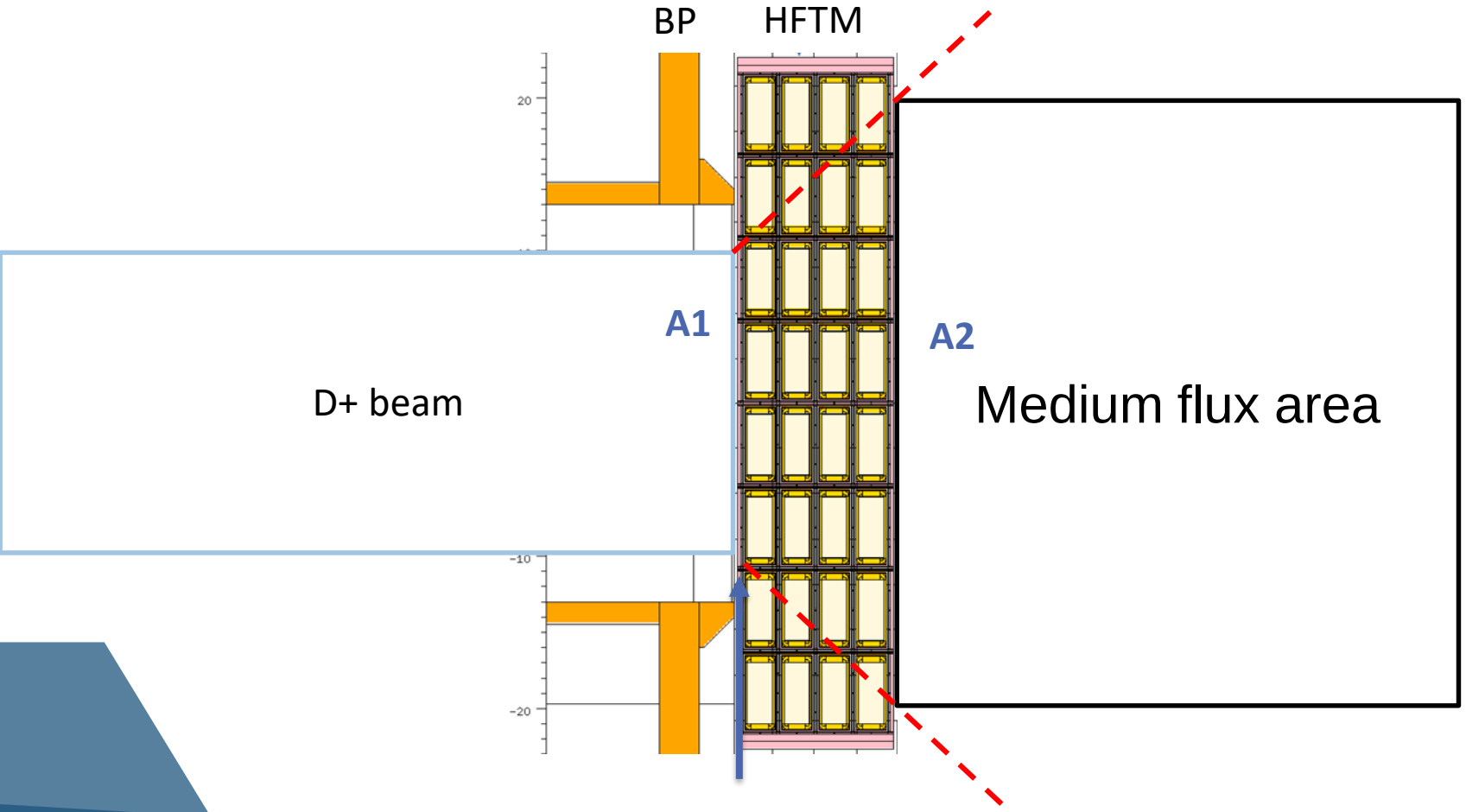


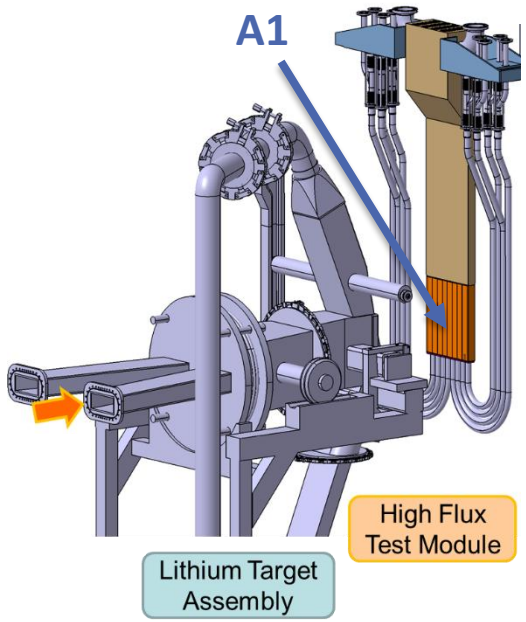
'footprint of beam on lithium curtain'
20 x 5 cm²



IFMIF-DONES test area

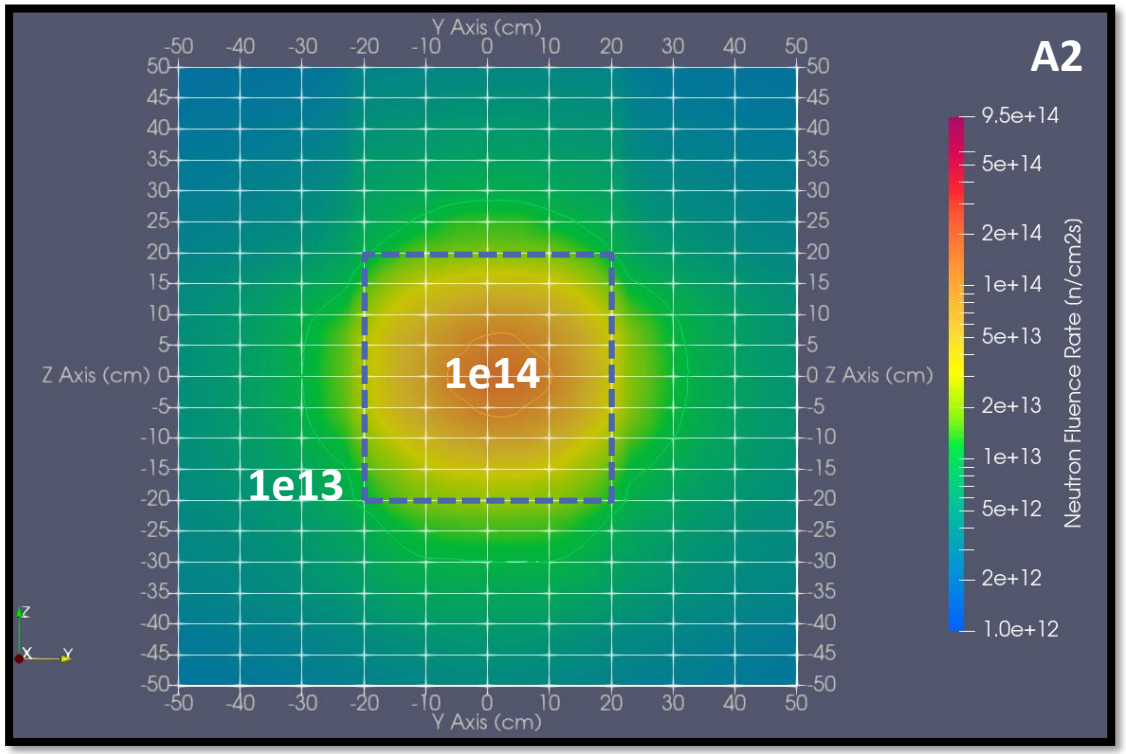
- However, the neutron field presents dispersion through the HFTM (distance ~10 cm) that results in a more homogeneous rad field at the beginning of the Medium Flux area



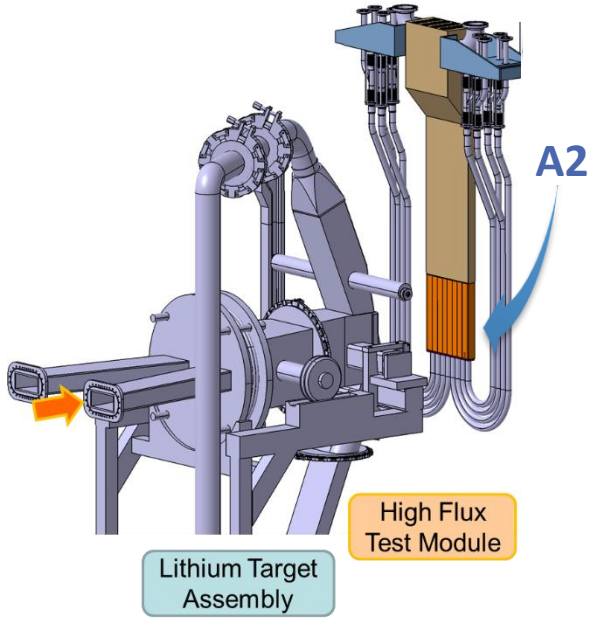


IFMIF-DONES test area

- However, the neutron field presents dispersion through the HFTM (distance ~10 cm) that results in a more homogeneous rad field at the beginning of the Medium Flux area



'footprint in the Medium Flux area'
40 x 40 cm²

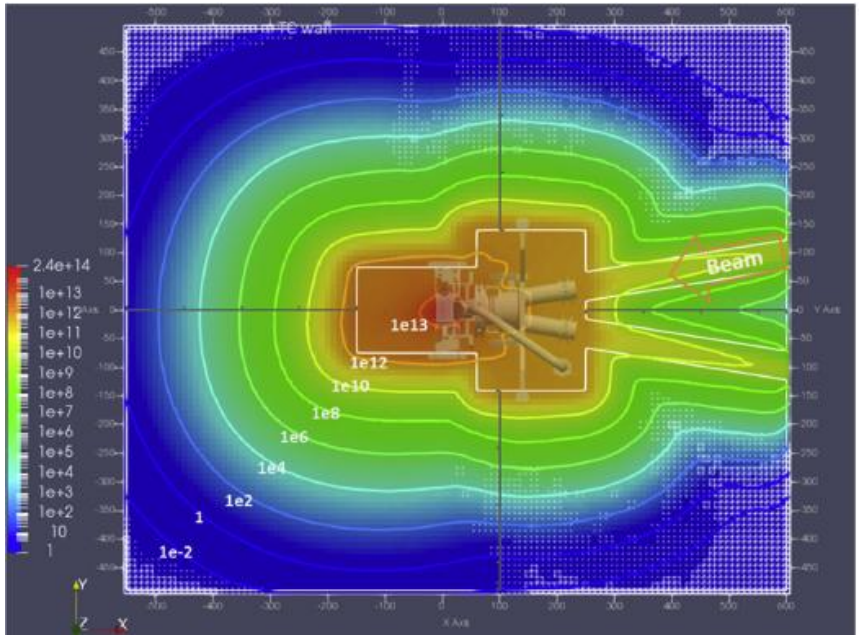


Behind the HFTM → 'effective area' from 100 cm² to **1600 cm²**

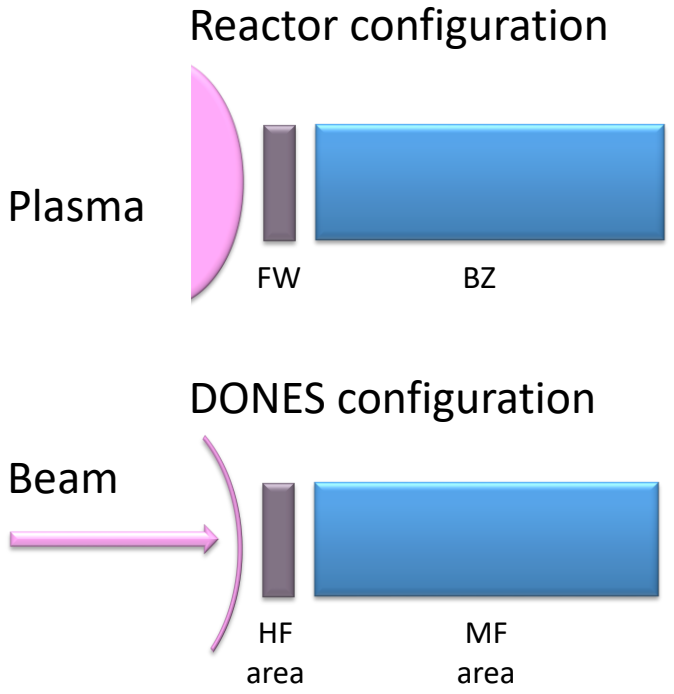
IFMIF-DONES test area

- Neutron flux distribution in and around the TC (left: horizontal cut)

$n/(cm^{-2}\ s^{-1})$



U. Fischer et al., 2019 *Fus. Eng. Des.* **146** 1276-1281



- Neutron axial gradient similar to the one in DEMO → TBC (results coming in the next slides)

The DONES medium flux area constitutes a perfect test bench

Conditions for Temperatures calculation:
He purge. Cycle of 60 minutes (50 min static, 10 min flowing at 1 m/s and renewed)
inlet temperature: 50ºC
inlet pressure: 0,3 bar absolute

He refrigeration. Continously flowing at 50 m/s
inlet temperature: 50ºC
inlet pressure: 3 bar absolute

Total nuclear heating: 11,2 W
Electric heating: 397 W

Distribution in He purge according to heat transfer mechanism:
Heat_conduction = 353.5 W
Heat_convection = 20.3 W
Heat_radiation = 29.1 W
Heat_estimate = 402.9 W (radiation heat in Experimental tube goes directly to He refrigeration around 6 W)

Iterative calculation of thickness of He purge gap required to comply with requirements

