Motivations for Spherical Torus research and initial results from NSTX Upgrade

Jonathan Menard (PPPL)
Thanks to many contributions from
ST community and NSTX-U Researchers

PPPL – MBG Auditorium

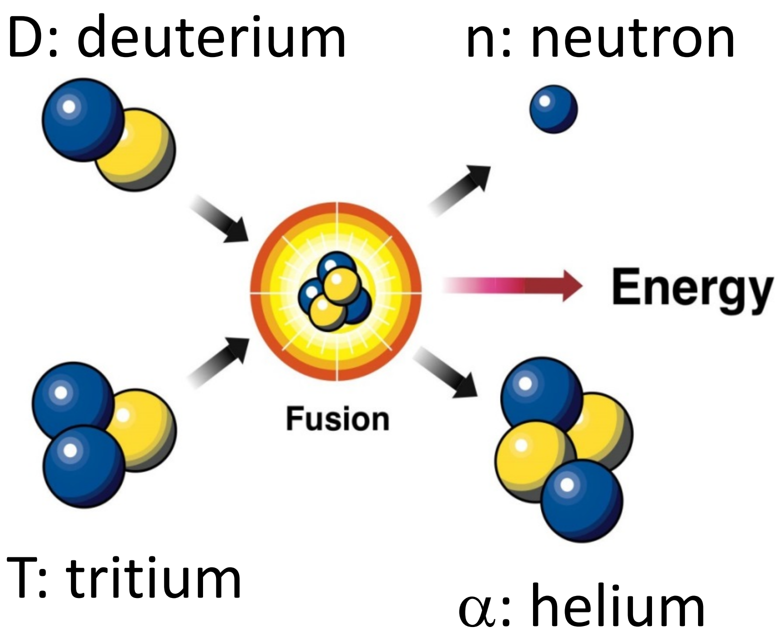
January 11, 2017

Outline

- Introduction
- Why spherical tori / tokamaks (STs)
- NSTX-U initial results, long-term directions
- Summary

Why fusion?

"D-T" fusion reaction:

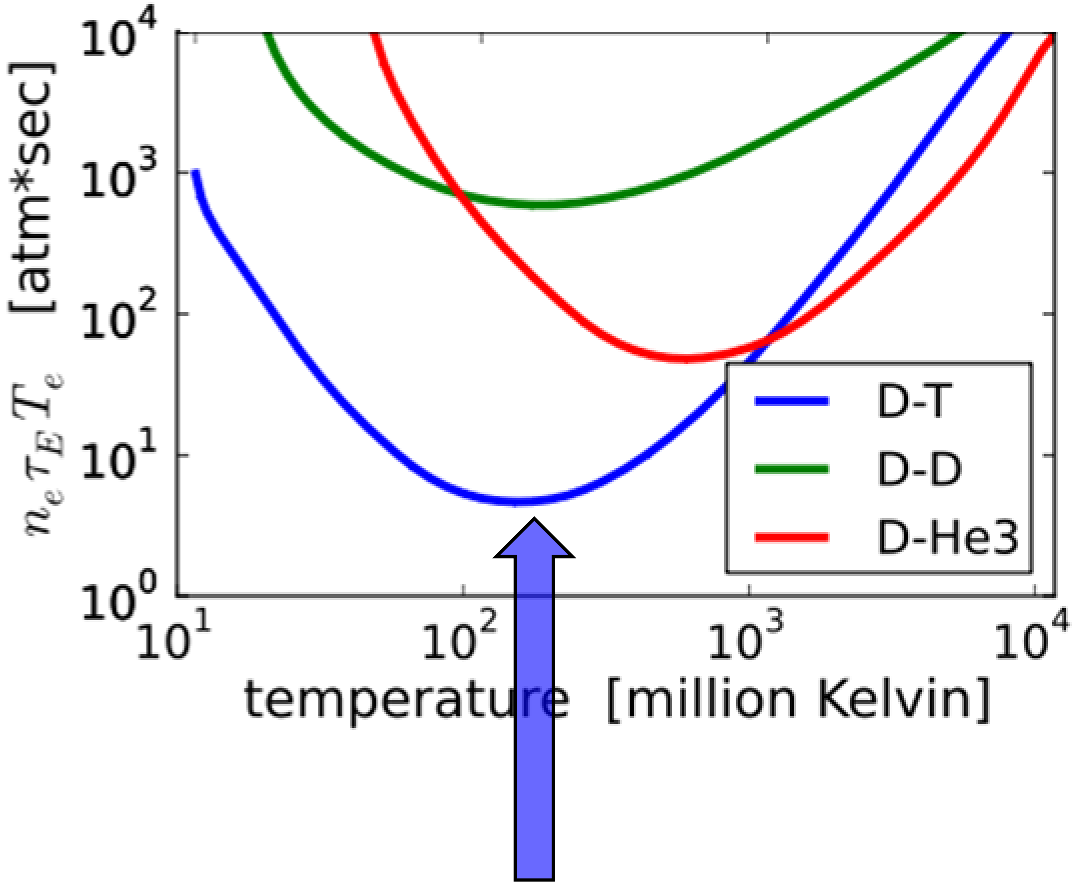


$$E = mc^2$$

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

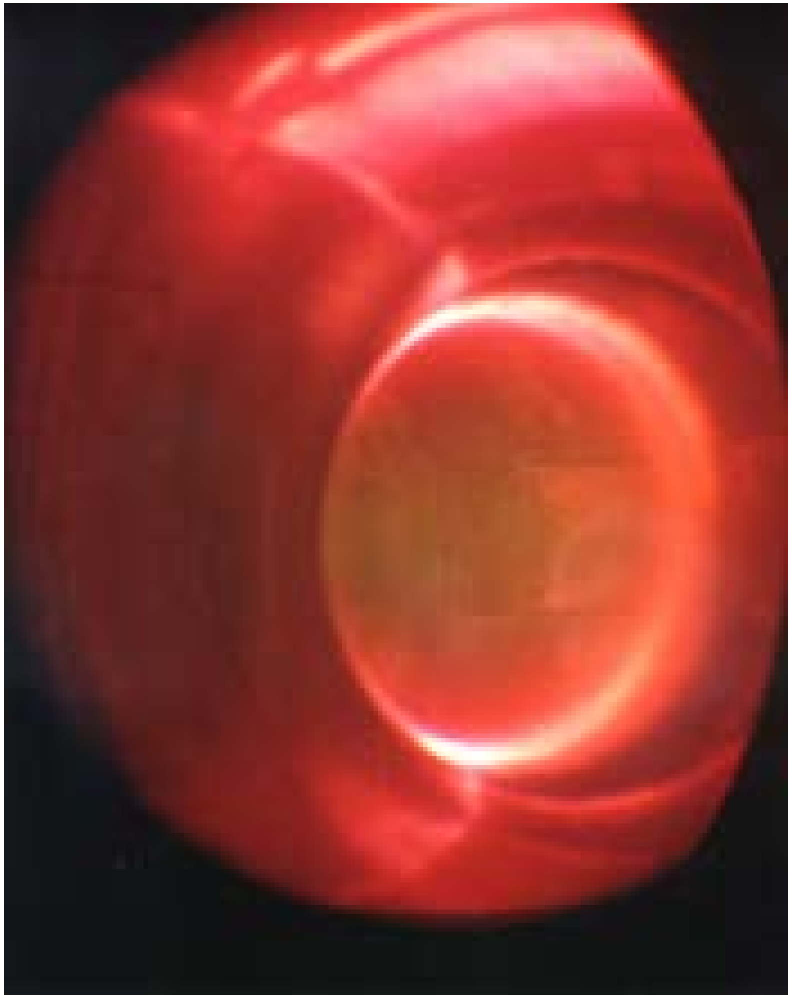
Fusion requires very high temperatures

Fusion difficulty
(pressure × confinement)



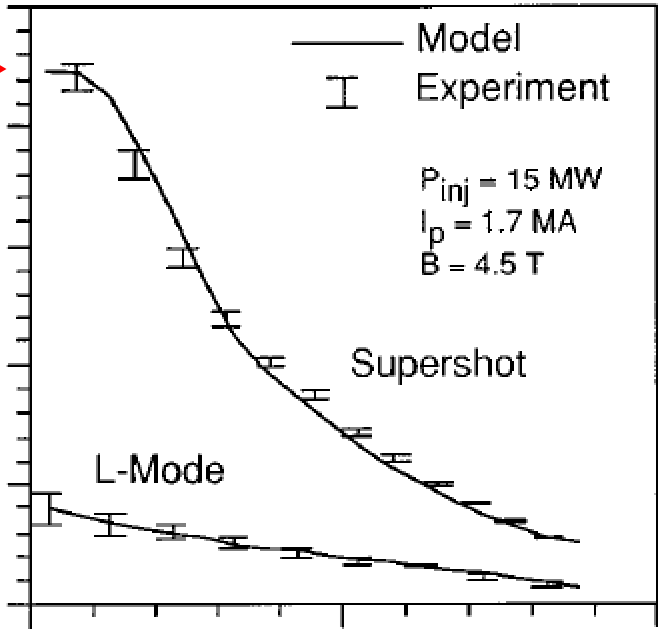
- Fusion is easiest here at 200 million °C (!!) (350 million °F)
  - Requires lowest pressure nT and energy confinement time τ_E
  - Minimum fusion "triple-product" value: 8 atmosphere-seconds

Magnetic fusion has already achieved the necessary very high temperatures!



TFTR at PPPL (1990's)

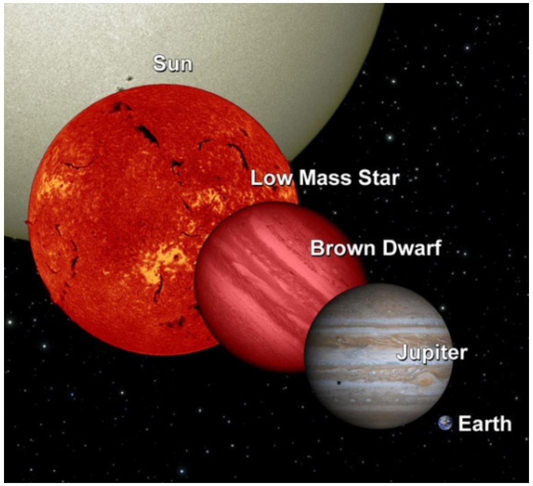
~250 million C →



core boundary

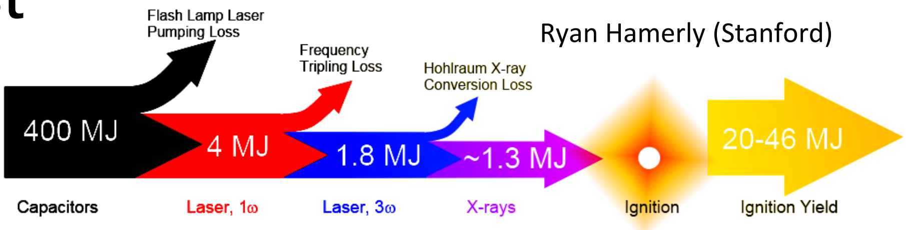
Magnetic fusion is arguably closest to ultimate goal of electricity generation

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



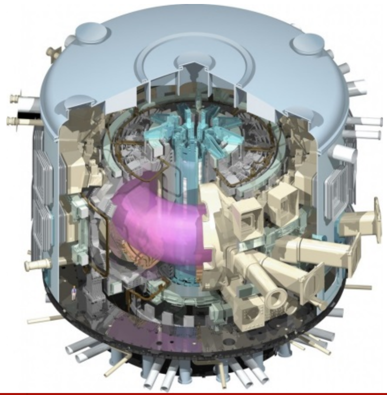
- Laser fusion ala NIF at best has $E_{fusion} / E_{electrical} \sim 5\%$
  - So far, 0.006% efficient

Ryan Hamerly (Stanford)



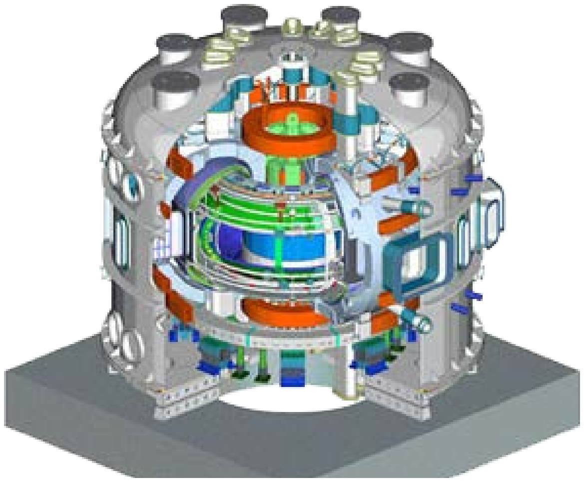
~25kJ fusion yield achieved

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



Tokamaks and stellarators are the leading configurations in magnetic fusion

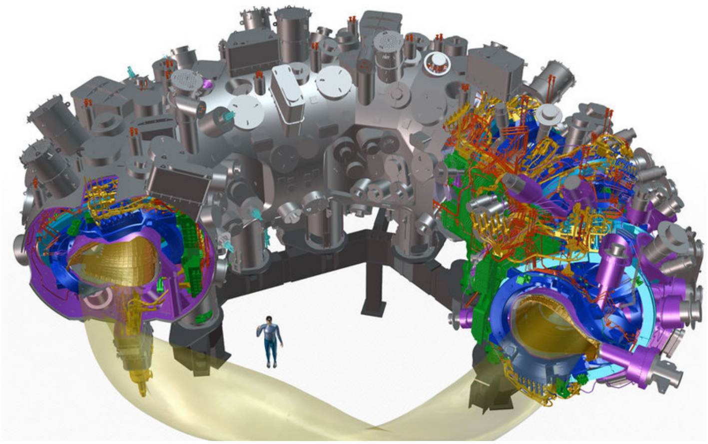
Superconducting tokamak



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



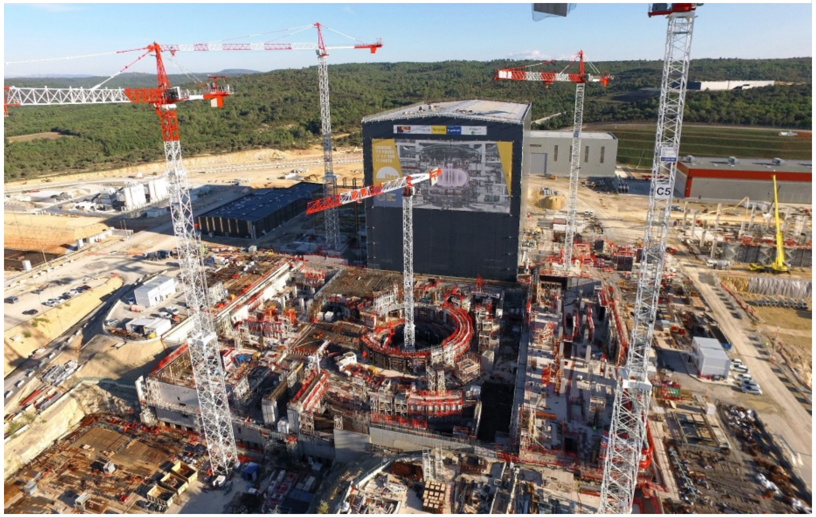
W7-X (Germany) – 1st run campaign in 2016

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

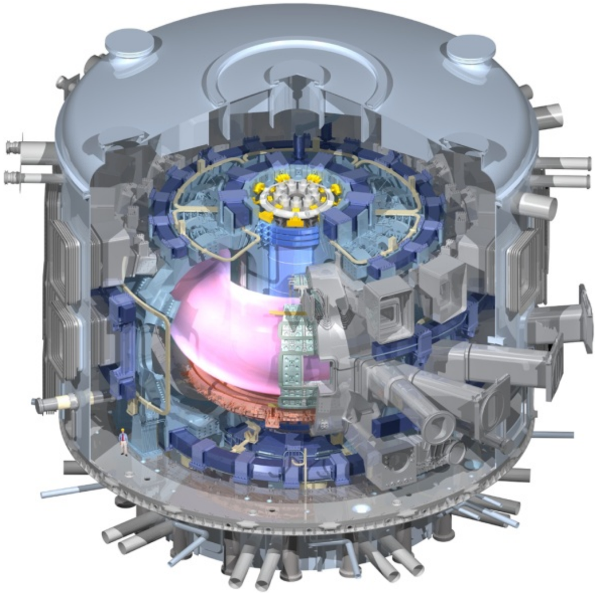
ITER will be first device to access “burning plasma”

- 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 $Q = P_{fusion} / P_{external\ heating} = 10$
- $Q = 10 \rightarrow P_{alpha} / P_{external} = 2$
- $P_{alpha} / P_{alpha\ +\ external} = 2 / 3 > 50\%$

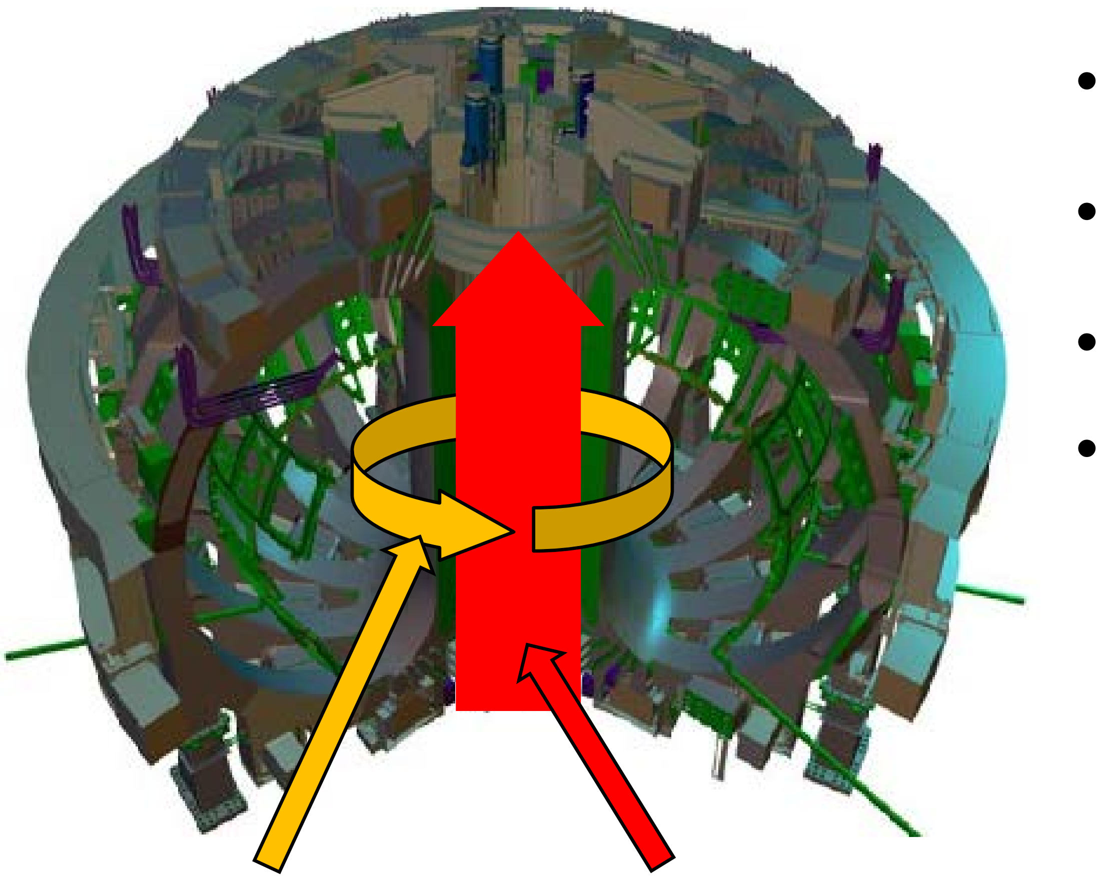
ITER under construction in Cadarache, France



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



ITER magnets will be largest ever built



- 18 toroidal field magnets
- 12 Tesla at coil
- Weight: 6500 tons
- 80,000 km of Nb3Sn 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

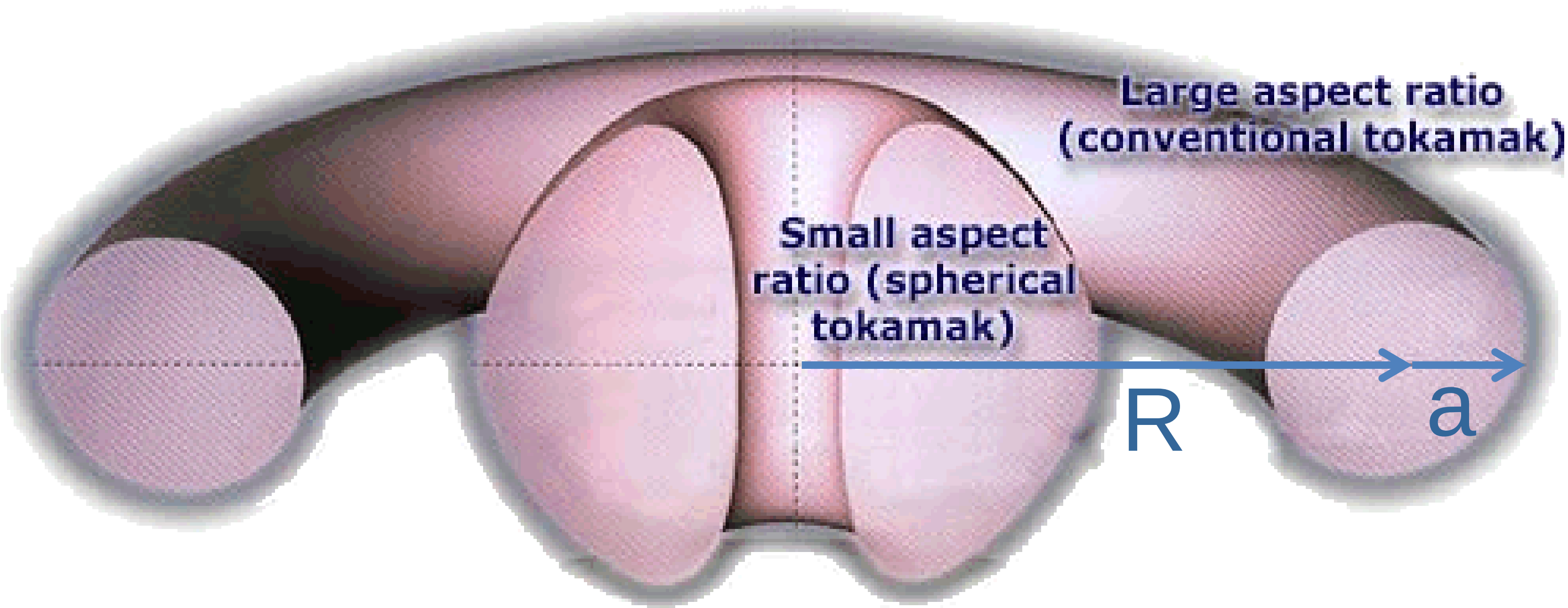
- Fusion power density $\propto$ (plasma pressure)^2
- $\beta \equiv$ plasma pressure / magnetic pressure = $p/(B^2/2\mu_0)$
- Maximum β limited by MHD instabilities
- B limited by magnet stress, cooling, quench
- Fusion power density $\propto$ $\boxed{\beta^2 B^4}$

Maximize / optimize this product

Aspect ratio is important free parameter

Aspect ratio $A = R / a$

R = major radius a = minor radius

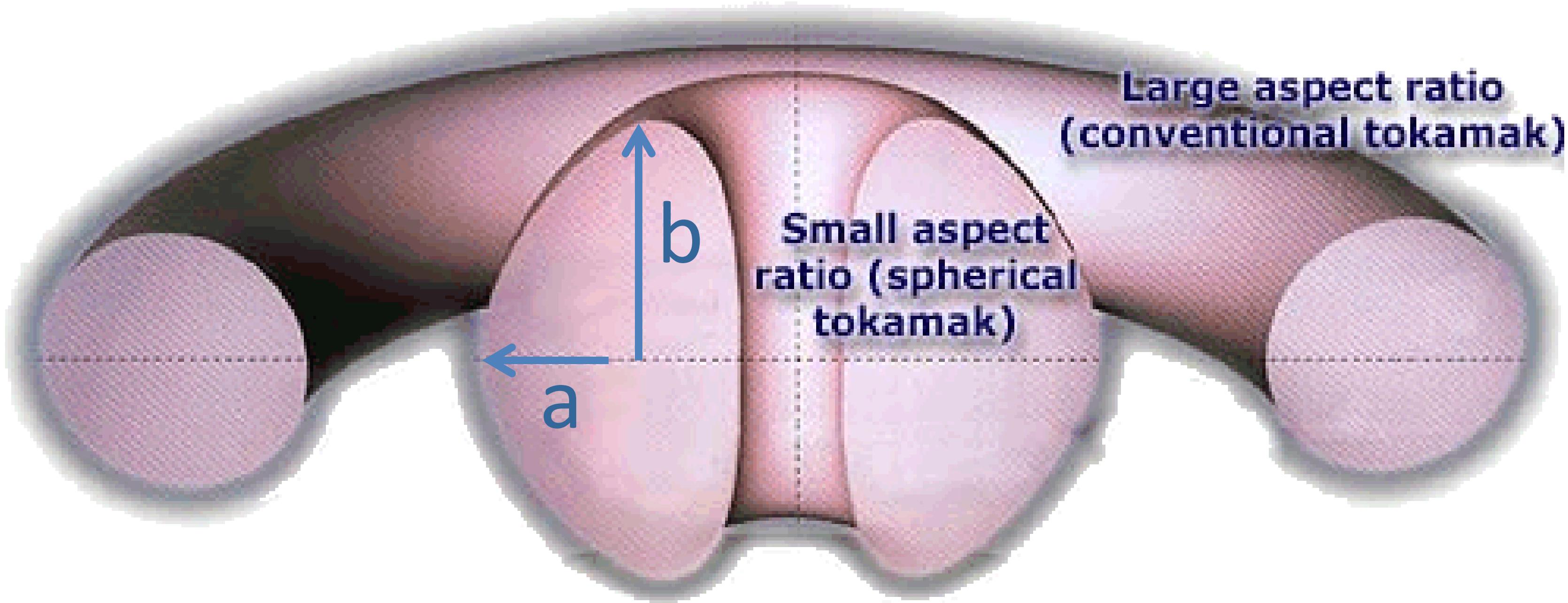


Spherical torus/tokamak (ST) has A = 1.1-2
Conventional tokamak typically A = 2.5-4

STs have higher natural elongation

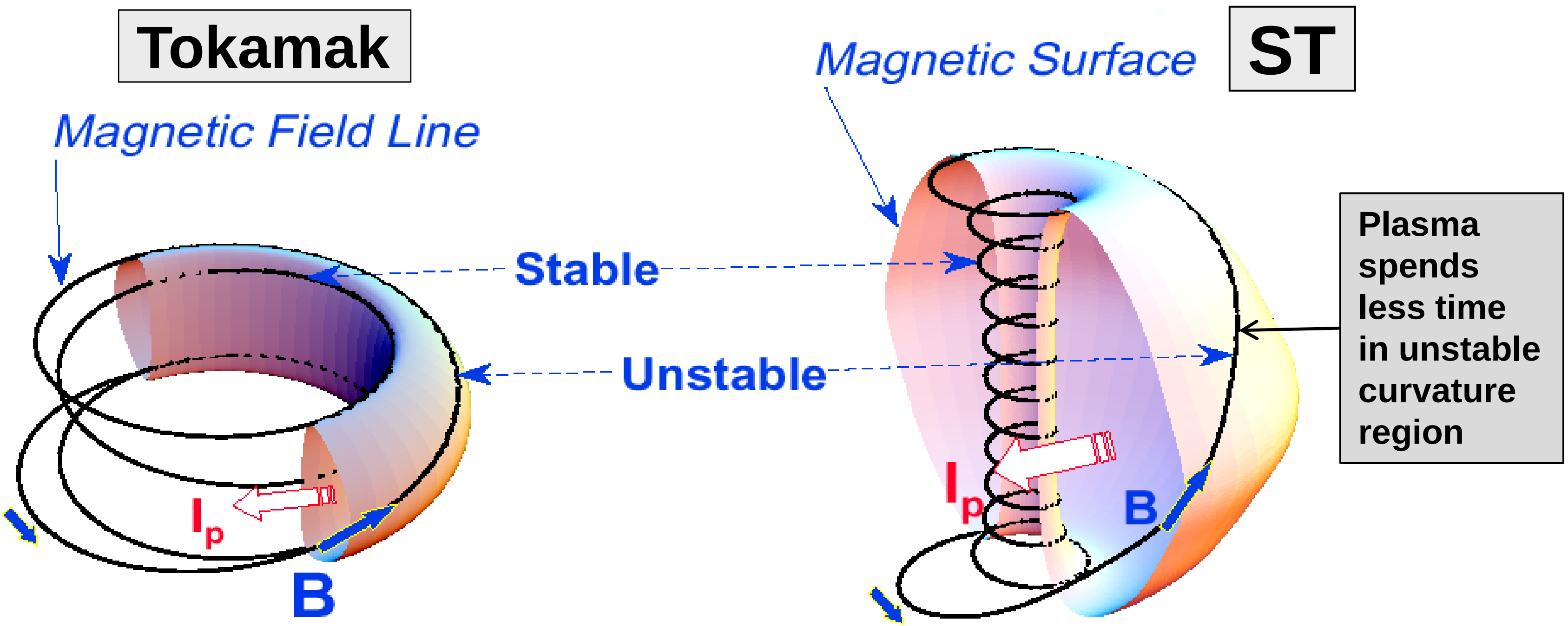
Elongation $\kappa = b / a$

b = vertical ½ height   a = minor radius



Higher elongation improves stability, confinement

Favorable average curvature improves stability



Tokamak

A ~ 2.5-4
κ = 1.5-2
β_T = 3-10%

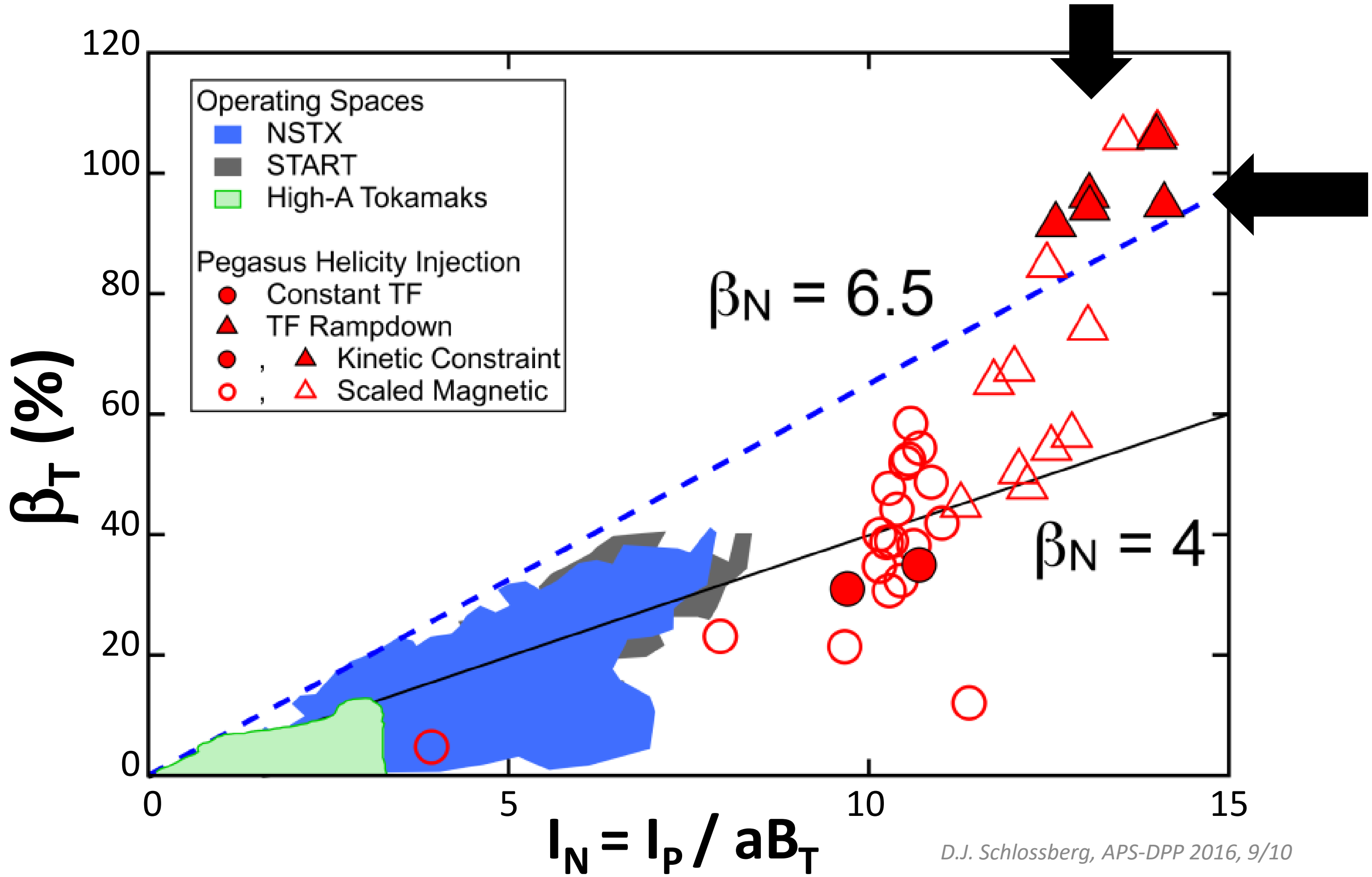
ST

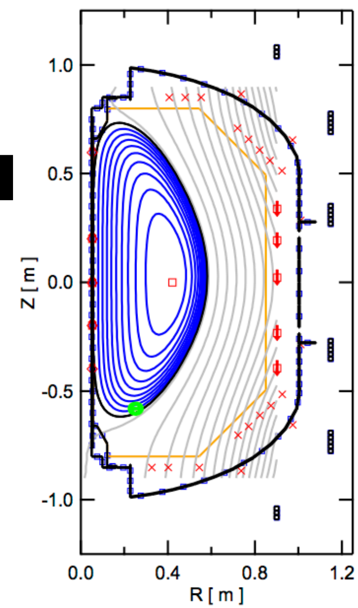
A ~ 1.3-2
κ = 2-3
β_T = 10-40%

Aspect Ratio A = R / a	**Elongation κ = 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%





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

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

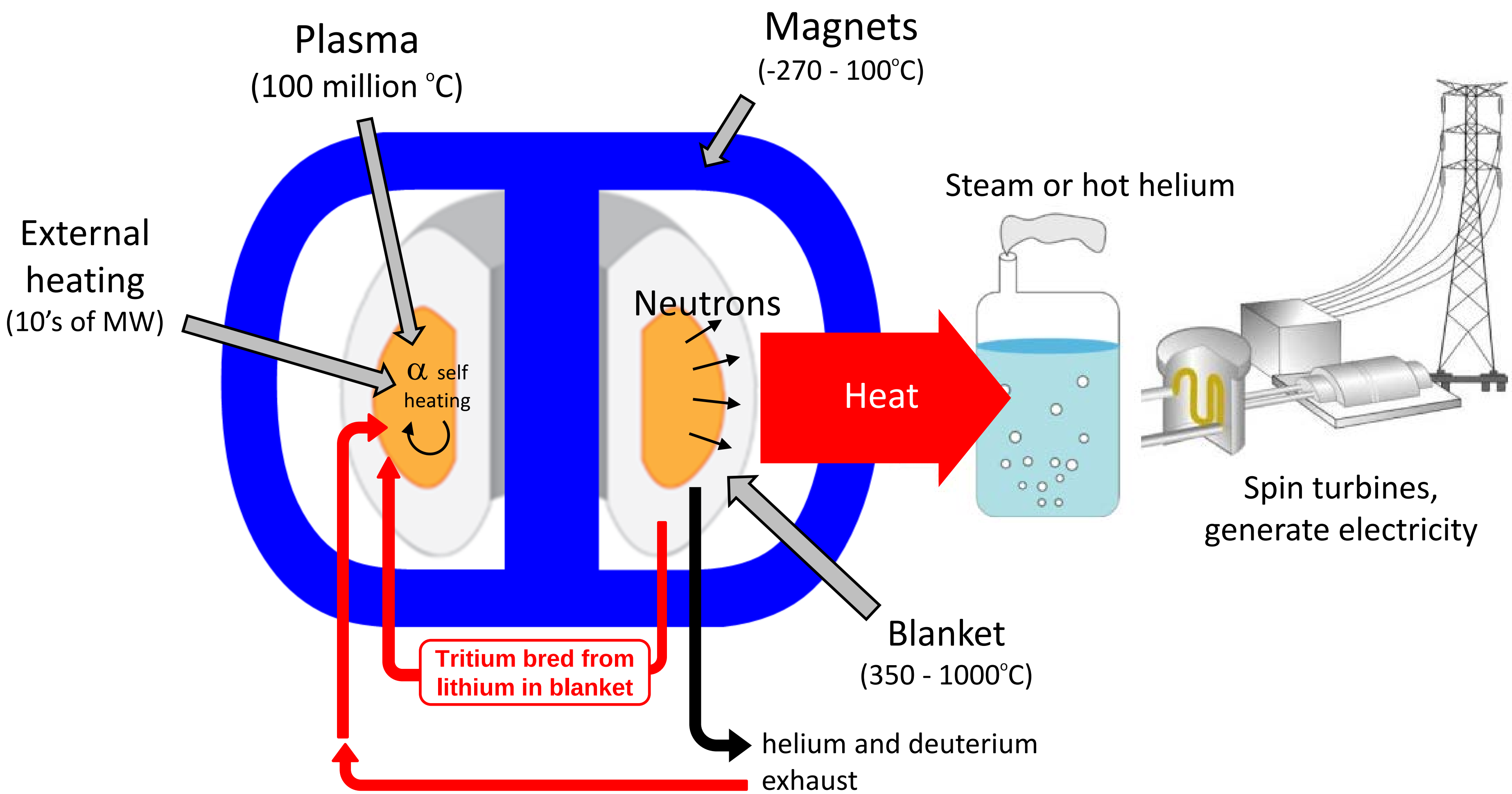
Why explore spherical torus/tokamak?

- 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

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:

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

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

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, $\alpha_P = 0.7$:

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

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

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

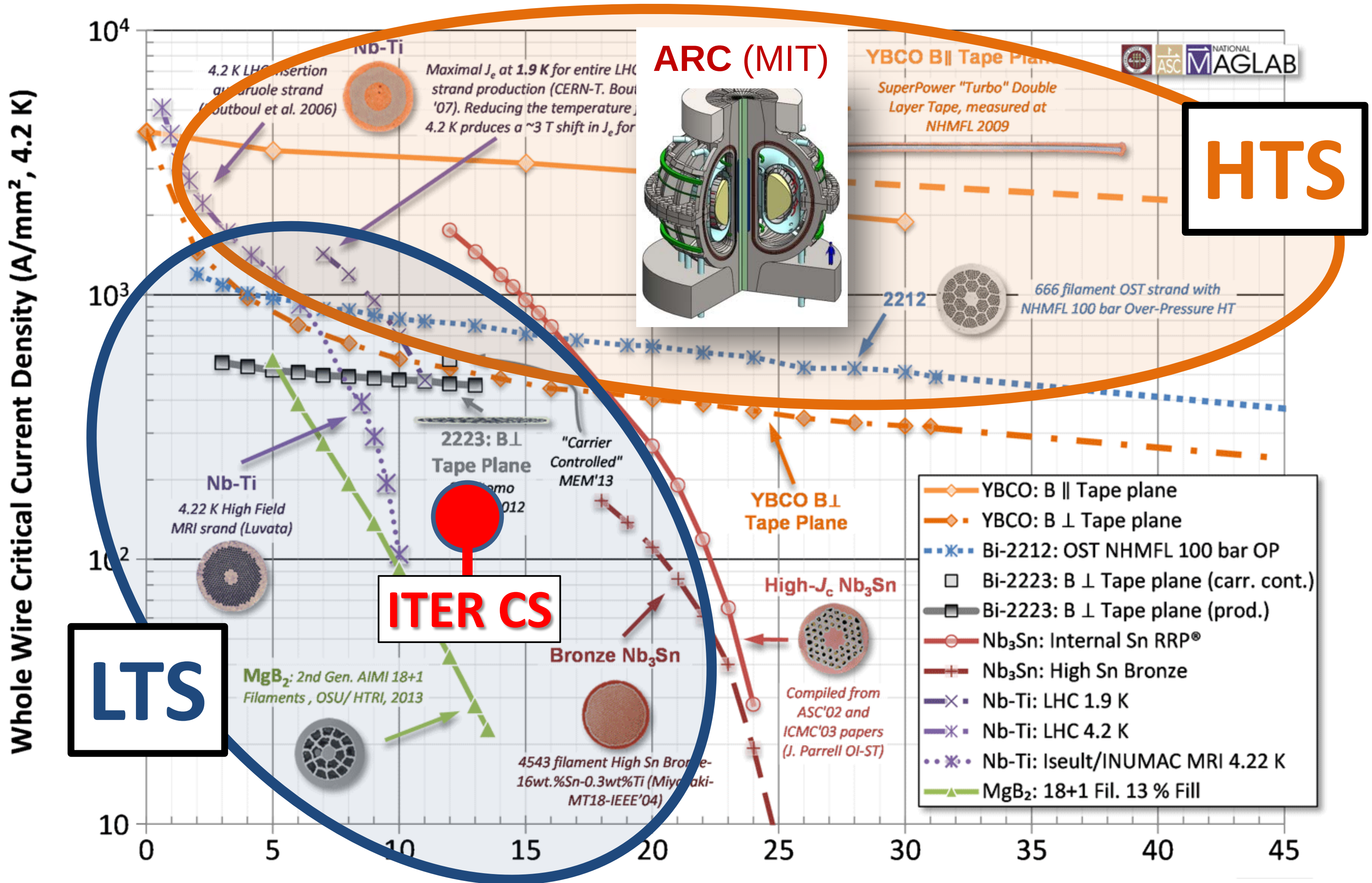
Use electrostatic gyro-Bohm τ_E scaling with no β degradation (NSTX, JET, DIII-D)

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

External current drive fraction

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

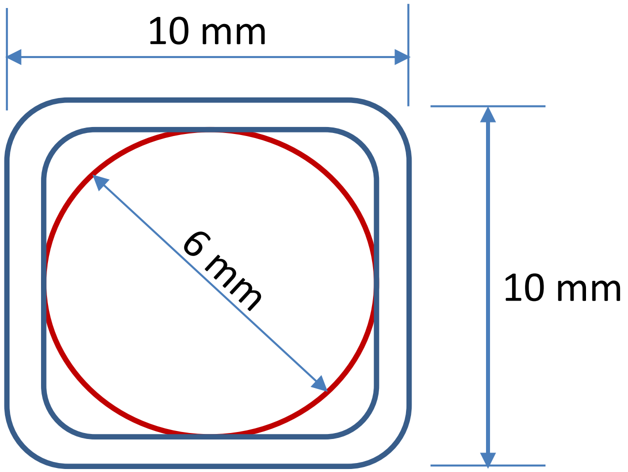
Current Density



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

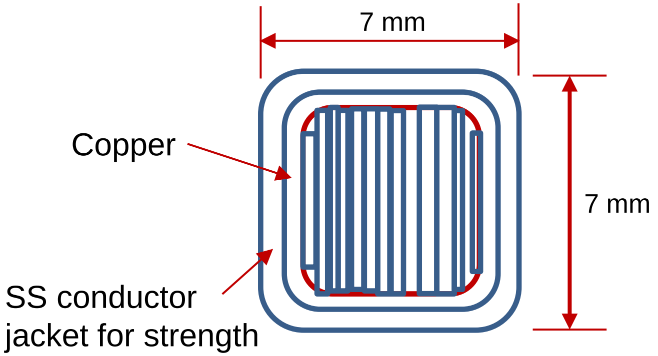


7 kA CORC (4.2K, 19 T) cable

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

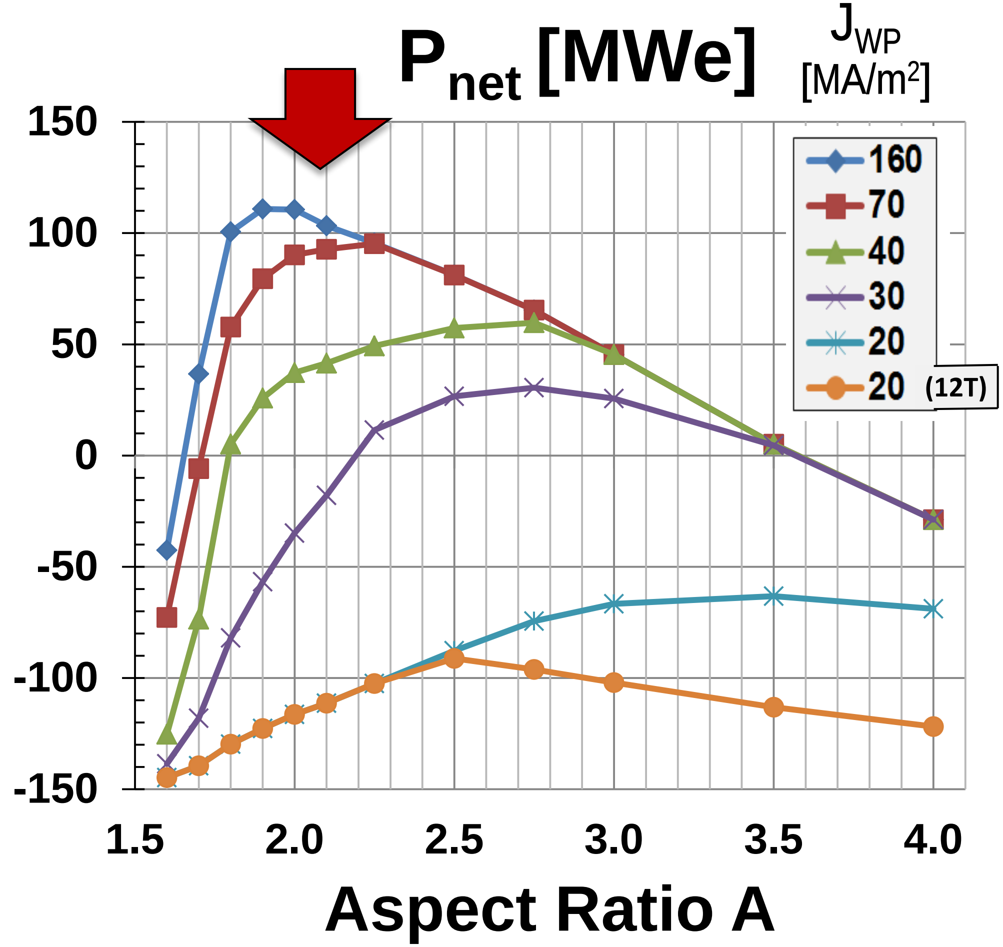


High current density HTS cable motivates consideration of lower-A tokamak pilot plants

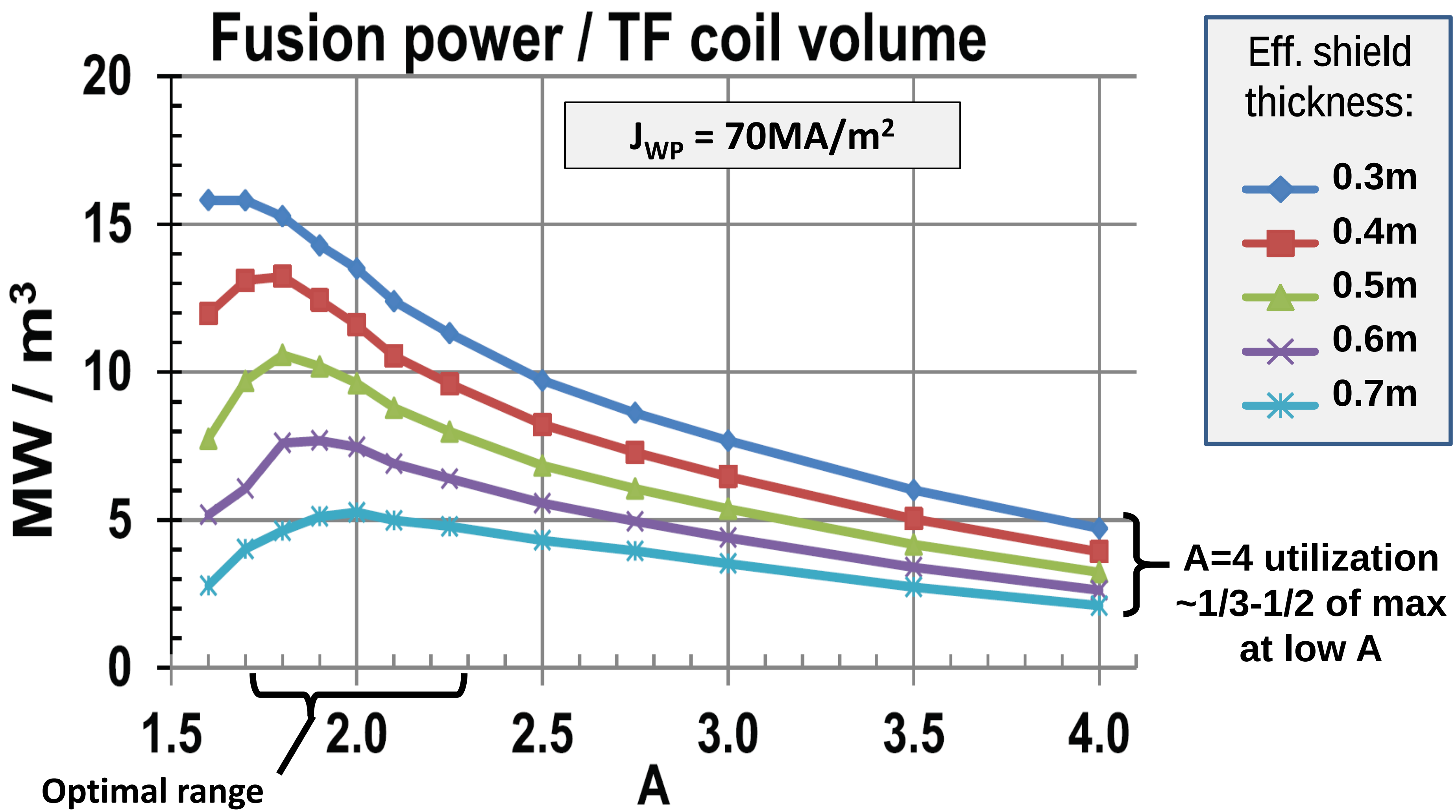
Fix plasma major radius R_0=3m, heating power P_{NNBI}=50MW

- 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**

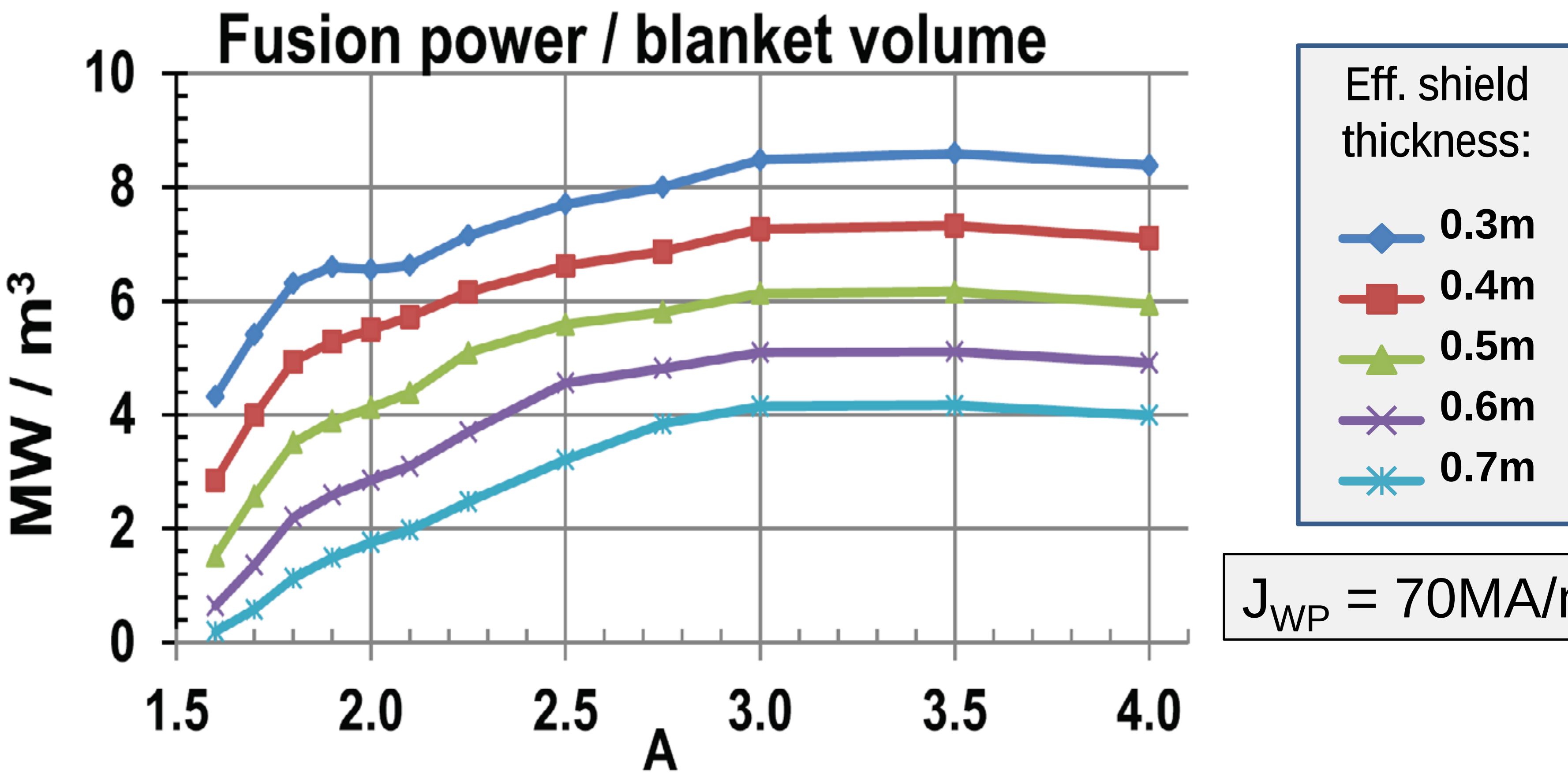
A ~ 2 attractive at high J_{WP}



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



A ≥ 3 maximizes blanket volume utilization

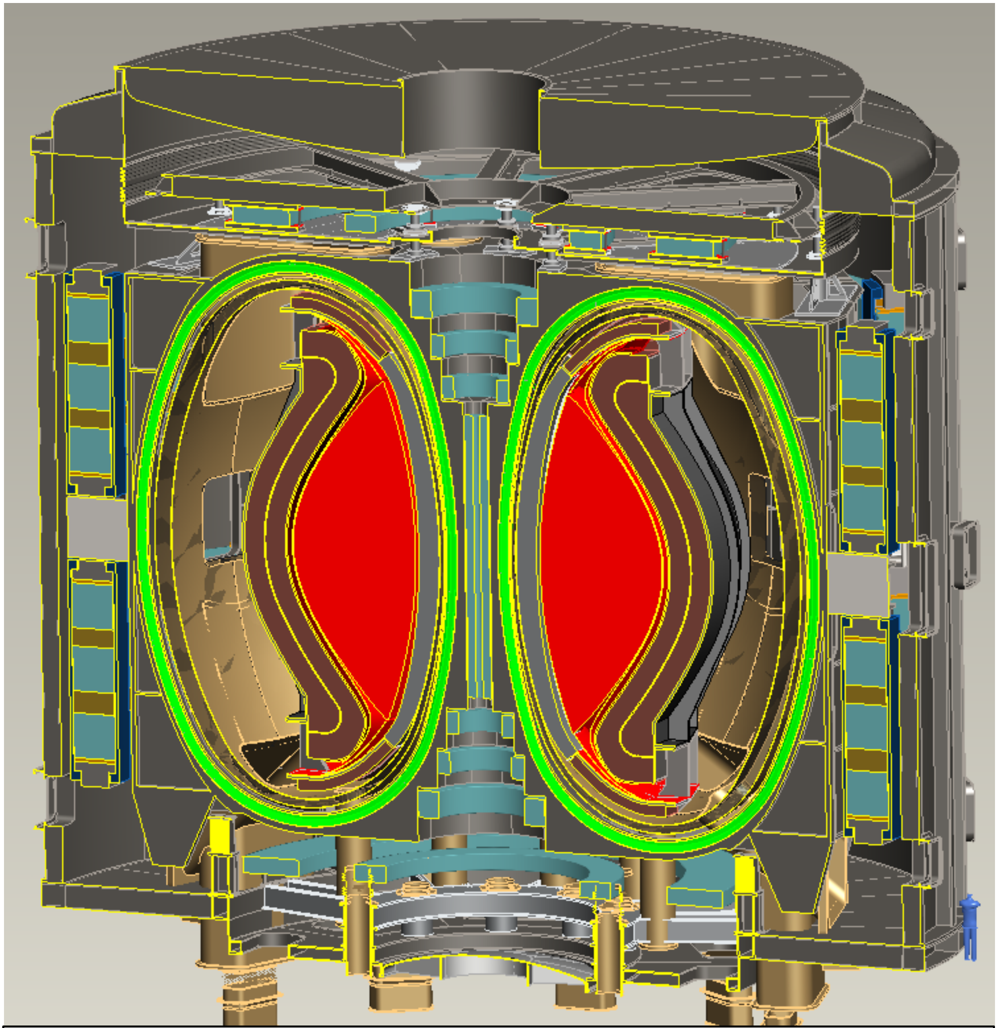


J_{WP} = 70MA/m^2

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
$\kappa = 2.5$, $\delta = 0.55$
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
$J_{WP} = 70 MA/m^2$
$B_{T\text{-}max} = 17.5T$
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?

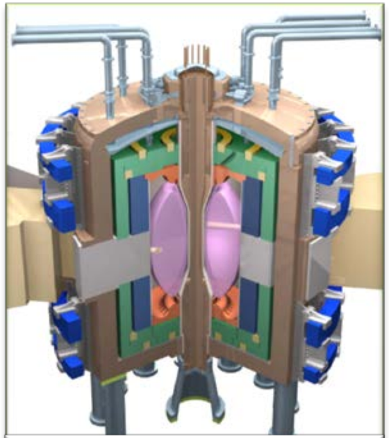
- Potentially attractive for electricity production
  - Requires accompanying magnet innovations
- High neutron wall loading in small device
  - Well suited for fusion nuclear component R&D
- Improve toroidal physics predictive capability
  - High β and high temperature at low collisionality
  - Understand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

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

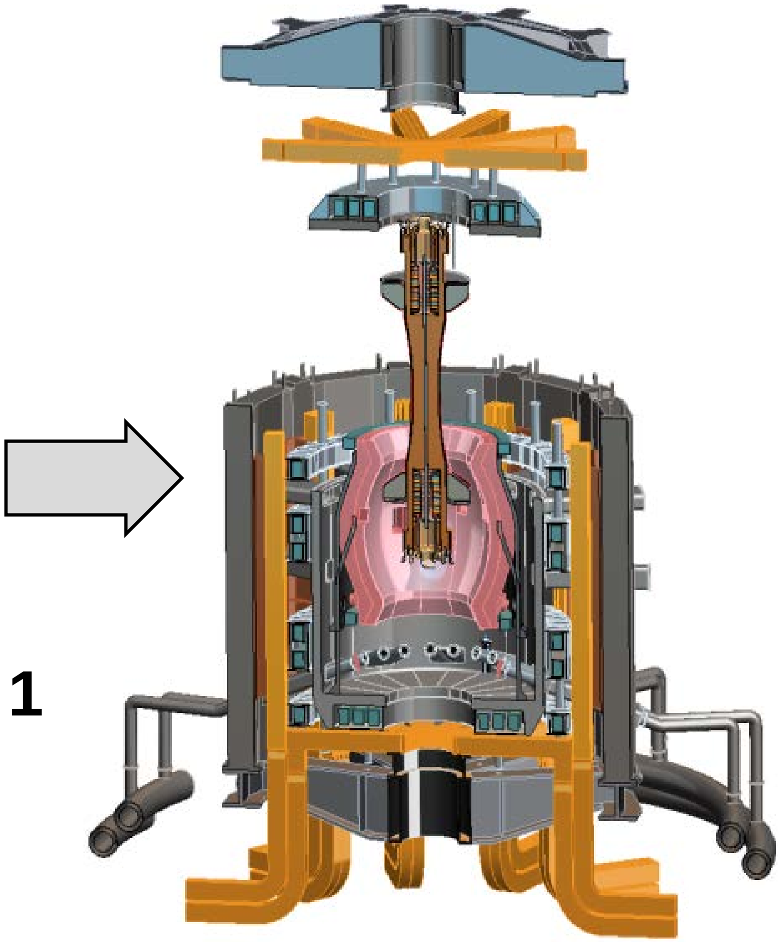
Y.-K.M. Peng (ORNL)

- **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

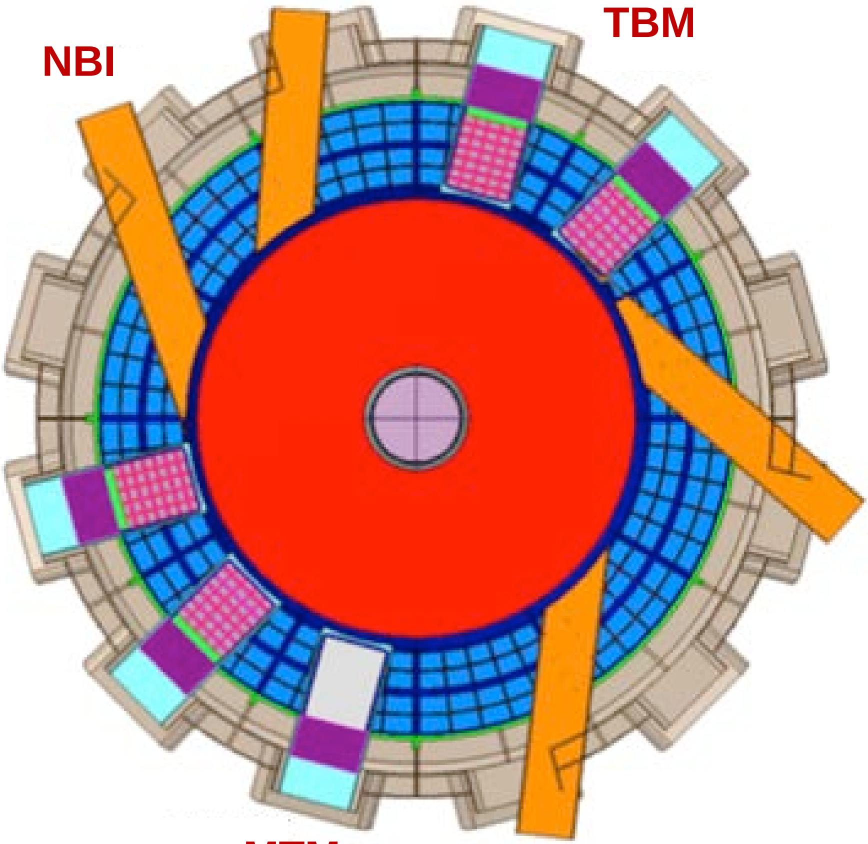
- **Projected to access high neutron wall loading at moderate R, P_{fusion}**
  - $W_n \sim 1\text{-}2\ MW/m^2$
  - $P_{fus} \sim 50\text{-}200MW$
  - $R \sim 0.8\text{-}1.8m$
- **Modular design, maintenance**
- **Tritium breeding ratio (TBR) near 1**
  - Requires sufficiently large R
  - Careful layout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

PPPL ST-FNSF concept



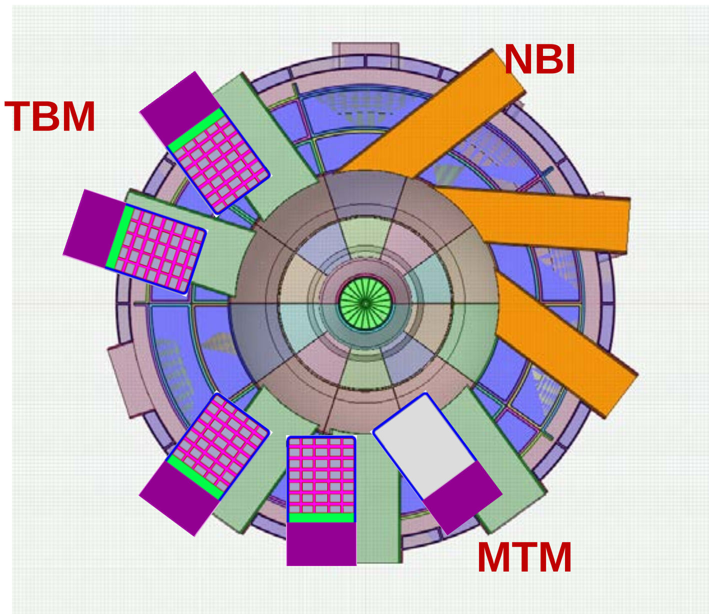
R ≥ 1.7m necessary for net breeding at A=1.7

R=1.7m: **TBR ≥ 1**



TBM = Test Blanket Module
MTM = Materials Testing Module

R=1.0m: **TBR < 1 (≈ 0.9)**



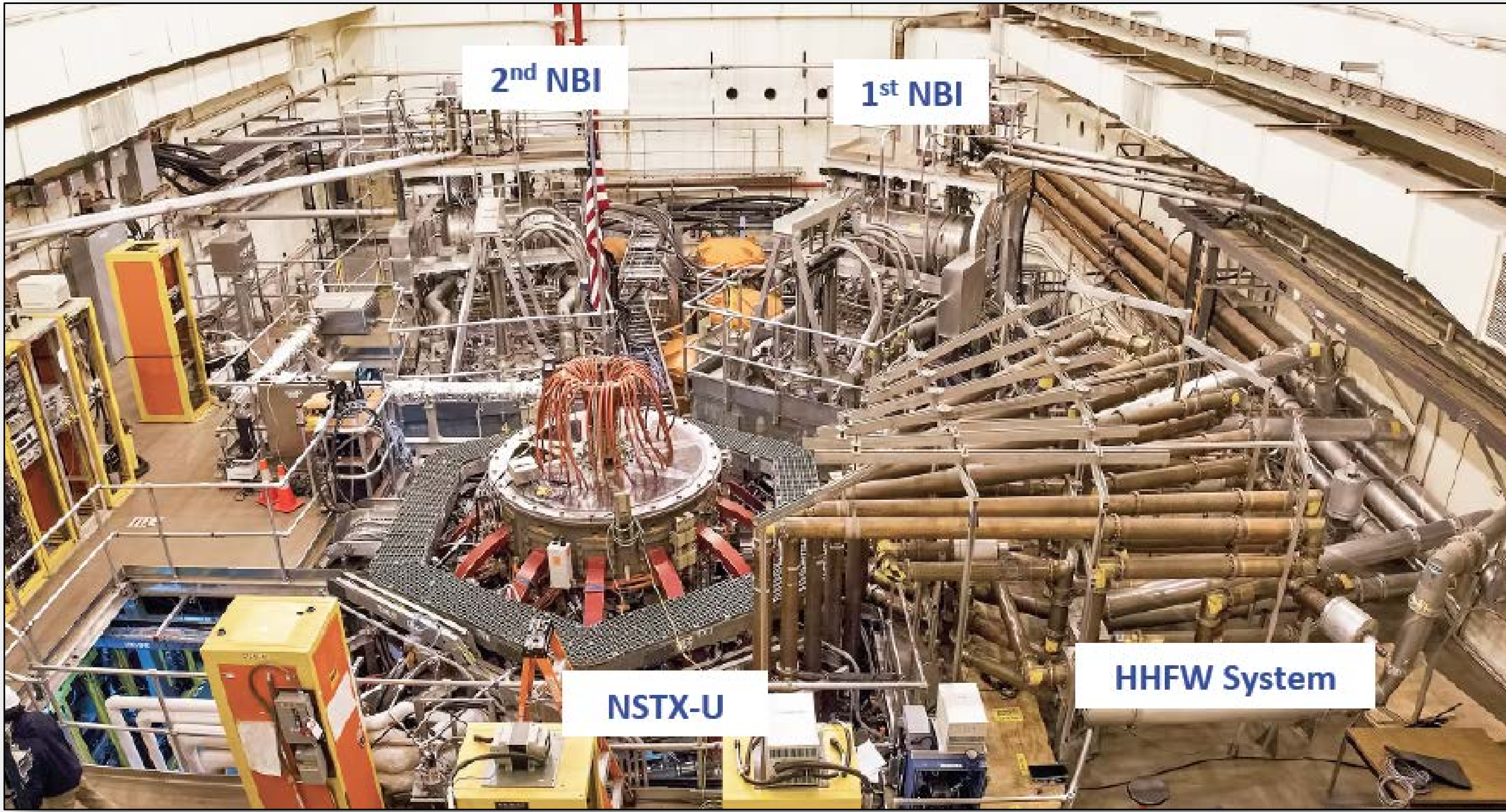
- Need to purchase Tritium from outside sources:
  - –$12-55M / full power year (FPY)

WISCONSIN
UNIVERSITY OF WISCONSIN–MADISON

Why explore spherical torus/tokamak?

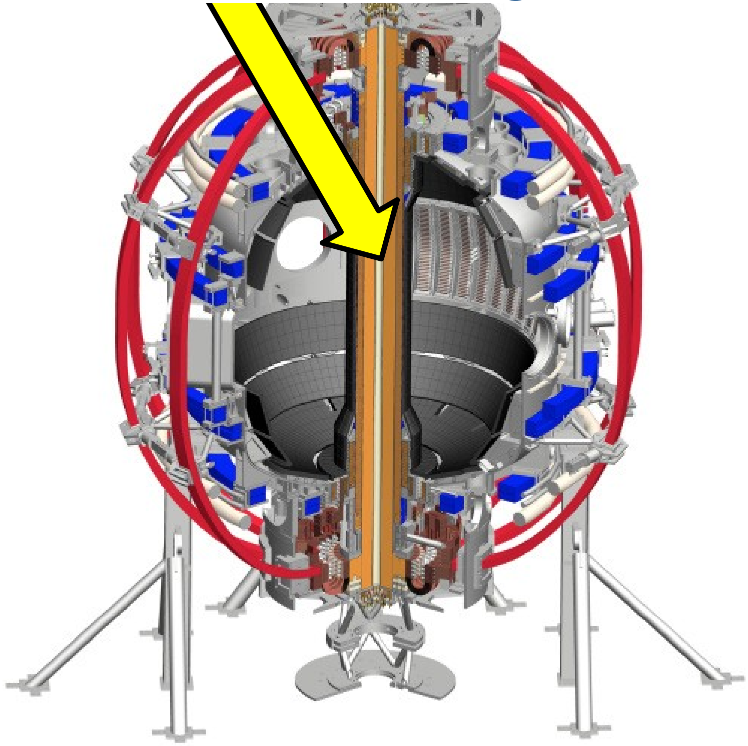
- Potentially attractive for electricity production
  - Requires accompanying magnet innovations
- High neutron wall loading in small device
  - Well suited for fusion nuclear component R&D
- Improve toroidal physics predictive capability
  - High β and high temperature at low collisionality
  - Understand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

NSTX Upgrade Device and Test Cell – Aerial View



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

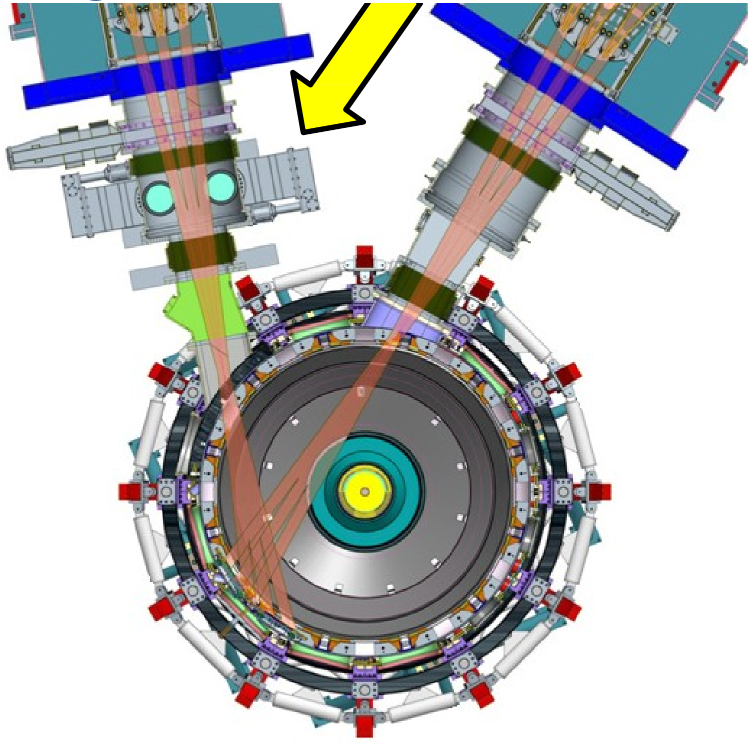
1. New Central Magnet



Higher T, low ν^* from low to high β

→ Unique regime, study new transport and stability physics

2. Tangential 2nd Neutral Beam

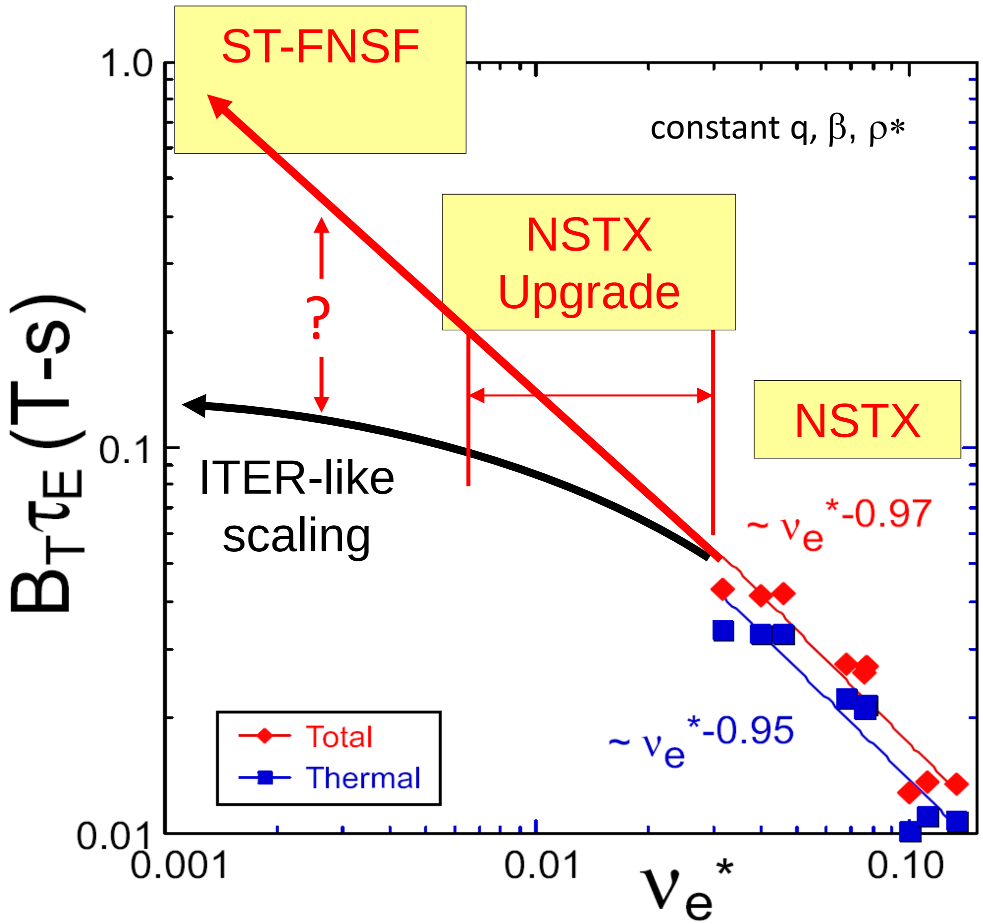


Full non-inductive current drive

→ 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?

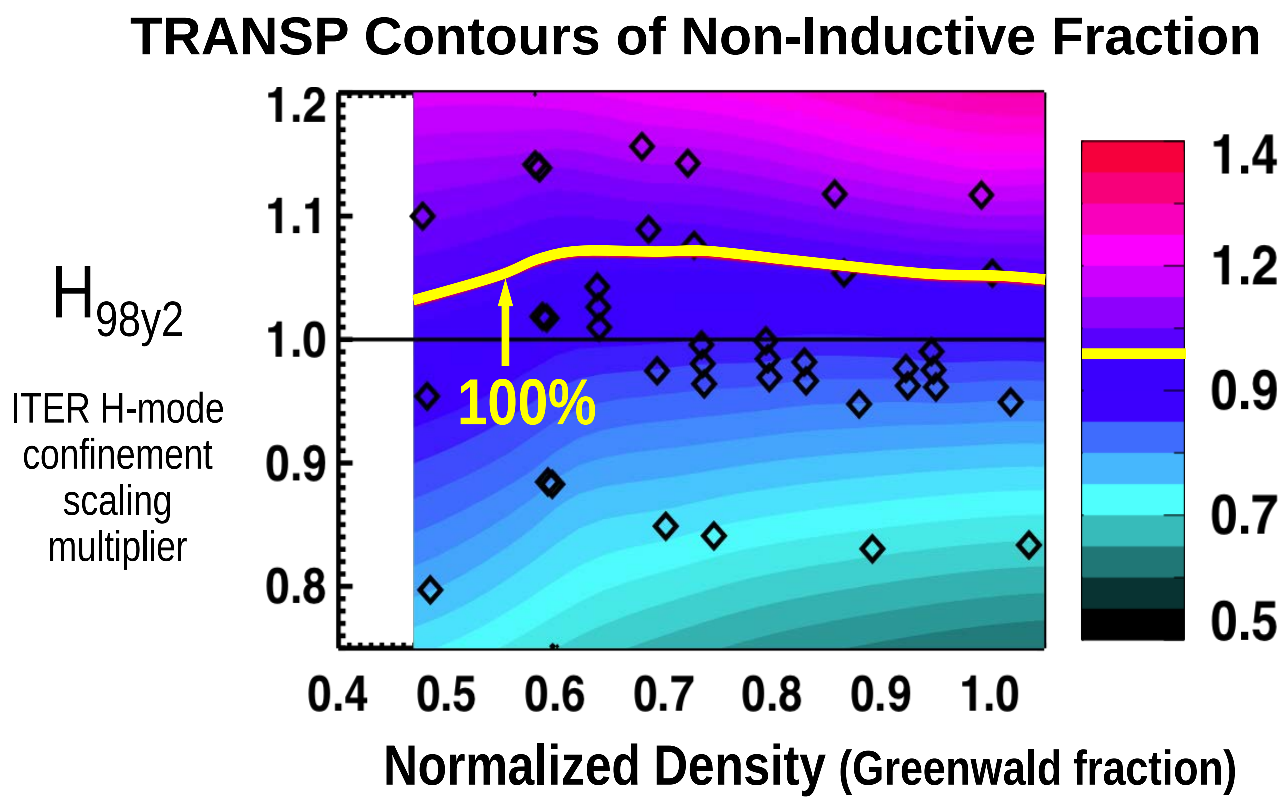


Normalized electron collisionality $\nu_e^* \propto n_e / T_e^2$

Favorable confinement results could lead to more compact ST reactors

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

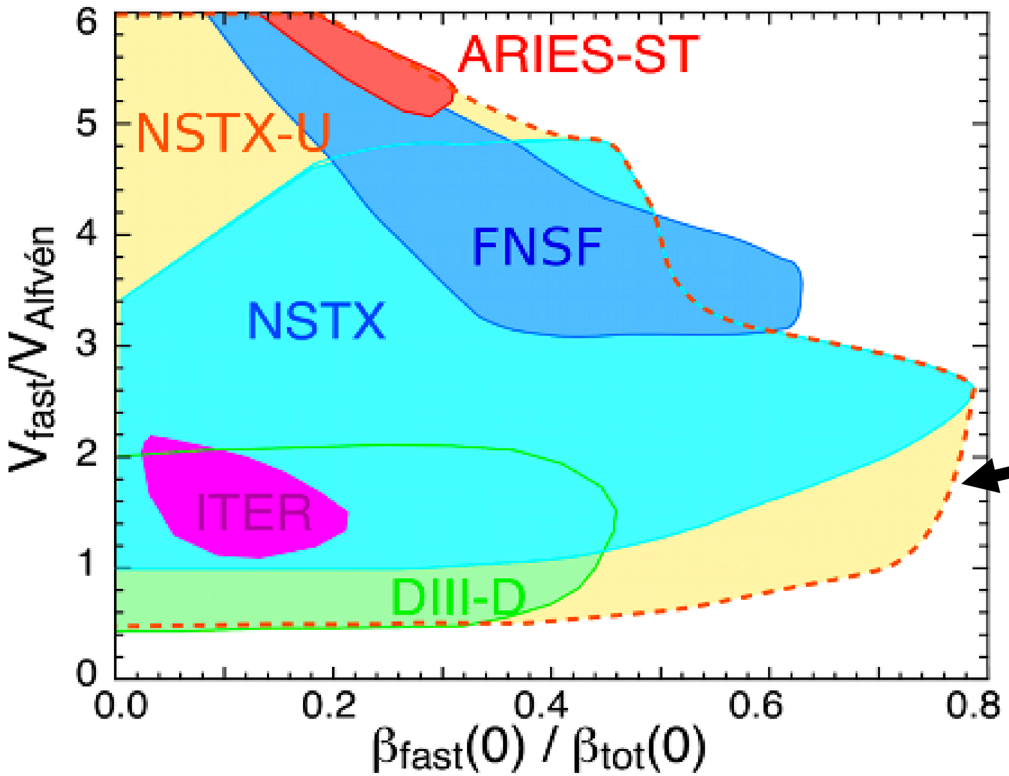
NSTX achieved 70% "transformer-less" current drive
Will NSTX-U achieve 100% as predicted by simulations?



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

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

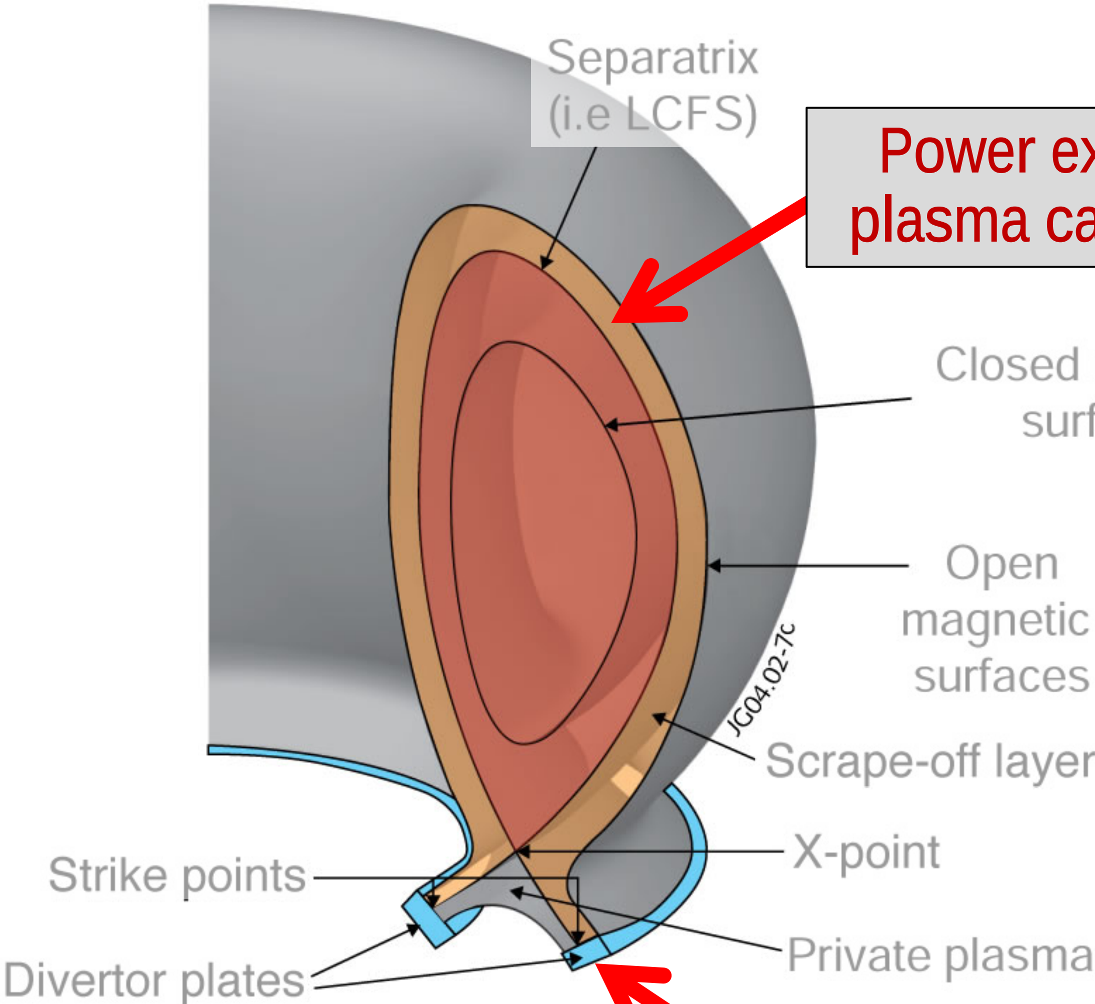
NBI-heated STs excellent testbed for α-particle physics



- 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?**

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



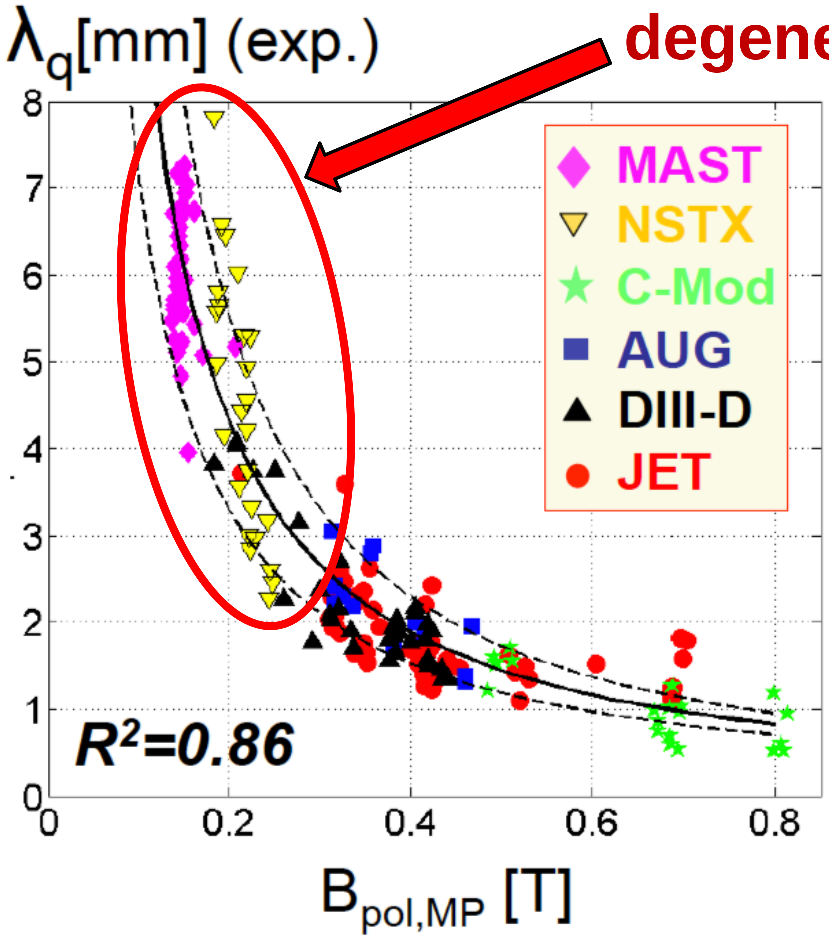
Power exhaust width outside main plasma can be very narrow (few mm)

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

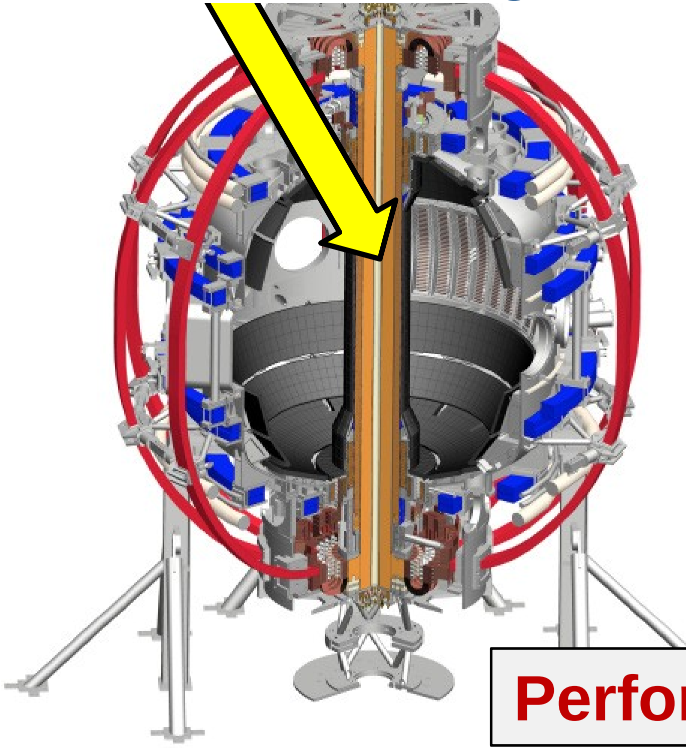


Issue: If peak heat flux in divertor region exceeds ~10 MW/m^2 → material damage

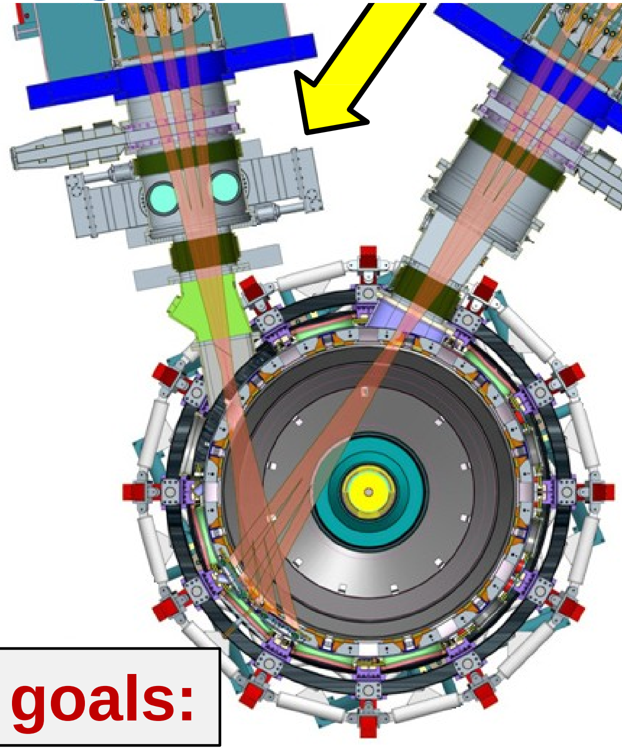
Wider heat-flux width may offset smaller R → maybe better than tokamak

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)

Outline

- Introduction
- Why spherical tori / tokamaks (STs)
- NSTX-U initial results, long-term directions
- Summary

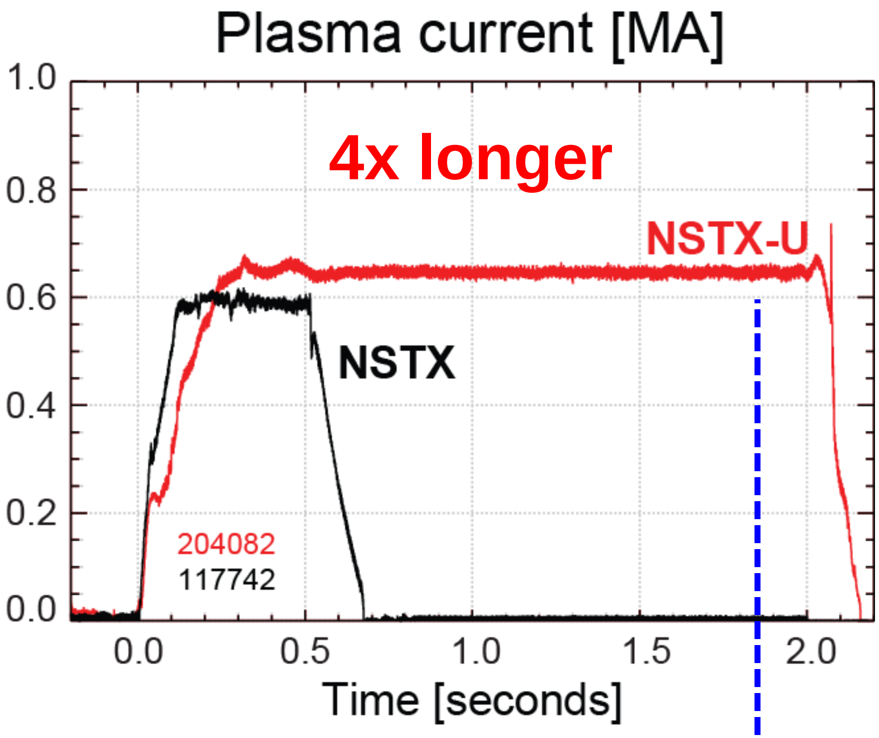
NSTX-U had scientifically productive 1st year

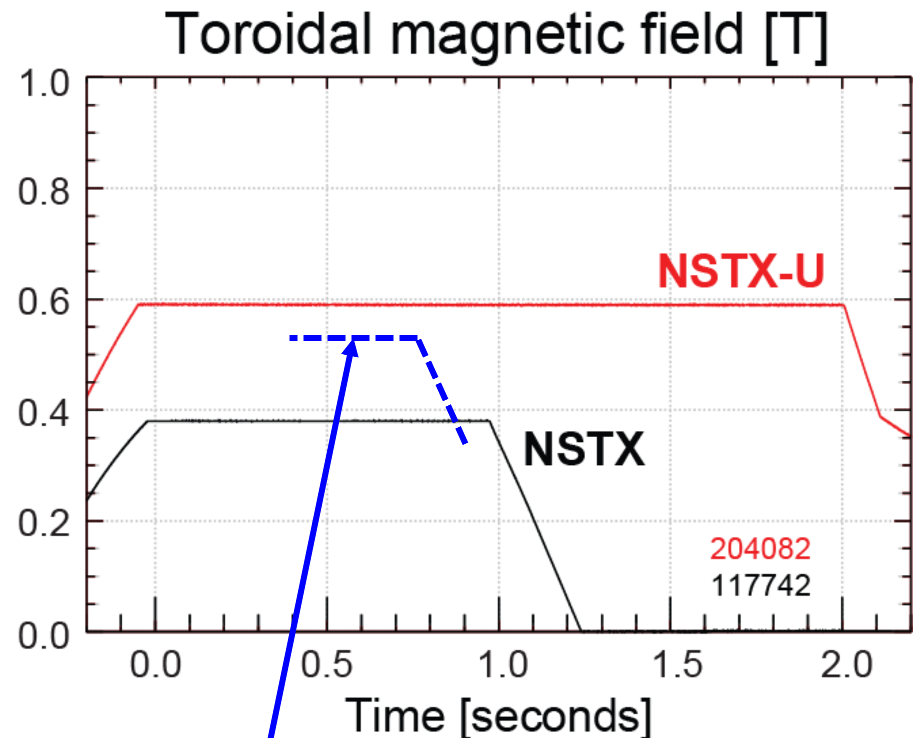
- Achieved H-mode on 8th day of 10 weeks of operation
- Surpassed magnetic field and pulse-duration of NSTX
- Matched best NSTX H-mode performance at ~1MA
- Identified and corrected dominant error fields
- Commissioned all magnetic and kinetic profile diagnostics
- Injected up to 12MW NBI power into 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 during 2018 – 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

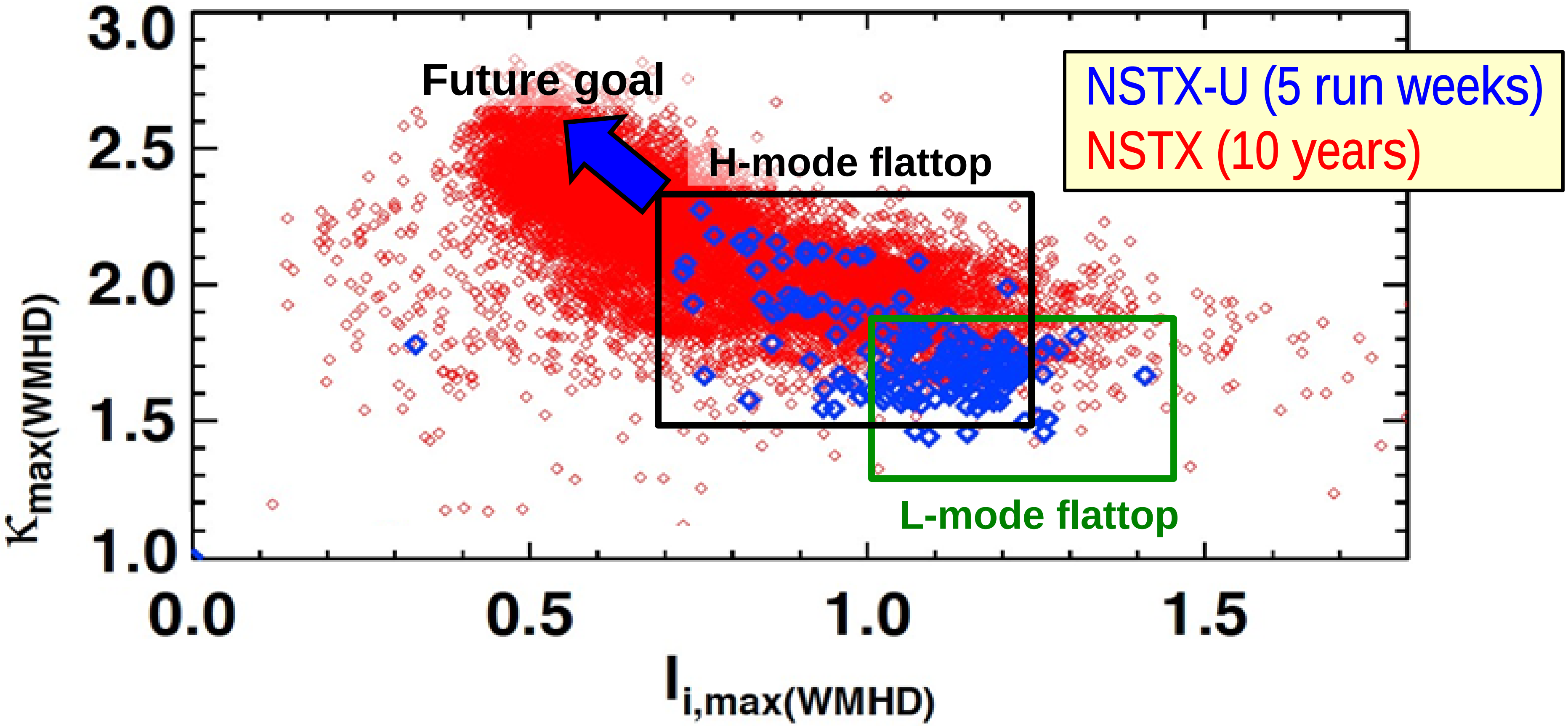




NSTX-U L-mode duration exceeds longest NSTX H-mode

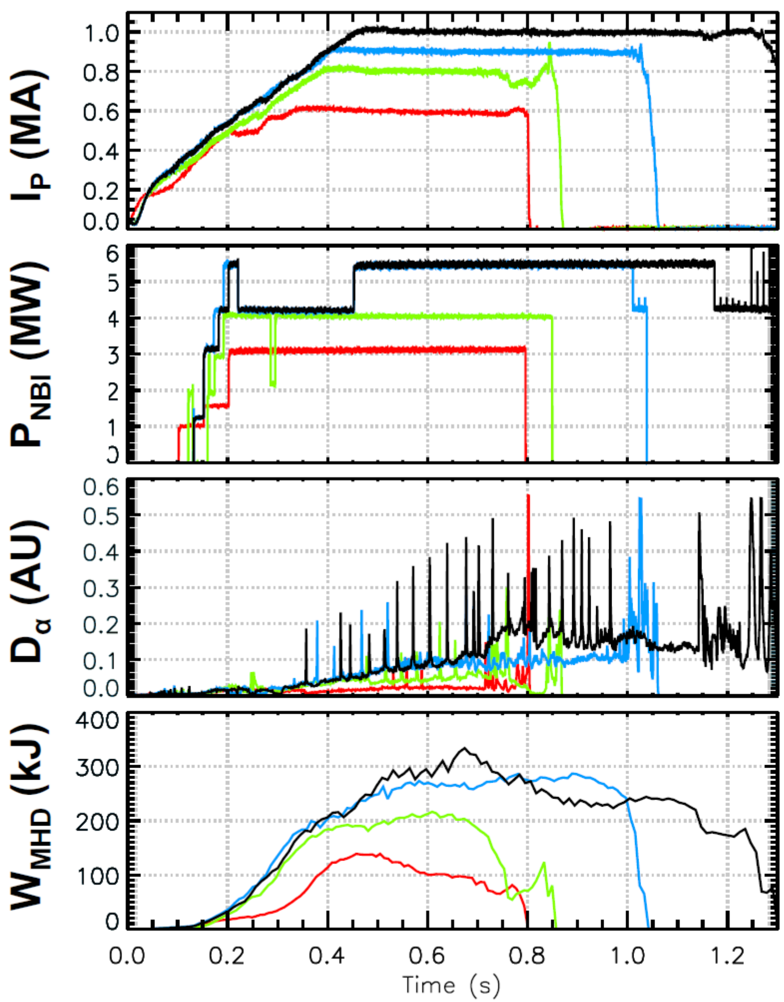
NSTX-U B_T > highest NSTX B_T

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

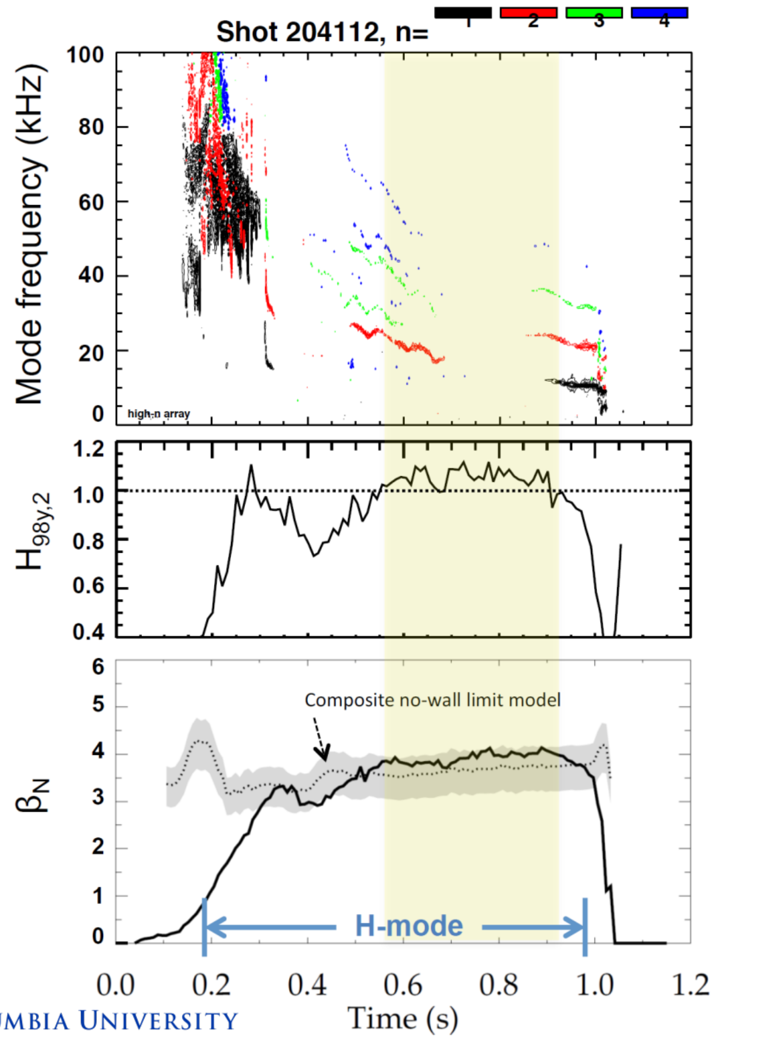


- **Goal: Internal inductance l_i = 0.5-0.7 → κ = 2.4-2.7**

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

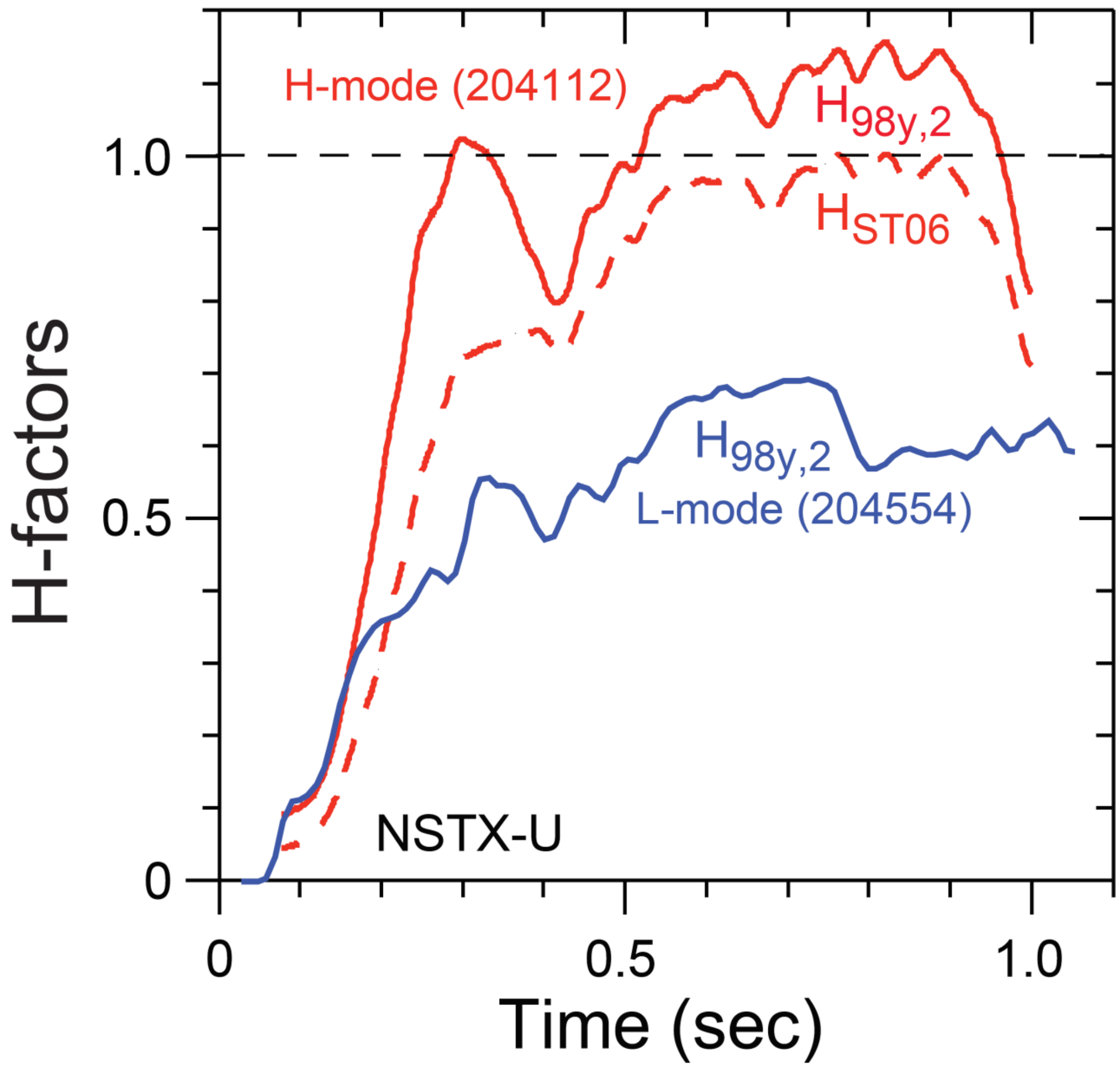


$H_{98} \geq 1$, $\beta_N \sim 3.5\text{-}4 \geq n=1$ no-wall limit



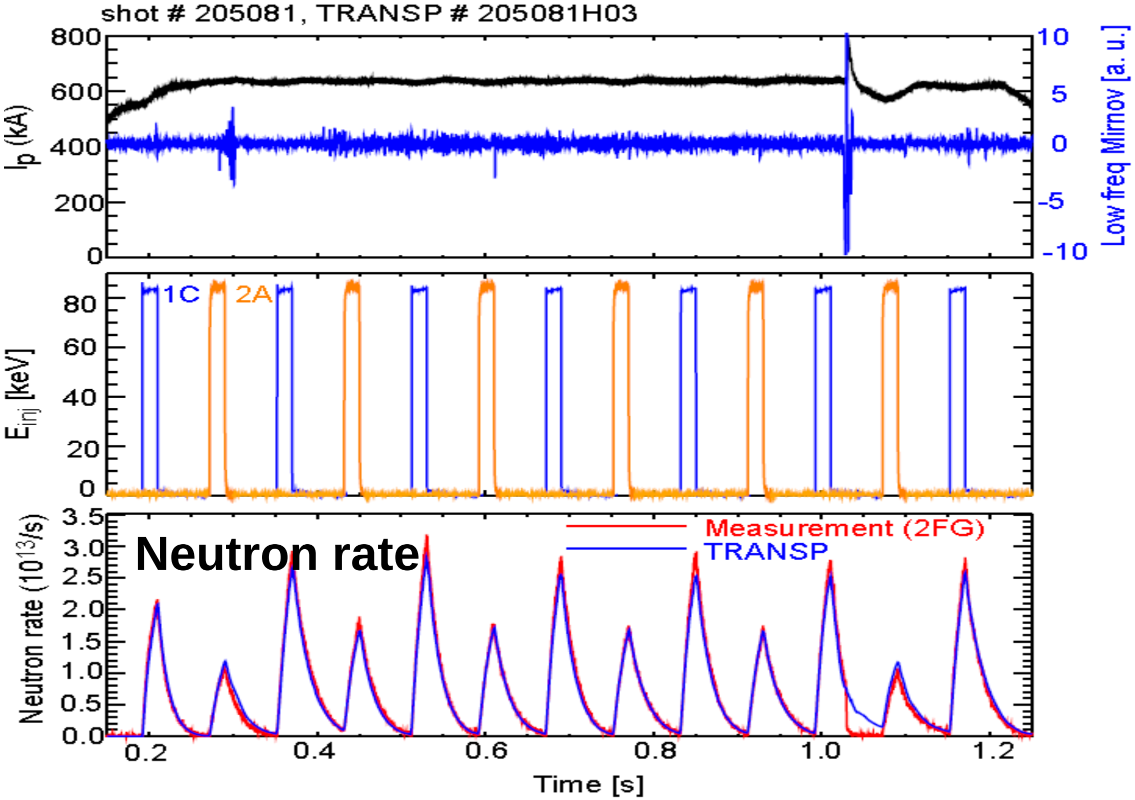
COLUMBIA UNIVERSITY IN THE CITY OF NEW YORK

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

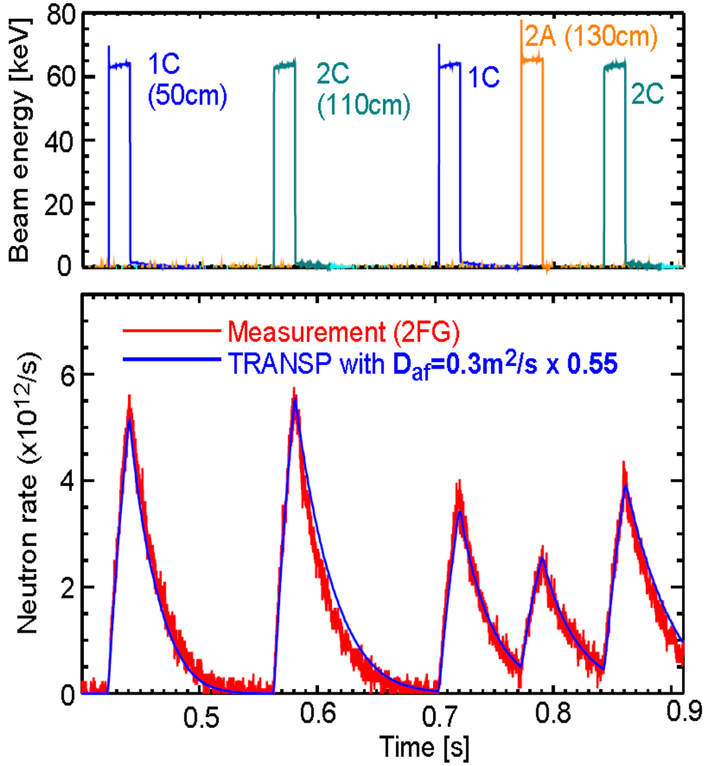


Fast-ion confinement measured to be at / near predicted values at low total NBI power ~1-2MW

E_{NBI} = 85keV



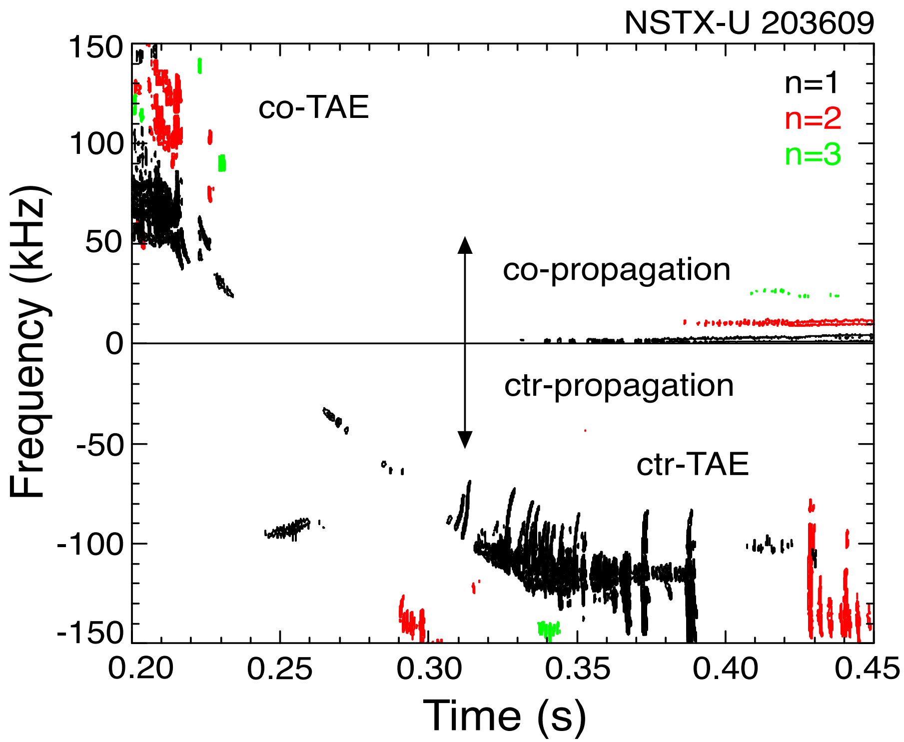
E_{NBI} = 65keV



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

UCIrvine

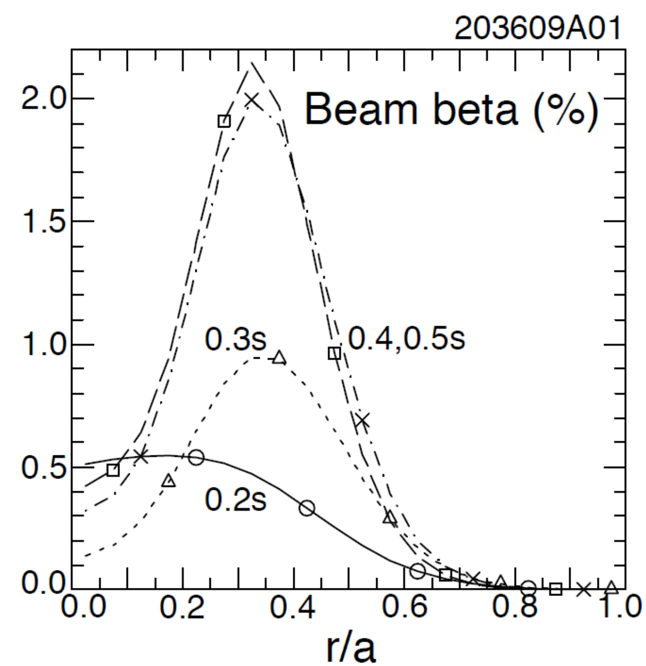
New: Most tangential NBI generates counter-propagating Toroidal Alfvén Eigenmodes (TAEs)



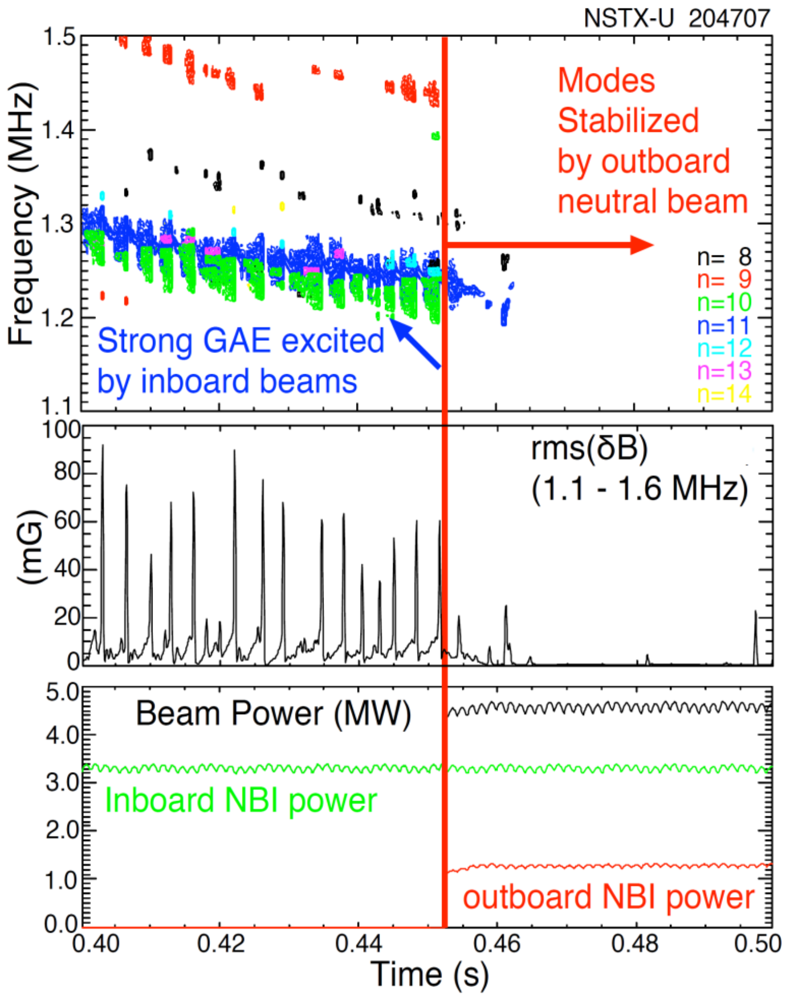
- Counter-propagating TAE predicted for **hollow** fast-ion profiles

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

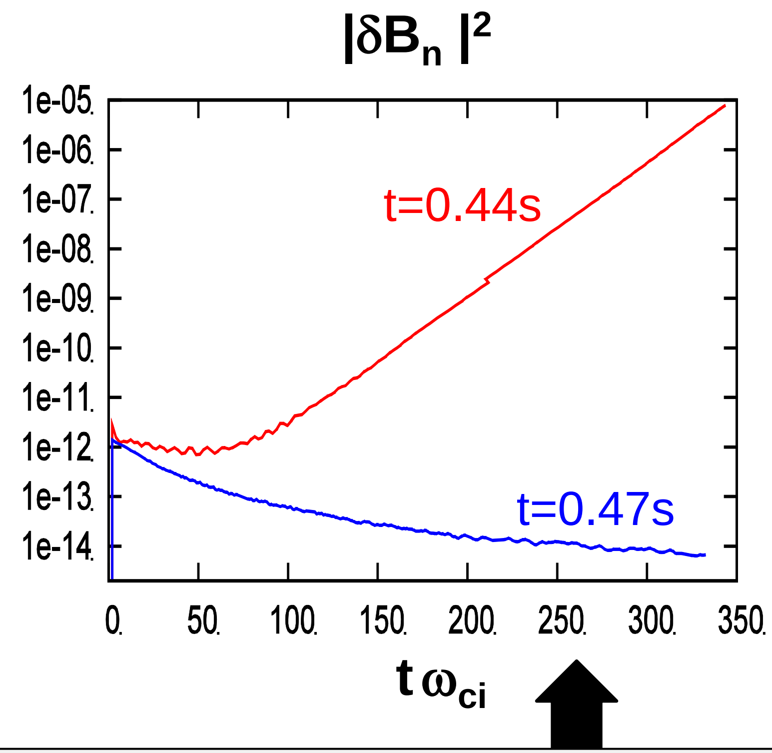
- TRANSP: As current builds up beam fast-ion beta profile predicted to become hollow
- **1st evidence of off-axis NBI in NSTX-U**



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

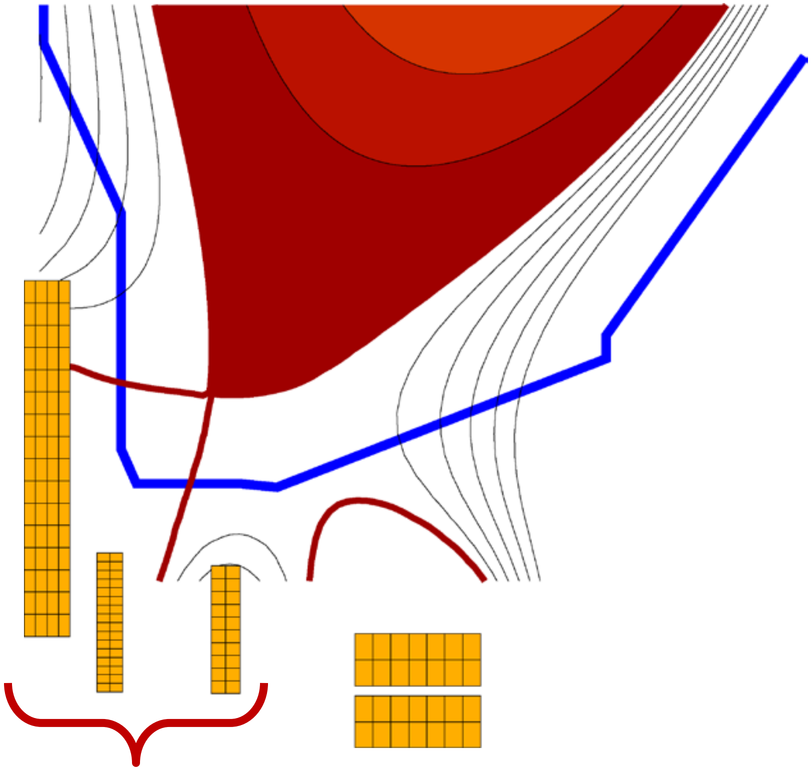
Goals for future NSTX-U operation

- 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

STs leading advanced divertor development

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

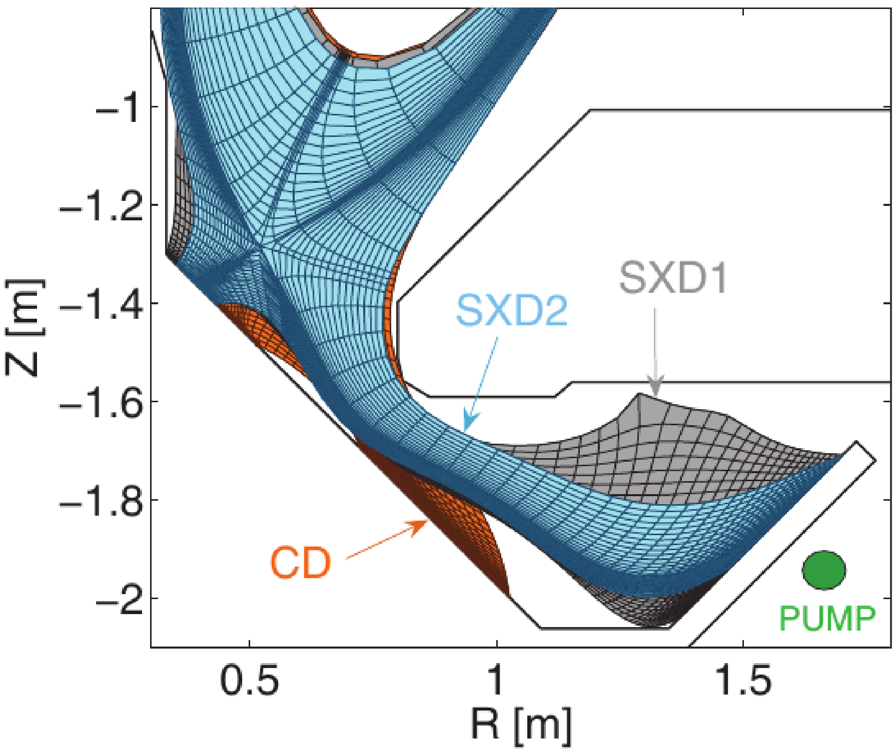
NSTX-U: Flared divertor using "snowflake/X" + radiation



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

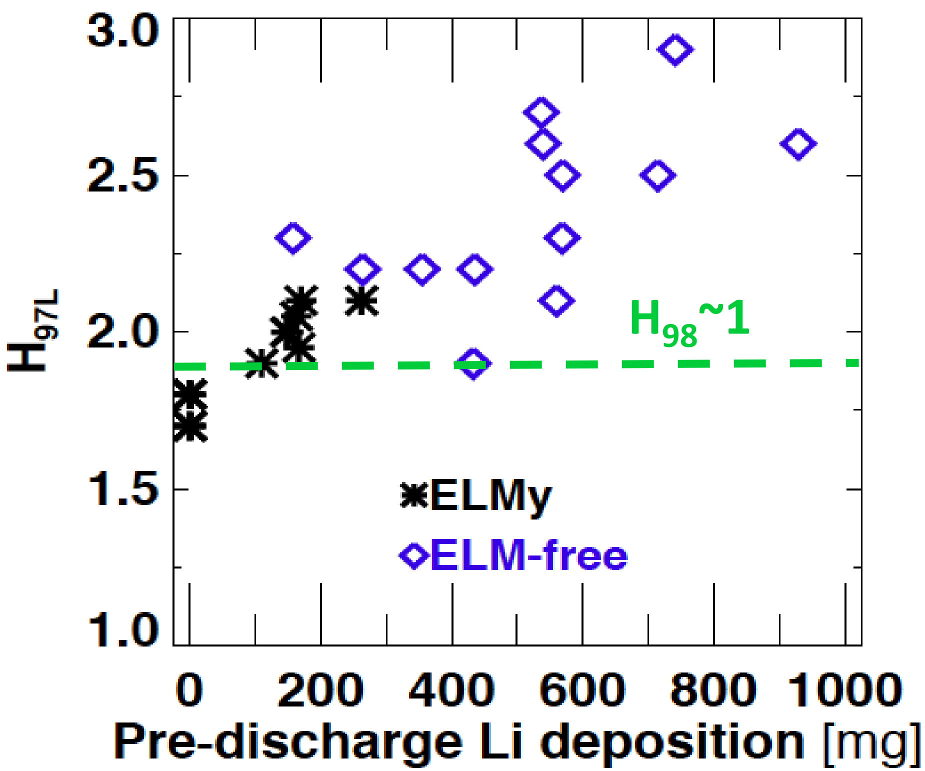


E. Havlickova, et al., Plasma Phys. Control. Fusion 56 (2014) 075008

STs investigating lithium (solid and liquid) walls to significantly increase energy confinement

NSTX (wider → higher pedestals)

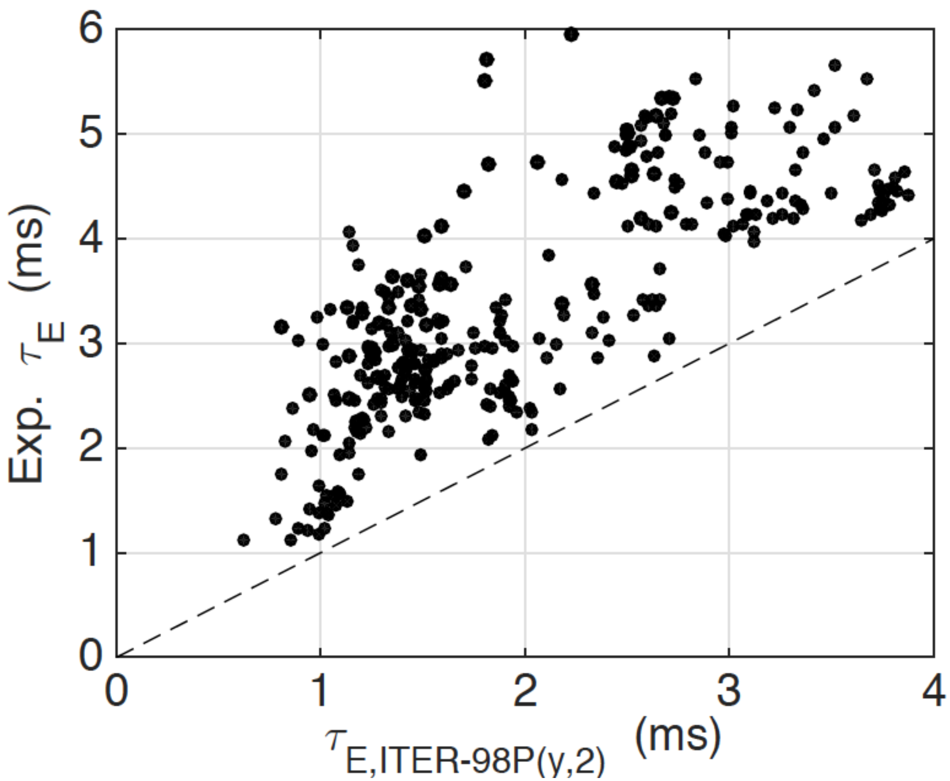
$H = 0.8 \rightarrow 1.4 \times$ ITER98P(y,2)



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

LTX (flatter → higher T profiles)

$H = 2\text{-}4 \times$ ITER98P(y,2)

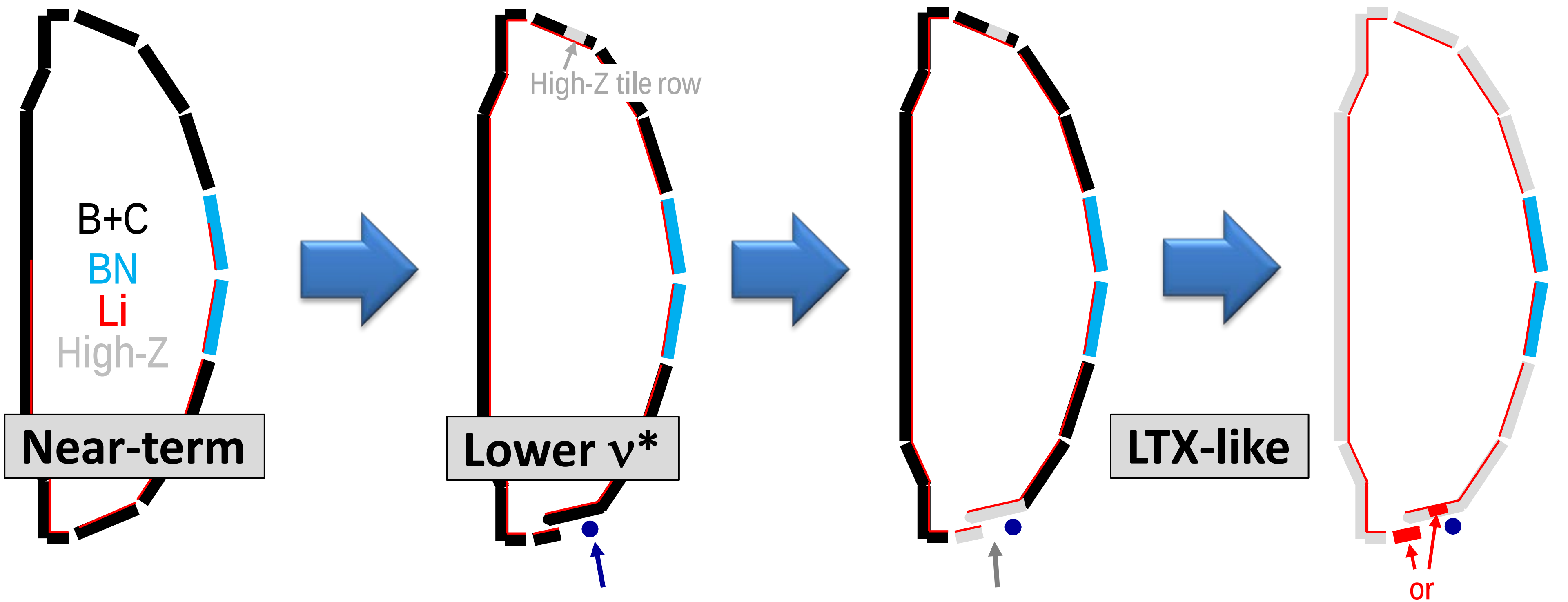


J.C. Schmitt, et al., Phys. Plasmas 22 (2015) 056112

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

Summary

- STs broaden our understanding of toroidal physics and enhance predictive capability for ITER & beyond
- ST potentially attractive as FNSF and Pilot Plant
- NSTX-U designed to be highest performance ST in world program - this is why the project is challenging
- Team is working very hard to ensure NSTX-U can run reliably at high performance & get back into operation!

Thank you!

Any questions?