Physics Basis for the ARIES-ST Power Plant

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Abstract — ARIES-ST, a fusion power plant design based on the spherical tokamak concept, has many attractive features, including high beta and power density, low magnetic field, and high self-driven current. The physics basis for choosing the design plasma parameters are described with respect to MHD equilibrium and stability, current drive, and startup. Implications for the fusion power core design are highlighted, and key directions for ST physics research are identified.

I. INTRODUCTION

The record plasma beta recently achieved on the START experiment [1] has advanced the concept of the spherical tokamak (ST) [2] as a potential power plant and volumetric neutron source. At the beginning of 1999, the ARIES Group has completed the design of ARIES-ST [3], a commercial fusion power plant based on this concept. This device has attractive features such as very high plasma β (~50%), low magnetic field (~2 T on axis) using normal conducting TF coils, high self-driven current fraction (~96%), and a highly elongated fusion power core that allows for a rapid vertical maintenance scheme.

For the ST-based fusion power plant to be economical, the plasma parameters must be optimized in order to offset the large resistive losses on the normal conducting TF coils. The plasma performance is governed by the simple relation:

$$f_{bs}\cdot\beta(\%) = A^{-1/2}\cdot C_{bs}\cdot (1+\kappa^2)/2\cdot(\beta_N/2)^2$$

where A is the aspect ratio, f_{bs} is the bootstrap current fraction, C_{bs}~0.6 is a profile dependent constant, κ is the plasma elongation and $\beta_N=\beta/(I_p/aB_o)$ is the normalized β. To minimize the costly CD power, it is desirable to focus on a class of equilibria with broad pressure profiles and high enough β values for which f_{BS}~1 and good profile alignment can be obtained. It is then clear that raising β_N and κ is pivotal in increasing β, and thus the fusion power density. To achieve both high β and near-unity f_{bs} with good profile alignment requires detailed control of the current and density/pressure profiles. Maximizing the center TF coil cross-section area is crucial in minimizing the resistive losses, thus eliminating room for an OH transformer in the center stack. Therefore, noninductive plasma startup is a central issue for the ST power plant operation

In this paper the physics basis for the ST concept as an attractive power plant is described. The results from our studies are used in determining the final design parameters for ARIES-ST [3]. The paper is divided into three parts, each dealing with an important aspect of the performance and operations of ST plasmas. Section II presents the results of detailed MHD studies of the plasma performance as a function of shape factors, aspect ratio and location of the stabilizing plates. The requirements for current drive (CD) and rotation generation, and the search for appropriate techniques are given in Sec. III. This is followed by Sec. IV on two proposed schemes for plasma and current startup without the use of an OH transformer. In Sec. V, we summarize our findings, and point out future directions for ST physics research.

II. EQUILIBRIUM AND STABILITY

II-A. Ballooning and Kink Stability

To maximize the plasma β for ARIES-ST, we carried out an extensive analysis of the MHD stability limit as a function of A, and plasma shape factors: κ, triangularity δ, and squareness ζ. We first studied the ballooning stability limits of fixed-boundary equilibria with f_{BS}=0.99, A=1.25-2.0 and with κ=1.6-2.8 at δ=0.45. The results [4][5] are summarized in Fig. 1 which indicates that increasing κ results in higher β_N and rapidly increasing β, that is peaked at A~1.4 and drops off gradually beyond. The square data point denotes the result for an optimized pressure profile and has a higher β_N (β) than an unoptimized curve at κ=2.8. At κ>2.8, this trend in β continues albeit at a slower rate of increase with κ.

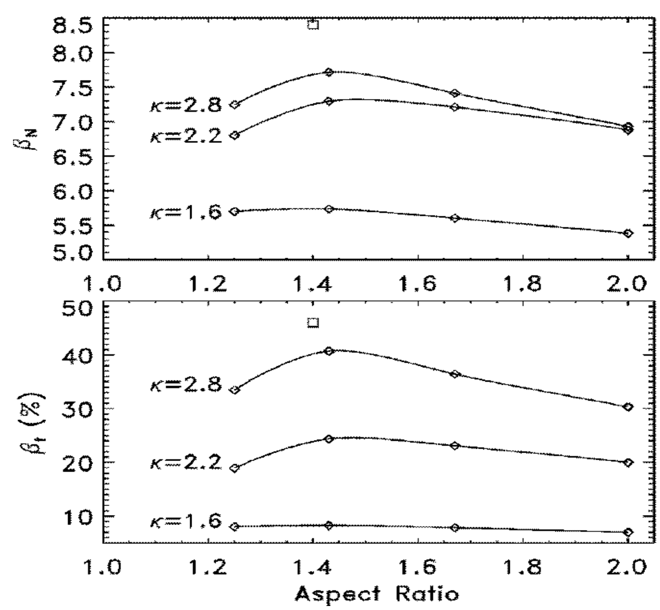

Fig. 1. Ballooning stability limits for 99% bootstrapped ST equilibria for various A and κ, and at fixed δ. [4] The square point is for optimized pressure profile at κ=2.8

We have also studied the kink stability of these ballooning stable equilibria in the presence of a close fitting conducting wall as a function of κ and δ at A=1.6. The more unstable the kink mode is, the closer the wall has to be to the plasma to stabilize it. The results are summarized in Fig. 2, where the wall separation requirements are determined by the n=6 kink mode for the equilibria studied. Decreasing κ and increasing δ (not shown) are both stabilizing for all low n modes. For very high κ (>3.6), the n=1 external kink mode limits the maximum stable elongation. To provide a comfortable margin for ARIES-ST, we choose κ=3.4 and raise δ to 0.64 at a nominal wall location of r_w/a=1.15, which should be well represented by the ferritic steel first wall.

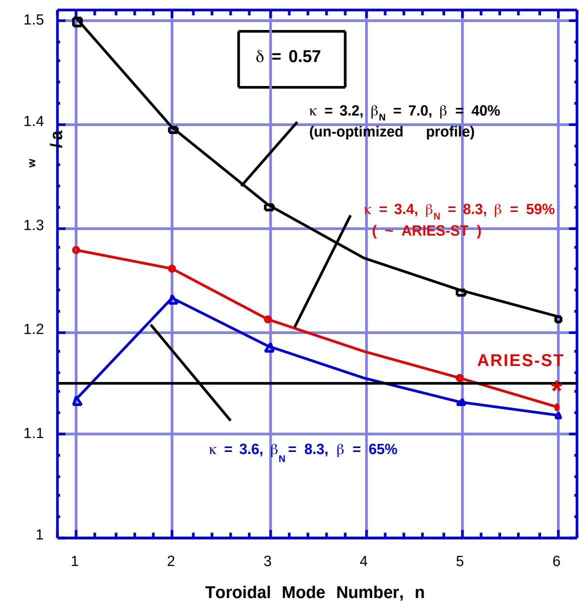

Fig. 2. Stabilizing wall location is determined by n=6 kink mode stability for κ< 3.6 at δ=0. 57 for A=1.6 ST equilibria

During this study, we found that the "squareness" of the plasma boundary shape, characterized by a parameter ζ, can be used to optimize the ballooning stable β. The parameter ζ can be applied separately to the inboard and outboard plasma boundary, given by: $X(\theta) = R_o + a\cos(\theta + \delta\sin\theta)$ and $Z(\theta) = \kappa a\sin(\theta + \zeta\sin 2\theta)$. As indicated in Table I for A=1.6 equilibria, the marginal β_N (β) improves from 7.2(40%) to 8.3(60%) as ζ is varied from –0.15 (unconstrained) to 0.1.

Table I.
Stability as a Function of Outboard "Squareness"

ζ	κ	δ	β_N	β(%)
-0.15	3.4	0.65	7.2	40
0.00	3.4	0.65	8.2	56
0.10	3.4	0.65	8.3	60

II-B. Poloidal Field Design and Vertical Stability

We have made improvements in a free-boundary equilibrium code for the ARIES-ST project to determine (i) the PF coil locations and currents, (ii) the poloidal flux topology for the divertor design, and (iii) the feasible plasma parameters and plasma shape combinations. These modifications are necessary in order to obtain numerical convergence for the class of ST equilibria examined, with high plasma pressure and very low internal inductance l_i (hollow current profile).

Free-boundary equilibrium calculations show that there are three distinct poloidal flux geometries that can exist outside the plasma, each of which has different heat flux deposition patterns. The pure limiter configuration (or natural divertor) has the X-point located at R=0, the plasma limited, and the outboard flux in the SOL splitting into two part, one intersecting the inboard wall and the other expanding vertically upward. The hybrid limiter configuration has the X-point located at R>0, the plasma still limited, and the outboard flux again splitting into two parts. The diverted configuration has the X-point at R>0, the plasma connected to the X-point, and the outboard flux in the SOL pinched and strictly extended outward. The double-null divertor configuration was chosen as the reference for ARIES-ST in order to avoid the high heat flux on the inboard wall.

The PF coil locations are determined along a specified contour, and are consistent with ports, the TF coils, neutral beamlines, and device maintenance. However, with the PF coils located outside the TF shell, we found that the amount of PF energy required for a ζ=0.1 boundary shape in Table I to be about ten times that for the unconstrained case (ζ=-0.15), which is costly. To lower the PF energy, we compromise by reducing ζ from 0.1 to 0.0 and mounting the PF coils on the inside of the TF shell. The final PF coil solution has a total of 14 coils and none on the inboard side.

The vertical stability analysis was difficult due to the extreme plasma elongation combined with very high β and very low l_i. The instability growth rate could only be inferred from a number of analyses, while the vertical position feedback control power requirement could be estimated but not simulated because of numerical difficulties for these plasmas. The fact that these instabilities are very non-rigid further complicates the analysis. We found that close fitting passive stabilizers, made of tungsten and located on the inboard and outboard top and bottom of the plasma, together with the steel wall, are quite sufficient for this purpose. Based on a random vertical displacement of 1.0 cm and a growth time of 70 ms, the required peak power for vertical control is estimated to be 105 MVA. The feedback coils would be located just behind the shield at 45° from the OB midplane.

II-C. Reference ARIES-ST Equilibrium

The aspect ratio of the ARIES-ST reference equilibrium is obtained by optimizing the value of the engineering Q, which is a balance between maximum fusion power output and minimum TF coil dissipation power. Detailed analysis with the systems code [3] yields an optimum aspect ratio of 1.6. The reference equilibrium is then given in Fig. 3, with A=1.6, κ=3.4, δ=0.64, ζ=0.0, β=56%, β_N=8.2, f_{BS}=0.99, R=3.2 m, B_o=2.1 T, I_p=28.4 MA, and $p_o/\langle p\rangle$=1.4, a very broad pressure profile.

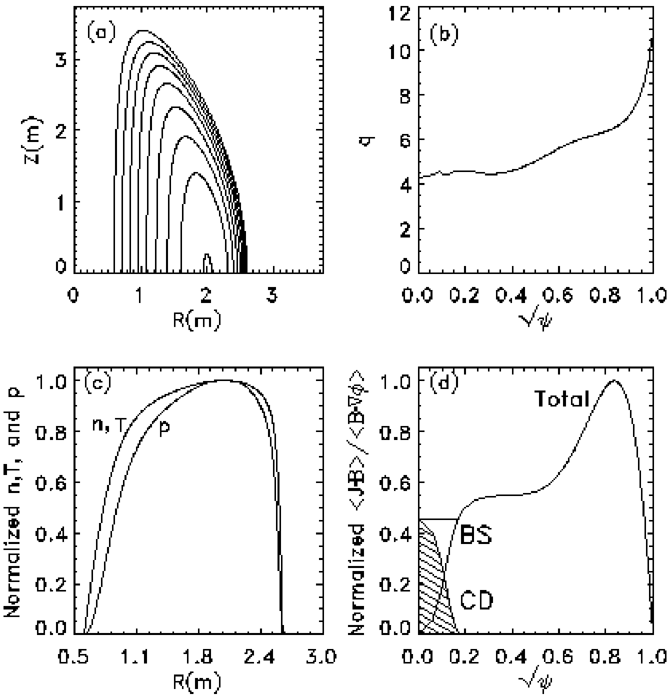

Fig. 3. Reference equilibrium for ARIES-ST.

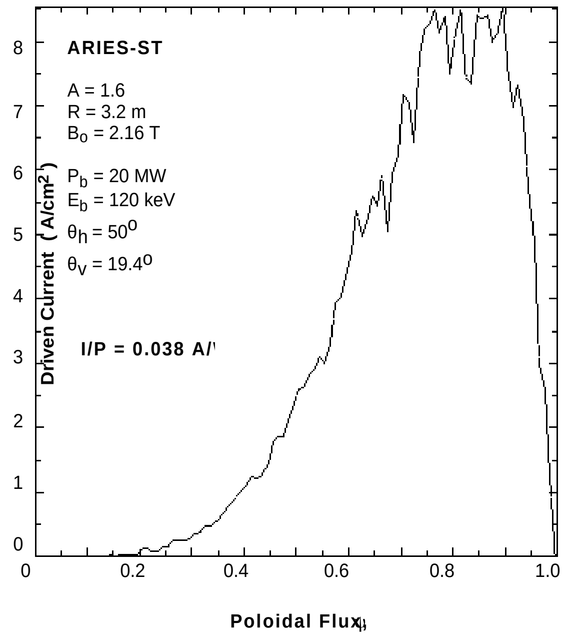

Fig.4. Off-axis neutral beam CD in ASRIES-ST.

III. CURRENT DRIVE AND PROFILE CONTROL

As indicated in Fig. 3, the reference equilibrium has both high β and pressure-driven current fraction of ~99% with the appropriate plasma profiles. In addition, this self-driven current profile is well aligned to the equilibrium profile, requiring only a tiny amount of seed current on-axis to be driven externally. Because of the high plasma dielectric constant, $(f_{pe}/f_{ce})^2>>1$, and/or the non-monotonic R-dependence of the magnetic field, most RF waves will not penetrate to the plasma axis, except low frequency fast waves with $f<f_{ci}$ throughout the plasma [8]. However, the problem for this CD technique is the large size of the antenna structure, which is estimated to be ~14 m in toroidal width in order to launch the desired spectrum. Neutral beams will require several MeVs of beam energy which are costly. Thus we invoke the so-called self-driven current due to potato particle orbits near the magnetic axis [7] to drive the on-axis seed current. An estimate indicates the local bootstrap fraction on axis is about 20% of that in the mid-plasma region, which should be adequate to sustain the equilibrium.

During the design study, we found that a small amount of CD in the outer plasma region can further enhance the ballooning β limit. Therefore, instead of relying entirely on the bootstrap effect to drive the off-axis current, we adopt the prudent approach of setting aside adequate external power to drive 5% or less of the total plasma current in order to maintain some current profile control and provide a wider stability margin. At the same time, we seek a CD technique that can also drive a toroidal plasma rotation efficiently for the purpose of stabilizing the kink mode. The latter requirement would exclude most of the RF current drive techniques that have been considered [8].

The technique of coaxial helicity injection (CHI) [9] has also been considered. One merit of this technique is the ohmic-like CD efficiency which is orders of magnitude better than that of the non-inductive methods. The major drawback is the lack of understanding of the relaxation processes that determine the current penetration and its profile, to allow us to extrapolate to the reactor regime. Further consideration of CHI should await increased theoretical understanding and experimental data, for example, in the National Spherical Tokamak Experiment at Princeton University.

For off-axis drive, we therefore propose to use tangential injection of neutral beams at moderate energies to drive current at ψ=0.8, where the bootstrap current is peaked, as shown in Fig. 4. Here, the beam energy is 120 keV which can be generated using conventional positive-ion-based sources, and the normalized bootstrap-aided CD efficiency is reasonable at 5.4×10^{20} A/W/m^2. In this specific example, 32 MW of NBI power is used to drive 5% of I_p. Using a rigid-body approximation and simple toroidal momentum balance, we estimated the driven rotation speed to be ~80 km/s which is ~4% of the Alfven speed.

IV. PLASMA STARTUP

Space limitation in the center column of the spherical tokamak makes it necessary to consider a plasma startup sequence without the OH transformer in ARIES-ST. In our design, we use a combined bootstrap and neutral beam current overdrive to ramp up the current to its full value. However, to implement this scheme, an initial target plasma with ~0.3 MA of current must first be formed. This can be accomplished by a combination of the flux swing available from the divertor coils located near the centerpost above and below the vacuum vessel, and a modest level of ECH power for preionization and low voltage breakdown. Extrapolating from DIII-D data, with the initial current channel in the outer part of the vessel, we found that at a loop voltage of 5.0 V, a current ramp rate of 1.0 MA/s would be achieved on ARIES-ST. To ramp to a current of 0.3 MA would take 0.3 s and a flux swing of 1.5 Vs from the PF coil system. However, a detailed self-consistent analysis has not been carried out.

Two plausible plasma startup sequences have been examined. Both scenarios start with a low initial current of ~0.3 MA, and maintain axisymmetric stability throughout the ramp-up. A modestly elongated small plasma is initially formed at the OB limiter which increases in size at constant κ and δ until the plasma is at full width at the 10.8 MA point. Subsequently κ and δ are increased to the final shape. The first sequence [A] assumes a standard H-mode confinement with H_{97H}=1.05 and will require 120 MW of NB power. The second sequence [B] considered requires confinement control with H_{97H}=1.55 at the 10.8 MA point, but only 50 MW of NB power. We choose B to be our base startup scenario.

A 0-D analysis has been carried out for the two startup sequences. A high β_p/A equilibrium is maintained in order to provide a high bootstrap fraction which, together with NBCD, provides the overdrive that ramps up the current. The evolution of power input from both NB and α-particles are plotted as a function of I_p in Fig. 5. Simultaneously the bootstrap and NB-driven currents are in Fig. 6. We note that the NB power is always set at 10% over the level required for steady state current, so that the startup sequence is very slow, taking 3-5 hours to complete. β_N and β_p are always maintained below the stability limits throughout the ramp-up, with β_N approaching limiting values only at the initial and final points. Also, NBCD is significant only at low β and small current plasma phase, and never exceeds 50% of the total current during ramp-up.

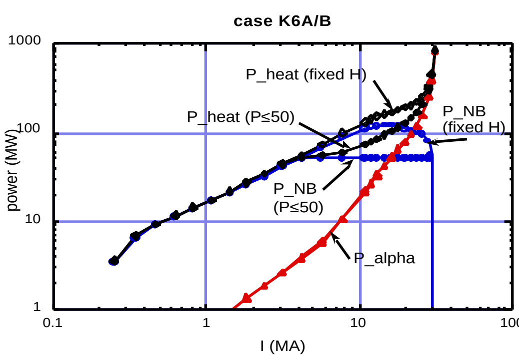

Fig. 5. Power input during startup on ARIES-ST.

V. SUMMARY AND DISCUSSION

The physics basis for the ARIES-ST power plant design has been presented in the critical areas of MHD stability, current drive and plasma startup. We learned from this design study that for the ST power plant to be economical, the plasma

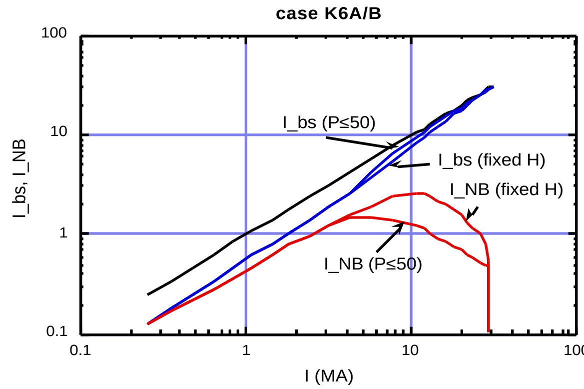

Fig. 6. Current ramp during startup on ARIES-ST.

must be nearly fully bootstrapped, unless steady state CHI is demonstrated and scaled favorably to the reactor regime. Strong shaping, wall stabilization of the kink mode with toroidal rotation, and a broad pressure profile are crucial for maximizing the plasma performance. Several outstanding issues have been identified. First, wall-stabilization of the kink mode over long time scales must be demonstrated experimentally. Second, the issue of whether or not α-heated plasmas will have broad pressure profiles like those desired for ARIES-ST must be addressed. Thirdly, we should demonstrate the feasibility of sustaining a steady-state plasma without on-axis current drive.

VI. ACKNOWLEDGMENTS

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