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Mini-workshop for planning NSTX Integrated Scenario Development (ISD) research for the FY09-13 5-year plan

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Presented by:

J.E. Menard, PPPL

Integrated Scenario Development Task Group

February 16, 2007
Princeton Plasma Physics Laboratory

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ISD 5yr plan mini-workshop agenda

1. Introductory remarks (1-1:30PM) – J. Menard
   1. Planning process and schedule
   2. NHTX overview, and relation to NSTX research
2. Integrated Scenario Modeling for NSTX 5yr Plan (1:30-2:00PM) – C. Kessel
3. CHI performance extension and system upgrades (2:00-2:30PM) – R. Raman
4. PF-only start-up research (2:30-3PM) – M. Ono
5. Development of Long Pulse Double Barrier Plasmas (3:00-3:30PM) – C. Bush
6. Discussion (3:30-5PM) – all
   1. TRANSP/TSC modeling needs
   2. Possible theory input
   3. Final questions to motivate discussion
   4. NBICD optimization in NHTX, and possibilities in NSTX (J. Menard)

Starting Preparation for the Next NSTX 5 Year Plan

Dec. 22, 2006	Initial brainstorming "kick-off" meeting on key research opportunities for the next 5 years
Jan. 15, 2007	Input from theory community for theory/modeling support
Feb. 2007	**The "Leads" to organize mini-meetings to identify key research opportunities**
March 2007	Develop preliminary upgrade cost estimates (manager's estimates)
April 2007	Develop draft plan for key approaches in support of opportunities
April 2007	Develop and review outline for the draft plan
April 2007	Team meeting to review approaches and opportunities
July 2007	Initial draft plan ready
August 2007	Team discussion of the initial draft plan
Mid-Sept, 2007	**Tokamak Planning Workshop at MIT**
Jan. 2008	NSTX PAC reviews the draft plan
Feb. 2008	Final draft plan ready for review by the team
April 1, 2008	Final plan (document) ready
1 wk before review	Final presentation material ready
~ May 2008 (TBD)	**New 5 Year Plan Review meeting**

Possible NSTX Strategy for the Next Five Years

NSTX to Address Scientific Issues Important for NHTX, ST-CTF, ITER, and Toroidal Fusion Plasmas

- **Establish physics basis for design and construction of National High-power advanced Torus eXperiment (NHTX)**
- **Explore physics of Spherical Torus / Spherical Tokamak to provide basis for attractive U.S. Component Test Facility (CTF) and Demo.**
- **Support preparation and resolution of the issue cards for burning plasma research in ITER using physics breadth provided by ST; support and benefit from "ITPA Specific" activities.**
- **Complement and extend tokamak physics experiments, maximizing synergy in investigating key scientific issues of toroidal fusion plasmas**

The development of advanced fusion reactors will require the integration of key areas of fusion science

- Four key requirements are well known:
  1. High thermal confinement, well confined α's
  2. High plasma beta
  3. Steady state operation
  4. **Solution for *reactor-level* high-heat-flux plasma-boundary interface**

- The integration of advanced-reactor-level high-heat-flux handling with high confinement, high β, and steady-state operation has not been demonstrated
  - and apparently will not be demonstrated by planned long-pulse devices

- **NHTX mission:**

  "To study the integration of high-confinement, high-beta, long-pulse non-inductive plasma operation with a fusion-relevant high-power plasma-boundary interface."

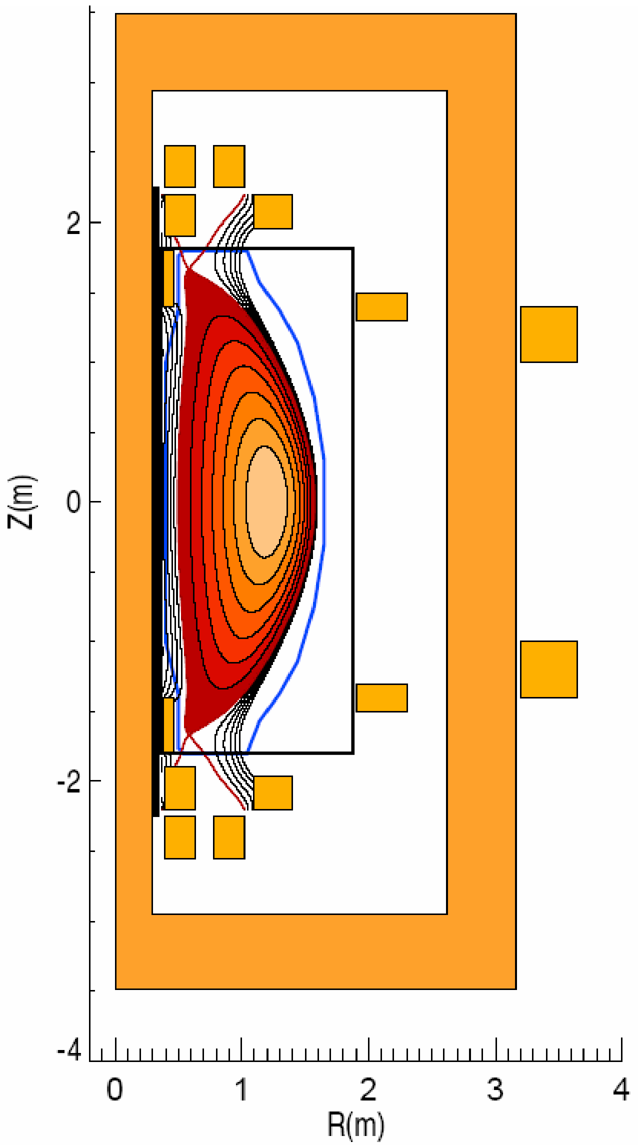
NHTX can lead the field in the integration necessary for successful CTF/FDF & Demo

JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	Upgrade Capability
LHD	3.90	0.60	10	3	0.11	10,000	–	H	Upgrade capability
SST-1	1.10	0.20	3	3	0.23	1000	0.2	H (D)	Initial heating
W7-X	5.50	0.53	10	2	0.09	1800	–	H	30MW for 10sec
NHTX	**1.00**	**0.55**	**50**	**50***	**1.13**	**1000**	**3.5**	**D (DT)**	**Initial heating**
ITER	6.20	2.00	150	24	0.21	400-3000	15.0	DT	Not for divertor testing
Component Test Facility Designs									
CTF (A=1.5)	1.20	0.80	58	48	0.64	weeks	12.3	DT	2 MW/m^2 neutron flux
FDF (A=3.5)	2.49	0.71	108	43	1.61	weeks	7.0	DT	2 MW/m^2 neutron flux
Demonstration Power Plant Designs									
ARIES-RS	5.52	1.38	514	93	1.23	months	11.3	DT	US Advanced Tokamak
ARIES-AT	5.20	1.30	387	74	0.85	months	12.8	DT	US Advanced Technology
ARIES-ST	3.20	2.00	624	195	0.99	months	29.0	DT	US Spherical Torus
ARIES-CS	7.75	1.70	471	61	0.91	months	3.2	DT	US Compact Stellarator
ITER-like	6.20	2.00	600	97	0.84	months	15.0	DT	ITER @ higher power, Q
EU A	9.55	3.18	1246	130	0.74	months	30.0	DT	EU "modest extrapolation"
EU B	8.60	2.87	990	115	0.73	months	28.0	DT	EU
EU C	7.50	2.50	794	106	0.71	months	20.1	DT	EU
EU D	6.10	2.03	577	95	0.78	months	14.1	DT	EU Advanced
SlimCS	5.50	2.12	650	118	0.90	months	16.7	DT	JA
CREST	7.30	2.15	692	95	0.73	months	12.0	DT	JA

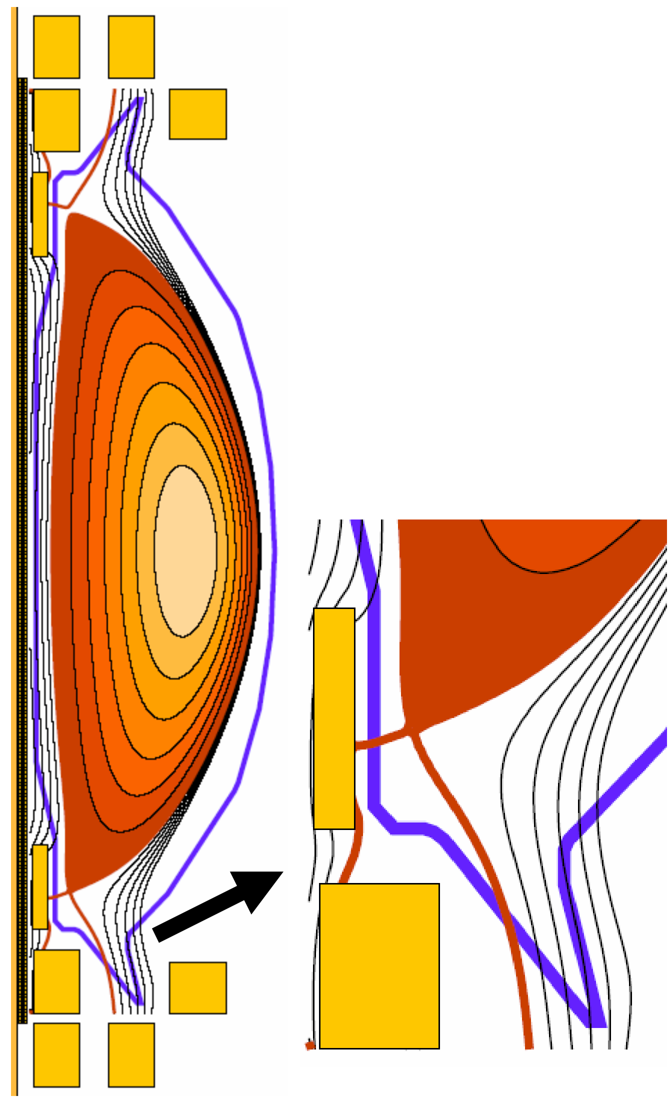
*** Flux compression, low R_X/R, SND, additional power allow higher heat flux.**

Single coil set supports range of divertor configurations

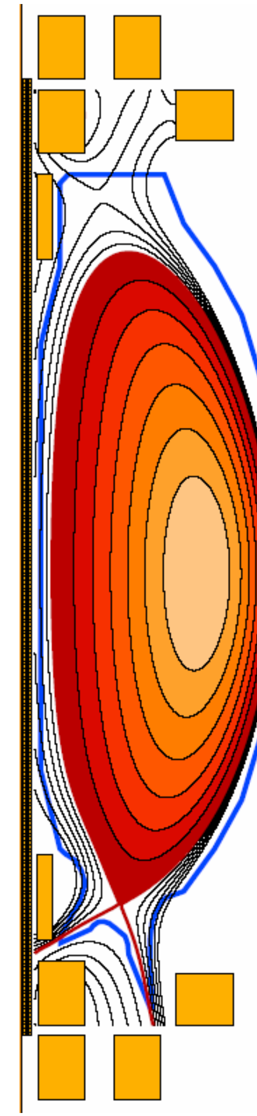
Open DN divertor



Pumped DND, JET-like

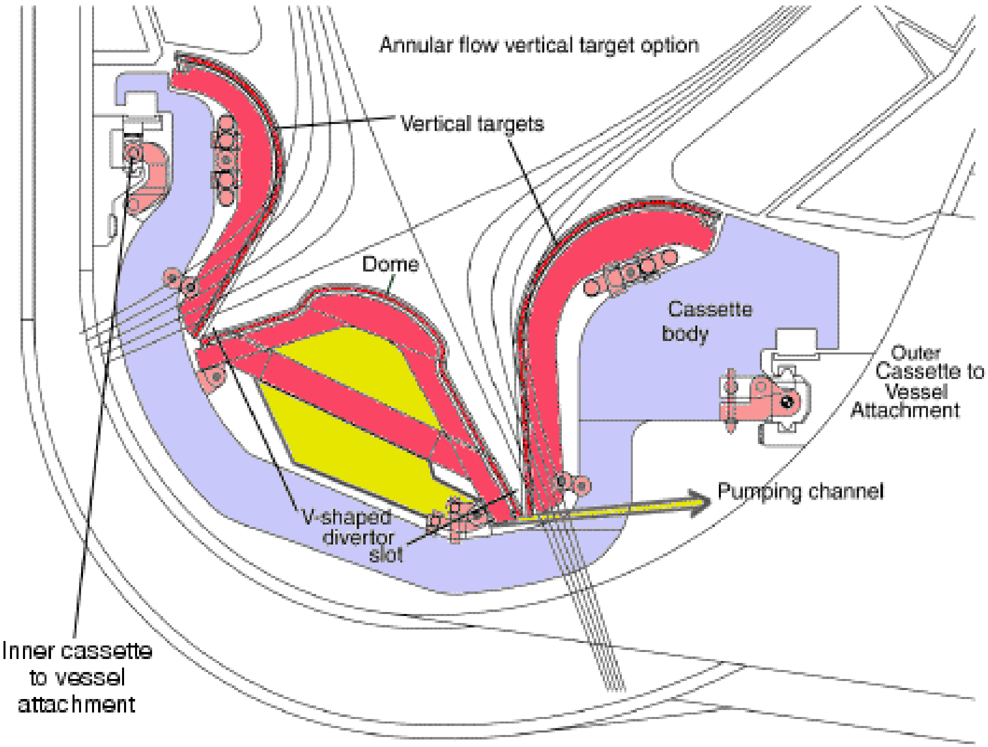


ITER-like LSN divertor

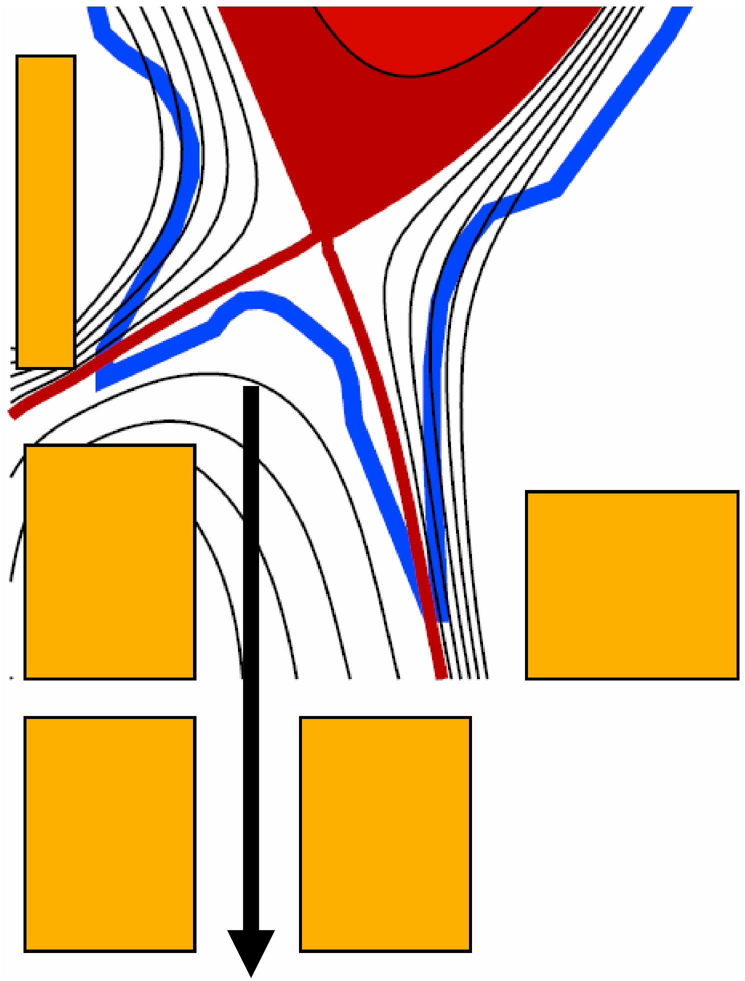


NHTX coil set supports ITER-like LSN divertor

ITER



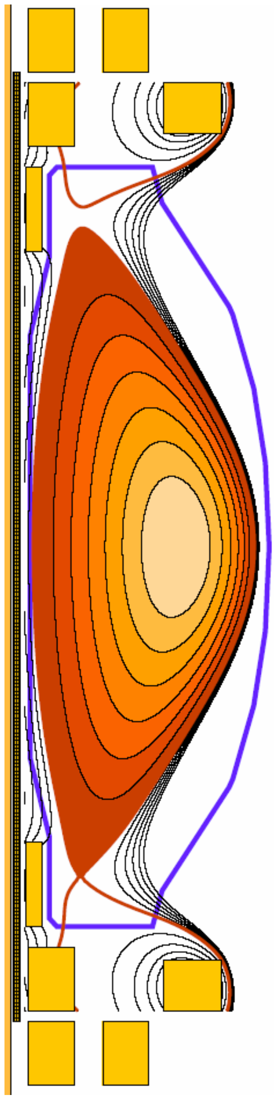
NHTX



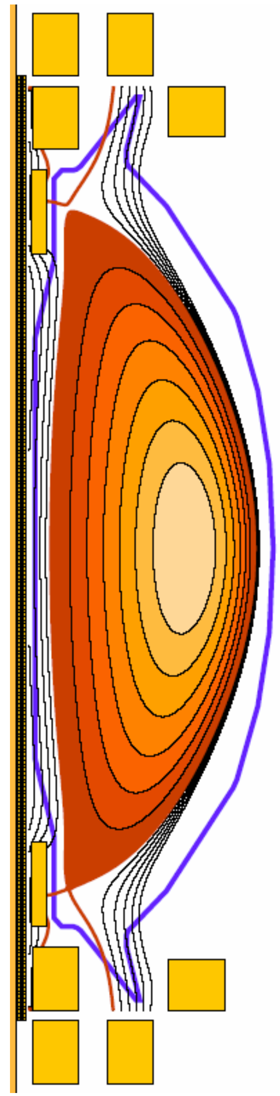
Pumping channel from dome

Coil set supports wide range of boundary shapes

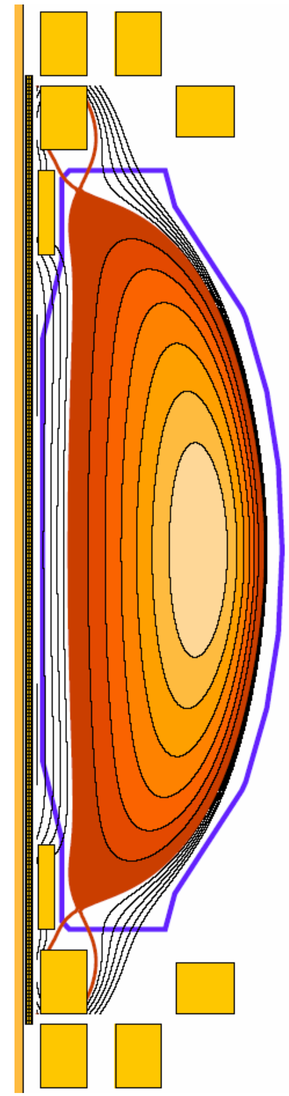
DND w/ negative squareness $\zeta \approx -0.15$



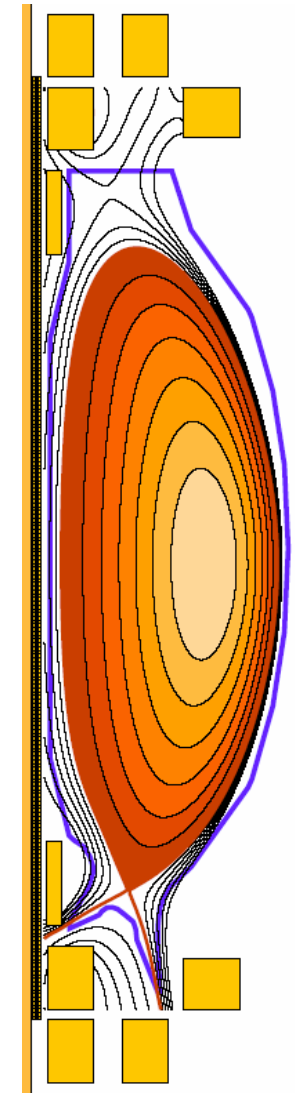
DND w/ near zero squareness



DND w/ positive squareness $\zeta \approx 0.25$



Example LSN shape



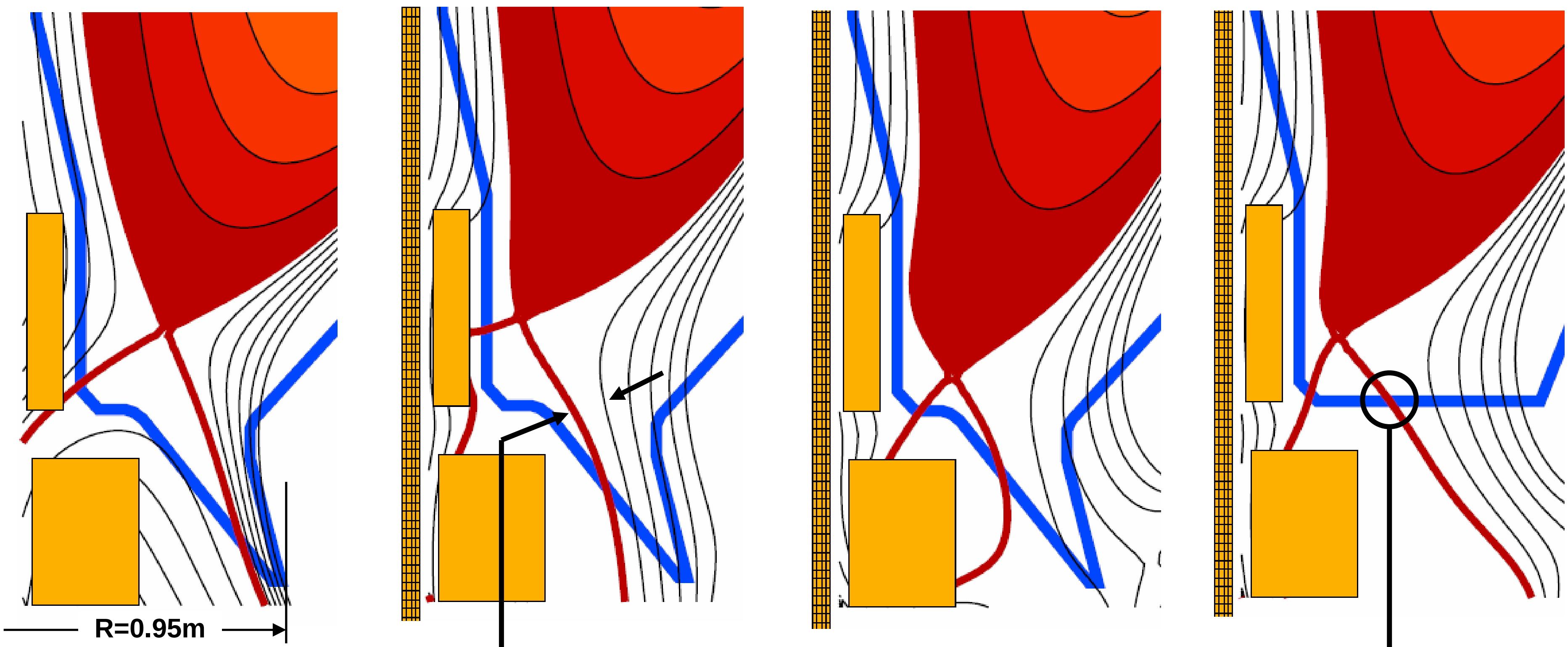
Divertor coil set supports wide range of flux expansion

Poloidal flux expansion factor $f_{exp} \equiv |\nabla\psi|_{mid\text{-}plane} \,/\, |\nabla\psi|_{strike\text{-}point}$

Poloidal B-field angle of incidence into target plate $\equiv \alpha_p$

Total B-field angle of incidence into target plate $\equiv \alpha_t$

f_{exp} = 2.8	f_{exp} = 9	f_{exp} = 17	f_{exp} = 35
α_p=22° α_t=5.1°	α_p=23° α_t=1.8°	α_p=25° α_t=1.0°	α_p=64° α_t=1.1°



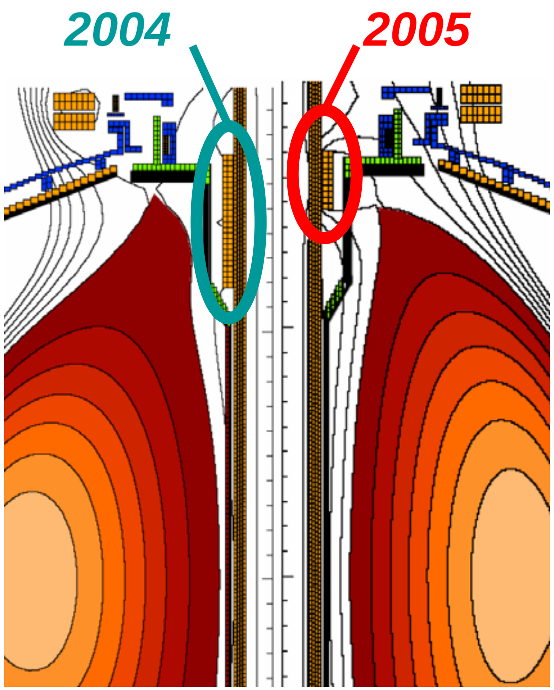
Flux contours have 5mm separation at midplane

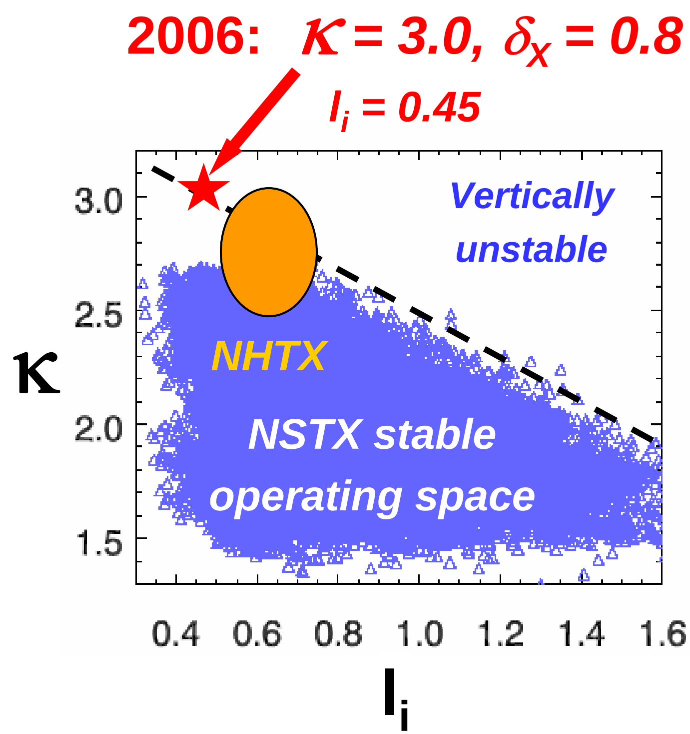
f_{exp}, α values computed at strike-point

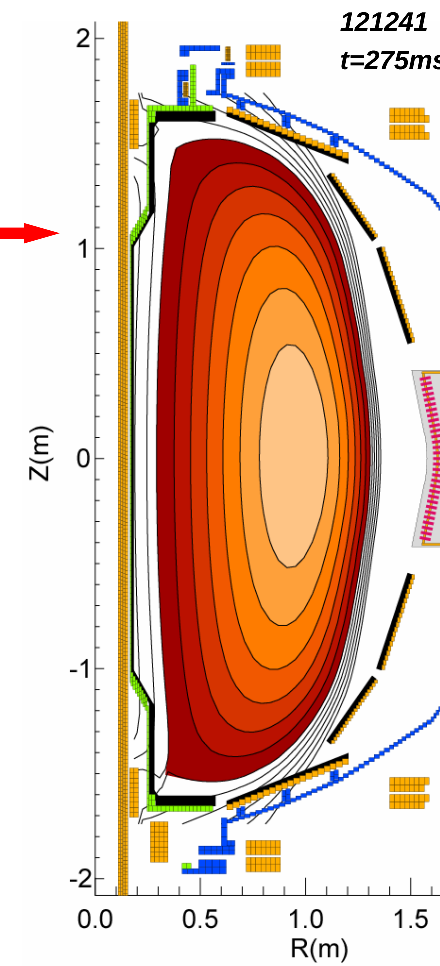
NHTX requires advanced control of high κ/δ boundary, strike point placement, and flux expansion

- NSTX: Sustained $\kappa \geq 2.8$ (reached $\kappa = 3$) for many τ_{WALL} using rtEFIT isoflux control
- High κ n=0 stability research important for NHTX and CTF/FDF design studies

Divertor coil upgrade







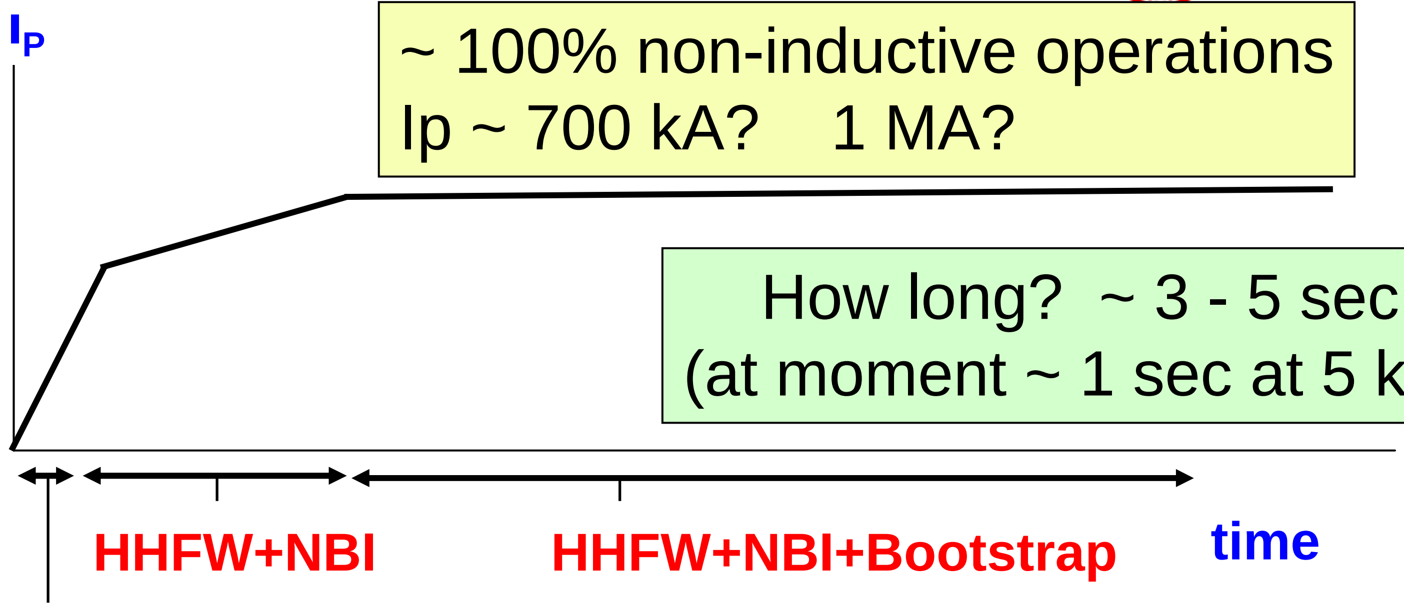
*Gates, et al., PoP **13** (2006) 056122*
*Gates, et al., NF **46** (2006) 17*

GENERAL ATOMICS

What plasma parameters should NSTX aim to establish physics basis for NHTX design?

~ 100% non-inductive operations
Ip ~ 700 kA?   1 MA?

How long? ~ 3 - 5 sec
(at moment ~ 1 sec at 5 kG)



CHI+EC/EBW
Or
PF-only?

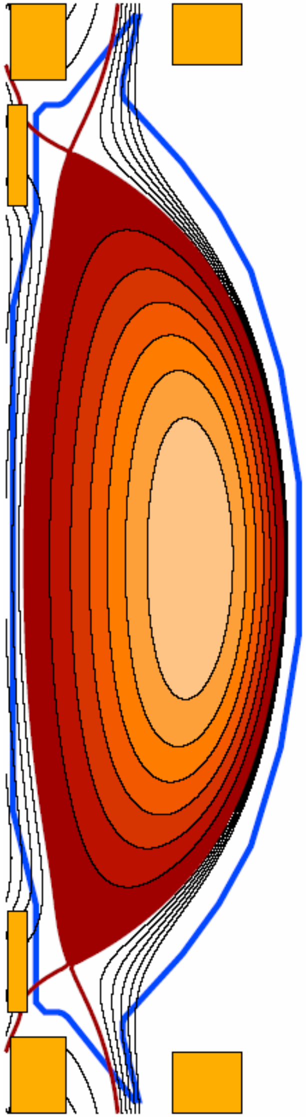
How much solenoid-free I_P can be achieved?
Can we achieve 500 kA?

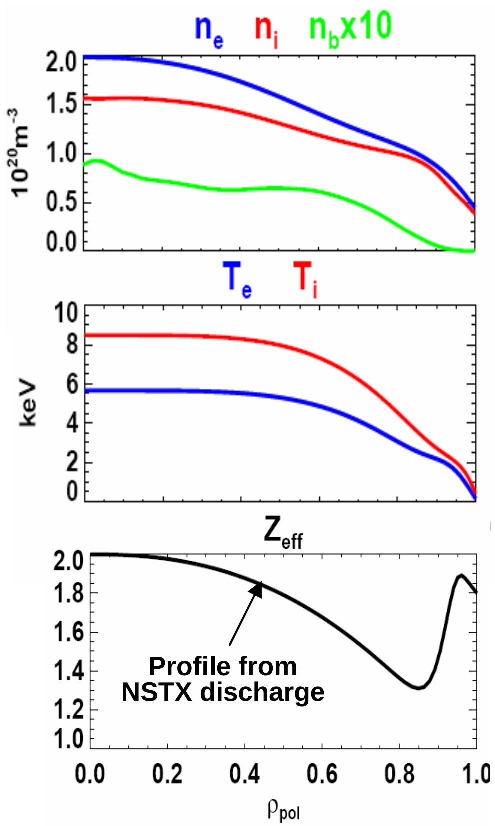
- Advanced control including RWM, EF, j(r) control?
- NTM control?
- ELMs - mitigation or small ELMs likely crucial
- Advanced divertor - liquid lithium divertor target, cryo...?

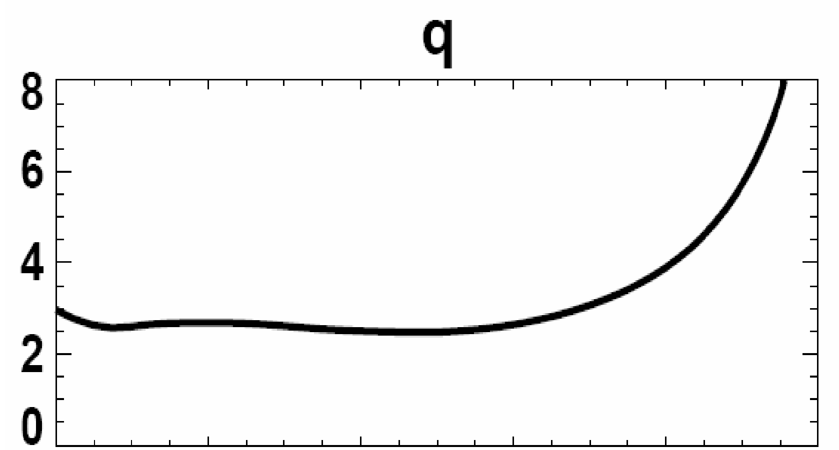
Parameter comparison of present NSTX high-f_{NI}, fully non-inductive NSTX target, and NHTX design

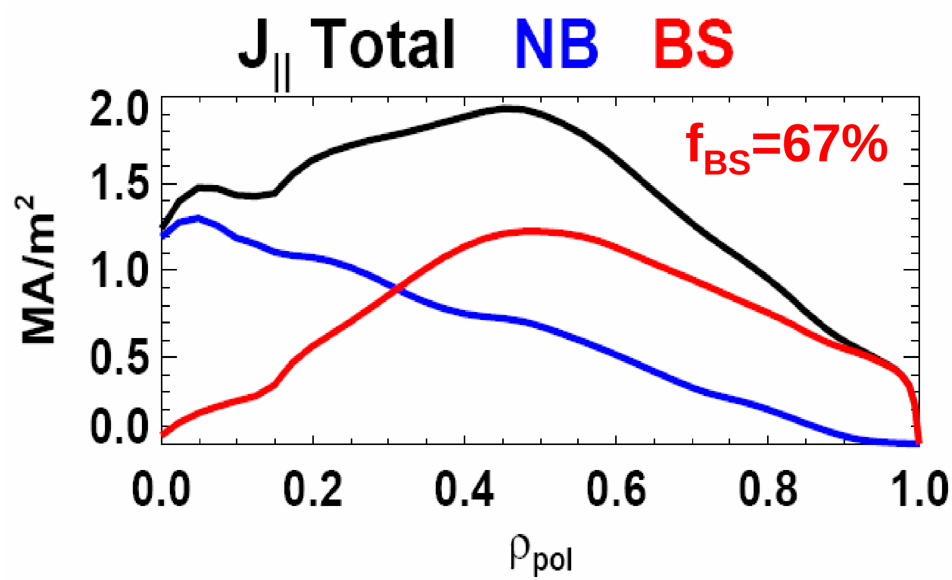
	NSTX 65% NI experiment TRANSP BS, NBI	NSTX full NI target TRANSP BS, NBI	NHTX full NI target 0D scaling analysis
A	1.55	1.65	1.8
IP (MA)	0.75	0.7	3 to 4
BT0 (T)	0.45	0.52	2.1
beta-N	5.6	6.7	4 to 5
beta-P	1.5	2.7	1 to 1.25
beta-T (%)	17	15	12 to 15
li	0.6	0.5	
kappa	2.3	2.6	2.7
delta-X lower	0.75	0.85	0.6
qmin	1.3	2.4	1 to 4
qstar	3.9	5.6	3.4
f-BS	0.55	0.85	0.6 to 0.75
f-NBICD	0.1	0.15	0.4 to 0.25
f-NI	0.65	1	1
HH98	1	1.3	1.3

NHTX: A=1.8, κ=2.85, I_P=3MA target plasma with self-consistent J(ρ) from NBI and BS with q_{MIN} > 2.4







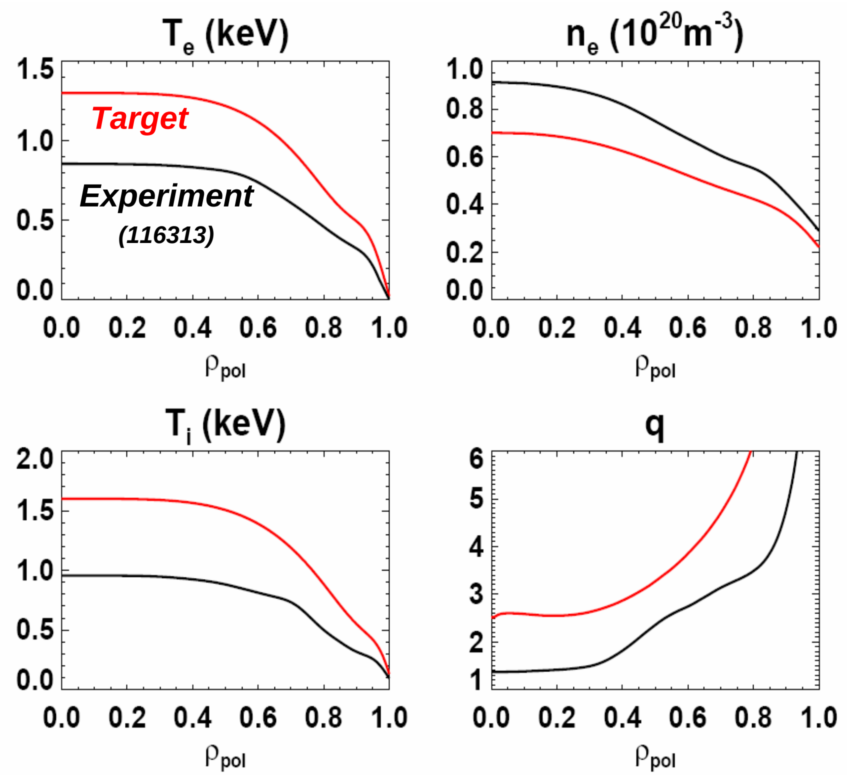


R_{TAN} = 115cm, Z_{TAN}= 0cm

NHTX goal for sustainment mission is full NI-CD for 60-1000s

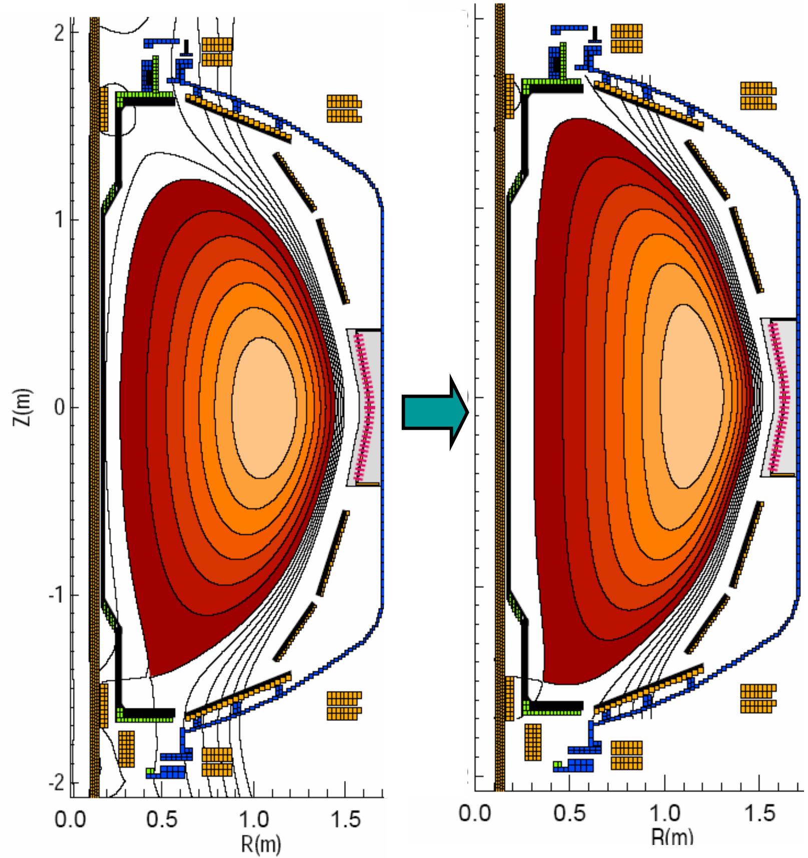
NSTX - extrapolation to full NI requires higher τ_E, q_{MIN}, κ, δ

- Need 60% increase in T, 25% decrease in n_e
  - **Lithium for higher τ_E & density control?**
    - **20% increase in thermal confinement**
    - **30% increase in HH_{98}**
  - **Core HHFW heating**
- Want $q_0 \approx q_{min} \approx 2.4 \Rightarrow$ higher with-wall limit



- Higher κ for higher q, β_P, f_{BS}
- High δ for improved kink stability

κ= 2.3, $\delta_{X\text{-}L}$ = 0.75, δR_{SEP} = -1cm | **κ= 2.6, $\delta_{X\text{-}L}$ = 0.85, δR_{SEP} = -2mm**



NHTX mission, and relation to NSTX

- NHTX aims to integrate fully non-inductive operation with high beta, high confinement, and **high-heat-flux solutions**

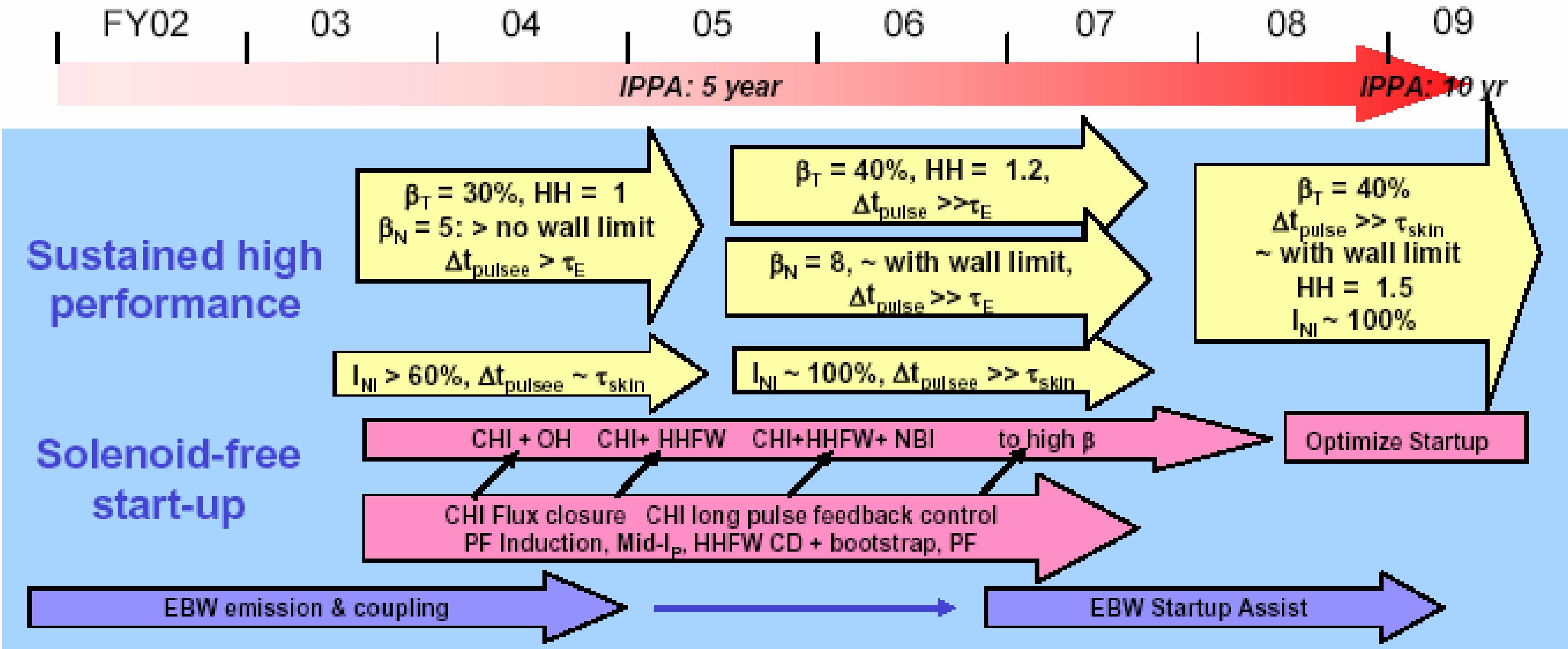
NSTX issues directly relevant to design/operation of NHTX:

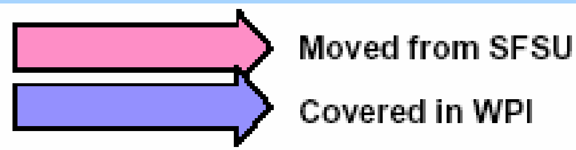
1. NSTX has not yet demonstrated 100% NI operation
2. NSTX diagnosis/control of divertor, SOL, pedestal incomplete

- **If major goal of the next 5 year period of NSTX operation is to support NHTX/CTF, should these 2 topics be our focus?**

- NSTX and NHTX also need to carry forward development of solenoide-free startup techniques needed for ST-CTF
  - Iron core transformer, PF-only, LHCD, NBICD, and CHI all possibilities
- Advanced control of shape, vertical stability, and MHD modes also critical to success of NHTX mission

Integrated Scenarios and Solenoid-Free Start-up





- Major accomplishments
  - Demonstrated sustained f_{NI} = 65%, β_N above no-wall limit, HH=1.1
  - Validated inductive and non-inductive CD models/diagnostics at low-A
  - Observe current redistribution from MHD – relevant to ITER hybrid mode
  - Produced 160kA closed-flux current using CHI start-up

Possible TSC/TRANSP modeling tasks

1. Updated Ip HHFW ramp-up modeling incorporating higher Bt=5 or 5.5kG results, and possible changes in launch spectrum, assuming we can/will modify antenna feeds.
2. Explore impact of early diverted/large bore plasma on q-profile evolution, stability, and NICD fraction
3. Explore impact of early HHFW heating on q-profile evolution, access to steady state, stability, and NICD fraction - using #2 above.
4. Expand range of scenarios/profiles which achieve fully non-inductive operation at moderate betan = 5-6 - scan HH, density, and beam CD diffusion/broadening to understand dependencies and trade-offs
5. Explore impact of addition of 100-200kA (?) of off-axis EBWCD to moderate betan fully NI scenario.
    1. How much does this increase beta?
    2. Is this approach extrapolable in time to high beta (30-40%) scenario which we are presently neglecting in our near-term plans?
6. Time-evolving simulations of NHTX plasmas in NSTX - higher A = 1.8-1.85 and lower delta=0.6.
    1. Is this stable? Can it influence NHTX design?

Possible input from theory – comments from Kessel

1. **Transport:** what gyrokinetic analysis is being done, what does it tell us, anything useful for predictive simulations of profiles?
2. **RF:** HHFW was done with ray-tracing and we can gather those results better, are people working on full wave HHFW?
    1. coupling analysis that is emerging with TOPICA and whether this can help us
    2. More EBW analysis, anything new from Harvey, new insight into this area
3. **NB/fast particles:** are we getting any analysis that involves scans to show the trends we might identify in the expts, in particular fast particle redistribution
4. **MHD:** a good 2D equivalent of the 3D structure would be nice, since I promised it a year ago or so, Bialek has agreed to work on this
5. **CHI, 3D MHD**: beyond CHI, consider theory work on plasma gun startup? extrapolation to NHTX?

Additional questions to motivate discussion

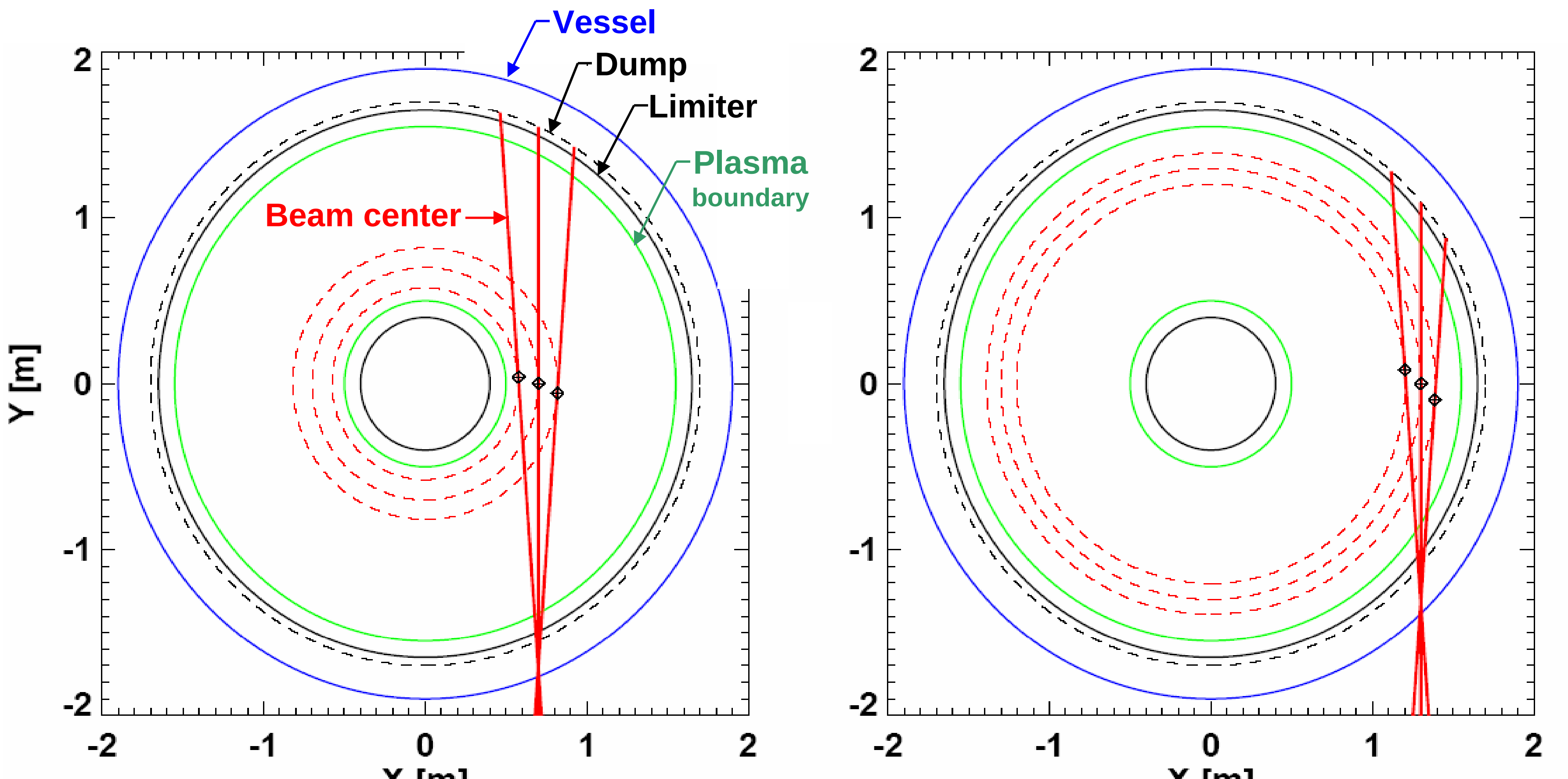
1. Assuming effective coupling, how could EBWCD contribute to sustaining high performance plasmas in NSTX and project to NHTX?
2. How do we link a solenoid-free startup plasma to a high performance plasma with high non-inductive fraction?
3. Can off-midplane control coils enable ELM mitigation in high-betan long-pulse scenarios?
4. Can operation with hot walls (in addition to Lithium) facilitate improved density control and prototype hot-wall operation in NHTX?
5. Can NBI reorientation improve the NBICD efficiency and provide a more stable steady-state q profile?

Can NBICD be optimized further for NSTX?

- For NHTX, NBI Z_{TAN} and R_{TAN} variations allow control of J_{NBICD}, and more current is driven for large R_{TAN}
  - Analyzing engineering tradeoffs of ΔR vs. ΔZ beam shift
- Will revisit possible advantages of NBI re-orientation as function of I_P and B_T for NSTX
  - Previous studies found no significant advantage at present NSTX current and field
  - If field and current are increased with centerstack upgrade, then beam realignment could become more advantageous.

NHTX: scan R_{TAN} within range $R_0 \pm 30$cm to assess NBICD efficiency and profiles

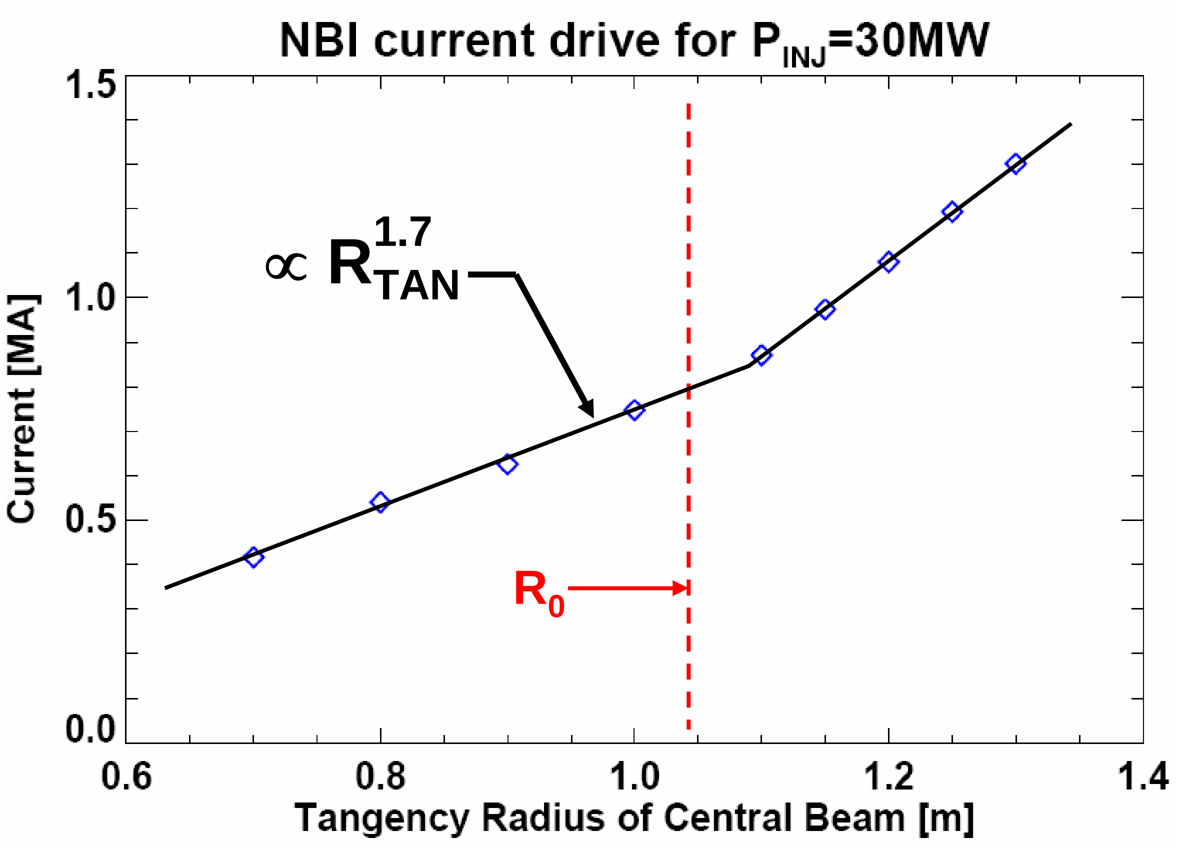
- Fix source cross-over radius at R_{CO} = 1.85m to be near vessel entrance
- Simulates horizontal beam-line swing with bellows near vessel



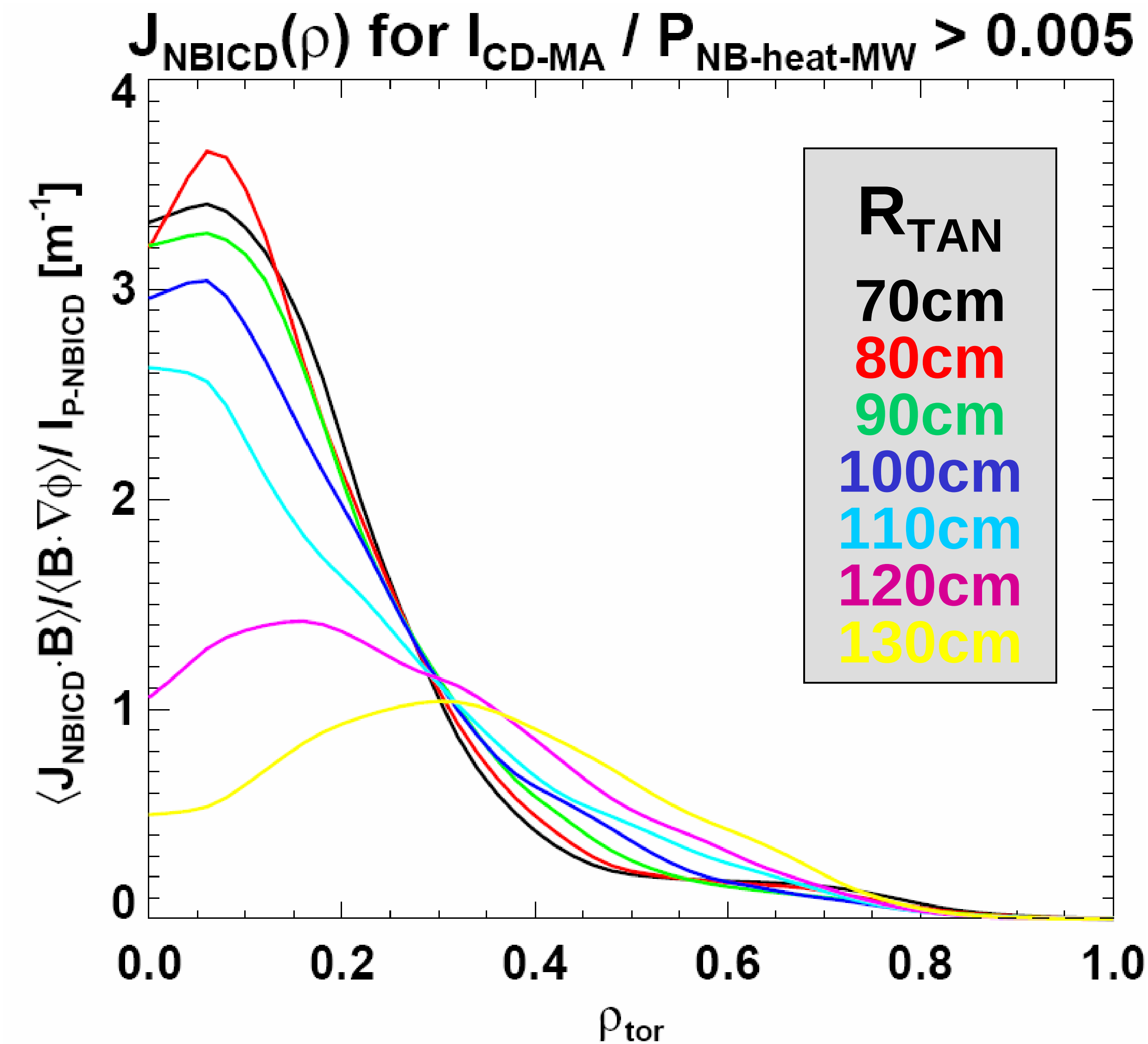
Tangency radius = 70cm for central source   Tangency radius = 130cm for central source

Driven current increases × 3 for R_{TAN}=0.7 → 1.3m and increases more quickly w/ radius for $R_{TAN} > R_0$

NBICD for $\overline{n}_e = 1.4\times10^{20}m^{-3}$, $\overline{T}_e$=4.2keV, f_{GW} = 0.43

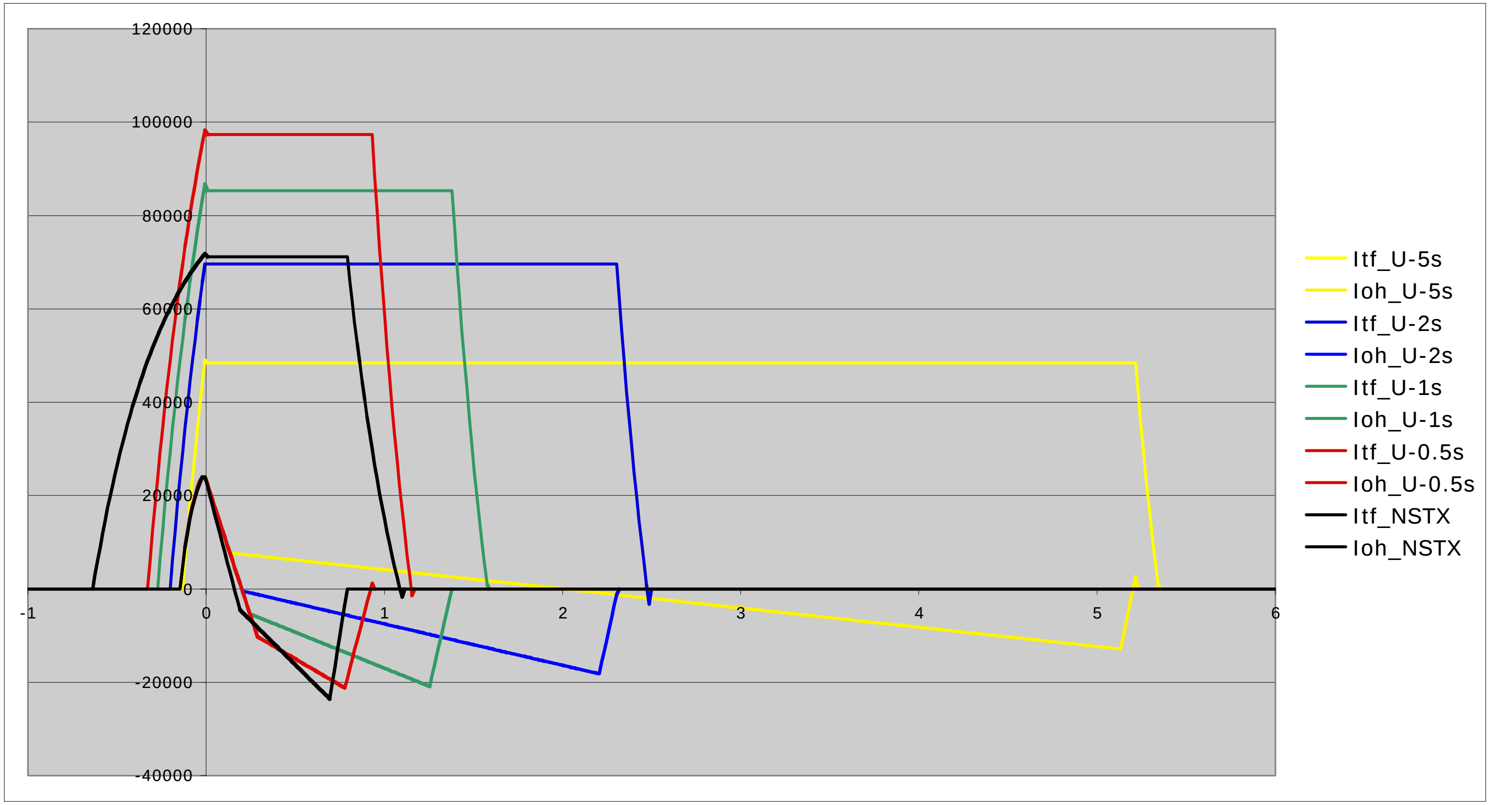


Beam tangency radius variation would enable control of core current and *q* profile



Discussion of merits/impact of NSTX CS upgrade

Tflat-->	0.5s	1s	2s	5s
Ip	1.5	1.3	1	0.7
Bt	0.82	0.72	0.59	0.41
%Flat Top Flux	51%	29%	20%	19%



- Still working to develop optimization approach
- Will look also at BeCu solution