U.S. DEPARTMENT OF ENERGY | Office of Science

NSTX-U

Motivations for advanced divertor system and materials research in next-step fusion facilities and NSTX Upgrade

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with contributions from

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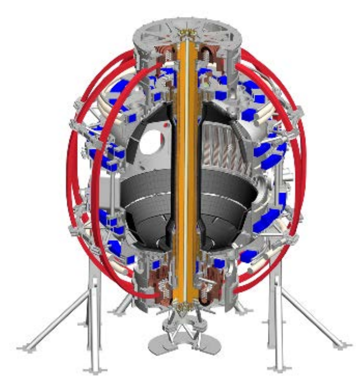
2017 Physics / Theoretical Seminar Series
Thursday, March 16, 2017 - 3:45 – 4:45 p.m.
Rosen Auditorium (TA-53, Bldg. 1)
Los Alamos National Laboratory

PPPL



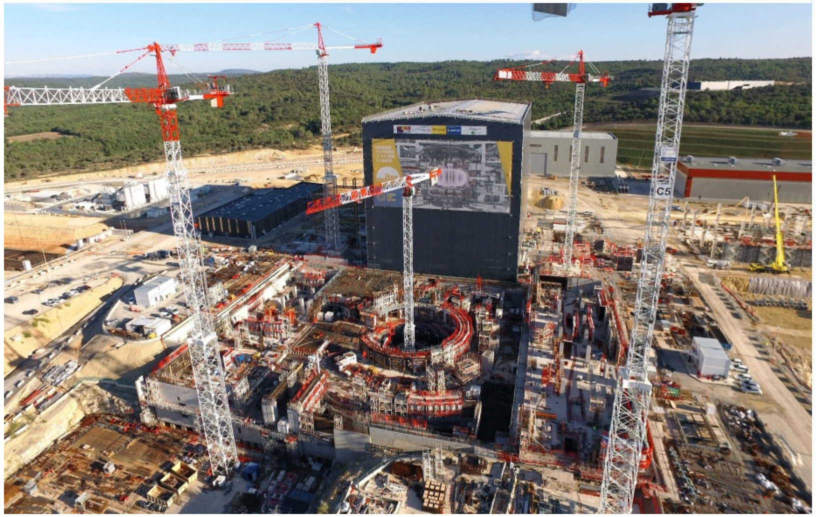
Outline

- Introduction
- Why spherical tori / tokamaks (STs)
- NSTX-U initial results and near-term plans
- Future: liquid metals in NSTX-U and beyond
- Summary

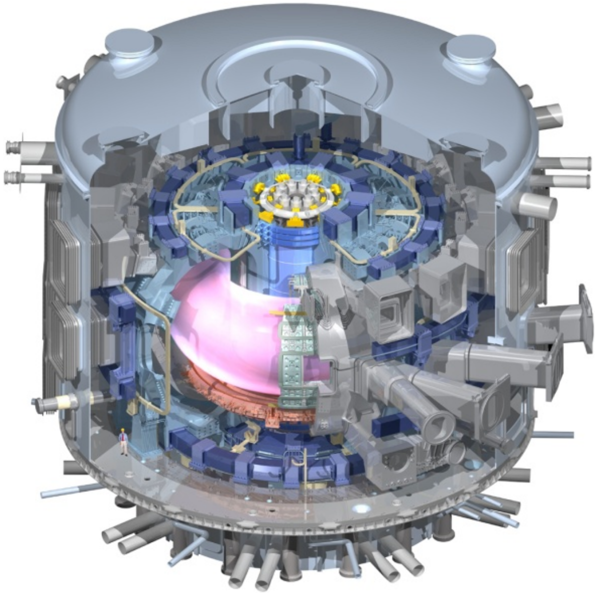
ITER will be first device to access “burning plasma”

- Burning plasma: majority of plasma heating power comes from fusion alpha particles from DT reactions
  - ➢ DT reaction energy split: 1/5 in alphas, 4/5 in neutrons
- ITER goal $Q = P_{fusion} / P_{external\ heating} = 10$
- $Q = 10 \rightarrow P_{alpha} / P_{external} = 2$
- $P_{alpha} / P_{alpha + external} = 2 / 3 > 50\%$

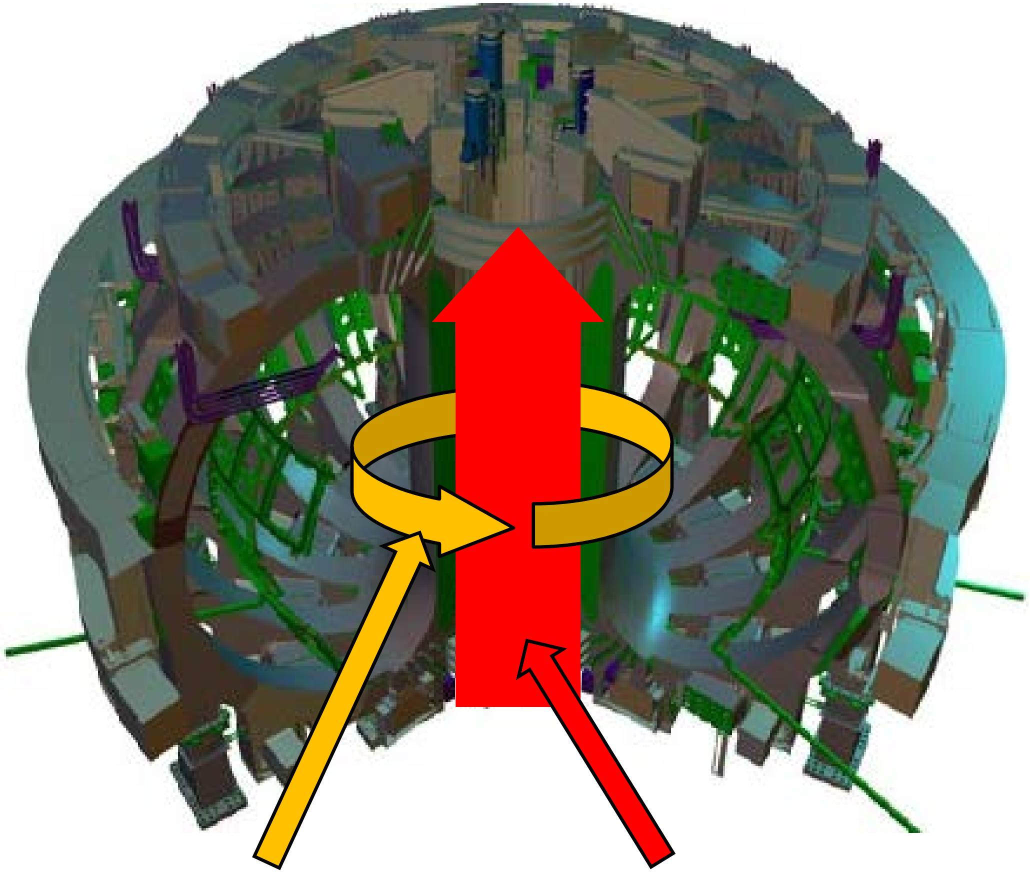
ITER under construction in Cadarache, France



A=3.1, R=6.2m, B_T=5.3T, I_P=15MA



ITER magnets will be largest ever built



Plasma current:
15 million amps

Toroidal field current
165 million amps

- 18 toroidal field magnets
- 12 Tesla at coil
- Weight: 6500 tons
- 80,000 km of Nb_3Sn superconducting strand in total length

These are large numbers

Perspective

- Studying burning plasmas is essential to fusion development, and ITER is presently the best approach

- But as we look beyond plasma self-heating toward economical electricity production, how might we improve?

Assuming cost $\propto$ size → need higher fusion power / volume = high fusion power density

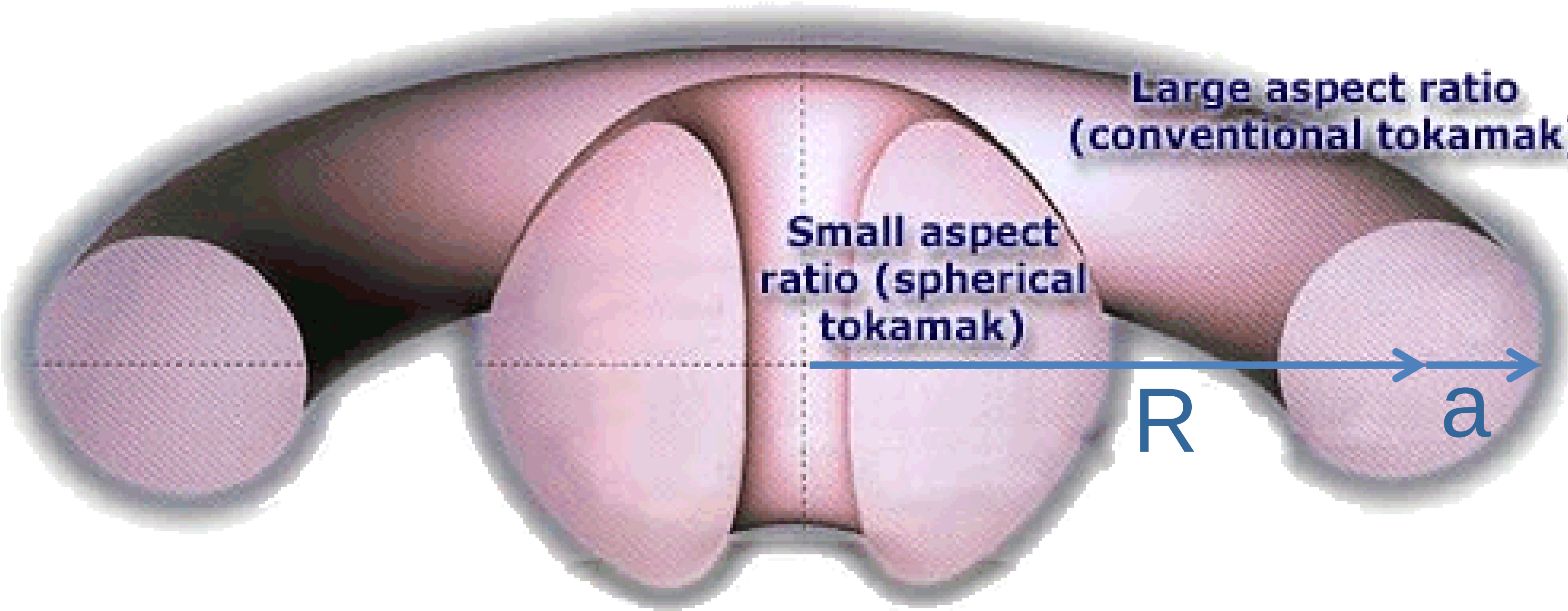
- Fusion power density $\propto$ (plasma pressure)^2
- $\beta \equiv$ plasma pressure / magnetic pressure = $p/(B^2/2\mu_0)$
- Maximum β limited by MHD instabilities
- B limited by magnet stress, cooling, quench
- Fusion power density $\propto$ $\boxed{\beta^2 B^4}$

Maximize / optimize this product

Aspect ratio is important free parameter

Aspect ratio A = R / a

R = major radius   a = minor radius

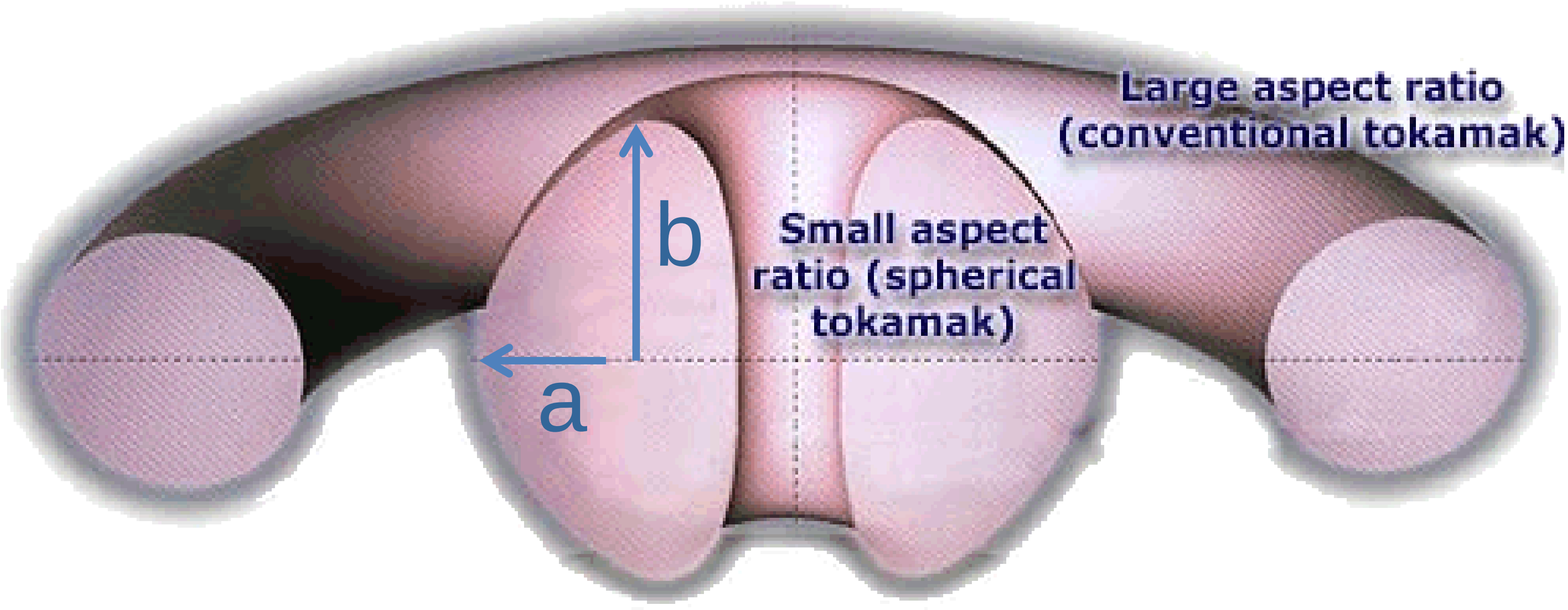


Spherical torus/tokamak (ST) has A = 1.1-2
Conventional tokamak typically A = 2.5-4

STs have higher natural elongation

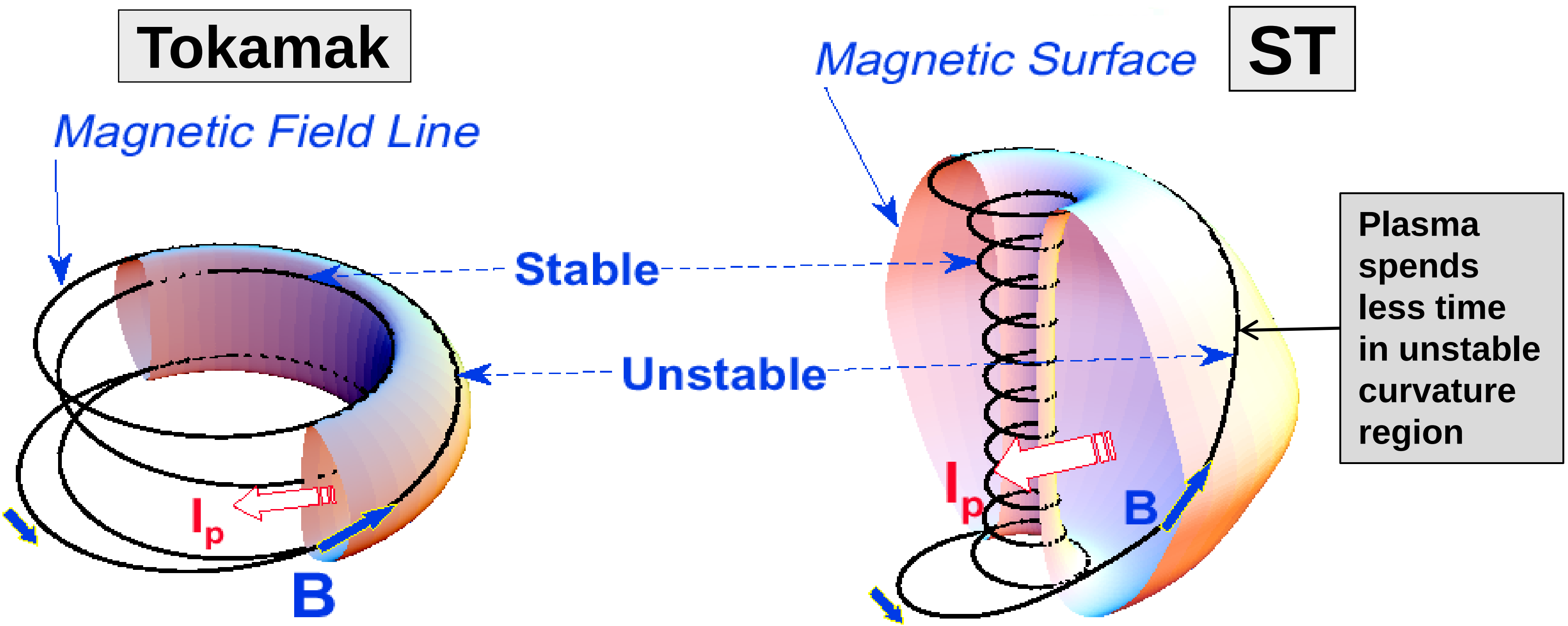
Elongation $\kappa = b / a$

b = vertical ½ height a = minor radius



Higher elongation improves stability, confinement

Favorable average curvature improves stability

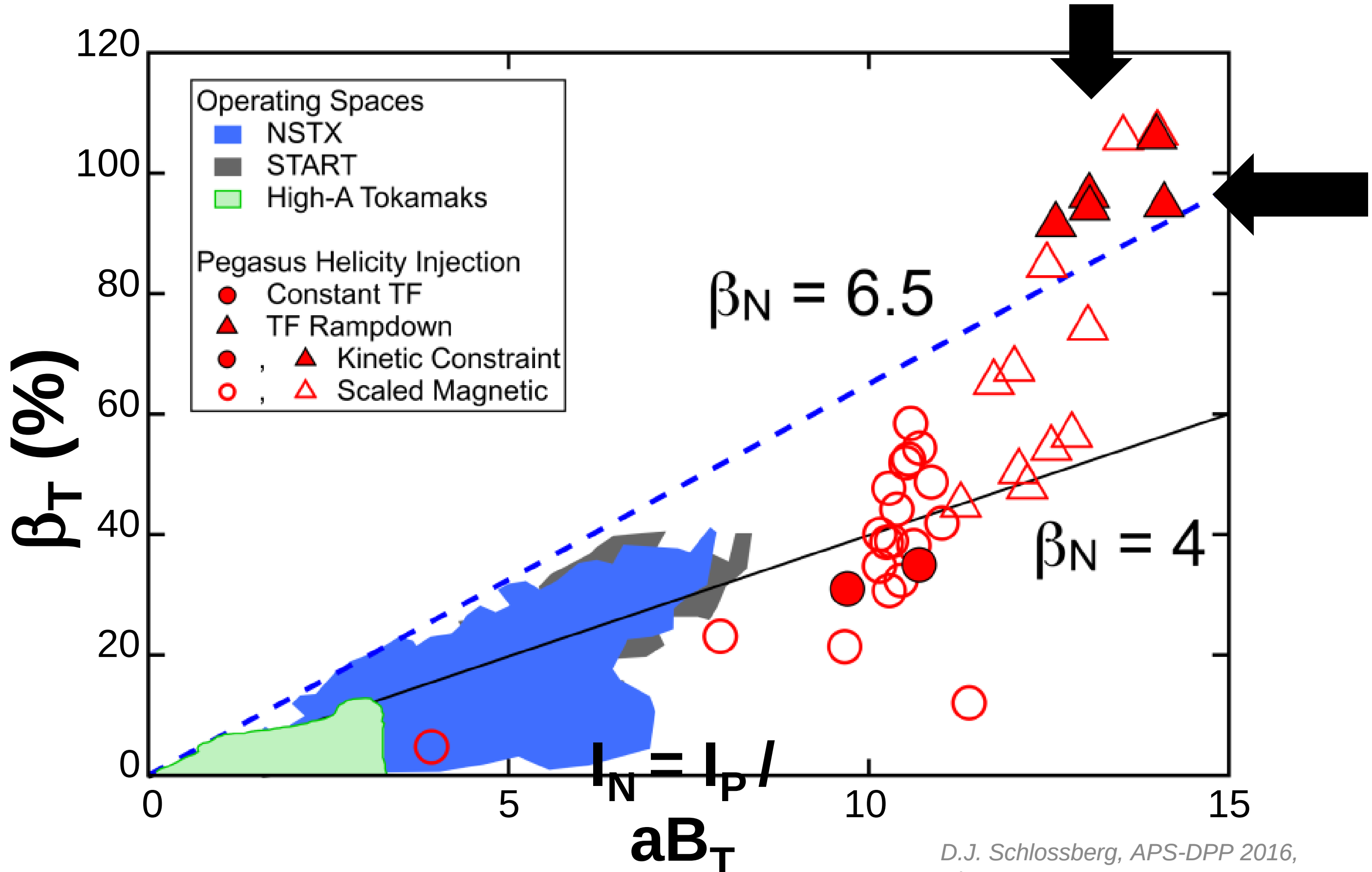


Tokamak	ST
A ~ 2.5-4	A ~ 1.3-2
κ = 1.5-2	κ = 2-3
β_T = 3-10%	β_T = 10-40%

Aspect Ratio $A = R/a$	Elongation $\kappa = b/a$	Toroidal beta $\beta_T = \langle p \rangle / (B_{T0}^2/2\mu_0)$

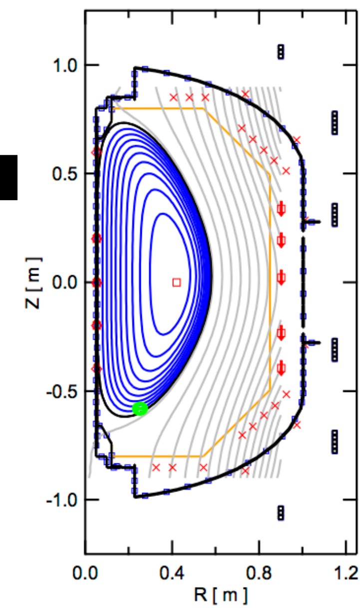
STs can access very wide range of β_T

Pegasus ST recently accessed β_t ~ 50-100%



D.J. Schlossberg, APS-DPP 2016, 9/10

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



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

Why explore spherical torus/tokamak?

- Potentially attractive for electricity production
  - Requires accompanying magnet innovations
- High neutron wall loading in small device
  - Well suited for fusion nuclear component R&D
- Improve toroidal physics predictive capability
  - High β and high temperature at low collisionality
  - Understand confinement, fast-ion physics for ITER

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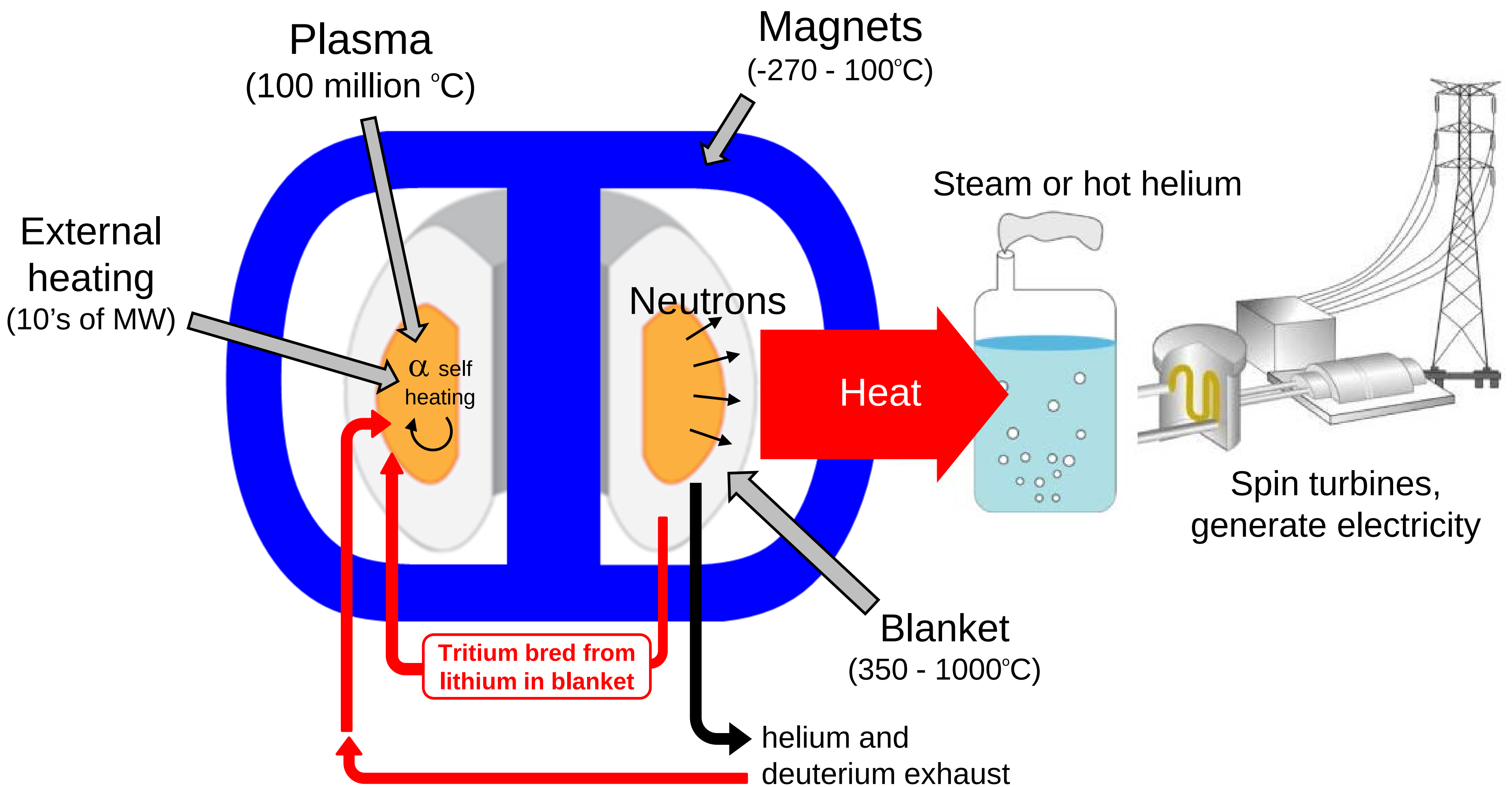
How would magnetic fusion make electricity?



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

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

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

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×P_{th}
- P_{sub} + $P_{control}$ = 0.04×P_{th}
- η_{aux} = 0.3
- $\eta_{CD} = I_{CD}R_0n_e/P_{CD} \approx 0.3\times10^{20}$ A/W/m^2

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

Gain is very strong function of confinement: $Q_{DT} \propto H^{2\rightarrow 5}$ from low → high gain

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

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

$$\tau_E \propto H\, I_P^{\alpha_I} B_T^{\alpha_B} n_e^{\alpha_n} P^{-\alpha_P} R^{\alpha_R} \kappa^{\alpha_\kappa} \epsilon^{\alpha_\epsilon} \qquad \varepsilon \equiv A^{-1}$$

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

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

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

Fix current, field, density, geometry, auxiliary power, α_P = 0.7:

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

Gain vs. physics & engineering constraints

- **Steady-state tokamaks:** current-driven kink limit less relevant
- Normalized β (β_N) and "bootstrap" fraction (f_{BS}) more important
- Relevant variables are $\boldsymbol{\beta_N / f_{BS}}$ and normalized density $\boldsymbol{f_{gw}}$ →

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

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

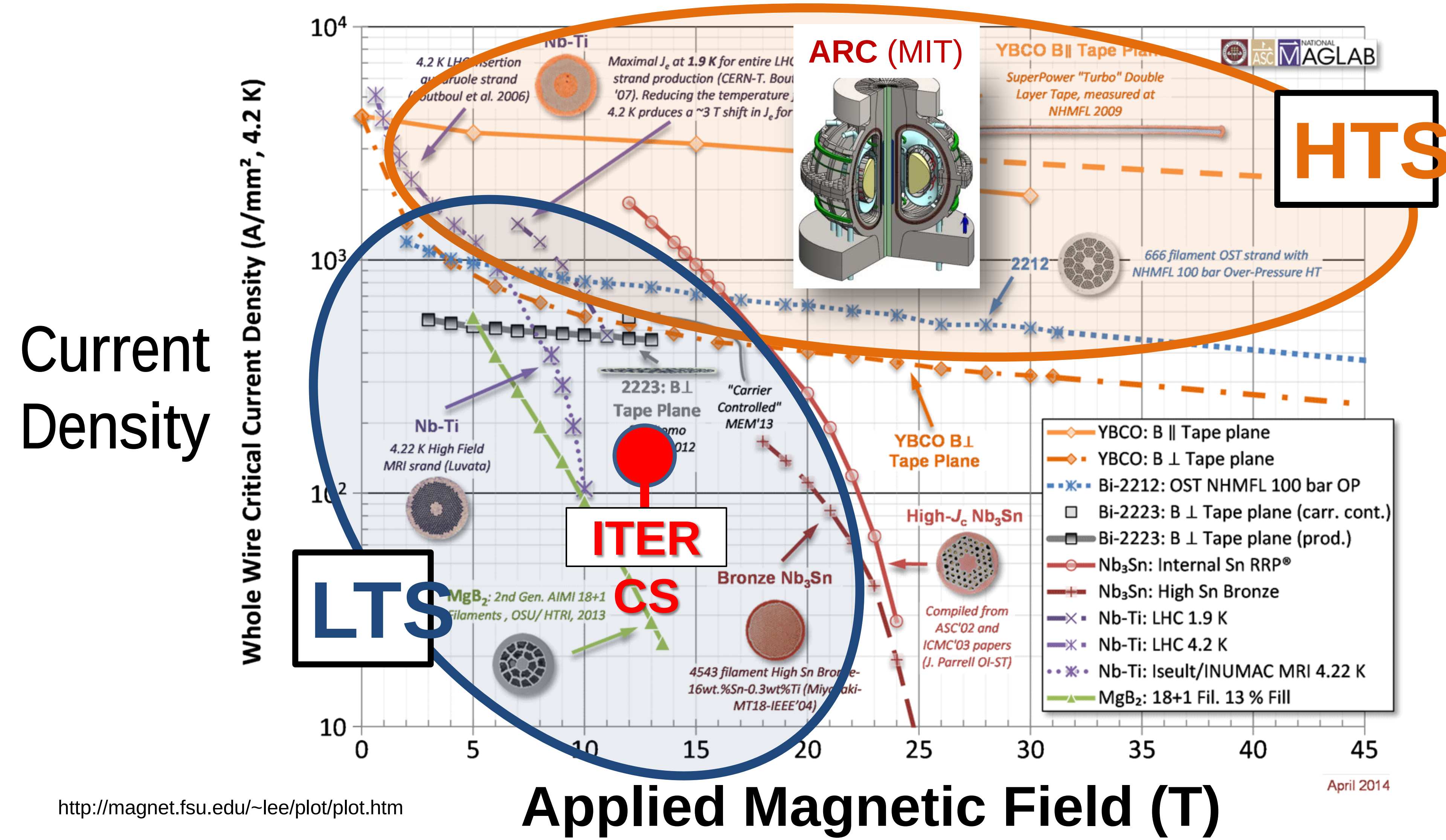
Use electrostatic gyro-Bohm τ_E scaling with no β degradation (NSTX, JET, DIII-D)

$$Q^*_{DT} \propto R^2 H^2 (1 - f_{CD})^{-2} f_{gw}^{0.7} \underbrace{B_T^3 \kappa^{3-5} \beta_N^2 \epsilon^{1.6}}_{\text{Need to optimize this product vs. aspect ratio}}$$

External current drive fraction

Need to optimize this product vs. aspect ratio

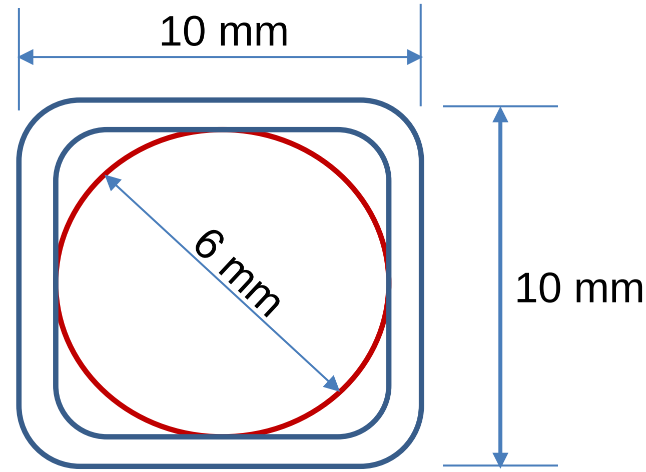
High temperature superconductors (HTS) could substantially expand fusion magnet performance



http://magnet.fsu.edu/~lee/plot/plot.htm

Cables formed from HTS tapes achieving high winding pack current density at high B

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

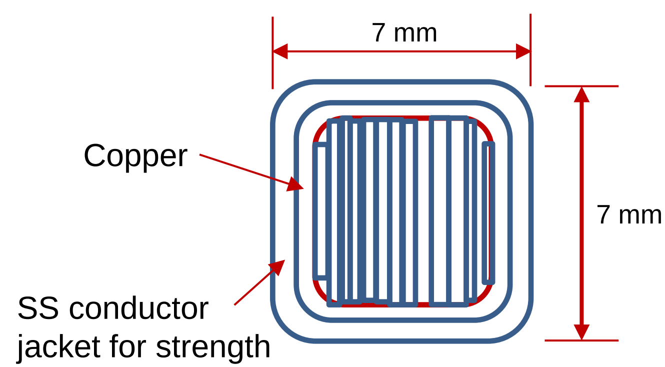


7 kA CORC (4.2K, 19 T) cable

Base cable: 50 tapes YBCO Tapes with 38 mm 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

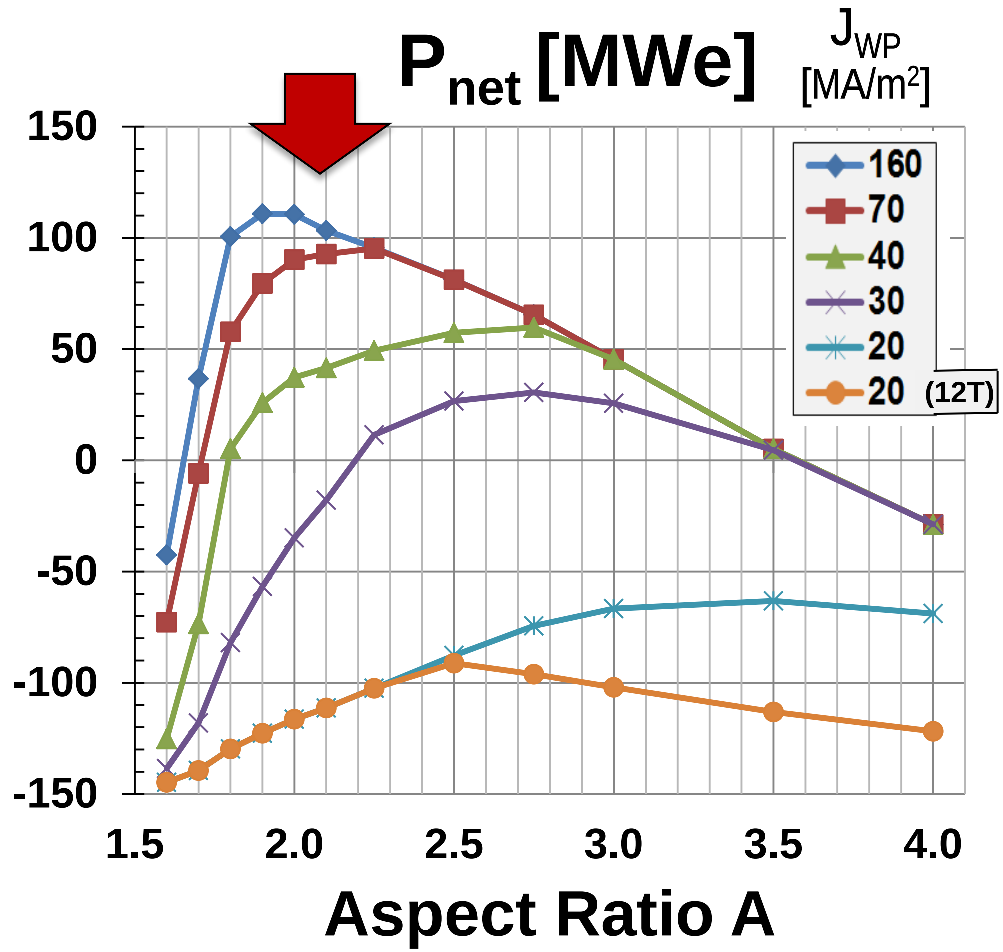


High current density HTS cable motivates consideration of lower-A tokamak pilot plants

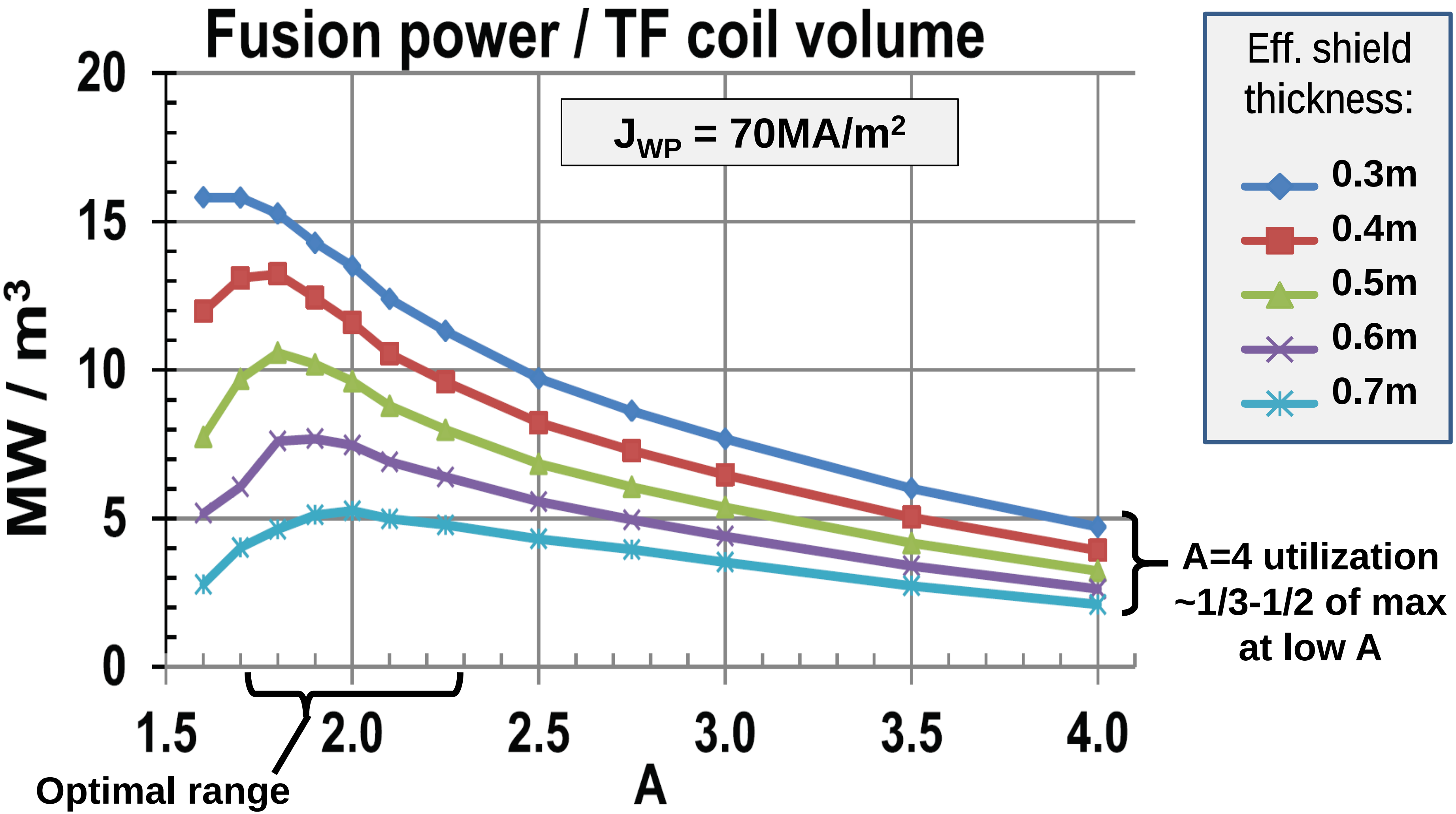
Fix plasma major radius R_0=3m, heating power P_{NNBI}=50MW

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

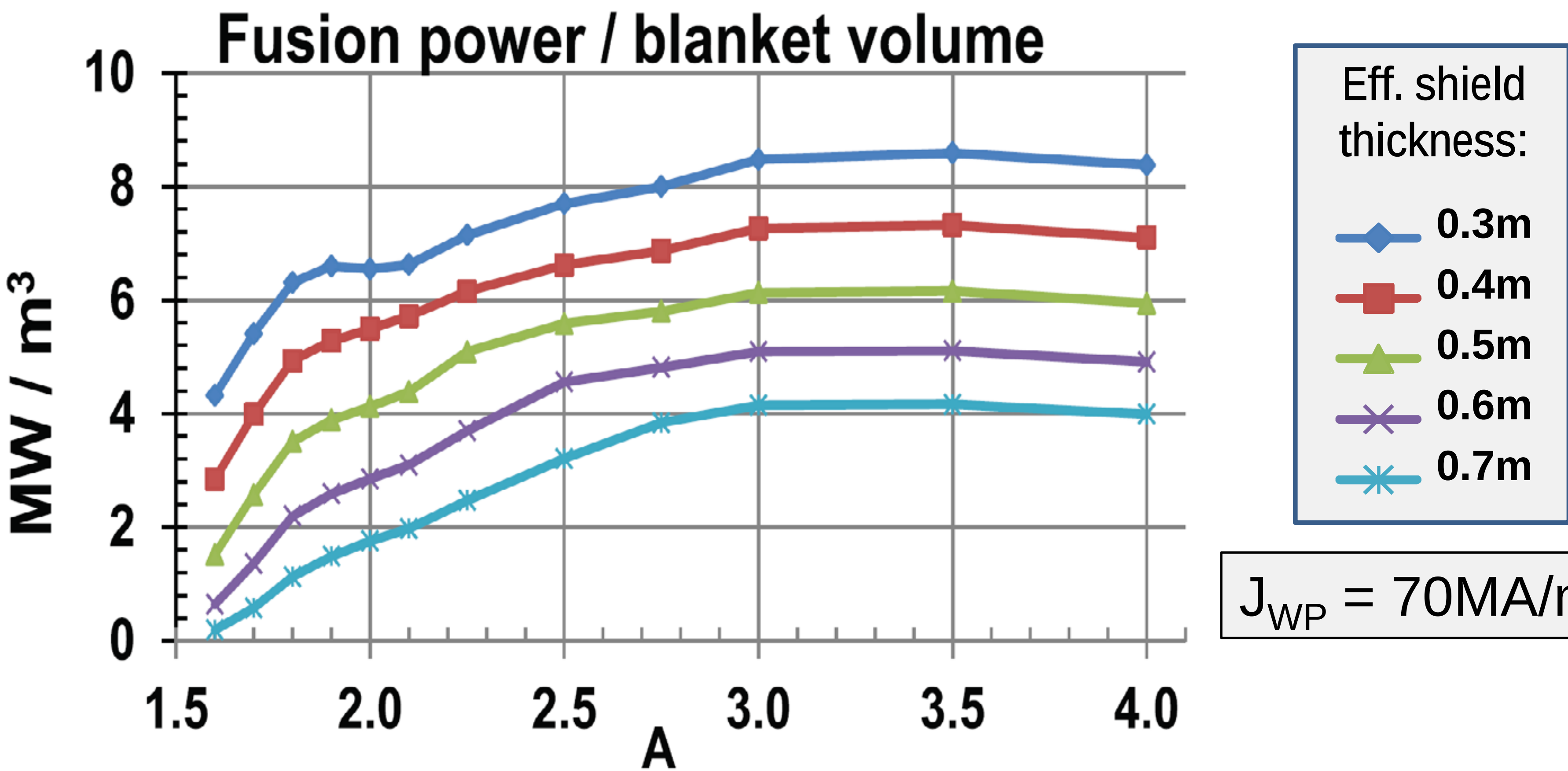
A ~ 2 attractive at high J_{WP}



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



A ≥ 3 maximizes blanket volume utilization

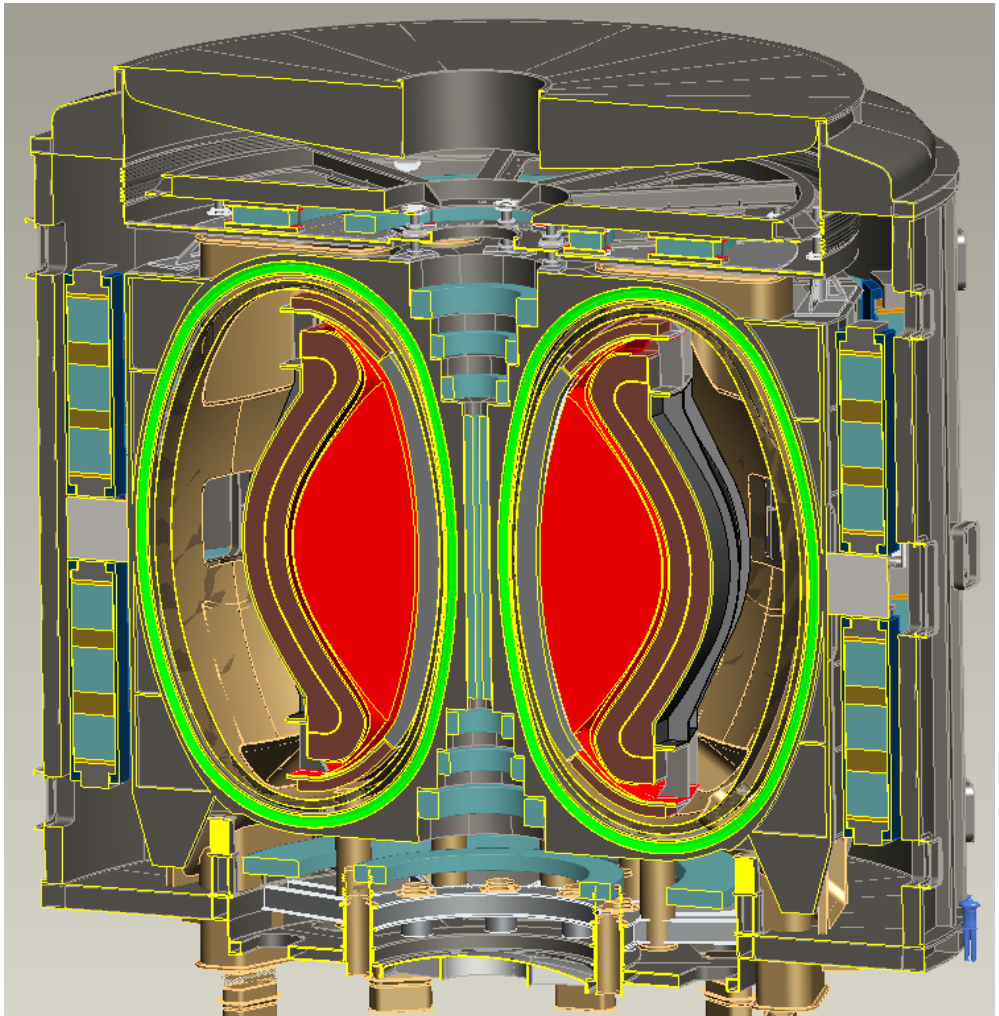


J_{WP} = 70MA/m^2

Which components - magnets, blankets, ... dominate cost ?

Costing of these (never-been-built) objects has large uncertainty

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



Cryostat volume ~ 1/3 of ITER

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

Startup I_P (OH) ~ 2MA
J_{WP} = 70MA/m^2
$B_{T\text{-}max}$ = 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

Why explore spherical torus/tokamak?

- Potentially attractive for electricity production
  - Requires accompanying magnet innovations

- High neutron wall loading in small device
  - Well suited for fusion nuclear component R&D

- Improve toroidal physics predictive capability
  - High β and high temperature at low collisionality
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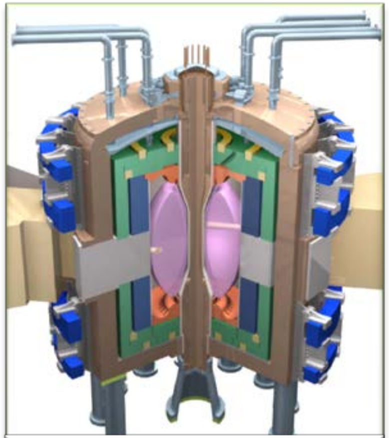
Fusion technology development is major challenge

Fusion Nuclear Science Facility (FNSF) could aid development

Need to develop reliable and qualified nuclear components unique to fusion:

- Divertor and plasma facing components
- Blanket and first wall
- Vacuum vessel and shielding
- Tritium fuel cycle
- Remote maintenance



ST-FNSF

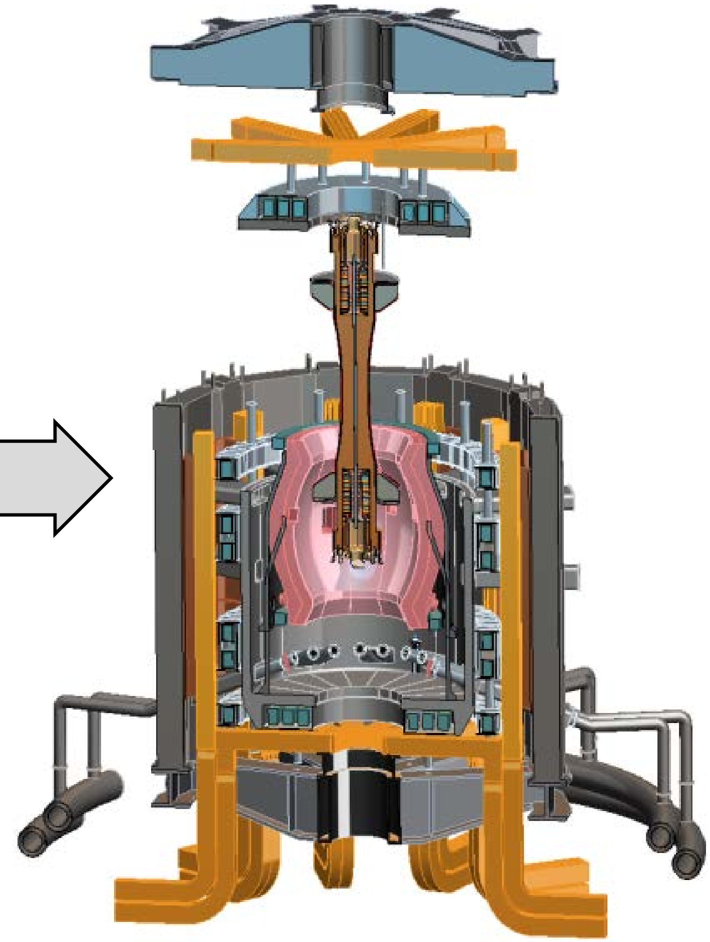
Y.-K.M. Peng (ORNL)

- **Without R&D, fusion components could fail prematurely, requiring long repair/down time.**
- This would cripple power plant operation
- FNSF can help develop reliable fusion components
- Such FNSF facilities must be: modest cost, low T, reliable

Design studies show ST potentially attractive as FNSF

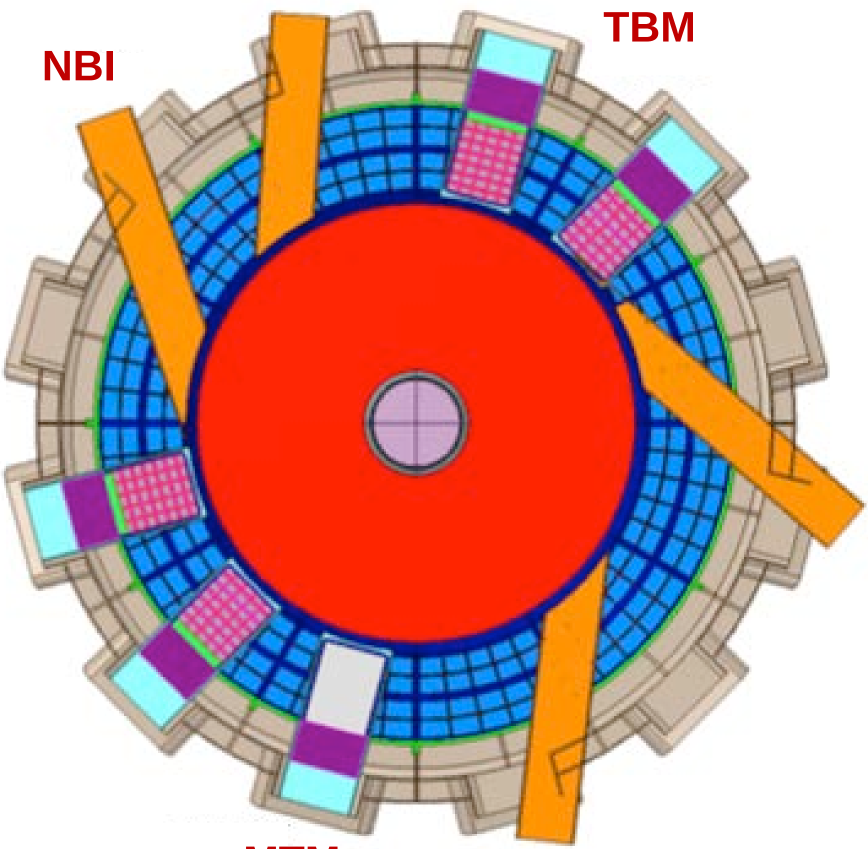
- **Projected to access high neutron wall loading at moderate R, P_{fusion}**
  - $W_n \sim 1\text{-}2\ MW/m^2$
  - $P_{fus} \sim 50\text{-}200MW$
  - $R \sim 0.8\text{-}1.8m$
- **Modular design, maintenance**
- **Tritium breeding ratio (TBR) near 1**
  - Requires sufficiently large R
  - Careful layout / design

PPPL ST-FNSF concept

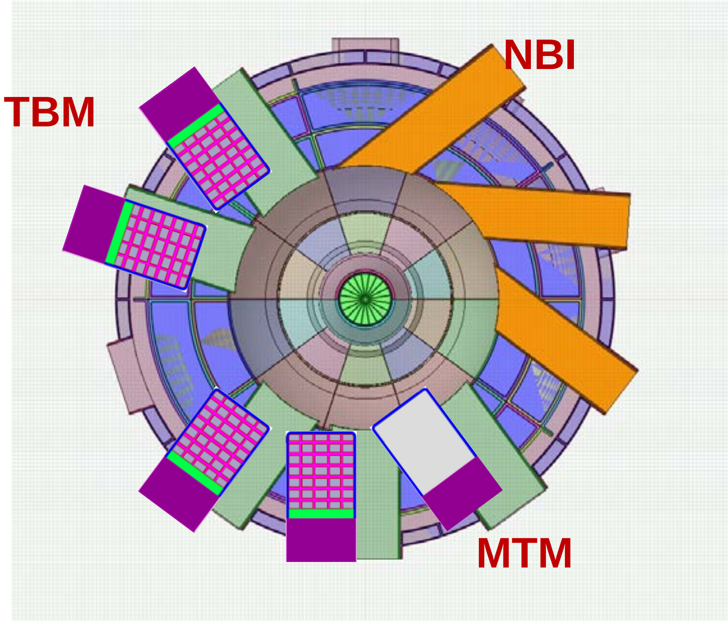


R ≥ 1.7m necessary for net breeding at A=1.7

R=1.7m: **TBR ≥ 1**



R=1.0m: **TBR < 1 (≈ 0.9)**



TBM = Test Blanket Module
MTM = Materials Testing Module

- Need to purchase Tritium from outside sources:
  – $12-55M / full power year (FPY)

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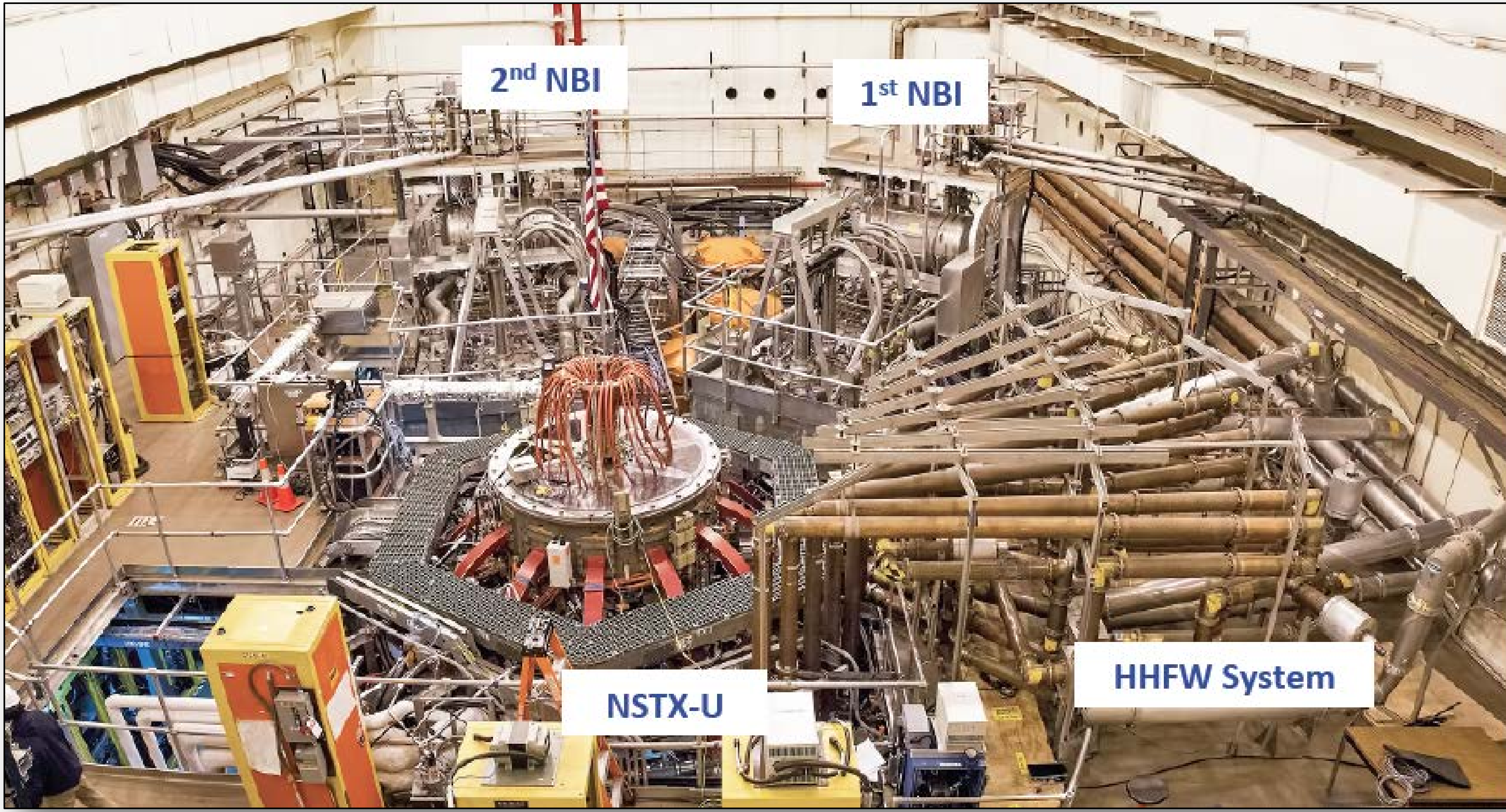
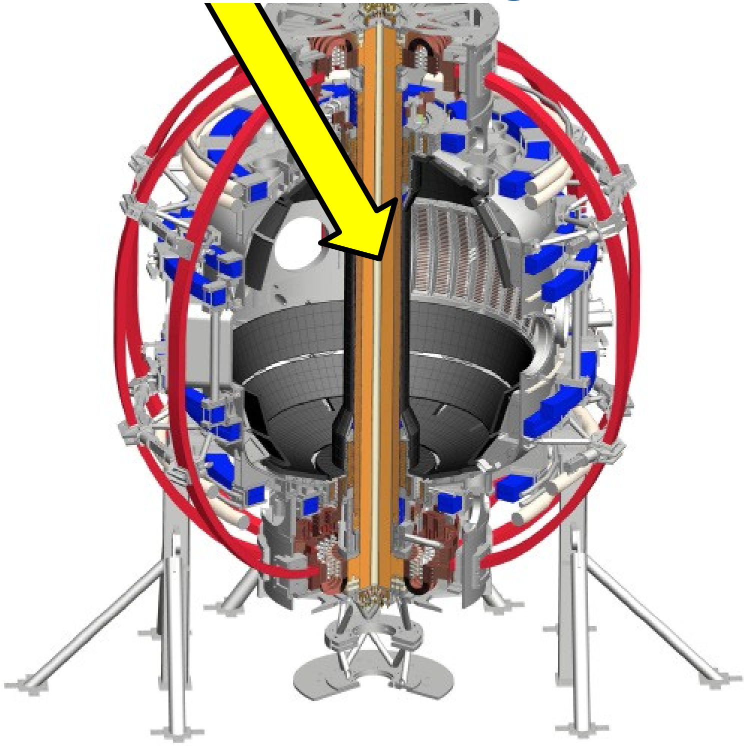
NSTX Upgrade Device and Test Cell – Aerial View



NSTX Upgrade will access new physics with 2 major new tools:

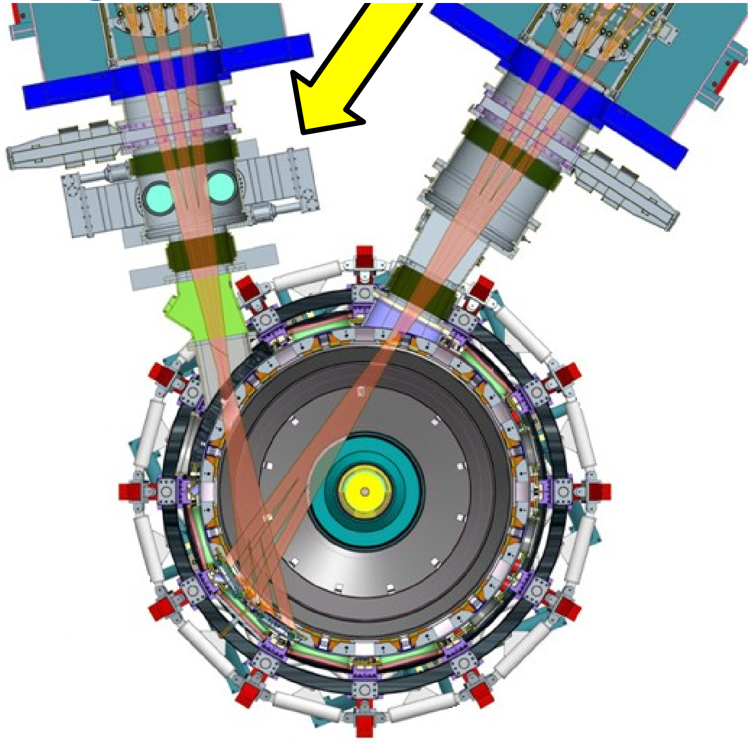
1. New Central Magnet



Higher T, low ν^* from low to high β

→ Unique regime, study new transport and stability physics

2. Tangential 2nd Neutral Beam



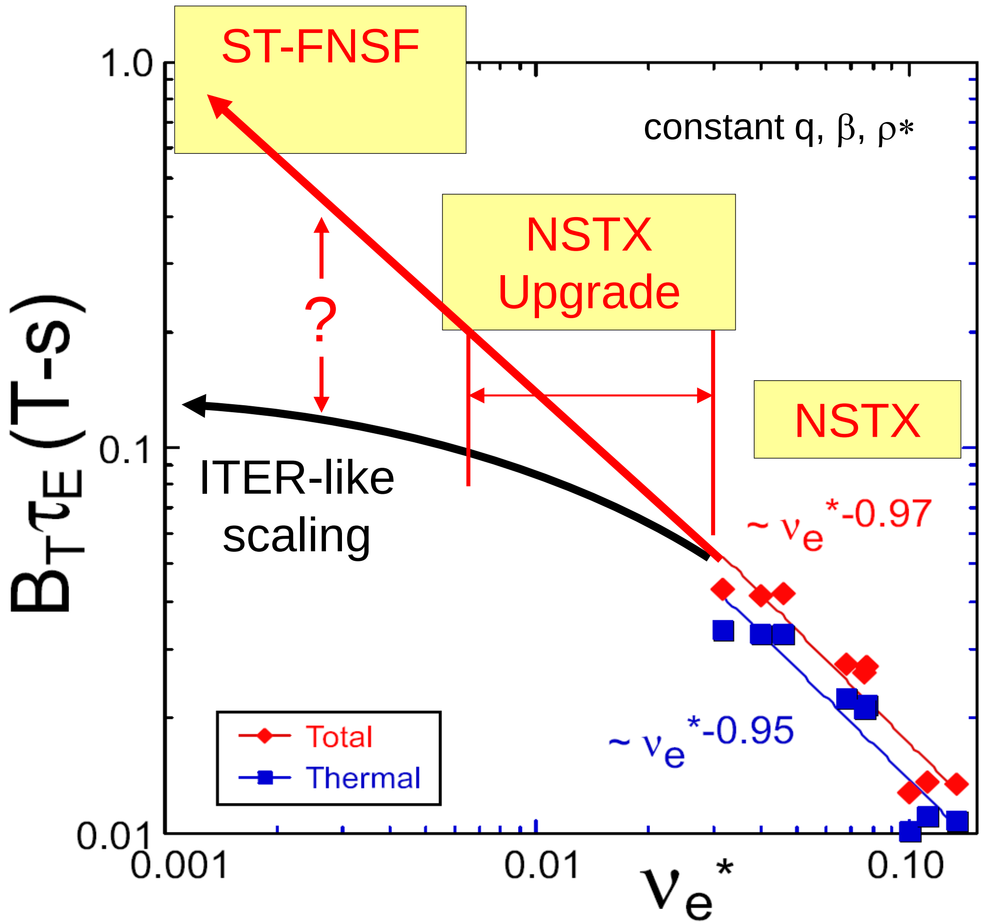
Full non-inductive current drive

→ Not demonstrated in ST at high-β_T

Essential for any future steady-state ST

NSTX / MAST confinement increased at higher T_e (!)

Will confinement trend continue, or look like conventional A?

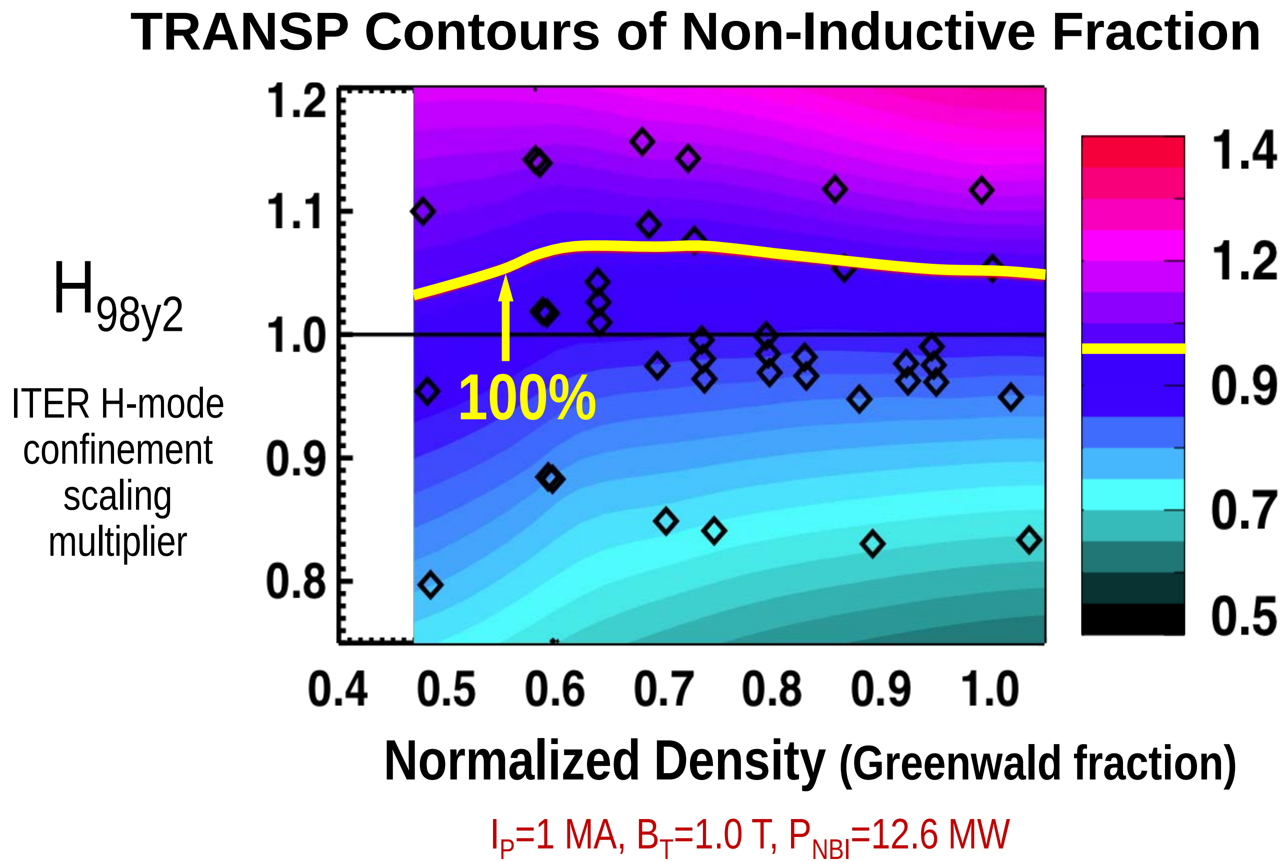


Normalized electron collisionality $\nu_e^* \propto n_e / T_e^2$

Favorable confinement results could lead to more compact ST reactors

Low ν^* → need higher plasma current, toroidal field, heating power, density control

NSTX achieved 70% “transformer-less” current drive
Will NSTX-U achieve 100% as predicted by simulations?



Steady-state operation required for ST, tokamak, or stellarator FNSF

Outline

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- NSTX-U initial results, long-term directions
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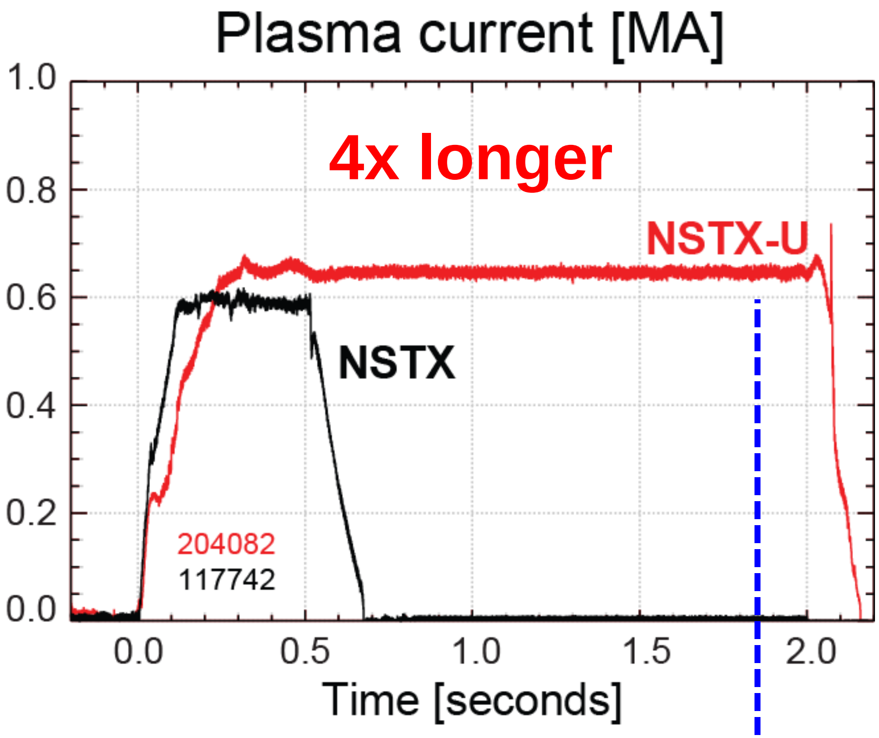
NSTX-U had scientifically productive 1st year

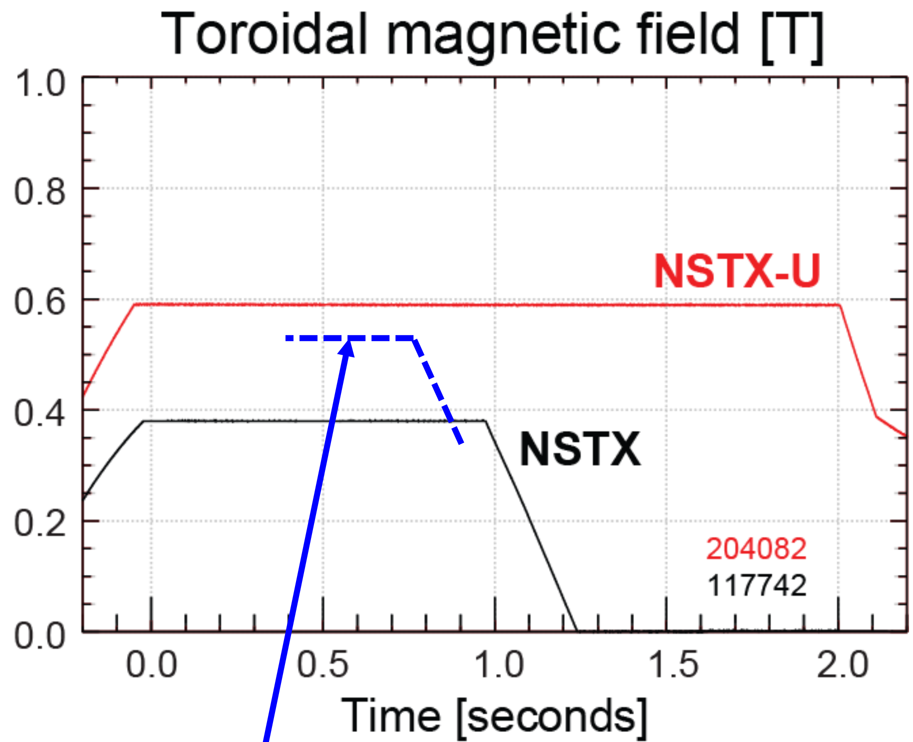
- Achieved H-mode on 8th day of 10 weeks of operation
- Surpassed magnetic field and pulse-duration of NSTX
- Matched best NSTX H-mode performance at ~1MA
- Identified and corrected dominant error fields
- Commissioned all magnetic and kinetic profile diagnostics
- Injected up to 12MW NBI power into armor by end of run
- Discovered new 2nd NBI modifies several fast-ion modes
- Implemented techniques for controlled plasma shut down, disruption detection, commissioned new tools for mitigation

- 2016 run ended prematurely due to fault in divertor PF coil
  - Coil forensics, Extent of Condition → new coil fab, other repairs
  - Aim to resume plasma operation by 2018/19 – but timing still TBD

NSTX-U has surpassed maximum pulse duration and magnetic field of NSTX

Compare similar **NSTX** / **NSTX-U** Boronized L-modes, P_{NBI}=1MW

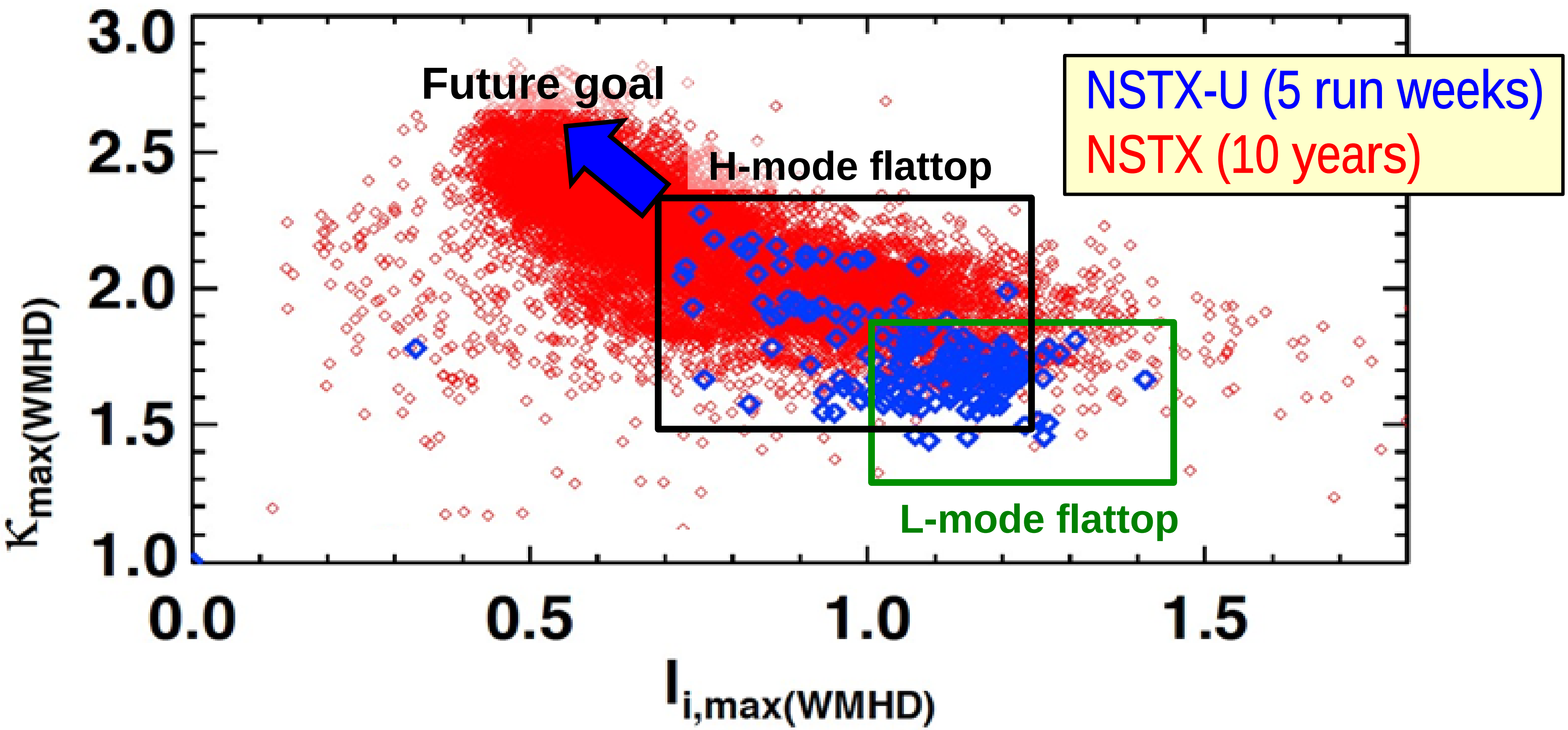




NSTX-U L-mode duration exceeds longest NSTX H-mode

NSTX-U B_T > highest NSTX B_T

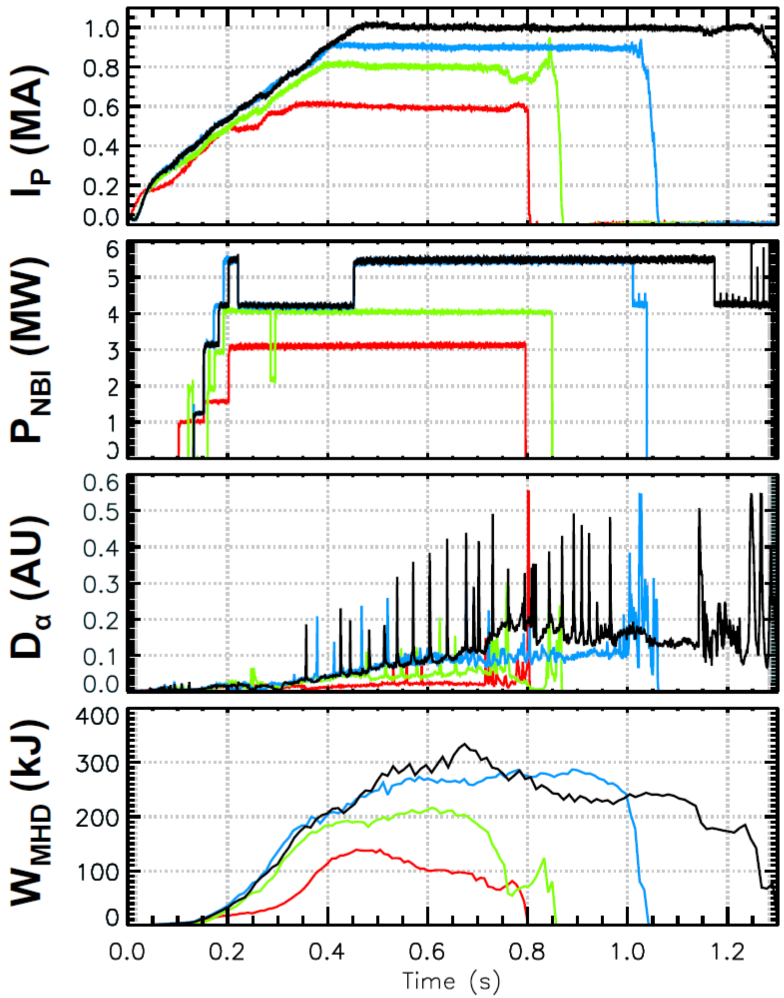
Accessed high elongation κ using progressively earlier H-mode and heating + optimized EFC



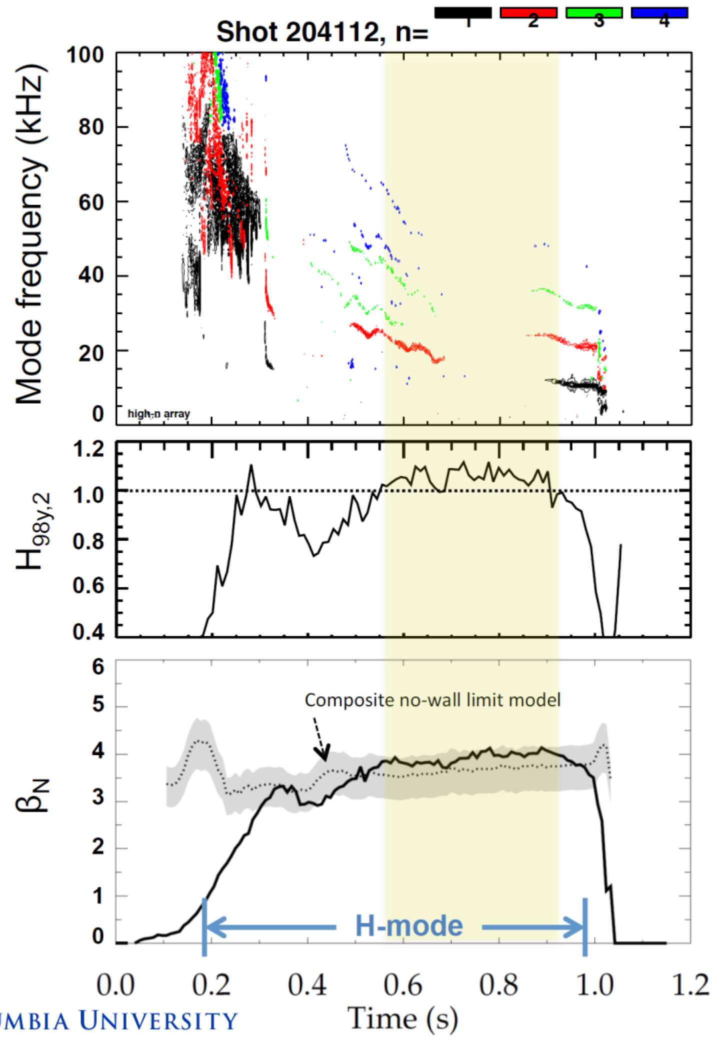
- **Goal: Internal inductance l_i = 0.5-0.7 → κ = 2.4-2.7**

Recovered ~1MA H-modes with performance comparable to best NSTX plasmas at similar current

202946 – no EFC 204112 – EFC v2
203679 – EFC v1 204118 – EFC v2

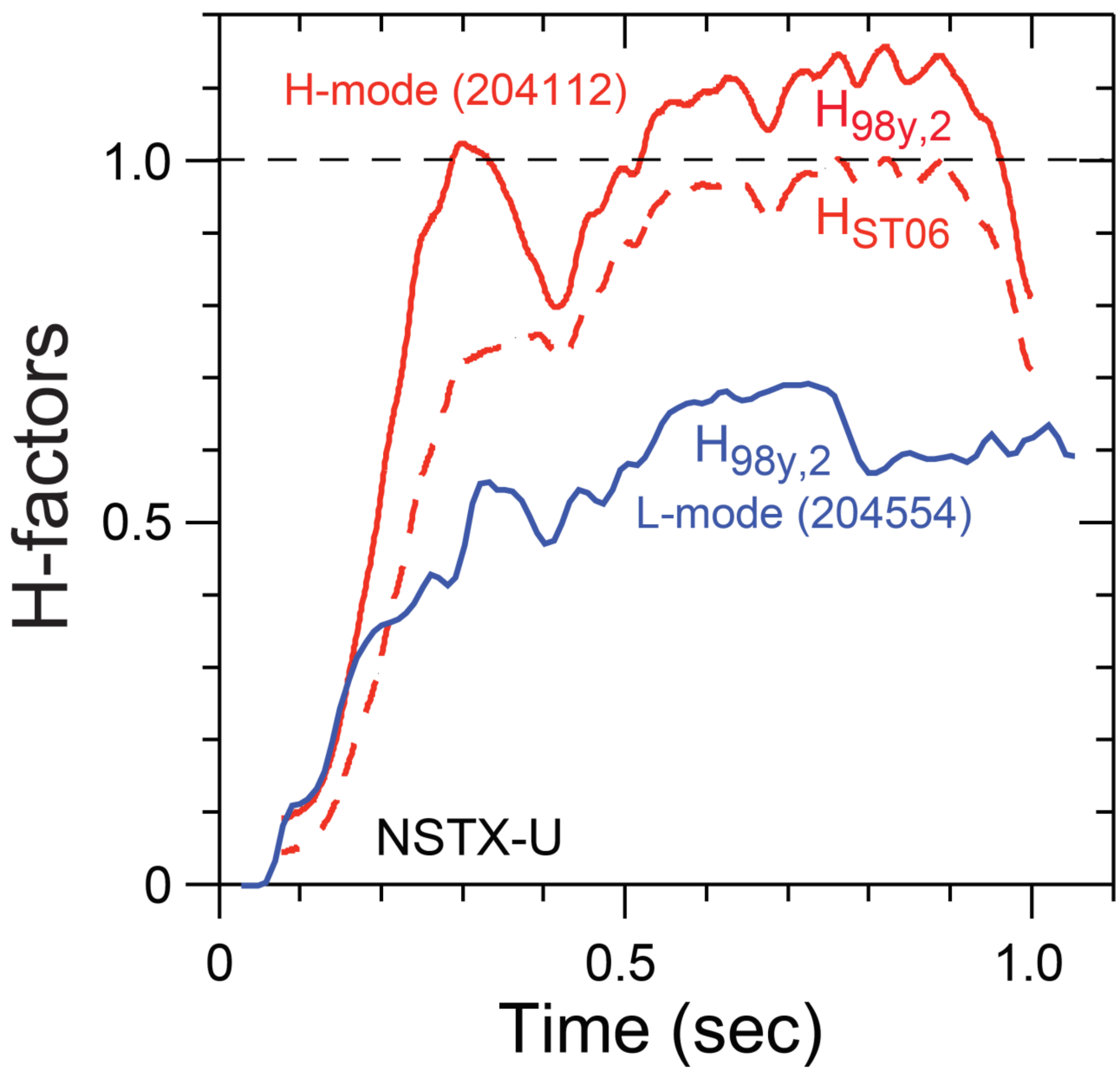


$H_{98} \geq 1$, $\beta_N \sim 3.5$-$4 \geq$ n=1 no-wall limit

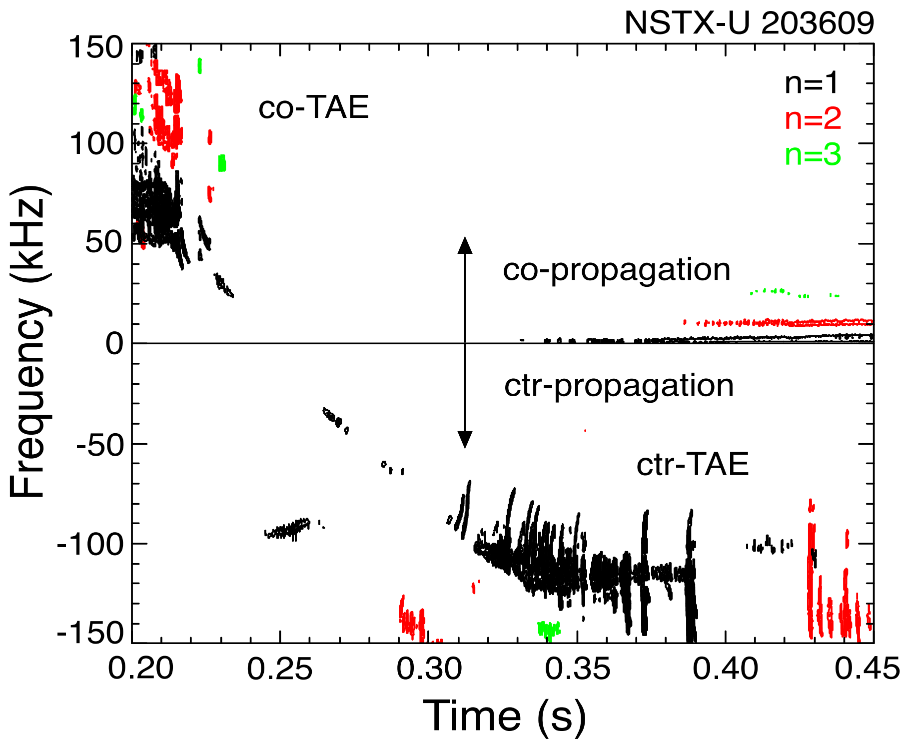


COLUMBIA UNIVERSITY IN THE CITY OF NEW YORK

H-mode confinement > ITER scaling, consistent with ST scaling (so far) – need higher I_P, B_T to test



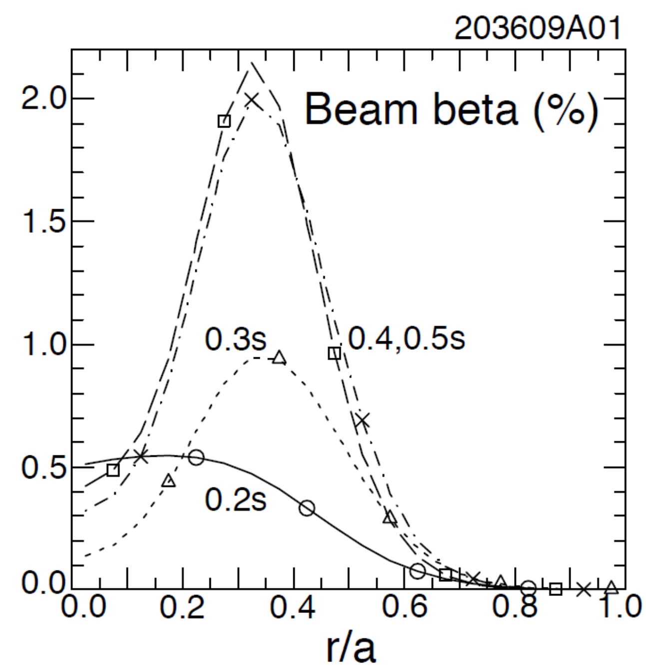
New: Most tangential NBI generates counter-propagating Toroidal Alfvén Eigenmodes (TAEs)



- Counter-propagating TAE predicted for **hollow** fast-ion profiles

*H.V. Wong, H. Berk, Phys. Lett. A **251** (1999) 126.*

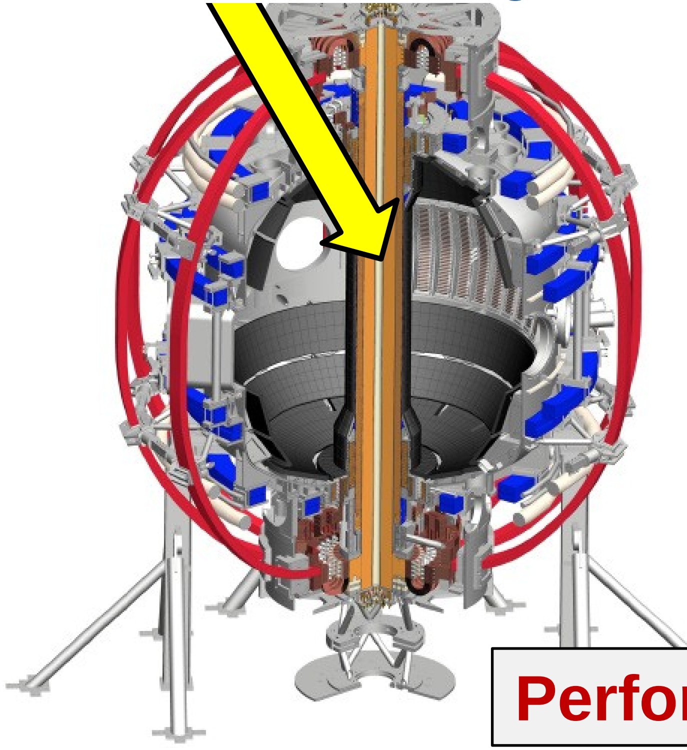
- TRANSP: As current builds up beam fast-ion beta profile predicted to become hollow



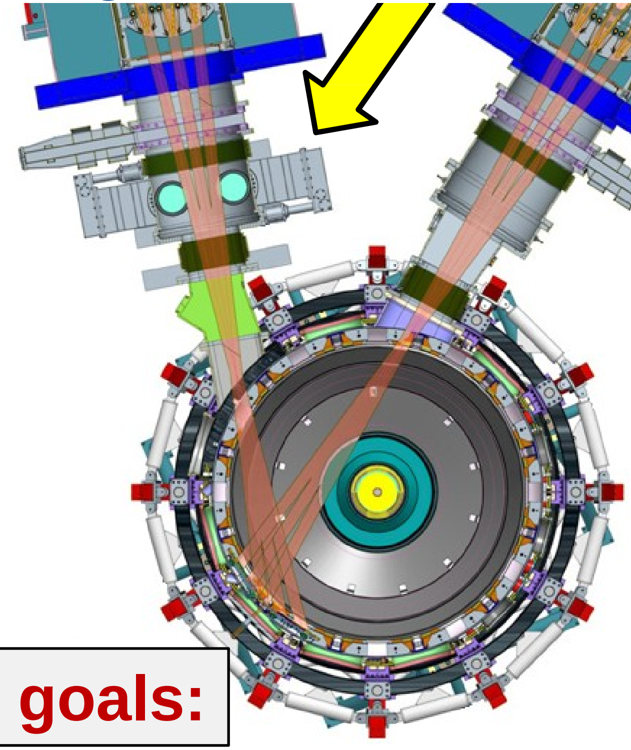
- **1st evidence of off-axis NBI in NSTX-U**

NSTX-U will have major boost in performance

1. New Central Magnet



2. Tangential 2nd Neutral Beam



Performance goals:

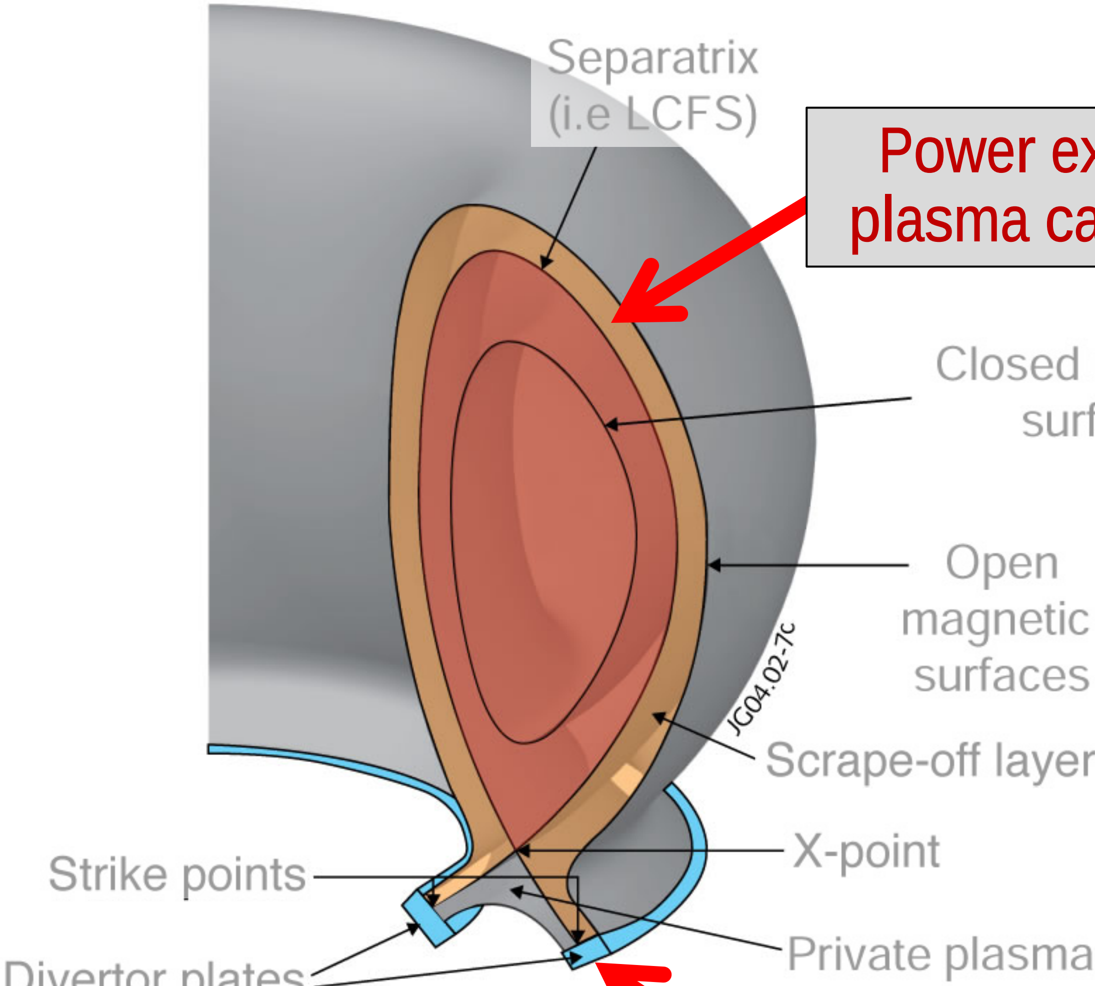
- ➢2× toroidal field (0.5 → 1T)
- ➢2× plasma current (1 → 2MA)
- ➢5× longer pulse (1 → 5s)

- ➢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)

Goals for future NSTX-U operation

- Increase field to 0.8-1T, current to 1.6-2MA, extend flat-top duration (H-mode) to 2-5s
- Assess global stability, energy confinement, pedestal height/structure, edge heat-flux width
- Characterize 2nd beam: heating, current drive, torque / rotation profiles, fast-ion instabilities
- Push toward full non-inductive current drive
- Test advanced divertor heat flux mitigation

All modern tokamaks / STs use a "divertor" to control where power and particles are exhausted



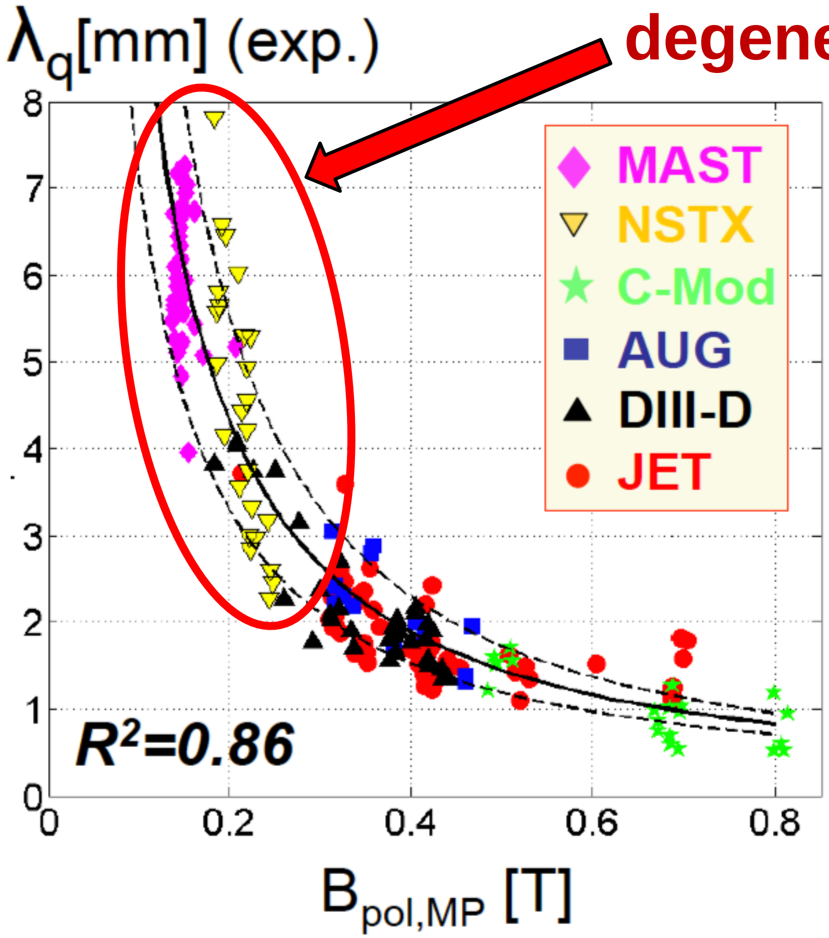
Power exhaust width outside main plasma can be very narrow (few mm)

Power / particles contact target plates here

Tokamak + ST data: power exhaust width varies as 1 / $B_{poloidal}$

Will previous ST trend continue at 2× I_P, B_P, B_T, power?

ST data breaks aspect ratio degeneracy of data set



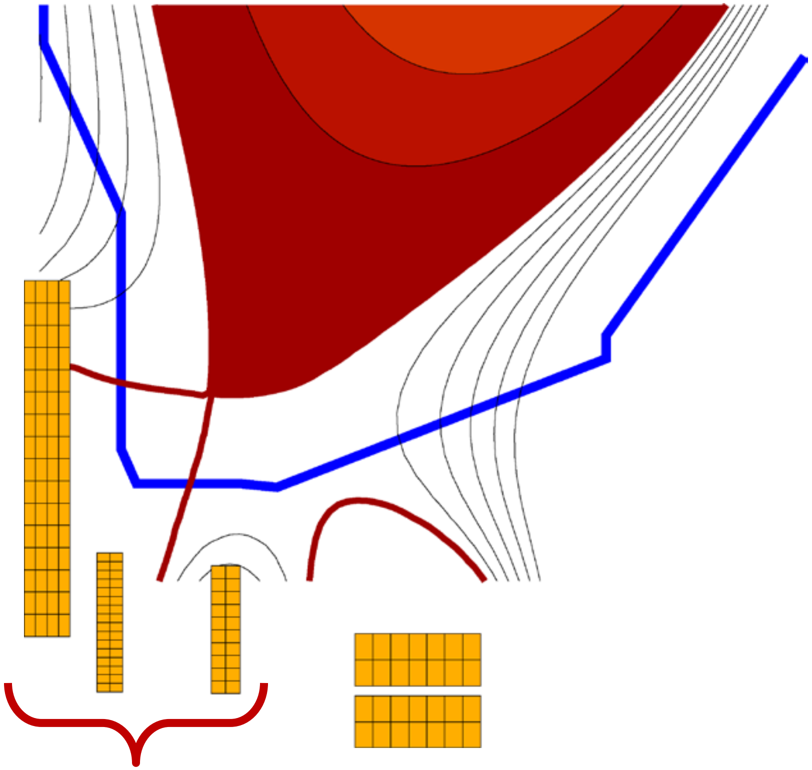
Issue: If peak heat flux in divertor region exceeds ~10 MW/m^2 → material damage

Wider heat-flux width may offset smaller R → maybe better than tokamak

STs leading advanced divertor development

NSTX-U / MAST-U will collaborate on 1st plasma, scenarios, divertors

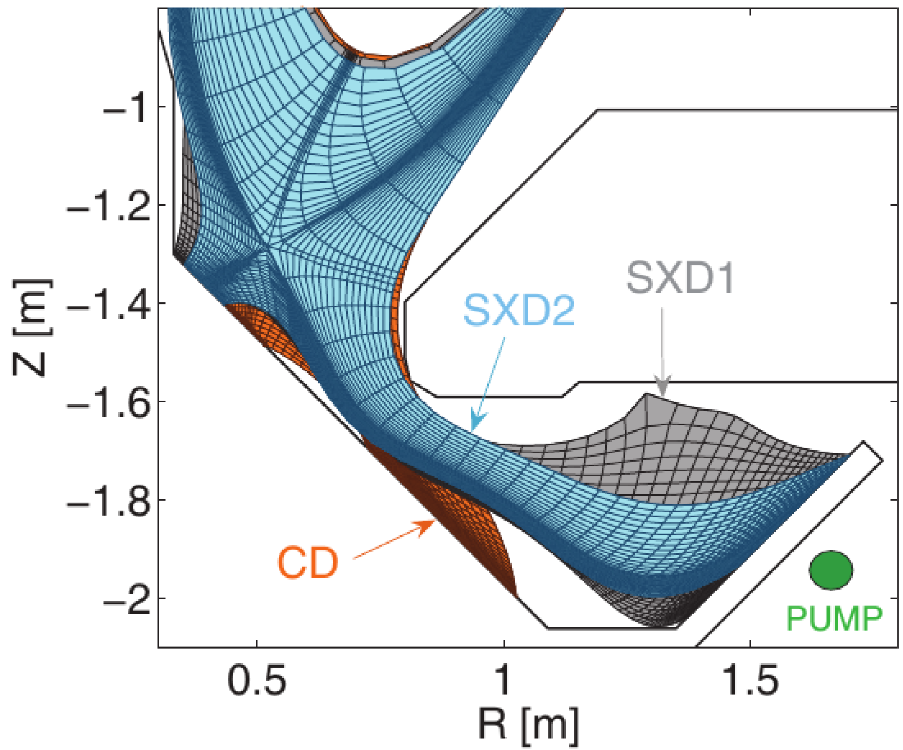
NSTX-U: Flared divertor using "snowflake/X" + radiation



New PF coils in NSTX-U central magnet

MAST-U will test range of divertors:

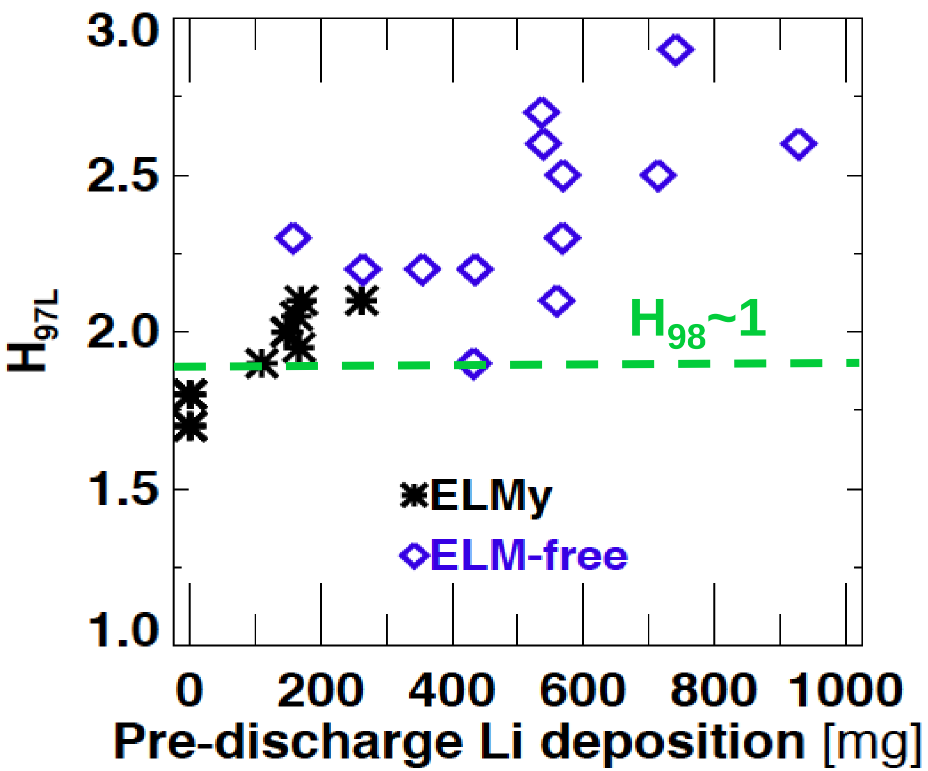
- Conventional, snowflake (not shown)
- Long-leg "Super-X" with variable flaring



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

NSTX (wider → higher pedestals)

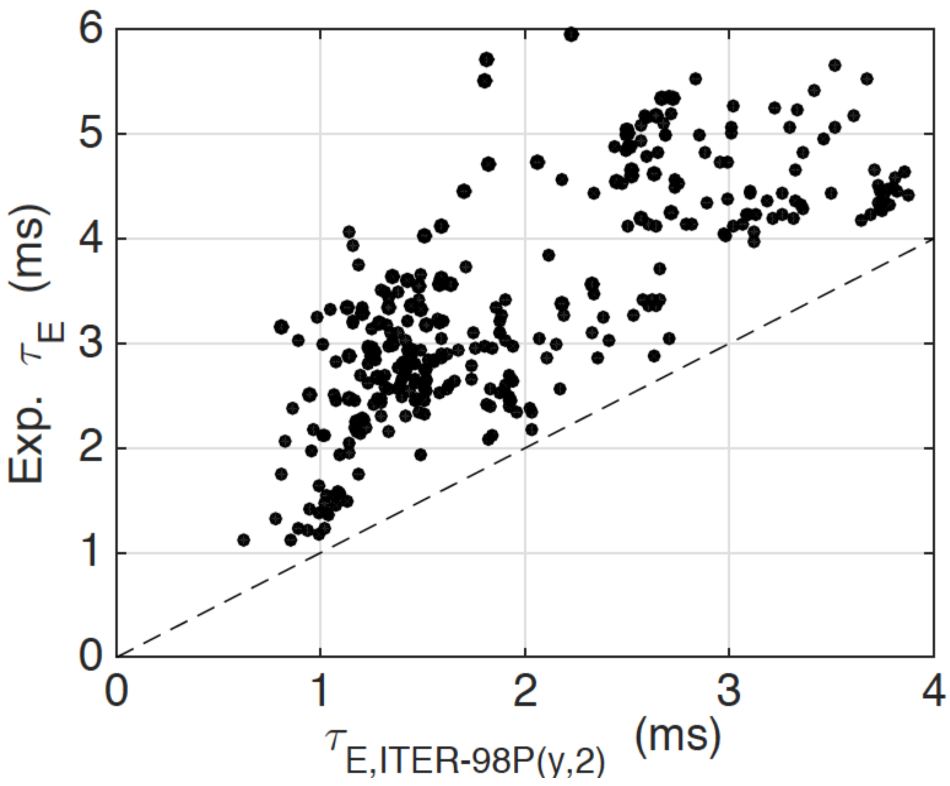
H = 0.8 → 1.4 × ITER98P(y,2)



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

LTX (flatter → higher T profiles)

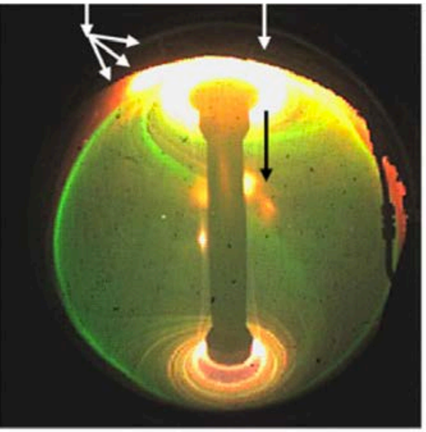
H = 2-4 × ITER98P(y,2)



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

Innovative gravitational dropper used to drop impurities and improve performance in edge of fusion devices

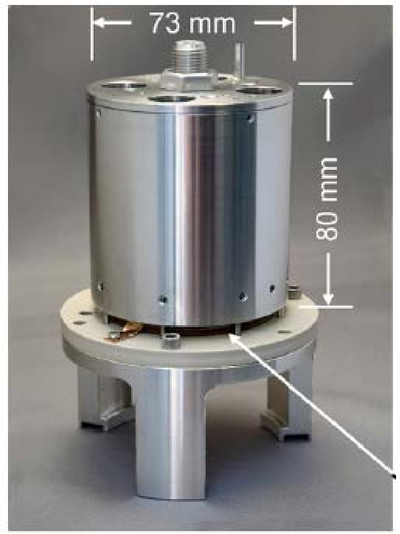
NSTX: ELM suppression



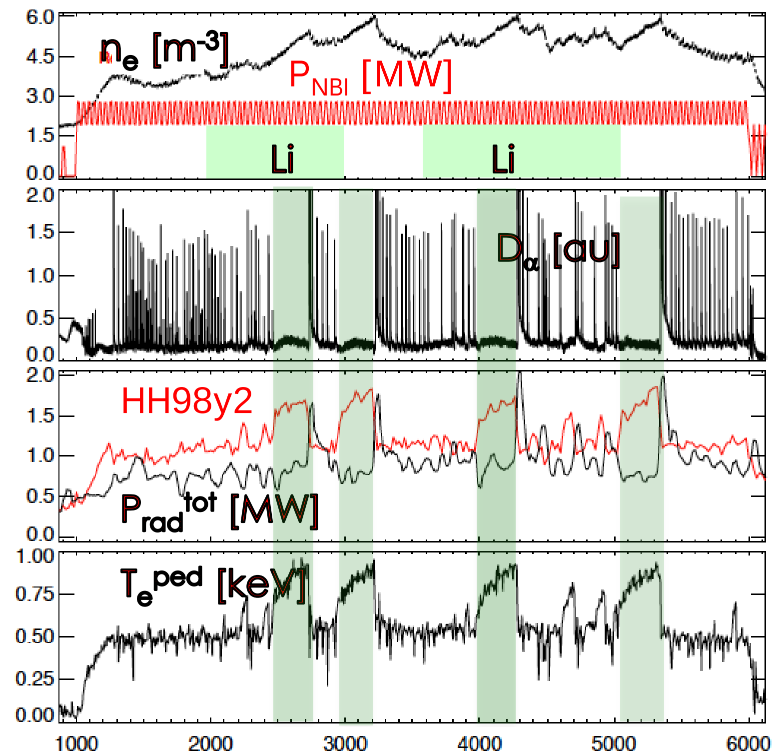
t = 655 ms

ELMs and MARFES vanish at t ~ 500 ms. Stored energy peaks and a sharp plasma edge with Li^{+1} radiative mantle develops.

D. Mansfield, FEDC **85** (2010) 890



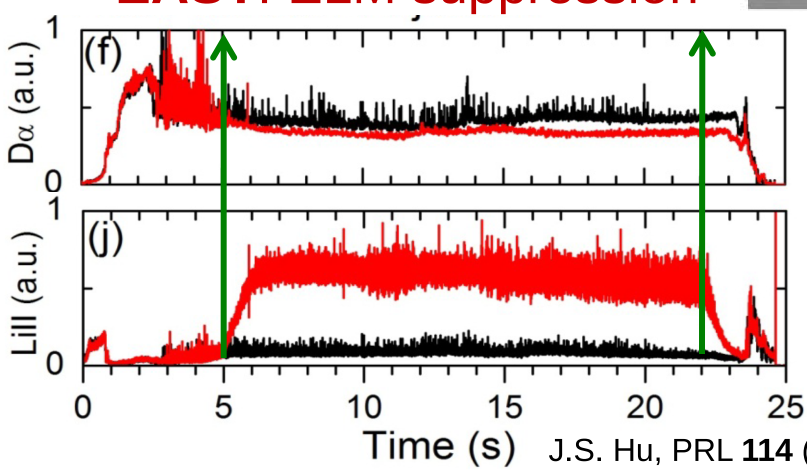
DIII-D: trigger very high confinement bifurcations



T. Osborne, Nucl. Fusion **55** (2015) 063018

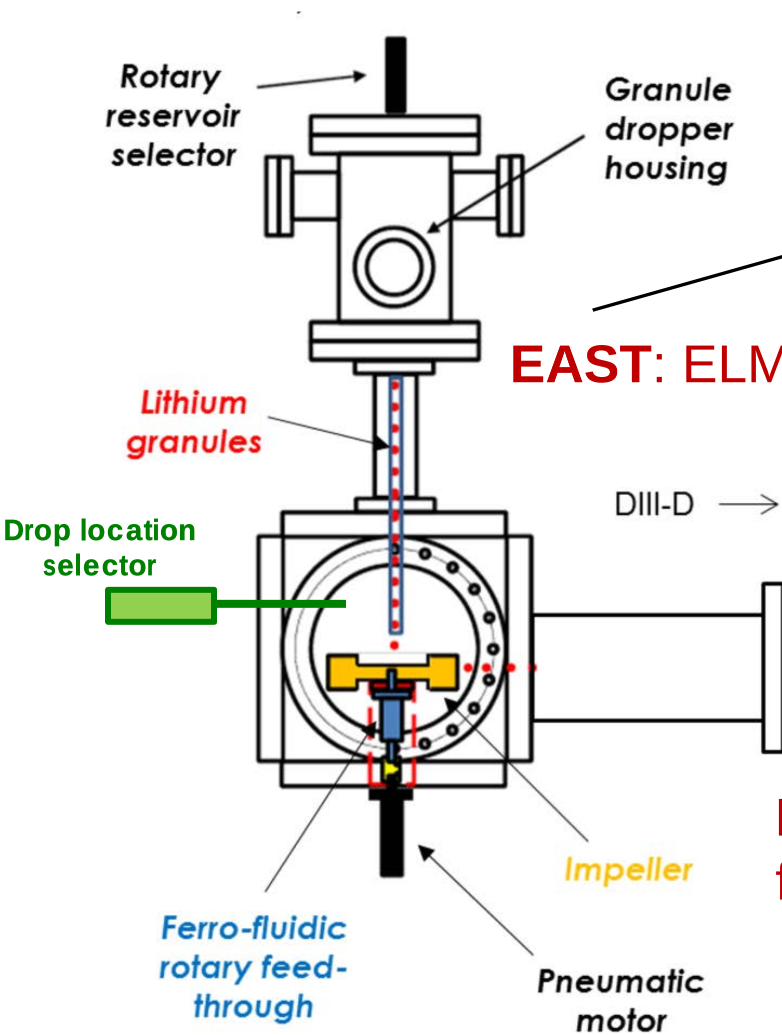
R. Maingi, H-mode Workshop, Oct. 2015

EAST: ELM suppression

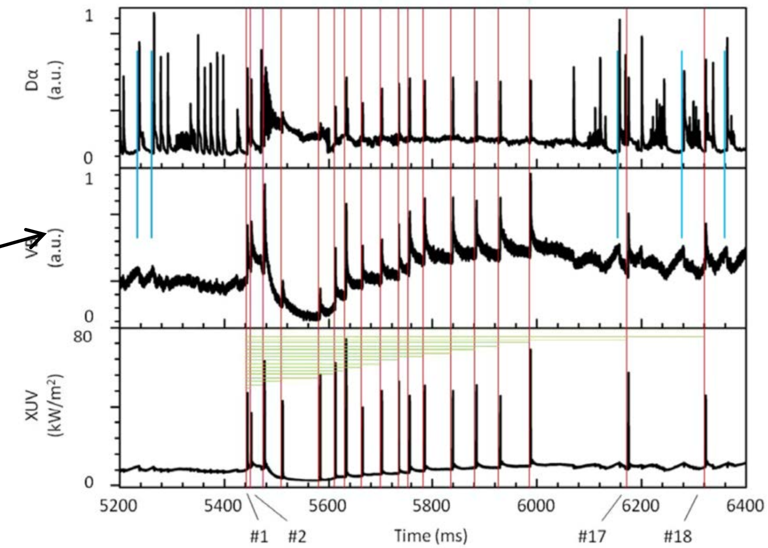


J.S. Hu, PRL **114** (2015) 055001

Impurity granule injector used to pace ELMs and test heat flux mitigation in tokamaks

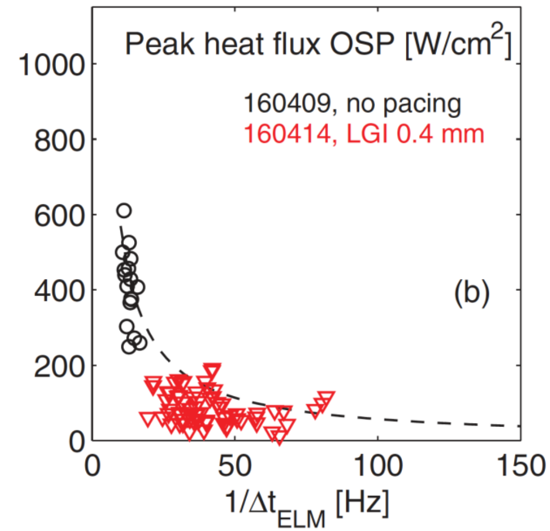


EAST: ELM pacing



D. Mansfield, NF **53**

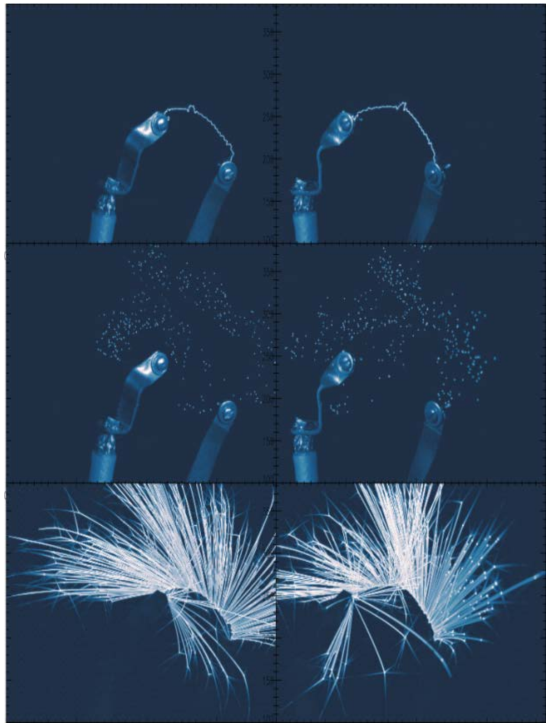
DIII-D: peak heat flux reduction



A. Bortolon, NF **56** (2016) 056008

LANL / NSTX-U Collaborations

- Zhehui Wang pursuing collaborations with NSTX-U
  - Existing and new applications of micropellet injection
    - 4D microparticle tracking (HTPD 2016)
  - Team-member of EAST PMI collaboration
    - Study granule injection, trajectories
- NSTX-U interested in new methods for liquid metal (LM) / lithium delivery
  - Including near divertor region
  - Possible means of power dissipation, LM replenishment on high-Z substrates
- Other LANL interests as part of NSTX-U high-Z / liquid metal plan?

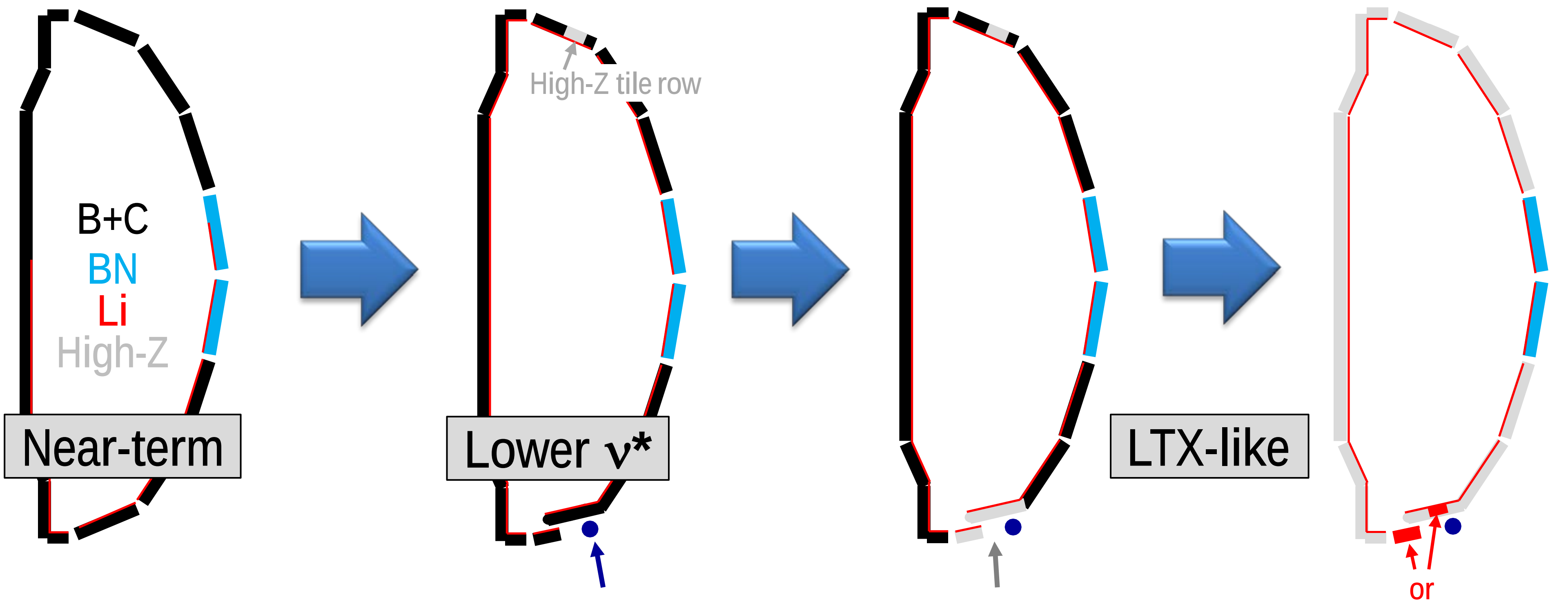


New 4D particle tracking demonstrated using an exploding-wire high-temperature microparticle source

NSTX-U long-term goals

- **5 year: Integrate high confinement + β_T + full non-inductive**
- **10 year: Assess compatibility with high-Z & liquid Li PFCs**

Possible progression of in-vessel / plasma-facing components:



C walls + Li coatings on bottom ½ of wall

Add divertor cryo-pump + full-wall Li coatings

Heatable C → high-Z for liquid Li divertor

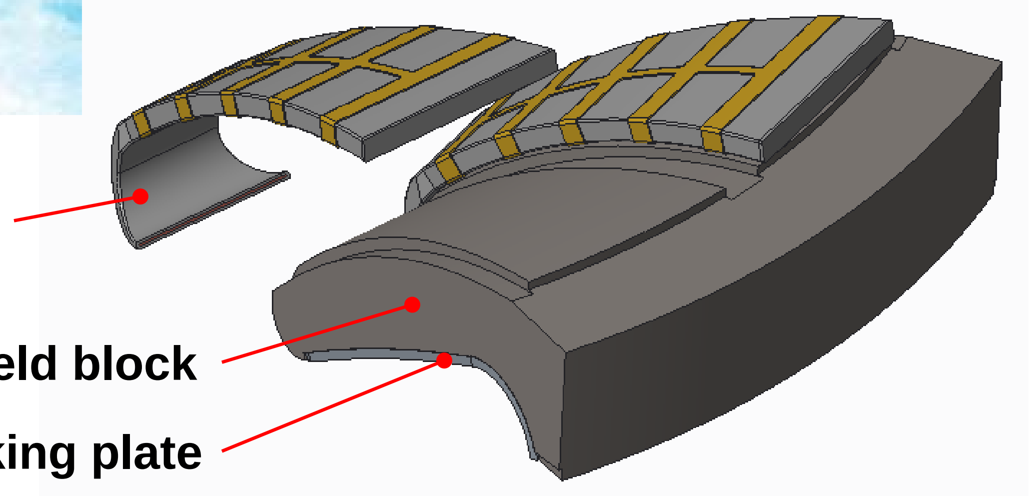
Flowing LM module? + heatable (?) high-Z walls

Why (flowing) liquid metals in divertor?

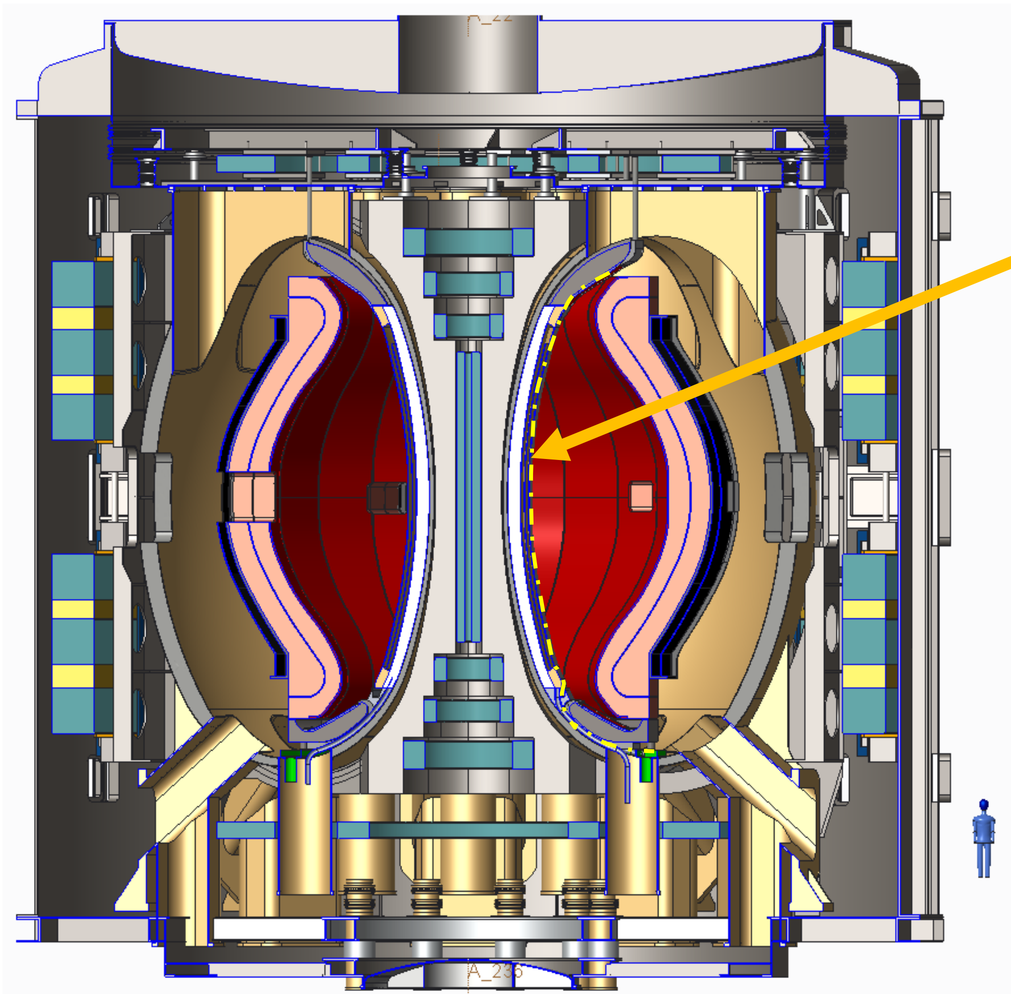
- Replenish divertor material lost to steady-state & transient erosion (ELMs, control excursions)
  - Potentially more resilient to disruptions (?)
- First-wall erosion materials will very likely end up in divertor → need active dust removal
- Dissipate higher heat fluxes than is possible with solids → widens divertor design space
- Lithium H/D/T pumping may increase energy confinement → important for compact Pilots
  - ELMs suppressed by Li in NSTX, EAST

Pilot Plants: Exploring liquid metal divertor concepts similar to flowing water curtain systems

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



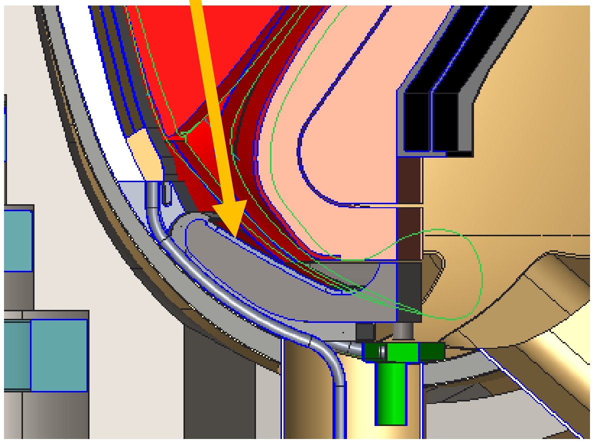
HTS ST-FNSF design with Li flow on divertor and inboard surfaces



Double null liquid metal divertor system

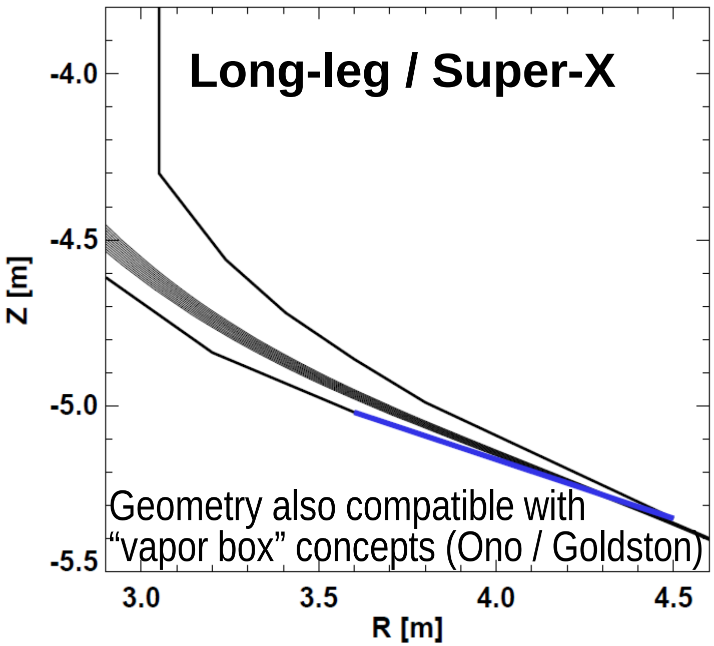
Li flows from upper divertor down the inboard wall, exiting just after the lower inboard divertor.

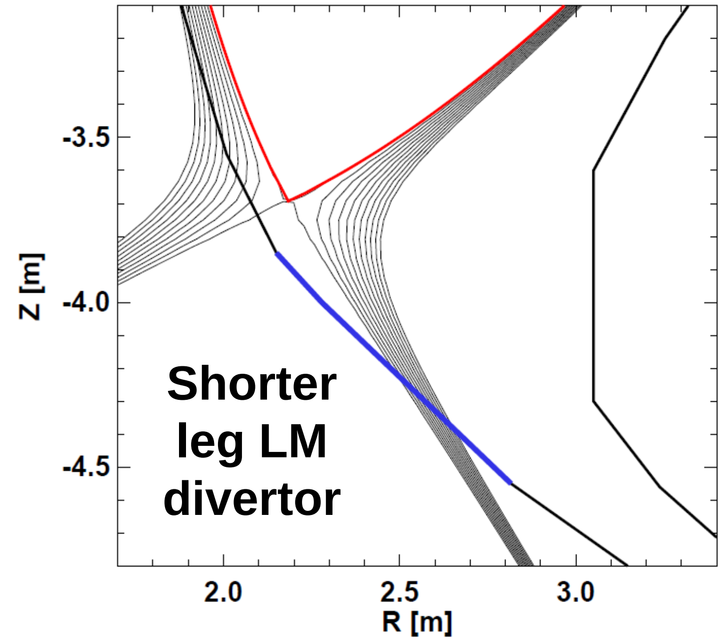
Separate Li cooling of lower divertor

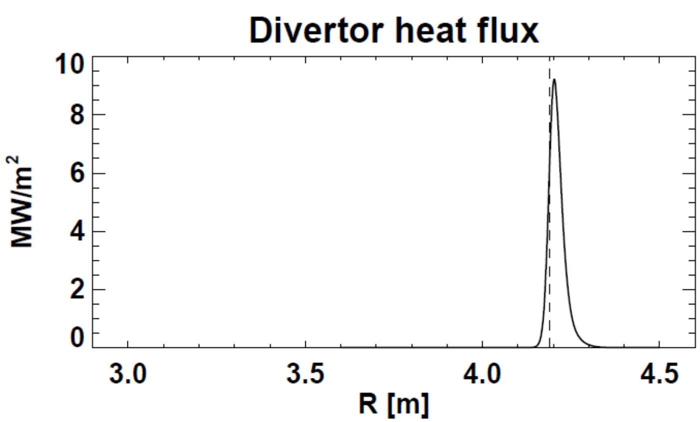


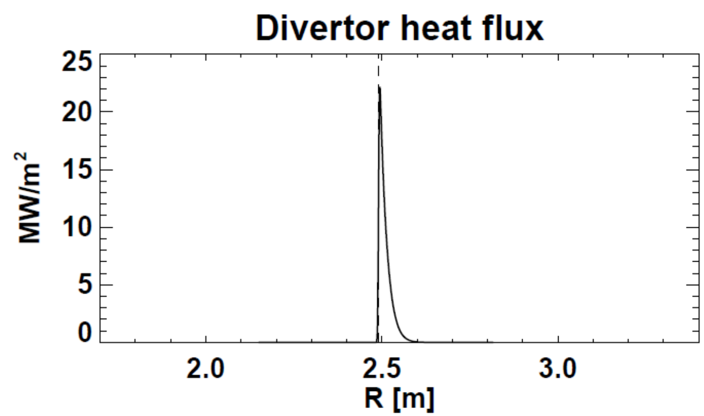
Thickness = 5-10 mm, flow speed ~5-10 m/s - talk to Dick Majeski for more details…

Another option: Li divertor with shorter outer leg
P_{div} = 9 → 21MW/m² for R_{strike} = 4.2m → 2.5m



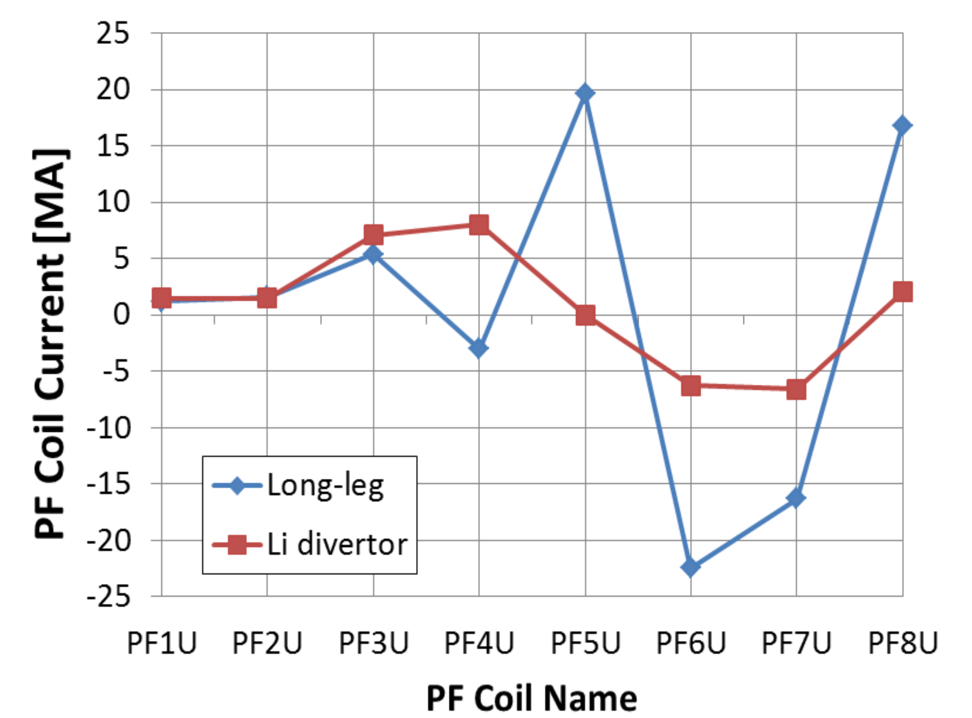


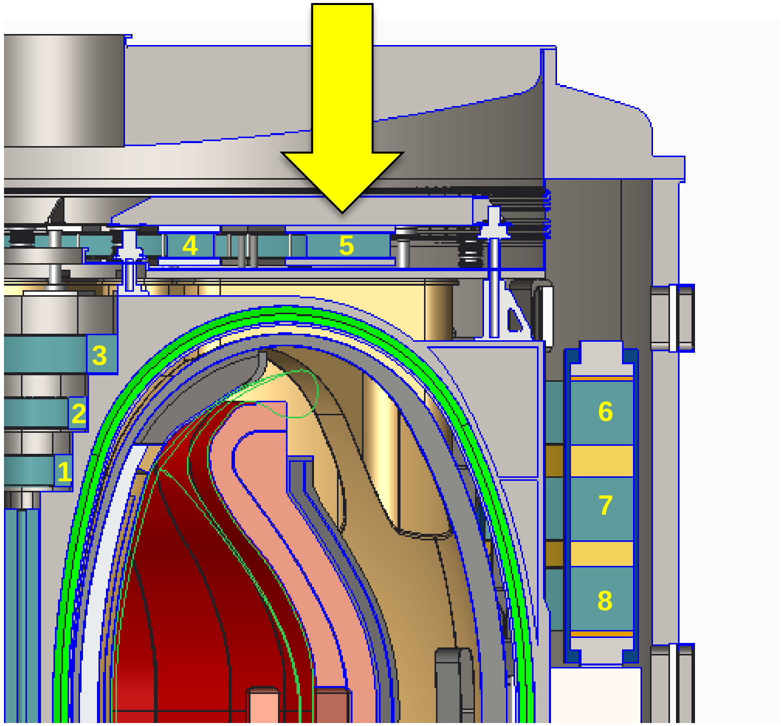




Benefits of shorter-leg LM high-heat-flux divertor:

- 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-factor

Summary

- Advanced superconductors open new possibilities for low aspect ratio tokamaks for fusion applications
- NSTX-U first run campaign scientifically productive
  - Restart in 1-2 years, depending on scope of Recovery effort
- NSTX-U will explore confinement and non-inductive sustainment in novel high-β / low collisionality regime
  - If results are favorable, ST applications could expand
- Longer-term, NSTX-U will address plasma-material-interface challenge by exploring liquid metal solutions

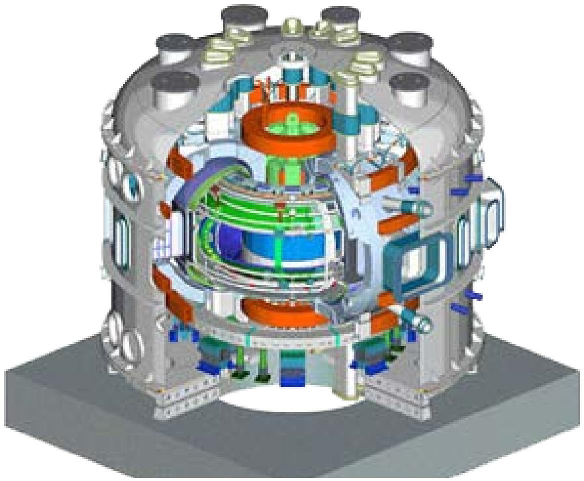
Thank you!

Any questions?

Backup

Tokamaks and stellarators are the leading configurations in magnetic fusion

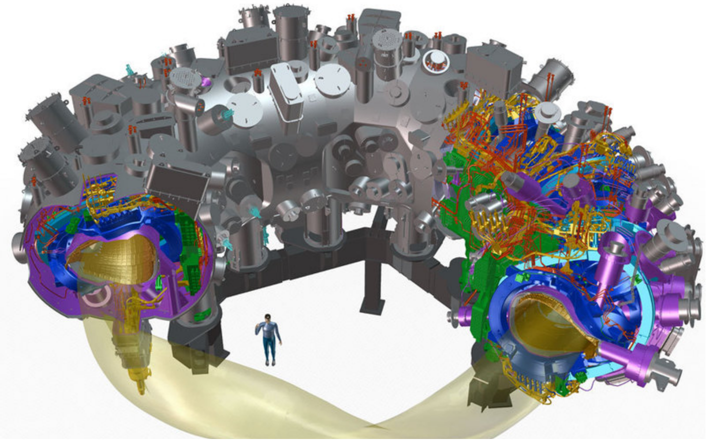
Superconducting tokamak



KSTAR (South Korea)

- Tokamak advantages:
  - Best confinement, closest to "breakeven"
  - Simpler planar coils and power/particle exhaust
- Disadvantages:
  - Must drive multi-mega-ampere plasma current
  - More prone to rapid loss of plasma = "disruption"
    - Potential show-stopper for tokamak power-plant

Superconducting stellarator



W7-X (Germany) – 1st run campaign in 2016

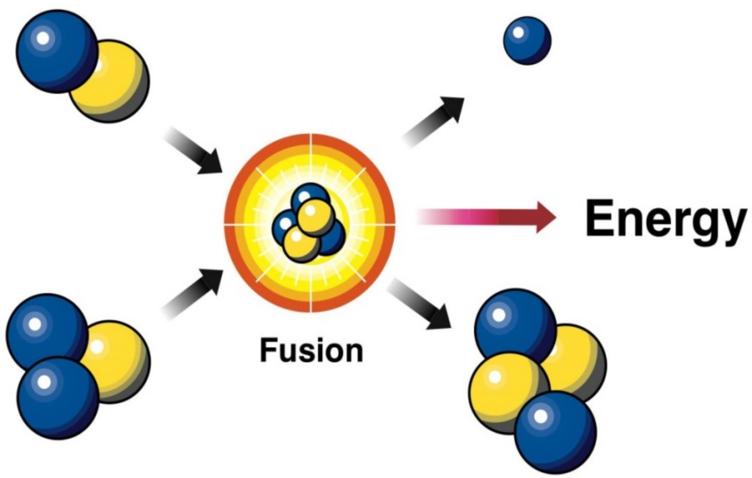
- Stellarator advantages:
  - No plasma current drive necessary
  - More stable, steady-state
- Disadvantages:
  - More complex coils and exhaust
  - Confinement < tokamaks (so far…)

Why fusion?

"D-T" fusion reaction:

D: deuterium

n: neutron



T: tritium

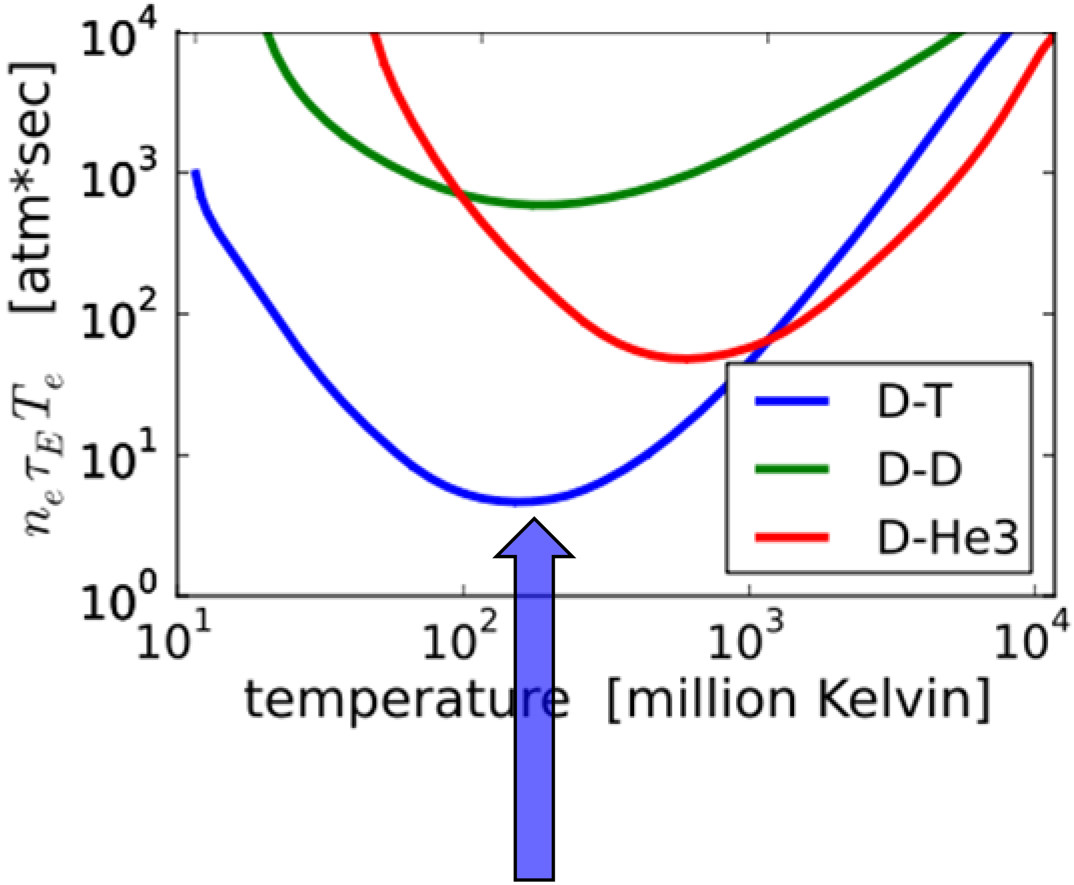
α: helium

$$E = mc^2$$

- High energy gain ≈ **1000 ×**
- No runaway reactions
- Abundant fuel supply
- Waste short-lived, low-level
- No CO_2 production

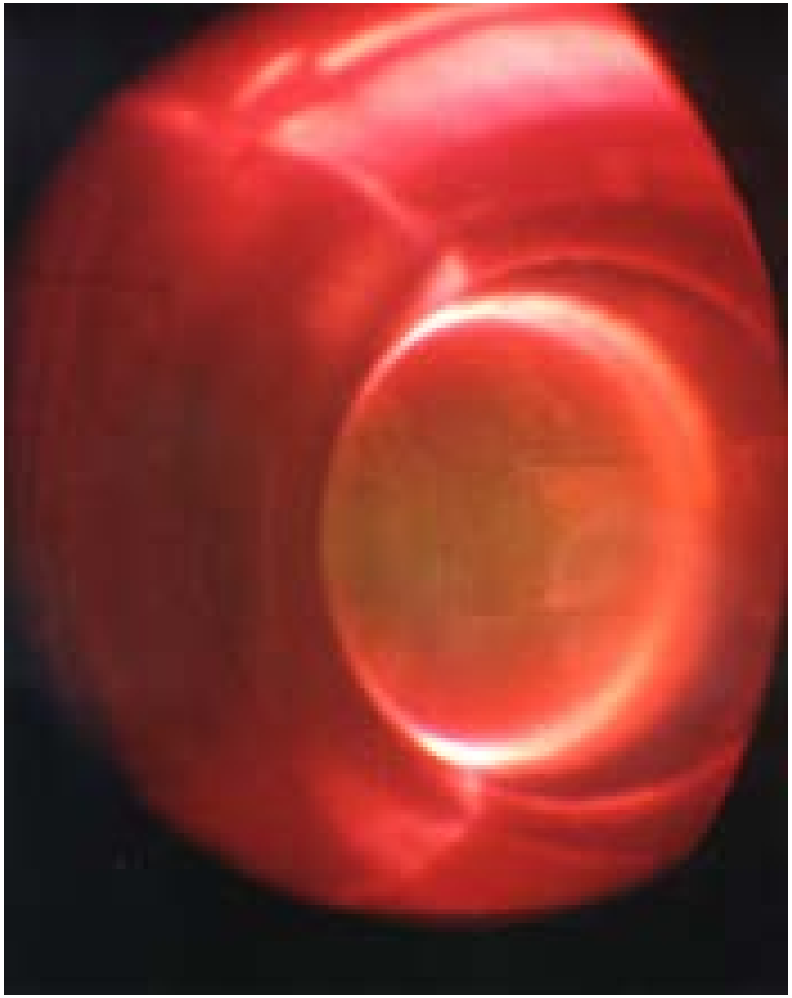
Fusion requires very high temperatures

Fusion difficulty
(pressure × confinement)

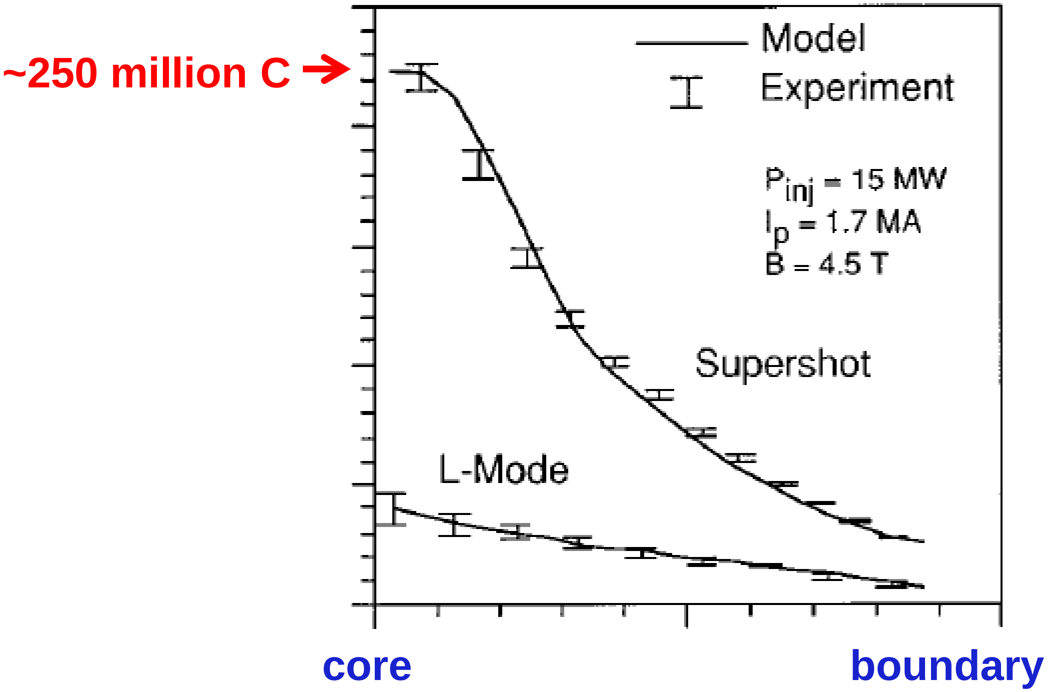


- Fusion is easiest here at 200 million °C (!!) (350 million °F)
  - Requires lowest pressure nT and energy confinement time τ_E
  - Minimum fusion "triple-product" value: 8 atmosphere-seconds

Magnetic fusion has already achieved the necessary very high temperatures!

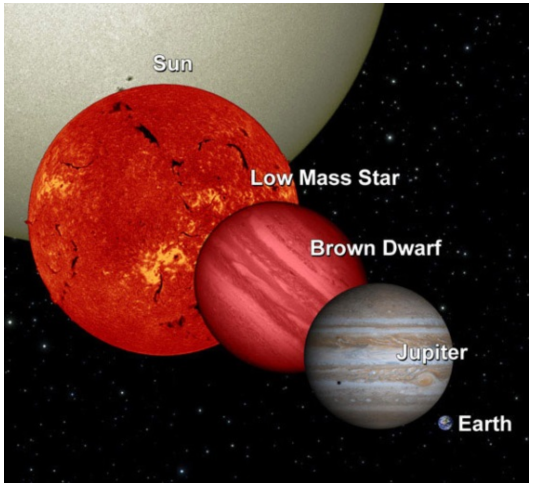


TFTR at PPPL (1990's)



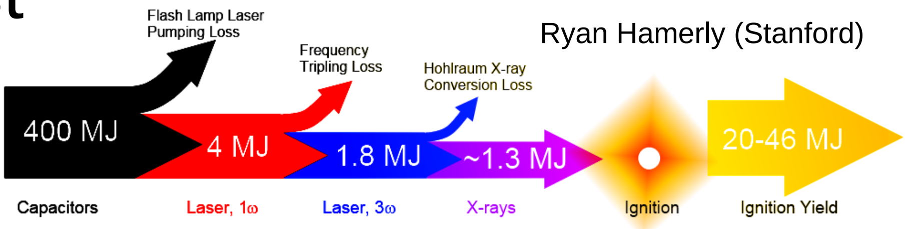
Magnetic fusion is arguably closest to ultimate goal of electricity generation

- Gravitational confinement fusion requires large device
  - Need 7-8% of mass of our sun
  - Approximately 10× diameter of Earth



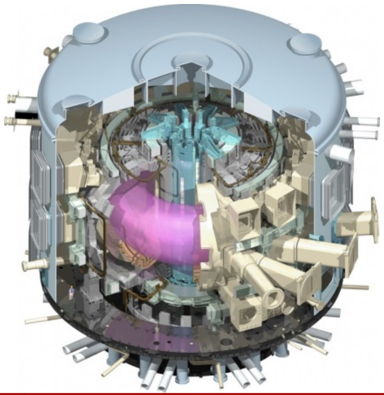
- Laser fusion ala NIF at best has $E_{fusion} / E_{electrical} \sim 5\%$
  - So far, 0.006% efficient

Ryan Hamerly (Stanford)

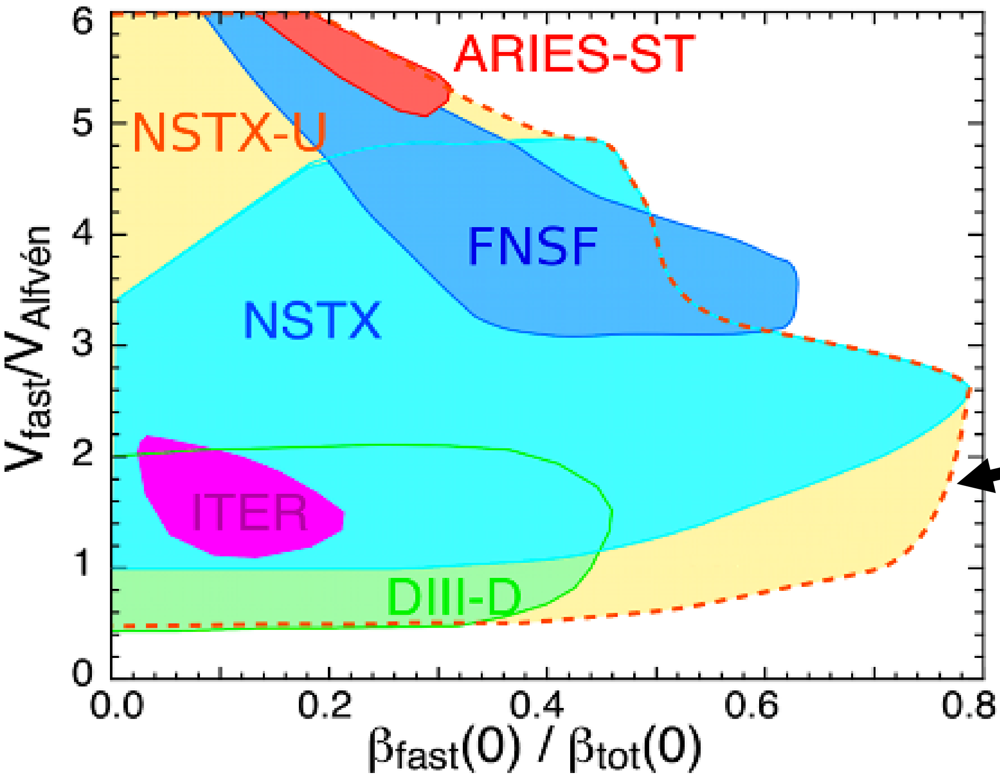


~25kJ fusion yield achieved

- Magnetic fusion in ITER:
  - Goal: 500MW fusion power for ≤ 600MW electrical input for 400s
  - ➢ Industrial levels of fusion power



NBI-heated STs excellent testbed for α-particle physics

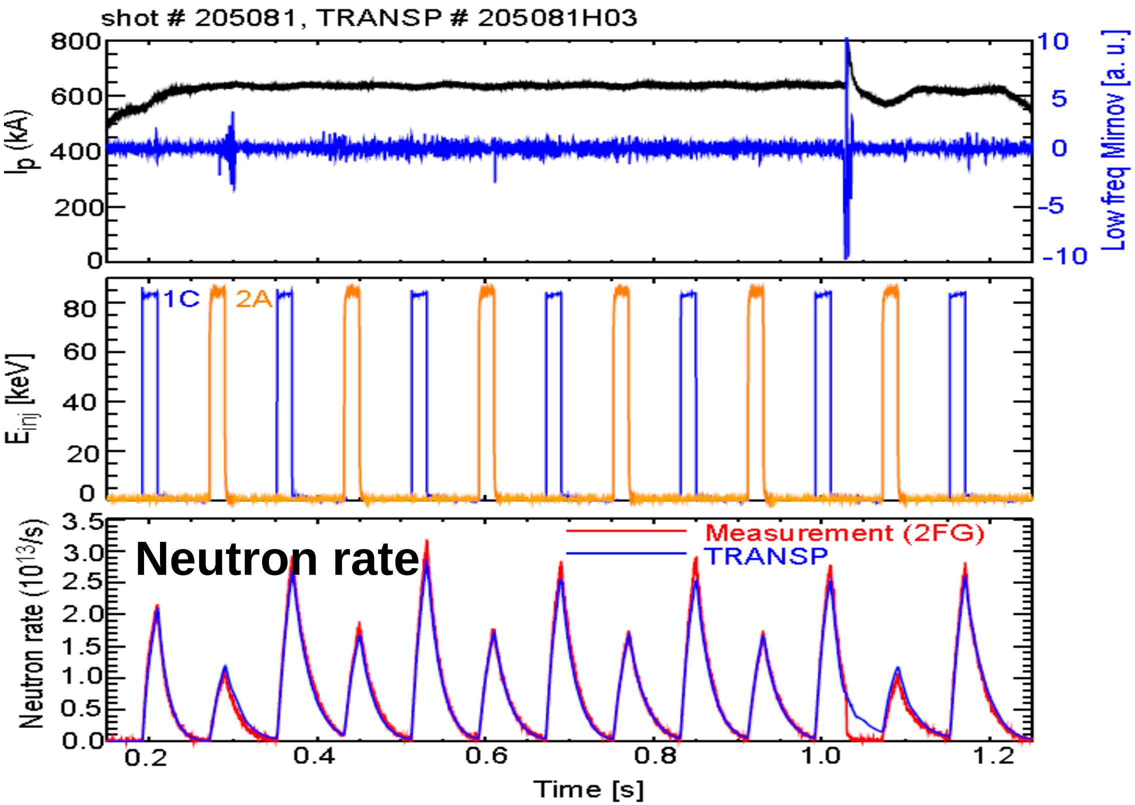


- NSTX-U: large fast-ion dynamic range spanning ST and conventional A
  - **Toroidal field 2× NSTX →** $V_{fast} < V_A$ → stabilize modes
  - **Tangential 2nd NBI →** very flexible fast-ion distribution
    - Vary pitch angle, pressure profile

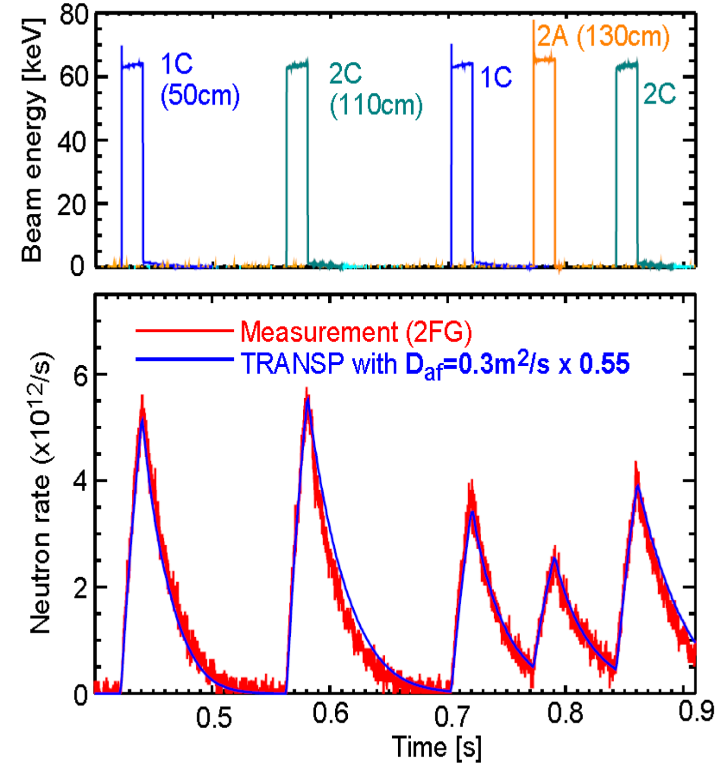
- **Can we find TAE-quiescent, high-performance regimes in NSTX-U?**
- **And predict fast-ion confinement for ITER scenarios?**

Fast-ion confinement measured to be at / near predicted values at low total NBI power ~1-2MW

E_{NBI} = 85keV



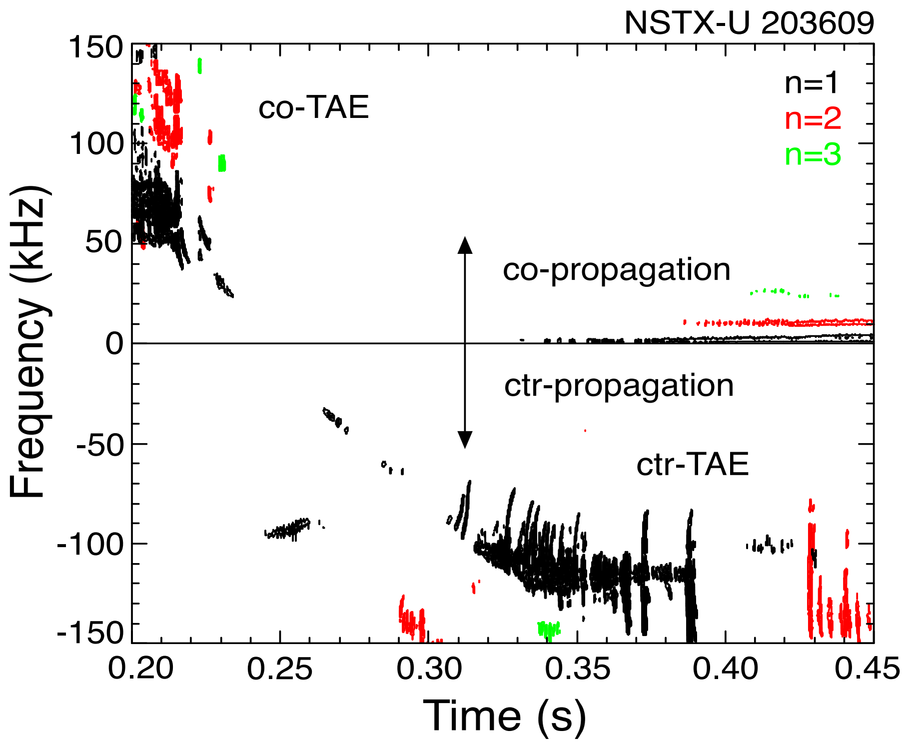
E_{NBI} = 65keV



- Good agreement between neutron measurement and TRANSP prediction
- Need small anomalous fast ion diffusivity (D_{af}=0.3m^2/s) for agreement

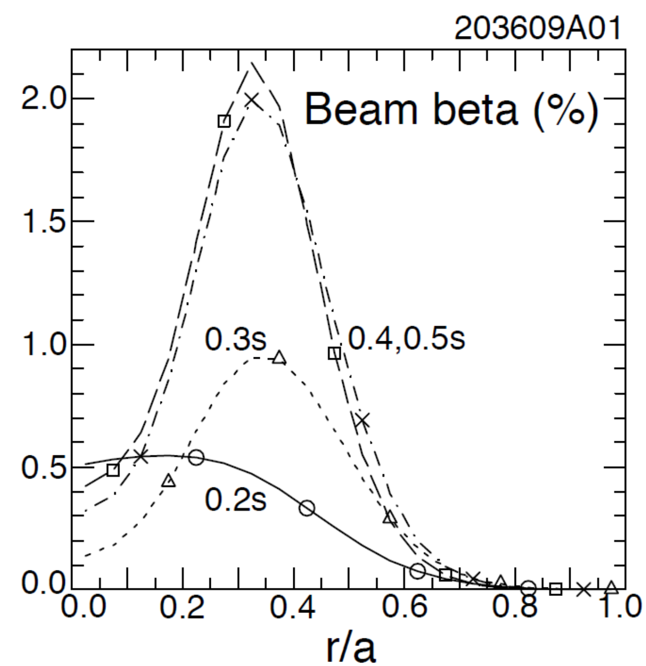
UCIrvine

New: Most tangential NBI generates counter-propagating Toroidal Alfvén Eigenmodes (TAEs)



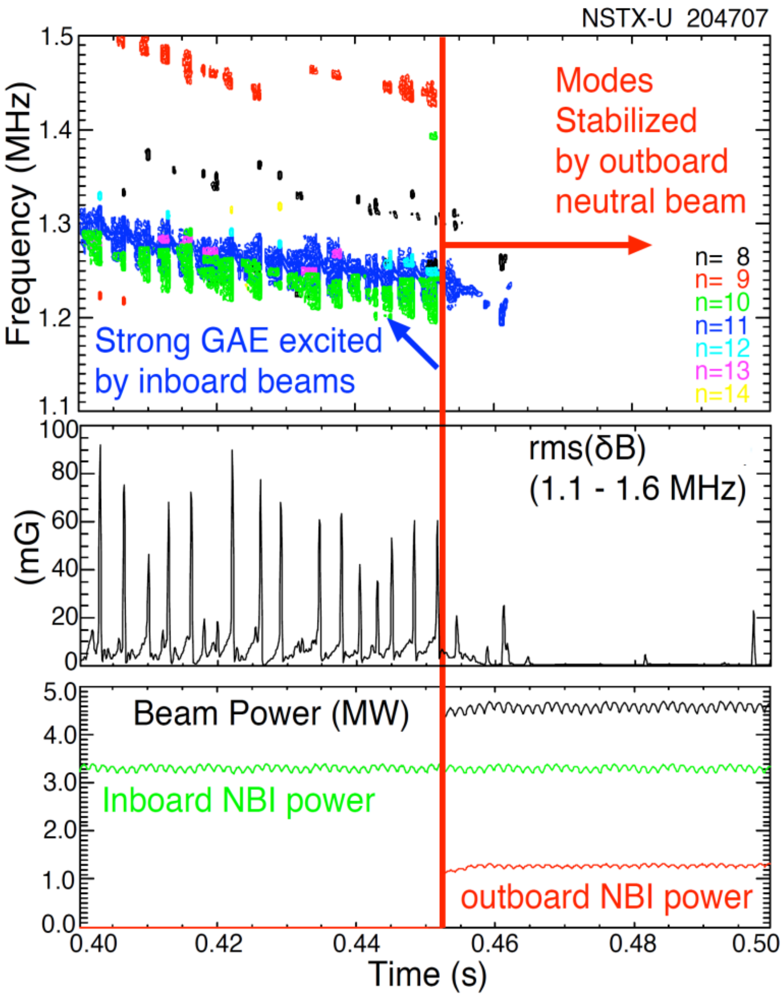
- Counter-propagating TAE predicted for **hollow** fast-ion profiles

H.V. Wong, H. Berk, Phys. Lett. A ***251*** *(1999) 126.*

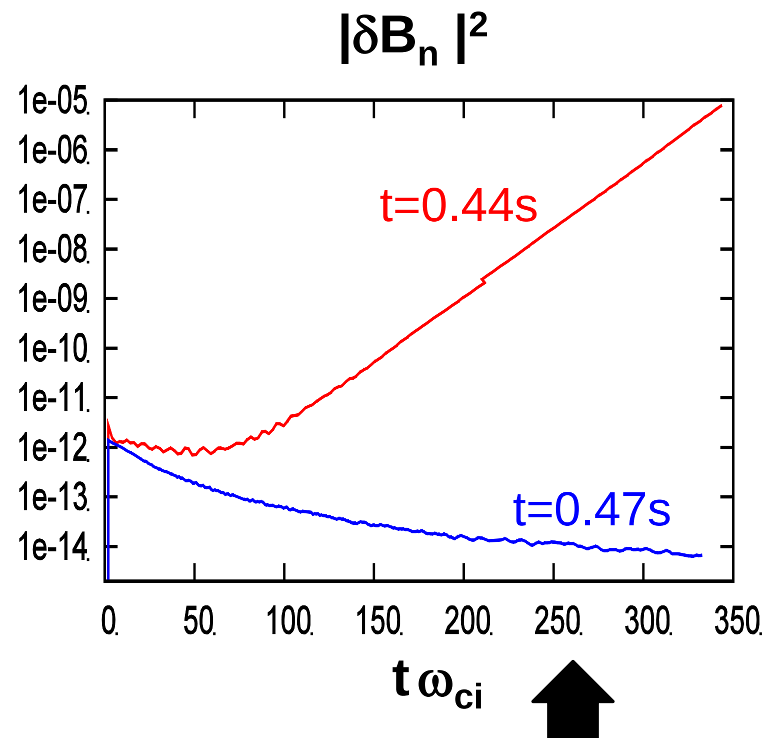


- TRANSP: As current builds up beam fast-ion beta profile predicted to become hollow
- **1st evidence of off-axis NBI in NSTX-U**

New: Tangential 2nd neutral beam suppresses Global Alfven Eigenmode (GAE) – consistent with simulation



HYM code simulation of #204707, n=10



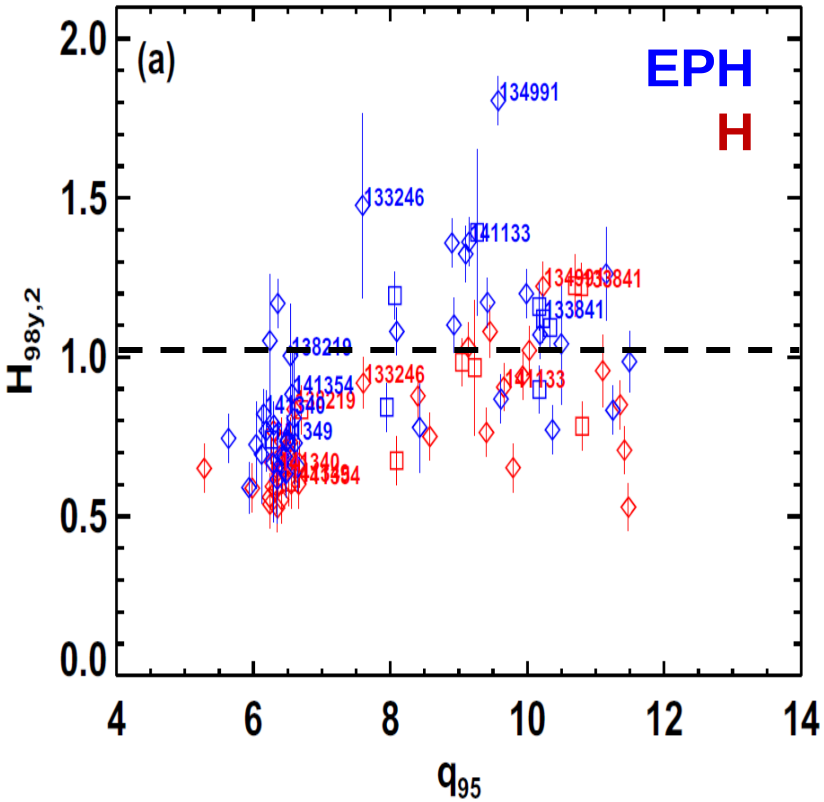
- HYM code: growth of n=10 counter-GAE from 1st NBI
- HYM: suppression of n=10 counter-GAE by 2nd NBI
- Most unstable *n*-number, mode ω consistent with HYM

New 2nd NBI already powerful tool for fast-ion mode physics

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



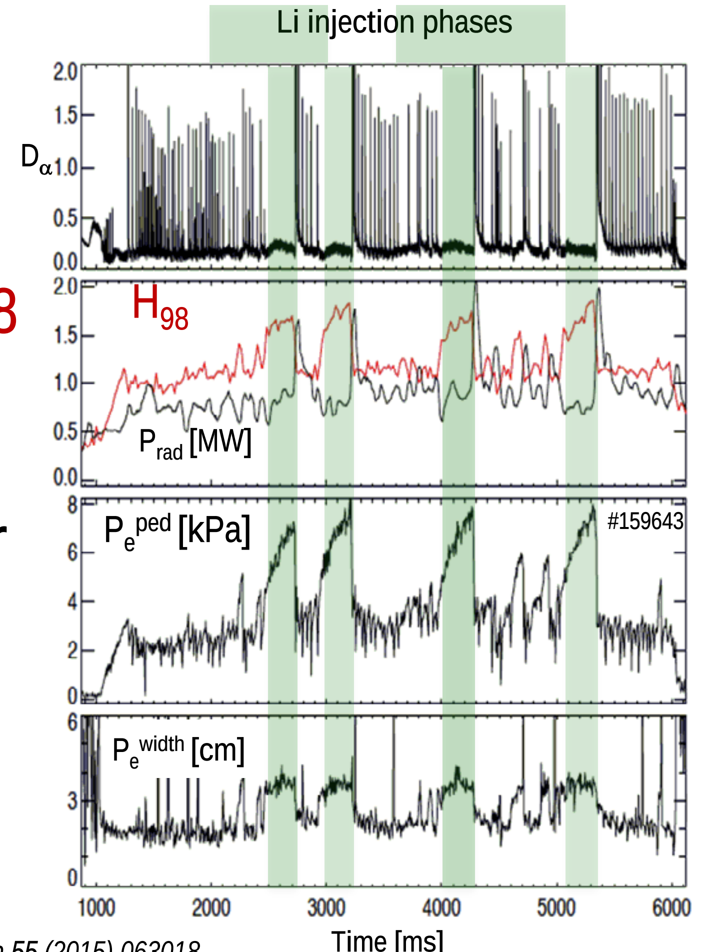
S. Gerhardt, et al., Nucl. Fusion 54 (2014) 083021

Lithium injection on DIII-D

H_{98}→1.5-1.8

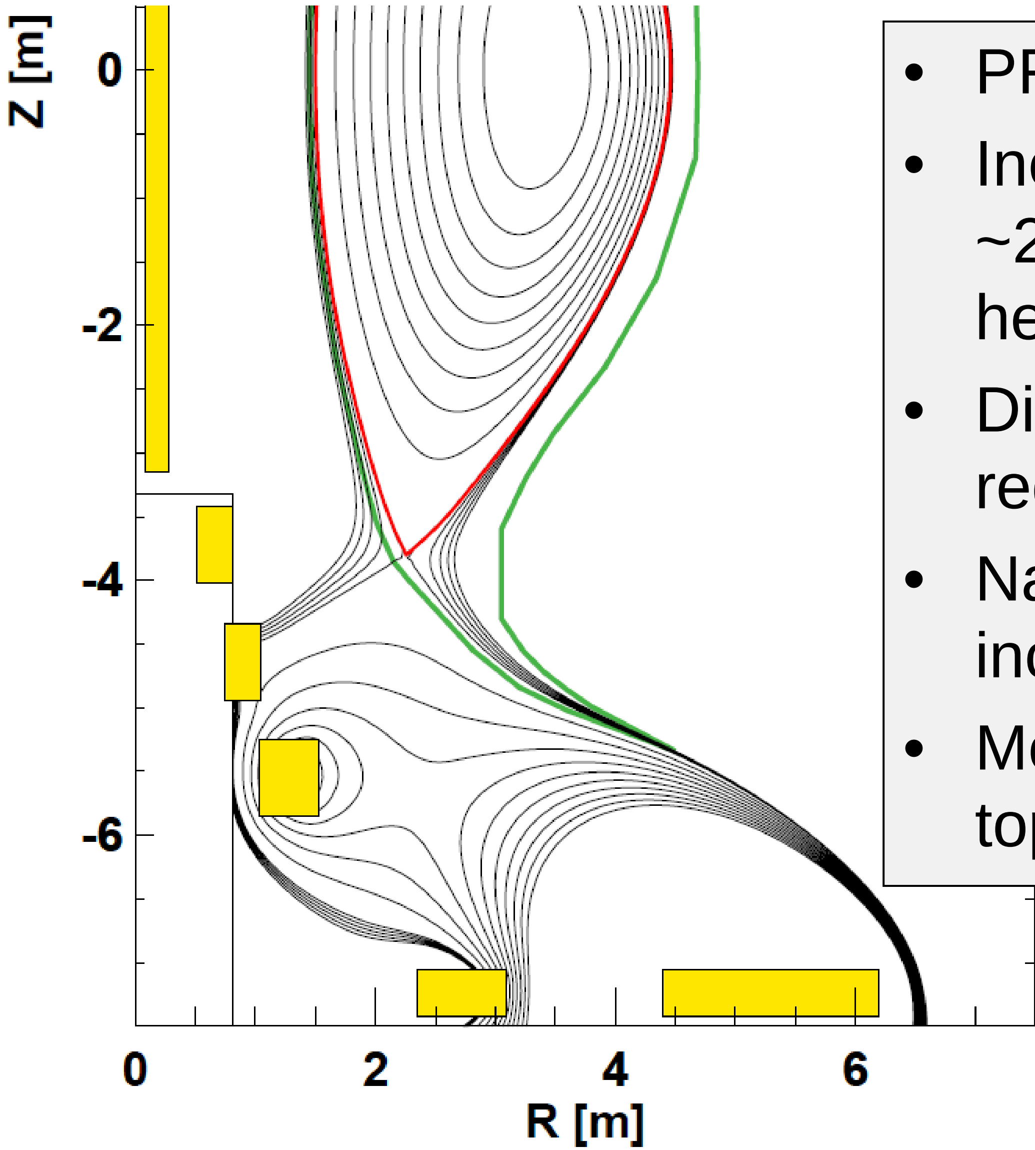
2× wider, 2-3× higher pedestal

(BCM: Bursty chirping mode)



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

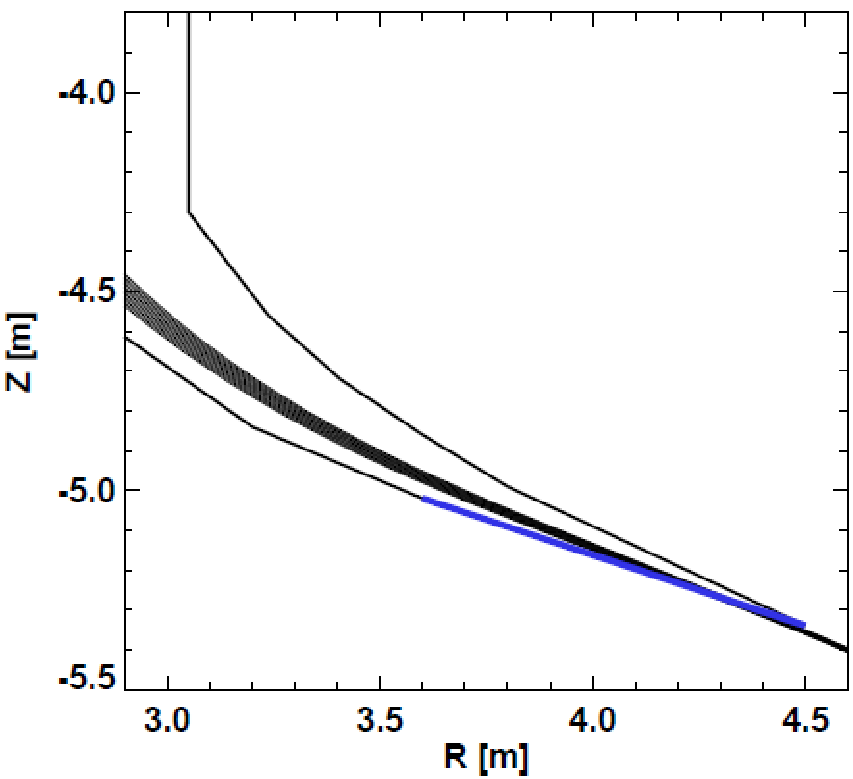
Base-design Pilot uses long-leg / deep-V slot divertor



- 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

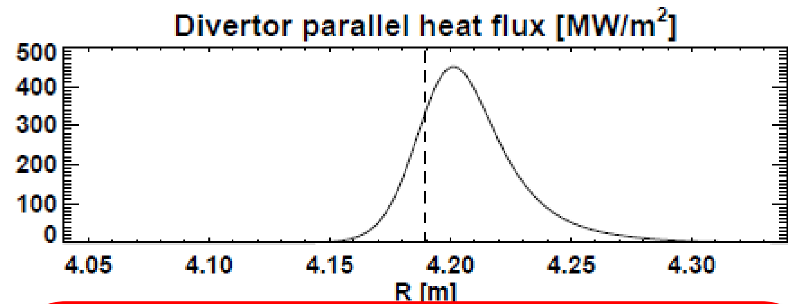
Long-leg / Super-X aids heat flux reduction

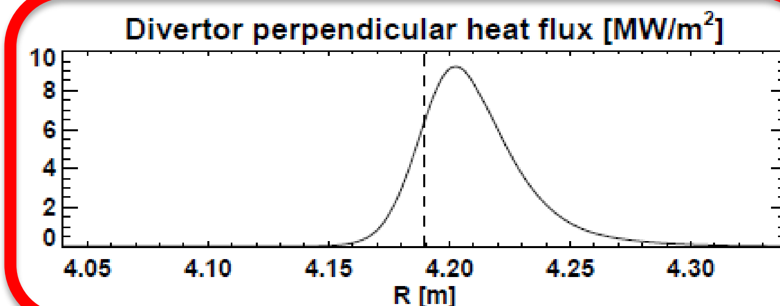
A=2 HTS TF FNSF/Pilot

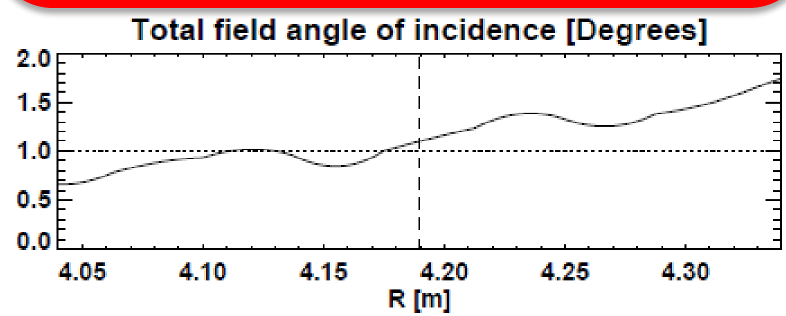


$\lambda_q \sim$ 1mm, **assume S $\approx \lambda_q$** (closed divertor)

(T. Eich NF 2013)



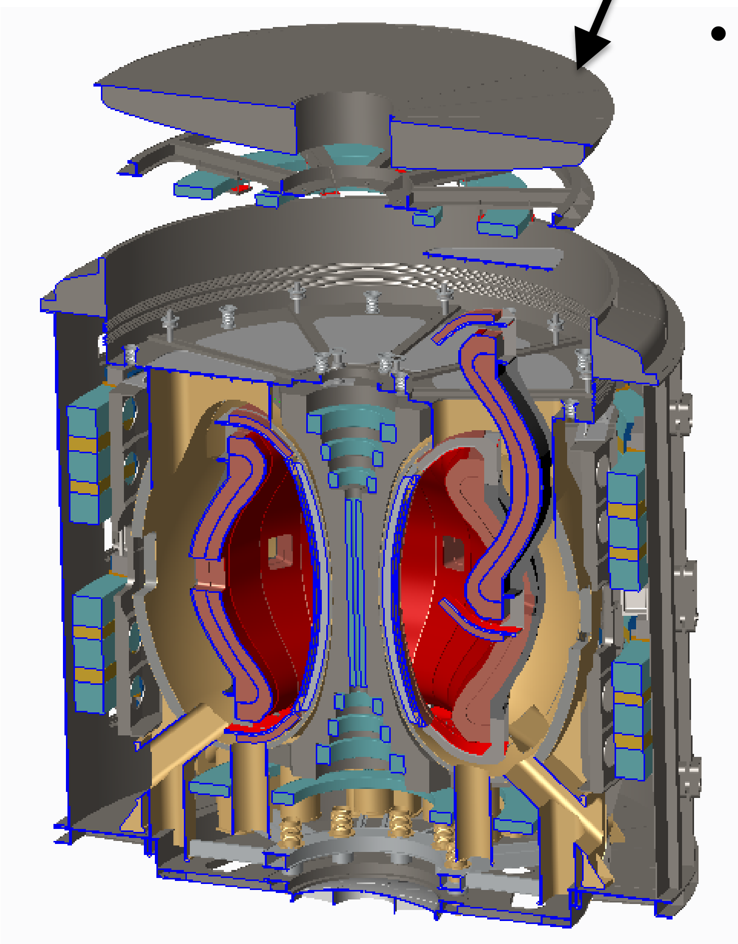




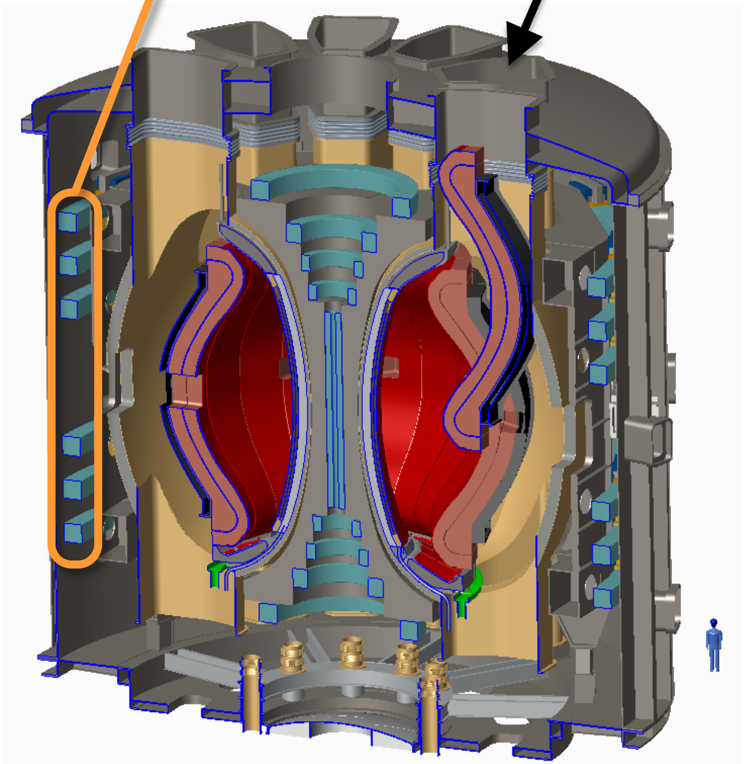
(Partial) detachment likely reduces peak $q_\perp$ by further factor of 2-7

Comparison of long vs. shorter-leg divertor Pilots:

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

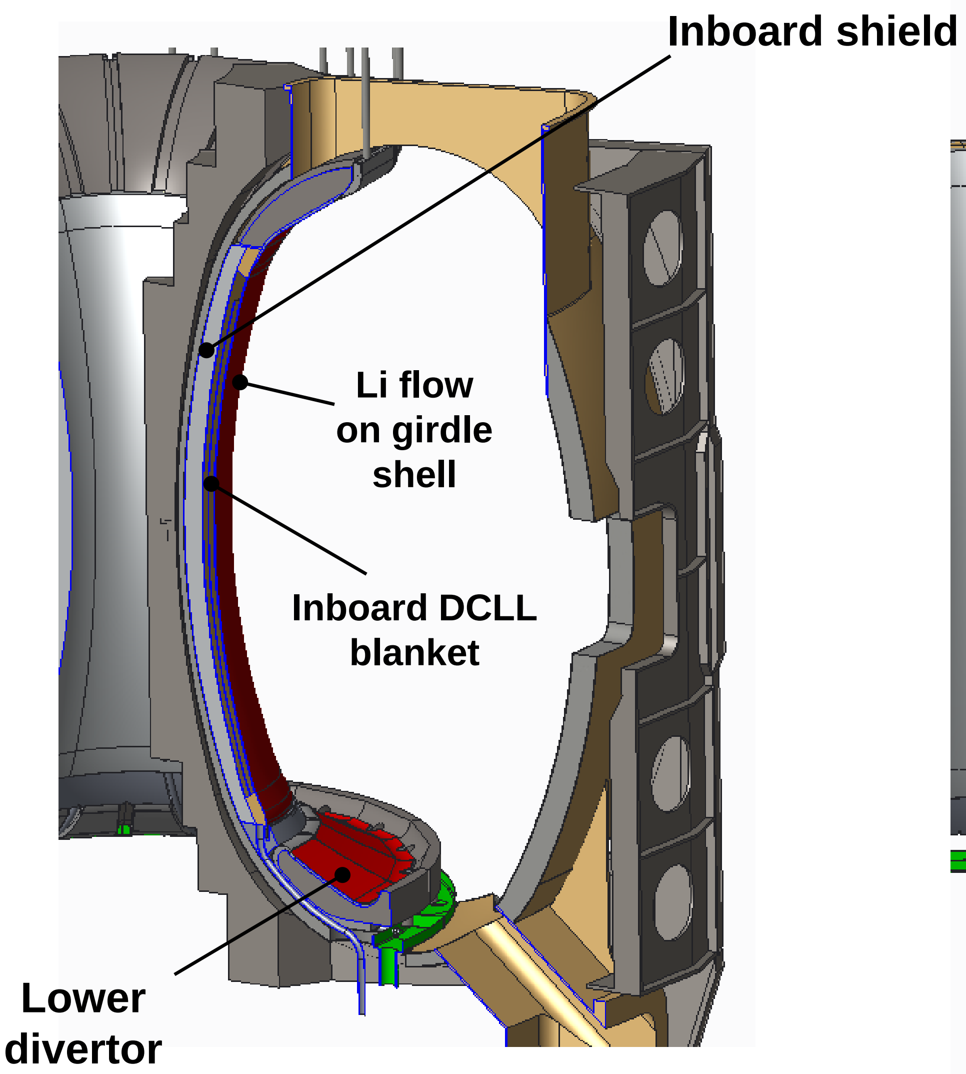


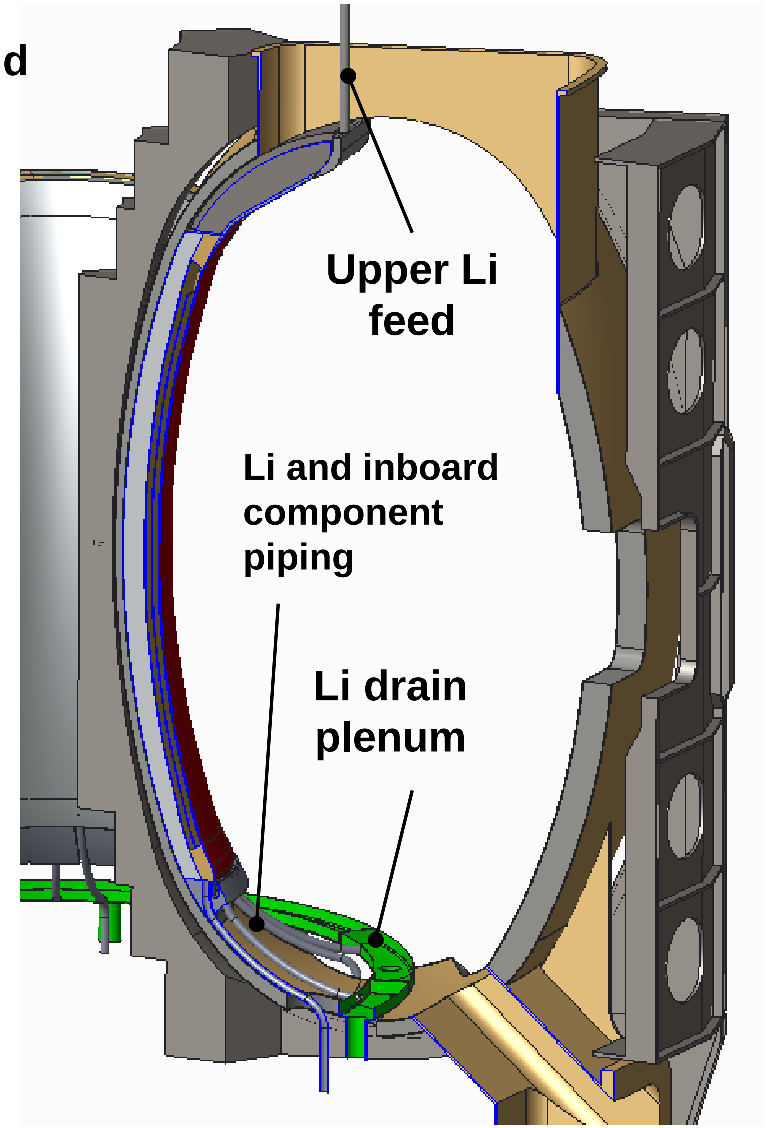
3m HTS ST-FNSF with long-leg/Super-X divertor



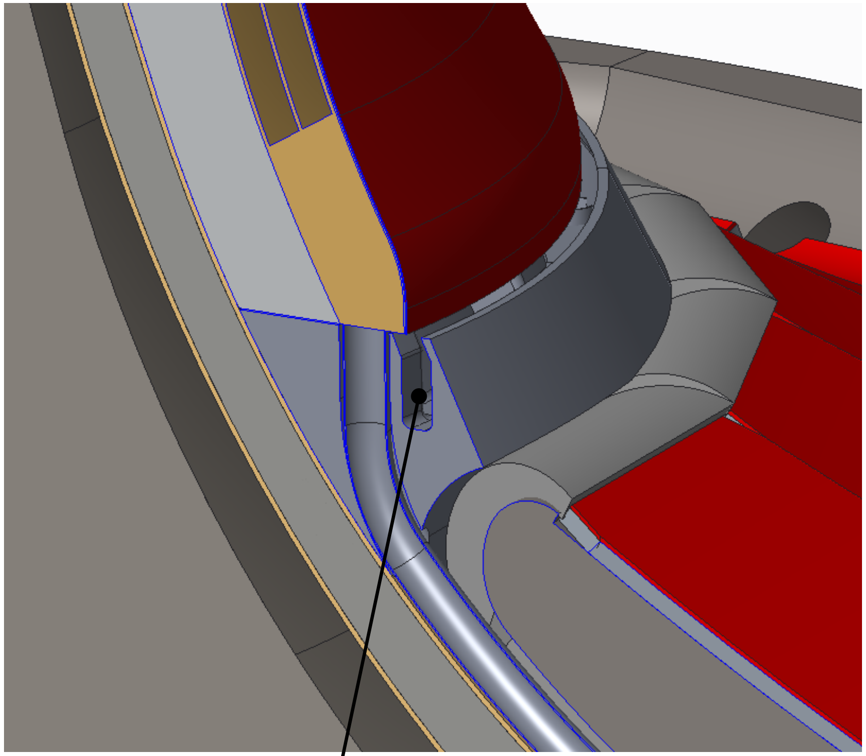
3m HTS ST-FNSF with Liquid Metal divertor

Local details of Li divertor / inboard FW



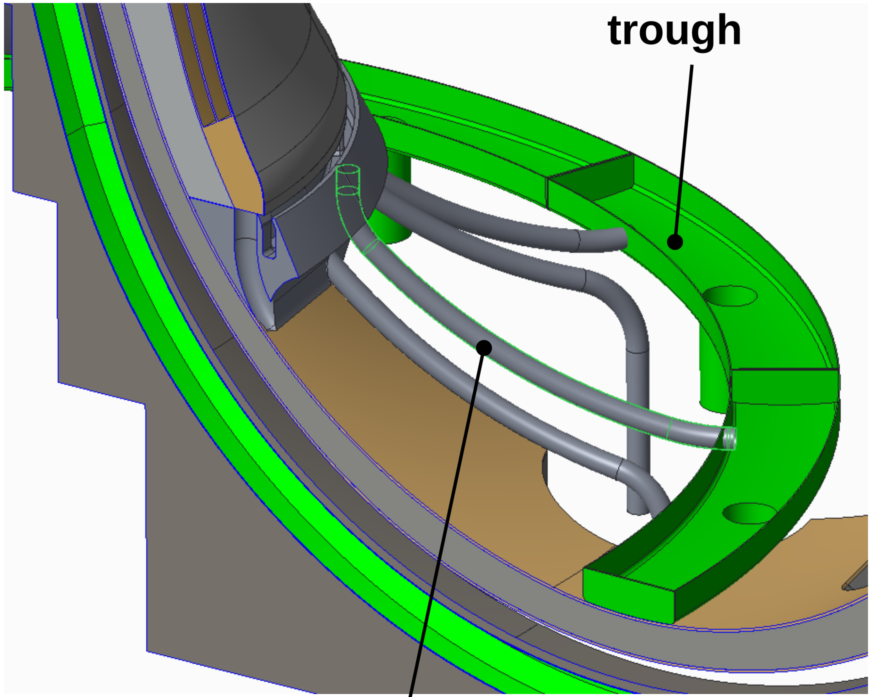


Lower Li containment system



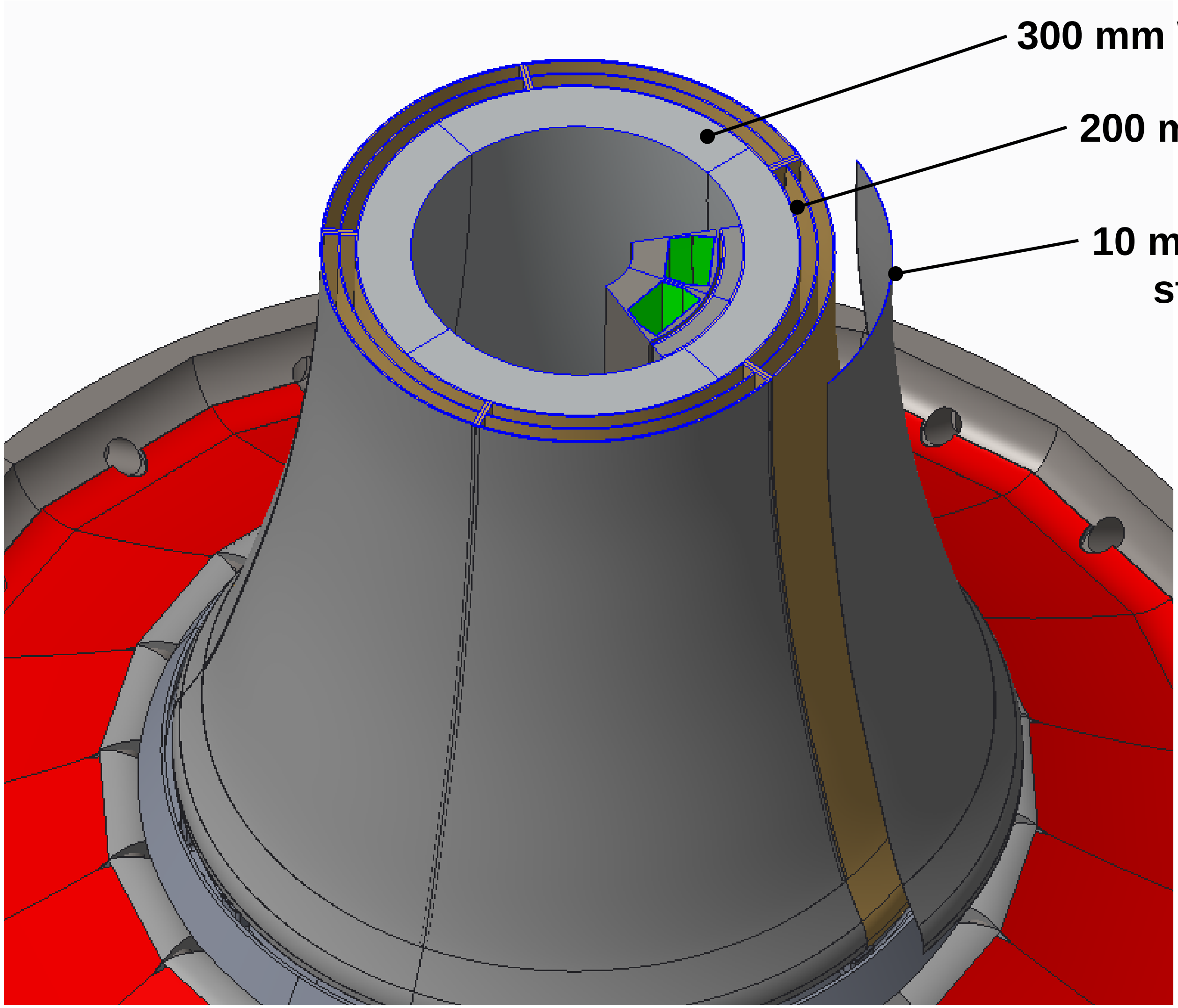
Li flows over inboard surface to a continuous trough that feeds ten Li drain lines.

Base Li return trough



One of ten 100 mm ID Li inboard drain lines

Inboard FW / DCLL / shield components



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