# Parametric studies of next-step spherical tokamaks using high-temperature superconductors*

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# Possible missions for next-steps

1. Integrate high-performance, steady-state, exhaust
   - Divertor test-tokamak - DTT

***Past (& future) PPPL Studies***

2. Fusion-relevant neutron wall loading
   - $\Gamma_n \sim 1\text{-}2\mathrm{MW/m^2}$, fluence: $\geq 6\mathrm{MW\text{-}yr/m^2}$
3. Tritium self-sufficiency
   - Tritium breeding ratio TBR $\geq 1$
4. Electrical self-sufficiency
   - $Q_{eng} = P_{electric} / P_{consumed} \sim 1$

***Recent / Present PPPL-led Studies***

5. Large net electricity generation
   - $Q_{eng} >> 1$, $P_{electric} = 0.5\text{-}1$ GWe

# What is optimal A for HTS FNSF / Pilot Plant?

- **Why high-temperature superconductors (HTS)?**
  - Higher current density & $B_T$, tolerates higher nuclear heating

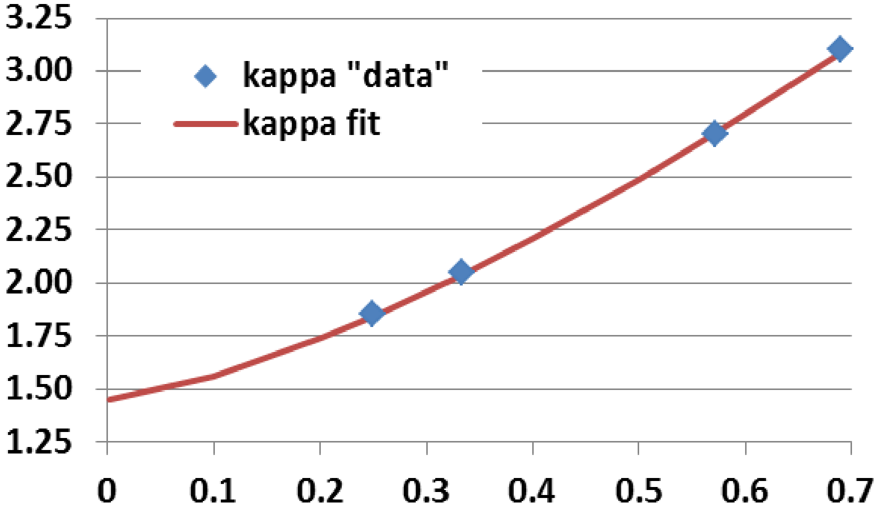
## Approach:

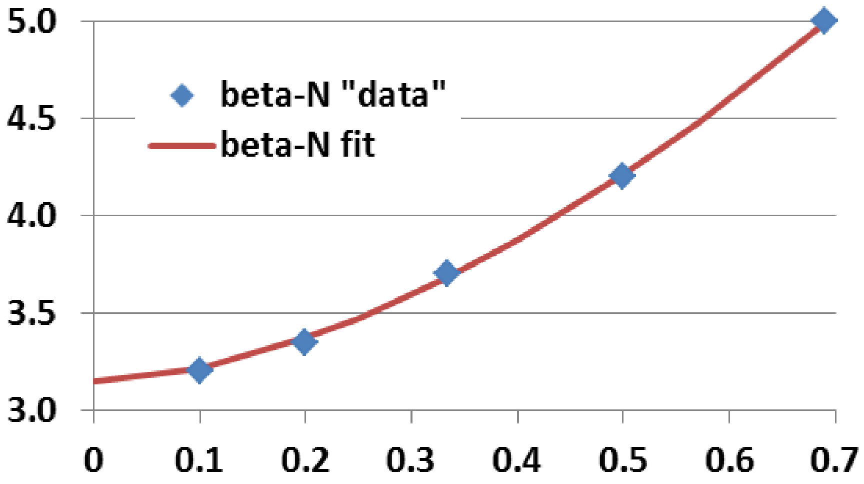
- Fix plasma major radius and heating power
  - Choose compact device ≤ $R_0$ = 3m to reduce cost
- Apply magnet and core plasma constraints
- **Vary aspect ratio from A = 1.6 to 4**
- **Vary HFS WC shield thickness: 30-70cm**
- Calculate achievable $Q_{DT}$ , $Q_{eng}$ , required $H_{98}$
- Assess various trade-offs

# Plasma constraints

- Fix plasma major radius at $R_0$ = 3m
  - Chosen to be large enough to allow space for HTS neutron shield and access $Q_{eng} > 1$
- Inboard plasma / FW gap = 4cm
- Use $\varepsilon$-dependent $\kappa(\varepsilon)$, $\beta_N$ $(\varepsilon)$ (see next slide)
- Greenwald fraction = 0.8
- q* not constrained
  - q* is better $\varepsilon$-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: $\kappa(\varepsilon)$, $\beta_N(\varepsilon)$





$\varepsilon = A^{-1}$

- NSTX data at low-A (+ NSTX-U/ST-FNSF modelling)
- DIII-D, EAST for higher-A
  - $\kappa \rightarrow 1.4$ 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 = 0.8 GPa
  - Maximum TF winding pack current density = 70 $MA/m^2$
  - OH at small R → higher solenoid flux swing for higher A
- Shielding / blankets
  - Assume HTS fluence limit of $3.5x10^{22}$ $n/m^2$
  - 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
    - See backup for assumed thicknesses
- 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

# HTS performance vs. field and fast neutron fluence

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

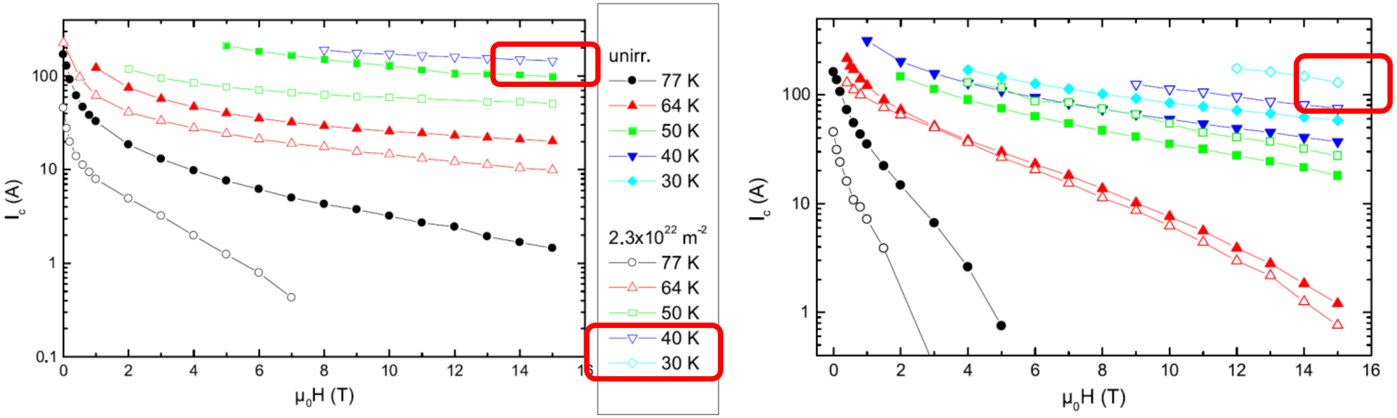


**Figure 6.** Critical currents (ASC-40) in magnetic fields applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) before and after irradiation to a fast neutron fluence of $2.3 \cdot 10^{22}$ $m^{-2}$.

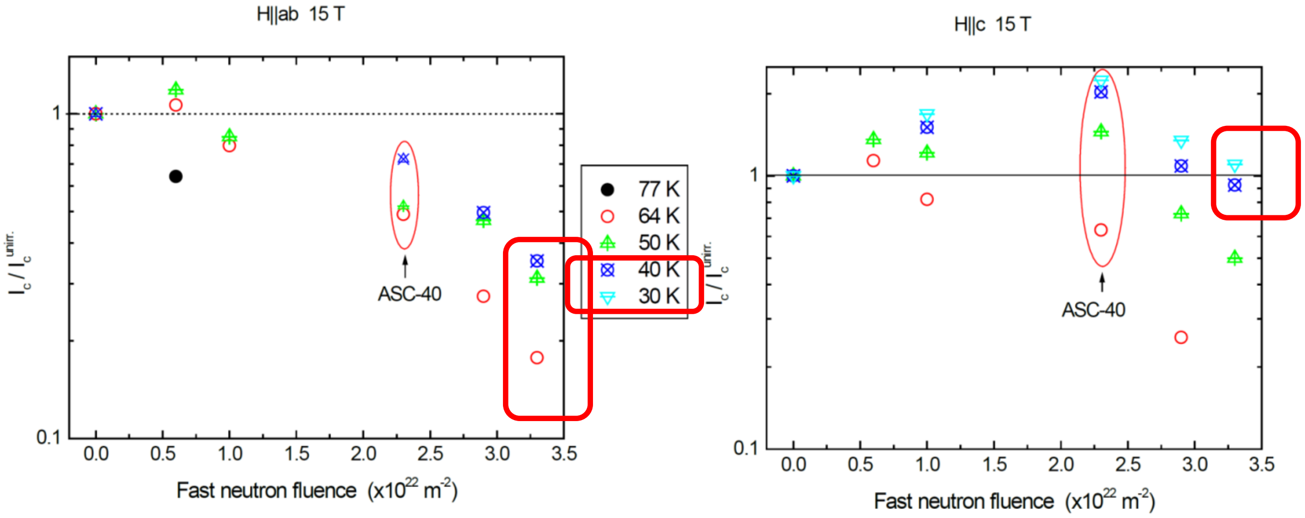


**Figure 8.** Normalized critical currents in a magnetic field of 15 T applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) as a function of neutron fluence.

# Simplified TF magnet design equations

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

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

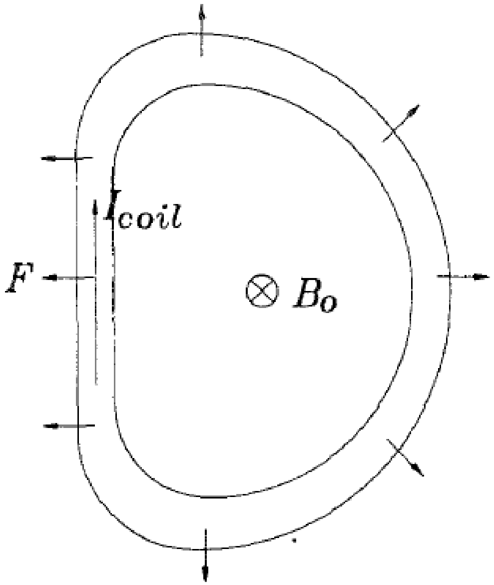


**Fig. 5.** Lorentz forces are normal to the conductor in the poloidal plane.

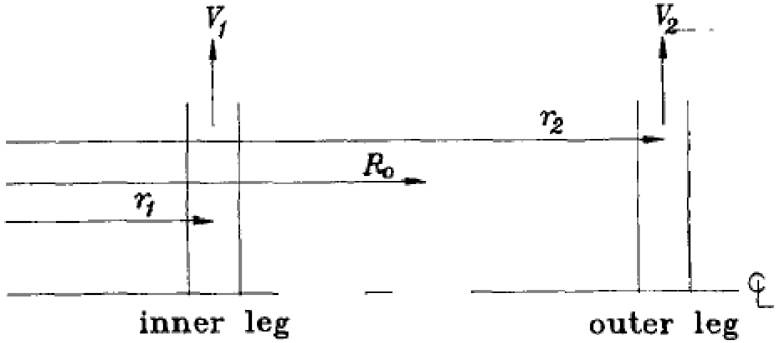


**Fig. 7.** Geometry for force and moment balances.

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

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

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

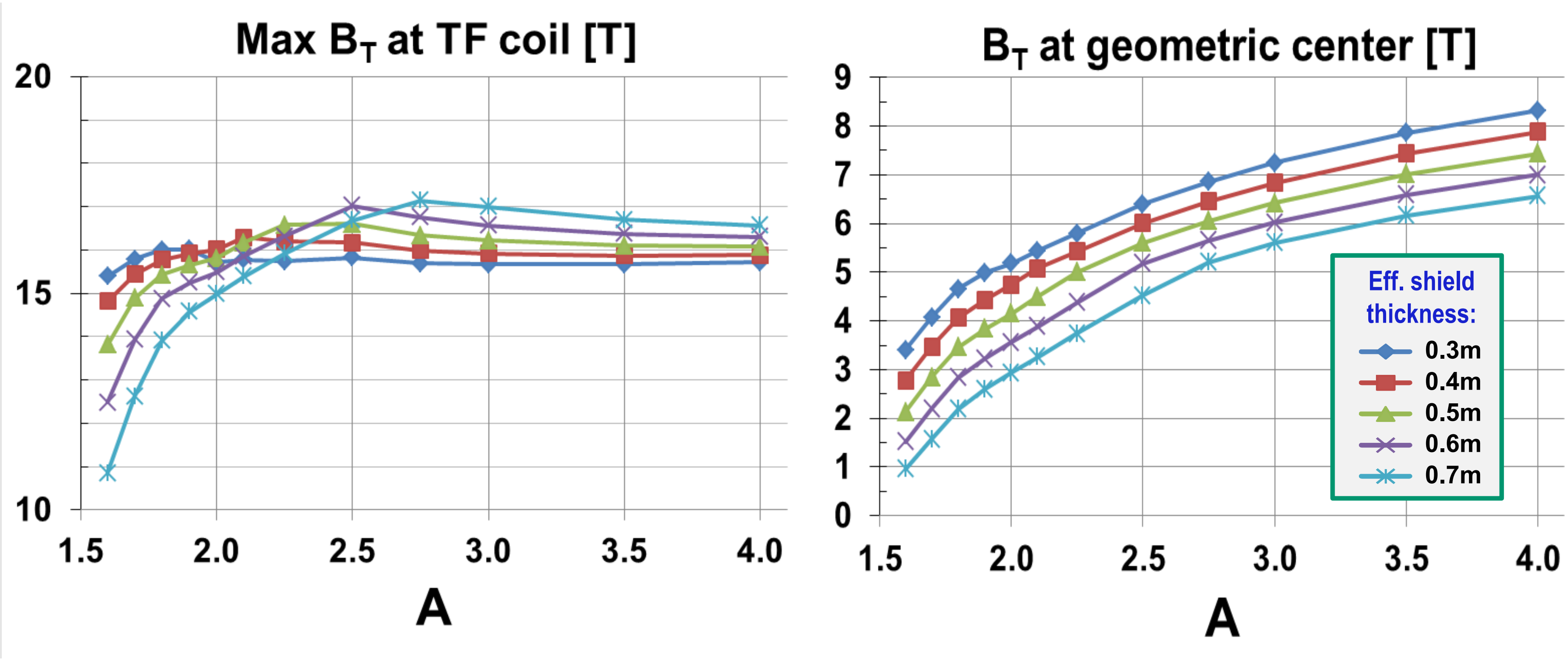
$$I_{\text{coil}} = \frac{5 \times 10^6 B_0 R_0}{N_{\text{coils}}}$$

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

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

# Simplified TF model used here projects to max $B_T$ at TF coil ~16-17T

- Assume winding pack provides no/little structural support
- Winding pack area fraction chosen to match stress & $J_{wp}$ limits



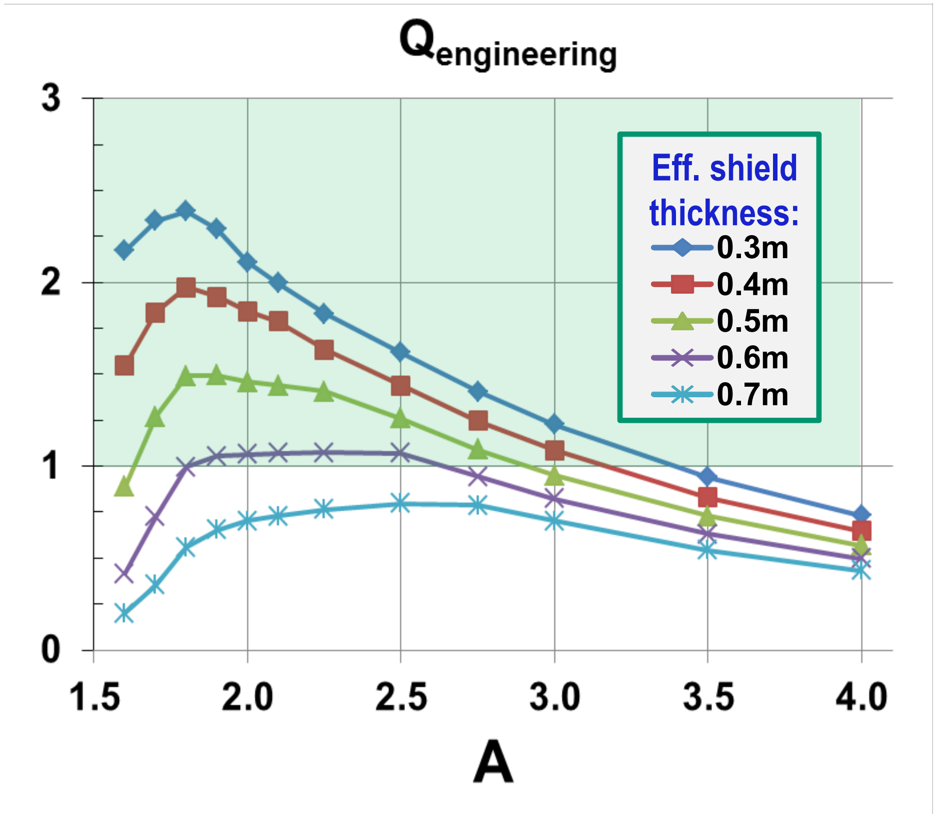
Max $B_T$ at TF coil [T]

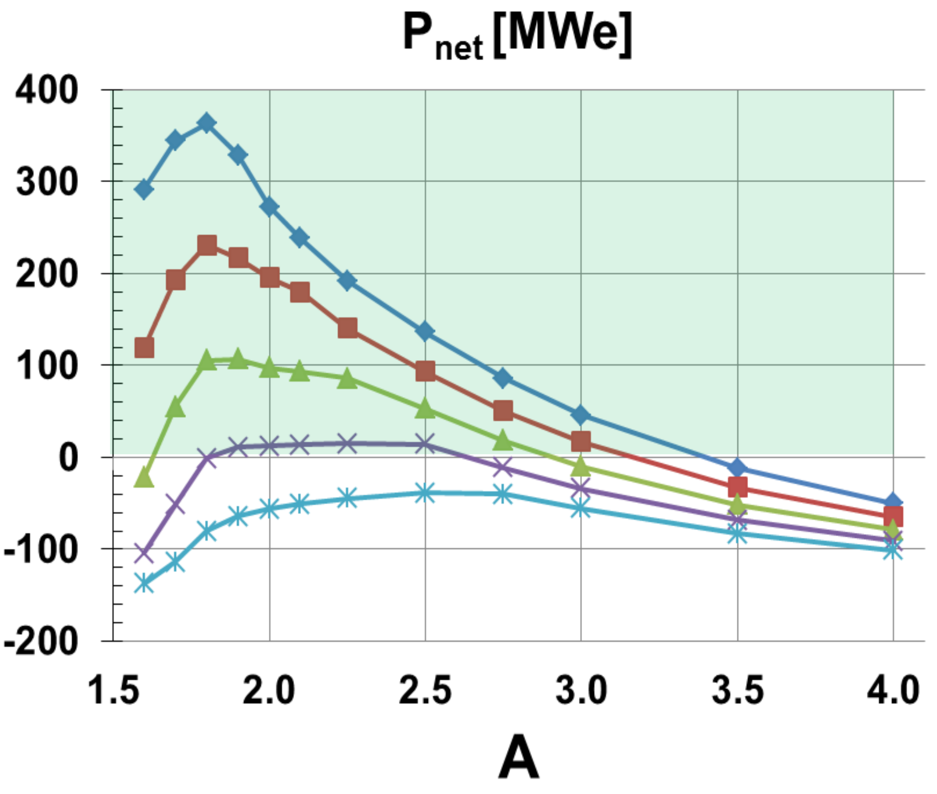
$B_T$ at geometric center [T]

Achievement of higher field limit (~17T) at coil could support **3x higher fusion power** vs. 13T limit of ITER-style magnets

# Maximum $Q_{DT}$ , $Q_{eng}$ achieved for A = 1.8-2.5

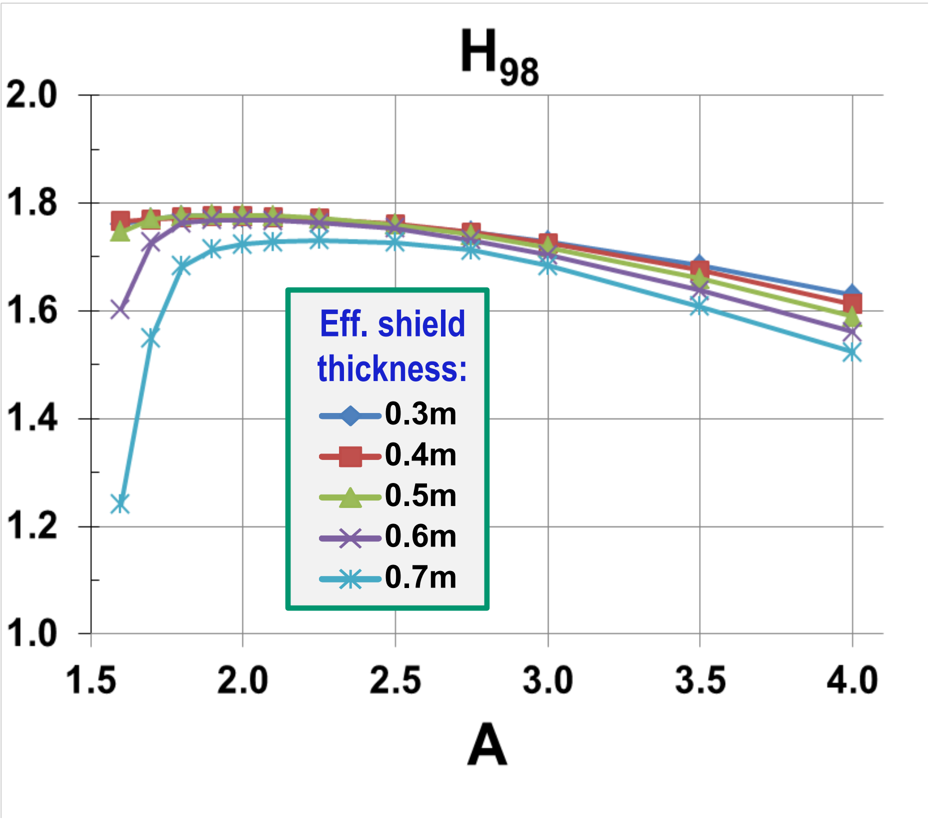
## $Q_{eng}$ ≥ 1 requires shielding thickness ≤ 60cm
## A ≈ 2 optimal for thinner shield cases

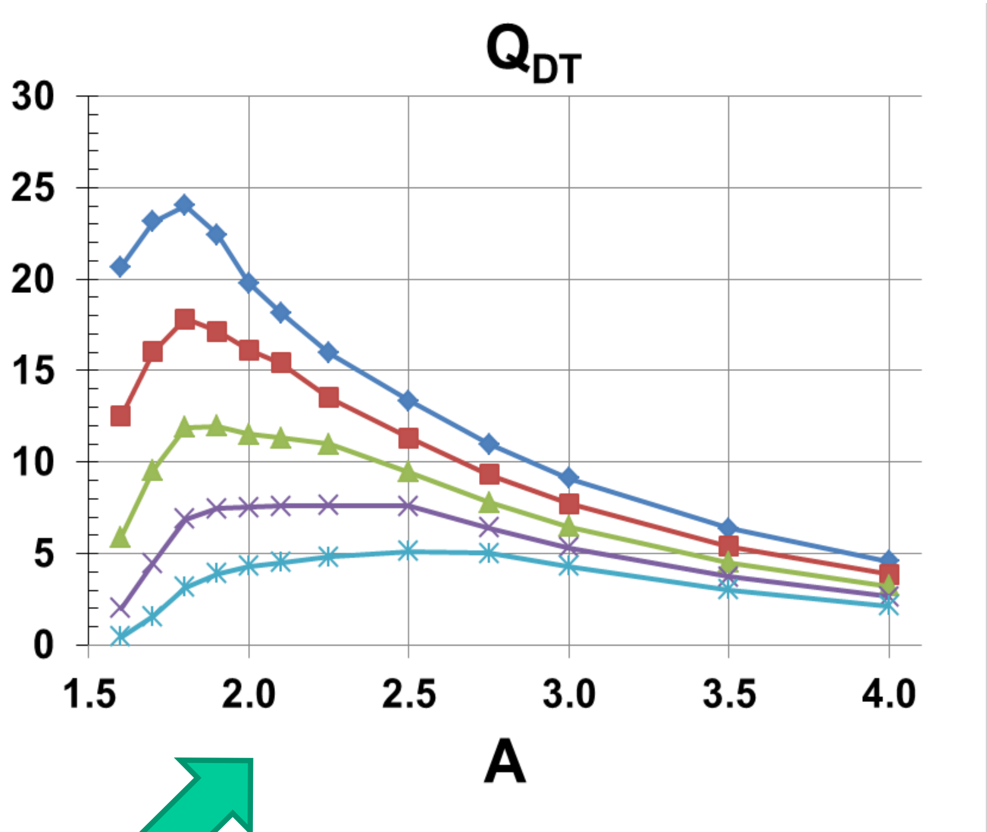




**Reminder: confinement multiplier not constant:**
**$H_{98}$ is adjusted to achieve full NICD for assumed $\kappa(\varepsilon)$, $\beta_N(\varepsilon)$**

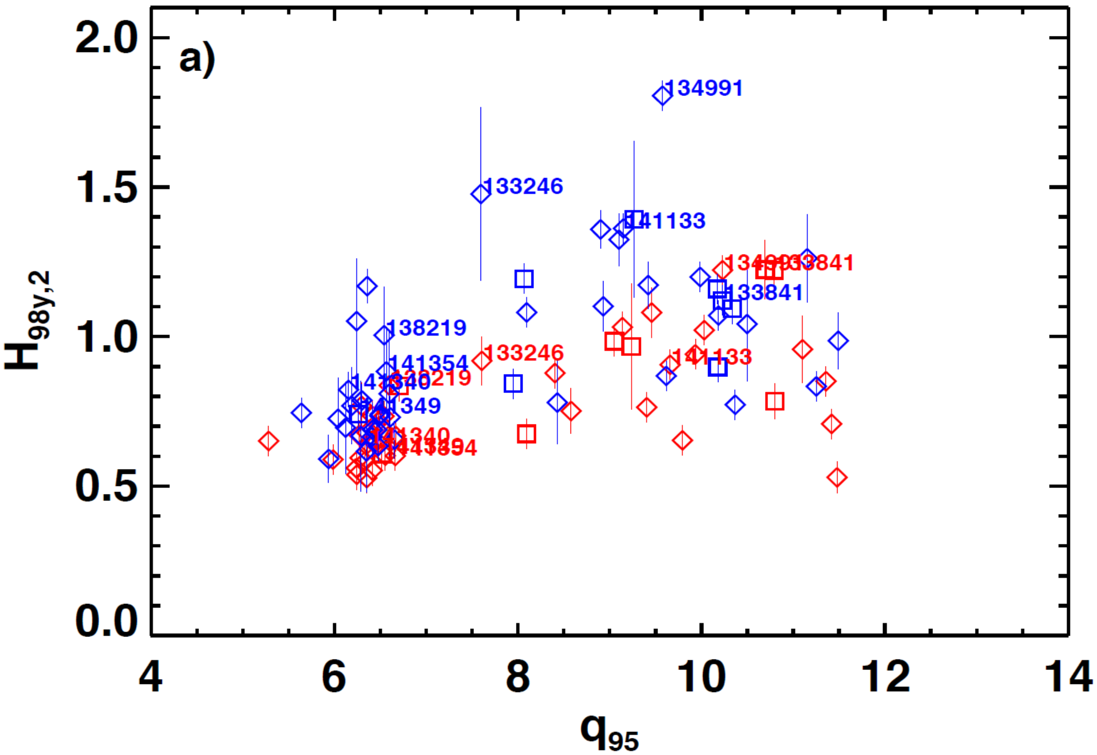
# Required $H_{98}$ is nearly constant ~ 1.75 for A = 1.8-2.5 → optimal A ≈ 2 is not a confinement scaling effect





**Fusion gain $Q_{DT} \geq 7$ needed for $Q_{eng} > 1$**

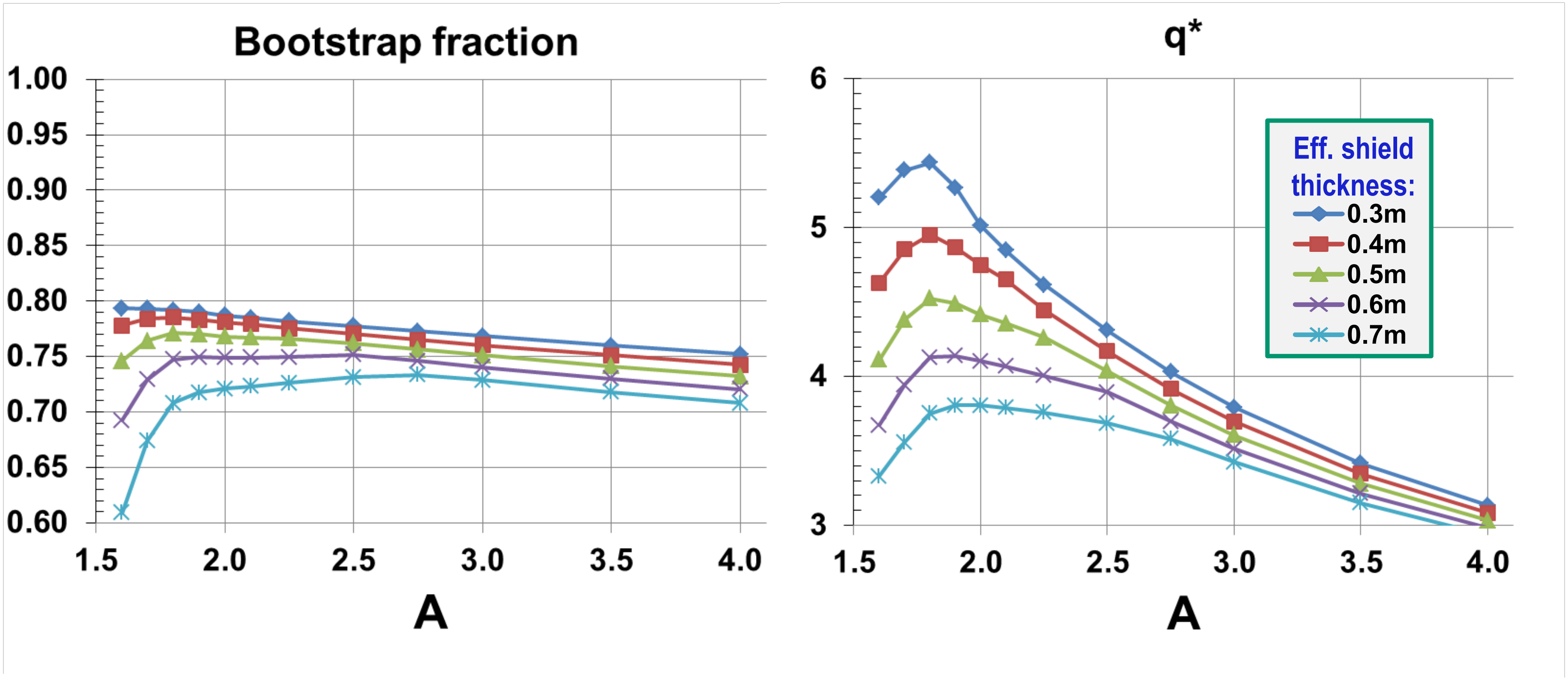
# A few “Enhanced pedestal” H-modes (EPH) in NSTX have accessed $H_{98}$ in the range of 1.5-2



- Highest $H_{98}$ in EPH appears to require:
  - Strong edge rotation shear (3D fields/edge island?)
  - Lithium wall coatings (lower edge recycling, $\nu^*$ )
- Often transient (EPH lost w/ ELM) – much more work needed to understand access and sustainment

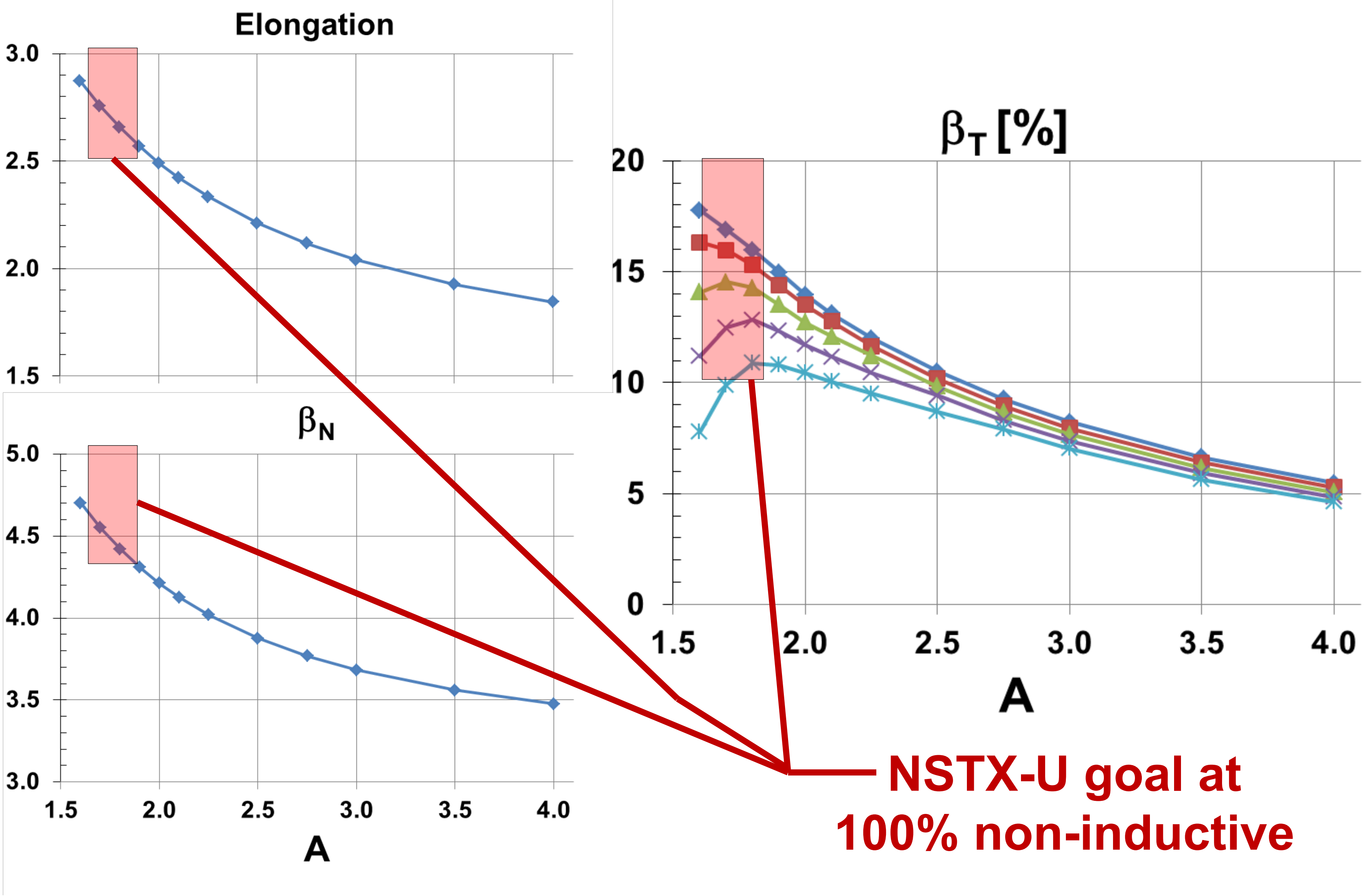
# Highest performance scenarios have $f_{BS}$ = 70-80% and q* ≥ 4-5 for shield thickness < 50cm

- **Should be acceptable from control/stability standpoint (?)**



- **Further, all scenarios have q* ≥ 3 (benefit of high $B_T$)**

# NSTX-U aims to access fully non-inductive plasmas relevant to FNSF / Pilot-plants with $\kappa$~2.6, $\beta_N$ ~ 4.5, $\beta_T$ = 12-15%

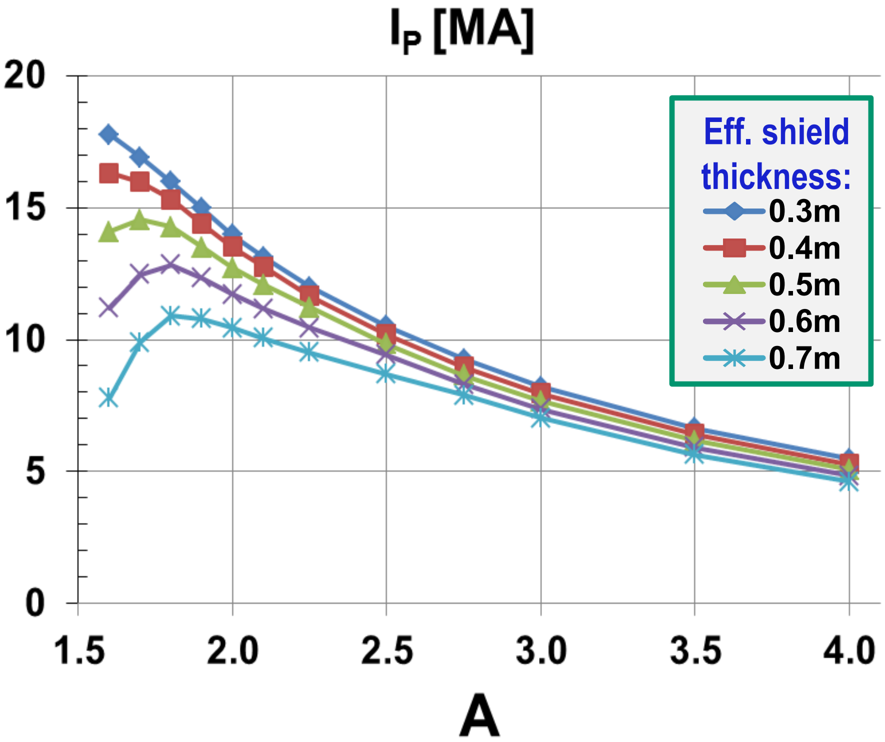


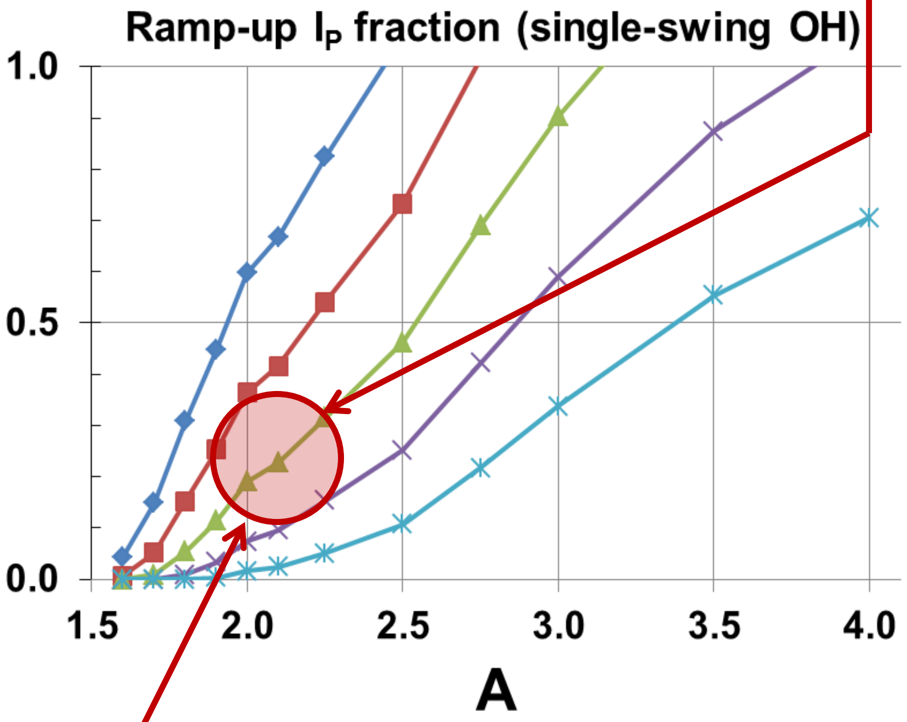
**NSTX-U goal at 100% non-inductive**

# A ≥ 2 enables inclusion of modest ohmic solenoid for plasma current start-up / initial ramp-up

**20T HTS solenoid → provide ~20-30% flat-top $I_P$ for A ≈ 2.1**

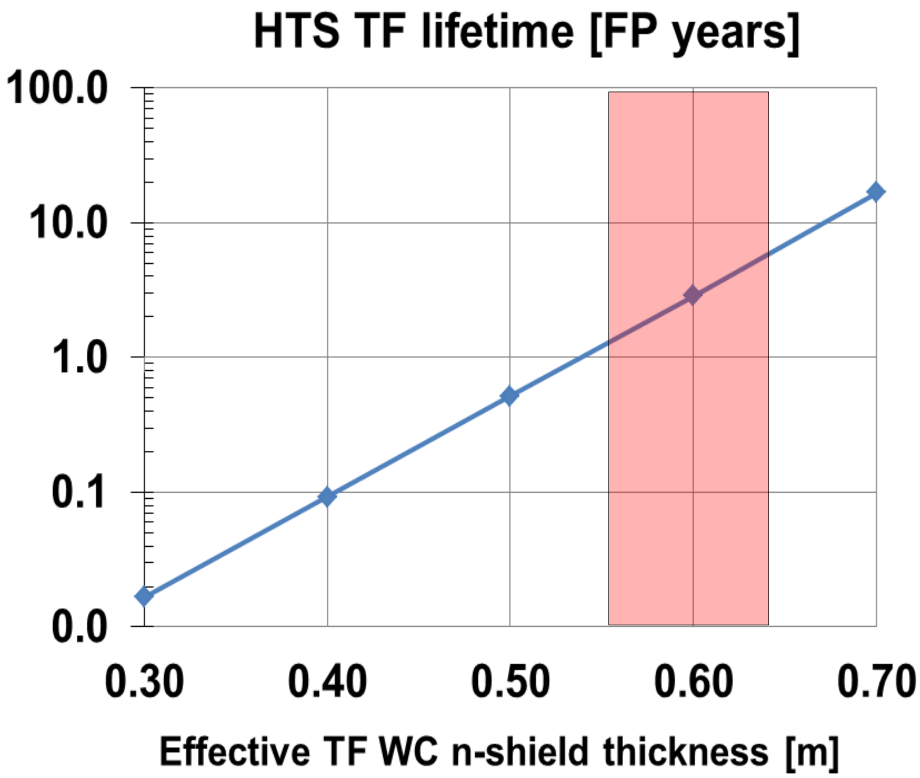
**Ramp-up fraction ~50-100% for A = 3-4**

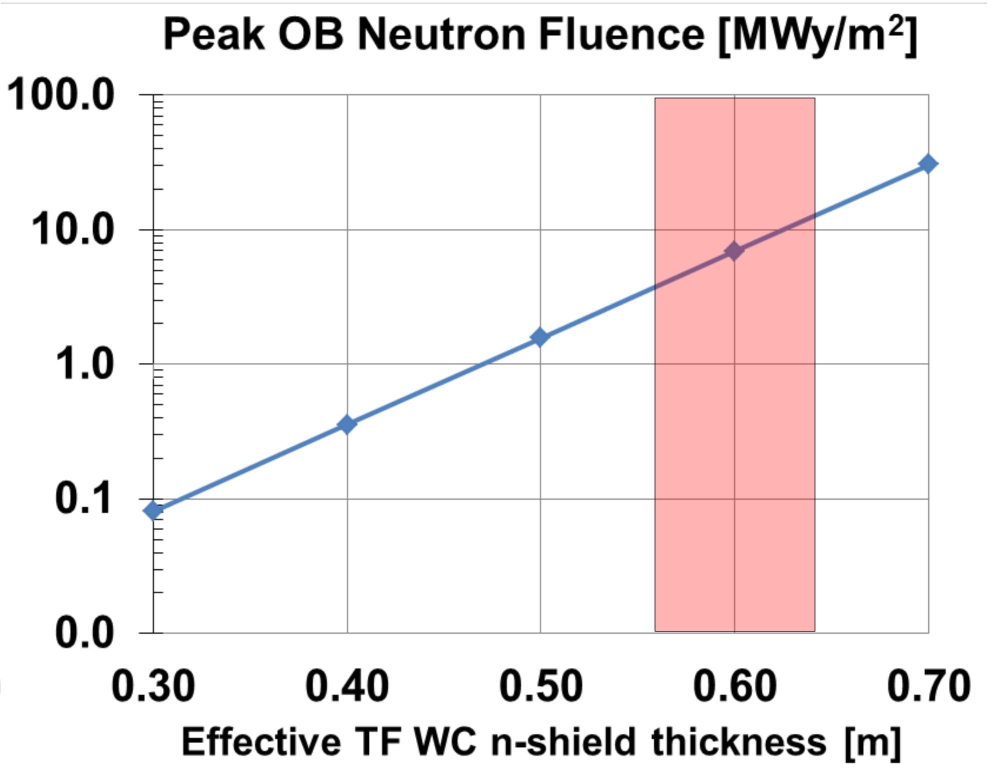




**2-3MA sufficient to absorb NNBI**

# HTS TF lifetime is very strong function of inboard shielding thickness


HTS TF lifetime [FP years]


Peak OB Neutron Fluence [MWy/m$^2$]

**Inboard shield + blanket equivalent to 60cm WC → 3FPY → 6-7MWy/m$^2$ → fulfill FNSF requirement**

# Selection of HTS-ST device goals and configuration

- Attempt to satisfy FNSF (fluence) ***and*** Pilot (net electric) goals:
  - ≥ 6 MWy/$m^2$ neutron wall loading (peak) at outboard midplane
  - $Q_{eng}$ ~ 1 – similar to previous PPPL Pilot Plant Study

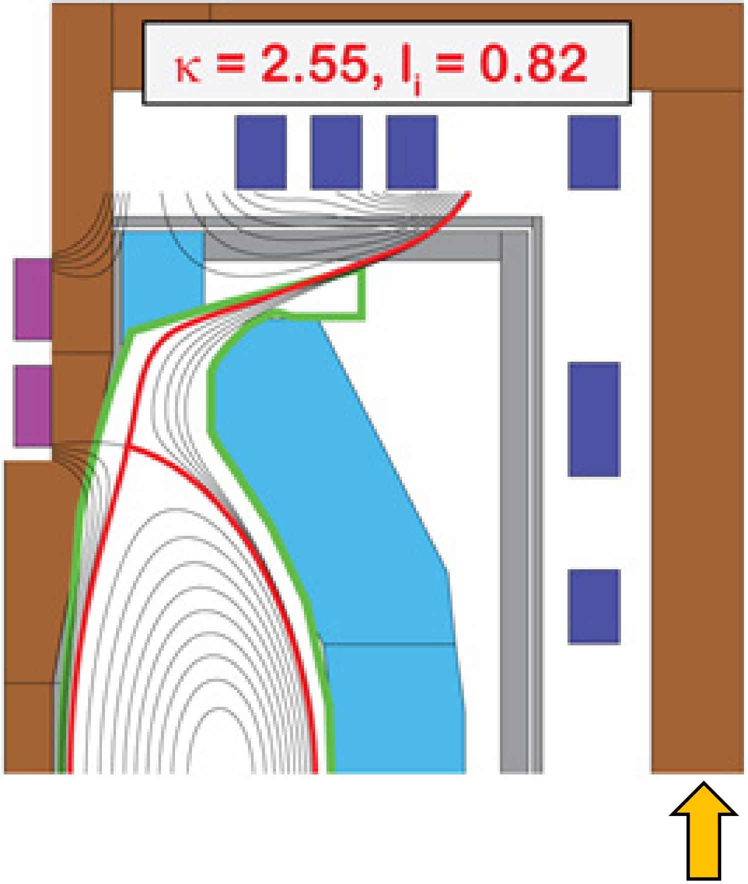
→ Shield equivalent to ~60cm WC, $\Delta/R = 0.2$ → $R_0 = 3$m

  - Assumes n-radiation damage limit of $3.5 \times 10^{22}/m^2$
    - HTS already tested to this damage fluence range

- With small / no inboard breeding, optimal A ~ 2.1-2.4
- But, for TBR ~ 1 probably need A ≤ 2 → chose / try A=2
- **Chosen design point (so far):**
  - R=3m, $B_T$ = 3.5-4.1T, A=2, $\kappa$=2.5, $\beta_N$ = 4.2 (~no-wall limit)
  - $H_{98y2}$ ~ 1.7, $H_{Petty}$ ~ 1.2-1.3, $H_{ST}$ ~ 0.7, $P_{fusion}$ ~ 500-600MW
  - 80% Greenwald fraction, 50MW of 0.5-0.7 MeV NNBI
  - $I_P$ = 12MA, double-swing of small OH provides ~ 2-3MA

# PF coil layout, long-leg divertor, vertical maintenance similar between Cu and HTS FNSFs

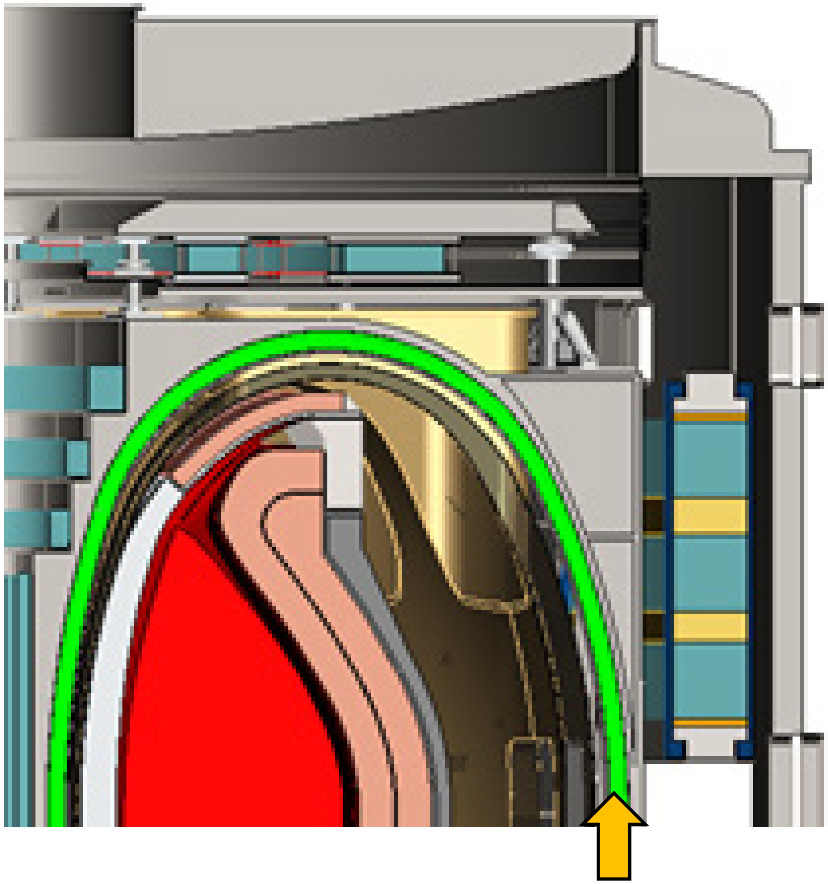
## A=1.7 Copper TF FNSF



**Outboard PF coils enclosed by TF coil**

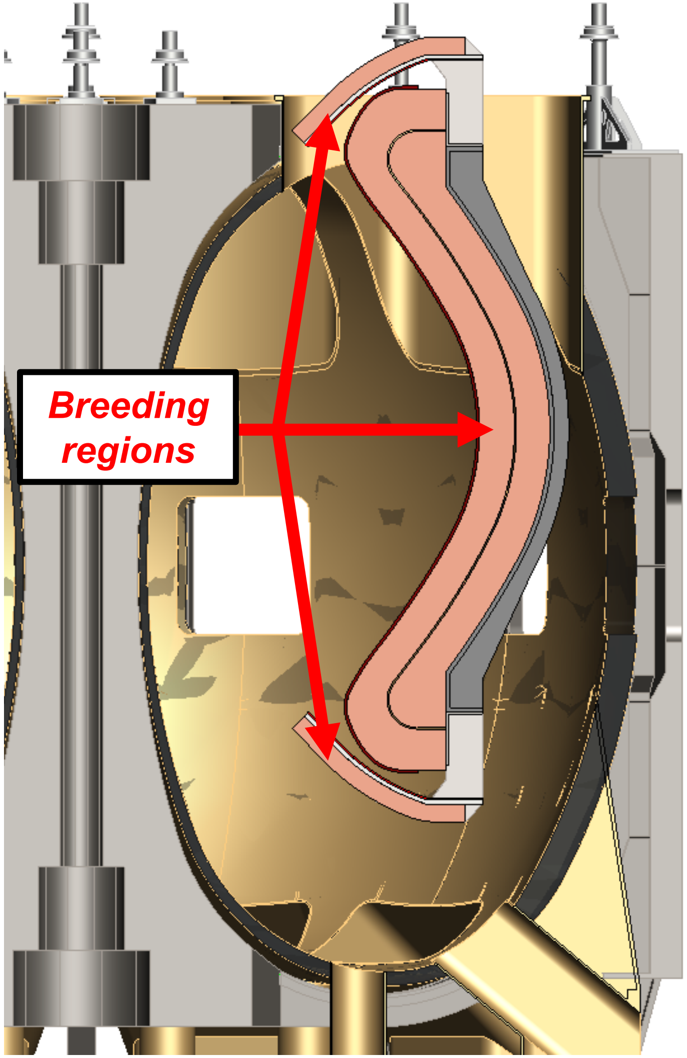
## A=2 HTS TF FNSF/Pilot

**VECTOR-like A, but with small CS**



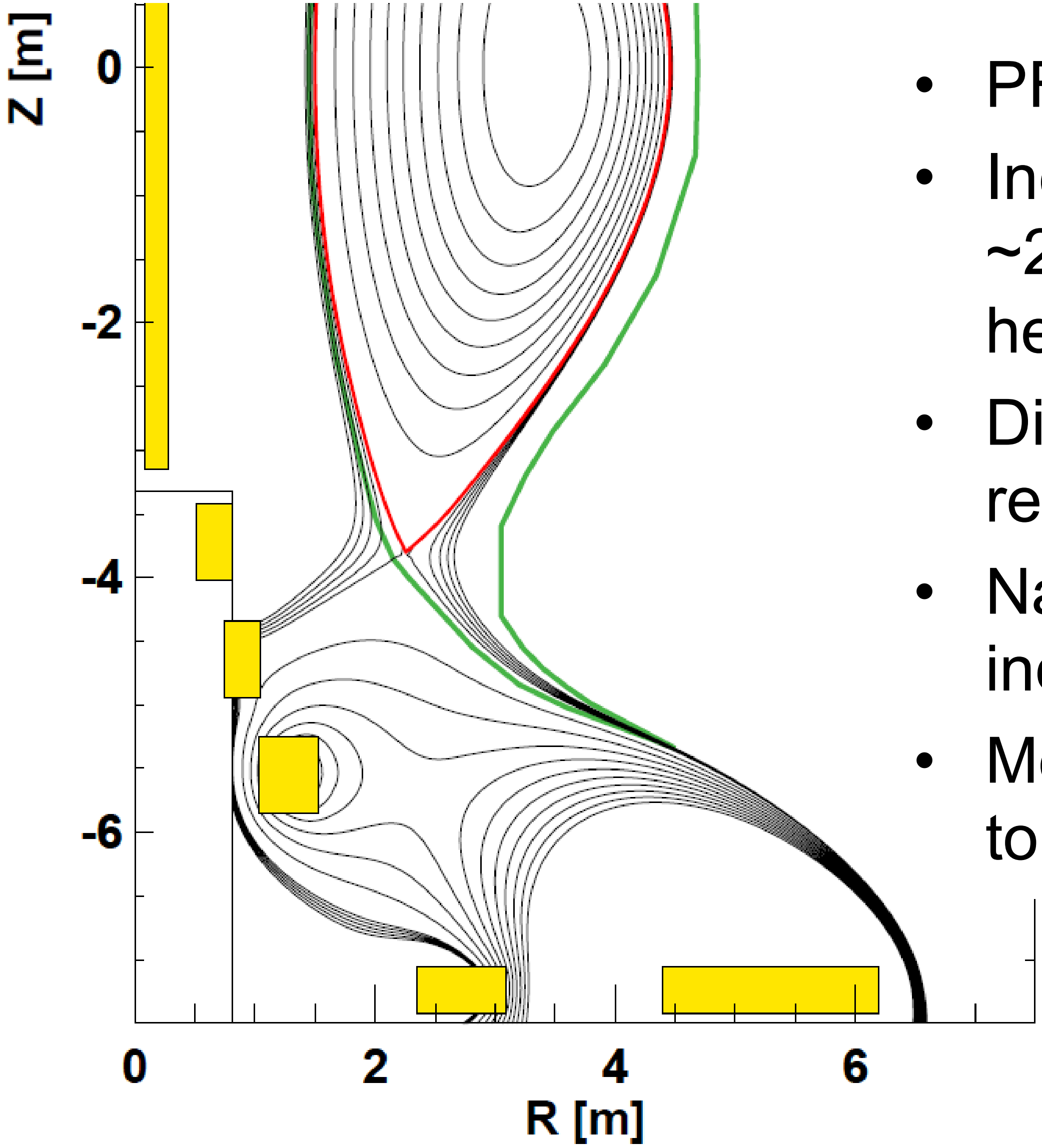
**All PF coils outside TF coil**

# Vertical port maintenance used for OB blanket and divertor modules via separate cryostat for upper PFs



- Potential advantages of this low-A configuration:
  - Reduced part count + no / small inboard breeding → simplified maintenance (?)
- Need to include some breeding at top + bottom
  - Similar to Cu ST-FNSF
- 2016 - will also study LM/Li wall and divertor compatibility with this HTS configuration

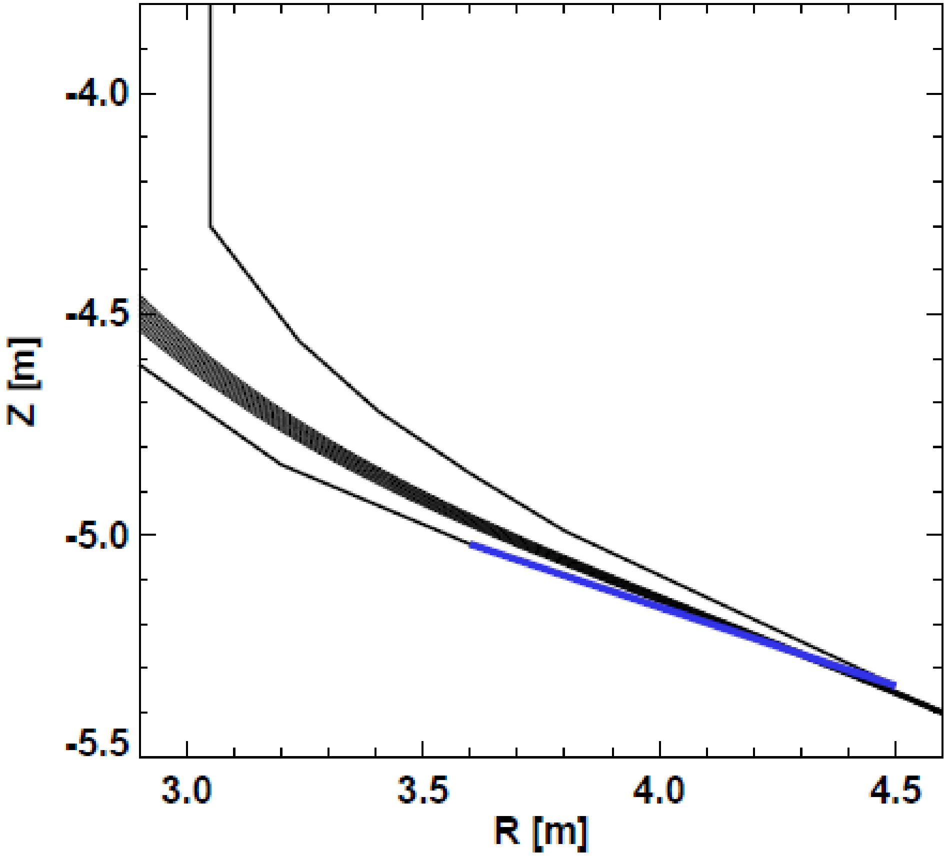
# Assessing long-leg / deep-V slot divertor



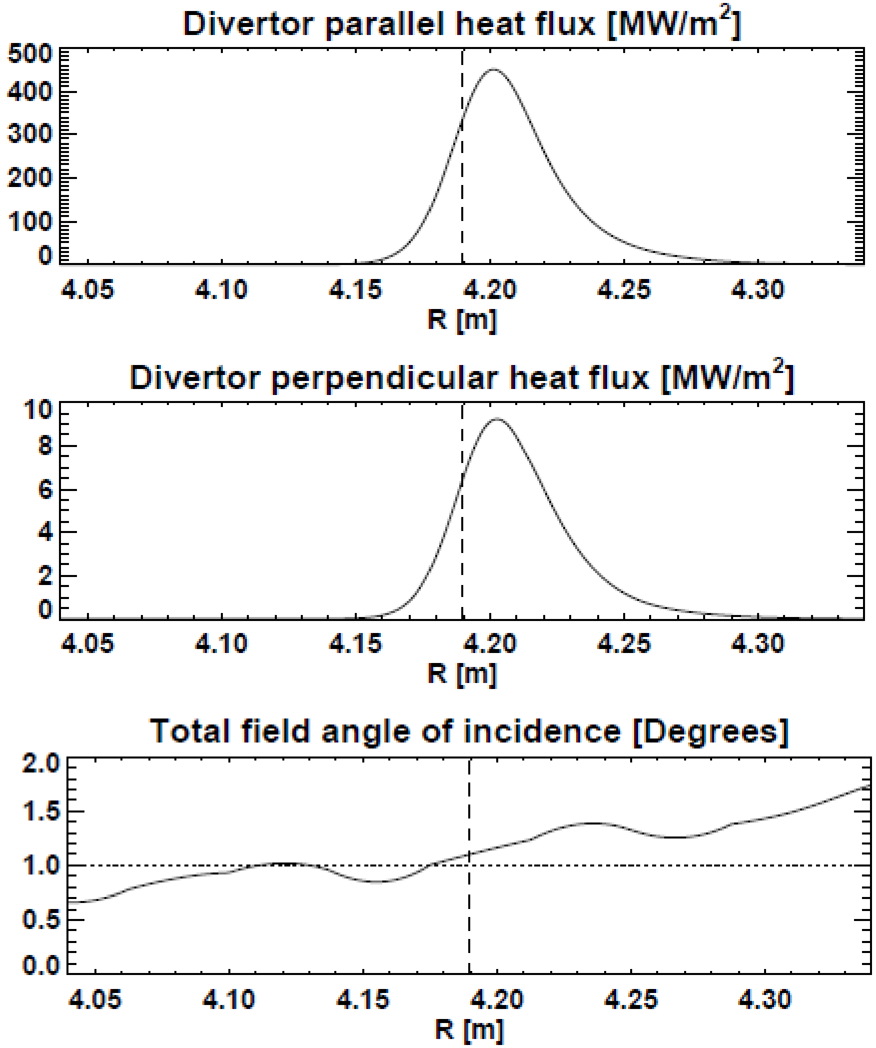
- PF coils outside TF
- Increase strike-point radius ~2× to reduce $q_{||}$ and peak heat flux
- Divertor PFCs in region of reduced neutron flux
- Narrow divertor aperture for increased TBR
- More space for breeding at top/bottom of device

# Long-leg / Super-X aids heat flux reduction

**A=2 HTS TF FNSF/Pilot**



$\lambda_q$ ~ 1mm, assume S ≈ $\lambda_q$ (closed divertor) (T. Eich NF 2013)


Divertor parallel heat flux [MW/m$^2$]

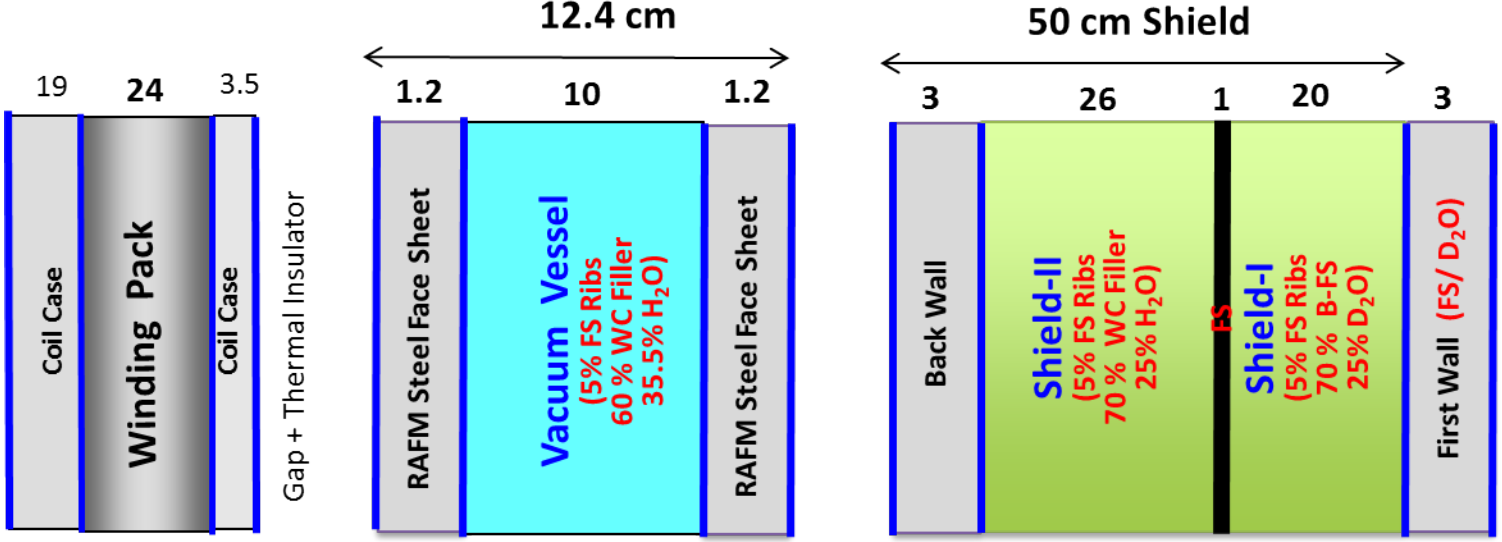
Divertor perpendicular heat flux [MW/m$^2$]

Total field angle of incidence [Degrees]

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

# A=2 HTS ST Shielding Assessment

- **Focus on inboard (IB) shield** - main functions are:
  - Protect IB magnet for machine lifetime (3.1 FPY)
  - Enhance OB breeding by reflecting neutrons to OB
  - Generate low decay heat to control temperature response during accident → avoid using WC filler near FW.
- Two-layer IB shield presents best option:



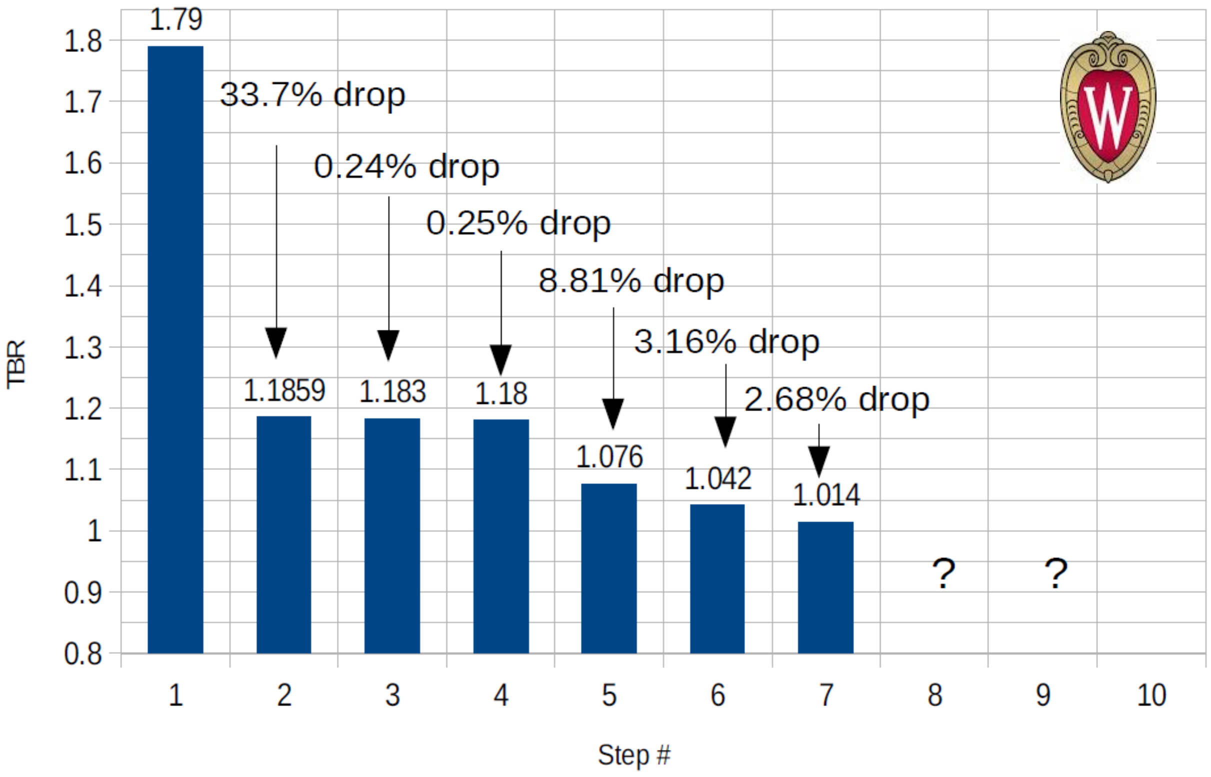
- 3-D analysis confirms radiation damage at IB magnet is near/below limits:

| | |
|---|---|
| – Peak fast n fluence to HTS ($E_n$ > 0.1 MeV) | $4.3 \times 10^{18}$ n / cm$^2$ |
| – Peak nuclear heating @ WP | 1.7 mW / cm$^3$ |
| – Peak dose to electrical insulator | $4 \times 10^9$ rads |
| – Total nuclear heating in IB magnet | 8.7 kW |

# TBR vs. blanket internal component assumption being evaluated step-by-step

**Steps:**

1. **1-D infinite Cylinder: 100% LiPb breeder with 90% enriched Li**
2. **$Li_{17}Pb_{83}$ confined to OB blanket region and blanket behind divertor**
3. **2 cm assembly gap between blanket modules**
4. **FS structure and FCI added to homogeneous mixture of blanket at top/bottom ends and behind divertor only**
5. **Materials assigned to 4 cm thick OB FW**
6. **Materials assigned to side, bottom/top, and back walls of blanket**
7. **IB and OB cooling channels**



**To be added:**

8. **SiC FCI**
9. **W Stabilizing shell**
10. **Penetrations**

**Expect final TBR ≈ 0.95-1 – Options to increase:**

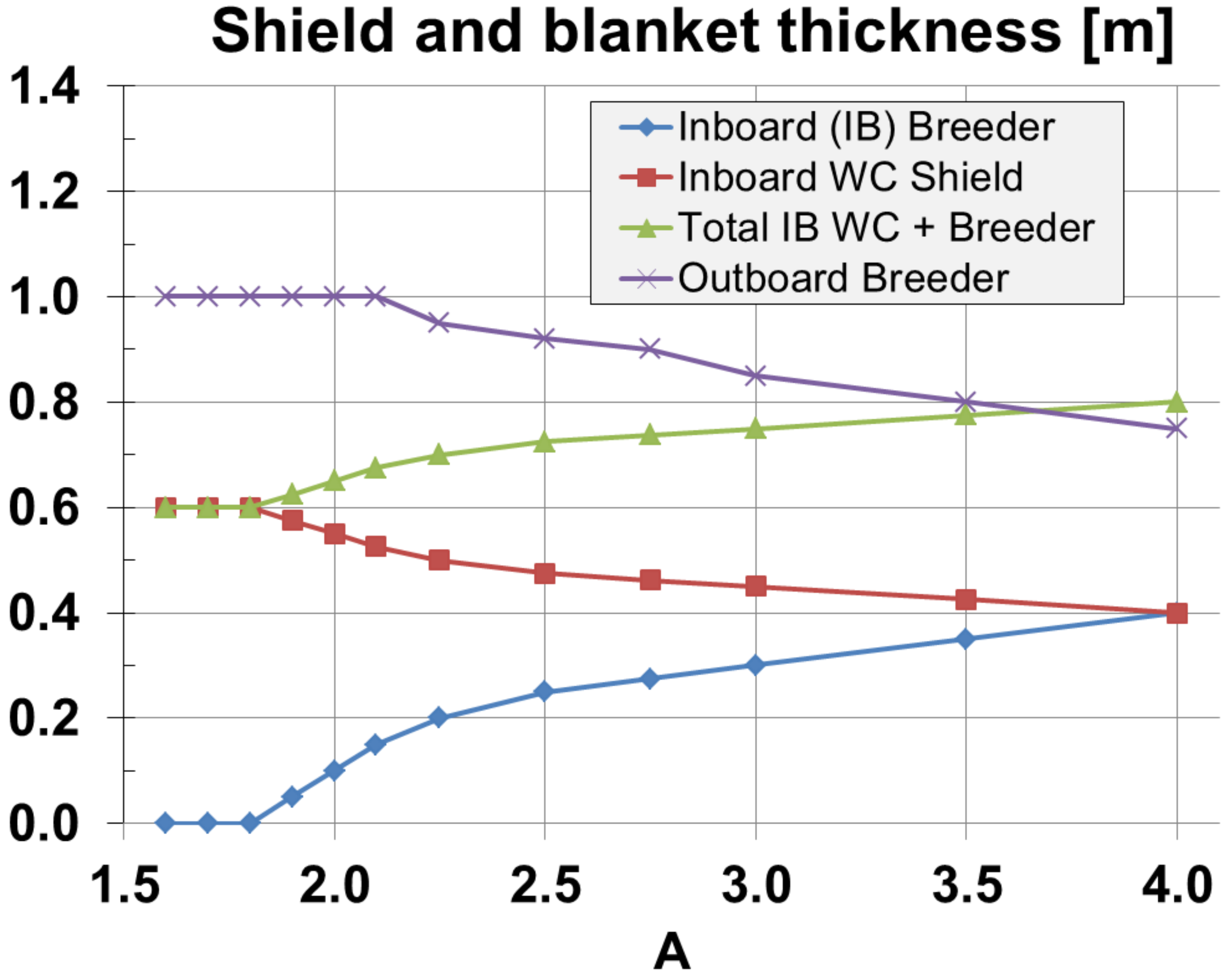
- **Thin inboard breeding region (assessing now)**
- **Reduce aspect ratio (reduces $Q_{eng}$, no CS)**

# Summary

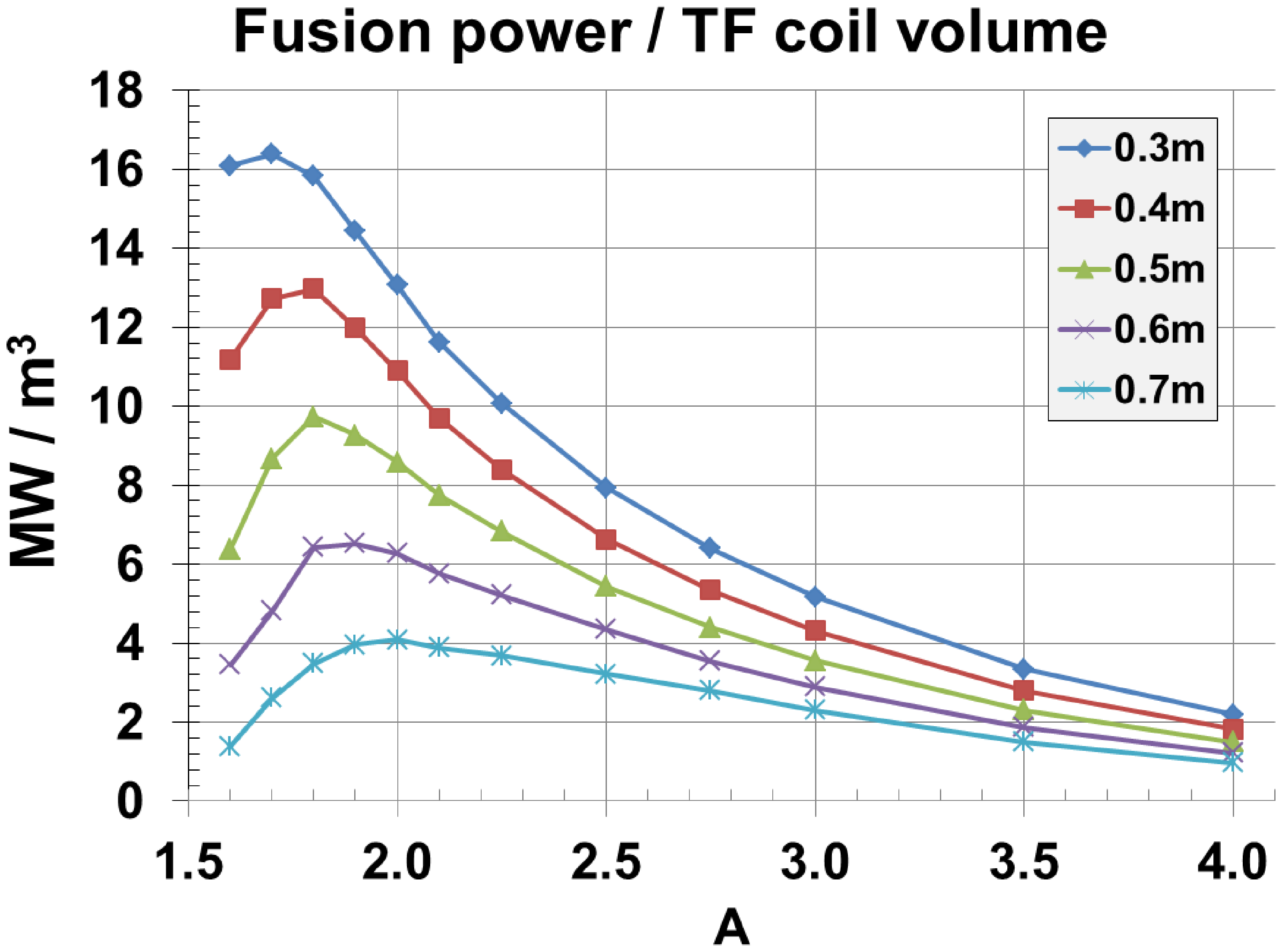
- $A \approx 2$ maximizes fusion performance at fixed $R_0$, $P_{aux/CD}$, normalized density for thin shield ($\Delta/R_0 < 20\%$)
  - A~2 likely requires thin (10cm?) blanket to achieve TBR~1
    - Note that A~1.8, R=1.7m ST-FNSF projects to TBR~1
  - $A \geq 2$ provides space for OH solenoid for $I_P$ start-up
  - $A \geq 2.7$-3 could provide full OH ramp-up
- High normalized confinement ($H_{98} \sim 1.5$-2) needed to achieve $Q_{eng} > 1$ for all "small" R=2.5-3m devices
- Performance/lifetime very sensitive to shield thickness
- 0.5-1MeV NNBI well matched to this device size
- HFS launch LHCD possible for (far)-off-axis CD
  - $A \approx 2$ with $B_T = \sim 4T \rightarrow 8T$ on HFS

# Backup

# Breeding blanket thickness assumptions



# Lower-A maximizes TF magnet utilization



# A = 2.5-3 maximizes blanket utilization