Progress and plans for NSTX Upgrade and prospects for next-step spherical tori

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On behalf of the NSTX-U Research Team

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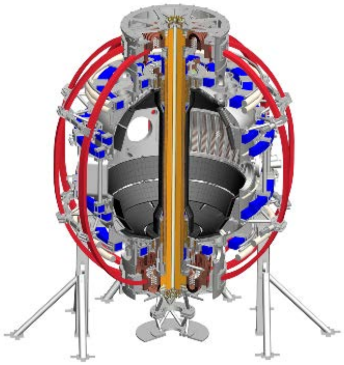
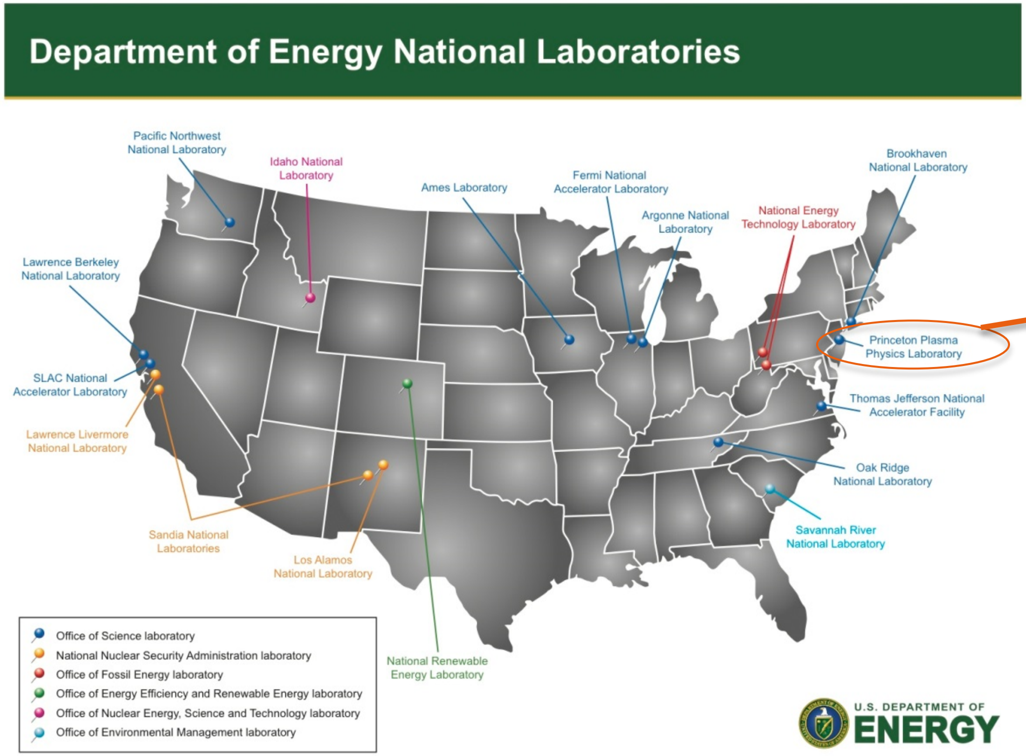


Outline

- Overview of PPPL, fusion, and plasma
- Plasma confinement issues
- What are tokamaks, stellarators, ITER?
- Motivation for studying spherical tokamaks
- NSTX Upgrade construction
- NSTX Upgrade scientific goals and questions
- Student and Faculty Opportunities at PPPL

PPPL: Princeton Plasma Physics Lab

- PPPL is one of 17 DoE national laboratories.
- We are managed by Princeton University but have a government mandate that focuses on fusion energy research and basic plasma science.

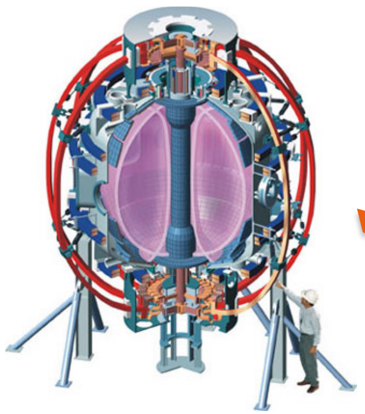


PPPL
PRINCETON PLASMA PHYSICS LABORATORY

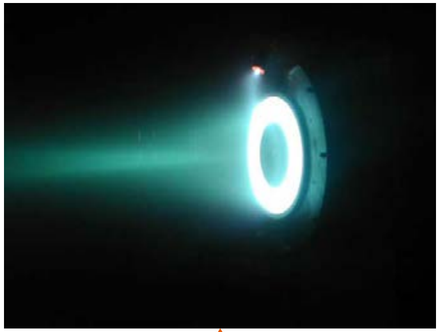
- ~550 employees
- $90-100M / year

At PPPL, we try to understand the many aspects of plasma physics

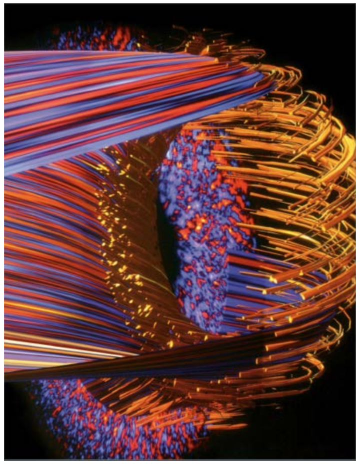
Fusion Research



Basic plasma physics

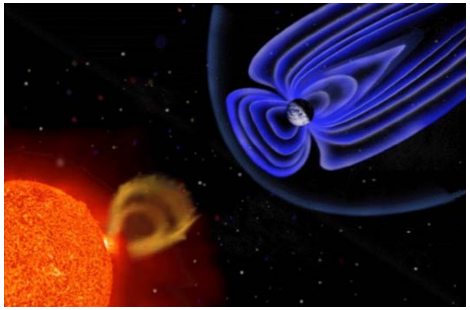


Plasma theory and simulation

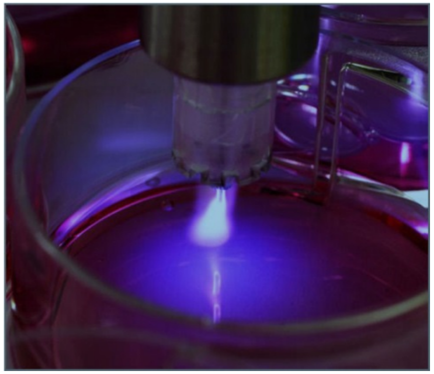


PPPL PRINCETON PLASMA PHYSICS LABORATORY

Astrophysical plasmas



Plasma in other fields (e.g. medicine)

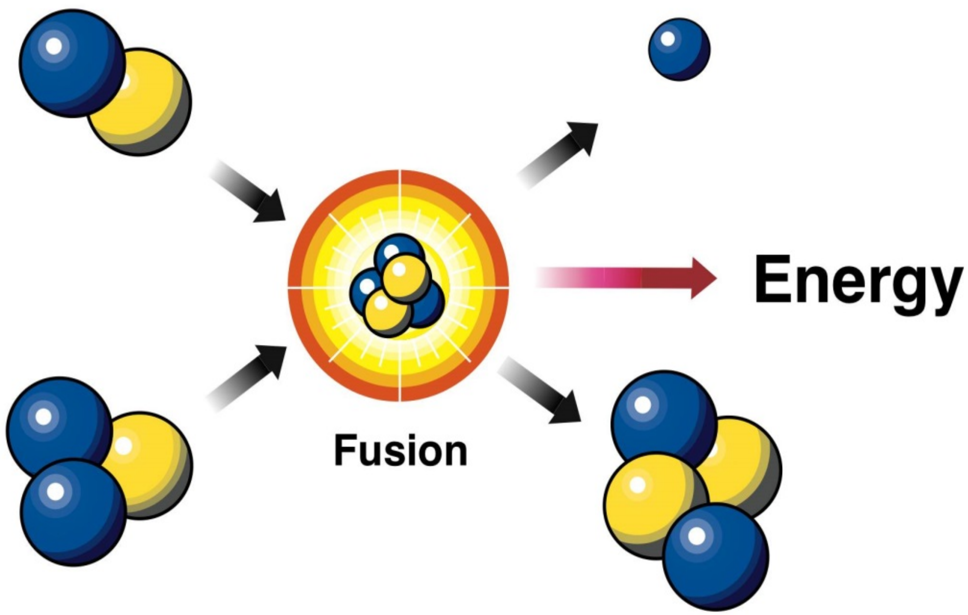


What is fusion?

“D-T” fusion reaction:

D: deuterium

n: neutron



T: tritium

α: helium

High energy gain
≈ 1000 ×

$E = mc^2$

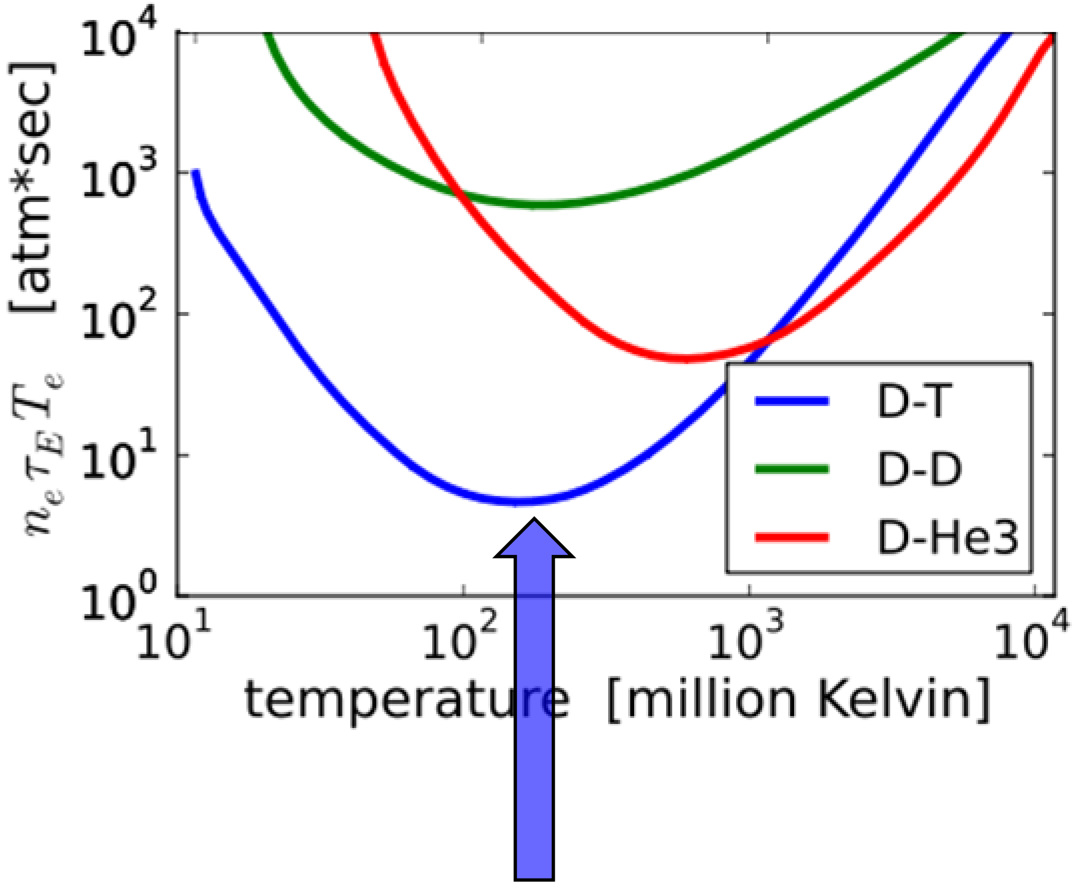
Advantages of fusion: safe, sustainable, high energy density, environmentally attractive

- **Cannot have runaway reaction**
  - **Only small amount of fuel present**
  - **If particles cool, fusion stops**
- **Abundant fuel supply**
  - **D from seawater: HDO, D/H = 1/6400**
  - **T bred from lithium in earth's crust**
- **High energy density**
  - **1 liter water = 500 liters gasoline**
- **Waste short-lived, low-level**
- **No CO_2 production**

Fusion requires very high temperatures

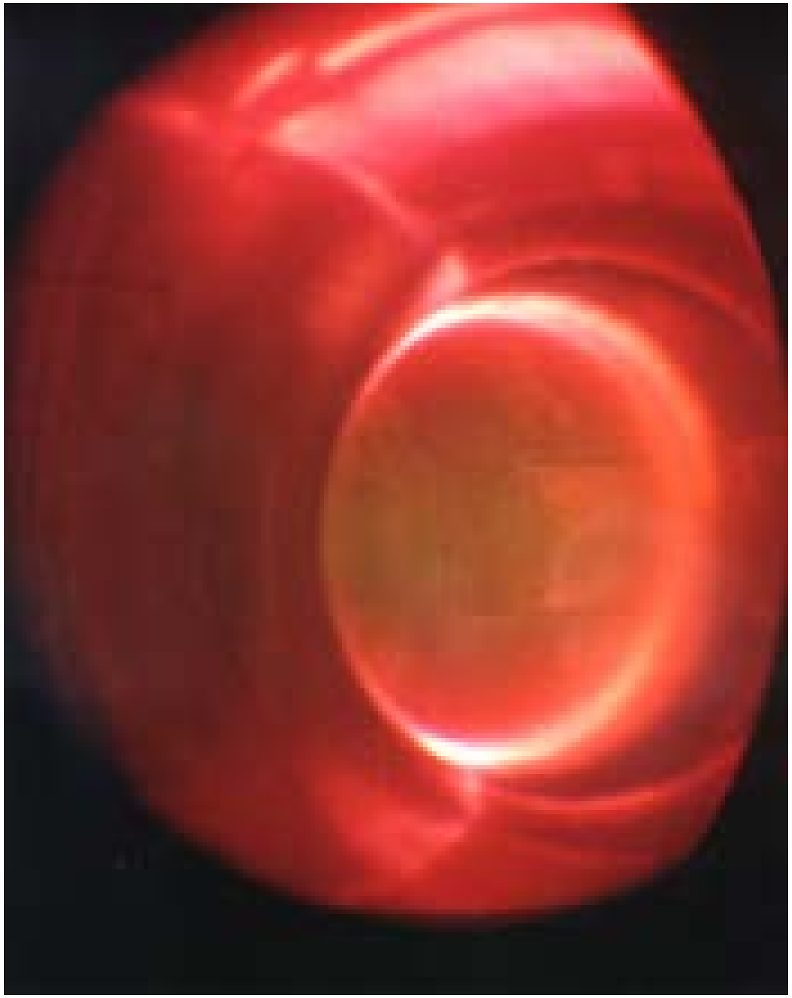
Fusion difficulty

(pressure × confinement)

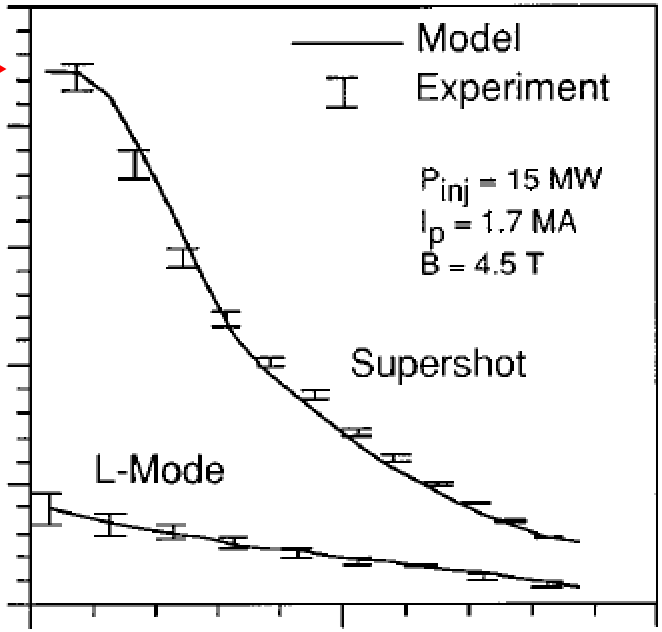


- Fusion is easiest here at 200 million °C (!!) (350 million °F)
  - Requires lowest pressure nT and energy confinement time τ_E
  - Minimum fusion "triple-product" value: 8 atmosphere-seconds

Magnetic fusion has already achieved the necessary very high temperatures!

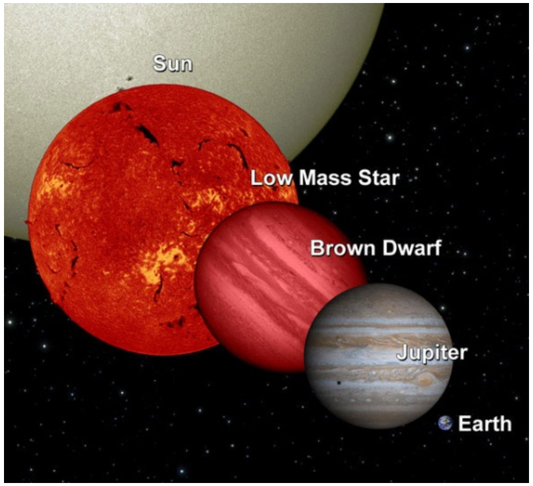


TFTR at PPPL (1990's)

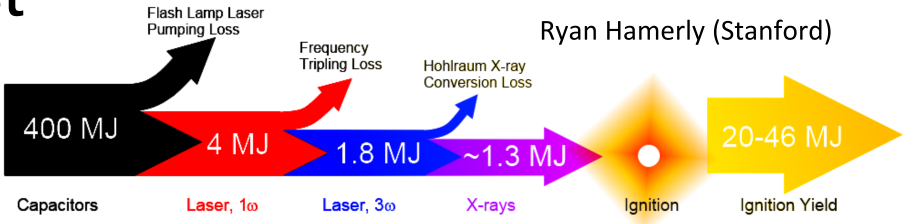


Magnetic fusion is arguably closest to ultimate goal of electricity generation

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



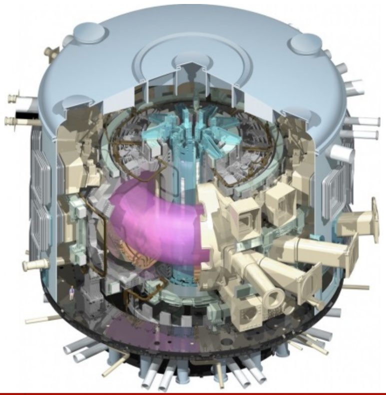
- Laser fusion ala NIF at best has $E_{fusion} / E_{electrical}$ ~ 5%
  - So far, 0.004% efficient



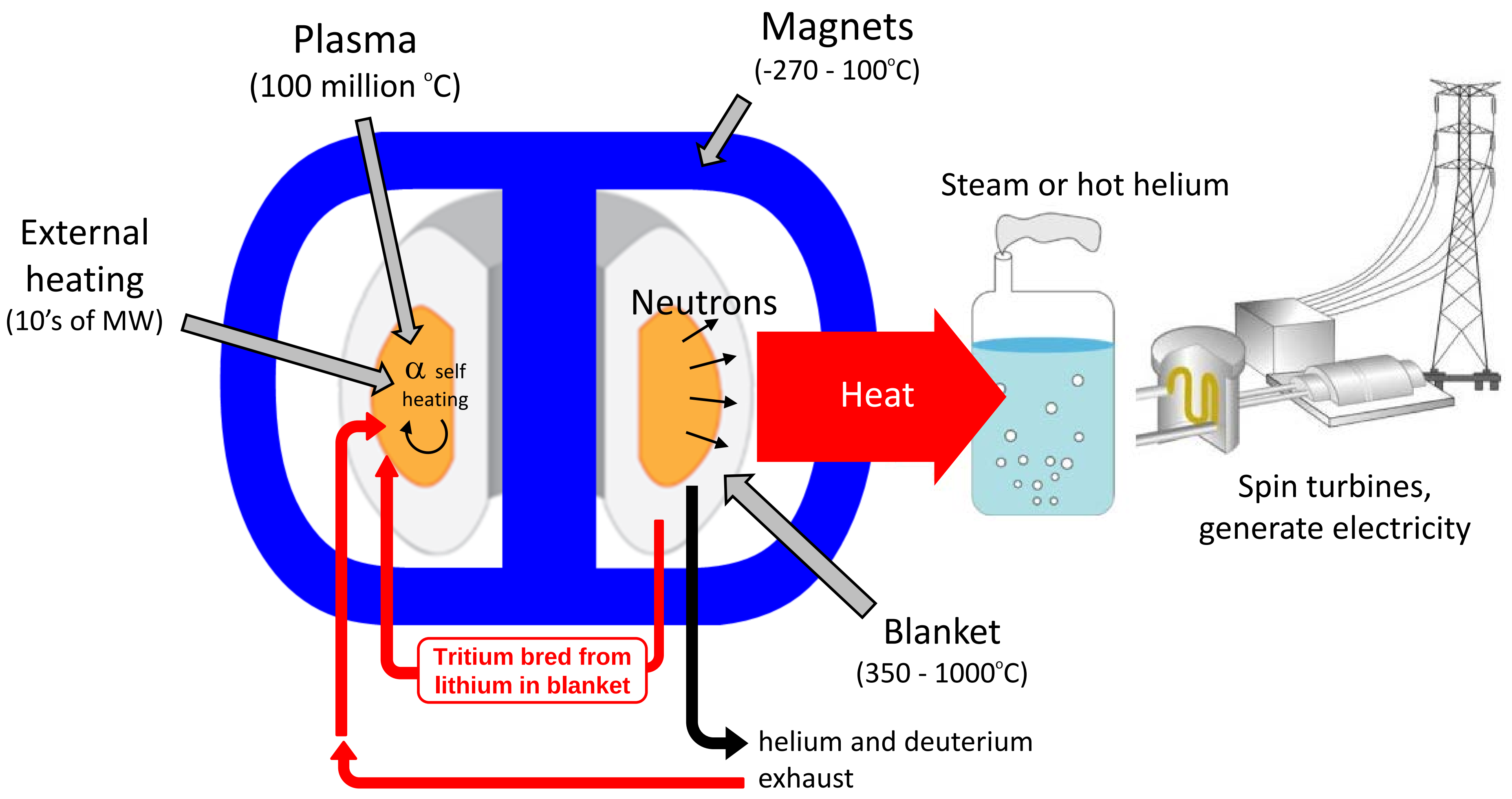
Ryan Hamerly (Stanford)

14kJ fusion yield achieved

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



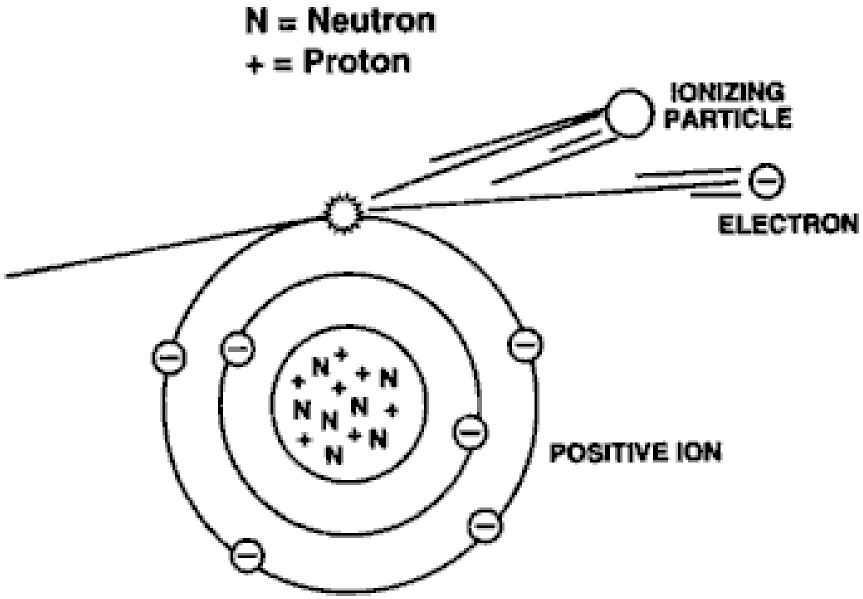
How would magnetic fusion make electricity?



How do we confine plasma?

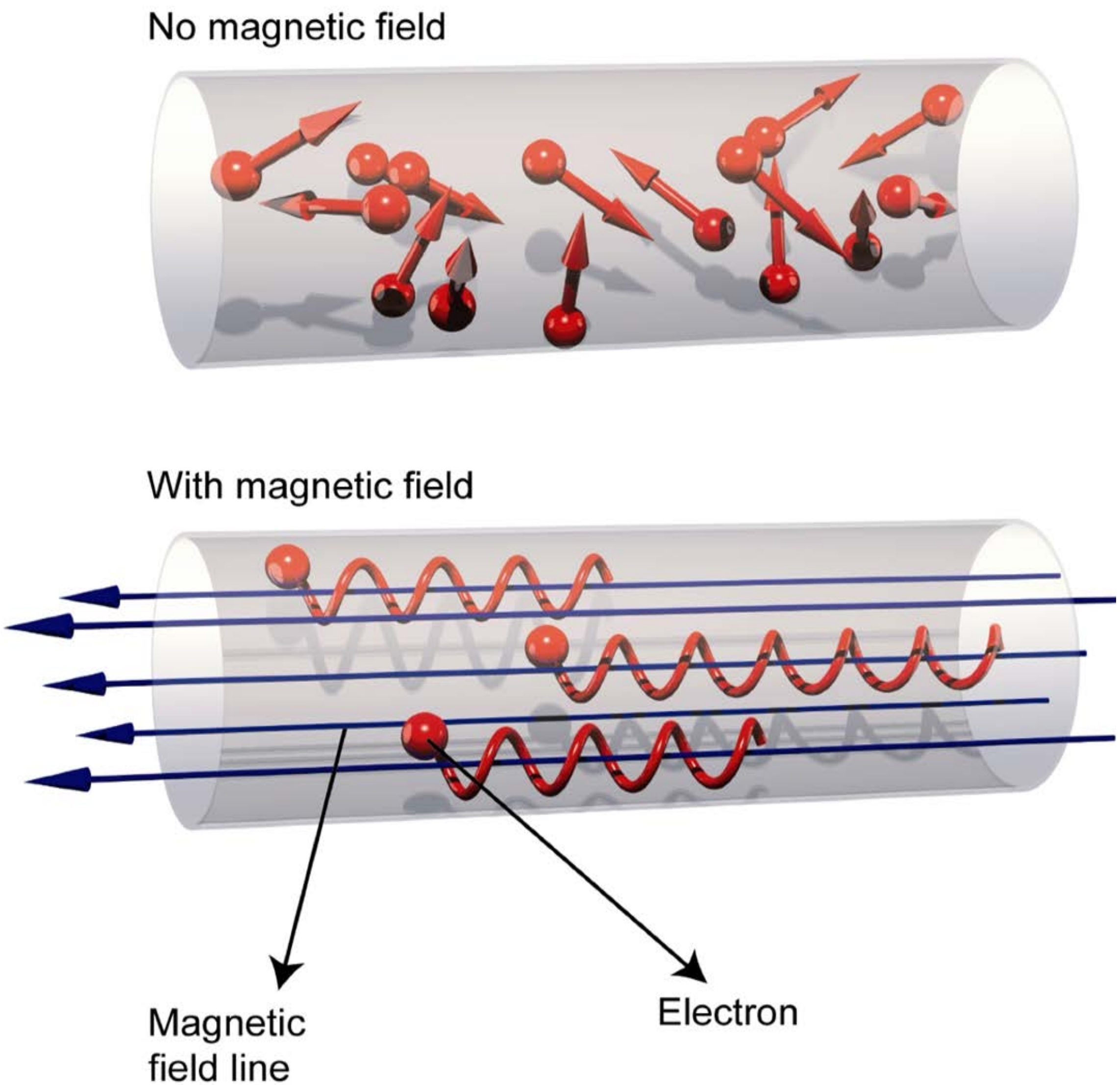
Plasma is a gas of charged particles: "Soup" of negatively charged electrons, positive ions

- At fusion temperatures, particles are so energetic that negatively charged (-) electrons are stripped from neutral atom leaving positively charged (+) ions



- One benefit of plasma state: charged particle motion can be manipulated by electric and magnetic fields

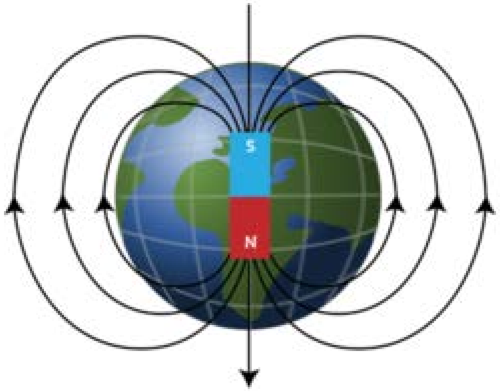
Charged particles confined by magnetic fields



- No magnetic field: Charged particles move freely in all directions
- Charged particles spiral around magnetic field "arrows", but move freely along the field
- Magnetic fusion goal: make field so particles never touch walls

Example magnetic fields in units of Tesla [T]

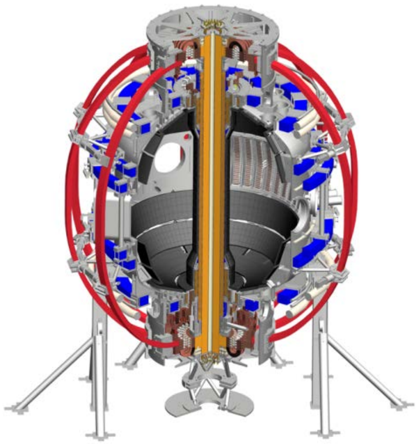
Earth: ~5 × 10^{-5} T



Refrigerator magnets

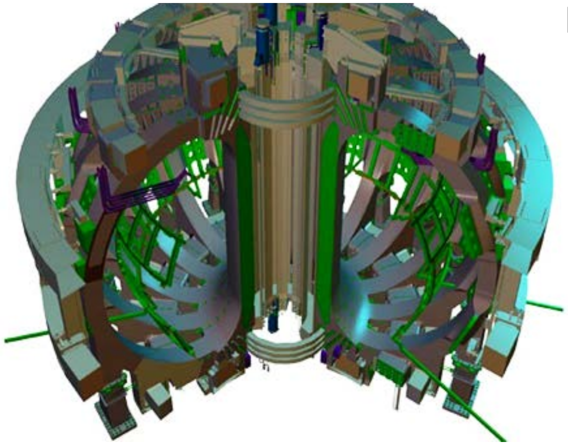
~1-5 × 10^{-3} T

PPPL's NSTX-U: 1 T



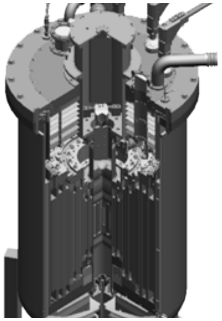
MRI: 0.5-3 T

ITER: 5.3 T



World Record: 100 T

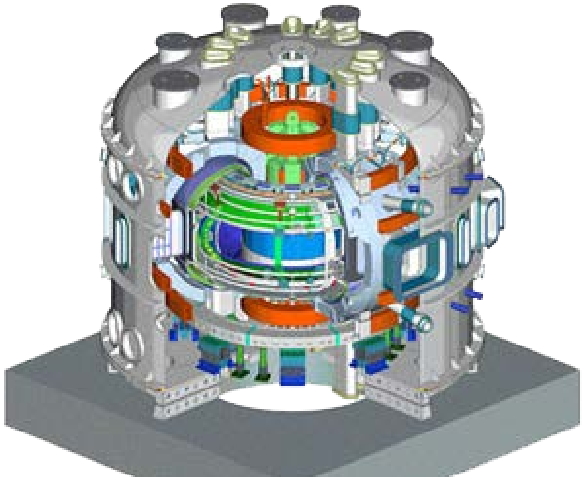
Non-destructive - for few milliseconds



National High Magnetic Field Laboratory

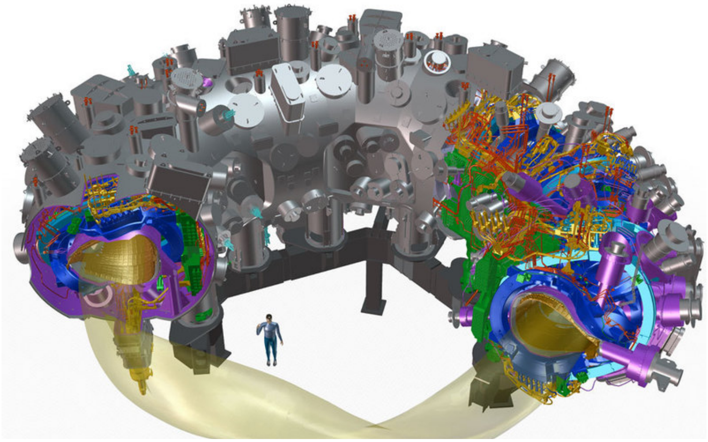
Tokamaks and stellarators are the leading configurations in magnetic fusion

Superconducting tokamak



KSTAR (South Korea)

Superconducting stellarator



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

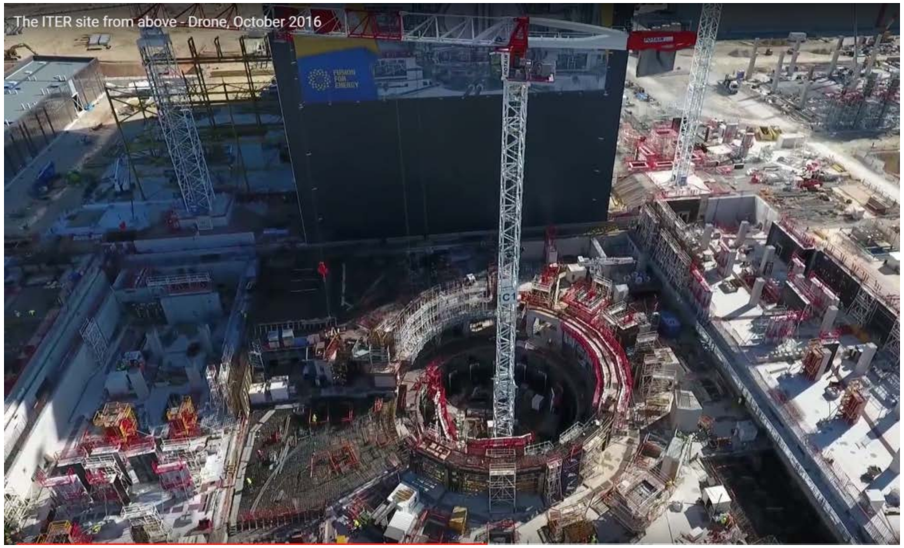
- Tokamak advantages:
  - Best confinement, closest to "breakeven"
  - Simpler planar coils and power/particle exhaust
- Disadvantages:
  - Must drive multi-mega-ampere plasma current
  - More prone to rapid loss of plasma = "disruption"

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

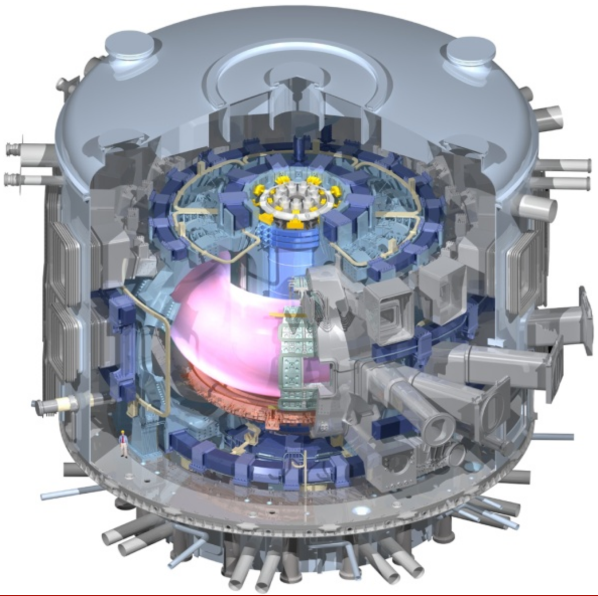
ITER will be first device to access "burning plasma"

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

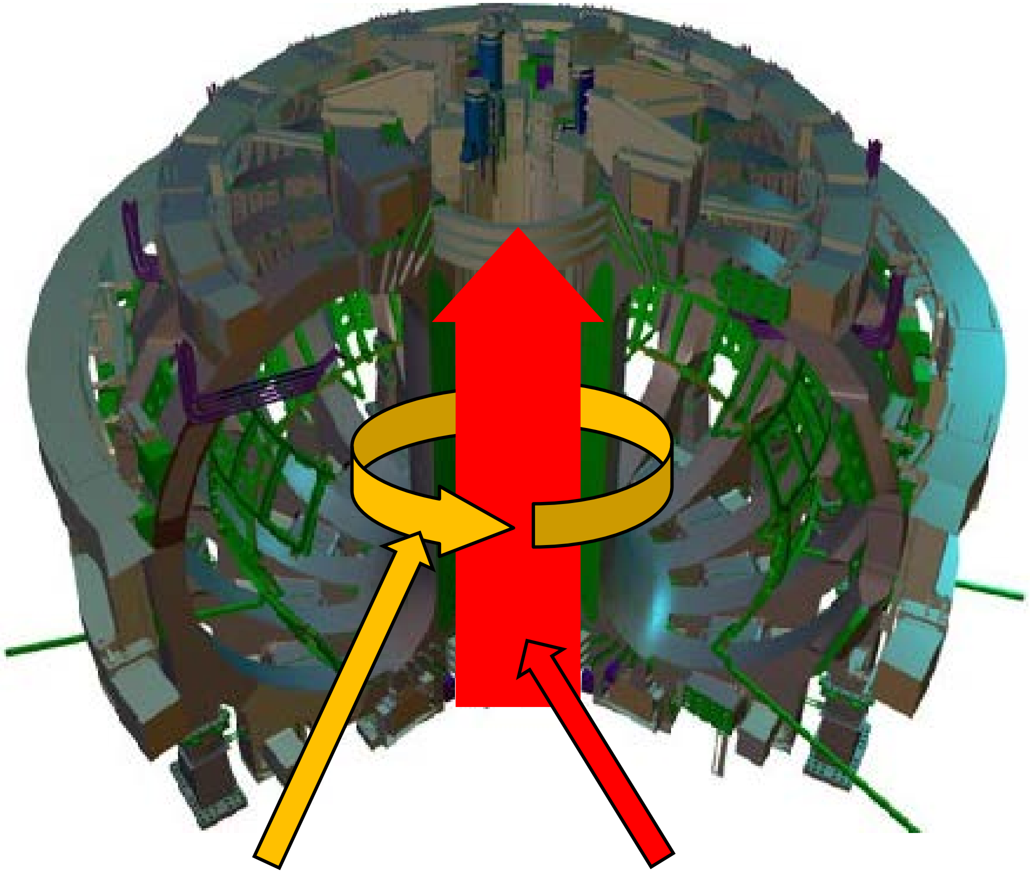
ITER under construction in Cadarache, France



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

Plasma current:
15 million amps

Toroidal field current
165 million amps

15 amps
(1800W)

Size of ITER driven largely by plasma confinement

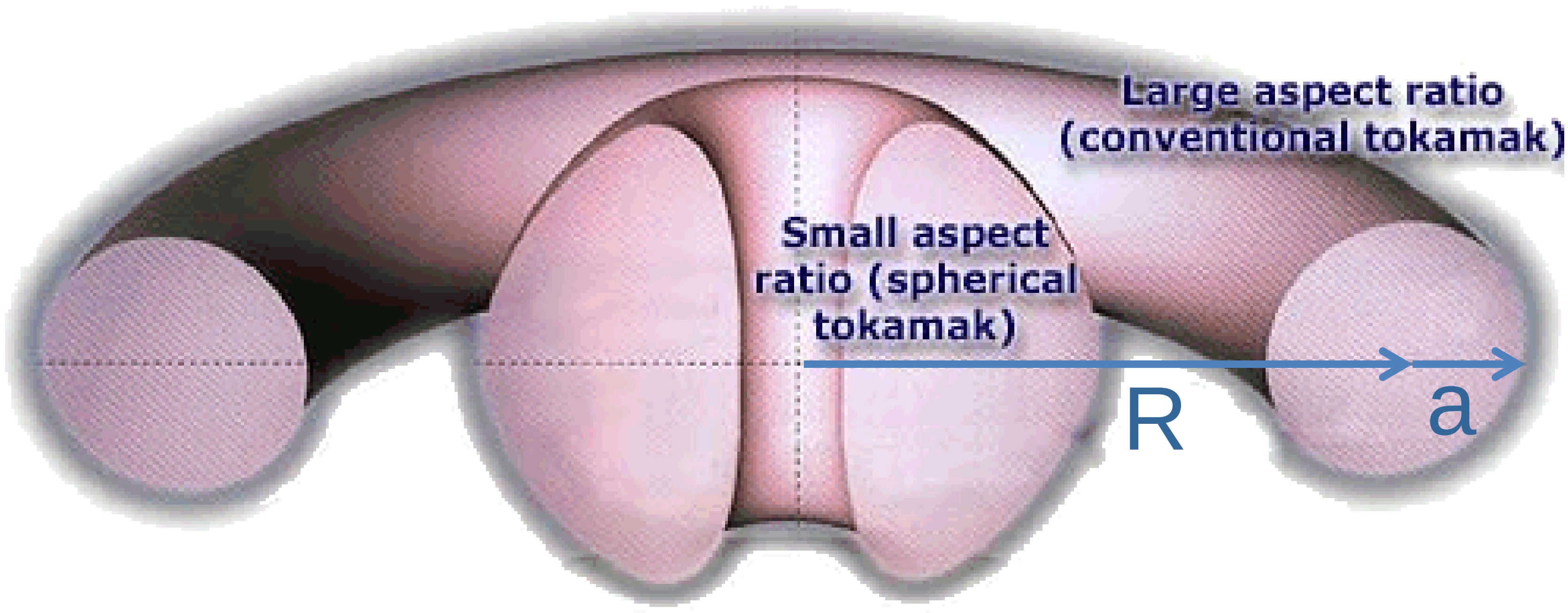
- Energy confinement scales with plasma current
- Large plasma current requires large toroidal field and/or plasma size for plasma to remain stable
- Current and confinement both scale with size
- **Can we make smaller devices with better confinement and smaller or cheaper magnets?**
- Such questions motivate exploring alternatives…

How might we possibly improve the conventional tokamak?

Aspect ratio is important free parameter

Aspect ratio A = R / a

R = major radius a = minor radius

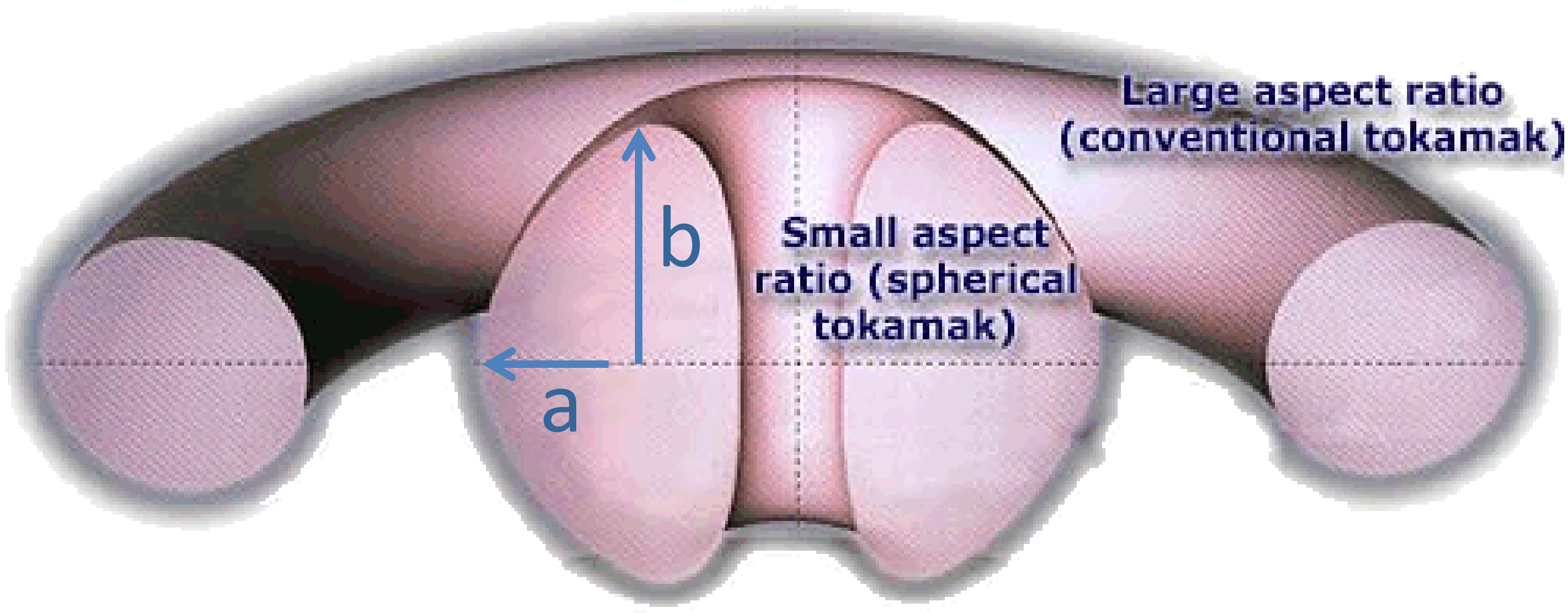


Spherical torus/tokamak (ST) has A = 1.1-2
Conventional tokamak typically A = 3-4

STs have higher natural elongation

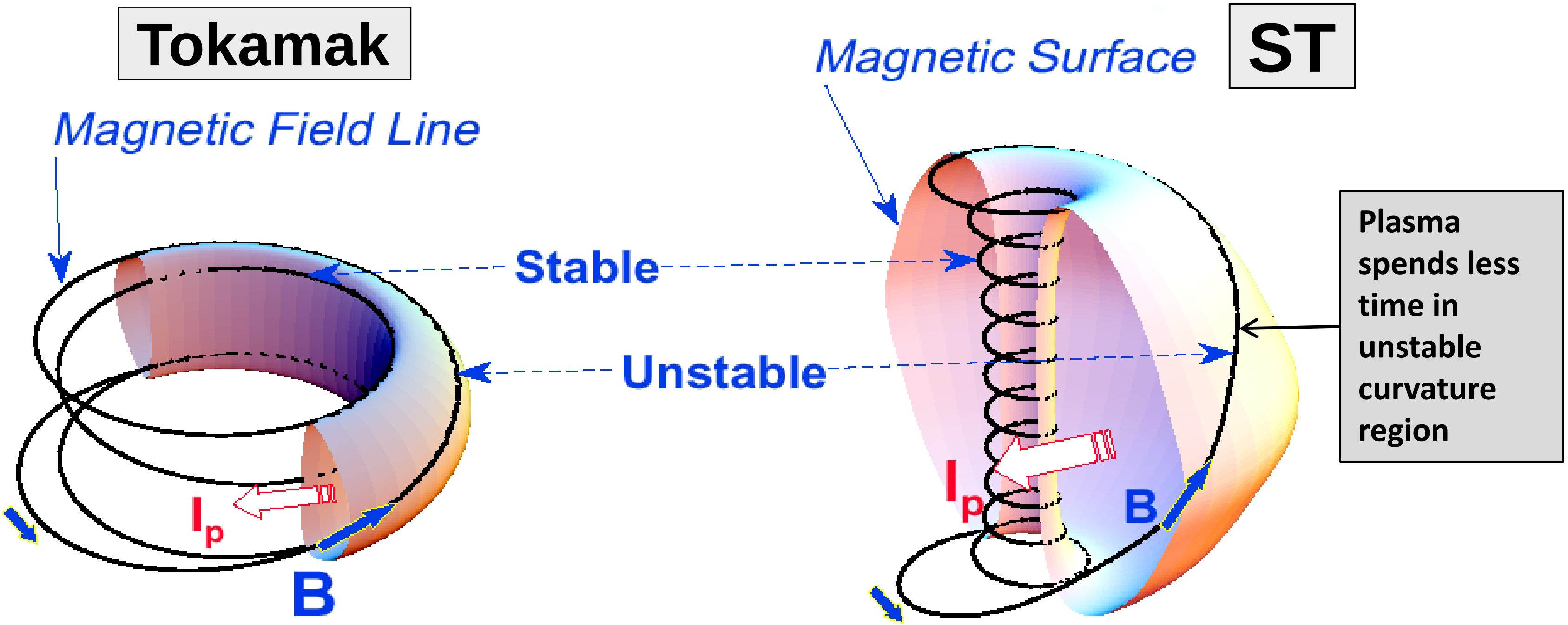
Elongation $\kappa = b / a$

b = vertical ½ height   a = minor radius



Higher elongation improves stability, confinement

Favorable average curvature improves stability



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

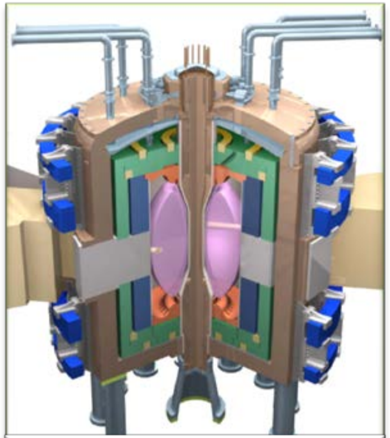
Aspect Ratio A = R /a	**Elongation κ = b/a**	**Toroidal beta $\beta_T = \langle p \rangle / (B_{T0}^2/2\mu_0)$**

Fusion technology development is major challenge

Fusion Nuclear Science Facility (FNSF) could aid development

Need to develop reliable and qualified nuclear and other components which are unique to fusion:

- Divertor, plasma facing components for exhaust
- Blanket and Integral First Wall
- Vacuum Vessel and Shield
- Tritium Fuel Cycle
- Remote Maintenance Components



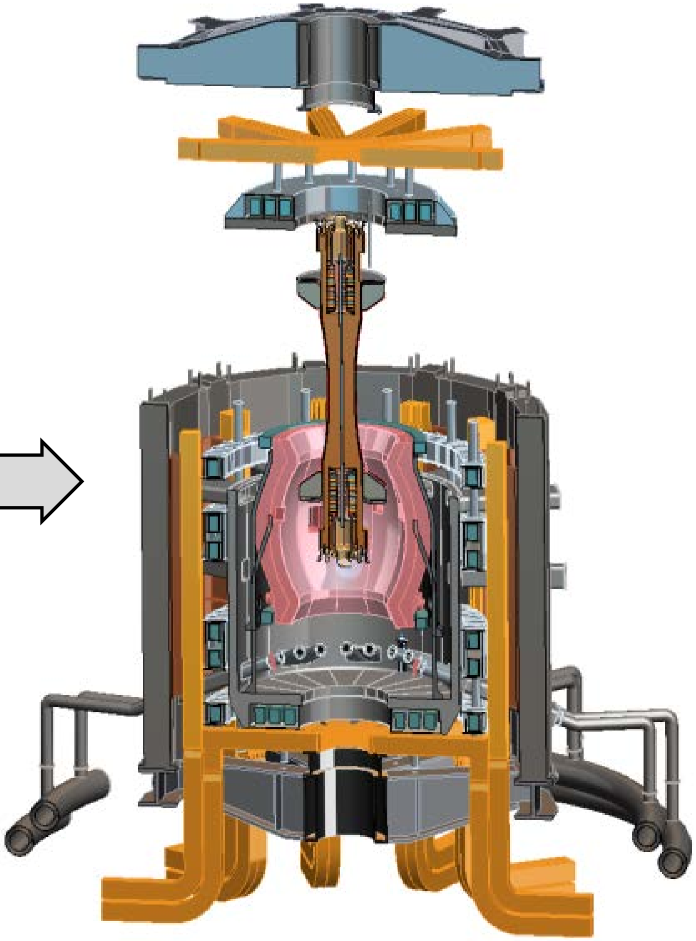
ST-FNSF

- 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, and reliable

Design studies show ST potentially attractive as FNSF

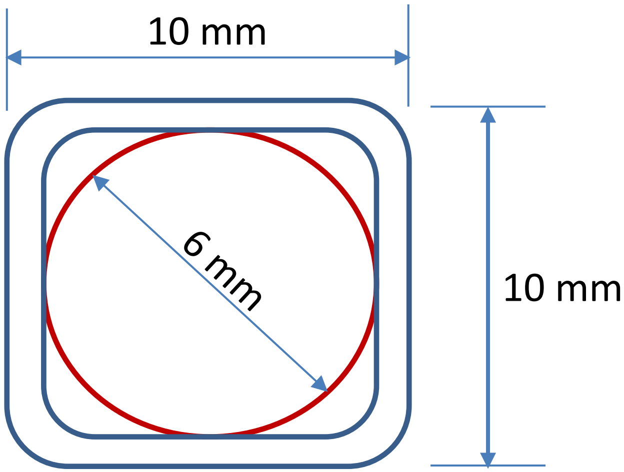
PPPL ST-FNSF concept

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



HTS cables using REBCO tapes achieving high winding pack current density at high B_T

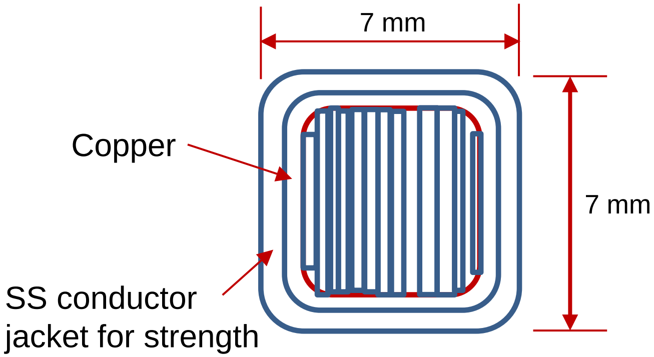
Conductor on Round Core Cables (CORC)
J_{WP} ~ 70MA/m^2 19T



7 kA CORC (4.2K, 19 T) cable

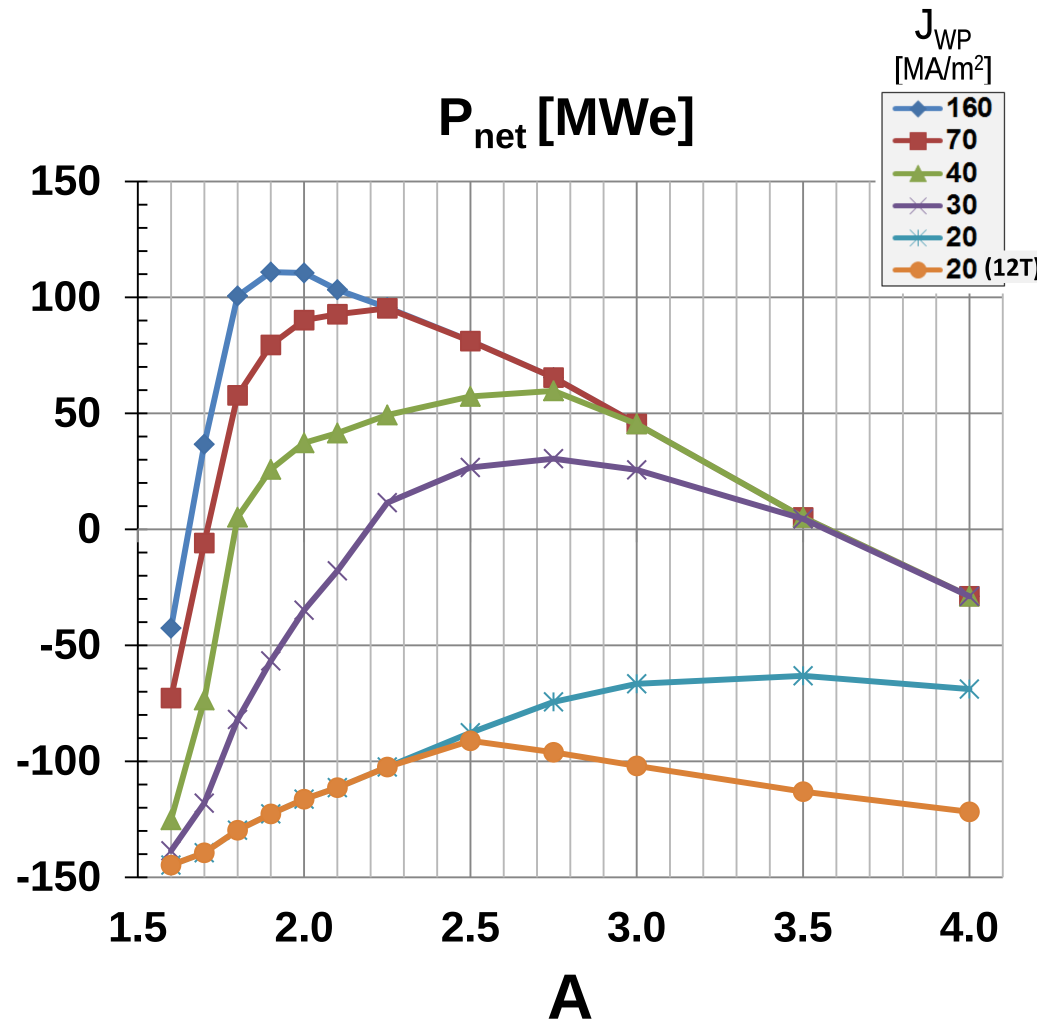
Base cable: 50 tapes YBCO Tapes with 38 mm substrate (Van Der Laan, HTS4Fusion, 2015)

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



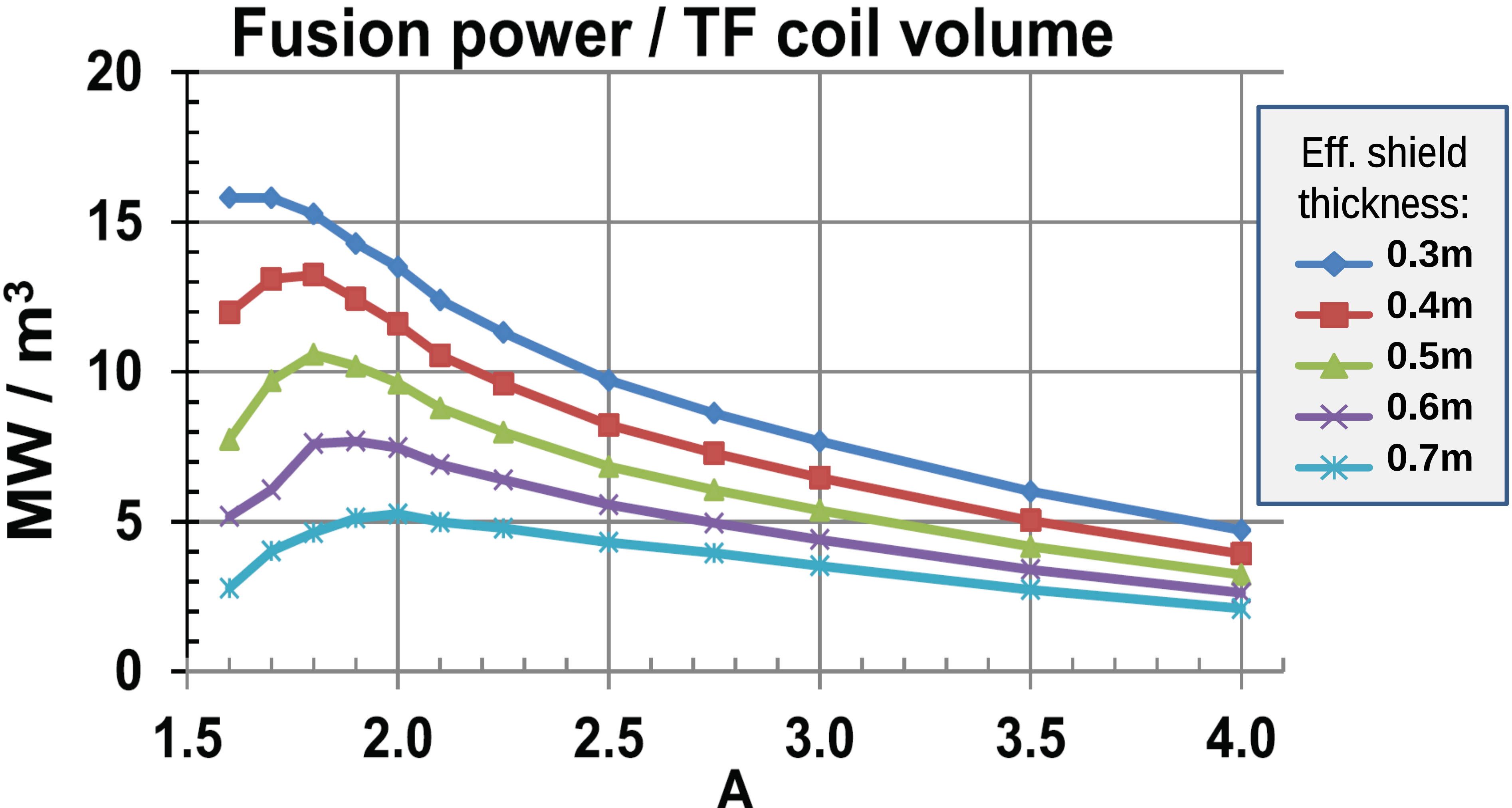
High current density HTS cable motivates consideration of low-A tokamak pilot plants

- 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**



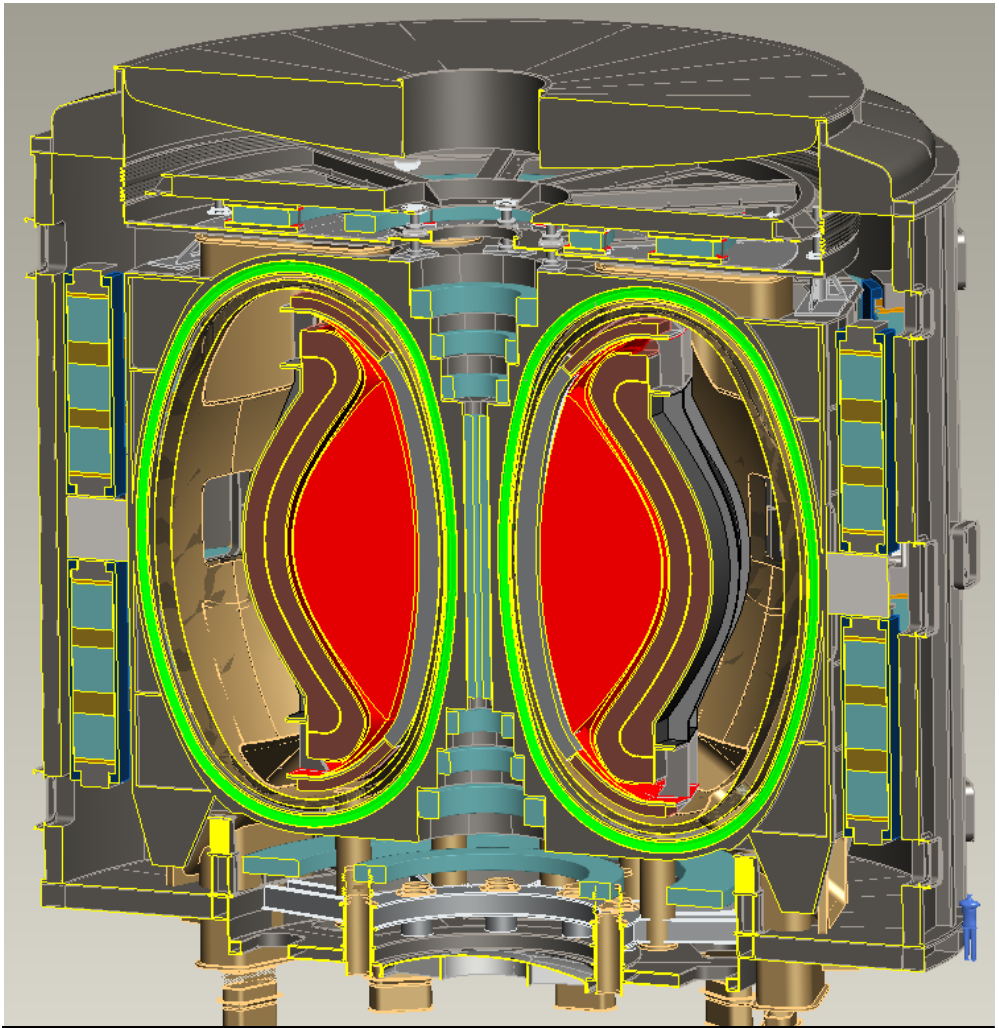
A ~ 2 attractive at high J_{WP}

A ≤ 2 maximizes TF magnet utilization



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

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-max} = 17.5T
No joints in TF
Vertical maintenance

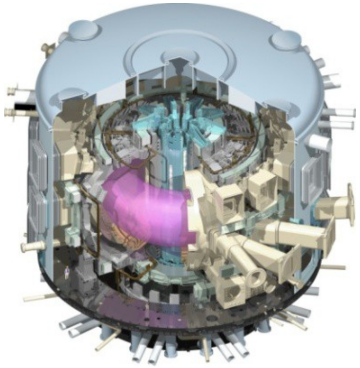
P_{fusion} = 520 MW
P_{NBI} = 50 MW, E_{NBI} = 0.5MeV
Q_{DT} = 10.4
Q_{eng} = 1.35
P_{net} = 73 MW

$\langle W_n \rangle$ = 1.3 MW/m^2
Peak n-flux = 2.4 MW/m^2
Peak n-fluence = 7 MWy/m^2

What are the goals of NSTX Upgrade (NSTX-U)?

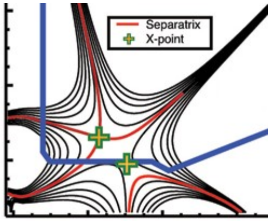
NSTX-U Mission Elements:

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



ITER

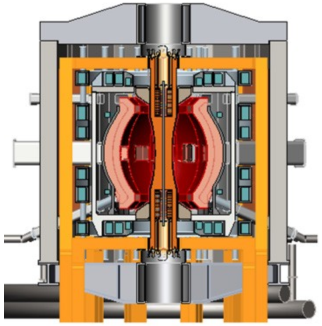
- Develop solutions for plasma-material interface (PMI)



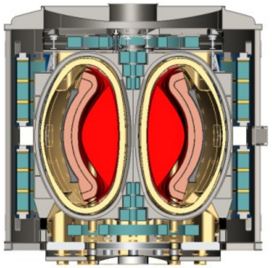
Snowflake/X

Liquid metals / Li

- Advance ST as Fusion Nuclear Science Facility and Pilot Plant

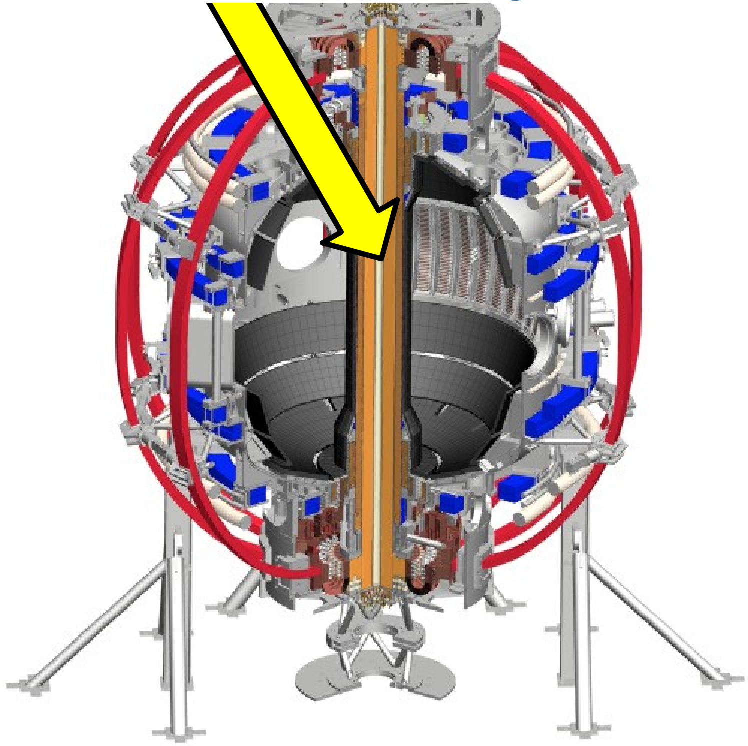


ST-FNSF / Pilot-Plant



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

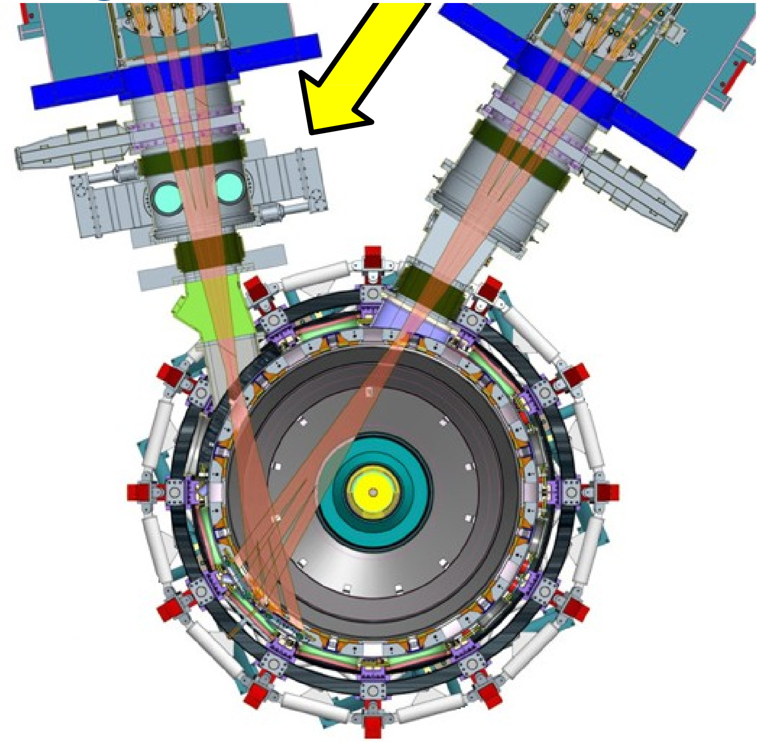
1. New Central Magnet



Higher T, low ν^* from low to high β

→ Unique regime, study new transport and stability physics

2. Tangential 2^{nd} Neutral Beam



Full non-inductive current drive

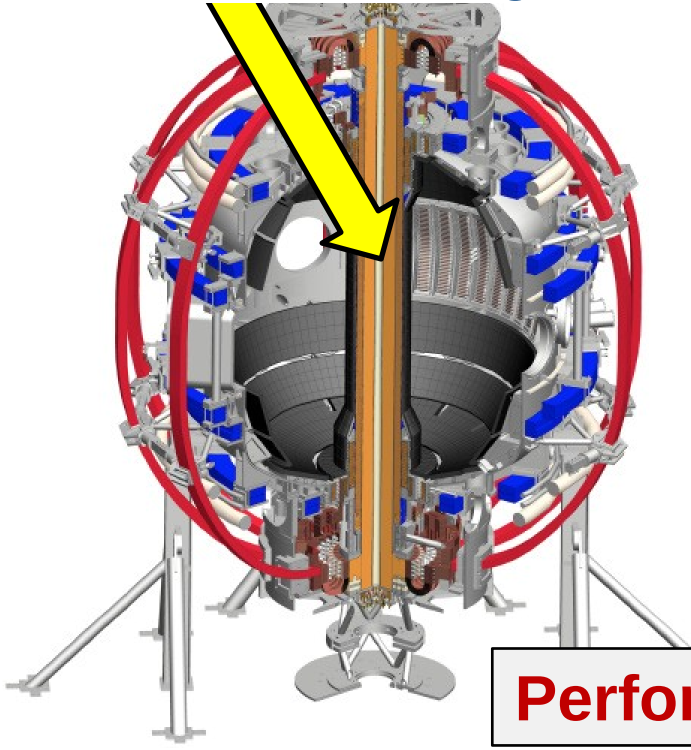
→ Not demonstrated in ST at high-β_T

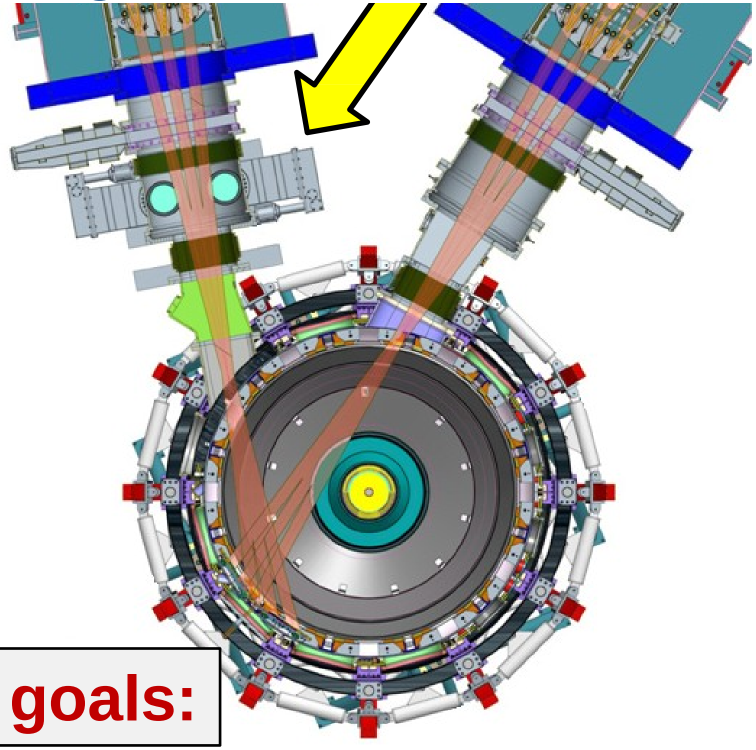
Essential for any future steady-state ST

NSTX-U will have major boost in performance

1. New Central Magnet

2. Tangential 2nd Neutral Beam





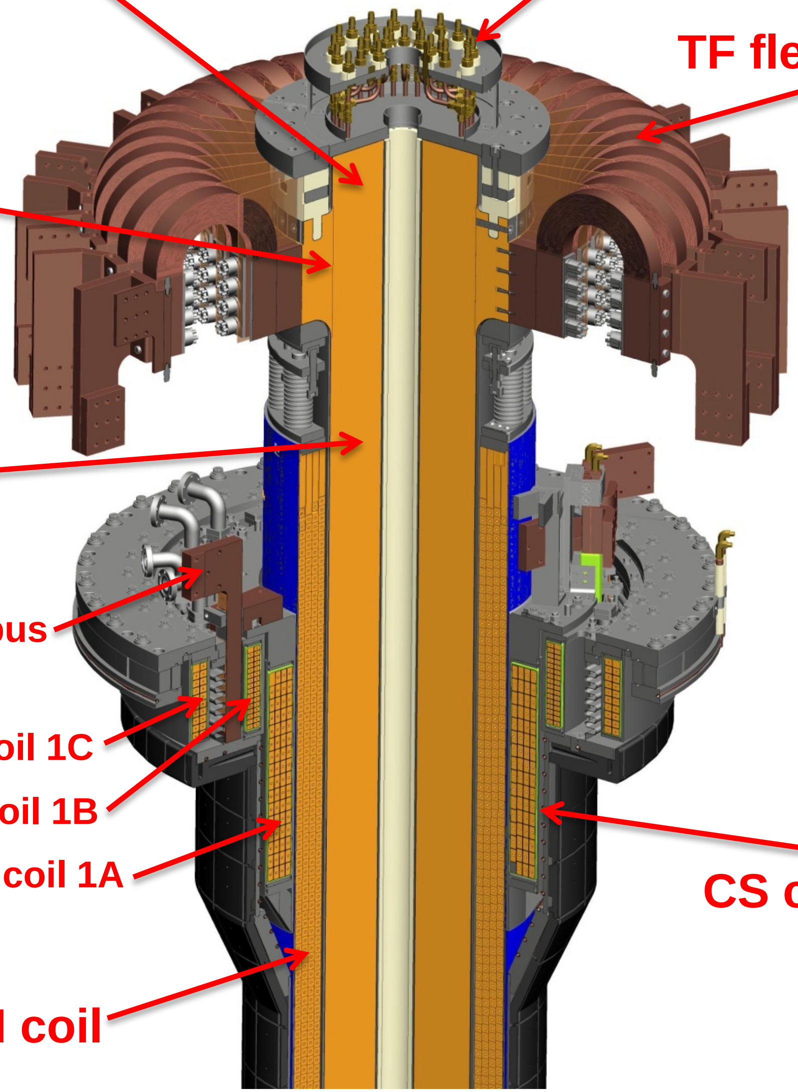
Performance goals:

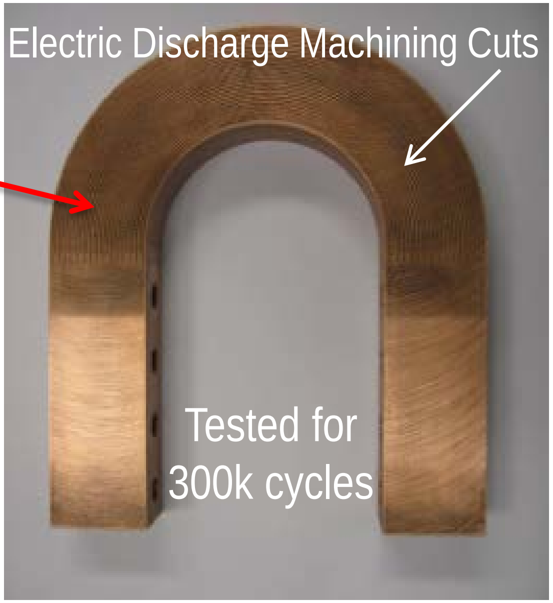
- ➢2× toroidal field (0.5 → 1T)
- ➢2× plasma current (1 → 2MA)
- ➢5× longer pulse (1 → 5s)
- ➢2× heating power (5 → 10MW)
  - Tangential NBI → 2× current drive efficiency
- ➢4× divertor heat flux (→ ITER levels)
- ➢Up to 10× higher $nT\tau_E$ (~MJ plasmas)

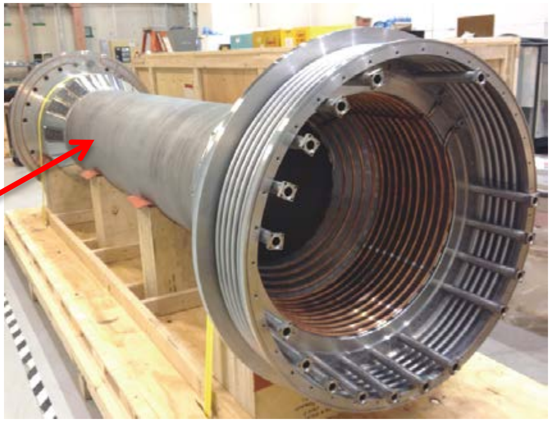
How was NSTX-U constructed?

New center-stack designed to handle increased forces

Identical 36 TF conductors and innovative flex-bus design

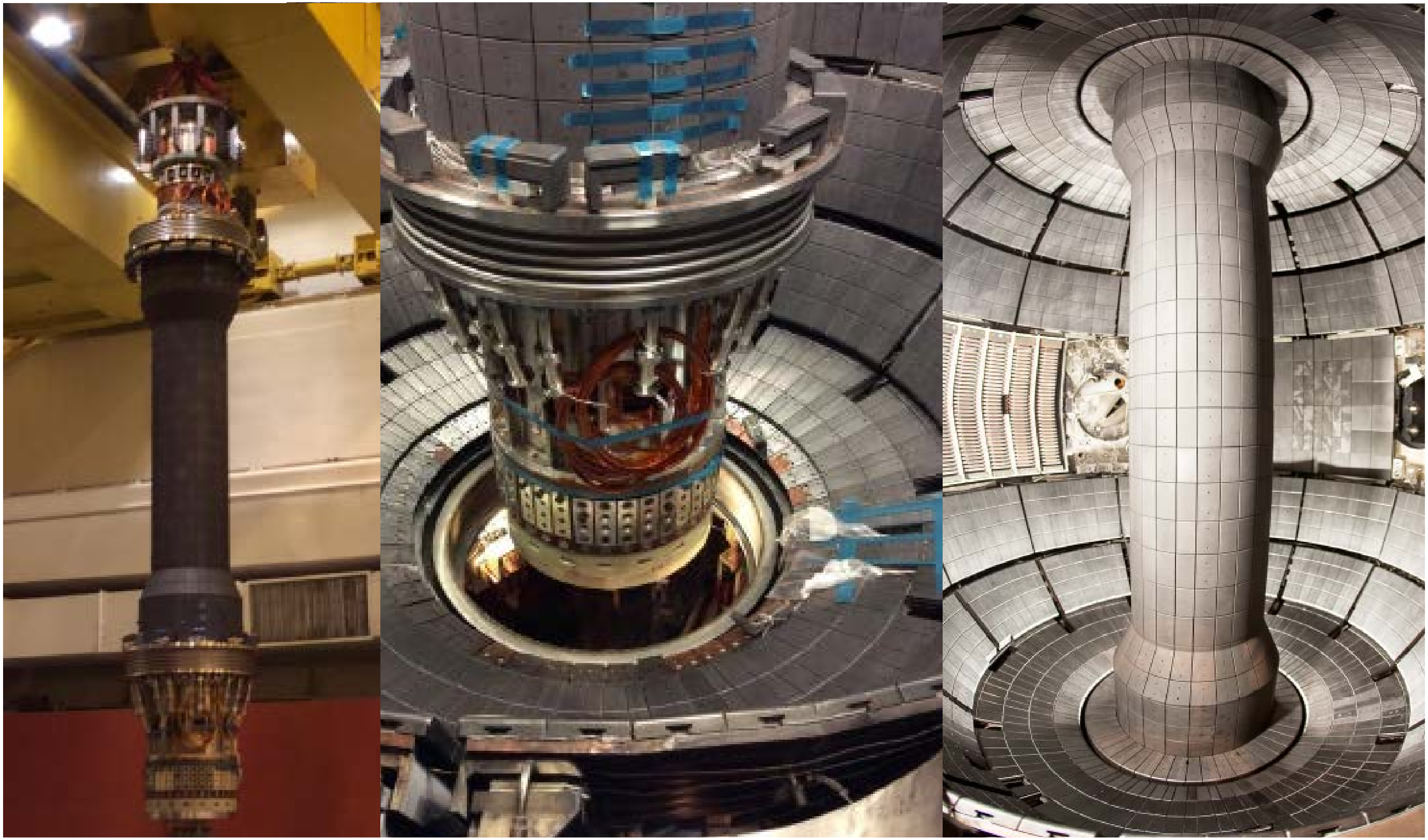






New Center-Stack installed in NSTX-U

Vacuum pump-down achieved in January, 2015



Relocated 2nd NBI beam line box from the TFTR test cell into the NSTX-U test cell

TFTR NBI beam box and components successfully tritium decontaminated

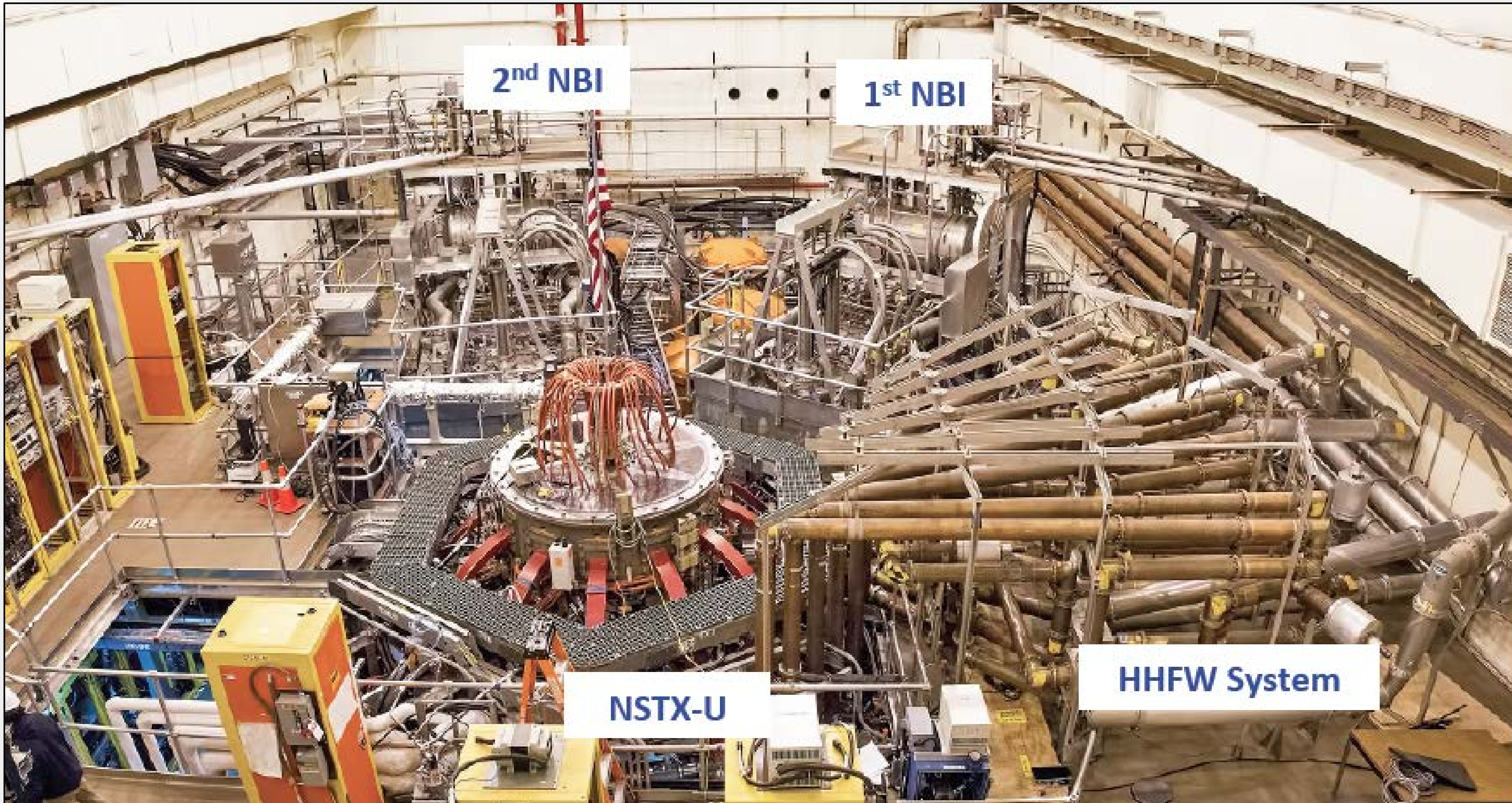
Beam Box being lifted over NSTX

Beam Box placed in its final location and aligned

Beam Box being populated with components

NSTX Upgrade Project Completed September 2015

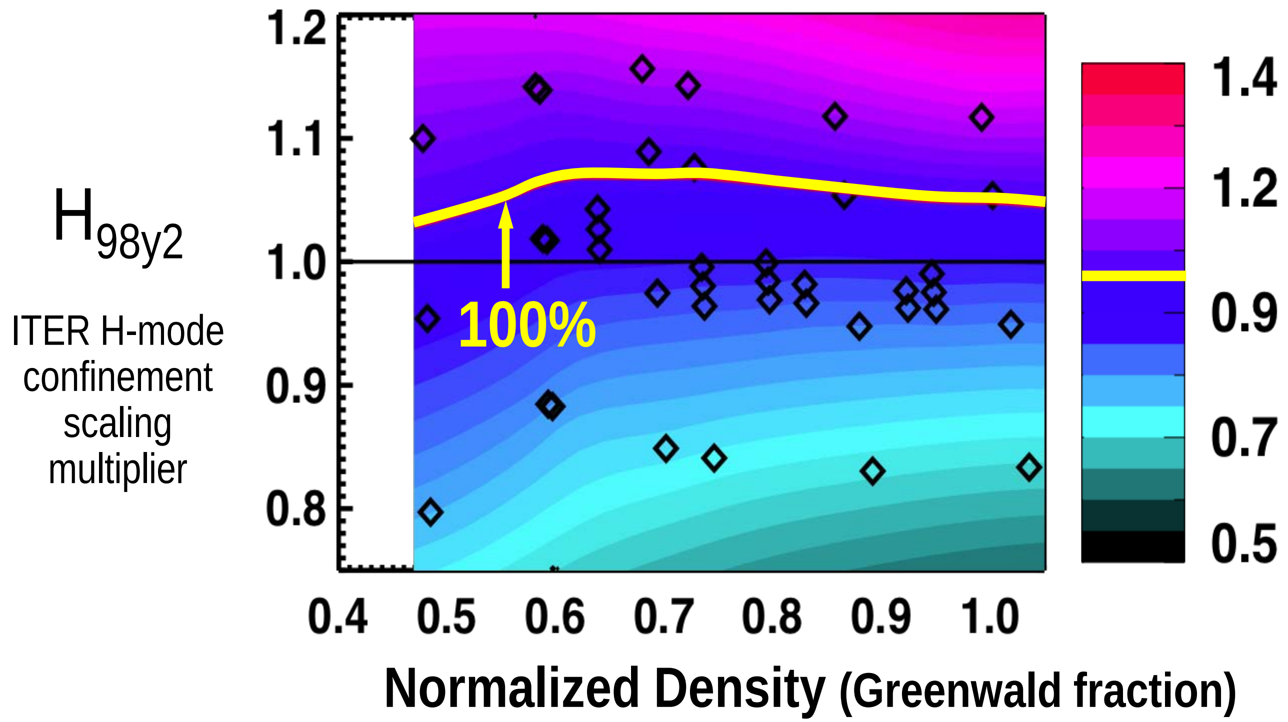
Test plasmas August 2015, Research plasmas December 2015



What key science questions will NSTX-U address?

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

TRANSP Contours of Non-Inductive Fraction

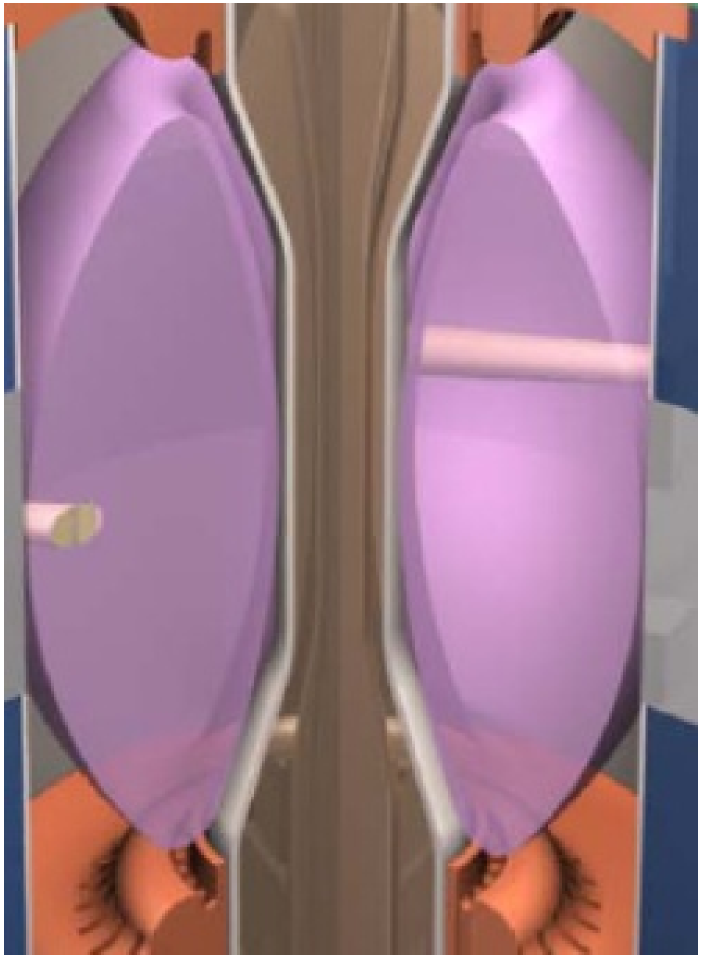


I_P=1 MA, B_T=1.0 T, P_{NBI}=12.6 MW

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

I_P Start-up/Ramp-up Critical Issue for ST-FNSF

Compact ST-FNSF has no/small central solenoid



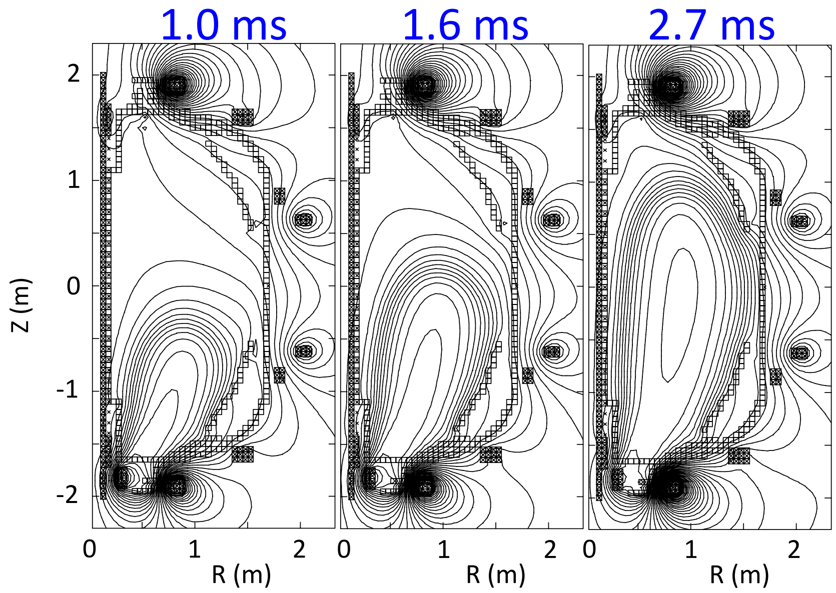
~ 1-2 MA of transformer-free start-up current needed for FNSF → 10-20% of total current

- Long-term major goal of NSTX-U: generate and sustain a high-performance plasma without using any transformer
  (this will not be easy…)

NSTX achieved 200kA (~20%) "transformer-less" start-up

Will NSTX-U achieve 400kA or more as per simulations?

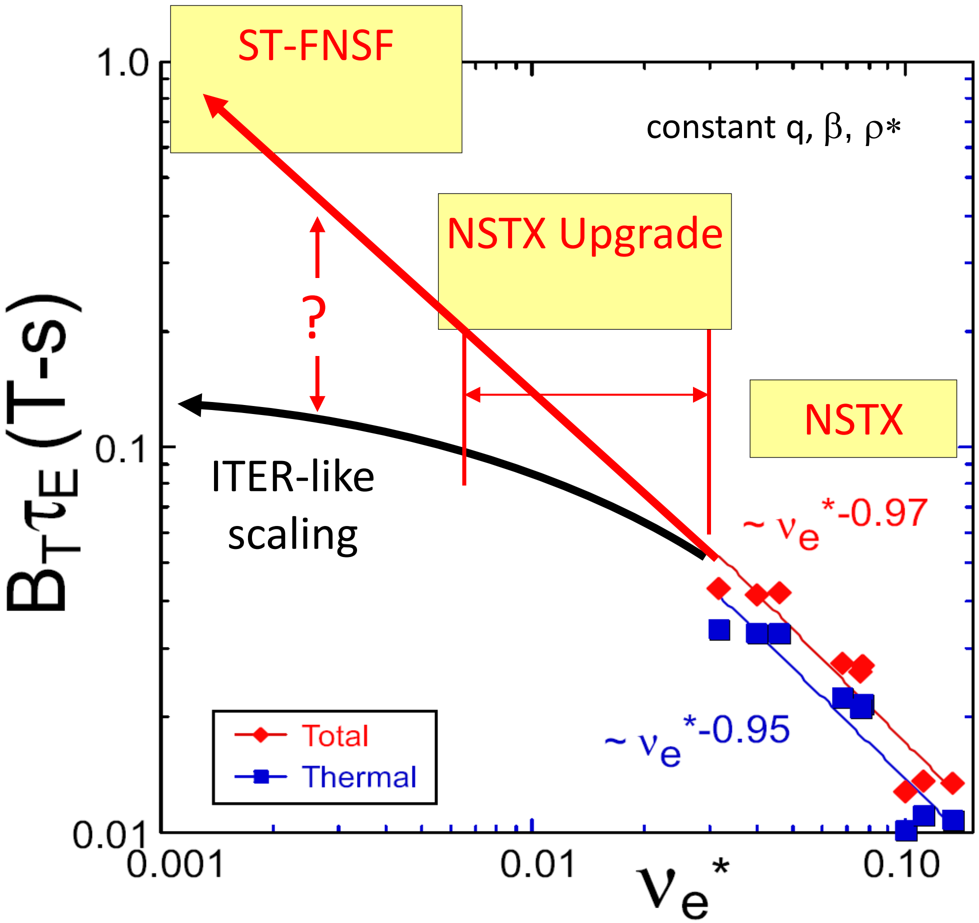
- TSC code (2D) successfully simulated helicity injection I_P ~200kA in NSTX



Additional electron heating likely required
Design of heating system is underway…

NSTX / MAST confinement increased at higher T_e (!)

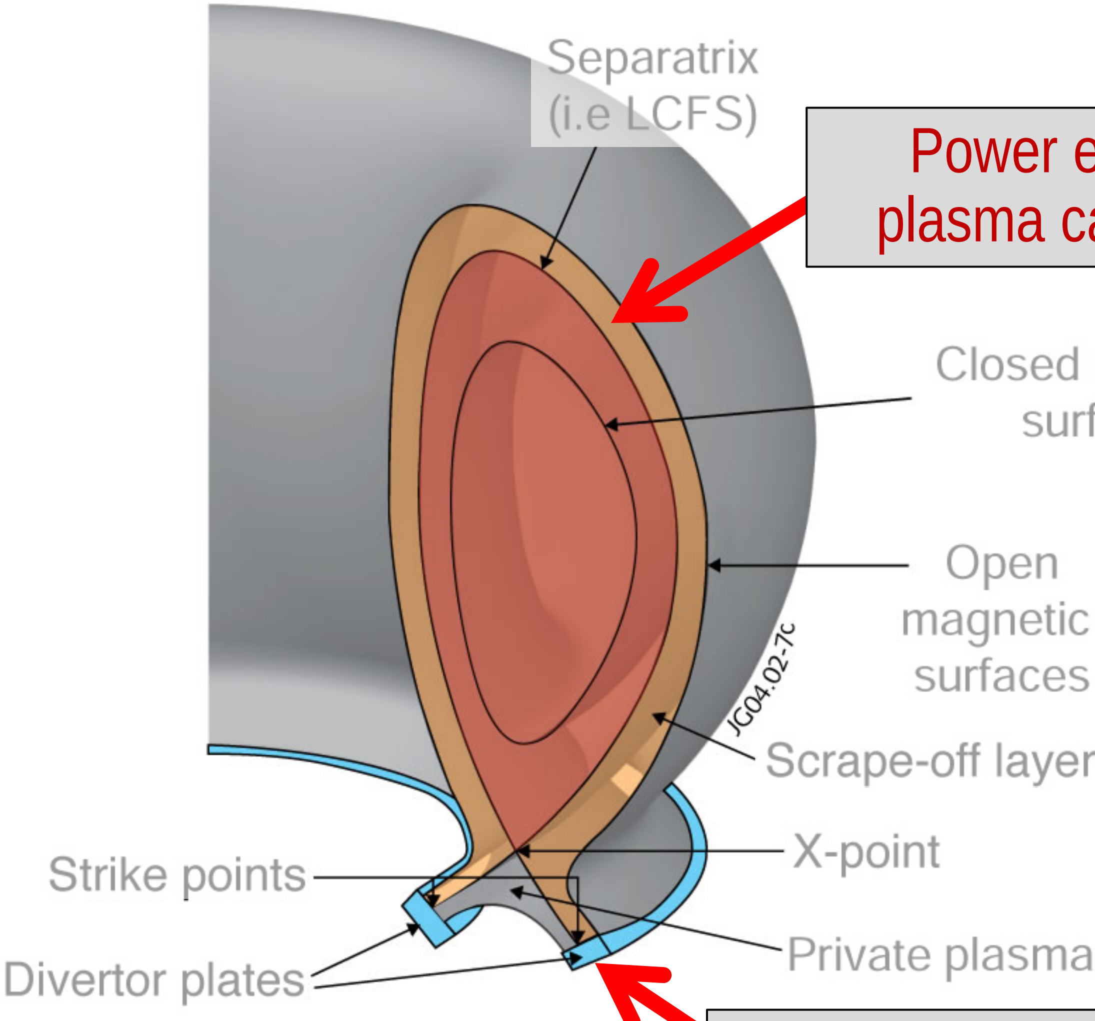
Will confinement trend continue, or look like conventional A?



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

Favorable confinement results could lead to more compact ST reactors

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



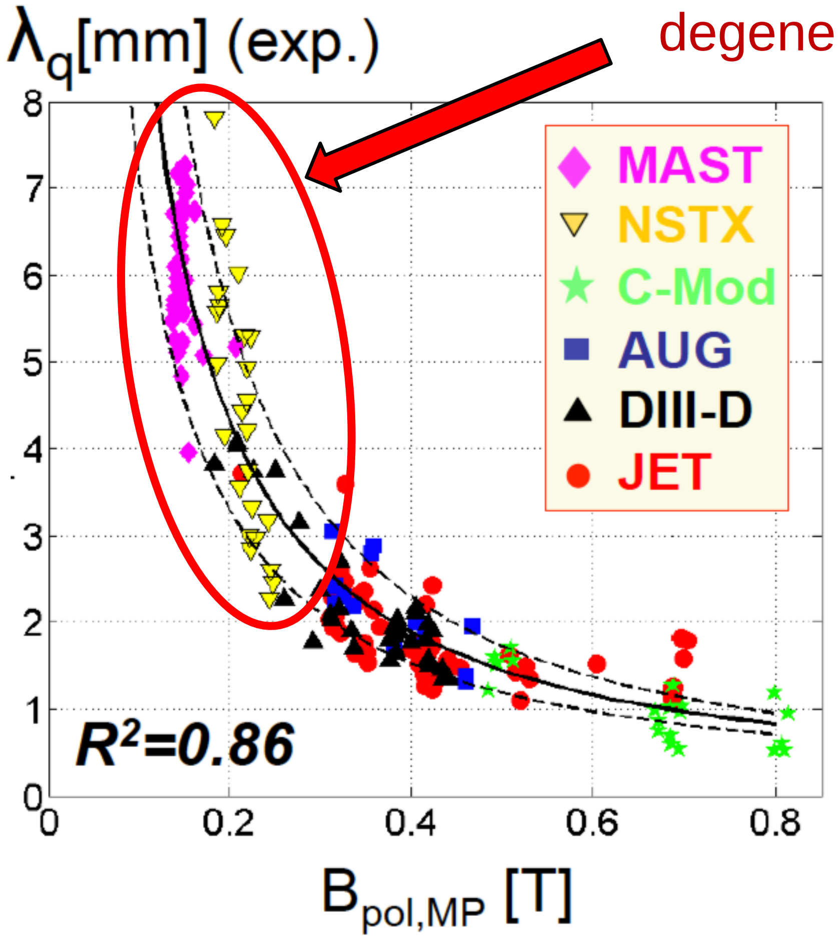
Power exhaust width outside main plasma can be very narrow (few mm)

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

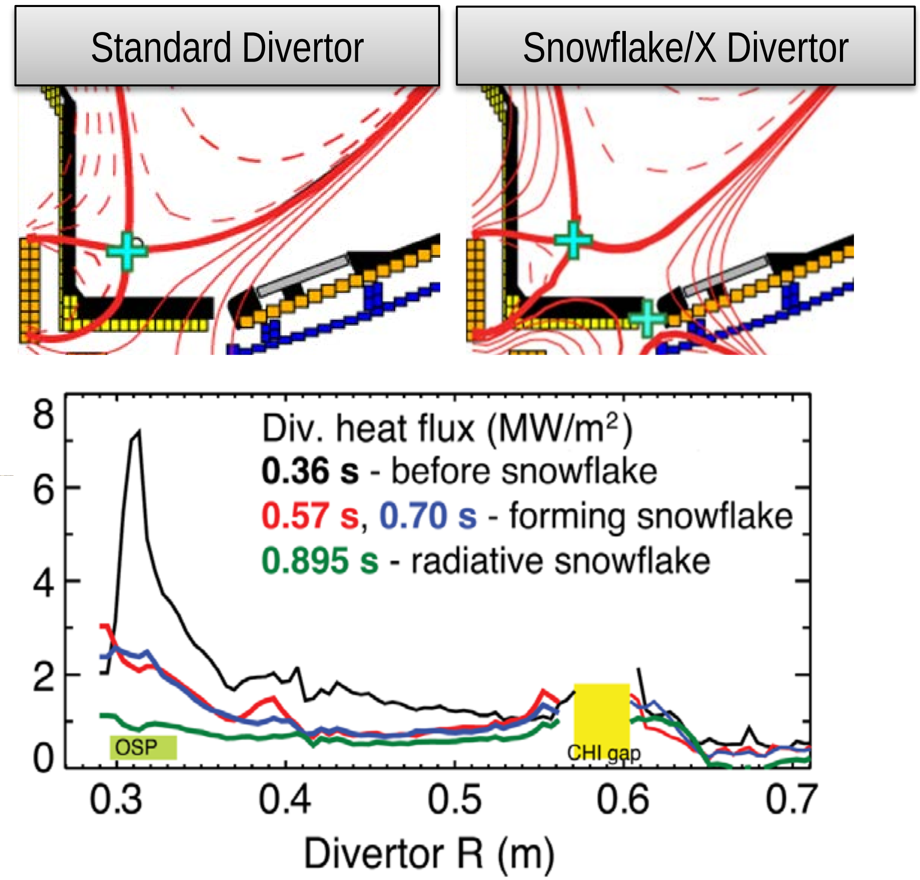


Issue: If peak heat flux in divertor region exceeds 10 MW/m^2 → material damage

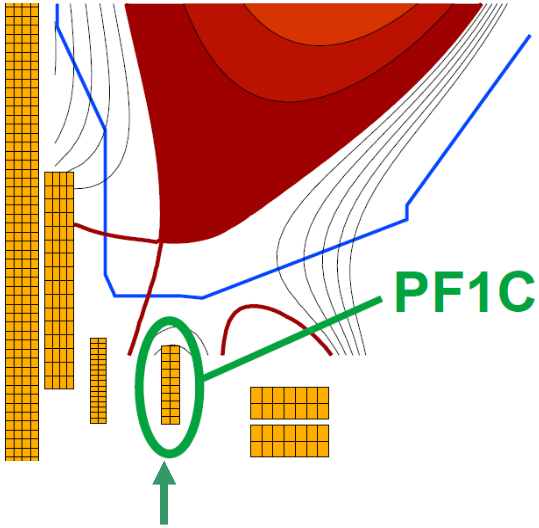
Wider heat-flux width may offset smaller R → maybe better than tokamak

NSTX-U will test ability of radiation and advanced divertors to mitigate very high heat-fluxes

- NSTX: reduced heat flux 2-4 × via radiation (partial detachment)
- Additional null-point in divertor expands field, reduces heat flux



NSTX-U peak heat fluxes will be up to 4-8× higher than in NSTX



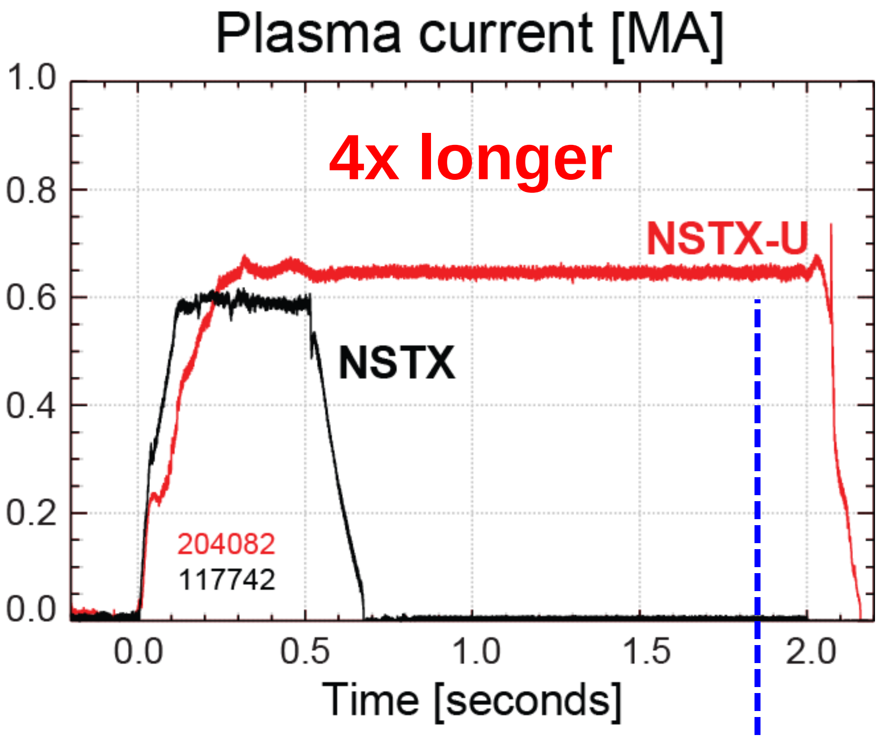
NSTX-U has additional coils for up-down symmetric snowflake/X, improved control

NSTX-U had scientifically productive 1st year

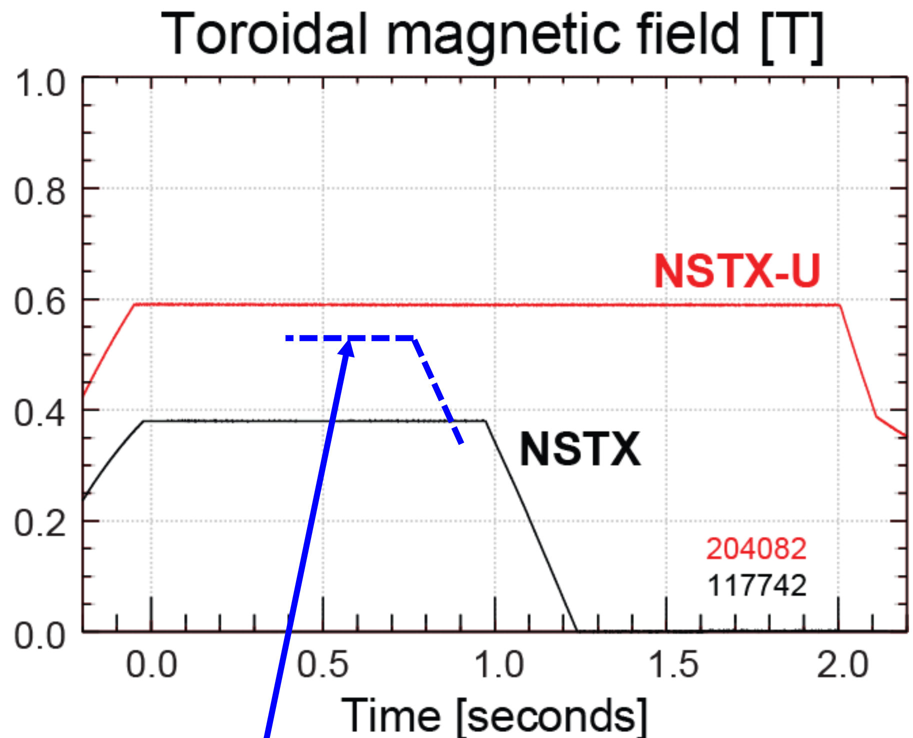
- Achieved H-mode on 8th day of 10 weeks of operation
- Surpassed magnetic field and pulse-duration of NSTX
- Matched best NSTX H-mode performance at ~1MA
- Identified and corrected dominant error fields
- Commissioned all magnetic and kinetic profile diagnostics
- New 2nd NBI suppresses Global Alfven Eigenmodes (GAE)
- 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, design (re)-reviews, preparing for new coil fabrication

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

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



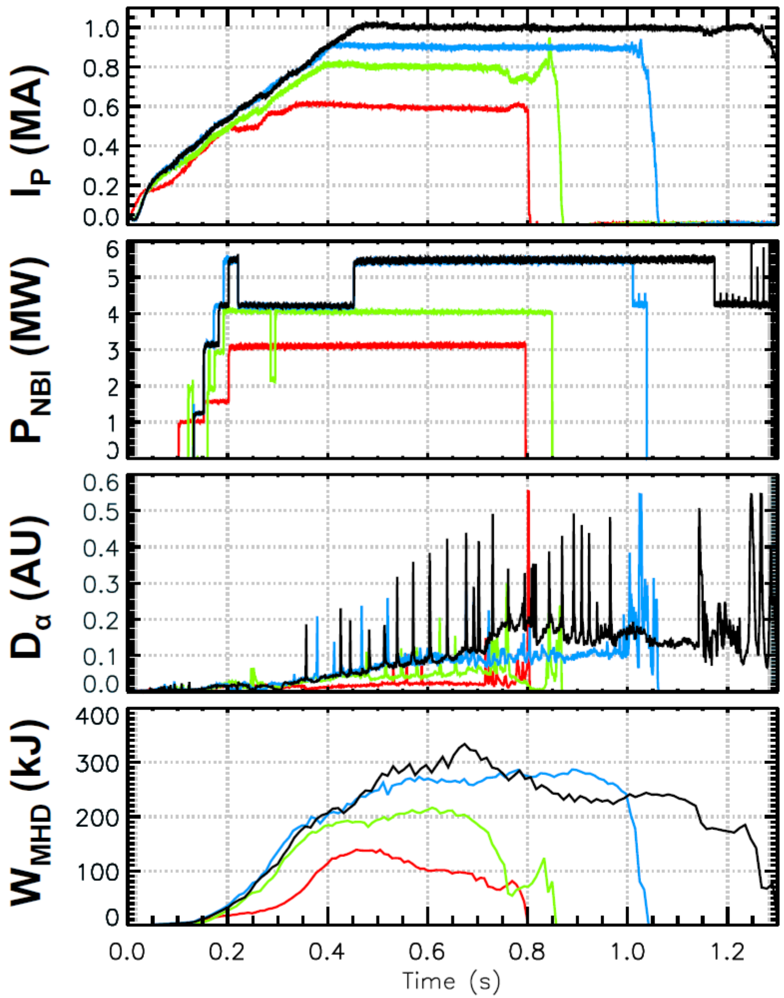
NSTX-U L-mode duration exceeds longest NSTX H-mode



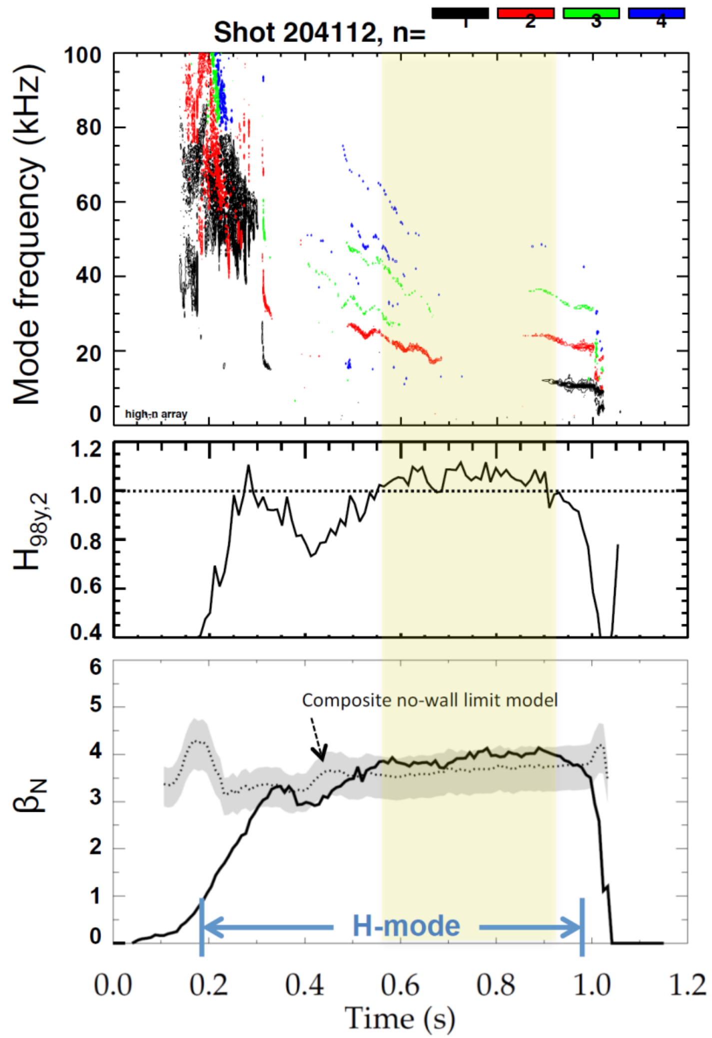
NSTX-U B_T > highest NSTX B_T

Recovered ~1MA H-modes with weak/no core MHD
(comparable to best NSTX plasmas at similar plasma current)

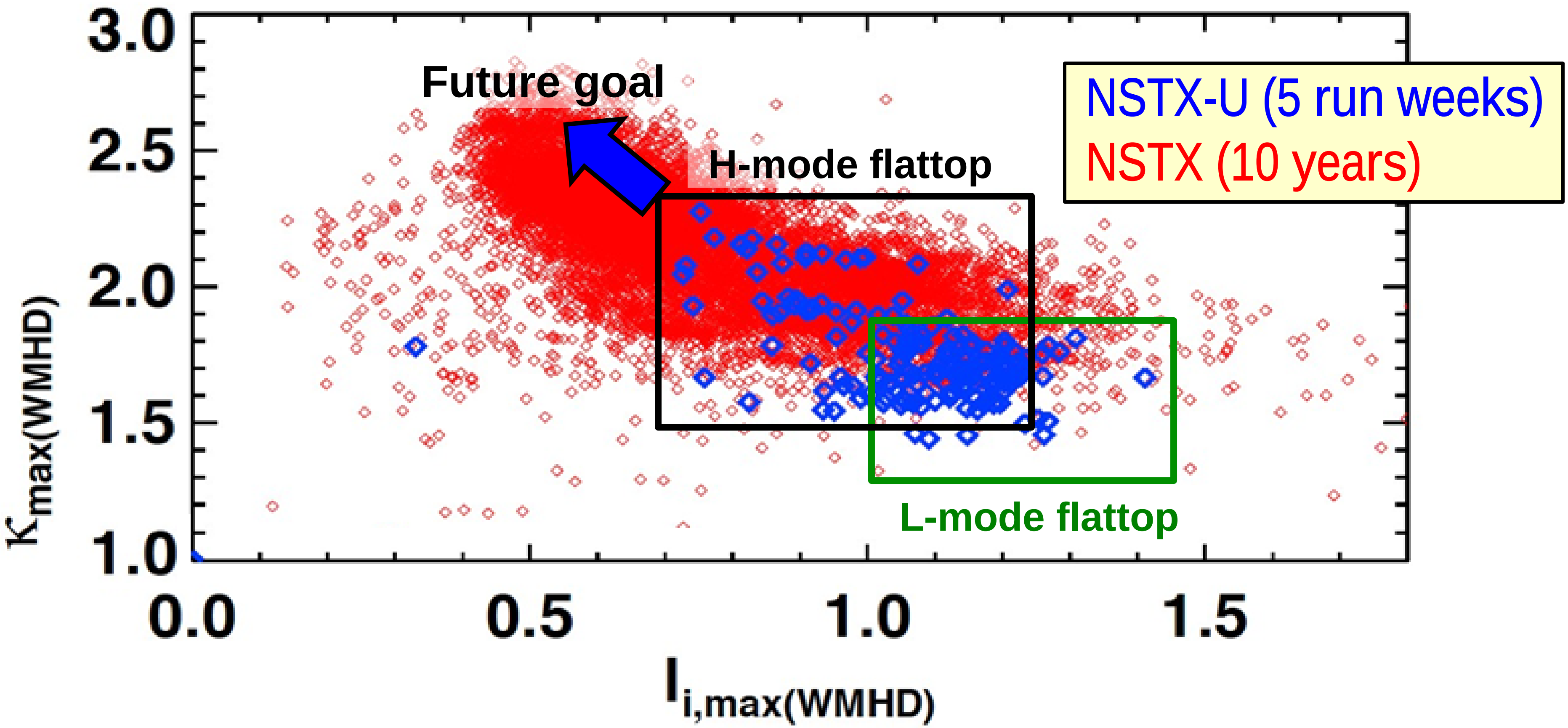
202946 – no EFC 204112 – EFC v2
203679 – EFC v1 204118 – EFC v2



$H_{98} \geq 1$, β_N ~ 3.5-4 ≥ n=1 no-wall limit



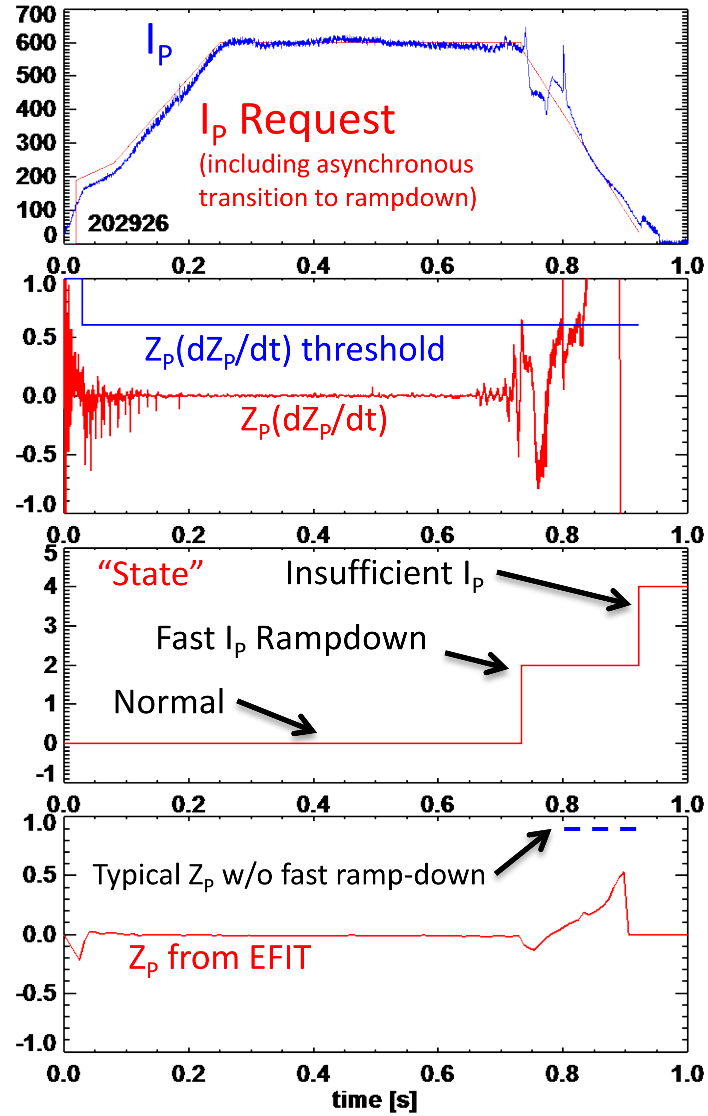
Accessed low l_i and high κ using progressively earlier H-mode and heating + optimized EFC



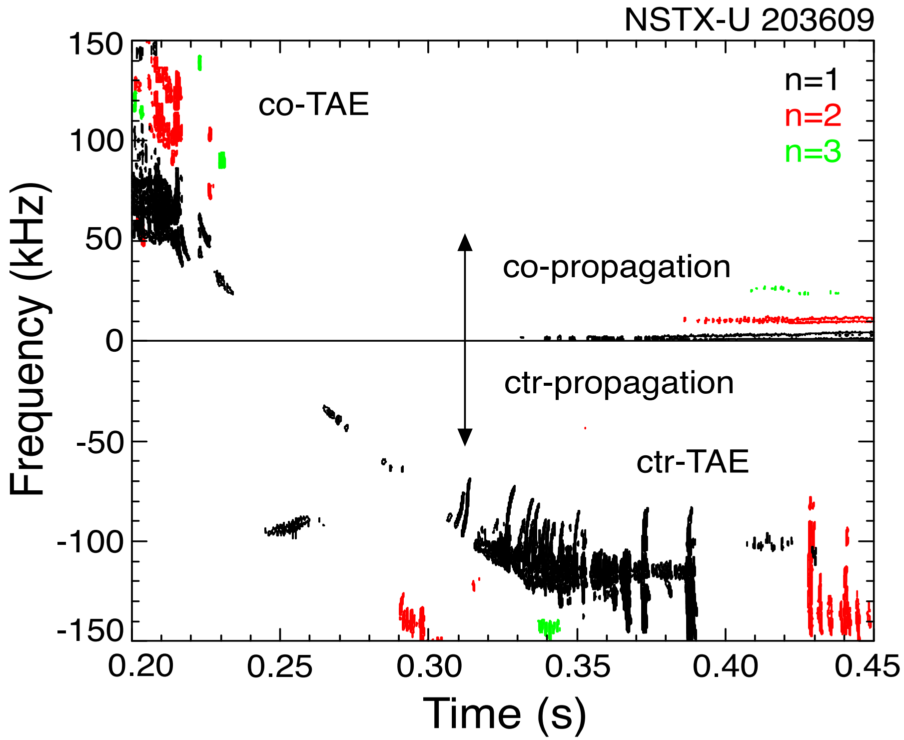
- **NSTX-U: Additional sensors improve estimation of Z, dZ/dt**
- **Goals for next run:**
  - **Access l_i = 0.5-0.7, κ=2.4-2.7, B_T = 0.75-1T, I_P = 1.5-2MA**

Implemented automated ramp-down for NSTX-U

- Plasma control system detects loss of control
  - Central solenoid coil near maximum allowed current
  - Vertical oscillations exceed threshold
  - **ABS**(I_p–$I_{p\ request}$) above threshold
- "State-machine" based:
  - Feedback control switches to new "states" that attempt to stably ramp-down the plasma

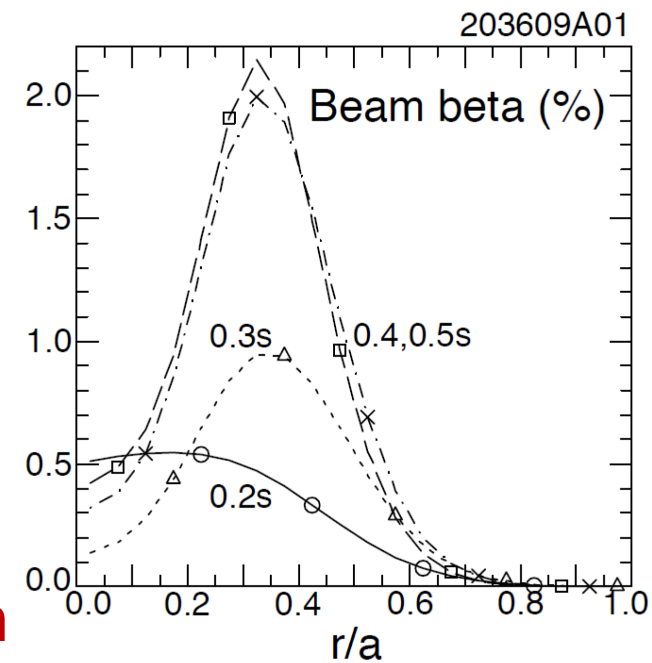


NSTX-U: Most tangential NBI generates counter-propagating Toroidal Alfvén Eigenmodes (TAEs)



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

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



- **1st evidence of off-axis NBI deposition**

Summary: NSTX-U strongly supporting advanced predictive capability, ITER, PMI, next-step STs

- Productive first year of operations on NSTX-U
- Advancing predictive capability for core, edge, PMI
- Developed attractive ST-FNSF / Pilot concepts
- Aim to resume NSTX-U operation following repairs

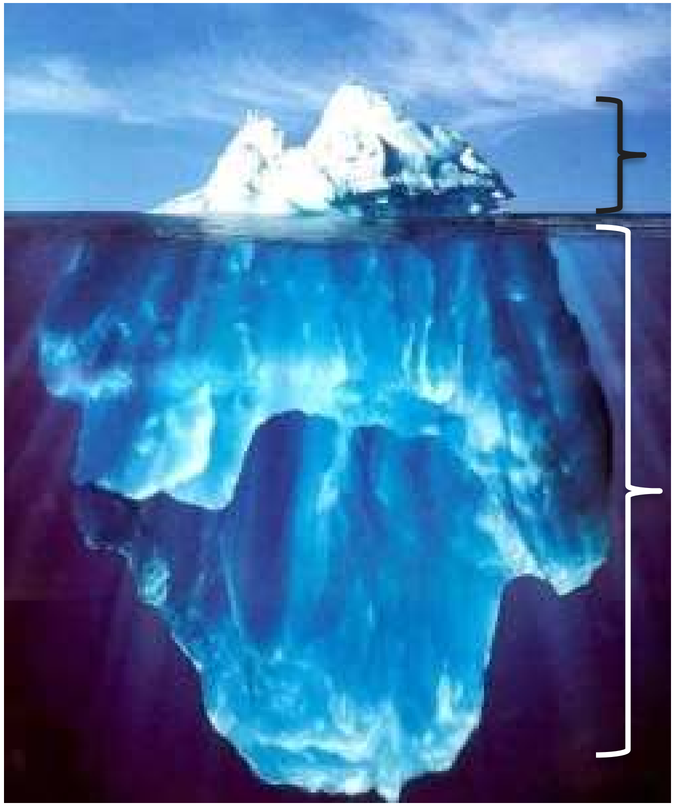
➢ **See research opportunities on next slides**

Research opportunities

- PPPL has a wide range of research activities, many opportunities for novel, impactful projects

- New projects coming
  - NSTX-U
  - FLARE
  - W7X collaboration
  - ITER
  - Theory core codes
  - Liquid walls
  - Plasma-nano

  ...

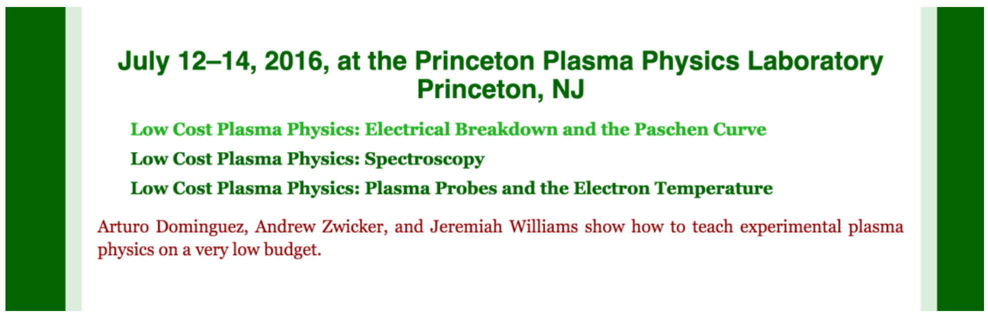


SULI Internship

- **Summer Undergraduate Laboratory Internship (SULI) & Community College Internship (CCI) programs**
  - Paid summer internship program
  - 1 week course intro to plasma/fusion
  - 9 week project based internship
  - Also available for fall/spring semester long internship
  - Summer 2017 applications are now open!

Faculty and Lecturer opportunities: ALPhA immersion and VFP

-**ALPhA immersion** is an NSF, AAPT funded 3 day workshop for college faculty and lecturers to develop “beyond first year” labs. We run 3 workshops to develop low cost plasma physics experiments using a DC discharge apparatus (total cost <$5k).



-The **Visiting Faculty Program (VFP)** provides the opportunity for a faculty member accompanied by one or two students to come to a national lab (e.g. PPPL) and work on a project with a host researcher for 10-weeks(time-frame coincides with the SULI/CCI programs).

Thank you!