DESIGN-POINT FINALIZATION FOR THE ARIES-ST POWER PLANT†

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ABSTRACT

Plasma physics and technology features of the low-aspect-ratio spherical tokamak (ST) combine to determine the characteristic performance of a projected DT-fueled, 1,000-MWe-class, commercial fusion power plant, designated ARIES-ST. Key atttributes of the final design point are summarized.

I. INTRODUCTION

The multi-institutional ARIES Team has completed a study of a magnetic-fusion 1,000-MWe-class central-station electric power plant based on the spherical tokamak (ST) approach, denoted ARIES-ST. Previous interim results [1,2] have evolved to form a final conceptual design point, summarized here.

The ARIES Systems Code (ASC) is the primary tool used here to integrate the physics, engineering, and cost considerations that characterize the ARIES-ST design window. The ASC is supported by more detailed physics and engineering analyses performed by the ARIES Team. The projected Cost of Electricity [COE (mill/kWeh)] is used as a figure of merit to identify optimal candidate design points and mediate tradeoffs among variables. With one major exception (see Section V. below), the costing basis of previous ARIES studies is retained.

II. PHYSICS BASIS

The physics basis of the ARIES-ST derives from extensive investigations of the MHD plasma equilibrium design space of this class of tokamaks, with a view toward maximizing the stable beta while maintaing high (near unity) bootstrap-current fractions, f_{BC} [3]. Representative results are summarized in Table 1 for a range of plasma aspect ratios, $A \equiv R_T/a_p$, where R_T is the major toroidal radius and a_p is the midplane plasma half-width. The highest beta values are accessible at the lower values of A, where the plasma vertical elongation is larger. A conservative standoff factor (~ 0.9) from the beta limit is introduced as a margin against plasma disruptions, lowering both the toroidal and poloidal betas, and reducing f_{BC} to about 95%. Passive conductors located behind the blanket act as a stabilizing 'wall' to provide vertical stability, without which the plasma vertical elongation and resultant beta and bootstrap current fraction are too low.

With the toroidal plasma current near 30 MA, it is important to have a a high bootstrap-current fraction, f_{BC}, to reduce the amount of external current-drive power($\simeq 28$ MW), provided by energetic neutral beams in the baseline ARIES-ST design. Options using rf sources have also been considered.

The ARIES-ST operates in steady state with density-weighted, volume-averaged plasma temperatures, $T_i \simeq T_e \simeq 16$ keV and $Z_{eff} \simeq 2$. Xenon impurity is added to increase the core radiation fraction ($f_{RAD} \simeq 0.30$) to help distribute the surface heat fluxes more evenly over the entire first wall. For purposes of ARIES-ST projections, a confinement-time enhancement factor, H, is monitored relative to several empirical tokamak transport scalings, the application of which to the ST regime is as yet uncertain.

Impurity control is provided by a double-null divertor configuration. The plasma triangularity, δ, reduces the available space at low aspect ratio for inboard divertor slots. A thin inboard scrape-off layer (SOL) allows the inboard plasma to 'lean' on the inboard first-wall.

III. ENGINEERING BASIS

The basic ST Fusion-Power-Core (FPC) configuration was established in recognition of the attractive features and corresponding limitations of the ST concept. The baseline configuration includes resistive toroidal-field coils (TFCs), allowed by access to high plasma beta and allowing a corresponding reduction of shield thickness, particularly on the inboard side. The resistive TFCs incorporate sliding joints to allow separate removal of the centerpost and facilitate maintenance. High plasma triangularity, δ, inhibits the radial flaring of the TFC centerpost, which can lower the current density and reduce Joule losses.

The water-cooled, Copper-alloy, single-turn toroidal-field-coil (TFC) centerpost is protected by a 20-cm-thick high-temperature shield that does not breed tritium. The tapered centerpost does not have up/down symmetry and is flared at one end only; flaring at both ends would result in trapping of the centerpost by the inboard shield. The benefit of flaring at the upper end is realized by adding

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Table 1. Physics Basis of the ARIES-ST[‡].

Plasma aspect ratio	$\mathbf{1.4}^{(a)}$	$\mathbf{1.6}^{(a)}$	$\mathbf{1.6}^{(b)}$	$\mathbf{1.6}^{(c)}$	$\mathbf{1.8}^{(a)}$
Plasma vertical elongation, κ	3.60	3.40	3.40	3.10	3.20
Plasma triangularity, δ	0.644	0.644	0.644	0.720	0.644
On-axis safety factor, q_o	4.51	4.48	4.33	1.76	4.53
Edge safety factor, q	18.19	10.65	10.97	22.21	7.84
Circularized safety factor, q_*	2.97	2.73	2.87	3.14	2.63
β_N (%)	8.830	8.350	8.200	5.600	7.830
Toroidal beta, β	0.741	0.601	0.560	0.296	0.465
Toroidal beta, $\beta^{(d)}$	0.667	0.541	0.504	0.266	0.419
Poloidal beta, $\beta_\theta{}^{(d)}$	1.653	1.694	1.697	1.266	1.668
Stability parameter, $\epsilon\beta_\theta$	1.181	1.059	1.061	0.791	0.927
Bootstrap-current fraction, f_{BC}	$\sim$1	$\sim$1	$\sim$1	0.9	$\sim$1
I_{TF}/I_p	0.835	1.111	1.172	1.515	1.516

[‡] *cf.* Refs. [3,4].

[a] High-beta series, including finite plasma squareness, ζ.

[b] Final beta case, with $\zeta = 0$, and a practical (inboard) PFC set.

[d] Including disruption-avoidance margin (0.9).

conductor material to the outboard TFC shell. Joule dissipation in the TFC system provides the major contribution to the plant recirculating power; specifically, Cu centerpost (222 MW), Al outboard shell (47 MW), Cu leads (27 MW), and power supplies (33 MW). The overall plant recirculating power fraction is an uncomfortable 34%.

Adequate (outboard-only) blanket performance in terms of tritium breeding is achieved, except at the higher aspect ratios. High-power-density operation suggests the use of a LiPb breeder/coolant, enriched in ^{6}Li to 60%. A low-activation ferritic steel structural materials are used to meet Class-C limits for radioactive waste disposal and to improve the safety performance under in the event of a Loss of Coolant Accident (LOCA), for example. The massive TFC leads are used as a heat sink in the event of an accident. The usual engineering considerations of providing an efficient thermal power cycle (gross efficiency $\sim$ 45%) in a dual (PbLi and He) coolant system, while handling first-wall ($\sim$ 1 MW/m^2) and divertor-plate ($\sim$ 5 MW/m^2) surface heat fluxes, leads to 14-MeV neutron wall loads near 4.1 MW/m^2 (average) and 5.7 MW/m^2 (peak at outboard midplane).

Plasma shaping and the field cancellation for the poloidal-field divertor (double null) is provided by superconducting poloidal field coils (PFCs) located inside the TFC outboard shell. A toroidally continuous shell, rather than discreet TFC legs, also serves as a vacuum vessel. The plasma triangularity reduces the available space at low aspect ratio for inboard divertor slots. A larger plasma elongation, κ, implies a taller centerpost with increased Joule dissipation.

The service life of the nuclear components of the ARIES-ST is set by the fluence life (18 MW a/m^2, corresponding to 200 dpa) divided by the peak 14-MeV wall load. The service life of the TFC centerpost is limited by radiation damage and resistivity growth, which is reduced by a $\sim$20-cm-thick inboard shield, as shown in Fig. 3 of Ref. [2]. The inboard shield captures useful thermal power at η_{TH}, but competes with the TFC-centerpost for valuable space in the inboard radial build.

V. RESULTS

The ASC is used to examine the ARIES-ST design window by varying the FPC physical size and magnetic-field requirements self-consistently in order to satisfy power balance requirements at a target (net) output power. Nominal blanket/shield thicknesses are incorporated into the radial build of the FPC. For small values of major radius, less space is available to accomodate the radius of the centerpost; rising current density results in larger Joule dissipation and higher recirculating power.

The FPC of the near-optimum $A = 1.6$ ARIES-ST FPC is illustrated schematically in Fig. 1. Not shown are the large TFC power supplies, the biological shield, or a fueling system (*e.g.*, pellet injector). Maintenance access is from below, as described in Refs.[5,6], including provision for replacement of the tapered centerpost.

The selection of the ARIES-ST design point is summarized in the results of Fig. 2. Using conventional ARIES costing assumptions and 'rules', a minimum-COE point is identified; smaller FPCs have higher Joule dissipation in the TFC and larger FPCs have lower values of

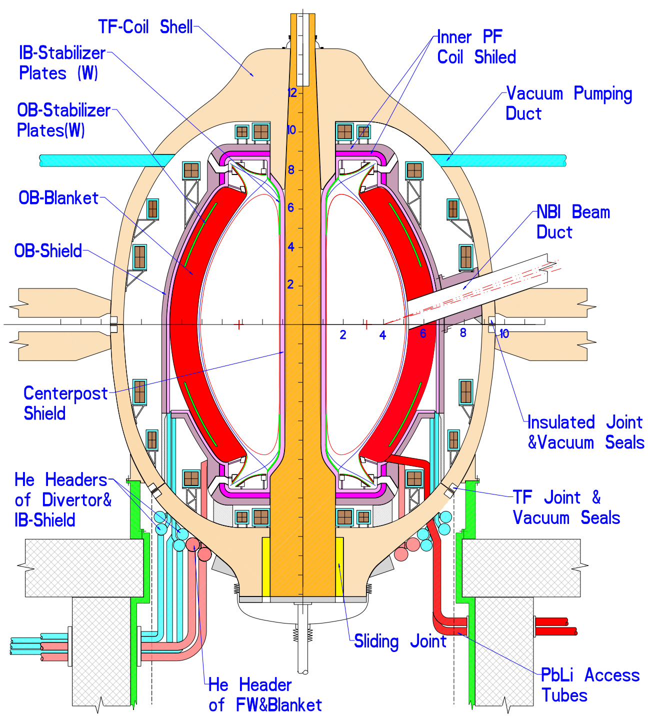

Fig. 1. Elevation view of the ARIES-ST Fusion Power Core (FPC) at A=1.6, for a 1,000 MWe (net) power plant.

mass power density (kWe/tonne), both tending to push the COE up. Application of cost savings resulting from application of advanced manufacturing [7] to the TFC *only*, results in a significantly lower cost curve and a shift of the minimum-COE point to larger major radius. The ARIES-ST plasma size for a target net electrical power output of 1,000 MWe was chosen such that $R_T = 3.2$ m, which choice (square) is slightly off-optimum under the conventional costing assumptions, but is closer to the optimal size under the low-cost-TFC assumption.

Parameters for the ARIES-ST conceptual design point are summarized in Table 2. Other details, particularly the direct cost breakdown, are too voluminous to be included in the present paper, but will be published elsewhere. While the COE can be lowered if the target net power is increased above 1,000 MWe, a larger FPC results and the advantages of vertical maintenance may be lost. The neutron wall load varies with poloidal location, the peak being located at the outboard equatorial area ($z \simeq 0$).

The costruction lead time is assumed to be six years, there being insufficient design detail in the ARIES series to use this parameter as a discriminator. A nominal plant factor, $p_f = 0.76$, is assumed in the calculation of the COE. A detailed allocation between forced outages and scheduled downtime was not made. Level of Safety Assurance (LSA) cost credits reward passive safety features

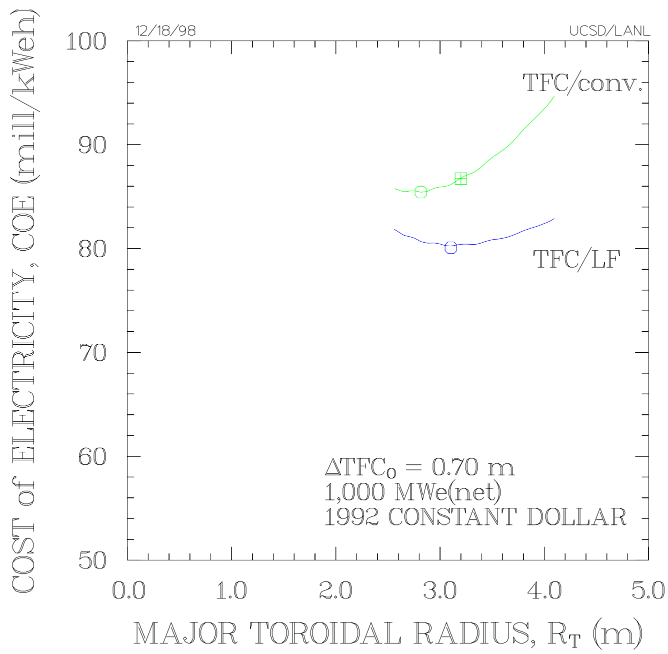

Fig. 2. Projected Cost of Electricity as a function of major toroidal radius, R_T, and $A = 1.6$ for the indicated fixed ARIES-ST parameters, comparing conventional ARIES costing results to new results incorporating advanced manufacturing *e.g.*, laser-forming (LF) of the TFC Cu centerpost and spray-casting of the TFC Al outboard shell. Minimum-COE points are denoted by circles.

or low source terms with lower direct costs, owing to the removal of some active emergency systems. A non-nuclear system would earn $LSA = 1$.

VI. SUMMARY

A systematic survey of the ST design space has identified candidate design points exhibiting various attributes and limitations of the concept as a commercial powerplant and provides the basis for detailed engineering analysis and integration by the ARIES Team. Good physics attributes, including high beta and high bootstrap-current fractions, are offset somewhat by a marginally acceptable recirculating power, owing to the substantial resistive dissipation in the toroidal-field system. An 'optimal' aspect ratio near $A = 1.6$ has been identified for the ARIES-ST, taking many features and constraints into consideration. Substantial potential direct-cost savings have been identified from novel low-cost manufacturing techniques to the toroidal-field-coil system (*i.e.*, Cu centerpost and Al outboard shell), which would have to be tested and scaled to the proposed application.

Table 2. 1,000-MWe(net) ARIES-ST Parameters.

Plasma aspect ratio, $A = R_T/a_p$	1.60
Major toroidal radius, R_T (m)	3.20
Plasma minor radius, a_p (m)	2.00
Plasma vertical elongation, κ_{95}	3.40
Plasma vertical elongation, κ_x	3.70
Plasma triangularity, δ_{95}	0.64
Plasma triangularity, δ_x	0.67
Circularized safety factor, q^*	2.88
Stability parameter, $\epsilon\beta_p$	1.06
Normalized beta, β_N (%)†	7.38
Toroidal beta, β (%)†	50.4
Poloidal beta, $\beta_p{}^\dagger$	1.70
Ion temperature, T_i (keV)	16.0
Electron temperature, T_e (keV)	16.5
Ion density, n_i ($10^{20}/\mathrm{m}^3$)	1.41
Electron density, n_e ($10^{20}/\mathrm{m}^3$)	1.58
Lawson parameter, $n_i\tau_E$ ($10^{20}\mathrm{s/m}^3$)	3.02
ITER-89P scaling multiplier, H_{89P}	2.83
ITER-97P scaling multiplier, H_{97P}	1.44
ITER-97H scaling multiplier, H_{97H}	1.17
ITER-98H scaling multiplier, H_{98H}	1.44
Plasma core radiation fraction, f_{RAD}	0.30
Plasma current, I_p (MA) [$f_{BC} \simeq 0.96$]	28.4
CD efficiency, γ_B (10^{20} A/W m^2)	5.22
CD power to plasma, P_{CD} (MW)	28
On-axis toroidal field, B_T (T)	2.1
Peak field at TF coil, B_{TF} (T)	7.4
TF-coil CP current density, j_{TF} (MA/m^2):	
-peak ($z \simeq 0$)	13
-average	8
TF-coil ohmic losses, P_{TF} (MW)	329
-Cu centerpost	222
-outboard Al shell	47
-leads	27
-TFC power supplies	33
Peak FW neutron load, $\hat{I}_w$ (MW/m^2)	5.6
Avg. FW neutron load, I_w (MW/m^2)	4.1
First-wall/blanket life, $I_w\tau$ (MWa/m^2)	18
Norm. divertor heat flux, P_{TR}/R_T (MW/m)	137
Blanket neutron energy multiplication, M_N	1.1
Thermal conversion efficiency, η_{TH}	0.45
Recirculating power fraction, $\epsilon(= 1/Q_E)$	0.34
Mass power density, MPD (kWe/tonne)	117
Thermal power, $P_{TH}(MWt)$	3,373
Gross electric power, $P_{ET}(MWe)$	1,518
Total direct cost, TDC (B$)	2.41
Total capital cost, TCC (B$)	4.65
Cost of electricity (mill/kWeh, 1992-$):	
total COE w/ safety credits (LSA=2)	80
total COE w/o safety credits (LSA=4)	92

† Includes disruption-avoidance factor (0.9).

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REFERENCES

[1] R. L. Miller for the ARIES Team, "Design-Point Determination for the Commercial Spherical Tokamak, ARIES-ST," Proc. 17th IEEE/NPSS Symposium on Fusion Engineering, 97CB36131, Vol. 2, 1031 (October 1997).

[2] R. L. Miller for the ARIES Team, "Overview of the Spherical Tokamak Power Plant Study: ARIES-ST," *Fusion Technology*, 34, 3, Part 2 (Nov. 1998) 364.

[3] J. E. Menard, S. C. Jardin, S. M. Kaye, C. E. Kessel and J. Manikam, "Ideal MHD Stability Limits of Low Aspect Ratio Tokamak Plasmas," *Nuclear Fusion*, 37, 5 (1997) 595.

[4] T. K. Mau, S. C. Jardin, C. E. Kessel, J. Menard, *et al.*, "Physics Basis for the ARIES-ST Power Plant," these Proceedings.

[5] X. R. Wang, M. S. Tillack, F. Najmabadi, S. Malang and the ARIES Team, "Configuration and Maintenance of the ARIES-ST Power Plant," these Proceedings.

[6] W. Reiersen, F. Dahlgren, H-M. Fan, C. Neumeyer, and I. Zatz, "The Toroidal Field Coil Design for ARIES-ST," these Proceedings.

[7] L. M. Waganer, "Innovative, Ultra-Low-Cost Fabrication Methods," these Proceedings.