NSTX

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U.S. DEPARTMENT OF ENERGY | Office of Science

Overview of Progress and Plans of the NSTX Facility*

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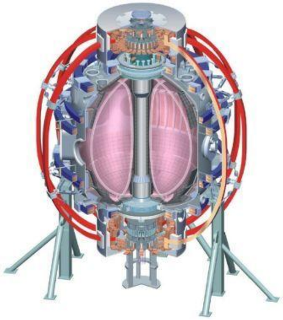
10th International Symposium on Fusion Nuclear Technology
11-16 September 2011
Portland, Oregon, USA



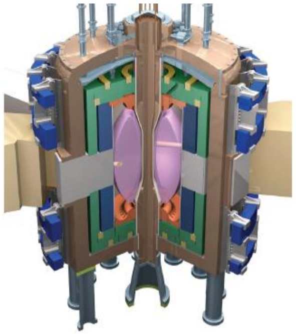
**This work supported by the US DOE Contract No. DE-AC02-09CH11466*

Outline

- **NSTX Mission**
- **Motivation for NSTX Upgrade**
- **NSTX Research Highlights and Upgrade Design Progress**
- **Summary**

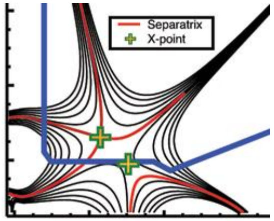
NSTX Mission Elements

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



ST-FNSF

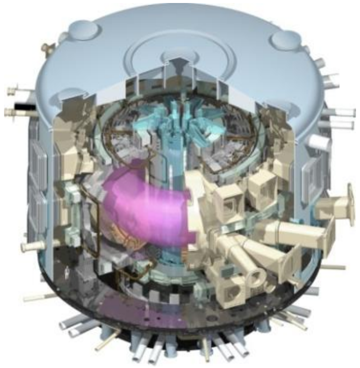
- **Develop solutions for plasma-material interface**



"Snowflake"

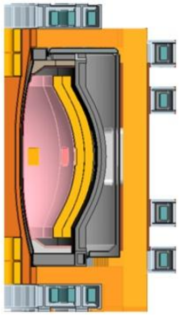
Lithium

- **Advance toroidal confinement physics for ITER and beyond**



ITER

- **Develop ST as fusion energy system**

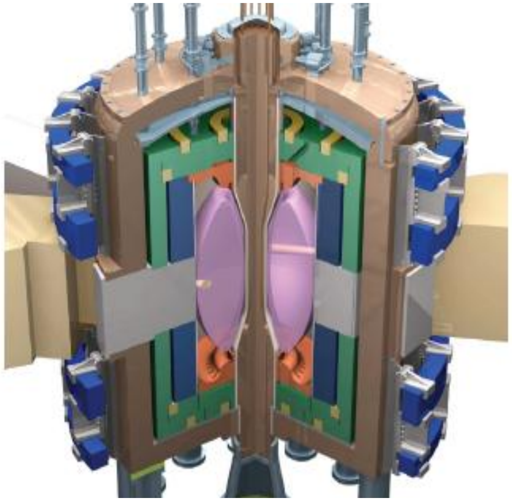


ST Pilot Plant

Mission of ST-FNSF

From M. Peng, ORNL

- **Provide a continuous fusion nuclear environment of copious neutrons to develop an experimental database on:**
  - Nuclear-nonnuclear coupling phenomena in materials in components for plasma-material interactions
  - Tritium fuel cycle
  - Power extraction



ST-FNSF

- **Complement ITER, prepare for component test facility (CTF):**
  - Low Q (≤ 3): 0.3 x ITER
  - Neutron flux ≤ 2 MW/m^2: 3 x
  - Fluence = 1 MW-yr/m^2: 5 x
  - t_{pulse} ≤ 2 wks: 1000 x
  - Duty factor = 10%: 3 x

Low-aspect-ratio "spherical" tokamak (ST) is most compact embodiment of FNSF

High-Priority Research Areas for ST-FNSF

ReNeW Thrust 16 (2009): “Develop the ST to advance fusion nuclear science”

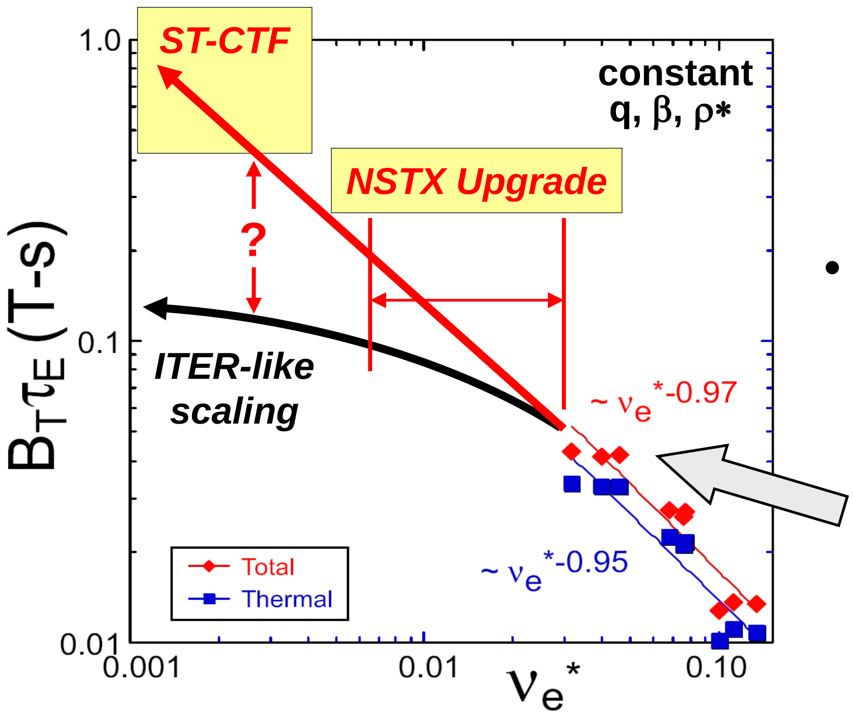
1. Develop **MA-level plasma current formation and ramp-up**
2. Advance **innovative magnetic geometries, first wall solutions**
3. Understand **ST confinement and stability** at fusion-relevant parameters
4. Develop **stability control techniques** for long-pulse, disruption-free ops
5. **Sustain current, control profiles** with beams, waves, pumping, fueling

6.Develop normally-conducting radiation-tolerant **magnets** for ST applications

7. **Extend ST performance** to near-burning-plasma conditions

This talk will focus on how NSTX and NSTX Upgrade address the ST-FNSF physics research needs (1-5) above

Access to reduced collisionality is needed to understand underlying causes of ST transport, scaling to next-steps



- Future ST's are projected to operate at 10-100x lower normalized collisionality ν*

Electron collisionality $\nu_e^* \propto n_e / T_e^2$

- Conventional tokamaks observe weak inverse dependence of confinement on ν*

STs observe much stronger ν* scaling

– Does favorable scaling extend to lower ν* ?

– What modes dominate e-transport in ST ?

- **NSTX H-mode thermal confinement scaling differs from higher aspect ratio scaling:**

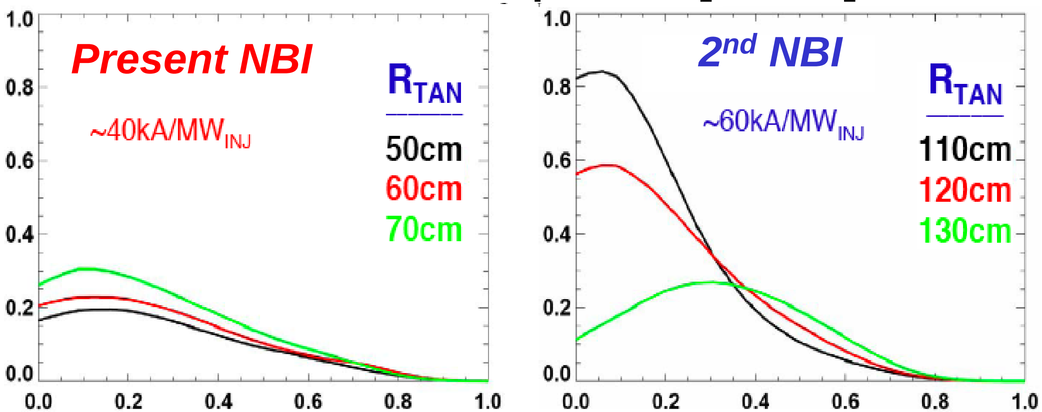
$\tau_{E,NSTX} \propto B_T^{0.9}\ I_P^{0.4}$ → strong B_T scaling $\quad \tau_{E,98y,2} \propto B_T^{0.15} I_P^{0.93}$ → weak B_T scaling

- **Upgrade: Double field and current for 3-6x decrease in collisionality → require 3-5x increase in pulse duration for profile equilibration**

Increased auxiliary heating and current drive are needed to fully exploit increased field, current, and pulse duration

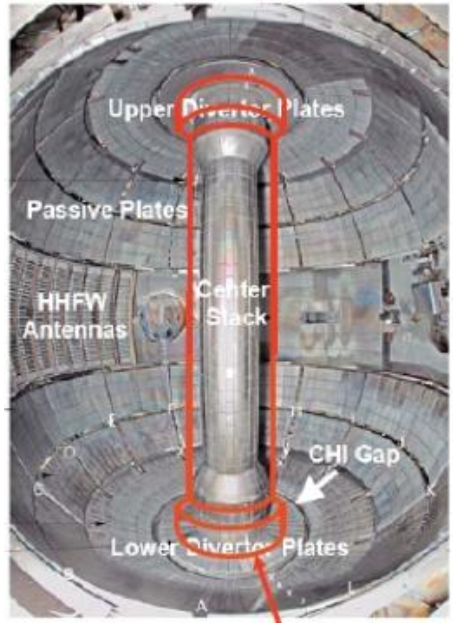
- Higher heating power to access high T and β at low collisionality
  - Need additional 4-10MW, depending on confinement scaling
- Increased external current drive to access and study 100% non-inductive
  - Need 0.25-0.5MA compatible with conditions of ramp-up and sustained plasmas
- **<u>Upgrade:</u> double neutral beam power + more tangential injection**
  - More tangential injection → up to 2 times higher efficiency, current profile control
  - ITER-level high-heat-flux plasma boundary physics capabilities & challenges

NBI current drive profiles [MA/m²]



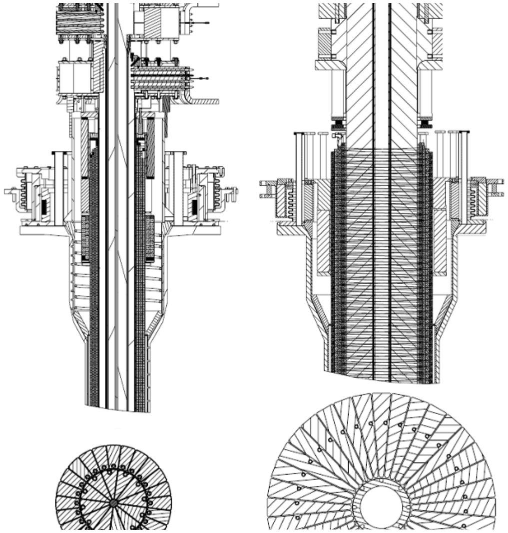
Normalized minor radius

NSTX Upgrade consists of two major elements that together bridge the device and performance gaps toward next-steps



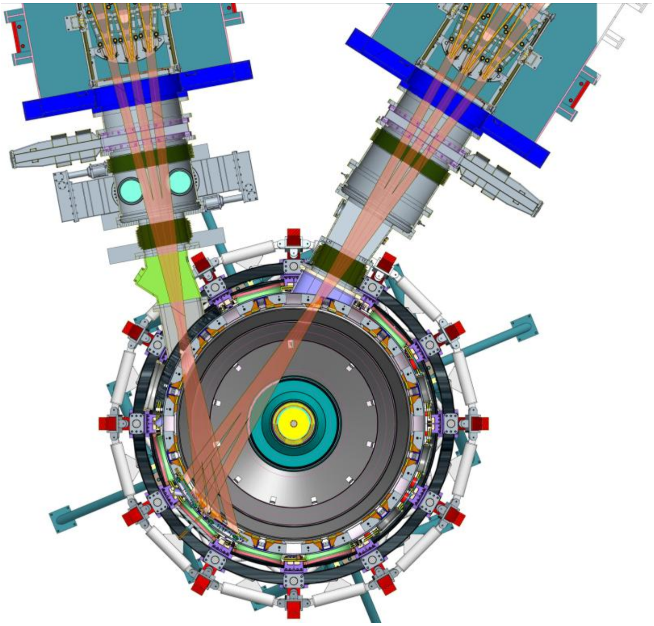
Outline of new center-stack (CS)

Present CS **New CS**



TF OD = 20cm **TF OD = 40cm**

New 2nd NBI **Present NBI**



	NSTX	NSTX Upgrade	Fusion Nuclear Science Facility
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.5
Plasma Current (MA)	1	2	4 → 10
Toroidal Field (T)	0.5	1	2-3
P/R, P/S (MW/m,m^2)	10, 0.2*	20, 0.4*	30 → 60, 0.6 → 1.2

** Includes 4MW of high-harmonic fast-wave (HHFW) heating power*

Center Stack Upgrade analysis and design are largely complete, and R&D activities are underway

B and J each increase 2x → electromagnetic forces increase 4x

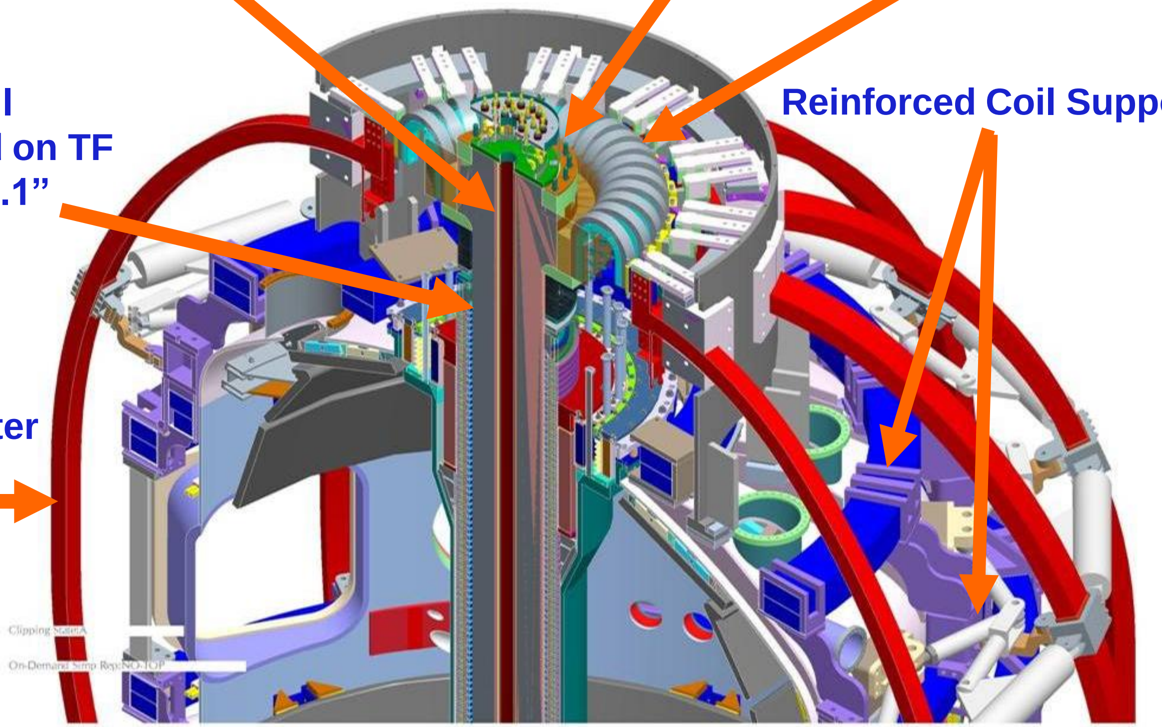
Simpler Inner TF design
(single layer of TF conductors)

Improved Joint Design

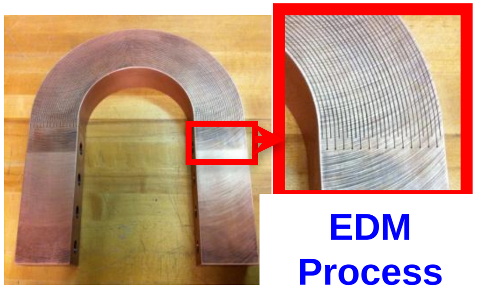
OH coil
wound on TF
(with 0.1” gap)

Reinforced Coil Supports

Existing
outer TF
WITH water
cooling



TF Flex Strap R&D

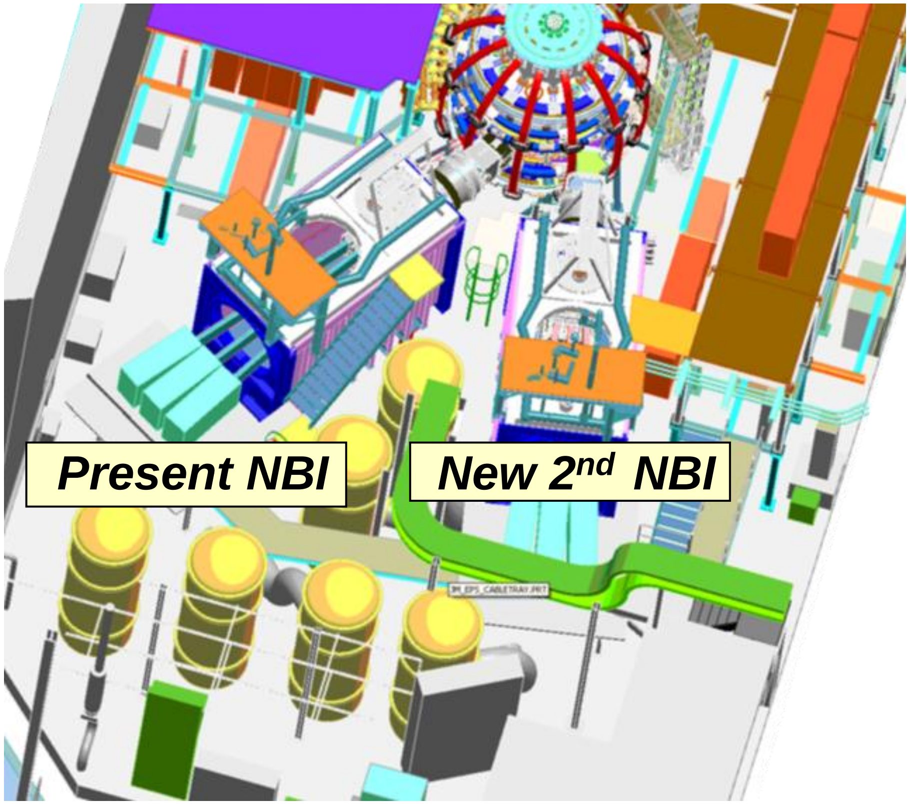


EDM Process

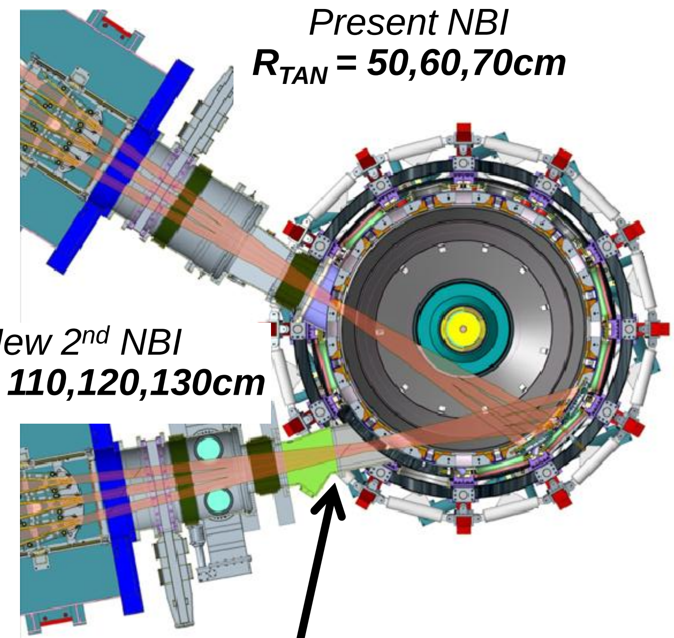
Successfully Tested to 300,000 cycles

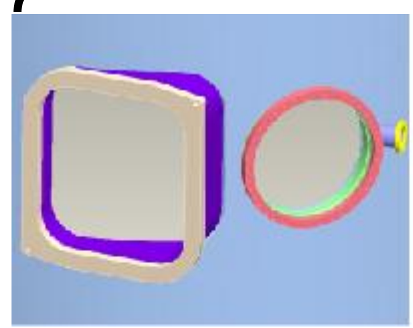
2nd NBI requires relocation of a TFTR NBI system to NSTX, diagnostic relocations, new port for more tangential NBI

NSTX

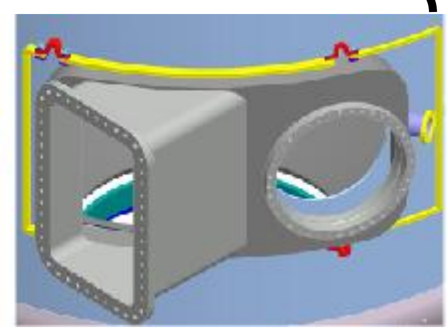


- **Decontamination of 2nd Beam line successfully completed in 2010**
- **Reassembly of 2nd Beam line has started**





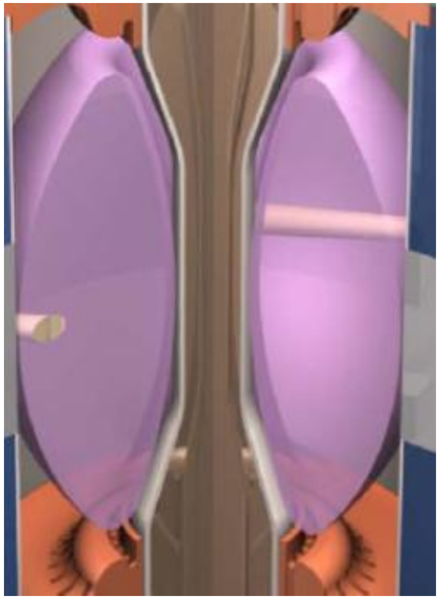
Original NBI Port

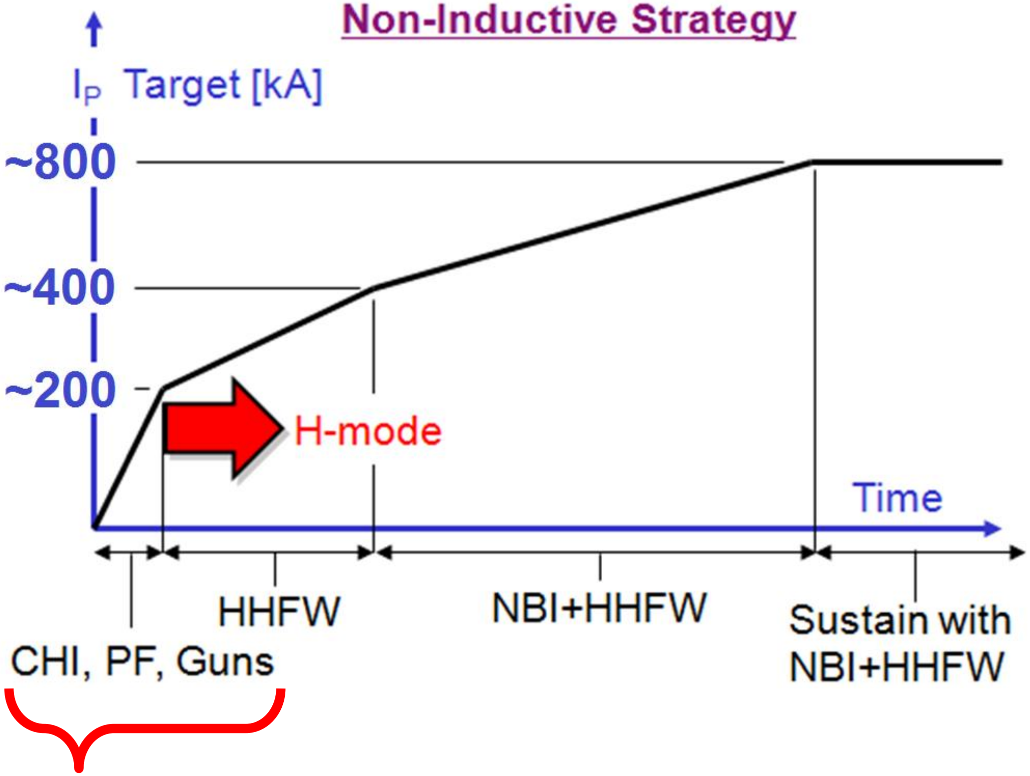


New NBI Port

Plasma initiation with small or no transformer is unique challenge for ST-based Fusion Nuclear Science Facility

ST-FNSF has no/small central solenoid

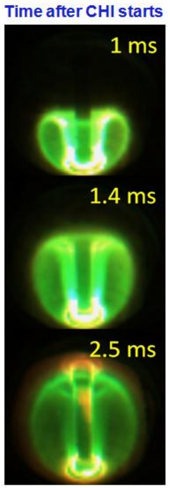


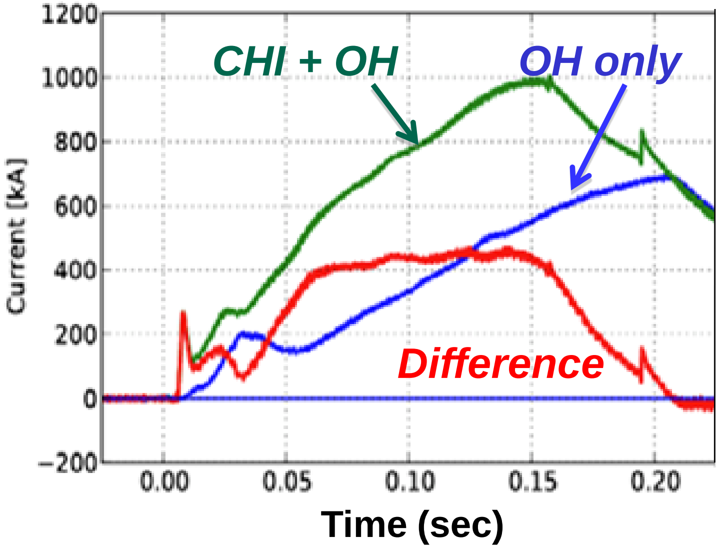


- Near-term NSTX Goal: Generate ~0.3-0.4MA full non-inductive start-up with Coaxial Helicity Injection + fast wave heating + NBI (need Upgrade)
- Upgrade goal: Provide physics basis for non-inductive ramp-up to high performance 100% non-inductive ST plasma → prototype FNSF

Achieved substantial progress on Coaxial Helicity Injection (CHI) and fast wave heating of low-current plasmas in 2010

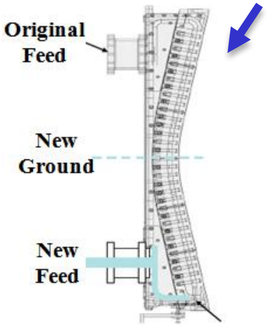
- **Early in shot: produce 150-200kA**
- **Generated 1MA using 40% less flux than induction-only case**
  - **Low $l_i \approx 0.35$, and high elongation > 2 → suitable for advanced scenarios**

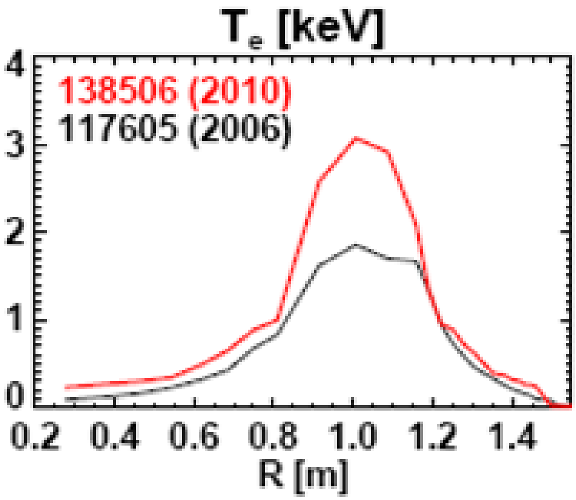




- CHI-driven current scales linearly with B_T → **2x higher in Upgrade**

- **Achieved high $T_e(0)$ ~ 3keV at I_P=300kA w/ only 1.4MW of HHFW**
  - **Previous best was $T_e(0)$ ~ 1.5keV at twice the RF power**
  - **Enabled by 2009 antenna upgrades**

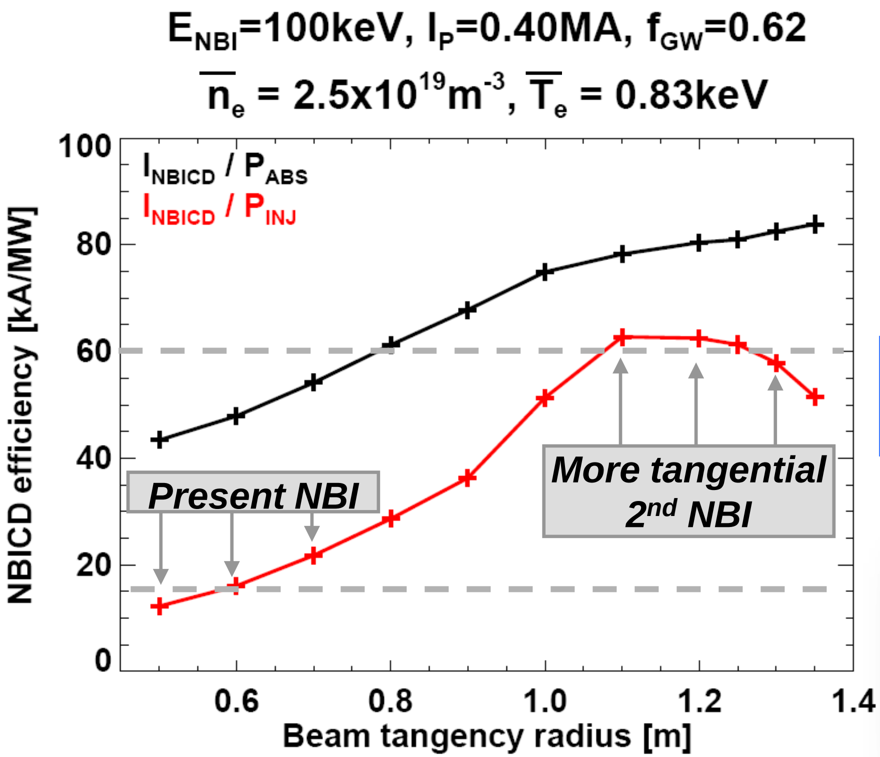


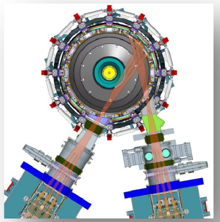


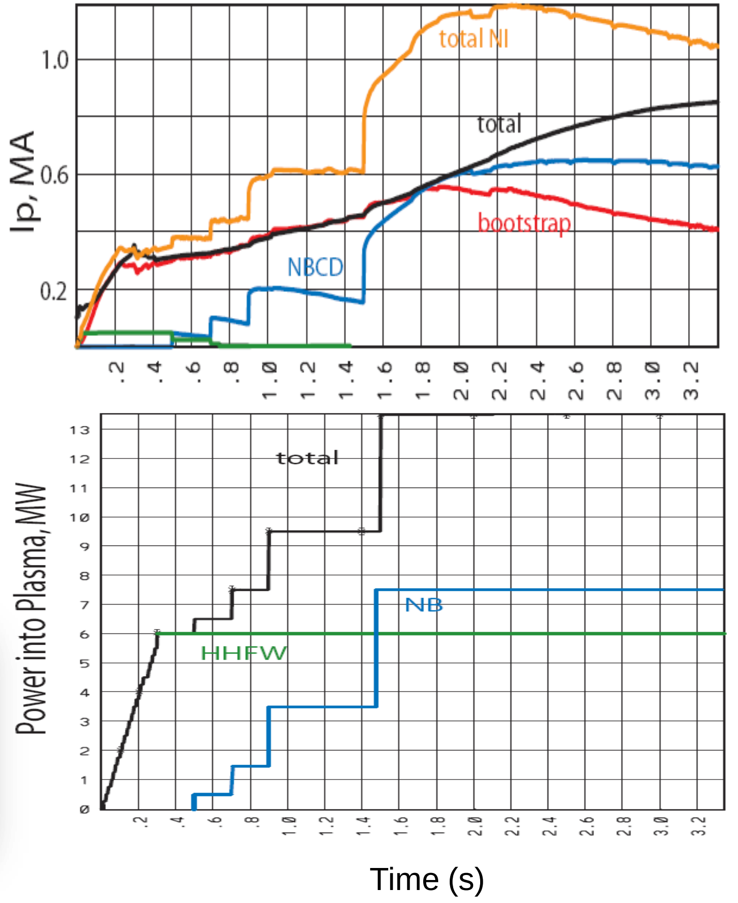
- Non-inductive fraction ~60-70% with 25-30% from RFCD from high $T_e(0)$
- **Projects to ~100% NI at P_{RF} = 3-4MW**

Non-inductive ramp-up from ~0.4MA to ~1MA projected to be possible with new CS + more tangential 2nd NBI

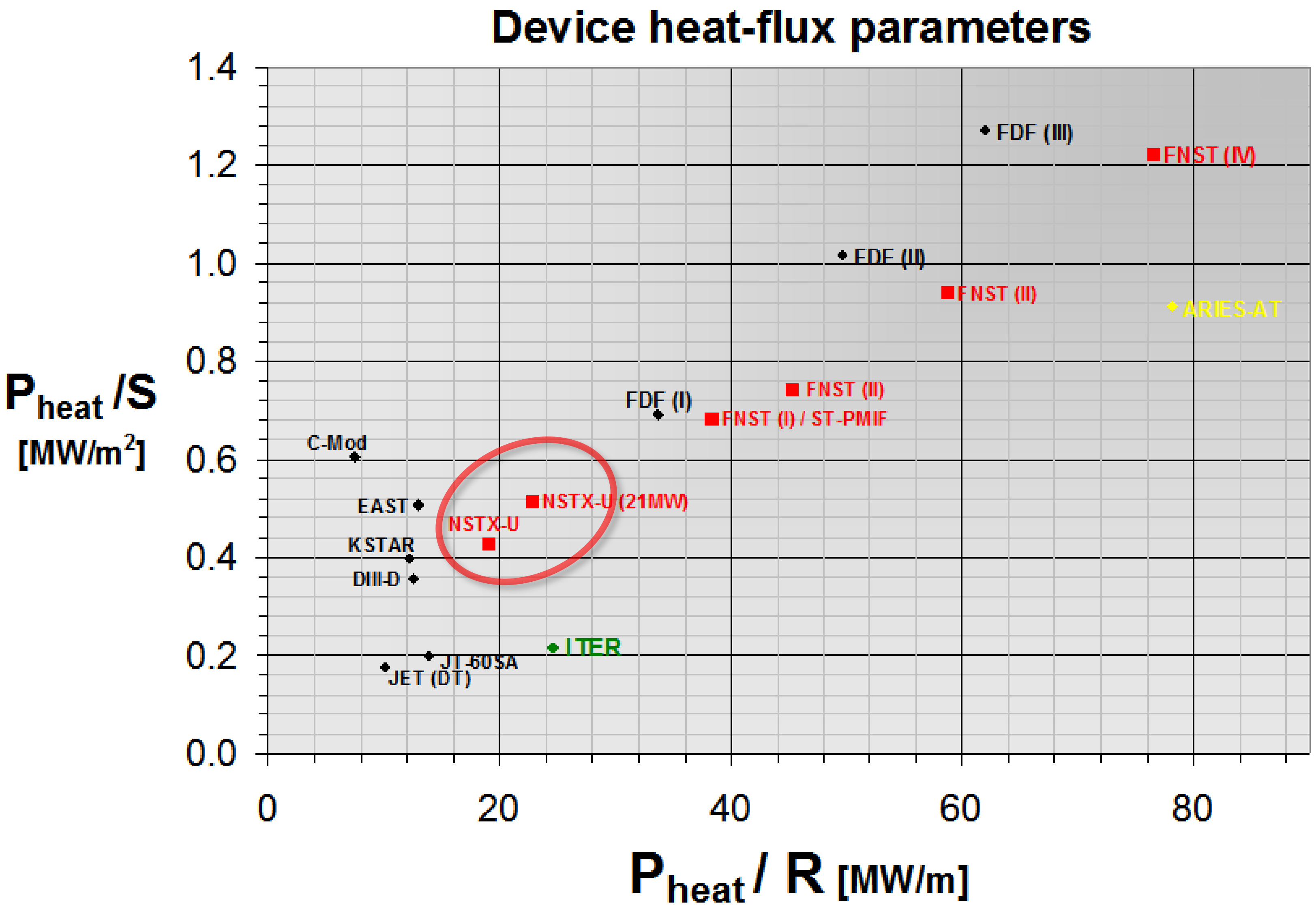
- **New CS provides higher TF (improves stability), 3-5s needed for J(r) equilibration**
- **More tangential injection provides 3-4x higher CD at low I_P:**
  - 2x higher absorption (40→80%) at low I_P = 0.4MA
  - 1.5-2x higher current drive efficiency





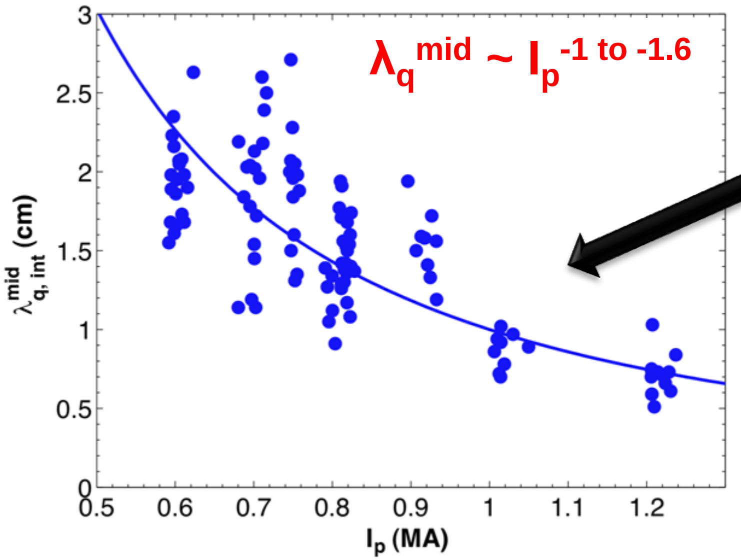


NSTX Upgrade will extend normalized divertor and first-wall heat-loads much closer to FNS and Demo regimes



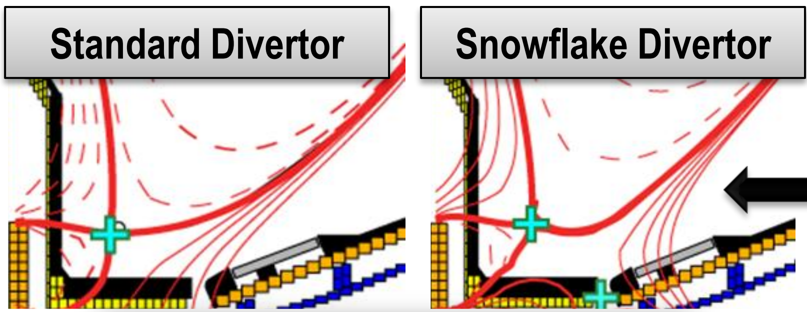
NSTX has contributed strongly to divertor heat flux width studies*, and is developing new heat-flux mitigation methods

**Joint Research Milestone (3 U.S. Facilities)*



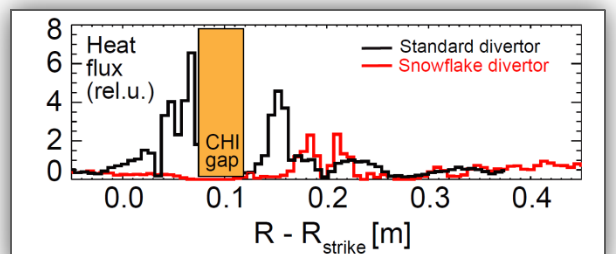
- Divertor heat flux width decreases with increased plasma current I_P
  - Potentially major implications for ITER

→ NSTX Upgrade with conventional divertor projects to very high peak heat flux up to 30-45MW/m²



- Divertor heat flux inversely proportional to flux expansion over a factor of five

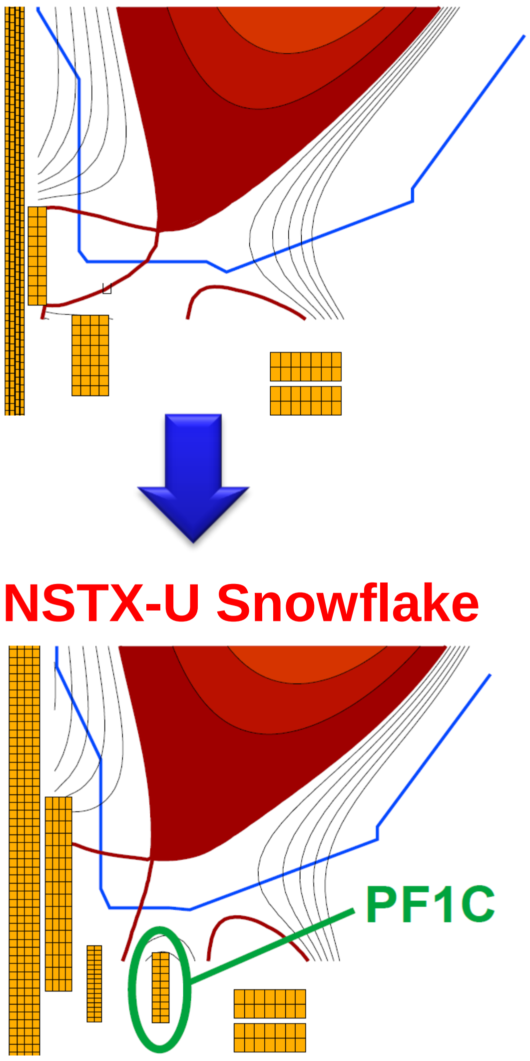
Snowflake→ high flux expansion 40-60, larger divertor volume and radiation

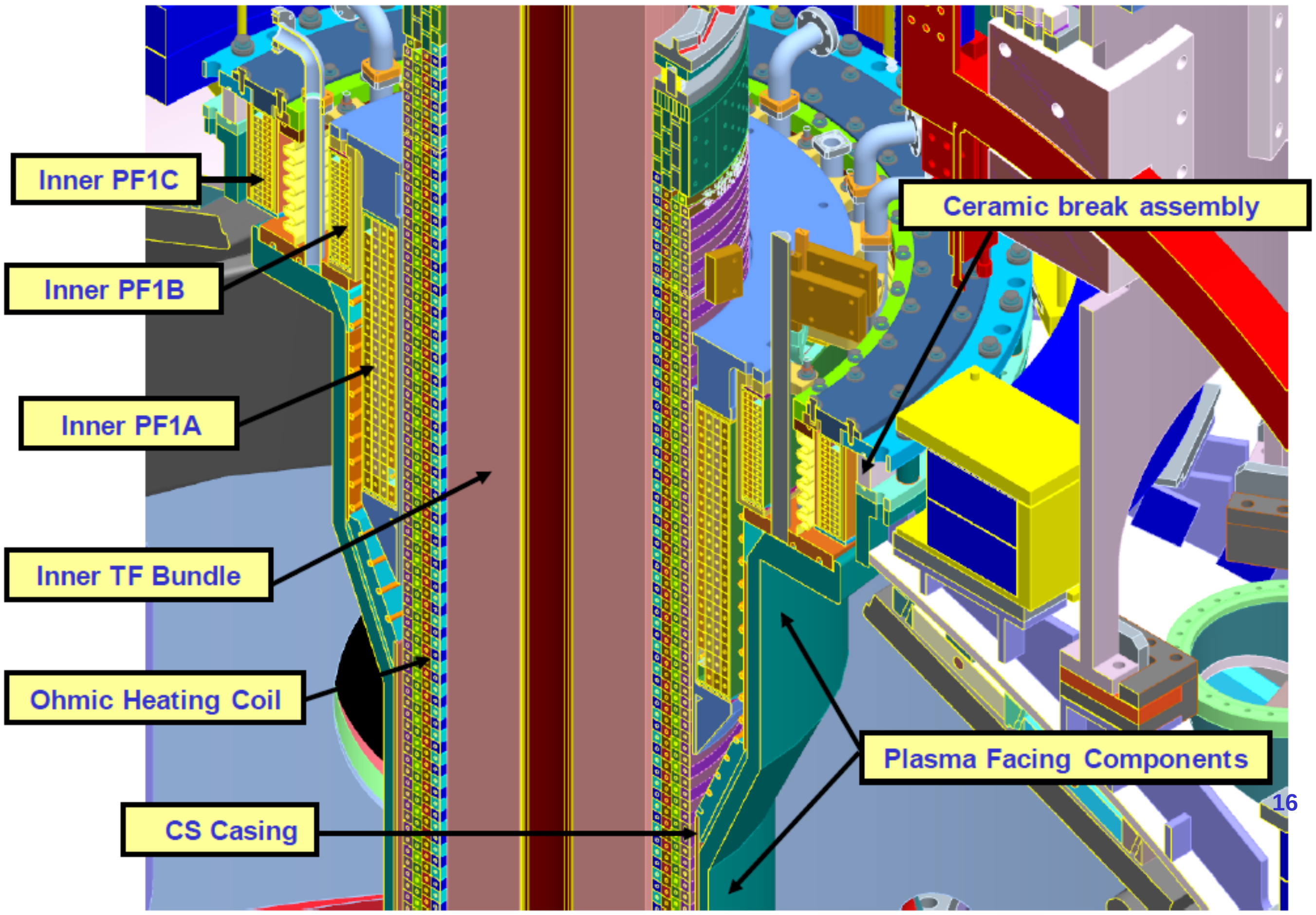


→ U/D balanced snowflake divertor projects to acceptable heat flux < 10MW/m² in Upgrade at highest expected I_P = 2MA, P_{AUX}=10-15MW

Upgrade CS design provides additional coils for flexible and controllable divertor including snowflake, and supports CHI

NSTX Snowflake

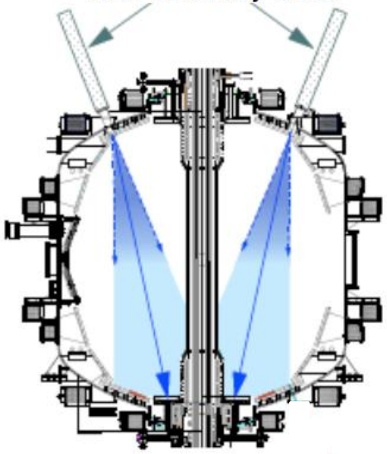




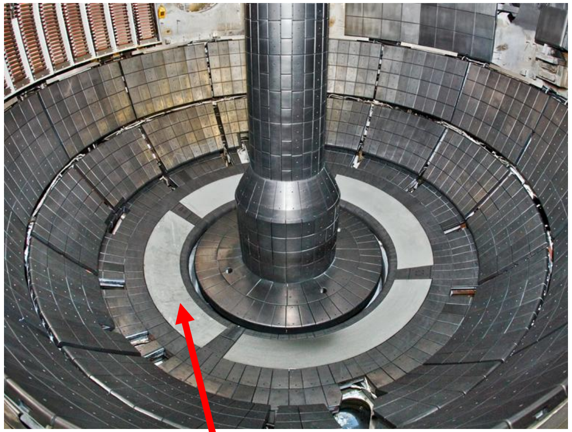
NSTX is a world leader in assessing lithium plasma facing components as a possible PMI solution for magnetic fusion

- **<u>Solid Li surface coatings</u>:** Pump D, increase confinement, stored energy, and pulse length, eliminate ELMs, reduce core MHD instabilities

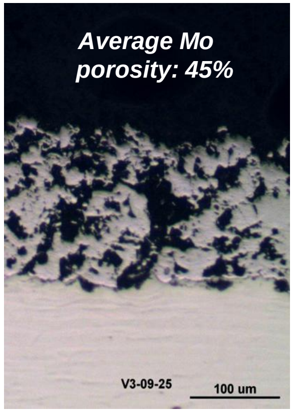
Dual Liquid Lithium Evaporator
For Li wall coatings
Now routinely used



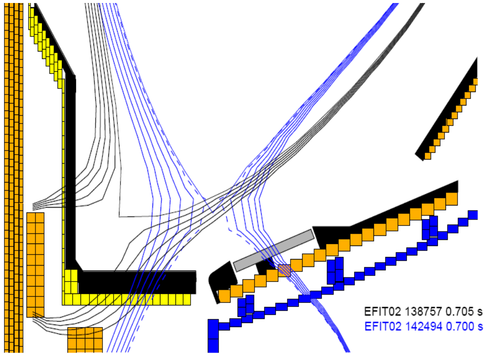
- **<u>Liquid Lithium Divertor (LLD) motivation:</u>**
  – Provide volume D pumping capacity (> solid Li coatings) for increased pumping and duration
  – Potential for handling high heat flux (longer term)



4 heatable LLD plates (Mo on Cu)
Surface temp: 160 – 350+ °C

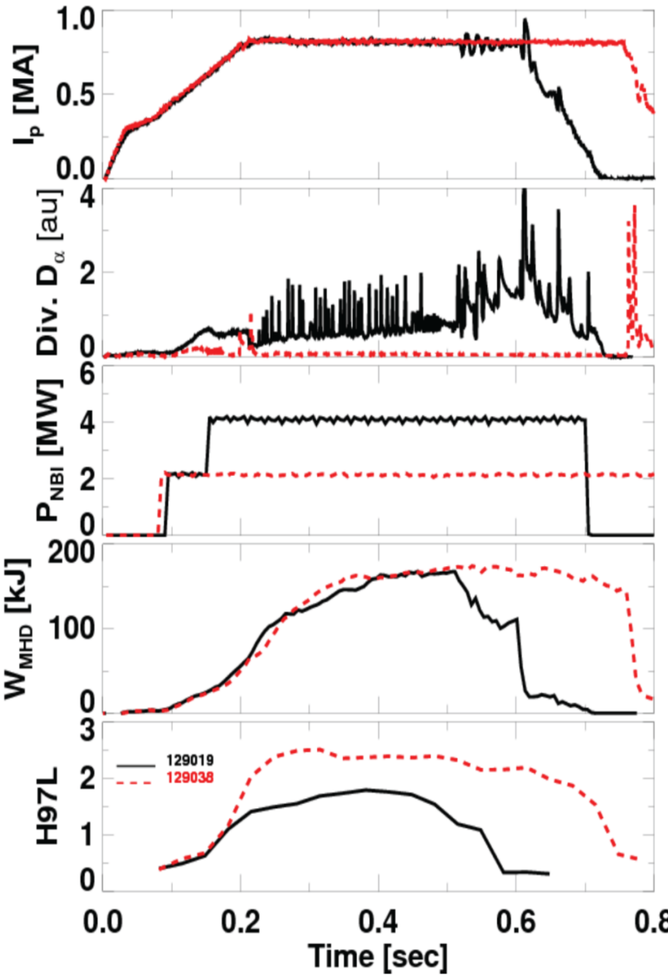


LLD surface cross section: plasma sprayed porous Mo

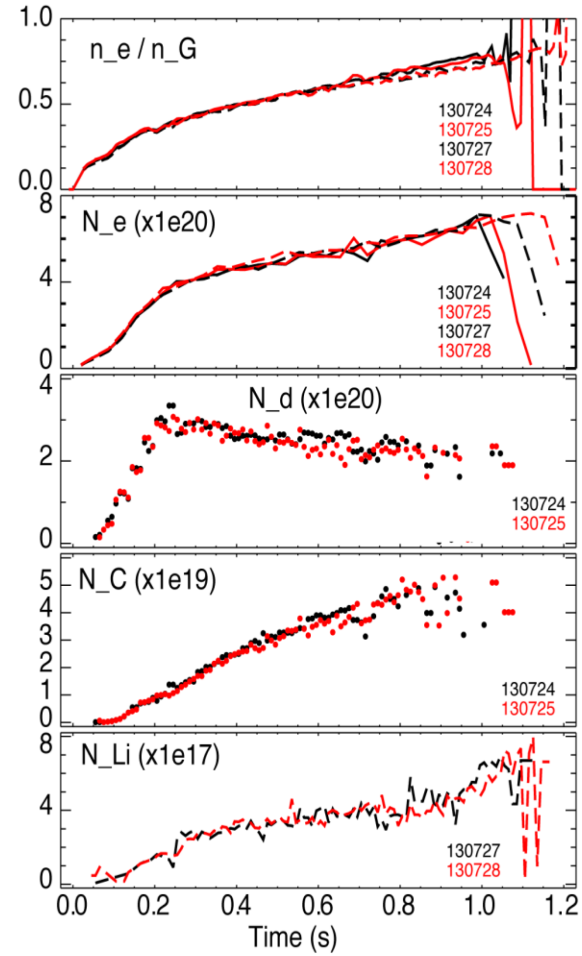


<u>Controlled scans of strike-point location:</u>
On inboard divertor
On LLD (outboard divertor)

Solid Li surface coatings: pump D, increase energy confinement, eliminate ELMs, but confine impurities too well



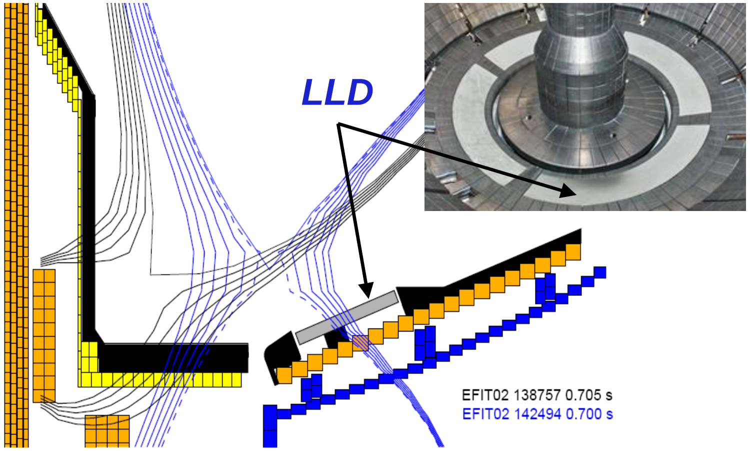
- ◀ Without Li
- ◀ With Li
- ◀ ELM-free, reduced divertor recycling
- ◀ Lower NBI to avoid β limit
- ◀ Similar stored energy
- ◀ H-factor increases up to 40%



- ◀ Monotonic increase in electron density
- ◀ Deuterium inventory pumped out
- ◀ But carbon inventory increases (Z_{eff} ~ 3-4)
- ◀ Li inventory rises slowly, concentration is very low

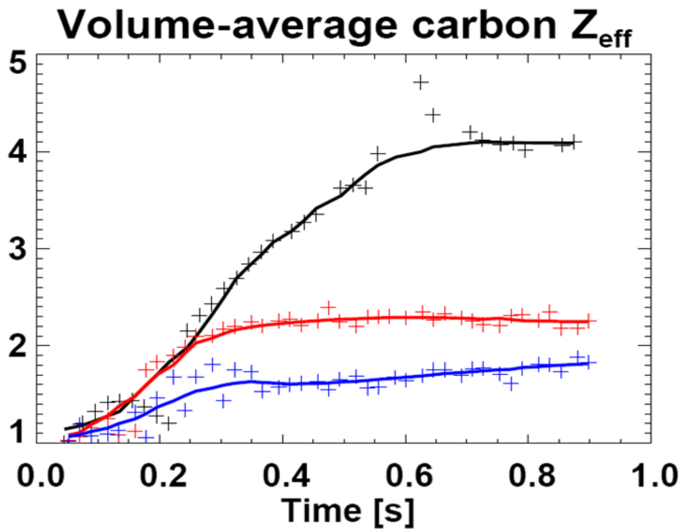
Based on these results, NSTX is shifting emphasis from D inventory control to C impurity reduction

2010: Operation with outer strike-point on Molybdenum LLD (coated w/ Li) successful, achieved high plasma performance



LLD FY2010 results:

- LLD did not increase D pumping beyond that achieved with LiTER
- No evidence of Mo from LLD in plasma during normal operation
- Operation with strike-point on LLD can yield <u>reduced</u> core impurities



◄Strike-point on inner C divertor (no ELMs)

◄Strike-point on LLD, $T_{LLD} < T_{Li\text{-}melt}$

◄Strike-point on LLD, $T_{LLD} > T_{Li\text{-}melt}$ (+ fueling differences)

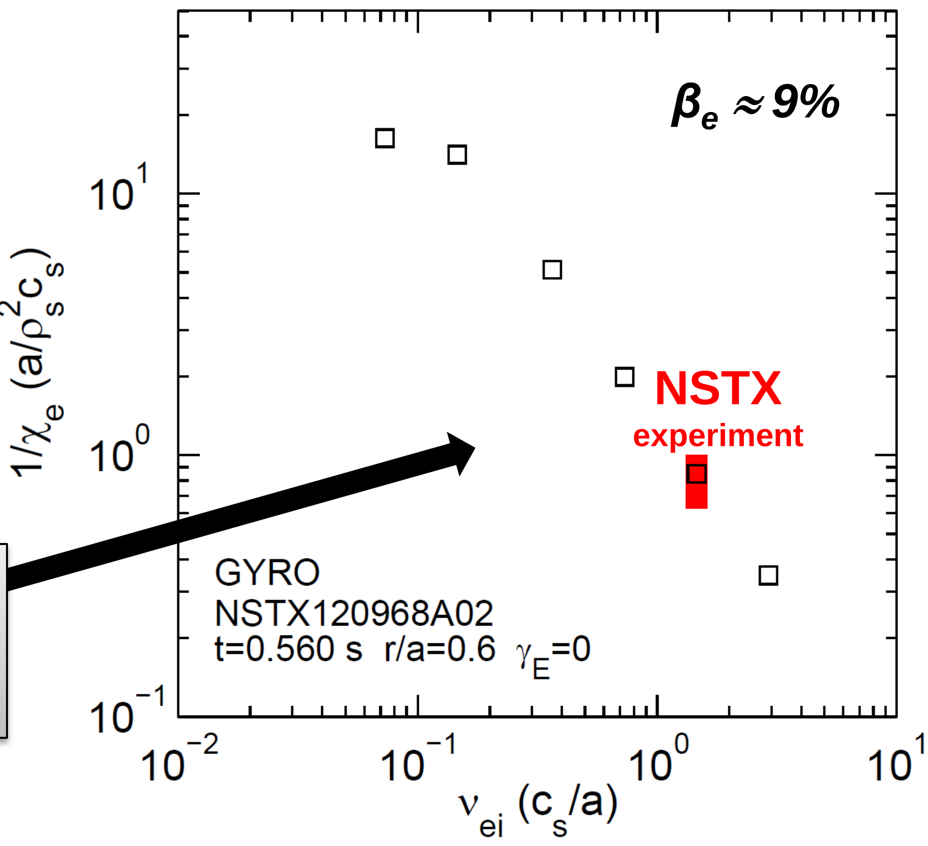
- No ELMs, no → small, small → larger

Li + plasma-facing component research will be continued, extended in NSTX-U

New NSTX turbulence simulations are advancing the understanding of ST energy confinement

- Non-linear gyrokinetic turbulence simulations of micro-tearing instabilities predict $\tau_E \propto 1/\chi_e \propto 1/\nu_e^*$

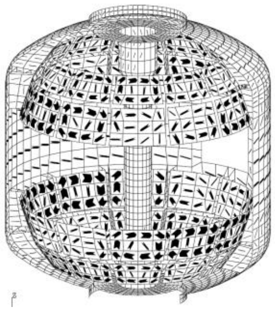
- Predominantly electromagnetic turbulence – result of high β
- Candidate explanation for ST confinement scaling observed on NSTX and MAST



Lower ν* accessible in Upgrade will clarify roles of micro-tearing vs. ETG, TEM in ST e-transport

NSTX is 1st tokamak to implement advanced resistive wall mode state-space controller, utilized it to sustain high $\beta_N \sim 6$

Full 3-D model

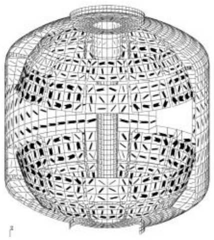


~3000+ states

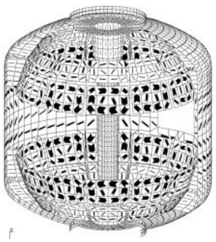
State reduction (< 20 states)

RWM eigenfunction (2 phases, 2 states)

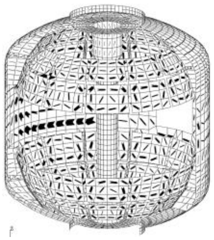
$(\hat{x}_1, \hat{x}_2)$



$\hat{x}_3$



$\hat{x}_4$



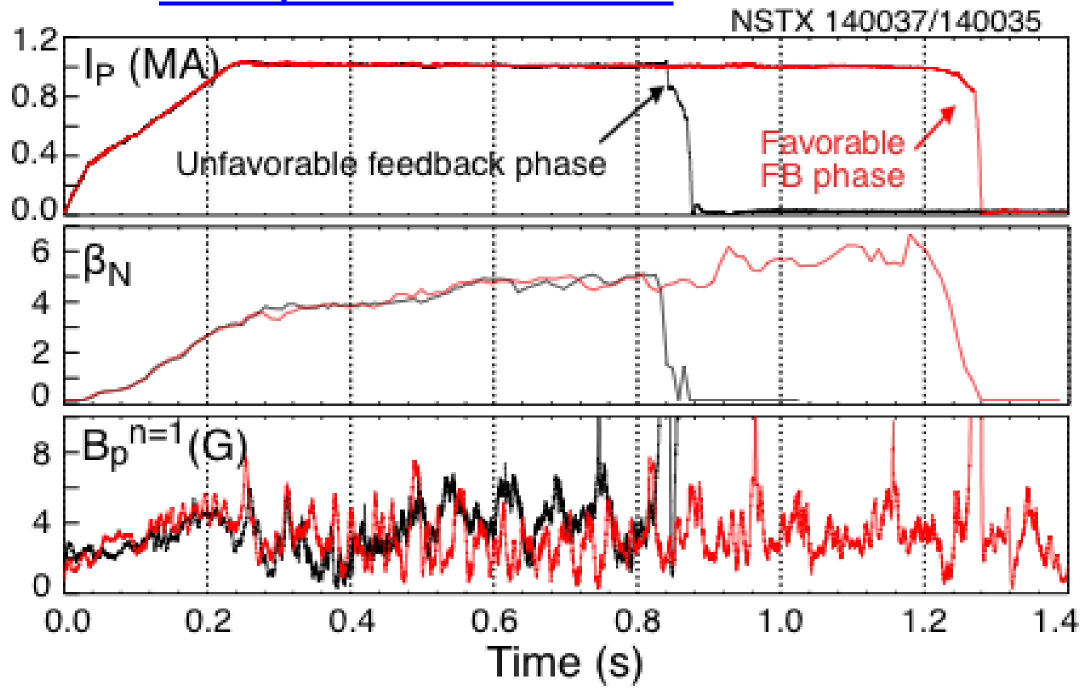
$\hat{x}_N$

truncate

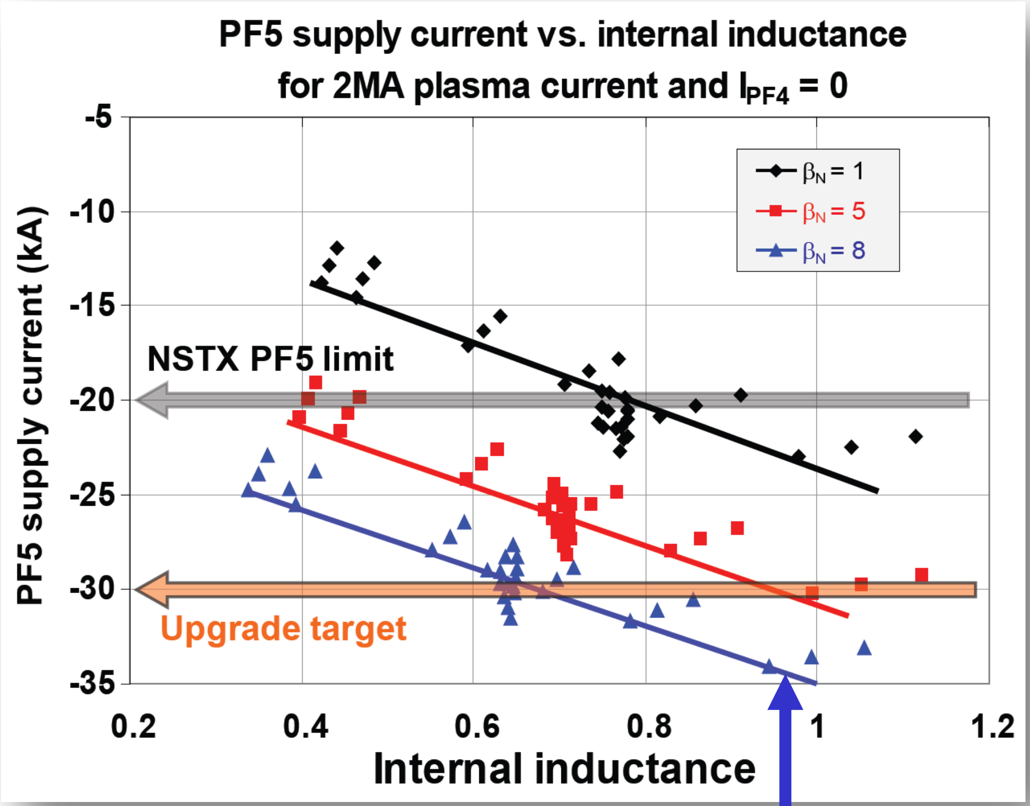
- *Device R, L, mutual inductances*
- *Instability B field / plasma response*
- *Modeled sensor response*

- **Controller can compensate for wall currents**
  - Including mode-induced current
  - Examined for ITER
- **Successful initial experiments**
  - Suppressed disruption due to n = 1 applied error field
  - Best feedback phase produced long pulse, β_N = 6.4, β_N / l_i = 13

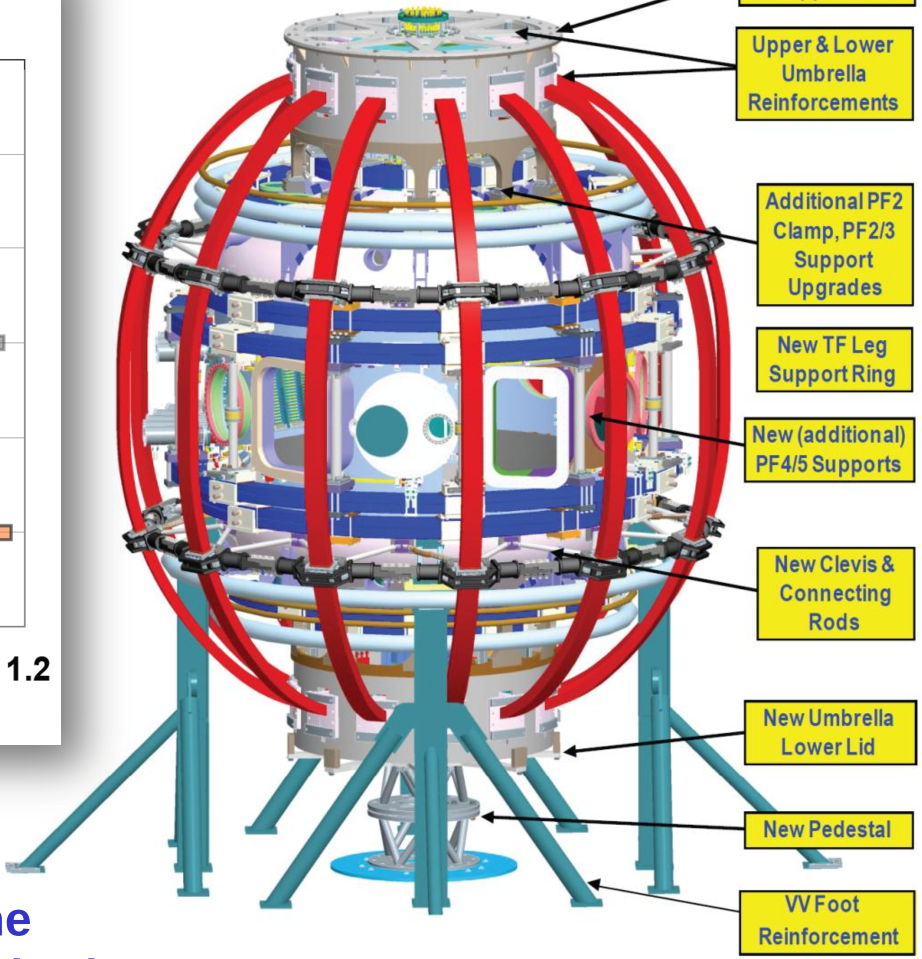
State space feedback with 12



Upgrade structural enhancements designed to support high β at full I_P = 2MA, B_T=1T: $\beta_N = 5$, $l_i \leq 1$ and $\beta_N = 8$, $l_i \leq 0.6$

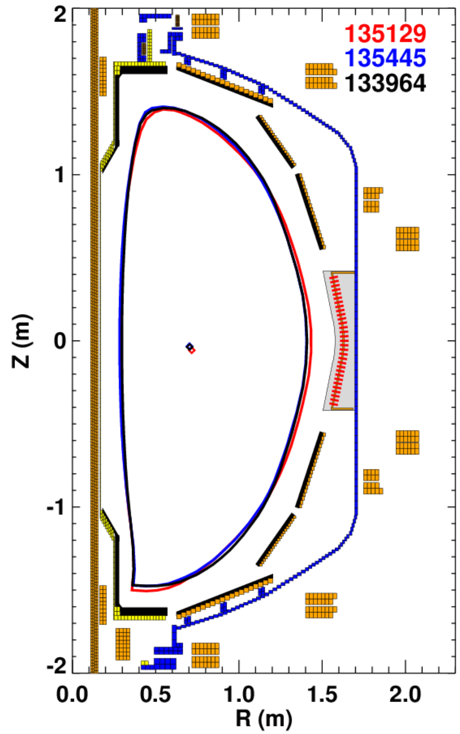


High l_i, high-β_N scenarios determine the maximum vertical field (PF5) current required



In 2009-10, NSTX demonstrated sustained high-elongation configurations over a range of currents and fields

High-β_T	Long Pulse	High-β_P
q=2.8*	*q*=3.9*	*q*=4.7*
B_T=0.44 T	B_T=0.38 T	B_T=0.48 T
I_P=1100 kA	I_P=700 kA	I_P=700 kA

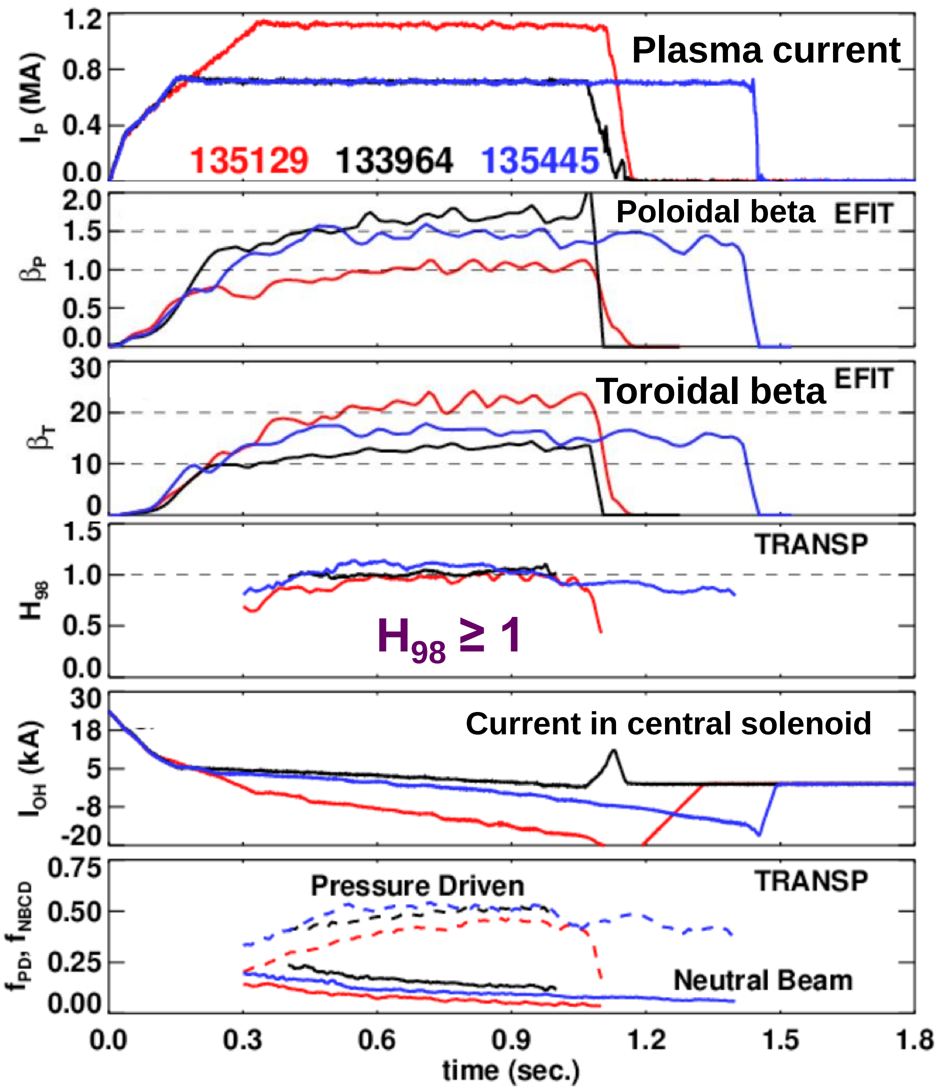


$$q^* = \frac{\varepsilon\left(1+\kappa^2\right)\pi a B_{T0}}{\mu_0 I_P}$$

κ ~ 2.6-2.7

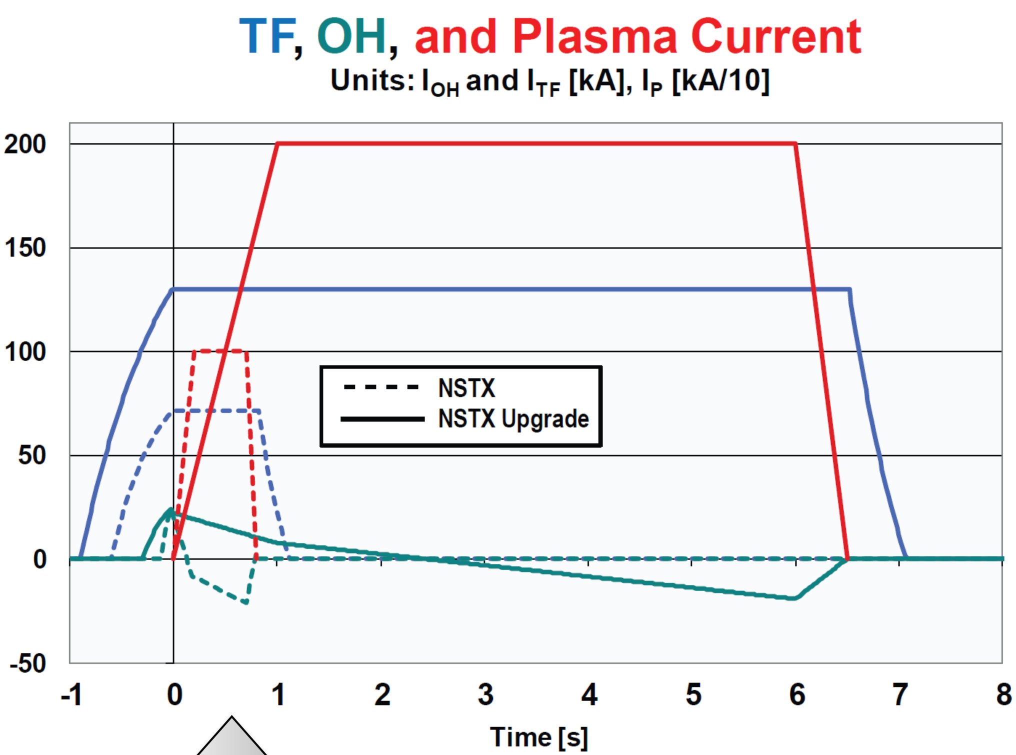
δ ~ 0.8

Double Null



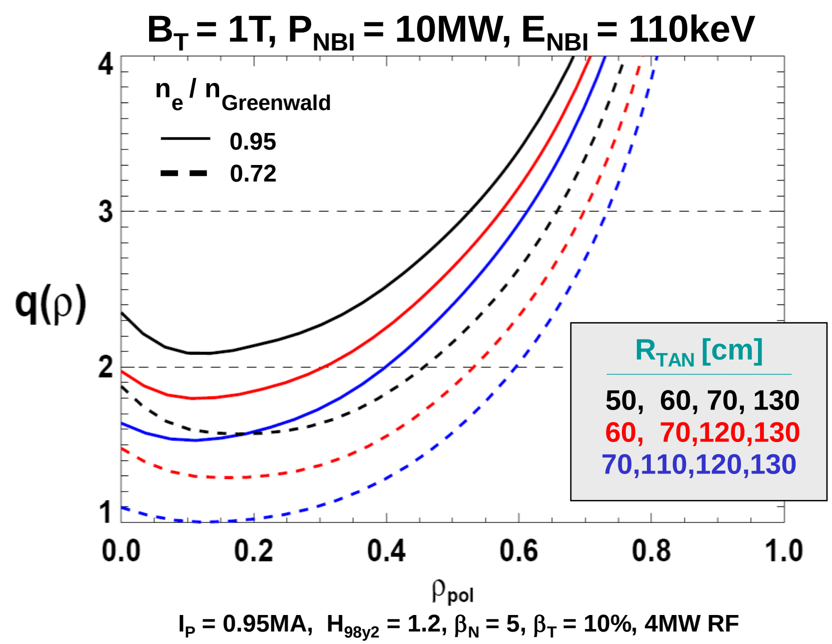
Pulse-lengths limited by OH, TF coil heating limits

NSTX Upgrade supports 5x longer pulses and 100% non-inductive current drive, ultimately with *q* profile control



Fully non-inductive scenarios with q(r) profile will be controllable by:

- NBI source tangency radius
- plasma density
- plasma position (not shown)

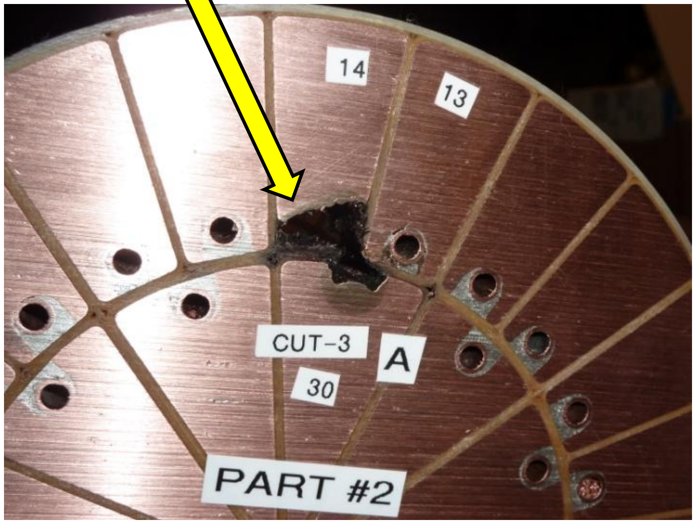


Changes from NSTX to NSTX Upgrade:

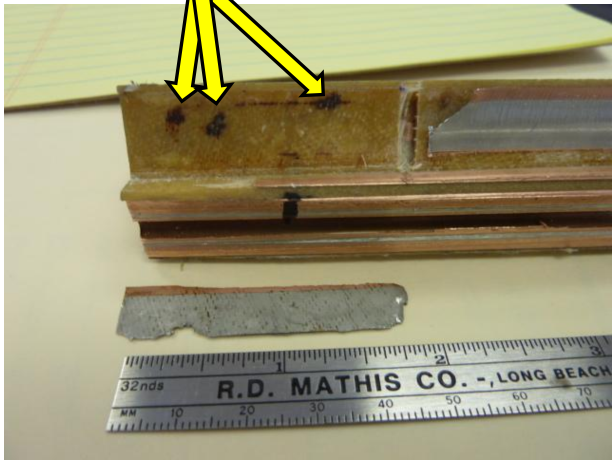
- I_P and B_T 2x higher, 3x OH flux, flat-top 5x longer, W_{TOT} up to 4x higher
- Minimum Aspect Ratio A = 1.3 → 1.5, inter-shot time increased ~2x

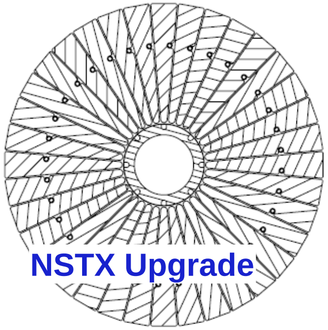
NSTX inner TF bundle experienced irreparable turn-to-turn electrical short at beginning of FY2011-12 run (July 20, 2011)

- Fault accessed by dissecting TF bundle



- Cause of fault traced to solder flux contamination of insulation





- **NSTX Upgrade TF design has several improvements:**
  - Single layer instead of 2 layers → reduced turn-to-turn voltage
  - VPI instead of B-stage (pre-preg) insulation
  - Lesson learned from TF fault: will use rosin-based (organic) flux instead of ZnCl-based flux, improve flux removal techniques

Plan: Start Upgrade ASAP, finish 6-9 mo. earlier than originally planned

Summary: NSTX and NSTX Upgrade strongly support FNSF development, Materials/PMI, and ITER

- **NSTX Research Highlights:**
  - CHI+OH plasma current savings up to 400kA, RF heating of low I_P to 3keV
  - Established divertor heat flux scalings, advancing snowflake divertor, Li
  - Non-linear simulations suggest micro-tearing may influence ST transport
  - High β_N ~ 6 sustained with advanced RWM control
  - Long-pulse plasmas developed – duration limited by magnet capabilities
- **NSTX Upgrade Progress:**
  - Design supports CHI/start-up, PMI, transport, high-β, 100% NICD research
  - New center-stack design and analysis complete – fabrication beginning
  - 2nd NBI relocation/installation – ready to begin during Upgrade outage
- **NSTX Upgrade Schedule:**
  - Project base-lined (CD-2) December 2010
  - Final Design Review held June 2011, CD-3 review to be held October 2011
  - NSTX Upgrade outage to begin late 2011
  - NSTX Upgrade first plasma → early 2014