

# ARIES-AT

## Extension of the Physics Basis for ARIES-RS

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## Improvements Over ARIES-RS

- using  $\geq 99\%$  flux surface from free-boundary plasma equilibria rather than assumed 95% flux surface
  - larger elongation and triangularity giving higher stable  $\beta$
- using more flexible pressure profile formulation
  - allows better bootstrap alignment and higher ballooning  $\beta$  limit
- higher triangularity, elimination of inboard slot divertor
  - allows higher  $\beta_N$  and higher  $I_p$ , resulting in large  $\beta$  increase (recall  $\beta = \beta_N(I_p/aB\tau)$ )
- higher elongation, moving vertical stabilizing shell closer to plasma
  - allows higher  $\beta_N$  and higher  $I_p$ , resulting in large  $\beta$  increase
- elimination of HHFW, and use of only LHCD for off-axis current drive
  - allows reduction in  $P_{CD}$  and reduces number of CD systems?

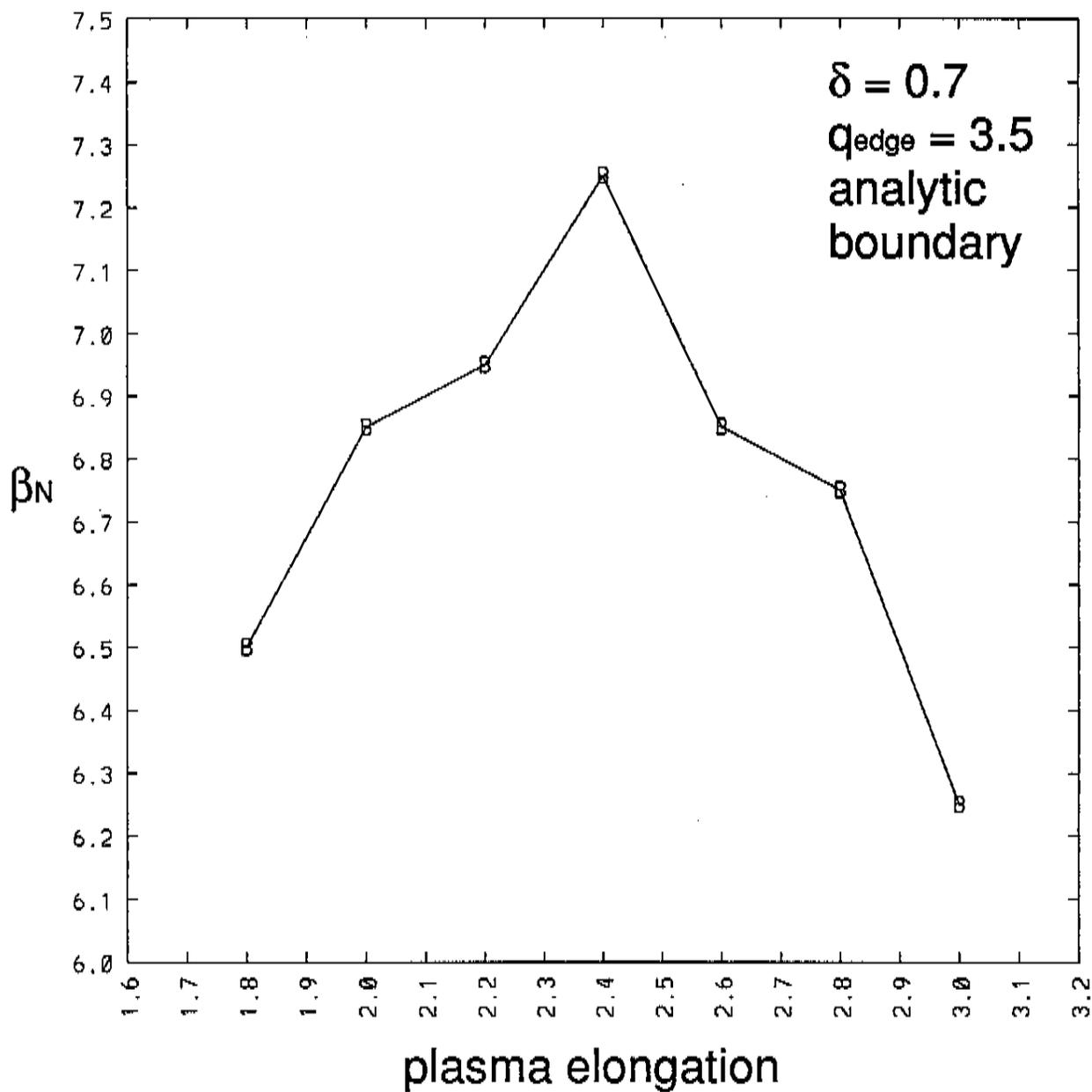
## Impact of Increased Plasma Elongation

- since the possibility of moving the vertical stabilizer shell closer to the plasma was raised, elongation studies were done
  - shell location has traditionally been in the gap between the blanket and shield, typically  $b/a \approx 0.5$
  - ARIES-AT blanket design would allow the shell to be at  $b/a \approx 0.25-0.35$
- elongation has a strong impact on  $\beta$ , partly from  $\beta_N$ , but more importantly from increases in plasma current

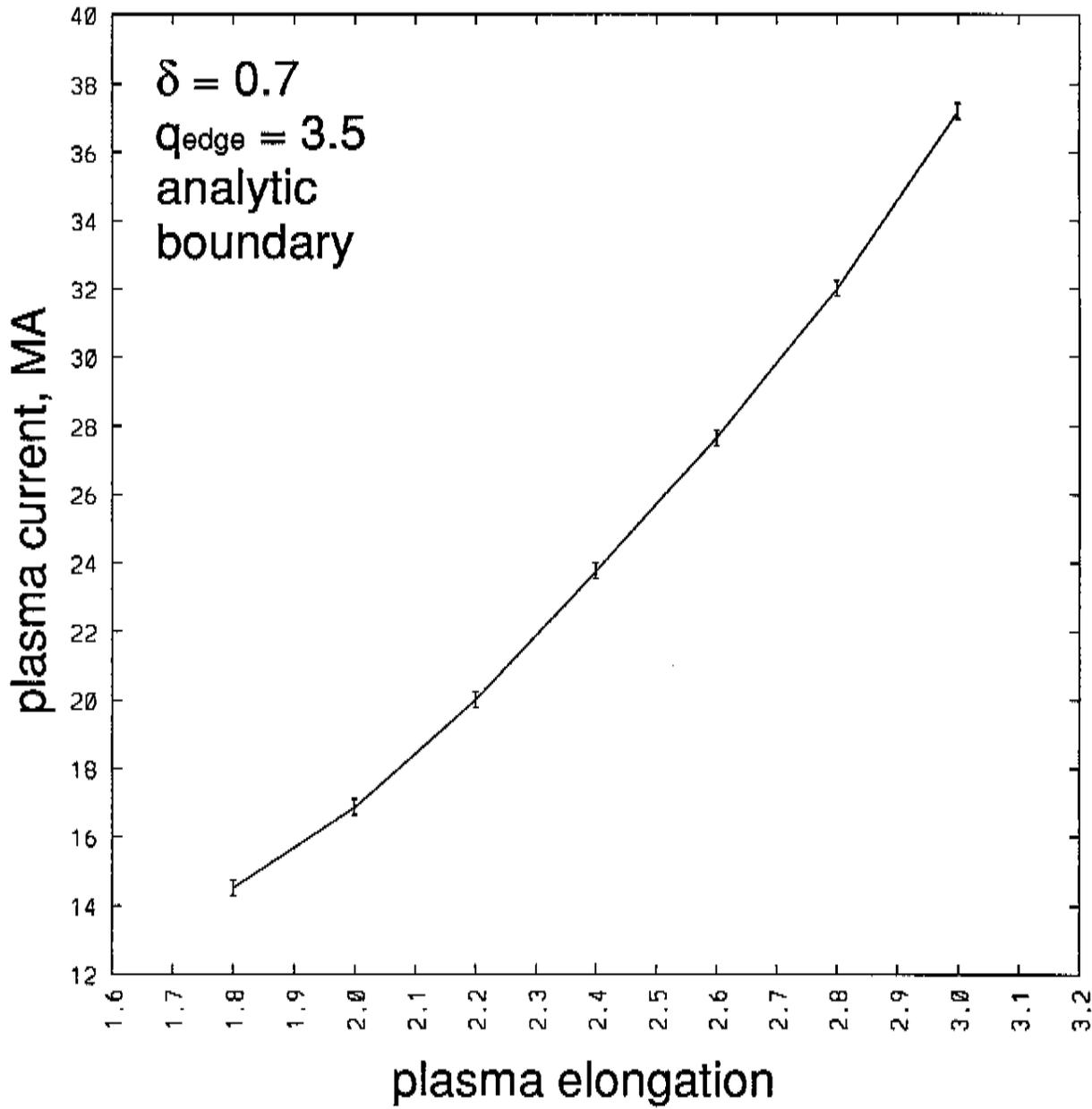
$$\beta = \beta_N (I_p/aB_T) = \beta_N/q_* \times 2.5 \epsilon (1 + \kappa^2)$$

- using analytic boundaries, elongation was scanned and  $n=\infty$  ballooning,  $n=1-6$  kink, and  $n=0$  vertical stability was examined, with the following constraints
  - $\delta = 0.7$
  - $q_{\text{edge}} = 3.5$
  - $A = 4.0$

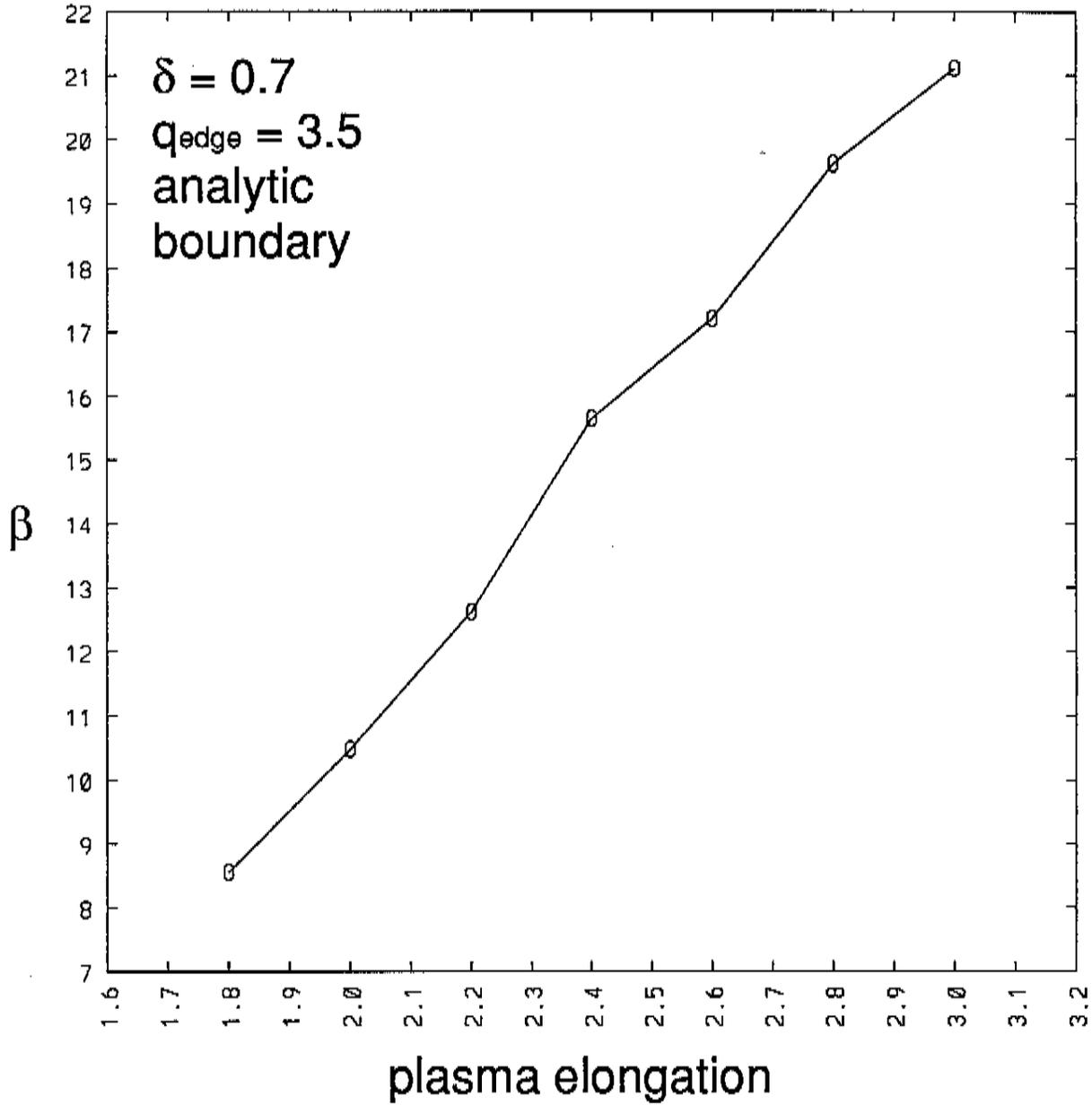
# $n=\infty$ ballooning stability ARIES-A1



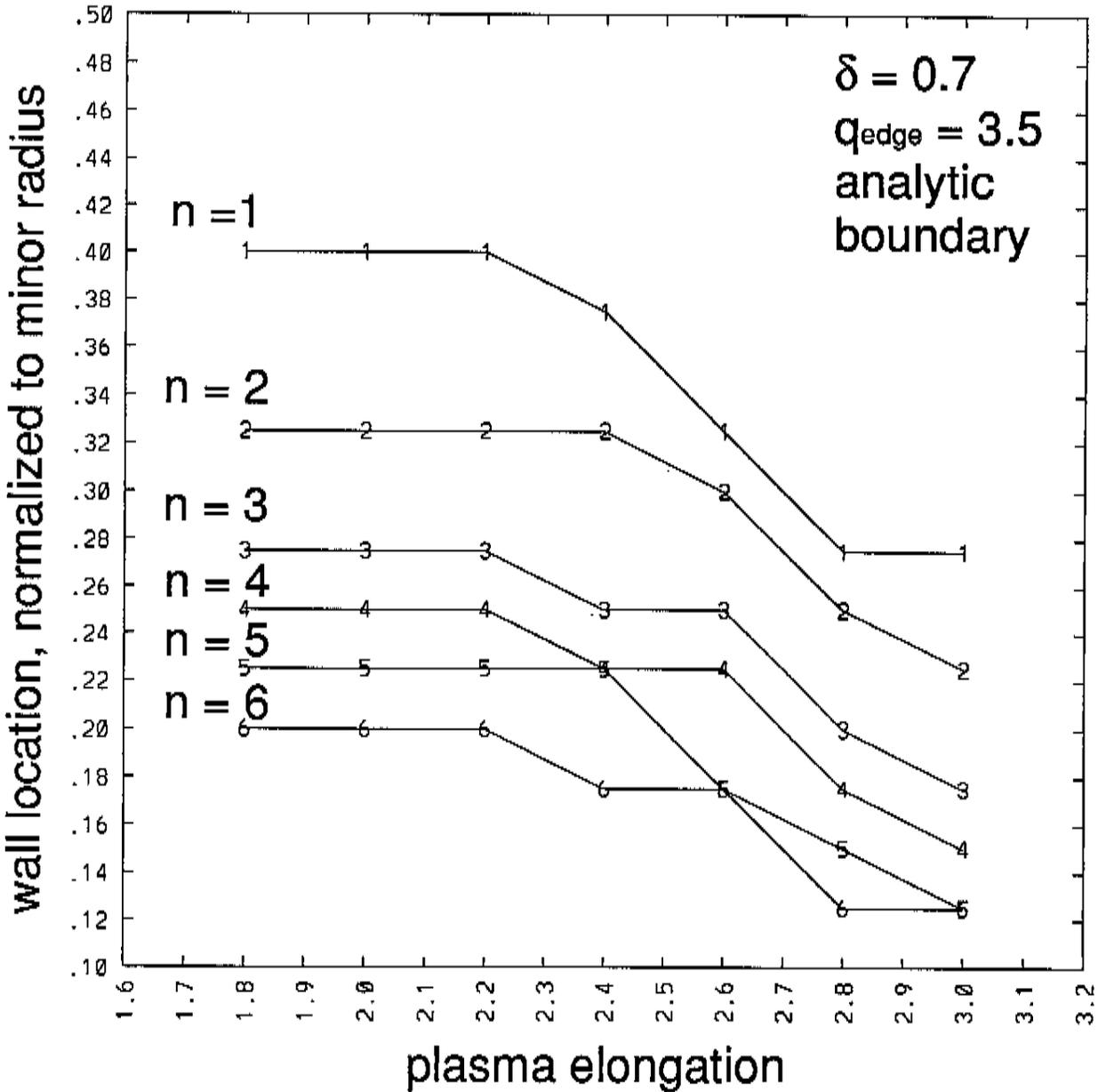
$$\rho = \rho_N \times (I_p / a BT)$$



$$\beta = \beta_N \times (I_p / a DT)$$



# $\eta = 1-0$ KINK STABILITY ARIES-A I



## Plasma Elongation, cont'd

- ballooning stability shows a peak in  $\beta_N$  around  $\kappa = 2.4$ , however, due to increasing  $I_p$  the  $\beta$  continues to increase
- kink stability shows that the strongly triangular plasmas have shifted the most unstable mode to higher toroidal mode number  $n \geq 6$ , and that beyond an elongation of 2.2-2.4 the required kink stabilizing wall location rapidly moves closer to the plasma
- vertical stability analysis indicates which plasma elongations are viable based on the distance between the plasma and the actual vertical stabilizing shell

$$b/a = (\Delta r(\text{SOL}) + \Delta r(\text{1st wall}) + \Delta r(\text{blanket})) / a$$

- from vertical stability analysis, the required stability factor could be obtained in the following cases

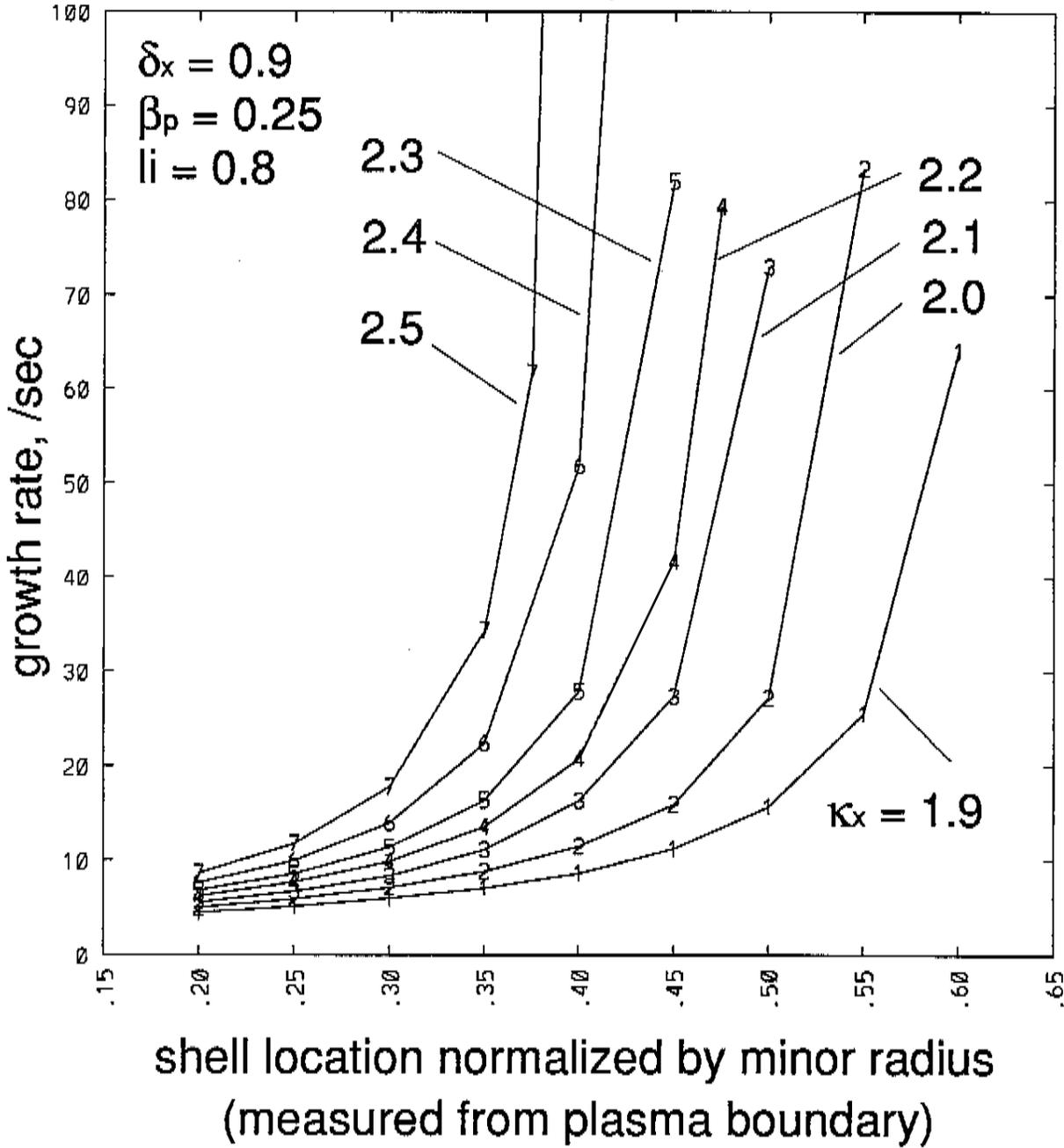
$$b/a = 0.25, \kappa = 2.5, \beta = 15.7\%$$

$$b/a = 0.30, \kappa = 2.3, \beta = 12.7\%$$

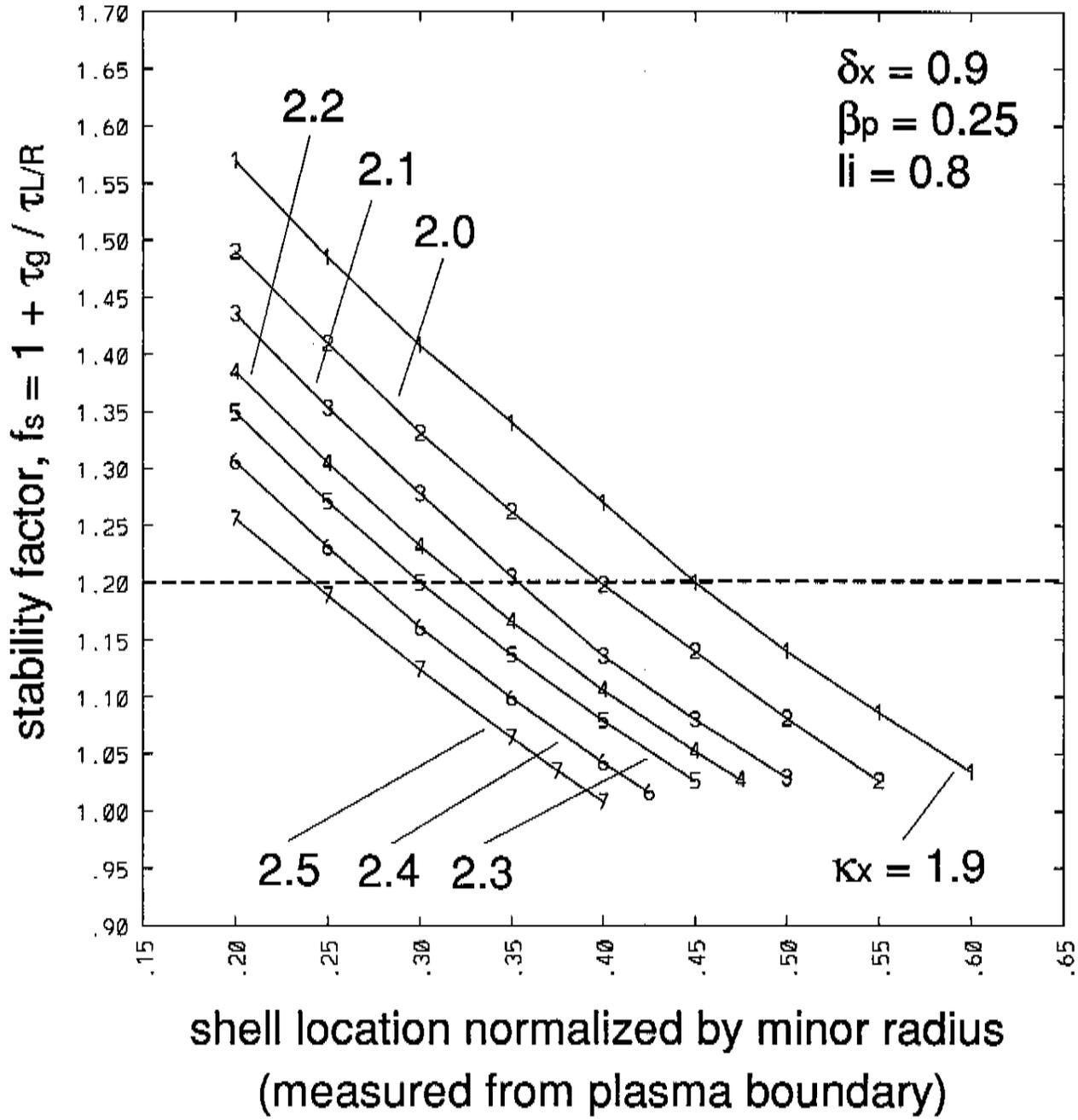
$$b/a = 0.35, \kappa = 2.1, \beta = 10.5\%$$

→ the actual case depends on the SOL width, in addition to engineering design

# Vertical Stability Scan for ARIES-A1 tungsten, 3.5 cm thick, $\rho = 8 \times 10^{-8}$ ohm-m



# Vertical Stability Scan for ARIES-AT

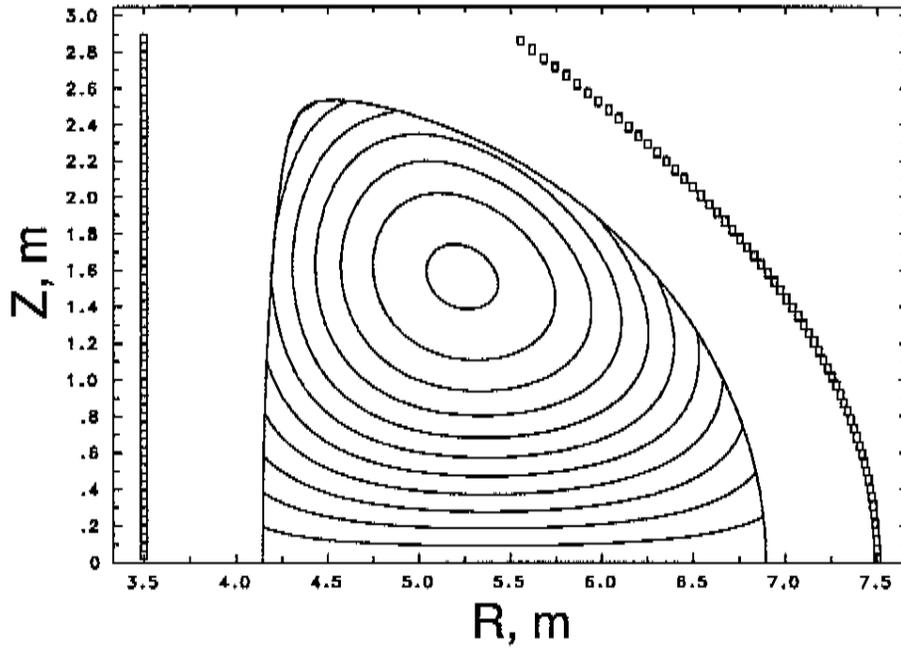


## Plasma Elongation, cont'd

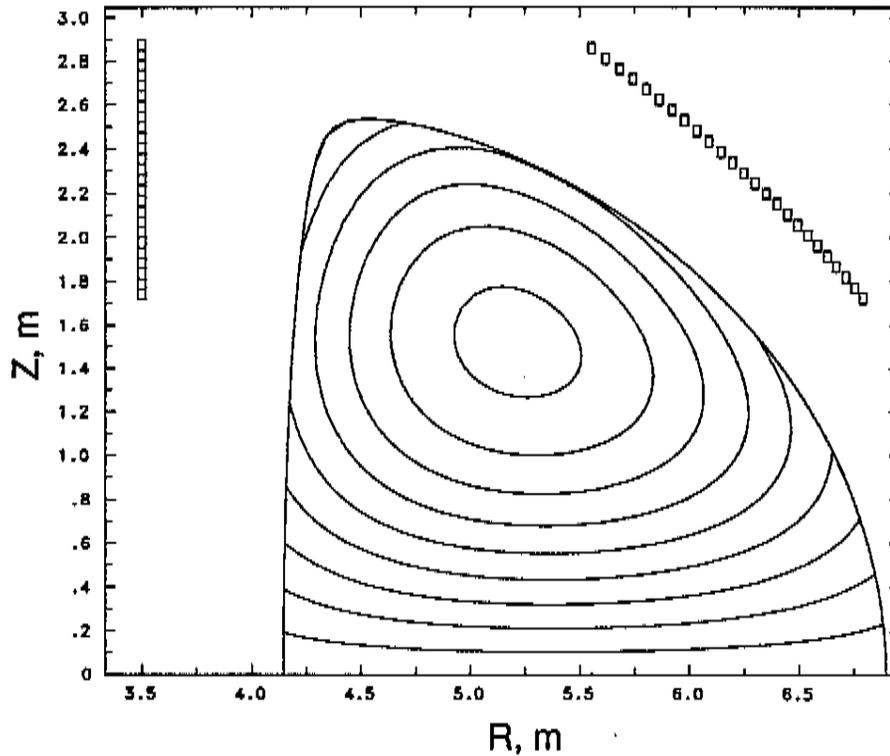
- the vertical stabilizing shell used in the analysis has a poloidally continuous shell on the outboard and on the inboard, with a gap at the divertor
- this shell will have to be reduced in poloidal extent to incorporate in the design, by removing the region closest to the midplane
- it is most likely that the elongation at the separatrix will not exceed 2.2-2.3, when the actual wall distance and poloidal coverage are accounted for, leading to a  $\beta$  increase of 35-45%
- the tungsten thickness is expected to be about 5.5-6.0 cm to account for the high temperature
- feedback control calculations will need to be done on the final configuration, with coils located behind the shield

# Vertical Stability / Elongation Scan

## Vertical Stabilizing Shell Used in Elongation Scans



## Vertical Stabilizing Shell Used in Power Plant Design



# $\kappa_x = 2.2$ Equilibrium

$I_p = 18.6$  MA

$B_T = 7.98$  T

$R = 5.52$

$a = 1.38$

$\kappa_x = 2.2$

$\delta_x = 0.9$

$\kappa_{99.5} = 2.12$

$\delta_{99.5} = 0.78$

$li(3) = 0.30$

$q_0 = 3.1$

$q_{min} = 2.34$

$q_{99.5} = 4.46$

$\beta_N = 6.86$

$\beta = 11.6\%$

$\beta_p = 2.54$

$I_{CD} = 1.35$  MA

$I_{self} = 17.25$  MA

$n(0)/\langle n \rangle = 1.24$

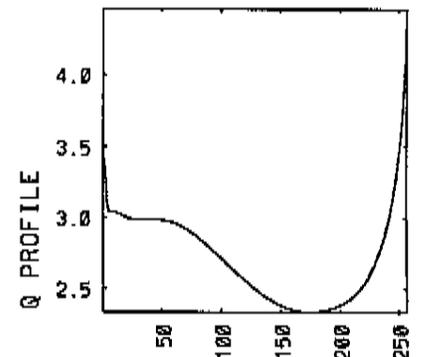
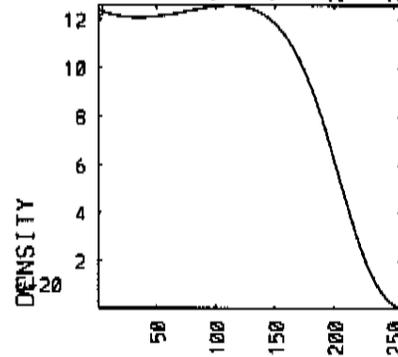
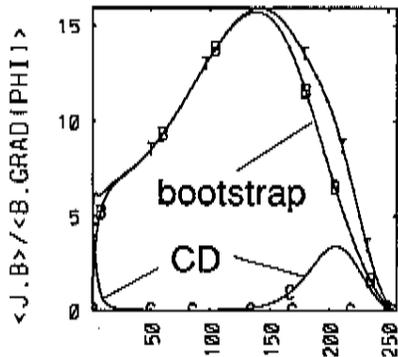
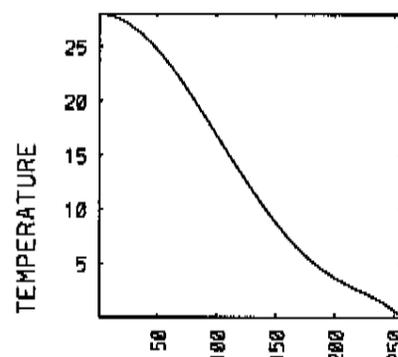
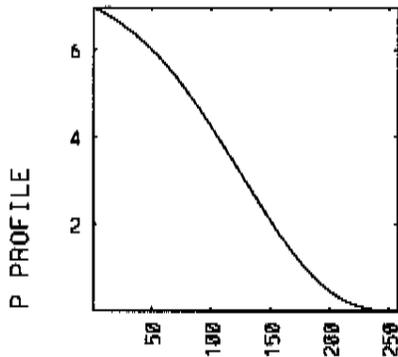
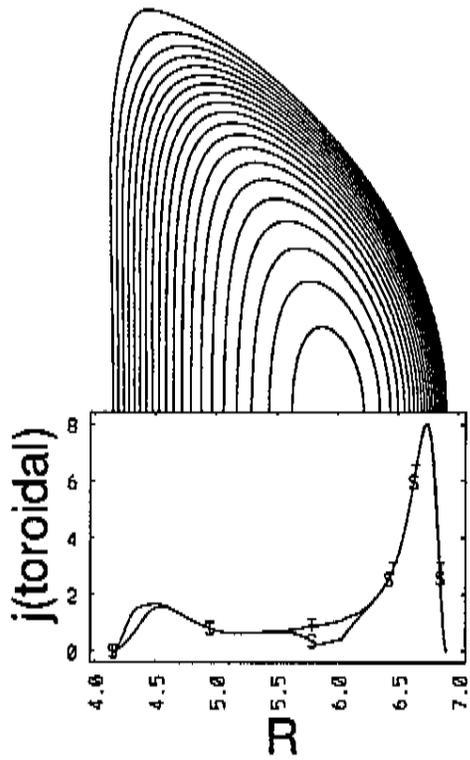
$T(0)/\langle T \rangle = 1.82$

$p(0)/\langle p \rangle = 1.89$

$T(0) = 28$  keV

$n(0) = 12.4 \times 10^{20}$

$n/n_{Gr} = 1.95$



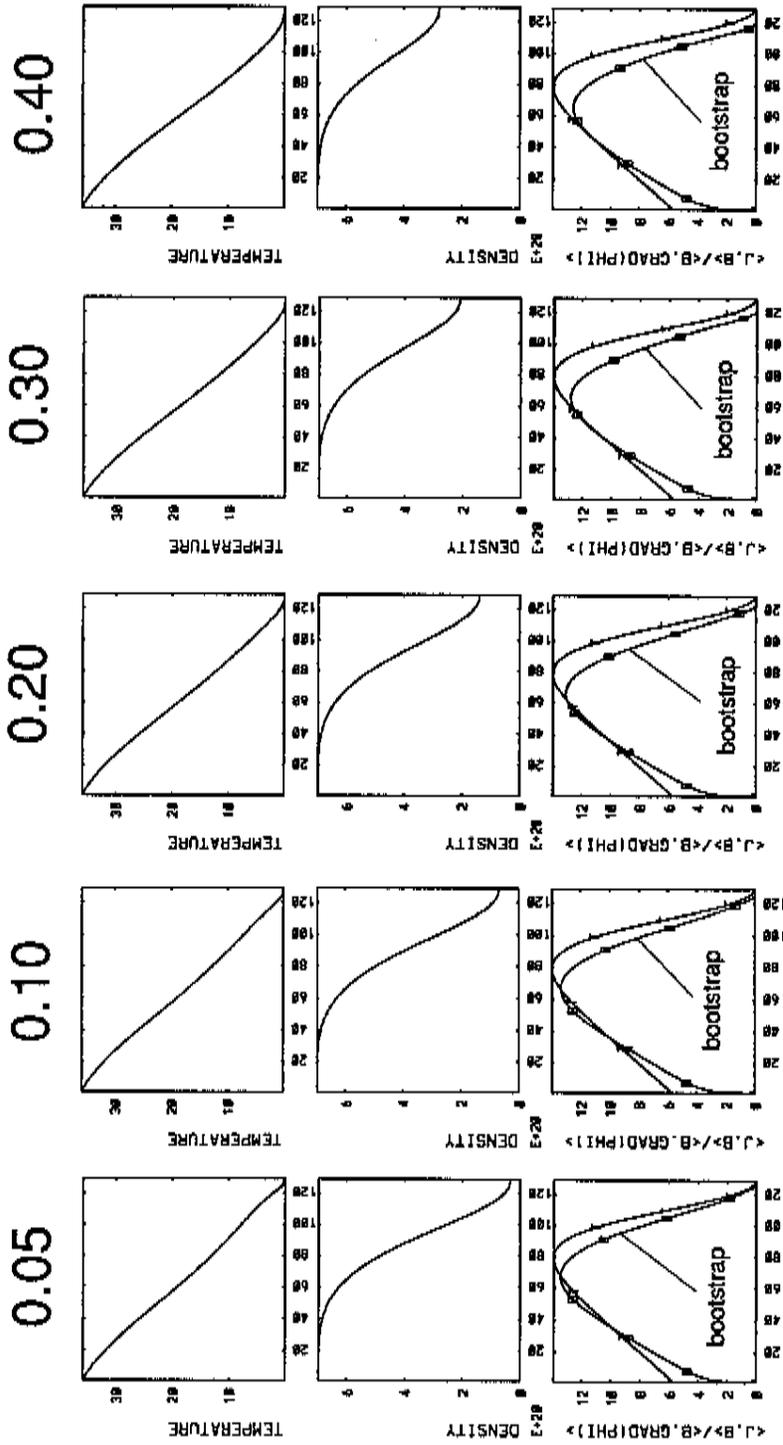
$\psi$

## Finite Edge Density

- including a finite edge density allows increased radiation from the plasma edge and leads to cooler plasma edge temperatures which are desirable for the divertor heat loads
- ARIES-RS found that to effectively radiate more power from the plasma (line radiation), to relieve the divertor heat load, we need edge densities  $n(a) > 0.4 \times n(0)$ , which lead to
  - strong bootstrap current reduction
  - increased CD requirement near the plasma edge
  - excessive increases in  $Z_{\text{eff}}$  in the core, further increasing CD power
- the solution was to use  $n(a) = 0.2 \times n(0)$ , neon as the impurity with  $n_{\text{Ne}}/n_e = 0.8\%$ , making  $Z_{\text{eff}} = 2.0$
- have examined  $\beta_N = 5.6, 6.0, \text{ and } 6.5$  cases with  $n(a)/n(0) = 0.2$  to find stable equilibria and CD requirement
  - off-axis CD has increased to about 1.2 MA in all cases

# Effect of Finite Edge Density on Bootstrap Current

$n(a)/n(0) =$



$\psi$

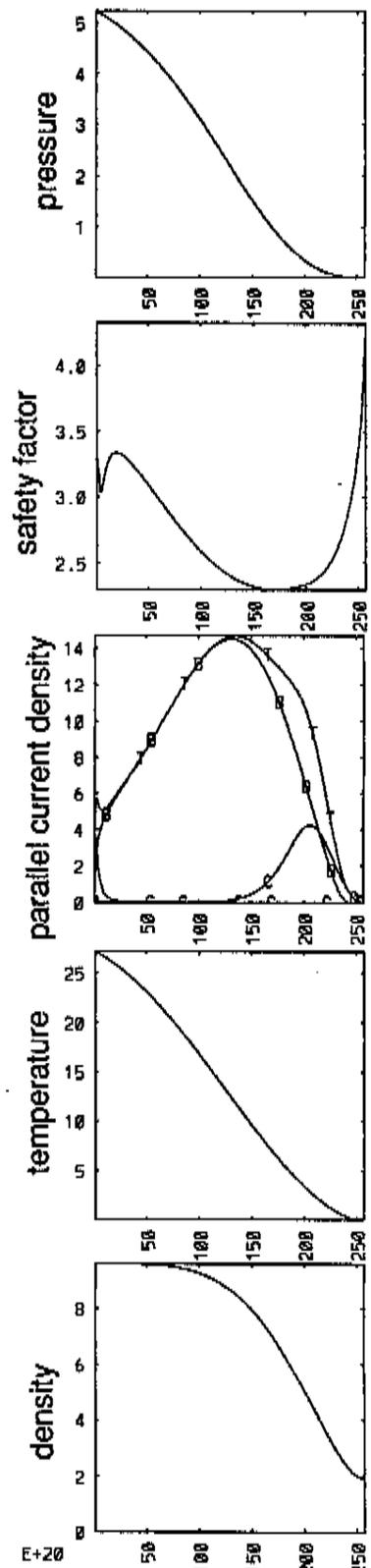
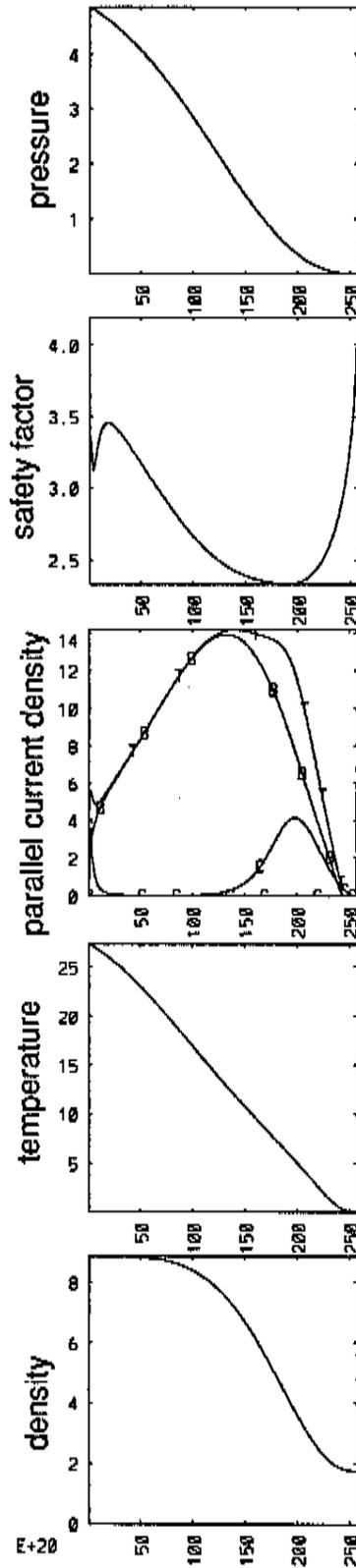
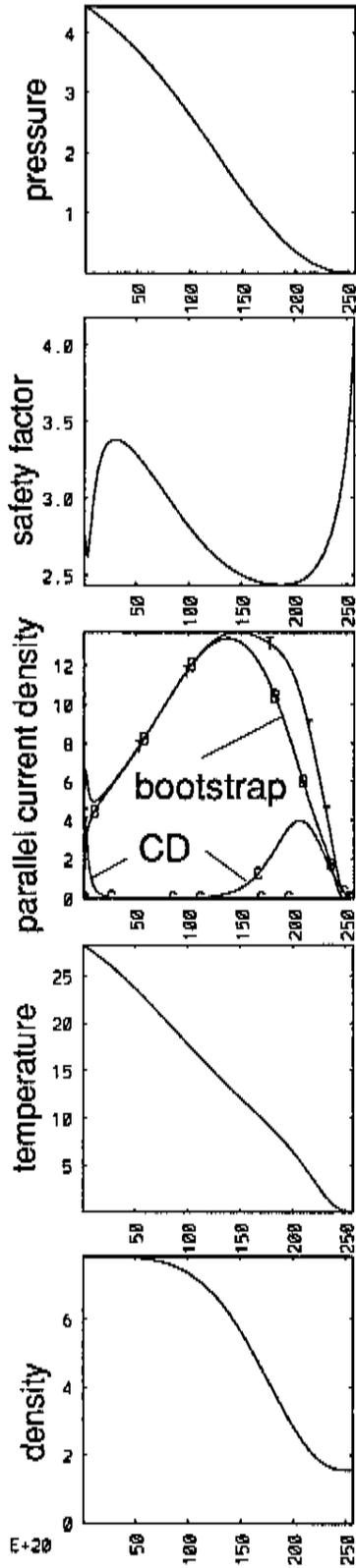
bootstrap current density is strongly reduced at the plasma edge, and  $I_{bs}$  is reduced overall

# Finite Edge Density Equilibria

$\beta_N = 5.6$

$\beta_N = 6.0$

$\beta_N = 6.5$

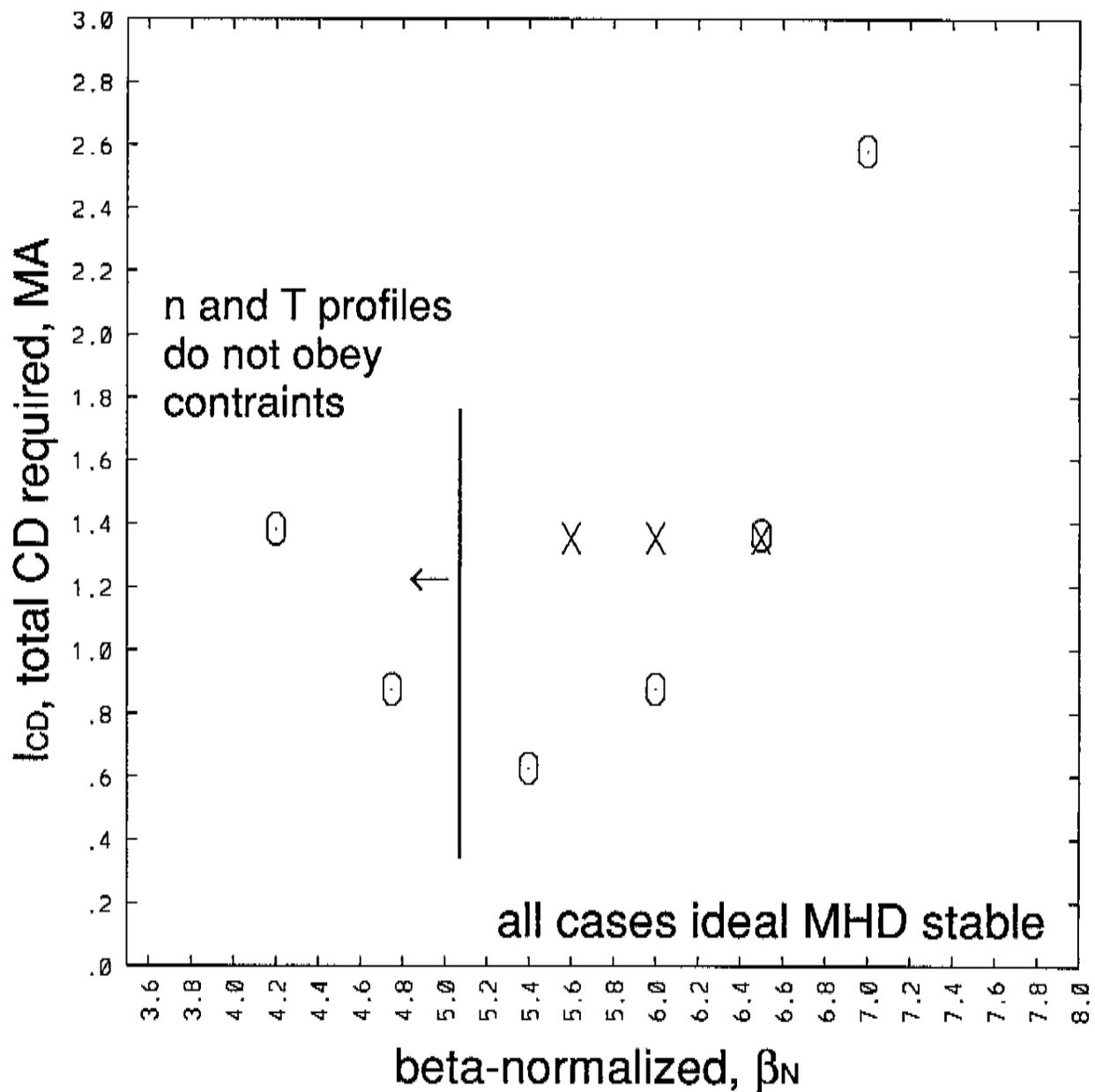


$\psi$

# Finite Edge Density Equilibria

O -  $n(a)=0$  equilibria

X -  $n(a)=0.2n(0)$  equilibria



## Exceeding the Greenwald Density Limit

- the Greenwald density limit,  $n_{Gr} \propto I_p/a^{**2}$ , is found to limit most tokamak experiments, however, power plants at high  $\beta$  require  $n > n_{Gr}$
- recent experiments indicate that the Greenwald density limit is tied to unstable MARFE formation, and the excessive radiation associated with them
- avoiding the formation of MARFE's or avoiding the MARFE moving into the main plasma has allowed TEXTOR and DIII-D to exceed  $n_{Gr}$  by up to a factor of 2
- this has important implications because our choice to go with high triangularity has eliminated an inboard slot divertor, so particles and heat directly hit inboard first wall

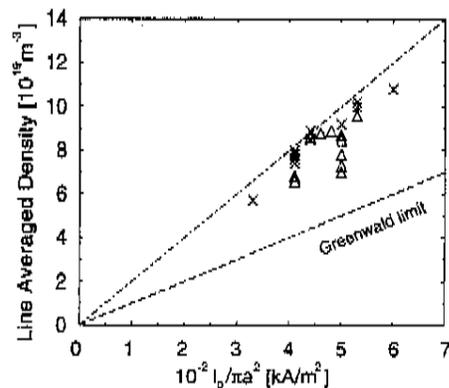


Figure 14. Dependence of density limit on the averaged plasma current density: disruptive density limit with  $P_{aux} = 1.9$  MW, crosses; non-disruptive high density discharges, triangles. The dashed line indicates the Greenwald limit, the chain line indicates double the Greenwald limit.

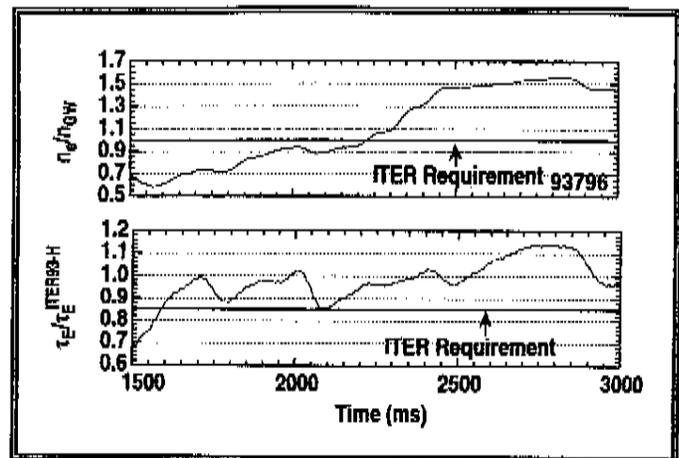


Fig. 2.4.5-2. Demonstration discharge with density  $\geq 1.5 \times$  Greenwald scaling and H-mode confinement.

- ARIES-AT will operate above the Greenwald density limit so we must have  $T_{edge} > 100$  eV and pumping on the inboard strike point region

## Plasma Transport Constraints

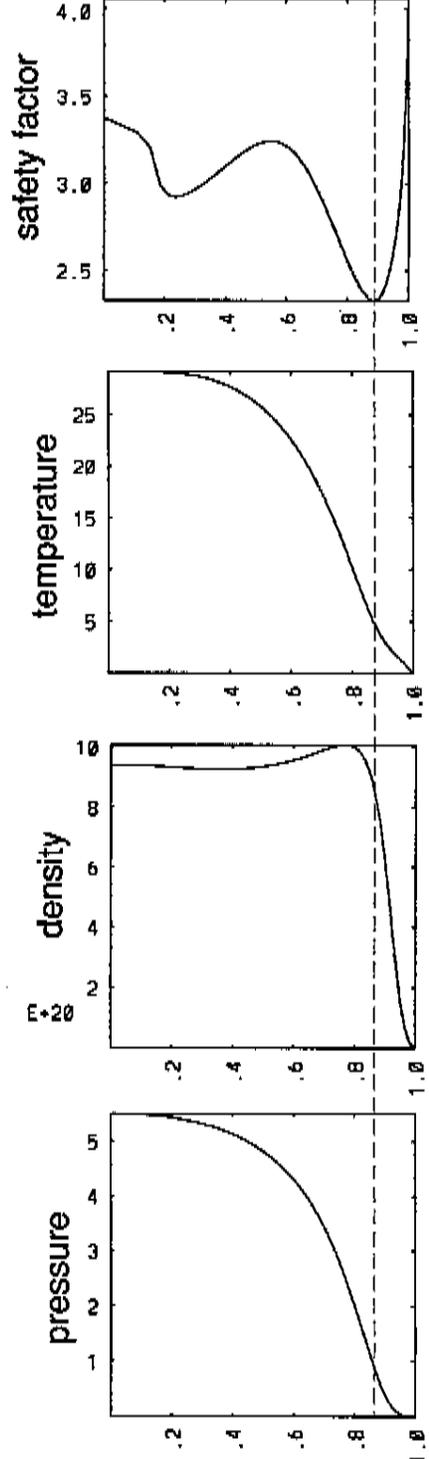
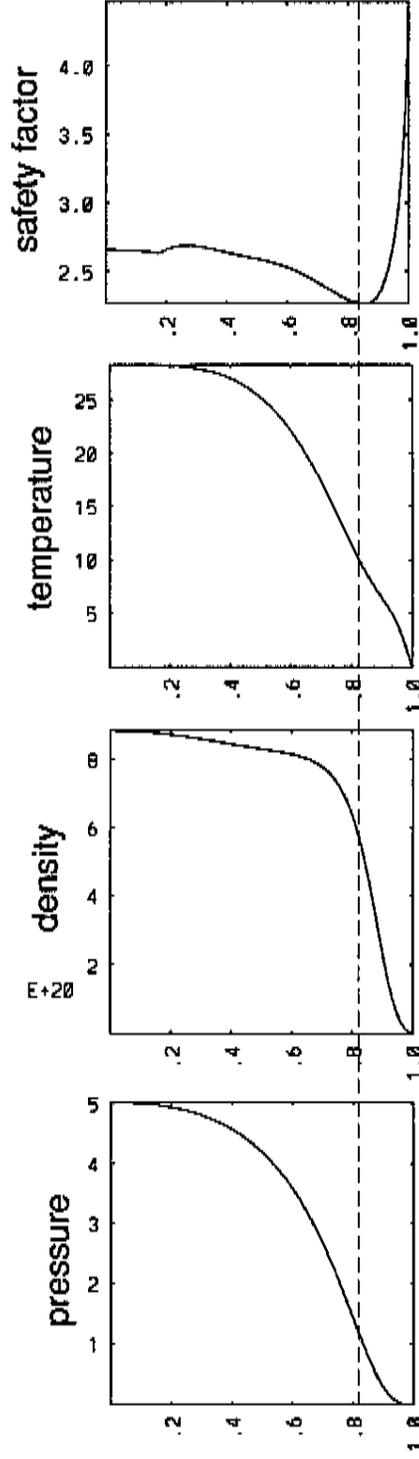
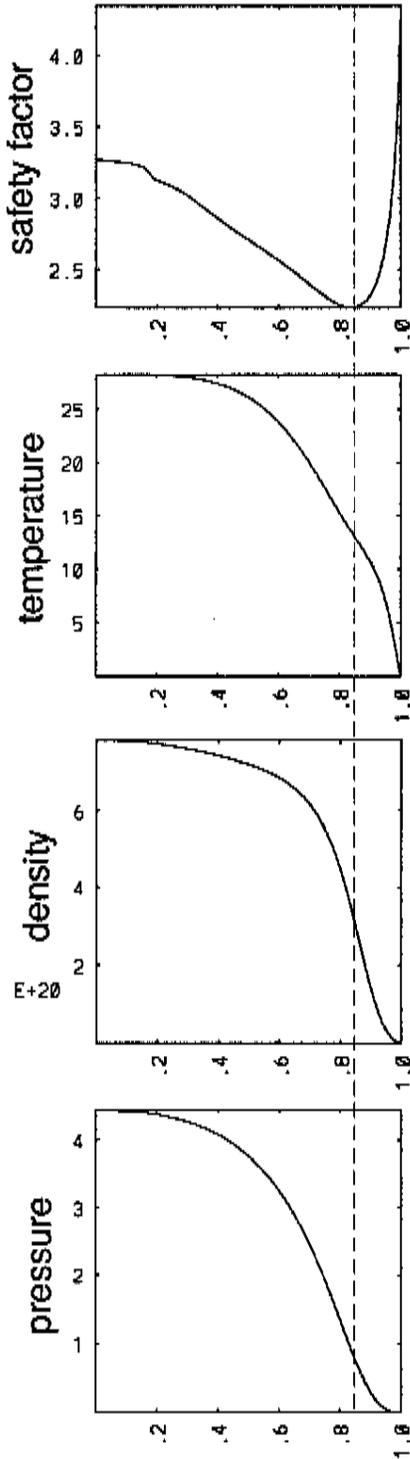
- in ARIES-RS the only constraint imposed on kinetic profiles ( $n, T$ ) was that the dominant gradient lie inside or around the  $q_{\min}$  location
- this constraint was used because there were, and still are, significant differences between RS profiles in various tokamak experiments
  - H-mode or L-mode edge
  - $T_e, T_i, n$  affected by ITB
  - heating schemes on various devices (access inside ITB)
- we are primarily interested in finding kinetic profiles that provide good bootstrap alignment and ideal MHD stability
  - the profiles used are not inconsistent with experimental observations
  - however, our gradients are spread out over a larger region than those observed on experiments, which would require some form of ITB control
- the characteristics of kinetic profiles outside of  $q_{\min}$  should be determined to give good bootstrap current and ideal MHD stability along with
  - neoclassical tearing mode stability
  - connection to divertor solution/radiated power fraction

# Density and Temperature Profile Assumptions

### ARIES-AT $\beta_N = 5.4$

### ARIES-AT $\beta_N = 6.0$

### ARIES-AT $\beta_N = 6.4$



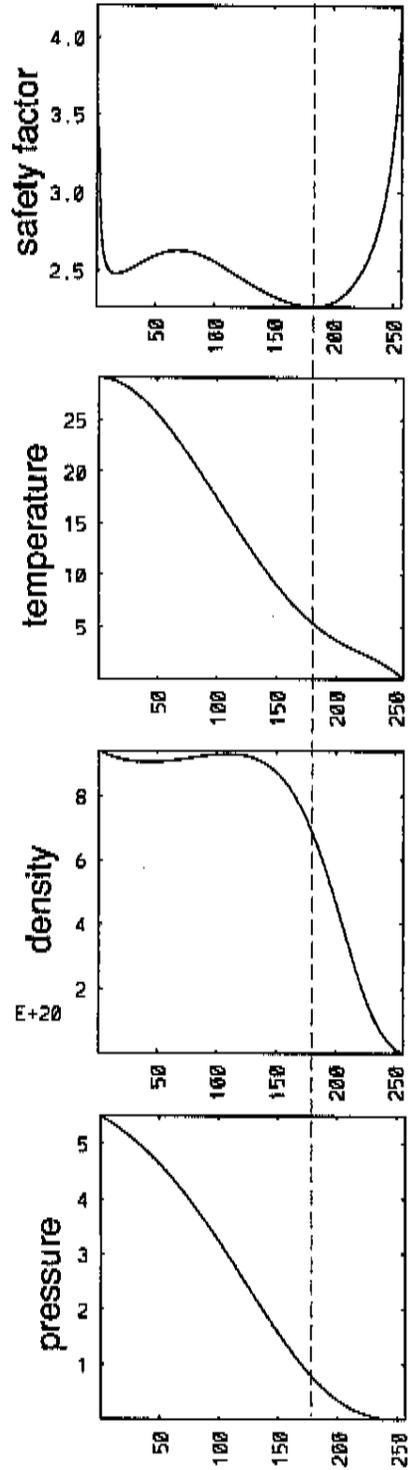
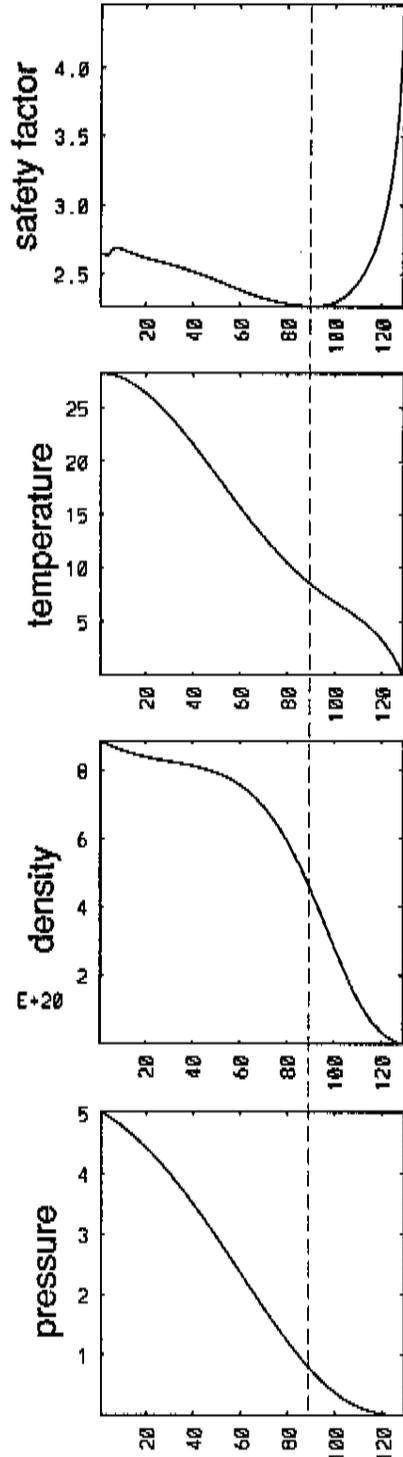
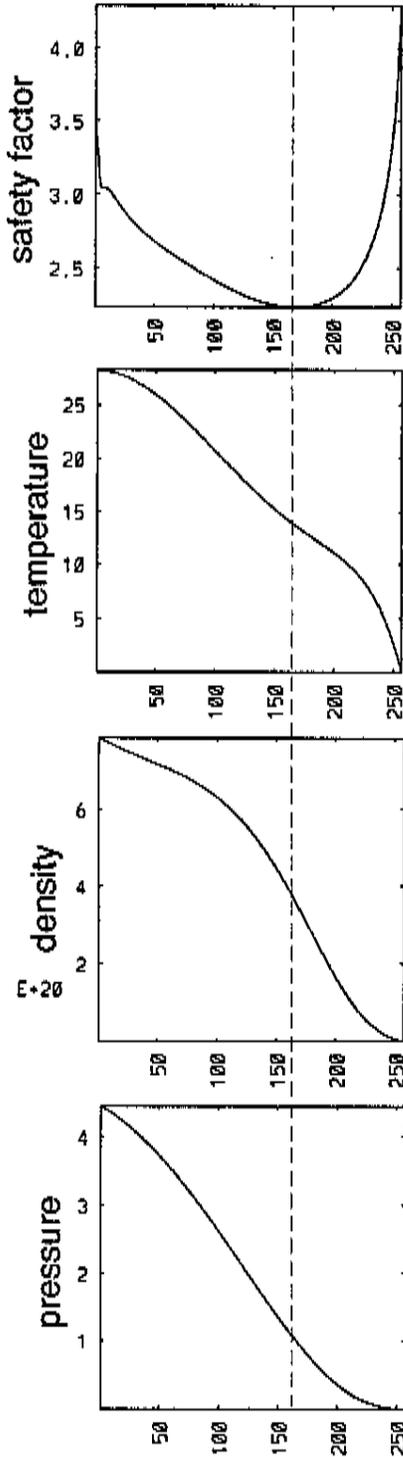
$\sqrt{V/V_0}$

# Density and Temperature Profile Assumptions

### ARIES-AT $\beta_N = 5.4$

### ARIES-AT $\beta_N = 6.0$

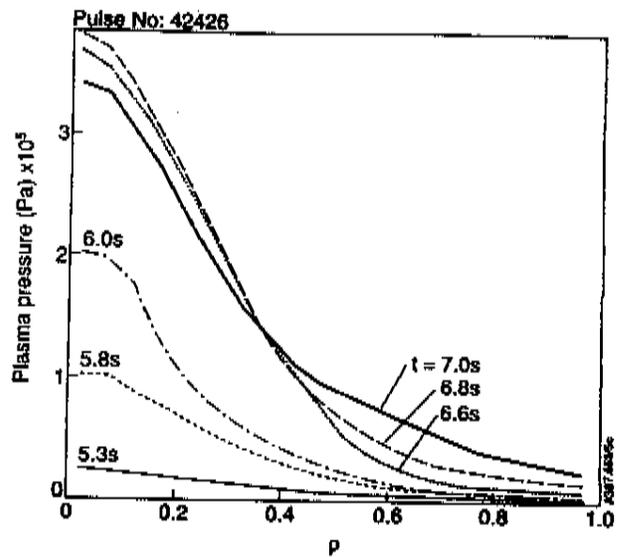
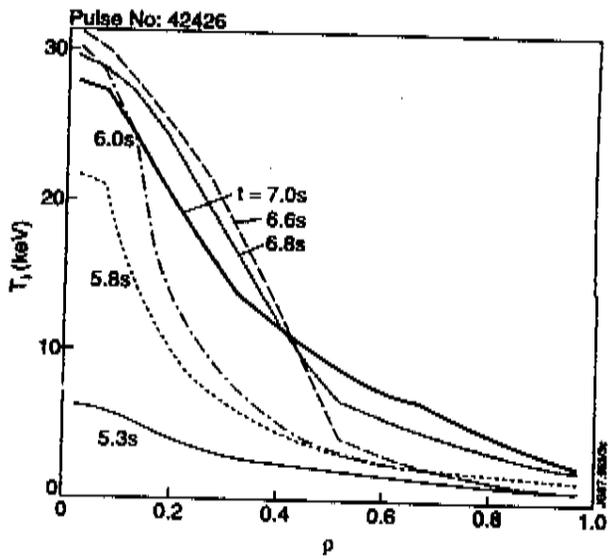
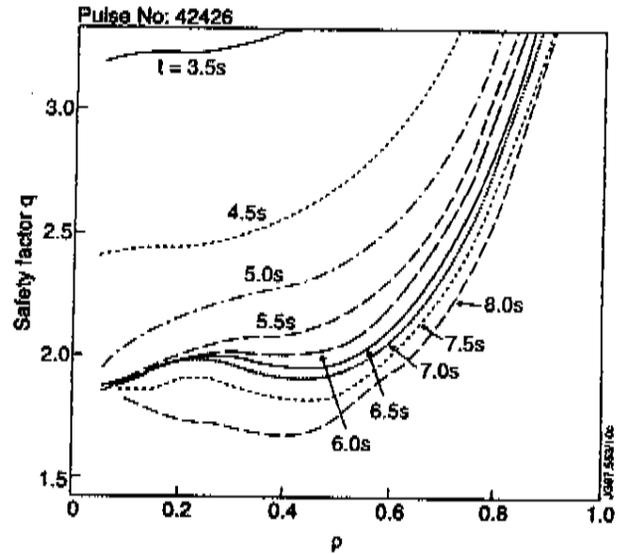
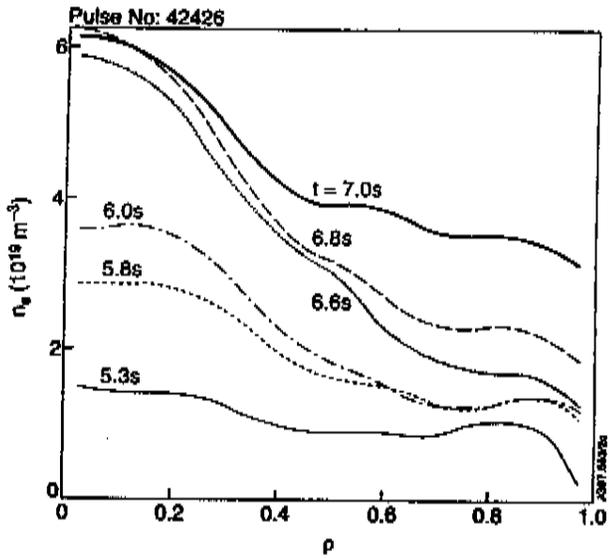
### ARIES-AT $\beta_N = 6.4$



$\psi$

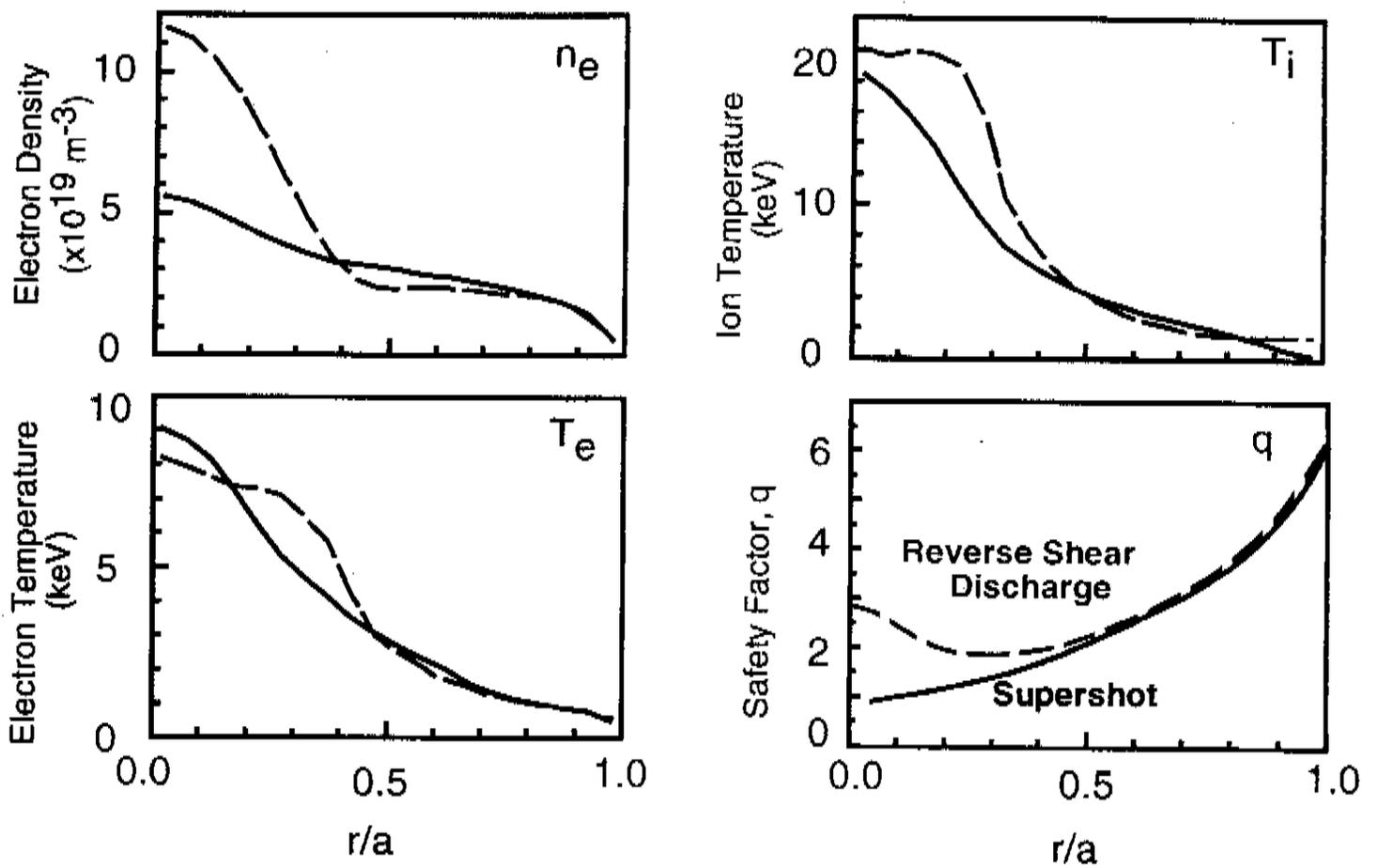
# Experimental RS Profiles

JET Optimized Shear



# Experimental RS Profiles

TFTR ERS



# Experimental RS Profiles

Tore Supra

DIII-D L-mode

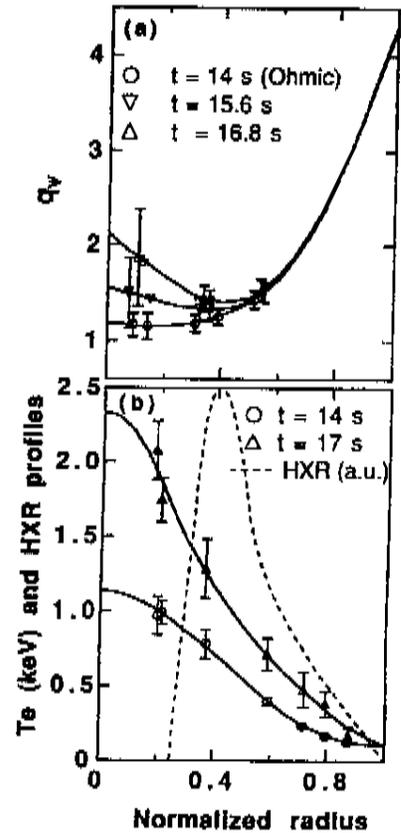
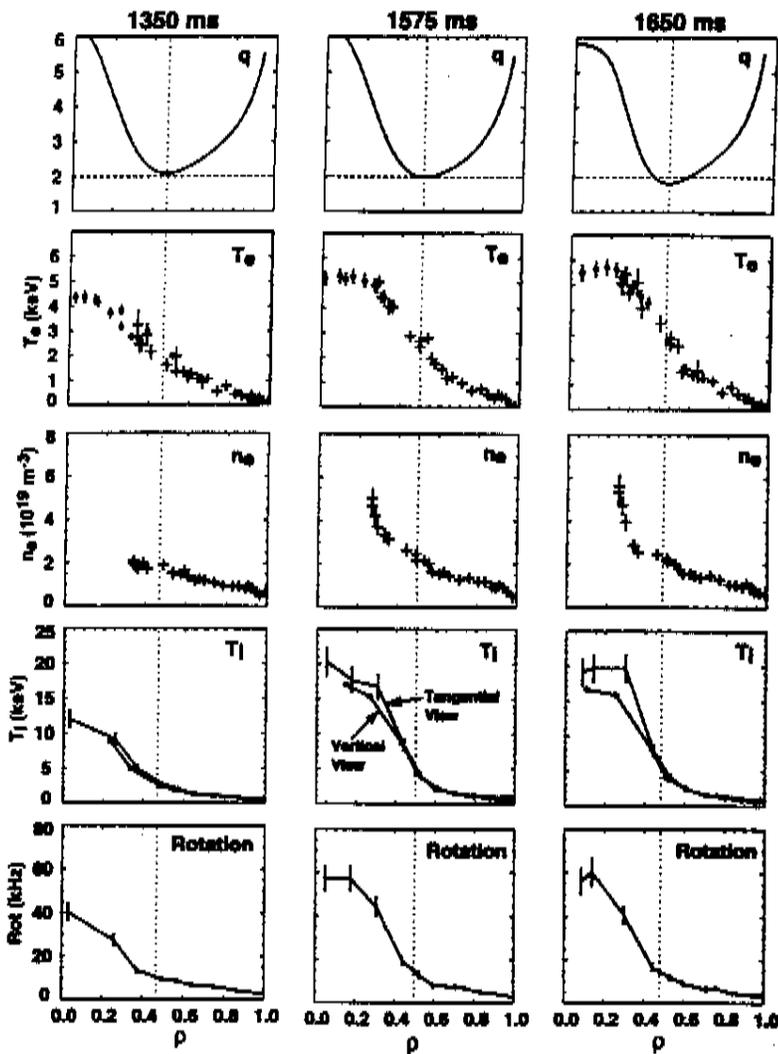


Figure 3. Measured radial profiles of a reversed central shear discharge ( $B_t = 1.3$  T,  $q_\psi(a) = 4.2$ ,  $I_p = 0.4$  A (shot 14409)): (a)  $q_\psi$ -profiles at three different times ( $t = 14.0, 15.6, 16.8$  s) from the ohmic phase ( $t = 14$  s) to stationary state with magnetic shear reversal ( $t \approx 16.8$  s); (b) electron temperature profiles (Thomson scattering) in ohmic (open circles) and LHCD (open triangles) phases together with the normalized Abel-inverted hard x-ray profile for photon energies  $h\nu = 75$  keV (dashed line).

Fig. 3. A time sequence of the measured profiles of  $q$  (MSE),  $T_e$  (Thomson scattering and horizontal ECE),  $n_e$  (Thomson),  $T_i$  and toroidal rotation (CER). The vertical dashed line represents the radius of  $q_{min}$ . The difference between the innermost vertical and tangential CER  $T_i$  measurements is due to differing systematic errors. The values plotted represent upper and lower bounds to the true  $T_i$ .

# Experimental RS Profiles

JT-60

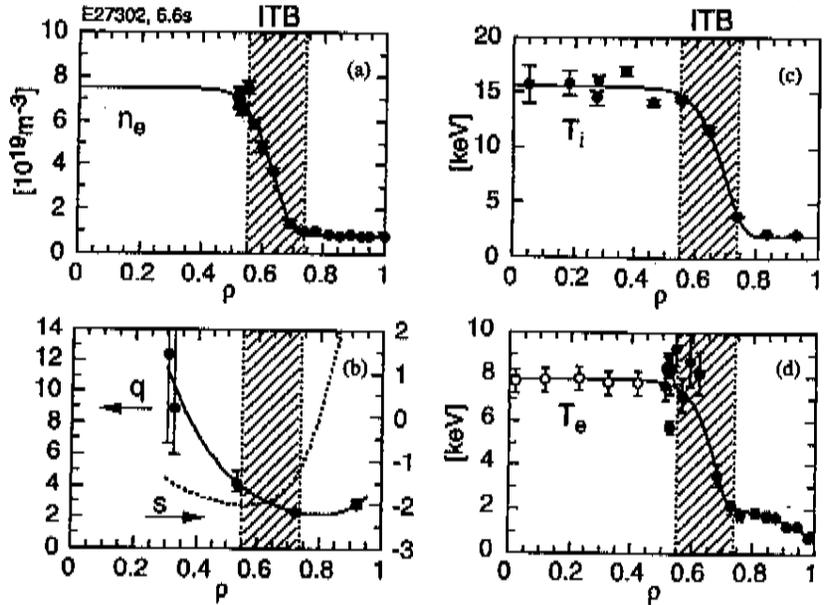


FIG. 3. Profiles for the discharge shown in Fig. 2 at  $t = 6.6$  s. (a) Electron density  $n_e$  measured by Thomson scattering (ruby laser). The solid curve is obtained to fit the interferometer data (one tangential and two vertical chords). (b)  $q$  and  $s$  (magnetic shear) profiles measured by MSE. (c)  $T_i$  measured by CXRS. (d)  $T_e$ ; the closed circles are measured by Thomson scattering (ruby laser) and the open circles by ECE.

ASDEX-U

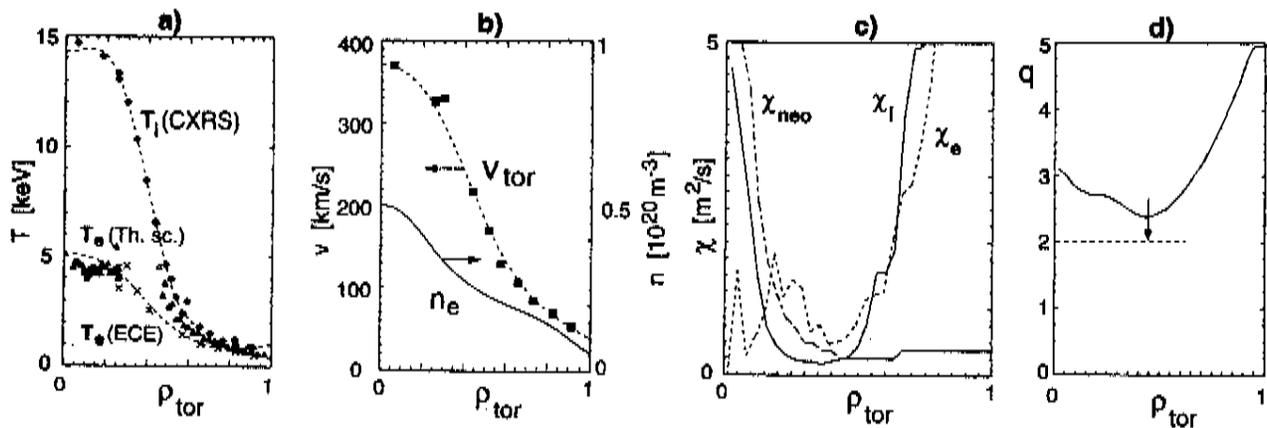


Figure 14. Radial profiles of ion and electron temperatures (ECE and Thomson scattering), electron density and toroidal rotational velocity ( $v_{tor}$ ) of a discharge with ITB, L mode edge and reversed shear with  $q_{min} > 2$  (discharge 10701, 0.84 s). The time point chosen is at maximum performance just before termination of the ITB. The ion and electron thermal conductivities ( $\chi_{i,e}$ ) from ASTRA analysis are compared with the neoclassical ion thermal conductivity ( $\chi_{neo}$ ) [10]. The calculated  $q$  profile is given in (d).

## Summary

- elongation studies show that  $\beta$  can be increased significantly
  - $\beta_N$ (ballooning) rises up to  $\kappa=2.4$ , and then drops
  - the kink stabilizing shell remains at  $0.2a$  and then drops rapidly beyond  $\kappa=2.2$
  - vertical stabilizing shells in the range  $0.25a-0.35a$  allow elongations in the range  $2.5-2.1$ , with  $\beta$ 's in the range of  $10.5-15.7\%$
  - likely configuration has  $\kappa = 2.2$ , and  $\beta = 11.6\%$
- finite edge densities allow reasonable divertor solutions, but affect bootstrap current and CD
  - used  $n(a)=0.2n(0)$  and found ideal MHD stable cases with increased off-axis CD
- we will be operating above the Greenwald density limit, and we have no inboard slot divertor to avoid strong recycling
  - recent experiments have demonstrated operation above the Greenwald density limit by avoiding MARFE's
  - we must have consistent story of how we will achieve this; inboard strike point pumping, to be demonstrated on DIII-D

## Summary, cont'd

- density and temperature profiles have been constrained only roughly up to now for ARIES-AT (and ARIES-RS)
  - our interests have been to optimize bootstrap current alignment and MHD stability
  - our assumed profiles are not inconsistent with experimental profiles, but have their gradients spread out more than those observed
  - we are assuming some form of ITB control to relax the gradients and spread them out
  - the profiles of density and temperature outside of  $q_{\min}$  are important as well, and will affect the divertor solution, CD, neoclassical tearing stability
- future work
  - begin to examine finite pressure gradient at the plasma edge (or just inside the edge)
  - poloidal field coil solution for ARIES-AT
  - aspect ratio effects ???