# Beryllium Erosion and Tritium Retention in ITER

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*ITER Organisation*

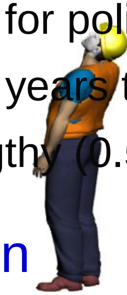
*with contributions from*
C. Björkas (U. Helsinki/FZJ), R. Doerner (UCSD)

# Outline

- Status of the ITER Project (3 slides)
- Overview of the ITER device (4 slides)
- Be erosion and deposition modeling activity at ITER (14 slides)
- Model predictions for tritium retention (4 slides)
- Rates used (4 slides)
- Summary (1 slide)

# ITER and scale

- **Size**: moving beyond the “human scale”
  - almost all components are “uncomfortably large”
  - can be a challenge for industry
- **Cost**: currently quoted at 15-20 billion US dollars (~7 billion $US in 2001)
  - difficult for politicians and national science budgets, especially right now
  - project risks are likely to increase as efforts are made to save money (and/or time)
- **Time**: 10 years for construction, 20 years of operation
  - again, the long timescales can be difficult for politicians (and physicists)
  - long timescales for manufacture (e.g. 5-7 years to build many of the components)
  - maintenance periods are difficult and lengthy (0.5-2 years)



SPLAT!

- **Complexity**: highly integrated components built in different places
  - large effort to manage “interfaces” and establish and enforce quality assurance (QA) procedures

THE “ITER PERSON”

- **Benefit**: an important step toward a (reasonably) clean, universally accessible source of energy

# Primary project goals / objectives

$$D + T \rightarrow n\,(14.1\,MeV) + \alpha\,(3.5\,MeV)$$

$$Q = \frac{fusion\ \ power}{input\ \ power} = \frac{P_n + P_\alpha}{P_{heating}} = Q_{scientific} \neq Q_{engineering}$$

*Physics*

Q = 10

Q → ∞

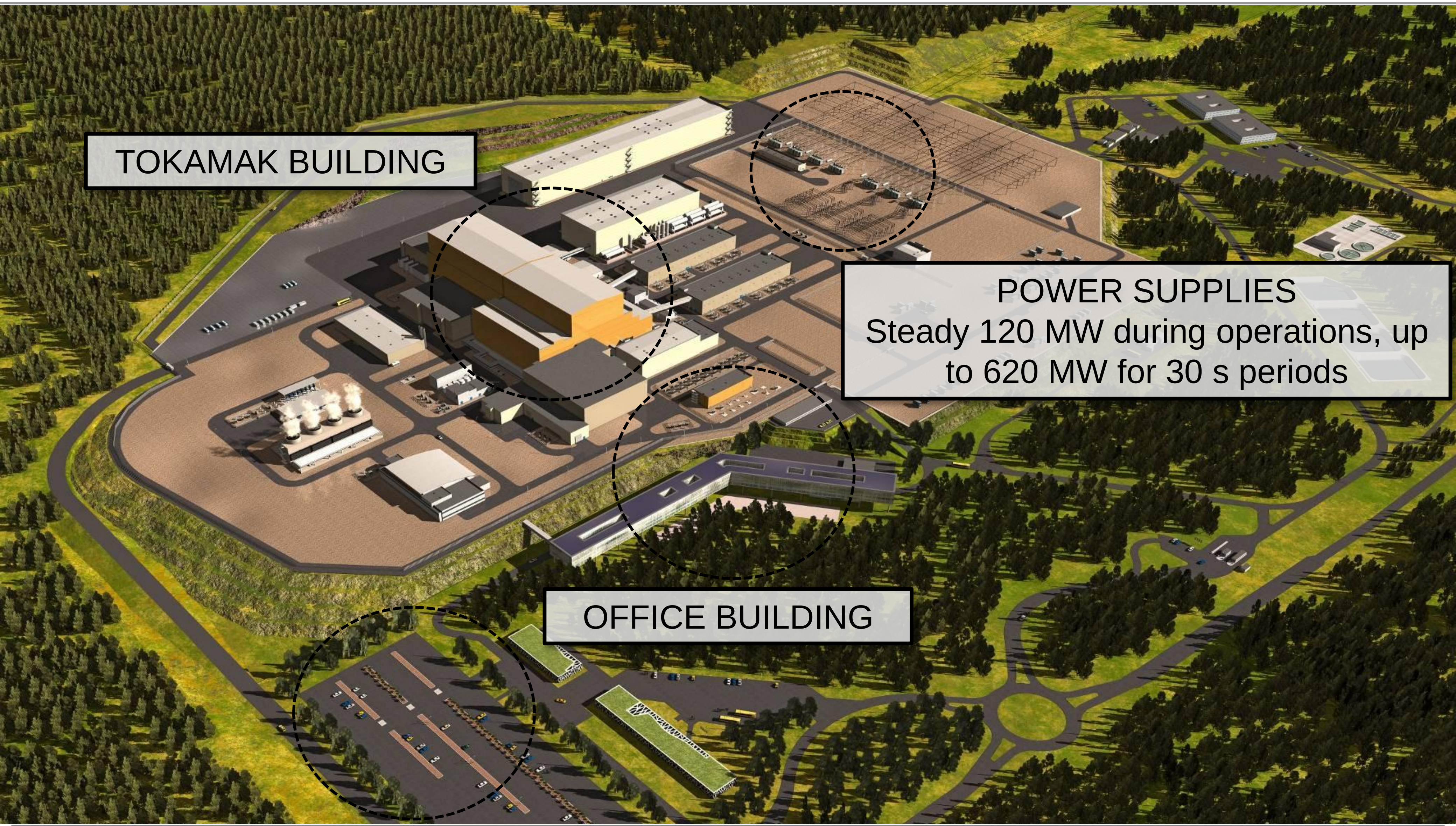
- Produc... h...
- Signifi... io... ver (50 MV... he...

*Technology*

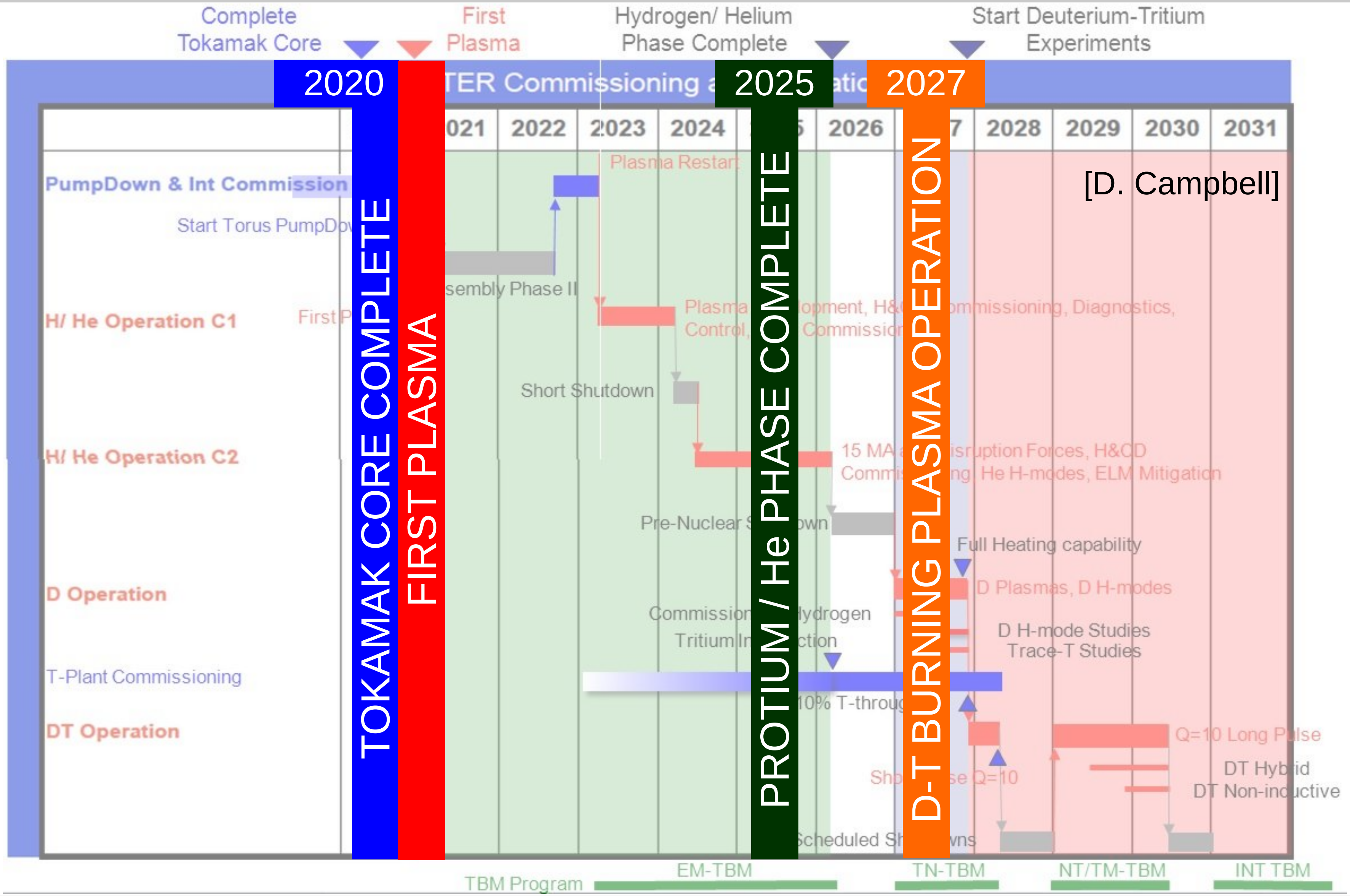
- Test co... er...
- Demon... ol...
- Test concepts for a tritium breeding module (production of tritium fuel)

# Current status of the ITER platform

# Completed site (2020)



# Current ITER experimental programme: D-T in 2027



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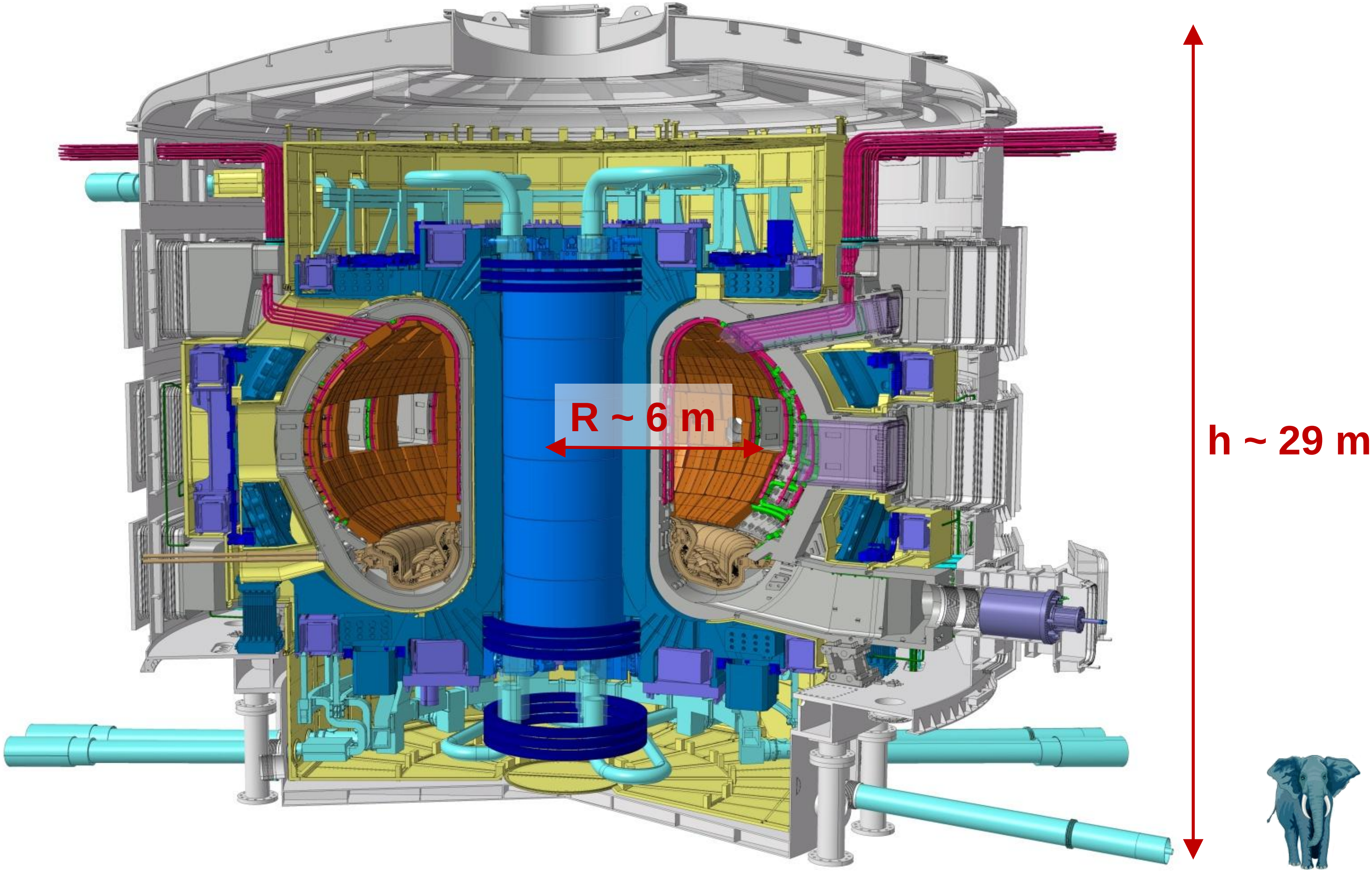
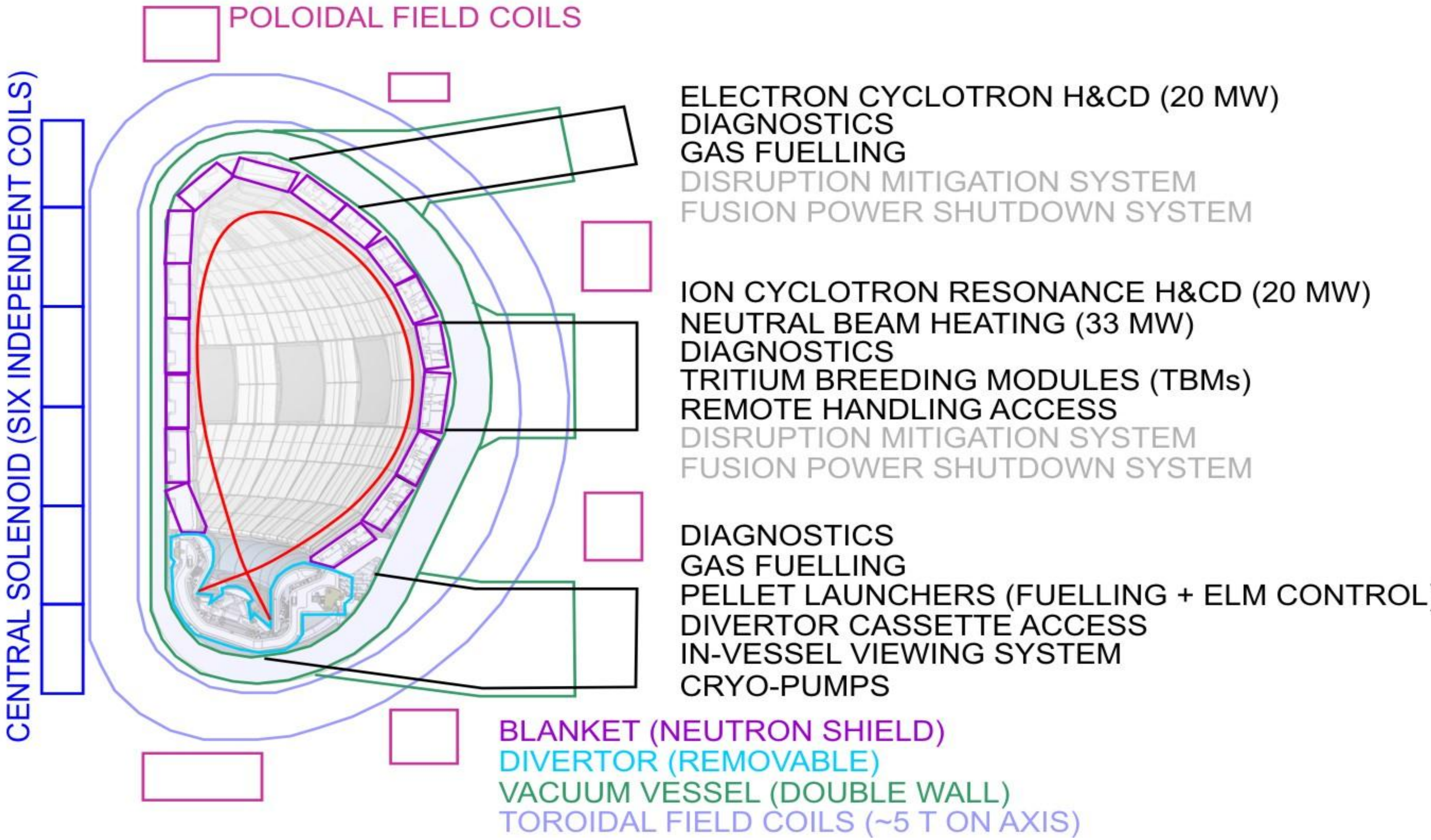
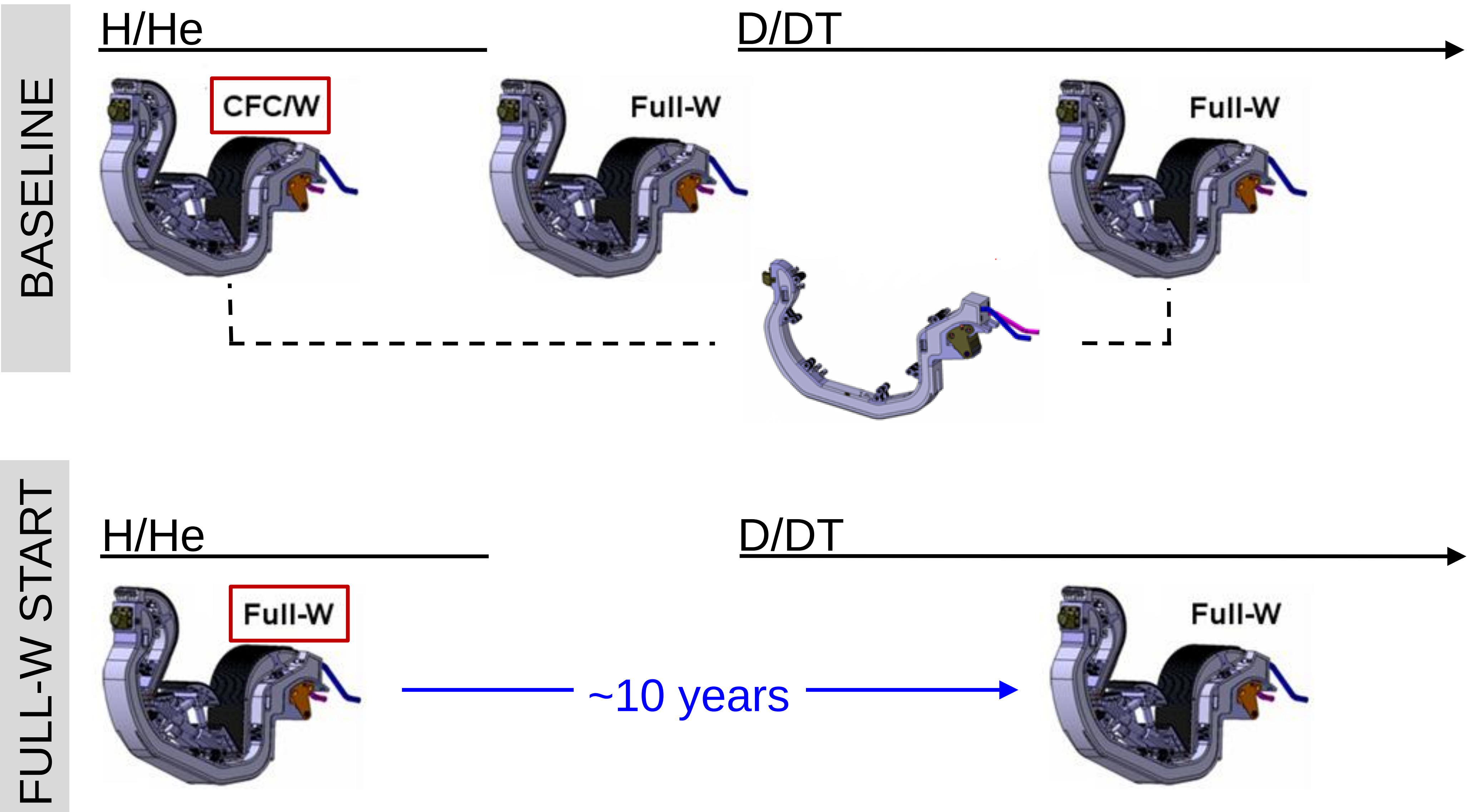
# The ITER tokamak



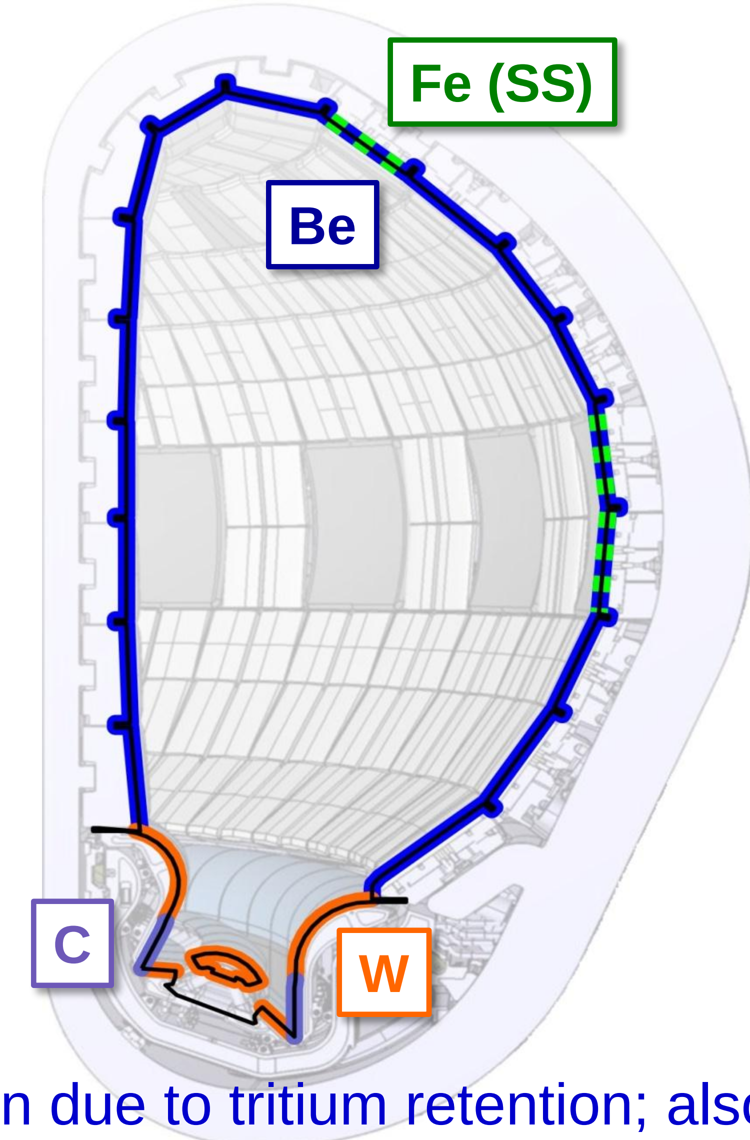
# Overview of major systems



# Divertor strategies



# Plasma facing materials




- No C during DT operation due to tritium retention; also a license condition

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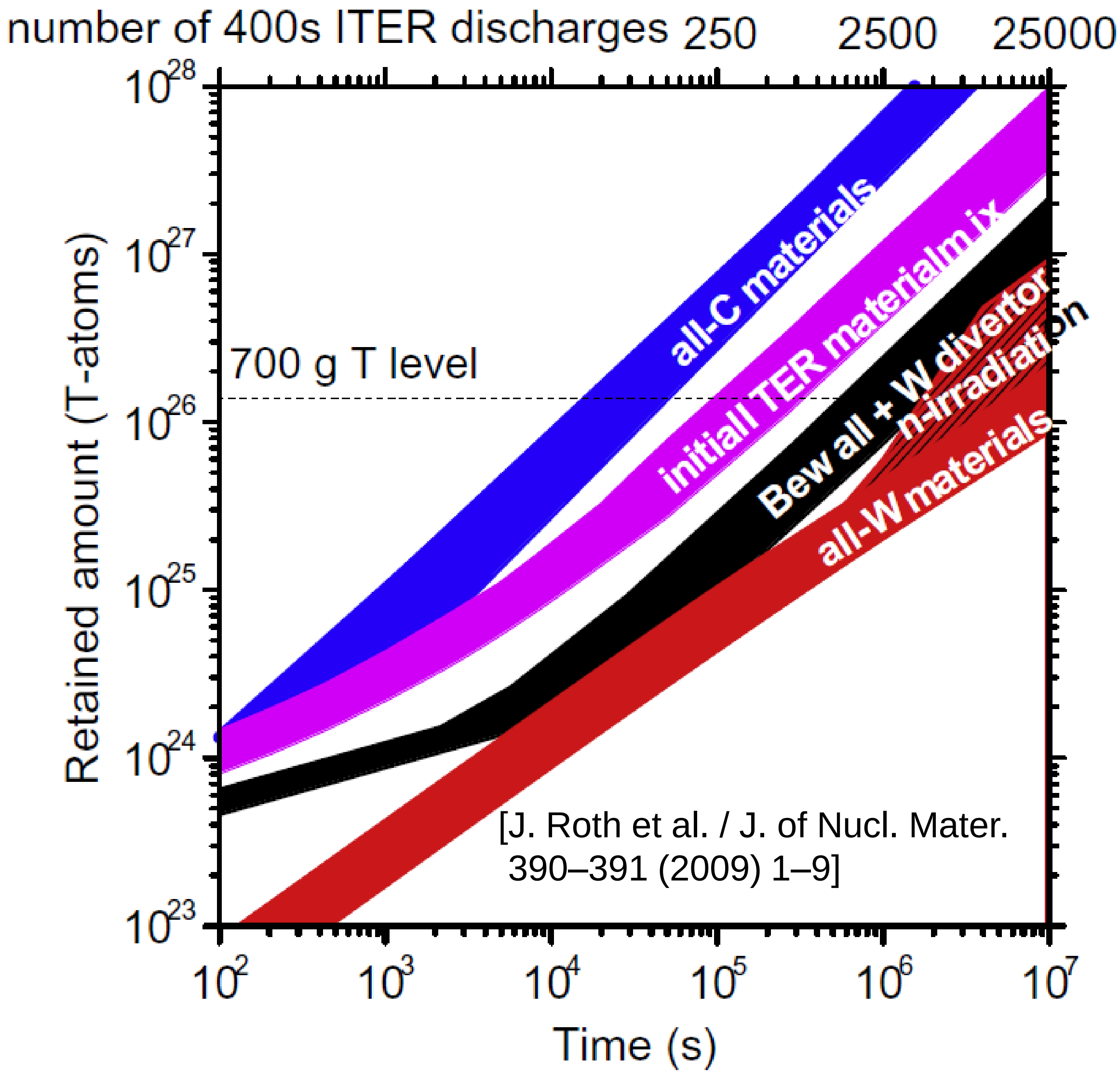
# Roth tritium retention estimates for ITER

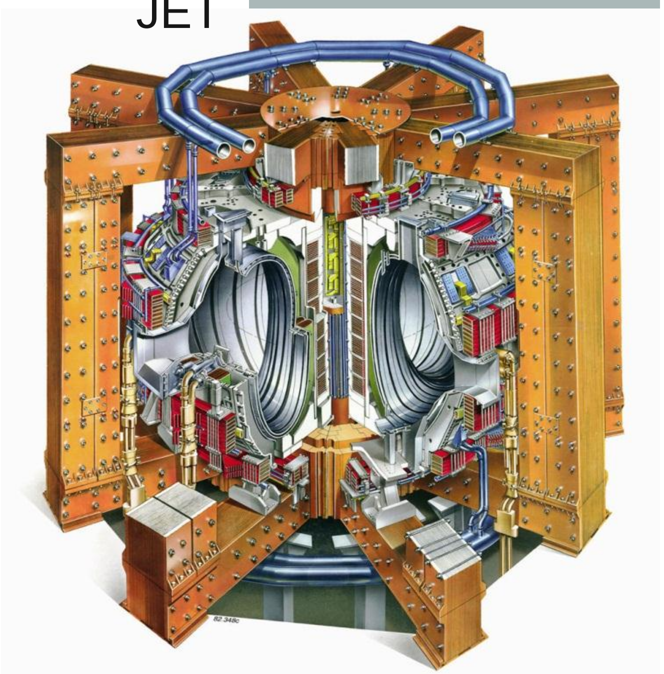


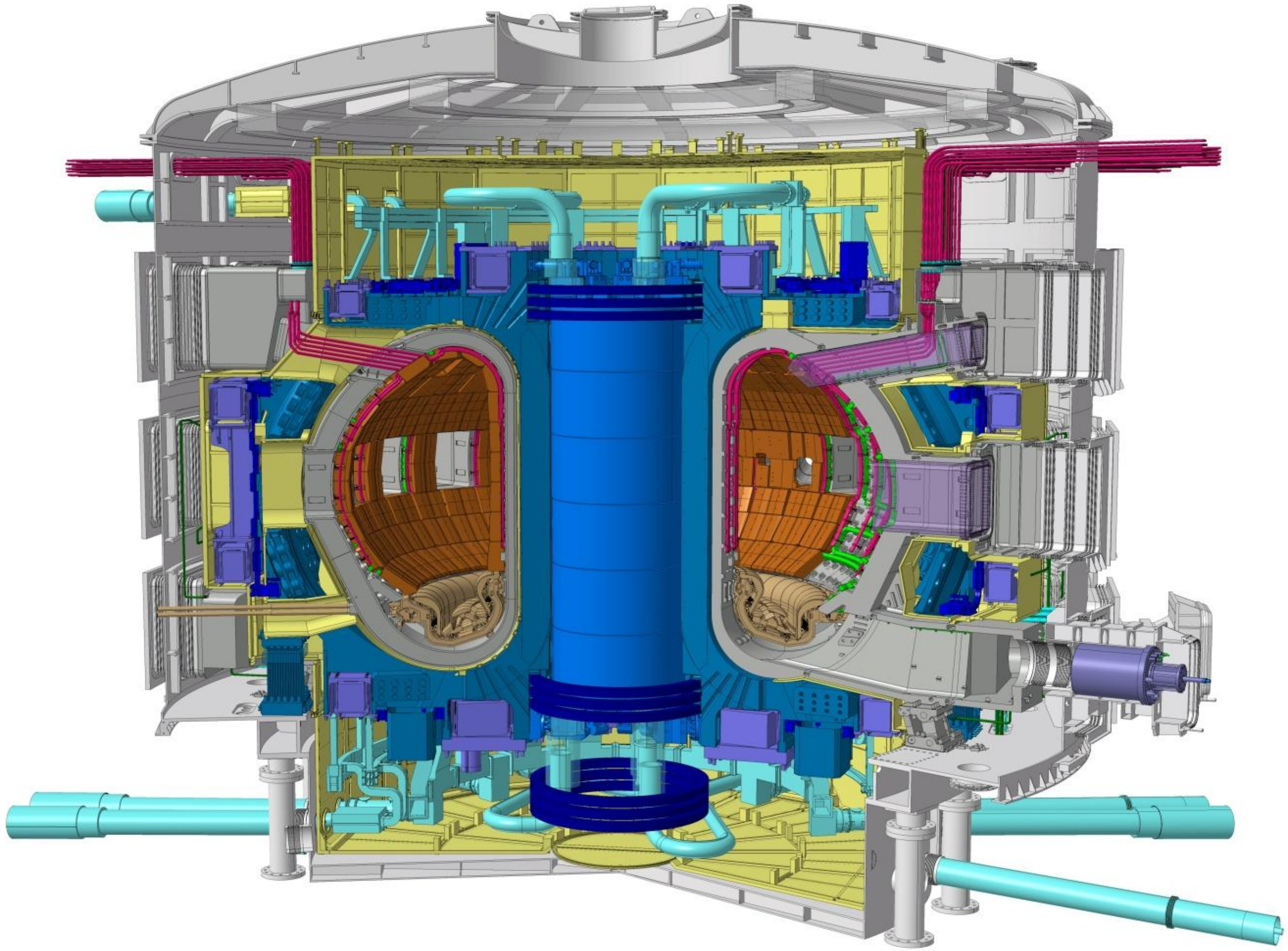
# Plasma flux to surfaces: plasma-wall contact

- Long-pulse, large size, and high density operation combine to give a *significant* increase in the ion fluence to the wall

DIVERTOR ION FL
(2000-2008, campa

JET

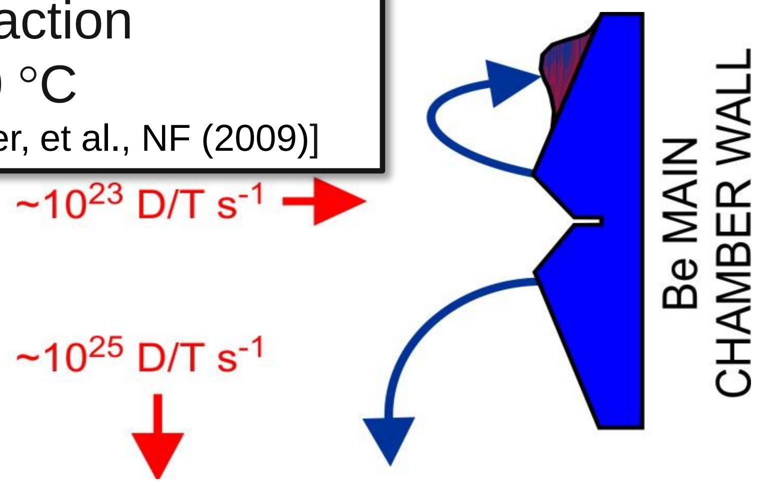




# Nuclear operation: tritium retention by co-deposition

- A 400 s Q=10 pulse will require <u>~50 g of T fuel</u>, but the maximum <u>mobilisable</u> in-vessel T inventory is <u>limited to 640 g</u> (+180 g in pumps, +180 g uncertainty)
  - nuclear safety (license) issue (and tritium is expensive too)

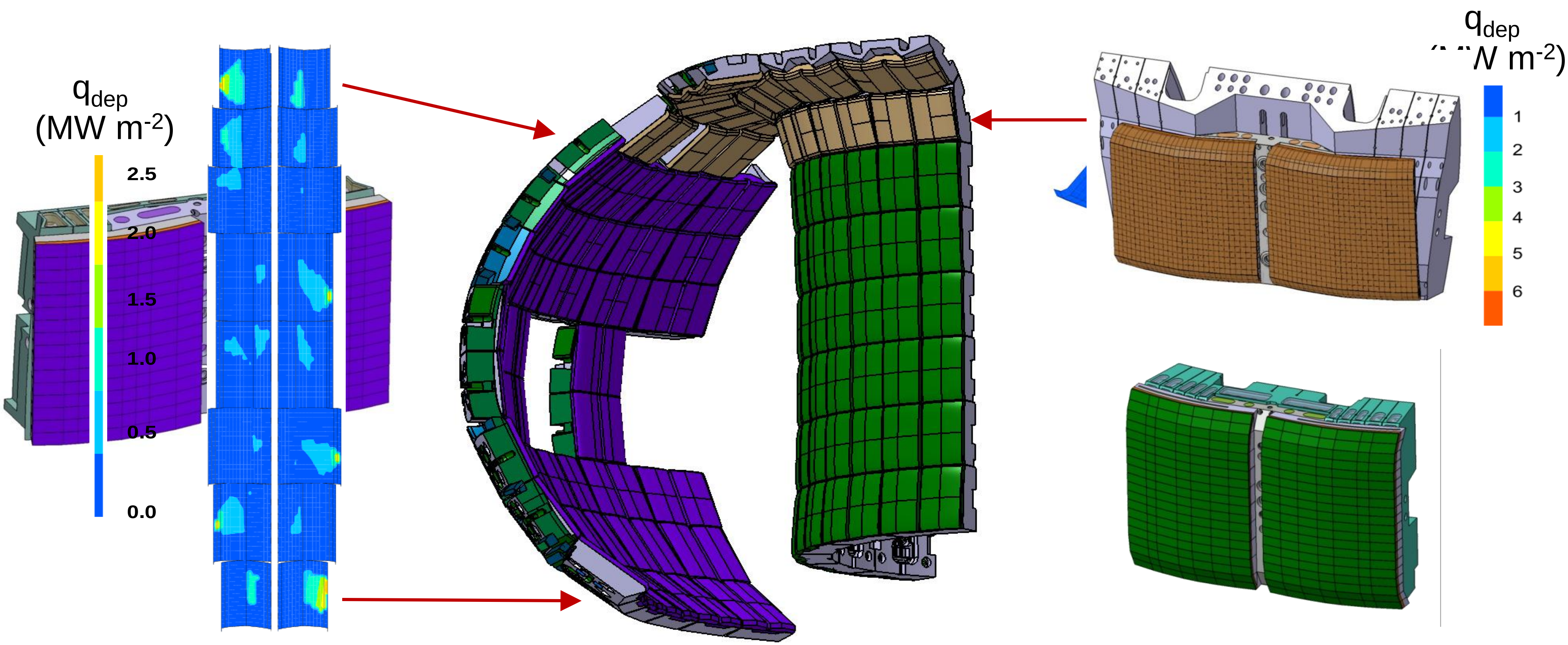
T:Be trapped fraction
~1:20 @ 200 °C
[G. De Temmerman, R. P. Doerner, et al., NF (2009)]




- Divertor bakeable to 350 °C, main wall to 240 °C → amount removed depends on surface temperature at deposition [J. Roth, 14th DivSOL ITPA, Korea (2010)]
  - (tritium trapped with carbon <u>cannot</u> be removed by a vacuum bake at 350 °C)
- Note: T:Be depends sensitively on deposition rate, incoming particle energy, and surface temperature → complex problem
  - efforts are underway to predict the level of T-retention [S. Carpentier, JNM, 2011]

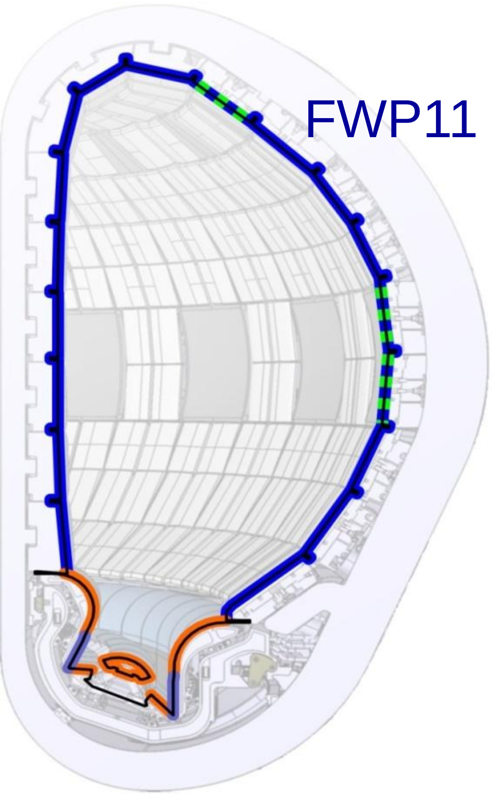
# Shaped ITER wall

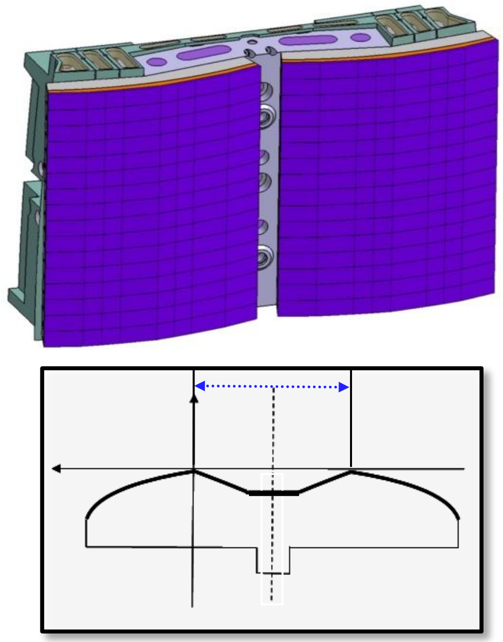
- Main chamber plasma-wall interaction is complicated by the close-fitting wall and shaped panels
  - plasma facing panels are replaceable, full change in 1-1.5 years (2-3 months for a single panel)

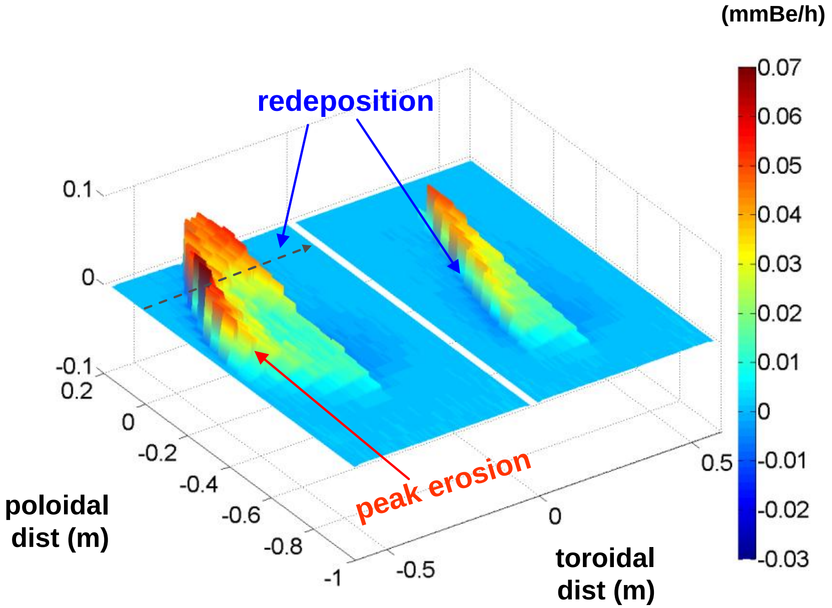


# Local quasi-2D erosion analysis of FWP11 with LIM

- 2D local model (i.e. only one panel represented), with 3D picture assembled from a series of "2D slides"
  - no transport of material across slices
  - benchmark against 3D ERO code [D. Borodin]

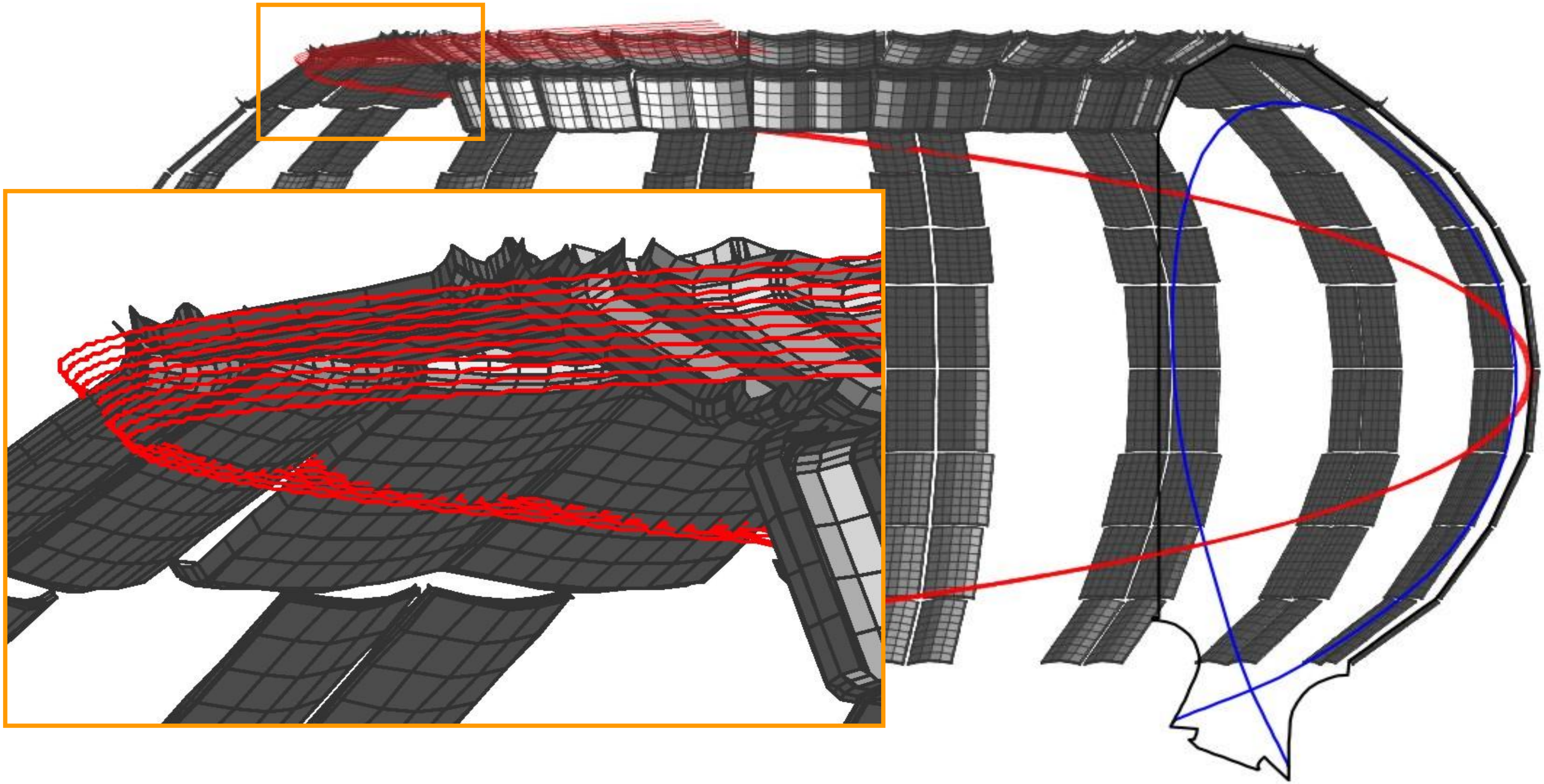


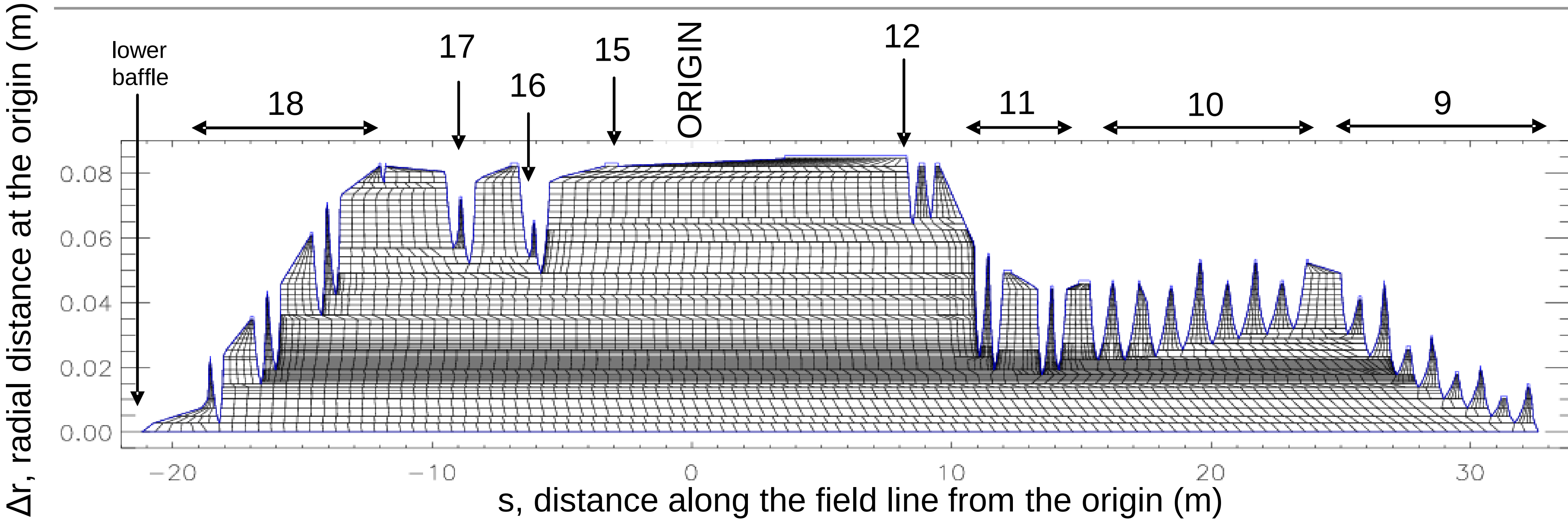


# “Medium-scale” modeling with DIVIMP

- DIVIMP with “ribbon grid” from field line tracing → quasi-3D representation of plasma contact with shaped wall panels, not fully local



# “Medium-scale” model grid and comparison with LIM




Differences between DIVIMP and LIM models:

✓ working directly with the equilibrium,

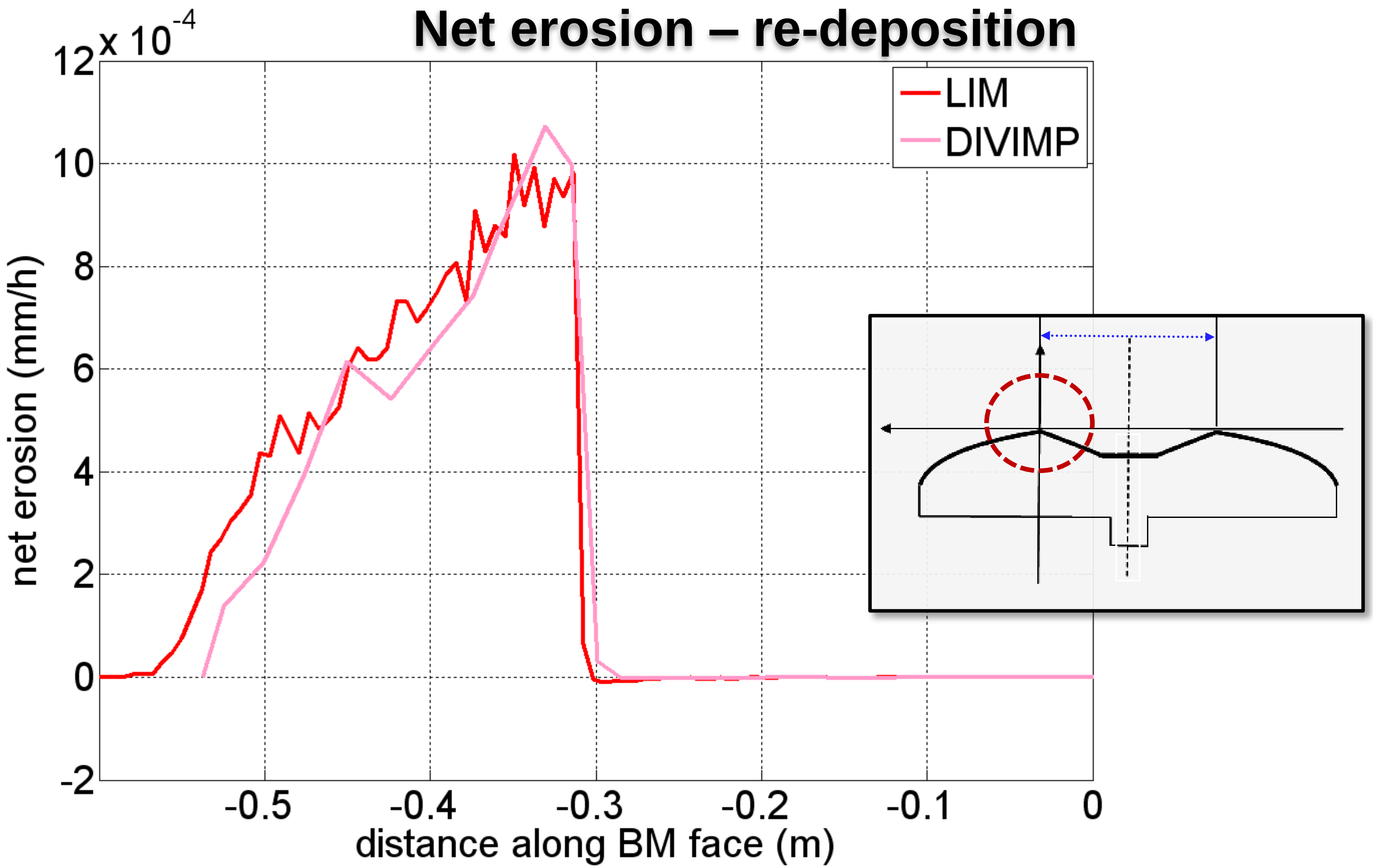
✓ identify all intersection points and keep track of the multiple PFRs (each field line can give many PFRs, depending on how often it intersects the wall),

✓ non-analytic calculation of BM-to-BM shadowing (wetted area),

✓can track impurity transport on the “medium scale”, i.e. between neighbouring BMs.

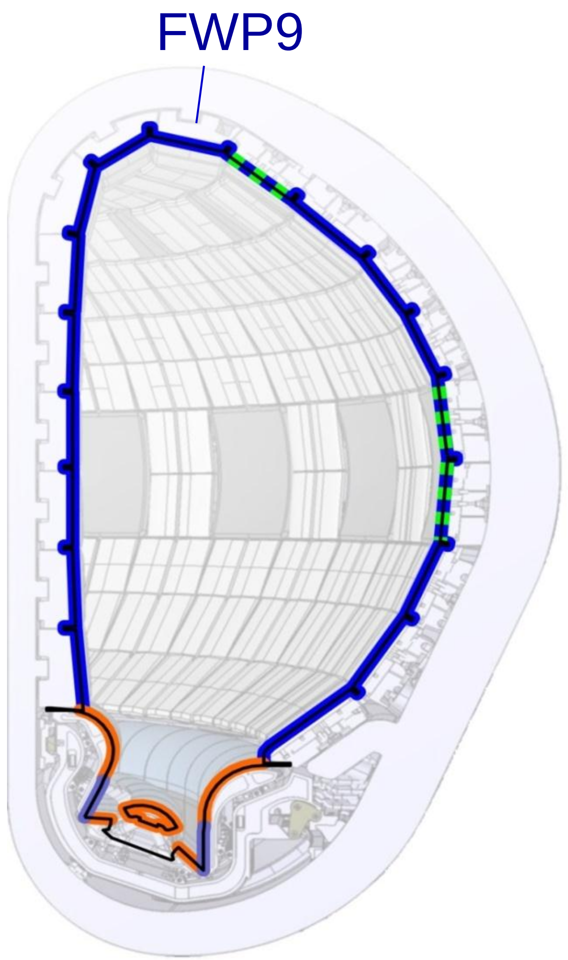
# “Medium-scale” modeling with DIVIMP

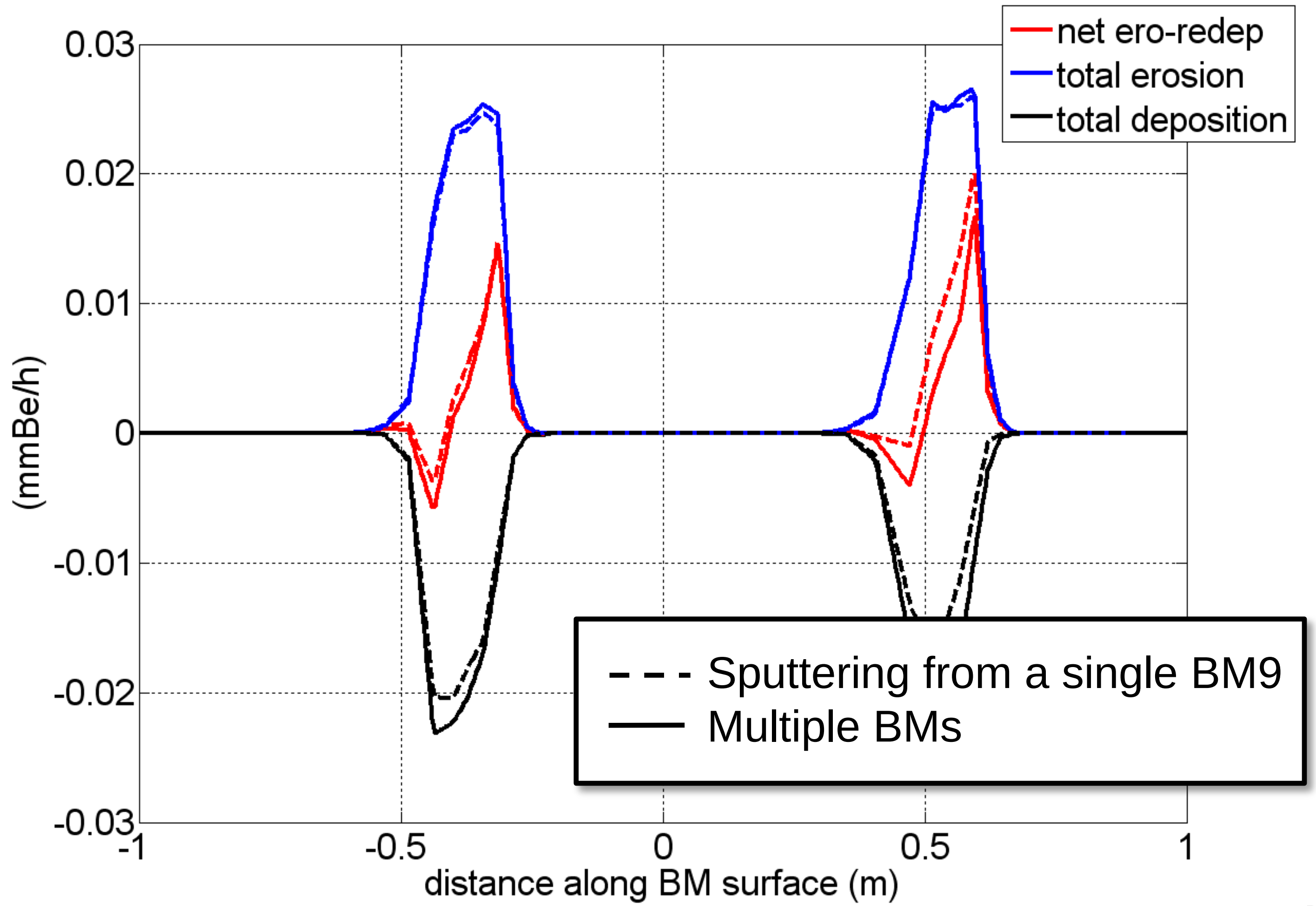
- Benchmarked against LIM for FWP11



# "Medium-scale" modeling with DIVIMP: upper panel

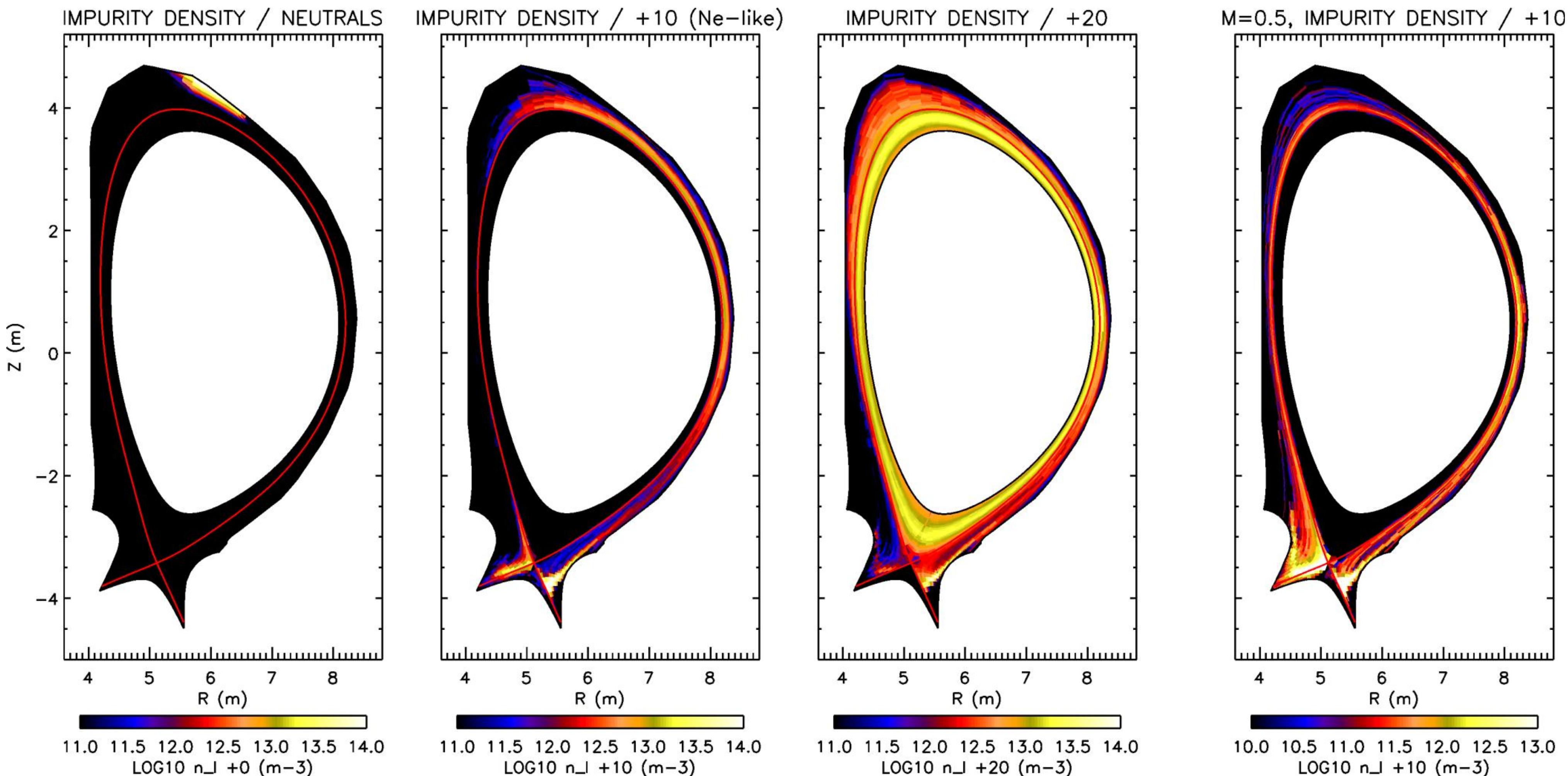
- Results: erosion and re-deposition on top FWP9: <u>local effect</u>
  - high density case, sputtering <u>BM9</u> row, $T_e$ = 10 eV, $T_i = 2.T_e$, no flow, $D_\perp = 3\ m^2\ s^{-1}$



# 2D global modeling with DIVIMP ("standard model")

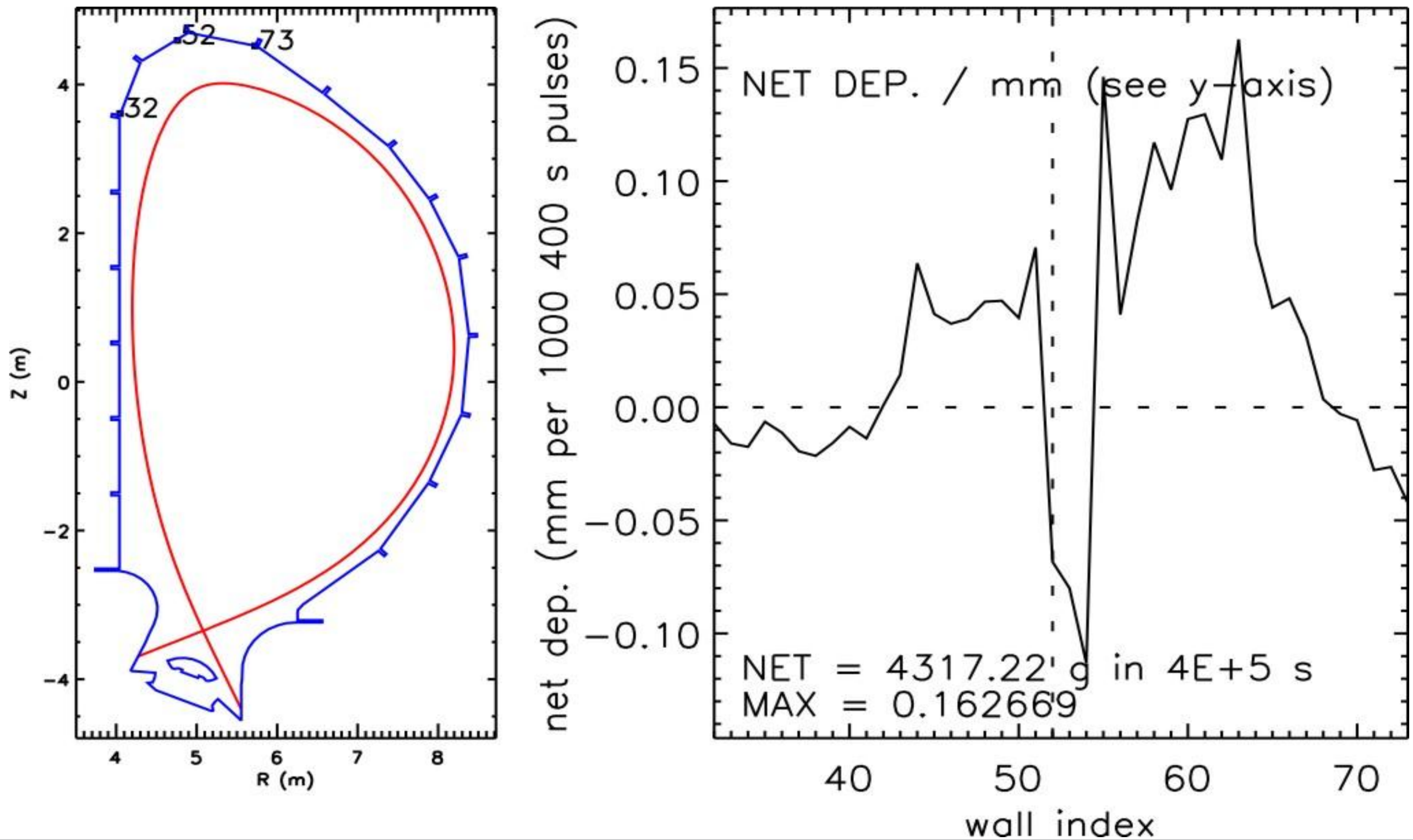
- Assume toroidal symmetry, i.e. 2D model where panel shape is not resolved
  - (example data shown here is for iron)



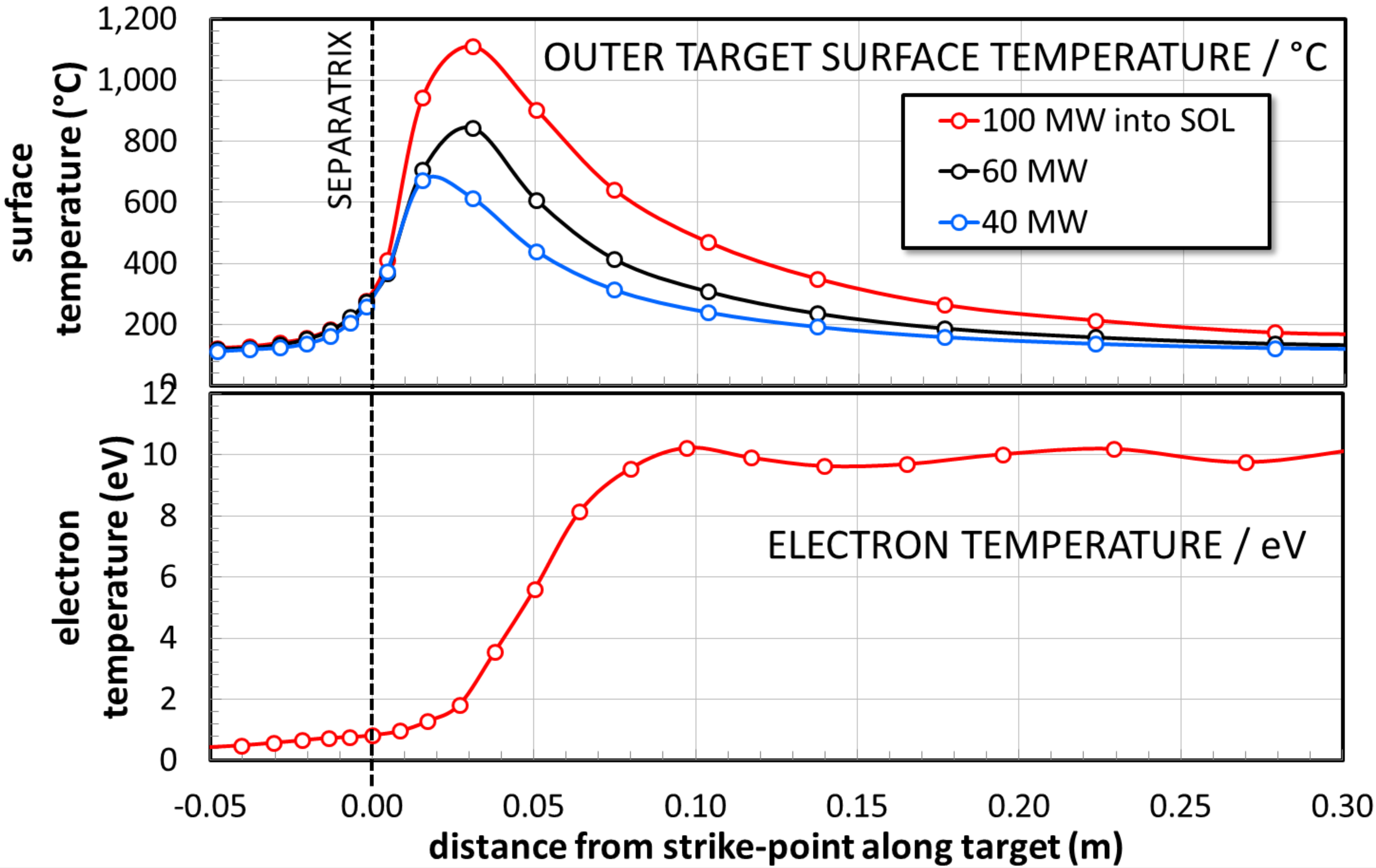
# 2D global modeling with DIVIMP: Be migration

- If upper panels are W, will they become coated with Be?
  - need improved material migration model and sputtering dynamics → <u>WALLDYN code</u> [K. Schmid,IPP]

Be DEPOSITION



# Surface temp. calculation with RACLETTE → Be evap.

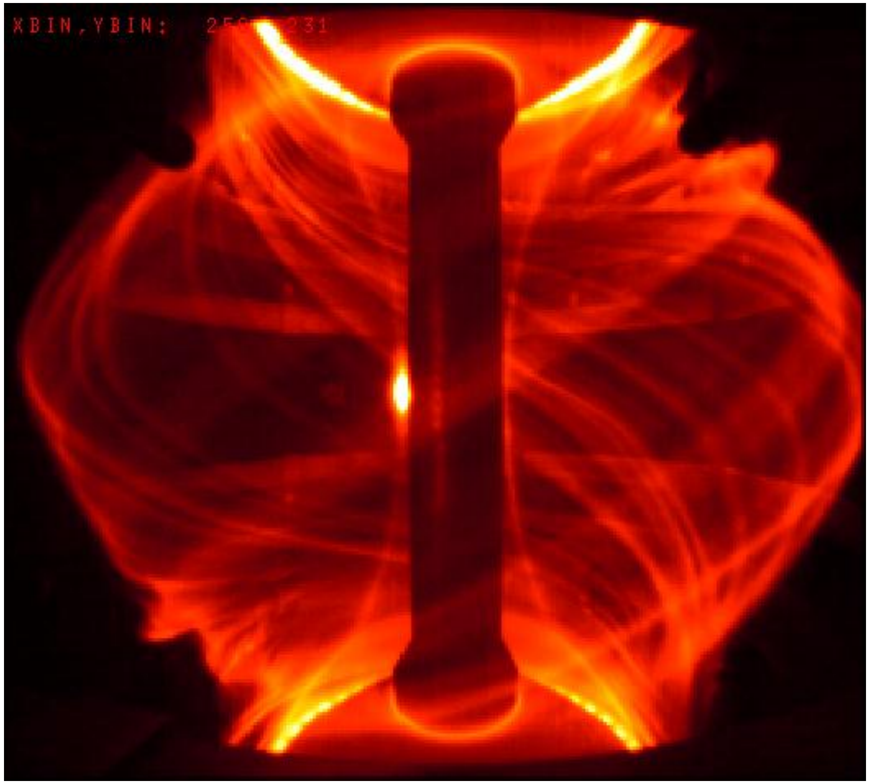


# 3D global transport

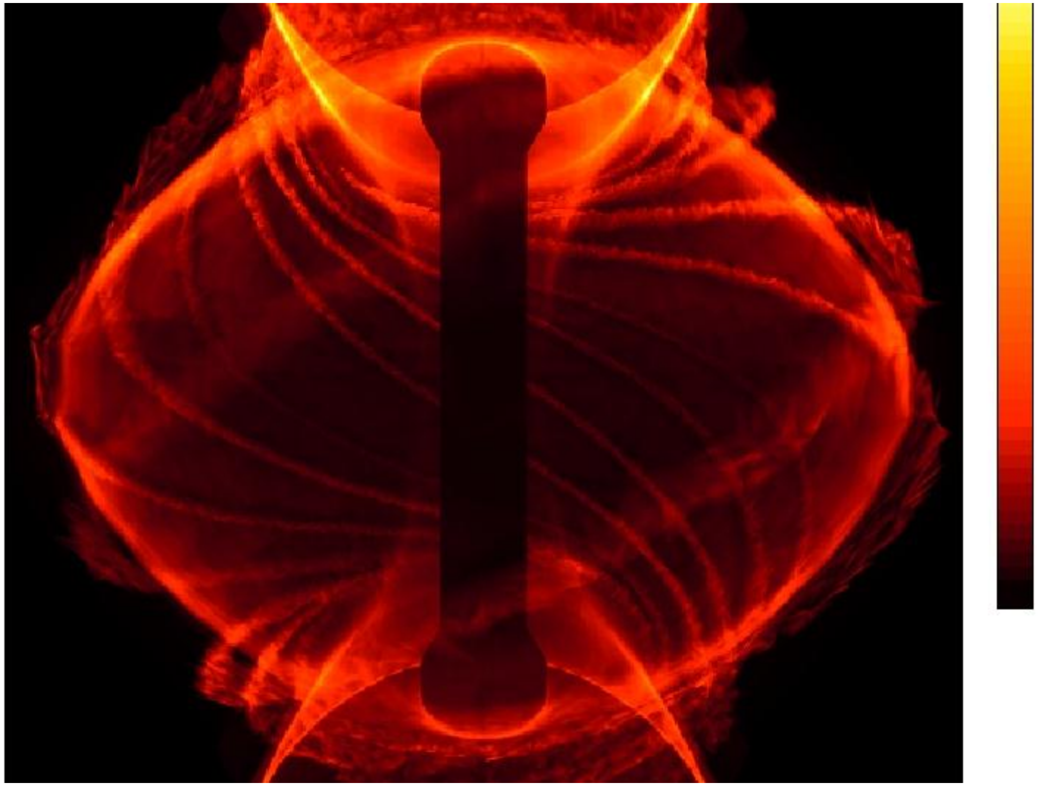
- Code extensions to 3D, time-dependent transport underway, i.e. semi-complete model
  - model framework established; post-doc starting next month; WALLDYN 3D extensions also planned

FAST CAMERA, UNFILTERED

MAST DISCHARGE 15622 @ 203.7 ms



OSM-EIRENE CODE



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# Estimates of first wall panel lifetime (circa. 2010)

- Large uncertainties in plasma specification and rates

**Plasma specifications** → **Modeling** ← **Sputtering Yields**

***high density:***
**Peak net erosion = 0.06 mm/h**

$Y_{eff}$ ~50%
*not applicable to lowest edge T° conditions*

**Peak ~ 0.0086 mm/h**
$Y_{eff}$ ~1% [PICSES-B]

***factor X24***

$Y_{eff}$ ~6-7% [Eckstein]

***factor x150***

***low density:***
**Peak net erosion = 0.0025 mm/h**

$Y_{eff}$ ~50% [JET]
**Peak = 0.021 mm/h**

**Peak ~ 0.0004 mm/h**
$Y_{eff}$ ~1%

**PFC lifetime: [~1500 – 225 000] discharges**
(assuming 10 mm Be thickness, steady-state phase ~ 400s)

– model framework established, post-doc starting next month, WALLDYN 3D exte

# Estimates of main chamber Be retention (circa. 2010)

IO (LIM) worst-case estimate: **3 gT/h** in the main chamber (wall fluxes ~ $2\ 10^{23}\ s^{-1}$)

**J. Roth estimates (PPCF 2008, JNM 2009)**

- Wall fluxes = [ $1.10^{23} – 5.10^{23}$] $s^{-1}$
- $Y_{eff}$ ~1-2 % (?)
- All Be eroded is assumed to be redeposited in PFC line-of-sight …
- … with fixed (D+T)/Be ~ 5%

➔ T-retention max ~ 0.4 gT/shot **(3.6 gT/h)**

(all machine)

**MIT working group on T-retention (April 2010)**

- Wall fluxes = [ $1.10^{23} – 1.10^{24}$] $s^{-1}$
- $Y_{eff}$ ~ 2 %
- Low density case => all the eroded material is assumed to be transported to the divertor
- High density case => 50% of material locally redeposited & associated codeposition not included
- (D+T)/Be = f ($T_{surf}$, $E_{imp}$, etc.)

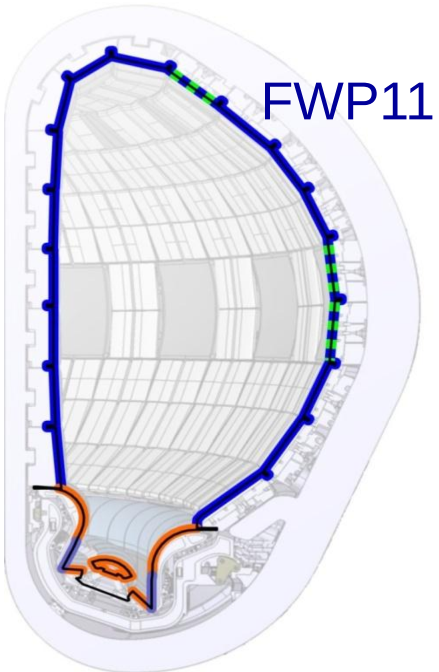
➔ T-retention max ~ 0.32 gT/shot **(2.9 gT/h)**

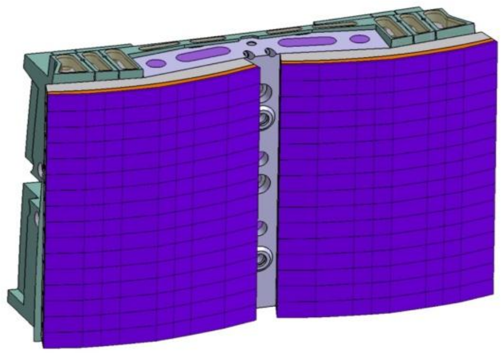
(only divertor)

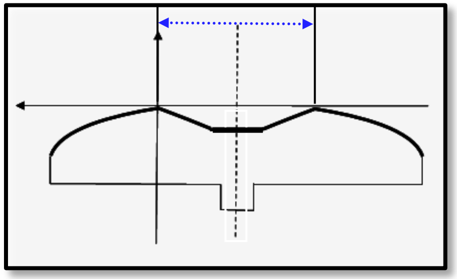
(final report can be found at: http://www.psfc.mit.edu/library1/catalog/reports/2010/10rr/10rr004/10rr004_full.pdf )

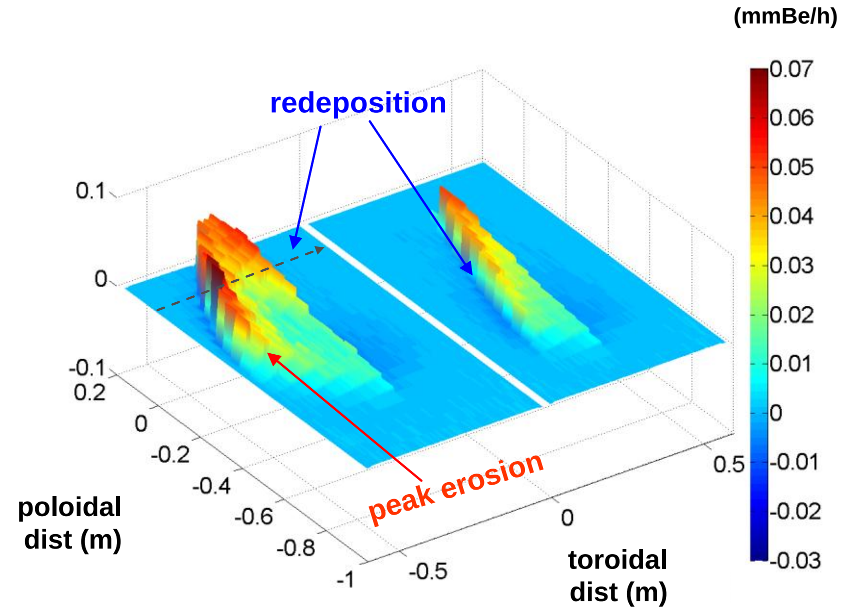
Caveat. The previous approaches and the one developed here are complementary but cannot be linearly combined to estimate a total in-vessel T-retention rate

# ITER shot limits from panel lifetime and T retention (LIM)








- $\langle Y_{eff} \rangle$ ~ 7%, ~50% particles locally re-deposited; net peak erosion ~0.06 mm/h
  ➔ PFC lifetime ~ 1500 shots (representative case)
- ~ 0.083 gT/h for one module ~ 3gT/h for 36 BM11-18 ➔ T-retention* limit ~ 1920 shots
- (assuming: 50:50 D:T plasma, maximum safety limit ~640g)

* 2D estimation of (D+T)/Be = f($T_{surf}$, $E_{imp}$, $\Gamma_D/\Gamma_{Be}$) [PICSES-B scaling law, G. De Temmerman, R> Doerner]

# Experimental benchmarking of model required!

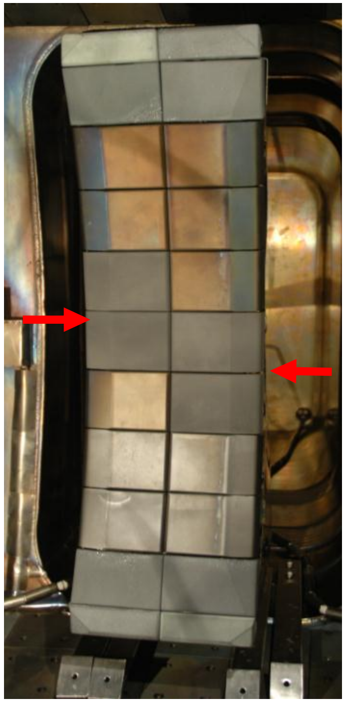
Codeposition ratios

Plasma specifications → Modelling ← Sputtering Yields

**Large uncertainties !**

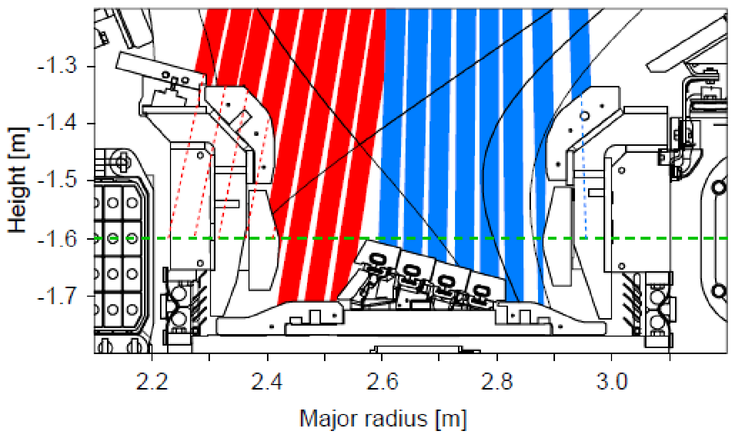
**In progress:**

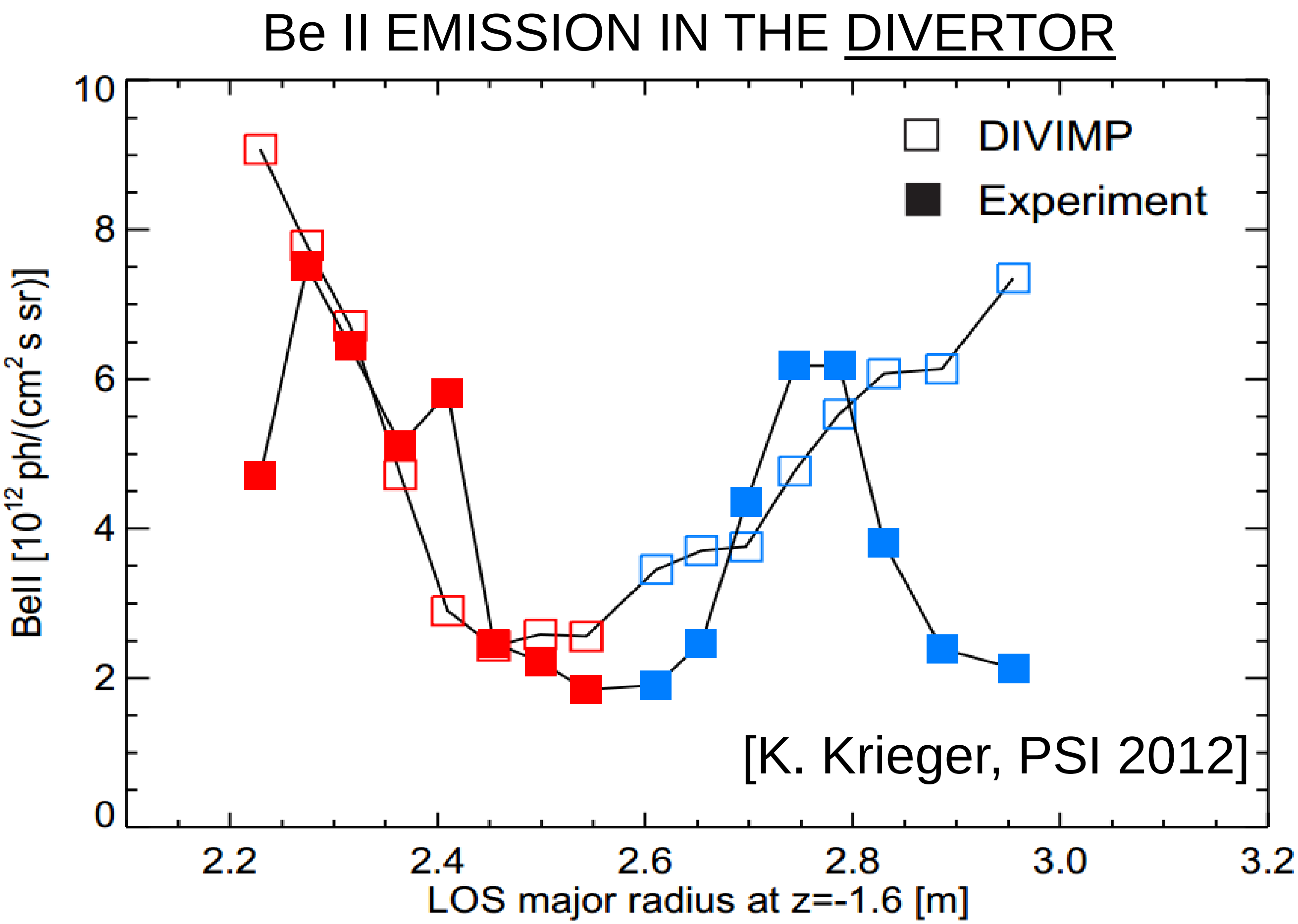
- ✓ Experimental benchmark (EAST dedicated experiments – start-up)
- ✓ JET ITER like wall migration



# JET ILW modeling of Be main wall erosion and transport

- Same DIVIMP model as used with ITER design work
  - preliminary results only, incomplete model (no CX sputtering for example)



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- Summary (1 slide)

# Rates

- The D→Be and Be→Be sputtering yields: W. Eckstein, Report IPP 9/132 (2002) Garching (angle-averaged yield)
  - which are consistent with accelerator erosion yield measurements [J. Roth et al., Fus. Eng. Design, 37 (1997) 465-480]
  - and JET divertor data [M.F. Stamp et al., J. Nucl. Mat. (2011), in press]

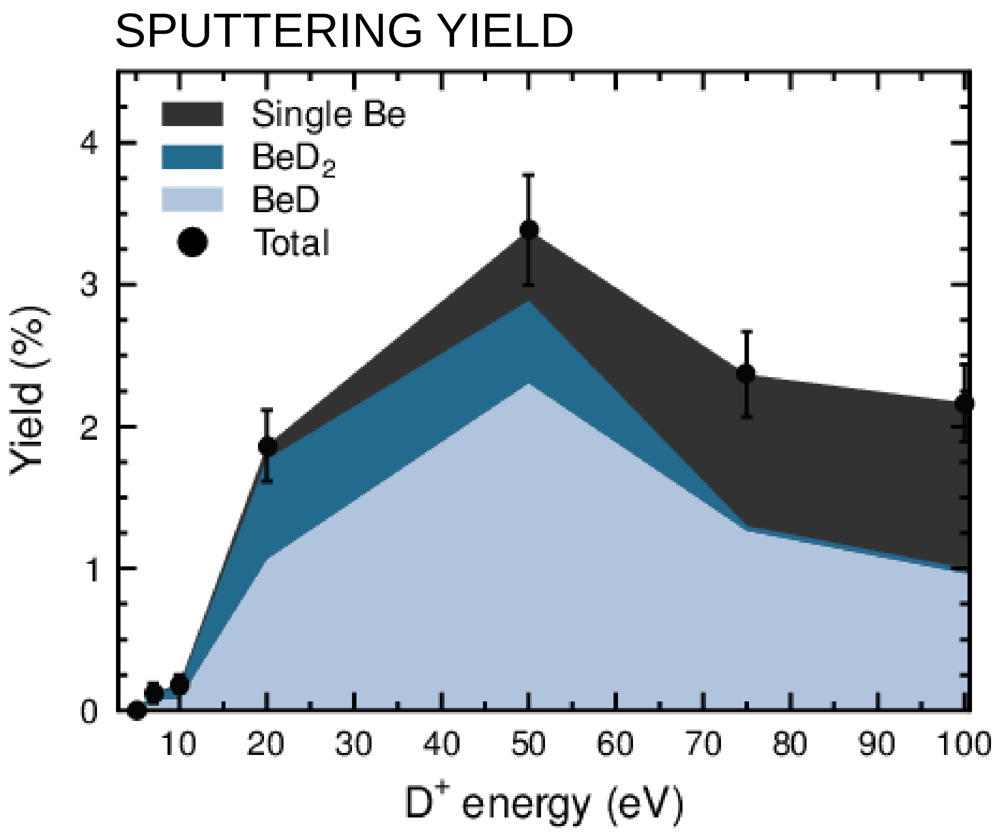
- Also rates x2 to account (approximately) for surface roughness

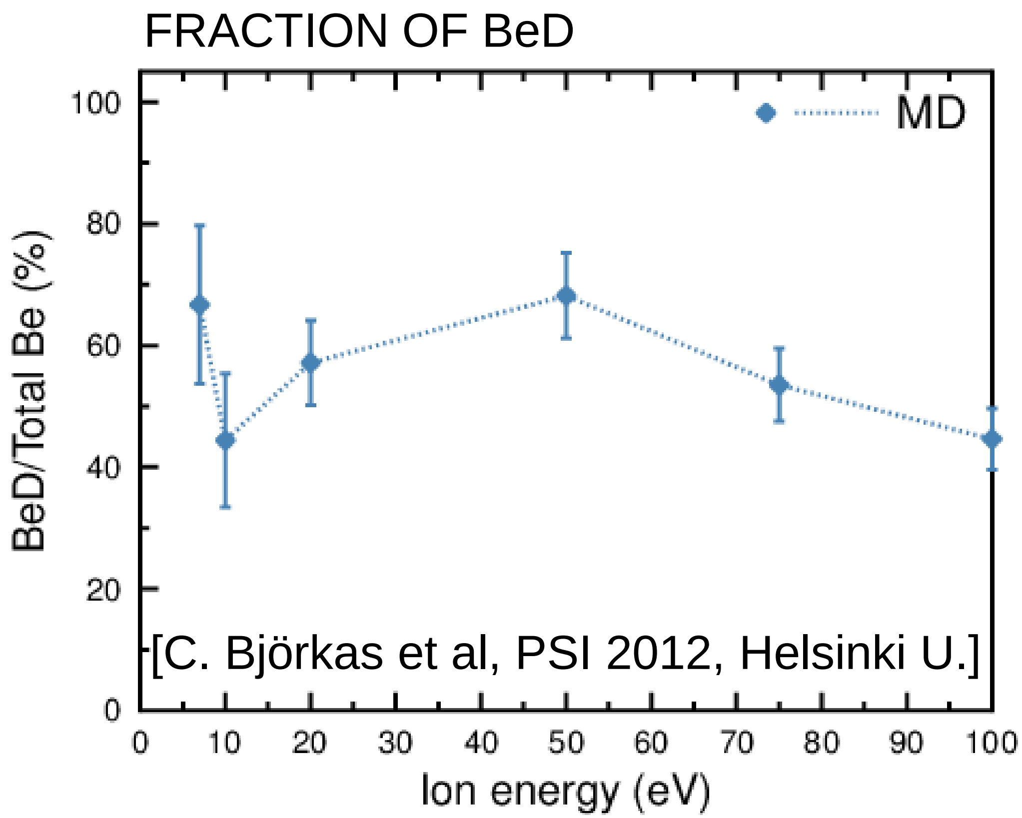
- Be molecular effects not included

- Some inconsistency with yields from accelerator measurements…

# Be MD sputtering rate calculations

- Molecular dynamics simulations are being performed to refine the code-calculated Be sputtering yields and estimate the molecular fraction

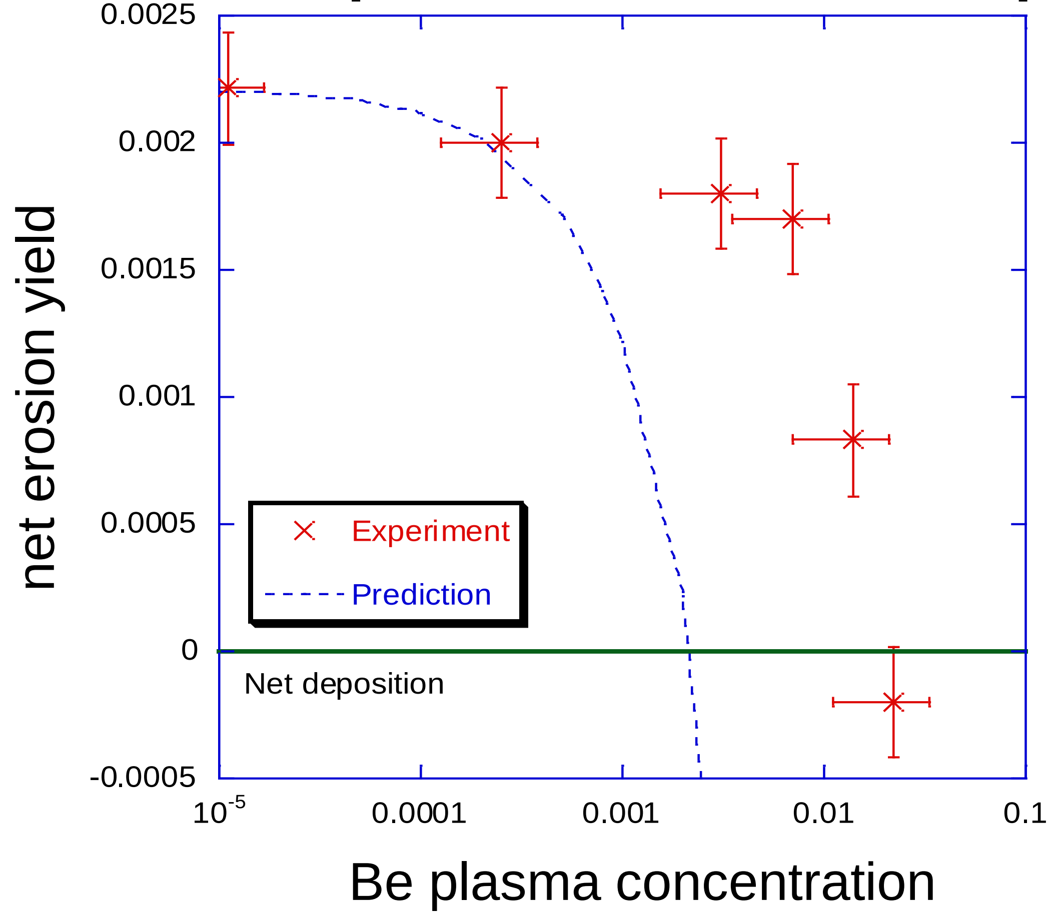


# Be sputtering rate: experimental results: PISCES-B

- Results suggest that the sputtering rate of re-deposited material has a significantly higher sputtering rate than the bulk atoms

[R. Doerner et al., PSI 2012, UCSD]



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# Summary

- ITER has attempted to estimate Be panel erosion, and the associated T co-deposition, using a “progression” of codes
- The initial local, quasi-2D calculations will eventually be superseded by fully 3D, time-dependent simulations that include the wall shape, but the timescales for completion of the work are uncertain
- Panel life time and tritium retention operational limits have a wide range and required further model refinements, improved rate data, and experimental benchmarking
- More accurate specification of the boundary plasma conditions is also needed