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SMR 1161/13

AUTUMN COLLEGE ON PLASMA PHYSICS

25 October - 19 November 1999

The Physics Basis for Spherical Tokamak Research

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These are preliminary lecture notes, intended only for distribution to participants.

The Physics Basis for Spherical Tokamak Research

Presented by J. Menard

(with **many** contributions from the NSTX Team)

Autumn College on Plasma Physics

ICTP

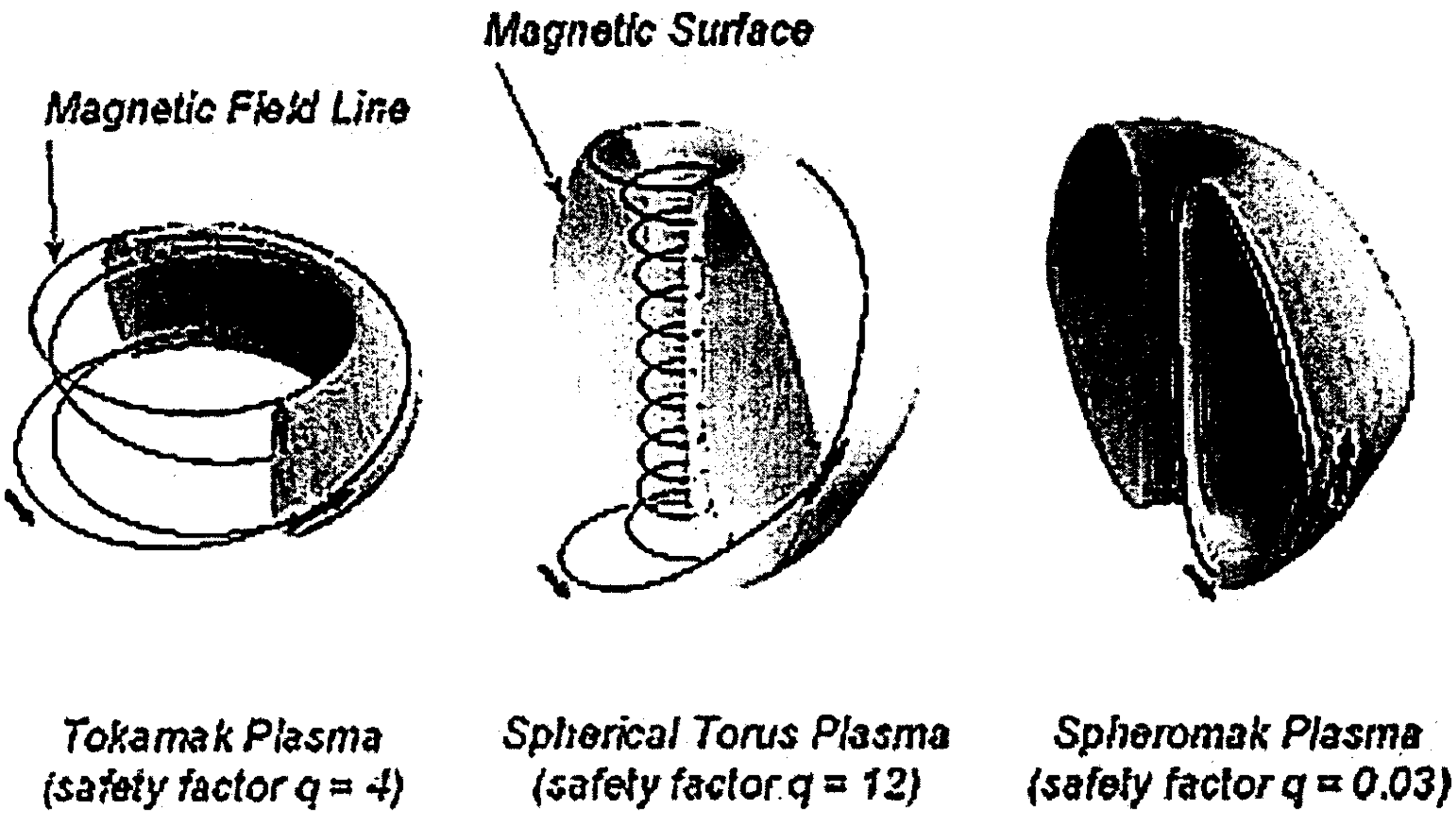
Trieste, Italy

November 1st, 1999

PPPL

PRINCETON PLASMA PHYSICS LABORATORY

The distinguishing feature of the spherical tokamak/torus (ST) is its unique field-line topology



The goal of ST research is to achieve the good confinement properties of the tokamak in a more magnetically efficient configuration (high-β)

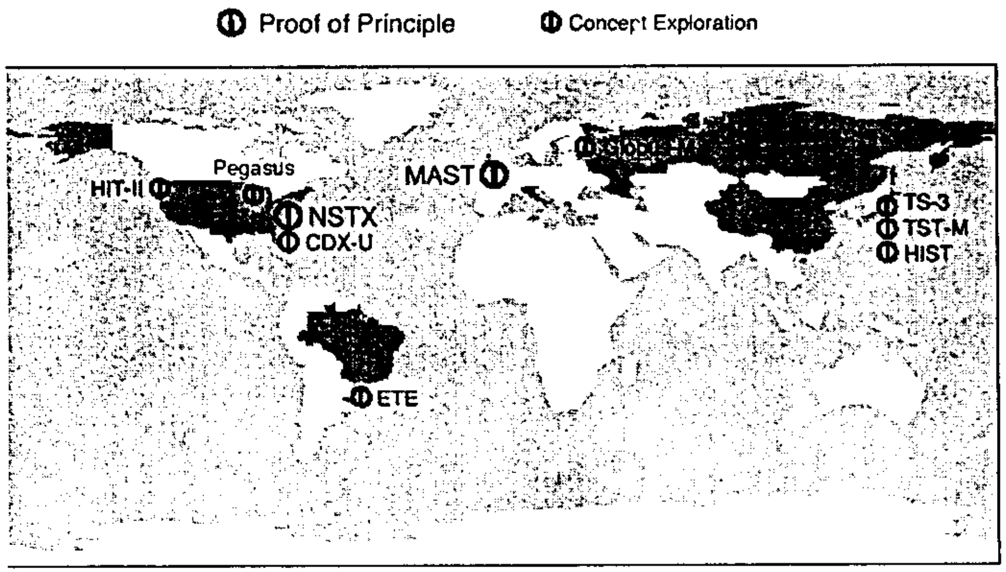
Spherical Tokamak Characteristics

- Low aspect ratio configuration: major radius/minor radius = **R/a ~ 1.1-1.6**
  - Leads naturally to high $\beta = 2\mu_0\langle p\rangle/B^2$ configuration $\Rightarrow$ more efficient reactor?
  - Configuration not yet studied at 1MA current level

- Different regimes/operational approaches to those of conventional R/a
  - Non-inductive start-up and current drive even more crucial
  - Potential enhancements in confinement from large flow shear
  - High-$\beta_{toroidal}$ and β_{Normal}
    - Improved MHD stability with and without conducting shell
    - Formation of absolute magnetic well
    - high bootstrap fraction possible
  - large mirror ratio, trapped particle fraction
  - high edge safety factor q_ψ at high current
    - reduced disruptivity?
  - Power handling issues in next step and reactor-sized device are serious

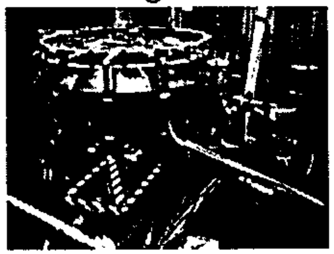
- Small STs have already realized success:
  - START (β_t = 40% [NBI], = 24% [OH], H-mode), CDX-U (RF studies), HIT-II (CHI)

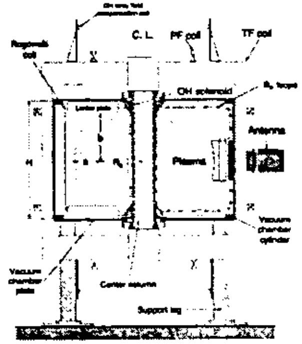
STs around the world

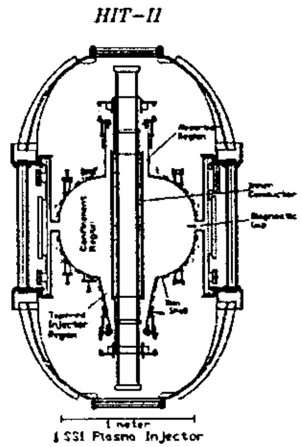


MAST

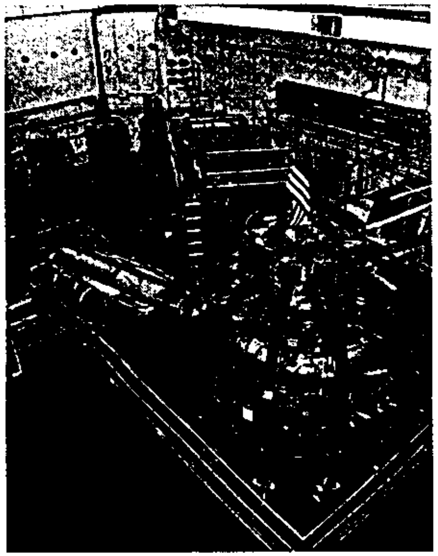
Pegasus





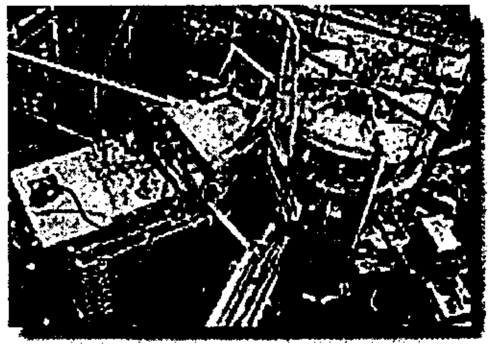


HIT-II



NSTX

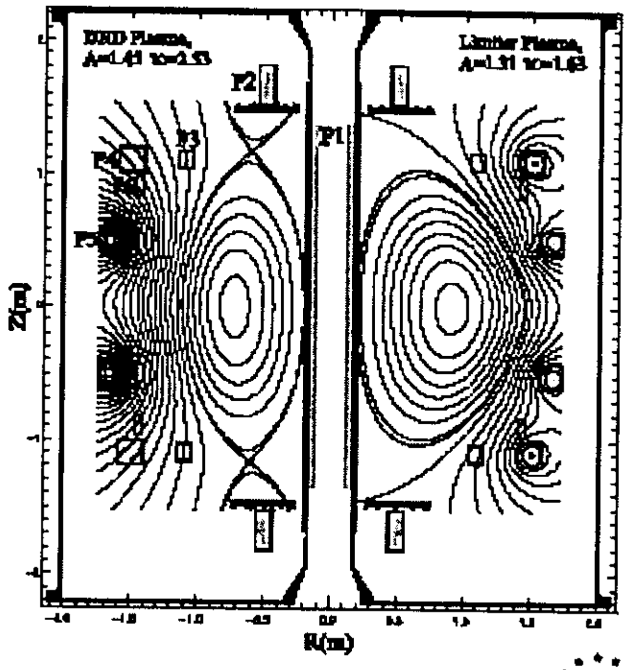
CDX-U



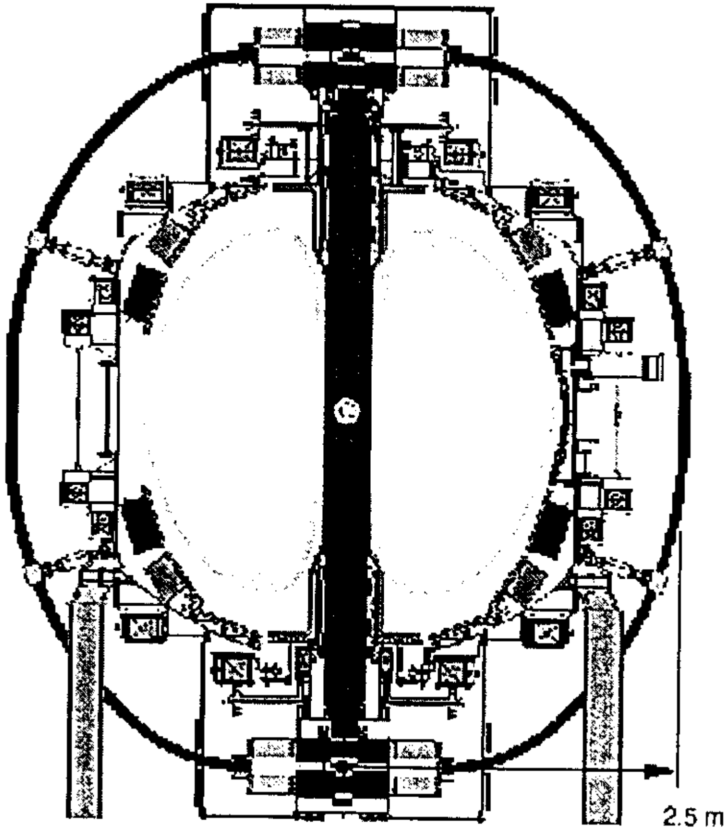
START

MAST and NSTX have complementary capabilities

MAST (U.K.)



UKAEA Fusion



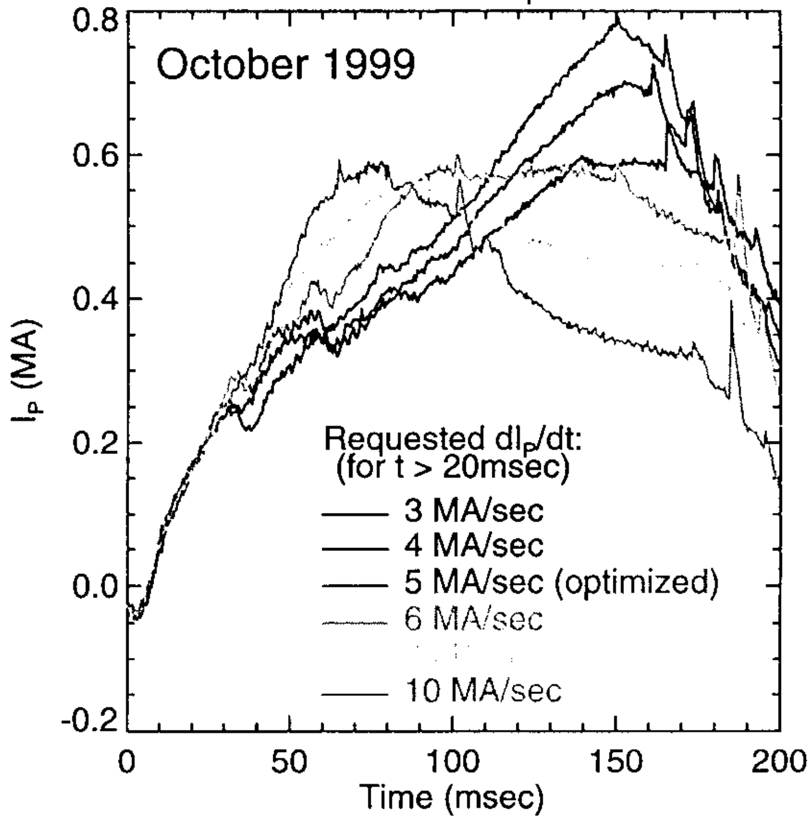
• Nearby Stabilizing Shell	No	Yes	(beta limits)
• Poloidal Field Coils	In-vessel	Ex-vessel	(plasma shaping flexibility)
• RF Heating&Current Drive	ECH	HHFW	(efficient sustainment)
• Plasma Current Startup	Compression	CHI	(eliminate solenoid)

⇒ Development of comprehensive database for Performance Extension step

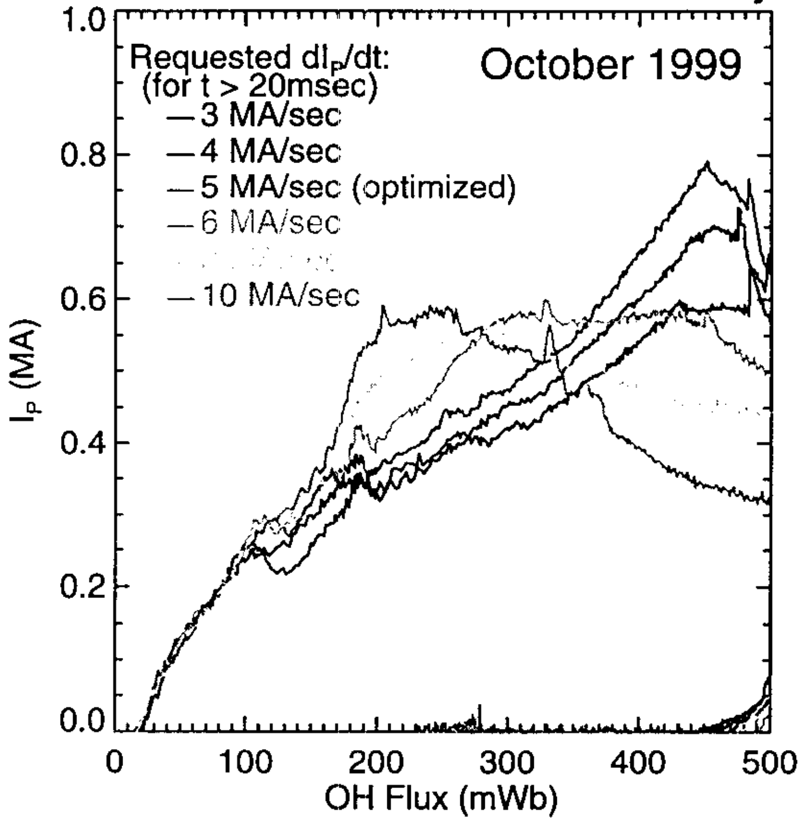
NSTX has achieved I_P=800kA using 75% of OH flux ⇒

can reach 1MA with short flat-top (as designed)

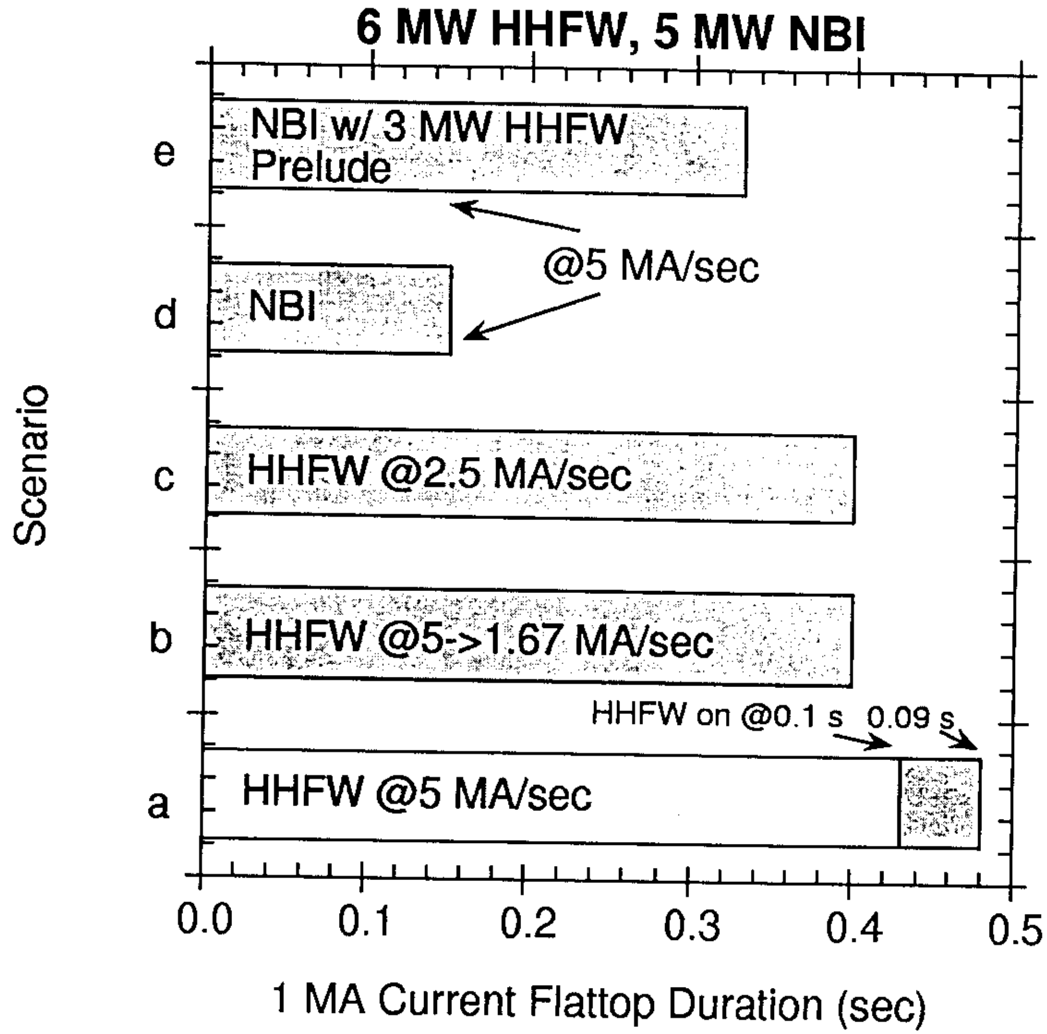
Maximum sustained dI_P/dt < 6MA/sec



450mWb of OH flux used efficiently



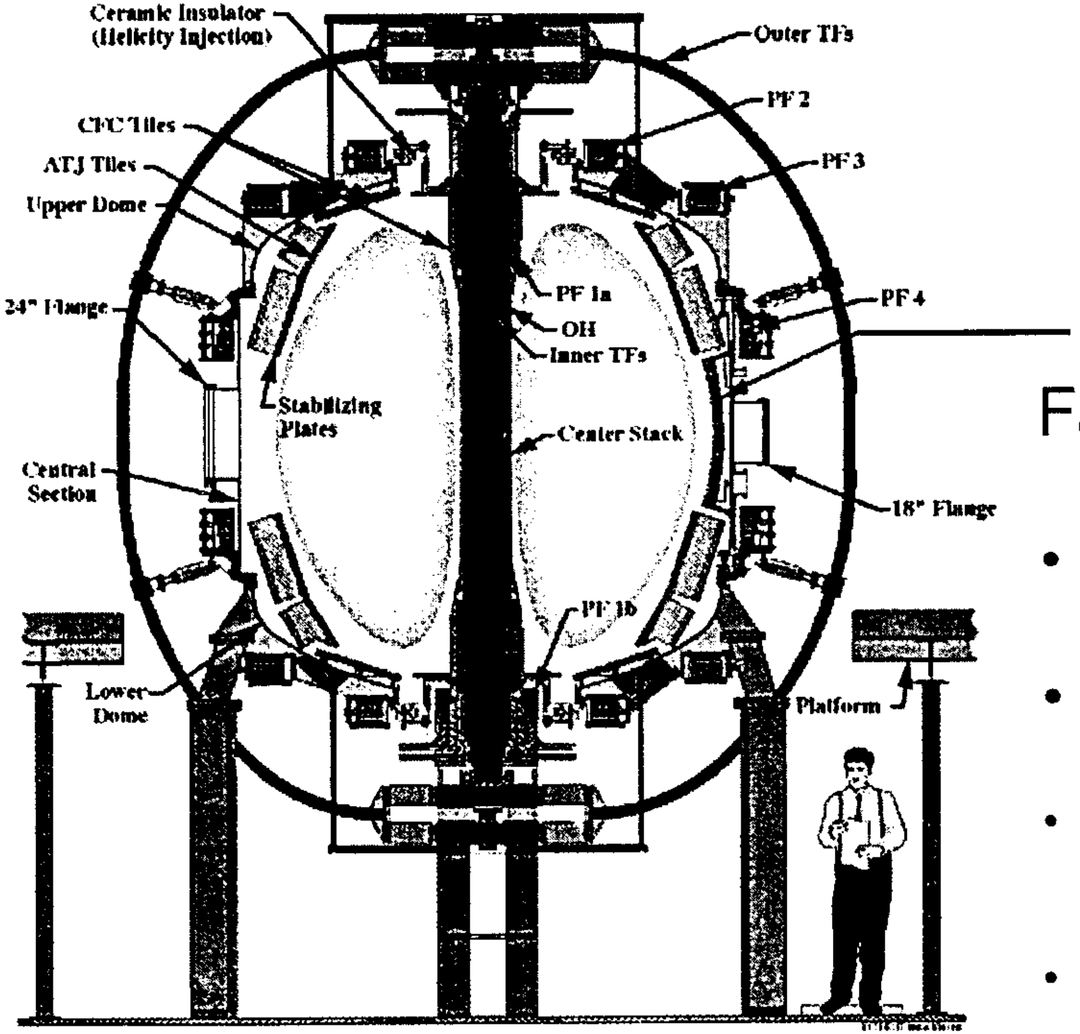
TSC simulations show heating during I_P ramp can significantly lengthen I_P flat-top



TSC = Tokamak Simulation Code (S. Jardin) - simulations by S. Kaye

The NSTX HHFW System

In collaboration with the Oak Ridge National Laboratory RF technology group



High-Harmonic
Fast Wave Antenna

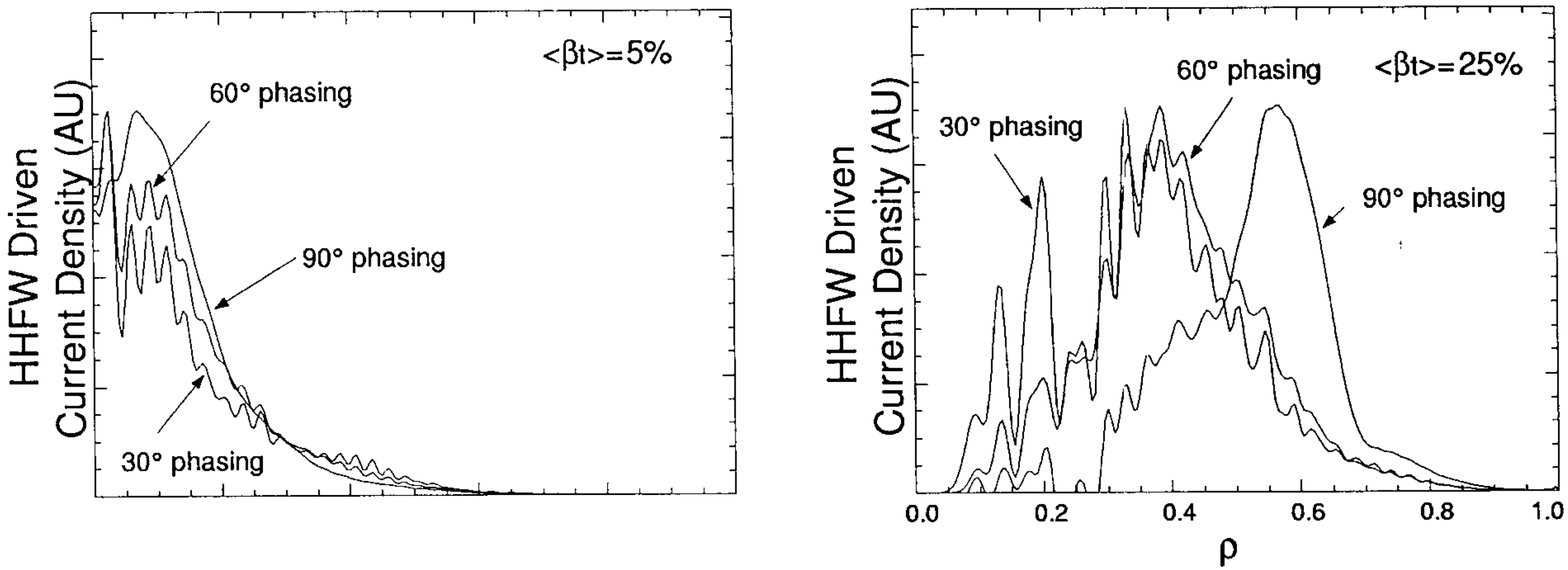
- 6MW at 30MHz
- 12 straps, 6 transmitters
- Antenna covers 90° of outer mid-plane
- Maximum $k_{||}$ of 15 m^{-1} (n_{ϕ}=24)

How will NSTX try to sustain high-β 1MA discharges and start up non-inductively?

- **6MW** of 30MHz high-harmonic fast waves
  - Should damp far off axis at high β with strap-phase controllable deposition location for heating and CD.
  - Pre-heating during the Ohmic ramp-up should reduce VS consumption while raising $\beta \Rightarrow$ longer flat-top.
  - HHFW should be able to couple to low T_e (300eV) plasmas for current drive and for BS over-drive start-up experiments, possibly producing a usable NBI target.
  - Broad p profile from off-axis heating may optimize MHD.
  - CDX-U experiments $\Rightarrow$ efficient coupling to HHFW even with significant misalignment between straps and B.

- ISSUES: Central heating virtually impossible at high β, thermal ion (and beam ion) wave absorption may be strong at sufficiently high ion temperature.

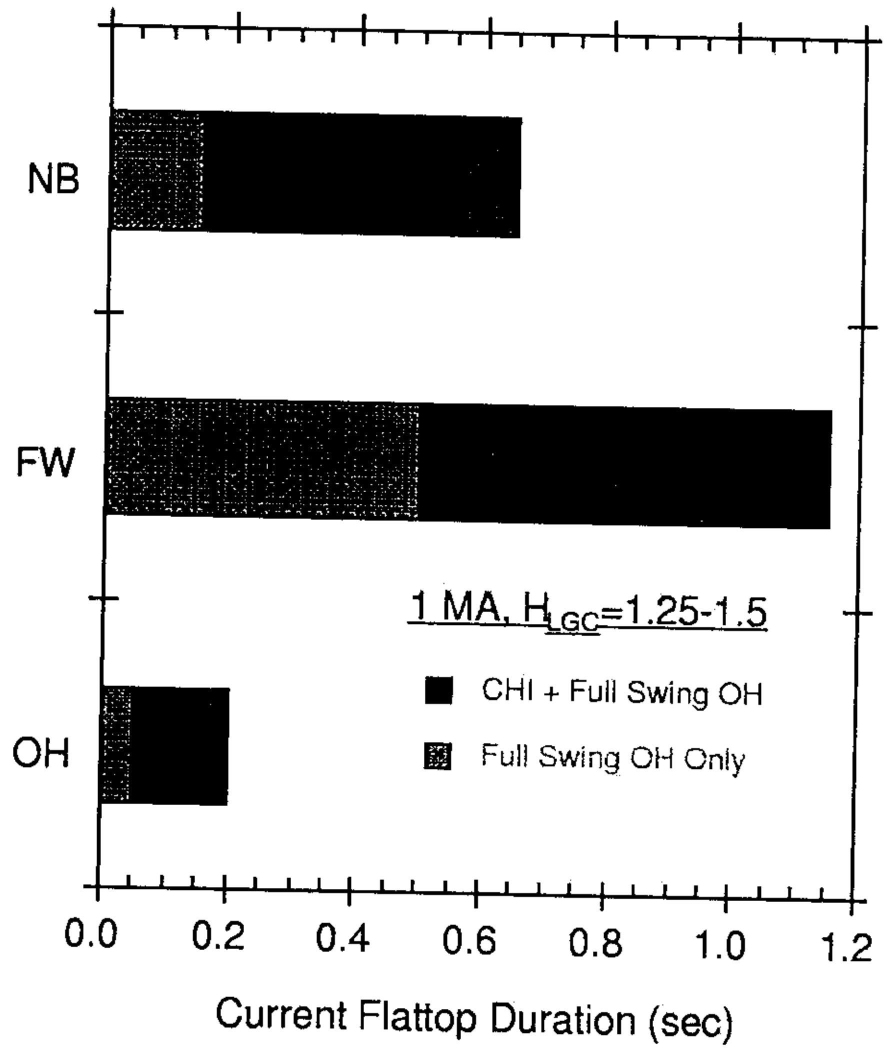
Simulations predict central heating and CD at low β, controllable off-axis deposition at high β



Results from PICES code - ORNL

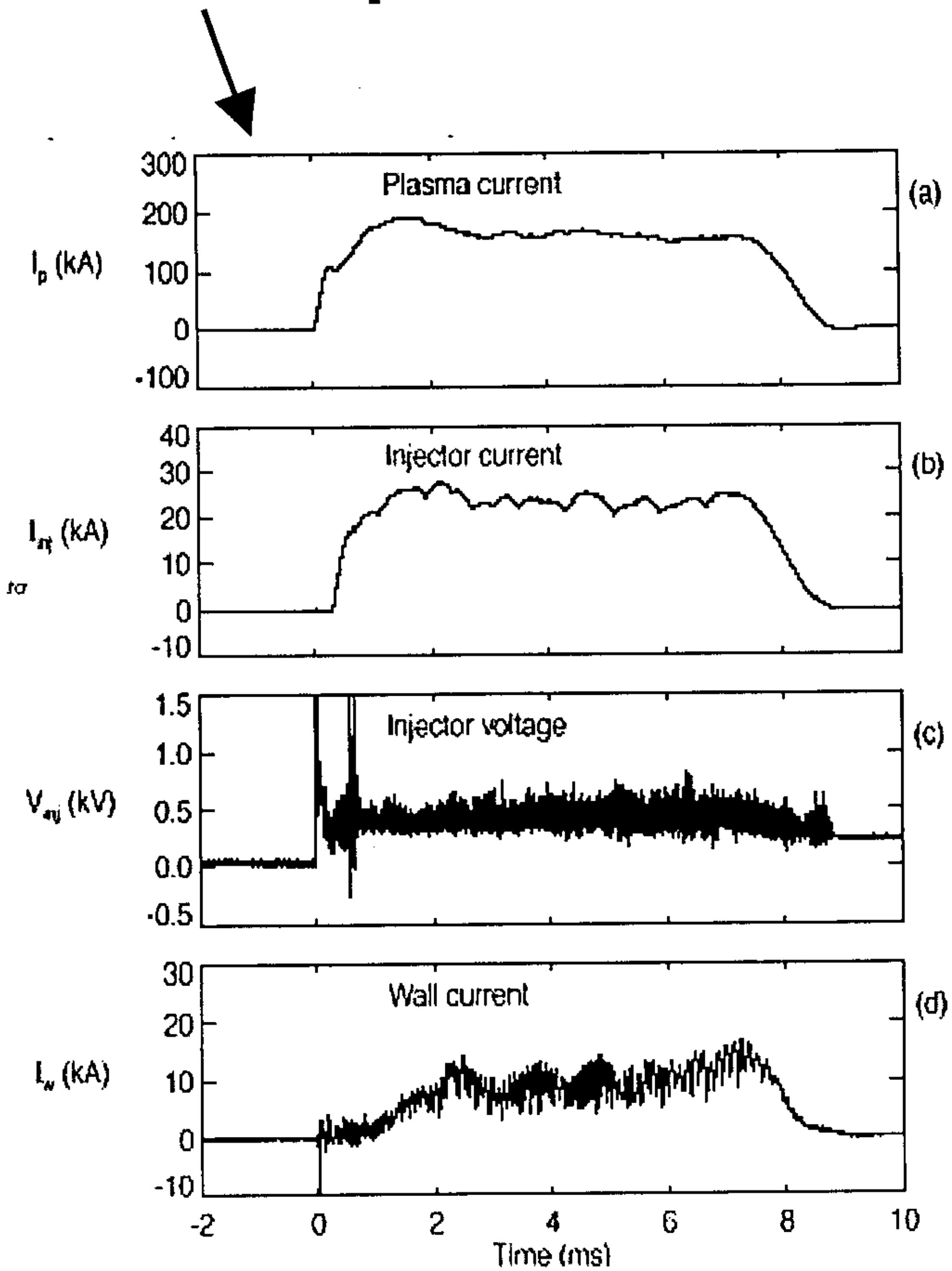
Single-pass fast wave absorption on electrons is a new regime for fusion experiments

Formation of 500kA Coaxial Helicity Injection (CHI) plasma could further and significantly lengthen the I_P flat-top in NSTX

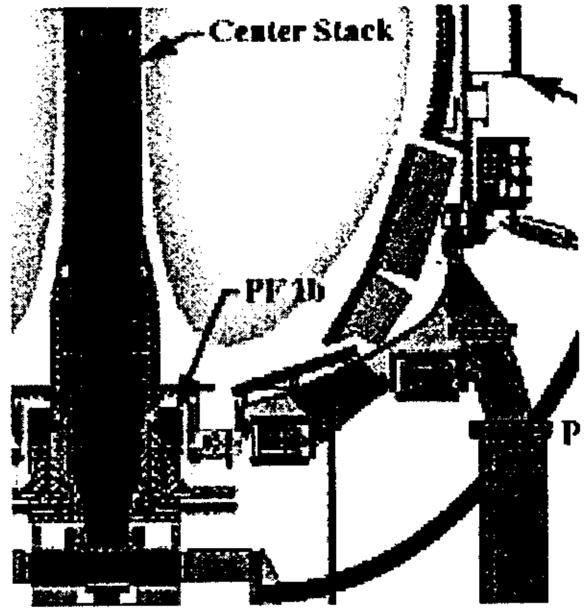


Results from TSC code - S. Kaye

HIT-II experiment has driven 150kA of CHI current



Fully non-inductive CHI startup to be first attempted on NSTX during November 1999.

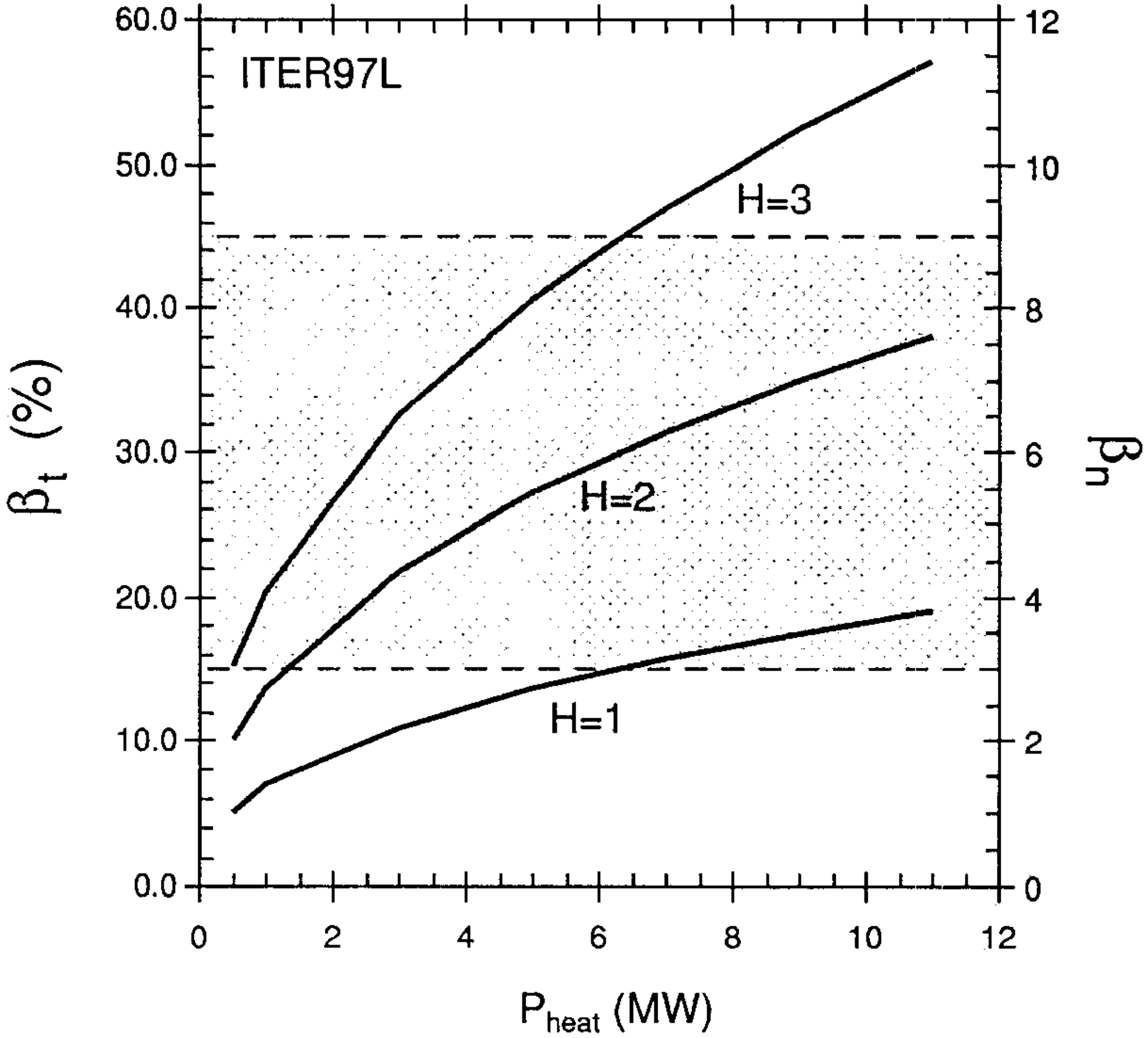


Other possible start-up techniques:
- ECH / EBW
- Inductive Compression

How will NSTX try to sustain high-β 1MA discharges and start up non-inductively?

- Coaxial Helicity Injection (CHI) plasma formation
  - How does it work?
    - First, the NSTX center-stack casing and inner divertor plate can be biased:
      - up to 1kV on NSTX
      - Injection current up to 50kA on NSTX
    - Lower single null divertor coil magnetically connects inner and outer divertor plates
      - bias drives force-free currents along B
      - $\Rightarrow$ toroidal current is generated.
    - After the current channel grows away from the injector,
      - hollow current profile flattens to a minimum energy state through reconnection
      - reconnection mediated by a large n=1 mode
    - The injector to plasma current gain is expected to be as high as ten
      - $\Rightarrow$ up to 500kA of toroidal plasma current.
- ISSUES:
  - Impurity introduction due to sputtering.
  - Lack of good magnetic surfaces and impurities $\Rightarrow$ cold, dense plasma.
  - The hope is that CHI plasmas will be hot enough to provide a HHFW target.
  - CHI into an already formed OH plasma will also be tried.

Both HHFW and NBI power may be needed to test stability limits, depending on confinement

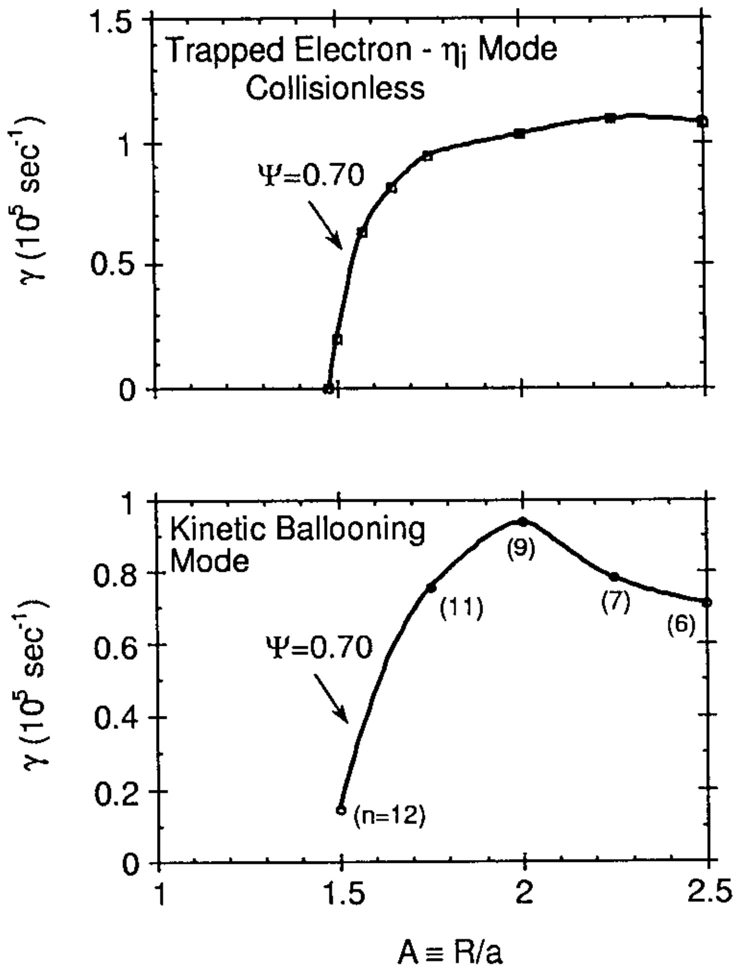


How will NSTX try to sustain high-β 1MA discharges and start up non-inductively?

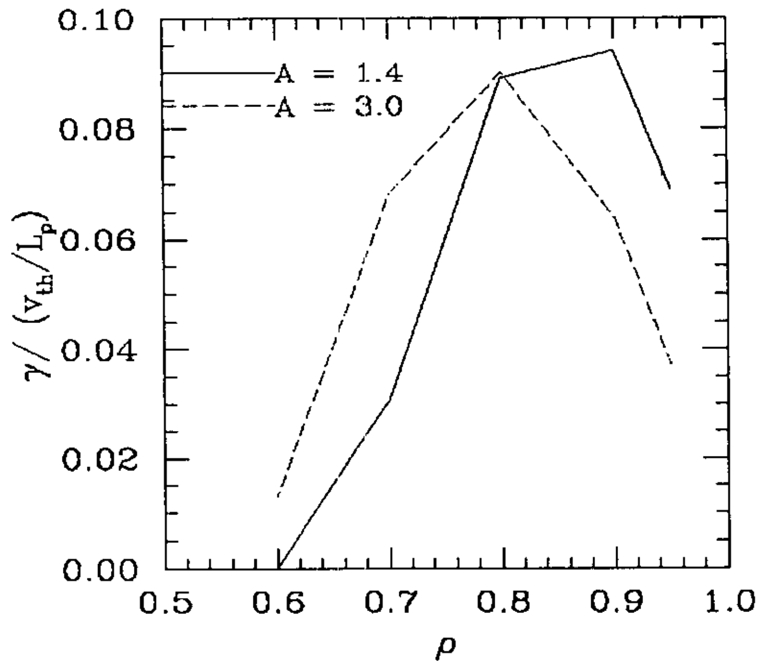
- **5MW** of 80keV Neutral Beams
  - Deposition profile should be centrally peaked and NBI should reliably heat Ohmic plasmas (NBI more proven than HHFW).
  - Counter losses so large only co-injection can be used. Can also be used for pre-heating during the Ohmic ramp-up but not as early as HHFW. Can also try BS+beam CD+HHFW ramp-up experiments.
  - Strong driven rotation should enhance ExB flow, turbulence suppression, and MHD stabilization.
- ISSUES: Peaked p profile not optimal for wall stabilization, fast particle modes(?), cannot separate rotation from heating.

Will low aspect ratio have improved confinement?

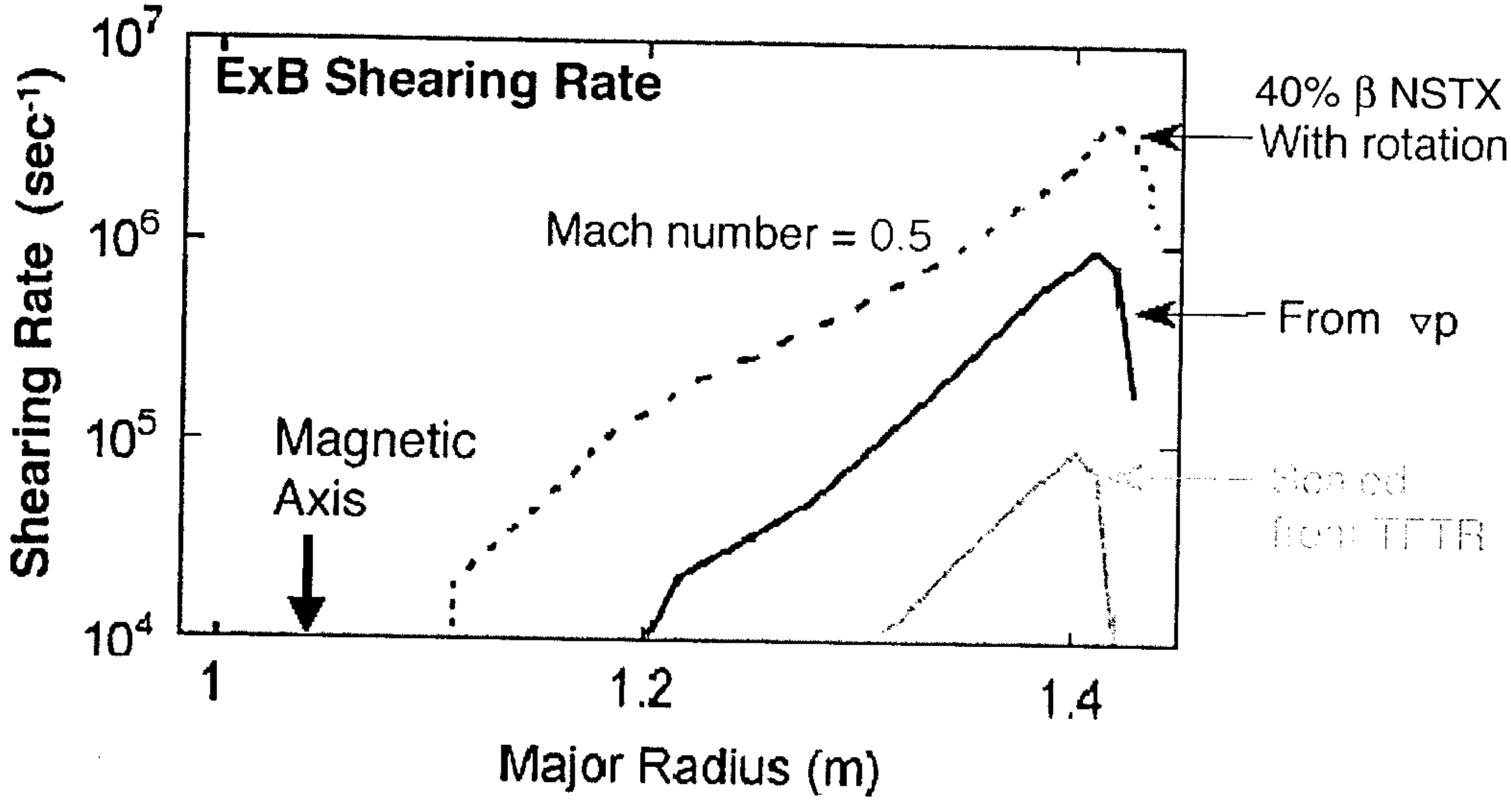
Suppression of micro-instabilities due to decrease in orbit-averaged bad curvature predicted (Rewoldt, 1996)



But, ITG growth rates comparable **near β-limit** (Kotschenreuther et al., 1999)

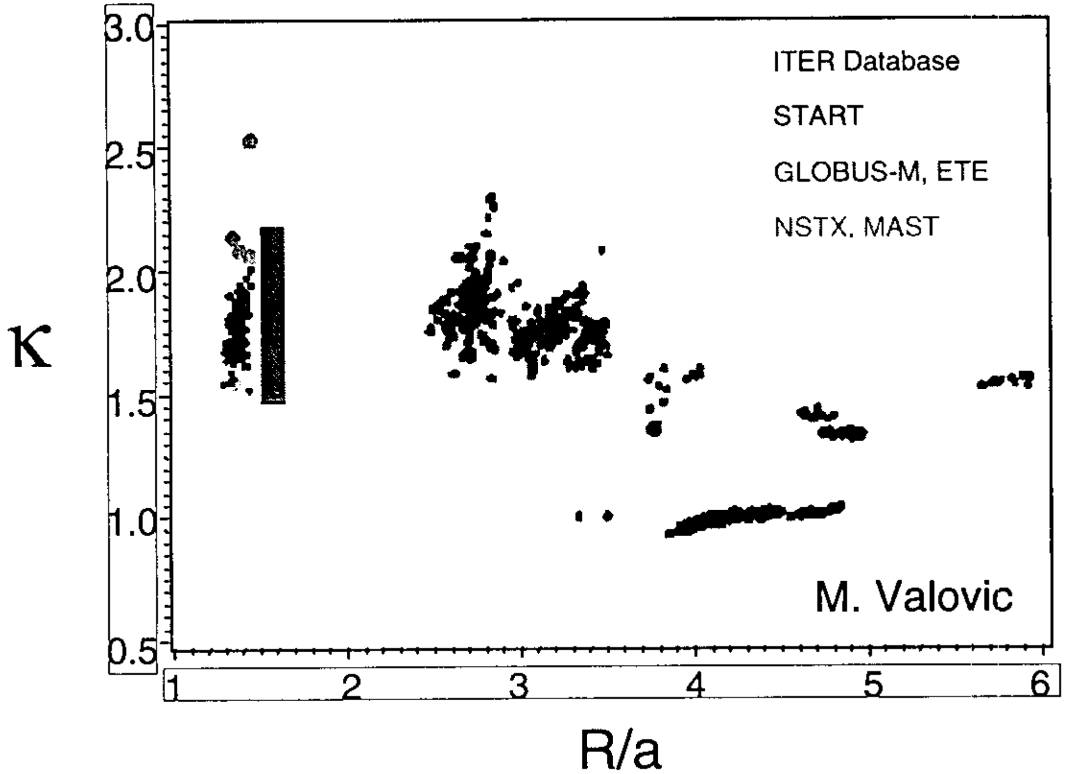


Consensus that flow-shear suppression of turbulence occurs when $\omega_{ExB} > \gamma$



Potential reduction in anomalous cross-field transport at low-A

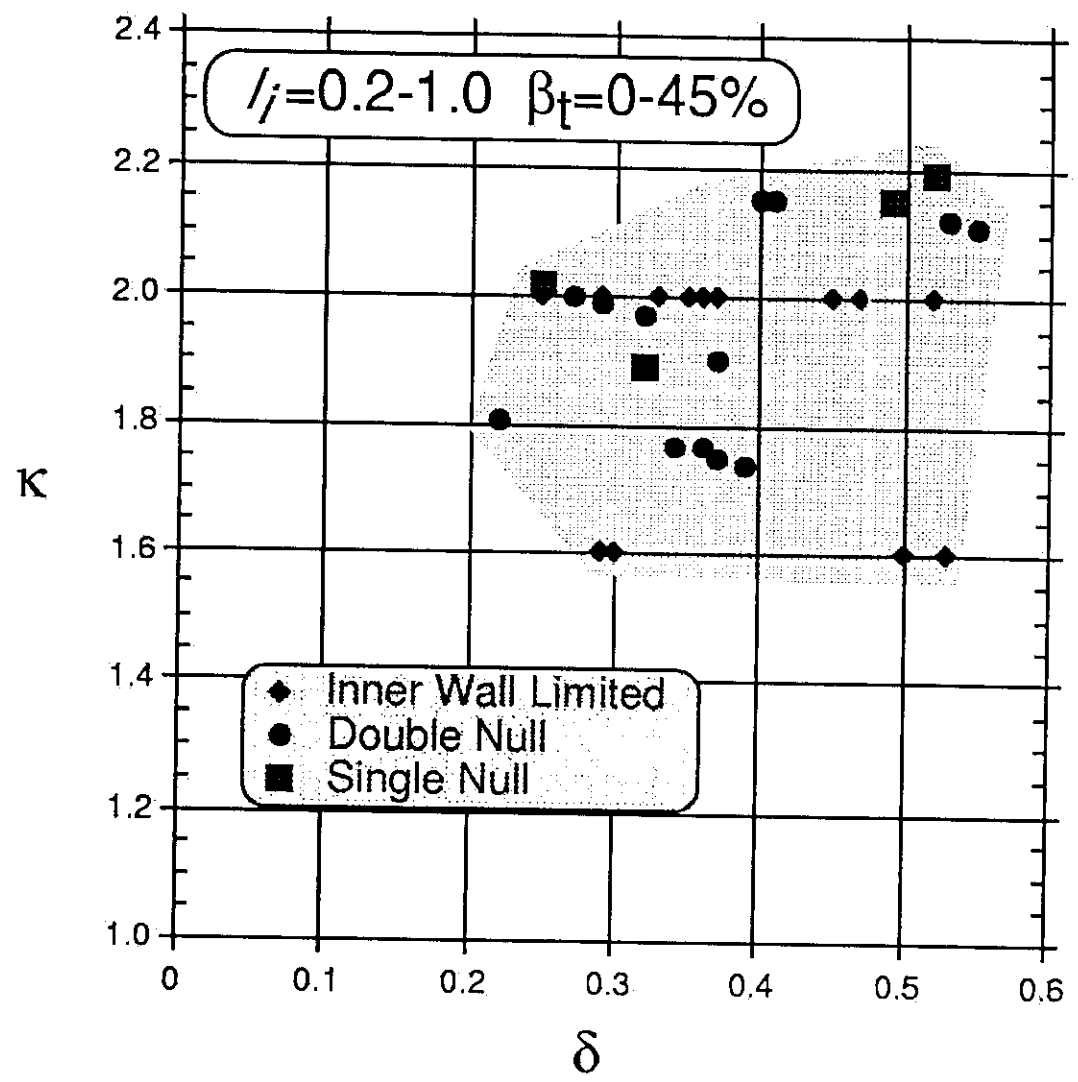
NSTX & MAST experiments will contribute to understanding of confinement and threshold physics



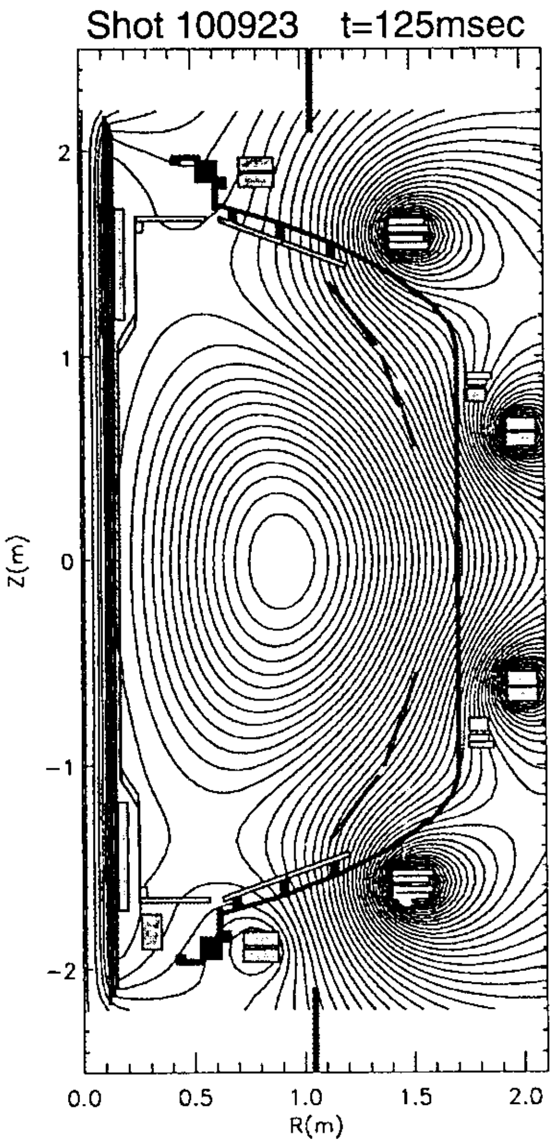
START results raise some concern

- Achieved H-mode access, but only with $P >> P_{threshold}$
- Likely caused by high edge neutral density influx from large vacuum chamber

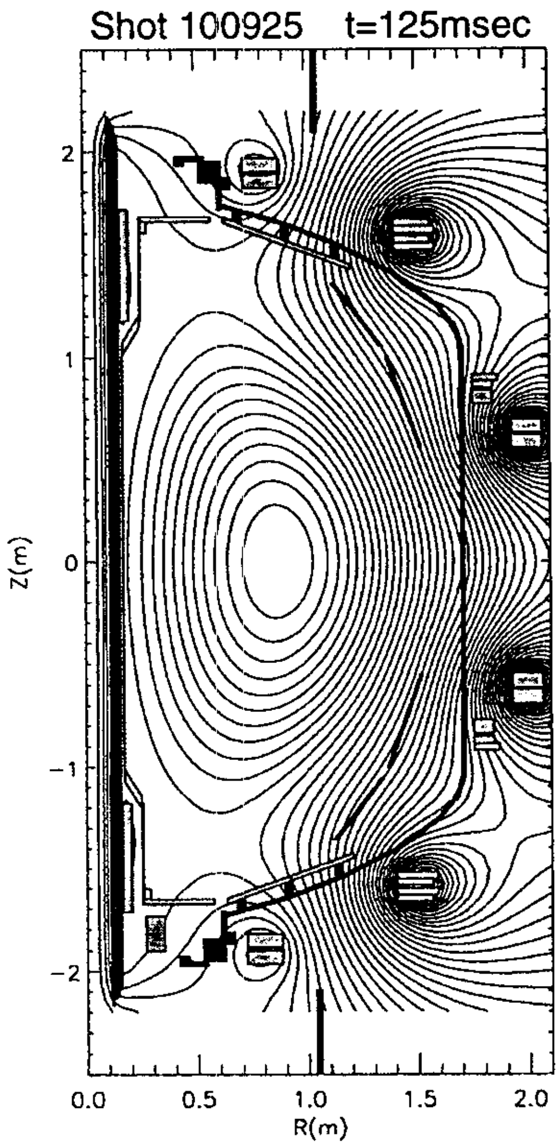
NSTX can test a wide variety of shapes, profiles, and divertor configurations to optimize performance



Lower SND and DND already achieved in NSTX



Lower single-null divertor plasma - 520kA



Double-null divertor plasma - 500kA

EFITs from S. Sabbagh - Columbia Univ.

Some reactor-relevant scalings for tokamaks and STs:

- $P_{fusion} \propto \beta^2 B^4 V_{plasma}$ where $\beta = 2\mu_0\langle p\rangle / B^2$
- Troyon scaling: $\beta(\%) < \beta_N I_P(MA) / aB \Rightarrow$ $\beta(\%) < 5\ \beta_N\ \varepsilon\ (1+\kappa^2)/2q^*$
  - β_N = normalized β, ε = inverse aspect ratio a/R
  - κ = elongation, q^* = kink safety factor
- Self driven (bootstrap) current fraction: $f_{BS} = I_{BS}/I_P \approx C_{BS}\ \varepsilon^{1/2}\ \beta_P$

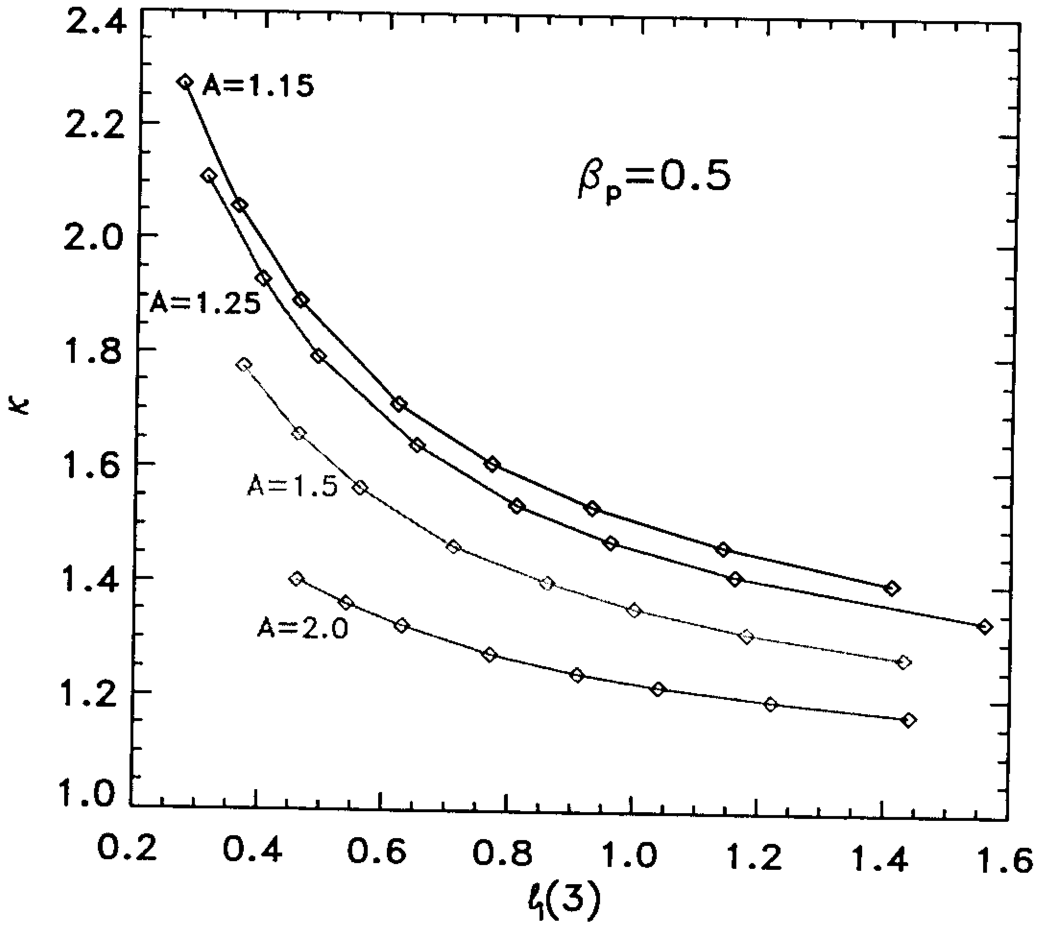
Scalings continued...

- Combining Troyon and BS scalings ⇒
  ⇒ $\beta(\%) < \varepsilon^{1/2}\, C_{BS}\, (1+\kappa^2)\, (\beta_N)^2 / 8\, f_{BS}$
- The ε dependence above is relatively weak and $f_{BS} \approx 1$ is required for steady-state operation ⇒
  - high κ strongly impacts β
  - high β_N is essential for high β

How do κ and β_N scale with aspect ratio?

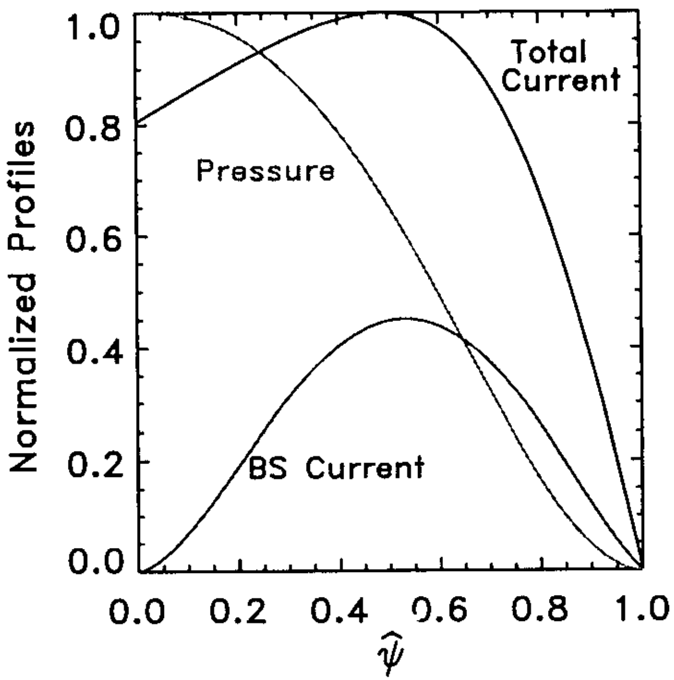
Natural elongation increases rapidly as $\varepsilon \rightarrow 1$ at low l_i

For high f_{BS} ST equilibria which have broad pressure profiles and $q(0) > 2$, the internal inductance $l_i < 0.3$ $\Rightarrow$ elongation in a pure vertical field (natural k) is approximately 2 $\Rightarrow$ much higher than for $A \geq 2$.



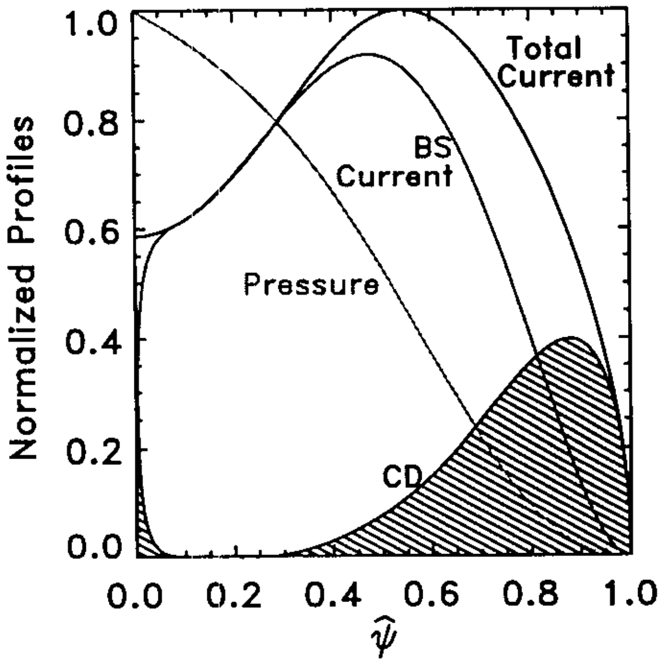
Theoretical β_N limit also increases significantly as $\varepsilon \rightarrow 1$ with and w/o active kink stabilization

A=1.26, κ=2.0, δ=0.45, β=31%
β_N =5.5, f_{BS} =40% , q(0)=1.9



Ballooning and kink stable without wall

A=1.26, κ=2.0, δ=0.45, β=40%
β_N =8.5, f_{BS} =77% , q(0)=2.85



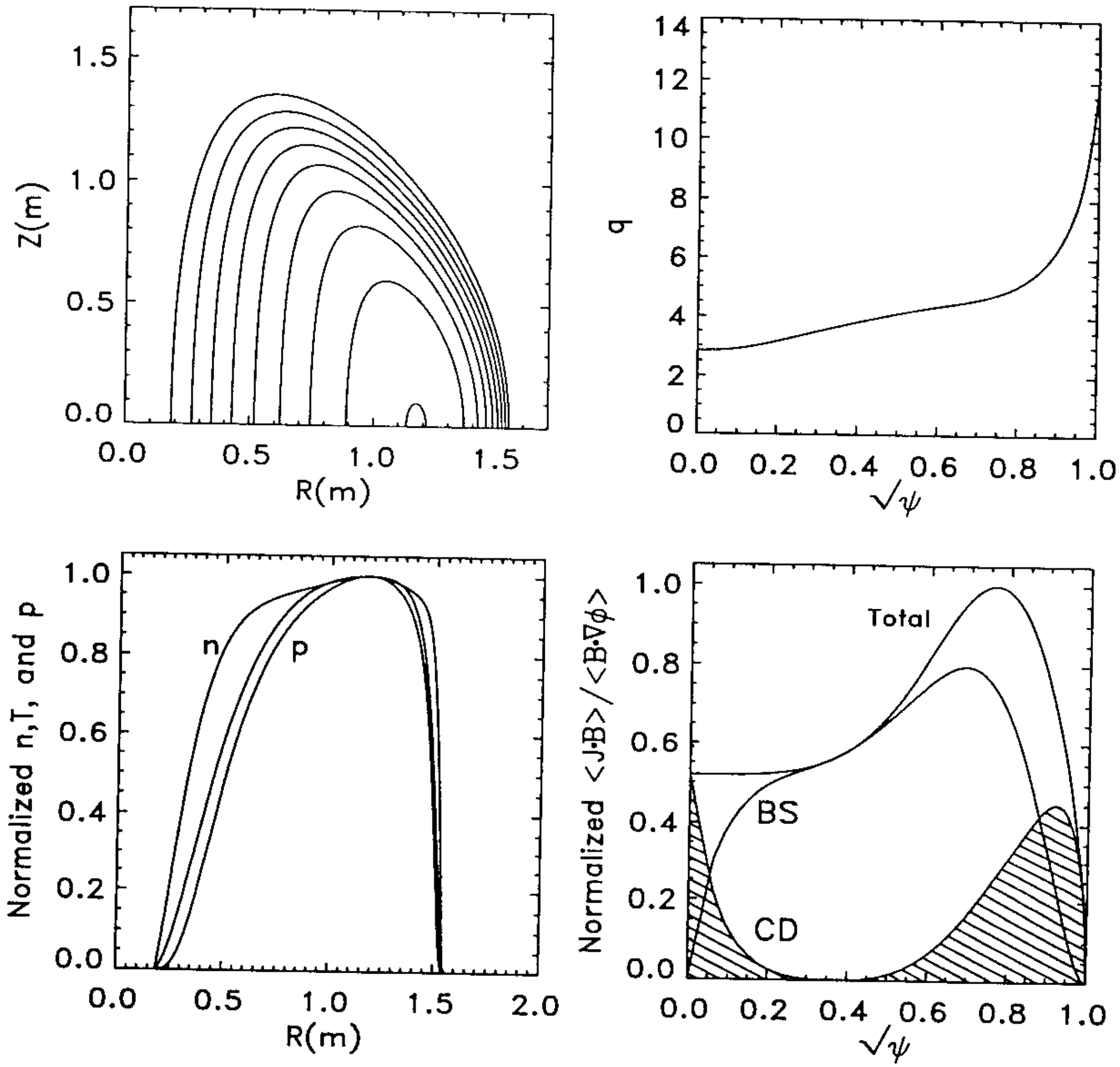
Stable with wall at b_{wall}/a=1.2

Note: corresponding high-A β_N limits are roughly 3.5 and 5.5
$\Rightarrow$ 50% increase in $\beta_N \Rightarrow \times$ 5 increase in P_{fusion}

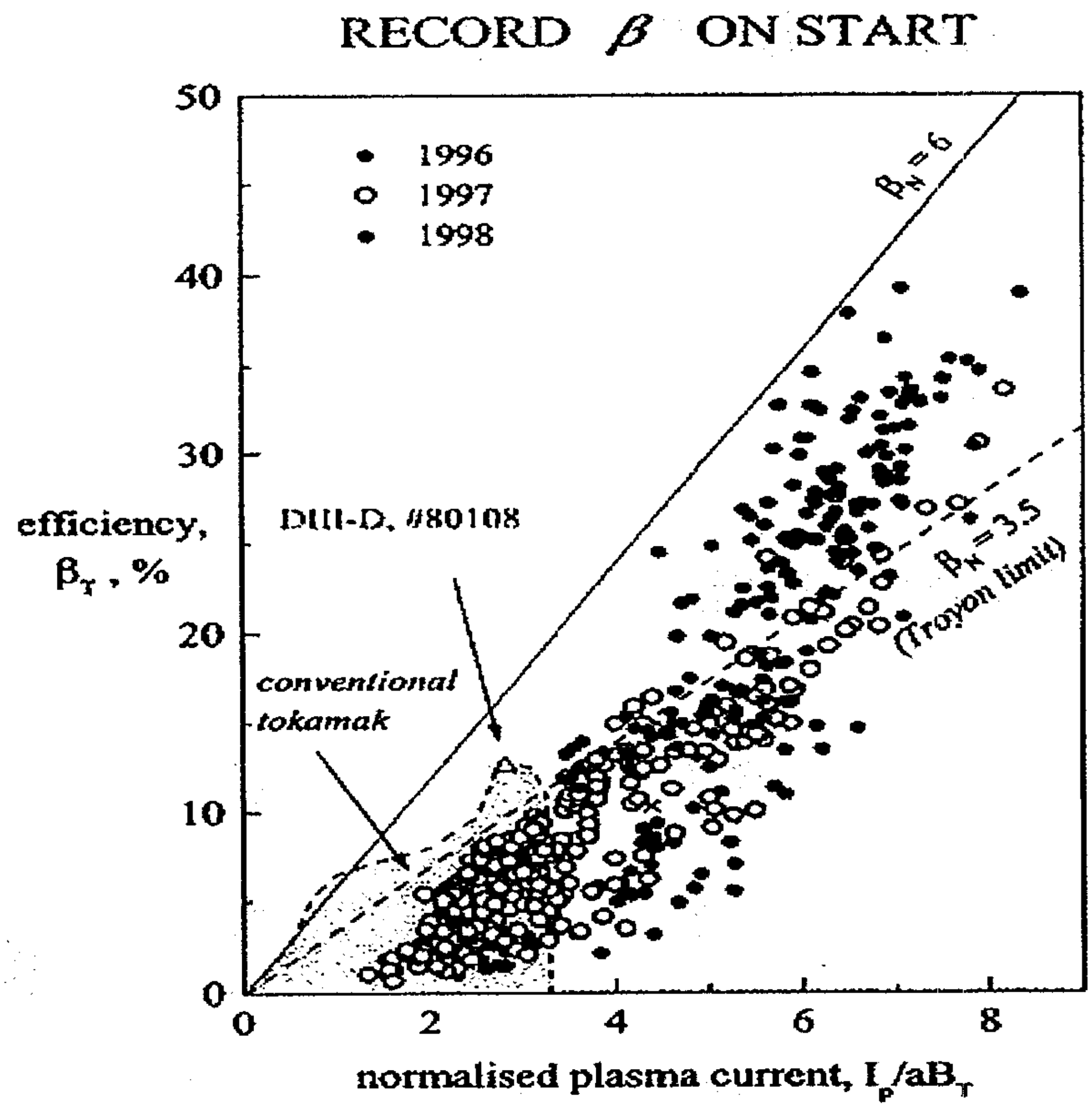
The target equilibrium for NSTX

- $R = 0.86$ m
- $a = 0.67$ m
- $k = 2.0$
- $\delta = 0.45$
- $A = 1.27$
- $I_p = 1$ MA
- $B_T = 0.3$ T
- $\beta_N = 8.1$

- **$\beta = 40.4\%$**
- **$f_{BS} = 70\%$**

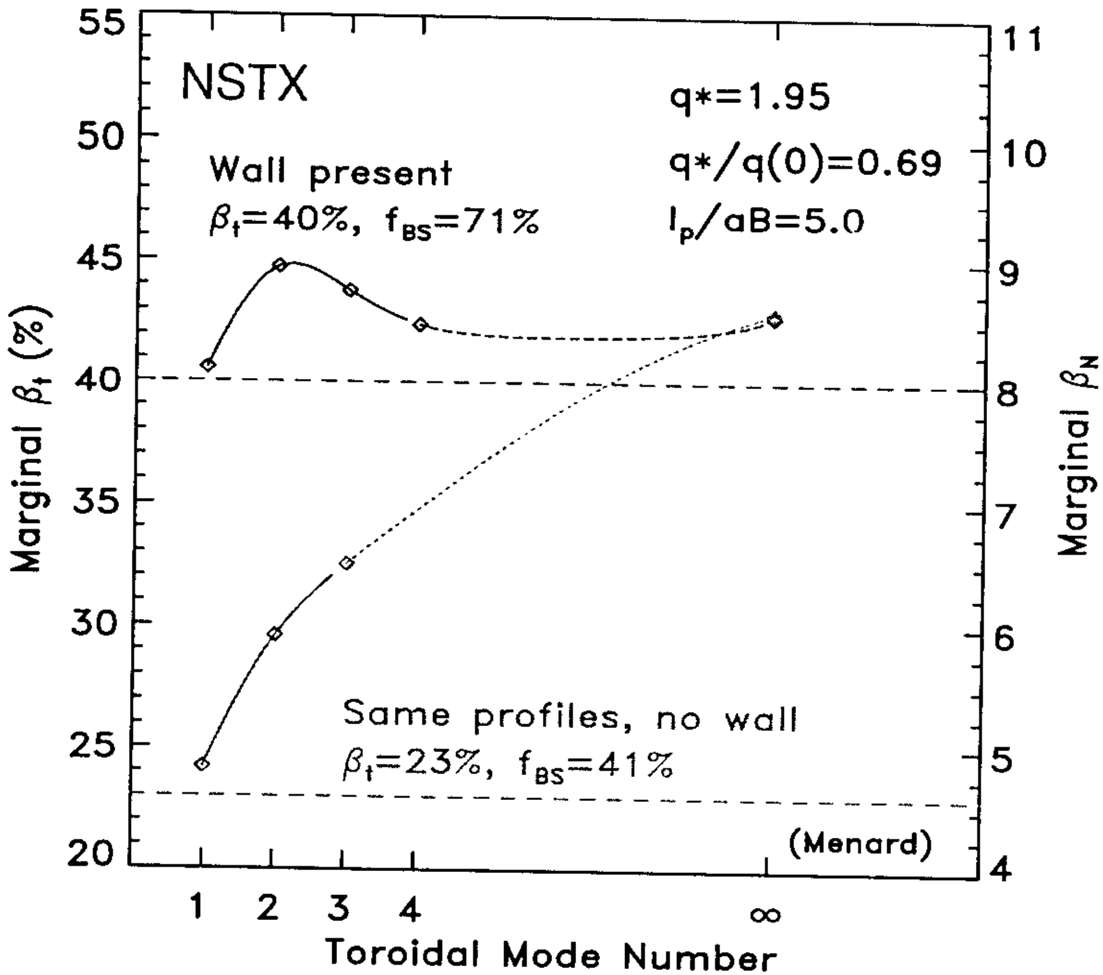


START record β_N is 5-6 at high β



(Figure taken from the START web page)

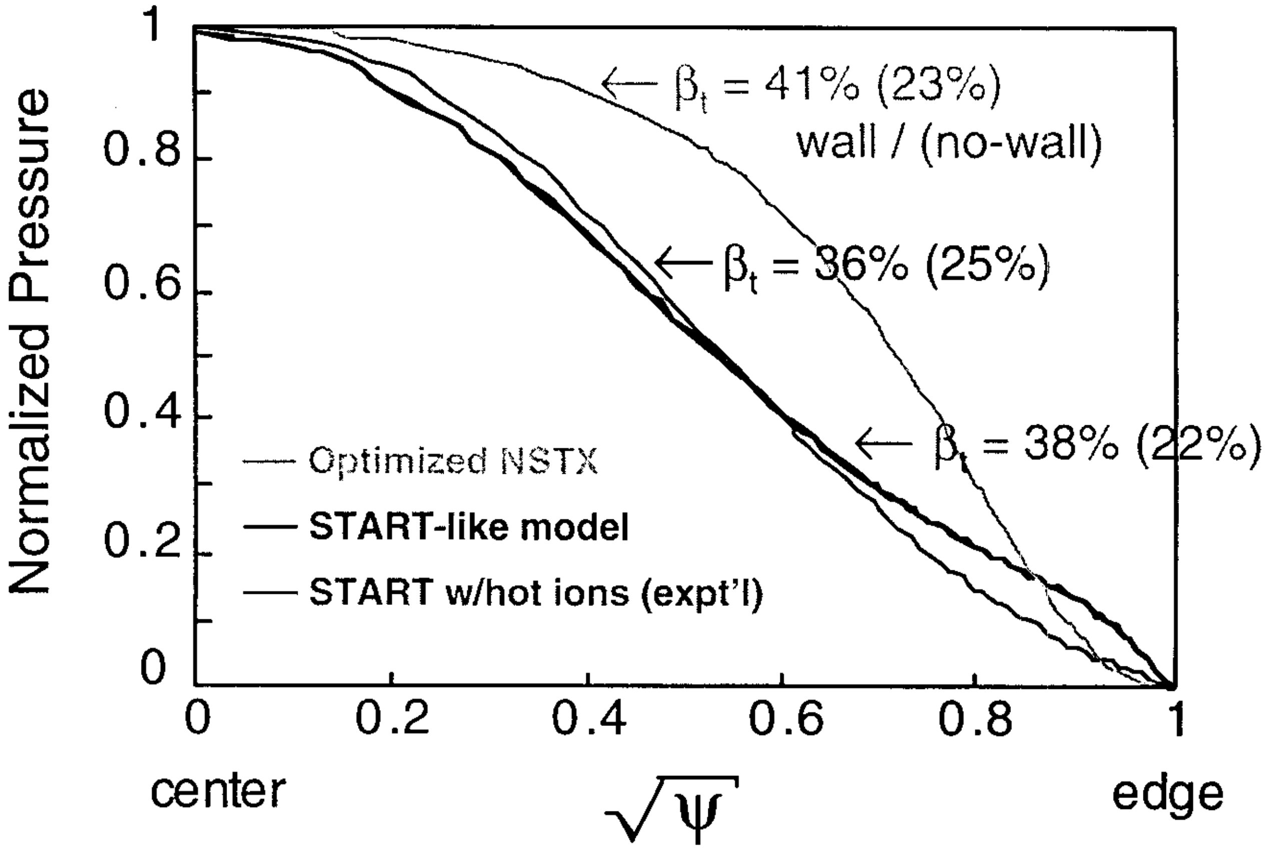
NSTX / MAST comparison allows assessment of influence of conducting wall on MHD stability



Conducting wall important for optimizing β **and** boostrap current (q_ψ~10)

- For NSTX, 30% edge CD required for full non-inductive sustainment (?)
- Role of resistive wall modes will be investigated

MHD stability relatively robust to variations in *p* and *J* profiles - still retain high-β_t with conducting walls

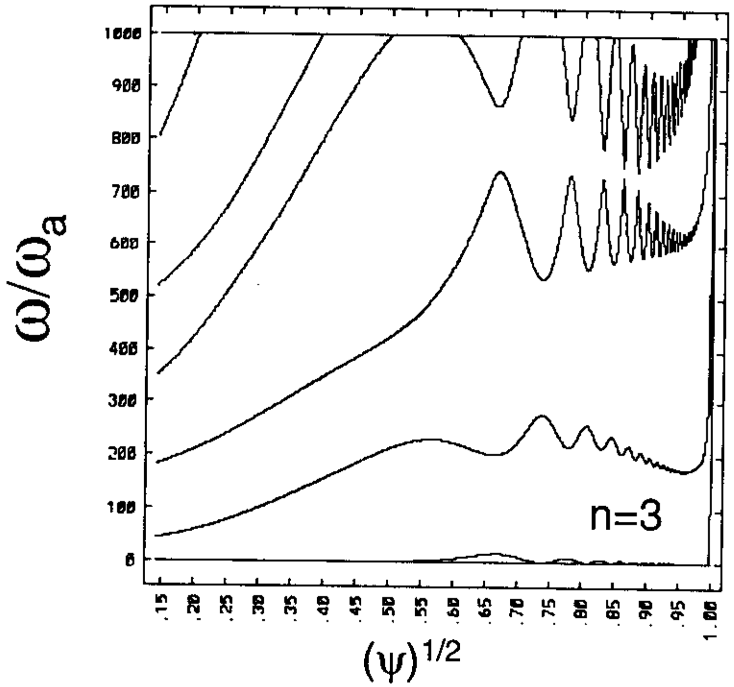


F. Paoletti,
S. Sabbagh

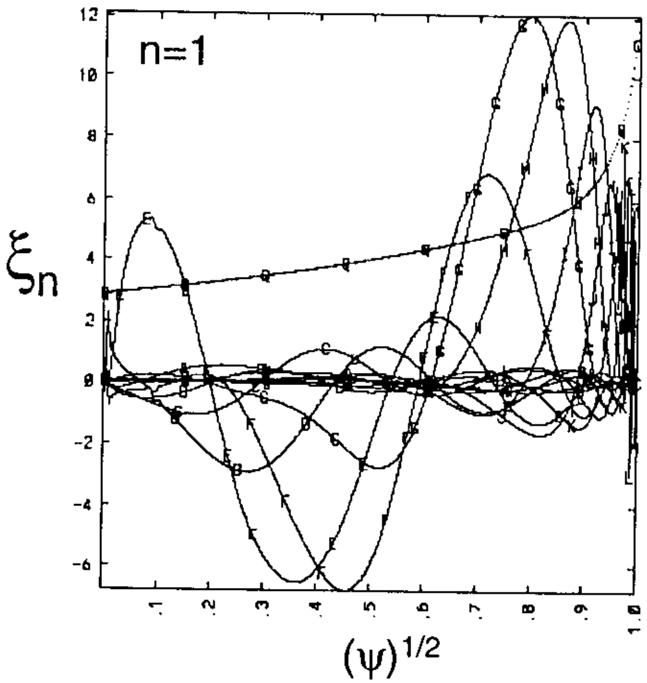
- Experimental p(r) more peaked than the optimized ones (beams)
- BS current alignment may be adversely effected
- Stability also relatively robust to 30% changes in q-profile ($\beta_t \rightarrow 32\%$)

80 keV D^0 NBI on NSTX will produce a super-Alfvenic fast ion population

- $v_{beam} > v_A$
- $\rho_{L\text{-}beam}/a \sim 1/5\text{-}1/3$
- $\Delta_{pol\text{-}beam}/a \sim 1/3\text{-}1/2$



- Large spectrum of modes for each n
- Modes exist for many n
- Broad, global structure
- TAEs exist in high-β plasmas



(N. Gorelenkov)

Modeling indicates little fast ion loss due to TAEs

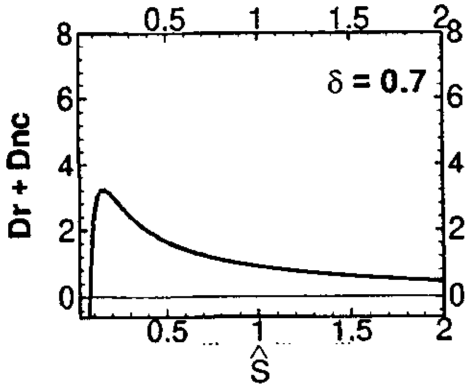
- Magnetic well and large $B_{\theta a}$ help particle confinement
- TAEs on START appear to be "benign" (McClements)

Neoclassical Tearing Modes are expected to be less virulent in STs than at conventional aspect ratio

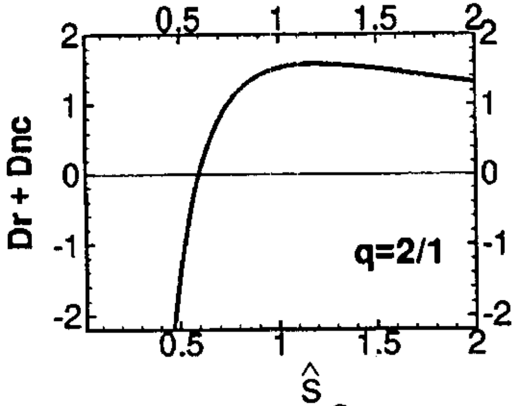
Island evolution: $$\frac{d}{dt} w = 1.22 \frac{\eta_{nc}}{\mu_0} \left(\Delta' + 4.6 \frac{D_{nc} + D_R}{w} \right)$$

$D_{nc} \sim \frac{\sqrt{\epsilon}\beta_\theta}{\hat{s}}$; $D_R \sim \frac{\beta}{\hat{s}^2}$ (at high β) $\hat{s} \equiv 2V/q(dq/dV)$

destabilizing stabilizing - large Pfirsch-Schluter current in ST



DIII-D β=3%



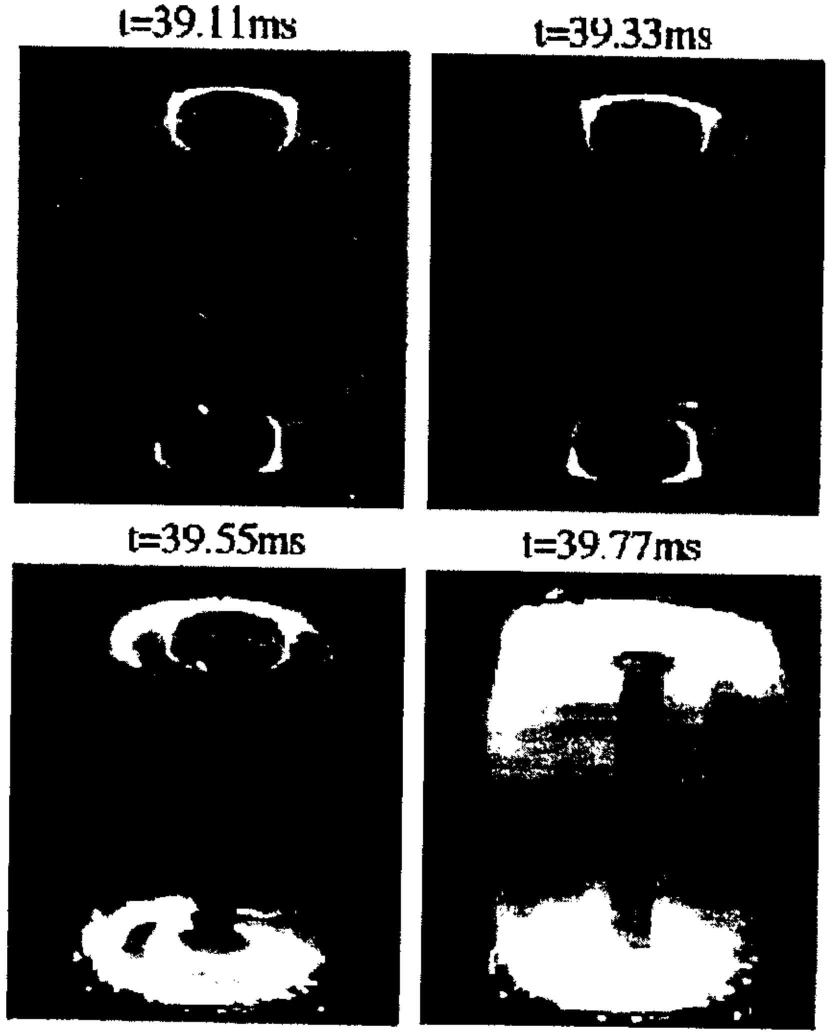
Pegasus β=15%

from Kruger et al.Phys. Plasmas, Vol. 5, No. 2

- Stability improved, island size reduced at low R/a?
- Effect of fast particles, kinetic effects on NTMs?

At sufficiently low edge q, STs can disrupt like standard tokamaks:

High-speed camera images of the boundary distortion caused by an n=1 kink instability near the end of a low $q^* \approx 2\text{-}3$ discharge on START.

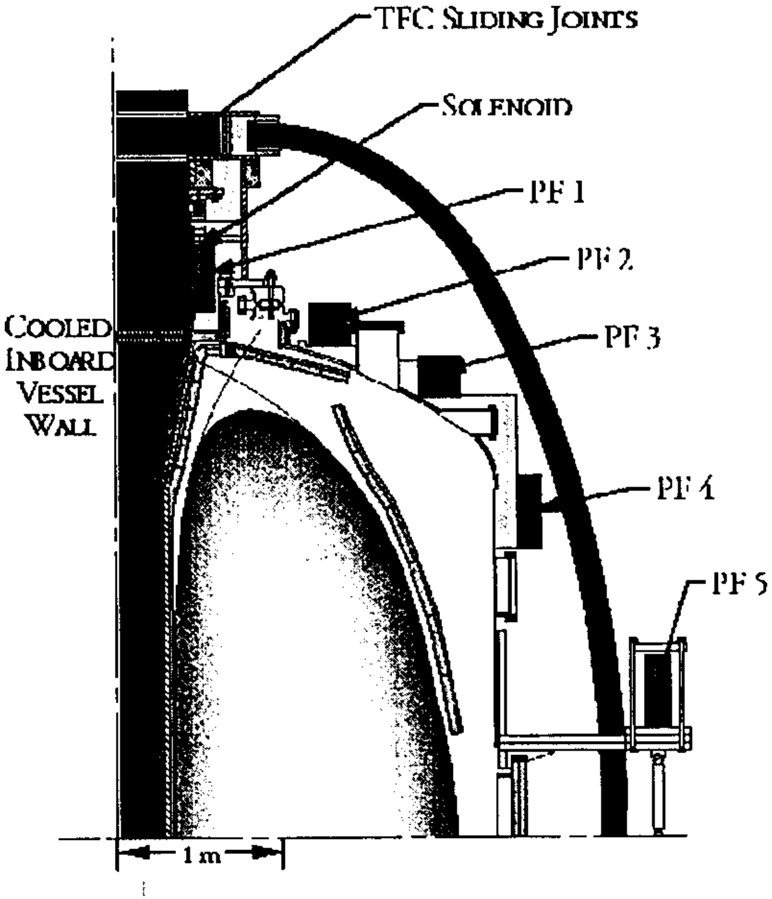


Highlights the desire to operate at higher q_{edge}

(Figure taken from D. Gates, et al., Phys. Plasmas, May 1998)

Understanding and resolving physics issues in present devices may allow aggressive follow-on experiments to be pursued

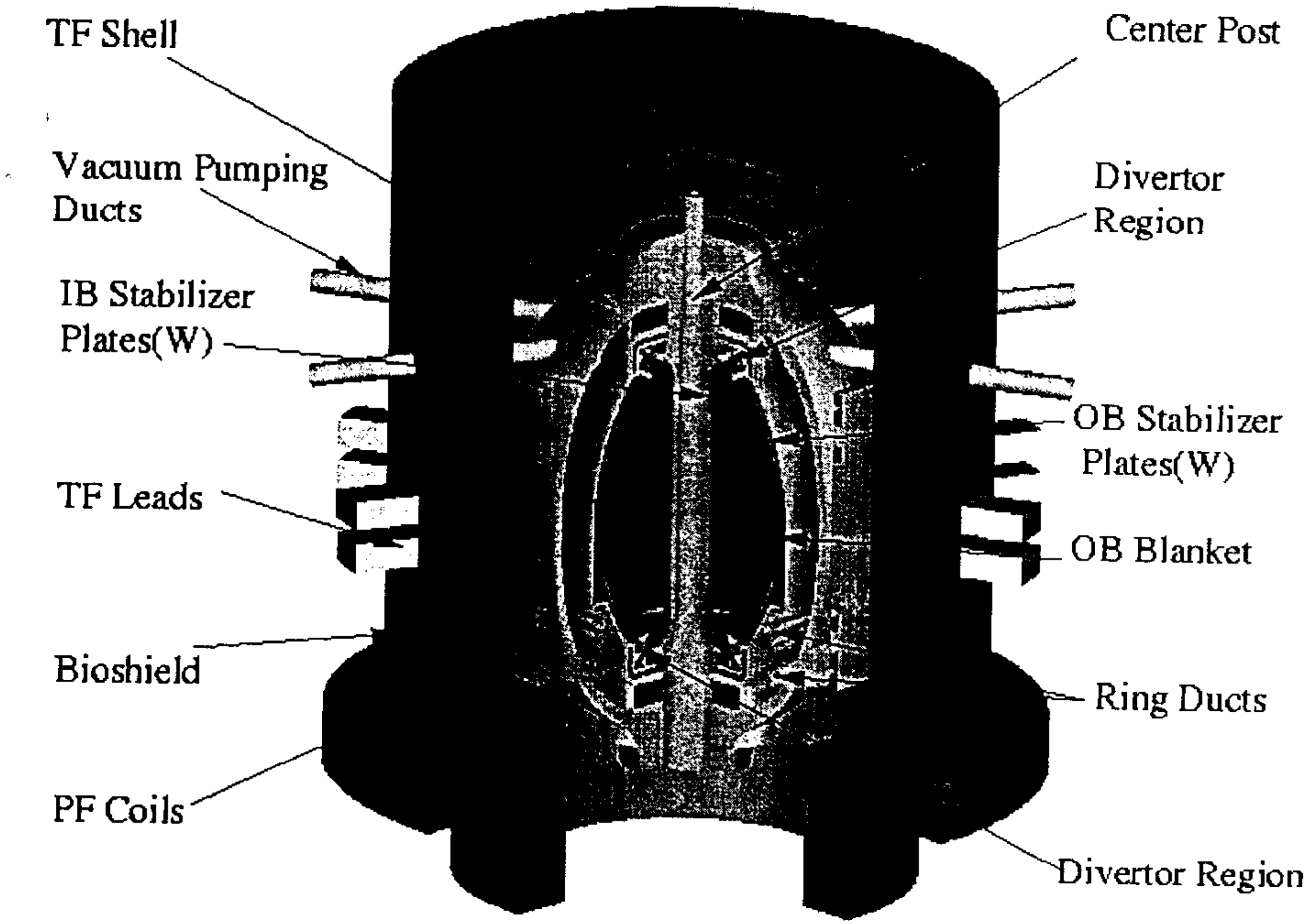
10 MA ST Performance Extension Test



Low-Q (max. neutrons)	High-Q
B_{T0}=1.67 T	1.67 T
R_0=1.2 m	1.2 m
a=0.86 m	0.86 m
κ=3	3
q_{95}=10	10
$<n_e>=1\times10^{20}$ m^{-3}	0.76×10^{20} m^{-3}
P_{aux}=35 MW	4 MW
β_t=23%	34%
β_n=3.5	5.0
Q=1	10
VNS application	Reactor application

Enhanced τ_E required: 3X increase in enhancement at constant I_p

Cutaway View of the ARIES-ST Reactor Core



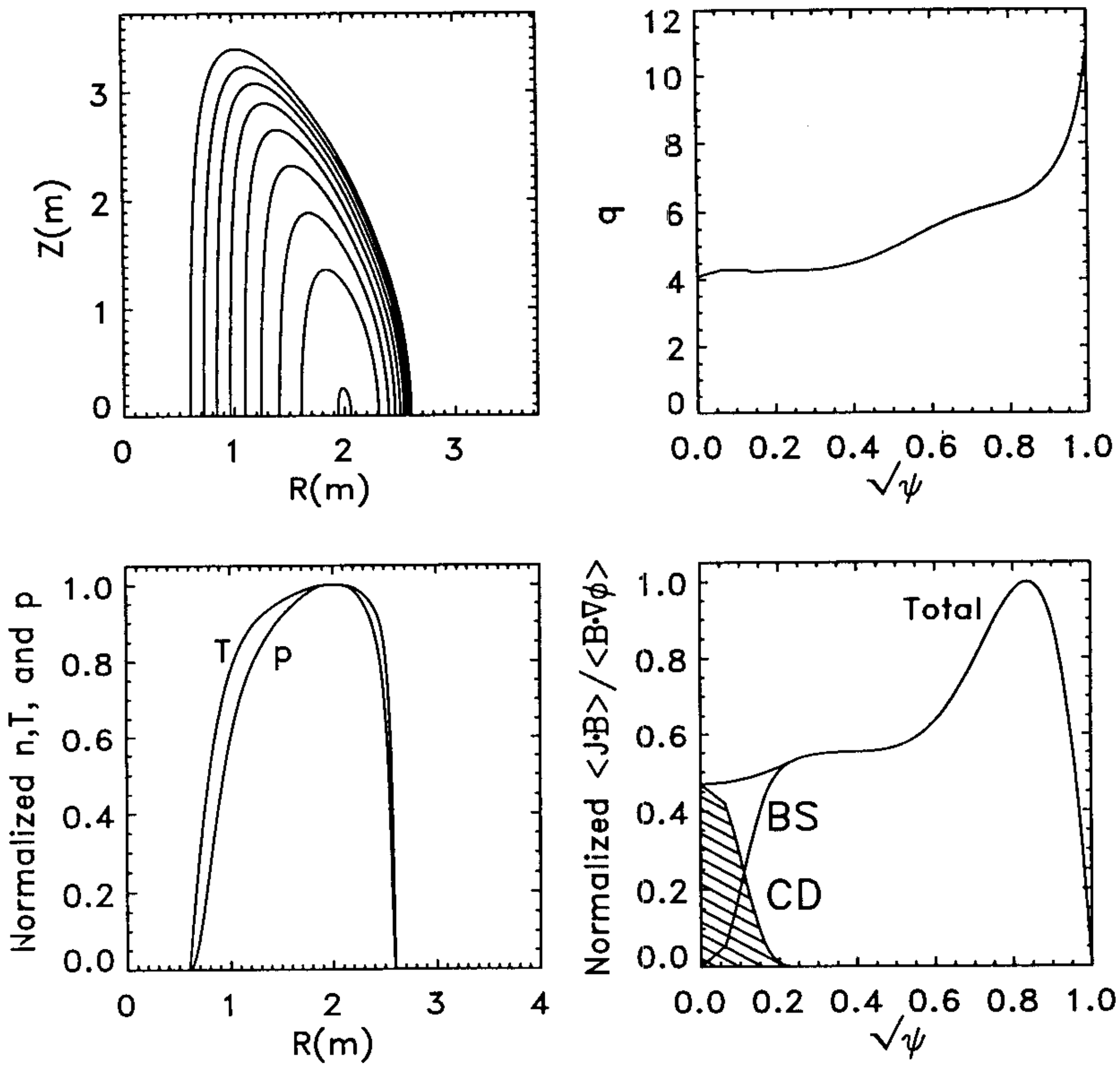
ARIES Team Public Information Site

ARIES-ST Equilibrium Parameters

$A = 1.6$
$\kappa = 3.4$
$\delta = 0.64$

$\beta = 56\%$
$\beta_N = 8.2$
$f_{BS} = 99\%$
$I_P = 35MA$

$p(0)/\langle p\rangle = 1.4$



How important is wall stabilization to ARIES-ST?

- Assuming good BS alignment is retained, removing the wall reduces the marginal β_N by 30-40% ($\beta_N = 5\text{-}6$) $\Rightarrow$
  - β drops by a factor of 2
- Lower β_N also reduces $f_{BS} \Rightarrow$
  - Must drive $\geq$ 20% of I_P non-inductively
- As for the advanced tokamak concept, suppression of pressure-driven kink modes is crucial

The boundary layer and divertor region in STs exhibit distinct differences from those at conventional aspect ratio

- High heat fluxes due to compact configuration (large P/R)

- Large mirror ratio ($R_m = B_{T,max}/B_{T,mp} \sim 4$ as compared to ~2 at R/a~3)
  - Modified velocity space distribution (smaller loss cone: $\alpha_L \sim 40° \rightarrow 30°$, $f_t \sim 0.75 \rightarrow 0.85$; beam-like features at midplane)
- Δ_p comparable to thermal ion Larmor radius
- High local β_t (electromagnetic effects important?)

Are differences significant in terms of driving or modifying X-SOL transport processes?

What is ratio of inboard to outboard SOL heat flux?

Summary

- ST research is just beginning: many physics (and engineering) issues need to be resolved to develop a successful approach to next step devices.

- Experiments in NSTX and MAST will contribute greatly to the understanding of the physics of:
  - Steady-state operation
    - HHFW, NBI, CHI
    - Bootstrap current
  - Confinement and transport
    - ExB shear stabilization physics
    - neoclassical transport
  - MHD
    - Ideal stability limits, resistive wall mode physics
    - Alfven eigenmodes and other velocity space instabilities
    - Tearing modes - resistive and neoclassical
  - Boundary physics
    - Power dispersal, impact of high divertor heat loads