National Spherical Torus eXperiment Upgrade

NSTX-U: An Essential Science Facility for US Fusion Innovation

National Academy Panel Visit to PPPL

S. Gerhardt, for the NSTX-U Team

April 11, 2018

Princeton Plasma Physics Laboratory

PPPL PRINCETON PLASMA PHYSICS LABORATORY

PRINCETON UNIVERSITY

Here is why we are excited for NSTX-U

- NSTX-U contributes to optimization of the tokamak concept though optimization of aspect ratio

- NSTX-U is a Key Component of the PPPL Programmatic Strategy Towards Fusion Power

- NSTX-U has a unique role within the world ST program

- NSTX-U will provide answers to critical fusion science questions early in its operation.

Outline

- Why study spherical tori/tokamaks (STs)
  → And why use NSTX-U for that mission.

- Brief Discussion of NSTX-U Operations and Recovery

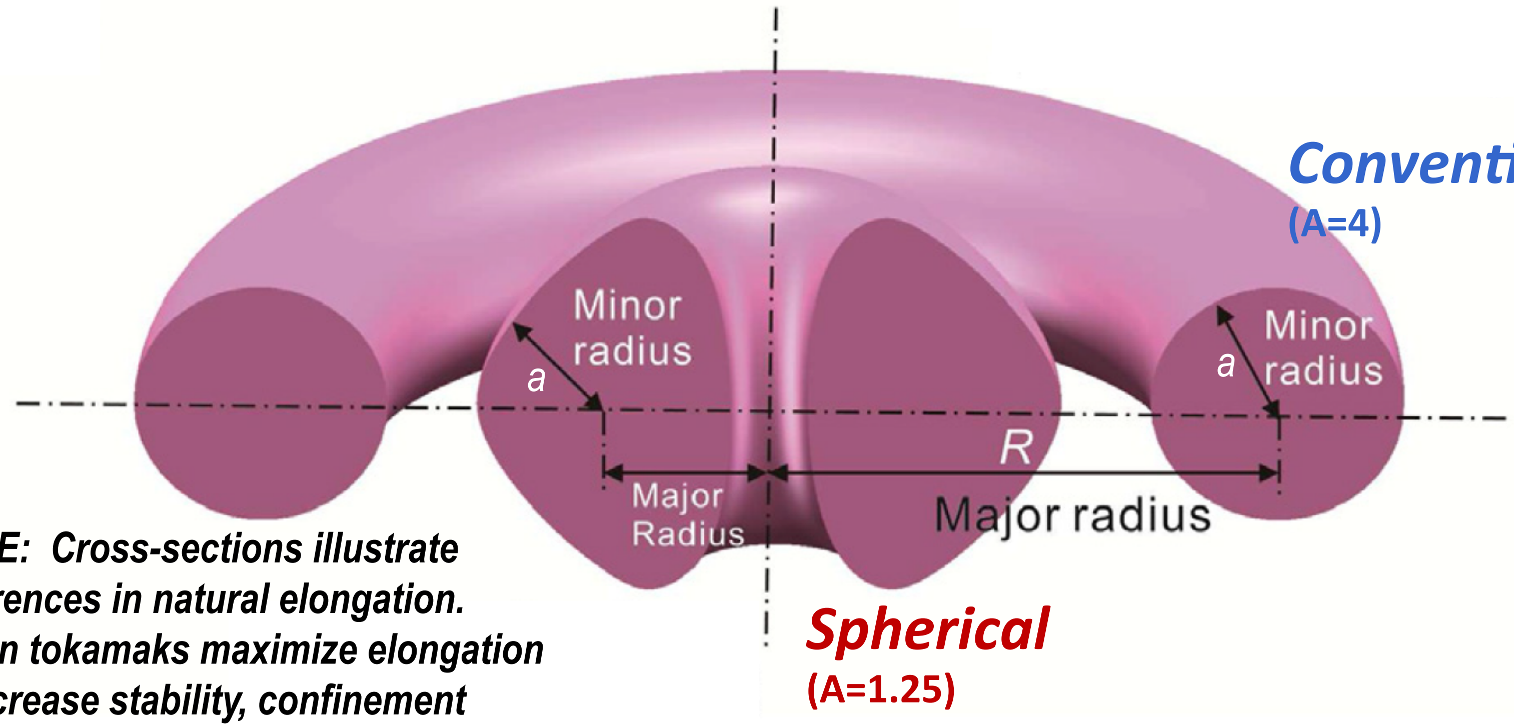
- Key Scientific Deliverable of the NSTX-U Program

- NSTX-U in the Larger ST/Fusion Context

Aspect ratio is important tokamak geometrical parameter that has not yet been fully explored

Aspect ratio A = R / a

R = Major radius *a* = Minor radius



NOTE: Cross-sections illustrate differences in natural elongation. All modern tokamaks maximize elongation to increase stability, confinement

Spherical tokamak / torus (ST) A < 2

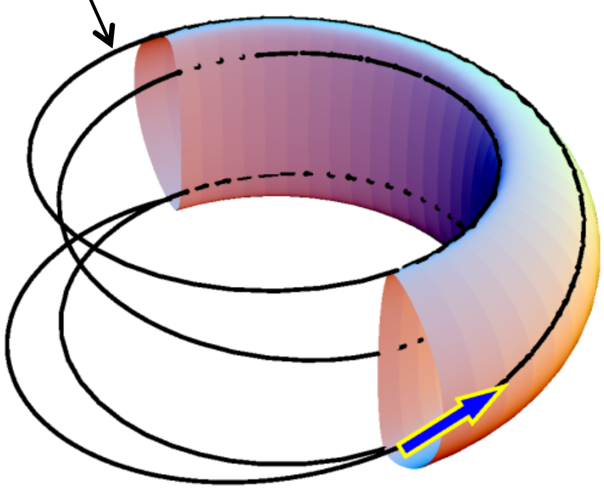
Conventional tokamak typically A > 2.5

Spherical shape increases plasma stability

- Key plasma physics and fusion parameter is normalized pressure = $\beta \propto p / B^2$

Plasma pressure (p) Magnetic pressure (B^2)

Magnetic field line



A=4, κ=2

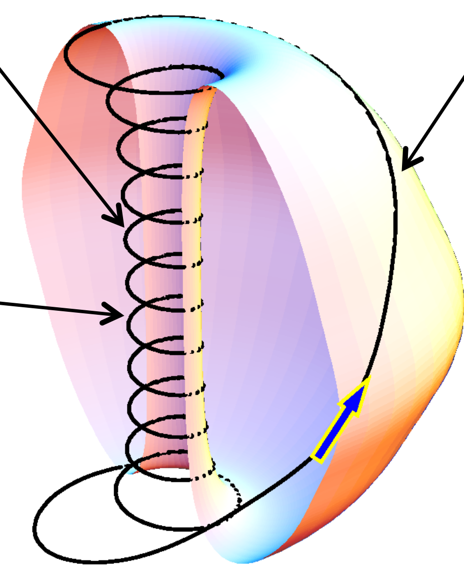
ST field line trajectory favors the stabilizing good curvature region

→Higher β

→Changes turbulence, confinement properties

Stable region

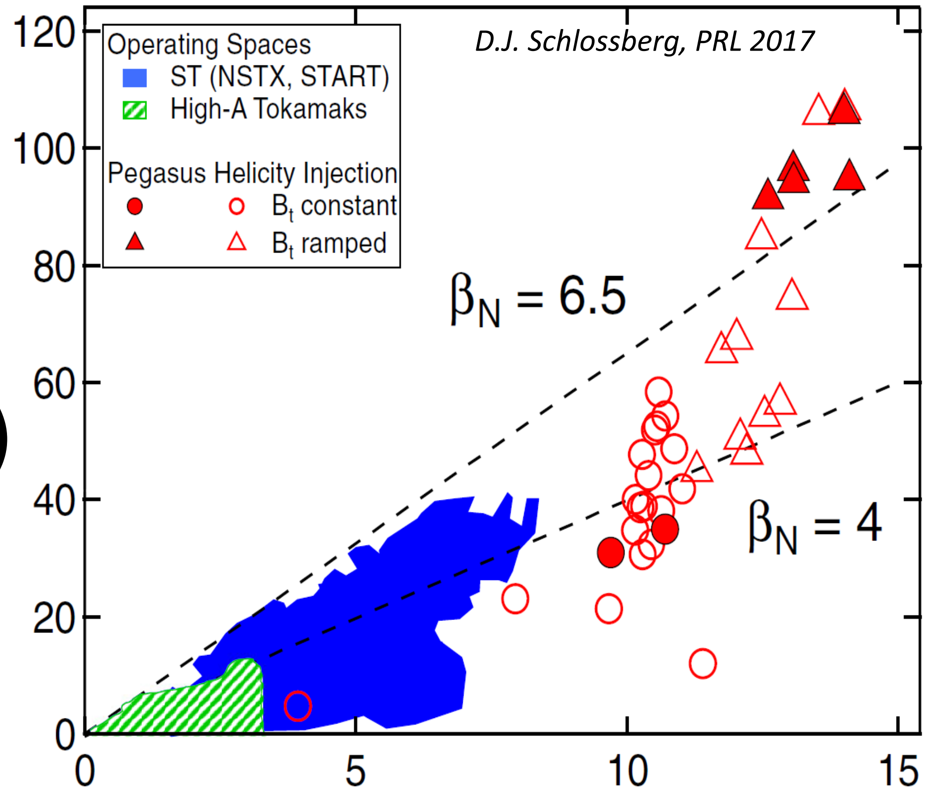
Unstable region



A=1.3, κ=3

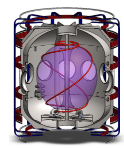
STs access very high toroidal β_t

β_t (%)



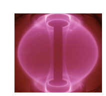
Normalized Current

$I_N = I_P / aB_T$ (MA / m·T)

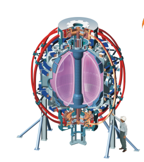


A=1.1-1.2
B=0.025T

Pegasus (US): β_t ~ 30-100%



A=1.25
B=0.15T



A=1.35
B=0.3T

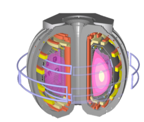
START (UK), NSTX (US): β_t ~ 30-40%

MAST (UK): β_t ~ 15-20%

A=1.45
B=0.4T

DIII-D (US): β_t ~ 12%

Conventional A=2.5, B=0.8T



Increasing A and B

STs can maintain a higher $\beta_N = \beta_t/(I_P/aB_T)$

Key Science Question: What Aspect Ratio and Field Maximizes the Pressure (βB^2) in Steady State?

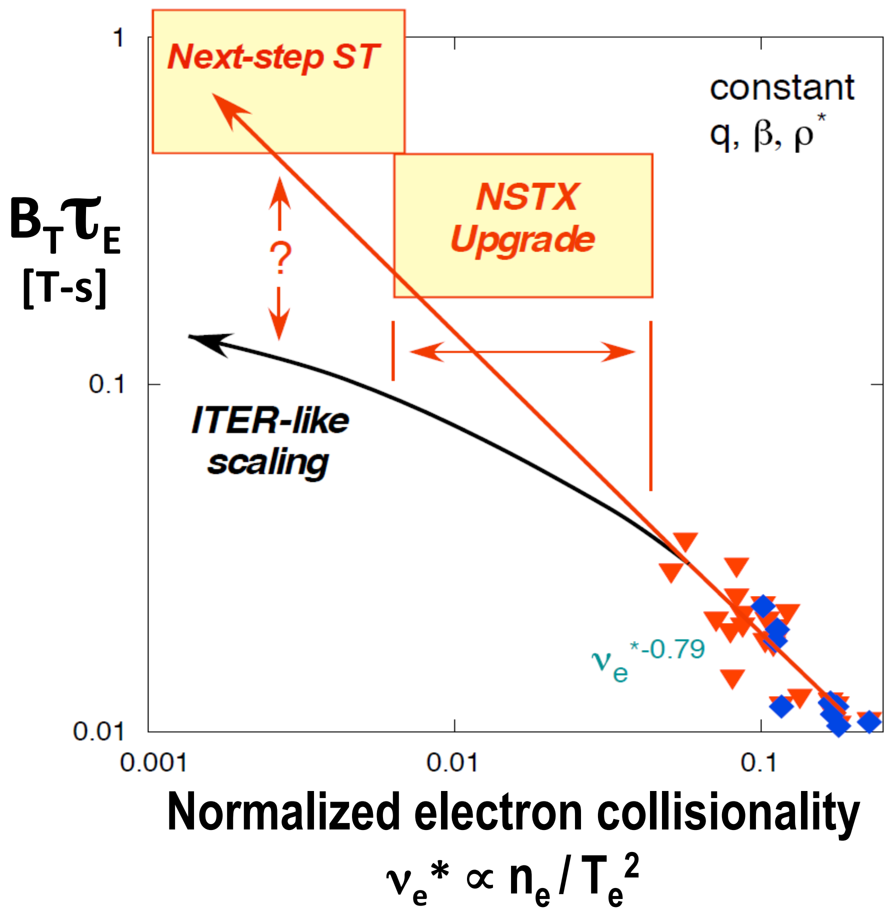
Data indicates ST confinement improves towards fusion relevant collisionality

ST scaling: $\tau_{E,th} \propto \nu_e^{*\,-0.8}\ \beta^{-0.0}$ **ITER basis:** $\tau_{E,th} \propto \nu_e^{*\,-0.1}\ \beta^{-0.9}$

Does favorable ST confinement scaling with ν* continue at lower ν^*?

- Only 20% difference in scaling values in NSTX regime
- NSTX-U will access higher T_e and lower ν^* to distinguish between scalings

Could this be a game changer for configuration optimization?

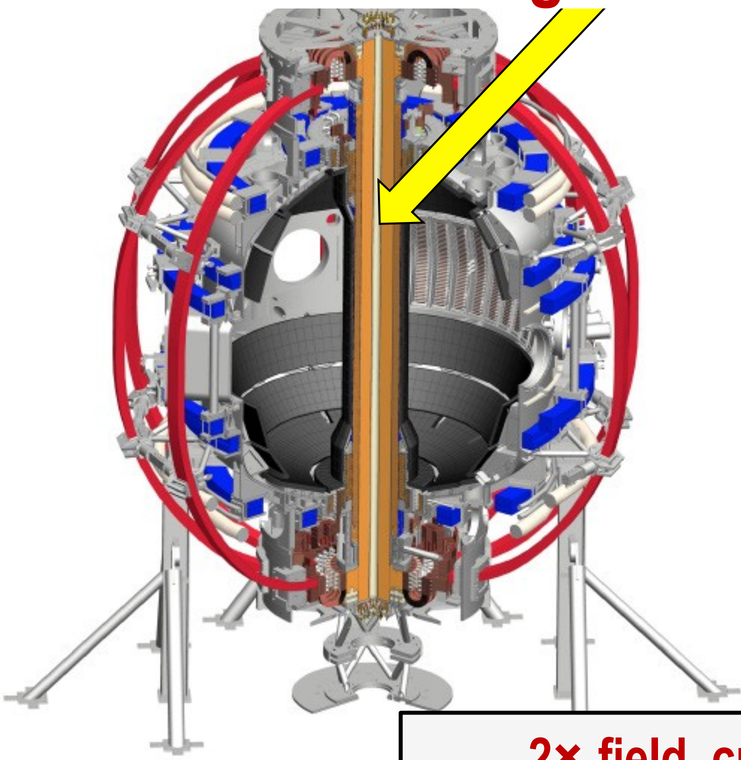


NSTX-U vital for addressing key ST / fusion questions

Will access new physics with 2 new tools:

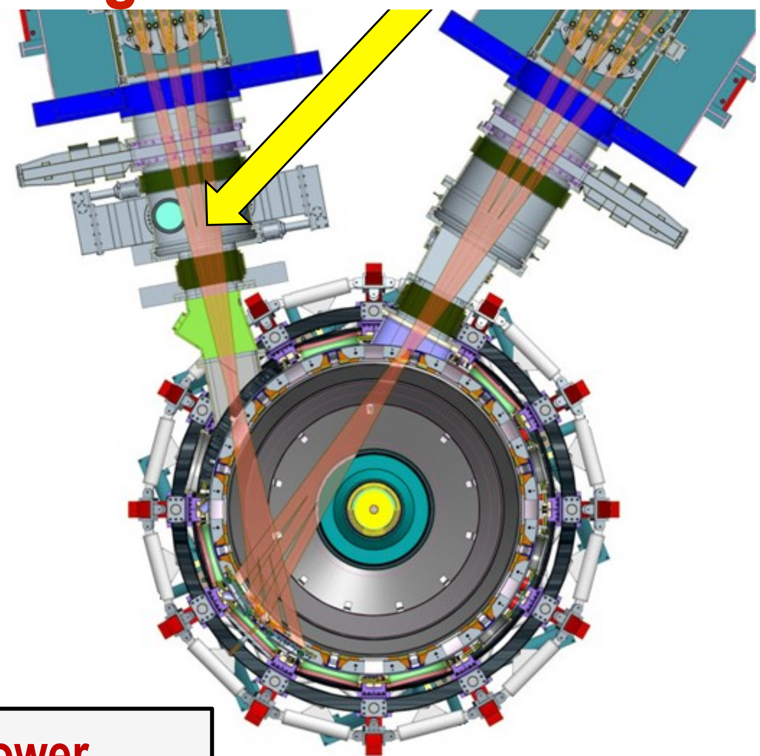
Highest normalized pressure at high T & Low Collisionality

1. New Central Magnet



Sustain plasma without transformer

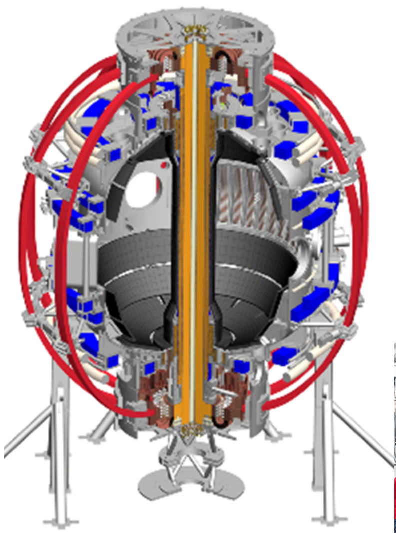
2. Tangential 2nd Neutral Beam



2× field, current, power
4× heat flux, 5× pulse length
Up to 10× higher $nT\tau_E$

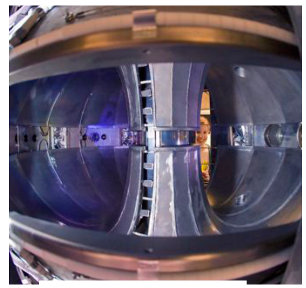
NSTX-U is one of the two largest facilities in a world-wide ST research program

NSTX-U, USA

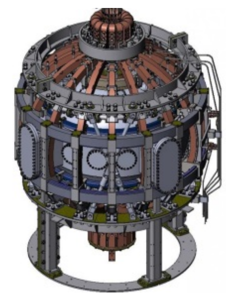


PEGASUS, USA

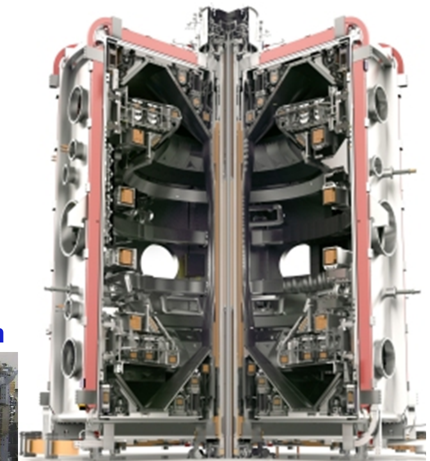
LTX-β / CDX-U, USA



GLOBUS-M2, Russia

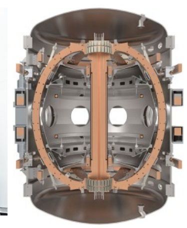


MAST-U, UK



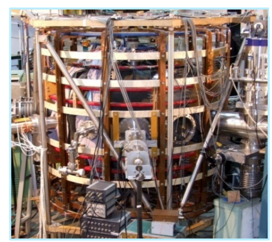
PI3, Canada

ST40, UK



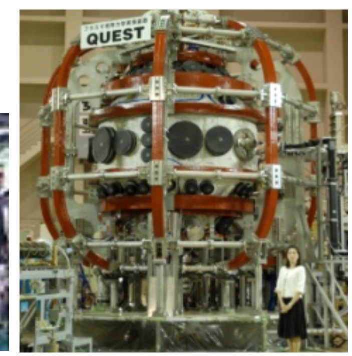
Proto Sphera, Italy

SUNIST, China



KTM, Kazakhstan

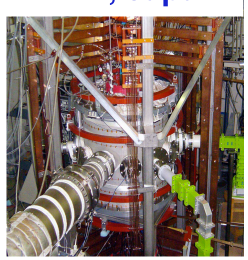
QUEST/CPD, Japan



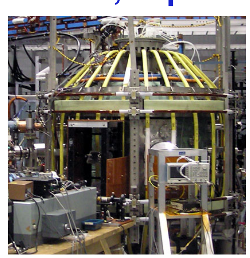
VEST, Korea

HIST, Japan

LATE, Japan



TST-2, Japan



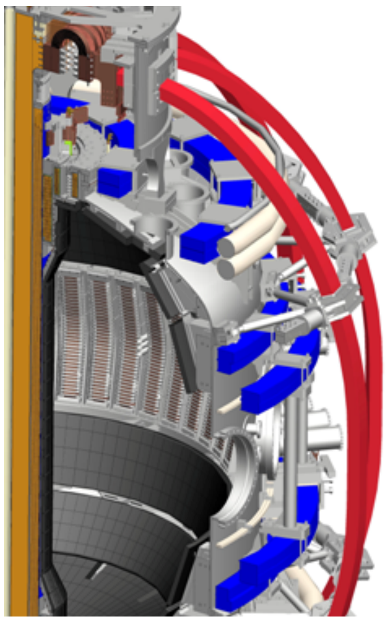
UTST, Japan

TS3/4, Japan

17 international facilities for ST research and broader fusion science

NSTX-U and MAST-U are the most capable devices in a world-wide ST research program

NSTX-U



MAST-U (UK)

Similar features:

- **Major radius R = 0.8-1m**
- **Plasma current up to 2MA**
- **Pulse durations 1s → up to 5s**
- **Strong neutral beam heating**

Complementary Research:

Core emphasis

- Highest plasma beta in large ST, facilitated by passive conductors and RWM control
- Only large ST with RF heating

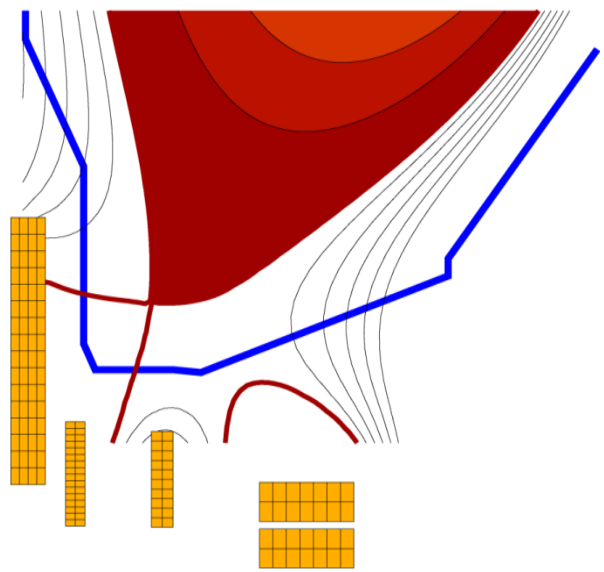
Boundary emphasis

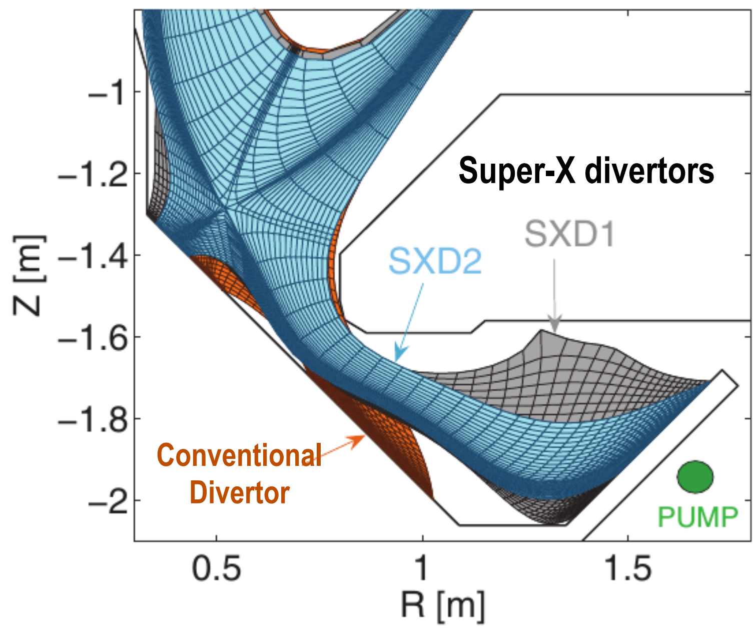
- "Long-leg" divertor for power exhaust research
- Only large ST with off-midplane 3D magnetic field coils

STs will provide leading contributions to development and understanding of advanced divertors

NSTX-U: Short-leg flared divertor + radiation to mitigate heat flux

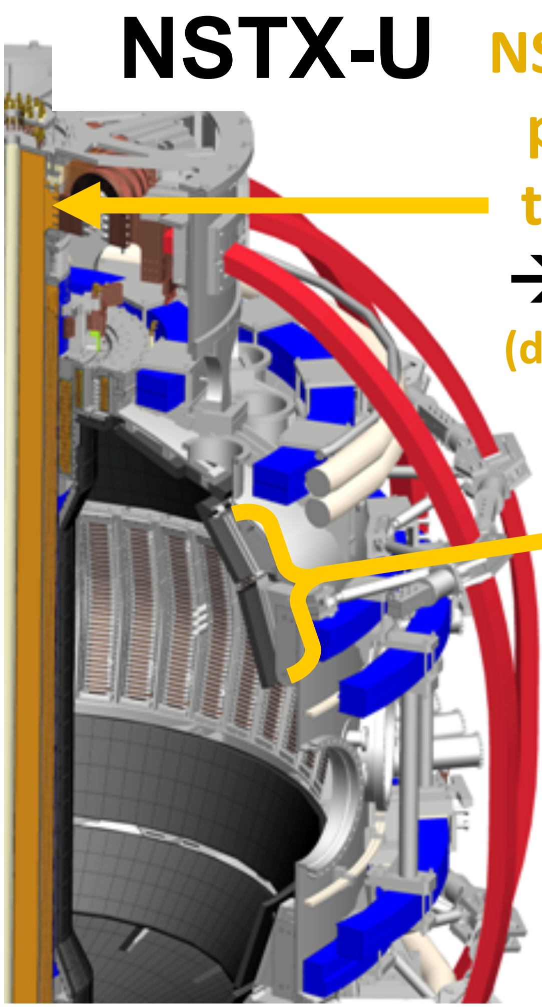
MAST-U: World-leading pumped long-leg + flexible flaring, radiation





Importance of this research elevated by research showing narrow scrape-off-layers at high plasma current (poloidal field)

NSTX-U design enables access to 2-3× higher plasma pressure, temperature than MAST-U



NSTX-U central magnet provides 1.5× higher toroidal field current

→ ~1.5 - 2× higher B_T^2

(depending on plasma shape)

Conducting plates can suppress global kink instabilities,

~1.5× higher β_T

$$p \propto \beta_T B_T^2$$

2-3× higher

Allows an extended range of collisionality, higher self-driven currents

Outline

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  → And why use NSTX-U for that mission.

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- NSTX-U in the Larger ST/Fusion Context

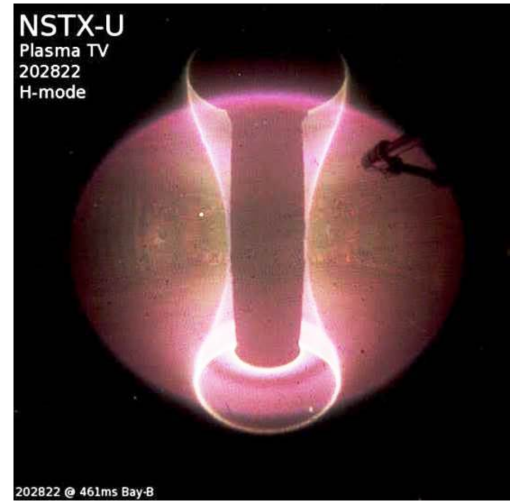
In FY2016, NSTX-U had scientifically productive 1st year

NSTX-U in high demand: ~4x more run-time requested than available

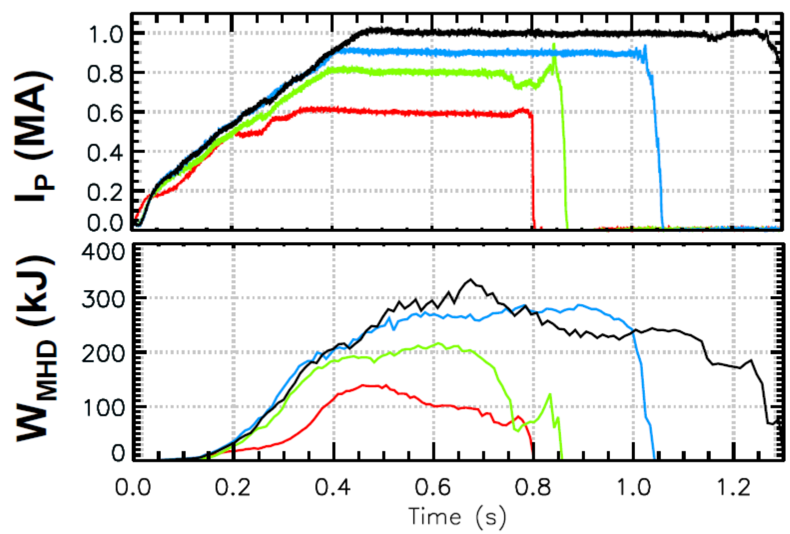
- Achieved H-mode on 8th day of 10 weeks of operation
- Surpassed magnetic field and pulse-duration of NSTX
- Matched best NSTX H-mode performance at ~1MA
- Identified, corrected dominant error fields
- Commissioned all magnetics, profile diagnostics
- Operated NB systems with up to 12MW NBI available
- Discovery: New 2nd NBI suppresses Alfvénic instabilities
  - APS invited, PRL 2017
- Implemented techniques for controlled plasma shut down, disruption detection / new tools for mitigation

Several papers on NSTX-U results published, in-press, or submitted

- **PF1A upper divertor coil failed, ending the run**



mid January early April
early March early April



NSTX-U Recovery will restore NSTX-U as a critical facility for fusion science research

- Investigations following the coil failure identified additional issues with the design of the device
- The “Recovery Project” scope was defined by an extensive Extent of Condition analysis during 2017, resulting in numerous changes to improve the reliability of the device, including:
  - New inner-PF coils with improved design and manufacture
  - New plasma facing component tiles
  - Improved vacuum boundary
- These technical activities in parallel with a complete overhaul of PPPLs engineering and quality procedures
- Aiming for project baselining in the fall of this year, with resumption of operations in 2020

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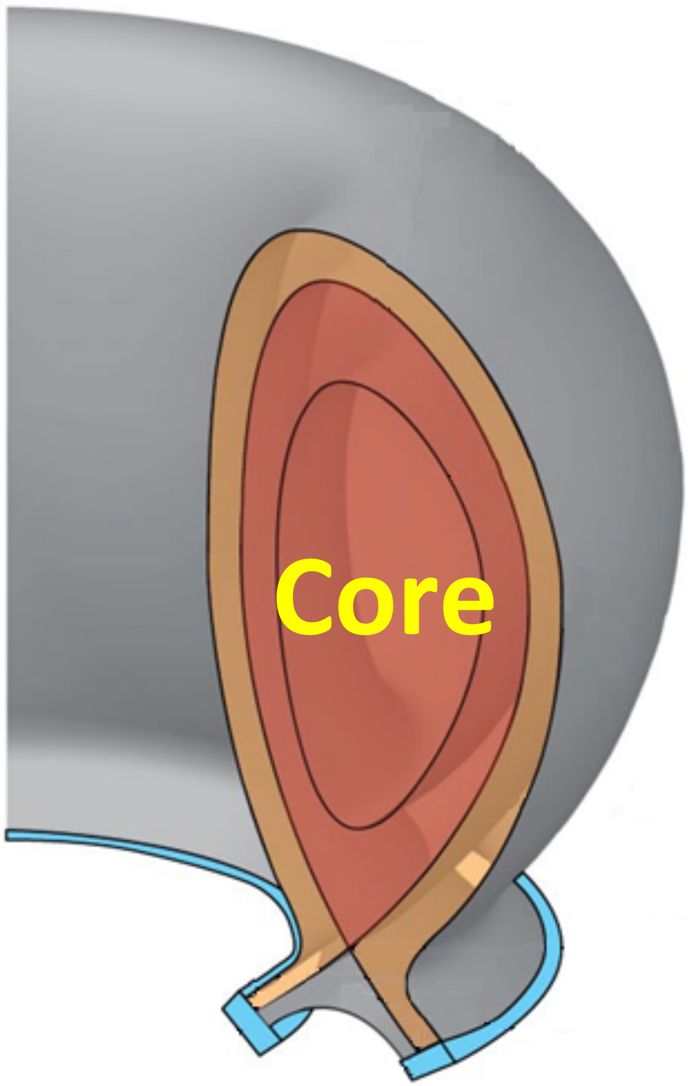
NSTX-U will address critical questions related to performance of the core plasma



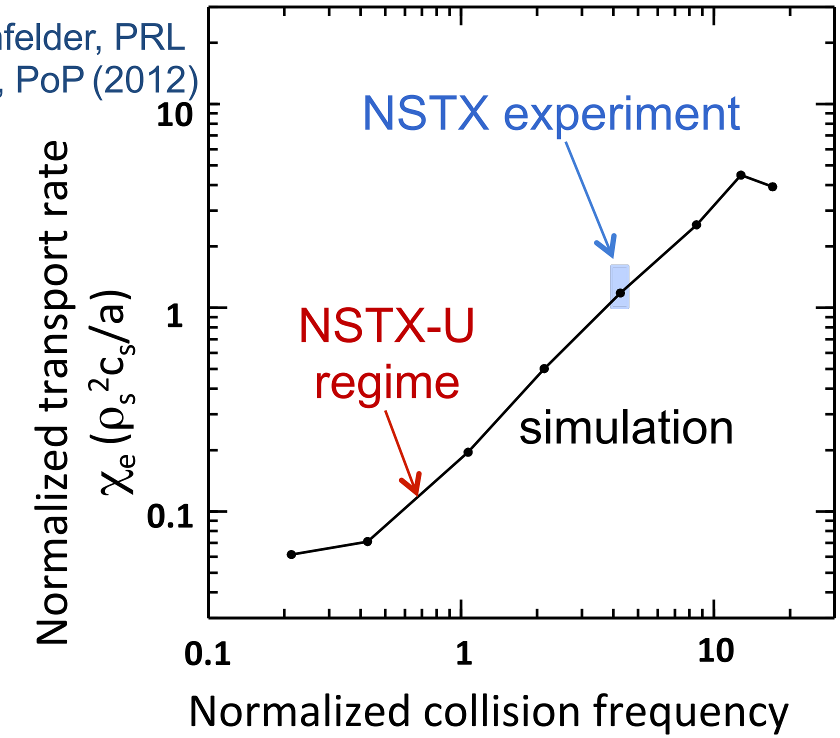
Key Questions:

- How will confinement and stability scale at fusion relevant plasma parameters?
- Can the ST plasma be sustained w/o a transformer?

NSTX-U will access and measure novel regimes of turbulent transport at lower aspect ratio & high beta

- **Experimental Result**: Confinement improves as collisionality is decreased
- Energy confinement scaling from "micro-tearing" modes simulated to be similar to NSTX results
  - Results from non-linear gyrokinetic codes (GYRO)
- Other microturbulent modes (KBMs) may become dominant at low collisionality?

Guttenfelder, PRL (2011), PoP (2012)



NSTX-U will deploy advanced turbulence diagnostics to understand transport

Key differences between high- and low-A:

Low-A: electromagnetic electron turbulent transport is dominant channel, ions neoclassical

Conventional-A: ion transport dominant, electrons and ions transport both dominated by electrostatic turbulence

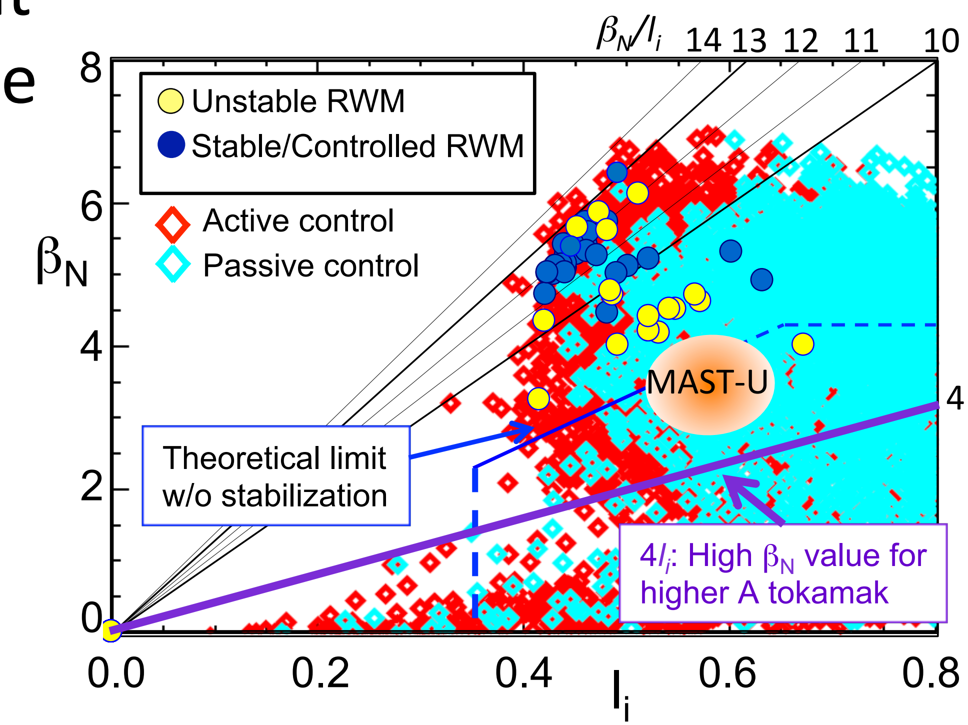
STs have the Potential For High Self-Driven Currents Due to Inherently Higher Stability

- Bootstrap Current: Current internally generated by the plasma pressure itself.

Plasma-driven current fraction

$$f_{BS} \sim A^{1/2} \left(\frac{\beta_N}{l_i} \right) q_{cyl}$$

A **= device aspect ratio**
l_i **= plasma internal inductance**
(Low value ➔ broad current profile)

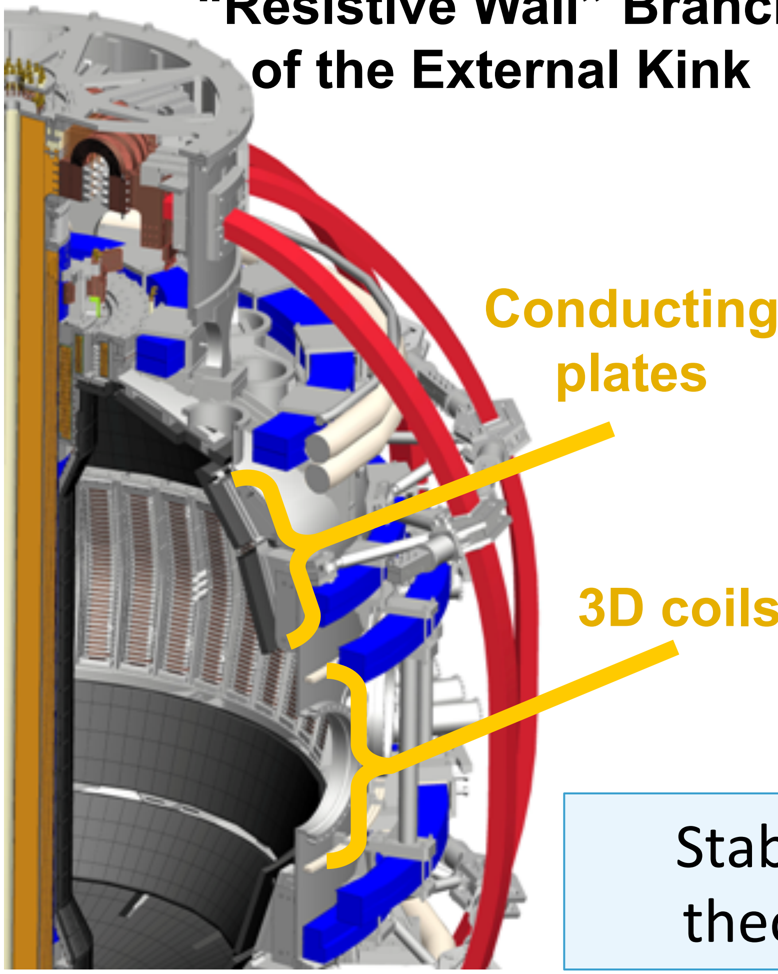


S. Sabbagh et al., Nucl. Fusion **53**, 104007 (2013)

ST plasmas can achieve very high values of β_N/l_i, with associated high non-inductive current

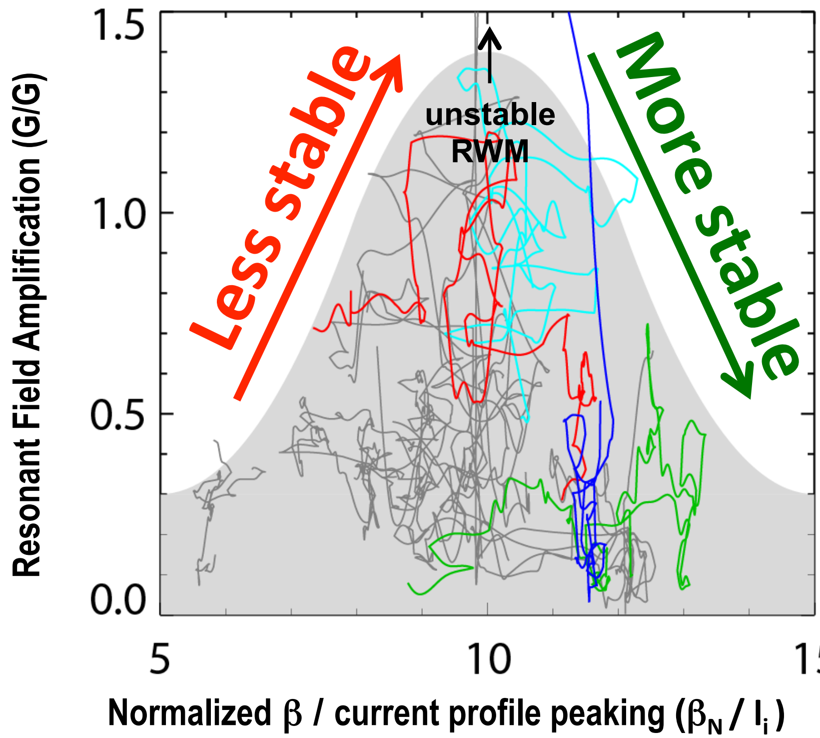
NSTX-U will be leading ST able to access, study, and exploit kinetic resistive wall mode (RWM) stabilization

Conducting Plates Result in the Slowly Growing "Resistive Wall" Branch of the External Kink



Stability Improvement at High β_N/l_i

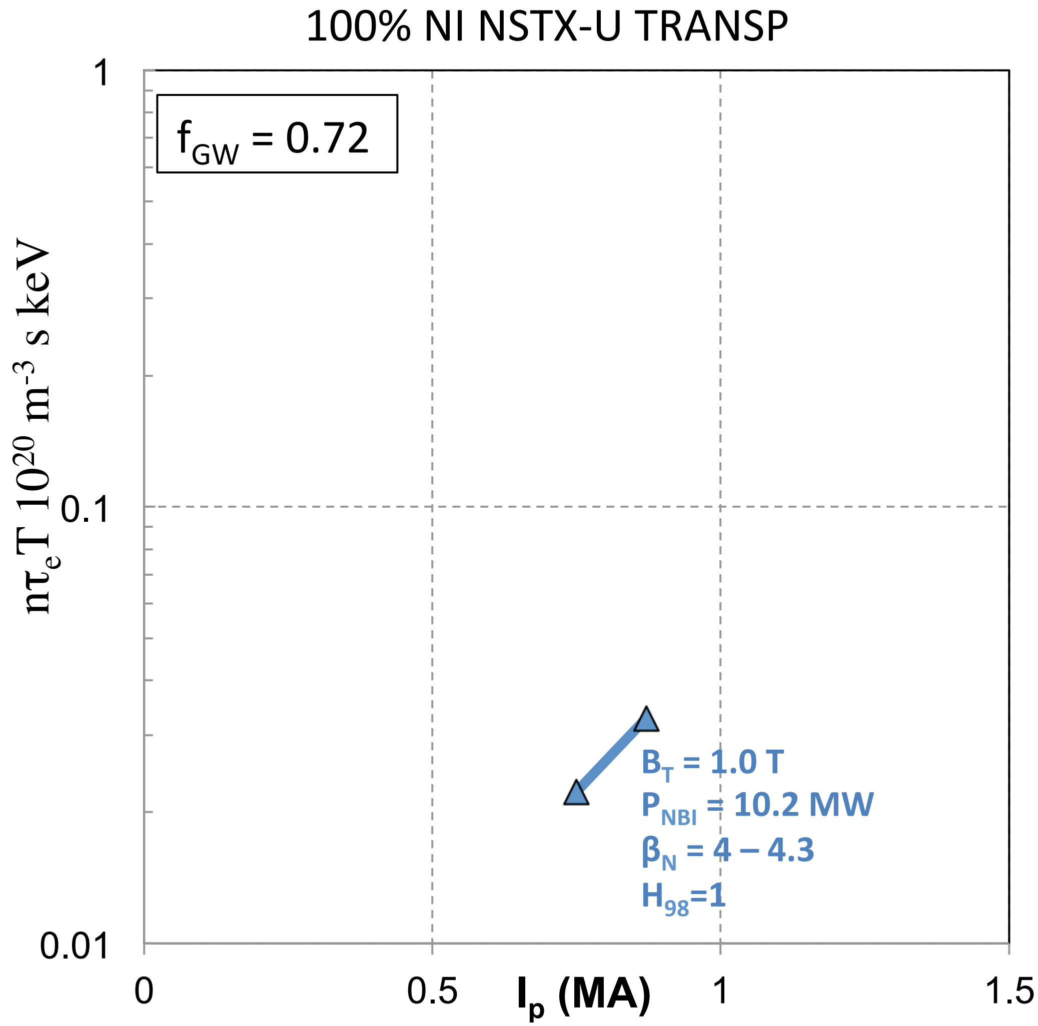
J. Berkery, et al., PoP **21** (2014) 156112



Stability results consistent with drift-kinetic theory including mode-particle resonances

NSTX-U Will Enable Physics Assessment of High-Current 100% Non-Inductive Scenarios

- Conventional aspect ratio H-mode scaling implies non inductive currents up to ~0.9 MA



Adapted from Gerhardt et al., NF 52 (2012) 083020

NSTX-U Will Enable Physics Assessment of High-Current 100% Non-Inductive Scenarios

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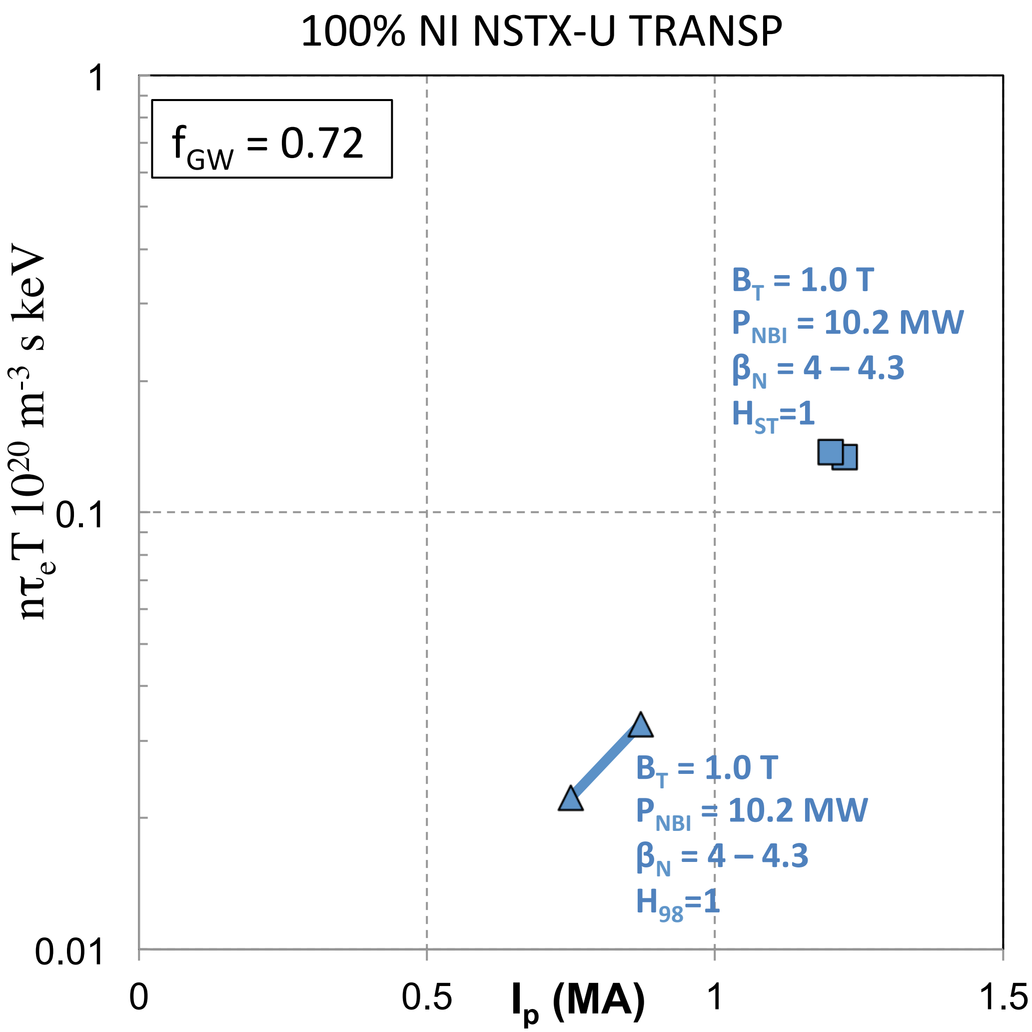
- ST-scaling implies significantly higher performance



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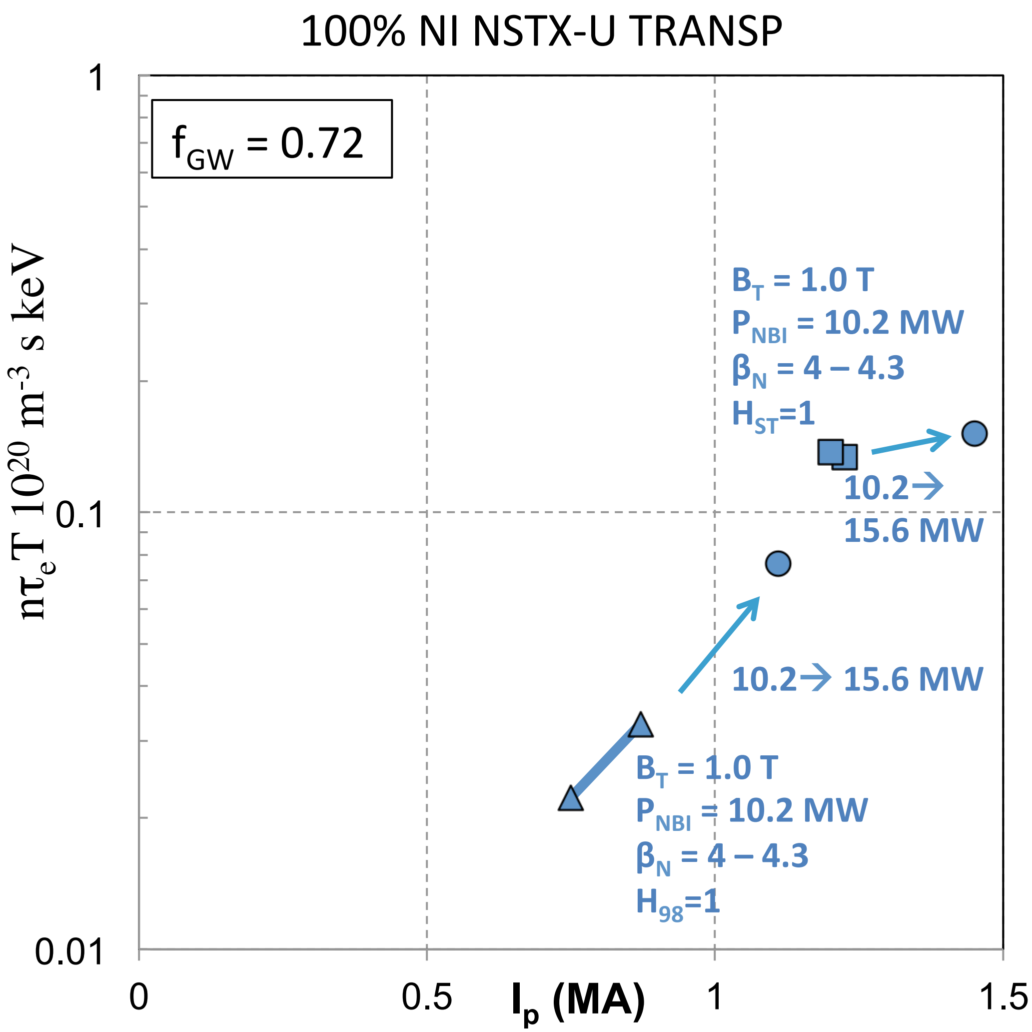
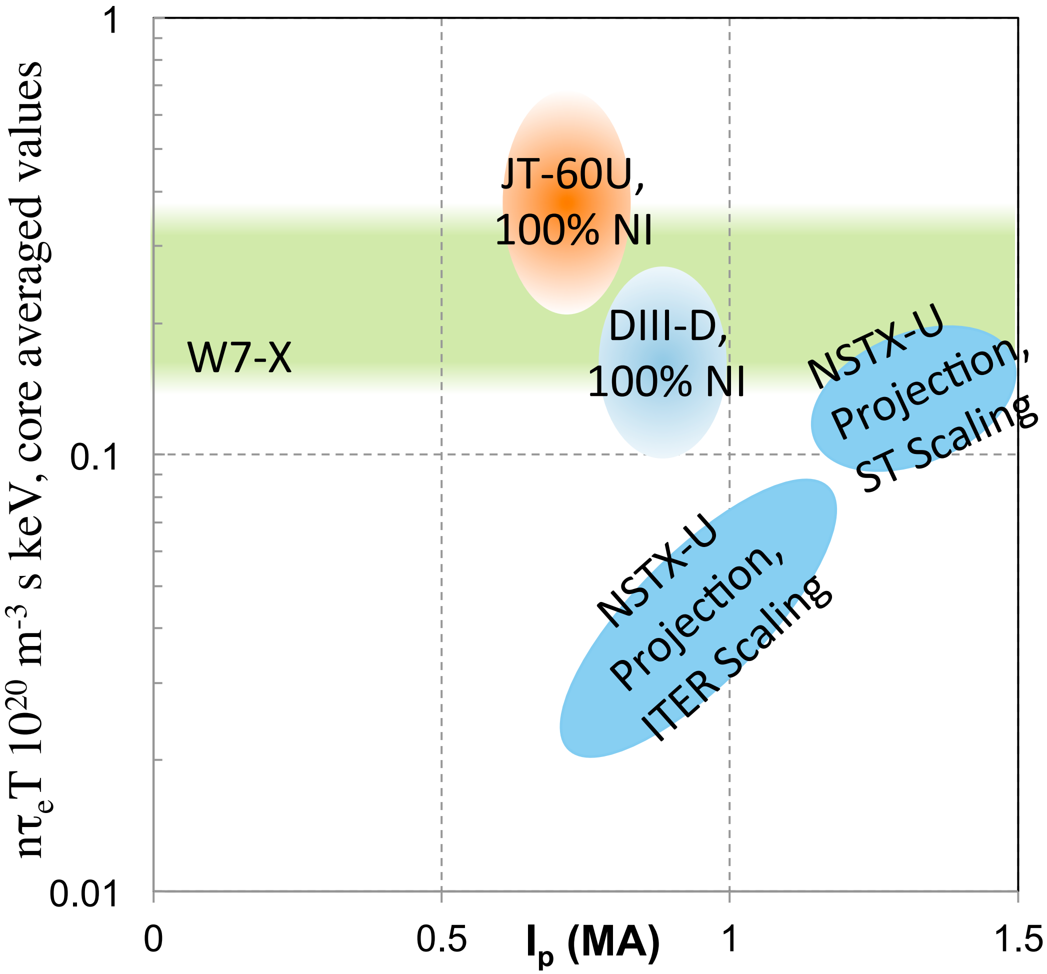


Adapted from Gerhardt et al., NF 52 (2012) 083020

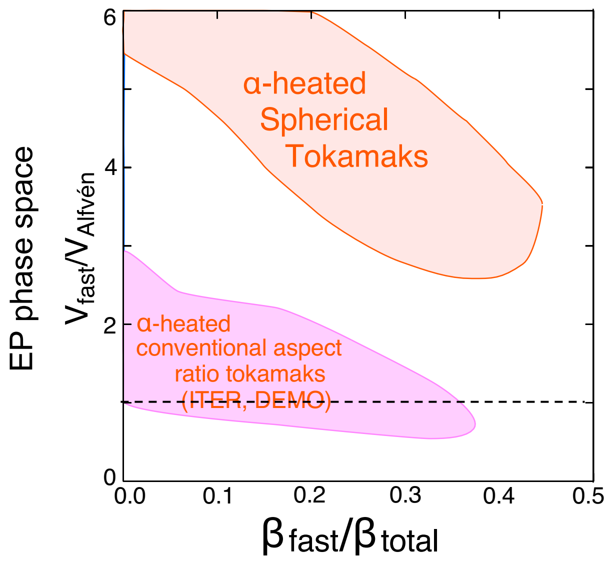
NSTX-U Will Enable Physics Assessment of High-Current 100% Non-Inductive Scenarios

- Conventional aspect ratio H-mode scaling implies non inductive currents up to ~0.9 MA
- ST-scaling implies significantly higher performance
- Performance will be similar to steady state operations in other devices.



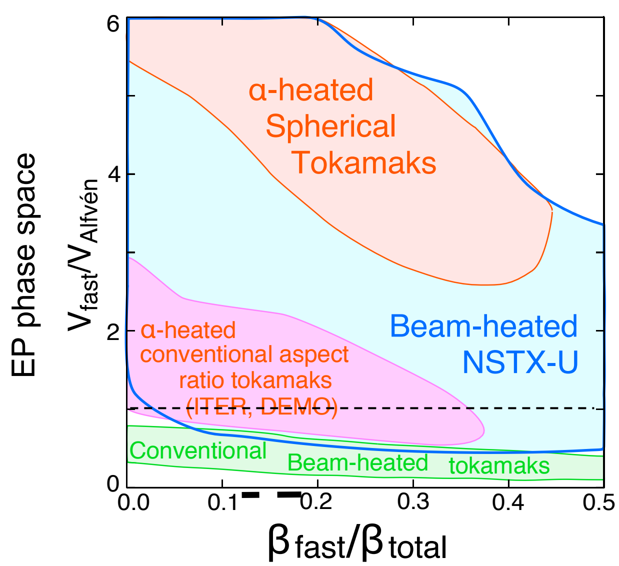
Adapted from Gerhardt et al., NF 52 (2012) 083020

NSTX-U is excellent testbed for studying α-particle physics applicable to burning plasmas, ITER



~ (EP-instability drive)/(EP-instability damping)

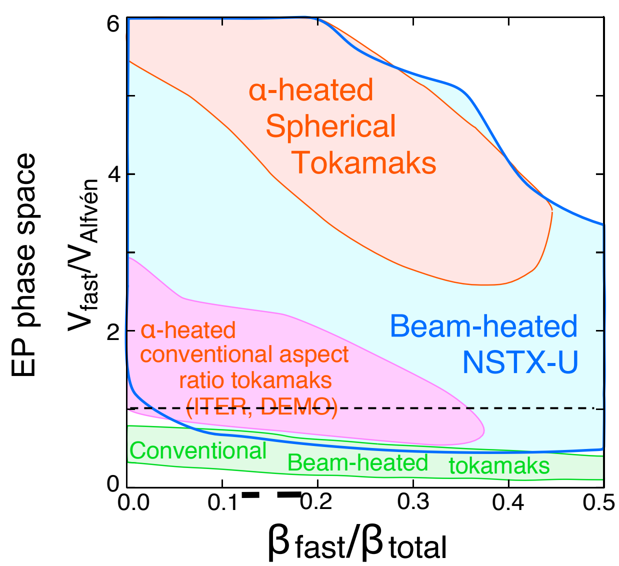
NSTX-U is excellent testbed for studying α-particle physics applicable to burning plasmas, ITER



~ (EP-instability drive)/(EP-instability damping)

- NSTX-U: Fast-ion dynamic range spanning ST and conventional A burning plasma regime
  - **B_T 2× NSTX** → stabilize modes ($V_{fast} < V_{Alfven}$)

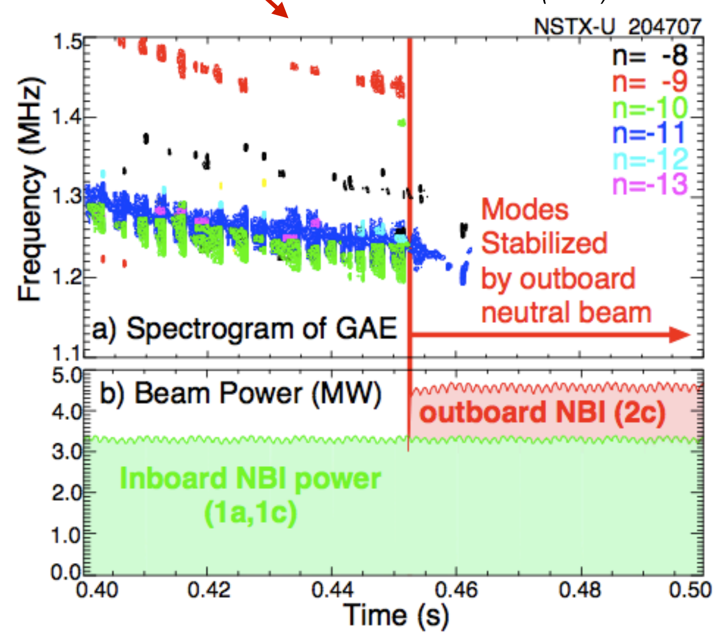
NSTX-U is excellent testbed for studying α-particle physics applicable to burning plasmas, ITER



~ (EP-instability drive)/(EP-instability damping)

- NSTX-U: Fast-ion dynamic range spanning ST and conventional A burning plasma regime
  - **B_T 2× NSTX** → stabilize modes ($V_{fast} < V_{Alfven}$)
  - **Tangential 2nd NBI** → flexible fast-ion distribution
    - Vary pitch angle, pressure profile → Assess mode stability as a function of ion phase space

E.D. Fredrickson, et al., Phys. Rev. Lett. ***18*** *265001 (2017)*

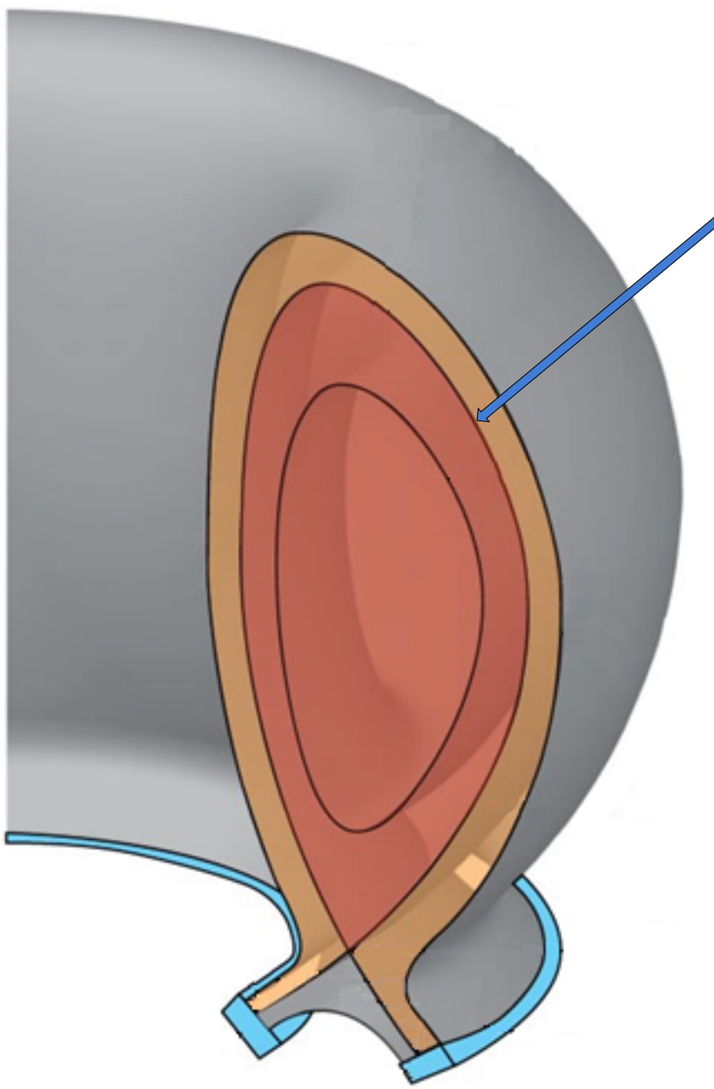


NSTX-U will:

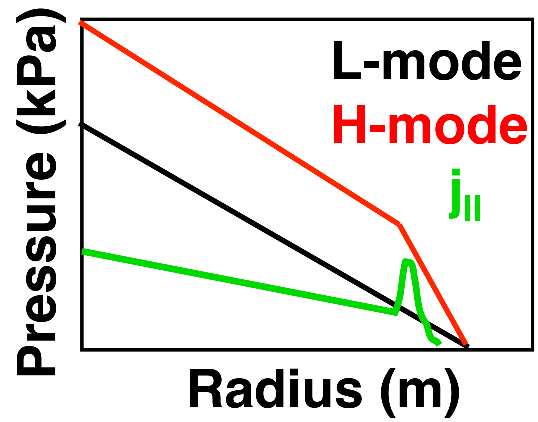
→ **Assess and utilize phase-space engineering to optimize the scenario performance**

→ **Extend predictive capabilities for energetic particle dynamics in burning plasma regime**

Optimization of the edge pedestal is critical for Fusion Performance



Pedestal is a narrow region of steep gradients immediately inside the separatrix



Key Questions:

- What physics sets the height and width of the pedestal?
- How can we control the pedestal?

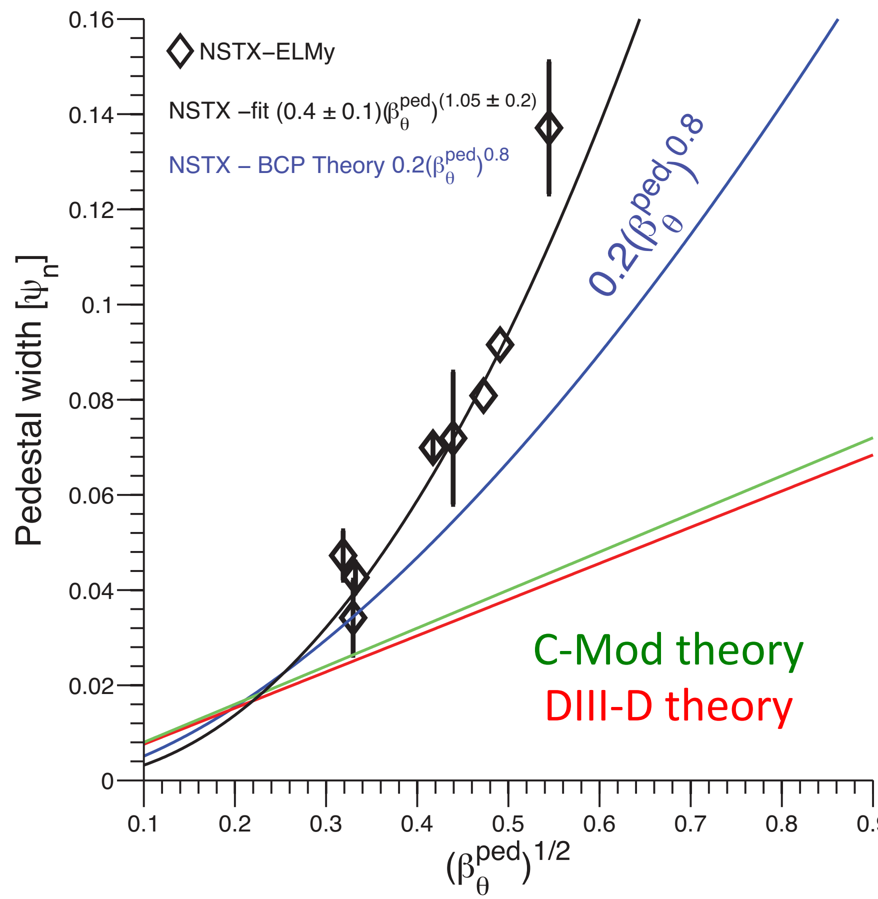
NSTX-U Offers Exciting Opportunities For Optimization of the Edge Pedestal

1: Pedestal Widths is Broader than Predicted by Standard Theory (EPED) for Conventional-A

→ Challenge to theory

→ Opportunity for optimization – improved pedestal stability

A. Diallo et al., Nucl. Fusion ***53*** *(2013) 093026*



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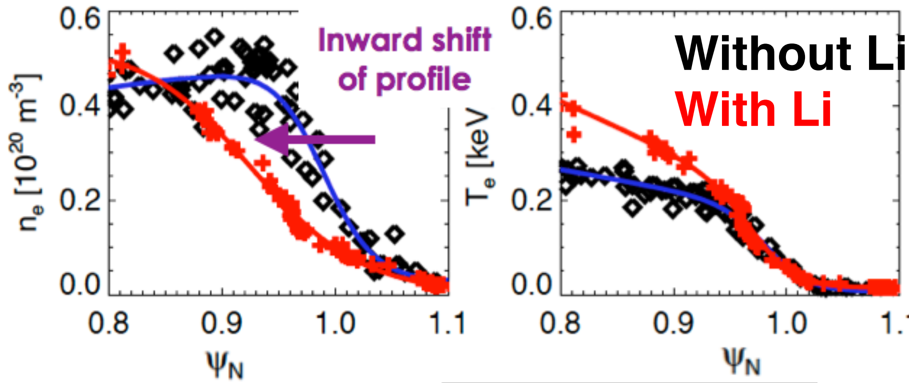
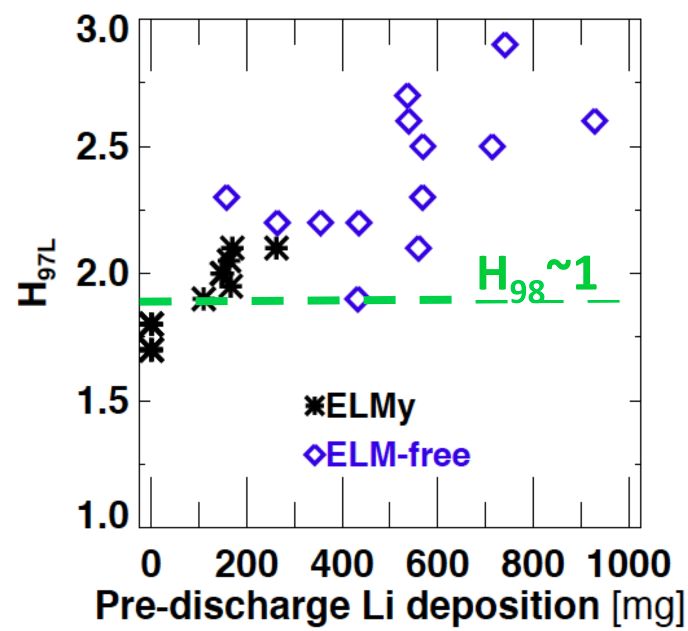
1: Pedestal Widths is Broader than Predicted by Standard Theory (EPED) for Conventional-A
→ Challenge to theory
→ Opportunity for optimization– improved pedestal stability

2: Lithium conditioning of PFCs
→ Broader profiles → Elimination of ELMs
→ Higher confinement

NSTX-U will have world leading capabilities to diagnose, study, and *control* the pedestal parameters.

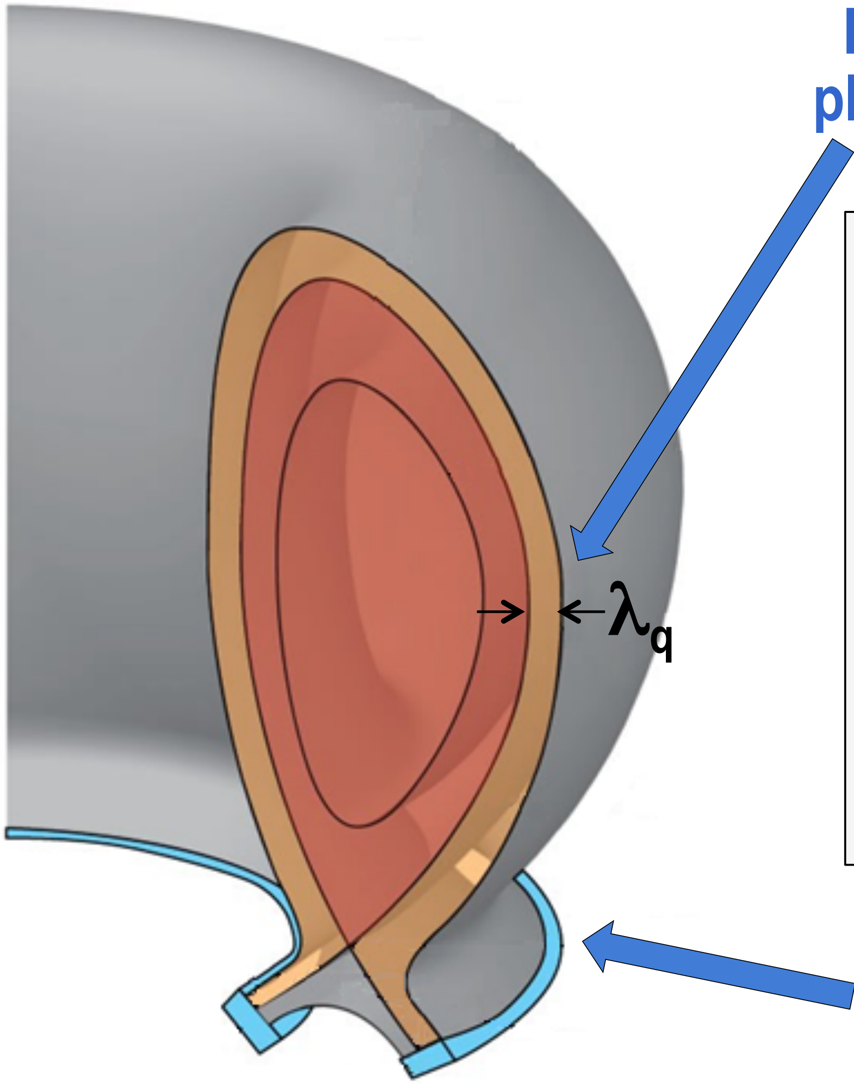
*R. Maingi et al., Nucl. Fusion **52** (2012) 083001*





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

Mitigation / control of edge heat and particle exhaust is critical issue for magnetic fusion



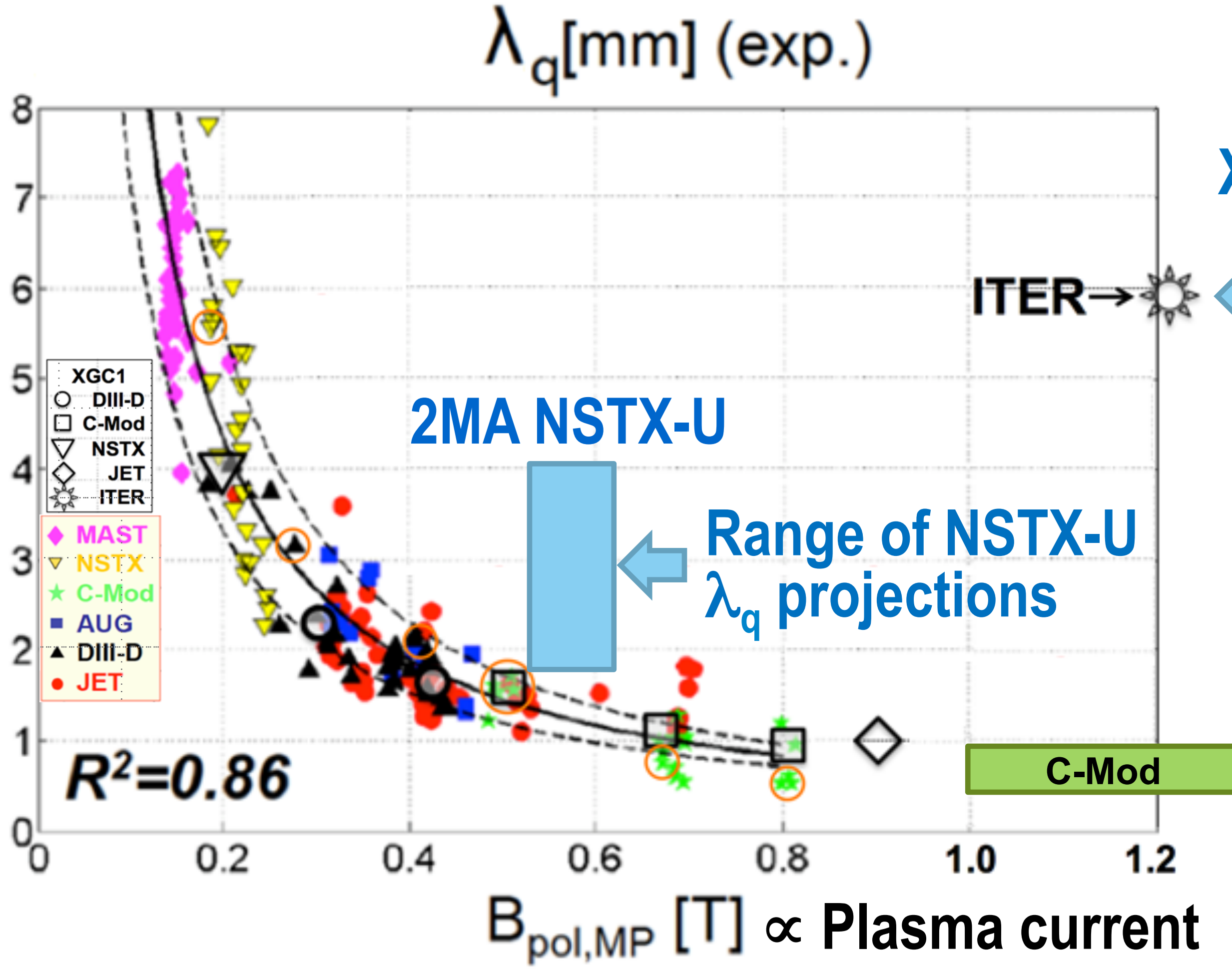
Power exhaust width outside main plasma can be very narrow (few mm)

Key Questions:

- How does the scrape-off-layer width (λ_q) scale with plasma parameters?
- How can the severe heat loads reaching the divertor be mitigated?

Edge power and particles are purposely guided away from plasma core using a magnetic "divertor"

NSTX-U will play important role in understanding how power exhaust width extrapolates to future devices



XGC1 simulations predict turbulence will widen edge heat flux in ITER

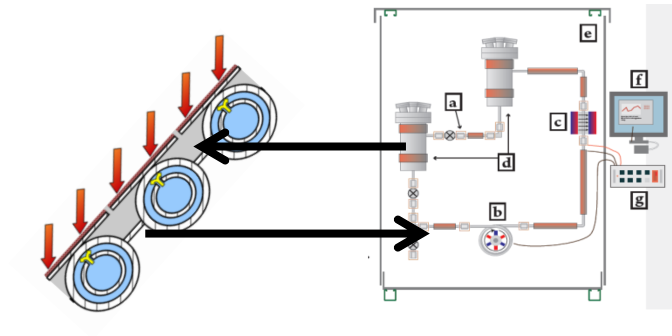
C.S. Chang et al 2017 Nucl. Fusion **57** 116023

XGC1 studies of NSTX-U initiated and ongoing

Preliminary predictions show λ_q ~ 3 mm (unpublished)

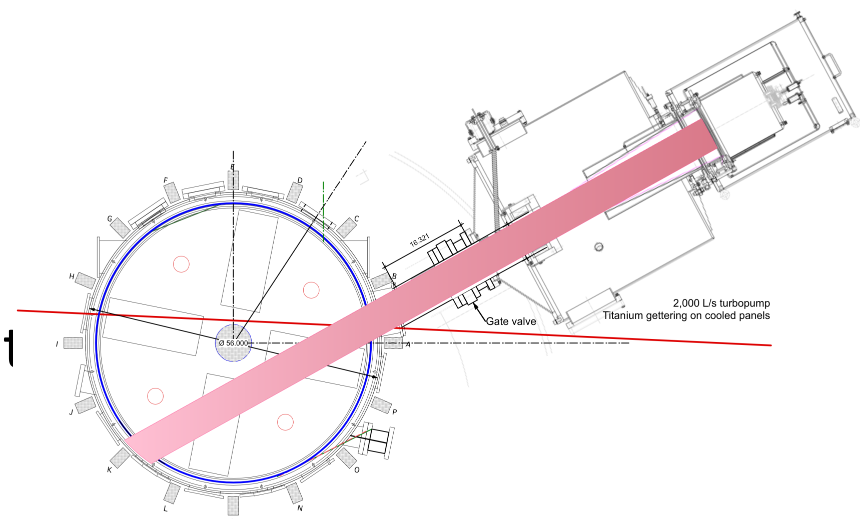
NSTX-U aims to explore liquid metal PFCs

- Need for high heat flux exhaust motivates long term research directions for NSTX-U
  - High-Z tiles with pre-filled Li
  - Flowing liquid lithium divertor with external feed into reservoir



High heat fluxes in NSTX-U (>20 MW/m^2 unmitigated in standard divertors) make NSTX-U an excellent facility for this research

- Part of a liquid metal PFC program at PPPL
  - LTX-β (static liquid Li pools)
  - EAST collaboration (coatings, slow flow liquid lithium limiter)



LTX-β

See talk by Jaworski tomorrow for more details on liquid-metal PFCs

Outline

- Why study "spherical tori/tokamaks" (STs)
  → And why use NSTX-U for that mission.
- Brief Discussion of NSTX-U Operations and Recovery
- Key Scientific Deliverable of the NSTX-U Program
- NSTX-U in the Larger ST/Fusion Context

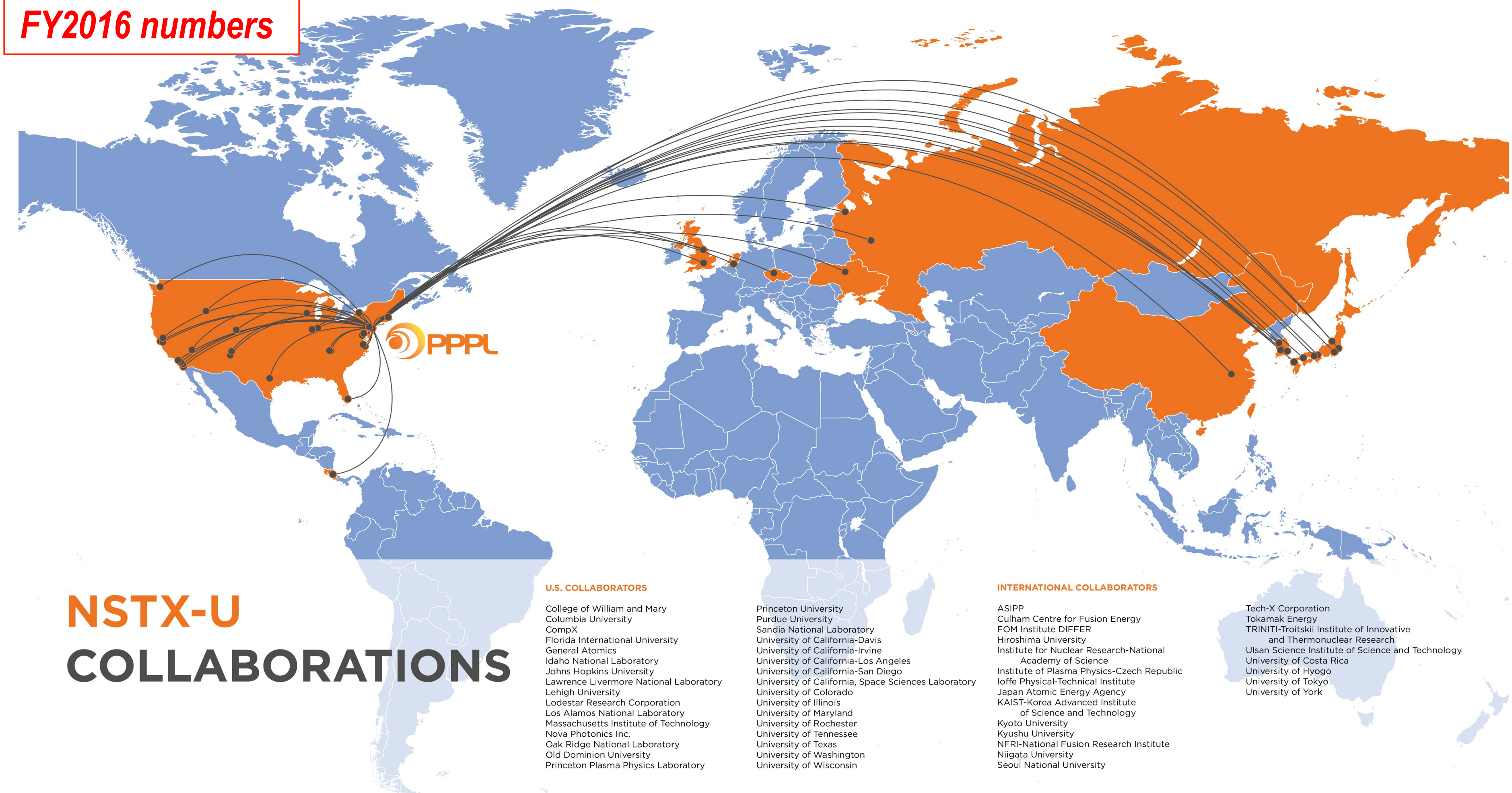
NSTX-U Research is Highly Collaborative, Facility Supports Wide range of Users

362 data users
40 international

54 collaborating institutions
32 domestic, 22 international

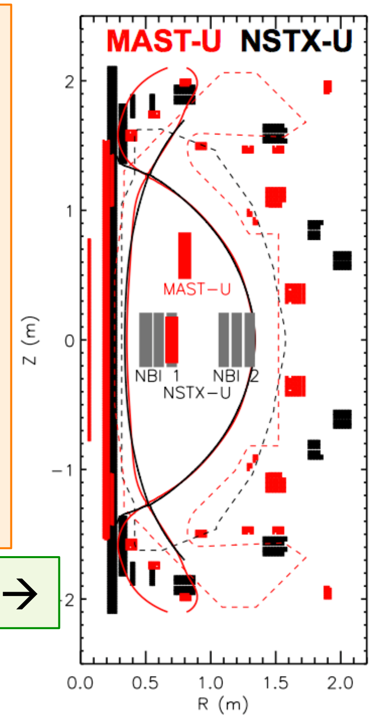
29 graduate students
25 post-doctoral researchers

FY2016 numbers



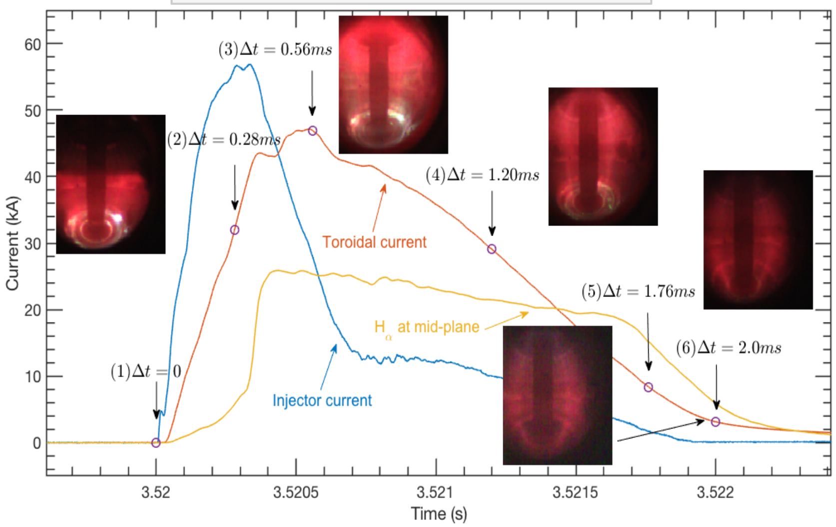
Collaborations Support NSTX-U Mission While Advancing Fusion Capabilities at Other Facilities

- Lithium use and related technologies have spread: EAST dropper, granule injector on EAST and DIII-D, collaborations with AUG and W7-X.
- Solenoid free start-up technologies developed on NSTX being used on QUEST.
- Scenario & equilibrium reconstruction development on MAST-U
- Profile and divertor control development on DIII-D

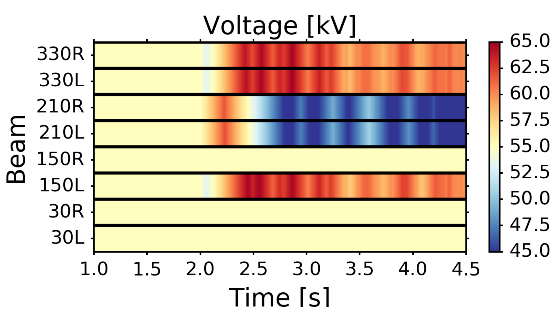


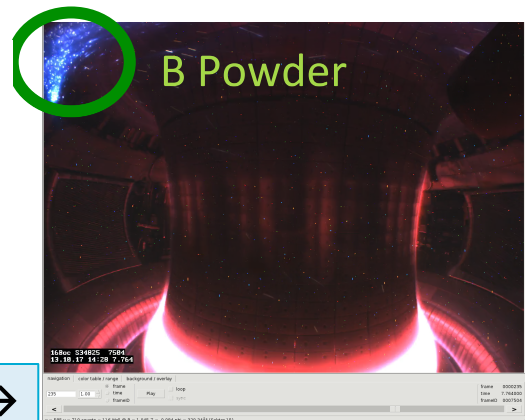
MAST-U and NSTX-U Shape Comparison →

Transient CHI on QUEST



Realtime Control of NB Voltage on DIII-D





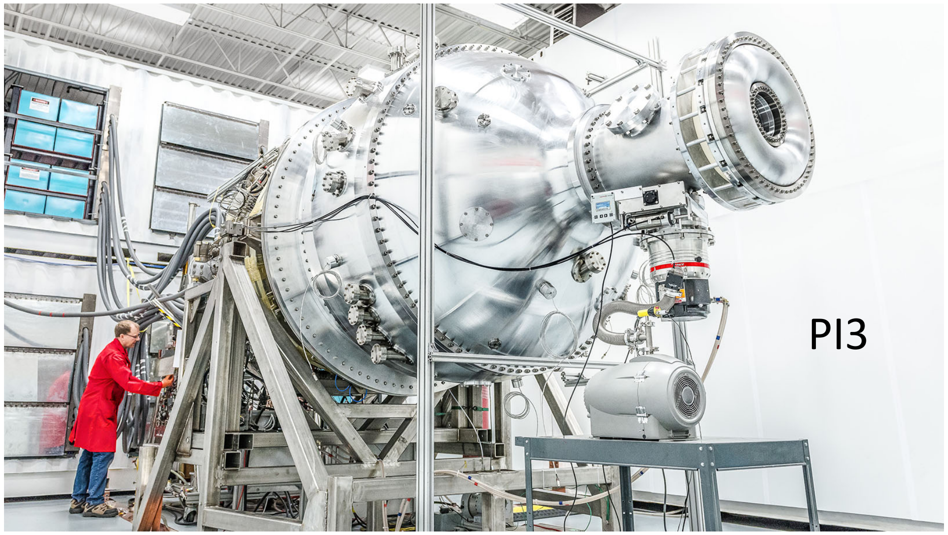
B powder dropping in AUG→

Private Companies are Pursuing ST Research

General Fusion - Canada

Future Plans: Compress plasmas via converging liquid metal liner.

ST is the most stable compact plasma configuration (FRCs, spheromaks, STs…)



PI3

These facilities focus on developing technology using ST platform − synergy with DoE research

Tokamak Energy - UK

ST-40: 2 MA, 3 T, R_0=0.4 m Cu magnets
Merging compression and reconnection formation scheme
Future Plans: HTS development and devices

ST-40

NSTX-U Addresses Key Science Issues from Compact Fusion System

- How will turbulence and transport scale to the low-collisionality regime?
  - →Key research enabled by increased magnetic field of the Upgrade
- Can the plasma current be sustained 100% non-inductively at fusion relevant β and with high confinement?
  - →Key research enabled by increased magnetic field and 2nd neutral beam
  - →Nearby conducting plates facilitate key low-A RWM stability research
- How do energetic particle stability and confinement properties trend into the burning plasma regime?
  - →2nd neutral beam and upgraded of TF allows fast-ion phase space to be comprehensively explored
  - →Can span burning plasma regimes at both low- and conventional A
- How can the power exhaust be handled in compact fusion systems?
  - →Exploring innovations in flared divertors and lithium systems for handling large divertor heat fluxes.

Summary

- NSTX-U contributes to optimization of the tokamak concept though optimization of aspect ratio

  → Potentially transformational behavior in STs when approaching fusion relevant plasma parameters
- NSTX-U is a Key Component of the PPPL Programmatic Strategy Towards Fusion Power

  → Supports our vision for innovation in compact high-β configurations with lower capital costs, materials innovation
- NSTX-U has a unique role within the world ST program

  → Unique focus on core physics & impact of Li on plasma behavior
- NSTX-U will provide answers to critical fusion science questions early in its operation.

  → Exciting science from the commissioning campaign