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MHD EQUILIBRIUM AND STABILITY CONSIDERATIONS FOR HIGH-ASPECT-RATIO ARIES-I TOKAMAK REACTORS*

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ABSTRACT

The requirements of an (1) external poloidal field coil (PFC) system with minimum stored energy, (2) double-null divertor plasmas with elongated D shape, (3) adequate passive stabilization of plasma vertical displacement by a vacuum vessel located behind the blanket zone, and (4) an enhanced plasma beta limit in the first stability regime are incorporated in the ARIES-I concept for a high-field tokamak reactor with high aspect ratio (A = 4.5). The plasma current and pressure profiles are also made consistent with enhanced bootstrap current I_{bs} and reduced current drive power by means of ion cyclotron wave (ICW) or neutral beam (NB) injection. These lead to plasmas characterized by an elongation κ_x of 1.8 to the divertor X-point, a triangularity δ_x of 0.7, a safety factor q on axis of $q_0 \approx 1.5$, a safety factor at the edge $q_{95} > 4$, a plasma beta $\beta \approx 2\%$, and a poloidal beta given by $\varepsilon\beta_p \approx 0.5$ ($\varepsilon = 1/A$). With a plasma current I_p of 11 MA, a toroidal field B_t of 13 T at the major radius R_0 of 6.5 m, and over 3.5 m of clearance between the PFCs and the plasma edge, the stored energy in the PFC system ranges from 20 GJ during plasma operation at low beta to 12 GJ during plasma operation at high beta.

INTRODUCTION

ARIES-I [1] is a tokamak reactor concept based on modest extrapolation from the near-term physics data base characterized by present ITER design assumptions [2], advanced technologies [3], attractive safety and environmental properties [4], and optimized reactor economy using costing assumptions projected for future tenth-of-a-kind reactors [5]. These latter characteristics engender requirements on the plasma design that lead to variations from ITER, which is a first-of-a-kind experimental device.

In the area of MHD equilibrium and stability, these different requirements include:

1. placing the PFCs at a distance of ≈2.5 times the plasma minor radius a away from the plasma edge, leading to large increases in the PFC stored energy,
2. placing passive conductor (such as the vacuum vessel) between the blanket and the shield at a distance of at least $0.6a$, leading to a reduced κ_x (= 1.8) and an increased δ_x (= 0.7) to ensure plasma vertical stability [6], which in turn lead to further increases in the PFC stored energy,
3. reducing the steady-state current drive power by limiting I_p to ≈10 MA while maintaining adequate H-mode plasma confinement [7],
4. raising the plasma $\varepsilon\beta_p$ to ≈0.5 to increase the bootstrap current fraction I_{bs}/I_p to ≈0.5 [8], and
5. enhancing the plasma beta in the first stability regime for plasma current profiles characterized by $q_0 \approx 1.5$ and $q_{95} > 4$, which are consistent with those producible by ICW or NB current drive [8]. Here q_{95} refers to the flux surface at 95% of the poloidal flux toward the divertor X-point.

Iterations with the current drive analysis, tokamak integration, and systems code calculations have led to the design parameters discussed here. This paper presents the results in the areas of PFC distribution, free-boundary MHD equilibria, PFC current and energy requirements, and MHD stability beta limit. Details of plasma vertical stability

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analysis for the ARIES-I plasma are presented in an accompanying paper [6].

PFC DISTRIBUTION

A recent study [9] of free-boundary divertor D-shaped plasmas showed that the PFC stored energy increases when κ_x is decreased for constant δ_x and constant distance between the PFC and the plasma edge. Since the plasma vertical stability is improved by lowering κ_x [10], it is important to minimize the increase in PFC stored energy as κ_x is reduced.

To this end, we study the dependence of the externally applied poloidal field on κ_x and δ_x. The analysis can be simplified by examining the lower-order multipole components of the poloidal field (nullapole, dipole, quadrupole, and hexapole), since these represent roughly the field properties of minimum stored energy. A set of free-boundary equilibria is calculated from fields composed of only these components, and a key result is shown in Fig. 1. It is seen that the desired δ_x depends strongly on κ_x when these multipole field components are used alone. When κ_x decreases from 2.2 to 1.8, δ_x must be increased by 0.3. It is also clear that, as δ_x increases under this restriction, the magnitude of the hexapole component increases relative to the quadrupole component. As a result, the divertor and D-shaping coils, located in the direction of the X-point of the plasma, are expected to carry relatively large currents.

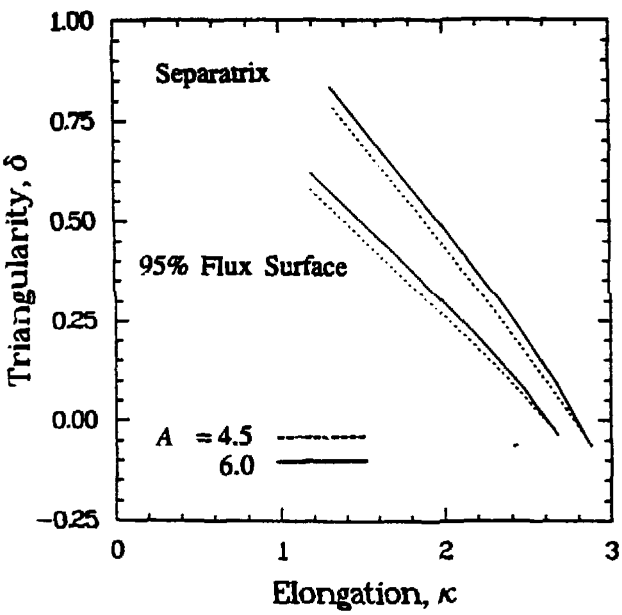

Fig. 1. The relationship between δ and κ for free-boundary plasmas calculated using only the dipole, the quadrupole, and the hexapole field components, for $\beta_p = 2.5$ and $A = 4.5$ and 6.

The overall distribution of the PFCs is then roughly determined, as shown in Fig. 2, which assumes six coil groups. This number is considered the minimum required because of the need for controlling the plasma position and shape (R_0, a, κ_x and δ_x), minimizing the stored energy, and providing some induction capability during plasma operation through varying plasma conditions [9]. It is clear that coil groups 1 and 2 contribute dominantly to the induction field (the nullapole), groups 1 and 6 to the vertical field (the dipole), groups 3 and 5 to the elongating field (the quadrupole), and groups 4 and 6 to the triangulating field (the hexapole). Higher-order multipoles exist in any PFC system with discrete coils. The PFCs are distributed so as to minimize the higher-order multipoles and thus the stored energy.

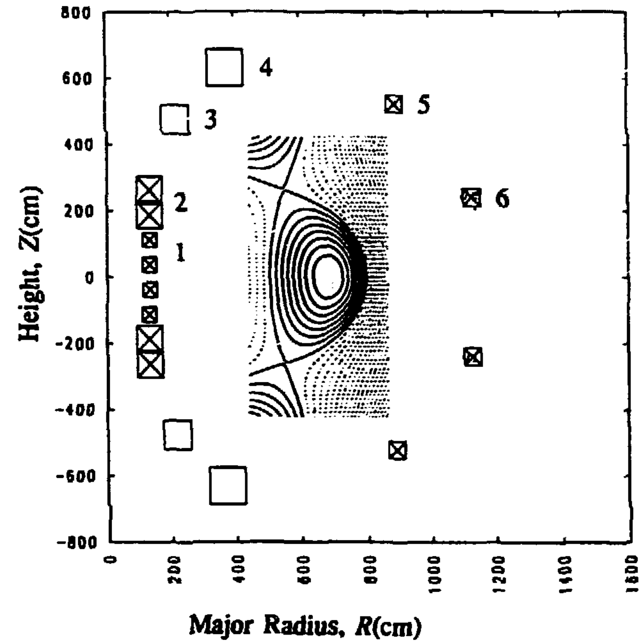

Fig. 2. Plasma equilibrium flux configuration and PFC placement.

FREE-BOUNDARY MHD EQUILIBRIA

Given the coil distribution, reference MHD equilibria for ARIES-I are computed using the VEQ code [9], which calculates free-boundary solutions for a given plasma position, shape, and linked poloidal flux while minimizing the stored energy. The plasma shape is chosen to have $\kappa_x = 1.8$ and $\delta_x = 0.7$ to allow for adequate vertical stabilization [6]. The plasma pressure and current profiles are consistent with an enhanced first stability beta and I_{bs}. The current profile should be maintainable solely by ICW or NB current drive [8]. Trade-offs among these requirements lead to a choice of profiles, that are close to the following pressure (p) and poloidal current profile (f) functions:

$$p'(x) = p_0(e^{-\alpha x} - e^{-\alpha})/(e^{-\alpha} - 1),$$

$$ff'(x) = \mu_0 R_0^2 p_0 (1/\beta_J - 1)(e^{-\gamma x} - e^{-\gamma})/(e^{-\gamma} - 1),$$

where x is the poloidal flux normalized to 1 within the plasma. The toroidal plasma current density is

$$J_t = Rp' + ff'/\mu_0 R,$$

where R is the major-radius variable. A reference equilibrium assuming $\alpha = -3$, $\gamma = -3$, and $\beta_J = 2.75$ is provided in Fig. 2 (the poloidal flux distribution), Fig. 3 [the $p(R)$ and $J_t(R)$ profiles], and Fig. 4 [the $q(x)$ profile]. Parameters of this reference equilibrium that are relevant to stability and current drive analyses are given in Table I.

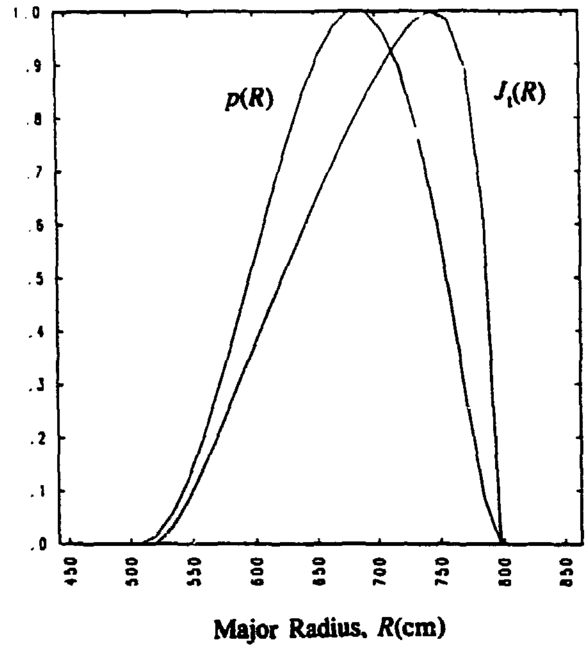

Fig. 3. Pressure and toroidal current density profiles along the major radius R for the plasma shown in Fig. 2.

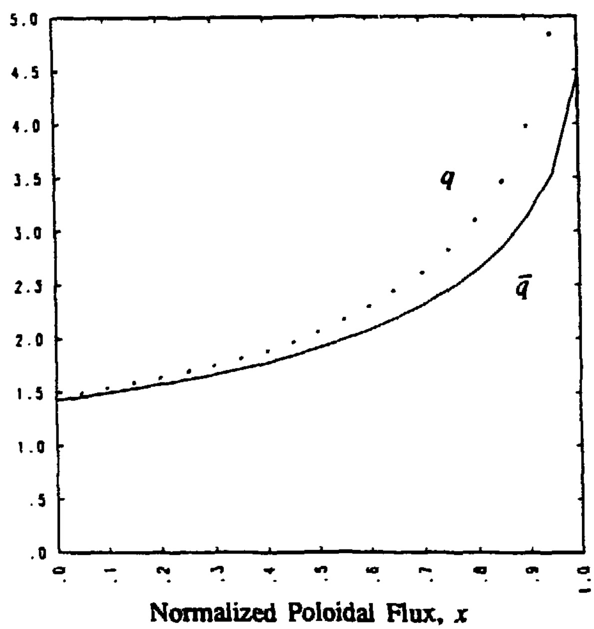

Fig. 4. The profiles of q and $\bar{q}$ (the average-field safety factor) along x, the normalized poloidal flux within the plasma edge.

This reference case provides an adjustment to the conditions that relate I_p to q, a, B_t, and the plasma shape parameters:

$$I_p\bar{q} = 5aB_t[\varepsilon(1.15 - 0.65\varepsilon)/(1 - \varepsilon^2)^2](1 + \kappa_x^2)/2,$$

where $\bar{q}$ is the average-field safety factor using the averaged poloidal field at the plasma edge. For the reference equilibrium, we also have

Table I. Plasma Parameters of a Reference Divertor MHD Equilibrium for the ARIES-I Concept

Parameter	Symbol (unit)	Value
Major radius	R_0 (m)	6.53
Minor radius	a (m)	1.45
External toroidal field at R_0	B_t (T)	13.0
Plasma current	I_p (MA)	11.1
Safety factor on axis	q_0	1.45
Average-field safety factor	$\bar{q}$	4.47
Safety factor at 95% flux	q_{95}	4.85
Average beta	β (%)	1.90
Poloidal beta	β_p	2.18
Elongation at x-point	κ_x	1.80
Elongation at 95% flux	κ_{95}	1.62
Triangularity at x-point	δ_x	0.70
Triangularity at 95% flux	δ_{95}	0.44
X-point location	R_x (m)	5.51
	Z_x (m)	2.61
Internal inductance	l_i	0.74

$$\delta_x/\delta_{95} = 1.59, \quad \kappa_x/\kappa_{95} = (1.13 - 0.08\varepsilon), \quad q_{95}/\bar{q} = 1.09$$

to relate the edge and 95% flux surface quantities.

Different forms of the profile functions can be used to produce equilibria nearly identical to this case in all its global parameters as given in Table I. The results of the free-boundary equilibrium and the PFC currents do not change significantly when these different profile functions are used as long as the global parameters remain unchanged.

PFC CURRENT AND ENERGY REQUIREMENT

The PFC cross sections shown in Fig. 2 assume an overall current density of 20 MA/m^2 for each coil. The maximum values of the PFC currents during plasma operation, together with the required distance from the toroidal field coils and their structure [11], contribute to determining the locations of the PFCs. It is therefore necessary to calculate the PFC current variations through typical conditions of the plasma during operation.

Since the ARIES-I tokamak assumes noninductive methods to assist startup of the plasma current [8], the amount of poloidal flux linkage between the plasma and the PFCs can be chosen to reduce the PFC stored energy. Some flexibility exists near the condition of minimum stored energy to vary the PFC currents and provide some degree of induction for plasma operation. The range of plasma parameters can therefore be characterized by three separate conditions, all at full plasma current with a fixed X-point location: low beta and low linked flux, high beta and low linked flux, and high beta and high linked flux. The PFC currents are listed in Table II for these cases. The maximum current for each coil is then estimated and used in sizing its cross sections and locating the coil as plotted in Fig. 2. These data are also used as input to the PFC design concept [12].

Table II. PFC currents and the maximum currents for each coil group shown in Fig. 2 at three typical operation conditions. (Coil groups 1 and 2 have 2 coils each.)

Operation conditions	I	II	III	Maximum
β (%)	0.62	1.90	1.90	
Linked flux (Wb)	39.2	50.5	90.1	
Stored energy (GJ)	19.8	13.2	12.2	
Coil group current (MA·turn)				
I_1	−14.8	−0.0	−7.5	15.0
I_2	−20.9	−12.0	−26.0	26.0
I_3	20.0	20.0	15.0	20.0
I_4	33.7	26.4	24.1	34.0
I_5	−10.7	−4.8	−5.2	11.0
I_6	−0.9	−5.3	−5.3	6.0

It is seen from Table II that an induction flux of about 40 Wb is available by varying the nullapole component of the PFC currents, leading only to a small (<10%) change in the stored energy from its minimum of about 12 GJ. However, the stored energy is significantly larger (about 20 GJ) at low beta and full I_p if the X-point is to be fixed during plasma heating to burn at high beta (*e.g.*, to satisfy the divertor operation requirements). This difference in the stored energy leads to a reactive power supply requirement of hundreds of megavoltamperes during a plasma heating time of 10–20 s. However, this requirement may not be necessary if the plasma can be heated to high beta during current ramp-up.

MHD STABILITY BETA LIMIT

The first stability regime requires that all ideal MHD modes be at least marginally stable in the absence of a conducting shell beyond the plasma edge [13]. While this requirement is broad in scope, it is usually adequate to examine only the high-n ballooning modes and the low-n (n = 1) kink modes to determine the stability beta limit. The intermediate-n ballooning modes (the "infernal" modes) are easily avoided by retaining small gradients in the q-profile near the plasma axis.

As an input to design trade-offs involving plasma shaping, profiles, A, and the beta limit, our study emphasizes clarifying the dependences of beta on A, κ_{95}, q_0, and q_{95}. We use only the traditionally successful profile functions for the analysis. This study is therefore limited in its scope, since several other parameters, such as δ, the q-profile, and the pressure profile, also affect the plasma beta limit. However, the study benefits from an extensive study of the beta limit recently carried out for ITER [14] and from reviews of the large body of information in the literature. Calculations are carried out for high A (4.5 and 6.0) ARIES-I plasmas using the PEST equilibrium and stability codes [15] to "fill in" data where needed. The combined data base of the stability analysis covers a range of A = 2.6–6.0, κ_{95} = 1.6–3.2, q_0 = 1.05–2.0, and $q_{95} \le 5$.

The p and q profile functions used include those optimized for JET plasmas [13] and those used in the ITER studies [14]:

$$p = p_0[(1 - y^{\alpha})^{\gamma} + p_1 y^{\zeta}(1 - p_2 y^{\eta})],$$

$$q = q_0 + q_1 y^{\lambda} + q_2 y^{\nu} \text{ or } q = q_0/(1 - \xi y^{\rho})^{\sigma}.$$

Here, y is the poloidal flux normalized to 95% of the X-point flux, p_0 determines beta, p_1 determines the profile, α = 1.5, γ = 2.5, ζ = 3, η = 1.2, and p_2 = 0 or 1. The first q-profile gives q_{95} (= $q_0 + q_1 + q_2$) = 3.1, where q_1 and q_2 are independent variables; λ = 6; and ν = 2. The second function for q has $\sigma = \ln(q_0/q_{95})/\ln(1 - \xi)$, ξ = 0.7, and ρ = 2, and is used for the case with $A \approx 6$.

The shape of the 95% flux surface is given by

$$R = R_0 + a\cos(\theta + \delta'\sin\theta), \quad Z = \kappa_{95} a \sin\theta,$$

where $\delta' \approx \delta_{95}$. The 95% flux surface is used in the stability analysis to avoid the numerical difficulties near the X-point, which are a subject of present investigations [16].

The results are summarized in Figs. 5 and 6, which indicate the dependence of beta-limit on κ_{95} (for $A \approx 6$) and the dependence of Troyon factor limit ($C_T = \beta a B_t/I_p$, in % m T/MA) [13] on A. From these, one obtains

$$C_T \approx 2.8[1 - 0.4(\kappa_{95} - 1)^2]/(1 - \varepsilon)^{1.5},$$

which gives $C_T \approx 3.5$ and $\beta \approx 2.06\%$ for the reference plasma parameters in Table I. It is important to note that this approximate scaling has a limited basis; its use should be limited to the profiles given here and to the range of parameters indicated above. For $A \approx 3$, it has also been shown that this beta limit remains relatively unchanged as long as l_i remains below 0.75.

Additional studies of the beta limit have also been carried out for plasmas using polynomial profiles and with parameters encompassing the reference case: I_p ranging from 16 to 8 MA, q_{95} from 3 to 6, and β_p from 1.4 to 3. The value of C_T is 3.1 to 3.2 as long as q_{95} is above 3.7. This result is considered conservative relative to the preceding indications. Design values of C_T = 3.2 (corresponding to β = 1.9%) and l_i = 0.74 are therefore adopted for ARIES-I (see Table I).

DISCUSSION

The results summarized in this paper, together with those of Ref. [6], provide a relatively sound basis for the plasma equilibrium and stability of the ARIES-I reactor concept. They also provide approximate scaling relationships of the beta limit that are useful in the systems trade-off studies [5] needed to choose the ARIES-I parameters. The key parameters produced by our study are given in Tables I and II. They are made consistent with the current drive requirements [8] after reducing C_T from the nearly stability-optimized value of about 3.5 to a more conservative limit of 3.2, which assumes relatively "mild" pressure and current profiles. It is felt that a value of $C_T = 3.5$ can also be made consistent with steady-state current drive, but may lead to more stringent requirements on current drive, given more detailed analysis.

Iterations with the ARIES-I design integration have led to the reference PFC configuration of minimum stored energy as shown in Fig. 2. This configuration provides adequate flexibility for maintaining proper plasma position, shape, and flux linkage during heating and burn operation. Our results suggest the study of various scenarios of plasma heating and current ramp-up to reduce the stored poloidal field energy required at low beta.

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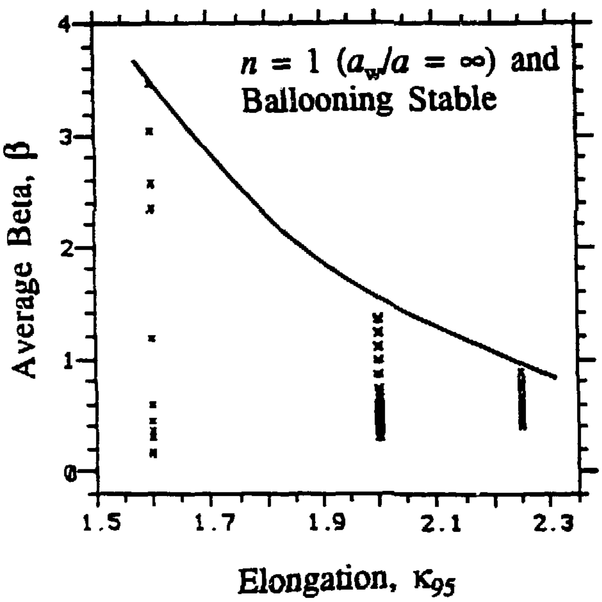

Fig. 5. Plasma beta values (x) of stable equilibria for κ_{95} = 1.6, 2.0, and 2.25 with $A \approx 6$ using the profile functions given.

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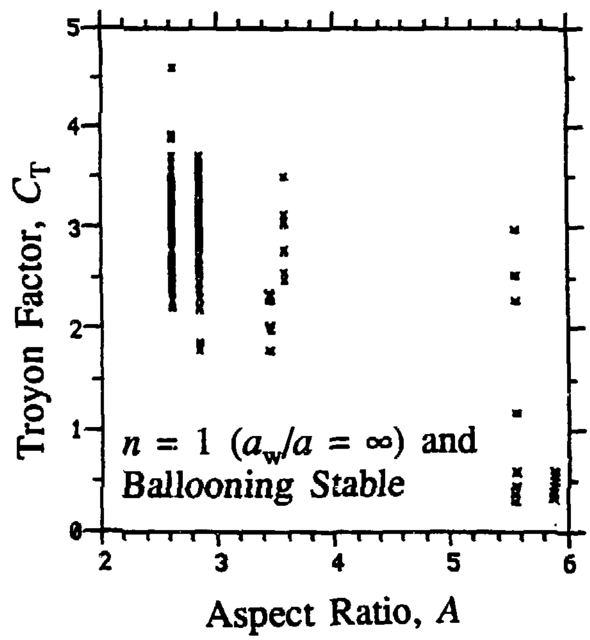


Fig. 6. Troyon factor (C_T) values (x) of stable equilibria for A = 2.6–6 with κ_{95} = 1.6–3.2, q_0 = 1.05–2.0, and $q_{95} \le 5$ using the profile functions given.

ARIES-I TOKAMAK REACTOR CONCEPT PHILOSOPHY
[Najmabadi, 21-I-01]

- Modest Extrapolation From the Near-Term Physics Data Base Assumed for ITER, such as
  - Enhanced first stability β limit
  - Adequate plasma vertical stability
  - Steady state current drive using ICW or NB
  - Somewhat higher bootstrap current fraction
  - Double-null, open divertor configuration

- Advanced Technologies, such as
  - 24-T superconductor with high strength material
  - Low activation material for support structure
  - Helium cooled solid breeder (Li_4SiO_4) blanket with SiC structure
  - Regenerative Rankine steam cycle with supercritical pressure and multiple reheat, 48% gross thermal efficiency

- Attractive Environment and Safety Characteristics

- Optimized Reactor Economy Using Aggressive Costing Assumptions

These lead to plasma design requirements that are different from ITER.

DESIGN REQUIREMENTS IN MHD EQUILIBRIUM AND STABILITY

1. Poloidal Field Coil (PFC) placement at a distance about 2.5 times the plasma minor radius a away from the plasma edge **[Grotz, 21-O-03]**,

2. Placement of passive conductor (as the vacuum vessel) between the blanket and the shield at a distance near or greater than 0.6a, leading to a reduced κ_x and an increased δ_x to insure plasma vertical stabilizability **[Bathke, 19-P-22]**,

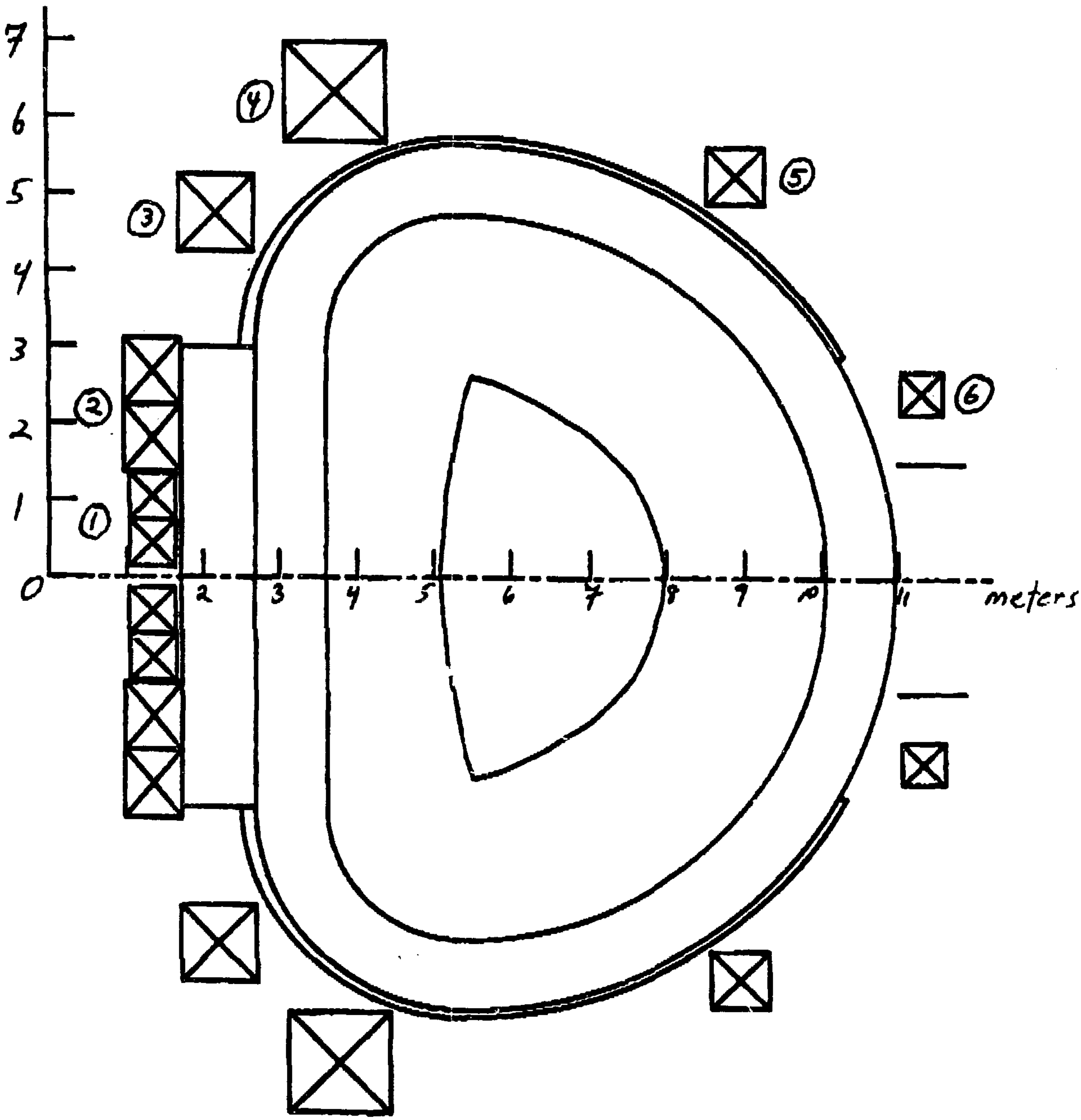
3. Reduction of the steady state current drive power by lowering I_p to close to 10 MA while still maintaining adequate plasma confinement in H-mode plasmas **[Miller, 21-O-02]**,

4. Raising the plasma $\varepsilon\beta_p$ to around 0.5 to increase the bootstrap current fraction I_{bs}/I_p to around 0.5 **[Mau, 08-O-02]**, and

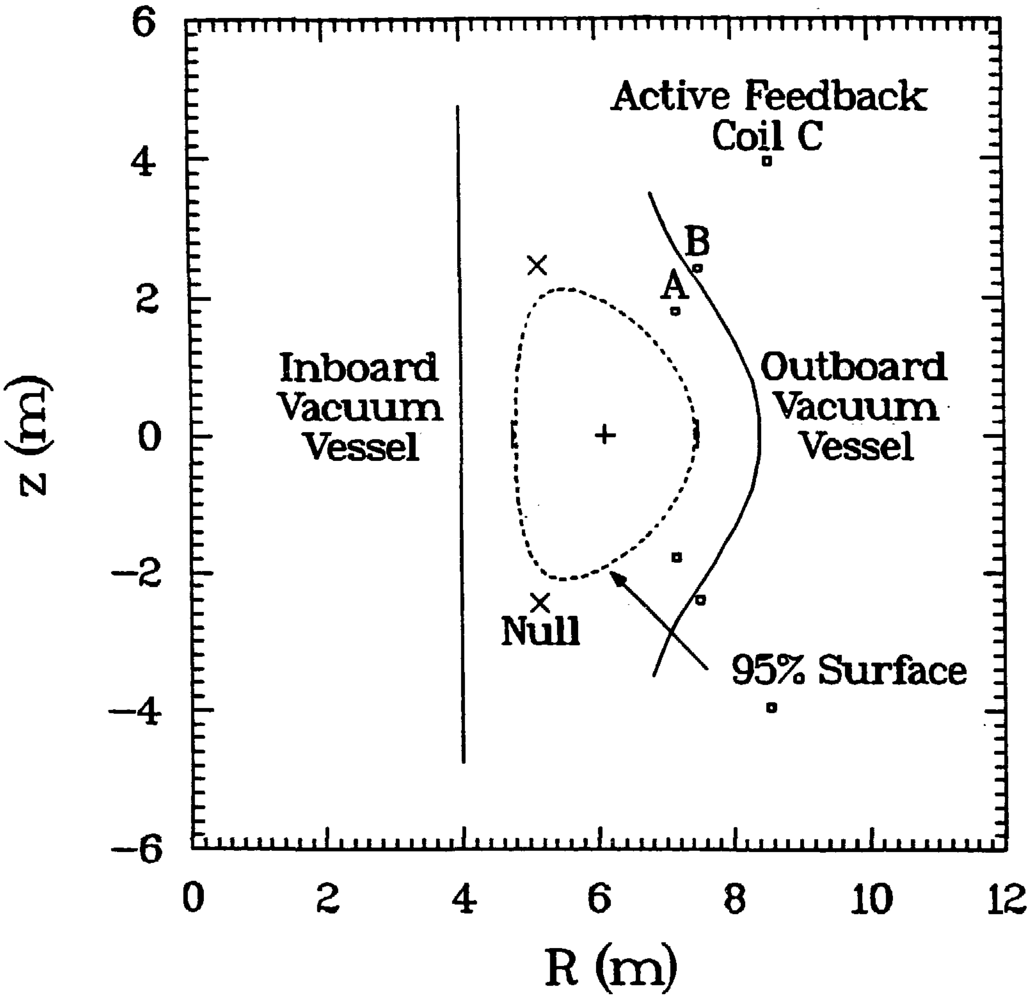
5. Enhancing the plasma β in the first stability regime using plasma current profiles characterized by $q_0 \approx 1.5$ and $q_{95} > 4$, which are consistent with those producible by ICW or NB current drive **[Mau, 08-O-02]**.

POLOIDAL FIELD COILS ARE LOCATED FAR (2.5*a*) FROM THE PLASMA EDGE

Elevation View of ARIES-I Reactor



VACUUM VESSEL AS PASSIVE STABILIZER IS ALSO FAR (0.6*a*) FROM THE PLASMA EDGE



Plasma elongation $\kappa_x = 1.8$ and triangularity $\delta_x = 0.7$ are needed adequate plasma vertical stability.

MHD EQUILIBRIUM AND STABILITY DESIGN

- Poloidal Field Coil distribution
- Free-Boundary MHD Equilibria
- Poloidal Field Coil Current and Energy Requirements
- MHD Stability Beta Limit

These areas have strong interaction with tokamak integration, current drive, and systems tradeoff.

PFC LOCATIONS ARE DETERMINED FOR MINIMUM STORED ENERGY AND ADEQUATE FLEXIBILITY IN PLASMA OPERATION

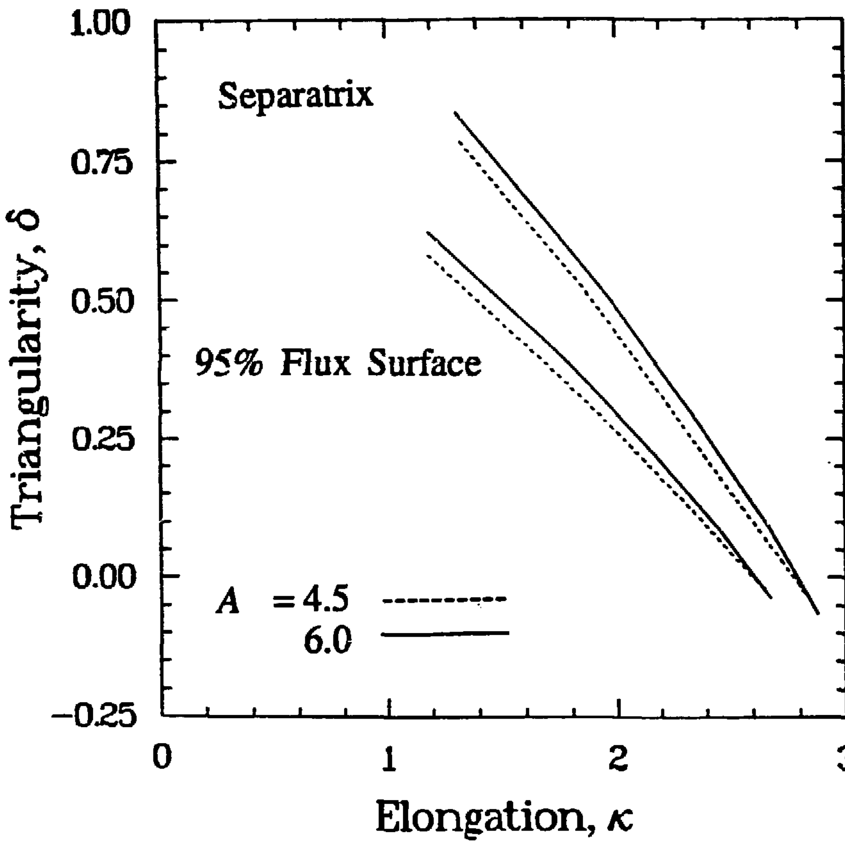
- For constant distance between PFC and plasma, stored energy **increases** when divertor plasma elongation κ_x is reduced from 2.5 [Strickler *et al.*, 1988].

- As κ_x is reduced, δ_x needs to be increased to reduce PFC stored energy.

- This increases the shaping and divertor coil currents (the hexapole field).

- 6 coil groups are needed to maintain plasma location and shape (R, a, κ_x, and δ_x), minimize the stored energy, and provide some induction.

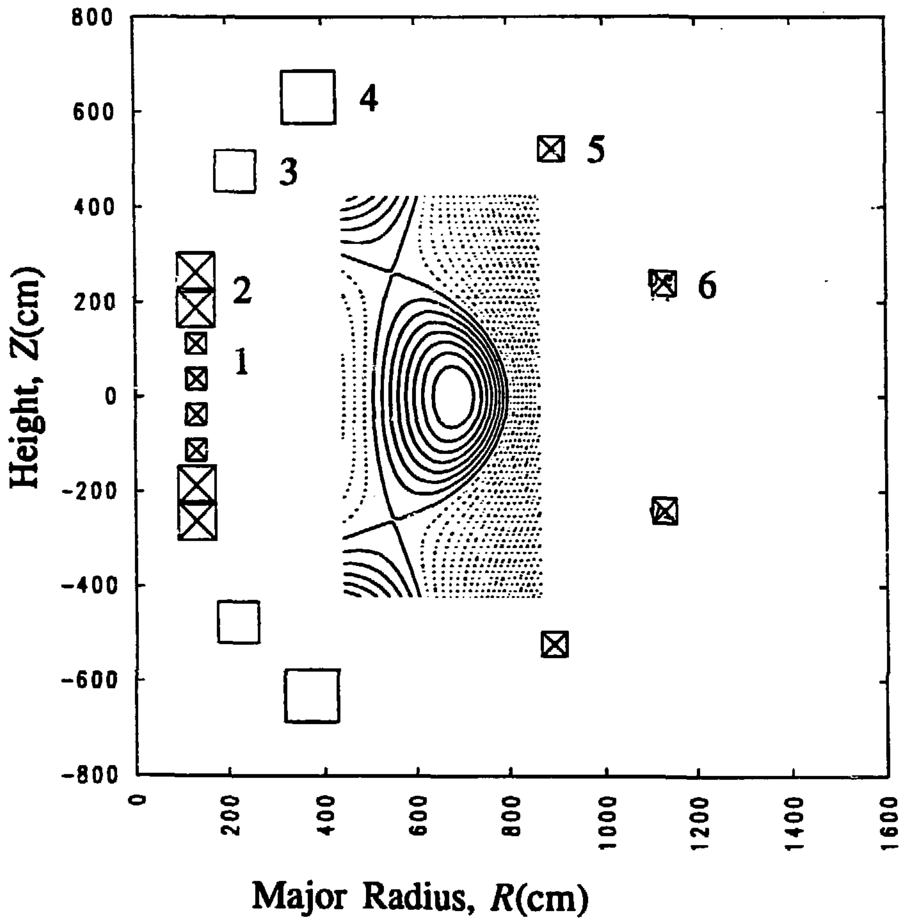
DIVERTOR TRIANGULARITY NEEDS TO INCREASE WHEN ELONGATION IS REDUCED TO MINIMIZE STORED ENERGY



The hexapole component increases relative to the quadrupole component when κ_x is reduced and δ_x is increased, leading to large shaping and divertor coil currents.

SIX COIL GROUPS ARE NEEDED TO HANDLE SIX PARAMETERS:

[R, a, κ_x, δ_x, linked flux, and stored energy]



RELATIVELY "MILD" FORMS OF PLASMA PROFILES ARE USED

- Pressure

$$p'(x) = p_0(e^{-\alpha x} - e^{-\alpha})/(e^{-\alpha} - 1),$$

- Poloidal Current f'

$$ff'(x) = \mu_0 R_0^2 p_0(1/\beta_J - 1)(e^{-\gamma x} - e^{-\gamma})/(e^{-\gamma} - 1),$$

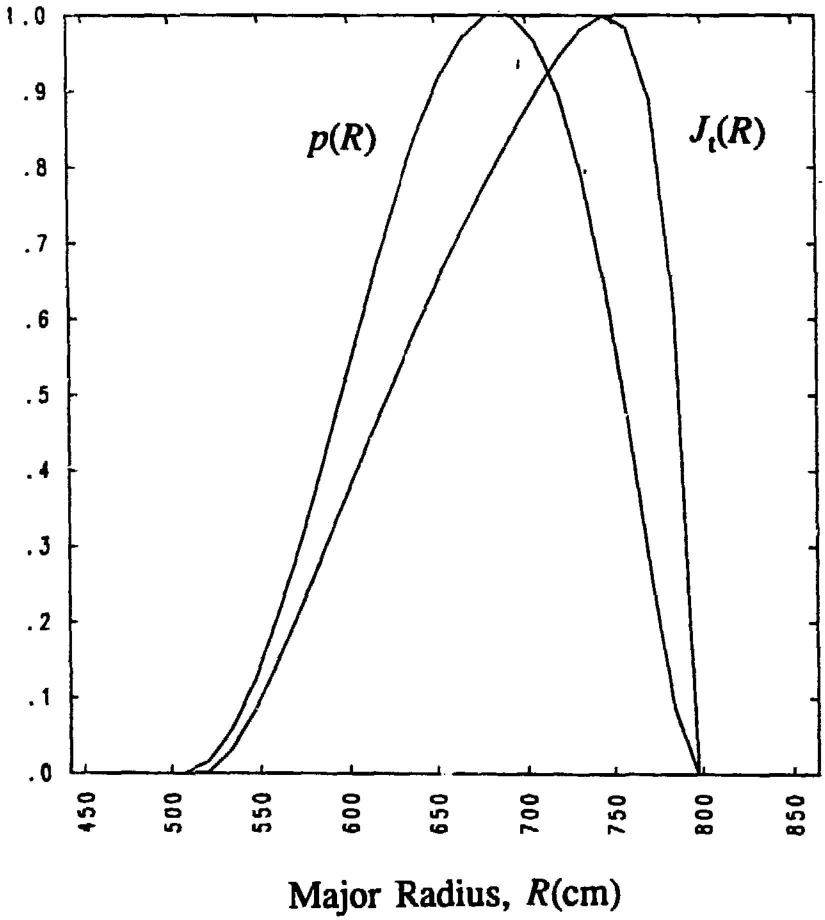
- Toroidal Current

$$J_t = Rp' + ff'/\mu_0 R,$$

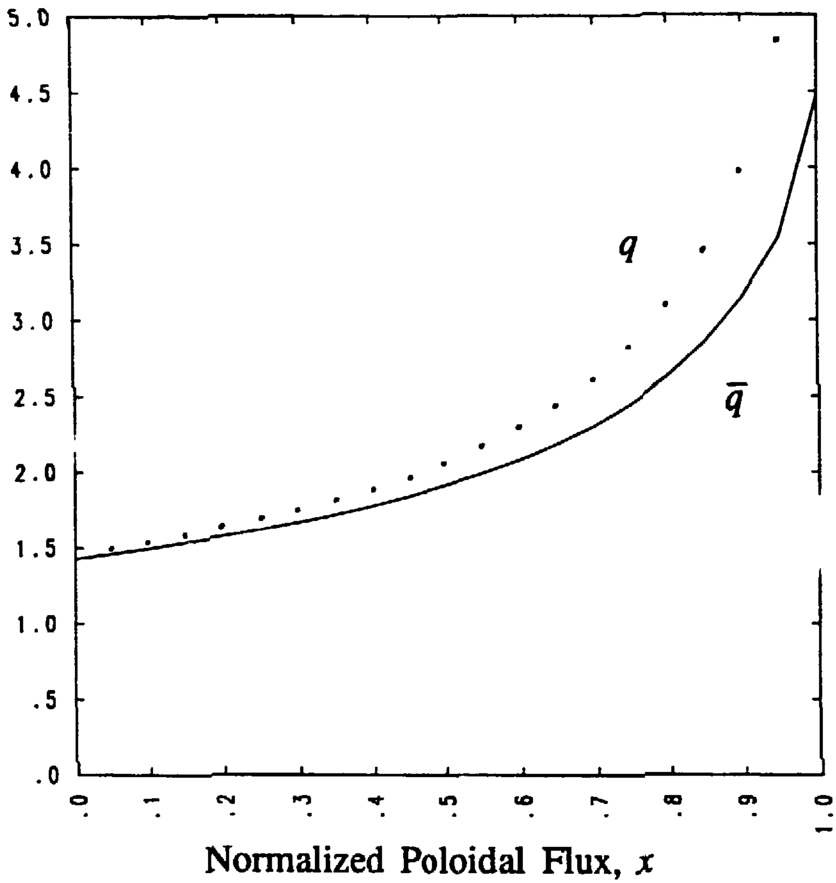
- Typically, $\alpha = -3$, $\gamma = -3$, and $\beta_J = 2.75$.

TOROIDAL CURRENT PROFILE (J_T) IS PRODUCIBLE BY ICW OR NB CURRENT DRIVE

[Mau, 08-O-02]



FOR PLASMA CURRENT NEAR 10 MA, SAFETY FACTOR q HAS HIGH VALUES (q_0 = 1.5 AND q_{95} = 4.9)



Low I_p reduces the current drive requirement. Low I_p also increases β_p, which increases the bootstrap current and further reduces current drive requirement.

DIVERTOR EQUILIBRIUM PARAMETERS FOR ARIES-I

Parameters	Symbol(Unit)	Value
Major Radius	R_0 (m)	6.53
Minor Radius	a (m)	1.45
External toroidal field at R_0	B_t (T)	13.0
Plasma current	I_p (MA)	11.1
Safety factor on axis	q_0	1.45
Safety factor at 95% flux	q_{95}	4.85
Average beta	β (%)	1.90
Poloidal beta	β_p	2.18
Elongation at x-point	κ_x	1.80
Elongation at 95% flux	κ_{95}	1.62
Triangularity at x-point	δ_x	0.70
Triangularity at 95% flux	δ_{95}	0.44
X-point location	R_x (m)	5.51
	Z_x (m)	2.61
Internal inductance	l_i	0.74

OPERATING CONDITIONS AND COIL CURRENTS

Operation Conditions	I	II	III	
β (%)	0.62	1.90	1.90	
Linked flux (Wb)	39.2	50.5	90.1	
Stored energy (GJ)	19.8	13.2	12.2	
Coil group current (MA-t)				maximum
I_1	−14.8	−0.0	−7.5	15.0
I_2	−20.9	−12.0	−26.0	26.0
I_3	20.0	20.0	15.0	20.0
I_4	33.7	26.4	24.1	34.0
I_5	−10.7	−4.8	−5.2	11.0
I_6	−0.9	−5.3	−5.3	6.0

- About 40 Wb induction flux is available without dictating large increases in the stored energy.

- Low β stored energy (20 GJ) much higher than at high β (13 GJ). Heating during current ramp up to reduce the former?

FIRST STABILITY CRITERIA:

MARGINAL STABILITY IN ABSENCE OF CONDUCTING SHELL

High-n Ballooning Modes

Low-n Kink Modes

Intermediate-n "Infernal" Modes

Only the ballooning modes and the $n = 1$ kink mode are analyzed.

The "infernal" modes can be stabilized by small q-gradients near the plasma axis.

WITH SPECIFIC ANALYSIS AT HIGH ASPECT RATIO AND USING ITER STABILITY DATA BASE, STABILITY INFORMATION OVER A WIDE RANGE ARE AVAILABLE

$$A = 2.6 - 6$$

$$\kappa_{95} = 1.6 - 3.2,$$

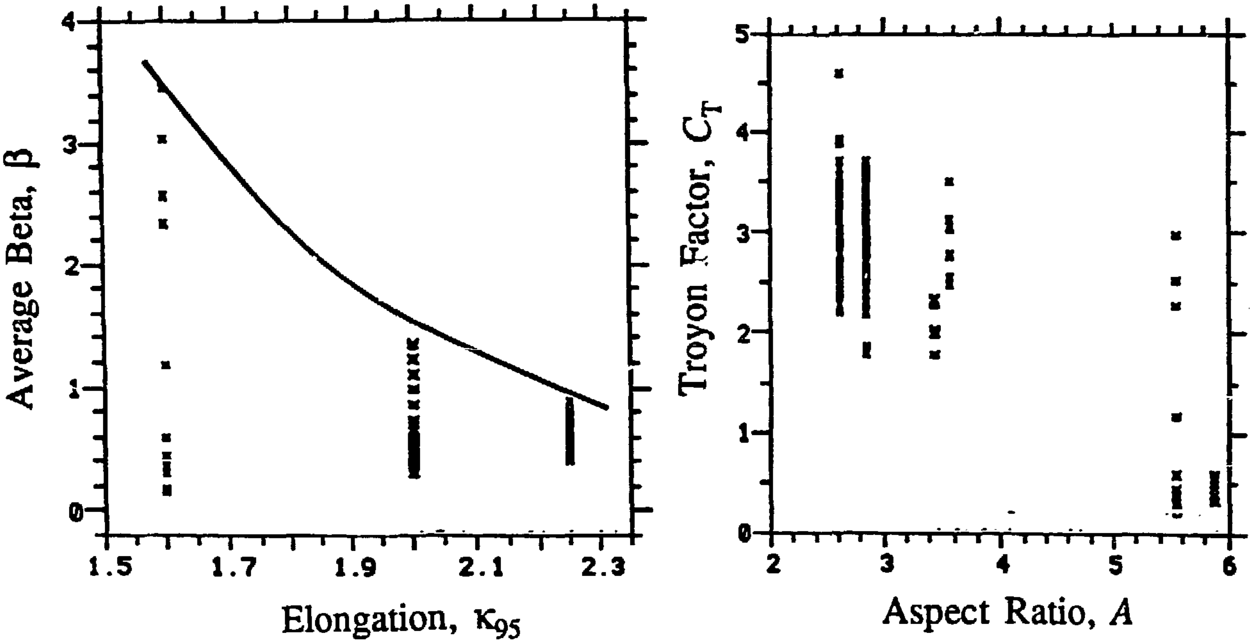
$$q_0 = 1.05 - 2.0, \text{ and}$$

$$q_{95} \leq 5.$$

Only profile functions that have been successful in analyzing β limit for JET and ITER are used.

DEPENDENCIES OF β LIMIT ON ELONGATION AND ASPECT RATIO

(Equilibria with Stable β)



Approximate scaling for the Troyon factor C_T (= $\beta a B_t/I_p$, in %mT/MA):

$$C_T \approx 2.8[1 - 0.4(\kappa_{95} - 1)^2]/(1 - \varepsilon)^{1.5},$$

indicating C_T = 3.5 for the ARIES-I configuration. We choose a more conservative value of 3.2 for the design.

DISCUSSION

- MHD equilibrium and stability analysis (including the vertical stability) results are on a relatively sound basis.
- The plasma profiles used are consistent with those producible by ICW or NB current drive.
- Beta limit assumed is somewhat lower than the optimized maximum.
- Poloidal field coils are arranged for minimum stored energy.
- They provide adequate flexibility to maintain plasma position and shape, and to provide some induction flux.

Concerns:

- Stored energy at low β significantly higher than at high β.
- Tradeoff between higher β limit and current drive requirements unclear.