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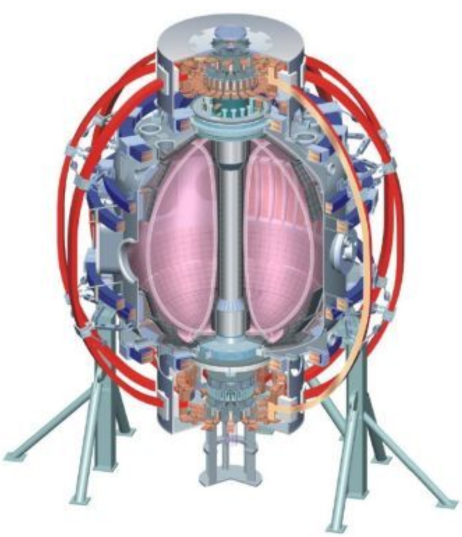
NSTX Research Program Overview for 2009-11 and Beyond

Jon Menard, PPPL

For the NSTX Research Team

NSTX PAC-25 Meeting
LSB B318, PPPL
February 18-20, 2009



College W&M
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U Quebec

NSTX 5 year plan for 2009-13 was favorably reviewed

- "Proposed research clearly aims to position the ST as a candidate for future high priority US research missions, as articulated in recent FESAC reports
  - High heat flux facility for PMI research, as embodied in NHTX
  - Nuclear component testing, as embodied in ST-CTF"

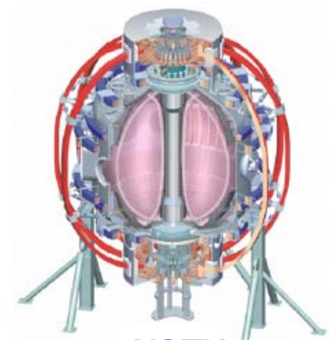
- "The panel agrees that the proposed research priorities address these missions
  - 100% non inductive current drive
  - Particle and heat flux control
  - Non inductive start up and ramp up
  - Sustained high beta operation"

- "The major facility upgrades are appropriately sequenced:
  1. The liquid lithium divertor (LLD) is an innovative approach to density control
     - Potential for high reward, but no guarantee LLD will provide necessary control
     - Measuring and modeling effects associated with lithium will be critical to understanding the science and projecting future applications.
     - It is not clear that there is sufficient attention paid to this in the proposal.
     - A backup strategy for density control should be better developed
  2. The center stack upgrade is very well motivated and should be installed as soon as possible
  3. The second neutral beam source is essential to take advantage of higher B_T and current capability from center stack upgrade"

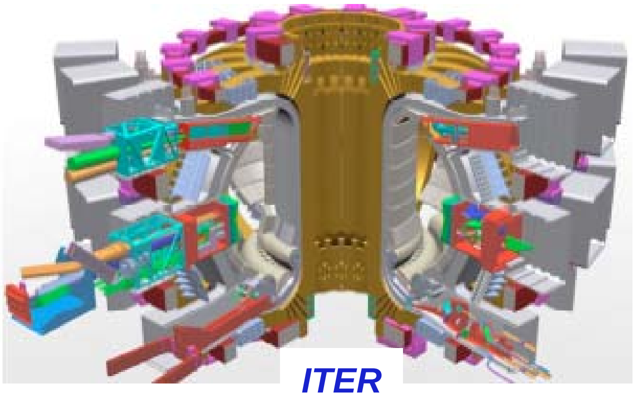
NSTX advances toroidal plasma science and burning plasma physics, and provides attractive near-term fusion options

NSTX Mission Elements:

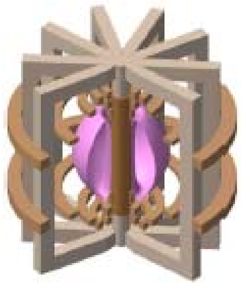
- Understand unique physics properties of ST
  - Assess impact of low A, high β, high v_{fast} / v_A, etc. on all aspects of toroidal plasma science
- Complement tokamak physics, support ITER
  - Exploit unique ST features to improve tokamak understanding, while also benefiting from tokamak R&D
- Establish attractive ST operating conditions for future fusion applications
  - **Long-term goal:** Understand and utilize advantages of the ST configuration for addressing key gaps between ITER performance and that needed for DEMO



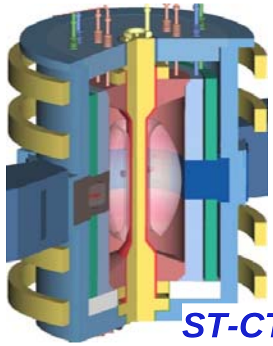
NSTX



ITER



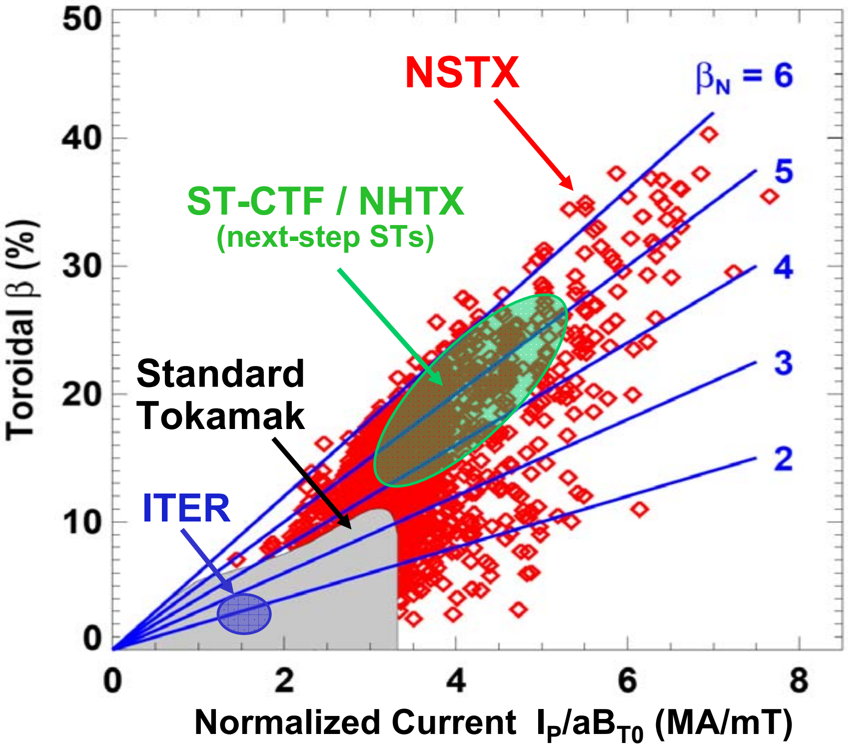
NHTX



ST-CTF

Pre-conceptual designs

NSTX creates stable, well diagnosed plasmas at high β enabling a wide range of toroidal physics studies

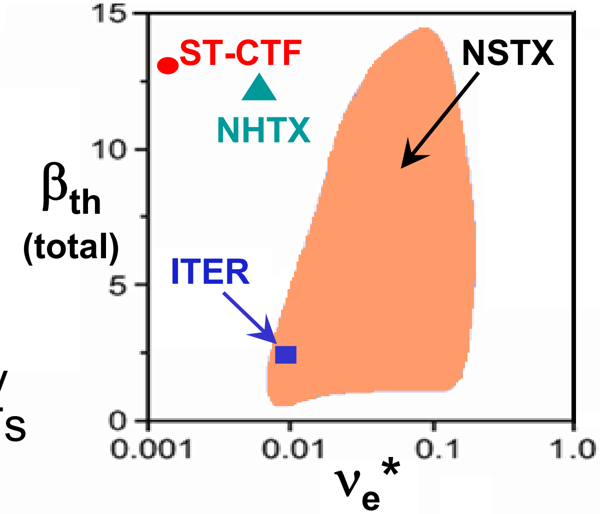


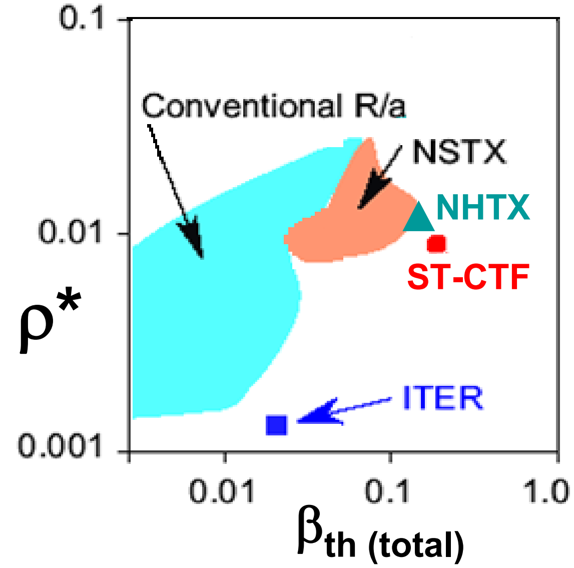
- ST accesses higher normalized current & higher normalized β

➡ higher $\beta_{Toroidal}$

(High β_N results in part from rotational stabilization of resistive wall mode)

- Access ITER-level ν*, extending confinement understanding to high β
- Next-step STs expected to operate at significantly lower ν* than present STs
- ST operates at higher ρ* than tokamaks / ITER - impacts thermal and fast-ion transport, MHD
- Extrapolation in ρ* from present STs to next-step STs is small

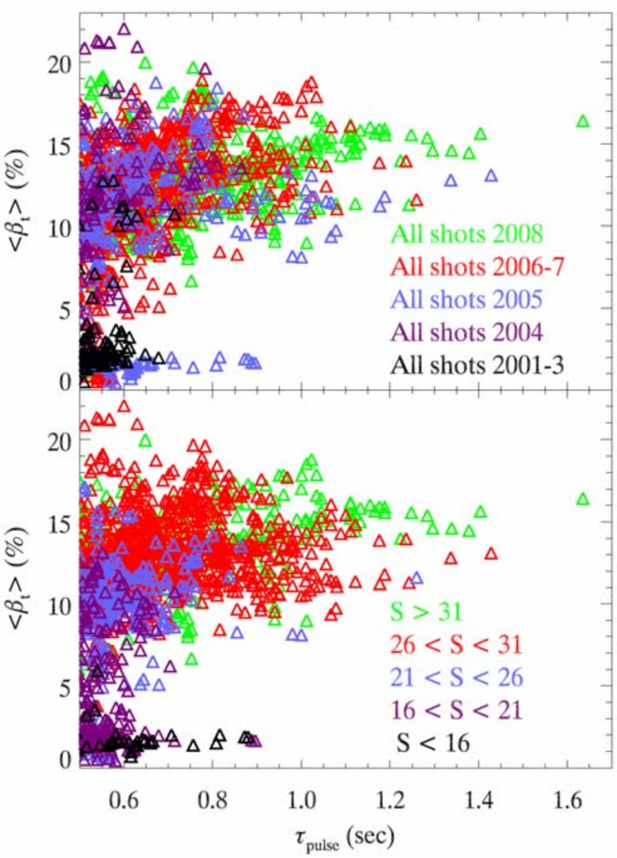




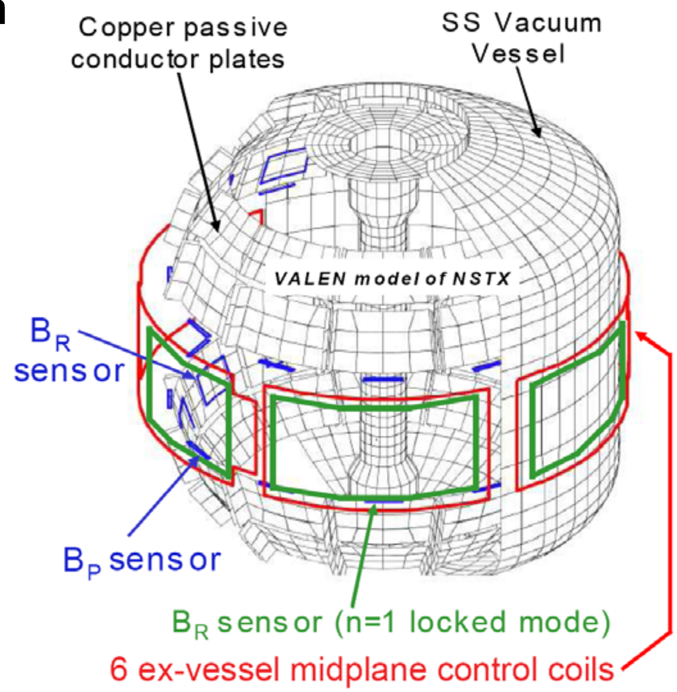
Improved control of plasma instabilities has significantly increased the duration of sustained high β in NSTX

Increased plasma shaping from improved n=0 control for high κ and δ operation **+ n ≥ 1 EF/RWM control =** **Duration of β_T > 15% increased factor of 4 from 2002 to 2008**

NSTX has sustained β_T needed for ST-CTF for 4 current redistribution times

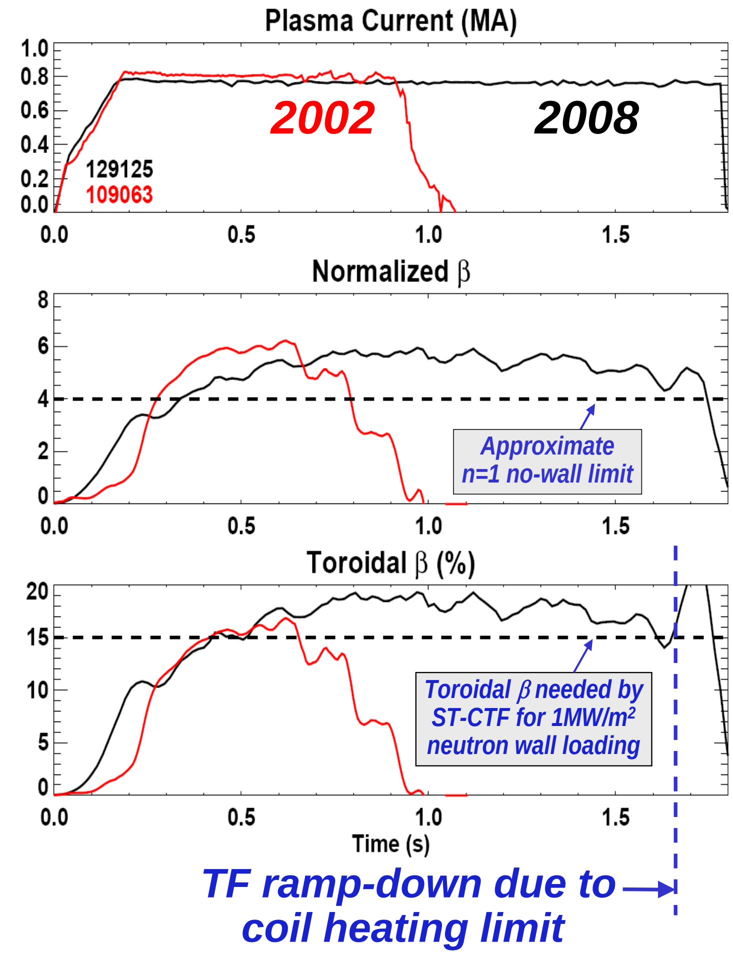


$S \equiv q_{95}\, I_P/aB_T$ [MA/mT]



Control coils also used to study:

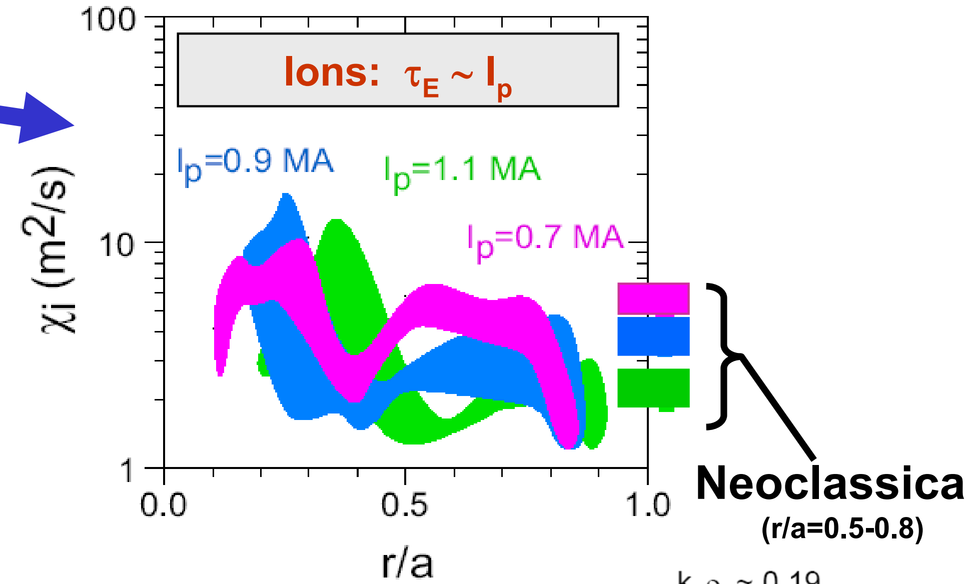
- Locked mode thresholds
- Resonant field amplification
- Rotation damping from NTV
- Anomalous momentum transport
- Pedestal transport and stability

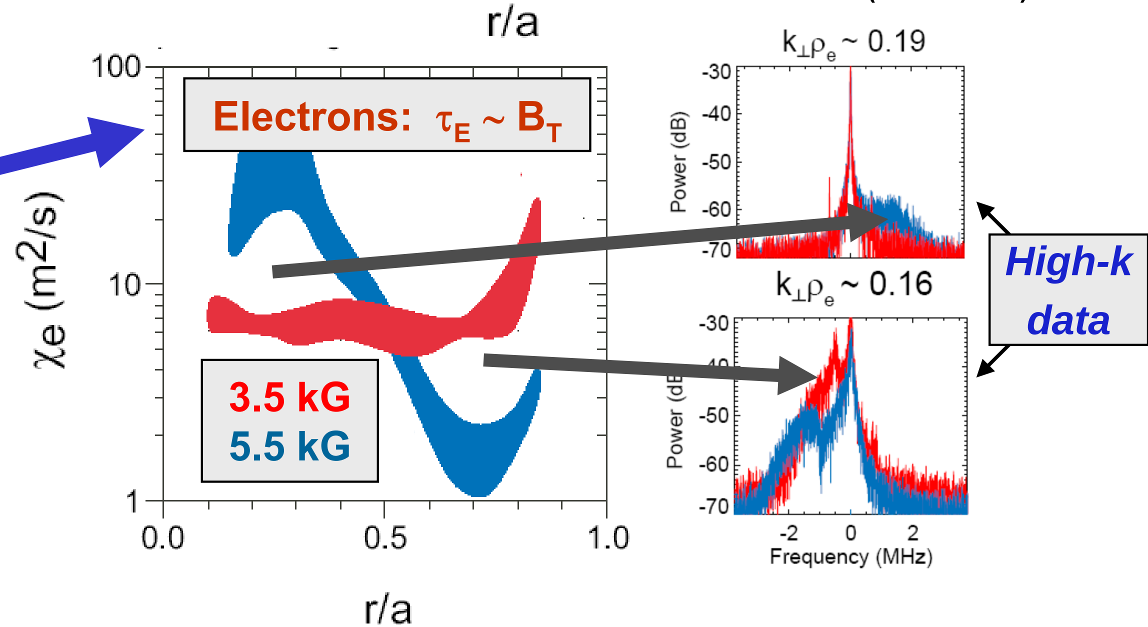


FY09-11 goals: Improve/characterize sustainment of high-β, understand RWM/NTV physics at lower ν*

NSTX is developing a deeper understanding of ion and electron energy transport for STs and tokamaks

- Ion $\tau_E \sim I_P$, consistent with neoclassical ion transport
  - Implies ion turb. suppressed by high E×B shear → possibility of isolating causes of e-transport
- Electron & ion τ_E scale differently, and different than at higher A:
  - Ion $\tau_E \sim I_P$, electron $\tau_E \sim B_T$
- High-k scattering data indicates χ_e correlated w/ high-k density fluctuations
  - Correlation holds both spatially and versus B_T
  - Consistent with ETG at large r/a (i.e. in T_e gradient region)





FY09-11 goals: **Measure low-k turbulence, understand modes responsible for anomalous e/i transport**

NSTX accesses broad range of fast ion parameters, and a broad range of fast particle modes

- Figure at right illustrates NSTX operational space, as well as projected operational regimes for: ITER (α's only), ST-CTF (α+NBI), ARIES-ST (α's)
- Also shown are parameters where typical fast particle modes (FPMs) have been studied.
- Conventional beam heated tokamaks typically operate with $V_{fast}/V_{Alfven} < 1$.
- CTF in avalanche regime motivates studies of fast ion redistribution
  - ITER with NBI also unstable to AE
- Higher ρ^* of NSTX compensated by higher beam beta

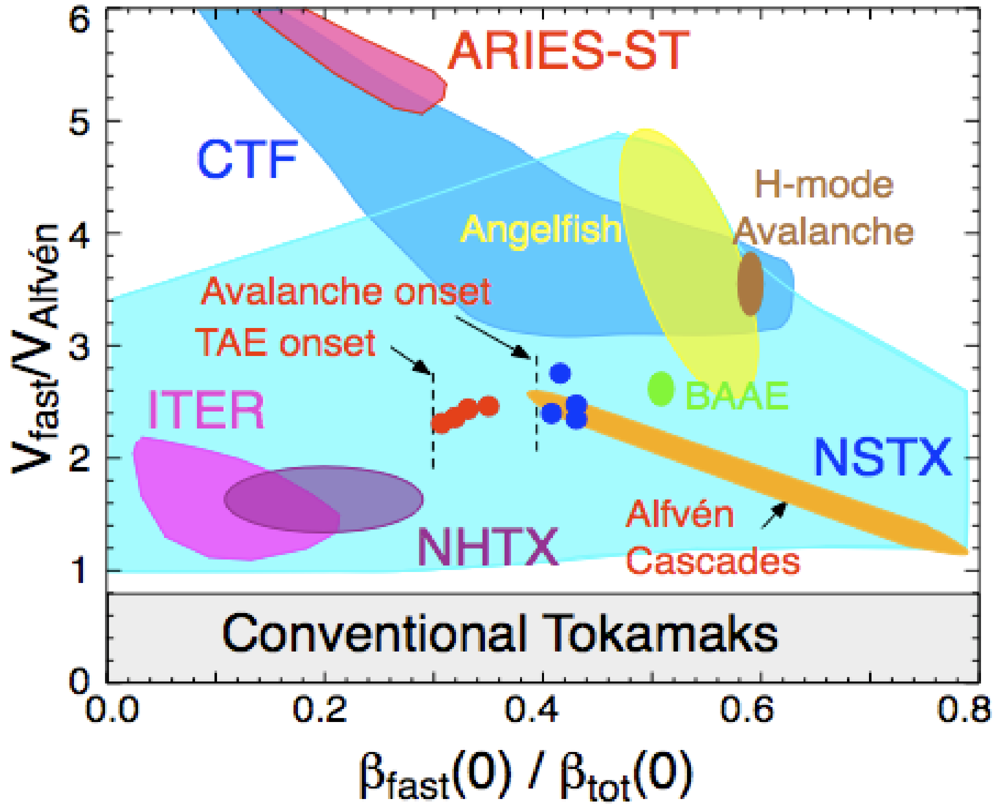
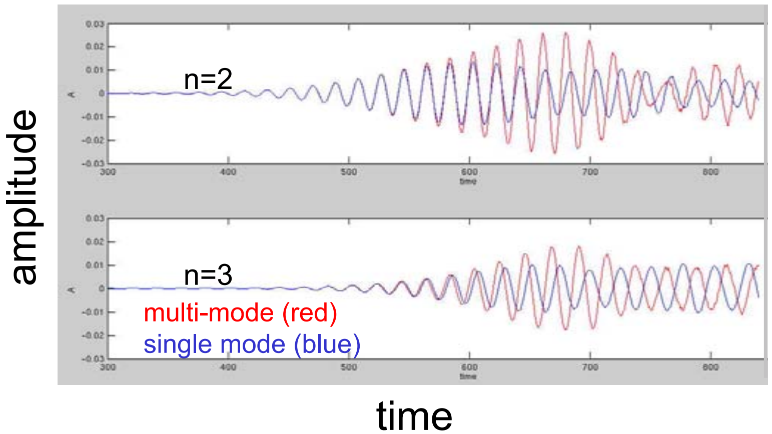
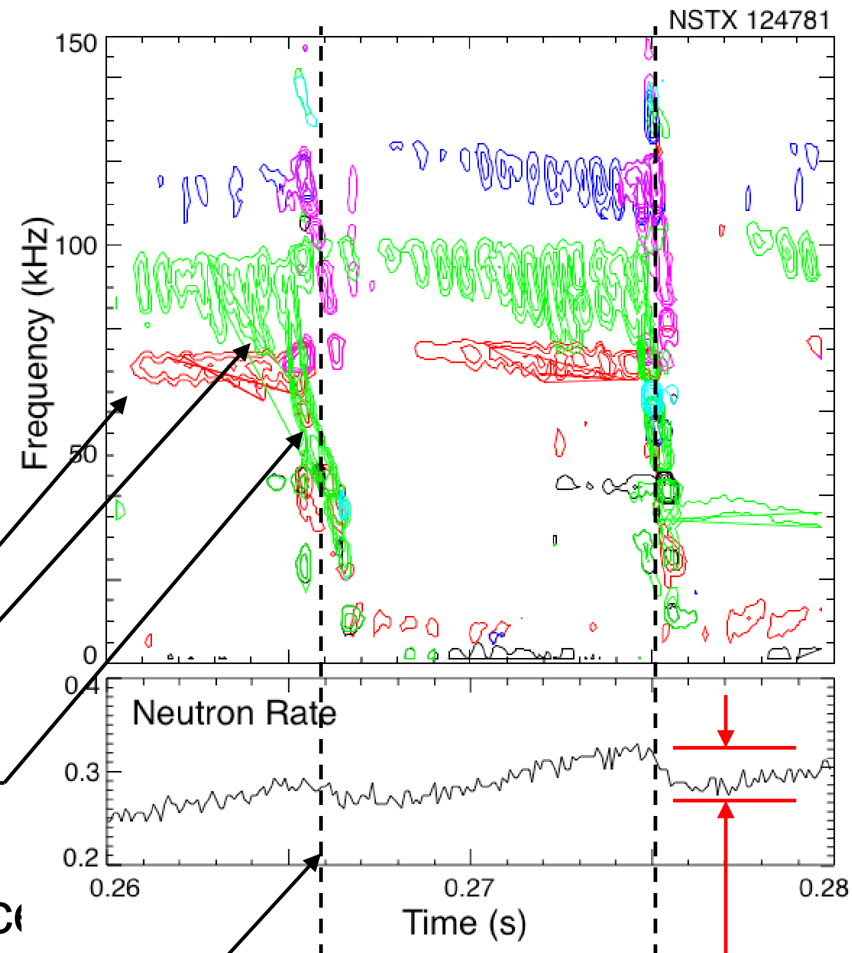


Figure above is simplified picture - there are other dependences, such as *q* profile, ρ^*

NSTX finds AE avalanches can induce fast-ion redistribution and/or loss - potentially important for ITER and ST-CTF

M3D-K simulations: overlapping resonances + multiple modes cause larger mode amplitude → larger fast-ion f(v) perturbation





Experiment:

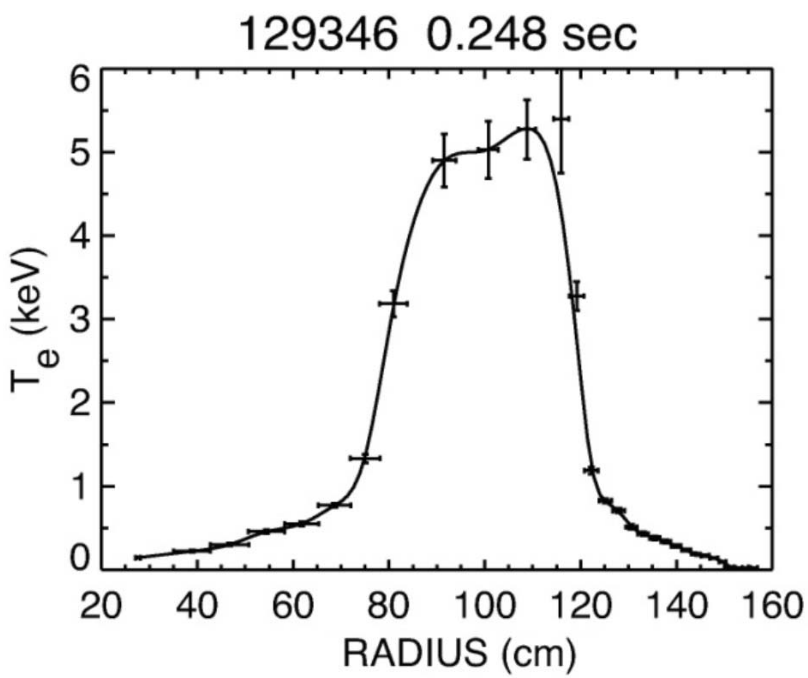
- As power is raised, first see AE
  - then chirping AE
  - then avalanches, multi-mode transport
- Avalanches are strong bursts of multiple AE modes (2 ≤ n ≤ 6) overlapping in space and frequency
- Avalanches correlate with neutron drops indicating fast ion redistribution and/or loss

15% drop in neutron rate

FY09-11 goals: Improve ξ(r) data and predictive capability for fast-ion transport, extend to H-mode

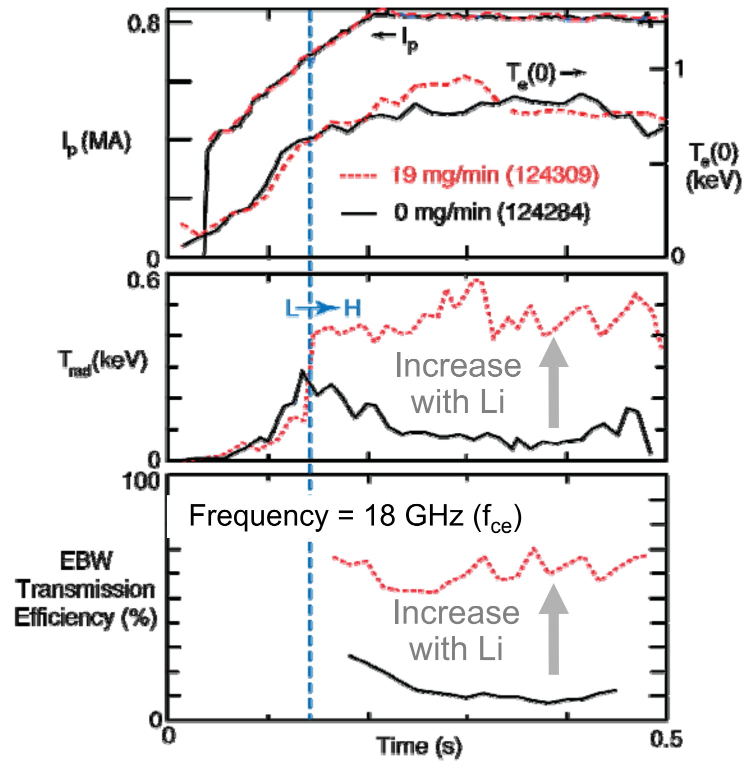
NSTX has improved the understanding and performance of wave heating & CD techniques in over-dense plasmas

- High-harmonic fast-wave (HHFW)
  - Discovered that surface waves reduce heating efficiency if density near antenna is too high
  - Control of edge density improves heating → record T_e = 5keV in NSTX achieved with HHFW



$n_e(0) = 1.5 \times 10^{19}\ m^{-3}$

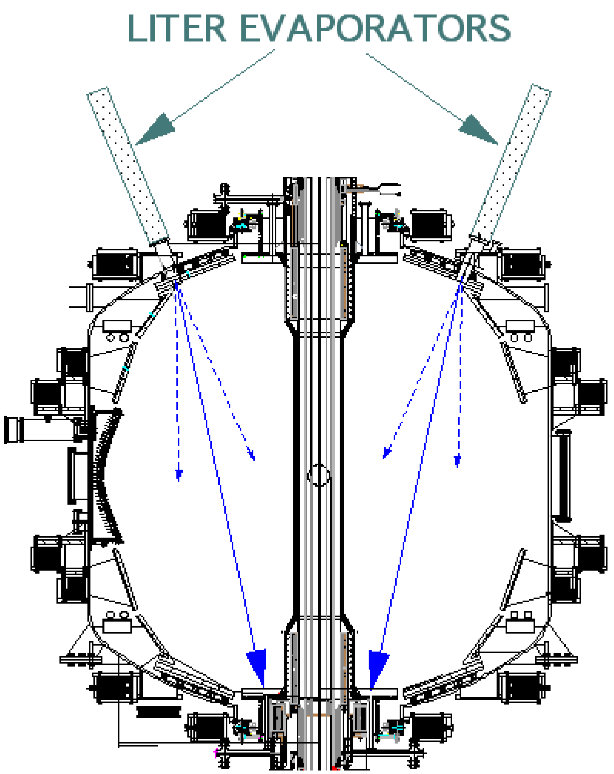
- Electron Bernstein Wave (EBW)
  - Discovered that collisional damping at mode conversion layer reduces coupling
  - Higher T_e at MC layer via Li-conditioning increases EBW transmission efficiency from 10% to 50-60% in H-mode→ Improved prospects for EBW as H&CD tool



FY09-11 goals: **Improve HHFW heating in D H-mode for ramp-up & sustainment, MAST EBW collaboration**

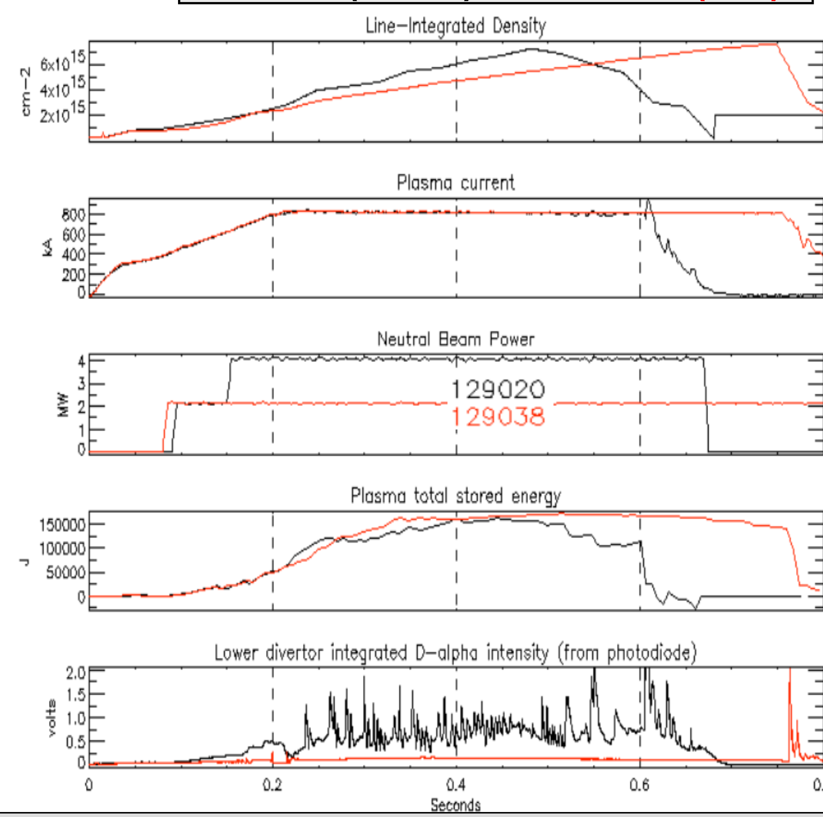
NSTX is unique in the world program in exploring lithium in a diverted H-mode plasma

- Dual Lithium evaporators (LITERs) provide complete toroidal coverage of lower divertor
  - Improved performance vs. 1 LITER
  - 2008: High-performance operation with **NO** between-shot He glow → increased shot-rate



- Reproducible ELM elimination from Li
  - Plasma density reduced
  - Pulse-length extended
  - At 800kA, power must be reduced to avoid β limit
  - Confinement time doubled (up to 80ms)
  - Large reduction in divertor D_{α} → reduced recycling

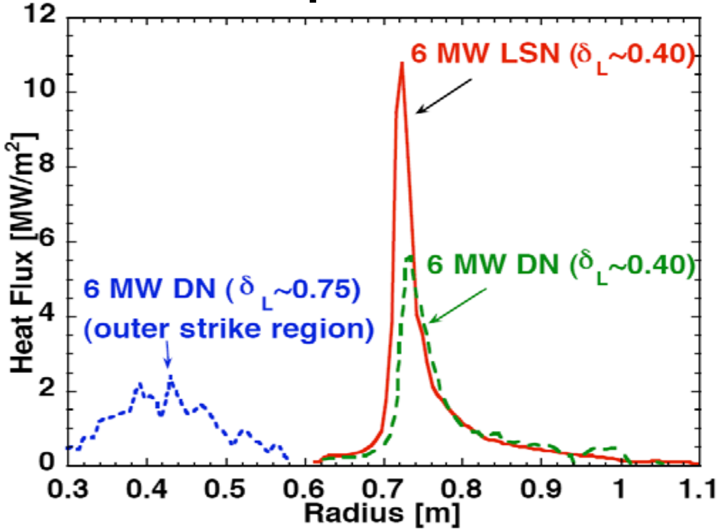
No Li (black) **With Li (red)**



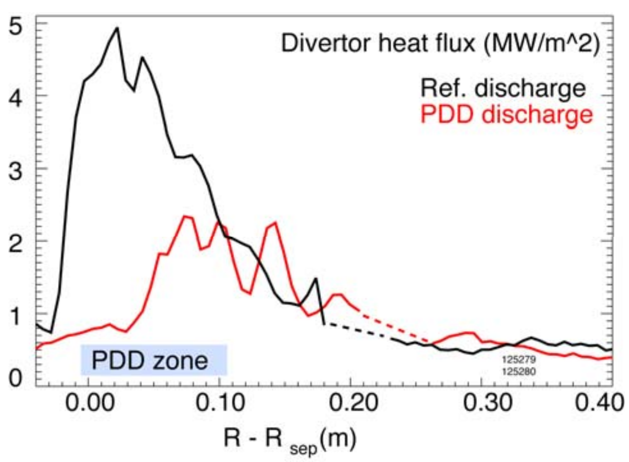
FY09-11 goals: Understand Li-plasma interaction, achieve density control with Liquid Lithium Divertor

NSTX accesses ITER-level divertor heat fluxes and is exploring mitigation of steady-state and transient heat fluxes

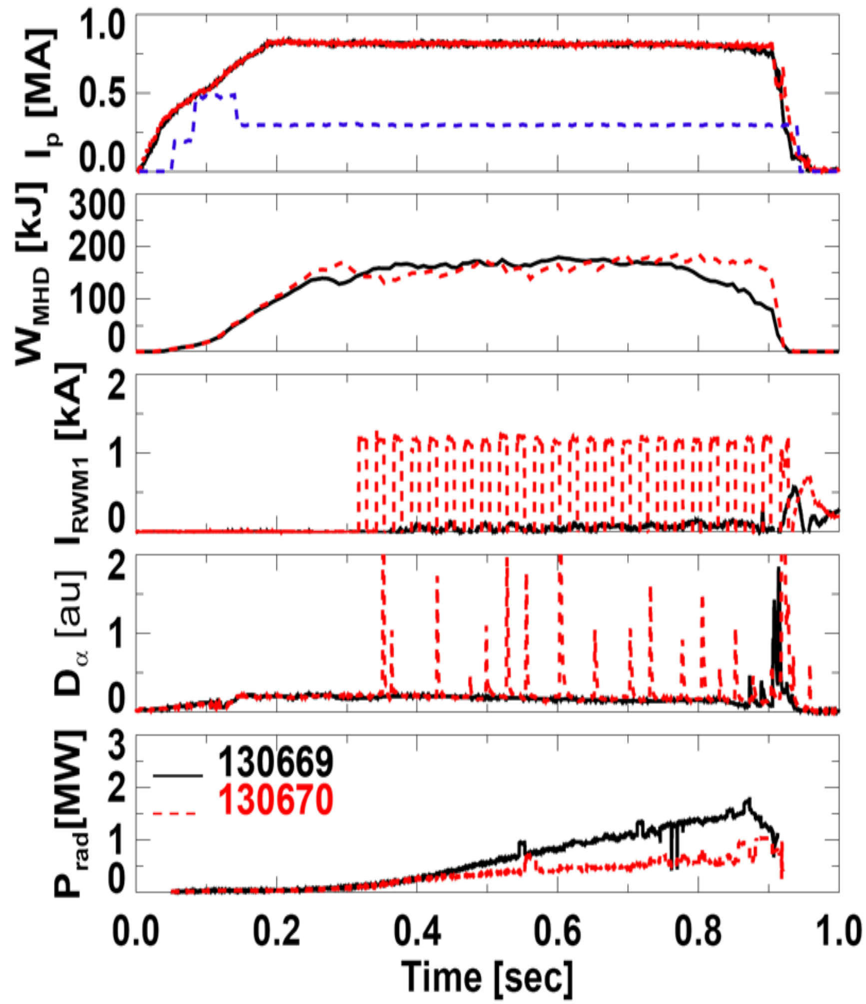
- Magnetic geometry strongly influences peak heat flux



- Partial detachment reduces peak heat flux



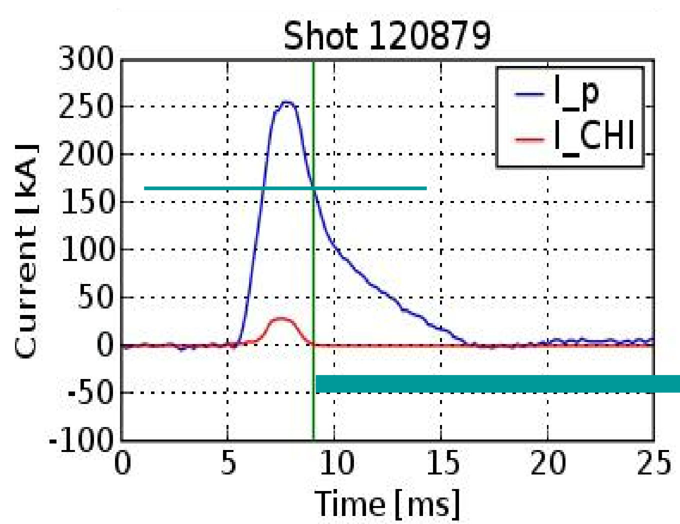
- Lithium conditioning can eliminate ELMs
- RMPs can controllablly trigger ELMs and expel impurities from Li-ELM-free plasmas

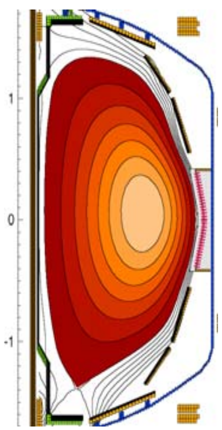


FY09-11 goals: Understand & develop steady-state heat-flux mitigation, ELM control for STs and ITER

NSTX is testing unique methods of non-solenoidal plasma current start-up and ramp-up for STs

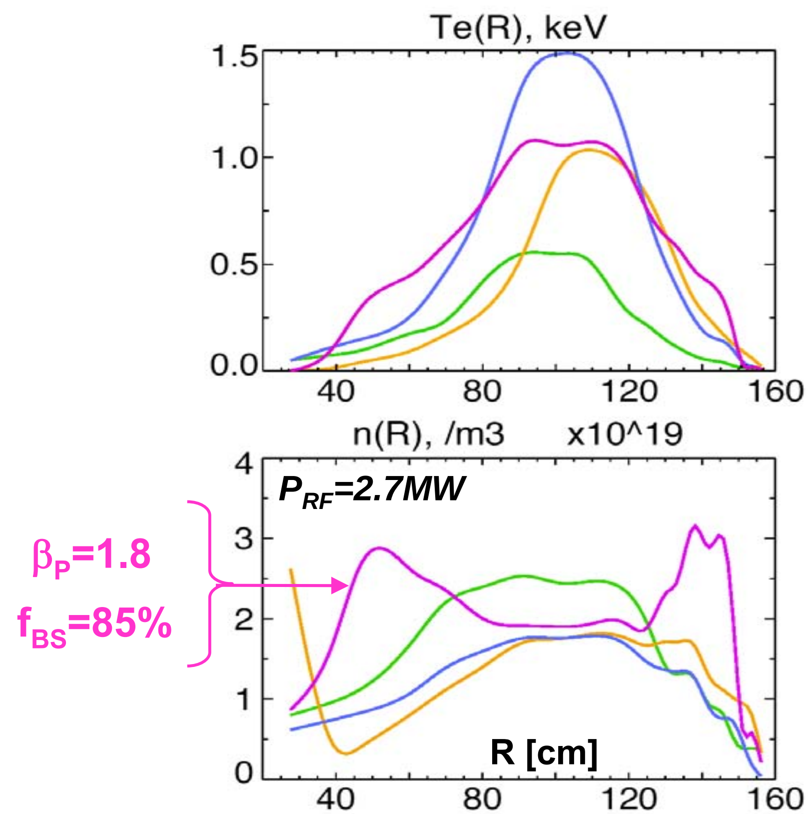
- Start-up: Coaxial Helicity Injection
  - Generated record closed-flux I_P=160kA
  - Demonstrated coupling to induction and compatibility with high performance H-mode
  - Higher I_P limited by lack of auxiliary heating, possibly impurities/divertor conditions





t=9ms

- Ramp-up: High Harmonic Fast Wave
  - HHFW heats 250kA plasma to T_e=1keV
  - Produces f_{BS}=85% H-mode plasma
  - Limited by antenna voltage stand-off, ELMs

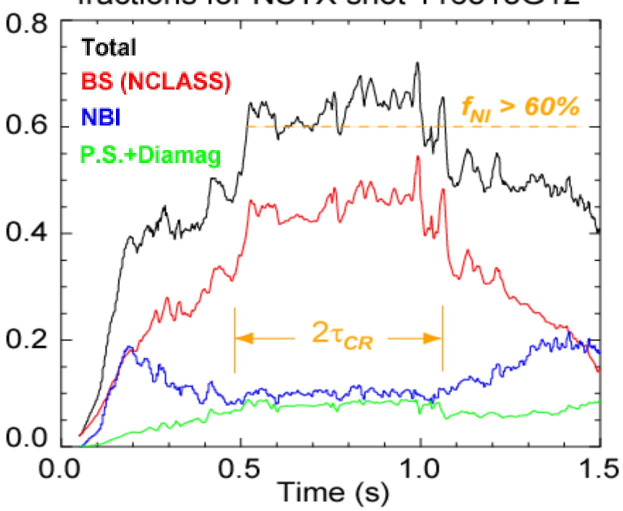


FY09-11 goals: Improve CHI start-up (LLD target plates, Li, absorber coils), high-P_{HHFW} for ramp-up

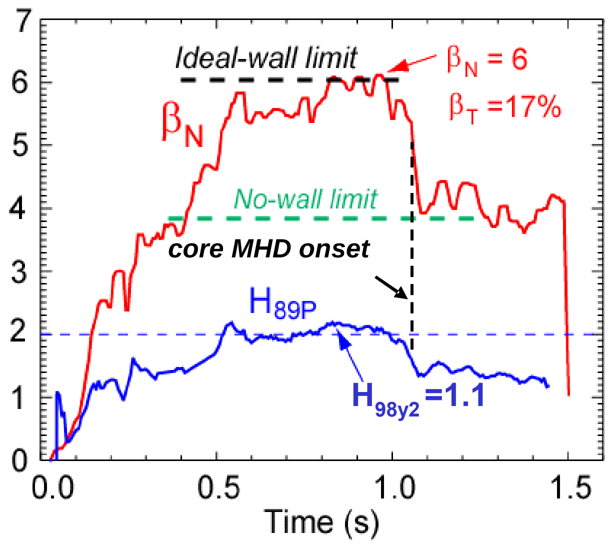
NSTX has developed and sustained scenarios with high non-inductive fraction and high normalized β

- f_{NICD} = 65%
- $f_{\nabla p}$ = 55%

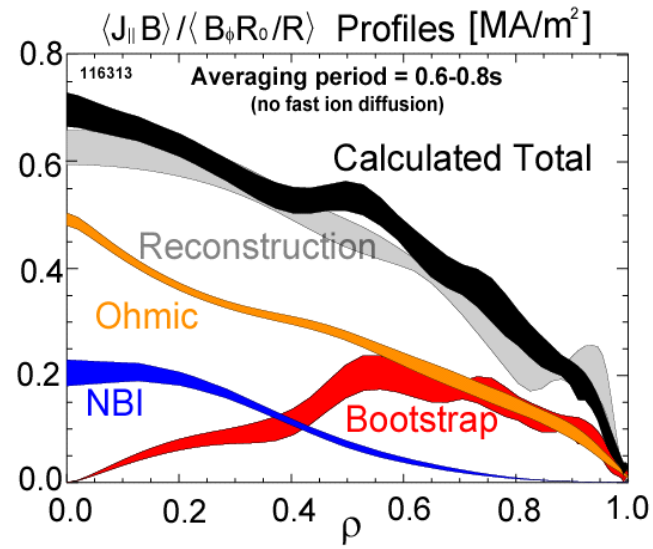
TRANSP non-inductive current fractions for NSTX shot 116313G12



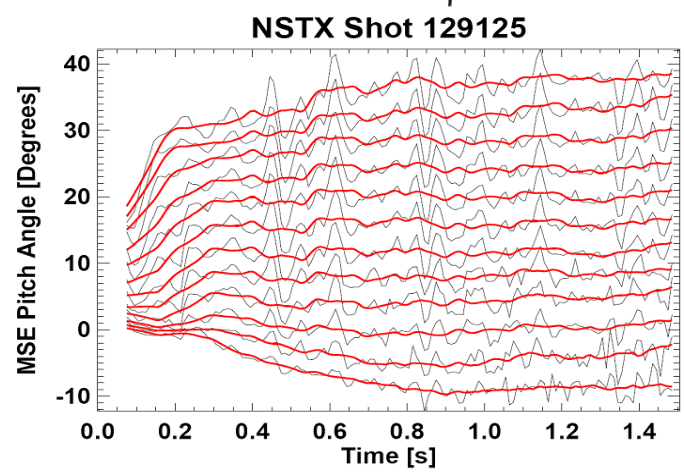
- β_N = 5.5-6
- H_{98} = 1-1.1
- $f_{Greenwald} \rightarrow 1$



Predicted and reconstructed J profiles are in agreement when MHD activity is weak



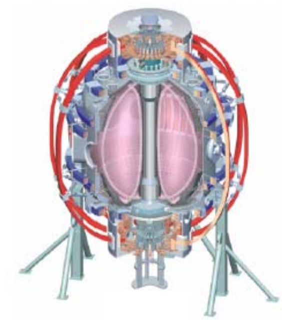
- Recent long-pulse discharges which avoid core rotating MHD activity exhibit J-profile equilibration
  - Spikes in MSE pitch angle are low-f MHD (early) and large ELMs (late)



FY09-11 goals: Density, β, RFA/RWM, ELM, impurity control for sustained & higher non-inductive fraction

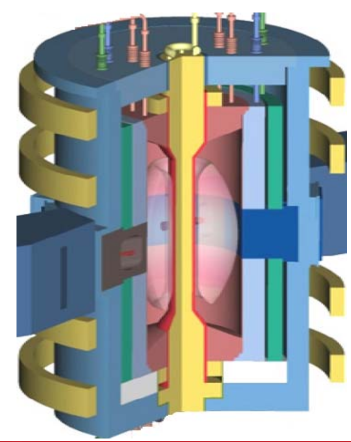
Near-term NSTX research, and longer-term NSTX Major Upgrades will prepare the U.S. to address FESAC Priorities, Gaps, and Opportunities

- NSTX:
  - Providing foundation for understanding ST physics and performance
- Upgraded NSTX:
  - Study high beta plasmas at reduced collisionality – important for further understanding confinement, stability, start-up, current drive
  - Assess full non-inductive current drive operation – needed for steady-state ST applications and ITER advanced operating scenarios
  - Prototype heat and particle exhaust solutions for next-step facilities



- Tame the plasma-material interface
  - Exploit intrinsic high heat flux of ST to understand boundary physics at fusion-relevant edge plasma conditions and heat/particle fluxes

- Advance fusion engineering science
  - Exploit high β, compactness of ST to achieve high neutron flux and fluence at reduced size and cost, reduced T consumption

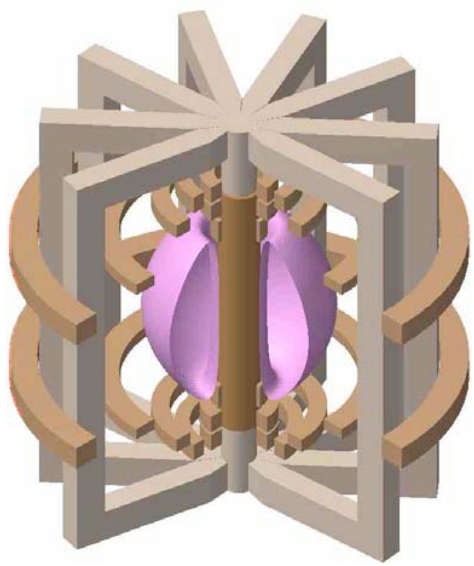


ST is attractive configuration for "Taming the plasma-material interface"

- FESAC-PP identified PMI issue as highest priority: "…solutions needed for DEMO not in hand, …require major extrapolation and substantial development"

Scientific mission of National High-power advanced Torus eXperiment (NHTX):
"Integration of a fusion-relevant plasma-material interface with stable sustained high-performance plasma operation"

- **PMI research and integration goals:**
  - Create/study DEMO-relevant heat-fluxes
  - Perform rapid testing of new PMI concepts
    - Liquid metals, X-divertor, Super-X divertor
  - PMI research at DEMO-relevant $T_{wall} \sim 600°C$
  - Plasma-wall equilibration: τ_{pulse} = 200-1000s
  - Develop methods to avoid T retention
  - Demonstrate compatibility of PMI solutions with high plasma performance:
    - High confinement without ELMs
    - High beta without disruptions
    - Steady-state, fully non-inductive
  - Study high β_N, f_{BS} for ST-DEMO and ST-CTF
  - Test start-up/ramp-up for ST-CTF and ST-DEMO



National High-power advanced Torus eXperiment (**NHTX**)

Baseline operating scenario:

P_{heat}	50MW
R_0	1m
A	1.8-2
κ	$\leq$ 3
B_T	2T
I_P	3-3.5MA
β_N	4.5
β_T	14%
n_e/n_{GW}	0.4-0.5
f_{BS}	$\approx$ 70%
f_{NICD}	100%
$H_{98Y,2}$	$\leq$ 1.3
E_{NB}	110keV
P/R	50MW/m
Solenoid	½ swing to full I_P

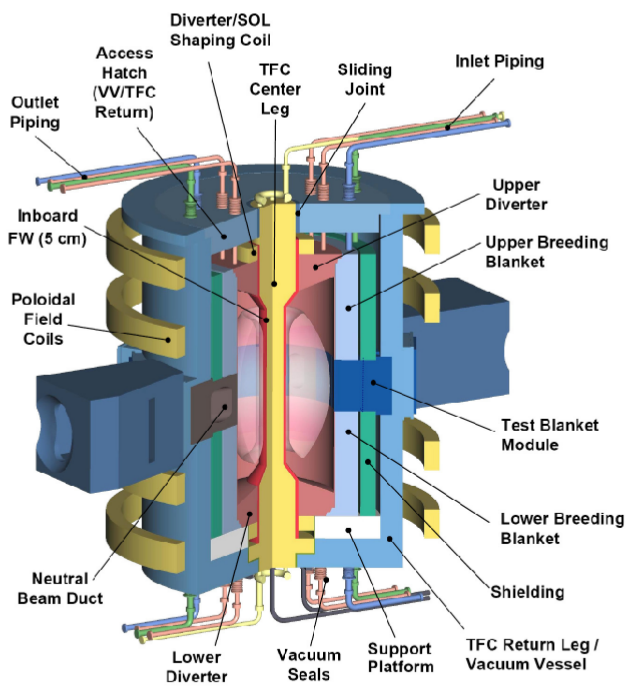
ST-based Component Test Facility (ST-CTF) is attractive concept for "Harnessing Fusion Power"

- **ST-CTF Required Conditions:**

From M. Peng APS-2007, based on NCT presentation to FESAC 8/7/2007

Performance metrics	ITER	Required Conditions	Demo Goals
Continuous operation	~hour	weeks	~months
14-MeV neutron flux on module (MW/m^2)	~0.8	1.0-2.0	~3
Total neutron fluence goal ($MW\text{-}yr/m^2$)	~0.3	6	~6-15
Duty factor goal	~1%	30%	~80%
Tritium self-sufficiency goal (%)	~0	~100	≥100

- **ST advantages for CTF:**
  - Compact device, high β
    - Reduced device cost
    - Reduced operating cost ($P_{electric}$)
    - Reduced T consumption
  - Simplified vessel and magnets
    - Fully modularized core components
    - Fully remote assembly/disassembly



ST-based **C**omponent **T**est **F**acility (**ST-CTF**)

W_L [MW/m^2]	**0.1**	**1.0**	**2.0**
R0 [m]		**1.20**	
A		**1.50**	
kappa		**3.07**	
qcyl	4.6	**3.7**	3.0
Bt [T]	1.13	**2.18**	
Ip [MA]	3.4	**8.2**	10.1
Beta_N	**3.8**		5.9
Beta_T	0.14	**0.18**	0.28
n_e [$10^{20}/m^3$]	0.43	**1.05**	1.28
f_{BS}	0.58	**0.49**	0.50
T_{avgi} [keV]	5.4	**10.3**	13.3
T_{avge} [keV]	3.1	**6.8**	8.1
HH98		**1.5**	
Q	0.50	**2.5**	3.5
$P_{aux\text{-}CD}$ [MW]	15	**31**	43
E_{NB} [keV]	100	**239**	294
P_{Fusion} [MW]	7.5	**75**	150
T M height [m]		**1.64**	
T M area [m^2]		**14**	
Blanket A [m^2]		**66**	
$F_{n\text{-}capture}$		**0.76**	
P/R [MW/m]	**14**	**38**	**61**
Solenoid		**Iron core or MIC solenoid for startup**	

FESAC Toroidal Alternates Panel (TAP) recently prioritized issues and gaps for the Spherical Torus (ST) for the ITER era

ST ITER-era goal: ***"Establish the ST knowledge base to be ready to construct a low aspect-ratio fusion component testing facility …. to inform the design of a demonstration fusion power plant"***

"Tier 1" issues and key questions from TAP, and NSTX goals:

1. **Startup and Ramp-Up:** Is it possible to start-up and ramp-up the plasma current to multi-MA levels using non-inductive current drive w/ minimal or no central solenoid?
   - NSTX goal: demonstrate non-inductive ramp-up and sustainment

2. **First-Wall Heat Flux:** What strategies can be employed for handling normal and off normal heat flux consistent with core and scrape-off-layer operating conditions?
   - NSTX goal: assess high flux expansion, detached divertors, liquid metals

3. **Electron Transport:** What governs electron transport at low-A & low collisionality?
   - NSTX goal: determine modes responsible for electron turbulent transport and assess the importance of electromagnetic (high β) and collisional effects

4. **Magnets:** Can we develop reliable center-post magnets and current feeds to operate reliably under substantial fluence of fusion neutrons?
   - NSTX goal: develop and utilize higher performance toroidal field magnet

Performance gaps between present and next-step STs

For NHTX, ST-CTF scenarios: reduce n_e, increase NBI-CD, confinement, start-up/ramp-up

For ST-DEMO scenarios: increase elongation, β_N, f_{BS}, confinement, start-up/ramp-up

Present high β_N & f_{NICD}	NSTX	NSTX Upgrade	NHTX	ST-CTF	ST-DEMO
A	1.53	1.65	1.8	1.5	1.6
κ	2.6-2.7	2.6-2.8	2.8	3.1	3.7
β_T [%]	14	10-16	12-16	18-28	50
β_N [%-mT/MA]	5.7	5.1-6.2	4.5-5	4-6	7.5
f_{NICD}	0.65	1.0	1.0	1.0	1.0
$f_{BS+PS+Diam}$	0.54	0.6-0.8	0.65-0.75	0.45-0.5	0.99
f_{NBI-CD}	0.11	0.2-0.4	0.25-0.35	0.5-0.55	0.01
$f_{Greenwald}$	0.8-1.0	0.6-0.8	0.4-0.5	0.25-0.3	0.8
H_{98y2}	1.1	1.15-1.25	1.3	1.5	1.3
ν^*_e	0.15	0.04	0.01	0.002	0.007
Dimensional/Device Parameters:					
Solenoid Capability	Ramp+flat-top	Ramp+flat-top	Ramp to full I_P	No/partial	No
I_P [MA]	0.72	1.0	3-3.5	8-10	28
B_T [T]	0.52	0.75-1.0	2.0	2.5	2.1
R_0 [m]	0.86	0.92	1.0	1.2	3.2
a [m]	0.56	0.56	0.55	0.8	2.0
I_P / aB_{T0} [MA/mT]	2.5	1.8-2.4	2.7-3.2	4-5	6.7

Near-term highest priority is to assess proposed ST-CTF operating scenarios

Gaps between present and future STs motivate NSTX scientific priorities and associated upgrades

1. Increase and understand beam-driven current at lower n_e, ν^*
   - Next-step STs ***require*** full NICD to achieve missions, and NBI-CD is largest gap
   - But, lower n_e, ν^* also impacts AE avalanches, transport, MHD, pedestal, ELMs

   → *Test increased NBI-CD with density reduction, higher T_e, higher NBI power*

2. Increase and understand H-mode confinement at low ν^*
   - ST energy confinement – in particular electron energy confinement - not sufficiently well understood to make extrapolation to next-steps with high confidence

   → *Determine modes responsible for transport, determine scaling vs. B_T, I_P, P_{HEAT}*

3. Demonstrate and understand non-inductive start-up and ramp-up
   - Non-inductive ramp-up essential to ST-CTF and ST-DEMO
   - Increased non-inductive start-up current must also be demonstrated

   → *Increase ramp-up heating power & current drive to test I_P ramp-up technique*

4. Demonstrate and understand means to "tame the plasma-material interface"
   - Short-pulse pumping needed near-term, longer-pulse pumping + heat-flux mitigation in upgrade
   - PMI solution for very high particle/heat/neutron flux needed for ST-CTF and ST-DEMO

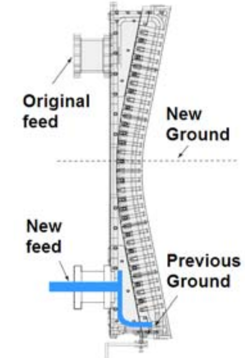
   → *Develop means for particle control in H-mode (such as LLD), extend to long pulse, higher P/R*

5. Sustain β_N and understand MHD near and above no-wall limit
   - Operation at no-wall limit assumed as baseline for NHTX and ST-CTF designs
   - Increased β_N, κ increases f_{BS}, β_T - would enhance ST-CTF, needed for ST-DEMO

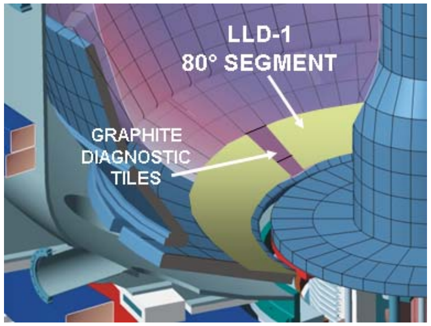
   → *Improve control of β, RWM/EF, rotation and q profiles to optimize stability*

Near-term (FY2009-11) upgrades support highest priorities and enable key research thrusts:

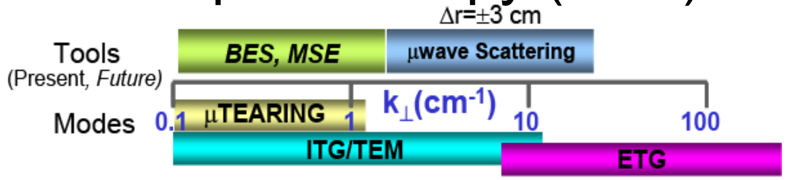
- FY2009-11: Upgraded fast wave heating for ramp-up, sustainment
  - Antenna modified to double RF power, ELM resilience for heating in H-mode
  - Ramp-up is critical issue for future ST devices
  - Utilize strong electron heating for self-generated “bootstrap” current ramp-up
  - Wave coupling/heating physics in advanced ST H-mode scenarios, ITER



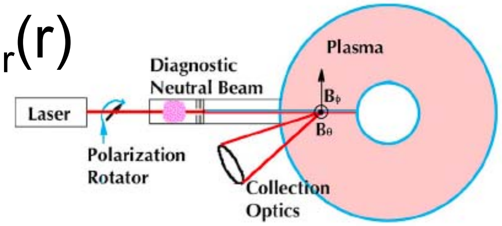
- FY10 - Access new physics regimes by utilizing a novel particle pumping technique: Liquid Lithium Divertor (LLD)
  - Will study impact of reduced collisionality and liquid lithium on edge energy and particle transport, and edge stability
  - Solid Li coatings previously led to transient pumping, improved energy confinement, ELM reduction/elimination



- FY10 – First low-k turbulence data w/ Beam Emission Spectroscopy (BES)
  - Expand turbulence measurements to cover full k-range
  - **Long term goal:** determine and understand modes responsible for anomalous transport



- FY11 – MSE-LIF for pitch angle & |B| w/o heating beam, $E_r(r)$
  - Greatly expanded flexibility for all topical science areas
  - Especially beneficial for HHFW, energetic particle research
    - q(r) during RF-only heating, reconstruct NBI fast-ion p from total – thermal



NSTX FY2009-11 Research Milestones
(base and incremental)

	FY2009	FY2010	FY2011
Expt. Run Weeks:	14 (20)	15 (20)	15 (20)
1) **Transport & Turbulence**			Study turbulence regimes responsible for ion and electron energy transport (formerly FY2010)
2) **Macroscopic Stability**	Understand physics of RWM stabilization & control vs. rotation	Assess sustainable beta and disruptivity near and above the ideal no-wall limit.	Assess sustained operation above the no-wall limit at reduced collisionality
3) **Boundary Physics**		Assess H-mode characteristics as a function of collisionality and lithium conditioning	Relationship between lithiated surface conditions and edge and core plasma conditions
4) **Wave-Particle Interaction**	Study how j(r) is modified by super-Alfvénic ion-driven modes	Characterize HHFW heating, CD, and ramp-up in deuterium H-mode *Joint milestone w/ solenoid-free TSG*	Assess predictive capability of mode-induced fast-ion transport
5) **Solenoid-free start-up, ramp-up**			
6) **Advanced Scenarios & Control**	Perform high-elongation wall-stabilized operation at lower n_e; Integrate MHD mode modification of j(r) into optimized operation	←	Dependence of integrated plasma performance on collisionality *(FY2010 incremental accelerates this by 1yr if LLD and/or HHFW achieve FY2010 goals)*

Joint Research Targets (3 US facilities):

	FY2009	FY2010	FY2011
	Particle control and hydrogenic fuel retention	Understanding of divertor heat flux, transport in scrape-off layer	TBD (*…Characterize H-mode pedestal structure…)*

Run time/schedule priority will be given to milestones

List below is prioritized based on relative importance of gaps

- Key capabilities & upgrades utilized for milestones are shown in (red)
- Note that all milestones below are "high priority", since milestones are allocated as much run-time as is needed (within reason) to achieve their goal

FY2009 Milestones

Joint	Particle control and hydrogenic fuel retention in tokamaks	(Li, sample probe)
1.	Perform high-elongation wall-stabilized operation at reduced n_e	(Li, NBI control)
2.	Study how j(r) is modified by super-Alfvénic ion-driven modes	(Fast-ion D-alpha)
3.	Understand physics of RWM stabilization and control vs. rotation	(NBI control, NTV braking)

FY2010 Milestones

Joint	Understanding of divertor heat flux, transport in scrape-off layer	(Div. bolom & fast IR, LLD)
1.	Characterize HHFW heating, CD, and I_P ramp-up in H-mode plasmas	(Upgraded HHFW)
2.	Assess pedestal characteristics and ELM stability as a function of ν^* & Li	(LLD, Li CHERs, sample probe)
3.	Assess sustainable β and disruptivity near and above ideal no-wall limit	(Improved β & mode control)

FY2011 Milestones

Joint	(TBD) Improve understanding of H-mode pedestal structure	(Higher-res MPTS)
1.	Dependence of integrated plasma performance on collisionality	(LLD, HHFW, MSE-LIF)
2.	Study turbulence regimes responsible for ion, electron energy transport	(BES, high k_θ, LLD)
3.	Relationship between lithiated surface & edge/core plasma conditions	(LLD, MAPP, div. spect.)

A "Lithium Research Thrust" has recently been formed to coordinate Li research on NSTX

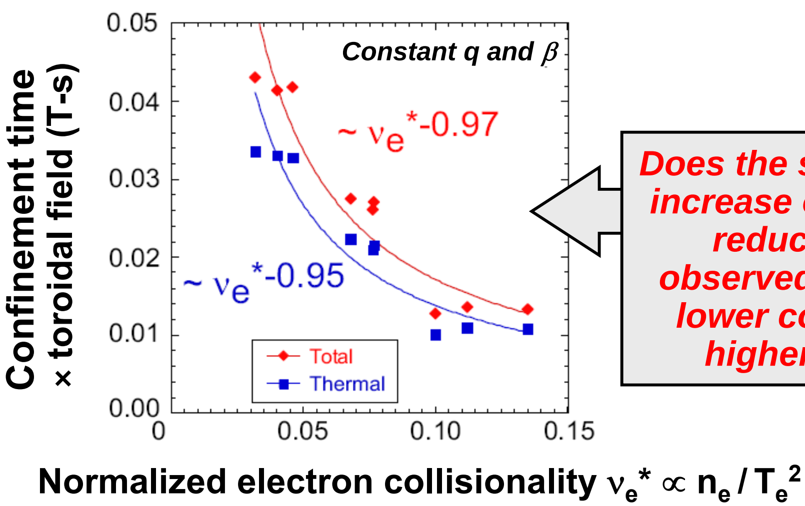
- Motivation:
  - Need to better understand underlying physics of how Li impacts plasma
    - And how plasma and non-Li PFCs impact Li PFCs
  - Impact of Li is cross-cutting
    - Cuts across all NSTX topical science areas, and several programs at PPPL
- Goals:
  - Develop integrated NSTX Li research plan – 3 yr time horizon (FY09-11)
    - Increase emphasis on Li diagnostics, theory, simulation support
  - Coordinate Li research plans between NSTX, LTX, theory
- Leadership:
  - C. Skinner (leader, NSTX rep – coordinates NSTX expt Li program)
  - R. Kaita (deputy – LTX rep, NSTX Li diagnostics)
  - D. Stotler (Li theory and modeling coordinator for NSTX and LTX)
  - Meet with all TSG leaders as needed to capture cross-cutting issues
- Time-line:
  - Initiate in parallel with TSGs this year, coordinate FY10 LLD research
  - Run-time allocation under discussion (estimate: 4 run days / year)

Decreased density and collisionality (from LLD and/or higher T_e) will likely impact physics and plasma performance across all topical science areas

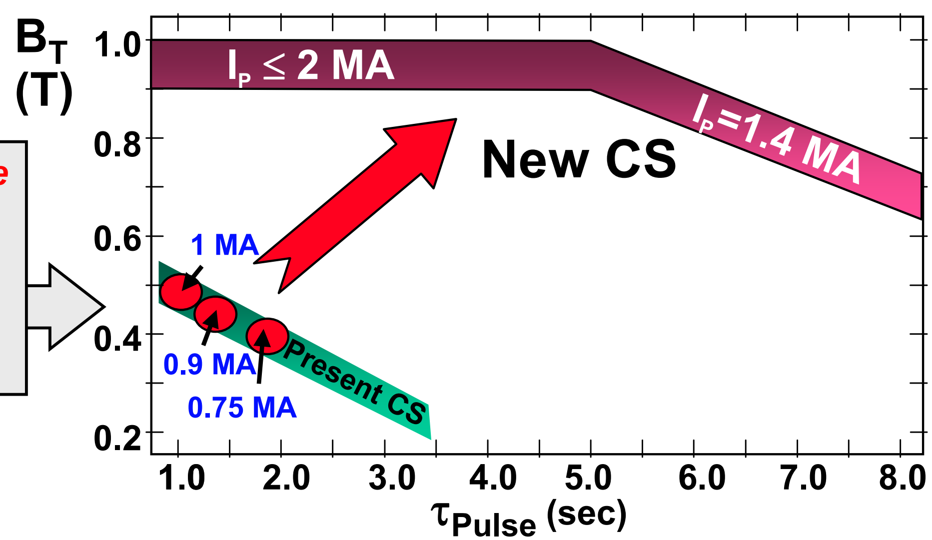
- Macroscopic Stability
  - RWM critical rotation and neoclassical viscous torques may increase at lower ν_i
- Transport & Turbulence
  - Underlying instabilities (micro-tearing, CTEM, and ETG) scale differently versus ν^*
  - If $T_e(r)$ is set by a critical ∇T_e, H-mode confinement may be reduced at reduced n_e
- Boundary Physics
  - ELM ΔW increases at lower ν_e * - could impact confinement, plasma purity, divertor
  - ELM stability may improve at lower ν_e^* - possible second-stability access
  - Detachment for heat flux reduction will be more challenging at reduced SOL density
- Wave-Particle Interactions
  - AE avalanches may be more easily triggered at reduced n_e due to increased fast-ion pressure fraction resulting in possible fast-ion redistribution and/or loss
- Plasma Start-up, Ramp-up, Sustainment
  - NBI-CD and RF-CD efficiency for ramp-up are increased at reduced n_e, increased T_e
  - ST-CTF scenarios rely on reduced n_e and increased T_e to increase NBI current drive efficiency to achieve 100% non-inductive current fraction.

Increased temperature and duration needed to address key issues for toroidal plasma science, ITER, and next-step STs

- Higher field and current enable access to higher temperature
- Higher temperature reduces collisionality and increases efficiency of non-inductive current-drive sources, and increases equilibration time
- New CS with B_T = 1T, I_P = 2MA (with induction), $t_{flat\text{-}top}$ = 5s would provide:
  - Longer pulse to assess RF ramp-up, 100% non-inductive sustainment at ~1MA
  - Higher field to stably accept high power for edge heat/particle transport studies
  - Extended range of field, current, β, collisionality to obtain unique data to aid development of first-principles understanding of turbulent transport
  - Magnet operation at ~1T (vs. 0.55T), within factor of 2 of next-step STs

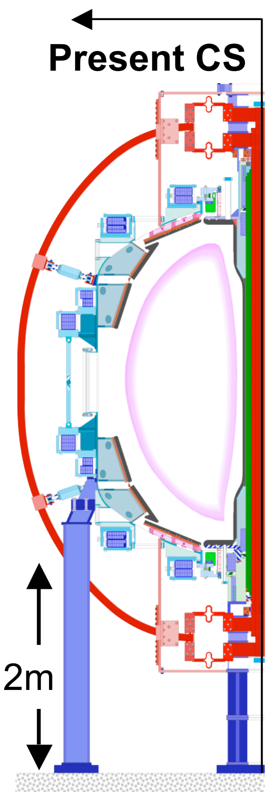


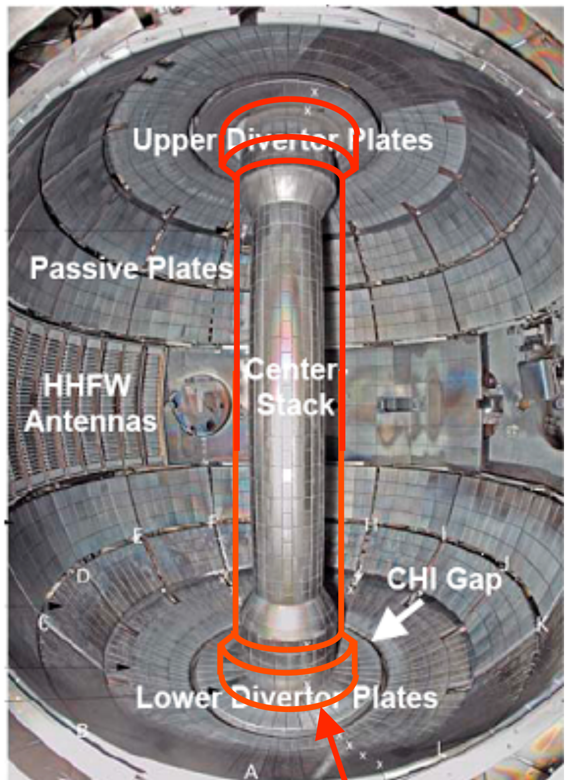
Does the strong and favorable increase of confinement with reduced collisionality observed in NSTX persist at lower collisionality and/or higher magnetic field?

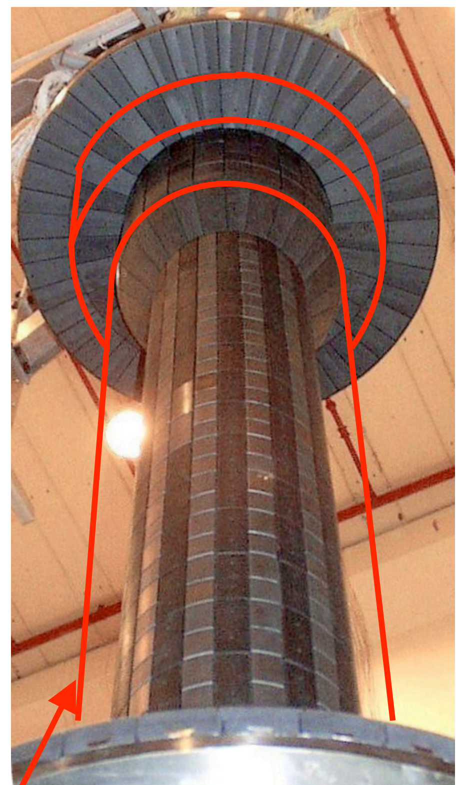


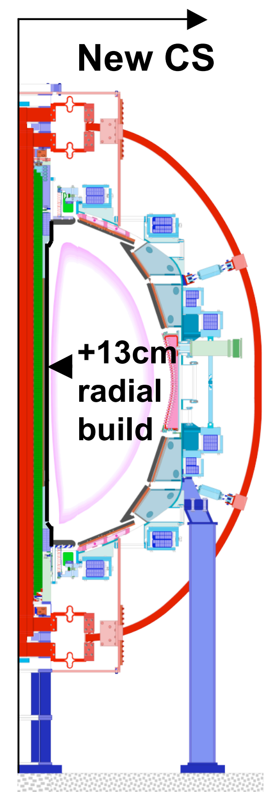
Modular design of NSTX enables removal of present CS and replacement with a new higher-performance CS

- Present CS has been removed and re-installed several times for maintenance and modifications
- New CS would have larger radius for increased conductor area and toroidal field current, while maintaining low aspect ratio A ~ 1.5
- Construction tolerance requirements are similar to present NSTX CS





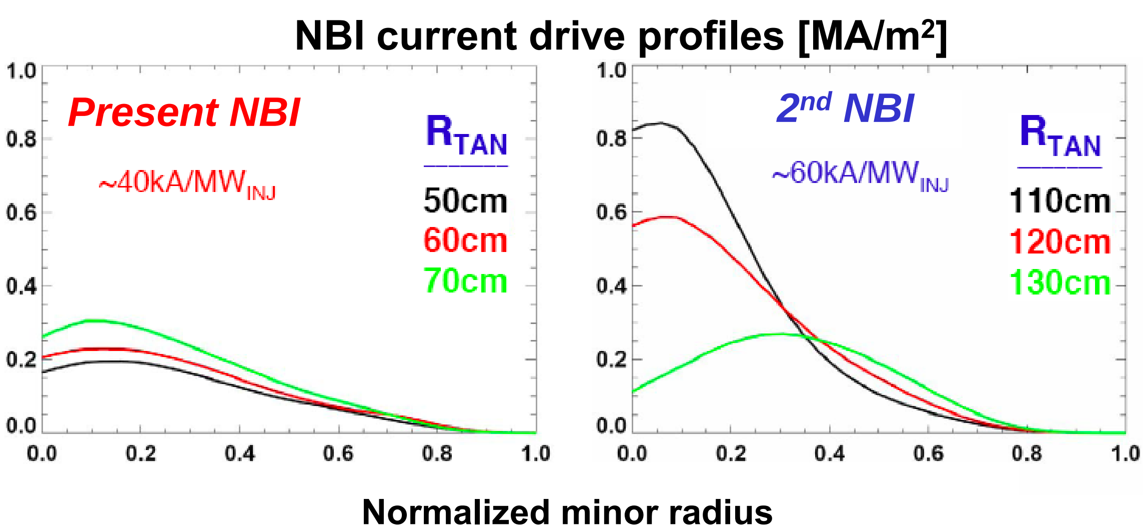


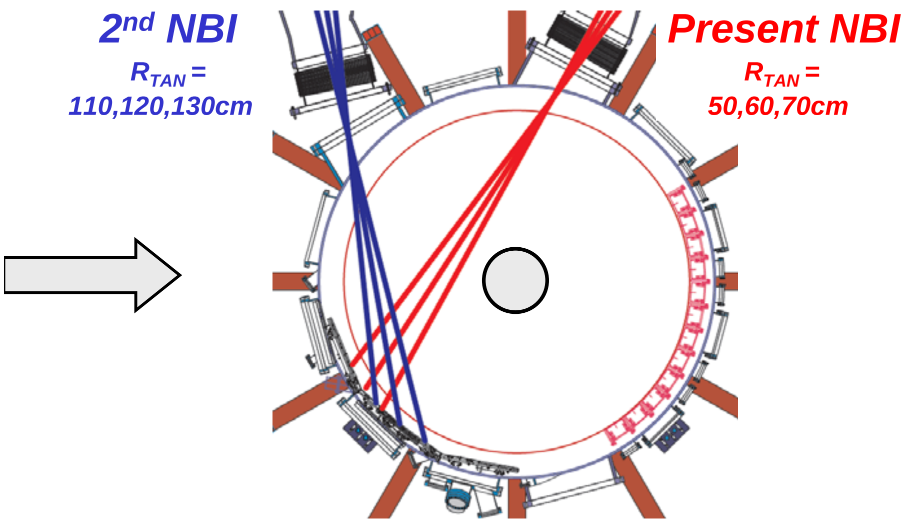


Approximate outline of new Center-Stack

More tangential 2nd NBI would enhance heating & current-drive for start-up, sustainment, heat-flux, transport studies

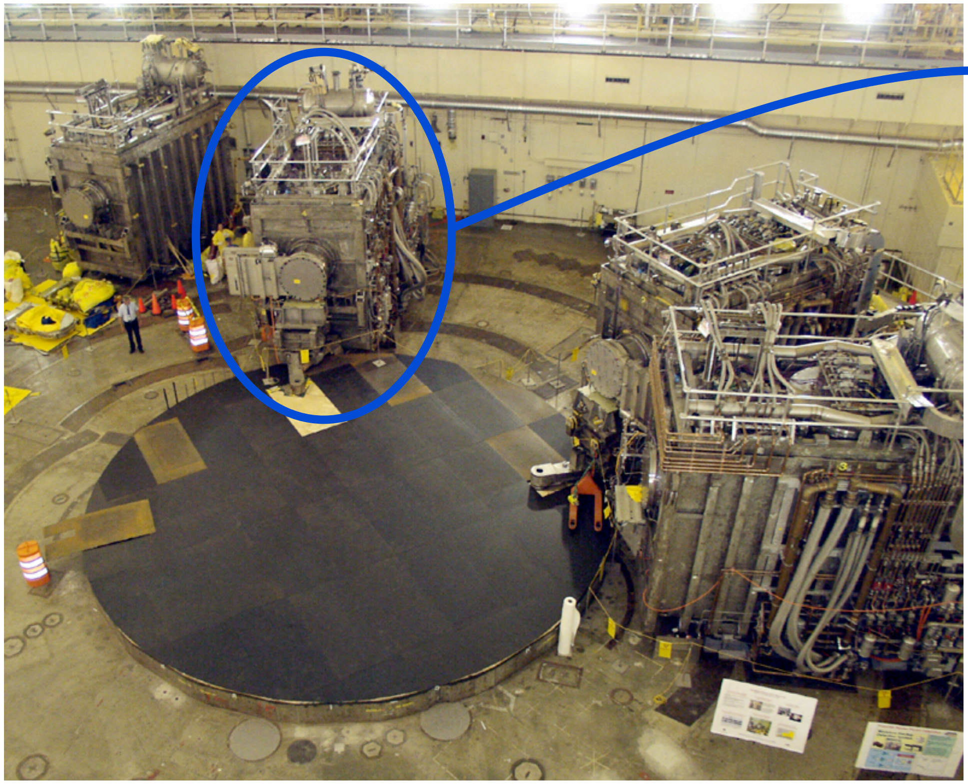
- More tangential 2nd NBI would provide:
  - Up to 2 times higher current-drive efficiency, and current profile control
  - Tests of NBI ramp-up to ~1MA
  - World-leading capabilities for plasma boundary physics at high heat flux
  - Increased heating power to access very high β at low collisionality – important for fundamental studies of transport and global stability
  - Overall, a highly flexible tool for toroidal physics research by varying current, heating, and torque profiles, and fast-ion distribution function $f(v_{||},v_{\perp})$



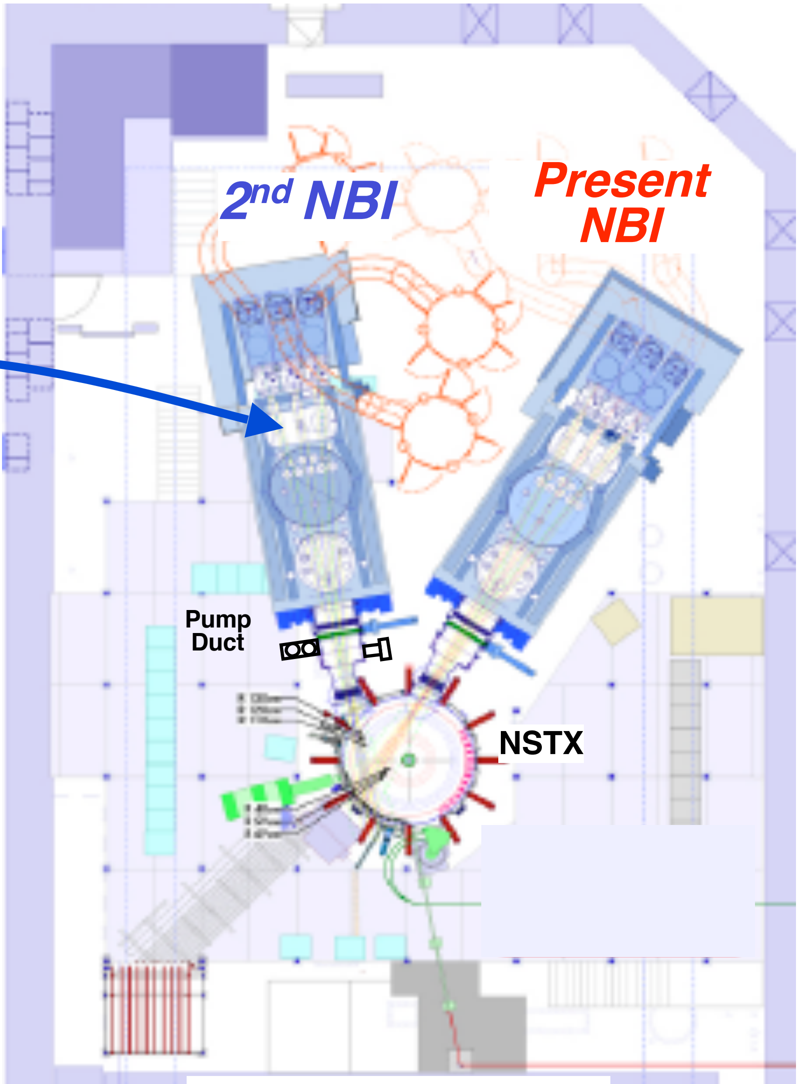


An NBI heating system available from TFTR could be moved to the NSTX test cell and installed next to the present NBI

- **PPPL has extensive experience operating, maintaining, refurbishing NBI**
- **NBI is well understood and has provided reliable heating to high β values in NSTX**

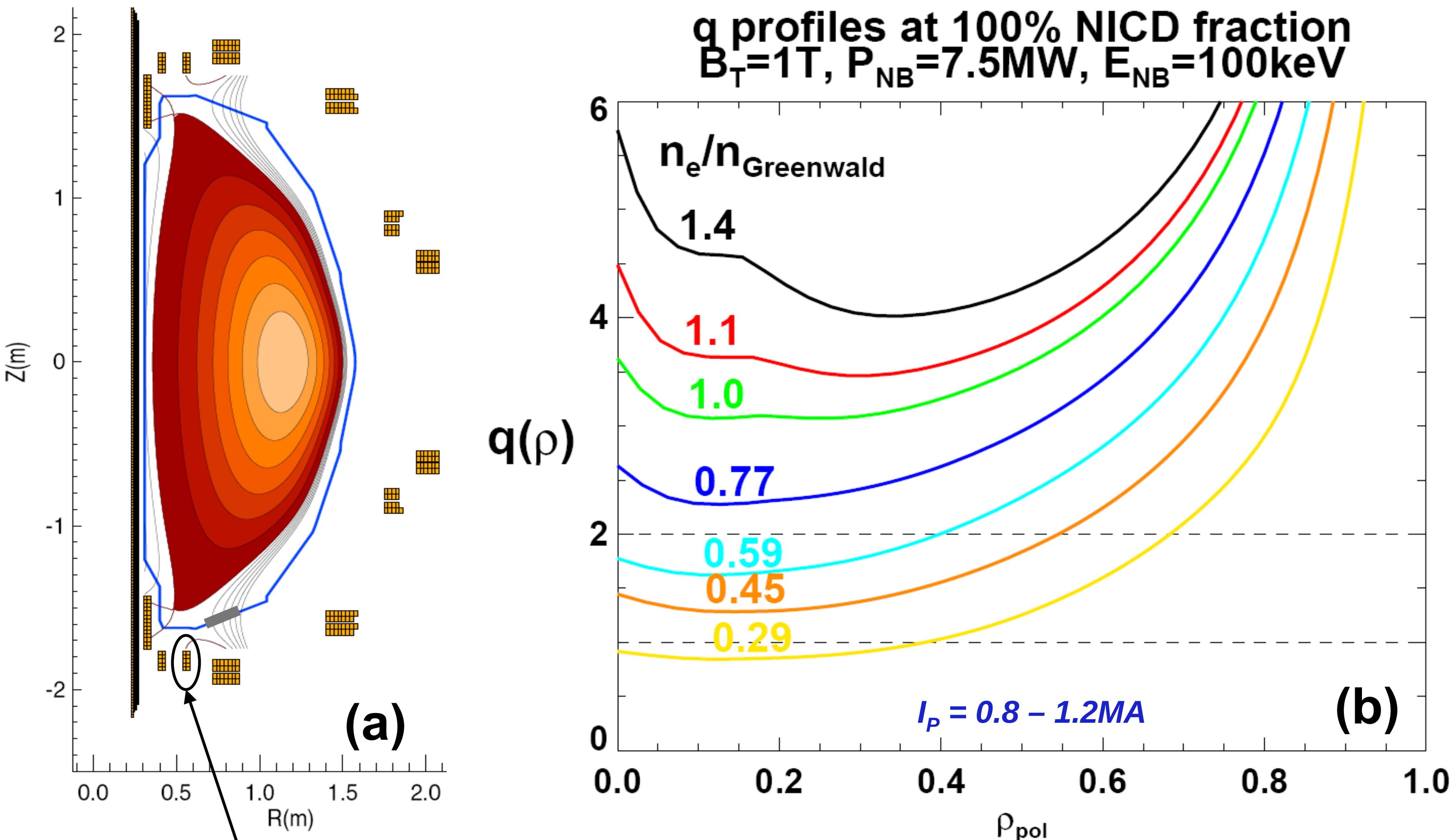


TFTR test cell



NSTX test cell

New CS will include additional PF coils for "X-divertor", and higher B_T enables q_{min} control using n_e to control NBI-CD



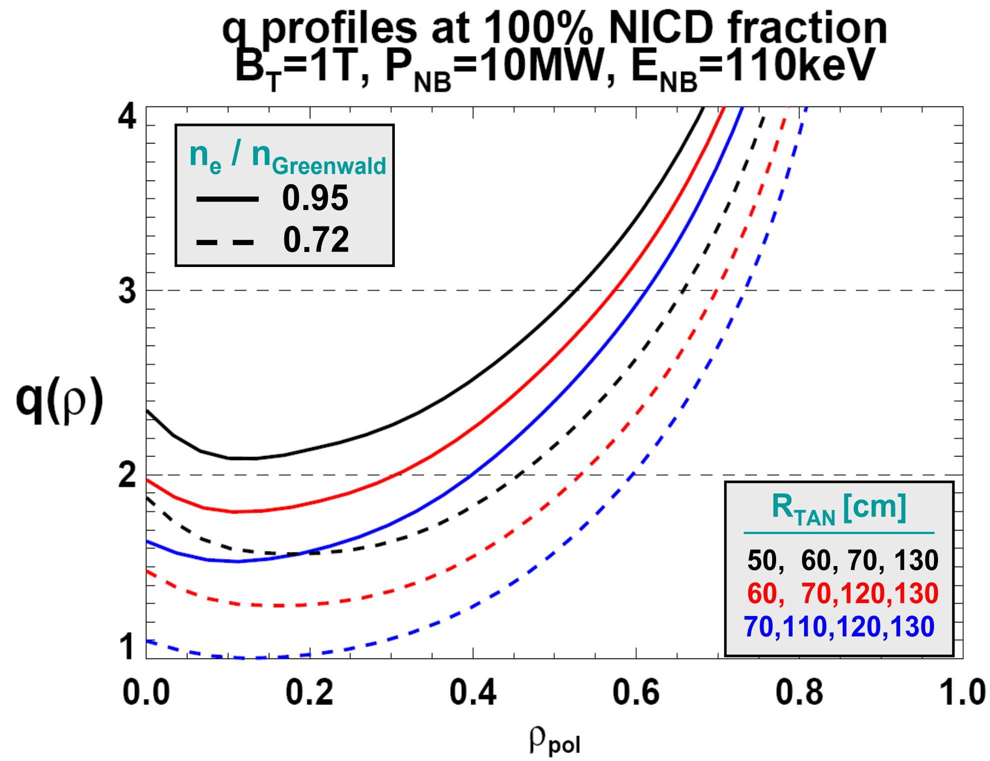
Additional divertor coils for very high flux expansion (up to 60) exhaust onto LLD in outboard "X-divertor" configuration

q_{min} > 1 achievable using existing 3 NBI sources (100keV, 7.5MW) + additional 4MW of HHFW for H_{98} = 1.2-1.4, β_N = 4.5-5, β_T = 10-12%

2nd NBI needed to support long-pulse (5s) fully non-inductive scenarios at high power at full TF (B_T = 1T)

- NBI duration 5s for 80kV → 5MW total per NBI, ~2s limit for ~7MW
  - 2nd NBI can double maximum power or double duration at fixed power

- Fully non-inductive scenarios require 7-10MW of NBI heating for $H_{98} \leq 1.2$
  - τ_{CR} will increase from 0.35 → 1s if T_e doubles at lower n_e, higher B_T
  - Need 3-4 τ_{CR} times for J(r) relaxation → 5s pulses → need 2nd NBI
  - f_{GW} > 0.7 needed at higher P_{NBI} to reduce core J_{NBICD} to maintain q_{min} > 1



Above: β_N=5, β_T=10%, I_P=0.95MA

β_N=6.1, β_T=16%, q_{min} > 1.3, I_P=1MA at B_T=0.75T also possible

2nd NBI + 1T → study transport, stability (especially NTM) of high q_{min} plasmas for NHTX, ST-CTF

2^{nd} NBI also needed to support long-pulse (5s) high-I_P partial-inductive scenarios at high-power at full TF (B_T = 1T)

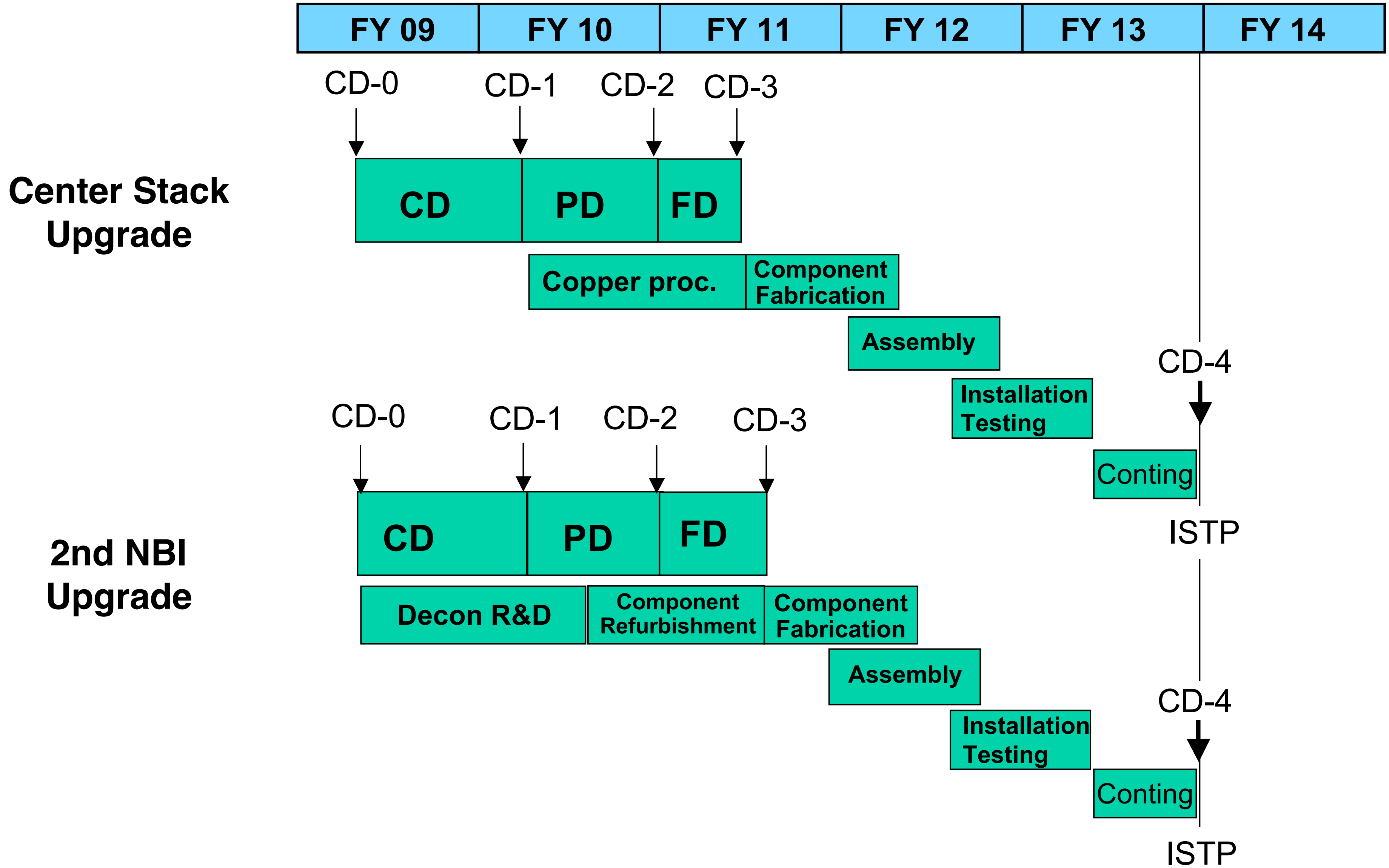
- Higher current expected to expand range of accessible T and ν^*
  - Accessible ν^* will depend on how confinement scales at higher field and current

- Access to higher current important for variety of physics issues – examples:
  - High-β_T physics at lower ν^* (RWM, NTV) – requires access to high I_P/aB_T
  - Core transport and turbulence at reduced ν^*, reduced $\chi_{i\text{-neoclassical}}$
  - Pedestal transport/stability, SOL width, heat flux scaling vs. current, …

- High I_P =1.6MA and B_T = 1T partially-inductively driven scenarios identified:
  - f_{NICD} = 65% with **q_{min} > 1**, β_N = 5, β_T =14%, NBI profile computed with TRANSP
    - Similar to present high NI-fraction discharges, but with 2× field and current
  - Higher current possible with present PF systems if l_i < 0.5
  - These scenarios also require ≥ 8MW of NBI heating power for $H_{98} \leq 1.2$

- Solenoid in new CS and PFs being designed to support 2MA plasmas for 5s

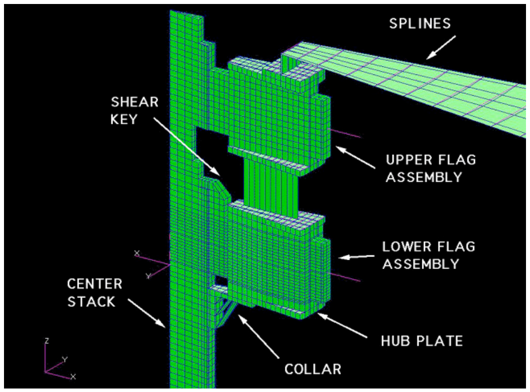
Design and R&D in FY 09-11 for CS & NBI Upgrades
Without Funding Constraints for FY 10 and beyond



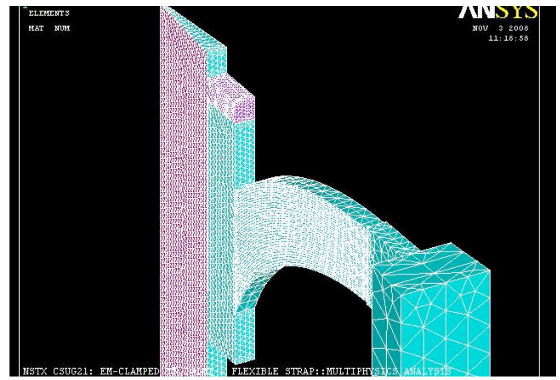
Conceptual Design to Retire Technical Risks
Perform Critical Designs and R&Ds Upfront

- **Reliable, robust CS TF joint design at 1T**

Present TF joint design

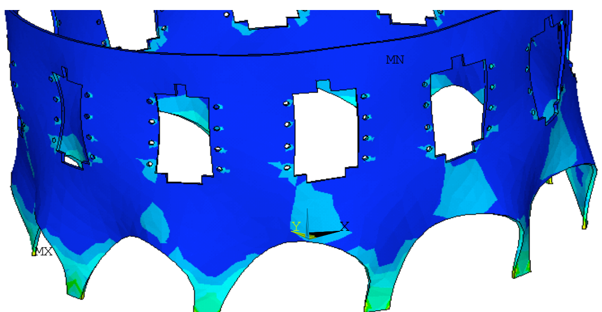


Candidate new TF joint design

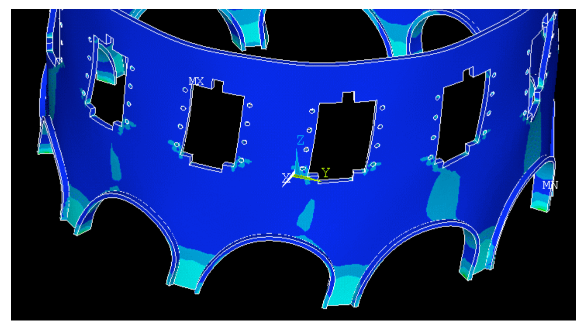


- Simplification
- 36 identical turns
- Self-closing joints

- **Appropriate structures enhancement for 1T, 2 MA operations**



Unbrella structure legs stressed excessively



Relatively simple structure enhancement brought down the stress to acceptable level

- **NBI beam box tritium decontamination**
  - Initial decontamination assessment has started in the TFTR Test Cell

NSTX participation in International Tokamak Physics Activity (ITPA) benefits both ST and tokamak/ITER research

Actively involved in 18 joint experiments – contribute/participate in 33 total

MHD, Disruption Control

• MDC-3	Joint experiments on neoclassical tearing modes (including error field effects)
• MDC-12	Non-resonant magnetic breaking
• MDC-14	Rotation effects on neoclassical tearing modes
• MDC-15	Disruption database development

Transport and Confinement

• TC-1 (was CDB-2)	Confinement scaling in ELMy H-modes: beta degradation
• TC-2 (was CDB-10)	Power ratio – Hysteresis and access to H-mode with H~1
• TC-4 (was CDB-12)	H-mode transition and confinement dependence on ionic species
• TC-10 (was TP-7)	Experimental ID of ITG, TEM and ETG turbulence + comparison w/ codes
• TC-15	Dependence of momentum and particle pinch on collisionality

Energetic Particles

• EP-2	Fast ion losses and Redistribution from Localized Losses

Pedestal and Edge Physics, Divertor, Scrape-off Layer

• PEP-6	Pedestal structure and ELM stability in DN
• PEP-19	Edge transport under the influence of resonant magnetic perturbations
• PEP-25	Inter-machine comparison of ELM control by magnetic field perturbations from midplane RMP coils
• DSOL-17	Cross machine comparisons of pulse-by-pulse deposition
• DSOL-21	Introduction of pre-characterized dust for dust transport studies in divertor and SOL

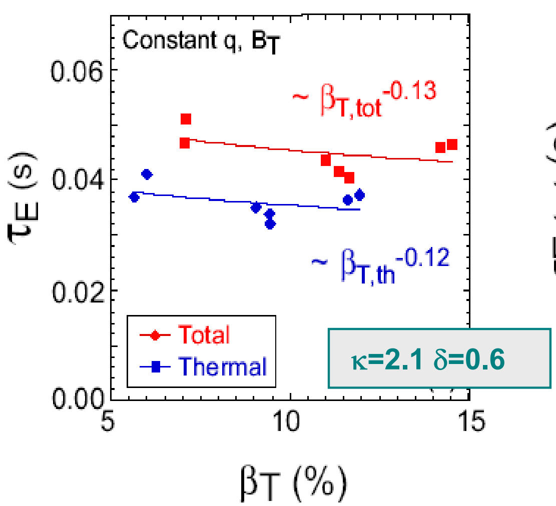
Integrated Operation Scenarios

• IOS-4.1	Access conditions for hybrid with ITER-relevant restrictions
• IOS-5.1	Ability to obtain and predict off-axis NBCD
• IOS-5.2	Maintaining ICRH coupling in expected ITER Regime.

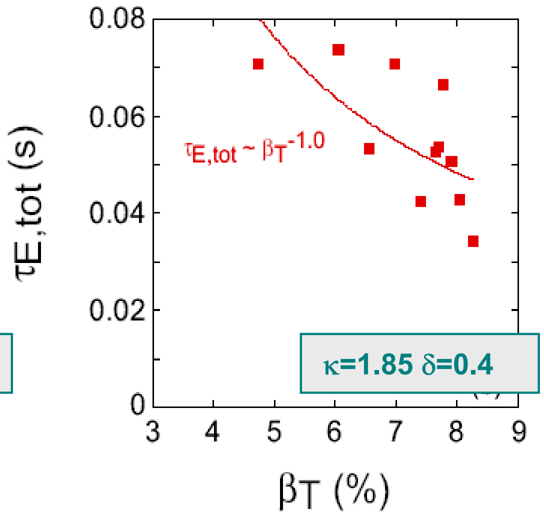
Previous examples of NSTX contributions to ITPA for ITER:

- **Transport:** β-dependence of H-mode confinement important to ITER advanced scenarios ($B\tau_{98y2} \sim \beta^{-0.9}$)
  - NSTX performed β-scan (factor of 2-2.5) at fixed q, B_T
  - Degradation of τ_E with β weak on NSTX for strongly shaped plasmas, stronger for more weakly shaped plasmas
  - Implies shape and/or ELM-type influences β dependence of H-mode confinement scaling

Small Type V ELMs

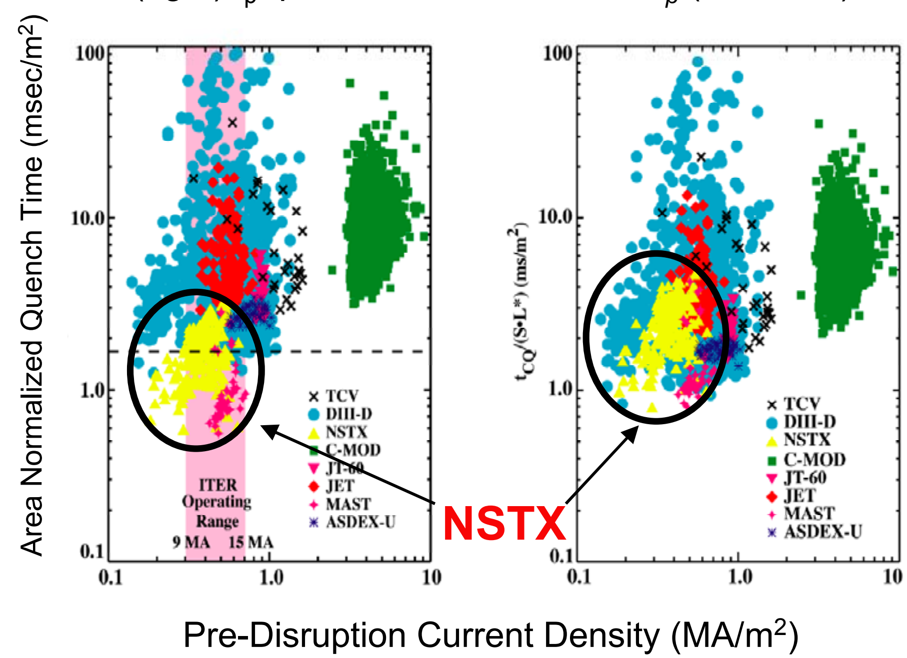


Type III/No → Type I ELMs



- **MHD:** Reduced normalized external inductance of low-A explains difference in I_P quench-rate
  - Implies tokamaks & STs have similar T_e during I_P quench phase (impurity radiation dominates dissipation of plasma inductive energy)

Area-normalized (left), Area and L_{ext}-normalized (right) I_p quench time vs. toroidal J_p (ITER DB)



NSTX FY09-11 support of ITER high priority research

- Impact of He (and possibly H) operation on H-mode
  - Important for H-phase of ITER operation
  - NSTX: Examine L→H threshold, global confinement, ELM stability
- ELM modification, suppression, control
  - Important for high fusion gain goal of ITER, essential for DEMO
  - Understand NSTX ELM modifications:
    - ELM stabilization with Lithium
    - ELM destabilization of with resonant magnetic perturbations (RMP)
  - RMP ELM control at lower q_{95}, reduced ν^* (HHFW, LLD), consider vertical jogs
- Validate neoclassical toroidal viscosity (NTV) flow damping theory
  - Important for minimizing mode locking during ITER RMP ELM control
  - NSTX: Additional expt/theory comparisons at varied ν^*, rotation, RMP spectrum
- Simulation of ITER test blanket module impact on plasma
  - Important for understanding impact of large predicted error fields
  - NSTX: Use EF/RWM coils to approximate TBM spectrum

NSTX will make world-leading contributions to ST development, and contribute strongly to ITER and fundamental toroidal science

- The FY09-11 plan:
  - Focuses research to address key gaps in extrapolating to next-step STs
    - Increase and understand beam-driven current at lower n_e, ν^* (also assess integration)
    - Increase and understand H-mode confinement at low ν^*
    - Demonstrate and understand non-inductive start-up and ramp-up
    - Demonstrate and understand means to "tame the plasma-material interface"
    - Sustain β_N and understand MHD near & above no-wall limit
  - Contains substantial and targeted upgrades:
    - FY09-11 – Improved HHFW for e-heating and CD for ramp-up & sustainment
    - FY10 – Liquid Lithium Divertor (LLD) for lower n_e and ν^*, BES for transport & Alfvén modes
    - FY11 – MSE-LIF for q(r) w/o heating NBI, |B| for total (& fast-ion) pressure, improved MPTS
- These plans and upgrades enable exciting new science in all topical science areas:
  - Measure & understand underlying instabilities that cause **anomalous energy transport**
  - Understand **RWM critical rotation and viscous torques** and dependence on lower ν_i
  - Understand role of ν^* and **Lithium on pedestal transport/stability** and divertor physics
  - Develop predictive capability for **fast-ion redistribution from multi-mode AE** for ST, ITER
  - Integrate CHI into normal ops, develop/understand I_P **ramp-up w/ HHFW BS overdrive**
  - **Push toward 100% non-inductive operation** by increasing NBI-CD w/ reduced collisionality