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Physics design of NHTX

National High-power advanced Torus eXperiment

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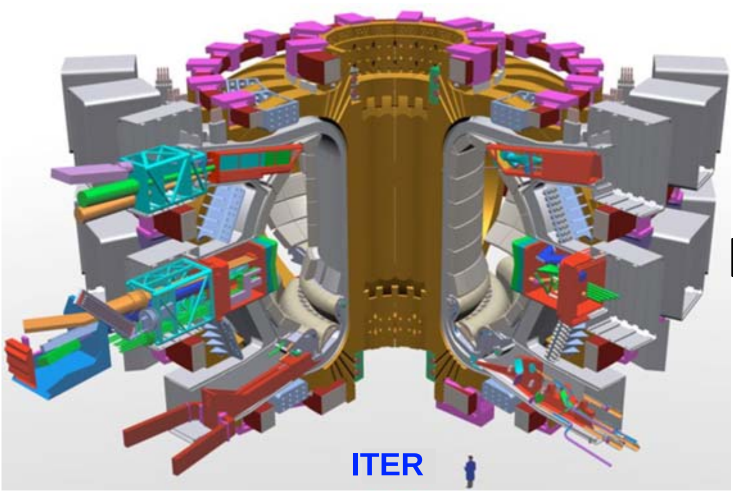
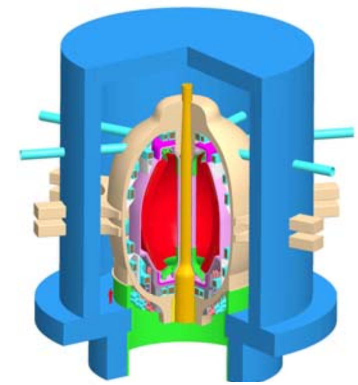
U.S. fusion program beginning an assessment of what concepts and initiatives are needed to extrapolate from ITER to Demo

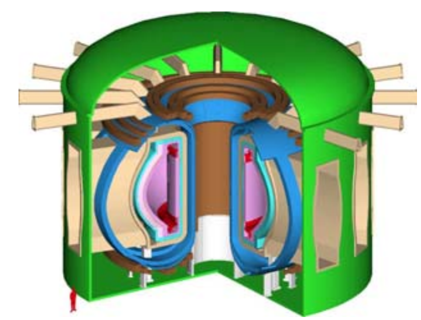


Next Step for U.S.?

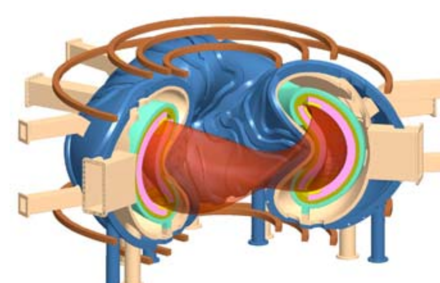
Simpler Magnets



ARIES-ST



ARIES-AT



ARIES-CS

Simpler Sustainment

- A few critical issues to be resolved for Demo:
  - Plasma-material interface - heat flux, T retention, etc.
  - Steady-state high-performance operation

This presentation will focus on a concept to develop integrated solutions to these 2 challenges

Existing plasma-material interface concepts are marginal for ITER, and are unacceptable for CTF/FDF and Demo

- High-heat-flux challenge
  - ITER divertor and first-wall marginal even without off-normal events
    - No demonstrated heat flux solution (at high plasma performance) for CTF/FDF and Demo
  - ELMs & disruptions can ablate/melt divertor, threaten first-wall & blankets in ITER
    - Disruptions & ELMs unacceptable for CTF/FDF and Demo
- Tritium retention challenge
  - Carbon erosion and re-deposition → up to 50:50 mix of C & DT in surface films
    - Erosion and neutron damage → Carbon unacceptable for CTF/FDF or Demo
  - Safety concerns limit ITER in-vessel mobilizable T inventory to < 350g
    - < 1000mins of accumulated ITER ops before limit is reached with only 3% retention
    - Potentially acceptable for ITER → but need to develop new clean-up techniques
    - Few % retention rate unacceptable for week/month long CTF/FDF or Demo operation
  - Tungsten or flowing lithium might reduce T retention to acceptable levels, but…
    - W can melt during ELMs & disruptions, sputtered mid-Z impurities, dust formation
    - High liquid metal vapor pressure at high temperature could pollute plasma

- These challenges motivate new research device with following capabilities:
  - Multiple divertor/PFC concepts, hot wall 1000°C, T operation, long pulse < 1000s
  - High beta, high confinement, fully non-inductive operation

Measured power scrape-off width independent of machine size → P/R is useful divertor heat-flux metric for comparing devices

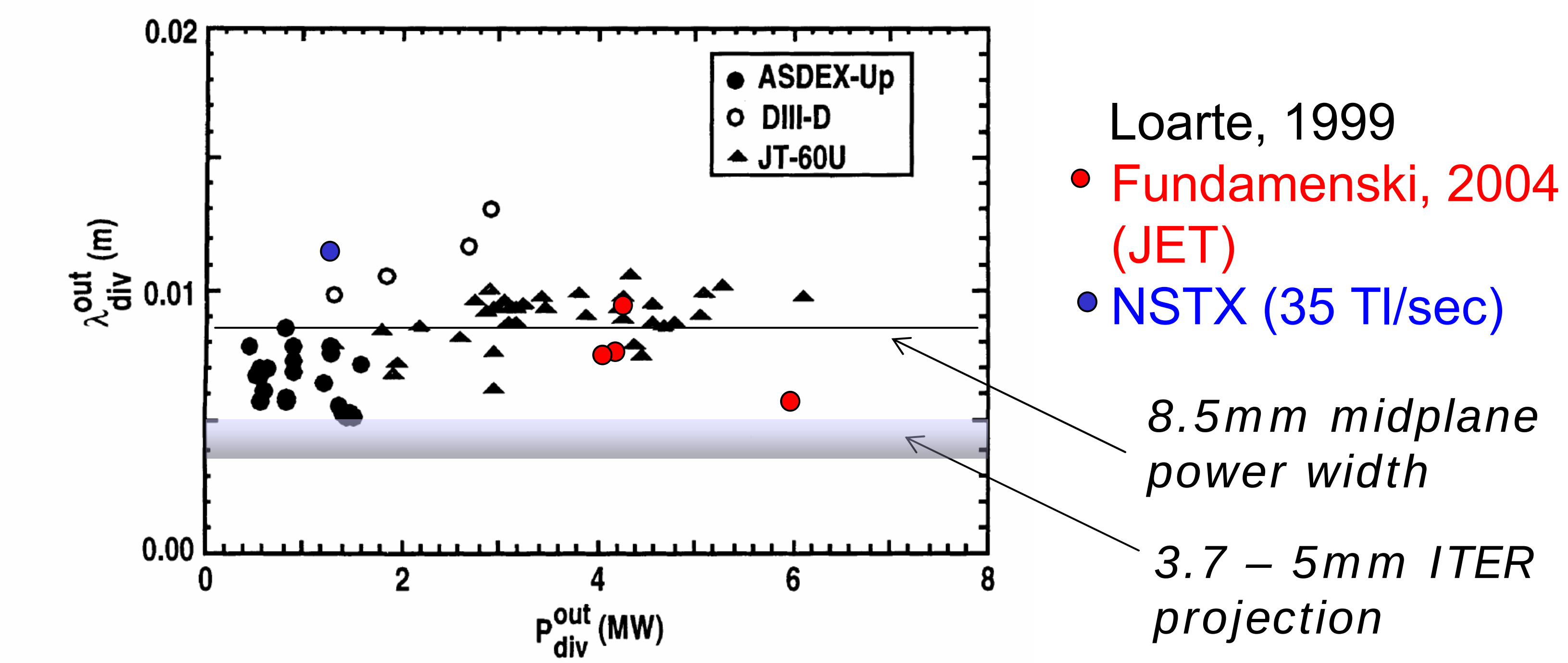


Fig. 5. Measured power deposition width versus divertor power for H-mode discharges without gas puff in the ITER power deposition database. (Mapped from strike point to outer mid-plane.)

First wall heat-flux challenge → P/S

NHTX can address integrated fusion science mission at heat-flux level of CTF/FDF, and extrapolates to Demo reactor heat-flux

Device	R	a	P_{in}	P_{in} / R	P_{in} / S	Pulse	I_p	Species	Comments
	(m)	(m)	(MW)	(MW/m)	(MW/m^2)	(sec)	(MA)		
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
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**	**200-1000**	**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	0.87	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

*** Flux compression, low R_X/R, SND, additional P_{AUX} → can achieve Demo level heat-fluxes**

The Integrated Fusion Science Mission of NHTX

National High-power advanced Torus eXperiment

To integrate a fusion-relevant plasma-material interface
with sustained high-performance plasma operation

NHTX will have the flexibility to study:

- **Multiple divertor geometries**
- **Tritium retention and high-T PFCs**
- **Multiple advanced solid materials**
- **Liquid surfaces**
- **Stellarator-like edge magnetic field**
- **Magnetically expanded strike zone**
- **Radiative edge zone**
- **Multiple plasma heating technologies**
- **INTEGRATED WITH A HIGH-PERFORMANCE PLASMA**

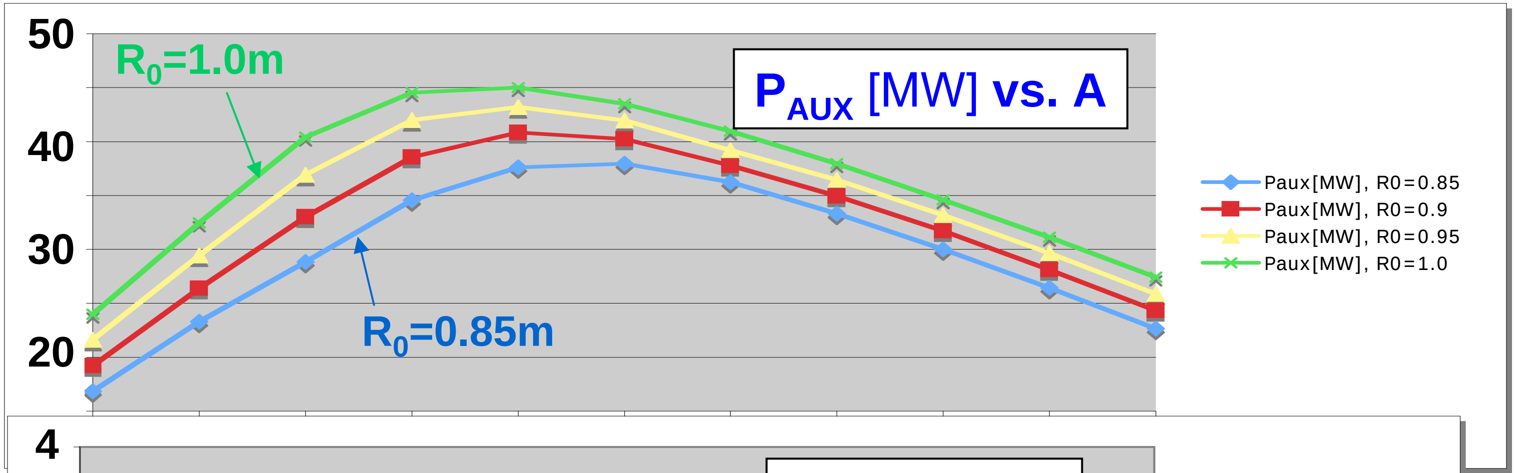
FTU Lithium Capillary Porous System (CPS)

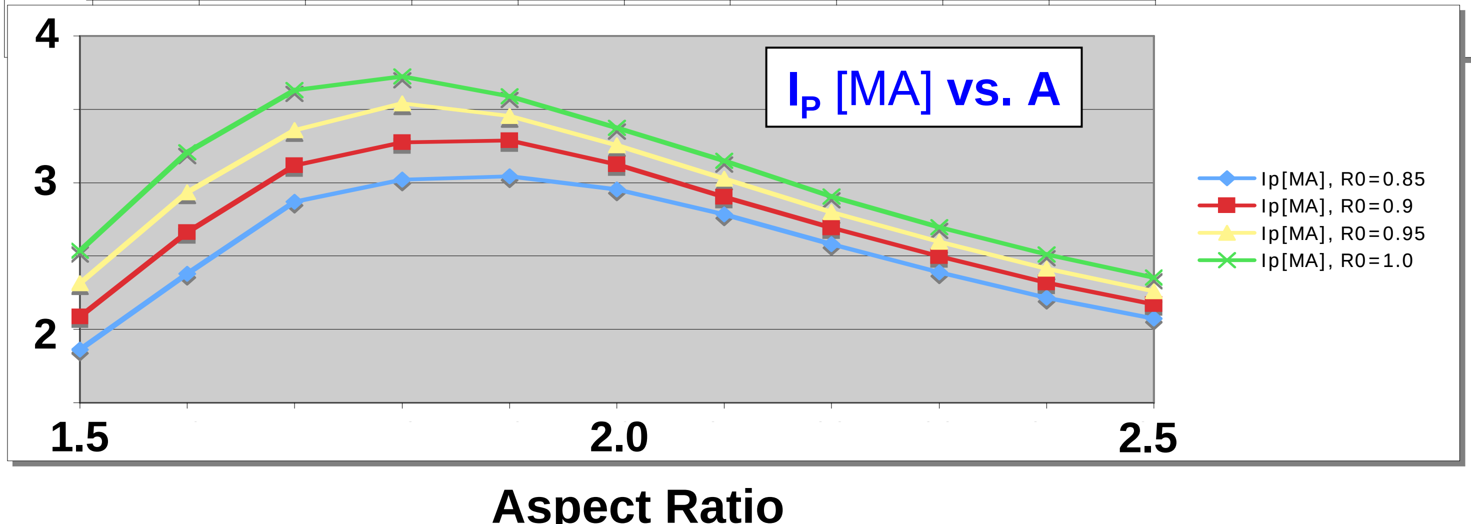
Such a device would:

Develop innovations **needed for integrated core and boundary science for later phases of ITER, for CTF/FDF, and for a Demo power plant – whether Tokamak, ST or Compact Stellarator.**

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 κ(A) and **n=1 no-wall limit** β_N(A) scalings
  - I_P from BS and NBI – additional LHCD, ECCD/EBW to be assessed





NHTX has uniquely high P_{in} / $P_{L\text{-}H}$ > 10 needed to test radiative solutions at f_{rad} > 90% for Demo

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

- Is high radiated power fraction to reduce divertor heat flux compatible with high performance?

- Is thermal instability problematic in burning plasma at high radiation fraction?

NHTX Heating and Current Drive

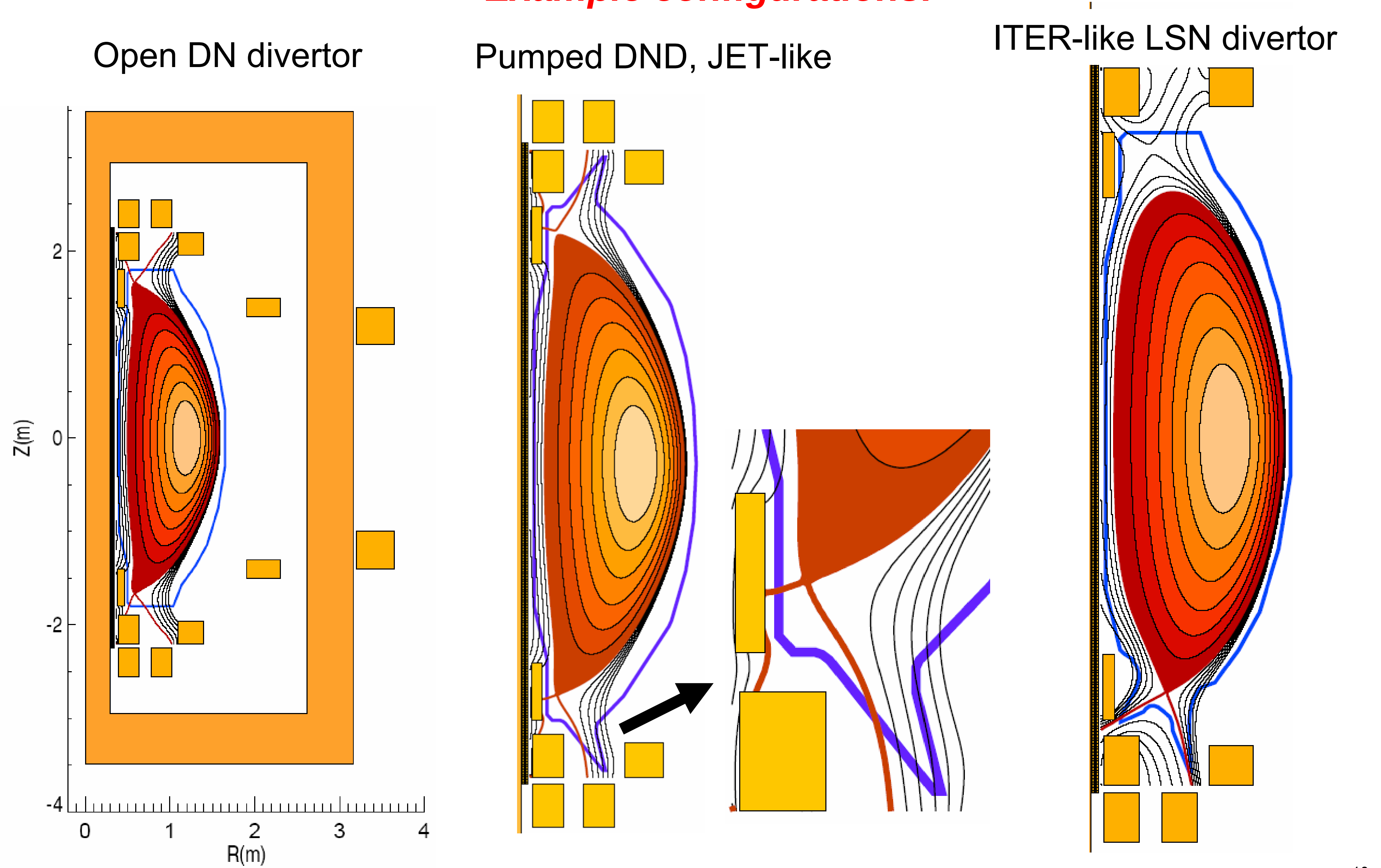
- Total auxiliary heating and current drive power = 50MW
  - Neutral beams: 32 MW, 110 kV D_0 NBI, steerable off axis
  - 18 MW RF – type to be determined
- Results from NSTX, C-MOD, DIII-D will be critical to selection of RF system(s)
  - EBWCD: High efficiency, remote coupling.
  - LHCD: High efficiency, intimate coupling.
  - ECCD: Inside-launch 120 GHz 2nd harmonic: lower efficiency, more complex access.
  - ICRF: Cost-effective electron or ion heating, 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)

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**
  - **HH_{98Y} = 1.3, β_N=4.5, β_T=15%, f_{BS}= 65%, f_{GW}=0.4-0.5**
  - Maximizes I_P, $I_P \times A$, and P/R for given magnet power
  - High 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 (ΔZ), diagnostics, access
- 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

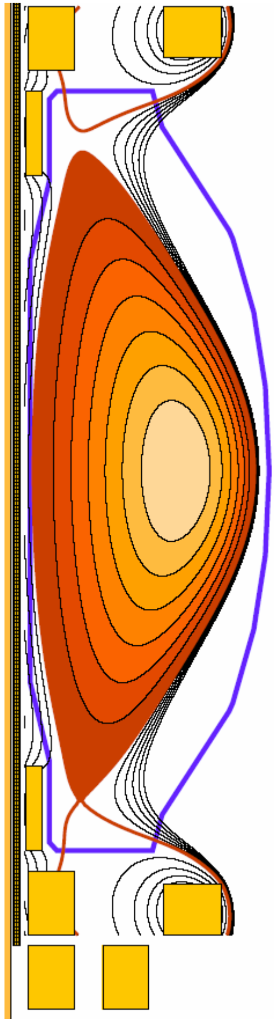
Example configurations:



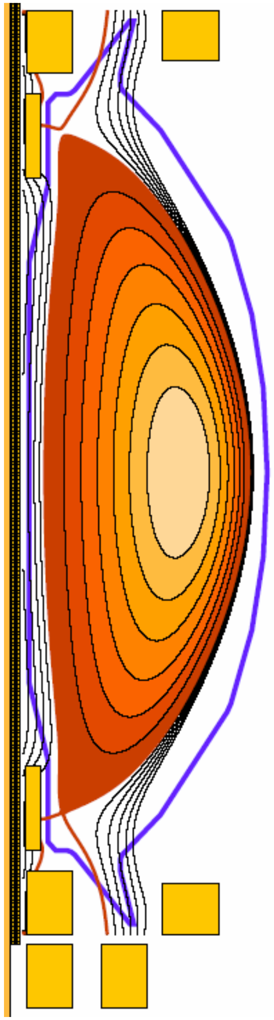
Coil set supports wide range of boundary shapes

- Shaping plays important role in determining global and ELM stability

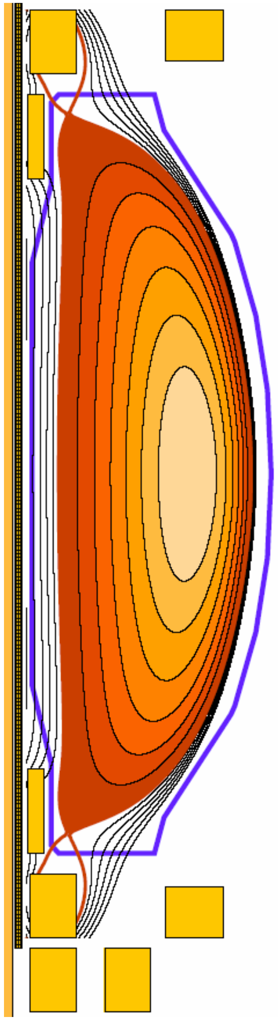
DND w/ negative squareness $\zeta \approx -0.15$



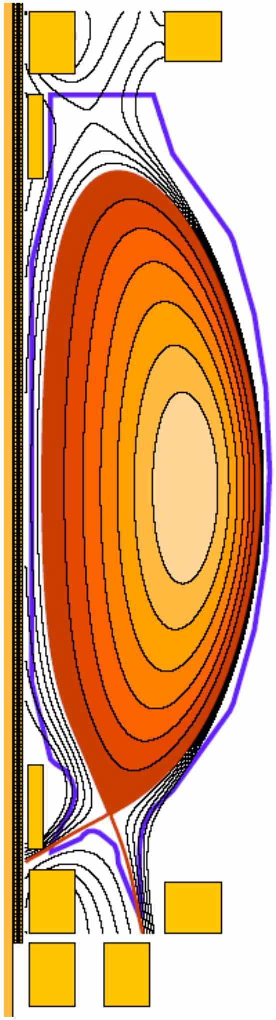
DND w/ near zero squareness



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



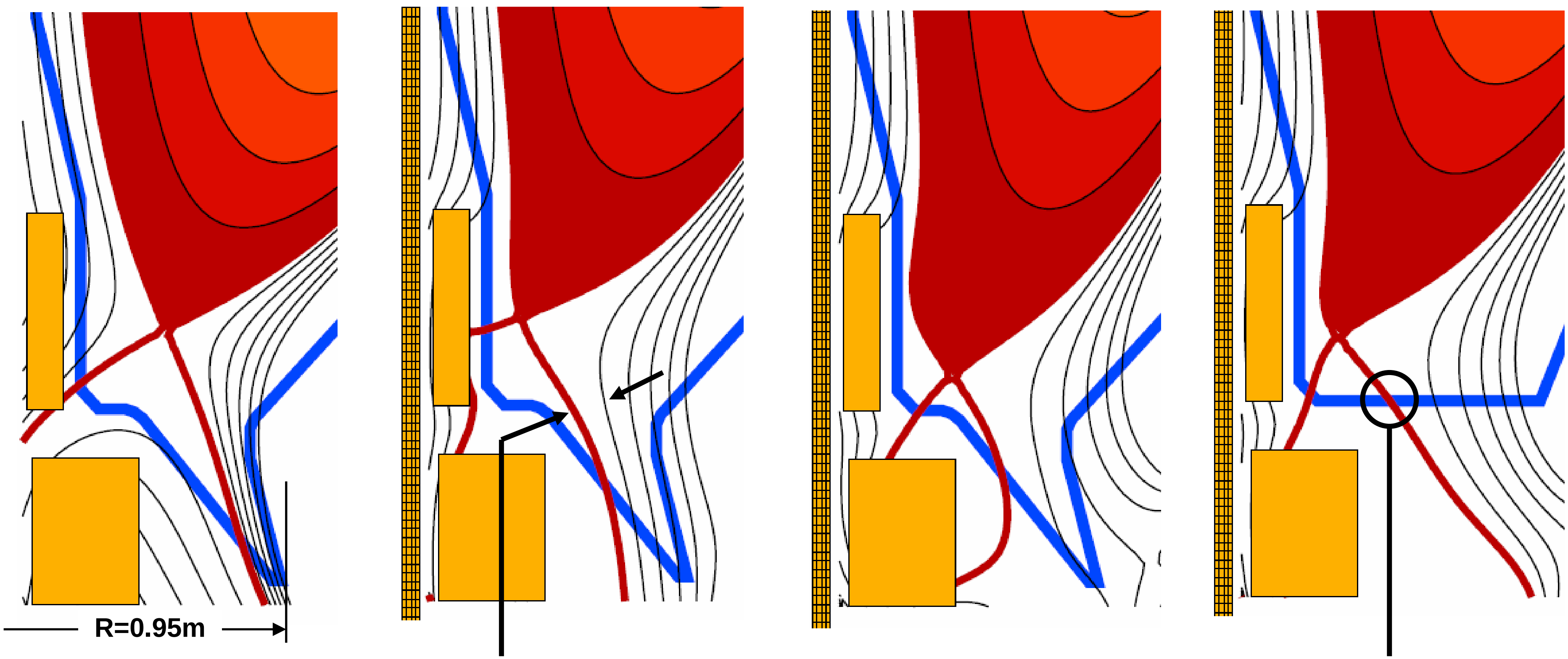
Example LSN shape



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 $\equiv \alpha_t$

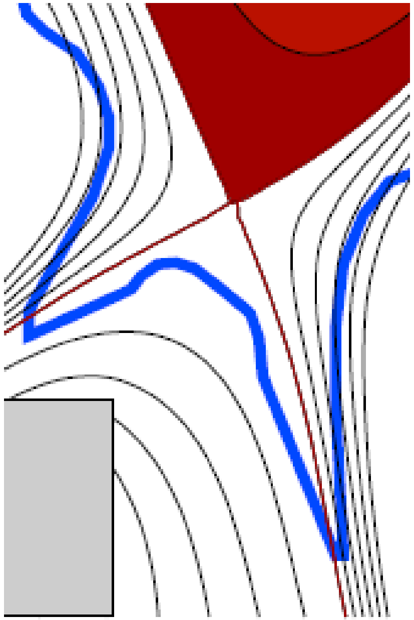
f_{exp} = 2.8	f_{exp} = 9	f_{exp} = 17	f_{exp} = 35
α_p=22° α_t=5.1°	α_p=23° α_t=1.8°	α_p=25° α_t=1.0°	α_p=64° α_t=1.1°



Flux contours have 5mm separation at midplane

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

NHTX can test wide range of divertor heat flux values

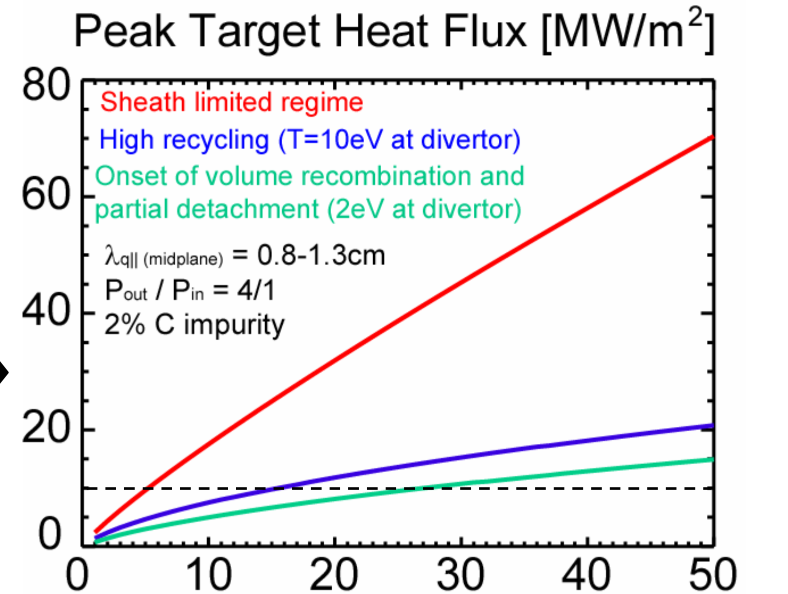


Note: ITER designed for $q_{div} \leq 10 MW/m^2$

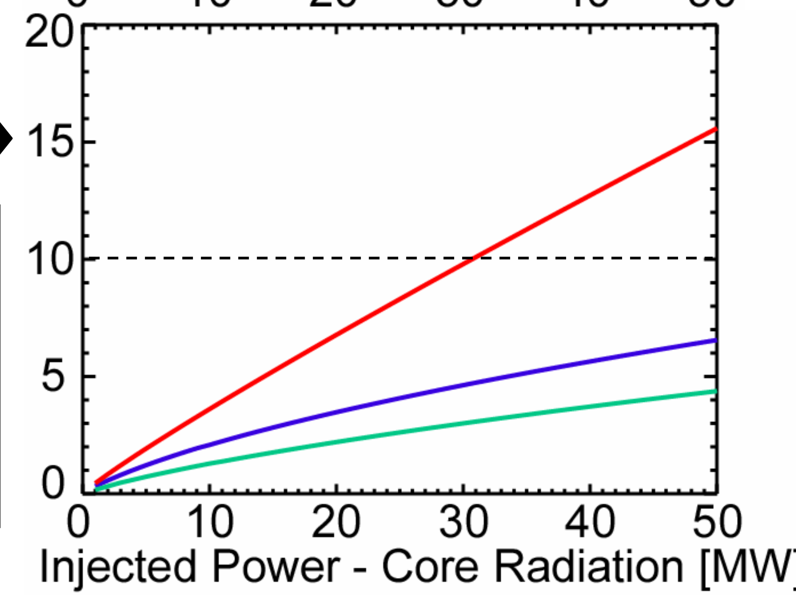
ITER-like div. - LSN

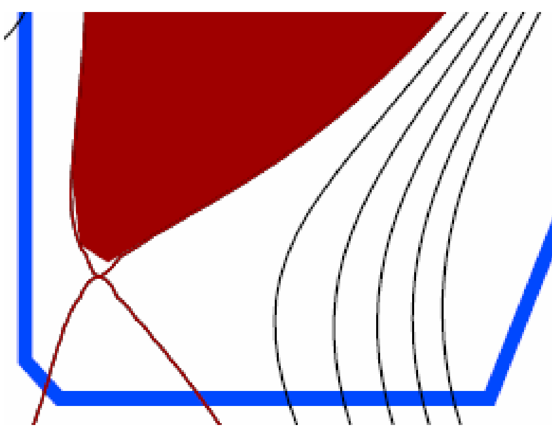
f_{exp} = 3 at strike-pt

Compatible with solid divertor material?



DND - f_{exp} = 35 at strike-point



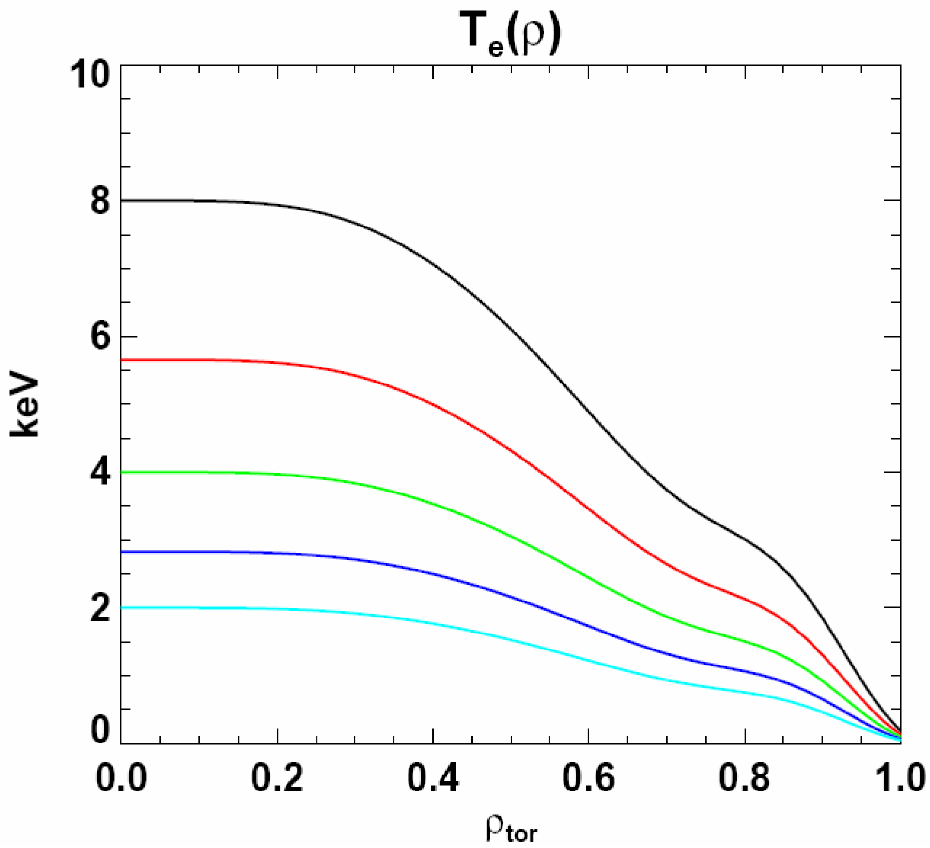


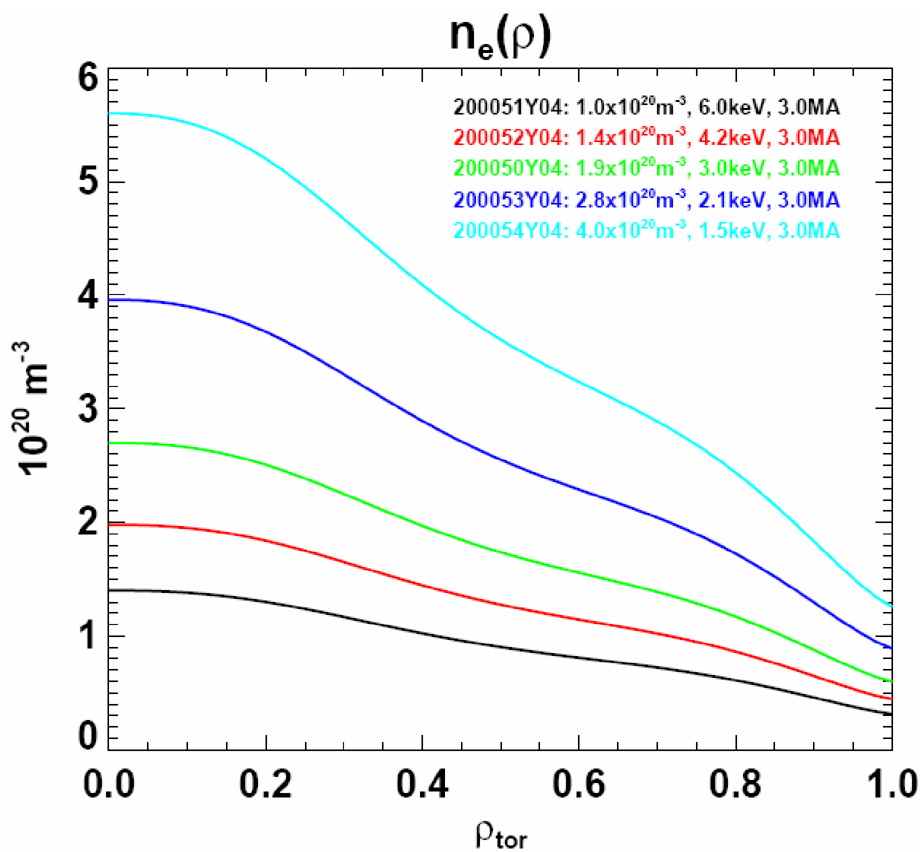
…but can one pump over large surface area?

Use slowly-flowing liquid lithium?

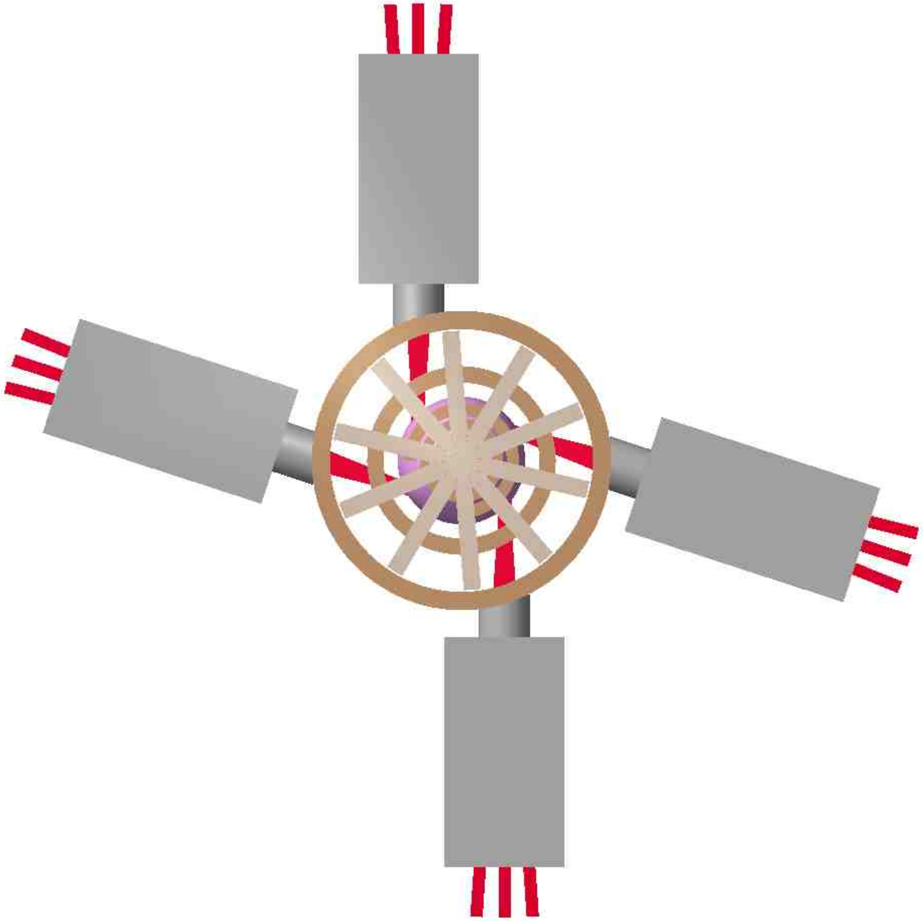
NBICD assessment w/ TRANSP uses thermal profile shapes based on high f_{NI} = 60-70% NSTX discharges

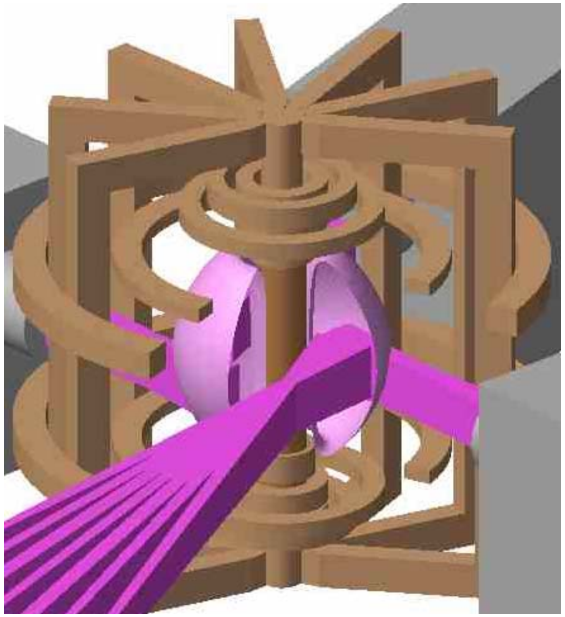
- Scale n_e, T_e profiles from 116313 - fixed T_i / T_e = 1.5, β_T=14%





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

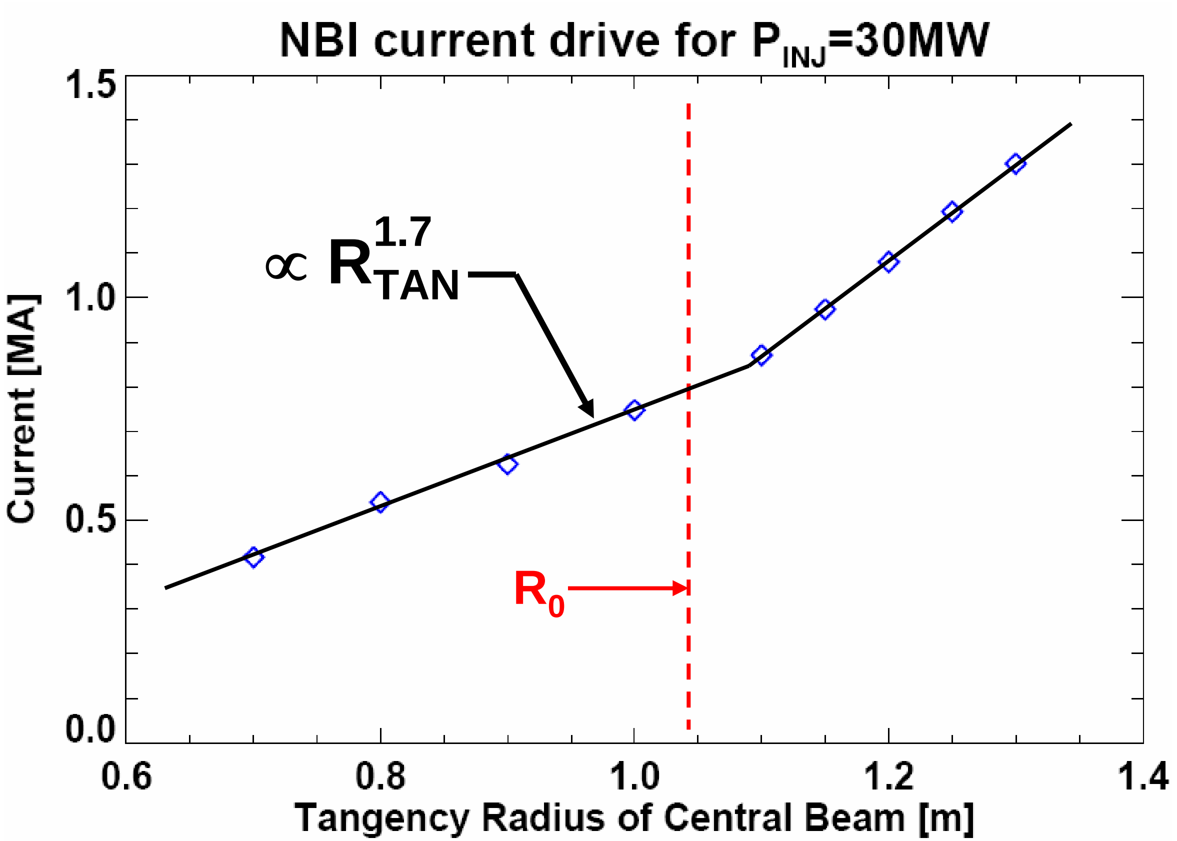




- R_{TAN} range = 1m $\pm$ 0.2m possible with cross-over point at vessel entrance

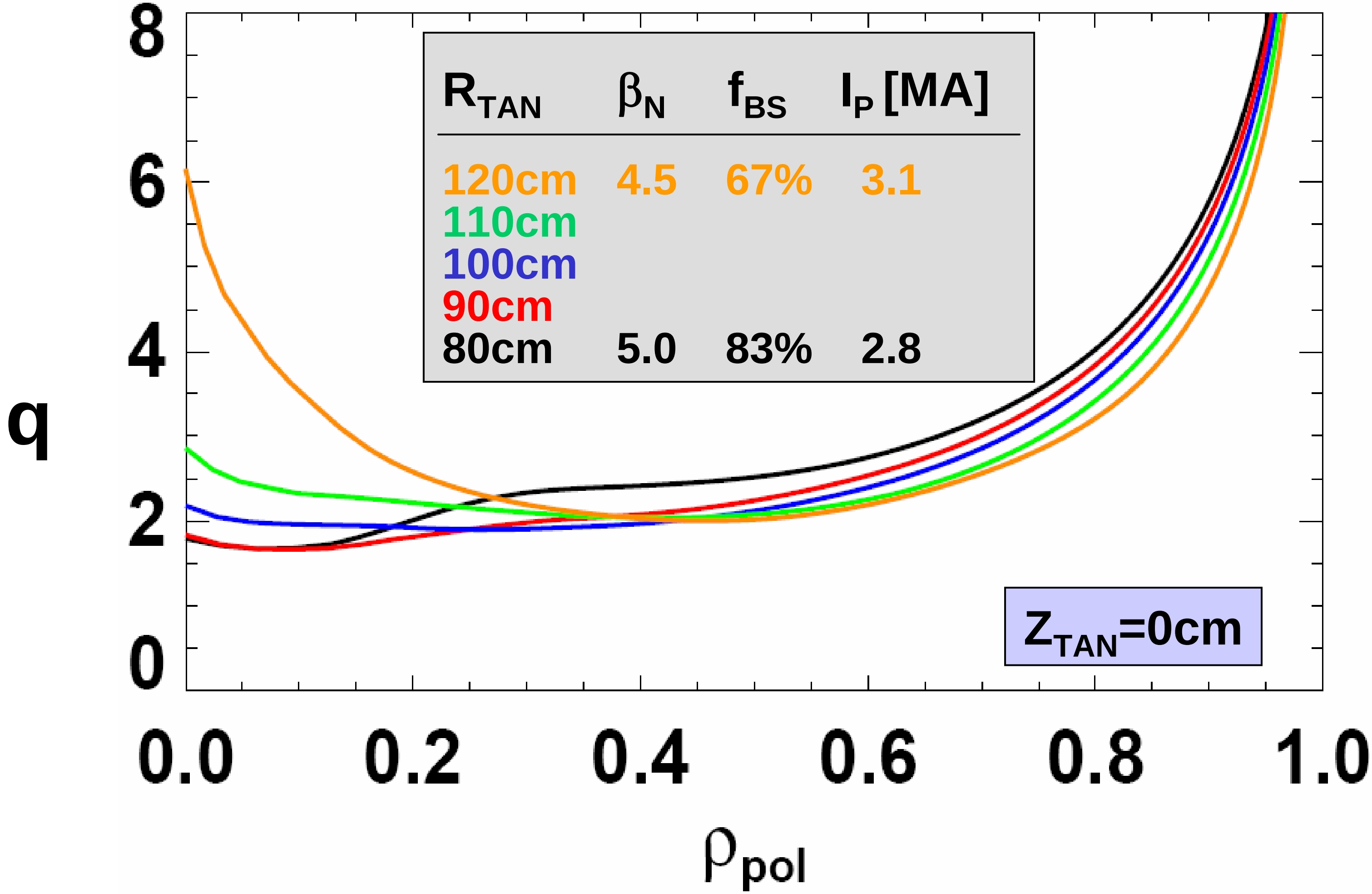
Driven current increases × 3 for R_{TAN}=0.7 → 1.3m and increases more quickly w/ radius for $R_{TAN} > R_0$

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



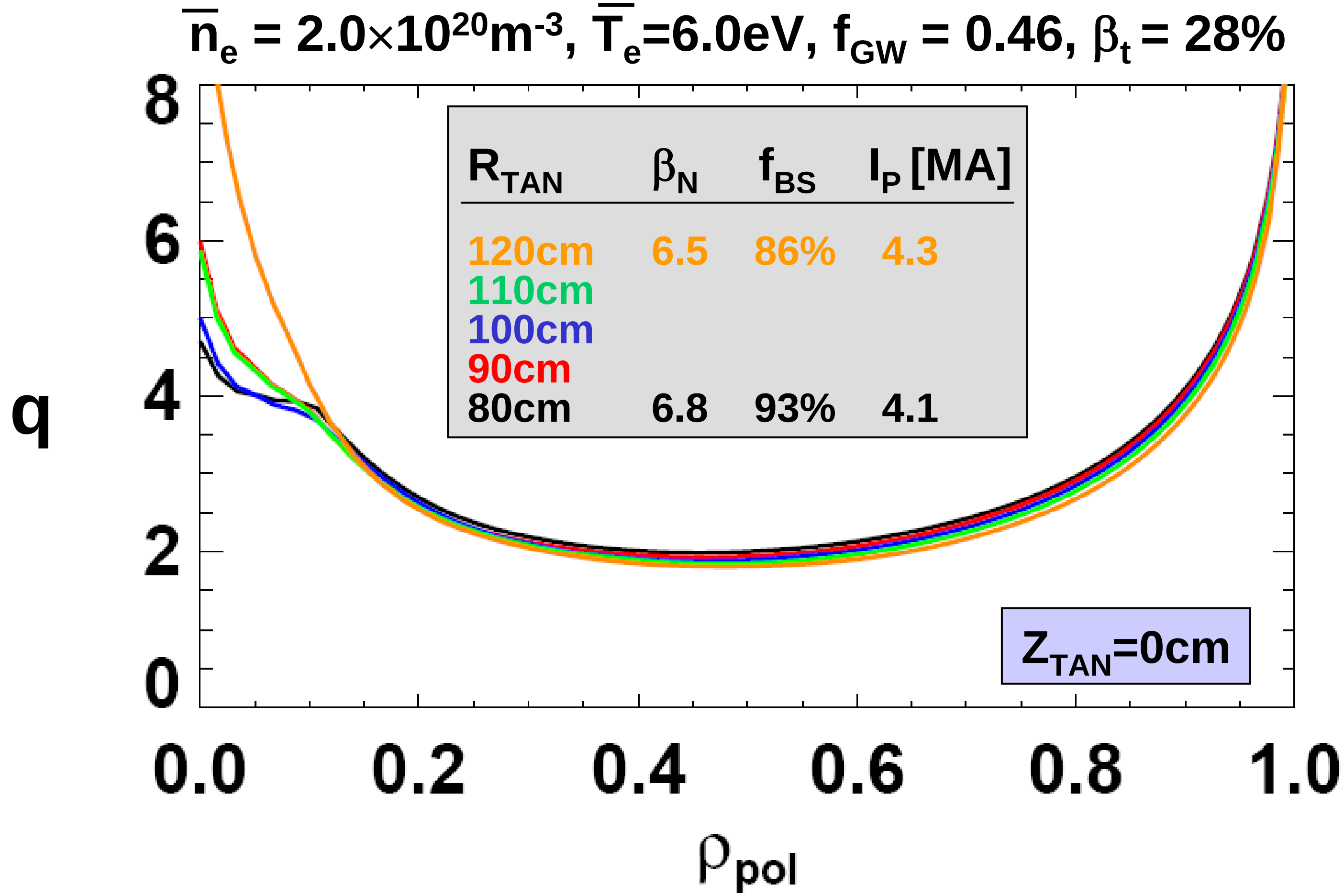
At design point, tangency radius of injection controls degree of shear reversal and radius of q_{MIN}

$\overline{n}_e = 1.4\times10^{20}m^{-3}$, $\overline{T}_e$=4.2keV, f_{GW} = 0.43, β_t = 14%



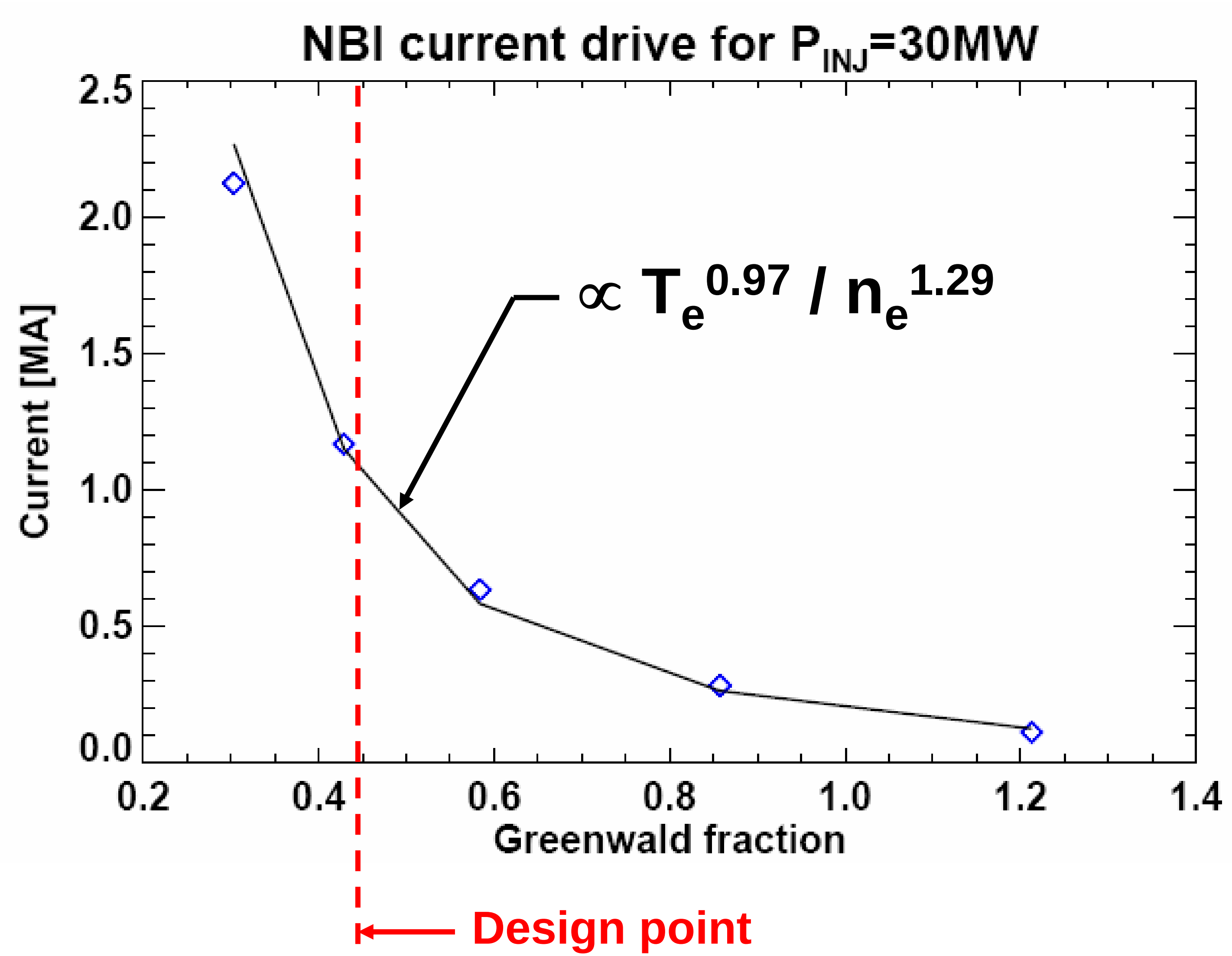
With sufficient confinement and/or P_{AUX}, NHTX can investigate high f_{BS} AT physics relevant to Demo

If β_t is doubled, bootstrap current dominates NBI-driven current, and R_{TAN} controls only q(0)



Next step: assess stability and sensitivity to n,T profile shapes

Ability to control density and operate at $f_{GW} < 0.5$ crucial for high NBICD efficiency

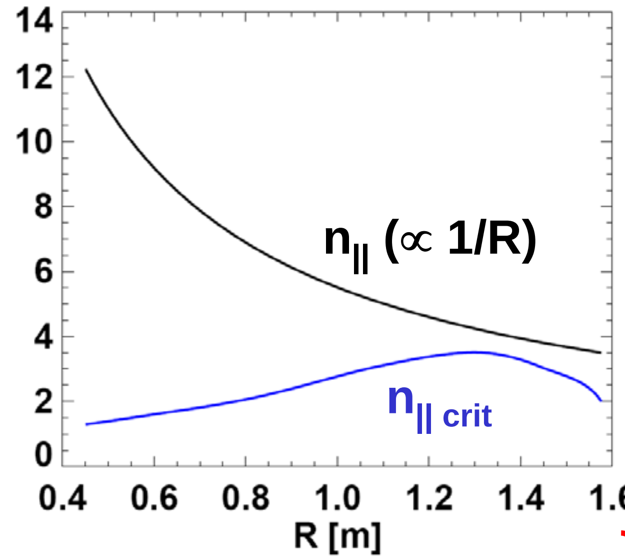


LHCD for lower-density operating points and current ramp-up appears promising

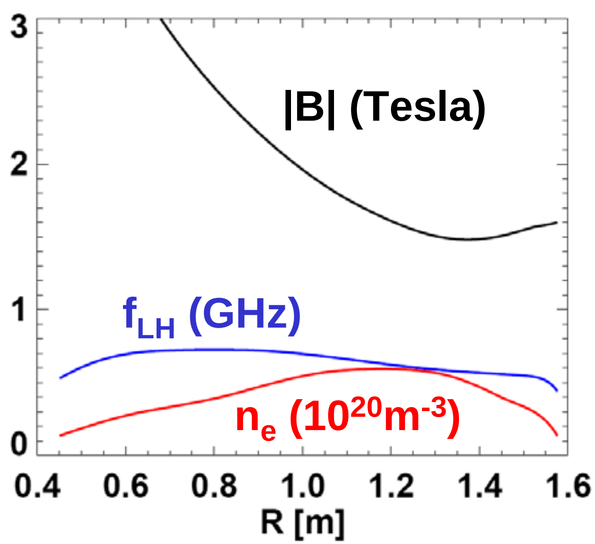
- Require $n_{||} > 3.5$ for $n_e(0) = 6\times10^{19} m^{-3}$
- Find v_{ph} / v_{the} = 1.5-3 for $T_e(0)$=3keV

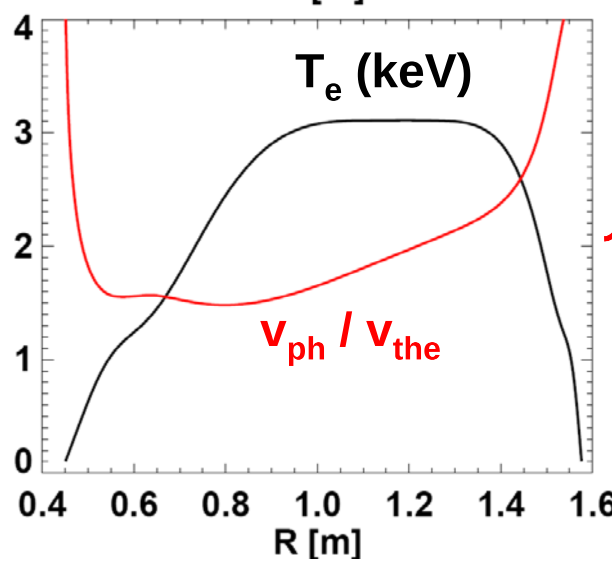
Core LHCD efficiency = 0.1 A/W

1MA of I_P for 10MW delivered



$$n_{\|crit} \approx \frac{\omega_{pe}}{|\Omega_e|} + \sqrt{S}$$





v_{ph} / v_{the} for high η_{CD}

$$\frac{I}{P} \simeq \frac{(v_{ph}/v_T)^2}{30} \left[\frac{T_{10}}{R_1 n_{14}}\right] \text{A/W}$$

Summary

- 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, full NICD
  - HH_{98Y} = 1.3, β_N=4.5, β_T=15%, $f_{BS} \geq$ 65%, f_{GW}=0.4-0.5
  - High 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

Sign-up