U.S. DEPARTMENT OF ENERGY | Office of Science

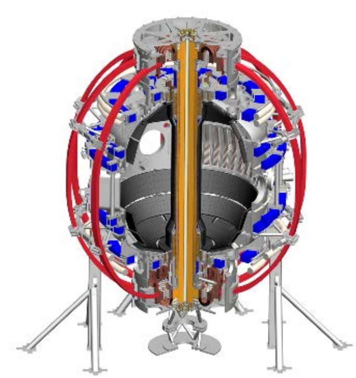
NSTX-U

Scientific Opportunities & Challenges for the upgraded National Spherical Torus eXperiment - NSTX-U

Jon Menard, PPPL

NSTX-U Program Director

Graduate Student Seminar
PPPL theory conference room
February 29, 2016



PPPL

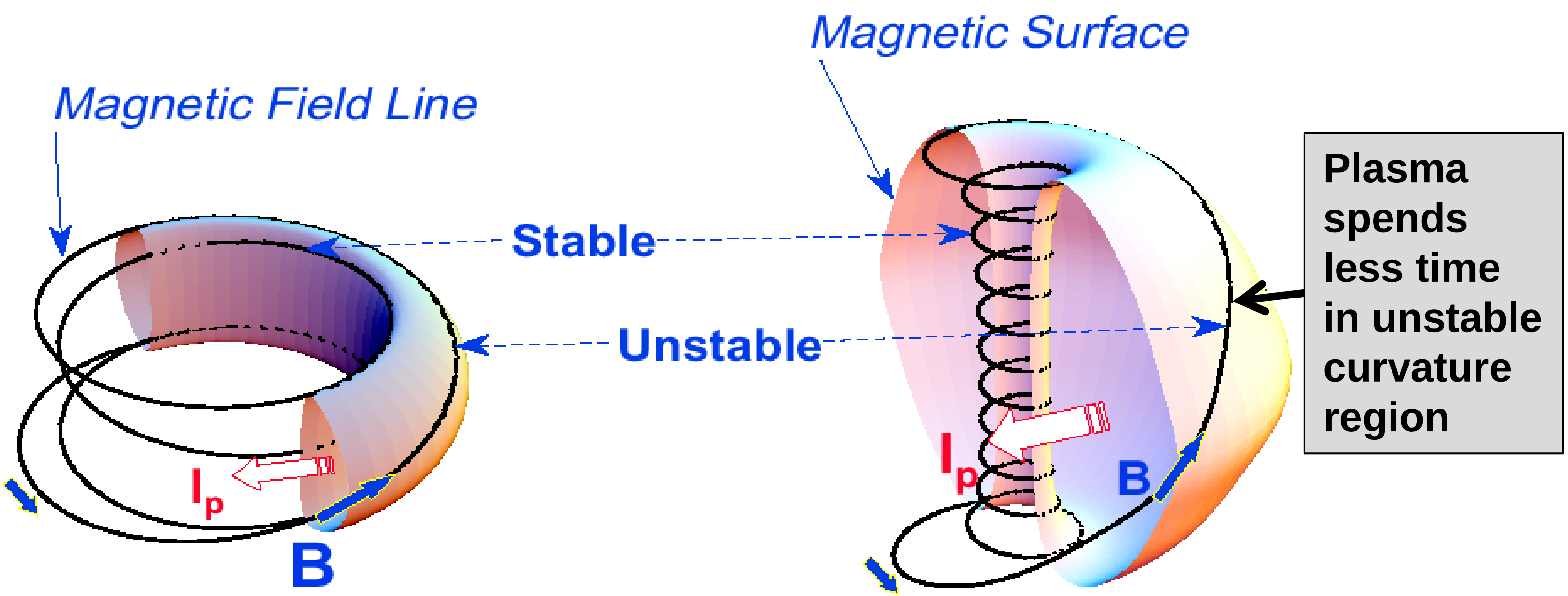
Outline

- Why study spherical tori / tokamaks?
- Mission and status of NSTX-U
- Results, opportunities, challenges

Tokamak aspect ratio is important free parameter

Favorable average curvature of ST improves stability

Conventional Tokamak | **“Spherical” Tokamak (ST)**



Conventional Tokamak:

$A \sim 3\text{-}5$

$\kappa = 1.5\text{-}2$

$\beta_T = 3\text{-}10\%$

Spherical Tokamak (ST):

$A \sim 1.1\text{-}2$

$\kappa = 2\text{-}3$

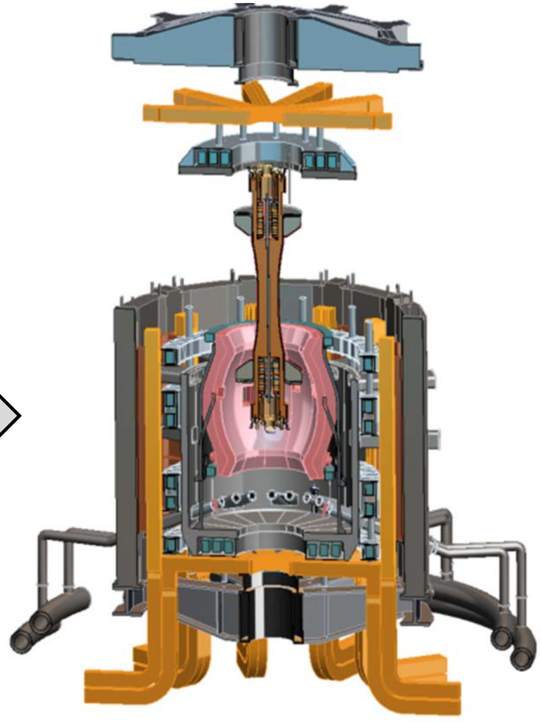
$\beta_T = 10\text{-}40\%$

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

Design studies show ST potentially attractive as Fusion Nuclear Science Facility (FNSF)

- FNSF can help develop reliable fusion nuclear components
  - Substantial integrated R&D, testing needed to develop components
  - An FNSF facility should be: modest cost, low T, and reliable

- ST-FNSF projected to access high neutron wall loading at moderate size and fusion power
  - $W_n \sim$ 1-2 MW/m^2 , R ~ 0.8-1.8m, $P_{fusion} \sim$ 50-200MW
- Modular, simplified maintenance
- Tritium breeding ratio (TBR) ≈ 1
  - Using only/primarily outboard breeding requires sufficiently large R, careful design

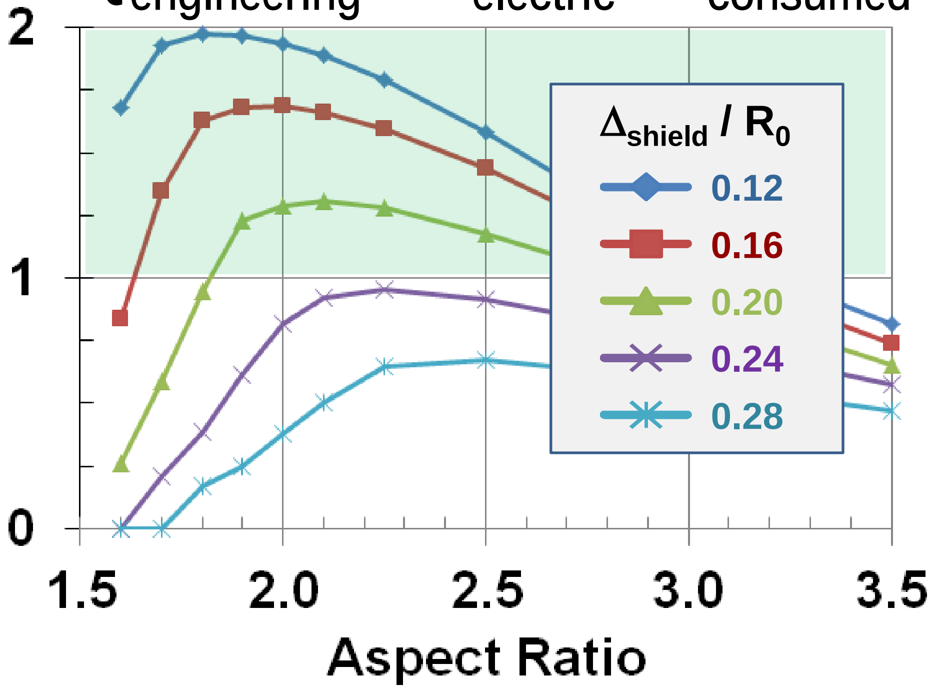


PPPL ST-FNSF concept

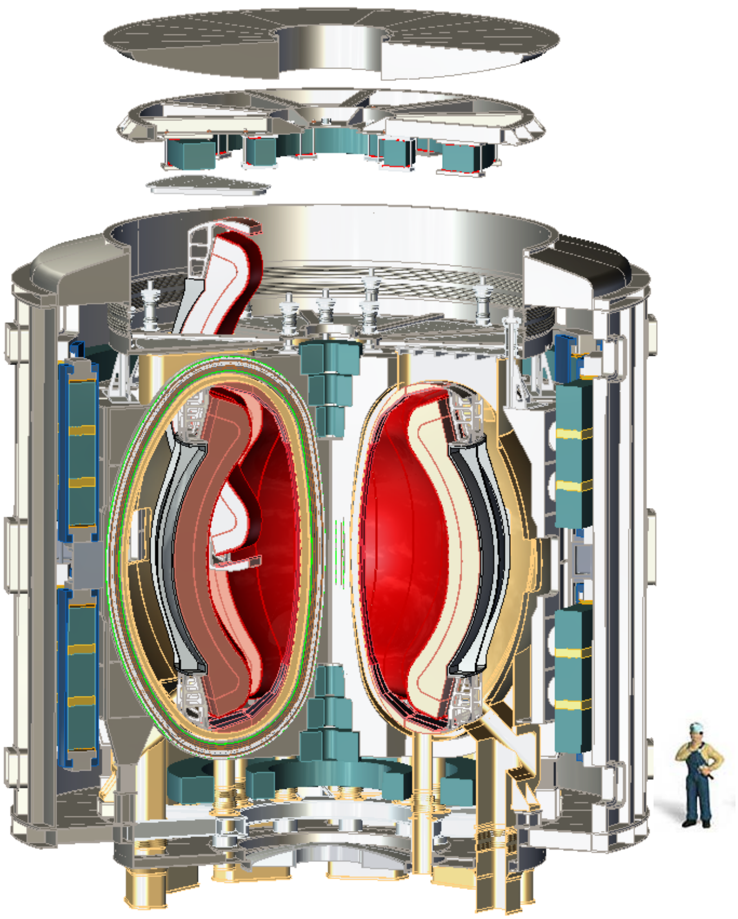
High Temperature Superconductors (HTS) attractive for ST*
(~10× lower magnet cooling power vs. Cu, less thermal shielding required)

Net electricity easiest near A=2 if:
- Inboard T breeding minimized / eliminated
- Central solenoid minimized / eliminated

$$Q_{engineering} = P_{electric} / P_{consumed}$$



Magnet lifetime increases exponentially with Δ_{shield}



Work supported by Tokamak Energy (UK) - 2014

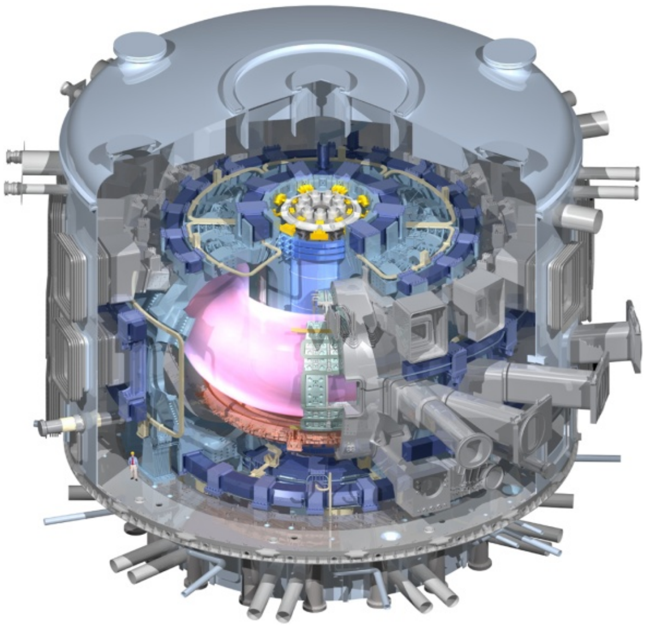
R_0 = 1.4m, B_T = 3.2-4T, I_P = 7-8MA, $P_{fusion\text{-}DT}$ = 100MW
H_{98y2} = 1.7-2, H_{ST} ~ 1, A = 1.8-2, κ ~ 2.5-2.7, δ ~ 0.5

Unique ST properties also support ITER

ST Extends Predictive Capability for ITER and Toroidal Science

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

Burning Plasma Physics - ITER



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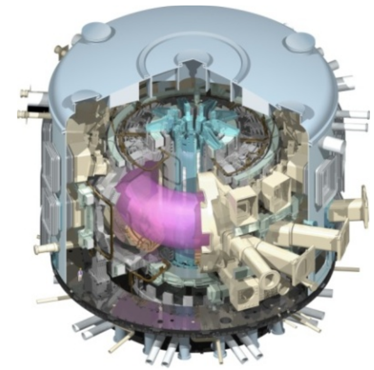
NSTX-U = National Spherical Torus eXperiment - Upgrade

Mission Elements:

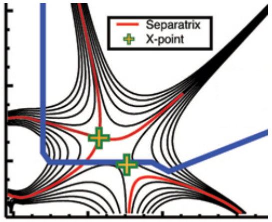
- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond



ITER

- Develop solutions for the plasma-material interface (PMI) challenge

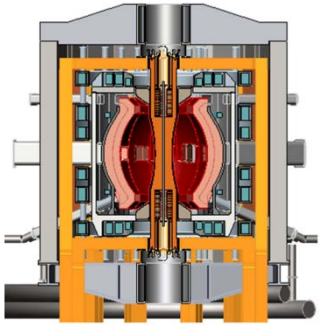
Liquid metals / Lithium



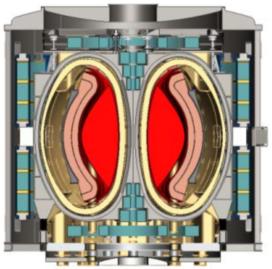
"Snowflake"

- Advance ST as candidate for Fusion Nuclear Science Facility (FNSF)

- Develop ST as fusion energy system

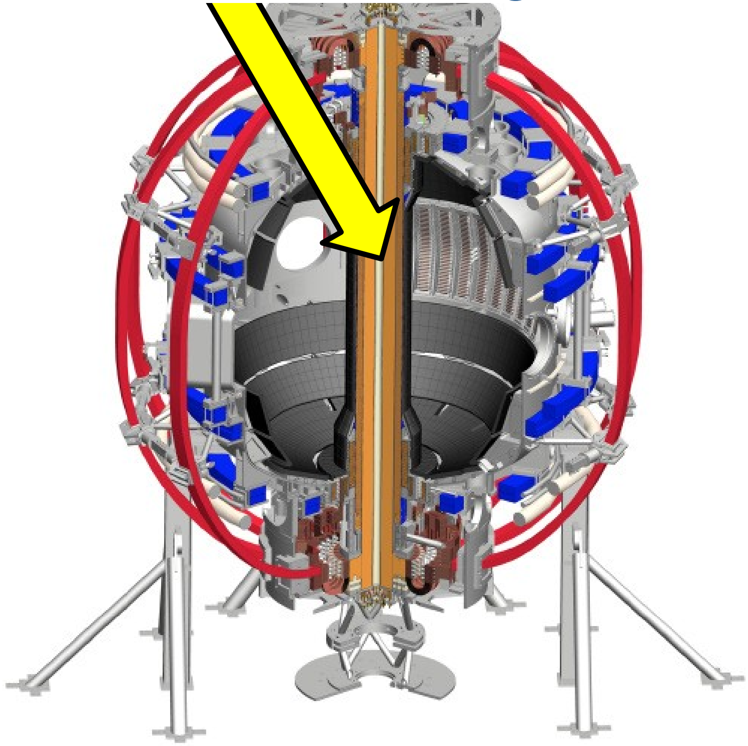


ST-FNSF / Pilot-Plant

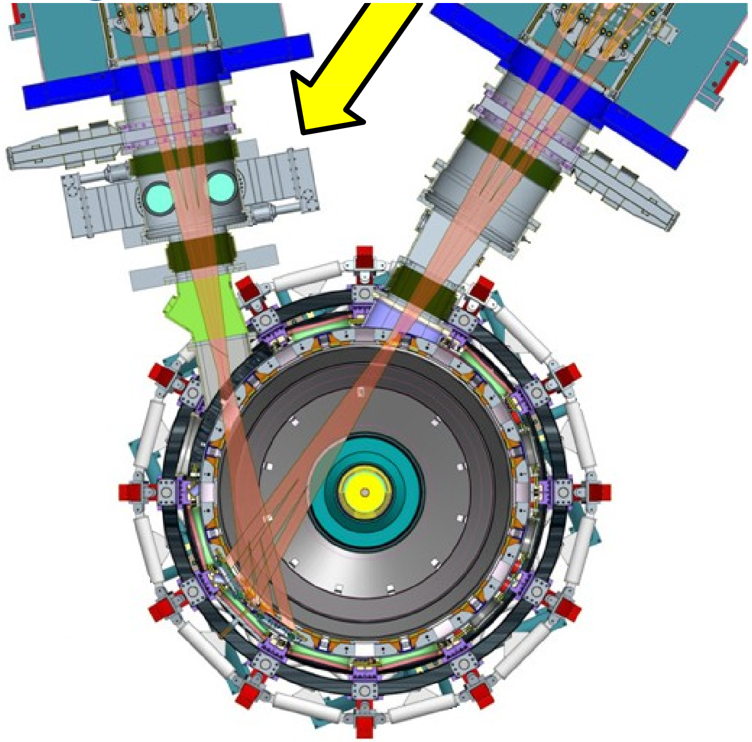


NSTX-U will access new physics with 2 major new tools:

1. New Central Magnet



2. Tangential 2nd Neutral Beam



Higher T, low collisionality at high β

→ Unique regime, study new transport and stability physics

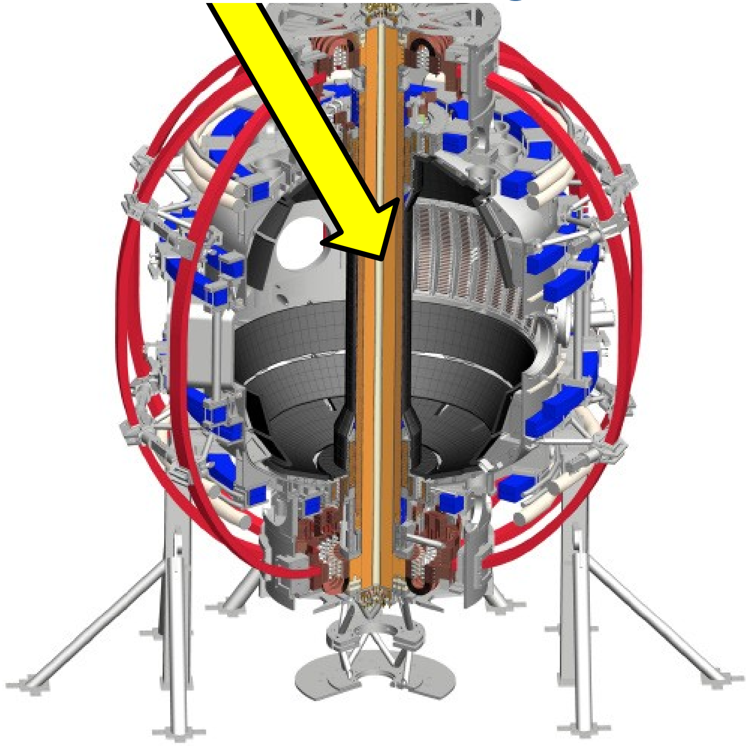
Full non-inductive current drive

→ Not demonstrated in ST at high-β_T

Essential for any future steady-state ST

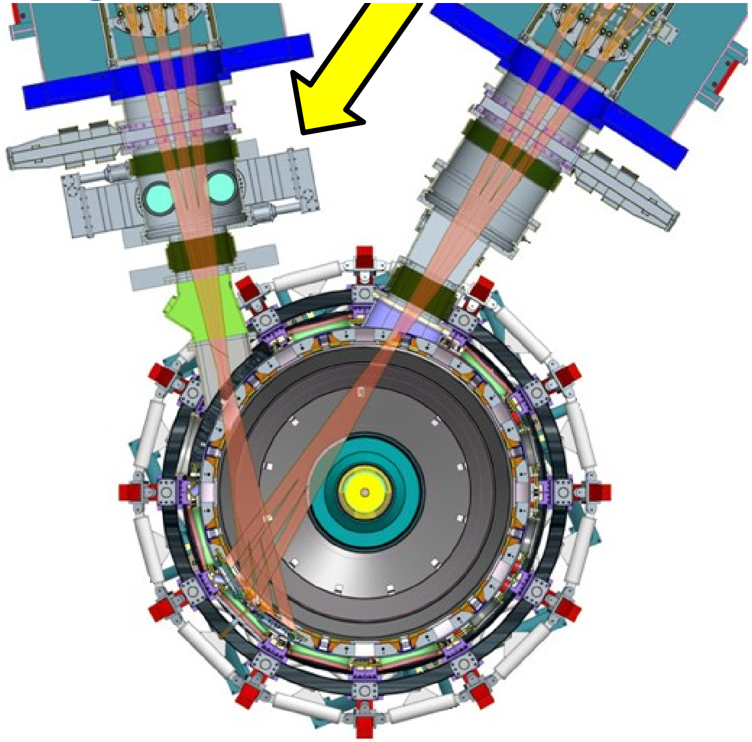
NSTX-U will have major boost in performance

1. New Central Magnet



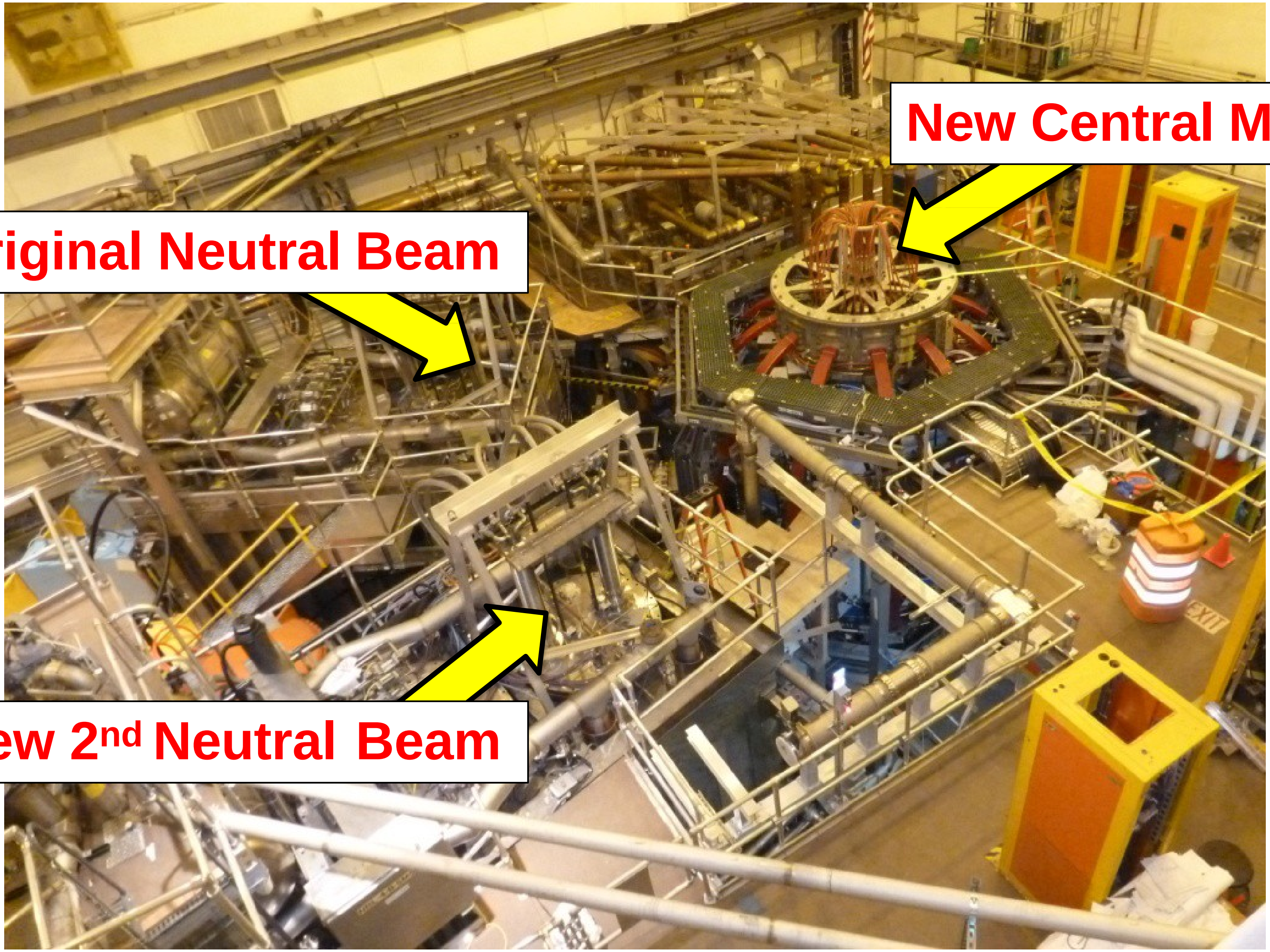
- ➢2× toroidal field (0.5 → 1T)
- ➢2× plasma current (1 → 2MA)
- ➢5× longer pulse (1 → 5s)

2. Tangential 2^{nd} Neutral Beam

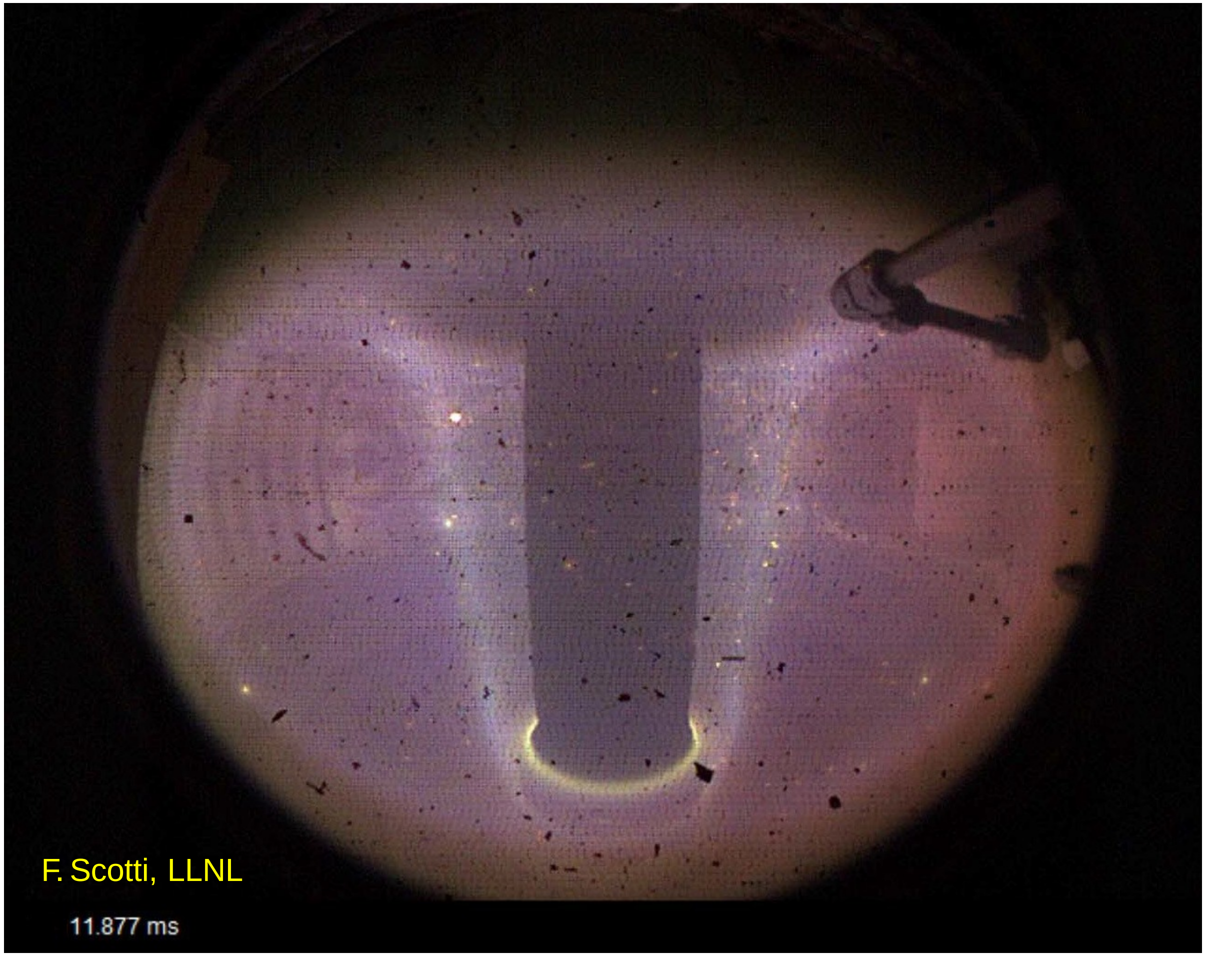


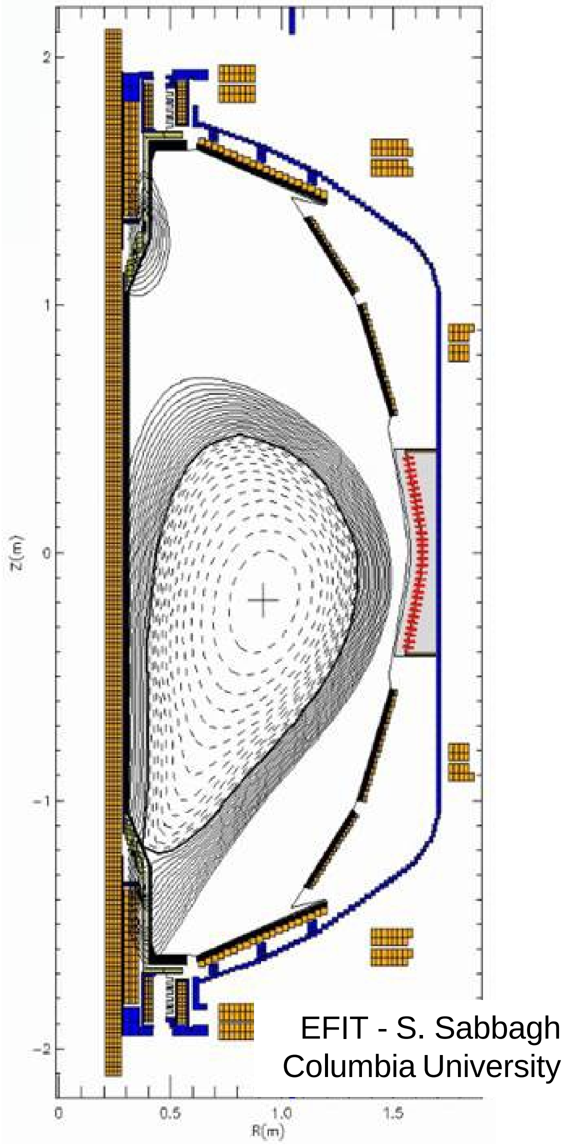
- ➢2× heating power (5 → 10MW)
  - Tangential NBI → 2× current drive efficiency
- ➢4× divertor heat flux (→ ITER levels)
- ➢Up to 10× higher $nT\tau_E$ (~MJ plasmas)

NSTX Upgrade project complete



Achieved 110kA test plasma August 10, 2015

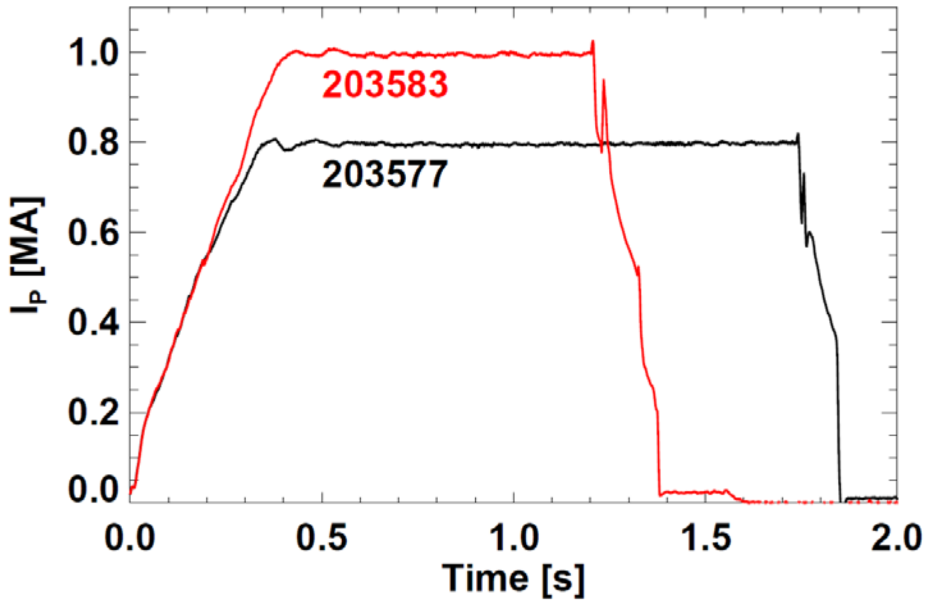




NSTX-U is now an operating facility

Plasma commissioning progressing rapidly thus far

- Completed wall conditioning (bake-out, boronization)
- Began operation in late December
- **Routinely making ~1-2s plasmas up to 1MA**

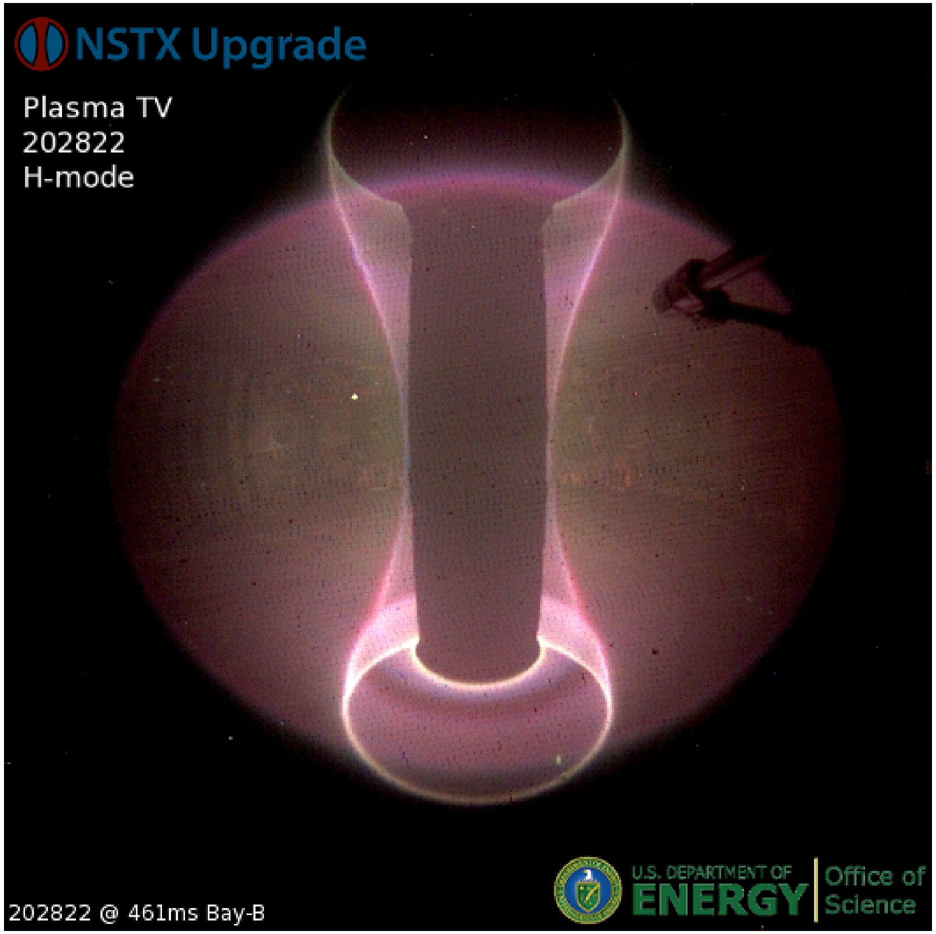


B_T already above highest NSTX B_T

- Optimizing control
  - I_P, gap, elongation κ
- Increasing κ, P_{NBI}
- Ongoing / next steps:
  - Optimize H-mode / timing
  - Measure/correct error fields
  - Access(ing) NSTX-levels of performance, then surpass

Plasma operations on NSTX-U is off to a great start – come join us!

- First days of NSTX-U produced H-mode and stationary diverted L-mode discharges
- Control and diagnostic capabilities established quickly
- Research program starting in parallel with commissioning activities



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NSTX-U Mission Elements
5 Highest Research Priorities

- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond
  1. Understand energy confinement at high normalized pressure (high β)
  2. Study energetic particle physics prototypical of burning plasmas

- Develop solutions for PMI challenge
  3. Dissipate high edge heat loads using expanded magnetic fields + radiation
  4. Compare performance of solid vs. liquid metal plasma facing components

- Advance ST as possible FNSF / Pilot Plant
  5. Form and sustain plasma current without transformer for steady-state ST

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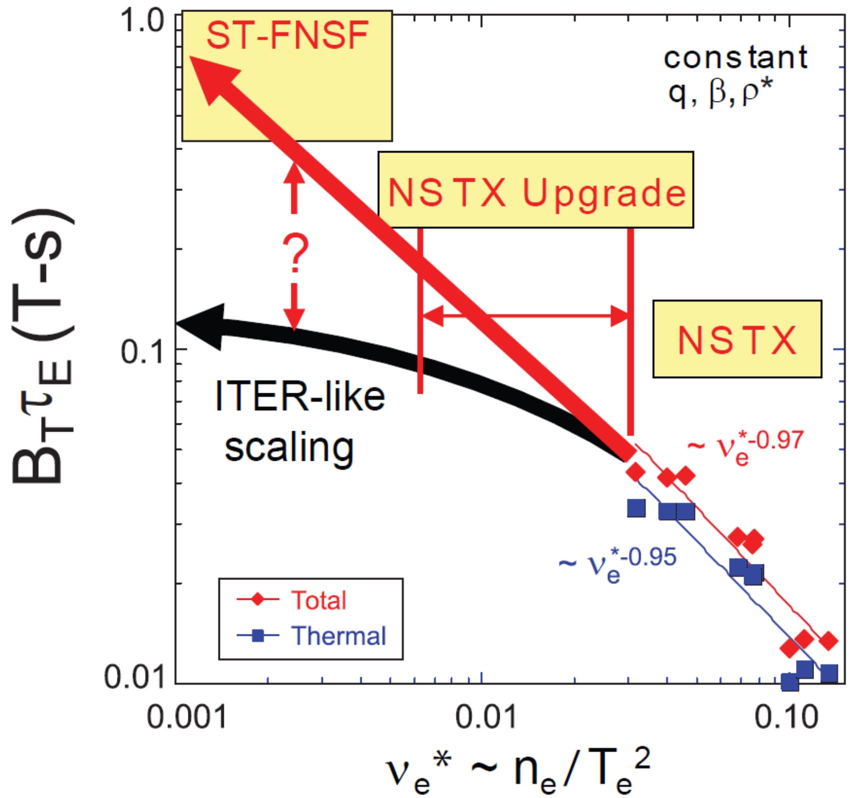
Favorable confinement trend with collisionality and β found in ST experiments

$\tau_{E,\,th} \propto \nu_{*e}^{-0.8}\ \beta^{-0.0}$	**ST scaling observed in NSTX and MAST**
$\tau_{E,\,th} \propto \nu_{*e}^{-0.1}\ \beta^{-0.9}$	**tokamak empirical scaling (ITER 98y,2)**

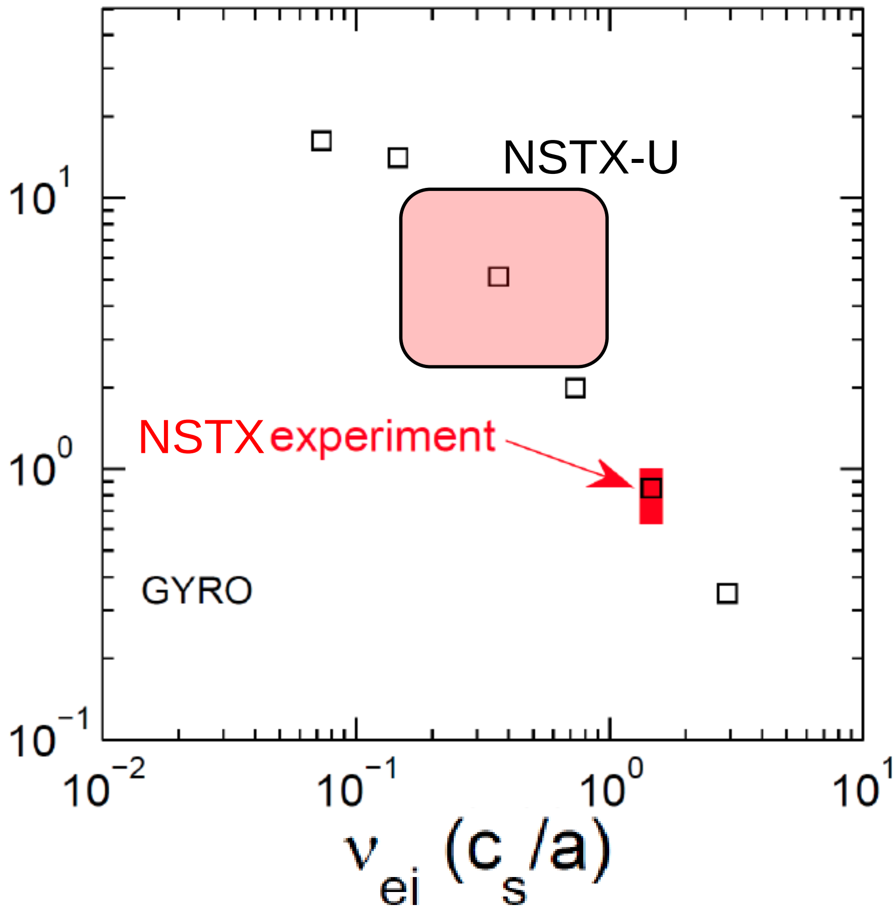


Promising scaling to ST-FNSF / Pilot, will trend continue on NSTX-U / MAST-U?

Electromagnetic effects may play important role in collisionality scaling of ST confinement

Micro-tearing $\tau_{E\text{-}e}$ vs. ν_{ei} similar to experiment

χ_e^{-1} $(a/\rho_s^2 c_s)$ $\propto \tau_{E\text{-}e}$



- High β → small-scale overlapping tearing modes → transport from magnetic turbulence (GYRO – W. Guttenfelder)
- Other electrostatic turbulence may also have similar ν^* scaling: Dissipative Trapped Electron Mode (GTS - W. Wang)

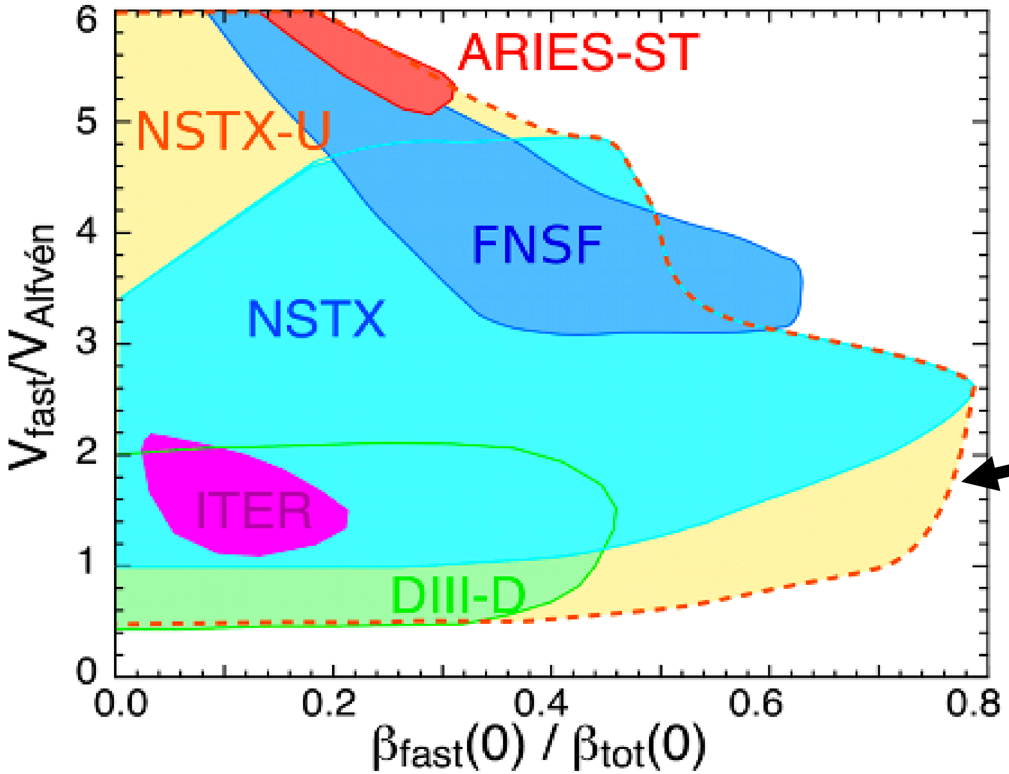
Will NSTX-U observe confinement improvement at lower collisionality?

Research opportunities

- What are the leading causes of anomalous electron transport in the ST, and tokamaks generally?
  - Will electrostatic turbulence (ETG, TEM) dominate over micro-tearing ($\chi_e \sim \nu$) at lower collisionality values of NSTX-U?
- Will ion thermal and impurity transport remain predominantly neoclassical?
  - Neoclassical diffusivities also scale $\sim \nu$

NBI-heated STs excellent testbed for α-particle physics

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

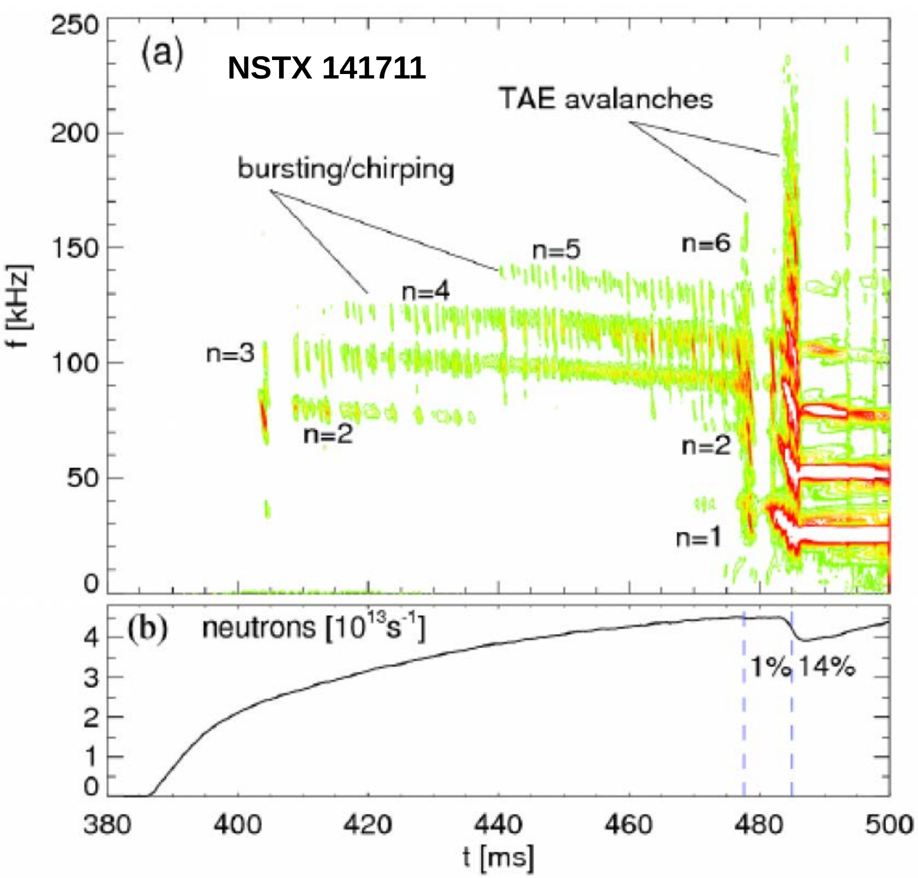


- NSTX-U: large fast-ion dynamic range spanning ST and conventional A
  - **Toroidal field 2× NSTX →** $V_{fast} < V_A$ → stabilize modes
  - **Tangential 2nd NBI →** very flexible fast-ion distribution
    - Vary pitch angle, pressure profile

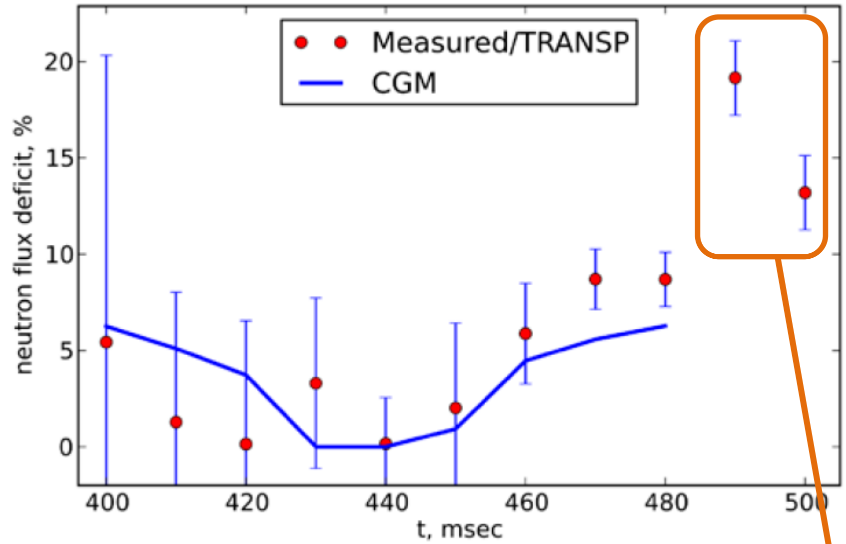
Can we find TAE-quiescent, high-performance regimes in NSTX-U?

"TAE avalanche" can cause energetic particle loss

Uncontrolled α-particle loss could cause reactor first wall damage



- Quasi-linear "Critical Gradient Model" (CGM) consistent with transport before avalanche



- Working towards fully non-linear multi-mode simulations (e.g. M3D-K) for avalanche phase

Research opportunities

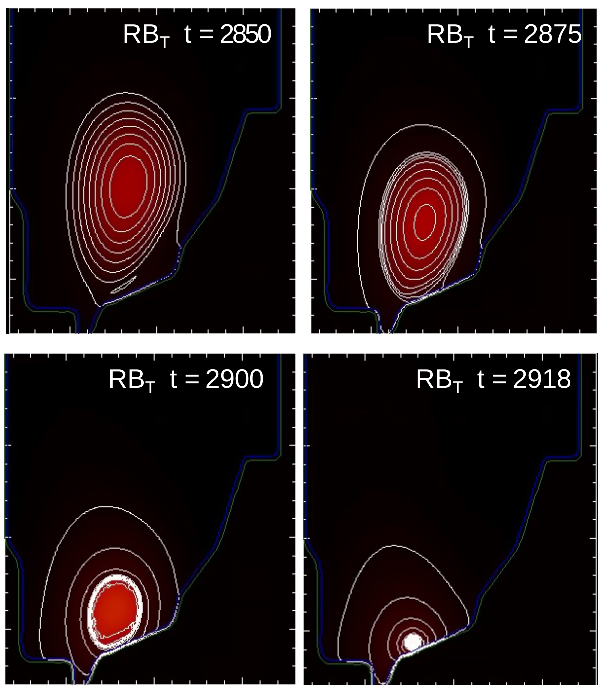
- Will reduced drive for fast-ion instabilities in NSTX-U reduce anomalous fast-ion transport?

- How will NSTX-U plasmas respond to more tangential neutral-beam / fast-ion injection?
  - Are current and momentum drive consistent with theory?

- What is role of high-f fast-ion instabilities (GAE/CAE) in core electron transport?
  - Is this physics unique to high-beta/ST?

NSTX-U aims to play leading role in understanding halo current dynamics, disruption mitigation physics

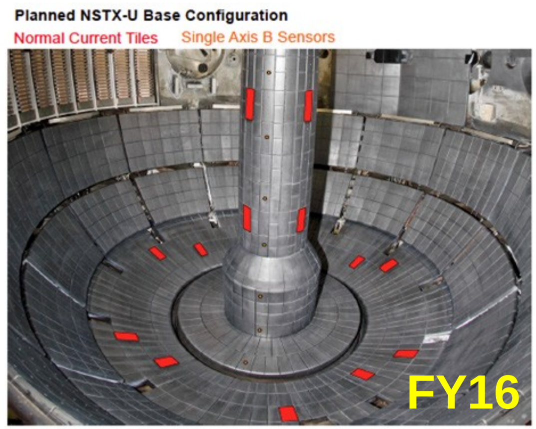
- Advanced non-linear MHD modelling of vertical displacement events (VDE) + halo currents with M3D-C[1]

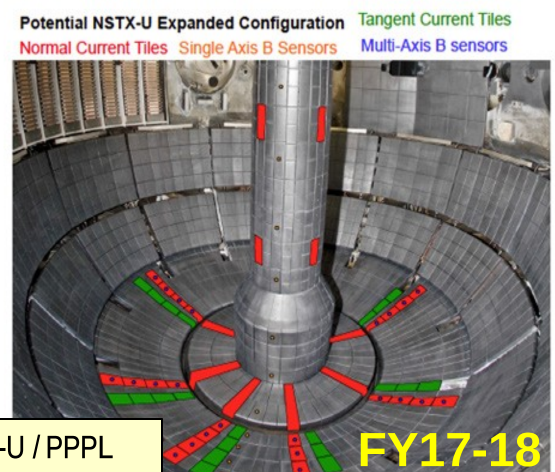


NSTX Discharge 132859

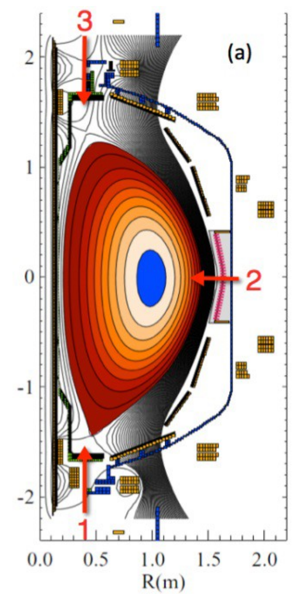
NSTX-U / PPPL Theory Partnership

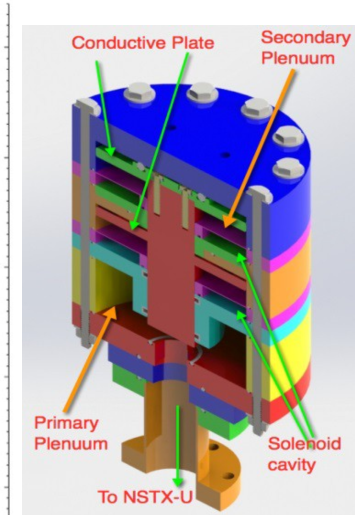
- Enhance measurements of halo-current dynamics





- Test ITER-like Massive Gas Injection (MGI) valves
  - Test poloidal dependence of density assimilation
  - First data expected FY16

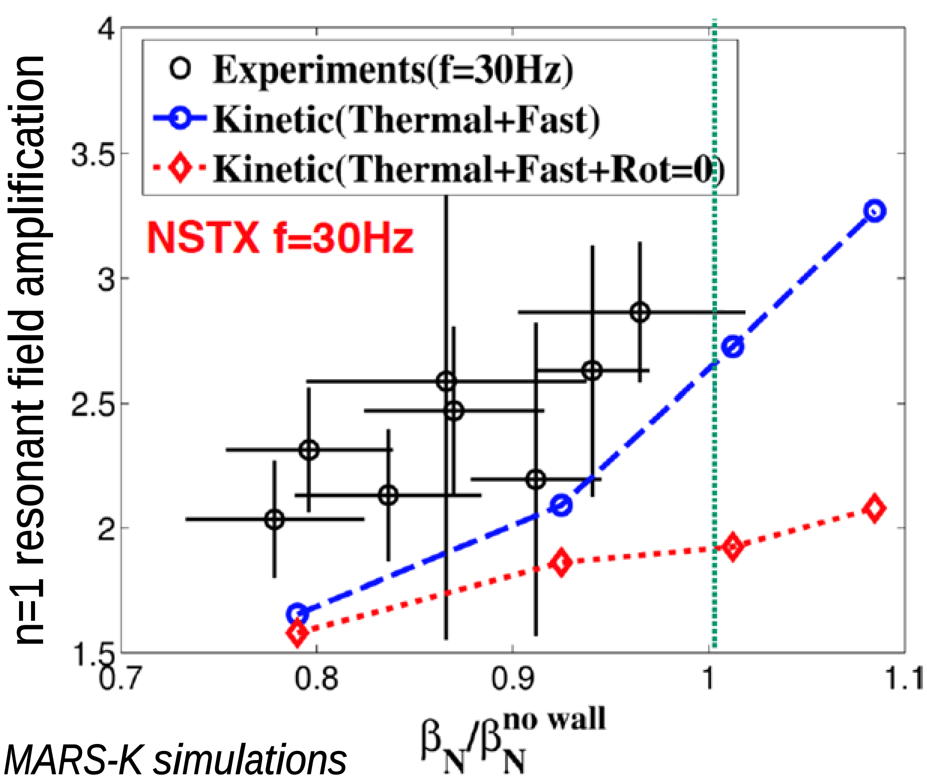




University of Washington

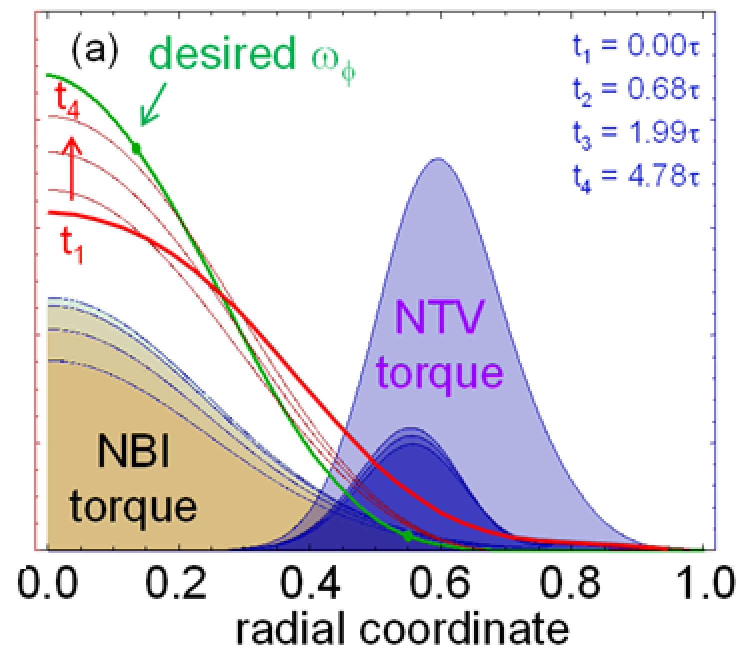
NSTX-U will play key role in understanding stability limits, developing disruption avoidance

- Developing understanding of kinetic effects in MHD stability
  - NSTX: Drift-kinetic damping, fast-ions, rotation all important



MARS-K simulations

- Developing advanced rotation control to optimize performance
  - Actuators: Beams, 3D fields (NTV)



Princeton, Columbia, PPPL collaboration

Research opportunities

- What are leading causes of disruptions in ST? and what are implications for next steps?
  - NSTX-U will implement advanced profile and instability controls – how much does this reduce disruptivity?
  - NSTX-U will perform in-depth study/diagnosis of “halo/hiro” currents, i.e. edge plasma and wall currents and forces induced by disruptions

- How do RWM stability and rotation damping from neoclassical toroidal viscosity (NTV) change at the lower (ultimately up to 10x lower) collisionality values of NSTX-U?
  - Enhanced kinetic damping for RWM?
  - Enhanced rotation damping in $1/\nu$ regimes?

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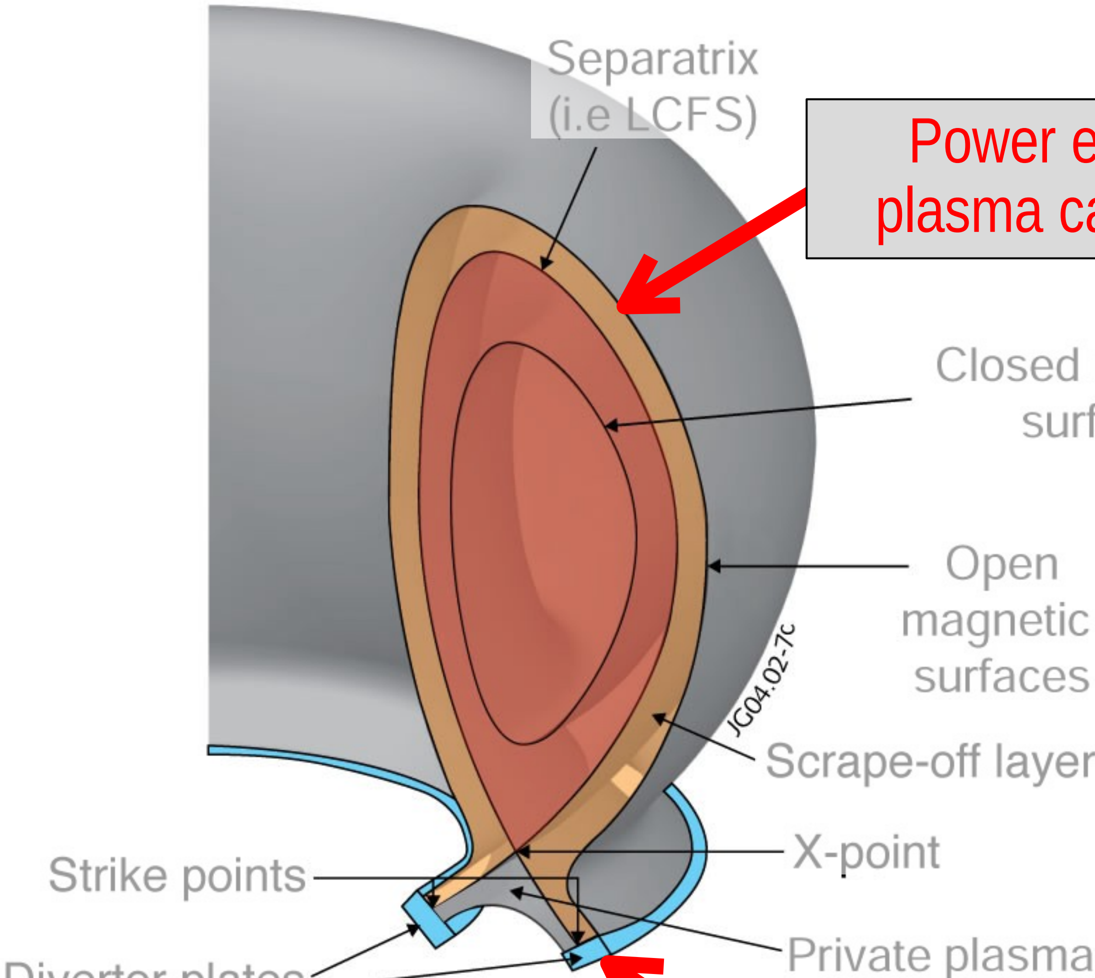
All modern tokamaks / STs use a “divertor” to control where power and particles are exhausted

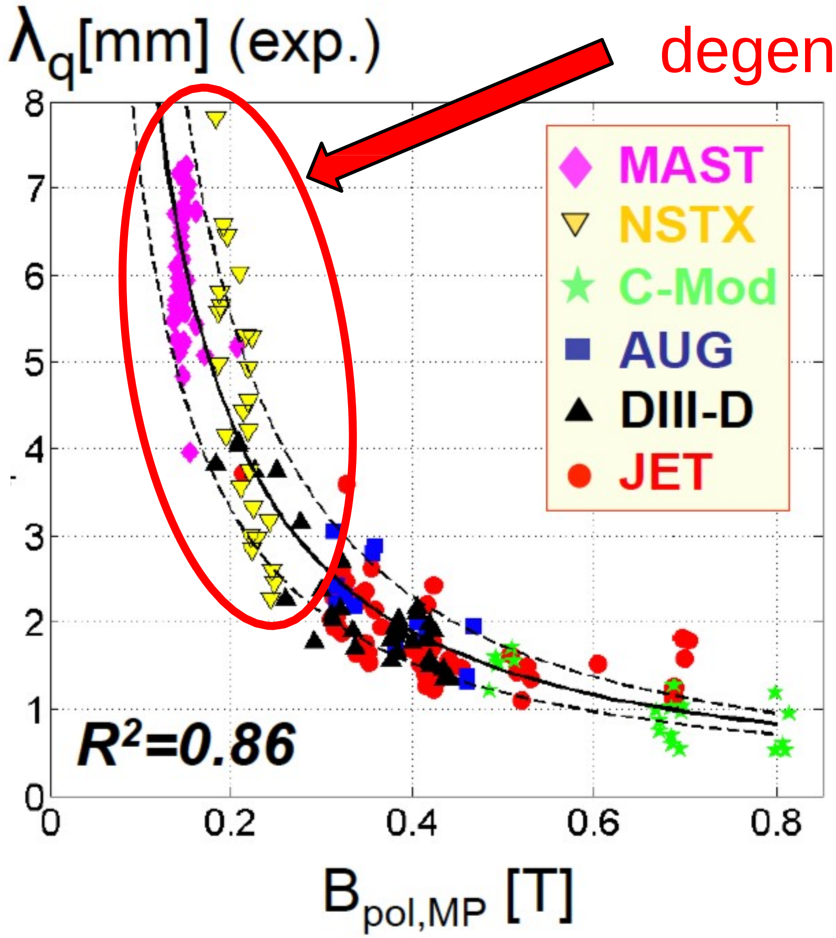


Power exhaust width outside main plasma can be very narrow (few mm)

Power / particles contact target plates here

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

ST data breaks aspect ratio degeneracy of data set

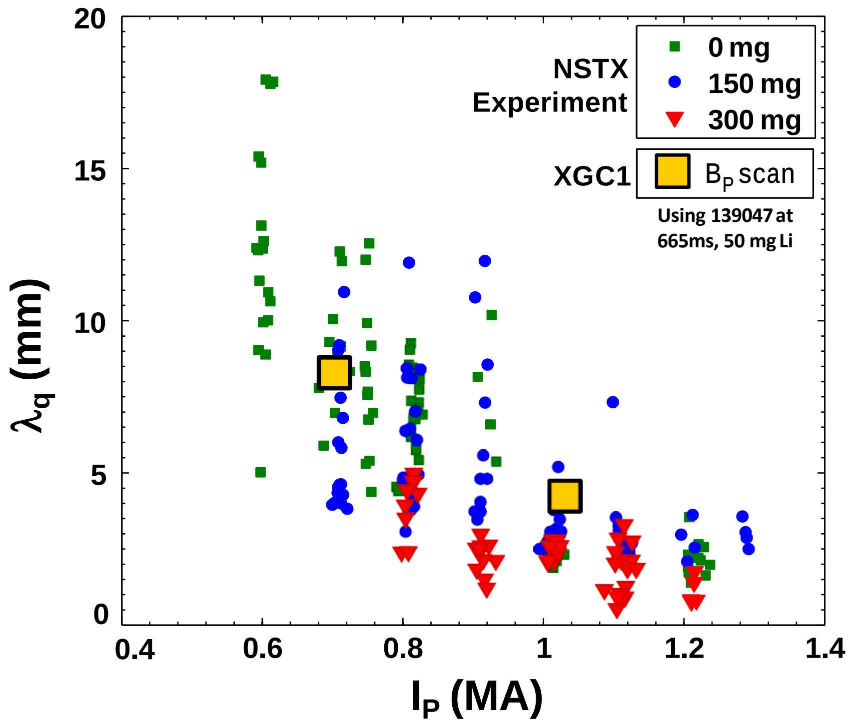


Will 1/$B_{poloidal}$ variation continue at higher I_P? What about detached conditions?

XGC1 simulations aiding in understanding of SOL heat flux width trends in NSTX

- Experiment shows contraction of SOL heat flux width at midplane with I_p as well as influence of Li conditioning

XGC1 w/ collisions → similar trends



Heat flux width determined primarily by neoclassical processes

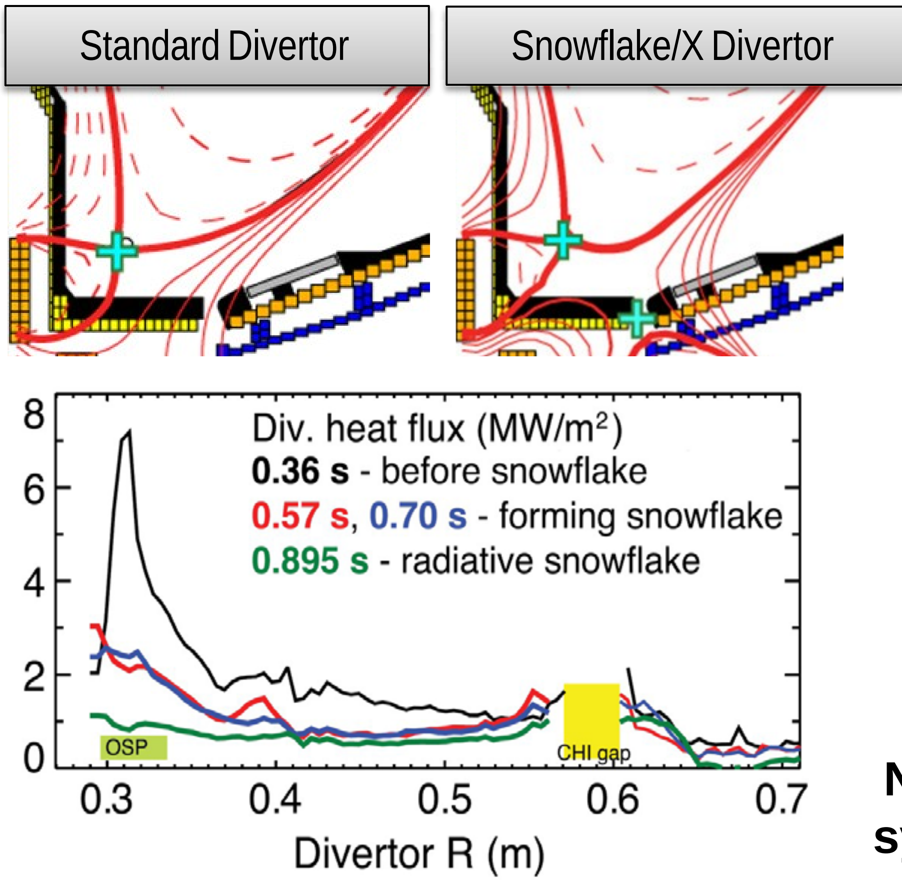
XGC-1:

- Full-f, global PIC, kinetic ions, fluid electrons *(kinetic electrons under development)*
- Good candidate for exascale computing initiative

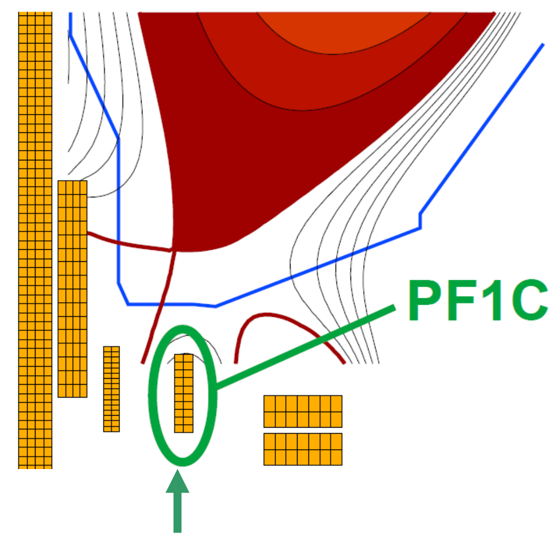
NSTX-U / PPPL Theory Partnership

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

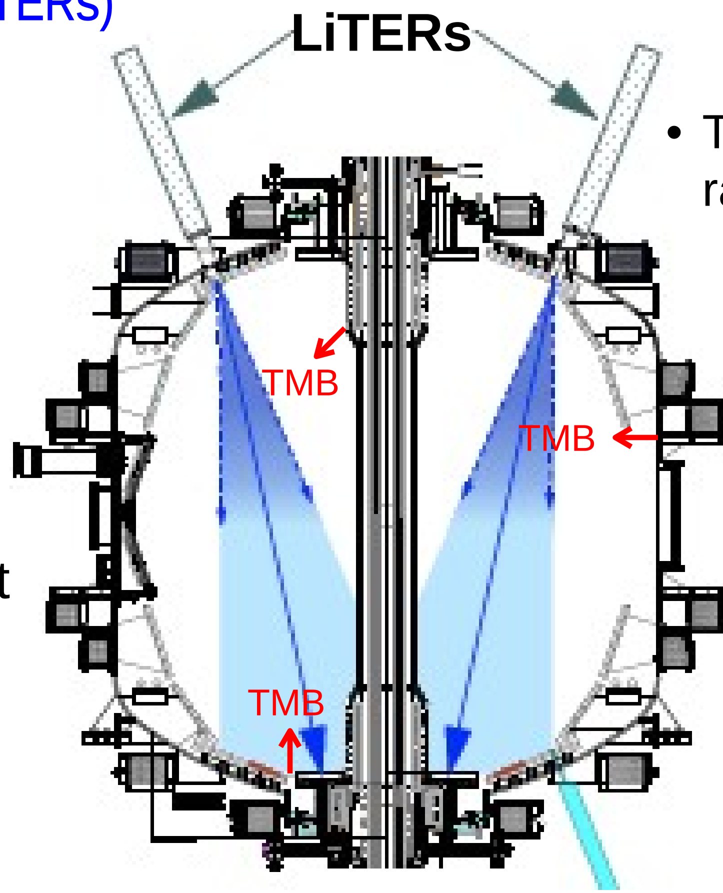
Research opportunities

- What sets heat flux width in open field line region?
  - Mostly drifts + collisions (i.e. neoclassical) or does turbulence play a role?
- How does divertor geometry (such as snowflake) influence the plasma edge properties?

NSTX-U will employ multiple particle control tools to access long-pulse scenarios

2: Lithium Evaporator (LiTERs)

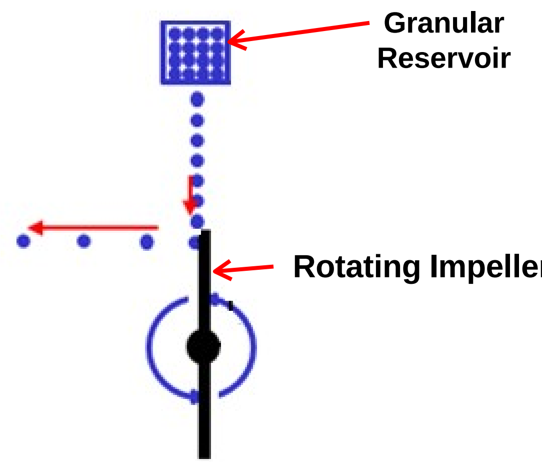
- Pumps deuterium
- Improves confinement
- Often stabilizes ELMs
  → can cause impurity accumulation



1: Boronization (TMB)

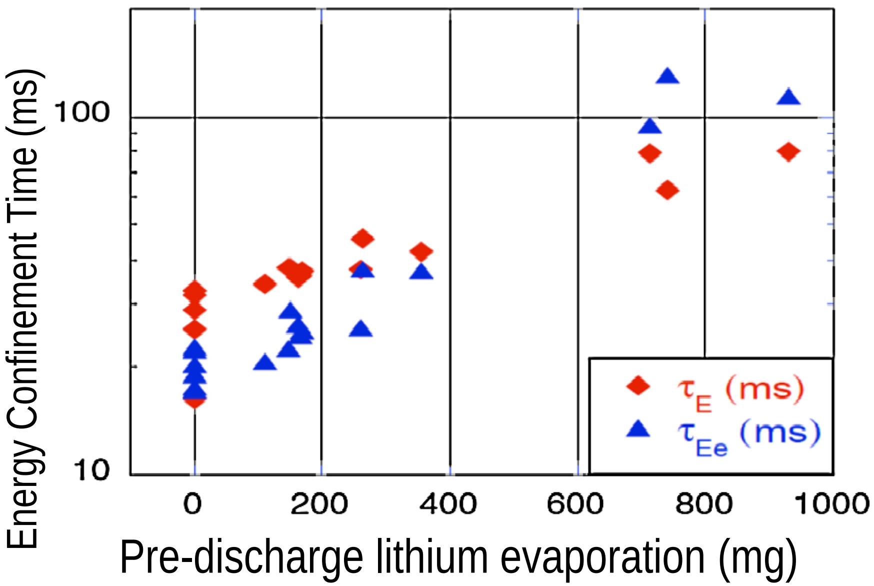
3: Granule injector (GI) for ELM pacing

- Test during FY16 run to trigger rapid ELMs, flush impurities



- Successfully tested on EAST and DIII-D
- NSTX-U will test several granule types: Li, B_4C, C
- $f_{injection}$ ~ up to 500 Hz

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



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

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

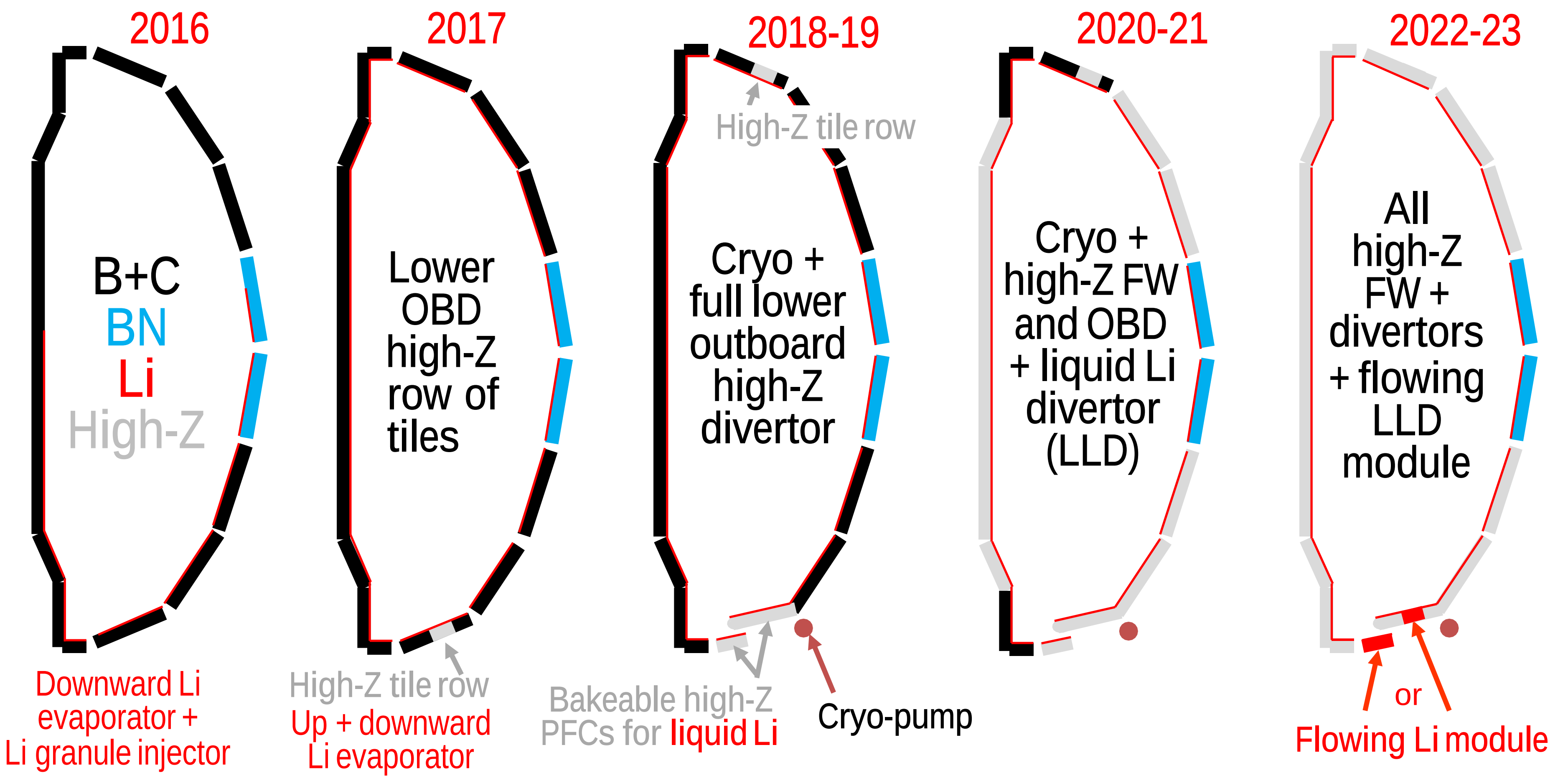
- NSTX-U will double Li-wall coverage with upward evaporators
- Will further assess contributors to confinement improvement:
  - Lower-recycling / reduced neutral source / higher T_e
  - Edge profile / turbulence changes
  - Influence of (low-Z) impurities in pedestal region

Research opportunities

- H-mode pedestal profiles determine overall fusion performance…. But what are dominant transport mechanism(s) in the H-mode pedestal region?
  - Neoclassical, kinetic ballooning, electron ∇T (ETG), other?
  - Ahmed Diallo's Early Career Award with "burst" Thomson Scattering will help determine fast time evolution, transport
- Lithium coatings substantially increased NSTX global confinement (1.4x ITER H) – important for next-steps
  - How high can we make energy confinement in NSTX-U?
  - 2x ITER H-mode scaling? What sets the limit?

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

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



Research opportunities

- How do Li and high-Z PFCs modify the edge region, and also the core performance?
  - NSTX-U will study "vapor shielding" regime to improve understanding of Li evaporation + radiation and impact on power exhaust / heat-flux mitigation / core performance

- Are liquid metals / Li effective and practical for mitigating plasma power exhaust?

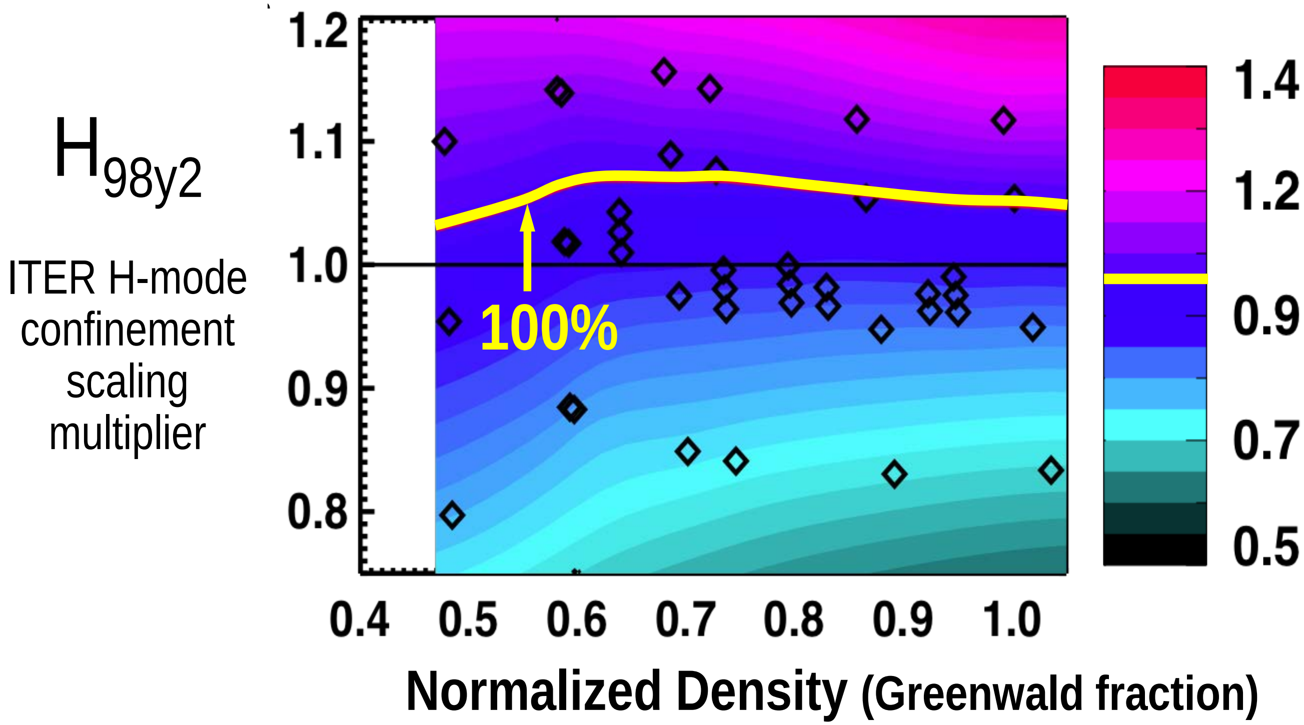
NSTX-U Mission Elements
5 Highest Research Priorities

- Explore unique ST parameter regimes to advance predictive capability - for ITER and beyond
  1. Understand energy confinement at high normalized pressure (high β)
  2. Study energetic particle physics prototypical of burning plasmas
- Develop solutions for PMI challenge
  3. Dissipate high edge heat loads using expanded magnetic fields + radiation
  4. Compare performance of solid vs. liquid metal plasma facing components
- Advance ST as possible FNSF / Pilot Plant
  5. Form and sustain plasma current without transformer for steady-state ST

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

NSTX achieved 70% "transformer-less" current drive

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

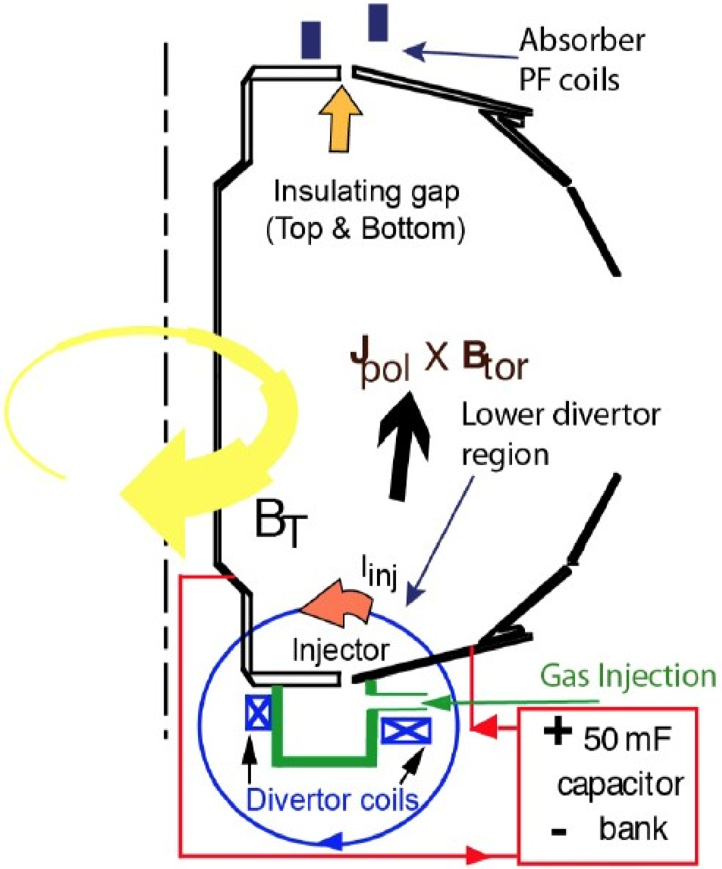


I_P=1 MA, B_T=1.0 T, P_{NBI}=12.6 MW

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

ST-FNSF may need solenoidless current start-up method
Coaxial Helicity Injection (CHI) effective for current initiation

CHI developed on HIT, HIT-II
Transferred to NSTX / NSTX-U

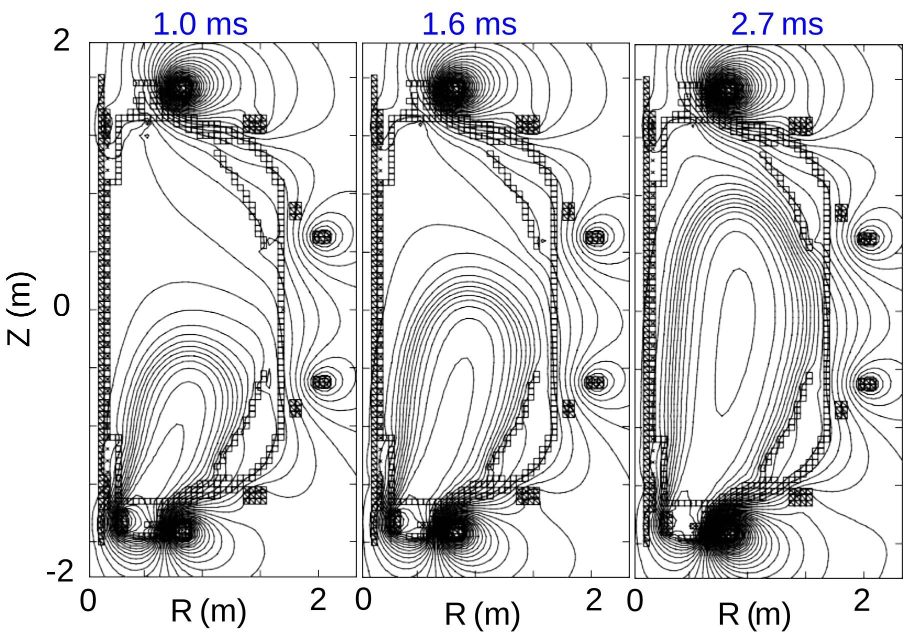


R. Raman et al., PRL 2006

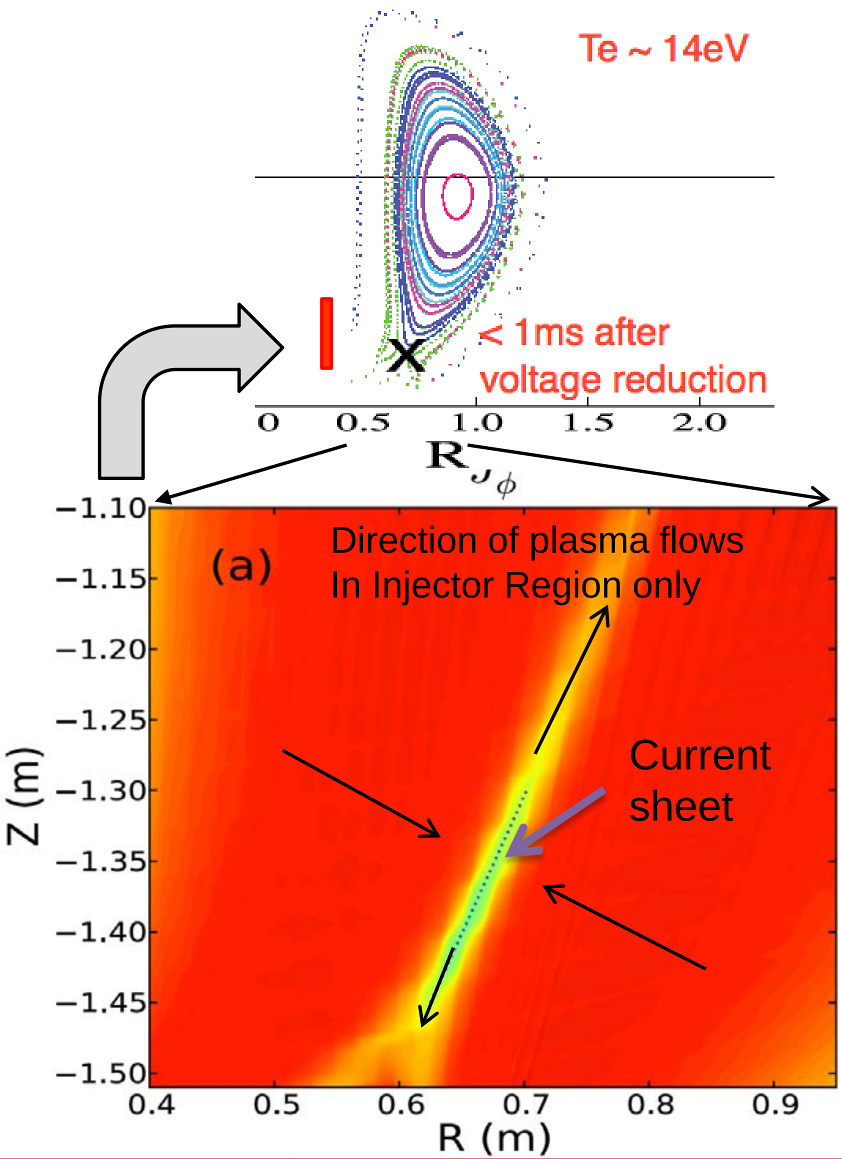
NSTX: 150-200kA closed flux current

NSTX-U: Project 300-400kA

TSC (axisymmetric 2D)
simulation of CHI startup



CHI in NSTX has resemblance to 2D Sweet-Parker reconnection (NIMROD simulations)



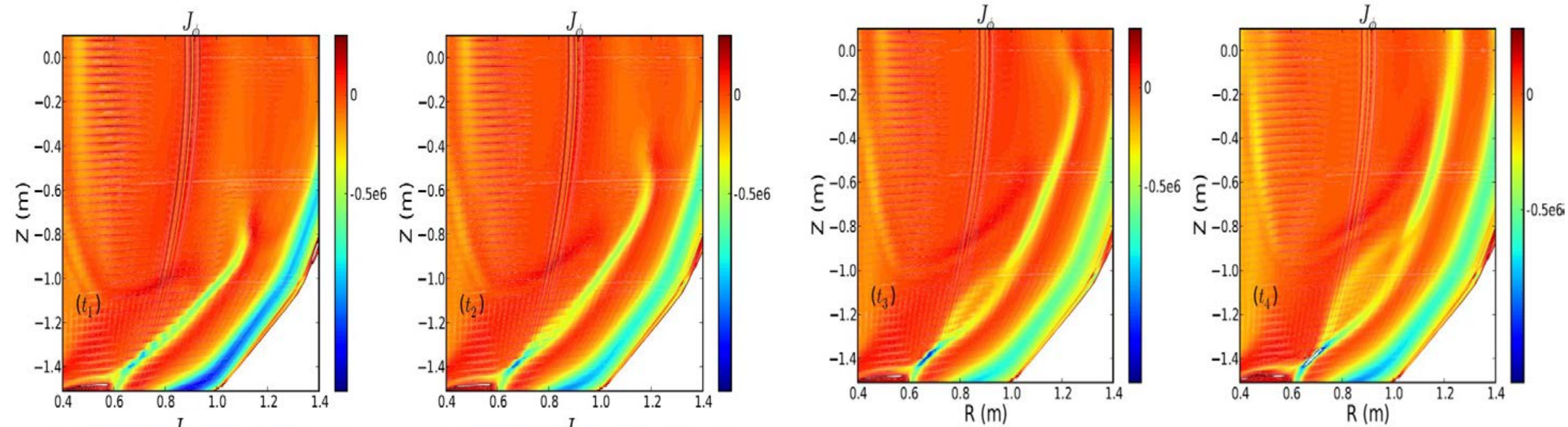
- Toroidal electric field generated in injector region by reduction of injector voltage and current
  - $E_{toroidal} \times B_{poloidal}$ drift brings oppositely directed field lines closer and causes reconnection, generating closed flux
- Elongated Sweet-Parker-type current sheet
- n > 0 modes / MHD not strongly impacting 2D reconnection

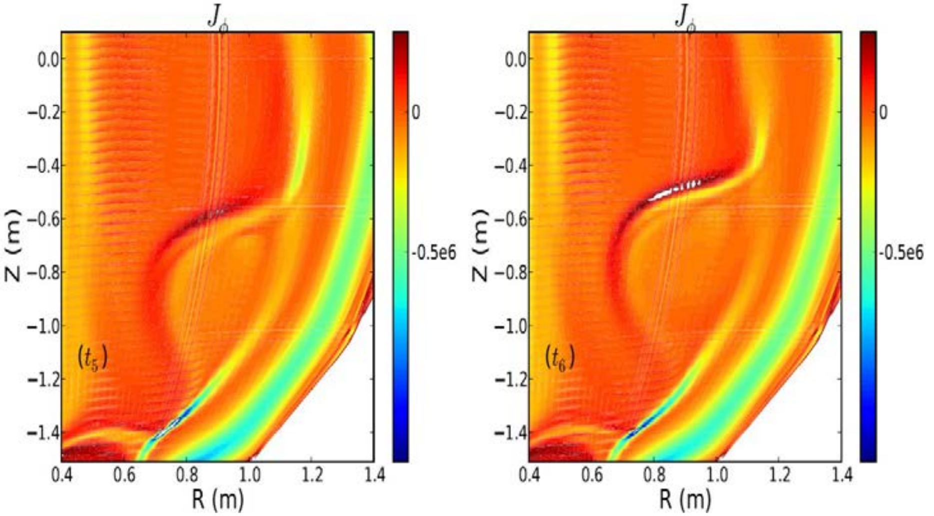
F. Ebrahimi, PoP 2013, PoP 2014

CHI current sheet unstable → plasmoids → merging

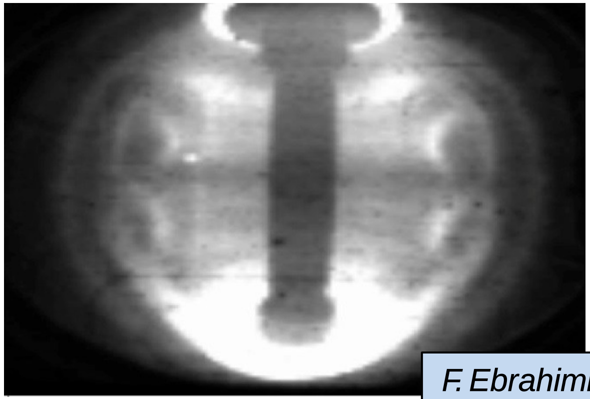
Possible lab observation of plasmoids → contribute to lab-astro

Current sheet shown in the lower half of the device.





Possible plasmoids during NSTX CHI



F. Ebrahimi , PRL 2015

NSTX-U will extend NSTX results, contribute to basic reconnection physics

Research opportunities

- Can NSTX-U really sustain all of its current without a transformer? If so, at what performance level?
- If NSTX-U succeeds at non-inductive sustainment, what level of non-inductive start-up and ramp-up is achievable?
  - Is it really possible to have a tokamak with NO solenoid?
  - Can we develop simple yet powerful enough models to simulate all this accurately?
    - Complex combination of current drive, transport, stability (TRANSP, TSC, + other models)
- What is fundamental reconnection physics in CHI?
  - How will plasmoid instabilities impact CHI, next-steps?

Summary

- NSTX-U will provide many opportunities to study toroidal confinement physics in new regimes:
  - Low aspect ratio, strong shaping, high β, low collisionality
  - Access to strong fast-ion instability drive, high rotation
  - Advanced divertors, lithium walls, high-Z PFCs
- The opportunities described here are just a small fraction of the research possibilities available!
- Please see next slide for people to contact for more

Contacts for 1st / 2nd year projects, thesis ideas

Boundary Science Group
R. Maingi, C.-S. Chang (deputy)

Pedestal Structure and Control
A. Diallo, R. Maingi
Theory/Modelling: J. Canik[1]
University Rep: D. Smith[2]

Divertor and Scrape-Off-Layer
V. Soukhanovskii[3], J.-W. Ahn[1]
Theory/Modelling: D. Stotler
University Rep: O. Schmitz[2]

Materials and PFCs
M. Jaworski, C. Skinner
Theory/Modelling: B. Wirth[4]
University Rep: J.-P. Allain[5]

Core Science Group
S. Kaye, S. Sabbagh[6] (deputy)

Macroscopic Stability
J. Berkery[6], J.-K. Park
Theory/Modelling: N. Ferraro
University Rep: D. Brennan[7]

Transport and Turbulence
W. Guttenfelder, Y. Ren
Theory/Modelling: W. Wang
University Rep: K. Tritz[8]

Energetic Particles
M. Podestá, D. Liu[9]
Theory/Modelling: N. Gorelenkov
University Rep: N. Crocker[10]

Integrated Scenarios Group
S. Gerhardt, R. Raman[11] (deputy)

Advanced Scenarios and Control
D. Battaglia, S. Gerhardt
Theory/Modelling: F. Poli
University Rep: E. Kolemen[7]

Solenoid-free Start-up & Ramp-up
D. Mueller, R. Raman[11]
Theory/Modelling: F. Ebrahimi[7]

Wave Heating and Current Drive
R. Perkins, J. Hosea
Theory/Modelling: N. Bertelli
University Rep: P. Bonoli[12]

Particle Control Task Force (FY2016-17)
Leader: J. Canik[1], **Deputy:** R. Maingi

Goal: Develop pumping and fueling tools, operating scenarios, and control systems to achieve main-ion and impurity density control for long-pulse

Disruption Prediction, Avoidance, and Mitigation (DPAM) Working Group
Leader: S. Sabbagh[6], **Deputy:** R. Raman[11]

Goal: Improve understanding of prediction, avoidance, and mitigation of disruptions in tokamaks, achieve acceptable levels of disruption frequency and severity

Collaborators in Science Program Leadership:

1	ORNL
2	University of Wisconsin
3	LLNL
4	UT Knoxville
5	University of Illinois
6	Columbia University
7	Princeton University
8	Johns Hopkins University
9	UC Irvine
10	UC Los Angeles
11	University of Washington
12	MIT

Backup

Latest run plan schedule for 2016

Goal is to operate 18 run weeks

➔FY16 budgets are favorable enough to support 18 weeks

➔ Want as much data as possible for IAEA synopses/meeting, APS-2016

- **December:** **0.5 run weeks (XMP)**
- **January:** **~ 2-3 run weeks (XMP, XP), PAC-37**
- **February:** **~ 2-3 run weeks (Ar-PS, LITER, LGI, MGI)**
- **March** **~ 3 run weeks**
  - **Mid-run assessment in March/April**
- **April – June** **9.5 run weeks, complete FY16 run**
- **July: Start outage: install high-k, high-Z tiles, …**
- **Resume operations winter 2017 for FY17: ~16 run weeks**

Summary of FY2016-18 NSTX-U Research Milestones

- **FY2016**
  - Obtain first data at 60% higher field/current, 2-3× longer pulse:
    - Re-establish sustained low l_i / high-κ operation above no-wall limit
    - Study thermal confinement, pedestal structure, SOL widths
    - Assess current-drive, fast-ion instabilities from new 2^{nd} NBI
- **FY2017**
  - Extend NSTX-U performance to full field, current (1T, 2MA)
    - Assess divertor heat flux mitigation, confinement at full parameters
  - Access full non-inductive, test small current over-drive
  - First data with 2D high-k scattering, prototype high-Z tiles
- **FY2018**
  - Study low-Z and high-Z impurity transport
  - Assess causes of core electron thermal transport
  - Test advanced q profile and rotation profile control
  - Assess CHI plasma current start-up performance

NSTX-U device performance progression:

- **1st year:** Limit forces to ½ way between NSTX and NSTX-U, and ½ of the design-point heating of any coil
  - Will permit up to ~5 second operation at B_T ~ 0.65T
- **2nd year goal:** Full field and current, coil heating to ¾ of limit
- **3rd year goal:** Full capability

Parameter	NSTX (Max.)	Year 1 NSTX-U Operations	Year 2 NSTX-U Operations	Year 3 NSTX-U Operations	NSTX-U Ultimate Goal
I_P [MA]	1.2	~1.6	2.0	2.0	2.0
B_T [T]	0.55	~0.8	1.0	1.0	1.0
Allowed TF I^2t [MA^2s]	7.3	80	120	160	160
I_P Flat-Top at max. allowed I^2t, I_P, and B_T [s]	~0.4	~3.5	~3	5	5

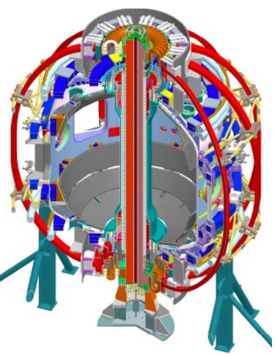
Five Year Facility Enhancement Plan (green – ongoing)

2015: Engineering design for high-Z tiles, Cryo-Pump, NCC, ECH

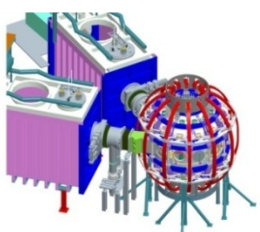
Fiscal Year:	2015	2016	2017	2018	2019
	Upgrade Outage	1.5 → 2 MA, 1s → 5s			
Run Weeks:		18	16	12-16	10-12

Major enhancements:
- ● Base funding
- ○ +15% incremental

New center-stack



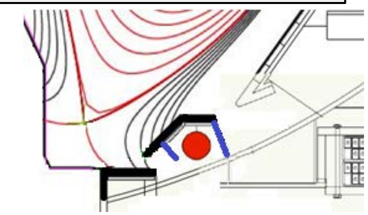
2nd NBI



Boundary Science + Particle Control

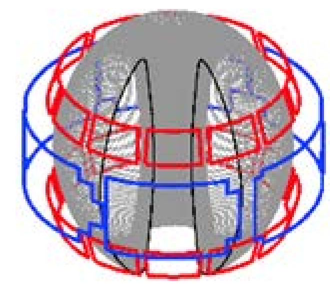
- Pulse-burst MPTS
- Boronization
- Li granule injector
- MAPP
- Upward LiTER
- High-Z tile row on lower OBD

- Lower divertor cryo-pump
- High-Z PFC diagnostics
- LLD using bakeable cryo-baffle



Core Science

- Upgraded halo sensors
- MGI disruption mitigation
- Laser blow-off
- 42 ch MPTS
- 48 ch BES
- MSE/LIF
- High k_θ
- Charged fusion product
- 4 coil AE antenna
- δB diag. - incremental
- Off-midplane 3D coils (NCC) →
- Enhanced MHD sensors
- DBS, PCI, or other intermediate-k
- Neutron collimator



Integrated Scenarios

Establish control of:

- Snowflake
- FIReTIP
- Rotation
- $\bar{n}_e$
- q_{min}
- Divertor P_{rad}
- HHFW limiter upgrade
- 1 MW ECH/EBW →
- Upgraded CHI for ~0.5MA
- 0.5-1 MA CHI
- up to 1 MA plasma gun

NSTX-U diagnostics available during first year

MHD/Magnetics/Reconstruction

Magnetics for equilibrium reconstruction
Halo current detectors
High-n and high-frequency Mirnov arrays
Locked-mode detectors
RWM sensors

Profile Diagnostics

MPTS (42 ch, 60 Hz)
T-CHERS: $T_i(R)$, $V_\phi(r)$, $n_C(R)$, $n_{Li}(R)$, (51 ch)
P-CHERS: $V_\theta(r)$ (71 ch)
MSE-CIF (18 ch)
MSE-LIF (20 ch)
ME-SXR (40 ch)
Midplane tangential bolometer array (16 ch)

Turbulence/Modes Diagnostics

Poloidal FIR high-k scattering (installed in 2016)
Beam Emission Spectroscopy (48 ch)
Microwave Reflectometer,
Microwave Interferometer
Ultra-soft x-ray arrays – multi-color

Energetic Particle Diagnostics

Fast Ion D_α profile measurement (perp + tang)
Solid-State neutral particle analyzer
Fast lost-ion probe (energy/pitch angle resolving)
Neutron measurements
Charged Fusion Product

New capability,
Enhanced capability

Edge Divertor Physics

Gas-puff Imaging (500kHz)
Langmuir probe array
Edge Rotation Diagnostics (T_i, V_ϕ, V_{pol})
1-D CCD H_α cameras (divertor, midplane)
2-D divertor fast visible camera
Metal foil divertor bolometer
AXUV-based Divertor Bolometer
IR cameras (30Hz) (3)
Fast IR camera (two color)
Tile temperature thermocouple array
Divertor fast eroding thermocouple
Dust detector
Edge Deposition Monitors
Scrape-off layer reflectometer
Edge neutral pressure gauges
Material Analysis and Particle Probe
Divertor VUV Spectrometer

Plasma Monitoring

FIReTIP interferometer
Fast visible cameras
Visible bremsstrahlung radiometer
Visible and UV survey spectrometers
VUV transmission grating spectrometer
Visible filterscopes (hydrogen & impurity lines)
Wall coupon analysis

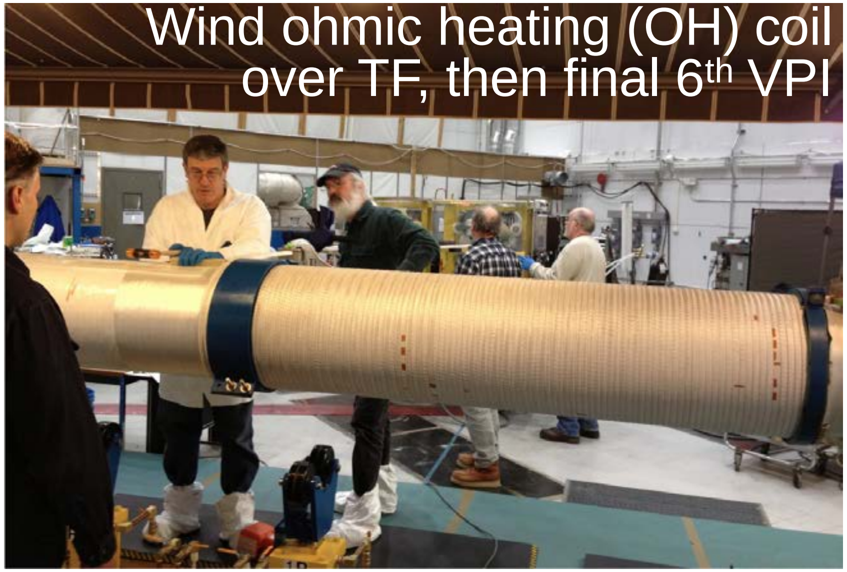
Central magnet construction was complex, multi-stage, multi-year effort

Built 4 toroidal field (TF) quadrants

4 vacuum pressure impregnations (VPI)

Combine quadrants (5th VPI)

Wind ohmic heating (OH) coil over TF, then final 6th VPI



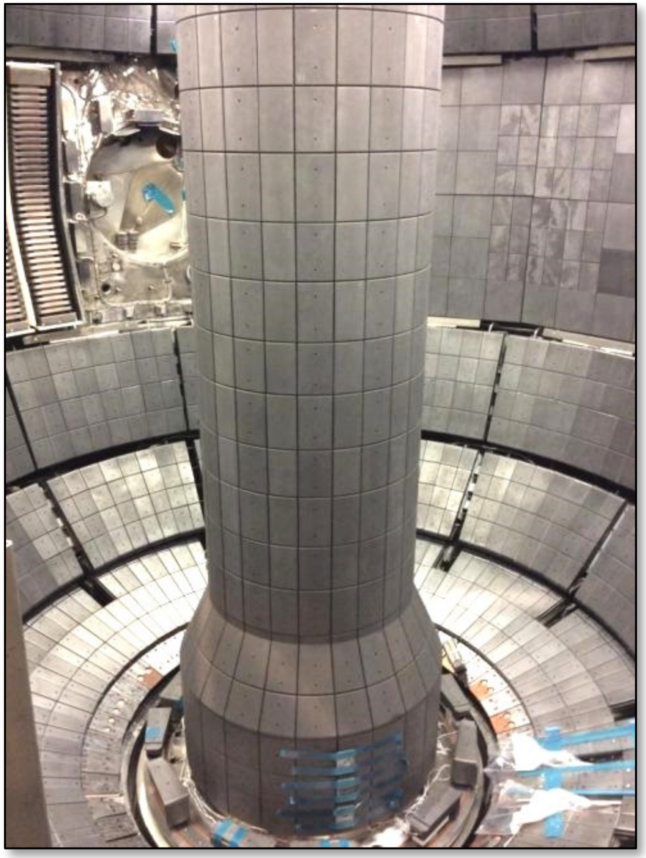
Install vacuum-tight casing over TF + OH bundle

Fabrication and installation of central magnet ultimately successful

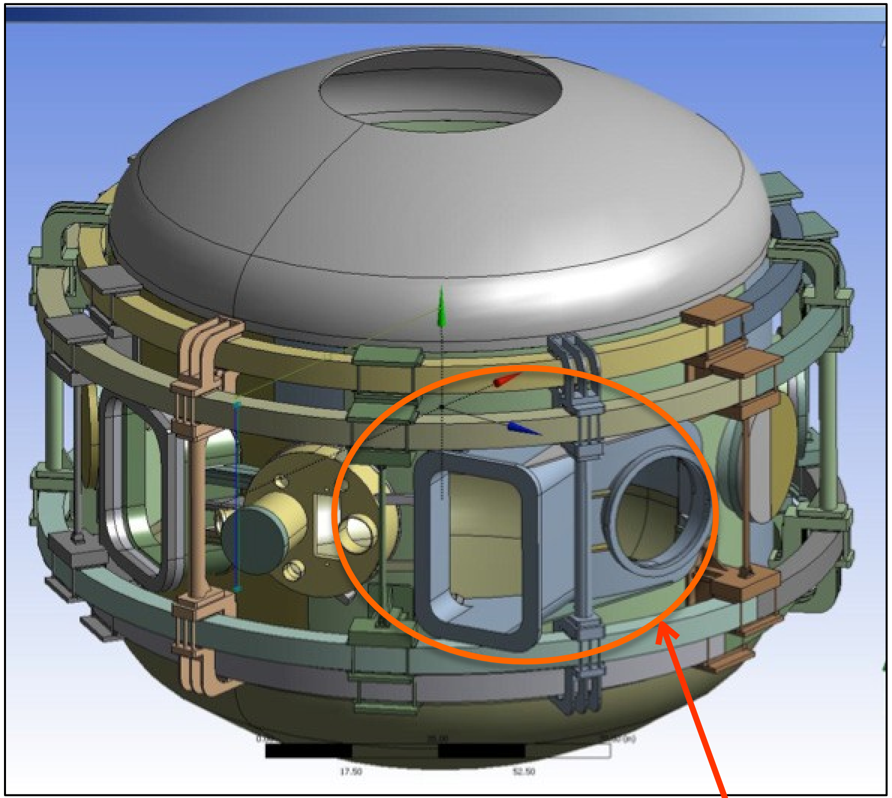
Over shield wall

Over the machine

Inside NSTX-U!

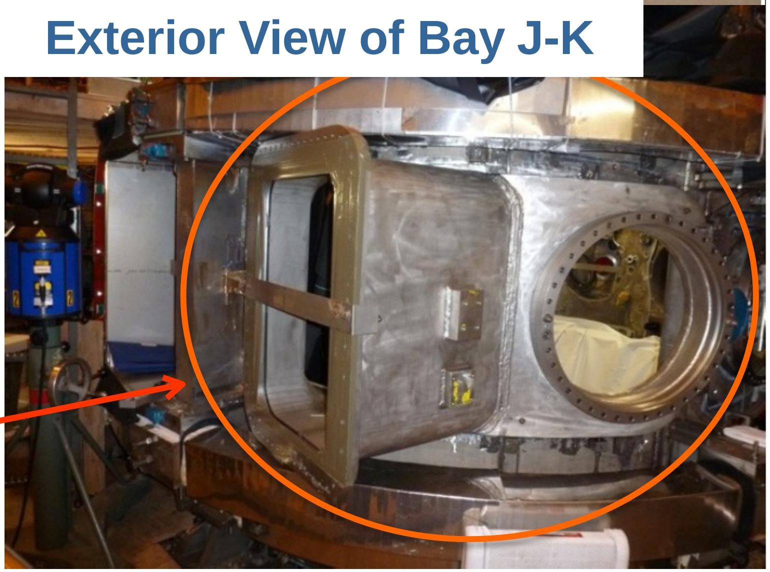


Tangential neutral beam required major vacuum vessel modifications



Interior View of Bay J-K

Exterior View of Bay J-K



JK cap

2^{nd} beam (from TFTR DT campaign) required T decontamination, NSTX test-cell re-arrangement

Beam Box being lifted over NSTX

Beam Box placed in its final location and aligned

Beam Box being populated with components

Design studies show ST potentially attractive as Fusion Nuclear Science Facility (FNSF) or Pilot Plant

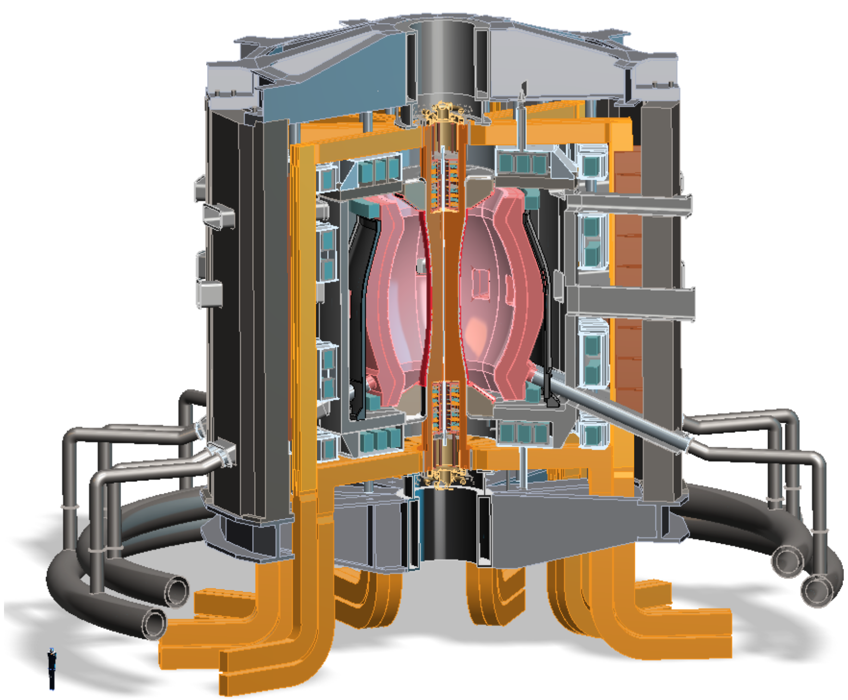
FNSF: Provide neutron fluence for material/component R&D (+ T self-sufficiency?)

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

FNSF with copper TF coils

$A=1.7$, $R_0 = 1.7m$, $\kappa_x = 2.7$

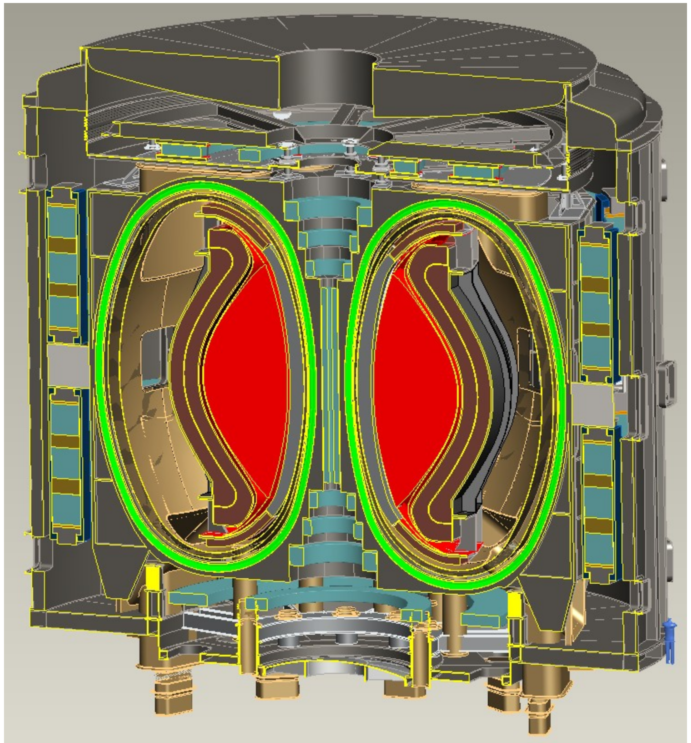
Fluence = $6MWy/m^2$, TBR ~ 1



FNSF / Pilot Plant with HTS TF coils

$A=2$, $R_0 = 3m$, $\kappa_x = 2.5$

$6MWy/m^2$, TBR ~ 1, Q_{eng} ~ 1



New physics regimes accessed at low aspect ratio → enhanced understanding of toroidal confinement physics

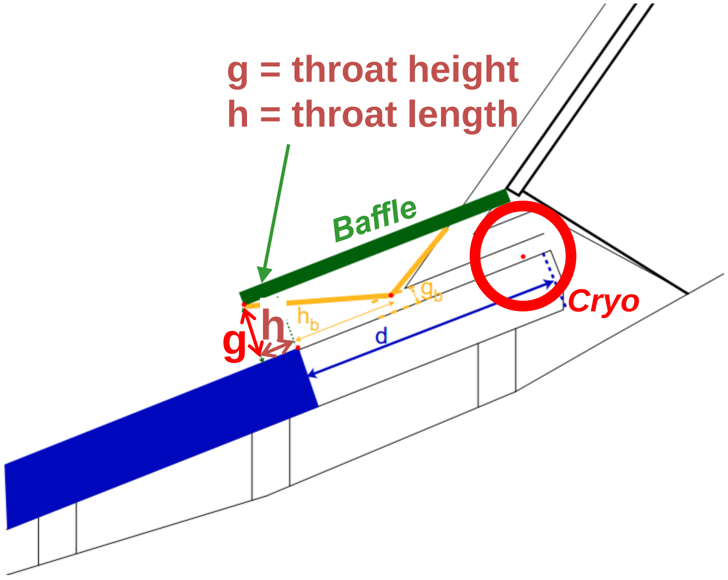
- Lower A → increased toroidicity → higher β, strong shaping
- Higher β → electromagnetic effects in turbulence, fast-particle modes, RF heating and Current Drive (over-dense plasmas)
- Higher fraction of trapped particles (low A), increased normalized orbit size (high β), and flow shear (due to low B, low A)→ broad range of effects on transport and stability
- Increased normalized fast-ion speed (high β) → simulate fast-ion transport/losses of ITER
- Compact geometry (small R) → high power/particle/neutron flux relevant to ITER, reactors

Backup - Facility Enhancements

Cryopump Physics Design Done in Collaboration with ORNL

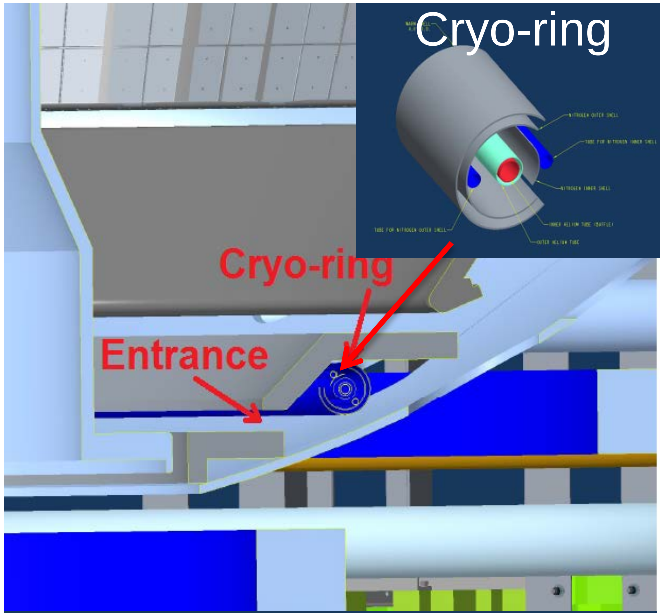
J. Canik, ORNL

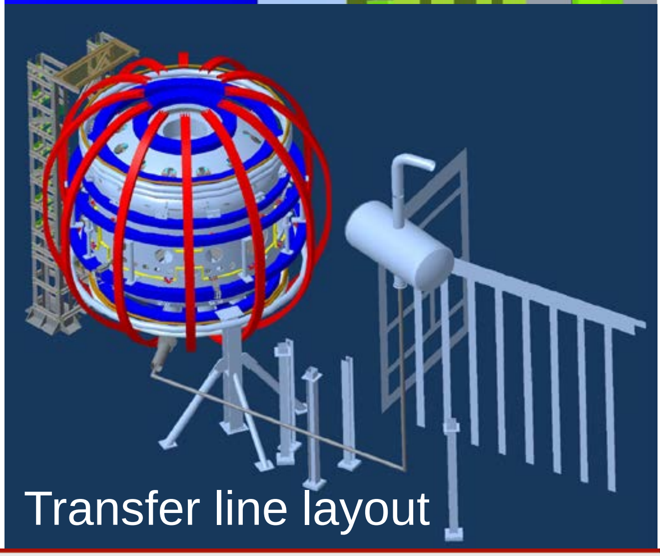
- Initial designs used semi-analytic models to determine pump geometry.
  - Conclusion on optimum geometry:
    - duct height h~2.5 cm,
    - length h~2 cm,
    - Radius of 0.72 m.
  - Allows pressures>1 mT over a range of plasma shapes and SoL widths.
  - Should allow the full beam fuelling to be pumped.
- Calculation then benchmarked against SOLPS



Physics Studies are Transferring to the Initial Engineering Design in Collaboration with MIT

- Initial in-vessel geometry has been laid out.
  - Pump radius, throat dimensions taken from the modeling.
  - Is a significant perturbation, requiring a rebuild of basically the entire lower outer divertor.
- Specification in progress for the Liquid He refrigerator.
  - Likely suitable model found
  - Location in room adjacent to NSTX-U has been identified.
- Ex-vessel liquid He plumping and transfer Dewar is under design.

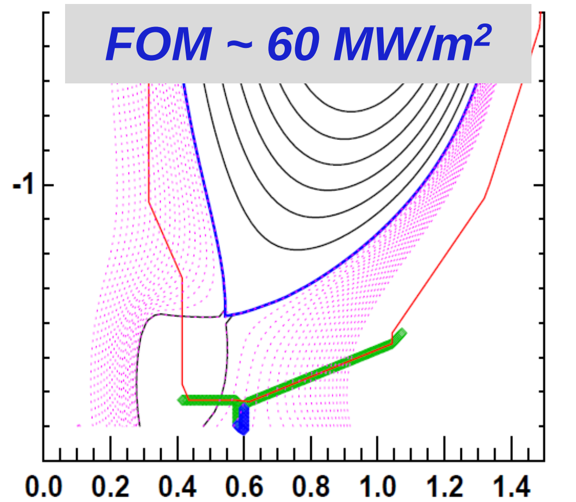




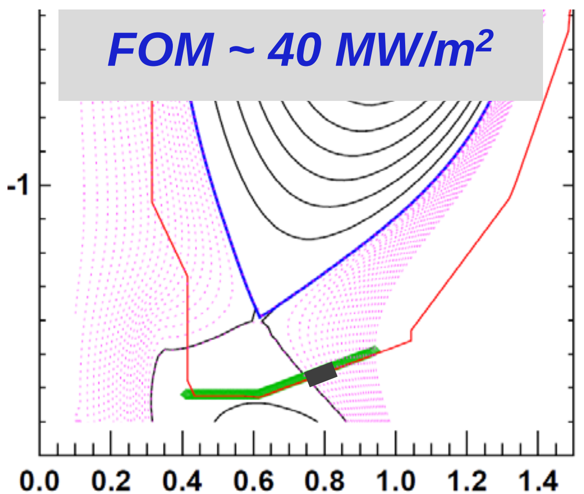
Outboard row of high-Z tiles can access high heat-flux, maintain operational flexibility

- Shape developed to perform dedicated tests on outboard PFCs
  - ISOLVER free-boundary solver utilized with specified β_N
  - 0D-analysis obtains heating power for assumed confinement multiplier H_{98y2}
- Zero-radiation power exhaust provides heat flux figure-of-merit (FOM)
  - FOM calculates incident power accounting for magnetic shaping only
  - High-Z shape FOM is 66% of similar full-power, high-triangularity scenario

High-performance discharge

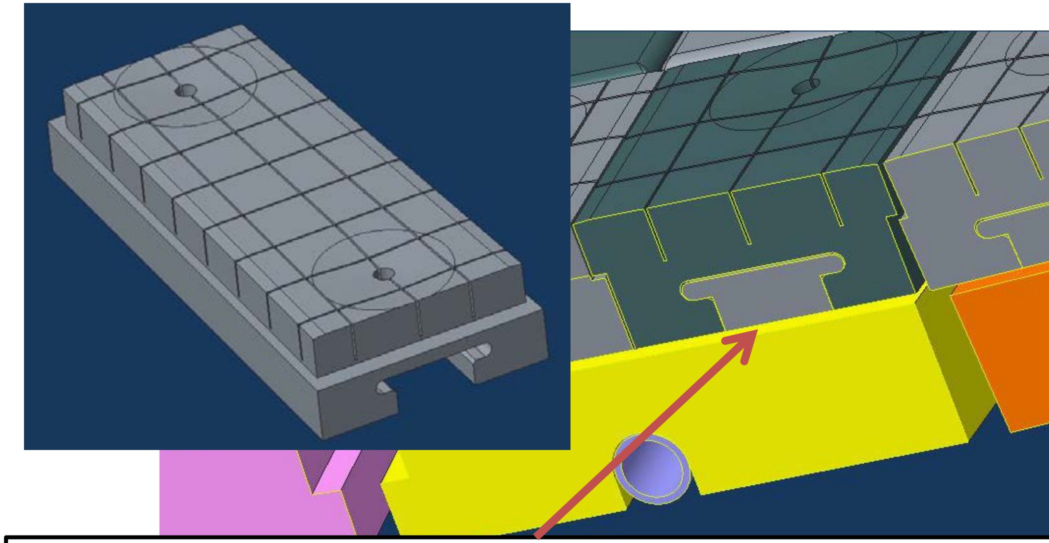


High-Z reference discharge



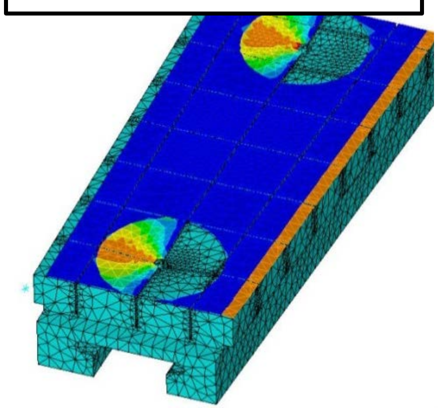
Single row of TZM molybdenum tiles will be installed for the FY-2017 run campaign

- Goals:
  - Initial assessment of plasma performance with high-Z PFC
  - Develop expertise in analysis, manufacture, installation with the new material
- Design constrained to by a one-for-one replacement of the existing tiles.
- Raw material procurement underway
- Edge and access-way chamfers introduced to reduce heat-flux peaking.

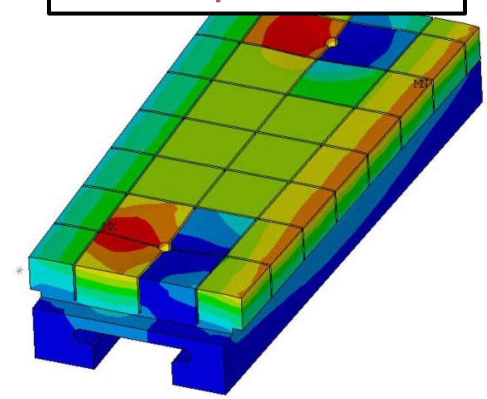


Seamless integration with existing mounting scheme minimizes installation time

Surface heat flux



Temperature

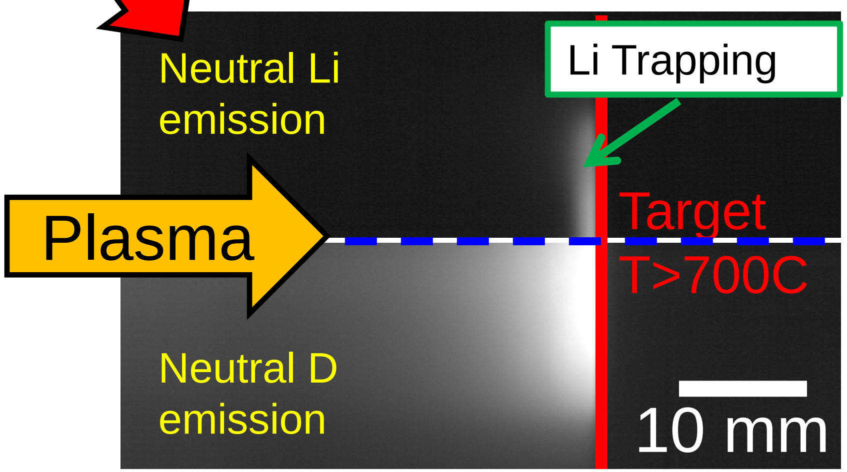


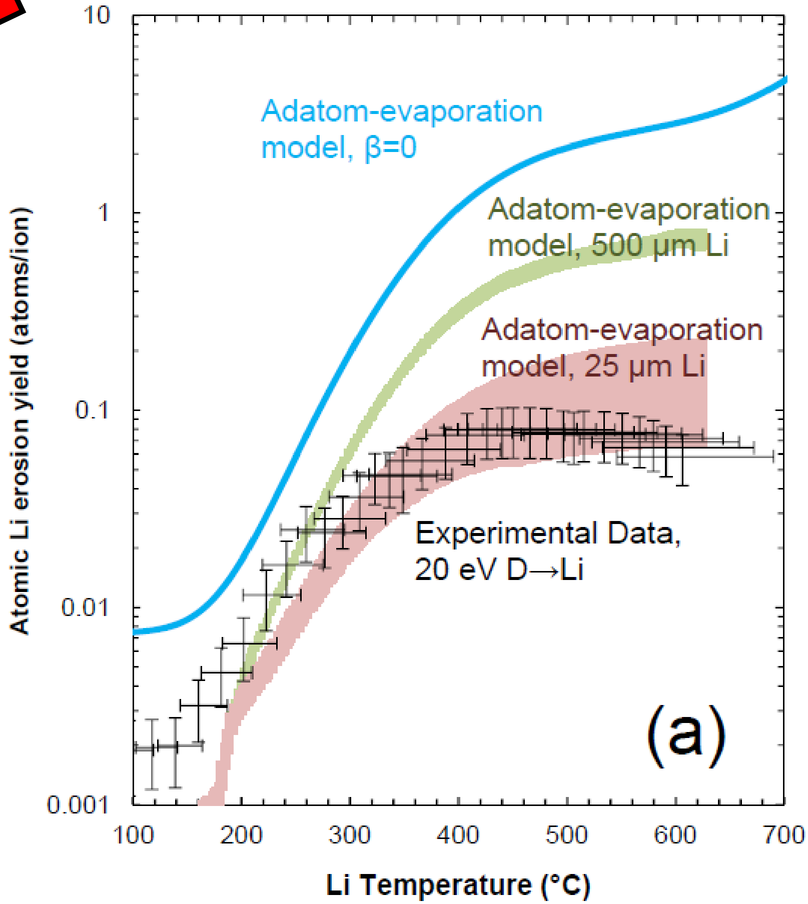
Suppressed erosion and trapping at target observed in linear plasma device (Magnum-PSI)

- Mixed-material effect reduces erosion due to LiD formation
- Plasma pre-sheath potential well large enough to retain eroded Li
- Significant implications for evaporative cooling concepts

T. Abrams 2014 PhD Princeton U., T. Abrams 2016 Nucl. Fusion, M. Chen 2016 Nucl. Fusion.

Li-I emission, t=2.5s

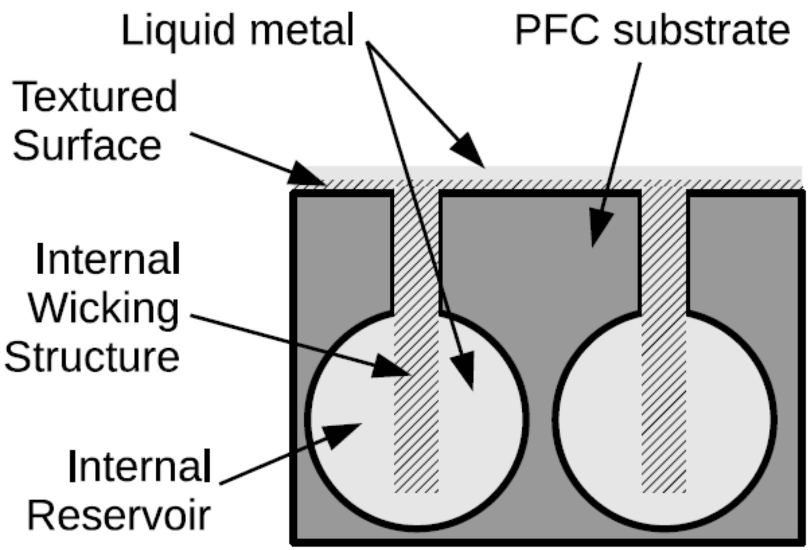


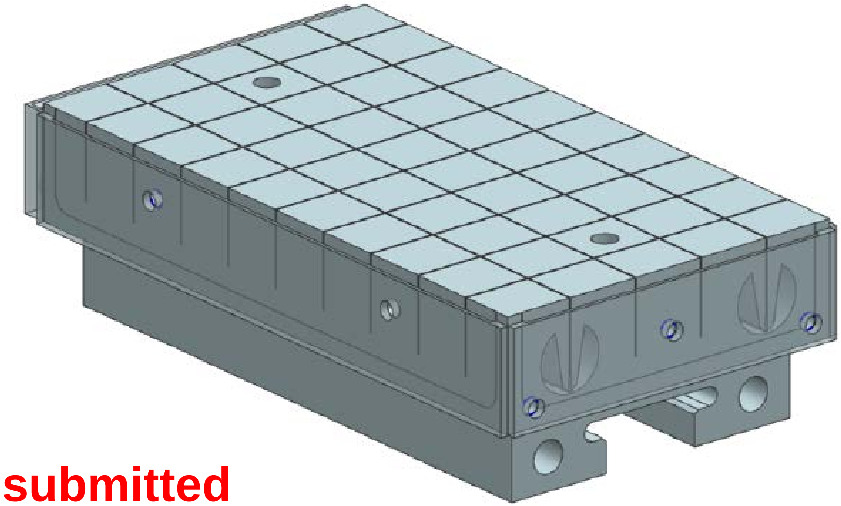


Jaworski, 3rd ISLA, 2013

Pre-filled target concept integrates Li reservoir with high-Z tile scheme

- Similar to CPS device but applicable as divertor PFC
- Utilizes wire-EDM fabrication to obtain complex geometry
- Emphasizes passive replenishment via capillary action

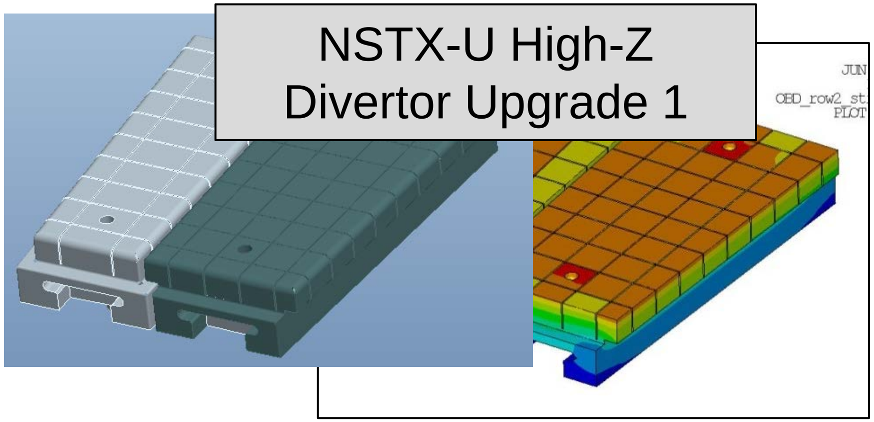




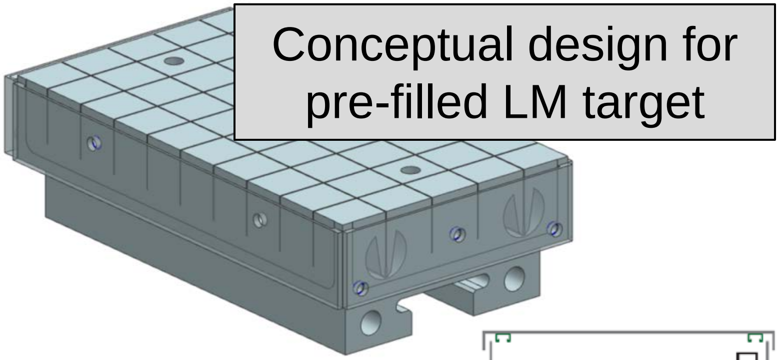
P. Rindt, TU/Eindhoven Thesis, Jaworski FED submitted

A three-step progression can achieve flowing, liquid metal PFCs

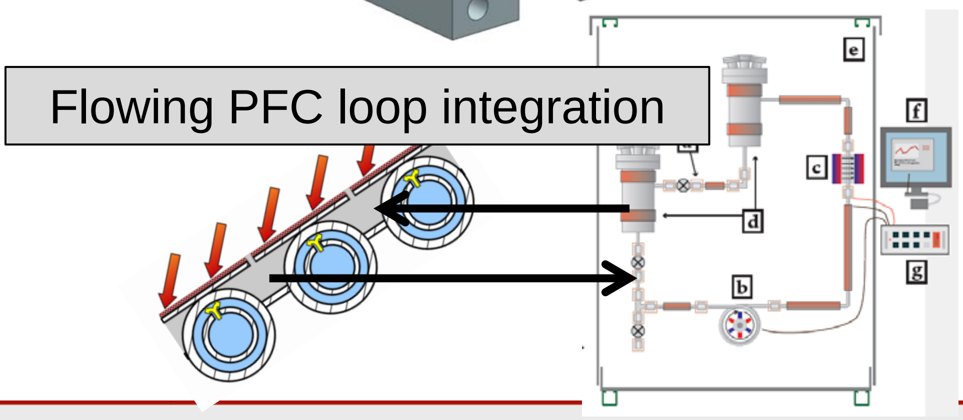
1. High-Z divertor tiles + LITER



2. Pre-filled liquid-metal target

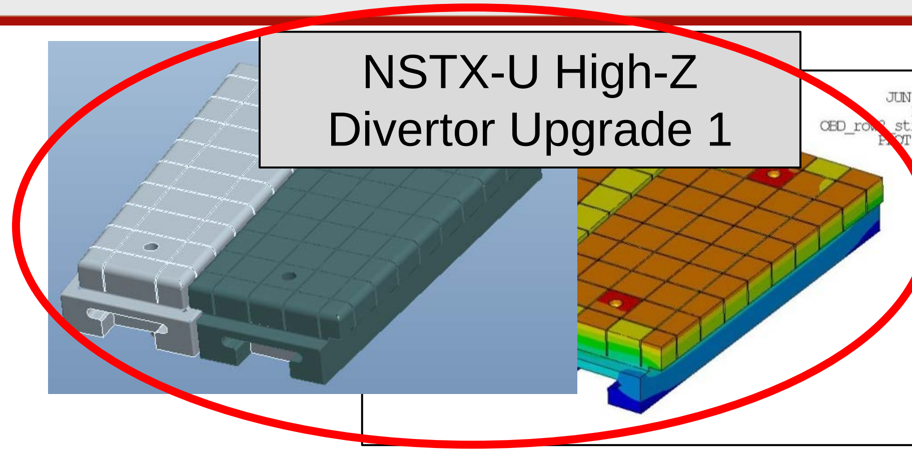


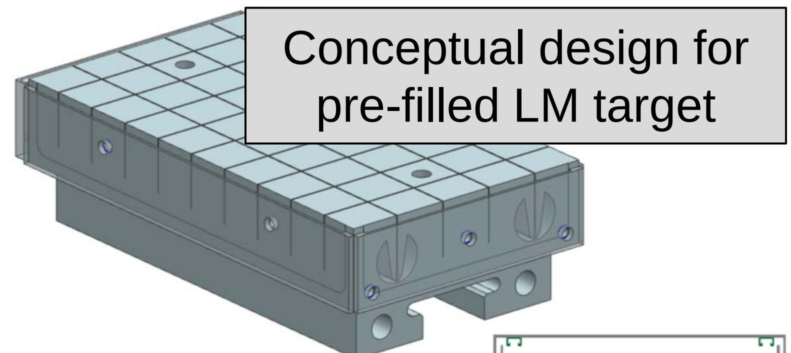
3. Flowing LM PFC

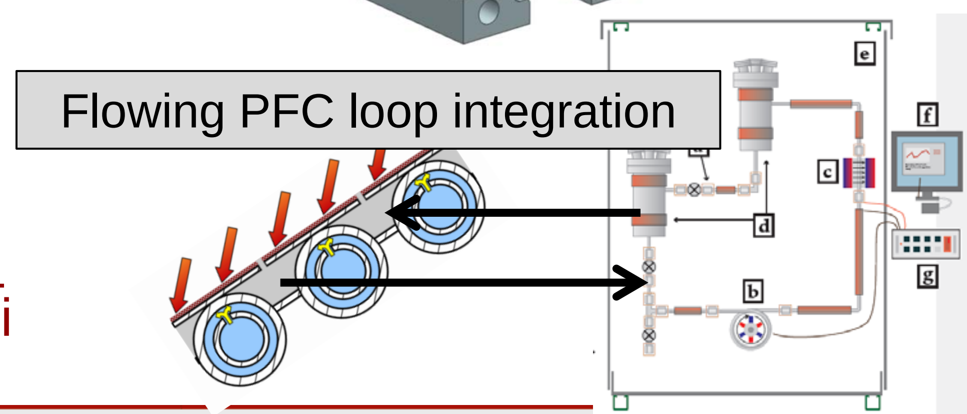


High-Z divertor tiles + Li evaporated coatings provide divertor analogue of Magnum-PSI experiments

1. High-Z divertor tiles + LITER
   - Technical goals:
     - Establish non-intercalating substrate for evaporated Li
     - Provide high-heat flux substrate for Li experiments
   - Scientific goals:
     - Quantify maintenance of Li on high-temperature substrate and protection of substrate
     - Re-examine suppression of erosion in high-flux divertor
     - Understand impact and core-edge compatibility of <u>high-temp. target</u> with limited inventory of Li

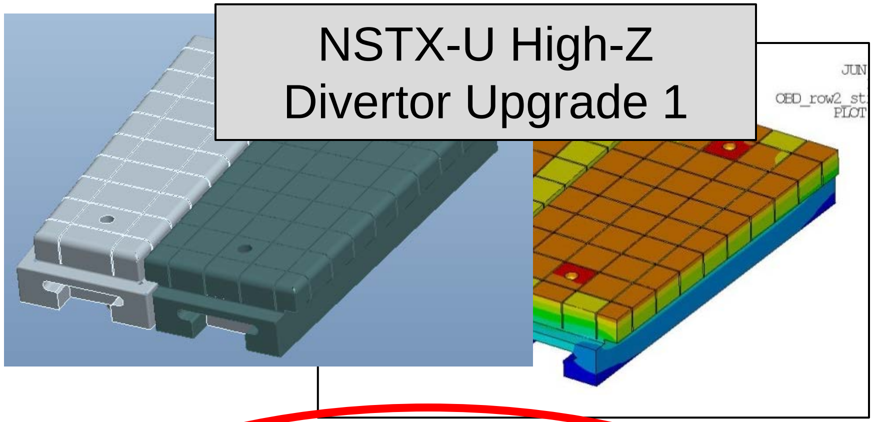


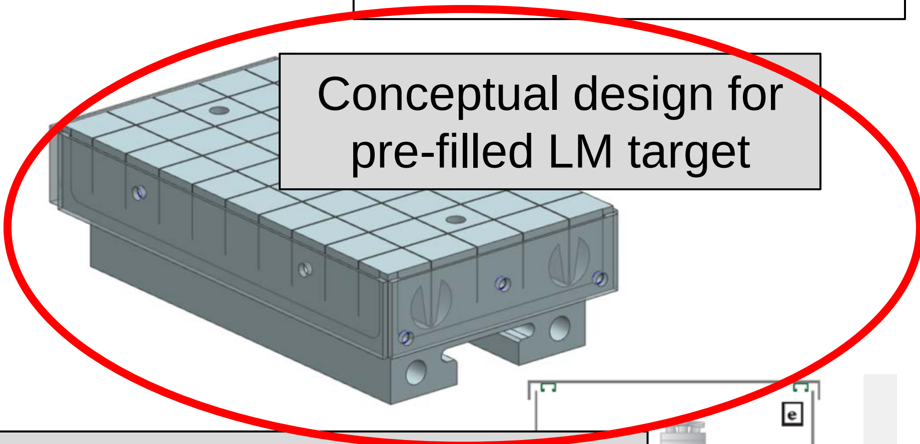


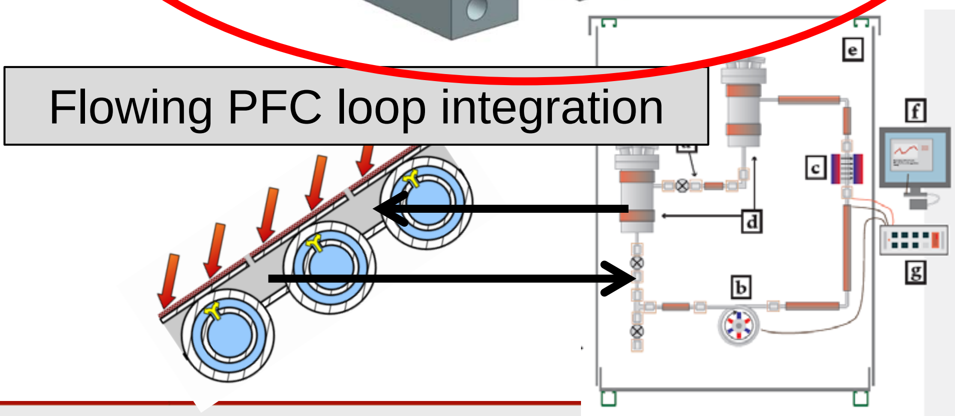


Pre-filled targets test LM coverage, resupply and impact of significant Li source

2. Pre-filled liquid-metal target
   - Technical goals:
     - Achieve introduction of Li in NSTX-U without evaporation
     - Realize complex target production as high-heat flux target
   - Scientific goals:
     - Test models of maintenance of LM wetting and coverage
     - Understand limits of LM passive resupply
     - Understand impact and core-edge compatibility of high-temp. target with **larger** inventory of Li

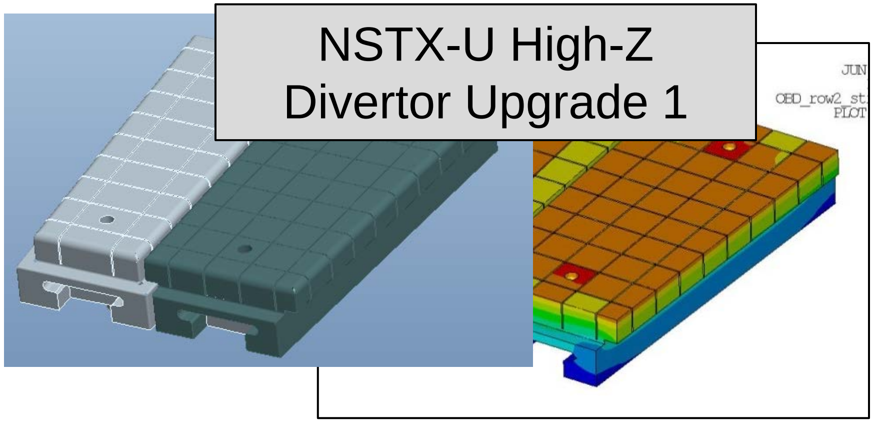




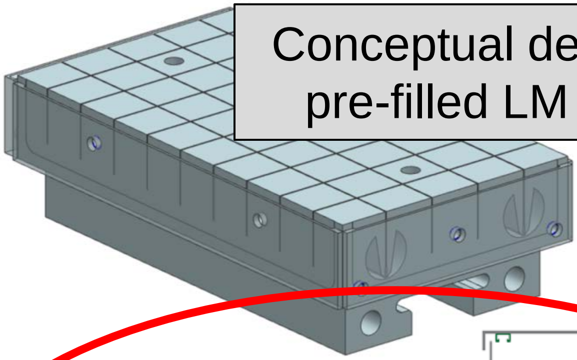


Final integration demonstrates LM introduction/extraction and inventory control

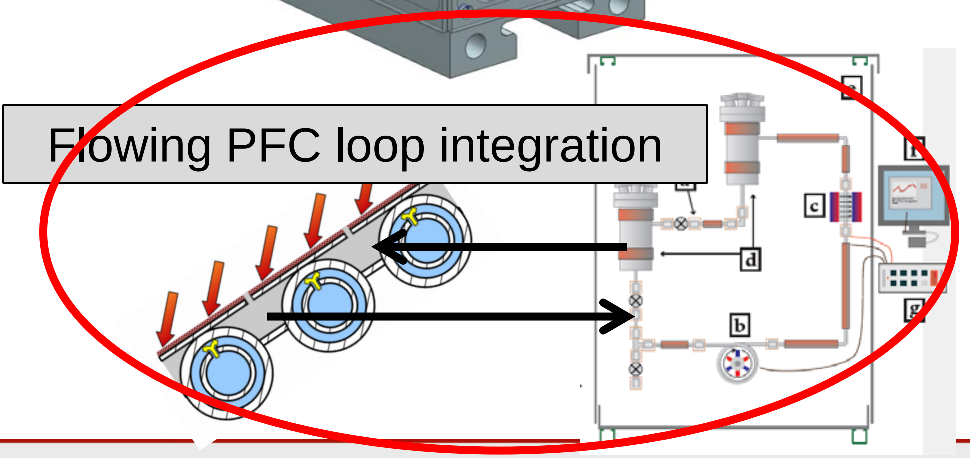
3. Flowing LM PFC
   - Technical goals:
     - Integrate parallel effort on loop technology with confinement experiment
     - Achieve active introduction and extraction from exp.
   - Scientific goals:
     - Assess material inventory control from LM target
     - Understand performance of passive + active replenishment techniques
     - Understand impact and core-edge compatibility of <u>high-temp. target</u>



NSTX-U High-Z Divertor Upgrade 1



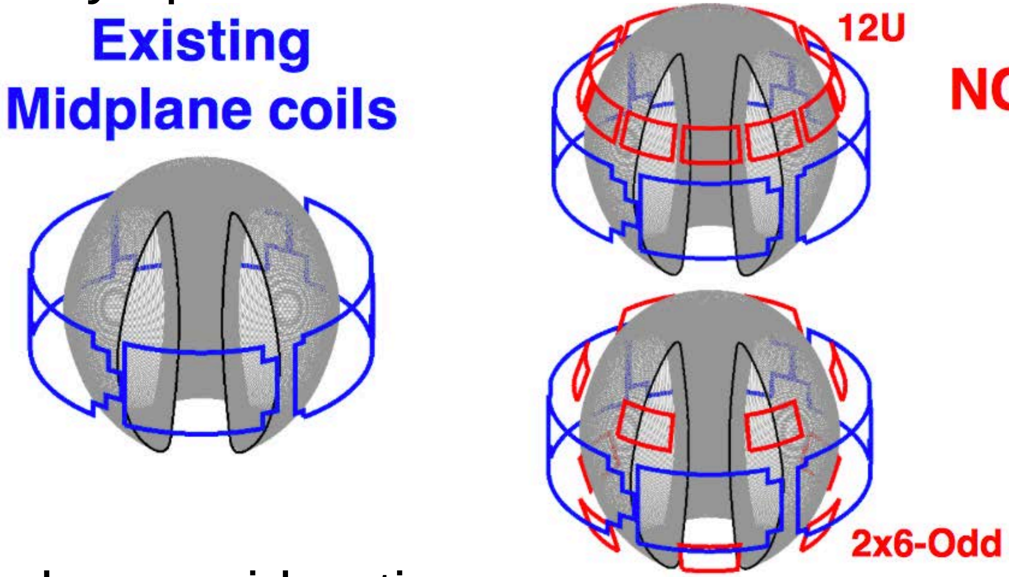
Conceptual design for pre-filled LM target



Flowing PFC loop integration

Non-Axisymmetric Control Coils (NCC) will dramatically improve NSTX-U 3D physics capabilities

- Three primary options considered for the NCC implementation:

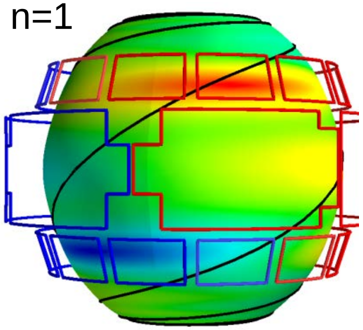


- Metrics under consideration:
  - n=1
    - RWM control
    - Ability to scan the relative ratio of resonant to non-resonant n=1 contributions
  - n>1
    - Variation of available NTV torque profiles.
    - NTV normalized to the Chirkov parameter
- Conclusion: 2x12 is best, 2x6-Odd is a good step in a staged implementation

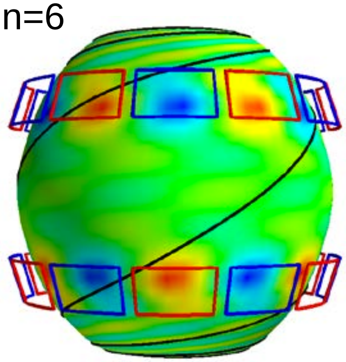
NCC physics design completed: Optimization for NTV braking performed with IPEC coupling matrix

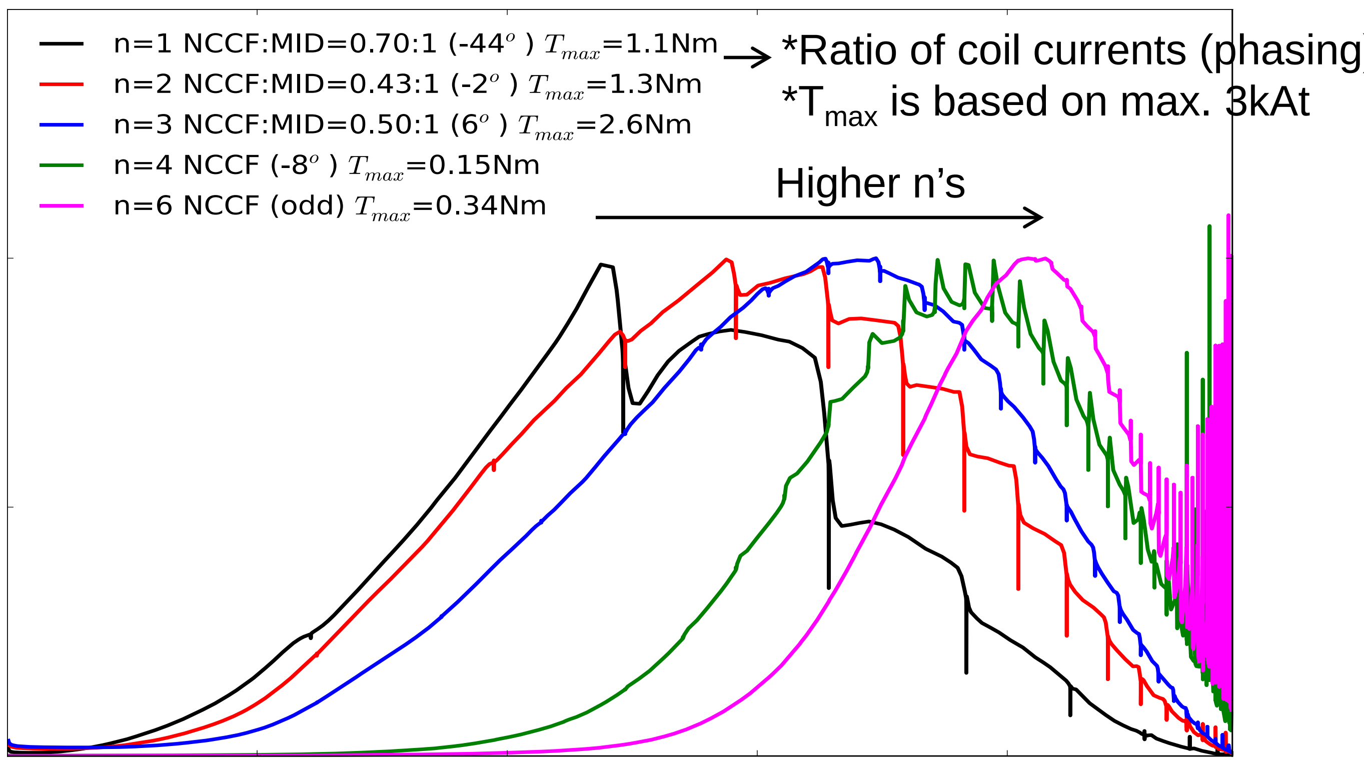
- NCC and midplane coils can be combined to remove the dominant resonant modes up to the second, giving the optimized NTV for core
  - NCC 2x12 provides n=1,2,3,4,6 optimized NTV, and 2x6 provides n=1,2,6
  - Optimized NTV can be used to control local torque with minimized resonance

n=1



n=6



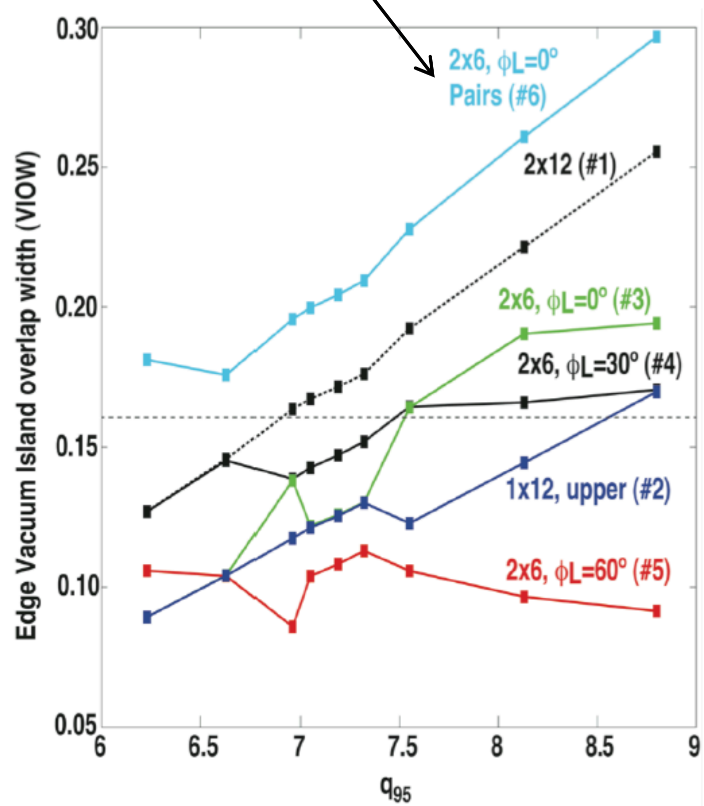


Study of RMP characteristics with NCC extended with TRIP3D (T. Evans) – 2x12 NCC (and 2x7) favorable for RMP

- Vacuum Island Overlap Width (VIOW) analysis shows full NCC 1kAt can produce sufficient VIOW in a wide range of q_{95}, but partial NCC needs more currents with low q_{95} targets
  - Also shows 2x7, with "one" more additional array upon partial NCC can provide the greater VIOW by toroidal coupling (n=2,4,9)

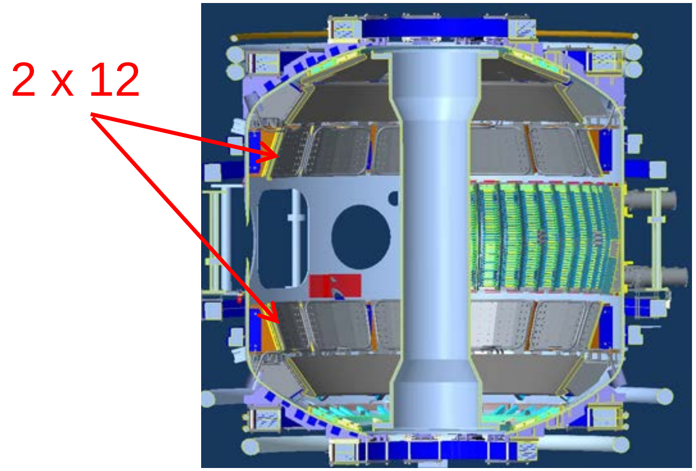
NCC Configurations Used for the plots Shown in Figures 1 and 2

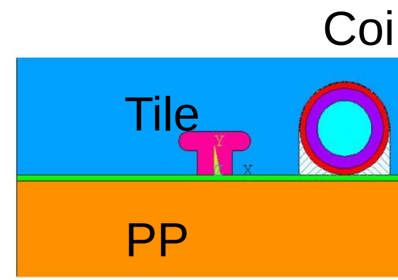
#	Description, Color	NCC Configuration Layout
1	2x12 dashed black line	+ + - - + + - - + + - - + + - - + + - - + + - -
2	1x12, upper solid blue line	+ + - - + + - - + + - - □ □ □ □ □ □ □ □ □ □ □ □
3	2x6, ϕ_L=0° solid green line	+ □ - □ + □ - □ + □ - □ + □ - □ + □ - □ + □ - □
4	2x6, ϕ_L=30° solid black line	+ □ - □ + □ - □ + □ - □ □ + □ - □ + □ - □ + □ -
5	2x6, ϕ_L=60° dashed red line	+ □ - □ + □ - □ + □ - □ - □ + □ - □ + □ - □ + □
6	2x7, ϕ_L=0° Pairs solid light blue line	+ □ - □ + - □ + □ - □ □ - □ + □ - + □ - □ + □ □



NCC is Undergoing Conceptual Design in Parallel with the Cryopump

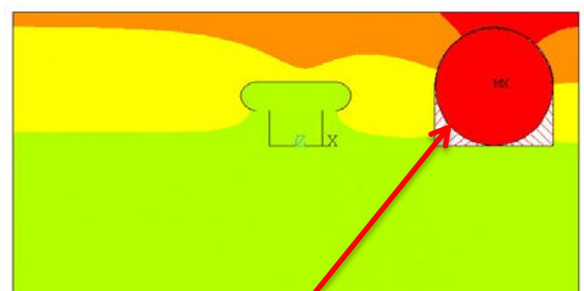
- Considering a mineral insulated conductor
- Order of 20' test sample is placed:
  - Both solid and hollow center conductors
- Goal is to assess
  - Manufacturability/formability,
  - electrical characteristics, including end-sealing methods,
  - fabrication lead time and cost.
- Hollow conductor will allow He cooling, but analysis indicates thermal ratcheting with solid conductor is likely manageable.





Assumptions In Solid Conductor Calculation

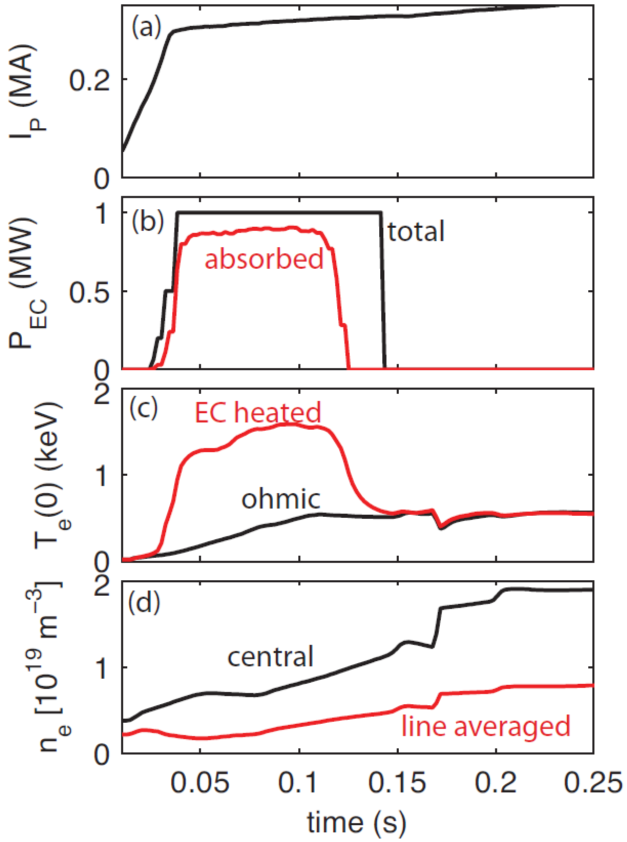
3 kA 5s pulse 1200 s repetition prate

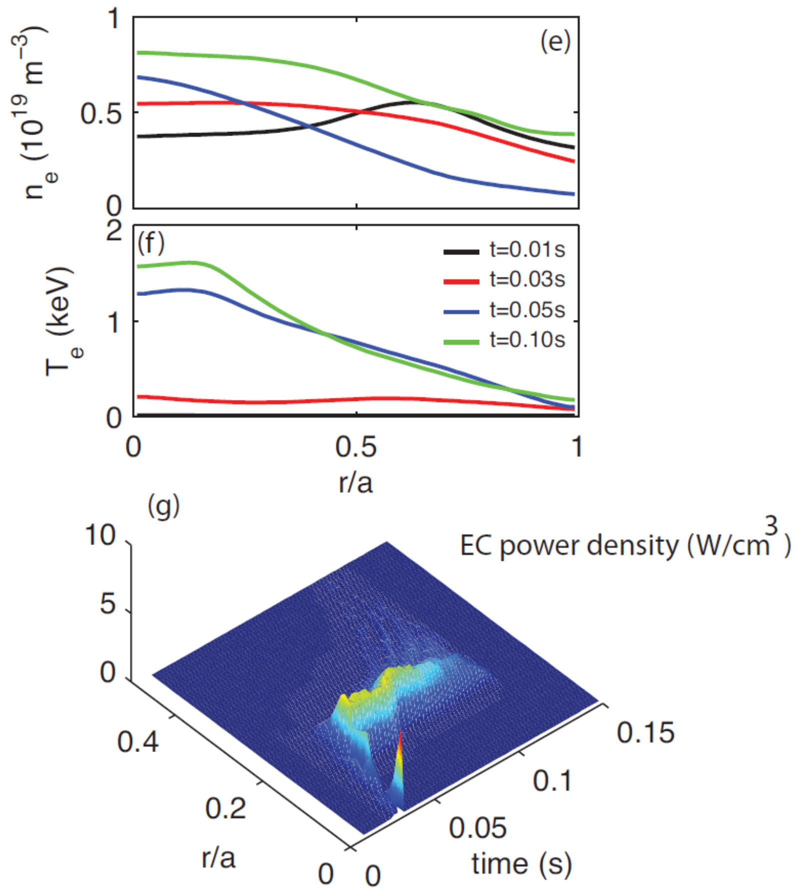


T_{max}=142 C after 8 hours

TRANSP modelling: ECH is game-changer for non-inductive ramp-up

Heats low temperature plasma to 1-1.5keV in ~30ms

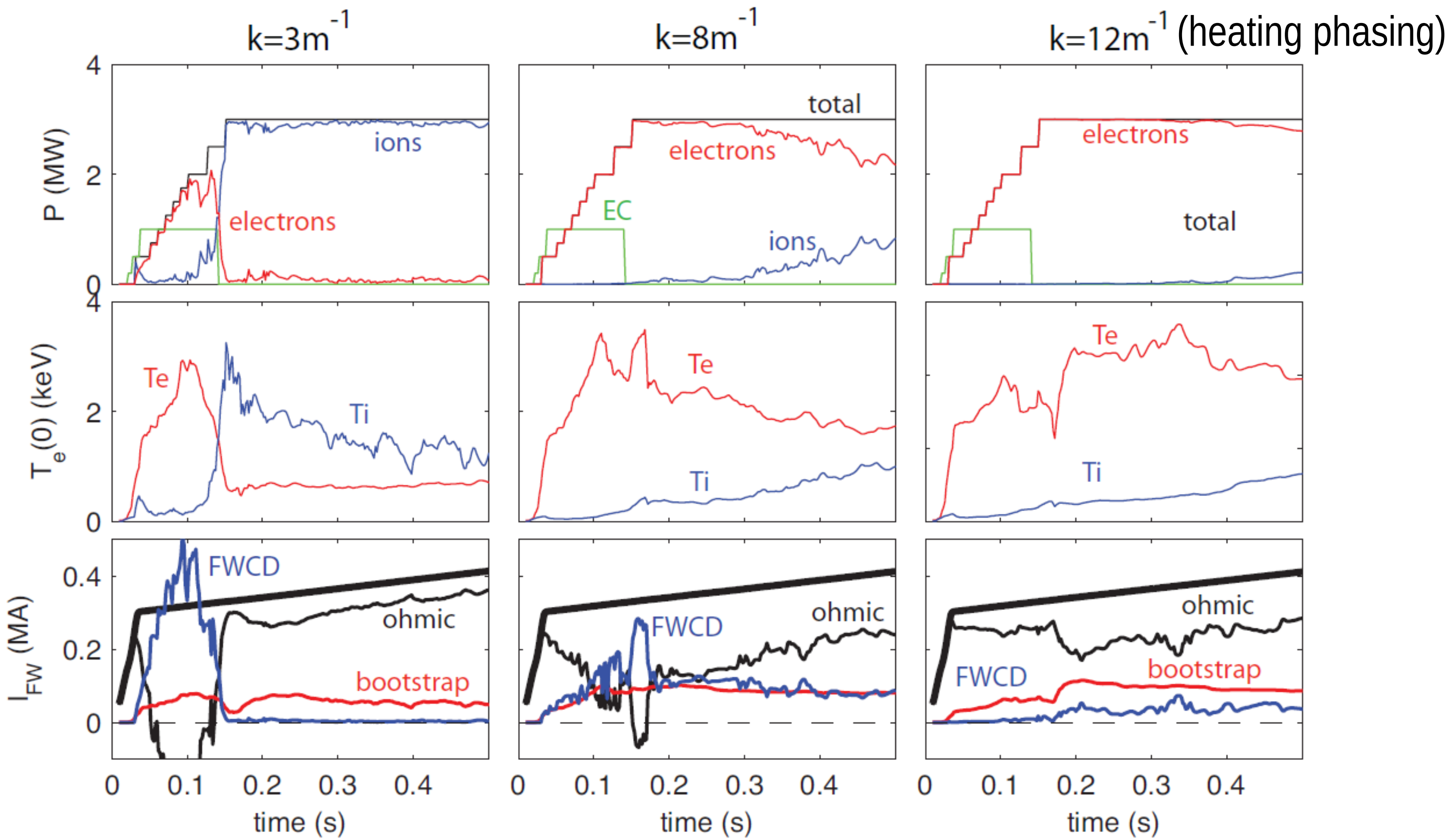




ECH accessibility limited to low density, but compatible with CHI

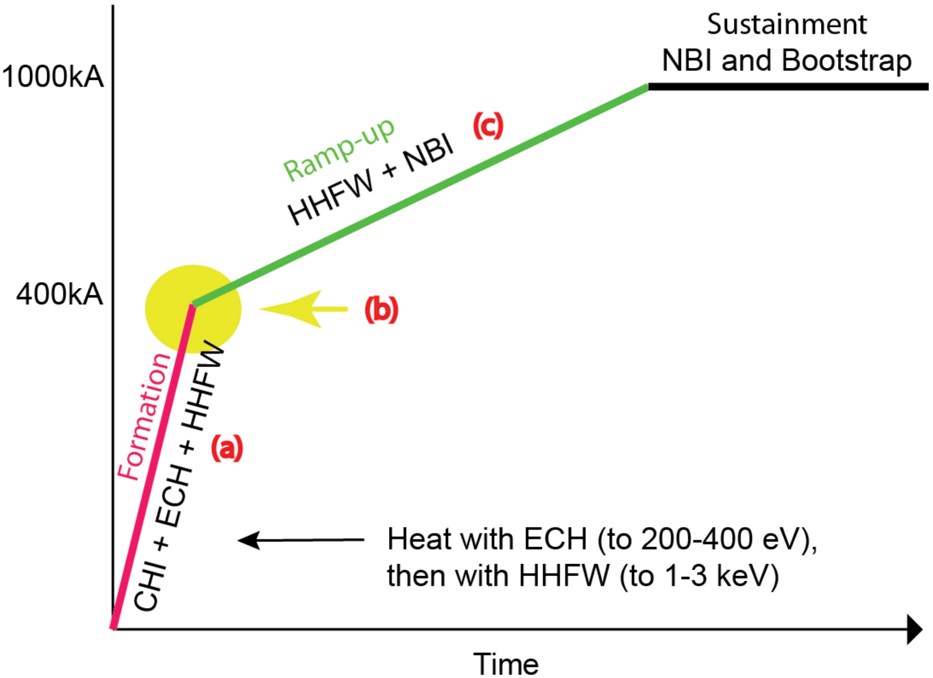
EC + FWCD synergistic for lowest FW phasing $k_\phi=3m^{-1}$
Half power needed to drive 400kA compared to no EC

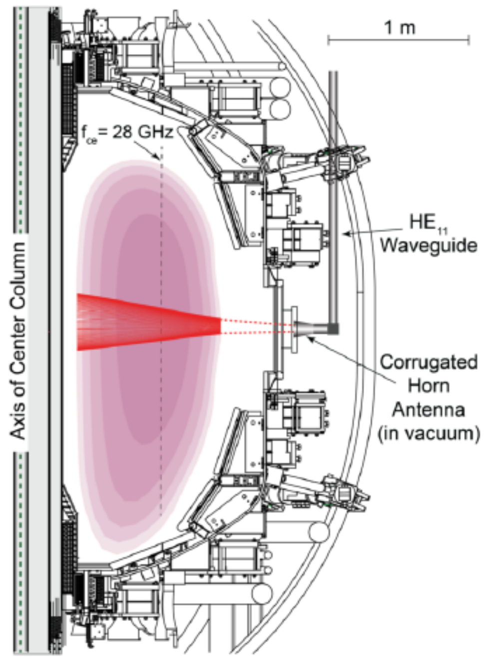
- ECH enables sustained T_e conditions for higher FW k_ϕ
- Need to optimize FW phasing during shot to sustain H&CD

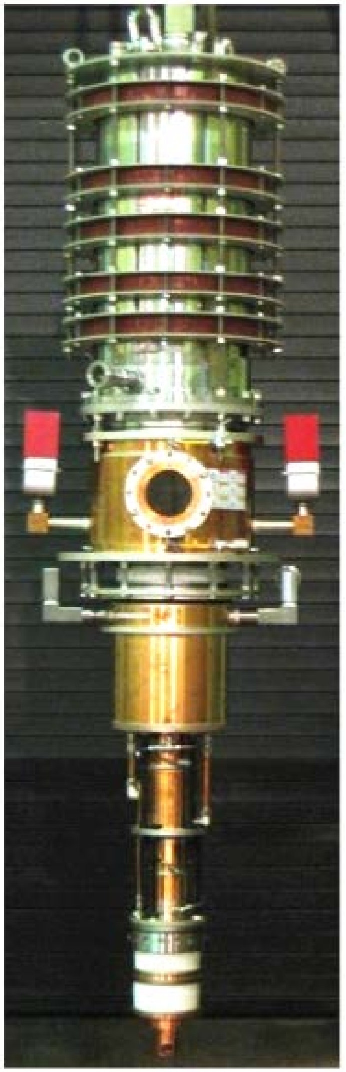


28 GHz Gyrotron System Will Facilitate Non-Inductive Startup Research

- Coupling CHI to NB overdrive will be aided by electron heating







- TSC simulations indicate 0.6MW of absorbed ECH power could increase T_e to ~400eV in 20ms
- 28 GHz, 2 MW tubes developed by Tsukuba University planned to provide this power.
- Have found location for gyrotron, appropriate commercial waveguide manufacturer.