NSTX-U is sponsored by the
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Power exhaust scenarios & control for projected high-power NSTX-U operation

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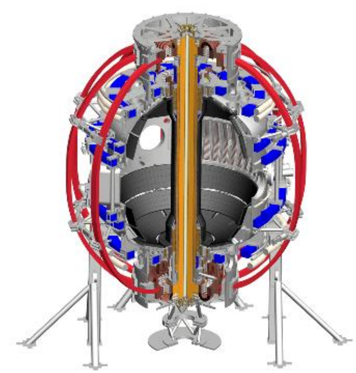
PPPL

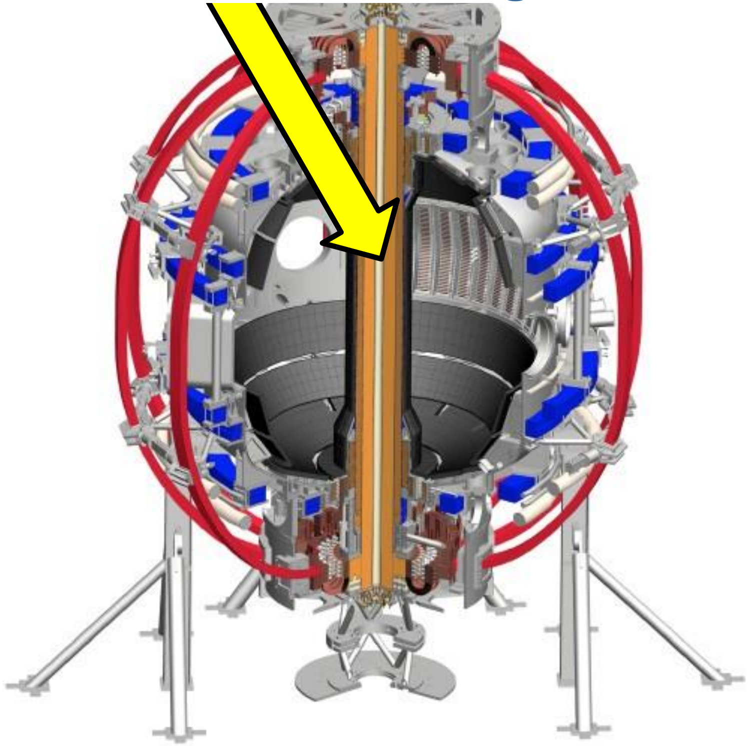


Abstract / Motivation

- An important goal of the NSTX Upgrade (NSTX-U) research program is to characterize energy confinement in the low-aspect-ratio spherical tokamak configuration over a significantly expanded range of plasma current, toroidal field, and heating power, while increasing flattop durations to ~5 seconds.
- However, the narrowing of the scrape-off layer at higher current combined with an improved understanding of expected halo-current loads has motivated a significant re-design of NSTX-U plasma facing components in the high-heat-flux regions of the divertor.
- In order to reduce the expected divertor heat flux to acceptable levels, a combination of mitigation techniques will be used: increased divertor poloidal flux expansion and divertor radiation, and controlled strike-point sweeping.
- The machine requirements for these various mitigation techniques are studied here using a newly implemented reduced heat-flux model.
- Systematic equilibrium scans are used to quantify the required divertor coil currents and to verify vertical stability for a range of plasma shapes.
- Free-boundary control schemes to constrain the strike-point location and field-line angle-of-incidence are also described.

NSTX facility upgraded to access new physics using 2 major new tools:

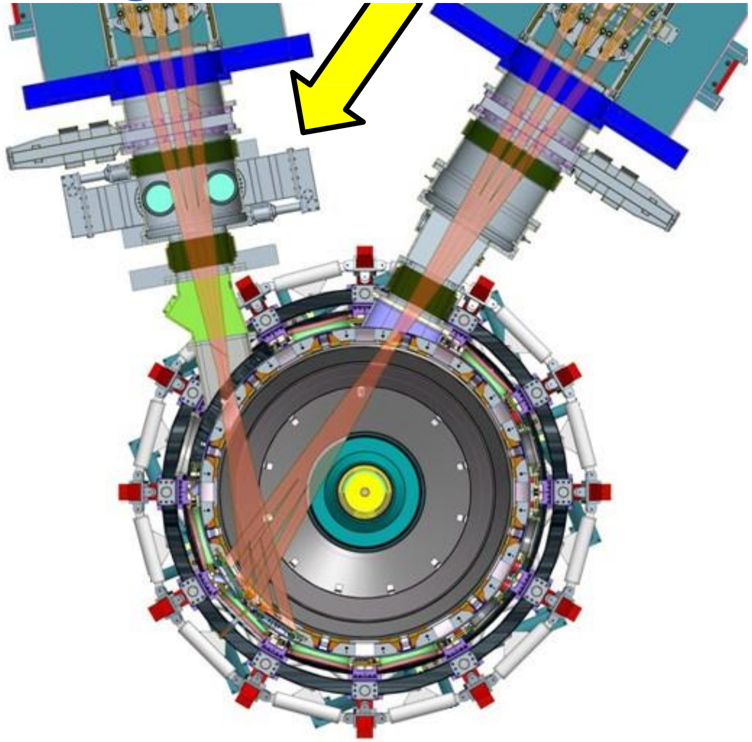
1. New Central Magnet



Higher T, low ν^* from low to high β

→ Unique regime, study new transport and stability physics

2. Tangential 2^{nd} Neutral Beam



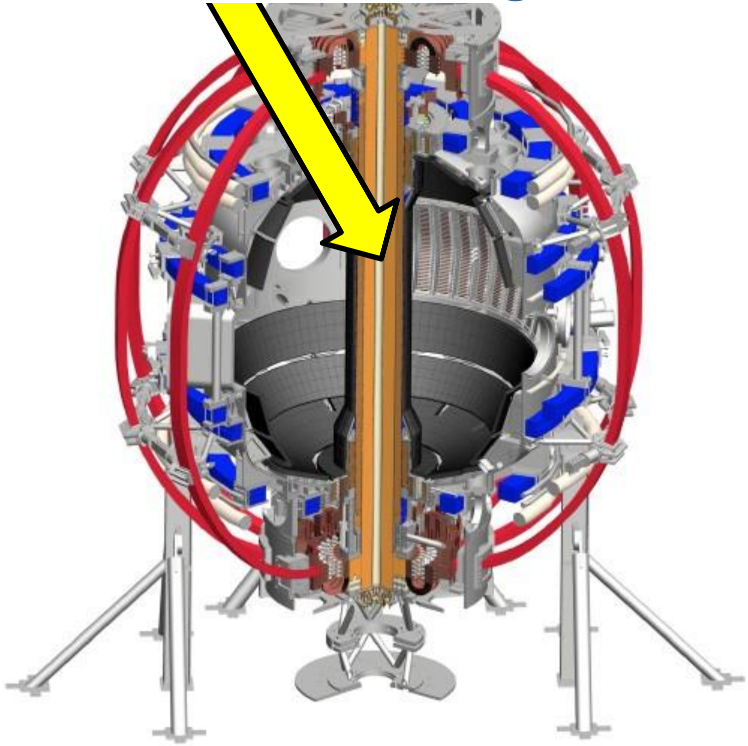
Full non-inductive current drive

→ Not demonstrated in ST at high-β_T

Essential for any future steady-state ST

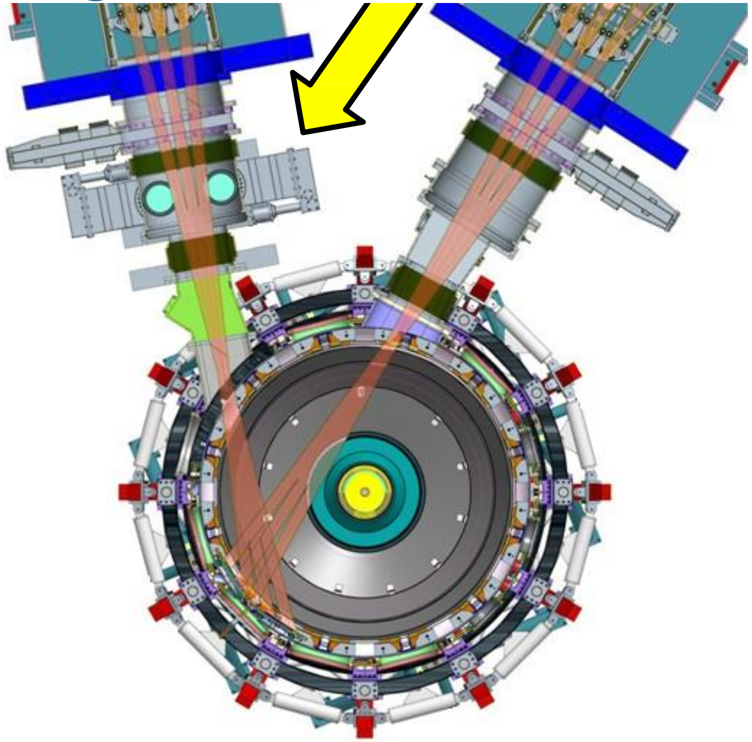
NSTX-U integrated performance goals

1. New Central Magnet



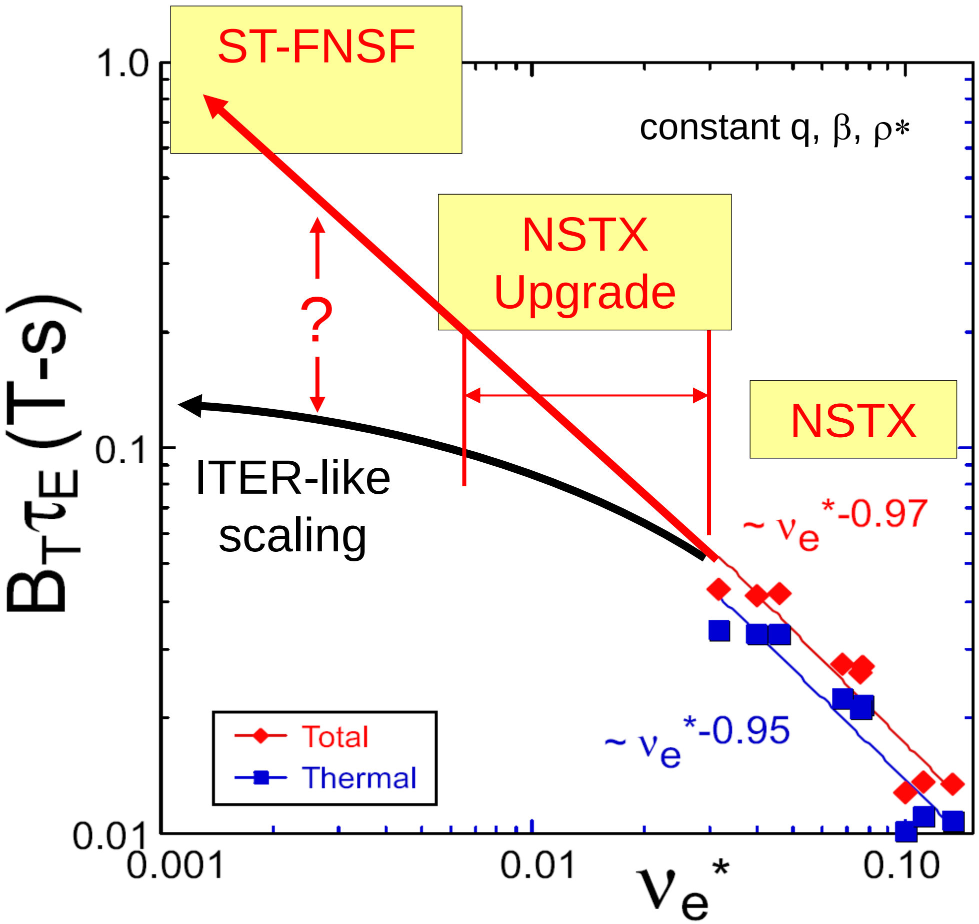
- ➢2× toroidal field (0.5 → 1T)
- ➢2× plasma current (1 → 2MA)
- ➢5× longer pulse (1 → 5s)

2. Tangential 2nd Neutral Beam



- ➢2× heating power (5 → 10MW)
- ➢4× divertor heat flux (→ ITER levels)
- ➢Up to 10× higher $nT\tau_E$ (~MJ plasmas) and energy injected into vessel

NSTX confinement increased at higher T_e



Normalized electron collisionality $\nu_e^* \propto n_e / T_e^2$

Will confinement trend continue, or look like conventional A?

Rationale for 1T, 2MA, 10MW requirement

- 2× increase in plasma current I_P and toroidal field B_T sufficient to differentiate between conventional (ITER 98y,2) and low-A (ST) H-mode confinement time τ at higher plasma temperature and/or density:

$$\tau_{98y,2} \propto I_{\mathrm{P}}^{0.93} B_{\mathrm{T}}^{0.15} \bar{n}_{\mathrm{e}}^{0.41} P_{\mathrm{Loss}}^{-0.69} R_0^{1.97} \varepsilon^{0.58} \kappa^{0.78}$$

$$\tau_{\mathrm{ST}} \propto I_{\mathrm{P}}^{0.57} B_{\mathrm{T}}^{1.08} \bar{n}_{\mathrm{e}}^{-0.44} P_{\mathrm{Loss}}^{-0.73}$$

- 2× heating power P needed to access high enough stored energy to access wide range of β at 2× higher field and current: $\beta_N \leq \sim 4.5\text{-}5.5$, $\beta_T \leq \sim 15\text{-}20\%$

NSTX-U magnet pulse duration requirement arises from goal of achieving current profile equilibration

- Current redistribution time $\tau_{CR} \propto T_e^{3/2}$ → longest for highest confinement, lowest density
- Physics ranges of interest:
  - Normalized confinement: H_{98y2} = 1-1.5, $H_{ST} \leq 1$
  - Normalized density f_{GW} = 0.5-1 – lower range for low ν^*
- Longest τ_{CR} ~1.8s - for H_{98y2} = 1.4-1.5, f_{GW} = 0.5-0.6
  - This H_{98y2} and f_{GW} combo accessed only transiently on NSTX
  - Significant physics R&D needed to access such plasmas
- Access then measure stationary profiles → Δt_{flat} ~$3\tau_{CR}$

- **$3\tau_{CR}$ ~ 5-5.5s → motivates goal of 5s I_P flat-top**

Achievable pulse-lengths and $\Delta t_{flat} / \tau_{CR}$

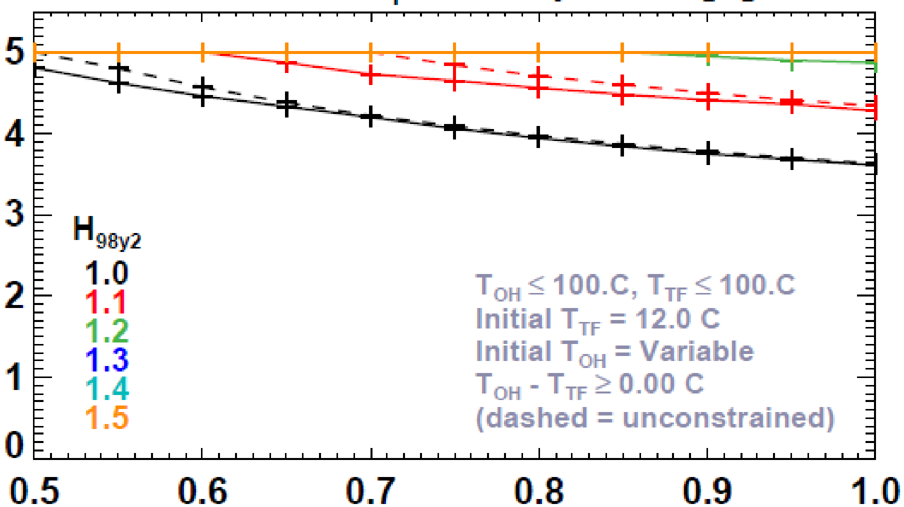
Constraint: maximum I_P flat-top duration = 5s (set by TF)

- $H_{98y2} \geq 1.2$: Δt_{flat} = 5s for all f_{GW}
- Δt_{flat} < 5s at lower H (high f_{GW})
- Note: NSTX-U accessed H_{98y2} ~ 1.1-1.15 during initial ops

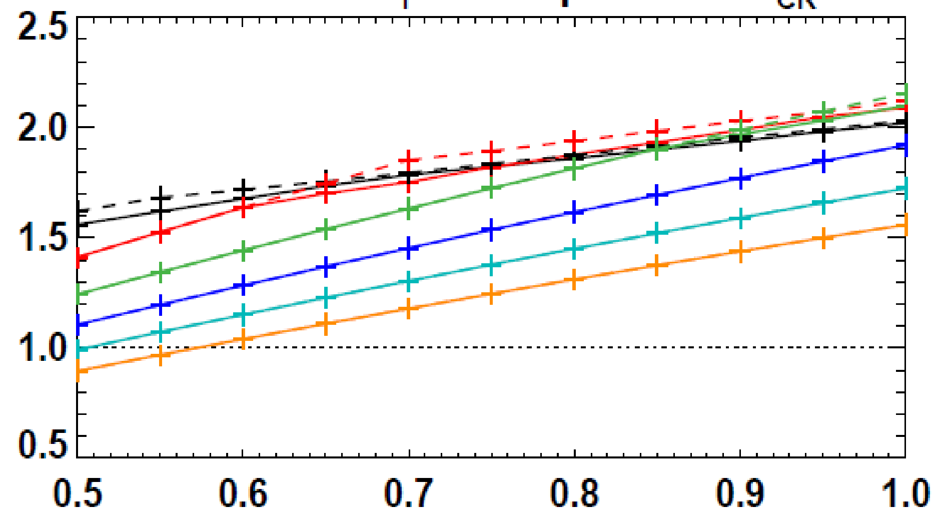
- Flat-top durations range from ~2.7 to ~6.3 τ_{CR}
- Nearly all cases (97%) meet or exceed ~3 τ_{CR}

I_P = 2.0MA, B_T = 1.0T, P_{NBI} = 8.0MW

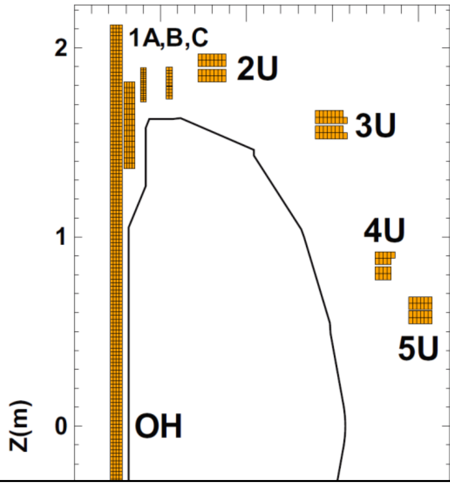
Maximum I_P flat-top time [s]

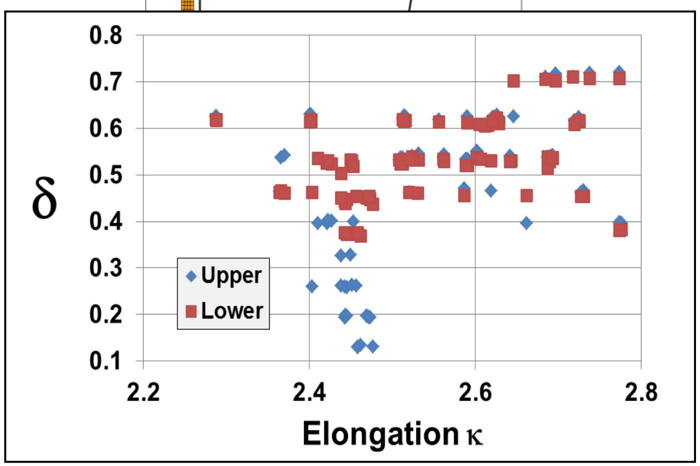


Maximum I_P flat-top time / $3\tau_{CR}$

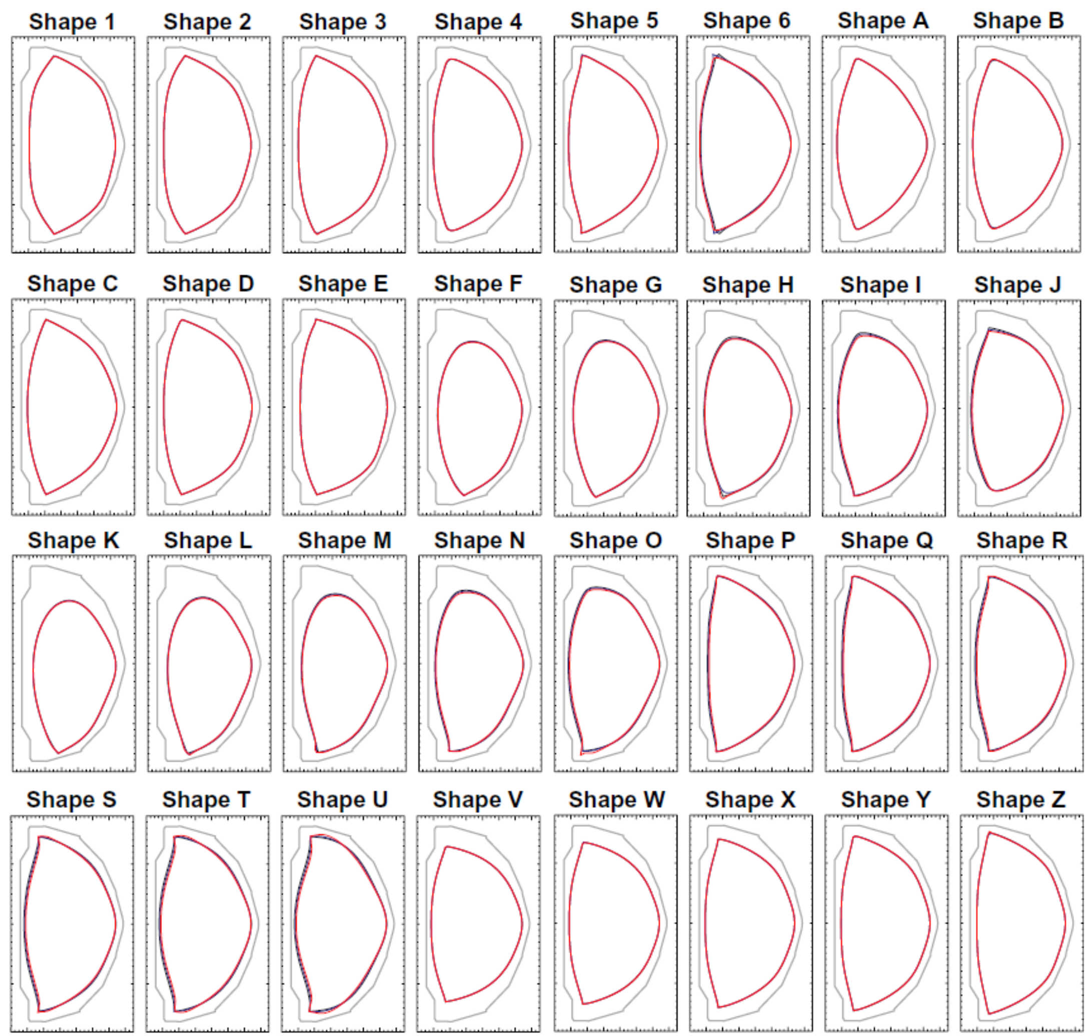


Boundary shaping flexibility drives PF coil and structural requirements for plasma operation

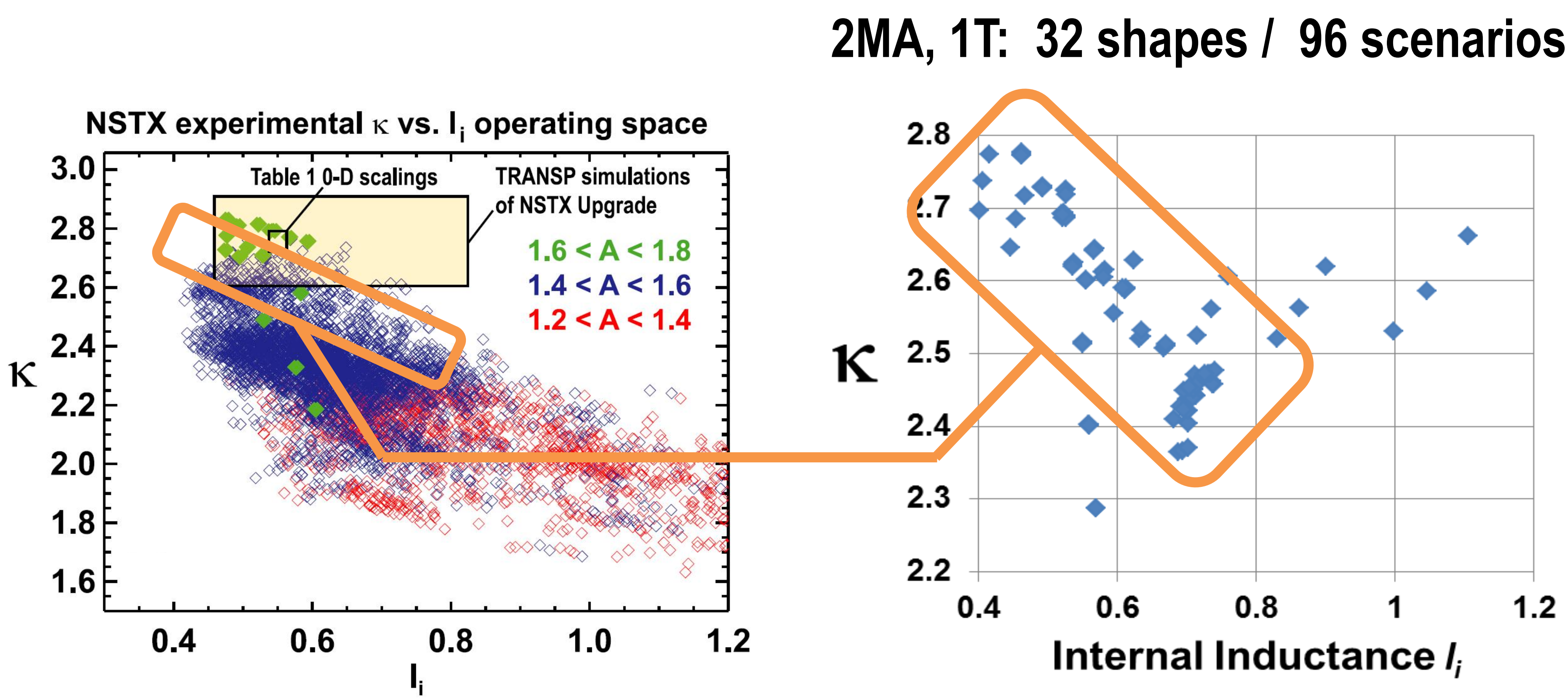




2MA, 1T: 32 shapes × 3 OH states = 96 equilibria



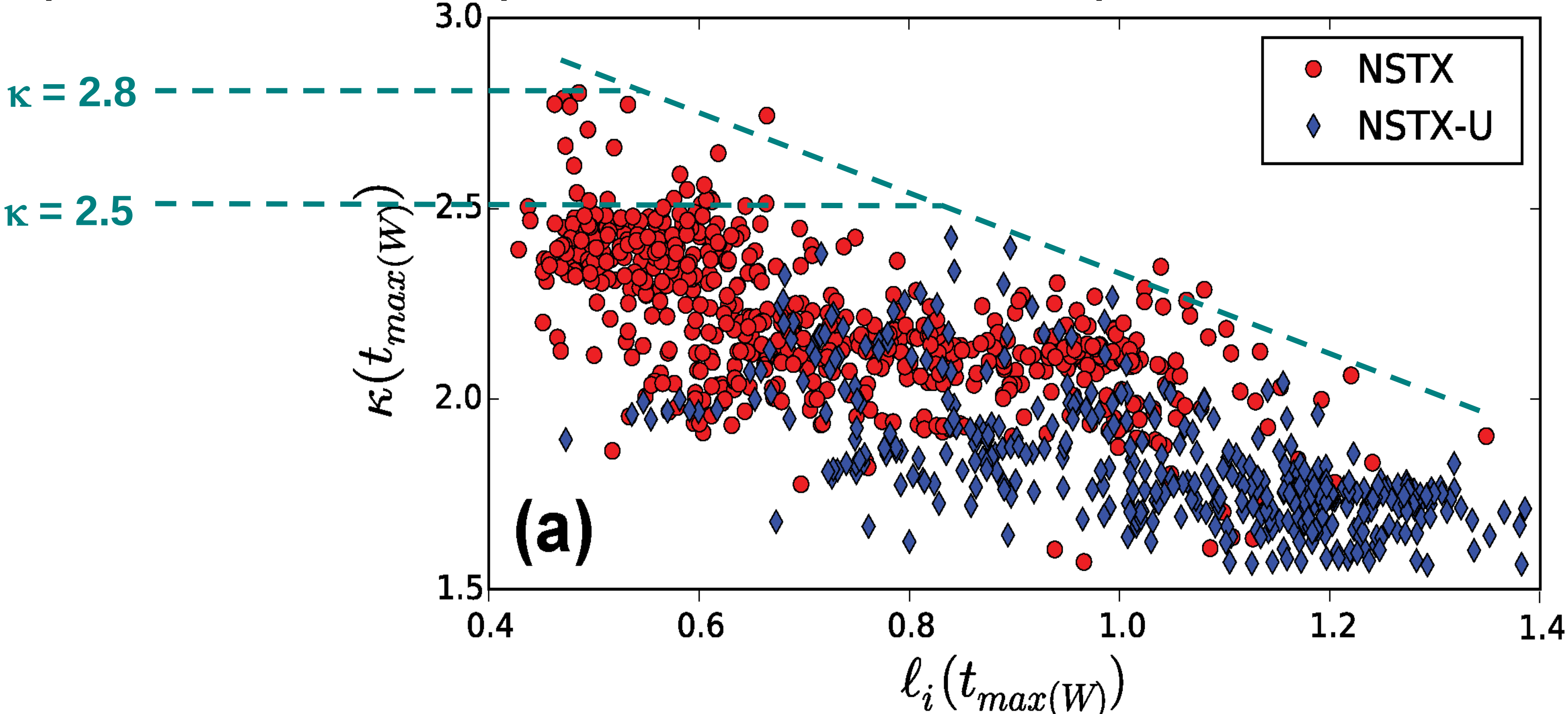
96 scenarios primarily higher-κ, lower-l_i high-performance H-modes (β_N ~ 5)



- Higher l_i ~1-1.5 L-modes not included in design requirements → L-mode I_P < 2MA, Δt_{flat} < 5s

Operating κ depends on range of stable internal inductance (& impacts power-exhaust solution)

- Internal inductance depends on bootstrap and NBI current profiles which depend on thermal transport



- Risk: No NSTX-U transport data yet at high B_T, I_P, κ
  - This confinement physics data is a key motivation for the Upgrade in the first place!

Reduced heat flux model

Parametric fits to divertor heat flux

- T. Eich, et al., Nucl. Fusion **53** (2013) 093031, Eqn 1

$$q(\bar{s}) = \frac{q_0}{2} \cdot \exp\left(\left(\frac{S}{2\lambda_q}\right)^2 - \frac{\bar{s}}{\lambda_q \cdot f_x}\right) \cdot \mathrm{erf}c\left(\frac{S}{2\lambda_q} - \frac{\bar{s}}{S \cdot f_x}\right) + q_{BG}$$

$$\bar{s} = s - s_0 = (R_{sep} - R) \cdot f_x$$

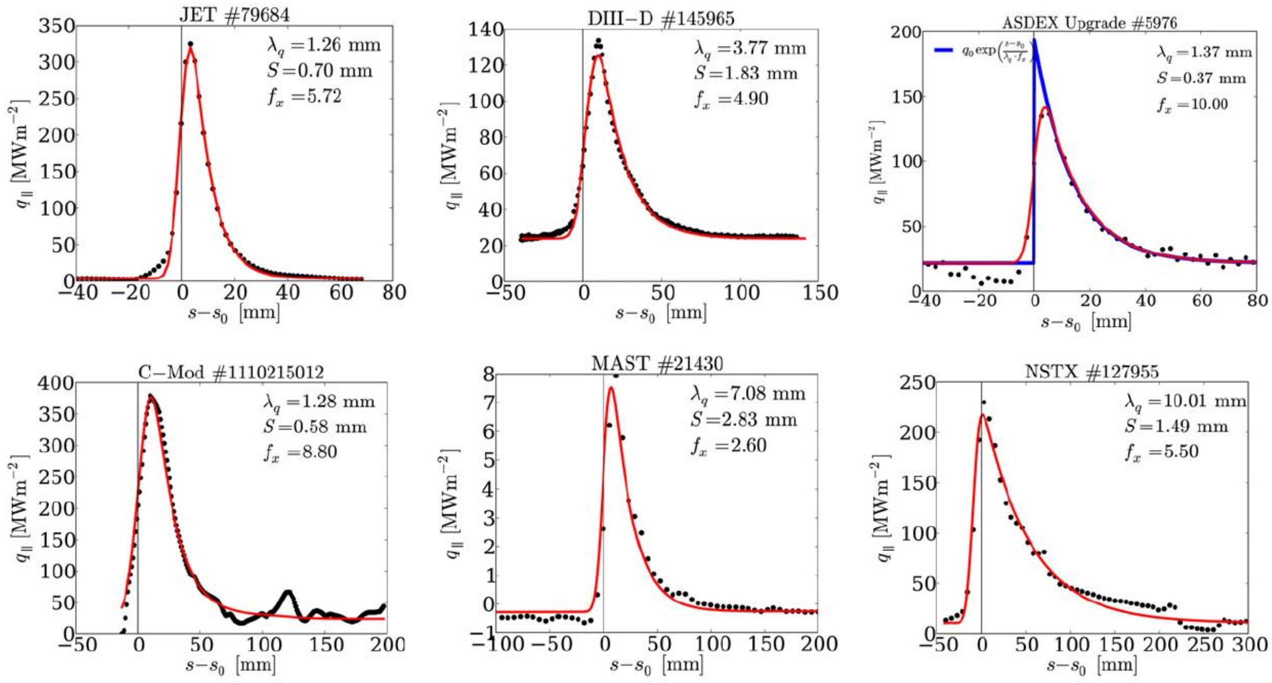


Figure 1. Typical outer target power parallel heat flux for each machine and result of fitting equation (1).

Model for SOL heat flux width λ_q

- T. Eich, et al., Phys. Rev. Lett. **107** (2011) 215001
  - Equations 5, 7-10 → Goldston heuristic drift model for λ_q

Map λ_m to outer midplane: $\lambda_m^* = \frac{R_{\rm geo}}{(R_{\rm geo} + a)} \frac{B_p}{B_p^{mp}} \lambda_m$

$$B_p = \frac{\mu_0 I_p}{2\pi a \sqrt{(1+\kappa^2)/2}}$$

$$\lambda_m = 2.02 \frac{f_{\rm AZ}}{\sqrt{(1+\kappa^2)}\epsilon^{1/8}} B_T^{-7/8} q_{\rm cyl}^{9/8} P_{\rm SOL}^{1/8}$$

$$q_{\rm cyl} = \frac{2\pi a \epsilon B_T}{\mu_0 I_p} \frac{(1+\kappa^2)}{2}$$

with λ_m in [mm], $P_{\rm SOL}$ in [MW], B_T in [T]

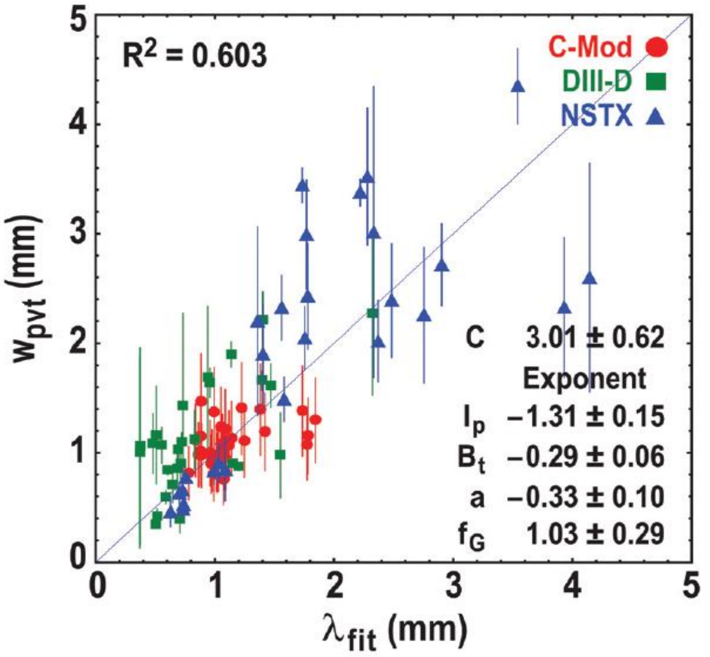
$$f_{\rm AZ} = \left(\frac{2\bar{A}}{1+\bar{Z}}\right)^{7/16} \left(\frac{Z_{\rm eff}+4}{5}\right)^{1/8}$$

$$\bar{Z} = \sum_i Z_i n_i / \sum_i n_i \qquad \bar{A} = \sum_i n_i A_i / \sum_i n_i$$

Use $\lambda_q = \lambda_m^*$

Data for private flux region width w_{pvt}=S

- M. Makowski et al., Phys. Plasma **19**, 056122 (2012)



- T. Eich, et al., Nucl. Fusion **53** (2013) 093031

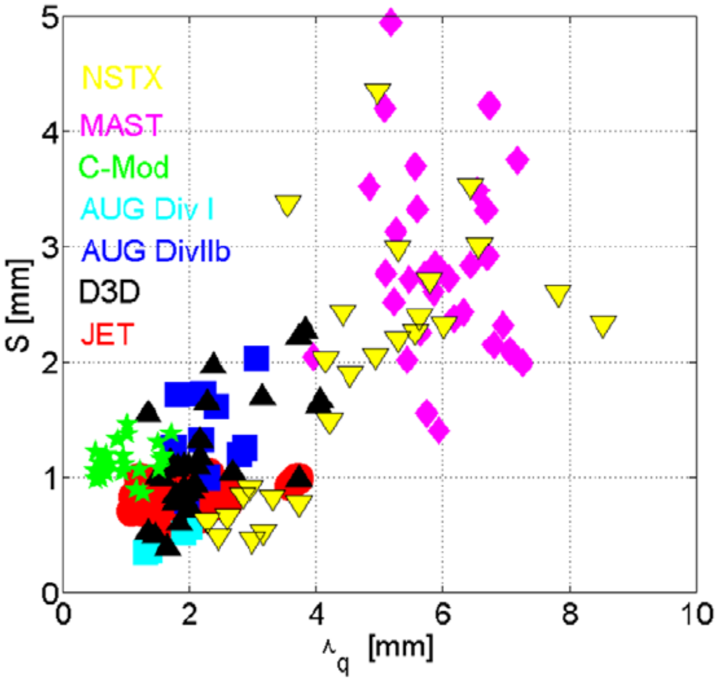


Table 5. Variation of mean power spreading factor, S and S/λ_q for the various devices.

	JET	DIII-D	AUG DivI	AUG DivII	C-Mod	MAST	NSTX
S (mm)	0.59–1.04	0.39–2.27	0.35–0.56	0.79–2.02	0.86–1.46	1.11–4.95	0.46–4.35
S/λ_q	0.26–0.81	0.24–1.14	0.26–0.28	0.40–0.94	0.67–2.32	0.17–0.95	0.15–0.95

Generalized divertor heat-flux model consistent with Eich parametric fitting

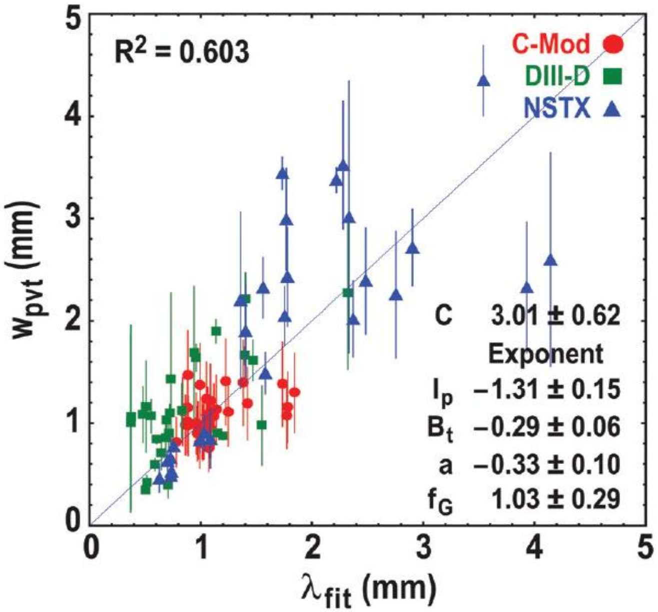
- SOL heat flux approximately field-aligned $\Rightarrow \vec{q} \approx \vec{q}_{||} = q_{||}\hat{b} = q_{||}\vec{B}/B$
- No SOL heat source $\Rightarrow \nabla \cdot \vec{q} = \vec{B} \cdot \nabla(q_{||}/B) = 0 \Rightarrow q_{||} = f(\psi)B$
- $q_{||} \equiv q_{||0}(B/B_0)\hat{q}(\hat{\psi})$ $\quad\quad \hat{q}(\hat{\psi}) \equiv 0.5\exp(\sigma_0^2 - \sigma)\mathrm{erfc}(\sigma_0 - \sigma/2\sigma_0)$
- $\sigma_0 \equiv S/2\lambda_q$ $\quad\quad \sigma \equiv \hat{s}/\hat{\lambda}_q$ $\quad\quad \hat{s} \equiv \hat{\psi} - 1$ $\quad\quad \hat{\psi} \equiv (\psi - \psi_{axis})/\Delta\psi$
- $\hat{\lambda}_q \equiv \lambda_q|\nabla\psi|_{omp}/\Delta\psi$ $\quad\quad \Delta\psi \equiv (\psi_{edge} - \psi_{axis})$
- Note: $q_{||0} \approx P_{div}B_0/(2\pi|\nabla\psi|_{omp}\lambda_q)$ for $\sigma_0 \rightarrow 0$
- Divertor surface normal unit vector $\equiv \hat{n} \Rightarrow q_{divertor} = (\hat{n} \cdot \hat{b})q_{||0}(B/B_0)\hat{q}(\hat{\psi})$
- Define total B-field angle of incidence $\theta_B \Rightarrow \hat{n} \cdot \hat{b} = \sin(\theta_B)$
- For $q_{divertor} =$ Eich $q(\bar{s}) = q_0\hat{q}(\bar{s}) \Rightarrow q_0 = \sin(\theta_B)q_{||0}(B/B_0)$

$$q_{divertor} = q_{||0}\hat{q}(\hat{\psi})(B/B_0)\sin(\theta_B)$$

Choice of S for NSTX-U calculations

- Options:
  - S_{Mak} = S from Makowski scaling ⟹
  - $S_{rel} = MIN(S / \lambda_q) \times \lambda_q = 0.15 \times \lambda_q$
    - $MIN(S / \lambda_q)$ = 0.15, 0.17 for NSTX, MAST
  - S_{fix} = fixed / constant value of S
- NSTX-U model uses combination of all these options as follows:
  - First set $S = MIN([S_{Mak}, S_{rel}])$
  - Then enforce $S_{min} \leq S \leq S_{max}$
    - S_{min} = 0.1mm, S_{max} = 0.5mm
  - S typically set by $S = S_{rel} \approx$ 0.2-0.3mm
  - $S = 0.15 \times \lambda_q \approx S_{mak}(f_G = 0.4, 2MA, 1T)$

  ➔ **Consistent w/ physics/ops goal $f_G \geq 0.5$**



Model for power conducted to divertor target

- P_{heat} = total heating power (ohmic + auxiliary + alpha)
- f_{rad} = fraction of heating power radiated from core
  - For NSTX-U projections assume $f_{rad} = 0.3$
- $P_{rad} = f_{rad} \times P_{heat}$ = power radiated from core
- $P_{sol} = P_{heat} \times (1-f_{rad})$ = power into SOL
- N_{div} = Number of in/out divertor legs connected to target
  - $N_{div} = 1$ for single null (SN), $N_{div} = 2$ for double null (DN)
- f_{obl} = fraction of power to outboard divertor leg(s)
  - For NSTX-U projections assume $f_{obl} = 0.8$ for DN, 0.65 for SN
- $f_{ibl} = (1-f_{obl})$ = fraction of power to inboard divertor leg
- $f_{leg} = f_{obl}$ or f_{ibl} = fraction of power to chosen divertor leg

$P_{div} = P_{sol}\, f_{leg} / N_{div}$ = power conducted to divertor target

NSTX example: low δ, I_P = 0.8MA

D. Gates *et al.* Phys. Plasmas 13 (2006) 056122

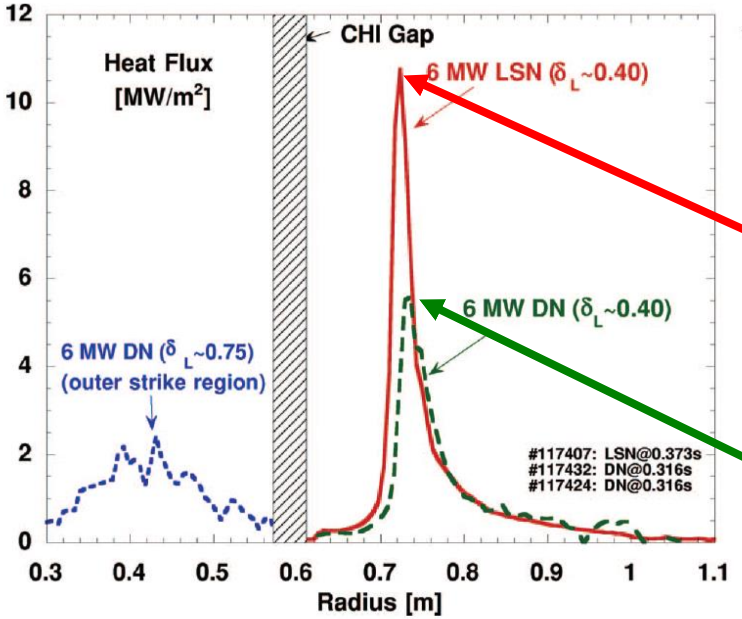
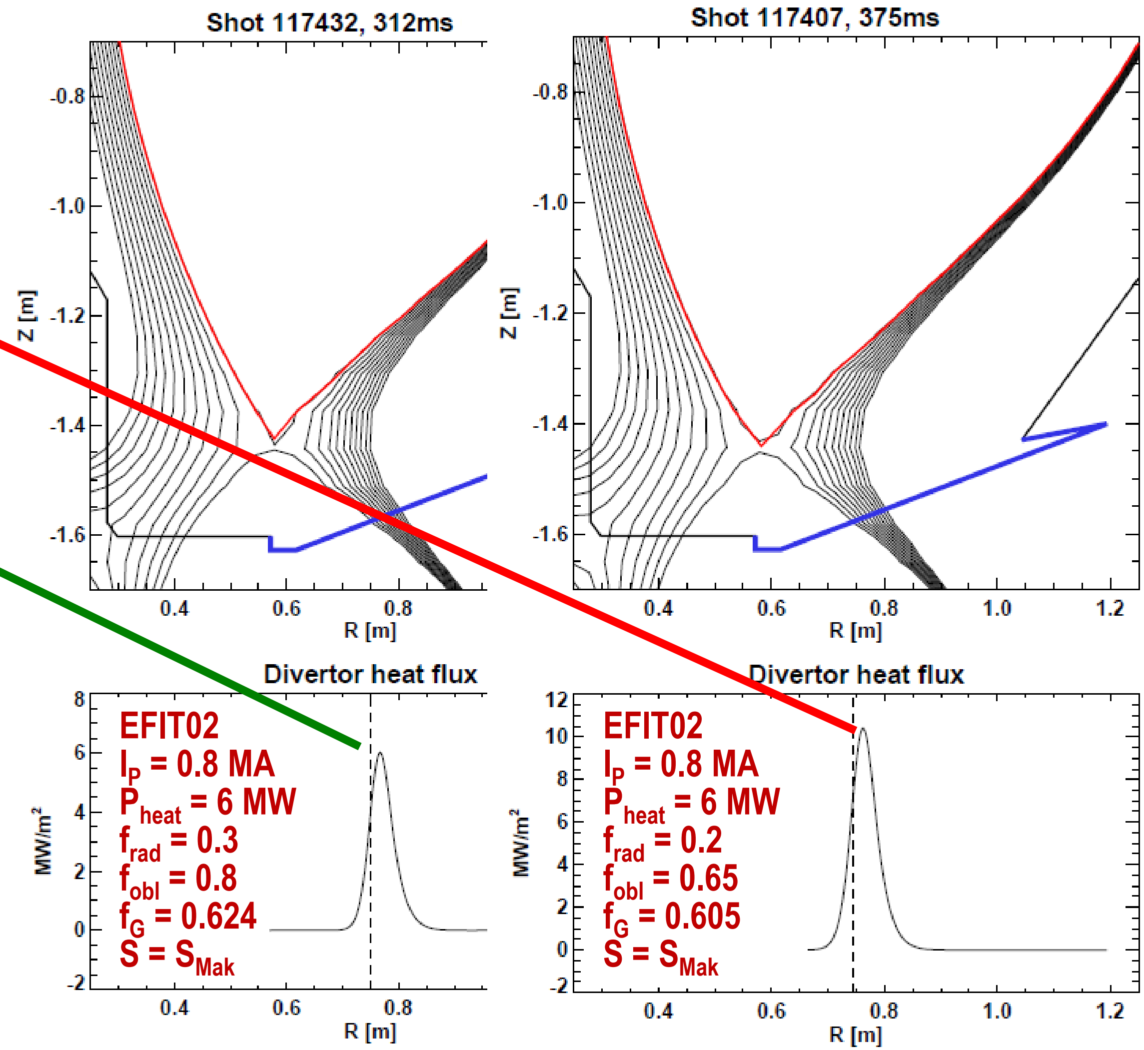


FIG. 13. (Color) The peak heat flux vs major radius for three different divertor configurations: (1) low δ lower single null, (2) low δ double null, and (3) high δ double null. All three configurations had identical heating power. The peak heat flux reduces by a factor of ~5.



NSTX example: high δ, I_P= 1.2MA

V.A. Soukhanovskii et al Nucl. Fusion 49 (2009) 095025

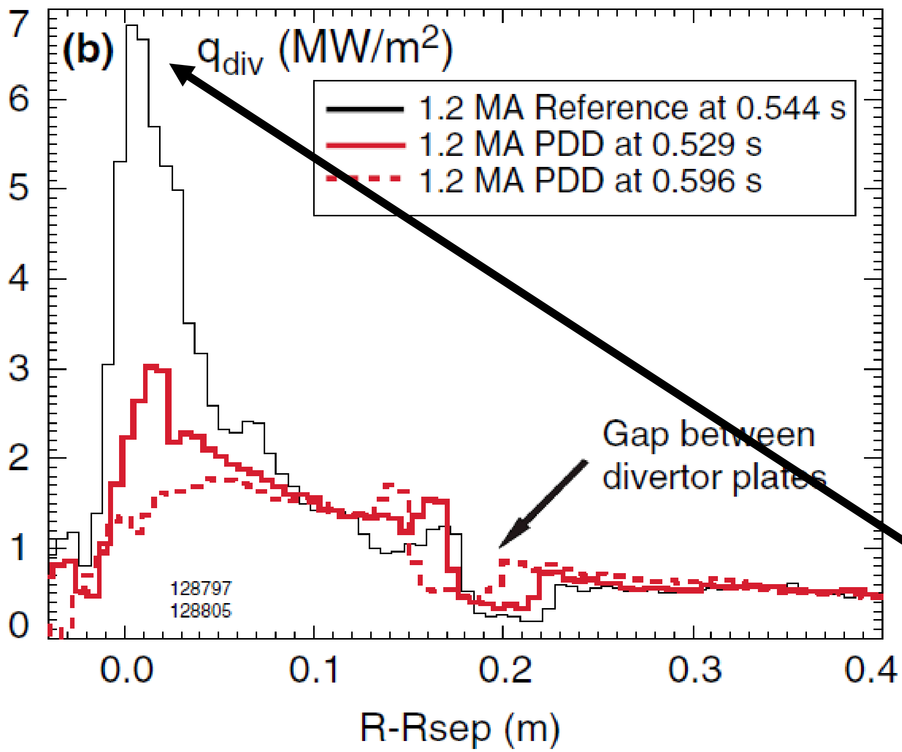
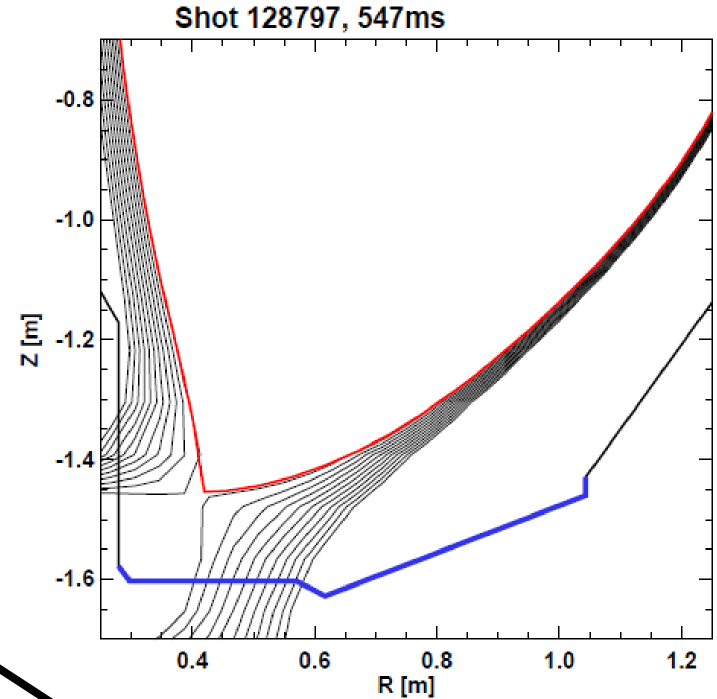
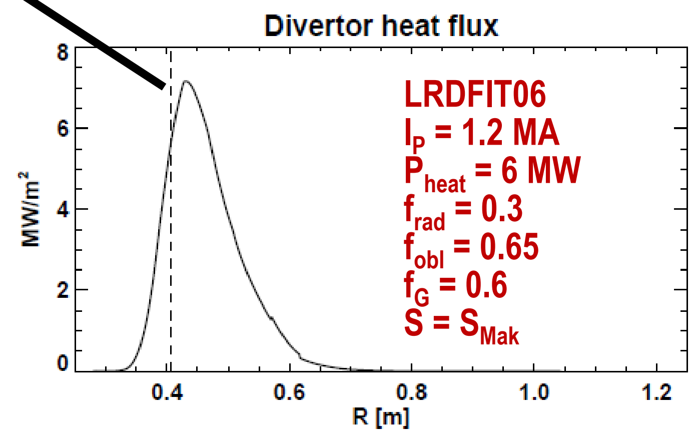


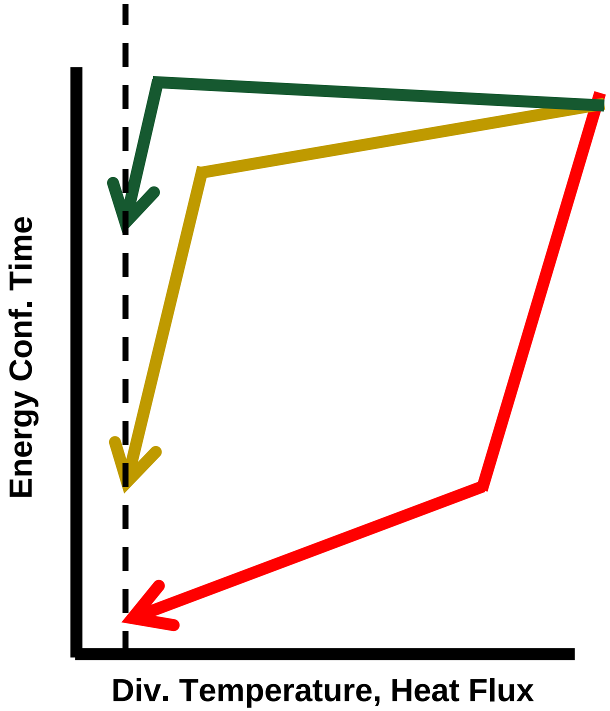
Figure 4. Divertor heat flux profiles in the reference and PDD discharges: (*a*) 1.0 MA and (*b*) 1.2 MA.





Strategies for Mitigating Heat Fluxes

- Increase poloidal flux expansion
  - changes the amount of wetted area on divertor, but also makes for shallow angles
- Strikepoint sweeping in time
  - use PF coils to move the strike point back and forth across the surface
- Increase radiation fraction *(30% assumed in models)*
  - contingency due to uncertainty of compatibility w/ physics goals
  - adding new divertor fueling locations to help us exploit radiative exhaust

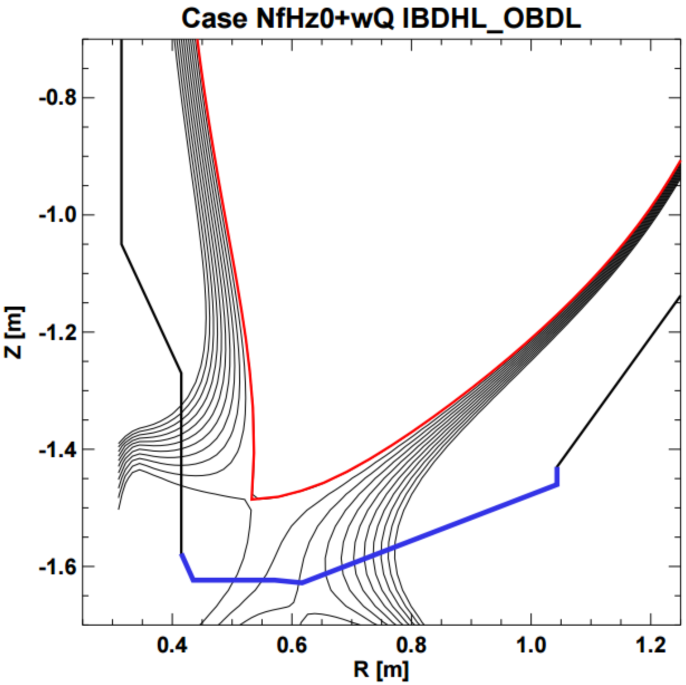


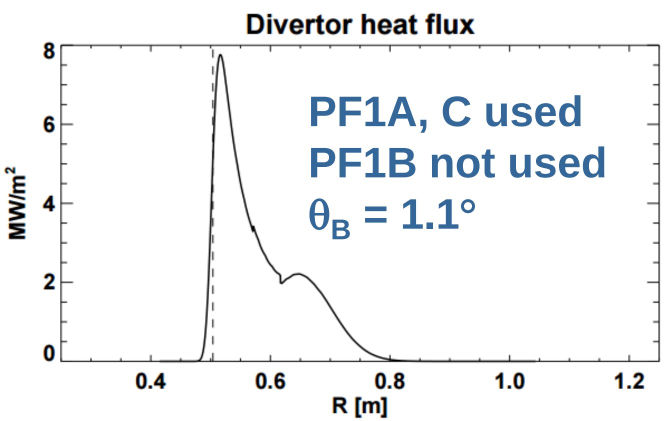
how we move on this plot is a research focus of the fusion program worldwide

Comments on comparison to NSTX and extrapolation to NSTX-U

- Have looked at very limited number of NSTX cases
  - f_{rad} = 0.2-0.3 and S = S_{Mak} ***might*** be reasonable scaling assumption for NSTX / NSTX-U
    - Peak heat fluxes can match, but exact profile shapes differ
    - There is substantial uncertainty in both f_{rad} and S_{Mak}
  - Need DIVSOL TSG to identify more cases for comparison
  - More detailed analysis of NSTX S-scaling would be valuable
- For scaling to NSTX-U, use more conservative (i.e. smaller) $S = S_{rel} = 0.15 \times \lambda_q$
- Detachment is option for reducing NSTX-U heat-flux
  - Showed reduction of $q_\perp$ by ~50-70% in NSTX
  - Prefer not to rely on detachment for NSTX-U scenarios
    - Beneficial to have more operating margin if PFCs will allow

NSTX-U projection example: high δ, I_P = 2MA with high flux expansion divertor

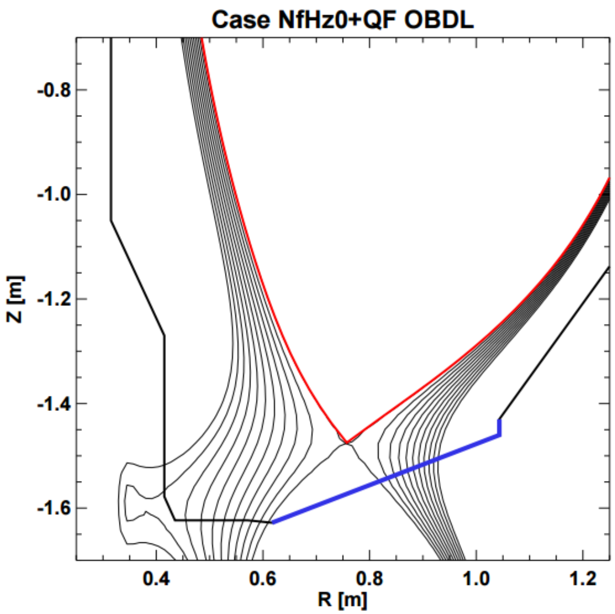


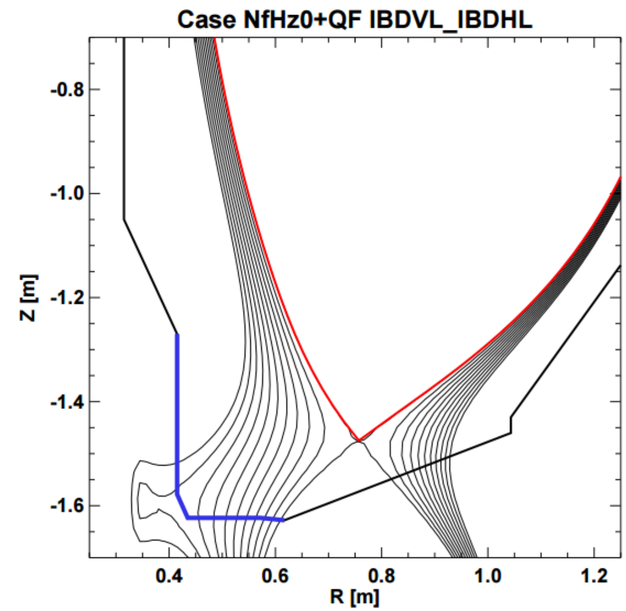


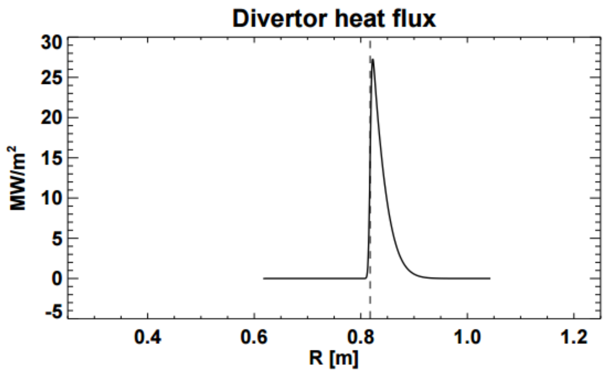
- A = 1.75, κ = 2.74, balanced DN
- I_P = 2MA, B_T = 1T, P_{heat} = 10MW
- $\lambda_{q\text{-mid}}$ =1.97mm, S/λ_q = 0.15
- Poloidal flux expansion = 36
  - Also assume B-field angle of incidence θ_B must be ≥ 1° (tile alignment / leading edge tolerance)
- Radiation fraction = 30%
- 80% of power to outboard
- 50-50 split between upper/lower
- P_{div} ~ 2.8MW to divertor target
- $q_{div\text{-peak}}$ = 7.8 MW/m^2

Example case from 96 with high I_{PF1A}

- **A=1.84, κ=2.5, $\delta_{U,L}$ = 0.193, 0.375, I_{OH}=0, $I_{PF1AU,L}$ = 15, 7kA**

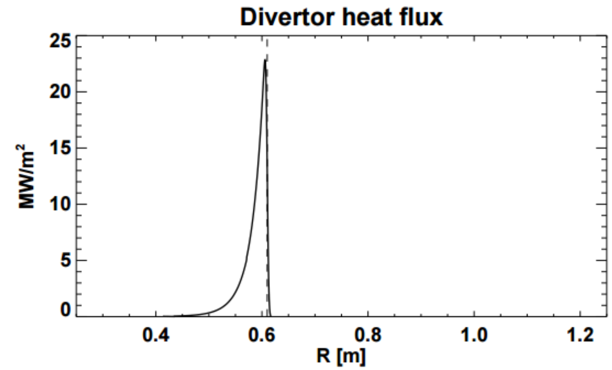






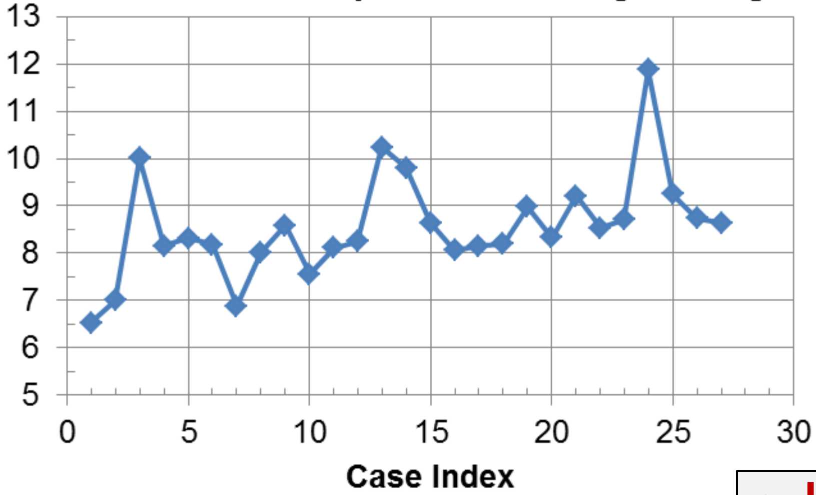
q_{peak} ~20-25MW/m²

Δt_{flat} < 1s without sweeping or other mitigation

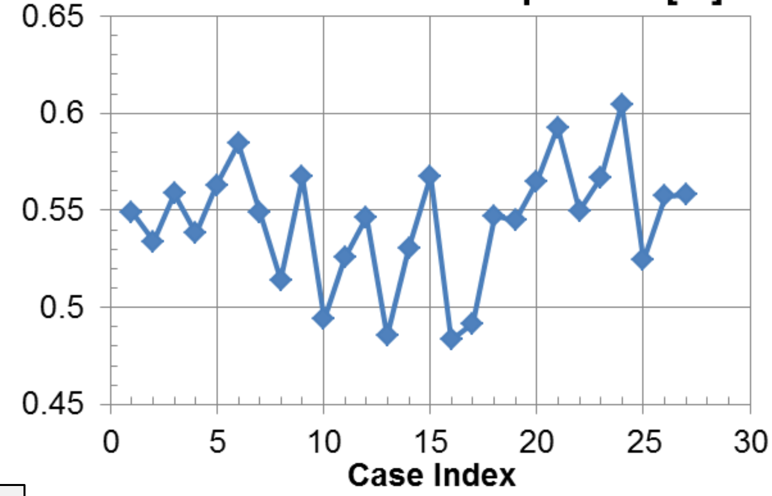


Example: Scan 1: No PF1B, use PF1C for high flux expansion
IBDH tile heat flux projections

Scan 1: IBDH peak heat flux [MW/m²]

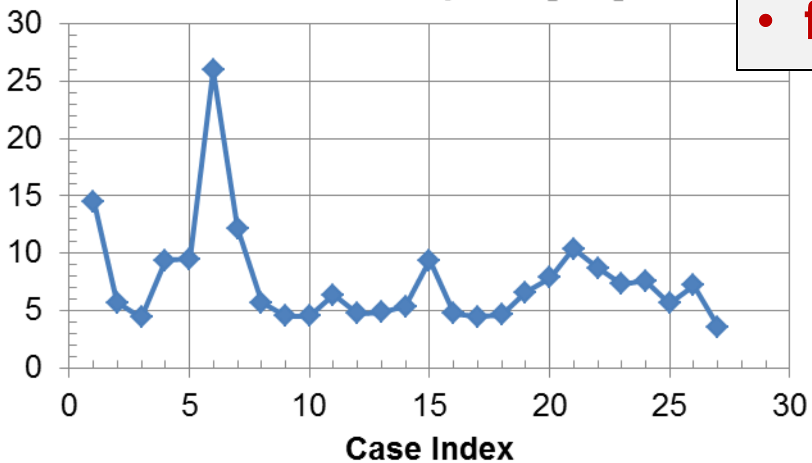


Scan 1: IBDH strike point R [m]

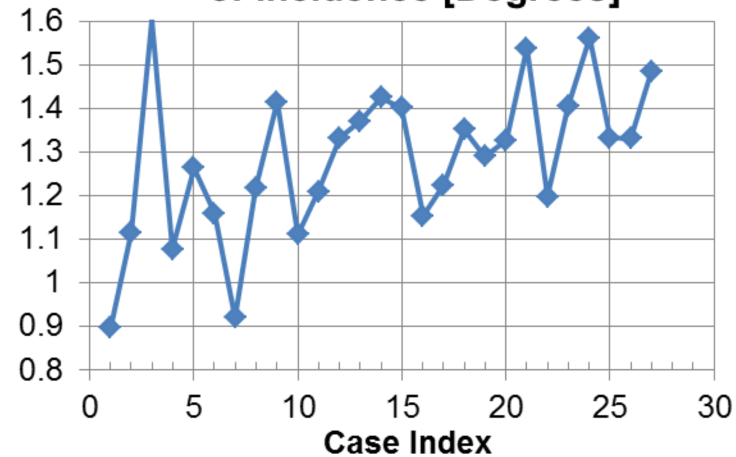


- HD model λ_q
- $S / \lambda_q = 0.15$
- $f_{rad} = 0.3$

Scan 1: IBDH heat flux e-folding width at strike-point [cm]



Scan 1: IBDH total B-field angle of incidence [Degrees]

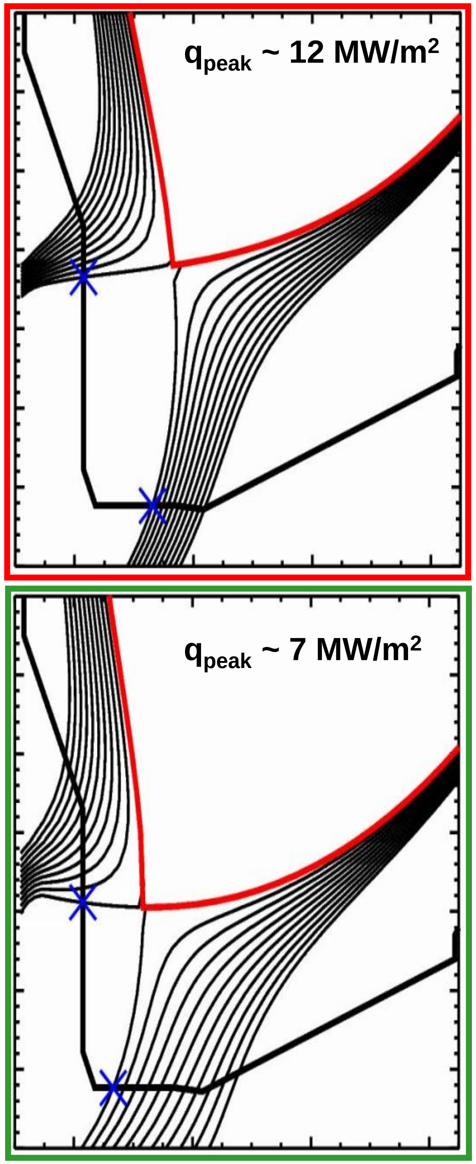


Elongation Impacts Use of Poloidal Flux Expansion

- Low-κ shapes have x-point far from PFCs have useful scientific value
  - I_p = 1 MA, P_{NBI} = 3 MW L-mode
  - flux expansion < 3, field line angles > 10 deg
  - high stationary heat flux (> 6 MW/m^2)

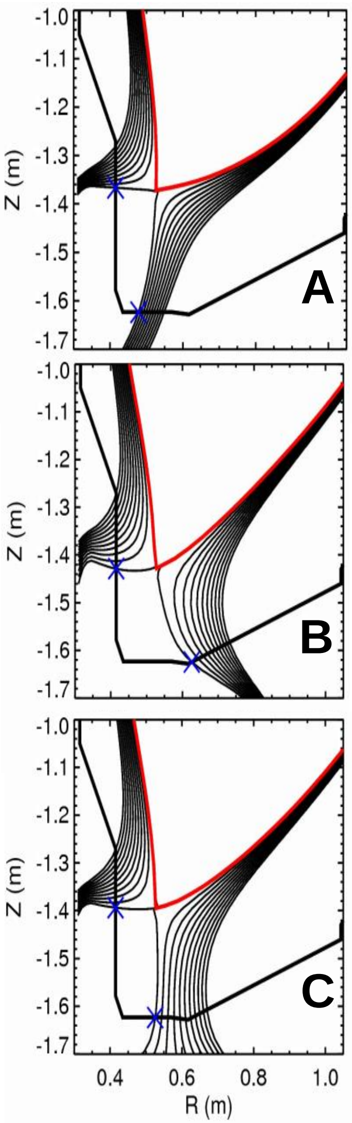
- Increasing kappa and moving x-point closer to targets can mean higher I_p, P_{NBI} are 'easier'
  - I_P=1.00 MA, B_T=0.75 T, 7.5 MW: q_{peak} ~ 12 MW/m^2
  - I_P=1.25 MA, B_T=0.75 T, 8.0 MW: q_{peak} ~ 7 MW/m^2

- Shape/heat flux coupling stronger in STs

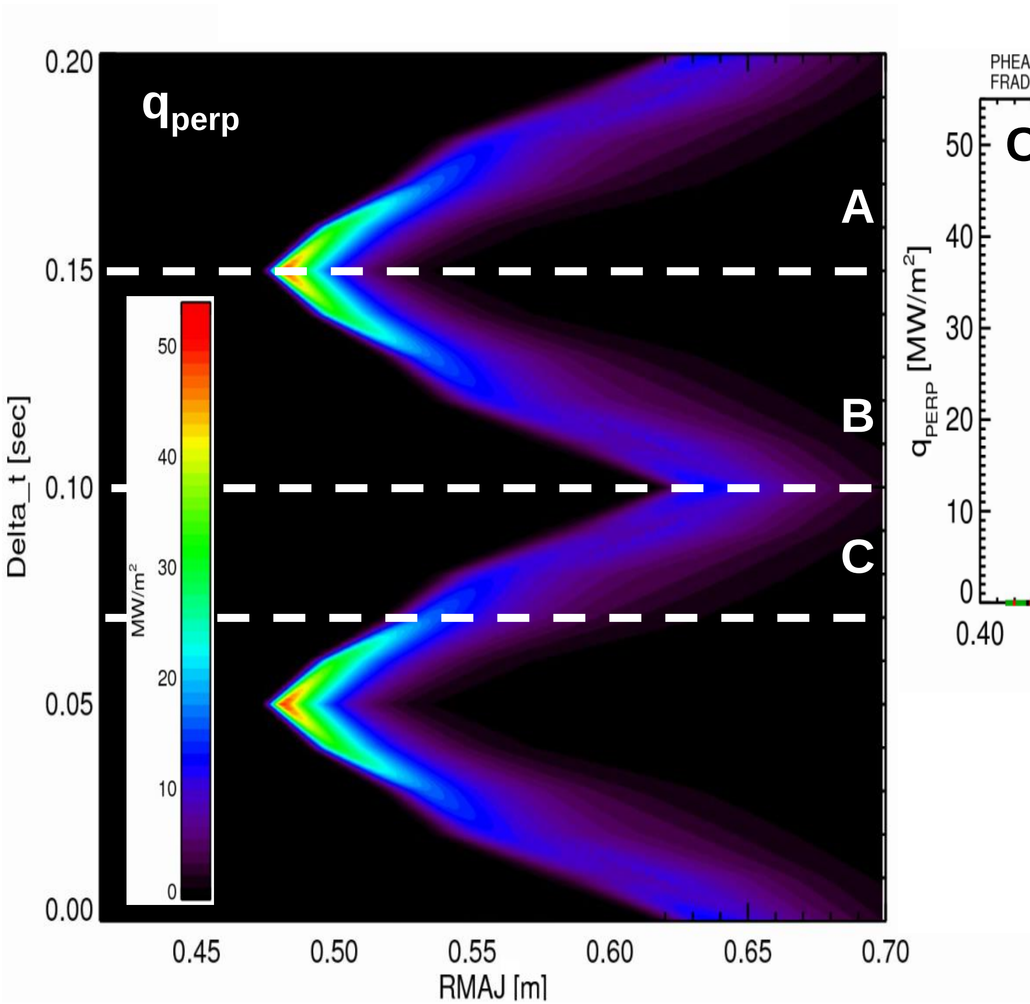


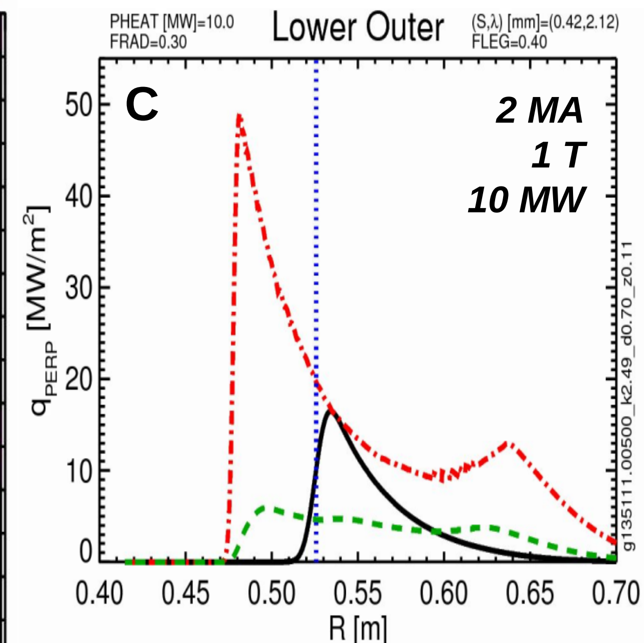
Initial Modeling of Sweeping Shows Benefits

(Implemented in reduced model by M. Reinke, ORNL)



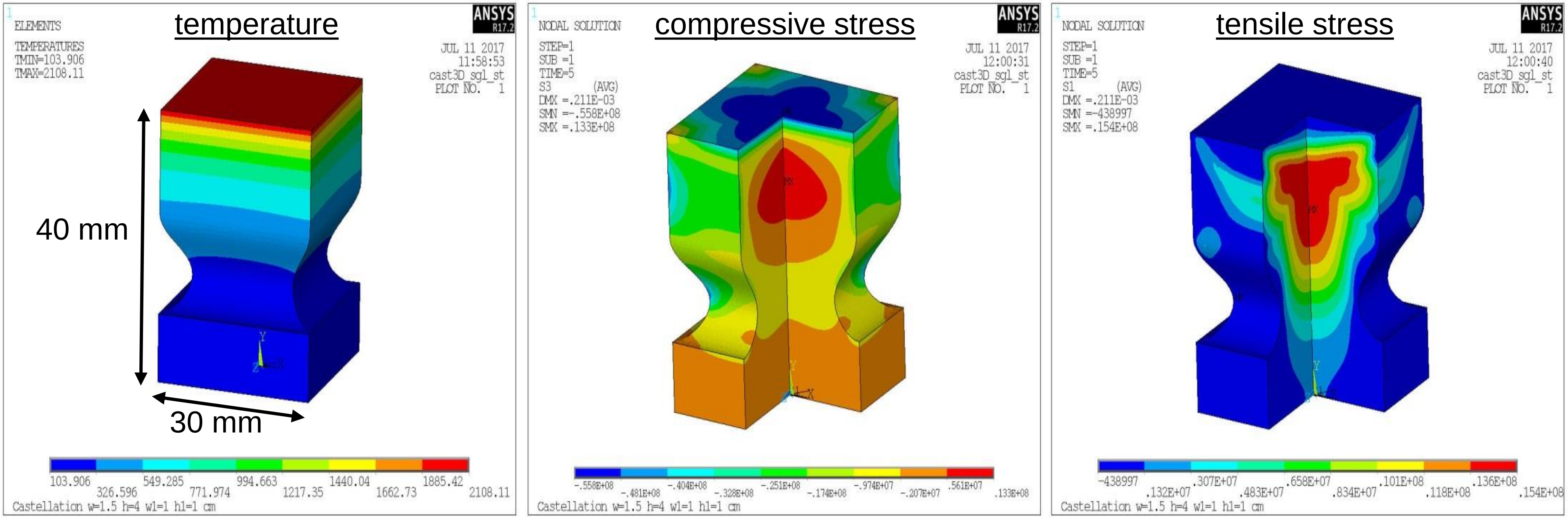
assemble individual equilibria and interpolate





Single Equilibrium
Peak Across Scan
Time Averaged

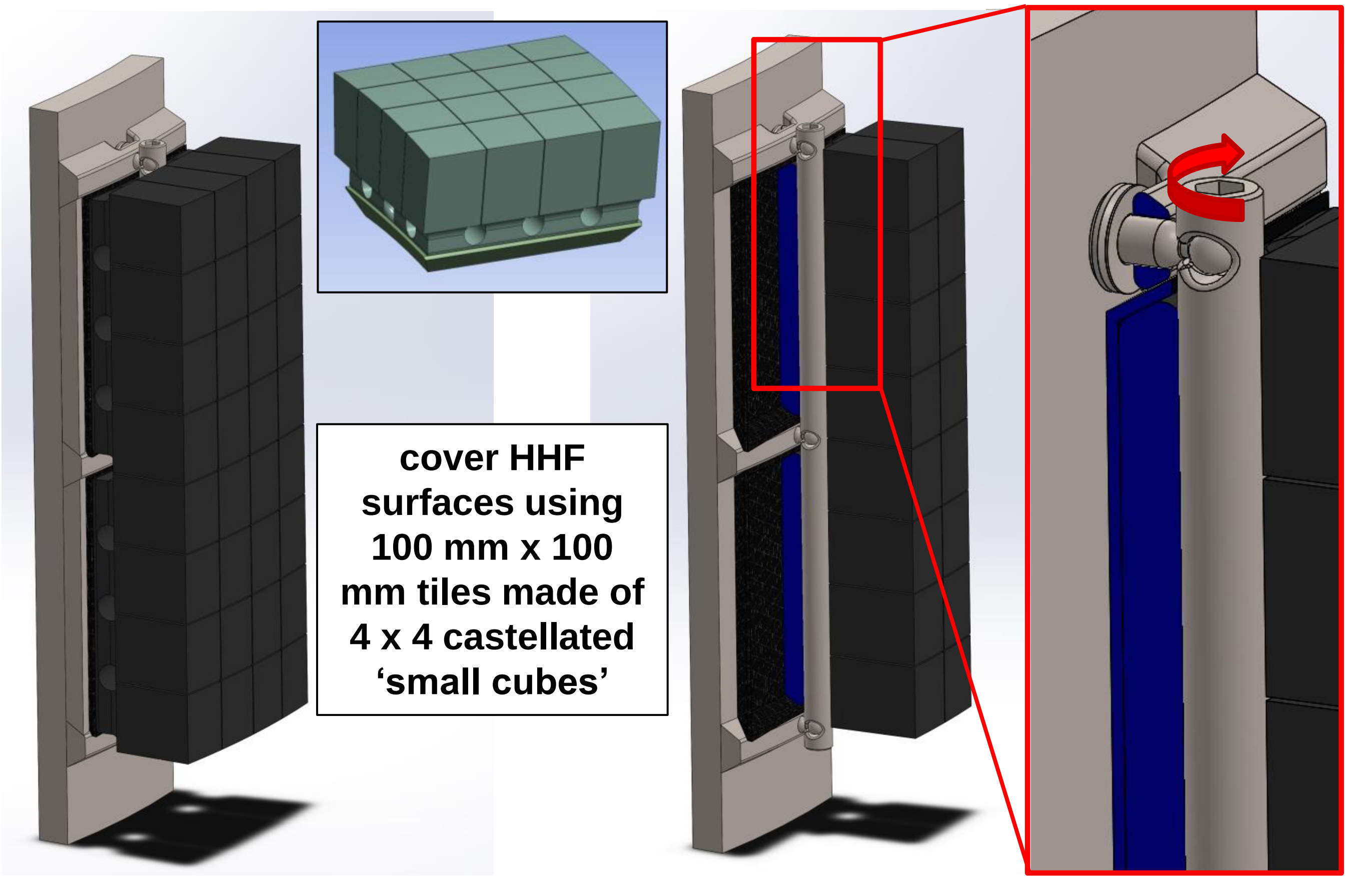
HHF Tile Designs Converged to ‘Small Cubes’



ANSYS simulation of 10 MW/m² for 5 sec onto isotropic graphite at normal incidence

- larger tiles ‘bow’, enhancing stress, small cubes relieve this by ‘mushrooming’
- design criteria using T_{limit} ~ 1600 ^{o}C, and allowable stresses of 50% material limit
- scoping simulation show T_{max} = 2100 ^{o}C, max compressive stress of 55.8 MPa (86% of allowable) and max tensile stress of15.4 MPa (51% of allowable)
  - example of design that is ‘temperature limited’ and not ‘stress limited’

Developing Designs Using Castellated Tiles



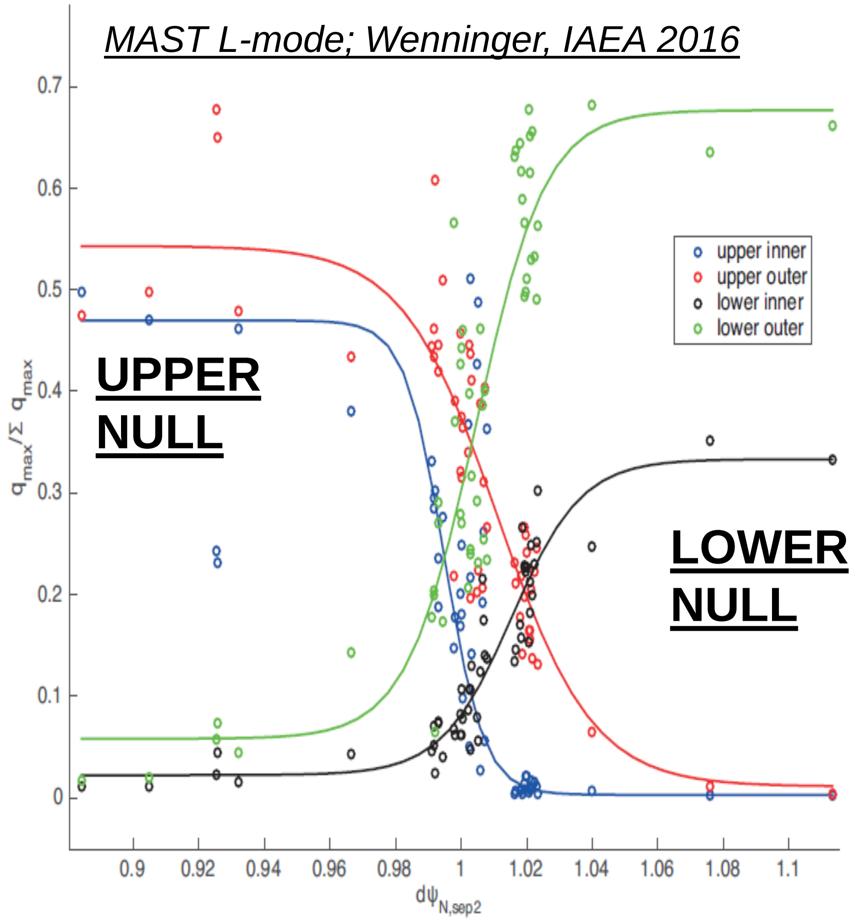
example vertical divertor

- Designs working to avoid front-surface holes in HHF regions
- Side-access through removing low-heat flux tiles w/ front surface holes
- Ex: **cam-like action** secures tiles against mounting plate/vessel
- Beginning designs for inter/intra-tile diagnostics

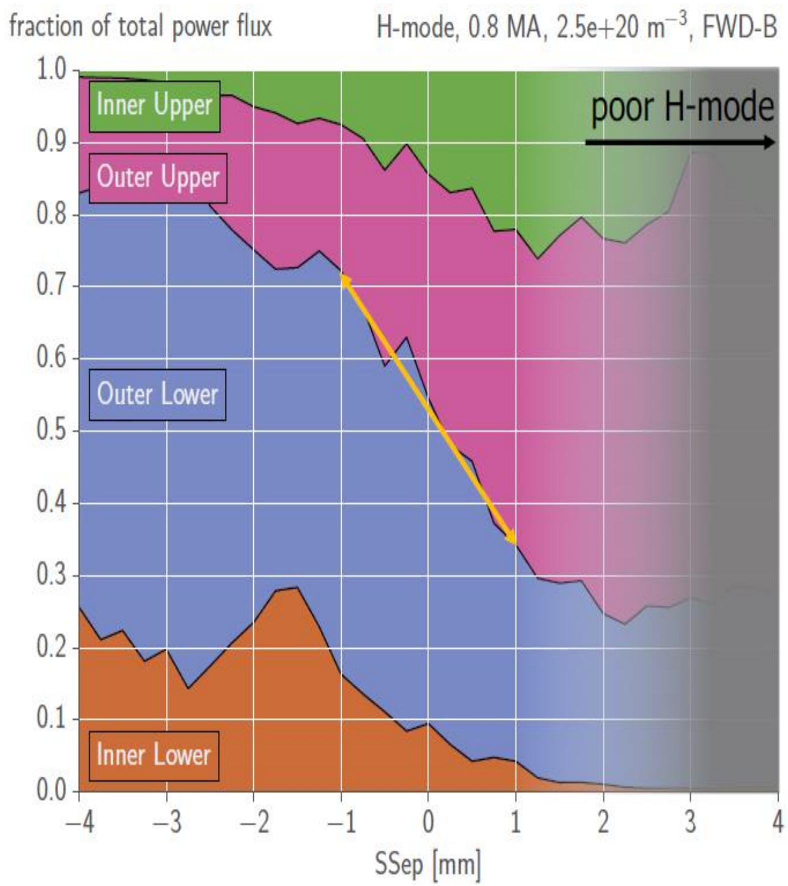
Empirically Motivated Divertor Power Sharing
(Implemented in reduced model by M. Reinke, ORNL)

- NSTX only measured the outer, lower divertor heat flux
  - model uses inner/outer split of 70/30 LSN and 55/45 USN, smooth transition in-between

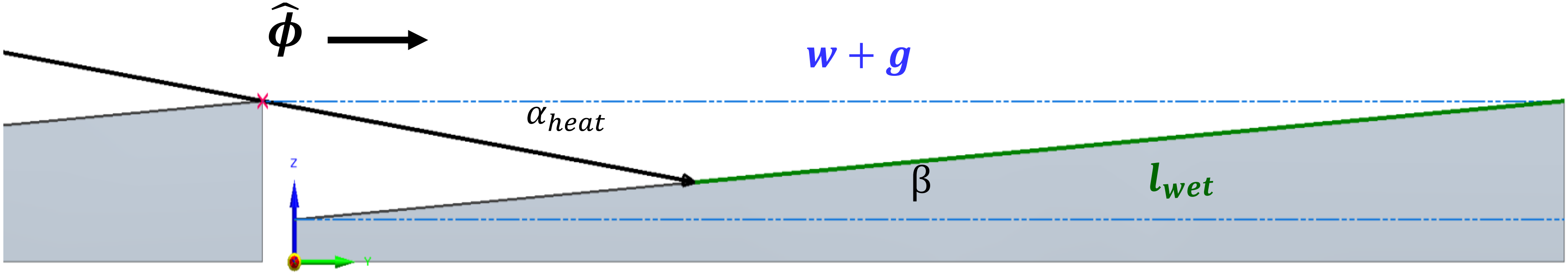
MAST L-mode; Wenninger, IAEA 2016

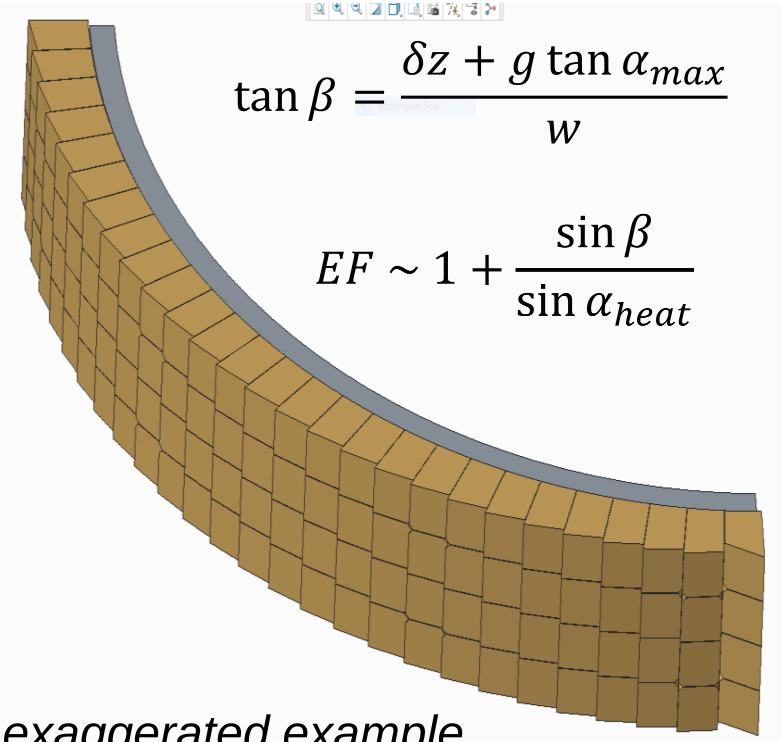


C-Mod H-mode; Brunner APS 2016



Tile Shaping Used to Avoid Leading Edges





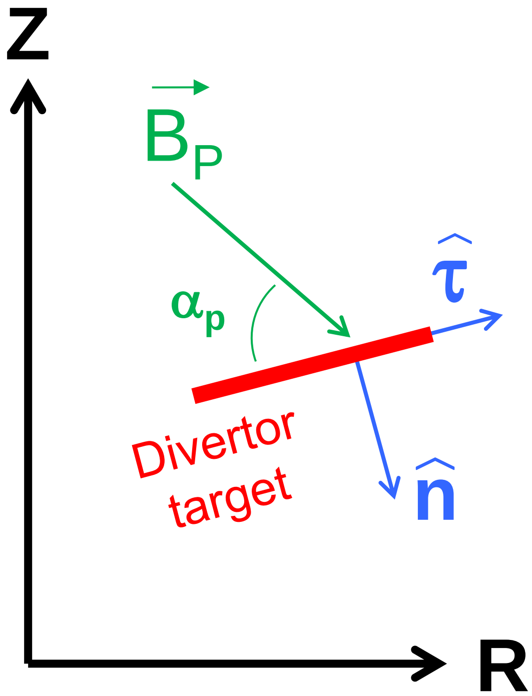
$$\tan\beta = \frac{\delta z + g\tan\alpha_{max}}{w}$$

$$EF \sim 1 + \frac{\sin\beta}{\sin\alpha_{heat}}$$

exaggerated example

- Small ramp toroidally, $\beta \sim 1^o$
  - driven by max field line expected, α_{max}
  - impacted by alignment uncertainty, δz
- This reduces wetted area (increases $\vec{B} \cdot \hat{n}$) : 'enhancement factor (EF)'
- Large poloidal expansion leads to shallow attack angles, $\alpha_{heat} \sim 1^o$

Method for simultaneously controlling strike-point position and angle of incidence



- $q_{divertor} = q_{||0}\hat{q}(\hat{\psi})(B/B_0)\sin(\theta_B)$
- $q_{divertor} = q_{||0}\hat{q}(\hat{\psi})(B/B_0)(\hat{n}\cdot\hat{b})$
- $q_{divertor} = q_{||0}\hat{q}(\hat{\psi})(B_\phi/B_0)\tan(\alpha_{heat})$
- $\tan(\alpha_{heat}) \equiv B_n/B_\phi \qquad B_n \equiv \vec{B}_P\cdot\hat{n}$
- Can also show that: $B_n = R^{-1}\partial\psi/\partial\tau$
- $\Rightarrow \partial\psi/\partial\tau = RB_\phi\tan(\alpha_{heat}) = \text{constant}$
- $\Rightarrow$ constrain flux gradient at strike-point
- also constrain poloidal flux at strike-point

Implemented in PPPL IDL-ISOLVER: Works well with all 3 divertor PF1 coils

Summary

- NSTX-U redesign of high-heat-flux PFCs motivated by projected narrowing of SOL at high I_P and improved understanding of expected halo-current loads
- Systematic equilibrium scans used to quantify the required divertor coil currents and to verify vertical stability for a range of plasma shapes
- Tile designs developed to handle high heat fluxes
- Free-boundary control schemes developed to constrain the strike-point location and field-line angle-of-incidence → supports use of fish-scaled tiles
- Can support core confinement studies at the highest current, power, duration projected for NSTX-U