Studies of Next-Step Spherical Tokamaks Using High-Temperature Superconductors

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U.S. DEPARTMENT OF ENERGY

PRINCETON UNIVERSITY

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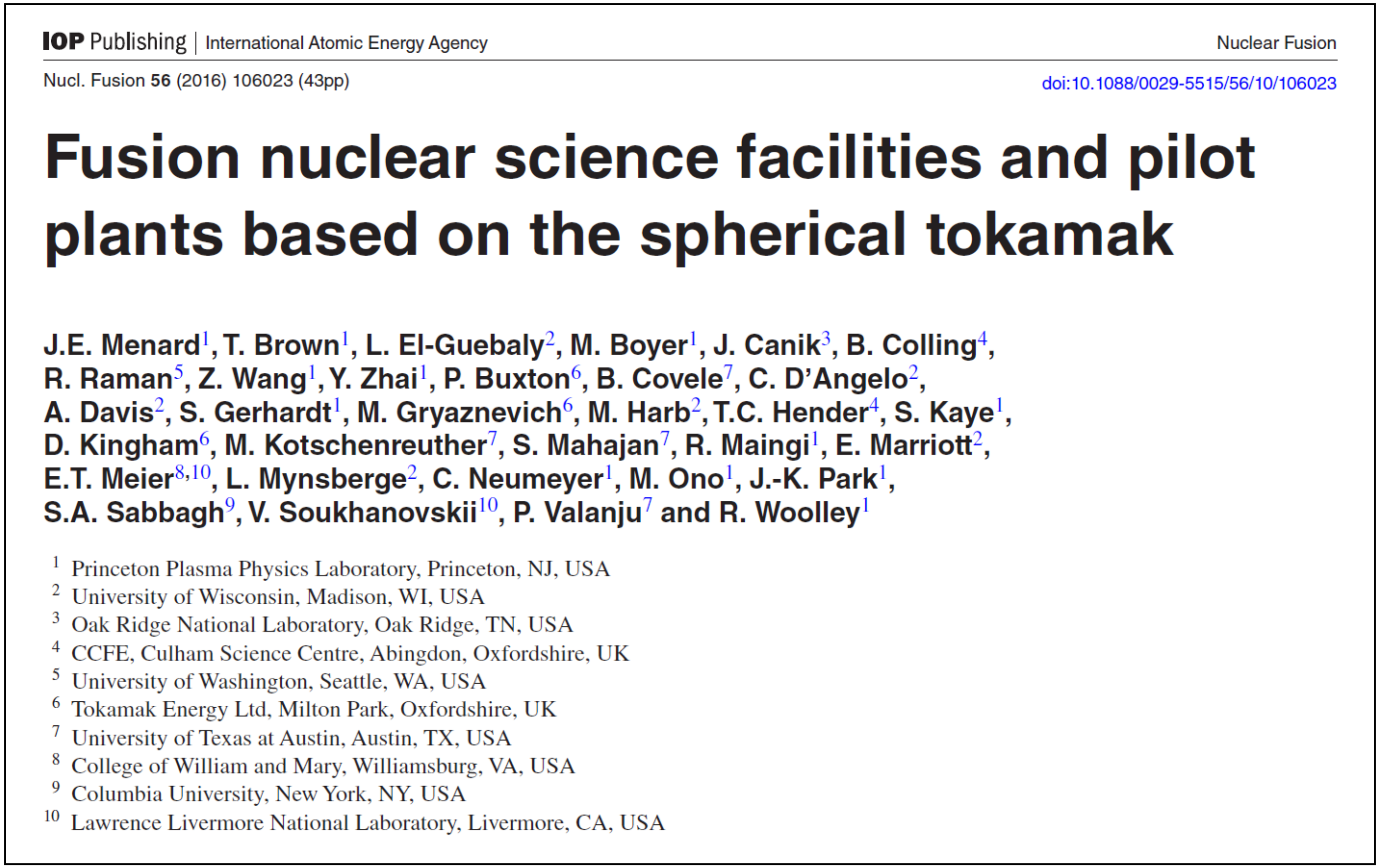
PPPL leading multi-institutional collaborative effort exploring low aspect ratio tokamak concepts



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Fusion nuclear science facilities and pilot plants based on the spherical tokamak

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Paper summarizing 5 year study of Cu and HTS STs recently published in Nuclear Fusion

Possible missions for next-steps

1. Integrate high-performance, steady-state, exhaust
    - ➢ Divertor test-tokamak - DTT
2. Fusion-relevant neutron wall loading
    - ➢ $\Gamma_n \sim$ 1-2MW/m^2, fluence: $\geq$ 6MW-yr/m^2
3. Tritium self-sufficiency
    - ➢ Tritium breeding ratio TBR ≥ 1
4. Electrical self-sufficiency
    - ➢ $Q_{eng} = P_{electric} / P_{consumed} \sim 1$
5. Large net electricity generation
    - ➢ $Q_{eng} >> 1$, $P_{electric} = 0.5\text{-}1$ GWe

This talk will discuss PPPL-led studies of how low-A "spherical" tokamaks could fulfill these missions

What is optimal A for HTS FNSF / Pilot Plant?

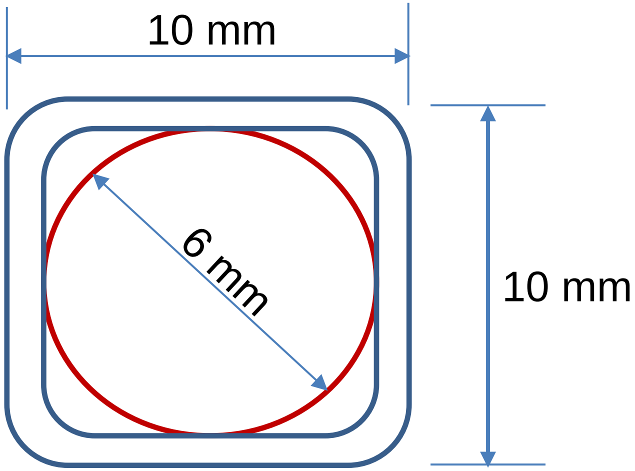
- P_{fus} / V ~ $\varepsilon(\beta_N \kappa B_T)^4$ at fixed bootstrap fraction
- β_N and κ increase at lower aspect ratio
- B_T decreases at lower A – depends strongly on:
  - Inboard shielding, HTS allowable field and current density

Approach:

- Fix plasma major radius and heating power (50MW)
  - R_0 = 3m – smallest size for $Q_{eng} > 1$ and high fluence
- Apply magnet & plasma constraints (see backup)
  - HTS strain: 0.3%, β_N: n=1 no-wall, κ: 0.95 × limit, $f_{GW} = 0.8$
- Vary aspect ratio from A = 1.6 to 4
- Vary HTS current density, peak field
  - Also scan inboard shielding thickness
- Compute Q_{DT}, Q_{eng}, and required H_{98} (*unconstrained*)

HTS cables using REBCO tapes achieving high winding pack current density at high B_T

Conductor on Round Core Cables (CORC)
J_{WP} ~ 70MA/m^2 19T

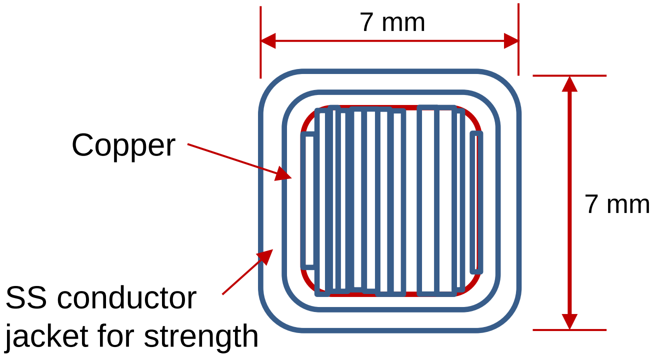


7 kA CORC (4.2K, 19 T) cable

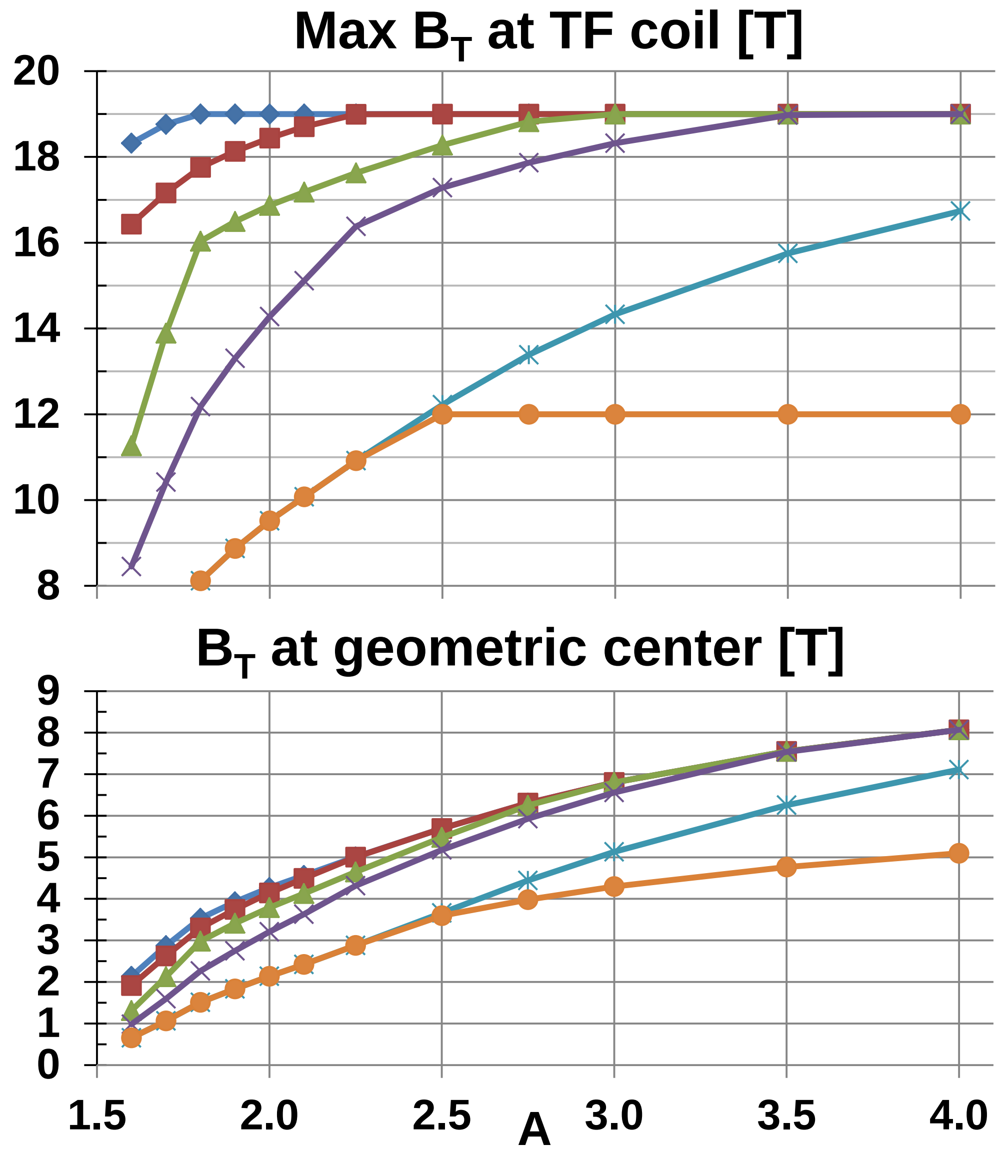
Base cable: 50 tapes YBCO Tapes with 38 μm substrate (Van Der Laan, HTS4Fusion, 2015)

Higher J_{cable} HTS cable concepts under development:

Base Conductor
He Gas Cooled
8kA,
J_{WP} ~ 160MA/m^2



High TF winding-pack current density required to access highest B_T at lower A



19T: Present CORC HTS limit

12T: ITER-like TF coil limit

(Nb_3Sn, 11.8T)

J_{WP} [MA/m^2]

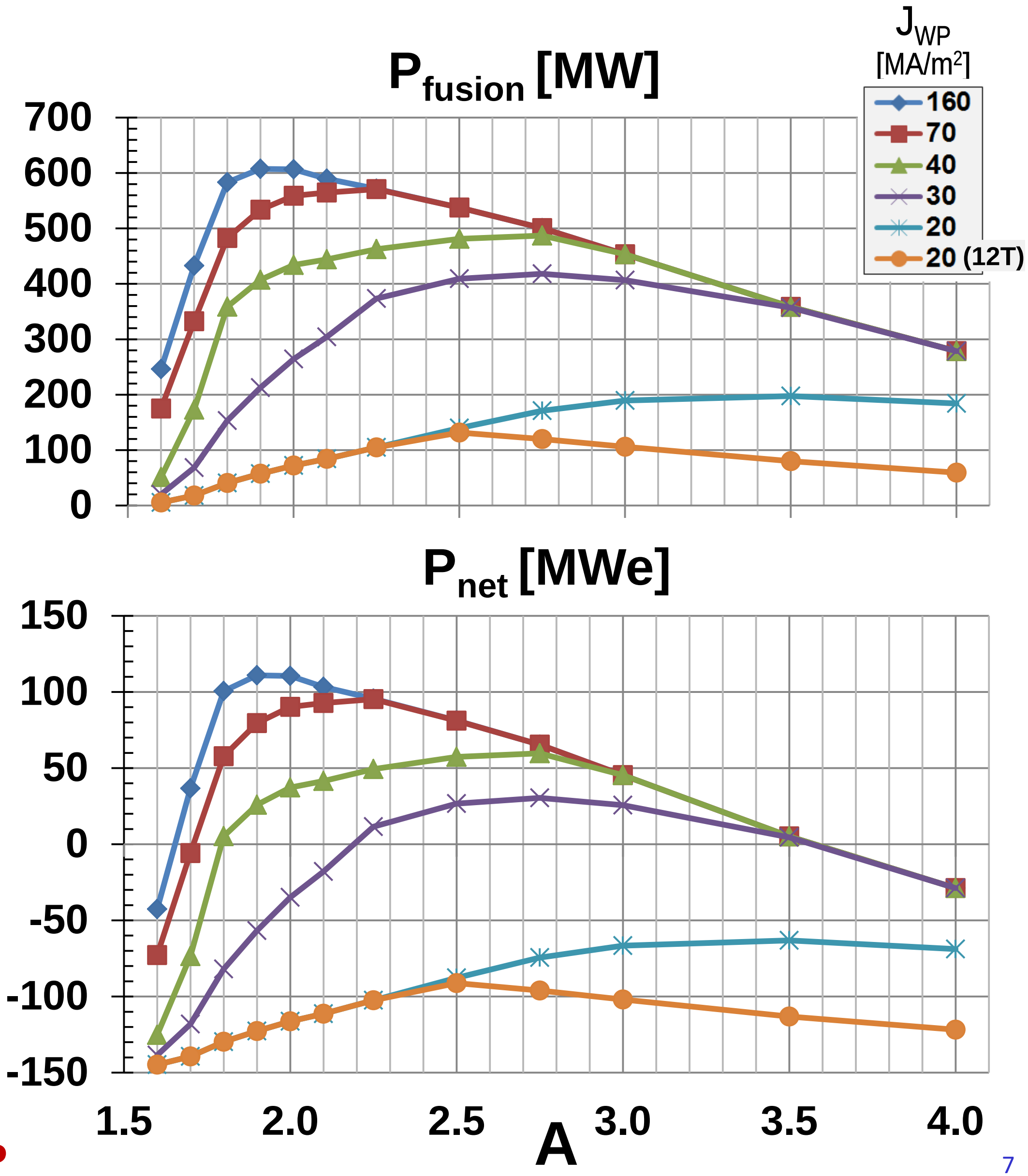
- ◆ 160
- ■ 70
- ▲ 40
- × 30
- ✳ 20
- ● 20

- Coil structure sized to maintain ≤ 0.3% strain on winding pack for all cases shown here
- Effective inboard WC neutron shield thickness = 60cm

High current density HTS cable motivates consideration of low-A tokamak pilot plants

- ITER-like TF constraints:
  - J_{WP}=20MA/m^2, B_{max} ≤ 12T
  - P_{fusion} ≤ 130MW
  - **P_{net} < -90MW**
- J_{WP} ~ 30MA/m^2, B_{max} ≤ 19T
  - P_{fusion} ~ 400MW
  - **Small P_{net} at A=2.2-3.5**
- J_{WP} ≥ 70MA/m^2,B_{max} ≤ 19T
  - P_{fusion} ~500-600MW
  - **P_{net} = 80-100MW at A=1.9-2.3**

A ~ 2 attractive at high J_{WP}

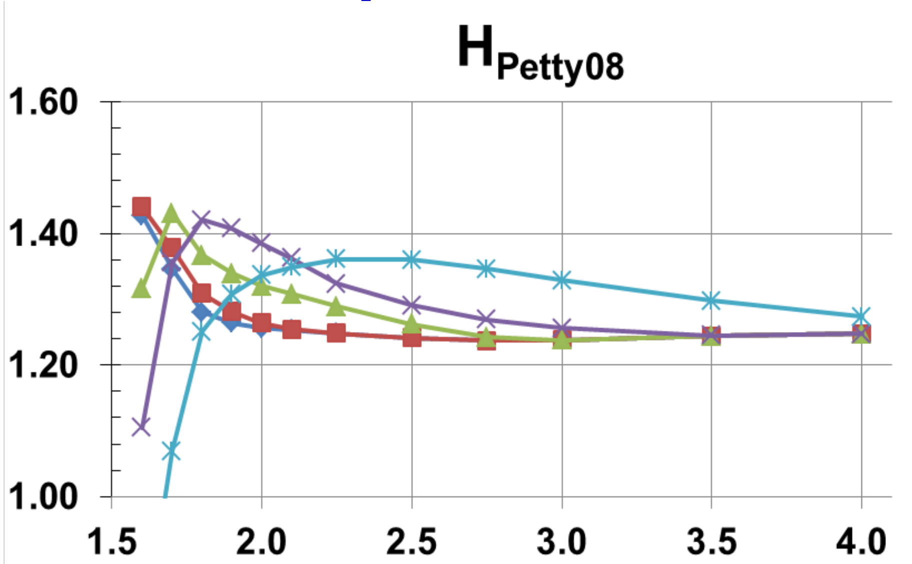


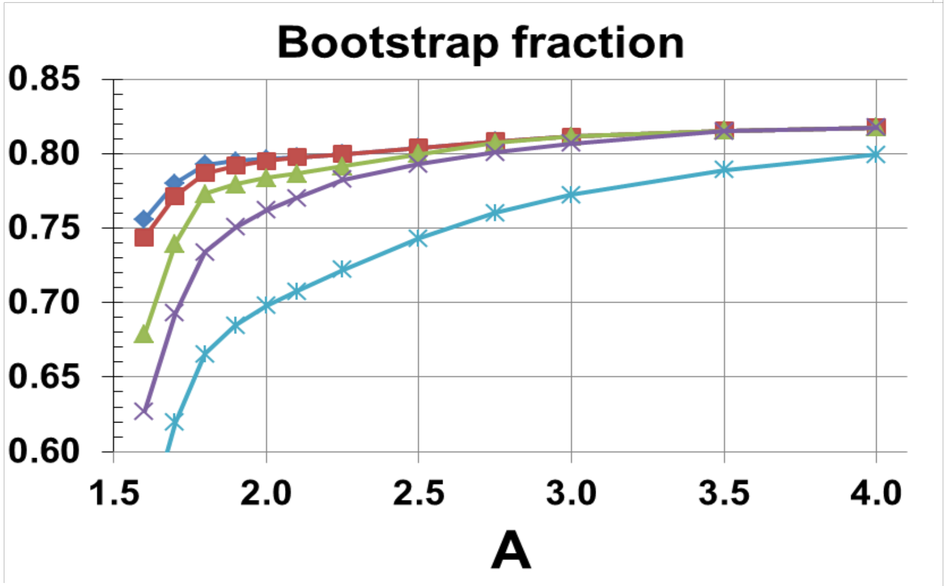
A ≥ 2 pilot plant scenarios have elevated H > 1, f_{BS} ~ 80%, I_P = 6-12MA

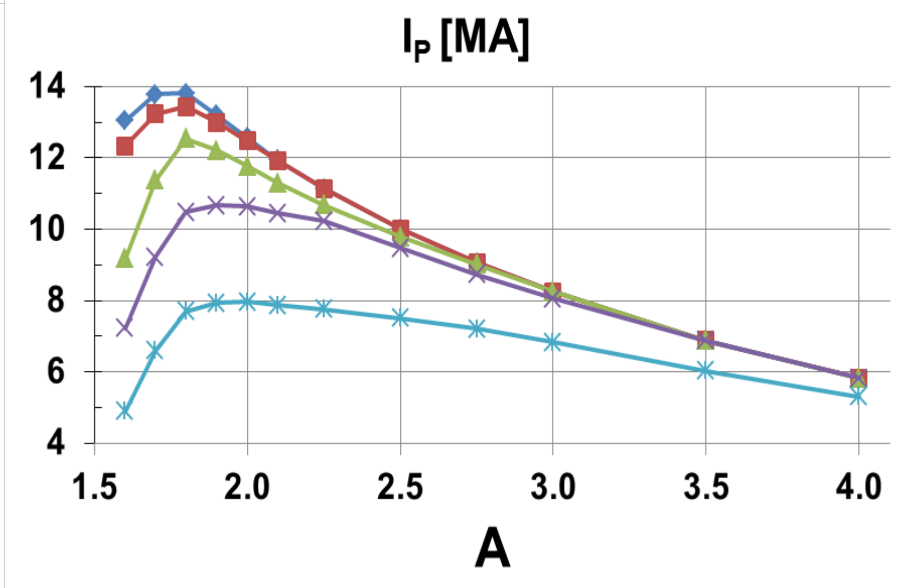
J_{WP} [MA/m^2]

- ◆ 160
- ■ 70
- ▲ 40
- × 30
- ✳ 20

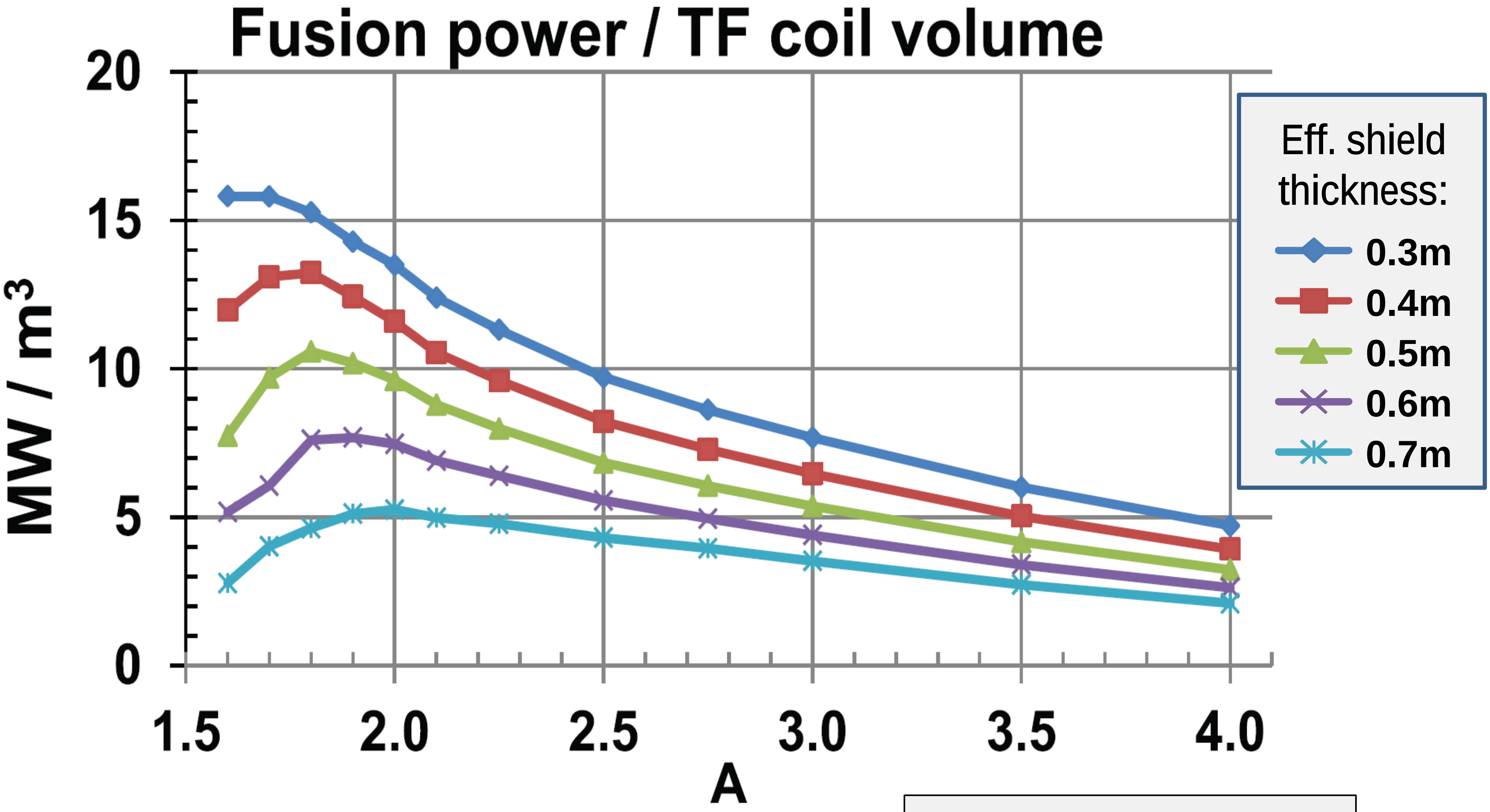
Effective inboard WC neutron shield thickness = 60cm





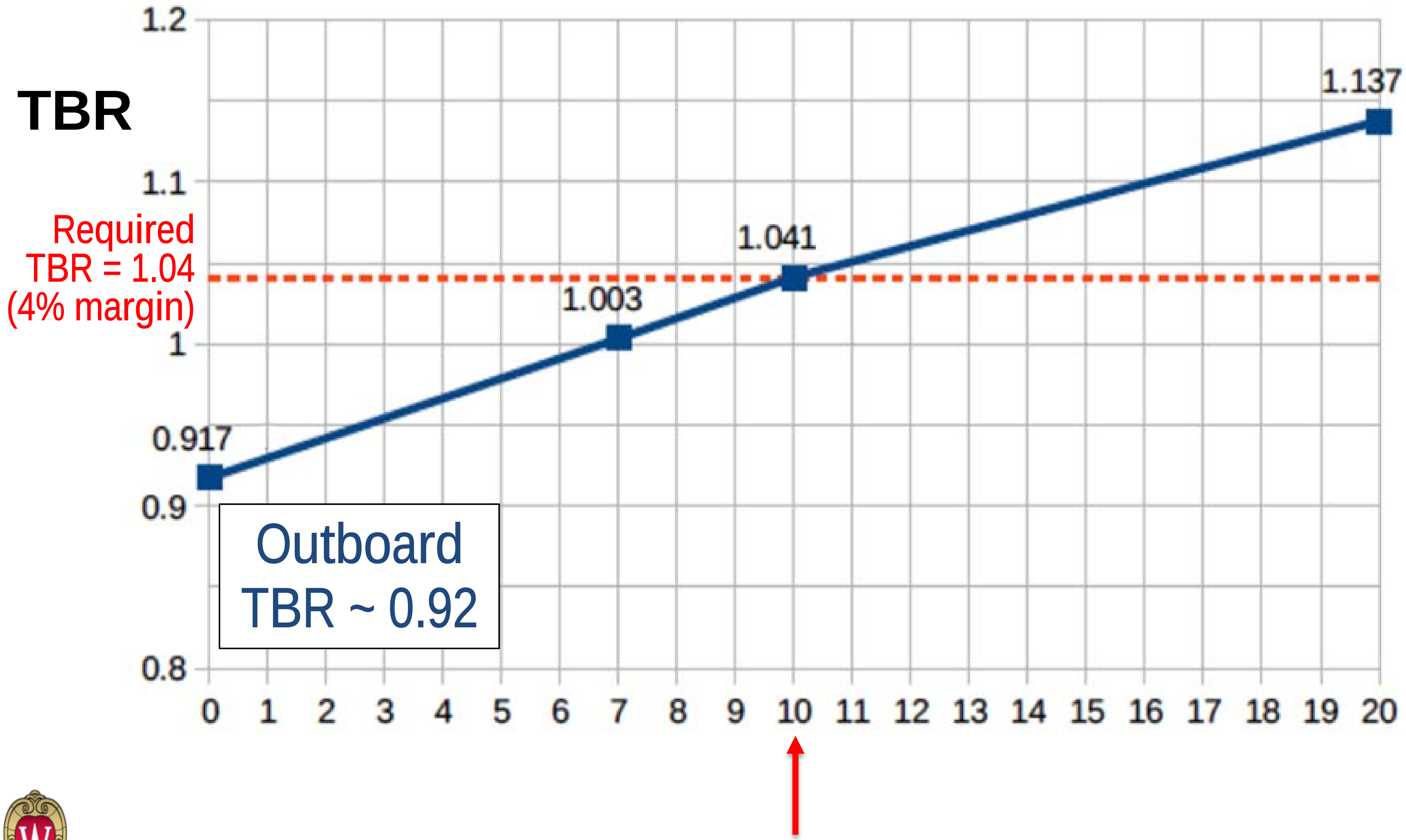


A ≤ 2 maximizes TF magnet utilization



J_{WP} = 70MA/m^2

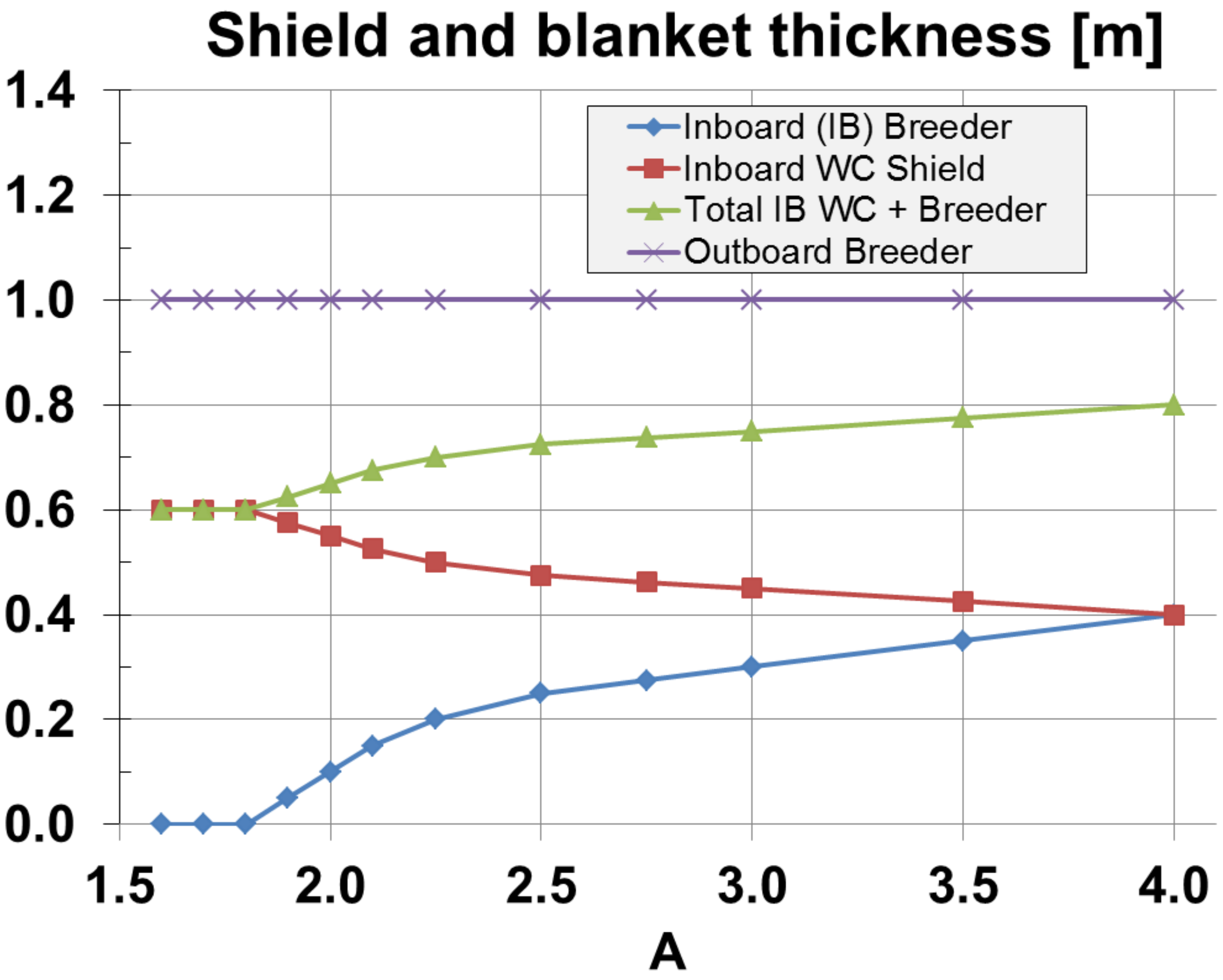
Need inboard breeding for TBR > 1 at A=2



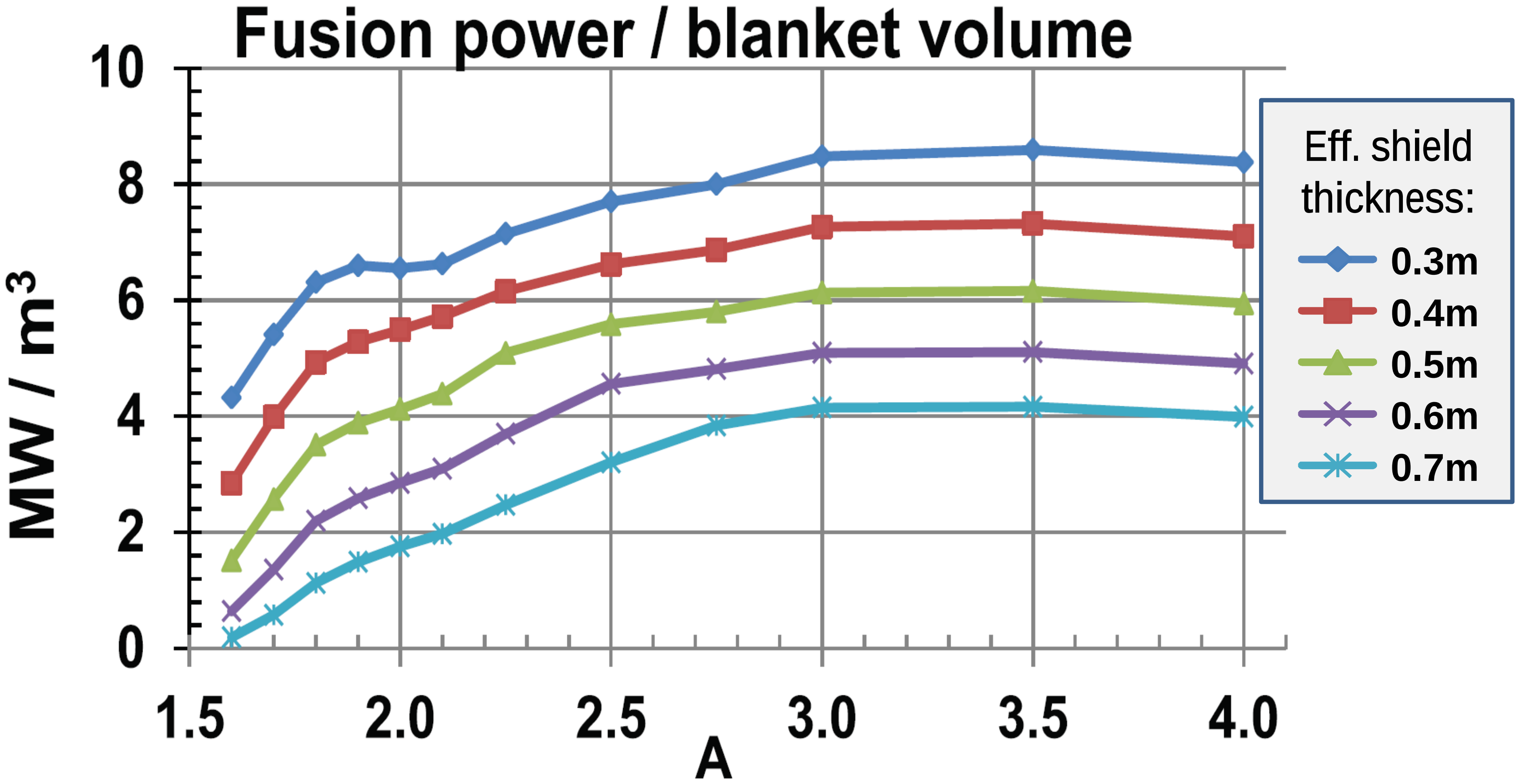
Outboard
TBR ~ 0.92

10cm IB shield sufficient for TBR > 1.04

Breeding blanket thickness model

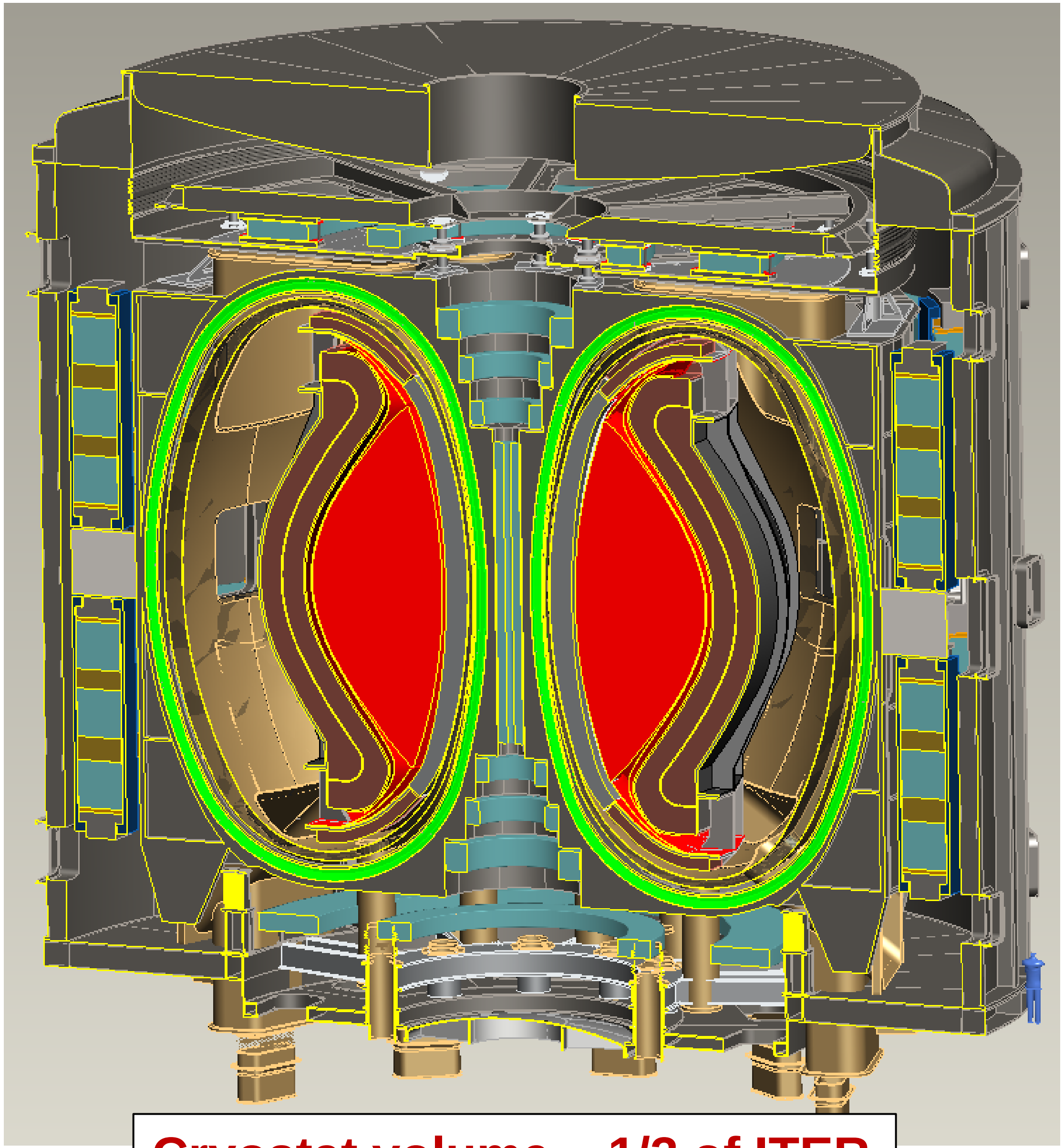


A ≥ 3 maximizes blanket utilization



J_{WP} = 70MA/m^2

A=2, R_0 = 3m HTS-TF FNSF / Pilot Plant



Cryostat volume ~ 1/3 of ITER

B_T = 4T, I_P = 12.5MA

κ = 2.5, δ = 0.55

β_N = 4.2, β_T = 9%

H_{98} = 1.8, $H_{Petty-08}$ = 1.3

f_{gw} = 0.80, f_{BS} = 0.76

Startup I_P (OH) ~ 2MA

J_{WP} = 70MA/m^2

B_{T-max} = 17.5T

No joints in TF

Vertical maintenance

P_{fusion} = 520 MW

P_{NBI} = 50 MW, E_{NBI} = 0.5MeV

Q_{DT} = 10.4

Q_{eng} = 1.35

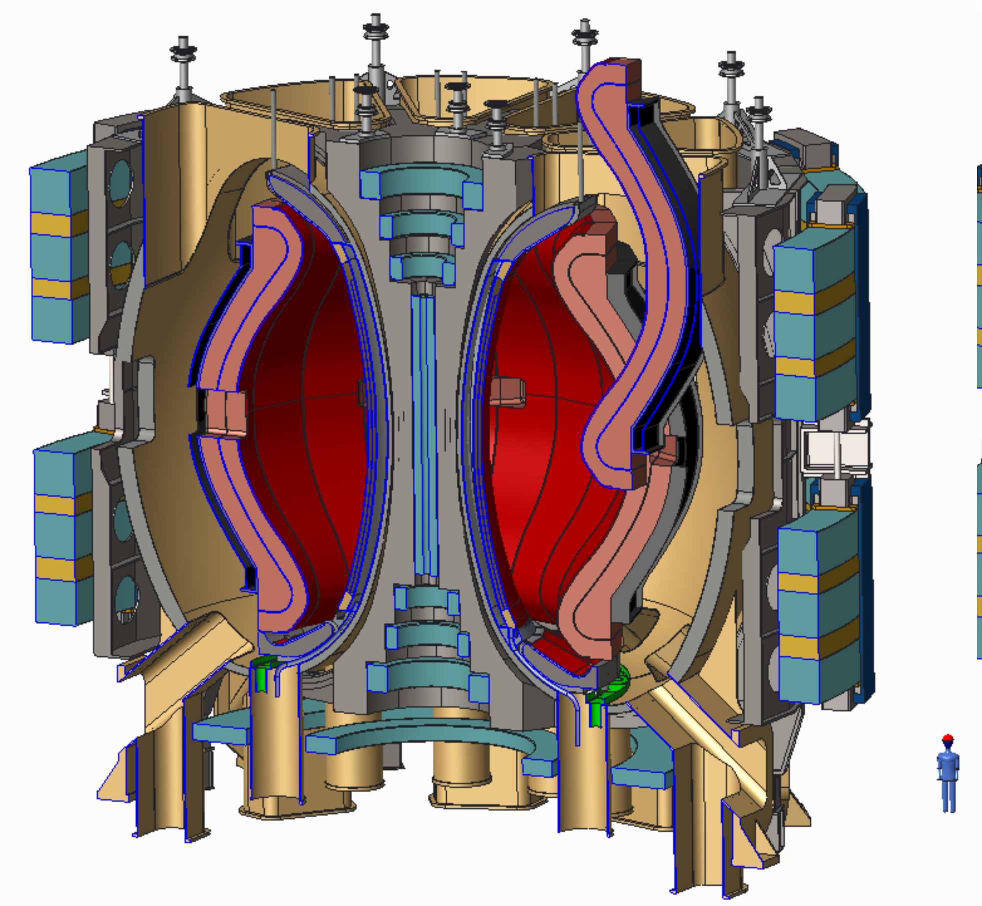
P_{net} = 73 MW

$\langle W_n \rangle$ = 1.3 MW/m^2

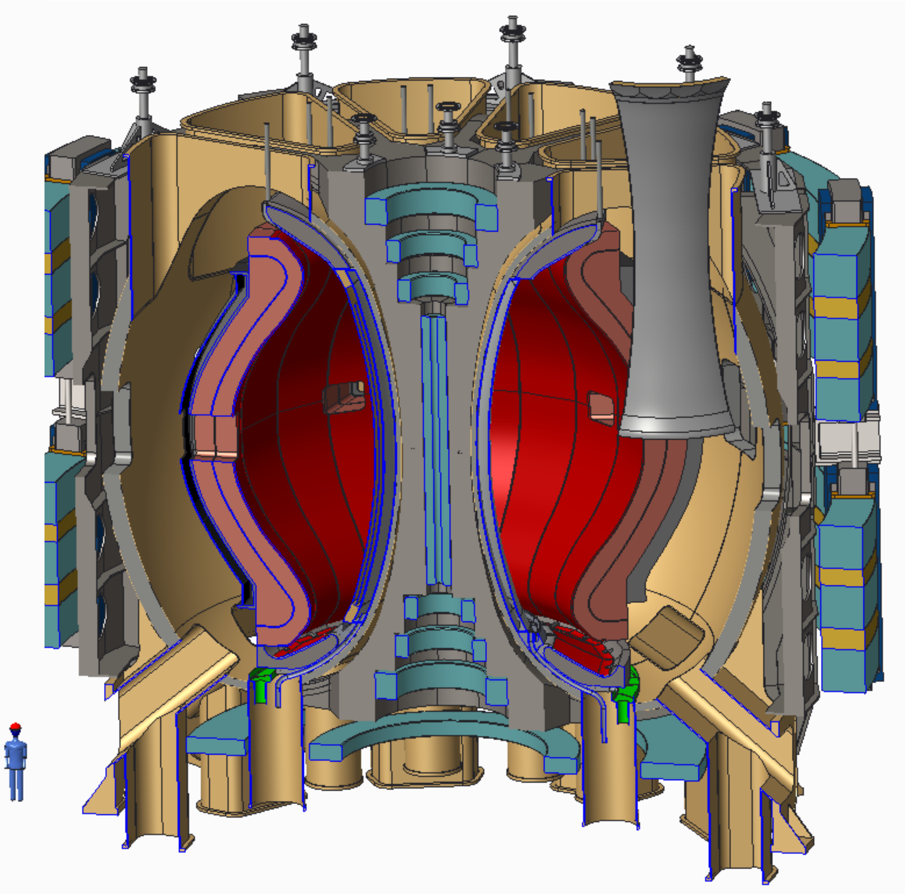
Peak n-flux = 2.4 MW/m^2

Peak n-fluence = 7 MWy/m^2

Inboard and outboard blanket vertical maintenance



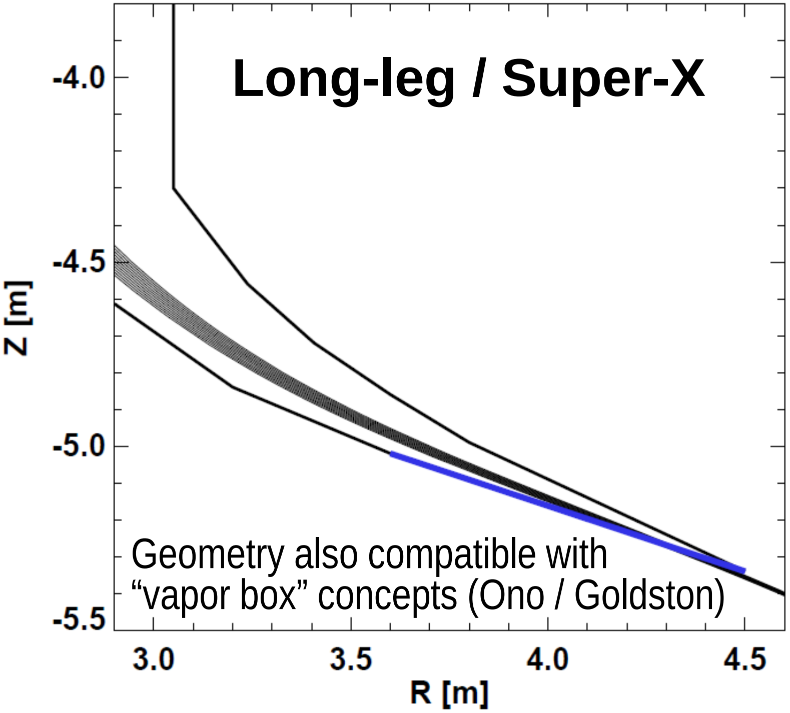
Outboard blanket removed



Inboard blanket removed once outboard blanket sectors removed – depending on the toroidal extent of the inboard blanket

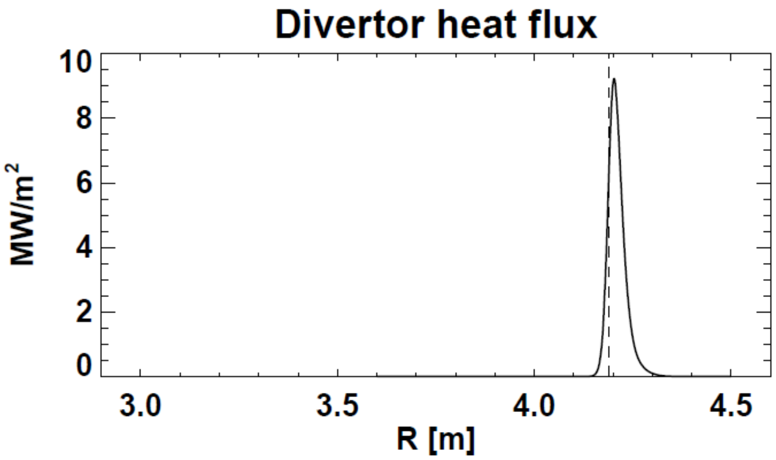
Long-leg divertor

P_{div} = 9MW/m^2, R_{strike} = 4.2m



Long-leg / Super-X

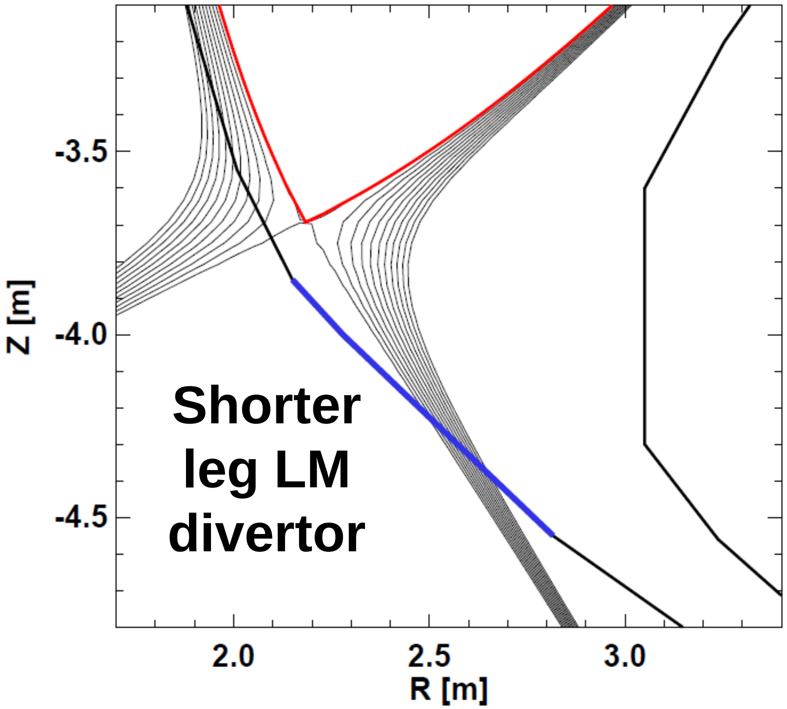
Geometry also compatible with “vapor box” concepts (Ono / Goldston)



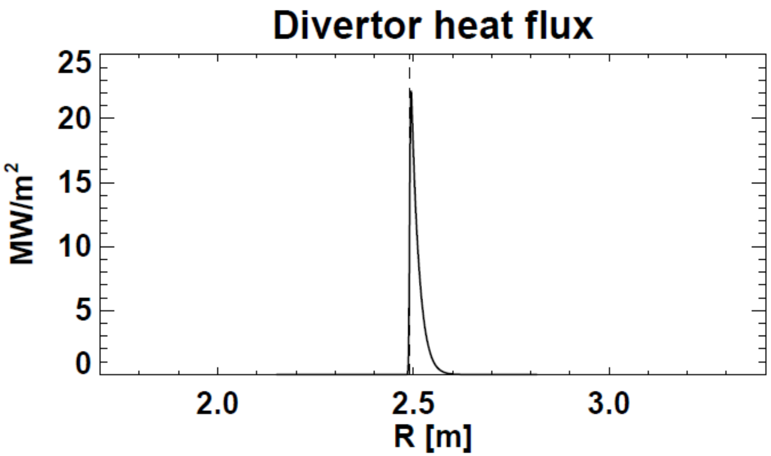
Divertor heat flux

Liquid metal (short-leg)

P_{div} = 21MW/m^2, R_{strike} = 2.5m



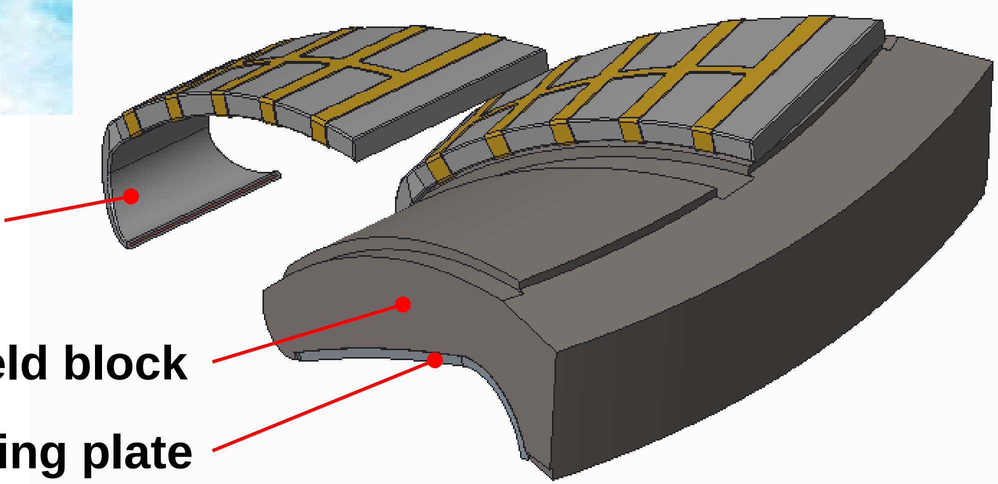
Shorter leg LM divertor



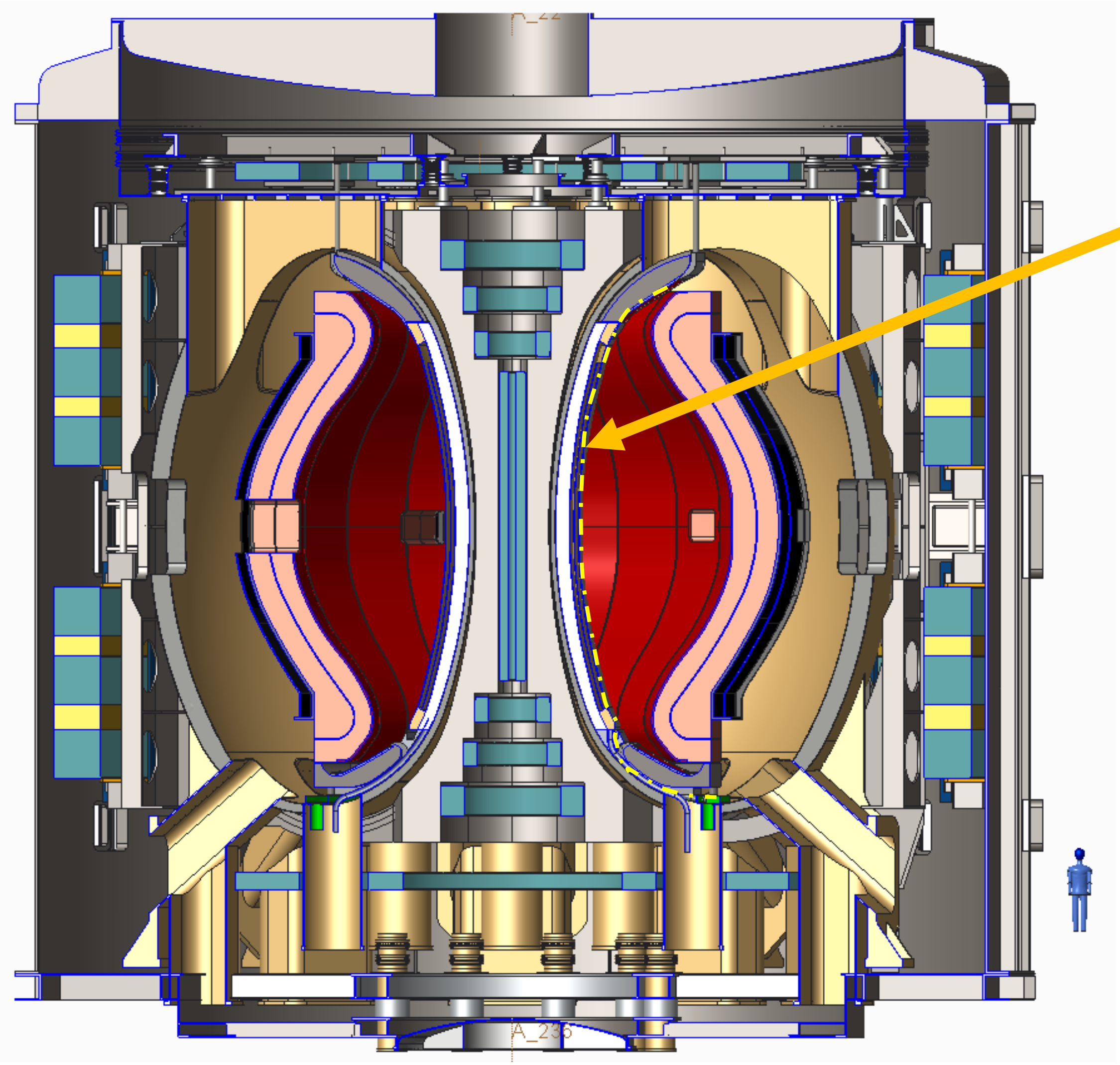
Divertor heat flux

Exploring liquid metal divertor design similar to flowing water curtain systems

LM injector system can be assembled in a single or double unit



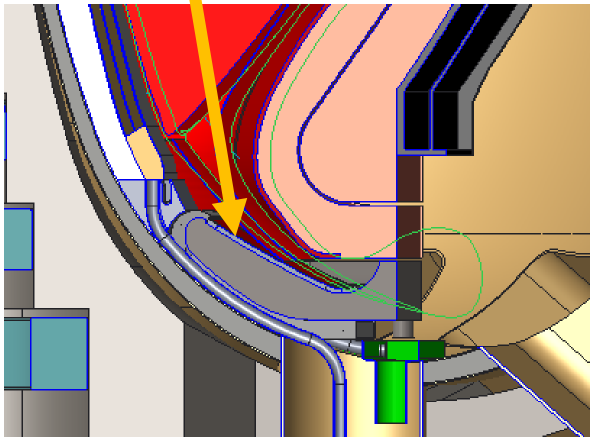
HTS ST-FNSF design with Li flow on divertor and inboard surfaces



Double null liquid metal divertor system

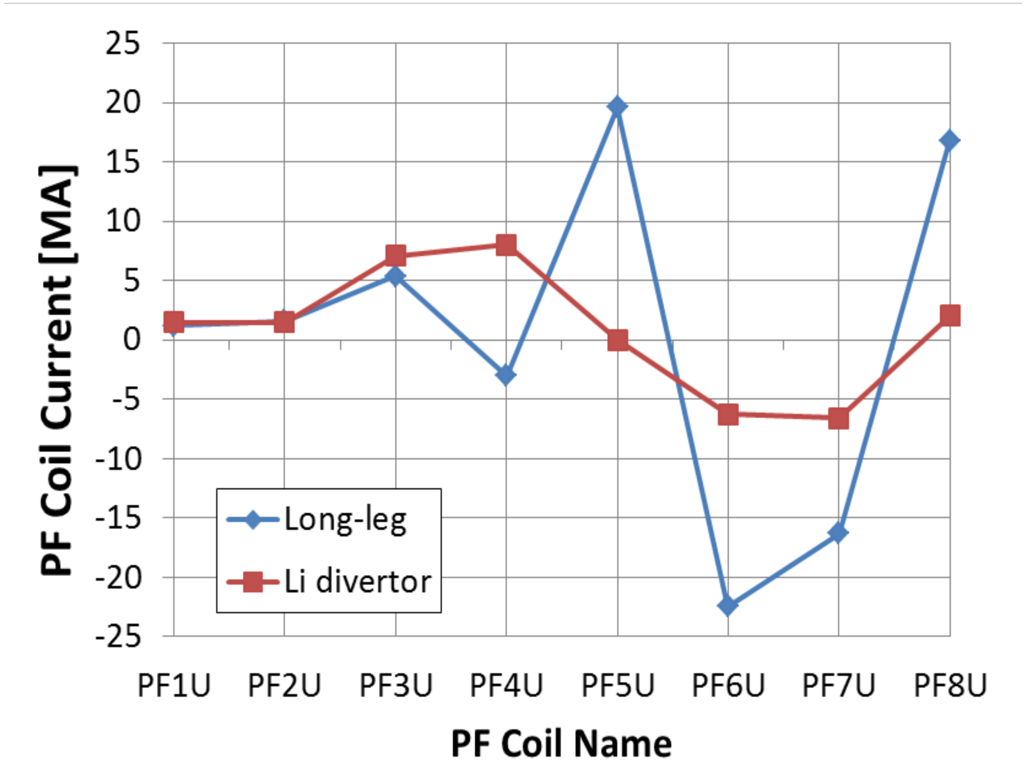
Li flows from upper divertor down the inboard wall, exiting just after the lower inboard divertor.

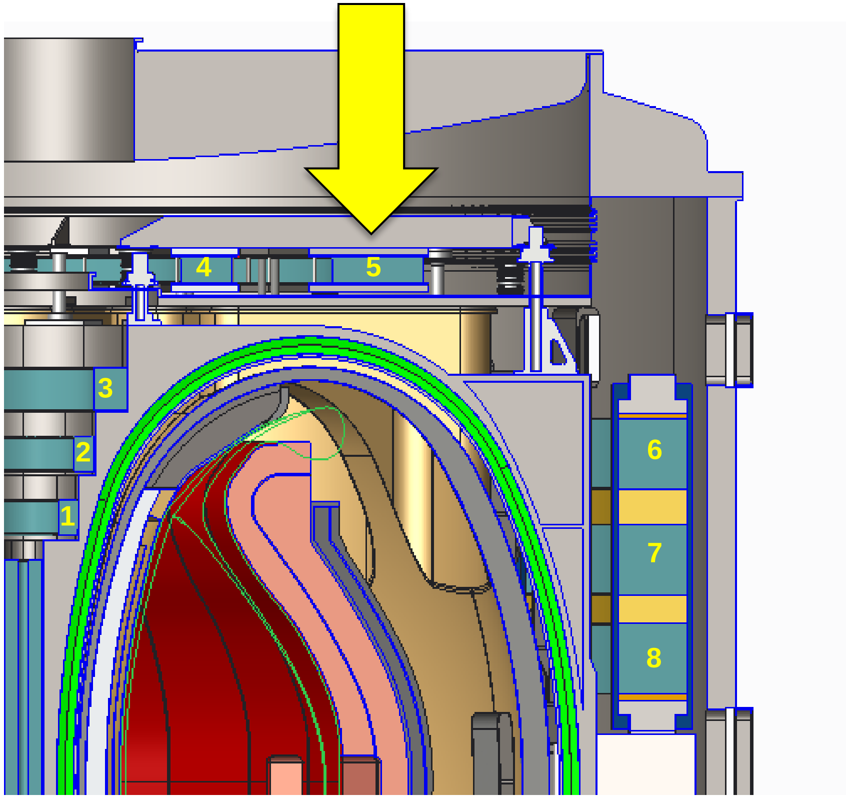
Separate Li cooling of lower divertor



Benefits of shorter-leg LM high-heat-flux divertor:

- Significantly reduce outboard PF coil current
  - Reduced PF size, force, structure
- Eliminate separate upper cryo-stat (for PF5U)





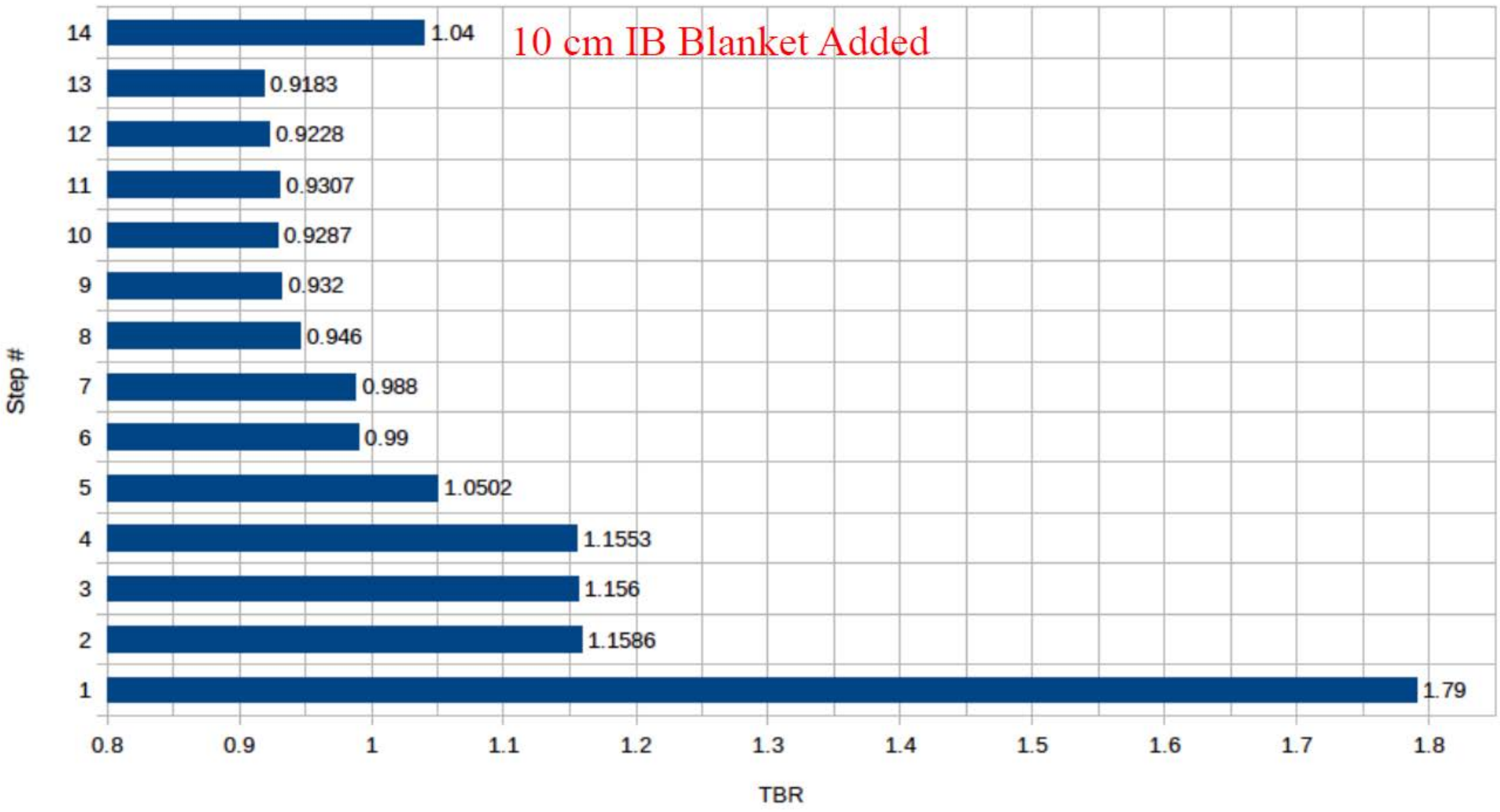
- **Li wall pumping could help increase H**

Summary

- Developed new self-consistent configurations for low-A FNSFs / Pilot Plants
  - Long-leg and/or LM divertor, T self-sufficient, only ex-vessel TF and PF coils, vertical maintenance
- Compact Pilot Plants achievable by combining improved stability of low-A + advanced magnets
  - Optimal A will be informed by results from NSTX-U and MAST-U and REBCO TF magnet development
- Liquid metal divertors for high heat flux could simplify cryostat, reduce coil currents/forces
  - Higher confinement from liquid Li also beneficial

Backup slides

Detailed breeding calculations completed for A=2



- Step 1- Infinite media of LiPb
- Step 2- LiPb confined to OB FW/blanket
- Step 3- Assembly gaps added
- Step 4- Homogeneous mixture of blanket in upper and lower ends of OB blanket
- Step 5- FW material added
- Step 6- Side, back, and front walls added
- Step 7- Cooling channels added
- Step 8- SiC FCI added
- Step 9- Stabilizing shells added
- Step 10- MTM only inserted (TBR relative to Step #9)
- Step 11- 4 TBMs only inserted (TBR relative to Step #9)
- Step 12- 4 NBIs only inserted (TBR relative to Step #9)
- Step 13- all MTM, 4 TBMs, and 4 NBIs inserted
- Step 14 – include inboard breeding blanket

Comparison of low-A FNSF / Pilot Plants

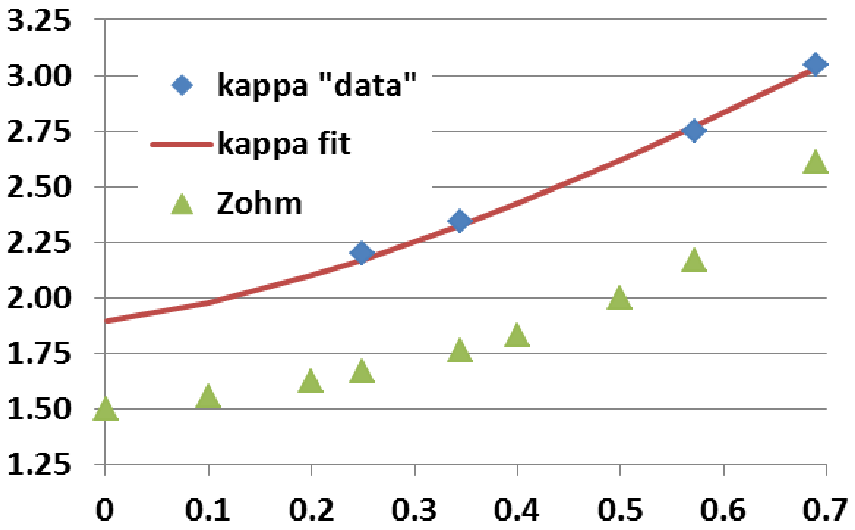
TF coil type	R [m]	A	Q_{eng}	Q_{DT}	TBR	Surf-avg n-fluence [MWy/m^2]	P_h / S [MW/m^2]	H_{98}	H_{Petty}	H_{ST}	κ_X	β_N	β_T [%]	f_{BS}	I_P [MA]	B_T [T]	P_{fus} [MW]
Copper	1	1.7	0.1	1.0	≤ 0.9	6	1.6	1.25	1.25	0.70	2.75	5	20	0.82	7.3	3.0	60
	1.7	1.7	0.15	2.0	1.0	≥ 6	0.9	1.25	1.1	0.72	2.75	4	16	0.76	11	3.0	160
REBCO	1.8	2	1	7.3	0	0.04	0.5	2.3	2.1	0.64	2.30	4	7.1	0.84	7.4	5	160
	3	2	1.3	10	1.0	4 - 6	0.5	1.8	1.3	0.69	2.50	4	8.7	0.76	13	4.0	510

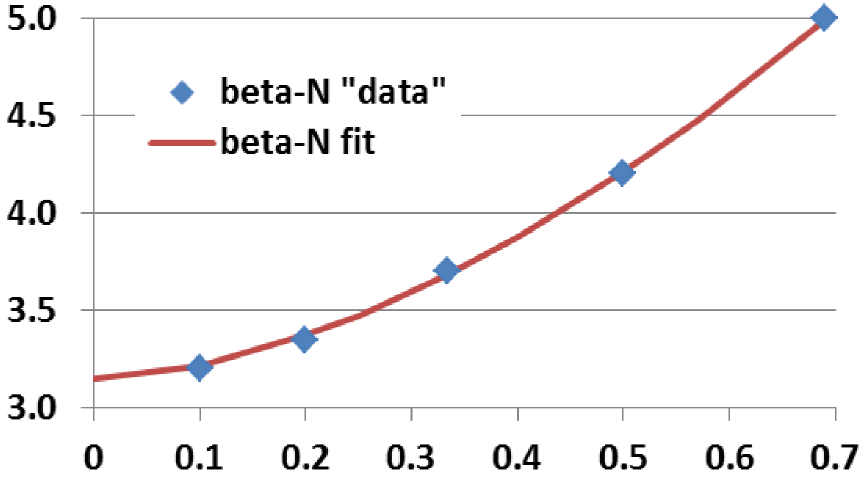
Plasma constraints

- Fix plasma major radius at R_0 = 3m
  - – Chosen to be large enough to allow space for HTS neutron shield and access $Q_{eng} > 1$ for range of A
- Inboard plasma / FW gap = 4cm
- Use ε-dependent $\kappa(\varepsilon)$, β_N (ε) (see next slide)
- Greenwald fraction = 0.8
- q* not constrained
  - – q* is better ε-invariant than q_{95} for current limit
  - – Want to operate with q* > 3 to reduce disruptivity
- 0.5 MeV NNBI for heating/CD – fixed P_{NBI} = 50MW
- H_{98y2} adjusted to achieve full non-inductive CD

Aspect ratio dependence of limits: $\kappa(\varepsilon)$, $\beta_N(\varepsilon)$

Pilot study uses $0.95 \times \kappa$ value shown here:





$\varepsilon = A^{-1}$

- NSTX data at low-A
  - Also NSTX-U/ST-FNSF modelling
- DIII-D, ARIES-AT for higher A
  - $\kappa \rightarrow 1.9$ for $A \rightarrow \infty$
- Profile-optimized **no-wall** stability limit at $f_{BS} \approx 50\%$
  - Menard PoP 2004
- $\beta_N \rightarrow 3.1$ for $A \rightarrow \infty$

$$\beta_T \propto A^{-1/2}\,(1+\kappa^2)\,\beta_N^2\,/\,f_{BS}$$

➡ $P_{fus} \propto \varepsilon\,[\kappa(\varepsilon)\,\beta_N(\varepsilon)\,B_T(\varepsilon)]^4$

Engineering constraints

- Magnet constraints
  - Maximum stress in TF magnet structure = 0.66 GPa
  - HTS tape/cable strain limit 0.3% (equivalent to 0.4 GPa)
  - Winding pack current density (CORC 2015) 70 MA/m^2
  - OH at small R → higher solenoid flux swing for higher A
- Shielding / blankets
  - HTS fluence limit: 3.5-5 x 10^{22} n/m^2
  - Shield:10x n-shielding factor per 15-16cm WC for HTS TF
  - Include inboard & outboard breeder thickness for TBR ~ 1
    - "Effective shield thickness" includes shield + DCLL blanket
- Electrical system efficiency assumptions:
  - 30% wall plug efficiency for H&CD - typical of NNBI
  - ≥ 45% thermal conversion efficiency - typical of DCLL
    - Also include pumping, controls, other sub-systems
    - See Pilot Plant NF 2011 paper for more details

Simplified TF magnet design equations

$$V_1 + V_2 = \frac{1}{2} B_0 R_0 I_{\mathrm{coil}} \ln\left(\frac{r_2}{r_1}\right) \qquad (25)$$

$$r_1 V_1 + r_2 V_2 = \frac{1}{2} B_0 R_0 I_{\mathrm{coil}} (r_2 - r_1) \qquad (26)$$

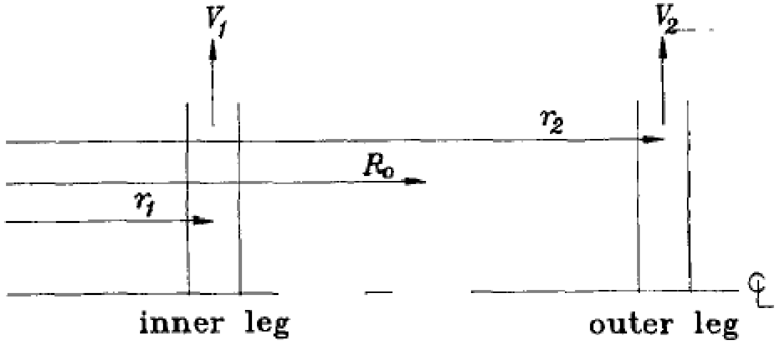


Fig. 7. Geometry for force and moment balances.

$$V_1 = \frac{F}{(r_2 - r_1)}(r_1 + r_2(k - 1))$$

$$V_2 = \frac{F}{(r_2 - r_1)}(r_2 - r_1(k + 1))$$

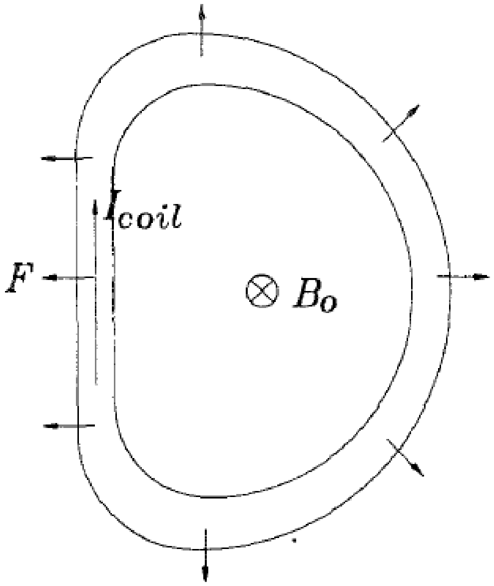


Fig. 5. Lorentz forces are normal to the conductor in the poloidal plane.

$$F \equiv \frac{1}{2} B_0 R_0 I_{\mathrm{coil}}$$

$$I_{\mathrm{coil}} = \frac{5 \times 10^6 B_0 R_0}{N_{\mathrm{coils}}}$$

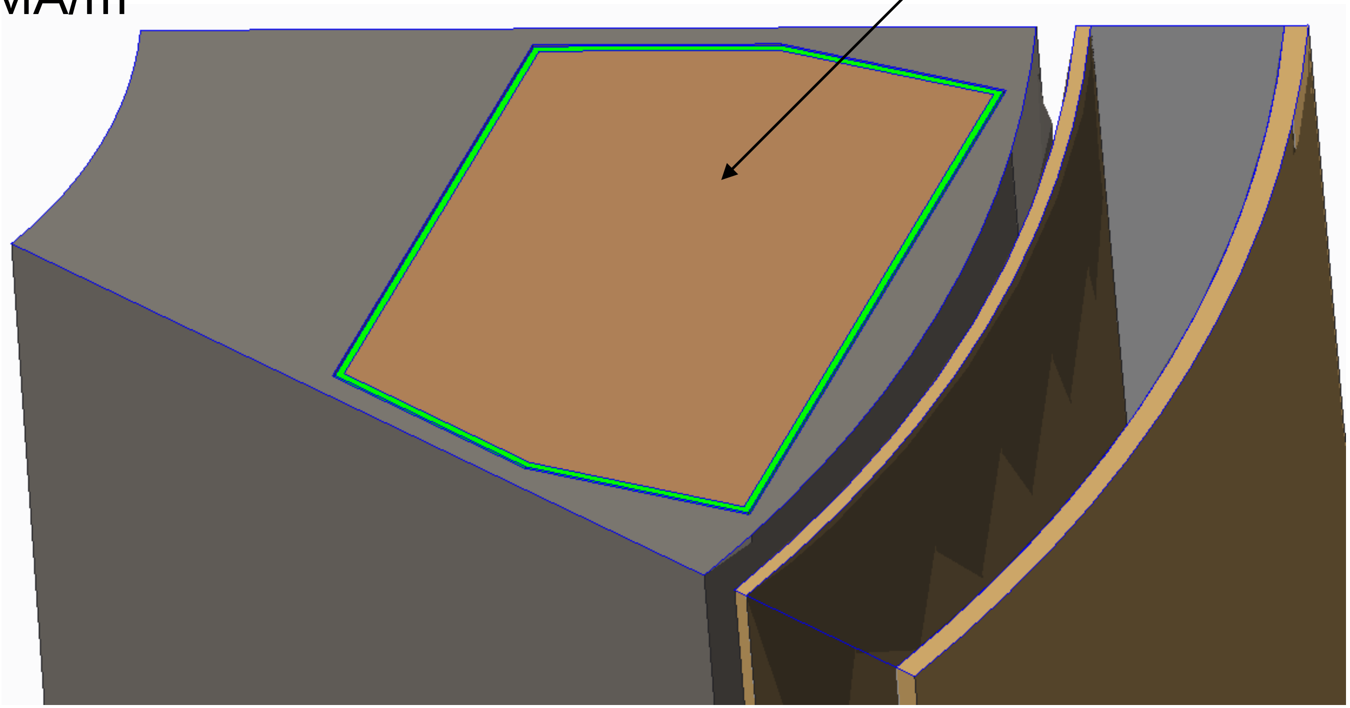
$$k \equiv \ln\left(\frac{r_2}{r_1}\right)$$

From J. Schwartz, Journal of Fusion Energy, Vol. 11, No. 1, 1992

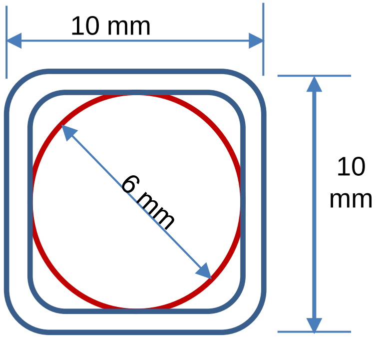
A=2, R_0 = 3m device TF inboard leg showing allocated space for case and winding

Current per coil: 6 MA
Winding Cd: 35.9 MA/m^2

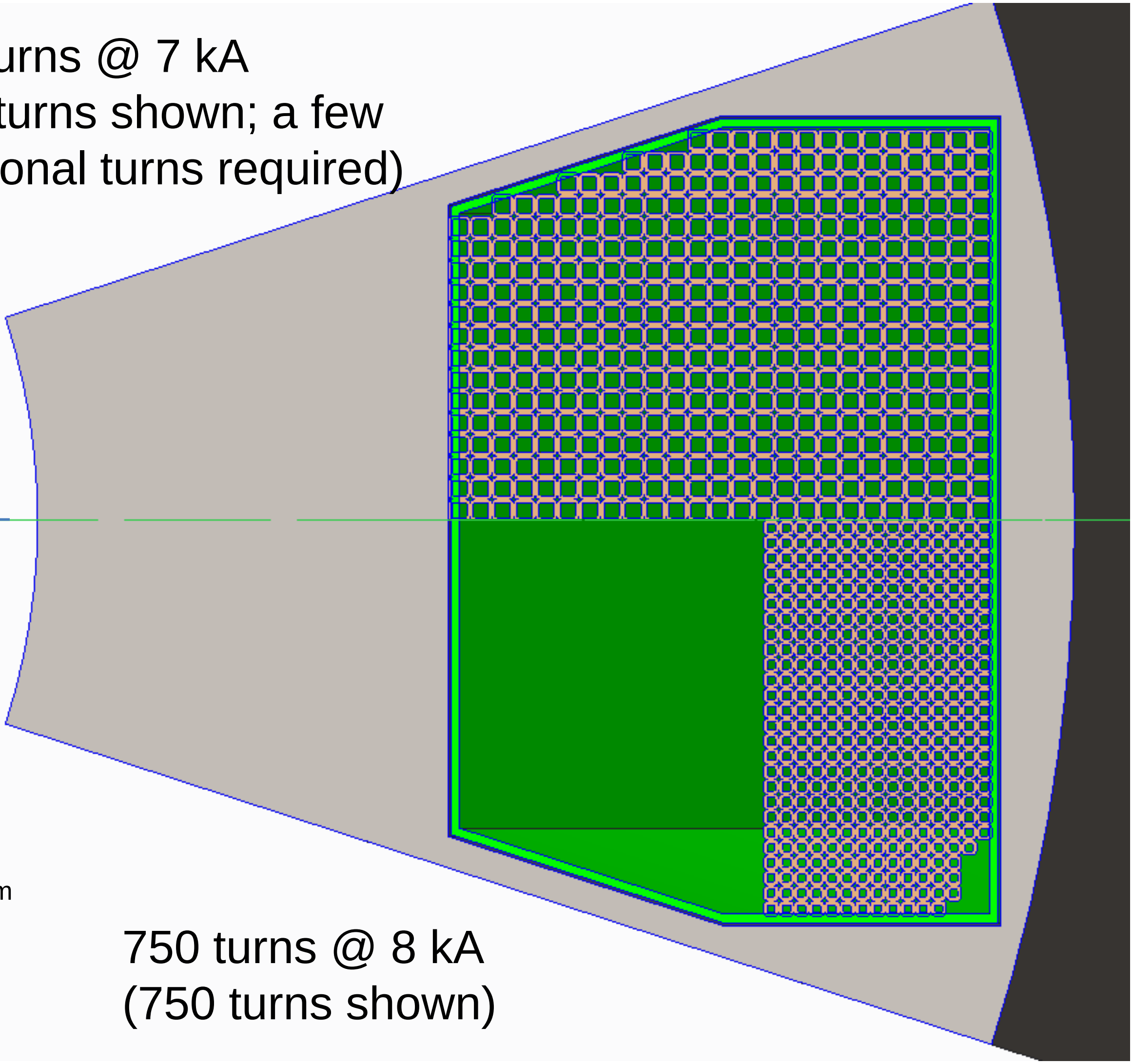
$0.167\ m^2$ winding area



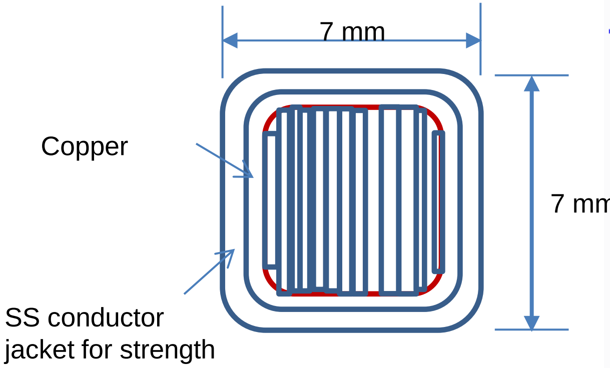
CORC Conductor – Achieved now



857 turns @ 7 kA
(848 turns shown; a few additional turns required)



Base Conductor – Helium Gas Cooled



750 turns @ 8 kA
(750 turns shown)

HTS performance vs. field and fast neutron fluence

Supercond. Sci. Technol. **28** (2015) 014005 R Prokopec *et al*

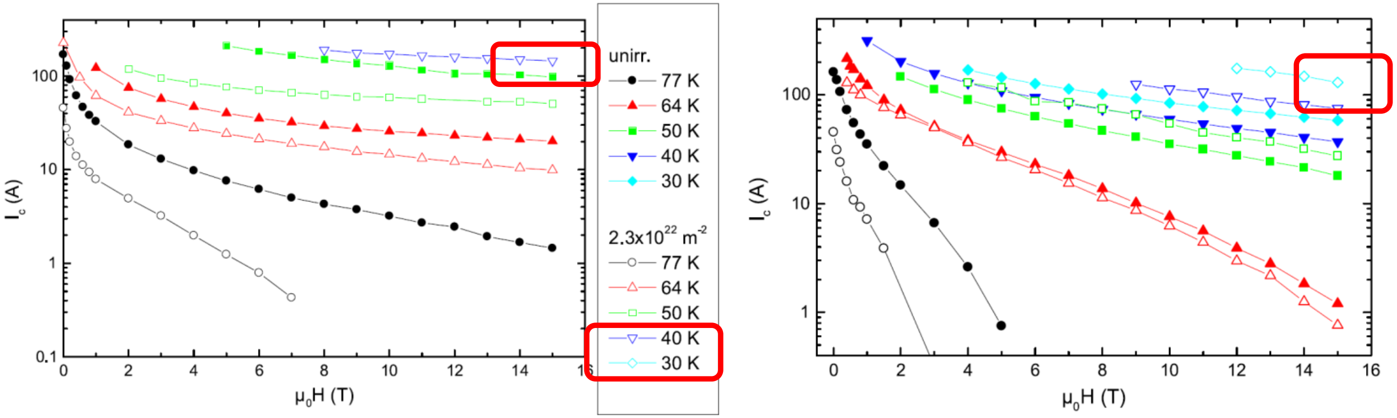


Figure 6. Critical currents (ASC-40) in magnetic fields applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) before and after irradiation to a fast neutron fluence of $2.3 \cdot 10^{22}$ m^{-2}.

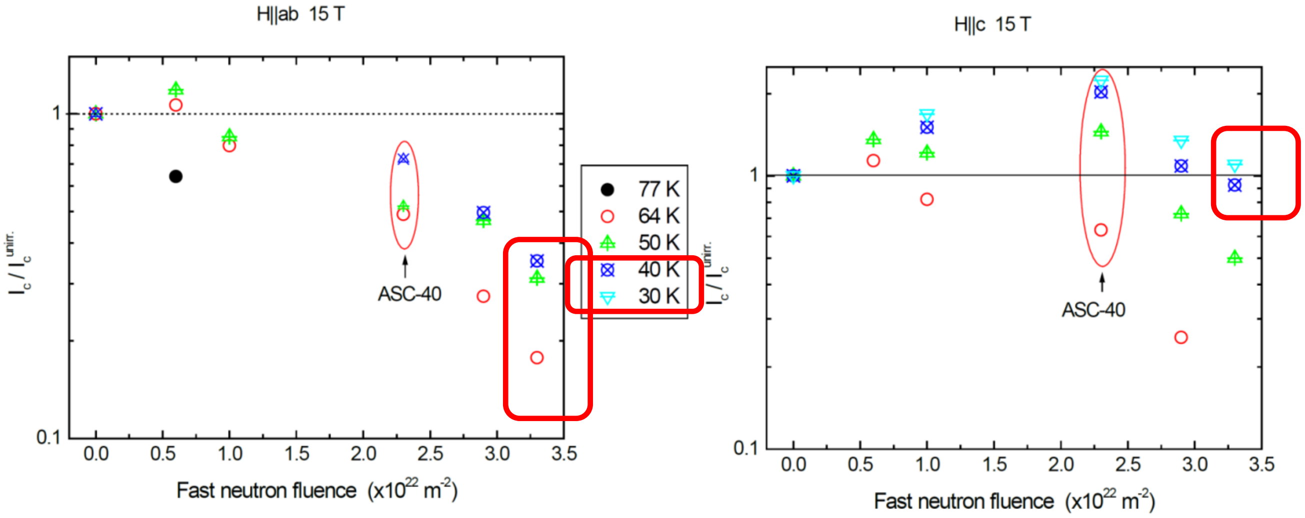
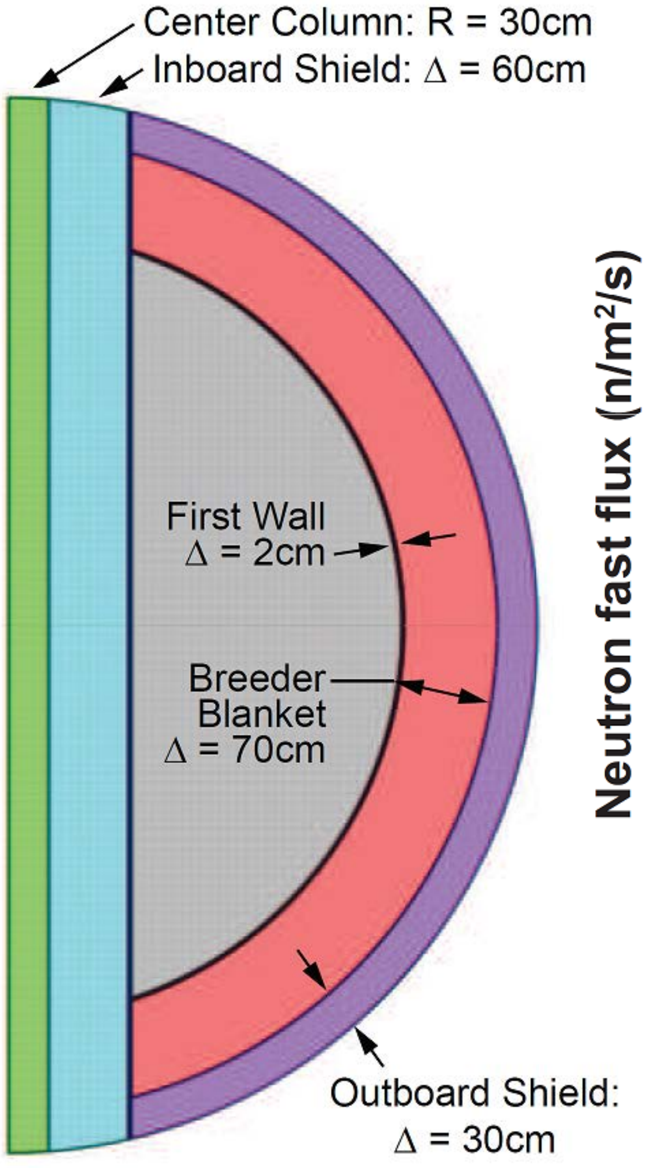
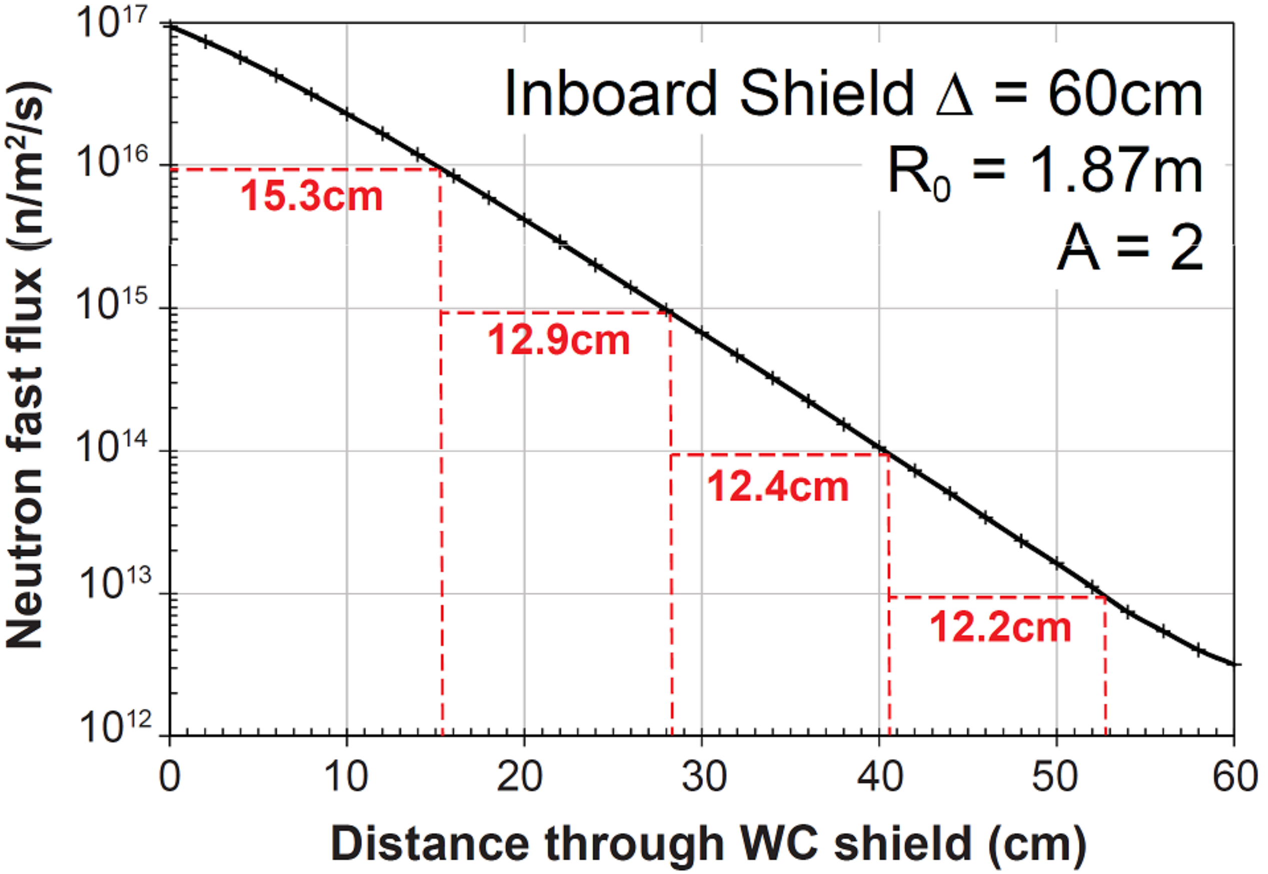


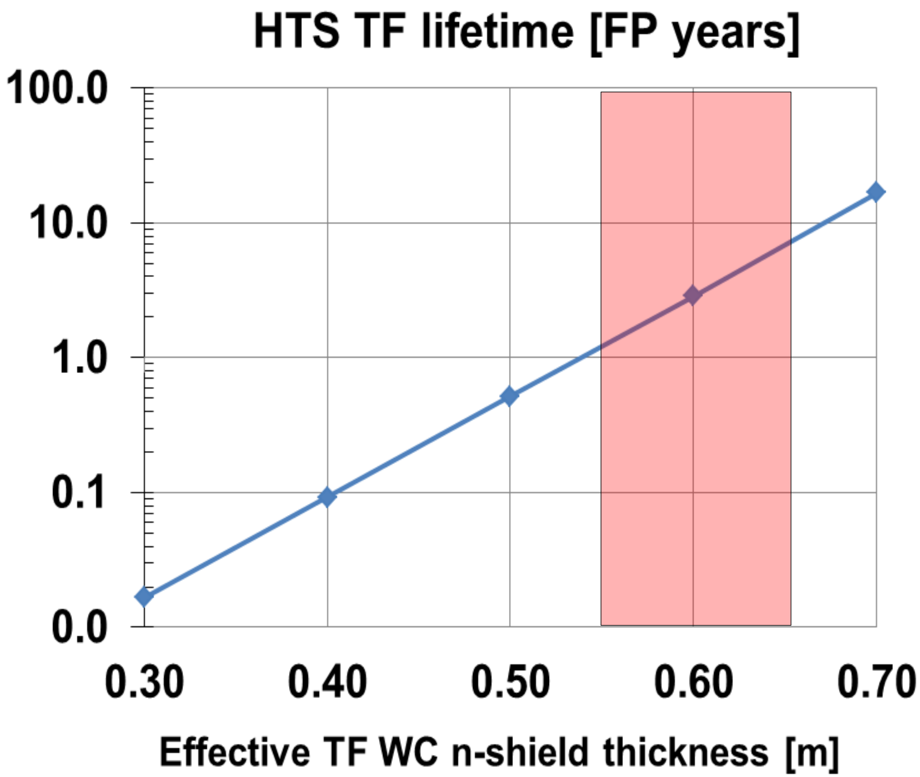
Figure 8. Normalized critical currents in a magnetic field of 15 T applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) as a function of neutron fluence.

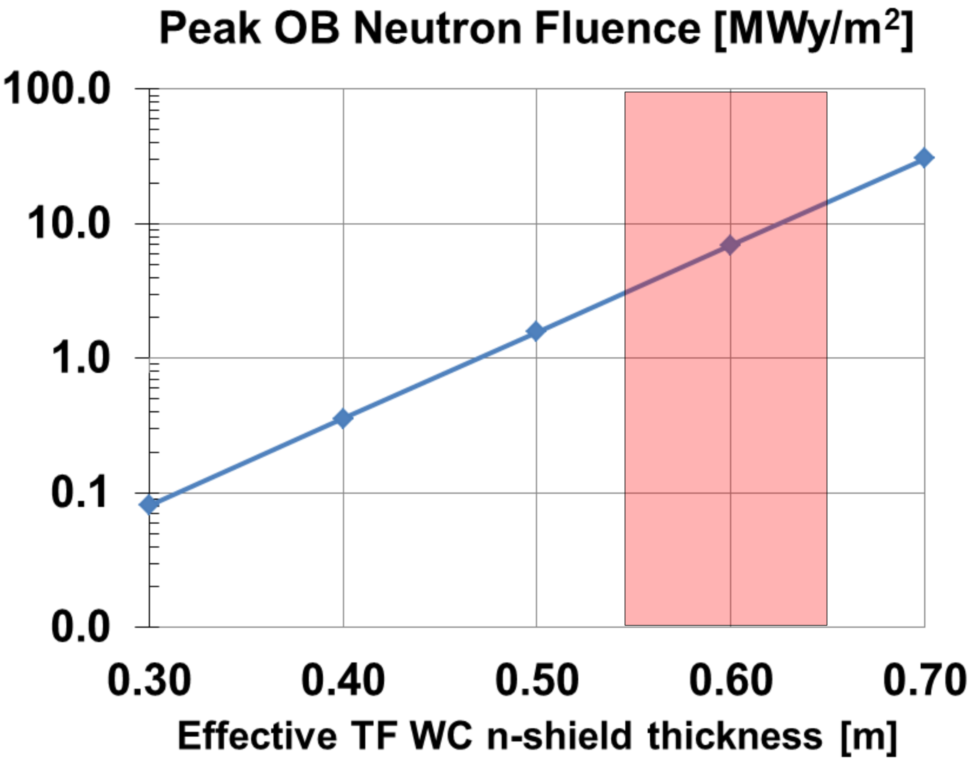
Neutronics analysis for HTS TF shielding





HTS TF lifetime is very strong function of inboard shielding thickness

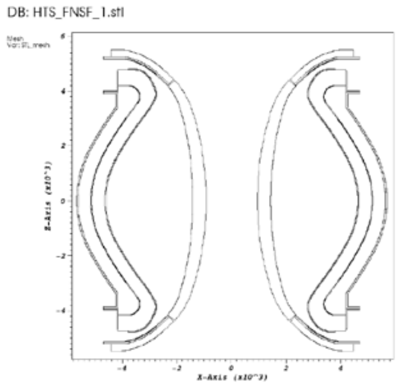

HTS TF lifetime [FP years]

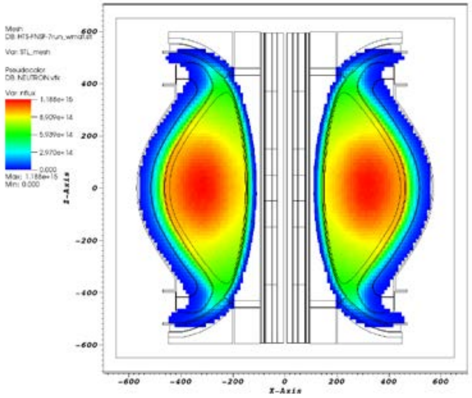

Peak OB Neutron Fluence [MWy/m²]

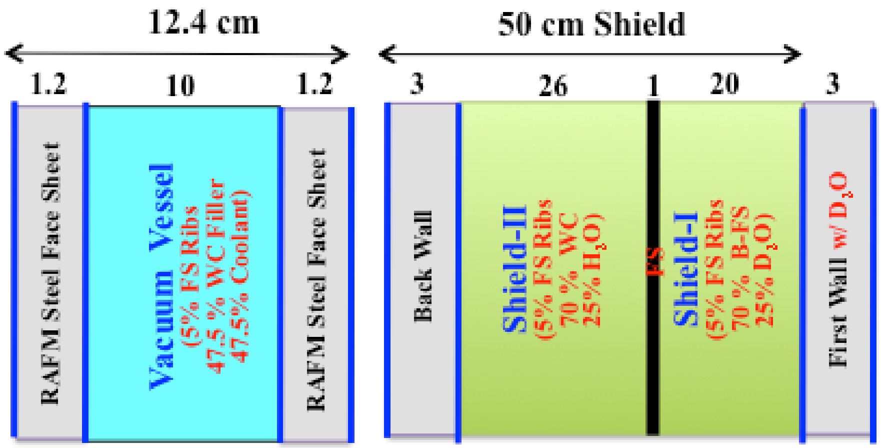
Inboard shield + blanket equivalent to 60**cm WC →**
3FPY → 6-7MWy/m² → fulfill FNSF requirement

3-D Neutronics Model

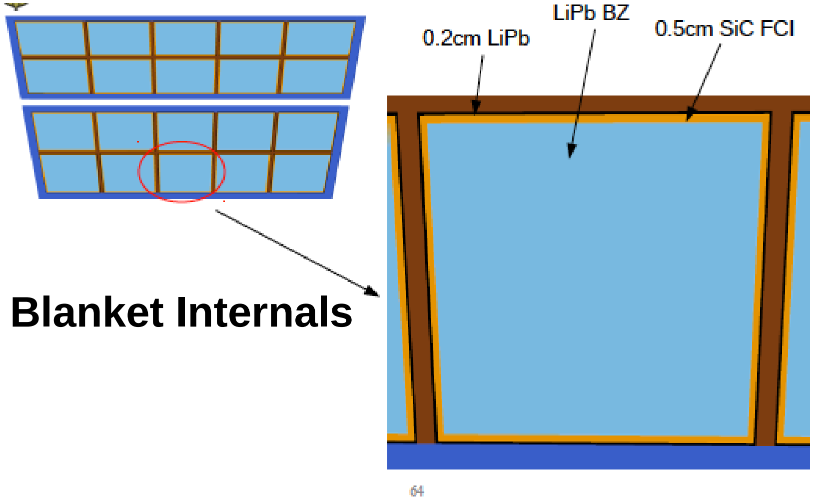
- Simple CAD Model outlining components (from T. Brown):
  - 10 cm IB modules
  - 20 cm OB modules
- R-Z neutron source
- IB shield and VV optimization and composition by UW.
- OB blanket internals and composition by UW
- Detailed CAD of blanket by UW
- UW DAGMC code couples CAD to 3D MCNP neutronics code.



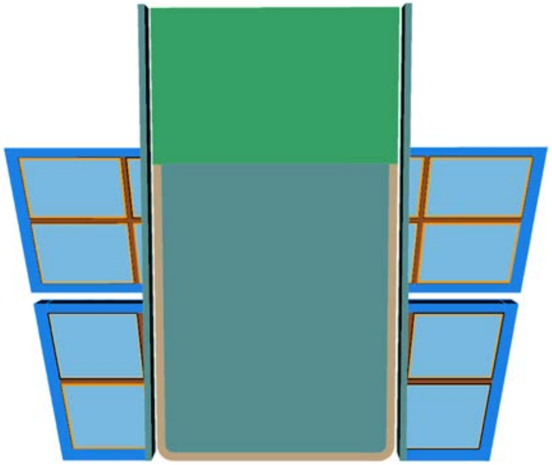




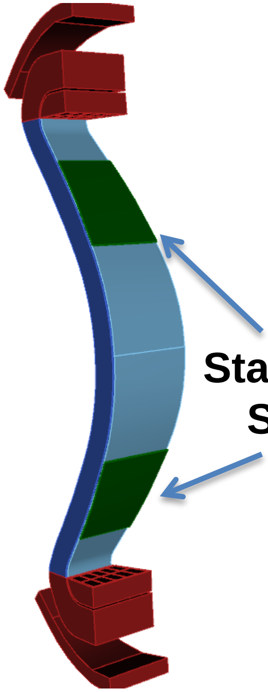
CAD Geometry of OB Blanket with Ports



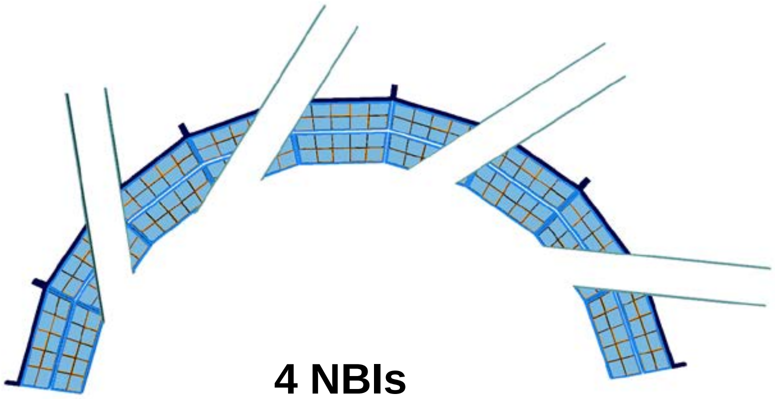
Detailed Blanket Internals



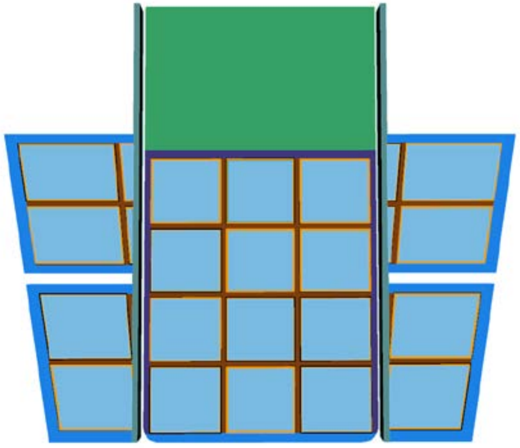
MTM



Stabilizing Shells

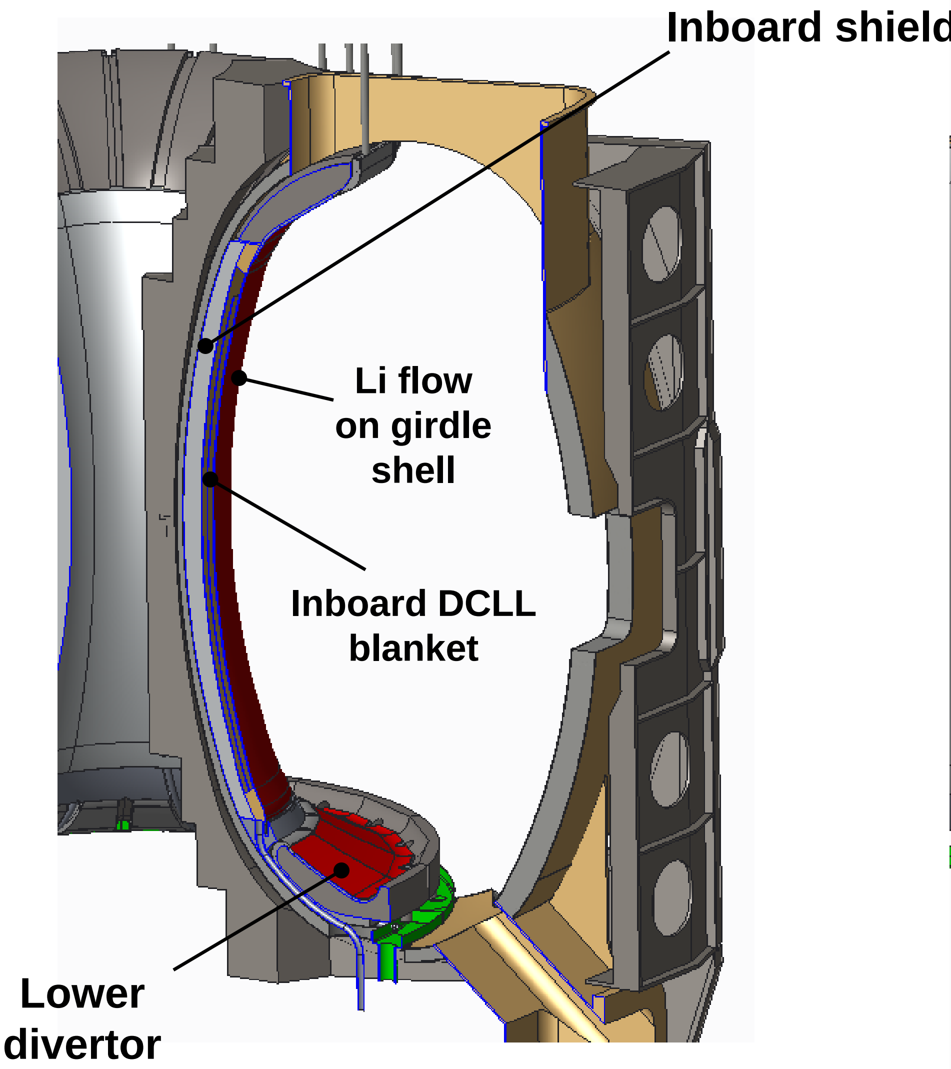


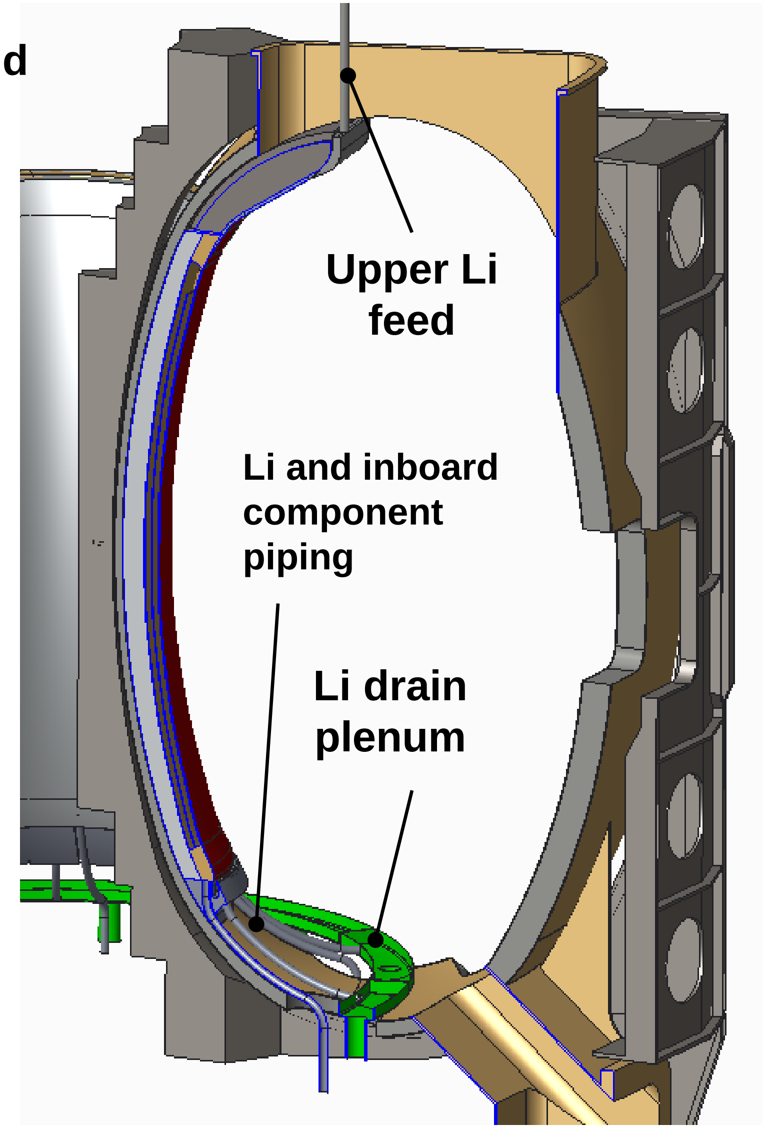
4 NBIs



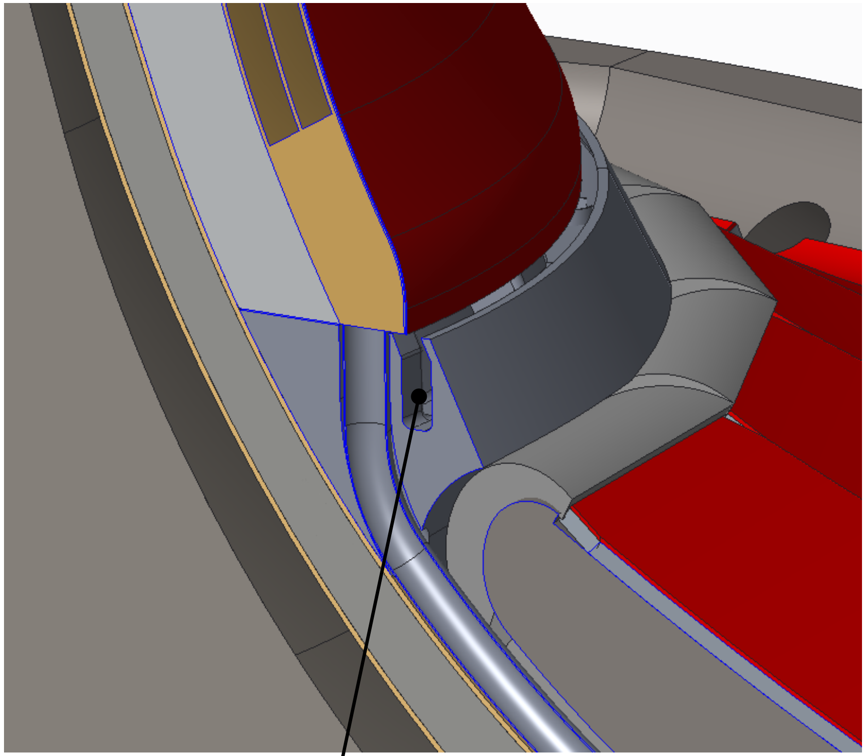
TBM

Local details of Li divertor / inboard FW



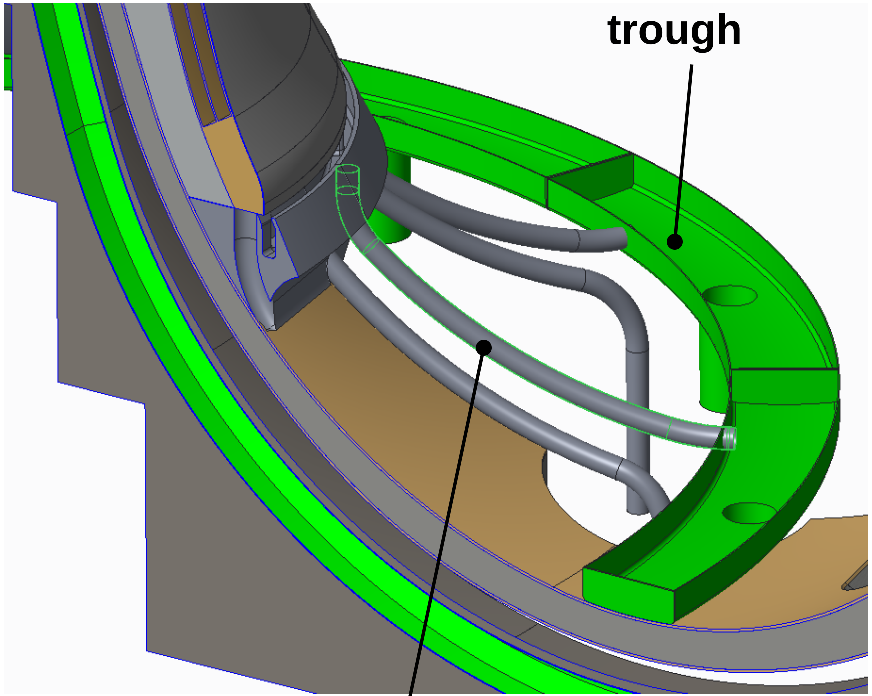


Lower Li containment system



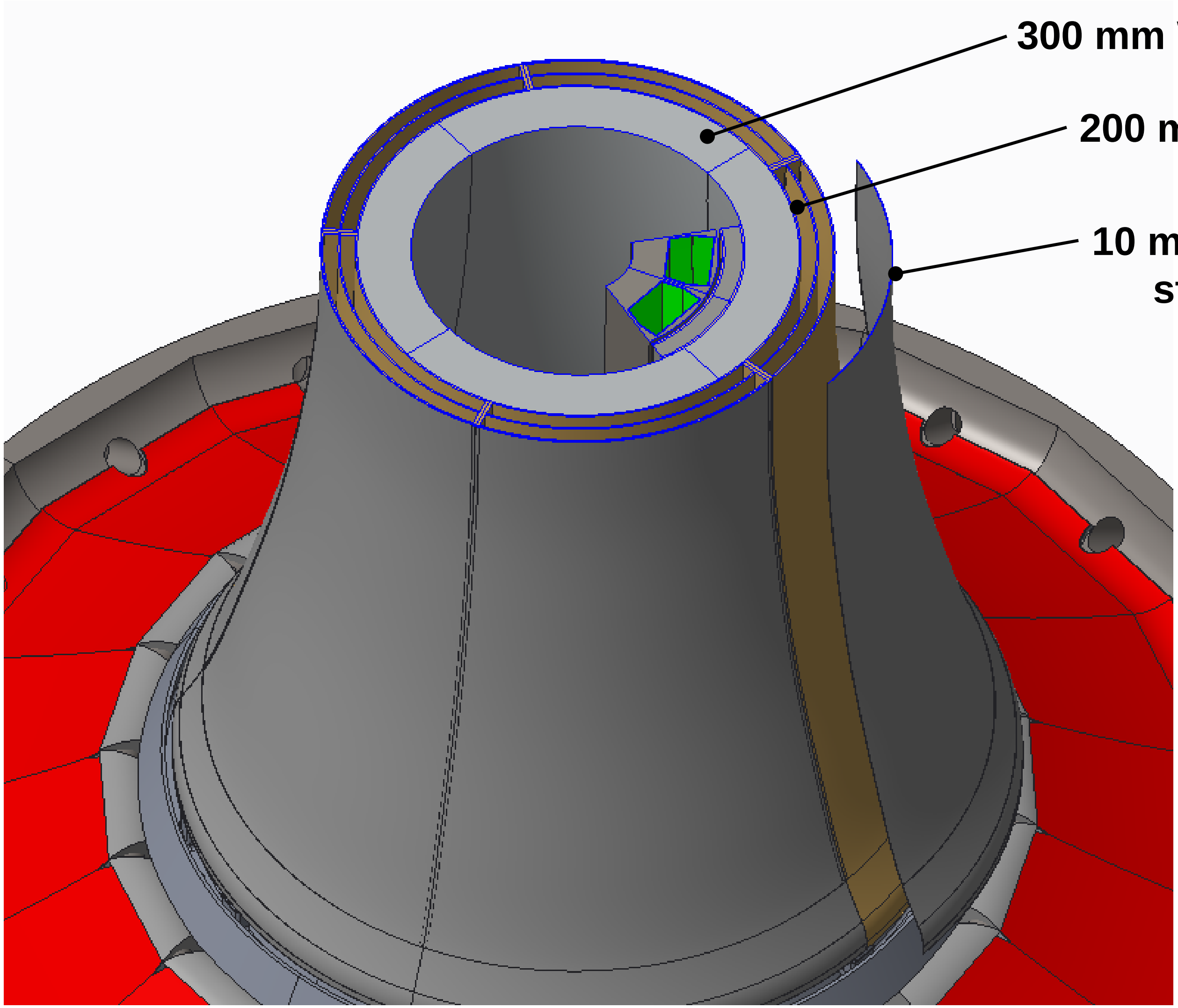
Li flows over inboard surface to a continuous trough that feeds ten Li drain lines.

Base Li return trough



One of ten 100 mm ID Li inboard drain lines

Inboard FW / DCLL / shield components



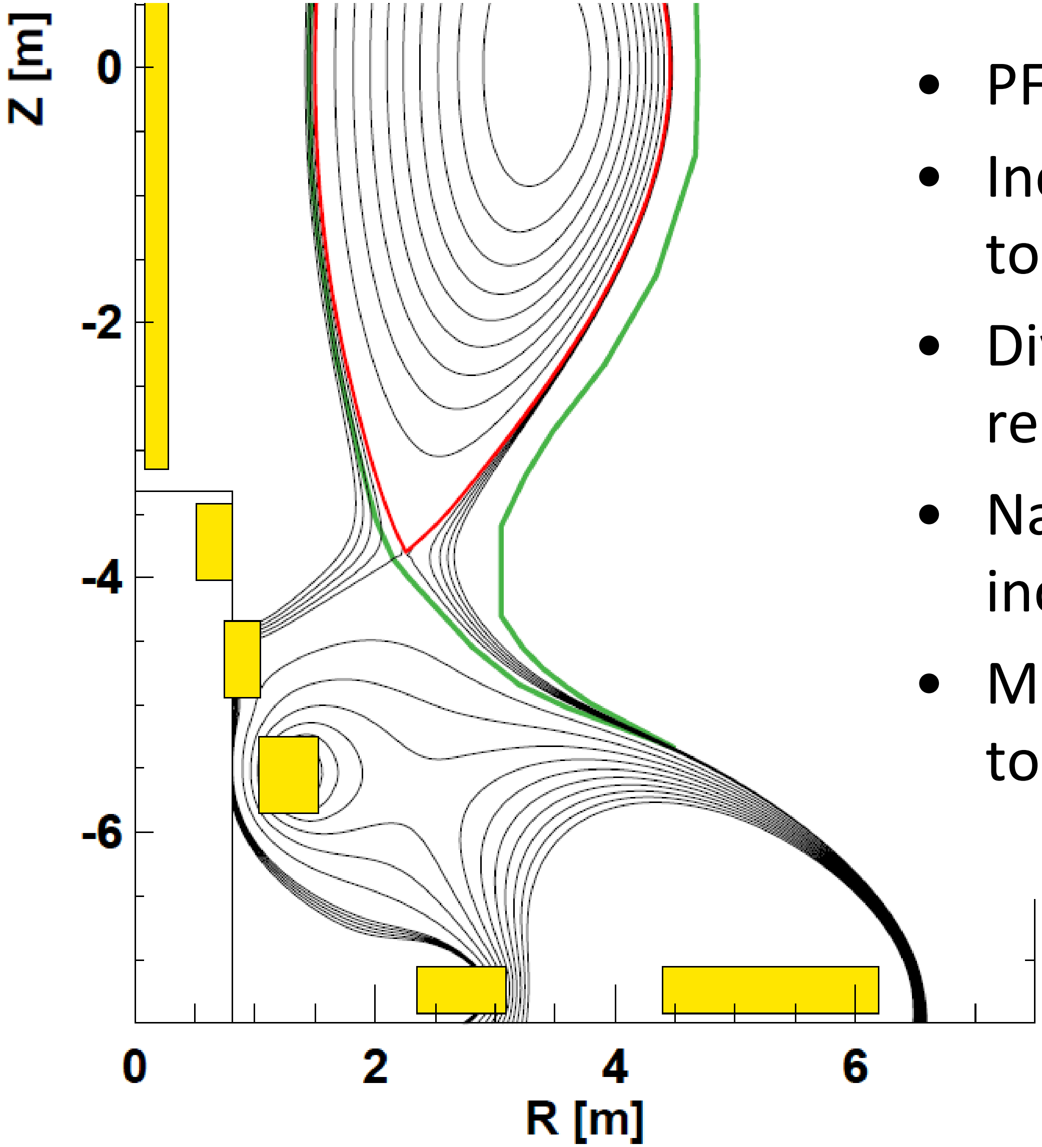
300 mm WC inboard shield

200 mm DCLL blanket system

10 mm Li surface metal structure (girdle)

As a first pass design it is assumed that these components will be electrically continuous …analysis will follow.

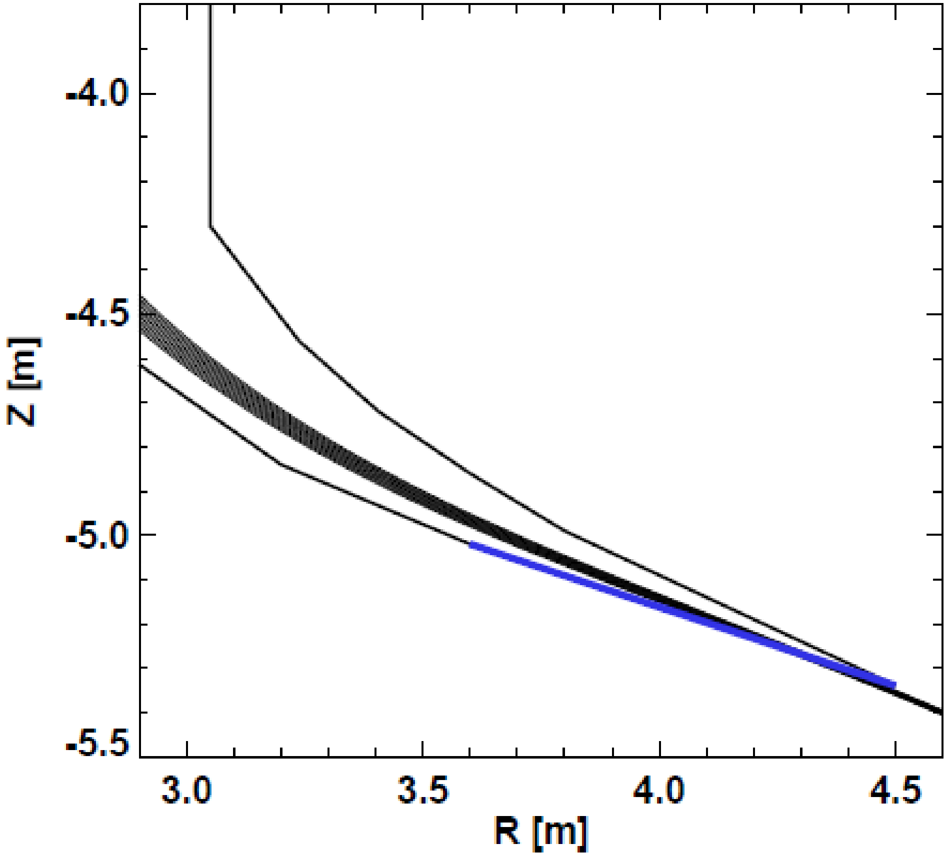
Assessing long-leg / deep-V slot divertor



- PF coils outside TF
- Increase strike-point radius ~2× to reduce $q_{||}$ and peak heat flux
- Divertor PFCs in region of reduced neutron flux
- Narrow divertor aperture for increased TBR
- More space for breeding at top/bottom of device

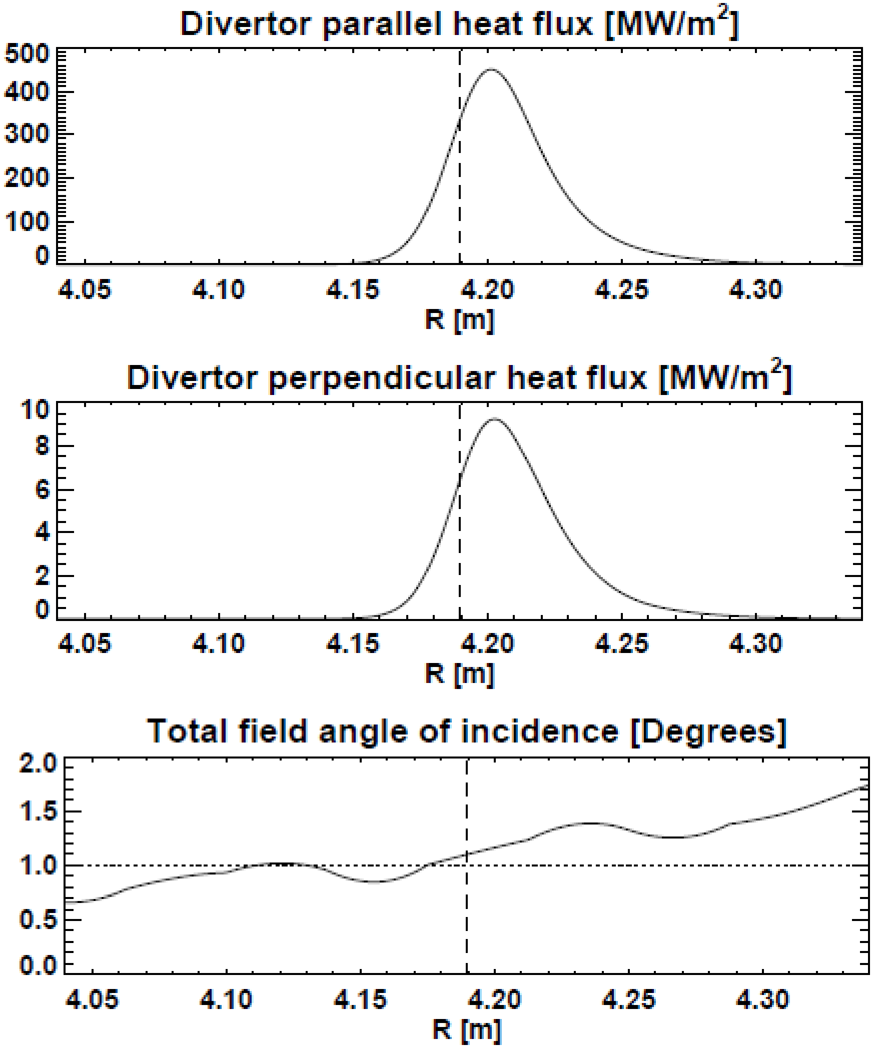
Long-leg / Super-X aids heat flux reduction

A=2 HTS TF FNSF/Pilot



λ_q ~ 1mm, **assume S ≈** λ_q (closed divertor)

(T. Eich NF 2013)



(Partial) detachment likely reduces peak $q_\perp$ by further factor of 2-4