

ARIES-ST design point selection

Ronald L. Miller*, The ARIES Team

Fusion Energy Research Program, University of California, San Diego 460 Eng. Building Unit II 9500, Gilman Drive Mail Code 0417, La Jolla, CA 92093-0417, USA

Abstract

The spherical tokamak (ST) regime of low-aspect-ratio (LAR) tokamak operation has provided an opportunity to consider a 1000-MWe(net) D–T power plant conceptual design, designated ARIES-ST. Physics attributes include high beta ($\sim 50\%$) together with high bootstrap current contribution ($\sim 95\%$). Neutral-beam current drive (~ 28 MW) supports steady-state operation. The plasma aspect ratio was selected to be $A \simeq 1.6$ leading to the choice of major plasma toroidal radius, $R_T = 3.2$ m and plasma half-width, $a = 2.0$ m. The usual engineering considerations of providing an efficient thermal power cycle (gross efficiency $\simeq 45\%$), while handling first-wall (~ 1 MW/m²) and divertor-plate (~ 5 MW/m²) surface heat fluxes, leads to 14-MeV neutron wall loads near 4.1 MW/m² (average) and 5.6 MW/m² (peak at outboard mid-plane) in a dual (PbLi and He) coolant system. The water-cooled toroidal-field-coil (TFC) centerpost is protected by a 20-cm (inboard) shield that does not breed tritium. 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%. For a plant capacity factor $\sim 76\%$, consistent with earlier ARIES studies, the COE is projected to be ~ 84 mill/kWe h (1992 constant \$), using similar unit-cost assumptions and some passive-safety cost credits. Inclusion of additional credits for advanced, low-unit-cost manufacturing based on plasma spray and related techniques (as applied to the TFC centerpost and outboard TFC shell only) lowers this COE estimate to ~ 79 mill/kWe h.

© 2002 Elsevier Science B.V. All rights reserved.

Keywords: Magnetic fusion; Spherical tokamak (ST); Spherical torus; Neutral beam current; Drive; Bootstrap current contribution; Power plant

1. Introduction

Interest in the spherical tokamak (ST) (or spherical torus) concept [1] is motivated by the potential for high-beta ($\beta \equiv$ ratio of plasma pres-

sure to confining magnetic-field pressure) operation with high self-generated bootstrap-current fractions, $f_{BC} \simeq 1$. The emerging physics basis [2], together with an engineering embodiment of the fusion power core (FPC) of the ST allows a conceptual projection of a 1000 MWe(net)-class, deuterium–tritium (D–T) fueled, central-station electric power plant.

As is the case with most conceptual fusion power-plant projections, a plasma physics model

* Corresponding author. Tel.: +1-858-534-7842; fax: +1-858-534-7716.

E-mail address: miller@fusion.ucsd.edu (R.L. Miller).

Table 1
Plasma radial profile inputs to the ASC

Aspect ratio, $A \equiv R_T/a_p$	1.4 ^a	1.6 ^{a,b,c}	1.8 ^a
Peak-to-average pressure, p_0/p ($= n_0 T_0/nT$)	1.373	1.385	1.404
Peak-to-average density, n_0/n	1.218	1.220	1.230
Separatrix-to-average density, n_s/n	0.05	0.05	0.05

Corresponding to cases summarized in Table 2.

^a High-beta series, including ‘squareness’ in plasma shaping.

^b Final beta case, without ‘squareness’.

^c No stabilizing wall.

is coordinated with an engineering embodiment of the FPC sufficient to establish a power flow model in the context of a central-station power cycle. A cost estimate is projected and used as a figure-of-merit in trade studies meant to steer the design in optimal directions and resolve certain configurational decisions and subsystem choices.

The present work builds upon earlier consideration of a low-aspect-ratio (LAR) candidate considered in the STARLITE study [3]. A previous, comprehensive study [4] of this concept assumed physics and current-drive (CD) characteristics prior to the availability of the emerging database.

The primary tool for the integration of physics, engineering, and costing aspects of magnetic-fusion power-plant is the ARIES Systems Code (ASC). Some features of the profile-averaged (0-dimensional), steady-state power-balance model used in the ASC are summarized in the next section as prelude to the engineering characterization of the ST FPC. A description of the ST power-plant design space leads to the selection of a reference design point, which was subjected to more detailed analysis by the ARIES Team. Results of some sensitivity studies are presented to highlight the key cost drivers.

2. Physics aspects

2.1. Plasma equilibrium

The ST plasma equilibrium has been examined separately [5] with a view toward maximizing the available beta (ratio of plasma pressure to confining magnetic-field pressure). This examination has

produced a series of discrete cases representing three values of plasma toroidal aspect ratio, A , in the range of interest. The β value has been found to scale as the inverse square root of A . Three values of A (1.4, 1.6, and 1.8) illustrate the physics performance of the ST.

The $A = 1.6$ case was identified as ‘near optimum’ with the other cases acting as bracketing outliers. The $A = 1.6$ case itself was considered at three levels: (1) high-beta (including ‘squareness’ in the plasma shaping, as with the other aspect-ratio cases), (2) a reference case (without squareness, consistent with modeled poloidal-field-coil (PFC) performance) at somewhat lower beta, and (3) a subcase with no stabilizing wall, forced to use lower plasma elongation, resulting in lower beta.

Relative to the conventional tokamak plasma at typically higher aspect ratio, the ST plasma accesses higher values of normalized beta, β_N , due to the favorable impact of increased magnetic shear in the edge plasma and the suppression of ballooning modes [5].

2.2. Plasma profiles

The following forms are assumed for the minor-radial density profiles for all species, the temperature profile for all thermal species, and the toroidal current-density profile:

$$n(x) = (n_o - n_s)(1 - x)^{\alpha_n} + n_s, \quad (1)$$

$$T(x) = T_o(1 - x)^{\alpha_T}, \quad (2)$$

$$j_\phi(x) = j_o(1 - x)^{\alpha_j}; \quad (3)$$

where $x \equiv (r/r_p)^2$ and the α s are fitting constants. A non-zero edge density, n_s , is included to lower the peak heat flux and plasma temperature at the

Table 2
Plasma equilibrium inputs to the ASC

$A \equiv R_T/a_p$	1.4 ^a	1.6 ^{a,b,c}			1.8 ^a
Plasma elongation, $\kappa_{\times}$	3.60	3.40	3.40	3.10	3.20
Plasma triangularity, $\delta_{\times}$	0.644	0.644	0.644	0.720	0.644
On-axis safety factor, q_0	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, β^d	0.667	0.541	0.504	0.266	0.419
Toroidal beta, β_θ^d	1.653	1.694	1.697	1.266	1.668
Stability parameter, $\varepsilon\beta_0$	1.181	1.059	1.061	0.791	0.927
Bootstrap-current fraction, f_{bc}	~ 1	~ 1	~ 1	0.9	~ 1
I_{TF}/I_P	0.835	1.111	1.172	1.515	1.516

^a High-beta series, including ‘squareness’ in plasma shaping.

^b Final beta case, without ‘squareness’.

^c No stabilizing wall.

^d Including disruption-avoidance margin (0.9).

divertor plate, but the presence of this density pedestal has been found to lower the efficiency of RF CD. The profile inputs to the ASC for ARIES-ST designs at representative values of plasma aspect ratio (PAR), $A \equiv R_T/a_p$, are summarized in Table 1 (R_T is the major toroidal plasma radius and a_p is the plasma half-width in the equatorial plane). These cases span a range of PAR $1.4 < A \equiv R_T/a_p < 1.8$ of interest to the present study. This range is somewhat higher than the the $A = 1.25$ case considered in the STARLITE study [3]. As will be seen below, the $A = 1.6$ case is the reference ARIES-ST choice. Difficulties in incorporating a PFC set that provides the higher-order ‘squareness’ contribution to plasma shaping were identified, so this variant was dropped. The case with no stabilizing wall was found to provide insufficient plasma beta and was also not emphasized.

2.3. Plasma beta

The plasma-surface-averaged poloidal magnetic field, $B_\theta \equiv \bar{B}_\theta(r_p)$, is produced by the plasma toroidal current, I_ϕ , and is given by:

$$B_\theta = \frac{\oint B_\theta d\ell}{\oint d\ell} = \frac{\mu_0 I_\phi}{2\pi a_p S}, \quad (4)$$

where $\mu_0 = 4\pi \times 10^{-7}$ H/m is the permeability of vacuum. The toroidal magnetic field, produced by external toroidal-field (TF) coils, provides the primary confining field for the plasma pressure, p , in a tokamak. The on-axis toroidal magnetic-field strength is $B_\phi(R_T) \equiv B_{\phi 0}$, the toroidal beta is $\beta \equiv 2\mu_0 p/B_{\phi 0}^2$, and the poloidal beta is $\beta_\theta \equiv 2\mu_0 p/B_{\phi 0}^2$. The plasma-edge safety factor, q , is approximated by:

$$q = \frac{2\pi}{\mu_0} C_q \frac{a_p B_{\phi 0}}{I_\phi} \frac{\varepsilon}{(1 - \varepsilon^2)^2} S^2, \quad (5)$$

where the coefficient $C_q \equiv C_q(\varepsilon\beta_\theta, q) \approx 1$. The plasma shape factor, S , represents the ratio of the length of the perimeter of the shaped (including elongation, triangularity, squareness) plasma to the perimeter of a circular plasma with the same value of a_p . The value of S is dominated by elongation, κ , such that $S \approx (1 + \kappa^2)/2$. The inputs to the ASC equilibrium model are derived from equilibrium and stability results, and are summarized in Table 2. From this set of physics results, the

intermediate $A = 1.6$ case was selected to provide representative performance. The reference (final) ARIES-ST case does not invoke ‘squareness’ in the plasma shaping, such that the value of beta is somewhat reduced. Wall stabilization of certain potential instabilities is invoked to raise the level of accessible beta values. Operation at some stand-off from the beta limit (here 90%) is invoked to reduce the probability of plasma disruptions, seen as an important operational concern for tokamaks, generally.

2.4. Power balance

The plasma particle- and power-balance module used in the ASC is based on the zero-dimensional (0-D), steady-state MakNTAU [6,7] code. The model used in this module is summarized in this subsection with a detailed description given in [6]. Profile effects are included in the model through radial averages over the profiles given by Eqs. (1)–(3).

The steady-state, zero-dimensional ion and electron power balance equations, written in terms of power density, are given by:

$$p_{CP_i} + p_{H_i} = p_{TR_i} + p_{ie}, \quad (6)$$

$$p_{CP_e} + p_{H_e} + p_{\Omega} + p_{ie} = p_{TR_e} + p_{BR} + p_{CY} + p_{LINE} \quad (7)$$

The charged-particle fusion-product power density, p_{CP} , is calculated by summing over D–³He, D–D, and D–T fusion reactions. The power density deposited in the ions and electrons, p_{CP_i} and p_{CP_e} , respectively, (assuming a specified prompt loss fraction) is calculated through a time integral of the slowing down rate of a test particle against the background ions over the time for the test particle to thermalize.

The heating power densities provided by sources external to the plasma are:

$$p_{H_i} = \frac{p_{CP}}{Q_p} f_{H_i}, \quad (8)$$

$$p_{H_e} = \frac{p_{CP}}{Q_p} (1 - f_{H_i}). \quad (9)$$

The fractional heating power going to the ions,

f_{H_i} , and the plasma gain, $Q_p \equiv P_{CP}/P_H$, are input from other plasma engineering modules. The term that partitions energy between the ions and electrons is given by p_{ie} . The transport energy losses are characterized by ion and electron energy confinement times:

$$p_{TR_i} = \frac{3}{2} k n_i T_i / \tau_{E_i}, \quad (10)$$

$$p_{TR_e} = \frac{3}{2} k n_e T_e / \tau_{E_e}. \quad (11)$$

where $k = 1.6021 \times 10^{-19}$ J/eV. The total transport energy confinement time is given by:

$$\tau_E = \frac{n_i T_i + n_e T_e}{(n_i T_i / \tau_{E_i}) + (n_e T_e / \tau_{E_e})}, \quad (12)$$

and a ratio of $\tau_{E_i}/\tau_{E_e} = 1$ is assumed. The plasma-energy confinement time expressed in terms of the net heating power, $P_{TR} \simeq P_{PH}(1 - f_{RAD})$, is:

$$\tau_E = \frac{W_p}{P_{PH}(1 - f_{RAD})}. \quad (13)$$

The ASC uses the particle- and power-balance calculations to establish the appropriate ion constituent fractions and the corresponding values of the Lawson parameter, $n_i \tau_{E_i}$, and τ_E . Once τ_E is known, a comparison can be made to any of a number of proposed empirical scaling laws [8,9]. It should be emphasized that a scaling law is not needed to derive any of the ARIES design points; τ_E is determined from the specification of P_E and Q_p that is set by the power requirements of the CD system. The value of τ_E required for a design point is compared with the various empirical scaling laws by means of a confinement multiplier, $H_j \equiv \tau_E / \tau_{E_j}^i$, where the superscript j denotes the particular scaling of interest. The standard scaling law used for comparison is the ITER-89P scaling [10] and is given by:

$$\tau_E^{89P} = \frac{3.8033 \times 10^{-6} I_{\phi}^{0.85} n_e^{0.1} B_{\phi 0}^{0.2} a^{0.3} R_T^{1.2} \kappa_{\times}^{0.5} A_i^{0.5}}{[P_{PH}(1 - f_{RAD})]^{0.5}}, \quad (14)$$

where A_i is the atomic mass (2.5 for a nominal 50:50 fuel mixture of either D–T or D–³He). Bremsstrahlung provides the dominant plasma

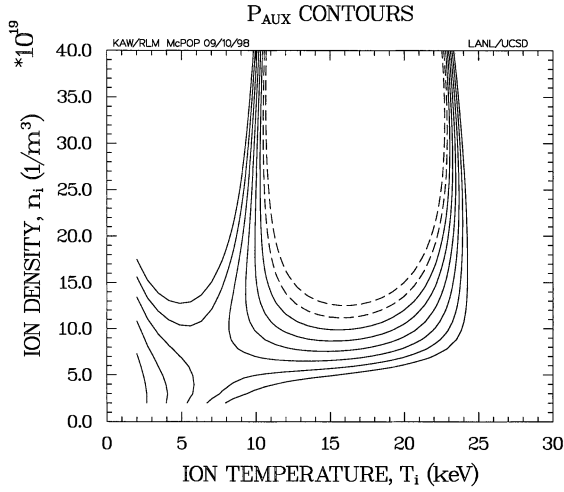


Fig. 1. ARIES-ST POPCON diagram. Solid contours indicate auxiliary heating power, P_{AUX} , in increments of 10 MW until 'ignition' ($P_{AUX} = 0$) at the dashed contours.

radiation loss channel, with cyclotron radiation being relatively small at the high betas and low magnetic-field strengths characteristic of the ARIES-ST.

The plasma-core radiation fraction is given by:

$$f_{RAD} \equiv \frac{P_{BR} + P_{CY}}{P_{CP} + P_H + P_{\Omega}}. \quad (15)$$

The core radiation fraction for the ARIES-ST plasma at $T \simeq 16$ keV with $Z_{eff} \simeq 2$, including the addition of Kr impurity, is $f_{RAD} \simeq 0.3$.

2.5. Current drive

A neutral beam (NB) system provides steady-state CD for the ARIES-ST with a beam energy, $E_B = 120$ keV. For purposes of the ASC, a scaling of NBCD efficiency, represented by γ_B , was developed [11] for the ranges: $1.6 < Z_{eff} < 2.2$ and $8 < T_{e0} < 24$ keV. The figure-of-merit, γ_B , improves as Z_{eff} increases and as central electron temperature, T_{e0} , increases. Efficient CD performance tends to favor higher plasma temperature, at the reduction of plasma density and fusion power density. The ARIES-ST benefits from a high bootstrap current fraction, $f_{bc} \simeq 0.95$.

2.6. Operating space

Plasma operating contour (POPCON) plots have been generated for the ARIES-ST using the several empirical scaling relations. The results assuming the ITER-89P relation are shown in Fig. 1. Thermally stable operation near $T_i \simeq 16$ keV is suggested by result; other scaling relations might shift this zone. A start-up trajectory from initially low values of density and temperature with an auxiliary heating input of ~ 50 MW is suggested by this operating space. The chord-averaged electron density relative to the Greenwald density limit is $\bar{n}_e/n_{GR} \simeq 0.75$ for the ARIES-ST.

3. Engineering aspects

A number of engineering considerations are incorporated into the ASC models of the ARIES-ST; these considerations interact with the plasma physics aspects to characterize the FPC and provide the basis for cost projections. Adequate tritium breeding is obtained from the out-board blanket provided that the Li in the LiPb eutectic is enriched to 60% ^6Li .

3.1. Geometric considerations

To begin with, several geometric invariants related to the physics cases of Table 2 (above) are summarized in Table 3. These results supersede those found in Table 2 of [14]. A tight PAR can be expected to restrict the inboard radial build with several adverse consequences. These effects include a small-radius centerpost with high(er) current density and Joule dissipation as well as elimination of an inboard tritium-breeding blanket in favor of a thin shield. High plasma elongation increases the length of the centerpost and increases Joule dissipation. High plasma triangularity restricts the possibility of increasing the radius (flaring) the centerpost away from the plasma mid-plane to reduce current density and Joule dissipation. The peak-to-average ratio of the 14-MeV neutron wall varies slowly among the several combinations of A and κ . The peak neutron wall load, occurring at

Table 3
ARIES-ST engineering invariants

$A \equiv R_T/a_p$	1.4 ^a	1.6 ^{a,b,c}			1.8 ^a
Plasma elongation, $\kappa_\times$	3.60	3.40	3.40	3.10	3.20
Plasma triangularity, $\delta_\times$	0.644	0.644	0.644	0.720	0.644
I_{TFC}/I_P	0.835	1.111	1.172	1.515	1.516
<i>Neutron wall load (NWL)</i>					
IB peak/total average NWL	0.871	0.887	0.891	0.993	0.901
OB peak/total average NWL, $\hat{I}_w/I_w$	1.361	1.370	1.380	1.514	1.379
IB normalized average NWL	0.584	0.611	0.613	0.600	0.639
IB average/IB peak NWL	0.670	0.689	0.688	0.643	0.709
OB normalized average NWL	1.083	1.100	1.101	1.108	1.111
OB average/OB peak NWL	0.795	0.803	0.797	0.731	0.807

Inboard (IB) and outboard (OB) peaks occur at respective equatorial mid-planes.

^a High-beta series, including ‘squareness’ in plasma shaping.

^b Final beta case, without ‘squareness’.

^c No stabilizing wall.

the outboard (OB) mid-plane, is used to gauge the in-service lifetime of the first-wall and blanket structures. The ARIES-ST first-wall/blanket structural fluence lifetime, $\tau_{\max}$, is taken to be 18 MWa/m², corresponding to ~ 200 dpa, a value lower than the 20 MWa/m² used for the STARLITE LAR case [3].

3.2. Toroidal and poloidal coils

Rather than a set of discreet TF coils (TFCs) typically seen in conventional tokamaks, the ARIES-ST uses a water-cooled monolithic center-post (CP) made from Cu alloy. The systems-level characterization of the CP is at the level of previous models [12,13]. The outboard return conductor is configured as a 0.70-m Al shell, rather than as discreet TFCs. The TFC components are operated as near-room-temperature resistive elements. The CP, at 110 °C, has a resistivity of 2.64E-08 Ω -m and the Al outboard shell, at 65 °C, has a resistivity of 2.64E-08 Ω -m. The conductor fractions of both components are 85%.

Massive Cu-alloy TFC current leads attach to the TFC outboard shell near the equatorial plane and provide a heat sink under accident conditions. In normal operation, the Joule dissipation in these leads is ~ 22 MW. The TFC leads are sized using the scaling summarized in Fig. 4. It is assumed that

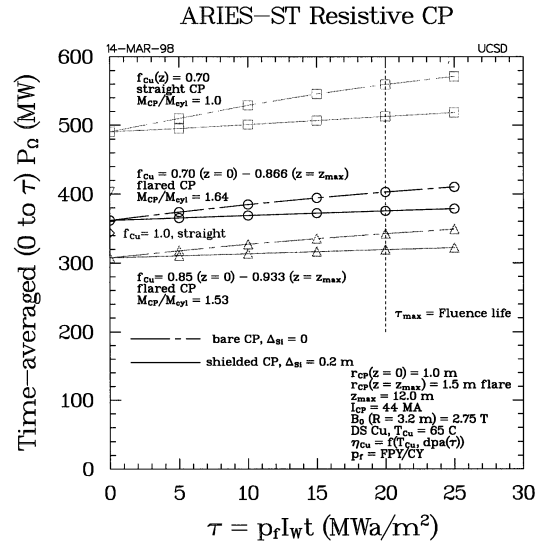


Fig. 2. Dependence of typical ST CP resistive power dissipation growth due to neutron-induced damage/transmutation (resistivity growth) and current profile redistribution as a function of exposure for representative cases (straight vs. flared and bare vs. shielded).

the TFC power supplies have internal dissipation consistent with a voltage drop equal to 1 V, resulting in an additional ~ 33 MW of dissipation as a component of recirculating power.

The low-temperature superconducting (LTS) PFCs are located inside the TFC shell in the low-

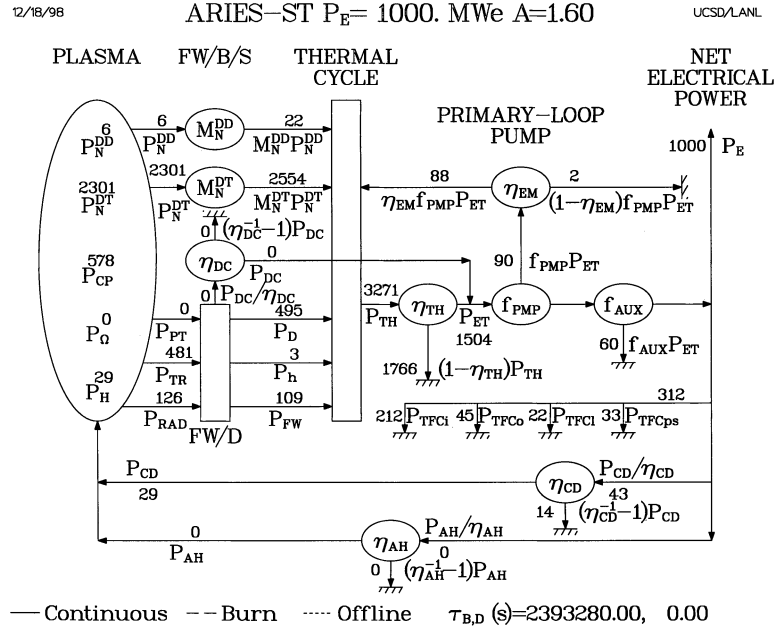


Fig. 3. Overall ARIES-ST power-flow diagram used in the ASC. Values of powers are given in MW. Minor differences between the entries on this figure and those of Tables 5 and 6 may exist.

field ARIES-ST. This positioning improves the coupling to the plasma for purposes of shaping. The ARIES-ST does not have an ohmic-heating coil set due to the lack of space at low aspect ratio.

3.3. Shielding considerations

The ARIES-ST uses a He-cooled steel inboard shield to capture useful thermal power, reduce the damage-induced resistivity growth of the Cu-alloy CP conductor, and aid in the meeting of shallow-burial (Class-C) waste-disposal criteria for the activated and expendable centerpost (CP). In a sense, a thicker CP conductor occupying the same annular space provides a measure of self-shielding of the CP core, but does not contribute to the power conversion cycle and adds to the radioactive waste stream. As shown in Fig. 2, a bare (unshielded) CP would typically experience a $\sim 14\%$ increase in resistive dissipation for an exposure of 20 MWa/m^2 compared with $\sim 7\%$ increase for a CP protected by a 0.30-m-thick inboard shield/manifold. Damage to the CP occurs in an outer

annular region with a thickness of a few tens of centimeters corresponding to a few neutron mean free paths; a radial and axial integration gives a volumetric average and a time average for the service life, τ , is used to represent the degraded performance terminated by CP replacement at ~ 3 full power years (FPY). The plant capacity factor, p_f , of course, characterizes the number of FPYs accumulated per calendar year (CY). Pending a more detailed availability and maintenance study, a nominal value, $p_f \simeq 0.76$, is assumed in the COE projections.

3.4. Power flow

The power flow, as modeled in the ASC, is illustrated in Fig. 3 for ARIES-ST. Given a stipulated target for the net electric-power output, P_E , the thermal-power output, P_{TH} , is determined for a nominal value of the thermal conversion efficiency, η_{TH} , such that $P_E = \eta_{TH}(1 - \varepsilon)P_{TH}$, where $\varepsilon = 1/Q_E$ is the recirculating power fraction and $Q_E \equiv P_{ET}/P_C$ is the engineering Q -value. The

gross electric-power output is $P_{ET} = \eta_{TH} P_{TH}$. A fraction $f_{AUX} = 0.04$ of P_{ET} ($P_{AUX} = f_{AUX} P_{ET}$) is allocated for auxiliary functions. A fraction f_{pump} of P_{ET} ($P_{pump} = f_{pump} P_{ET}$) is allocated for primary-loop pumping power.

The engineering gain, Q_E , can be written as:

$$Q_E = \frac{1}{\varepsilon} = \eta_{TH} \times \frac{M_N P_N + P_{CP} + P_H + P_\Omega + 0.90 P_{pump}}{P_{TFC} + P_{AUX} + P_{pump} + (P_{CD} + P_{AH})/\eta_{CD}} \quad (16)$$

The gross thermal conversion efficiency, η_{TH} , is 45% for the ARIES-ST. The neutron energy multiplication, M_N , is found from neutronics modeling to be ~ 1.11 and is used to determine the useful thermal power, P_{TH} , recovered from the blanket, divertors, and high-temperature shields. It is assumed that $0.98 P_{pump}$ is recoverable as useful thermal power in the primary coolant loop.

4. Systems assessment

The role of systems analysis for the ARIES-ST power plant is to identify high-leverage parameters affecting the conceptual design. Representative values for parameters of the FPC that minimize the projected cost-of-electricity (COE) are identified with reasonable self-consistency.

4.1. Systems models

The plasma equilibrium (e.g. magnetic topology, β , etc.), CD, and other physics models are based upon the analyses of the equilibrium provided by the Physics effort. Together with the approximate engineering models, the ARIES-ST FPC is characterized in sufficient detail to provide a conceptual design. A cost breakdown structure (CBS) provides the basis for a direct cost estimate, consistent with previous ARIES and other studies. Scaling estimates for building volumes and other balance of plant (BOP) items are used.

Table 4
Magnet-system unit costs (1992 \$)

Component	Mass density (kg/m ³)	Unit cost (\$ per kg)
<i>Low-temperature superconductors (LTS)^{a,b}</i>		
Bi	7500	103.0
Ternary Nb ₃ Sn	7500	97.8
Nb ₃ Sn	7500	92.7 ^c
NbTi	7500	82.8 ^c
<i>Resistive coils</i>		
Resistive Cu ^{d,e}	7300	65.4
ARIES-ST Cu CP ^f	7565	7.05
ARIES-ST outer TFC shell ^f	2295	5.52

^a Compare with 6462–7108 kg/m³ and 39–35 \$/kg for TFCs and 7900 kg/m³ and 62 \$/kg for PFCs, cf. STARFIRE [15].

^b Compare with 7900 kg/m³ for PFCs cf. STARFIRE [15].

^c [17–19].

^d cf. TITAN [20].

^e Compare with 7900 kg/m³ and 62 \$/kg, cf. GENEROMAK

[16].
^f [21].

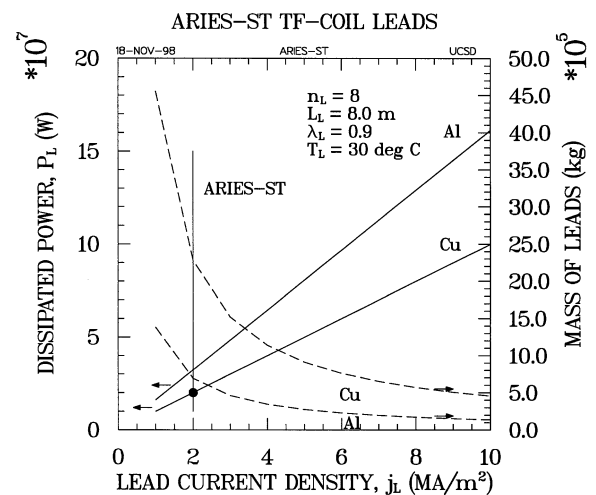


Fig. 4. Power-dissipation and mass trade-off of the ARIES-ST TFC resistive leads, comparing aluminium to the reference copper case.

The ASC costing algorithms limit the capacity of a single turbine to 1360 MWe. For the same target electrical output of 1.0 GWe(net) as pre-

Table 5
ARIES-ST power-plant physics parameters

Plasma major toroidal radius, R_T (m)	3.20
Plasma minor radius, a_p (m)	2.00
<i>Plasma vertical elongation</i>	
κ_X	3.40
κ_{95}	3.75
<i>Plasma triangularity</i>	
δ_X	0.64
δ_{95}	0.67
PAR, $A = R_T/a_p$	1.60
On-axis safety factor, q_0	4.33
Circularized safety factor, q^*	2.87
Plasma-edge safety factor, q	10.97
Troyon coefficient, C_T (Tm/MA)	0.07
Plasma beta, β	0.50
Plasma poloidal beta, β_p	1.70
Stability parameter, $\epsilon \beta_p$	1.06
Peak-to-average density, n_0/n	1.23
Peak-to-average temperature, T_0/T	1.14
Normalized edge density, n_E/n	0.05
Ion temperature, T_i (keV)	16.00
Electron temperature, T_e (keV)	16.47
Ion density, n_i (10^{20} per m^3)	1.41
Electron density, n_e (10^{20} per m^3)	1.58
Particle-to-energy confinement time ratio, τ_p/τ_E	1.00
Ion-to-electron energy confinement time ratio, τ_{E_i}/τ_{E_e}	1.00
Lawson parameter, $n_i \tau_E$ (10^{20} s/ m^3)	3.02
ITER-89P scaling [10] multiplier, H_{89H}	3.00
ITER-98P scaling [10] multiplier, H_{98H}	1.47
On-axis toroidal field, $B_{\phi 0}$ (T)	2.08
Radiation fraction, f_{RAD}	0.30
Plasma current, I_p (MA)	28.49
Bootstrap-current fraction, f_{BC}	0.96
CD power to plasma, P_{CD} (MW)	27.79
<i>Current-drive efficiencies</i>	
I_{EX}/P_{CD} (mA/W)	43.42
γ (10^{20} A/W m^2)	5.22

vious ARIES designs, the gross electrical power, P_{ET} , of the ARIES-ST exceeds this threshold, requiring two turbine-generator sets at reduced capacity. Extrapolation of the single-turbine scaling to larger units (consistent with large, recent fission plants) results in the savings of ~ 3 mill/kWeh in the COE values reported here.

Space constraints preclude a full exposition of the cost database here. Unit costs for the FPC components are derived from recent ARIES studies. Since the ARIES-ST coil configuration is

unconventional, it is appropriate to report the coil costing basis in Table 4. Using consensus financial assumptions, total capital costs are calculated and the Cost of Electricity (COE, mill/kWe h) is projected. Assumptions include a plant factor, $p_f \simeq 0.76$, and a 6-year construction lead time. All costs are reported in 1992 \$. The base configuration of the ARIES-ST FPC is shown in Fig. 5. The inboard first wall conforms to a flux surface and serves as a limiter to the plasma surface inboard of the null. Vertical space is provided above the null for a divertor chamber that captures the plasma exhaust from the outboard plasma surface. The centerpost cross section expands above the equatorial plane as allowed by the first wall and inboard shield to provide maximum shaping of the centerpost conductor in order to reduce coil current density and Joule dissipation. The ARIES-ST CP is not up-down symmetric; the flaring of the upper end has been incorporated into the Al outboard shell to allow a lowering of the CP separate from the other FPC components. This tapered CP is not trapped by the inboard shield annulus.

4.2. Design survey

The most basic design parameter is the physical size of the tokamak. Typically, the ASC determines the plasma minor radius that minimizes the COE for a specified PAR and ion temperature with an internal scan of minor radius. The COE for the ARIES-ST might be expected to display a minimum in minor radius resulting from the trade-off of maximizing the FPC mass power density (MPD) and of maximizing the engineering gain ($1/\epsilon$, where ϵ is the recirculating power fraction). Decreasing the plasma minor radius decreases the blanket, shield, and coil masses and increases the MPD. For an ST-based power plant, the centerpost dissipation dominates the recirculating-power losses that determine the engineering gain. Decreasing the plasma minor radius increases the ohmic dissipation in the resistive TF coils, because the centerpost cross-sectional area decreases with decreasing plasma minor radius, and the centerpost resistance and ohmic dissipation then increase. Determination of a self-consistent PFC

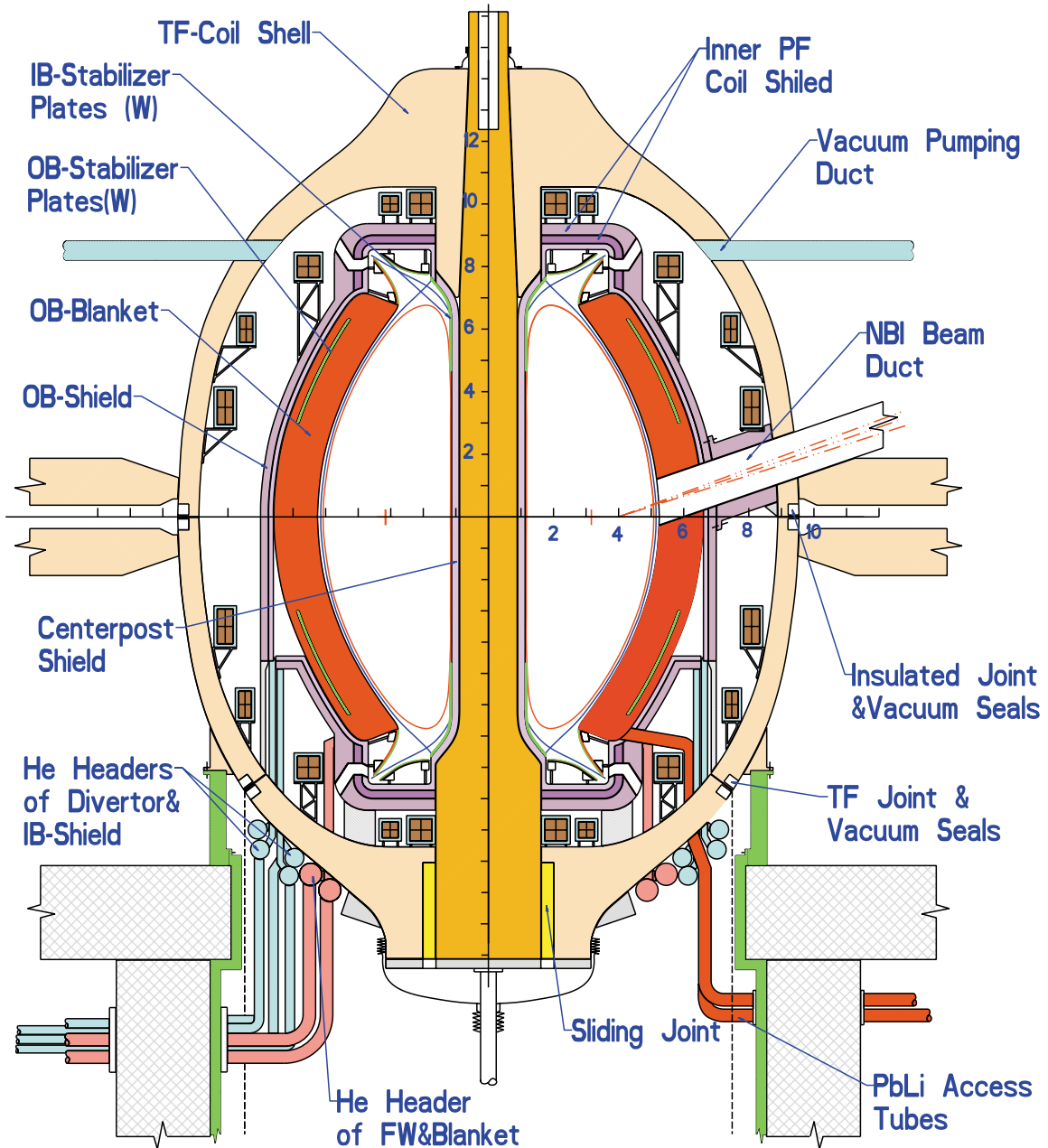


Fig. 5. Elevation view of the ARIES-ST FPC.

configuration may inhibit the realization of this minimum-COE design point. For the fixed PAR, $A \equiv R_T/a_p$, cases of Tables 1–3, plasma minor radius, a_p , is used by the ASC as a marching variable. Results are summarized in Fig. 6.

4.3. Design point determination

The selection of the ARIES-ST design point is summarized in the results of Fig. 7. The resolution of the PAR cases is coarse such that it is not

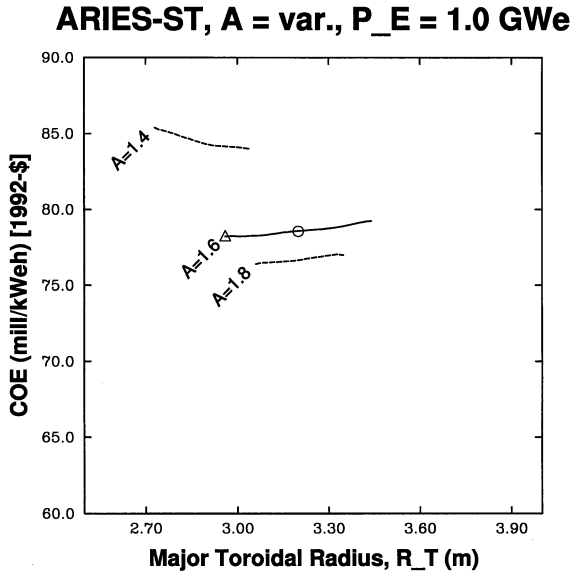


Fig. 6. Projected COE as a function of the major toroidal radius, R_T , for 1-GWe(net) ARIES-ST power plants with PAR, $A = R_T/a_p$, invoking advanced ('low-cost') manufacturing techniques cf. [21]. The reference ARIES-ST with $A = 1.6$ and $R_T = 3.2$ m is indicated by the open circle. The smallest accessible value of R_T with slightly reduced COE and higher power density and higher wall loads is indicated by the open triangle.

practical to identify minimum-COE performance in the vicinity of $A = 1.6$, which is taken as the baseline case. The reference ARIES-ST major plasma radius was taken to be $R_T = 3.2$ m.

Summary lists of parameters for the reference design used in the systems analysis are given in Tables 5–7. This final design point supercedes the interim results of [22–24]. The confinement-time enhancement factors, H , reported in Table 5 are calculated from empirical scaling relations derived from a database emphasizing conventional tokamaks. Since not enough information is available to characterize a ST-specific transport scaling, the H factors reported here should be considered suggestive only.

Level of Safety Assurance (LSA) factors have been used to give credit to fusion systems for improvements related to passive safety [25]. Nuclear systems without such credits reflect N-stamp costs consistent with present fission power plants ($LSA = 4$). At the other end of this scale, $LSA = 1$ represents non-nuclear (fossil) costs. ARIES-ST

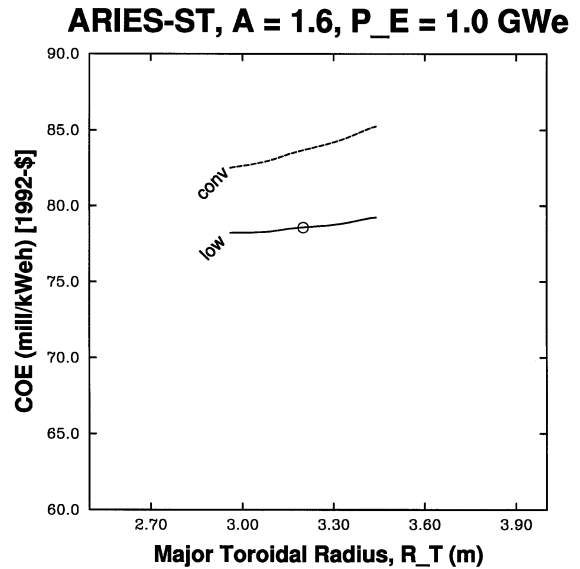


Fig. 7. Projected COE as a function of the major toroidal radius, R_T , for 1-GWe(net) ARIES-ST power plants with PAR, $A = 1.6$, comparing the impact of conventional ('conv') unit costs for the TFC system with advanced ('low') manufacturing techniques cf. [21]. The reference ARIES-ST with $R_T = 3.2$ m is indicated by the open circle.

has been designated to achieve $LSA = 2$. The COE values for the four LSA cases are summarized in Table 8 assuming conventional unit costs and again in Table 9, assuming the low-unit-cost advanced manufacturing [21] is applied to the Cu TF centerpost and outboard Al TF shell. These large components would be fabricated on-site. The periodic need for replacement centerposts implies the regular fabrication of these items.

4.4. Sensitivities

The sensitivity of the $A = 1.6$ ARIES-ST projected COE to the assumed thickness of the inboard shield, Δ_{Si} , is illustrated in Fig. 8. Not shown in these results is the increased resistivity growth for shields thinner than the reference case ($\Delta_{Si} = 0.20$ m) that would increase the Joule dissipation and penalize the COE.

The sensitivity of the $A = 1.6$ ARIES-ST projected COE to average first-wall 14-MeV neutron wall load is illustrated in Fig. 9. For fixed net electrical power output, $P_E = 1.0$ GWe, higher

Table 6
ARIES-ST power-plant engineering parameters

Field at coil, $B_{\phi c}$ (T)	7.43
Centerpost resistive losses (MW)	222
Total TF resistive losses (MW)	329
TF-coil current density, j_{TF} (MA/m ²)	7.77
Magnetic-field energy, W_B (GJ)	17.86
Total specific energy, W_B/M_c (MJ/t)	4.67
Fusion power, P_F (GW)	2.98
Neutron power, P_N (GW)	2.38
Average neutron wall load, I_w (MW/m ²)	4.10
Peak neutron wall load, $\hat{I}_w$ (MW/m ²)	5.58
Neutron fluence life-FW/B, $I_w\tau$ (MW year/m ²)	18.00
Average blanket energy multiplication	1.12
Average first-wall heat flux, q_w (MW/m ²)	0.33
Blanket power density, P_{TH}/V_{BLK} (MW/m ³)	4.78
CD system efficiency, η_{CD}	0.68
Thermal conversion efficiency, η_{TH}	0.45
Thermal power, $\dot{P}_{TH}$ (GWth)	3.37
Auxiliary site power, P_{AUX} (MW)	60.78
Primary loop pumping power (MW)	91.18
Gross electrical power, P_{ET} (GWe)	1.52
Net electrical power, P_E (GWe)	1.00
Recirculating-power fraction, $\varepsilon = 1/Q_E$	0.34
Plasma gain, $Q_p = P_F/P_{CD}$	108.05
Engineering gain, Q_E	2.93
Net plant efficiency, $\eta_p = \eta_{TH}(1 - \varepsilon)$	0.30
Plant capacity factor, p_r	0.76
<i>Masses (t)</i>	
First wall	13
Blanket and reflector	814
Shield	2545
TF coils	2904
PF coils	919
PF-coil spares	0
FPC	8,613
MPD (kWe/t)	54.97
LSA	2

(lower) wall loads occur as the FPC size is reduced (increased). The trade-off between recirculating power and MPD results in a nearly flat COE.

4.5. Superconducting TFC

The ARIES-ST study concentrated on the resistive TFC embodiment. The conventional wisdom has been that for ST systems with tight PAR, A , there would be insufficient space on the inboard side to provide adequate shielding (1–1.5 m) of the inboard TFC legs, assuming conventional low-temperature-superconductor (LTS) TFCs.

Table 7
ARIES-ST Power-plant economic parameters (1992 \$)

Account number	Account title	Million \$
20	Land and land rights	10.6
21	Structures and site facilities	370.8
22	Reactor plant equipment	1224.1
22.1.1	Blanket and reflector	50.2
22.1.2	Shield	102.6
22.1.3	Magnets	113.1
22.1.3.1	TFC set	16.0
22.1.3.3	PF coils	72.3
22.1.4	Supplemental heating systems	212.5
22.1.5	Primary structure and support	37.4
22.1.6	Reactor vacuum systems	48.1
22.1.7	Power supplies	71.6
22.1.8	Impurity control (divertor)	8.8
22.1.9	Direct energy conversion system	0.0
22.1.10	ECRH breakdown system	4.4
22.1	Reactor equipment	648.7
22.2	Main heat transfer and transport	358.1
22.[3-7]	Other	
23	Turbine plant equipment	339.6
24	Electric plant equipment	125.4
25	Miscellaneous plant equipment	77.9
26	Heat rejection system	64.3
27	Special materials	108.9
90	Total direct cost	2321.7
91	Construction services and equipment	278.6
92	Home office engineering and services	120.7
93	Field office engineering and services	139.3
94	Owner's costs	429.1
96	Project contingency	555.1
97	Interest during construction (IDC)	635.1
98	Escalation during construction (EDC)	0
99	Total capital cost	4479.7
[90]	Unit direct cost, UDC (\$/kWe)	2321.7
[94]	Unit base cost, UBC (\$/kWe)	3289.4
[99]	Unit total cost, UTC (\$/kWe)	3844.5
	Capital return (mill/kWe h)	65.04
[40-47,51]	O&M (1.4 %) (mill/kWe h)	9.28
[50]	Blanket replacement (mill/kWe h)	3.71
	Decommissioning (mill/kWe h)	0.50
[02]	Fuel (mill/kWe h)	0.05
	LSA = 2 COE (mill/kWe h)	78.57

The ASC was not able to ‘run to completion’ and solve for a self-consistent FPC in this range of inboard shield thicknesses. At about 0.5 m, however, a case was found that is generally consistent with this conventional wisdom. Despite the fact that there is the possibility of trading the additional direct cost of a LTS TFC against the costs

Table 8
ARIES-ST ($A = 1.6$) with conventional unit costing

LSA rating ^a	4	3	2	1
Capital return	78.0	71.57	69.66	58.57
O&M	10.91	10.09	9.28	7.64
SCR	4.48	4.40	4.17	3.85
Fuel	0.05	0.05	0.05	0.05
Decommissioning	1.00	0.75	0.50	0.25
Total COE (mill/kWe h) ^b	94.63	86.86	83.65	70.36

^a LSA, cf. [25].

^b Constant 1992 \$.

Table 9
ARIES-ST ($A = 1.6$) with advanced (low) unit costing

LSA rating ^a	4	3	2	1
Capital return	73.02	66.63	65.04	54.34
O&M	10.91	10.09	9.28	7.64
SCR	3.98	3.91	3.71	3.42
Fuel	0.05	0.05	0.05	0.05
Decommissioning	1.00	0.75	0.50	0.25
Total COE (mill/kWe h) ^b	88.97	81.43	78.57	65.69

cf. [21].

^a LSA, cf. [25].

^b Constant 1992 \$.

associated with recirculating power in the resistive system, even this optimistic LTS ST is seen to be less competitive than the range of resistive STs shown in Fig. 10. The large LTS-ST FPC is also noted to be incompatible with the ARIES-ST vertical maintenance scheme. No engineering attention was applied to this LTS ST case.

5. Summary and conclusions

Work on the ARIES-ST extends the preliminary assessment of the LAR case considered in the STARLITE Study for a 1-GWe (net) ARIES-ST. The interaction of engineering considerations with a broader set of physics cases suggests consideration of a PAR near $A = 1.6$, in part to accommodate a thin inboard shield. Joule dissipation on the resistive TF-coil results in an uncomfortably large recirculating power fraction. The application of advanced, low-unit-cost manufacturing techni-

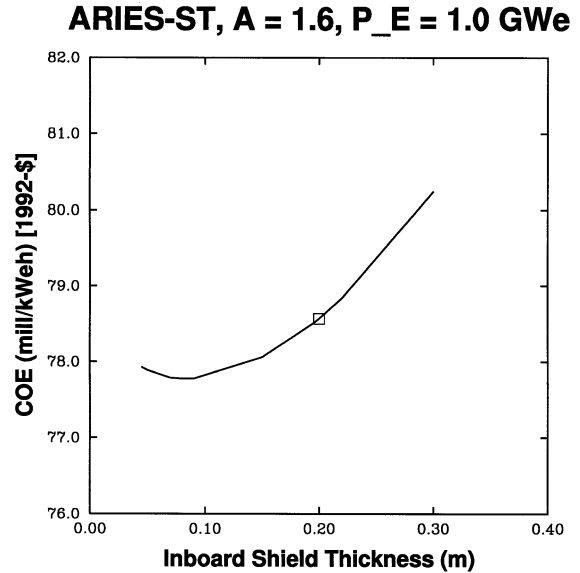


Fig. 8. Sensitivity of the projected COE of the 1-Gwe(net) ARIES-ST power plant with PAR, $A = 1.6$ to the thickness of the inboard shield, Δ_{si} . The penalty of additional neutron-damage-induced CP resistivity growth is not included for the thinner shields. The reference ARIES-ST with $\Delta_{si} = 0.20$ m is indicated by the open square.

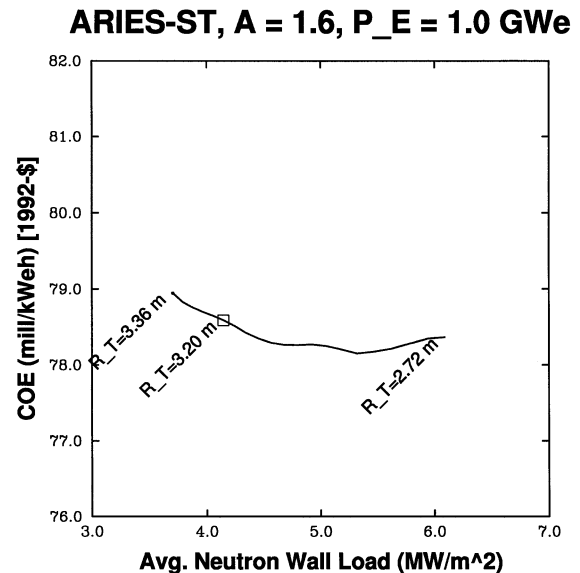


Fig. 9. Sensitivity of the projected COE of the 1-Gwe(net) ARIES-ST power plant with PAR, $A = 1.6$ to the average first-wall neutron load, I_w (MW/m^2). The reference ARIES-ST at $R_T = 3.20$ m is indicated by the open square.

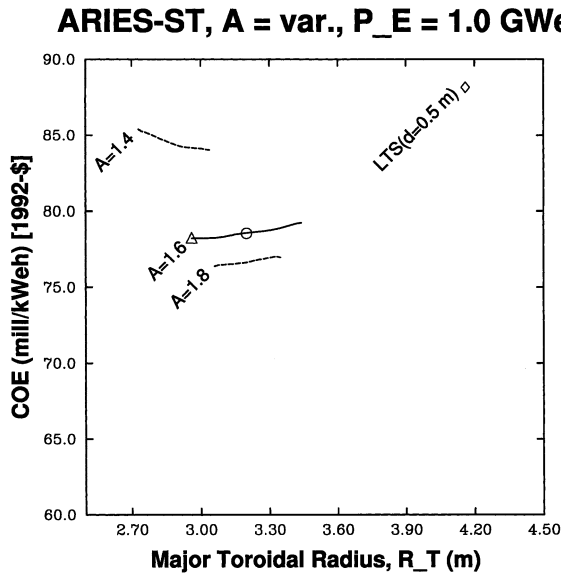


Fig. 10. Comparison of an optimistic, thin-inboard-shield ST power plant with $A = 1.6$ using LTS TFC technology, indicated by the diamond, with the resistive-coil ARIES-ST design window.

ques seems particularly well suited to the TFC centerpost and outer TFC shell to help offset the costs associated with high recirculating power. Access to higher beta would allow additional performance improvements.

Acknowledgements

This work was supported by the US Department of Energy, Office of Fusion Energy Sciences. Useful discussions with C.G. Bathke (LANL) and K.A. Werley (LANL) are acknowledged gratefully.

References

- [1] Y.K.M. Peng, et al., Physics progress toward compact tokamak reactors with normal conducting toroidal field coils, *Plasma Physics and Controlled Nuclear Fusion Research* (IAEA 1995) 2 (1994) 643.
- [2] M. Gryaznevich, et al., Achievement of record β in the START Spherical Tokamak, *Physics Review Letters* 80 (18) (1998) 3972.
- [3] F. Najmabadi, the ARIES Team, The STARLITE Study: Assessment of the Options for Tokamak Power Plants, University of California, San Diego Report UCSD-ENG-005, 1997.
- [4] R.L. Miller, R.A. Krakowski, C.G. Bathke, C. Copenha-ver, N.M. Schnurr, A.G. Engelhardt, T.J. Seed, R.M. Zubrin, Advanced Tokamak Reactors Based on the Spherical Torus (ATR/ST), Los Alamos National Laboratory Report LA-10740-MS, 1986.
- [5] S. Jardin, Physics basis for spherical torus power plants, *Fusion Engineering and Design* PII: S0920-3796(02)00303-4.
- [6] K.A. Werley, Reversed Field Pinch Ignition Physics, Los Alamos National Laboratory Document LA-UR-89-4093, 1989.
- [7] K.A. Werley, Reversed field pinch ignition requirements, *Nuclear Fusion* 31 (1991) 576.
- [8] ITER Conceptual Design Interim Report, ITER Documentation Internal Series No. 7, International Atomic Energy Agency, Vienna, 1989.
- [9] N.A. Uckan, Tokamak confinement projections and performance goals, *Fusion Technology* 15 (1989) 391.
- [10] D.E. Post, K. Borrass, J.D. Callen, S.A. Cohen, J.G. Cordey, F. Engelmann et al., ITER Physics, International Thermonuclear Experimental Reactor documentation series No. 21, November 1990.
- [11] T.K. Mau, S.C. Jardin, C.E. Kessel, J.E. Menard, R.L. Miller, F. Najmabadi, V.S. Chan, L.L. Lao, T.W. Petrie, P.A. Politzer, A.D. Turnbull, Physics basis for the ARIES-ST power plant, Proceedings of IEEE Eighteenth Symposium on Fusion Engineering, Albuquerque, NM (October 1999) IEEE No. 99CH37050, IEEE, Piscataway, NJ, 1999, p. 45.
- [12] L.L. Snead, D. Steiner, Center conductor design for the compact Spherical Torus reactor, Proceeding of IEEE Thirteenth Symposium on Fusion Engineering, vol. 2, 89CH2829-9, 1989, p. 890.
- [13] R.D. Stambaugh, V.S. Chan, R.L. Miller, M.J. Schaffer, The spherical tokamak path to fusion power, *Fusion Technology* 33 (1) (1998) 1.
- [14] R.L. Miller for the ARIES Team, Design-point determination for the commercial Spherical Tokamak, ARIES-ST, Proceedings IEEE Seventeenth Symposium on Fusion Engineering, San Diego, CA (October 1997) IEEE No. 97CB36131, IEEE, Piscataway, NJ, 1997 p. 1031.
- [15] C.C. Baker, M.A. Abdou, R.M. Arons, A.E. Bolon, C.D. Boley, J.N. Brooks et al., STARFIRE—a Commercial Tokamak Fusion Power Plant Study, Argonne National Laboratory Report ANL/FPP-80-1, September 1980.
- [16] J.G. Delene, R.A. Krakowski, J. Sheffield, R.A. Dory, GENEROMAK Fusion Physics, Engineering, and Costing Model, Oak Ridge National Laboratory Report ORNL/TM-10728, June 1988.
- [17] S.L. Thomson, Cost Algorithms for Superconducting Magnets, Oak Ridge National Laboratory Document FEDC-M-88-ITER-001, October 1988.

- [18] J.R. Miller, S.L. Thomson, Cost Algorithms for Superconducting Magnets, October 1988 (in press).
- [19] L. Bromberg, D.R. Cohn, J. Schwartz, J.E.C. Williams, Costing of the Toroidal Field Coils for ARIES-I, Massachusetts Institute of Technology, private communication, March 1989.
- [20] F. Najmabadi, R.W. Conn, R.A. Krakowski, K.R. Schultz, D. Steiner, the TITAN Research Group, The TITAN Reversed-Field-Pinch Reactor Study—the Final Report, University of California at Los Angeles, General Atomics, Los Alamos National Laboratory, and Rensselaer Polytechnic Institute, UCLA Report, PPG-1200, 1990.
- [21] L.M. Waganer, Innovative, ultra-low cost fabrication methods, Proceedings of the IEEE Eighteenth Symposium on Fusion Engineering Albuquerque, NM (October 1999) IEEE No. 99CH37050, IEEE, Piscataway, NJ, 1999 p. 77.
- [22] R.L. Miller, Overview of the spherical tokamak power plant study: ARIES-ST, *Fusion Technology* 34 (3) (1998) 364 (Part 2).
- [23] R.L. Miller, Design-point finalization for the ARIES-ST power plant, Proceedings IEEE Eighteenth Symposium on Fusion Engineering, Albuquerque, NM (October 1999) IEEE No. 99CH37050, IEEE, Piscataway, NJ, 1999 p. 41.
- [24] R.L. Miller, Which are the competitors for a fusion power plant, *Fusion Engineering and Design* 49–50 (2000) 33.
- [25] J.P. Holdren, D.H. Berwald, R.J. Budnitz, J.G. Crocker, J.G. Delene, R.D. Endicott, M.S. Kazimi, R.A. Krakowski, B.G. Logan, K.R. Schultz, Exploring the competitive potential of magnetic fusion energy: the interaction of economics with safety and environmental characteristics, *Fusion Technology* 13 (1) (1988) 7.