Impact of physics and technology innovations on compact tokamak fusion pilot plants

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Possible missions for fusion 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: ≥ 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
   - $Q_{eng} >> 1$, $P_{electric}$ = 0.5-1 GWe

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This talk: assess possible innovations to achieve these missions in a single and more compact tokamak device

Outline

- Scalings for electricity gain
- Scalings for DT fusion gain
- Possible innovations for higher gain
- Example low-A pilot plant concept
- Summary

Electricity gain Q_{eng} determined primarily by engineering efficiencies and fusion gain

$$Q_{eng} \equiv \frac{\text{Electricity produced}}{\text{Electricity consumed}} = \frac{\eta_{th}(M_n P_n + P_\alpha + P_{aux} + P_{pump})}{\frac{P_{aux}}{\eta_{aux}} + P_{pump} + P_{sub} + P_{coils} + P_{control}}$$

$$Q_{eng} = \boxed{\eta_{th}\,\eta_{aux}\,Q\times}\frac{(4M_n + 1 + 5/Q + 5P_{pump}/P_{fus})}{5(1+\eta_{aux} Q P_{extra}/P_{fus})}$$

η_{th} ≡ thermal power conversion efficiency

η_{aux} ≡ injected power wall plug efficiency

Q ≡ P_{fus} / P_{aux} = fusion power / auxiliary power

Parameter Assumptions:

- $M_n = 1.1$, $P_{pump} = 0.03 \times P_{th}$
- $P_{sub} + P_{control} = 0.04 \times P_{th}$
- $\eta_{aux} = 0.3$
- $\eta_{CD} = I_{CD} R_0 n_e / P_{CD} \approx 0.3 \times 10^{20}$ A/W/m^2

For more details see J. Menard, et al., Nucl. Fusion 51 (2011) 103014

Fusion gain $Q_{DT} \propto H^{2\rightarrow 5}$ from low → high gain

$$\text{Fusion power density} \equiv \Gamma_{DT} = n_D n_T \langle \sigma v \rangle_{DT} E_{DT} \propto p^2$$

$$P_{fusion} \propto (P\,\tau_E)^2/V$$

$$\tau_E \propto H\, I_P^{\alpha_I} B_T^{\alpha_B} n_e^{\alpha_n} P^{-\alpha_P} R^{\alpha_R} \kappa^{\alpha_\kappa} \epsilon^{\alpha_\epsilon}$$

$$P = P_{aux}(1 + \lambda_{DT} Q_{DT}) \qquad Q_{DT} \equiv P_{fusion}/P_{aux} \quad \lambda_{DT} = 0.2$$

$$Q^*_{DT} \equiv Q_{DT}/(1 + \lambda_{DT} Q_{DT})^{2(1-\alpha_P)}$$

$$\propto H^2 I_P^{2\alpha_I} B_T^{2\alpha_B} n_e^{2\alpha_n} P_{aux}^{1-2\alpha_P} R^{2\alpha_R - 3} \kappa^{2\alpha_\kappa - 1} \epsilon^{2\alpha_\epsilon - 2}$$

Fix current, field, density, geometry, auxiliary power, $\alpha_P = 0.7$:

$Q_{DT} \le 1 \rightarrow Q_{DT} \approx Q^*_{DT} \propto H^2$ $\qquad$ $Q_{DT} >> 1 \rightarrow Q_{DT} \propto {Q^*_{DT}}^{2.5} \propto H^5$

Stability limits and other operating constraints

- ***n* = 0 stability limit:** elongation $\kappa \leq \kappa_{max}(l_i, \varepsilon, \beta_P, \text{wall})$
- ***n* > 0 stability limits:**
  - **Pressure:** $\beta_N \equiv \beta_T\ aB_{T0} / I_P$ [%mT/MA] $< \beta_{N\text{-}max}(\varepsilon, \kappa, \delta, \text{profiles})$
  - **Current:** $q^* \equiv \pi a^2 B_{T0}(1+\kappa^2)/\mu_0 R_0 I_P > 2$
- **Density limit:** $n_e < n_{Greenwald} = 10^{20}\text{m}^{-3}\ I_P\,[\text{MA}] / \pi a[\text{m}]^2$
- **Steady-state:** $I_{Plasma} = I_{bootstrap\ (BS)} + I_{external\ current\ drive\ (CD)}$
  - $f_{BS} = C_{BS}\ \varepsilon^{1/2}\ \beta_P \qquad \beta_P\ \beta_T \propto \beta_N^2\ G(\kappa) \qquad G(\kappa) \propto \kappa$ or $(1+\kappa^2)/2\ \sim \kappa^2$
  - Fraction of external current drive = $f_{CD} = 1 - f_{BS}$

Gain vs. physics and engineering constraints

- For steady-state, current limit is weaker constraint than high f_{BS} → no *q** dependence → relevant variables are ***β_N* / *f_{BS}*** and ***f_{gw}***:

Exponent	98y2	Petty-08
C_β	2.68	2.14
C_B	2.98	2.74
C_{gw}	0.82	0.64
C_P	-0.38	0.06
C_R	1.98	2.04
C_κ	5.92	5.04
C_ε	1.54	1.61

C. Petty, et al., Phys. Plasmas 15 (2008) 080501

$$Q^*_{DT} \propto H^2(\beta_N/f_{BS})^{c_\beta} B_T^{c_B} f_{gw}^{c_{gw}} P_{aux}^{c_P} R^{c_R} \kappa^{c_\kappa} \epsilon^{c_\epsilon}$$

Choose electrostatic gyro-Bohm Petty-08 with no β degradation (JET, DIII-D, NSTX)

$$Q^*_{DT} \propto R^2 \overbrace{H^2(1-f_{CD})^{-2} f_{gw}^{0.7}}^{\text{Optimize: confinement, current drive vs density}} \overbrace{B_T^3 \kappa^{3-5} \beta_N^2 \epsilon^{1.6}}^{\text{aspect ratio}}$$

Potential Innovation Areas for Compact Pilot

- Aspect Ratio – Reduced A → higher β_N and κ
- Magnet Technology – HTS for higher B_T, $J_{\text{winding-pack}}$
- Confinement – Optimize edge transport barrier
- Current Drive – Negative NBI, new RF techniques
- Divertors – Long-leg, liquid metals
- Blankets – Liquid metal, high efficiency

Potential Innovation Areas for Compact Pilot

- Aspect Ratio
- Magnet Technology
- Confinement
- Current Drive
- Divertors
- Blankets

Optimize this combination first

Assess R_0 = 3m → smallest size for $Q_{eng} > 1$, high fluence

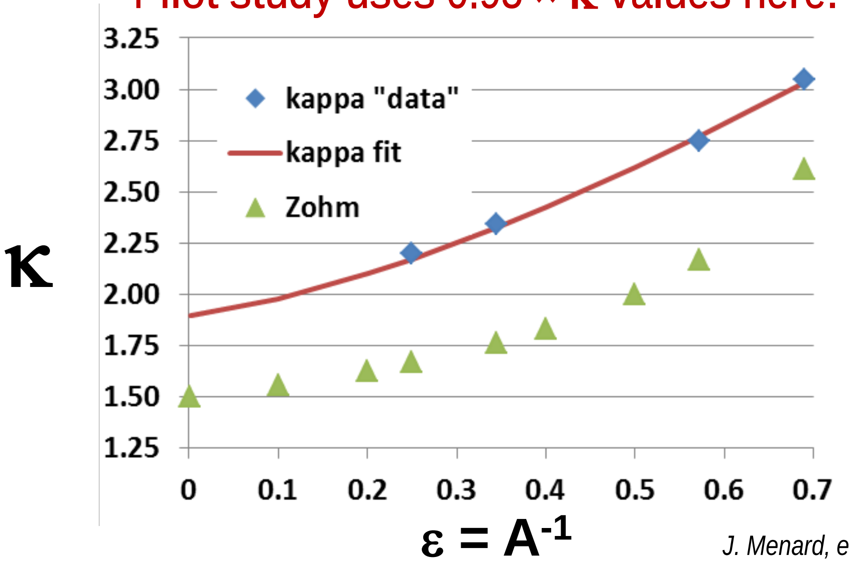
Optimization Approach

- Fix plasma major radius, heating power (P_{NNBI}=50MW)
  - R_0 = 3m – smallest size for Q_{eng} > 1 + high fluence ~ $6MWy/m^2$
- Vary aspect ratio from A = 1.6 to 4
- Include blanket/shield model to achieve TBR~1 for all A
- Apply magnet (see backup) and plasma constraints
  - HTS strain: 0.3%, $\beta_N(\varepsilon)$ n=1 no-wall, $\kappa(\varepsilon)$: 0.95×limit, f_{GW} = 0.8
  - Vary HTS current density, peak field
  - Also scan inboard shielding thickness
- Compute Q_{DT}, Q_{eng}, and required H_{98} (*unconstrained*)

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

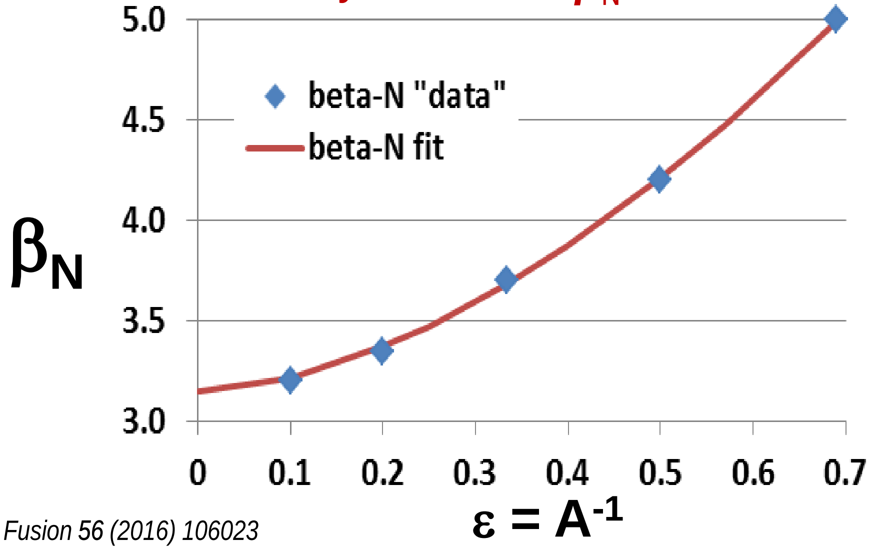
- **NSTX data (+ST-FNSF models) at low-A, DIII-D, ARIES-AT for high A**
  - $\kappa \rightarrow 1.9$ for $A \rightarrow \infty$

Pilot study uses $0.95 \times \kappa$ values here:



- **Profile-optimized no-wall stability limit at $f_{BS} \approx 50\%$ *(Menard PoP 2004)***
  - $\beta_N \rightarrow 3.1$ for $A \rightarrow \infty$

Pilot study uses $1.0 \times \beta_N$ values here:

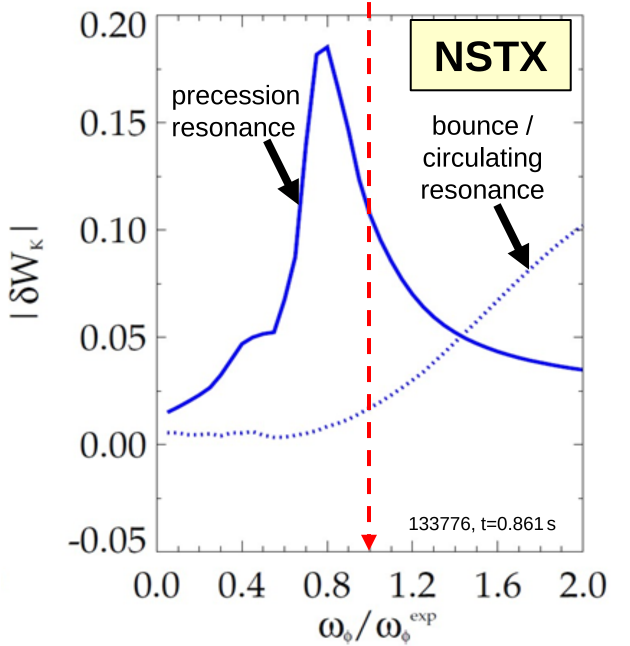


J. Menard, et al., Nucl. Fusion 56 (2016) 106023
H. Zohm, et al., Nucl. Fusion 53 (2013) 073019

Rotation profile control may provide stable operation near and above n=1 no-wall limit

- NSTX, DIII-D show resonant damping can stabilize resistive wall mode (RWM)

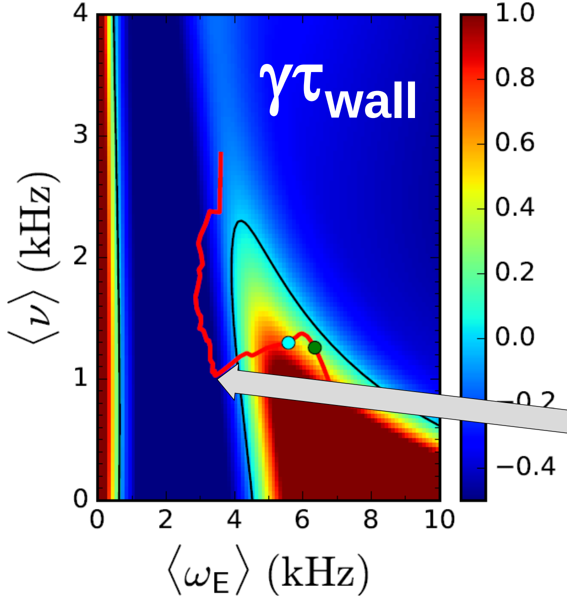
Expt. rotation for RWM-stable plasma

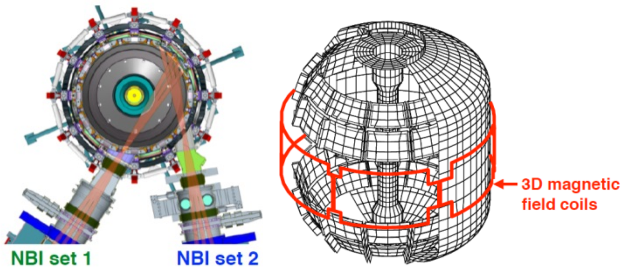


S. Sabbagh (Columbia University) – APS-DPP 2014

Reduced drift-kinetic MHD RWM stability map

YI2.00005 - J. Berkery (Columbia University)



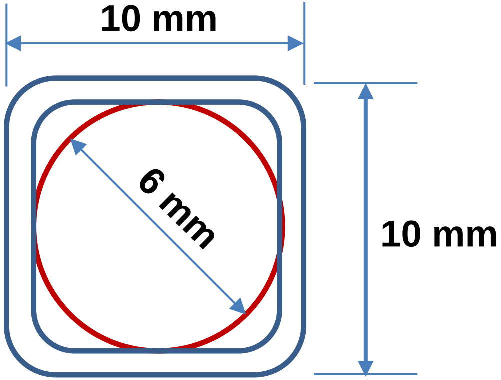


Use neutral beams, rotation damping from 3D fields, other actuators to control rotation for RWM stability

BI2.00005 - I. Goumiri (Princeton University)

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 at 19T

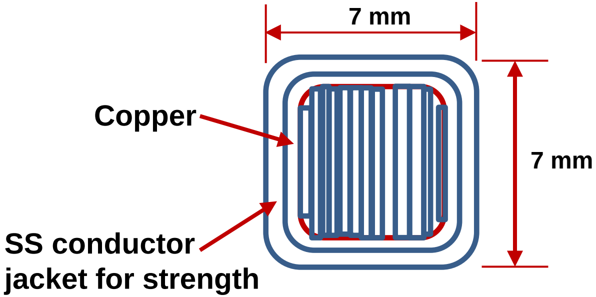


7 kA CORC (4.2K, 19 T) cable

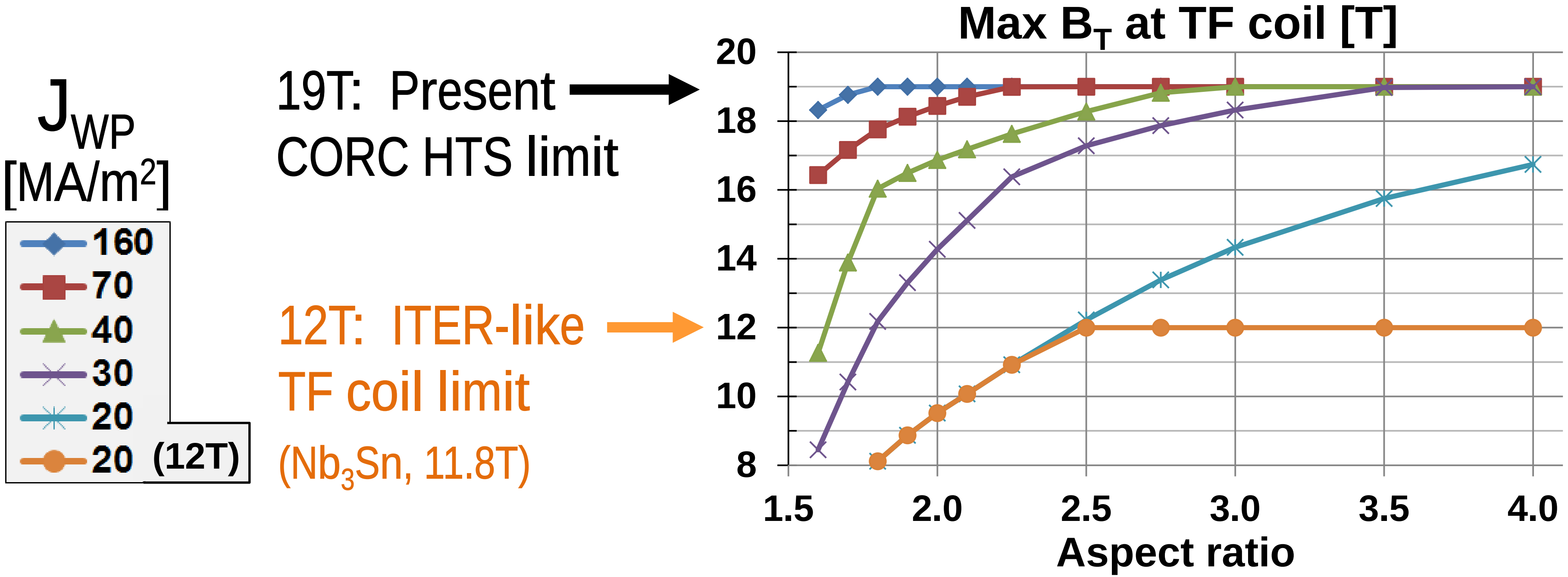
Base cable: 50 tapes YBCO Tapes with 38 μm substrate

(Van Der Laan, HTS4Fusion, 2015)

Higher J_{cable} HTS cable concepts under development: → **Base Conductor He Gas Cooled 8kA, J_{WP} ~ 160MA/m^2**



At lower A, high TF winding-pack current density enables access to maximum allowed B_T at coil



19T: Present CORC HTS limit

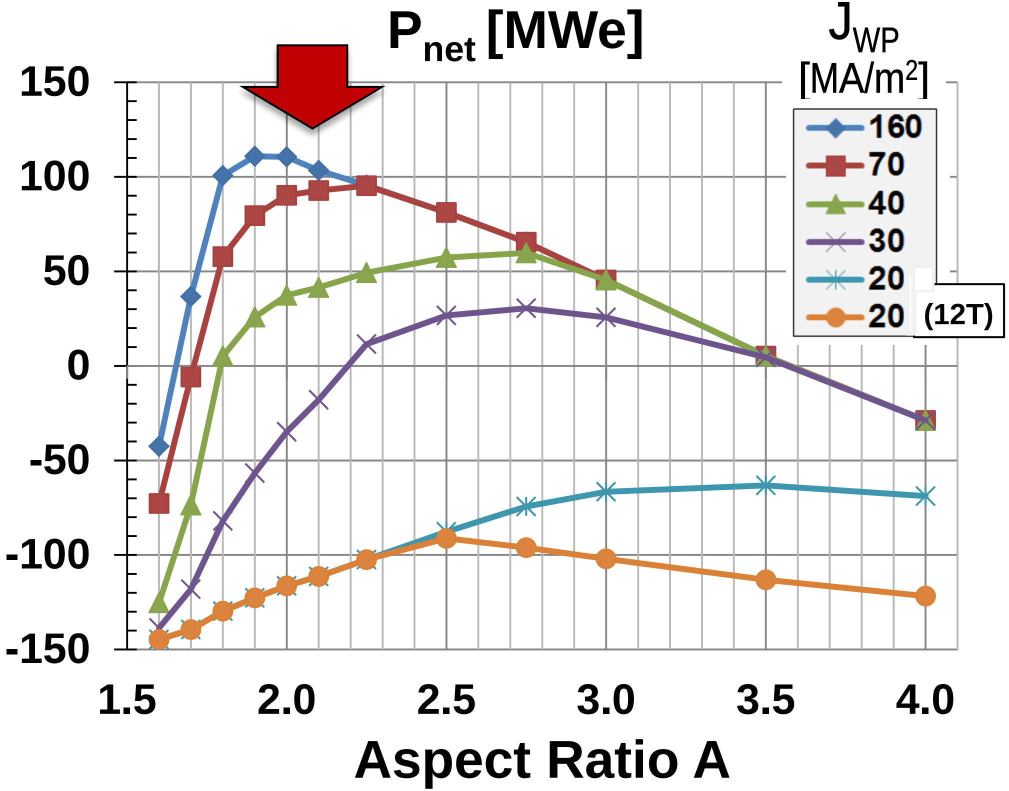
12T: ITER-like TF coil limit (Nb_3Sn, 11.8T)

- Coil structure sized to maintain ≤ 0.3% strain on winding pack
- Effective inboard tungsten carbide (WC) neutron shield thickness = 60cm

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

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

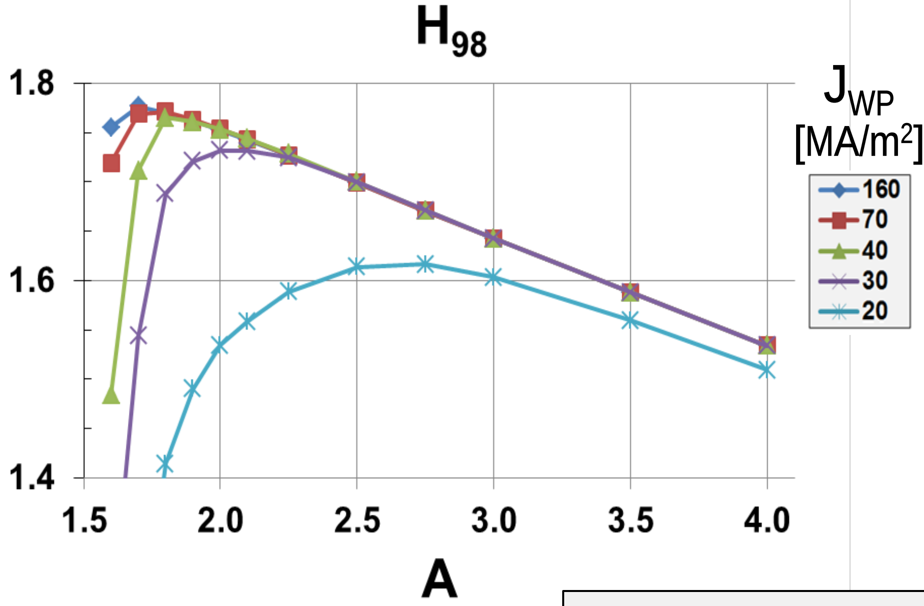
A ~ 2 attractive at high J_{WP}

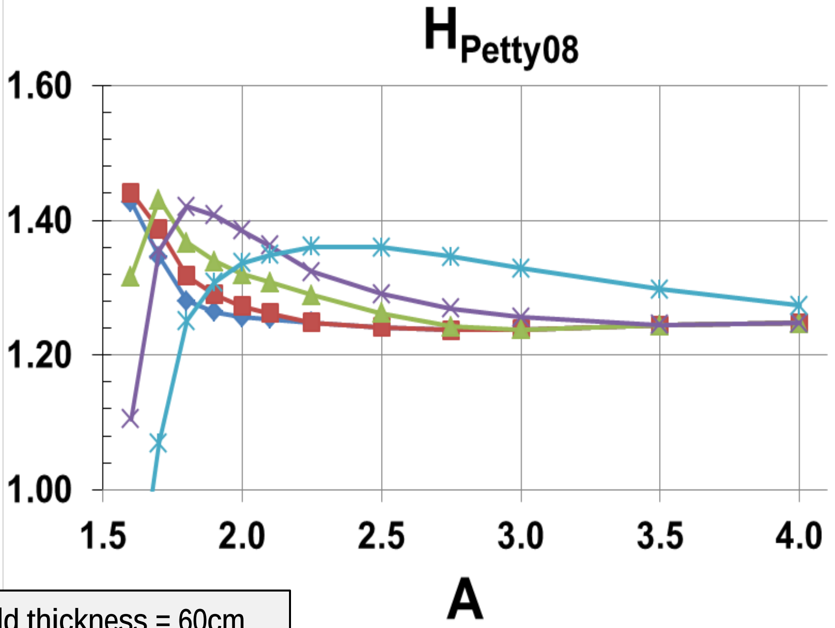


R=3m Pilot Plants require elevated H values

H_{98y2} = 1.5-1.8

$H_{Petty\text{-}08}$ = 1.25-1.4



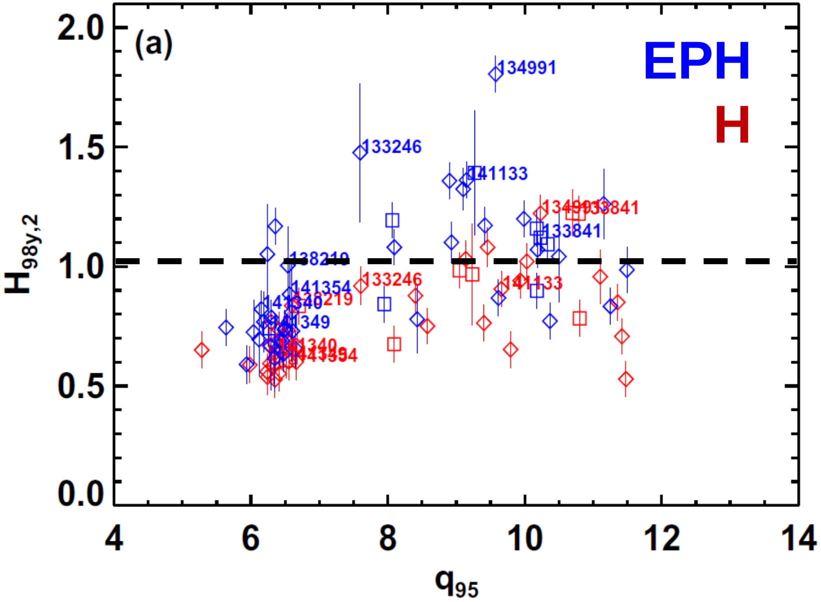


Effective inboard WC n-shield thickness = 60cm

Increased edge rotation shear, wider and higher pedestal can increase normalized confinement ~1.5×

NSTX: Enhanced Pedestal H-mode

Higher edge v_ϕ shear (+Li) → H_{98}=1.3-1.8

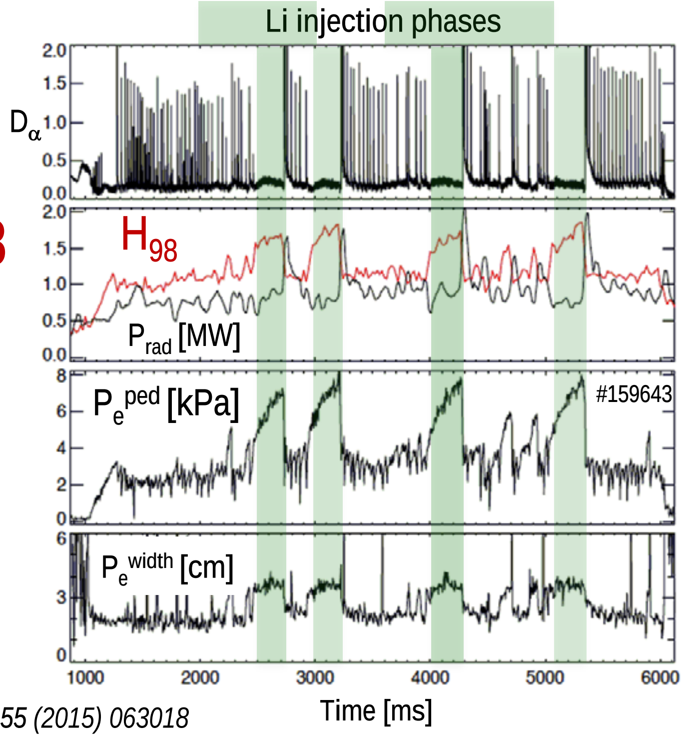


S. Gerhardt, et al., Nucl. Fusion 54 (2014) 083021

H_{98}→1.5-1.8

2× wider,
2-3× higher
pedestal
(BCM: Bursty chirping mode)

Lithium injection on DIII-D

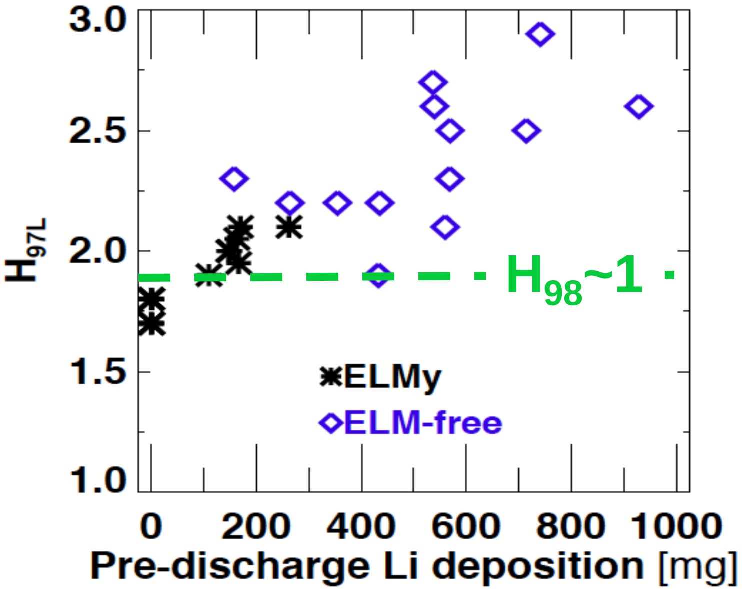


T. Osborne, et al., Nucl. Fusion 55 (2015) 063018

Li (solid and liquid) PFCs can increase confinement

NSTX (wider → higher pedestals)

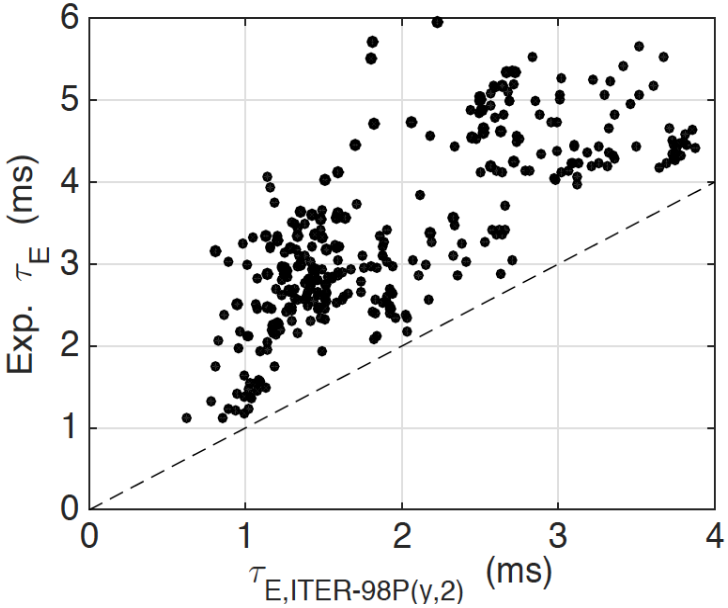
H_{98y2} increased from 0.8 → 1.4



D.P. Boyle, et al., J. Nucl. Mater. 438 (2013) S979

LTX (flatter → higher T profiles)

2-4x improvement over ITER98P(y,2)



J.C. Schmitt, et al., Phys. Plasmas 22 (2015) 056112

High-efficiency current drive options exist but need additional R&D and demonstration (1)

Current drive (CD) efficiency target for Pilot Plant:

$\eta_{CD} \equiv n_e I_{CD} R / P$

At least 2×10^{19} A/W/m^2

Prefer > 3×10^{19} A/W/m^2

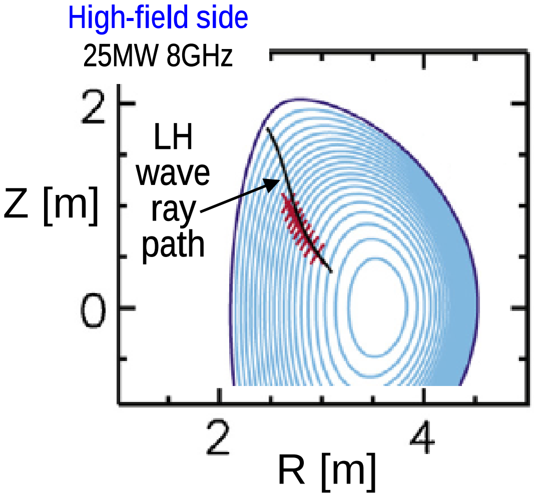
to keep $f_{BS} \leq \sim 80\%$

ARC (MIT)

Inboard Launch Lower Hybrid

$\eta_{CD} \sim 3.6\times10^{19}$ A/W/m^2

HFS LHCD needs expt test



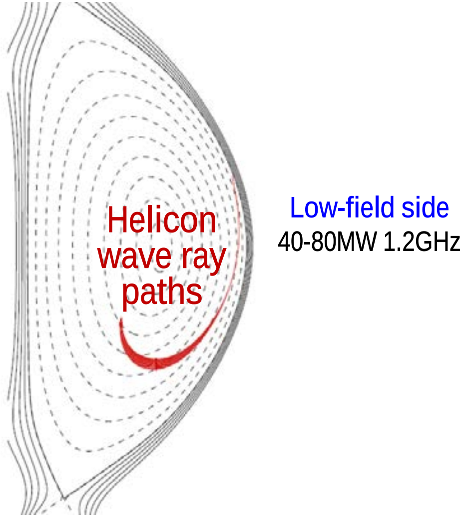
B. Sorbom, et al., *Fus. Eng. Design*, 100, (2015) 378

FNSF-AT (GA)

Helicon Wave

$\eta_{CD} \sim 2.1\times10^{19}$ A/W/m^2

Helicon tests: DIII-D, KSTAR



R. Prater, et al., *Nucl. Fusion* **54** (2014) 083024

High-efficiency current drive options exist but need additional R&D and demonstration (2)

HTS ST-FNSF/Pilot

Electron Bernstein Wave (EBW)

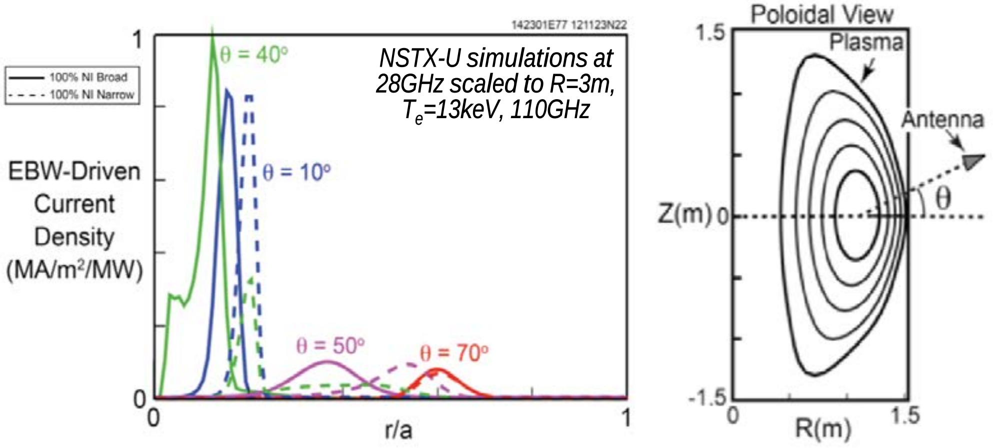
$\eta_{CD} \sim 3\times10^{19}$ A/W/m^2

Ongoing / future tests: QUEST / NSTX-U, MAST-U

Negative Neutral Beam Injection (NNBI)

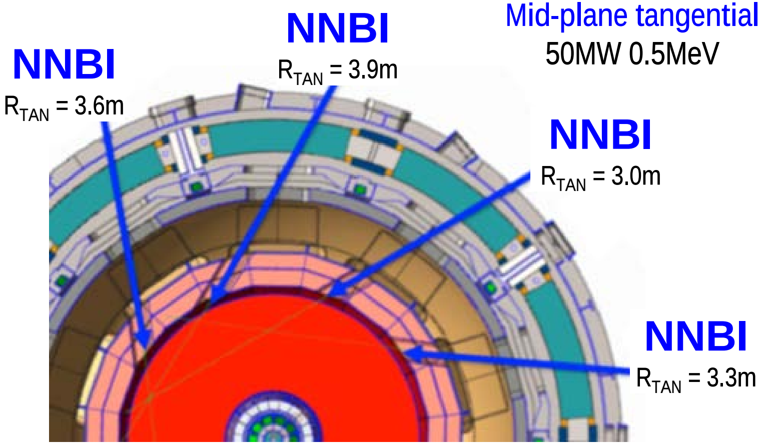
$\eta_{CD} \sim 3.5\times10^{19}$ A/W/m^2

Test on JT-60SA, leverage ITER NNBI



NSTX-U simulations at 28GHz scaled to R=3m, T_e=13keV, 110GHz

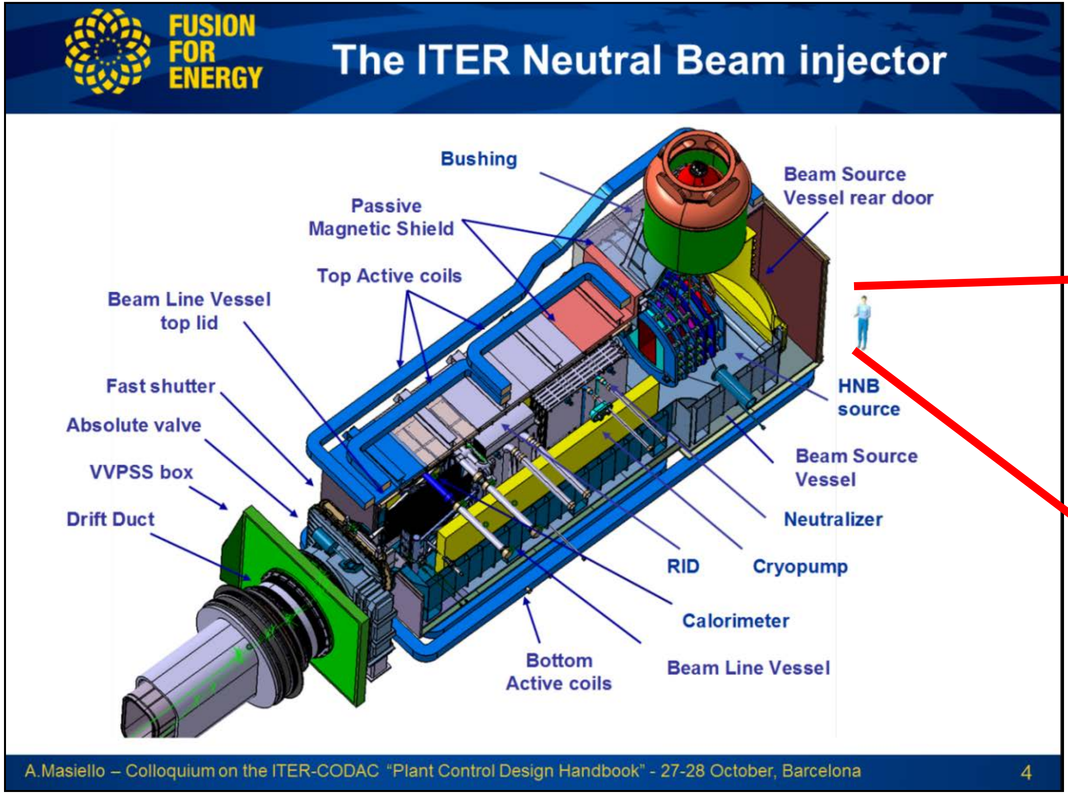
G. Taylor, et al., EPJ Web of Conferences 87, 02013 (2015)



Mid-plane tangential
50MW 0.5MeV

L. El-Guebaly, et al., Energies, 9 (2016) 632

Despite ITER's beam ions being "too light", Ron was impressed by the scale of ITER NBI

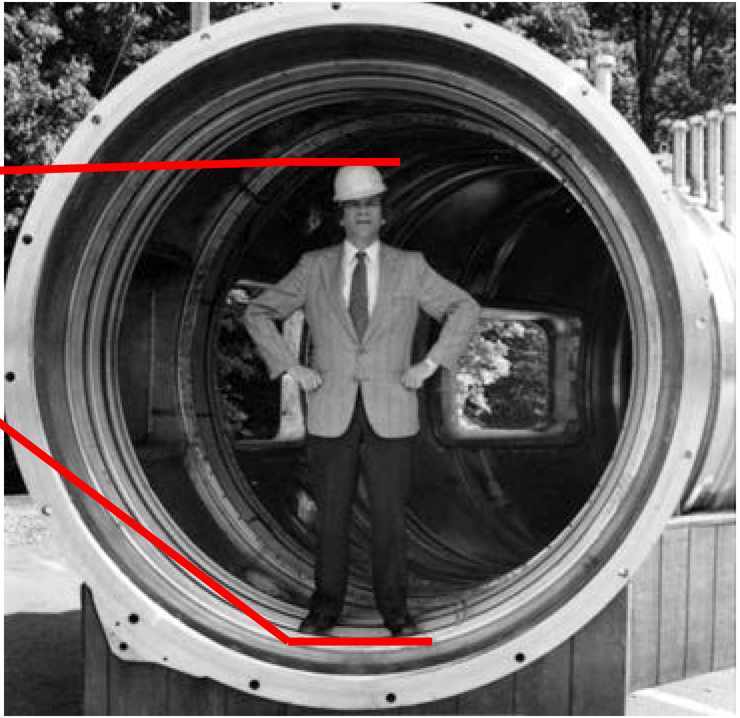


The ITER Neutral Beam injector

A.Masiello – Colloquium on the ITER-CODAC "Plant Control Design Handbook" - 27-28 October, Barcelona

Plasma neutralization (≈80% efficient), laser photo-detachment (≈95% efficient) could improve η_{aux} from 30 → 40-50%

Ron Davidson inside mock-up of TFTR vacuum vessel

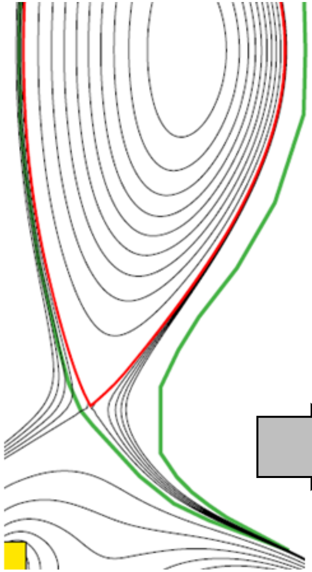


Potential Innovation Areas for Compact Pilot

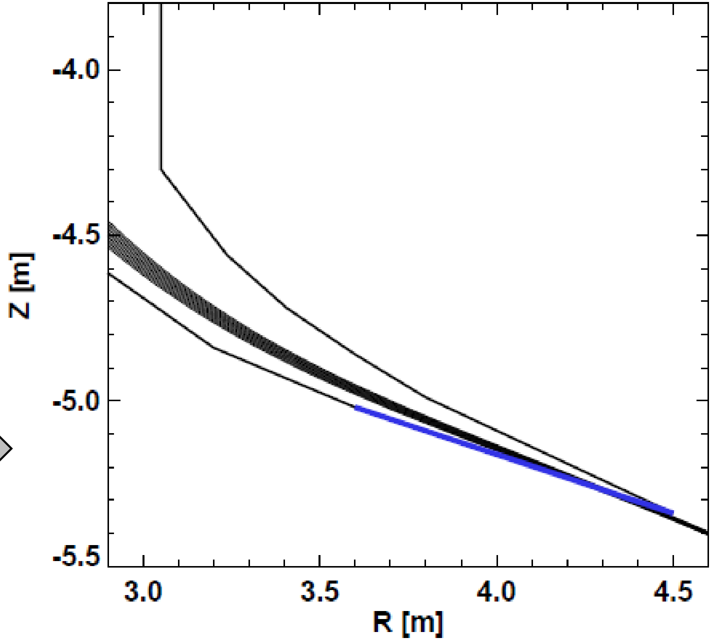
- Aspect Ratio
- Magnet Technology
- Confinement
- Current Drive
- Divertors
- Blankets

Long-leg / Super-X aids heat flux reduction

A=2 HTS TF FNSF/Pilot



$f_{rad}=0.8$, $f_{obd}=0.8$, $N_{div} = 2$



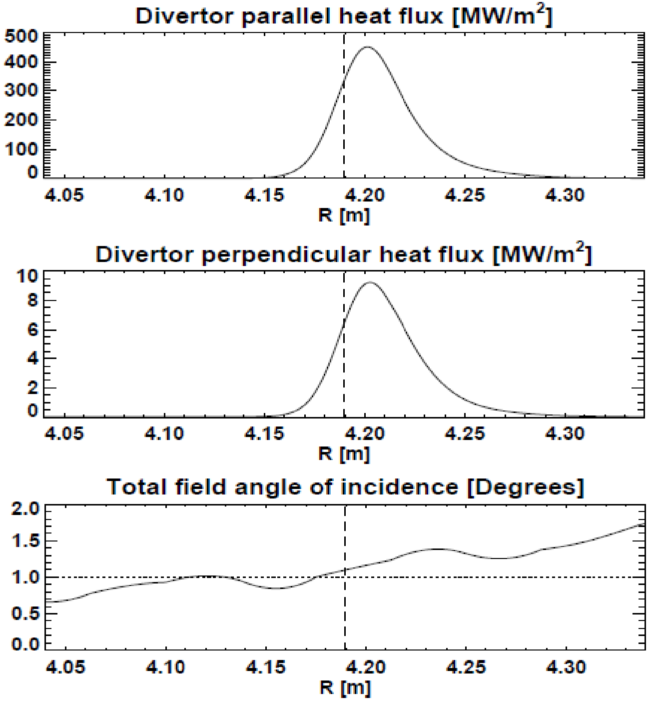
$\lambda_q \sim$ 1mm, **assume S ≈ λ_q** (closed divertor)

T. Eich, et al., Nucl. Fusion ***53*** *(2013) 093031*

Divertor parallel heat flux [MW/m^2]

Divertor perpendicular heat flux [MW/m^2]

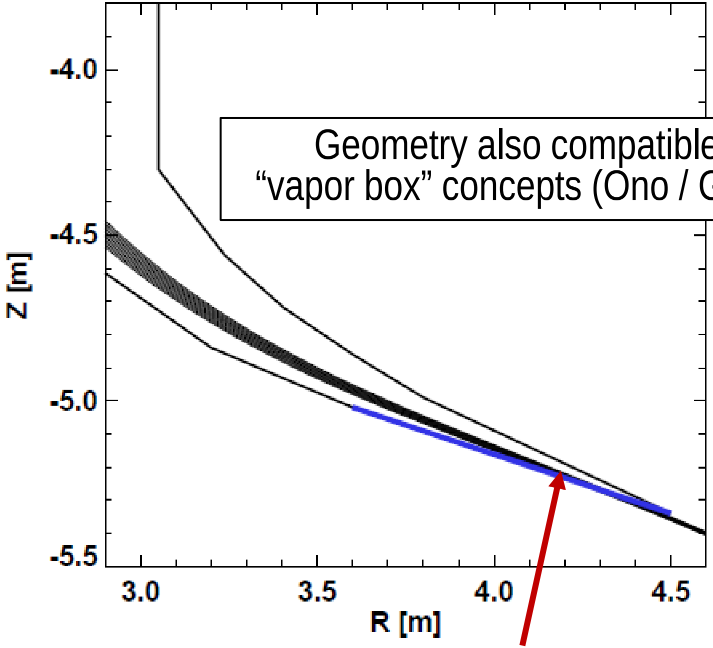
Total field angle of incidence [Degrees]



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

Another option: use fast-flowing Liquid Metal (LM) divertor for high heat-flux mitigation

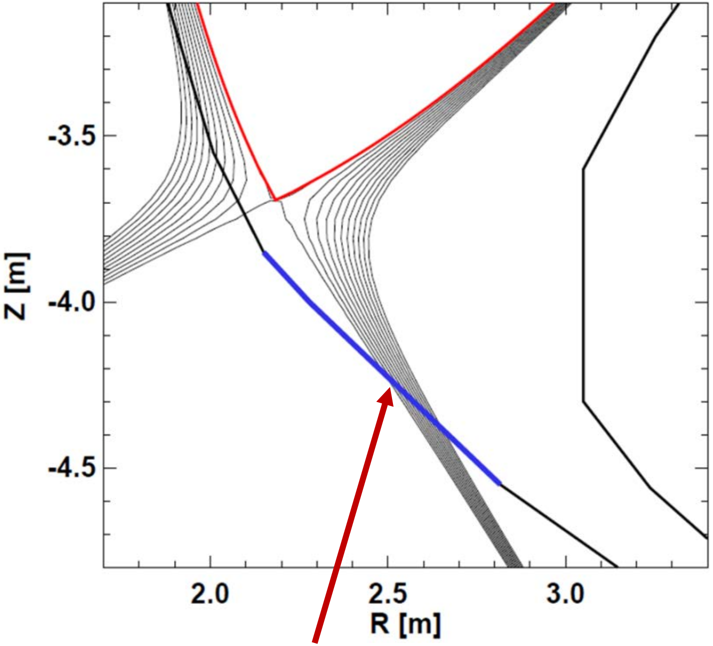
Long-leg / Super-X divertor



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

9 MW/m^2

Shorter leg LM divertor

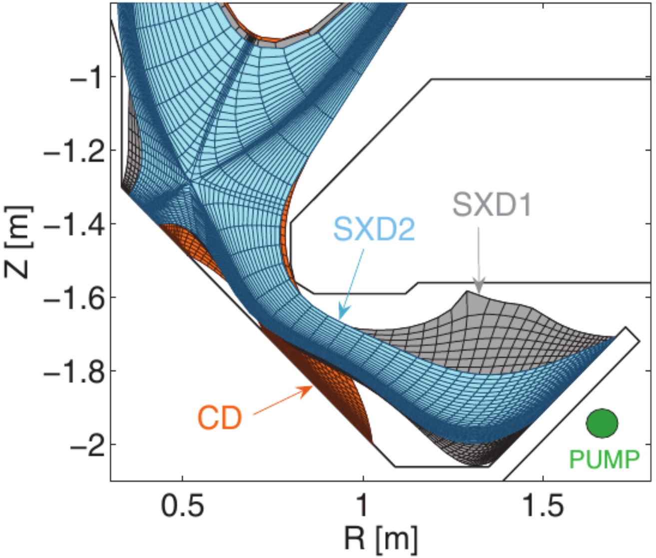


21 MW/m^2

Advanced divertors under active development

MAST-U will test range of divertors:

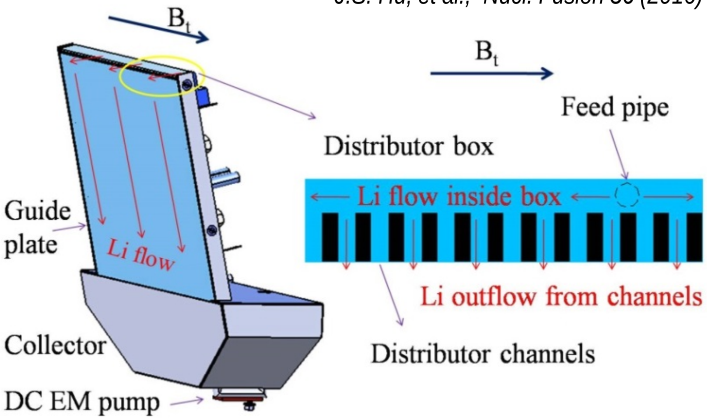
- Conventional, snowflake (not shown)
- Long-leg "Super-X" with variable flaring



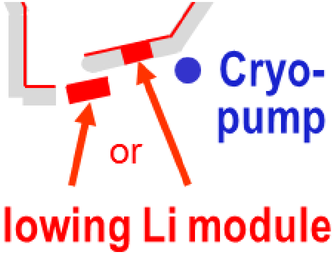
E. Havlickova, et al., Plasma Phys. Control. Fusion 56 (2014) 075008

EAST testing flowing liquid Li limiter:

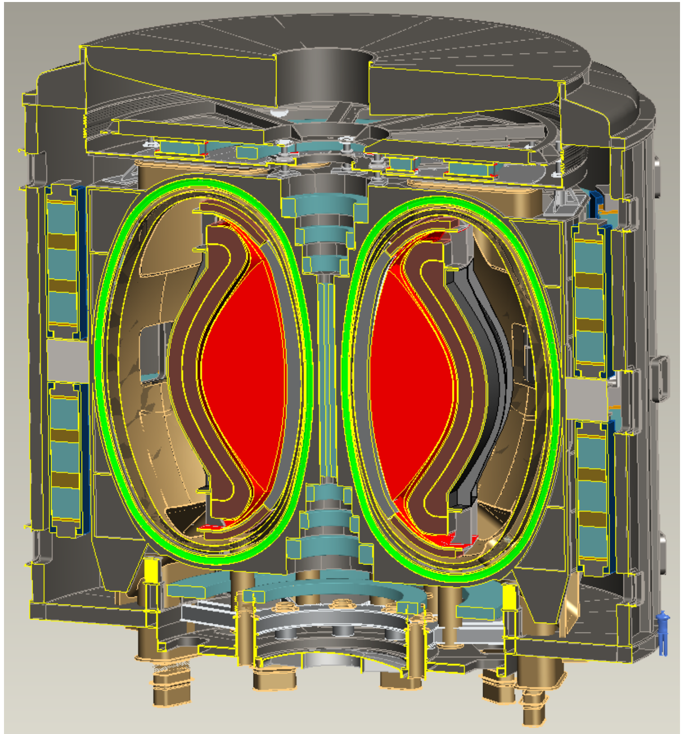
J.S. Hu, et al., Nucl. Fusion 56 (2016) 046011



EAST results will inform possible NSTX-U LM divertor



Example: A=2, R_0 = 3m HTS-TF Pilot Plant

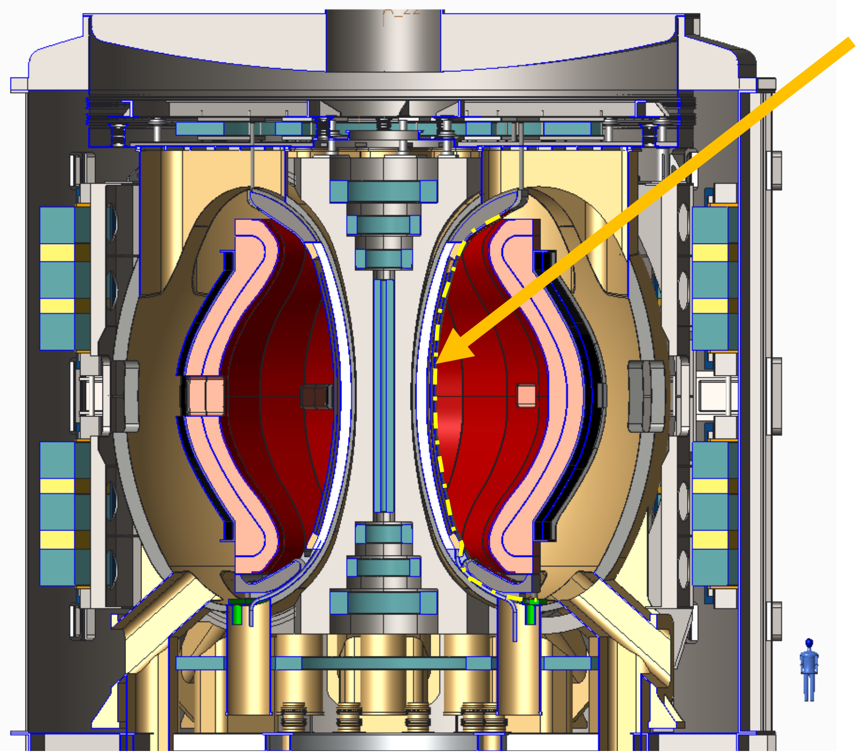


Cryostat volume ~ 1/3 of ITER

B_T = 4T, I_P = 12.5MA	P_{fusion} = 520 MW
κ = 2.5, δ = 0.55	P_{NBI} = 50 MW
β_N = 4.2, β_T = 9%	E_{NBI}=0.5MeV
H_{98} = 1.75, $H_{Petty\text{-}08}$ = 1.3	Q_{DT} = 10.4
f_{gw} = 0.80, f_{BS} = 0.76	
Startup I_P (OH)~2MA	Q_{eng} = 1.35
J_{WP} = 70MA/m^2	P_{net} = 73 MW
$B_{T\text{-}max}$ = 17.5T	$\langle W_n \rangle$ = 1.3 MW/m^2
No joints in TF	Peak n-flux = 2.4 MW/m^2
Vertical maintenance	Peak n-fluence: 7MWy/m^2

J. Menard, et al., Nucl. Fusion 56 (2016) 106023

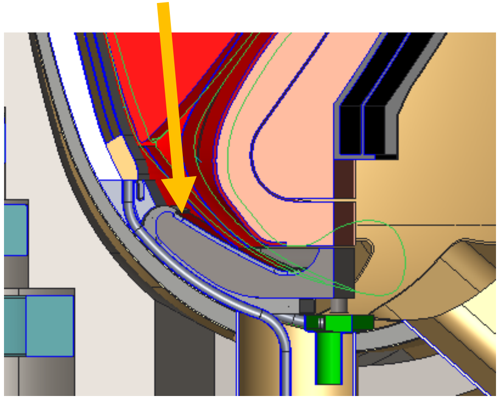
Low-A HTS design with Li flow on divertor and inboard surfaces



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

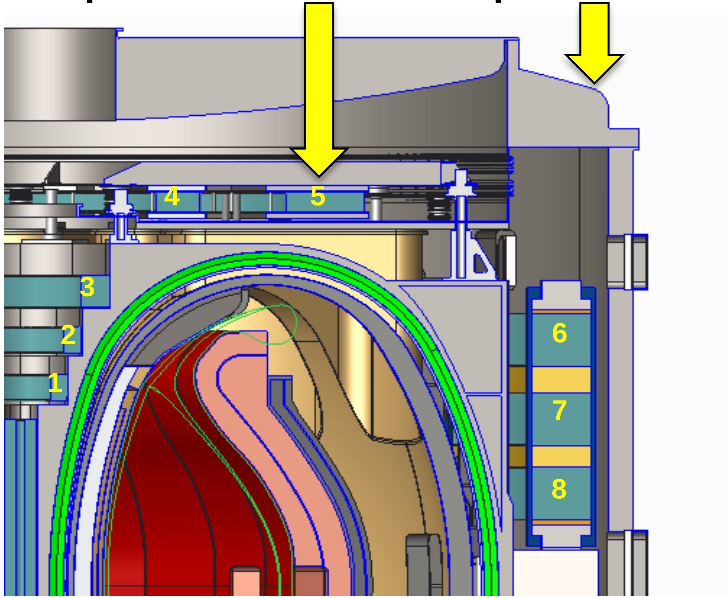


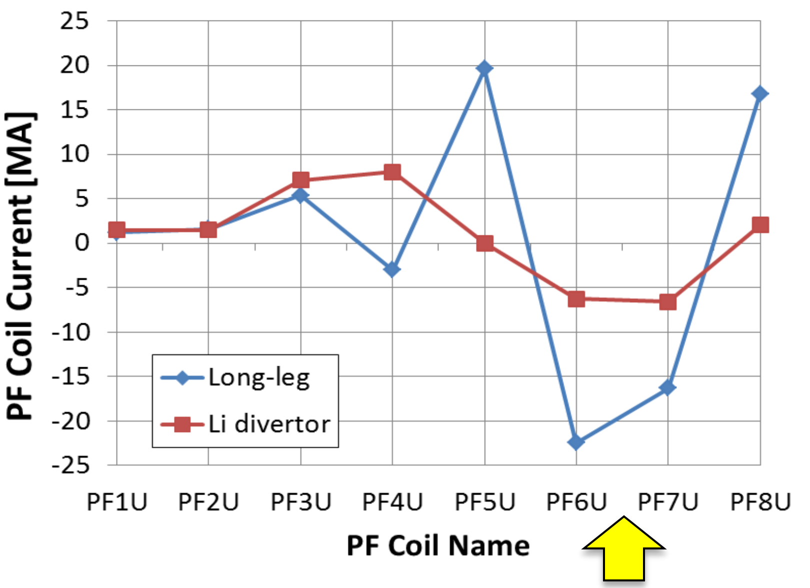
T. Brown, et al., submitted to Nuclear Fusion

LM thickness = 5-10 mm, flow speed ~5-10 m/s

Benefits of shorter-leg liquid metal divertor:

- No top PF coil or separate cryo-stat → simplified maintenance

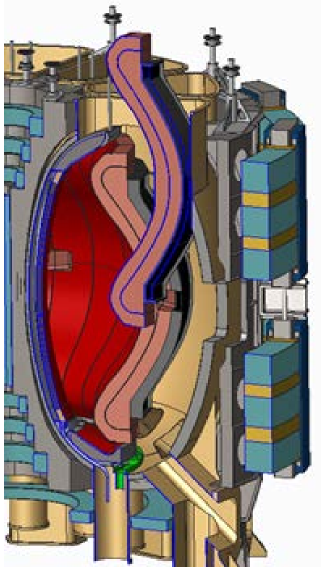




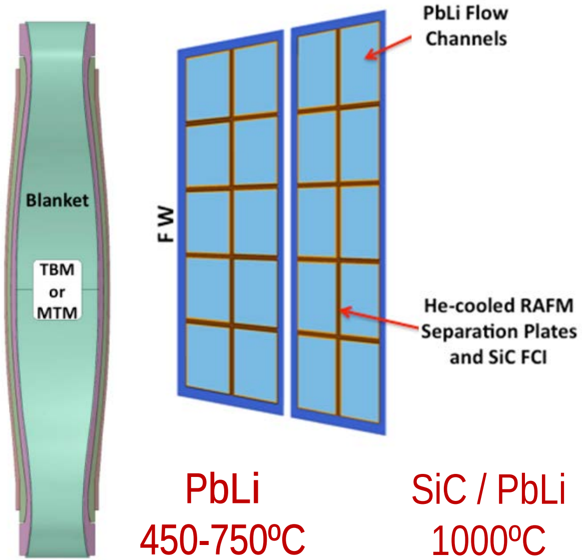
- Significantly reduce outboard PF coil current, force, structure
- If liquid lithium, wall pumping could help increase H-factor

Liquid metal blankets offer potential for high thermal efficiency, modular design

HTS ST-FNSF/Pilot



L. El-Guebaly, et al., Energies, 9 (2016) 632

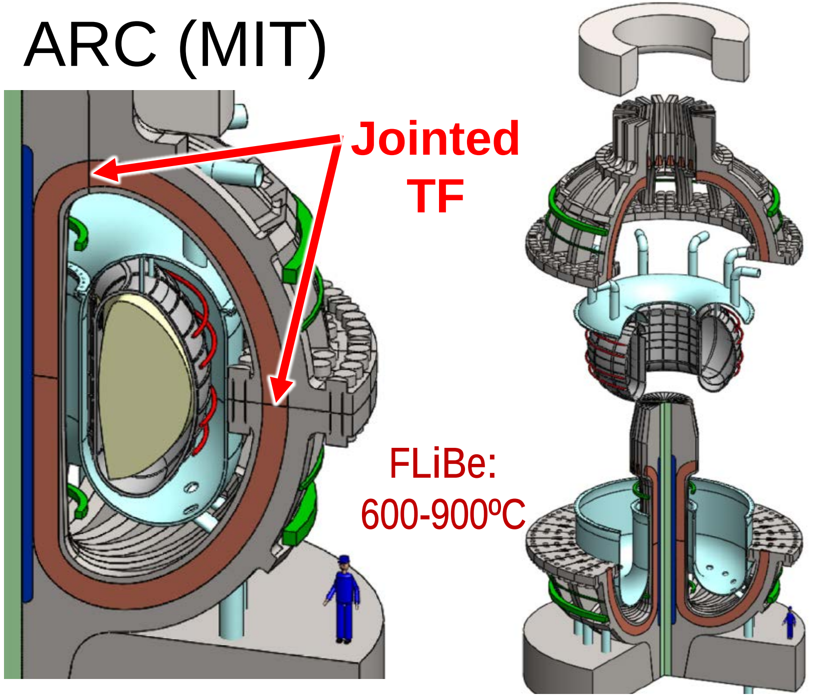


PbLi
450-750ºC

SiC / PbLi
1000ºC

Dual-coolant Lead-Lithium (DCLL) blankets, 20 vertical sectors: η_{th} **= 30-45%** (55% SiC/SiC)

ARC (MIT)



Jointed TF

FLiBe:
600-900ºC

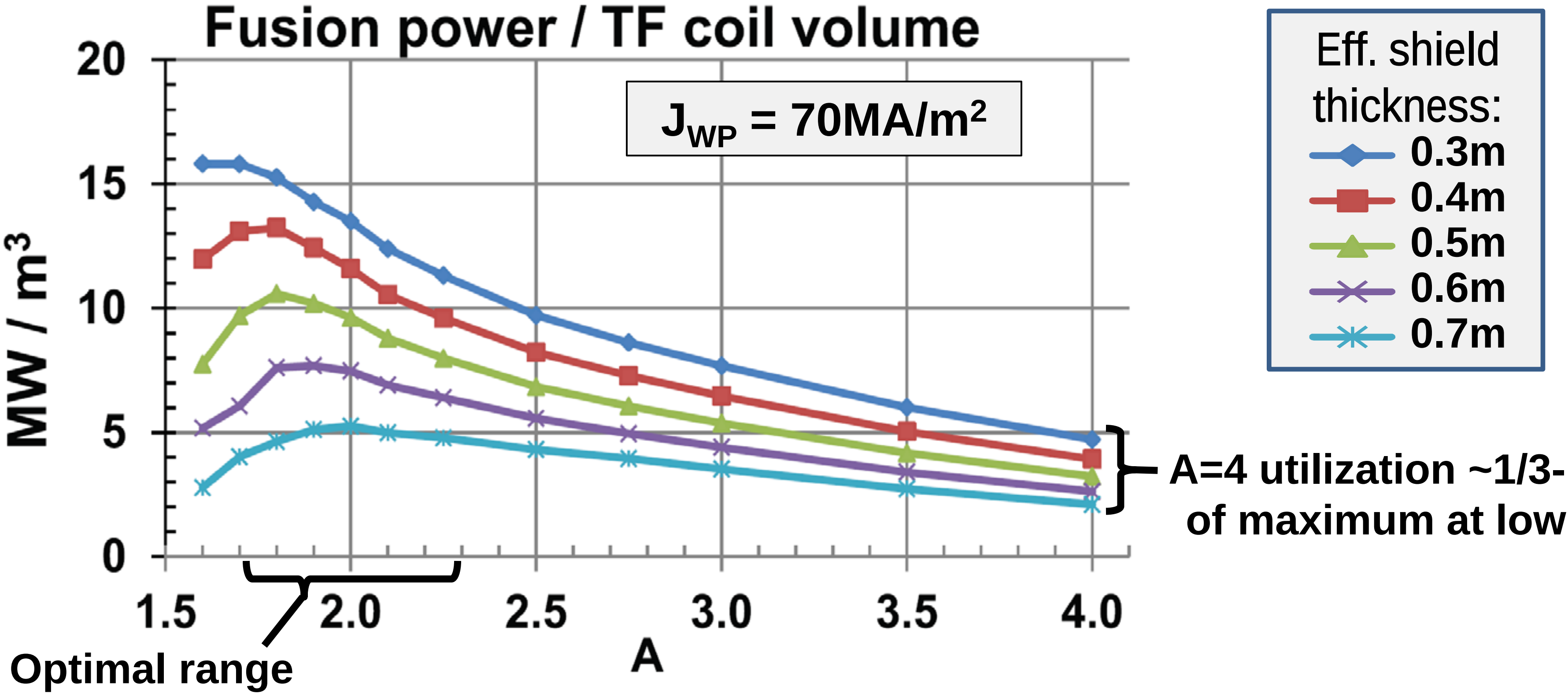
FLiBe liquid immersion blanket, single component/removable: η_{th} **= 40-50%**

Summary: Compact fusion Pilot Plants possible with improved technology and physics operating regimes

- Rare Earth BCO (REBCO) HTS TF magnet development
- Lower A to improve stability – informed by NSTX-U/MAST-U
- Pedestal confinement control via optimized velocity-shear, width
  - Li wall pumping to provide pedestal width control, flatter T profiles
- More efficient bulk current drive to facilitate steady-state
- Long-leg / Super-X for heat-flux mitigation for narrow SOL
- Liquid metals to exhaust heat, particles, eroded materials
  - Simplify PF coil layout and vertical maintenance strategy
- High-temp, efficiency modular or pool liquid-metal blankets

Backup slides

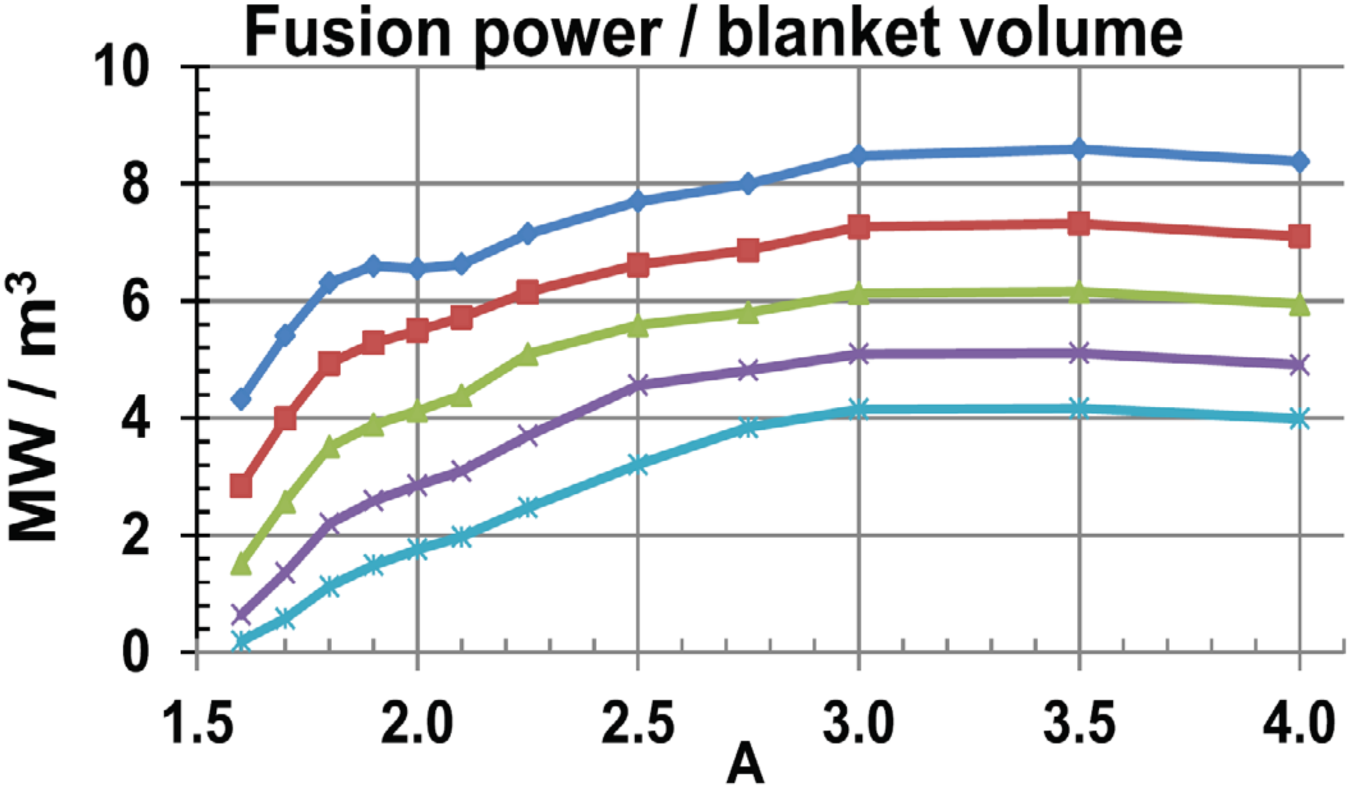
A = 1.8-2.3 maximizes TF magnet utilization, and TF will be significant fraction of core cost



A=4 utilization ~1/3-1/2 of maximum at low A

Optimal range

A ≥ 3 maximizes blanket volume utilization

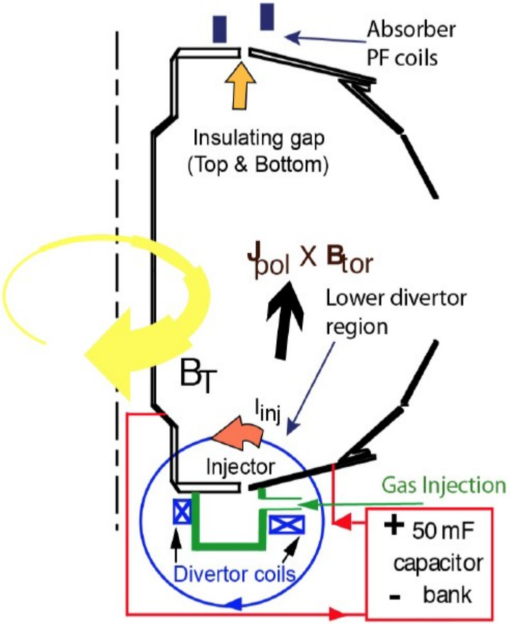


$J_{WP} = 70MA/m^2$

ST-FNSF may need solenoidless current start-up method

Coaxial Helicity Injection (CHI) effective for current initiation

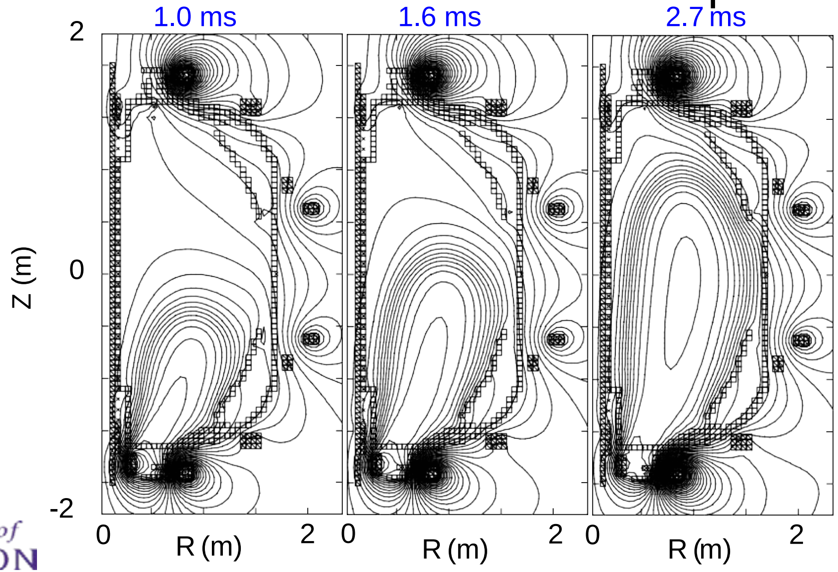
CHI developed on HIT, HIT-II
Transferred to NSTX / NSTX-U



R. Raman et al., PRL 2006

NSTX: 150-200kA closed flux current
NSTX-U: CHI projects to 300-400kA

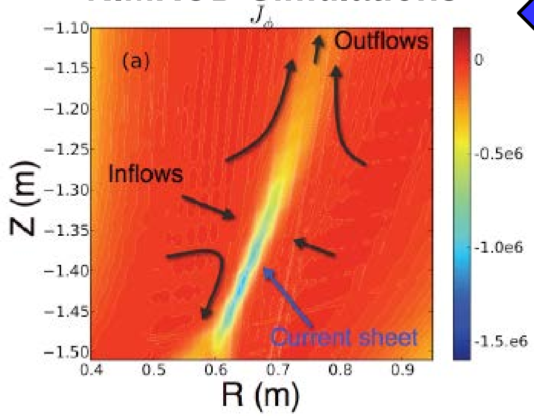
TSC axisymmetric
simulation of CHI startup

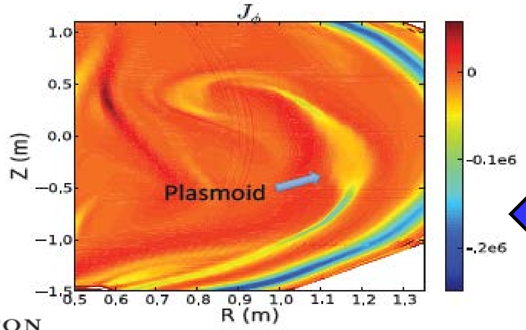


UNIVERSITY of WASHINGTON

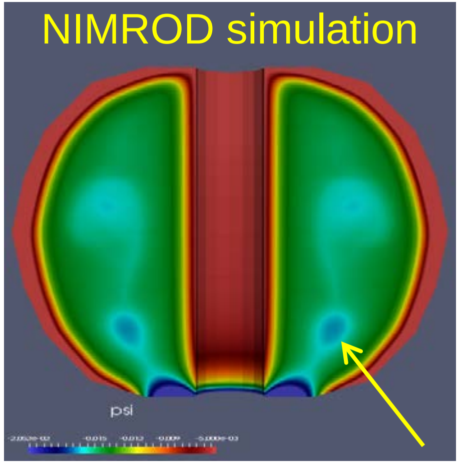
NIMROD simulations of CHI at high Lundquist number → plasmoid-mediated reconnection assists in flux closure

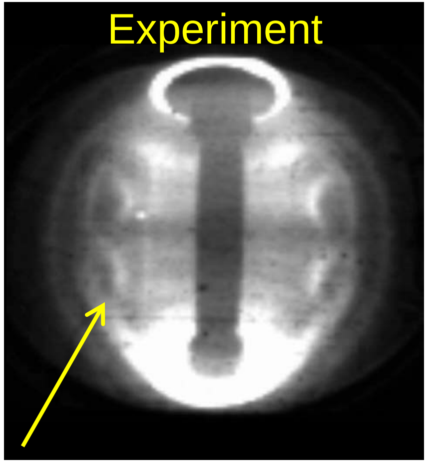
NIMROD Simulations





Sweet Parker (S-P) reconnection in the injector region at low Lundquist number





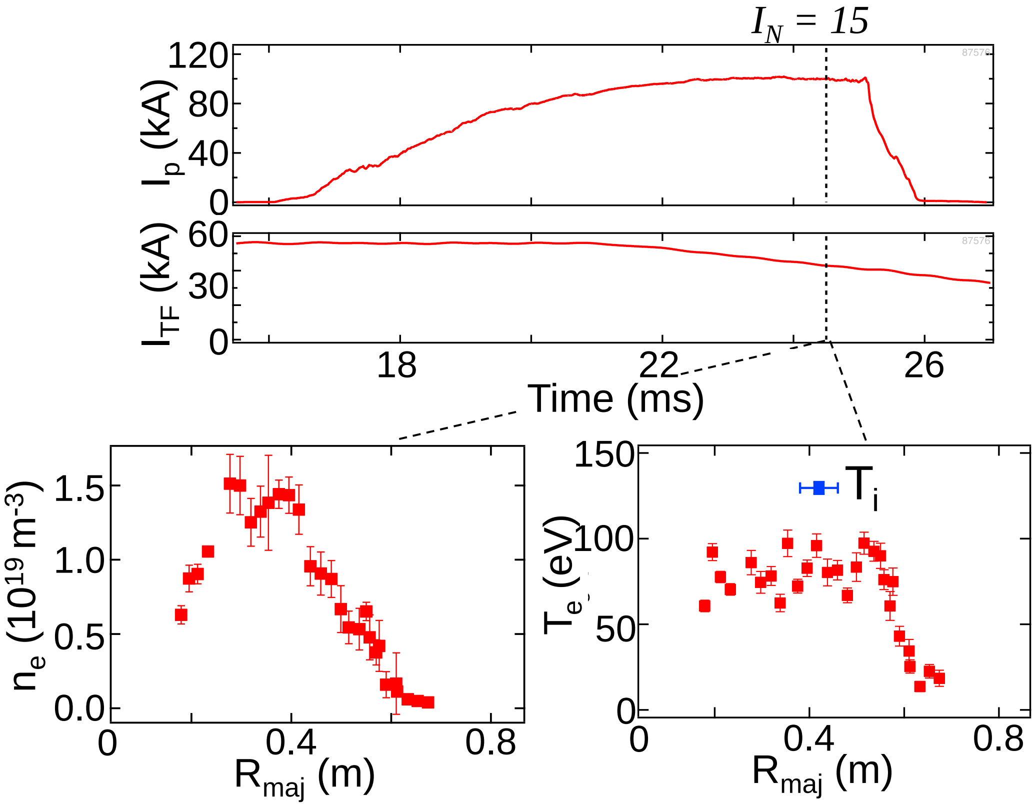
Plasmoids

- Plasmoids seen in NSTX CHI
- Plasmoids potentially important for extrapolating CHI to FNSF

F. Ebrahimi, et al., PRL (2015), NF (2016)

HFS Injection at low TF Provides Non-Solenoidal Sustainment at High I_N

- HFS LHI development campaign provides unique operation space
  - Low $I_{TF} \sim 0.6\, I_p$
  - $I_N = 5A\frac{I_p}{I_{TF}} > 10$ accessible
- Enables high β_t access[1]
  - Aided by anomalous ion heating
- Kinetic constraints on magnetic equilibrium fits[2]
  - $P_{tot}(0)$
  - Edge location defined by T_e profiles



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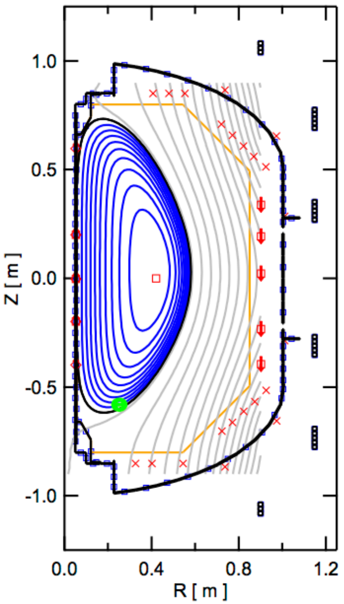
[1] M.W. Bongard, et al. NP10.52 Poster Wed morning

[2] G.M. Bodner, et al. NP10.54, Poster Wed. morning

LHI-Produced Plasmas at low B_t Provide High β_t

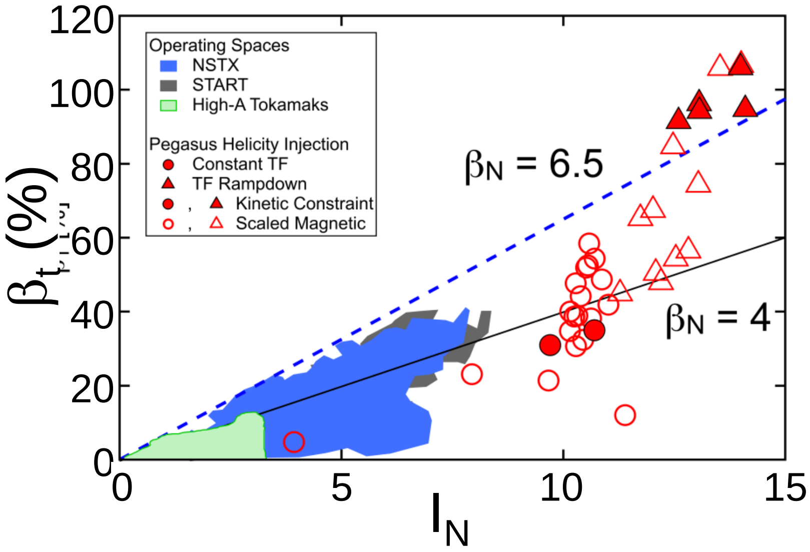
- Sample magnetic reconstruction at t = 24.5 ms, using kinetic constraints



Equilibrium Parameters
Shot 87332, 24.50 ms

I_p	102 kA	R_0	0.317 m
β_t	0.95	a	0.263 m
ℓ_i	0.22	A	1.21
β_p	0.45	κ	2.6
W	545 J	δ	0.54
B_{T0}	0.0249 T	q_{95}	7.24

- β_T for sustained, low-ℓ_i, high-κ, LHI-driven plasmas



[1] M.W. Bongard, et al. NP10.52 Poster Wed morning

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

Simplified TF magnet design equations

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

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

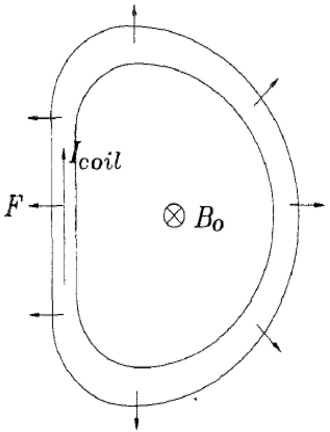


Fig. 5. Lorentz forces are normal to the conductor in the poloidal plane.

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

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

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

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

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

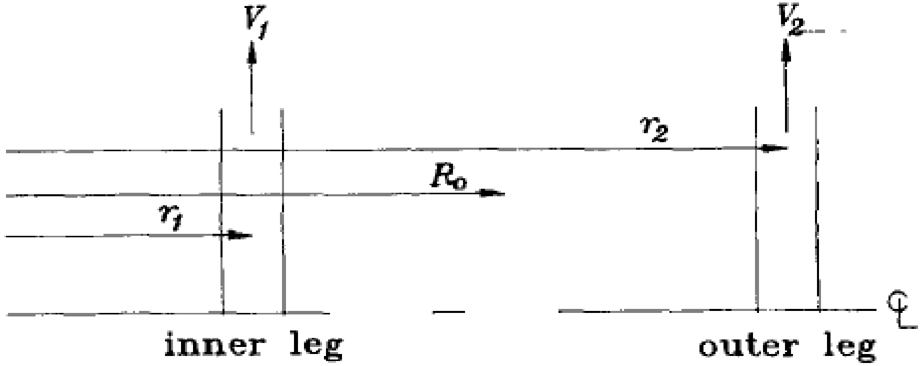
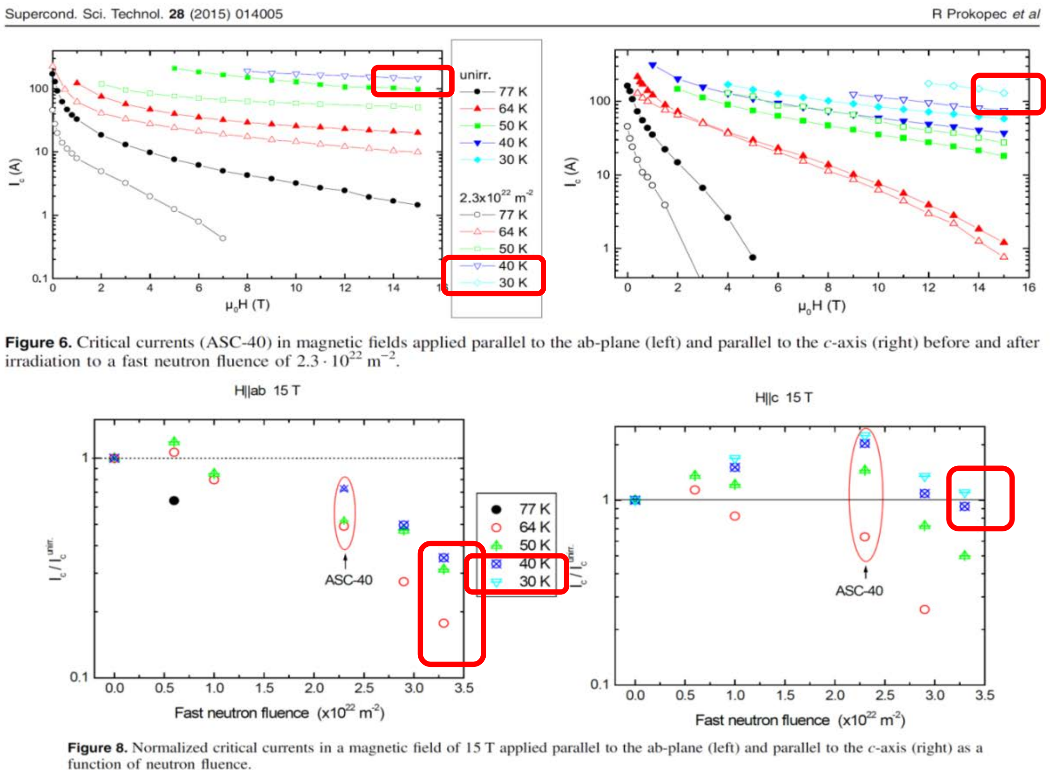


Fig. 7. Geometry for force and moment balances.

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

HTS performance vs. field and fast neutron fluence

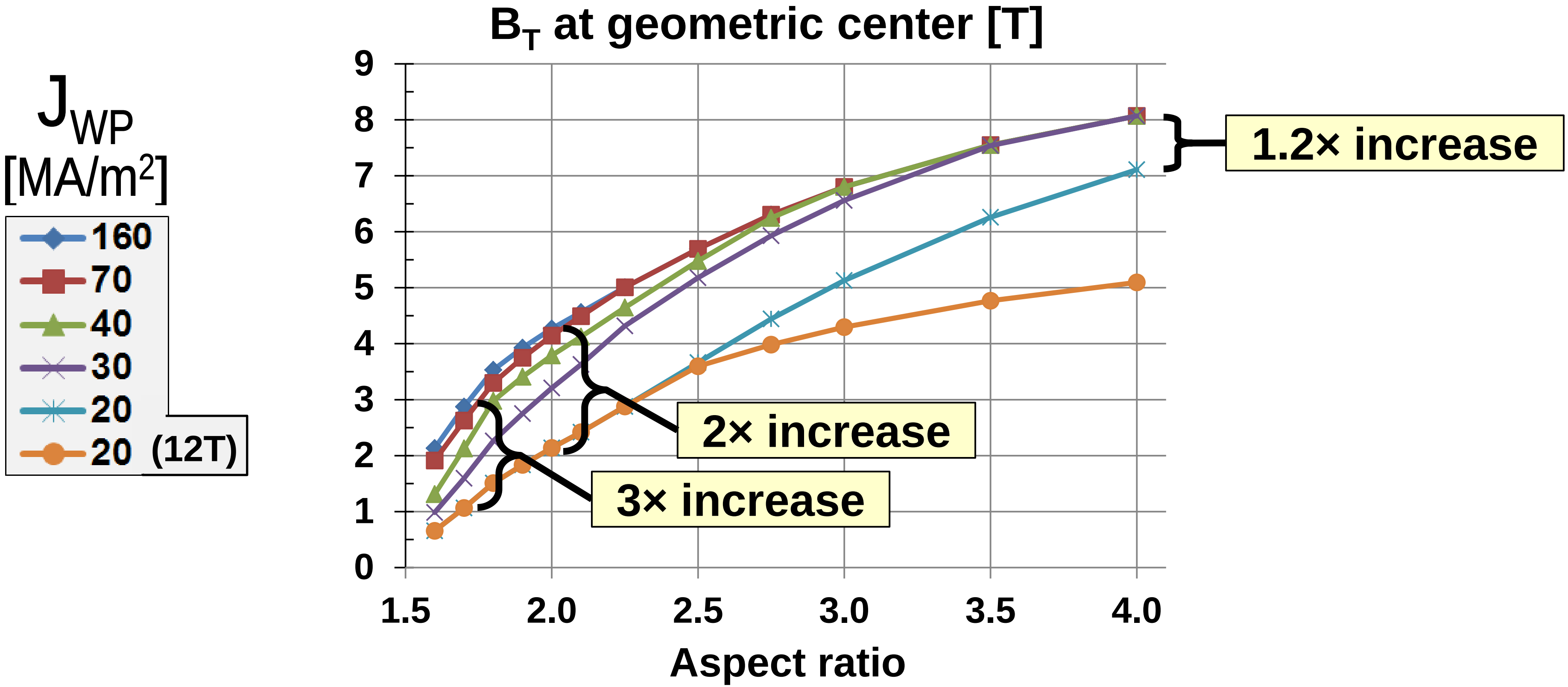


Supercond. Sci. Technol. **28** (2015) 014005 R Prokopec *et al*

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.

Increased TF winding-pack current density can increase B_T in plasma 2-3× at low A



1.2× increase

2× increase

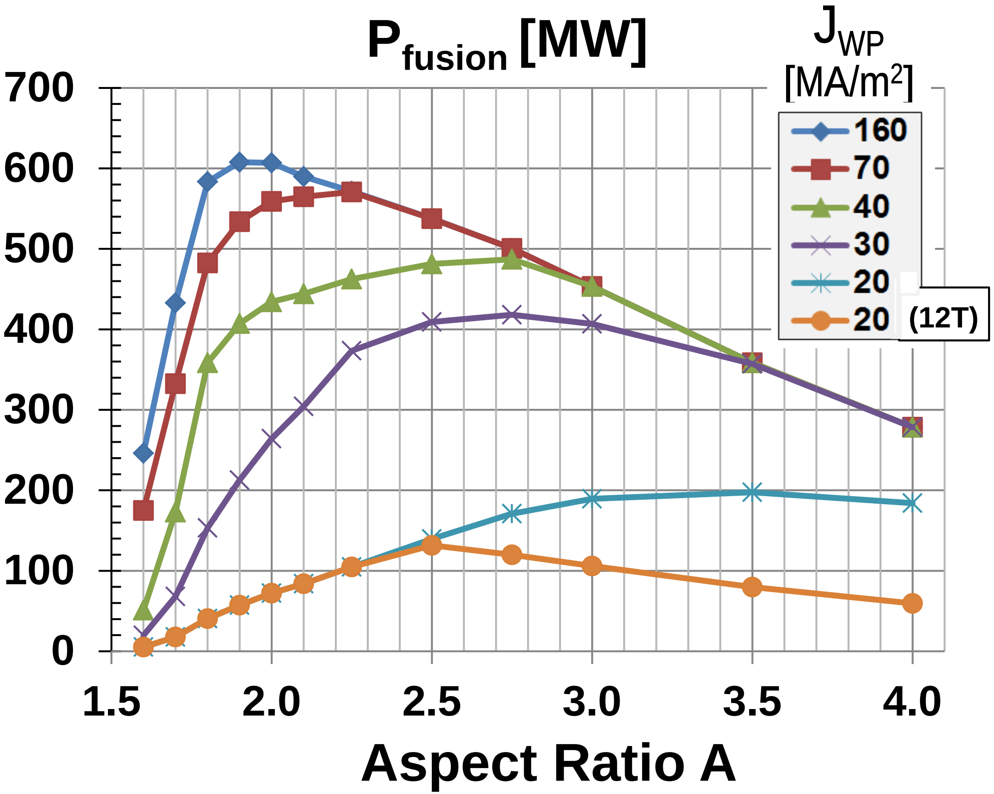
3× increase

High current density HTS toroidal field coils could enable access to high fusion power in R~3m device

- ITER-like TF constraints:
  - J_{WP}=20MA/m^2, $B_{max} \leq$ 12T
  - $P_{fusion} \leq$ 130MW

- J_{WP} ~ 30MA/m^2, $B_{max} \leq$ 19T
  - P_{fusion} ~ 400MW

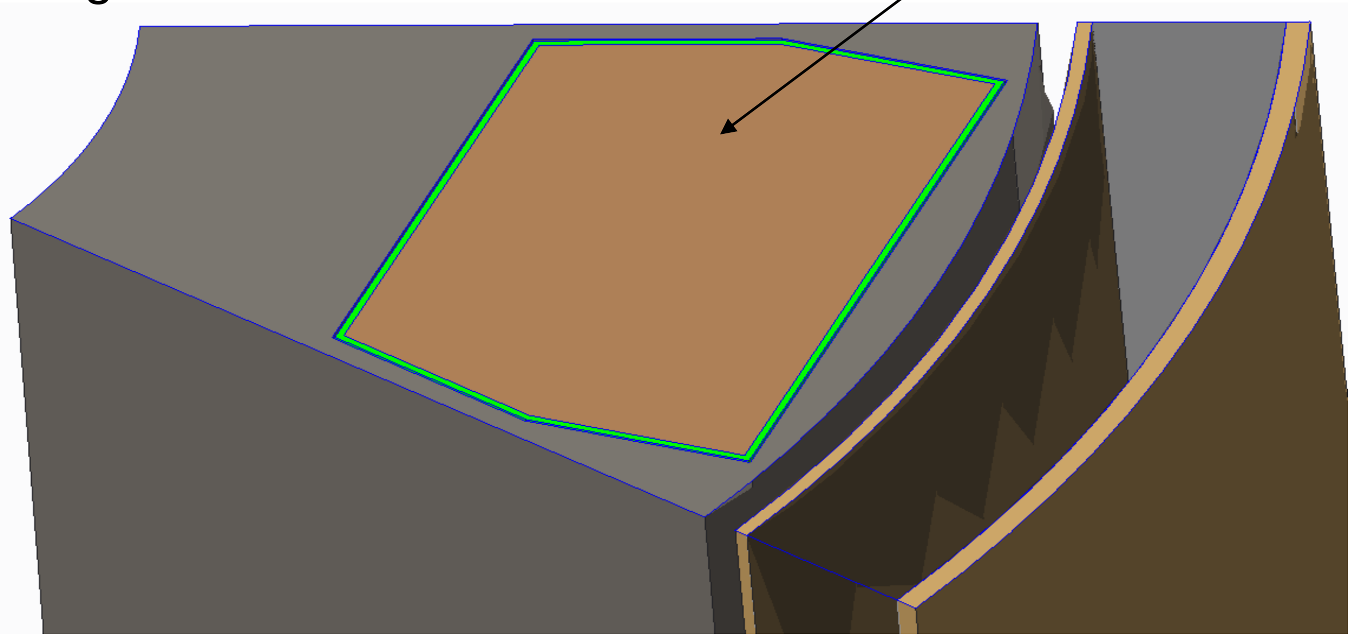
- $J_{WP} \geq$ 70MA/m^2,$B_{max} \leq$ 19T
  - P_{fusion} ~500-600MW



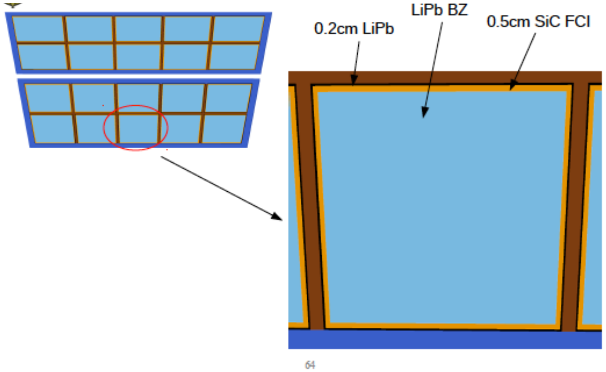
A=2, R_0 = 3m device TF inboard leg

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

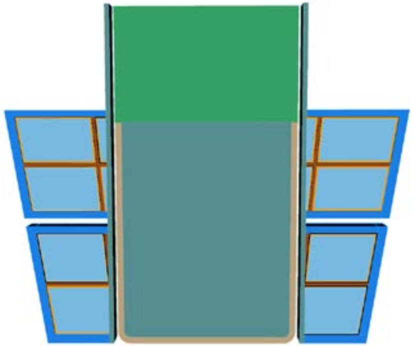
0.167 m^2 winding area



CAD Geometry of OB Blanket with Ports

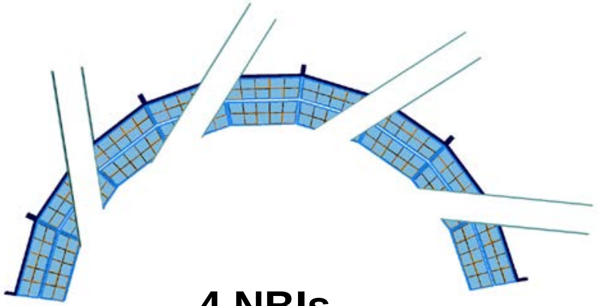


Detailed Blanket Internals

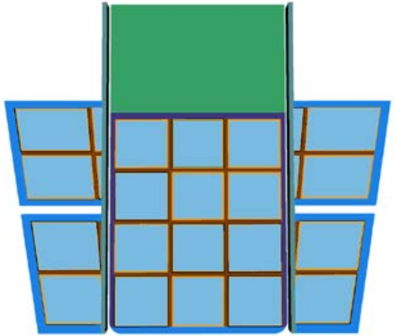


MTM

Stabilizing Shells

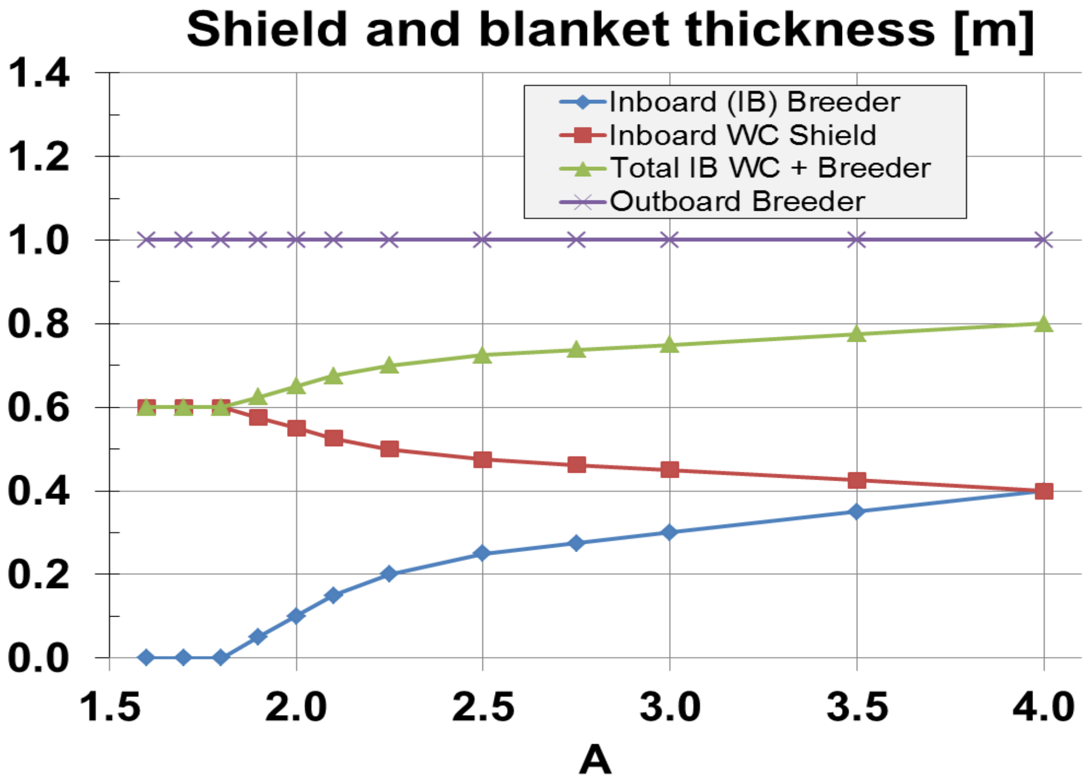


4 NBIs

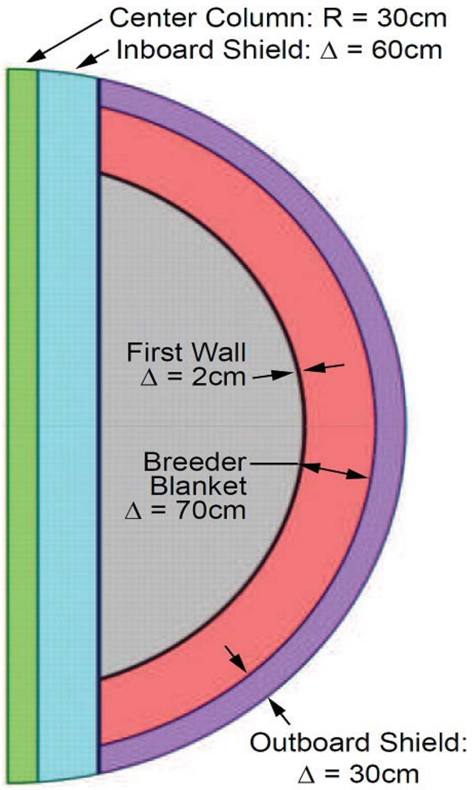


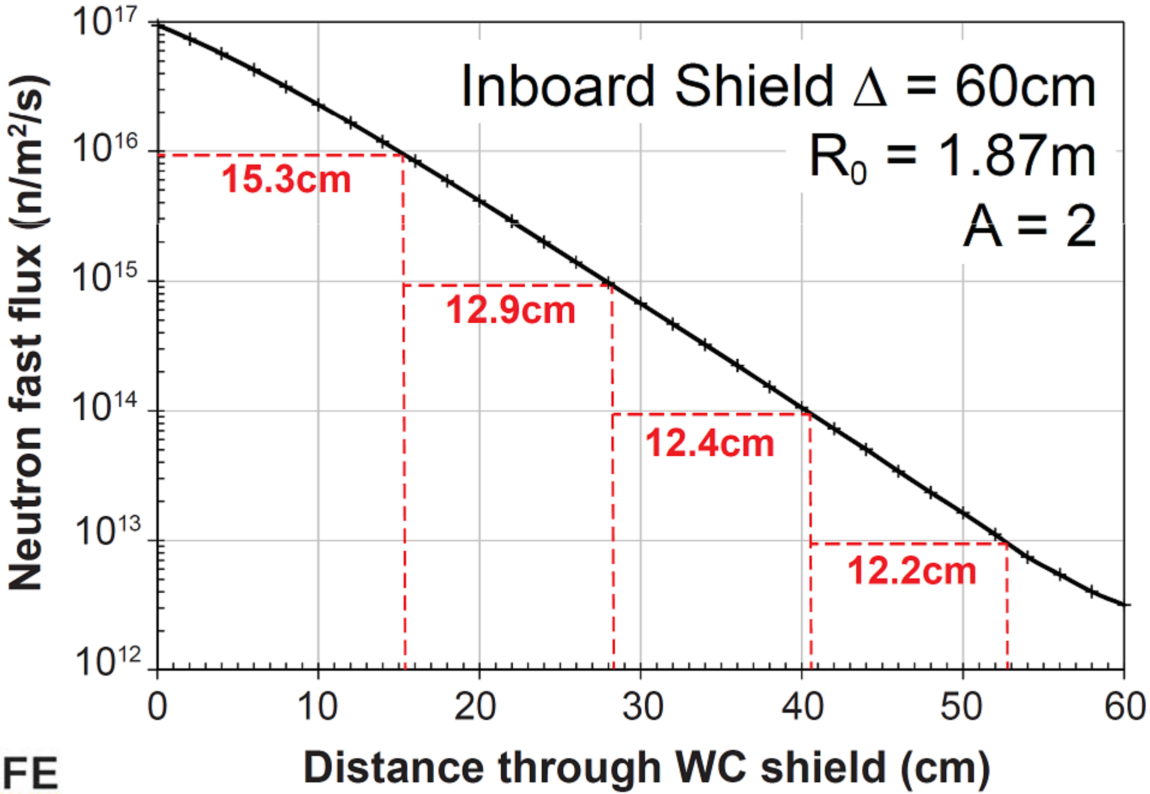
TBM

Model blanket and shield thickness vs. A



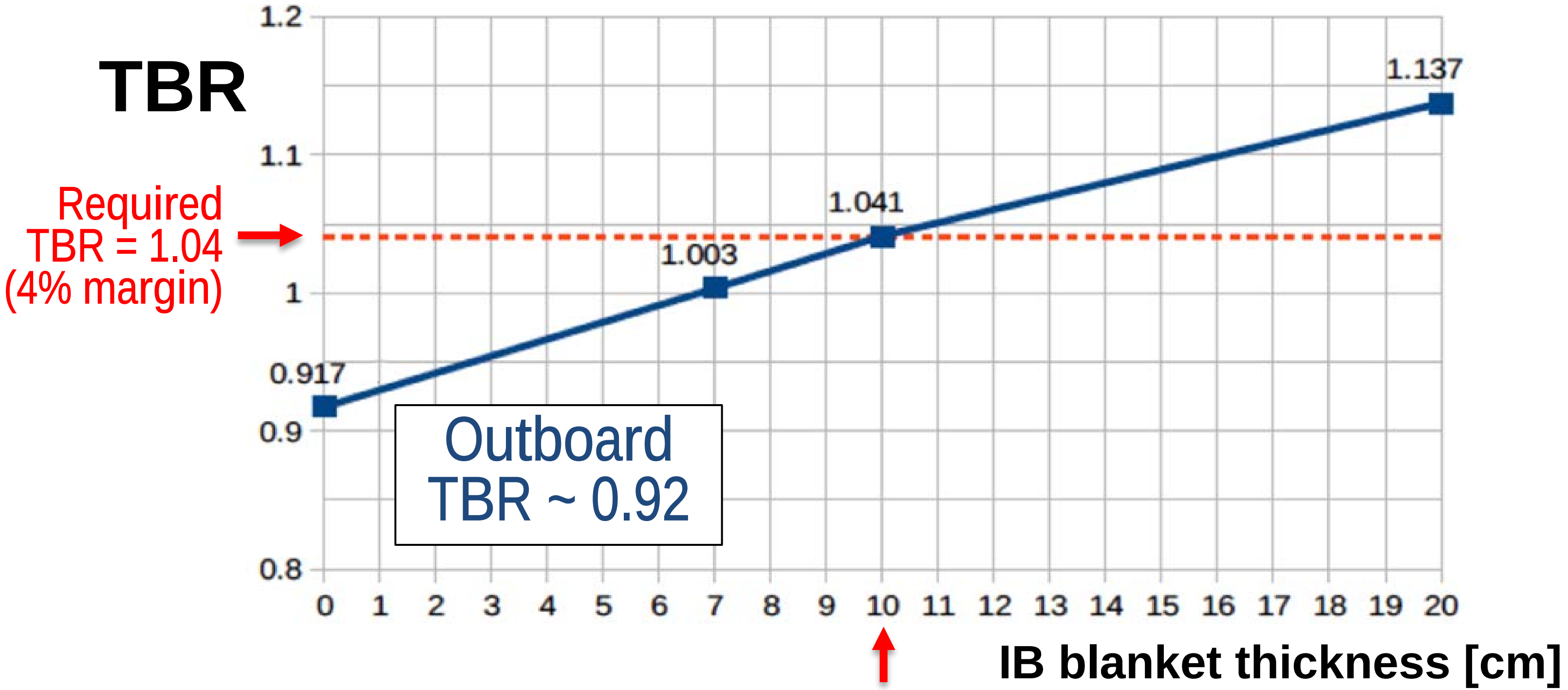
Neutronics analysis for HTS TF shielding





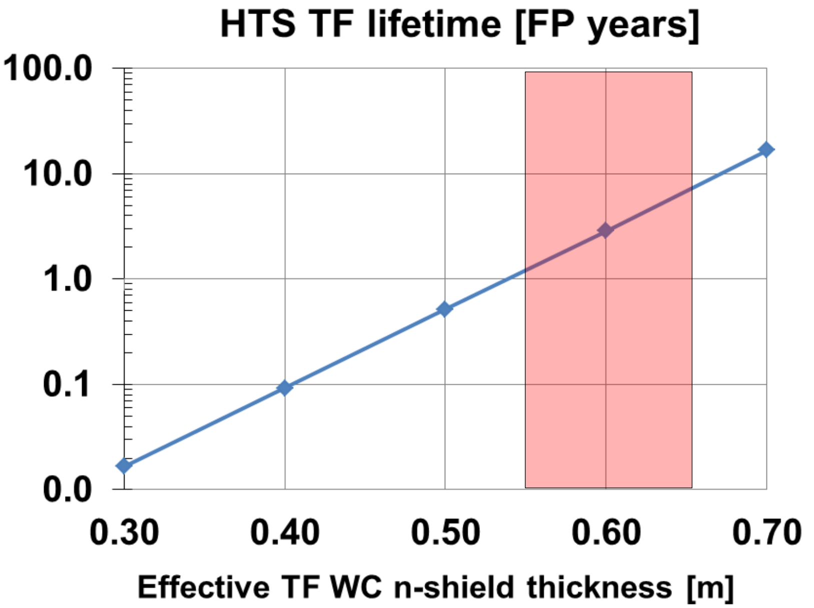
CCFE

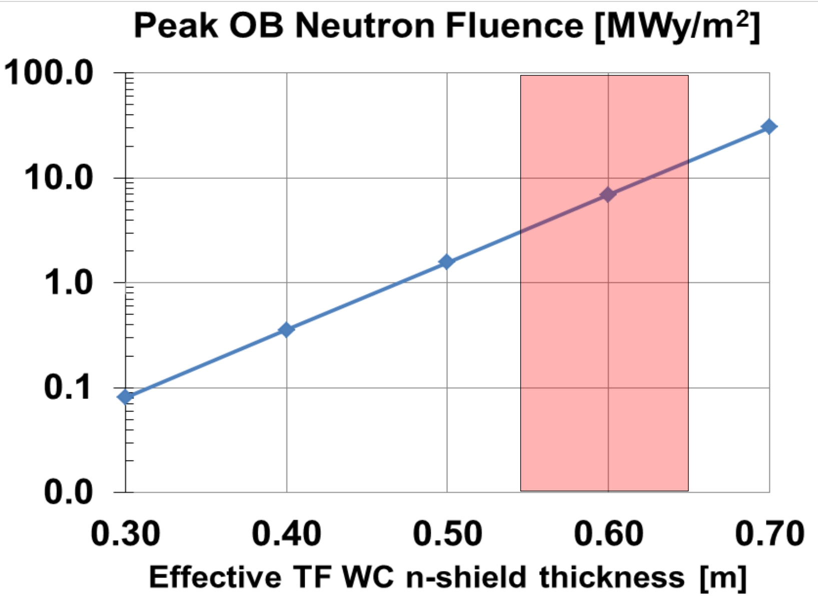
Need inboard breeding for TBR > 1 at A=2



10cm IB blanket sufficient for TBR > 1.04

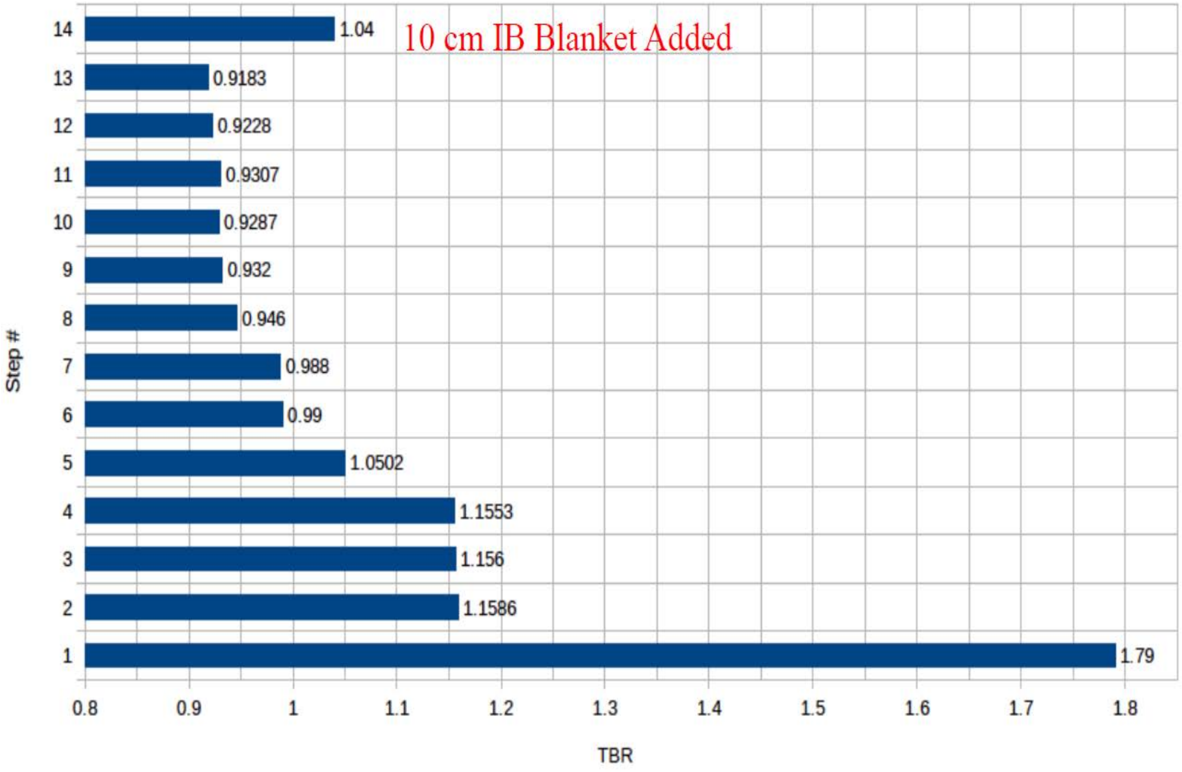
HTS TF lifetime is very strong function of inboard shielding thickness





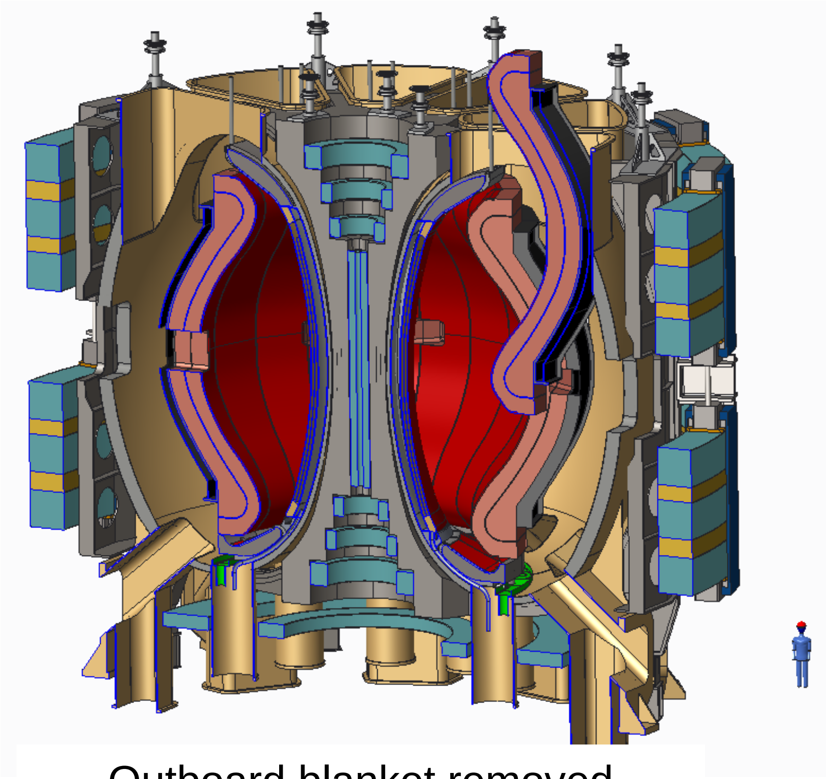
Inboard shield + blanket equivalent to 60cm WC → 3FPY → 6-7MWy/m^2 → fulfill FNSF requirement

Detailed breeding calculations completed for A=2

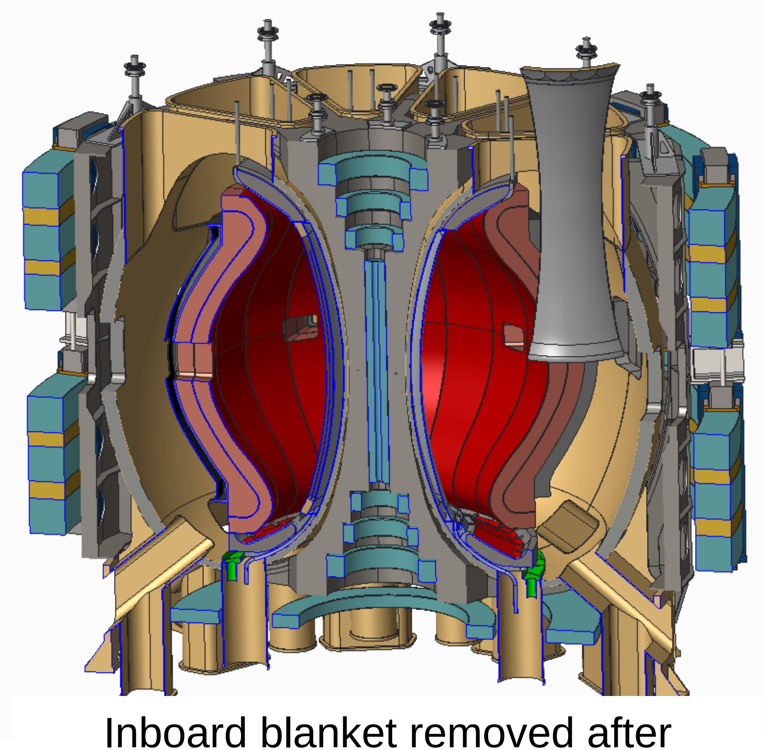


- Step 1- Infinite media of LiPb
- Step 2- LiPb confined to OB FW/blanket
- Step 3- Assembly gaps added
- Step 4- Homogeneous mixture of blanket in upper and lower ends of OB blanket
- Step 5- FW material added
- Step 6- Side, back, and front walls added
- Step 7- Cooling channels added
- Step 8- SiC 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

Outboard / inboard blanket vertical maintenance

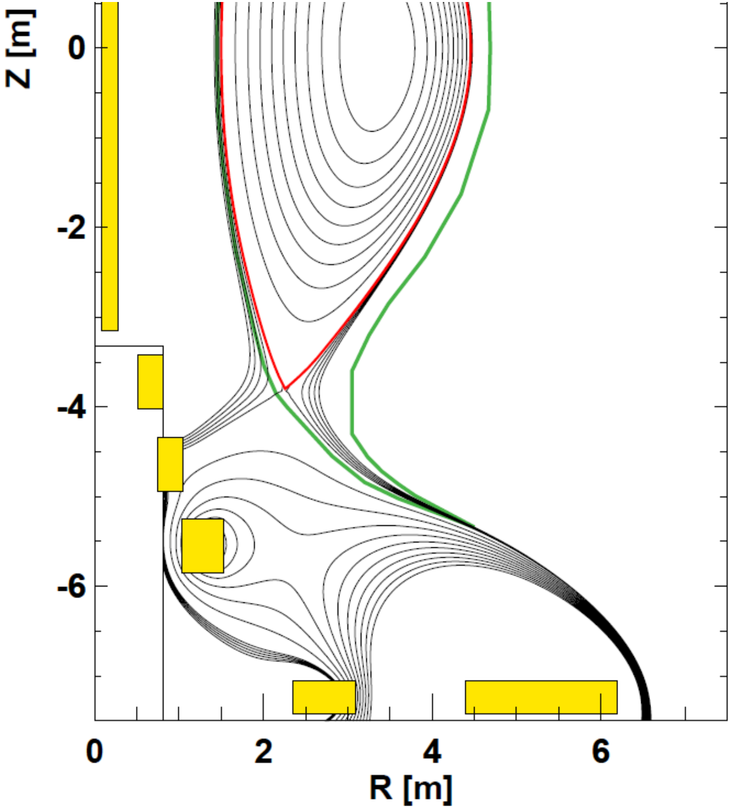


Outboard blanket removed



Inboard blanket removed after outboard blanket sectors removed

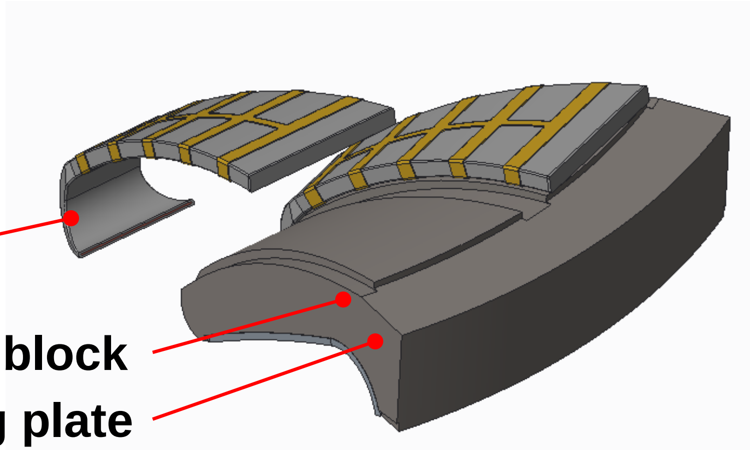
Long-leg / deep-V slot divertor



- 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 and bottom of device

Pilot Plant study exploring liquid metal divertor design similar to flowing water curtain systems

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



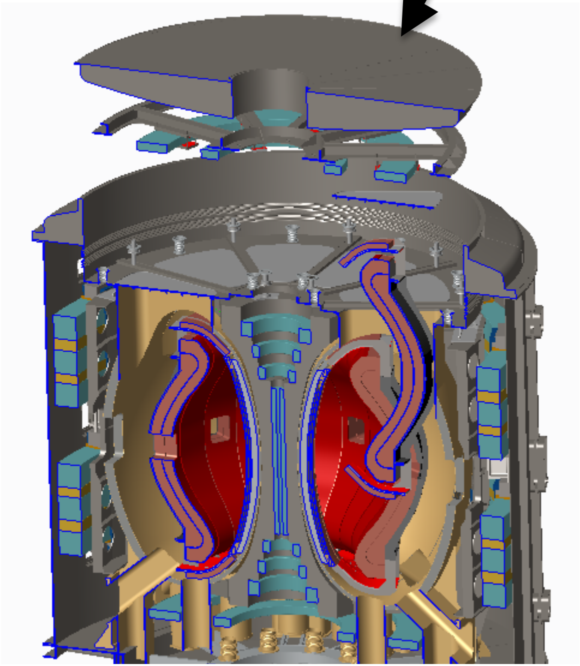
LM containment structure

Shield block

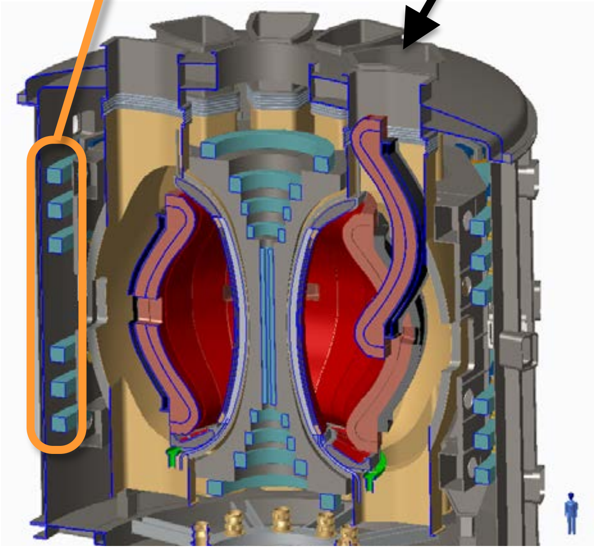
Ferritic steel backing plate

Comparison of long vs. shorter-leg divertor Pilots:

- Simplify vertical maintenance
- Reduce outboard PF currents
  → can use LTS PF coils

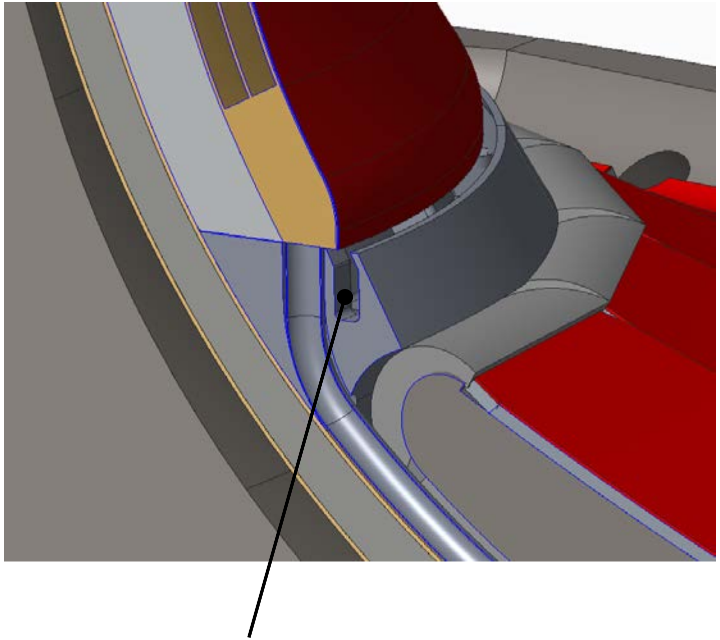


Long-leg/Super-X divertor

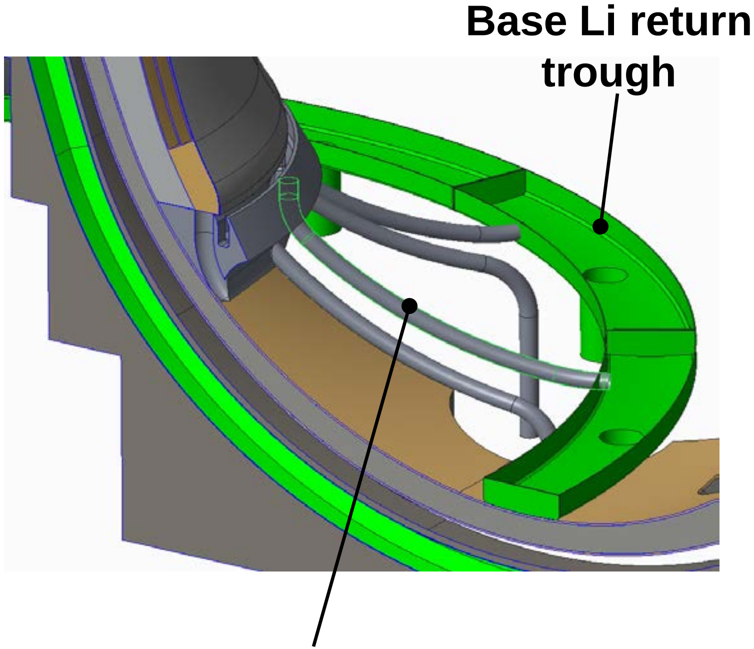


Liquid Metal divertor

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

Local details of Li divertor / inboard FW