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

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PPPL PRINCETON PLASMA PHYSICS LABORATORY

Multi-institutional 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 $\geq$ 1
4. Electrical self-sufficiency
   - $Q_{eng} = P_{electric} / P_{consumed} \sim 1$
5. Large net electricity generation
   - $Q_{eng} >> 1$, $P_{electric}$ = 0.5-1 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?

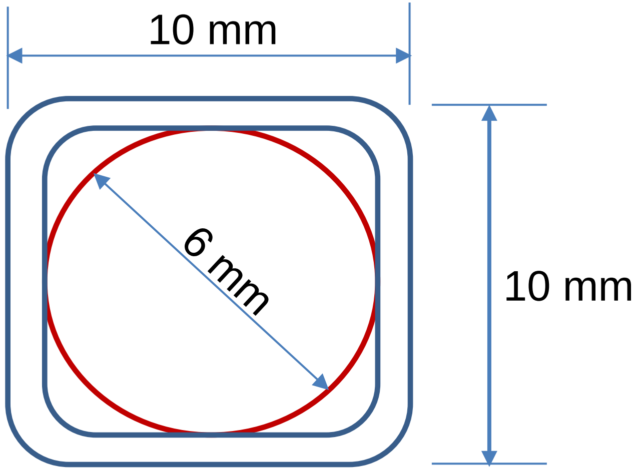
- $P_{fus} / V \sim \varepsilon(\beta_N \kappa B_T)^4$ at fixed bootstrap fraction
- β_N and κ increase at lower aspect ratio
- But B_T decreases at lower A – depends strongly on:
  - Thickness of inboard shielding and breeding blanket
  - HTS allowable field and current density

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 $\times$ 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

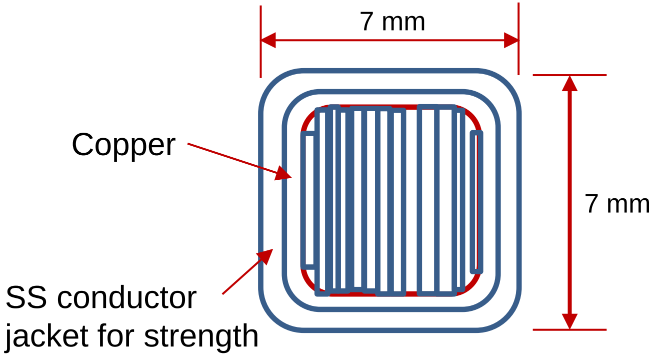


Base cable: 50 tapes YBCO Tapes with 38 μm substrate
(Van Der Laan, HTS4Fusion, 2015)

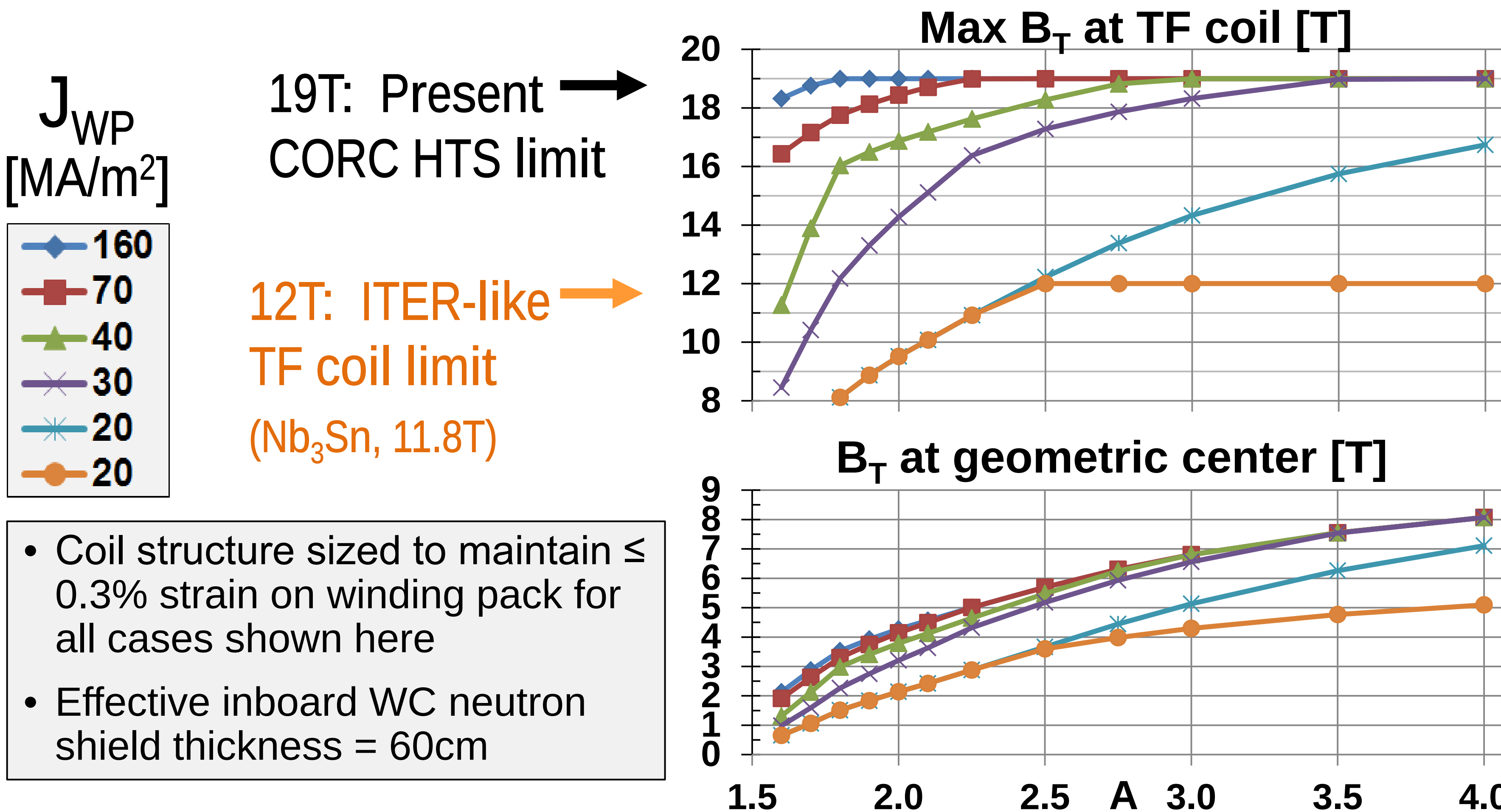
7 kA CORC (4.2K, 19 T) cable

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

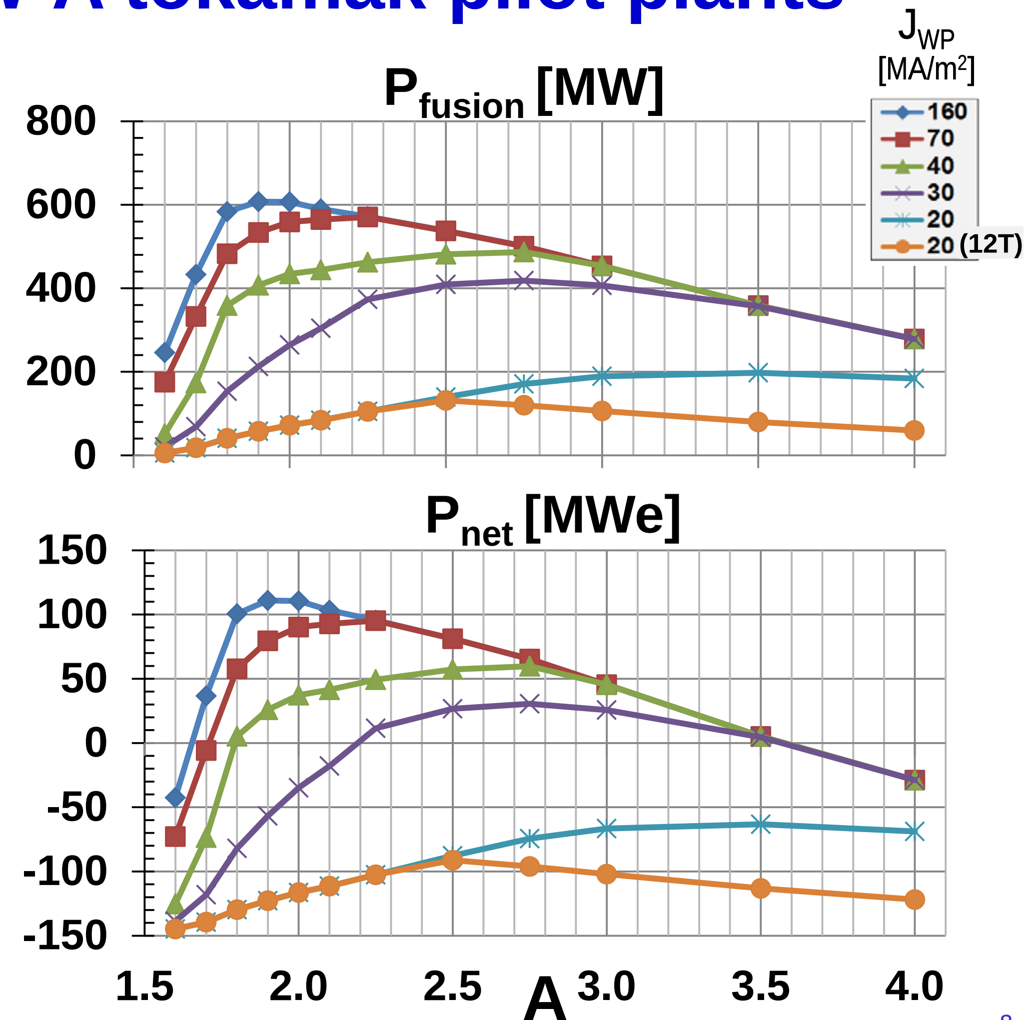


- Coil structure sized to maintain ≤ 0.3% strain on winding pack for all cases shown here
- Effective inboard WC neutron shield thickness = 60cm

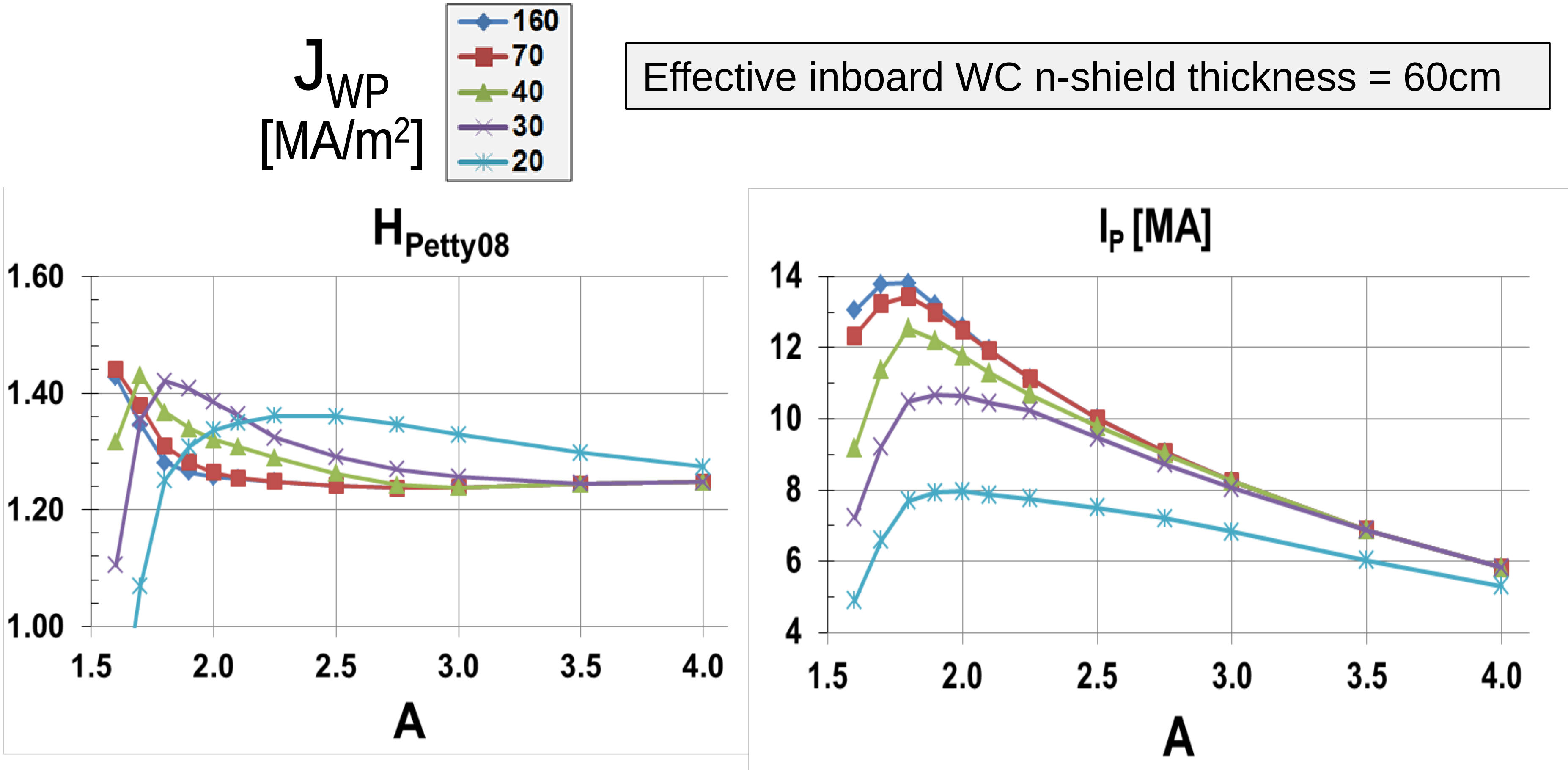
High current density HTS cable motivates consideration of low-A tokamak pilot plants

- 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**

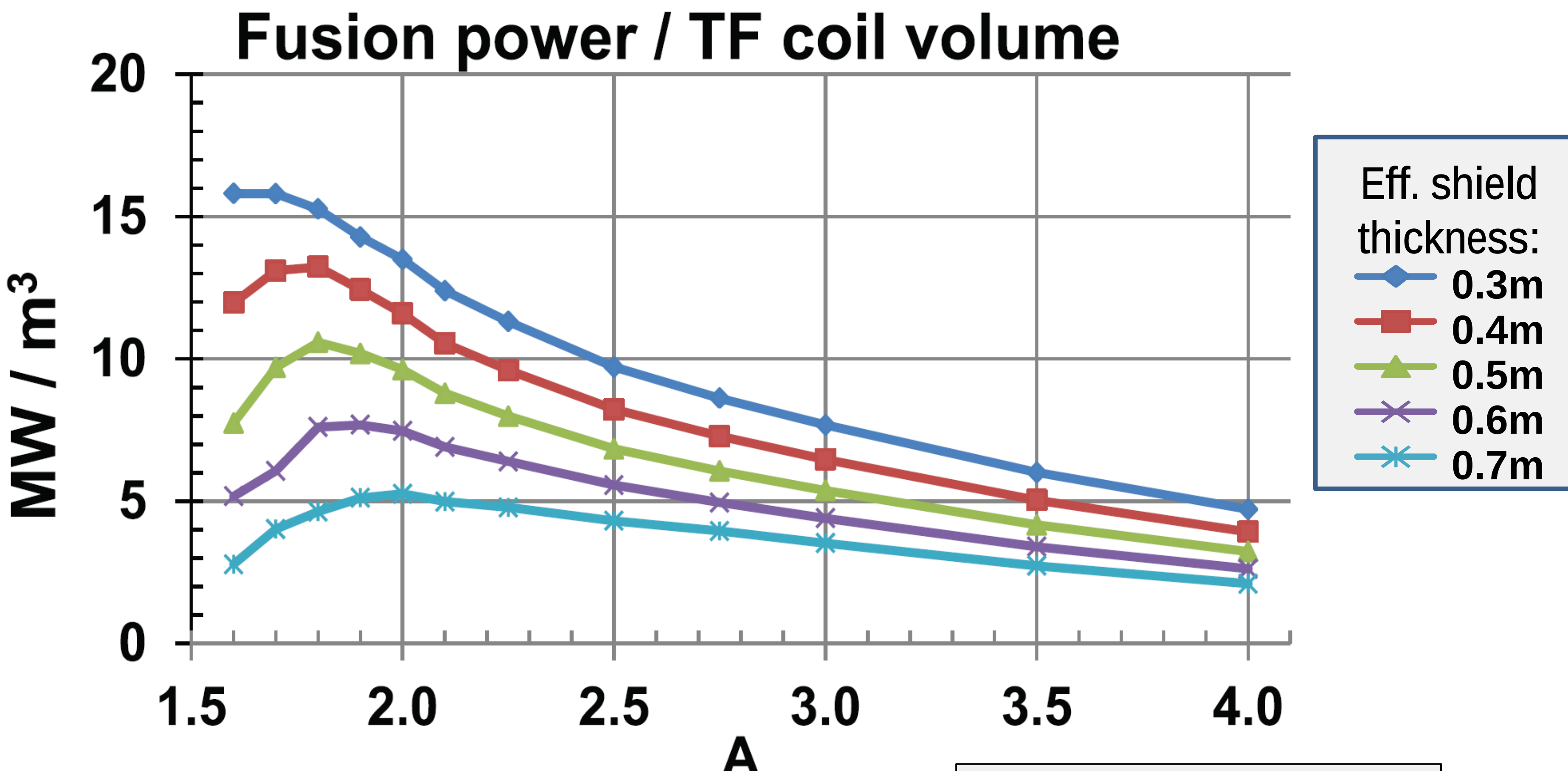
A ~ 2 attractive at high J_{WP}



A ≥ 2 pilot plant scenarios have elevated H > 1, I_P = 6-12MA, f_{BS} ~ 80% (not shown)

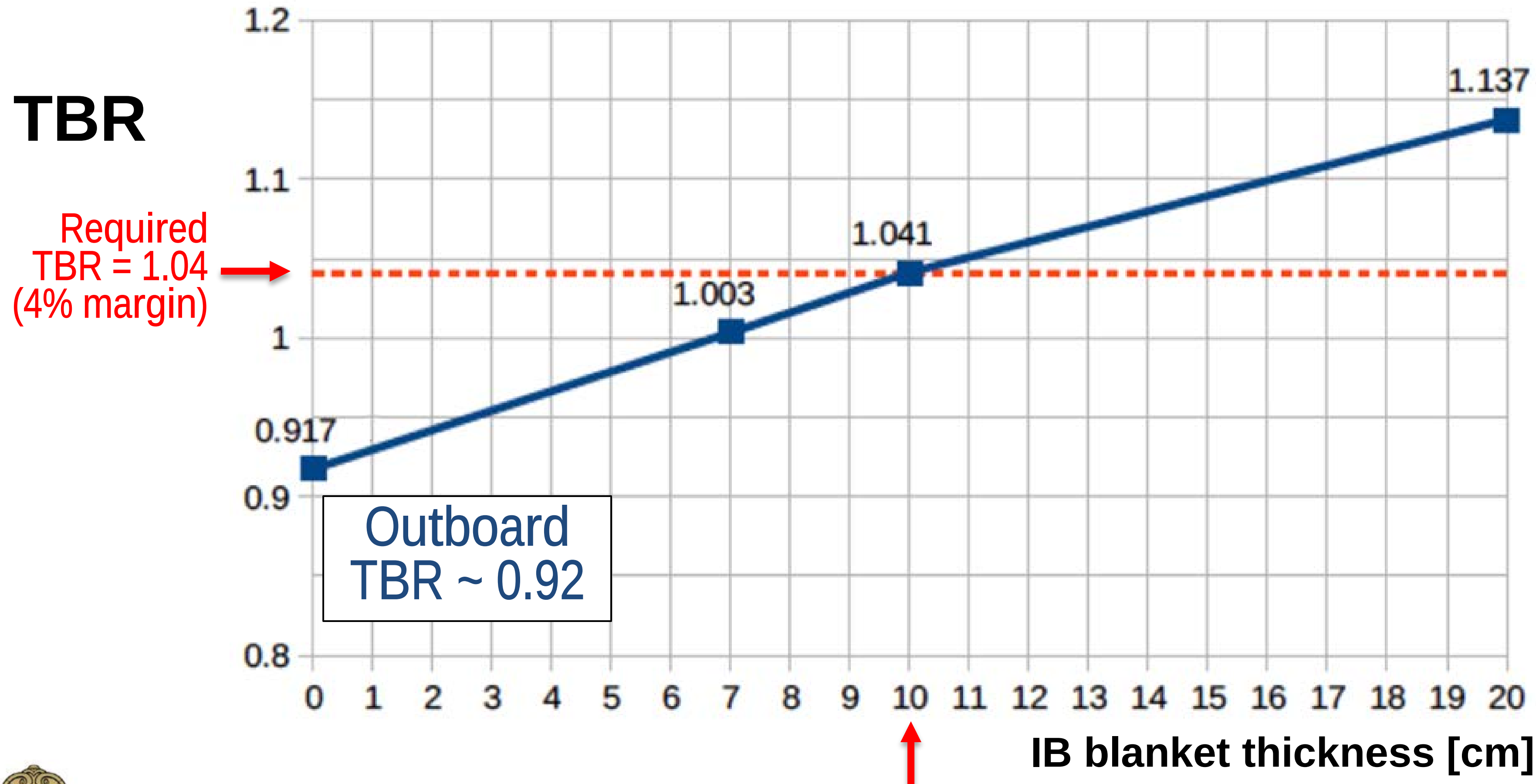


A ≤ 2 maximizes TF magnet utilization



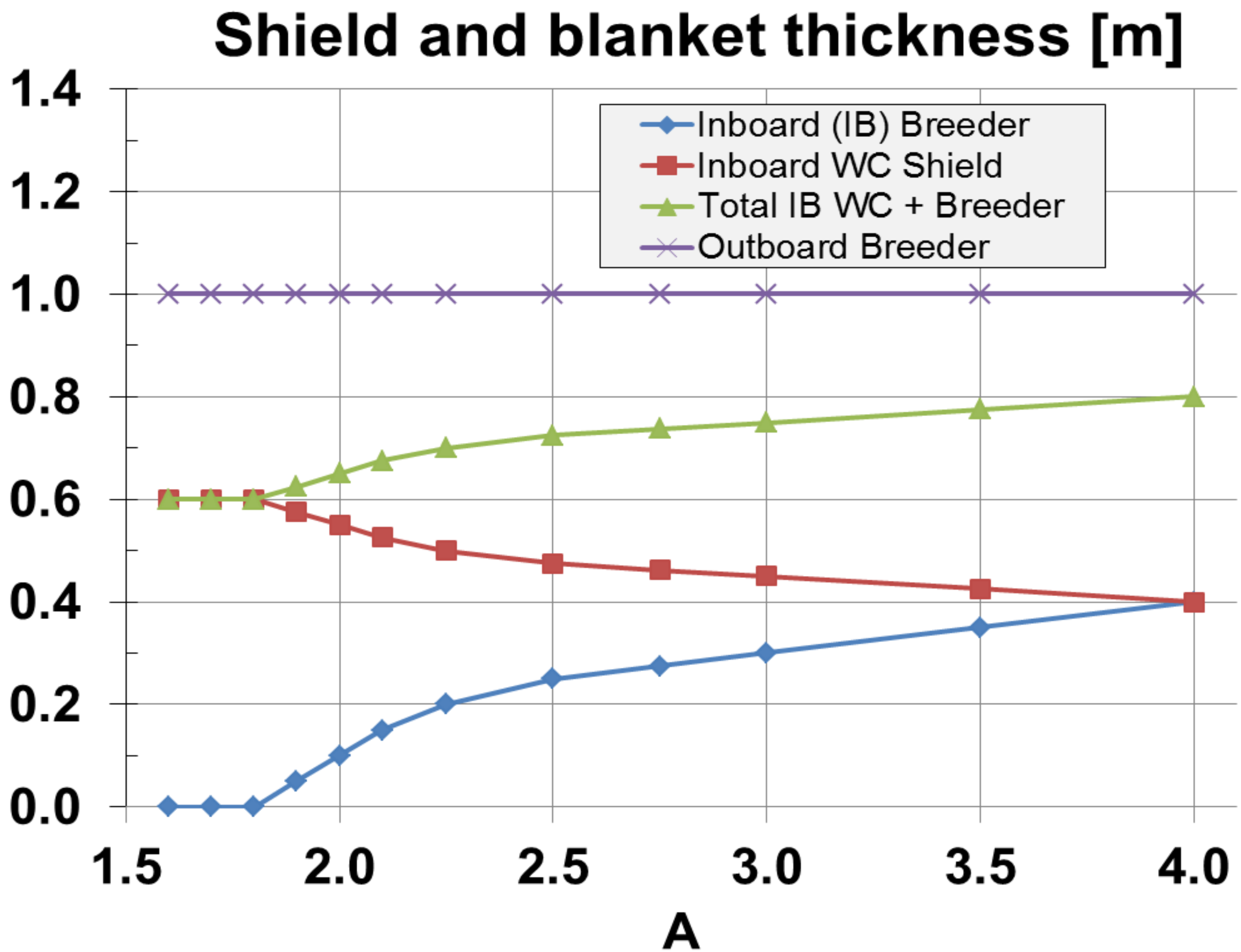
J_{WP} = 70MA/m^2

Need inboard breeding for TBR > 1 at A=2

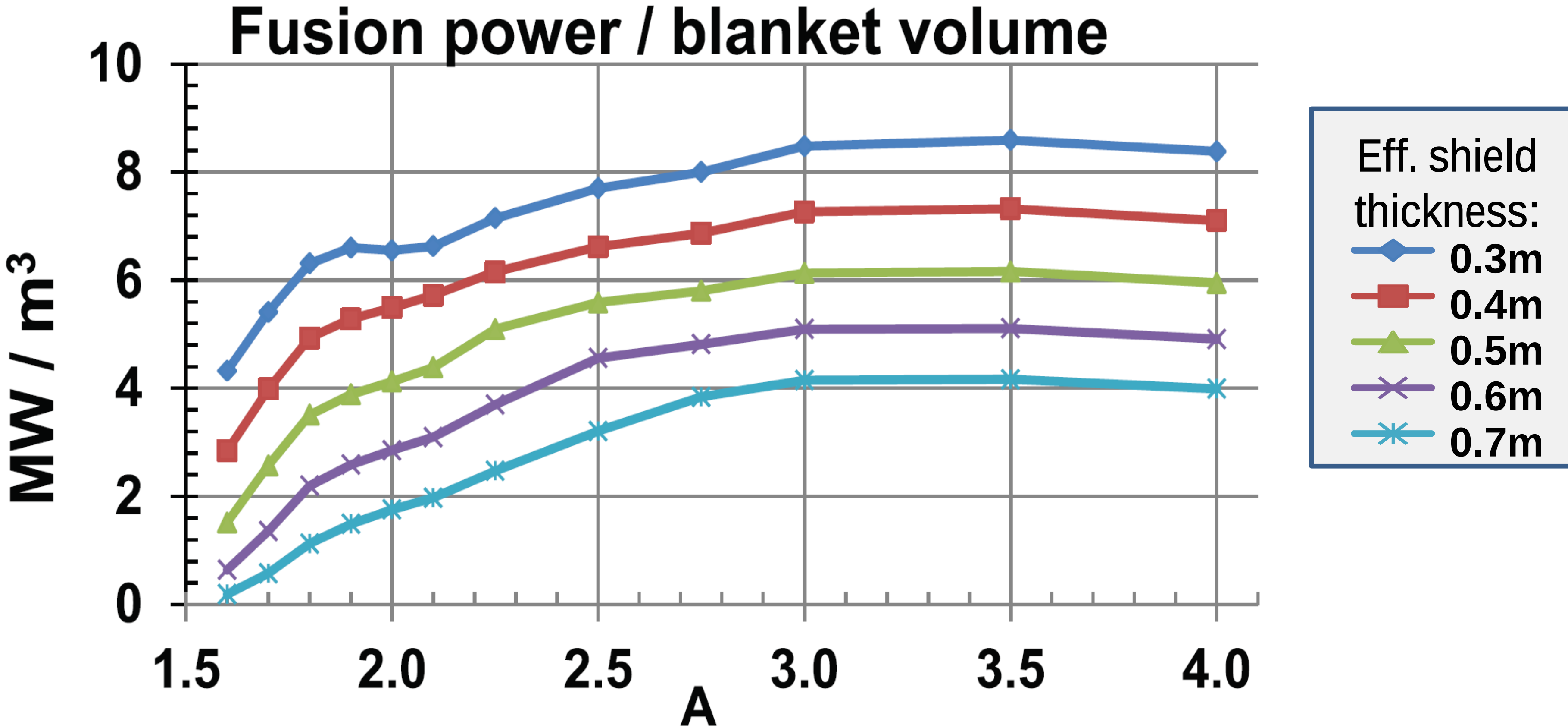


10cm IB blanket sufficient for TBR > 1.04

Model blanket and shield thickness vs. A

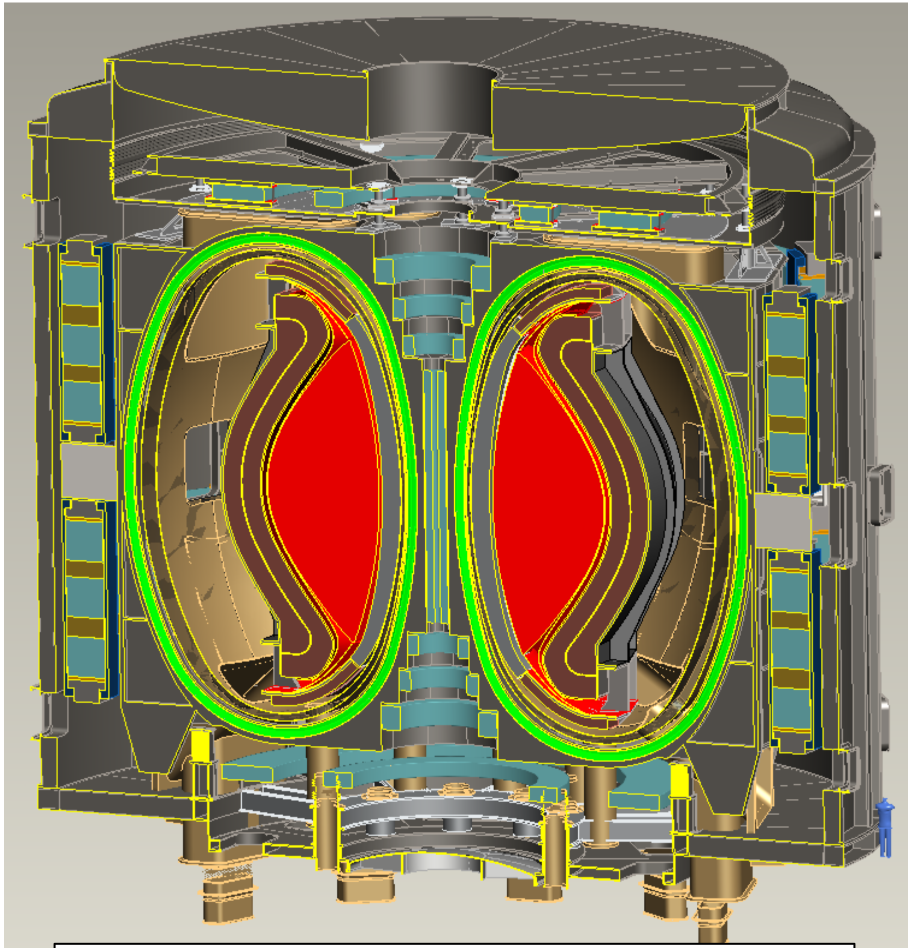


A ≥ 3 maximizes blanket utilization



J_{WP} = 70MA/m^2

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



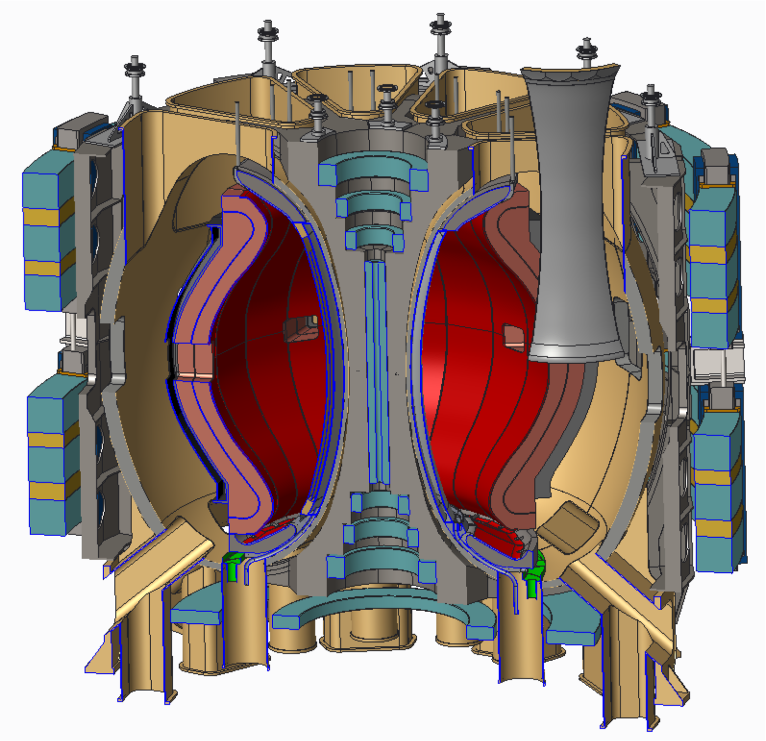
Cryostat volume ~ 1/3 of ITER

B_T = 4T, I_P = 12.5MA
κ = 2.5, δ = 0.55
β_N = 4.2, β_T = 9%
H_{98} = 1.8, $H_{Petty\text{-}08}$ = 1.3
f_{gw} = 0.80, f_{BS} = 0.76

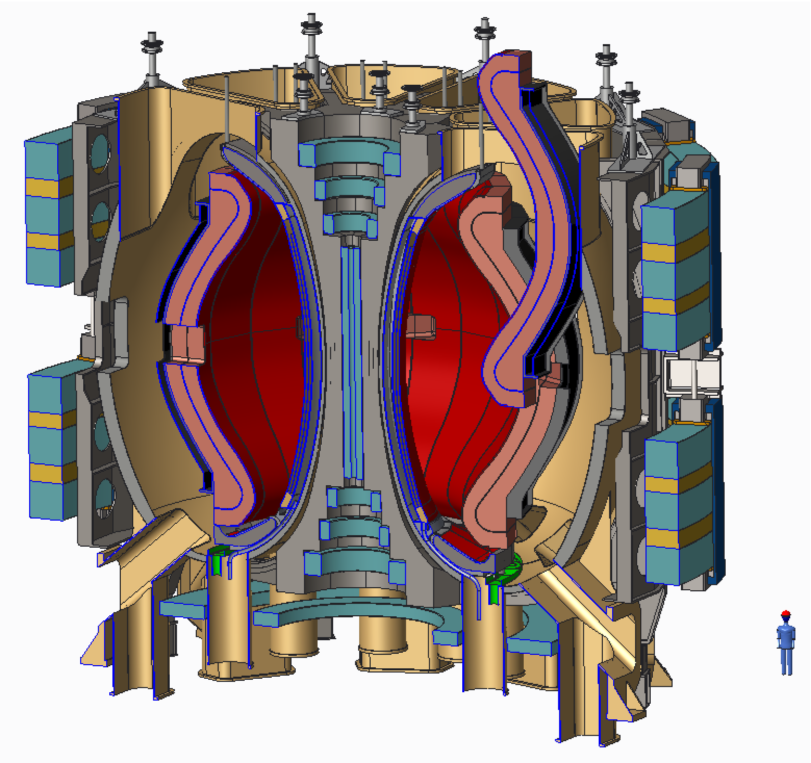
Startup I_P (OH) ~ 2MA
J_{WP} = 70MA/m^2
$B_{T\text{-}max}$ = 17.5T
No joints in TF
Vertical maintenance
P_{fusion} = 520 MW
P_{NBI} = 50 MW, E_{NBI} = 0.5MeV
Q_{DT} = 10.4
Q_{eng} = 1.35
P_{net} = 73 MW

$\langle W_n \rangle$ = 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

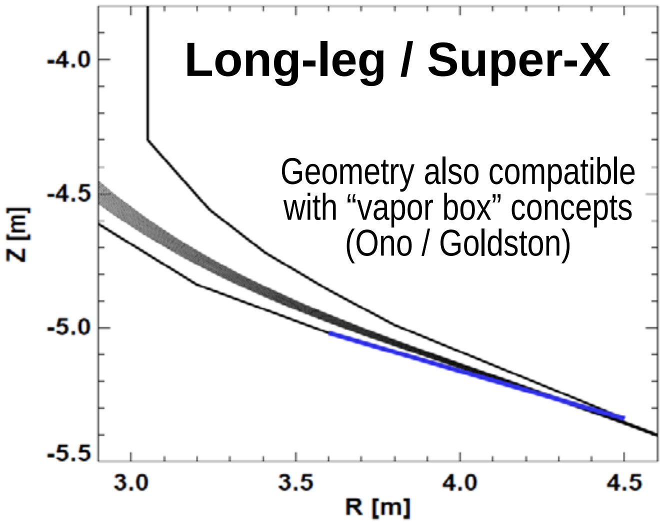


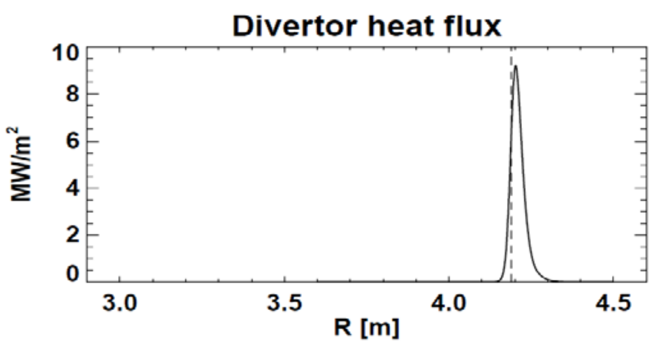
Inboard blanket removed once outboard blanket sectors removed – depending on the toroidal extent of the inboard blanket

Long-leg divertor

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

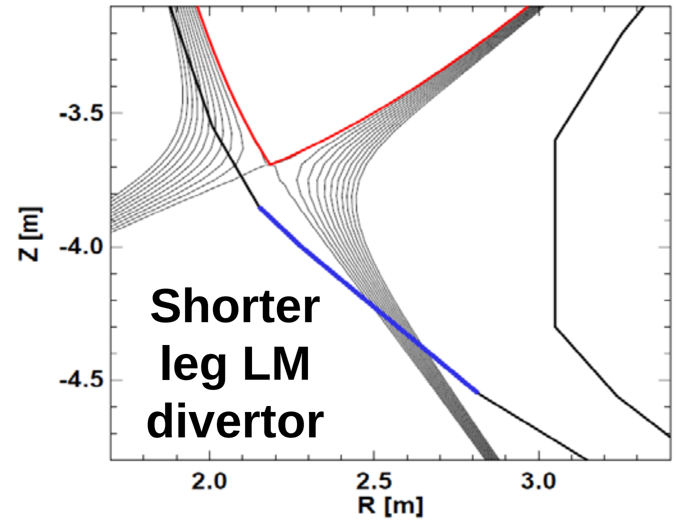
Detachment can further reduce q_{div} ~ 2-3x

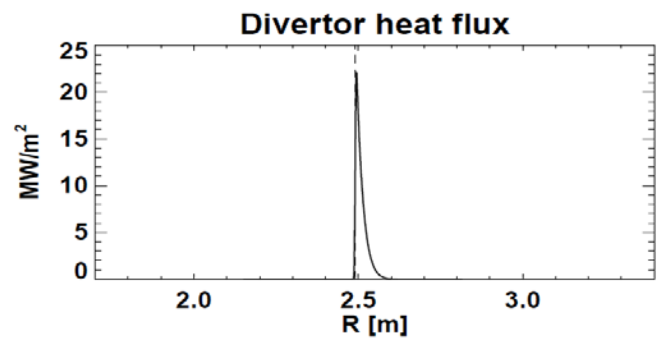




Liquid metal (short-leg)

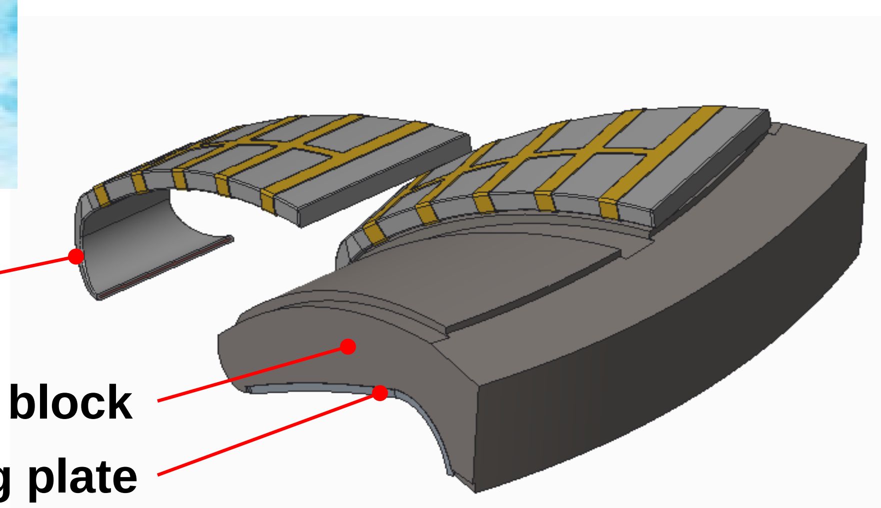
q_{div} = 21MW/m^2, R_{strike} = 2.5m





Exploring liquid metal divertor design similar to flowing water curtain systems

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

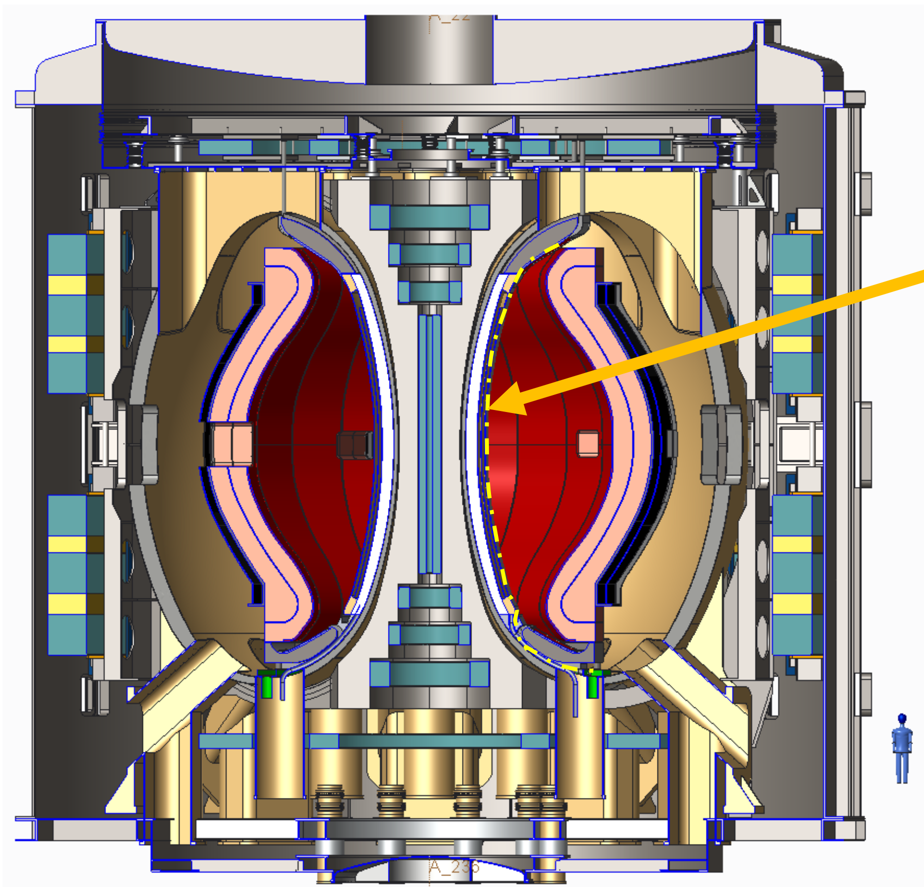


LM containment structure

Shield block

Ferritic steel backing plate

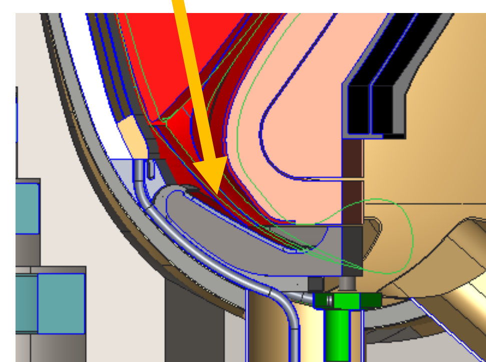
HTS ST-FNSF design with Li flow on divertor and inboard surfaces



Double null liquid metal divertor system

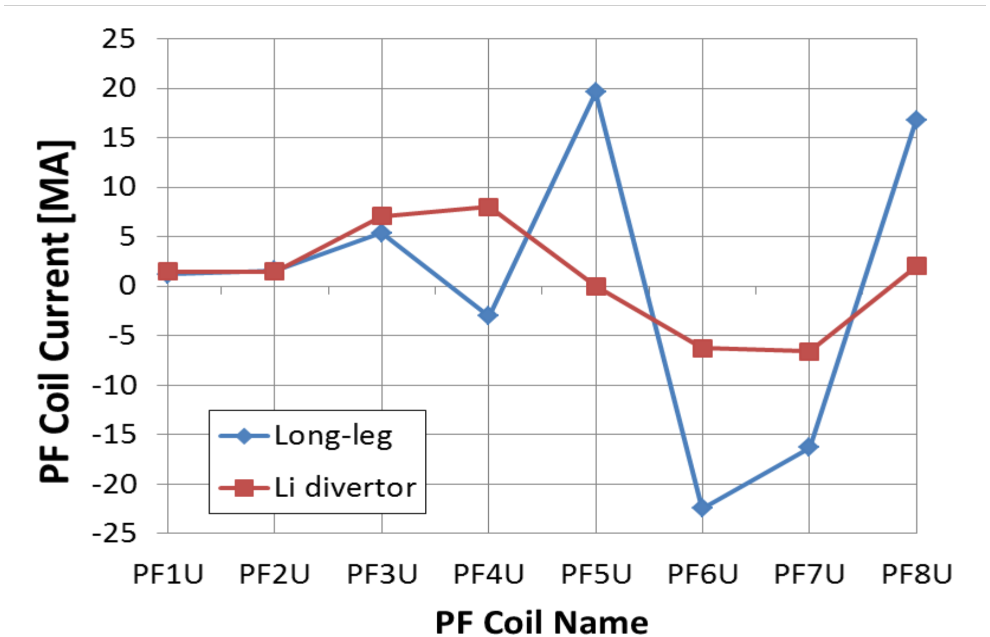
Li flows from upper divertor down the inboard wall, exiting just after the lower inboard divertor.

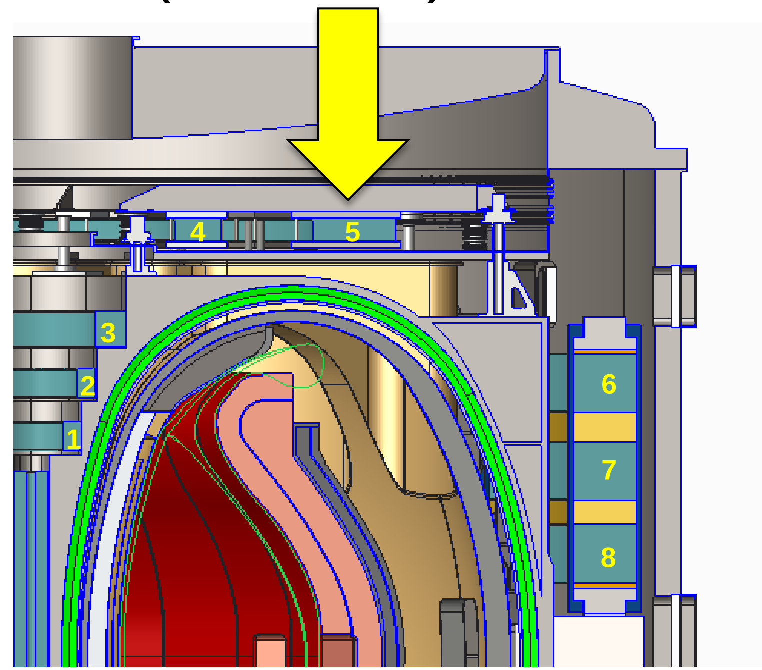
Separate Li cooling of lower divertor



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