U.S. DEPARTMENT OF ENERGY | Office of Science

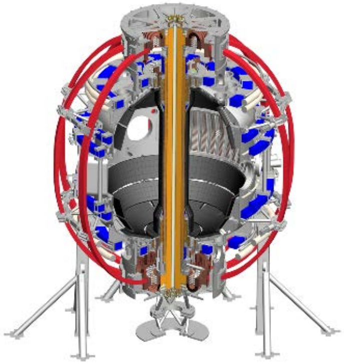
NSTX-U

Key physics issues and opportunities for next-step spherical torus devices

Jonathan Menard (PPPL)

18th International Congress on Plasma Physics
Kaohsiung Exhibition Center, Kaohsiung, Taiwan
June 27 – July 1, 2016

PPPL



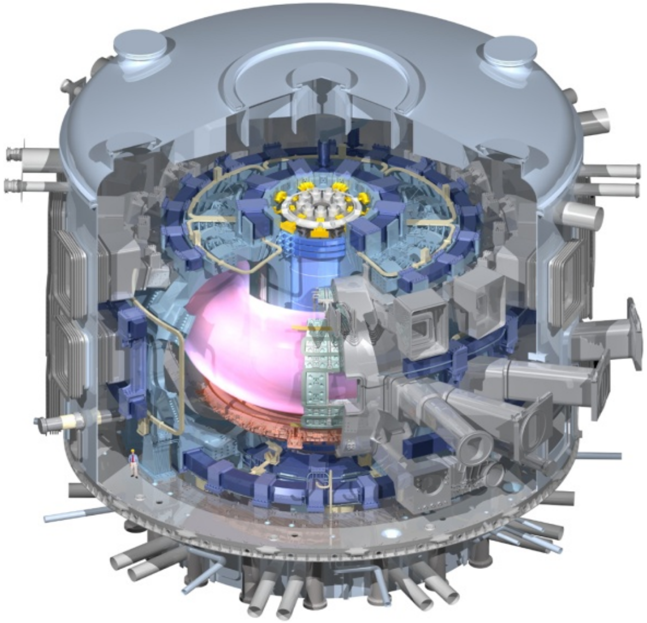
Outline

- Motivations for studying “spherical” tokamak (ST)
- Mission elements of NSTX-U Research Program
- NSTX-U research commissioning status
- Key physics issues for ST
  - Energy transport
  - High-beta stability
  - Power exhaust
  - Current sustainment and start-up

ST research extends predictive capability for ITER and toroidal confinement science

- High β physics, rotation, shaping extend stability, transport knowledge
- NBI fast-ions in present STs mimic DT fusion product parameters in ITER → study burning plasma science
- STs can more easily study electron scale turbulence at low collisionality → important for all magnetic fusion

Burning Plasma Physics - ITER



Recent design studies show ST potentially attractive as Fusion Nuclear Science Facility (FNSF) and Pilot Plant

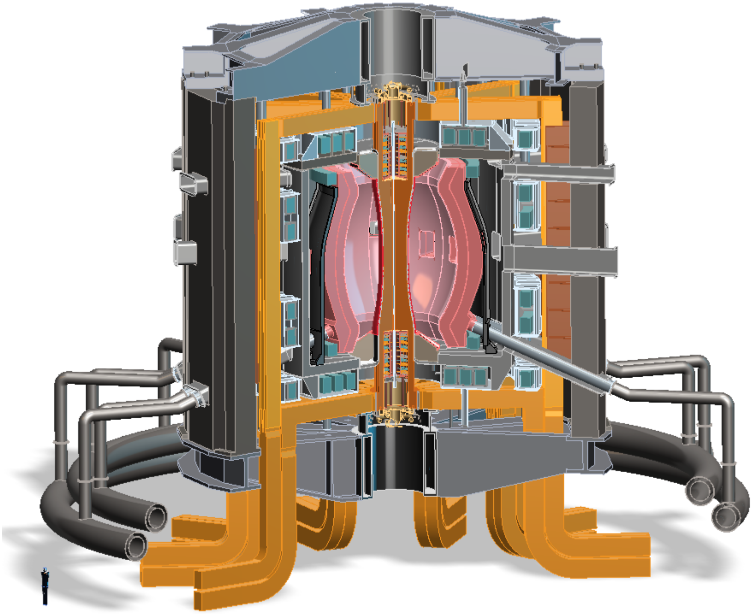
FNSF: Provide neutron fluence for material/component R&D (+ T self-sufficiency?)

Pilot Plant: Electrical self-sufficiency: $Q_{eng} = P_{elec} / P_{consumed} \geq 1$ (+ FNSF mission?)

FNSF with copper TF coils

$A=1.7$, $R_0 = 1.7$m, $\kappa_x = 2.7$, $B_T=3$T

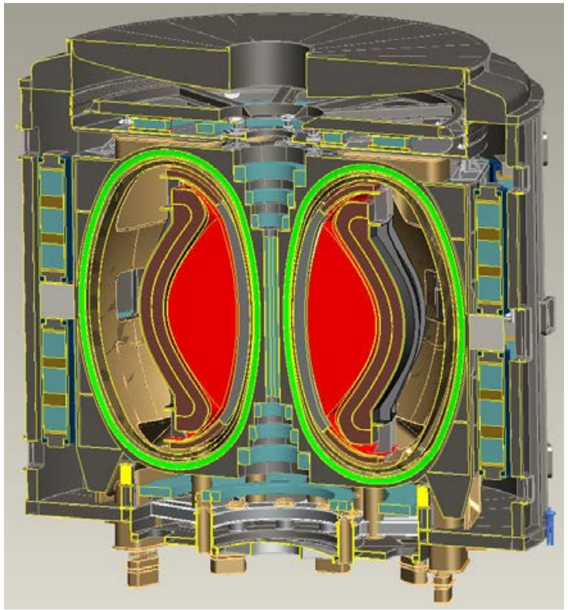
Fluence = 6MWy/m², TBR ~ 1



FNSF / Pilot Plant with HTS TF coils

$A=2$, $R_0 = 3$m, $\kappa_x = 2.5$, $B_T = 4$T

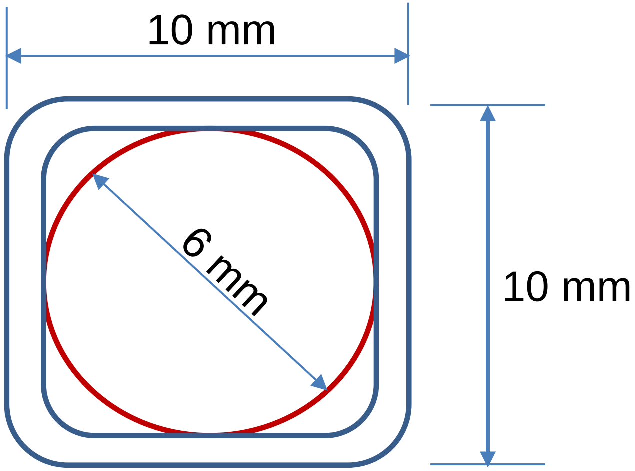
6MWy/m², TBR ~ 1, Q_{eng} ~ 1



Designs integrate ST higher κ , β_N and advanced divertors (+ HTS TF for Pilot Plant)

Beyond high-B_T capability, HTS cables using REBCO tapes achieving very high winding pack current density

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

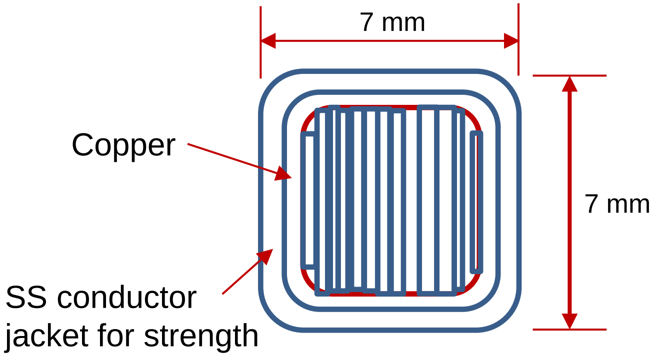


7 kA CORC (4.2K, 19 T) cable

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

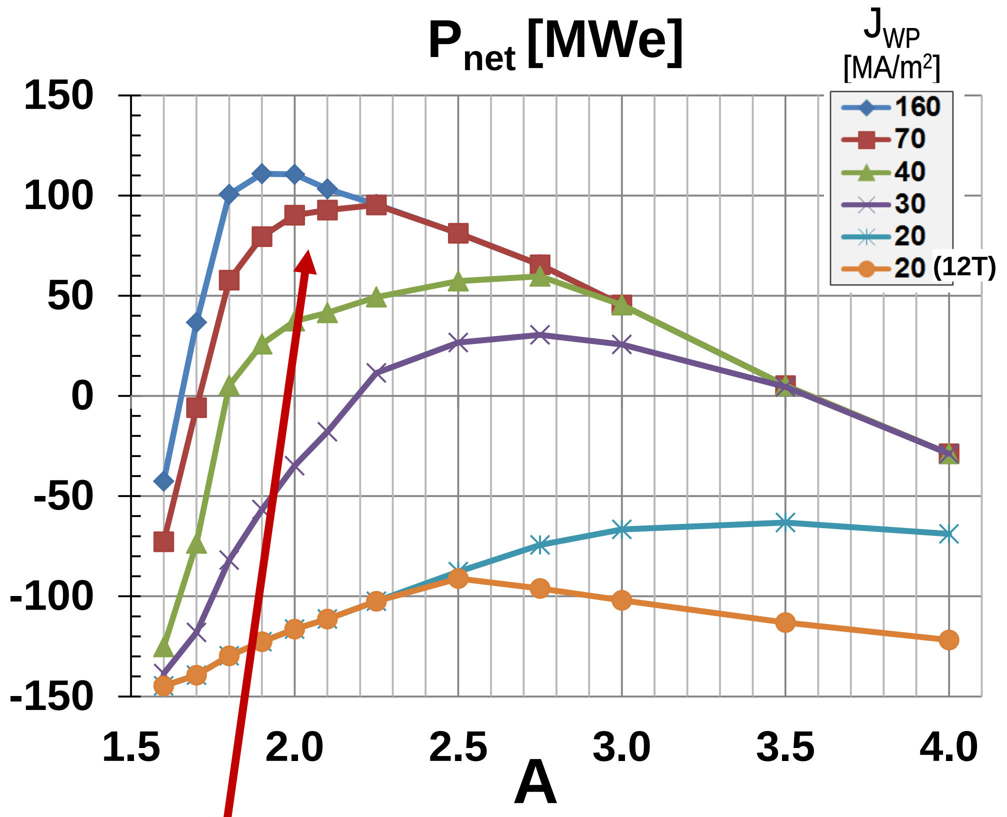


High current density HTS → more compact TF magnets → lower-A tokamak pilot plants

Pilot: R=3m, P_{NBI} = 50MW, f_{GW}=0.8, f_{NICD} = 1, $\beta_N=\beta_{N\text{-no-wall}}(\varepsilon)$, $\kappa=\kappa(\varepsilon)$, H unconstrained, η_{th} = 0.45

- 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}



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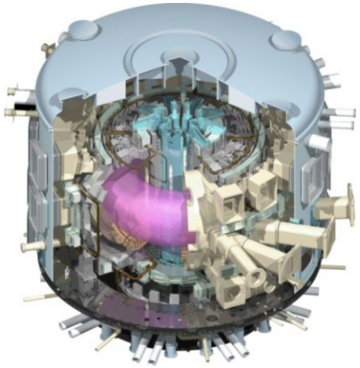
NSTX-U Mission Elements:

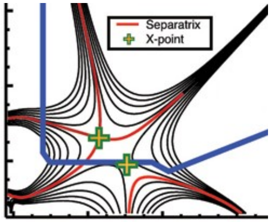
- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond



ITER

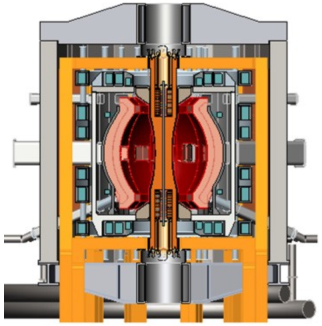
- Develop solutions for plasma-material interface (PMI)

Liquid metals / Lithium

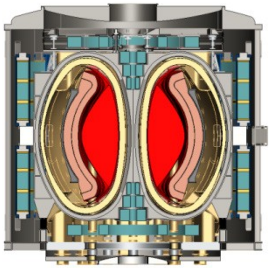


Snowflake/X

- Advance ST as Fusion Nuclear Science Facility and Pilot Plant

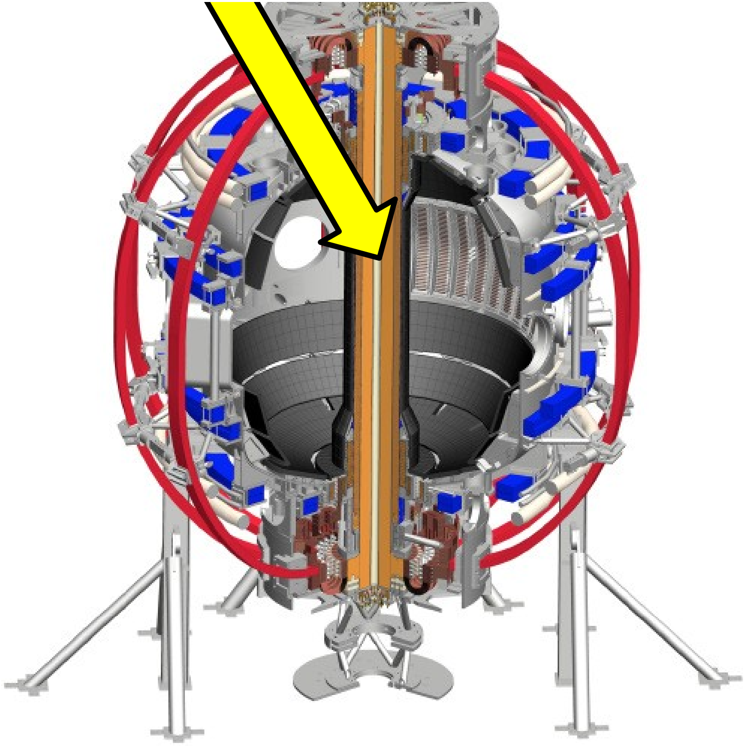


ST-FNSF / Pilot-Plant

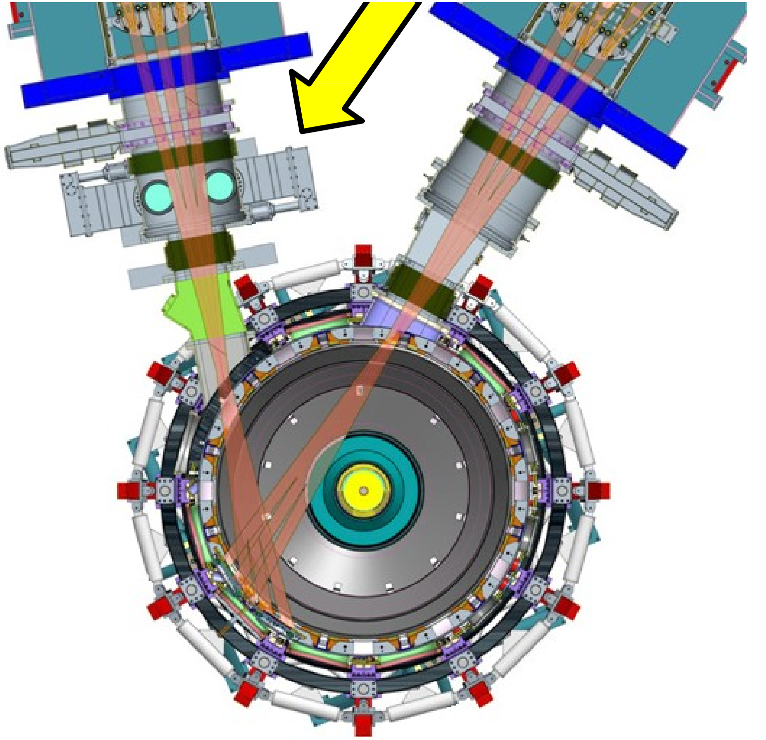


NSTX-U will access new physics with 2 major new tools:

1. New Central Magnet



2. Tangential 2nd Neutral Beam



Higher T, low ν^* from low to high β

→ Unique regime, study new transport and stability physics

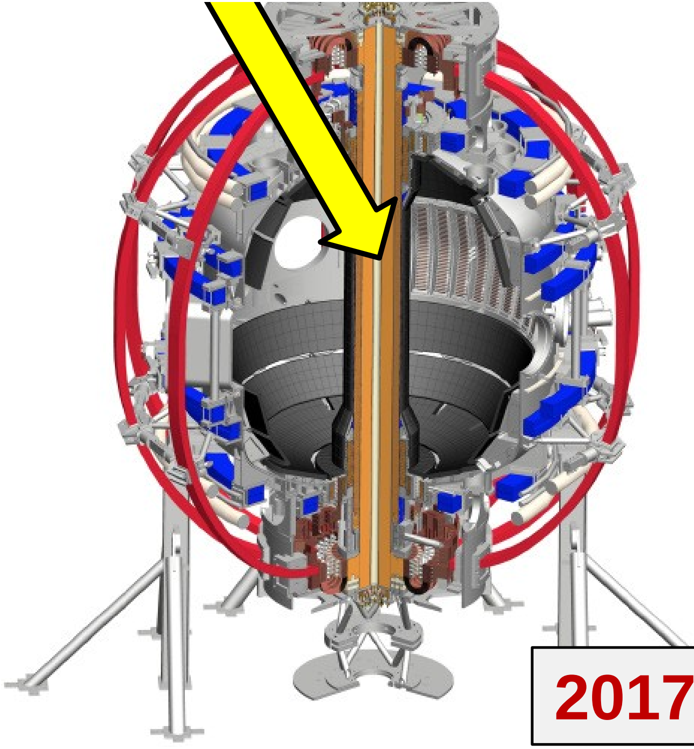
Full non-inductive current drive

→ Not demonstrated in ST at high-β_T

Essential for any future steady-state ST

NSTX-U will have major boost in performance

1. New Central Magnet



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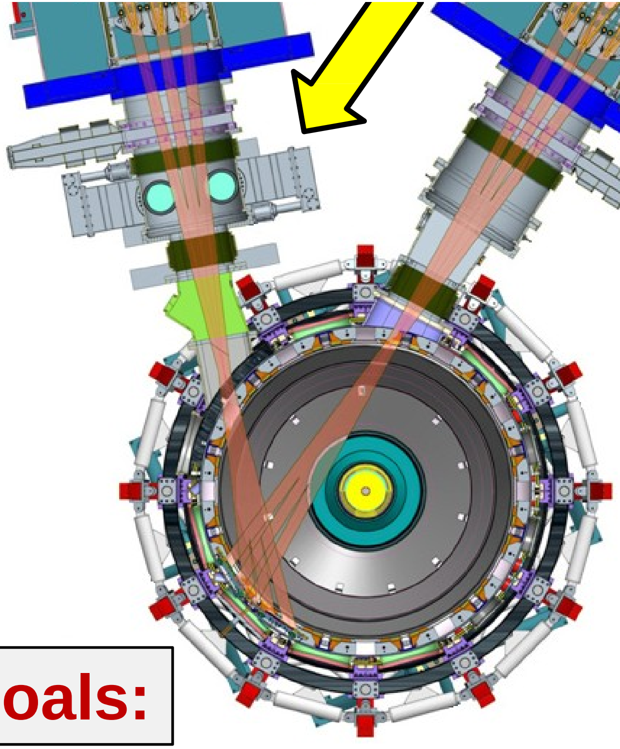
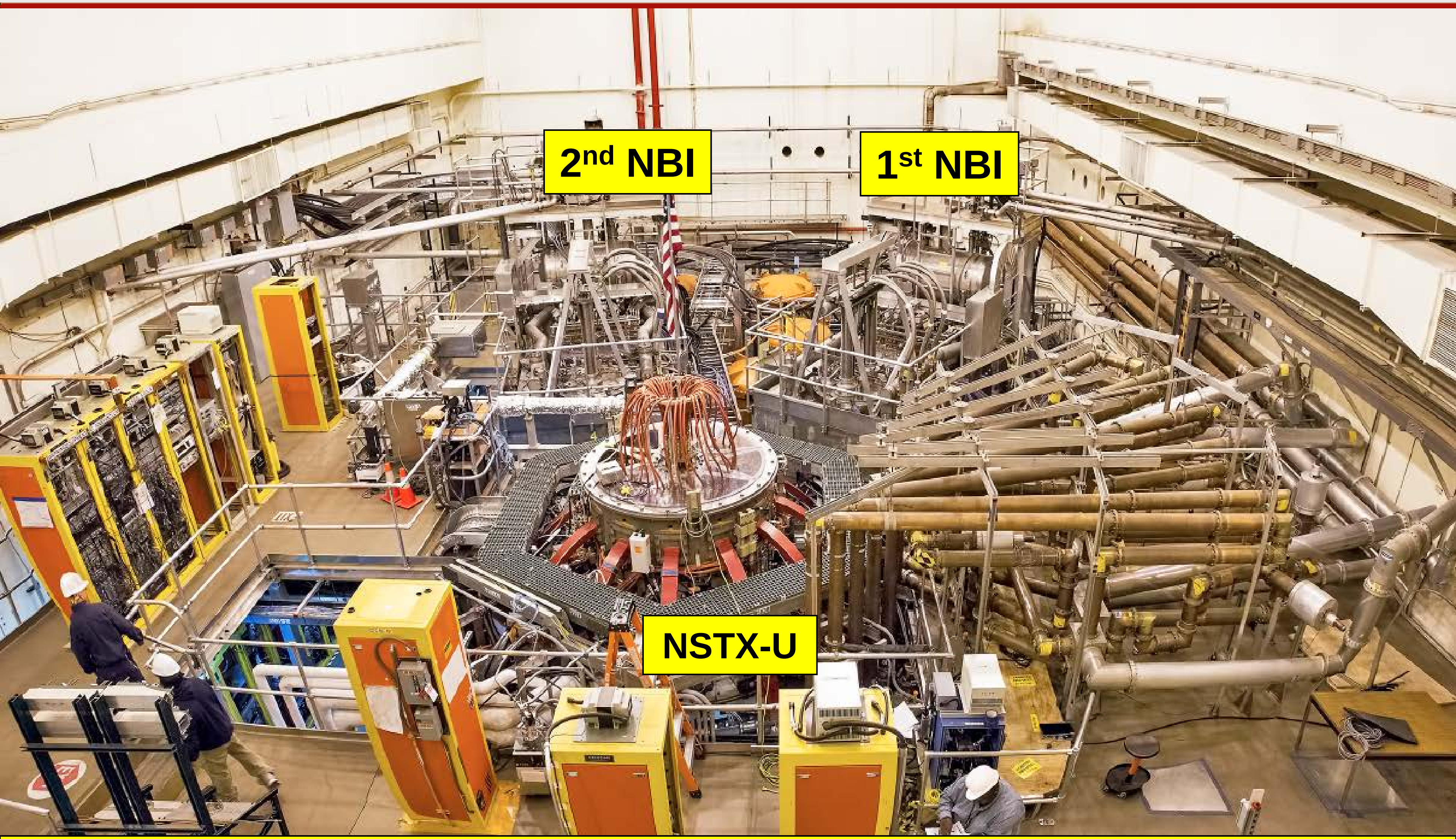


2017-2018 goals:

- ➢2× toroidal field (0.5 → 1T)
- ➢2× plasma current (1 → 2MA)
- ➢5× longer pulse (1 → 5s)

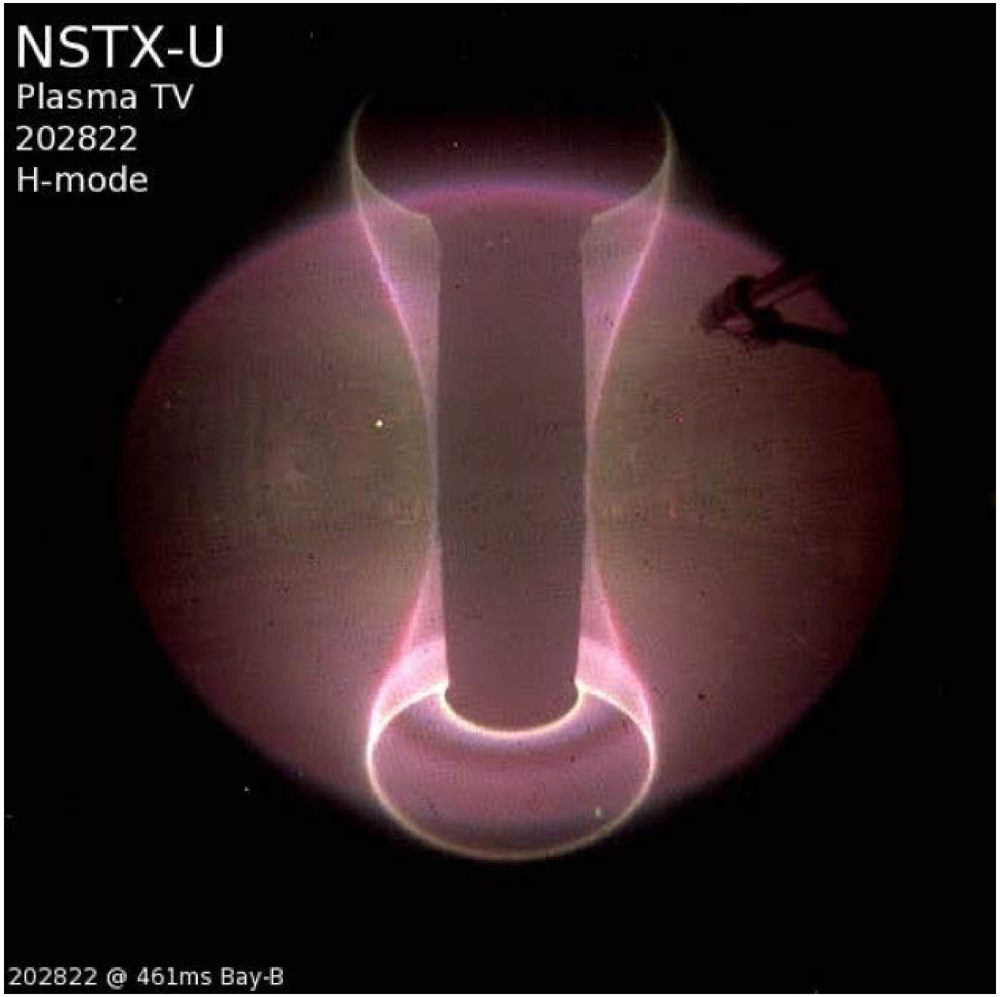
- ➢2× heating power (5 → 10MW)
  - Tangential NBI → 2× current drive efficiency
- ➢4× divertor heat flux (→ ITER levels)
- ➢Up to 10× higher $nT\tau_E$ (~MJ plasmas)

NSTX-U: Operational Research Facility



All six NBI sources now operational, supporting high performance experiments

NSTX-U research initiated January 2016

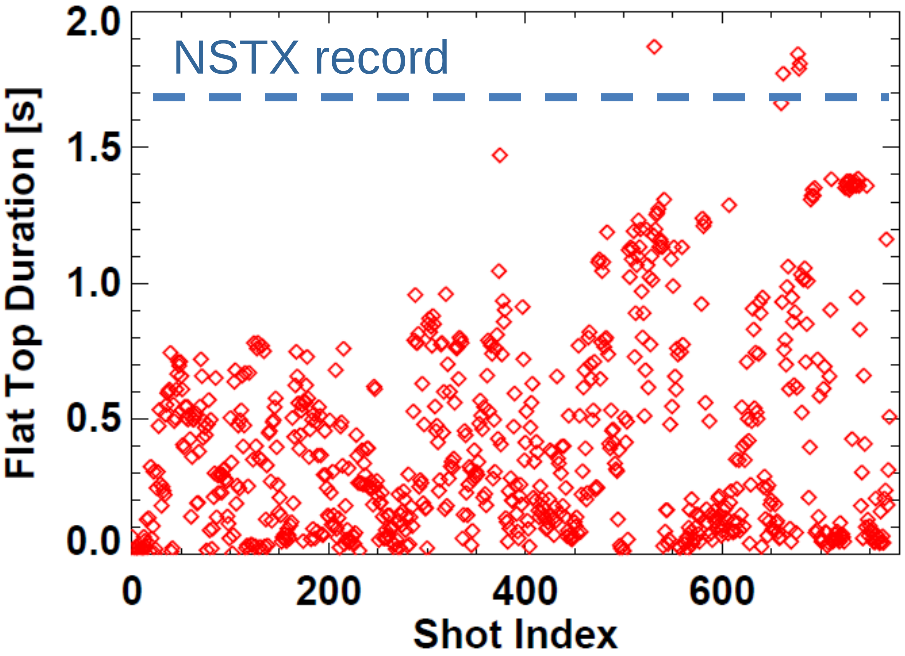


H-mode access achieved during first 2 weeks of operation

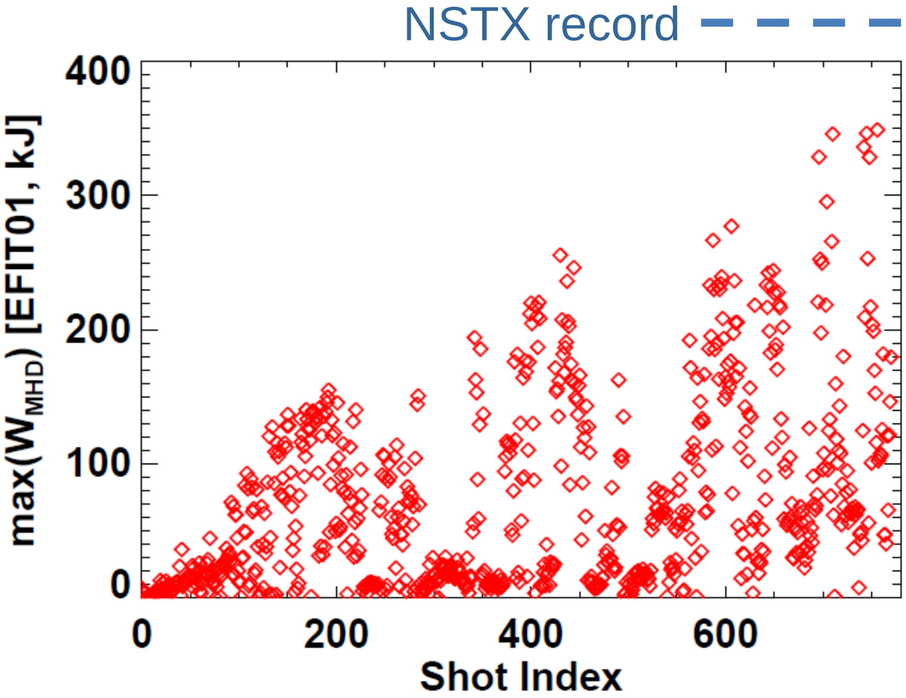
NSTX-U plasma commissioning status

- ~9 run weeks of ops, I_P = 0.5–1MA, boronized PFCs
- Nearly all shots at B_T = 0.6-0.65T > NSTX max = 0.55T
- All 6 NBI injected into plasmas, 5-10MW routinely available

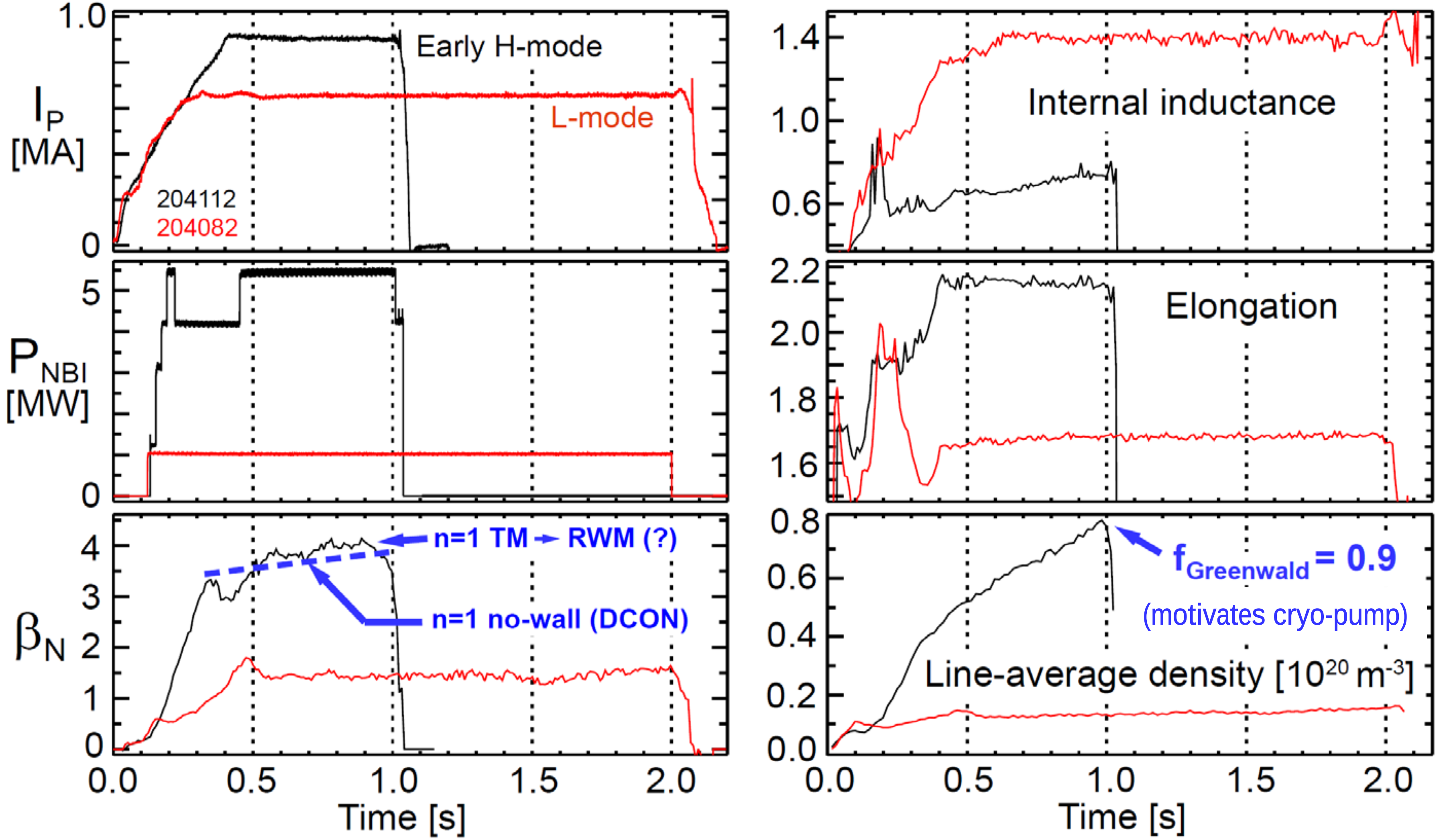
Exceeded NSTX L & H-mode pulse-lengths using 1MW L-modes (0.65, 0.8MA)



Approaching NSTX W_{TOT} record ~ 430kJ



Examples of high-performance L and H-modes achieved thus far in NSTX-U



NEXT: Increase κ to avoid tearing, active RWM control, trigger ELMs (ΔR_{sep}, granule injector)

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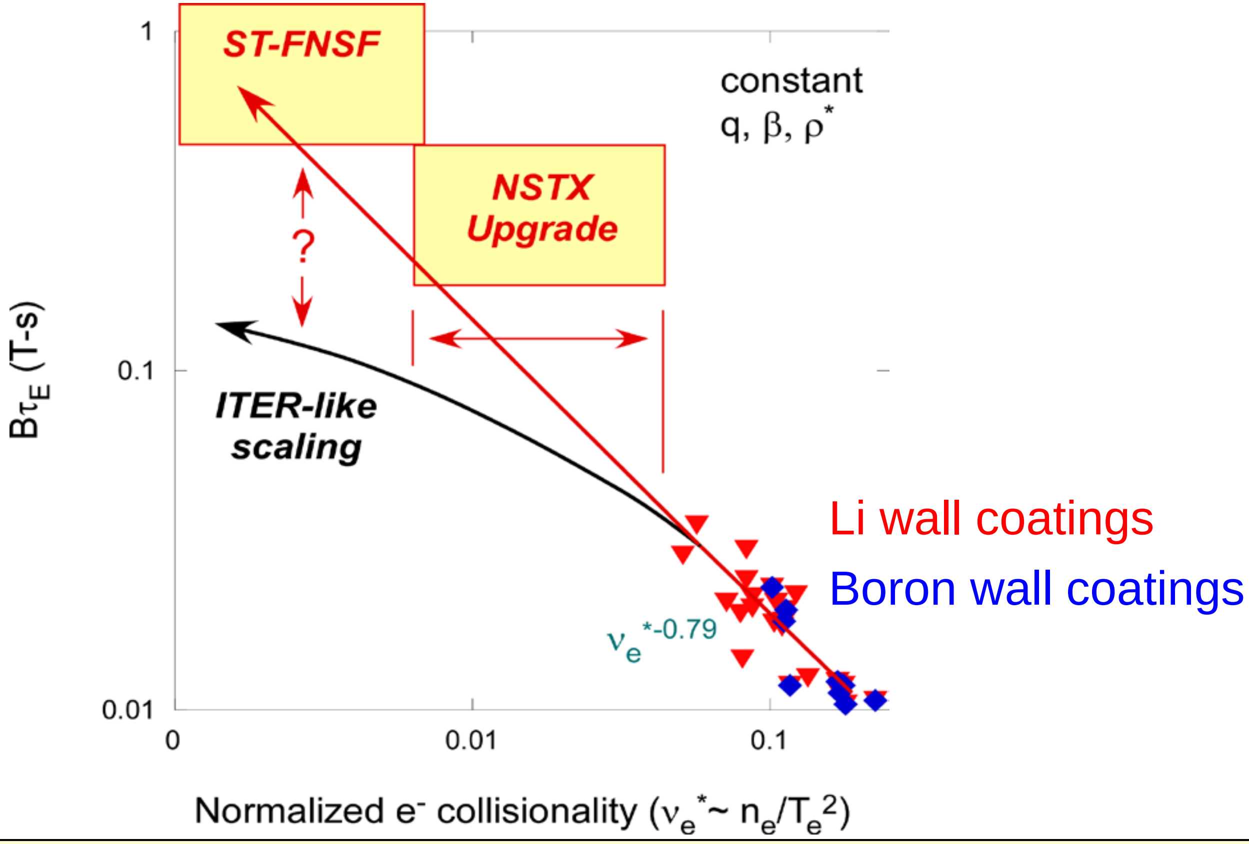
Favorable confinement trend with collisionality and β found in ST experiments

ST scaling observed in NSTX and MAST: $\tau_{E,\,th} \propto \nu_{*e}^{-0.8}\ \beta^{-0.0}$

Tokamak empirical scaling (ITER 98y,2): $\tau_{E,\,th} \propto \nu_{*e}^{-0.1}\ \beta^{-0.9}$



Promising scaling to ST-FNSF / Pilot, will trend continue on NSTX-U / MAST-U?

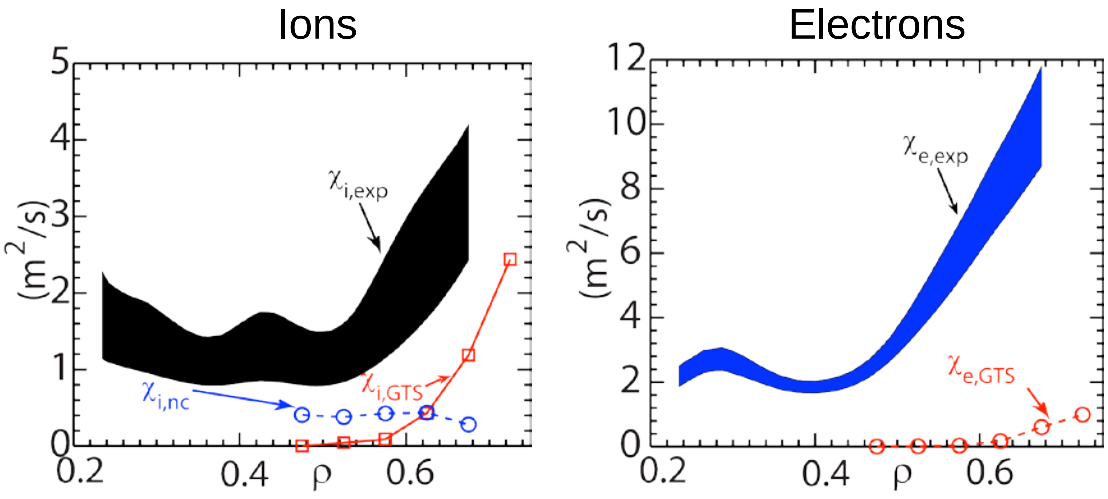
NSTX: Global non-linear GTS gyrokinetic simulations have identified multiple low-k turbulence transport mechanisms

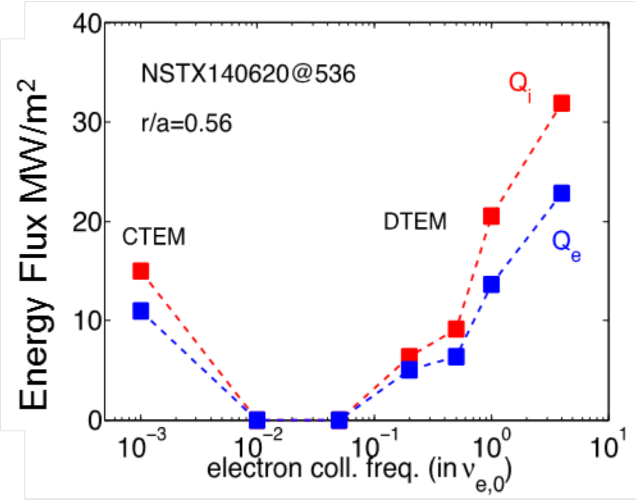
Strong flow shear can destabilize Kelvin-Helmholtz instability

- Non-linear global GTS simulations
- K-H + ITG + Neo ion transport within factor of 2 of expt'l level
- Cannot account for electron transport

Recent GTS simulations have shown possible role of DTEM in contributing to observed favorable collisionality scaling ($B\tau_{th} \sim \nu_{*e}^{-0.8}$)

- In addition to microtearing
- Synergy with DIII-D work

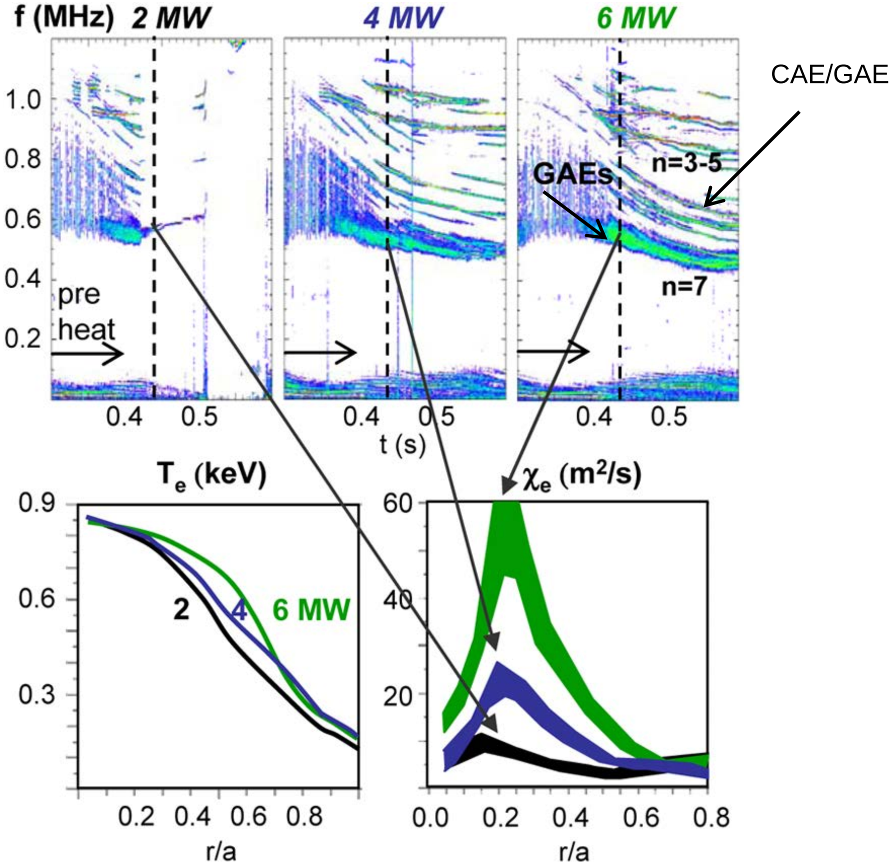




W. Wang et al., NF Letters, PoP (2015)

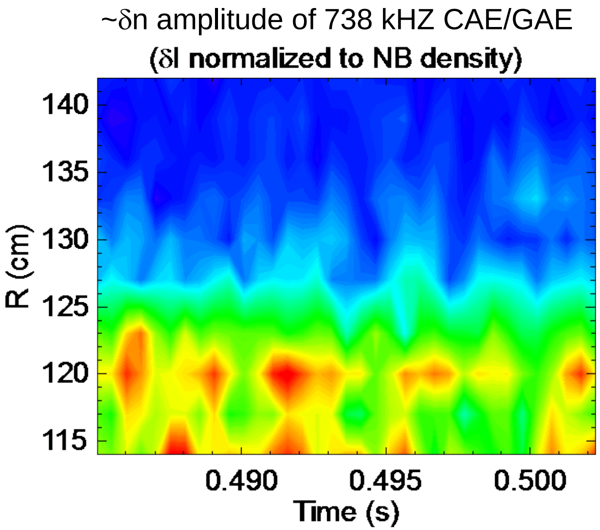
NSTX: Large inferred anomalous core electron transport in presence of CAE/GAEs

- Observation of high frequency Compressional/Global Alfven Eigenmodes (CAE/GAE) modes in plasma core associated with flattening of T_e profile (Stutman, Tritz - JHU)
  - High level of transport (10-100 m^2/s) inferred assuming classical beam physics



Stutman

BES spectra show mode amplitudes peaking from R=115 to 120 cm (r/a~0.2 to 0.3), in region of enhanced transport


~δn amplitude of 738 kHZ CAE/GAE

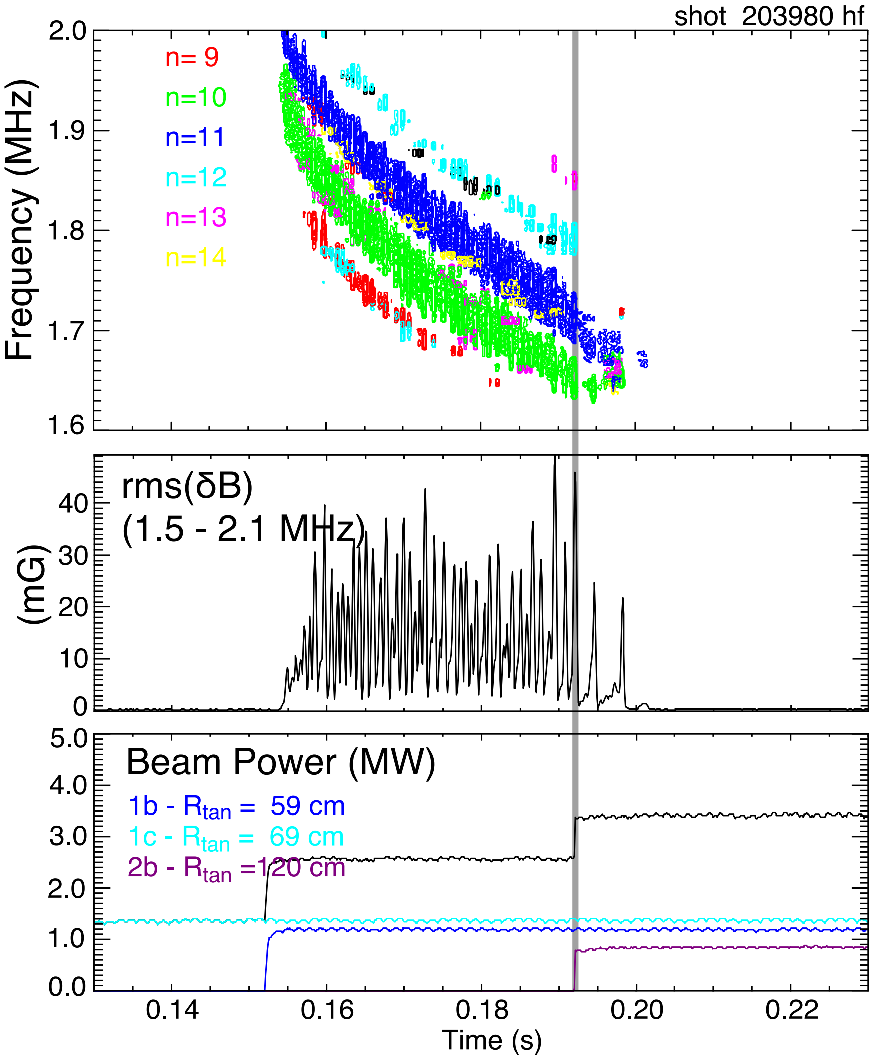
(δI normalized to NB density)

Smith

Is enhanced transport the full picture?

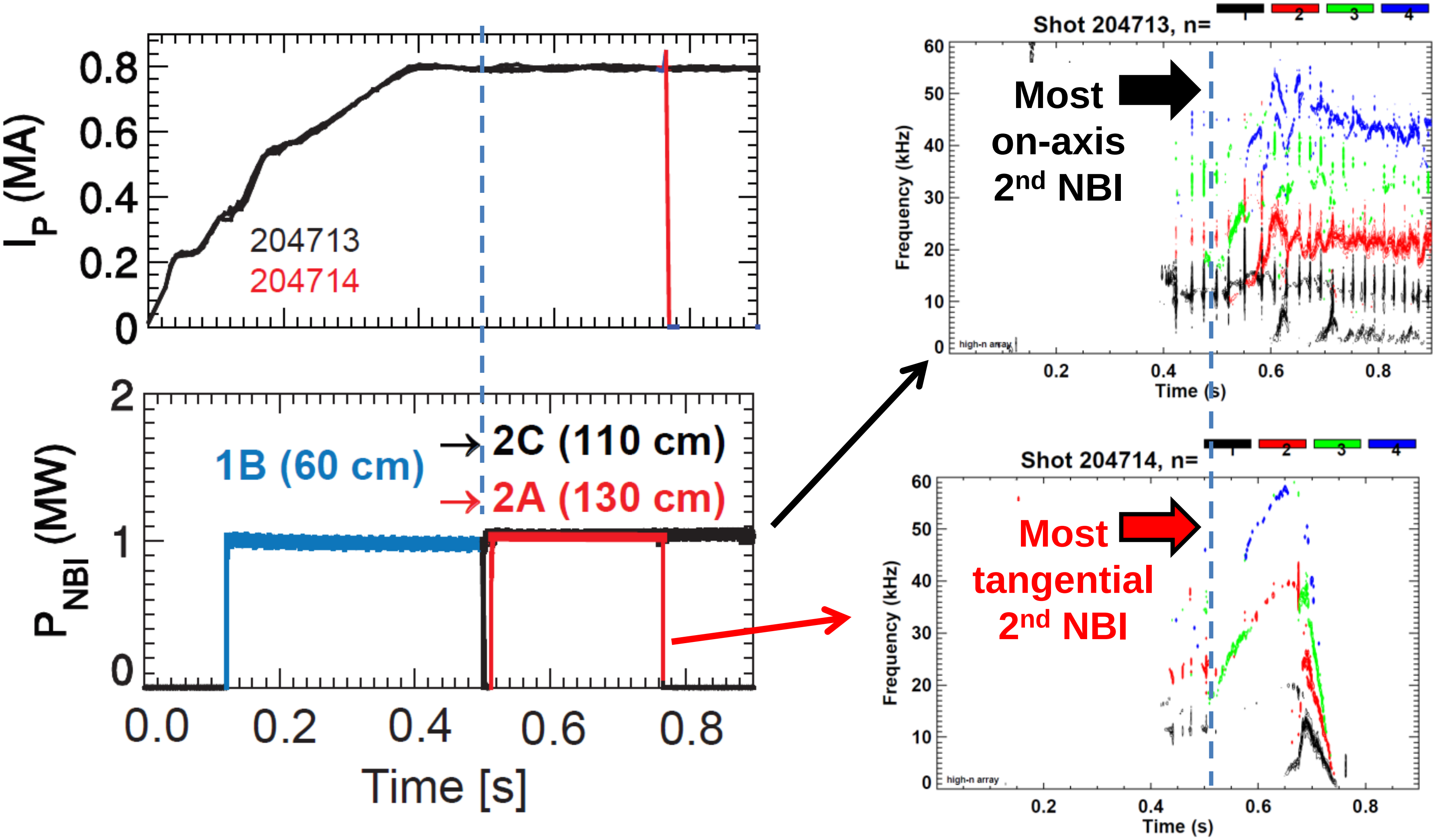
New NSTX-U result: suppression of counter-propagating GAE observed for large R_{TAN} 2nd NBI

- Top panel: GAE excited by inboard sources 1B / 1C
- Injection of new outboard source 2B starts at 0.192s → suppression of GAE
  - Suppression also with 2A, 2C
- Observations consistent with model of cyclotron-resonant drive of GAE
- Will investigate whether GAE absence impacts electron thermal transport



→ **2nd NBI already powerful new tool for Fast Ion and AE physics**

New 2nd NBI: L-mode core sawtooth and tearing dynamics change with source tangency radius



Most on-axis 2nd NBI

Most tangential 2nd NBI

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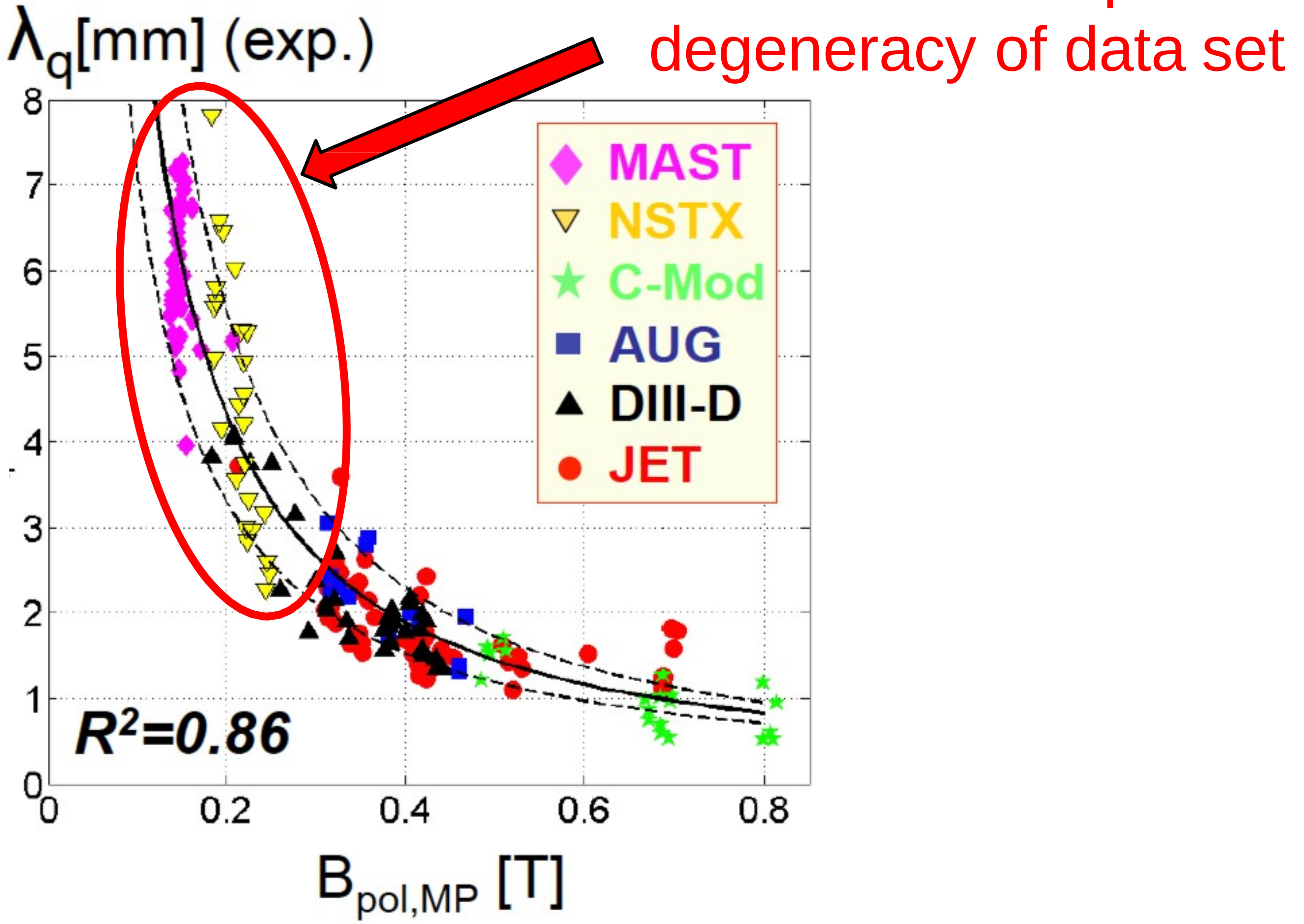
Dedicated tokamak + ST experiments found power exhaust width varies as 1 / $B_{poloidal}$



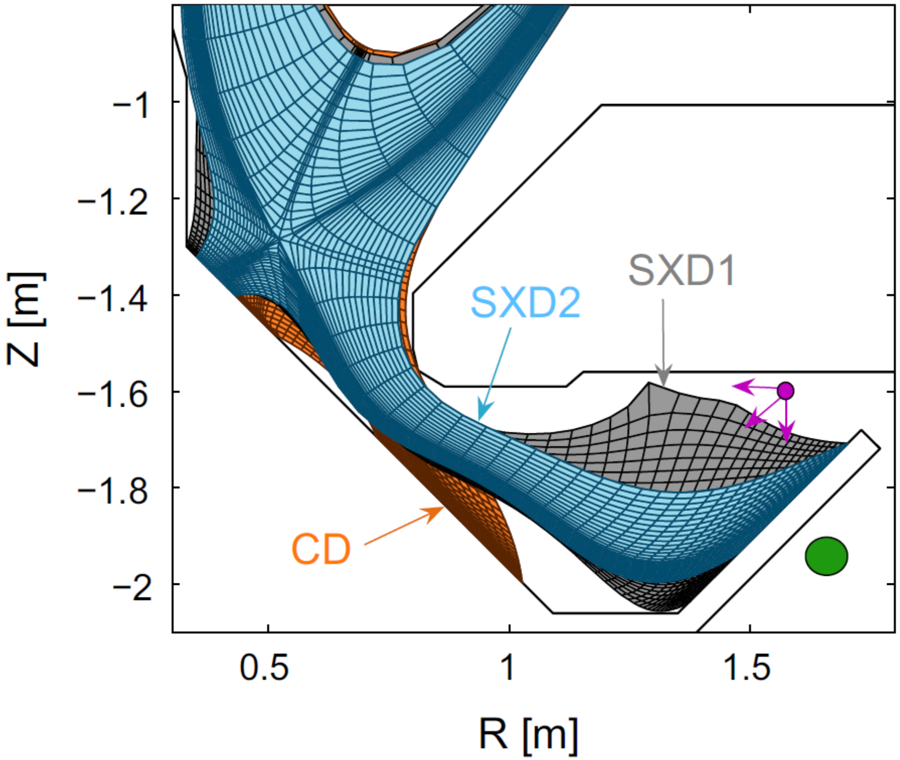
Will 1/$B_{poloidal}$ variation continue at higher I_P? What about detached conditions?

MAST-U and NSTX-U will test radiation and advanced divertors for mitigating high heat-fluxes

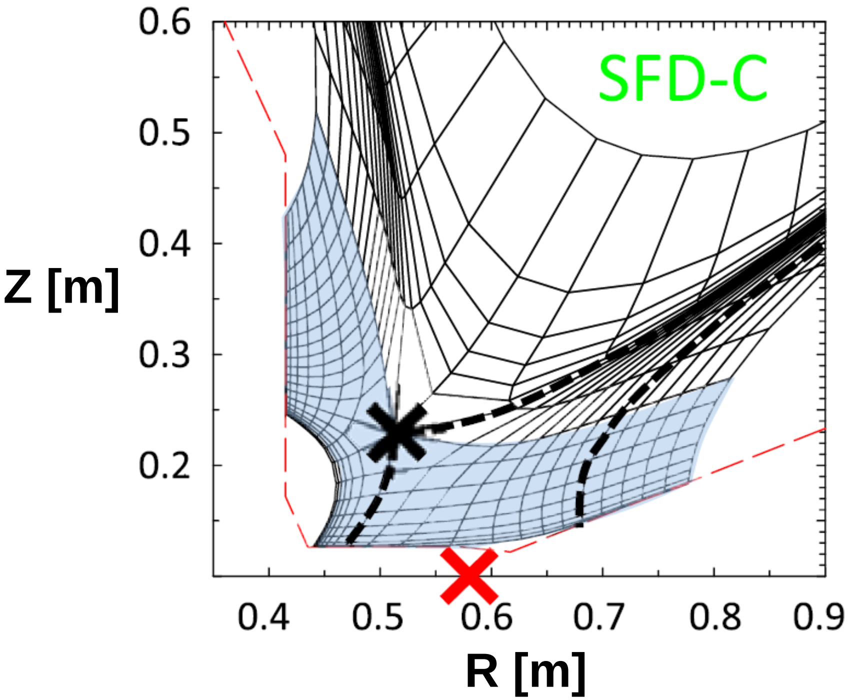
- Assess impact of flux expansion, line length on detachment

- MAST-U: Conventional (CD), Snowflake (SFD), Super-X (SXD)

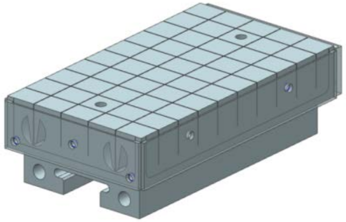
Divertor geometry



- NSTX-U: Snowflake (SFD) / X-D



- Longer-term:
  C PFCs→ Liquid Li on high-Z substrates



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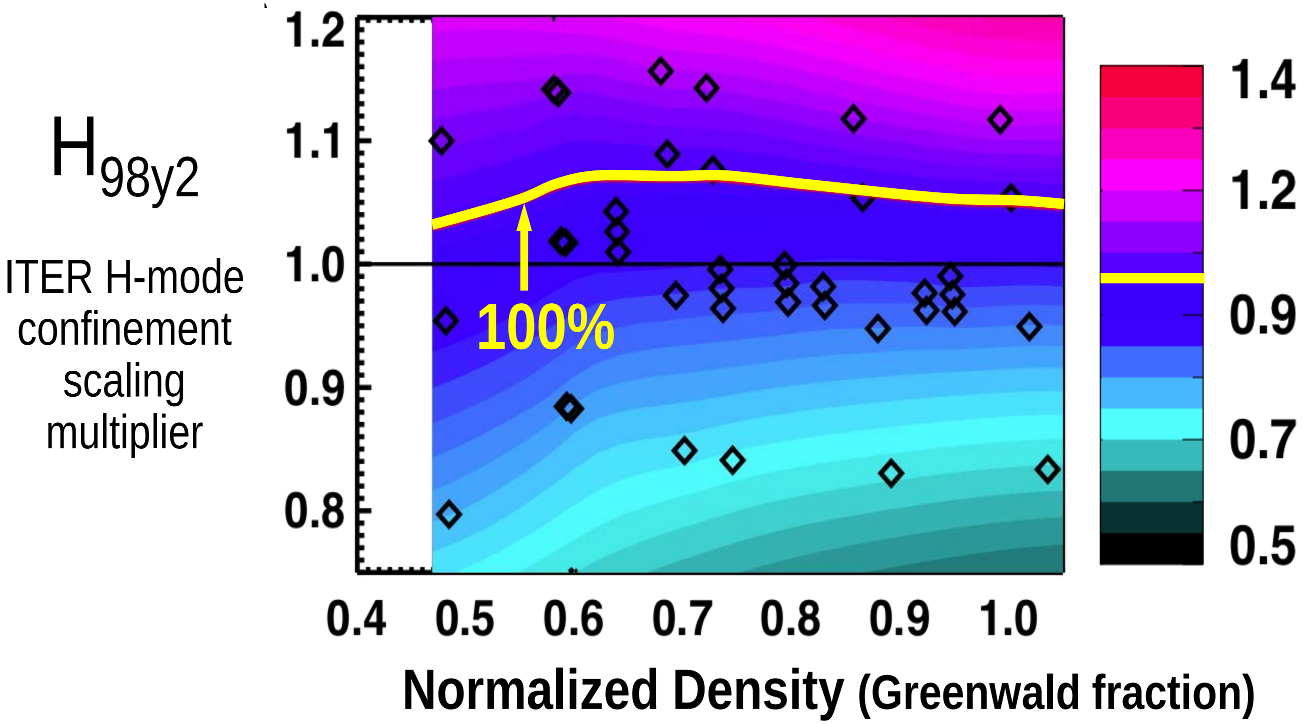
Steady-state operation required for ST/AT FNSF or Pilot Plant

NSTX achieved 70% “transformer-less” current drive

NSTX-U designed to achieve 100% (TRANSP):



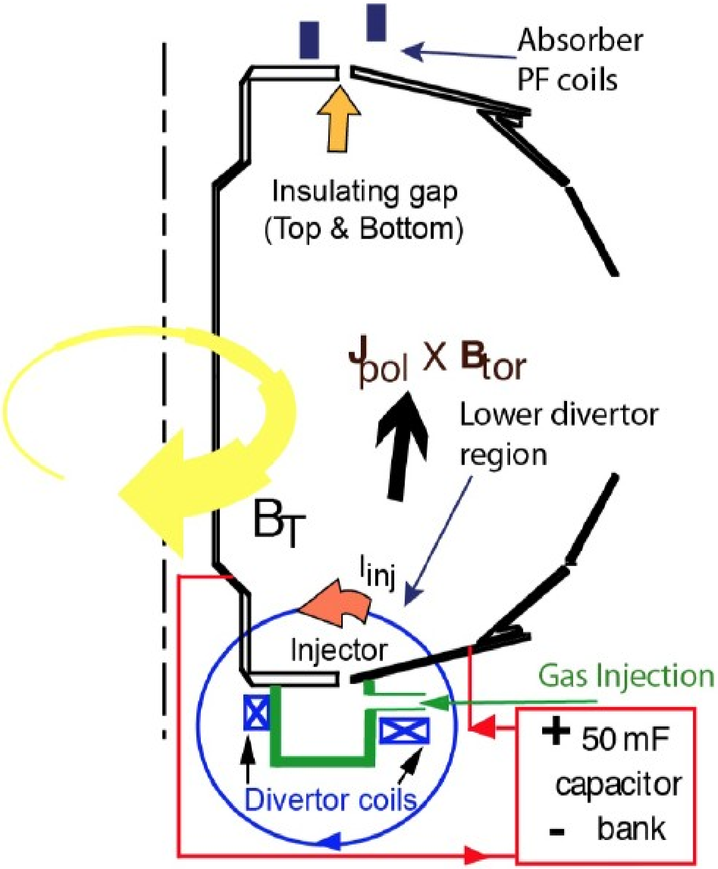
I_P=1 MA, B_T=1.0 T, P_{NBI}=12.6 MW

Will NSTX-U achieve 100% as predicted by simulations?

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

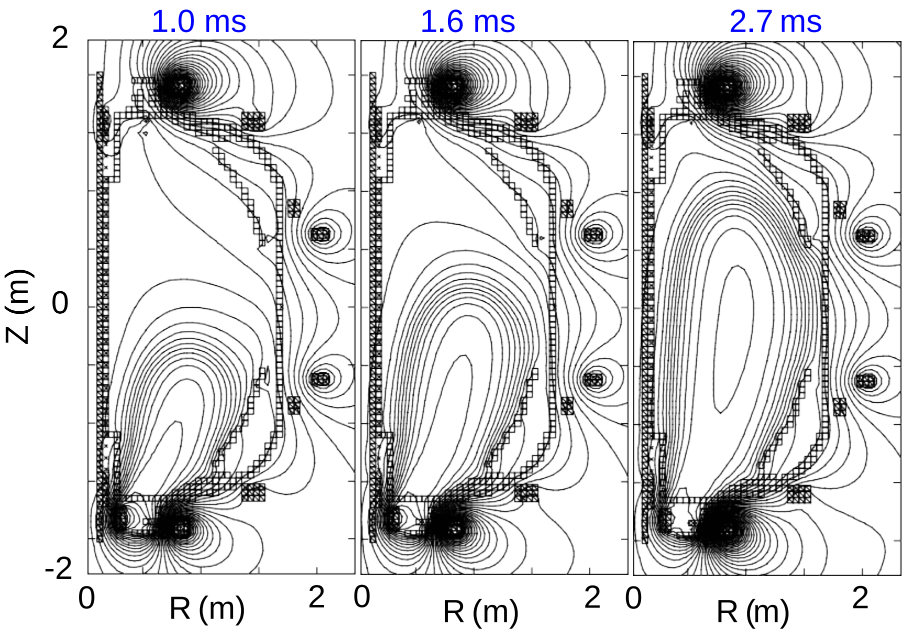


R. Raman et al., PRL 2006

NSTX: 150-200kA closed flux current

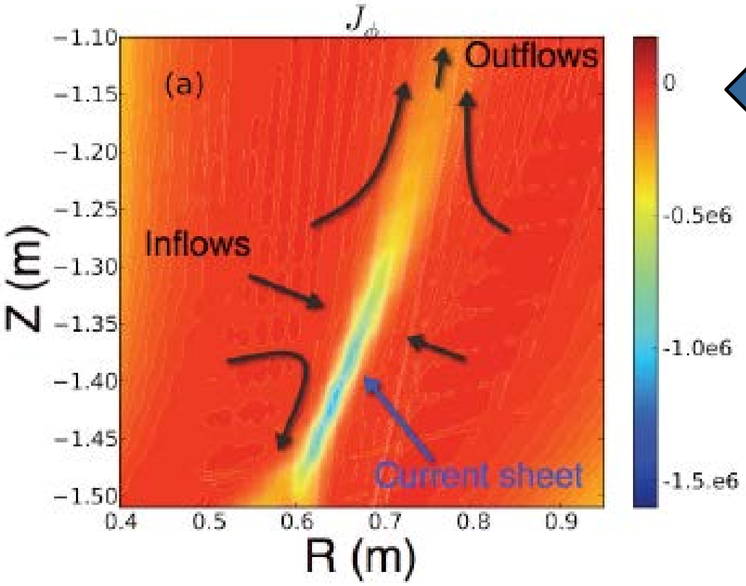
NSTX-U: CHI projects to 300-400kA
FNSF: CHI blanket electrodes: 2MA

TSC axisymmetric
simulation of CHI startup

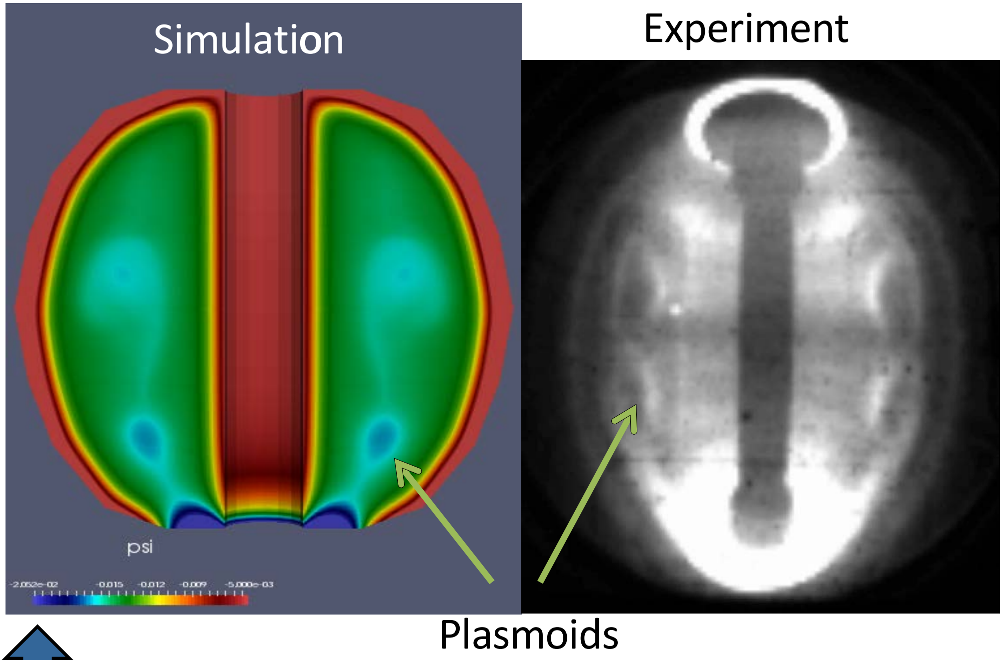


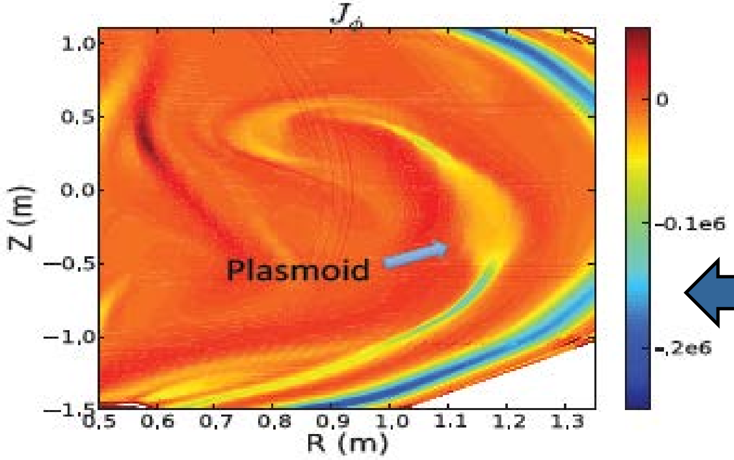
What about 3D effects?

NIMROD simulations: plasmoid-mediated reconnection assists flux closure at high Lundquist number



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





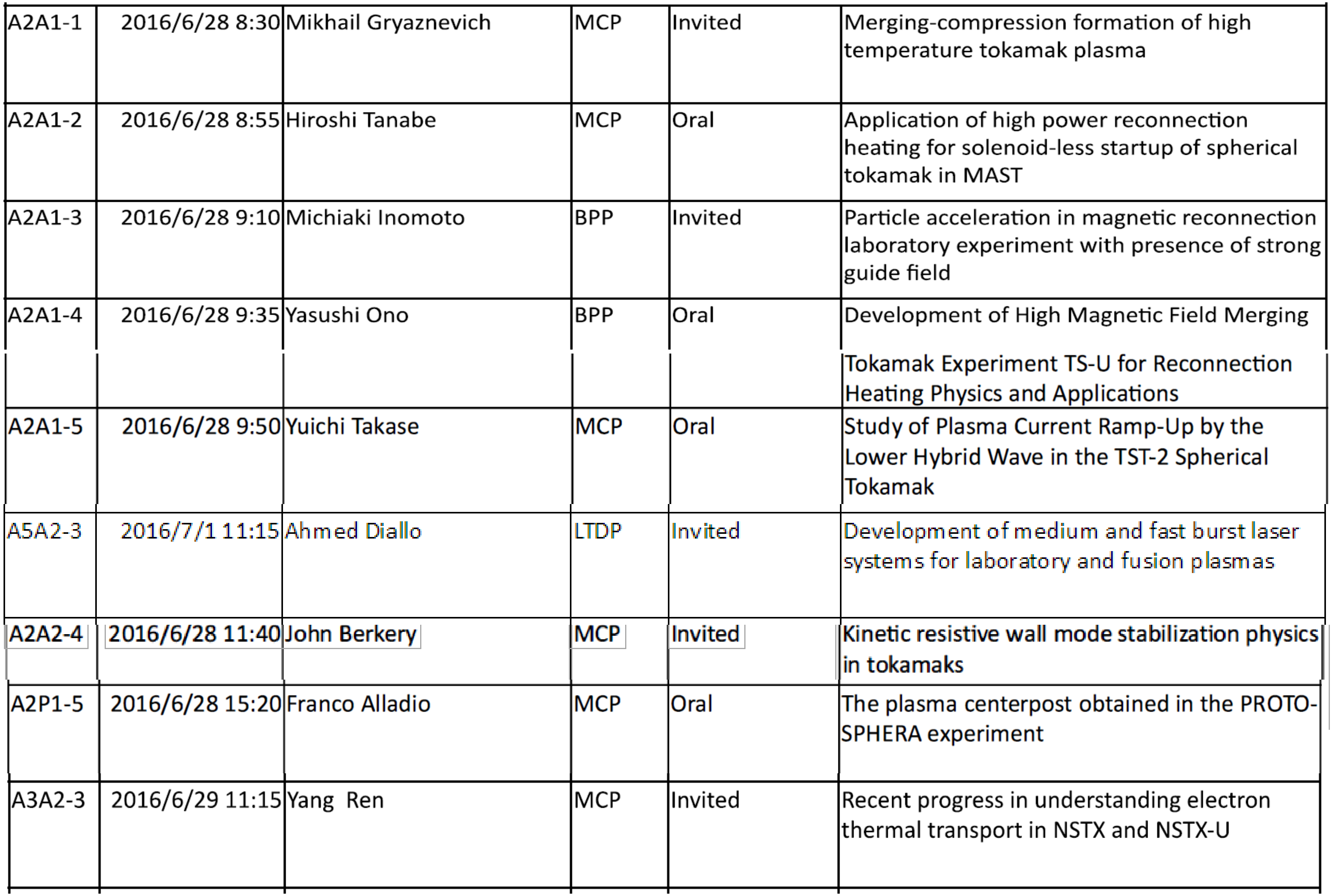
F. Ebrahimi, et al., PRL (2015)

- At high Lundquist number the S-P current sheet is plasmoid unstable
- **Plasmoids seen in modelling before found/observed in CHI data!**

Summary: ST research making leading contributions to fusion energy development

- Support burning plasma research by expanding understanding of energetic particle, thermal transport
- Will explore performance and implications of advanced divertor configurations, liquid metals
- ST promising as Fusion Nuclear Science Facility
  - High J_{WP} of HTS → enables compact lower-A Pilot Plant
- NSTX-U operational now!
- MAST-U first plasma next year!

See more ST-related talks this week

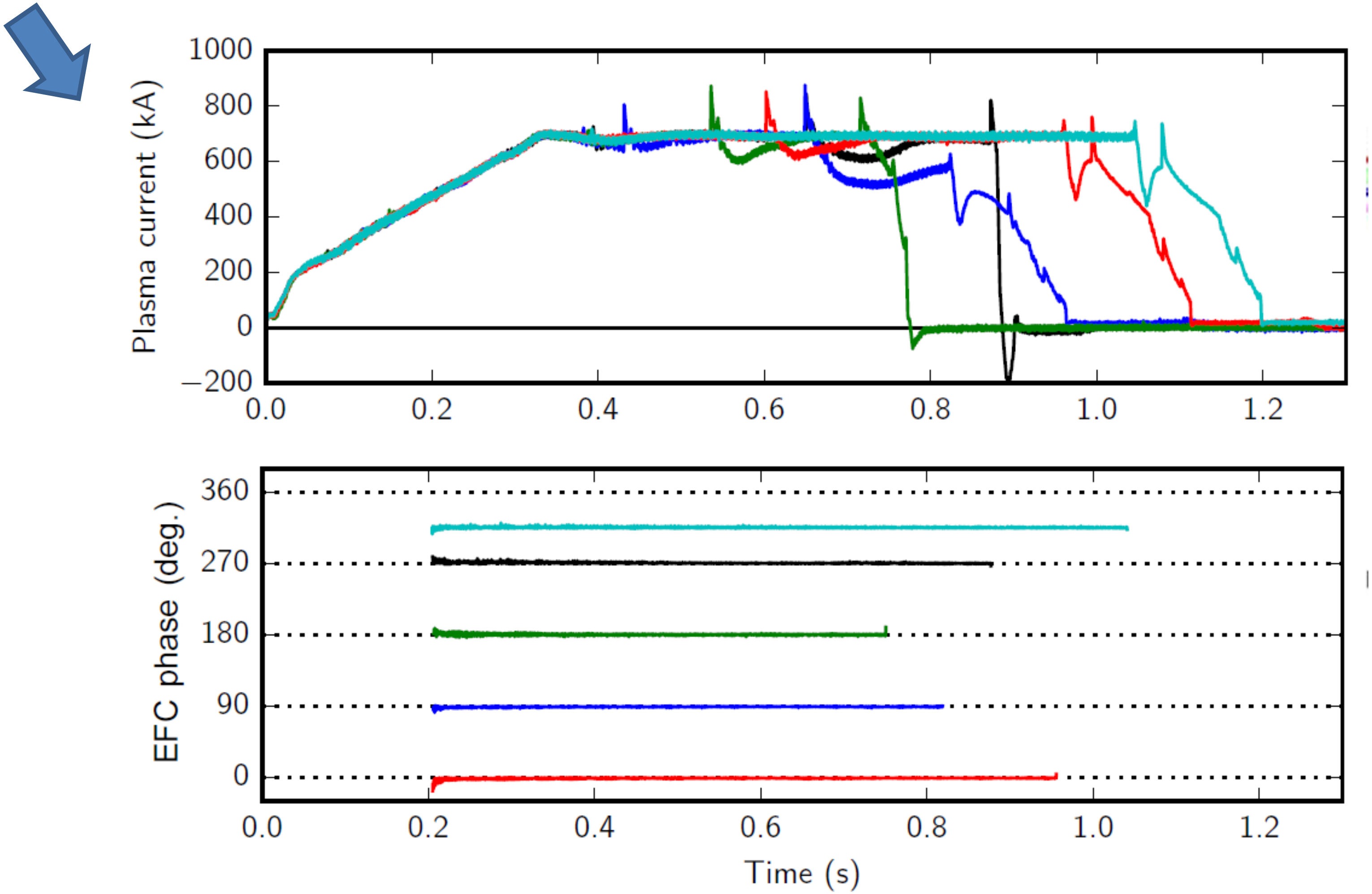


A2A1-1	2016/6/28 8:30	Mikhail Gryaznevich	MCP	Invited	Merging-compression formation of high temperature tokamak plasma
A2A1-2	2016/6/28 8:55	Hiroshi Tanabe	MCP	Oral	Application of high power reconnection heating for solenoid-less startup of spherical tokamak in MAST
A2A1-3	2016/6/28 9:10	Michiaki Inomoto	BPP	Invited	Particle acceleration in magnetic reconnection laboratory experiment with presence of strong guide field
A2A1-4	2016/6/28 9:35	Yasushi Ono	BPP	Oral	Development of High Magnetic Field Merging Tokamak Experiment TS-U for Reconnection Heating Physics and Applications
A2A1-5	2016/6/28 9:50	Yuichi Takase	MCP	Oral	Study of Plasma Current Ramp-Up by the Lower Hybrid Wave in the TST-2 Spherical Tokamak
A5A2-3	2016/7/1 11:15	Ahmed Diallo	LTDP	Invited	Development of medium and fast burst laser systems for laboratory and fusion plasmas
A2A2-4	2016/6/28 11:40	John Berkery	MCP	Invited	Kinetic resistive wall mode stabilization physics in tokamaks
A2P1-5	2016/6/28 15:20	Franco Alladio	MCP	Oral	The plasma centerpost obtained in the PROTO-SPHERA experiment
A3A2-3	2016/6/29 11:15	Yang Ren	MCP	Invited	Recent progress in understanding electron thermal transport in NSTX and NSTX-U

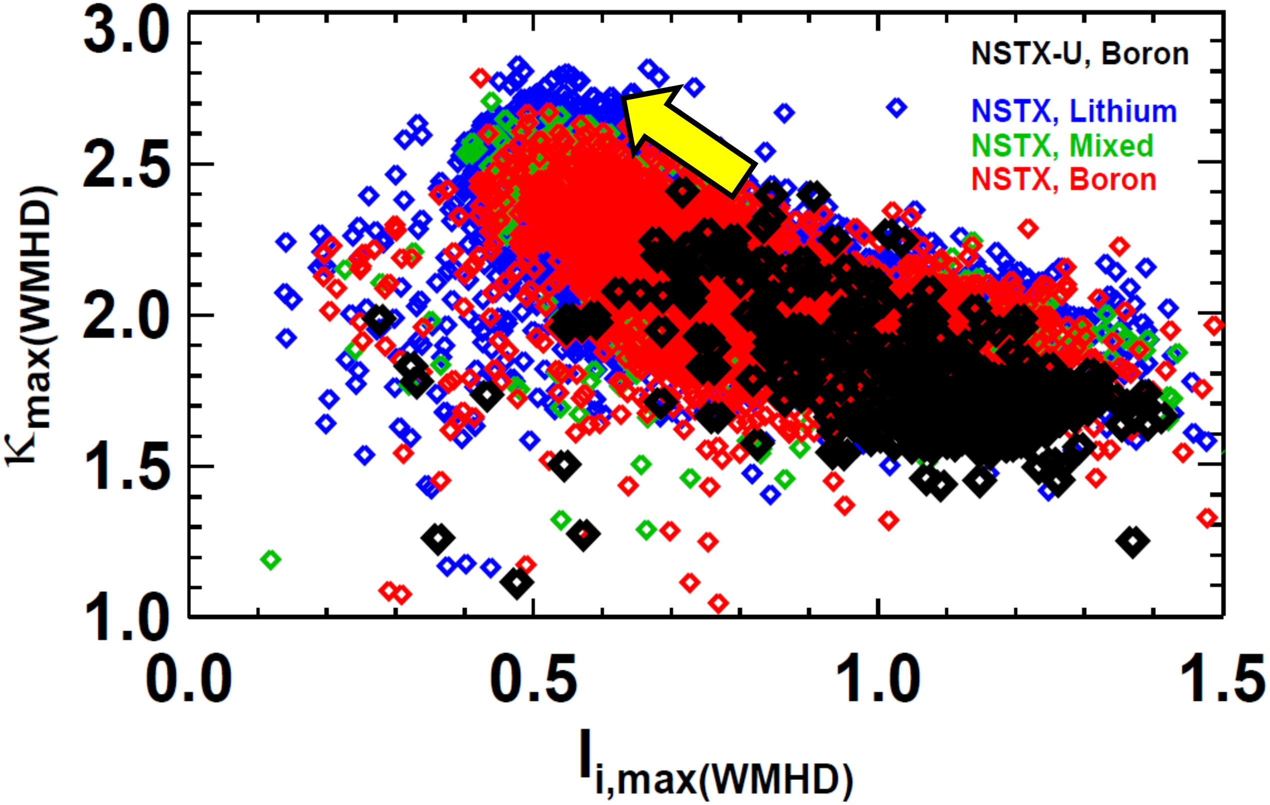
Backup Slides - Physics

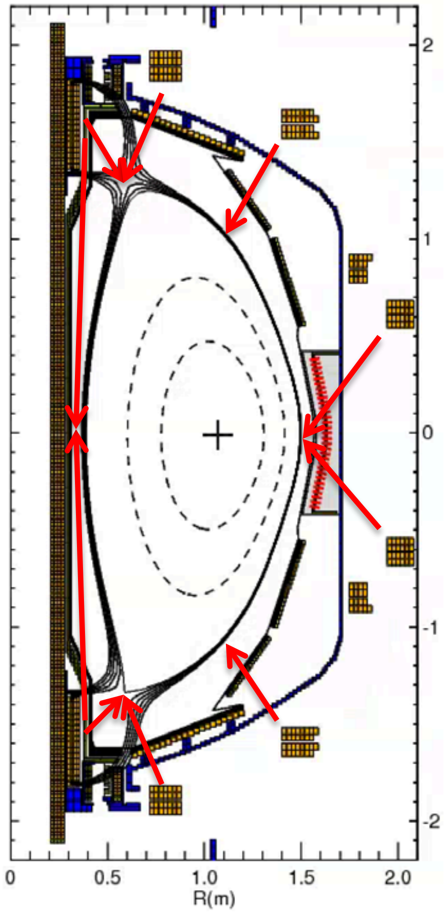
Optimal n=1 error field correction amplitude and phase identified to maximize pulse length, discharge performance

- Dominant error-field source: PF5 vertical field coils
- Long-pulse L-modes used to identify optimal correction amplitude, phase



On path to high I_P without tearing modes by elevating q_{min} with early heating + H-mode → l_i=0.5-0.6, κ=2.5-2.7





- Matching NSTX κ at same l_i but at higher A
  - Real-time EFIT / ISOFLUX (GA collaboration)
  - Also utilizing improved vertical motion detection

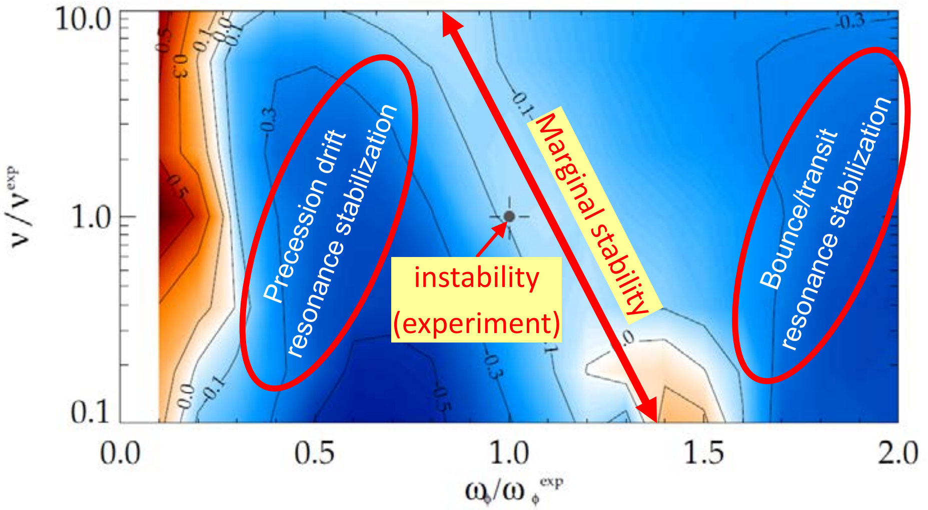
NSTX-Theory collaborations have led the advancement in kinetic global mode stability physics

RWM High-beta stability physics theory

$$\gamma\tau_w \simeq -\frac{\delta W_\infty + \delta W_K}{\delta W_b + \delta W_K}$$

(Betti, Berkery, Sabbagh)

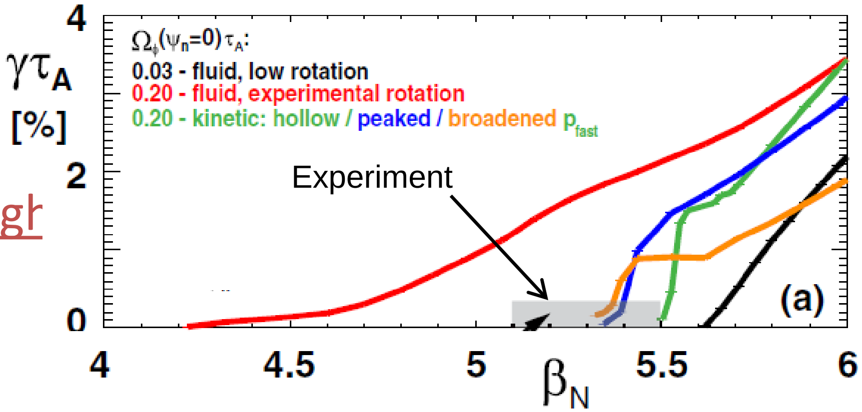
MISK: RWM High-beta stability (Berkery, Sabbagh



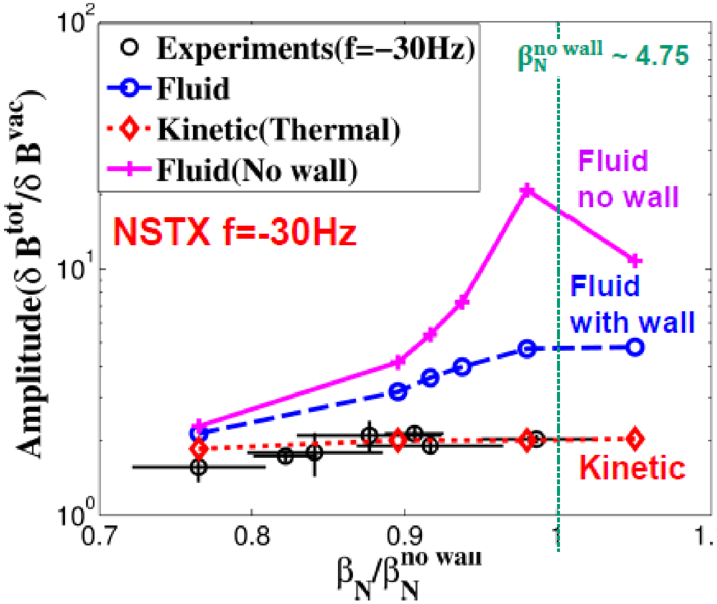
RWM physics with M3D-C^1 (Ferraro)

- Resistive wall implemented

MARS-K: Ideal wall stability (Menard)

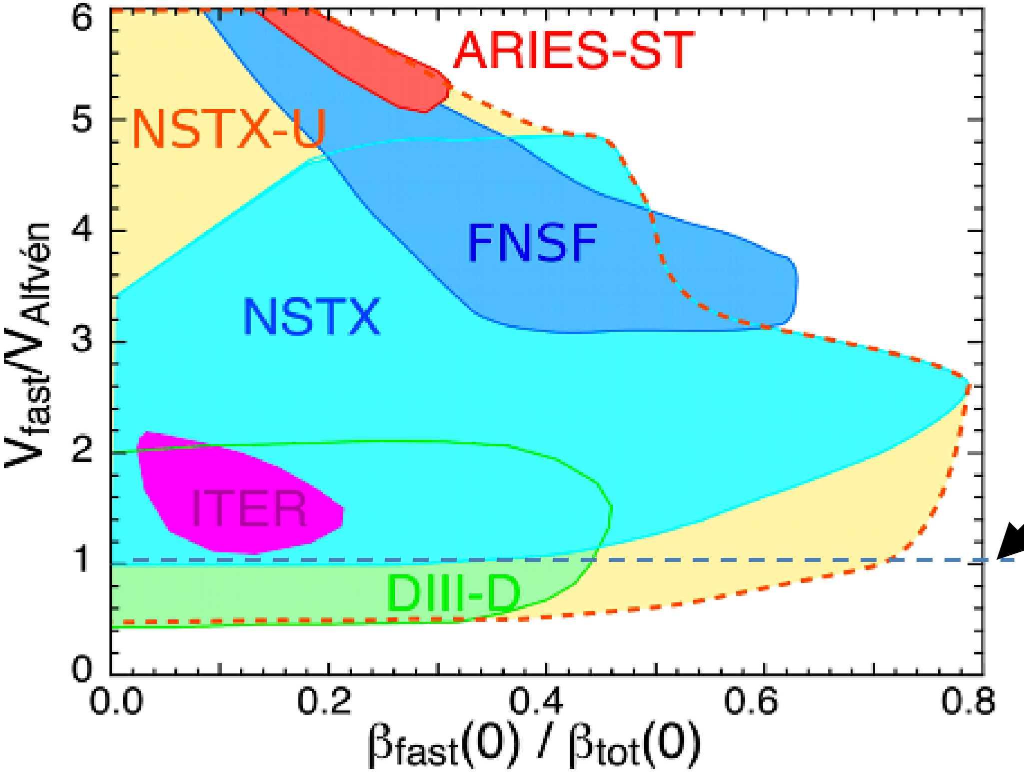


MARS-K: Resonant field amp. (Wang)



NBI-heated STs excellent testbed for α-particle physics

- α-particles couple to Alfvénic modes when $V_\alpha > V_{Alfvén} \sim \beta^{-0.5} C_{sound}$
- $V_{fast} > V_A$ condition easily satisfied in high-β ST with NBI heating

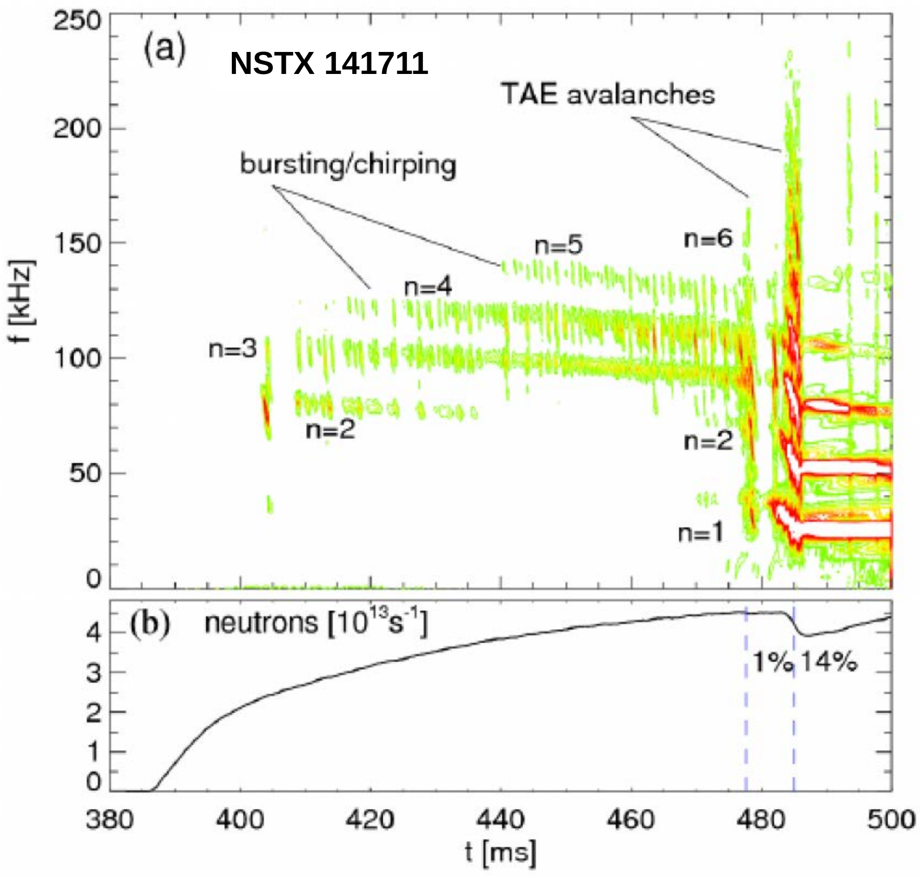


- NSTX-U: large fast-ion dynamic range spanning ST and conventional A
  - **Toroidal field 2× NSTX →** $V_{fast} < V_A$ → stabilize modes
  - **Tangential 2nd NBI →** very flexible fast-ion distribution
    - Vary pitch angle, pressure profile

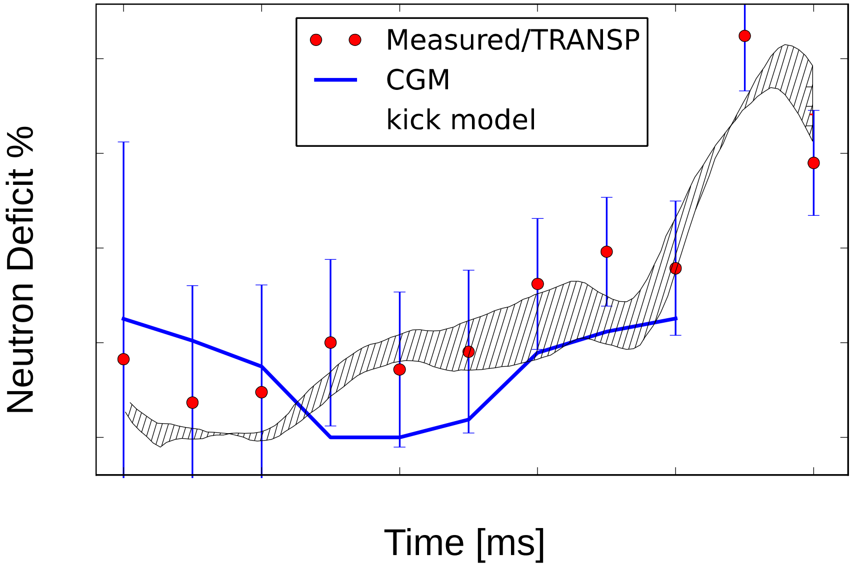
Can we find TAE-quiescent, high-performance regimes in NSTX-U?

"TAE avalanche" can cause energetic particle loss

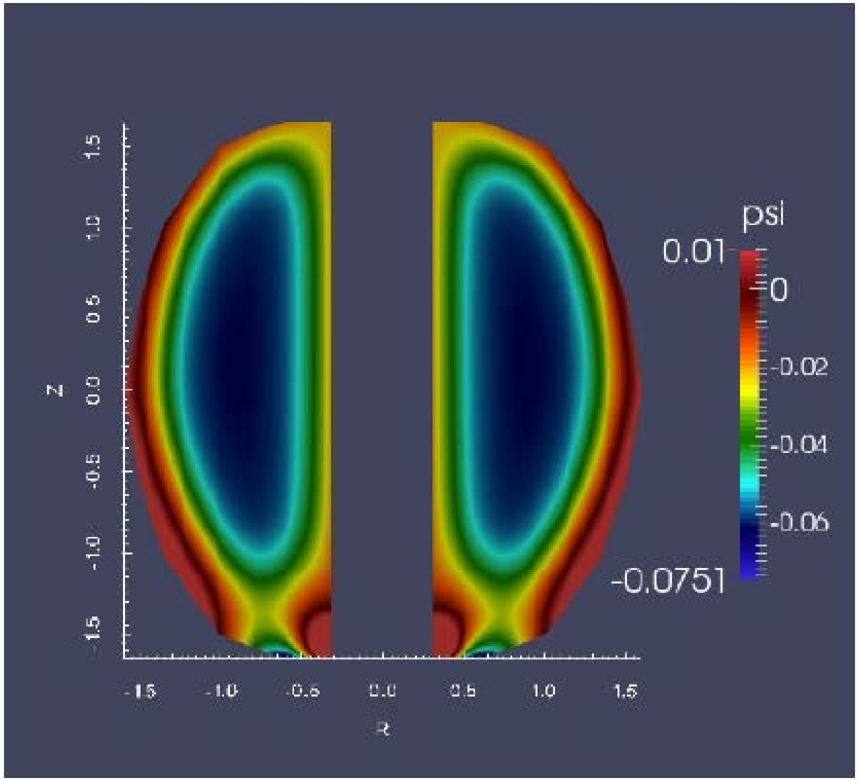
Uncontrolled α-particle loss could cause reactor first wall damage

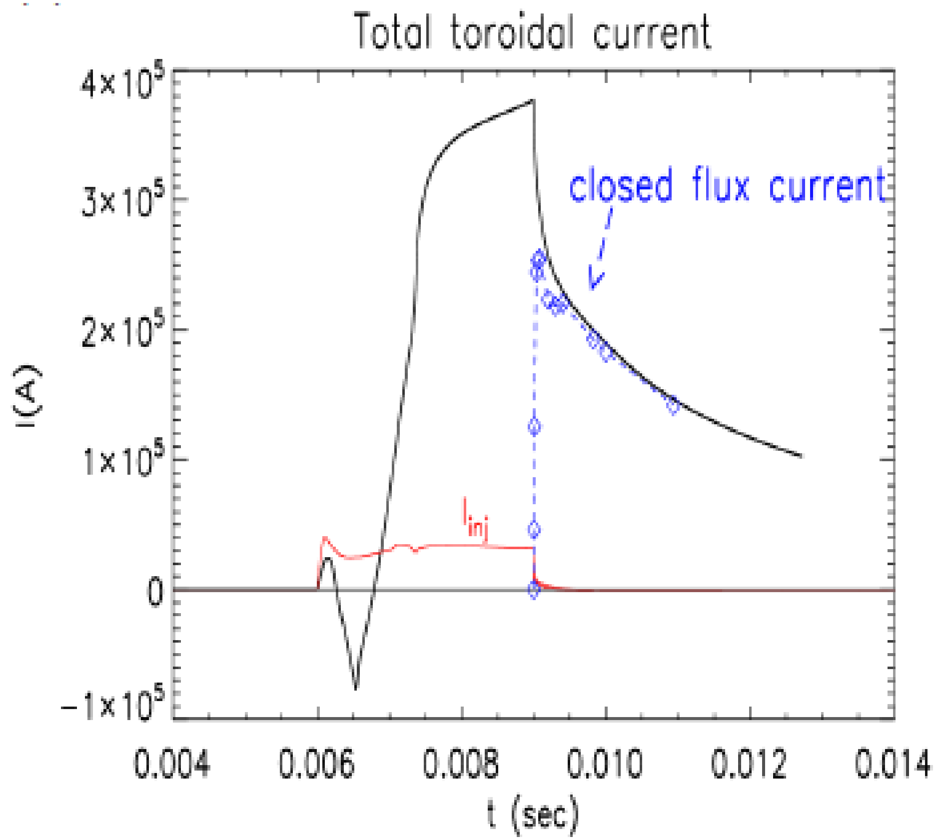


- Quasi-linear "Critical Gradient Model" (CGM) consistent with transport before avalanche
- "Kick" model (ΔP_ϕ vs ΔE) predicts neutron decrement even during large avalanche events



NSTX-U NIMROD projection: high fraction of open flux converted to closed flux with narrow injector flux footprint

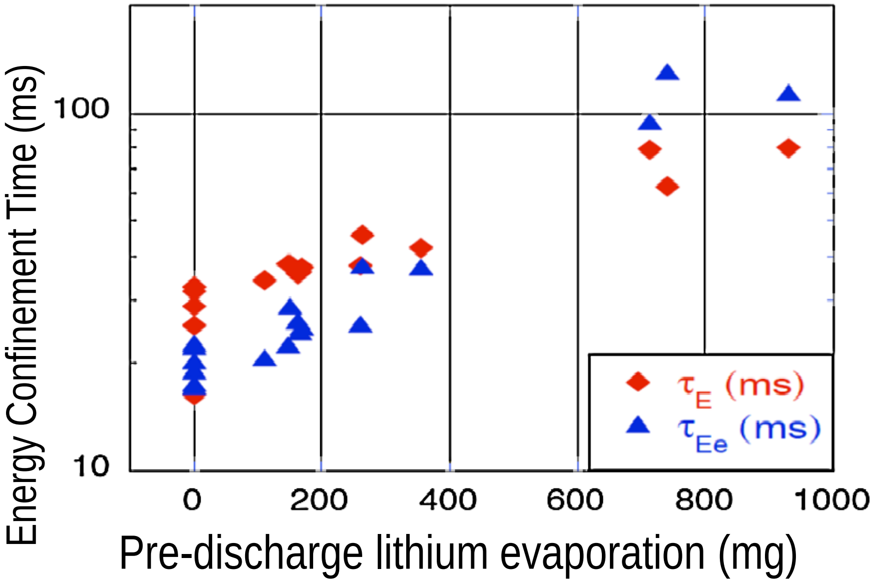




- CHI in NSTX-U configuration naturally has a narrower injector flux foot-print due to improved Injector coil positioning
- Due to higher Lundquist number in NSTX-U CHI simulations, closed flux surfaces form even during the actively injected phase

NIMROD Simulations **F. Ebrahimi, et al., accepted in NF Letters**

Plasma confinement increased continuously with increasing Li coatings in NSTX – what is limit?



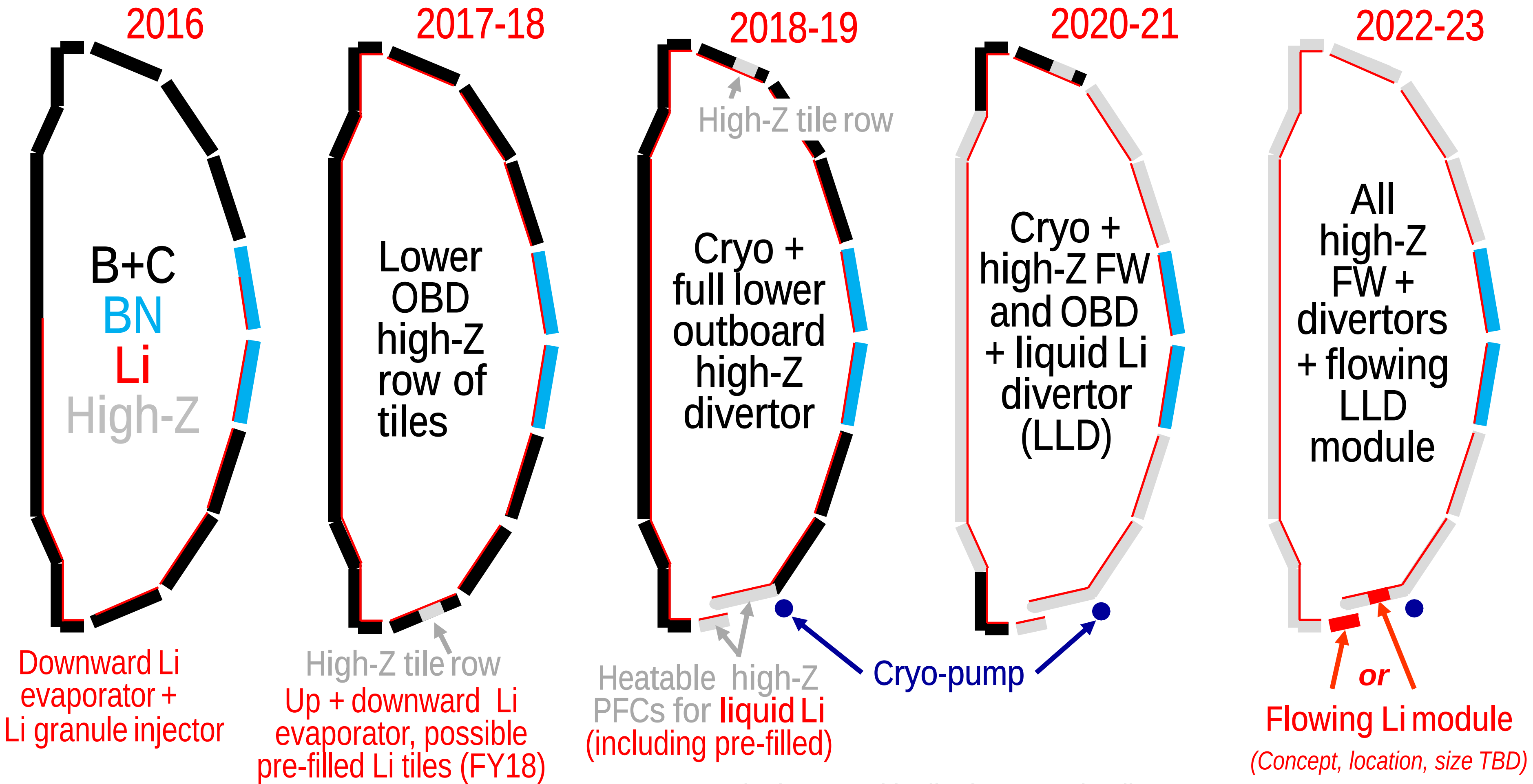
R. Maingi, et al., PRL 107 (2011) 145004

- Global parameters improve
  - H_{98y2} increases ~0.9 → 1.4
  - No core Li accumulation
- High H critical for compact FNSF / Pilot Plants

- NSTX-U will double Li-wall coverage with upward evaporators
- Will further assess contributors to confinement improvement:
  - Lower-recycling / reduced neutral source / higher T_e
  - Edge profile / turbulence changes
  - Influence of (low-Z) impurities in pedestal region

NSTX-U boundary / PFC plan: add divertor cryo-pump, transition to high-Z wall, study flowing liquid metal PFCs

- **5yr goal: Integrate high τ_E and β_T with 100% non-inductive**
- **10yr goal: Assess compatibility with high-Z & liquid lithium PFCs**



2016

B+C
BN
Li
High-Z

Downward Li evaporator + Li granule injector

2017-18

Lower OBD high-Z row of tiles

High-Z tile row

Up + downward Li evaporator, possible pre-filled Li tiles (FY18)

2018-19

High-Z tile row

Cryo + full lower outboard high-Z divertor

Heatable high-Z PFCs for liquid Li (including pre-filled)

Cryo-pump

2020-21

Cryo + high-Z FW and OBD + liquid Li divertor (LLD)

2022-23

All high-Z FW + divertors + flowing LLD module

or

Flowing Li module

(Concept, location, size TBD)

See Maingi, Jaworski talks for more details

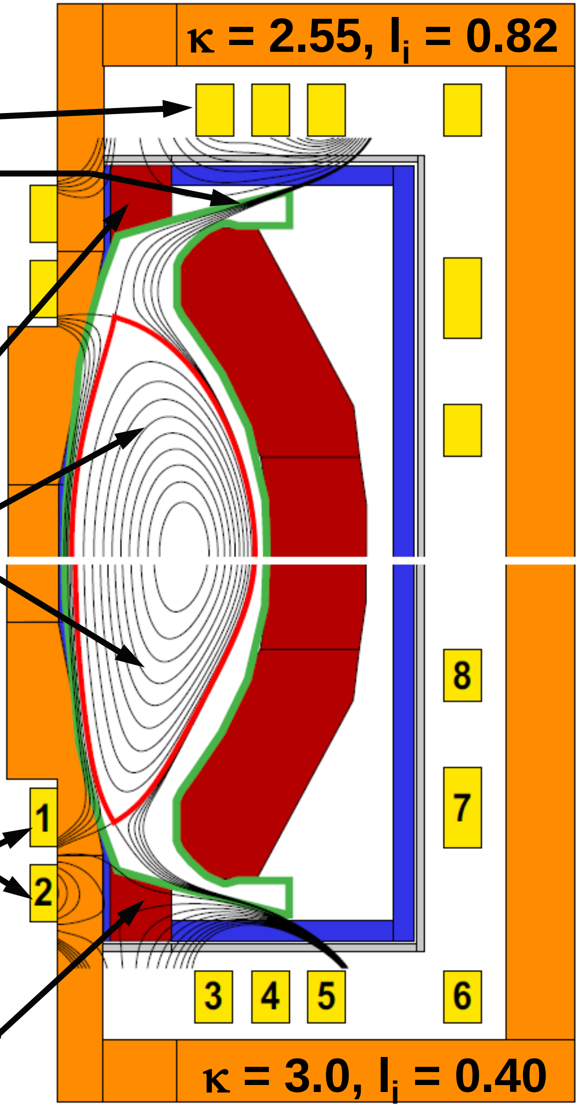
Backup Slides – Cu TF FNSF

Identified self-consistent configuration for power exhaust, equilibrium flexibility, breeding, maintenance

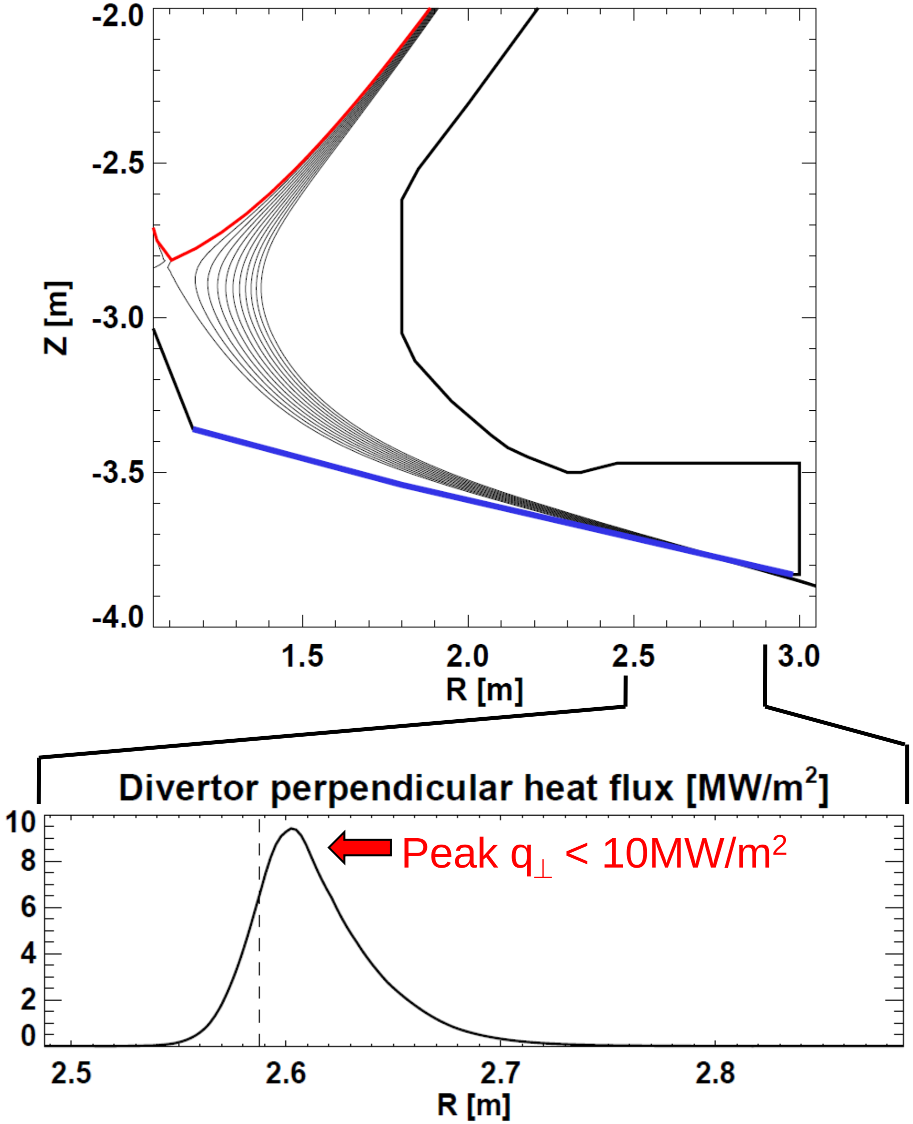
Components: TF coil PF coil Vessel Shield Blanket

- All equilibrium PF coils outside vacuum vessel
- Increased strike-point radius reduces B, $q_{||}$

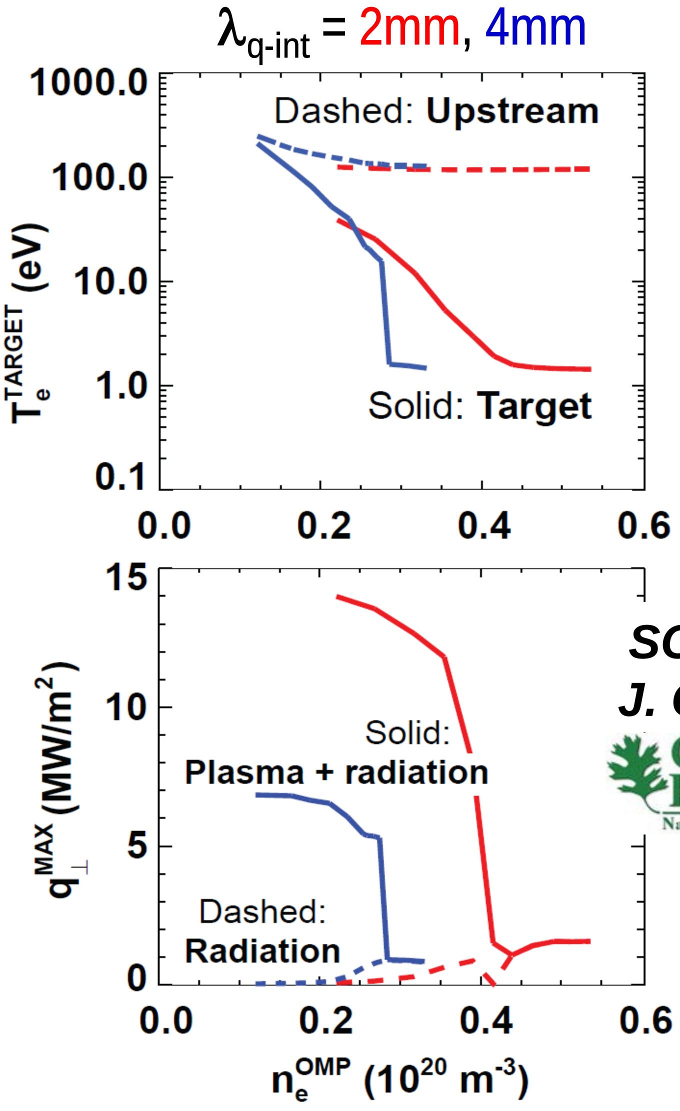
  Strike-point PFCs also shielded by blankets
- **2^{nd} X-point increases SOL line-length**
- **PF coil set supports wide range of l_i: 0.4-0.8**
  - Elongation and squareness change with l_i variation
  - Fixed strike-point R, controllable **B** angle of incidence (0.5-5˚)
- No central solenoid in this design
- Divertor coils in TF coil ends for equilibrium, high δ
- Breeding at top/bottom important for maximizing TBR



Long-leg divertor reduces heat flux 3×to ~10MW/m^2 Also promotes detachment → additional 5-10×reduction

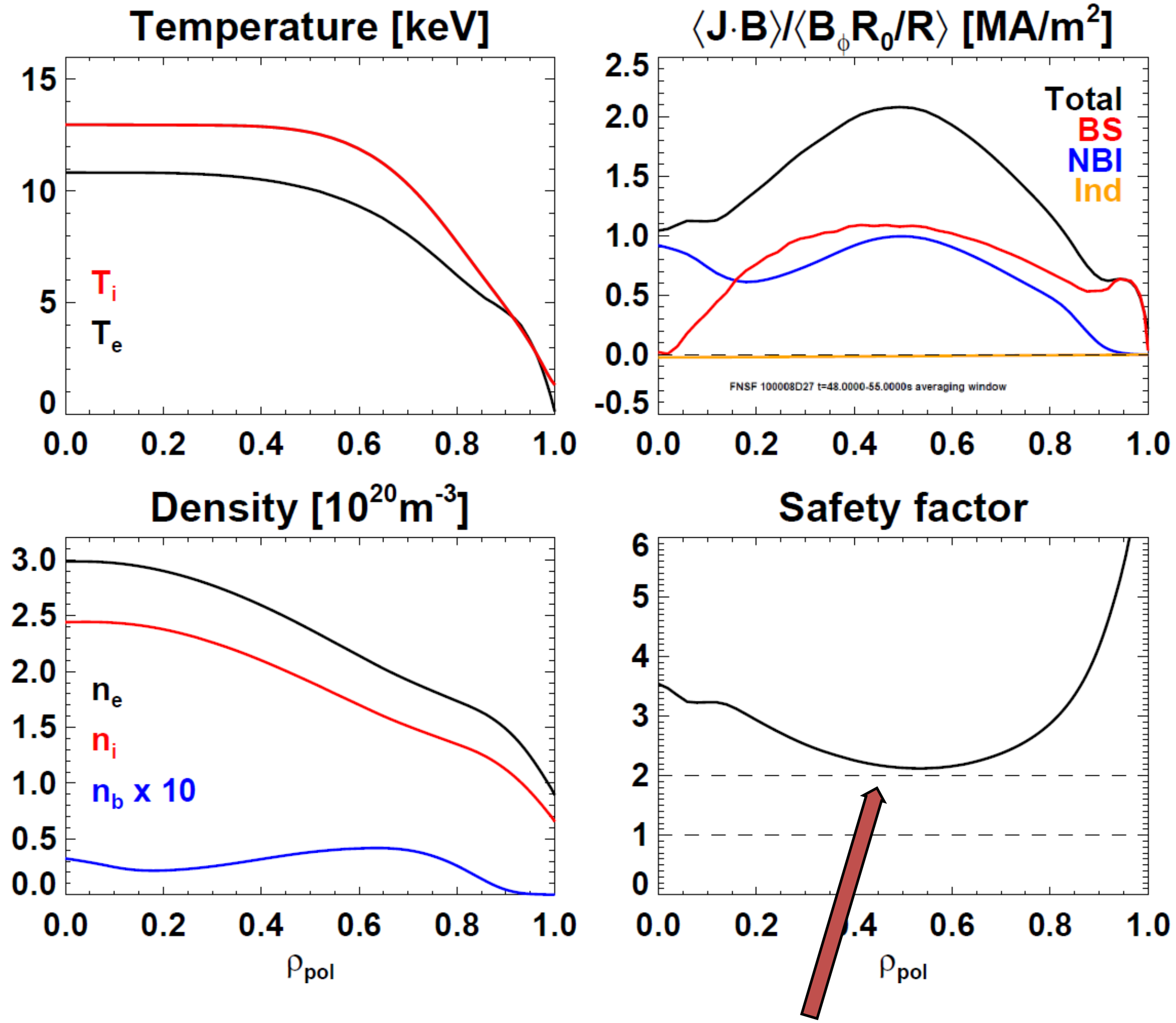


Peak $q_\perp < 10MW/m^2$



SOLPS
J. Canik

Used free-boundary TRANSP/NUBEAM to specify NBI and simulate 100% non-inductive plasmas with Q_{DT} ~2



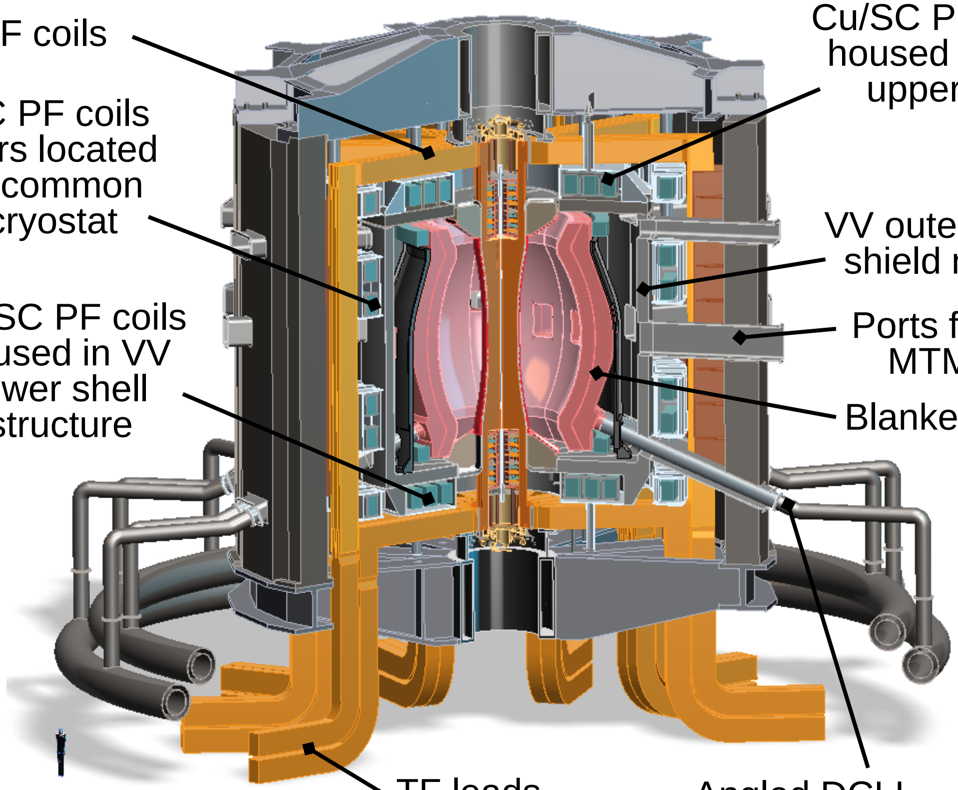
- Neoclassical χ_{ion}
- n_e / $n_{Greenwald}$ = 0.7
- $H_{98,y2}$ = 1.4
- I_P = 8.9MA, B_T = 2.9T
- f_{NICD} = 100%, f_{BS} = 65%
- P_{NNBI} = 80MW (0.5MeV)
- P_{fus} = 200MW (50-50 DT)
  - 2.6% alpha bad orbit loss
- Q_{DT} = 2.5
- β_N = 5.5, W_{tot} = 58MJ
  - W_{fast} / W_{tot} = 14%

- Maintain q_{min} > 2, q(0) / q_{min} controllable via R_{tan} and density

Developed detailed CAD models for 2 different sizes

(R = 1m and 1.7m – most analysis done for R=1.7m configuration)

Design features



TF coils

SC PF coils pairs located in common cryostat

Cu/SC PF coils housed in VV lower shell structure

TF leads

Cu/SC PF coils housed in VV upper lid

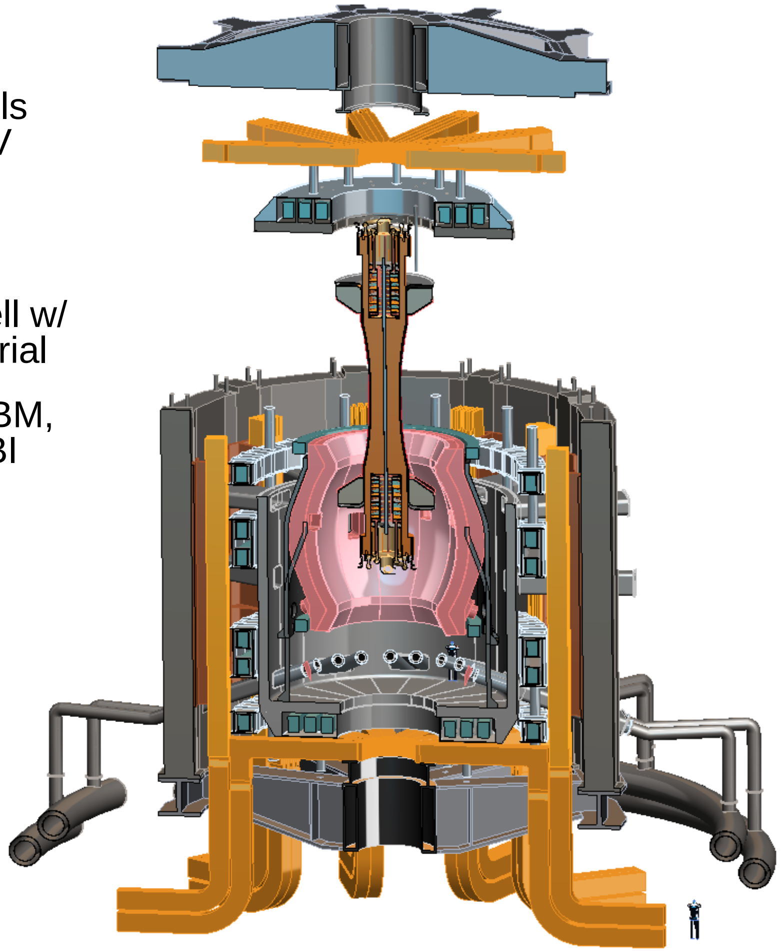
VV outer shell w/ shield material

Ports for TBM, MTM, NBI

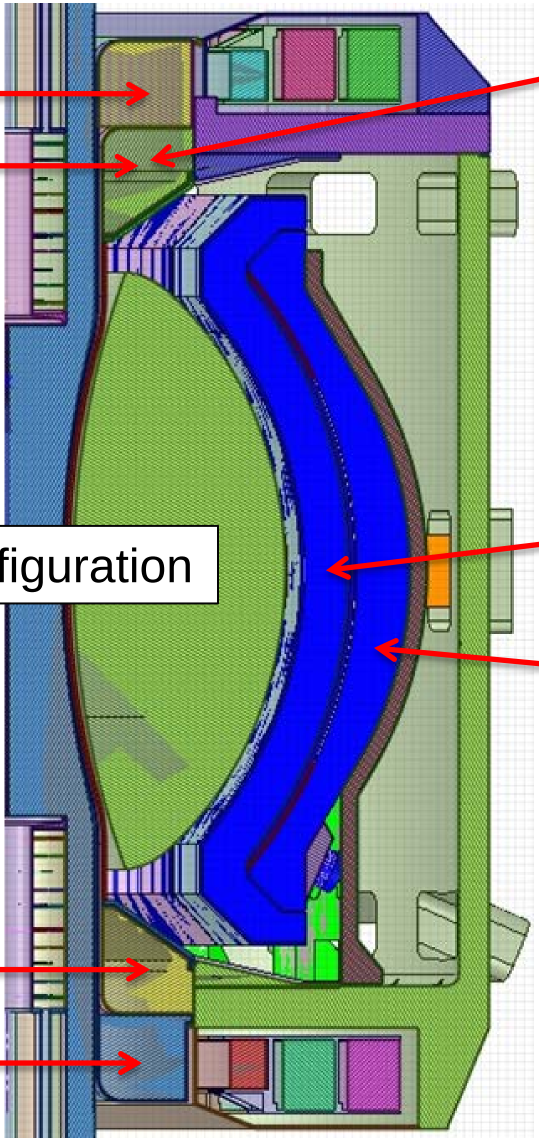
Blankets

Angled DCLL concentric lines to external header

Vertical maintenance



Conformal blankets + breeding at top/bottom important for tritium breeding ratio TBR ~ 1



R=1.7m configuration

ΔTBR = +0.07

Inner Blanket Segment = **0.81**

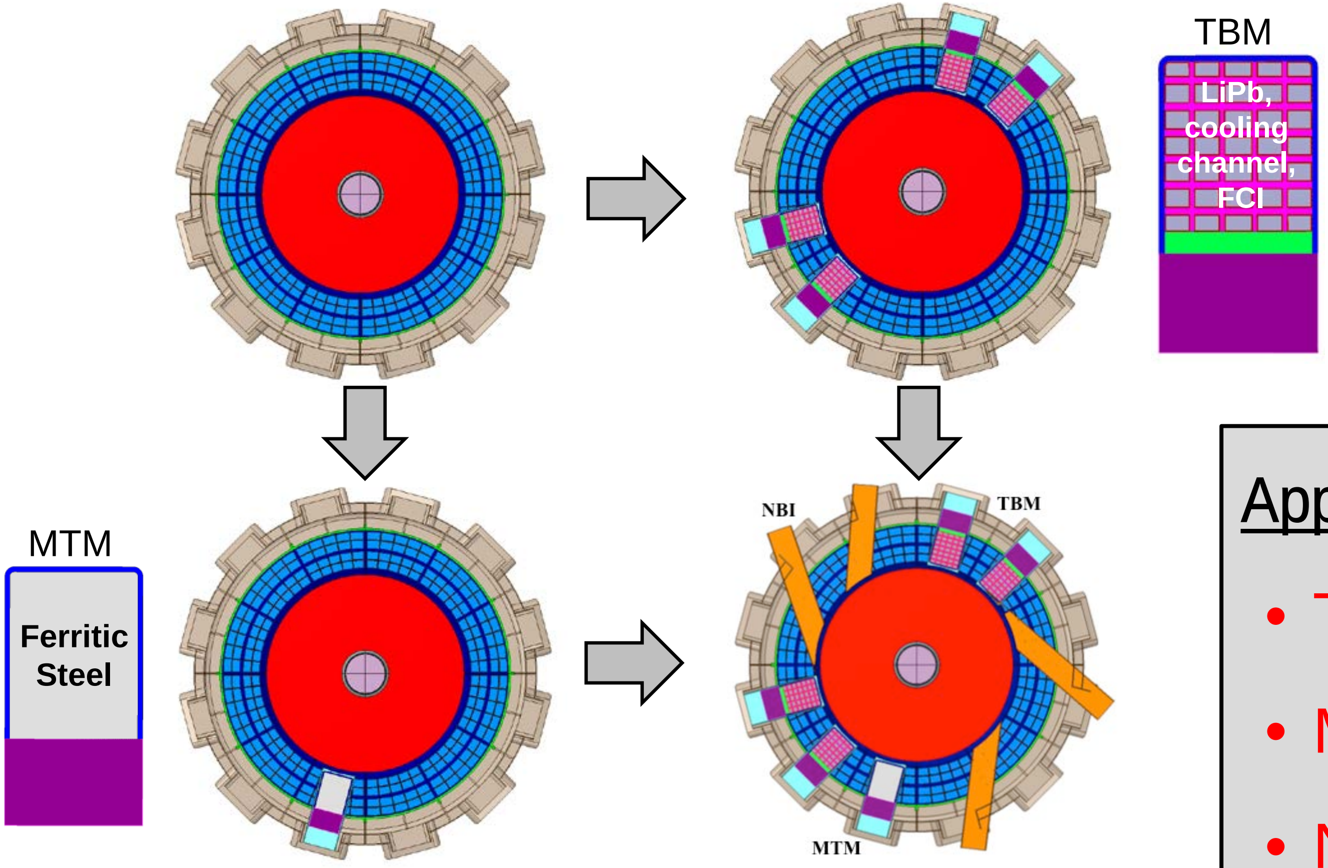
Outer Blanket Segment = **0.15**

Total TBR ~ 1.03 with no penetrations or ports
(heterogenous outboard blanket)

Quantified impact of TBM, MTM, NBI ports on TBR

No ports or penetrations,
homogeneous breeding zones:
TBR = 1.03

Add 4 Test Blanket
Modules (TBMs)
TBR = 1.02 (ΔTBR = -0.01)



1 Materials Test Module (MTM)
TBR = 1.01 (ΔTBR = -0.02)

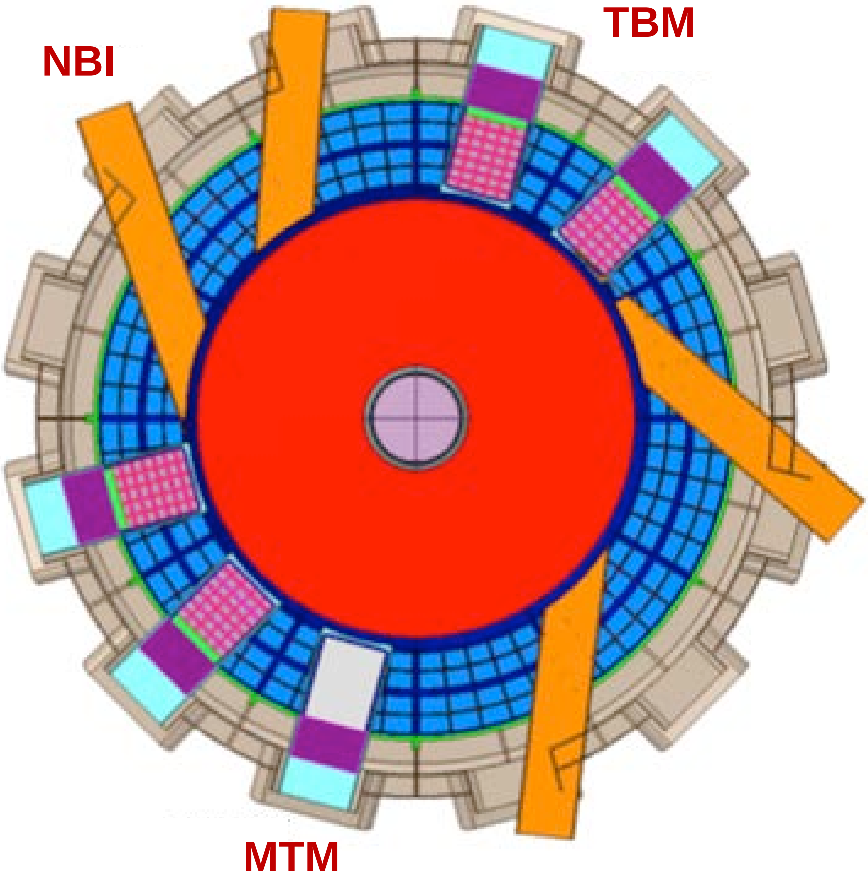
4 TBM + 1 MTM + 4 NBI
TBR = 0.97

Approx. ΔTBR per port:

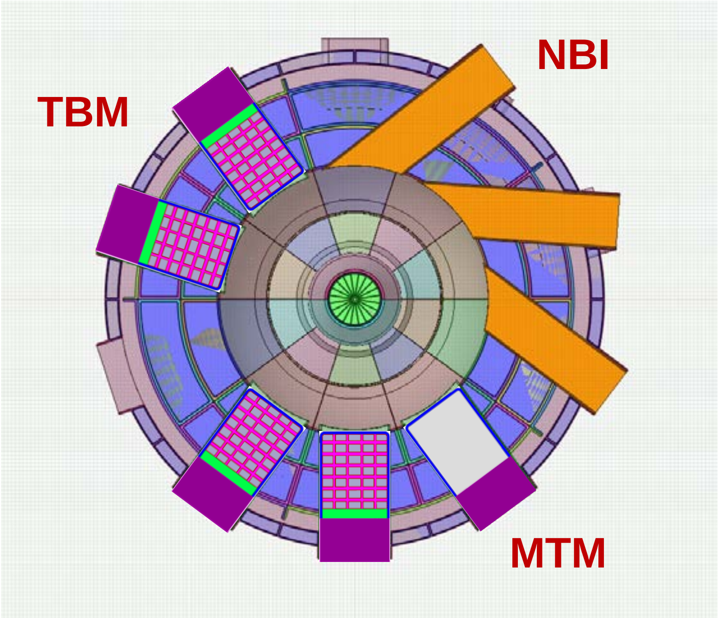
- TBM: -0.25%
- MTM: -2.0%
- NBI: -0.75%

Find R ≥ 1.7m necessary for TBR ≥ 1 at A=1.7

R=1.7m: **TBR ≥ 1**

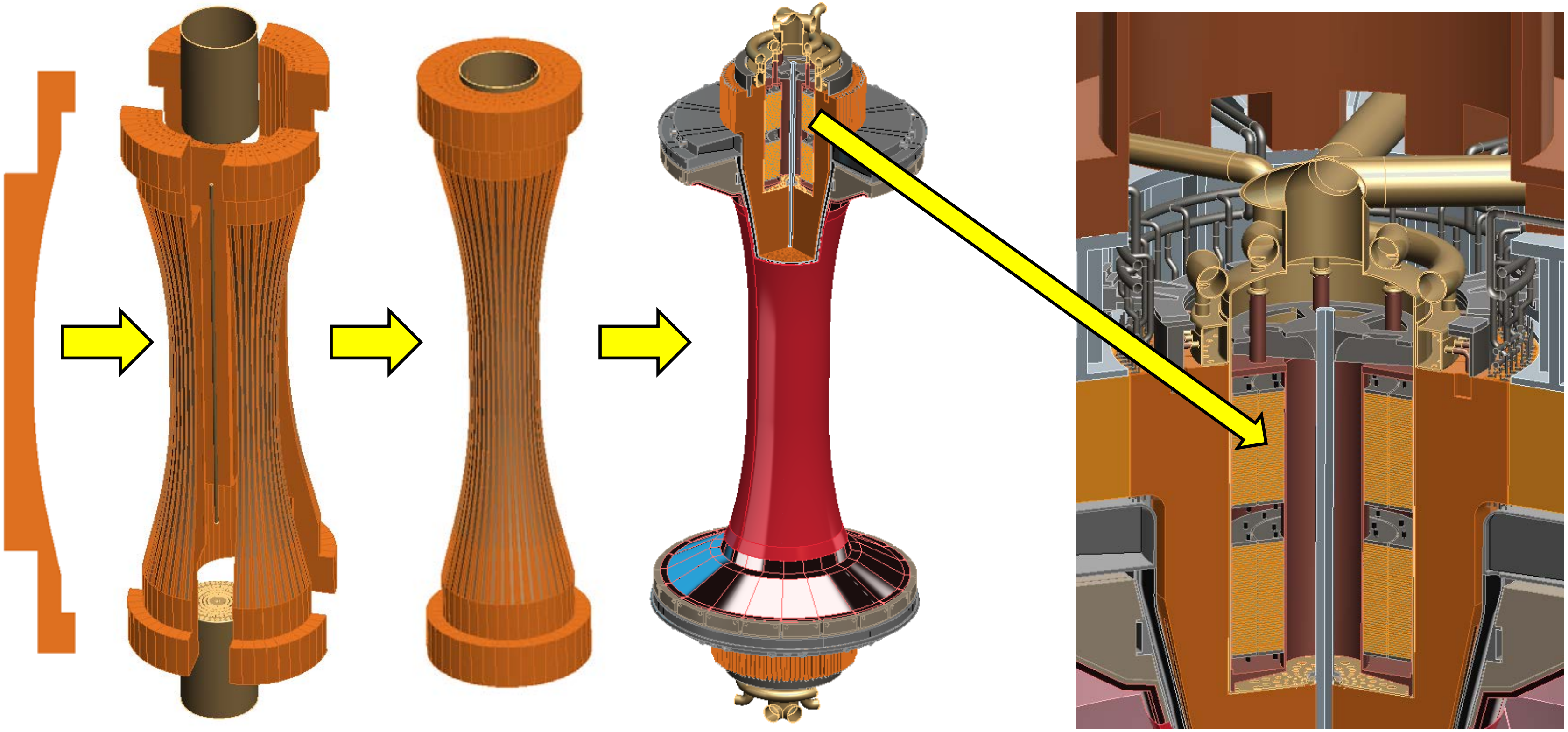


R=1.0m: **TBR < 1 (≈ 0.9)**



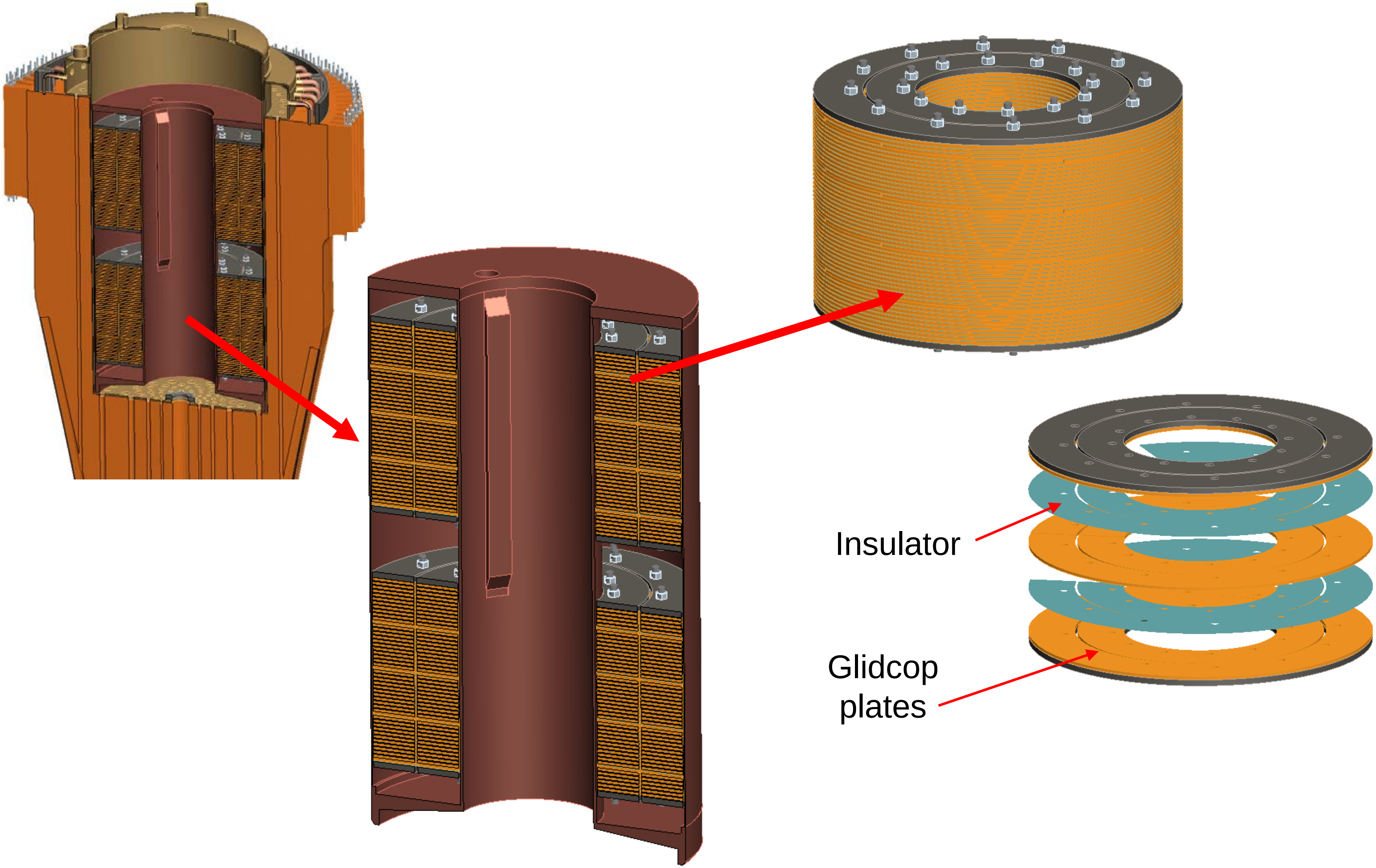
- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

FNSF center-stack can build upon NSTX-U design and incorporate NSTX stability results



- **Like NSTX-U, use TF wedge segments (but brazed/pressed-fit together)**
  - Coolant paths: gun-drilled holes or grooves in side of wedges + welded tube
- **Bitter-plate divertor PF magnets in ends of TF achieve high triangularity**
  - **NSTX data:** High $\delta > 0.55$ and shaping $S \equiv q_{95}I_P/aB_T > 25$ minimizes disruptivity
  - Neutronics: MgO insulation can withstand lifetime (6 FPY) radiation dose

Bitter coil insert for divertor coils in ends of TF



Insulator

Glidcop plates

MgO insulation appears to have good radiation resistance for divertor PF coils

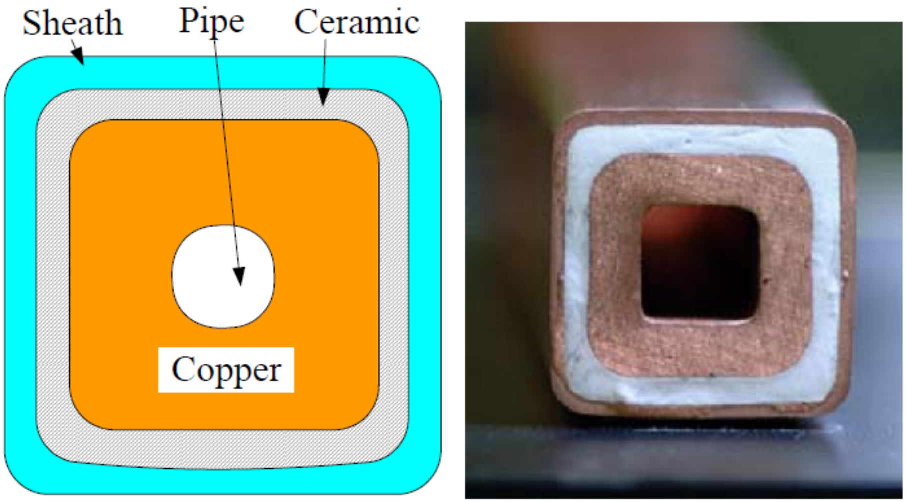
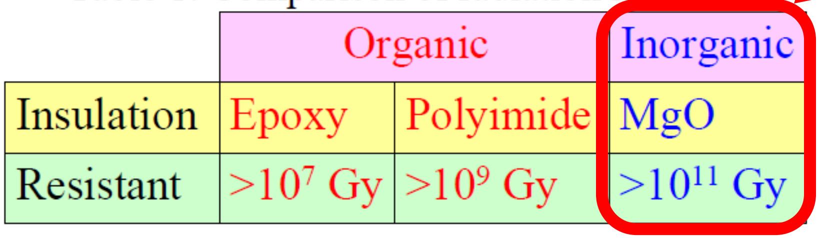


Fig. 3 Cross section of MIC

Table 1: Comparison of radiation resistant



	Organic		Inorganic
Insulation	Epoxy	Polyimide	MgO
Resistant	$>10^7$ Gy	$>10^9$ Gy	$>10^{11}$ Gy

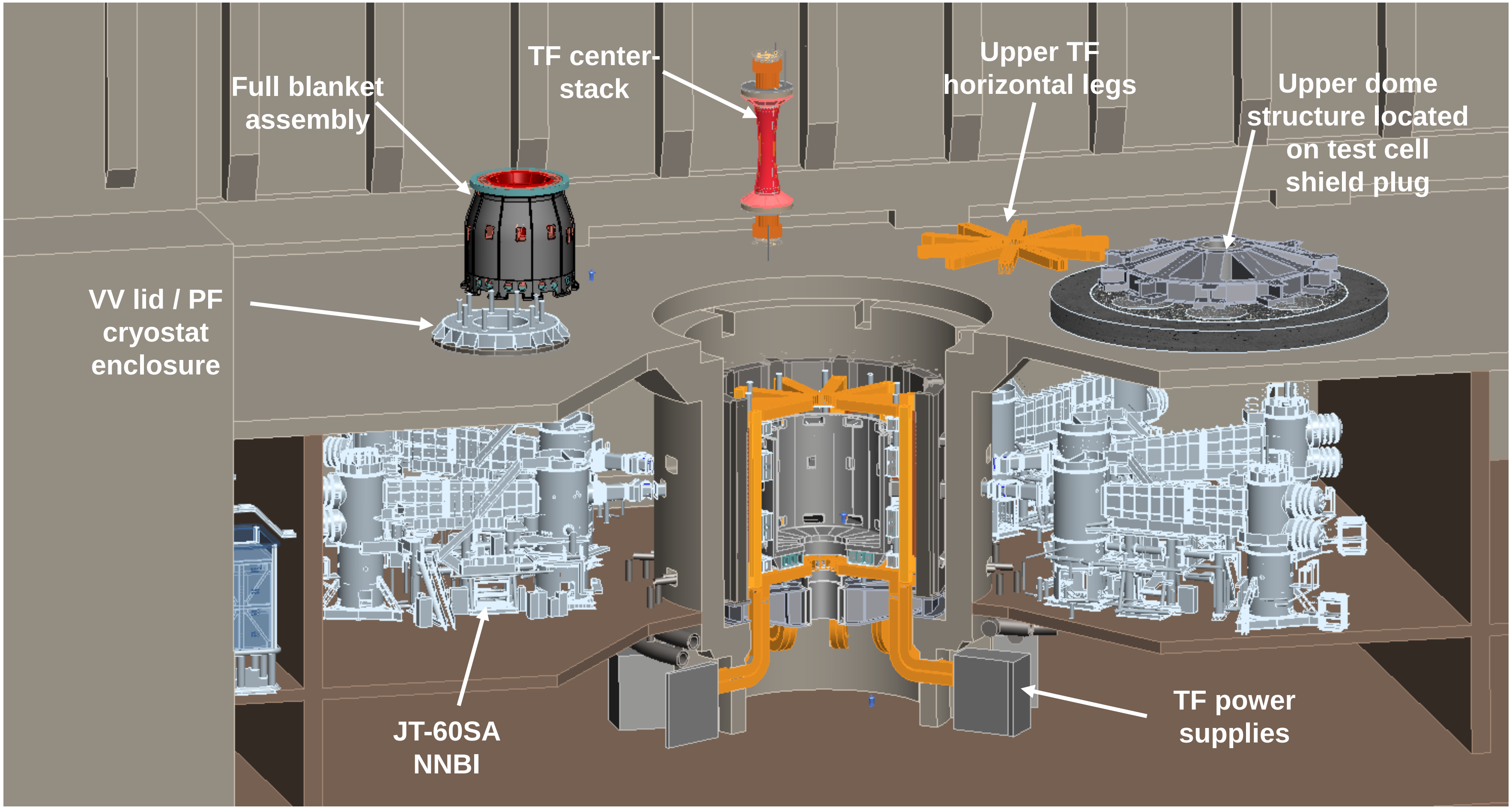
Proceedings of the 5th Annual Meeting of Particle Accelerator Society of Japan
and the 33rd Linear Accelerator Meeting in Japan (August 6-8, 2008, Higashihiroshima, Japan)

R&D of a Septum Magnet Using MIC coil

Kuanjun Fan [1,A)], Hiroshi Matsumoto [A)], Koji Ishii [A)], Noriyuki Matsumoto [B)]
[A)] High Energy Accelerator Research Organization (KEK)
1-1 OHO, Tsukuba, Ibaraki, 305-0801, Japan
[B)] 2NEC/Token

R=1.7m ST-FNS facility layout

using an extended ITER building



Backup Slides – HTS TF FNSF

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
- B_T decreases at lower A – depends strongly on:
  - Inboard shielding, HTS allowable field and current density

Approach:

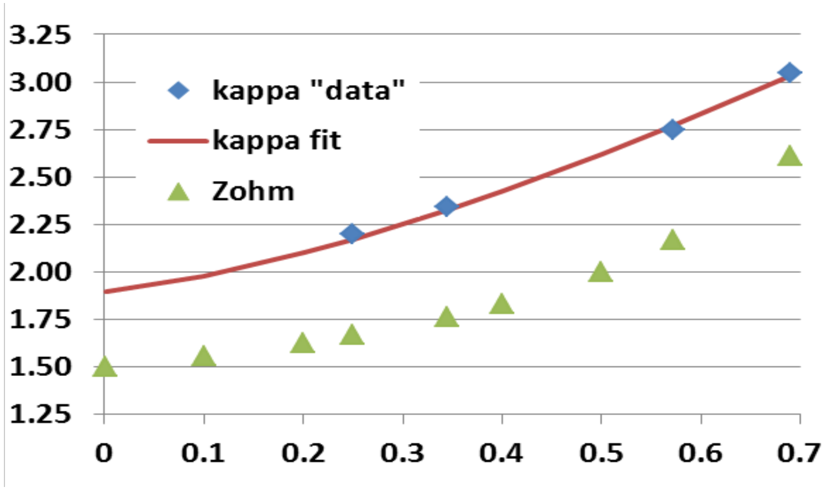
- Fix plasma major radius and heating power (50MW)
  - R_0 = 3m – smallest size for $Q_{eng} > 1$ and high fluence
- Apply magnet & plasma constraints (see backup)
  - HTS strain: 0.3%, β_N: n=1 no-wall, κ: 0.95×limit, $f_{GW} = 0.8$
- Vary aspect ratio from A = 1.6 to 4
- Vary HTS current density, peak field
  - Also scan inboard shielding thickness (not shown)
- Compute Q_{DT}, Q_{eng}, and required H_{98} (*unconstrained*)

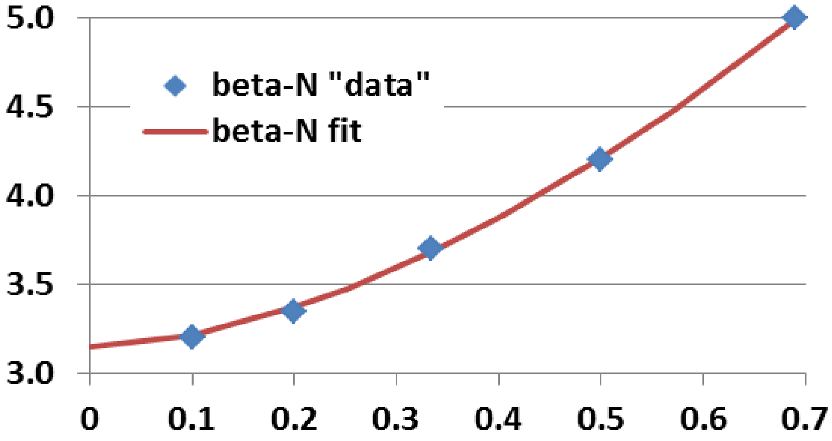
Engineering constraints

- 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: 3.5×10^{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

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

Pilot study uses $0.95\times\boldsymbol{\kappa}$ 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 $A \rightarrow \infty$

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

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

Simplified TF magnet design equations

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

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

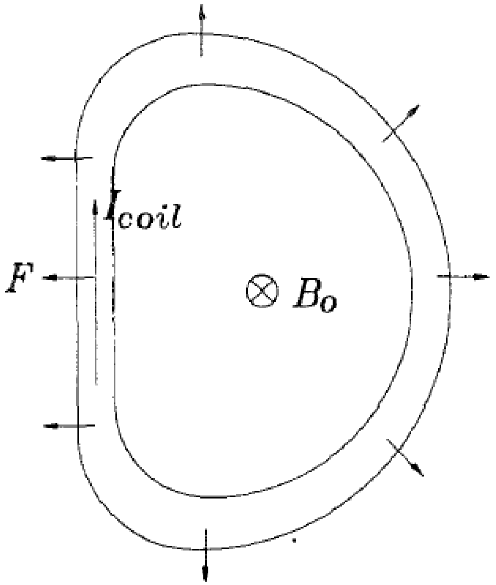


Fig. 5. Lorentz forces are normal to the conductor in the poloidal plane.

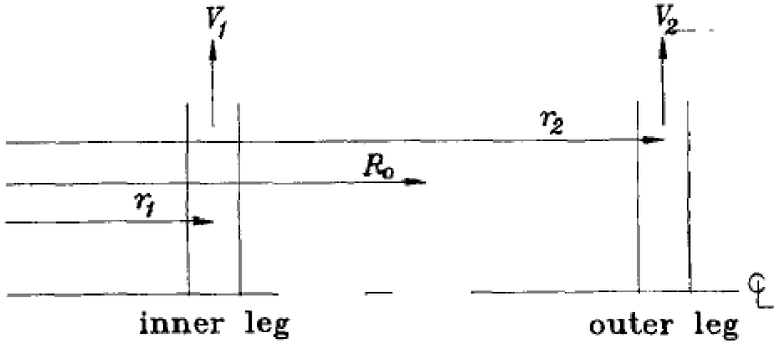


Fig. 7. Geometry for force and moment balances.

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

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

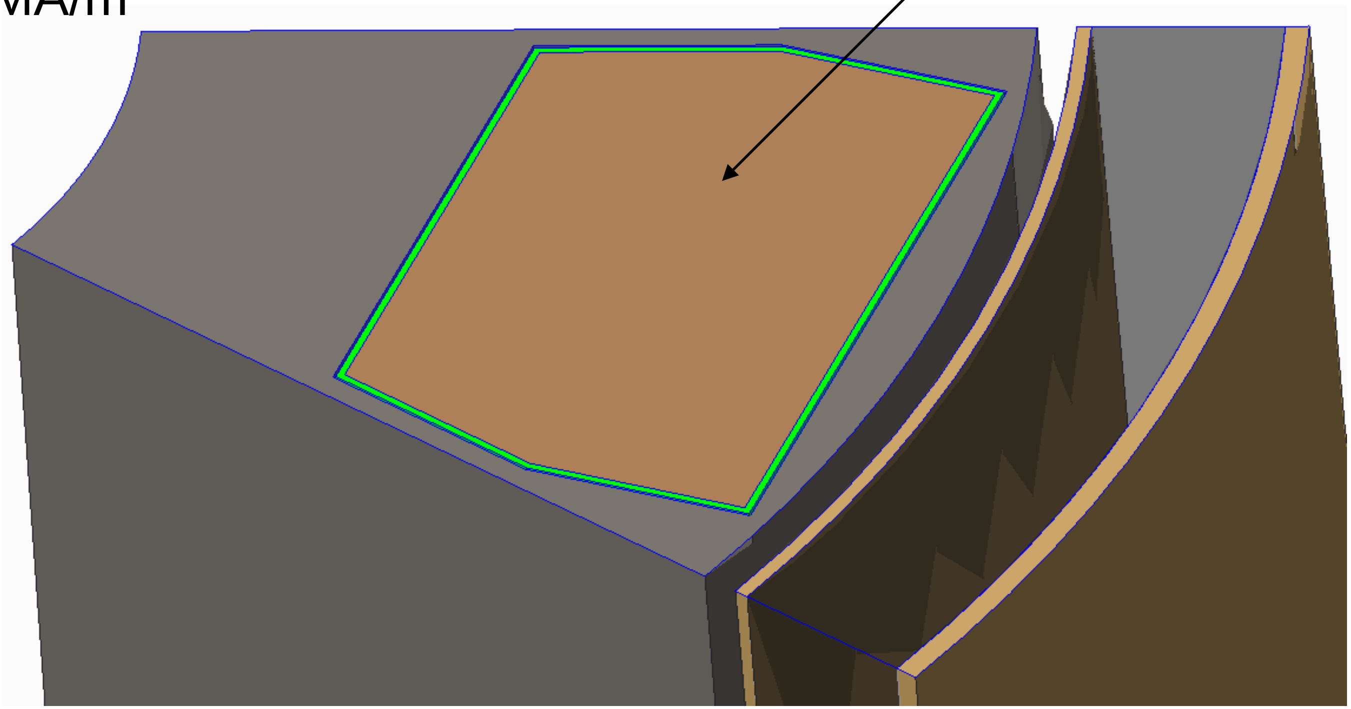
$$F \equiv \frac{1}{2} B_0 R_0 I_{\text{coil}}$$

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

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

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

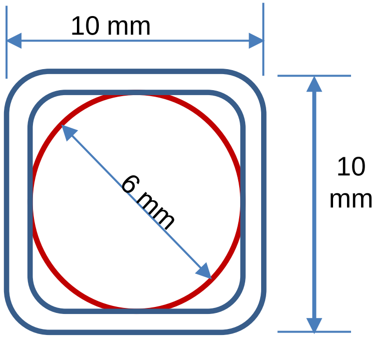
A=2, R_0 = 3m device TF inboard leg showing allocated space for case and winding



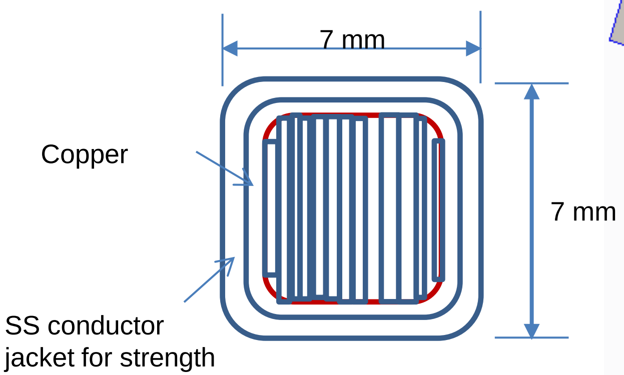
Current per coil: 6 MA
Winding Cd: 35.9 MA/m^2

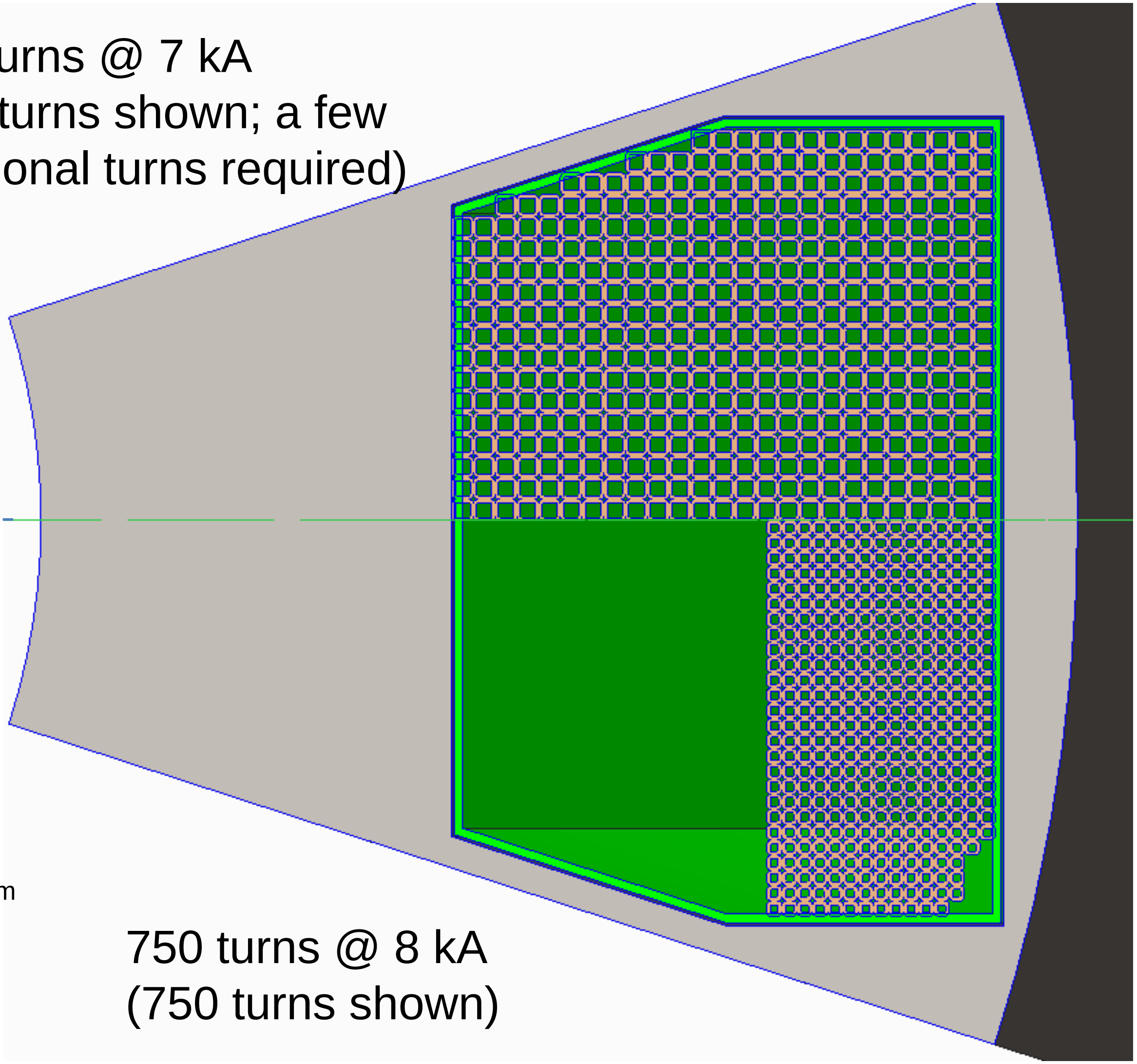
$0.167\ m^2$ winding area

CORC Conductor – Achieved now



Base Conductor –
Helium Gas Cooled





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

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*

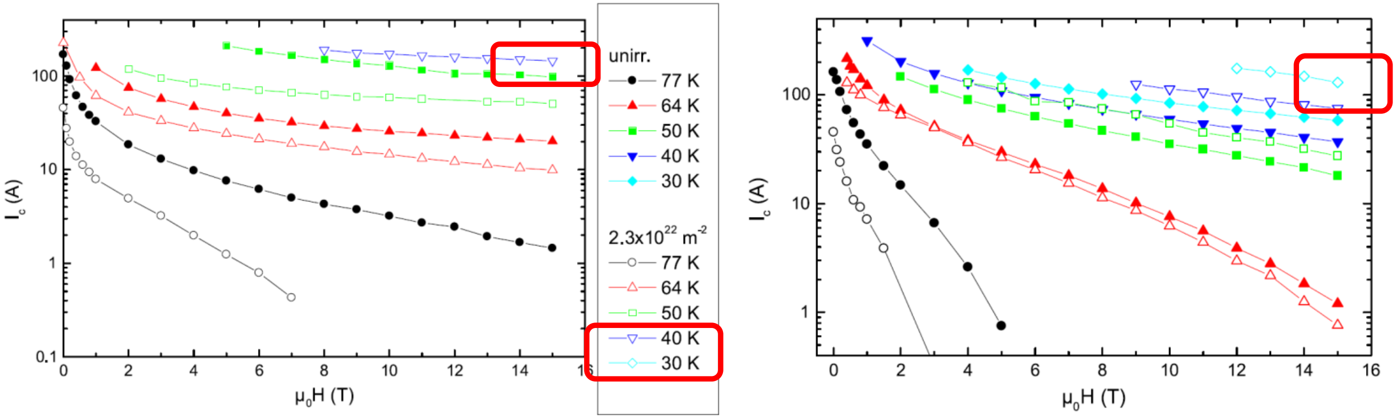


Figure 6. Critical currents (ASC-40) in magnetic fields applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) before and after irradiation to a fast neutron fluence of $2.3 \cdot 10^{22}$ m^{-2}.

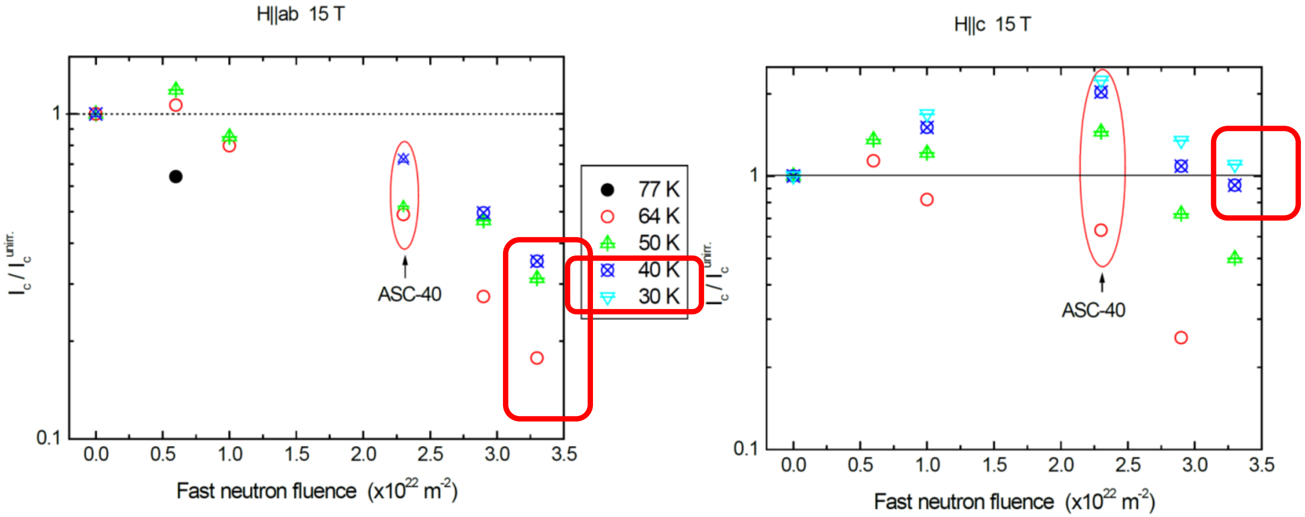
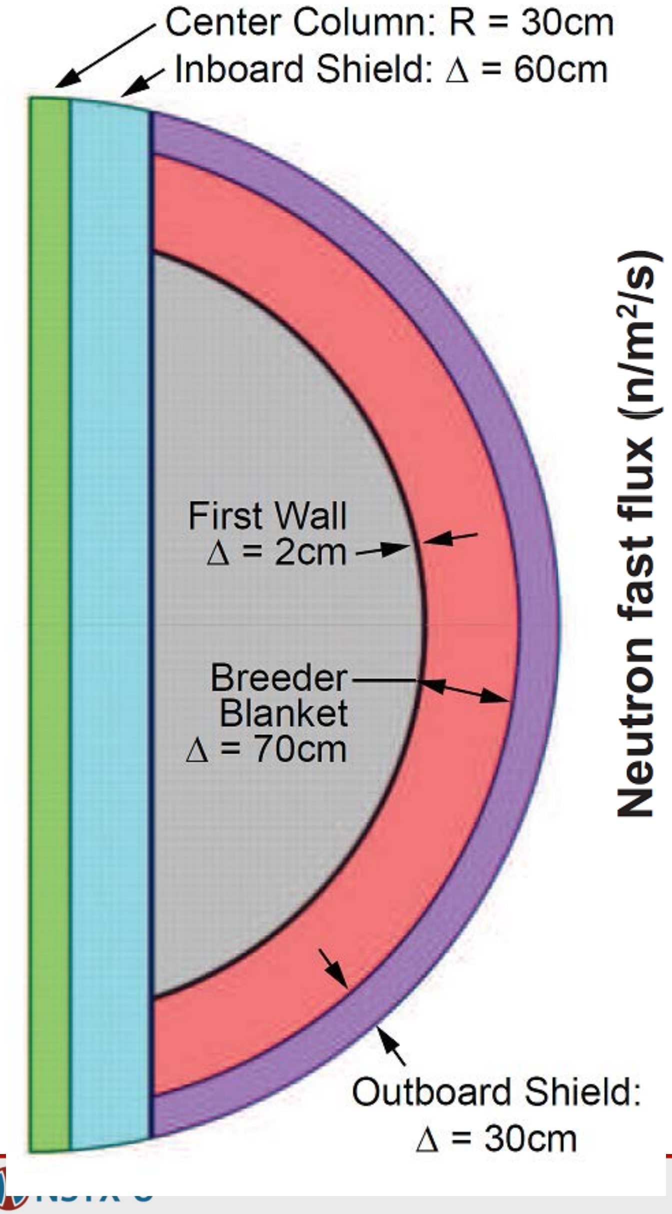
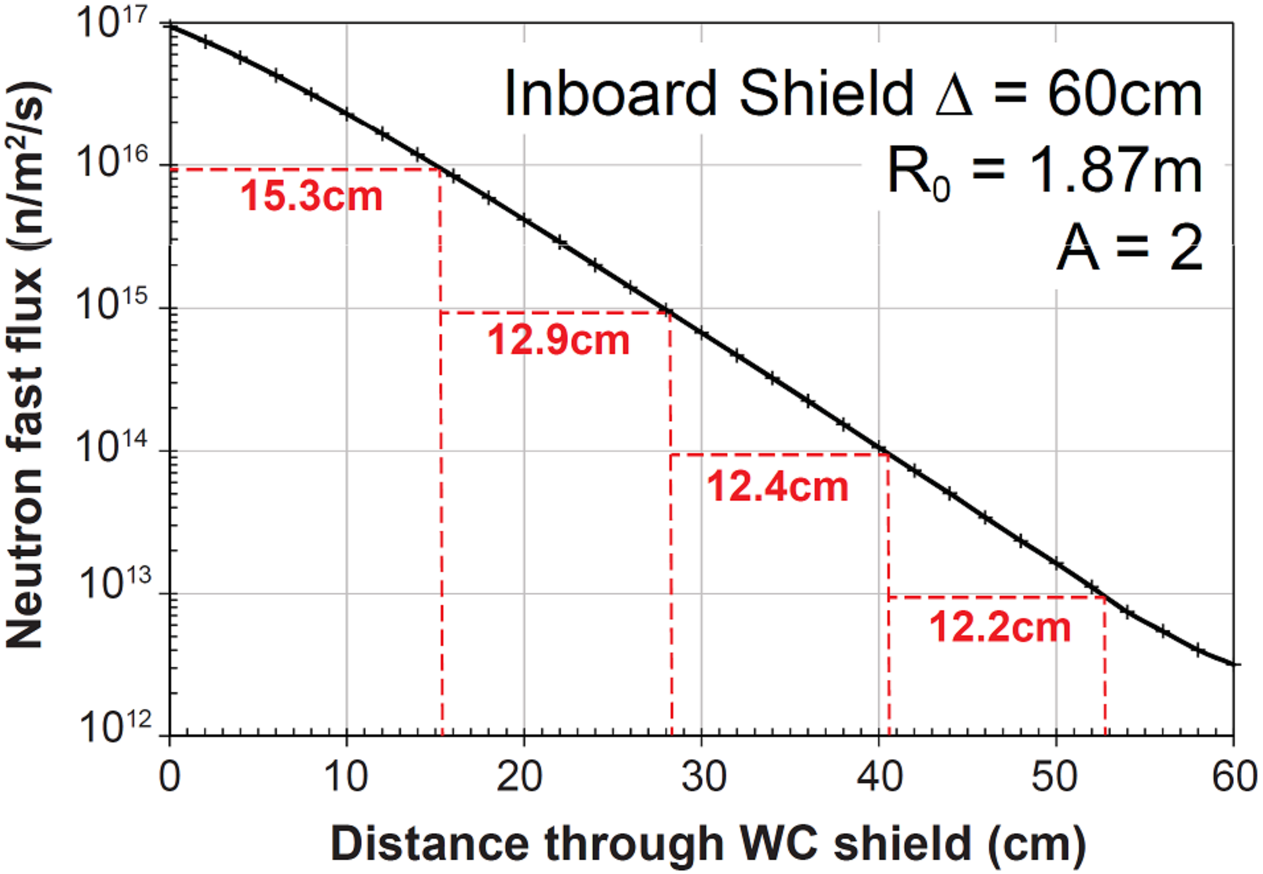


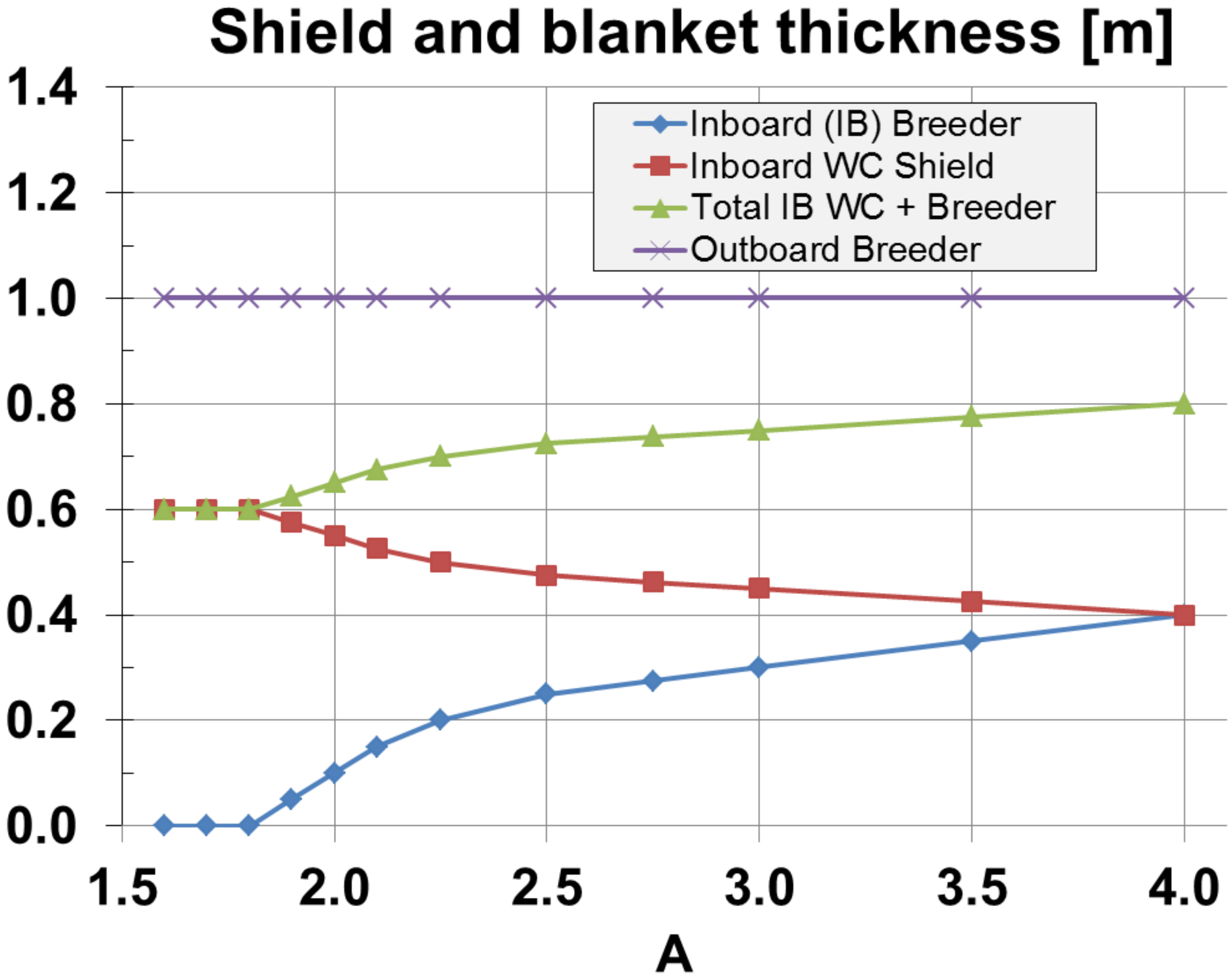
Figure 8. Normalized critical currents in a magnetic field of 15 T applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) as a function of neutron fluence.

Neutronics analysis for HTS TF shielding





Breeding blanket thickness model



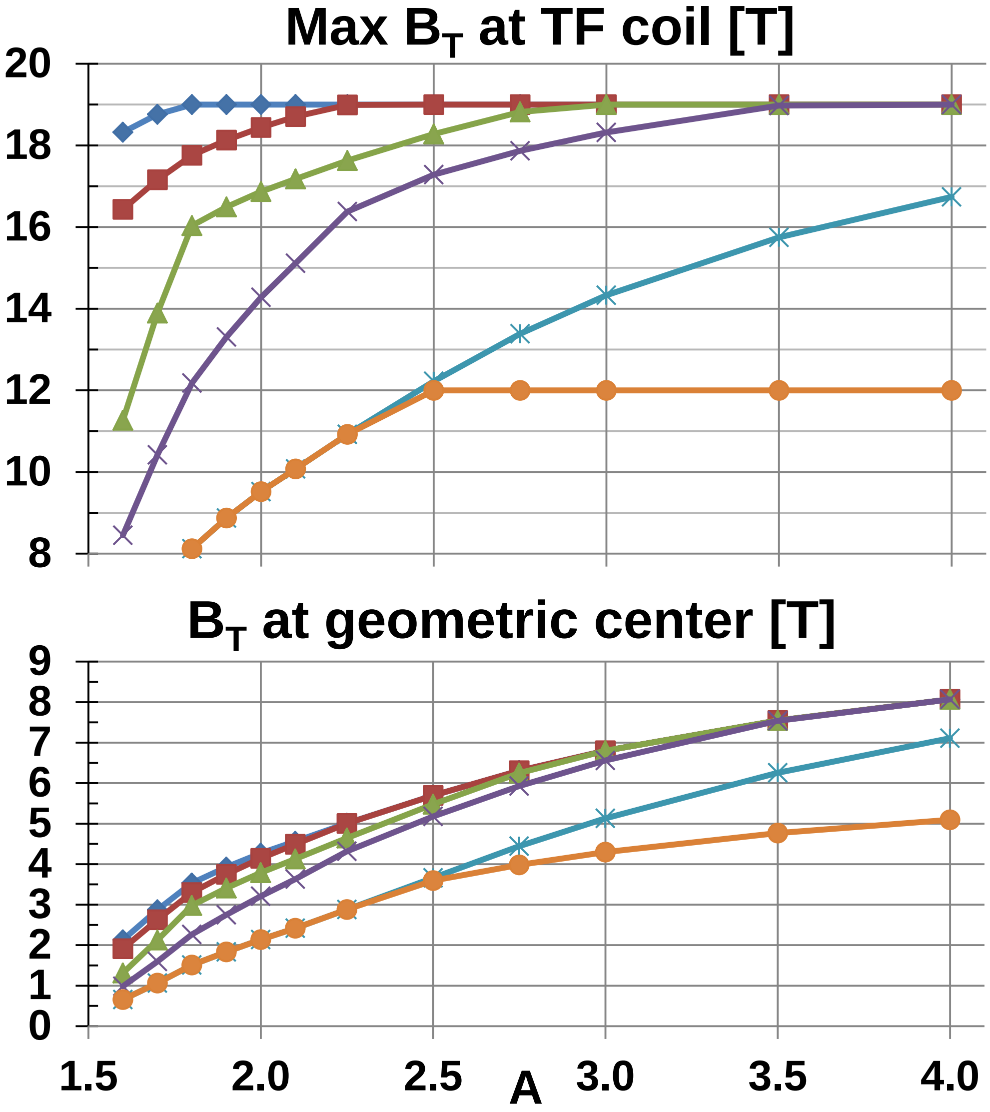
High TF winding-pack current density required to access highest B_T at lower A

J_{WP}
$[MA/m^2]$

◆ 160
■ 70
▲ 40
× 30
✳ 20
● 20

19T: Present CORC HTS limit →

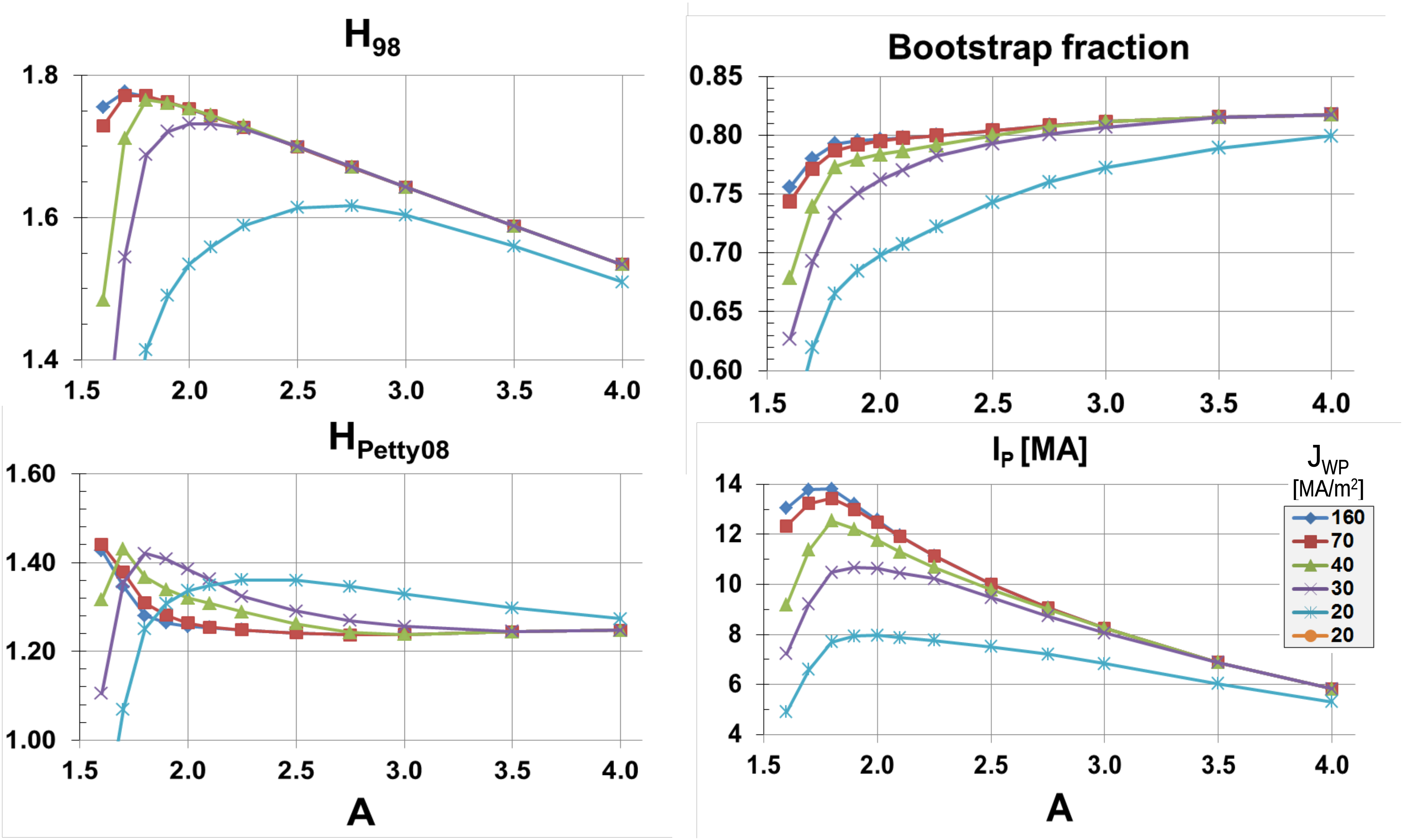
12T: ITER-like TF coil limit →
(Nb_3Sn, 11.8T)


Max B_T at TF coil [T]

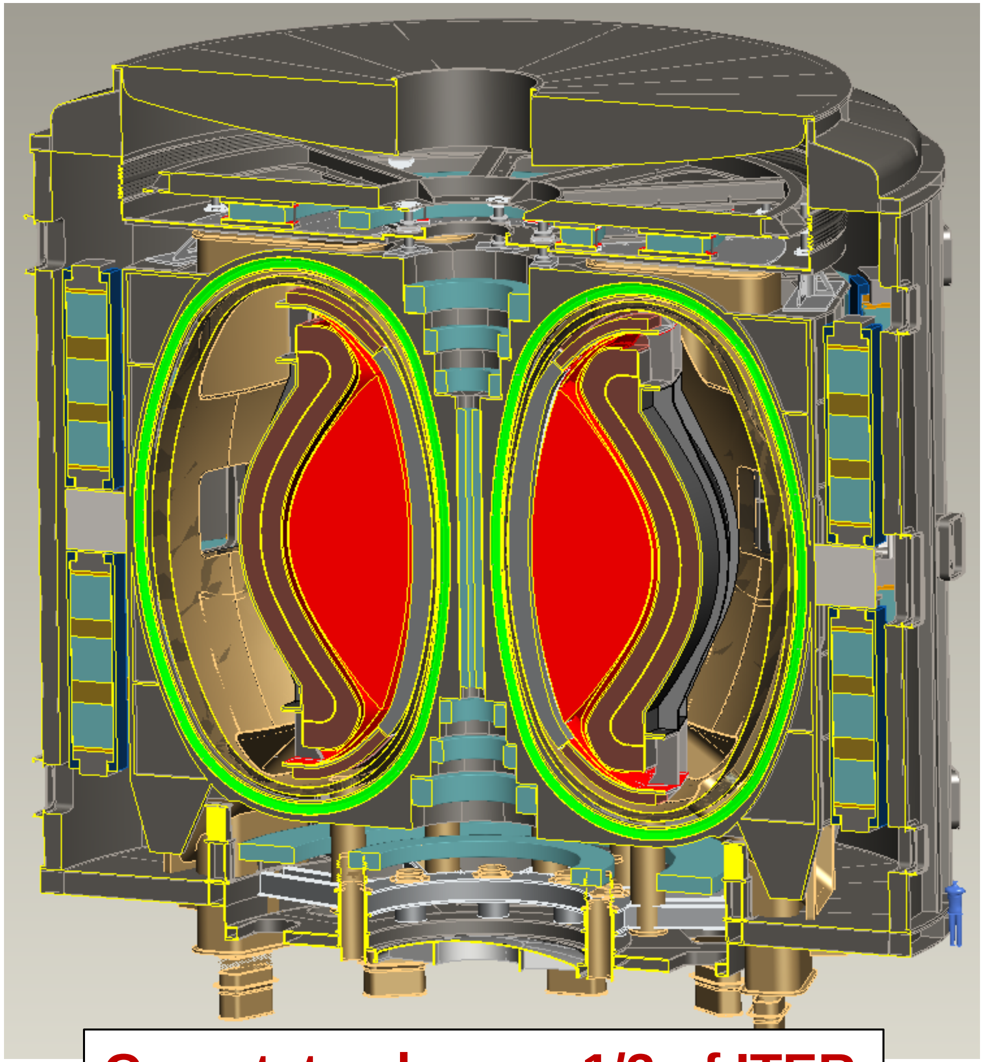
B_T at geometric center [T]

Coil structure sized to maintain ≤ 0.3% strain on winding pack for all cases shown here

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



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



Cryostat volume ~ 1/3 of ITER

B_T = 4T, I_P = 12.5MA

$\kappa = 2.5$, $\delta = 0.55$

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

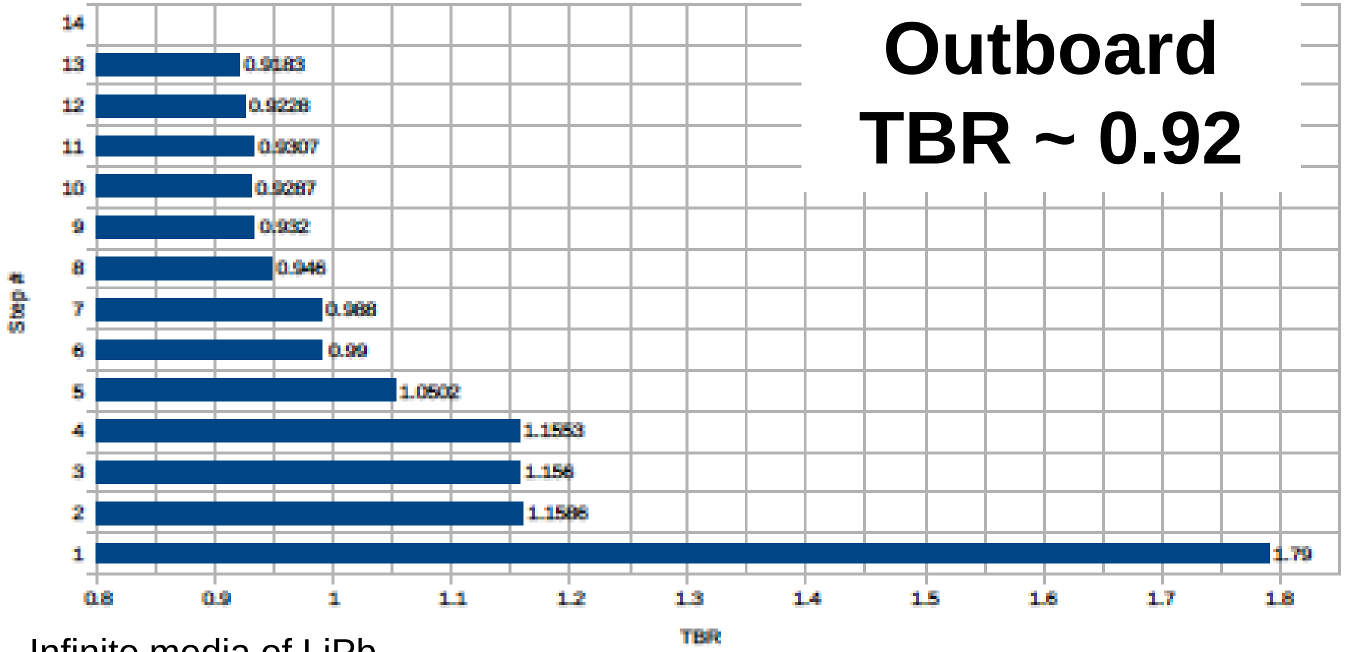
P_{net} = 73 MW

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

Peak n-flux = 2.4 MW/m^2

Peak n-fluence = 7 MWy/m^2

Breeding calculations nearly complete for A=2



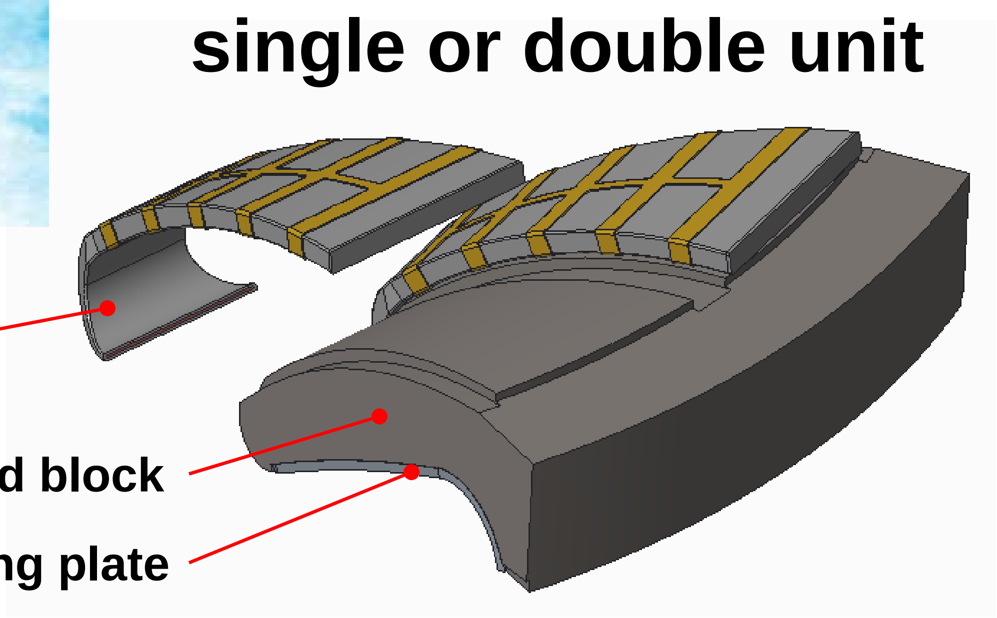
Outboard TBR ~ 0.92

- 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

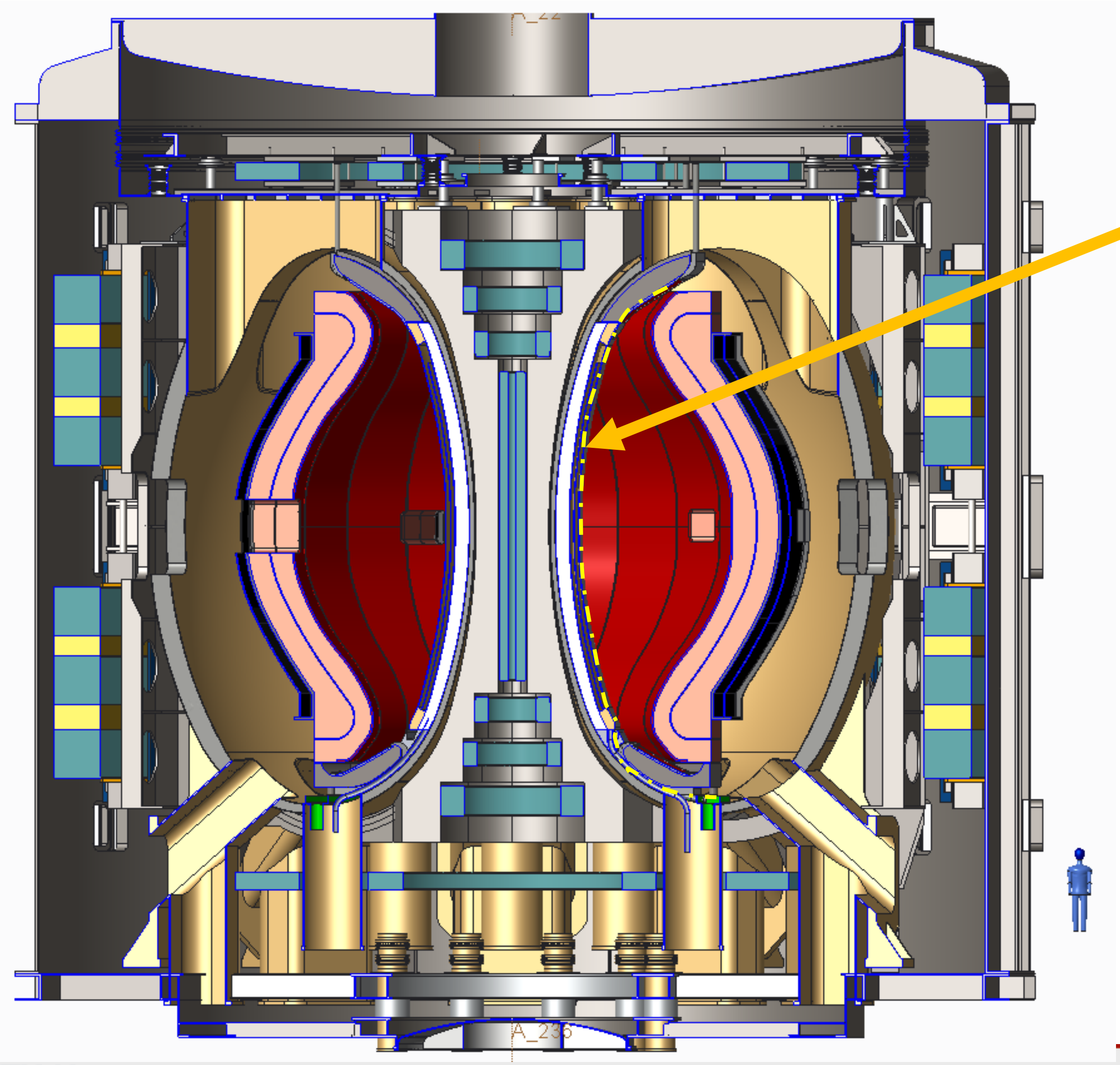
Ongoing: Thin inboard blanket (10 cm) should provide TBR > 1

Exploring liquid metal divertor design similar to flowing water curtain systems

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



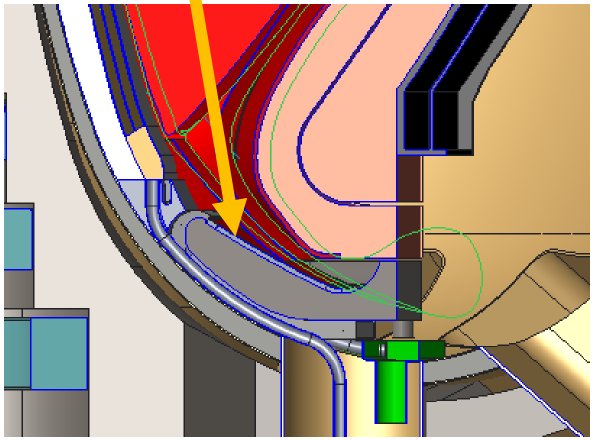
HTS ST-FNSF design with Li flow on divertor and inboard surfaces



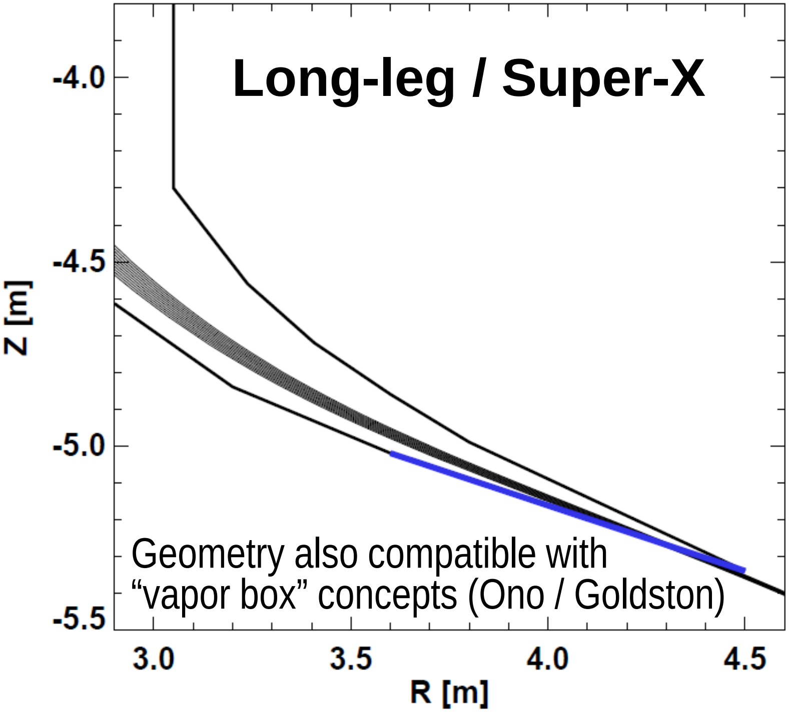
Double null liquid metal divertor system

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

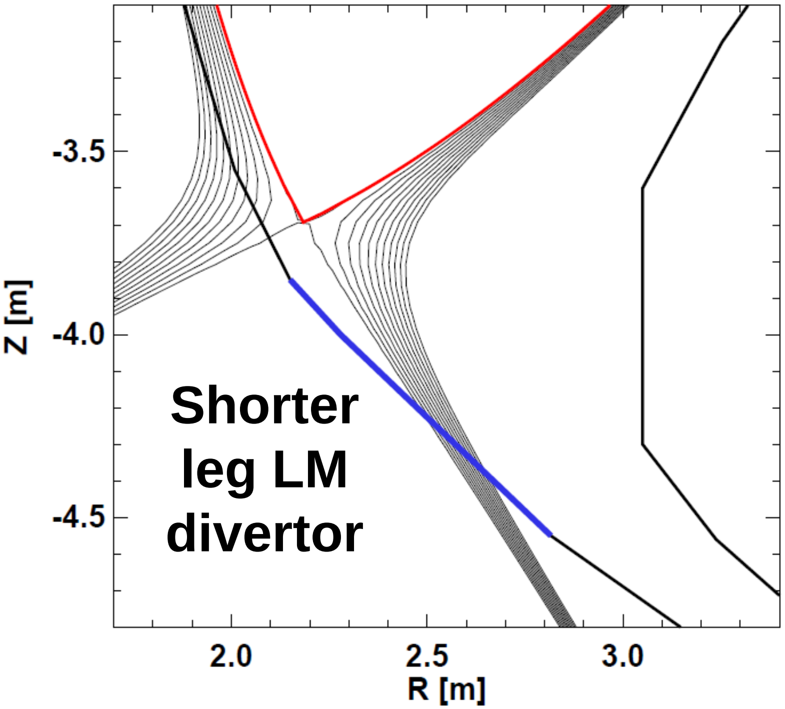
Separate Li cooling of lower divertor

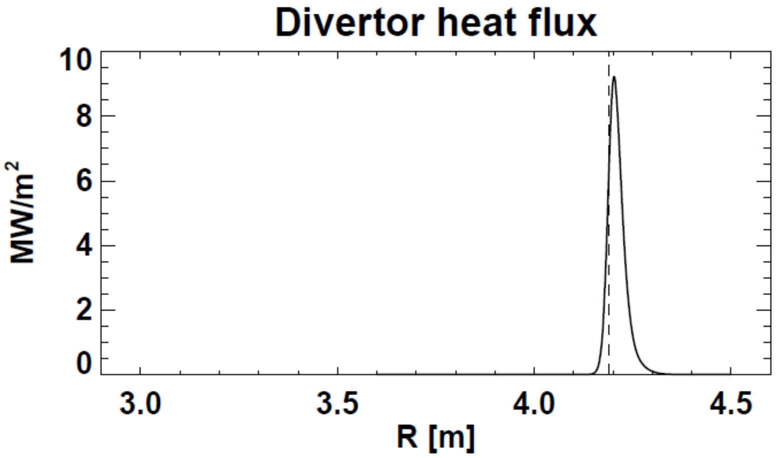


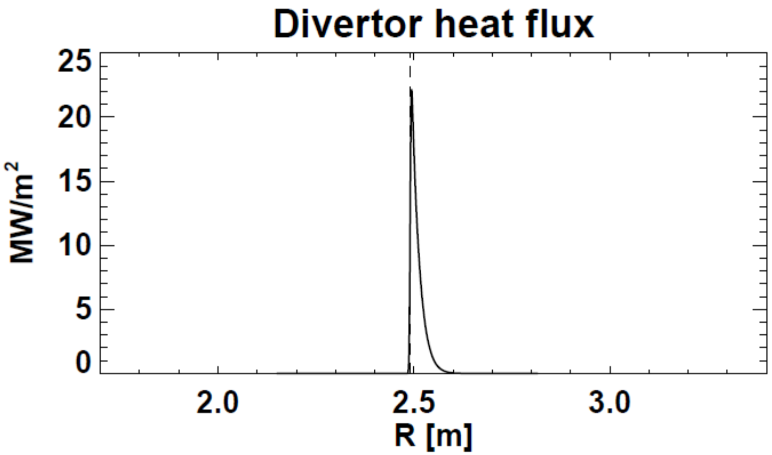
Another option: Li divertor with shorter outer leg
$P_{div} = 9 \rightarrow 21 MW/m^2$ for $R_{strike} = 4.2m \rightarrow 2.5m$



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

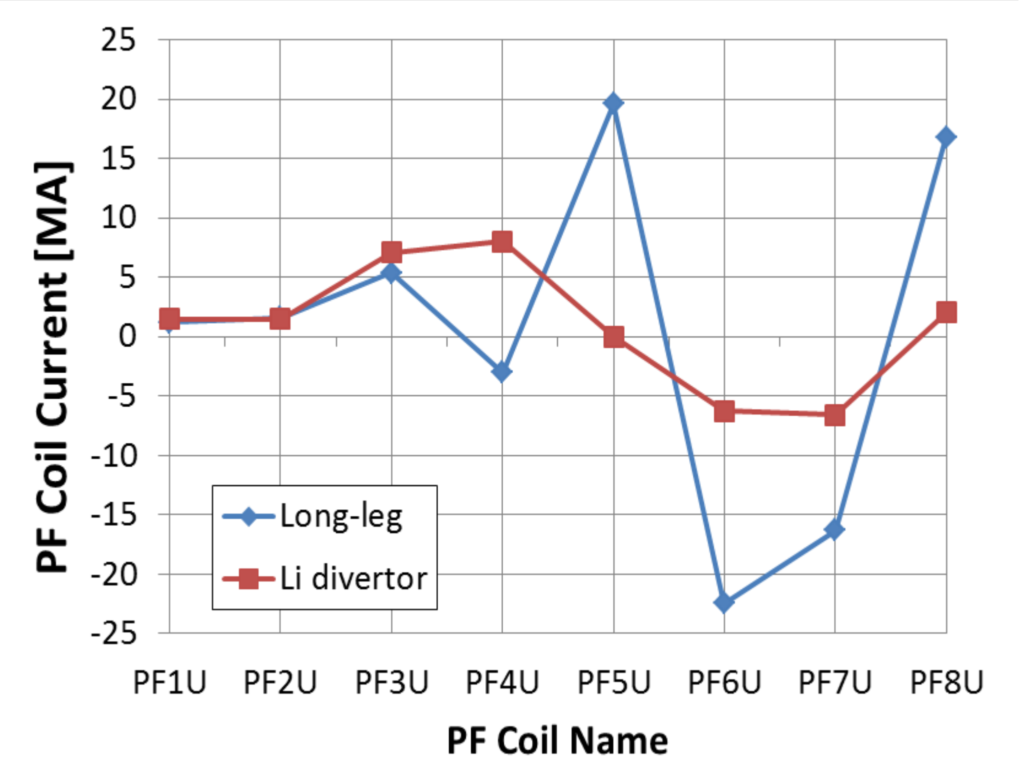


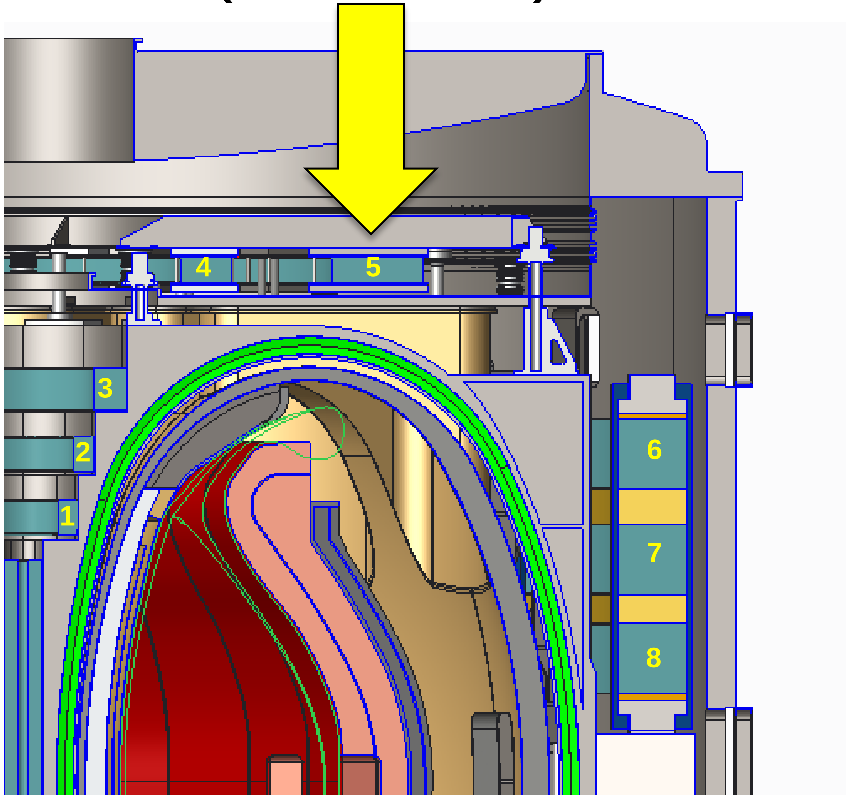




Benefits of shorter-leg LM high-heat-flux divertor:

- 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 (see Maingi talk)