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Physics and Engineering Design Considerations for NHTX

National High-power advanced Torus eXperiment

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General Atomics

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.**

NHTX Heating and Current Drive

- Neutral beams: 32 MW, 120 kV D_0 NBI, steerable off axis
- 18 MW RF – type to be determined
- Results from NSTX, C-MOD, DIII-D will be critical to selection of RF system(s)
  - EBWCD: High efficiency, remote coupling.
  - Inside-launch 120 GHz 2nd harmonic ECCD: lower efficiency, more complex access.
  - LHCD: High efficiency, intimate coupling.
- 2MA bootstrap current at operating point
- For confidence in 3.5 MA steady-state operation, desirable to be able to drive ~ 1.5 MA with beams + RF (R_0 = 1m)

Beyond high P/R, NHTX provides high $P/P_{L\rightarrow H}$ required for testing radiative power dispersal techniques

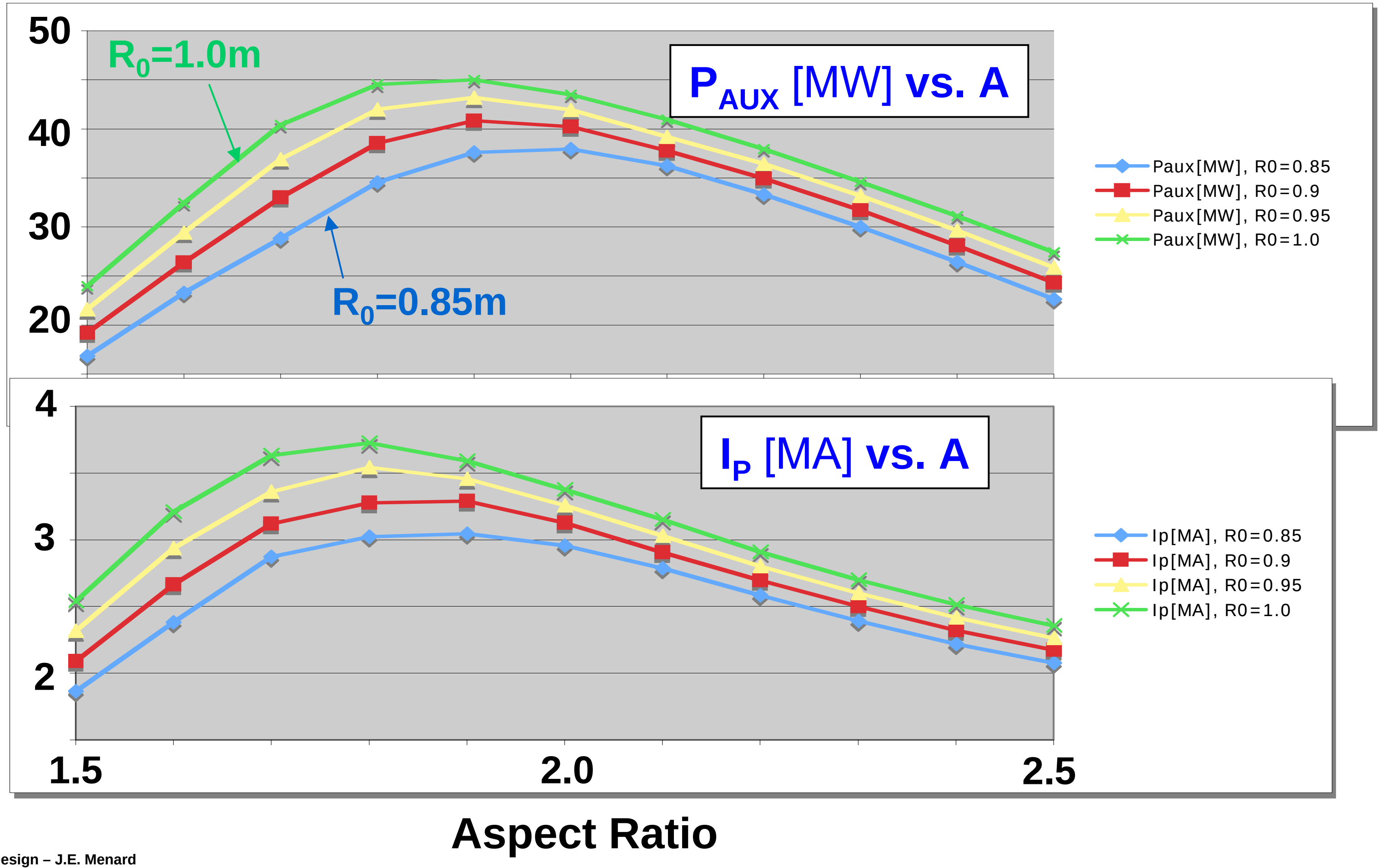
- Can fusion plasmas operate at high τ_E and β with 90% core radiated power, to reduce divertor heat flux?
- Physics test requires input power exceeding H-mode threshold power by a very large factor ~ 10.
- NHTX has unique capability to test the Demo-relevant physics in this area:

	$P_{in}/P_{L\rightarrow H}$ @ $0.85\times n_{gw}$
ITER	**3.6**
JT-60SA	**4.9**
ARIES-AT	**11**
NHTX	**12**

The solution to the power-dispersal problem has order-unity impact on CTF/FDF and Demo design

Systems code identifies optimal aspect ratio A=1.8-2 based on NHTX mission and design

- A=1.8-2 maximizes P/R and I_P (or $I_P \times A$) at fixed magnet power
  - Fixed HH_{98y2}=1.3, use $\kappa(A)$ and no-wall $\beta_N(A)$ scalings
  - I_P from BS and NBI – additional LHCD, ECCD/EBW to be assessed

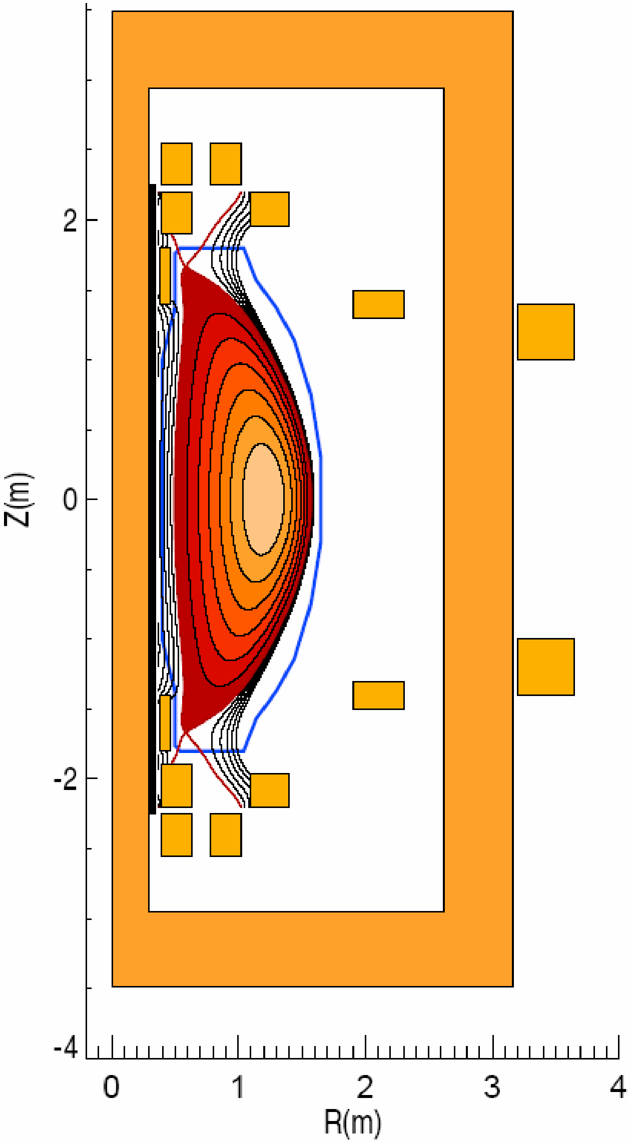


Overview of NHTX design progress

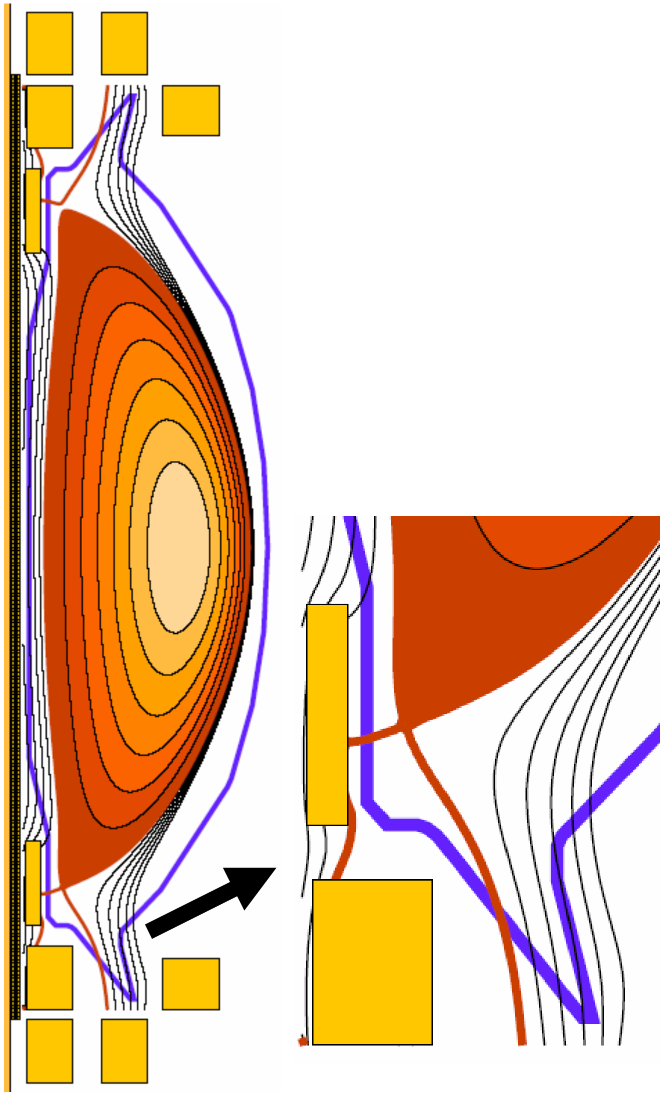
- Systems code has identified favorable design point:
  - A=1.8-2, R_0=1m, I_P=3-4MA, B_T=2T, κ=2.7-3, fully non-inductive
  - Maximizes I_P, $I_P \times A$, and P/R for given magnet power
  - HH_{98Y} = 1.3, β_N=4.5, β_T=15%, f_{BS}= 65%, f_{GW}=0.4-0.5
  - Higher β possible with Ω_ϕ & feedback stabilization of RWM
- Favorable PF coil configuration identified
  - Divertor flexibility without PF coil modification
  - Strong shaping flexibility (κ, δ, squareness, flux expansion)
  - Large midplane vertical gap for beam steering via ΔZ, and diagnostics
- NBI current drive efficiency & profiles studied with TRANSP
  - R_{TAN} and Z_{TAN} variations allow for J_{NBI} profile control
  - NBICD scalings used in systems code are reasonable

Single coil set supports range of divertor configurations

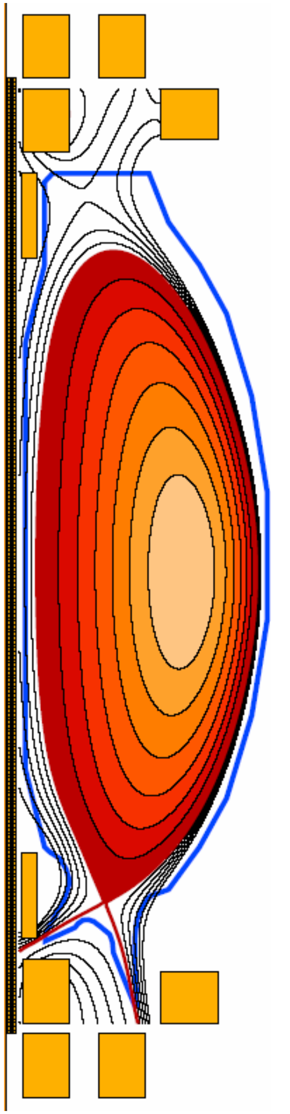
Open DN divertor



Pumped DND, JET-like

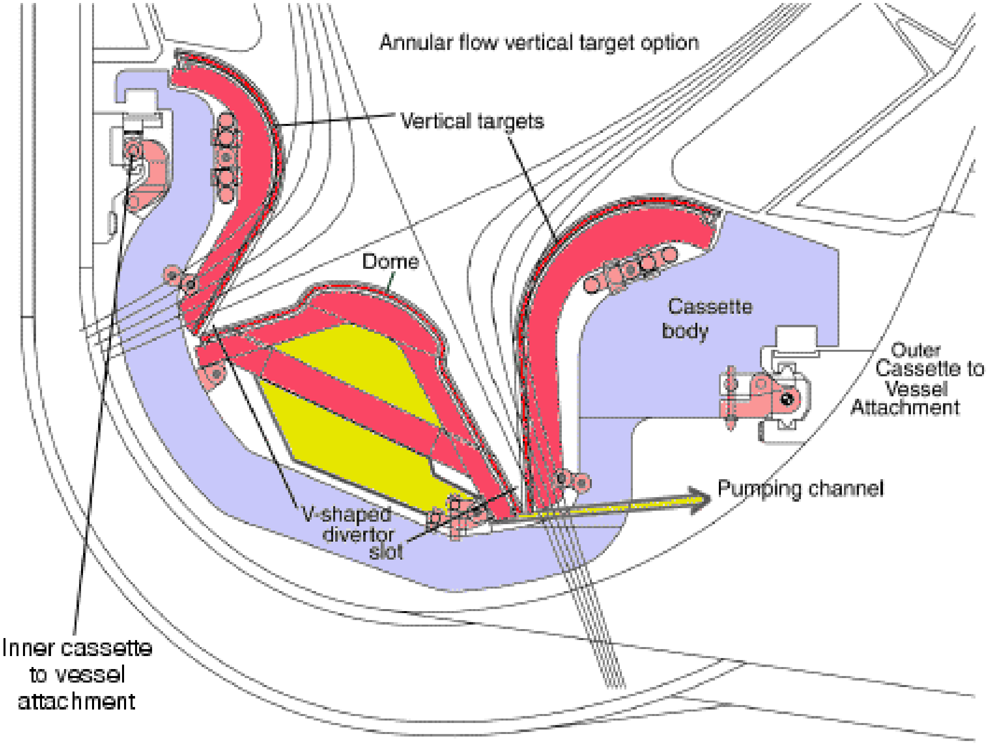


ITER-like LSN divertor

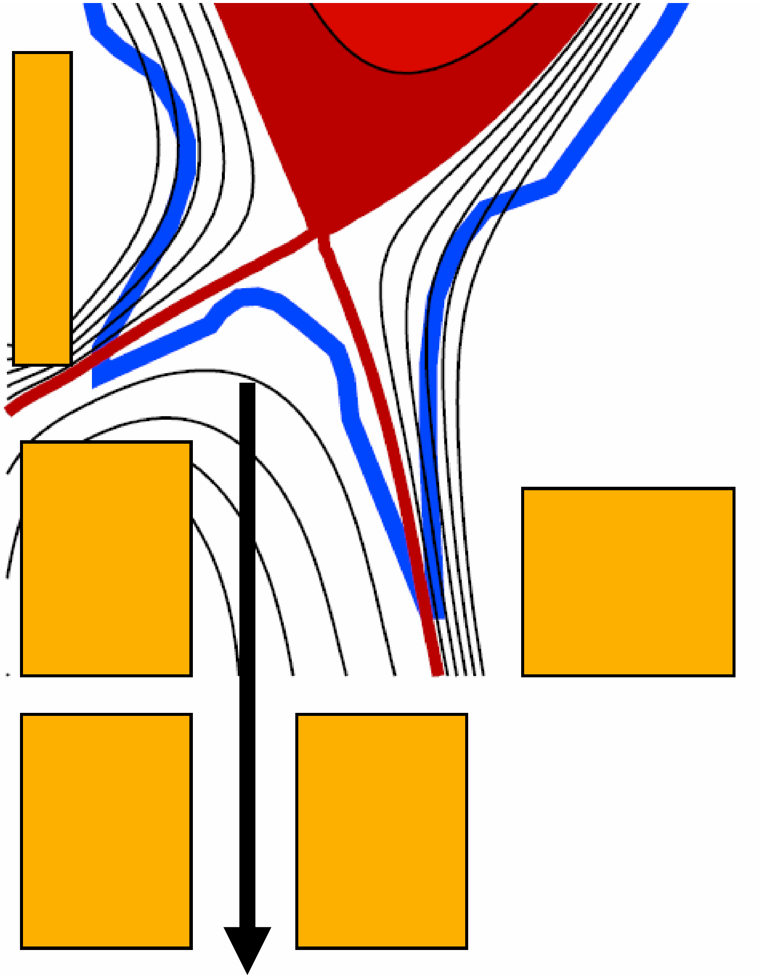


NHTX coil set supports ITER-like LSN divertor

ITER

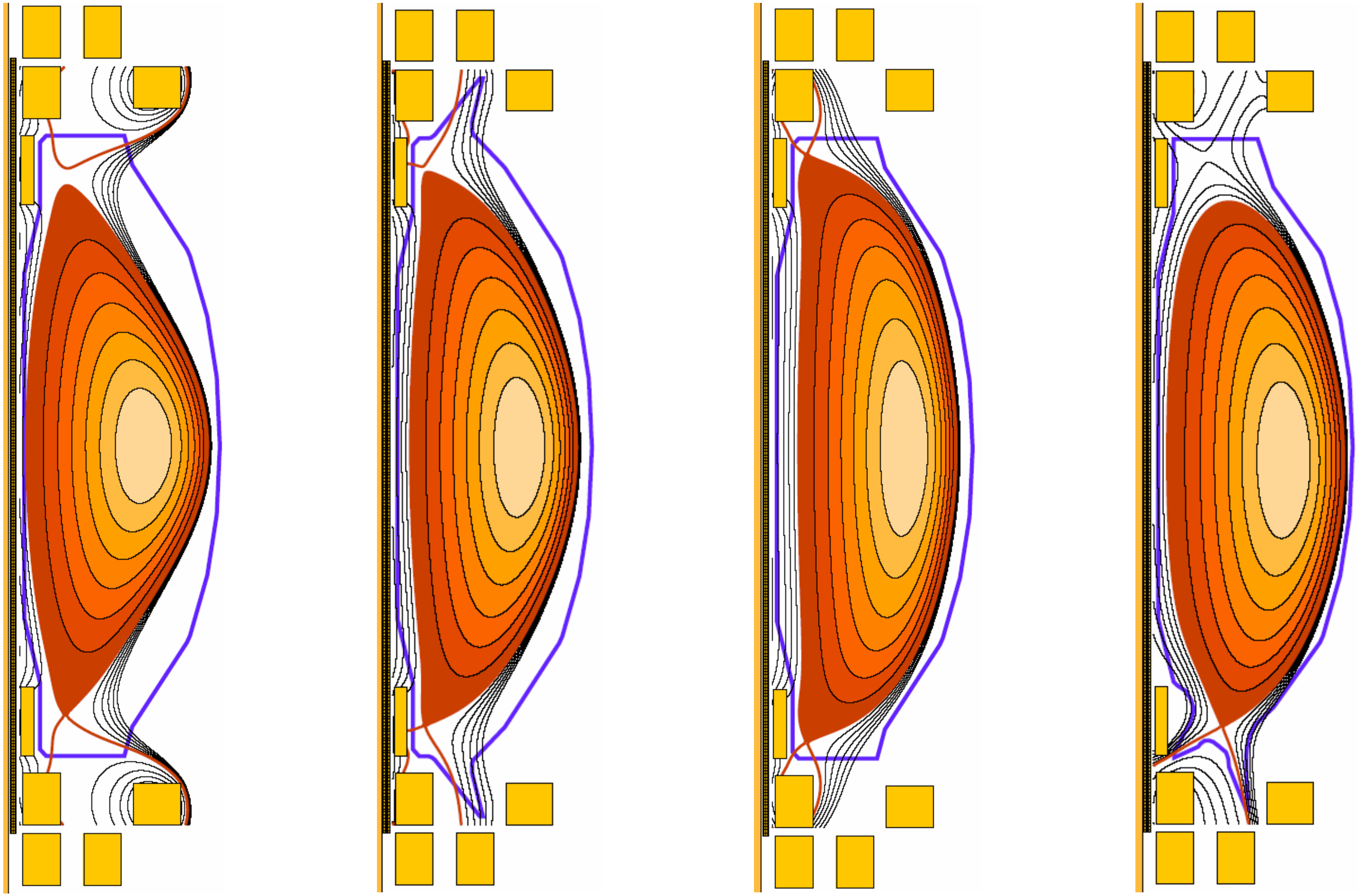


NHTX



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 $f_{exp} \equiv |\nabla\psi|_{\text{mid-plane}} / |\nabla\psi|_{\text{strike-point}}$

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

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

$f_{exp} = 2.8$

$\alpha_p=22^\circ$ $\alpha_t=5.1^\circ$

$f_{exp} = 9$

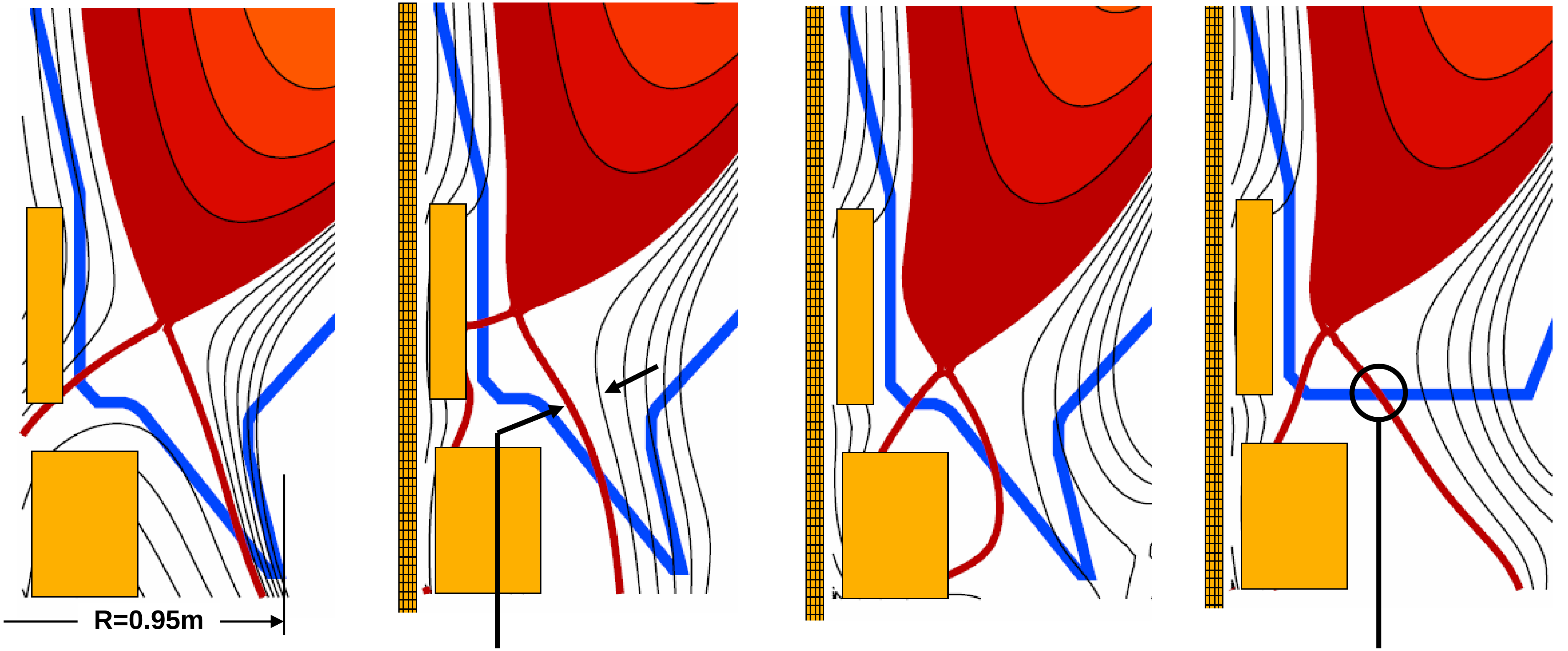
$\alpha_p=23^\circ$ $\alpha_t=1.8^\circ$

$f_{exp} = 17$

$\alpha_p=25^\circ$ $\alpha_t=1.0^\circ$

$f_{exp} = 35$

$\alpha_p=64^\circ$ $\alpha_t=1.1^\circ$



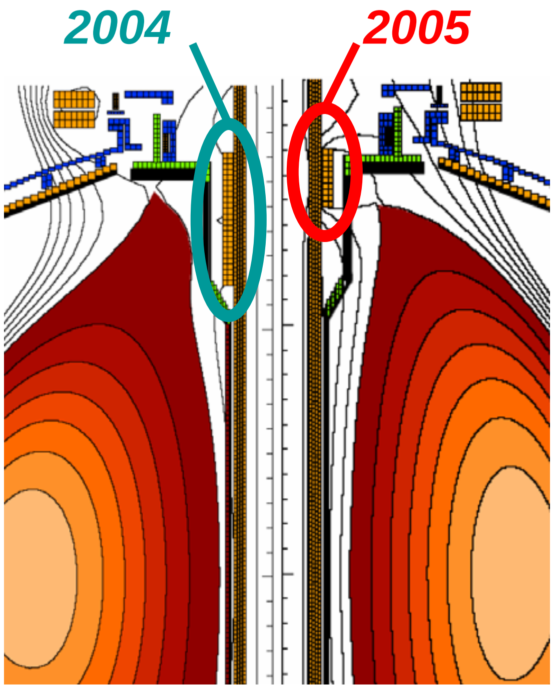
Flux contours have 5mm separation at midplane

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

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

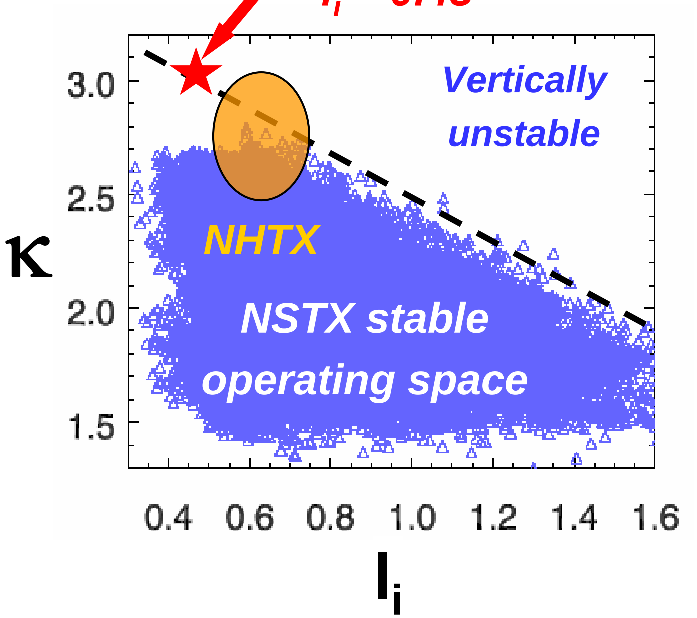
- NSTX: Sustained $\kappa \geq 2.8$ (reached $\kappa = 3$) for many τ_{WALL} using rtEFIT isoflux control
- High κ n=0 stability research important for NHTX and CTF/FDF design studies

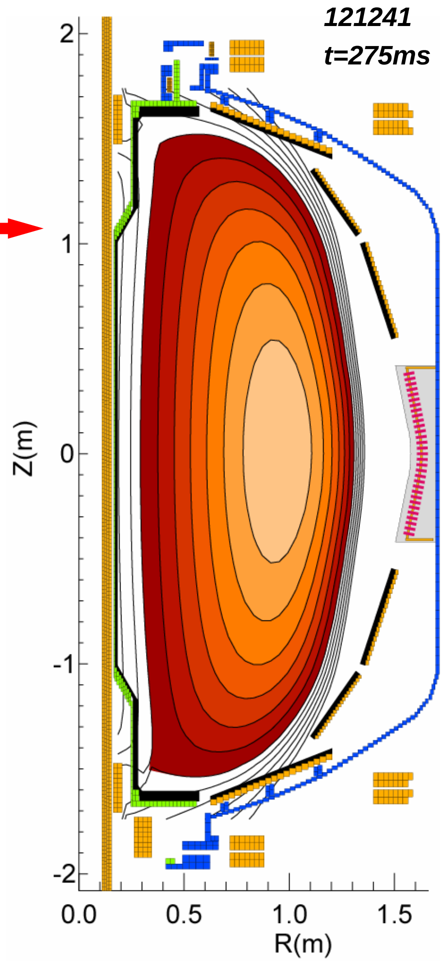
Divertor coil upgrade



2006: $\kappa = 3.0$, $\delta_X = 0.8$

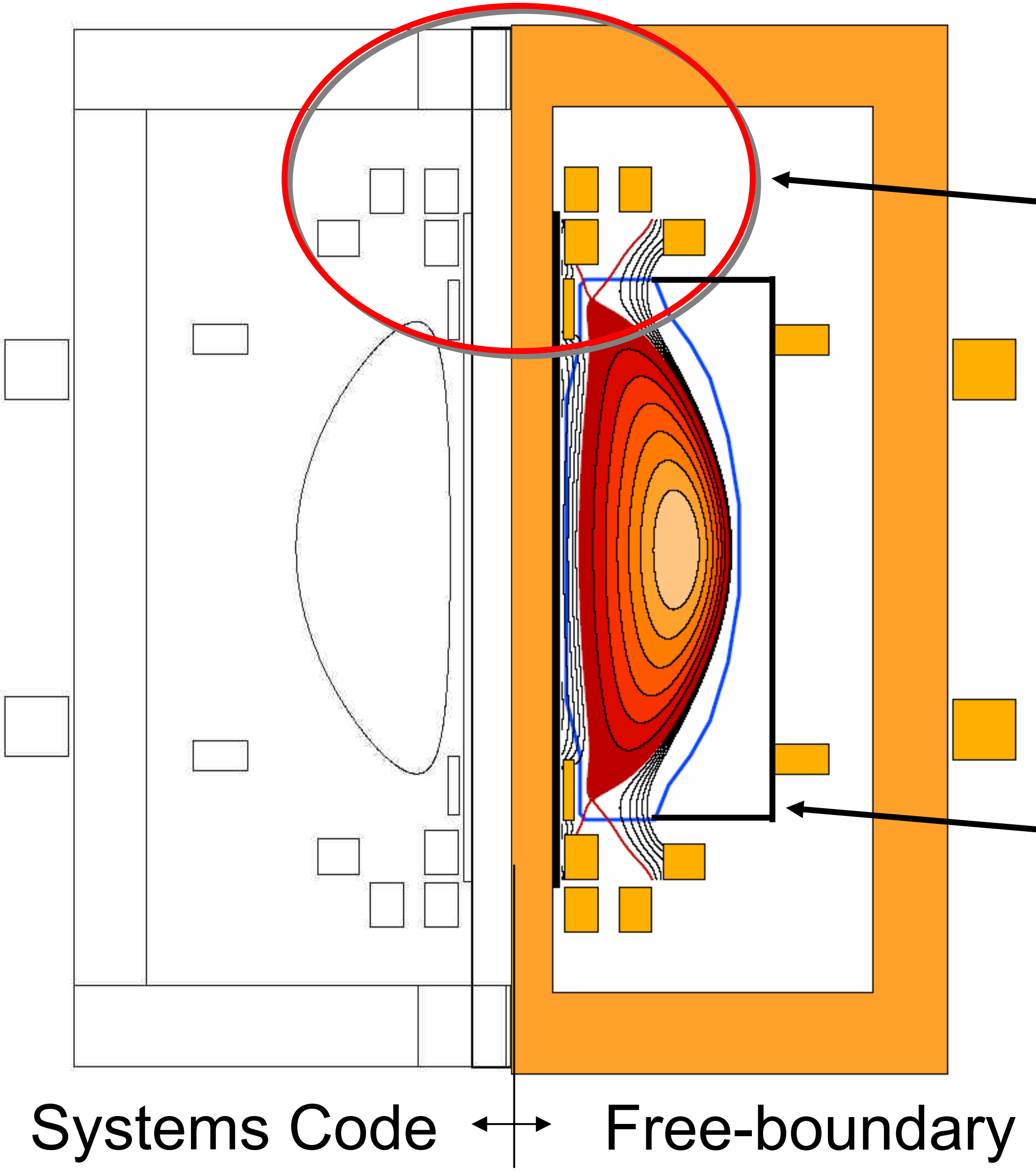
$l_i = 0.45$





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

Many engineering issues remain to be addressed



Critical area for physics and engineering design:

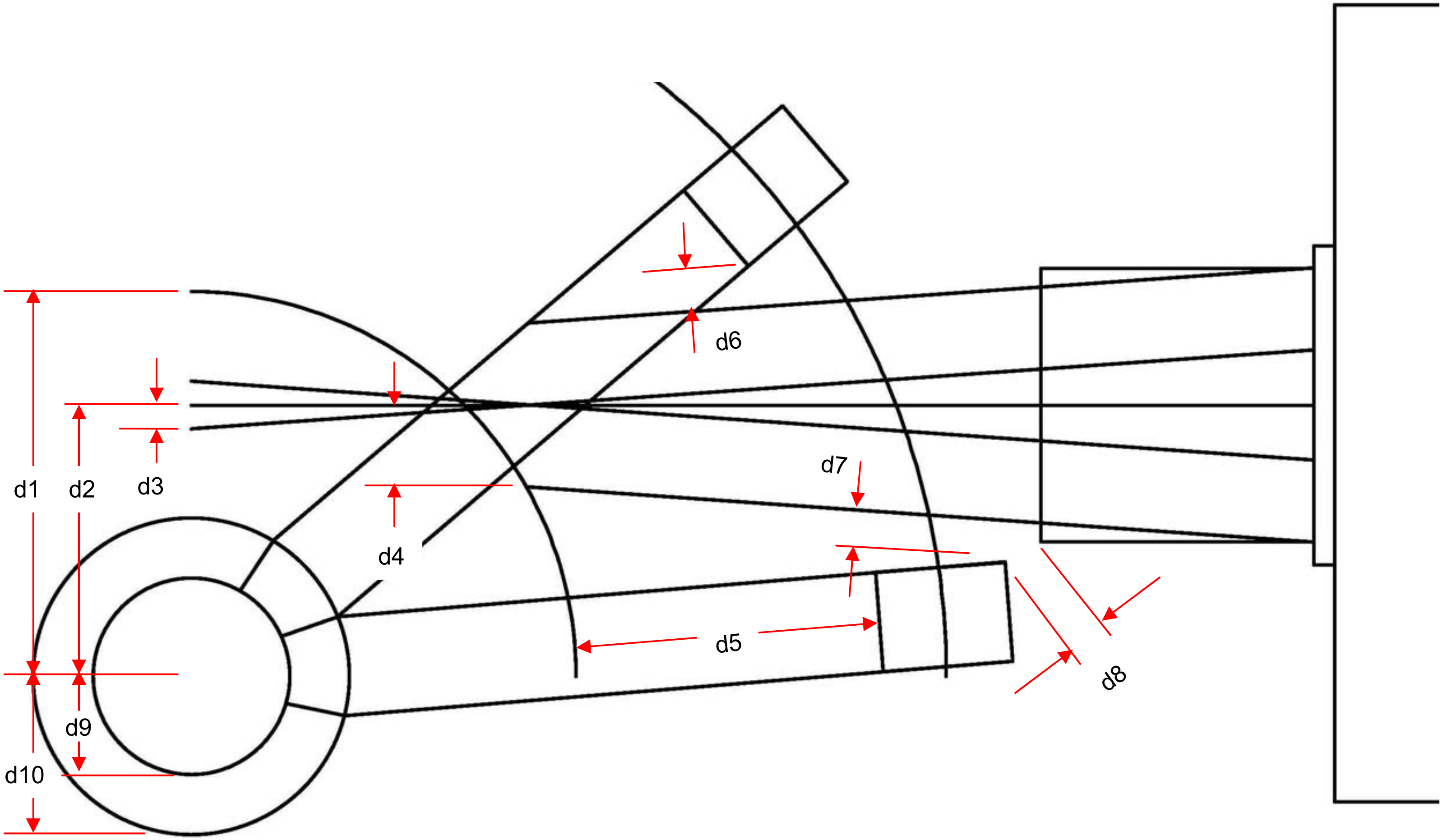
- Space for many divertor options
- Manifolds for heat removal
- B_{PF} & J_{PF} optimization
- TF jointing and structure

Cylindrical vacuum vessel has potential advantages:

- Facilitates top (crane) access
- Simplicity of fabrication

Systems Code ↔ Free-boundary equilibrium

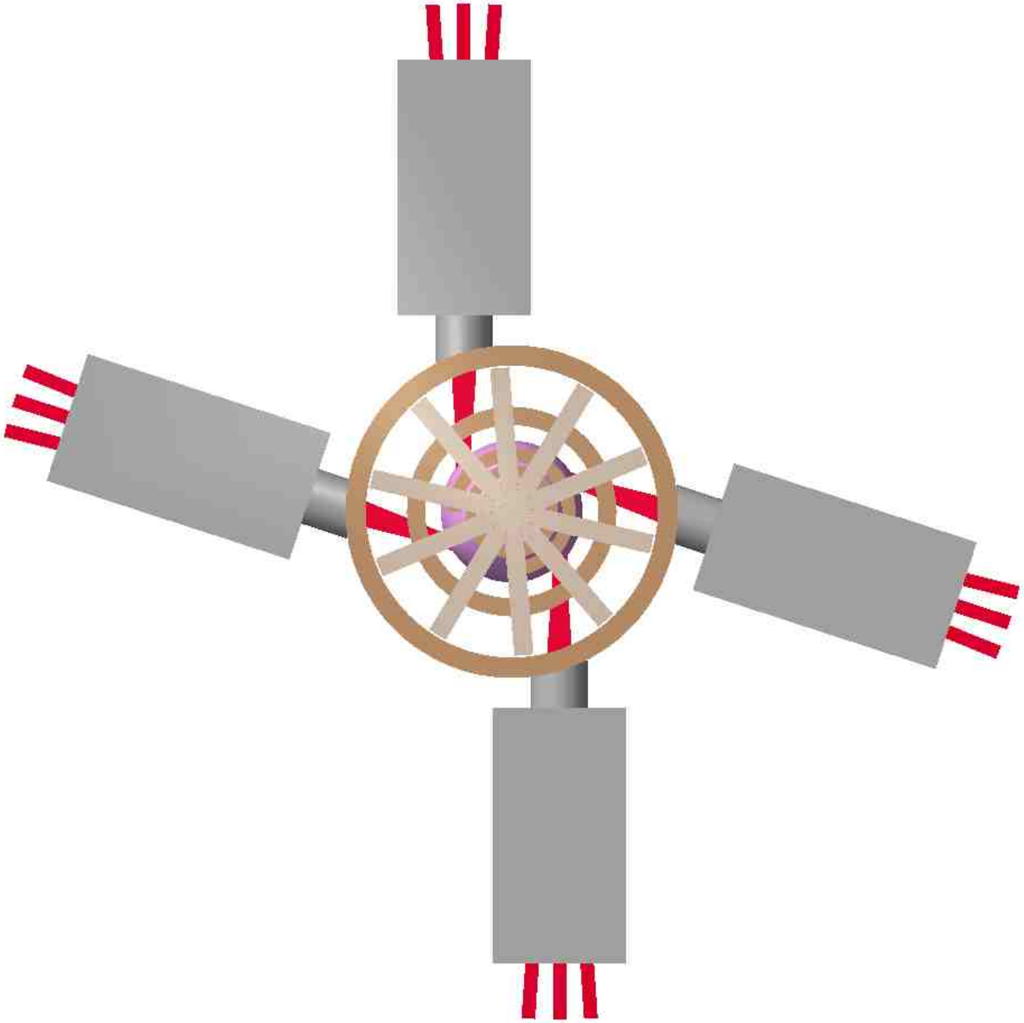
Systems code incorporates NBI geometry, TF ripple < 0.5%, and J_{TF} limits into TF outer leg layout and sizing



d1 = R0+a
d2 = R_tangency
d3 = beam centerline spacing around R_tangency
d4 = extent of beam duct w.r.t. beam centerline

d5 = gap R0+a to TF outer leg
d6, d7 = gaps TF outer legs to beam duct
d8 = gap TF outer leg to beam nozzle
d9 = radius of TF inner leg
d10 = radius of TF outer leg taper

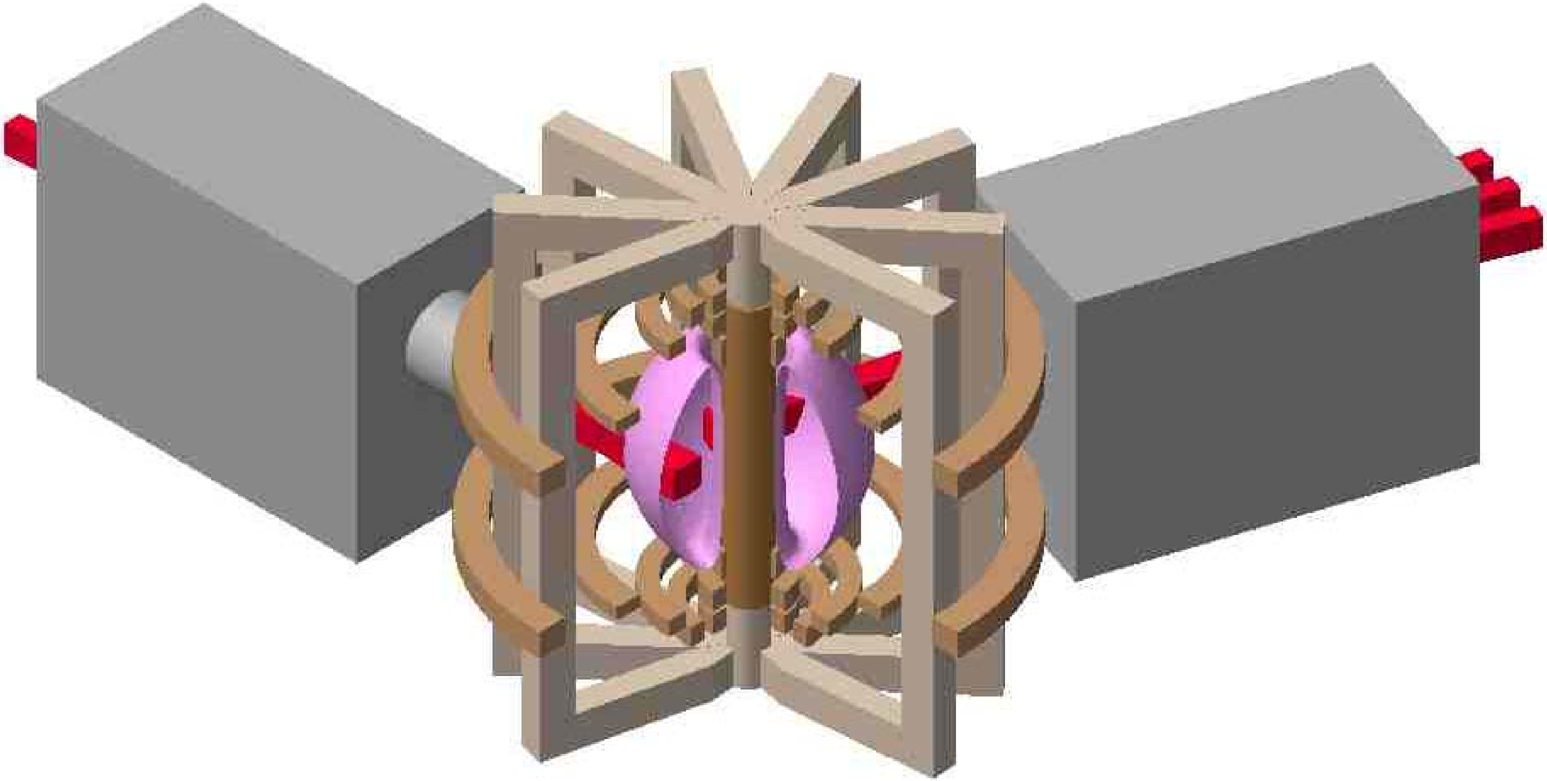
TF coil layout (10 coils) and sizing allows for R_{TAN} variation of NBI for J-profile control



- Assessing trade-offs between vertical shift and tangency radius variation
- Both provide broadened and/or off-axis current drive allowing J-profile control
- R_{TAN} variation from just inside R_0 to 30cm outside looks most favorable for CD

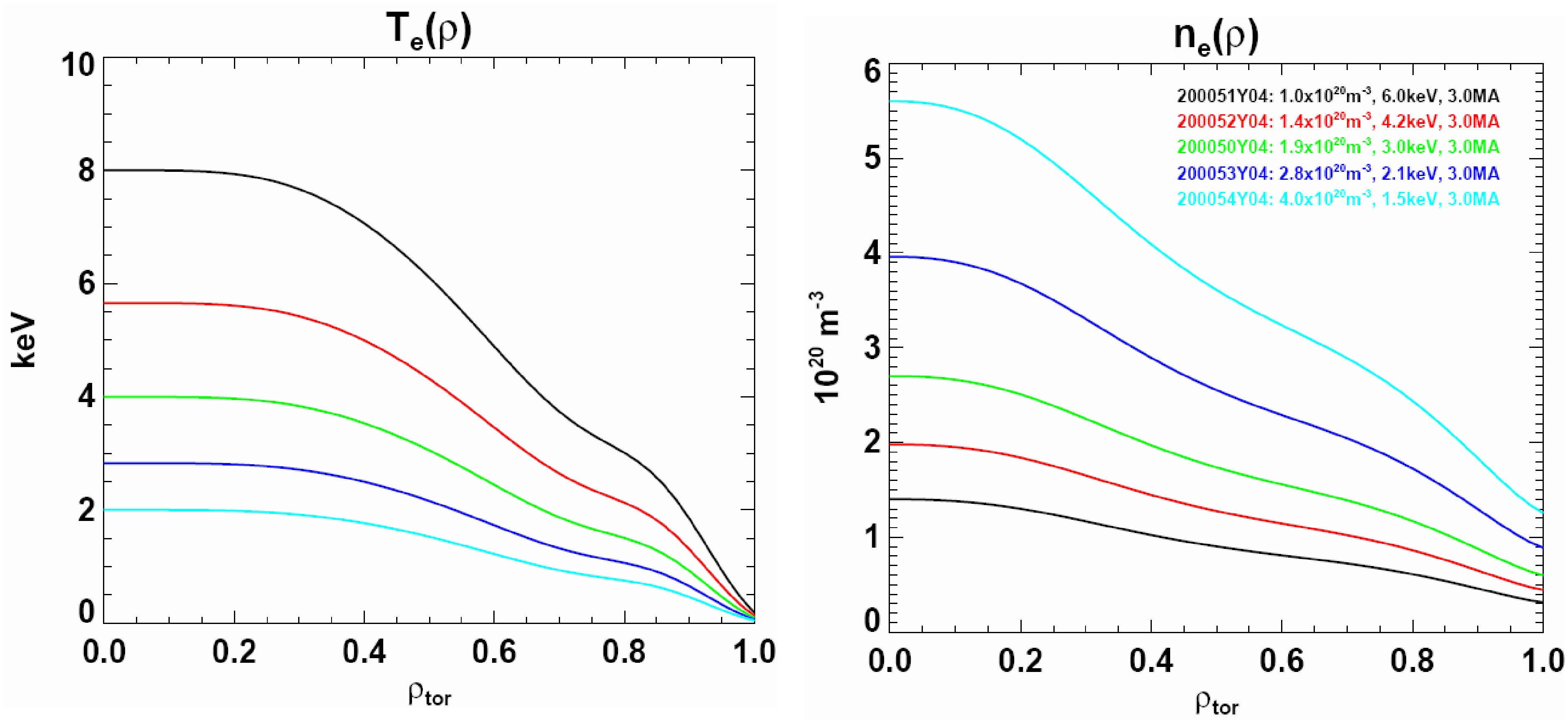
Large vertical gap between outer PF coils allows for vertical shifting of NBI for J-profile control

- High κ capability requires outer-most PFs to be outside TF
- If R_{TAN} variation is chosen, these PFs could have smaller R
  - Reduces PF power consumption, but…
  - Lose accessibility of large vertical midplane gap



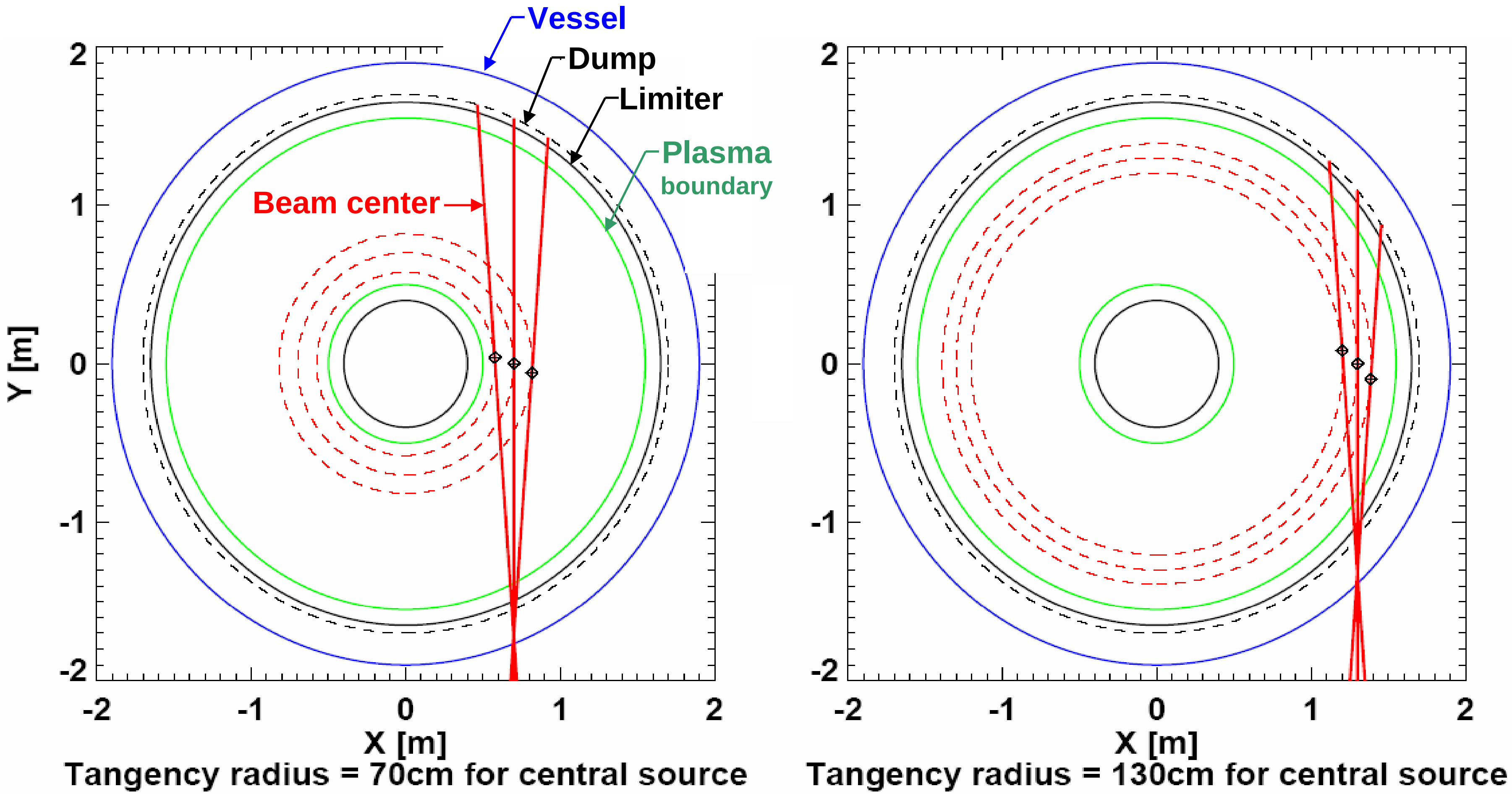
NBICD assessment w/ TRANSP uses thermal profile shapes based on high f_{NI} = 60-70% NSTX discharges

- Scale n_e, T_e profiles from 116313 - fixed T_i / T_e = 1.5, β_T=14%



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

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

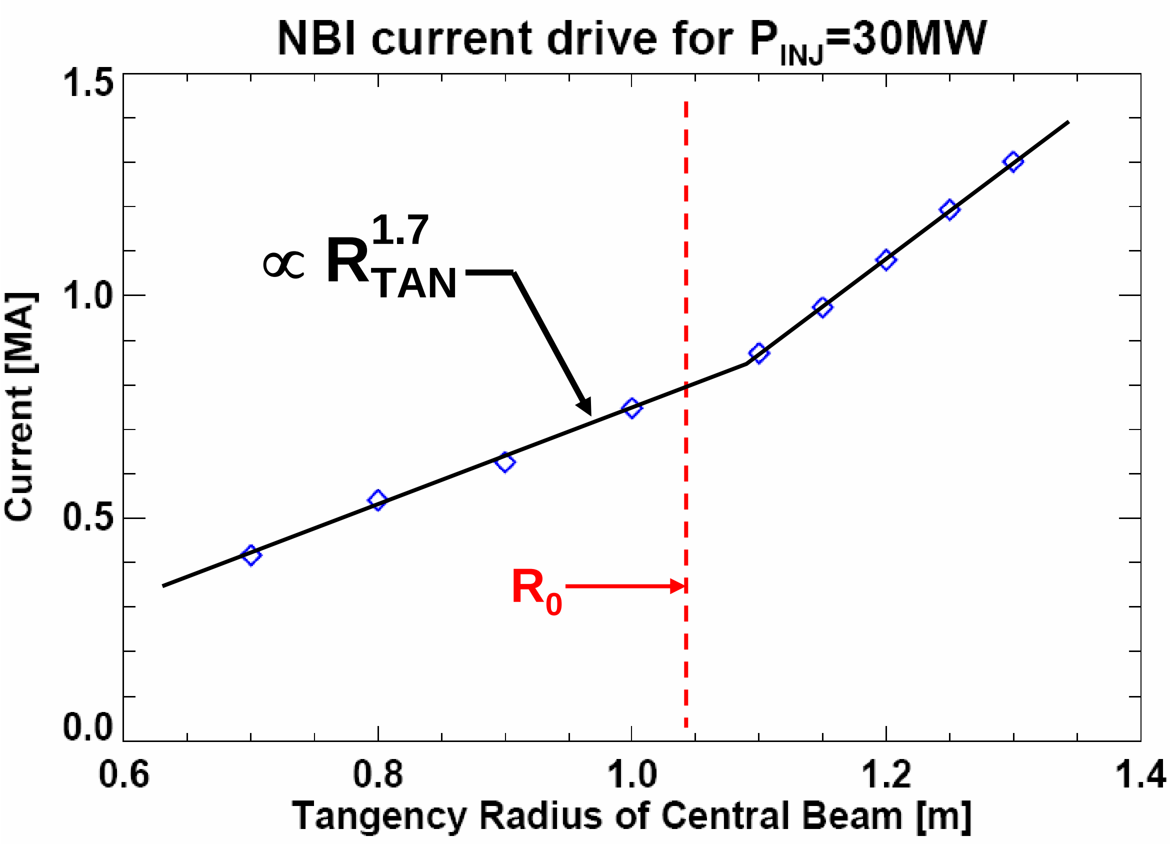


Tangency radius = 70cm for central source

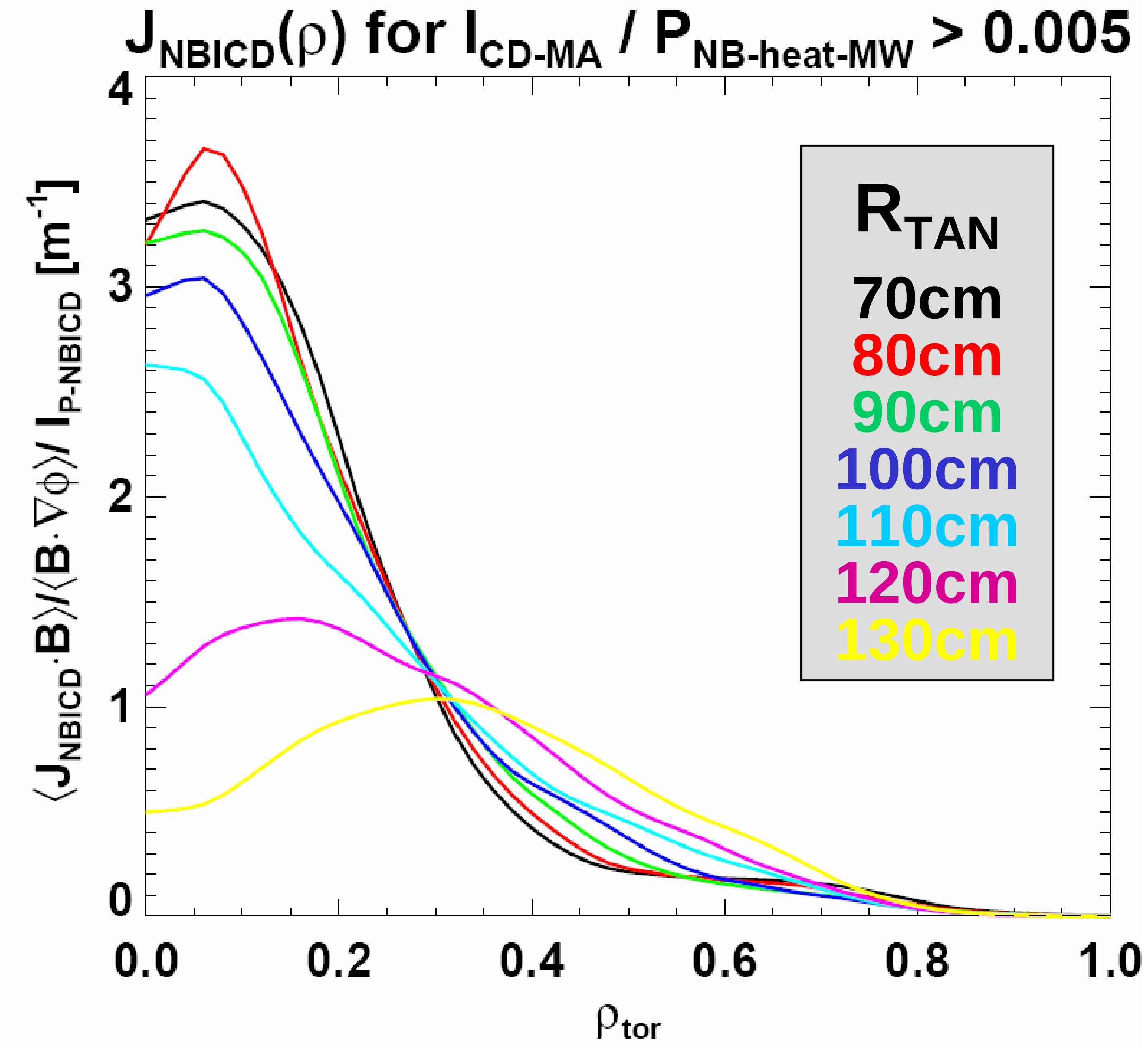
Tangency radius = 130cm for central source

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

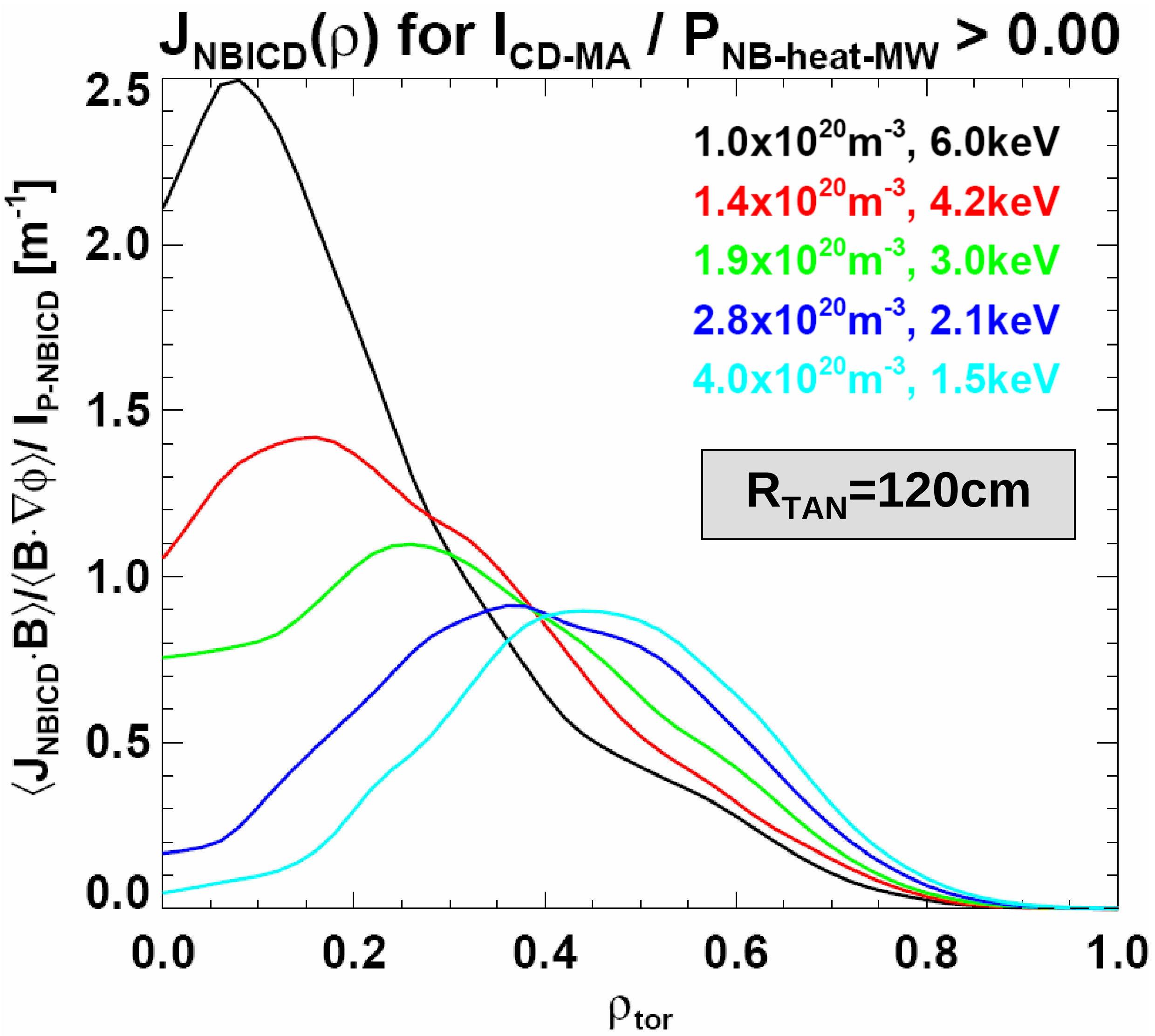
NBICD for $\overline{n}_e = 1.4\times10^{20}\text{m}^{-3}$, $\overline{T}_e$=4.2keV, f_{GW} = 0.43



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

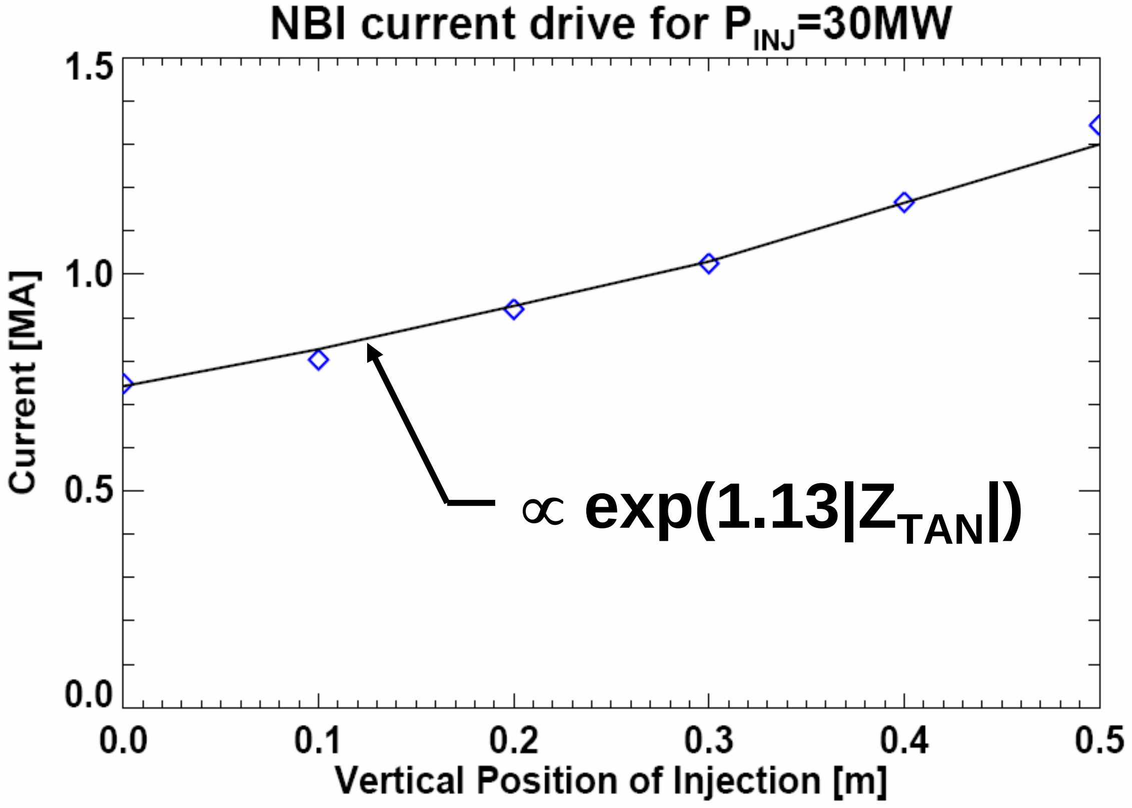


For outboard beam tangency radius, driven current profile broadens significantly at high density

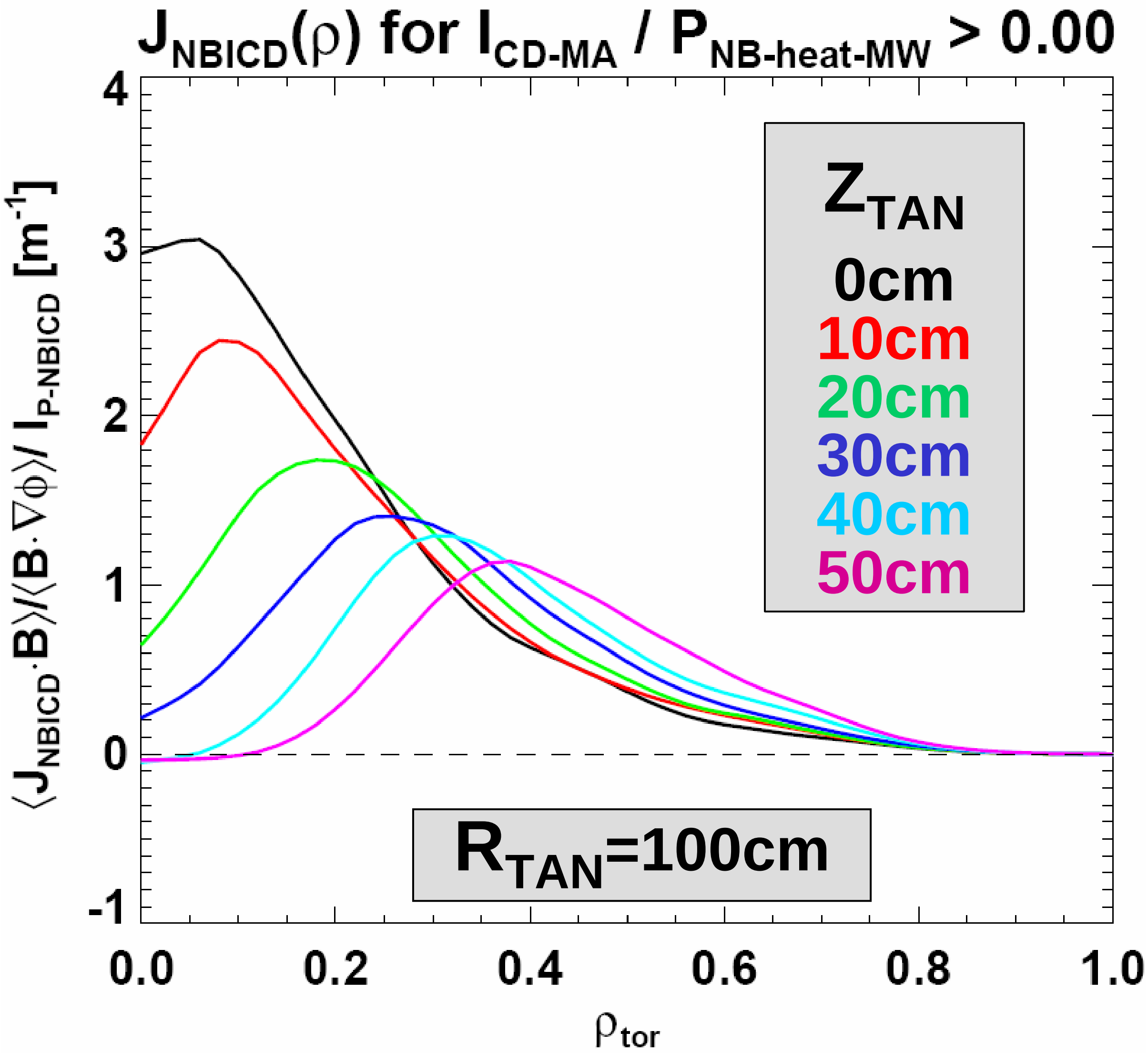


Driven current increases × 1.8 for Z_{TAN}=0.0 → 0.5m for R_{TAN} = 1.0m

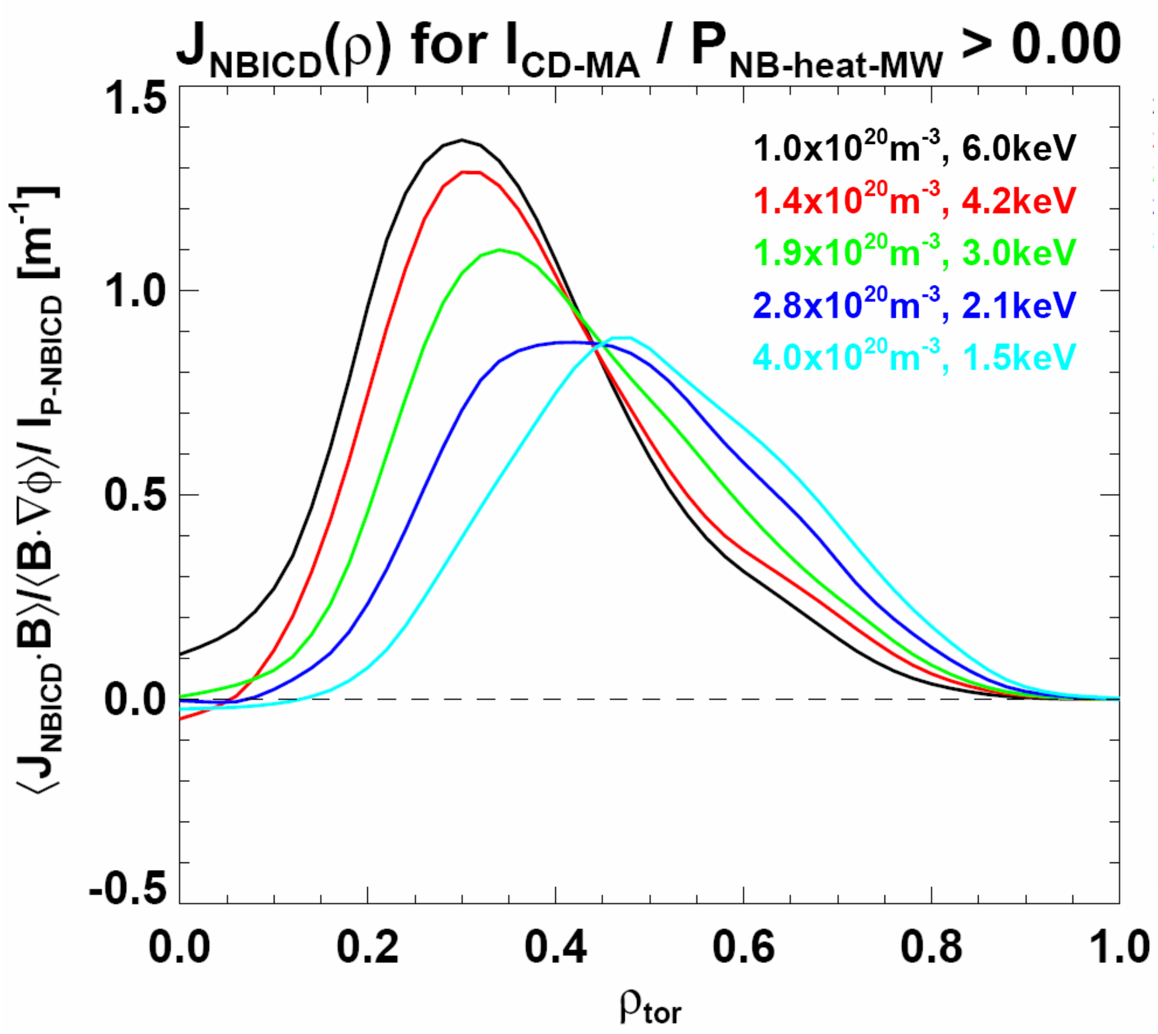
NBICD for $\overline{n}_e = 1.4\times10^{20}m^{-3}$, $\overline{T}_e$=4.2keV, f_{GW} = 0.43



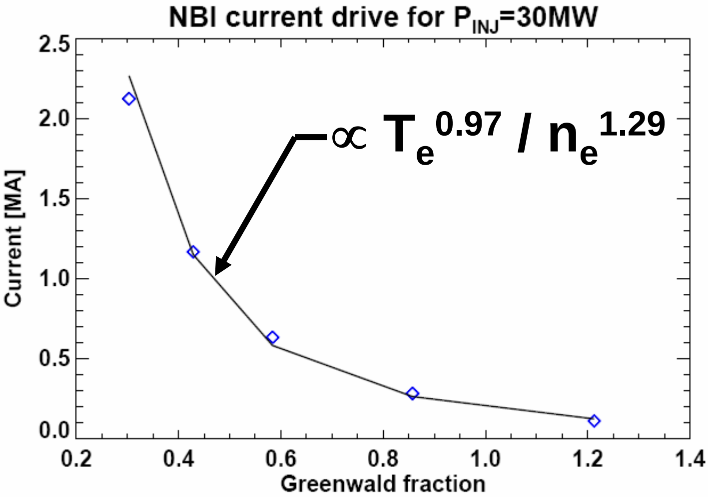
Beam vertical position (Z_{TAN}) variation would also enable control of core current and *q* profile



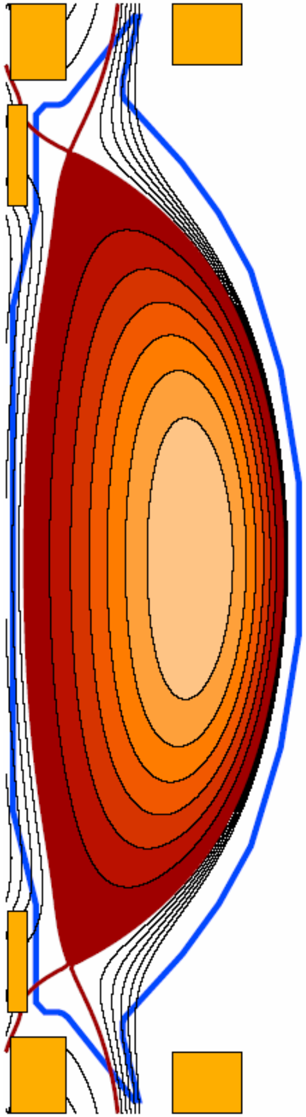
For vertically shifted beams, driven current profile shape remains hollow for all densities tested

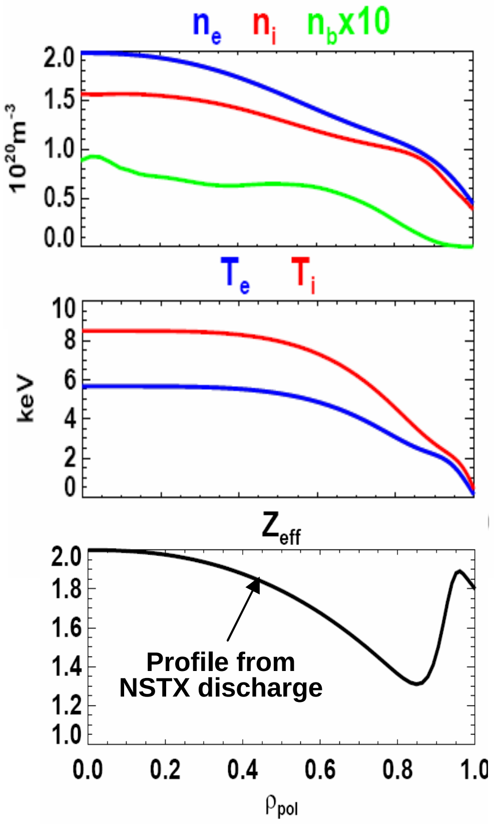


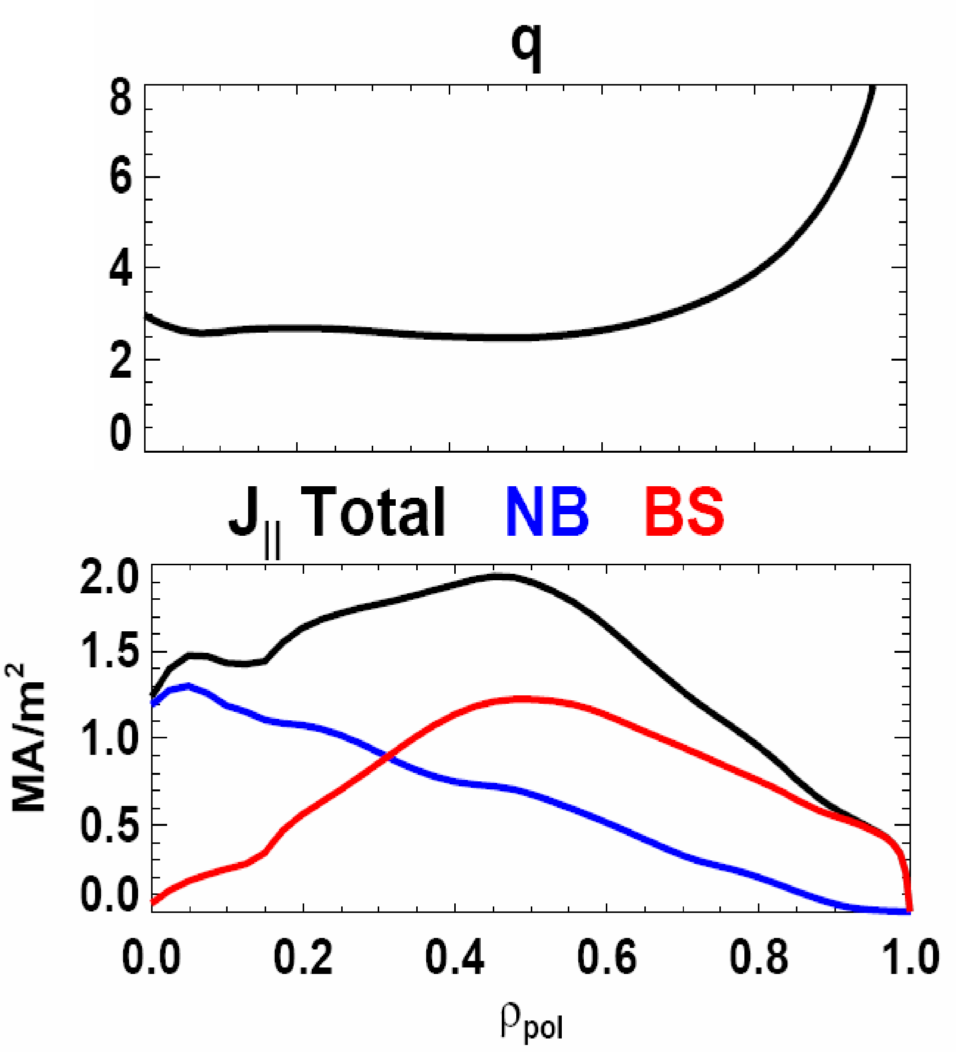
Z_{TAN}=40cm
R_{TAN}=100cm



$A=1.8$, $\kappa=2.85$, $I_P=3$MA target plasma with self-consistent $J(\rho)$ from NBI and BS with $q_{MIN} > 2.4$







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

Summary

- Systems code has identified favorable design point:
  - $A=1.8\text{-}2$, $R_0=1m$, $I_P=3\text{-}4MA$, $B_T=2T$, $\kappa=2.7\text{-}3$, full NICD
  - $HH_{98Y} = 1.3$, $\beta_N=4.5$, $\beta_T=15\%$, $f_{BS} \geq 65\%$, $f_{GW}=0.4\text{-}0.5$
  - Higher β possible with Ω_ϕ & feedback stabilization of RWM
- Favorable coil geometry found for maximum flexibility
  - Divertor flexibility critical element of NHTX mission
- NBI Z_{TAN} and R_{TAN} variations allow control of J_{NBICD}
  - Analyzing engineering tradeoffs of ΔR vs. ΔZ beam shift
- Beginning studies of additional heating & CD sources
  - Up to 18MW of additional RF power

Backup slides

Systems Code Method

- XL-based - uses non-linear optimizer ("Solver")
- Jardin/Kessel algorithms used for NSST were starting point for Systems Code
- Continued evolution with Peng, Rutherford, Kessel for CTF studies
    - See PPPL Report 4165 "Spherical Torus Design Point Studies"
- Engineering & physics algorithms tailored to suit NHTX

Physics Assumptions in Systems Code

A	1.5-3.0	100% flux surfaces
R0	0.9-1.0m	
kappa	3.674/SQRT(A)	Goldston
delta	0.6	Fixed
qcyl	4/3*(12.259-13.58*A+6.4286*A^2-1.0417*A^3)	Multiple of Menard
beta_N	<= limit 6.43-1.02*A	Fit to Menard no-wall limit
α_n=α_T	(0.64-0.3/A)/2	Menard model
peaking factor (pf)	ŗ(1-(r/a)^2)^α_n*(1-(r/a)^2)^α_T	
kBS	0.344+0.195*A	Menard model
fBS	Beta_P*kBS*pf^0.25/SQRT(A)	
Confinement	Ti=Te, HH98=1.3	Also examined Ti .ne. Te w/HHe=0.7-1.3
Solenoid Flux	85% Hirshman-Neilson flux, ramp-up only	85% factor matches formula to Menard data
Non-inductive CD	Bootstrap + NBI (4*8=32MW) @ 110keV	
Paux	32MW (NBI) + 6MW (RF) = 38MW	Beta limited
NBI alignment	Normalized to 90,100,110 cm tangency for R0=0.95m, A=1.8 case	Kaye
PF Currents	Amp-turns scaled from Menard equilibrium @ 3MA (A=1.8)	

Engineering Assumptions in Systems Code

TF Inner Leg Heating	Jcu_avg <= 5.75kA/cm^2	dz=kappa*a+1.425m, packing fraction f(Jcu_avg,dZ) based on KCOOL, v=10m/s, Tcu_max=100C
TF Inner Leg Stress	Radial stress <=138MPA	Insulation shear stress is tracked
TF Outer Leg Heating	Minimize J but maximizing CSA of outer legs within available space, considering NBI alignment	
TF Outer Leg Stress	Not Modeled	
OH Heating	G-function adiabatic	dz=f(kappa*a)
OH Stress	Hoop stress <=138MPA	
PF Heating	Jcu_avg <= 2.5kA/cm^2	KCOOL analysis assumes conductor area per turn 1.5*CSA of existing PF coils, 10 turns per cooling path, 15kA per turn
PF Stress	Not Modeled	
Center Stack Casing (VV) Heating and Radial Build	25% of Paux impinges on CS over dZ=2*kappa*a	Radial build based on heat flux, ferritic steel w/15% cooling fraction, 400C, 4MPa He cooling at 150m/s
PFC Heating	Not Modeled	
PFC Stress	Not Modeled	
Transrex Capacity	15kA/PSS, 3.25kA rms	Irms is limiting (Trep~20min)
MG	TF/PF/OH Loads W<=4.5GJ, CCV on during pulse	
Grid	NBI/MG/BOP Loads P<=200MW	Approved by PSE&G for TPX, requires local D-site substation and p.f. correction
Cooling Water Systems	Total flow requirement based on total energy dissipation, rep rate limited by 20MW heat removal	60-10=50C rise typ. deltaT